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LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AHU	Air Handling Unit
ASTM	American Society for Testing and Materials
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
DBE	Design Basis Event
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GSI	Generic Safety Issue
HVAC	Heating, Ventilation, and Air Conditioning
HEPA	High Efficiency Particulate Air
HPSI	High Pressure Safety Injection
ITS	Improved Technical Specifications
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MTC	Moderator Temperature Coefficient
NDTT	Nil-Ductility Transition Temperature
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management & Resources Council
PORV	Power-Operated Relief Valve
PSAR	Preliminary Safety Analysis Report
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protective System
RVCH	Reactor Vessel Closure Head
RWT	Refueling Water Tank
SACM	Societe Alsacienne De Constructions Mecaniques De Mulhouse
SBO	Station Blackout
SER	Safety Evaluation Report

LIST OF ACRONYMS

SG	Steam Generator
SIT	Safety Injection Tank
SRO	Senior Reactor Operator
TMI	Three Mile Island
TRM	Technical Requirements Manual
VCT	Volume Control Tank

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

Construction of Calvert Cliffs Units 1 and 2 was authorized by the Atomic Energy Commission (AEC) by issuance of Construction Provisional Permits CPPR-63 and CPPR-64 in Docket Numbers 50-317 and 50-318 on July 7, 1969. Unit 1 went into commercial operation in May 1975, and Unit 2 in April 1977.

On July 11, 1967, the AEC published in the Federal Register the Proposed General Design Criteria for Nuclear Power Plants. Prior to the issuance of the construction permit, Calvert Cliffs submitted the Preliminary Safety Analysis Report (PSAR) in which was reflected this plant's design intent based on these criteria. Design and construction proceeded accordingly. The Final Safety Analysis Report (FSAR) was submitted in support of the application for a license to operate the plant. Revision 0 of the Updated Final Safety Analysis Report was submitted in July 1982, and has been periodically revised since then.

Subsequent to the initial startup of both units, 10 CFR Part 50, Appendix A containing 64 general design criteria was issued. These criteria reflected the original 70 criteria with revisions and regrouping. Design changes and modifications for Calvert Cliffs are evaluated for consistency with the proposed criteria except where specific Appendix A criteria have been required by the Nuclear Regulatory Commission (NRC).

The Nuclear Steam Supply System (NSSS) for both units is identical, utilizing pressurized water reactors supplied by Combustion Engineering, Inc. (CE). The NSSS includes a control element assembly (CEA)-type reactor core with two steam generators (SGs), two reactor coolant loops and four reactor coolant pumps (RCPs). The geometry of the core is essentially identical to that used for the Main Yankee Atomic Power Station (Docket Number 50-309). The reactor coolant loops are very similar to those in the Palisades Plant (Docket number 50-255). The SGs are Babcock & Wilcox, Canada replacement steam generators. The replacement reactor vessel closure heads (RVCHs) were supplied by Babcock & Wilcox, Canada.

An initial license was requested to operate each of the facilities at a core thermal output of 2,560 megawatts (MWt). An increase in power to 2700 MWt was authorized by license amendments (References 1 and 2). Rated thermal power was once again increased to 2737 MWth as part of a measurement uncertainty recapture modification which was approved by Reference 3. Site parameters and major systems and components, including the engineered safety features (ESFs) and containment structures, have been evaluated for operation at the higher power level. The postulated incidents considered in Chapter 14 are also evaluated at the higher power level.

Over the time the plant has been operated, numerous modifications have been made. In some cases, these changes were submitted to, and approved by, the NRC through the license amendment process. In other cases, changes were implemented under the provisions of 10 CFR 50.59 with notification to the Commission after the fact.

Each revision to the Updated Final Safety Analysis Report is intended to reflect, within the limitations of the report format, the configuration and operation of the plant at the end of the refueling outage preceding the revision date, as required by 10 CFR 50.71.

On the basis of the information presented in this FSAR and referenced material, Calvert Cliffs Nuclear Power Plant (CCNPP) concludes that CCNPP Units 1 and 2 were designed and constructed and are operated without undue risk to the health and safety of the public.

1.1.1 REFERENCES

1. Letter from D. K. Davis (NRC) to A. E. Lundvall, Jr. (BGE), dated September 9, 1977, Amendment No. 24 to Facility Operating License No. DPR-53 for Unit No. 1

2. Letter from D. K. Davis (NRC) to A. E. Lundvall, Jr. (BGE), dated October 19, 1977, Amendment No. 9 to Facility Operating License No. DPR-69 for Unit No. 2

3. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), dated July 22, 2009, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC Nos. MD9554 and MD9555) (Amendment Nos. 291/267)

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 PLANT SITE

The site for the Calvert Cliffs Nuclear Power Plant (CCNPP) consists of approximately 962 acres on the western shore of the Chesapeake Bay, in Calvert County, about 10-1/2 miles southeast of Prince Frederick, Maryland (Figure 1-1). The site is characterized by a minimum exclusion radius of 1,150 meters, remoteness from population centers, an abundant supply of cooling water, and favorable conditions of hydrology, geology, seismology and meteorology. The nearest population center is Washington, DC, which is approximately 45 miles to the northwest of the site (Figures 1-1 and 2.2-1).

1.2.2 PLANT ARRANGEMENT

The Turbine Building for the CCNPP is oriented parallel and adjacent to the shoreline of the Chesapeake Bay with the twin Containment Structures and the Auxiliary Building located on the west, or landward, side of the Turbine Building. The service building and the intake and discharge structures are on the east, or bay side, of the Turbine Building (Figure 1-2).

Each Containment Structure houses an NSSS, consisting of a reactor, SGs, RCPs, a pressurizer, and some of the reactor auxiliaries which do not normally require access during power operation. Each Containment Structure is served by a polar crane.

The Turbine Building houses the two turbine generators, condensers, feedwater heaters, condensate and feed pumps, turbine auxiliaries, and certain of the switchgear assemblies.

The Auxiliary Building houses the waste treatment facilities, ESF components, heating and ventilating system components, the Fairbanks Morse emergency diesel generators (EDGs), switchgear, laboratories, offices, laundry, Control Room, spent fuel pool, and new fuel storage facilities. Fuel transfer to and from the containment is through a fuel transfer tube.

A safety-related building houses the Societe Alsacienne De Constructions Mecaniques De Mulhouse (SACM) emergency diesel generator. The fuel oil storage tank and auxiliary equipment for this diesel generator are also housed in this building (Figures 1-31 through 1-34). An augmented quality building houses the SACM Station Blackout diesel generator. Auxiliary equipment for this diesel generator is housed in this building (Figures 1-35 through 1-37).

The Independent Spent Fuel Storage Installation, located on Road C-1, is described in its own Updated Safety Analysis Report.

Plant layouts are shown in Figures 1-4 through 1-37.

1.2.3 REACTOR

The reactor of either unit is a pressurized light water cooled and moderated-type fueled by slightly enriched uranium dioxide. The uranium dioxide is in the form of pellets and is contained in Zircaloy, ZIRLO, or M5® tubes fitted with welded end caps. These fuel rods are arranged into fuel assemblies each consisting of 176 fuel rods arranged on a 14-rod square matrix. Space is left in the fuel rod array to allow for the installation of five guide tubes. These guide tubes provide for the smooth motion of CEA fingers. The assembly is fitted with end fittings and spacer grids to maintain fuel rod alignment and to provide structural support. The end fittings are also drilled with flow holes to provide for the flow of cooling water past the fuel tubes.

The reactor is controlled by a combination of chemical shim and solid absorber. The solid absorber is boron carbide in the form of pellets contained in Inconel tubes. Five tubes of absorber form a CEA (i.e., four tubes in a square matrix plus a central tube). The five tubes are connected together at the tops by a yoke which is, in turn, connected to the control element drive mechanism (CEDM) extension shaft. Each CEA is aligned with and is inserted into a guide tube in the fuel assembly.

Chemical shim control is provided by boric acid dissolved in the coolant water. The concentration of boric acid is maintained and controlled as required by the Chemical and Volume Control System (CVCS).

The reactor core rests on the core support plate assembly which is supported by the core support barrel. The core support barrel is a right circular cylinder supported from a machined ledge on the inside surface of the vessel flange forging. The support plate assembly transmits the entire weight of the core to the core support barrel through a structure made of beams and vertical columns. Surrounding the core is a shroud which serves to limit the coolant which bypasses the core. An upper guide structure, consisting of an upper support structure, CEA shrouds, a fuel alignment plate and a spacer ring, serves to support and align the upper ends of the fuel assemblies, prevents lifting of the fuel assemblies in the event of a loss-of-coolant accident (LOCA), and maintains spacing of the CEAs.

1.2.4 REACTOR COOLANT SYSTEM

The Reactor Coolant System (RCS) of each unit consists of two closed heat transfer loops in parallel with the reactor vessel. Each loop contains one SG and two pumps to circulate coolant. An electrically heated pressurizer is connected to one loop hot leg. The coolant system is licensed to operate at a power level of 2,737 MWt to produce steam at a pressure of 888 psia with no plugged SG tubes.

The reactor vessel, loop piping, and SG plenums are fabricated of low alloy steel, clad internally with stainless steel. The pressurizer surge line and RCPs are fabricated from stainless steel and the SG tubes are fabricated from Inconel.

Overpressure protection is provided by power-operated relief valves (PORVs) and spring-loaded safety valves connected to the pressurizer. Safety and relief valve discharge is released under water in the quench tank where the steam discharge is condensed.

The two SGs are vertical shell and U-tube SGs each of which produces approximately $6x10^6$ lb/hr of steam. Steam is generated in the shell side of the SG and flows upward through moisture separators. Steam outlet moisture content is less than 0.05%.

The reactor coolant is circulated by four electric motor-driven, single-suction, centrifugal pumps. Each pump motor is equipped with a nonreverse mechanism to prevent reverse rotation of the pump.

1.2.5 CONTAINMENT

The Containment Structure uses a pre-stressed concrete design. The structure is in the form of a vertical right cylinder with a dome and a flat base. The interior of the structure is lined with carbon steel plate for leak tightness. Inside the structure, the reactor and other NSSS components are shielded with concrete. An unlined steel ventilation stack is attached to the outside of the Containment Structure and extends to an elevation about 10' above the top of the containment dome. Access to portions of the Containment Structure during power operation is permissible.

The Containment Structure, in conjunction with ESFs, is designed to withstand the internal pressure and coincident temperature resulting from the energy released in the event of the LOCA associated with operation at rated thermal power plus uncertainty. The design conditions for the structure are an internal pressure of 50 psig, a coincident concrete surface temperature of 276°F and a leak rate of 0.16% by weight per day at design temperature and pressure.

1.2.6 ENGINEERED SAFETY FEATURES SYSTEMS

Separate ESF systems for each unit in conjunction with separate containment systems protect the public and plant personnel from accidental release of radioactive fission products, particularly in the unlikely event of a LOCA. These safety features function to localize, control, mitigate, and terminate such incidents to hold exposure levels below applicable guidelines.

The ESF systems are:

The safety injection systems [including High Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), and the Safety Injection Tanks (SITs)];

The containment cooling systems, consisting of the Containment Spray System (CSS), and the Containment Air Recirculation and Cooling System;

The Containment Penetration Room Ventilation System;

The Containment Iodine Removal System.

For each unit, four SITs are provided, each connected to one of the four reactor inlet lines. Each tank has a volume of 2,000 ft^3 containing 1,000 ft^3 of borated water at refueling concentration and 1,000 ft^3 of nitrogen at 200 psig. In the event of a LOCA, the borated water is forced into the RCS by the expansion of the nitrogen. The water from three tanks adequately cools the entire core. In addition, borated water is injected into the same nozzles by two LPSI and two (three pumps are available) HPSI pumps taking suction from the refueling water tank (RWT). For maximum reliability, the design capacity from the combined operation of one HPSI and one LPSI pump provides adequate injection flow for any LOCA; in the event of a Design Basis Event (DBE), at least one HPSI and one LPSI pump will receive power from the emergency power sources if normal power is lost and one of the EDGs is assumed to fail. Upon depletion of the RWT supply, the HPSI pump suctions automatically transfer to the containment sump and the LPSI pumps are shutdown. One HPSI pump has sufficient capacity to cool the core adequately at the start of recirculation. During recirculation, heat in the recirculating water is removed in the shutdown cooling heat exchangers by the operation of the CSS (see below). Further, the suction of the HPSI pumps may be manually aligned so as to inject sub-cooled water from the shutdown cooling heat exchangers directly in the RCS for core cooling.

All LPSI and HPSI pumps are located outside the Containment Structure to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during plant operation.

The CSS supplies cool, borated water which reduces the temperature and pressure of the containment atmosphere. The pumps take suction initially from the RWT. Long-term cooling is based on suction from the containment sump through the recirculation lines. In the recirculation mode of operation, heat is transferred from the recirculating borated water via the shutdown cooling heat exchangers to the Component Cooling System and ultimately to the Chesapeake Bay water via the component cooling heat exchangers.

The Containment Air Recirculation and Cooling System is also designed to provide capability for reducing the temperature and pressure of the containment atmosphere. The cooling coils and fans are sized to provide adequate containment cooling at DBE conditions without assistance from other containment heat removal systems. The heat is transferred to the Service Water System.

Operation of the penetration room exhaust system ensures that radioactive materials discharging from the containment atmosphere following a LOCA are filtered prior to reaching the environment. The penetration room may be maintained at a negative pressure relative to the Containment following a LOCA. The penetration room ventilation system is equipped with particulate filters and charcoal adsorbers.

The containment iodine removal system recirculates containment air through charcoal adsorbers to remove iodine from the containment atmosphere.

1.2.7 REACTOR PLANT PROTECTION, CONTROL, AND INSTRUMENTATION SYSTEMS

1.2.7.1 Reactor Protection

Reactor parameters are maintained within acceptable limits by the inherent self-controlling characteristics of the reactor, by CEA positioning, by boron content of the reactor coolant and by operating procedures. The function of the Reactor Protective System (RPS) is to provide reactor operators with audible and visual alarms when any reactor parameter approaches the preset limits for safe operation. Should pre-selected limits be reached, the RPS initiates reactor shutdown to prevent unsafe conditions for plant personnel and equipment and to the general public.

The RPS is divided into four channels, each receiving trip signals from separate sensors when the relevant parameter reaches a preset level. If any two of these four channels receives coincident signals, the power supply to the magnetic jack CEDM is interrupted allowing the control elements to drop into the core to shut down the reactor. The protective system is completely independent of, and separate from, the control system.

1.2.7.2 Reactor Control

The RCS provides for start-up and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The NSSS is capable of following a ramp change from 15% to 100% power at a rate of 5% per minute and at greater rates over smaller load change increments up to a step change of 10%. The control is accomplished by manual control. The temperature control program provides a demand temperature which is a function of power. This temperature is compared with the coolant average temperature; if the temperatures are different, the CEAs are adjusted until the difference is within the prescribed control band. Regulation of the reactor coolant temperature in accordance with this program maintains the secondary steam pressure within operating limits and matches reactor power to load demand.

The reactor is controlled by a combination of CEAs and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, xenon effects and fuel burnup. Additions of boric acid also provide an increased shutdown margin during the initial loading and subsequent refuelings.

Control Element Assembly movement provides changes in reactivity for shutdown or power changes. The CEAs are actuated by CEDMs mounted on the reactor vessel head. The CEDMs are designed to permit rapid insertion of the CEAs into the reactor core by gravity. Control Element Assembly motion is initiated manually.

The pressure in the RCS is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce pressure variations caused by expansion and contraction of the reactor coolant due to reactor system temperature changes.

1.2.7.3 Instrumentation

The nuclear instrumentation includes out-of-core and incore neutron flux detectors. Ten channels of excore instrumentation monitor the neutron flux and provide reactor protection and control signals during start-up and power operation. Four of the channels monitor the neutron flux from the start-up range through the full power range, and six channels monitor the neutron flux from within the start-up range through the full power range. Of the latter, four are used for reactor protection and two for reactor control.

The incore monitors consist of self-powered rhodium neutron detectors and thermocouples to provide information on neutron flux distribution and temperature in the core. The Unit 1 and Unit 2 instrumentation consists of 35 incore detector assemblies.

The process instrumentation monitoring includes those critical channels which are used for protective action. Additional temperature, pressure, flow and liquid level monitoring is provided, as required, to keep the operating personnel informed of plant conditions, and to provide information from which plant processes can be evaluated and/or regulated.

Instrument signals penetrating the containment are electronic. Instrument signals for the remaining plant instruments are either electronic or pneumatic depending on the function to be served.

The plant gaseous and liquid effluents are monitored for radioactivity. Activity levels are displayed and off-normal values are annunciated. Area monitoring stations are provided to measure radioactivity at selected locations in the plant.

1.2.8 ELECTRICAL SYSTEMS

The CCNPP includes two generating units, the ratings of which are 1,020,000 kVa, 0.9 PF, 25 kV, for Unit 1 and 1,011,900 kVa, 0.9 PF, 22 kV, for Unit 2. Each generator delivers power to the 500 kV switchyard through two 810,000 kVa main step-up power transformers. Three 500 kV transmission lines connect to the switchyard and transmit the plant output to the network.

The plant distribution system utilizes voltage levels of 13.8 kV, 4.16 kV, 480 Volt, and 120/208 Volt. The system is designed to provide reliable power for normal operation and safe shutdown of the plant. Auxiliary and start-up power will be supplied by two service transformers rated at 500/14 kV and 60/80/100 mVa. Each transformer is capable of supplying the total auxiliary load of both units simultaneously. One service transformer is connected to each 500 kV bus in the switchyard.

Four 125 Volt DC systems provide continuous emergency power for control, vital instrumentation, emergency lighting, vital 120 Volt AC loads, and computers. Both units share a 250 Volt DC system which supplies power to the emergency lube and seal oil pumps. Separate battery systems are provided for substation control, relaying, microwave, telemetering, and communications.

A total of four EDGs, two dedicated to each unit, are provided to supply power to the ESF loads. Three of these EDGs are Fairbanks Morse diesels with generators rated at 0.8 PF and 4160 Volts and continuous ratings of 3000 kW. The fourth is an SACM diesel generator (Diesel Generator 1A) with a continuous rating of 5400 kW, 0.8 PF, and 4160 Volts. Although the Fairbanks Morse and SACM diesels have different continuous ratings, either of the two EDGs dedicated to a unit is capable of supplying all of the ESF loads for the associated bus. In addition, an augmented-quality Station Blackout SACM diesel generator with a continuous rating of 5400 kW, 0.8 PF, and 4160 Volts is installed. This generator can be aligned to any of the four 4160 Volt emergency buses to support, SBO, or ESF loads, if necessary.

1.2.9 AUXILIARY SYSTEMS

1.2.9.1 Chemical and Volume Control System

The purity level in the RCS is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the RCS is cooled in the regenerative heat exchanger. From there, the coolant flows to the letdown heat exchanger and then through a filter and demineralizer where corrosion and fission products are removed. It is then sprayed into the volume control tank (VCT), and returned to the RCS by the charging pumps through the regenerative heat exchanger.

The CVCS automatically controls the rate of coolant removed from the RCS to maintain the pressurizer level within the prescribed control band, thereby compensating for changes in volume due to coolant temperature changes. The VCT is sized to accommodate coolant inventory changes resulting from load changes from hot standby to full power. Using the VCT as a surge tank decreases the quantity of liquid and gaseous waste which otherwise would be generated.

Reactor Coolant System make-up water is taken from the demineralized water storage system and from the two concentrated boric acid tanks. The boric acid solution in these tanks is maintained at a temperature which prevents crystallization. The make-up water is pumped through the regenerative heat exchanger into the reactor coolant loop by the charging pumps.

Boron concentration in the RCS can be reduced by diverting the letdown flow away from the VCT to the Reactor Coolant Waste Processing System and using demineralized water for coolant make-up (feed and bleed).

When the boron concentration in the RCS is low, the feed and bleed procedure described above would generate excessive volumes of waste to be processed; therefore, the CVCS is equipped with an ion exchanger which is loaded with deborating resin to reduce boron concentration late in cycle life.

1.2.9.2 Shutdown Cooling System

The Shutdown Cooling System is used to reduce the temperature of the reactor coolant at a controlled rate from 300°F to a refueling temperature of ≤ 140°F and to maintain the proper reactor coolant temperature during refueling.

The Shutdown Cooling System utilizes the LPSI pumps to circulate the reactor coolant through two shutdown cooling heat exchangers, returning it to the RCS through the LPSI header. Component cooling water is used to cool the shutdown cooling heat exchangers.

1.2.9.3 Component Cooling System

The Component Cooling System consists of three pumps, two saltwater-cooled heat exchangers, interconnecting piping, valving and controls. The corrosion-inhibited, demineralized water of this closed system is circulated through the component cooling heat exchangers where it is cooled to a design temperature of 95°F by saltwater with a maximum design inlet temperature of 90°F. Component cooling water temperature may reach as high as 120°F during a LOCA and during plant cooldown.

Typical items cooled by component cooling water are:

- Shutdown cooling heat exchangers
- Letdown heat exchanger
- RCP seals and lube oil cooler
- HPSI pump seals
- LPSI pump seals
- Waste gas compressors aftercooler
- Waste evaporators

All ESF equipment connected to this system and requiring cooling water are fed by flow paths arranged in parallel to each other.

A component cooling water head tank floats on the system and absorbs the volumetric changes due to temperature changes to which the water in the closed system is subjected.

A chemical additive tank is piped and valved to the system in such a way that the corrosion inhibitor concentration can be increased during normal operation as required.

During normal plant operation, only one of the three pumps and one of the two heat exchangers are required for cooling service.

During normal shutdown, two of the three pumps and both of the heat exchangers are utilized for cooling.

For a LOCA one of the three pumps and both of the heat exchangers can provide the necessary cooling.

1.2.9.4 Fuel Handling and Storage Systems

The fuel handling systems provide for the safe handling of fuel assemblies and CEAs and for the required assembly, disassembly, and storage of the reactor vessel head and internals. These systems include a polar crane and a refueling machine located inside containment above the refueling pool, the fuel transfer carriage, the upenders, the fuel transfer tube, a fuel handling machine in the spent fuel storage room, and various other devices used for handling the reactor vessel head and internals.

The spent fuel pool, located in the Auxiliary Building, consists of two halves. Both new fuel and spent fuel may be stored in either half of the pool. Dry storage for

new fuel is provided near the spent fuel pool in the new fuel storage racks. A spent fuel handling machine is provided for manipulation of the spent fuel.

Spent fuel may be stored at the Independent Spent Fuel Storage Installation. The NUHOMS dry storage system is used for the transfer and storage of spent fuel. The system includes storage canisters, a transfer cask, lifting yoke and transfer trailer. A detailed description of the components and transfer operations is discussed in the Independent Spent Fuel Storage Installation Safety Analysis Report.

1.2.9.5 Sampling System

The sampling system consists of three subsystems: reactor coolant sampling, radioactive waste systems sampling, and turbine plant sampling. These subsystems provide the means for determining chemical and radiochemical conditions of the process fluids used in the plant.

The turbine plant sample station is located in the Turbine Building. This station contains pressure reducing valves, cooling equipment, pressure, temperature and flow control regulators, valves, piping, grab sample sink, and continuous pH, oxygen, and conductivity monitors, and indicators. An annunciator is located in the Control Room to alarm on abnormal conditions at the sampling station.

The reactor coolant and radioactive waste subsystems sample stations are located in the Auxiliary Building. The radioactive waste sampling station which is common to both units is located in the Unit 1 sample room. Each sample room contains the piping, valves, and cooling equipment necessary to reduce the pressure and temperature of the sample fluid or gas to acceptable levels for grab sampling or collection in a sample bomb. The sample streams are radioactive or potentially radioactive and may contain boric acid. All grab samples and bomb samples are taken to the chemistry laboratory for analysis.

1.2.9.6 Cooling Water Systems

The exhaust steam of the main turbine and SG feed pump turbines is condensed by circulating water. Six circulating water pumps per unit, having a combined volumetric capacity of 1,200,000 gpm take suction from and discharge to the Chesapeake Bay through a three-shell condenser (Figures 1-3A and 1-3B). The circulating water system is designed to maintain condenser back pressure at 2" Hg absolute with a 70°F injection temperature.

Centrifugal displacement-type vacuum pumps maintain a siphon on the condenser circulating water system and permit the circulating water pumps to operate at minimum total dynamic head based on friction drop through the system.

The saltwater cooling system provides bay water to the component cooling heat exchangers, the service water heat exchangers, and the Emergency Core Cooling System (ECCS) pump room air coolers. There are three vertical centrifugal pumps per unit, only two of which are to be on-line during normal plant operation. These pumps take suction from the circulating water intake structure and discharge through the heat exchangers to the Chesapeake Bay.

The Saltwater Chemical Addition System serves both the Unit 1 and Unit 2 Saltwater Systems to minimize the marine fouling of piping and heat exchanger surfaces. This system has the ability to inject approved chemicals into each saltwater header, as necessary.

In the Circulating Water System, a mechanical condenser tube cleaning system is used instead of chemical addition to minimize fouling on the heat exchange surfaces.

1.2.9.7 Plant Ventilation Systems

All areas are heated, cooled, and ventilated in different ways depending upon the peculiarities of each.

Normally the containment atmosphere is cooled using water in fan-coil heat exchangers. After a LOCA these coolers, plus a containment filtering system, will reduce the radioactivity concentration, temperature, and pressure of the containment atmosphere to a safe level. A forced outside air purging system, whose discharge is filtered to eliminate contaminants, is provided to protect personnel entering the containment.

Separate (forced) supply and exhaust ventilating systems are provided in the Auxiliary Building. The exhaust air is forced through high efficiency particulate air (HEPA) filters, then mixed with outside air before it is discharged into the atmosphere. Hot water unit heaters are used for standby and auxiliary heating. The access control areas in this building have a combination heating, cooling, and ventilating system which makes use of a direct expansion-type water chiller with an air-cooled condenser.

The Control Room and the cable spreading room are incorporated into a single air conditioning system, serving Units 1 and 2. The air handling and refrigeration equipment is redundant, but the ductwork is not. During the post-LOCA period, upon a high radiation signal from the Control Room air monitor or a safety injection actuation signal initiation from either Unit, a portion of the recirculated air is shunted through HEPA and charcoal filters to provide for a reduction in the Control Room airborne radioactivity concentration. A chiller is also provided which can supply chilled water to a second set of cooling coils in the air handling equipment. For Appendix R events, this chiller can also be used in conjunction with fan coil units to provide an alternate source of cooling to the Control Room and cable spreading room. The battery rooms are ventilated using air from the access control area.

In the Turbine Building, supply and exhaust fans provide year-round ventilation plus cooling in summertime. Heat is provided by hot water unit heaters.

The office and conference rooms of the service buildings have year-round cooling, heating, and ventilating systems. Floor mounted fan-coil units and suspended air handling units work in conjunction with a direct expansion type water chiller and air cooled condenser. Summer-winter ventilation is provided for the balance of these buildings. Heat is provided through use of hot water coils and in some cases electric heating elements, except in office spaces where only electric heaters are installed.

Due to the high output of heat from plant equipment, the intake structure pump room is cooled during both summer and winter using natural and forced air circulation. Hot water unit heaters are used for shutdown periods.

The Heating, Ventilation, and Air Conditioning (HVAC) System for the safety-related Diesel Generator Building is divided into safety-related and non-safety-related portions. While the emergency diesel generator is not in operation, the non-safety-related ventilation provides cooling to the Diesel Generator Building

Control Room, Battery Room, 1E Switchgear Room, and non-1E Electrical Panel Room using a constant volume, direct-expansion cooling air handling unit (1A-AHU-1). During diesel generator operation, the safety-related ventilation system provides cooling to the Diesel Generator Room using only outdoor ambient air. The areas serviced by the safety-related HVAC system are heated by safety-related electrical duct heaters. Both the non-safety-related AHU (1A-AHU-1) and the safety-related supply and exhaust fans share a section of common ductwork to supply and exhaust these rooms. Interlocks are provided to ensure that both the non-safety-related AHU (1A-AHU-1) and the safety-related fans do not operate at the same time. Another non-safety-related AHU (1A-AHU-2) also serves the Maintenance Shop, hallway, Fuel Oil Storage Tank Room, and Future Expansion Room.

The HVAC System for the Station Blackout (SBO) Diesel Generator Building includes four augmented-quality fans, each thermostatically controlled, which are provided to exhaust air from the Diesel Generator Room. The SBO Diesel Generator Building HVAC System also includes an augmented-quality AHU (0C-AHU-1) to provide conditioned air to the Control Room. The augmented-quality AHU (0C-AHU-2) provides supply and exhaust ventilation to the Switchgear Room and only supply ventilation to the Battery Room, Cable Spreading Area, and Fuel Tank Room. The Cable Spreading Area, Fuel Tank Room, and Diesel Generator Room are exhausted by the Basement and Tank Room Exhaust Fan (0C-F-6). A separate fan (0C-F-5) provides exhaust ventilation for the Battery Room.

1.2.9.8 Plant Fire Protection System

The plant Fire Protection System is supplied with well water pumped from ground wells into storage tanks. Two fire pumps take suction from these tanks and supply fire protection systems, including sprinkler systems, deluge systems, hose stations and hydrants. In addition to water systems, gaseous and foam systems are provided to accommodate special requirements for various classes of hazards.

These systems were provided to fulfill requirements of the NRC, the Insurer, and corporate policy. The fire protection systems are designed following the guidance of the applicable National Fire Protection Association.

One electric motor-driven and one diesel engine-driven fire pump, each having a capacity of 2,500 gpm at a discharge pressure of 125 psig, take suction from the two 500,000-gallon capacity pretreated water storage tanks and discharge to the 12" fire main header. Each tank reserves 300,000 gallons explicitly for fire protection which cannot be withdrawn for other non-emergency plant uses. A jockey pump with a capacity of 30 gpm at a discharge pressure of 129 psig will maintain the fire system to make-up for minor losses. A make-up pump takes suction from plant service water mains and discharges 215 gpm at 125 psig to the fire system to meet intermittent usage of water for purposes other than fire protection.

Consideration is given to the use of noncombustible and fire-resistant materials throughout the facility, particularly in areas containing critical portions of the plant such as the Containment Structure, Control Room, and components of the ESF systems. Fire walls and stair towers are provided to segregate portions of the plant for safe ingress and egress.

Hydrants and hose stations are strategically located to provide primary and back-up protection. Automatic fire suppression systems consisting of water spray

sprinklers, pre-action sprinklers, deluge systems, Halon 1301 total flooding systems, and manually-actuated foam systems are provided for primary fire protection. In addition, automatic fire and smoke detection systems are installed to provide surveillance in safety-related and unattended locations. A sufficient number of portable extinguishers, placed at key locations, can be used for extinguishing limited magnitude fires.

1.2.9.9 Auxiliary Steam System

Normally, the plant heating steam is extracted from Unit 1 or 2 hot reheat. In case both units are not operating, heating steam is supplied to the plant hot water generators by one of two auxiliary boilers. Each oil-fired auxiliary boiler has a steam generating capability of 125,000 lbs/hr at a pressure of 180 psig when supplied with 180°F feedwater and approximately 10,000 lbs/hr of fuel oil.

The hot water generators are connected to the plant heating system. Each generator has a capacity of 25,050,000 Btu/hr when supplied with 27,067 lbs/hr of hot reheat steam or pressure reduced auxiliary boiler steam of 65 psig.

For a unit start-up, one of the auxiliary boilers also supplies steam for the unit pre-operational deaeration of the condenser hotwell condensate. The deaeration takes place by direct contact. The excess condensate generated by adding the auxiliary boiler steam to the main unit condensate is circulated to the auxiliary steam system deaerator. The auxiliary boiler feed pumps take suction from the storage portion of the deaerator and discharge the condensate back to the respective operating auxiliary boiler.

1.2.10 STEAM AND POWER CONVERSION SYSTEM

The turbine generator for Unit 1 is furnished by the General Electric Company and the Unit 2 turbine generator is furnished by the Westinghouse Electric Corporation. Each turbine is an 1,800 RPM tandem compound, six-flow exhaust, indoor unit.

Under nominal steam conditions of approximately 865 psia and 528°F at the stop-valves inlet, and with the turbines exhausting condenser pressure of 2" Hg absolute, the Unit 1 generator produces approximately 913,719 kW and the Unit 2 generator produces approximately 950,285 kW at the generator terminals. Turbine output corresponds to an NSSS thermal power level of approximately 2,700,000 kW (Unit 1) and 2,750,000 kW (Reactor Power plus RCP Heat Load – Unit 2).

The condensate and feedwater system of each unit consists of three condensate pumps, one gland steam condenser, five demineralize columns, six precoat filter columns, three external heater drain coolers, three first and second stage feedwater heaters, three condensate booster pumps, two third, fourth and fifth stage feedwater heaters, two turbine-driven feed pumps and two sixth stage feedwater heaters.

Normally, the feed pump turbines are driven by steam from the hot reheat. At low turbine generator loads, main steam or auxiliary steam is used to drive the feed pump turbines. All turbines exhaust into their respective unit condenser.

1.2.11 WASTE PROCESSING SYSTEMS

The Waste Processing Systems, which are shared by Units 1 and 2, provide controlled handling and disposal of liquid, gaseous and solid wastes. Gaseous and liquid waste discharges to the environment are controlled to comply with the limits set by 10 CFR Part 20.

a. Reactor Coolant Waste Processing System

Reactor coolant from the CVCS and from the reactor coolant drain tanks is processed by the Reactor Coolant Waste Processing System, which is comprised of filters, degasifiers, ion exchangers, evaporators, receiver tanks, and monitor tanks. The coolant is first purified by the filters, degasifiers, and ion exchangers.

The evaporators are used to reconcentrate the boric acid. The concentrate is normally returned to the boric acid storage tank, but if the activity is high, or if the solution is chemically unsuitable for reuse, the concentrate is processed in the solid Waste Processing System and transported to an offsite disposal facility. The distillate from the evaporators is monitored to ensure proper radioactivity limits are not exceeded and then discharged to the circulating water system.

b. Miscellaneous Waste Processing System

Miscellaneous liquid wastes from the Auxiliary Building are filtered and stored in the miscellaneous waste receiver tank. The miscellaneous waste ion exchanger is used to purify the miscellaneous waste before it enters the monitor tank. If the radioactivity level of the liquid in the monitor tank is found to be high, the waste can be recycled through the ion exchanger or sent to the Reactor Coolant Waste Processing System.

The liquid in the monitor tank is sampled to ensure proper radioactivity limits are not exceeded prior to discharge to the circulating water system.

c. Waste Gas Processing System

Waste gases are collected in the vent header and the waste gas surge tank. One of the two waste gas compressors is used to compress the gas for storage in one of the three waste gas decay tanks. After decay, the gas in the waste gas decay tanks is sampled to ensure proper radioactivity limits are not exceeded, and then is released to the plant vent at a controlled rate.

d. Solid Waste Treatment

Spent demineralizer resins, filters, and evaporator concentrates which are not to be recycled, are transported to the waste disposal bay. These resins are dewatered in accordance with the CCNPP Process Control Program and stored, onsite, or shipped in an appropriately shielded container to an offsite disposal facility, or offsite vendor for processing.

Low activity wastes (dry active waste) such as contaminated laundry, rags, and paper may be shredded or compacted and/or packaged for removal from the plant for burial or shipment to an offsite processor.

1.3 COMPARISON WITH OTHER PLANTS

Table 1-1 presents a summary of the original design characteristics of the CCNPP. The table includes similar data for Maine Yankee Unit 1, Turkey Point Units 3 and 4 and Palisades Unit 1. Bechtel Power Corporation (Bechtel) and CE are identified as contractors in Section 1.8. The Palisades plant is included in the table because its coolant system is similar to that of Calvert Cliffs, and because both Bechtel and CE were Palisades contractors. Maine Yankee is selected because its core is similar to that of Calvert Cliffs and it is a plant of vintage similar to Calvert Cliffs with which CE is associated. In particular, the reactor regulating system, the reactor coolant pressure regulating system, and the pressurizer level regulating system are essentially identical in design and function to the Maine Yankee systems except in the adaptation to the lesser number of similar inputs for a two-loop plant. Turkey Point is included because it is another comparable plant with which Bechtel is associated.

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

HYDRAULIC AND THERMAL DESIGN PARAMETERS	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
Total Core Heat Output, MWt	2560	2440	2200	2200
Total Core Heat Output, Btu/hr	8740×10^6	8328×10^6	7479×10^6	7509×10^6
Heat Generated in Fuel, %	97.5	97.5	97.4	97.5
Maximum Overpower, %	12	12	12	12
System Pressure, Nominal psia	2250	2250	2250	2100
System Pressure, Minimum Steady State, psia	2200	2200	2220	2050
Hot Channel Factors, Overall				
Heat Flux, Fq	3.00	2.89	3.23	3.80
Enthalpy Rise, F_{WH}	1.65	1.62	1.77	2.51
DNB Ratio at Nominal Conditions	2.18	2.45	1.81	2.00
Coolant Flow				
Total Flow Rate, lb/hr	122×10^6	122×10^6	101.5×10^6	125×10^6
Effective Flow Rate for Heat Transfer, lb/hr	117.5×10^6	117.5×10^6	97.0×10^6	121.25×10^6
Effective Flow Rate for Heat Transfer, ft^2	53.5	53.5	41.8	58.7
Average Velocity Along Fuel Rods, ft/sec	13.6	13.9	14.3	12.7
Average Mass Velocity, lb/hr-ft^2	2.20×10^6	2.9×10^6	2.32×10^6	2.07×10^6
Coolant Temperatures, °F				
Nominal Inlet	543.5	538.9	546.2	545
Maximum Inlet due to Instrumentation Error and Deadband, °F	548	546	550.2	548
Average Rise in Vessel, °F	52	51.1	55.9	46
Coolant Temperatures, °F				
Average Rise in Core, °F	54	53.1	58.3	47
Average in Core, °F	570.4	565.4	575.4	568.5
Average in Vessel	569.5	564.4	574.2	568
Nominal Outlet of Hot Channel	643	636	642	642.8
Average Film Coefficient, Btu/hr-ft^2-°F	5240	5300	5400	4860
Average Film Temperature Difference, °F	33.5	33	31.8	30

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
Heat Transfer at 100% Power				
Active Heat Transfer Surface Area, ft^2	48,416	47,700	42,460	51,400
Average Heat Flux, Btu/hr-ft^2	176,000	170,200	171,600	142,400
Maximum Heat Flux, Btu/hr-ft^2	527,900	502,300	554,200	541,200
Average Thermal Output, kW/ft	5.94	5.74	5.5	4.63
Maximum Thermal Output, kW/ft	17.5	16.7	17.9	17.6[c]
Maximum Clad Surface Temperature at Nominal Pressure, °F	657	657	657	648
Fuel Center Temperature, °F				
Maximum at 100% Power	3780	3640	4030	4040
Maximum at Over Power	4070	3940	4300	4350
CORE MECHANICAL DESIGN PARAMETERS				
Thermal Output, kW/ft at Maximum Over Power	19.6	18.7	20.0	19.7[c]
Fuel Assemblies				
Design	CEA	CEA	RCC	Cruciform
Rod Pitch, in.	0.58	0.580	0.563	0.550
Cross-section Dimensions, in.	7.98x7.98	7.98x7.98	0.563	8.1135x8.1135
Fuel Weight (as UO_2), lbs	207,269	203,934	176,200	210,524
Total Weight, lbs	282,570	279,235	226,200	295,800
Number of Grids per Assembly	8	8	7	8
Fuel Rods				
Number	36,896	36,352	32,028	43,168
Outside Diameter, in.	0.440	0.440	0.422	0.4135
Diametral Gap, in.	0.0085	0.0085	0.0065	0.0065
Clad Thickness, in.	0.026	0.026	0.0243	0.022
Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Fuel Pellets				
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered
Diameter, in.	0.3795	0.3795	0.367	0.359
Length, in.	0.650	0.650	0.600	0.600

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
Control Assemblies				
Neutron Absorber	B_4C/SS/Cd-In-Ag	B_4C/SS/Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%)
Cladding Material	Inconel	Inconel	304 SSO Cold Worked	Stainless
Clad Thickness, in.	0.040	0.040	0.019	0.016
Control Assemblies				
Number of Assemblies, full/part length	77/8	77/8	53	4 1/4 Cruciform Rods
Number of Rods per Assembly	5	5	20	117 Tubes per Rod
Core Structure				
Core Barrel ID/OD, in.	148/149.75	148/149.75	133.875/137.875	149.75/152.5
Thermal Shield ID/OD, in.	None	156/162	142.625/148.0	None
NUCLEAR DESIGN DATA				
Structural Characteristics				
Core Diameter, inches (Equivalent)	136.0	136.0	119.5	136.71
Core Height, inches (Active Fuel)	136.7	136.7	144	132
Reflector Thickness & Composition				
Top - Water plus steel, in.	10	10	10	10
Bottom - Water plus steel, in.	10	10	10	10
Side - Water plus steel, in.	15	15	15	15
H_2O/U, Unit Cell (Cold)	3.44	3.44	4.18	3.50
Number of Fuel Assemblies	217	217	157	204
UO_2 Rods per Assembly, unshimmed/shimmed			204	212/208
Batch A	176	176		
Batch B	164	160		
Batch C	(176/164/164)	(176/164/160)		
Structural Characteristics				
Performance Characteristics Loading Technique	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	3 Regions Non-Uniform	3 Batch Mixed Central Zone
Fuel Discharge Burnup, MWD/MTU				
Average First Cycle	13,775	13,795	13,000	10,180
First Core Average	22,550	30,000	24,500	17,600

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
Feed Enrichments wt%				
Region 1	2.05	2.01	1.85	1.65
Region 2	2.45	2.40	2.55	2.08/2.54
Region 3	2.99	2.95	3.10	2.54/3.20
Control Characteristics				
Effective Multiplication (beginning of life)				
Cold, No Power, Clean	1.194	1.170	1.180	1.212
Hot, No Power, Clean	1.152	1.129	1.38	1.175
Hot, Full Power, Xe Equilibrium	1.094	1.075	1.077	1.111
Control Assemblies Material	B_4C/SS-Cd-In-Ag	B_4C/SS-Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%)
Number of Control Assemblies	85	85	53	45 Cruciform
Number of Absorber Rods per CEA (or RCC) Assembly	5	5	20	117 Tubes Welded to Form 13.5 in. Span
Total Rod Worth (Hot), %	≥9.6	≥9.9	7	8.6
Boron Concentrations				
To shut reactor down with no rods inserted, clean, Cold/Hot, ppm	1120/1095	945/935	1250/1210	1180/1210
To control at power with no rods inserted, clean/equilibrium xenon, ppm	960/725	820/590	1000/670	1070/830
Kinetic Characteristics, Ranger Over Life Moderator Temperature Coefficient k/k/F	-0.20×10^{-4} to -1.96×10^{-4}	-0.04×10^{-4} to 2.20×10^{-4}	$+0.3\times10^{-4}$ to -3.5×10^{-4}	-0.08×10^{-4} to -2.25×10^{-4}
Moderator Pressure Coefficient, /psi				
Hot, Operating				
Beginning-of-Life				
End-of-Cycle	$+0.3\times10^{-6}$ to 2.6×10^{-6}	$+0.65\times10^{-6}$ to $+2.39\times10^{-6}$	-0.3×10^{-6} to $+3.4\times10^{-6}$	$+0.10\times10^{-6}$ to $+1.7\times10^{-6}$

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
Moderator Pressure Coefficient, Void, /% Void				
Hot, Operating				
Beginning-of-Life	$-0.1x10^{-3}$	$-0.41x10^{-3}$ to	$+0.5x10^{-3}$ to	$0.06x10^{-3}$ to
End-of-Cycle	$-1.3x10^{-3}$	$1.43x10^{-3}$	$-2.5x10^{-3}$	$-1.0x10^{-3}$
Doppler Coefficient[d]	$-1.46x10^{-5}$	$-1.45x10^{-5}$	$-1.0x10^{-5}$	$-1.56x10^{-5}$
k/k/F	$-1.06x10^{-5}$	$-1.07x10^{-5}$	$1.6x10^{-5}$	$-1.46x10^{-5}$
REACTOR COOLANT SYSTEM - CODE REQUIREMENTS				
Reactor Vessel[f]	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Steam Generator				
Tube Side	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Shell Side	ASME III Class A	ASME III Class A	ASME III Class C	ASME III Class A
Pressurizer	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class C
Pressurizer Relief (or Quench) Tank	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C
Pressurizer Safety Valves	ASME III	ASME III	ASME III	ASME III
Reactor Coolant Piping	ANSI B 31.7	ANSI B 31.1	ANSI B 31.1	ANSI B 31.1
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL				
Operating Pressure, psig	2235	2235	2235	2085
Reactor Inlet Temperature, °F	544.5	540	546.2	545
Reactor Outlet Temperature, °F	599.4	592.8	602.1	591.1
Number of Loops	2	3	3	2
Design Pressure, psig	2485	2485	2485	2485
Design Temperature, °F	650	650	650	650
Hydrostatic Test Pressure (cold), psig	3110	3110	3107	3110
Total Coolant Volume – cu.ft.	11,101	11,026	9,088	10,809

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS				
Material	SA-533, Grade B, Class 1, low alloy steel, internally clad with Type 304 austerities SS equivalent	SA-533, Grade E Class 1 steel, forgings-A-508-64 Class 2, cladding-weld deposited 304 SS equivalent	SA-302, Grade B, low alloy steel internally clad with type 304 austerities SS equivalent	SA-302, Grade B low alloy steel internally clad with Type 304 austerities SS equivalent
Design Pressure, psig	2485	2485	2485	2485
Design Temperature, °F	650	650	650	650
Operating Pressure, psig	2235	2235	2235	2085
Inside Diameter of Shell, in.	172	172	155.5	172
Outside Diameter Across Nozzles, in.	253	266-5/8	236	254
Overall Height of Vessel and Enclosure Head to Top of CRDM Nozzle, ft.-in.	41-11-3/4	42-1-3/8	41-6	40-1-13/16
Minimum Clad Thickness, in.	1/8	1/8	5/32	3/16
Number of Units	2	3	3	2
Type	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator
Tube Material	Inconel	Inconel	Inconel	Inconel
Shell Material	SA-533, Gr. B, Class 1 and SA-516 Gr 70	SA-533, Gr. B, Class 1 and SA-516 Gr 70	Carbon Steel	Carbon Steel
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS				
Tube Side Design Pressure, psig	2485	2485	2485	2485
Tube Side Design Temperature, °F	650	650	650	650
Tube Side Design Flow, lb/hr	61×10^6	40.67×10^6	33.93×10^6	62.5×10^6
Shell Side Design Pressure, psig	985	985	1085	985
Shell Side Design Temperature, °F	550	550	556	550
Operating Pressure, Tube Side, Nominal, psig	2235	2235	2235	2085
Operating Pressure, Shell Side, Maximum, psig	885	885	1020	885
Maximum Moisture at Outlet at Full Load, %	0.2	0.2	1/4	0.2

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[(a)]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[(b)]	TURKEY POINT UNITS 3 & 4[(b)]	PALISADES UNIT 1
Hydrostatic Test Pressure, Tube Side (cold), psig	3110	3110	3107	3110
Steam Pressure, psia, at full power	850	815	745	770
Steam Temperature, °F, at full power	525.2	520.3	510	513.8
Number of Units	4	3	3	4
Type	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge
Design Pressure, psig	2485	2485	2485	2485
Design Temperature, °F	650	650	650	650
Operating Pressure, nominal psig	2235	2235	2235	2085
Suction Temperature, °F	543.4	538.9	546.5	545
Design Capacity, gpm	81,200	108,000	89,500	83,000
Design Head, ft.	300	290	260	260
Hydrostatic Test Pressure, (cold), psig	3110	3110	3107	3110
Motor Type	A-C Induction Single Speed	A-C Induction Single Speed	A-C Induction Single Speed	A-C Induction Single Speed
Motor Rating, hp	7200 (cold)	9000	6000	6250 (cold)
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING				
Material	SA516-gr 70 with SS clad	SA516-gr 70 with SS clad	Austenitic SS	SA516-gr 70 clad with SS
Hot Leg - ID, in.	42	33.5	29	42
Cold Leg - ID, in.	30	33.5	27 1/2	30
Between Pump & Steam Generator - ID, in.	30	33.5	31	30

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[(a)]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[(b)]	TURKEY POINT UNITS 3 & 4[(b)]	PALISADES UNIT 1
CONTAINMENT SYSTEM PARAMETERS UNIT 1				
Type	Steel-lined, pre-stressed post tensioned concrete cylinder, curved dome roof	Steel-lined, reinforced concrete flat bottom and hemispherical dome	Steel-lined pre-stressed post tensioned concrete cylinder, hemispherical domed roof	Steel-lined pre-stressed post tensioned concrete cylinder, hemis-cylinder curved dome roof
Design Parameters				
Inside Diameter, ft.	130	135	116	116
Height, ft.	181-2/3	169-1/2	169	190
Free Volume, ft^3	2,000,000	1,855,000	1,550,000	1,600,000
Reference Incident Pressure, psig	50	55	59	55
Concrete Thickness, ft.				
Vertical Wall	3-3/4	4-1/2	3-3/4	3
Dome	3-1/4	2-1/2	3-1/4	2-1/2
Containment Leakage Prevention & Mitigation Systems	Leak-tight pene-tration & continuous steel liner. Automatic isolation where required. The exhaust from penetration rooms to vent	Leak-tight pene-tration & continuous steel liner. Automatic isolation where required.	Leak-tight pene-tration & continuous steel liner. Automatic isolation where required.	Leak-tight pene-tration & continuous steel liner. Automatic isolation where required.
Gaseous Effluent Purge	Discharge through vent	Discharge through stack	Through particulate filter and monitors part of main exhaust system.	Discharge through stack

TABLE 1-1

COMPARISON OF ORIGINAL PLANT CHARACTERISTICS[a]

	CALVERT CLIFFS UNITS 1 & 2	MAINE YANKEE[b]	TURKEY POINT UNITS 3 & 4[b]	PALISADES UNIT 1
ENGINEERED SAFETY FEATURES				
Safety Injection System				
No. of High Head Pumps	3	3 (Charging)	4 (Shared)	3
No. of Low Head Pumps	2	2	2	2
Containment Fan Coolers				
No. of Units	4	6	3	3
Air Flow Capacity, each at emergency condition, cfm	55,000	Not Applicable	25,000	25,000
Containment Spray No. of Pumps	2	3	2	2
Emergency Power Diesel Generator Units	3 total for both units[e]	2	2 total for both units	4
Safety Injection Tanks, Number	4	3	3	4

(a) The current design characteristics for Calvert Cliffs Units 1 and 2 may differ from those shown in this table.

(b) The values listed for these plants were taken from public documentation.

(c) Based on total heat output of the core rather than heat generated in the fuel alone.

(d) Values shown are for hot, zero power/beginning-of-life, full power conditions.

(e) The current design characteristics show four total emergency power diesel generators for both units.

(f) See Table 4-9 for the design code of the replacement RVCH.

1.4 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR DESIGN

The principal architectural and engineering criteria for design of the plant are summarized below.

1.4.1 PLANT DESIGN

Principal structures and equipment which may serve either to prevent incidents or to mitigate their consequences are designed, fabricated and erected in accordance with applicable codes to withstand the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature, and other deleterious natural phenomena which could be reasonably assumed to occur at the site during plant lifetime. Units 1 and 2 are sufficiently independent so that the safety of one unit will not be impaired in the unlikely event of an incident in the other unit. Principal structures and equipment are sized for the maximum expected NSSS and turbine outputs.

Redundancy is provided in reactor and safety systems so that no single failure of any active component of the systems can prevent the action necessary to avoid an unsafe condition. The plant is designed to facilitate inspection and testing of systems and components whose reliability is important to the protection of the public and plant personnel.

Provisions are made to protect against the hazards of such events as fires or explosions.

Systems and components which are significant from the standpoint of nuclear safety are designed, fabricated, and erected to quality standards commensurate with the safety function to be performed.

1.4.2 REACTOR

The following apply to the reactor of either unit:

a. The reactor is of the pressurized water type, designed to produce steam to drive a turbine generator. The reactor was initially licensed and operated at a core thermal output of 2,560 MWt; the license was later amended and the reactor now operates at 2,737 MWt.

b. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy, ZIRLO, or M5® tubes.

c. Minimum departure from nucleate boiling ratio (DNBR) during normal operation and anticipated transients will not be below that value which could lead to fuel rod failure. The maximum fuel center line temperature evaluated at the design overpower condition will be below that value which could lead to fuel rod failure. The melting point of the UO_2 will not be reached during normal operation and anticipated transients.

d. Fuel rod clad is designed to maintain cladding integrity throughout fuel life. Fission gas release within the rods and other factors affecting design life are considered for the maximum expected exposures.

e. The reactor and control systems are designed so that any xenon transients will be adequately damped.

f. The reactor is designed to accommodate the anticipated transients safely and without fuel damage.

g. The RCS is designed and constructed to maintain its integrity throughout expected plant life. Appropriate means of test and inspection are provided.

h. Power excursions which could result from any credible reactivity addition incident will not cause damage to the pressure vessel either by deformation or rupture, or impair operation of the ESF.

i. Control element assemblies are capable of holding the core subcritical at hot zero power conditions with adequate margin following a trip, even with the most reactive rod stuck in the fully withdrawn position.

j. The CVCS is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain an adequate shutdown margin during maximum design rate RCS cooldown following a reactor trip. The system is independent of the CEA system.

k. The combined response of the fuel temperature coefficient, the moderator temperature coefficient (MTC), the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

l. Automatic and redundant reactor trips are provided to prevent anticipated plant transients from producing fuel or clad damage.

1.4.3 REACTOR COOLANT AND AUXILIARY SYSTEMS

Heat removal systems are provided which can safely accommodate core heat output. Each of these heat removal systems is designed to provide reliable operation under all normal and expected transient circumstances.

1.4.4 CONTAINMENT STRUCTURE

The Containment Structure, including the associated access openings and penetrations, is designed to contain the pressures and temperatures resulting from a LOCA in which the following occur:

a. The total energy contained in the RCS water is assumed to be released into the containment through a double-ended break of one of the reactor coolant pipes adjacent to the reactor vessel outlet nozzle.

b. External electric power is lost simultaneously.

c. Heat is transferred from the reactor to the containment by water supplied from the Safety Injection System.

d. Either the containment air recirculation subsystem or the containment spray subsystem functions.

e. The containment ESF do not operate until 30 seconds following the incident.

Means are provided for pressure and leak rate testing of the entire containment system including provisions for leak rate testing of individual piping and electrical penetrations that rely on gasketed seals, sealing compounds, or expansion bellows.

1.4.5 ENGINEERED SAFETY FEATURES

The design for either unit incorporates redundant ESF systems. These, in conjunction with the containment systems, ensure that the release of fission products, following any credible LOCA, will not exceed the guidelines set forth in 10 CFR 50.67. The ESF systems include: (a) independent systems, each with redundant features, to remove heat from the Containment Structure in order to reduce containment pressure; (b) a Safety Injection System to limit fuel and cladding damage to an amount which would not interfere with adequate emergency core cooling (ECC) and to limit metal-water

reactions to negligible amounts; and (c) a system to remove radioactive iodine for the post-incident containment atmosphere. The ESF are designed for all break sizes in the RCS piping up to and including the double-ended rupture of the largest reactor coolant pipe.

1.4.6 PROTECTION, CONTROL, AND INSTRUMENTATION SYSTEMS

Interlocks and automatic protective systems are provided along with administrative controls to insure safe operation of the plant.

An RPS is provided which initiates reactor trip if the reactor approaches an unsafe condition.

Sufficient redundancy is installed to permit periodic testing of the RPS so that failure or removal from service of any one protective system component or portion of the system will not preclude reactor trip or other safety action when required.

1.4.7 ELECTRICAL SYSTEMS

Normal, standby, and emergency sources of auxiliary power are provided to assure both the safe and orderly shutdown of the plant and the ability to maintain a safe shutdown condition under all credible circumstances.

1.4.8 WASTE PROCESSING AND RADIATION PROTECTION

The waste treatment systems are designed so that the discharge of radioactivity to the environment is in accordance with the requirements of 10 CFR Part 20.

The plant is provided with a centralized Control Room having adequate shielding to permit occupancy during all credible accident conditions.

The radiation shielding in the plant and the radiation control procedures ensure that operating personnel do not receive radiation exposures in excess of the applicable limits of 10 CFR Part 20 during normal operation and maintenance.

1.4.9 FUEL HANDLING AND STORAGE

Fuel handling and storage facilities are provided for the safe handling, storage and shipment of fuel and will preclude accidental criticality.

1.4.10 NIL DUCTILITY TRANSITION TEMPERATURE

Components of the RCS are designed and will be operated so that no deleterious pressure or thermal stress will be imposed on the structural materials. Consideration is given to the ductile characteristics of the materials at low temperature.

1.4.11 FIELD RUNNING OF 2" AND SMALLER DIAMETER PIPE

All 2" and smaller piping with the exception of portions of the charging, letdown, and some few branch connections tied into the Safety Injection System (these systems are classified in ANSI B 31.7 Class I and are routed by the engineering office) were field-run during plant construction and initial modification work.

All piping for field-run essential systems (Section 1.8.1, II.E.4.2), including all ESF, were routed by experienced piping designers. This piping was routed and support points selected in accordance with the field installation manual. The spacing of supports and type of support used are such that the combination of stresses due to thermal, dead load, and seismic does not exceed the allowable stresses. Prior to installation, all field piping isometric drawings were routed to the engineering office for

comments. After installation and before start-up, surveillance by the field quality assurance personnel and by the respective engineering specialist group was performed to ensure that all piping is installed per the design drawings. Prior to start-up, each system was checked off showing that all hangers and supports are located as designed.

During start-up, each system was observed under conditions which simulate operating conditions. This included the starting and stopping of pumps and opening and closing of valves.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

1.5.1 INTRODUCTION

The design of the Calvert Cliffs Nuclear Plant is based upon concepts which have been successfully applied in the design of pressurized water reactor power plants. However, certain programs of theoretical analysis or experimentation (constituting "research and development" as defined in the Atomic Energy Act, as amended, and in AEC or NRC regulations) have been undertaken to aid in plant design and to verify the performance characteristics of plant components and systems. This section describes the results and status of those analytical and test programs which were conducted or in progress at the time of operating license application including experimental production and testing of models, devices, equipment and materials. No attempt is made to include all pertinent research and development programs conducted since operating license application.

In carrying out these programs, information which is derived from research and development activities of the AEC or NRC and other organization in the nuclear industry has been taken into account.

1.5.2 CEA TESTING

1.5.2.1 Critical Experiments

An experimental program was completed to confirm techniques for calculating CEA worth and local nuclear peaking associated with the fuel assembly design. The work was performed in the CRX facility of the Westinghouse Reactor Evaluation Center at Waltz Mill, Pennsylvania, between June and August 1967. The basic core configuration was a 30x30 square array of Zr-4 clad UO_2 fuel rods with an enrichment of about 2.7 wt% U-235; fuel rods were removed to create internal water holes or channels to accommodate absorber elements.

The experiments demonstrated that the current CE methods of analysis (Section 3.0) accurately predict the CEA worth and local peaking in the small critical assemblies. This lends support to the use of these methods in the design of the Calvert Cliffs core.

The significant conclusions which were drawn on the basis of the test program include the following:

a. The standard CE design methods are capable of calculating clean, room temperature, critical lattices (lattices which contain no CEAs, water slots, or other heterogeneities) to an accuracy of within 0.03% reactivity on the average;

b. The worths of various arrays of cylindrical absorbers containing boron carbide were predicted within 2% of the worth on the average, with errors for individual cases ranging from +6% to -2.2% of the worth. The arrays had worths ranging from 6 to 8% reactivity;

c. In assemblies containing water holes, the calculated and measured power peaking agree within 2%. Occasional differences between calculated and measured power of up to 4% are seen, but only in fuel rods of low power, usually near the reflector of these highly buckled, small cores.

1.5.2.2 Mechanical Testing

A series of tests were completed on single and dual CEAs to satisfy the following objectives:

a. To determine the mechanical and functional feasibility of the CEA concept;

b. To experimentally determine the relationship between CEA drop time and CEA drop weight, annular clearance between CEA fingers and guide tubes, and coolant flowrate within the guide tube;

c. To experimentally determine the relationship between flowrate and pressure drop within the guide tube as a function of CEA axial position and of finger-to-guide tube clearance;

d. To determine the effects on drop time of adding a flow restriction or of plugging the lower end of a guide tube (as might occur under accident conditions);

e. To determine the effects of misalignment within the CEA guide tube system on drop time.

Both types of CEAs were tested by operation for over 1000 hours at reactor operating temperature, pressure, and water chemistry with flows in excess of those anticipated in the reactor. This test and other tests demonstrated that the five-finger CEA concept is mechanically and functionally feasible and that the CEA meets criteria established for drop time under the most adverse condition. Both types of CEAs were examined after conclusion of these tests and no significant wear was observed. At no point were there any wear marks in excess of .001" in depth. The testing also verified that the analytical model used for predicting drop time gives uniformly conservative results.

The effects on drop time of all possible combinations of frictional restraining forces in the CEDM, angular and radial misalignment of the CEDM, misalignments between the CEA and the guide tubes of as much as 0.4", misalignments of the CEA was experimentally investigated and defined. The conditions tested simulated all the effects of tolerance buildup, dynamic loadings, and thermal effects. The tests demonstrated that misalignments and distortions in excess of those expected from tolerance buildup or any other anticipated cause would still result in acceptable drop times.

The results of the cold CEA testing were analytically extended into the hot operating range. Further testing of a design verification nature was performed scheduled on a complete CEDM-CEA system in the hot test facility at operating plant temperature, pressure, and flow.

1.5.3 CONTROL ELEMENT DRIVE MECHANISM TESTING

The development of the magnetic jack CEDM was carried out over a period of approximately three years. As such, it was a closely coupled program consisting of design, testing and fabrication. The early design effort resulted in a prototype which was tested extensively and was modified during the testing process to improve operation. The results of this early testing led to the design and fabrication of a second improved prototype mechanism. This mechanism was again tested extensively in a manner similar to the first, and from this series of tests coupled with design modifications that Maine Yankee prototype mechanism resulted. The specifications for the Calvert Cliffs CEDM was similar to that used for the construction of the Maine Yankee mechanisms.

The following design requirements were imposed on the CEDMs with regard to operation under severe service conditions:

a. The pressure housing of the CEDM, for the replacement RVCH, is designed for service as a Class 1 appurtenance and code stamped for

service at 2,500 psia and 650°F. Normal CEDM operating conditions are 2,250 psia and 608°F.

b. The CEDMs are designed to withstand the combined mechanical loads associated with normal and abnormal operating pressures, temperatures and transients, plus the maximum earthquake, with no loss of function.

c. When DBE reactor coolant pipe rupture loads are added to the loads given in (b) above, the CEDM need not function normally during the application of the DBE loads, but must prevent ejection of its CEA from the core and be capable of normal operation following the DBE.

d. The CEDM is capable of performing its normal function after an inactive period of one month with the CEDM in the hold mode and the plant at normal pressure and temperature.

e. Under the ambient conditions inside the containment building following a postulated main pipe rupture (DBE), the CEDM is capable of driving the CEA to fully inserted position from the fully withdrawn position, and transmitting all position indication signals for 15 minutes after rupture occurs.

f. The CEDM is capable of withstanding complete loss of cooling service for a four-hour period with the plant at normal operating temperature and pressure. CEDM operation under these conditions is restricted to the "HOLD" and "SCRAM" modes. Upon restoration of cooling service the CEDM is capable of normal operation.

The CEDM must be capable of passing the following tests:

a. Single CEA - Accelerated life tests of at least 30,000' of travel and 200 full-height gravity drops at simulated reactor operating conditions and ambient external conditions.

b. Dual CEA - Accelerated life test of at least 15,000' of travel and 200 full-height gravity drop tests at simulated reactor operating conditions and ambient external conditions.

After installation and prior to operation, each CEDM was tested in the field to ascertain that the system, as constructed, meets all of the design requirements, as discussed in Section 13.1.

1.5.4 FUEL ASSEMBLY DESIGN

1.5.4.1 Prototype Tests

Full size Zircaloy fuel assemblies filled with depleted UO_2 have undergone high temperature (600°F), high pressure (2,250 psig) full flow tests in typical reactor chemistry coolant. These tests were performed in connection with a full-size CEA and a CEDM.

Full-size prototype fuel assemblies loaded with depleted UO_2 were subjected to mechanical testing to evaluate their reaction to applied loads. Axial and lateral loading of assemblies supported in air between simulated upper and lower support plates, as well as free end twisting and lateral motion typical of refueling operation, were performed. Previous testing of a similar nature on CE fuel assemblies had shown them to be stable and mechanically sound for all expected reactor operating, casualty and refueling conditions. These tests were repeated for the Calvert Cliffs type fuel assemblies to verify that no significant mechanical characteristics changed.

In 1966, a series of single-phase tests on coolant turbulent mixing was run on a "prototype" fuel assembly which was geometrically similar to the Palisades assembly. The model enabled determination of flow resistances and vertical subchannel flow rates using pressure instrumentation and the average level of eddy flow using dye-injection and sampling equipment. The tests yielded the value of inverse Peclet number characteristic of eddy flow (0.00366). The value was shown during the course of the tests to be insensitive to coolant temperature and to vertical coolant mass velocity. The design value of the inverse Peclet Number was established at 0.0035 on the basis of the experimental results.

As part of a CE-sponsored research and development program, a series of single phase dye injection mixing tests were conducted in 1968. The tests were performed on a model of a portion of a CEA-type fuel assembly which was sufficiently instrumented to enable measurement (via a data reduction computer program) of the individual lateral flows across the boundaries of twelve subchannels of the model. Although these tests were not intended for that purpose, some of the test results could be used to determine the average level of turbulent mixing in the reference design assembly. The inverse Peclet number calculated from the average of 56 individual turbulent mixing flows (two for each subchannel boundary) obtained from the applicable data was 0.0034. With respect to general turbulent mixing, therefore, the more recent study on the CEA assembly verifies the constancy of the inverse Peclet number for moderately different fuel assembly geometries and confirms the design value of that characteristic.

1.5.4.2 Assembly Flow Distribution Tests

Velocity and static pressure measurements were made in an oversize model of a CEA fuel assembly in order to determine the flow distributions present in that geometry. The effect of the distributions on thermal behavior and margin were evaluated, where necessary, with the use of CE's CORAL code, which is an extensively revised version of the COBRA thermal and hydraulic code. Subjects investigated include the following:

a. Assembly inlet flow distribution, as affected by the core support plate and lower end fitting flow hole geometry. The flow distribution was measured and indicated that the desirable uniform condition is achieved within 10% of core height. The effect of the initial non-uniform condition on thermal behavior was analyzed;

b. Assembly inlet flow distribution as affected by a blocked core support plate flow hole. The flow distribution was measured and indicated that flow has recovered to at least 50% of the uniform nominal value at an elevation corresponding to 10% of core height. The effect of the non-uniform flow pattern on thermal behavior was analyzed;

c. Flow distribution within the assembly, as affected by complete blockage of one to nine subchannels. The flow distributions were measured and indicated very little upstream effect of such blockage, followed by recovery to normal subchannel flow conditions within 10 to 15% of core height, depending upon the number of subchannels blocked;

d. Flow distribution below the upper end fitting as affected by the upper end fitting and fuel bundle alignment plate flow hole geometry and by the presence of the CEA shroud. Measurement of the flow pattern in the absence of the shroud showed no appreciable upstream effect of the flow holes in the active core region.

1.5.4.3 DNB Testing on the Mark V CEA Fuel Assembly, 1969-1970

In 1968, CE initiated a series of tests at Columbia University on the departure from nucleate boiling (DNB) phenomenon. One purpose of the tests was to obtain experimental DNB data for verifying the combined accuracy of the thermal and hydraulic COSMO design code and the empirical W-3 DNB correlation in predicting the DNB condition for the CEA fuel assembly.

The tests were conducted on a nine-foot long exact scale portion of a Mark Volt CEA fuel assembly, consisting of one guide tube and 21 electrically-heated "fuel" rods arranged five-by-five. There were three distinct test sections, one with a 7' heated length and a uniform lateral power distribution, and one with a 4' heated length and a non-uniform lateral power distribution. The axial power distribution was uniform for all test sections. Test conditions comprised a coolant inlet temperature range of 450°F to 650°F, a mass velocity range of $1x10^6$ to $3x10^6$ lb/hr-ft^2 and a system pressure range of 1,500 to 2,200 psia.

Approximately 90 data points were obtained from all three test sections. The COSMO/W-3 combination was used for predicting the corresponding Critical Heat Flux values for the experimental conditions. The measure of the accuracy of prediction was defined as the average value of the ratio of experimental to predicted Critical Heat Flux. The value was 0.983 with a sample standard deviation of 0.58; these compare satisfactorily with corresponding values in the literature. The result implies that COSMO and W-3 are acceptable by present standards for describing DNB in the CEA geometry.

The remainder of the DNB analytical and experimental program was devoted in part to further aspects of predicting DNB for the reference design assembly. This program was comprised of:

a. Refinement of the W-3 correlation or development of a new correlation to reduce the statistical error attendant on the prediction;

b. Investigation of the case of the small systematic deviation;

c. Investigation of DNB behavior over a wider range of system pressure and flow conditions.

1.5.4.4 Dynamic Loop and Vibration Testing

Considerable testing was performed to evaluate the effects of assembly and fuel rod vibration or fretting. Dynamic loop testing under simulated reactor operating conditions and mechanically-induced autoclave vibration tests were carried out.

Over 18,000 hours of test time were accumulated on subsize assemblies and over 14,000 hours on full-size test assemblies in dynamic test loops. Test conditions duplicated reactor temperature, water chemistry, pressure, and flow velocity. Intentional cross flow and forced bundle vibrations were used to accentuate any vibration between the fuel rods and the spacer grids. In addition, the spacer grid spring tabs were individually set to simulate relaxed spring conditions.

In addition to the dynamic loop tests, forced vibration tests were also performed. Fuel rods supported by spacer grids were vibrated at various frequencies and amplitudes. The tests were conducted in a static autoclave at operating pressure and temperature. Test variables included, in addition to the vibration frequency and amplitude, spring preset of the spacer grids, and time under test. The spring tabs were varied from design interference fits to gaps. These tests did not

reproduce reactor flow conditions but were designed to develop trends in the degree of fretting as a function of the test variables.

Even under unreasonably severe conditions (i.e., high frequencies, large amplitudes, and gaps between the grid spring and fuel tube) no serious fretting was observed in these tests.

1.5.5 MODERATOR TEMPERATURE COEFFICIENT

Analytical studies were completed as part of the detailed plant design to define the least negative MTC for the Calvert Cliffs reactor. The factors which affect the MTC are discussed in Section 3.0 of this report.

Analyses of MTC for the Connecticut Yankee reactor compared with measurements made during the course of the start-up experiments are shown Table 1-2. It will be observed from the data that the measured coefficient is at most 0.16×10^{-4} $\Delta\rho/^{\circ}F$ more positive than the calculated value. This good agreement lends confidence in the ability of the methods used to predict MTCs.

1.5.6 FUEL ROD CLADDING

A substantial amount of information was generated in the course of CE's continuing test program on Zr-4 cladding.

Creep collapse tests on unsupported Zr-4 specimens with t/OD ratios of between 0.050 and 0.071 were performed at 650°F and 750°F. All tests were performed at an external pressure of 2400 psia. Results of tests performed at 750°F show that specimens with a t/OD of 0.059 (the reference design of Calvert Cliffs) collapse between 100 and 1000 hours at this temperature. Tests conducted at 650°F show collapse will occur between 5000 and approximately 30,000 hours.

Zircaloy-4 specimens supported with plenum springs and mandrels with machined defects to simulate chipping and separated pellets were creep collapse tested at 750°F. Zircaloy-4 specimens with plenum springs were creep collapse tested at 650°F. Tests at 650°F after 19,000 hours showed no indications of the cladding deforming into the spaces between the springs. Specimens tested at 750°F accumulated in excess of 10,000 hours and the cladding showed some deformation into the intentional defects. Grooves 3/16" wide representing separated pellets caused a maximum of 2.0 mils deformation of the clad into the groove. Grooves 1/16" wide showed no measurable deformation after 10,000 hours at 750°F.

Long-term corrosion tests were performed on Zr-4 fuel cladding under simulated reactor coolant conditions. Results of weight gain and hydrogen pickup were evaluated with respect to results obtained using demineralized water. The tests were conducted in coolant which had approximately 1100 ppm boron and less than 10 ppm NH_4OH added for pH control. Results after 12,000 hours of test at 650°F showed no effect of the coolant additives on corrosion rates or hydrogen pickup over similar tests conducted in demineralized water. Specimens tested included as-received Zr-4 tubing and 750°F steam autoclaved samples. Similar tests performed using LiOH as a pH control additive accumulated in excess of 8000 hours. Under the test conditions, the corrosion characteristics exhibited by the material were equivalent to those observed in demineralized water.

Dynamic corrosion tests were also conducted at 600°F to determine the effect of contamination on non-autoclaved material corrosion rates. Samples intentionally contaminated with dilute acid and machine oils were tested. Results after 4,300 hours of

tests showed no deleterious effect of these conditions on the corrosion behavior of the Zr-4. Samples tested included as-received and 750°F steam autoclaved Zr-4 tubing.

In addition to the corrosion studies mentioned above, CE participated in two studies being conducted by American Society for Testing and Materials (ASTM).

The first study, "Corrosion Testing Zr-4 and its Effect on Hydrogen Absorption," was performed by the ASTM G Committee (Corrosion of Metals), Subcommittee No. VIII (Corrosion of Zirconium in Water Systems).

The second study, "Task Force on Hydride Orientation," was performed by the ASTM B10 Committee (Reactive and Refractory Metals and Alloys), Subcommittee No. II (Zirconium and Hafnium).

Areas of concern to the committee included methods of hydriding and effect of fuel tubing fabrication on platelet orientation.

Calvert Cliffs began phasing in the Westinghouse ZIRLO cladding starting with Unit 1 Cycle 16 (Batch 1V). The AREVA M5® cladding is first used beginning with Unit 2 Cycle 19 (Batch 2Z) and Unit 1 Cycle 21 (Batch AB).

1.5.7 REACTOR VESSEL FLOW TESTS

Tests were conducted with one-fifth scale models of CE reactors to determine hydraulic performance. The first tests were performed for the Palisades plant which has a RCS similar to that of Calvert Cliffs. The test investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps operating and various part-loop configurations. Air was used as the test medium.

Similar one-fifth scale model tests were performed for Maine Yankee, which has a core similar to that of Calvert Cliffs. These tests were conducted in a cold water loop. All components for the model were geometrically similar to those in the reactor except for the core where 217 cylindrical core tubes were substituted for the fuel bundles. The core tubes contained orifices to provide the proper axial flow resistance.

Combustion Engineering, Inc. also conducted tests on a one-fourth scale model of the Fort Calhoun reactor using air as the test medium.

Flow characteristics for Calvert Cliffs were determined by taking into consideration similarities between Calvert Cliffs and other CE reactors, in conjunction with the experimental data from the flow model programs.

1.5.8 INCORE INSTRUMENTATION TESTS

Tests on incore thermocouples and flux detectors were performed to insure that the instrumentation will perform as expected at the temperature to be encountered and that it does not excessively vibrate and cause excessive wear or fretting. Cold flow testing was completed; no adverse vibrations or wear effects were encountered. Hot flow testing was also completed; after 2,000 hours at 590°F and 2,100 psig in a test loop, no breach of mechanical integrity was observed.

Mechanical tests of the insertion and removal equipment and instrumentation were performed to determine the necessary forms and procedures. The top entry incore instrumentation design provides a means of eliminating the need for handling instrument assemblies separately, thus minimizing down-time and personnel exposure. A full scale

mock-up was built to accommodate three incore instrumentation thimble assemblies. Major components and subassemblies of the mock-up included:

a. An incore instrumentation test assembly, including the upper guide structure support plate, three thimble guide sleeves, fuel alignment plate, three fuel bundle guide tubes, and the core support plate.

b. A thimble assembly consisting of the instrument plate, three incore instrumentation thimbles and the lifting sling.

c. An upper guide tube, with the guide tube attached to the thimble extension and the detector cable partially inserted in the guide tube.

Insertion and withdrawal tests were performed to determine the frictional forces of a multi-tube instrument thimble assembly during insertion and withdrawal from a set of fuel bundles. This test simulated the operation that will be performed during the refueling of the reactor. To determine whether jamming of the thimbles would occur during this operation, bending loads were applied to the thimble assembly by tilting the instrument plate 0.5° increments up to a total of 5° from horizontal. Guide tubes were filled with water. The assembly was raised and lowered approximately five times for each tilt setting. Results showed no discernible difference in the friction forces for the various tilt settings, however, the friction forces varied during withdrawal and insertion, reaching a maximum value of 8 lbs. The tests demonstrated that the repeated insertion and withdrawal of incore instrumentation thimble assemblies into the fuel bundle guides can be accomplished with reasonable insertion forces.

Life cycle tests were performed to determine if the frictional forces increase as a result of 40 insertions and withdrawals. An automatic timer was installed in the crane electrical circuitry to automatically cycle the thimble assembly between the fully inserted and withdrawn position. The instrument plate was set for 5° tilt and the assembly was cycled 60 times. The insertion and withdrawal forces were measured during the first and last five cycles. No discernible difference was noticed.

An off-center lift test was performed to determine if the thimble assembly can be withdrawn from the core region while lifting the assembly from an extreme off-center position. For a lifting point 11" off-center, insertion was accomplished without incident. The flexibility of the thimble is such that jamming of the assembly due to off-center lifting does not occur.

Cable insertion tests were performed to determine forces required to completely insert and withdraw a detector cable from the incore instrumentation thimble assembly. The guide tube routing included several 5" radius bends. The detector cable was passed through the wet guide tubing and into the thimble. For 540° of 5" radius bends, an insertion force of 15 lbs and a withdrawal force of 37 lbs was required. This force is reasonable for hand insertion.

1.5.9 MATERIALS IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed on the inside wall of the vessel to monitor the Nil Ductility Transition Temperature (NDTT) of the material during reactor operating lifetime. Details of the program are given in Section 4.1.5.

TABLE 1-2

ANALYSIS OF MODERATOR TEMPERATURE COEFFICIENTS IN THE CONNECTICUT YANKEE REACTOR AT START-OF-LIFE

REACTOR TEMPERATURE (°F)	DISSOLVED BORON CONCENTRATION (ppm)	ROD WORTH INSERTED (% $\Delta\rho$)	MODERATOR TEMPERATURE COEFFICIENT (10^{-4}/°F)	
			CALCULATED	MEASURED
260	2040	-	0.46	0.57
560	2305	-	0.84	1.00
551	2045	1.8	0.37	0.47
561	1730	4.5	-0.23	-0.25
551	1610	5.6	-0.30	-0.30

1.6 ACRS SPECIAL INTEREST ITEMS

1.6.1 GENERAL

This section describes the status of programs conducted for the investigation of items which were identified by the Advisory Committee on Reactor Safeguards (ACRS) as being of special interest and pertaining to all large water-cooled power reactors.

In carrying out these programs, information derived from research and development activities of the AEC or NRC and other organizations in the nuclear power industry was considered.

1.6.2 QUALITY ASSURANCE

The Baltimore Gas & Electric Company has traditionally retained full responsibility and maintained close control over all aspects of the design and construction of its power plants. This background of experience has been used by BGE in establishing a comprehensive quality assurance program to assure that the Calvert Cliffs Units 1 and 2 are designed, fabricated, and constructed in accordance with the requirements of applicable specifications and codes. The BGE program starts with the initial plant design and is continued through all phases of equipment procurement, fabrication, erection, construction, and plant operation. The program provides for review of specifications to assure that quality control requirements are included and for surveillance and audits of the manufacturing and construction efforts to assure that the specified requirements are met.

A summary description of the Calvert Cliffs quality assurance program is contained in the Quality Assurance Topical Report. This program fully meets the guidelines established by 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

1.6.3 FAILED FUEL DETECTION

Early detection of gross failure of fuel elements is important in limiting the consequences of fuel element failure. Early detection permits early application of protective action.

Combustion Engineering, Inc. has evaluated the following instruments for possible application as a failed fuel monitor:

a. Delayed-neutron monitor;

b. Ion exchange iodine monitor;

c. Cerenkov-detector monitor;

d. Gaseous-fission-product detector;

e. Differential gamma monitor;

f. Gross gamma plus specific isotope monitor.

Based on this instrument evaluation and a study of the expected fission and corrosion product activities in the reactor coolant, it has been concluded that the gross gamma plus specific isotope monitor provides a simple and reliable means for early detection of fuel failures.

The design bases of the detection system include the following:

a. Trends in fission product activity in the RCS are used as an indication of fuel element cladding failures. The minimum detectable activity is specified as 10^{-4} μc/cc I^{135};

b. There is a time delay of less than five minutes before the activity, emitted from a fuel-element-cladding failure, is indicated by the instrumentation. This time delay is a function of the location of the monitor;

c. The information obtained from this system is not used for automatic protective or control functions or for detecting the specific fuel assembly (or assemblies) which has failed;

d. The high activity alarm is supplemented with radiochemical analysis of the reactor coolant for fission products to provide positive identification for a fuel element failure.

The location and operation of the detector, designated as the process radiation monitor, is described in Section 9.1.3.

1.6.4 REACTOR VESSEL THERMAL SHOCK

Large quantities of ECC water are available to flood the core region in the event of a major LOCA. The Calvert Cliffs design uses a section of each of the RCS cold legs to conduct the water from the safety injection nozzles to the reactor vessel. This water then flows into the downcomer annulus and into the lower plenum of the reactor vessel before flooding the core itself. Analytical investigations were performed to provide assurance that the resultant cooling of the irradiated inner surface of the thick-walled reactor vessel will not induce or propagate cracks sufficient to cause the reactor vessel to fail.

A detailed analysis of the reactor response to thermal shock was performed. A report describing the results of the investigation, "Thermal Shock Analysis on Reactor Vessels Due to Emergency Core Cooling System Operation," A-68-9-1, March 15, 1968, was prepared and submitted to the AEC as part of Amendment 9 to the Maine Yankee license application (AEC Docket No. 50-309). The conclusion as stated in the report was that the CE reactor vessels are capable of sustaining the thermal shock imposed by ECCS operation without gross failure.

Further work was performed to refine the surface heat transfer coefficient and the brittle fracture model used in the evaluation. Combustion Engineering, Inc. first made an extensive review of temperature-quench data obtained during the heat treatment of several heavy section steel plates. With this background, additional quench tests were planned and conducted to develop experimental heat transfer coefficients to be used in the analysis of this problem. These tests were performed on a plate approximately 2'x2'x1/2" thick and instrumented with eleven thermocouples. The plate was heated to 550°F and quickly lowered into an agitated (turbulent) water bath at 80°F (nearly duplicating the temperature conditions present in the reactor). The temperature of all thermocouples were recorded throughout the cooldown of the plate. Subsequently, the transient temperature data were compared to a heat transfer computer model of the plate to obtain an effective heat transfer coefficient. A detailed report covering this work entitled, "Experimental Determination of Limiting Heat Transfer Coefficients During Quenching of Thick Steel Plates in Water," A-68-10-2, December 13, 1968, was submitted to the AEC and made part of the public record. The report concludes that an effective heat transfer coefficient of 300 Btu/hr-ft^2 F provides a realistic upper limit for thick steel plates quenched in highly agitated room temperature water.

Subsequent development of a more realistic brittle-fracture model involved the development of a finite-element analysis computer program. The finite element method was used to compute the stress near the tip of hypothetical axial and circumferential cracks in the vessel. The stress intensity factor for thermal, pressure and residual stress loadings was computed as a function of crack depth. A detailed report entitled, "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel during Emergency

Core Cooling," A-70-19-2, January 1970, has been submitted to the AEC and is part of the public record. Accuracy of the finite element solution was confirmed by comparisons between a boundary collocation solution and the intensity factor as a function of crack depth for an edge cracked tensile specimen. The results confirm the conservatism of the approach used by CE and verify that cracks in the vessel will not grow during the thermal shock transient associated with ECC operation.

1.6.5 BLOWDOWN FORCES ON CORE AND REACTOR COOLANT SYSTEM COMPONENTS

In the event of a large break, the RCS would depressurize rapidly, developing local pressure differences and forces in excess of normal operating loads. The WATERHAMMER computer program was utilized to define the hydraulic transient during the initial subcooled portion of the blowdown; the MODFLASH-2 computer program was used to calculate the pressure variations during the saturated portion of the blowdown. The loadings on the system components were then calculated from the pressure forces so obtained.

The reactor vessel internals were evaluated on the basis of these transient loadings. All critical components were designed to withstand these loads so that the core will be kept in place and that there will be no significant interference with the subsequent cooling of the core. The analysis reported in Section 14.15 shows that the Calvert Cliffs reactor meets these design requirements; in addition, the analysis shows that all of the CEAs except those adjacent to the outlet nozzle nearest the break can be inserted into the core following the accident.

In order to further refine the capability in this area, CE is assessing the possible use of third generation computer programs like FLASH-3 and RELAP-3B. The progress of AEC- or NRC-sponsored experimental programs like the LOFT program is being monitored, and the information derived is being used to confirm the adequacy of the analytical techniques currently in use and under development.

1.6.6 EFFECT OF FUEL ROD FAILURE ON ECCS PERFORMANCE

Experimental results have indicated that the core conditions after a LOCA (high internal gas pressure and increasing clad temperature which decreases clad tensile strength) can induce deformation of the fuel cladding in the time interval between blowdown and refill. The deformation takes the form of localized swelling of a fuel rod continuing until the clad perforates or until the temperature transient is reversed. Analytical and experimental programs were undertaken by CE to provide assurance that this deformation will not significantly affect the ability of the ECCS to prevent fuel melting.

The analytical work program established the conditions within the core during the transient for various break sizes. The parameters which determine the extent of fuel deformation are the internal gas pressure and the clad temperature transient (the temperature, the rate of change and the duration). The experimental program was designed to establish the correlation between the variable (internal gas pressure and the temperature transient) and the extent of clad deformation and perforation.

The testing indicated the degree of clad deformation which may take place before perforation, as a function of clad heating rate and internal pressure. The clad swelling was observed to be a localized phenomenon, and the clad perforation was observed as a longitudinal split less than 1/2" long and 3/16" wide.

1.7 IDENTIFICATION OF CONTRACTORS

Calvert Cliffs Nuclear Power Plant, Inc. retains full responsibility for the engineering and design of facilities, purchase of equipment, construction, and operation of the CCNPP. The procedure followed during construction was similar to that which has been used by BGE for most of its generating facilities now in service or under construction.

Baltimore Gas & Electric Company carried out its responsibilities either by performing the work with its own staff or by in-depth involvement in work delegated to its major contractors. Such responsibilities were divided internally within the Company as follows:

The Electric Engineering Department had the overall responsibility for the design of the plant. Procurement was the responsibility of the Purchasing and Stores Department. The Electric Construction Department was responsible for all site construction activities. The Electric Production Department was responsible for preoperational testing and initial testing as well as operation and maintenance of the plant. Shop inspections and witness testing were the responsibility of the Electric Production Department, assisted by members of the Electric Test Department, under the direction of the Quality Assurance Engineer. All other departments of BGE were available as needed to assist in the design and construction of the plant.

Baltimore Gas & Electric Company engaged CE to design, manufacture, and deliver to the site two complete NSSSs and to design and fabricate the initial core loads of fuel and two reload batches for each reactor. Combustion Engineering, Inc. also furnished technical and professional supervision for erection, initial fuel loading, testing, and initial start-up of the two NSSSs. Replacement steam generators and replacement RVCHs were provided by Babcock & Wilcox, Canada for Units 1 and 2.

Bechtel Associates, an affiliate of the Bechtel Power Corporation, was engaged as the Architect-Engineer for this project and as such performed engineering and design work for the balance of the plant equipment, systems, and structures not included under CE's scope of supply. Bechtel Associates prepared specifications, subject to BGE's approval, for all material, equipment, and systems which were purchased. Bechtel Power Corporation also provided qualified inspectors for shop inspections. Baltimore Gas & Electric Company contracted with Bechtel Power Corporation to perform the on-site construction of the entire plant.

The firm of Dames and Moore was retained as a consultant in the fields relating to site acceptability; namely, population and land use, meteorology, geology, seismology, hydrology, local shoreline protection, and hurricane effects.

NUS Corporation was retained as a general nuclear and radiological consultant and assisted BGE in the area of environmental radiological monitoring.

MPR Associates was retained to assist BGE in all aspects of the Quality Assurance Program. Their exact function in the Program is detailed in Appendix 1A.

A number of consultants participated in studies in connection with the use of Chesapeake Bay water for condenser cooling purposes. Sheppard T. Powell and Associates performed studies of the physical and chemical characteristics of the bay. Under the direction of Dr. Ruth Patrick, the Academy of Natural Sciences of Philadelphia performed extensive studies in the field of marine ecology. The Alden Research Laboratories of the Worchester Polytechnic Institute performed studies on a hydraulic model of a 34-mile portion of the Chesapeake Bay in the area of the plant site. In addition, Dr. John C. Geyer of the Johns Hopkins University served as a general water consultant. All of these studies were coordinated by BGE.

1.8 GENERIC ISSUES

1.8.1 THREE MILE ISLAND ACTION ITEMS

On March 28, 1979 a serious accident occurred in Unit 2 of Three Mile Island (TMI) Nuclear Power Plant in Pennsylvania. In response to the lessons learned from the accident, the NRC issued NUREG-0578, "TMI-2 Lessons Learned Task Force Report (Short-Term and Final)" and NUREG-0737, "Clarification of TMI Action Plan Requirements". What follows in this section is a list of the action items from NUREG-0737 and the BGE actions taken to respond to each of them. Baltimore Gas and Electric Company provided to the NRC the implementation status of each of the TMI action items identifying each item as either complete or not applicable to Calvert Cliffs or, in the case of three items, gave the completion status (Reference 1).

The action items are divided into two sections. The first is a statement of the NRC requirement. Documents appearing at the end of this section are NRC issuances on the same subject in addition to NUREG-0737. The second section is the BGE response to the item and the NRC approval. At the end of this section may be identified some sections of the UFSAR where more information can be found. Likewise, where there is information on the subject in the plant Technical Specifications, the words "Technical Specifications" appear. PLEASE NOTE: These references are starting points. They do not represent all the places that information may be found.

I.A.1.1 SHIFT TECHNICAL ADVISOR

Provide an on-shift technical advisor to the shift supervisor. This person may serve more than one unit if qualified on both units. A bachelor's or equivalent degree in a scientific or engineering discipline and plant-specific training is required. (Generic Letter 86-04)

Baltimore Gas and Electric Company submitted the training and qualifications of a Shift Technical Advisor (Reference 2). The NRC concurred with the program, but did not concur with the use of non-technical degree SROs in the program (Reference 3). Baltimore Gas and Electric Company later restated the position that equivalent qualifications were allowed to substitute for a technical degree (Reference 4). Plant procedures and Technical Specifications define the requirements for the on-shift presence of the Shift Technical Advisor and the qualifications necessary for holding the position.

I.A.1.2 SHIFT SUPERVISOR RESPONSIBILITIES

Delegate the non-safety duties of the shift supervisor to another position.

In 1979, the duties of tagging authority and operations refueling outage coordinator were assigned to other positions. This item was reported complete to the NRC (Reference 1). (Sections 12.1.1, 12.1.3; Technical Specifications)

I.A.1.3 SHIFT MANNING

Provisions governing shift staffing shall be included in plant administrative procedures. These procedures shall also restrict the use of overtime for personnel who perform safety-related functions (e.g., senior reactor operators [SROs], reactor operators, health physicists, auxiliary operators, Instrument and Control technicians and key maintenance personnel. (Generic Letter 82-12)

Baltimore Gas and Electric Company took exception to the NRC overtime rules and informed the NRC that procedures had been established to describe the requirements for shift manning and to control overtime for shift operators

(Reference 5). The NRC accepted the shift manning procedure (Reference 6). The issuance of Generic Letter 82-12 superseded the NUREG-0737 rules on shift manning. Baltimore Gas and Electric Company responded with a license amendment request which stated BGE compliance with the generic letter. The NRC approved the license amendment (Reference 7). (Technical Specifications)

I.A.2.1 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

An applicant for SRO who does not have an engineering degree must have been a Control Room operator for four years with one year as a licensed operator. An applicant who is a degreed engineer must have two years of nuclear plant experience as an engineer, participate in an SRO training program and have three months on shift as an SRO in training.

Modifications to the SRO training program were submitted to the NRC (References 8 and 9). The NRC approved the BGE approach to Item I.A.2.1.4, "Upgrading RO and SRO Training" (Reference 10). (Section 12.2.1.6; Technical Specifications)

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

Permanent operations instructors must demonstrate SRO qualifications and be enrolled in the appropriate requalification program until the operations training program is accredited.

The operations training program was first accredited by the National Academy for Nuclear Training in 1984 and has been continuously accredited since then. (Sections 12.1.1, 12.1.3, 12.2.1.6; Technical Specifications)

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMS

Simulator exams must be included as part of licensing examinations. (Generic Letter 81-29)

Baltimore Gas and Electric Company informed the NRC that simulator examinations are included in the BGE licensing examinations (Reference 11).

I.C.1 SHORT-TERM ACCIDENT AND PROCEDURES REVIEW

Analyses must be performed to address transients and accidents and inadequate core cooling. Technical guidelines must be developed from these analyses. Emergency operating procedures (EOPs) must be upgraded to the level of the technical guidelines and a writer's guide must be provided. Upgraded EOPs must be implemented and personnel trained on them. (Generic Letter 83-23)

Calvert Cliffs adopted the Combustion Engineering Emergency Procedures Guidelines (CEN-152) for the development of EOPs (Reference 12). The guidelines contain the EOP Writer's Guide, the EOP Verification/Validation Plan and the EOP Training Plan. The NRC accepted the guidelines for implementation with some comments (References 13 and 14). Baltimore Gas and Electric Company submitted the EOP Procedures Generation Package for NRC review (Reference 15). The NRC gave a critique of the package (Reference 16), recommended that it be reviewed by BGE for compliance with the concepts set forth in CEN-152, and stated that the revision did not need to be submitted to the NRC.

I.C.2 SHIFT AND RELIEF TURNOVER PROCEDURES (NUREG-0578 Item 2.2.1.c)

Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the on-coming and off-going individuals responsible for command of the operations in the Control Room. Supplementary checklists and shift logs should be developed for the entire operations organization, including instrument technicians, auxiliary operators and maintenance personnel.

Shift turnover is controlled by plant procedures which requires a turnover checklist.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITIES

Plant procedures shall clearly define the duties, responsibilities and authority of shift supervisors and operators. The highest level of plant management shall periodically reissue a directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant on his or her shift and that clearly establishes that person's authority.

The duties, responsibilities and authority of the operators and shift supervisors are defined in plant procedures. (Section 12.1.1)

I.C.4 CONTROL ROOM ACCESS

Limit access to the Control Room to those individuals responsible for the direct operation of the plant, certain technical supervisors and certain NRC personnel. Establish authority of the person in charge of the Control Room and line of authority and responsibility in the Control Room in an emergency.

Control Room access and watchstander authority and responsibilities are defined in plant procedures.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE

Carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. Implement procedures to ensure that important information on operating experience relating to plant safety inside and outside the plant is continually supplied to operators and others.

Baltimore Gas and Electric Company proposed to create the Plant Operating Experience Assessment Committee to satisfy this requirement (Reference 17). Approval was given by the NRC (Reference 6). Subsequently, BGE proposed to dissolve the committee (Reference 18) and form a new entity called the Industry Operating Experience Review Unit. The unit would perform the same operating experience review function as the committee. This change was approved by the NRC (Reference 19). (Section 12.1.3)

I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Establish an effective system for verifying the correct performance of operating activities.

Baltimore Gas and Electric Company stated that plant procedures would control activities associated with operating activities (Reference 17). Some activities are done by tagging equipment with the knowledge and concurrence of a senior

licensed person and including independent verification. The NRC found the program to be acceptable (Reference 6).

I.D.1 CONTROL ROOM DESIGN REVIEWS

Conduct a detailed Control Room design review considering human engineering requirements and correct discrepancies resulting from the review. The objective of this action is to improve the ability of nuclear power plant operators to prevent accidents by improving the information provided to them. (NUREG-700, Generic Letter 82-33)

This was a long-term project. Baltimore Gas and Electric Company submitted the Control Room Design Review program plan (Reference 20), as well as a supplementary report (Reference 21). The BGE approach to resolving this issue was approved by the NRC (Reference 22). (Technical Specifications)

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Install a safety parameter display which displays to operating personnel the minimum set of parameters which defines the safety status of the plant. (NUREG-0696, Generic Letter 82-33)

Baltimore Gas and Electric Company presented the plan for installing the Safety Parameter Display System (References 23 and 24). The NRC approved the system for installation (Reference 25). Baltimore Gas and Electric Company later sent a summarizing report to the NRC (Reference 26). (Section 7.5.5.3)

II.B.1 REACTOR COOLANT SYSTEM VENTS

Install high point vents, remotely operated from the Control Room, in the ARCS and the reactor vessel head. The important safety function of the vents is to enhance core cooling. The vents must not lead to an unacceptable increase in the probability of a LOCA or a challenge to containment integrity.

The proposed design and operating procedure guidelines were presented to the NRC (References 27 and 28). Later, the NRC stated this TMI requirement was superseded by 10 CFR 50.44(c)(3)(iii) and, therefore, the item was considered complete (Reference 29). The work was completed on December 18, 1987. (Section 4.1.2, 4.1.3.6, Technical Specifications)

II.B.2 PLANT SHIELDING

Conduct a radiation and shielding review of plant areas around systems that may contain highly radioactive material as result of an accident. Identify vital areas and equipment. Design and install shielding where the review shows it is necessary to allow access to vital areas and equipment. (Generic Letter 83-37)

The NRC approved the plant modifications and inspected them (References 30 and 31). (Sections 5.1.5.6, 5A.6, 11.2.1, 11.2.2.5)

II.B.3 POST-ACCIDENT SAMPLING

Provide the capability for personnel to obtain a sample of reactor coolant and containment atmosphere under accident conditions. Personnel must be able to take these samples with limited radiation exposure.

Baltimore Gas and Electric Company detailed compliance with the PASS requirements of this item (Reference 32). Compliance was approved by the

NRC (Reference 33). Baltimore Gas and Electric Company later proposed a different approach to meeting the requirements of this item (Reference 34). The NRC accepted the modified system (Reference 35). The NRC subsequently determined (Reference 103) that the PASS was not needed to support emergency response decision making during the initial phases of an accident. They no longer require dedicated equipment to perform PASS functions and the PASS sampling requirements are eliminated from the licensing basis. However, the NRC believes that there are benefits in having post-accident information previously provided by the PASS. Therefore, we committed to maintaining contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump and containment atmosphere, and for monitoring radioactive iodines released to offsite environs. In addition, we committed to maintain a capability for classifying fuel damage events at the Alert level threshold. (Table 5-3; Sections 9.6.2.2, 7.3.2.2, 7.5.8, 9.6.3, 11.2.1)

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Develop and implement a program which teaches the use of installed equipment and systems to control and mitigate accidents. This training must be presented to operations personnel from the plant manager through licensed operators.

The NRC observed the BGE operator training and approved the implementation of the program (Reference 36). Development of the training program was approved by the NRC (Reference 37).

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

Test the ARCS relief and safety valves to ensure they will operate under expected conditions created by design basis transients and accidents.

Baltimore Gas and Electric Company informed the NRC (Reference 38) of participation in a PWR utilities response to the NRC recommendations for safety and relief valve testing. The NRC endorsed the BGE response (Reference 39), leaving only operability testing of the PORV block valves as an open item. Technical Specifications require periodic testing of the PORV block valves but do not satisfy this open item. (Section 12.2.1.6)

II.D.3 VALVE POSITION INDICATION

Provide the operator with unambiguous indication of ARCS safety and relief valve position so that appropriate operator actions can be taken.

Baltimore Gas and Electric Company proposed to utilize acoustic monitors with indication in the Control Room to satisfy this requirement (Reference 40). This arrangement was reviewed and accepted by the NRC (Reference 41).

II.E.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Perform a simplified Auxiliary Feedwater (AFW) System reliability analysis, review of the AFW System using the criteria of Standard Review Plan 10.4.9, and reevaluate the design basis of the AFW flowrate.

The NRC detailed the requirements for this plant (Reference 42) and approved the BGE responses (Reference 43). What follows here are the requirements on which BGE took action.

Short-Term Recommendations

X.2.3.1

GS-2 Perform routine inspections to verify locked open valves are still open.

Originally BGE instituted a Technical Specification surveillance to verify the position of these locked valves. In Technical Specification Amendment Nos. 178 and 152 the inspection of locked valves was dropped because the valves are locked and access to the lock is controlled by procedure. These factors give assurance that the valve will stay in the designated position.

GS-4 Provide plant operators with emergency procedures for transferring to an alternate AFW supply.

Baltimore Gas and Electric Company proposed that the direction for transferring between condensate storage tanks be included in plant operating instructions (Reference 44). The NRC approved this as part of a license amendment (Reference 43).

GS-5 Without an AC power source, provide the required AFW flow for at least two hours from one AFW train.

Baltimore Gas and Electric Company stated the following: 1) that the only components in the steam-driven trains that would lose power on loss of AC power are the motor-operated steam supply valves to the AFW turbines (Reference 45); 2) that there are procedures instructing the operators to manually open the valves on loss of AC power (Reference 46); and 3) that emergency lighting is provided in the area of the handles for these valves (Reference 44). The NRC approved these responses (Reference 43). (Modified by Recommendation GL-3)

GS-6 Confirm AFW flow path availability after a train has been out of service for periodic testing or maintenance.

Baltimore Gas and Electric Company proposed a change to the Technical Specifications to require that the flow path be verified after a cold shutdown of 14 days or greater (Reference 47). Baltimore Gas and Electric Company also proposed that any valve in the flow path that has been repositioned must be returned to the original position and the position must be independently verified (Reference 44). The NRC approved the responses (Reference 43).

GS-8 Install a system to automatically initiate AFW.

Baltimore Gas and Electric Company stated that all criteria had been met (Reference 44). The NRC approved the response (Reference 43). Installation of an automated system was approved as a control-grade installation by the NRC (Reference 43). This approval included only the automatic start of the AFW pumps. The safety-grade system was evaluated under Recommendation GL-1.

Additional Short-Term Recommendations

2.4.2

1. Provide redundant AFW primary supply level indications and low level alarms in the Control Room. The alarm setpoint should allow the operator at least 20 minutes to anticipate the need to make up water or to transfer to an alternate supply.

   Baltimore Gas and Electric Company stated that the primary source of AFW (Condensate Storage Tank No. 12) has redundant level indication and the alarm in the Control Room will provide 20 minutes warning (Reference 45). The NRC approved the response (Reference 43). (Sections 10.3.2 and 10.3.3; Bechtel Design Criteria 4.3.5, Bullet 4)

2. Perform a 72-hour endurance test of all AFW system pumps. After a cool-down, run the pumps for one hour.

   This test was conducted on Pump 11. Following that test, the NRC changed the requirement to a 48-hour test. Pump Nos. 12, 21 and 22 were tested to the new standard. The results of the tests were evaluated as acceptable by the NRC (Reference 43).

3. Provide safety-grade indication of AFW flow to each SG in the Control Room. The flow instrument channels shall be powered from the emergency busses consistent with emergency power diversity requirements of Auxiliary Systems Branch Technical Position 10-1 of Standard Review Plan, Section 10.4.9.

   Baltimore Gas and Electric Company stated that the flow indication system was upgraded to safety-related (Reference 17). The NRC found this to be acceptable (Reference 43).

4. Provide a dedicated individual at the manual AFW valves when they are shut for testing.

   There is a technical specification requirement for the presence of this individual.

Long-Term Recommendations

2.4.3

GL-1 Provide a system to automatically start AFW flow. Design and install the system to meet safety-grade requirements.

Control grade circuitry to automatically initiate feedwater flow was installed and was approved by the NRC (Reference 43). The upgrade of the circuitry to safety-related for automatic initiation and for flow indication was approved by the NRC (Reference 48). (Sections 7.2.3.4 (Low SG Water Level), 7.4.5.2 (Manual Operation); 7.10 (Actuation), 7.12 (Diverse AFW Actuation System), 10.3 (AFW System), 14.4.2.2 (Full Power Case), 14.6 (Loss Of Feedwater Flow Event), and 14.10.2 (Loss of Non-Emergency Power); Technical Specifications)

GL-2 Install a redundant parallel flow path for the water supply to the AFW system.

Baltimore Gas and Electric Company requested an exemption to this requirement, citing the other changes being made to the system and the results of a cost and reliability study (Reference 49). The study showed that the addition of this requirement to the other modifications would result in a cost increase with little gain in reliability. The NRC agreed (Reference 105).

GL-3 Provide for automatic initiation of at least one AFW pump, its associated flow path and essential instrumentation. Maintain flow for at least two hours independent of AC power.

Automatic initiation is discussed in Recommendation GL-1. Baltimore Gas and Electric Company agreed to replace the motor-operated valves in the steam supply to the AFW pump turbine with air-operated, fail open valves (Reference 45). Flow can be maintained independent of AC power for two hours as discussed in Section 10.3.3. These modifications were found acceptable by the NRC (Reference 43).

Other Long-Term Recommendations

1. Environmentally qualify motor operated steam inlet valves and other equipment affected by main steam and feedwater line breaks.

   As stated in GL-3, the steam supply valves were changed to air operated control valves. Baltimore Gas and Electric Company stated that the solenoid valves associated with the control valves were environmentally qualified (Reference 44). This was approved by the NRC (Reference 43).

2. Because the steam supply to the AFW pump turbines comes from a single pipe source and the discharge from the pumps is directed to a single pipe, the NRC recommended an evaluation of the systems to determine if any changes were necessary to protect the SGs from boiling dry or to prove that the plant could be brought to a safe shutdown.

   Baltimore Gas and Electric Company stated that plant design and procedures were reviewed to ensure that they maintained the capability to provide adequate flow to the SGs in the case of an AFW pipe break (Reference 44). This was approved by the NRC (Reference 43).

   The NRC required flow design basis information for design basis transients and accidents, therefore, the minimum long-term flow rate was reanalyzed (References 51 and 52).

II.E.1.2 AUXILIARY FEEDWATER INITIATION AND FLOW

1. Design and install a safety-grade AFW system with the following features:

   A. The design shall provide for the automatic initiation of the AFW System;

   B. The automatic initiation signals and circuits shall be designed to withstand a single failure;

   C. The initiating signals and circuits shall be testable;

   D. The initiating signals and circuits shall be powered from the emergency busses;

E. Manual capability to initiate the AFW System from the Control Room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function;

F. The electrically operated pumps and valves shall be included in the automatic actuation of the loads onto the emergency busses;

G. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW System from the Control Room; and

2. Provide for safety-grade indication of feedwater flow in the Control Room.

   See Item GL-1 in Section II.E.1.1 of this discussion.

Baltimore Gas and Electric Company described modifications proposed to meet the control grade automatic start requirement (Reference 53); provided additional information on instrument power sources (Reference 54); and committed to installation of a third AFW train with electrically-driven pumps in each unit, as well as safety-grade AFW flow initiation and automatic initiation systems. Baltimore Gas and Electric Company later provided a more detailed description of the AFW proposed changes (Reference 55).

Compliance with this requirement was approved by the NRC (Reference 48). The Technical Specifications give limiting conditions for operation and surveillance requirements for the automatic operation of the AFW System. Although it was not relied upon by the NRC in their review, Bechtel Design Criteria (Reference 56) provides further information on how BGE met the requirements to ensure adequate flow, provide unit separation, provide auxiliary shutdown capability, improve overall system reliability and incorporate human engineering considerations into control boards and panels.

II.E.3.1 EMERGENCY POWER FOR PRESSURIZER HEATERS

Provide the capability to supply the pressurizer heaters from emergency power in a timely manner consistent with the with safety-related devices. Provide training for operators in the use of the heaters in natural circulation.

Two sets of pressurizer heaters on each unit are connected to emergency busses. The emergency power supply is also a technical specification requirement which was approved by the NRC (Reference 57). Operator training was implemented and was reviewed by the NRC (Reference 58). Baltimore Gas and Electric Company informed the NRC that instructions to the operators regarding timely connection of the heaters to emergency busses are in the Calvert Cliffs emergency procedures and that training programs are implemented. At the same time, BGE also stated that the interfaces between the heaters and the emergency busses are safety-related (Reference 59). The NRC acknowledged receipt of this information (Reference 60).

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

Revise and review procedures used for the control of combustible gas.

Operating and test procedures are reviewed and updated periodically.

Control of hydrogen in Containment during and following a DBE is no longer required. On March 2, 2004, the NRC issued a license amendment that allows removal of the hydrogen recombiners and hydrogen analyzers from the Technical Specifications (Reference 102). Since control of hydrogen in Containment is no longer necessary, the hydrogen recombiners and procedures for them are not required to be maintained. The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond DBE. Maintenance and testing procedures for the analyzers continue to be maintained.

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

The approval of related Technical Specification license amendments (Reference 61) concluded the NRC's review of Item II.E.4.2.

1. Containment isolation systems shall have diverse isolation parameters.

   Baltimore Gas and Electric Company stated that all containment penetrations by non-essential systems are locked shut, close on containment isolation signal, close on safety injection actuation signal, close on SG isolation signal or close on containment radiation signal (Reference 40). (Sections 5.2, 7.3.2.2 and 9.8.2.2; Technical Specifications)

2. Define essential and non-essential systems.

   Baltimore Gas and Electric Company defined essential fluid systems as those that are actively required during the early stages of an accident to control and mitigate the consequences of an accident such that exposure to offsite individuals is not in excess of the limits specified in 10 CFR Part 20 (References 62 and 63). The only exception to the basis for the "essential" designation of essential systems, CEN-125, is containment pressure sensing. Baltimore Gas and Electric Company determined that this system is necessary to monitor containment pressure throughout the accident evolution and is, therefore, an essential system.

3. All non-essential systems shall be automatically isolated.

   See Number 1 in this item. (Table 5-3, Section 5.2)

4. Design of the isolation scheme will not permit reopening of the containment isolation valves without operator action.

   Baltimore Gas and Electric Company stated that the system, as designed, does not allow automatic reset of the containment isolation valves and administrative procedures and controls are in place to ensure continued operability of the isolation system (Reference 40).

5. The isolation signal setpoint pressure that initiates containment isolation for nonessential penetrations must be the minimum compatible with normal conditions.

   Baltimore Gas and Electric Company provided the NRC with a description of the containment isolation pressure setpoint (Reference 2). The setpoint is also supported by BGE design calculations (Reference 64).

6. Containment purge valves that do not satisfy the operability criteria in Branch Technical Position CSB 6-4 (attached to Standard Review Plan 6.2.4) must be sealed closed as defined in Standard Review Plan 6.2.4, item II.3.f during Modes 1-4. Sealed closed valves must be under administrative control to ensure they cannot be inadvertently opened.

   Baltimore Gas and Electric Company applied for a license amendment to stipulate that the containment purge valve operators would be disconnected in Modes 1-4 and the valves would be verified shut (Reference 65). This was approved by the NRC (Reference 61). (Tables 5-3 and 7-4, Technical Specifications)

7. Containment purge valves must close on a high radiation signal.

   The NRC stated the valves were not required to have a high radiation closure signal for containment isolation (Reference 61). The logic behind this position was that the valves would be closed in Modes 1-4 and therefore automatic closure was not necessary. (Table 5-3, Section 7.3.2.2, Table 7-4, Technical Specifications)

II.F.1 <u>GASEOUS EFFLUENT MONITORS</u>

1. Provide monitors that are capable of measuring concentrations of noble gas fission products in plant gaseous effluents during normal operations and during and following an accident. Monitor all potential accident release paths. The detection range shall be from HALERU [As Low As Reasonably Achievable] concentrations to a maximum of 105 micro Ci/cc of XE-133.

   Baltimore Gas and Electric Company proposed to incorporate the main vent noble gas wide range gas effluent monitors into the Technical Specifications (References 66, 67, and 68). The NRC concurred with this request and stated that the Technical Specification part of this item was satisfied. Baltimore Gas and Electric Company notified the NRC that the noble gas monitors were installed in the main steam headers (Reference 69). The NRC found this acceptable (Reference 70). (Sections 11.2.3.2.1, 11.2.3.2.11, 11.2.3.2.12; GL 83-37; Technical Specifications)

2. Provide the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. An administrative program should be established, implemented and maintained to ensure this capability (Generic Letter 83-37).

   Baltimore Gas and Electric Company reported that a new wide range noble gas monitor had been installed and that procedures for its use were implemented (Reference 66). The NRC approved the BGE request (Reference 68). (Section 11.2.3.2.1; Technical Specifications).

3. A minimum of two containment high-range radiation monitors shall be provided for each unit. The monitors shall be physically separated and function to measure radiation within the reactor containment during and following an accident.

   The NRC conducted an inspection of the high-range radiation monitors and closed the item (Reference 71). The NRC acknowledged the

installation and approved the related technical specifications (Reference 68). (Section 11.2.3.3, Table 11-11; Technical Specifications)

4. Continuous indication of each unit's containment pressure shall be provided in the Control Room during power operation, startup and hot standby. (Generic Letter 83-37) The NRC also required that the measurement and indication capability include three times the design pressure of the containment and minus 5 psig, including a requirement that indication meet the design provisions of Regulatory Guide 1.97, including qualification, redundancy and testability.

   The NRC declared the equipment and installation requirements of this item were satisfied (Reference 7). The NRC conducted an inspection of the containment pressure measurement and indication and closed the item (Reference 72). The NRC acknowledged the installation and approved the related technical specifications (Reference 68). (Sections 4.3.2.1, 7.3.2.2, 7.5.8; Technical Specifications)

5. Continuous indication of each unit's containment water level shall be provided in the Control Room during power operation, startup and hot standby. (Generic Letter 83-37) The instrumentation shall consist of a wide-range indicator covering the range from the bottom of the containment to an elevation equivalent to a 600,000 gallon capacity and a narrow-range indicator which covers the range from the bottom of the containment to the top of the containment sump. The NRC also required that the wide-range instruments shall meet the requirements of Regulatory Guide 1.97 and the narrow-range instruments shall meet the requirements of Regulatory Guide 1.89 (Reference 73).

   In 1984, BGE advised the NRC of the status of this item. The current status can be found in Section 7.5.8. (Section 4.3.2.1; Technical Specifications; Regulatory Guide 1.97)

6. Continuous indication and recording of each unit's containment hydrogen concentration shall be provided in the Control Room within 30 minutes of the initiation of safety injection.

   The NRC declared that the equipment and installation requirements of this item were satisfied (Reference 7). (Section 6.8.1; Technical Specifications; Regulatory Guide 1.97)

   The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond DBE (Reference 102).

II.F.2 <u>INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING</u>

Provide instrumentation which gives an unambiguous, easy to interpret indication of inadequate core cooling.

In 1984 and 1985, BGE submitted the results of a review of post-accident monitoring instrumentation of which inadequate core cooling is a part (References 74 and 75). The NRC found the instrumentation to be acceptable (Reference 76). Baltimore Gas and Electric Company later revised the 1984 submittal (Reference 77). Section 7.5.9 of the UFSAR contains information resulting from that revision. (Technical Specifications; Regulatory Guide 1.97)

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Motive and control components for the PORVs, PORV block valves and pressurizer level indication must be supplied from vital power supplies when offsite power is not available.

Baltimore Gas and Electric Company stated that the motive components of the PORVs and the PORV block valves are connected to vital 480 Volt motor control centers, which are powered by the emergency diesel generators on loss of offsite power (Reference 40). The control components for the valves are connected to the same motor control centers with a 125 Volt battery backup. Two of the pressurizer level indicators are connected to vital DC busses and one is connected to offsite AC power with emergency diesel generator backup. The NRC found this acceptable (Reference 58). (Section 4.2.2, Technical Specifications)

II.K.2 ANALYSIS AND EVALUATION

II.K.2.13 A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

An analysis (CEN-189) was performed. The analysis was endorsed by BGE (Reference 78) and approved by the NRC (Reference 79). (Section 4.1.4.5.4)

II.K.2.17 Analyze the potential for voiding in the ARCS during anticipated transients.

An analysis (CEN-199) was performed and endorsed by BGE (Reference 80). In that letter exception was taken to the 20-hour cooldown period before depressurization for entry into shutdown cooling given in CEN-199. A 17.5 hour period was proposed instead. The analysis and the BGE exception were approved by NRC (Reference 81). (Sections 3.2.3.4 and 3.2.3.6)

II.K.2.19 Provide a benchmark analysis of sequential feedwater flow to the SGs following a loss of feedwater.

The NRC declared that no action was required for CE-designed NSSSs (Reference 50).

II.K.3 B&O TASK FORCE FINAL RECOMMENDATIONS

II.K.3.1 Provide a system that uses the PORV block valves, in an automated mode, to protect against a small-break LOCA.

Baltimore Gas and Electric Company declared that an automated PORV isolation system would present serious challenges to plant safety and was, therefore, not necessary at this plant (Reference 82). The NRC agreed (Reference 83).

II.K.3.2 Submit a report documenting the various actions taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV and show how those actions constitute sufficient improvement in reactor safety.

An analysis (CEN-145) was performed. The analysis was endorsed by BGE (Reference 84) and approved by the NRC (Reference 83).

II.K.3.3 Report safety valve and PORV failures and challenges.

Technical Specifications used to require all such events be reported annually. The Technical Specifications were amended (reference 104) to remove this requirement.

II.K.3.5 Automatic trip of the RCPs during a LOCA.

An analysis (CEN-268) of a strategy called "trip two/leave two" was performed. Baltimore Gas and Electric Company endorsed this strategy, which was approved for Calvert Cliffs by the NRC (References 85 and 86). (Section 7.2.3.3, Table 14.1-3; Generic Letters 83-10a, 83-10b and 86-06)

II.K.3.17 Report on EGGS outages for the five years prior to issuance of the TMI Action Items.

The report was submitted to the NRC and accepted (Reference 87).

II.K.3.25 Determine the effect of loss of cooling water to the RCP seal coolers due to loss of offsite power.

Baltimore Gas and Electric Company presented argument that automatic initiation of RCP seal cooling on loss of offsite power is not necessary (Reference 88). The NRC approved this position (Reference 89).

II.K.3.30 Revise, document and submit the small-break LOCA analysis to show compliance with 10 CFR Part 50, Appendix K.

An analysis (CEN-203) was performed. The analysis was endorsed by BGE (Reference 90). The NRC advised BGE that use of the topical was satisfactory (Reference 91).

II.K.3.31 Submit the small-break LOCA analysis required by II.K.3.30 to show compliance with 10 CFR 50.46.

The NRC approved the analysis for II.K.3.30 and stated that further analysis to satisfy this item was not necessary (Reference 91).

III.A <u>IMPROVING EMERGENCY PREPAREDNESS</u>

III.A.1.2 Establish a technical support center separate from but in close proximity to the Control Room. Establish an operational support center separate from the Control Room and other emergency response facilities (Reference 92). (NUREG-0696)

Baltimore Gas and Electric Company presented a conceptual design for the Technical Support Center and the Emergency Operations Facility (Reference 93). In addition, BGE provided a progress report for the Emergency Operations Facility (Reference 94). The NRC gave partial approval in 1983 (Reference 95). Final approval of the

facilities was to be the subject of an appraisal by the NRC but their appraisal program was canceled and the appraisal was not conducted. (Sections 7.5.5.2, 7.8.2.6, 12.6.2.1)

III.A.2 Upgrade emergency plans to provide reasonable assurance that adequate measures can and will be taken in the event of a radiological emergency. Additionally, NUREG-0654 required that plants have the capability to take meteorological measurements from primary and backup systems. (Regulatory Guide 1.97)

The NRC found the upgrade of the emergency plans satisfactory and the onsite and offsite emergency preparedness adequate (Reference 96). (Section 12.6)

The NRC approved the meteorological data upgrades (Reference 97). (Section 2.3.7)

On November 30, 2004 the Patuxent River Naval Air Station discontinued staffing the weather station on a 24/7 schedule. A 10 CFR 50.54(q) evaluation was performed to replace the backup meteorological information provided by Patuxent River Naval Air Station. The new description states that the Emergency Response Plan Implementing Procedures provide instructions for accessing backup meteorological data in the event the primary meteorological data becomes unavailable.

III.D <u>PRIMARY COOLANT OUTSIDE CONTAINMENT</u>

III.D.1.1 Implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. The NRC asked for a review of potential release paths due to design and operator deficiencies (Reference 73). This requirement was a result of an incident at North Anna, Unit 1. (NUREG-0578)

The program, in the form of a Surveillance Test Procedure and a procedure for dumping the reactor coolant drain tank into the containment sump, was approved by the NRC (Reference 41). Baltimore Gas and Electric Company conducted the review and declared that no changes to the plant or procedures were necessary (Reference 98). (Technical Specifications)

The systems involved are:

Safety Injection
Containment Spray
Shutdown Cooling
Containment Sump Recirculation
Containment Atmosphere Sampling
Reactor Coolant Sampling

III.D.3.3 Provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas of the facility where personnel may be present during an accident. Effective monitoring of increasing iodine levels under accident conditions must include portable monitoring instruments using sample media that will collect iodine selectively over xenon.

Baltimore Gas and Electric Company declared that all requirements of this item were being met (Reference 17). The NRC accepted that statement (Reference 99). (Sections 4.3.3.1, 11.2.3.2.1; Technical Specifications)

III.D.3.4 Assure that Control Room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the plant can be safely operated or shut down under design accident conditions.

Control Room ventilation is capable of automatic isolation from outside air and filtered to reduce airborne radioactivity concentration. A Control Room Habitability Study was conducted by BGE and reported to the NRC (Reference 2). An evaluation of the available self-contained breathing apparatus and Control Room infiltration was presented to the NRC in 1982 (Reference 100). The NRC concluded that the BGE response to this item was acceptable (Reference 101). Baltimore Gas and Electric Company's current method of complying with General Design Criterion 19 is discussed in Section 9.8.2.3, Auxiliary Building Ventilating Systems. (Sections 1.2.9.7, 1.4.8, 1C.4, 7.6.2, 11.2.3.2.8; Technical Specifications)

1.8.2 STATION BLACKOUT

On July 21, 1988, 10 CFR Part 50 was amended to include a new section 50.63, "Loss of All Alternating Current Power" (Station Blackout [SBO]). The SBO rule requires that each of the light water cooled nuclear power plants be able to withstand and recover from an SBO of specified duration. It also identifies the factors which must be considered in specifying the SBO duration. Regulatory Guide 1.155, "Station Blackout," describes a method acceptable to the NRC for meeting the requirements of 10 CFR 50.63. The Regulatory Guide references NUMARC 87-00, Revision 1 as an acceptable means of evaluating our response to the SBO rule. Our response to the SBO rule is briefly described below.

<u>Station Blackout Duration</u>

Based on several factors including, the expected frequency of severe weather, the expected frequency of a loss of offsite power, the onsite power configuration and the addition of a fourth safety-related diesel generator, we initially determined that the rule required us to be able to cope with an SBO for four hours. With the addition of a non-safety-related diesel generator for SBO response, we are required to cope with an SBO for one hour (the maximum time assumed to start and load the non-safety-related diesel). Only one unit is assumed to be in an SBO condition. The scenario that the rule proposes is as follows. Both units are at full power when offsite power is lost. Three diesel generators fail to start. The fourth diesel generator starts and loads the shutdown loads for one unit. The other unit is in an SBO. Restoration of AC power after a blackout was assumed to be from an onsite diesel generator. This was because restoration of offsite sources could take in excess of four hours for major grid blackouts.

<u>Ability to Cope with a Station Blackout</u>

We evaluated the ability of either unit to cope with an SBO. The Regulatory Guide requires the evaluation of several factors. Each of these factors is described below. Although the initial evaluation was for four hours with one Unit in a blackout condition, the installation of the non-safety related DG allowed us to change our coping duration to one hour. The information presented below is based on the initial four hour coping duration.

Condensate Inventory

We determined that the minimum permissible condensate storage tank level, per the Technical Specifications, provides more than enough water for four hours of decay heat removal for one unit.

Class 1E Battery Capacity

Battery capacity calculations verify that each of the four 125 VDC Class 1E batteries have capacity to carry SBO loads for at least one hour. This is sufficient for SBO since the AAC power source (0C Diesel Generator) will be available within one hour to supply the battery chargers. Shutdown loads and equipment needed to start a diesel generator and close its breakers were included in the load profile. The four battery duty cycle calculations utilize a 15% design margin in accordance with IEEE-485 recommendation.

Compressed Air

Air-operated valves relied upon to cope with an SBO for four hours can be operated manually. Valves requiring manual operation are identified in plant procedures.

Loss of Ventilation

Detailed room heatup calculations were performed for nine different areas of the plant. These calculations resulted in modifications to the Control Room ceiling and the battery room heating, ventilation and air conditioning system. Operability of the necessary equipment in these rooms was verified.

Containment Isolation

Containment isolation valves which must be closed or cycled during an SBO event are capable of being operated without onsite or offsite power. Valve position indication is provided if necessary.

Reactor Coolant Inventory

An analysis was done to confirm that we can maintain adequate reactor coolant inventory for four hours during an SBO event. The analysis assumed the maximum Technical Specification leakage and 25 gpm from each RCP (four pumps).

Procedures

Existing plant procedures address AC power restoration, severe weather response, and the plant response to an SBO. These procedures will be updated as plant conditions change.

Modifications

Plant modifications were required to comply with 10 CFR 50.63. The major modification is the addition of one safety-related and one non-safety-related diesel generator to our onsite distribution system. These modifications were completed and are described in Section 8.4.

1.8.3 MAINTENANCE RULE

Title 10 CFR 50.65 "Requirements for Monitoring the effectiveness of Maintenance at Nuclear Power Plants" was issued on July 10, 1991. Utilities were required to be in full compliance by July 10, 1996. The purpose of the Maintenance Rule is to insure that structures, systems, and components of nuclear power plants are maintained such that plant equipment will perform its intended function when required.

Nuclear Energy Institute [formerly Nuclear Management & Resources Council (NUMARC)] formed a utility group to develop a guideline (NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) to assist utilities in implementing 10 CFR 50.65. NUMARC 93-01 was developed with input from the NRC and later endorsed by the NRC in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" as an acceptable method to meet the requirements of 10 CFR 50.65. Calvert Cliffs used NUMARC 93-01 to implement the requirements of the Maintenance Rule. Existing programs were used when available to meet the requirements of the Rule. New processes were developed and included in the appropriate program procedure to ensure compliance.

1.8.4 INDIVIDUAL PLANT EXAMINATION

In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded that existing plants posed no undue risk to the public health and safety. However, the Commission recognized that systematic examinations were beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. Therefore, each plant was requested to perform such a systematic examination of internally initiated events in Generic Letter 88-20 and report the results to the NRC. One of the purposes of the examination is to determine whether modifications to hardware and procedures were necessary to reduce the frequency of severe accidents or to mitigate their consequences. Supplement 1 to the Generic Letter provided additional guidance concerning the method to be used in the plant examination. Supplement 2 to the Generic Letter addressed severe accident management strategies that could be used in the plant examination. These strategies were developed by the NRC based on experience gained in reviews of probabilistic risk assessments. Calvert Cliffs was requested to evaluate these or similar strategies during our plant examination. Supplement 3 of the Generic Letter provided additional insights about the performance of pressurized water reactor containments. These insights could be used, if appropriate, during the individual plant examination. Supplement 4 addressed the need to evaluate plant vulnerabilities and response to external events. Risk assessments at that time indicated that the risk from external events could be a significant contributor to core damage in some instances. Finally, Supplement 5 to Generic Letter 88-20 provided updated guidance concerning seismic hazard estimates for many plants so licensees could determine the appropriate level of examination for their plants.

The NRC's purpose in requesting these evaluations was for licensees to: (1) develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at the plant under full-power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material

releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

In response to these requests, Calvert Cliffs performed an individual plant examination for internally initiated events. Evaluated events included (but are not limited to) transients, LOCAs, anticipated transient without scram, internal flooding, steam generator tube rupture, and interfacing system LOCAs. In addition, the examination searched for decay heat removal vulnerabilities. Based on that examination, the NRC concluded that the resolution of Unresolved Safety Issue A-45, Decay Heat Removal Reliability, was acceptable and closed this issue for Calvert Cliffs. The NRC also concluded that Calvert Cliffs' individual plant examination was complete and the results were reasonable given Calvert Cliffs' design, operation, and history. As a result, the NRC concluded that Calvert Cliffs' individual plant examination process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and Calvert Cliffs has met the intent of Generic Letter 88-20, including Supplements 1, 2, and 3.

Calvert Cliffs also performed an individual plant examination for externally initiated events. This examination included initiating events such as, fires, earthquakes, high winds, and other external events. As part of the plant examination process, a number of generic safety issues (GSI's) were identified and addressed. The NRC has reviewed the response for these GSI's and considers them resolved for Calvert Cliffs: GSI–103, Design for Maximum Probable Precipitation; GSI–57, Effects of Fire Protection System Actuation on Safety Related Equipment; GSI–147, Fire Induced Alternate Shutdown/Control Room Panel Indications; GSI–148, Smoke Control and Manual Fire-Fighting Effectiveness; GSI-156, Systematic Evaluation Program; and GSI–172, Multiple System Responses Program. The NRC found that the Calvert Cliffs individual plant examination for external events was complete and the results were reasonable given Calvert Cliffs' design, operation, and history. As a result, the NRC concluded that the integrated plant evaluation for external events process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and Calvert Cliffs has met the intent of Generic Letter 88-20, Supplements 4 and 5.

1.8.5 GENERIC LETTER 2008-01, MANAGING GAS ACCUMULATION

Gas accumulation in water systems can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Reference 106 to address the issue of gas accumulation in certain systems. The NRC requested that each licensee evaluate its Emergency Core Cooling System (ECCS), decay heat removal system (Shutdown Cooling [SDC] system), and containment spray (CS) system licensing basis, design, testing, and corrective actions to ensure that gas accumulation is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality are identified.

In Reference 107 the following systems were determined to be in the scope of the generic letter for Calvert Cliffs:

- Safety Injection (SI) system., i.e., ECCS
- SDC system
- CS system
- Relevant flow path in the charging system when used for High Pressure Safety Injection (HPSI) - not applicable to Unit 1 due to existing high point vents.

For the purposes of this issue, the term ECCS refers to the combination of SI and SDC systems. The relevant portion of the charging system includes that piping used for providing a HPSI flow path for the hot leg injection via pressurizer spray post-loss-of-coolant-accident (LOCA).

The following are considered gas intrusion mechanisms:

1) The formation of gas upstream of normally shut valves in SI discharge piping caused by leak-by of nitrogen saturated water from the safety injection tanks to the lower pressure SI discharge piping.
2) The "stripping" of gas out of solution due to leakage of the RCS/SI boundary check valves.
3) Gas coming out of solution due to dynamic pressure drops.
4) Gas intrusion due to human error.

In-leakage through vent valves, valve packing, mechanical pump seals, threaded pipe connections, and gasketed flanges, is not considered a valid source of gas intrusion since all piping in the scope of the GL is under positive gauge pressure (excluding the dry CS piping at higher elevations in the Containment).

References 108 and 109 acknowledged that no gas voids were found at any location during the confirmatory walkdowns and evaluations performed on the subject systems.

Reference 110 submitted a license amendment request to add Surveillance Requirements to verify that the locations susceptible to gas accumulation are sufficiently filled with water and to provide allowances which permit performance of the verification.

In Reference 111, the NRC issued a change to the TS to add new Surveillance Requirements to the TS for the SDC system, the ECCS and the CS system. These SRs require periodic verification that gas has not accumulated in the associated piping to a degree that would render the system inoperable.

1.8.6 REFERENCES

1. Letter from G. C. Creel (BGE) to T. E. Murley (NRC), dated April 18, 1989, Implementation Status of TMI Action Plan Items
2. Letter from C. H. Poindexter (BGE) to D. G. Eisenhut (NRC), dated December 30, 1980, Response to NUREG-0737
3. Letter from R. A. Clark (NRC) to A. E. Lundvall (BGE), dated July 9, 1982, NUREG-0737 Item 1A.1.1, Shift Technical Advisor
4. Letter from J. A. Tiernan (BGE) to D. H. Jaffe (NRC), dated May 16, 1986, Generic Letter 86-04, Policy Statement on Engineering Expertise on Shift
5. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated January 31, 1981, Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Response to NUREG-0737
6. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated November 3, 1981, TMI Action Plan Items I.A.1.3.1, I.C.5, and I.C.6, As Described in NUREG-0737
7. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BGE), dated April 21, 1983, Amendment Nos. 82 and 65 to Facility Operating License Nos. DPR-53 and DPR-63 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2

8. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated June 18, 1980, Additional TMI-2 Related Requirements

9. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated May 28, 1982, NUREG-0737, Items I.A.2.1 & II.B.4

10. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated November 24, 1982, Safety Evaluation by the Office of Nuclear Reactor Regulation, Items I.A.2.1.4 and II.B.4.1 of NUREG-0737

11. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated April 19, 1982, Post-TMI Requirements

12. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated November 13, 1981, Generic Letter 81-21 Response

13. Letter from D. G. Eisenhut (NRC) to All Licensees, dated July 29, 1983, Safety Evaluation of Emergency Procedure Guidelines (Generic Letter 83-23)

14. Letter from J. A. Zwolinski (NRC) to R. W. Wells (CE), dated April 16, 1985, Supplement 1 to Safety Evaluation Report for CEN-152

15. Letter from A. E. Lundvall (BGE) to J. R. Miller (NRC), dated March 14, 1984, EOP Procedures Generation Package

16. Letter from D. G. McDonald, Jr. (NRC) to G. C. Creel (BGE), dated February 15, 1990, Procedures Generation Package – Safety Evaluation

17. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated December 15, 1980, Response to NUREG-0737

18. Letter from G. C. Creel (BGE) to NRC Document Control Desk, dated June 12, 1992, Proposed Change to Quality Assurance Program

19. Letter from C. H. Bettenhausen (NRC) to G. C. Creel (BGE), dated June 30, 1992, 10 CFR 50.54 Quality Assurance Program Change Review

20. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated September 1, 1983, Detailed Control Room Design Review

21. Letter from J. A. Tiernan (BGE) to NRC Document Control Desk, dated June 30, 1988, Supplemental Detailed Control Room Design Review Summary Report (TAC Nos. 56110 and 56111)

22. Letter from D. G. McDonald, Jr. (NRC) to G. C. Creel (BGE), dated March 27, 1990, Supplement 1 to NUREG-0737, Item I.D.1 – Detailed Control Room Design Review

23. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated June 6, 1984, SPDS Safety Analysis

24. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated February 4, 1985, Safety Parameter Display System (SPDS)

25. Letter from A. C. Thadani (NRC) to J. A. Tiernan (BGE), dated March 24, 1986, NUREG-0737, Item I.D.2 Safety Parameter Display System

26. Letter from J. A. Tiernan (BGE) to NRC Document Control Desk, dated September 30, 1988, Safety Parameter Display System Report

27. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated January 4, 1980, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned)

28. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated August 11, 1981, Response to NUREG-0737 Item II.B.1

29. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated September 7, 1983, NUREG-0737, Item II.B.1, Reactor Coolant System Vents

30. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated August 16, 1984, NUREG-0737 Item II.B.2.2, Plant Shielding at Calvert Cliffs Unit Nos. 1 & 2

31. Letter from R. W. Starostecki (NRC) to A. E. Lundvall, Jr. (BGE), dated June 25, 1984, Inspection Report 50-317/84-05, 50-318/84-05

32. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated November 30, 1982, TMI Action Plan Item II.B.3

33. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated February 12, 1985, Safety Evaluation for TMI Item II.B.3

34. Letter from A. E. Lundvall, Jr. (BGE) to A. C. Thadani (NRC), dated October 31, 1985, NUREG-0737, Item II.B.3, Post-Accident Sampling

35. Letter from D. H. Jaffe (NRC) to J. A. Tiernan (BGE), dated May 6, 1986, NUREG-0737, Item II.B.3, Post-Accident Sampling

36. Letter from R. W. Starostecki (NRC) to A. E. Lundvall, Jr. (BGE), dated March 8, 1982, Combined Inspection 50-317/82-04, 50-318/82-04

37. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated November 24, 1982, TMI Items I.A.2.1.4, Upgrading of RO and SRO Training; and II.B.4.1, Training for Mitigation of Core Damage

38. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated April 6, 1982, TMI Action Plan, Item II.D.1

39. Letter from S. A. McNeil (NRC) to G. C. Creel (BGE), dated July 18, 1989, Safety Evaluation Concerning Conformance to NUREG-0737, Item II.D.1, Relief Valve and Safety Valve Testing

40. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated October 19, 1979, Follow-up Action Resulting from TMI-2 Incident (Lessons Learned)

41. Letter from R. W. Reid (NRC) to A. E. Lundvall, Jr., (BGE) dated April 7, 1980, Evaluation of Implementation of Category “A” Lessons Learned

42. Letter from D. G. Eisenhut (NRC) to A. E. Lundvall, Jr. (BGE), dated November 7, 1979, NRC Requirements for Auxiliary Feedwater Systems at Calvert Cliffs Nuclear Power Plant Units 1 & 2

43. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated May 8, 1981, Safety Evaluation Report for Amendment Nos. 54 and 37 to Facility Operating License Nos. DPR-53 and DPR-63 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 & 2

44. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated March 9, 1981, NRC Requirements for Auxiliary Feedwater System

45. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated December 13, 1979, NRC Requirements for Auxiliary Feedwater System at Calvert Cliffs Nuclear Power Plant

46. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated January 26, 1981, NRC Requirements for Auxiliary Feedwater System

47. Letter from A. E. Lundvall, Jr. (BGE) to H. R. Denton (NRC), dated January 22, 1980, Request for Amendment to Operating Licenses

48. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated June 21, 1982, TMI Action Item II.E.1.2, Auxiliary Feedwater (AFW) Automatic Initiation and Flow Indication, for Calvert Cliffs Units 1 & 2

49. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated March 5, 1981, NRC Requirements for Auxiliary Feedwater System

50. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated June 29, 1981, Elimination of Item II.K.2.19 (Sequential Auxiliary Feedwater flow Analysis) Requirement for Licensees with NSSS Designed by W and CE

51. BGE Calculation A-85-009, Calvert Cliffs Natural Circulation Cool Down Auxiliary Feedwater Flow Rate, July 15, 1985

52. Letter from J. A. Tiernan (BGE) to A. C. Thadani (NRC), dated January 20, 1986, Request for Amendment, AFW Flow

53. Letter from A. E. Lundvall, Jr. (BGE) to R. W. Reid (NRC), dated November 9, 1979, Follow-up Actions Resulting from TMI-2 Incident

54. Letter from A. E. Lundvall, Jr. (BGE) to R. W. Reid (NRC), dated November 23, 1979, Follow-up Actions Resulting from TMI-2 Incident Auxiliary Feedwater System Modification

55. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated March 28, 1980, NRC Requirements for Auxiliary Feedwater Systems

56. Letter from J. C. Ventura (Bechtel) to R. F. Ash (BGE), dated October 16, 1981, Auxiliary Feedwater Design Criteria

57. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated April 21, 1981, Issuance of Amendment Nos. 53 and 36 to Facility Operating License Nos. DPR-53 and DPR-63

58. Letter from E. J. Brunner (NRC) to A. E. Lundvall, Jr. (BGE), dated January 15, 1981, Combined Inspection 50-317/80-16 and 50-318/80-15

59. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated September 3, 1993, Operator Examination Report Nos. 50-317/93-11 and 50-318/93-11, Reply to Unresolved Item Nos. 50-317/93-11-01 and 50-318/93-11-01

60. Letter from G. W. Meyer (NRC) to R. E. Denton (BGE), dated March 13, 1994, Combined Inspection Report 50-317/93-11 and 50-318/93-11

61. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BGE), dated February 1, 1982, Issuance of Amendment Nos. 65 and 47 to Facility Operating License Nos. DPR-53 and DPR-63

62. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated November 20, 1979, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned Short Term)

63. Letter from A. E. Lundvall, Jr. (BGE) to R. W. Reid (NRC), dated February 29, 1980, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned)

64. Design Calculations I-90-307 and I-90-119

65. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated December 7, 1981, Request for Amendments

66. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated June 29, 1984, Request for Amendment

67. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated April 9, 1984, Request for Amendment

68. Letter from D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BGE), dated February 22, 1985, Issuance of Amendment Nos. 99 and 81 to Facility Operating License Nos. DPR-53 and DPR-63

69. Letter from A. E. Lundvall, Jr. (BGE) to A. C. Thadani, dated April 16, 1986, Request for Amendment

70. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BGE), dated August 6, 1986, Issuance of Amendment Nos. 120 and 102 to Facility Operating License Nos. DPR-53 and DPR-63

71. Letter from R. W. Starostecki (NRC) to A. E. Lundvall, Jr. (BGE), dated March 25, 1983, NRC Resident Inspector Report 50-317/83-05, 50-318/83-05

72. Letter from T. T. Martin (NRC) to N. J. Bowmaker (BGE), dated June 22, 1983, Combined Inspection Nos. 50-317/82-28 and 50-318/82-24

73. Letter from H. R. Denton to All Reactor Licensees, dated October 30, 1979, Discussion of TMI Lessons Learned Short-Term Requirements

74. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated December 1, 1984, Regulatory Guide 1.97 Review

75. Letter from A. E. Lundvall, Jr. (BGE) to E. J. Butcher, Jr. (NRC), dated July 1, 1985, Inadequate Core Cooling Instrumentation

76. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BGE), dated February 9, 1987, Safety Evaluation of the Inadequate Core Cooling System for Calvert Cliffs Units 1 and 2

77. Letter from J. A. Tiernan (BGE) to NRC Document Control Desk, dated August 9, 1988, Regulatory Guide 1.97 Review Update

78. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated January 28, 1982, Pressurized Thermal Shock

79. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated June 5, 1984, NUREG-0737, II.K.2.13, Thermal-Mechanical Report

80. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated October 19, 1983, Natural Circulation Cooldown

81. Letter from J. R. Miller (NRC) A. E. Lundvall, Jr. (BGE), dated October 24, 1983, Review of NUREG-0737 Item II.K.2.17, Voiding in the Reactor Coolant System During Transients

82. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated August 11, 1981, Response to NUREG-0737, Item II.K.3.1

83. Letter from J. R. Miller (NRC) to A. E. Lundvall, Jr. (BGE), dated October 12, 1983, NUREG-0737, Item II.K.3.1 – Automatic PORV Isolation and II.K.3.2 – Report on PORVs for Calvert Cliffs Units 1 and 2

84. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated February 20, 1981, Response to NUREG-0737

85. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated July 25, 1983, Automatic Trip of Reactor Coolant Pumps

86. Letter from S. A. McNeil (NRC) to G. C. Creel (BGE), dated July 3, 1989, Closure of Generic Letter 86-06 Associated Actions for Calvert Cliffs Units 1 and 2

87. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated August 8, 1983, Safety Evaluation for TMI Action Item II.K.3.17

88. Letter from A. E. Lundvall, Jr. (BGE) to J. R. Miller (NRC), dated November 30, 1984, Reactor Coolant Pump Seal Integrity Following Loss of Offsite Power

89. Letter from E. J. Butcher (NRC) to A. E. Lundvall, Jr. (BGE), dated October 7, 1985, NUREG-0737, Item II.K.3.25, Reactor Coolant Pump Seal Integrity Following Loss of Offsite Power

90. Letter from A. E. Lundvall, Jr. (BGE) to E. J. Butcher (NRC), dated June 28, 1985, Small Break LOCA Method

91. Letter from E. J. Butcher (NRC) to A. E. Lundvall, Jr. (BGE), dated June 28, 1985, NUREG-0737, Item II.K.3.30 Closure

92. NRC Generic Letter 81-10 to All Licensees, dated February 18, 1981, Post-TMI Requirements for the Emergency Operating Facility

93. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated June 1, 1981, Emergency Response Facilities Conceptual Design

94. Letter from R. C. L. Olsen (BGE) to R. A. Clark (NRC), dated December 3, 1982, Emergency Operations Facility

95. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated August 12, 1983, Emergency Operations Facility Review

96. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated May 11, 1983, NUREG-0737, Item III.A.2.1 Emergency Plan Upgrade to Meet the Requirements of Appendix E to 10 CFR Part 50

97. Letter from S. A. McNeil (NRC) to J. A. Tiernan (BGE), dated October 17, 1986, NUREG-0737, Item III.A.2.2, Emergency Response Capability

98. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated February 1, 1980, Follow-up Action Resulting from TMI-2 Incident Lessons Learned

99. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated March 9, 1982, NUREG-0737, Item III.D.3.3, Improved In-Plant Iodine Instrumentation Under Accident Conditions

100. Letter from A. E. Lundvall, Jr. (BGE) to R. A. Clark (NRC), dated July 27, 1982, NUREG-0737 Item III.D.3.4

101. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), September 3, 1982, Review of NUREG-0737 Item III.D.3.4

102. Letter from G. S. Vissing (NRC) to G. Vanderheyden (CCNPP), dated March 2, 2004, Elimination of Requirements for Hydrogen Recombiners and Hydrogen Monitors

103. Letter from R. V. Guzman (NRC) to G. Vanderheyden (CCNPP), dated September 15, 2004, Elimination of Post-Accident Sampling System (PASS) Sampling Requirements

104. Letter from G. S. Vissing (NRC) to P. E. Katz (CCNPP), dated July 16, 2003, Amendment re: Revision to the Administrative Controls Section of the Technical Specifications

105. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated June 29, 1981, Safety Evaluation for Recommendation GL-2

106. NRC Generic Letter 2008-01; Managing Gas Accumulation In Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems, January 11, 2008 (ML072910759)

107. Letter from J. A. Spina (CCNPP) to Document Control desk (NRC), dated October 14, 2008, Nine-Month Response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems" (ML082900149)

108. Letter from T. E. Trepanier (CCNPP) to Document Control Desk (NRC), dated June 12, 2009, Nine-Month Supplemental (Post-Outage) Response to NRC Generic Letter 2008-01 (ML091670262)

109. Letter from G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated June 21, 2010, Nine-Month Supplemental (Post-Outage) Response to NRC Generic Letter 2008-01 (ML101740435)

110. Letter from D. T. Gudger (EGC) to Document Control Desk (NRC), dated July 10, 2014, Application to Revise Technical Specifications to Adopt "Generic Letter 2008-01, Managing Accumulation," using the Consolidating Line Item Improvement Process

111. Letter from A. N. Chereskin (NRC) to D. T. Gudger (EGC), dated July 30, 2015, Issuance of Amendments Regarding Implementation of TSTF-523, "Generic Letter 2008-01, Managing Gas Accumulation" (Amendments 313/291)

APPENDIX 1A
QUALITY ASSURANCE PROGRAM FOR DESIGN AND CONSTRUCTION

LIST OF EFFECTIVE PAGES

APPENDIX 1A

1A QUALITY ASSURANCE PROGRAM FOR DESIGN AND CONSTRUCTION

Appendix 1A, Quality Assurance Program for Design and Construction, is historical information about the construction of the plant. Appendix 1A has been removed from the Safety Analysis Report and has been sent to Plant History. An image of the contents of the appendix can be accessed in the NORMs Records system under Document ID (DOC ID) "Appendix-1A."

APPENDIX 1B
QUALITY ASSURANCE PROGRAM FOR THE OPERATIONS PHASE

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APPENDIX 1B
QUALITY ASSURANCE PROGRAM FOR THE OPERATIONS PHASE

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LIST OF ACRONYMS

ANSI	American National Standards Institute
CCNPP	Calvert Cliffs Nuclear Power Plant
SRO	Senior Reactor Operator
QA	Quality Assurance

APPENDIX 1B

1B QUALITY ASSURANCE PROGRAM FOR THE OPERATIONS PHASE

The Quality Assurance Program for the Calvert Cliffs Nuclear Power Plant is the latest version of the Quality Assurance (QA) Topical Report.

In addition the specific commitments in the Quality Assurance Topical Report, Calvert Cliffs has the following commitments and site-specific exceptions to the following Regulatory Guides and American National Standards Institute (ANSI) Standards.

1B.1 REGULATORY GUIDE 1.8

Personnel Selection and Training (September 1975).

This endorses ANSI N18.1 (March 8, 1971). Calvert Cliffs Takes two exceptions to ANSI N18.1, as follows:

Item 1

Requirement

Paragraph 4.2.2 states that at the time of initial core loading or appointment to the active position, the Operations Manager shall hold a Senior Reactor Operator's (SRO) License.

Paragraph 3.2.1 states that positions at the functional level of Manager are those to which are assigned broad responsibilities for direction of major aspects of a nuclear power plant. This functional level generally includes the plant manager (plant superintendent, or other title), his line assistants, if any, and the principal members of the operating organization reporting directly to the plant manager and having overall responsibility for operation of the plant or for its maintenance or technical service activities.

Response

Calvert Cliffs has two positions in its organization, Manager-Operations and General Supervisor-Shift Operations. Neither of these positions needs to individually meet all of the requirements of both Paragraphs 3.2.1 and 4.2.2. The Manager-Operations will satisfy paragraph 3.2.1 and most of 4.2.2 except that he will not maintain an SRO license. Instead, the Manager-Operations will hold or have held an SRO license. The General Supervisor-Shift Operations will hold and maintain an SRO license. The General Supervisor-Shift Operations satisfies Paragraph 4.2.2, but he does not satisfy 3.2.1 because he does not report directly the plant manager.

Reason

The Manager-Operations will hold or have held an SRO license, as opposed to having a license at the time of appointment to the position. He will have an excellent understanding of plant operations. The General Supervisor-Shift Operations will not only hold an SRO license at the time of appoint to the position, but he will maintain the license. The General Supervisor-Shift Operations directly supervises the operating shift organization, whereas the Manager-Operations is also responsible for procedure development, modifications acceptance, and operations/ maintenance coordination. The Manager-Operations' level of supervision does not require current in-depth and plant specific knowledge that results from maintaining an SRO license.

Item 2

Requirement

Paragraph 3.2.2 states that supervisors are persons principally responsible for directing the actions of operators, technicians, or repairmen. Those positions usually designated as intermediate and first line supervisors are included in this category.

Paragraph 4.3.2 states that supervisors not requiring Atomic Energy Commission licenses shall have a high school diploma or equivalent and a minimum of four years of experience in the craft or discipline he supervises.

Response

Calvert Cliffs has three supervisory positions in its organization - Supervisors, and in some cases Assistant General Supervisors and General Supervisors - that are organizationally equivalent (when supervising technicians/repairmen) to the positions described in paragraph 3.2.2 of ANSI N18.1 (March 8, 1971). All these individuals need not possess the four years of craft/discipline experience required by paragraph 4.3.2. Instead, at least the first line supervisor shall possess four years experience in the craft/discipline he supervises while other supervisors in the organization may be selected to fill supervisory positions based on possessing a minimum of an Associate's Degree, with four years or related technical experience, and demonstrated supervisory ability. Additionally, all first line and intermediate supervisors shall have at least a high school diploma or equivalent.

Reason

To provide a balanced and broad base of supervisory ability within the site organizations made up of technicians/repairmen, it is desirable to include as supervisors both individuals with extensive craft/discipline experience accrued through field work and individuals with related education and experience who have demonstrated the ability to effectively supervise.

1B.2 REGULATORY GUIDE 1.16

Reporting of Operating Information - Appendix A Technical Specifications (Revision 4, August 1975).

Calvert Cliffs is committed to this standard, as modified by the Calvert Cliffs Technical Specifications.

1B.3 REGULATORY GUIDE 1.26, REVISION 3

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.

Calvert Cliffs takes the following alternative:

Calvert Cliffs' Quality Assurance program is applied to structures, systems, components, and activities that have been designated safety-related because they prevent accidents or mitigate the consequences of postulated accidents that could cause undue risk to the health or safety of the public. The QA program is also applicable to designated non-safety-related structures, systems, components, activities, and services as required by regulations. Designated non-safety-related program requirements are based on a graded approach to Quality Assurance required to meet applicable regulatory designated requirements and guidance. The level of QA program controls placed on designated non-safety-related items are defined in QA program documents and/or implementing procedures. Controls have been established for specifying on a Quality List all safety-related structures, systems, components, and activities that are subject to the requirements of the QA program.

1B.4 REGULATORY GUIDE 1.37

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 16, 1973).

This endorses ANSI N45.2.1 (February 26, 1973). Calvert Cliffs takes one exception to ANSI N45.2.1, as follows:

Requirement

Subsection 3.2 outlines requirements for demineralized water.

Response

Calvert Cliffs specifications for demineralized water are different from the specifications outlined in the standard.

Reason

Calvert Cliffs specifications for demineralized water are consistent with guidelines provided by the Nuclear Steam Supply System supplier. Calvert Cliffs specifications are generally more restrictive than those specified by ANSI N45.2.1.

1B.5 REGULATORY GUIDE 1.38

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Revision 2, May 1977).

This endorses ANSI N45.2.2 (December 20, 1972). Calvert Cliffs takes seven exceptions to ANSI N45.2.2, as follows:

Item 1

Requirement

Subsection 2.4 could be interpreted to mean that on-site and off-site personnel who perform any inspection, examination, or testing activities related to the packing, shipping, receiving, storage, and handling of items for nuclear power plants shall be qualified in accordance with ANSI N45.2.6.

Response

Calvert Cliffs requires that only persons who are responsible for approving items for acceptance shall be qualified in accordance with Regulatory Guide 1.58 (which endorses ANSI N45.2.6) and that personnel who verify that storage areas meet requirements will be qualified to either Regulatory Guide 1.58 (which endorses ANSI N45.2.6) or ANSI N45.2.23.

Reason

Our receipt inspection procedures require persons who approve items for acceptance to be qualified in accordance with Regulatory Guide 1.58 (which endorse ANSI N45.2.6). Quality and Performance Assessment assessors verify that storage areas meet requirements. All other inspection, examination, and testing activities are subject to review by persons qualified to Regulatory Guide 1.58 (which endorses ANSI N45.2.6).

Item 2

Requirement

The second sentence of Subsection 2.4 requires that:

Off-site inspection, examination, or testing shall be audited and monitored by personnel who are qualified in accordance with ANSI N45.2.6.

Response

Calvert Cliffs uses personnel qualified in accordance with ANSI N45.2.23 to perform auditing and monitoring functions.

Reason

The qualification requirements for auditors cannot always be met by persons qualified to Regulatory Guide 1.58 (which endorses ANSI N45.2.6).

Item 3

Requirement

Subsection 2.7 requires that activities covered by the Standard shall be divided into four levels, though recognizing that within the scope of each level there may be a range of controls depending on the importance of the item to safety and reliability.

Response

1. The level of protective measures defined by Subsection 2.7 are applied to Basic Component Purchases.
2. Engineering or Supply Chain personnel will determine the level of protective measures to be applied to Commercial Grade purchases.

Reason

Calvert Cliffs' position is as follows:

1. For Commercial Grade items, it is not always possible to assign a level of classification in accordance with ANSI N45.2.2, as many items are purchased after they have been packaged by the manufacturer and shipped to his local agent, the wholesaler.
2. Experience has shown that the level of protection assigned to Commercial Grade items by vendors is adequate.

Item 4

Requirement

Subsection 3.0 specifies detailed requirements for packing items for each level defined in Subsection 2.7.

Response

Calvert Cliffs has replaced Section 3.0 with the following:

1. Packaging for Shipment to Calvert Cliffs Nuclear Power Plant (CCNPP)

   Engineering or Supply Chain personnel shall ensure that procurement documents for Basic Component and Commercial Grade item purchases either indicate that the normal methods of packaging and shipment used by industry in general are acceptable for the items being procured or specify the level of protection assigned to the item and the requirement that the vendor conform to applicable requirements for items in that classification defined in Regulatory Guide 1.38, Revision 2 - March 1977.

2. The normal methods of packaging used by the industry in general are acceptable for items being procured as Commercial Grade.

3. Packaging for Storage by CCNPP

   In general, the packaging used by the vendor to ship items for all types of purchases to CCNPP need not be retained after the item is received by CCNPP, provided that the item is stored in an area that meets the requirements for a storage area for the level of protection assigned to the item. Special or unique items, however, may require special protective measures. For such unusual items, the Department that initiated the purchase, together with Engineering or Supply Chain personnel shall identify if any of the requirements of Section 6.4.2 of ANSI N45.2.2-1972 apply.

<u>Reason</u>

1. This substitution will ensure that the item will receive adequate protection during shipment and storage, thus eliminating unnecessary restrictions and enabling CCNPP to use commercial sources to the utmost.
2. Experience shows that industrial practices for packaging Commercial Grade items are adequate for most applications.

<u>Item 5</u>

<u>Requirement</u>

Section 4.0 defines shipping requirements related to the protection levels assigned to items.

<u>Response</u>

Calvert Cliffs has replaced Section 4.0 with the following:

1. Shipping to CCNPP

   Calvert Cliffs will invoke the requirements for shipping specified in Section 4.0 of ANSI N45.2.2-1972 on Basic Component purchases only when Engineering or Supply Chain personnel have specified in procurement documents that the item shall be packaged in conformance with ANSI N45.2.2, Section 3.8.

   Calvert Cliffs will not invoke the requirements of ANSI N45.2.2-1972, Section 4.0, on Commercial Grade item purchases.

2. Shipping from CCNPP

   Items shipped from CCNPP need not conform to any of the requirements of ANSI N45.2.2, but the organization that packs and handles the item shall provide roughly the same level of protection that the item was given during shipment to CCNPP.

<u>Reason</u>

If engineering personnel have determined that the vendor's methods of packaging are acceptable, they have already determined that the suppliers's methods of shipping are adequate. As items are shipped from CCNPP only for repair, the detailed requirements specified in Section 4.0 of ANSI N45.2.2 are not necessary.

<u>Item 6</u>

<u>Requirement</u>

Subsection 6.4 gives detailed requirements for care of items in storage, according to the protection levels assigned to the items.

<u>Response</u>

Calvert Cliffs does not require items to be stored in the packing used for shipment if the storage level in the area provides the same protection as the level of packing assigned to the items. Caps, covers, etc., will be required only if specified by Engineering or Supply Chain personnel during the procurement process. If an item is taken from one storage area to another, however, the persons who move it are responsible for ensuring, as applicable, that additional packing is supplied to give adequate protection during transportation.

Reason

The degree of protection given an item during storage should be tailored to the importance of the item to safety and the probability of deterioration during storage; to base storage requirements purely on the categories in Subsection 2.7 of ANSI N45.2.2-1972 is impractical. Calvert Cliffs requires Engineering and Supply Chain personnel to specify requirements more closely related to the actual function of items and to storage conditions.

Item 7

Requirement

Subsection 7.3.3 requires compliance with a series of ANSI documents.

Response

Calvert Cliffs controls for the use of hoisting equipment are compatible with the Standards listed in Subsection 7.3.3 of ANSI N45.2.2, although at the discretion of the Plant General Manager, they need not be compatible with documents referred to in these documents.

Reason

Lower-level documents referred to in the documents listed in Subparagraph 7.3.3 will not necessarily affect the ability of CCNPP personnel to properly handle safety-related items and could lead to confusion.

1B.6 REGULATORY GUIDE 1.54

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (June 1973).

This endorses ANSI N101.4 (November 28, 1972). Calvert Cliffs takes an exception to ANSI N101.4, as follows:

Requirement

Section 1.2 specifies applicability requirements for the Standard.

Response

Calvert Cliffs requires that only activities performed inside containment structures and related to protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces shall conform to applicable Section of ANSI N101.4.

Reason

Deterioration of protective coatings applied to surfaces outside containment structures would have no detrimental effects on the safe operation of the plant.

1B.7 REGULATORY GUIDE 1.68

Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors (November 1973).

Calvert Cliffs is committed to this standard. The preoperational and initial startup test program was completed prior to each Unit startup and is described in Section 13.1.

1B.8 REGULATORY GUIDE 1.94

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Revision 1, April 1976).

In lieu of this Regulatory Guide, Calvert Cliffs conforms to ANSI N45.2.5, Draft 3, Revision 1 (November 1973).

APPENDIX 1C
AEC PROPOSED GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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APPENDIX 1C
AEC PROPOSED GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

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LIST OF ACRONYMS

ASTM	American Society for Testing and Materials
GDC	General Design Criteria
LOCA	Loss-of-Coolant Accident
NDTT	Nil-Ductility Transition Temperature
SACM	Societe Alsacienne De Constructions Mecaniques De Mulhouse

APPENDIX 1C

1C.0 AEC PROPOSED GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

1C.1 INTRODUCTION

On July 10, 1967, the Atomic Energy Commission published the proposed General Design Criteria (GDC) for Nuclear Power Plants. These 70 criteria were issued for public comment. In 1971, the final version of the GDC was made law, as Appendix A to 10 CFR Part 50. This version contained 64 criteria. The purpose of the final GDC is to "… establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." The final GDC contained in Appendix A to 10 CFR Part 50 are to be used for applying for a construction permit.

Baltimore Gas and Electric Company obtained a construction permit on July 7, 1969, to construct both Calvert Cliffs Units 1 and 2. The construction permit was obtained from the Atomic Energy Commission after their review of our Preliminary Safety Analysis Report which contained an assessment of our compliance with the draft GDC. By the time the final GDCs were issued, we already had our construction permit and, therefore, are one of the "… plants for which construction permits have been issued …" The final GDC were not used to establish the principal design criteria for Calvert Cliffs.

In 1971, Baltimore Gas and Electric Company submitted the Final Safety Analysis Report for Calvert Cliffs Units 1 and 2. Included in the Final Safety Analysis Report was an assessment of the plant design against the draft GDC. That assessment has not been updated since 1974.

For convenience, the draft GDC are reproduced in this appendix as they appeared in 1967. The 1974 assessment of plant design against these draft GDC is not included since it is significantly out-of-date. Calvert Cliffs was designed and constructed to meet the intent of the draft GDC. Modifications to the facility are evaluated in accordance with 10 CFR 50.59 to assess consistency with the current licensing basis (including the draft GDC, as applicable).

In 1996, Baltimore Gas and Electric Company completed the addition of a new safety-related Societie Alsacienne De Construction Mechaniques Del Melhouse (SACM) diesel generator. This emergency diesel generator was installed to comply with the Station Blackout Rule (10 CFR 50.63). Calvert Cliffs Nuclear Power Plant compliance with the Station Blackout Rule is described in Section 1.8.2. The new safety-related diesel generator (Emergency Diesel Generator 1A), its support systems, and the building which houses it, were designed to the GDC that appear in Appendix A to 10 CFR Part 50. The specific design criteria that applies to Emergency Diesel Generator 1A and its support systems are:

Criterion 1	Criterion 17
Criterion 2	Criterion 18
Criterion 3	Criterion 44
Criterion 4	Criterion 45
Criterion 5	Criterion 46
Criterion 13	

1C.2 OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

CRITERION 4 - SHARING OF SYSTEMS (Category 4)

Reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

1C.3 PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

1C.4 NUCLEAR AND RADIATION CONTROLS

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR Part 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems together with associated equipment shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

1C.5 RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failure resulting from a single event shall be treated as a single failure.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation, and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of protection function.

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

1C.6 REACTIVITY CONTROL

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

1C.7 REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagation type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) the provisions for control over service temperature and irradiation effects which may require operational restrictions.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil-ductility transition temperature (NDTT) of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDTT of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with American Society for Testing and Materials (ASTM) E-185-66 shall be provided.

1C.8 ENGINEERED SAFETY FEATURES

A. GENERAL REQUIREMENTS

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the system, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power system shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident (LOCA).

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

B. EMERGENCY CORE COOLING

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a LOCA, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a LOCA and is not lost during the entire period this function is required following the accident.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 47 - TESTS OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

C. CONTAINMENT

CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

CRITERION 50 - NIL-DUCTILITY TEMPERATURE REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load-carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above NDTT.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

D. CONTAINMENT PRESSURE REDUCING

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEM (Category A)

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

E. AIR CLEANUP

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided for in-situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

CRITERION 65 - TESTING OF OPERATION SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

1C.9 FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR Part 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

1C.10 PLANT EFFLUENTS

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

FIGURE 1-1 SITE PLAN

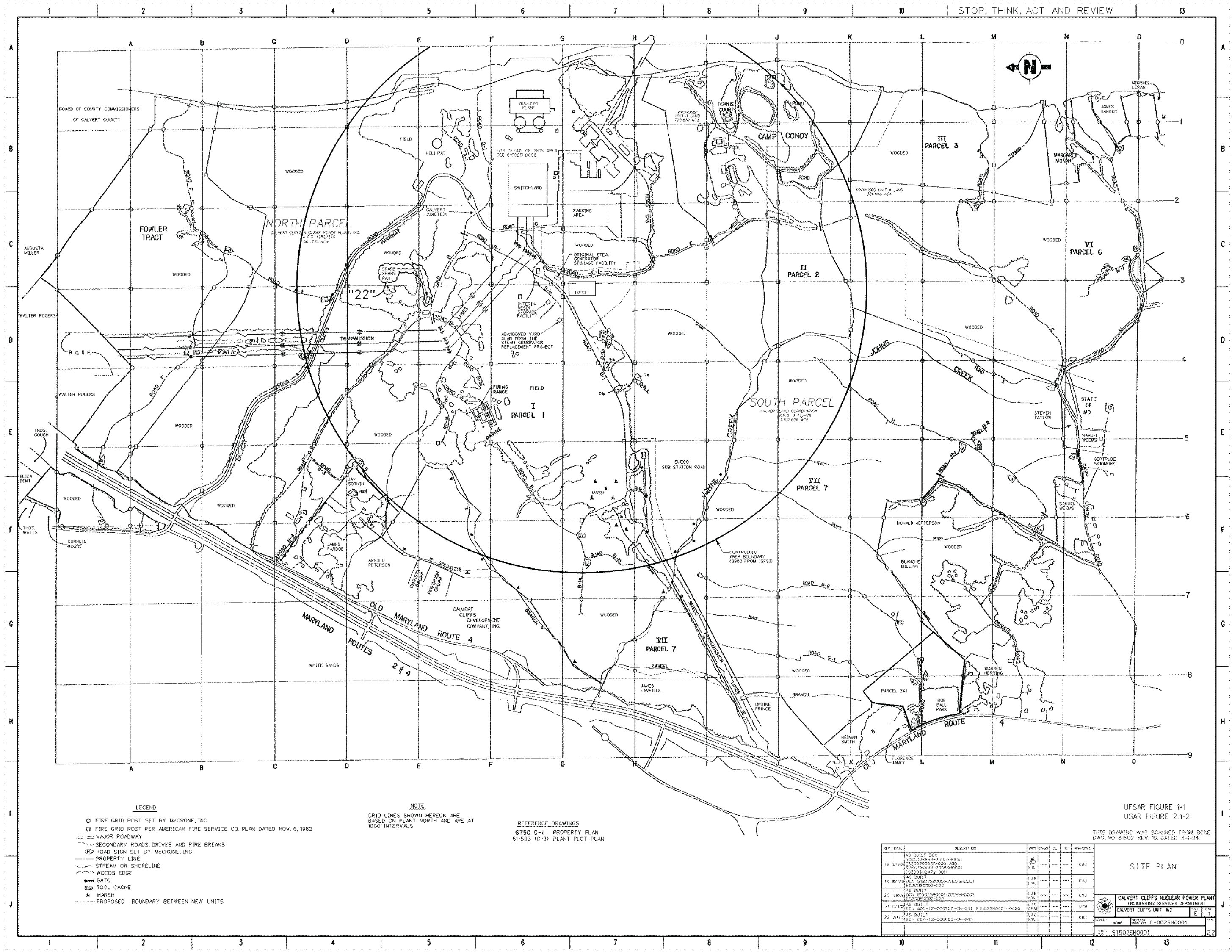
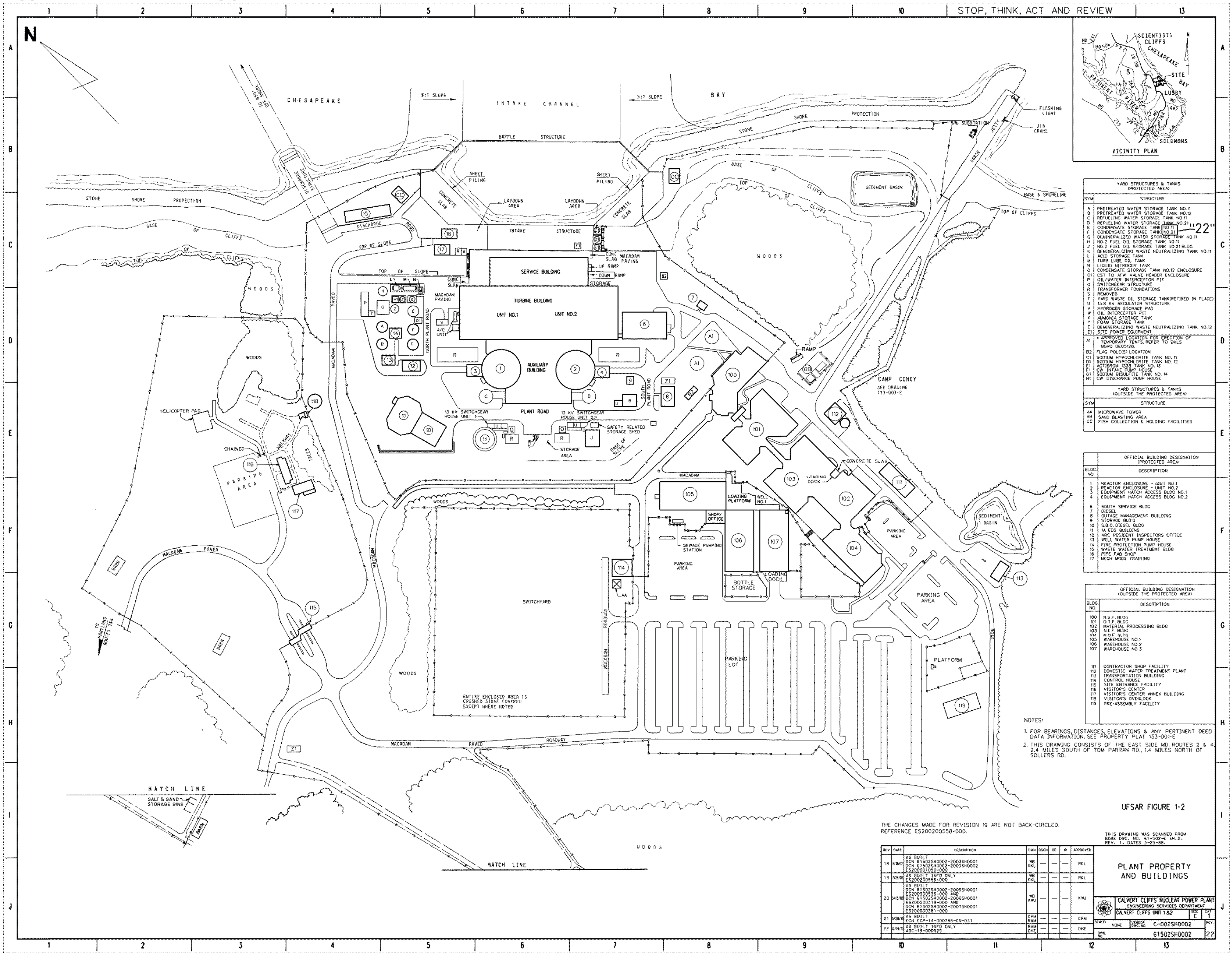
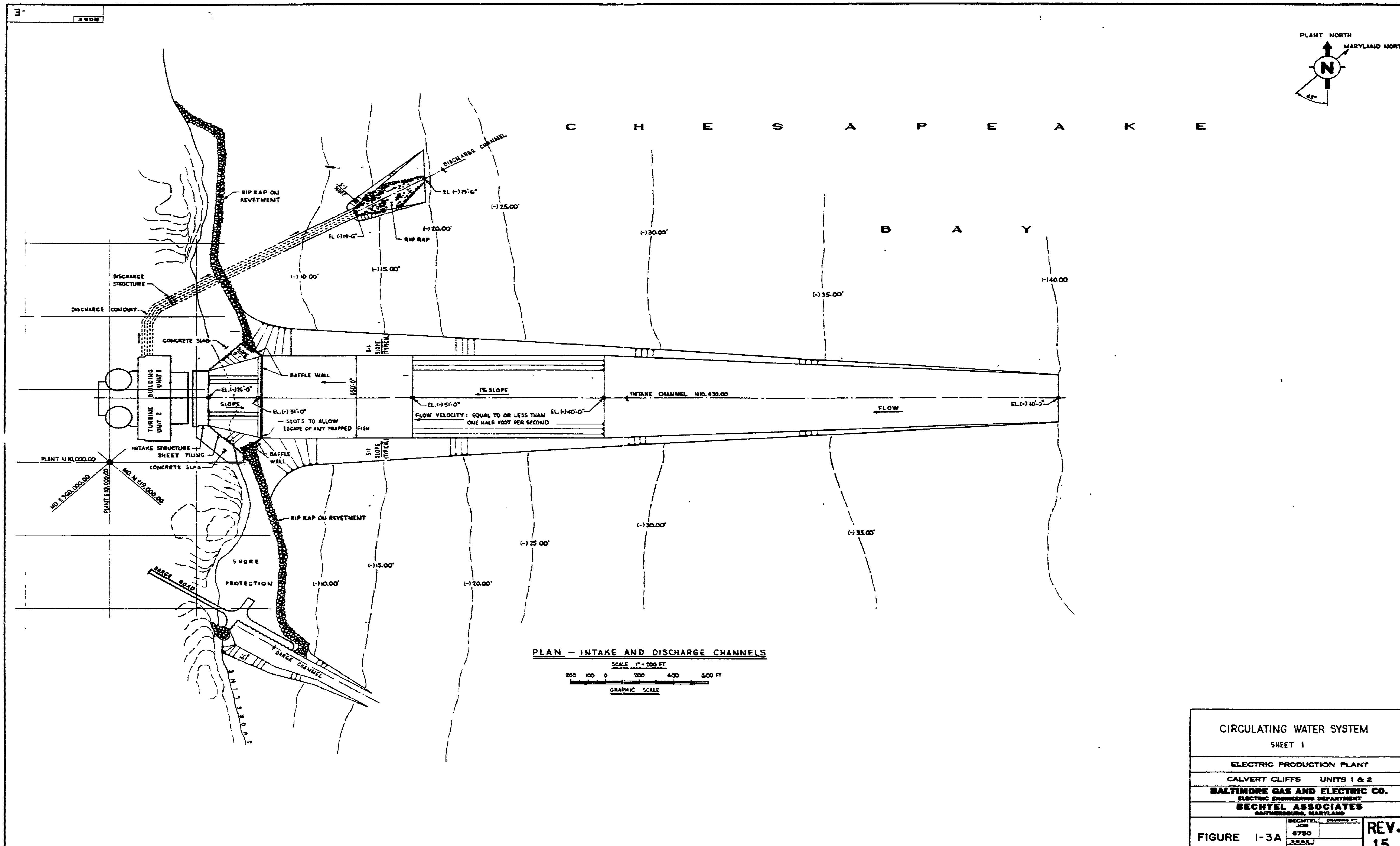
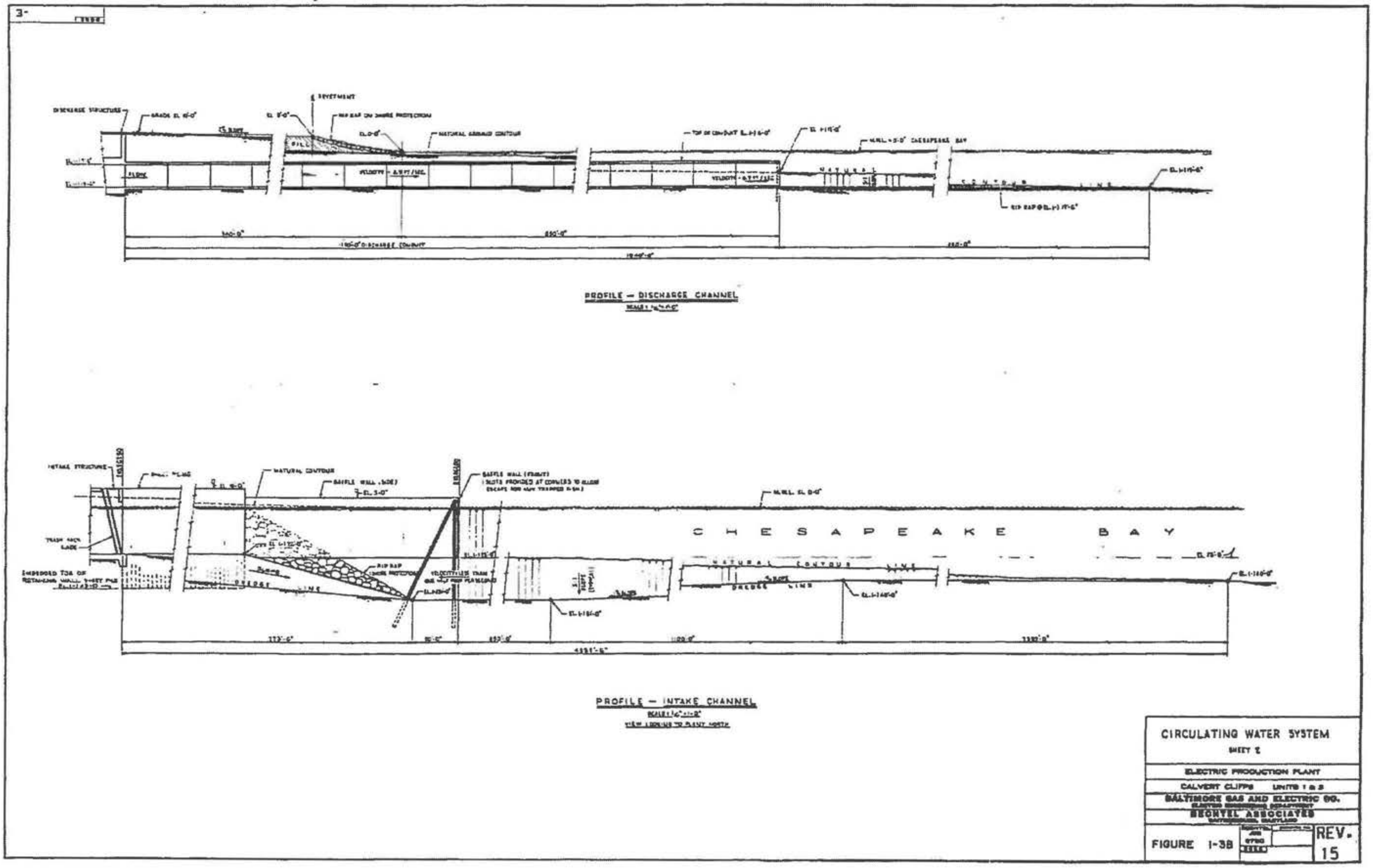
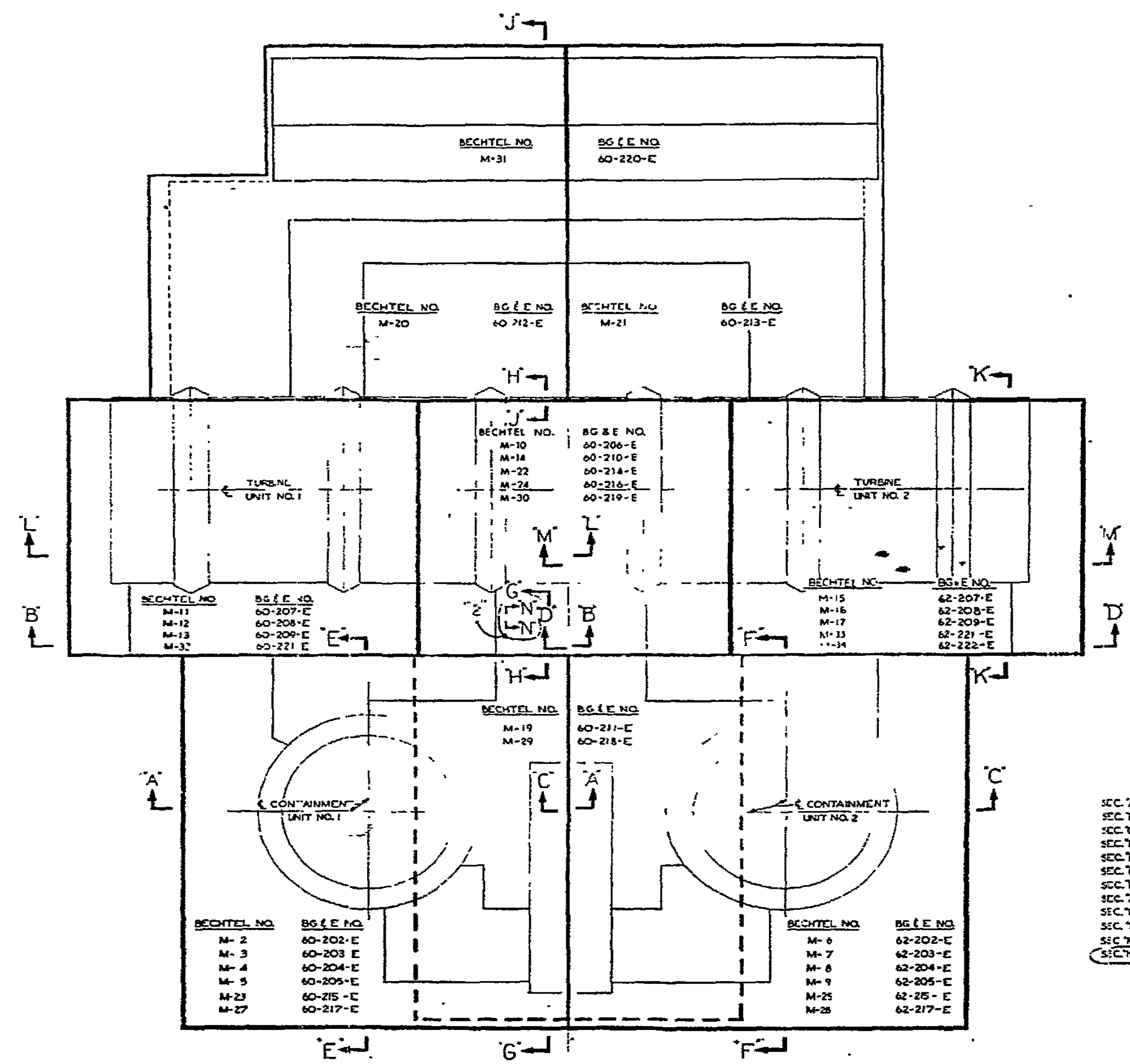


FIGURE 1-2 CALVERT CLIFFS SITE PLAN




PLANT NORTH
MARYLAND NORTH
C H E S A P E A K E
B A Y
DISCHARGE CHANNEL
RIP RAP ON REVETMENT
RIP RAP
DISCHARGE STRUCTURE
DISCHARGE CONDUIT
CONCRETE SLAB
BAFFLE WALL
TURBINE BUILDING UNIT 1 UNIT 2
INTAKE STRUCTURE
SHEET PILING
CONCRETE SLAB
1% SLOPE
INTAKE CHANNEL N 10,430.00
FLOW
SLOTS TO ALLOW ESCAPE OF ANY TRAPPED FISH
FLOW VELOCITY: EQUAL TO OR LESS THAN ONE HALF FOOT PER SECOND
RIP RAP ON REVETMENT
SHORE PROTECTION
BARGE ROAD
BARGE CHANNEL
SHORELINE
PLAN – INTAKE AND DISCHARGE CHANNELS
SCALE 1" = 200 FT
GRAPHIC SCALE
CIRCULATING WATER SYSTEM
SHEET 1
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNITS 1 & 2
BALTIMORE GAS AND ELECTRIC CO.
BECHTEL ASSOCIATES
FIGURE 1-3A
REV. 15


PROFILE – DISCHARGE CHANNEL
C H E S A P E A K E B A Y
PROFILE – INTAKE CHANNEL
CIRCULATING WATER SYSTEM
ELECTRIC PRODUCTION PLANT
BALTIMORE GAS AND ELECTRIC CO.
BECHTEL ASSOCIATES
FIGURE 1-3B
REV. 15



SECTION DRAWINGS

	BECHTEL NO.	BG & E NO.
SEC. "A"-"A"	M-23	60-215-E
SEC. "B"-"B" & "D"-"D"	M-24	60-216-E
SEC. "C"-"C"	M-25	62-215-E
SEC. "E"-"E"	M-27	60-217-E
SEC. "F"-"F"	M-28	62-217-E
SEC. "G"-"G"	M-29	60-218-E
SEC. "H"-"H"	M-30	60-219-E
SEC. "J"-"J"	M-31	60-220-E
SEC. "K"-"K"	M-34	62-222-E
SEC. "L"-"L"	M-32	60-221-E
SEC. "M"-"M"	M-33	62-221-E
SEC. "N"-"N"	M-34	60-210-E

REV NO.	DATE	DESCRIPTION	DRN.	CHK.
2		AS BUILT INFO ONLY DCR 84-N-420, (FCR 83-1005)		
1		DELETED SEC "K-K" ON M-30 & ADDED M 34		
0		ISSUED FOR CONSTRUCTION	ES	WHS

EQUIPMENT LOCATION DRAWINGS - KEY PLAN

ELECTRIC PRODUCTION PLANT

CALVERT CLIFFS UNIT 1 & 2

BALTIMORE GAS AND ELECTRIC CO.
ELECTRIC ENGINEERING DEPARTMENT

BECHTEL ASSOCIATES

FSAR FIG NO. 1-4

BECHTEL JOB 6750

DRAWING NO. M-1

60-201-E

REV 2

FOR INFORMATION ONLY

FSAR FIG. NO. 1-4

Revision 4

[redacted]

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SITE AND ENVIRONMENT

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FIGURE

CHAPTER 2
SITE AND ENVIRONMENT

LIST OF FIGURES

FIGURE

CHAPTER 2
SITE AND ENVIRONMENT

LIST OF ACRONYMS

AEC	Atomic Energy Commission
ANA	Annapolis, MD
ASTM	American Society for Testing and Materials
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
DCA	Washington National Airport
ESSA	Environmental Science Services Administration
LNG	Liquified Natural Gas
MLW	Mean Low Water
MRI	Meteorology Research, Inc.
MSL	Mean Sea Level
NATC	Naval Air Test Center
NHK	
NHT	Normal High Tide
NPDES	National Pollution Discharge Elimination System
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
PAX	Patuxent
PMH	Probable Maximum Hurricane
PPSP	
PSAR	Preliminary Safety Analysis Report
RIC	Byrd Field, Richmond
SSE	Safe Shutdown Earthquake
STUB	
YMCA	Young Men's Christian Association

2.0 SITE AND ENVIRONMENT

2.1. GENERAL DESCRIPTION

This section presents data on the site and environs for the Calvert Cliffs Nuclear Power Plant (CCNPP). These data were used to establish a basis for the selection of design standards for the plant and to determine the adequacy of concepts for controlling routine and accidental release of radioactive effluents to the environment. A series of studies (population and land use, meteorology, geology, hydrology, and seismology) has been conducted.

The site is located in Calvert County, MD, approximately 10-1/2 miles southeast of Prince Frederick, MD, and on the west bank of the Chesapeake Bay.

Cooling water for the plant is drawn from and returned to the Chesapeake Bay.

The exclusion area around the plant has a minimum radius of 1150 meters. The distance to the nearest permanent residence is approximately one mile. A summer camp presently included in the plant site was abandoned in December 1971. The closest major metropolitan area is Washington, DC, approximately 45 miles to the northwest.

The structures are founded on Miocene sediments of the Coastal Plain Physiographic province. The foundation materials presented no special problems in design or construction. Foundations have been designed in accordance with the considerations discussed in the report on subsurface conditions and foundations, Section 2.7.

2.2 POPULATION AND LAND USE

2.2.1 GENERAL

This section of the report presents the results of a population and land-use study. References 1 through 14 used for this study include United States Census data for Maryland, Virginia, Delaware, and Washington, DC; planning reports for various areas of Maryland; maps and aerial photographs of the site and surrounding area; and discussions with various individuals (References 15 through 18). A list of references is presented in Section 2.2.7.

2.2.2 LOCATION

The site is located in Calvert County, MD, on the west bank of the Chesapeake Bay, approximately 10-1/2 miles southeast of Prince Frederick, MD. It originally covered an area of approximately 1135 acres and was owned by Baltimore Gas and Electric Company (BGE). The site boundary is posted and a fence has been erected around the immediate plant area.

This site originally included Camp Conoy, a summer camp previously operated by the Baltimore YMCA. The camp was operated through December 1971, at which time it was abandoned. The camp was used by BGE for various recreational purposes.

Camp Bay Breeze, also a summer camp, is two miles to the southeast and has a seasonal population of approximately 140. Nearby communities include: Calvert Beach and Long Beach, approximately 3 miles to the northwest; Cove Point, approximately 4-1/2 miles to the southeast; Chesapeake Ranch Estates, approximately 6 miles to the south-southwest; and the Patuxent Naval Air Test Center (NATC), approximately 10 miles to the south. Cultural features in the region and area are shown on Figures 2.2-1 and 2.2-2, Regional Map and Site Vicinity Map, respectively. The low population zone as defined in 10 CFR 100.2(b) is shown on Figure 2.2-13.

The metropolitan centers closest to the site are: Washington, DC, approximately 45 miles to the northwest; Baltimore, MD, approximately 60 miles to the north; Richmond, VA, approximately 80 miles to the southwest; and Norfolk, VA, approximately 110 miles to the south.

2.2.3 PRESENT POPULATION

The estimated 1970 population density is shown on Figure 2.2-3, Regional Map, Showing Present and Future Population Density 0-50 Miles, and on Figure 2.2-4, Site Vicinity Map, Showing Present and Future Population Density 0-10 Miles. Estimated 1970 population distribution is shown on Figure 2.2-5, Regional Map, Showing Present and Future Population Distribution 0-50 Miles, and on Figure 2.2-6, Site Vicinity Map, Showing Present and Future Population Distribution 0-10 Miles.

The 1970 population estimates are predicated upon 1970 census data and extrapolation of past population trends for cities, towns, election districts, and minor civil divisions. The population within each of these various political subdivisions is assumed to be uniformly distributed. All estimates include both seasonal and permanent population. (Seasonal population estimates are based on housing data from the United States Census of Housing and on residence classification data.) Population estimates within a five-mile radius of the site are based on a count for houses shown on the 1959 Calvert County General Highway Map, assuming four people per house. The estimates were extrapolated to 1970, based on the growth history of the area. House counts have been confirmed by recent aerial photographs for the immediate vicinity of the site.

The population estimates indicate that the site and surrounding area are sparsely populated with the exception of localized areas along the coast of Chesapeake Bay which attract many summer residents. The summer seasonal residents account for approximately 20% of the total population within 10 miles of the site. The region surrounding the site is predominantly rural in character. Table 2-1 lists the communities within 30 miles of the site with 1970 populations greater than 1,000.

The total 1970 population, including seasonal residents, within 10 miles of the site was estimated to be 16,827. The population is distributed throughout the area and includes many small communities with population less than 1,000. The County Seat, Prince Frederick, is located 10-1/2 miles northwest of the site and has a total population of about 605. The population of the larger communities located within 10 miles of the site is presented in Table 2-2.

A new community, known as Chesapeake Ranch Estates, is located 6 miles south-southeast of the site. The present (1970) permanent population of this development is approximately 180. In the summer, the population is approximately 1,000 during the week, and reaches a maximum of approximately 2,000 on weekends.

All of the above populations are included in the data presented on Figures 2.2-3 through 2.2-6.

The character of the area begins to change from rural to suburban as the major population center of metropolitan Washington, DC, is approached. As indicated in Table 2-3, Accumulative Population Summary - 1970, the rate of change is greatest within 10 to 20 miles of Washington, DC, more than 30 miles from the site.

The sector containing the maximum 1970 population is bounded by the west and northwest radial lines as shown on Figure 2.2-5. The estimated population of this sector, out to 50 miles from the site, is approximately 1,160,000. The data show that 96.6% of the present population in this sector is located more than 30 miles from the site. The population estimates also indicate that 97% of the 2010 population in this sector will be located more than 30 miles from the site.

2.2.4 FUTURE POPULATION

Estimates of population density and distribution for the year 2010 are presented on Figures 2.2-3 through 2.2-6. The population estimates are based on an extrapolation of the past growth history of the region and on future population estimates made by the Maryland State Planning Department.

The published future population estimates extend through 1985. These were extrapolated to 2010 for purposes of this study. In areas where the present population was less than 50, estimates of the 2010 population are based on an anticipated population density of 250 persons per square mile. This method of computation was necessary only for certain areas within 10 miles of the site. Estimates of the future development of Chesapeake Ranch Estates indicate a maximum future population of 28,000. This estimate is included in the data presented on Figures 2.2-3 through 2.2-6.

With continued growth of the Washington, DC metropolitan area, moderate population gains can be expected in the outlying regions, including portions of northern Calvert County and eastern Charles County which are within 15 to 25 miles of the site. Considerable population gains are expected in the area near Washington and Baltimore as part of the growth of the Boston to Washington "megalopolis." Table 2-4 presents the accumulative 2010 population within various distances from the site.

2.2.5 LAND USE

In 1959, 85,400 acres (61%) of the land in Calvert County was devoted to farms, 51,200 acres (36.5%) to forests, and 3,500 acres (2.5%) to other uses. Of the 3,500 acres not used for farms or occupied by forests, 79% was residential; 5% was commercial; 3% was industrial; and 13% was devoted to public and semi-public use.

Dairy farming is of minor importance in Calvert County. In 1959, there were five dairy farms in the county and in 1964 the number decreased to one. There is, however, no dairy farm within a 5 mile radius of the plant.

As stated previously, approximately 61% of the land in Calvert County was devoted to farms in 1959. In 1964, it declined to approximately 53%. The amount of harvested cropland declined slightly over this period, from 16,800 acres to 16,100 acres. The majority of the harvested cropland was used for growing tobacco, corn, and hay, as shown in Table 2-5, Agricultural Land Use - 1959 and 1964.

Within 25 miles of the site, over 90% of the land area is located in Calvert, Charles, Dorchester, and St. Mary's Counties. In 1959, approximately 512,700 acres (49%) of the land in these counties was devoted to farms. In 1964, it declined to approximately 45%. The amount of harvested cropland declined over this period from 180,000 acres to 170,500 acres. The major crops grown in the four-county area are shown in Table 2-6, Agricultural Land Use - 1959 and 1964.

With continued population growth, it is anticipated that the percentage of land devoted to farms will continue to decline and will be accompanied by increased residential and commercial use. However, the overall character of the area is expected to remain essentially rural.

The waters adjacent to the site are used for commercial fishing, primarily for shellfish, such as clams, oysters, and crabs. Calvert County accounted for approximately 2% of the State's total fish catch in 1963.

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2.2.6 SUMMARY

The site is located in an undeveloped, sparsely populated area. The present population within 30 miles of the site is small. However, moderate increases, on the order of 1.5%/year, are estimated over the next 40 years. At present, more than 90% of the

population within 50 miles of the site is located at distances greater than 30 miles. This trend is expected to continue for the expected life of the CCNPP.

At the present time, the major portion of the land in the area surrounding the site is devoted to agricultural and forest uses. Although the amount of land devoted to farming is declining, agriculture should continue to be a primary land use during the life of the proposed nuclear plant. Small increases in the amount of land devoted to residential and commercial use will occur with increased population growth.

The waters of the Chesapeake Bay are now and should remain a source of sea food, primarily clams, oysters, and crabs.

From a population and land-use standpoint, the site is suitable for the location of a nuclear power plant.

2.2.7 REFERENCES

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3. Maryland State Planning Department Newsletter, July 1967, Volume XX, No. 4

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6. Regional Planning Council, Baltimore, Maryland, September 1965 - Regional Plan Alternatives, Publication 5

7. United States Bureau of the Census, 1964 Census of Agriculture, Statistics for the State and Counties, Maryland, Volume 1, Part 23, United States Government Printing Office, Washington, DC

8. United States Bureau of the Census, 1950 Census of Housing, Volume 1 General Characteristics, Parts 2, 3, and 6, United States Government Printing Office, Washington, DC

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10. United States Bureau of the Census, 1970 Census of Population, Volume 1, Characteristics of the Population, Parts 9,10, 22, and 46, United States Government Printing Office, Washington, DC

11. United States Bureau of the Census, U.S. Census of Population: 1970, Number of Inhabitants -- Delaware, 1971

12. United States Bureau of the Census, U.S. Census of Population: 1970, Number of Inhabitants -- Maryland, 1971

13. United States Bureau of the Census, U.S. Census of Population: 1970, Number of Inhabitants -- Virginia, 1971

14. United States Department of Commerce, Bureau of the Census, Election Districts, 1960 Minor Civil Division Maps for the States of Delaware, Maryland and Virginia

15. Chesapeake Ranch Club; Lusby, MD; Mr. Jetmore

16. Girl Scouts Council of the Nations Capital; Arlington, VA; Mr. Slover

17. Patuxent NATC; Lexington Park, MD; Housing Personnel

18. Young Men's Christian Association; Baltimore, MD; Mr. Moss

19. Final Environmental Impact Statement, Increased Flight and Related Operations in the Patuxent River Complex, Patuxent River, Maryland: Department of the Navy, Naval Air Warfare Center Aircraft Division, December 1998

20. Letter from Ms. D. M. Skay (NRC) to Mr. C. H. Cruse (CCNPP), dated August 29, 2001, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Aircraft Hazards Analysis (TAC Nos. MA7229 and MA7230)

TABLE 2-1

COMMUNITIES WITHIN 30 MILES OF THE SITE WITH POPULATION OF 1,000 OR GREATER

COMMUNITY	POPULATION[a] 1940	1950	1960	1970	DISTANCE AND DIRECTION FROM SITE (miles)
Patuxent NATC	[c]	[c]	l,900[b]	2,100[b]	10-S
Lexington Park	[c]	[c]	7,039	9,136	12-S
Leonardtown	668	1,017	1,281	1,406	14-SW
Cambridge	10,102	10,351	12,239	11,600	21-ENE
St. Michaels	1,309	1,470	1,484	1,456	26-NNE
Waldorf	[c]	[c]	1,048	7,368	27-WNW
Easton	4,428	4,836	6,337	6,809	30-NE
La Plata	488	780	1,214	1,561	30-WNW

[a] Based on United States Census Bureau Statistics.

[b] Estimated.

[c] Population less than 1,000. Exact number not available.

TABLE 2-2

POPULATION OF COMMUNITIES NEAR THE SITE

COMMUNITY	ESTIMATED 1970 POPULATION	DISTANCE AND DIRECTION FROM SITE
Calvert Beach and Long Beach	935	3 NW
Cove Point	385	4-1/2 SE
Kenwood	275	5-1/2 NW
Scientists Cliffs	660	7 NNW
Solomons	350	8 S
Dares Beach	825	10 NNW
Prince Frederick	605	10-1/2 NW

TABLE 2-3

ACCUMULATIVE POPULATION SUMMARY - 1970

RADIAL DISTANCE FROM SITE (miles)	**ACCUMULATIVE POPULATION**
5	3,425
10	16,827
20	83,495
30	188,755
40	518,825
50	2,305,635

TABLE 2-4

ACCUMULATIVE POPULATION SUMMARY - 2010

RADIAL DISTANCE FROM SITE (miles)	ACCUMULATIVE POPULATION
5	11,253
10	59,750
20	187,470
30	379,830
40	1,040,750
50	4,757,810

TABLE 2-5

AGRICULTURAL LAND USE - 1959 AND 1964
(Calvert County)

CROP	PERCENT OF HARVESTED CROPLAND 1959	1964
Tobacco	42	44
Corn	27	27
Hay	16	15
Small Grains	8	8
Other	7	6

TABLE 2-6

AGRICULTURAL LAND USE – 1959 AND 1964
(Calvert, Charles, Dorchester, and St. Mary's Counties)

	PERCENT OF HARVESTED CROPLAND	
CROP	**1959**	**1964**
Corn	28	30
Soybeans	25	22
Small Grains	17	17
Tobacco	12	12
Hay	10	8
Other	8	10

2.3 METEOROLOGY

2.3.1 INTRODUCTION

This section summarizes the meteorological studies that have been conducted since the start of the engineering and design of the CCNPP. The meteorological studies performed include work in the following main categories, listed in chronological order:

a. Preliminary Data Collection
b. Initial Site Weather Data Program
c. Special Vertical Wind Standard Deviation Tests
d. Land-Sea Wind Speed Investigation
e. Extended Onsite Penetration Wind Study
f. Calculation of Incident and Routine Long-Term Relative Concentrations

2.3.2 PRELIMINARY DATA COLLECTION

Proximal long-term weather station data were used from the Patuxent NATC - PAX (now Patuxent Naval Air Station - NHK) for periods of record from 1955-1960, and 1949-1964. In addition, meteorological data from Washington National Airport (DCA); Byrd Field, Richmond, VA, (RIC); and Annapolis, MD, (ANA) were used to evaluate the frequency of various weather parameters and certain meteorological extremes, respectively. See Regional Map, Figure 2.2-1. Also, statistical data for severe weather parameters were obtained from numerous official records issued by the Environmental Science Services Administration (ESSA), Department of Commerce, Asheville, NC.

The following weather information from the above sources was evaluated and related to the Calvert Cliffs Nuclear Plant Site: Tornadoes, Freezing Precipitation, Tropical Storms, Hurricanes, and Diffusion Conditions.

2.3.2.1 Tornadoes

Five tornadoes were observed during the period 1953-1962 in the general vicinity of a single latitude-longitude square near the proposed plant site. The mean annual frequency was 0.5 tornadoes per year and the probability of a tornado striking a single point within a single latitude-longitude square near Calvert Cliffs, using a method originally derived by H.C.S. Thom of ESSA, was calculated to be 3.75×10^{-4}. The recurrence frequency was calculated to be once about every 2,700 years.

2.3.2.2 Thunderstorms

Thunderstorm day statistics indicate that about 40 thunderstorms per year can be expected in the area. Fifteen years of records at Patuxent showed 814 observations of thunderstorm activities. From these data one can calculate the average duration of a thunderstorm to be 1.356 hours for a point. A study of 10 years of records for transmission subtransmission feeders was conducted. This study showed that transmission and subtransmission feeder losses were 4 minutes and 423 minutes, respectively, due to storms in a 10-year period. The subtransmission feeders covered an area of approximately 180 square miles.

2.3.2.3 Freezing Precipitation

The Patuxent NATC records (1949-1964) list 910 hours of snow and 265 hours of frozen or freezing precipitation, other than snow, for a total of 1175 hours (or 70,500 minutes) in 15 years. Interpolating for a 10-year span yields 47,000 minutes. The outages due to snow and/or freezing precipitation were 182 minutes and 122 minutes in 10 years, for transmission and subtransmission feeders, respectively. It is interesting to note that 9 of 12 outages occurred during

a single snowstorm in March 1958. Certain design changes were made as a result of this storm and it is unlikely that outages of this magnitude would again occur.

2.3.2.4 Tropical Storms and Hurricanes

Approximately one hurricane per year poses a threat to the area, and about one hurricane every 10 years produces a significant effect. Northeasters, or extratropical storms, also can influence the area in terms of flooding of low-lying land. The detrimental effects of northeasters are considerably less than those postulated for hurricanes in the site area.

2.3.2.5 Preliminary Diffusion Climatology

The frequency of various Pasquill classes of diffusion was initially assessed through the use of the familiar Pasquill-Turner method. The proximal Patuxent NATC data were used for a five-year period of record, which yielded the following results:

Pasquill Condition	Annual Percent Occurrence
A and B	2.6
C	10.4
D (day)	35.0
D (night)	28.2
E	11.8
F	8.0
G	4.0

Since it is possible to take advantage of offshore waterways in considering a site boundary, it was considered reasonable to limit discussion interest and calculations to onshore winds at the Calvert Cliffs site.

The onshore wind directions, by sector, are as follows:

a. North
b. North-Northeast
c. Northeast
d. East-Northeast
e. East
f. East-Southeast

The frequency of all winds from these directions was documented (over a five-year period) to be:

a.	Patuxent NATC	23.0%
b.	Washington National Airport	24.0%
c.	Byrd Field, Richmond, VA	30.0%
d.	Annapolis, MD	24.0%

Based on these total wind frequency samples, it was calculated that the frequency of inversion winds associated with onshore flow was as follows:

	Station	Pasquill "E"	Pasquill "F and G"
a.	Patuxent NATC	2.04%	1.35%
b.	Annapolis	1.97%	3.40%

In order to confirm the initial conclusions drawn above and to get a first approximation of typical Pasquill "F" conditions at the site, two additional station records were examined in detail. Five years of records from RIC and DCA were examined and a computer program was written to produce the frequency of winds equal to or less than X knots for the 0100 EST hour of the day when the cloud cover was equal to or less than .4 coverage. Values for wind speeds from 10 knots to calm were documented. The results indicated that the average Pasquill wind speed for RIC was 1.88 m/sec, while that for DCA was found to be 2.02 m/sec. The frequency of these conditions with onshore wind directions was found to be 2% at RIC and 3% at DCA. Since neither of these stations had as good exposure as would be anticipated for the Calvert Cliffs site due to the unrestricted fetch over the Chesapeake Bay, it was deemed conservative to select a wind speed of 1.5 m/sec as a typical onshore Pasquill "F" site condition.

In general, the site’s low-level winds under a temperature inversion drain toward the Chesapeake Bay. It would not be possible for a ground-released effluent to get to the minimum site boundary under these conditions, and highly improbable that the ground release could get to the other inland boundaries due to terrain slope and other effects.

Wind persistence maxima for all wind sectors based on five-year record summaries at Annapolis, MD were as follows:

1 Sector	3 Sectors	5 Sectors
48 hrs	140 hrs	220 hrs

Washington National, Byrd Field, and Patuxent showed less persistent winds

MAXIMUM WIND PERSISTENCE FOR ONSHORE WINDS IN A SINGLE SECTOR (5 YEARS OF RECORD)

Station	Pasquill Condition	Maximum Persistence
DCA	1-3 knots winds	6 hrs
RIC	1-3 knots winds	6 hrs
PAX	"E" and "F"	12 hrs
ANA	"E" and "F"	12 hrs
PAX	All Speeds	27 hrs
ANA	All Speeds	37 hrs

Onsite low-level diffusion measurements were made at two primarily coastal locations during the periods from September 14, 1967, through November 9, 1968 at site N1W; and November 9, 1967 and through November 9, 1968, at site S1W. See Figure 2.3-1, Figure 2.3-2, and Table 2-10 for station locations and the description of meteorological instrumentation, respectively. In addition, temperature gradient approximations were made using two inland ground-level thermograph stations at the site at locations approximately 120’ above mean sea level (MSL) and 40’ MSL. These data also extended from November 9, 1967 to November 9, 1968.

The two coastal sites were selected initially because they:

a. offered good to excellent exposure to onshore winds; and
b. offered the only initial long-term exposure to winds unmodified by terrain and extensive tree cover.

The results of these onsite data comparison evaluations indicated the following:

a. Frequency of inversions derived from
   1. onsite data 31%
   2. long-term data 24%
b. Air drainage was toward the Bay under inversion conditions.
c. Average wind speed during inversion conditions was 2.6 MPS.
d. Standard deviation of horizontal wind direction (σ_θ) during worst, single-season wind sector inversion conditions averaged 6.6°.
e. When wind speed decreased, σ_θ increased, in general.
f. For on/or along-shore winds, the average value of $\sigma_\theta \bar{\mu}$ was 0.209 rad meters/sec.
g. χ/Q values at the 0.5% level of all conditions was 1.17x10^{-4} sec/m^3 for the 1150 meter minimum site boundary.

2.3.3 SPECIAL VERTICAL WIND STANDARD DEVIATION TESTS

2.3.3.1 General

Two sets of special diffusion tests were conducted at the Calvert Cliffs site. In both cases, both horizontal and vertical standard deviations of the wind conditions were measured.

2.3.3.2 Test Set 1 (October 17 to November 1, 1968)

In order to simulate actual reactor site location data, a standard anemometer was set up on a 40' bluff at Camp Conoy - just south of the reactor site. The anemometer permitted recording of wind direction and wind horizontal direction and its deviation. In addition, an σ_e meter was installed to evaluate the vertical standard deviations during this period.

The wind sensors were about 10' above the cliff area and about 40' inland. Results of σ_e Test Set 1 were as follows:

	Onshore Inversion Wind σ_e	Offshore Inversion Wind σ_e	"Neutral" Winds σ_e
Cases	16	122	157
$\bar{\sigma}_e$	13°	8°	14.3°
Lowest σ_e	1°	1°	1°
Cases <5°	1	35	9

2.3.3.3 Test Set 2 (February 11 through 20, 1969)

A second set of readings was taken during this period at Station 2 (about 2000' from the coastline). The companion statistics for Test Set 2 were follows:

	Onshore Inversion Wind σ_e	Offshore Inversion Wind σ_e	"Neutral" Winds σ_e
Cases	36	28	104
$\bar{\sigma}_e$	10°	13°	6°
Lowest σ_e	6°	2°	4°
Cases <5°	0	5	1

These readings were also taken about 10 to 12' above the ground, but with an unobstructed trajectory from an onshore viewpoint.

2.3.3.4 Conclusions

The sigma e values measured during these two test series both indicated that

a. Onshore inversion winds tend to produce near-neutral (Pasquill "D") σ_e values.
b. Offshore inversion winds tend to produce lower standard deviations than onshore cases near the coast, but somewhat larger inland.
c. Only one case in the total showed σ_e values as low as Pasquill "F".

2.3.4 LAND-SEA WIND SPEED INVESTIGATION

There was some concern expressed that the wind speed for onshore flow at the Station 4 (S1W) site was not representative for inland locations because the anemometer was in an area that is subject to a "Venturi" effect when the wind direction is onshore. In order to explore this possibility, this study compared the wind speed and diffusion values at the Station 4 (S1W) site to those on a raft anchored about one mile offshore.

For approximately one month of data, the diffusion parameter ($\sigma_\theta\bar{\mu}$) (STUB) was compared at each site where simultaneous onshore flow occurred at the sites. The average wind speeds at the two sites were also compared. Table 2-11 gives the results of the 256 simultaneous onshore winds and compares them to the classical Pasquill inversion classification values. The data indicated the following:

a. Only 1 observation of 256 at S1W gave $\sigma_\theta\bar{\mu}$ value equivalent to Pasquill "F".
b. Wind speeds were generally lower at S1W than at the raft, but wind deviations were larger.
c. The only possible Venturi effect noted at S1W was when the wind was onshore and the speeds were 3 mph or less.

2.3.5 THE PENETRATION ONSHORE WIND STUDY

The primary purpose of this extended meteorological investigation at the Calvert Cliffs site was to further refine the atmospheric dispersion parameters obtained from the initial site weather data program for use in the calculation of the relative concentration, χ/Q, at the site boundary nearest the reactor. Of secondary importance was to examine any anomalous flow features detected at the site and discuss its relevance to site diffusion characteristics.

Three inland meteorological stations were set up along with Station 4 (S1W). All four stations became active January 10, 1969 at the Calvert Cliffs site. In addition, temperature gradient systems were installed at Stations 2 and 4. See Figure 2.3-1 and Table 2-12 for station locations and instrumentation. A computer program was developed to analyze the wind flow across the site using the simultaneous wind observations from the four stations as input. The wind speed at each of the four stations was conservatively read to the lowest whole mph.

Standard techniques for evaluation of short-term releases (Pasquill "F", wind direction invariant, $\bar{\mu}$ = 1 MPS), were compared with measured parameters to determine, within conservative limits, the proper values applicable to this specific location.

The low percent probability level of $\sigma_\theta\bar{\mu}$ values was considered over the area collectively. The procedure was to select only those hours when the wind at Station 1 (K) was blowing onshore and also where at least two of the stations a $\sigma_\theta\bar{\mu}$ value of ≤ 0.200 existed.

Results for the one year extended study showed inversion conditions for 35% of the total observations, neutral conditions for 47%, and lapse for 17%, with 1% of the observations

missing. The winds showed a definite tendency to drain offshore during inversions; for the onshore winds, nearly 18% were in the neutral category, 9% in the unstable, and less than 4% in the stable category. The cumulative frequency distribution by wind speed category of on/or along-shore inversion winds for the four stations is given in Table 2-13 in terms of the total observations.

2.3.6 CALCULATION OF INCIDENT AND ROUTINE LONG-TERM RELATIVE CONCENTRATIONS

Two types of relative concentration calculations are of interest at the Calvert Cliffs site. The first are the 0-2 hour, 2-24 hour, and 1-30 day values which are used to determine the resulting radiation exposure from all of the postulated incidents. The second type is that pertinent to routine gaseous releases at the site.

2.3.6.1 Calculation of the Zero to Two-Hour Relative Concentration

For the first two hours following a postulated "maximum hypothetical accident," the relative concentration is calculated by the Gifford wake model for a ground release:

$$\frac{\chi}{Q} = \frac{1}{\bar{\mu}\left(\pi \sigma_y \sigma_z + cA\right)}$$

where:

$\frac{\chi}{Q}$	=	relative concentration, seconds/m^3
$\bar{\mu}$	=	average wind speed, meters/sec
$\sigma_y\sigma_z$	=	standard deviations of the distributed material in the lateral and vertical directions, in meters
c	=	wake factor (dimensionless)
A	=	cross-sectional area of structure from which material is presumed to be released, square meters

From the data in Table 2-13 it was determined that 5% of the time the on/or along-shore winds at Station 1 had speeds of 3.2 MPS or less; the comparable speeds at the 5% level for Stations 2, 3, and 4 are 1.1 MPS, 1.7 MPS, and 2.1 MPS, respectively. The average of the four stations at the 5% level is 2.0 MPS. This shows that relatively strong flow is available for on/or along-shore inversion wind directions even at the 5% frequency level at the site.

The 0-2 hour relative concentration was evaluated at various frequency levels of the statistic STUB, the product of sigma theta and u-bar, using a very conservative technique. The technique was to select the average of the two lowest of the four simultaneous values of STUB observed for on/or along-shore winds, and to array these averages in the order of frequency of occurrence. Assuming that the wind speed was one meter per second, the corresponding values of were tabulated, and the corresponding values of σ_θ for a distance of 1150 meters (the distance to the nearest site boundary) were selected. A wake factor of cA = 0.5x1640 M^2 = 820 M^2, and a σ_z value of 24 meters were used. The relative concentrations are shown in Table 2-14 for the 1% through 10% frequency levels.

The value of σ_z = 24 meters was selected as being compatible with the Pasquill "E" category for the 1% STUB level, using a wind speed of one meter per second. The previously referred to measurements of σ_e showed that a selection of Pasquill "E" for the vertical fluctuations was highly conservative.

A value for the 0-2 hour χ/Q of 1.3×10^{-4} sec/m^3 was selected for the radiation exposure calculations in Chapter 14 resulting from the containment wall release pathway. Meteorological conditions resulting in this value or higher for the 0-2 hour relative concentration will occur less than 5% of the time. For releases from the plant vent stack, main steam gooseneck, and refueling water tank vent, a 0-2 hour χ/Q of 1.44×10^{-4} sec/m^3 was calculated based on a zero cross-sectional area.

2.3.6.2 Calculation of the 2-24 Hour and 1-30 Day Average Relative Concentrations

Average relative concentrations for periods of 10 hours, 12 hours, and 29 days were calculated utilizing the onsite data acquired at Calvert Cliffs. No credit was taken for the wake factor of the plant structure and a minimum site boundary of 1150 meters was assumed in all 16 sectors.

The meteorological station with the lowest STUB value, Station 4 (S1W), was selected for this study. No Pasquill class with more diffusion than Pasquill "C" (slightly unstable) was considered and a ground-release accident model was assumed. As was done with the 0-2 hour χ/Q, the 2-24 hour and the 1-30 day values were also selected at the 5% frequency level. The resulting values were as follows:

Time Period	5% Probability Level χ/Q at 1150 meters (sec/m^3)
2-24 hrs	9.10×10^{-6}
1-30 days	2.70×10^{-6}

The 5% values are shown as a function of distance on Figure 2.3-3 for all of the incident-related time periods.

2.3.6.3 Calculation of Routine Long-Term Concentrations

The average annual relative concentrations, χ/Q which are applicable to routine venting or other routine operational gaseous effluent releases, have been determined for the final annual data record in accordance with the following equations:

$$\frac{\chi}{Q(i,D)} = \frac{\sqrt{2}}{\sqrt{\pi}} \sum_{p=A}^{G} \frac{R(p)}{\sigma_z(p,D)} \frac{1}{DB}$$

Where Rρ =

$$\sum_{k=1}^{K} \frac{0.01 f(k)}{\mu(k)}$$

$\frac{\chi}{Q(i,D)}$ = relative concentration (sec meter^{-3}); at a distance D (meters) from the effluent source; in direction sector i

p = Pasquill class (A through G)

f(k) = percent frequency wind blows toward sector i, within speed interval k, during Pasquill class condition p

$\mu(k)$ = Wind speed value representative of speed class interval, k, MPS

$\sigma_z(p,D)$ = vertical dispersion coefficient, meters, for Pasquill class p, at distance D

B = spread of wind sector, radians = $\pi/8$, for 22-1/2° sectors.

These equations and resultant calculations are appropriate for evaluating ground releases over longer time intervals. They do not include a wake factor term.

The Isopleths of the average annual concentration, shown in Figure 2.3-4 were calculated using the wind data and ΔT Pasquill class data of the final annual record. The maximum average on-shore relative concentration is 2.2×10^{-6} seconds meter^{-3} in the southeast sector at a distance of 1300 meters, which occurs as a result of the northwest winds and associated stability conditions. The site boundary in this direction is 2100 meters (Figure 1-1).

2.3.6.4 Average Annual Concentration at the Milk Samples Location

Milk samples were obtained from a location 4.2 miles southwest of the reactor site, during the period December 23, 1971 through June 5, 1976. Since this time, no samples have been available in the area.

The model used in the above section has been applied to this location. The average annual χ/Q is 7.0×10^{-8} and occurs with a northeast wind.

2.3.6.5 Continuing Studies

Additional studies were made to further refine the diffusion parameters. Included in these studies was an analysis of the data obtained at Station 2 (IS) between November 12, 1971 and November 11, 1972. This analysis showed that the diffusion characteristics specified in Section 2.3.6 are conservative.

Comparisons were made of ΔT data from the 12’ to 48’ system installed on the pole at Station 2 (IS) and the ΔT data from the "Sky Needle" 30’ to 98’ system also located at Station 2. The 12’ to 48’ system was continued in operation until September 1974.

A comparison study was made during the summer of 1974 to determine the correlation between meteorological data obtained from the "Sky Needle" 30’ to 98’ system at Station 2 and data obtained from the microwave tower system. The results of this study have been evaluated, and the remaining meteorological systems at Stations 2 and 4 were discontinued. The "Sky Needle" system was taken out-of-service August 14, 1975.

2.3.7 METEOROLOGICAL MEASUREMENT SYSTEMS

In accordance with the requirements of NUREG-0654 and Generic Letter 82-33 (Supplement 1 to NUREG-0737), a meteorological tower (Figure 1-1) was installed to provide the essential parameters used in support of dose assessment calculations for emergency preparedness. The meteorological tower and instrumentation design meets the intent of Safety Guide 23, February 1972, and Regulatory Guide 1.97, Revision 3, for primary meteorological measurements systems.

The instrumentation on the meteorological tower is described in Table 2-12. Signals from the wind and temperature sensors are transmitted to the plant Control Room where ΔT, W_s, W_d, σ_θ, and rain water level can be continuously monitored by the operator. The meteorological tower, located at the end of Road B-1, has been operational since April 1982. Subsequently, the Technical Specifications were amended to designate the new meteorological system as the "primary" meteorological system as addressed in Regulatory Guide 1.23, Revision 1, and the old microwave tower became a backup system. The meteorological instrumentation on the old microwave tower was taken out-of-service in the fall of 1993. The current primary and backup meteorological

measurement systems are described in the Emergency Response Plan and its implementing procedures.

2.3.8 INVESTIGATION OF RELATIVE CONCENTRATION FREQUENCIES USING THERMAL STABILITY PARAMETERS

During the investigation of the meteorological conditions at Calvert Cliffs, the almost universal acceptance of sigma theta to define diffusion qualities was questioned. This was in part due to the uncertainties of the sigma theta measurements in defining vertical plume growth. Also with winds at 2 to 3 mph or less, the measurement of sigma theta becomes difficult. Yet, in evaluations of the accident hazards, the periods of low wind speeds are the most critical. For these reasons the need for a Calvert Cliffs diffusion climate evaluation which does not depend upon sigma theta measurements was assessed.

2.3.8.1 The Requirement For Additional Meteorological Evaluation at Calvert Cliffs

The Calvert Cliffs site analyses in Section 2.3.6 use sigma theta measurements to define horizontal plume growth only. The uncertainties of the relationships between these measurements and vertical plume growth do not, therefore, cloud the validity of these analyses. Further, the 5% worst weather conditions of most concern for the accident evaluations are those with on-shore winds at low speeds. With on-shore directions conservatively defined to include nine 22-1/2° sectors, NW through SE clockwise, at Station 2 (IS) at 12’ above grade, on-shore winds at 3 mph or less occur 12% of the time. This 12% frequency includes the unstable and neutral as well as the stable (winds have subsequently been measured at 33’ above grade. At this elevation, on-shore winds of 3 mph or less occur less than 5% of the time.) It is unlikely, therefore, that the analysis based on sigma theta measurements are significantly biased by difficulties of measuring sigma theta at low wind speeds. Nevertheless, to remove the residual uncertainties in 1969 Baltimore Gas and Electric Company began to measure and record vertical temperature gradients near the ground for use in classifying site stability characteristics into inversion, neutral, and unstable conditions.

2.3.8.2 The Weather Data for the Independent Evaluation

The vertical temperature gradient (ΔT) was measured continuously between 12 and 50’ above grade at Station 2 from 1969 through September 14, 1974. Concurrently, an MRI 2040 wind instrument was installed at 33’ above grade (as opposed to the prior wind instrumentation at 12’ above grade) to measure wind speed and direction and values. Hourly averages of wind speed and direction, and one-an-hour 20 minute averages of were recorded. There were two sources of σ_θ data available with the MRI 2040 wind instrument, sigma meter readings and wind range measurements. The data were compared and wind range measurements, divided by six to obtain σ_θ, in accordance with the standard procedures, gave uniformly-lower values at the smaller σ_θ readings. The range measured σ_θ values were, therefore, used in this analysis because they provide more conservative estimates of the site diffusion quality.

Subsequent to the initiation of this program, it became an accepted practice to classify stability conditions into the standard Pasquill classes by the use of ΔT values in accordance with the following table of values.

PASQUILL CLASS	ΔT° C/100 meters
A	≤ -1.9
B	-1.9 to -1.7
C	-1.7 to -1.5
D	-1.5 to -0.5
E	-0.5 to +1.5
F	+1.5 to +4.0
G	≤ +4.0

To take advantage of this accepted practice, the validity, for Pasquill classification purposes, of the 12 to 50' ΔT data has been investigated by a comparison with concurrently observed 12 to 97' data, as shown in Figure 2.3-5 and Table 2-15.

For the shallower layer, 2 to 3% more of the observations fell in the critical Pasquill E, F and G classes, and 2 to 3% less in classes B, C, and D.

Because the atmospheric layer upward from 30' above grade was becoming the standard layer for determination of thermal stability Pasquill classes, the validity of the 12 to 50' layer data was further investigated by a comparison with concurrently observed 30 to 97' ΔTs at Station 2, as shown in Figure 2.3-6. On this figure, the dashed line is the line showing equal lapse rates for both layers. It is apparent that the assignment of Pasquill classes using the 12 to 50' layer ΔTs is very conservative in comparison with the use of the standard layer based at 30' above grade.

Data observed at Station 2, from November 1969 through October 1970, were selected for the Primary Year of Record. There were gaps in this record caused by equipment malfunctions. To complete the record and eliminate a potential seasonal bias, 1971 data were added to it, thereby creating the Final Annual Record. These added data are limited to dates and hours of the day which coincide with the data gaps in the Primary Year of Record.

2.3.8.3 The Zero to Two-Hour Relative Concentration Determined by ΔT and σ_θ Parameters

Relative concentrations for each hour of the final annual record have been calculated using the equations in Section 2.3.6.1 but with the uncertainties associated with σ_θ measurements eliminated. σ_y and σ_z values were fixed by the Pasquill classes, as before. However, two sets of Pasquill Classes were defined; one set based upon the ΔT measurements using the table in Section 2.3.8.2 and the other set using the measurements as recommended in Meteorology and Atomic Energy. All vertical dilution factors (σ_z), plus those horizontal dilution factors (σ_y) associated with wind speeds at 3 mph or less, were determined by the ΔT Pasquill classes. The horizontal dilution factors (σ_y) associated with winds greater than 3 mph were determined by the σ_θ Pasquill Classes. On-shore and along shore wind directions were conservatively selected to include the nine sectors NW through SE, clockwise. The hourly relative concentration values occurring with these wind direction were ranked and placed in a cumulative frequency distribution in accordance with the accepted practice at coastal sites for evaluation of accident conditions as shown in Figure 2.3-7. The relative concentration which is exceeded only 5% of the time during the year is $1.3\text{x}10^{-4}$

seconds per cubic meter. In view of the very conservative nature of the 12 to 50' ΔT data, as evidenced in Figure 2.3-6, and of the conservatism of the σ_θ data as evidenced by a comparison with the concurrently observed sigma meter readings, this 5 percentile relative concentration value is conservative indeed.

It is concluded that, considering both the ΔT and σ_θ data observed at the Calvert Cliffs site, a relative concentration of 1.3×10^{-4} seconds per cubic meter is a very conservative value, and is appropriate for the 0-2 hour accident evaluations. This relative concentration is equivalent to a meteorological condition which may be defined as Pasquill E and a wind at 1.4 MPS.

Details of the concurrent values of wind speed and direction and ΔT and σ_θ Pasquill Classes for the Final Annual Record are presented in Figure 2.3-10, Sheets 1 through 14. The same data are presented in Figure 2.3-10, Sheets 13 through 28 except that the 1971 data observed after the Primary Year of Record have been omitted.

2.3.8.4 A Critique of the Data Record for the Independent Evaluation

A calendar of data availability is presented in Table 2-30. It can be seen from this table that the Primary Year of Record, November 1969 through October 1970, provides the most complete 12-consecutive-month data record during the November 1969 through October 1971 period.

To fill in the data gaps which might be the cause of a seasonal bias in the Primary Year of Record, 1971 data coincident with the dates and times-of-day of the data gaps have been added in this Final Annual Record used in the analysis. A calendar of data in this Final Annual Record is presented in Figure 2.3-8, which shows the dates for which no data is available from November 1969 through October 1971. The sequence of overall data availability is presented in Figures 2.3-8 and 2.3-9.

The Final Annual Record data is quite complete with less than 10% missing observations; 8% occurring in consecutive-day lots, and 2% in periods of less than a day duration. The consecutive-day lots range from four to six days duration, occurring in January, April, August, and November. Because of the distribution of this missing data, it is very unlikely that it has contributed a seasonal or diurnal bias to the data record.

The data added to the Primary Year of Record to fill in its gaps, thereby producing the Final Annual Record, were added to ensure that a potential seasonal bias in the data record was eliminated. Data were added only to replace data lost because of equipment malfunctions, and they were only added to the extent that 1971 data, coincident with the dates and times-of-the-day of the equipment malfunction, were available. Although added data constitute 20% of the Final Annual Record, they could not create a bias in the record.

2.3.9 RECENT DATA COLLECTION

The following sections summarize the meteorological studies that were conducted to obtain information for use in the design of the Diesel Generator Building for Diesel Generator 1A.

2.3.9.1 Strong Winds

As illustrated in Reference 2, the average velocity for CCNPP's "fastest mile" of wind with a mean return period of 100 years is 100 mph. Reference 2 used

records of the fastest mile as published by the United States Weather Bureau from data obtained at airport stations.

2.3.9.2 Snow Storms

Monthly snowfall depth data from the weather stations at Baltimore, Maryland (1958 to 1989) and the Patuxent River Naval Air Station, Lexington Park, Maryland (1976 to 1992) were used to estimate the 100-year ground snow pack level at the CCNPP site. Frequency analyses were performed on the monthly snowfall records for the months of December, January, February, March, and the combined snowfall total for the months of January and February. The snowfall total for the combined months of January and February, 59", was chosen to represent the 100-year snow pack on the ground.

2.3.10 REFERENCES

1. Meteorology and Atomic Energy 1968, USAEC Division of Technical Information
2. S. C. Hullister, The Engineering Interpretation of Weather Bureau Records for Wind Loading on Structures, Cornell University, Ithaca, NY

TABLE 2-10

FIRST YEAR ONSITE METEOROLOGICAL STATIONS AND INSTRUMENTATION CALVERT CLIFFS NUCLEAR POWER PLANT

DESIGNATION	LOCATION	ELEVATION	PERIOD	INSTRUMENTATION
Station 1[(a)] "N1W" North	N11,916 E10,403	100’ MSL +10’ Mast	09/14/67-11/11/68	Packard Bell (Beckman-Whitley, Inc.) Model K-100 with Quick-D Vane Wind System
			09/14/67-11/15/67	Cassella Thermograph
			12/14/67-11/11/68	Standard US Weather Bureau Rain and Snow Gauge
Station 4 "S1W" Conoy South	N8,400 E1,060,000	90’ MSL +50’ Mast	11/09/67 Use Discontinued, Date Unknown	Packard Bell Electronics Corporation (Beckman-Whitley, Inc.) Model K-101 with Quick-D Vane Wind System
Station UT1 Upper	N10,000 E8,162	120’ MSL +4’ Shelter	11/15/67 to 12/31/68	Cassella Thermograph installed in standard US Weather Bureau Cotton-Region type shelter
Station LT1 Lower	N8,642 E9,590	40’ MSL +4’ Shelter	11/15/67 to 12/31/68	Same as Station UT1
Test Site Camp Conoy	N7,600 E1,055,000 N7,625 E1,000,000	40’ MSL +12’ Masts 60’ MSL	10/17/68 11/01/68	Meteorology Research, Inc. (MRI) Mechanical Weather Station Model 1072 with rain gauge; (2) MRI vector vane Sigma Meter Model 1053 and (3) MRI Mechanical Weather Station Model 1071

(a) Temporary Location

TABLE 2-11

RESULTS OF 256 SIMULTANEOUS ONSHORE WINDS IN RAFT STUDY AS COMPARED TO CLASSICAL PASQUILL INVERSION CLASS VALUES

SITE PASQUILL CLASS	AVG. $\sigma_\theta \bar{\mu}$ (rad m/sec)	Ave. $\bar{\mu}$ (m/sec)	AVERAGE (degrees)
Raft	0.434	4.23 (9.5 mph)	7.2
Station 4 (S1W)	0.492	3.41 (7.6 mph)	8.3
Classical "F"	0.044	1.00 (2.2 mph)	2.5
Classical "E"	0.175	2.00 (4.5 mph)	5.0

TABLE 2-12

ONSITE METEOROLOGICAL SYSTEMS AND INSTRUMENTATION CALVERT CLIFFS NUCLEAR POWER PLANT

DESIGNATION	LOCATION	ELEVATION	PERIOD	INSTRUMENTATION
Microwave Tower	N9,770 E8,809	75' MSL +40' & 125' & 220'	8/8/73 - Fall 1993	125' & 200' MRI 2040 Wind Diffusion System
			8/8/73 - Fall 1993	40', 125' & 200' Weathermeasure Corporation Aspirated Radiation Shields with Rosemount Sensors (Temperature Gradient System)
			8/23/73 - Fall 1993	125' Weathermeasure Corporation Dewpoint System
Meteorological Towers				
a. Primary Tower	N10,560 E7,710	110' MSL +33' & 197'	1982 - Current	197' & 33' Wind Sensors
			1982 - Current	197' & 33' Temperature Sensors
			1982 - Fall 1995	33' Dewpoint Sensor
b. Backup Tower	N10,422 E7,709	110' MSL +33'	1982 - Current	0' Rain Gauge
			2005 - Current	33' Wind Sensor
			2005 - Current	33' Temperature Sensor
Station 1 "K" Knoll	N10,895 E10,435	48' MSL +12' Mast	1/3/69 to 11/4/70	Meteorology Research, Inc. (MRI) Mechanical Weather Station, Model 1072 Wind System with Precipitation Gauge
Station 2 "IS" Inner South	N9,530 E8,720	48' MSL +12' Mast	1/9/69 to 1/12/70	MRI Mechanical Weather Station, Model 1071
			2/11/69 to 2/20/69	MRI Vector Vane Sigma Meter Model 1053
		48' MSL +12' & 49.5'	5/15/69 to 9/4/74	Temperature Gradient System, Packard Bell Corp. (Beckman-Whitley) Model 327 Aspirated Radiation Shields
		48' MSL +12' Mast	6/1/69 to 8/7/69	MRI 2040 Wind Diffusion System
		48' MSL +33'	8/7/69 to 5/15/71	MRI 2040 Wind Diffusion System

TABLE 2-12

ONSITE METEOROLOGICAL SYSTEMS AND INSTRUMENTATION CALVERT CLIFFS NUCLEAR POWER PLANT

DESIGNATION	LOCATION	ELEVATION	PERIOD	INSTRUMENTATION
		48’ MSL +33’ & 97’	5/15/71 - 8/14/75 9/29/71 - 8/14/75 7/17/71 - 8/14/75	MRI 2040 Wind Diffusion System Temperature Gradient System. Weathermeasure Corp. Aspirated Radiation Shields with Rosemount Sensors Beckman-Whitley Model WS-101 Quick Vane Wind System
			3/13/72 - 1/11/74	Gill Anemometer Bivane
Station 3 "BW" Boundary West	N12,375 E6,735	115’ MSL +10’ Mast	1/9/69 to 1/11/70	MRI Mechanical Weather Station Model 1071
Station 4 "S1W" Conoy South	N8,500 E10,550	90’ MSL +50’ Mast +12’ & 49’ Mast	1/10/69 to 5/13/75 This station was shut down for reworking during the summer of 1971. It was reactiviated 9/1/71.	Beckman-Whitley Model WS101 Wind System Packard Bell (Beckman-Whitley) Model 327 Aspirated Radiation Shields

TABLE 2-13

CUMULATIVE FREQUENCY DISTRIBUTION, PERCENT OF TOTAL OBSERVATIONS, FOR ON/ALONG-SHORE INVERSION WINDS

	STATION			
SPEED CLASS meters/sec	**1 (K)**	**2 (IS)**	**3 (BW)**	**4 (S1W)**
0.01 - 0.50	0.30%	2.52%	1.40%	0.55%
0.51 - 1.00	0.82	4.92	2.92	1.54
1.01 - 2.00	2.46	7.66	6.11	4.91
2.01 - 3.00	4.28	9.66	7.94	8.53
3.01 - 4.00	5.78	10.67	8.55	11.01
4.01 - 5.00	7.44	11.23	8.68	12.59
5.01 - 6.00	8.01	11.36	8.71	13.02
6.01 - 8.00	8.57	11.37	8.71	14.14
8.01 -10.00	8.87	11.37	8.71	14.17
10.01	8.98	11.37	8.71	14.17
	8291[a]	8386[a]	8399[a]	6743[a]

[a] Number of valid observations in each record.

TABLE 2-14

LOW-FREQUENCY χ/Q VALUES FOR ON-ALONG SHORE INVERSION WINDS AT CALVERT CLIFFS NUCLEAR STATION

% LEVEL OF OCCURRENCE	STUB (radian-M/sec)	σ_y (M)(1150M)	σ_z (M)(1150M)	χ/Q (0-2 hrs) (sec/m^3)
1	.097	59	24	1.89×10^{-4}
2	.130	66	24	1.72×10^{-4}
3	.158	74	24	1.56×10^{-4}
4	.185	83	24	1.41×10^{-4}
5	.208	92	24	1.29×10^{-4}
6	.228	103	24	1.17×10^{-4}
7	.243	110	24	1.09×10^{-4}
8	.258	116	24	1.045×10^{-4}
9	.273	124	24	9.85×10^{-5}
10	.287	133	24	9.20×10^{-5}

TABLE 2-15

THE FREQUENCY OF CONCURRENTLY OBSERVED ΔT VALUES FROM THE 50-12 FT AND 97-12 FT LEVELS ABOVE GRADE

97-12 ft	50-12 ft ΔT														
ΔT	≤0.5	-0.4	-0.3	-0.2	-0.1	0	+0.1	0.2	0.3	0.4	0.5	0.6	0.7	+0.8	≥0.9
≤0.9	924	52	41	61	46	68	20	39	20	10	7	9	3	1	15
-0.8	12	8	5	5	3	3	0	1	0	0	0	2	1	1	0
-0.7	11	4	6	4	3	0	1	0	1	0	1	0	1	0	1
-0.6	19	6	9	8	1	3	0	1	1	2	1	3	0	2	6
-0.5	16	12	9	9	2	16	2	0	0	0	0	0	2	2	2
-0.4	14	16	11	8	3	21	6	4	2	0	1	0	2	1	3
-0.3	6	12	11	13	5	34	8	3	2	0	0	0	0	1	1
-0.2	4	4	3	9	2	12	6	2	1	0	0	1	1	0	2
-0.1	4	2	5	9	4	9	2	3	1	1	0	0	0	0	2
0	4	3	4	1	3	9	3	4	2	0	0	0	0	0	4
+0.1	3	1	5	1	1	4	1	6	2	0	1	1	1	0	3
0.2	0	0	0	4	0	3	2	2	2	0	1	0	1	0	5
0.3	0	0	0	1	0	0	2	1	3	1	1	0	0	1	6
0.4	0	0	0	0	0	3	0	2	1	2	1	1	0	0	2
0.5	1	1	0	0	0	0	1	3	1	1	1	1	0	0	6
0.6	1	0	1	3	1	3	1	2	2	0	2	1	1	1	10
0.7	0	0	1	0	0	2	1	1	0	0	0	0	0	0	6
0.8	1	0	0	0	0	0	0	0	1	0	0	5	0	2	5
0.9	0	0	0	0	0	1	2	1	1	3	1	0	1	1	5
1.0	0	0	0	1	0	0	0	0	0	1	3	0	1	1	7
1.1	0	0	1	0	0	1	0	1	0	0	0	1	2	0	9
1.2	0	0	0	0	0	0	0	1	1	0	0	0	1	2	5
1.3	0	0	0	0	0	0	0	1	0	2	0	0	0	2	8
1.4	0	0	0	0	0	0	0	0	0	1	1	1	0	0	5
1.5	0	0	0	1	0	0	0	0	0	0	1	0	0	0	6
1.6	0	0	0	0	0	0	0	1	0	0	2	2	1	0	7
1.7	0	0	0	0	0	0	0	0	0	0	1	0	0	1	8
1.8	0	0	0	0	0	0	0	0	0	0	0	0	1	0	6
≥1.9	0	0	1	2	1	5	3	1	3	0	7	4	12	7	215

Observations were made between May 14, 1971 and September 29, 1971.
Tables 2-16 through 2-29 were deleted, see Figure 2.3-10.

TABLE 2-30

CALENDAR OF METEOROLOGICAL DATA AT STATION 2 FROM NOVEMBER 1969 THROUGH OCTOBER OF 1971

Number of Days with 12 or more hours of valid data.

PARAMETER	NOV 69	DEC 69	JAN 70	FEB 70	MAR 70	APR 70	MAY 70	JUN 70	JUL 70	AUG 70	SEP 70	OCT 70
ΔT	23	30	24	28	31	14	20	30	31	28	30	28
Wind Dir	27	31	31	28	31	30	31	26	31	27	8	0
Sigma Theta	27	31	31	28	31	30	31	26	31	27	8	0
Wind Speed	30	31	31	27	31	30	31	26	31	27	15	4
All	23	30	24	27	31	14	20	26	31	24	8	0
Running 12-Month Totals												
ΔT	--	--	--	--	--	--	--	--	--	--	--	317
Wind Dir	--	--	--	--	--	--	--	--	--	--	--	301
Sigma Theta	--	--	--	--	--	--	--	--	--	--	--	301
Wind Speed	--	--	--	--	--	--	--	--	--	--	--	314
All	--	--	--	--	--	--	--	--	--	--	--	258

PARAMETER	NOV 70	DEC 70	JAN 71	FEB 71	MAR 71	APR 71	MAY[(a)] 71	JUN 71	JUL 71	AUG 71	SEP 71	OCT 71
ΔT	30	27	30	28	30	30	31	20	30	24	30	29
Wind Dir	14	14	26	28	20	19	28	20	26	25	30	29
Sigma Theta	14	14	26	28	20	19	28	20	26	25	30	29
Wind Speed	21	31	22	26	20	19	28	28	29	24	27	29
All	8	12	16	26	19	19	28	17	25	16	27	29
Running 12-Month Totals												
ΔT	324	321	327	327	326	343	354	344	343	339	339	340
Wind Dir	288	271	266	266	255	244	241	235	230	228	250	279
Sigma Theta	288	271	266	266	255	244	241	235	230	228	250	279
Wind Speed	305	305	296	295	284	273	270	272	270	267	279	304
All	243	225	217	216	204	209	217	208	202	194	213	242

(a) Much of the wind data from May through September 1971 was observed at 100’ above grade.

2.4 GEOLOGY

2.4.1 INTRODUCTION AND SUMMARY

This section of the report presents the results of the geologic phase of the environmental study. This phase of the study included a geologic investigation of the site and surrounding area, a review of pertinent geologic literature (References 1 through 23), and interviews with personnel from government agencies and private organizations (References 24 through 33). Subsurface geologic conditions within the site were investigated in detail by exploratory borings.

The site is underlain by approximately 2,500’ of southeasterly dipping sedimentary strata of Cretaceous and Tertiary age. Underlying these sediments are crystalline and metamorphic rocks of Precambrian and Early Paleozoic age.

Sediments of the Chesapeake Group of Miocene age underlie the proposed plant area to a depth of about 200’. The material in this group consists of essentially horizontally-stratified sandy and clayey silt with occasional interbeds of sand and shells. It is relatively impervious and dense and provides adequate foundation support for the nuclear power plant. The Miocene sediments are underlain by dense, relatively pervious glauconitic sand and silt of Eocene age.

No known or suspected faults are present in the sedimentary strata underlying the site. The closest known faults are located in the Piedmont Province in Western Maryland, approximately 50 miles from the site.

The site is considered satisfactory, from a geologic standpoint, for construction and operation of a nuclear power plant.

2.4.2 REGIONAL GEOLOGY

2.4.2.1 Physiography

The site lies within the Coastal Plain Physiographic Province about 50 miles east of the Fall Zone. The Fall Zone separates the low-lying gently rolling terrain of the Coastal Plain from the higher relief of the Piedmont Physiographic Province. The provinces are shown on Figure 2.4-1, Regional Physiographic Map.

The Coastal Plain in Maryland is a low plain rising from sea level to about Elevation +250’ at the Fall Zone. Relief in the region ranges generally from about 20 to 100’. The regional slope of the Coastal Plain is to the east at approximately 1.5 ft/mile. The topography of the region is characterized by a series of broad, step-like terraces. The terraces are successively less dissected by stream erosion from west to east. The region is well drained by a large number of small streams.

2.4.2.2 Stratigraphy

The general geologic characteristics of the region are shown on Figure 2.4-2, Regional Geologic Map. The Piedmont Province consists of a complex of igneous and metamorphic rocks of Precambrian and Early Paleozoic age with areas of sedimentary and igneous rocks of Triassic age. Beneath the Coastal Plain Province these rocks are concealed by younger strata of Cretaceous and Tertiary age. The buried surface of the basement igneous and metamorphic rocks slopes to the southeast at about 50 ft/mile. In the vicinity of the site, the surface of the basement complex is located approximately 2,500’ below sea level. The Cretaceous and Tertiary strata consist of sedimentary deposits of silt, clay, sand, and gravel which exhibit considerable lateral and vertical variations in lithology and

texture. The strata form a wedge-shaped mass which thickens to the southeast and pinches out to the northwest toward the Fall Zone.

A generalized geologic cross-section of the Coastal Plain is presented on Figure 2.4-3, Regional Geologic Section. A detailed description of the stratigraphy at the site is presented on Figure 2.4-4, Geologic Columnar Section - Site Area.

2.4.2.3 Structure

The thick sedimentary strata of the Coastal Plain in the vicinity of the site have remained essentially undeformed since they were deposited up to 135 million years ago. They are believed to have been affected only by slow regional crustal downwarping during their deposition. No known faults have been identified within the Cretaceous and Tertiary sedimentary deposits in the site area. Some local, very shallow folds have been recognized in the Coastal Plain sediments about 40 miles south of the site. These structures are possibly related to depositional conditions rather than to post-depositional tectonic activity. The strata exposed for many miles along the Chesapeake Bay shoreline show no visible signs of faulting or deformation.

There is no known fault or geologic evidence of faulting in the deep crystalline rocks in the area. The absence of deformation in the overlying sediments indicates that no major faults are present in the area. Significant tectonic features of the region are shown on Figure 2.4-5, Regional Tectonic Map.

The closest known faults to the site are more than 50 miles to the west in the Precambrian and Early Paleozoic rocks of the Piedmont Physiographic Province. The rocks in the Piedmont are highly folded, and many zones of major faulting have been identified. Most earthquake activity in the region can be related to them. Some of these faults, the closest of which are located about 60 miles southwest of the site, theoretically could be projected beneath the Coastal Plan strata toward the general location of the site. However, such faults are local rather than regionally continuous and appear to be associated with individual fault troughs containing Triassic sediments. Concealed local faults of this type in the basement rock may be responsible for part of the minor earthquake activity in the Coastal Plain of Maryland.

2.4.2.4 Geologic History

The recognizable geologic history of the region begins with the deposition of Paleozoic sediments on a Precambrian granitic and metamorphic basement complex. Thick sequences of sedimentary rocks, which accumulated during the Cambrian and Ordovician Periods of geosynclinal deposition were subsequently uplifted, folded, faulted, and metamorphosed during the late Paleozoic Period of mountain building. This activity was followed by another period of uplift along the axis of the Appalachian Mountain chain at the end of the Triassic Period.

Slow regional downwarping of the Coastal Plain started during Early Cretaceous time and continued intermittently through Tertiary time. South and east of the Fall Zone the Piedmont was depressed below sea level providing a base on which the sediments were deposited. Several periods of submergence and emergence resulted in alternate deposition and erosion of continental and marine deposits throughout Cretaceous and Tertiary times.

Near the end of the Tertiary Period (Pliocene time) the area is believed to have been above sea level. This resulted in erosion of the sediments deposited

previously during Early Pliocene and Late Miocene time, so that Miocene sediments are presently exposed in the site area.

During Early Pleistocene time, the ocean advanced westward to the Fall Zone, completely covering the Coastal Plain. Fluctuating sea levels, occurring during Pleistocene time, resulted in alternating periods of erosion and deposition along what are now the major terraces and scarps of the region. A veneer of Pleistocene soils covers most of Coastal Plain. At present, the land is again being submerged by a very slow rise of the sea level.

2.4.3 SITE GEOLOGY

2.4.3.1 General

The site is located on the west shore of the Chesapeake Bay in an area characterized by densely wooded, low, flat to gently rolling terrain of low to moderate relief. Ground surface elevations at the site range from sea level to about +130', with an average Elevation of approximately +100'. Nearly vertical cliffs, over 100' high in places, are located along the shore of the Chesapeake Bay. The plant is located in an area near the east edge of the site where the preexisting ground Elevation was about +65'. The final grade Elevation is about +45'.

2.4.3.2 Surficial Deposits

The upland areas of the site (areas above Elevation +70') are underlain by sediments of Pleistocene age. These sediments consist primarily of silt and sand, and as encountered at the boring locations, range up to about 50' in thickness. The portion of the site below Elevation +70', which includes the plant area, is underlain by relatively impervious sediments of the Chesapeake Group of Miocene age. The contact between the Pleistocene and Miocene sediments is relatively even and slopes very gently toward the southeast. The surficial geology of the site is shown on Figure 2.4-6, Site Geologic Map.

2.4.3.3 Subsurface Deposits

The details of the subsurface geology were investigated primarily by means of ten exploratory borings at the locations shown on Figure 2.4-7, Plot Plan.

The borings ranged in depth from 146' to 332' and were drilled with truck-mounted rotary drilling equipment. Data were obtained from the borings through continuous observation of drill cuttings and examination of undisturbed samples collected by Dames & Moore geologists and engineers.

The soil samples were obtained at intervals in each boring ranging between 3-1/2 and 15', utilizing the Dames & Moore soil sampler illustrated on Figure 2.4-8, Soil Sampler Type U. A few samples were obtained using a standard split-spoon sampler. The number of hammer blows required to drive the sampler a distance of 1' into undisturbed material is recorded in the column entitled "blow count" on the left side of each boring log. The energy used to advance the samplers was greater than that in a standard penetration test, resulting in generally lower blow counts.

All samples were examined and logged in the field and then shipped to Dames & Moore's New York office for further examination and appropriate laboratory testing. Detailed descriptions of the materials encountered in the borings are shown on Figures 2.4-9A through 2.4-9J, Logs of Borings. The type of sampler used and data relative to the energy used to advance the sampler are presented on the logs

of borings. The depth of ground water after completion of drilling operations and the date on which the borings were completed also are presented on the logs.

The site is underlain by a relatively simple sequence of strata, which is shown on Figures 2.4-10A, 2.4-10B, and 2.4-10C, Geologic Sections A-A, B-B, and C-C, respectively. Details of the strata exposed along the shore of the Chesapeake Bay are illustrated on Figure 2.4-11, Schematic Cliff Section, Plant Area.

The Chesapeake Group is approximately 270' in thickness and occurs between Elevation +70' and Elevation -200'. It is composed primarily of gray and green, fine sandy and clayey silt which is relatively impervious. Occasional interbeds of sand and small shells are present, particularly in the upper portion of the group. The upper 15 to 30' of the Chesapeake Group, where exposed in the plant area, have been highly oxidized by weathering.

The Chesapeake Group in the region has been divided from top to bottom into the St. Mary's formation, the Choptank formation, and the Calvert formation. For purposes of this study, these formations are essentially identical.

Eocene deposits consisting of about 350' of dense, relatively pervious, green, interbedded glauconitic sands and silts with some clays are present below Elevation -200'. The uppermost Eocene deposit, the Piney Point formation, is approximately 40' thick and is composed primarily of glauconitic sand. Because this formation is continuous and distinctive, it provided an excellent horizon for correlation stratigraphy at the site. The contact between the Chesapeake Group and the Piney Point formation occurs at about Elevation -200' and is essentially horizontal throughout the site.

The deeper sediments underlying the Piney Point formation (below an Elevation of about -240') were not investigated, but they have been identified in nearby water wells. The names and descriptions of these formations are shown on Figure 2.4-4.

No evidence of faulting was observed at the site in surface outcrops, in the borings, or in the results of the geophysical surveys. A good correlation of subsurface stratigraphy was obtained between the borings. The strata exposed along several miles of the western Chesapeake Bay shoreline in the vicinity of the site show no visible deformation. A view of the slightly dipping strata is shown on Figure 2.4-12, Cliff Face Photograph - Plant Site Vicinity.

A poorly developed crack pattern, believed to be related to desiccation, is exposed in outcrops of the Chesapeake Group strata. These cracks are noticeable in places along the cliffs facing the Chesapeake Bay in material where the effects of weathering are pronounced.

2.4.4 SHORE EROSION

The cliffs bordering the Chesapeake Bay along the east side of the site have receded due to shoreline erosion at a maximum rate of about 2' per year. This rate of erosion was calculated from records of measurements made along the shore from 1848 to 1945. The measurements were updated to 1967 by means of recent topographic maps and aerial photographs.

The data indicate that the shoreline at the site receded a maximum of 200' between 1848 and 1945. The average rate of recession along various sections of the eroded coast, including part of the site, has ranged up to 2.1' per year. Changes along the shoreline are shown on Figure 2.4-13, Shoreline Changes.

A field check on the rate of shore erosion was provided by an unidentified monument located near the southeast corner of the site. The inscription on this monument reads: "The bank was 55' from this line[1] in August 1936." On September 8, 1967, the bank was only 36' from the monument. Therefore, 19' of bank recession has occurred at this location since 1936, representing an average of 0.6' per year.

Shoreline recession along the site is due mainly to wave erosion, particularly storm waves, undercutting the cliff. This results in sloughing of the overlying material. Generally, only the surficial 1'-2' of the cliff face slough at any one time. In the proximity of localized jointing, up to 5' of the cliff face may slough at one time.

Records show that 645 acres of land along a 31.3 mile section of the shoreline in Calvert County were lost between 1848 and 1945 due to erosion. However, during the same interval of time, 115 acres were gained by redeposition.

Approximately 3700 lineal ft of shore protection has been placed in front of the plant area, as shown on Figure 1-3A. The shore protection consists of onsite material placed in front of the cliffs and faced with filter cloth and layered riprap.

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2.5 HYDROLOGY

2.5.1 INTRODUCTION AND SUMMARY

This section of the report presents the results of the surface and ground water hydrology phase of the environmental study performed by Dames & Moore. Research for this phase included a review of available pertinent hydrologic literature (References 1 through 5) and interviews with representatives of government agencies and other individuals possessing knowledge of the local area (References 6 through 10).

A study of the hydrologic features of the site and the surrounding area was conducted. This study included an inventory of water wells, an inspection of surface drainage features, a study of aquifers, measurement of ground water levels in exploratory borings, and an analysis of the depth, direction, and rate of ground water movement.

The site is well drained and not susceptible to flooding. Surface runoff is moderately high and accounts for about 35% of the total annual precipitation. Average annual precipitation in the region ranges from about 40.6" at the Patuxent NATC to about 44" at Prince Frederick. A drainage divide extends across the site in a general north-south direction. The area east of the divide (20% of the site) drains into the Chesapeake Bay, whereas the area to the west drains into local tributaries and eventually into the Patuxent River. The plant is located east of the divide where surface drainage is toward the Chesapeake Bay.

The plant area is underlain by over 200' of relatively impermeable deposits which effectively confines the underlying artesian aquifers and minimizes their possible contamination by the downward percolation of an accidentally discharged contaminated liquid. The vertical component of ground water movement through the Chesapeake Group is upward. This precludes the possibility of contamination of the aquifers due to downward percolation of a contaminant.

Most of the potable water used in the region is obtained from the artesian aquifers underlying the Chesapeake Group. The aquifers are composed of glauconitic sand and silt of the Piney Point, Nanjemoy, and Aquia formations. The piezometric surfaces of these water-bearing formations slope to the southeast at about 2' per mile. Based upon this hydraulic gradient and coefficients of permeability for these formations, the estimated average rate of natural ground water movement is less than 1" per day.

A limited quantity of potable water is obtained from shallow wells completed in the surficial Pleistocene deposits which overlie the Chesapeake Group throughout most of the area surrounding the site. The areas in which these materials are utilized as a source of water are up-gradient from the plant and cannot be affected by the accidental release of contaminated liquids at or below the ground surface in the plant area.

The possibility of adversely affecting the available ground water resources or existing wells in the area, by the construction and operation of a nuclear facility, is remote. The hydrologic characteristics of the site are favorable for the construction and operation of a nuclear power plant.

2.5.2 SURFACE HYDROLOGY

2.5.2.1 General

Calvert County is a peninsula bounded on the east by the Chesapeake Bay and on the west by the Patuxent River. The area is characterized by gently rolling terrain with a dendritic drainage pattern. A drainage divide extends longitudinally across the county. The county is well drained by a relatively large number of streams, although most are less than seven miles long. Many streams have moderately

steep valley walls, while others form estuaries to the Patuxent River. Swampy areas and tidal flats are common along the coastal areas.

Stream flow in Calvert County is measured at two gauging stations maintained by the U.S. Geological Survey. Their locations are shown on Figure 2.5-1, Map of Area Showing Surface Hydrology. The gauge on St. Leonard Creek is a continuous recording station, while the Hellen Creek station is a partial recording site. Average monthly discharges at the continuous recording station are presented in Table 2-31, Average Monthly Discharge at Gauging Station on St. Leonard Creek (1958-1964).

The average runoff measured at the St. Leonard gauge from 1958 to 1964 was 15-1/2"/year. The average annual precipitation for the same period as measured at Prince Frederick, about 10-1/2 miles north of the site, was about 44". Thus, runoff accounts for about 35% of the total precipitation. Evapotranspiration also accounts for a large portion of the annual precipitation. Relatively little precipitation percolates into the surficial materials to recharge the phreatic surface.

The reason for the high runoff and low infiltration probably can be attributed to the impermeable nature of the Miocene subsoils which retard downward percolation of water. The surficial soils are Pleistocene or Recent deposits which are relatively pervious. Rainfall absorbed by them is soon discharged as stream runoff or lost through evapotranspiration. Many lowland areas of the county are not mantled by Pleistocene deposits and the relatively impermeable Miocene deposits are exposed. Precipitation falling on these areas is discharged almost immediately as surface runoff.

2.5.2.2 Site Conditions

The topography at the site is gently rolling with steeper slopes along stream courses. Local relief ranges up to about 130'. The site is well drained by short, intermittent streams. A drainage divide, which is generally parallel to the coastline, extends across the site as shown on Figure 2.5-1. The area to the east of the divide comprises about 20% of the site and includes the plant area. This area drains into the Chesapeake Bay. The western area is drained by tributaries of Johns Creek and Woodland Branch, which flow into St. Leonard Creek and subsequently into the Patuxent River. Grading performed during construction has not substantially altered the present drainage system.

The average Elevation of the site is about 100' above mean sea level. The site occupies the head-water area of several small drainage basins, and is not subject to flooding. It is possible that high intensity rain storms may cause water to back up in some valleys due to local constrictions in the stream beds, but this would be a temporary situation. The plant area has an Elevation of about +45' and has a storm drain system to handle runoff. High water levels in the bay due to storm conditions are discussed in Section 2.8.3.

Site grading in the vicinity of the Diesel Generator Buildings provides a system of swales that direct overland flow of the probable maximum precipitation runoff without producing drainage or flooding problems for the buildings. The system of swales direct runoff to the Chesapeake Bay without any dependence on the site's storm drain system. The results of the runoff and backwater analyses indicate that during the probable maximum precipitation storm the swale system at the Diesel Generator Building site will convey the surface runoff with a maximum water level of 44.8' above sea level near the Diesel Generator Buildings. This water level is below the floor grade of the Diesel Generator Buildings, which is 45.5' above sea

level, and thus precludes the potential for flooding of the Diesel Generator Buildings during the probable maximum precipitation.

2.5.3 GROUND WATER HYDROLOGY

2.5.3.1 Regional Conditions

Ground water occurs in the surficial soils and is tapped by many shallow dug and driven wells. Ground water in deeper aquifers occurs under artesian conditions. These aquifers, the Piney Point, Nanjemoy, and Aquia formations, are separated from the surficial deposits by an aquiclude averaging about 270' in thickness. Recharge to these aquifers occurs in their outcrop areas about 15 to 30 miles west of the site. The geologic position of these aquifers relative to other formations in Calvert County is presented in Table 2-32, Geologic Units in Calvert County.

The hydrologic characteristics of the Piney Point, Nanjemoy, and Aquia formations are discussed in greater detail in the following subsections.

2.5.3.1.1 Piney Point Formation

The Piney Point formation consists of glauconitic sand interspersed with shell beds and a little clay. Well cuttings and particle-size analyses indicate that the aquifer is composed mainly of medium to fine sand. The formation occurs as a wedge-shaped geologic unit and is known only in Southern Maryland. It is about 30' thick in the vicinity of the site and increases in thickness to the southeast.

The Piney Point formation is widely utilized as a source of ground water in Calvert County and adjoining St. Mary's County. It is estimated that more than 500 domestic wells in these two counties tap the Piney Point and the underlying Nanjemoy formations. These two aquifers are hydrologically connected. The yields of domestic wells generally range from about 3 to 20 GPM. At the Patuxent NATC, located about 10 miles south of the site, large-capacity wells tap the Piney Point aquifer and the upper part of the Nanjemoy formation. These wells each yield as much as 190 GPM. The specific capacities of 25 selected wells in these formations range from 0.1 to 3.3, and average 1.2 GPM/ft of drawdown.

2.5.3.1.2 Nanjemoy Formation

The lower part of the Nanjemoy formation consists of an impermeable red clay known as the Marlboro Clay. The remainder of the formation consists chiefly of greensand, but contains some clayey greensand. A limited number of particle-size analyses indicate that the sand is predominantly medium- to fine-grained.

The Nanjemoy formation is an important aquifer in Calvert County where it is tapped by several hundred wells. Most wells are completed in the permeable water bearing sands occurring in the uppermost 80' of the formation and yield less than 10 GPM. The specific capacities of 11 wells in Calvert County tapping this aquifer range from 0.2 to 2.4, and average 0.8 GPM/ft of drawdown. On the basis of water level recovery measurements made during a pumping test, a coefficient of transmissibility of approximately 2,000 GPD/ft has been computed. The field coefficient of permeability was 66 GPD/ft^2. The results of a similar test in Prince George's County indicate coefficients of transmissibility ranging from 260 to 840 GPD/ft.

2.5.3.1.3 Aquia Formation

The Aquia formation is characterized by an abundance of glauconitic sand with some quartz sand and clay. The thickness of permeable sandy beds in the formation ranges up to slightly more than 40’ in parts of Calvert County. Particle-size analyses of nine samples at the Aquia formation show that the sand is medium- to fine-grained.

The most productive wells tapping this formation are at the Patuxent NATC. Yields of individual wells range from 125 to 350 GPM. The specific capacities of eight of these wells range from 0.8 to 4.2 and average 2.5 GPM/ft of drawdown. The results of six pumping tests indicate field coefficients of permeability ranging from 130 to 1,340 GPD/ft^2. Coefficients of transmissibility determined from the tests range from 5,500 to 33,000 GPD/ft^2.

Plant wells tap this formation.

2.5.3.1.4 Water Levels

The artesian head of the three principal aquifers in Calvert County is generally above sea level. The effect of tidal fluctuations on water levels is noticeable in two observation wells, completed in the Nanjemoy and Piney Point formations, at Solomons Island about 7 miles south of the site. Recorder charts from these wells, which are 248 and 493’ deep, respectively, show semi-diurnal fluctuations of about 1/2’.

The approximate configurations of the piezometric surfaces of the Aquia and Nanjemoy formations are shown on Figure 2.5-2, Piezometric Surfaces in Calvert County. The regional hydraulic gradient in the vicinity of the proposed plant site is to the southeast. However, local minor variations occur. The cone of depression at the southern end of Calvert County is the result of ground-water extraction at the Patuxent NATC. Records of observation wells maintained by the U.S. Geological Survey indicate that water levels in the area have remained essentially unchanged since 1963. If the future rate and amount of ground-water extraction in the area is not significantly changed, it is likely that the cone of depression will remain constant and the existing hydraulic gradient in the vicinity of the site will be maintained.

2.5.3.2 <u>Water Use</u>

2.5.3.2.1 General

Nearly all potable water used in Calvert County is from subsurface sources. Since little industry is located in this area, the major use of water is for domestic and agricultural purposes.

2.5.3.2.2 Public Water Supplies

In 1967, there were 12 towns in Calvert County with public water supplies. The output from these systems is relatively small, but increases substantially in the summer to accommodate the seasonal population increase. Data concerning the public water supplies are presented in Table 2-33, Public Supply Wells in Calvert County. The locations of these supplies are shown on Figure 2.5-3, Public Water Supplies in Calvert County.

2.5.3.2.3 Private Wells

Most domestic water supplies in Calvert County are obtained from private wells greater than 300' in depth. In some instances, other wells are less than 50' deep and are of limited capacity. The locations of the deep wells, in the vicinity of the site, are shown on Figure 2.5-4, Map of Area, Showing Known Water Wells. Information pertaining to these wells is presented in Table 2-34, Known Water Wells, Vicinity of Site.

Shallow dug or driven wells are not tabulated or shown on Figure 2.5-4. The shallow wells will not be affected by changes in the ground water regimen at the site since they are at a higher elevation than the proposed plant grade.

Wells numbered 2 & 10 are located on BG&E property and were previously owned by YMCA. One is in use supplying water to our recreational pool facility for authorized personnel including BGE employees and the Red Cross. The same aquifer supplies water to this well and three others located close to the reactors on the plant site. Output from the three onsite wells referred to in Section 2.5.3.3 is pumped to two storage tanks from the well water treatment building.

There is no comprehensive source of public information on dug wells in Calvert County. The Calvert County Health Department records and retains dug-well records for only five years. Permits are not required by the Maryland Department of Water Resources nor by the Maryland Department of Health. Several dug wells are listed in Reference 2 and Section 2.5.4.2; however, accurate public information is not available for most of them. It is believed that dug wells in the area are upgradient from the plant site or are across drainage area boundaries.

2.5.3.3 Site Conditions

The depth of ground water at the site was measured in piezometers installed in seven of the Dames & Moore exploratory borings. The piezometers consisted of small-diameter steel pipe equipped with a well point, or perforated PVC pipe. They were installed in borings DM-1, DM-2, DM-3, DM-5, DM-7, DM-8, and DM-9 immediately after completion of the drilling operations. The locations of the borings are shown on Figure 2.4-7, Plot Plan. The water level recorded in each piezometer is shown on the Log of Borings, Figures 2.4-9A through 2.4-9J in Section 2.4, Geology.

An in-situ soil percolation test was performed at the site in Miocene soils typical of those underlying the proposed plant. The test was conducted in a one-foot square hole in accordance with the procedure used for the Corps of Engineers Soil Absorption Test (Reference 5). Results of this test indicate a permeability of less than 1 GPD/ft^2. The location of the test is shown on Figure 2.4-7.

Representative samples extracted from the exploratory borings were subjected to a laboratory testing program in order to evaluate the permeability characteristics of the natural soils and the physical properties of the material for correlation purposes. The laboratory program included the following tests:

a. moisture and density determinations;
b. particle-size analyses;
c. permeability tests; and
d. cation exchange and X-ray diffraction analyses.

The moisture and density determinations were performed on undisturbed samples for correlation purposes. The results of these determinations are shown on the Log of Borings in Section 2.4, Geology.

Selected soil samples were tested in order to measure their grain-size distribution. The results of these analyses were used in evaluating soil permeability and for classification purposes. The results are presented on Figures 2.5-5A and 2.5-5B, Particle Size Analyses.

Two permeability tests were performed on materials typical of those underlying the plant. The results of these tests, which were performed in accordance with the American Society for Testing and Materials (ASTM) procedures, are presented in Table 2-35, Laboratory Permeability Tests.

The clay mineral content and the total cation exchange capacity of seven selected soil samples was analyzed. The results of these tests are presented in Table 2-36, Cation Exchange and X-ray Diffraction Analyses.

Data obtained from the geologic exploratory borings indicate that a large portion of the site is mantled by relatively permeable Pleistocene soils. These soils have been eroded from a portion of the site exposing the Chesapeake Group which includes the St. Mary's, Choptank, and Calvert formations. The Chesapeake Group consists of about 270' of impervious sandy and clayey silts of Miocene age. Underlying this material are the Piney Point, Nanjemoy, and Aquia formations of Eocene age.

The Pleistocene deposits consist mainly of silts and sands which have fairly good infiltration characteristics. A few domestic wells in the area obtain water from this material, but are up-gradient from the plant and cannot be affected by a change in the ground water regimen at the site. Grain-size analyses of the surficial soil samples indicate a maximum permeability coefficient of about 400 GPD/ft^2. The elevation of the phreatic surface changes with the surface topography and can be expected to fluctuate slightly as a result of climatic changes. The water table occurs generally within 30' of the ground surface. East of the topographic divide, the direction of ground water movement is toward the Chesapeake Bay. The direction of ground water flow west of the divide is toward the existing stream valleys. Piezometers installed in the borings show that the ground water gradient at the site is generally less than 1%. The average rate of ground water flow probably does not exceed a few feet per day.

The underlying impervious sandy and clayey silts of the Chesapeake group extend to about 200' below mean sea level. A percolation test conducted near the plant indicates a permeability of less than 1 GPD/ft^2. Particle-size analyses indicate that the permeability of the Chesapeake Group averages about 3 GPD/ft^2. The rate of ground water movement is extremely low (much less than 1" per day). The formation is an aquiclude which effectively confines the underlying artesian aquifers. Regional studies by the U.S. Geological Survey have shown that the head in the artesian aquifers is above sea level. The result is vertical upward leakage through the Chesapeake Group. The rate of leakage is extremely low because of the low permeability of the Miocene sediments.

At the site, the combined thickness of the aquifers within the Piney Point and Nanjemoy formations is about 80'. They occur at Elevations ranging between 200' and 300' below mean sea level and are separated from the deeper Aquia formation by a layer of clay (Lower Nanjemoy) about 150' thick. The general

direction of ground water movement in the Aquia formation is toward the southeast with a piezometric gradient of about 2' per mile.

Grain-size analyses of samples of the Piney Point formation collected at the site indicate a permeability of about 150 GPD/ft^2. This value is probably typical of both the Piney Point and Nanjemoy aquifers. It is estimated that the permeability coefficient of the Aquia formation may be on the order of 1,000 GPD/ft^2. The computed rate of flow of ground water through these aquifers ranges from about .07 to .004' per day. The possibility of accidental contamination of the Eocene aquifers beneath the site is remote because; (a) the aquifers are covered by over 200' of relatively impervious soils and, (b) the vertical component of ground water movement is upward.

Cation exchange and X-ray diffraction analyses were performed on seven samples ranging in depth from 5 to 115' in order to evaluate the cation retention characteristics in the vicinity of the site. The cation exchange capacity of these soils is relatively high and would effectively absorb radioactive cations.

Three wells have been developed for plant use. They extend to a depth of approximately 640' and are each capable of producing 300 GPM from the Aquia formation. Casings for the three plant wells are continuous and sealed with grout to the top of the screens. The effect of pumping these wells will be to create a cone of depression in the Aquia formation such that the direction of groundwater flow will be toward the site rather than away from it. This will further minimize the possibility of lateral migration of any possible release of contaminated liquids beyond the site boundary. For other aquifers, the upward component of groundwater movement and the overlying aquiclude prevent contamination due to downward percolation.

2.5.4 REFERENCES

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6. U.S. Geological Survey Water Resources Division; Towson, MD; Mr. James Weigle, Mr. Wayne E. Webb
7. Maryland State Department of Health; Baltimore, MD; Mr. Charles Gross
8. Maryland Department of Geology Mines and Water Resources; Baltimore, MD; Mr. H. Hansen
9. Maryland Division of Water Resources; Annapolis, MD; Dr. Bruce Martin
10. Numerous local residents; Vicinity of Site

TABLE 2-31

AVERAGE MONTHLY DISCHARGE AT GAUGING STATION ON ST. LEONARD CREEK (1958-64)[a]

January	1.74"	May	1.35"	September	0.48"
February	1.92"	June	0.88"	October	0.56"
March	2.46"	July	1.00"	November	1.23"
April	2.04"	August	0.83"	December	0.99"

(a) Expressed in inches of runoff.

TABLE 2-32

GEOLOGIC UNITS IN CALVERT COUNTY

GEOLOGIC UNIT	APPROXIMATE RANGE IN THICKNESS (ft)	PHYSICAL CHARACTERISTICS	WATER BEARING PROPERTIES
Pleistocene surficial deposits	0-150	Silt and sand with some clay gravel	Yields small quantities of water to relatively shallow dug or driven wells.
Chesapeake Group: St. Mary's, Choptank, and Calvert Formations	30-325	Sandy and clayey silt with inter-bedded sand and fossiliferous layers	An aquiclude. Yields small supplies of water to a few dug wells.
Piney Point Formation	0-60	Glauconitic sand	Yields up to 200 GPM are reported from drilled wells. An important aquifer in Calvert County.
Nanjemoy Formation	40-240	Glauconitic sand with clayey layers. Basal part is red or gray clay	Yields of individual wells reported up to 60 GPM. An important aquifer in Calvert County.
Aquia Formation	30-200	Green to brown glauconitic sand	Yields up to 300 GPM reported from wells. An important aquifer in Southern Maryland.
Brightseat Formation	0-40	Gray to dark gray micaceous silty and sandy clay	Not known to be an aquifer in Southern Maryland.
Monmouth and Matawan Formations	20-135	Sandy clay and sand, dark gray to black, with some glauconite	Not a major aquifer in Southern Maryland, but yields up to 50 GPM have been reported.
Magothy Formation	0-40	Light-gray to white sand and gravel with interbedded clay layers	A few wells reportedly yield up to 1,000 GPM but average yields are considerably less. This aquifer is not used in Calvert County because of its depth.
Raritan Formation	100	Interbedded sand and clay with iron-stone nodules	Yields up to a few hundred GPM reported. Not utilized in Calvert County due to depth.
Patapsco Formation	100-650	Interbedded sand, clay, and sandy clay	Large-diameter wells yield up to 1,000 GPM. Not used in Calvert County because of depth.
Arundel Clay Formation	25-200	Red, brown, and gray clay	Not generally a water-bearing formation.

TABLE 2-32

GEOLOGIC UNITS IN CALVERT COUNTY

GEOLOGIC UNIT	APPROXIMATE RANGE IN THICKNESS (ft)	PHYSICAL CHARACTERISTICS	WATER BEARING PROPERTIES
Patuxent Formation	100-450+	Chiefly gray and yellow sand with interbedded clay	Yields of several hundred GPM reported. Not used in Calvert County due to great depth.
Precambrian	Unknown	Gneiss, granite, gabbro, meta-gabbro, quartz diorite, and granitized schist	Yields moderate supplies of ground water, generally not more than 50 GPM. Not used in Calvert County because of its great depth.

TABLE 2-33

PUBLIC SUPPLY WELLS IN CALVERT COUNTY (1967)

TOWN	POPULATION SERVED[a]	NUMBER OF CONNECTIONS	AVERAGE OUTPUT (mgd)	WELL	TOTAL DEPTH (ft)	DIA (in)
Calvert Beach	60	15	.006	1	475	5
Chesapeake Beach	500	155	.05	1[b]	400	8
				2	400	8
				3	400	6
				4[b]	400	2
				5	400	2
Chesapeake Ranch Estates	150			1	400	4
				2	750	4
				3	750	4
Dares Beach	600	175	.05	1[b]	217	1 1/2
				2	210	2 1/2
Hunting Hills	30	8	.003	1		4
Kenwood Beach	250	62	.025	1	365	1 ½
Long Beach	720	180	.07	1	525	3
				2	500	4
				3	475	4
				4	500	4
Prince Frederick	125	35	.02	1	552	8
St. Leonard	60	14	.006	1	550	5
Scientists Cliffs	120	185	.025	1	240	6
Western Shores				1	325	3
White Sands[c]	160 (approx.)	40	.012	1	402	6
	Comm.	2	.015	2	315	2 1/2

(a) Does not include seasonal increases, with the exception of Chesapeake Ranch.

(b) Not in use.

(c) Appropriate permits data per Maryland Department Natural Resources ground water.

NOTES:
1) Information based on 1963 data from United States Public Health Service and Maryland Department of Health.
2) Total population of a community is not necessarily served by the public water supply.

TABLE 2-34

KNOWN WATER WELLS, VICINITY OF SITE

OWNER	WELL NUMBER	WELL DEPTH (ft)	WELL DIAMETER (in)	AMOUNT OF CASING (ft)	YIELD (GPM)
K. C. Gerard	1	365	4	280	22
BGE	2	585	6	560	60
Yacht Club	3	315	2 1/2	189	25
H. Krellen	4	285	2	180	15
E. Zinn	5	384	4	237	25
B. Foot	6	399	2 1/2	252	20
H. C. Wilder	7	399	2 1/2	273	20
H. J. Mishou	8	404	2 1/2	277	*
D. Wood	9	315	2	273	5
BGE	10	540	6	520	10
R. C. Hall	11	274	2 1/2	*	50
G. D. Wait	12	252	2	*	3
D. Adams	13	300	2 1/2	*	*
William Rekar	14	461	3	*	*
E. Bowen	15	525	3	*	*
E. Daniels	16	450	*	*	*
W. Jenkin	17	435	*	140	*
Knotty Pine Bar & Grill	18	*	*	*	*
Bay Breeze Camp	19	340	*	*	*
Mrs. Faron	20	*	*	*	*
Mrs. Moran	21	*	*	*	*
*	22	365	2 1/2	*	*
Mr. McQueen	23	300	*	*	*
Mr. Street	24	375	*	*	*
P. Andjiano	25	360	*	*	*
Mr. Bancroft	26	*	*	*	*
Mr. Wohlgenmuth	27	*	*	*	*
Mrs. Mansfield	28	*	*	*	*
*	29	*	*	*	*
*	30	*	*	*	*
*	31	*	*	*	*

* Information not available.

NOTE: Well locations are shown on Figure 2.5-4.

TABLE 2-35

LABORATORY PERMEABILITY TESTS

BORING	DEPTH (ft)	SOIL TYPE	DRY DENSITY (lbs/ft^3)	COEFFICIENT OF PERMEABILITY (ft/day)
DM-6	45	Sandy silt with shells	93	3.18
DM-8	75	Sandy silt	82	0.86

TABLE 2-36

CATION EXCHANGE AND X-RAY DIFFRACTION ANALYSES

BORING	DEPTH (ft)	SOIL TYPE	GRADATION IN % FINER[a]				% OF TOTAL CLAY MINERALS[b]			TOTAL CATION EXCHANGE CAPACITY[c]	
			.074	.048 (in millimeters)	.005	.002	Illite	Montmorillonite and Mixed Clay Minerals	Chlorite	Test 1	Test 2
DM-1	45	Gray Silty Clay	98	44	23	17	30	50	20	23.8	24.5
DM-6	43 1/2	Green Silty Sand	28	25	21	15	40	60	-	10.0	11.0
DM-6	115	Green Clayey Silt	98	91	44	31	30	50	20	24.3	27.3
DM-8	30	Green Silty Sand	28	25	22	15	35	65	-	13.8	12.8
DM-8	45	Green Clayey Silt	90	80	45	35	20	60	20	25.0	30.6
DM-9	5	Reddish-Brown Sandy Clay	80	65	38	34	-	-	100	10.0	10.3
DM-10	47	Gray Sandy Silt	59	22	16	11	20	80	-	14.4	10.0

[a] Soil samples soaked for 24 hours in 0.4% hours in 0.4% sodium hexametaphosphate before hydrometer analysis.

[b] X-ray diffraction analyses of minus 2 micron material.

[c] Because of the $CaCO_3$ in some of the samples, the ammonium acetate method was used. Total Cation Exchange was determined on the minus 40 micron material.

NOTE: Swell observed in samples DM-1 (45’), DM-6 (115’), and DM-8 (45’).

2.6 SEISMOLOGY

2.6.1 INTRODUCTION AND SUMMARY

This section of the report presents the results of the engineering seismology phase of the environmental study. This phase of the study included literature research to compile a record of the seismicity of the area (References 1, 2, 6, and 8), evaluation of the geologic structure and tectonic history of the region, a program of dynamic laboratory soil testing, and analyses to evaluate the response of the foundation materials to earthquake-type loading. Field geophysical studies were performed to evaluate the in-situ dynamic properties of the foundation materials.

The site is located in a region which has experienced infrequent and minor earthquake activity. Most of the reported earthquakes are related to known faulting more than 50 miles west of the site in the Piedmont Physiographic Province. No known faults occur in the vicinity of the site. The closest earthquake (Intensity VII) which caused any structural damage occurred about 80 miles southwest of the site. Several minor shocks (no greater than Intensity V) have been reported within 50 miles of the site. Because of very limited data, it is not possible to determine whether or not these were of tectonic origin.

The foundations of major plant structures are established in dense Miocene soil which will not undergo reduction in strength or increased settlement under safe shutdown earthquake (SSE) conditions.

Significant earthquake ground motion is not expected at the site during the economic life of the nuclear facility. On a conservative basis, the power plant was designed to respond elastically, with no loss of function, to horizontal ground accelerations as high as 8% of gravity and vertical ground accelerations as high as 5-1/3% of gravity.

For safe shutdown of the reactor, a maximum horizontal ground acceleration of 15% of gravity was used in the design. A maximum vertical acceleration of 10% of gravity was used in the design. These accelerations are based on an assumed Intensity VII earthquake originating near the site.

The results of the regional study of seismicity and tectonics show that the aforementioned ground accelerations are conservative. Therefore, the nuclear power plant designed to these parameters meets all safety requirements.

It is not expected that the plant will be subjected to a significant tsunami effect. The maximum expected tsunami would not result in more than minor wave action at the site and, thus, was not significant in the design.

Geological and geophysical investigations performed since the late 1970s have led to speculation that many large Cretaceous and Cenozoic fault zones may exist in the Atlantic Coastal Plan. The closest of these are the Stafford and Brandywine fault systems, located about 45 and 29 miles west of the site, respectively.

An update of the seismic activity within 200 miles of the site focuses on two areas. The first is the occurrence of any seismic events since the last investigation that would be of greater significance than those previously considered. The second area is the evaluation of the earthquake catalog data that would lead to significant changes to the original SSE assessment. A significant earthquake catalog was developed by investigators and consultants of the Electric Power Research Institute in 1988 and the National Center for Earthquake Engineering Research in 1992. An independent evaluation of regional seismicity, performed by Bechtel Power Corporation in 1992, found that while more seismic events have been cataloged than were previously considered in the Updated

Final Safety Analysis Report, none of the additional events were larger or were in new or significantly different areas than those used to develop the SSE design basis. The basic premise of SSE specifications at Calvert Cliffs, an intensity MMI VII earthquake in the Atlantic Coast Plain province in the vicinity of the site, remains unchanged.

Reference 13 discusses the results of additional analyses of vibratory ground motion which were performed to determine the design ground acceleration level for the Diesel Generator Buildings. This supplemental information applies to the power block as well as to the Diesel Generator Buildings.

2.6.2 TECTONICS

The site is located in the Coastal Plain Physiographic Province. This province is bounded on the east by the Atlantic Ocean, and on the west by the Fall Zone and the Piedmont Physiographic Province. The Coastal Plain consists of easterly dipping Cretaceous and Tertiary sediments which are about 2,500' thick at the site. Crystalline basement rock outcrops near the Fall Zone about 50 miles west of the site. A graphical representation of the subsurface conditions at the site is shown on Figure 2.6-1, Columnar Section, Showing Geophysical Data.

On the basis of regional data, the Cretaceous and Tertiary sediments are undeformed. The absence of folding and faulting in the sedimentary strata indicates that displacements along unknown faults which may be present in the basement have been negligible.

No known faults occur within the basement rock or sedimentary deposits in the vicinity of the site. The closest known fault systems are found in the rocks of the Piedmont, more than 50 miles west and northwest of the site. The Piedmont Province consists of igneous and metamorphic rocks of Precambrian and early Paleozoic Age, with areas of sedimentary and igneous rocks of Triassic Age. Major tectonic activity that has occurred in the Mid-Atlantic Region can be related to known faults in the Piedmont Province.

2.6.3 SEISMICITY

The site is situated in a region which has experienced only infrequent minor earthquake activity. No shock within 50 miles of the site has been large enough to cause significant structural damage. Since the region has been populated for over 300 years, it is probable that all earthquakes of moderate intensity, say VI or greater, would have been reported during this period. It is very likely that all earthquakes of Intensity V or greater which occurred within the last 200 years have been reported.

The first report of earthquake occurrence in the general area of the site dates back to the late 18th Century. Since then, only 14 earthquakes with epicentral intensities of V or greater on the Modified Mercalli[a] Scale have been reported within about 100 miles of the site. None of these shocks was greater than Intensity VII. Few were of high enough intensity to cause structural damage and only one of these shocks can be considered more than a minor disturbance. This was an Intensity VII shock near Wilmington, DE in 1871 about 100 miles from the site. A list of earthquakes of Intensity V or greater with epicenters located within a distance of about 100 miles of the site is presented in Table 2-37, Significant Earthquakes within 100 miles of the site. Several smaller earthquakes, which are significant because of their proximity to the site, are also included in Table 2-37. The locations of these and other earthquakes in the region surrounding the site are shown on Figure 2.6-2, Epicentral Location Map. Several small shocks are shown on the Epicentral Location Map, but not indicated in Table 2-37. Little information is available regarding these shocks. The indicated epicentral locations are those suggested by G.P. Woollard (Reference 12).

Most of the reported earthquakes in the region have occurred in the Piedmont Physiographic Province west of the Fall Zone. The closest approach of the Fall Zone to the site is about 50 miles. These shocks were generally related to known faults in the Piedmont rocks.

There have been several large shocks with epicenters in the Coastal Plain, some of which were damaging. The largest of these is the Charleston, South Carolina, earthquake of 1886, which has an epicentral intensity of about IX. Geological and seismological research in the meizoseismal vicinity of Charleston appear to support the view that the Charleston earthquake occurred in association with a specific seismogenic structure located near Charleston, and there is no need to consider a site intensity of MMI X for seismic design at Calvert Cliffs.

While the Giles County, Virginia Seismic Zone is just over 200 miles from the Calvert Cliffs site, it is relevant to discuss this seismically active zone in that it was the location of a MMI VII intensity earthquake on May 31, 1897. However, the seismic activity around Giles County appears to be very distinct in character as compared to seismicity in the nearer Central Virginia Seismic Zone. There appears to be no reason to assume the 1897 Giles County earthquake is applicable to the Calvert Cliffs site seismicity.

The largest earthquake in the Coastal Plain close enough to the site to be of significance in the current study occurred in 1927 near the northern New Jersey coast, about 180 miles northeast of the Calvert Cliffs site. The epicentral intensity of this earthquake was VII. Three shocks were felt over an area of about 3,000 square miles from Sandy Hook to Tom’s River. Highest intensities were felt from Asbury Park to Long Branch where several chimneys fell, plaster cracked, and articles were thrown from shelves. This shock has not been related to any known geologic feature.

An earthquake which occurred near Wilmington, DE, in 1871 is the largest reported earthquake within 100 miles of the proposed plant site. It is not possible to accurately locate the epicenter of this shock with the limited data available, but it is probable that the shock occurred along the Fall Zone about 100 miles northeast of the site. The epicentral intensity of this shock is rated at VII. At Wilmington, chimneys toppled and windows broke. Damage was also reported at Newport, New Castle, and Oxford, DE. The earthquake was felt over a relatively small area of northern Delaware, southeastern Pennsylvania, and southwestern New Jersey.

Only one earthquake of Intensity V or greater has been reported within 50 miles of the proposed plant site. This shock, which had a rated epicentral intensity of V, caused no structural damage. Its epicenter was located near Seaford, DE, about 45 miles northeast of the site.

Several small shocks have been reported in the Coastal Plain in the region surrounding the site. Four such shocks were reported east and south of the site. Several other small shocks were reported in the vicinity of Annapolis, MD, northwest of the site. These reported earthquakes were considered in this investigation; however, available data regarding these shocks are limited, and it is impossible to estimate their maximum intensities or to precisely locate their epicenters.

None of these earthquakes caused structural damage and they are of interest only in that they may indicate the possible presence of unidentified faulting in the basement rock of the Coastal Plain. However, it is possible that some of these reports may refer to relatively distant earthquakes which were felt in eastern Maryland. Furthermore, it is also possible that these shocks resulted from causes other than tectonic activity. The probable epicenters of these small shocks are shown on the Epicentral Location Map, Figure 2.6-2.

An independent evaluation of regional seismicity performed in 1992, to support design of the Diesel Generator Building, identified more earthquakes than were previously identified. However, these additional earthquakes were neither larger nor in different areas than those used to develop the original SSE design basis.

2.6.4 GEOPHYSICAL INVESTIGATIONS

2.6.4.1 General

Geophysical studies were performed at the Calvert Cliffs site in order to evaluate the dynamic properties of the foundation materials. The dynamic soil properties are used in evaluating the response of the foundation materials to earthquake loading.

A seismic refraction survey and an uphole velocity survey were performed in order to measure the velocity of compressional wave propagation at the site. The shear wave velocity was estimated from the field measurements of surface waves (predominantly Rayleigh waves). Micromotions were measured in order to indicate the pattern of vibration at the site due to background "noise." These measurements assist in estimating the natural period of vibration at the site. Laboratory shockscope tests were performed for correlation with the field measurements.

Shear and compressional wave velocity measurements were derived for the upper strata during the geophysical investigations. Compressional wave velocities for the deeper strata near the site were measured during a geophysical survey performed by Ewing and Worzel in 1943. Shear wave velocities were computed from these data using an estimated Poisson's ratio. Geophysical data for the entire stratigraphic section and presented on Figure 2.6-1.

2.6.4.2 Refraction Seismic Survey

Seismic refraction surveys were performed along two lines, 2,000' and 2,100' in length, as shown on Figure 2.4-7, Plot Plan. The purpose of these surveys was to evaluate the compressional wave velocities of the sediments underlying the site. The work was conducted with an Electro-Technical Labs M4E seismograph. Dynamite charges, from 10 to 50 lbs at each end of the seismic lines, were used as a source of energy. The charges were buried at depths of 25 to 60'.

Geophones were located at intervals ranging from 20 to 100' along each line. Data from these field measurements indicate that the velocity of compressional wave propagation in the upper surficial Pleistocene silts and sands is about 2,200 fps, and in the Miocene sandy and clayey silt about 5,900 fps.

Extensive geophysical surveys in the Coastal Plain were made in 1943 by Ewing and Worzel. These surveys included measurements to the crystalline basement rock at a point several miles south of the site. The results of these measurements and the measurements made at the site by Dames & Moore are summarized in Table 2-38 Geophysical Data.

2.6.4.3 Uphole Seismic Velocity Survey

An uphole seismic survey was performed in Boring DM-4 in the proposed plant area. The purpose of this survey was to correlate the compressional wave velocities of the materials in the plant area with the compressional wave velocities measured by the refraction survey approximately 1,000' to the northwest.

The uphole seismic survey was performed using the Electro-Technical Labs ER-75-12 seismograph. The seismic energy was provided by either blasting caps or charges of one to three ounces of dynamite detonated at 10’ intervals in the boring to a maximum depth of 148’. Geophones were placed at 5 to 60’ intervals at distances up to 220’ from the boring. The results of this survey are presented on Figure 2.6-3, Uphole Seismic Survey.

2.6.4.4 Shear Wave Measurements

The velocity of shear wave propagation was evaluated at the site. The compressional wave and shear wave velocities were used to compute Poisson’s ratio and the dynamic properties of the soil. The shear waves were estimated from surface waves (predominantly Rayleigh waves) measured with a Sprengnether velocity meter. These measurements indicate that the velocity of shear wave propagation at the site is about 1,600 fps in the Miocene sediments.

2.6.4.5 Micromotion Measurements

Micromotion measurements were made at three locations at the site using the Dames & Moore Microtremor Equipment. The micromotion measuring stations are shown on Figure 2.4-7, Plot Plan. The equipment used is a highly sensitive recording device capable of magnifying ground motions up to 150,000 times and is accurate over a frequency range of 1 to 30 CPS.

The microtremor records indicate a predominant period of background vibration on the order of one-half to three-quarters of a second. The low intensity levels are consistent with what would be expected in a reasonably dense material. The predominant ground period and intensity of ambient motion at the site will present no special problems in design of the facility.

2.6.4.6 Laboratory Shockscope Tests

Several representative samples of the soil underlying the site were tested in the Shockscope. The Shockscope is an instrument developed by Dames & Moore to measure the velocity of propagation of compressional waves in soil. The velocity of compressional wave propagation observed in the laboratory was correlated with the field measurements and used as an aid in evaluating the dynamic elastic properties.

In the Shockscope test, the samples were subjected to physical impulses under a range of confining pressures while the time necessary for the shock wave to travel the length of the sample was measured using an oscilloscope. The velocity of compressional wave propagation was then computed. The results of these tests are presented in Table 2-39, Shockscope Test Data.

2.6.4.7 Diesel Generator Building Siting Surveys

Additional analyses of vibratory ground motion were performed to support design of the Diesel Generator Buildings. A significant earthquake catalog was developed by investigators and consultants of Electric Power Research Institute and the National Center for Earthquake Engineering Research. A subsequent independent evaluation of this regional seismicity was performed by Bechtel Power Corporation in 1992.

2.6.5 ASEISMIC DESIGN

2.6.5.1 Foundations

The foundations of the major plant structures are established in the Miocene sandy and clayey silts of the Chesapeake Group. These soils are apparently preconsolidated as a result of deposition and subsequent erosion of younger sediments, as well as desiccation and increase in effective pressure caused by lowering of the water table. Some appurtenant structures at the site are founded in the surficial Pleistocene silt and sand which overlies the Miocene sediments.

2.6.5.2 Operating Basis Earthquake (No Loss of Function)

On the basis of the seismic history of the area, it does not appear likely that the site will experience significant earthquake ground motion during the economic life of the proposed facility. The nuclear power plant was conservatively designed to respond elastically, with no loss of function, to horizontal earthquake ground accelerations of 8% of gravity, and vertical earthquake ground accelerations of 5-1/3% of gravity. It is not believed that this level of ground motion will be exceeded at the site during an earthquake similar to any historical event. This ground acceleration is considerably greater than what might be expected due to an Intensity V shock (Magnitude 4 on the Richter Scale[a]) close to the site, or to an Intensity VII (Magnitude 5) shock at an epicentral distance of about 15 to 20 miles.

2.6.5.3 Safe Shutdown Earthquake (Safe Reactor Shutdown)

For a safe shutdown of the reactor, the facility was designed using a horizontal ground acceleration of 15% of gravity at foundation level, and a corresponding vertical ground acceleration of 10% of gravity. It is not believed that this level of earthquake ground motion would be exceeded during the maximum potential earthquake. The magnitude of vertical ground motion was estimated on the assumption that vertical particle motions due to compressional waves have magnitudes of about one-half to two-thirds of the horizontal particle motions due to shear waves.

The SSE for the site is considered to be a shock similar to one of the following:

a. A shock equivalent to the Intensity VII, 1871 Wilmington earthquake as close to the site as its related geologic structure. This earthquake was probably a Magnitude 5. It is likely that this earthquake was related to faulting in the Piedmont west of the Fall Zone. However, since it is impossible to precisely locate the epicenter of this shock from the limited available data, and since the earthquake was felt in portions of the Coastal Plain, it was considered that the epicenter of this shock may have been located somewhat east of the Fall Zone.

b. A shock equivalent to the Intensity VII northern New Jersey earthquake of 1927 occurred close to the site. This shock occurred in the Coastal Plain and has not been related to known geologic structure. Therefore, the conservative assumption was made that it could occur along a hypothetical geologic structure in the basement rock near the site. This earthquake was probably about a Magnitude 5 to 5-1/2.

c. A shock equivalent to the Intensity IX Charleston earthquake of 1886 recurring at or near the original epicenter. An Intensity IX (Magnitude 7)

[a] Earthquake magnitudes in this section refer to the magnitude scale developed by Dr. C. F. Richter. The magnitude scale is a means of indicating the size of an earthquake based on instrumental records. The magnitude scale is further described in Section 2.6.8.

earthquake centered near Charleston, about 480 miles southwest of the site, would not be of significance at the site.

Based on the foregoing statements, the very conservative assumption has been made that the SSE would be a shock as large as Intensity VII (Magnitude 5 to 5-1/2) originating in the basement rock close to the site.

Although a later, independent evaluation of region seismicity performed in 1992 (in support of the Diesel Generator Project) identified more recent earthquakes than those presented here, these were not larger nor were they in different areas than those used to develop the original SSE basis.

2.6.5.4 Response Spectra

Ground motion response spectra are presented on Figures 2.6-4 and 2.6-5, Response Spectra-Operating Basis Earthquake (OBE) and Response Spectra-SSE, respectively. These spectra conform to the average spectra developed by Dr. G. W. Housner for the frequency range higher than about 0.33 CPS. These average spectra were originally presented in TID-7024 (Reference 5). The spectra presented herein are based on a later revision by Dr. Housner, presented for the H. B. Robinson Nuclear Power Plant of Carolina Power and Light Company (Reference 9). The spectra for frequencies lower than about 0.33 CPS were prepared utilizing data suggested by Dr. N. M. Newmark (Reference 7).

The spectra have been normalized to a horizontal ground acceleration of 8% of gravity for the OBE and 15% of gravity for the SSE.

The response spectra indicate the estimated response of a structure subjected to earthquake ground motion. The spectra are presented over a range of frequencies corresponding to the natural frequencies of the various structural elements and represent the maximum amplitude of motion in the various elements of the structure for typical degrees of damping.

The digitalized El Centro Earthquake (1940, East-West), normalized to a ground acceleration of 0.08g horizontally and 0.053g vertically, acting simultaneously, was used in the analysis of Category I equipment. See paragraph 5.1.3.2 for description of dynamic response spectra.

2.6.6 TSUNAMIS

The occurrence of tsunamis is infrequent in the Atlantic Ocean. Other than the tidal fluctuation recorded on the New Jersey shore during the Grand Banks earthquake of 1929, there has been no record of tsunamis on the northeastern United States coast. The earthquake of November 18, 1929, on the Grand Banks about 170 miles south of Newfoundland, resulted in a tsunami which struck the south end of Newfoundland, about 750 miles northeast of the Massachusetts Coast. This tsunami occurred at a time of abnormally high tide and resulted in some loss of life and destruction of property. The effect of this tsunami was recorded on tide gauges along the east coast of the United States as far south as Charleston, SC. A tidal fluctuation of approximately nine-tenths of one foot was noted at Atlantic City, NJ and Ocean City, MD (Reference 11).

The Lisbon earthquake of November 1, 1755, produced great waves which contributed heavily to destruction on the coast of Portugal. These waves were noticeable in the West Indies. It has been reported that the Cape Ann, MA, earthquake of November 18, 1755, caused a tsunami in Saint Martin’s Harbor in the West Indies. However, there is no record of tsunami occurrence along the east coast of the United States at this time, and it appears that the Saint Martin’s Harbor report actually refers to the tsunami caused by the

Lisbon earthquake, which occurred less than three weeks before the Cape Ann shock. Some tsunami activity has occasionally followed earthquakes in the Caribbean, but none of these was reported in the United States (References 4 and 10).

It is not believed that the site will be subjected to a significant tsunami effect. The maximum expected tsunami would result in only minor wave action, and the maximum expected storm wave effect, discussed in Section 2.8.3, Hurricane Tidal Effects, was a more critical factor in the design.

2.6.7 MODIFIED MERCALLI INTENSITY SCALE

The Modified Mercalli Intensity scale of 1931 is described in Table 2-40. The intensity scale is a means of indicating the relative size of an earthquake in terms of its perceptible effect. The intensities presented in this report indicate the damage caused by an earthquake at its epicenter.

2.6.8 RICHTER MAGNITUDE SCALE

Magnitude Scale is a means of indicating the size of an earthquake based on instrumental records.

Dr. C. F. Richter developed a magnitude scale which is based on the maximum recorded amplitude of a standard seismograph located at a distance of 100 kms from the source of a shallow earthquake. The magnitude is defined by the relationship $M=\log A - \log A_o$. In this equation, A is the recorded trace amplitude for a given earthquake at a given distance written by the standard instrument, and A_o is the trace amplitude for a particular earthquake selected as a standard. The zero of the scale is arbitrarily fixed to fit the smallest recorded earthquakes. The largest known earthquake magnitude is on the order of 8-3/4. This magnitude is the result of observations and not an arbitrary scaling. The upper magnitude limit is not known, but is estimated to be about 9.

Empirical relationships between earthquake magnitude and energy release have been developed by several investigators (Reference 10). There is no exact relationship between earthquake magnitude and energy for large earthquakes, and these empirical relationships should be considered no more than approximations.

2.6.9 REFERENCES

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3. M. Ewing and L. Worzel, 1948, Explosion Sounds in Shallow Water, Geological Society of America, Memoir 27
4. B. Gutenberg and C.F. Richter, 1954, Seismicity of the Earth and Associated Phenomena, Princeton University Press, Princeton
5. G.W. Housner, 1963, Response of Structures to Earthquake Ground Motion, Nuclear Reactors and Earthquakes (TID-7024), United States Atomic Energy Commission, Division of Technical Information
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7. N.M. Newmark, 1969, Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards, Proceedings of the International Atomic Energy Agency Panel on Aseismic Design and Testing of Nuclear Facilities, Tokyo, Japan, 1967

8. Preliminary Determination of Epicenters - (Card Series 1966 through 1967) United States Department of Commerce, Coast and Geodetic Survey, Washington, DC

9. Preliminary Safety Analysis Report, H.B. Robinson Nuclear Power Plant, Carolina Power and Light Company

10. C.F. Richter, 1958, Elementary Seismology, W.H. Freeman and Company, San Francisco, CA

11. United States Earthquakes (Serial Publications, 1928 through 1965) United States Department of Commerce, Coast and Geodetic Survey, Washington, DC

12. G.P. Woollard, 1958, Areas of Tectonic Activity in the United States As Indicated By Earthquake Epicenters, Transactions of the American Geophysical Union, Volume 39

13. Letter from R.E. Denton (BGE) to Document Control Desk (NRC), dated December 18, 1992, Emergency Diesel Generator Project-Civil Engineering Design Report

TABLE 2-37

SIGNIFICANT EARTHQUAKES WITHIN 100 MILES OF THE SITE

YEAR	DATE	TIME	INTENSITY	LOCATION	N. LAT.	W. LONG.	AREA FELT (sq. mi.)	DISTANCE FROM SITE (miles)
1733	June 14	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1758	April 24	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1774	Feb. 21	14:00	VI	Richmond, VA	37 1/2	77 1/2	--	80
1833	Aug. 27	06:00	VI	Central Virginia	37 3/4	78	52,000	90
1871	Oct. 9	09:40	VII	Wilmington, DE	39 3/4	75 1/2	--	100
1875	Dec. 22	23:45	VI	Near Richmond, VA	37 1/2	77 1/2	50,000	80
1876	June 19	--	[a]	Vicinity of Annapolis, MD	--	--	--	--
1879	Mar. 25	19:30	IV-V	Northern Delaware	39 3/4	75 1/2	600	100
1883	Mar. 11	18:57	IV-V	Harford County, MD	39 1/2	76 1/2	Local	80
	Mar. 12	00:00	IV-V	Harford County, MD	39 1/2	76 1/2	Local	80
1885	Jan. 2	21:16	V	Frederick County,	39 1/2	77 1/2	3,500	80
1897	Dec. 18	18:45	V	Ashland, VA	37 3/4	77 1/2	7,500	75
1906	May 8	12:41	V	Seaford, DE	38 3/4	75 3/4	400	45
1908	Aug. 23	04:30	V	Powhatan, VA	37 1/2	78	450	95
1919	Sept. 5	21:46	VI	Front Royal, VA	38 3/4	78 1/4	--	95
1930	Jan. 18	--	IV[a]	Pines of the Sermon, MD	--	--	--	--
1930	Nov. 1	01:34	I-III[a]	Anne Arundel County, MD	39.0	76.5	Local	--
1949	May 8	06:01	IV-V	Richmond, VA	37 1/2	77 1/2	1,800	80
1966	May 31	06:19	IV-V	Central Virginia	37.6	78.0	--	100

[a] Several small shocks in Maryland are included in this table. Little information is available regarding these reports, and the indicated epicenters are uncertain. See text of report for discussion.

TABLE 2-38

GEOPHYSICAL DATA

STATION	SURFICIAL SEDIMENTS (PLEISTOCENE) COMPRESSIONAL WAVE VELOCITY (fps)	SURFICIAL SEDIMENTS (PLEISTOCENE) THICKNESS (ft)	UNCONSOLIDATED SEDIMENTS (TERTIARY) COMPRESSIONAL WAVE VELOCITY (fps)	UNCONSOLIDATED SEDIMENTS (TERTIARY) THICKNESS (ft)	INTERMEDIATE SEDIMENTS (CRETACEOUS)[a] COMPRESSIONAL WAVE VELOCITY (fps)	INTERMEDIATE SEDIMENTS (CRETACEOUS)[a] THICKNESS (ft)	BASEMENT ROCK COMPRESSIONAL WAVE VELOCITY (fps)	BASEMENT ROCK DEPTH (ft)
Solomons Shoal[b]	--	--	5900	3080	--	--	15,170	3130
Solomons Deed[b]	--	--	6080	1070	6980	1900	18,100	3080
Site[c]	2200	40	5500	--	--	--	--	--
Site[c]	--	--	5900	--	--	--	--	--

(a) These measurements refer to a "masked" arrival and the results are questionable.

(b) Adapted from Ewing and Worzel (Reference 3).

(c) Measurements by Dames & Moore.

TABLE 2-39

SHOCKSCOPE TEST DATA

BORING	DEPTH (ft)	CONFINING PRESSURE (lbs/ft^2)	COMPRESSIONAL WAVE VELOCITY (fps)
DM-2	5	0	1,000
		2000	1,200
		4000	1,400
		6000	1,700
DM-9	15	0	1,200
		2000	1,300
		4000	1,500
		6000	1,700
DM-1	30	0	1,400
		2000	1,500
		4000	1,800
		6000	2,100
DM-10	68	0	2,600
		2000	2,600
		4000	3,200
		6000	3,200
DM-10	111	0	2,600
		2000	2,600
		4000	3,000
		6000	3,000
DM-10	156	0	1,800
		2000	1,800
		4000	1,900
		6000	1,900
DM-10	211	0	1,600
		2000	1,700
		4000	1,700
		6000	1,700
DM-10	256	0	2,100
		2000	2,100
		4000	2,200
		6000	2,200
DM-10	271	0	2,000
		2000	2,200
		4000	2,300
		6000	2,600

TABLE 2-40

MODIFIED MERCALLI INTENSITY (DAMAGE) SCALE OF 1931 (Abridged)

I.	Not felt except by a very few under especially favorable circumstances. (I Rossi-Forel Scale)
II.	Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing. (I to II Rossi-Forel Scale)
III.	Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motorcars may rock slightly. Vibration like passing of truck. Duration estimated. (III Rossi-Forel Scale)
IV.	During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing motorcars rocked noticeably. (IV to V Rossi-Forel Scale)
V.	Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop. (V to VI Rossi-Forel Scale)
VI.	Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight. (VI to VII Rossi-Forel Scale)
VII.	Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motorcars. (VIII Rossi-Forel Scale)
VIII.	Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving motorcars disturbed. (VIII+ to IX Rossi-Forel Scale)
IX.	Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken. (IX+ Rossi-Forel Scale)
X.	Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks. (X Rossi-Forel Scale)
XI.	Few, if any, (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.
XII.	Damage total. Waves seen on ground surface. Lines of sight and level distorted. Objects thrown upward into the air.

2.7 SUBSURFACE AND FOUNDATIONS

2.7.1 INTRODUCTION

This summarizes the results, analyses, and evaluation of the subsurface and foundation investigations. The field exploration and the laboratory testing were done under the supervision and direction of Bechtel Associates. These studies included site and area reconnaissance, field supervision of the boring operations, a review of pertinent literature, and the foundation analysis and evaluation. The initial graphic boring logs and laboratory test data cited were presented in the Preliminary Safety Analysis Report (PSAR). Subsequent data and subsurface profiles are presented here.

The Civil Engineering Design Report for the Emergency Diesel Generator Project describes: (1) exploration, (2) the properties of subsurface materials, (3) groundwater conditions, (4) response of soil and rock to dynamic loading, (5) liquefaction potential, and (6) static and dynamic stability at the site of the Diesel Generator Buildings.

2.7.2 EXPLORATION

2.7.2.1 Field Reconnaissance

The geologic field work for this site was started concurrently with a drilling program. The site reconnaissance was a continuation of the field work done in the early part of 1967. Local soil outcrops were examined on the Chesapeake Bay bluffs and the soils, types, orientation, and variations noted.

In addition to the site reconnaissance, the area was examined as the drilling progressed. Also, geological information was gained from the Maryland Geological Survey and various publications (References 1, 2, 3, 4, and 6). The geologic information gained from the site reconnaissance and literature review was used in conjunction with the borings to prepare the site geology portion of this section. Also, Section 2.4, Geology, gives a comprehensive presentation of the site and regional geology.

2.7.2.2 Boring and Sampling Investigations

In June and July 1967, a preliminary foundation exploration for the proposed nuclear power plant was conducted at the site by BGE. The field program included five borings, B1 through B5, and numerous split-spoon soil samples.

The field exploration for the PSAR and plant design began with the initial reconnaissance of the site by personnel from BGE and Bechtel Associates. Subsurface exploration started on August 17, 1967, and was completed on September 22, 1967. This exploration included drilling 22 soil test borings in the vicinity of the plant site and one geologic boring on a bluff approximately 2,000' north of the site for the Maryland Academy of Science.

During November and December 1968, a supplementary subsurface investigation was undertaken. A series of 18 borings were drilled in the plant area. The primary purpose of the supplementary program was to provide additional information for the plant foundation analyses.

In order to obtain additional information necessary for the design of the Intake Structure and switchyard foundations, a final subsurface investigation was undertaken during spring 1969. Five test borings were drilled in switchyard area and borings WB-1 through WB-35 were drilled from a barge offshore of the Intake Structure. The location plan of the borings is shown on Figures 2.7-1 to 2.7-3.

During all field investigations, a geologist or soil engineer from Bechtel Associates continuously supervised and inspected drilling operations and modifications in the boring programs as deemed necessary.

Split-spoon samples and undisturbed Shelby tube samples were obtained at desired sampling intervals. All samples were initially visually inspected and classified in the field. Samples were then either forwarded to Bechtel Associates' Washington Area Engineering Office for further examination, or to a soils laboratory for testing. The boring logs subsequent to the PSAR are shown on Figures 2.7-4 to 2.7-26.

2.7.2.3 Geotechnical Investigations to Support Diesel Generator Building Siting

Two field investigations were conducted, one in 1980/1981 and another in 1992, which assessed the geotechnical conditions at the proposed site of the Diesel Generator Buildings. In 1980/1981, a field investigation was performed to assess the North Parking Area's suitability as a location for a generic Category I structure. The investigation consisted of sample borings, cone penetration soundings, and observation wells. In 1992, a second field investigation was performed which consisted of sample borings, dilatometer soundings, a crosshole seismic survey, and field soil resistivity tests. In both investigations, standard penetration test samples were obtained in accordance with ASTM D 1586. Thin-walled tube samples were obtained in general accordance with ASTM D 1587. The crosshole seismic survey was performed in accordance with the requirements of ASTM D 4428/4428M-84 by using the "preferred method." Field soil resistivity testing was conducted using the "Wenner four-electrode method" in accordance with ASTM G 57. Additional details about the geotechnical investigations can be found in the Civil Engineering Design Report for the Diesel Generator Project (Reference 29).

2.7.3 SITE CONDITIONS

2.7.3.1 Area Geology

A geology summary for the purpose of understanding the foundation evaluation is presented herein. The main geologic presentation is in Section 2.4.

The CCNPP site lies in the Atlantic Coastal Plain physiographic region of Maryland. It is in an area of sedimentary deposits formed by the ancient rivers which carried large quantities of solids from the northern and western uplands of the Piedmont and Appalachian physiographic provinces into the once larger Atlantic Ocean. These deposits were formed in both a freshwater (fluvial) and a saltwater (marine) environment.

The upper deposits in this coastal plain area are the Recent and Pleistocene deposits of tan and brown silts, sands and clays with some inclusions of seashell fossils. Below the Pleistocene deposits lie the older Miocene sediments. The Miocene deposits represent the soils of significant interest for the foundations of this power plant. The foundation properties of both the Pleistocene and Miocene are more than adequate for the plant loads. The crystalline basement rocks are approximately 2500' below the present ground surface.

2.7.3.2 Soil Conditions

For foundation engineering purposes, the soils at the site can be divided into an upper zone and a lower zone. These soils are a mixture of marine and fluxial deposits. Each zone has both continuous and discontinuous strata with isolated pockets or lenses of slightly different materials. The soils are non-uniform

sedimentary deposits of silty sands, sandy silts, clayey sands and sandy clays with layers of shell fossils.

The upper zone is generally yellowish tan, brownish tan and light brown in color. This color is caused by oxidation of the mineral constituents. The soil is firm to dense in consistency. The predominant soil types are sandy silts and silty sands. The average thickness is 18' and varies in elevation dependent upon the topography. This zone is primarily the Pleistocene deposit.

The lower zone, the Miocene deposit, is greenish gray in color with several occurrences of medium to light gray soil at the top of the zone. The soil below +5' MSL is very dense to extremely dense with a few lenses, which are isolated but dense in consistency. The major soil types are sandy silts, silty sands, and slightly clayey sands. The lower part of this zone is classified as fine sands and silts.

The upper zone of soil can support light loads, on the order of 2000 to 3000 psf, with a small amount of anticipated consolidation, while the lower zone can support heavy loads on the order of 15,000 to 20,000 psf with slight consolidation.

The original groundwater surface was between +15' and +20' MSL in the plant area; however, a permanent pipe drain system, subsurface drain system, surrounding the plant will maintain the ground water below Elevation +16'. Additional information concerning groundwater appears in Section 2.5. Subsurface profiles are shown on Figures 2.7-27 and 2.7-28.

2.7.4 LABORATORY TESTING

The laboratory testing program provided the soils' physical characteristics for foundation design. The testing was conducted in accordance with currently accepted procedures (References 7 through 12).

The testing program was divided into three parts, to determine the soil parameters under static, dynamic, and remolded (fill) conditions. The laboratory program included: grain size and specific gravity tests to determine particle size and distribution; Atterberg limit tests to determine soil plasticity characteristics; consolidation tests, to determine the soil settlement characteristics; unconfined compression and static triaxial shear tests to aid in the evaluation of foundation bearing capacity and slope stability analysis; dynamic triaxial shear tests to determine the dynamic properties used in the evaluation of liquefaction potential of foundation materials; compaction tests; and numerous moisture-density, void ratio and relative density determinations.

2.7.5 STRUCTURAL DATA

The foundations for the Turbine Building, Auxiliary Building, Containments, Turbine-Generators, and Circulating Water System are mat foundations on the Miocene soils. Individual bearing capacities were required because such a value depends on load, elevation of foundation, settlement tolerance, foundation size, and proximity to other loads. The final design bearing loads are as follows:

Structure	Contact Pressure
Containment Structure Mat	8000 psf
Auxiliary Building Mat	8000 psf
Turbine Pedestal Mat	5000 psf
Turbine Building Column Footings	5000 psf
Intake & Discharge Structure Mat	2500 psf

In all the above cases, the allowable soil bearing capacity exceeds the contact pressure. Two groups of structures, the circulating water structures (i.e., intake and discharge structures), and the switchyard structures have been studied since the PSAR and are discussed below.

2.7.5.1 Circulating Water Structures

The intake structure is located between the Turbine Building and shoreline, and is approximately 90 x 385’ in size. The total effective load due to the structure is approximately 42,000 tons. As a result, net soil pressures due to the structure will be approximately 2500 psf. The size and total loads were increased due to design changes necessitated by the structure being changed from a partial to a total Category I Structure. A 300’ segment of anchored sheet piling extends from the intake structure to the inside intake channel at the shoreline. The invert inlet elevation of the intake structure is Elevation -26’ MSL and the elevation at the junction of the anchored sheet piling and the intake channel is -51’ MSL. The approximate slope of the excavation from the intake inlet to the channel junction is approximately 10 horizontal to 1 vertical. The channel extends 4500’ offshore from the shoreline with 5 horizontal to 1 vertical side slopes excavated in the dense, silty slightly, cemented sand.

The discharge facility is located north of the plant. It consists of four conduits extending from the Turbine Building to a point 850’ offshore. The top of the conduit is Elevation -6’ MSL and the invert is at Elevation -19.5’ MSL at the point of discharge. The conduit will be constructed and buried by cut and cover methods. The plans of the intake and discharge scheme are shown on Figure 2.7-29.

2.7.5.2 Switchyard Structures

The switchyard area is located approximately 500’ west of the plant area. The north half of the yard is in a cut area, the maximum cut being approximately 30’. The south half of the yard is a fill area with a maximum thickness of about 20’.

Analyses show an allowable soil bearing capacity of 2000 psf in the fill area and 3000 psf in the cut excavated area. Drilled piers were used where higher loads and/or uplift conditions required deep foundations.

2.7.6 FOUNDATION EVALUATION

The soils at this site are suited for the construction of the plant. The upper zone, Pleistocene soils, will support light loads of 2000 to 3000 psf without adverse settlements. The lower zone, the Miocene, is exceptionally dense and will support heavy foundation loads on the order of 15,000 to 20,000 psf.

2.7.6.1 Site Excavation and Earthwork

The general site grading and excavation was done with conventional earth moving equipment. Soil compaction requirements were prepared and based on the proposed utilization of a filled area. In general, the bearing capacity of the fill was not the controlling engineering parameter; but rather, it was found that settlement controls.

A maximum settlement of 1" was the limit set in the computations to determine the contact pressure for 10’x10’ and smaller footings to be placed on the compacted fills.

The following criteria were formulated for the various loading conditions based on the soil test data for the proposed fill materials.

Fill Compaction (minimum)	Areas Where Criterion Is Used
85% Standard Proctor (ASTM D698; AASHO T-99)	Shore protection landfill except within 100' of bulkhead or anchor sheet piling and within 25' of culverts.
90% Standard Proctor	General, nonstructure supporting, fill areas and plant parking lot lower than 5' below finished grade.
95% Standard Proctor	Shore protection landfill within 100' of bulkhead or anchor sheet piling and within 25' of culverts and switchyard fill.
97% Standard Proctor	Structural backfill areas supporting facilities with footings 10'x10' or smaller with contact pressure of 4000 psf or less.
95% Modified Proctor (ASTM D1557: AASHO T-180)	Roadway embankments and subgrades.
95% Modified Protector (ASTM D155; AASHO T-180)	Structural fill for the Diesel Generator Buildings (consisting of well graded, sound, dense and durable crushed stone).
100% Modified Proctor	Structural backfill areas supporting facilities with footings 10'x10' or smaller with contact pressures of 5000 psf or less.

The above criteria covered the majority of the backfill conditions. Unique conditions of footing size and load were evaluated on an individual basis.

The minor plant excavation and embankment slopes were constructed according to the following tabulation:

Slope Height	Temporary Slope	Permanent Slope
0-30'	1:1	1 1/2:1
30-50'	1 1/4:1	2:1

These slopes have a factor of safety in excess of 1.5. In addition to the minor slopes, five other slopes adjacent to the plant were evaluated. Attached, at the end of this section, are the five cross-sections of the slopes around the plant. The locations of these slopes are indicated on Figure 2.7-30, "Slope Cross-Sections at Plant."

These cross-sections show the range of topographic conditions that exist. The backfill on the north, west, and south sides of the plant is to Elevation 45', and on the east side to approximately Elevation 45' but sloping to the Chesapeake Bay.

Stability analyses (Reference 24) were made for the design slopes shown on the cross-sections. The slopes shown on Section DD and EE are the maximum in height within the immediate plant area. These slopes have a safety factor in excess of 1.5. The other slopes around plant area are flatter or of less height; therefore, by inspection, safety factors greater than 1.5 can be assigned to these slopes. All of the slopes are acceptable for permanent slopes (Reference 22). Two dynamic slope stability analysis methods were used to evaluate the safety of the slopes for conditions resulting from the SSE.

In the conventional method of dynamic slope stability analysis (Reference 24), the severity of the earthquake is expressed by relating the ground acceleration to the acceleration of gravity as a percentage. The horizontal severity is 8% g and the vertical severity is taken as two-thirds of this, or 5.3% for the OBE. For the SSE these values are 15% g for the horizontal severity and 10% g for the vertical severity. During the earthquake, all parts of the mass of soil are assumed to be acted on by a steady vertical and horizontal force, equal to unit weight times acceleration, in addition to all other forces to which the slope is subjected. These forces act in the direction of instability. With these two forces determined, the analysis was completed similarly to the conventional static analysis.

The other analysis method used was the procedure proposed by N.M. Newmark (Reference 23). This is fully explained in the cited reference; therefore, it is not discussed here.

The factor of safety for the dynamic conditions is approximately equal for both methods of analysis. These design slopes have a factor of safety of 1.3 or more, which is acceptable (References 13 and 22).

Also, an extensive slope stability analysis of the intake channel and structure was undertaken with the assistance of a computer program. The factor of safety was computed using the Swedish slip circle method of analysis. Both static and the SSE dynamic conditions were analyzed. The intake channel slopes away from the intake structure at a gradual slope of approximately 10 horizontal to 1 vertical. The minimum factors of safety for the intake structure were computed to be 2.7 and 1.6 for the static and dynamic conditions, respectively. The minimum factors of safety for the intake channel side slopes, which are 5 horizontal to 1 vertical, were computed to be 6.5 for static conditions and 2.0 for dynamic conditions. The minimum factors of safety for the critical circle perpendicular to the anchored sheet pile section of the intake were computed to be 3.0 for the static conditions and 1.6 for the dynamic conditions.

During investigations done to support siting the Diesel Generator Building, the stability of the western slope of the site was evaluated to determine a factor of safety under both static and dynamic conditions. For the static condition, an analysis was made of both the total and effective stress cases. Actual conditions will fall somewhere in between the two. Since dynamic conditions are developed under undrained conditions, only the total stress case was evaluated for the dynamic conditions. The Simplified Bishop method of computing the factor of safety was used to perform the slope analysis. The results of the evaluation are summarized as follows:

Condition	Factor of Safety
Static, Total Stress	1.72
Static, Effective Stress	1.74
Dynamic, Total Stress	1.22

Results of an earlier analysis performed on the western slope yielded similar results. The results of both analyses demonstrate that an adequate factor of safety exists against mass failure of the western slope of the site under static and dynamic conditions.

Since the crib wall adjacent to the Diesel Generator Buildings was not seismically designed, it is possible that some localized failure may occur during a seismic event. The postulated worst case is a complete failure of the crib wall, which

would result in fill material sloughing against the wall of the Diesel Generator Building. The resulting wedge of fill material would reach a maximum height of 7.5' above grade. The static lateral loadings which result from this failure were found to be enveloped by the loading imposed by the design basis tornado. The west wall of the Safety-Related Diesel Generator Building is, therefore, designed for the fill loading under dynamic conditions.

2.7.6.2 Plant Foundations

The foundation elevations, original ground surface and amount of stress unloading by excavation are shown below:

Structure		Average Ground Elevation	Foundation Elevation	Average Excavation Unloading
North Containment Structure		+75' MSL	-1' MSL	8400 psf
South Containment Structure		60	-1	6600
Auxiliary Building	West End	70	-14	8300
	East End	70	-19	8850
North Turbine Building		60	-11	7300
South Turbine Building		40	-11	4900
Intake Structure		80	-30	10800
Discharge Structure		20	-27	4050

Generally, the weight of soil removed by site grading and pit excavation for the structures is greater than the loads imposed by plant construction. This verified the results of the analyses made using the triaxial shear data, i.e., that bearing capacity is no problem. The ultimate bearing capacity of the foundation strata is in excess of 80,000 psf. The allowable bearing capacity is in excess of 15,000 psf.

In addition to bearing capacity, settlement of the proposed structures was also considered. The settlement of the foundations can be divided into two categories: (1) elastic settlement; and (2) time-dependent or hydrodynamic settlement.

Elastic expansion of the confined soil occurred as a result of excavation unloading. This resulted in a slight upward movement. During construction, the soil moved downward as load was applied. This elastic movement is small and was complete when construction was completed. It had no effect on the structures or function of the plant. The excavation unloading and structural loading caused a small change in void ratio. This change allowed a very small amount of hydrodynamic settlement to occur. The time-dependent or hydrodynamic settlement will be very small or negligible because the structural load is either less than the overburden removed, or only slightly greater than the removed overburden weight. Considering the types of soils present, contact pressures of 1500 to 2000 psf greater than the overburden removed would not result in large consolidation settlements. The magnitude of maximum possible post-construction settlement is 1/2".

The excavation for the power plant structures was below the water table. Conventional dewatering was done to maintain a dry and stable condition during construction.

2.7.6.3 Liquefaction Potential

If a loose, saturated sandy soil (a soil with less than 10 to 15% silt and clay fines and less than 200 to 300 psf cohesion) is subjected to ground vibrations, as during an earthquake, it tends to compact and decrease in volume. If the soil cannot drain during the rapid load fluctuations imposed by an earthquake, there is a buildup in pore pressure until it is equal to the overburden pressure. The effective stress then becomes zero, the soil loses its strength, and develops a "quick" or liquefied condition. If this condition is of a general areal extent and the pressure not otherwise relieved, it can cause a flow or bearing capacity failure (References 14, 16, 17, 18, 19, 20, and 21).

For the evaluation of liquefaction potential at this site, data were used from the dynamic triaxial testing, standard penetration resistances from the borings, in-place density determinations and geologic origin of the sedimentary soils at the site. All of these data showed that the soil at the site was not of a liquefaction potential. The dynamic tests showed exceptional strength under constant cyclic stress.

Other characteristics also support that there is no liquefaction potential at this site. The amount of material passing the No. 200 sieve in the gradation analysis (the silt and clay fines) and the amount of cohesion observed in the static triaxial shear tests supports the conclusion that there is no potential to liquefy. This is concluded because the soil is not truly cohesionless or reasonably suited to be susceptible to the liquefaction phenomenon. The last significant indicator that a liquefaction potential does not exist is the geologic origin of the site soils. The areas where liquefaction is believed to have happened are areas of flacial outwash, recent alluvium, or loose artificial fills (Reference 15). The Calvert Cliffs site soils are preconsolidated deposits several million years old.

During the siting of the Diesel Generator Buildings, the liquefaction potential of various strata of the North Parking area were evaluated using the standard penetration test blow counts obtained during subsurface explorations. Factors of safety against liquefaction were computed for all standard test blow counts performed for the loose-to-medium dense granular sand strata (considered to be the most susceptible to liquefaction). These computed factors of safety ranged from 1.3 to 2.4 with a median value of 1.8. Previous analyses performed in 1981 indicated a minimum factor of safety of 1.37 against liquefaction based upon standard penetration test blows, and between 1.6 and 2.0 based upon laboratory cyclic triaxial tests data. The results indicated that the site of the Diesel Generator Buildings has an adequate factor of safety against liquefaction under design earthquake conditions.

2.7.6.4 Lateral Earth Pressure

The lateral earth pressures for the walls of this plant were evaluated for both the static and dynamic conditions.

The rigidity of the walls and the fact that backfilling was done after the walls were framed at the top do not allow sufficient movement for developing the active earth pressure case.

Therefore, the at-rest condition was developed. For convenience of design, the earth pressure has been converted to an equivalent fluid pressure method. This utilizes the Ranking approach which is a conservative estimate of lateral earth pressures. The earth pressure has been determined based on the characteristics

of the material stockpiled for use as backfill. This backfill was sand, silty sand, and gravely silty sand.

The equivalent fluid unit weight above the water table is 47 lbs/ft^3. Below the water table, the equivalent fluid unit weight is 85 lbs/ft^3. The pressure distribution will be hydrostatic.

The dynamic earth pressures were considered for this plant. The analysis was based on work by N. M. Newmark, Y. Ishii, et al., K. Terzaghi, and the Corps of Engineers (References 25, 26, 27, and 28, respectively). These references provided at-rest coefficients of earth pressure which are dependent on the magnitude of the earth shock acceleration. Based on this information, an increase of the equivalent fluid unit weight should be 10% and 17% for the OBE and SSE, respectively.

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2.8 CHESAPEAKE BAY STUDIES

The Calvert Cliffs site is located on the western shore of the Chesapeake Bay approximately at the mid-point of its 195 mile length. The Chesapeake Bay is the largest tidal estuary on the Atlantic Coast, roughly comparable in size to Lake Ontario. Its width ranges from 3 miles to 35 miles. Major tributaries include the Susquehanna, Patapsco, Choptank, Patuxent, Potomac, Rappahannock, York, and James Rivers.

The site is located on the western shore of the Bay, approximately 10 miles north of the Patuxent River, and 25 miles south of the Choptank River which enters from the eastern side of the Bay. At the site, the Bay is approximately 6 miles wide. Water depths up to 110' are found in the main channel of the Bay at the site, with depths of 6, 12, and 18' found at distances of 750, 1200, and 1500' offshore, respectively.

2.8.1 USES OF THE BAY

The Chesapeake Bay in the general vicinity of the site is utilized for fisheries' resources, navigation, and recreation. Shellfish (oysters and clams), crabs, and finfish are taken commercially from the area. A natural oyster bar of 680 acres had extended in front of the site. However, in 1969, BGE relocated approximately 500 acres of the bar to a location in the Patuxent River.

Navigational interests, both ocean-going and local Bay vessels, utilize the Chesapeake Bay extensively. Major shipping channels are situated some 3500' or more from the Calvert Cliffs site.

Recreational use of the Bay in the vicinity of the site is primarily boating and sport fishing. Water-contact recreation at the site location is negligible.

2.8.2 COMPREHENSIVE STUDY PROGRAM

The need to control and minimize adverse effects on the Chesapeake Bay from the construction and operation of the Calvert Cliffs Plant was recognized at the start of the project planning. In keeping with this objective and to assure conformance with the water quality standards of the State of Maryland, shortly after the site was purchased a team of research consultants was assembled to develop the information needed for guiding the design, construction and operation of the plant. The study program and its findings have been discussed frequently with appropriate State and Federal agencies during the course of the work.

The principal consultants were Sheppard T. Powell Associates, Academy of Natural Sciences of Philadelphia, Alden Research Laboratories of Worchester Polytechnic Institute, Dr. John C. Geyer of Johns Hopkins University and NUS Corporation of Rockville, MD. During studies concerning water quality in 1979-1981 Ecological Analysts, Inc. and J. E. Edinger Associates, Inc. were added to the team of consultants.

The effect of the condenser cooling water discharge on temperatures in the Chesapeake Bay was studied by the Alden Research Laboratories through operation of hydraulic models which simulated flows in a 34 mile stretch of the Bay and the areas within a few thousand feet from the plant site. The objective of the studies was to determine the optimum arrangement of the cooling water system to achieve rapid dispersion of effluents and minimize water temperature variations in the area of plant influence. Results of tests on the final design arrangement were reported in December 1969.

In order to provide a baseline for assessment of the effects of plant operation on the aquatic environment, the Academy of Natural Sciences of Philadelphia engaged in a seven-year program of compiling a comprehensive inventory of the population, species,

and condition of the aquatic life in the area, and detailed information on the physical, chemical, and bacteriological characteristics of the Chesapeake Bay near Calvert Cliffs. These studies have been completed for five years of operating conditions.

In accordance with the requirements of the Calvert Cliffs PSAR and the conditions of the Atomic Energy Commission (AEC) construction permits, BGE initiated in 1969 work on the design and development of the environmental radiological monitoring program for Calvert Cliffs. Concurrent with the design, development and operation of the monitoring program, BGE in contract with NUS initiated several studies to assess the potential radiological impact of expected radioeffluents from Calvert Cliffs.

A review of the results of these studies was made in conjunction with other known environmental data. The purpose of this review was to ascertain the significance of the various exposure pathways and to identify the "potential critical pathways" in the area of the facility. The results of this review and the mandatory compliance requirements, based on the AEC limitations on 1971 (proposed 10 CFR Part 50, Appendix I), determined the scope of the monitoring program described in Section 2.9.

In June 1974, BGE received discharge permits from both the Water Resources Administration of the State of Maryland and from the Environmental Protection Agency authorizing the discharge of heated condenser cooling water into the Chesapeake Bay. In June 1976, BGE received a National Pollution Discharge Elimination System (NPDES) permit which consolidated the previous two permits. The NPDES permit is periodically renewed on a schedule established by the Maryland Department of the Environment.

On July 31, 1974, the AEC issued a facility operating license authorizing the power operation of CCNPP in accordance with a set of Environmental Technical Specifications. The Non-Radiological section of the Environmental Technical Specification (Appendix B) was amended by the Nuclear Regulatory Commission (NRC) by deletion of four sections dealing with environmental monitoring on March 25, 1981. Some of the aquatic programs had been completed and the remainder are the responsibility of the NPDES program of the State of Maryland.

The above-mentioned documents require that the plant is to be operated in such a manner as to ensure compliance with the State of Maryland Water Quality Regulations, and also require that monitoring programs are conducted to determine the effects of the operation of the plant on the physical, chemical and biological characteristics of the Chesapeake Bay. In order to implement these requirements, the Academy of Natural Sciences of Philadelphia and Radiation Management Corporation of Philadelphia are continuing the programs initiated before the operation of the plant. Baltimore Gas and Electric Company has performed programs to study specialized areas, such as the impingement of organisms on the traveling screens and the passage of organisms through the condenser cooling water system. The results of these studies indicate the plant meets State mixing zone criteria, the entrainment does not impact a spawning or nursing area of consequence, and the impingement cannot be cost effectively mitigated any further. The PPSP has concurred with these findings.

2.8.3 HURRICANE TIDAL EFFECTS

2.8.3.1 Historic Storms and Tides

Historic accounts of early hurricanes affecting the Chesapeake Bay area date back to the 17th Century. Early chronologies of tidal flooding record extreme events which occurred in August 1667, October 1749, September 1769 and July 1788. In his report (Reference 1) on hurricane tides and tidal flooding in the Chesapeake Bay area, the District Engineer, Baltimore Corps of Engineers District, notes that U.S. Weather Bureau records show at least 80 tropical hurricanes or their

remnants have affected the bay area in the 75 year period since 1889. By far the most destructive hurricane in recent years to affect the Chesapeake Bay region was the hurricane of August 23, 1933. Other notable storms were hurricanes "Hazel" in October 1954, "Connie" and "Diane" in August 1955 (only 5 days apart), and "Donna" in September 1960. The "Great Atlantic Hurricane" of September 1944, which passed some 50 miles offshore of Chesapeake Bay, was also a storm of major size and intensity. As noted above, the relative frequency of hurricane occurrence for this area is slightly more than one hurricane per year. Numerous studies have been made of the more significant hurricanes to have affected the Middle Atlantic and New England States (References 2, 3, 4, 8, 9, and 11) in which their paths, intensities, forward speeds, resulting tides and other associated phenomenon have been well documented. In general, record hurricanes passing over or near Chesapeake Bay have had central pressures of from 27.8 to 28.5"; peak wind speeds over the ocean approaching 100 mph, and maximum winds over the bay area of up to 75 mph. Following recurvature in the middle latitudes the forward speed of these storms has ranged from 10 to 36 knots. Northeast storms also affect the Chesapeake Bay area, however, because of the general orientation of the bay the magnitude of tides reached in the Bay is not as great as those generated by record hurricanes. The northeast storm of March 6-8, 1962, which resulted in 4.9’ mean low water (MLW) tide in the lower Potomac River, was about the worst experienced along the Atlantic Coast.

2.8.3.2 Tides and Storm Surges

Normal Tides - Normal tides in the bay area are the semidiurnal type having two highs and lows roughly every 23-1/2 hours, with a higher high and lower low as a daily occurrence. Information contained in Reference 5 shows normal and spring tide ranges at various locations in Chesapeake Bay as follows:

	Mean Tidal Range (ft)	Spring Tidal Range (ft)
Kiptopeke Beach (Ocean)	2.7	3.2
Hampton Roads	2.5	3.0
Cape Charles Harbor (Bay)	2.4	2.8
Point Lookout	1.2	1.4
Cove Point	1.2	1.4
Taylors Island	1.3	1.5
Oxford	1.4	1.6

The mean and spring tide ranges to be expected at the site are 1.2 and 1.4’, respectively. The time occurrence of daily high and low tides within the Chesapeake Bay, related to time of occurrence at Hampton Roads and Baltimore, is also given in Reference 5. From the data given in Table 2 of that reference, it was ascertained that the travel time of high and low tide occurrence from the Bay entrance to the site area is approximately 5 hours.

Storm Surges and Extreme High Tides - Storm surges and extreme high tides have been recorded at numerous locations in Chesapeake Bay and in the various rivers entering the Bay. Plate 2 of Reference 1 shows peak tide elevations above MLW of 8.2’ at Solomons Island near the mouth of Patuxent River and 7.4’ at Point Lookout at the mouth of the Potomac River. Tide levels of 4.1’ and 5.1', respectively, were recorded at these locations in the October 1954 hurricane. In the August 23, 1933 hurricane, a peak tide of about 8.5' (MLW) occurred at Norfolk, VA. A generalized time-frequency curve was developed for the

Chesapeake Bay area in Reference 1 (Plate 3) utilizing observed hurricane surge elevations. A reproduction of that relation is shown on Figure 2.8-1, Generalized Time-Frequency Curve. The tide elevations noted above include the cumulative effects of tidal surge, pressure effect, local wind effect, wave effect (in open Bay areas) and the astronomical tidal component. The contribution of the latter can add as much as 3' to the total recorded hurricane tide height (Hampton Roads) if the peak ocean surge entering the Bay coincides with a peak spring tide condition. The time of translation of tides up the Bay, noted above, is also an important consideration in determining the coincidence of both ocean and Bay peak surge heights with normal and spring high and low tides. Analysis of observed hurricane tide hydrographs in References 2 and 3 shows this effect to be quite pronounced. For example, in the August 1949 hurricane, which passed to the west of Chesapeake Bay, below-normal tides were experienced at Hampton Roads and Norfolk. Similarly, in the June 1945 hurricane, which passed east of the bay, a peak tide of 3' was recorded at Hampton Roads, whereas a low tide of over a foot below normal was recorded at Baltimore.

2.8.3.3 Tide and Storm Surge Analysis

A comprehensive investigation of hurricane surge problems for the Chesapeake Bay area was made using parameters from Memorandum HUR 7-97, "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States" (Reference 13).

2.8.3.4 Probable Maximum Hurricane

Parameters describing the maximum probable hurricane were selected from Reference 13 at the approximate latitude of the Chesapeake Bay entrance (36.1°). Definition of each of those parameters for a maximum probable hurricane is given below.

Central Pressure (P_o) - Minimum central pressures in hurricanes passing over or near the Chesapeake Bay area have been as low as the 27.88" of mercury for the September 1944 hurricane. Except for hurricanes occurring within the last several decades, sufficient information on central pressures to establish a reliable pressure-frequency relationship for the area is not available. The standard project hurricane derived in Reference 6 as hurricane "B" had a pressure anomaly of 2.2" of mercury which would mean a central pressure of about 27.75" of mercury. A probable maximum hurricane (PMH) was derived in Reference 10 for the New York Bay area. The pressure and wind patterns constructed for that storm are shown on Figure 43 of that report. The minimum pressure of that hurricane, when opposite the entrance to Chesapeake Bay, is 27.02" of mercury. A central pressure of 26.94" was selected.

Asymptotic Pressure (P_n) - A value of 30.92" was taken from the envelope curve shown on Figure 6 of Reference 13 to represent the peripherical pressure of the PMH.

Radius of Maximum Winds (R) - A value of 30 statute miles was used for this parameter, being considered representative of severe storm occurrences in the general area. Its use results in a storm of reasonable size for transposition purposes. That value lies between moderate and large radius values recommended in Table 1 of Reference 13.

Forward Speed (T) - The value selected for forward speed is 23 mph, a moderate speed of translation. The forward speed of the storm affects not only the peak 30'-overwater wind speed, but also the height of peak ocean tide at or near shore and

the shape of the resulting hydrograph. In the case of Chesapeake Bay, the forward speed is especially important in that it is related to the development of surge elevation within the Bay, the speed of the free wave up the Bay, and the resulting surge height at the plant site. A very slow moving storm would permit the Bay surge to crest at the site before the maximum effect of crosswinds could reinforce and increase that height. A fast moving storm would result in the converse.

Maximum Winds at Radius R would be 124.7 mph; adding half the forward speed results in a peak isovel wind speed of 136.2 mph.

Path - The path selected for the PMH. is shown in Figure 2.8-2. It would approach the coast from the east, curving northward on passing inland west of Chesapeake Bay.

Parametric relationships describing the wind speed profile, pressure profile, pressure effect profile and basic wind data used in constructing the isovel pattern for the PMH were derived using a computer program developed and employed by personnel of the Jacksonville District Corps of Engineers, and run on a GE 415 Computer. The output of the program can be seen on Tables 2-41 and 2-42. Methods used to derive the PMH conform to those given in Reference 13. Graphical representation of the overwater wind profile, the pressure and pressure effect profiles can be seen on Figure 2.8-3. An isovel pattern was constructed for transposition purposes using data given in Table 2-42. That pattern is shown on Figure 2.8-4.

2.8.3.5 Tidal Surge Computations

General - Procedures used in the tidal surge analysis for the open ocean across the Continental Shelf are those described in the U.S. Army Coastal Engineering Research Center publication, "Shore Protection - Planning and Design," Technical Report No. 4 (Reference 14). Formula (1-65) shown on page 140 of that report was used for the computations. Peak tide at the Chesapeake Bay entrance will occur at time T_o. Basic offshore depths, fetch, wind speed data, and Cos c values, together with general map features from wind speed data, and Cos c values, together with general map features from C&GS Chart No. 1222 can be seen on Figure 2.8-5. Peak winds in the zone of maximum winds were oriented over the shallow Bay entrance channel area to obtain the maximum surge height. Based on the selected forward speed, the surge hydrograph at the coast would have about a 12- to 14-hour rise from slightly above normal tides to the peak surge at T_o. This is based on a comparison of storm features of the PMH with that of the August 1933 hurricane which affected the area. Data for that storm and its resulting tide can be found in Table 1 and Figure 2 of Reference 6.

Procedures - An offshore bottom profile along the fetch noted on Figure 2.8-5 is plotted on Figure 2.8-6. Average depths offshore along a fetch of about 88 statute miles range from 22' to 800'. The normal tide relations used are as follows:

a. At Hampton Roads:

   Mean Tide Range = 2.5' MLW
   Normal High Tide (NHT) = 1.2' + MLW

b. At Cove Point (near Plant Site):

   Mean Tide Range = 1.2' MLW
   NHT = 0.6' MLW

The basic assumptions made for the surge computations are as follows:

a. PMH surge at bay entrance coincident with NHT at entrance.

b. Beginning elevation of surge computations
E1 (Mile 105) = NHT + PE
E1 = 1.2' + 1.5' = 2.7' MLW

c. Five-mile fetch lengths used with miles shown on Tables 2-43 and 2-44 as mid-point.

d. Average depth values, $\bar{d}$, taken from offshore depth profile, Figure 2.8-6, at mid-points of 5-mile reaches.

e. Values of μ (wind speed) from isovel pattern at 5-mile fetch increments.

f. Surge computation procedures from Reference 14 (Formula 1-65, page 140).

g. Pressure effect values taken from PMH parameters, Figure 2.8-3.

h. PMH approach speed = 23 mph. Speed of free wave in Chesapeake Bay for 40 to 50' average bay depth = 24 to 27 mph.

$\left(v = \sqrt{gD}\right)$

i. Assume PMH forward speed overland increases slightly from 23 mph to speed of free wave in bay. Storm accompanies surge up the bay to plant site.

j. Distance from bay entrance to Calvert Cliffs plant site = 110 miles (statute). At 24-27 mph, the surge would travel from bay entrance to plant site in 4+ to 5 hours.

k. Normal high tide will occur at plant site 4+ to 5 hours after NHT in channel entrance. Coincident peak surge at NHT will thus occur at both locations, with 4+ to 5-hour time difference.

l. Overland reduction in PMH intensity - used factors given in Table 2a, Reference 13; at T+5 hours reduction factor = 80% μ max at site area = 136x.80 = 110 mph.

It was assumed that NHT at shore would occur coincident with the peak hurricane surge. Basic data along the fetch for time T_o are given in Table 2-43; those data were employed in the computational procedures and formula as shown on Table 2-44.

Results - The peak tidal surge elevation that would occur at the Bay entrance was computed to be 18.67' MLW (17.32' MSL). It should be noted that wave effect was not considered to be applicable. Water depths in the channel entrance to the Bay would be on the order of 40' (22' depth + 18' surge). That depth would sustain a 30-32' wave which would move into the bay area to break farther inland.

Chesapeake Bay Surge Analysis - A comprehensive analysis of the interrelationships involved in tidal surge movement up Chesapeake Bay was presented in Reference 6. In that report, it was shown that a reduction in ocean surge occurs in its passage into and up the Bay due to the comparative dimensions and hydraulic characteristics of the entrance channel and the various sections of the Bay between Hampton Roads and Baltimore. The results of that analysis were utilized for prediction purposes. The relationship shown on Figure 15 of that report relates the maximum surge on the open coast to that to be

expected in the southern portion of the Bay. The relation extends within the range of the computed PMH surge elevation. Using the mean prediction curve and the value of 18.67' MLW, the value of 13.2' ± 0.8' would be obtained for the surge Elevation of 14.0' MLW. Movement of the surge up the Bay to the plant site area will occur at approximately the speed of the free wave in the Bay (about 24 to 27 mph depending on depth changes) and at a speed coincident with the speed of the hurricane. The presence of large rivers with added storage volume was found in Reference 6 to result in a further minor reduction in surge height in its passage up the Bay. Table IV of that report indicates a factor of 0.96 times the surge elevation in the lower Bay will give the value of the surge elevation to be expected in the vicinity of the plant site. Using that factor gives a surge Elevation of 13.44' MLW (14.0x0.96). That elevation represents the height of the surge in the Bay as it moves northward past the plant site. To that value must be added the additional effect of hurricane winds blowing from east to west across the Bay, and the effect of coincident occurrence of NHT at the site, plus any wave effect.

<u>Surge Elevation at Plant Site</u> - Movement of the PMH inland and overland will result in a reduction in intensity and wind speed. Table 2 of Reference 12 lists the reduction factors to be applied with respect to travel time overland. At T_{+5} hours, a factor of 80% is considered applicable. Wind directions slightly ahead of the zone of maximum winds will be oriented generally east-to-west over the Bay in the vicinity of the plant site at the time the peak surge reaches that area. An evaluation of wind speed and direction was made for that condition. Wind speeds of 115 to 120 mph (117 mph average) were found to be applicable for the wind direction and fetch conditions shown on Figure 2.8-7. An effective crosswind of 94 mph (117x0.80) was, therefore, used to compute the additional height of Bay setup. An average bottom profile shown on Figure 2.8-8 was constructed using data from Figure 2.8-7. The total fetch length is approximately 10 Statute miles. Preliminary estimates indicated the node line along the fetch would be at about Mile 4.0 east of the western Bay shore. A summary of setup computations is given in Table 2-45. The computed setup elevation in the vicinity of the plant site was determined to be 15.21' MLW. A value of 1' was added to that elevation for estimated wave effect, giving a total peak surge elevation at the site of 16.21' MLW (15.6' MSL).

<u>Wave Analysis</u> - The significant wave height that can be expected to occur in the vicinity of the plant site during the PMH peak surge will be a function of wind speed, water depth, and length of available fetch (Reference 7). Evaluation of average water depth with fetch length in the Bay offshore indicates a 50'-depth for about 7 miles; a 40'-depth for about 9 miles. Using a wind speed of 94 mph and Figures 1-40 and 1-42 of Reference 14, results in a significant wave height of 11.4'; with a corresponding wave period of 9 seconds (Figure 1-43). The wave will break in approximately 14-1/2' of water. The height of the wave above still-water level would be 6.8' (11.4x0.6). Added to the peak surge Elevation of 16.2', the elevation of the top of that wave, <u>unbroken</u>, would be 23.0' MLW.

<u>Wave Run-up</u> - The maximum wave run-up elevation at the intake structure was previously calculated to be 28.1' MLW or 27.5' MSL. A series of scale model tests was performed at the University of Florida with an adverse slope on the top of the pump room wall. Six tests were performed to represent still-water levels of 16.2', 17.2' and 18.2', each with and without the baffle wall. Results (Figure 2.8-9), indicate that there will be no overtopping of the intake structure.

The calculated run-up elevations on the slopes north and south of the intake structure are well below the plant grade of Elevation 45'.

2.8.3.6 Wave Run-up at Intake Structure

Introduction - The saltwater cooling pumps, which are essential for safe shutdown of the CCNPP, are housed in the intake structure. Since these are Class I components, it was decided to design the enclosure as a Category I structure for seismic, tornado, and hurricane conditions. The hurricane surge calculations in Section 2.8.3.5 are based on implementation of References 13 and 14. Using a number of conservative assumptions and the results of a series of model wave run-up tests at the University of Florida, it is concluded that the structural integrity of the intake structure will be maintained under PMH conditions. Thus, the saltwater cooling pumps can continue to operate under PMH conditions.

Configuration of the Intake Structure - The intake structure has an open deck at Elevation 10.0' MSL on the Bay side. The deck is about 50' wide and has openings for the trash rakes and racks, stop logs, and traveling screens.

Behind this open deck is an enclosure housing the circulating water pumps and saltwater cooling pumps. The roof of the pump room is at Elevation 28.5' MSL and has watertight hatches to provide access to the pumps for maintenance. An intake structure air supply unit is mounted on each saltwater pump hatch, and an air exhaust vent is mounted on each circulating water pump hatch. To minimize entry of moisture into the pump room, each air supply unit and air exhaust vent housing is provided with louvers designed for high moisture separation efficiency. Similarly, a separate ventilation system draws outside air through the service building (west wall) to minimize water entry. The personnel door located at the north end of the intake structure is of a watertight design.

Model Tests - The configuration of the intake structure is not one of the classic profiles which has been tested in the past. That is, it is not a curved or vertical wall, a stepped wall, or a riprap-covered wall like those shown in Figures 3-6 through 3-10 in Reference 14. However, T. E. Haeussner prepared a report predicting a maximum run-up Elevation of 28.1' MSL based on the curves of Figure 3-6, with an appropriate reduction factor for the case of waves breaking on the front edge of the intake structure.

There were questions on this approach and it was decided to perform two-dimensional model tests at the University of Florida to determine the run-up elevation. These tests were run with a prototype wave period of 9.0 seconds and wave height of 11.4', the significant wave height. Both of these values appear in Section 2.8.3.5. Three still-water Elevations were tested: 16.2', 17.2', and 18.2' MSL. The lowest of these, 16.2' MSL, was 0.6' higher than the value calculated in Section 2.8.3.5, due to confusion between MSL and MLW. The higher still-water elevations were to account for some differences of opinion as to the appropriately-conservative value.

During the tests, considerable overtopping resulted in the early runs, so the adverse slope (upper 5' of pump room sloping out at 20% angle) was added. This adverse slope appeared to eliminate the overtopping. Even with the still-water at 18.2' MSL, an analysis of the test runs indicates that the run-up carries only to Elevation 26.5' MSL, a run-up of 8.3'. Figure 2.8-9 shows the output of the final test runs. A color film was available for viewing. Table 2-46 is a summary of the model test results compared with run-up computed from Figure 3-6 in Reference 14. The highest model/computed ratio is 0.50.

Predicted Wave Run-up - There have been questions as to the conservatism of basing wave run-up on the run-up from the significant wave. Therefore, the results

of the University of Florida tests were used as a predictive tool along with the wave height corresponding to H1, the average of the highest 1% of waves. This is the wave height recommended for design of rigid structures in Reference 14. In Reference 15, C.L. Bretschneider found the expression

$$\frac{H_{max}}{H_s} = \left(145 \frac{gd}{U^2}\right)^{0.1}$$

where:

H_{max}	=	maximum wave height
H_s	=	significant wave height
g	=	acceleration of gravity, 32.2 ft/sec^2
d	=	depth, in ft
U	=	wind velocity, in ft/sec (138 for Calvert Cliffs)

Solving this expression gives H_{max} = 1.27 H_s for 40' depth, and H_{max} = 1.30 H_s for 50' depth. Fitting these values to a normal curve approximation yields H_1 - 1.23 H_s for 50' 1.23 (11.4') = 14.0. Thus, the design wave for the intake structure is 14.0' high.

As explained previously, the still-water levels tested at the University of Florida were 0.6' high. Using a still-water Elevation of 17.6' MSL, which was calculated by the AEC's consultant, using the 14.0' wave with a period of 9 seconds and using the curves of Reference 14, Figure 3-6 with a reduction factor of 0.50 (the highest model/computed ratio in Table 2-46), the calculated wave run-up is to Elevation 27.1' MSL, 1.4' below the pump house roof.

As a comparison with the approach, the following check was made, again based on model test results:

Run-up with SWL of 17.2' MSL	=	7.3'
Run-up with SWL of 18.2' MSL	=	8.3'
Interpolating, run-up for 17.6' MSL	=	7.7'
Multiplying by H_1/H_s (1.23)	=	9.5' for 14.0' wave
Add still-water level	=	17.6'
Run-up for 14.0' wave	=	27.1' MSL

Analysis was also made of the same conditions with periods of 5 and 12 seconds and resulted in somewhat less run-up. There was a question of model scale effects. The predicted run-up of 9.5' came to 1.4' below the pump house roof, so that a run-up of 10.9' would not overtop the structure. (10.9/9.5) = 1.15. Thus, the structure allows for model scale effects of 15%. This is a greater increase in run-up than predicted from use of the method of effective slope outlined in Reference 14.

Structural Analysis of the Intake Structure - The intake structure has been analyzed for seismic and tornado loadings. In addition, although it is not expected that the pump house portion of the structure will see breaking or broken waves, the intake structure has been analyzed for the following hurricane loading conditions:

a. Nonbreaking wave for water to the top of the roof, Elevation 28.5' MSL; still-water level of 17.6' MSL; and wave periods of 5, 9, and 12 seconds.

b. Broken wave for a wave height of 14.0'; still-water level of 17.6' MSL; and wave periods of 5, 9, and 12 seconds.

c. Breaking wave for the highest wave which could continue unbroken across the front edge of the structure. Since the still-water level is 17.6' MSL and

the deck Elevation is 10.0' MSL, the controlling depth is 7.6'. Thus, the maximum wave height is 0.78x(7.6) or 5.9'. This condition also has been examined for periods of 5, 9, and 12 seconds.

For each of the above loading conditions, the analysis shows that structural integrity will be maintained.

Additional Considerations - Curbs a minimum of 6" high are provided around the roof hatches. The roof and hatch covers are designed for a live loading of 250 psf. The louvered housings for intake structure air supply units and exhaust vents are designed for a live load of 100 psf.

The baffle wall in the intake channel is designed for conditions less than PMH (85 mph wind with a 9-second period, 10.5' high wave).

However, the Florida tests indicate that there would be no overtopping of the intake structure whether or not the baffle wall is in place during the PMH. An analysis showed that even if sections of the wall came loose, they would not damage or block the intake structure. In addition, the saltwater cooling pumps are redundant (two out of three required for each unit) and each can take suction from either of two screen wells.

The concrete stop logs were stored in a recess to the south of the pump house. Thus, they were not considered to be a missile for the intake structure design.

Scour at the front edge of the structure is not expected since there is a very low velocity past this point (about 1 ft/sec), and since the foundation soil is a dense silty sand or sandy silt.

There will be no resonant vibration of the structure due to the waves. The structure's natural frequency is about 3 CPS.

Conclusions - The following conclusions are drawn from the studies performed:

a. The predicted wave run-up is to Elevation 27.1' MSL, 1.4' below the pump room roof elevation.

b. The structural integrity of the intake structure will be maintained during hurricane conditions.

c. The saltwater cooling pumps can continue to operate under hurricane conditions.

2.8.3.7 Extreme Low Tide Considerations

Normal Tides and Tidal Datum - Normal tides in Chesapeake Bay are semidiurnal, or of the mixed type with two highs and two lows occurring daily; and with a higher high and lower low tide level as a daily occurrence. Information available in Reference 5 for Taylors Island, located across the Bay from the site, gives a mean range of 1.3', a spring range of 1.5', and a mean tide level of 0.6'. The latter value indicates that sea level is 0.6' above low water datum. The extreme annual range is approximately 2.3', occurring in December.

Low Tide Considerations - Various factors affect and, to a large extent, control the value of extreme low tide elevation at the site. Essentially, they are as follows:

a. Hurricane wind direction, duration and intensity in storms passing offshore of Chesapeake Bay. Counterclockwise rotating winds from the northeast to

northwest over the Bay will prevail for such a storm path and have the greatest effect in lowering Bay tide levels.

b. The location of the plant site with respect to the total length of the Bay.

c. Depth of water in the Bay, especially along available west-east tide fetches for generation of maximum set-down at shore.

d. General orientation of the Bay, length of the Bay and degree of curvature of the longitudinal axis corresponding to wind streamline curvature of hurricane winds over the Bay.

The general orientation of the Chesapeake Bay is north-south, its length from Baltimore to Norfolk is about 165 miles. The site is about two-thirds of that distance from Norfolk. Records of hurricane tides, shown on path and tide maps in References 2, 3, and 5, indicate that for hurricanes passing offshore of the Bay, maximum drawdown in the Bay occurs at the windward end, i.e., Baltimore, becoming less with distance southward toward the mouth of the Bay. For example, in Hurricane Donna of September 1960 (Figure 3, page 188 of Reference 5), the drawdown at Baltimore was to -2.6' MLW; to -1.5' at Annapolis; to -1.0' at Solomons; and to -0.8' at Portsmouth. Analyses of other hurricanes with similar wind conditions over the Bay, such as the October 1944, September 1947, and October 1954 hurricanes, show much the same variation in tidal drawdown within the Bay. Because of its location, Calvert Cliffs does not experience the maximum effects of Bay drawdown, as is indicted by observed data. The maximum drawdown elevation presumably occurred in Hurricane Donna, and is estimated to have been about -1.2' MLW (1.7' MSL). The extreme low tide elevation occurrence believed possible at Calvert Cliffs is predicated on an occurrence of the PMH on a path offshore of Chesapeake Bay, similar to that of Hurricane Donna. Correlating wind intensity over the Bay of the PMH with that of Donna and the drawdown elevations experienced in Donna, an estimated drawdown Elevation of -3.0' could be expected to occur at Baltimore with a value of -1.6' at the site. With a counterclockwise wind shift from north to west over the Bay, an additional setdown of about 0.5' could be expected to occur, giving an extreme low tide Elevation of -2.1' MLW (-2.7' MSL) at the site.

The predicted extreme low tide Elevation is -3.6' MSL (-3.0' MLW). However, the plant has been designed for -4.0' MSL and can continue to operate with an extreme low water Elevation of -6.0' MSL. The top of the saltwater pump intakes is at Elevation -9.5' MSL.

2.8.4 REFERENCES

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4. Inter-Agency Committee Report, The Resources of the New England - New York Region, Part Two, Chapter XXXIX, Special Subjects Regional, Hurricanes, Volume 4, December 1954

5. Tide Tables, 1967 East Coast, North and South American, U.S. Department of Commerce, ESSA, Coast and Geodetic Survey

6. Miscellaneous Paper No. 3-59, Hurricane Surge Predication for Chesapeake Bay, BEB, September 1959
7. Technical Memorandum No. 83, Reid, R. D, Approximate Response of Water Level on a Sloping Shelf to a Wind Fetch Which Moves Toward Shore, BEB, June 1956
8. N.H.R.P. Report No. 50, part II, Proceedings of the Second Technical Conference on Hurricanes, U.S. Weather Bureau, March 1962 (pp 341-354)
9. N.H.R.P. Report No. 14, B. I. Miller, On the Maximum Intensity of Hurricanes, U.S. Weather Bureau, December 1957
10. Technical Memorandum No. 120, The Predication of Hurricane Storm-Tides in New York Bay, BEB, August 1960
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15. Technical Memorandum No. 46, Field Investigations of Wave Energy Loss in Shallow Water Ocean Waves, BEB, September 1954

TABLE 2-41

BASIC INFORMATION FOR CONSTRUCTING PMH ISOVEL PATTERN

DISTANCE FROM CENTER	OVERWATER WIND PROFILE	PRESSURE EFFECT	PRESSURE
7.5	42.05	4.45	27.01
15.0	81.16	3.92	27.48
22.5	111.92	3.34	27.99
30.0	124.71	2.87	28.40
40.0	110.14	2.39	28.82
50.0	96.07	2.05	29.12
60.0	87.81	1.79	29.35
70.0	81.24	1.58	29.53
80.0	74.47	1.42	29.68
90.0	68.90	1.29	29.79
110.0	62.11	1.08	29.97
130.0	56.48	0.94	30.10
150.0	51.50	0.82	30.20
170.0	47.03	0.73	30.28
190.0	42.96	0.66	30.34
210.0	39.22	0.60	30.39
230.0	35.74	0.55	30.43
250.0	32.48	0.51	30.47
R	T	P_o	P_n
30.00	23.00	26.94	30.92

TABLE 2-42

ANGLES MEASURED FROM LINE OF FORWARD MOTION (USED FOR CONSTRUCTING ISOVEL PATTERN)

DIST.	25	55	85	115	145	175	205	235	265	295	325	355
7.5	42.0	47.8	52.0	53.5	52.0	47.8	42.0	36.3	32.1	30.5	32.1	36.3
15.0	81.2	86.9	91.1	92.7	91.1	86.9	81.2	75.4	71.2	69.7	71.2	75.4
22.5	111.9	117.7	121.9	123.4	121.9	117.7	111.9	106.2	102.0	100.4	102.0	106.2
30.0	124.7	130.5	134.7	136.2	134.7	130.5	124.7	119.0	114.7	113.2	114.7	119.0
40.0	110.1	115.9	120.1	121.6	120.1	115.9	110.1	104.4	100.2	98.6	100.2	104.4
50.0	96.1	101.8	106.0	107.6	106.0	101.8	96.1	90.3	86.1	84.6	86.1	90.3
60.0	87.8	93.6	97.8	99.3	97.8	93.6	87.8	82.1	77.8	76.3	77.8	82.1
70.0	81.2	87.0	91.2	92.7	91.2	87.0	81.2	75.5	71.3	69.7	71.3	75.5
80.0	74.5	80.2	84.4	86.0	84.4	80.2	74.5	68.7	64.5	63.0	64.5	68.7
90.0	68.9	74.7	78.9	80.4	78.9	74.7	68.9	63.2	58.9	57.4	58.9	63.2
110.0	62.1	67.9	72.1	73.6	72.1	67.9	62.1	56.4	52.2	50.6	52.2	56.4
130.0	56.5	62.2	66.4	68.0	66.4	62.2	56.5	50.7	46.5	45.0	46.5	50.7
150.0	51.5	57.2	61.5	63.0	61.5	57.2	51.5	45.7	41.5	40.0	41.5	45.7
170.0	47.0	52.8	57.0	58.5	57.0	52.8	47.0	41.3	37.1	35.5	37.1	41.3
190.0	43.0	48.7	52.9	54.5	52.9	48.7	43.0	37.2	33.0	31.5	33.0	37.2
210.0	39.2	45.0	49.2	50.7	49.2	45.0	39.2	33.5	29.3	27.7	29.3	33.5
230.0	35.7	41.5	45.7	47.2	45.7	41.5	35.7	30.0	25.8	24.2	25.8	30.0
250.0	32.5	38.2	42.4	44.0	42.4	38.2	32.5	26.7	22.5	21.0	22.5	26.7

TABLE 2-43

BASIC DATA - PMH - FETCH

FETCH MILE	μ (mph)	Cos c VALUE	μ Cos c $= \mu_x$	$\overline{\mu\mu_{\bar{x}}}$ $(mph)^2$	DIST. FOR P_e (miles)	P_e (ft)	ΔP_e (ft)	DEPTH @X (ft MLW)	
X = 0	116	0.42	49	5,684	39	2.42	-.10	0	Hampton Roads
5	122	0.52	63	7,690	37	2.52	-.13	19	
10	127	0.60	76	9,650	34	2.65	-.10	23	
15	131	0.70	92	12,040	32	2.75	-.11	25	
17.5									Bay Entrance
20	133	0.80	106	14,100	30	2.86	0	24	
25	136	0.90	122	16,600	30	2.86	0	32	
30	136	0.93	127	17,300	30	2.86	0	46	
35	136	0.97	132	17,950	30	2.86	0	60	
40	136	0.98	133	18,100	30	2.86	0.11	62	
45	135	0.95	128	17,300	32	2.75	0.10	65	
50	130	0.93	121	15,700	34	2.65	0.13	70	
55	126	0.88	111	14,000	37	2.52	0.10	76	
60	120	0.85	102	12,230	39	2.42	0.14	82	
65	115	0.80	92	10,620	43	2.28	0.11	90	
70	110	0.75	83	9,130	46	2.17	0.12	99	
75	105	0.70	73	7,660	50	2.05	0.12	100	
80	100	0.65	65	6,500	54	1.93	0.08	106	
85	97	0.60	58	5,630	57	1.85	0.05	112	
90	94	0.56	53	4,980	60	1.80	0.13	130	
95	90	0.52	47	4,230	65	1.67	0.09	180	
100	87	0.50	44	3,828	70	1.58	0.07	300	
105	84	0.45	38	3,200	74	1.51	-	800	-Deep Water

TABLE 2-44

PMH SURGE COMPUTATION - OCEAN TO BAY ENTRANCE

E_1 = 2.7

χ (Mid Point)	$\bar{d}$ (ft)	ΔPe (ΔS_1) (ft)	$\bar{d} + S_1$ (ft)	$\bar{d}_T$ (ft)	$\overline{\mu\mu_{\bar{x}}}$ 2 (mph)	ΔS_i (ft)	S_2 (ft)	E_2 (ft MLW)
105	800	0.07	800.+	803	32,000	0.024	0.024	2.724’
100	300	0.07	300.+	303	3,828	0.077	0.10	2.80’
95	180	0.09	180.+	183	4,230	0.128	0.23	2.93’
90	130	0.13	130.+	133	4,980	0.227	0.46	3.16’
85	112	0.05	112.+	116	5,630	0.296	0.76	3.40’
80	106	0.08	106.+	110	6,500	0.36	1.12	3.82’
75	100	0.12	100.+	104	7,660	0.45	1.57	4.27’
70	99	0.12	99.+	104	9,130	0.53	2.10	4.80’
65	90	0.11	90.+	95	10,620	0.68	2.78	5.48’
60	82	0.14	83	89	12,230	0.83	3.61	6.31’
55	76	0.10	77	84	14,000	1.01	4.62	7.32’
50	70	0.13	71	79	15,700	1.21	5.83	8.53’
45	65	0.10	66	75+	17,300	1.39	7.22	9.92’
40	62	0.11	62.+	75	18,100	1.46	8.68	11.38’
35	60	0	60.+	74	17,950	1.47+	10.15	12.85’
30	46	0	46.+	61	17,300	1.72	11.87	14.57’
25	32	0	32.+	48	16,600	2.10	13.97	16.67’
20	24	0	24.+	43	14,100	2.00	15.97	18.67’

S_0 (Max.) - Surge Elevation at Bay Entrance (Mile 17.5) - 18.67' MLW or 17.32' MSL.

TABLE 2-45

BAY SETUP COMPUTATIONS AT PLANT SITE

Used Parametric Relationship (Corps of Eng. - Jax. Dist)

Based on Formula: $S = \frac{F_{\lambda}\tau_s}{\lambda D}$ (N factor not included)

SETUP PORTION E_1 = 13.44' + 0.6' = 14.04'

17

Fetch F miles	υav mph	Dav ft	W/T Slope(1) ft/M1.	Setup S ft	$\sum_s$ ft	$D+\frac{\sum S}{2}$ ft	Setup S(2) ft	$\sum_s$ ft	E_2 ft MLW
2.70	94	56.4	0.20	+0.54	0.54	56.7	0.51	0.51	14.55'
0.30	94	40.4	0.20	0.06	0.60	40.7	0.07	0.58	14.62'
0.65	94	28.4	0.38	0.25	0.85	28.8	0.24	0.82	14.86'
0.35	94	19.4	0.54	0.19	1.04	19.9	0.19	1.01	15.05'
0.15	94	8.4	1.40	0.21	1.25	9.0	0.16	1.17	15.21'
4.15									

Surge in bay + crosswind effect	=	15.21'
Added wave effect	=	1.00' (est.)
Total Tide	=	16.21' MLW (15.6' MSL)

TABLE 2-46

WAVE RUN-UP AT INTAKE STRUCTURE

Comparison of Model and Computer Wave Run-up

	STILL-WATER ELEVATION (MSL)	COMPUTED RUN-UP (T.R. 4, Figure 3-6)	MODEL RUN-UP	MODEL/ COMPUTED RATIO
16.2'	14.8'	6.7'	0.45	
	17.2'	16.2'	7.3'	0.45
	18.2'	16.7'	8.3'	0.50

2.9 ENVIRONMENTAL RADIATION MONITORING

2.9.1 GENERAL

The objectives of the radiological environmental monitoring program at CCNPP are to:

- Measure actual radiation exposure to the general population at the fence line and beyond.
- Observe any sudden or unexpected rise in radiation levels in the vicinity of the plant.
- Document for legal and regulatory purposes actual radiation exposure levels and radionuclide concentrations in air, Bay surface water, sediment, fish, invertebrates, and vegetation.
- Provide monitoring services in emergency situations.

In order to fulfill these objectives, the radiological environmental monitoring program must differentiate between naturally-occurring and artificially-introduced radioactivity in the environment, and between plant-related and unrelated radioactivity. The radiological monitoring program is carried out in two phases: preoperational and operational.

2.9.2 PREOPERATIONAL RADIATION MONITORING

In accordance with the requirements of the Calvert Cliffs Safety Analysis Report and the conditions of the construction permits, BGE initiated work in 1969 on the design and development of the radiological environmental monitoring program for Calvert Cliffs. Concurrent with the design and development of the monitoring program, BGE, in contract with NUS, initiated several studies to assess the potential dose impact of expected radio-effluents from Calvert Cliffs. These studies addressed the following topics:

- Build-up of radionuclides in the aquatic environment;
- Relative biological significance of radionuclides;
- Estimate of potential dose to a maximum exposed individual via seafood ingestion;
- Estimate of potential immersion dose from noble gases, and thyroid inhalation dose to a hypothetical individual at the site boundary;
- Estimate of potential adult-thyroid and child-thyroid dose via the air-pasture-cow-milk pathway; and,
- Potential dose to population within 50-mile radius.

A review of the results of these studies was made in conjunction with other environmental data. The purpose of this review was to ascertain the significance of the various exposure pathways, and to identify the "potential critical pathways" in the area of the facility. The results of this review and the mandatory requirements based on the regulatory limitations on dose, radiation/radioactivity levels as published on June 7, 1971 (10 CFR Part 50, Appendix I) determined the design of the monitoring program.

The preoperational phase provided both seasonal and annual information about the distribution of natural radioactivity in the region, defined the ambient gamma-radiation levels, and obtained baseline data for some of the more important radionuclides, both natural and man-made.

2.9.3 OPERATIONAL RADIATION MONITORING

With the issuance of the operating license for Calvert Cliffs Unit 1 on August 1, 1974, BGE began the operational phase of the monitoring program.

Between 1974 and 1985, the program was carried out based on the environmental monitoring network designed in the preoperational phase. On February 22, 1985, the

NRC issued the Technical Specifications associated with the environmental monitoring program to assure the compliance with the provisions of 10 CFR Part 20, 10 CFR Part 50, 40 CFR Part 190, and NUREG-0472. The new operational program started on March 1, 1985, as per Table 2-47 and Figure 2.9-1.

In its present form, the radiological environmental monitoring program requires sufficient sample locations, types of samples, and analytical sensitivities which, in conjunction with the preoperational and background data, permit verification of the effectiveness of station radio-effluent control. The program provides data on changes in use of unrestricted areas and meets quality assurance criteria. The results of the program provide an indication of a measurable change, if any, in radiation and radioactivity in the environment, and provide reasonable assurance that the releases are within the limits specified in the Offsite Dose Calculation Manual for plant operation. The program is periodically reviewed to determine any changes that may be warranted in its content.

TABLE 2-47

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION	23 routine monitoring stations (DR1-DR23) either with two or more dosimeters, or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY (DR1-DR9)[(1)]; an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR10-DR18); the remaining stations (DR19-DR23) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 area to serve as a control station	At least quarterly	Gamma dose at least quarterly
2. AIRBORNE Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 SITE BOUNDARY locations, in different sectors of the highest calculated annual average ground-level D/Q[(1)] 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	Radioiodine Canister I-131 analysis weekly Particulate Sampler: Gross beta radioactivity analysis following filter change; Gamma isotopic analysis of composite (by location) quarterly

TABLE 2-47

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. WATERBORNE			
a. Surface	1 sample at intake area (Wa1) 1 sample at discharge area (Wa2)	Composite sample over 1-month period	Gamma isotopic analysis monthly Composite for tritium analysis quarterly
b. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wb1)	Semi-annually	Gamma isotopic analysis semi-annually
4. INGESTION			
a. Fish and Invertebrates	3 samples of commercially and/or recreationally important species (2 fish species and 1 invertebrate species) in vicinity of plant discharge area (Ia1-Ia3) 3 samples of same species in areas not influenced by plant discharge (Ia4-Ia6, Patuxent River)	Sample in season, or semi-annually if they are not seasonal	Gamma isotopic analysis on edible portions
b. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near the site boundary at 2 different locations of highest predicted annual average groundlevel D/Q (Ib1-Ib6)[(1)]	Monthly during growing season	Gamma isotopic and I-131 analysis
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction (Ib7-Ib9)	Monthly during growing season	Gamma isotopic and I-131 analysis

(1) Exception to these locations is in the South Sector where DR7, A1, Ib4, Ib5 and Ib6 are located approximately 0.7 km from the release point. This location is conservative with respect to the site boundary, which is located approximately 2.1 km from the release point.

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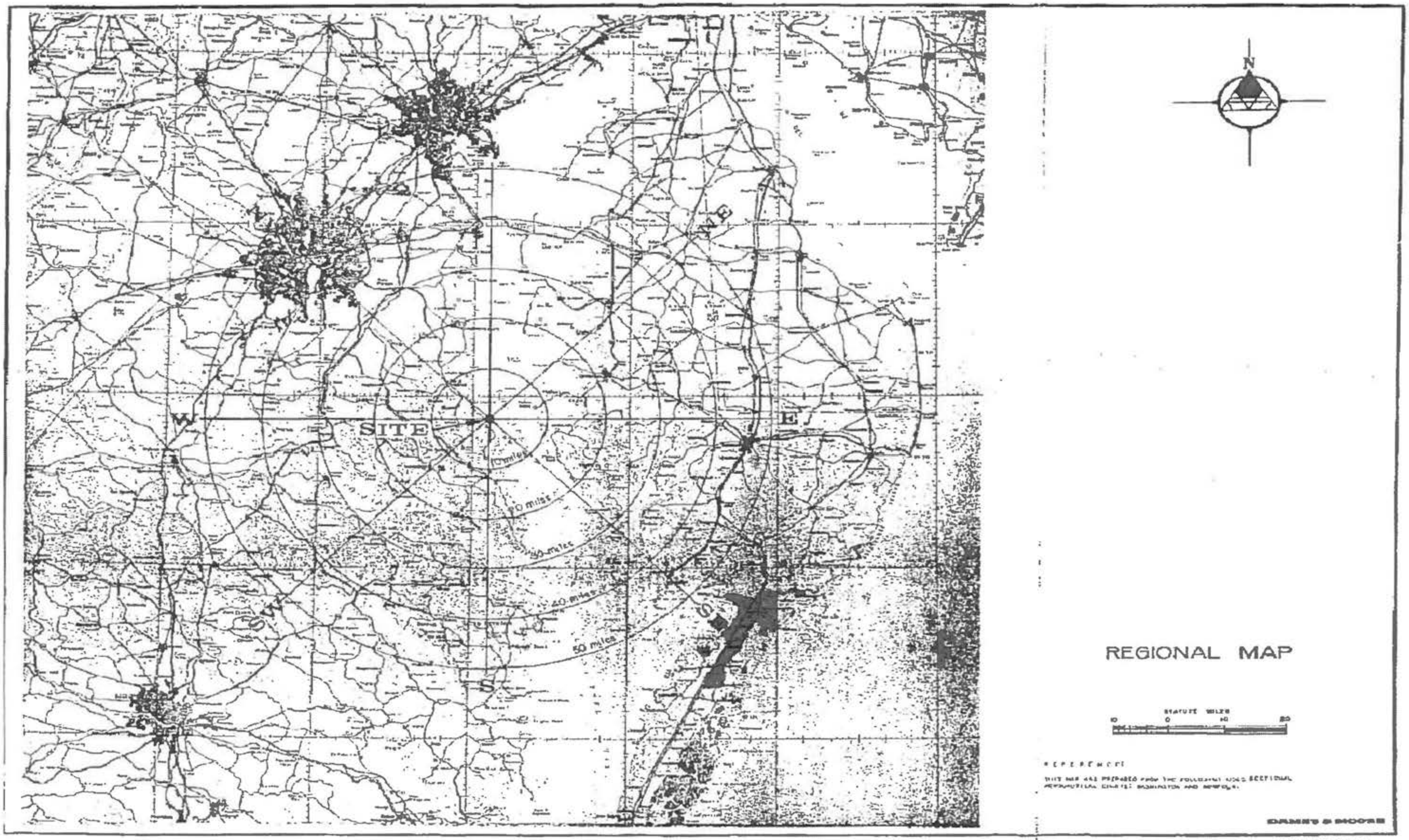
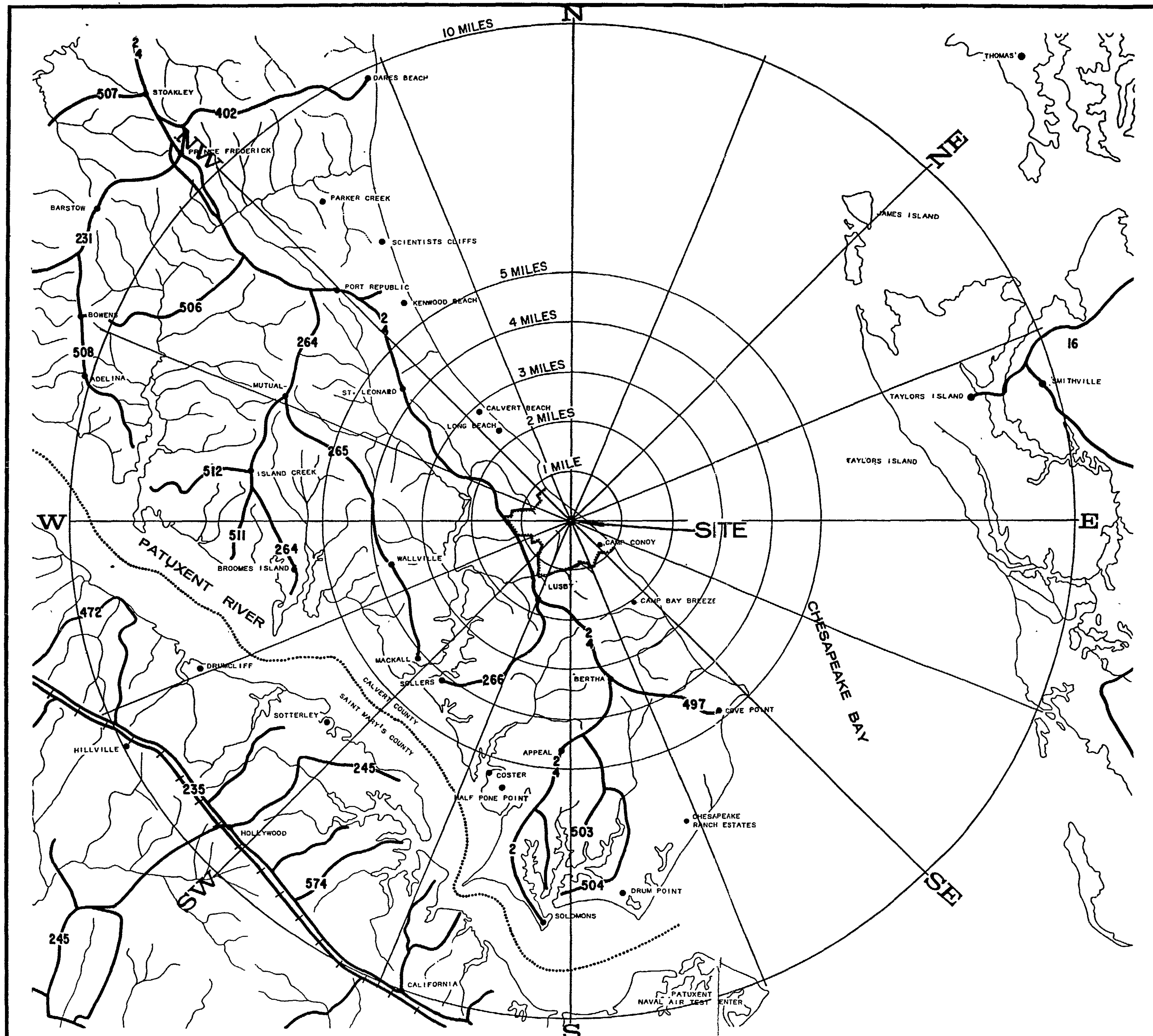

SITE
N
E
S
W
NE
SE
10 miles
20 miles
30 miles
40 miles
50 miles
REGIONAL MAP
STATUTE MILES
10
0
10
20

FIGURE 2.2-1



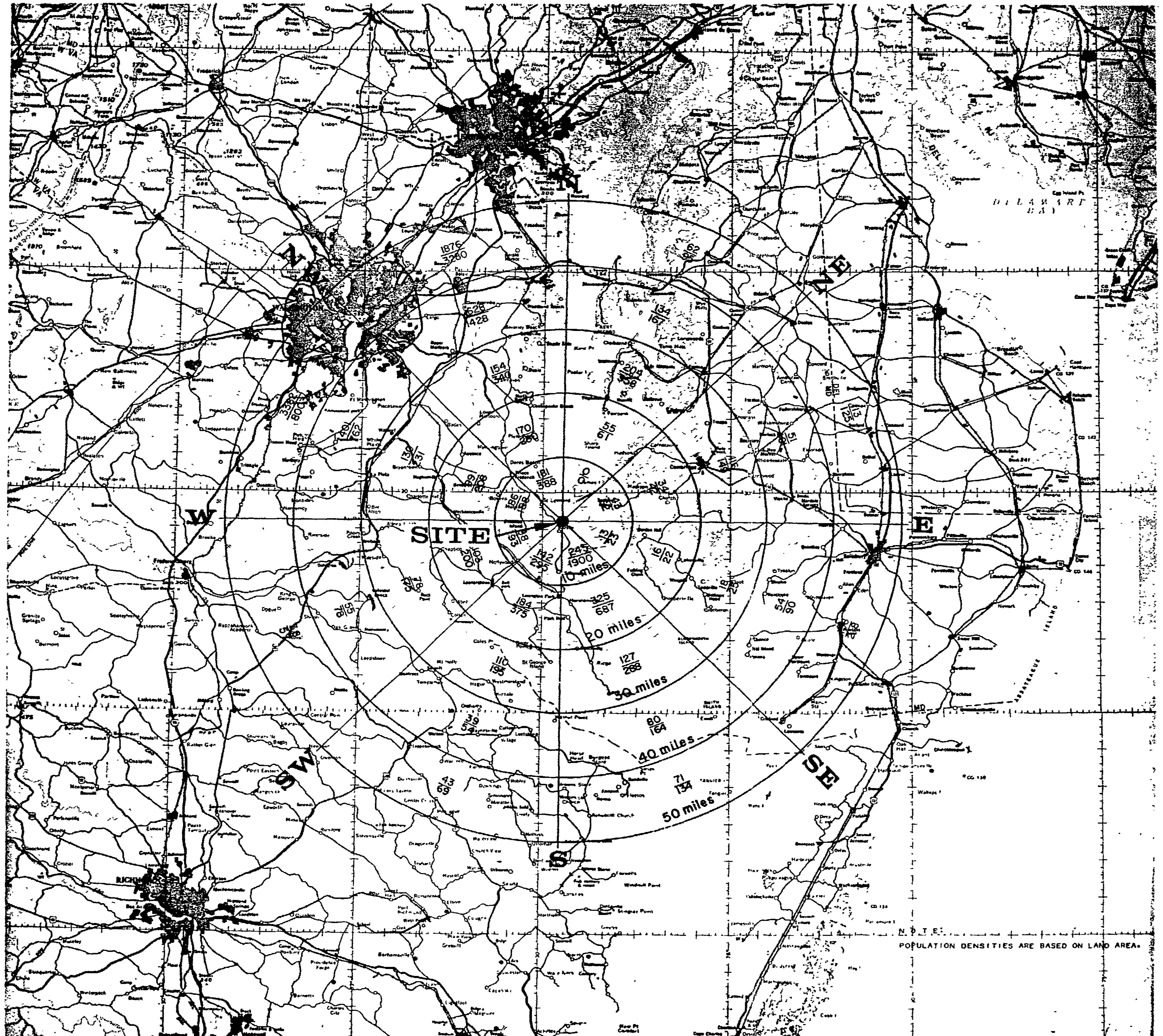
SITE VICINITY MAP

STATUTE MILES
1 0 1 2

REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF USGS WASHINGTON, D.C.-MARYLAND AND VIRGINIA 1957 TOPOGRAPHIC MAP.

DAMES & MOORE

FIGURE 2.2-2



KEY

52 / 78 = ESTIMATED 1970 POPULATION PER SQUARE MILE / ESTIMATED 2010 POPULATION PER SQUARE MILE

REGIONAL MAP

SHOWING PRESENT AND FUTURE POPULATION DENSITY 0-50 MILES

STATUTE MILES

10 0 10 20

REFERENCE:

THIS MAP WAS PREPARED FROM THE FOLLOWING USGS SECTIONAL AERONAUTICAL CHARTS: WASHINGTON AND NORFOLK.

REVISION 5

DAMES & MOORE

FIGURE 2.2-3

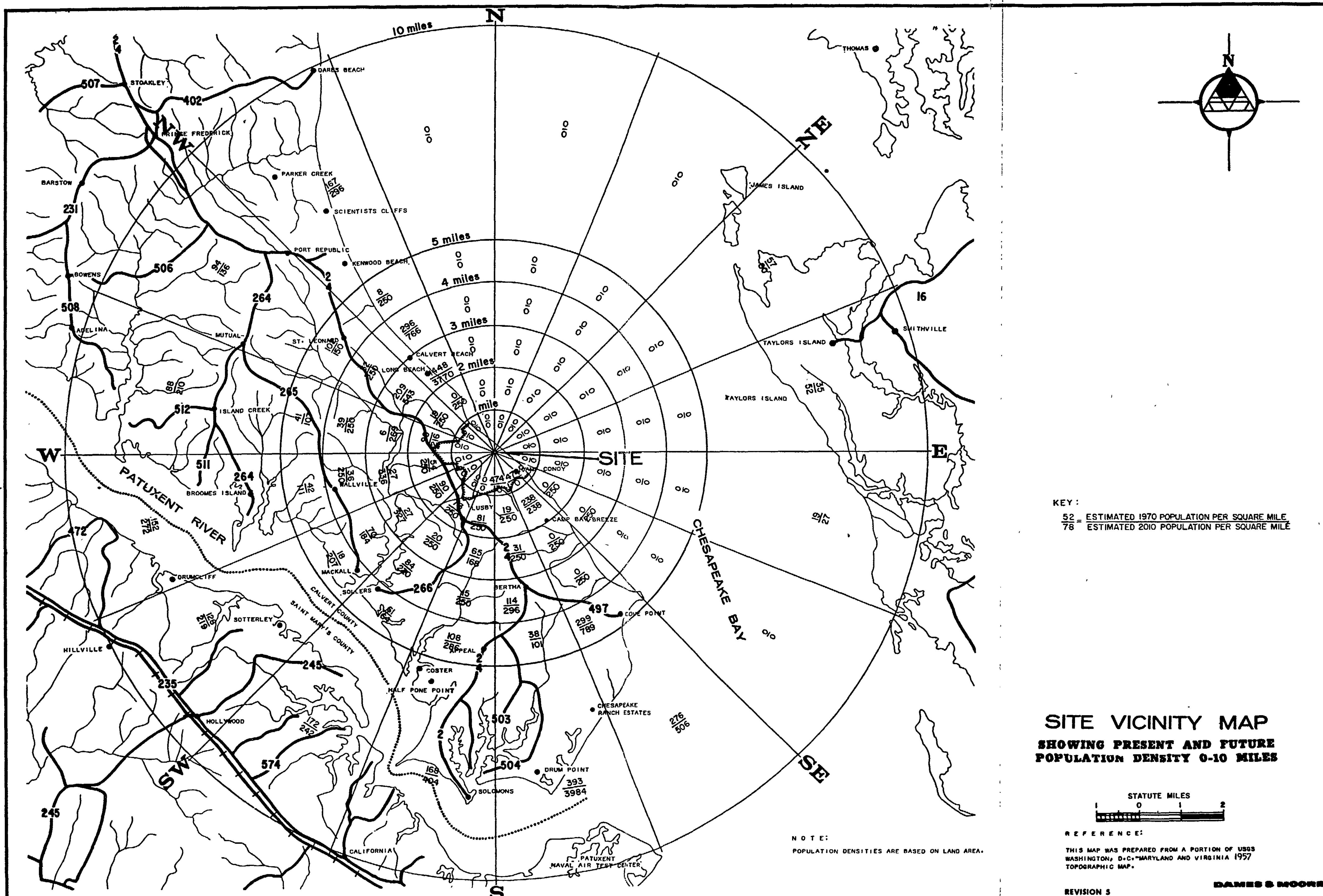
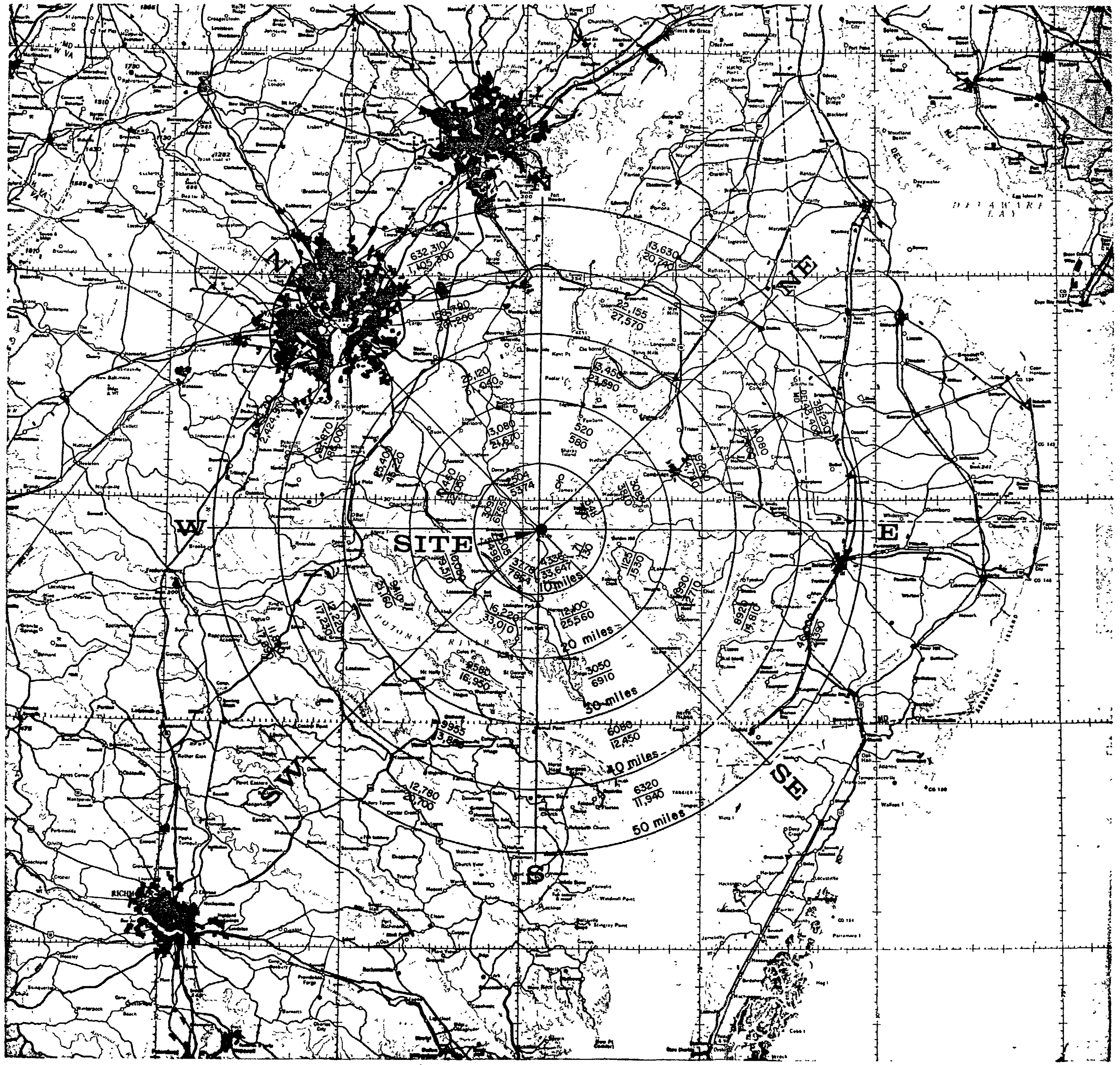

KEY:
52/78 = ESTIMATED 1970 POPULATION PER SQUARE MILE / ESTIMATED 2010 POPULATION PER SQUARE MILE
SITE VICINITY MAP
SHOWING PRESENT AND FUTURE POPULATION DENSITY 0-10 MILES
STATUTE MILES
NOTE: POPULATION DENSITIES ARE BASED ON LAND AREA.
REFERENCE: THIS MAP WAS PREPARED FROM A PORTION OF USGS WASHINGTON, D.C.-MARYLAND AND VIRGINIA 1957 TOPOGRAPHIC MAP.
DAMES & MOORE
REVISION 5
CHESAPEAKE BAY
PATUXENT RIVER
SITE

FIGURE 2.2-4



KEY:
125 = ESTIMATED 1970 POPULATION
261 = ESTIMATED 2010 POPULATION

NOTE:

RADIUS IN MILES		10	20	30	40	50
ACCUMULATIVE TOTAL	1970	16,827	83,495	188,755	518,825	2,305,635
	2010	60,048	187,768	380,128	1,041,048	4,758,108

REGIONAL MAP

SHOWING PRESENT AND FUTURE POPULATION DISTRIBUTION 0-50 MILES

STATUTE MILES
10 0 10 20

REFERENCE:

THIS MAP WAS PREPARED FROM THE FOLLOWING USGS SECTIONAL AERONAUTICAL CHARTS: WASHINGTON AND NORFOLK.

REVISION 5

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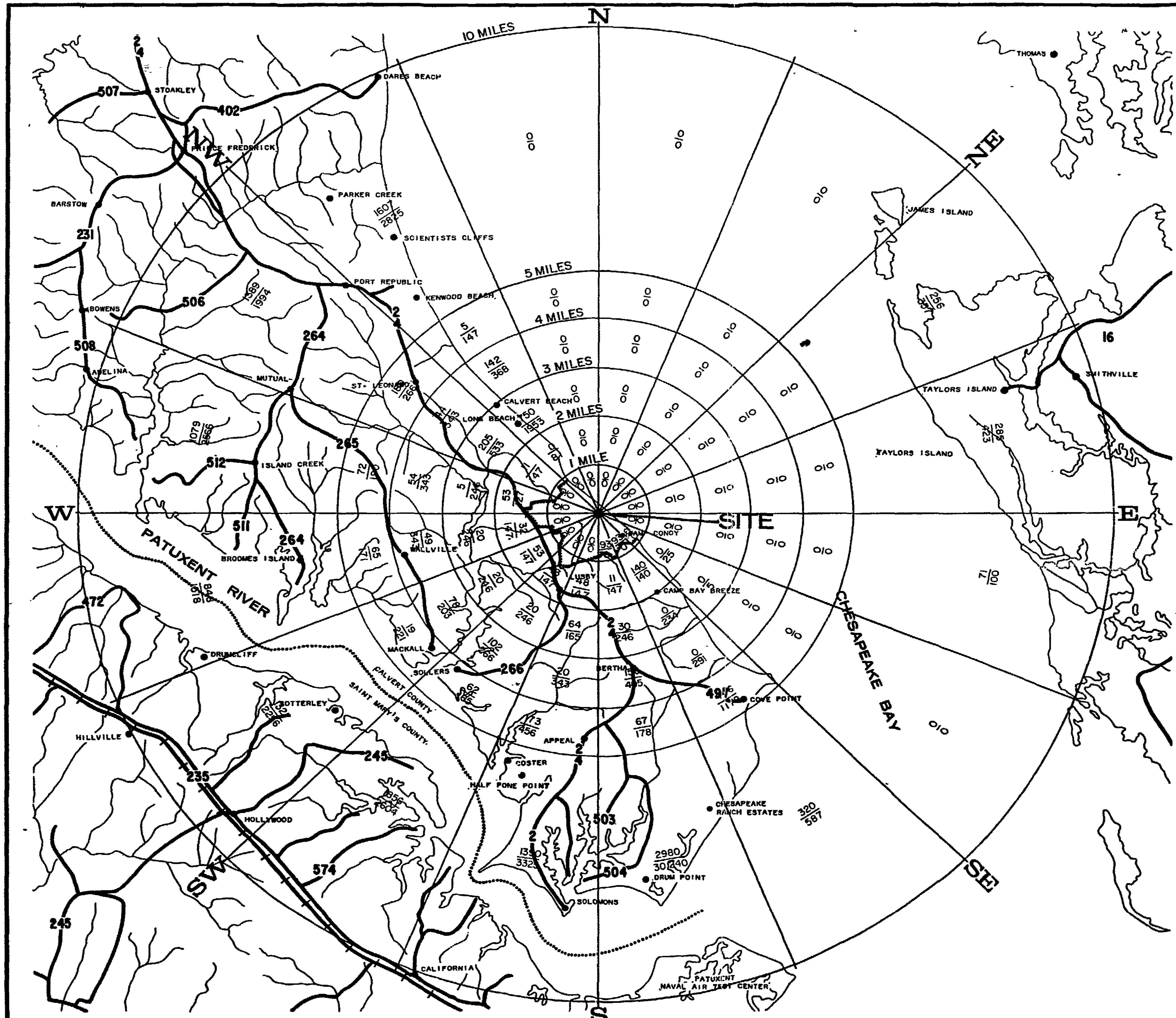
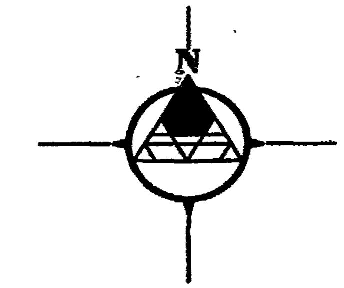
FIGURE 2.2-5





KEY:

210 / 580 = ESTIMATED 1970 POPULATION / ESTIMATED 2010 POPULATION

NOTE:

RADIUS IN MILES		1	2	3	4	5	10
ACCUMULATIVE TOTAL	1970	186	577	1691	2336	3425	16,827
	2010	0	1255	5375	8280	11,253	60,048

SITE VICINITY MAP

SHOWING PRESENT AND FUTURE POPULATION DISTRIBUTION 0-10 MILES

STATUTE MILES

1 0 1 2

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS WASHINGTON, D.C.-MARYLAND AND VIRGINIA 1957 TOPOGRAPHIC MAP.

REVISION 5

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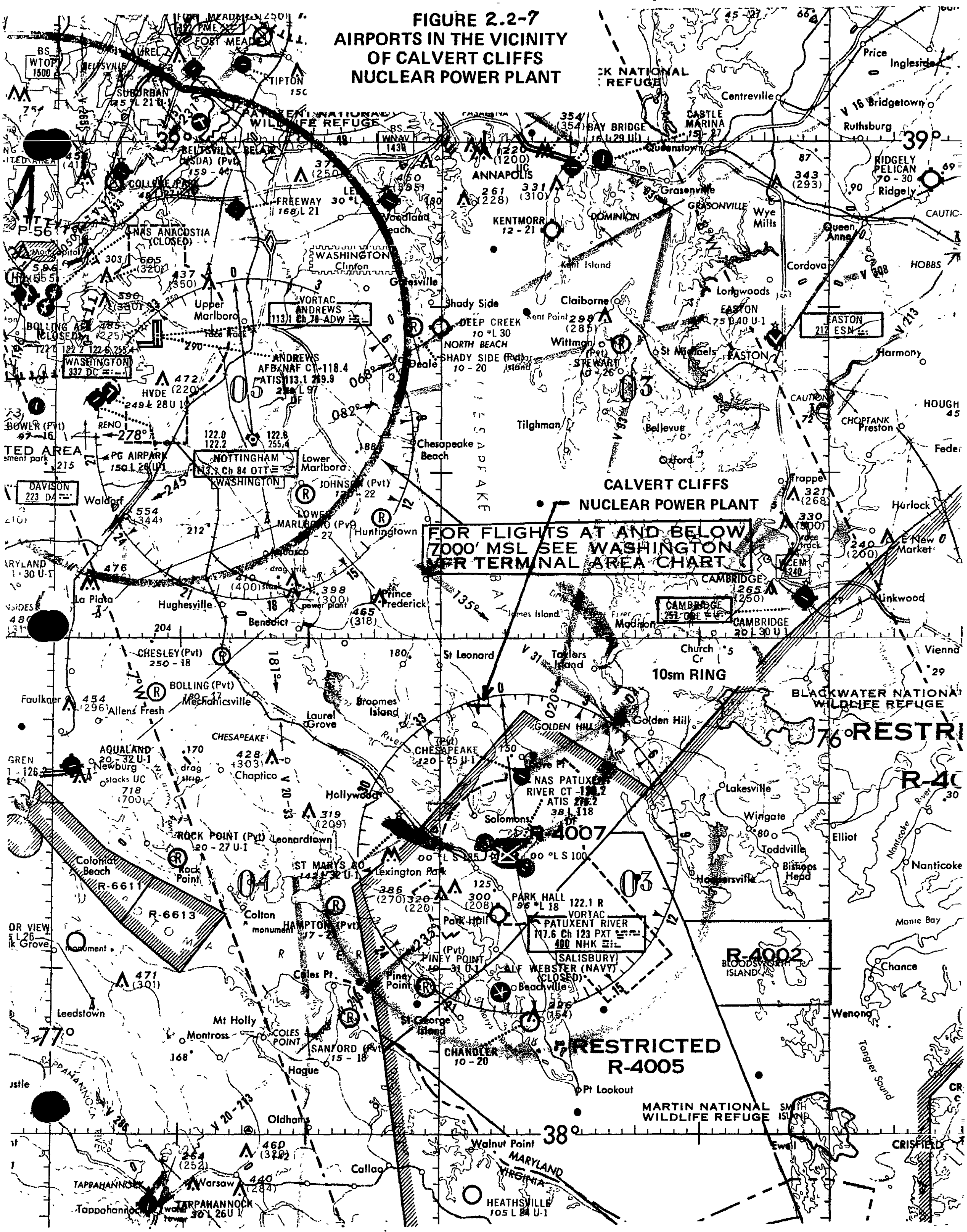
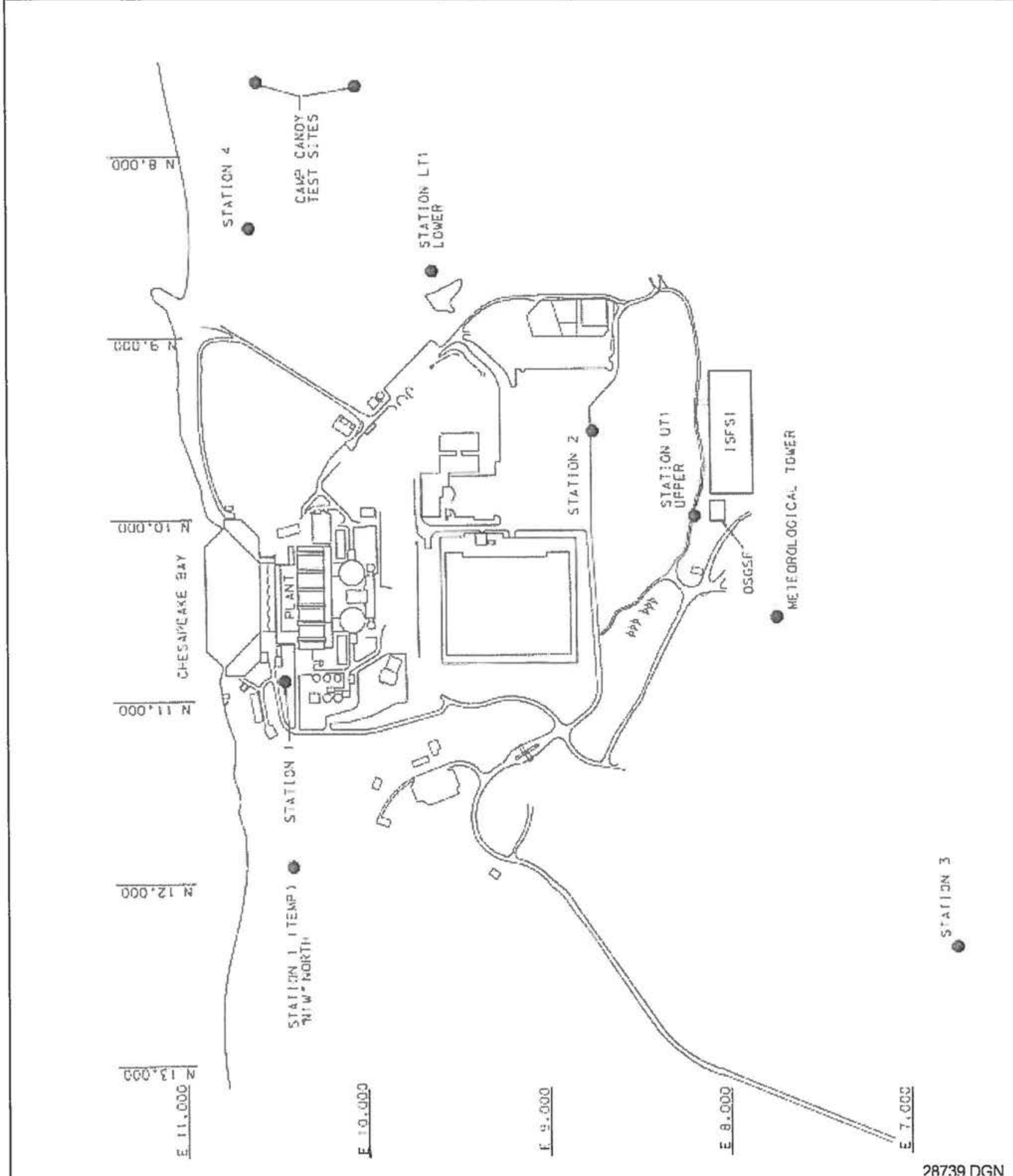
FIGURE 2.2-6

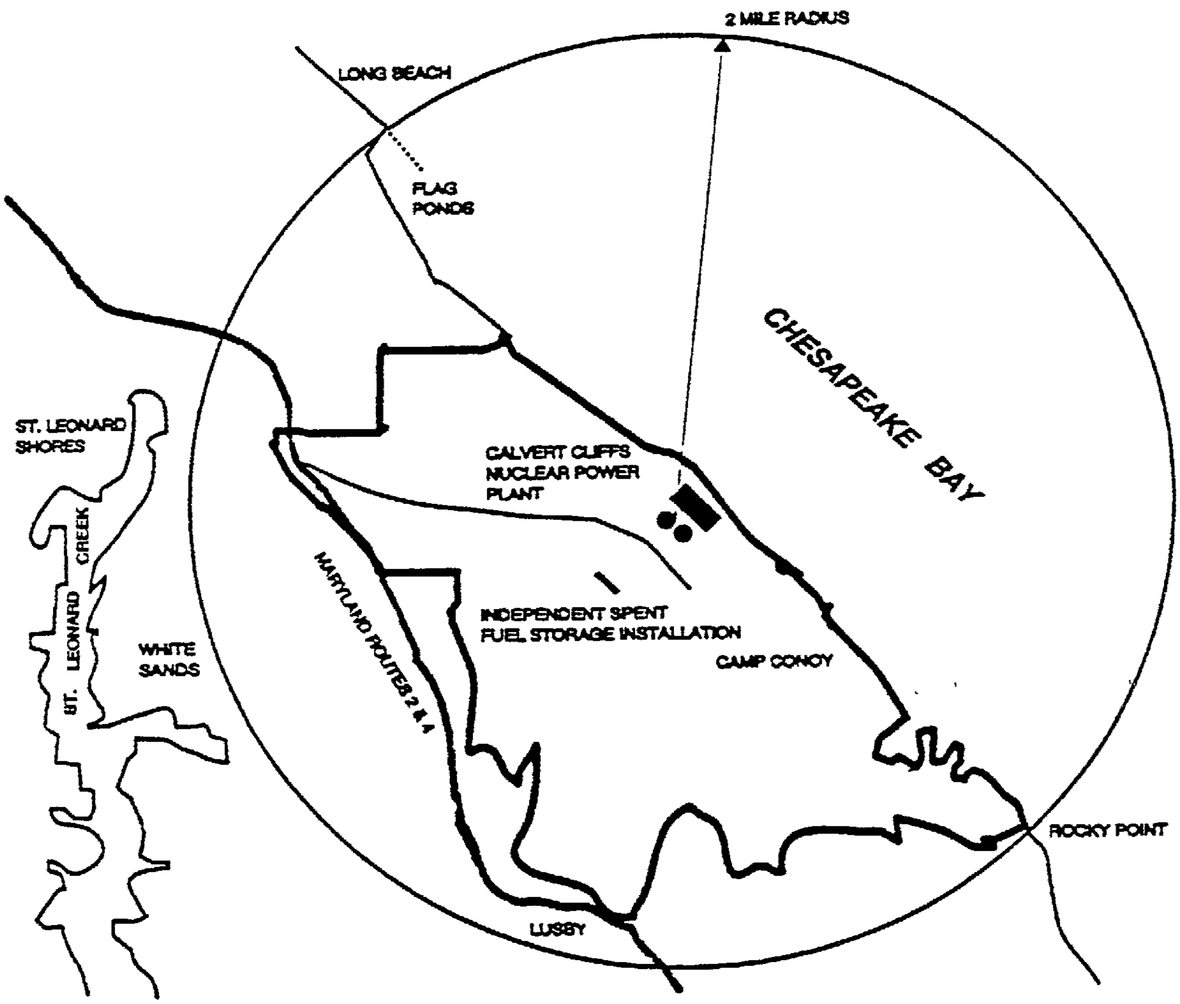
2.2-7 AIRPORTS IN THE VICINITY OF CALVERT CLIFFS NUCLEAR POWER PLANT





Calvert Cliffs Nuclear Power Plant	METEOROLOGICAL INSTRUMENTATION LOCATIONS	Figure 2.3-1 Revision 32

2.2-13 LOW POPULATION ZONE



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOW POPULATION ZONE	Figure 2.2-13 Rev. 18

FIGURE 2.3-2
METEOROLOGICAL INSTRUMENTATION LOCATION MAP

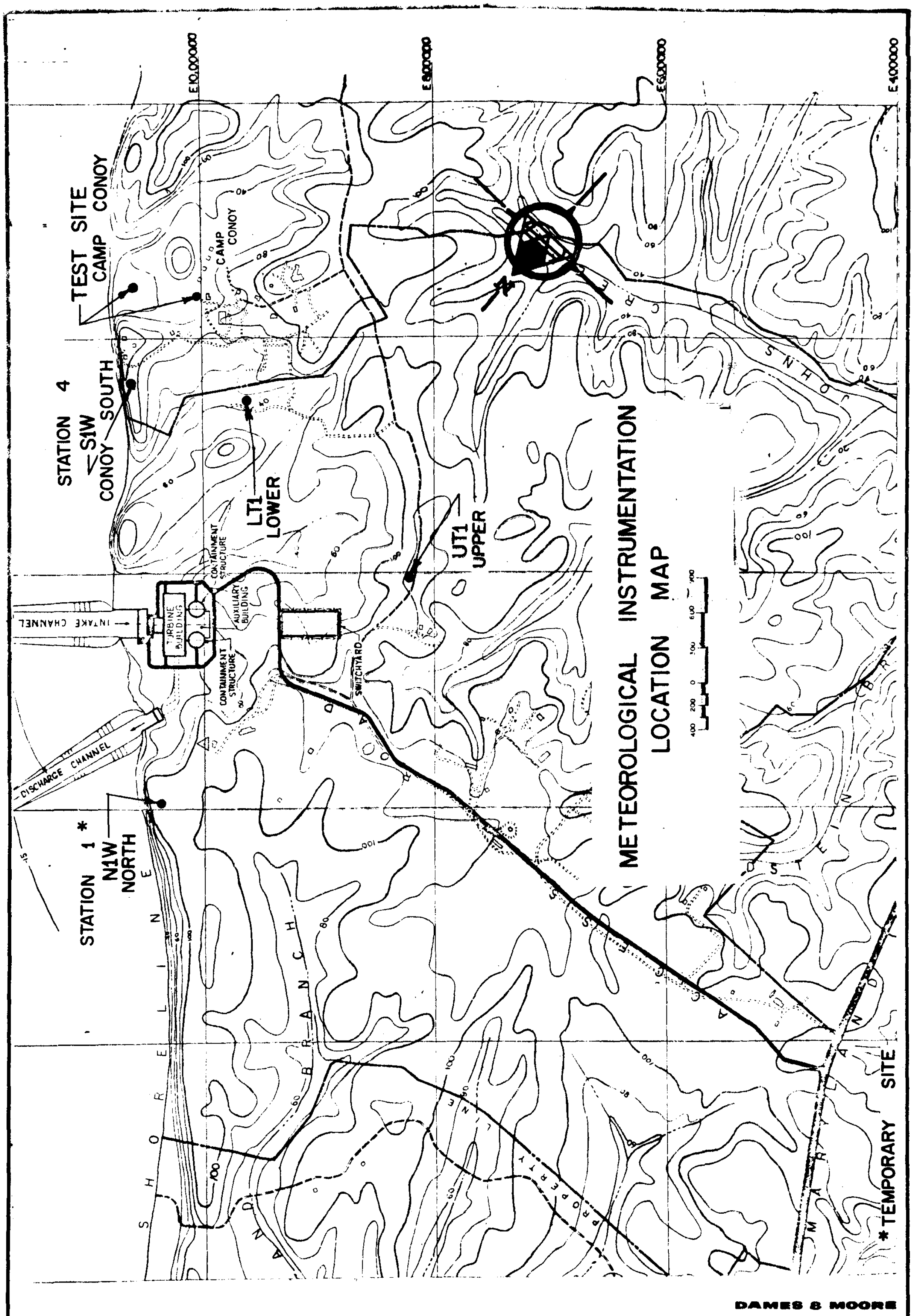
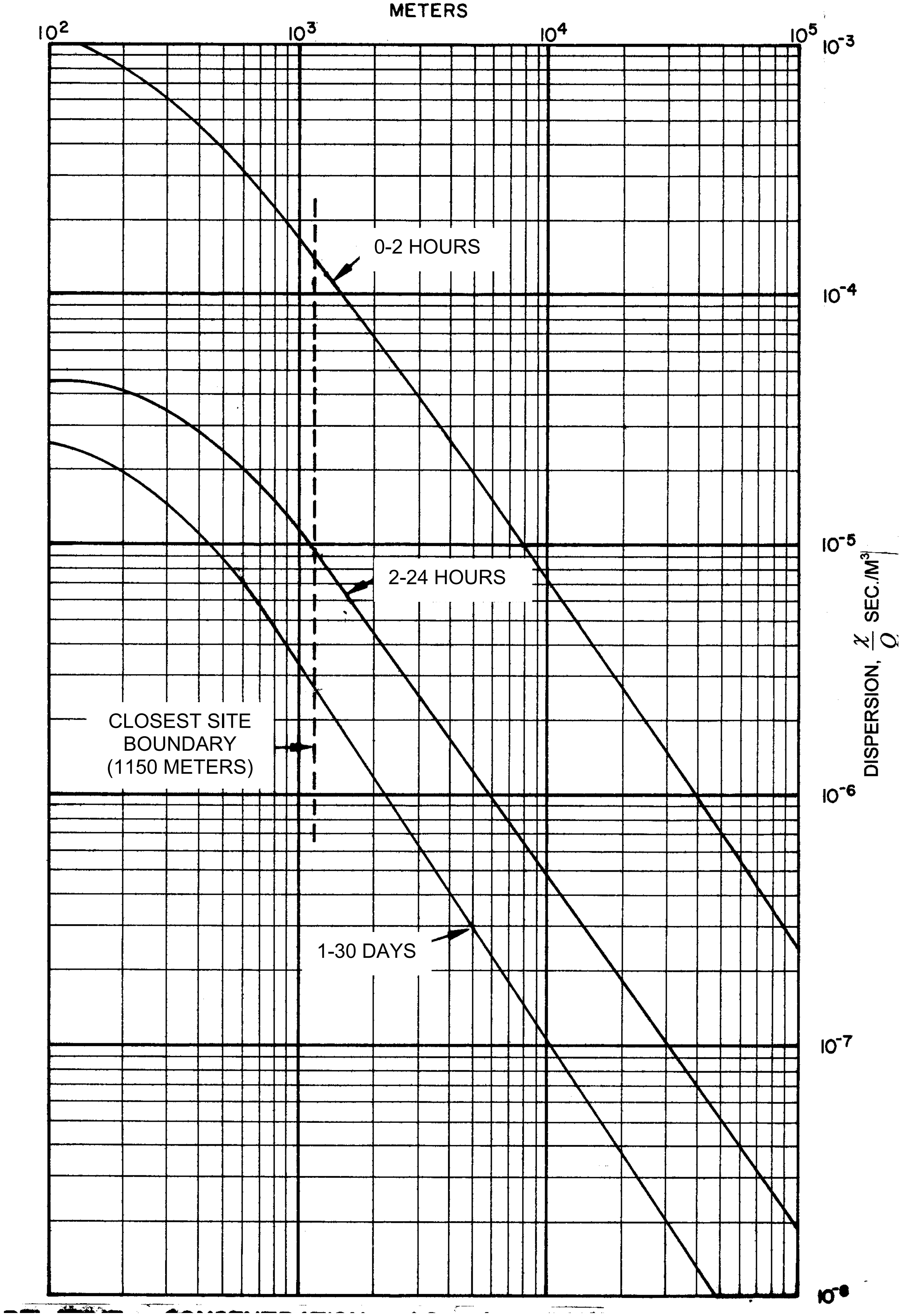
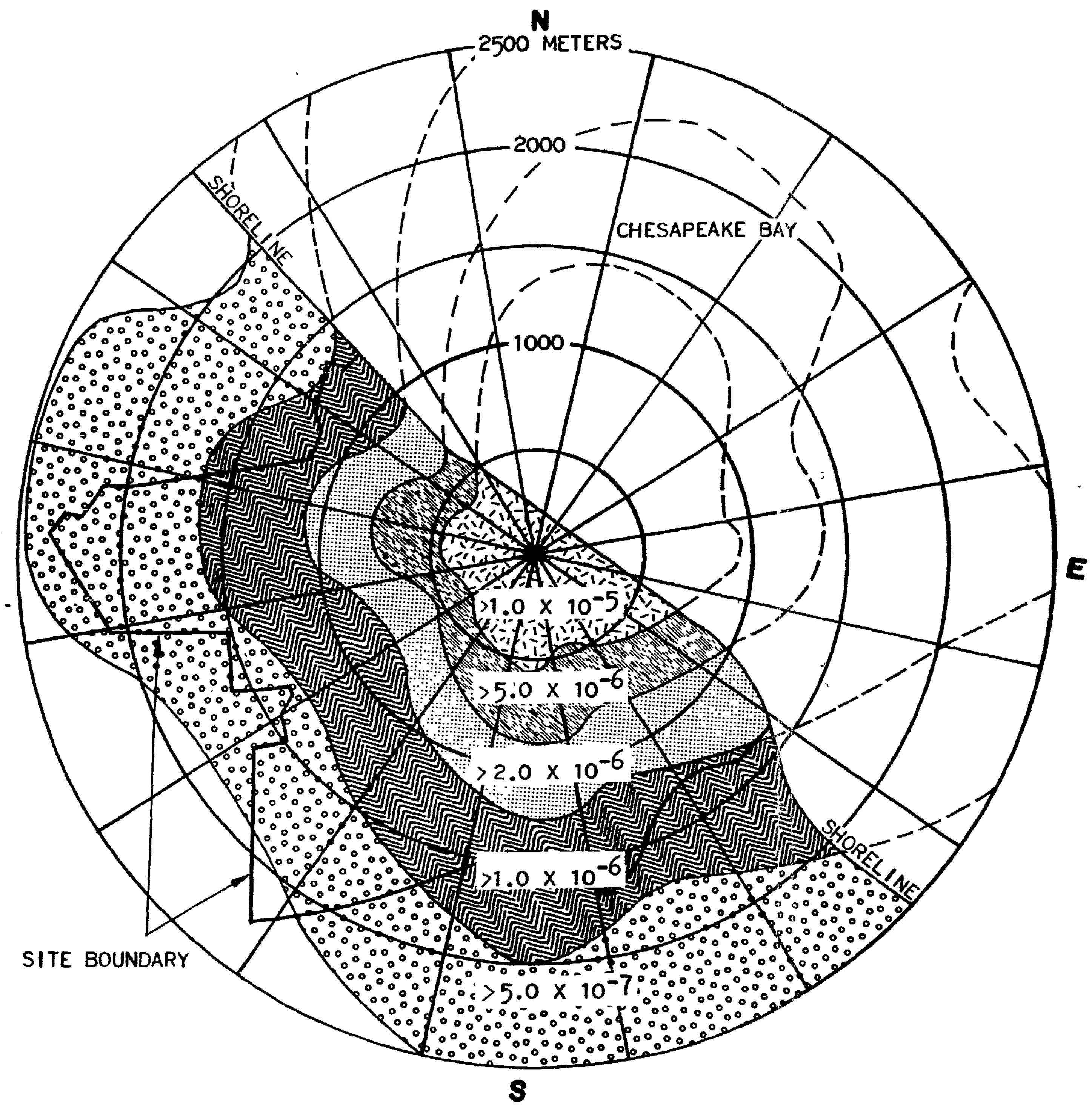


FIGURE 2.3-3
RELATIVE CONCENTRATION AS A FUNCTION OF DISTANCE FROM RELEASE POINT



DAMES & MOORE
PLATE



AVERAGE ANNUAL VENTING RELATIVE CONCENTRATION

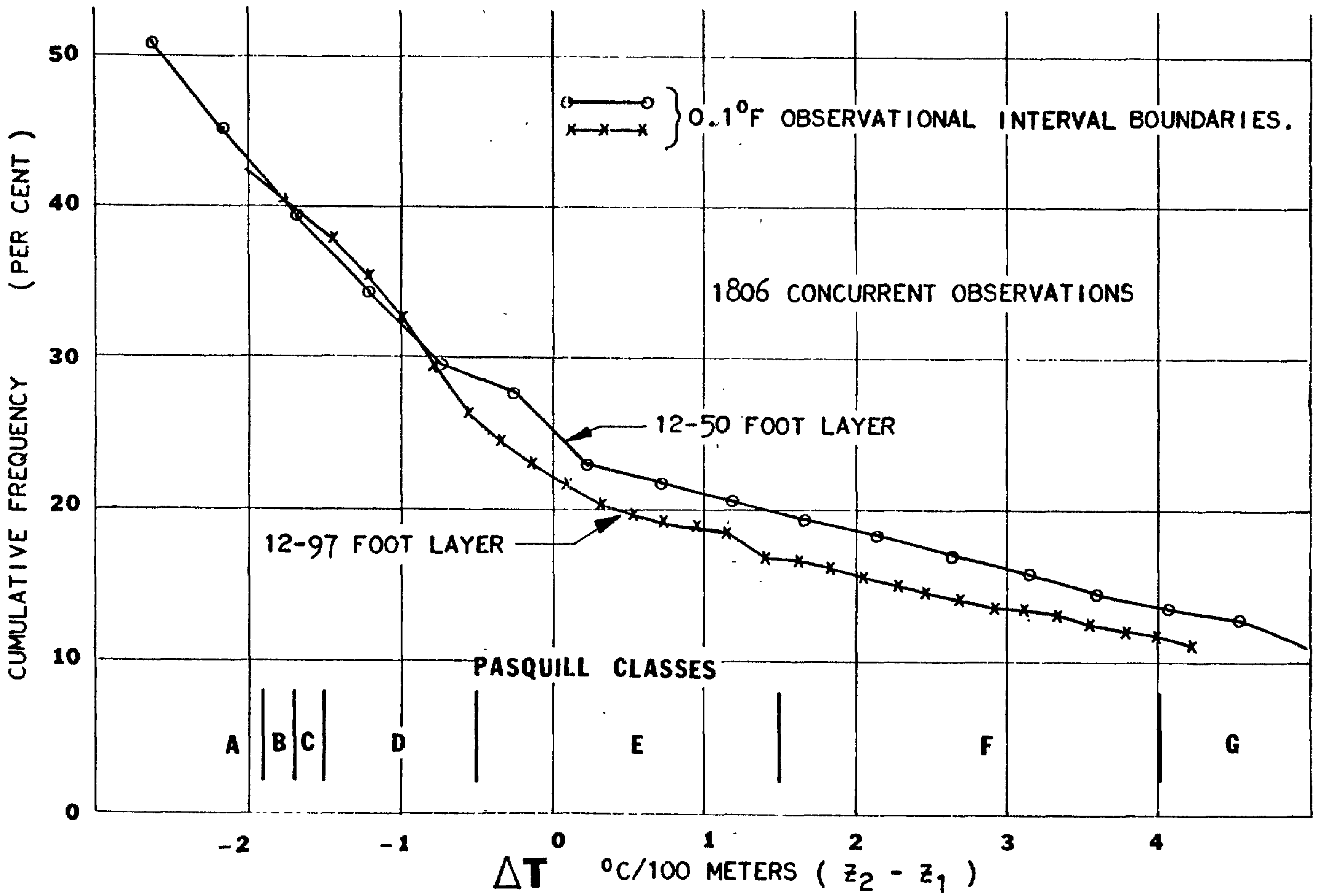
FIGURE 2.3 - 4

(Rev. 3/3/72)

DAMES & MOORE

FIGURE 2.3-5

A COMPARISON OF 12-50 FEET ABOVE GRADE ΔT VALUES WITH THOSE BETWEEN 12-97 FEET.



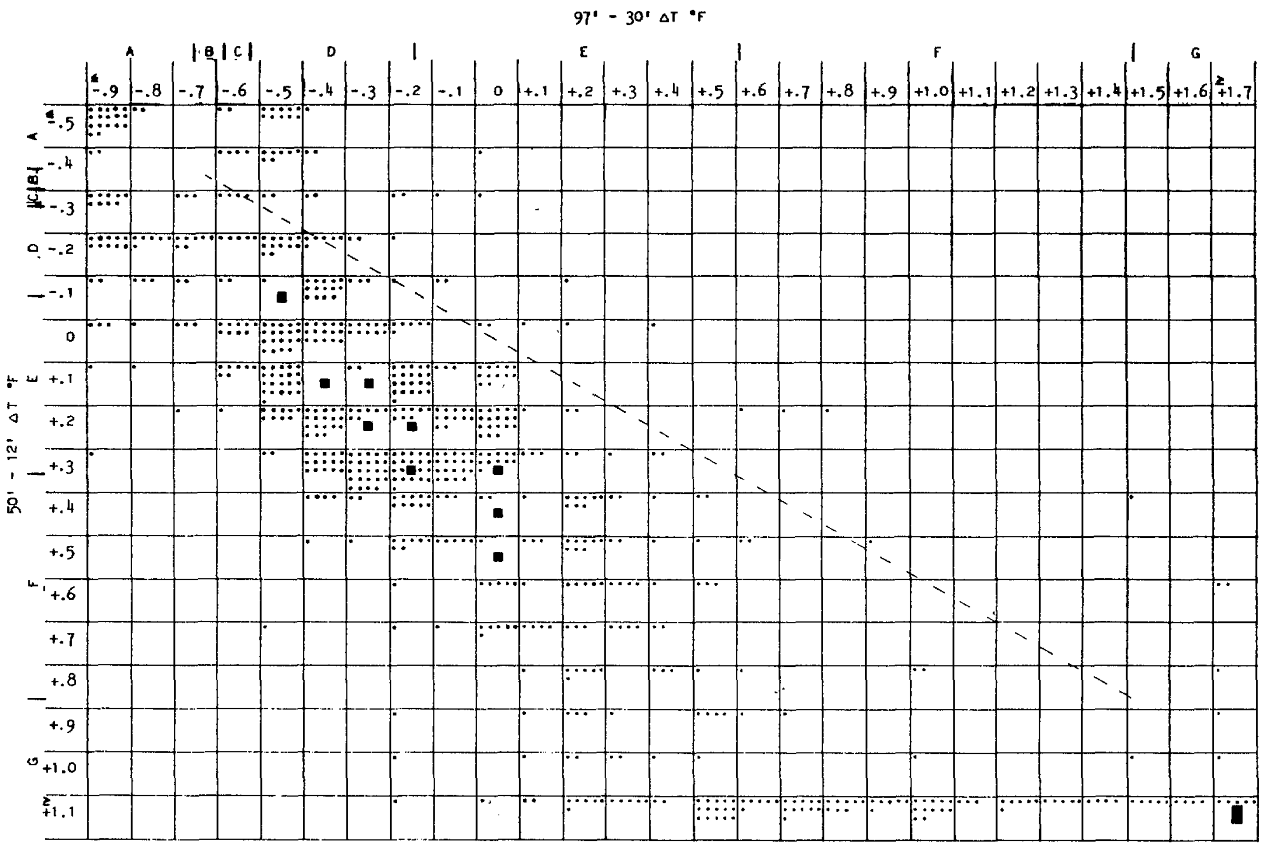
Rev. 10/22/71

DAMES & MOORE

2.3-5 A COMPARISON OF 12-50 FEET ABOVE GRADE ΔT VALUES WITH THOSE BETWEEN 12-97 FEET

Rev.0

2.3-6 CONCURRENT OBSERVATIONS OF VERTICAL THERMAL GRADIENTS AT THE 50-12 FOOT AND THE 97-30 FOOT LEVELS ABOVE GRADE



■ = 25 OCCURRENCES

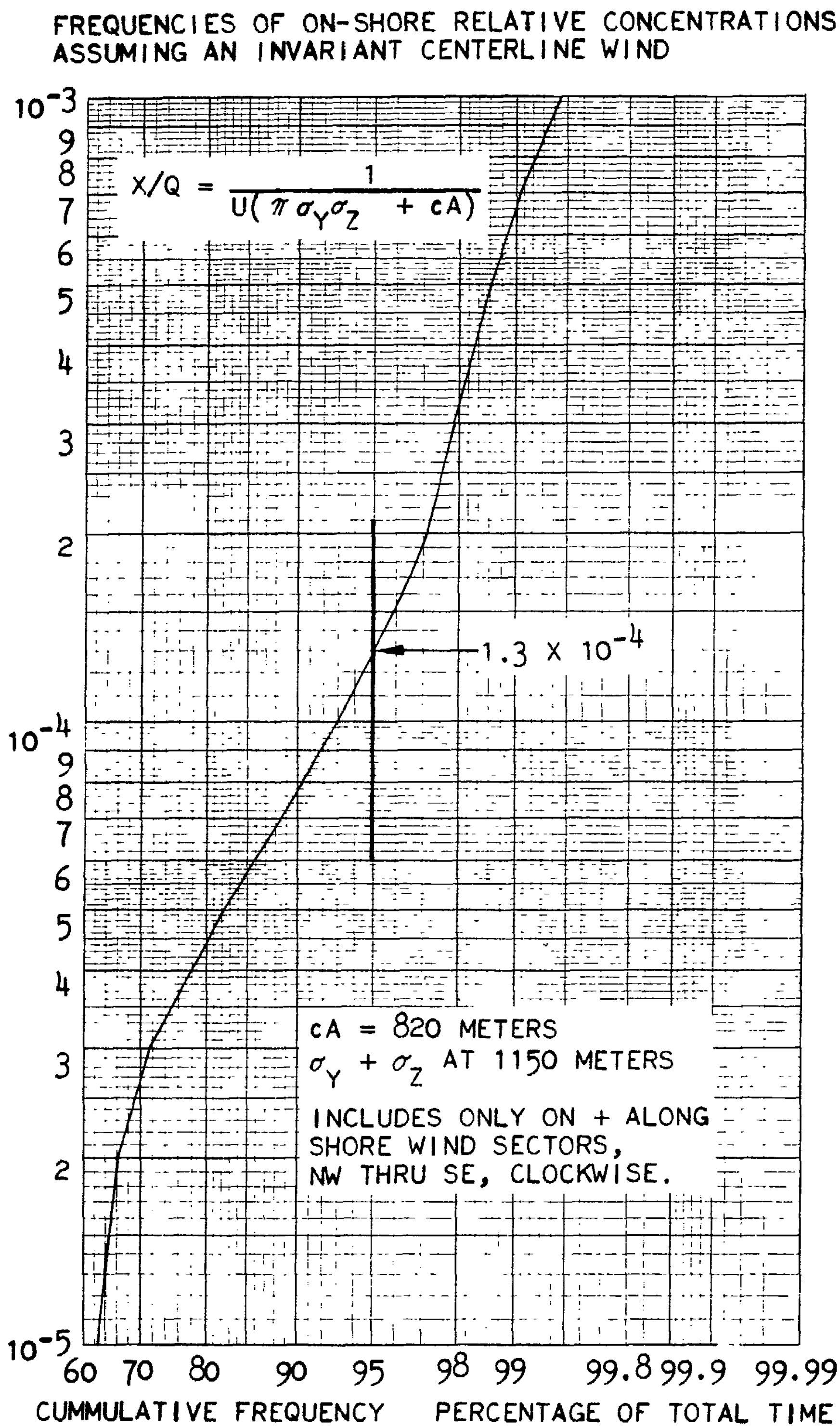
PERIOD OF RECORD: 11/12/71 TO 12/28/71 (1006 OBSERVATIONS)

CONCURRENT OBSERVATIONS OF VERTICAL THERMAL GRADIENTS
AT THE 50 - 12 FOOT AND THE 97 - 30 FOOT LEVELS ABOVE GRADE.

FIGURE 2.3-6

(Rev. 3/3/72)

DAMES & MOORE



σ_Z DETERMINED BY 12 - 50 FT. ΔT PASQUILL CLASSES
σ_Y DETERMINED BY 12 - 50 FT. ΔT PASQUILL CLASSES AT WIND $\leq$3 MPH AND BY 33 FT. LEVEL σ_θ PASQUILL CLASSES AT WINDS >3 MPH

(Rev. 3/3/72)

FIGURE 2.3 - 7

DAMES & MOORE

DATA CALENDAR
FINAL ANNUAL RECORD

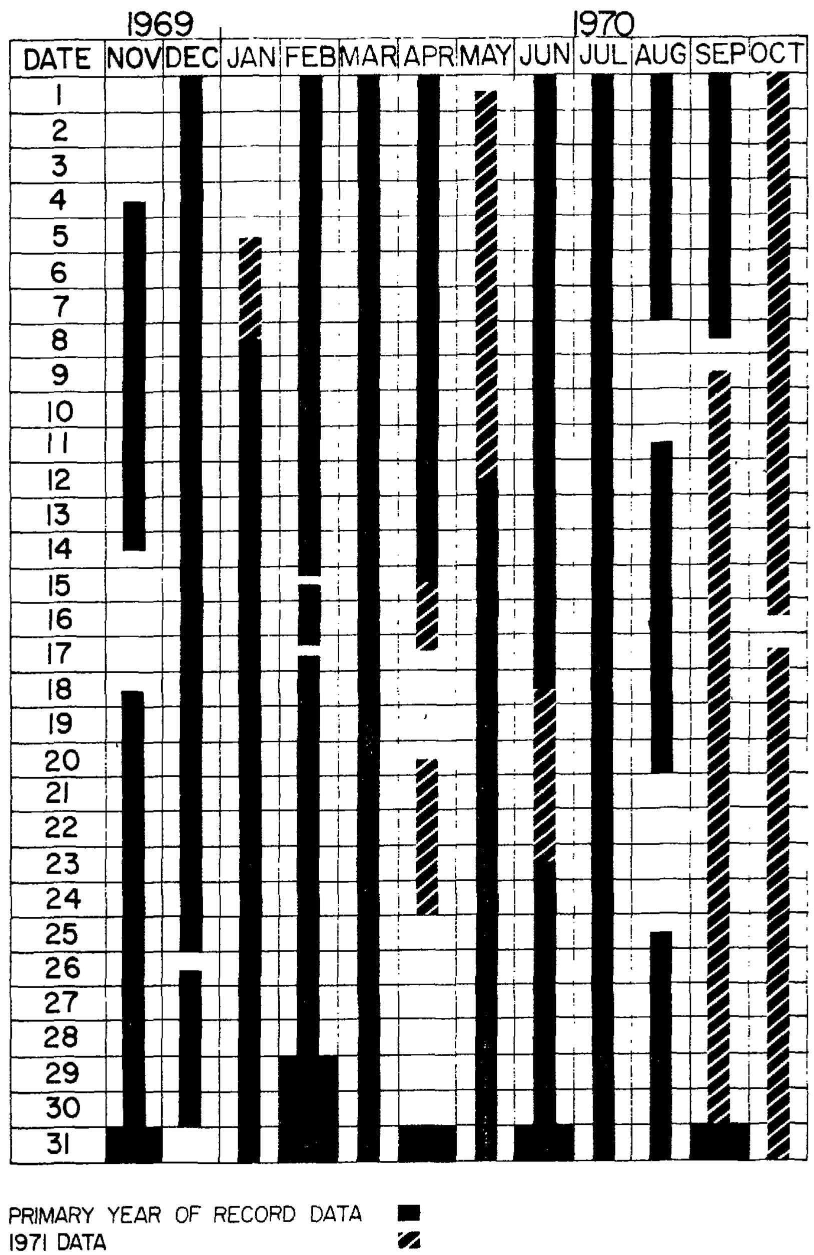
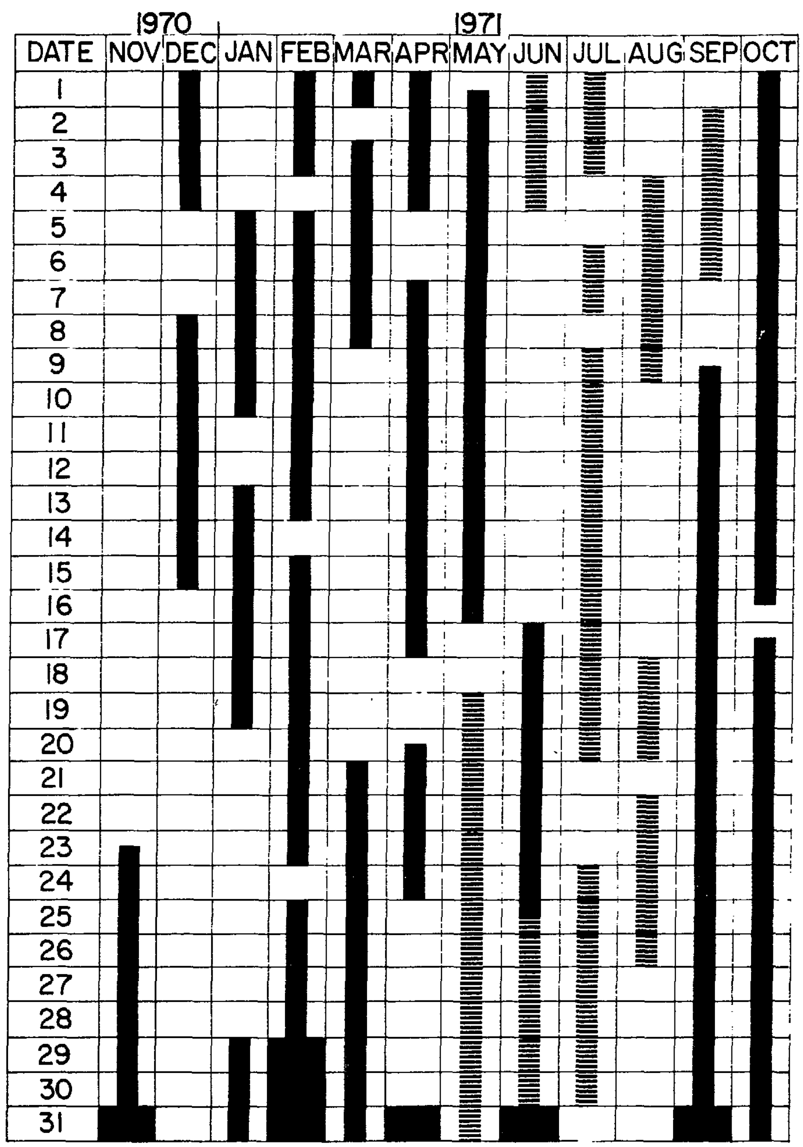


FIGURE 2.3-8

(Rev. 3/3/72)

DAMES & MOORE

DATA CALENDAR
NOVEMBER 1970 THROUGH OCTOBER 1971



33 FT LEVEL WIND, AND 50-12 FT LEVEL ΔT DATA ARE NECESSARY FOR X/Q ANALYSIS

ALL NECESSARY DATA AVAILABLE - - - - - - - - - - - - - - - - - ■

50-12 FT ΔT DATA AVAILABLE 100 FT LEVEL WIND DATA ARE - - ≡
AVAILABLE INSTEAD OF 33 FT LEVEL DATA

FIGURE 2.3-9

(Rev. 3/3/72)

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	1	0	0	0	0	0	0	0	0	0	0	0	1	.90
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SSE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
S	0	1	0	1	1	2	2	0	0	0	0	0	7	4.94
SSW	0	0	1	3	1	5	1	0	0	0	0	0	11	4.80
SW	0	0	1	6	2	1	0	0	0	0	0	0	10	3.86
WSW	0	0	4	2	4	4	0	0	0	0	0	0	14	4.04
W	0	2	2	4	5	6	0	1	0	0	0	1	21	4.71
WNW	0	0	1	0	0	0	1	0	0	0	0	0	2	4.75
NW	0	0	0	1	1	0	0	0	0	0	0	0	2	4.05
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
N	0	0	0	0	0	0	0	0	0	0	0	0	0	1
TOTAL	1	3	10	18	14	18	4	1	0	0	0	1	70	4.37

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.75
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	1	0	0	0	0	0	0	0	1	4.50
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
SSE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
S	0	0	0	1	2	5	1	0	0	0	0	0	9	5.02
SSW	1	1	0	1	0	0	0	0	0	0	0	0	3	1.87
SW	0	1	0	0	0	1	0	0	0	0	0	0	2	3.45
WSW	0	0	0	2	1	0	0	0	0	0	0	1	4	5.67
W	0	0	0	0	3	1	1	0	0	2	1	2	10	7.94
WNW	0	0	0	1	0	1	0	1	0	0	0	0	3	5.30
NW	0	0	1	0	0	0	0	0	0	1	0	0	2	5.60
NNW	0	0	0	0	0	0	1	0	0	1	0	0	2	8.05
N	0	0	0	0	0	0	0	0	0	0	0	0	0	1
TOTAL	1	2	2	6	9	8	3	1	0	4	1	3	40	5.57

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	3	2	1	0	0	0	0	0	6	5.18
NE	0	0	1	2	1	0	0	0	0	0	0	0	4	3.68
ENE	0	0	0	1	2	2	0	0	0	0	0	0	5	4.70
E	0	0	0	1	0	2	0	0	0	0	0	0	3	4.83
ESE	0	0	2	0	0	0	1	0	0	0	0	0	3	3.93
SE	0	0	0	1	0	0	0	0	0	1	0	0	2	6.50
SSE	0	0	0	1	2	0	0	1	2	0	0	1	7	7.19
S	0	0	0	0	2	0	1	0	0	0	0	0	3	4.97
SSW	0	2	0	1	0	0	1	0	0	0	0	0	4	3.00
SW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.70
WSW	0	0	2	0	1	0	0	0	0	0	0	1	4	5.38
W	0	1	1	2	2	0	1	0	1	1	1	2	12	6.77
WNW	0	0	2	2	3	2	2	0	2	1	0	1	15	6.80
NW	0	0	0	0	1	0	2	4	1	0	2	1	11	7.95
NNW	0	0	0	1	2	4	0	0	1	1	0	1	10	6.47
N	0	0	0	0	1	3	3	0	0	0	0	0	7	5.74
TOTAL	0	3	8	13	20	15	12	5	7	4	3	7	97	5.95

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	0	3	7	5	7	5	1	2	1	0	32	6.10
NE	0	0	1	7	10	3	6	2	2	0	0	0	31	5.23
ENE	0	0	1	0	10	2	3	0	1	0	1	0	18	5.40
E	0	0	0	3	8	4	5	2	3	0	0	0	25	5.83
ESE	0	1	1	1	2	4	2	3	1	1	0	0	16	5.68
SE	0	0	1	3	3	2	0	1	1	1	0	0	12	5.85
SSE	0	1	3	2	8	1	1	1	5	0	0	0	23	5.21
S	0	3	2	2	1	3	2	4	0	0	1	0	18	4.20
SSW	0	2	1	0	2	1	2	1	0	0	0	0	9	4.57
SW	1	3	0	1	1	2	0	1	0	0	0	0	9	3.58
WSW	0	1	3	0	0	0	0	0	0	4	0	2	10	7.82
W	0	1	2	3	2	0	1	1	0	3	1	3	17	6.84
WNW	0	0	0	3	3	3	3	0	1	0	2	4	19	7.50
NW	0	0	1	1	4	5	3	4	5	4	5	12	44	8.83
NNW	0	0	0	1	2	7	4	3	4	6	1	10	38	8.87
N	0	0	0	5	3	6	7	10	3	2	2	3	41	7.10
TOTAL	1	13	16	35	66	48	46	30	27	23	14	34	361	6.61

Figure 2.3-10, sheet 1
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS A CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	2	4	12	13	12	10	8	0	5	3	69	6.62
NE	0	0	0	0	5	3	5	4	2	0	0	0	19	6.25
ENE	0	0	0	0	3	4	4	1	0	0	0	0	12	5.82
E	0	0	0	3	1	7	2	2	1	1	0	0	17	5.82
ESE	0	0	0	0	3	1	1	3	2	1	0	0	11	6.96
SE	0	0	0	1	4	3	2	2	3	1	1	0	17	6.74
SSE	0	0	2	2	1	3	0	3	1	1	0	0	13	5.64
S	0	1	0	1	1	0	2	0	0	0	0	0	5	4.56
SSW	0	0	0	0	0	0	0	1	1	2	0	0	4	9.00
SW	1	0	0	0	2	0	3	1	0	1	1	0	9	6.57
WSW	0	1	1	2	1	0	0	0	0	2	0	0	7	4.97
W	0	0	1	1	3	0	1	1	0	1	1	0	9	6.01
WNW	0	0	0	2	4	0	1	2	0	2	1	2	14	7.14
NW	0	0	0	0	0	2	2	2	4	2	11	11	34	10.26
NNW	0	0	0	0	0	3	6	1	2	8	7	12	39	9.83
N	0	0	0	0	7	18	20	11	10	5	3	10	84	7.70
TOTAL	1	2	6	16	47	57	61	44	34	27	30	38	363	7.42

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	1	0	0	1	3	7	1	2	2	3	20	8.28
NE	0	0	0	4	0	2	1	1	2	1	0	0	11	6.07
ENE	0	0	0	2	1	0	0	0	0	0	0	0	3	4.13
E	0	0	0	1	0	1	0	0	0	0	0	0	2	4.40
ESE	0	0	0	0	0	1	0	0	1	0	0	0	2	7.20
SE	0	0	0	0	0	0	0	1	0	0	0	0	1	7.70
SSE	0	0	0	1	0	0	0	1	0	0	0	0	2	5.35
S	2	0	1	0	1	0	0	2	0	0	0	0	6	3.73
SSW	0	1	0	0	0	1	1	0	0	0	0	0	3	4.43
SW	0	1	0	0	0	0	0	0	0	0	0	0	1	2.00
WSW	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	1	0	0	0	0	0	0	1	2	10.15
NNW	0	0	0	0	2	1	0	0	3	4	1	10	21	10.32
N	0	0	0	0	3	1	0	3	5	6	6	8	32	9.82
TOTAL	2	2	2	9	8	8	5	15	12	13	9	22	107	8.22

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	0	0	3	0	0	1	0	0	1	5	8.06
NE	0	0	0	0	0	1	0	1	2	1	0	0	5	8.14
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.20
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	1	2	3	0	0	6	8.65
N	0	0	0	1	2	6	2	2	2	1	1	2	19	7.20
TOTAL	0	0	1	1	2	11	2	4	7	5	1	3	37	7.49

PASQUILL A (FROM AEC/DELTA T CRITERIA,50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	1	3	8	24	24	23	22	12	4	6	7	136	6.69
NE	0	0	2	13	17	9	12	8	8	2	0	0	71	5.74
ENE	1	0	1	3	17	8	7	1	1	0	1	0	40	5.81
E	0	0	0	10	12	15	7	4	4	1	0	0	53	5.55
ESE	0	1	3	2	5	7	5	7	6	2	0	0	38	6.83
SE	0	0	1	5	8	5	2	4	4	3	1	0	33	6.16
SSE	0	1	7	6	11	4	2	6	8	1	0	1	47	5.54
S	2	5	3	6	8	10	7	7	0	0	1	0	49	4.84
SSW	1	6	2	4	3	6	6	2	1	2	0	0	33	4.81
SW	2	5	1	8	5	4	4	4	1	1	1	0	36	4.79
WSW	0	2	11	8	8	5	3	1	0	6	1	7	52	5.79
W	0	4	6	11	19	10	4	3	2	8	4	10	81	6.31
WNW	0	0	3	8	12	6	6	4	3	3	4	8	59	6.89
NW	0	0	3	2	8	7	7	11	14	10	19	26	107	9.03
NNW	0	0	0	3	7	15	18	8	15	25	10	37	138	9.83
N	0	0	0	7	16	35	34	27	20	15	12	23	189	7.77
CALM													0	
TOTAL	6	25	46	104	180	170	143	119	99	83	68	119	1136	6.89

Figure 2.3-10, sheet 2
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS A CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.80
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	1	0	0	0	0	0	0	0	0	0	1	2.80
SSW	0	1	0	0	1	0	0	0	0	0	0	0	2	3.30
SW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	1	0	0	0	1	0	0	0	0	0	0	2	3.65
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	1	0	0	0	0	1	0	0	2	6.75
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	0	2	3	0	2	1	0	0	0	1	0	0	9	3.97

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.80
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	1	0	1	1	1	0	0	0	0	0	0	4	3.95
WNW	0	0	0	0	0	1	0	0	0	0	0	0	1	5.80
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	0	2	0	1	1	2	0	0	0	0	0	0	6	3.90

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	1	0	0	0	0	2	5.65
NE	0	0	0	0	1	0	1	0	0	0	0	0	2	5.30
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.50
SE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.60
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
WNW	0	0	0	0	1	1	0	0	0	0	0	0	2	5.30
NW	0	0	0	1	0	0	0	1	1	0	0	0	3	6.70
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	2	0	0	0	0	0	0	2	5.40
TOTAL	0	0	0	1	3	5	2	2	1	0	0	0	14	5.77

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	2	1	0	2	0	0	0	0	5	6.08
NE	0	0	0	1	1	1	0	0	0	0	2	0	5	6.78
ENE	0	0	0	1	0	0	1	0	0	0	0	0	2	5.15
E	0	0	0	0	0	1	3	1	0	0	0	0	5	6.36
ESE	0	0	0	0	1	0	1	0	1	1	0	0	4	7.35
SE	0	0	0	0	0	1	1	0	1	0	0	0	3	7.00
SSE	0	0	0	1	0	0	2	1	0	0	0	0	4	6.22
S	0	1	0	0	0	0	0	0	0	1	0	0	2	5.45
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	1	0	0	0	0	0	0	0	1	0	0	2	5.65
W	0	0	0	0	1	0	0	0	0	0	0	1	2	8.25
WNW	0	0	0	2	1	0	0	0	0	0	0	0	3	3.57
NW	0	0	0	0	0	2	0	2	0	0	0	3	7	9.61
NNW	0	0	0	0	0	0	0	1	1	0	1	2	5	10.96
N	0	0	0	0	1	0	2	1	0	1	1	0	6	7.62
TOTAL	0	2	0	5	7	6	10	8	3	4	4	6	55	7.25

Figure 2.3-10, sheet 3
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS B CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	3	0	0	0	0	0	4	5.95
NE	0	0	0	1	0	1	0	0	0	0	0	0	2	4.90
ENE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.70
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.70
ESE	0	0	0	0	1	0	1	0	0	0	0	0	2	5.45
SE	0	0	0	0	0	0	0	1	0	0	0	0	1	7.70
SSE	0	0	0	0	0	0	0	0	1	1	0	0	2	9.30
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.70
SW	0	0	0	0	0	0	0	0	0	0	1	0	1	11.00
WSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	1	0	0	0	0	1	7.30
NW	0	0	0	0	0	0	2	1	1	0	1	3	8	9.90
NNW	0	0	1	0	0	1	0	1	2	2	1	3	11	9.95
N	0	0	0	2	1	3	1	1	2	0	0	1	11	6.65
TOTAL	0	0	1	4	5	6	7	5	6	3	3	7	47	7.66

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	2	0	0	0	0	0	0	0	2	4.40
NE	0	0	0	1	1	0	0	0	0	1	0	0	3	6.13
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	1	0	0	0	1	0	0	0	0	0	2	4.15
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	1	0	0	0	1	8.30
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.90
NNW	0	0	0	0	0	0	0	0	1	0	0	1	2	14.00
N	0	0	0	1	0	1	3	0	0	0	2	1	8	8.08
TOTAL	0	0	2	2	3	1	4	0	2	1	2	2	19	7.31

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.50
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.90
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	1	1	13.30
N	0	0	0	0	0	1	1	1	0	0	0	0	3	6.43
TOTAL	0	0	0	1	1	1	1	1	0	0	0	1	6	6.83

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	6	1	3	3	0	0	0	0	13	5.71
NE	0	0	0	3	4	2	1	0	0	1	2	0	13	5.94
ENE	0	0	0	1	1	0	1	0	0	0	0	0	3	5.00
E	0	0	1	0	0	2	4	1	0	0	0	0	8	5.72
ESE	0	0	1	0	2	1	2	0	1	1	0	0	8	6.07
SE	0	0	0	0	0	2	1	1	1	0	0	0	5	6.90
SSE	0	0	0	1	0	0	2	1	2	1	0	0	7	7.40
S	0	1	1	0	0	0	0	0	0	1	0	0	3	4.57
SSW	0	2	0	1	2	0	0	0	0	0	0	0	5	3.40
SW	0	0	1	0	0	0	0	0	0	0	1	0	2	6.85
WSW	0	1	0	1	0	0	0	0	0	1	0	0	3	4.83
W	0	2	0	1	2	2	1	0	0	0	0	1	9	5.08
WNW	0	0	0	2	2	2	0	1	0	0	0	0	7	4.91
NW	0	0	1	1	1	2	2	4	2	1	1	6	21	8.71
NNW	0	0	1	0	0	1	0	2	4	2	2	8	20	10.92
N	0	0	0	3	2	7	7	3	2	1	3	2	30	7.10
CALM													0	
TOTAL	0	6	6	14	22	22	24	16	12	9	9	17	157	7.01

Figure 2.3-10, sheet 4
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS B CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(1S)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
ENE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
E	0	0	0	0	0	0	0	0	0	0	0	0	0	*
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
S	0	0	2	0	2	1	0	0	0	0	0	0	5	4.00
SSW	0	0	0	2	0	1	0	0	0	0	0	0	3	4.07
SW	0	0	0	2	1	1	1	0	1	0	0	0	6	5.33
WSW	0	0	3	0	1	1	0	0	0	0	0	0	5	3.34
W	0	0	3	3	0	2	1	0	0	0	0	0	9	3.91
WNW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
NW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
N	0	0	0	0	0	0	0	0	0	0	0	0	0	*
TOTAL	0	0	8	8	6	6	2	0	1	0	0	0	31	4.18

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
NE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.90
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
E	0	0	0	0	0	0	0	0	0	0	0	0	0	*
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
S	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.60
WSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.80
W	0	0	0	1	1	0	0	0	0	0	0	0	2	4.05
WNW	0	1	1	0	0	1	1	0	0	0	0	0	4	4.07
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
N	0	0	0	0	0	0	0	0	0	0	0	0	0	*
TOTAL	0	1	1	4	3	1	1	0	0	0	0	0	11	4.04

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.30
NE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.80
ENE	0	0	0	2	0	1	0	0	0	0	0	0	3	4.07
E	0	0	1	0	1	0	0	0	0	0	0	0	2	3.85
ESE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.40
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	*
S	0	1	0	0	0	0	0	0	0	0	0	0	1	2.00
SSW	0	1	0	0	1	0	0	0	0	0	0	0	2	2.85
SW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.80
WSW	0	0	0	0	0	1	0	0	0	0	0	0	1	5.80
W	0	0	0	0	1	0	0	0	0	0	0	0	1	4.30
WNW	0	0	0	0	0	2	0	1	0	0	0	0	3	6.00
NW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
N	0	0	0	1	0	0	0	0	0	0	0	1	2	7.95
TOTAL	0	2	2	3	5	5	1	1	0	0	0	1	20	4.75

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	1	0	0	2	0	0	1	5	6.00
NE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.90
ENE	0	0	0	1	0	1	0	0	0	0	0	0	2	4.40
E	0	0	1	1	6	0	2	2	1	0	0	0	13	5.36
ESE	0	0	0	0	1	1	1	0	0	0	0	0	3	5.40
SE	0	0	0	0	0	3	1	0	0	0	0	1	5	7.84
SSE	0	1	0	0	1	0	0	0	1	1	0	0	4	5.97
S	1	0	0	0	0	0	1	0	0	0	0	0	2	3.40
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	*
W	0	0	0	0	2	0	0	0	0	0	0	0	2	4.55
WNW	0	0	0	0	0	1	2	1	1	0	0	0	5	7.10
NW	0	0	0	0	0	0	2	0	1	2	1	7	13	10.45
NNW	0	0	0	2	1	0	0	0	1	0	0	1	5	7.16
N	0	0	0	1	0	0	0	0	1	0	0	0	2	5.95
TOTAL	1	1	1	6	13	7	9	3	8	3	1	10	63	6.94

Figure 2.3-10, sheet 5
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS C CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	2	2	1	1	1	0	1	1	0	9	6.30
NE	0	0	0	0	3	0	0	0	0	0	1	0	4	6.08
ENE	0	1	0	2	1	0	0	1	0	0	0	0	5	4.20
E	0	0	1	0	0	1	0	0	0	0	0	0	2	3.85
ESE	0	0	0	0	0	1	2	0	0	0	0	0	3	6.20
SE	0	0	0	0	0	0	0	0	1	2	0	0	3	9.10
SSE	0	0	0	0	0	0	0	2	0	1	2	0	5	9.28
S	1	0	0	0	0	0	1	0	1	0	0	0	3	5.23
SSW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.60
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
NW	0	0	0	1	0	1	0	1	2	4	0	7	16	10.22
NNW	1	0	0	1	1	1	1	0	1	0	4	6	16	10.24
N	0	0	0	0	0	0	1	1	1	1	0	1	5	9.42
TOTAL	2	1	1	7	7	5	7	6	6	9	8	14	73	8.85

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	2	1	3	0	0	0	0	0	1	7	5.74
NE	0	0	0	0	0	0	0	0	0	1	1	0	2	10.45
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.10
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.90
NW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.30
NNW	0	0	0	0	1	0	0	0	0	1	1	2	5	12.12
N	0	0	0	1	3	1	0	0	2	0	3	4	14	9.27
TOTAL	0	0	1	4	5	5	1	0	2	2	5	7	32	8.41

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	0	0	0	0	0	0	1	0	1	2	11.70
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	1	0	0	0	0	0	0	0	1	4.30
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.40
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	2	0	0	0	0	1	1	4	9.20
TOTAL	0	0	0	0	1	3	0	0	0	1	1	2	8	8.74

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	4	5	5	1	1	1	2	1	3	23	6.79
NE	0	0	0	2	5	1	0	0	0	1	2	0	11	6.04
ENE	0	1	0	6	1	2	0	1	0	0	0	0	11	4.18
E	0	0	3	1	8	1	2	2	1	0	0	0	18	4.97
ESE	0	0	1	0	1	3	3	0	0	0	0	0	8	5.29
SE	0	0	0	0	0	4	1	0	1	2	0	0	8	7.11
SSE	0	1	0	1	0	3	0	1	0	2	4	0	12	7.45
S	1	1	2	0	4	4	1	1	1	0	1	0	16	5.03
SSW	0	1	0	1	2	0	0	0	0	0	0	0	4	3.37
SW	0	0	0	3	2	1	1	0	1	0	0	0	8	5.02
WSW	0	1	2	1	2	0	0	0	0	0	0	0	6	3.85
W	0	0	1	5	2	2	1	1	0	0	0	0	12	4.46
WNW	0	0	0	0	3	6	2	1	0	1	0	1	14	6.34
NW	0	0	0	1	1	1	4	1	3	6	1	14	32	9.69
NNW	1	0	0	3	3	1	1	0	2	1	5	9	26	10.01
N	0	0	0	3	3	3	1	1	5	1	4	8	29	9.07
CALM													0	
TOTAL	2	5	9	31	42	37	16	10	15	16	18	39	238	7.07

Figure 2.3-10, sheet 6
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS C CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.50
NE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	0	1	1	0	0	1	0	0	0	0	0	3	4.13
SE	0	0	0	0	1	2	0	0	0	0	0	0	3	5.37
SSE	0	0	1	1	2	0	0	0	0	0	0	0	4	3.55
S	0	2	7	4	4	6	0	0	0	0	0	1	24	4.10
SSW	0	2	2	8	15	3	1	0	0	0	0	0	31	4.14
SW	0	6	7	10	10	2	0	0	0	0	0	0	35	3.48
WSW	1	3	5	13	9	2	0	1	0	0	0	0	34	3.61
W	2	3	6	9	3	2	3	0	0	0	0	0	28	3.58
WNW	0	0	0	0	3	2	0	1	0	0	1	0	7	5.93
NW	0	2	1	1	1	0	0	0	0	0	0	0	5	2.95
NNW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.40
N	0	0	0	0	0	0	0	0	0	0	0	0	0	1
TOTAL	3	19	31	48	48	19	5	2	0	0	1	1	177	3.82

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	4.25
NE	0	0	0	1	1	0	0	0	0	0	0	0	2	4.20
ENE	0	0	0	0	0	1	0	0	0	0	0	0	1	6.00
E	0	0	1	0	0	0	0	0	0	0	0	0	1	2.50
ESE	0	0	0	0	0	1	0	1	0	0	0	0	2	6.65
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SSE	1	0	1	2	0	0	0	0	0	0	0	0	4	2.55
S	0	2	6	1	3	4	2	1	1	0	0	1	21	4.75
SSW	0	2	3	2	1	1	0	0	0	0	0	0	9	3.02
SW	0	1	3	0	0	2	0	0	0	0	0	0	6	3.20
WSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.50
W	0	2	1	1	0	2	0	0	0	0	0	0	6	3.48
WNW	0	0	2	0	0	3	2	0	1	1	1	0	10	6.36
NW	0	0	0	1	0	0	0	0	0	1	0	1	3	8.63
NNW	0	1	0	0	1	0	0	0	0	0	0	0	2	3.45
N	0	0	0	0	0	0	0	0	0	0	0	0	0	1
TOTAL	1	8	17	10	7	14	4	2	2	2	1	2	70	4.51

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	2	0	0	1	0	1	0	0	0	4	5.60
NE	0	2	0	2	1	0	2	1	0	3	0	0	11	5.94
ENE	0	0	0	2	3	0	0	0	0	0	0	0	5	4.12
E	0	0	0	4	2	1	1	0	0	0	0	0	8	4.37
ESE	0	0	1	1	3	3	1	2	1	0	0	0	12	5.49
SE	1	0	0	1	1	2	1	0	2	0	0	0	8	5.54
SSE	0	0	1	1	3	1	3	3	0	0	0	4	16	7.34
S	1	3	3	4	5	2	6	1	0	1	0	0	26	4.48
SSW	1	2	0	1	1	0	0	1	0	0	0	0	6	3.16
SW	0	0	1	1	0	2	0	0	0	0	1	0	5	5.62
WSW	0	1	1	4	5	0	0	0	0	0	1	0	12	4.22
W	0	3	3	1	2	1	0	3	0	0	0	1	14	4.54
WNW	0	0	2	2	2	4	4	2	1	4	0	0	21	6.06
NW	0	0	2	3	7	2	3	2	1	3	3	2	28	6.69
NNW	0	1	2	2	0	0	0	0	1	3	0	1	10	6.31
W	0	0	0	1	1	0	1	0	0	0	0	0	3	4.50
TOTAL	3	12	16	32	36	18	23	15	7	14	5	8	169	5.53

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	2	3	1	0	2	2	0	2	1	3	16	7.06
NE	0	1	0	5	3	4	3	2	2	0	0	0	20	5.19
ENE	0	1	1	6	3	3	1	1	1	0	0	0	19	4.55
E	0	0	5	13	6	10	5	3	1	0	0	0	43	4.78
ESE	0	1	1	2	6	4	3	5	1	0	0	0	23	5.49
SE	0	1	1	3	4	7	6	5	2	0	0	0	29	5.75
SSE	0	0	5	1	4	9	6	7	5	6	5	6	54	7.60
S	0	0	3	3	5	2	2	5	2	0	1	0	23	5.73
SSW	0	0	0	0	4	1	1	1	1	0	0	0	8	5.77
SW	0	2	2	0	0	1	1	1	0	1	0	1	9	5.72
WSW	0	1	1	1	1	0	0	0	0	0	1	1	6	5.76
W	0	0	0	2	1	3	2	1	0	0	0	1	10	6.10
WNW	1	0	0	3	7	5	5	1	2	0	0	1	25	5.58
NW	0	1	0	0	9	6	5	5	3	7	6	19	63	6.96
NNW	0	0	0	0	1	1	1	3	3	4	2	3	18	9.03
N	0	1	2	4	5	3	2	3	4	2	3	5	34	7.13
TOTAL	1	9	23	48	60	61	45	45	27	22	19	40	400	6.61

Figure 2.3-10, sheet 7
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS D CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	1	6	3	5	3	1	3	3	3	6	34	7.58
NE	1	0	0	4	3	3	2	5	3	2	1	0	24	6.39
ENE	0	0	0	4	5	0	0	4	0	1	0	0	14	5.51
E	0	0	2	7	4	3	3	3	1	0	0	0	23	4.97
ESE	0	0	2	0	2	4	4	2	2	3	0	0	19	6.57
SE	0	0	0	0	1	3	4	5	2	1	3	1	20	7.74
SSE	0	1	1	2	6	9	1	5	7	5	1	2	40	6.93
S	1	1	0	2	1	3	2	1	2	1	0	0	14	5.49
SSW	0	0	0	0	0	1	1	0	2	0	0	0	4	7.37
SW	0	0	0	0	0	0	0	0	1	0	2	2	5	10.80
WSW	0	0	0	0	1	0	0	1	1	0	0	1	4	8.20
W	0	0	1	2	0	0	0	0	0	0	0	0	3	2.97
WNW	0	1	1	1	1	1	2	3	0	0	0	1	11	6.03
NW	0	0	0	1	2	6	4	4	6	3	10	30	66	10.62
NNW	0	0	1	1	5	2	2	5	11	14	5	40	86	11.17
N	0	1	2	6	5	4	7	6	3	2	6	18	60	8.13
TOTAL	2	4	11	36	39	44	35	45	44	41	31	95	487	8.37

(Rev. 3/3/72)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	2	2	3	4	0	5	6	3	5	30	8.43
NE	0	0	0	2	1	1	0	3	1	0	1	0	9	6.47
ENE	0	0	0	1	0	0	0	1	0	1	0	0	3	6.67
E	0	0	0	0	1	1	1	0	0	0	0	0	3	5.53
ESE	0	0	0	0	0	1	1	1	0	0	0	0	3	6.37
SE	0	0	1	0	0	1	0	0	0	1	0	0	3	5.50
SSE	0	2	0	0	1	0	0	0	1	1	2	0	7	6.63
S	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	1	0	0	1	10.00
WSW	0	0	0	0	0	0	0	0	1	0	0	0	1	8.20
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
NW	0	0	0	0	0	0	0	0	2	0	0	1	3	9.93
NNW	0	0	0	1	2	3	0	2	0	3	4	6	21	9.58
N	0	0	2	3	2	1	1	1	5	6	4	7	32	9.83
TOTAL	0	2	4	9	9	11	8	8	15	19	14	19	118	8.48

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	2	1	1	1	1	1	4	1	5	17	8.96
NE	0	0	0	0	0	0	3	0	1	3	1	0	8	8.36
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.80
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	2	2	12.95
NNW	0	0	0	0	1	1	1	0	0	0	1	2	6	9.53
N	0	0	0	0	5	1	1	2	0	1	1	8	19	9.77
TOTAL	0	0	1	2	7	3	6	3	2	8	4	17	53	9.26

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-18 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	2	3	17	8	9	9	6	7	15	8	20	104	7.68
NE	1	3	3	15	10	8	9	10	9	7	2	1	78	5.99
ENE	0	1	2	15	9	6	8	5	0	2	0	0	48	4.90
E	0	0	8	25	12	14	11	6	2	0	0	0	78	4.76
ESE	1	2	4	4	8	12	11	10	4	3	0	0	59	5.68
SE	1	0	2	4	8	14	14	10	5	2	3	3	66	6.50
SSE	1	4	7	7	18	21	11	15	14	12	4	13	127	6.93
S	3	8	20	13	20	11	10	4	5	2	0	2	98	4.59
SSW	2	4	3	11	20	7	5	1	3	0	1	0	57	4.62
SW	1	7	15	11	14	8	2	1	2	2	3	3	69	4.73
WSW	2	4	10	18	13	6	1	2	2	1	1	2	62	4.36
W	1	7	11	14	9	8	6	2	0	0	0	2	60	4.23
WNW	0	4	9	8	11	15	11	10	4	4	1	3	80	5.74
NW	0	3	3	5	16	15	18	12	12	14	10	55	165	9.31
NNW	0	1	3	4	8	8	5	6	15	24	10	54	138	10.38
N	0	2	6	16	16	10	12	14	14	17	15	37	163	8.34
CALM													0	
TOTAL	13	58	109	189	208	172	131	114	98	105	66	195	1446	6.80

Figure 2.3-10, sheet 8
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS D CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.90
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.40
E	0	0	1	0	0	0	0	0	0	0	0	0	1	2.60
ESE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.20
SE	0	1	0	1	0	0	0	0	0	0	0	0	2	2.50
SSE	1	1	4	6	4	0	1	0	0	0	0	0	19	3.37
S	2	7	26	15	22	4	7	1	0	0	0	0	84	3.69
SSW	0	10	25	17	20	10	1	1	0	0	1	0	85	3.70
SW	1	8	14	12	20	4	1	2	0	0	0	0	62	3.68
WSW	0	10	10	10	8	5	0	0	0	0	0	0	43	3.28
W	0	5	8	10	18	6	1	1	1	0	0	0	44	3.99
WNW	0	5	5	2	5	5	2	1	2	2	0	1	30	4.96
NW	0	0	4	1	1	1	0	1	0	1	0	0	9	4.64
NNW	0	1	0	1	2	1	0	0	0	0	0	0	5	3.66
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	4	49	97	78	95	36	13	7	3	3	1	1	387	3.77

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.10
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.60
ENE	0	0	0	0	0	1	1	0	0	0	0	0	2	5.95
E	0	1	0	3	0	0	0	0	0	0	0	0	4	3.12
ESE	1	2	0	0	1	0	0	0	0	0	0	0	4	2.17
SE	0	0	1	1	1	1	1	0	0	0	0	0	5	4.46
SSE	0	3	9	7	3	2	1	2	1	0	0	0	28	3.87
S	1	7	13	13	20	7	2	2	0	0	0	1	66	3.99
SSW	1	6	8	8	7	4	3	1	0	0	0	0	38	3.62
SW	0	5	5	2	1	1	0	0	0	0	0	0	14	2.83
WSW	3	5	5	1	2	0	0	0	0	0	0	0	16	2.17
W	8	2	5	1	1	0	1	0	0	0	1	0	13	3.28
WNW	0	1	2	4	4	5	0	1	1	1	0	0	19	4.66
NW	0	1	1	0	3	4	1	1	4	0	0	1	16	6.31
NNW	0	2	0	1	2	2	0	0	0	0	1	0	8	4.50
N	0	0	0	0	1	0	0	0	0	0	0	0	1	4.90
TOTAL	8	35	49	48	47	27	10	7	6	1	2	2	236	3.91

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	2	0	1	2	2	3	3	1	0	0	15	6.05
NE	2	0	0	2	1	1	2	1	4	3	0	0	16	6.55
ENE	0	0	2	1	2	0	2	3	2	0	0	0	12	5.92
E	0	2	2	4	1	2	3	0	3	0	0	0	17	4.97
ESE	1	3	1	2	1	2	2	2	0	0	0	0	14	4.19
SE	0	0	1	3	1	3	3	0	0	0	0	1	12	5.37
SSE	2	4	16	11	17	8	6	2	1	2	1	1	71	4.46
S	3	8	20	15	20	16	7	3	0	0	1	0	93	4.13
SSW	1	10	7	6	6	8	4	2	0	0	0	0	44	3.77
SW	0	5	6	0	3	0	0	0	0	0	0	0	14	2.76
WSW	0	1	6	1	1	2	1	0	0	0	0	0	12	3.37
W	1	5	5	10	5	2	1	1	0	0	0	0	30	3.53
WNW	1	1	4	13	4	3	6	2	2	4	2	2	44	5.67
NW	0	0	4	2	3	8	9	5	2	1	3	1	38	6.37
NNW	0	2	0	6	4	4	1	0	0	1	0	0	18	4.51
N	0	2	1	6	3	2	0	1	4	1	0	1	21	5.46
TOTAL	12	43	77	62	73	63	49	25	21	13	7	6	471	4.70

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	2	2	1	1	2	2	3	4	4	0	22	7.23
NE	0	1	4	3	2	2	1	6	8	11	2	1	41	7.31
ENE	1	0	5	2	4	1	5	1	3	1	1	2	26	6.01
E	0	5	3	5	6	4	7	5	2	0	1	0	38	5.1
ESE	0	2	4	6	7	6	7	1	1	0	0	0	34	4.8
SE	0	1	5	6	6	8	6	2	3	0	0	0	37	5.0
SSE	0	3	19	27	15	14	14	11	8	3	6	12	132	5.99
S	2	4	10	14	13	8	3	2	1	3	1	4	65	4.84
SSW	0	7	8	7	3	3	1	3	0	0	0	0	32	3.67
SW	2	4	6	2	1	0	0	0	0	0	0	0	15	2.35
WSW	1	2	1	1	1	0	0	0	1	1	0	0	8	4.07
W	0	5	8	6	2	0	1	2	0	0	0	0	24	3.38
WNW	0	2	4	15	13	10	3	7	2	3	2	3	64	5.63
NW	0	1	3	9	22	16	16	10	10	12	9	20	128	7.32
NNW	0	0	0	2	1	4	6	3	5	2	4	3	30	7.94
N	0	1	1	4	2	3	1	1	3	3	3	1	23	6.97
TOTAL	7	38	83	111	99	60	73	56	50	43	33	46	719	5.92

Figure 2.3-10, sheet 9
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS E CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	2	7	5	8	5	3	7	10	6	6	3	8	70	6.55
NE	1	0	1	6	1	7	2	2	2	3	0	4	29	6.56
ENE	0	0	5	7	3	1	1	2	0	0	1	5	25	6.18
E	3	0	5	8	3	3	3	1	1	0	0	PO	27	4.04
ESE	0	2	1	7	3	6	3	4	1	2	0	0	29	5.38
SE	0	2	2	5	2	3	4	1	3	2	1	0	25	5.60
SSE	0	2	4	9	10	13	11	15	6	5	4	5	84	6.57
S	1	4	3	6	6	1	5	3	1	3	0	2	35	5.26
SSW	1	3	2	3	2	0	0	0	0	0	0	0	11	2.86
SW	2	0	2	1	1	0	0	0	0	0	0	0	6	2.65
WSW	0	2	1	1	0	0	0	0	0	0	0	0	4	2.47
W	1	0	1	2	4	0	1	0	1	0	0	0	10	4.25
WNW	0	0	1	9	7	3	3	2	1	0	0	3	29	6.03
NW	0	1	2	2	9	17	30	28	18	11	16	28	162	8.35
NNW	0	2	0	2	2	10	7	6	15	12	17	19	92	9.19
N	0	3	3	10	7	4	8	7	6	2	7	13	70	7.62
TOTAL	11	28	38	86	65	71	85	81	61	46	49	87	708	6.99

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	2	1	5	1	5	2	3	3	3	25	7.96
NE	0	0	1	2	4	0	4	1	4	0	0	0	16	5.94
ENE	0	0	1	3	0	0	0	0	0	0	0	0	4	3.28
E	1	0	1	1	0	1	0	0	0	0	0	0	4	3.20
ESE	0	0	0	0	2	0	0	0	0	0	0	0	2	4.80
SE	0	2	0	0	1	0	0	0	0	1	0	0	4	4.80
SSE	0	1	1	2	0	1	2	1	1	0	0	0	9	5.13
S	0	0	1	2	2	0	0	0	0	0	0	0	5	3.76
SSW	1	0	0	0	0	0	0	0	0	0	1	0	2	5.45
SW	1	0	0	1	0	0	0	0	0	0	0	0	2	2.10
WSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
W	1	1	0	0	0	0	0	0	0	0	0	0	2	1.00
WNW	0	0	2	1	0	1	0	0	0	0	0	0	4	3.80
NW	0	0	0	2	0	3	2	3	0	1	3	2	16	7.82
NNW	1	1	0	2	3	6	6	2	5	7	7	24	64	9.95
N	0	1	3	6	9	5	5	2	4	3	9	20	67	8.41
TOTAL	5	6	10	24	23	22	20	14	16	15	23	49	227	7.81

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	1	2	2	4	2	1	3	17	8.31
NE	0	0	0	1	0	1	0	1	1	0	0	0	4	6.48
ENE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	1	0	0	0	0	0	0	1	5.70
SW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
WSW	1	0	1	0	0	0	0	0	0	0	0	0	2	1.50
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	1	0	0	0	1	0	0	0	2	6.85
NNW	0	0	0	0	1	2	3	2	4	1	1	11	25	10.83
N	0	0	2	2	1	1	5	1	3	3	4	9	31	9.55
TOTAL	1	0	5	4	5	6	10	6	13	6	6	23	85	9.01

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-18 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9.	10	11	>11	TOTAL	MEAN SPEED
NNE	4	6	9	13	10	12	16	20	22	16	11	14	153	7.08
NE	3	1	4	14	7	11	10	12	17	18	3	4	104	6.64
ENE	1	0	13	14	18	1	8	7	6	1	2	7	78	5.88
E	4	8	12	20	11	11	12	6	6	0	1	0	91	4.58
ESE	1	9	7	15	17	15	11	8	2	2	0	0	87	4.78
SE	0	7	11	16	10	16	11	3	7	3	1	0	85	4.98
SSE	2	11	58	65	51	35	34	32	17	9	13	16	343	5.45
S	9	32	70	70	79	43	26	13	3	6	3	7	361	4.30
SSW	3	35	52	40	34	26	7	8	0	0	1	0	206	3.61
SW	4	25	31	20	21	4	1	2	2	0	0	0	110	3.24
WSW	4	18	23	16	17	5	1	0	2	0	1	0	87	3.26
W	8	19	32	28	22	9	4	6	2	0	1	0	131	3.57
WNW	2	6	16	40	33	25	17	11	9	10	5	7	181	5.54
NW	0	3	14	17	42	50	58	47	35	26	32	52	376	7.58
NNW	1	9	0	15	17	28	22	20	29	23	32	55	251	8.69
N	0	7	10	24	23	14	19	11	19	12	24	42	205	7.92
CALM													1	
TOTAL	47	196	362	427	406	303	257	206	178	126	130	204	2844	5.77

Figure 2.3-10, sheet 10
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS E CONDITION IN THE FINAL ANNUAL RECORD

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	1	0	0	0	0	0	0	0	0	0	0	0	1	.90
SSE	0	2	3	0	1	0	0	0	0	0	0	0	6	2.67
S	0	8	10	1	7	2	1	0	1	0	0	0	30	3.23
SSW	1	3	8	2	3	0	0	0	0	0	0	0	17	2.59
SW	0	5	5	0	0	0	0	0	1	0	0	0	11	2.68
WSW	0	3	3	1	0	0	0	0	0	0	0	0	7	2.24
W	0	6	7	5	0	0	0	0	0	0	0	0	18	2.44
WNW	0	2	3	0	3	0	0	0	0	0	0	0	8	3.02
NW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.50
NNW	1	0	0	0	1	0	0	0	0	0	0	0	2	2.60
N	0	0	1	0	0	0	0	0	0	0	0	0	1	2.50
TOTAL	3	29	40	10	15	2	1	0	2	0	0	0	102	2.77

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.10
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.10
SSE	0	3	3	1	0	0	0	0	0	0	0	0	7	2.30
S	0	10	7	4	3	0	0	0	0	0	0	0	24	2.59
SSW	3	10	5	1	2	0	0	0	0	0	0	0	21	2.01
SW	3	1	2	0	1	0	0	0	0	0	0	0	7	1.69
WSW	1	0	2	0	0	0	0	0	0	0	0	0	3	1.90
W	0	0	2	3	0	0	0	0	0	0	0	0	5	3.18
WNW	0	1	2	1	3	0	0	0	0	0	0	0	7	3.36
NW	0	0	2	0	1	0	0	0	0	0	0	0	3	3.40
NNW	0	1	0	0	1	0	0	0	0	0	0	0	2	2.70
N	0	0	0	0	0	0	0	0	0	0	0	0	0	1
TOTAL	7	26	25	12	11	0	0	0	0	0	0	0	81	2.48

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	2	0	0	1	0	0	0	0	0	0	0	3	2.50
SE	0	0	1	0	1	0	0	0	0	0	1	0	3	5.77
SSE	0	5	6	3	1	1	0	0	0	0	0	0	16	2.77
S	1	15	15	5	7	3	4	0	0	0	0	0	50	3.18
SSW	1	11	7	0	4	1	0	0	0	0	0	0	24	2.37
SW	2	8	8	0	0	0	0	0	0	0	0	0	18	1.86
WSW	1	2	3	0	1	1	0	1	0	0	0	0	9	3.30
W	0	6	6	3	0	0	0	0	0	0	0	0	15	2.43
WNW	1	1	3	3	4	3	0	0	0	0	0	0	15	3.62
NW	1	0	1	0	2	1	0	0	1	0	0	0	6	4.52
NNW	1	1	1	1	0	1	0	1	0	0	0	0	6	3.48
N	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
TOTAL	9	51	51	15	22	11	4	2	1	0	1	0	167	2.9[illegible]

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	1	1	1	0	0	0	0	0	0	0	0	0	3	1.43
NE	0	0	1	3	0	1	0	0	0	0	0	0	5	3.80
ENE	0	0	0	2	0	0	0	0	0	0	0	0	2	3.60
E	1	0	0	0	0	0	0	0	0	0	0	0	1	.80
ESE	0	0	0	0	0	0	0	0	0	2	0	0	2	9.20
SE	0	1	2	1	0	0	0	0	0	1	1	1	7	5.90
SSE	0	3	11	4	1	3	0	0	0	0	0	0	22	3.20
S	1	5	9	9	3	1	1	1	0	2	0	0	32	3.68
SSW	3	9	7	4	1	1	0	0	0	0	0	0	25	2.36
SW	3	7	5	1	0	0	0	0	0	0	0	0	16	1.8[illegible]
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
W	1	6	10	2	2	2	2	0	0	0	0	0	25	3.13
WNW	1	0	4	5	3	1	1	1	1	0	0	0	17	4.16
NW	0	0	1	1	7	6	1	0	2	0	2	3	23	6.60
NNW	1	0	0	0	2	2	0	0	0	0	0	0	5	4.04
N	0	1	2	1	0	0	0	0	2	0	0	0	6	4.57
TOTAL	12	33	53	33	19	17	5	2	5	5	3	4	191	3.75

Figure 2.3-10, sheet 11
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS F CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	1	0	0	2	1	0	1	0	1	0	0	7	4.77
NE	1	0	2	0	0	2	0	0	0	0	0	0	5	3.34
ENE	1	1	0	0	0	0	0	0	0	0	0	0	2	.80
E	0	2	3	0	0	0	0	0	0	0	0	0	5	2.12
ESE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
SE	0	1	1	0	0	0	0	0	0	0	0	0	2	1.75
SSE	0	2	7	2	4	2	0	0	0	0	0	0	17	3.42
S	0	10	5	2	1	0	0	0	0	0	1	0	19	2.71
SSW	1	4	2	0	0	0	2	0	0	0	0	0	9	2.69
SW	1	2	2	0	0	0	0	0	0	0	0	0	5	1.80
WSW	3	1	1	1	0	0	0	0	0	0	0	0	6	1.62
W	4	2	0	2	0	1	1	0	0	0	0	0	10	2.46
WNW	0	1	2	6	4	1	1	0	0	1	0	0	16	4.16
NW	1	2	0	2	7	7	7	3	1	2	0	4	36	6.44
NNW	0	1	1	0	2	1	0	1	0	2	2	1	11	7.04
N	1	3	0	2	0	0	0	2	0	0	1	0	9	4.12
TOTAL	14	33	27	19	20	15	11	7	1	6	4	5	162	4.11

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	1	0	0	0	1	8.30
NE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.60
ENE	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
E	0	1	0	1	0	0	0	0	0	0	0	0	2	2.55
ESE	0	0	0	0	0	0	0	0	0	1	0	0	1	9.10
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	2	0	0	0	0	0	0	0	0	0	0	0	2	.85
S	1	0	1	0	0	0	0	0	0	0	0	0	2	1.70
SSW	1	1	0	0	0	0	0	0	0	0	0	0	2	.80
SW	1	2	0	0	0	0	0	0	0	0	0	0	3	1.10
WSW	1	0	0	0	0	0	0	0	0	0	0	0	1	.80
W	0	0	0	0	0	1	0	0	0	0	0	0	1	6.00
WNW	0	1	1	0	0	0	0	0	0	0	0	0	2	2.85
NW	0	1	0	0	0	1	0	0	0	1	0	0	3	5.87
NNW	0	0	0	0	0	0	0	0	1	0	1	0	2	9.50
N	0	0	1	0	0	0	1	1	1	1	0	0	5	6.74
TOTAL	7	7	3	1	0	2	1	1	3	3	1	0	29	3.92

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	4.20
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	1	0	0	0	0	0	0	0	0	0	0	0	1	.20
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	1	0	0	0	0	0	0	0	0	0	0	0	1	.40
S	2	0	0	0	0	0	0	0	0	0	0	0	2	.30
SSW	1	0	0	0	0	0	0	0	0	0	0	0	1	.40
SW	1	0	1	0	0	0	0	0	0	0	0	0	2	1.50
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	1	1	0	0	0	0	0	0	0	2	4.00
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
N	0	0	0	0	0	1	2	0	0	0	0	1	4	8.08
TOTAL	6	0	1	2	3	1	2	0	0	0	0	1	16	3.63

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	3	2	1	1	3	1	0	1	1	1	0	0	14	3.96
NE	1	1	3	4	0	3	0	0	0	0	0	0	12	3.37
ENE	2	1	0	2	0	0	1	0	0	0	0	0	6	2.65
E	1	3	3	1	0	0	0	0	0	0	0	0	8	2.06
ESE	1	2	2	0	1	0	0	0	0	3	0	0	9	4.44
SE	1	2	4	2	1	0	0	0	0	1	2	1	14	4.72
SSE	4	17	27	11	6	5	0	0	0	1	0	1	72	3.03
S	5	47	49	17	22	3	6	2	0	2	0	0	153	2.97
SSW	11	39	22	9	15	1	2	0	0	0	0	0	99	2.39
SW	11	24	23	2	2	0	0	1	0	0	0	0	63	2.01
WSW	6	8	9	3	3	1	0	1	0	0	0	0	31	2.49
W	3	20	23	16	3	5	3	0	0	0	0	0	73	2.89
WNW	2	7	15	20	17	5	3	1	1	1	0	0	72	3.75
NW	2	3	4	5	18	15	8	3	4	3	2	7	74	6.06
NNW	3	3	2	1	7	4	0	2	1	2	3	1	29	5.28
N	1	4	4	3	1	1	3	3	3	1	1	1	26	5.28
CALM													1	
TOTAL	58	183	191	97	99	44	26	14	10	15	8	11	756	3.40

Figure 2.3-10, sheet 12
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS F CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.60
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	1	0	0	0	0	0	0	0	1	4.50
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.90
S	1	4	4	4	1	0	0	0	0	0	0	0	14	2.60
SSW	1	6	2	3	0	0	0	0	0	0	0	0	12	2.28
SW	3	5	3	1	0	0	0	0	1	0	0	0	13	2.39
WSW	0	7	3	0	1	0	0	0	0	0	0	0	11	2.15
W	1	6	5	1	0	0	0	0	0	0	0	0	13	2.02
WNW	0	2	4	1	0	0	0	0	0	0	0	0	7	2.61
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.10
N	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
TOTAL	6	32	22	11	3	0	0	0	1	0	0	0	75	2.35

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	1	0	0	0	0	0	0	0	0	0	2	1.55
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	1	1	0	0	0	0	0	0	0	0	2	2.60
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.70
SSE	0	2	0	0	0	0	0	0	0	0	0	0	2	1.65
S	0	10	4	0	3	0	0	0	0	0	0	0	17	2.41
SSW	5	8	2	2	1	0	0	0	0	0	0	0	18	1.79
SW	3	5	2	0	0	0	0	0	0	0	0	0	10	1.55
WSW	1	10	1	1	0	0	0	0	0	0	0	0	13	1.65
W	2	7	5	1	0	0	0	0	0	0	0	0	15	1.69
WNW	1	4	5	1	0	0	0	0	0	0	0	0	11	2.16
NW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.40
N	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
TOTAL	13	47	22	6	6	0	0	0	0	0	0	0	94	2.01

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.80
E	0	2	0	2	0	0	0	0	0	0	0	0	4	2.72
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.50
SSE	0	5	1	1	1	0	0	0	0	0	0	0	8	2.14
S	4	23	12	6	5	0	0	0	0	0	0	0	50	2.24
SSW	2	29	8	8	1	0	0	0	0	0	0	0	48	1.64
SW	1	15	6	1	0	0	0	0	0	0	0	0	23	1.82
WSW	1	12	7	0	0	0	0	0	0	0	0	0	20	1.88
W	5	7	12	2	0	0	0	0	0	0	0	0	26	2.09
WNW	0	4	6	3	1	0	0	0	0	0	0	0	14	2.50
NW	0	0	2	3	3	0	0	0	1	0	0	0	9	4.03
NNW	0	0	1	1	0	0	1	0	0	0	0	0	3	4.43
N	0	0	2	0	1	0	0	0	0	0	0	0	3	3.10
TOTAL	13	98	58	28	12	0	1	0	1	0	0	0	205	2.20

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	2	1	0	0	0	0	0	0	0	0	3	2.97
NE	1	1	0	0	1	1	0	0	0	0	0	0	4	2.92
ENE	0	0	3	1	1	0	0	0	0	0	0	0	5	3.26
E	0	2	1	0	0	1	0	0	0	0	0	0	4	2.97
ESE	0	1	1	0	0	0	0	0	0	0	0	0	2	1.80
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.40
SSE	2	4	3	3	0	0	0	0	0	0	0	0	12	2.00
S	3	23	35	11	4	0	0	1	0	0	0	0	77	2.49
SSW	5	55	21	3	0	1	0	0	0	0	0	0	85	[illegible]
SW	2	37	19	2	0	0	0	0	0	0	0	0	60	1.97
WSW	3	10	5	1	0	0	0	0	0	0	0	0	19	1.77
W	3	14	13	4	0	1	1	0	0	0	0	0	36	2.35
WNW	1	1	5	3	0	2	0	0	0	0	0	0	12	3.05
NW	1	1	0	2	3	0	0	0	0	0	0	0	7	3.20
NNW	0	0	0	3	1	1	1	1	0	0	0	0	7	4.93
N	0	2	0	0	0	0	0	0	0	0	0	0	2	1.50
TOTAL	21	152	108	34	10	7	2	2	0	0	0	0	336	2.24

Figure 2.3-10, sheet 13
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS G CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970
(WITH 1971 DATA SUBSTITUTIONS)

(Rev. 3/3/72)

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	1	0	0	0	0	0	0	0	0	0	2	1.85
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.40
ENE	0	0	1	0	1	0	0	0	0	0	0	0	2	3.60
E	1	0	1	0	0	0	0	0	0	0	0	0	2	1.85
ESE	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
SE	0	2	2	0	0	0	0	0	0	0	0	0	4	2.37
SSE	0	3	3	0	0	0	0	0	0	0	0	0	6	2.02
S	3	23	24	4	0	0	0	0	0	0	0	0	54	2.08
SSW	5	52	17	2	0	0	0	0	0	0	0	0	76	1.76
SW	7	38	16	2	0	0	0	0	0	0	0	0	63	1.79
WSW	3	11	8	2	1	1	0	0	0	0	0	0	26	2.12
W	4	10	2	2	3	1	1	0	0	0	0	0	23	2.40
WNW	3	1	0	6	2	0	1	0	0	0	1	0	14	3.71
NW	0	0	0	3	1	1	0	0	0	0	0	1	6	5.43
NNW	0	1	1	0	1	0	0	0	1	1	0	0	5	5.38
N	2	2	0	0	0	0	0	0	0	0	0	0	4	1.02
TOTAL	29	144	76	22	9	3	2	0	1	1	1	1	259	2.16

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	2	0	0	0	0	0	0	0	0	0	2	2.40
ESE	2	3	0	0	0	0	0	0	0	0	0	0	5	1.20
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.80
SSE	1	0	0	0	0	0	0	0	0	0	0	0	1	.30
S	2	9	2	0	0	0	0	0	0	0	0	0	13	1.58
SSW	2	16	3	0	0	0	0	0	0	0	0	0	21	1.60
SW	8	24	6	1	0	0	0	0	0	0	0	0	39	1.52
WSW	2	5	0	1	0	0	0	0	0	0	0	0	8	1.55
W	1	3	0	0	0	1	0	0	0	0	0	0	5	2.22
WNW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.30
NW	0	0	1	0	1	1	0	0	0	0	0	0	3	4.33
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
N	2	0	0	0	0	1	0	0	0	0	0	0	3	2.33
TOTAL	20	62	14	2	2	3	0	0	0	0	0	0	103	1.71

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	2	0	0	0	0	0	0	0	0	0	0	0	2	.15
E	0	4	1	0	0	0	0	0	0	0	0	0	5	1.96
ESE	3	0	0	0	0	0	0	0	0	0	0	0	3	.37
SE	0	3	0	0	0	0	0	0	0	0	0	0	3	1.63
SSE	0	2	0	0	0	0	0	0	0	0	0	0	2	1.45
S	0	0	1	1	0	0	0	1	0	0	0	0	3	4.53
SSW	7	16	3	0	0	0	0	0	0	0	0	0	26	1.38
SW	11	35	3	0	0	0	0	0	0	0	0	0	49	1.43
WSW	1	4	0	0	0	2	1	0	0	0	0	0	8	2.95
W	1	4	0	0	0	1	0	0	0	0	0	0	6	2.17
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
N	3	1	0	0	0	0	0	0	0	0	0	0	4	.67
TOTAL	28	69	8	1	0	3	1	1	0	0	0	0	111	1.60

PASQUILL G (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	1	6	1	0	0	0	0	0	0	0	0	9	2.36
NE	1	1	0	1	1	1	0	0	0	0	0	0	5	3.02
ENE	2	0	5	3	2	0	0	0	0	0	0	0	12	2.72
E	1	8	5	2	1	1	0	0	0	0	0	0	18	2.47
ESE	6	4	1	0	0	0	0	0	0	0	0	0	11	1.06
SE	0	9	2	0	0	0	0	0	0	0	0	0	11	1.89
SSE	3	17	7	4	1	0	0	0	0	0	0	0	32	1.92
S	13	92	82	26	13	0	0	2	0	0	0	0	228	2.31
SSW	27	163	56	12	2	0	2	1	0	0	0	0	263	1.81
SW	35	158	56	8	0	0	2	1	1	0	0	0	261	1.79
WSW	11	59	24	8	4	3	1	0	0	0	0	0	110	2.05
W	17	51	38	11	9	4	2	0	0	0	0	0	132	2.32
WNW	5	13	20	14	4	3	1	1	0	0	1	0	62	2.99
NW	1	1	3	8	9	2	1	0	2	1	0	1	29	4.64
NNW	0	2	2	4	4	1	2	1	1	1	0	0	18	4.73
N	7	5	3	1	1	1	0	0	0	0	0	0	18	1.79
CALM													1	
TOTAL	131	604	310	103	51	16	11	6	4	2	1	1	1240	2.17

Figure 2.3-10, sheet 14
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS G CONDITION IN THE FINAL ANNUAL RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)

DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
S	0	1	0	0	1	2	2	0	0	0	0	0	6	5.10
SSW	0	0	0	0	1	5	1	0	0	0	0	0	7	5.73
SW	0	0	0	5	1	1	0	0	0	0	0	0	7	4.06
WSW	0	0	3	2	2	3	0	0	0	0	0	0	10	4.00
W	0	2	1	2	2	6	0	0	0	0	0	0	13	4.31
WNW	0	0	1	0	0	0	1	0	0	0	0	0	2	4.75
NW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.50
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	0	3	6	11	7	17	4	0	0	0	0	0	48	4.46

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.75
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
S	0	0	0	1	2	4	1	0	0	0	0	0	8	5.00
SSW	1	1	0	0	0	0	0	0	0	0	0	0	2	1.15
SW	0	0	0	0	0	1	0	0	0	0	0	0	1	5.10
WSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.90
W	0	0	0	0	2	1	0	0	0	0	0	0	3	4.80
WNW	0	0	0	1	0	1	0	0	0	0	0	0	2	4.15
NW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
NNW	0	0	0	0	0	0	0	0	0	1	0	0	1	9.50
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	1	1	2	3	6	7	1	0	0	1	0	0	22	4.38

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	3	0	0	0	0	0	0	0	3	4.43
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.70
ENE	0	0	0	0	2	0	0	0	0	0	0	0	2	4.40
E	0	0	0	1	0	1	0	0	0	0	0	0	2	4.65
ESE	0	0	0	0	0	1	1	0	0	0	0	0	2	5.60
SE	0	0	0	1	0	1	0	0	0	1	0	0	3	6.27
SSE	0	0	0	0	2	0	0	1	1	0	0	0	4	6.27
S	0	0	0	0	2	0	0	0	0	0	0	0	2	4.25
SSW	0	2	0	1	0	0	0	0	0	0	0	0	3	1.63
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
W	0	1	0	0	1	0	0	0	0	0	0	0	2	3.25
WNW	0	0	1	1	2	2	1	0	1	1	0	0	9	5.66
NW	0	0	0	1	0	0	2	4	1	0	2	0	10	7.59
NNW	0	0	0	1	0	2	0	0	0	1	0	1	5	7.36
N	0	0	0	0	1	1	0	0	0	0	0	0	2	5.25
TOTAL	0	3	2	7	13	8	4	5	3	3	2	1	51	5.67

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	0	0	8	5	5	1	0	0	0	0	20	5.30
NE	0	0	0	5	6	4	4	0	2	0	0	0	21	5.30
ENE	0	0	0	0	7	2	1	0	1	0	0	0	11	5.25
E	0	0	0	2	7	3	4	3	3	0	0	0	22	6.04
ESE	0	0	0	0	2	2	0	3	2	1	0	0	10	6.91
SE	0	0	1	3	0	0	0	1	1	1	0	0	7	5.40
SSE	0	0	3	2	3	0	1	1	4	0	0	0	14	5.45
S	0	2	2	1	0	1	0	0	0	0	0	0	6	3.02
SSW	0	0	0	0	1	0	1	0	0	0	0	0	2	5.55
SW	1	3	0	0	0	0	0	0	0	0	0	0	4	1.30
WSW	0	1	2	0	0	0	0	0	0	0	0	0	3	2.20
W	0	1	2	1	2	0	1	0	0	0	0	0	7	3.74
WNW	0	0	0	3	4	2	2	0	1	0	0	0	12	5.07
NW	0	0	1	0	3	1	3	5	5	4	4	8	34	8.67
NNW	0	0	0	0	0	5	1	2	2	5	1	8	24	9.71
N	0	0	0	2	1	2	4	8	2	2	1	0	22	7.11
TOTAL	1	9	11	19	44	27	27	24	23	13	6	16	219	6.44

Figure 2.3-10, sheet 15
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS A CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

(Rev. 3/3/72)

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	3	4	8	9	6	6	0	3	2	42	6.68
NE	0	0	0	0	4	1	4	3	0	0	0	0	12	6.10
ENE	0	0	0	0	3	4	1	0	0	0	0	0	8	5.16
E	0	0	0	2	0	5	2	2	1	1	0	0	13	6.81
ESE	0	0	0	0	2	0	1	3	1	1	0	0	8	7.27
SE	0	0	0	1	1	3	2	2	3	1	0	0	13	6.94
SSE	0	0	0	2	0	2	0	2	1	1	0	0	8	6.40
S	0	1	0	1	0	0	0	0	0	0	0	0	2	2.50
SSW	0	0	0	0	0	0	0	0	0	1	0	0	1	9.40
SW	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
WSW	0	1	0	0	1	0	0	0	0	0	0	0	2	3.20
W	0	0	1	1	3	0	0	1	0	0	0	0	6	4.47
WNW	0	0	0	2	3	0	0	1	0	1	1	0	8	6.01
NW	0	0	0	0	0	1	1	2	2	2	10	7	25	10.29
NNW	0	0	0	0	0	2	1	0	0	6	5	9	23	10.49
N.	0	0	0	0	4	13	16	10	7	4	3	10	67	8.80
TOTAL	1	2	2	12	25	39	37	32	21	18	22	28	239	7.64

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	1	2	5	1	2	2	2	16	6.37
NE	0	0	0	4	0	2	1	1	1	0	0	0	9	5.34
ENE	0	0	0	1	1	0	1	0	0	0	0	0	3	5.20
E	0	0	0	1	0	1	1	0	0	0	0	0	3	4.97
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	1	1	0	0	0	2	7.65
S	1	0	0	0	0	0	0	0	0	0	0	0	1	.60
SSW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.60
SW	0	1	0	0	0	0	0	0	0	0	0	0	1	2.00
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	1	1	16.20
NNW	0	0	0	0	0	1	0	1	3	4	1	9	19	10.67
N	0	0	0	0	3	1	1	2	5	6	5	8	31	9.73
TOTAL	1	2	0	6	5	6	6	10	11	12	8	20	87	8.65

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-18 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	2	0	0	1	0	0	1	4	8.57
NE	0	0	0	0	1	1	0	0	0	0	0	0	2	5.15
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.20
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	1	2	3	0	0	6	8.65
N	0	0	0	1	2	5	2	2	1	1	1	2	17	7.16
TOTAL	0	0	1	1	3	9	2	3	4	4	1	3	31	7.26

PASQUILL A (FROM AEC/DELTA T CRITERIA, 50-18 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	1	4	17	16	16	12	8	2	5	5	87	6.68
NE	0	0	0	10	11	8	9	4	3	0	0	0	45	5.48
ENE	0	0	0	1	13	6	3	0	1	0	0	0	24	5.15
E	0	0	0	6	7	11	7	5	4	1	0	0	41	5.93
ESE	0	0	0	1	4	3	2	6	4	2	0	0	22	6.84
SE	0	0	1	5	1	4	2	3	4	3	0	0	23	6.38
SSE	0	0	5	4	5	2	1	5	7	1	0	0	30	5.78
S	1	4	2	3	5	7	3	0	0	0	0	0	25	4.14
SSW	1	4	0	1	2	5	2	0	0	1	0	0	16	4.37
SW	2	4	0	5	1	2	0	0	0	0	0	0	14	2.98
WSW	0	2	7	2	4	3	0	0	0	0	0	0	18	3.45
W	0	4	4	4	10	7	1	1	0	0	0	0	31	4.19
WNW	0	0	2	7	9	5	4	1	2	2	1	0	33	5.44
NW	0	0	2	2	3	2	6	11	8	6	16	16	72	9.12
NNW	0	0	0	1	0	10	2	4	7	20	7	27	78	9.94
N	0	0	0	3	11	22	23	22	15	13	10	20	139	8.20
CALM													0	
TOTAL	4	19	24	59	103	113	81	74	63	51	39	68	698	6.91

Figure 2.3-10, sheet 16
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS A CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
NE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	1	0	0	1	0	0	0	2	7.10
SE	0	0	0	0	0	0	1	1	0	0	0	0	2	7.40
SSE	0	0	0	0	0	0	0	0	0	1	0	0	1	9.60
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.10
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	1	1	0	0	0	0	0	0	0	2	4.25
WNW	0	0	0	0	1	1	0	0	0	0	0	0	2	5.30
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	2	0	0	0	0	0	1	1	4	7.88
N	0	0	0	1	1	0	0	0	0	0	0	0	2	4.85
TOTAL	0	0	0	3	7	2	1	1	1	1	1	1	18	6.18

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.50
ESE	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.50
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	1	0	0	0	0	1	7.60
NW	0	0	0	0	0	0	1	0	0	0	0	1	2	8.80
NNW	0	0	0	0	0	0	0	0	0	0	0	1	1	19.00
N	0	0	0	0	1	0	0	0	0	0	0	0	1	4.70
TOTAL	0	0	0	0	2	1	2	1	0	0	0	2	8	8.24

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.50
NE	0	0	0	1	0	0	0	0	0	0	1	0	2	7.00
ENE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.40
ESE	0	0	0	0	0	0	0	0	0	1	0	0	1	9.50
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	1	1	0	0	0	0	0	0	0	0	2	3.40
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.80
NW	0	0	1	0	0	1	0	1	0	0	0	0	3	5.20
NNW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.60
N	0	0	0	0	1	0	1	1	0	0	0	0	3	6.53
TOTAL	0	0	2	4	2	2	2	2	0	1	1	0	16	5.61

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	1	0	0	0	0	2	6.05
NE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.95
ENE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.30
E	0	0	1	0	0	0	2	0	0	0	0	0	3	5.23
ESE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.50
SE	0	0	0	0	0	0	0	0	0	0	1	0	1	10.10
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.90
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	1	0	0	1	0	1	0	0	0	0	0	3	4.07
WNW	0	0	0	0	0	1	0	1	0	0	0	0	2	8.55
NW	0	0	0	0	0	0	0	0	0	0	0	1	1	12.70
NNW	0	0	0	0	0	0	0	1	1	1	0	3	6	11.28
N	0	0	0	1	0	1	3	1	0	0	0	0	6	6.08
TOTAL	0	1	1	2	4	2	8	4	1	1	1	4	29	7.08

Figure 2.3-10, sheet 17
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS B CONDITION IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)

DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	0	1	1	0	0	0	0	4	5.77
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.40
ENE	0	0	0	0	0	1	0	0	0	0	0	0	1	6.00
E	0	0	0	0	0	1	0	0	0	0	0	0	1	5.10
ESE	0	0	0	0	0	1	0	0	0	0	0	0	1	6.00
SE	0	0	0	0	0	1	0	0	1	0	0	0	2	7.10
SSE	0	0	0	0	0	0	1	0	1	0	0	0	2	7.90
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	2	0	0	0	0	0	0	0	2	4.55
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	1	1	1	5	8	12.24
NNW	0	0	1	0	0	0	0	0	1	0	1	2	5	11.10
N	0	0	0	2	0	4	1	0	0	0	0	0	7	5.14
TOTAL	0	0	1	4	3	8	3	1	4	1	2	7	34	8.00

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	1	1	0	0	0	0	2	6.90
NE	0	0	0	1	0	1	1	0	0	1	0	0	4	6.40
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
ESE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.40
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.70
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.50
NW	0	0	0	0	0	0	0	1	0	0	0	0	1	7.10
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	1	0	0	2	0	0	1	1	2	7	9.23
TOTAL	0	1	0	2	1	1	6	2	0	2	1	2	18	7.21

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.60
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	1	0	0	0	0	0	1	2	9.60
N	0	0	0	0	0	0	0	0	1	0	0	0	1	8.70
TOTAL	0	0	1	0	0	1	1	0	1	0	0	1	5	7.36

PASQUILL B (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	3	0	3	3	0	0	0	0	10	6.01
NE	0	0	1	4	2	1	1	0	0	1	1	0	11	5.30
ENE	0	0	0	1	0	1	1	0	0	0	0	0	3	5.43
E	0	0	1	0	0	3	3	0	0	0	0	0	7	5.40
ESE	0	0	0	0	2	2	2	0	1	1	0	0	8	6.46
SE	0	0	0	0	0	1	1	1	1	0	1	0	5	7.62
SSE	0	0	0	0	0	0	2	0	1	1	0	0	4	8.08
S	0	0	1	1	0	0	0	0	0	0	0	0	2	3.40
SSW	0	1	0	1	1	0	0	0	0	0	0	0	3	3.83
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	0.
WSW	0	0	0	0	2	0	0	0	0	0	0	0	2	4.55
W	0	1	0	1	2	0	1	0	0	0	0	0	5	4.14
WNW	0	0	0	0	2	2	1	2	0	0	0	0	7	6.09
NW	0	0	1	0	0	1	1	2	1	1	1	7	15	10.06
NNW	0	0	1	1	2	1	0	1	2	1	2	8	19	10.34
N	0	0	0	5	3	5	7	2	1	1	1	2	27	6.60
CALM													0	
TOTAL	0	2	5	15	19	17	23	11	7	6	6	17	128	7.11

Figure 2.3-10, sheet 18
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS B CONDITION IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)

DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.40
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	1	0	3	1	0	0	0	0	0	0	5	4.32
SSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.20
SW	0	0	0	2	3	1	0	0	0	0	0	0	6	4.30
WSW	0	0	1	0	1	0	0	1	0	0	0	0	3	4.70
W	0	0	0	2	0	0	1	0	0	0	0	0	3	4.60
WNW	0	0	0	0	1	1	0	1	0	0	0	0	3	5.80
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	0	0	2	5	9	4	1	2	0	0	0	0	23	4.62

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	1	1	0	0	0	0	0	0	2	5.35
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	1	0	0	0	0	0	0	0	0	0	1	2.40
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
WNW	0	0	1	0	0	2	1	0	0	0	0	0	4	4.95
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	0	0	2	1	1	3	1	0	0	0	0	0	8	4.51

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.60
ENE	0	0	0	2	0	0	0	0	0	0	0	0	2	3.55
E	0	0	0	2	0	0	0	0	0	0	0	0	2	3.70
ESE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.40
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	1	0	0	1	1	0	0	0	0	0	0	3	3.90
SSW	0	1	0	0	1	0	0	0	0	0	0	0	2	2.65
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	1	1	0	0	0	0	0	0	0	2	4.00
W	0	0	0	0	0	0	0	1	0	0	0	0	1	7.10
WNW	0	0	0	0	0	1	0	0	0	1	0	0	2	7.65
NW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.10
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	1	1	12.00
TOTAL	0	2	1	5	3	3	1	1	0	1	0	1	18	4.92

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	1	2	8.40
NE	0	0	0	2	1	0	0	0	0	0	0	0	3	3.77
ENE	0	0	0	2	0	1	0	0	0	0	0	0	3	4.10
E	0	0	1	2	6	2	2	2	1	0	0	0	16	5.23
ESE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.20
SE	0	0	0	0	0	3	2	0	0	0	0	0	5	6.24
SSE	0	0	0	0	0	1	0	0	1	1	1	2	6	9.98
S	0	0	0	0	0	1	0	1	0	0	0	0	2	6.15
SSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	1	0	1	0	0	0	1	3	8.13
NW	0	0	0	0	0	0	1	0	1	2	1	7	12	10.80
NNW	0	0	0	2	1	0	0	0	1	0	0	1	5	7.16
N	0	0	0	0	2	0	1	1	2	0	0	0	6	6.42
TOTAL	0	0	1	8	12	10	6	5	6	3	2	12	65	7.16

Figure 2.3-10, sheet 19
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS C CONDITION IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	2	1	0	1	0	0	0	0	5	5.30
NE	0	0	0	0	2	0	0	0	0	0	1	0	3	6.57
ENE	0	0	0	2	1	0	0	1	0	0	0	0	4	4.97
E	0	0	1	0	0	2	1	0	0	0	0	0	4	5.08
ESE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.30
SE	0	0	0	0	0	0	0	2	0	0	1	1	4	9.50
SSE	0	1	0	1	0	0	0	1	0	1	0	1	5	6.66
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	1	0	0	0	0	0	0	0	0	1	3.70
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	2	1	0	6	9	11.78
NNW	1	0	0	1	0	0	1	0	0	0	0	5	8	11.39
N	0	0	0	0	1	0	1	1	0	1	0	1	5	8.56
TOTAL	1	1	1	6	6	4	3	6	2	3	2	14	49	8.32

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	2	0	0	0	0	0	1	5	6.12
NE	0	0	0	0	0	0	0	0	0	1	1	0	2	10.45
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	1	0	1	10.30
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	1	0	0	0	0	0	1	6.30
NNW	0	0	0	0	1	0	0	0	0	1	1	2	5	12.12
N	0	0	0	1	3	1	0	0	2	0	3	4	14	9.27
TOTAL	0	0	0	2	5	3	1	0	2	2	6	7	28	9.23

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	0	0	0	0	0	1	0	1	3	9.07
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	2	0	0	0	0	1	1	4	9.25
TOTAL	0	0	0	1	0	2	0	0	0	1	1	2	7	9.17

PASQUILL C (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	3	4	3	0	1	0	1	0	3	15	6.74
NE	0	0	0	2	3	1	0	0	0	1	2	0	9	6.41
ENE	0	0	0	7	1	1	0	1	0	0	0	0	10	4.33
E	0	0	2	4	6	4	3	2	1	0	0	0	22	5.05
ESE	0	0	1	0	0	2	0	0	0	0	0	0	3	4.30
SE	0	0	0	0	0	4	2	2	0	0	1	1	10	7.46
SSE	0	1	0	1	0	1	0	1	1	2	2	3	12	8.72
S	0	1	1	0	5	4	0	1	0	0	0	0	12	4.69
SSW	0	1	0	0	3	0	0	0	0	0	0	0	4	3.62
SW	0	0	1	2	3	1	0	0	0	0	0	0	7	4.03
WSW	0	0	1	1	2	0	0	1	0	0	0	0	5	4.42
W	0	0	0	4	0	0	1	1	0	0	0	0	6	4.63
WNW	0	0	1	0	1	5	1	2	0	1	0	1	12	6.41
NW	0	0	0	0	0	0	3	0	3	3	1	13	23	10.76
NNW	1	0	0	3	2	0	1	0	1	1	1	8	18	10.42
N	0	0	0	1	6	3	2	2	4	1	4	7	30	8.67
CALM													0	
TOTAL	1	3	7	28	36	29	13	14	10	10	11	36	198	7.21

Figure 2.3-10, sheet 20
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS C CONDITION IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)

DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.50
NE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	1	0	0	0	0	0	0	0	0	0	1	2.80
ESE	0	0	0	1	0	0	1	0	0	0	0	0	2	5.05
SE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.90
SSE	0	0	1	2	2	0	0	0	0	0	0	0	5	3.64
S	1	1	8	4	2	3	0	0	0	0	0	1	20	3.75
SSW	0	1	1	9	15	3	0	0	0	0	1	0	30	4.37
SW	0	4	8	6	11	3	0	0	1	0	0	0	33	3.80
WSW	1	1	7	10	7	2	0	0	0	0	0	0	28	3.49
W	0	3	5	8	4	5	2	0	0	0	0	0	27	3.90
WNW	0	0	1	0	0	1	0	1	0	0	1	0	4	6.57
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.40
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	2	11	33	41	48	17	3	1	1	0	2	1	154	3.92

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	4.85
NE	0	0	0	2	2	0	0	0	0	0	0	0	4	3.95
ENE	0	0	0	0	0	2	0	0	0	0	0	0	2	5.90
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	1	0	0	0	1	0	1	0	0	0	0	3	4.93
SE	0	0	0	0	0	0	1	0	0	0	0	0	1	6.30
SSE	1	0	1	2	0	0	0	0	0	0	0	0	4	2.55
S	0	2	5	1	3	5	1	0	1	0	0	1	19	4.66
SSW	0	1	2	0	1	1	0	0	0	0	0	0	5	3.50
SW	0	1	2	1	0	1	0	0	0	0	0	0	5	3.18
WSW	1	0	0	2	0	0	0	0	0	0	0	0	3	2.77
W	0	1	1	1	1	1	0	0	0	0	0	0	5	3.66
WNW	0	1	3	0	0	2	0	0	0	1	0	0	7	4.83
NW	0	0	0	1	1	0	0	0	1	1	0	1	5	7.64
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	2	7	14	11	9	13	2	1	2	2	0	2	65	4.38

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	0	1	1	0	0	0	0	0	0	0	2	3.95
NE	0	2	0	2	1	0	2	1	0	1	0	0	9	5.08
ENE	0	0	1	2	1	0	0	0	0	0	0	0	4	3.68
E	0	0	0	2	0	1	0	0	0	0	0	0	3	4.20
ESE	1	0	0	1	2	2	0	2	1	0	0	0	9	5.39
SE	1	0	0	1	2	3	2	0	1	0	0	1	11	5.83
SSE	0	1	1	1	5	3	2	2	1	0	0	2	18	6.14
S	1	2	1	3	5	2	5	0	0	0	0	0	19	4.38
SSW	1	1	0	0	0	0	1	0	0	0	0	0	3	3.00
SW	0	0	1	1	1	1	0	0	0	0	0	0	4	4.20
WSW	0	0	1	3	2	3	0	0	0	0	0	0	9	4.17
W	0	2	1	3	3	1	1	1	0	0	0	0	12	4.17
WNW	0	0	2	4	2	4	2	2	2	2	0	1	21	6.00
NW	0	0	2	3	4	3	2	3	1	3	2	2	25	6.80
NNW	0	0	2	0	0	0	0	0	0	3	0	0	5	6.96
N	0	0	0	3	1	0	1	0	2	0	0	1	8	6.12
TOTAL	4	8	12	30	30	23	18	11	8	9	2	7	168	5.43

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
	UPPER CLASS INTERVALS OF WIND SPEED (MPH)													
NNE	0	0	2	2	1	1	1	2	0	0	0	0	9	4.92
NE	0	1	1	4	3	3	3	1	1	5	0	0	17	4.81
ENE	0	1	0	7	2	3	2	0	0	0	0	0	15	4.43
E	0	0	5	11	8	6	6	3	0	0	0	0	39	4.65
ESE	0	1	1	2	5	6	5	3	1	0	0	0	24	5.47
SE	0	0	0	0	0	5	4	5	2	0	0	0	16	6.68
SSE	0	1	3	0	4	6	7	7	3	5	1	6	43	7.58
S	1	0	4	2	6	0	1	0	0	0	0	0	14	3.86
SSW	0	0	0	0	3	1	0	0	0	0	0	0	4	4.77
SW	0	0	2	0	0	1	0	1	0	0	0	0	4	4.47
WSW	0	0	0	1	0	0	0	0	0	0	0	0	1	3.40
W	0	0	0	1	3	3	1	0	0	0	0	0	8	5.06
WNW	0	1	0	2	5	6	7	3	2	1	0	1	28	6.20
NW	0	1	0	0	8	7	5	4	5	7	7	19	63	9.04
NNW	0	0	0	0	1	1	0	1	2	3	1	3	12	9.24
N	0	0	1	5	2	2	1	1	4	1	0	3	20	6.60
TOTAL	1	6	19	37	51	51	43	31	20	17	9	38	317	6.50

Figure 2.3-10, sheet 21
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS D CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	2	1	7	1	3	2	0	1	1	4	3	25	6.64
NE	1	0	0	4	3	2	2	5	3	2	1	1	24	6.64
ENE	0	1	1	4	5	1	0	4	0	0	0	0	16	4.76
E	0	0	1	8	1	1	3	3	1	0	0	0	18	5.03
ESE	0	0	2	0	2	4	6	2	2	3	0	0	21	6.56
SE	0	0	0	0	2	1	6	3	3	2	2	0	19	7.36
SSE	0	0	1	1	6	10	1	5	7	5	3	1	40	7.16
S	1	1	0	1	1	0	1	0	0	0	0	0	5	3.28
SSW	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
SW	0	0	0	0	1	0	0	0	0	0	1	0	2	7.55
WSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.50
W	0	0	0	2	0	0	1	0	0	0	0	0	3	4.40
WNW	0	0	1	2	2	1	1	3	0	0	0	1	11	6.16
NW	0	0	0	0	2	1	5	1	4	4	6	31	54	11.60
NNW	0	0	1	0	3	2	2	4	11	10	4	37	74	11.33
N	0	1	1	6	3	4	4	5	1	3	1	11	40	8.24
TOTAL	3	5	9	35	33	30	34	35	33	30	22	85	354	8.39

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	3	1	3	3	1	3	5	2	4	25	8.03
NE	0	0	0	2	1	0	0	3	3	0	1	0	10	7.03
ENE	0	0	0	1	0	0	0	0	0	1	0	0	2	6.60
E	0	0	0	0	1	1	1	0	0	0	0	0	3	5.83
ESE	0	0	0	0	0	2	1	1	0	0	0	0	4	6.05
SE	0	0	0	0	0	0	0	0	0	2	0	0	2	9.25
SSE	0	1	0	0	1	1	1	0	1	1	1	0	7	8.71
S	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	1	0	0	0	1	8.80
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	1	0	0	1	1	0	0	0	0	0	3	4.93
NW	0	0	0	0	0	0	0	0	0	0	0	1	1	13.80
NNW	0	0	0	0	1	3	1	1	2	3	2	7	20	10.52
N	0	0	2	4	3	2	1	2	4	5	3	11	37	9.87
TOTAL	0	1	4	10	8	13	9	8	14	17	9	23	116	8.48

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	1	2	1	1	1	4	1	6	18	9.23
NE	0	0	0	0	0	1	2	0	1	3	0	0	7	8.01
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	1	0	0	0	0	0	0	0	1	4.30
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	1	0	0	0	0	0	0	1	5.40
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	2	2	12.95
NNW	0	0	0	0	1	0	2	0	0	0	1	3	7	10.33
N	0	0	0	1	5	1	2	2	1	1	1	8	22	9.45
TOTAL	0	0	0	2	8	5	7	3	3	8	3	19	58	9.26

PASQUILL D (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	3	3	15	6	9	7	4	5	10	7	13	82	7.26
NE	1	3	8	14	10	6	9	10	8	6	2	1	78	5.99
ENE	0	2	2	14	8	6	2	4	0	1	0	0	39	4.67
E	0	0	7	21	11	9	10	6	1	0	0	0	65	4.74
ESE	1	2	3	4	9	15	13	9	4	3	0	0	63	5.83
SE	1	0	0	1	5	10	13	8	6	4	2	1	51	6.78
SSE	1	3	7	6	18	20	11	14	12	11	5	9	117	6.81
S	4	6	19	11	17	10	8	0	1	0	0	2	78	4.11
SSW	2	3	3	9	19	5	1	0	0	0	1	0	43	4.13
SW	0	5	13	8	13	6	0	1	1	0	1	0	48	3.98
WSW	2	1	8	16	10	5	0	0	1	0	0	0	43	3.71
W	0	6	7	15	11	10	5	1	0	0	0	0	55	4.13
WNW	0	2	8	8	9	15	11	9	4	4	1	3	74	5.92
NW	0	1	2	4	15	11	12	8	11	15	15	56	150	9.63
NNW	0	0	3	1	6	6	5	6	15	19	8	50	119	10.67
N	0	1	4	19	14	9	9	10	12	10	5	34	127	8.40
CALM													0	
TOTAL	12	38	91	166	181	152	116	90	81	83	47	169	1226	6.79

Figure 2.3-10, sheet 22
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS D CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.90
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.40
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.80
SE	0	1	0	1	0	0	0	0	0	0	0	0	2	2.50
SSE	1	0	4	7	4	0	1	0	0	0	0	0	17	3.45
S	1	5	26	15	23	7	7	1	0	0	0	0	85	3.86
SSW	0	7	25	15	14	9	0	1	0	0	0	0	71	3.57
SW	0	8	12	15	16	3	1	2	0	0	0	0	57	3.70
WSW	0	8	9	11	9	5	0	0	0	0	0	0	42	3.41
W	1	3	9	18	11	5	2	1	1	0	0	0	45	3.97
WNW	0	2	4	2	5	5	2	0	2	2	0	1	25	5.32
NW	0	0	2	2	1	0	0	1	0	1	0	0	7	4.90
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	3	35	91	61	64	34	13	6	3	3	0	1	354	3.83

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	0	1	4.10
NE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	1	1	1	0	0	0	0	0	0	0	0	3	2.57
ESE	0	1	0	0	1	0	0	0	0	0	0	0	2	3.25
SE	0	0	1	1	1	1	0	0	0	0	0	0	4	4.00
SSE	0	3	8	7	3	2	1	2	1	0	0	0	27	3.90
S	1	6	13	12	19	5	2	2	0	0	0	1	61	3.99
SSW	0	5	7	5	7	3	2	1	0	0	0	0	30	3.63
SW	0	4	3	1	1	1	0	0	0	0	0	0	10	2.83
WSW	2	2	5	1	2	0	0	0	0	0	0	0	12	2.47
W	2	2	2	1	0	1	1	0	0	0	1	0	10	3.61
WNW	0	0	1	3	3	5	2	1	2	1	1	0	19	5.95
NW	0	0	2	0	2	3	1	1	3	0	0	1	13	6.40
NNW	0	3	0	0	0	0	0	0	0	0	1	0	4	3.78
N	1	0	0	0	1	0	0	0	0	0	0	0	2	2.45
TOTAL	6	27	43	33	41	21	9	7	6	1	3	2	199	4.05

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	2	1	1	1	0	0	0	1	0	0	6	4.48
NE	2	0	0	2	0	1	0	0	0	2	0	0	7	5.00
ENE	0	0	1	1	1	0	0	1	0	0	0	0	4	4.57
E	0	1	0	1	3	0	1	0	0	0	0	0	8	3.79
ESE	0	3	2	2	1	2	2	1	0	0	0	0	13	4.00
SE	0	0	1	3	0	1	2	0	1	0	0	0	8	4.95
SSE	1	3	15	10	15	6	7	3	1	2	0	2	65	4.66
S	3	8	19	18	17	15	8	4	0	1	1	0	88	4.86
SSW	1	6	2	3	6	6	3	3	0	0	0	0	30	4.17
SW	0	5	4	0	3	0	0	0	0	0	0	0	12	2.76
WSW	0	0	5	1	3	0	1	0	0	0	0	0	10	3.61
W	1	5	4	8	5	1	0	1	0	0	0	0	26	3.45
WNW	0	1	4	11	4	3	1	3	1	5	2	1	43	5.84
NW	0	0	3	1	6	5	7	4	2	1	4	1	34	6.53
NNW	0	3	0	2	0	1	0	0	1	0	0	0	7	3.80
N	0	1	0	4	3	2	0	0	0	0	0	0	10	3.98
TOTAL	8	36	64	62	68	45	39	20	6	12	7	4	371	4.59

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	1	1	1	0	0	0	1	0	1	0	6	5.08
NE	0	1	1	3	1	2	1	2	0	1	0	0	12	5.14
ENE	0	0	5	2	3	0	0	0	1	0	0	0	11	3.84
E	1	4	1	3	2	6	5	3	0	0	0	0	25	4.74
ESE	0	2	4	2	5	4	4	2	0	0	0	0	23	4.75
SE	0	2	2	3	4	7	4	0	1	0	0	0	23	4.68
SSE	0	3	21	27	16	15	13	11	6	1	0	9	132	5.77
S	2	4	0	13	9	5	2	2	1	2	0	1	52	4.32
SSW	0	6	3	0	0	1	2	0	0	0	0	0	12	3.02
SW	2	3	3	1	1	0	0	0	0	0	0	0	10	2.16
WSW	0	1	1	1	2	0	0	0	0	0	0	0	5	3.24
W	0	5	8	6	2	0	1	2	0	0	0	0	24	3.38
WNW	1	0	4	16	14	9	2	5	3	2	2	2	60	5.41
NW	0	0	3	9	23	16	15	10	9	12	10	20	127	7.56
NNW	0	0	0	0	0	2	5	4	3	1	4	1	20	8.13
N	0	2	1	3	2	4	1	0	0	0	1	1	15	5.20
TOTAL	7	33	66	90	85	74	55	41	27	19	26	34	557	5.64

Figure 2.3-10, sheet 23
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS E CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)

DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL E (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	2	4	2	1	1	3	1	6	3	5	0	3	31	6.38
NE	1	0	0	2	1	6	2	1	1	1	0	0	15	5.55
ENE	0	0	1	5	1	0	0	1	0	0	0	0	8	4.01
E	3	0	5	3	2	2	2	0	0	0	0	0	17	3.49
ESE	0	1	1	2	2	2	3	4	1	2	0	0	18	6.07
SE	0	2	0	3	0	5	1	0	3	3	1	0	18	6.35
SSE	0	2	3	8	7	4	5	13	4	2	4	3	55	6.50
S	1	3	2	3	4	2	1	0	0	0	0	0	16	3.56
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	0	0	2	0	0	0	0	0	0	0	0	0	2	2.95
WSW	0	2	1	0	0	0	0	0	0	0	0	0	3	2.17
W	1	0	2	2	2	0	0	0	0	0	0	0	7	2.63
WNW	0	1	1	7	6	3	4	2	1	0	0	3	28	6.07
NW	0	2	1	2	10	16	30	32	19	10	17	28	167	8.26
NNW	0	2	0	3	2	4	2	6	14	12	15	21	81	9.80
N	0	2	2	6	3	1	5	2	3	1	2	11	38	8.21
TOTAL	8	21	23	47	41	48	56	67	49	36	39	69	504	7.33

PASQUILL E (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	4	2	3	3	3	3	1	20	6.07
NE	0	0	1	0	3	0	4	1	2	0	0	0	11	6.01
ENE	0	0	0	1	0	0	0	1	0	0	0	0	2	5.35
E	0	0	1	0	0	1	0	0	0	0	0	0	2	4.25
ESE	0	0	0	0	2	0	0	0	0	0	0	0	2	4.20
SE	0	1	0	0	1	1	0	0	0	0	0	0	3	3.77
SSE	0	1	1	1	0	0	0	1	1	0	0	0	5	4.80
S	0	0	0	2	2	0	0	0	0	0	0	0	4	4.80
SSW	1	0	0	0	0	0	0	0	0	0	0	0	1	.80
SW	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
WSW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.60
W	1	1	0	0	0	0	0	0	0	0	0	0	2	1.00
WNW	0	0	2	1	0	0	0	0	0	0	0	0	3	3.13
NW	0	0	0	2	0	3	1	3	1	1	3	2	16	7.94
NNW	1	0	0	3	4	5	5	3	3	6	10	23	63	9.87
N	0	0	3	3	8	4	5	1	3	4	6	15	58	8.55
TOTAL	4	3	8	13	22	18	17	13	13	14	22	41	188	8.07

PASQUILL E (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	2	0	2	2	4	2	1	2	16	8.02
NE	0	0	0	1	0	0	1	1	1	0	1	0	5	7.30
ENE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.30
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
S	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSW	0	0	0	0	0	1	0	0	0	0	0	0	1	5.70
SW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WSW	1	0	1	0	0	0	0	0	0	0	0	0	2	1.50
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	1	0	0	0	1	0	0	0	2	6.85
NNW	0	0	0	0	2	3	2	2	4	1	1	10	[illegible]	10.47
N	0	0	0	1	1	1	[illegible]	1	3	3	4	[illegible]	[illegible]	9.67
TOTAL	1	0	4	1	7	[illegible]	11	[illegible]	[illegible]	[illegible]	7	22	94	8.99

PASQUILL E (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	3	4	5	5	7	8	5	11	11	11	5	6	81	6.80
NE	3	1	2	9	5	9	8	5	4	4	1	0	51	5.62
ENE	0	0	8	9	6	0	0	3	1	0	0	0	27	4.07
E	4	6	10	8	7	9	8	3	0	0	0	0	55	4.08
ESE	0	8	7	6	11	8	9	7	1	2	0	0	59	4.86
SE	0	6	4	11	6	15	7	0	5	3	1	0	58	5.15
SSE	2	12	52	60	47	27	27	30	15	5	12	14	303	5.35
S	8	26	68	57	74	37	20	9	1	3	1	2	306	4.07
SSW	2	24	37	23	27	20	7	5	0	0	0	0	145	3.66
SW	3	20	24	17	21	4	1	2	0	0	0	0	92	3.27
WSW	3	13	22	14	17	5	1	0	0	0	0	0	75	3.19
W	6	14	25	29	20	[illegible]	4	4	1	0	1	0	114	3.57
WNW	1	4	16	40	32	25	18	11	9	10	5	7	178	5.68
NW	0	2	11	16	43	43	54	51	35	25	34	52	366	7.70
NNW	1	8	0	8	8	15	14	15	25	20	31	55	200	9.40
N	0	5	8	17	18	12	17	4	8	8	13	37	147	8.08
CALM													1	
TOTAL	37	155	299	329	349	245	200	160	116	91	104	173	2258	5.75

Figure 2.3-10, sheet 24
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS E CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	1	0	0	0	0	0	0	0	0	0	0	0	1	.90
SSE	0	2	3	0	1	0	0	0	0	0	0	0	6	2.67
S	0	6	10	1	7	2	1	0	1	0	0	0	28	3.36
SSW	1	2	6	1	3	0	0	0	0	0	0	0	13	2.70
SW	0	5	3	0	0	0	0	0	1	0	0	0	9	2.78
WSW	0	2	2	1	0	0	0	0	0	0	0	0	5	2.36
W	0	4	8	4	0	0	0	0	0	0	0	0	16	2.62
WNW	0	2	2	0	3	0	0	0	0	0	0	0	7	3.13
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	1	0	0	0	0	0	0	0	0	0	0	0	1	.90
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	3	23	34	7	14	2	1	0	2	0	0	0	86	2.86

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.10
SSE	0	3	2	1	0	0	0	0	0	0	0	0	6	2.28
S	0	9	7	3	3	0	0	0	0	0	0	0	22	2.58
SSW	3	6	5	1	2	0	0	0	0	0	0	0	17	2.14
SW	2	0	2	0	1	0	0	0	0	0	0	0	5	2.08
WSW	0	0	2	0	0	0	0	0	0	0	0	0	2	2.35
W	0	0	2	3	0	0	0	0	0	0	0	0	5	3.18
WNW	0	0	2	1	2	0	0	0	0	0	0	0	5	3.48
NW	0	0	1	0	1	1	0	0	0	0	0	0	3	4.27
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	5	18	23	10	9	1	0	0	0	0	0	0	66	2.59

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	2	0	0	1	0	0	0	0	0	0	0	3	2.50
SE	0	0	0	0	1	0	0	0	0	0	1	0	2	7.45
SSE	0	4	8	3	0	1	0	0	0	0	0	0	18	[illegible]
S	1	10	13	8	7	3	4	0	0	0	0	0	43	3.31
SSW	1	11	5	0	4	1	0	0	0	0	0	0	[illegible]	[illegible]
SW	1	6	4	0	0	0	0	0	0	0	0	0	11	1.77
WSW	0	1	3	0	1	0	0	0	0	0	0	0	5	2.76
W	0	5	6	3	0	0	0	0	0	0	0	0	14	2.49
WNW	1	1	3	3	4	3	0	0	0	0	0	0	15	[illegible]
NW	1	0	0	1	1	3	1	0	1	0	0	0	8	5.18
NNW	1	1	1	0	0	0	0	0	0	0	0	0	3	1.73
N	1	0	0	0	1	0	0	0	0	0	0	0	2	2.30
TOTAL	7	41	43	15	20	11	5	0	1	0	1	0	144	3.01

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	1	0	0	0	0	0	0	0	0	2	3.05
NE	0	0	1	1	0	1	0	0	0	0	0	0	3	4.03
ENE	1	0	0	1	0	0	0	0	0	0	0	0	2	2.35
E	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
ESE	0	0	0	0	0	0	0	0	0	2	0	0	2	[illegible]
SE	0	1	1	0	0	0	0	0	0	1	1	1	[illegible]	7.06
SSE	0	2	10	2	1	3	0	0	0	0	0	0	18	3.19
S	1	5	9	8	3	1	1	0	0	0	0	0	28	3.10
SSW	2	7	3	3	1	0	0	0	0	0	0	0	16	[illegible]
SW	3	6	5	1	0	0	0	0	0	0	0	0	15	[illegible]
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	4	10	2	1	0	0	0	0	0	0	0	17	2.63
WNW	0	0	3	5	3	1	1	1	1	0	0	0	15	4.06
NW	0	0	[illegible]	[illegible]	[illegible]	[illegible]	1	0	1	0	0	0	[illegible]	[illegible]
NNW	1	0	0	0	0	0	0	0	0	0	0	0	[illegible]	[illegible]
N	0	1	1	0	0	0	0	0	0	0	0		[illegible]	[illegible]
TOTAL	8	[illegible]	45	[illegible]	15	12	3	1	[illegible]	[illegible]	1	[illegible]	143	3.52

Figure 2.3-10, sheet 25
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS F CONDITIONS IN THE PRIMARY YEAR OF RECORD

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	0	0	1	0	0	0	0	0	0	0	2	2.50
NE	1	0	0	0	0	1	0	0	0	0	0	0	2	2.75
ENE	0	1	1	0	0	0	0	0	0	0	0	0	2	1.80
E	0	0	1	0	0	0	0	1	0	0	0	0	2	4.70
ESE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.10
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.10
SSE	0	1	4	1	0	1	0	0	0	0	0	0	7	3.11
S	0	7	4	2	1	0	0	0	0	0	1	0	15	2.97
SSW	0	4	1	0	0	0	0	0	0	0	0	0	5	1.60
SW	1	2	0	0	0	0	0	0	0	0	0	0	3	1.17
WSW	3	1	0	1	0	0	0	0	0	0	0	0	5	1.50
W	3	2	0	2	0	1	0	0	0	0	0	0	8	2.21
WNW	0	1	2	8	4	0	0	0	0	1	0	0	16	3.69
NW	1	1	0	1	5	8	3	0	1	2	0	3	25	6.46
NNW	0	1	0	0	1	1	2	1	0	0	1	1	8	6.79
N	0	4	0	1	1	0	0	1	0	0	0	1	8	4.74
TOTAL	10	26	14	16	13	12	5	3	1	3	2	5	110	4.05

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	1	0	0	0	0	0	1	0	2	7.20
NE	0	1	0	0	1	0	0	0	0	0	0	0	2	3.25
ENE	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	1	0	0	1	9.10
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	2	0	0	0	0	0	0	0	0	0	0	0	2	.25
S	1	0	1	0	0	0	0	0	0	0	0	0	2	1.70
SSW	1	1	0	0	0	0	0	0	0	0	0	0	2	.80
SW	1	2	0	0	0	0	0	0	0	0	0	0	3	1.10
WSW	1	0	0	0	0	0	0	0	0	0	0	0	1	.80
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	1	1	0	0	0	0	0	0	0	0	0	2	2.25
NW	0	1	0	0	0	1	1	0	0	1	0	0	4	5.55
NNW	0	0	0	0	0	1	0	0	0	1	0	0	2	7.45
N	0	0	1	0	0	0	1	1	1	0	0	0	4	6.15
TOTAL	7	6	3	0	2	2	2	1	1	3	1	0	28	3.81

PASQUILL F (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA; RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.60
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	1	0	0	0	0	0	0	0	0	0	0	0	1	.40
S	1	0	0	0	0	0	0	0	0	0	0	0	1	.50
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SW	1	0	0	0	0	0	0	0	0	0	0	0	1	.10
WSW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
W	0	0	0	0	0	0	0	0	0	0	0	0	0	I
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	1	1	0	0	0	0	0	0	0	2	4.00
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	1	0	0	0	0	0	0	1	5.80
TOTAL	3	0	0	2	1	1	0	0	0	0	0	0	7	2.63

PASQUILL F (FROM AEC/DELTA T CRITERIA, 50-12 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	1	2	2	0	0	0	0	0	1	0	7	4.16
NE	1	1	1	1	1	2	0	0	0	0	0	0	7	3.44
ENE	2	1	1	1	0	0	0	0	0	0	0	0	5	1.66
E	0	0	2	0	0	0	0	1	0	0	0	0	3	3.63
ESE	0	2	1	0	1	0	0	0	0	3	0	0	7	5.30
SE	1	2	1	1	1	0	0	0	0	1	2	1	10	5.54
SSE	3	12	23	7	2	5	0	0	0	0	0	0	52	2.76
S	4	37	44	19	21	6	6	0	1	0	1	0	139	3.08
SSW	8	31	20	5	10	1	0	0	0	0	0	0	75	2.26
SW	9	21	18	1	1	0	0	0	1	0	0	0	51	1.97
WSW	4	4	7	2	1	0	0	0	0	0	0	0	18	2.14
W	3	15	26	14	1	1	0	0	0	0	0	0	60	2.56
WNW	1	5	13	17	16	4	1	1	1	1	0	0	60	3.79
NW	2	2	1	3	12	19	6	0	3	3	0	5	56	6.00
NNW	3	2	1	0	3	2	2	1	0	1	1	1	17	4.96
N	0	5	2	1	2	1	1	2	1	0	0	1	16	4.64
CALM													1	
TOTAL	43	140	162	74	74	41	16	5	7	9	5	8	584	3.30

Figure 2.3-10, sheet 26
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS F CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL A (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.80
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
E	0	0	0	0	1	0	0	0	0	0	0	0	1	4.50
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SSE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.90
S	1	4	3	4	1	0	0	0	0	0	0	0	13	2.60
SSW	0	5	2	3	0	0	0	0	0	0	0	0	10	2.45
SW	2	5	2	1	0	0	0	0	1	0	0	0	11	2.53
WSW	0	5	3	0	1	0	0	0	0	0	0	0	9	2.21
W	1	5	3	0	0	0	0	0	0	0	0	0	9	1.81
WNW	0	0	3	0	0	0	0	0	0	0	0	0	3	2.93
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	1
N	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
TOTAL	4	25	17	9	3	0	0	0	1	0	0	0	59	2.43

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL B (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	0	1	0	0	0	0	0	0	0	0	0	2	1.55
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	1	1	0	0	0	0	0	0	0	0	2	2.60
E	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.70
SSE	0	2	0	0	0	0	0	0	0	0	0	0	2	1.65
S	0	9	4	0	3	0	0	0	0	0	0	0	16	2.44
SSW	4	7	2	2	1	0	0	0	0	0	0	0	16	1.89
SW	3	3	1	0	0	0	0	0	0	0	0	0	7	1.29
WSW	0	7	1	1	0	0	0	0	0	0	0	0	9	1.80
W	0	6	5	1	0	0	0	0	0	0	0	0	12	2.12
WNW	1	4	5	1	0	0	0	0	0	0	0	0	11	2.16
NW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	4.40
N	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
TOTAL	9	39	21	6	6	0	0	0	0	0	0	0	81	2.10

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL C (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	1	0	0	0	0	0	0	0	0	0	1	2.70
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
ENE	0	0	0	1	0	0	0	0	0	0	0	0	1	3.20
E	0	1	0	1	0	0	0	0	0	0	0	0	2	3.60
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.50
SSE	0	4	1	1	1	0	0	0	0	0	0	0	7	2.23
S	2	19	11	6	4	0	0	0	0	0	0	0	42	2.30
SSW	1	28	8	2	1	0	0	0	0	0	0	0	40	1.88
SW	0	14	6	1	0	0	0	0	0	0	0	0	21	1.88
WSW	0	10	7	0	0	0	0	0	0	0	0	0	17	1.91
W	0	6	12	2	0	0	0	0	0	0	0	0	20	2.40
WNW	0	0	5	3	2	0	0	0	0	0	0	0	10	3.21
NW	0	0	2	1	1	0	0	0	1	0	0	0	5	4.16
NNW	0	0	1	1	0	0	1	0	0	0	0	0	3	4.43
N	0	0	2	0	1	0	0	0	0	0	0	0	3	3.10
TOTAL	3	83	56	19	10	0	1	0	1	0	0	0	173	2.09

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-12 FEET)
PASQUILL D (FROM AEC/SIGMA THETA CRITERIA, RANGE USED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (MPH) 1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	2	1	0	0	0	0	0	0	0	0	3	2.97
NE	1	1	0	0	1	0	0	0	0	0	0	0	3	2.07
ENE	0	0	3	1	1	0	0	0	0	0	0	0	5	3.26
E	0	2	1	0	0	1	0	0	0	0	0	0	4	2.97
ESE	0	1	1	0	0	0	0	0	0	0	0	0	2	1.80
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	1
SSE	1	3	2	3	0	0	0	0	0	0	0	0	9	2.23
S	3	22	32	10	4	0	0	0	0	0	0	0	71	2.42
SSW	2	50	20	3	0	0	0	0	0	0	0	0	75	1.86
SW	2	35	19	2	0	0	0	0	0	0	0	0	58	1.87
WSW	0	10	5	1	0	0	0	0	0	0	0	0	16	1.94
W	1	13	13	3	0	0	0	0	0	0	0	0	30	2.20
WNW	1	1	3	1	0	2	0	0	0	0	0	0	10	3.11
NW	1	1	0	1	2	0	0	0	0	0	0	0	5	2.98
NNW	0	0	0	2	0	1	0	0	0	0	0	0	3	4.07
N	0	2	0	0	0	0	0	0	0	0	0	0	2	1.50
TOTAL	12	141	101	30	[illegible]	4	0	0	0	0	0	0	396	2.18

Figure 2.3-10, sheet 27
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING
ΔT PASQUILL CLASS G CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE

WIND FREQUENCY DISTRIBUTION

(FREQUENCY IN NUMBER OF OCCURRENCES)

CALVERT CLIFFS PLANT SITE: STATION 2(IS)
DATA PERIOD: NOVEMBER 1, 1969 THROUGH OCTOBER 31, 1970

(Rev. 3/3/72)

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL E (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.50
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	1	0	1	0	0	0	0	0	0	0	2	3.60
E	1	0	1	0	0	0	0	0	0	0	0	0	2	1.85
ESE	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
SE	0	2	1	0	0	0	0	0	0	0	0	0	3	2.17
SSE	0	1	3	0	0	0	0	0	0	0	0	0	4	2.35
S	2	19	19	2	0	0	0	0	0	0	0	0	42	2.04
SSW	4	48	16	2	0	0	0	0	0	0	0	0	70	1.77
SW	6	34	16	1	0	0	0	0	0	0	0	0	57	1.78
WSW	2	8	5	2	1	1	0	0	0	0	0	0	19	2.27
W	2	4	2	2	3	0	0	0	0	0	0	0	13	2.55
WNW	2	1	0	5	1	0	0	0	0	0	1	0	10	3.66
NW	0	0	0	3	1	1	0	0	0	0	0	1	6	5.43
NNW	0	1	0	0	0	0	0	0	0	1	0	0	2	5.30
N	2	1	0	0	0	0	0	0	0	0	0	0	3	.83
TOTAL	22	120	64	17	7	2	0	0	0	1	1	1	235	2.12

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL F (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ESE	1	1	0	0	0	0	0	0	0	0	0	0	2	1.15
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
S	2	9	3	0	0	0	0	0	0	0	0	0	14	1.68
SSW	2	15	3	0	0	0	0	0	0	0	0	0	20	1.62
SW	2	20	5	1	0	0	0	0	0	0	0	0	28	1.67
WSW	2	5	0	0	0	0	0	0	0	0	0	0	7	1.30
W	0	1	0	0	0	0	0	0	0	0	0	0	1	1.70
WNW	0	1	0	0	0	0	0	0	0	0	0	0	1	1.30
NW	0	0	1	0	1	1	0	0	0	0	0	0	3	4.33
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
N	0	0	0	0	0	1	0	0	0	0	0	0	1	5.60
TOTAL	9	52	18	1	2	2	0	0	0	0	0	0	78	1.80

PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FEET)
PASQUILL G (FROM AEC/SIGMA THETA CRITERIA) RANGE USED)

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
ENE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
E	0	1	0	0	0	0	0	0	0	0	0	0	1	1.90
ESE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SE	0	0	0	0	0	0	0	0	0	0	0	0	0	I
SSE	0	1	0	0	0	0	0	0	0	0	0	0	1	1.60
S	0	0	1	0	0	0	0	0	0	0	0	0	1	2.20
SSW	2	13	3	0	0	0	0	0	0	0	0	0	18	1.68
SW	1	34	3	0	0	0	0	0	0	0	0	0	38	1.60
WSW	0	4	0	0	0	0	0	0	0	0	0	0	4	1.45
W	0	2	0	0	0	0	0	0	0	0	0	0	2	1.60
WNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	I
N	0	0	0	0	0	0	0	0	0	0	0	0	0	I
TOTAL	3	55	7	0	0	0	0	0	0	0	0	0	65	1.61

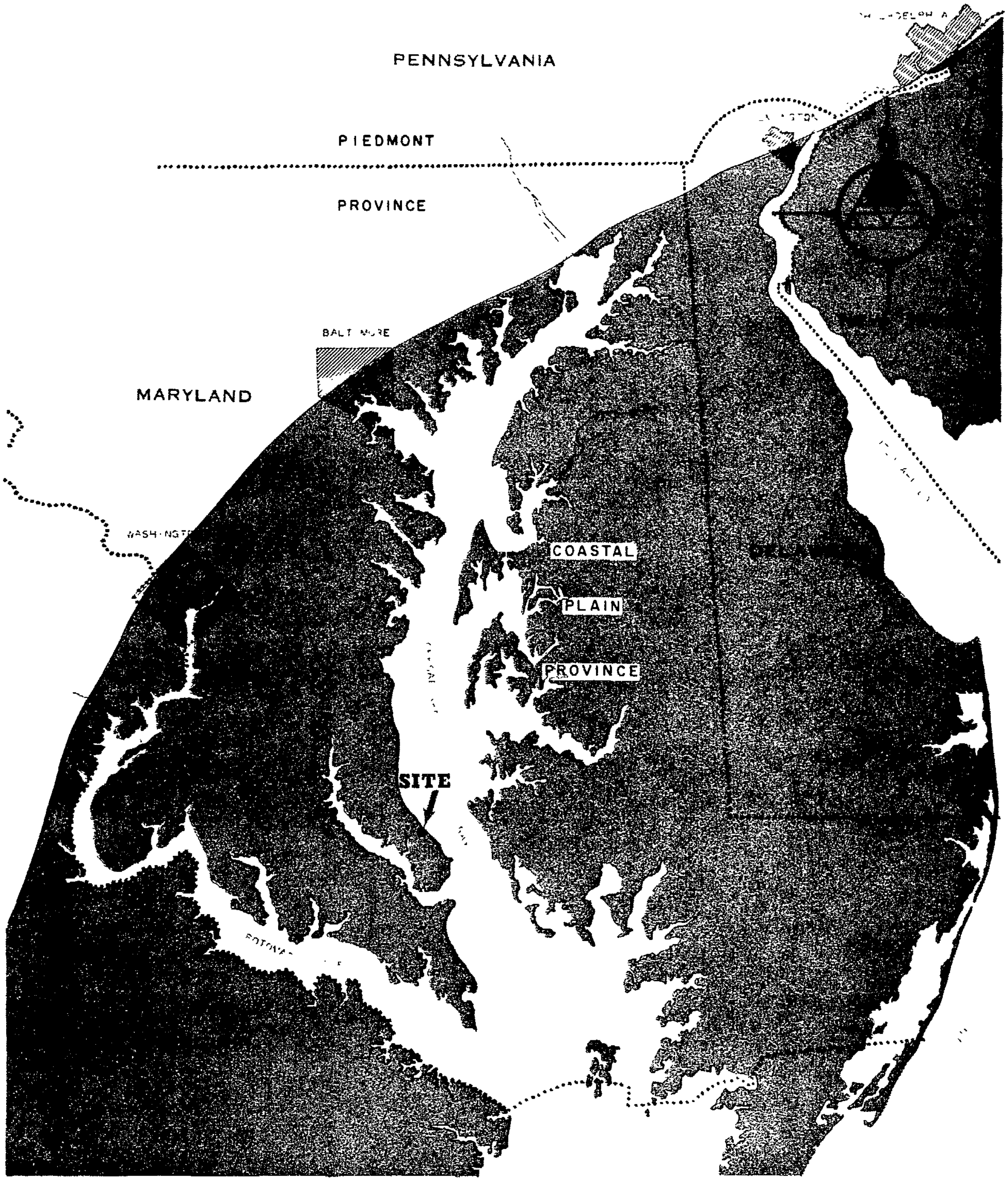
PASQUILL G (FROM AEC/DELTA T CRITERIA,50-18 FT.)
WINDS AT 33 FEET ABOVE GRADE

SECTOR	1	2	3	4	5	6	7	8	9	10	11	>11	TOTAL	MEAN SPEED
NNE	1	1	5	1	0	0	0	0	0	0	0	0	8	2.37
NE	1	1	0	0	1	0	0	0	0	0	0	0	3	2.07
ENE	0	0	5	3	2	0	0	0	0	0	0	0	10	3.23
E	1	4	2	1	1	1	0	0	0	0	0	0	10	2.60
ESE	2	2	1	0	0	0	0	0	0	0	0	0	5	1.38
SE	0	4	1	0	0	0	0	0	0	0	0	0	5	1.94
SSE	1	12	6	4	1	0	0	0	0	0	0	0	24	2.16
S	10	82	73	22	12	0	0	0	0	0	0	0	199	2.27
SSW	15	166	54	12	2	0	0	0	0	0	0	0	249	1.83
SW	16	145	52	6	0	0	0	0	1	0	0	0	220	1.79
WSW	4	49	21	4	2	1	0	0	0	0	0	0	81	1.95
W	4	37	35	8	3	0	0	0	0	0	0	0	87	2.23
WNW	4	7	16	12	3	2	0	0	0	0	1	0	45	2.97
NW	1	1	3	5	6	2	0	0	1	0	0	1	20	4.31
NNW	0	1	1	3	2	1	1	0	0	1	0	0	10	4.55
N	2	3	3	1	1	1	0	0	0	0	0	0	11	2.48
CALM													0	
TOTAL	62	515	278	82	36	8	1	0	2	1	1	1	987	2.12

Figure 2.3-10, sheet 28
Revision 18

THE DISTRIBUTION OF WIND SPEED, DIRECTION, AND σ_θ PASQUILL CLASSES DURING ΔT PASQUILL CLASS G CONDITIONS IN THE PRIMARY YEAR OF RECORD

DAMES & MOORE



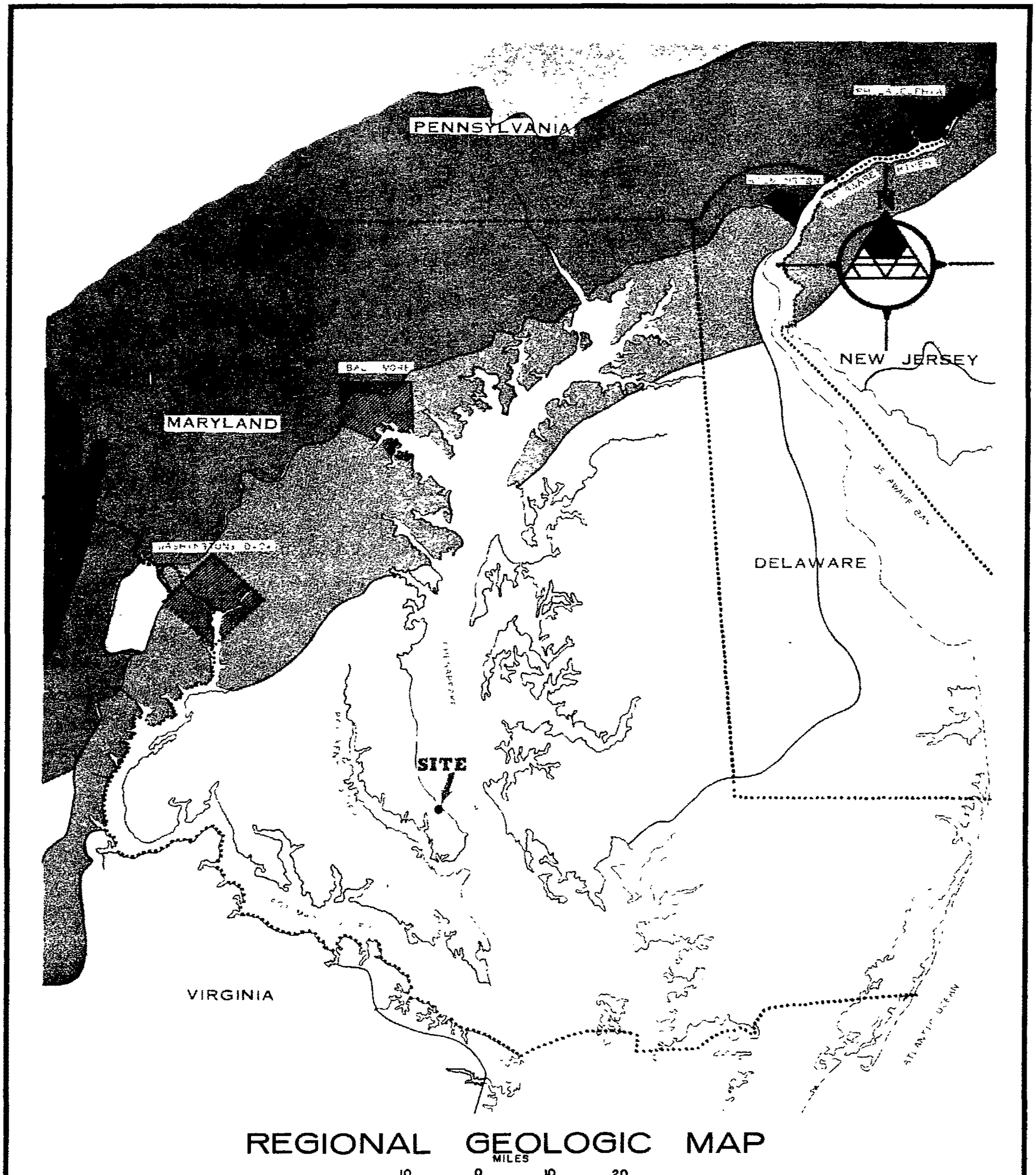
MILES
10 0 10 20

REFERENCE:
DEPARTMENT OF GEOLOGY, MINES AND WATER RESOURCES, STATE OF MARYLAND BULLETIN 18, 1957

REGIONAL PHYSIOGRAPHIC MAP

DAMES & MOORE

FIGURE 2.4-1



REGIONAL GEOLOGIC MAP

KEY:

- QUATERNARY SEDIMENTARY DEPOSITS
- TERTIARY SEDIMENTARY DEPOSITS– INCLUDES SOME PLEISTOCENE SEDIMENTS
- CRETACEOUS SEDIMENTARY DEPOSITS
- TRIASSIC SEDIMENTARY ROCKS
- PALEOZOIC SEDIMENTARY ROCKS
- PRECAMBRIAN METAMORPHIC AND IGNEOUS ROCKS
- TERTIARY AND PALEOZOIC IGNEOUS ROCKS

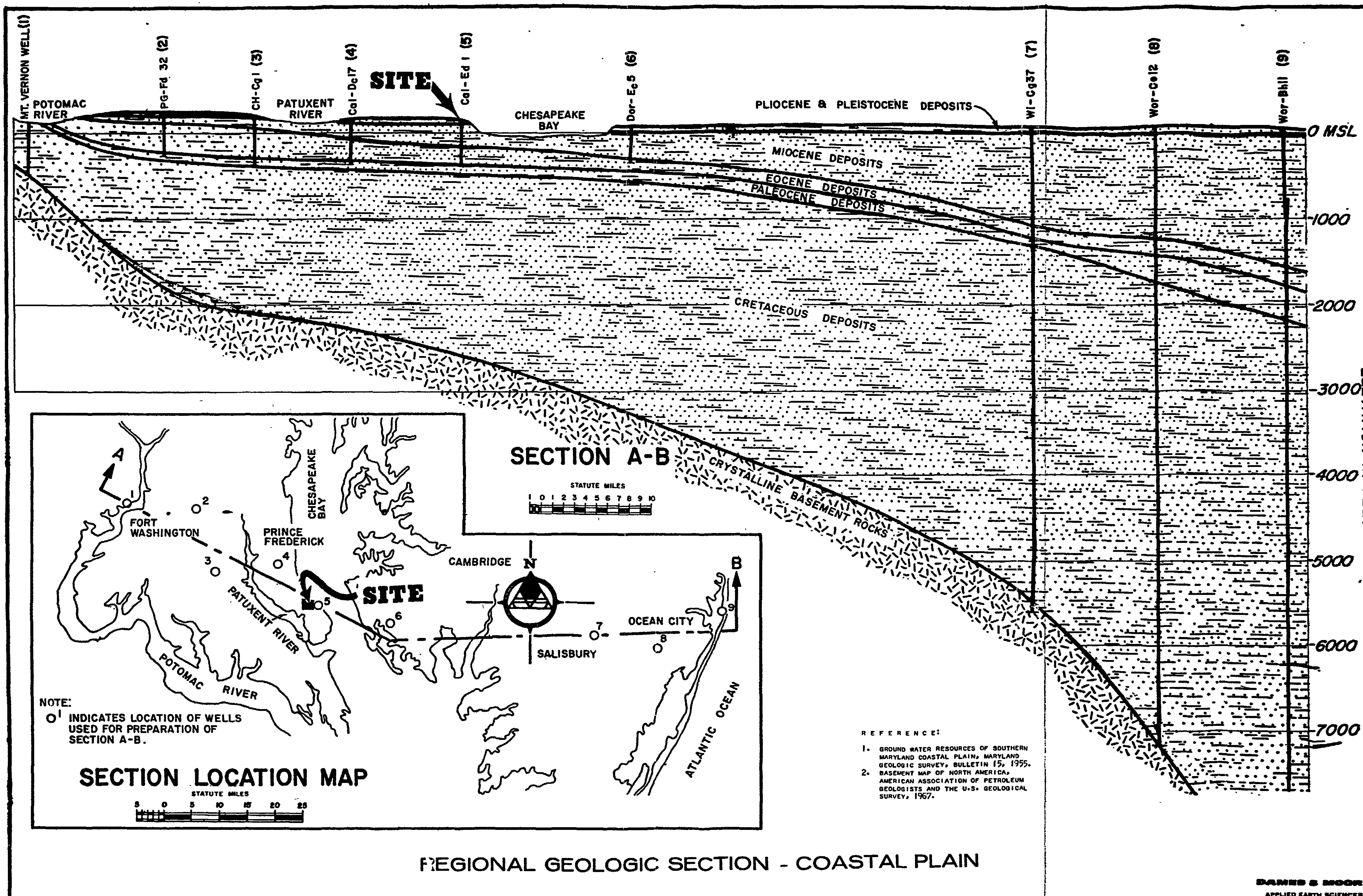
REFERENCES:

1) KINNEY, D.M., 1966, NATIONAL ATLAS– GEOLOGY; UNITED STATES GEOL. SURV., MAP.

2) U.S.G.S., 1932, GEOLOGIC MAP OF UNITED STATES, REPRINTED 1960.

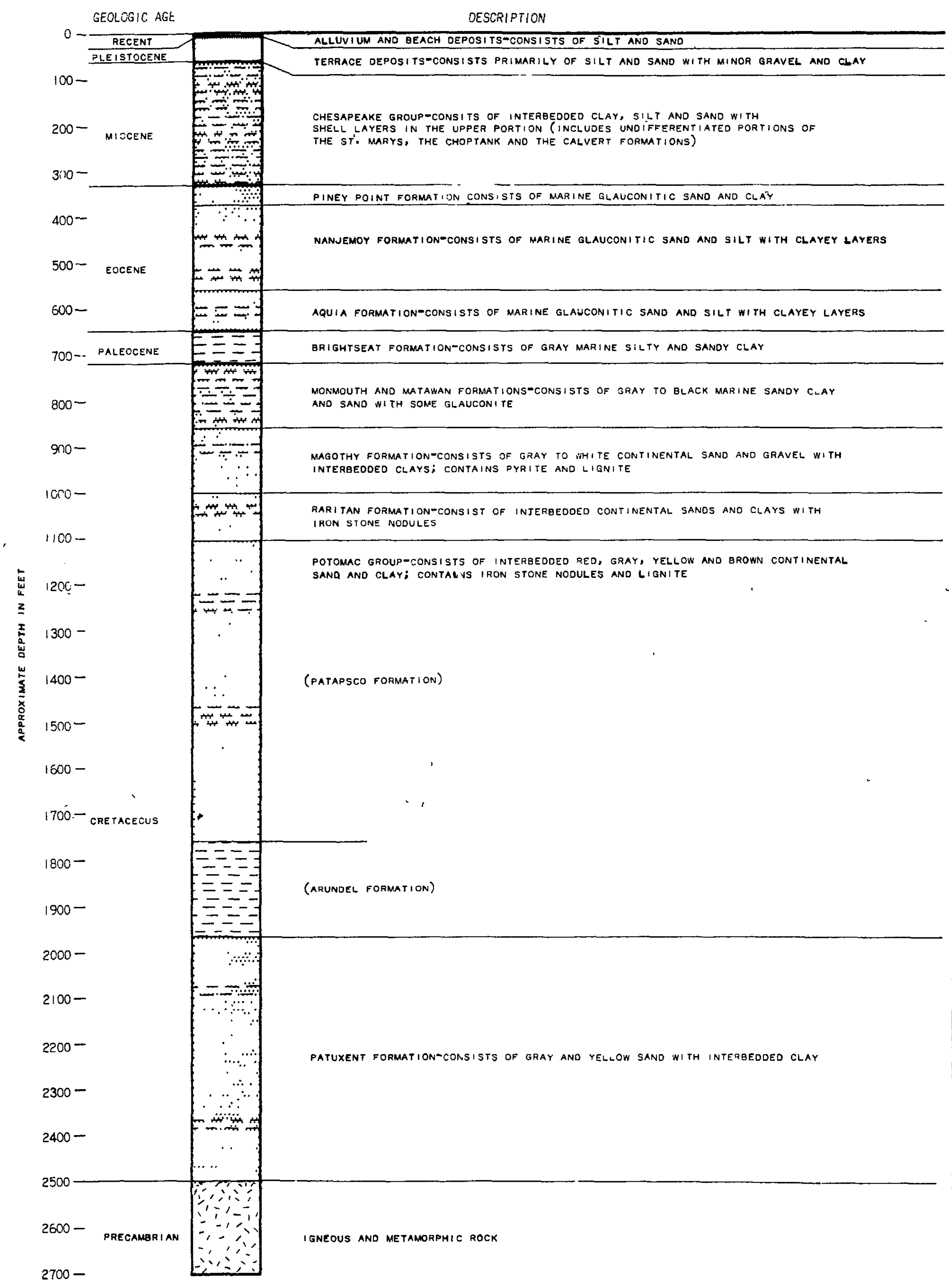
DAMES & MOORE

FIGURE 2.4-2



REGIONAL GEOLOGIC SECTION - COASTAL PLAIN

FIGURE 2.4-3

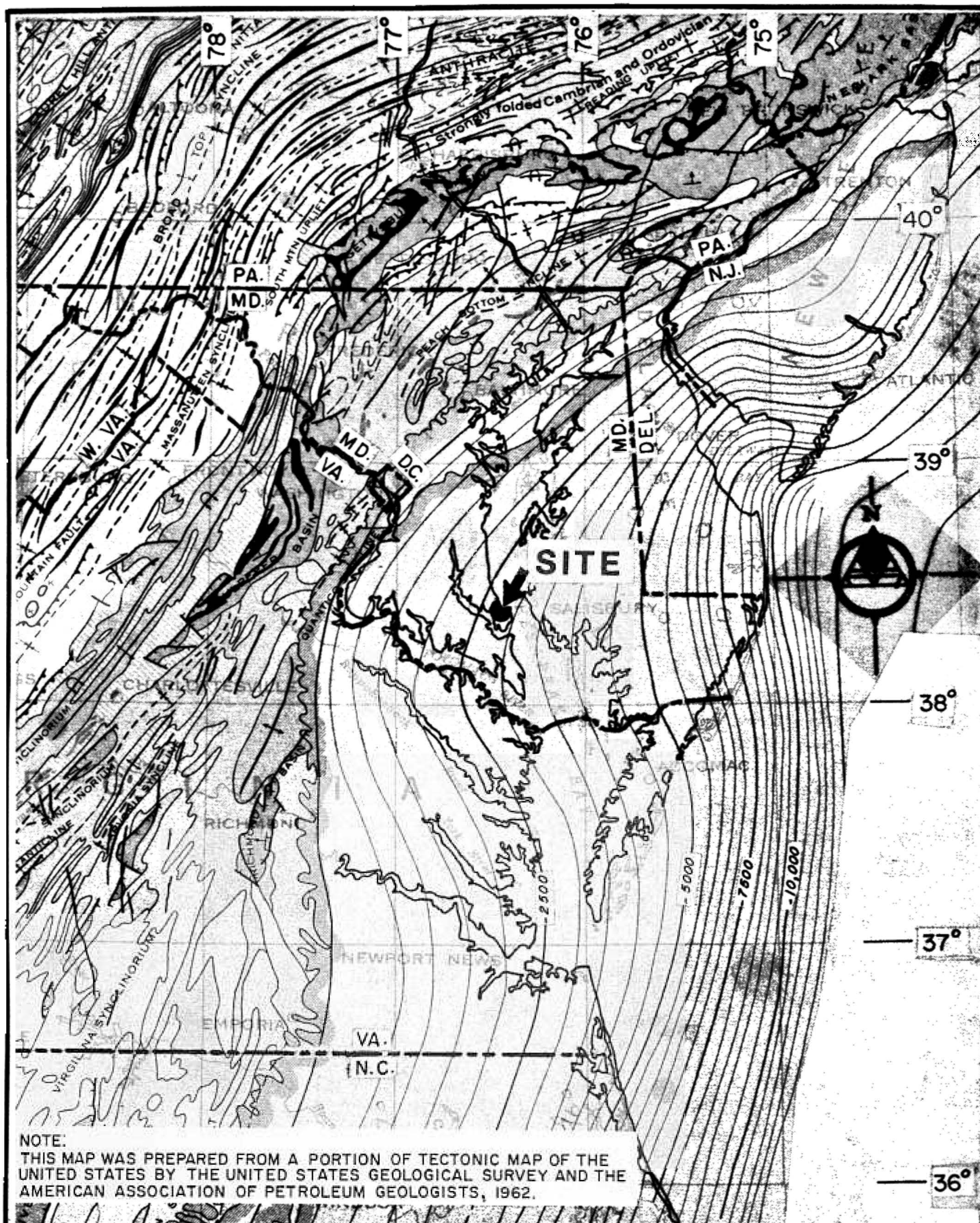


REFERENCE: GROUND-WATER RESOURCES OF THE SOUTHERN MARYLAND COASTAL PLAIN, MD. GEOLOGICAL SURVEY BULLETIN 15, 1955.

GEOLOGIC COLUMNAR SECTION - SITE AREA

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-4



LEGEND:

- ANTICLINAL AXIS
- SYNCLINAL AXIS
- DOME
- NORMAL FAULT
- THRUST FAULT
- CONTOURS ON TOP OF BASEMENT ROCKS

REGIONAL TECTONIC MAP

MILES

10 0 10 20 30 40 50

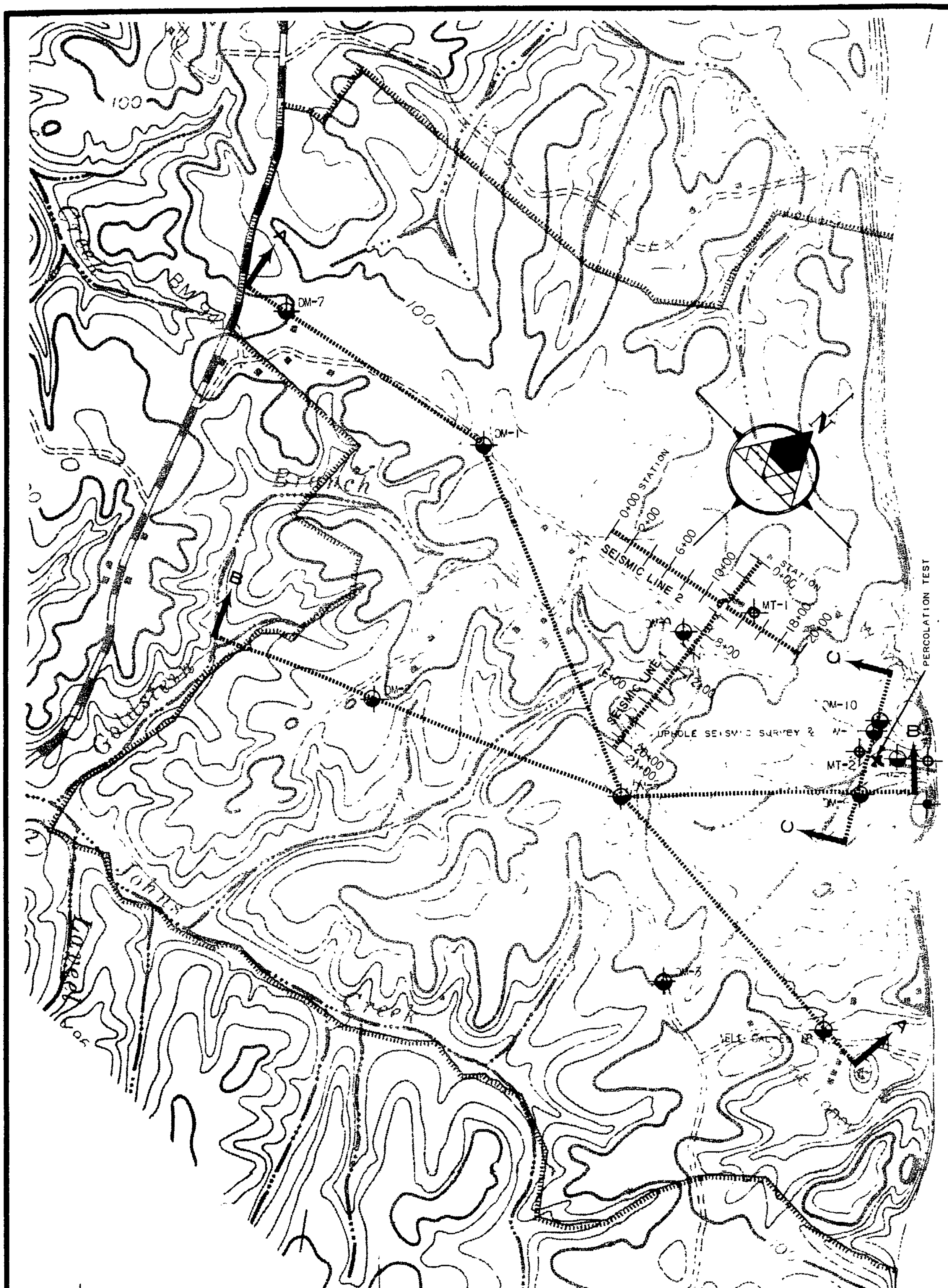
DAMES & MOORE

APPLIED EARTH SCIENCES

FIGURE 2.4-5


100
BM
DM-7
DM-1
Branch
100
B
28
River
KEY
RECENT
BEACH DEPOSITS
SAND
PLEISTOCENE
TERRACE DEPOSITS
SAND
MIOCENE
CHESAPEAKE GROUP
SAND
LOCATION OF DAMES & MOORE BORING
LOCATION WHERE CLIFF WAS LOGGED AS SHOWN ON FIGURE
REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF USGS., COVE POINT, MARYLAND - QUADRANGLE, 1943.
SITE GEOLOGIC MAP
FEET
600
0
600
1200
APPLIED EARTH SCIENCES

FIGURE 2.4-6



LOCATION OF DAMES & MOORE BORING

LOCATION WHERE CLIFF WAS LOGGED AS SHOWN ON FIGURE 2.4-11

LOCATION OF MICROTREMOR STATION

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS., COVE POINT, MARYLAND - QUADRANGLE, 1943.

PLOT PLAN

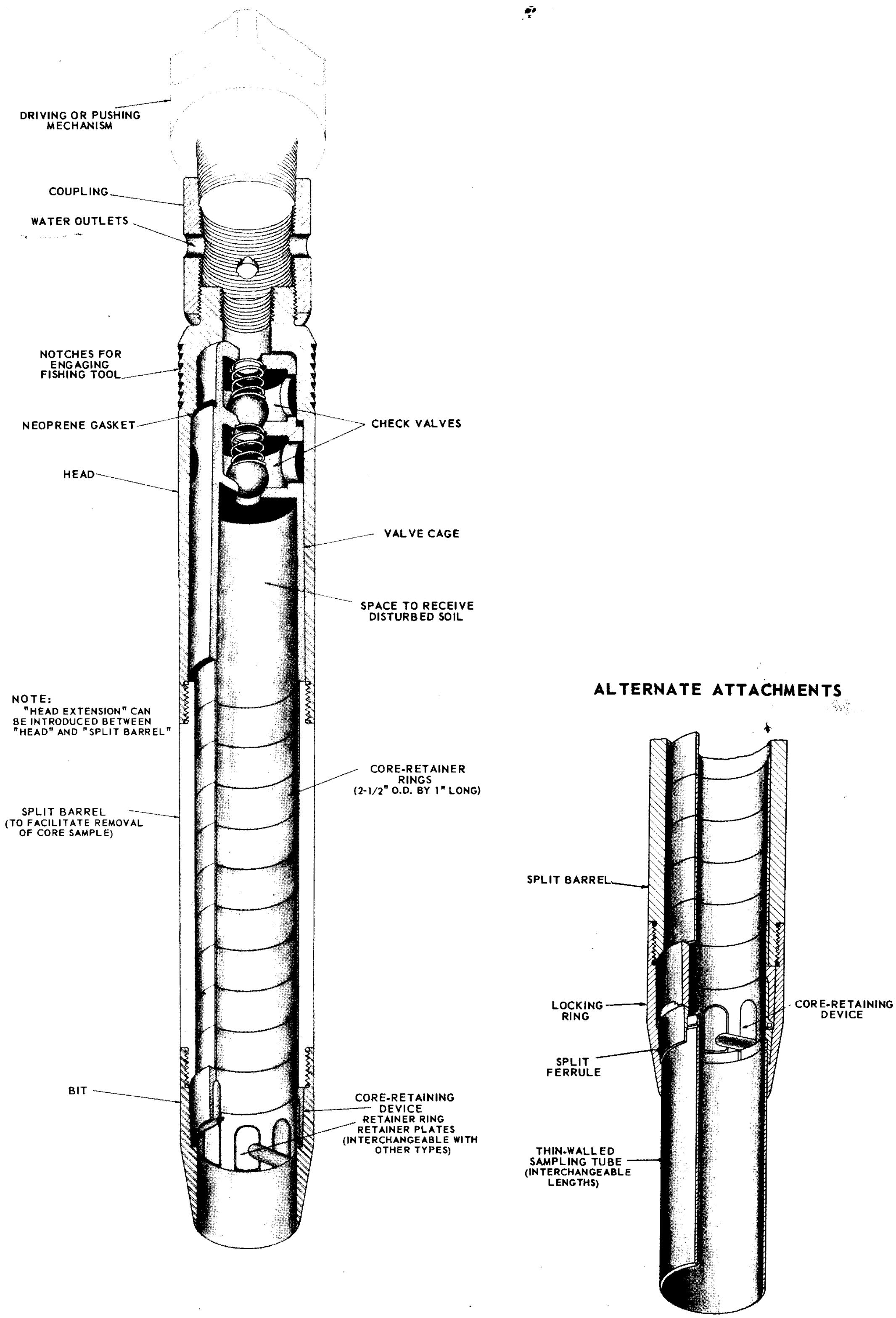
FEET

600 0 600 1200

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-7

FIGURE 2.4-8
SOIL SAMPLER TYPE U FOR SOILS DIFFICULT TO RETAIN IN SAMPLER
U.S. PATENT NO. 2,318,062

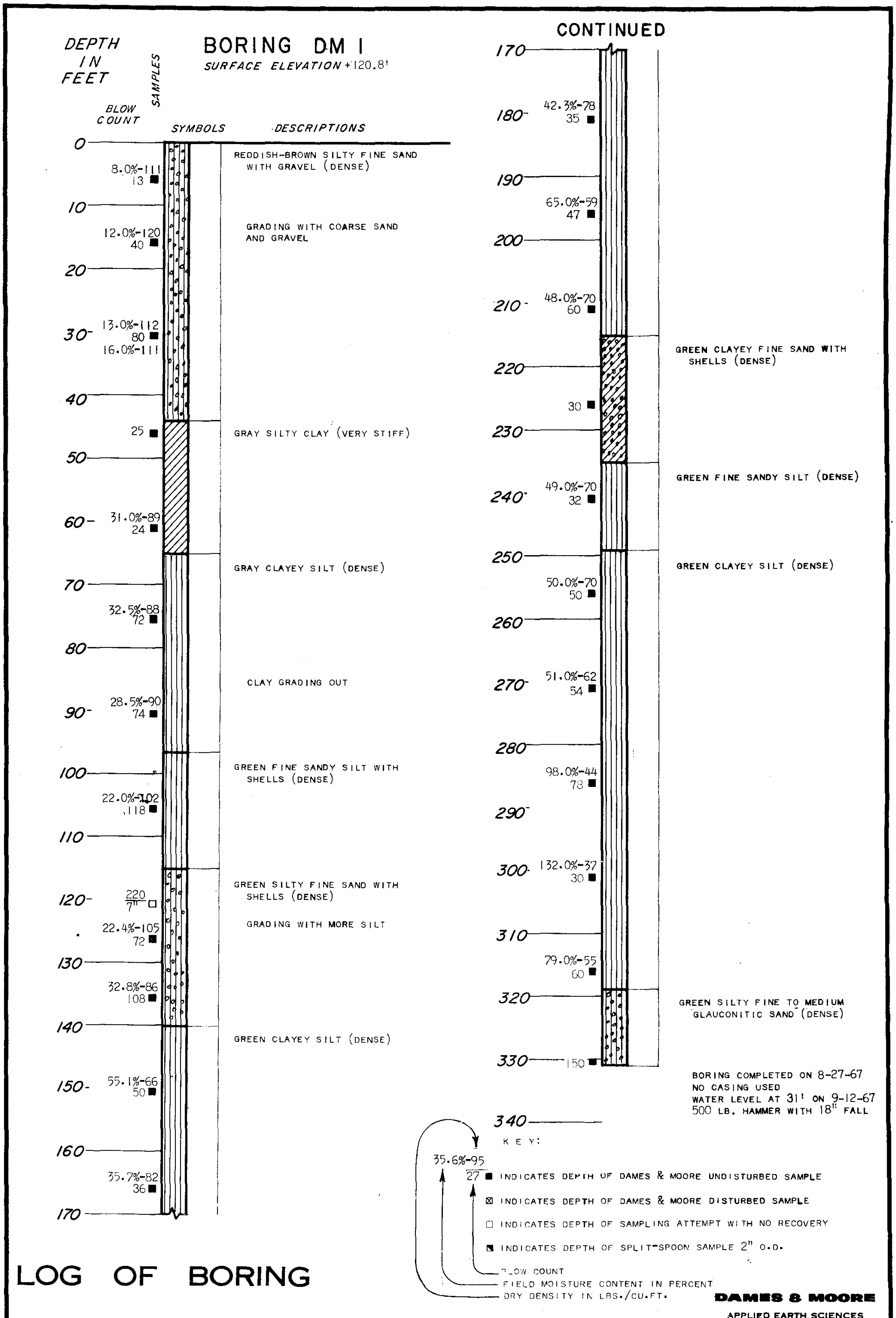


SOIL SAMPLER TYPE U

FOR SOILS DIFFICULT TO RETAIN IN SAMPLER
U. S. PATENT NO. 2,318,062

DAMES & MOORE

FIGURE 2.4-9A
LOG OF BORING



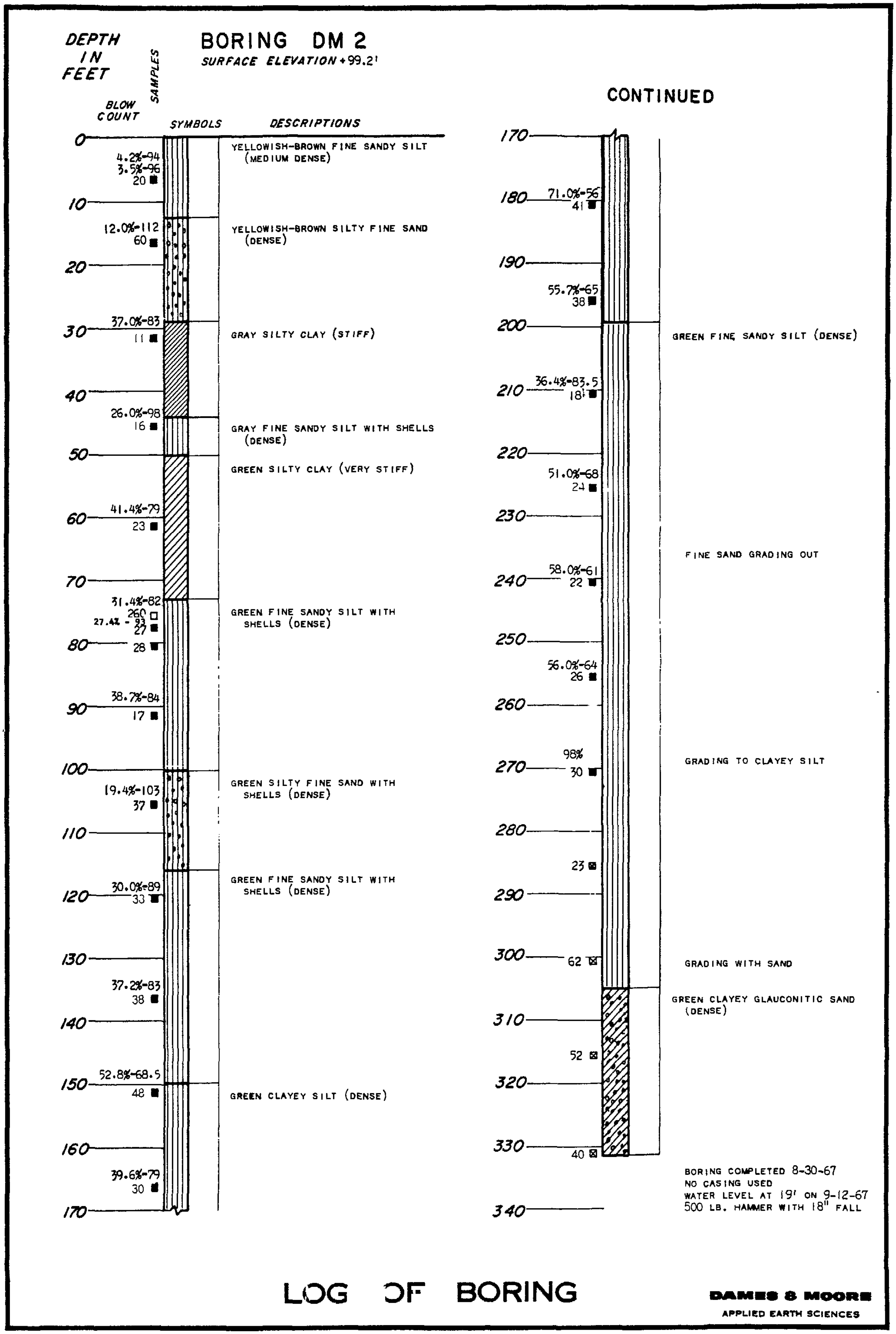
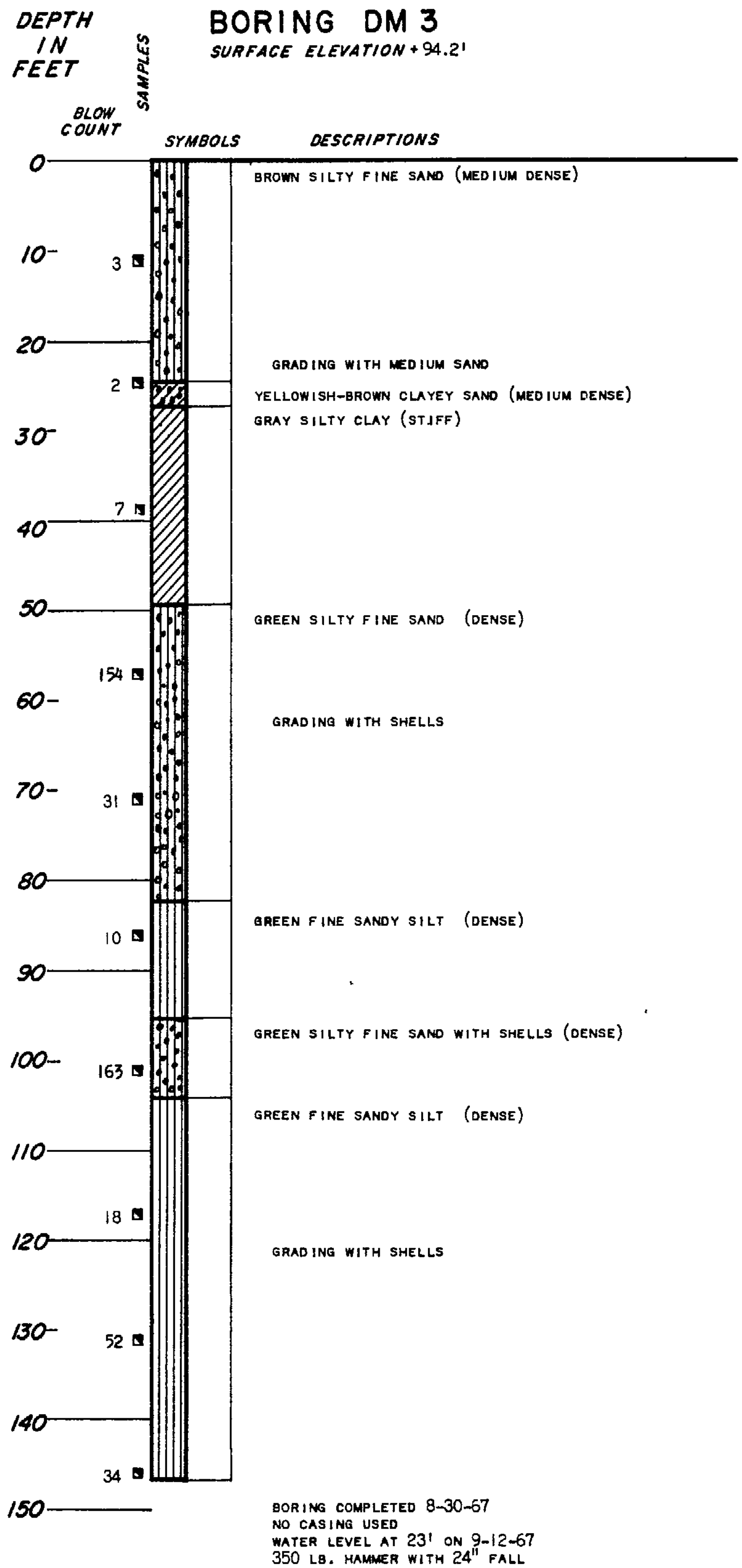

BORING DM 2
SURFACE ELEVATION +99.2'
DEPTH IN FEET
BLOW COUNT
SAMPLES
SYMBOLS
DESCRIPTIONS
YELLOWISH-BROWN FINE SANDY SILT (MEDIUM DENSE)
YELLOWISH-BROWN SILTY FINE SAND (DENSE)
GRAY SILTY CLAY (STIFF)
GRAY FINE SANDY SILT WITH SHELLS (DENSE)
GREEN SILTY CLAY (VERY STIFF)
GREEN FINE SANDY SILT WITH SHELLS (DENSE)
GREEN SILTY FINE SAND WITH SHELLS (DENSE)
GREEN FINE SANDY SILT WITH SHELLS (DENSE)
GREEN CLAYEY SILT (DENSE)
CONTINUED
GREEN FINE SANDY SILT (DENSE)
FINE SAND GRADING OUT
GRADING TO CLAYEY SILT
GRADING WITH SAND
GREEN CLAYEY GLAUCONITIC SAND (DENSE)
BORING COMPLETED 8-30-67
NO CASING USED
WATER LEVEL AT 19' ON 9-12-67
500 LB. HAMMER WITH 18" FALL
LOG OF BORING
DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-9B



LOG OF BORING

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-9C

Rev.0

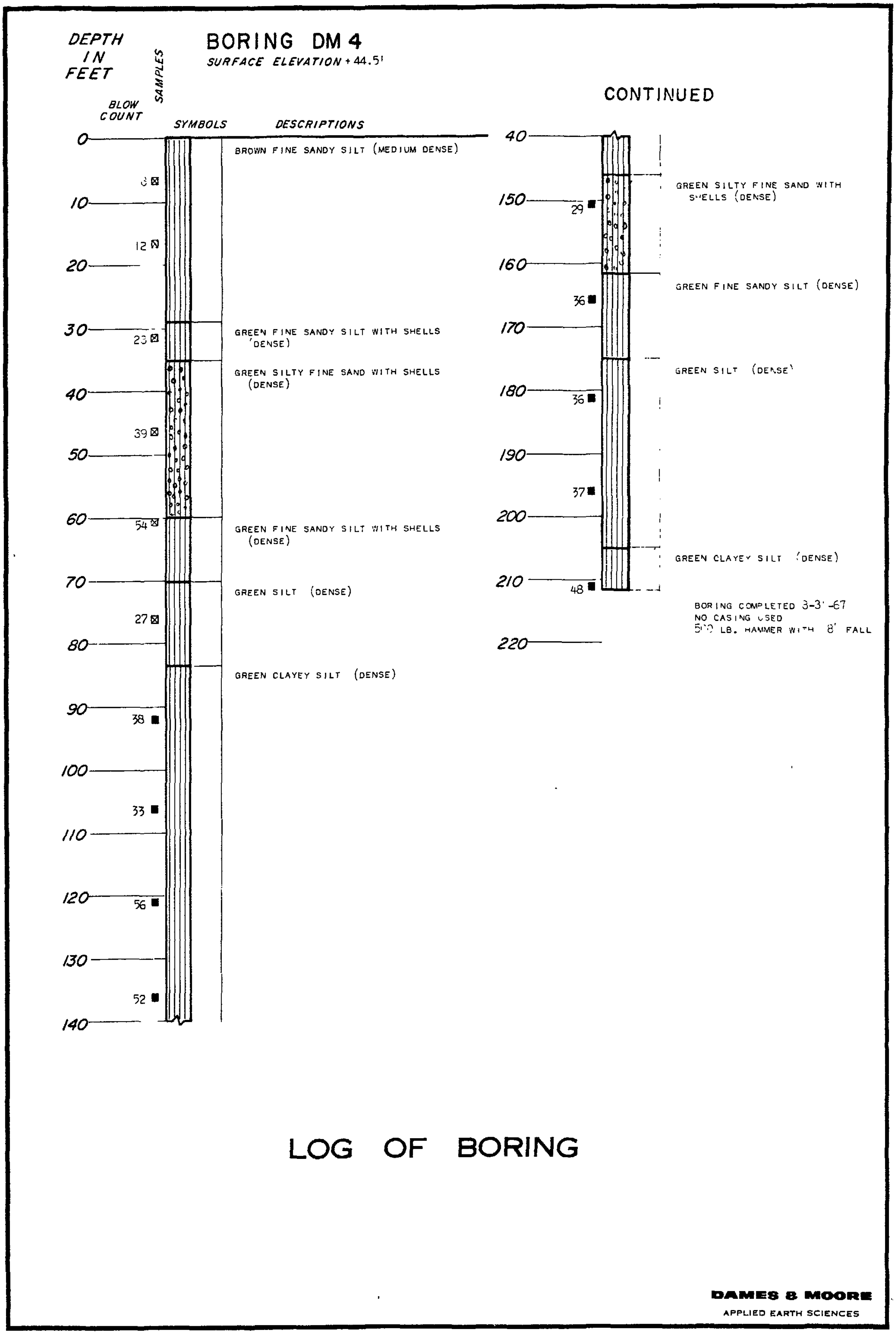
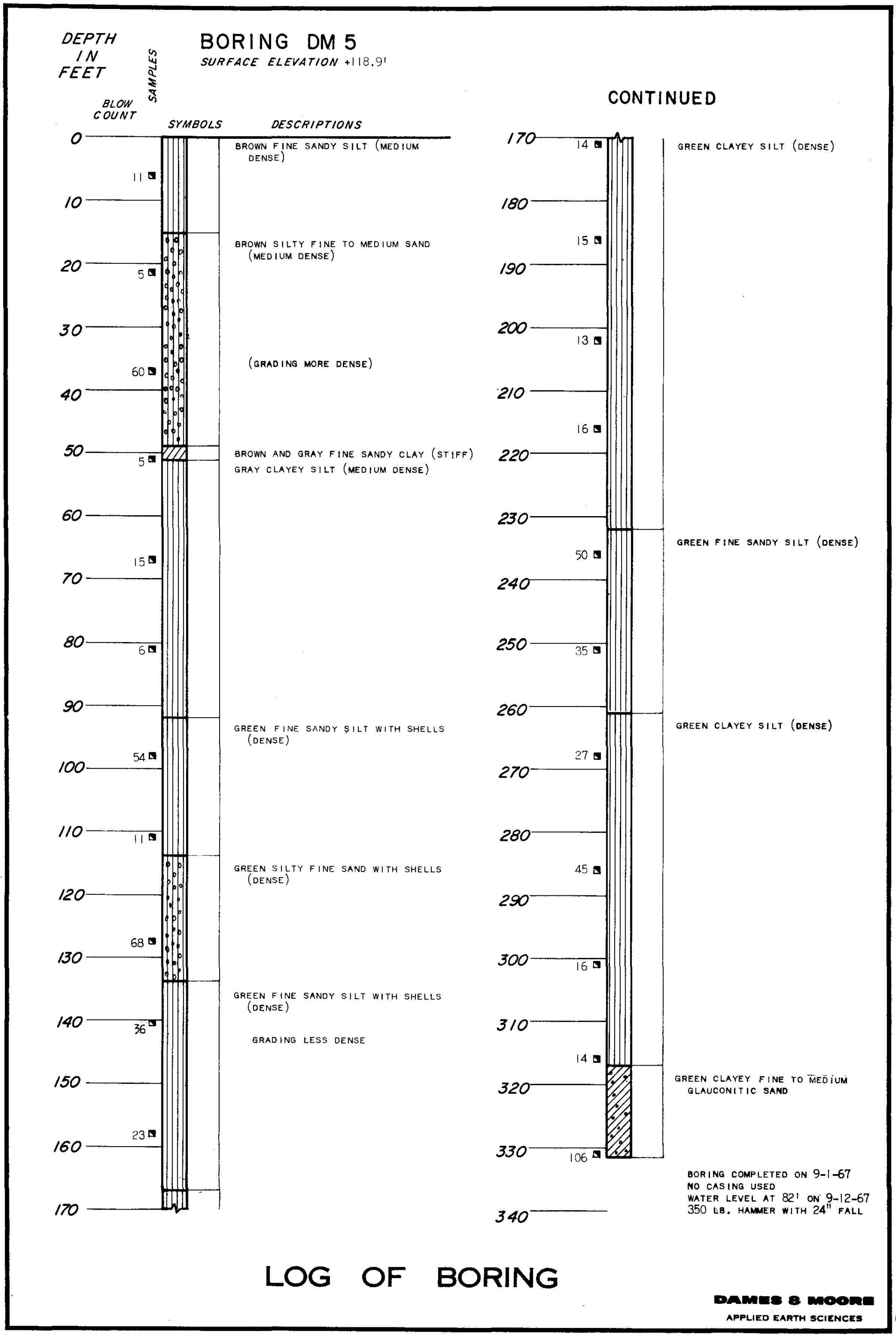

BORING DM 4
SURFACE ELEVATION + 44.5'
DEPTH IN FEET
BLOW COUNT
SAMPLES
SYMBOLS
DESCRIPTIONS
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
12
23
39
54
27
38
33
56
52
BROWN FINE SANDY SILT (MEDIUM DENSE)
GREEN FINE SANDY SILT WITH SHELLS (DENSE)
GREEN SILTY FINE SAND WITH SHELLS (DENSE)
GREEN FINE SANDY SILT WITH SHELLS (DENSE)
GREEN SILT (DENSE)
GREEN CLAYEY SILT (DENSE)
CONTINUED
150
160
170
180
190
200
210
220
29
36
36
37
48
GREEN SILTY FINE SAND WITH SHELLS (DENSE)
GREEN FINE SANDY SILT (DENSE)
GREEN SILT (DENSE)
GREEN CLAYEY SILT (DENSE)
BORING COMPLETED 3-3'-67
NO CASING USED
LB. HAMMER WITH 8' FALL
LOG OF BORING
DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-9D

FIGURE 2.4-9E
LOG OF BORING



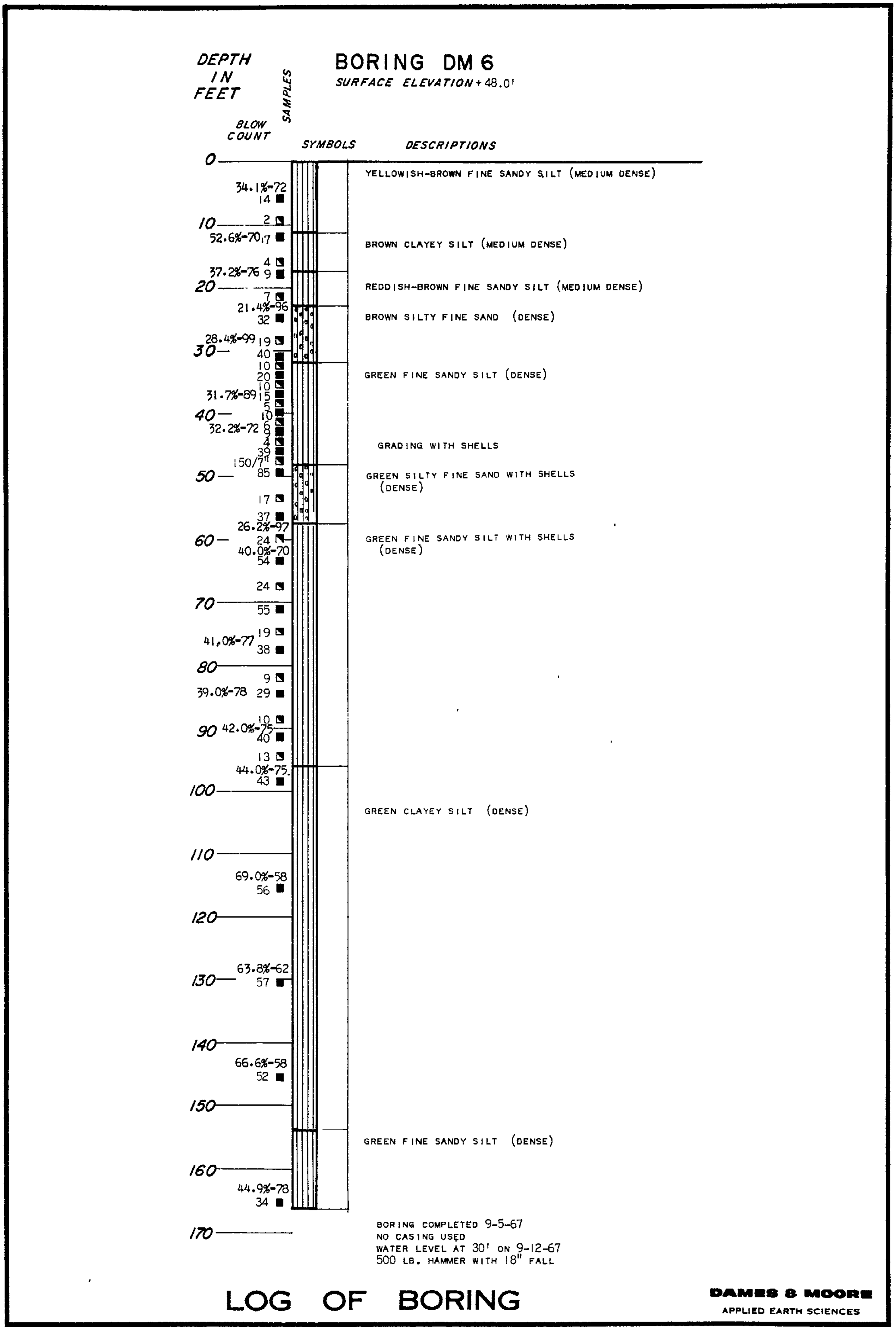


FIGURE 2.4-9F

Rev.0

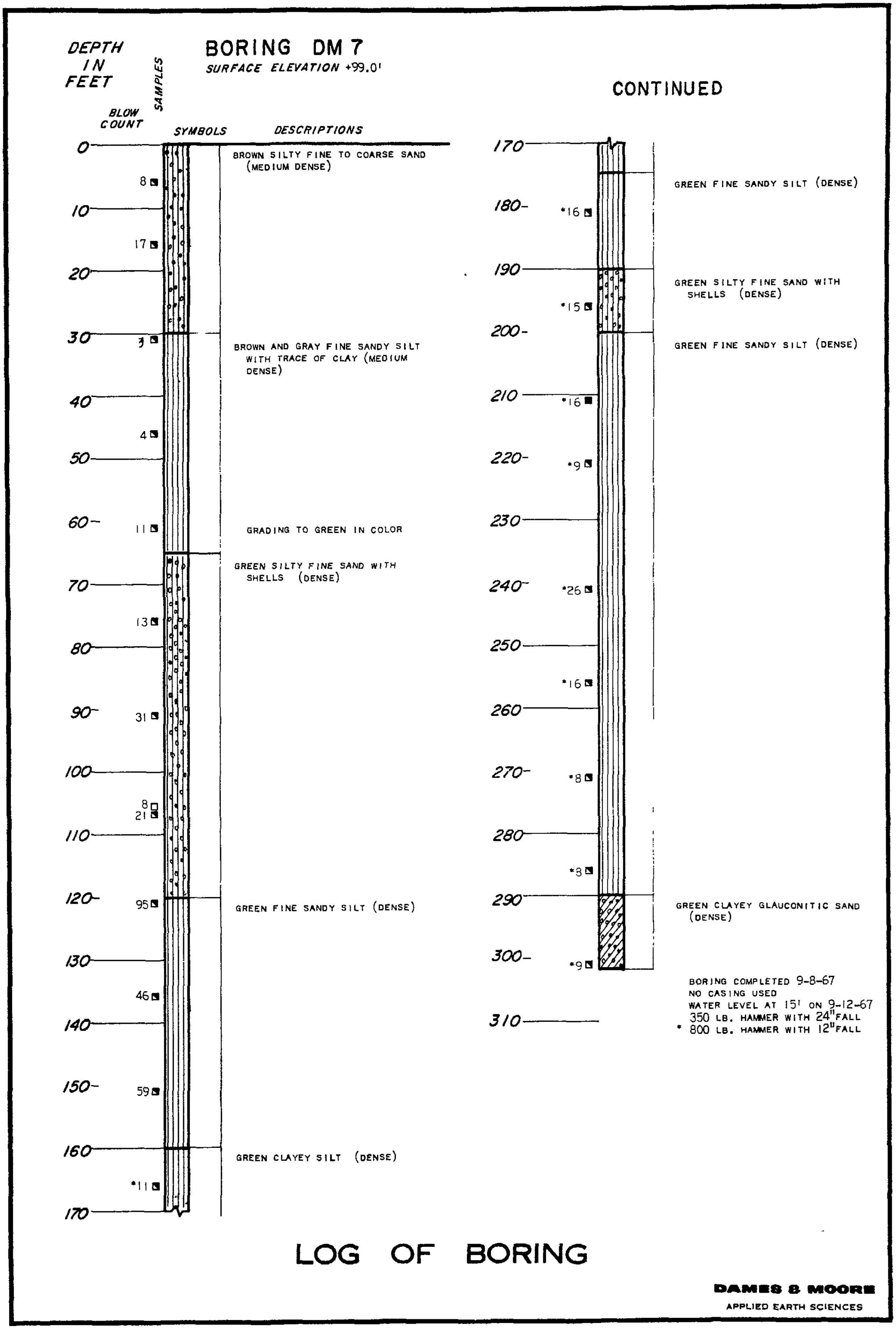
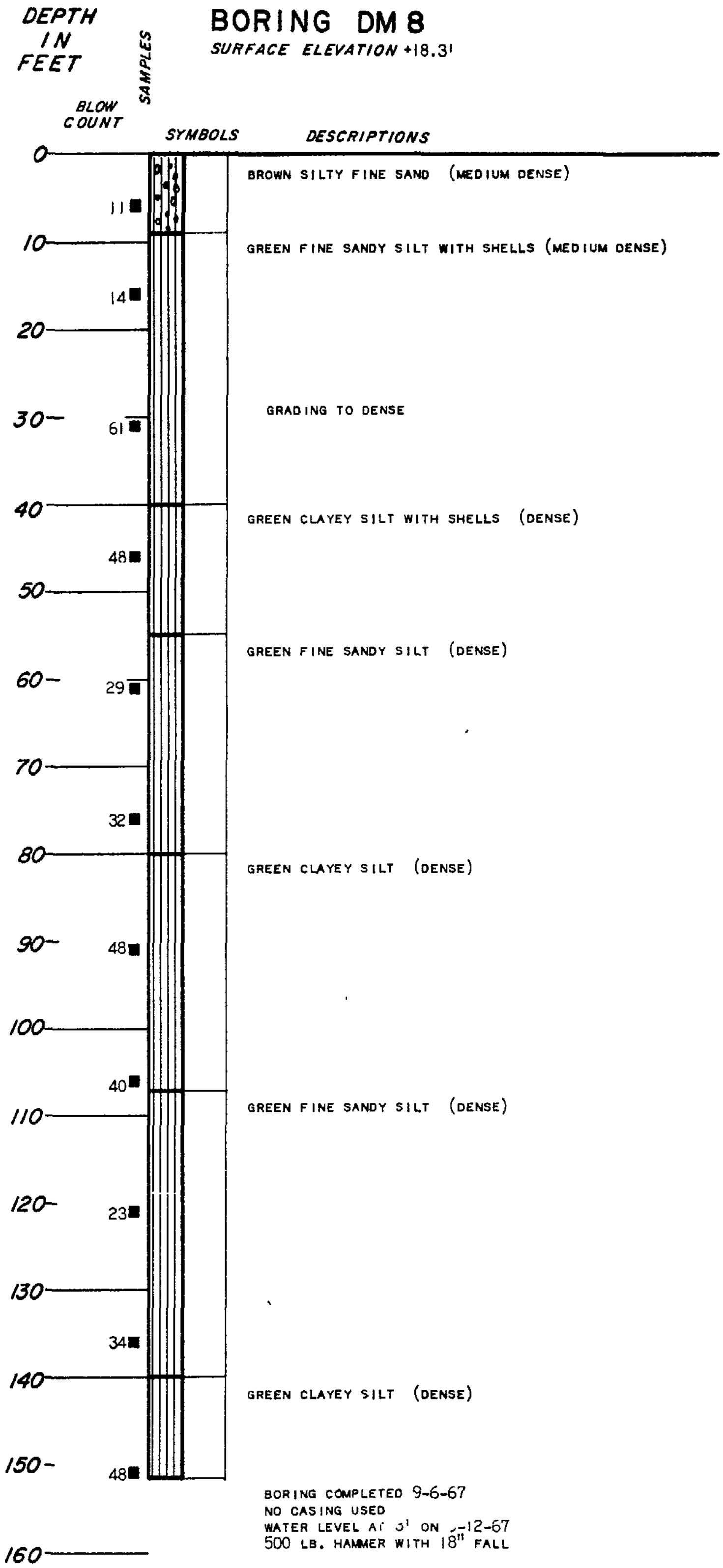


FIGURE 2.4-9G

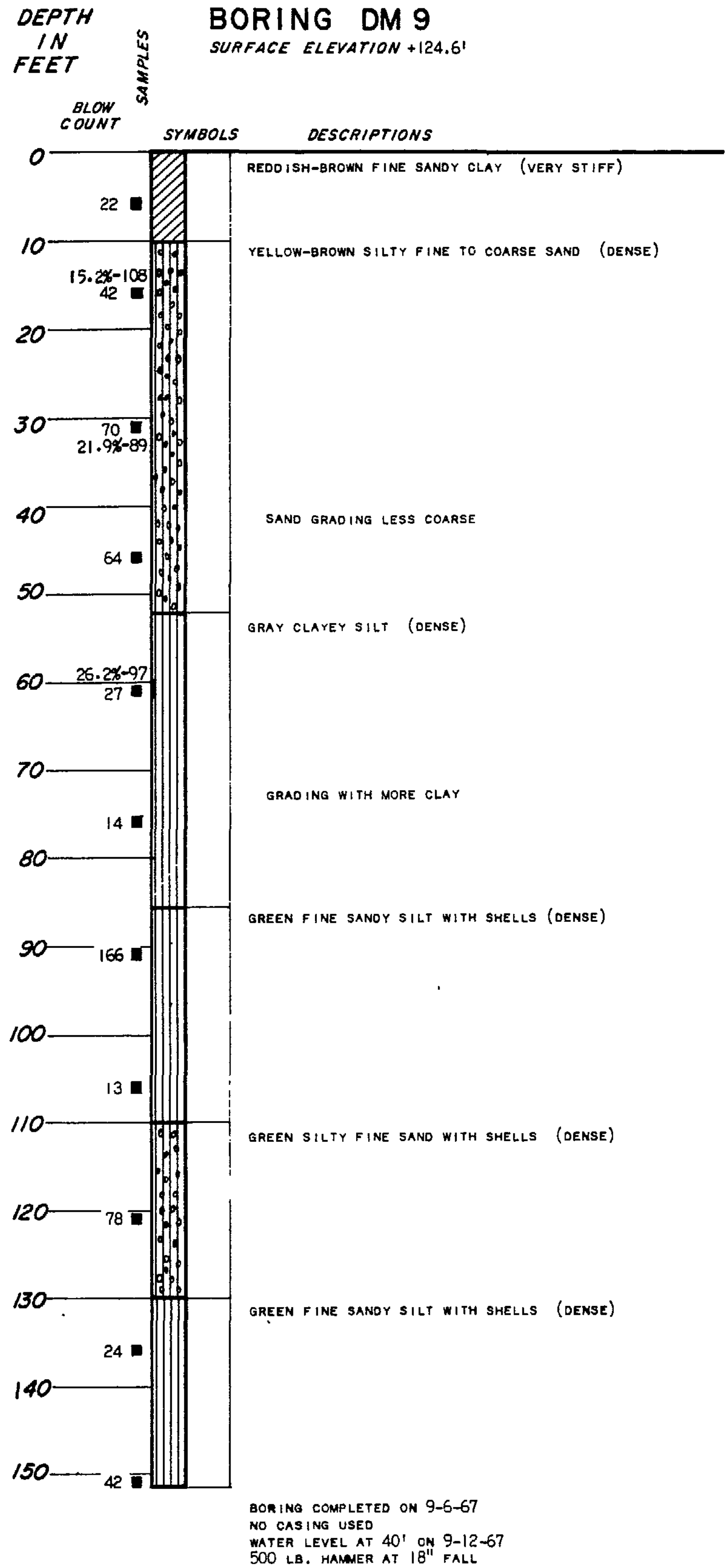
Rev.0



LOG OF BORING

DAMES & MOORE
APPLIED EARTH SCIENCES

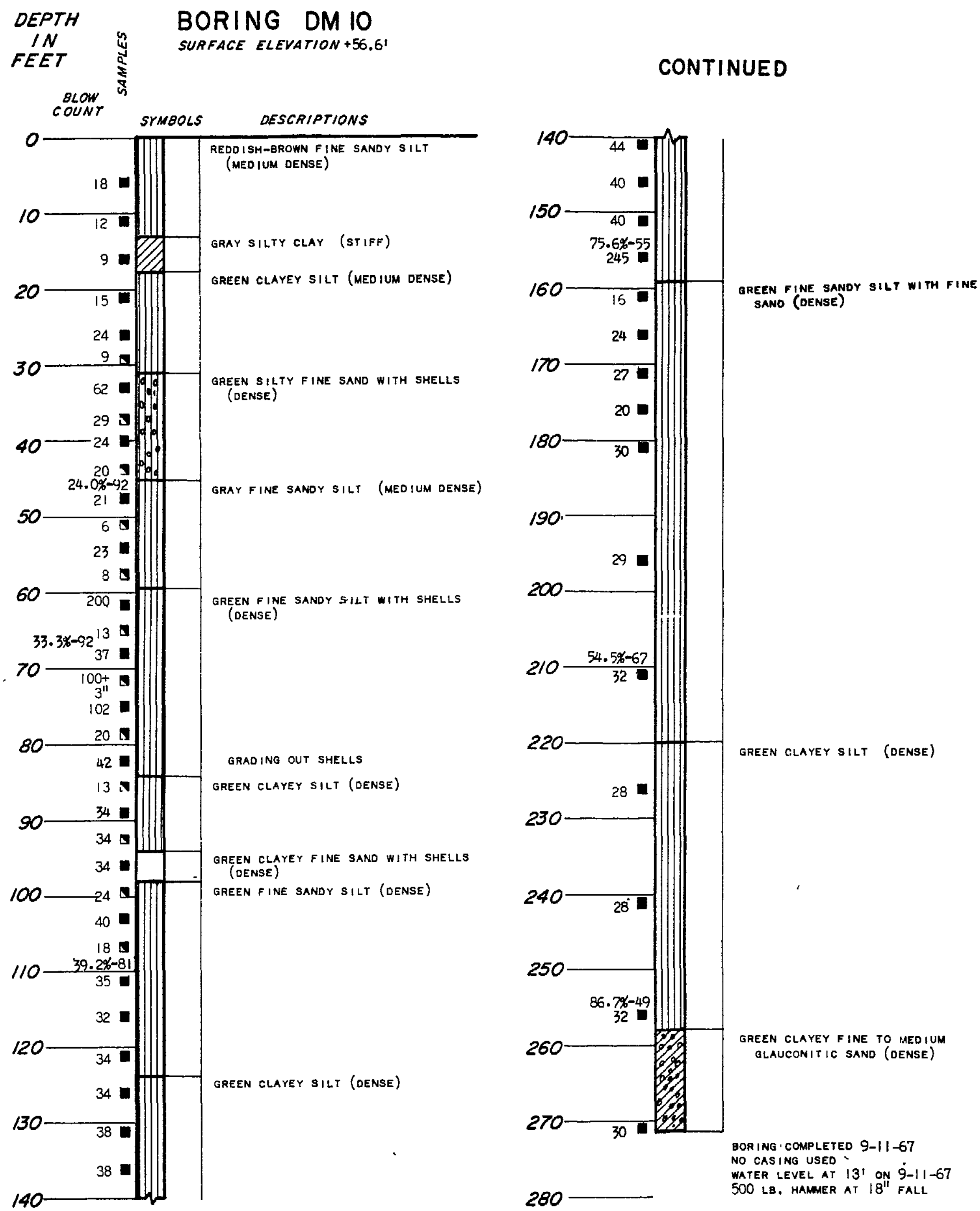
FIGURE 2.4-9H



LOG OF BORING

DAMES & MOORE
APPLIED EARTH SCIENCES

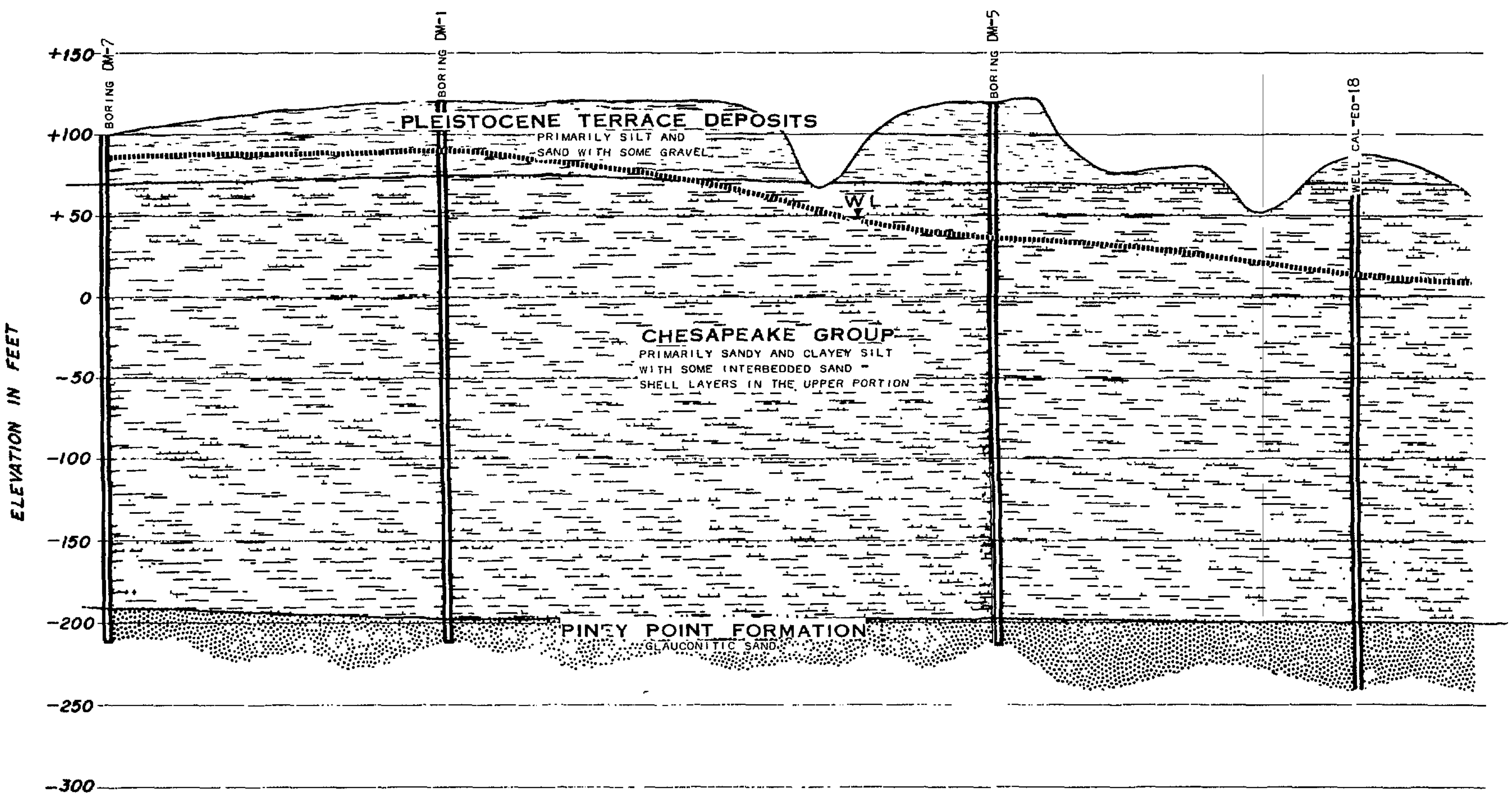
FIGURE 2.4-9I



LOG OF BORING

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-9J



GEOLOGIC SECTION A-A

HORIZONTAL SCALE IN FEET

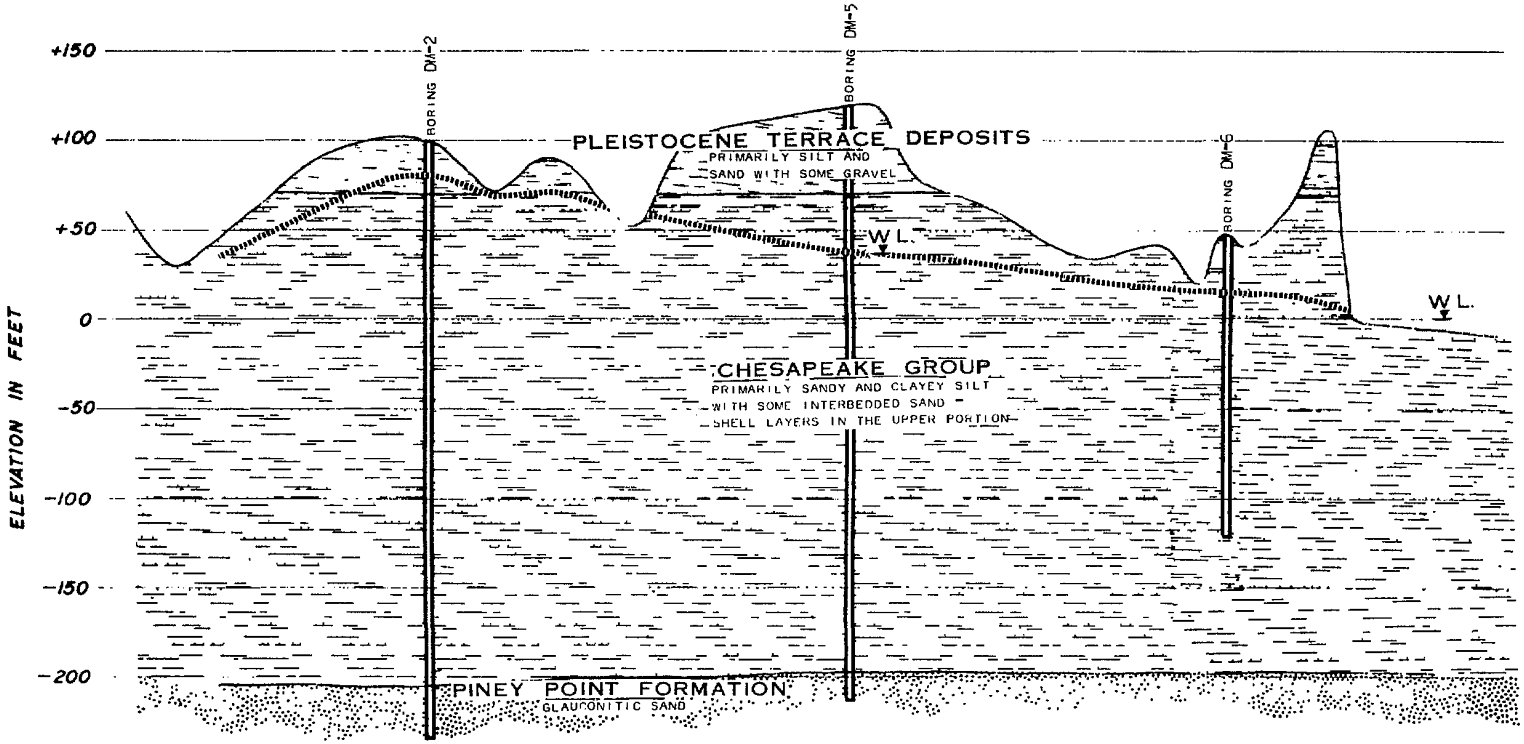
500 0 500 1000

DAMES & MOORE

FIGURE 2.4-10A

Rev.0

2.4-10A GEOLOGIC SECTION A-A



GEOLOGIC SECTION B-B

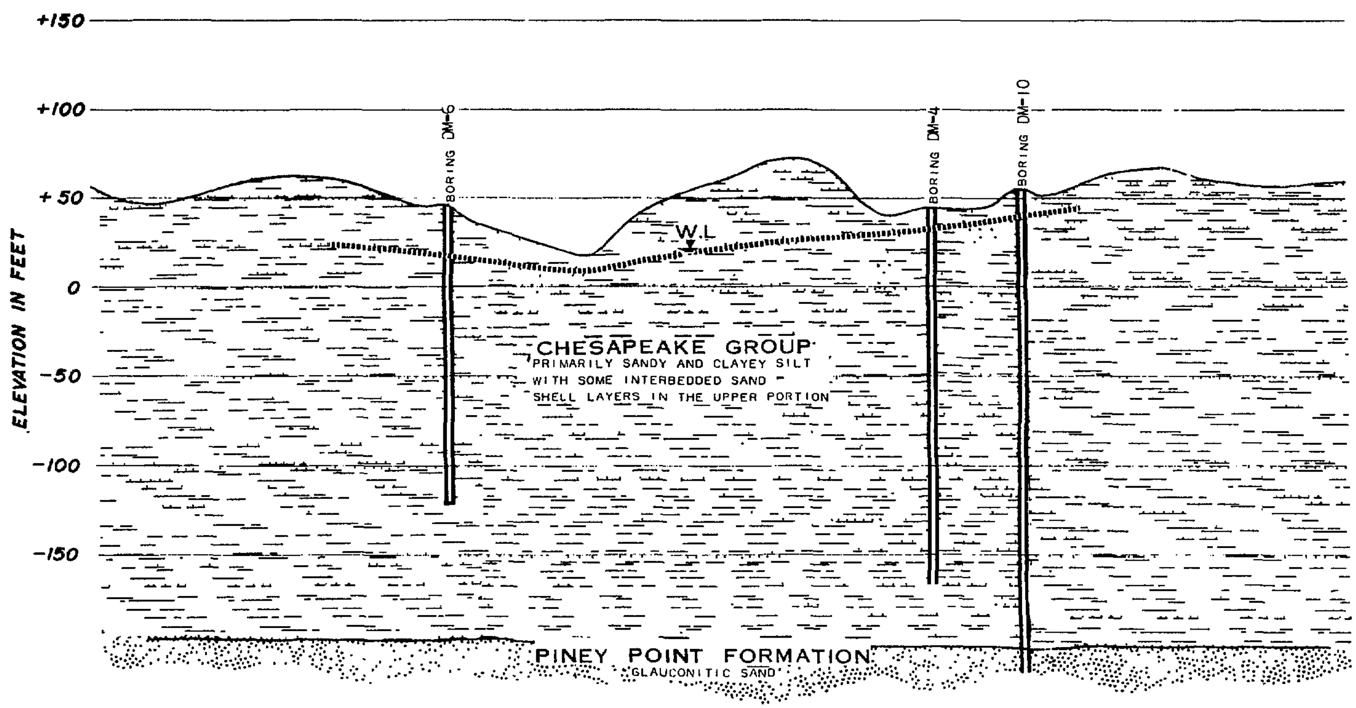
HORIZONTAL SCALE IN FEET

500 0 500 1000

DAMES & MOORE

FIGURE 2.4-10B

Rev.0



GEOLOGIC SECTION C-C

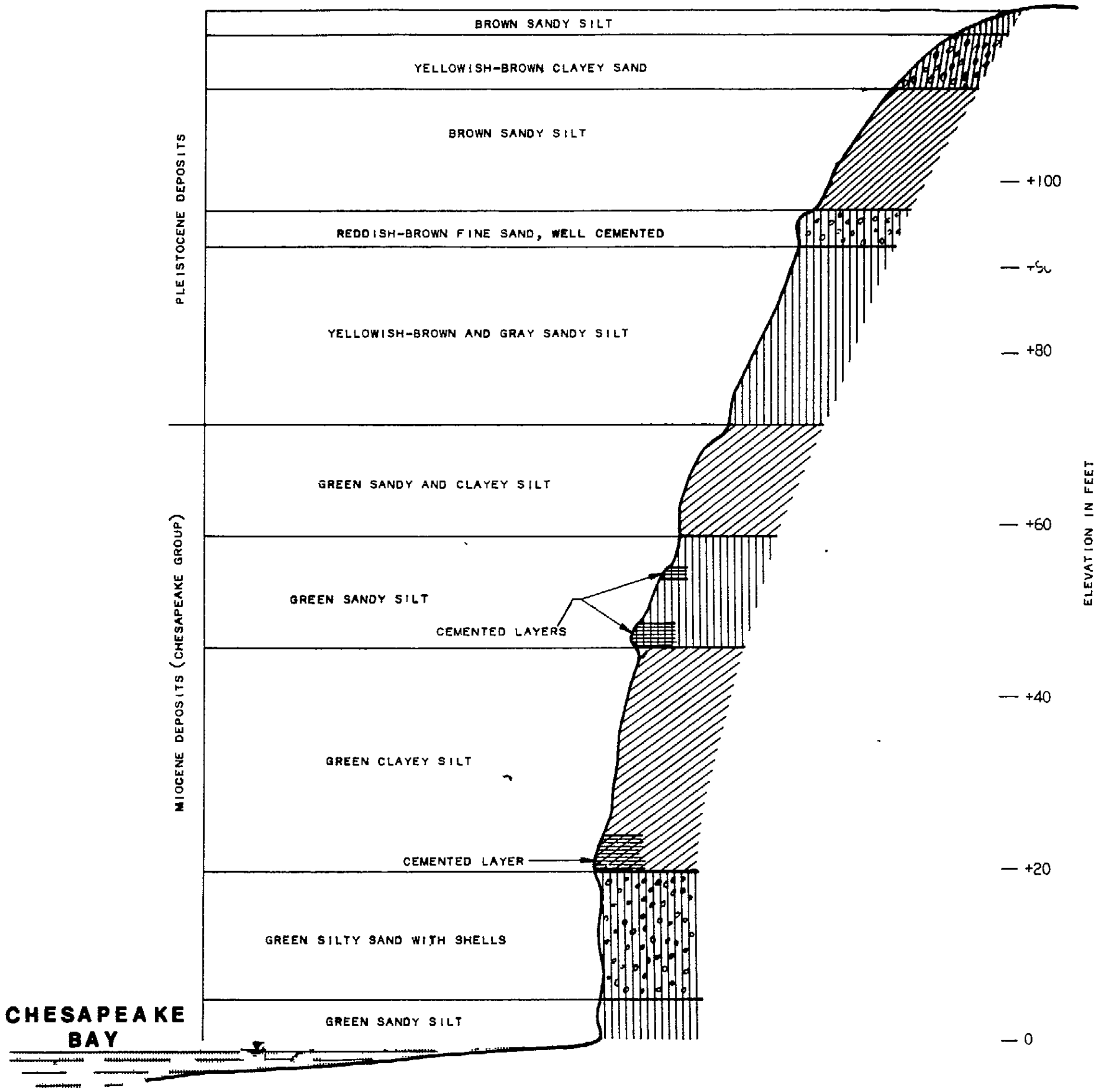
HORIZONTAL SCALE IN FEET

50 0 50 100 150 200 250

DAMES & MOORE

FIGURE 2.4-10C

Rev.0



SCHEMATIC CLIFF SECTION

PLANT AREA

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-11

CLIFF FACE PHOTOGRAPH

PLANT SITE VICINITY

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.4-12

FIGURE 2.4-13
SHORELINE CHANGES

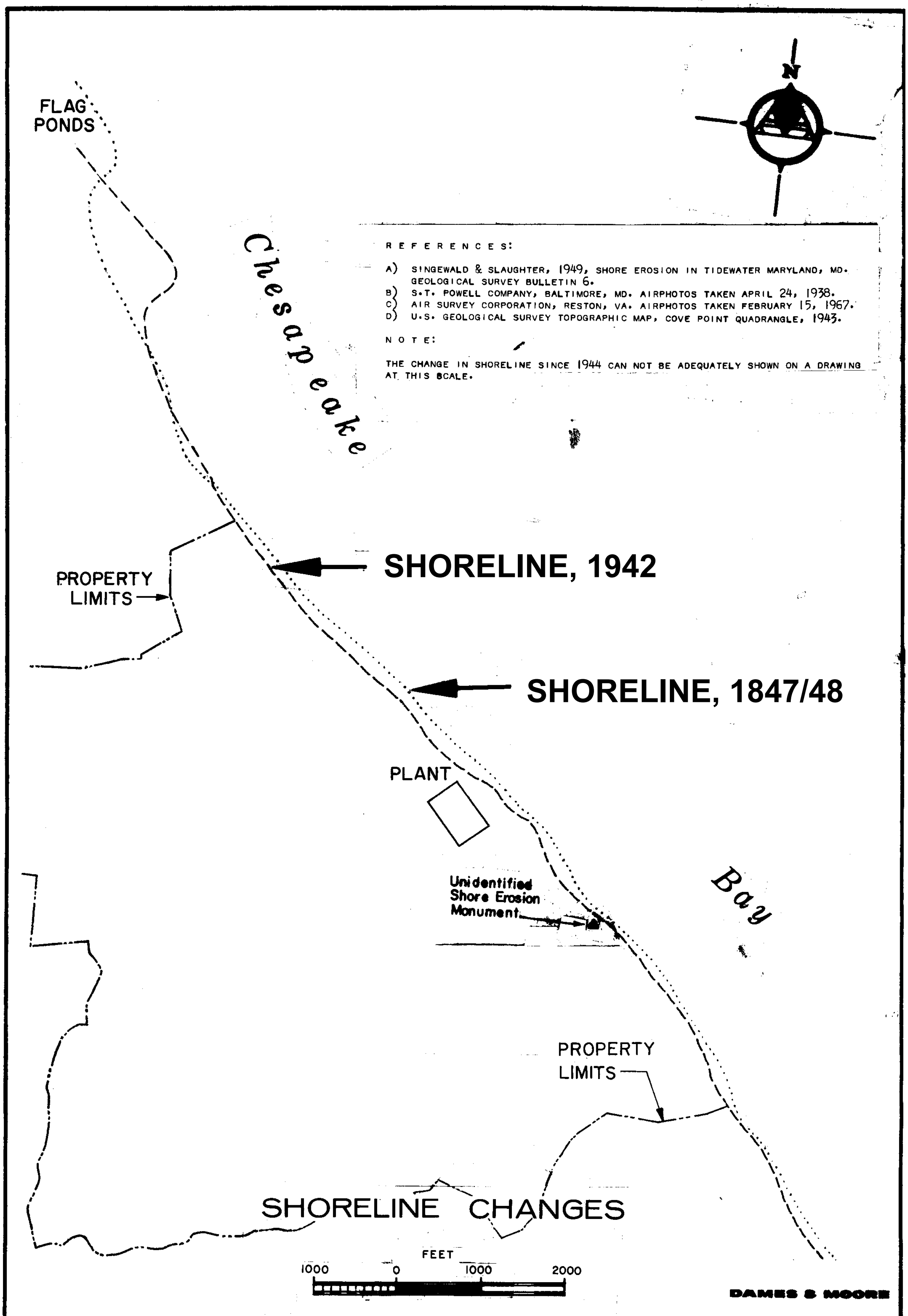
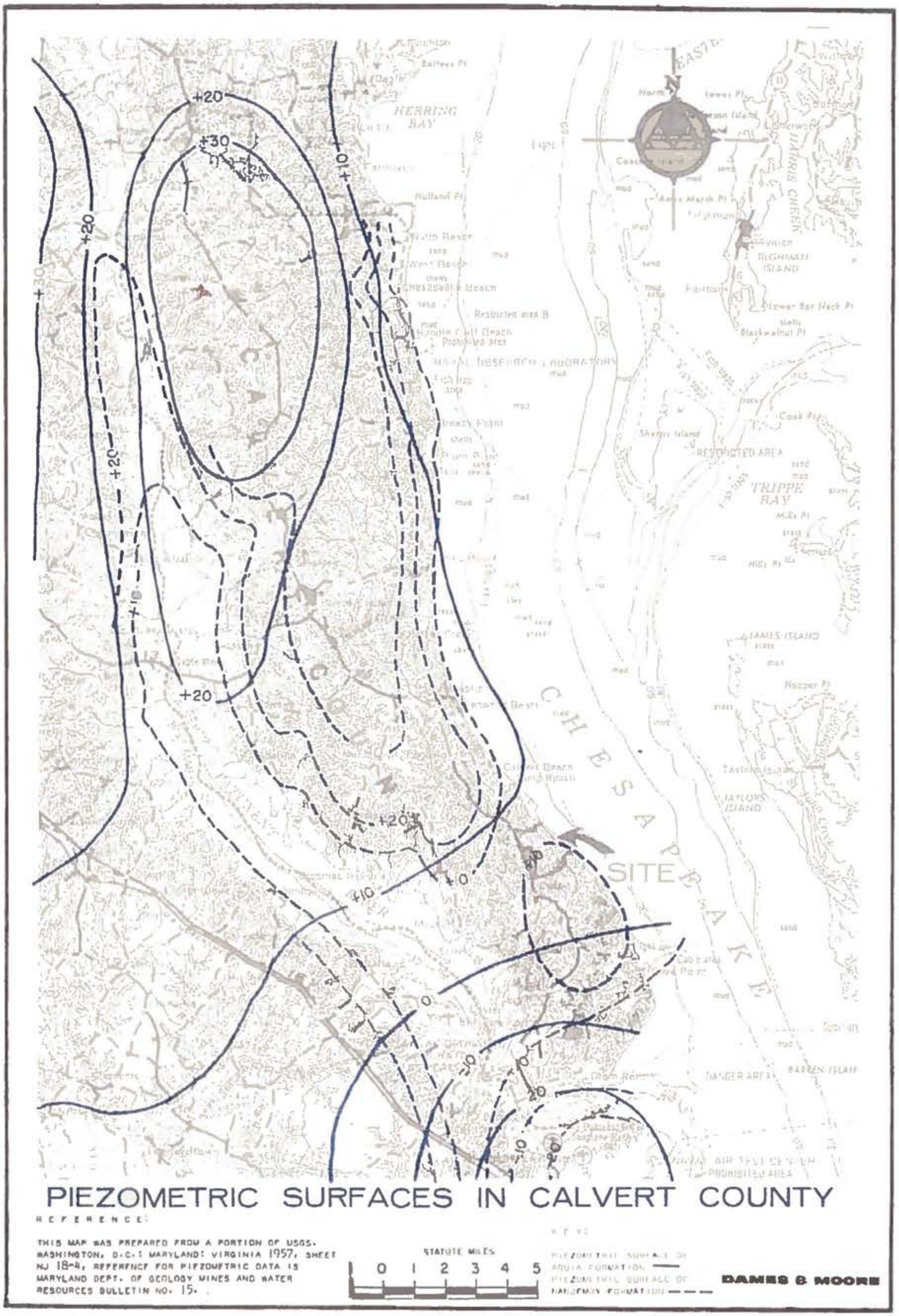
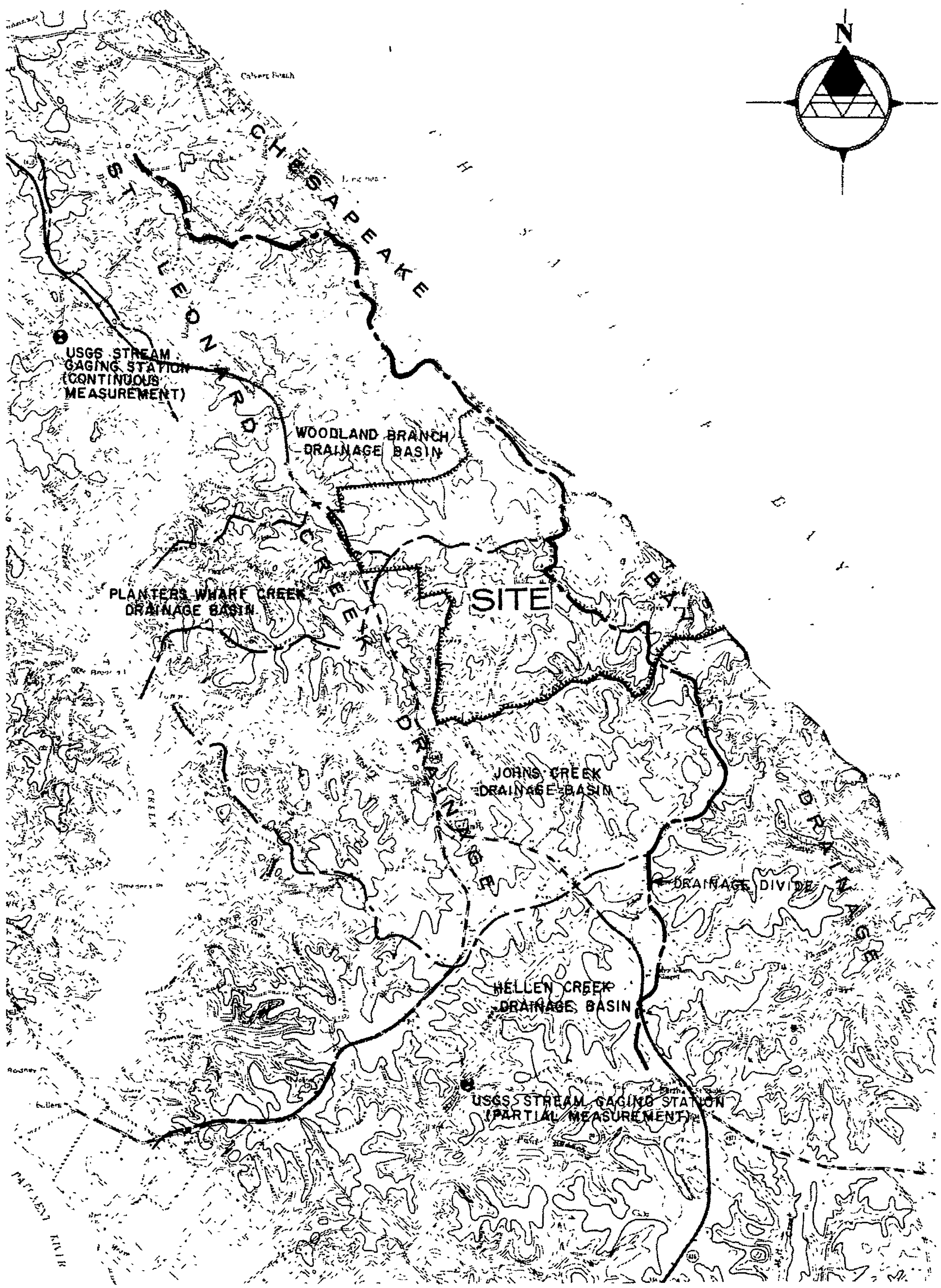


FIGURE 2.5-2
PIEZOMETRIC SURFACES IN CALVERT COUNTY





MAP OF AREA

SHOWING SURFACE HYDROLOGY

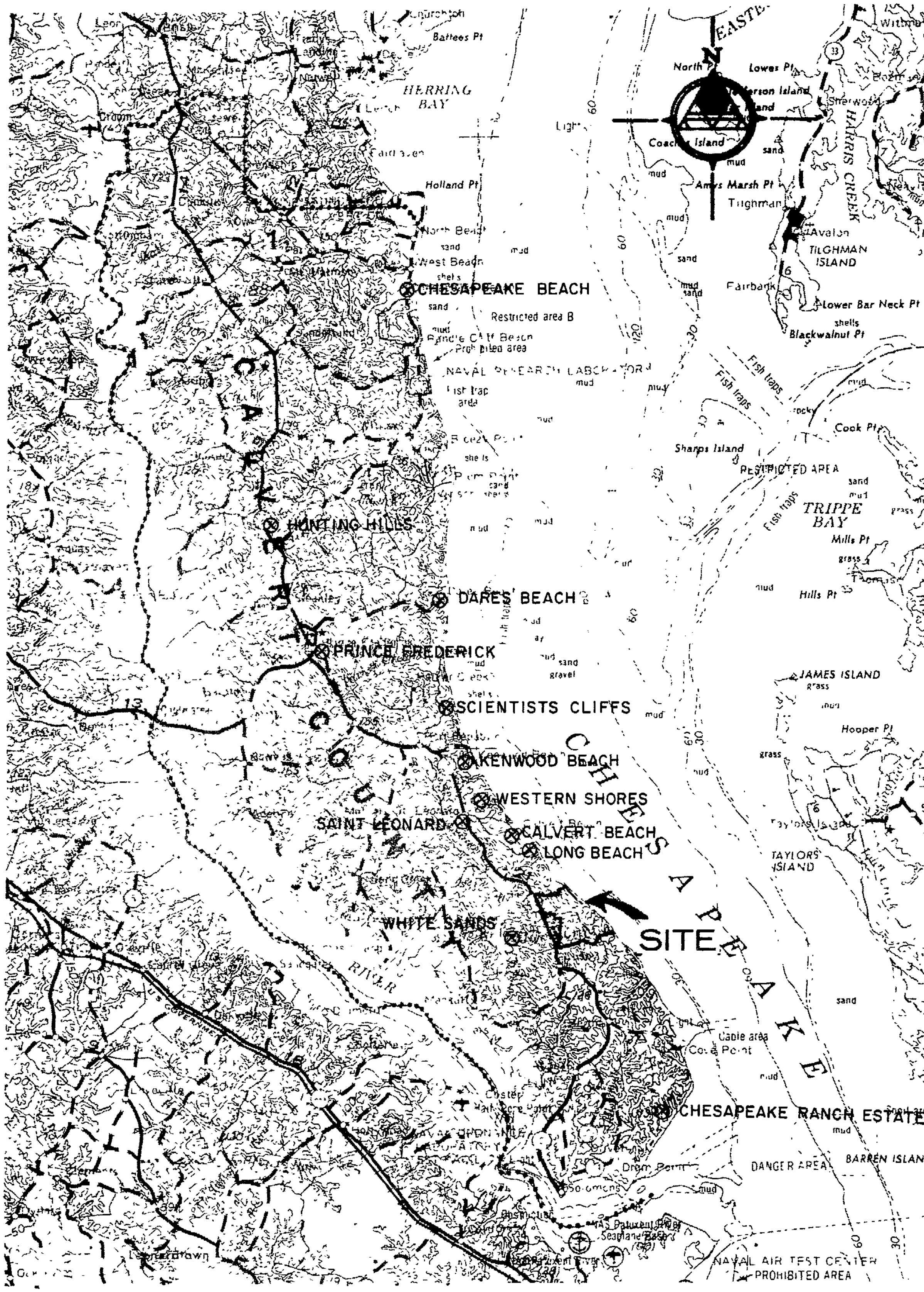
REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS QUADRANGLE COVE POINT, MARYLAND, 1943.

FEET
1000 0 2000 4000

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.5-1



PUBLIC WATER SUPPLIES IN CALVERT COUNTY

REFERENCE:

THIS MAP WAS PREPARED FROM A PORTION OF USGS. WASHINGTON, D.C.: MARYLAND: VIRGINIA 1957, SHEET NJ 18-4, REFERENCE FOR PIEZOMETRIC DATA IS MARYLAND DEPT. OF GEOLOGY MINES AND WATER RESOURCES BULLETIN NO. 15.

STATUTE MILES

1 0 1 2 3 4 5

DAMES & MOORE

FIGURE 2.5-3

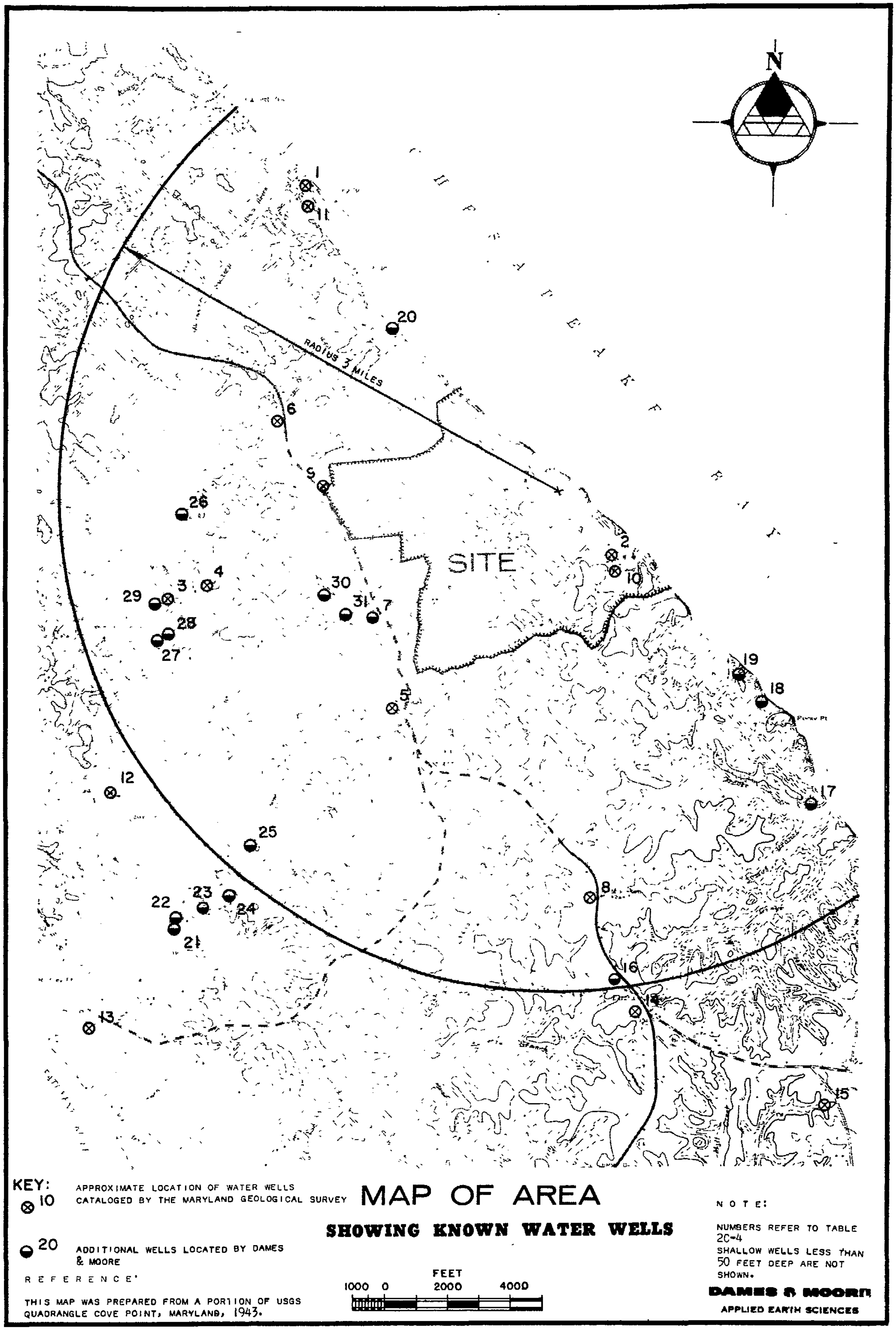
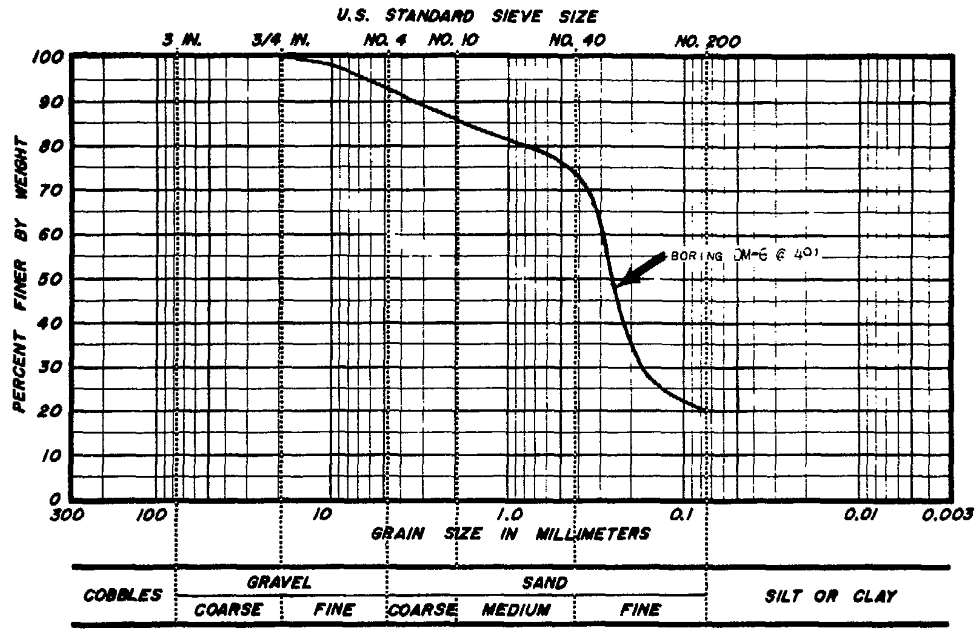
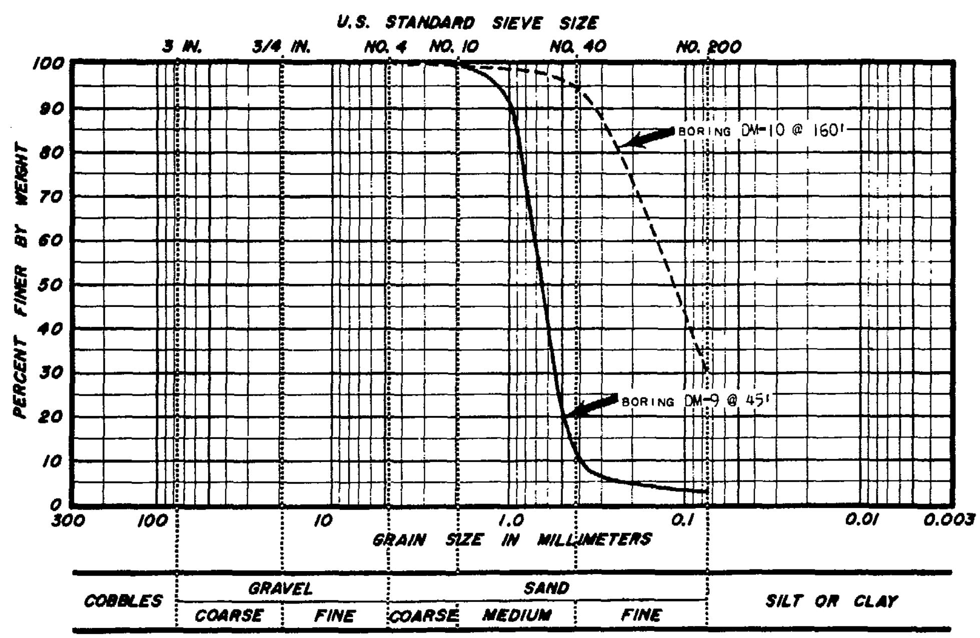

N
RADIUS 3 MILES
SITE
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
KEY:
10 APPROXIMATE LOCATION OF WATER WELLS CATALOGED BY THE MARYLAND GEOLOGICAL SURVEY
20 ADDITIONAL WELLS LOCATED BY DAMES & MOORE
REFERENCE:
THIS MAP WAS PREPARED FROM A PORTION OF USGS QUADRANGLE COVE POINT, MARYLAND, 1943.
MAP OF AREA
SHOWING KNOWN WATER WELLS
FEET
1000 0 2000 4000
NOTE:
NUMBERS REFER TO TABLE 2C-4
SHALLOW WELLS LESS THAN 50 FEET DEEP ARE NOT SHOWN.
DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.5-4

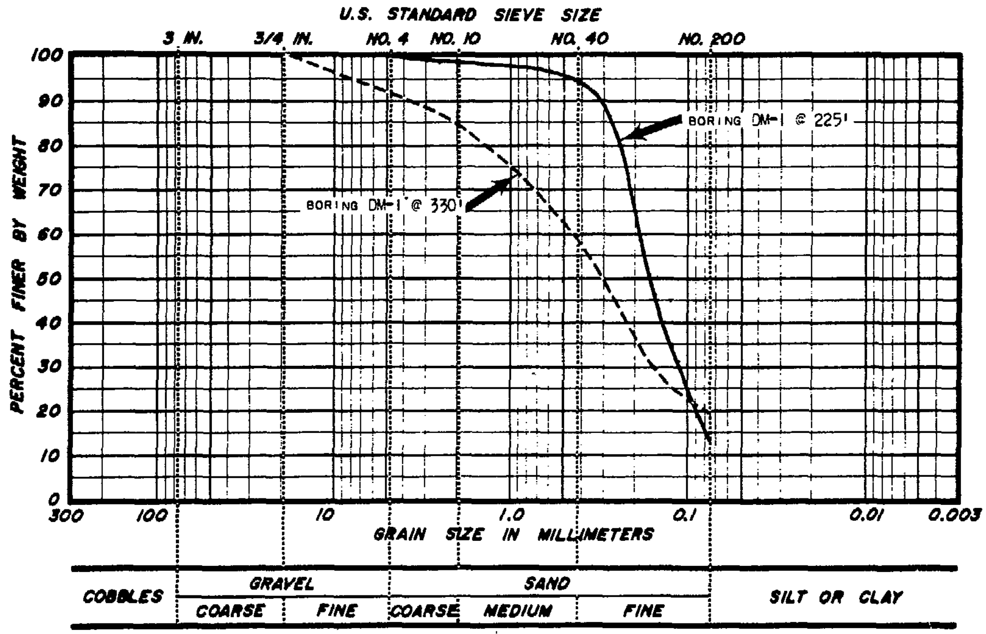


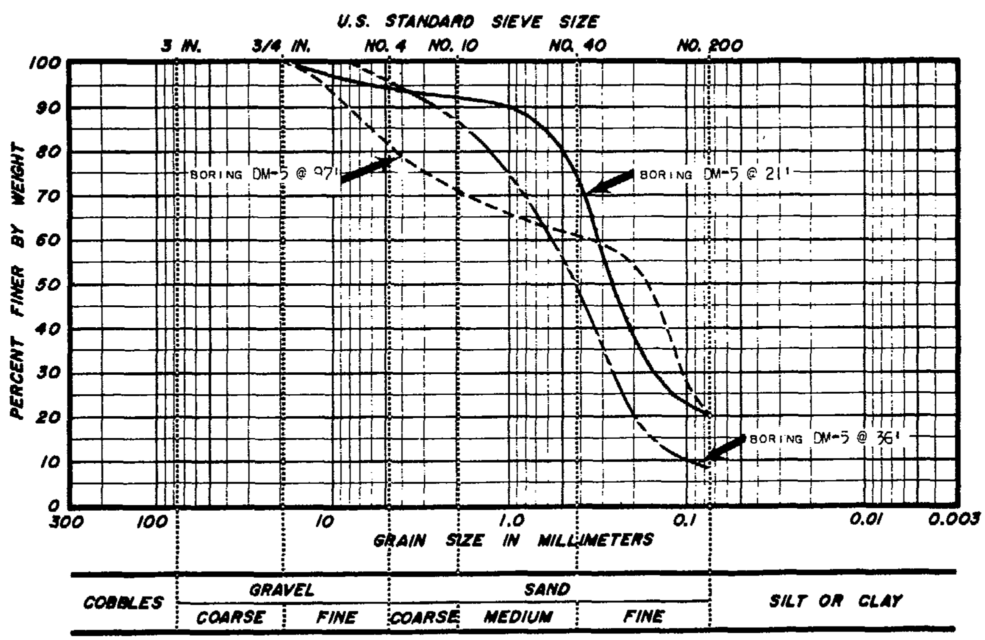


PARTICLE SIZE ANALYSES

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.5-5A

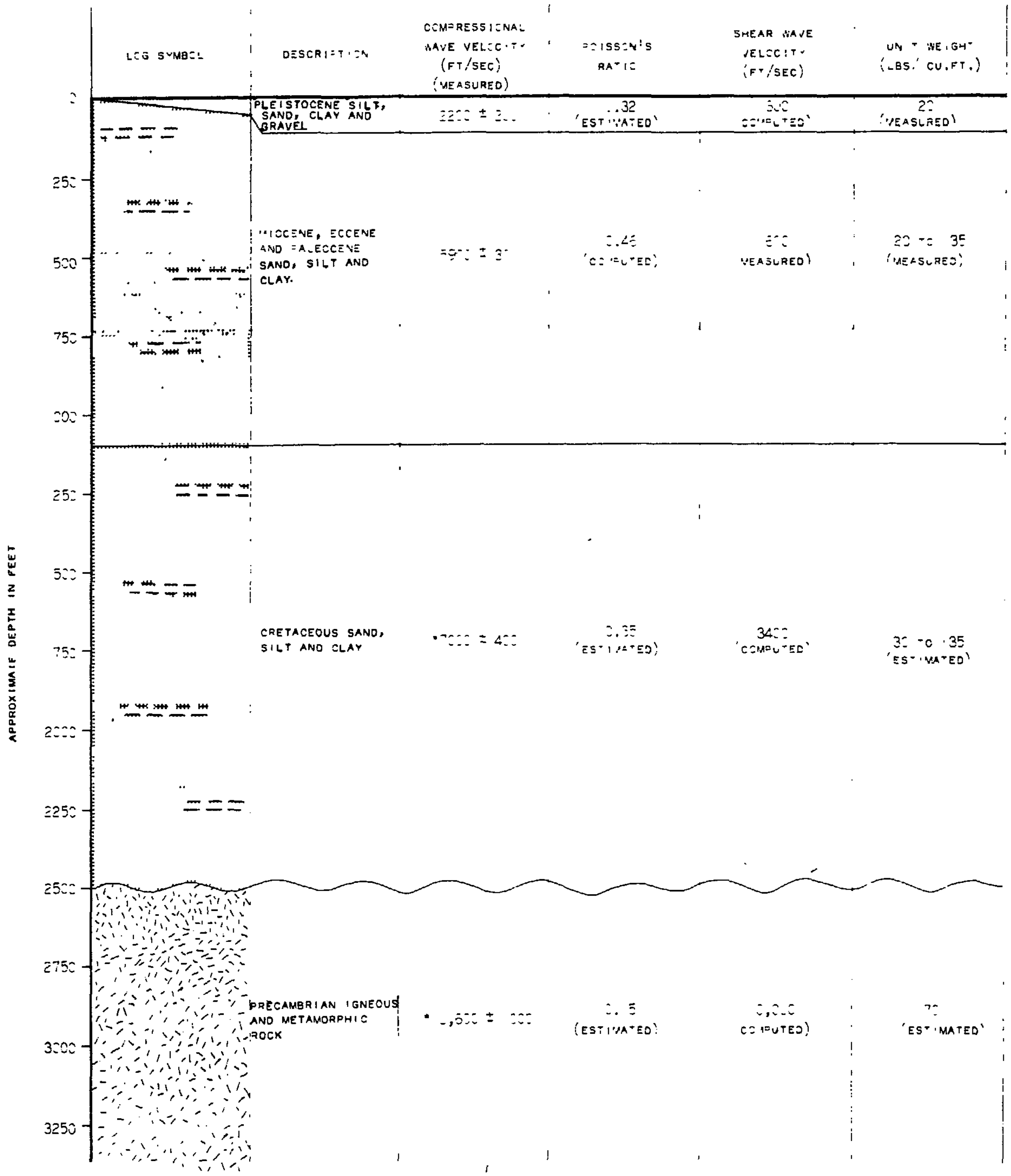




PARTICLE SIZE ANALYSES

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.5-5B

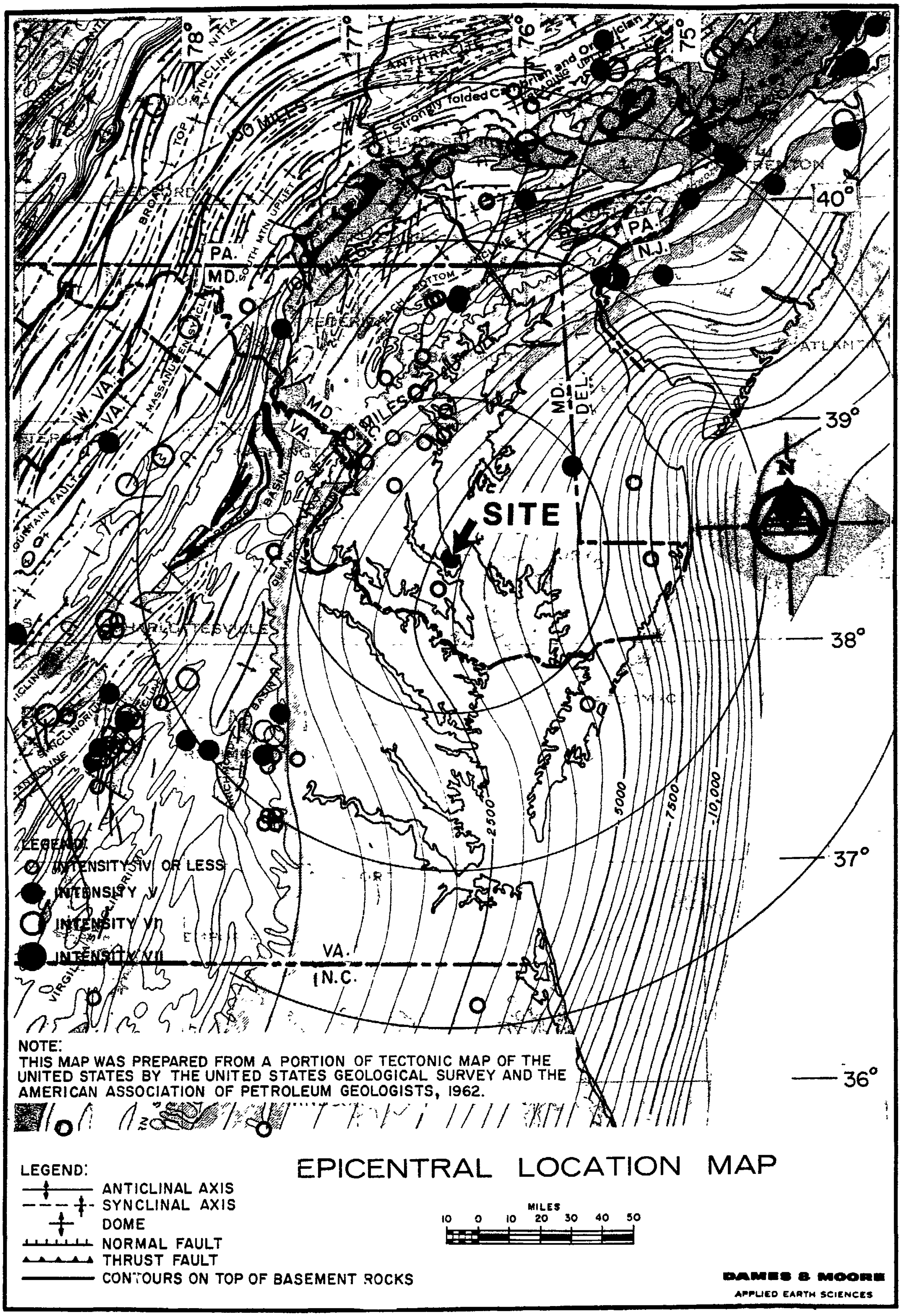


* TAKEN FROM EWING M. AND WORZEL J. (1948) EXPLOSIONS SOUNDS IN SHALLOW WATER, GEOLOGICAL SOCIETY OF AMERICA MEMOIR 27

COLUMNAR SECTION
SHOWING GEOPHYSICAL DATA

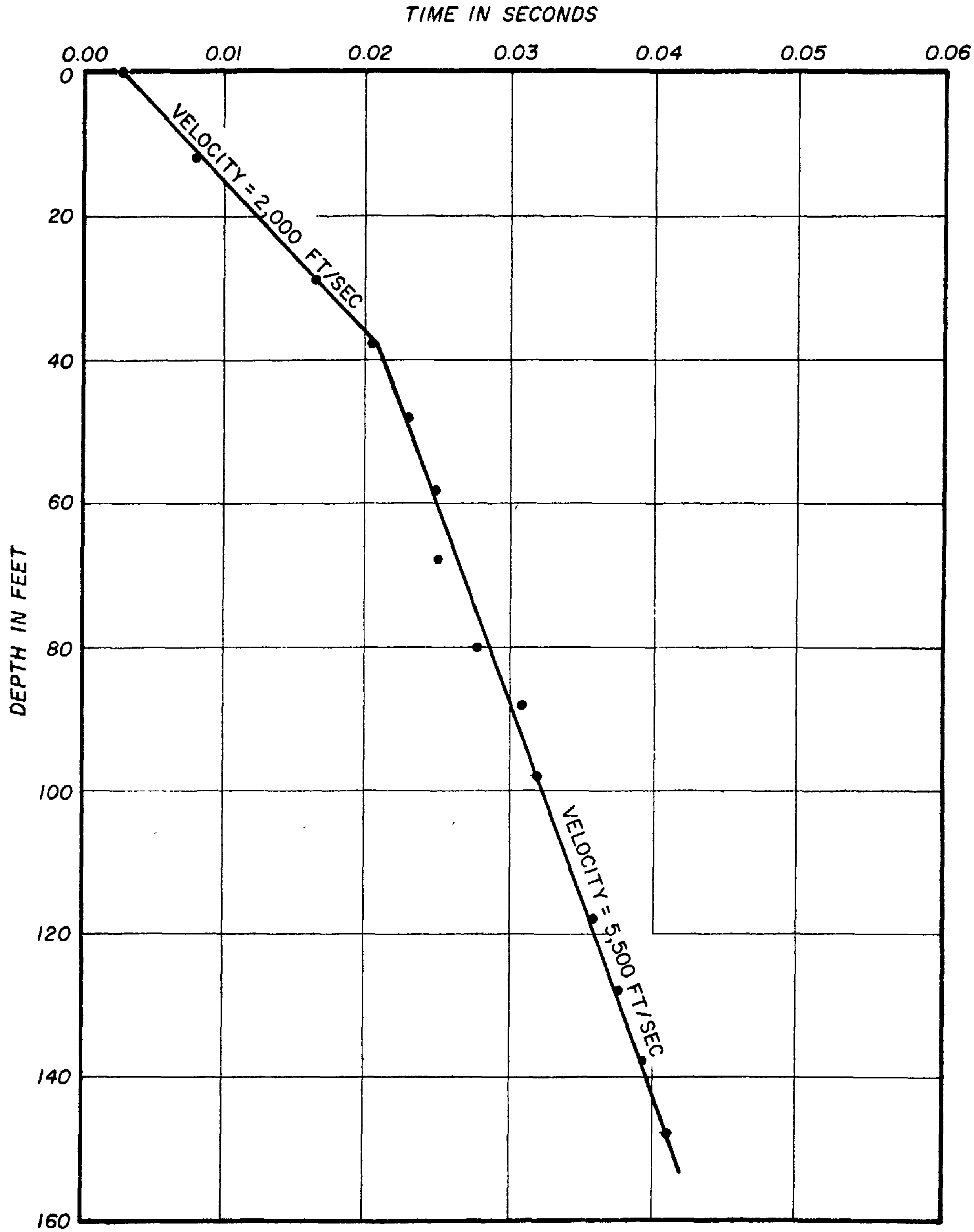
DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.6-1



EPICENTRAL LOCATION MAP

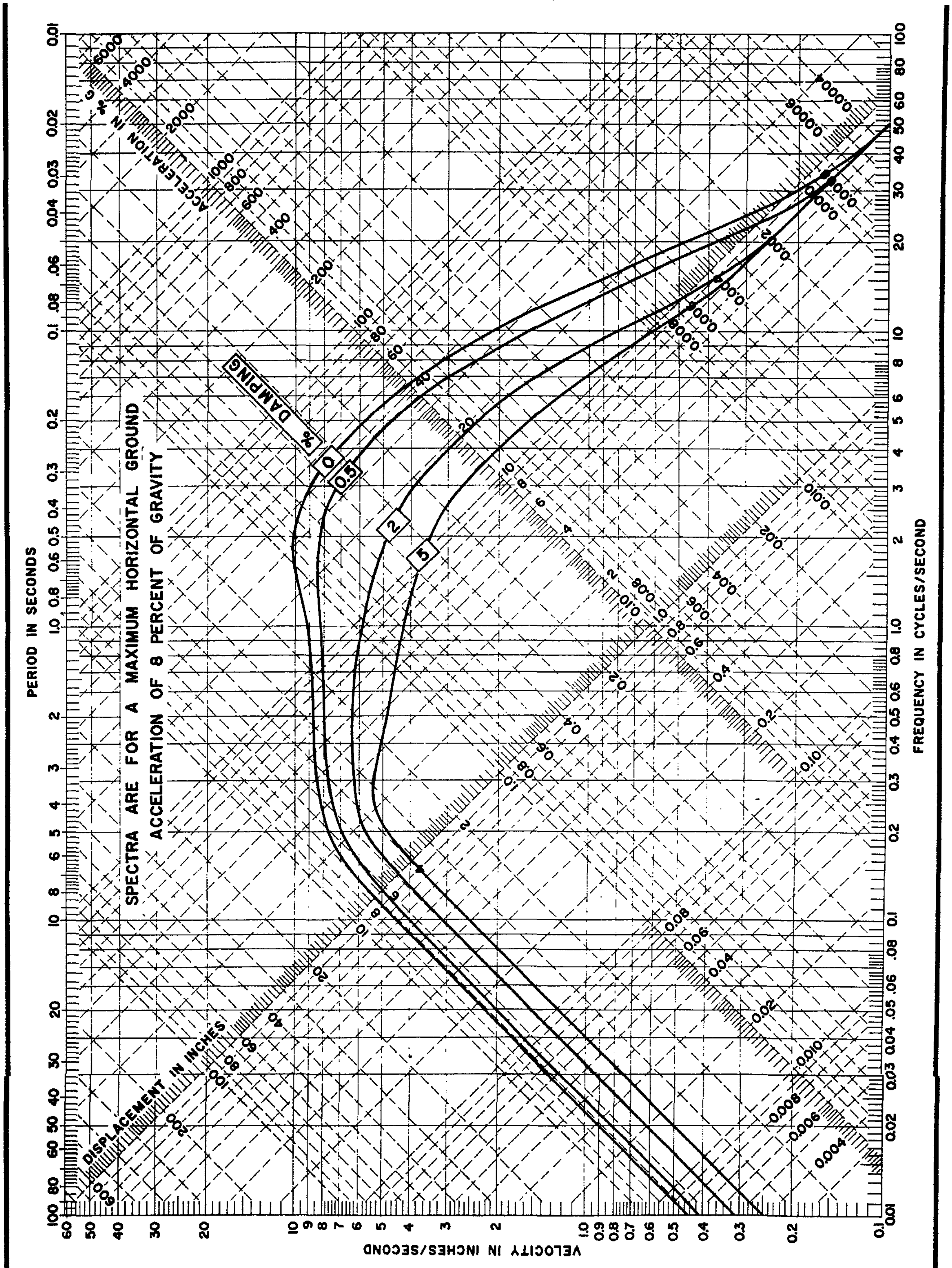
FIGURE 2.6-2



UPHOLE SEISMIC SURVEY

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.6-3
Rev.0



RESPONSE SPECTRA
OPERATING BASIS EARTHQUAKE

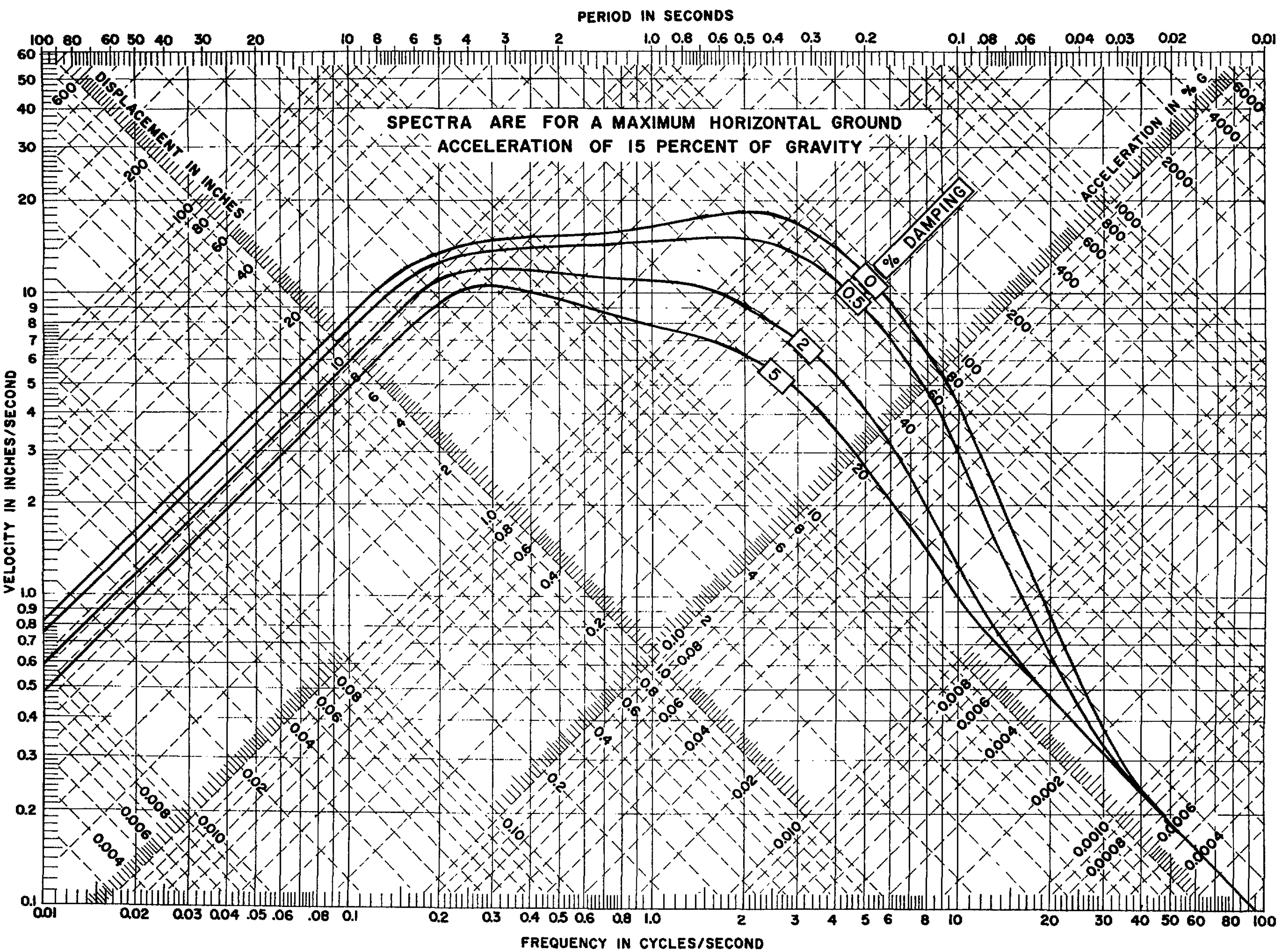
DAMES & MOORE

FIGURE 2.6-4

Rev.0

RESPONSE SPECTRA

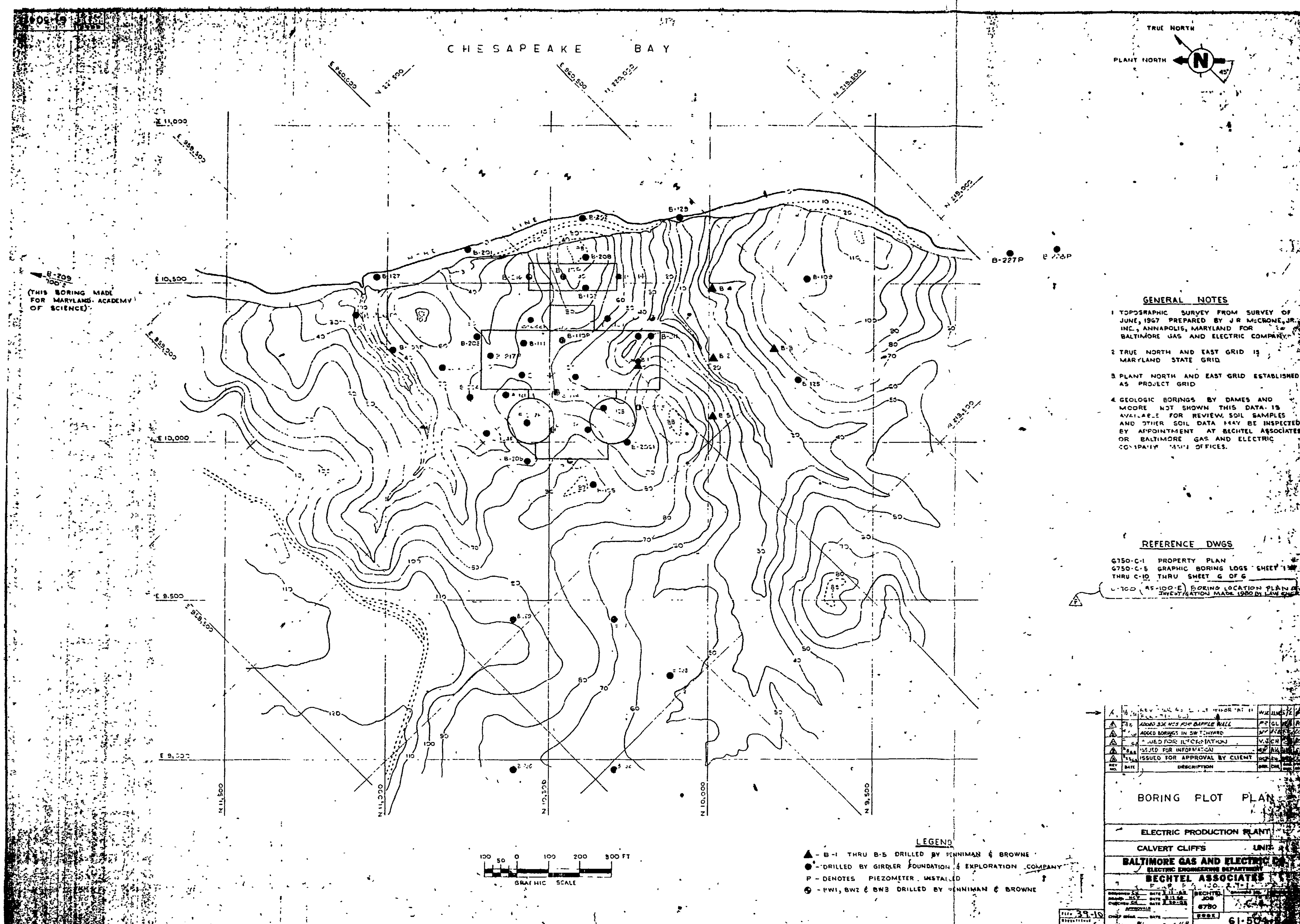
DESIGN BASIS EARTHQUAKE



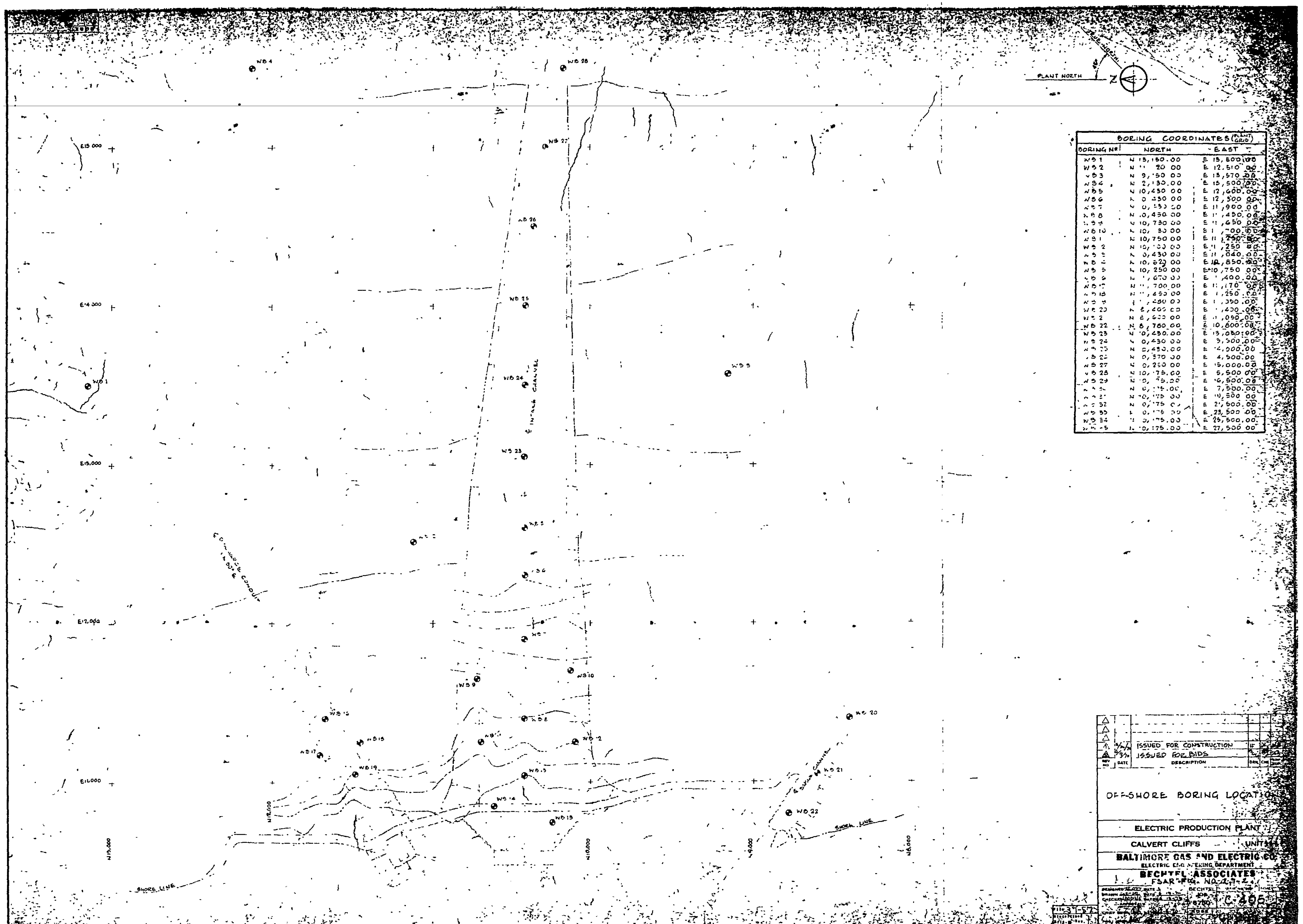
DAMES & MOORE

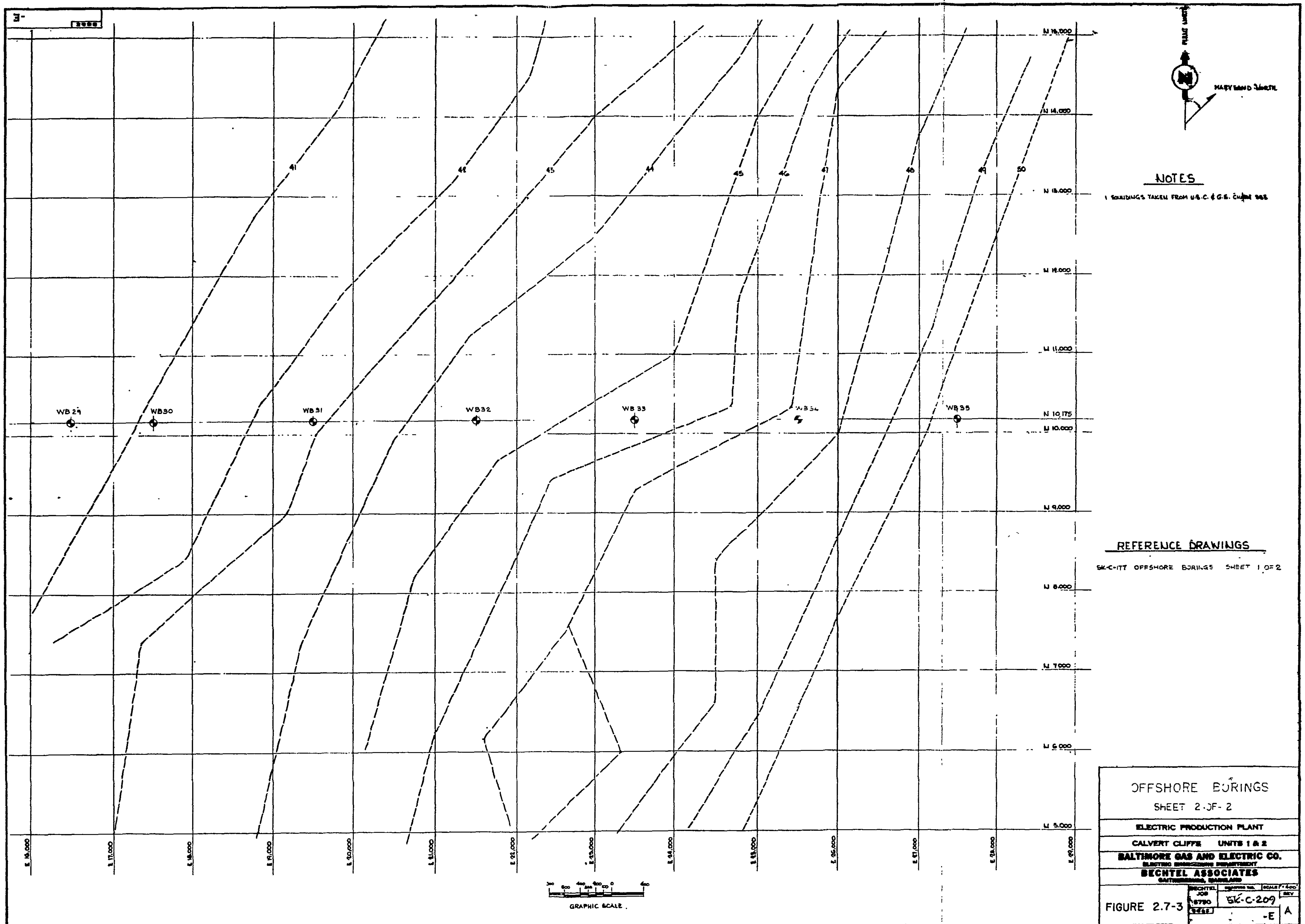
FIGURE 2.6-5

2.7-1 BORING PLOT PLAN



2.7-2 OFFSHORE BORING LOCATIONS




NOTES
REFERENCE DRAWINGS
WB 29
WB30
WB 31
WB32
WB 33
WB 35
N 10,175
N 10,000
GRAPHIC SCALE
OFFSHORE BORINGS
SHEET 2 OF 2
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNITS 1 & 2
BALTIMORE GAS AND ELECTRIC CO.
BECHTEL ASSOCIATES
6750
SK-C-209
A
FIGURE 2.7-3

GRAPHIC BORING LOGS

BALTIMORE GAS AND ELECTRIC COMPANY
CALVERT CLIFFS NUCLEAR POWER PLANT

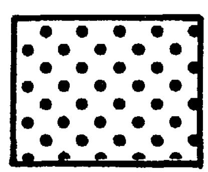
SUPPLEMENTARY SITE BORINGS

GENERAL NOTES

1. ELEVATIONS REFERENCED TO MEAN SEA LEVEL.
2. NUMBER ADJACENT TO BORING LOG IS STANDARD PENETRATION RESISTANCE.
3. STANDARD PENETRATION RESISTANCE IN BLOWS/FOOT OF A 140 LB. WEIGHT FREE FALLING 30 INCHES.
4. BORINGS BY GIRDLER FOUNDATION AND EXPLORATION CO., JUNE 1969.

LEGEND

Greenish Gray Silty SAND



Greenish Gray Sandy SILT

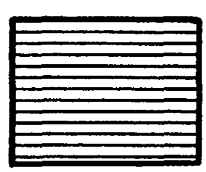


FIGURE 2.7-4

2.7-5 GRAPHIC BORING LOGS, BALTIMORE GAS AND ELECTRIC COMPANY, CALVERT CLIFFS NUCLEAR POWER PLANT, SUPPLEMENTARY SITE BORINGS

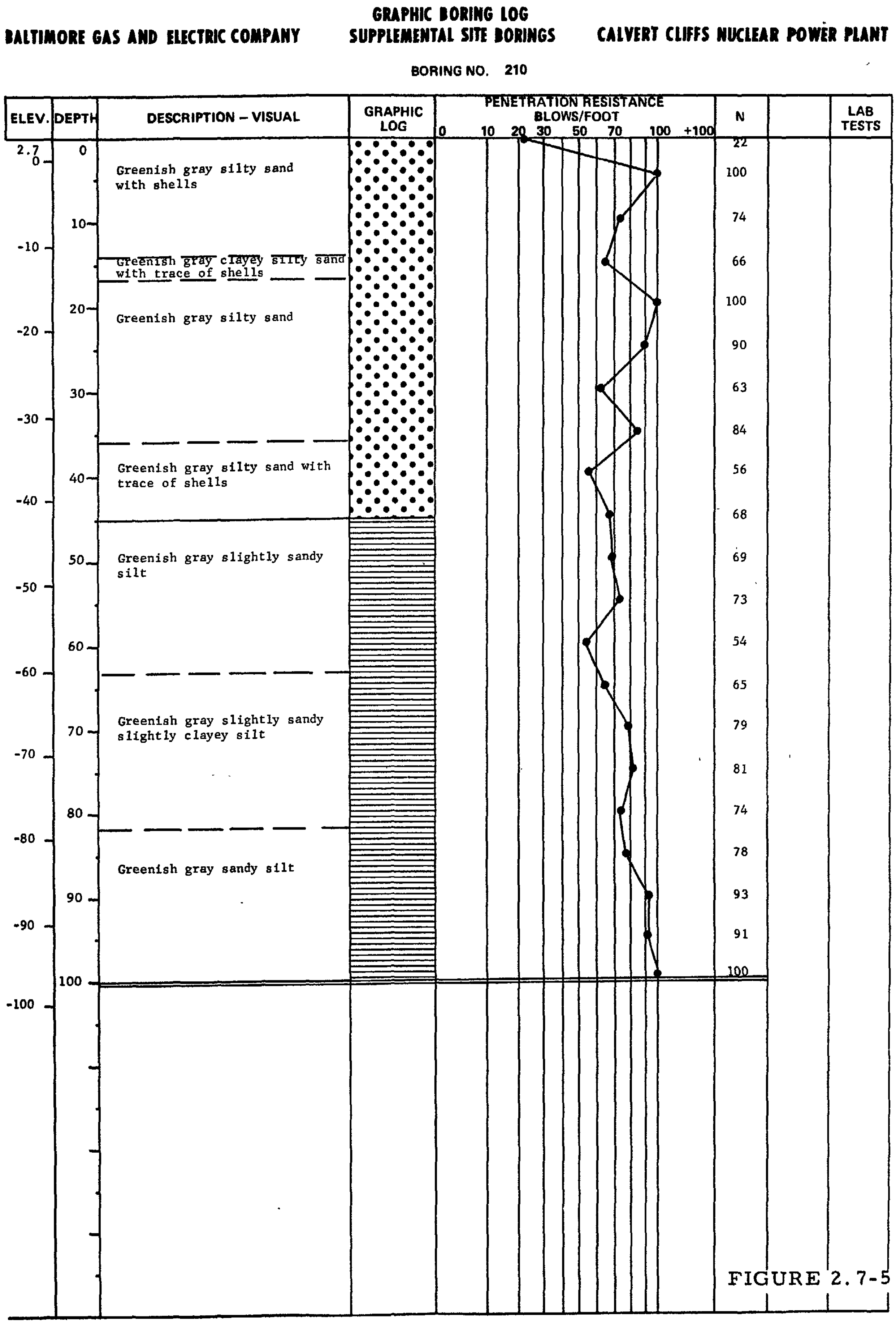


FIGURE 2.7-5

BALTIMORE GAS AND ELECTRIC COMPANY

GRAPHIC BORING LOG
SUPPLEMENTAL SITE BORINGS

CALVERT CLIFFS NUCLEAR POWER PLANT

BORING NO. 211

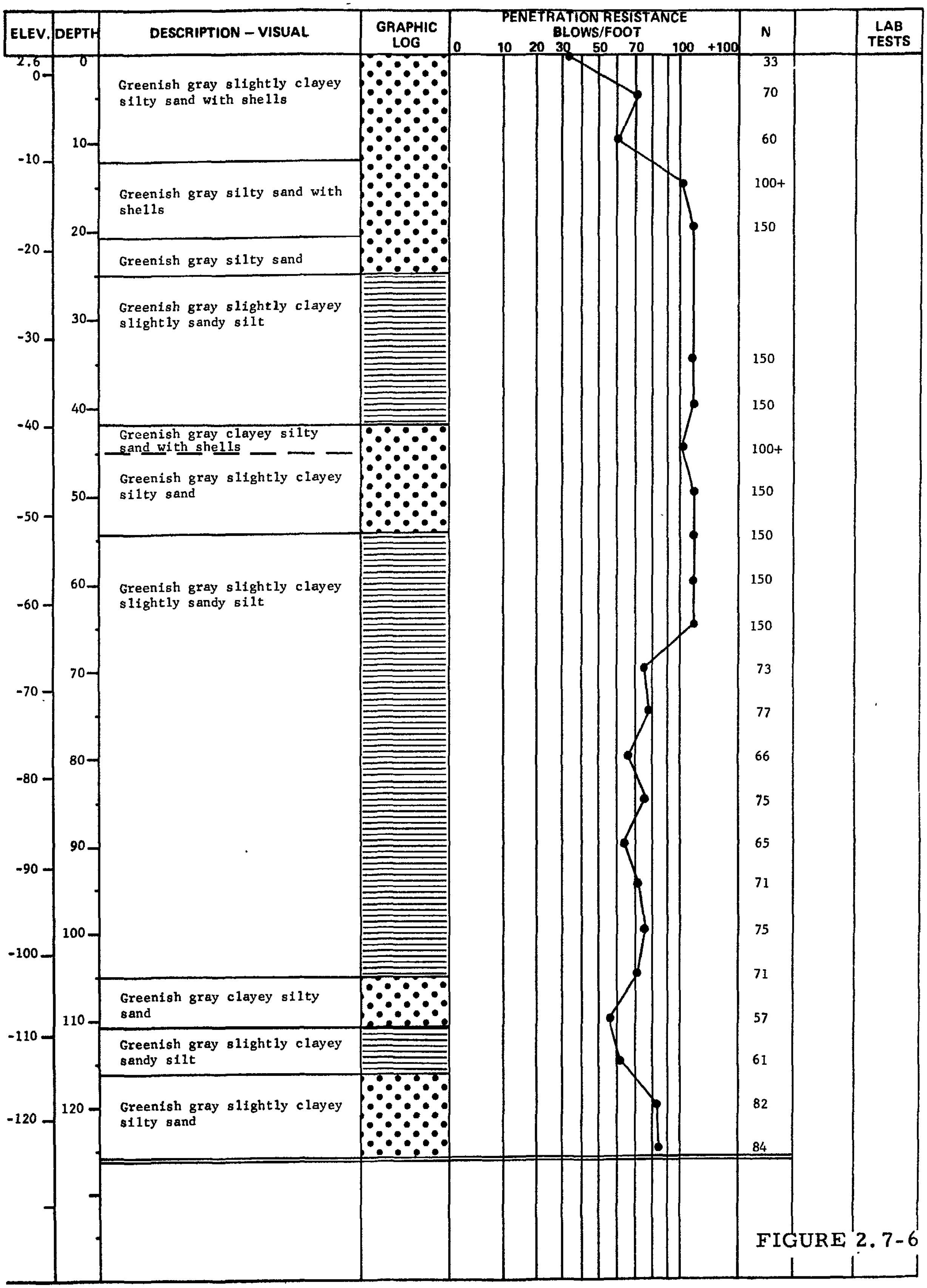


FIGURE 2.7-6

BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 212

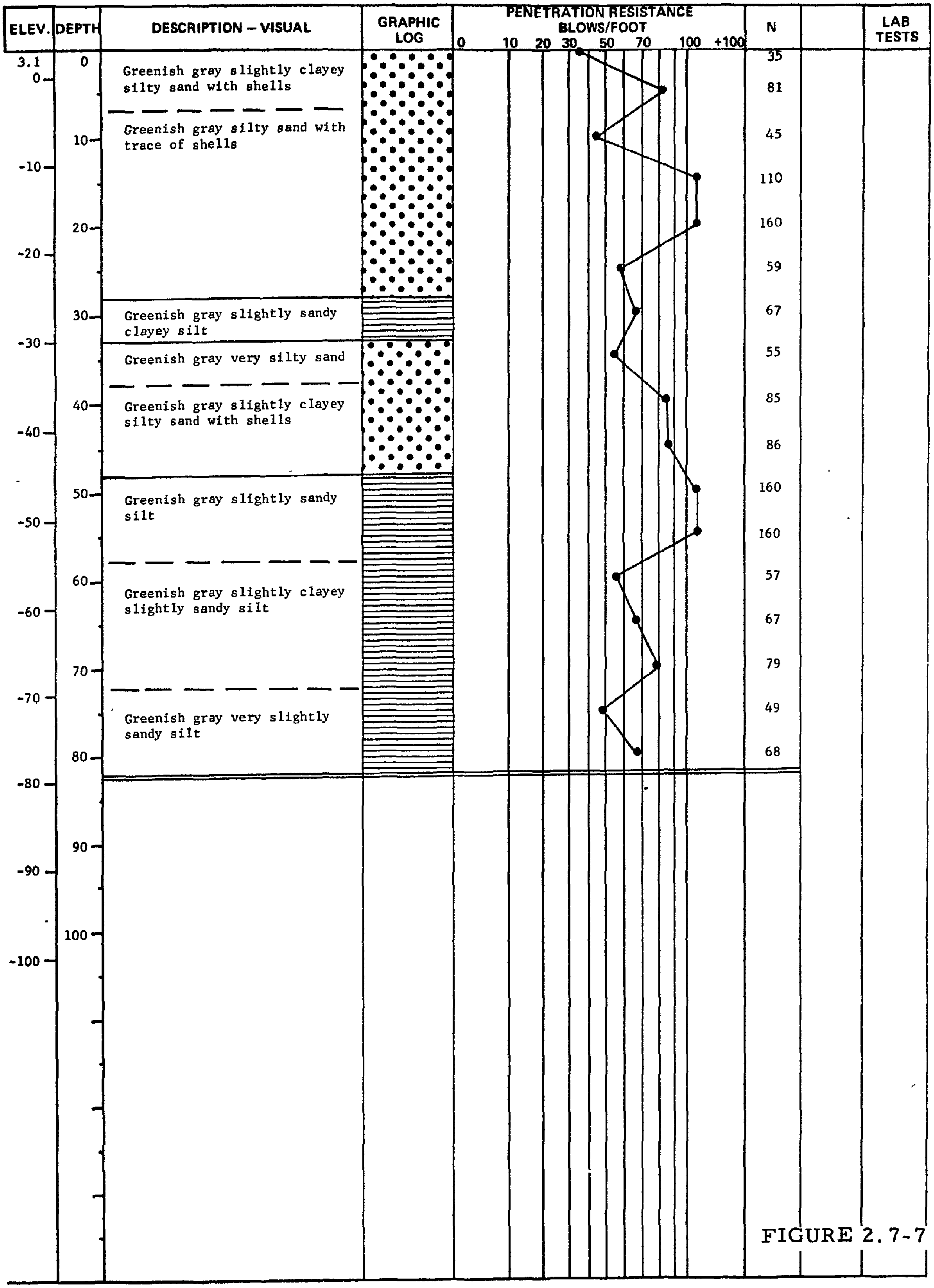


FIGURE 2.7-7

BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 213

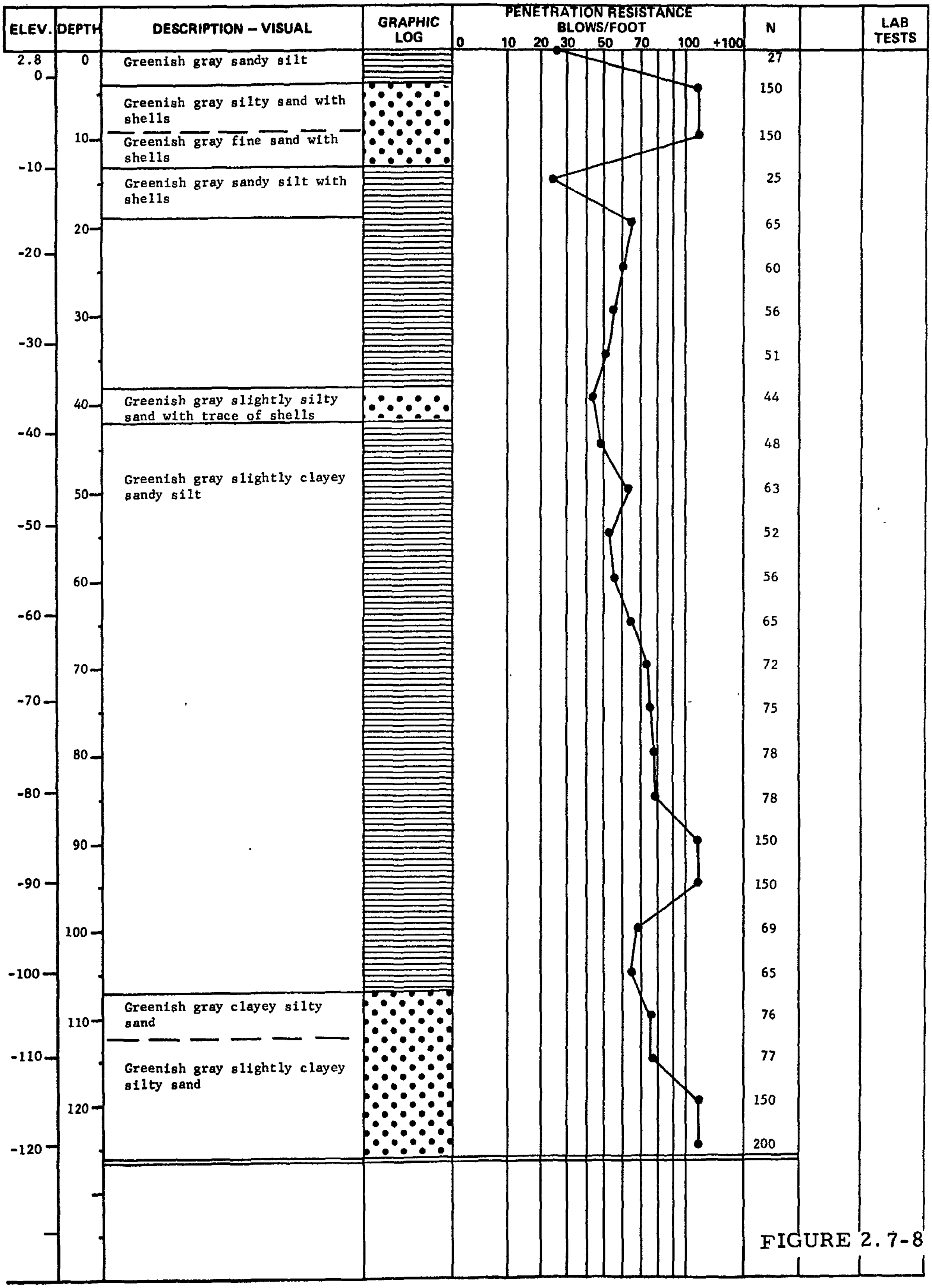


FIGURE 2.7-8

GRAPHIC BORING LOG

BALTIMORE GAS AND ELECTRIC COMPANY **SUPPLEMENTAL SITE BORINGS** **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 214

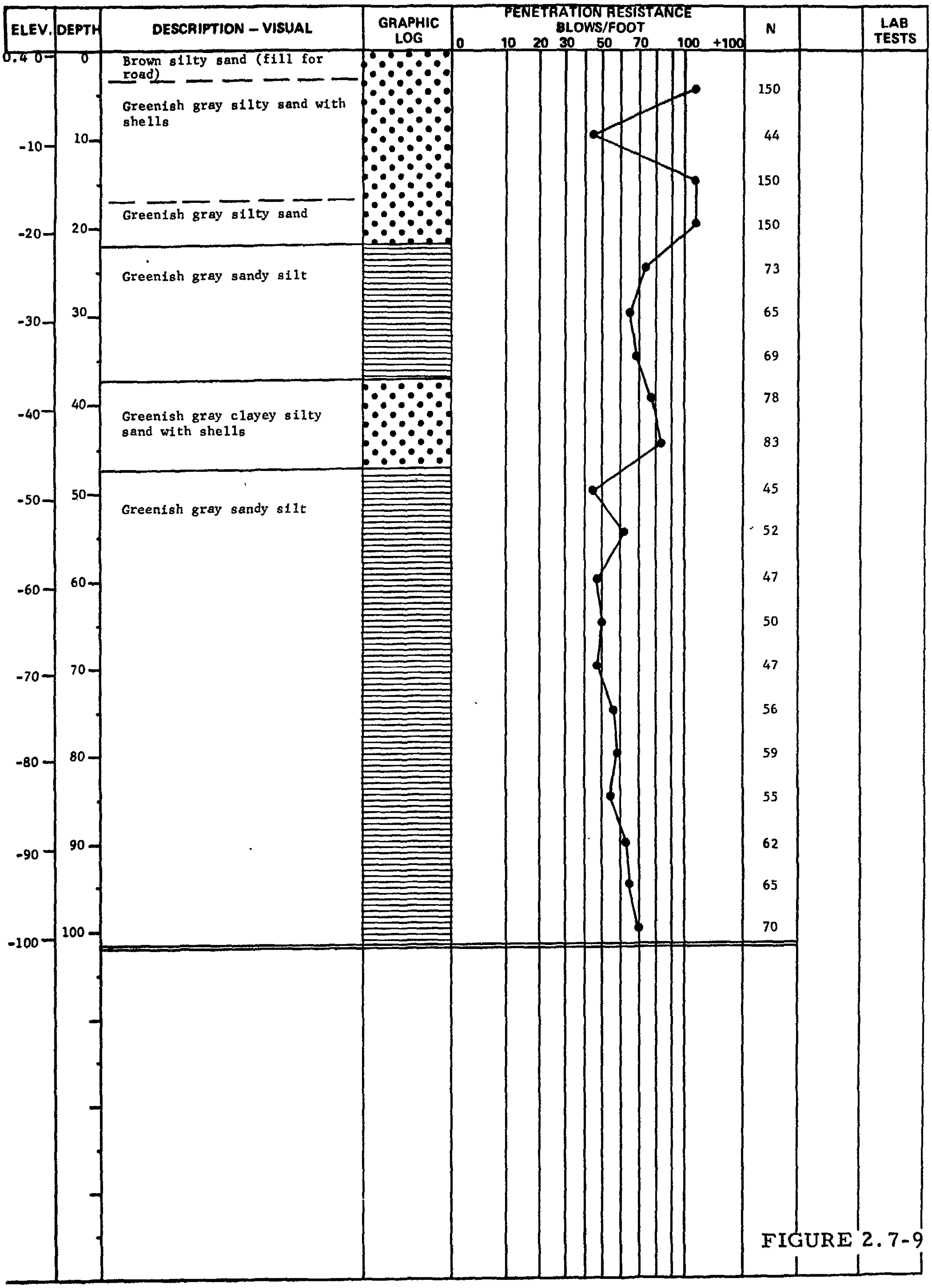
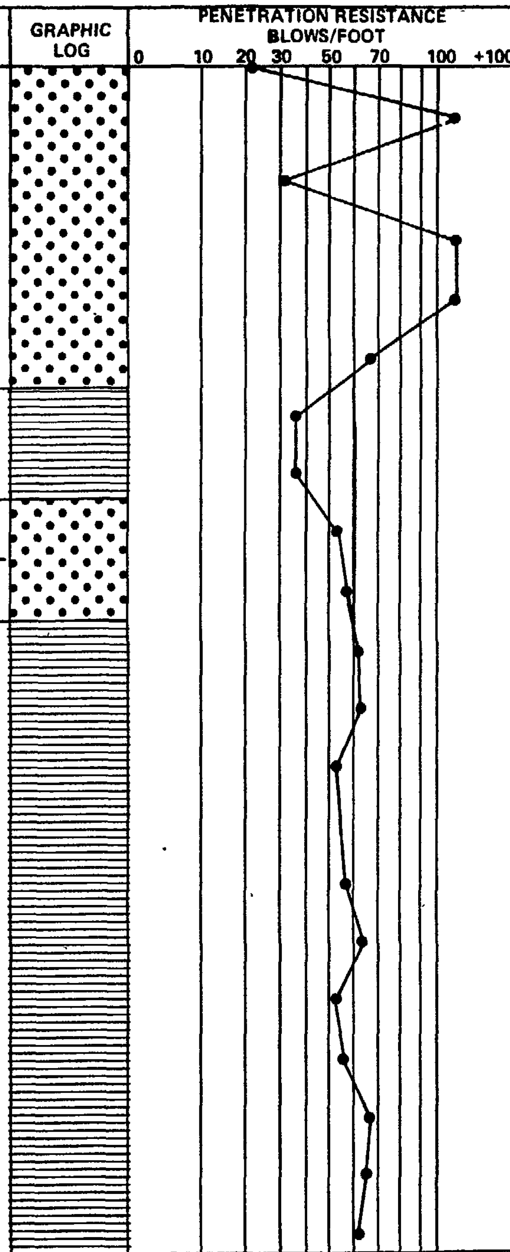


FIGURE 2.7-9

BALTIMORE GAS AND ELECTRIC COMPANY — **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** — **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 215

ELEV.	DEPTH	DESCRIPTION – VISUAL
0.2 0	0	Greenish gray silty sand with shells
-10	10	
-20	20	Greenish gray silty sand with trace of shells
-30	30	Greenish gray sandy silt with trace of shells
-40	40	Greenish gray quite silty sand
		Greenish gray silty sand with shells
-50	50	Greenish gray fine sandy silt
-60	60	
-70	70	
-80	80	
-90	90	
-100	100	



N: 22, 150, 31, 150, 200, 67, 36, 36, 52, 57, 61, 63, 53, 57, 63, 52, 57, 67, 65, 62

LAB TESTS

FIGURE 2.7-10
Rev. 9-10-71

BALTIMORE GAS AND ELECTRIC COMPANY — **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** — **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 216

ELEV.	DEPTH	DESCRIPTION – VISUAL	N
0.4 0	0	Greenish gray silty sand with shells	52
			56
-10	10		61
			150
-20	20	Greenish gray sandy clayey silt with shells	65
		Greenish gray silty sand	69
-30	30		78
			72
-40	40	Greenish gray silty sand with shells	66
			62
-50	50	Greenish gray sandy silt	63
			60
-60	60		48
			51
-70	70		69
			67
-80	80		59
			62
-90	90		67
			63
-100	100		59

PENETRATION RESISTANCE BLOWS/FOOT: 0 10 20 30 50 70 100 +100

LAB TESTS

FIGURE 2.7-11

BALTIMORE GAS AND ELECTRIC COMPANY — **GRAPHIC BORING LOG** **SUPPLEMENTAL SITE BORINGS** — **CALVERT CLIFFS NUCLEAR POWER PLANT**

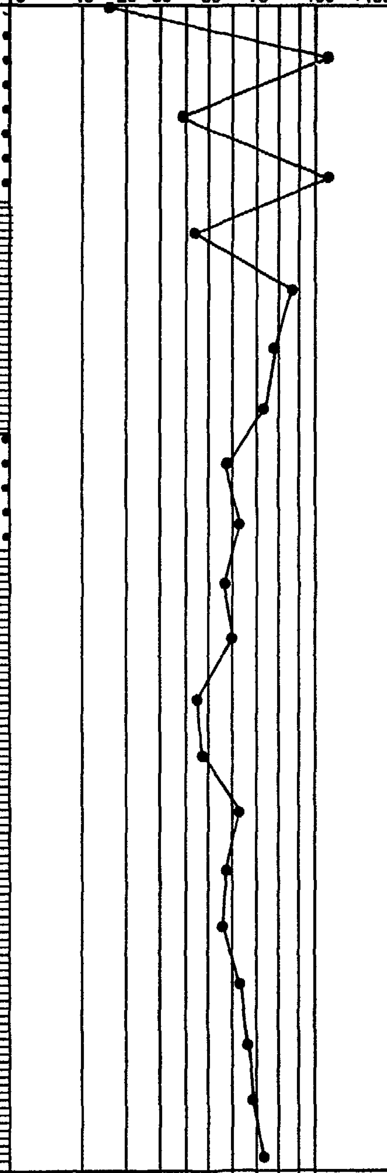
BORING NO. 217

ELEV.	DEPTH	DESCRIPTION – VISUAL	GRAPHIC LOG
2.2	0	Greenish gray silty sand with shells	
		Greenish gray sandy silt with shells	
		Greenish gray silty sand with shells	
		Greenish gray sandy silt	

Elevation scale: 0, -10, -20, -30, -40, -50, -60, -70, -80, -90, -100

Depth scale: 0, 10, 20, 30, 40, 50, 60, 70, 80, 90, 100

PENETRATION RESISTANCE BLOWS/FOOT



N: 17, 125, 39, 125, 43, 87, 79, 73, 58, 62, 57, 60, 45, 48, 62, 59, 58, 63, 67, 69, 73

LAB TESTS

FIGURE 2.7-12

GRAPHIC BORING LOG

BALTIMORE GAS AND ELECTRIC COMPANY SUPPLEMENTAL SITE BORINGS CALVERT CLIFFS NUCLEAR POWER PLANT

BORING NO. 218

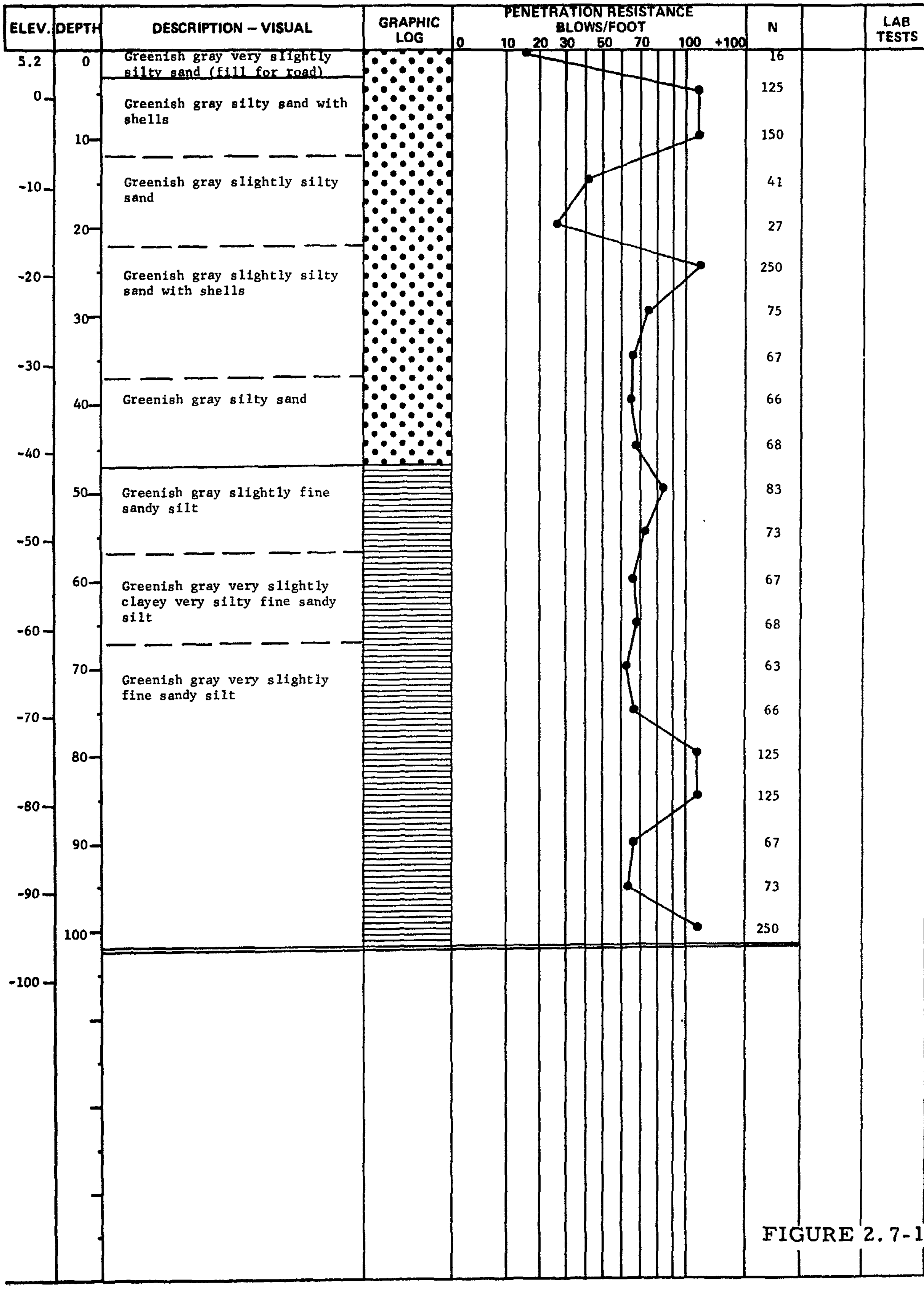


FIGURE 2.7-13

BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 219

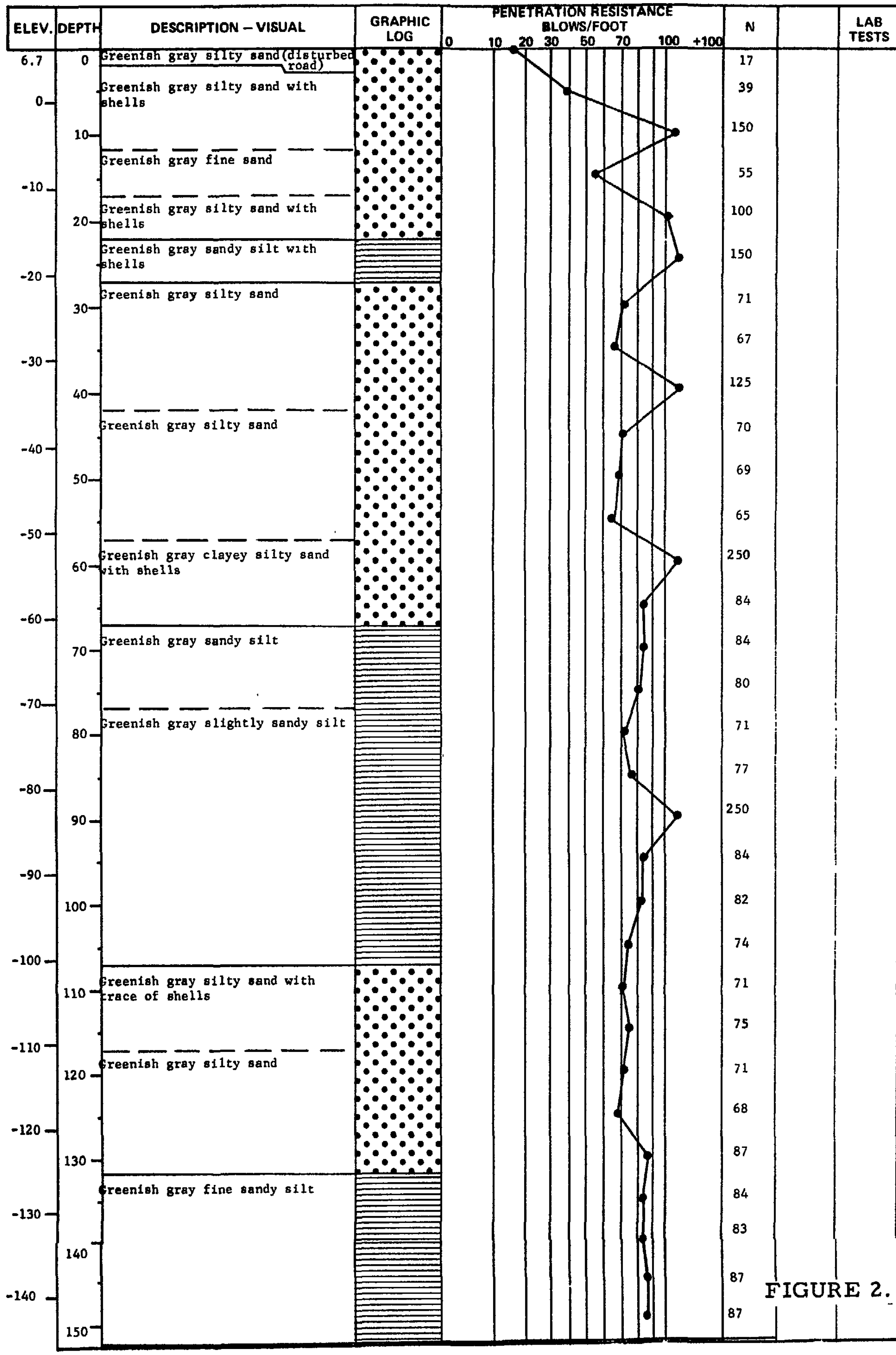


FIGURE 2.7-14

2.7-15 GRAPHIC BORING LOGS, BALTIMORE GAS AND ELECTRIC COMPANY, CALVERT CLIFFS NUCLEAR POWER PLANT, SUPPLEMENTARY SITE BORINGS

GRAPHIC BORING LOG

BALTIMORE GAS AND ELECTRIC COMPANY | **SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 220

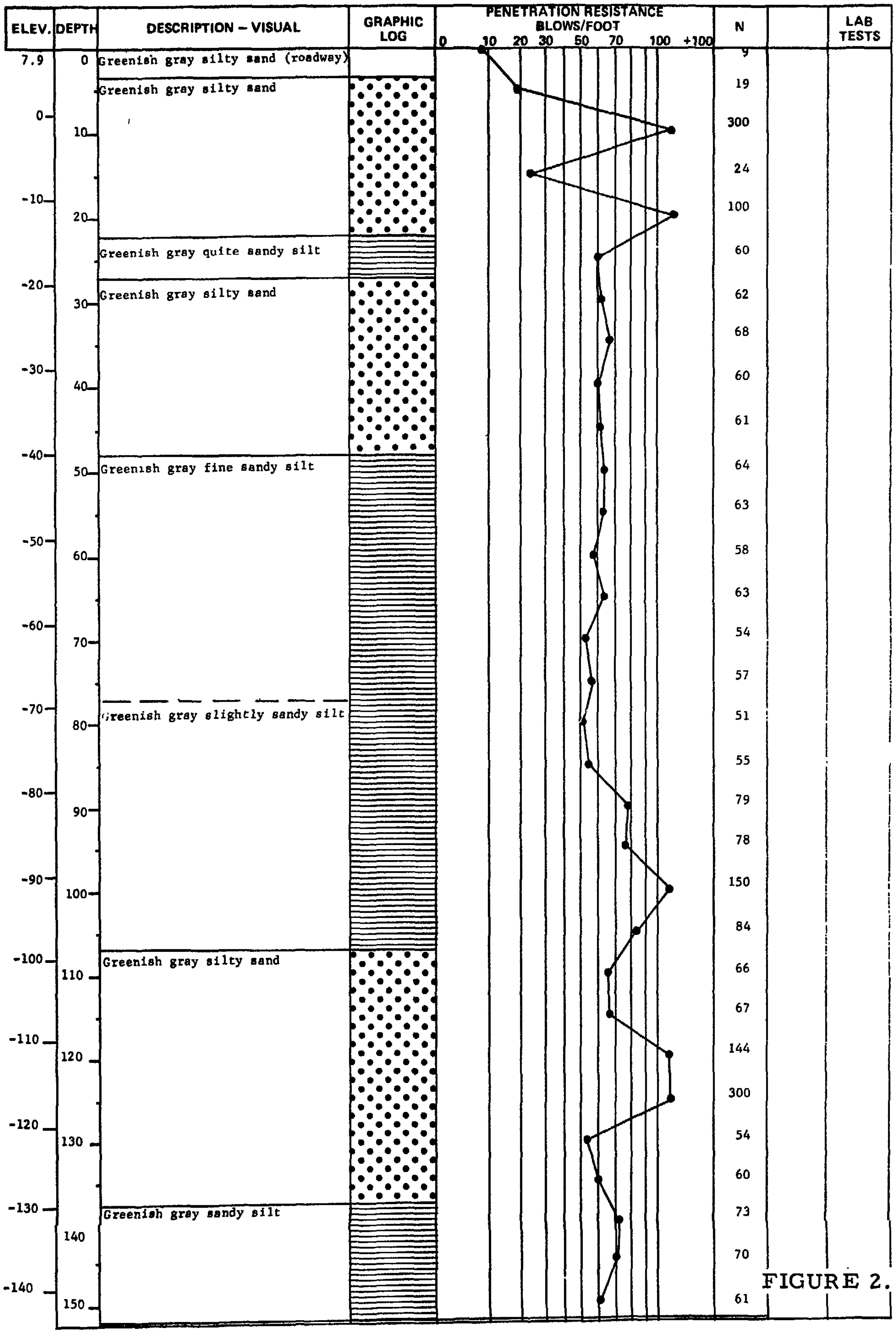


FIGURE 2.7-15

BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 221

PENETRATION RESISTANCE BLOWS/FOOT: 0 10 20 30 50 70 100 +100

ELEV.	DEPTH	DESCRIPTION – VISUAL	N	LAB TESTS
7.0	0	Greenish gray silty sand	9	
			29	
0				
	10	Greenish gray silty sand with shells	125	
			27	
-10				
	20		125	
		Greenish gray fine sandy clayey silt with shells	85	
-20				
	30		51	
			125	
-30				
	40		140	
		Greenish gray fine sandy silt with shells	125	
-40				
	50	Greenish gray fine sandy silt	200	
			144	
-50				
	60		44	
		Greenish gray silty sand	53	
-60				
	70		60	
		Greenish gray slightly sandy silt	66	
-70				
	80		72	
			66	
-80				
	90		67	
			73	
-90				
	100		79	
			79	
-100		Greenish gray silty sand		
	110		144	
			300	
-110				
	120		82	
		Greenish gray silty fine sand	86	
-120				
	130	Greenish gray slightly fine sandy silt	200	
			300	
-130				
			200	
	140	Greenish gray slightly fine sandy silt		
			500	
-140				
	150		144	

FIGURE 2.7-16

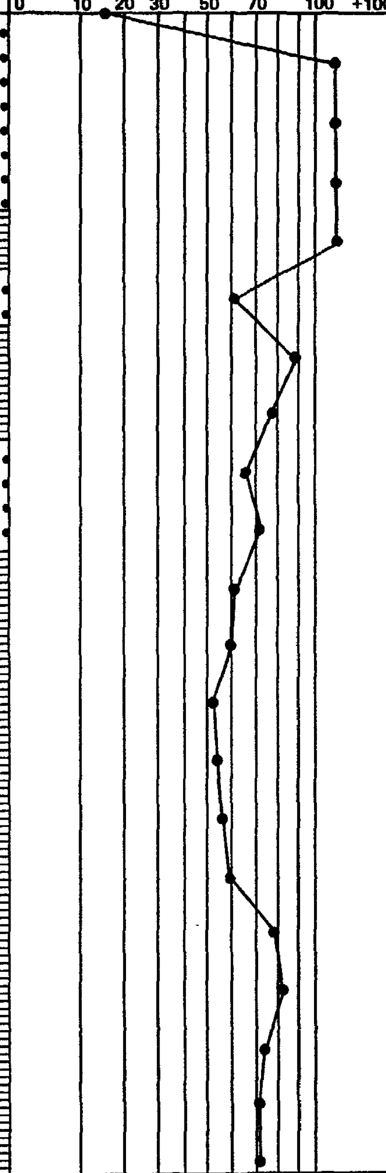
BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 222

ELEV. 1.5 (Depth 0) to -100 (Depth 100)

DESCRIPTION – VISUAL
Greenish gray silty sand with shells
Greenish gray slightly clayey silty sand with shells
Greenish gray slightly silty sand with shells
Greenish gray fine sandy silt with trace shells
Greenish gray silty sand
Greenish gray sandy silt with slight trace shells
Greenish gray silty sand with shells
Greenish gray sandy silt
Greenish gray slightly clayey fine sandy silt
Greenish gray sandy silt

PENETRATION RESISTANCE BLOWS/FOOT



N
16
150
170
150
150
61
89
79
67
71
61
60
53
54
56
59
79
82
74
71
71

LAB TESTS

FIGURE 2.7-17

BALTIMORE GAS AND ELECTRIC COMPANY | **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** | **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 223

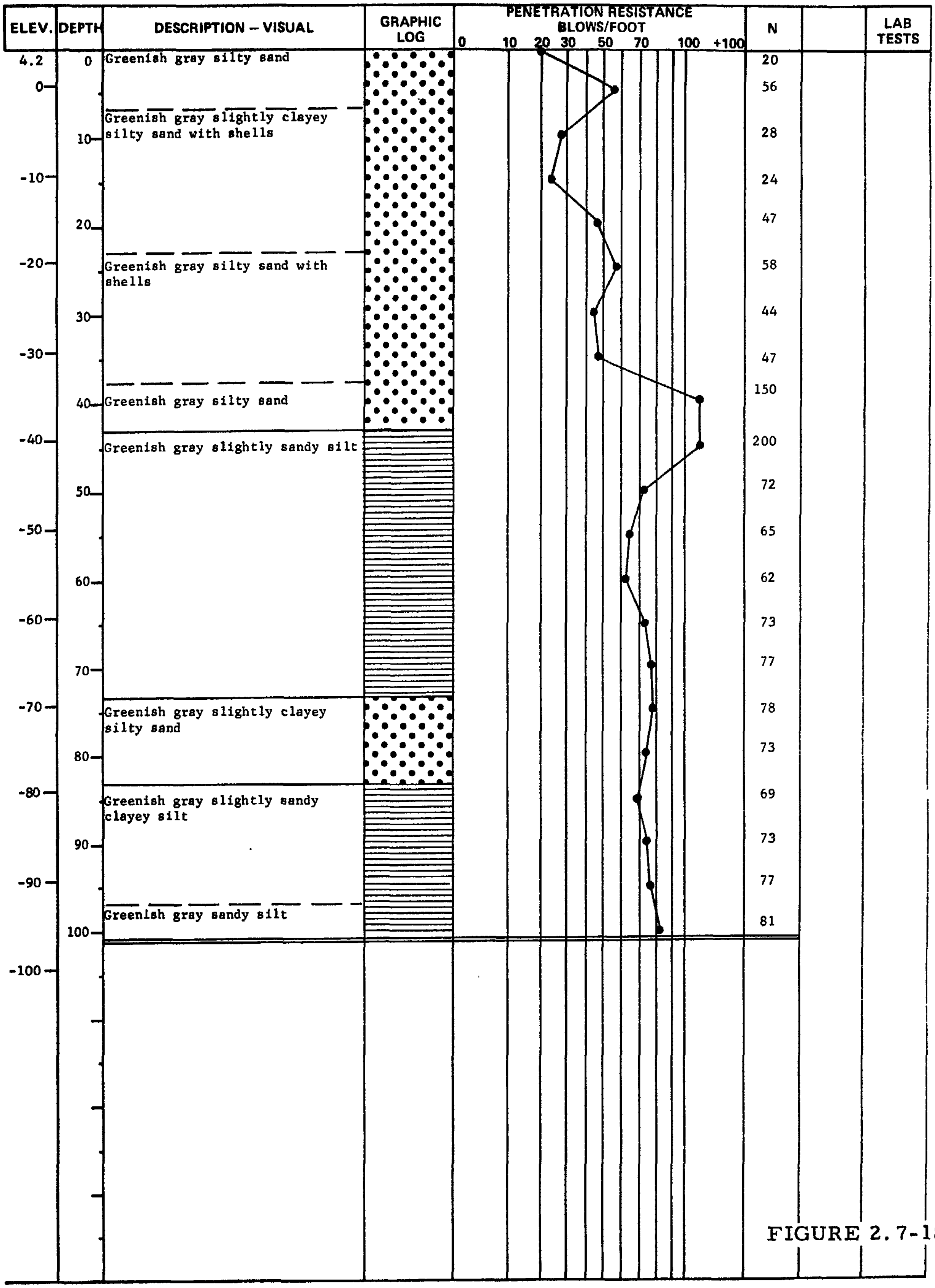


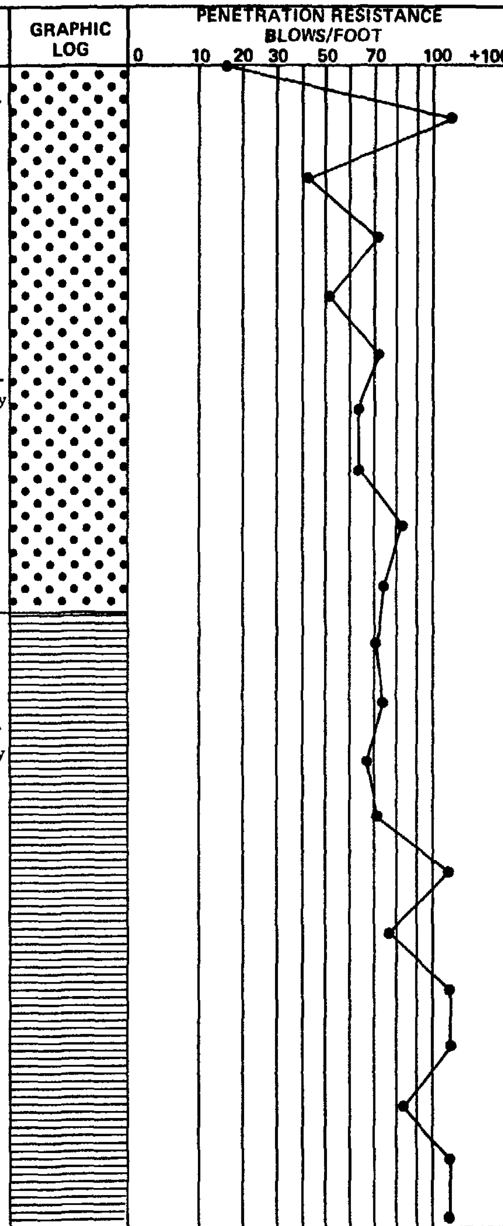
FIGURE 2.7-18

BALTIMORE GAS AND ELECTRIC COMPANY — **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** — **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 224

ELEV. 3.7 at DEPTH 0

DESCRIPTION – VISUAL
Greenish gray silty sand
Greenish gray silty sand with shells
Greenish gray very slightly clayey silty sand with trace of shells
Greenish gray fine sandy silt
Greenish gray very slightly clayey sandy silt
Greenish gray sandy silt



N
17
150
42
71
51
71
64
64
82
74
70
74
68
71
150
77
200
150
83
150
250

LAB TESTS

ELEV. scale: 0, -10, -20, -30, -40, -50, -60, -70, -80, -90, -100

DEPTH scale: 0, 10, 20, 30, 40, 50, 60, 70, 80, 90, 100

FIGURE 2.7-19

2.7-20 GRAPHIC BORING LOGS, BALTIMORE GAS AND ELECTRIC COMPANY, CALVERT CLIFFS NUCLEAR POWER PLANT, SUPPLEMENTARY SITE BORINGS

GRAPHIC BORING LOG

BALTIMORE GAS AND ELECTRIC COMPANY **SUPPLEMENTAL SITE BORINGS** **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 226

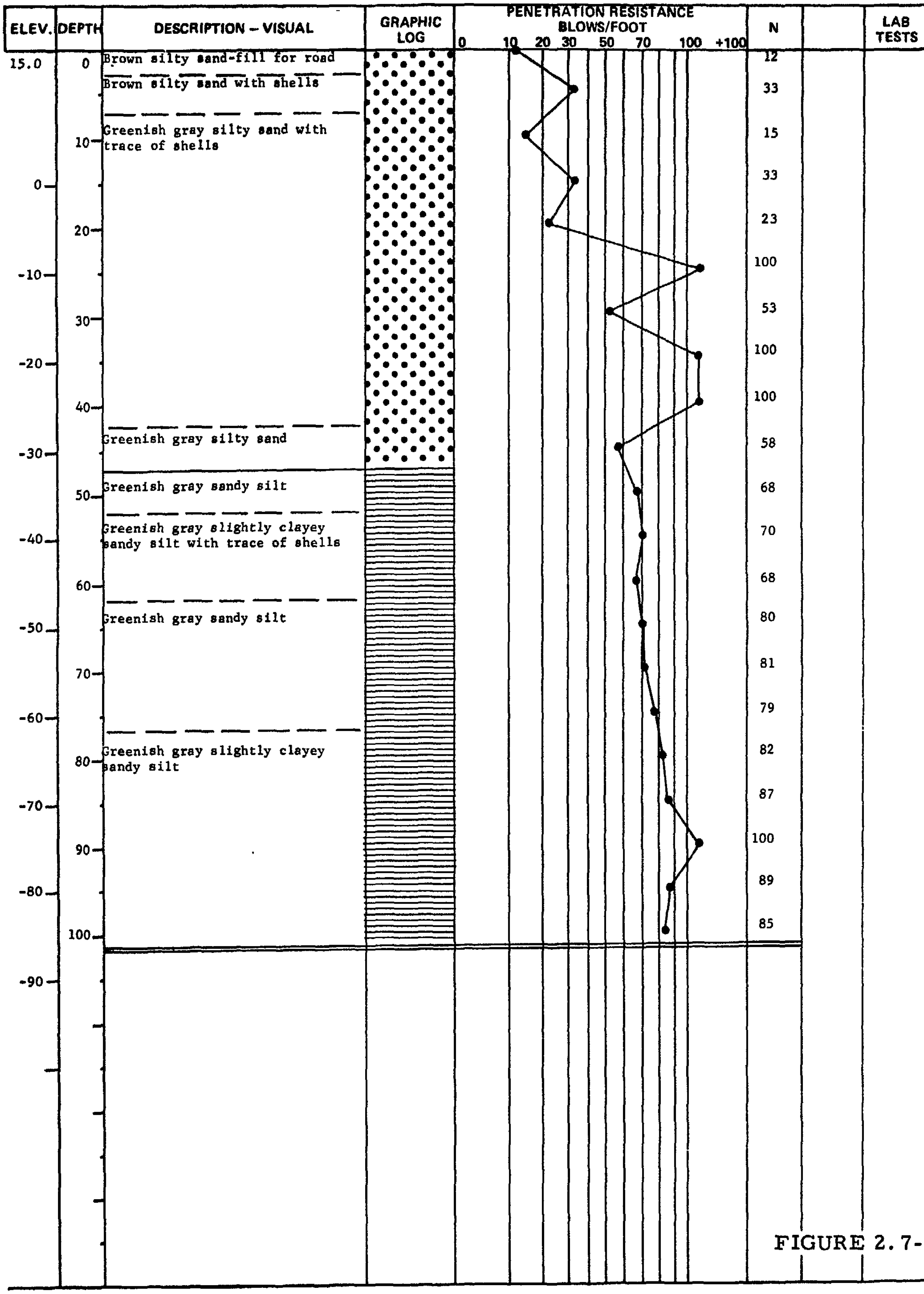


FIGURE 2.7-20

BALTIMORE GAS AND ELECTRIC COMPANY — **GRAPHIC BORING LOG SUPPLEMENTAL SITE BORINGS** — **CALVERT CLIFFS NUCLEAR POWER PLANT**

BORING NO. 227

ELEV.	DEPTH	DESCRIPTION – VISUAL	GRAPHIC LOG	PENETRATION RESISTANCE BLOWS/FOOT (0 10 20 30 50 70 100 +100)	N	LAB TESTS
2.0	0	Brown slightly silty sand with trace shells (beach sand)			21	
0		Greenish gray silty sand			18	
	10	Greenish gray slightly clayey silty sand with trace of shells			13	
-10					100	
	20				41	
-20					100	
	30	Greenish gray slightly clayey fine sandy silt			64	
-30					68	
	40	Greenish gray quite silty sand			50	
-40					57	
	50	Greenish gray fine sandy silt			70	
-50					75	
	60				81	
-60		Greenish gray very slightly clayey sandy silt			82	
	70				86	
-70						
	80					

FIGURE 2.7-21

GRAPHIC BORING LOG

BALTIMORE GAS AND ELECTRIC COMPANY — SUPPLEMENTAL SITE BORINGS — CALVERT CLIFFS NUCLEAR POWER PLANT

BORING NO. 228

ELEV.	DEPTH	DESCRIPTION – VISUAL	N
1.5	0	Brown medium sand (beach sand)	14
		Greenish gray slightly clayey silty sand with shells	13
			100
		Greenish gray silty sand with shells	100
	20	Greenish gray silty sand	69
			62
			65
			63
	40	Greenish gray slightly clayey sandy silt	43
			51
			79
			78
			79
			81
	70		100

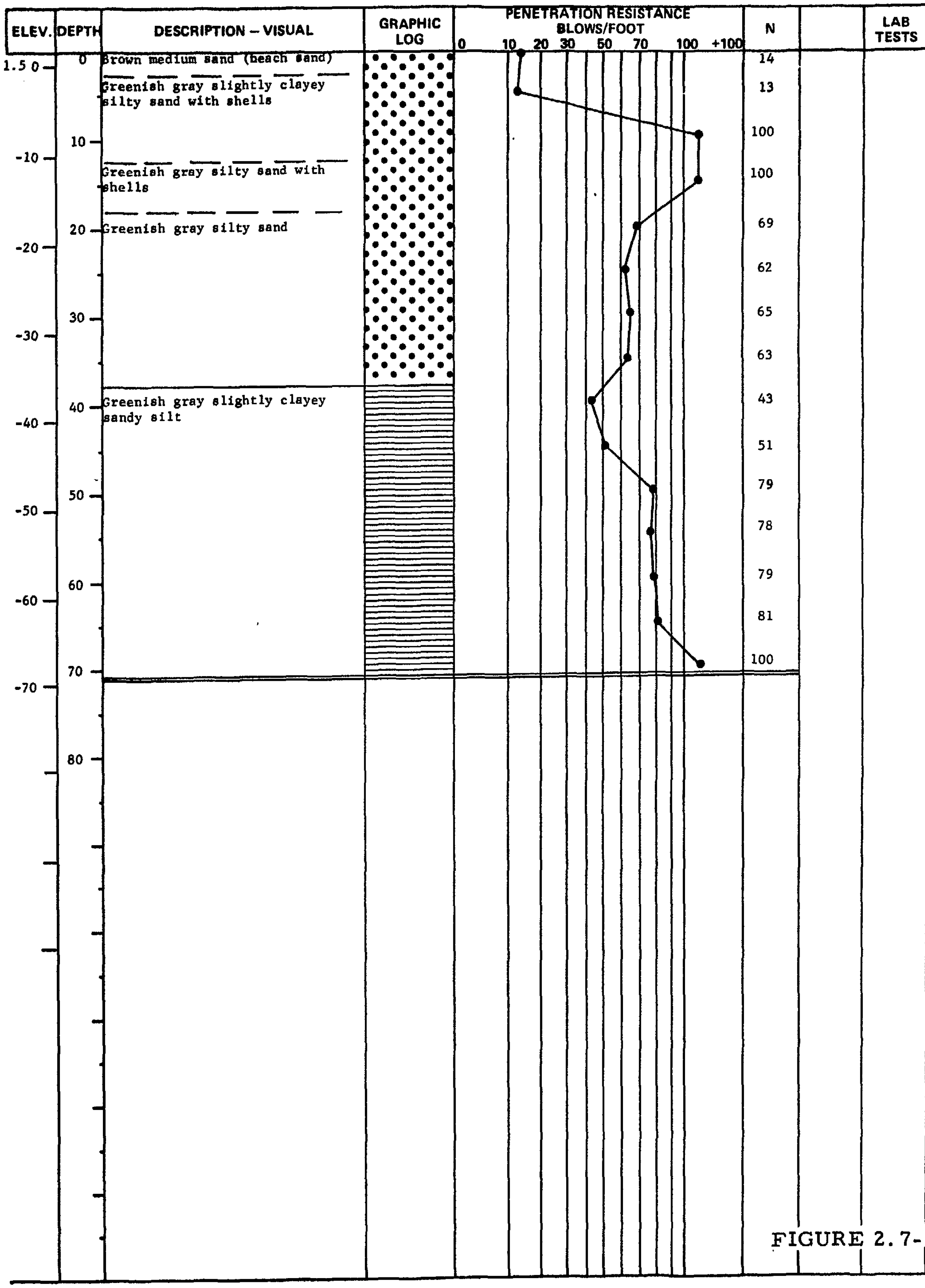
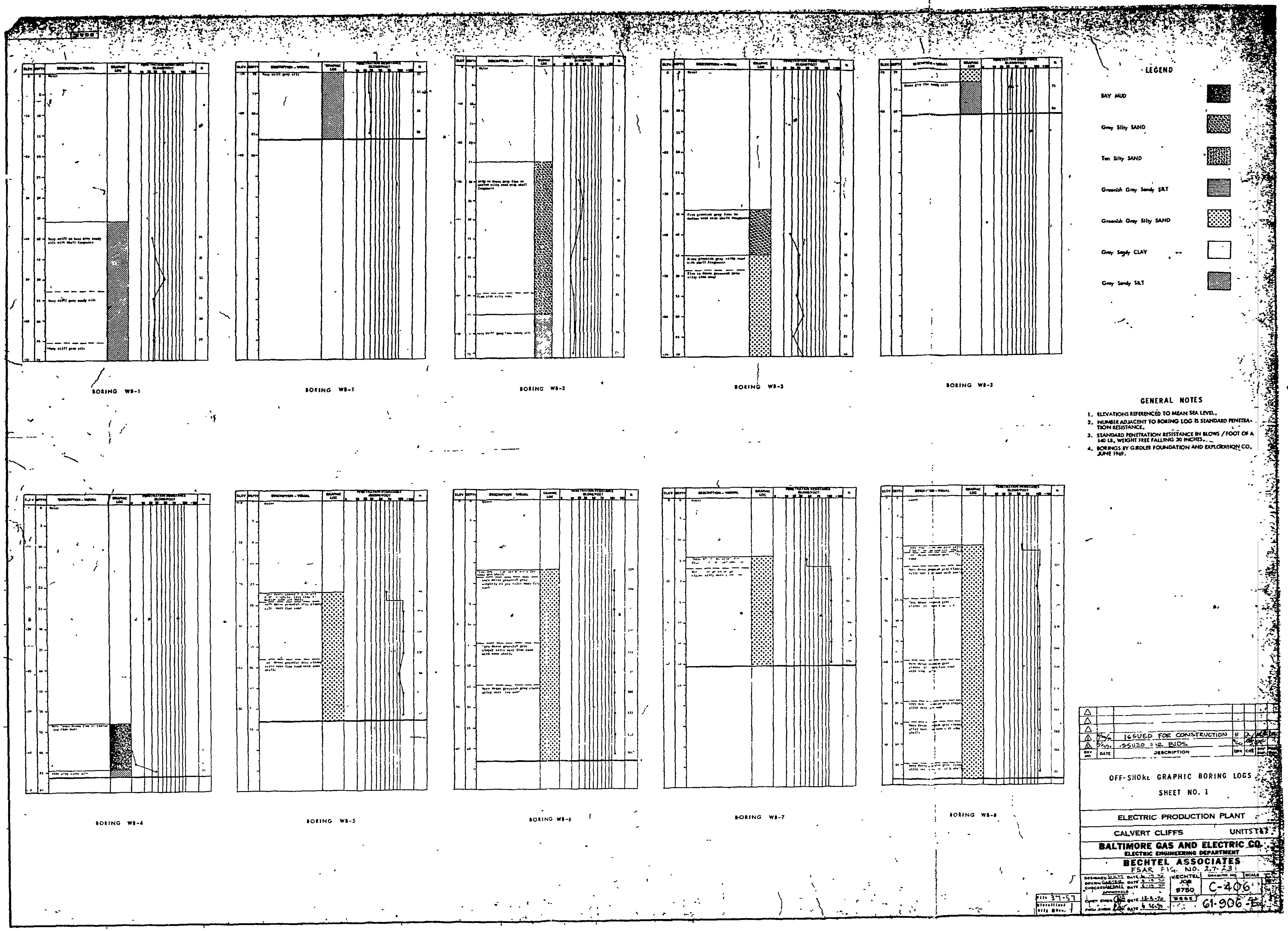


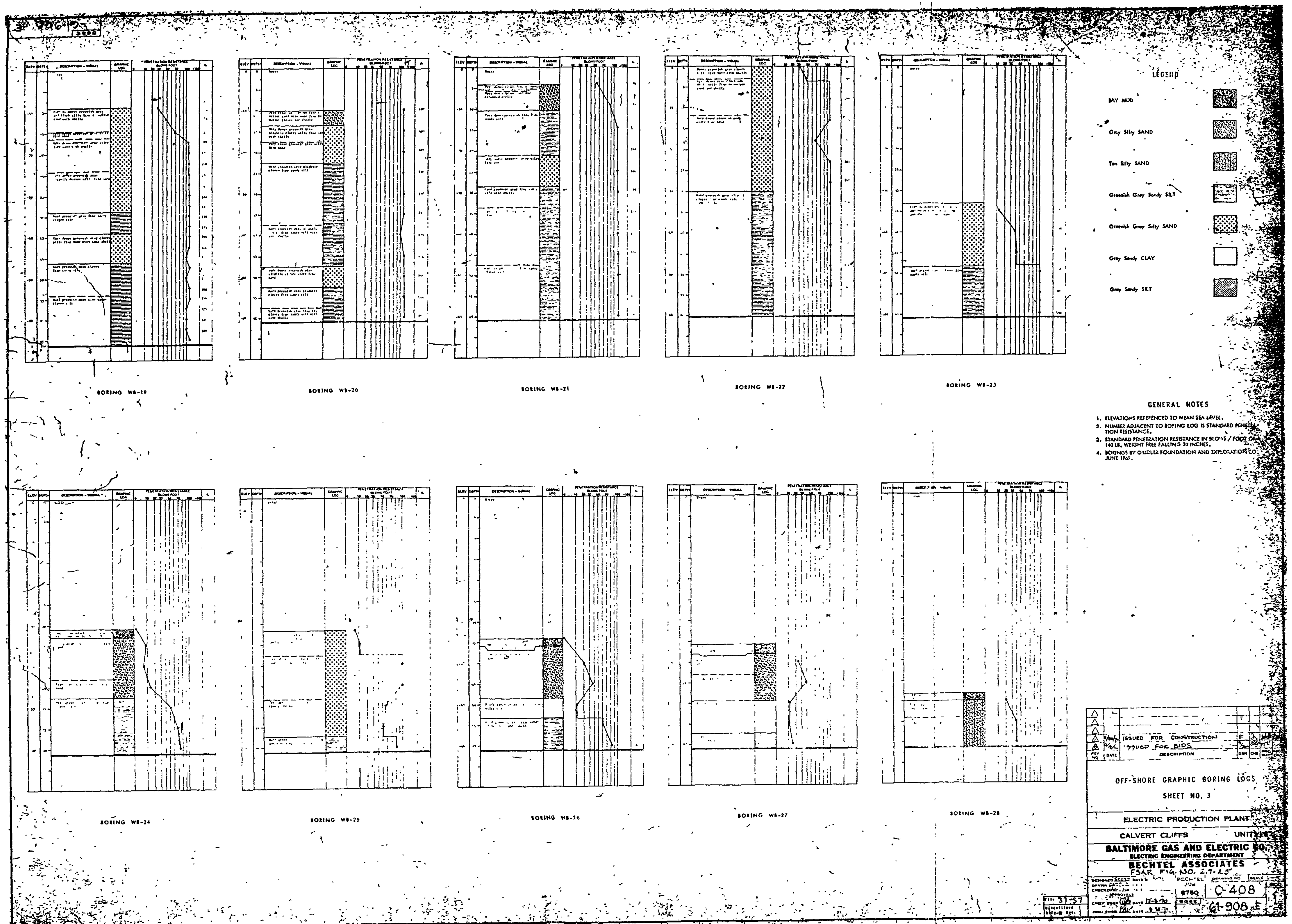
FIGURE 2.7-22

2.7-23 OFF-SHORE GRAPHIC BORING LOGS (Sheet 1)

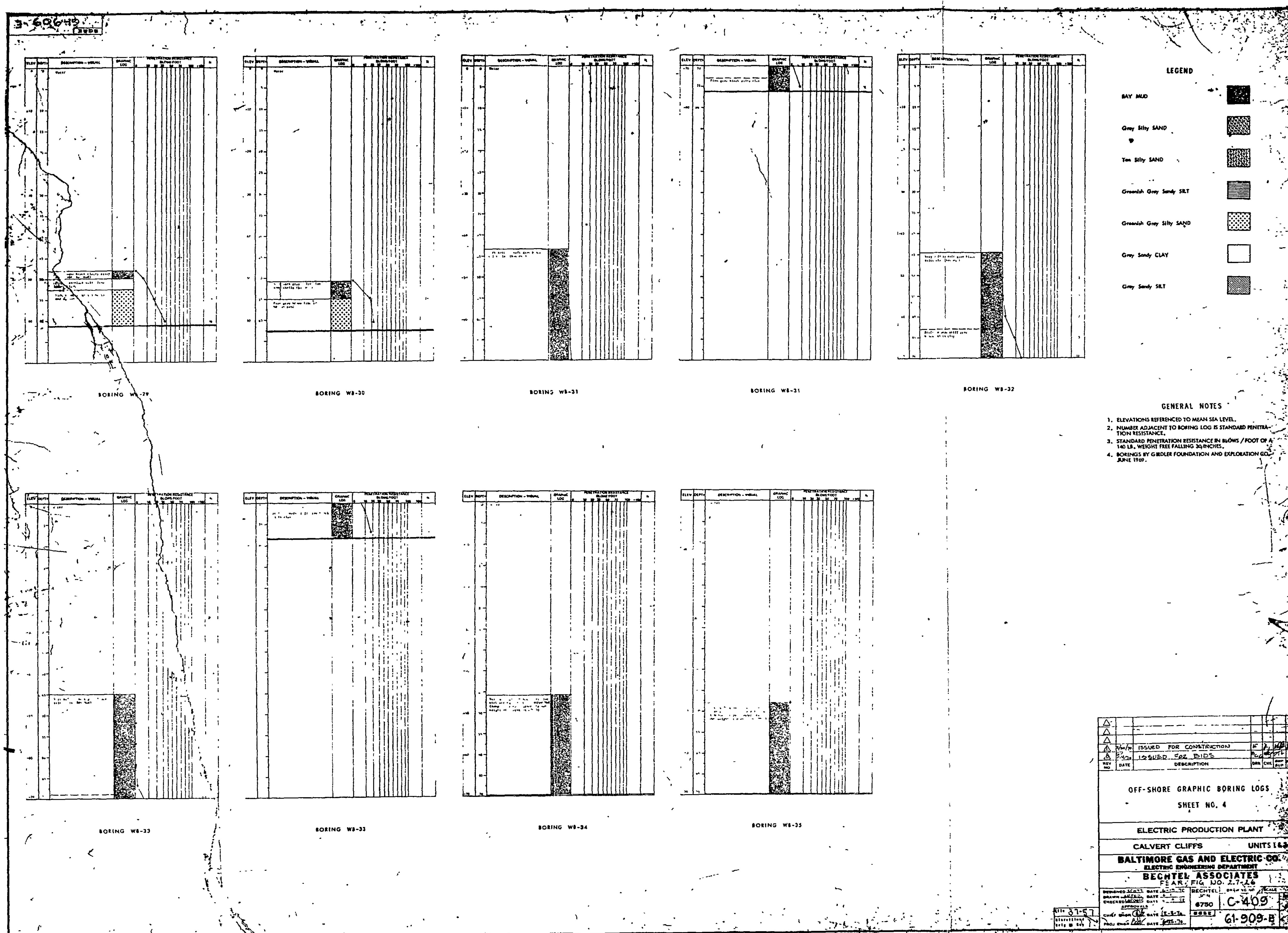


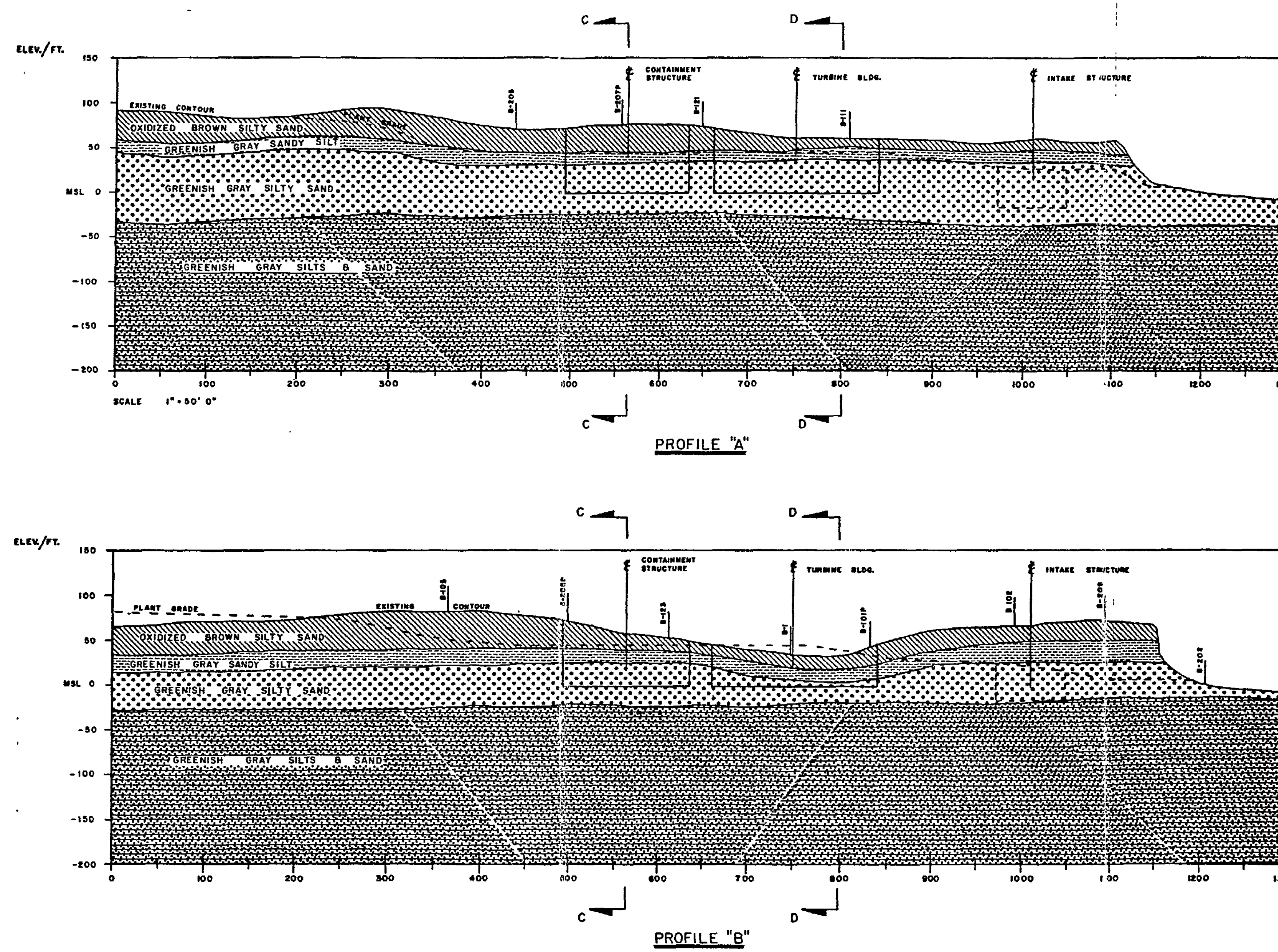

BORING WB-9
BORING WB-10
BORING WB-11
BORING WB-12
BORING WB-13
BORING WB-14
BORING WB-15
BORING WB-16
BORING WB-17
BORING WB-18
LEGEND
BAY MUD
Gray Silty SAND
Tan Silty SAND
Greenish Gray Sandy SILT
Greenish Gray Silty SAND
Gray Sandy CLAY
Gray Sandy SILT
GENERAL NOTES
1. ELEVATIONS REFERENCED TO MEAN SEA LEVEL.
2. NUMBER ADJACENT TO BORING LOG IS STANDARD PENETRATION RESISTANCE.
3. STANDARD PENETRATION PESISTANCE IN BLOWS / FOOT OF A 140 LB. WEIGHT FREE FALLING 30 INCHES.
4. BORINGS BY GIDDLER FOUNDATION AND EXPLORATION CO. JUNE 1969.
ISSUED FOR CONSTRUCTION
ISSUED FOR BIDS
REV NO
DATE
DESCRIPTION
OFF-SHORE GRAPHIC BORING LOGS
SHEET NO. 2
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS
UNITS
BALTIMORE GAS AND ELECTRIC CO.
ELECTRIC ENGINEERING DEPARTMENT
BECHTEL ASSOCIATES
FSAR FIG NO. 2.7-24
BECHTEL JOB 6780
DRAWING NO.
C-407
61-907
File 37-53

2.7-25 OFF-SHORE GRAPHIC BORING LOGS (Sheet 3)



Rev.0


BORING WB-29
BORING WB-30
BORING WB-31
BORING WB-31
BORING WB-32
BORING WB-33
BORING WB-33
BORING WB-34
BORING WB-35
LEGEND
BAY MUD
Grey Silty SAND
Tan Silty SAND
Greenish Grey Sandy SILT
Greenish Grey Silty SAND
Grey Sandy CLAY
Grey Sandy SILT
GENERAL NOTES
1. ELEVATIONS REFERENCED TO MEAN SEA LEVEL.
2. NUMBER ADJACENT TO BORING LOG IS STANDARD PENETRATION RESISTANCE.
3. STANDARD PENETRATION RESISTANCE IN BLOWS / FOOT OF A 140 LB. WEIGHT FREE FALLING 30 INCHES.
4. BORINGS BY GIRDLER FOUNDATION AND EXPLORATION CO. JUNE 1969.
ISSUED FOR CONSTRUCTION
ISSUED FOR BIDS
REV NO
DATE
DESCRIPTION
DRN
CHK
OFF-SHORE GRAPHIC BORING LOGS
SHEET NO. 4
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS
BALTIMORE GAS AND ELECTRIC CO.
ELECTRIC ENGINEERING DEPARTMENT
BECHTEL ASSOCIATES
FSAR, FIG NO. 2.7-26
BECHTEL
6750
C-409
61-909-B
37-57

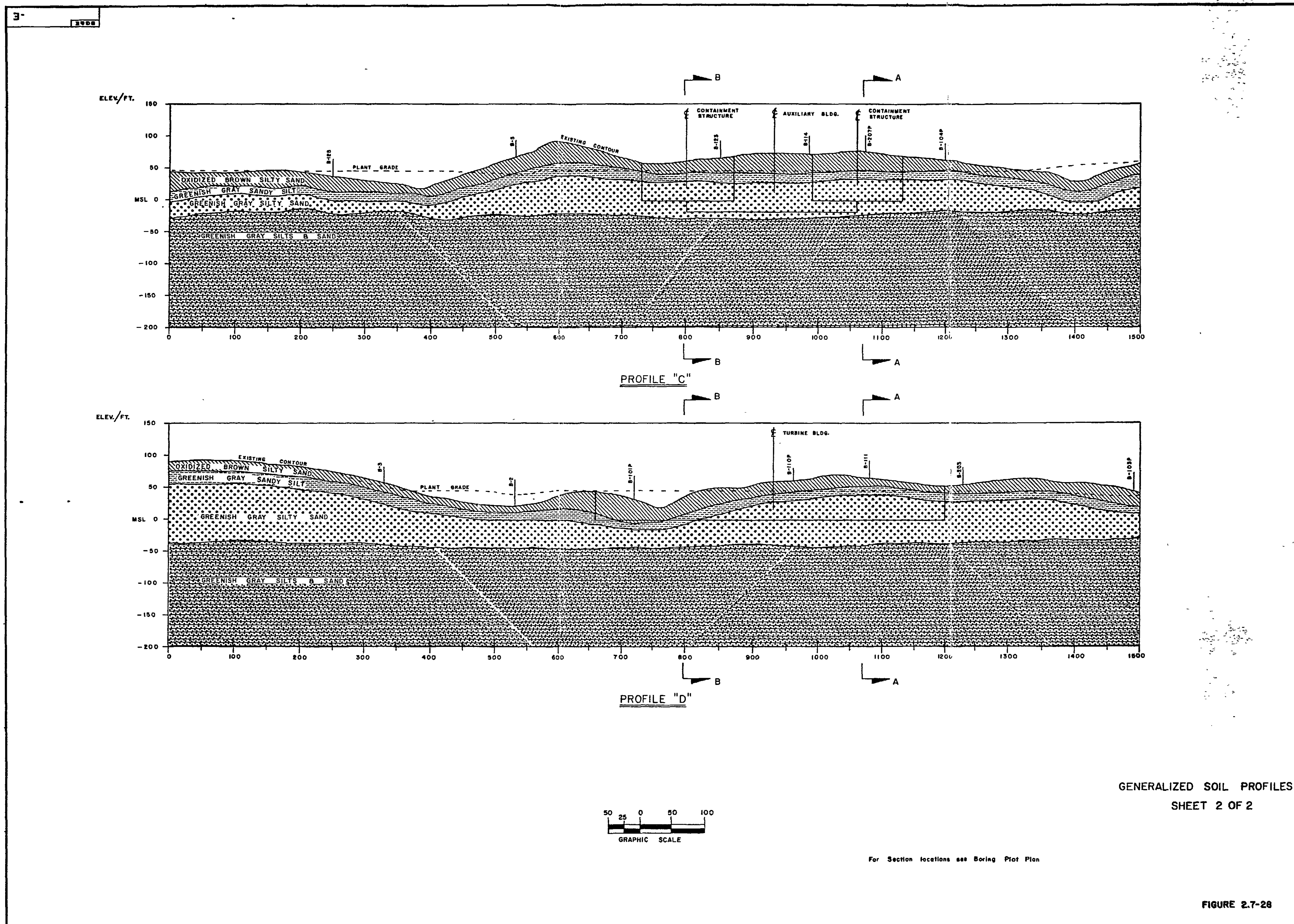


GENERALIZED SOIL PROFILES

SHEET 1 OF 2

For Section locations see Boring Plot Plan

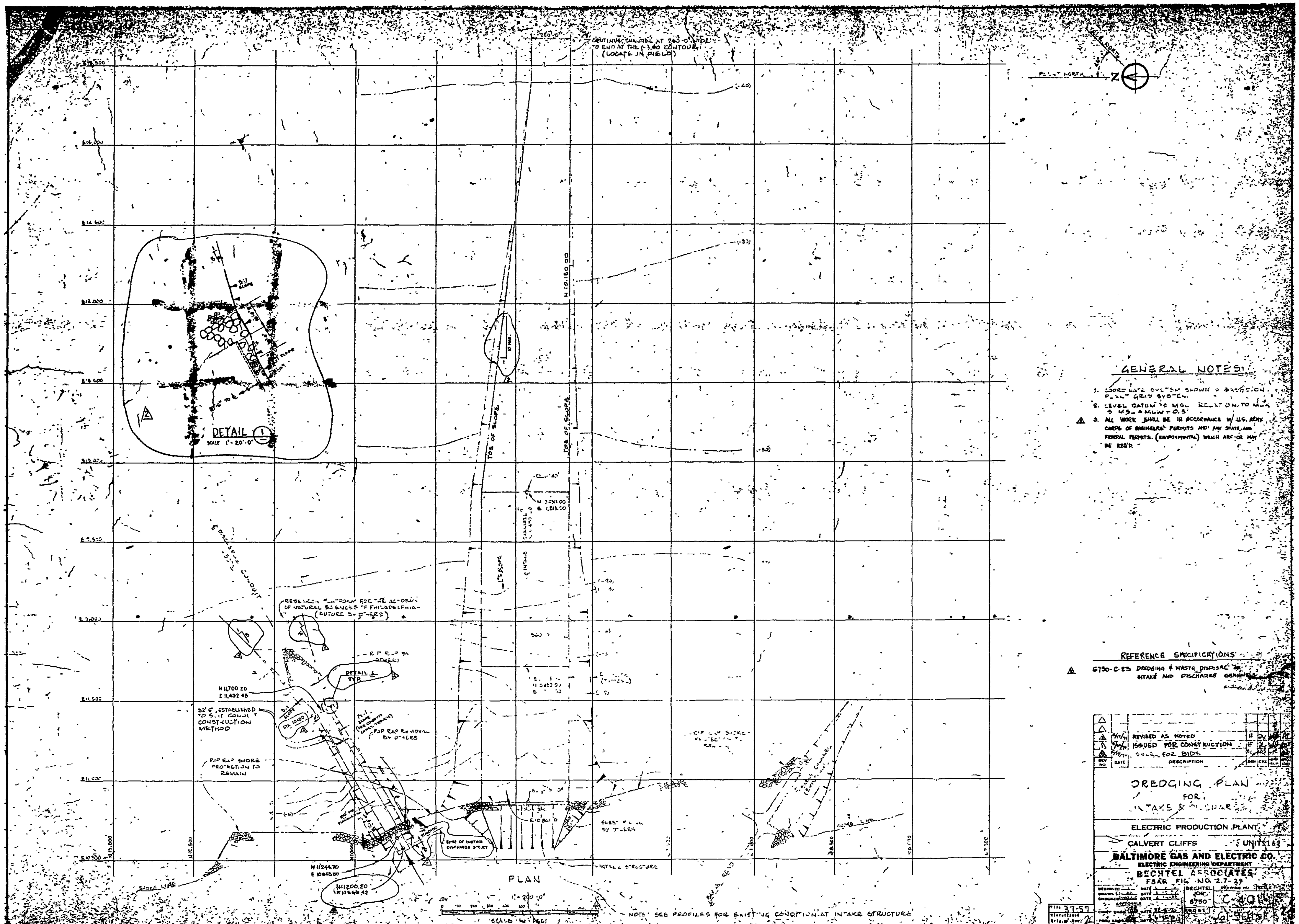
FIGURE 2.7-27

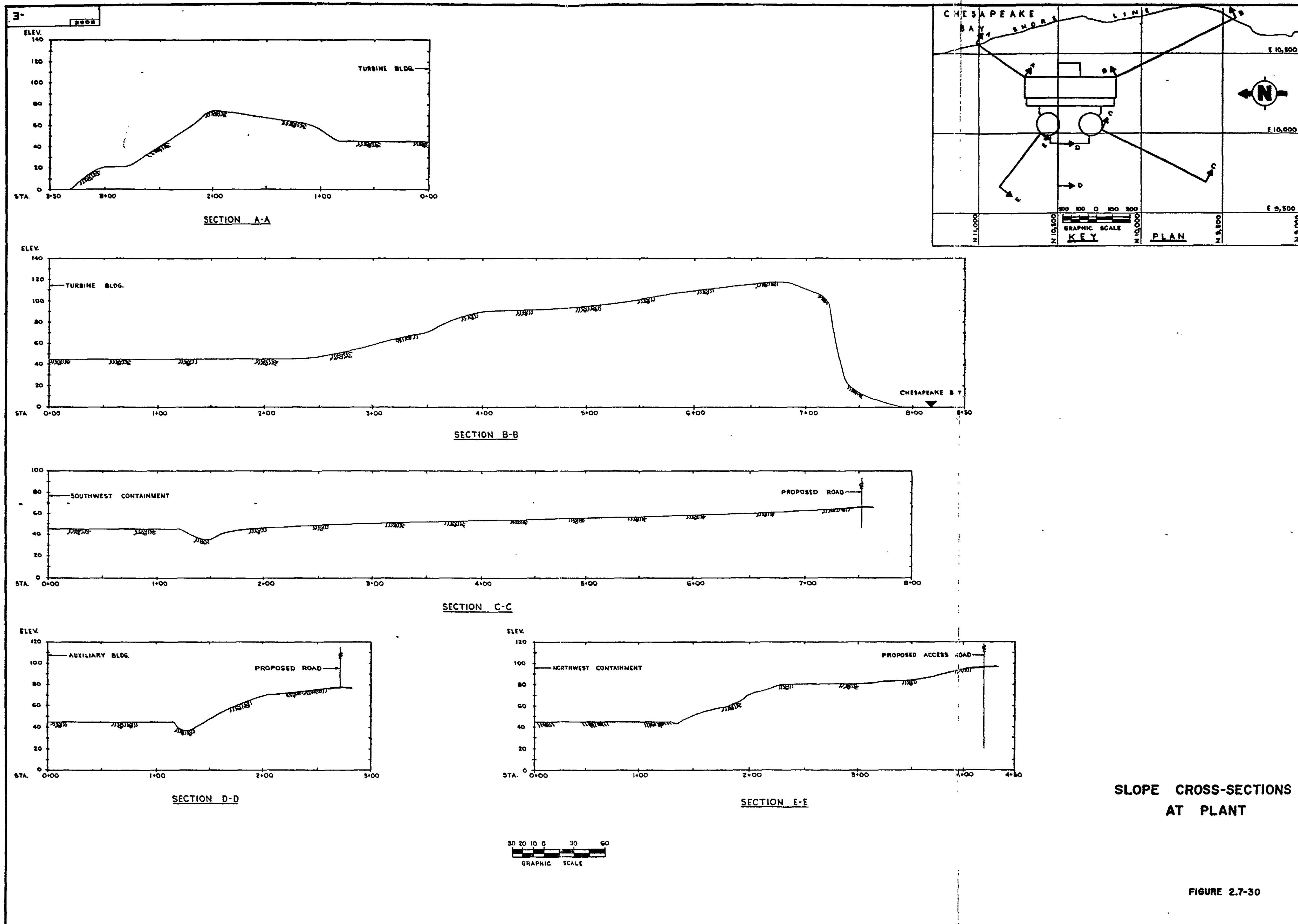


GENERALIZED SOIL PROFILES

SHEET 2 OF 2

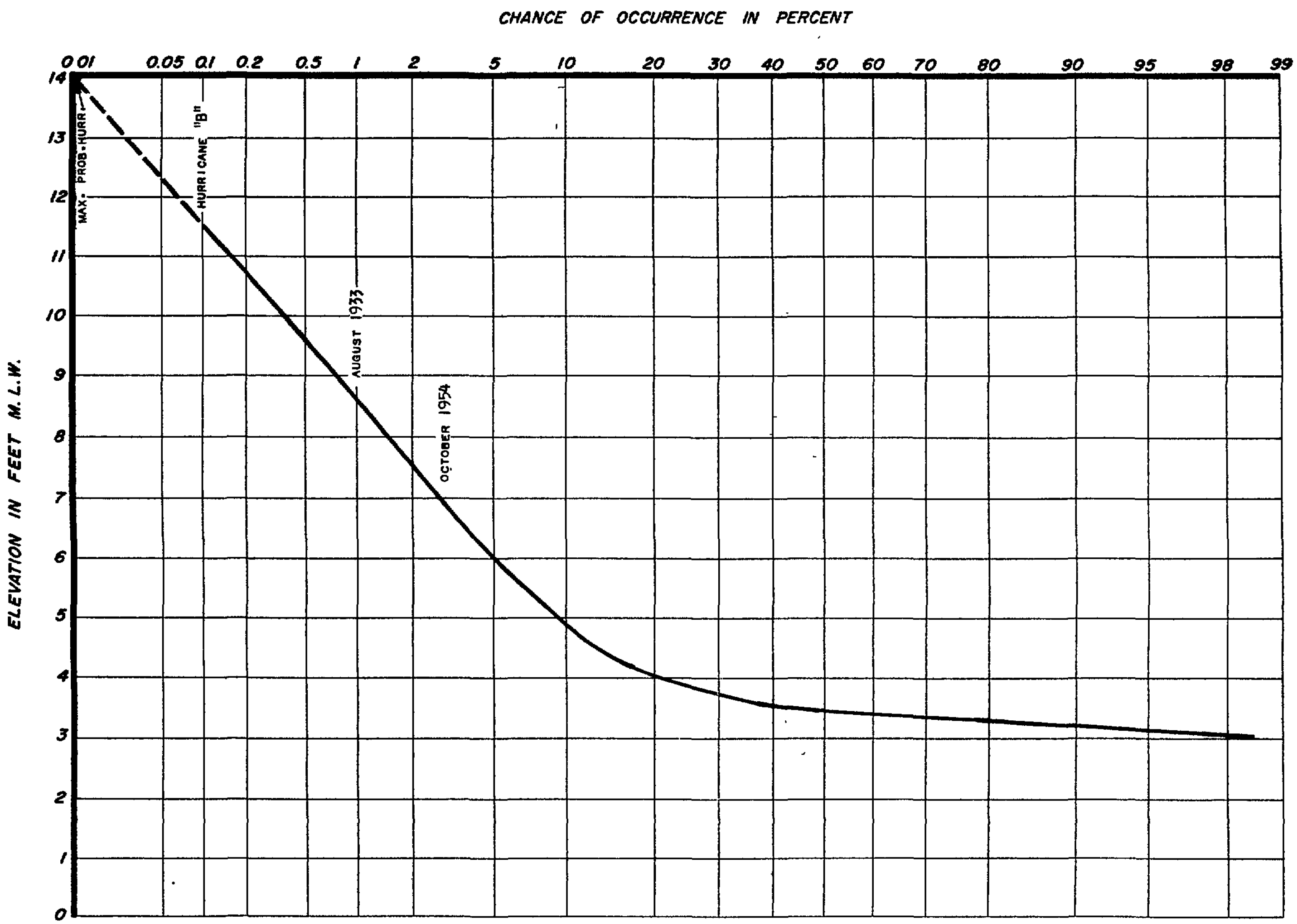
FIGURE 2.7-28


GENERAL NOTES:
REFERENCE SPECIFICATIONS
REVISED AS NOTED
ISSUED FOR CONSTRUCTION
DREDGING PLAN
FOR
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS
BALTIMORE GAS AND ELECTRIC CO.
BECHTEL ASSOCIATES
FSAR FIG. NO. 2.7-29
DETAIL
PLAN
TOP OF SLOPE
RIP RAP SHORE PROTECTION TO REMAIN
NOTE: SEE PROFILES FOR EXISTING CONDITION AT INTAKE STRUCTURE



SLOPE CROSS-SECTIONS AT PLANT

FIGURE 2.7-30



GENERALIZED TIME FREQUENCY CURVE

NOTE:
REPRODUCED FROM PLATE 3 OF H. DOC. NO. 350 88 CONGRESS, 2ND. SESS. "TIDEWATER PORTIONS OF PATUXENT, POTOMAC AND RAPPAHANNOCK RIVERS, INCLUDING ADJACENT CHESAPEAKE BAY SHORELINES"

DAMES & MOORE
APPLIED EARTH SCIENCES

FIGURE 2.8-1

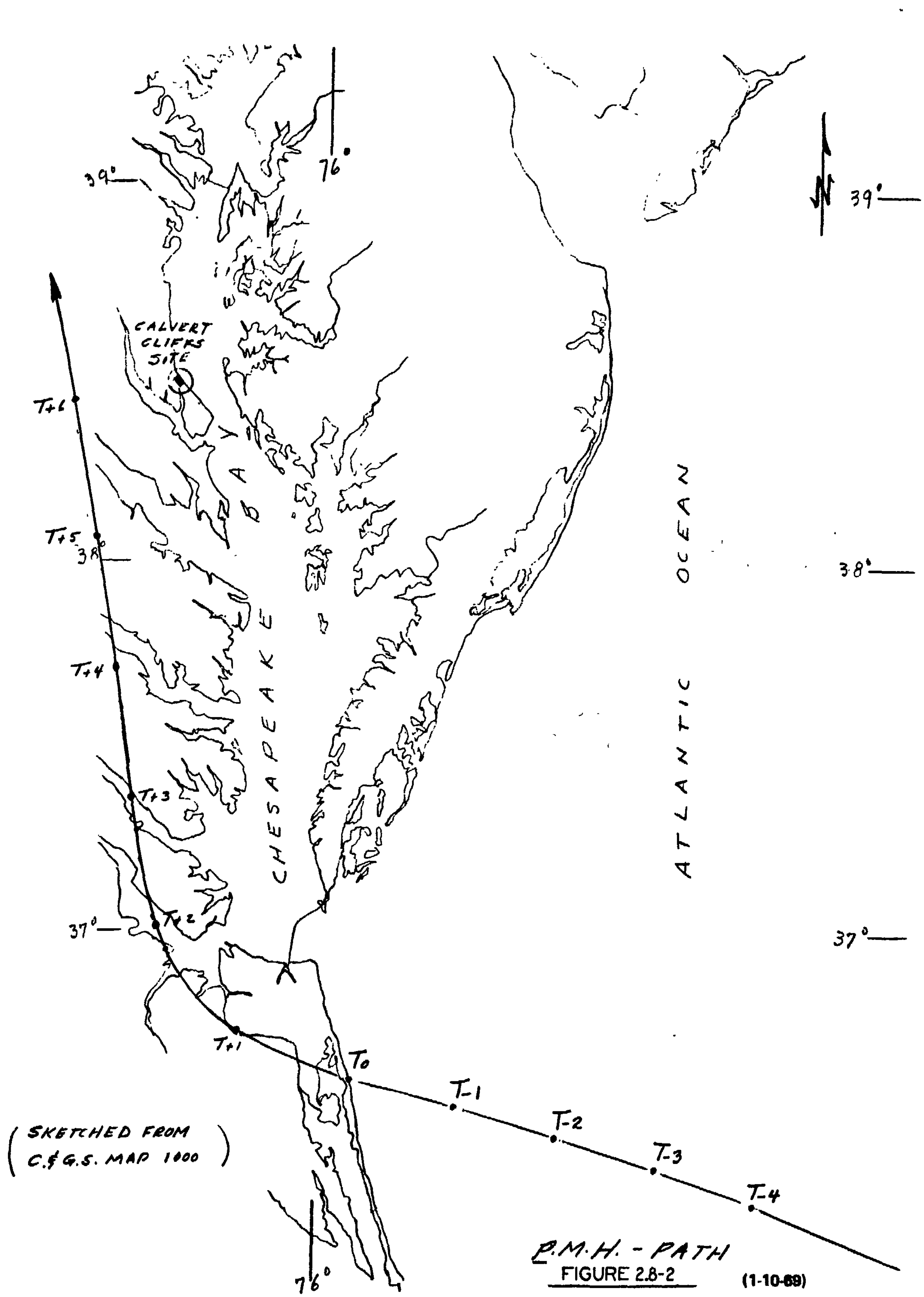


FIGURE 2.8-2 (1-10-69)

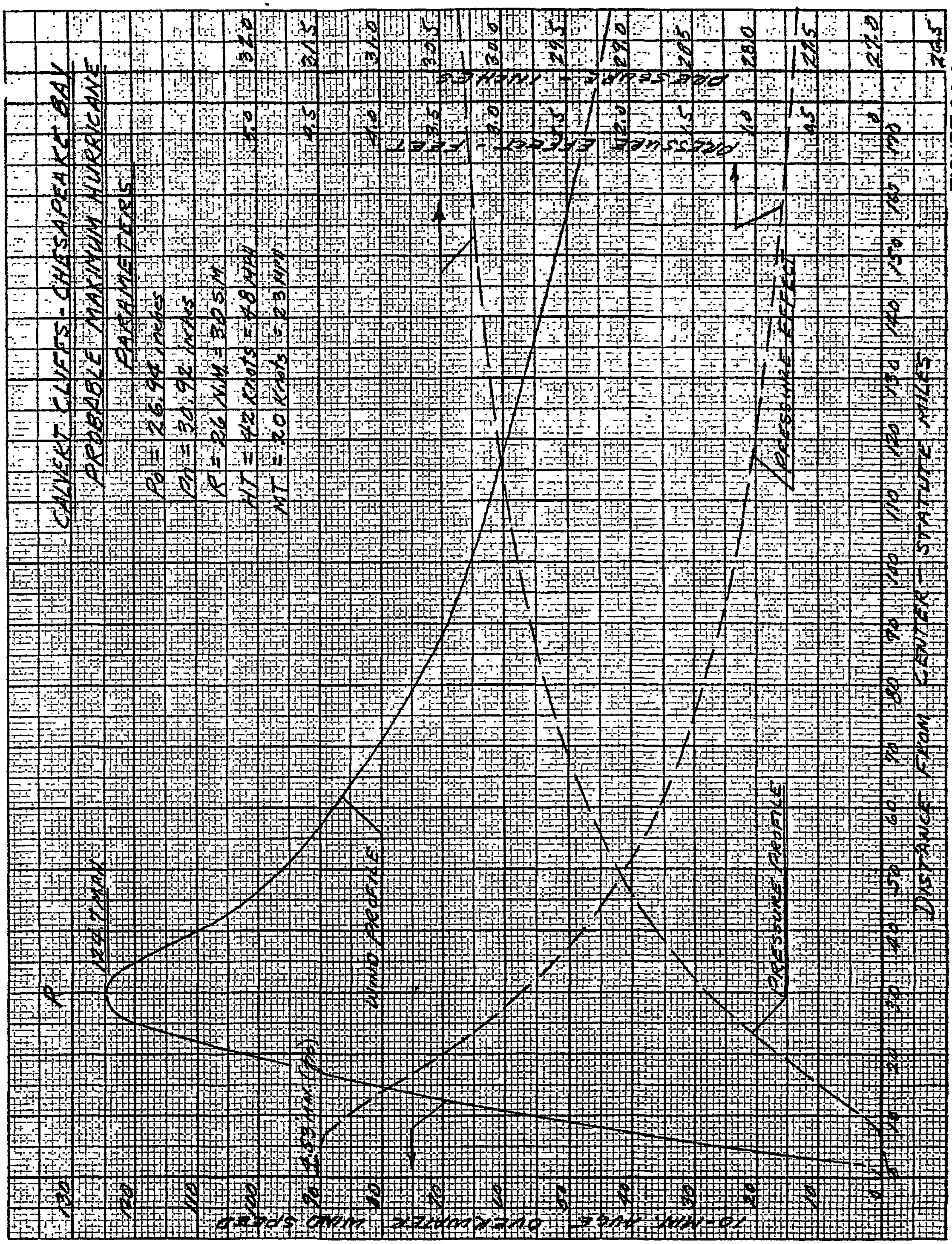

CALVERT CLIFFS - CHESAPEAKE BAY
PROBABLE MAXIMUM HURRICANE
PARAMETERS
WIND PROFILE
PRESSURE PROFILE
PRESSURE EFFECT
PRESSURE EFFECT - FEET
PRESSURE - INCHES
10-MIN. AVE. OVERWATER WIND SPEED
DISTANCE FROM CENTER - STATUTE MILES

FIGURE 2.8-3

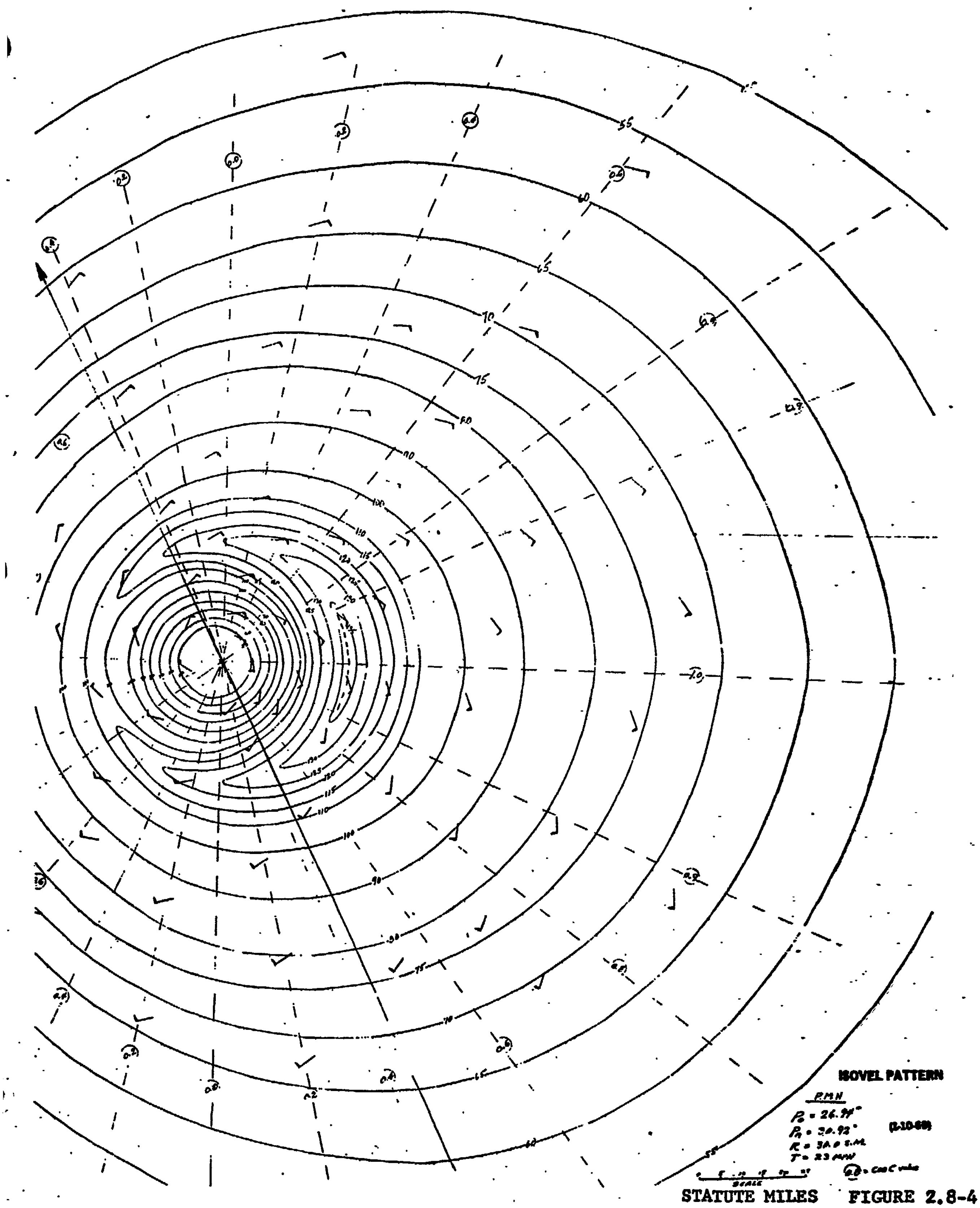
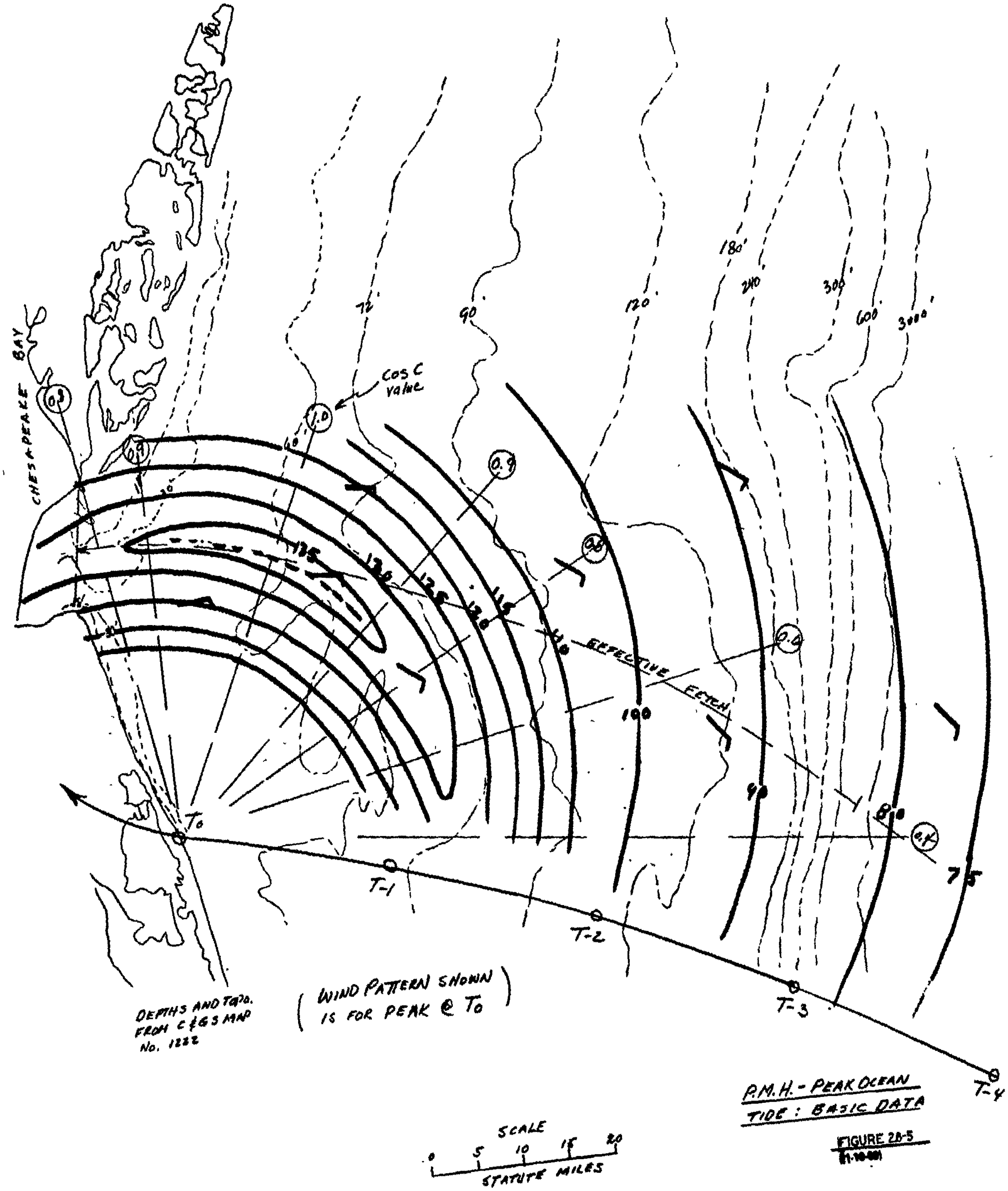


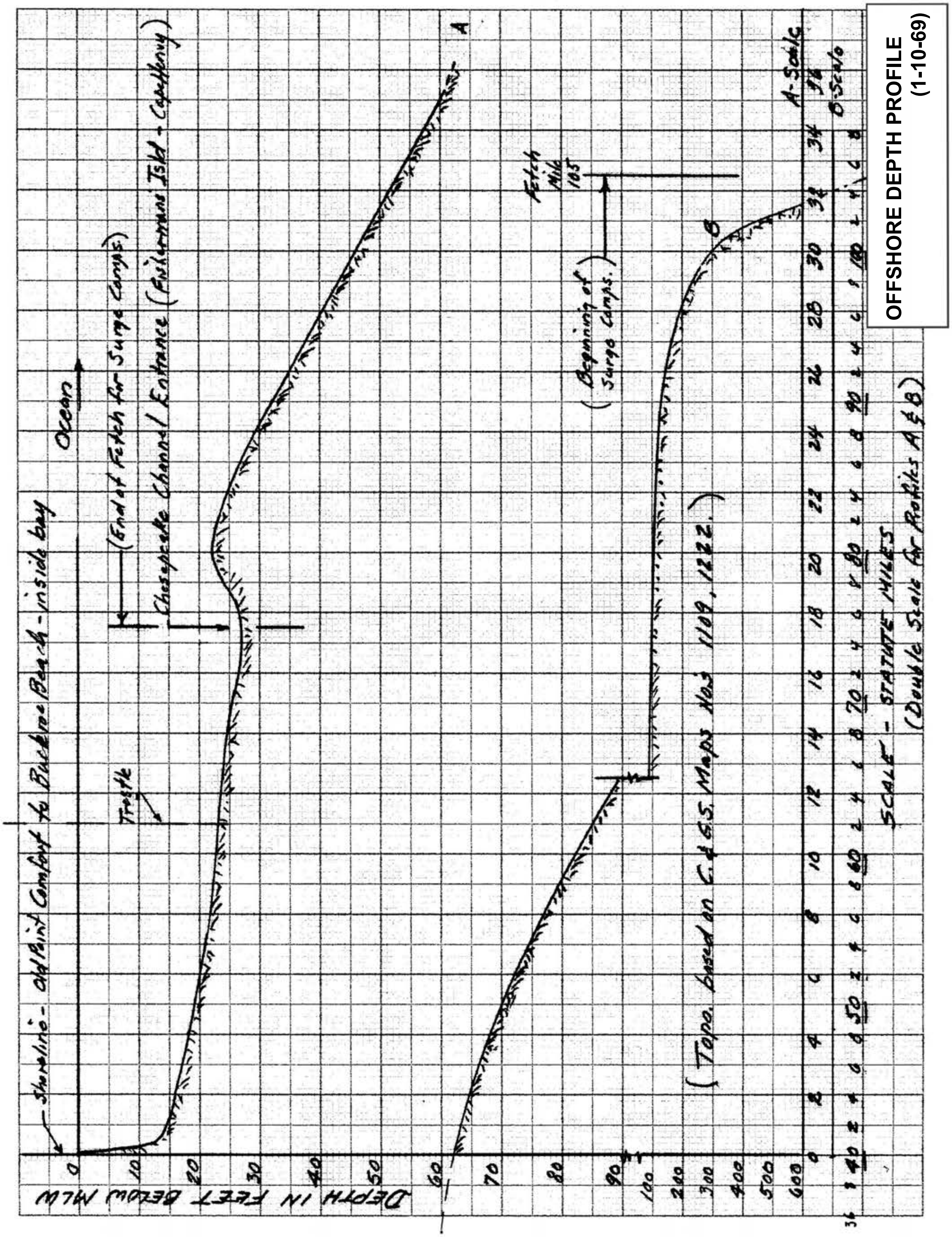
FIGURE 2.8-4



P.M.H. - PEAK OCEAN TIDE : BASIC DATA

FIGURE 2.8-5

FIGURE 2.8-6
OFFSHORE DEPTH PROFILE



2.8-7 WIND DIRECTION FOR FETCH

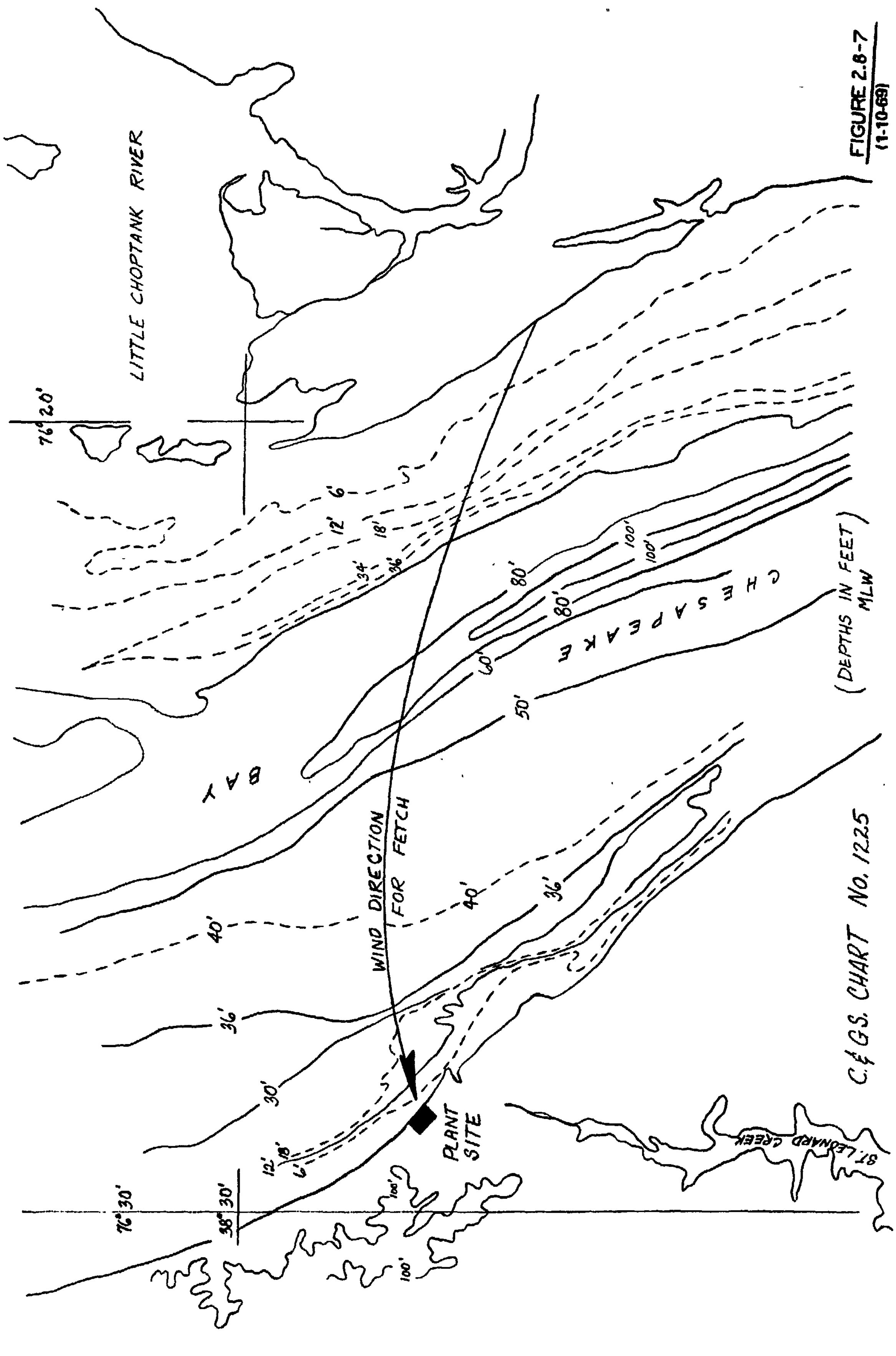


FIGURE 2.8-7
(1-10-69)

WIND DIRECTION FOR FETCH

FIGURE 2.8-8
BOTTOM PROFILE CHESAPEAKE BAY

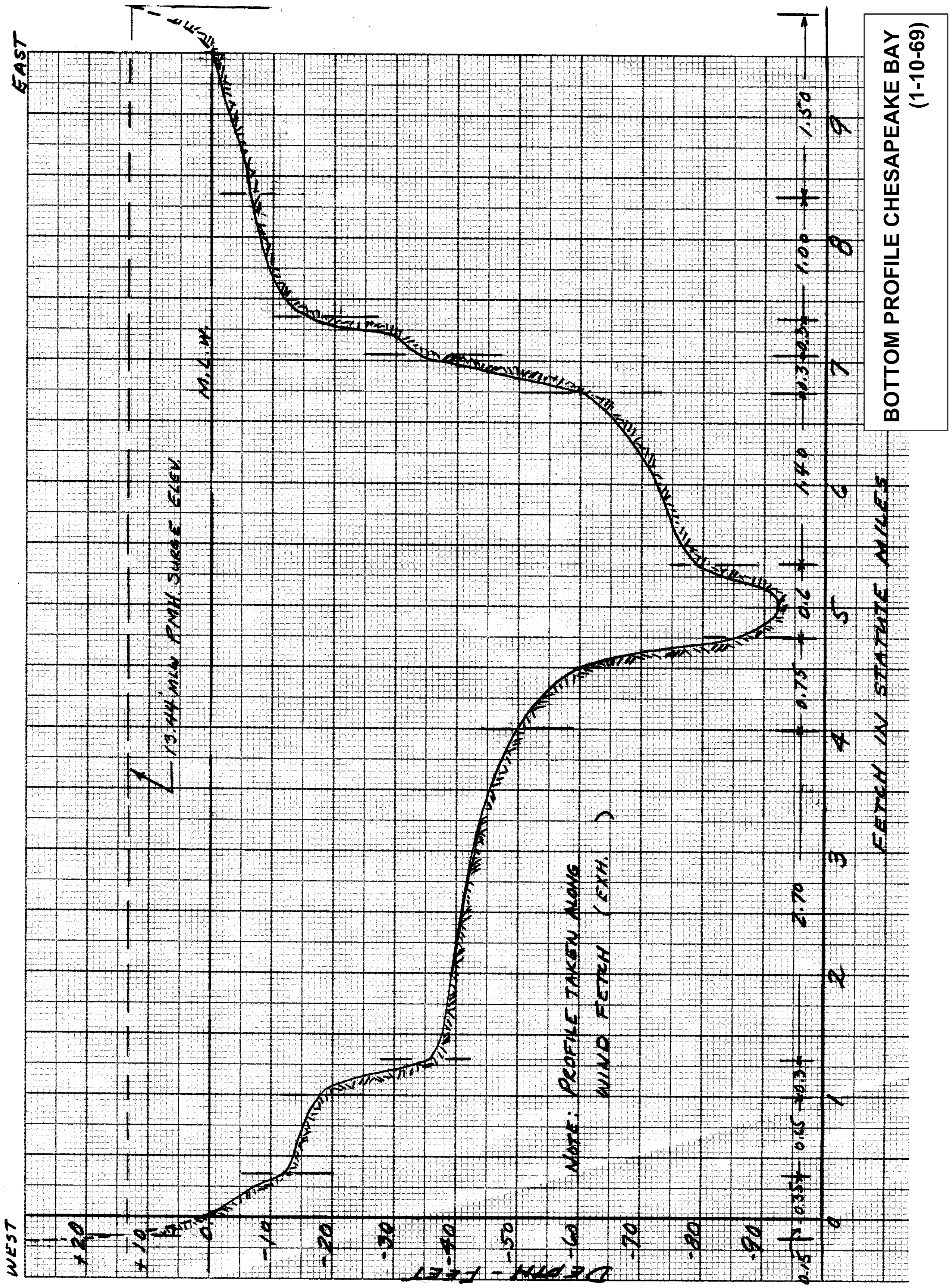
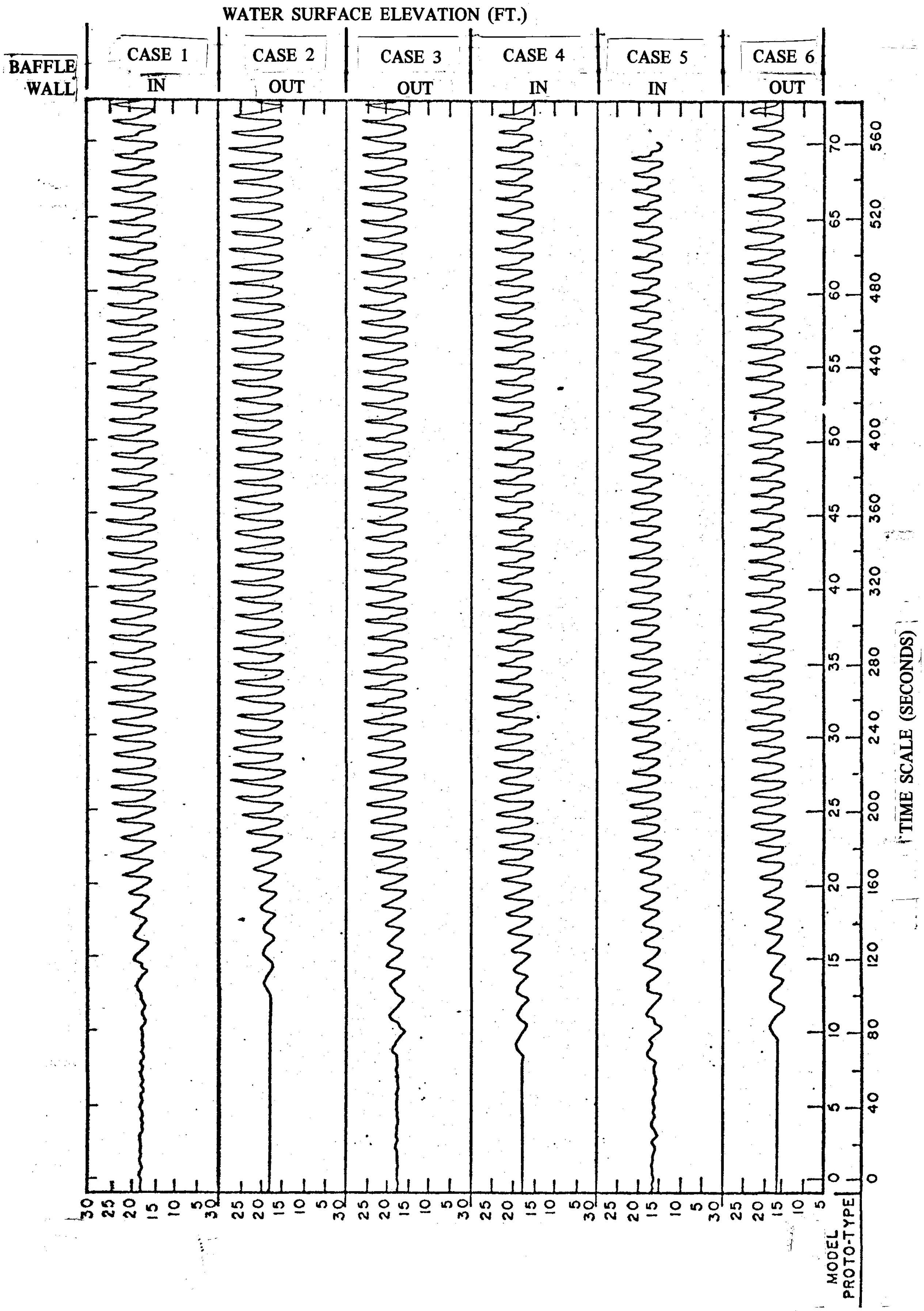


FIGURE 2.8-9
MODEL TEST RESULTS – WAVE RUNUP AT INTAKE STRUCTURE



2.9-1 CCNPP ENVIRONMENTAL SAMPLING SITES

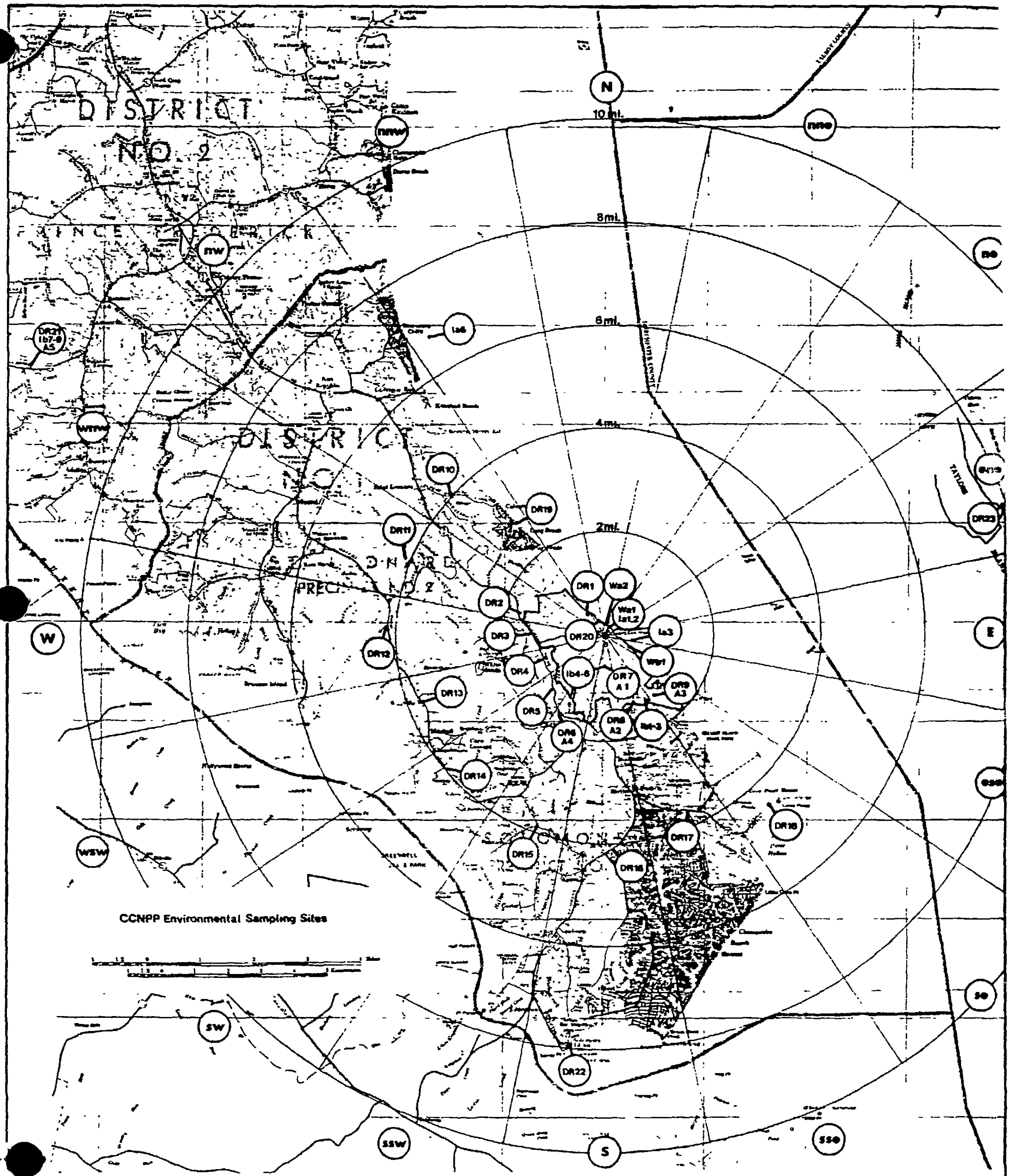


FIGURE 2.9-1 REV.11 1/91

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REACTOR

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FIGURE

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FIGURE

LIST OF ACRONYMS

ABB	Asea Brown Boveri, Inc.
ANF	Advanced Nuclear Fuel
AOO	Anticipated Operational Occurrence
APD	Axial Power Distribution
ARI	All Rods Inserted
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BGE	Baltimore Gas and Electric Company
BOC	Beginning of Cycle
BOL	Beginning of Life
BPR	Burnable Poison Rods
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEDS	Control Element Drive System
CHF	Critical Heat Flux
CVCS	Chemical and Volume Control System
DBE	Design Basis Event
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ENDF	Evaluated Nuclear Data File
EOC	End of Cycle
EOL	End of Life
ESCU	Extended Statistical Combination of Uncertainties
ESFAS	Engineered Safety Feature Actuation Signal
FANP	Framatome Advanced Nuclear Power (now AREVA)
FTC	Fuel Temperature Coefficient
GTFS	Guide Tube Flux Suppressor
HTP	High Thermal Performance
HMP	High Mechanical Performance
ICI	Incore Instrumentation
IFBA	Integral Fuel Burnable Absorber
LCO	Limiting Conditions for Operation
LEF	Lower End Fitting
LFA	Lead Fuel Assemblies
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LPD	Local Power Density
LSBR	Large Seed Blanket Reactor
LSSS	Limiting Safety System Setting
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MRR	Most Reactive Rod
MTC	Moderator Temperature Coefficient

LIST OF ACRONYMS

NEM	Nodal Expansion Method
NRC	Nuclear Regulatory Commission
PCI	Pellet-Clad Interaction
PDF	Probability Distribution Function
PDIL	Power Dependent Insertion Limit
PLCEA	Part Length Control Element Assembly
PLHR	Peak Linear Heat Rate
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPS	Reactor Protective System
RSS	Root-Sum-Square
SAFDL	Specified Acceptable Fuel Design Limit
SCU	Statistical Combination of Uncertainties
SS	Stainless Steel
T-H	Thermal Hydraulics
TD	Theoretical Density
TM/LP	Thermal Margin/Low Pressure
UGS	Upper Guide Structure
UO_2	Uranium Oxide
VAP	Value Added Pellet
VBWR	Vallecitos Boiling Water Reactor
ZrB_2	Zirc Diboride

3.0 REACTOR

3.1 GENERAL DESIGN SUMMARY

Both of the Calvert Cliffs reactors are of identical design. Consequently, reference throughout this section is made to a single reactor and, unless otherwise noted, implies either Unit 1 or 2.

The reactor is of the pressurized water type, using two reactor coolant loops. A vertical cross-section of the reactor is shown in Figure 3.1-1. The reactor core is composed of 217 fuel assemblies and 77 Control Element Assemblies (CEAs).

The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 136" and an active height of 136.7".

The fuel assembly consists of 176 rods (pins) and 5 guide tubes. The pins may contain fuel and/or a neutron poison. The assembly is held together by spacer grids and is closed at the top and bottom by end fittings.

Lateral support and positioning of the fuel rods within an assembly is provided by spacer grids. The spacer grids are welded to five full-length guide tubes. The guide tubes provide channels which guide the CEAs over their entire length of travel and form the longitudinal structure of the assembly. In selected fuel assemblies the central guide tube houses incore instrumentation (ICI). Design characteristics of demonstration or lead fuel assemblies are discussed in Section 3.7.

The fuel is low enrichment uranium dioxide (UO_2) in the form of ceramic pellets clad in Zircaloy or ZIRLO for Westinghouse fuel (as part of an advanced cladding test program, some fuel pins in Batches 2NT, 1RT, 2TF, and 2TW utilize cladding other than Zircaloy or ZIRLO) tubes which are welded into a hermetic enclosure. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, AREVA fuel uses M5® alloy cladding. Initially the fuel was managed in a three-cycle, mixed central zone, fuel management plan (Figure 3.4-3). Starting with Unit 2 Cycle 8 and Unit 1 Cycle 10, the 24-month cycle core utilized low leakage fuel management. Starting with Unit 1 Cycle 11 and Unit 2 Cycle 10, low fluence fuel management is employed to reduce the fluence on the critical vessel weld. Low fluence fuel management includes replacement of fresh fuel located on the core flats with once or twice-burned fuel. In Unit 1 Cycle 11 and Cycle 12 low fluence fuel management also included the addition of Guide Tube Flux Suppressors (GTFSs) in selected assemblies near the periphery. Sufficient margin is provided to ensure that power peaks are minimized.

The reactor coolant enters the upper section of the reactor vessel, flows downward between the reactor vessel wall and the core barrel, passes through the flow skirt where the flow distribution is equalized and into the lower plenum. The coolant then flows upward through the core, removing heat from the fuel rods, exits from the reactor vessel and passes through the tube side of the vertical U-tube steam generators where heat is transferred to the secondary system. The reactor coolant pumps return the coolant to the reactor vessel.

The reactor internals support and orient the fuel assemblies, the CEAs, and the incore instrumentation and guide the reactor coolant through the reactor vessel. The reactor internals also absorb static and dynamic loads and transmit the loads to the reactor vessel flange. They will safely perform their functions during normal operating, upset, emergency, and faulted conditions. The internals are designed to safely withstand forces due to deadweight, handling, temperature and pressure differentials, flow impingement, vibration, and seismic acceleration.

Reactivity control is provided by two independent systems: (1) the Control Element Drive System (CEDS) which controls CEA motion, and (2) the Chemical and Volume Control System (CVCS) which is used to control the Reactor Coolant System (RCS) boric acid concentration.

Boric acid dissolved in the coolant is used as a neutron absorber to provide long-term reactivity control. In order to reduce the boric acid concentration required at Beginning of Life (BOL) operating conditions and lower power peaking, mechanically fixed burnable poison rods (BPRs) may be provided in certain fuel assemblies. Originally, the neutron poison was boron carbide which is dispersed in alumina pellets; the pellets are clad in Zirconium alloy to form rods which are similar to the fuel rods. Gadolinia and erbium oxide, mixed into fuel pellets, are also being used as a neutron burnable absorber. Beginning with Unit 2 Cycle 16 and Unit 1 Cycle 18 Zirc Diboride (ZrB_2), applied as a coating on the fuel pellets, was used as a neutron burnable absorber. Poison rods are also called shims. Beginning in Unit 2 Cycle 19 and Unit 1 Cycle 21 with AREVA fuel, Gadolinia (Gd_2O_3) mixed into the fuel is used as the neutron burnable absorber.

The CEAs consist of five Inconel tubes filled with neutron absorbers. Four tubes are assembled in a square array around the central fifth tube. A spider joins the tubes at the upper end. The hub of the spider couples the CEA to the drive assembly. The CEAs are activated by magnetic jack Control Element Drive Mechanisms (CEDMs) mounted on the reactor vessel head.

The maximum reactivity worth of the CEAs and the associated reactivity addition rate are limited by system design to prevent sudden large reactivity increases. The design restraints are such that reactivity increases do not result in violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, nor disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

Control Element Assemblies are moved in groups to satisfy the requirements of shutdown, power level changes, and operational maneuvering. The control system is designed to produce power distributions that are within the acceptable limits of overall nuclear heat flux factor and Departure from Nucleate Boiling Ratio (DNBR). The Reactor Protective System (RPS) and administrative controls ensure that these limits are not exceeded.

In order to assure control of axial power distribution (APD), particularly in the event of axial xenon oscillation, eight CEAs designated as Part Length CEAs (PLCEA) were initially installed. They have since proved unnecessary and were removed along with their extension shafts. Control Element Assembly guide tube plugs were inserted into the locations previously occupied by the PLCEAs. They have also proved unnecessary and were removed before Unit 1 Cycle 8 and Unit 2 Cycle 7.

3.2 DESIGN BASIS

3.2.1 PERFORMANCE OBJECTIVES

The full-power thermal rating of the core is 2737 MWt. The physics, thermal and hydraulic information presented in this section is based on this power level.

3.2.2 DESIGN OBJECTIVES

The reactor core, together with its control systems and the RPS, is designed to function over its lifetime without exceeding fuel damage limits of excessive fuel temperature, cladding strain, and cladding stress (Section 3.2.3) during normal operating conditions and Design Basis Events (DBEs).

In the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in power. At the beginning of cycle (BOC) a slightly positive Moderator Temperature Coefficient (MTC) may occur. If power oscillations occur, their magnitude will be such that the fuel damage limits are not exceeded.

Reactivity control is provided by two independent systems: (1) the CEDS, and (2) the CVCS. The CEDS controls short-term reactivity changes and is used for rapid shutdown. The CVCS is used to compensate for long-term reactivity changes and can make the reactor subcritical without the benefit of the CEDS. The design of the core and the RPS prevents fuel damage limits from being exceeded for any single malfunction in either of the reactivity control systems.

The maximum reactivity addition rate from the withdrawal of the CEAs is limited by the core excess reactivity, CEA worth, and CEDS design. These limitations prevent sudden large reactivity increases. The design restraints are such that reactivity increases will not result in exceeding the fuel damage limits, rupture of the reactor coolant pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

3.2.3 DESIGN LIMITS

3.2.3.1 Nuclear Design Limits

The design of the core is based upon the following nuclear limitations:

a. The limitation on fuel burnup is determined by material design, mechanical design and nuclear considerations. The mechanical integrity of the fuel remains satisfactory beyond the planned discharge burnup.

b. In the power operating range, the effect of the prompt inherent nuclear feedback characteristic [Fuel Temperature Coefficient (FTC)] compensates for rapid increases in power.

c. CEAs are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce power distributions that are within the acceptable limits of overall Nuclear Heat Flux Factor (F_q^N) and DNBR. The RPS and administrative controls ensure that these limits are not exceeded.

d. Axial xenon oscillations, when they occur, will be manually controlled by regulating CEAs using information provided by the neutron flux detectors. The xenon oscillation period, about one day, allows ample time for operator action before the RPS trip setpoint is exceeded.

3.2.3.2 Reactivity Control Design Limits

The control system and operating procedures provide for adequate control of the core reactivity and power distributions such that the following limits are met:

a. Sufficient CEAs are withdrawn to provide an adequate shutdown reactivity margin;
b. The shutdown margin is maintained with the highest worth CEA assumed stuck in its fully withdrawn position;
c. The CVCS is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain the shutdown margin during a RCS cooldown at the design rate following a reactor trip.

3.2.3.3 Thermal and Hydraulic Design Limits

The principal basis of the thermal and hydraulic design is to avoid thermally-induced fuel damage during normal operation, and Design Basis Event (DBE). It is recognized that there is a small probability of limited fuel damage in certain unlikely situations as discussed in Chapter 14.

The following corollary thermal and hydraulic design bases are established, but violation of either is not necessarily equivalent to fuel damage:

a. There is a high confidence level that Departure from Nucleate Boiling (DNB) is avoided during normal operation and DBEs. This is achieved by setting a design lower limit on the Minimum Departure from Nucleate Boiling Ratio (MDNBR) calculated according to the Asea Brown Boveri, Inc. (ABB)-NV correlation for each cycle. Starting with Unit 1 Cycle 17, the ABB-TV correlation was used. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the high thermal performance (HTP) correlation was used to determine DNBR for AREVA fuel.
b. The melting point of the UO_2 fuel is not reached during normal operation nor during DBEs.

The RPS provides for automatic reactor trip before these design limits are exceeded.

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.2.3.4 Mechanical Design Limits

The reactor internals are designed to perform their functions safely during steady state conditions and DBEs. The internals can safely withstand the forces due to deadweight, handling, system pressure, flow-induced pressure drop, flow impingement, temperature differential, shock, and vibration. The structural components satisfy stress values given in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III.

The following limitations on stresses or deformations are employed to ensure capability of a safe and orderly shutdown in the combined event of earthquake and major loss-of-coolant accident (LOCA). For reactor vessel internal structures, the

stress criteria are given in Table 3.2-1. The intent of the limits in this table is as follows:

a. Under design loading plus design earthquake forces the critical reactor vessel internal structures are designed within the stress criteria established in the ASME B&PV Code, Section III, Article 4;
b. Under normal operating loadings plus maximum hypothetical earthquake forces, the design criteria permits a small amount of local yielding;
c. Under normal operating loadings plus reactor coolant pipe rupture loadings plus maximum hypothetical earthquake forces, permanent deformation is permitted by the design criteria.

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lbs/in^2.

Material	S_y[(a)]	S_u	S_D	S_L
SA 106B	25.4	60.0[(b)]	25.4	36.9
SA 533B	41.4	80.0[(b)]	41.4	54.3
304 SS	17.0	54.0[(c)]	18.35	29.3
316 SS	18.5	58.2[(c)]	22.2	31.7

(a) From ASME B&PV Code, Section III, at 650°F
(b) Minimum value at room temperature, which is approximately the same at 650°F for ferritic materials
(c) Estimated

S_u = Minimum tensile strength of material at temperature
S_L = $S_y + (1/3)(S_u - S_y)$
S_y = Tabulated yield at temperature from ASME B&PV Code, Section III
S_D = Design stress

To properly perform their functions, the critical reactor internal structures are designed to satisfy the additional deflection limits described below, in addition to the stress limits given in Table 3.2-1.

Under normal design loadings plus design earthquake forces or normal operating loadings plus maximum hypothetical earthquake forces, deflections are limited so that the CEAs can function and adequate core cooling is maintained. Under normal operating loadings plus maximum hypothetical earthquake forces plus pipe rupture loadings, the deflection design criteria depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main reactor coolant lines, deflections are limited so that the core is held in place, the CEAs function normally, and adequate core cooling is maintained. Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. For pipe breaks larger than the above, the criteria are that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. For these major pipe break sizes, CEA insertability is not required to achieve shutdown because the rapid voiding during the ensuing blowdown and the subsequent refill with the borated safety injection water ensures adequate shutdown margin for the reactor. For the larger break sizes, critical components are restrained from buckling by further limiting the stress levels to two-thirds of the stress level calculated to produce buckling
.

3.2.3.5 Fuel Assembly Design Limits

The fuel assemblies are designed to maintain their structural integrity under steady state conditions, DBEs, normal handling loads, shipping stresses, and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components including welds are highly resistant to the corrosive action of the reactor environment.

The fuel rod design accounts for external pressure, differential expansion of fuel and clad, fuel swelling, clad creep, fission and other gas releases, thermal stress, pressure and temperature cycling, and flow-induced vibrations. The structural criteria are based on the following:

a. The maximum primary stress during steady state operation, expected transients, and depressurization is limited to two-thirds of the minimum yield strength of the material at operating temperature.
b. The predicted total strain of the cladding at the End of Life (EOL) is less than 1.0%.

AREVA has performed the mechanical design analyses starting with the Unit 2 Cycle 19 and Unit 1 Cycle 21 fuel assembly design. These evaluations used the Nuclear Regulatory Commission (NRC)-approved mechanical analysis codes and methodology to demonstrate compliance with the NRC-approved design criteria. (References 1, 2, 3, and 4)

3.2.3.6 Control Element Assembly Design Limits

The CEAs are designed to maintain their structural integrity under all steady state conditions, DBEs and handling, shipping and refueling loads. Thermal distortion, mechanical tolerances, vibration and wear of the CEA are all accounted for in the design. Clearances and corresponding fuel assembly alignment are established so that possible accumulation of mechanical tolerances and thermal distortion will not result in frictional forces that could prevent reliable operation of the system. The structural criteria are based on limiting the maximum stress intensity to those values specified in Section III of the ASME B&PV Code.

The clearance between the CEA fingers and the guide tubes is designed for actuating within the prescribed time under steady state conditions, during DBEs, under maximum hypothetical earthquake, and temperature conditions in combination with various factors which cause a reduction in diametral clearance. These factors include adverse dimensional tolerances, bowing and twisting of CEA and guide tubes and possible enlargement of the poison rod diameter due to swelling of B_4C pellets at maximum burnup conditions. The design diametral change due to swelling of B_4C is based on the pellets being rigid and the high strength clad offers no restraint to pellet diametral growth.

The core is designed to limit deflections so that the core is held in place. The CEAs function and adequate core cooling is maintained even under:

a. Normal design loadings plus design earthquake forces;
b. Normal operating loadings plus maximum hypothetical earthquake forces plus a pipe break no larger than the equivalent diameter largest line connected to the main reactor coolant lines.

If the equivalent diameter of the pipe break is larger than the largest coolant line, the core is designed so that fuel is held in place to permit core cooling and adequate coolant flow is maintained.

Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. If the equivalent diameter of the pipe break is larger than condition b above, the core is designed so that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. For these major pipe breaks, CEA insertion is not required to achieve shutdown because the rapid voiding during blowdown and the refilling of the vessel with borated safety injection water ensures adequate shutdown margin for the reactor.

The speed at which the CEAs are inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation (Chapter 7). For conditions that require a rapid shutdown of the reactor, the CEDM holding coils deenergize to allow the CEAs to drop into the core. The reactivity is reduced during such a CEA drop at a rate sufficient to prevent exceeding fuel damage limits. A CEA automatic drive-down capability after a reactor trip is not required. During a trip, the RPS opens the trip circuit breakers, deenergizing the CEDM holding coils allowing the CEAs to drop by gravity into the core. To drive down a CEA stuck in the fully withdrawn position, the operator must first clear the trip condition and manually close the trip circuit breaker. Therefore, a drive-down feature would introduce the possibility of a failure which would prevent power from being removed from the CEDM holding coils. The safety analysis (Chapter 14) assumes the CEA of highest reactivity worth sticks in the fully withdrawn position.

The CEDM pressure housings are an extension of the reactor vessel, providing a part of the reactor coolant boundary, and are, therefore, designed to meet the requirements of the ASME B&PV Code, Section III, Nuclear Vessels. Pressure and thermal transients as well as steady state loadings were considered in the design analysis.

3.2.4 REFERENCES

1. ANF-88-133(P)(A), Revision 0 and Supplement 1, “Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU,” December 1991
2. XN-NF-82-06(P)(A), Revision 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” October 1986
3. BAW-10133(P)(A), Revision 1, Addendum 1 and Addendum 2, “Mark-C Fuel Assembly LOCA-Seismic Analysis,” October 2000
4. EMF-92-116(P)(A), Revision 0, Supplement 1 (P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs,” February 2015

TABLE 3.2-1

PRIMARY STRESS LIMITS FOR CRITICAL REACTOR VESSEL INTERNAL STRUCTURES

LOADING COMBINATIONS	**ALLOWABLE STRESSES**
Design Loading Plus Design Earthquake Forces	$P_m \le S_m$ $P_b + P_L \le 1.5S_m$
Normal Operating Loading Plus Maximum Hypothetical Earthquake Forces	$P_m \le S_D$ $P_b \le 1.5\left(1-\left(\frac{P_m}{S_D}\right)^2\right)S_D$
Normal Operating Loadings Plus Maximum Hypothetical Earthquake Forces Plus Pipe Rupture Loadings	$P_m \le S_L$ $P_b \le 1.5\left(1-\left(\frac{P_m}{S_L}\right)^2\right)S_L$

where:
LEGEND

P_m	=	Calculated Primary Membrane Stress, psi
P_b	=	Calculated Primary Bending Stress, psi
P_L	=	Calculated Primary Local Membrane Stress, psi
S_m	=	Tabulated Allowable Stress Limit at Temperature from ASME B&PV Code, Section III or ANSI B31.7, psi
S_y		Tabulated Yield Strength at Temperature, ASME B&PV Code, Section III, psi
S_D	=	Design Stress, psi
S_D	=	S_y (for ferritic steels), psi
S_D	=	$1.2S_m$ (for austenitic steels), psi
S_L	=	$S_y + 1/3\ (S_u - S_y)$, psi
S_u	=	Tensile Strength of Material at Temperature, psi

3.3 MECHANICAL DESIGN

3.3.1 SUMMARY

The reactor core and internals are shown in Figure 3.1-1. A cross-section of the reactor core and internals is shown in Figure 3.3-1. Mechanical design features of the reactor internals, the CEDMs and the reactor core are described below. Mechanical design parameters are listed in Tables 3.3-1, 3.3-2, 3.3-3, 3.3-4, and 3.3-5.

The fuel for Unit 2 is essentially identical to that of Unit 1. After the first cycle of Unit 1, a number of minor refinements (shown in Tables 3.3-1 and 3.3-2) were incorporated for the purpose of improving overall fuel performance. The principal changes are that the pellet density has been increased slightly and the overall pellet geometry modified. The increased pellet density, along with improvements in pellet microstructure, has the effect of improving the in-pile dimensional stability of the pellet, thereby lessening the adverse effect of in-pile densification on gap conductance and axial gap formation. The reduced pellet length-to-diameter (L/D) ratio and the use of chamfered pellets have the effect of reducing the severity of interaction between the pellets and the clad. Also, the fuel has been modified to permit replacement of fuel rods. These refinements represent standard practice among Combustion Engineering, Inc. (CE) reactors like the Calvert Cliffs design. Beginning in Unit 2 Cycle 19 and Unit 1 Cycle 21, the fuel is provided by AREVA.

3.3.2 CORE MECHANICAL DESIGN

The core approximates a right circular cylinder with an equivalent diameter of 136" and an active height of 136.7". It consists of Zircaloy-4 or ZIRLO (as part of an advanced cladding test program, some fuel pins in Batches 2NT, 1RT, 2TF, and 2TW utilize other cladding materials) clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, AREVA fuel uses the M5® alloy cladding. In addition, there are BPRs in certain fuel batches. The fuel rods are grouped into 217 assemblies. The enrichment of each batch of fuel is shown in Tables 3.3-1 and 3.3-2.

Short-term reactivity control is provided by 77 CEAs. The CEAs are guided within the core by the guide tubes which are integral parts of the fuel assemblies.

3.3.2.1 Fuel Rod Mechanical Design

The fuel rods consist of slightly enriched UO_2 cylindrical ceramic pellets. The first cycle fuel rod is shown in Figure 3.3-2. Recent Westinghouse fuel rod designs are shown in Figures 3.3-3A, 3.3-3B, and 3.3-3C. Originally, a round wire Type 302 stainless steel (SS) compression spring, and an alumina spacer disc were located at each end of the fuel column, all clad within a seamless Zircaloy-4 or ZIRLO tube with a Zircaloy-4 cap welded at each end. As part of an advanced cladding test program, some fuel pins in Batches 2NT, 1RT, 2TF, and 2TW utilize other cladding materials. Beginning with Unit 1 Cycle 12 and Unit 2 Cycle 11, the upper alumina spacer disc was removed. Beginning with the Unit 1 Cycle 16 Batch 1V rods manufactured at the Columbia facility, the lower alumina spacer disk was removed. The fuel rods manufactured by Hematite are evacuated and internally pressurized with helium to compensate for the pressure difference across the clad, minimizing clad collapse. The fuel rods built at Columbia are not evacuated prior to being pressurized with Helium. Helium, an inert gas, is chosen as the pressurizing medium because of its thermal conductivity. Cladding creep-collapse time for fuel was analyzed for each cycle until Unit 1 Cycle 8 and Unit 2 Cycle 7. Analysis of modern pressurized water reactor (PWR) fuels has demonstrated that the clad collapse time is significantly longer than its expected useful life. Therefore, cycle-specific clad collapse time is not calculated.

Each fuel rod assembly includes a unique serial number. The unique serial number ensures traceability of the fabrication history of each fuel rod. The fuel cladding is cold worked and stress-relief annealed Zircaloy-4 or ZIRLO seamless tubing. The actual tube forming process consists of a series of cold working and annealing operations.

The UO_2 pellets are dished and chamfered on both ends in order to better accommodate thermal expansion and fuel swelling. The pellet length to diameter ratio and the use of chamfered pellets decrease the interaction between the pellet and the clad. However, because the pellet dishes and chamfers constitute about 3% of the pellet, stack height density is reduced. The stack height density and pellet dimensions are given in Tables 3.3-1 and 3.3-2.

The compression spring, located at the top of the fuel pellet column, maintains the column in its proper position during handling and shipping. The alumina spacer disc at the lower end of the fuel rods with magnetic force welds is to protect the weld from radial strain induced by pellet swelling, while the upper spacer disc prevents UO_2 chips, if present, from entering the plenum region. Beginning with Unit 1 Cycle 12 and Unit 2 Cycle 11, the upper alumina (Al_2O_3) spacer disc was removed. Starting with the Unit 1 Cycle 16 rods built in Columbia, the lower alumina spacer was eliminated. The plenum spring is a low volume plenum spring. This provides greater margin between the EOL internal pin pressure and the rods mechanical design limit than earlier designs. The fuel rod plenum, initially pressurized with helium, provides space for axial thermal expansion of the fuel column and accommodates the gaseous fission products. The greater portion of the fission gas remains in the pellet lattice and does not contribute to the rod internal pressure.

Beginning in Unit 2 Cycle 19 and Unit 1 Cycle 21, the fuel is provided by AREVA and the general design is similar. The cladding is a zirconium alloy, M5®. The fuel column is sintered UO_2 pellets, 136.7” long (nominally) with a plenum spring at the top of the rod column. Each rod is pressurized with helium and sealed with caps welded at each end. The fuel column continues to have low enriched axial blanket pellets at both ends. The AREVA fuel rod is shown in Figures 3.3-3 and 3.3-16.

3.3.2.2 Burnable Poison Rod Mechanical Design

Fixed burnable poison (neutron absorbing) rods are included in selected fuel assemblies to reduce the BOL MTC. They replace fuel rods at selected locations. The various sheets of Figure 3.3-4 show assembly configurations for various fuel bundles. The poison rods are mechanically similar to fuel rods, but contain a column of burnable poison pellets instead of fuel pellets. The poison material consists of alumina with uniformly dispersed boron carbide particles. Mechanical design parameters are listed in Table 3.3-3.

The balance of the column contains Zircaloy-4 pellets. The BPR plenum spring produces a smaller preload on the pellet column than that in a fuel rod because of the lighter material in the poison pellets.

Each BPR includes a unique serial number and a batch identification mark. The serial number is used to record fabrication information for each component in the rod assembly. It ensures traceability of the fabrication history of each rod. The batch identification mark provides a visual check on the pellet boron concentration during fuel assembly fabrication.

For Unit 1 Cycle 10, four lead test assemblies containing Gadolinia as a burnable absorber were introduced. Twelve of the 176 fuel bearing rods in each of the test

assemblies contain a mixture of 10 wt% Gd_2O_3 and natural (not enriched in U-235) UO_2 pellets stacked over an active length of 122.7" within the rod. The top and bottom 7" of the column contain natural UO_2 pellets without Gadolinia. The test assembly poison rod pellets are otherwise mechanically identical to fuel pins.

For Unit 2 Cycle 9, four demonstration assemblies containing Erbium as a burnable absorber were introduced. Each assembly consists of 80 standard pins at 4.3 wt% U-235, 52 standard pins at 3.4 wt% U-235, and 44 Erbium bearing pins. The fuel stack in each Erbium bearing fuel pin consists of a central 115.7" region containing 3.4 wt% U-235, 0.9 wt% Er_2O_3, UO_2/Er_2O_3 pellets and two 10.5" cutback regions, one at each end of the stack containing standard 3.4 wt% U-235 pellets.

For Unit 2 Cycle 10, Batch 2M burnable absorber pins consist of a 115.7" central region containing the burnable material B_4C with two 10.5" cutback regions containing Al_2O_3, one at each end of the stack. This change is being made to enhance thermal margin by lowering the axial peak at BOC.

Beginning with Unit 1 Cycle 12 and Unit 2 Cycle 11, selected fuel pins contain erbia (Er_2O_3) as the integral burnable absorber (in lieu of B_4C). The erbium fuel pins consist of a central region containing the burnable absorber mixed with UO_2 at the batch enrichment and a cutback region at the upper and lower ends of the fuel rods. The cutback region consists of UO_2 at the batch enrichment, and enhances the thermal margin by lowering the core average axial peak.

Beginning with Unit 2 Cycle 16 and Unit 1 Cycle 18, selected fuel pins contain ZrB_2 as the integral burnable absorber (in lieu of erbia). The ZrB_2 fuel pins consist of a central region containing the burnable absorber. The ZrB_2 is applied as a very thin coating on the outside surface of select UO_2 fuel pellets. The ZrB_2 rods have a poison cutback region at the upper and lower ends of the fuel rods.

Beginning in Unit 2 Cycle 19 and Unit 1 Cycle 21, the fuel is provided by AREVA, and uses Gadolinia, dispersed in the UO_2 fuel as a burnable poison.

3.3.2.3 <u>Fuel Assembly Mechanical Design</u>

The fuel assembly (Figure 3.3-5) consists of 176 fuel rods and poison rods, 5 guide tubes, 5 guide tube sleeves (except as noted by Section 3.7), 8 fuel rod spacer grids, upper and lower end fittings (LEFs), and a hold-down device (Figure 3.3-6). The guide tubes, spacer grids, and end fittings form the structural frame of the assembly. The four outer guide tubes are mechanically attached to the end fittings and the spacer grids are welded to all five guide tubes.

The fuel rod spacer grids for the Westinghouse fuel (Figures 3.3-7, 3.3-7A, 3.3-7B, and 3.3-16) maintain the fuel rod pitch over the length of the rod. The grid provides positive lateral restraint to the fuel rod but only frictional restraint axially. The grids are fabricated from preformed Zircaloy, interlocked in an egg crate fashion, and welded together. The grid supports each fuel rod by two cantilever tab springs or two I-springs and four arches. The springs press the rod against the arches to restrict relative motion between the grids and the fuel rods. The spring and arch positions are reversed from grid to grid to provide additional movement restrictions. The perimeter strips contain features designed to prevent hang up of grids during a refueling operation. The eight Zircaloy-4 spacer grids are welded to each Zircaloy-4 guide tube at eight locations, four on the upper face of the grid and four on the lower face of the grid, where the spacer strips contact the guide tube surface.

The Westinghouse fuel assembly upper end fitting consists of a 304 SS flow plate, a SS hold-down plate, five machined posts, and five Inconel X-750 compression springs. The upper end fitting attaches to the guide tubes to serve as an aligning and lifting device for each fuel assembly. The flow plate is attached to the top ends of the guide tubes and is designed to prevent excessive axial motion of the fuel rods. Inconel X-750 is selected for the compression springs because of its previous use for coil springs and good resistance to relaxation during operation. The hold-down plate, together with the compression springs, comprise the hold-down device. The hold-down plate is axially movable. It is loaded by the compression springs and held down by the fuel alignment plate. The spring load combines with the fuel assembly weight to counteract upward hydraulic forces. The determination of upward hydraulic forces includes factors accounting for flow maldistribution, fuel assembly component tolerances, oxide buildup, drag coefficient, and bypass flow. The springs are sized and the spring preload selected such that a net downward force of at least 150 pounds will be maintained for all normal and anticipated transient flow and temperature conditions. The design criteria limit the maximum stress under the most adverse tolerance conditions to below yield strength of the spring material. The maximum stress occurs during cold conditions and decreases as the reactor heats up. The reduction in stress is due to a decrease in spring deflection resulting from differential thermal expansion between the Zircaloy fuel bundles and the SS internals.

The Westinghouse fuel LEF consists of an Inconel grid welded to a cast SS plate which has flow holes and four support legs. The support legs also serve as alignment posts. Precision-drilled holes in the support legs mate with four core support plate alignment pins, thereby properly locating the lower end of the fuel assembly.

Beginning with Batch 1D and 2D fuel and continuing with subsequent assemblies the bottom spacer grid is used in lieu of a mechanical retention grid to laterally locate the bottom of the fuel rods. The grid allows for removal and replacement of rods. The four outer guide tubes have a widened region at the upper end which contains an internal thread.

Beginning with Batch 1K and 2J fuel and continuing with subsequent assemblies, the height of the LEF was shortened by shortening the support legs. The overall lengths of the guide tubes were increased to compensate for the shorter LEF. The elevations of the Inconel grid and the uppermost Zircaloy grid were changed to maintain their same relative elevations with respect to previous assemblies.

In Batch 2L, a debris-resistant LEF design was used in which a 3x3 array of small flow holes replaces each of the large flow holes of the previous design. Also, wherever possible, additional small holes are added to minimize the increase in the pressure drop of the small hole LEF design, relative to the previous design.

Beginning with Batch 1N and 2M, the fuel incorporates the GUARDIAN™ fuel assembly design to entrap debris. The GUARDIAN™ design employs a redesigned Inconel spacer grid and redesigned rods that have longer, solid Zircaloy-4 lower end caps. Changes incorporated into the GUARDIAN™ fuel assembly include an increase in length of the lower end caps, an increase in the length of the fuel and burnable absorber rods, and a decrease in the length of the plenum regions. The length of the guide tubes is increased to maintain the shoulder gap, and the height of the upper end fitting is reduced to maintain the overall length of the fuel bundle. The change in height of the upper end fitting is accomplished by decreasing the compression region for the hold-down spring without a change in dimension of the

spring. The height of the LEF is reduced and "T" stanchions are added to aid fuel handling.

Beginning with Batch 1N and Batch 2M, Zircaloy spacer grids are redesigned to allow fuel rods located along the periphery of the fuel bundle to receive more coolant flow. This is performed through an increase in the outer pin cell size by enlargement of the outside envelope of the spacer grid assembly.

The externally-threaded end of each guide post passes through a hole in the flow plate and is torqued into the internally-threaded guide tube. When assembled, the flow plate is secured between flanges on the guide tubes and on the guide post. The connection with the upper end fitting is locked with a mechanical crimp. Each outer guide tube has, at its lower end, a welded Zircaloy-4 fitting. Either a threaded portion of this fitting passes through a hole in the fuel assembly LEF and is secured by a Zircaloy-4 nut and a SS locking ring, or a fitting with an internal thread engages with a hole in the LEF and is secured by a SS bolt and locking ring. The locking ring is tack welded to the LEF in four places.

The central guide tube inserts into sockets in the upper and lower end fittings and is thus retained laterally by the relatively small clearance at these locations. The upper end fitting socket is created by the center post which is threaded into the lower cast flow plate and tack welded in two places. The LEF socket is machined out of the LEF casting. There is no positive axial connection between the central guide tube and the end fittings.

A SS guide tube sleeve (except as noted by Section 3.7), located in the upper region of the guide tube/post, prevents guide tube wear. Fretting wear was caused by coolant turbulence inducing vibratory motion in the CEAs which rubbed against the guide tube wall. Significant wear has been found to be limited to the relative soft Zircaloy-4 guide tube because the Inconel-625 cladding on the CEAs provides a relatively hard wear surface. Beginning with Unit 1 Cycle 3, (Unit 2 Cycle 2) and continuing in subsequent cycles, SS sleeves were installed in fuel assembly guide tubes with significant wear, and in fuel assembly guide tubes under most CEAs. In addition to the installation of sleeves in guide tubes to prevent wear, some assemblies were fabricated for Unit 2 Cycle 2, Unit 1 Cycle 4, and Unit 1 Cycle 5 with reduced flow guide tubes to reduce CEA vibration.

The sleeve is of slightly cold worked 304 SS, chrome plated on the inside surface. The chrome plating provides resistance to wear without the risk of promoting wear in the CEA Inconel cladding. The nominal wall thickness is adequate for free movement of the CEA and does not significantly increase the maximum CEA drop time. To secure the sleeves in the guide tube, the lower ends of the sleeves are expanded radially so that the guide tubes are permanently expanded. The lower third of the sleeve is also expanded outward so that the outside of the sleeve contacts the guide tube.

Beginning with Batch 1K and 2J fuel assemblies a modified short-sleeve design is used. This allows for reconstitution of the assemblies without having to remove and reinstall the guide tube sleeves. All new fuel assemblies are sleeved with the short-sleeve design (Reference 3).

The five guide tubes have the effect of ensuring that bowing or excessive swelling of the adjacent fuel rods or poison rods cannot result in obstruction of the CEA pathway. This is so because:

a. There is sufficient clearance between the fuel rods and the guide tube surface to allow an adjacent fuel rod to reach rupture strain without contacting the guide tube surface.
b. The guide tube, having considerably greater diameter and wall thickness (and at a lower temperature) than the fuel rod, is considerably stiffer than the fuel rods and would, therefore, remain straight, rather than be deflected by contact with the surface of an adjacent fuel rod.

Therefore, the bowing or swelling of fuel rods would not result in obstruction of the control element channels such as could hinder CEA movement.

The fuel assembly design enables reconstitution (i.e., removal and replacement of fuel rods or poison rods) of an irradiated fuel assembly. The fuel rod and poison rod lower end caps are conically shaped to ensure proper insertion within the fuel assembly grid cage structure. The upper end caps are designed to enable remote grappling of the fuel rod or poison rod for purposes of removal and handling. The five posts may be untorqued and removed from the guide tubes, allowing the removal of the upper end fitting assembly as one unit (a hold-down plate, a flow plate, five posts and five springs) with a single tool. This removal provides access to the fuel rods and poison rods for replacement or servicing. Before loading into the core, the threaded joints which mechanically attach the upper end fitting to the guide tubes are properly torqued and locked.

A unique serial number on each fuel assembly upper end fitting enables verification of fuel enrichment and orientation of the fuel assembly. Indication is also provided on the LEF to ensure preservation of fuel assembly orientation in the event of upper end fitting removal.

The lower end of each rod has a serial number to provide a means of identifying the pellet enrichment, pellet lot, and fuel stack weight. In addition, a quality control program specification requires that measures be established for the identification and control of materials, components, and partially fabricated subassemblies. These means provide assurance that only acceptable items are used and also provides a method of relating an item or assembly from initial receipt through fabrication, installation, repair, or modification to an applicable drawing, specification, or other pertinent technical document.

For the AREVA design, the spacer grids are the Zircaloy-4 HTP™ spacers at all elevations except the bottom spacer. The bottom spacer is an Alloy 718 high mechanical performance (HMP™) spacer. The upper end fitting is the standard AREVA reconstitutable design that has been used at other CE14 units. The lower end fitting is the FUELGUARD™ design used to provide resistance against debris entering the fuel assembly. Features such as the capability to reconstruct fuel assemblies, individual rod and bundle identification are maintained. The springs are sized and the spring preload selected such that a net downward force will be maintained for all normal and anticipated transient flow and temperature conditions. The AREVA fuel assembly is shown in Figure 3.3-16.

3.3.2.4 Control Element Assembly Mechanical Design

The CEA (Figures 3.3-8, 3.3-9A, and 3.3-9B and Table 3.3-4) consists of five Inconel 625 tubes (fingers) loaded with a stack of cylindrical neutron absorber pellets. The absorber material is boron carbide (B_4C), with the exception of the lower portion of the four corner fingers (original design) and some center fingers (new design) which contain silver indium cadmium (Ag-In-Cd). The silver indium cadmium material reduces clad strain which radiation-induced swelling of boron carbide might cause.

The AREVA full strength CEA rod has a slightly different configuration in the lower portion of the rod. The AREVA full strength CEA rod design contains a stack support that resides within the annulus of the silver indium cadmium (Ag-In-Cd) stack. This stack support is comprised of a support column that passes through the Ag-In-Cd annulus and a support platform, upon which the B_4C column rests. The stack support prevents the weight of the B_4C column and plenum spring preload from compressing the Ag-In-Cd stack which is susceptible to deformation through creep during operation; a significant contributor to clad strain. The stack support reduces the creep mechanism of the lower absorber and thereby reduces cladding strain.

Above the poison pellet column is a plenum which provides expansion volume to limit the internal pressure from the gases released from the boron carbide such that the primary stress does not exceed the yield strength of the cladding material at operating conditions. The plenum contains a hold-down spring which restrains the absorber material against longitudinal movement while allowing for differential expansion between the absorber and the clad. The spring also maintains the position of the absorber material during shipping and handling.

Each finger is sealed by one Inconel 625 nose cap welded at the bottom and one Inconel 625 end fitting at the top. The end fittings are attached to a spider hub structure in a square array with one finger centrally located. The spider provides rigid support for the control elements. The spider provides a point of attachment for coupling the CEA to the CEA extension shaft. A unique serial number is on each hub to provide identification.

During normal operation all of the CEAs are normally in the fully withdrawn position. Mechanical reactivity control is achieved by vertically maneuvering the positions of the CEA groups by the magnetic jack CEDMs. Each CEDM is positioned on the reactor vessel closure head and is coupled to the CEA by the CEA extension shaft.

There are 37 single CEAs and 20 dual CEAs. Each dual CEA consists of two single CEAs connected to a single extension shaft and carried by a single CEDM. Considering the 20 dual CEAs as 40 single CEAs gives an overall equivalent of 77 single CEAs in the core (Figures 3.3-10 and 3.3-11). The center CEA in group 5 is weakened in absorption capability.

In the withdrawn position the CEA resides in the Upper Guide Structure (UGS), enclosed in CEA shrouds. The shrouds provide guidance and protect the CEA and the extension shaft from coolant cross flow. Within the core, each CEA finger travels in a Zircaloy guide tube. The guide tubes are part of the fuel assembly structure and ensure proper orientation of the CEAs with respect to the fuel rods.

When the extension shaft is released by the CEDM, gravity causes the CEA to insert into the full length of the fuel assembly. The four outer guide tubes of each assembly have a reduced diameter lower section which allows for hydraulic buffering action to

slow down the CEAs near the end of their travel. The CEA velocity is decreased to minimize impact. There is a small bleed hole on the side of the buffer section of the guide tube which prevents pressure buildup and allows some coolant flow. This hydraulic damping action is augmented by a spring arrangement attached between the central CEA post and the hub. When fully inserted, the CEAs rest on the central post of the fuel assembly upper end fitting.

A prototype CEA was installed in Unit 2 at the BOC 3. The changes from standard design included a change in cladding material (from Inconel to SS), reconstitutable fingers, and a change in material for the tips of the poison fingers from Ag/In/Cd to B_4C. The size of the B_4C pellets used in the tips was decreased from the pellet size used for the remainder of the rod length. A metal liner was added to prevent any B_4C fragments from collecting in the high flux tip. This CEA was discharged at the End of Cycle (EOC) 8.

In Unit 1, nine CEAs were replaced for Cycle 8 and the rest were replaced for Cycle 9. Eight CEAs (FLCEA2, FLCEA5) were replaced in Unit 2 Cycle 7. The replacement CEAs have essentially the same design as the original components with the exceptions that replacement CEAs have reconstitutable corner fingers, and have Ag-In-Cd tips in all fingers (with the exception of the weakened center CEA in Group 5).

For Unit 2 Cycle 8 and Unit 1 Cycle 10, the first 24-month cycles, the weakened CEA in the center CEA position was replaced with a less weak CEA (FLCEA5) with all reconstitutable fingers.

For Unit 2 Cycle 9, the 69 remaining full-strength, old-style CEAs (with B_4C to the bottom of the center finger) were replaced. The replacements (FLCEA1) were non-reconstitutable and have Ag-In-Cd tips in all fingers. In addition, the weakened center CEA was replaced (Unit 2 Cycle 9 and Unit 1 Cycle 11) with a weakened CEA containing SS in the bottom of each of the four weak fingers (FLCEA6) instead of a Zircaloy slug. The Zircaloy slug was found to be subject to hydriding, in this application.

For Unit 2 Cycle 14, the reduced strength re-constitutable CEA (FLCEA6) was replaced with an equivalent reduced strength non-reconstitutable CEA (FLCEA7).

For Unit 1 Cycle 16, the reduced strength re-constitutable CEA (FLCEA6) was replaced with an equivalent reduced strength non-re-constitutable CEA (FLCEA7). Additionally, eight full strength re-constitutable CEAs with 8" Ag-In-Cd poison stacks (FLCEA2) were replaced with eight full strength non-reconstitutable CEAs with 12" Ag-In-Cd poison stacks (FLCEA8).

For Unit 2 Cycle 15, 12 full length CEAs (10 of the standard design and 2 with reconstitutable corner fingers) were replaced with full strength, non-reconstitutable CEAs with 12” Ag-In-Cd poison stacks (FLCEA8).

For Unit 1 Cycle 17, 68 full-length full-strength re-constitutable CEAs with 8” Ag-In-Cd poison stacks (FLCEA2) were replaced with full-length full-strength non-reconstitutable CEAs with 12” Ag-In-Cd poison stacks (FLCEA8). All of the Unit 1 full-length full-strength CEAs are of the 12” Ag-In-Cd poison stack design.

For Unit 2 Cycle 16, 64 full-length full-strength CEAs with 8” Ag-In-Cd poison stacks were replaced with full-length full-strength CEAs with 12” Ag-In-Cd poison stacks.

All of the Unit 2 full-length full-strength CEAs are of the 12” Ag-In-Cd poison stack design.

For Unit 2 Cycle 18, 2 full-length full-strength CEAs with 12” Ag-In-Cd poison stacks were replaced with full-length full-strength CEAs with 8” Ag-In-Cd poison stacks.

For Unit 1 Cycle 20, 2-full length full-strength CEAs with 12” Ag-In-Cd poison stacks were replaced with full-length full-strength CEAs with 8” Ag-In-Cd poison stacks.

For Unit 2 Cycle 19, all of the full-length full-strength CEAs are of the 12” Ag-In-Cd poison stack design.

For Unit 1 Cycle 21, the center CEA is a full-length part-strength CEA of the 8” Ag-In-Cd poison stack design, and the remaining 76 CEAs are of the full-length full-strength 12” Ag-In-Cd poison stack design.

For Unit 2 Cycle 20, the center CEA is a full-length part-strength CEA of the 8” Ag-In-Cd poison stack design, and one CEA is of the full-length full-strength 8” Ag-In-Cd poison stack design. The remaining 75 CEAs are of the full-length full-strength 12” Ag-In-Cd poison stack design.

There are two approved for use AREVA CEA designs, full-length part-strength and full-length full-strength. The center CEA is a full-length part-strength CEA of the 12.5" Ag-In-Cd poison stack design. The remaining 76 CEAs are of the full-length full-strength 12.5" Ag-In-Cd poison stack design.

3.3.2.5 Neutron Source Design

A discrete neutron source was required for a quick, safe startup of the original core. Two plutonium-238/antimony-beryllium (Pu/Sb-Be) neutron sources were located in guide tubes of peripheral fuel assemblies. The discrete neutron sources are not necessary for restart, and were removed from the reactor for Unit 1 Cycle 9 and Unit 2 Cycle 8.

3.3.2.6 Guide Tube Flux Suppressor Design

For Unit 1 Cycle 11 and Cycle 12, GTFSs were installed into selected peripheral assemblies. The basic design of the GTFSs is identical to that of the CEA fingers with regard to the B_4C pellets, Al_2O_3 spacer pellets, and the Inconel 625 cladding (Table 3.3-4). The active core region consists of 116.2" of B_4C with 10.25" of Al_2O_3 spacers at each end.

3.3.2.7 Test Capsule Assembly Design

The Test Capsule Assembly Program is being conducted to evaluate the effects of irradiation at reactor temperatures on materials being considered for advanced spacer grid spring designs. TCA-1, TCA-2, and TCA-3 (inserted beginning with Unit 1 Cycle 12) consist of, from top to bottom, a holddown assembly, an upper extension tube, 7 capsules connected axially by 6 connecting tubes, and a lower extension tube with an endplug. TCA-4 and TCA-5 (inserted beginning with Unit 1 Cycle 13) consist of, from top to bottom, a holddown assembly, an upper extension tube, containing an unused test capsule, 6 test capsules connected axially by 6 connecting tubes, and a lower extension tube with a bottom endplug that contains a test capsule that is used. The SS holddown assembly, located entirely above the core, is similar to the holddown assembly for a flux suppressor and, like a flux suppressor, is designed to preload the capsule assembly against the bottom of the

guide tube in which it resides. The capsules, extension tubes, and connecting tubes are fabricated from Inconel CEA tubing and bar stock material. All tubular sections have holes which allow the free ingress and egress of reactor coolant. The upper extension tube is sized to position the used capsules in the middle 80% of the core. The lower extension tube is designed to extend into the buffer region of the outer guide tube in order to center the capsule assembly and to prevent lateral movement. The purpose of the connecting tubes is to facilitate separation of the capsules from one another using a shearing tool in the spent fuel pool.

In Unit 1 Cycle 12, three test capsules were placed in the outer guide tubes of three separate once-burned fuel assemblies.

Four test capsules were placed in the outer guide tubes of four separate fresh fuel assemblies in Unit 1 Cycle 13. Two of these capsules were from Unit 1 Cycle 12 and two are new capsules.

Two test capsules were placed in the outer guide tubes of two separate fuel assemblies in Unit 1 Cycle 14. Both of these capsules were reinserted from Unit 1 Cycle 13. The 2 test capsules (TCA-3 and TCA-5) were discharged at the end of Unit 1 Cycle 14.

3.3.2.8 ZIRLO Cladding (Westinghouse Fuel)

In the late 1990s, Calvert Cliffs identified clad spallation phenomena on its 2nd cycle high duty fuel. That fuel used the CE standard OPTIN cladding material. OPTIN is an Optimized Process Low Tin cladding that falls within the overall Zircaloy-4 material specification. In order to eliminate the spallation phenomena, Calvert Cliffs elected to switch to an alternate clad material that has better water-side corrosion properties. The alternate cladding material selected is the Westinghouse standard ZIRLO clad material. ZIRLO is a Westinghouse proprietary modification of Zircaloy-4 material achieved by reducing the tin and iron content, eliminating the chromium content, and adding niobium. Calvert Cliffs began to phase in the use of ZIRLO cladding starting with some of the rods for Unit 1 Cycle 16 (Batch 1V).

Westinghouse submitted a topical report (Reference 4) to the NRC. On September 12, 2001, the NRC issued a safety evaluation report to approve the use of ZIRLO cladding material in CE reactors. The NRC authorized full batch implementation of ZIRLO cladding without lead test fuel assemblies, but placed the following restrictions on the use of ZIRLO:

a. The corrosion limit, as predicted by the best-estimate model, will remain below 100 microns for all locations of the fuel.
b. All the conditions listed in the safety evaluations for all the CENPD methodologies used for ZIRLO fuel analysis will continue to be met, except that the use of ZIRLO cladding in addition to Zircaloy-4 cladding is now approved.
c. All CENP methodologies will be used only within the range for which ZIRLO data was acceptable and for which the verifications discussed in Reference 4 and responses to requests for additional information were performed.
d. Until data is available demonstrating the performance or ZIRLO cladding in CE designed plants, the fuel duty will be limited for each CE designed plant with some provision for adequate margin to account for variations in core design (e.g., cycle length, plant operating conditions, etc.). Details of this condition will be addressed on a plant specific basis during the approval to use ZIRLO in a specific plant.

e. The burnup limit for this approval is 60 GWD/MTU.

3.3.2.9 M5 Cladding (AREVA Fuel)

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, new AREVA fuel uses M5® cladding. AREVA submitted a topical report (Reference 5) to the NRC. The NRC issued a safety evaluation report to approve the use of M5® cladding material in CE reactors with the following restrictions:

a. The corrosion limit, as predicted by the best-estimate model, will remain below 100 microns for all locations of the fuel.
b. All the conditions listed in the safety evaluations for all the FANP methodologies used for M5® fuel analysis will continue to be met, except that the use of M5® cladding in addition to Zircaloy-4 cladding is now approved.
c. All FANP methodologies will be used only within the range for which M5® data was acceptable and for which the verifications discussed in References 5 or 6 was performed.
d. The burnup limit for this approval is 62 GWD/MTU.

3.3.2.10 Axial Blankets

Beginning with Unit 2 Cycle 16 and Unit 1 Cycle 18, the top and bottom 6 inches of pellets in all new fuel pins contain low enriched (2.6 w/o) fuel. This feature reduces axial neutron leakage and increases fuel economics.

All Zirc diboride fuel pins contain axial blankets with annular holes that remove approximately 25% of the volume of the pellet. The annular holes provide additional volume for gas production as a result of the boron coating being converted into helium gas.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the AREVA fuel uses 6” of low enriched (≤2.0 w/o) axial blankets on the top and bottom of non-gadolina-bearing fuel rods. Twelve inches of low enriched (≤2.0 w/o) axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

3.3.2.11 Radial Enrichment Zoning

Unit 2 Cycle 16 and Unit 1 Cycle 18 saw the introduction of radial enrichment zoning. In these cycles, eight pins adjacent to each CEA guide tube (40 in all) and three pins at each assembly corner (12 in all) contained a lower enrichment than the other fuel pins. Beginning in Unit 2 Cycle 17, some of the subbatches are as described above, and in others, three enrichments are used. The three enrichment patterns are intended primarily to reduce calculated steaming rates.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the AREVA fuel uses two-enrichment radial zoning with the eight pins adjacent to each CEA guide tube (40 in all) and three pins at each assembly corner (12 in all) containing a lower enrichment than the other fuel non-Gadolinia pins.

3.3.3 REACTOR INTERNAL STRUCTURES

The reactor internals are designed to support and orient the reactor core fuel assemblies and CEAs, absorb the CEA dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and guide incore instrumentation.

The internals are designed to safely perform their functions during all steady state conditions and during DBEs. The internals are designed to safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, vibration and seismic acceleration. All reactor components are considered Category I for seismic design. The reactor internals design provides limits of deflection where functionally required. The structural components satisfy stress values given in the ASME B&PV Code, Section III. Certain components have been subjected to a fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation.

The components of the reactor internals are divided into three major parts:

a. The core support barrel,
b. The lower core support structure (including the core shroud), and
c. The UGS (including the CEA shrouds and the ICI guide tubes).

The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the pressure vessel (Figure 3.1-1).

3.3.3.1 Core Support Assembly

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the lower support structure, and the core shroud. The major material for the assembly is Type 304 SS. The core support barrel supports the core support assembly.

The upper flange of the core support barrel rests on a ledge in the reactor vessel flange. The lower flange of the core support barrel supports and positions the lower support structure. The core support plate transmits the weight of the core to the core support barrel by means of vertical columns, an annular skirt, and beam structure. The core support plate provides support and orientation for the fuel assemblies. The core shroud, which provides lateral support for the peripheral fuel assemblies, is also supported by the core support plate. The lower end of the core support barrel is restrained radially by six snubbers.

3.3.3.2 Core Support Barrel

The core support barrel approximates a right circular cylinder with a nominal inside diameter of 148" and a minimum wall thickness of 1-3/4". It is suspended by a 4" thick flange from a ledge on the pressure vessel. The core support barrel supports the lower support structure upon which the fuel assemblies rest. Press fitted into the flange of the core support barrel are four alignment keys located 90° apart. The reactor vessel, the closure head, and the UGS assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment and to prevent excess motion between these components in the vessel flange region.

Since the core support barrel is about 27' long and is supported only at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers (Figure 3.3-12), are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally spaced double lugs around the circumference and are the grooves of a "tongue-and-groove" assembly; the pressure vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. The pressure vessel tongues have bolted, lock welded Inconel X shims and the core support barrel grooves are hard faced with Stellite to minimize wear.

With this design, the internals may be viewed as a beam with supports at the furthest extremities. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted by this design.

3.3.3.3 Core Support Plate and Support Column

The core support plate aligns the fuel assemblies and directs coolant flow through them. It is a 147" diameter, 2" thick, Type 304 SS plate with the necessary machined flow distributor holes for the fuel assemblies. Fuel assembly locating pins (four for each assembly) are shrink-fitted into the support plate. An annular skirt, columns, and support beams are located between the support plate and the bottom of the core support barrel. They provide a support for this plate and transmit the core load to the bottom flange of the core support barrel.

3.3.3.4 Core Shroud

The core shroud (Figure 3.3-13) provides an envelope for the core and limits the amount of coolant bypass flow. The shroud is 152-1/2" tall and 147-5/16" in diameter. The shroud consists of two Type 304 SS ring sections, aligned by means of radial shear pins and attached to the core support plate by eight Type 348 SS tie rods. A gap is maintained between the core shroud outer perimeter and the core support barrel in order to provide some coolant flow upward between the core shroud and core support barrel. This minimizes thermal stresses in the core shroud and eliminates stagnant pockets.

The gap between the outside of the peripheral fuel assemblies and the shroud is maintained by eight tiers of stiffening plates attached to the shroud. In locations where mechanical connections are used, bolts and pins are lock welded. All bolts are designed to be captured in the event of fracture. The bolt heads are trapped by lock bars or lock welds and the bolt bodies are trapped by incomplete tapping of holes. Holes are provided in the core support structure to allow coolant to flow upward between the core shroud and the core support barrel, thereby minimizing thermal stresses in the shroud and eliminating stagnant pockets.

3.3.3.5 Flow Skirt

The Inconel flow skirt is a 3,500 pound right circular cylinder, perforated with 2-11/16 in. diameter holes, and reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced machined sections which are welded to the bottom head of the pressure vessel.

3.3.3.6 Upper Guide Structure Assembly

The UGS assembly (Figure 3.3-14) consists of:

a. The upper support structure;
b. Sixty-five CEA shrouds;
c. A fuel assembly alignment plate; and,
d. An expansion compensating ring.

The UGS assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel assemblies from being lifted

out of position during a severe accident condition, and protects the CEAs from the effect of coolant cross flow in the upper plenum. The UGS is handled as one unit during installation and is removed for refueling.

The upper end of the UGS assembly is a support plate welded to a grid array of 24" deep beams and a 24" deep cylinder which encloses, and is welded to the ends of the beams. The periphery of the plate contains four accurately machined and located alignment keyways, equally spaced at 90° intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid structure aligns and supports the upper end of the CEA shrouds.

The CEA shrouds extend from the fuel assembly alignment plate to an Elevation about 3' above the support plate. There are 45 single-type shrouds. These consist of cylindrical upper sections welded to integral bottom sections, which are shaped to provide flow passages for the coolant passing through the alignment plate while shrouding the CEAs from cross flow. Also, there are 20 dual-type shrouds which, in configuration, consist of two single-type shrouds connected by a rectangular section shaped to accommodate the dual CEAs. The bottoms of the shrouds are bolted to the fuel assembly alignment plate. At the UGS support plate, the single shrouds are connected to the plate by spanner nuts which permit axial adjustment. The spanner nuts are tightened to the proper torque and lock-welded. The dual shrouds are welded to the upper plate.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and the lower ends of the CEA shrouds, as well as support the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate align the fuel assemblies (Figure 3.3-6). The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hard-faced pins protruding from the core shroud to limit lateral motion of the UGS assembly during operation. The alignment plate load and the weight of a fuel assembly produce a net downward force to counteract upward hydraulic forces for normal operating conditions and all DBEs. The fuel assembly alignment plate would capture the core and limit upward movement in the event of an accident.

A holddown ring acts as a shim between the reactor vessel flange and the UGS. It resists axial upward movement of the UGS assembly. This arrangement accommodates axial differential thermal expansion between the core barrel flange, UGS flange, the reactor vessel flange mating surface and head flange recess. The UGS also supports the incore instrumentation guide tubes. The tubes are conduits which protect the incore instrumentation and guide them during removal and insertion operations.

3.3.4 CONTROL ELEMENT DRIVE MECHANISM

3.3.4.1 Design

The CEDM is of the magnetic jack-type drive. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 137" stroke (Figure 3.3-15). The design of the CEDM is identical to that for Maine Yankee (Reference 2).

The CEDM drives the CEA within the reactor core and indicates the position of the CEA with respect to the core. The speed at which the CEA is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor

operation. For conditions that require a rapid shutdown of the reactor, the CEDM coils are deenergized, allowing the CEA and the extension shaft to drop into the core by gravity. The reactivity is reduced during such a drop at a rate sufficient to control the core under any operating transient or accident condition.

The CEA is decelerated at the end of the drop by the buffer section of the CEA guide tubes.

Originally, 65 CEDMs (61 CEDMs on the replacement reactor vessel closure head) were mounted on flanged nozzles on top of the reactor vessel closure head. Eight CEDMs were nonscrammable and were connected to the PLCEAs which have been removed (4 spare CEDMs are installed in the replacement reactor vessel closure head). Each CEDM extension shaft is connected to a CEA by a locked coupling. The weight of the CEAs and CEDMs is carried by the vessel head.

The CEDM is designed to handle dual or single CEAs. The total stroke of the drive is 137". The maximum withdrawing speed of CEDMs is 30" per minute for single CEAs and 20" per minute for dual CEAs. The maximum allowed time from receiving a trip signal to the essentially fully inserted position of the CEA is specified in the Technical Specifications.

a. CEDM Pressure Housing

   Each CEDM housing is attached to the reactor vessel head nozzle by means of a threaded joint and seal welded. It need not be removed since all servicing of the CEDM is performed from the top of the CEDM housing. This opening is closed by means of a threaded cap and omega seal weld.

   The CEDM upper housing design and fabrication conforms to the requirements of the ASME B&PV Code, Section III, for Class 1 appurtenances for the replacement reactor vessel closure head). The housing is designed for steady state conditions as well as all anticipated pressure and thermal transients.

b. Magnetic Jack Assembly

   The magnetic jack motor assembly fits into the CEDM housing through an opening in the top of the housing. This integral unit carries the motor tube, lift and hold pawls, and magnets. Electrical coils positioned around the CEDM housing supply the drive power. The CEDMs are cooled by forced air which maintains CEDM coil temperature below 350°F. Loss of cooling air will not prevent the CEDM from releasing the CEAs when a reactor trip is initiated. A description of the air circulation system is presented in Chapter 9.

   The upper housing cap is threaded into the CEDM housing and seal welded after the CEDM motor assembly is inserted. This cap supports the position indication housing which encloses the CEDM extension shaft.

   The lifting operation consists of a series of magnetically-operated step movements. Two sets of mechanical latches engage a notched drive shaft. To prevent excessive latch wear, a means has been provided to unload the latches during the engaging and disengaging operations.

   The magnetic force is obtained from large DC magnet coils mounted on the outside of the motor tube. Power for the electromagnets is obtained from

two separate supplies. A control programmer actuates the stepping cycle and positions the CEA by a forward or reverse stepping sequence. The CEA is held stationary by energizing one coil at a reduced current while all other coils are deenergized. The CEAs are tripped upon interruption of electrical power to all coils.

c. Position Indication

Three separate means are provided for transmitting CEA position indication.

The first method utilizes the electrical pulses from the magnetic coil power programmer. The second method utilizes reed switches and a voltage divider network mounted on the CEDM to provide an output voltage proportional to CEA position. The third method utilizes three pairs of reed switches spaced at discrete locations within a position transmitter assembly. A permanent magnet built into the drive shaft actuates the reed switches one at a time as it passes by them. CEA position instrumentation is discussed in detail in Chapter 7.

d. Control Element Assembly Disconnect

The CEA connects to the drive shaft extension with an internal collet-type coupling at its lower end. Coupling is performed before the vessel head is installed. In order to disengage the CEA from the drive shaft extension, a tool is attached to the top end of the drive shaft when the reactor vessel head (along with all the CEDMs) has been removed.

By pulling up on the spring-loaded operating rod in the center of the drive shaft, a tapered plunger is withdrawn from the center of the collet-type gripper causing it to collapse due to axial pressure from the CEA, thus permitting removal of the coupler from the CEA. Releasing the operating rod plunger after the coupler has been withdrawn from the CEA expands the coupler to a diameter that prevents recoupling to the CEA. At this point, the drive shaft buffer is resting on the positive stop in the CEA shroud. The drive shafts, uncoupled from the CEAs, are removed along with the UGS (when the UGS is removed from the vessel).

3.3.5 REFERENCES

1. Deleted
2. Maine Yankee Final Safety Report, Docket No. 50-309
3. Letter from A.E. Scherer (CE) to C.O. Thomas (NRC), "CEA Guide Tube Wear Sleeve Modification," LD-84-043, August 3, 1984
4. CENPD-404-P-A, "Implementation of ZIRLO Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001
5. BAW-10240P-A, Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods," May 2004
6. BAW-10227P-A, Revision 01, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ASSEMBLY AVERAGE ENRICHMENT wt% U-235	NUMBER OF B_4C SHIMS PER ASSEMBLY	INITIAL SHIM LOADING wt% B_4C	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
1A	2.05	0	---	395	176
1B	2.45	12	2.9	368	164
1C	2.99	0	---	395	176
1C+	2.99	12	1.1	369	164
1C.	2.99	12	.68	368	164
1D	3.03	0	---	388	176
1D/	2.73	0	---	388	176
1E	3.03	0	---	388	176
1E/	2.73	0	---	387	176
1F	3.03	0	---	389	176
1F/	2.73	0	---	389	176
1G	3.65	0	---	388	176
1G/	3.03	8	3.03	371	168
1H	4.00	0	---	389	176
1H/	3.55	8	3.03	372	168
1J	4.05	0	---	389	176
1J*	3.40	0	---	389	176
1K	4.05	0	---	389	176
1K*	3.40	0	---	388	176
1L	4.05	0	---	388	176
1L*	3.40	0	---	388	176
1M	4.08	0	---	393	176
1M*	4.08	12	4.09	365	164
1MX	3.85	0	---	377	176
1N	4.20	0	---	393	176
1NX	4.20	4	4.04	383	172
1N/	4.20	8	4.04	373	168

Batch 1MX is Advanced Nuclear Fuel (ANF) demonstration fuel with 12 Gd_2O_3 (10 wt%) fuel bearing (natural uranium) poison rods per assembly.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ASSEMBLY AVERAGE ENRICHMENT wt% U-235	NUMBER OF ERBIUM SHIMS PER ASSEMBLY	INITIAL SHIM LOADING wt% Er_2O_3	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
1P0	4.30	0	0.0	392	176
1P1	4.30	20	2.0	392	176
1P2	4.30	44	2.0	391	176
1P3	4.30	60	2.0	390	176
1R0	4.48	20	2.00	391	176
1R1	4.48	44	2.00	391	176
1R2	4.48	68	2.00	390	176
1RT	4.00 (VAP)	44	1.75	408	176
1S0	4.30	0	0.0	393	176
1S1	4.30	20	2.0	393	176
1S2	4.30	44	2.0	392	176
1S3	4.30	68	2.0	391	176
1T0	4.28 (VAP)	0	0.0	408	176
1T1	4.28 (VAP)	20	1.75	408	176
1T2	4.28 (VAP)	44	1.75	407	176
1V0	4.25 (VAP)	0	0.0	410	176
1V1	4.25 (VAP)	44	1.75	408	176
1V2	4.25 (VAP)	60	1.75	407	176
1W0	4.25 (VAP)	0	0.00	409	176
1W1	4.25 (VAP)	20	2.00	409	176
1W2	4.25 (VAP)	44	2.00	407	176
1W3	4.25 (VAP)	60	2.00	406	176
1W4	4.25 (VAP)	60	2.00	406	176

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF ZrB_2 SHIMS PER ASSEMBLY	INITIAL SHIM LOADING (mg-B-10/ inch)	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
1X1	2.6/4.0/4.5 (VAP)	44	3.29	407	176
1X2	2.6/4.0/4.5 (VAP)	52	3.29	407	176
1X3	2.6/4.0/4.5 (VAP)	64	3.29	406	176
1X4	2.6/4.0/4.5 (VAP)	76	3.29	406	176
1X5	2.6/4.0/4.5 (VAP)	96	3.29	405	176
1X7	2.0 (VAP)	0	0	409	176
1Z1	4.95/4.65/2.60 (VAP)	0	0	409	176
1Z2	4.95/4.65/2.60 (VAP)	28	3.29	408	176
1Z3	4.95/4.65/2.60 (VAP)	44	3.29	407	176
1Z4	4.95/4.65/4.00/2.60 (VAP)	64	3.29	406	176
1Z5	4.95/4.65/4.00/2.60 (VAP)	76	3.29	406	176
1Z6	4.95/4.65/4.00/2.60 (VAP)	96	3.29	405	176
AA1	4.95/4.00/2.60 (VAP)	28	3.29	407	176
AA2	4.95/4.00/2.60 (VAP)	52	3.29	406	176
AA3	4.95/4.55/4.00/2.60 (VAP)	64	3.29	406	176
AA4	4.95/4.55/4.00/2.60 (VAP)	76	3.29	405	176
AA5	4.95/4.55/4.00/2.60 (VAP)	96	3.29	404	176
2X7	2.00 (VAP)	0	0	409	176

VAP Value Added Pellet

Batch 2X7 assemblies were purchased as spare assemblies for Unit 2 Cycle 18. They were not used for U2C18; hence they are being employed for U1C20.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
AB1	4.60/4.00/2.00 (3.60/2.60)	4/12	4/8	407	176
AB2	4.60/4.00/2.00 (3.60/3.20)	4/12	4/6	408	176
AB3	4.60/4.00/2.00 (3.60)	12	4	409	176

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
AC1	4.87/4.20/2.00 (3.60/2.80)	4/12	4/8	407	176
AC2	4.87/4.20/2.00 (3.20)	12	6	408	176
AC3	4.87/4.20/2.00 (3.60)	8	4	409	176

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
AD1	4.90/4.30/2.00 (3.60/2.50)	4/12	4/8	407	176
AD2	4.90/4.30/2.00 (3.60/3.20)	4/12	4/6	408	176
AD3	4.90/4.30/2.00 (3.60)	12	4	409	176
AD4	4.90/4.30/2.00 (3.60/3.20)	4/4	4/6	409	176

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 1

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1A	69	0	12,144
1B	80	960	13,108
1C	40	0	7,040
1C+	16	192	2,624
1C.	12	144	1,968
TOTALS	217	1,296	36,896

In Cycle 1, Batch B included three test assemblies. Each contains four SS rods as well as twelve poison rods.

UNIT 1 CYCLE 2

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1B	77	924	12,620
1C	40	0	7,040
1C+	16	192	2,624
1C.	12	144	1,968
1D	48	0	8,448
1D/	24	0	4,224
TOTALS	217	1,260	36,924

In Cycle 2, Batch B included two test assemblies. Each contains four SS rods as well as twelve poison rods.

UNIT 1 CYCLE 3

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1A	40	0	7,040
1B	1	12	160
1C	32	0	5,632
1D	48	0	8,448
1D/	24	0	4,224
1E	48	0	8,448
1E/	24	0	4,224
TOTALS	217	12	38,176

In Cycle 3, Batch B is a test assembly. In addition to the twelve poison rods, it contains four SS rods.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 4

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1B*	1	12	160
1D	48	0	8,488
1D/	24	0	4,224
1E	48	0	8,448
1E/	24	0	4,224
1F	48	0	8,448
1F/	24	0	4,224
TOTALS	217	12	38,176

* This is the test assembly. In addition to the twelve poison rods, it contains four SS rods, one in each corner.

UNIT 1 CYCLE 5

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1D	1	0	175
1E	48	0	8,448
1E/	4	0	704
1F	48	0	8,448
1F/	24	0	4,224
1G	40	0	7,040
1G/	52	416	8,736
TOTALS	217	416	37,775

In Cycle 5, the Batch 1D fuel assembly contains one SS rod.

UNIT 1 CYCLE 6

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1D	1	0	176
2D	8	0	1,408
1F	44	0	7,743
1G	40	0	7,040
1G/	52	416	8,736
1H	40	0	7,040
1H/	32	256	5,376
TOTALS	217	672	37,519

Batch F includes one test assembly (SCOUT) that contains an SS rod.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 7

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2B	12	144	1,968
2D/	12	0	2,112
1E/	12	0	2,112
1F	5	0	877
1G	40	0	7,038
1H	40	0	7,040
1H/	32	256	5,376
1J	48	0	8,448
1J*	16	0	2,816
TOTALS	217	400	37,787

Batch F includes one test assembly (SCOUT) that contains three SS rods.

Batch G includes four test assemblies (PROTOTYPE) that contains two stainless steel rods.

UNIT 1 CYCLE 8

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2E	4	0	704
1F	1	3	173
1G	4	3	701
1H/	32	256	5,376
1H	40	0	7,040
1J*	16	0	2,816
1J	48	0	8,448
1K*	24	0	4,224
1K	48	0	8,448
TOTALS	217	262	37,930

The Batch F assembly is a test assembly (SCOUT) that contains three stainless steel rods.

The four Batch G test assemblies (PROTOTYPE) contain three SS rods.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 9

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2E	24	0	4,224
1G	4	2	702
1H	1	0	176
1J*	16	0	2,816
1J	48	2	8,446
1K*	24	2	4,222
1K	48	2	8,446
1L*	12	0	2,112
1L	40	0	7,040
TOTALS	217	8	38,184

All non-fuel rods in Cycle 9 contain SS.

In Batch 1G, one test assembly (PROTOTYPE) contains two SS rods. Prior to Cycle 9, one SS rod was replaced with a test rod from SCOUT.

Batch 1J includes one assembly with two stainless rods.

Batch 1K includes two assemblies with a total of two stainless rods.

Batch 1K* includes two assemblies with a total of two stainless rods.

UNIT 1 CYCLE 10

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1K	48	2	8,446
1K*	21	2	3,694
1L	40	2	7,038
1L*	12	2	2,110
1M	16	0	2,816
1M*	76	912	12,464
1MX	4	0	704
TOTALS	217	920	37,272

Batch 1L includes one assembly with two SS rods.

Batch 1L* includes two assemblies with a total of two SS rods.

Batch 1MX is ANF demonstration fuel with 12 Gd_2O_3 (10 wt%) fuel bearing (natural uranium) poison rods per assembly.

Batches 1K and 1K* have four assemblies with one SS rod in each.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 11

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1K*	1	0	176
1L	36	3	6,333
1M	16	2	2,814
1M*	76	923	12,453
1MX	4	0	704
1N	12	0	2,112
1NX	20	80	3,440
1N/	52	416	8,736
TOTALS	217	1,424	36,768

Batch 1L includes two assemblies with a total of three SS rods.

Batch 1M includes one assembly with a total of two SS rods.

Batch 1M* includes five assemblies with a total of eleven SS rods.

Batch 1MX is ANF demonstration fuel with 12 Gd_2O_3 (10 wt%) fuel bearing (natural uranium) poison rods per assembly.

UNIT 1 CYCLE 12

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1K*	1	0	176
1L	4	1	703
1M	16	2	2,814
1M*	20	245	3,275
1MX	4	0	704
1N	12	0	2,112
1NX	20	80	3,440
1N/	52	416	8,736
1P0	16	0	2,816
1P1	12	0	2,112
1P2	8	0	1,408
1P3	52	0	9,152
TOTALS	217	744	37,448

Batch 1L includes one assembly with a total of one SS rod.

Batch 1M includes one assembly with a total of two SS rods.

Batch 1M* includes one assembly with a total of five SS rods.

Batch 1MX is ANF demonstration fuel with 12 Gd_2O_3 (10 wt%) fuel bearing (natural uranium) poison rods per assembly.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 13

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2J*	1	0	176
1N	8	0	1,408
1NX	16	64	2,752
1N/	16	128	2,688
1P0	16	0	2,816
1P1	12	0	2,112
1P2	8	0	1,408
1P3	52	0	9,152
1R0	24	0	4,224
1R1	28	0	4,928
1R2	32	0	5,632
1RT	4	0	704
TOTALS	217	192	38,000

UNIT 1 CYCLE 14

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2J*	5	0	880
1M*	4	48	656
1P0	16	0	2,816
1P1	12	0	2,112
1P2	8	0	1,408
1R0	24	0	4,224
1R1	28	0	4,928
1R2	32	0	5,632
1RT	4	0	704
1S0	24	0	4,224
1S1	4	0	704
1S2	40	0	7,040
1S3	16	0	2,816
TOTALS	217	48	38,144

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 15

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1L*	4	0	704
1NX	4	16	688
1RT	2	2	350
1R2	6	1	1,055
1R0	24	0	4,224
1S3	16	0	2,816
1S2	40	0	7,040
1S1	4	0	704
1S0	24	0	4,224
1T2	60	0	10,560
1T1	4	0	704
1T0	28	0	4,928
2J*	1	0	176
TOTALS	217	19	38,173

Assembly 1RT1 has 2 stainless steel rods.

Assembly 1R222 has 1 stainless steel rod.

UNIT 1 CYCLE 16

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1V0	24	0	4,224
1V1	48	0	8,448
1V2	24	0	4,224
1T0	28	0	4,928
1T1	4	0	704
1T2	60	0	10,560
1S0	20	0	3,520
1S1	4	0	704
1S2	4	0	704
2J*	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 17

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1W0	20	0	3,520
1W1	12	0	2,112
1W2	36	0	6,336
1W3	16	0	2,816
1W4	4	0	704
1V0	24	0	4,224
1V1	48	0	8,448
1V2	24	0	4,224
1T0	20	0	3,520
1T1	4	0	704
1T2	8	0	1,408
1L*	1	0	176
TOTALS	217	0	38,192

UNIT 1 CYCLE 18

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
X1	32	0	5,632
X2	8	0	1,408
X3	12	0	2,112
X4	24	0	4,224
X5	20	0	3,520
X7	1	0	176
W0	20	0	3,520
W1	12	0	2,112
W2	36	4	6,332
W3	16	0	2,816
W4	4	0	704
V0	16	0	2,816
V1	16	0	2,816
TOTALS	217	4	38,188

Batch W2 includes three assemblies with a total of four stainless rods.

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 19

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
1Z1	4	0	704
1Z2	8	0	1,408
1Z3	12	0	2,112
1Z4	24	0	4,224
1Z5	40	0	7,040
1Z6	8	0	1,408
1X1	32	1	5,631
1X2	8	0	1,408
1X3	12	0	2,112
1X4	20	0	3,520
1X5	20	0	3,520
1W0	8	0	1,408
1W1	8	0	1,408
1W2	8	0	1,408
2V4	1	0	176
2TF	2	0	352
2TW	2	0	352
TOTALS	217	1	38,191

UNIT 1 CYCLE 20

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
AA1	12	0	2,112
AA2	8	0	1,408
AA3	12	0	2,112
AA4	36	0	6,336
AA5	20	0	3,520
2X7	4	0	704
1Z1	4	0	704
1Z2	8	0	1,408
1Z3	12	0	2,112
1Z4	24	0	4,224
1Z5	40	0	7,040
1Z6	8	0	1,408
1X1	24	0	4,224
1W0	4	0	704
2V4	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

UNIT 1 CYCLE 21

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
AB1	48	0	8,448
AB2	16	0	2,816
AB3	32	0	5,632
AA1	12	0	2,112
AA2	8	0	1,408
AA3	12	0	2,112
AA4	36	0	6,336
AA5	20	0	3,520
2X7	4	0	704
1Z1	4	0	704
1Z2	8	0	1,408
1Z3	4	0	704
1Z4	4	0	704
1X4	1	0	176
1W1	4	0	704
1W2	4	0	704
TOTALS	217	0	38,192

UNIT 1 CYCLE 22

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
AC1	56	0	9,856
AC2	16	0	2,816
AC3	24	0	4,224
AB1	48	0	8,448
AB2	16	0	2,816
AB3	32	0	5,632
AA1	12	0	2,112
AA2	8	0	1,408
2X1	4	0	704
1X1	1	0	176
TOTALS	217	0	38,192

UNIT 1 CYCLE 23

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
AD1	36	0	6336
AD2	20	0	3520
AD3	16	0	2816
AD4	24	0	4224
AC1	56	0	9856
AC2	16	0	2816
AC3	24	0	4224
AB3	24	0	4224
BA5	1	0	176
TOTALS	**217**	**0**	38192

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

PARAMETER	BATCHES A	B	C	D	E	F	G
Active Length, inches	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3795	.3795	.3795	.3765	.3765	.3765	.3765
Pellet Length, inches	.650	.650	.650	.450	.450	.450	.450
Pellet Density, g/cc	10.193	10.193	10.193	10.385	10.385	10.385	10.385
Stack Height Density, g/cc	10.054	10.054	10.054	10.018	10.046	10.046	10.046
Clad ID, inches	.3880	.3880	.3880	.3840	.3840	.3840	.3840
Clad OD, inches	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.026	.026	.026	.028	.028	.028	.028
Diametral Gap, inches	.0085	.0085	.0085	.0075	.0075	.0075	.0075

PARAMETER	BATCHES H	J	K	L	M	MX[(a)]	N	P
Active Length, inches	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3765	.3765	.3765	.3765	.3765	.3700	.3765	.3765
Pellet Length, inches	.450	.450	.450	.450	.450	.425	.450	.450
Pellet Density, g/cc	10.385	10.385	10.385	10.385	10.385	10.302	10.439	10.439
Stack Height Density, g/cc	10.046	10.046	10.046	10.046	10.046	10.180[(b)]	10.100	10.100
Clad ID, inches	.3840	.3840	.3840	.3840	.3840	.378	.3840	.3840
Clad OD, inches	.440	.440	.4400	.440	.440	.440	.440	.440
Clad Thickness, inches	.028	.028	.0280	.028	.028	.031	.028	.028
Diametral Gap, inches	.0075	.0075	.0075	.0075	.0075	.008	.0075	.0075

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

PARAMETER	**BATCHES**						
	R	**RT**	**S**	**T**	**V**	**W**	**X**
Active Length, inches	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3765	.3810	.3765	.3810	.3810	.3810	.3810
Pellet Length, inches	.450	.456	.450	.456	.456	.456	.456
Pellet Density, g/cc	10.439	10.467	10.439	10.467	10.467	10.467	10.467
Stack Height Density, g/cc	10.12	10.31	10.17	10.31	10.31	10.31	Regular 10.32 Annular 7.82
Clad ID, inches	.3840	.3880	.3840	.3880	.3880	.3880	.3880
Clad OD, inches	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.028	.026	.028	.026	.026	.026	.026
Diametral Gap, inches	.0075	.0070	.0075	.0070	.0070	.0070	.0070

PARAMETER	**BATCHES**			
	Z	**AA**	**AB**	**AC**
Active Length, inches	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3810	.3810	0.3805	0.3805
Pellet Length, inches	.456	.456	0.476 (Central) 0.545 (Blanket)	0.476 (Central) 0.545 (Blanket)
Pellet Density, g/cc	10.467	10.467	10.5216	10.5216
Stack Height Density, g/cc	Regular 10.32 Annular 7.82	Regular 10.32 Annular 7.82	10.3743 (UO_2) 10.2277 (4 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)	10.3743 (UO_2) 10.2277 (4 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)
Clad ID, inches	.3880	.3880	0.387	0.387
Clad OD, inches	.440	.440	0.440	0.440
Clad Thickness, inches	.026	.026	0.0265	0.0265
Diametral Gap, inches	.0070	.0070	0.0065	0.0065

TABLE 3.3-1

UNIT 1 BATCH-RELATED DATA

PARAMETER	**BATCHES**
	AD
Active Length, inches	136.7
Pellet Diameter, inches	0.3805
Pellet Length, inches	0.476 (Central) 0.545 (Blanket)
Pellet Density, g/cc	10.5216
Stack Height Density, g/cc	10.3743 (UO_2) 10.2277(4 wt% Gd_2O_3) 10.1565(6 wt% Gd_2O_3) 10.0867(8 wt% Gd_2O_3) 10.3953 (Blanket)
Clad ID, inches	0.387
Clad OD, inches	0.440
Clad Thickness, inches	0.0265
Diametral Gap, inches	0.0065

(a) ANF Demonstration Assemblies.

(b) Pellet envelope includes both UO_2 and Gd_2O_3.

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ASSEMBLY AVERAGE ENRICHMENT wt% U-235	NUMBER OF B_4C SHIMS PER ASSEMBLY	INITIAL SHIM LOADING wt% B_4C	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
2A	2.05	0	---	396	176
2B	2.45	12	2.9	370	164
2C	2.99	0	---	397	176
2C+	2.99	12	1.1	369	164
2C.	2.99	12	.7	369	164
2D	3.03	0	---	388	176
2D/	2.73	0	---	388	176
2E	3.03	0	---	389	176
2E/	2.73	0	---	389	176
2F	3.65	0	---	390	176
2F/	3.03	8	3.03	371	168
2G	4.00	0	---	389	176
2G/	3.55	8	3.03	372	168
2H	4.05	0	---	389	176
2H*	3.40	0	---	388	176
2J	4.05	0	---	390	176
2J*	3.40	0	---	390	176
2K	4.08	0	---	390	176
2K*	4.08	12	4.09	362	164
2K/	4.08	8	4.09	372	168
2L	4.30	0	---	389	176
2LX	4.30	4	4.09	380	172
2L/	4.30	8	4.09	371	168
2L*	4.30	12	4.09	363	164
2LE	3.81	0	---	389	176
2M	4.00	0	---	392	176
2M1	4.00	4	4.09	384	172
2M2	4.00	8	4.09	375	168
2M3	4.00	12	4.09	366	164

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ASSEMBLY AVERAGE ENRICHMENT wt% U-235	NUMBER OF ERBIUM SHIMS PER ASSEMBLY	INITIAL SHIM LOADING wt% Er_2O_3	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
2N0	4.48	0	0	392	176
2N2	4.48	20	1.75	392	176
2N4	4.48	44	1.75	390	176
2N6	4.48	68	1.75	389	176
2NT	4.00	44	1.75	408	176
2P0	4.48	0	0	393	176
2P1	4.48	20	2.00	393	176
2P2	4.48	44	2.00	391	176
2R0	4.48	0	0	393	176
2R1	4.48	20	1.75	392	176
2R2	4.48	44	1.75	391	176
2R3	4.48	68	1.75	390	176
2S0	4.28 (VAP)	0	0	410	176
2S1	4.28 (VAP)	20	1.75	408	176
2S2	4.28 (VAP)	44	1.75	408	176
2S3	4.28 (VAP)	68	1.75	407	176
2T0	4.25 (VAP)	0	0	410	176
2TF	4.26 (FANP)	0	0	412	176
2TW	4.25 (VAP)	0	0	409	176
2T1	4.25 (VAP)	20	2.0	409	176
2T2	4.25 (VAP)	44	2.0	408	176
2T3	4.25 (VAP	68	2.0	407	176

FANP – Framatome Advanced Nuclear Power (AREVA)

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF ZrB_2 SHIMS PER ASSEMBLY	INITIAL SHIM LOADING (mg-B-10/ inch)	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
2V0	2.6/4.1/4.6 (VAP)	0	0	411	176
2V1	2.6/4.1/4.6 (VAP)	44	3.35	408	176
2V2	2.6/4.1/4.6 (VAP)	52	3.35	407	176
2V3	2.6/4.1/4.6 (VAP)	64	3.35	406	176
2V4	2.6/4.1/4.6 (VAP)	76	3.35	405	176
2V5	2.6/4.1/4.6 (VAP)	96	3.35	404	176

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF ZrB_2 SHIMS PER ASSEMBLY	INITIAL SHIM LOADING (mg-B-10/ inch)	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
2W1	2.6/4.50/4.95 (VAP)	28	3.29	408	176
2W2	2.6/4.50/4.95 (VAP)	64	3.29	406	176
2W3	2.6/3.95/4.50/ 4.95 (VAP)	52	3.29	407	176
2W4	2.6/3.95/4.50/ 4.95 (VAP)	64	3.29	406	176
2W5	2.6/3.95/4.50/ 4.95 (VAP)	76	3.29	406	176
2W6	2.6/3.95/4.50/ 4.95 (VAP)	96	3.29	405	176
2X1	4.60/4.20/2.60 (VAP)	28	3.29	409	176
2X2	4.60/4.20/2.60 (VAP)	52	3.29	408	176
2X3	4.95/4.60/4.20/ 2.60 (VAP)	64	3.29	407	176
2X4	4.95/4.60/4.20/ 2.60 (VAP)	76	3.29	407	176
2X5	4.95/4.60/4.20/ 2.60 (VAP)	96	3.29	406	176
2X6	4.20/4.00/2.60 (VAP)	64	3.29	408	176

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
2Z1	4.88/4.34/2.00	0	N/A	410	176
2Z2	4.88/4.34/2.00 (4.40)	4	2	410	176
2Z3	4.88/4.34/2.00 (4.40/3.42)	4/12	2/6	408	176
2Z4	4.88/4.34/2.00 (2.93)	16	8	407	176
2Z5	4.88/4.34/2.00 (2.93)	12	8	408	176

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
BA1	4.60/4.00/2.00	0	N/A	410	176
BA2	4.60/4.00/2.00 (3.60)	4	4	410	176
BA3	4.60/4.00/2.00 (4.00/3.60)	4/12	2/4	409	176
BA4	4.60/4.00/2.00 (3.60/3.20)	4/12	4/6	408	176
BA5	4.60/4.00/2.00 (4.00/3.20)	8/12	2/6	408	176
BA6	4.15/3.60/2.00 (3.20/2.40)	4/8	4/8	408	176
BA7	4.15/3.60/2.00 (3.20/2.60)	4/12	4/6	408	176

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
BB1	4.92/4.32/2.00 (3.60)	8	4	409	176
BB2	4.92/4.32/2.00 (4.40/2.95)	4/4	2/6	409	176
BB3	4.92/4.32/2.00 (3.60/2.95)	4/8	4/6	408	176
BB4	4.92/4.32/2.00 (3.60/2.95)	4/12	4/8	407	176
BB5	4.92/4.32/2.00 (4.40/2.95)	8/12	2/8	407	176

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

BATCH DESIGNATION	INITIAL ENRICHMENTS wt% U-235	NUMBER OF Gd_2O_3 RODS PER ASSEMBLY	INITIAL Gd_2O_3 LOADING wt%	AVERAGE WEIGHT OF URANIUM PER ASSEMBLY Kg U	FUEL RODS PER ASSEMBLY
BC1	4.90/4.30/1.60 (3.60)	8	4	409	176
BC2	4.90/4.30/1.60 (3.60/3.20)	4/4	4/6	409	176
BC3	4.90/4.30/1.60 (3.60/3.20)	4/8	4/6	408	176
BC4	4.90/4.30/1.60 (3.60/2.90)	4/12	4/8	407	176
BC5	4.90/4.30/1.60 (3.60/2.90)	4/12	6/8	407	176

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 1

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2A	69	0	12,144
2B	80	960	13,120
2C	40	0	7,040
2C+	16	192	2,624
2C.	12	144	1,968
TOTALS	217	1,296	36,896

UNIT 2 CYCLE 2

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2B	65	780	10,660
2C	40	0	7,040
2C+	16	192	2,624
2C.	12	144	1,968
2D	48	0	8,448
2D/	36	0	6,336
TOTALS	217	1,116	37,076

UNIT 2 CYCLE 3

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2B	1	12	164
2C	40	0	7,040
2C+	16	192	2,624
2C.	12	144	1,968
2D	48	0	8,448
2D/	36	0	6,336
2E	48	0	8,448
2E/	16	0	2,816
TOTALS	217	348	37,844

UNIT 2 CYCLE 4

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2D	25	0	4,400
2E	48	0	8,448
2E/	16	0	2,816
2F	40	0	7,040
2F/	88	704	14,784
TOTALS	217	704	37,488

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 5

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2D	13	0	2,288
2F	40	0	7,040
2F/	88	704	14,784
2G	48	0	8,448
2G/	28	224	4,704
TOTALS	217	928	37,264

UNIT 2 CYCLE 6

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
1E/	8	0	1,408
2D/	20	0	3,520
2D	1	0	176
2F	40	0	7,040
2G	48	0	8,448
2G/	28	224	4,704
2H	48	0	8,448
2H*	24	0	4,224
TOTALS	217	224	37,968

UNIT 2 CYCLE 7

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	TOTAL SHIMS	TOTAL FUEL RODS
2D	1	0	176
2E	8	0	1,408
2G	48	0	8,448
2G/	28	224	4,704
2H	48	0	8,448
2H*	24	0	4,224
2J	40	0	7,040
2J*	20	0	3,520
TOTALS	217	224	37,968

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 8

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2H*	21	1	3,695
2H	48	7	8,441
2J*	20	0	3,520
2J	40	1	7,039
2K/	44	352	7,392
2K*	28	336	4,592
2K	16	0	2,816
TOTALS	217	697	37,495

Batch 2H* includes one assembly with one SS rod.

Batch 2H includes three assemblies with a total of seven SS rods.

Batch 2J includes one assembly with one SS rod.

UNIT 2 CYCLE 9

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2L	16	0	2,816
2LX	20	80	3,440
2L/	24	192	4,032
2L*	28	336	4,592
2LE	4	0	704
2K	16	0	2,816
2K/	44	355	7,389
2K*	28	339	4,589
2J	36	1	6,335
2H*	1	0	176
TOTALS	217	1,303	36,889

Batch 2K/ includes two assemblies with a total of three SS rods.

Batch 2K* includes one assembly with three SS rods.

Batch 2J includes one assembly with one SS rod.

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 10

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2M	12	0	2,112
2M1	16	64	2,752
2M2	20	160	3,360
2M3	40	480	6,560
2L[a]	16	0	2,816
2LX[a]	20	80	3,440
2L/[a]	24	192	4,032
2L*[a]	28	336	4,592
2LE[a,b]	4	0	704
2K[a]	16	0	2,808[f]
2K/[a]	16	128	2,688
2J*[c]	4	0	704
2H*[d]	1	0	174[e]
TOTALS	217	1,440	36,742

---

[a] Carried over from Unit 2, Cycle 9.

[b] Erbium demonstration assembly.

[c] Reinserted, discharged at End of Unit 2, Cycle 8.

[d] Reinserted, discharged at End of Unit 2, Cycle 7.

[e] The center assembly contains two SS replacement rods.

[f] Eight fuel rods were replaced by SS replacement rods in Batch 2K during the Cycle 9 to Cycle 10 refueling outage.

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 11

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
N0	12	0	2,112
N2	8	0	1,408
N4	16	0	2,816
N6	48	0	8,448
NT	4	0	704
M	12	0	2,112
M1	16	64	2,752
M2	20	160	3,360
M3	40	480	6,560
L	16	0	2,816
LX	12	48	2,064
LT	4	0	704
J	4	0	704
L	4	0	704
J*[a]	1	2	174
TOTALS	217	754	37,438

[a] Batch J* includes one assembly with a total of two SS rods.

UNIT 2 CYCLE 12

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
P0	24	0	4,224
P1	8	0	1,408
P2	60	0	10,560
N0	12	0	2,112
N2	8	0	1,408
N4	16	0	2,186
N6	48	0	8,448
NT	4	0	704
M	12	0	2,112
M1	16	64	2,752
N	4	0	704
M*	4	48	656
K1	1	0	176
TOTALS	217	112	38,080

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 13

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
R0	8	0	1,408
R1	12	0	2,112
R2	32	0	5,632
R3	40	0	7,040
P0	24	0	4,224
P1	8	0	1,408
P2	60	0	10,560
N0	12	0	2,112
N2	8	0	1,408
N4	4	0	704
K*	4	48	656
J*	5	0	880
TOTALS	217	48	38,144

UNIT 2 CYCLE 14

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2S0	8	0	1,408
2S1	24	0	4,224
2S2	24	0	4,224
2S3	36	0	6,336
2R0	8	0	1,408
2R1	12	0	2,113
2R2	32	0	5,632
2R3	40	0	7,040
2P0	23	0	4,048
2P1	8	0	1,408
1L*	1	0	176
1RT	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 15

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
T0	20	0	3,520
T1	4	0	704
T2	40	0	7,040
T3	20	0	3,520
TF	4	0	704
TW	4	0	704
S0	8	0	1,408
S1	24	0	4,224
S2	24	0	4,224
S3	36	0	6,336
R0	8	0	1,408
R1	12	0	2,112
R2	12	0	2,112
J*	1	0	176
TOTALS	217	0	38,192

UNIT 2 CYCLE 16

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2V0	12	0	2,112
2V1	28	0	4,928
2V2	4	0	704
2V3	16	0	2,816
2V4	12	0	2,112
2V5	20	0	3,520
2T0	20	0	3,520
2T1	4	0	704
2T2	40	0	7,040
2T3	20	0	3,520
2TF	4	0	704
2TW	4	0	704
2S0	8	0	1,408
2S1	24	0	4,224
1L*	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 17

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
W1	20	0	3,520
W2	4	0	704
W3	8	0	1,408
W4	12	0	2,112
W5	36	0	6,336
W6	16	0	2,816
V0	12	0	2,112
V1	28	0	4,928
V2	4	0	704
V3	16	0	2,816
V4	9	0	1,584
V5	20	0	3,520
1V0	2	0	352
1V1	2	0	352
T0	12	0	2,112
T2	8	0	1,408
T3	6	0	1,056
TW	2	0	352
TOTALS	217	0	38,192

UNIT 2 CYCLE 18

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2X1	12	0	2,112
2X2	12	0	2,112
2X3	28	0	4,928
2X4	8	0	1,408
2X5	32	0	5,632
2X6	4	0	704
2W1	20	0	3,520
2W2	4	0	704
2W3	8	0	1,408
2W4	12	0	2,112
2W5	35	0	6,160
2W6	16	0	2,816
2V0	11	0	1,936
2V1	13	0	2,288
1X4	1	0	176
1X7	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 19

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
2Z1	8	0	1,408
2Z2	12	0	2,112
2Z3	28	0	4,928
2Z4	44	0	7,744
2Z5	4	0	704
		0	
2X1	12		2,112
2X2	12	0	2,112
2X3	28	0	4,928
2X4	8	0	1,408
2X5	32	0	5,632
2X6	4	0	704
2W1	12	0	2,112
2W2	4	0	704
2W4	7	0	1,232
2V1	1	0	176
2V4	1	0	176
TOTALS	217	0	38,192

UNIT 2 CYCLE 20

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
BA1	12	0	2,112
BA2	8	0	1,408
BA3	32	0	5,632
BA4	12	0	2,112
BA5	20	0	3,520
BA6	4	0	704
BA7	12	0	2,112
2Z1	8	0	1,408
2Z2	12	0	2,112
2Z3	24	0	4,224
2Z4	44	0	7,744
2Z5	4	0	704
2X1	8	0	1,408
2X2	8	0	1,408
2X3	8	0	1,408
1X1	1	0	176
TOTALS	217	0	38,192

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

UNIT 2 CYCLE 21

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
BB1	12	0	2,112
BB2	8	0	1,408
BB3	20	0	3,520
BB4	28	0	4,928
BB5	28	0	4,928
BA1	12	0	2,112
BA2	8	0	1,408
BA3	32	0	5,632
BA4	12	0	2,112
BA5	17	0	2,992
BA6	4	0	704
BA7	12	0	2,112
2Z1	8	0	1,408
2Z2	8	0	1,408
2Z3	8	0	1,408
TOTALS	217	0	38,192

UNIT 2 CYCLE 22

ASSEMBLY BATCH	NUMBER OF ASSEMBLIES	NON-FUEL RODS	TOTAL FUEL RODS
BC1	12	0	2,112
BC2	8	0	1,408
BC3	20	0	3,520
BC4	20	0	3,520
BC5	36	0	6,336
BB1	12	0	2,112
BB2	8	0	1,408
BB3	20	0	3,520
BB4	28	0	4,928
BB5	24	0	4,224
BA1	12	0	2,112
BA2	8	0	1,408
BA3	4	0	704
BA5	1	0	176
2Z3	4	0	704
TOTALS	217	0	38,192

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

PARAMETER	**BATCHES**							
	A	**B**	**C**	**D**	**E**	**F**	**G**	**H**
Active Length, inches	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3805	.3805	.3805	.3765	.3765	.3765	.3765	.3765
Pellet Length, inches	.450	.450	.450	.450	.450	.450	.450	.450
Pellet Density, g/cc	10.412	10.412	10.412	10.385	10.385	10.385	10.385	10.385
Stack Height Density, g/cc	10.039	10.043	10.039	10.046	10.046	10.046	10.046	10.046
Clad ID, inches	.3880	.3880	.3880	.3840	.3840	.3840	.3840	.3840
Clad OD, inches	.440	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.026	.026	.026	.028	.028	.028	.028	.028
Diametral Gap, inches	.0075	.0075	.0075	.0075	.0075	.0075	.0075	.0075

PARAMETER	**BATCHES**							
	J	**K**	**L**	**M**	**N**	**NT**	**P**	**R**
Active Length, inches	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3765	.3765	.3765	.3765	.3765	.3810	.3765	.3765
Pellet Length, inches	.450	.450	.450	.450	.450	.456	.450	.450
Pellet Density, g/cc	10.385	10.385	10.385	10.439	10.439	10.467	10.439	10.439
Stack Height Density, g/cc	10.046	10.046	10.046	10.100	10.12	10.31	10.12	10.12
Clad ID, inches	.3840	.3840	.3840	.3840	.384	.388	.384	.384
Clad OD, inches	.440	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.028	.028	.028	.028	.028	.026	.028	.028
Diametral Gap, inches	.0075	.0075	.0075	.0075	.0075	.0070	.0075	.0075

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

PARAMETER	**BATCHES**				
	S	**T(Westinghouse)**	**T(FANP)**	**V**	**W**
Active Length, inches	136.7	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3810	.3810	.3805	.3810	.3810
Pellet Length, inches	.456	.456	.435	.456	.456
Pellet Density, g/cc	10.467	10.467	10.522	10.467	10.467
Stack Height Density, g/cc	10.31	10.31	10.39	Regular 10.32 Annular 7.82	Regular 10.32 Annular 7.82
Clad ID, inches	.3880	.3880	.3870	.3880	.3880
Clad OD, inches	.440	.440	.440	.440	.440
Clad Thickness, inches	.026	.026	.0265	.026	.026
Diametral Gap, inches	.0070	.0070	.0065	.0070	.0070

PARAMETER	**BATCHES**			
	X	**Z**	**BA**	**BB**
Active Length, inches	136.7	136.7	136.7	136.7
Pellet Diameter, inches	.3810	0.3805	0.3805	0.3805
Pellet Length, inches	.456	0.476 (Central) 0.545 (Blanket)	0.476 (Central) 0.545 (Blanket)	0.476 (Central) 0.545 (Blanket)
Pellet Density, g/cc	10.467	10.5216	10.5216	10.5216
Stack Height Density, g/cc	Regular 10.32 Annular 7.82	10.3743 (UO_2) 10.3003 (2 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)	10.3743 (UO_2) 10.3003 (2 wt% Gd_2O_3) 10.2277 (4 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)	10.3743 (UO_2) 10.3003 (2 wt% Gd_2O_3) 10.2277 (4 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)
Clad ID, inches	.3880	0.3870	0.3870	0.3870
Clad OD, inches	.440	0.440	0.440	0.440
Clad Thickness, inches	.026	0.0265	0.0265	0.0265
Diametral Gap, inches	.0070	0.0065	0.0065	0.0065

TABLE 3.3-2

UNIT 2 BATCH-RELATED DATA

PARAMETER	BATCHES BC
Active Length, inches	136.7
Pellet Diameter, inches	0.3805
Pellet Length, inches	0.476 (Central) 0.545 (Blanket)
Pellet Density, g/cc	10.5216
Stack Height Density, g/cc	10.3743 (UO_2) 10.2277 (4 wt% Gd_2O_3) 10.1565 (6 wt% Gd_2O_3) 10.0867 (8 wt% Gd_2O_3) 10.3953 (Blanket)
Clad ID, inches	0.387
Clad OD, inches	0.440
Clad Thickness, inches	0.0265
Diametral Gap, inches	0.0065

TABLE 3.3-3

BURNABLE POISON ROD DATA

UNITS 1 AND 2

BATCH	B,C+,C.	1G/,1H/ 2F/,2G/	2K/,2K*,1NX, 1N/,1M*,2L*, 2L/,2LX	1MX	2LE	2M	1P	2N	2NT	2P
Active Length	122.7	122.7	122.7	122.7	115.7	115.7	108.7	112.7	112.7	112.2
Pellet Diameter	.376	.362	.362	.370	.3765	.362	.3765	.3765	.3810	.3765
Clad ID	.388	.384	.384	.378	.384	.384	.384	.384	.388	.384
Clad OD	.440	.440	.440	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.026	.028	.028	.031	.028	.028	.028	.028	.026	.028
Diametral Gap, inches	.012	.022	.022	.008	.0075	.022	.0075	.0075	.0070	.0075

BATCH	1R	1RT	2R	1S	1T	2S	1V	2T	1W	2V
Active Length	114.2	114.2	112.7	114.2	112.2	112.7	112.2	112.2	114.2	114.7
Pellet Diameter	.3765	.3810	.3765	.3765	.3810	.3810	.3810	.3810	.3810	.3810
Clad ID	.384	.388	.384	.384	.388	.388	.388	.388	.3880	.388
Clad OD	.440	.440	.440	.440	.440	.440	.440	.440	.440	.440
Clad Thickness, inches	.028	.026	.028	.028	.026	.026	.026	.026	.026	.026
Diametral Gap, inches	.0075	.0070	.0075	.0075	.0070	.007	.007	.007	.0070	.0070

BATCH	1X	2W	1Z	2X	AA	2Z	AB	BA	AC
Active Length	116.7	116.7	116.7	116.7	116.7	112.7	112.7	112.7	112.7
Pellet Diameter	.3810	.3810	.3810	.3810	.3810	0.3805	0.3805	0.3805	0.3805
Clad ID	.388	.388	.388	.388	.388	0.3870	0.387	0.387	0.387
Clad OD	.440	.440	.440	.440	.440	0.440	0.440	0.440	0.440
Clad Thickness, inches	.026	.026	.026	.026	.026	0.0265	0.0265	0.0265	0.0265
Diametral Gap, inches	.0070	.0070	.0070	.0070	.0070	0.0065	0.0065	0.0065	0.0065

TABLE 3.3-3

BURNABLE POISON ROD DATA

UNITS 1 AND 2

BATCH	BB	AD	BC
Active Length	112.7	112.7	112.7
Pellet Diameter	0.3805	0.3805	0.3805
Clad ID	0.387	0.387	0.387
Clad OD	0.440	0.440	0.440
Clad Thickness, inches	0.0265	0.0265	0.0265
Diametral Gap, inches	0.0065	0.0065	0.0065

All dimensions in inches.

Batch 1 MX is ANF demonstration fuel with 12 Gd_2O_3 (10 wt%) fuel bearing (natural uranium) poison rods per assembly.

Batch 2LE is Erbium bearing demonstration fuel with 44 Er_2O_3 (0.9 wt%) fuel bearing (3.40 wt% U-235) rods per assembly.

TABLE 3.3-4

CONTROL ELEMENT ASSEMBLY DATA

UNITS 1 AND 2

All Dimensions are Nominal and are in inches

CEA TYPE	**FLCEA8** Non-reconstitutable	**FLCEA7** Non-reconstitutable	**FLCEA10** Non-reconstitutable	**FLCEA9** Non-reconstitutable
Number	76(1)	1(2)	76(1)	1(2)
Clad Thickness	0.040	0.040	0.040	0.040
Clad OD	0.948	0.948	0.948	0.948
Diametral Gap B4C/UAIC/LAIC/USS/LSS	.008(3)	.008(3)	.008/.012/.017/NA/NA	.008/.012/.017/.118/.012
Corner Element Pitch	4.64	4.64	4.64	4.64
Pellet Type	B4C/AIC	AL2O3/SS/B4C/AIC	B4C/UAIC/LAIC	B4C/UAIC/LAIC/USS/LSS
Pellet Diameter	0.86/0.86	0.85/0.86/0.86/0.86	0.86/0.856/0.851	0.86/0.856/0.851/0.75/0.856

Note (1) up to 76 FLCEA1, FLCEA2, FLCEA8 or FLCEA10
Note (2) up to 1 of FLCEA7 or FLCEA9
Note (3) diametral gap is .008 regardless of pellet type
AIC: Ag-In-Cd (Silver-Indium-Cadmium)
UAIC: Upper AIC
LAIC: Lower AIC
SS: Stainless Steel
USS: Upper SS
LSS: Lower SS

TABLE 3.3-4

CONTROL ELEMENT ASSEMBLY DATA

UNITS 1 AND 2

GUIDE TUBE FLUX SUPPRESSOR DATA

UNIT 1 (CYCLES 11 AND 12)

PARAMETER	GTFS
Number	24(U1)
Clad Thickness	.040
Clad OD	.948
Diametral Gap	.008
Pellet Type	Al_2O_3/B_4C
Pellet Diameter	.85/.86

NOTE: All dimensions in inches except where noted.

TABLE 3.3-5

CORE RELATED DATA

UNIT 1 AND UNIT 2

CORE ARRANGEMENT

Number of Fuel Assemblies in Core, Total	217
Number of CEAs	77
Total Number of Fuel Rods and Non-Fuel Rods	38,192
CEA Pitch, min, inches	11.57
Spacing Between Fuel Assemblies, Fuel Rod Surface to Surface, inches	.20
Spacing, Outer Fuel Rod Surface to Core Shroud, inches	.204
Hydraulic Diameter, Nominal Channel, feet	.044
Total Flow Area (Excluding Guide Tubes), ft^2	53.5
Total Core Cross-section Area, ft^2	101.1
Core Equivalent Diameter, inches	136
Core Circumscribed Diameter, inches	143.3
Core Volume, ft^3	1151

3.4 NUCLEAR DESIGN AND EVALUATION

3.4.1 SUMMARY

This section summarizes the nuclear characteristics of the core and discusses the important design parameters which are of significance to the performance of the core during transient and steady state operation. A discussion of the nuclear design methods employed and comparisons with experiments which support the use of these methods is included. Summaries of nuclear parameters are presented in Table 3.4-1. Design limits for shutdown margin, reactivity coefficients, reactivity insertion rates, and power distribution are discussed in the appropriate sections.

Fuel enrichment and BPR distributions are shown in Table 3.3-1, Table 3.3-2, and Figure 3.3-4.

Physical features of the lattice, fuel assemblies, and CEAs are described in Section 3.3.2. The soluble boron insertion rates are sufficient to compensate for the maximum reactivity addition due to xenon burnout and normal plant cooldown.

3.4.2 REACTIVITY AND CONTROL REQUIREMENTS

At the beginning of each cycle, the core is loaded with sufficient fuel to generate essentially full power for the cycle length. This results in built-in excess reactivity (the reactivity present in the reactor with all control material withdrawn from the core) which must be sufficient to compensate for the reactivity lost during the cycle due to:

a. Fuel burnup;
b. Fission product buildup; and,
c. Negative reactivity feedbacks.

The excess reactivity must be stable and controlled through the cycle to permit power operations while maintaining the ability to rapidly shut down the reactor if necessary (Table 3.4-2).

Excess reactivity is controlled during the cycle by adjusting both the position of the CEAs and the concentration of boric acid dissolved in the RCS. The CEAs permit rapid changes in reactivity, as required for reactor trip, and may be used to compensate for changes in moderator temperature, fuel temperature, and moderator density associated with changes in power level.

Adjustment of the boric acid concentration is used to control the relatively slow reactivity changes associated with plant heatup and cooldown, fuel burnup, certain xenon variations, and slow power level changes. The use of boric acid dissolved in the reactor coolant makes it possible to maintain the CEAs in an essentially fully-withdrawn position during full power operation, thus minimizing distortions in power distribution. Although the boric acid system reduces reactivity relatively slowly, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of normal cooldown and xenon decay. Table 3.4-1 lists the predicted concentrations of natural boron required to maintain the first cycle critical under various conditions, assuming all CEAs to be fully withdrawn. The hot full power, equilibrium xenon, BOC boron concentration predictions for the present cycles are also given in Table 3.4-1.

Design criteria require reactivity control and stability. In a stable reactor, a reactivity perturbation during steady state operation leads to another steady state. This desirable, self-limiting characteristic is due to negative feedback of reactivity. The parameters used to quantify feedback are the reactivity coefficients which relate changes in core reactivity to variations in fuel and/or moderator conditions. Verifications of predicted nuclear

parameters are conducted at the beginning of each cycle during startup testing (Chapter 13). Results of these tests verify prediction of the Moderator Temperature Coefficient. For each DBE in the Safety Analysis (Chapter 14), suitably conservative reactivity coefficient values are used. Values assumed in the transient analyses are listed in Chapter 14.

3.4.2.1 Fuel Temperature Coefficient

The FTC reflects the change of core reactivity per degree change in average fuel temperature. A change in fuel temperature affects the density of the fuel pellet and the nuclear density of the uranium in the pellet, thus changing the probability of interaction with a neutron. A fuel temperature change also modifies the reaction rates in uranium in both the thermal and epithermal neutron energy regimes.

The Doppler effect is the principal contributor to the change in reaction rate with fuel temperature in the epithermal range. This effect results from the increased (non-fissioning) neutron absorption in U-238 with increasing fuel temperature. This increase in neutron absorption rate with fuel temperature causes a negative FTC since the temperature increase decreases the fission rate. In the thermal energy regime, a change in reaction rate with fuel temperature arises from the effect of temperature dependent scattering properties of the fuel matrix on the thermal neutron spectrum. In typical PWR fuels containing strong resonance absorbers such as U-238 and Pu-240, the component of the FTC arising from the Doppler effect is more than a factor of ten larger than the thermal energy component.

The variation of FTC with temperature predicted for the first cycle is shown in Figure 3.4-1. The predicted first cycle hot, full power FTC was -1.06×10^{-5} $\Delta\rho/°F$ which is approximately equivalent to -1.49×10^{-3} $\Delta\rho/(kW/ft)$.

3.4.2.2 Moderator Temperature Coefficient

The MTC relates changes in reactivity to changes in the moderator average temperature and includes the effects of temperature on the moderator density.

Typically, an increase in the moderator temperature causes a decrease in moderator density and therefore less neutron thermalization which reduces core reactivity. When an increase in moderator temperature causes a decrease in reactivity, the core has a negative temperature coefficient. This adds to the stability since a temperature increase will reduce reactor power and vice versa.

When sufficient boron is present in the moderator, a reduction in moderator density also causes a reduction in the boron density in the core, thus producing a positive contribution to the MTC. One core design objective is to limit the MTC to values near zero, if positive, or to slightly negative values (the MTC positive limits are listed in the Technical Specifications). In order to limit the dissolved boron concentration and its positive contribution to the MTC, BPRs (shims) may be provided in the cycle design (Tables 3.3-1 and 3.3-2). The reactivity control provided by the shims makes possible a reduction in the dissolved boron concentration and therefore a reduction in the MTC.

Moderator Temperature Coefficient values for various core conditions during the first cycle and for the current cycle are given in Table 3.4-1. As shown in the table, the least negative value at full power conditions occurs in the unrodded core when the dissolved boron content is at its maximum. The MTC becomes more negative at the EOC due mainly to the reduction in the dissolved boron content with burnup.

CEA insertion provides a negative contribution to the coefficient since a corresponding reduction in the dissolved boron content is required to maintain the reactor critical.

The effects of plutonium and fission products on the MTC are small when compared to the effects of dissolved boron changes. The buildup of fission product xenon supplies a positive contribution to the MTC for a constant boron concentration. However, when the dissolved boron concentration is reduced by the reactivity equivalent of the xenon, MTC becomes more negative.

The change in MTC as a function of boron concentration is almost linear and was about $+0.16\times10^{-4}$ $\Delta\rho$/F per 100 ppm soluble boron for the first cycle.

3.4.2.3 Moderator Pressure Coefficient

The Moderator Pressure Coefficient is the change in reactivity per unit change in RCS pressure. Since an increase in pressure increases the water density, the pressure coefficient is opposite in sign to the temperature coefficient. The reactivity effect of increasing the pressure is reduced in the presence of dissolved boron because an increase in coolant density also increases the boron density. The Moderator Pressure Coefficient decreases as the RCS boron concentration increases. The calculated pressure coefficients for the beginning and end of the first cycle at full power were $+0.3\times10^{-6}$ $\Delta\rho$/psi and $+2.6\times10^{-6}$ $\Delta\rho$/psi, respectively. The Moderator Pressure Coefficient was measured to be $-.5\times10^{-6}$ $\Delta\rho$/psi at the beginning of Unit 1 Cycle 1 with a nominal RCS temperature of 450°F by changing pressure in the 1100 psia to 2250 psia range. The pressure coefficient of reactivity is relatively insignificant and is several orders of magnitude smaller than the MTC.

3.4.2.4 Moderator Void Coefficient

The occurrence of small amounts of local subcooled boiling in the reactor during full power operation may result in small steam bubbles (voids). These are called voids because they contain almost no moderating nuclei. The average void fraction is the fraction by volume of the moderator that is in void form and it is substantially less than 1% at normal operating conditions. The change in reactivity associated with these voids in the moderator is the Void Coefficient of Reactivity. An increase in voids reduces the moderator density and decreases core reactivity. The presence of soluble boron tends to add a positive contribution to the void coefficient because an increase in voids results in a reduction in boron density in the core. The calculated values at the beginning and at the end of the first cycle were -0.1×10^{-3} $\Delta\rho$/% void and -1.3×10^{-3} $\Delta\rho$/% void, respectively.

3.4.2.5 Power Coefficient

The Power Coefficient is the change in core reactivity per percent change incore power level. Although all of the previously mentioned coefficients (FTC, MTC, Moderator Pressure Coefficient and the Moderator Void Coefficient) contribute to the Power Coefficient, only the MTC and the FTC are significant. To determine the change in reactivity with power, it is necessary to know the change in the average moderator temperature and effective fuel temperature with power.

The average moderator temperature is a linear function of power. The effective fuel temperature is dependent on both power level and burnup. Due to fuel pellet cracking and fission gas release with irradiation, this functional relationship changes during the cycle.

The Power Coefficient can be obtained from the following equation:

$$\frac{dRho}{dP} = \left(\frac{dRho}{dT_f}\ x\ \frac{dT_f}{dP}\right) + \left(\frac{dRho}{dT_m}\ x\ \frac{dT_m}{dP}\right)$$

The first term of the equation provides the fuel temperature contribution to the Power Coefficient. The first term is the product of FTC of reactivity and the effective change of fuel temperature with respect to power. The second term in the equation provides the moderator contribution to the Power Coefficient. The first factor is the MTC and the second factor is a constant since the moderator temperature is a linear function of power.

Since the factors $dRho/dT_f$ and $dRho/dT_m$ are functions of one or more independent variables (e.g., burnup, temperature, soluble boron content, xenon worth, and CEA insertion), the total Power Coefficient, dRho/dP, also depends on these variables.

Plots of the calculated FTC and a plot of the predicted Power Coefficient for the beginning of the first cycle are shown in Figures 3.4-1 and 3.4-2, respectively. The full power value of the Power Coefficient for the unrodded first cycle is $-1.49 \times 10^{-3}\ \Delta\rho/(kW/ft)$. The Power Coefficient becomes more negative with burnup due to the increasing negative FTC and MTC.

3.4.3 SHUTDOWN REACTIVITY CONTROL

The reactivity worth requirement of all the CEAs is determined by:

a. Shutdown reactivity margin;
b. Power defect (including moderator voids); and,
c. CEA bite.

The total worth of all CEAs provides adequate shutdown at full power even if the CEA with the most worth is stuck in the fully withdrawn position. Table 3.4-2 compares available CEA reactivity with the various required reactivity components at BOC and EOC for the first cycle. The table also lists the current cycle's most limiting values of reactivity worths and allowances. Individual components are discussed below.

3.4.3.1 Shutdown Reactivity Margin

The shutdown margin requirement is based on the reactivity requirements for the most limiting postulated accident. This requirement varies throughout core life as a function of fuel depletion, boron concentration, and RCS temperature.

Sufficient CEA worth must be available for rapid insertion to ensure that:

a. The reactor can be made subcritical from all operating conditions.
b. Reactivity transients associated with postulated accident conditions are controllable.

A steam line break or excess load event (Chapter 14) at EOC requires the maximum CEA shutdown margin due to the large, rapid cooldown and the large, negative MTC. These accidents provide the basis for the shutdown margin Technical Specification with RCS temperature above 200°F.

Allowances of 2.0% and 2.4% $\Delta\rho$ at the beginning of first cycle and at the end of first cycle, respectively, were made for the predicted shutdown margin and safety

feature allowances at hot, zero power conditions. The current cycle values of CEA worths and allowances for the limiting event are listed in Table 3.4-2.

3.4.3.2 Power Defect

The power defect is the reactivity change in the core from hot zero power to a higher power. During a reactor trip, this increase in reactivity must be compensated for by the CEAs.

The power defect increases as the power level is increased and results from the following changes:

a. Fuel Temperature Variation
b. Moderator Temperature Variation
c. Moderator Voids Variation

These three reactivity variations are described below.

a. Fuel Temperature Variation

The reactivity increase that occurs when the fuel temperature decreases from its full (or other) power value to its zero power value is primarily due to changes in epithermal absorption resonance's (the Doppler effect) in U-238. As exposure accumulates, concentrations of plutonium increase making the FTC more negative. However, pellet swelling and clad creepdown associated with increasing exposure improves the heat transfer which decreases the fuel temperature. The competing effects of a more negative FTC and decreasing fuel temperature tends to minimize the burnup dependence of the fuel temperature defect. The Unit 1 Cycle 1 reactivity increase associated with a fuel temperature decrease from full power to zero power was 1.7% $\Delta\rho$ at BOC and 1.8% $\Delta\rho$ at EOC.

b. Moderator Temperature Variation

The average reactor coolant temperature increases with increasing power. This decreases the moderator density and causes a reactivity change which is usually negative (except at very high boron concentrations). The moderator temperature variation allowance is large enough to compensate for any reactivity increase that may occur when the moderator temperature decreases from full power to zero power. This reactivity increase, which is primarily due to the negative MTC, is largest at the EOC when the soluble boron concentration is near zero and the moderator coefficient is strongly negative. At BOC, when the MTC is less negative, the reactivity change associated with the moderator temperature change is smaller.

c. Moderator Voids Variation

Increasing the power level causes a decrease in reactivity resulting from formation of small steam bubbles (voids) due to local boiling. The average void content in the core is very small and is estimated to be less than 1% at full power. As with the moderator temperature effect, the maximum increase in reactivity from full to zero power occurs at EOC when the least amount of dissolved boron is present. At BOC the void coefficient is essentially zero.

3.4.3.3 Control Element Assembly Bite and Power Dependent Insertion Limits

Control element assembly bite is the CEA reactivity worth permitted to be inserted in the core when critical for power shaping and to compensate for minor variations

in moderator temperature, boron concentration, xenon concentration, and power level.

The substantially smaller power defects for shutdown initiated at lower power levels allow a reduction in the CEA worth required to be available for shutdown. The corresponding increase in allowed CEA insertion is reflected by the transient Power Dependent Insertion Limits (PDILs) of the Technical Specifications. The transient PDIL restricts the amount of CEA insertion into the core while at power. The amount of CEA insertion permitted by PDIL provides sufficient reactivity for control while ensuring the minimum shutdown requirement is maintained in the withdrawn CEAs.

The allowance for first cycle CEA bite was 0.1% $\Delta\rho$. In addition, 0.1% $\Delta\rho$ was allowed for the first cycle for compensating fuel depletion effects between adjustments of the dissolved boron concentration. The current allowances for CEA bite are reflected in the Technical Specification transient PDIL curves.

3.4.3.4 Shutdown Conditions

Boric acid is used to provide a large margin for shutdown and refueling. After a normal shutdown or reactor trip, boric acid is injected into the RCS to compensate for reactivity increases caused by normal cooldown and xenon decay. Although the boric acid system reduces reactivity slowly, compared to CEAs, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of cooldown and xenon decay. The boron concentration established for refueling is listed in the Technical Specifications. This boron concentration provides more than adequate negative reactivity to maintain the shutdown condition.

3.4.4 CONTROL ELEMENT ASSEMBLY PATTERN, OPERATIONS, AND WORTHS

The CEA is described in Section 3.3.2.4 and shown in Figures 3.3-8, 3.3-9A and 3.3-9B. The CEAs are designated as Regulating CEAs or Shutdown CEAs. The Regulating CEAs are divided into five groups (1 through 5). The Shutdown CEAs are divided into three groups (A, B, and C). The locations of all the CEAs in one of four symmetrical core quadrants are shown on Figure 3.3-10.

All CEAs within a particular group are designated to be withdrawn or inserted nearly simultaneously. For startups the Shutdown CEA groups are withdrawn without overlap; then the Regulating groups are withdrawn with overlap.

The PDILs specify the maximum permitted CEA insertion as discussed in Section 3.4.3.3. The PDIL curve for the current cycle of each Unit is in the Technical Specifications. The PDIL curve illustrates the Regulating CEA Groups insertion order (5-4-3-2-1) with overlap between successive groups. The typical CEA withdrawal procedure is as follows:

a. With the reactor subcritical, Shutdown Group A is fully withdrawn, followed by Shutdown Group B and then Shutdown Group C.
b. Withdrawal of regulating CEAs commences starting with Regulating Group No. 1. CEA withdrawal continues with the prescribed overlap of Regulating Groups 2 through 5.

All CEAs are inserted for cold shutdown conditions and are essentially fully withdrawn during full power steady state operation. Reactivity insertion rates are discussed in Section 3.4.5. The allowable CEA misalignment within any CEA group is specified in Technical Specifications and intentional misalignment of CEAs within a group for the purpose of power shaping is not allowed.

The accidents involving CEAs are analyzed with conservative assumptions as discussed in Chapter 14. The CEA withdrawal accident is analyzed with the maximum calculated differential reactivity insertion rate resulting from a sequential CEA bank withdrawal with overlap. The CEA drop accident is analyzed by selecting the dropped CEA that maximizes the increase in the radial peaking factor. The typical reactivity insertion during a reactor scram is presented in Chapter 14. This reactivity insertion is computed by static or space time axial models and is used for all accidents which are terminated by a scram.

3.4.5 REACTIVITY INSERTION RATES

Normal operating practices require reactivity changes to accommodate power level changes, fuel depletion, temperature control, etc. The principal reactivity control mechanisms are CEA Regulating Groups and boration/dilution. The analysis of CEA withdrawal events (Chapter 14) shows that maximum CEDM speed results in a differential reactivity per inch and consequence which will not exceed the Specified Acceptable Fuel Design Limits (SAFDLs). The analysis of the boron dilution event (Chapter 14) for the worst case of initial refueling in the drained-down mode shows that adequate time exists to take corrective measures.

Reactivity addition rates due to control rods vary with the CEA group, CEA group position, RCS temperature, dissolved boron concentration, fuel depletion, power level, and power distribution. Reactivity addition rates due to changes in dissolved boron vary with CEA insertion, temperature, fuel depletion, and power level due primarily to their effect on the dissolved boron concentration. Both spectral and spatial self-shielding effects are involved.

3.4.6 POWER DISTRIBUTION

3.4.6.1 <u>General</u>

The core is designed and the reactor is operated to maintain a relatively uniform power distribution. Significant deviations from the expected power distribution are restricted by the Limiting Safety System Settings (LSSSs) [e.g., Axial Shape Index (ASI)] and the Limiting Conditions for Operation (LCOs) [e.g., F_r^T, Peak Linear Heat Rate (PLHR) and T_q]. These operating limits are bounded by sufficient thermal margin to prevent DNB and fuel/clad melting during Anticipated Operational Occurrences (AOOs).

3.4.6.2 <u>Objective</u>

A stable uniform power distribution increases thermal margin by minimizing peak heat flux and enthalpy rise. The relative power distribution is approximately the ratio of the maximum local power to the core average power. The power peaking factors (F_r^T, ASI, and T_q) are minimized by design to allow operational flexibility to deviate from the nominal core power distribution without encroaching upon the LSSSs and LCOs.

3.4.6.3 <u>Fuel Management and Operations</u>

The peaking factors are most strongly influenced by the core loading which is therefore designed to reduce the inherent power peaking. To accomplish this goal in reload core designs, the arrangement of fresh and depleted assemblies together increases the power sharing. Reload batches may have multiple enrichments. Fuel assemblies located in a low neutron leakage or high power density region may have BPRs. Figures 3.4-3 through 3.4-5 show the assembly location for the first and current cycles. The expected power distributions at selected burnups and CEA insertions for the first and current cycles are shown in Figures 3.4-6 through

3.4-22. Figures 3.4-23 through 3.4-26 show the effect of CEA insertion on power peaking and distribution for the first cycle.

The radial power distribution and fuel depletion are almost insensitive to reactor operations due to the high order of radial symmetry maintained and the negative radial and azimuthal stability factor. The APD, however, is subject to nonuniform axial temperatures, burnups, control rods, and xenon oscillations. Prudent use of CEAs will control undesirable axial power oscillations and the PDILs (first cycle shown in Figure 3.4-27) on CEA position prevent the expected burnup distribution from being negated by excessive CEA use.

3.4.6.4 Power Peaking Limits

Specified Acceptable Fuel Design Limits require that the core power distribution does not result in either fuel/clad melt or a DNB. Assurance that SAFDLs are not exceeded is obtained through the RPS and the Technical Specifications which enforce LCO. Limiting Conditions for Operation are established such that the initial conditions assumed in the analysis of AOOs and postulated accidents are conservative with respect to allowed reactor conditions. However, during certain AOOs the margin to fuel design limits deteriorates either in a manner that is undetected by the RPS or in such a manner that the RPS would not act before some margin loss has occurred. Therefore, the reactor must be operated such that losses in margin do not result in exceeding SAFDLs before the RPS restores the reactor to a safe condition. Limiting Conditions for Operation assure that sufficient initial margin to overcome margin losses exists.

3.4.6.5 Power Distribution Monitoring Capability

Neutron flux detectors are provided both within the active core (incore) and outside (excore) the reactor vessel. The RPS continuously monitors the excore detectors and reactor coolant variables to determine whether the plant is being operated within the LSSSs. Indicators are provided to solicit operator action prior to reaching an LSSS. In the event that conditions reach an LSSS, the RPS will automatically initiate a reactor trip.

Incore detectors provide the detailed power distributions necessary for Technical Specification surveillance of power peaks and core data trends.

The 35 incore self-powered rhodium detector strings are placed in the center CEA guide tube of selected assemblies. Each detector string has four, 40 cm long, rhodium detectors located at approximately 20, 40, 60, and 80% core height.

3.4.7 REACTOR STABILITY

3.4.7.1 General

Pressurized water reactors with negative overall power coefficients are inherently stable with respect to power oscillations. Therefore, this discussion will be limited to xenon-induced power distribution oscillations.

Xenon-induced oscillations occur as a result of rapid perturbations to the power distribution which cause the xenon and iodine fission product distributions to be out of phase with the perturbed power distribution. This results in a shift in the iodine and xenon distribution that causes the power distribution to change in an opposite direction from the initial perturbation and, thus, initiate oscillations. The magnitude of the power distribution oscillation can either increase or decrease with time. Thus, the core can be considered to be either unstable or stable with respect to these oscillations. Xenon stability analyses on previous Calvert Cliffs cores

indicate that any radial and azimuthal xenon oscillations induced in the core would be damped. Axial xenon oscillations, however, could exhibit instabilities during later portions of the cycle in the absence of appropriate control action. Before discussing the methods of analysis and control, it is appropriate to reiterate several important aspects of the xenon oscillation phenomenon.

a. The time scale for the oscillations is long and any induced oscillation typically exhibits a period of about one day.
b. Xenon oscillations are readily detectable as discussed below.
c. As long as the initial power peak associated with the perturbation initiating the oscillation is acceptable, the operator has time, in the order of from hours to days, to take appropriate remedial action before the allowable peaking factors are exceeded.

3.4.7.2 Method of Analysis

A xenon oscillation may be described by the following equation:

$$\phi(\bar{r},t) = \phi_o(\bar{r}) + delta\ \phi_o(\bar{r}) e^{bt} \sin(\omega t + \sigma)$$

where:

$\phi(\bar{r},t)$	is the space-time solution of the neutron flux
$\phi_o(\bar{r})$	is the initial fundamental flux
delta $\phi_o(\bar{r})$	is the perturbed flux mode
b	is the stability index
ω	is the frequency of the oscillation
σ	is a phase shift

The stability of a reactor can be characterized by a stability index or a damping factor which is defined as the natural exponent which describes the growing or decaying amplitude of the oscillation.

A positive stability index (b) indicates an unstable core. A zero or a negative value indicates stability for the oscillatory mode being investigated. The stability index is generally expressed in units of inverse hours, so that a value of -0.01/hr would mean that the amplitude of each subsequent oscillation cycle decreases by about 25% for a period of about 30 hours for each cycle. Xenon oscillation modes can be classified into three general types: radial, azimuthal, and axial.

3.4.7.3 Radial Stability

A radial xenon oscillation consists of a power shift inward and outward from the center of the core to the periphery. This oscillatory mode is generally more stable than an azimuthal mode.

To confirm that the radial mode is extremely stable, for the first cycle a space-time calculation was run for a reflected, zoned core 11' in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth of 0.4% in reactivity placed in the central 20% of the core for 1 hr. Following removal of the perturbation, the resulting oscillation was followed in 4-hr time steps for a period of 80 hours. The resulting oscillation died out very rapidly with a damping factor of about -0.06/hr. If this damping coefficient is corrected for a finite time mesh, it would become even more strongly convergent. On this basis, it is concluded that radial oscillations are highly unlikely.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core, whose behavior is more closely followed by the excore detectors. Primary reliance is placed on these detectors for the detection of any xenon oscillations.

3.4.7.4 Azimuthal Stability

An azimuthal oscillation consists of an X-Y power shift from one side of the reactor core to the other. Azimuthally-symmetric operation and design practices and a negative stability index ensure proper azimuthal power distribution.

3.4.7.5 Axial Stability

Axial xenon oscillations consist of a power shift between the top and bottom of the reactor core. This type of oscillation may be unstable toward the EOC.

3.4.7.6 Detection and Control of Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the excore flux monitoring instrumentation. As indicated earlier, oscillations in modes such as radial, which would allow the center of the core to behave independently from the peripheral portions of the core, are highly unlikely and this lends support to reliance on the excore detectors for this purpose.

Although the primary response of these detectors will be to the power in the peripheral fuel assemblies, the lower modes of any induced oscillations will affect the power shapes in these peripheral assemblies. Therefore, azimuthal or axial flux tilts can be observed and identified with the use of incore or excore instrumentation and appropriate remedial action can be taken.

In addition, the incore flux monitoring instrumentation is used to verify the correlation between indications from the excore detectors and the space-dependent flux distribution within the core.

The reactor is operated in such a manner as to avoid inducing sizable spatial perturbations. As was discussed previously, radial and azimuthal xenon oscillations are expected to be damped in the Calvert Cliffs cores. Axial oscillations, however, may be undamped in the latter stages of core life. These unstable xenon oscillations require control action to prevent them from building in magnitude; however, they are very slow acting and thus leave time for appropriate control strategies to be determined. The part-length CEAs initially installed to control axial oscillations were removed since full-length CEAs have proved to be effective in controlling all xenon oscillations.

3.4.8 NEUTRON FLUX AT PRESSURE VESSEL

The original design of the reactor vessel considers the fast neutron fluence (neutron energy greater than 1 MeV) to the inner wall of the vessel. The fluence was determined by combining the results of the computer code P3MG1 (Reference 7) and SHADRAC (Reference 2). A detailed neutron transport analysis using the discrete ordinate computer code DOT-4 (Reference 10) is periodically performed to determine the fast neutron fluence on the vessel. The analysis considers the neutron flux from previous fuel cycles, and projected low fluence core design. The method is verified by examination of surveillance capsules. Beginning with Unit 1 Cycle 11 and Unit 2 Cycle 10, low fluence fuel management is used to reduce the fluence at the vessels' beltline region welds.

3.4.9 ANALYTICAL METHODS

3.4.9.1 General

Calvert Cliffs reactor cores are designed using a series of calculations to determine the energy and spatial dependent neutron flux, the integral or differential core reactivity, and the power distribution. The computer code DIT (Reference 4) calculates the energy dependent flux, the spatial dependent flux, and the flux weighted assembly-wide cross-sections necessary to perform core-wide calculations. Originally, the computer code PDQ (Reference 3) was used to calculate a few group pin power distribution. Starting with Unit 2 Cycle 8 and Unit 1 Cycle 10 the fine-mesh code MC (Reference 4) replaced PDQ. The computer code ROCS (Reference 4) was used to calculate a coarse-mesh two- or three-dimensional power distribution and the core averaged reactivity coefficients.

Starting with Unit 2 Cycle 16 and Unit 1 Cycle 18, the PARAGON and ANC computer codes are used for nuclear design analysis. PARAGON (Reference 11) calculates the energy dependent flux, the spatial dependent flux, and the flux weighted assembly-wide cross-sections, and the pin peak reconstruction factors necessary to perform core-wide calculations. The computer code ANC (References 12, 13, and 14) calculates a coarse and fine mesh two- or three-dimensional power distribution and the core averaged reactivity coefficients. Previously the computer codes DIT and ROCS (Reference 4) were used for these purposes. Both PARAGON and ANC have been extensively benchmarked to a wide variety of measurements including those taken on several past cycles of the Calvert Cliffs units.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21 with AREVA methods, the MICBURN-3/CASMO-3G and PRISM computer codes are used for nuclear design analysis. MICBURN-3/CASMO-3G (Reference 15) calculates the cross-sections, discontinuity factors, and heterogeneous form functions for the core simulator code, PRISM. The computer code PRISM (Reference 15) calculates the core-wide power distribution in three dimensions. MICBURN-3/CASMO-3G and PRISM have been extensively benchmarked to a wide variety of measurements including those taken on several past cycles of the Calvert Cliffs units.

MICBURN-3 is a multigroup one-dimensional transmission probability code which calculates the microscopic burnup in Gadolinium-loaded fuel containing initially homogeneously distributed poison. These cross-sections, as a function of absorber number density, are input to CASMO-3.

CASMO-3G is a multi-group, two-dimensional transport theory code for burnup calculations on assemblies. CASMO-3G is capable of modeling the geometry of the Calvert Cliffs cores including non-symmetric fuel bundles. The microscopic depletion is calculated in each fuel rod and burnable absorber rod. The output consists of cross-sections, discontinuity factors, and heterogeneous form functions for the core simulator code, PRISM.

The PRISM code performs core-wide two-group calculations. It uses pin power reconstruction to establish the individual rod histories and reactivities. The reactor core for Calvert Cliffs is modeled as 4 radial nodes and 32 axial nodes per assembly. With this reactor model, axial effects, including predicted values of LHR, F_r^T, and F_z can be studied. Thermal hydraulic feedback and axial exposure distribution effects on power shapes, rod worths, and cycle lifetime are explicitly included in the PRISM analysis.

The computer code INCA (Reference 5) was used to perform the on-line incore calculations. In Unit 2 Cycle 9 and Unit 1 Cycle 11, INCA is replaced by CECOR 3.3 for power distribution surveillance. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, CECOR is replaced by the AREVA core monitoring system POWERTRAX for power distribution surveillance. Satisfactory comparisons between these measured data and the predictions validate the design procedures.

3.4.9.2 Coarse-Mesh Diffusion Calculations - Westinghouse only

Coarse-mesh calculations are performed by ANC. This code contains a nodal solution to the diffusion equation to obtain a high degree of accuracy with a low number of spatial mesh points. The reactor core is typically modelled in ANC as four radial nodes and 20 to 30 axial nodes per fuel assembly. The axial and radial reflector regions are explicitly represented as additional nodes that are attached to the core boundary. ANC also contains equilibrium thermal models required to correctly determine the neutronic impact of changes in the spatial distributions of fuel temperatures and moderator density.

3.4.9.3 Power Distribution Monitoring

During normal operating conditions, signals proportional to the rhodium activation rates are obtained from the incore detectors. These signals are related to local power by use of calculated signal-to-power conversion factors for the appropriate core conditions. The measured signals are corrected for background, calibration, and depletion.

The POWERTRAX system replaces CECOR starting with Unit 2 Cycle 19 and Unit 1 Cycle 21. POWERTRAX provides a method of synthesizing detailed three dimensional assembly and peak-pin power distributions. This method is described below.

a. Calculated 3-D Nodal Power and Signal Distributions

The calculated nodal power and detector signals are provided by the POWERTRAX system nodal simulator.

b. Measured Powers at Operable Detector Locations

Measured powers at the operable detector locations are generated by multiplying the calculated nodal power by the corresponding ratio of the measured to calculate detector signals. The calculation is performed at all axial detector levels.

c. Measured Powers at other Locations

Other locations include those detector axial levels in un-instrumented assembly locations and those locations with failed detectors. The measured nodal powers at other locations are computed using the relation between the calculated nodal powers at these locations and at the locations with operable detectors.

d. Radial Power Distributions

A radial power distribution is a combination of the measured powers at operable detector locations and measured power at other locations.

e. Axial Power Profile

The axial power profiles are derived from the axial profiles from the 3-D nodal simulator calculation and adjusted by the differences in the measured and

calculated radial power distribution described above. These adjustments are made by modulating the calculated nodal power distribution with the ratio of inferred segment powers to the calculated nodal powers.

f. Final Normalization

The resulting 3-D power distribution is then re-normalized to the core thermal power to produce the final inferred 3-D nodal power distribution.

g. Peak Pin Power Distributions

Using the calculated nodal power distribution provided in step a, F_r (F_r-total is the maximum average pin power integrated over the entire core height; F_r-unrodded is the maximum average pin power integrated over the unrodded portion of the core) and pin-to box factor (PF-total is calculated for each assembly as the F_r-total value divided by the measured assembly average power; PF-unrodded is calculated for each assembly as the F_r-unrodded value divided by the average of the measured nodal powers averaged over the unrodded planes) can be obtained.

h. Azimuthal Power Tilt

The azimuthal power tilt is computed by determining the maximum values and locations of the maximum values from quadrant power tilt calculations for the upper and lower halves of the reactor core at each degree of rotation angle for 360° rotationally.

3.4.10 REFERENCES

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TABLE 3.4-1

NUCLEAR PARAMETERS

I. FIRST CYCLE NUCLEAR PARAMETERS	
<u>Control Characteristics</u>	
k_{eff}, BOL, No Control Element Assemblies or Dissolved Boron with BPRs in	
Cold (68°F)	1.194
Hot (532°F), Zero Power	1.152
Hot (572°F), Full Power	1.128
Hot, Equilibrium Xe, Full Power	1.094
<u>Total CEA Worth, %</u>	
BOL	
Hot (572°F)	9.8
Cold (68°F)	5.7
EOC	
Hot (572°F)	9.7
Cold (68°F)	5.6
<u>Dissolved Boron</u>	
Dissolved Boron Content for Criticality, ppm, (CEAs withdrawn, BOL)	
Cold (68°F)	1120
Hot (532°F), Zero Power, Clean	1095
Hot (572°F), Full Power	960
Hot (572°F), Equilibrium Xe, Full Power	725
Dissolved Boron Content for Refueling, ppm	1720
Boron Worth, ppm/%	
Hot (572°F)	86
Cold (68°F)	69
<u>Reactivity Coefficients (CEAs Withdrawn Unless Otherwise Indicated)</u>	
MTC, $\Delta\rho$/°F	
Hot (572°F)	
BOC, 960 ppmb	$-0.20\text{x}10^{-4}$
BOC, 847 ppmb (1% CEAs In)	$-0.49\text{x}10^{-4}$
EOC, Zero ppmb	$-1.96\text{x}10^{-4}$
EOC, Zero ppmb (1% CEAs In)	$-2.20\text{x}10^{-4}$
Cold (68°F) Zero Power	
BOC, 1120 ppmb	$-0.06\text{x}10^{-4}$
FTC, $\Delta\rho$/°F	
Hot, Zero Power	$-1.46\text{x}10^{-5}$
Full Power	$-1.06\text{x}10^{-5}$
Moderator Void Coefficient, $\Delta\rho$/% Void	
Hot, Operating, BOL	$-0.1\text{x}10^{-3}$
EOC	$-1.3\text{x}10^{-3}$
Moderator Pressure Coefficient, $\Delta\rho$/psi	
Hot, Operating, BOL	$+0.3\text{x}10^{-6}$
EOC	$+2.6\text{x}10^{-6}$

TABLE 3.4-1

NUCLEAR PARAMETERS

II. CURRENT CYCLE NOMINAL NUCLEAR CHARACTERISTICS

	UNIT 1 CYCLE 23	UNIT 2 CYCLE 22
Dissolved Boron, ppm		
Dissolved Boron Content for Criticality, CEAs Withdrawn, Hot Full Power, Equilibrium Xenon, BOC	1425	1419
Boron Worth, ppm/% $\Delta\rho$		
Hot Full Power, BOC	147	148
Hot Full Power, EOC	106	107
Moderator Temperature Coefficient (CEAs Withdrawn), 10^{-4} $\Delta\rho$/°F Hot Full Power, Equilibrium Xenon		
BOC	-0.39	-0.41
EOC	-2.85	-2.84

TABLE 3.4-2

CEA REACTIVITY WORTH AND ALLOWANCES, (% $\Delta\rho$)

I. UNIT 1 FIRST CYCLE VALUES

	BOC	EOC
Fuel Temperature Variation	1.7	1.8
Moderator Temperature Variation	0.7	1.4
Moderator Voids	0.0	0.1
CEA Bite and Boron Deadband	0.2	0.2
Shutdown Margin and Safety Features		
Allowance	2.0	2.4
Total Reactivity Allowances	4.6	5.9
Stuck CEA Allowance	1.9	2.2
Calculated CEA Worth at 572°F		
(77 Full-Length CEAs)	9.8	9.7
Uncertainty Allowance and Margin	3.3	1.6

II. CURRENT CYCLE LIMITING VALUES FOR EXCESS SHUTDOWN MARGIN

	UNIT 1 CYCLE 23	UNIT 2 CYCLE 22
Limiting Condition	EOC HFP	EOC HFP
Control Rod Worth		
ARI (All rods inserted)	8.360	8.283
MRR (Most reactive rod)	1.518	1.484
PDIL	0.175	0.173
(ARI-MRR-PDIL)*0.9	6.000	5.963
Positive Reactivity Insertion		
Power Defect	2.198	2.221
Axial Flux Redistribution	0.234	0.177
Coolant Void Effects	0.050	0.050
Total Positive Reactivity Insertion	2.481	2.448
Shutdown Margin		
(ARI-MRR-PDIL)*0.9 - Total Positive Reactivity Insertion	3.519	3.515
Required Shutdown Martin	3.500	3.500
Excess Shutdown Margin	0.019	0.015

3.5 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

3.5.1 GENERAL

This section presents the thermal and hydraulic characteristic data and design methodology. The objective of the thermal and hydraulic design of the reactor is to ensure that the core can meet steady state and transient performance requirements without violating the design bases. The principal thermal and hydraulic design bases are related to DNB, fuel/clad melting, and hydraulic loading. Instrument and control uncertainties, delays between parameter changes, RPS trip signals, and initiation of CEA movement are involved in the transient calculations. The RPS monitors and trips the reactor upon sensing an adverse condition. The Engineered Safety Feature Actuation Signal (ESFAS) provides automatic corrective action when operating parameters exceed their setpoints. The Safety Analysis determines the setpoints such that the design bases will not be exceeded during AOOs and most postulated accidents. The Safety Analysis (Chapter 14) discusses each of the DBEs in detail. A summary of the key thermal and hydraulic parameters is presented in Table 3.5-1.

3.5.1.1 Cycle Summaries

The following cycle summaries provide a brief synopsis of major changes associated with each cycle.

a. Unit 1

1. Cycle 2

   The core power level was increased from 2560 MWt to 2700 MWt.

   Several design changes to the type D fuel improved the thermal performance and were included in the performance analysis. The parameters that affected the gap conductance, such as decreased pellet/cladding gap and the increased pellet density that decreased the effects of densification, were responsible for the improved thermal performance of the type D fuel.

2. Cycle 3

   Excessive CEA guide tube wear was identified. In order to reduce the wear on the guide tubes and CEA fingers, SS sleeves were inserted into the top end of some of the guide tubes. The sleeves protect the guide tubes against wear by CEA fingers.

   Part-length CEAs were also removed to reduce guide tube wear and dummy CEA plugs were inserted to minimize the increase in bypass flow.

   Minimum Departure from Nucleate Boiling Ratio was calculated using TORC/CE-1. TORC was used to generate the LCO in the Technical Specifications and was also used for all AOOs and postulated accidents. The TORC thermal hydraulics code replaces COSMO-INTHERMIC.

   TORC uses the CE-1 DNBR correlation, whereas previous cycles used COSMO-INTHERMIC, which uses the W-3 DNBR correlation (References 2, 3, 4, and 5).

   The detailed version of TORC (Reference 2) is a benchmarking code. The simplified version (Reference 3) runs considerably faster but is set

to be more conservative when benchmarked against the detailed version.

Mechanical design and power distribution uncertainty factors used in the calculation of thermal margin were previously combined multiplicatively. In Cycle 3, these factors:

F_q^e = Engineering Factor,

F_q^n = Nuclear Factor,

F_q^f = Fuel Rod Bowing Factor and,

F_q^p = Poison Rod Factor

were combined using a root-sum-square (RSS) technique. The RSS technique is appropriate for combining random uncertainties. The multiplicative technique is appropriate for combining systematic/dependent uncertainties.

Augmentation factors were calculated using the FATES (Reference 1) fuel rod model.

3. Cycle 4

The guide tubes of reload fuel assemblies were either sleeved or their flow holes were changed. The sleeves protect the guide tubes against wear. The flow hole modification reduced coolant flow and thereby reduced the CEA vibration which caused the guide tubes to wear. Irradiated fuel assemblies previously resident in CEA locations, but not having sleeves, were sleeved in order to regain structural margin.

Fuel assemblies exceeding the 24,000 MWD/MTU rod bow penalty threshold were placed in core locations where their power density was sufficiently low to offset rod bow penalties on the MDNBR limit.

4. Cycle 5

The RPS was modified to include an asymmetric steam generator transient protection trip function. The trip function originates from the thermal margin/low pressure (TM/LP) logic and trips the reactor for those AOOs associated with secondary system malfunctions which would result in asymmetric primary loop temperatures. The limiting event is the loss of load to one steam generator caused by the closure of a single main steam isolation valve.

5. Cycle 6

The TORC/CE-1 thermal design code has been replaced by the CETOP/CE-1 code (Reference 12). The treatment of core system parameter uncertainties on the DNBR SAFDL has been changed from the deterministic approach to statistical combination of uncertainties (SCU) (References 13, 14, and 15). The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors to define a new design limit on CE-1 MDNBR at the 95/95 confidence/probability level.

The performance of the fuel has been analyzed using FATES-3 (Reference 17), a fuel performance code.

The analysis with these methodology changes resulted in a MDNBR limit of 1.23.

6. Cycle 7

   The effects of fuel rod bowing on DNBR margin were evaluated using the methods described in Reference 11.

7. Cycle 8

   The PLCEA plugs were removed for Cycle 8 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling outage operations. An assessment concluded that the removal of the CEA plugs from all eight partial length rod locations has insignificant effect on the thermal and hydraulic design.

   The axial fuel densification factor is reduced from 1.01 to 1.002. A negative bias of 15% is added to the FTC data used in the safety analysis to establish consistency with the bias in the ROCS/DIT topical. This bias is used conservatively by selective application.

8. Cycle 9

   The fuel thermal performance calculations used FATES3B (Reference 18) which is an updated version of the FATES3 (Reference 17) fuel evaluation model. The statistically derived DNBR limit was reduced from 1.23 to 1.21. This reduction resulted from Nuclear Regulatory Commission (NRC) approval of a reduced CE-1 DNBR limit for CE's 14x14 fuel.

9. Cycle 10

   The DNBR SAFDL calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors using the extended statistical combination of uncertainties (ESCU) methodology of Reference 19. This combination is used to derive an overall uncertainty allowance which, when used with the CE-1 critical heat flux (CHF) correlation DNBR design limit of 1.15 for 14x14 fuel, provides a 95/95 probability/confidence level of assurance against DNB occurring during steady state operation and AOOs. The statistically derived ESCU uncertainty allowance includes a 0.006 DNBR rod bow penalty which accounts for the adverse effects of rod bowing on CHF for 14x14 fuel with burnup not exceeding 45 GWD/T.

10. Cycle 11

    The fresh assemblies (Batch N) in Cycle 11 employ the GUARDIAN™ debris-resistant fuel design and large envelope Zircaloy grids. This fuel design results in a greater hydraulic resistance than the debris-resistant LEF design of the Batch 2L fuel used in Unit 2 Cycle 9. This increase in hydraulic resistance results in a slight decrease in the inlet flow for the Batch N fuel. The TORC and CETOP models used in the DNB analysis account for this flow reduction.

11. Cycle 12

    The fresh assemblies (Batch P) in Cycle 12 employ the GUARDIAN™ design with an improved top grid design. The top laser-welded grid introduces a backup arch in each grid cell in addition to the existing backup arches in the peripheral cell locations. This results in a greater hydraulic resistance in the Batch P assemblies. The TORC and CETOP models used in the DNB analysis account for the hydraulic resistance of each assembly type.

12. Cycle 13

    The standard fresh assemblies (Batch 1R) in Unit 1 Cycle 13 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1R fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 13 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 13 core.

13. Cycle 14

    The standard fresh assemblies (Batch 1S) in Unit 1 Cycle 14 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1S fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 14 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 14 core.

14. Cycle 15

    The standard fresh assemblies (Batch 1T) in Unit 1 Cycle 15 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1T fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 15 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 15 core.

    Mid-cycle the ABB-NV CHF correlation was approved. Therefore, both the CE-1 and ABB-NV correlations are applicable for DNB analysis for Unit 1 starting in Cycle 15. Calculations done with the new correlation were implemented mid-cycle.

15. Cycle 16

    The 96 fresh Turbo assemblies (Batch 1V) both with ZIRLO and OPTIN cladding in Unit 1 Cycle 16 employ the same straight strip GUARDIAN™ grid design as earlier Batches S and T. Batch 1V Turbo fuel also utilizes two bottom and one top straight strip advanced no-vane Zircaloy spacer grids and five intermediate advanced Zircaloy spacer grids with mixing vanes. The Unit 1 Cycle 16 DNB analysis explicitly accounts for the resistance of each assembly type in Unit 1 Cycle 16 mixed core.

    ABB-NV and ABB-TV CHF correlations have been developed applicable to Westinghouse standard and Turbo types of fuel assemblies, respectively, in Reference 20. The ABB-TV correlation has a better CHF performance compared to the ABB-NV correlation due to the presence of mixing vane grids in Turbo type fuel assemblies.

    Because of the higher hydraulic resistance of the Turbo fuel assemblies compared to the standard fuel assemblies, Turbo assemblies lose flow to standard fuel assemblies along the height of the core. This loss of

flow from Turbo fuel assemblies to the surrounding standard fuel assemblies in mixed core configuration such as in Unit 1 Cycle 16 is explicitly accounted for in the TORC DNB analysis.

Dual bundle tests have shown that the TORC prediction of this diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly is fairly close with the test results. However, in order to conservatively compensate for any minor adverse effect on DNB margin assessment of Turbo type fuel assemblies due to small differences between the test results and TORC prediction of the diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly, a margin neutral approach has been adopted for Unit 1 Cycle 16 DNB analysis. Based on this approach, Turbo fuel assemblies in Cycle 16 have been conservatively treated as standard fuel assemblies and the DNB margin assessment for Cycle 16 has been performed using ABB-NV CHF correlation documented in Reference 20. In other words, no margin credit has been taken for the ABB-TV CHF correlation that is applicable to Turbo fuel.

16. Cycle 17

Eighty-eight fresh assemblies were installed for Unit 1 Cycle 17 (batch designation 1W).

Batch 1W is the third batch of VAP for Unit 1. Erbia remains the burnable absorber, the Erbia fuel pins have cutback regions of 10.5 inches at the top of the rod and 12.0 inches at the bottom.

Batch 1W is the second batch of the Turbo fuel assembly design for Unit 1. All but four assemblies utilized the same design and the same grid cage design as the U2C15 Westinghouse LFAs, see Section 3.7.3.13. The four different assemblies do not have the increased backup-arch length and have been given a unique sub-batch identifier. All of the fuel was manufactured by Westinghouse at their Columbia, SC facility. The cladding material for all fresh fuel is ZIRLO™.

As Unit 1 contains approximately 85% of the Turbo fuel assembly design, transient analyses were updated to utilize the ABB-TV CHF correlation. Since the Turbo fuel has a non-mixing vane lower axial section and an upper section with mixing grids, both the ABB-NV and the ABB-TV CHF correlations are applied in the safety analysis.

17. Cycle 18

The fresh assemblies (Batch 1X) for Unit 1 Cycle 18 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 18 also employs the third full batch of the Turbo advanced grid design for Unit 1 and is the first Unit 1 core to contain all Turbo fuel assemblies.

18. Cycle 19

    Ninety-six fresh assemblies (Batch 1Z) were loaded for Unit 1 Cycle 19. All assemblies utilized in Cycle 19 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

19. Cycle 20

    The rated thermal power was increased from 2700 MWt to 2737 MWt.

    Ninety-two fresh assemblies (eighty-eight Batch AA and four Batch 2X7) were loaded for Unit 1 Cycle 20. All assemblies utilized in Cycle 20 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

20. Cycle 21

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

21. Cycle 22

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

22. Cycle 23

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

b. Unit 2

1. Cycle 2

    The core power level was increased from 2560 MWt to 2700 MWt.

    The following modifications made to Unit 1 Cycles 3 and 4 were also made to the design for Unit 2 Cycle 2:

    a) SS sleeves were installed in CEA guide tubes of selected reload fuel assemblies;

b) The size and number of CEA guide tube flow holes were modified on other selected fuel assemblies;

c) Part-length CEAs were removed and replaced by CEA plugs;

d) Minimum Departure from Nucleate Boiling Ratio was calculated using TORC/CE-1; and

e) Mechanical design and power distribution uncertainty factors, used in the calculation of thermal margin, which were previously combined multiplicatively were combined using an RSS technique.

2. Cycle 3

All fuel assemblies placed in CEA locations had SS sleeves installed in the guide tubes in order to prevent guide tube wear.

Augmentation factors were calculated using the FATES (Reference 1) fuel rod model.

Fuel assemblies exceeding the 24,000 MWD/MTU rod bow penalty threshold were placed in core locations where their power density was sufficiently low to offset rod bow penalties on the MDNBR limit.

3. Cycle 4

The RPS was modified to include the asymmetric steam generator transient protection trip function. The trip function originates from the TM/LP logic and trips the reactor for those AOOs associated with secondary system malfunctions which would result in asymmetric primary loop temperatures. The most limiting event is the loss of load to one steam generator caused by the closure of a single main steam isolation valve.

4. Cycle 5

The TORC/CE-1 thermal design code has been replaced by the CETOP/CE-1 Code (Reference 12). The treatment of core system parameter uncertainties has been changed from the deterministic approach to SCU (References 13, 14, and 15). The DNBR SAFDL calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors to define a new design limit on CE-1 MDNBR at the 95/95 confidence/probability level.

The analysis with these methodology changes resulted in a MDNBR limit of 1.23.

5. Cycle 6

The effects of fuel rod bowing on DNBR margin were evaluated using the methods described in Reference 11.

6. Cycle 7

The statistically derived DNBR limit was reduced from the value of 1.23 to a value of 1.21. The reduction results from NRC approval of a reduced CE-1 DNBR limit for CE's 14x14 fuel. At the time the SCU analysis was approved for the Calvert Cliffs units, NRC review of the

applicability of the CE-1 CHF correlation to rods with nonuniform APDs was incomplete. An interim CE-1 DNBR limit of 1.19 was thus used in the original SCU analysis. In the review of CE's nonuniform APD topical report, the NRC reduced the CE-1 DNBR limit from 1.19 to 1.15 for 14x14 fuel. The SCU DNBR limit was correspondingly reduced from 1.23 to 1.21. The 1.21 SCU DNBR limit includes the following penalties imposed by the NRC in their review of the SCU analysis.

- Critical heat flux correlation cross validation penalty (5% increase in standard deviation of CHF correlation uncertainty distribution).
- T-H code uncertainty penalty (5%, equal to two standard deviations).

The 1.21 SCU DNBR limit also includes a 0.006 DNBR rod bow penalty which accounts for the adverse effects of rod bowing on CHF for 14x14 fuel with burnup not exceeding 45 GWD/T.

The axial fuel densification factor was reduced from 1.01 to 1.002 to make it consistent with existing calculations.

The PLCEA plugs were removed for Cycle 7 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling outage operations.

7. Cycle 8

   The thermal performance of the fuel was evaluated using the FATES3B (Reference 18) fuel evaluation model.

8. Cycle 9

   The fresh assemblies (Batch L) in Cycle 9 have small flow hole LEF plates, which result in greater hydraulic resistance than the LEF plates of the irradiated fuel. This increase in hydraulic resistance will result in a slight decrease in the inlet flow for these Batch L assemblies. The cycle specific TORC and CETOP models used in the Cycle 9 DNB analyses accounted for this flow reduction.

   The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors using the ESCU methods (Reference 19). This combination is used to derive an overall uncertainty allowance which, when used with the CE-1 CHF correlation design limit of 1.15 for 14x14 fuel, provides a 95/95 probability/confidence level of assurance against DNB occurring during steady state operation or AOOs.

9. Cycle 10

   The fresh assemblies (Batch 2M) in Cycle 10 employ the GUARDIAN™ debris resistant fuel design. This fuel design results in a greater hydraulic resistance than the Batch 2L fuel which trapped debris by employing small flow holes in the lower end fitting.

10. Cycle 11

The standard fresh assemblies (Batch 2N) in Unit 2 Cycle 11 are the second Unit 2 batch to employ the GUARDIAN™ design. For Batch 2N, a straight strip GUARDIAN™ grid design was introduced, replacing the previously employed wavy GUARDIAN™ grid design. The new straight strip GUARDIAN™ grid design has lower hydraulic resistance than the Batch 2M wavy strip GUARDIAN™ grid design.

Laser welded wavy strip intermediate Zircaloy spacer grids were also introduced for the Batch 2N fuel, replacing the previously employed TIG welded wavy strip intermediate Zircaloy spacer grids. The laser welded intermediate grids have a slightly lower hydraulic resistance than the TIG welded grids. The Cycle 11 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 2 Cycle 11 core.

11. Cycle 12

The standard fresh assemblies (Batch 2P) in Unit 2 Cycle 12 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 2P fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The perimeter strips have small guide holes to match pins on the grid assembly weld fixture. This is to ensure more consistent alignment of the grid strips within the fixture during welding. The Unit 2 Cycle 12 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 2 Cycle 12 core.

12. Cycle 13

The standard fresh assemblies (Batch 2R) in Unit 2 Cycle 13 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 2R fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The DNB analysis for Unit 2 Cycle 13 explicity accounted for the resistance of each assembly type.

13. Cycle 14

The fresh assemblies (Batch 2S) in Unit 2 Cycle 14 employ the laser-welded, straight-strip GUARDIAN™ grid design. The Batch 2S fuel also utilizes laser-welded wavy-strip intermediate Zircaloy spacer grids. The DNB analysis for Unit 2 Cycle 14 explicitly accounted for the resistance of each assembly type. The ABB-NV DNB correlation was used instead of the CE-1 correlation.

14. Cycle 15

The fresh assemblies (Batch 2T) manufactured by Westinghouse in Unit 2 Cycle 15 employ the laser-welded, straight strip GUARDIAN™ grid design. Unit 2 Cycle 15 also contains the first full batch of the Turbo advanced grid design for Unit 2. Unit 1 received the first full batch of the Turbo advanced grid design at Calvert Cliffs in Unit 1 Cycle 16. The Turbo grid features include mixing vanes (at five of the eight spacer grid locations) and new rod retention device known as I-springs (at all eight spacer grid locations). The Unit 2 Cycle 15 DNB analysis explicitly

accounts for the resistance of each assembly type in the Unit 2 Cycle 15 mixed core.

ABB-NV and ABB-TV CHF correlations have been developed applicable to Westinghouse standard and Turbo types of fuel assemblies, respectively, in Reference 20. The ABB-TV correlation has a better CHF performance compared to the ABB-NV correlation due to the presence of mixing vane grids in Turbo type fuel assemblies.

Because of the higher hydraulic resistance of the Turbo fuel assemblies compared to the standard fuel assemblies, Turbo assemblies lose flow to standard fuel assemblies along the height of the core. This loss of flow from Turbo fuel assemblies to the surrounding standard fuel assemblies in mixed core configuration such as in Unit 2 Cycle 15 is explicitly accounted for in the TORC DNB analysis.

Dual bundle tests have shown that the TORC prediction of this diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly is fairly close with the test results. However, in order to conservatively compensate for any minor adverse effect on DNB margin assessment of Turbo type fuel assemblies due to small differences between the test results and TORC prediction of the diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly, a margin neutral approach has been adopted for Unit 2 Cycle 15 DNB analysis. Based on this approach, Turbo fuel assemblies in Cycle 15 have been conservatively treated as standard fuel assemblies and the DNB margin assessment for Cycle 15 has been performed using ABB-NV CHF correlation documented in Reference 20. In other words, no margin credit has been taken for the ABB-TV CHF correlation that is applicable to Turbo fuel.

Batch 2T also contains LFAs from Westinghouse and FANP/AREVA. See Section 3.7.3.13 for detailed discussion on the Unit 2 Cycle 15 LFA.

In order to accommodate growth of the guide tube thimbles, the center guide tube recess hole in the lower end fitting of the fuel assembly was made approximately 1-7/8 inches deeper. This was accomplished via a process known as electric discharge machining and was performed or the fresh fuel after delivery.

15. Cycle 16

    The fresh fuel assemblies (Batch 2V) for Unit 2 Cycle 16 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 16 also contains the second full batch of the Turbo advanced grid design for Unit 2. The TORC DNB analysis explicitly accounts for the difference in the hydraulic resistance between the Turbo and standard fuel assemblies. This is the second cycle for the Westinghouse and AREVA LFAs.

16. Cycle 17

    The fresh assemblies (Batch 2W) for Unit 2 Cycle 17 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 17 also employs the third full batch of the Turbo advanced grid design for Unit 2 and is the first Unit 2 core to contain all Turbo fuel assemblies.

17. Cycle 18

    The rated thermal power was increased from 2700 MWt to 2737 MWt.

    Ninety-six fresh assemblies (Batch 2X) were loaded for Unit 2 Cycle 18. All assemblies utilized in Cycle 18 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

18. Cycle 19

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal-hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with the addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

19. Cycle 20

    One-hundred fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal-hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with the addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

20. Cycle 21

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

21. Cycle 22

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

3.5.2 THERMAL AND HYDRAULIC DESIGN BASES

Avoidance of thermally- or hydraulically-induced fuel damage during normal steady state operation and during AOOs is the principal thermal and hydraulic design basis. In order to satisfy the design basis for reactor operation, the following design limits are established, but exceeding these limits will not necessarily result in fuel damage. The RPS provides for automatic reactor trip and the ESFAS provides other corrective action before these design limits are violated for AOOs. However, there is a small probability of limited fuel damage for certain other DBEs discussed in Chapter 14.

3.5.2.1 Minimum Departure from Nucleate Boiling Ratio

The minimum allowed DNBR provides at least a 95% probability with a 95% confidence that DNB does not occur on a fuel rod having the calculated MDNBR during steady state operation and AOOs. The ABB-NV correlation coupled with the CETOP code provides at least this probability and confidence. The DNBR limit may be modified to account for the possibility of fuel rod bow at burnups in excess of 45,000 MWD/MTU (Table 3.5-1). Starting with Unit 1 Cycle 17, the ABB-TV correlation was used in conjunction with the ABB-NV correlation to make DNB determinations.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP™ correlation was used to determine DNBR for the AREVA fuel assemblies. The burnup limit for fuel rod bow impact on DNBR is greater than fuel assembly burnups for all anticipated core designs. Therefore, it is not necessary to modify the HTP™ correlation limit due to the effects of fuel rod bow.

3.5.2.2 Fuel Design Basis

a. Fuel Melt

The UO_2 melting point will not be reached during steady state operation and AOOs. For Westinghouse fuel assemblies, the UO_2 melting point is 5080°F unirradiated, and reduced by 58°F per 10,000 MWD/MTU burnup and reduced by 10.4°F for each weight percent of erbia of the maximum core erbia loading. For AREVA fuel assemblies, a bounding value of 4595°F was used for the UO_2 melting point. This value bounds all anticipated limiting fuel burnup distributions and Gadolinia concentrations. The thermal and hydraulic parameters which influence the fuel centerline temperature include maximum linear heat rate (LHR), coolant velocity, pressure, temperature, clad temperature, fuel-to-gap conductance, fuel burnup, and UO_2 temperature.

b. Fuel Cladding Integrity

The fuel design bases for fuel clad integrity and fuel assembly integrity are given in Section 3.2.3.5. Thermal and hydraulic parameters that influence the fuel integrity include maximum LHR, core coolant velocity, coolant temperature, clad temperature, fuel-to-clad gap conductance, fuel burnup, and UO_2 temperature.

The cladding minimizes deformation from external hydraulic pressure or internal gas/pellet pressure. Excessive contraction of the clad may lead to power spiking and excessive expansion may significantly decrease the flow channel area.

Conformance with the design limits and conformance with the design bases are sufficient to ensure fuel clad integrity, fuel assembly integrity, and the avoidance of thermally- or hydraulically-induced fuel damage for steady-state operation and AOOs.

3.5.2.3 Hydraulic Stability

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Permissible flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.5.3 STATISTICAL COMBINATION OF UNCERTAINTIES

The input data required for a detailed thermal-hydraulic analysis can be defined by type: (1) system parameters which describe the physical system and are not monitored during reactor operation; and (2) state parameters, which describe the operational state of the reactor and are monitored during operation. There is a degree of uncertainty in the value used for each of the input parameters used in the design safety analyses. This uncertainty has been handled in the past by assuming that each variable affecting DNB is at the extreme most adverse limit of its uncertainty range. The assumption that all factors are simultaneously at their most adverse values leads to conservative restrictions in reactor operation.

Beginning with Unit 1 Cycle 6, a new methodology was applied to statistically combine uncertainties in the calculation of new limits for Calvert Cliffs. These limits will ensure that neither the DNB nor fuel centerline melt design bases will be violated. The methodology is presented in three parts (References 13, 14, and 15). Part 1 (Reference 13) describes the application of the SCU to the development of the local power density (LPD) and TM/LP LSSSs. These are used in the analog RPS to protect against fuel centerline melt and DNB, respectively. Part 2 (Reference 14) uses SCU methods to develop a new DNBR limit which accommodates system parameter uncertainties. Part 3 (Reference 15) uses SCU methods to define LCOs.

For Unit 1 Cycle 10, an improved method was used for statistically combining uncertainties for the CE calculated TM/LP LSSS and DNB LCO. The extended combination of uncertainties (ESCU) methodology (Reference 19) is a modification of the SCU methodology (References 13, 14, and 15).

With the introduction of AREVA fuel assemblies for Unit 2 Cycle 19 and Unit 1 Cycle 21, a new method for the statistical combination of uncertainties was used to verify the TM/LP LSSS, the LPD LSSS, the DNB LCO, and LPD LCO. This methodology is described in Section 14.1.4.1 and Reference 23.

3.5.4 REACTOR HYDRAULICS

3.5.4.1 Coolant Flow

The minimum coolant flow at full power is shown in Table 3.5-1. Coolant enters the four inlet nozzles and flows down through the annular plenum between the reactor vessel and the core support barrel. Coolant continues through the flow skirt to the plenum below the core lower support structure. Pressure losses in the skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower core plate and flows axially through the fuel assemblies. After passing through the core, the coolant flows past the fuel alignment plate and into the region outside the CEA shrouds. From this region the coolant flows across the CEA shrouds and passes out through the outlet sleeves on the core barrel to the two outlet nozzles.

The principal core bypass routes (coolant flow paths other than through the fuel assemblies and next to the fuel rods) are direct inlet-to-outlet coolant flow at the joint between the core support barrel sleeve and the outlet nozzle and the flow in the radial reflector region between the core shroud and core support barrel. A small portion flows into the guide tubes in the fuel assemblies. The flow through the guide tubes has been modified by a reduction in size of the flow holes in the bottom of the guide tubes in other fuel assemblies. The coolant required to cool the CEAs flows in the annulus between the CEA and the guide tube and into the region outside the

CEA shrouds. A similar but smaller leakage will occur around the restriction to flow at the upper end of those guide tubes without CEAs. The design limits the total guide tube flow and core bypass flow to a maximum of 3.9% of total reactor vessel flow as compared to the calculated bypass flow shown in Table 3.5-2.

3.5.4.2 Pressure Losses

The irrecoverable pressure losses from the inlet to outlet nozzles are calculated using standard loss coefficient methods and information from flow model tests. The nominal design pressure losses are listed in Table 3.5-3.

3.5.4.3 Partial Flow Operation

The plant operates with all four Reactor Coolant Pumps functioning. Partial pump operation will only occur during transients prior to trip and is discussed in Chapter 14. The most limiting partial pump operation is during a Seized Rotor Event which is more limiting than a four-pump Loss of Coolant Flow Event.

Partial pump operation creates an unbalanced inlet and outlet nozzle flow rate and a core inlet flow maldistribution. Furthermore, unbalanced steam generator flow rates create a greater possibility of temperature nonuniformities at the core inlet due to incomplete mixing of the incoming coolant.

3.5.5 MAXIMUM CORE TEMPERATURE

The maximum core temperature occurs at the center of the hottest pellet. The temperature drops radially across the pellet, gap, clad and coolant film. Heat transfer correlations relate the physical properties, heat flux, and temperature drops. The different physical geometry and properties necessitate separate correlations. The Jens-Lottes/Dittus-Boelter equation, clad conductivity, gap conductance, and pellet conductance relate the temperature drops across the coolant film, cladding, gap, and pellet, respectively.

AREVA fuel was added to the core starting in Unit 2 Cycle 19 and Unit 1 Cycle 21. For AREVA fuel, the maximum fuel centerline temperature reached during the event is explicitly calculated using heat structure in the S-RELAP5 model which represents the hot node in the core for either a UO_2 rod or Gadolinia rod, whichever is limiting for the fuel centerline melt.

3.5.6 DEPARTURE FROM NUCLEATE BOILING

3.5.6.1 Design Approach to Departure from Nucleate Boiling

The margin to DNB at any point in the core is expressed in terms of the DNBR. The DNBR is defined as the ratio of the heat flux predicted to produce DNB at specific local conditions to the calculated local heat flux at the same local conditions. At some point in the core the DNBR is a minimum and, at this point, the margin to DNB for the core is evaluated. The following items are important in determining the core margin to DNB:

a. the coolant inlet conditions (e.g., pressure, temperature, and velocity distribution),
b. the geometry,
c. the power level,
d. the nuclear power distribution,
e. the analytical methods used to predict local coolant conditions, and
f. the correlation used to predict DNB heat flux.

Correlations of DNB are derived from experimental data and reduced to key parameters. Correlations for DNB are intended only to predict actual DNB and, therefore, the concept of DNB ratio can be misleading if one attempts to associate a physical meaning rather than a statistical meaning. Because of the uncertainties associated with predicting DNB there is a finite probability that if a channel is operated at a specified DNB ratio greater than unity based on a particular correlation, it will be at or above its DNB heat flux. Therefore, the proper interpretation of DNB ratio is that it is a measure of the probability that DNB would occur in the particular design situation to which the DNB correlation is applied. This interpretation assumes that all operating parameters are known precisely and that the probability being evaluated is only that associated with the correlation. The approach used in design is to select core operating conditions and analytical methods in such a way that there is a very small probability that the actual hot subchannel coolant conditions are more severe than the calculated conditions used as input to the DNB correlation. Starting with Unit 1 Cycle 17, the ABB-TV correlation was used in conjunction with the ABB-NV correlation to make DNB determinations.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP correlation (Reference 24) was used to determine DNBR for the AREVA fuel assemblies. The DNBR limit accounts for state and system parameter uncertainties.

3.5.6.2 Evaluation of Margin to DNB

DNBR analysis is performed over a wide range of coolant parameters to determine the envelope in which the DNBR is at least greater than the SAFDL (Table 3.5-1). The inlet coolant flow distribution used was empirically developed from scale models.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the XCOBRA-IIIC code was used to calculate the core thermal-hydraulic conditions for the AREVA assemblies. These conditions are then used with the HTP correlation to determine DNBR for the AREVA assemblies.

3.5.7 VAPOR FRACTION

At steady state the reactor is operated at a negative coolant quality (subcooled temperature). Therefore, vapor formation is minimized and, for the most adverse steady state conditions core vapor fraction is less than 0.1%. To avoid the possibility of DNB resulting from local flow oscillations, a conservative vapor fraction limit prevents flow instabilities. The limits to assure stable flow are based on avoiding flow regime changes in the hot channel that could affect the flow pressure drop characteristics and cause an instability. A thermal margin trip will occur before the flow instability limit is reached; thus DNB resulting from flow oscillations is prevented.

AREVA fuel was added to the core starting from Unit 2 Cycle 19 and Unit 1 Cycle 21. Flow instability is not part of the AREVA evaluation and no changes to the flow stability are expected with the transition to AREVA fuel.

3.5.8 THERMAL AND HYDRAULIC EVALUATION

The margin to CHF or DNB is expressed in terms of DNBR.

3.5.8.1 Statistical Analysis of Hot Channel Factors

Random variations from nominal values in enrichment, pellet density, pellet diameter and clad diameter, will affect the engineering heat flux factor on heat flux. Similar random variations in heat flux, as well as rod diameter, pitch, and bow contribute to the enthalpy rise factor. The calculation of these factors uses randomly-collected inspection data on "as-manufactured" fuel assemblies. Statistically random and independent constituent uncertainties are combined statistically.

3.5.8.2 Fuel Temperature Conditions

An assessment of the fuel centerline temperature has been made. The results demonstrate that a significant margin to centerline fuel melting will exist over the normal range of plant operation.

3.5.8.3 Flow Stability

Flow oscillations of significant amplitude may be sustained in some channels when heat is added to two-phase flow in parallel channels. This possibility is mitigated by the low vapor fraction during steady state and AOOs.

The two-phase flow regimes may be classed as separated or homogeneous. Separated flow is annular or slug. Homogeneous flow is bubbly or froth flow. For homogeneous flow, the channel pressure drop continuously increases with increasing flow rate or increasing vapor fraction. A change in the flow regime to separated flow results in a change in the flow characteristics and flow oscillations in the parallel channels are then possible. Cross flow tends to damp the oscillations and tends to make the open channel array stable when parallel closed channels would not be stable.

3.5.8.4 AREVA Fuel Assemblies

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP correlation was used to determine DNBR for the AREVA fuel assemblies. The HTP correlation is described in Reference 24 and is applicable to the following operating conditions and nominal range of fuel design parameters:

Parameter	**Minimum Value**	**Maximum Value**
Pressure, psia	1385	2425
Local Mass Flux, mlb/hr/ft^2	0.498	3.573
Inlet Enthalpy (Btu/lb)	382.3	649.9
Local Quality	---	0.515
Fuel Rod Diameter (in)	0.360	0.440
Fuel Rod Pitch (in)	0.496	0.580
Axial Spacer Span (in)	10.5	26.2
Hydraulic Diameter (in)	0.4571	0.5334
Heated Length (ft)	8.0	14.0

Based on the overall core conditions calculated at selected times during a transient evaluation, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single “channel”) linked to a detailed thermal-hydraulic model of the limiting assembly

(representing each subchannel by a single "channel"). The limiting assembly DNBR calculations are performed using an approved AREVA DNB correlation.

The use of XCOBRA-IIIC is limited to the "snapshot" mode when used for transients and is restricted from use in LOCAs and other calculations with flow reversal and recirculation as per the NRC licensing restrictions (Reference 25). This mode is based on a series of steady-state calculations for input over a series of time steps.

3.5.9 REFERENCES

1. CENPD-139-P-A and CENPD-139, Supplement 1, Rev. 1, "CE Fuel Evaluation Model Topical Report," July 1974
2. CENPD-161-P-A, "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986
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4. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September 1976
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23. EMF-1961(P)(A), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000
24. EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 2005

25. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)

TABLE 3.5-2

REACTOR COOLANT FLOWS IN BYPASS CHANNELS

BYPASS ROUTE	UNIT 1 PERCENT OF TOTAL REACTOR FLOW	UNIT 2 PERCENT OF TOTAL REACTOR FLOW
Outlet nozzle clearances	0.75	0.75
Alignment keyways	0.10	0.10
Core shroud annulus	0.61	0.61
Guide tubes	2.04	2.04
Total bypass	3.51	3.51

NOTE: The Unit 1 and Unit 2 bypass flows are based on a full core consisting of Turbo fuel assemblies that are conservative for mixed core configurations. Units 1 and 2 bypass flow accounts for 35 ICI thimbles. The total design bypass flow rate used for safety analysis is typically 3.9% which accounts for uncertainties and the increase in hydraulic resistance due to postulated crud buildup.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the guide tube bypass flow will decrease from 2.04% to 2.02% as the core transitions from a full core of Westinghouse Turbo fuel to a full core of AREVA fuel.

TABLE 3.5-3

DESIGN REACTOR PRESSURE LOSSES

	VELOCITY, ft/sec	UNIT 1[a] PRESSURE LOSS, psid	UNIT 2[a] PRESSURE LOSS, psid
Inlet Nozzle and 90° Turn	42.0	5.7	5.7
Lower Plenum	7.5	9.6	9.6
Core	15.5	16.2	16.2
Core Outlet to Outlet Nozzle	46.8	6.7	6.7
TOTAL		38.2	38.2

(a) The values presented are valid for 2737 MWt, 2250 psia, 396,125 gpm, 548°F inlet temperature, and a full core of AREVA HTP™ fuel. Bounding analyses are also performed for the limiting conditions on power, pressure, flow, and temperature with mixed cores using both HTP™ and Turbo fuel.

3.6 ORIGINAL FUEL DESIGN EVALUATION

3.6.1 FUEL DESIGN AND ANALYSIS

The fuel rod cladding is designed to satisfy the design bases given in Section 3.2.3.5. The effects of irradiation on UO_2 and cladding materials are considered in the design calculations. The predicted effects of anticipated transients are also considered in the design process.

As stated in Section 3.2.3.5, the fuel rod cladding is designed to the following bases:

Basis 1

Maximum primary stress during steady state operation, expected transients, and depressurization is limited to two-thirds of the minimum yield strength of the material at operating temperature.

Basis 2

Predicted permanent hoop strain of the cladding at the end of fuel life is less than 1.0%.

These bases are conservative and the calculations used to demonstrate their satisfaction were conducted for limiting cases using limiting assumptions. This is considered advisable in the prediction of long-term fuel behavior under irradiation.

Maximum tensile stress in the fuel cladding occurs during a depressurization transient near EOL when internal gas pressure is highest. Clad thickness is such that under the anticipated transient conditions, this stress does not exceed two-thirds of the unirradiated value of yield stress of the clad material at its operating temperature. An unirradiated value is used for conservatism.

The satisfaction of Basis 2, the long-term total strain limit, was demonstrated as follows:

a. Clad stress-strain behavior was based on a stress analysis which includes the effect of creep. The loads considered were those due to fuel thermal and fission growth, fission gas pressure and external coolant pressure.

b. The fuel thermal and fission growth was calculated considering the fuel as a solid rod with unrestrained thermal expansion and a volumetric growth rate of 0.16% for 10^{20} fissions/cm^3 (Reference 1), and a LHR of 17.5 kW/ft. The fission gas pressure was calculated for a 31% fission gas release (which was based on the derivation of Lewis (Reference 2) considering the change in plenum volume due to the thermal expansion and growth of the rod).

c. The analysis was based upon an incremental approach, which divided the three-year fuel life span into discrete time intervals and evaluated the clad stress and strain, including the effect of creep, during these intervals. The relation between the incremental creep and the actual stress state is expressed by the Prandtl-Reuss formulae. The basis for creep is given by the von Mises criterion and the relation between creep rate and generalized stress is that given by Holmes (Reference 3). The rapidly convergent iterative technique was employed to solve the resulting non-linear equations.

d. For the nominal fuel-to-clad gap, at about 775 hours after BOL, the fuel has expanded to completely fill the fuel-to-clad gap and to restore the clad to a circular shape after its initial collapse onto the fuel. The fuel was subsequently assumed to swell unrestrained with the clad following. Based upon this conservative assumption, the final strain after three years service was 0.5%. That is, for

average fuel-to-clad gap at peak power density, Basis 2 was satisfied without credit for fuel strain under load.

e. For the most adverse initial condition, i.e., minimum clad ID, maximum pellet OD coincident with the point of maximum power density which was assumed to be sustained over lifetime, application of the unrestrained fuel growth model resulted in a computed strain at the end of the third cycle (EOC3) of 0.8%. However, as is well known (References 4, 5, and 6), the effect of restraint from the exterior, cooler regions of the fuel pellet, the clad, and the external pressure result in a significant limitation on radial swelling with corresponding flow of pellet material into the dish provided.

These analyses were conducted throughout with design BOL power density, although it was known that for fuel in its third burnup cycle, LPD would be substantially below these values. Thus, the LPD increase which might be associated with overpower transients near end of fuel life was conservatively considered. Further consideration of EOL power density is provided in subsequent paragraphs together with a summary of data justifying the maximum linear heat ratings and peak burnups. Table 3.6-1 contains typical maximum linear heat ratings as a function of burnup. The maximum linear heat rating for the first core was 17.5 kW/ft at BOL. The maximum heat rating near EOC3 was 14.9 kW/ft, resulting in a BOL/EOC3 ratio of 1.18. This was greater than the value of 1.12 for the ratio of maximum transient to steady state heat ratings. Thus, use of BOL power densities in these calculations for EOC3 transients provided considerable margin.

Studies by Notely, et al (References 5 and 6) in which 27 fuel elements were irradiated without failure, reported measured clad strains up to 3.33%.

In a series of experimental element irradiations, Westinghouse (Reference 4) reported strain values at failure for Zr-4 clad fuel elements of 0.78 to 2.6% depending on the fuel properties assumed. Also, Lustman (Reference 7) noted that failures in pile have occurred at strain values between 0.5 to 1.0%. However, these results are based on relatively low Zr-4 cladding temperatures as compared to contemporary, large, commercial PWRs. It is known (Reference 8) that permissible strain values for Zircaloy increase above 650°F. In the zone of interest, the average Zr-4 cladding temperature is about 720°F; this should result in increased ductility and thus a higher strain limit to failure.

For the AREVA design, compliance was demonstrated using the NRC-approved methodology using the RODEX2 code.

3.6.2 ANALYSIS OF BURNUP AND LINEAR HEAT RATINGS

Prior to a discussion of the experimental bases for justifying the initial maximum linear heat ratings and burnups, it is necessary to relate these parameters so that they may be viewed in the proper perspective. The maximum linear heat rating was reached but not exceeded only during approximately the first 28,000 MWD/MTU of peak burnup. The maximum linear heat rating decreased with additional burnup beyond this value.

Typical values at the time of initial design are shown in Table 3.6-1, which contains an analysis of burnup, total nuclear peaking factors, and the corresponding maximum linear heat rating (including consideration of the combination of total nuclear and mechanical peaking factors), for the most adverse equilibrium core.

Table 3.6-2 contains a comparison of maximum heat ratings for a number of plants of that period. Peak linear heat ratings for this plant were consistent with current practice and were considered as slightly conservative with respect to a number of the designs.

Although it was believed that fuel rods could operate satisfactorily with a small amount of fuel melting, the initial design did not permit fuel melting even under conditions imposed by anticipated transients. Cycle 1 design offered considerable margin with respect to the core linear heat rating of 24 kW/ft for melting (BOL value; typical EOC3 value was about 23 kW/ft), even when expected transients (112%) were considered.

3.6.3 SUMMARY OF PERTINENT FUELS IRRADIATION INFORMATION

The LHRs specified in this section are as they appeared in the referenced literature and represent total core heat rates.

3.6.3.1 High Linear Heat Rating Irradiations

The determination of the effect of linear heat rating and fuel-cladding gap on the performance of Zircaloy-clad UO_2 fuel rods was the object of two experimental capsule irradiation programs conducted in the Westinghouse Test Reactor (WTR) (Reference 9). In the first program, 18 rods containing 94% TD UO_2 pellets were irradiated at 11, 16, 18 and 24 kW/ft with cold diametral gaps of 0.006", 0.012" and 0.025". The wall thickness to diameter ratio (t/OD) of the Zircaloy-cladding was 0.064 which is slightly higher than the value of 0.059 of Cycle 1. Although these irradiations were short duration (about 40 hours), significant results applicable to Cycle 1 design were obtained. No significant dimensional changes were found in any of the fuel rods. Only one rod, which operated at 24 kW/ft with an initial diametral gap of 0.025", experienced center melting. Rods which operated at 24 kW/ft with cold gaps of 0.006" and 0.012" did not exhibit center melting on these bases. The initial gap of 0.0085" and the maximum linear heat ratings for this design (Table 3.6-1) provided adequate margin against center melting even when 12% overpower conditions were considered. These results also indicated that an initial diametral gap of 0.0085" was adequate to accommodate radial thermal expansion without inducing cladding dimensional changes, even at 24 kW/ft. This margin with respect to thermal expansion, decreased with increasing burnup at a rate of 0.16% ΔV per 10^{20} fissions/cm^3. However, the linear heat rating also diminished with burnup (Table 3.6-1). Since the diametral thermal expansion (assuming BOL maximum heat ratings) is almost twice as great as the swelling diametral growth (on the EOC3 burnup), these data added considerable weight to the conservative treatment of the influence of transients on fuel element integrity.

Further substantiation of the capability of operation at maximum linear heat ratings in excess of those in the first cycle design was obtained from later irradiation tests in WTR (Reference 9). Thirty-eight-inch long and 6" long fuel rods were irradiated at linear heat ratings of 19 kW/ft and 22.2 kW/ft to burnups of 3450 and 6250 MWD/MTU. The cold diametral gaps in these Zircaloy-clad rods containing 94% dense UO_2 were 0.002", 0.006" and 0.012". The cladding t/OD was 0.064. No measurable diameter changes were noted for the 0.006" or 0.012" diametral gap. Only small changes were observed for the rods with a 0.002" diametral gap.

Additional successful radiations had been performed with SS cladding in Saxton at 23 kW/ft and in Plum Point at 22 to 25 kW/ft.

3.6.3.2 Shippingport Blanket Irradiations

Zircaloy-clad fuel rods operated successfully (three defects had been observed which were a result of fabrication defects) in the Shippingport blanket with burnups of about 37,000 MWD/MTU and maximum linear power ratings of about 13 kW/ft (References 9, 10, and 11). Although higher linear heat ratings at lower burnups would be experienced, swelling (primarily burnup-dependent) and thermal

expansion (linear heat rating dependent) provide the primary forces for fuel cladding strain at the damage limit. Thus, Shippingport irradiations demonstrated that Zircaloy-clad rods with a cladding t/OD comparable to that for this plant (0.059) should successfully contain the swelling associated with 37,000 MWD/MTU burnup, while at the same time containing the radial thermal expansion associated with heat ratings of the time. Irradiation test programs in support of Shippingport in in-reactor loads demonstrated successful operation of burnups of 40,000 MWD/MTU and linear heat ratings of about 11 kW/ft with cladding t/OD ratios as low as 0.053 (compared with 0.059 for this plant) (Reference 12).

3.6.3.3 NRX Irradiations (AECL - Canada)

Eleven Zircaloy-clad, large diameter fuel elements (approximately .750" OD) with clad thicknesses of .016", .024", and 0.037" (t/OD = .021, .031, and .047 corresponding to TD percentages of 94.3, 94.3 and 93.7, respectively) were irradiated in the NRX pressurized loop facility of AECL, Canada (Reference 13) at loop pressures of 2000 to 3000 psi. The cold diametral gaps for the test elements were .0035" and .0040", and the fuel was UO_2 sintered pellets (0.700" diameter) loaded in an argon atmosphere.

The elements were operated for 535 full power days to an average burnup of 10,280 MWD/MTU at a maximum linear power output of 14.8 kW/ft. These elements experienced 308 power cycles. No failures were reported for these elements, and the final dimensions of the rods were reported to be virtually unchanged from pre-irradiation values.

The successful operation of these elements with considerable lower clad-to-diameter ratios than those for Cycle 1 demonstrated the capability of safe operation of Zircaloy-clad elements with thin cladding for many power cycles.

Additional tests on similar elements were then in progress at NRX involving test elements with UO_2 and (U, Pu) O_2 (PuO_2 = 2.4 wt%) at average linear heat ratings of 11.4 and 17.2 kW/ft. Those elements had accumulated burnups of 6,400 and 28,700 MWD/MTU without failure.

3.6.3.4 Saxton Irradiations

UO_2-PuO_2 fuel rods containing pellets of 94% TD and clad with Zircaloy-4 had been successfully irradiated in Saxton to burnups approaching 25,000 MWD/MTU at 16 kW/ft under USAEC Contract AT(30-1)-3385 (Reference 14). The t/OD of the cladding was 0.059 which is equivalent to the Cycle 1 design. The amount of PuO_2, 6.6%, was considered as insignificant with respect to providing any differences in performance when compared with that for UO_2. In fact, the higher thermal expansion coefficient for this PuO_2-UO_2 composition than that for UO_2 would induce greater cladding strain under equivalent irradiation conditions. Subsequent tests on two of the above rods (18,600 MWD/MTU at 10.5 kW/ft) successfully demonstrated the capability of these rods to undergo power transients from 16.8 kW/ft to 18.7 kW/ft.

3.6.3.5 Vallecitos Boiling Water Reactor - Dresden

The combined Vallecitos Boiling Water Reactor (VBWR) - Dresden irradiation of Zircaloy-clad oxide pellets (Reference 15 and 16) provided additional confidence with respect to the design conditions for the fuel rods for Cycle 1 core. Ninety-eight rods irradiated in VBWR to an average burnup of about 10,700 MWD/MTU were assembled in fuel assemblies and irradiated in Dresden to a peak burnup

greater than 48,000 MWD/MTU. The reported maximum heat ratings for these rods was 17.3 kW/ft, which occurred in VBWR. The t/OD cladding ratio of 0.052, pellet TD of 95%, and the external pressure of about 100 psi are conditions which are all in the direction of less conservatism with respect to fuel rod integrity when compared with the design values of 0.059 cladding t/OD ratio and an external pressure of 2250 psi. Ten of these VBWR - Dresden rods representing maximum combinations of burnup, heat rating and pellet density had been selected for detailed destructive examinations as part of an AEC program. The remaining 88 rods were returned to Dresden and successfully irradiated to the termination of the program.

3.6.3.6 Large Seed Blanket Reactor Rods

Two rods operated in the B-4 loop at the Materials Testing Reactor provided a very interesting simulation for contemporary PWR designs (Reference 4, 17, and 18). Both rods were comprised of 95% TD pellets with dished ends clad in Zircaloy. The first of these, No. 79-2, was operated successfully to a burnup of 12.41×10^{20} f/cc (approximately 48,000 MWD/MTU) through several power cycles which included linear power from 5.6 to 13.6 kW/ft. The second fuel pin, No. 79-25, operated successfully to 15.26×10^{20} f/cc (approximately 60,000 MWD/MTU). The basic difference in this rod was the 0.028" wall thickness, as compared to 0.016" (t/OD 0.058) in the first rod. All other parameters were essentially identical. The linear heat rating ranged from 7.1 to 16.0 kW/ft. After the seventh interim examination, the rod operated at a peak linear power of 12.9 kW/ft at a time when the peak burnup was 49,500 MWD/MTU. These high burnups were achieved with fuel elements which were assembled by shrinking the cladding onto the fuel. This indicated that a comparable irradiation of the fuel elements for this reactor would allow a considerable increase in swelling life at a given clad strain.

One additional rod irradiated in Materials Testing Reactor as part of the Large Seed Blanket Reactor (LSBR) series (rod 79-18) demonstrated the effect of clad restraint on the swelling behavior of a UO_2-Zircaloy-clad rod (Reference 19). A starting fuel density of 81.4% of theoretical was used in conjunction with a zero cold gap and a 0.060 cladding t/OD ratio. The rod was irradiated to 49,000 MWD/MTU with no measurable change in rod diameter.

3.6.3.7 Central Melting in Big Rock

As part of a Joint U.S. - Euratom Research and Development Program, Zircaloy-clad UO_2 pellet rods with 95% of TD had been irradiated under conditions designed to induce central melting in the Consumers Big Rock Point Reactor (Reference 20). The test included 0.7" diameter fuel rods (cladding t/OD = 0.061, fuel-to-clad gap of about 0.011") at maximum linear heat ratings of about 27 kW/ft and 22 kW/ft with peak burnups up to 20,000 MWD/MTU. Results of these irradiations provided a basis for incorporating linear heat ratings well in excess of those calculated for this reactor (Reference 21). These results showed that the presence of localized regions of fuel melting were not catastrophic to the fuel assembly.

3.6.3.8 Peach Bottom 2

General Electric (GE) had successfully irradiated fuel pins of the Peach Bottom 2 design to burnups in excess of 42,000 MWD/MTU at peak linear heat ratings of 23 kW/ft. An interim examination at 32,500 MWD/MTU indicated a satisfactory condition (Reference 22).

3.6.4 EVALUATION

It was concluded from the above information that heat ratings as high as 23 to 24 kW/ft could be achieved in the fuel elements without fuel centerline melting. Linear heat ratings in the Cycle 1 core design fell significantly below this limit even at the 112% overpower condition.

Heating ratings and burnups for this design were well demonstrated by the existing technology. Nevertheless, it was felt fruitful to consider the question of what constitutes a fuel element failure. For one, the cladding must be violated. On the subject of the influence of expected transients, a conservative analysis had been presented of the factors which influence cladding performance during such transients. The fuel rod cladding was designed on a conservative basis and the calculations considered limiting cases and limiting assumptions. Consideration of peaking factor reductions shown in Table 3.6-1 increased the conservatism of these analyses.

The analyses had been conducted throughout with design BOL power density, although it was known that for fuel in its third burnup cycle, LPD would be substantially below these values. Thus, the LPD increase which might be associated with overpower transients near end of fuel life had been conservatively considered. Cladding integrity had been demonstrated even under these adverse conditions. Consideration of peaking factor decreases noted in Table 3.6-1 made this analysis even more conservative.

Present heat rating limits are based on LOCA/Emergency Core Cooling System stored energy considerations and are included in Section 14.17.

3.6.5 REFERENCES

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TABLE 3.6-1

TYPICAL PEAK BURNUP - MAXIMUM HEAT RELATIONSHIP

MAXIMUM LOCAL EXPOSURE MWD/MTU	TOTAL NUCLEAR PEAKING Factor	MAXIMUM HEAT RATING kW/ft
24,200	2.86	17.5
24,200 - 36,000	2.86	17.5
36,000 - 48,500	2.42	14.9

TABLE 3.6-2

COMPARISON OF MAXIMUM HEAT RATINGS

REACTOR	kW/ft
Maine Yankee	16.7
Fort Calhoun	17.1
Calvert Cliffs, Unit 1	17.5
Calvert Cliffs, Unit 2	17.5
Hutchinson Island, Unit 1	17.8
Millstone Unit 2	17.8
Turkey Point	17.3
Surrey	17.5
Prairie Island	17.4
Three Mile Island	17.5
Oconee	17.5
Indian Point, Unit 2	18.5
Diablo Canyon	18.9
Browns Ferry	18.5
Sequoyah	18.8
San Onofre, Units 2 and 3	18.5

3.7 SUPPLEMENTARY FUEL DESIGN AND EVALUATION

3.7.1 FUEL ROD DESIGN EVALUATION

3.7.1.1 Mechanical Design Evaluation

a. Clad Creepdown/Creep-Collapse

Historical Perspective

Clad creepdown is the phenomenon caused by inward stresses on the cladding (caused by the difference in external and internal pressure) in combination with effects from temperature and neutron fluence. If the clad slowly ingresses toward the pellet stack, it would reduce gap size and increase gap conductance. Densification of fuel pellets leads to the formation of axial gaps in the fuel stacks and a loss of support for the cladding in these locations. Creepdown and subsequent collapse into axial gaps induced by fuel densification is called creep-collapse.

The minimum time to collapse for CE Zircaloy-clad fuel was calculated by the CEPAN computer code (Reference 13). The experimental database used for modeling creep collapse consists of measurements made on fuel rods irradiated in a CE reactor. The analytical creep correlation used in the model was fit to this data. That correlation leads to the time to collapse predictions given by CEPAN.

Beginning with Unit 1 Cycle 6, improvements were made in the modeling technique. These improvements (Reference 26) revised the method for establishing uncertainties in cladding geometrical parameters (diameters, thicknesses, etc.) used in the collapse analysis, and provided new criterion for the occurrence of collapse.

Present Analysis

Analysis of the phenomena of interpellet gap formation and clad collapse in modern PWR fuel rods (i.e., nondensifying fuel in prepressurized tubes), demonstrates that the collapse time for modern fuel is significantly larger than its expected useful life. This conclusion is discussed in an EPRI-sponsored report (Reference 28) and is based upon both empirical data covering several vendors' fuel and an analytical evaluation of the propensity for clad collapse into a postulated gap of finite length. Based upon the conclusion and recommendation of this report, cycle-specific clad collapse analyses are not necessary for modern CE manufactured fuel. A cycle-specific calculation was not prepared beginning with Unit 1 Cycle 8, and Unit 2 Cycle 7.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the RODEX2 fuel rod analysis code is used to evaluate the cladding creepdown and creep collapse for AREVA fuel. Creep collapse and the subsequent potential for fuel failure are avoided in the fuel system design by eliminating the formation of axial gaps in the fuel column. AREVAs licensing criterion for preventing cladding collapse is to maintain a radial gap large enough to prevent pellet hang up and, therefore, axial gap formation. The maximum cladding circumferential creep and ovalization, up to the time of maximum densification, are computed to demonstrate that a radial gap between pellet and cladding is maintained. The evaluation is performed using the approved RODEX2 code (References 33 and 34) and the COLAPX code on a cycle specific basis beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21. The RODEX2 code is used to provide initial in-reactor fuel rod

conditions to COLAPX. The COLAPX code calculates the cladding ovality changes (flattening) and creep deformation of the cladding as a function of time. M5® properties were incorporated, as appropriate, into these codes in Reference 36.

b. Fuel Rod Bowing

Fuel rod bowing is the phenomenon whereby a curvature of the rod is experienced, changing the thermal-hydraulic and neutronics characteristics of the region, and potentially affecting the mechanical performance of the fuel. It is primarily caused by a combination of rod axial growth and spacer grid restraint. Rod axial growth is largely the result of normal Zircaloy/ZIRLO growth, although some enhancement due to stresses caused by Pellet-clad Interaction (PCI) is possible. Fuel design changes to decrease the effects of PCI (chamfering, reduced length/diameter, etc.) tend to reduce PCI contributions to rod axial growth, and to bowing of the rod.

Fuel rod bowing leads to variations in the flow characteristics and neutronics of the affected region. Neutronic changes due to enhanced/decreased moderation (depending on the direction of bow) and, in the case of bowed BPRs, enhanced/decreased thermal neutron absorption, lead to changes in local LHRs. Flow changes due to opening/closing of channels can lead to changes in the margin to DNB. Both of these effects are analyzed in Reference 14, and are shown to be within existing margins.

Mechanical performance of the fuel itself can be affected by rod bowing. Rods bowed toward each other may come into contact. Restriction of flow in this region can enhance clad corrosion. Flow induced vibration of the rods may cause fretting wear. Present fuel designs are such that bowing to this extent is not experienced. Only a small reduction in channel size is seen, as compared to the channel closure necessary for mechanical degradation.

The effects of fuel rod bowing on DNBR are discussed in Section 3.5.3.2.b.8.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the methodology described in Reference 35 was used to evaluate fuel rod bowing for the AREVA assemblies.

c. Shoulder Gap Closure

The fuel assembly shoulder gap is defined as the axial gap between the top of a fuel rod and the bottom surface of the upper end fitting. During irradiation, the gap becomes smaller due to differences in irradiation-induced growth and thermal expansion of fuel rods and guide tubes (fuel rod growth has an interactive component related to pellet-clad interaction). Complete closure of the gap would result in additional stresses on the fuel, enhancing rod bowing. Therefore, it is important to ensure that the BOL gap is large enough to preclude gap closure by the EOL of the fuel. The model for Zircaloy growth is presented in Reference 15. Reference 32 discusses the impact of ZIRLO clad. Shoulder gap is reviewed using SIGREEP (Reference 31).

AREVA fresh fuel reloads will use M5® clad fuel rods with Zircaloy-4 guide tubes. The growth of the fuel rods is assessed using M5® cladding growth correlations. The upper bound fuel rod growth is considered in conjunction with the lower bound assembly growth along with the manufacturing tolerances that would result in the minimum fabricated clearance.

3.7.1.2 Fuel Thermal Design Evaluation

a. Introduction

The Combustion Engineering fuel rod thermal Analysis code, FATES (Reference 17), is used in the fuel evaluation model to predict fuel rod temperature distributions, fuel-clad gap conductance, rod internal pressures, and fuel rod stored energy. The effects of fuel densification and the subsequent formation of axial gaps are taken into account to calculate augmentation factors used to modify linear heat generation rate values. The densification process itself is modeled. Fission gas production is predicted by FATES and, using the internally-modeled temperatures, fission gas release is predicted as well.

Reference 17 compares FATES results with experimental data from in-reactor and out-of-reactor tests to show the conservatism in the model as well as to show the validity of the modeling techniques used.

Beginning with Unit 1 Cycle 6, an improved version of FATES, entitled FATES3 (Reference 27) was used for fuel thermal design evaluation. While a great number of the models remained unchanged, revisions included the models for 1) fission gas release, 2) fuel swelling, 3) closed gap conductance, 4) fuel relocation, 5) cladding axial growth [a calculation previously performed via Reference 15], and 6) plenum gas temperature. Additionally, the code was modified to include an annular fuel pellet geometry. This modeling modification required changes to the models for fuel pellet temperature distribution, fuel pellet thermal expansion, and rod internal void volume calculations.

Beginning with Unit 1 Cycle 9, an improved model, FATES3B (Reference 29), is used for fuel thermal design evaluation. FATES3B incorporates changes to the fission gas release model, specifically related to burnup dependence, kinetics of grain growth, and fission gas release calculation.

Reference 32 documents a modification to FATES3B for implementation of ZIRLO clad material.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the approved AREVA fuel rod thermal analysis code, RODEX2 (References 33 and 34), is used to evaluate AREVA fuel. The RODEX2 code incorporates models to describe the gas generation and release, swelling, densification, and cracking in the pellet, gap conductance, radial thermal conduction, free volume, gas pressure internal to the fuel rod, fuel and cladding temperatures and deformations, and cladding corrosion. The calculations are performed on a time incremental basis with conditions being updated at each calculated increment.

b. Fuel Densification and Swelling

1. Fuel Densification

The FATES model includes correlations to account for burnup-dependent fuel densification. This phenomenon is different from fuel densification caused by high temperatures and thermal gradients experienced by the fuel (Reference 18). The rate and extent of burnup-dependent densification varies with original fuel density and microstructure. Original Calvert Cliffs fuel was of a relatively unstable, densifying type. Subsequent design changes have resulted in a more stable, non-densifying fuel.

The primary concerns regarding fuel densification were:

a) Decrease in pellet diameter (increased gap size) which lowers gap conductivity and increases fuel temperatures and stored energy.

b) Decrease in pellet length, which increases the Linear Heat Generation Rate along the fuel rod.

c) Decrease in pellet length coupled with pellet cocking, which leads to axial gaps in the fuel pellet stack. Augmentation factors for LHGR were based on the formation of these axial gaps.

Burnup-dependent densification is a factor only at BOL of fuel. The terminal density is predicted to be achieved at a burnup of 4000 MWD/MTU. The terminal density is determined in one of two ways, depending on the type of fuel (densifying/non-densifying) (Reference 17).

The effects of densification on the Safety Analyses are presented in Reference 19. As the fuel will begin to swell after 4,000 MWD/MTU, some of the effects of densification (axial and radial shrinkage) will be reversed by the swelling process.

The RODEX2 fuel rod analysis code, used beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21 for AREVA fuel, contains both time and burnup dependent fuel densification correlations which are applied in the densification model for Uranium and Gadolinia fuel. The densification of the fuel is a phenomenon that is only observed at BOL, this is reflected in both the burnup and time dependent densification correlations. The densification model calculates the change in volume per unit volume or dilatation of the fuel material along with radial displacement, axial fuel stack length change, and change in the void fraction.

2. Fuel Swelling

Fuel swelling refers to the change in pellet volume which occurs as a result of the buildup of porosity and accumulation of fission products with increasing burnup. During the first 4,000 MWD/MTU of exposure, the densification mechanism prevails, but after the point where the fuel reaches its terminal density, the fuel begins to expand. Fuel swelling is said to be unrestrained until the time of pellet-clad contact, after which it is said to be restrained. The FATES model (Reference 17) contains a correlation for the rates of diametral and axial swelling while swelling is unrestrained. A new, restrained swelling rate is assumed after contact.

The FATES3 model (Reference 27), used beginning with Unit 1 Cycle 6, incorporated a lower swelling rate than that incorporated into the original FATES model. This new rate is based on results of recently published data and post-irradiation measurements made on Calvert Cliffs Unit 1 fuel.

The RODEX2 fuel rod analysis code, used beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21 for AREVA fuel, has models that account for the phenomenological swelling processes (solid swelling and gaseous swelling). The swelling of the fuel material contributes to the radial deformations, axial fuel column length changes, filling of dish volumes, and is related to the fabricated porosity and available crack volume; the models incorporated into RODEX2 take into account such design variables. The swelling models in RODEX2 also take into account restraint of the fuel (due to pellet-to-clad contact) as well as an incubation period in which nondensified porosity is utilized by swelling.

3. Fuel Pellet Relocation

During irradiation, the fuel pellet cracks radially (and reheals in a distorted shape) and the pellet pieces approach the clad, decreasing gap size and enhancing gap conductivity. This improved heat transfer due to relocation reduces centerline temperature and stored energy.

The FATES3 model incorporates a modeling change with regard to fuel relocation after the pellet-clad gap has closed.

Additionally, the FATES3 model explicitly treats the pellet-clad interface, allowing for calculation of pellet-clad interfacial pressure and gap conductance after contact occurs. The previous FATES model used preassigned maximum values for these parameters after pellet-clad contact occurred.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the RODEX2 fuel rod analysis code is used for fuel thermal design evaluations of AREVA fuel. The RODEX2 code contains models that account for fuel relocation prior to and after contact between pellet and cladding. The radial displacements caused by pellet cracking are factored into calculations of the effective width of the open gap, which in turn is used to establish the gap conductance.

c. Linear Heat Generation Rate Augmentation Factor

Historical Perspective

One former concern regarding fuel densification was the decrease in pellet length. This allegedly decreases the overall pellet stack length, increasing the LHR. Presumably the pellets will tend to settle at the bottom of the rod, but interference from clad creepdown and pellet cocking and lockup may lead to the formation of axial gaps in the fuel column. It was assumed that these gaps could cause local power peaking in that axial region in surrounding rods, because the loss of neutron absorption in the gap outweighs the loss of fission. If the clad collapses completely into the gap (less likely in pre-pressurized fuel), local peaking is enhanced further due to the replacement of the gap void with moderator.

Present Analysis

The local power peaking is dependent on the number of gaps as well as the size of gaps, both of which are modeled in FATES. A peaking factor was determined, called the augmentation factor, which is defined as the ratio of peak augmented power to peak unaugmented power. The peak augmented power was determined statistically so that there is a 95% certainty that no more than one rod will be at a higher power.

Analysis of the phenomenon of interpellet gap formation in modern PWR fuel (Reference 28) demonstrates that the increased power peaking associated with the small interpellet gaps found in modern, i.e., pre-pressurized and non-densifying, fuel is insignificant compared to the uncertainties in the safety analyses and Technical Specifications. Consequently, augmentation factors used for interpellet gap formation were eliminated from the analysis beginning with Unit 1 Cycle 8 and Unit 2 Cycle 7.

The RODEX2 fuel rod analyses, performed for AREVA fuel beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, use a conservative engineering heat flux augmentation factor to perform the fuel thermal design evaluations (Reference 35). The factor is based on a 95/95 statement including pellet and pellet lot variations in enrichment, as-sintered pellet density, and pellet diameter. The factor includes conservative allowances for in-reactor densification.

d. Fission Gas Release

1. Fission Gas Generation

Products that remain within the fuel matrix after fissioning of U-235 include primarily unstable isotopes with mass numbers ranging from 72 to 160. Each of these will experience an average of three stages of radioactive decay before being converted into a stable nucleus. This results in over 200 isotopes of 30 or more different elements present as fission products within the fuel pellets. Xenon and krypton, two of the stable gaseous elements liberated from the fuel matrix, are assumed to comprise the fission gas. FATES models the amount of fission gas generated in the fuel. The fuel rod is divided into axial nodes, and the gas generated in that node is calculated as a function of local burnup.

Beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21, the RODEX2 fuel rod analysis code is used for fuel thermal design evaluations of AREVA fuel. The RODEX2 code uses a bounding, power dependent fission gas generation rate per unit of energy produced that is applied to each axial region of the fuel column.

2. Fission Gas Release

While gas generation is strictly a function of burnup, the release of the fission gas from the fuel matrix is dependent on temperature and on temperature gradient. At low temperatures, recoil and knockout are the primary release mechanisms, as the relatively low energy of the gas precludes diffusion-type movement in the matrix. Recoil release is the direct release from the matrix to the free space directly as an energetic fission fragment, and knockout release is release resulting from the impact of another energetic fission fragment. At higher temperatures, the diffusion of gases within the grains is significant, and by diffusing to grain boundaries, gases can escape to cracks or to porosity already

present in the fuel matrix. At high temperatures, gas bubbles are driven along the thermal gradient and released to the free space.

Grain growth also plays an important part in fission gas release, and is itself strongly temperature dependent. At fairly high temperatures the pores initially present in the fuel begin migrating along the thermal gradient toward the center of the pellet. These pores leave a trail of small gas bubbles which form the boundary of a columnar grain extending radially from the pellet center. As the pores move inward they collect fission gas from the fuel matrix and grain boundaries, and eventually deposit it in the center of the pellet, forming, with other pores, the central void. Migration of these pores leave behind a crystal structure which is more dense than the original microstructure. The growth rate and extent of growth of columnar grains plays an important part in the fission gas release mechanism.

FATES modeled fission gas release in the following manner. The columnar grain growth boundary temperature T_g was determined by the equation:

$\int_{400°C}^{T_g} k_{95} dT$ = 42W / cm (Reference 17)

where: k_{95} is the thermal conductivity of 95% td UO_2

The FATES model combined the work of Notley and MacEwan with that of Lewis.

The original Lewis model is of the form:

$$\%Release = \frac{a\int_{T_s}^{1000} k_{dt} + b\int_{1000}^{1300} k_{dt} + c\int_{1300}^{1000} k_{dt} + d\int_{1000}^{T_c} k_{dt}}{\int_{T_s}^{T_c} k_{dt}}$$

where k is a function of temperature.

Notley and MacEwan showed the effect of columnar grain growth on release. The temperature bands in the Lewis model were retained in the FATES model, but the limits of the bands were changed to incorporate the temperature bands defined for the growth of columnar grains. A burnup-dependent correction factor was applied to the first three integrals of the FATES model to account for the gas diffusion mechanism which prevails in regions operating below the columnar grain growth boundary temperature.

The release function was calculated for each axial node. To obtain the accumulated fission gas release, the axial fractional releases were summed. No allowances were made in FATES for re-absorption of fission gas or any re-entry into the fuel matrix once the gas has been released.

A comparison of FATES predictions with experimental data is presented in Reference 17. Overall results are shown to be conservative.

Beginning with Unit 1 Cycle 6, the FATES 3 (Reference 27) model for fission gas release was used. The burnup and fuel microstructural (grain size) effects on fission gas release were more implicitly modeled in FATES 3. A restriction was placed on the effective grain size of the

fuel for FATES 3 analyses. This restriction was burnup-dependent, and had no effect on the model at low burnups. At higher burnups, the restriction acted by incorporating a smaller grain size into the gas release model yielding higher predicted releases, which was conservative.

Beginning with Unit 1 Cycle 9, the FATES3B (Reference 29) model for fission gas release is used. The imposed grain size restriction is removed based on recent high burnup, high temperature fission gas release data. The new model increases the burnup dependence of fission gas release, i.e., it predicts higher releases at high burnup while not significantly affecting lower burnup release predictions. The grain growth model is modified in FATES3B, as is the calculated gas release following grain growth.

The RODEX2 fuel rod analysis code, used beginning with Unit 2 Cycle 19 and Unit 1 Cycle 21 for AREVA fuel, contains a physically based fission gas release model. The model is based on several physical mechanisms, described in References 33 and 34, that are active in producing the release of the fission gas in the fuel. The importance of each mechanism is dependent upon the operational history of the fuel, the fuel design and the structure of the fuel material. The release model involves a two-stage release process of gas being released from the grains and accumulating in the grain boundary region and then this gas being released to the free volumes in the fuel rod as the gas concentration in the boundary increases. The phenomena incorporated into the gas release evaluation model are:

1. Release to the open porosity by a direct recoil mechanism
2. Release to the grain boundary by grain boundary sweeping due to grain growth
3. Release to the free volume due to columnar grain formation
4. Diffusion to the grain boundary controlled by a re-solution barrier
5. Release from the grain boundaries to the interconnected free gas volume when the boundary concentration barrier is exceeded

3.7.1.3 License Conditions with AREVA RODEX2 Methodology

Use of the AREVA RODEX2 methodology is restricted by the following NRC imposed license conditions from Reference 37. These license conditions are required to compensate for the RODEX2 methodology which does not explicitly model degraded fuel thermal conductivity nor adequately account for modeling uncertainties.

a) A reduction of the rod internal pressure limit is required to compensate for the RODEX2 methodology. Cycles which rely on the RODEX2 methodology must ensure that predicted maximum rod internal pressure in AREVA fuel remains below the steady-state system pressure.

b) A reduction of the LHGR fuel centerline melt safety limit is required to compensate for the RODEX2 methodology. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 kW/ft.

3.7.2 DESIGN EVALUATION OF OTHER FUEL ASSEMBLY COMPONENTS

3.7.2.1 Burnable Poison Rod Design Evaluation

a. Introduction

Fixed BPRs (neutron absorbing) are included in selected fuel assemblies to reduce the BOL MTC. They replace fuel rods at selected locations. The poison rods are mechanically similar to fuel rods, but contain a column of burnable poison pellets instead of fuel pellets. The poison pellets consist of alumina with uniformly dispersed boron carbide particles.

b. BPR Hydriding and Bowing

At the end of Unit 1 Cycle 1, it was noticed that a BPR end cap had become detached from its rod in one assembly. A subsequent detailed inspection revealed extensive hydriding of several BPRs, with subsequent failure of some cladding. A significant amount of rod bowing/Zircaloy growth was also seen. Analysis of rods examined showed that they were acceptable for reinsertion for Cycle 2 use as scheduled.

1. Hydriding

Hydride-induced failure occurs primarily due to initial moisture in the pellets. At high temperatures, the water dissociates and hydrogen gas accumulates in the gap. This hydrogen interacts with the Zircaloy-clad, forming hydride blisters, embrittling the clad and allowing subsequent perforation of the clad walls. An analysis of the neutronic and thermal hydraulic effects of hydride blisters, perforation of the clad, and loss of some exposed poison material by erosion, as well as a discussion of the possibility of fuel rod fretting caused by material dispersed in the coolant, is given in Reference 3.

2. Bowing

Bowing in BPRs is largely a result of grid restraint, rod axial growth, and pellets becoming cocked and lodged against the clad wall resulting in localized stresses. The rod axial growth is, in part, the irradiation-induced growth of Zircaloy, but is enhanced by outward pressure on the clad inner surface by poison pellet swelling. Clad creepdown, caused by the inward pressure on the clad due to coolant pressure, limits pellet movement resulting in pellet cocking and lockup. These stresses, together with those resulting from spacer grid restraint, when coupled with axial elongation of the rod itself, result in rod bow.

c. Burnable Poison Rod Improvements

Improved designs were used in manufacturing BPRs subsequent to Unit 1 Cycle 1. Pellet moisture content was maintained at a lower level during manufacturing to prevent hydriding and pellet/clad design was modified (chamfered pellet, smaller pellet, thicker clad wall, larger gap) to decrease PCI which can induce bowing and rod elongation.

3.7.2.2 CEA Guide Tube Evaluation

The CEA guide tubes form part of the structural frame of the fuel assembly and provide channels which guide the CEAs over their entire length of travel. The center guide tube houses the incore instrumentation assemblies and irradiation samples/surveillance capsules in selected assemblies. One of the corner guide tubes housed the neutron source assemblies in each of two peripheral assemblies.

The neutron sources were removed beginning with Unit 1 Cycle 9 and Unit 2 Cycle 8. For Unit 1 Cycle 11 and Cycle 12, GTFSs were inserted into the guide tubes of selected peripheral assemblies.

The guide tubes are constructed of Zircaloy-4 and are integrated into the assembly as described in Section 3.3.2.3. The Zircaloy-4 is softer than the Inconel 625 cladding on the CEA, which can result in significant wear of the guide tube as a result of CEA vibration caused by turbulent coolant flow. As a result, SS sleeves have been installed in fuel assembly guide tubes that show wear or which house CEAs. In Unit 1 Cycle 8 and Unit 2 Cycle 7, a short-sleeve design was implemented which is considered the permanent fix to CEA guide tube wear (Reference 30). All new fuel assemblies will contain the short-sleeve.

The sleeves are constructed of Type 304 SS and extend 15.375" into the guide tubes. They are chrome-plated on the wear surface to provide resistance to wear without promoting wear on the CEA cladding (Reference 5). The sleeving program was initiated with Unit 1 Cycle 3, Unit 2 Cycle 2 and has been maintained since.

Several additional measures have been taken to mitigate guide tube wear. As described in Section 3.3.2.6, PLCEAs were replaced with CEA plugs which retain the dynamic operating characteristics of the region (flow rate, pressure drops), but prevent vibration wear. The fingers of the plugs extend only 5" into the top of the fuel assembly and are positioned by a leaf spring to prevent vibration. The CEA plugs were installed beginning in Unit 1 Cycle 3, Unit 2 Cycle 2 and removed at the end of Unit 1 Cycle 7, Unit 2 Cycle 6.

In Unit 2 Cycle 2, Unit 1 Cycle 4, and Unit 1 Cycle 5, some assemblies were fabricated with small flow hole guide tubes to reduce flow induced CEA vibration (Reference 6). This design has since been discontinued.

AREVA fuel was added to the core starting in Unit 2 Cycle 19 and Unit 1 Cycle 21. The guide tubes are similar to the Westinghouse design described here. The guide tubes are constructed of Zircaloy-4 with 22” wear sleeves constructed of Type 304 stainless steel.

3.7.3 DEMONSTRATION PROGRAMS

3.7.3.1 Introduction

In an effort to provide data on fuel design and performance, several experimental programs have been established. These programs include the SCOUT and PROTOTYPE assemblies, a prototype CEA design, surveillance capsules, a joint CE/EPRI fuel performance evaluation program, a separate CE program involving the irradiation of test fuel rods, and the ANF demonstration assemblies.

3.7.3.2 CE/EPRI Fuel Performance Evaluation

Three Batch B fuel assemblies in the initial core of Unit 1 were modified to provide fuel performance data in a variety of areas. A major modification was the reconstitutability feature to facilitate pin removal for inspection.

One assembly was intended for one cycle of irradiation, while the other two were intended for two and three cycles of irradiation, respectively. Each assembly contained fueled and non-fueled test rods. The non-fueled rods were included to obtain information concerning Zircaloy-clad creep. Steel mandrels took the place of fuel pellets to act as support for the clad, prohibiting complete collapse. The fueled rods were of varying enrichment, pellet geometry, fill pressure, and pellet

microstructure. Pellet density was varied using different pore-formers to determine the effect of pellet microstructure on the densification characteristics of the fuel. Fill pressures were varied from 150 psig to 450 psig to learn the effects of fuel pressure on clad creepdown. Higher enriched fuel rods were included to gain information concerning clad creepdown at higher clad temperatures and the performance of fuel rods operating at higher LHR.

At the end of Unit 1 Cycle 1, all three assemblies were inspected and BT01 and BT02 were disassembled. Pins from each were examined for growth, visual appearances, etc. (Reference 20). BT01 was discharged as planned and BT02 was reassembled and reinserted along with BT03 for Cycle 2. At the EOC 2, BT02 and BT03 were inspected and disassembled and pins from each inspected (Reference 20). BT02 was discharged as planned and BT03 was reassembled and reinserted for Cycle 3. After Cycle 3, BT03 was inspected and disassembled and pins were removed for inspection (Reference 21). Six pins from BT03 were sent for examination in a hot cell and were replaced with six two-cycle rods from BT02. BT03 was then reinserted for Cycle 4. At the EOC 4, BT03 was inspected, disassembled and pins were removed for inspection (Reference 22). BT03 was not returned to the core for Cycle 5, but 13 pins (5 of which were 3-cycle rods and 8 were 4-cycle rods) along with one 2-cycle non-fueled rod from BT02 were inserted into a bundle from another test program (Section 3.7.3.3) for a fifth cycle. Results from analysis concerning densification and swelling, as well as fission gas release, performed on the pins sent to the hot cell, are documented (References 23, 24, and 25).

3.7.3.3 CE Irradiation of Test Fuel Rods

Three Unit 1 Batch D assemblies (D042, D047, D048) used initially in Unit 1 Cycle 2 contained graphite coated rods. Other than graphite coating on the cladding inner surface, there was no difference between the design of the test fuel and that of standard Batch D fuel. Fuel assembly D042 contained only graphite rods, while the other two assemblies each had one-fourth of their rods graphite coated. All three assemblies were irradiated for three cycles, and D047 was reinserted into Unit 1 Cycle 5 for a fourth cycle. It served as a carrier for the 14 test pins from BT03/BT02 (Section 3.7.3.2). D047 was removed at the EOC 5, and D042 was reinserted into Cycle 6 for its fourth cycle of irradiation. D042 was removed at the EOC 6 after its fourth cycle of irradiation.

3.7.3.4 SCOUT Program

One Unit 1 Batch F Assembly (F048) was designated as SCOUT. It was designed as a high burnup demonstration assembly. It contained 15 rods of various non-standard designs (Reference 8), and 5 well-characterized standard design rods for comparison. It was discharged at the EOC 8 after five cycles of irradiation.

3.7.3.5 PROTOTYPE Program

Four Unit 1 Batch G assemblies (G003, G004, G006, G008) were designated as PROTOTYPE assemblies and were initially inserted in Cycle 5. They were designed to obtain data concerning fuel performance on significant numbers of rods of standard and non-standard design in conjunction with the extended burnup program. There are several factors to consider in looking at the ability of fuel to withstand prolonged exposure, such as fission gas release and clad stresses. These factors were the bases of the experimental fuel designs used in PROTOTYPE (Reference 10) as well as in SCOUT. Before returning the PROTOTYPE assemblies to the core for their third cycle of irradiation in Cycle 7, two segmented test rods were removed from one of the assemblies and replaced with two SS rods.

The PROTOTYPE assemblies remained in the core for Cycle 9, their fifth cycle of irradiation. The PROTOTYPE assemblies were removed at the EOC 9.

3.7.3.6 Materials Surveillance Specimens

Three surveillance capsules were inserted in the Unit 1 Cycle 5 core in order to obtain data on the material properties of irradiated Inconel 625. These specimens resided in the center guide tubes of assemblies in high flux areas. One capsule was removed at the EOC 5, another at the EOC 6, and the last one at the EOC 7.

In the initial Unit 1 core, each of the three test assemblies (Section 3.7.3.2) contained a surveillance capsule in the center guide tube. These capsules contained Zircaloy and Zr-Mo-Si alloy specimens. One capsule was removed after one cycle, and another after two cycles. The third and final capsule was removed at the EOC 6.

3.7.3.7 Prototype CEA

A prototype CEA was installed in Unit 2 at the BOC 3. The changes from standard design included a change in cladding material (from Inconel to SS), reconstitutable fingers, and a change in material for the tips of the poison rods from Ag/In/Cd to B_4C. The size of the B_4C pellets used in the tips was decreased from the pellet size used for the remainder of the rod length. A metal liner was added to prevent any B_4C fragments from collecting in the high flux tip.

3.7.3.8 ANF Demonstration Assemblies

Four new demonstration assemblies, designated as Batch 1MX, were loaded into Unit 1 Cycle 10. These assemblies were manufactured by ANF and contain gadolinium as the burnable poison material. Each assembly consists of 164, 4.08 wt% U-235 enriched fuel pins and 12 fuel-bearing gadolinium poison pins. The poison pins contain natural uranium and 10 wt% Gd_2O_3. The mechanical configuration of the MX assemblies is essentially the same as the other Batch 1M assemblies. The purpose of installing the ANF assemblies in the Unit 1 core for Cycle 10 is to qualify an alternate source of supply for 24-month fuel assemblies. The ANF assemblies were reinserted into Unit 1 Cycle 11 for their second cycle of irradiation.

3.7.3.9 Erbium Demonstration Assemblies

For Unit 2 Cycle 9, four demonstration assemblies containing Erbium as a burnable absorber were introduced. Each assembly consists of 80 standard pins at 4.3 wt% U-235, 52 standard pins at 3.4 wt% U-235, and 44 Erbium bearing pins. The fuel stack in each erbium bearing fuel pin consists of a central 115.7" region containing 3.4 wt% of U-235, 0.9 wt% Er_2O_3, UO_2/Er_2O_3 pellets and two 10.5" cutback regions, one at each end of the stack containing standard 3.4 wt% U-235 pellets. The major incentives of erbium as a burnable absorber is an increase in core thermal margin and lower local peaking through the decrease in nonfuel bearing discrete burnable absorbers. The erbium assemblies were reinserted into Unit 2 Cycle 10 for their second cycle of irradiation.

3.7.3.10 Test Capsule Assemblies

The Test Capsule Assembly Program is being conducted to evaluate the effects of irradiation at reactor temperatures on materials being considered for advanced spacer grid spring designs.

In Unit 1 Cycle 12, three test capsules were placed in the outer guide tubes of three separate once-burned fuel assemblies.

Four test capsules were placed in the outer guide tubes of four separate fresh fuel assemblies in Unit 1 Cycle 13. Two of these capsules were from Unit 1 Cycle 12 and two are new capsules.

Two test capsules were placed in the outer guide tubes of two separate fuel assemblies in Unit 1 Cycle 14. Both of these capsules were from Unit 1 Cycle 13. Test capsules TCA-3 and TCA-5 were discharged at the end of Unit 1 Cycle 14.

3.7.3.11 Lead Fuel Assemblies for Unit 2 Cycle 11

Four Lead Fuel Assemblies were loaded in Unit 2 Cycle 11. The Lead Fuel Assemblies, designated Batch 2NT, have larger pellet diameter, larger pellet length, slightly reduced clad thickness, greater stack height density and shorter rod length. Zircaloy-2P is used for 216 fuel rods. The remaining rods are standard Zircaloy-4 clad. The Batch 2NT LFAs were re-inserted for a second cycle of operation in Unit 2 Cycle 12 and were discharged to the spent fuel pool after completion of that cycle.

3.7.3.12 Batch 1RT Lead Fuel Assemblies

Four Lead Fuel Assemblies (LFAs) designated as Batch 1RT were loaded into Unit 1 Cycle 13. These assemblies all have a larger pellet diameter, larger pellet length, slightly reduced clad thickness, greater stack height density and shorter rod length.

Five cladding variants were used in 176 rods in two of the Batch 1RT Lead Fuel Assemblies (12 rods of Zircaloy-2P, 20 rods of Zircaloy-4F, 60 rods of Zirconium Alloy E, 24 rods of Zirconium Alloy C, and 60 rods of Anikuloy™).

Two of the Batch 1RT LFAs utilize an advanced assembly design. From the bottom to the top of the assembly, the advanced assembly design contains the laser welded straight strip GUARDIAN™ grid, two laser welded straight strip intermediate spacer grids, four laser welded straight strip mixing grids, and a laser welded straight strip end grid. The advanced assembly design also contains an advanced spring design and a locking guide tube to upper flow plate design.

In Unit 1 Cycle 14, the Batch 1RT LFAs were carried over for their second cycle of irradiation. The LFA set was split up and asymmetrically loaded into the Unit 1 Cycle 14 core. The asymmetric loading (2 LFAs on the core periphery, and 2 LFAs in the core interior) was performed to gather data about Batch 1RT LFA performance on the core periphery.

In Unit 1 Cycle 15, Batch 1RT LFAs (1RT1 and 1RT3) were located on the core periphery for a second cycle in a row to gather data about Turbo test grids performance. Batch 1RT LFAs (1RT2 and 1RT4) were discharged to the spent fuel pool following Unit 1 Cycle 14.

In Unit 2 Cycle 14, the reconstituted LFA 1RT4 was located on the core periphery to gather data about Turbo test grids and advanced cladding material performance. The original LFA 1RT4 was reconstituted with fuel rods from LFA 1RT2, after Unit 1 Cycle 14. The original LFA 1RT4 assembly was reconstituted because the corrosion performance of Anikuloy, Zircaloy-2P, and Zirconium Alloy C claddings were not better than the standard OPTIN cladding. During the reconstitution, 1RT4 fuel rods with Anikuloy, Zircaloy-1P, and Zirconium Alloy C claddings were replaced with LFA 1RT2 fuel rods with OPTIN and Zirconium Alloy E claddings. The reconstituted LFA 1RT4 contains fuel rods with OPTIN, Zirconium Alloy E, and Zircaloy-4F claddings.

3.7.3.13 Framatome and Westinghouse Lead Fuel Assemblies for Unit 2 Cycle 15

Eight LFAs were loaded into Unit 2 Cycle 15. Four assemblies were manufactured by Westinghouse, and they were designated as 2TW, and four assemblies were manufactured by FANP, and they were designated as 2TF. The purpose of the LFAs is to test fuel cladding variants, and in the case of FANP, to evaluate an alternate fuel supplier.

In the Westinghouse LFAs, the standard ZIRLO cladding is present along with three test claddings. In the FANP LFAs, their standard M5® cladding is present. The test claddings (e.g., low tin ZIRLO and M5®) in the LFA do not meet the NRC definition of Zircaloy-4 or ZIRLO. An explicit submittal for each set of LFAs was made. The NRC reviewed each LFA submittal and approved the use of the LFAs for peak rod burnups up to 60,000 MWD/MTU. Neither LFAs contain any burnable absorbers.

The Westinghouse LFAs utilize the standard Turbo grid cage design, with one minor design change where the back-up arch on the perimeter Turbo grid strip is slightly longer. This change further improves the grid to rod fretting performance and does not impact the flow, structural, or mechanical characteristics of the grid.

The FANP LFAs for the Calvert Cliffs Unit 2 reactor will be the CE 14x14 design. The bundle uses nine Zircaloy-4 grid spacers of the high thermal performance design. The lower tie plate is the FUELGUARD™ design, and the upper tie plate is the standard, reconstitutable FANP design for CE 14x14 fuel. The high thermal performance spacer was generically reviewed and accepted by the NRC and has been used for reload designs for CE, Westinghouse, and Kraft-werke Union reactors since 1991. The FUELGUARD™ lower tie plate has also been used in reload designs for CE, Westinghouse, and boiling water reactor designs. The reconstitutable upper tie plate design has been in use for reloads for CE plants since the early 1980s. Except for the changes to the fuel rod described in the following paragraphs, the LFA fuel bundle design has been used in reloads for other CE 14x14 plants. An illustration of this design is shown in Figure 3.7-1.

Each fuel bundle contains four corner guide tubes, one center guide tube/instrument tube, and 176 fuel rods. The corner guide tubes in the LFAs have the same nominal inside diameter/outside diameter (ID/OD) and dashpot design as used for the standard CE 14x14 reload fuel supplied by FANP. The elevations of the features (e.g., weep holes, upper sleeve attachment, etc.), except for the total length, are the same as has been used on other CE 14x14 reload designs. Similarly, the center guide tube has the same nominal ID/OD as has been used on other CE 14x14 designs and as the co-resident fuel. The height and elevations are established to be compatible with the Calvert Cliffs core plate separation distance, the co-resident fuel and the FANP manufacturing processes.

The fuel rod design for Calvert Cliffs uses a 136.7 inch fuel column of uranium dioxide pellets. The rod consists of cladding, an upper end cap, a lower end cap,

fuel pellets, and a plenum spring. The differences between the Calvert Cliffs lead assemblies and the standard FANP reload design for CE designed 14x14 plants are changes to the fuel rod design.

Specifically, the rod changes are:

* the cladding material used for the fuel rod is M5® instead of Zircaloy-4;
* the cladding inner diameter is increased by 0.003 inches to 0.387 inches;
* the pellet diameter is increased by 0.0035 inches to 0.3805 inches;
* the pellet density is 96% TD instead of 95.35% TD;
* the initial rod internal pressure will be increased from 315 psig to 375 psig; and
* the cladding length is increased by about 0.2 inches.

The increased cladding length provides more plenum volume, but requires the plenum spring to be modified to accommodate the longer plenum. The cladding OD is unchanged and is the same as the standard CE 14x14 reload fuel supplied by FANP and the same as the co-resident fuel. The lengths of the end caps will be the same as used for the standard CE 14x14 reload design.

All eight LFAs resided in symmetric core locations for both Unit 2 Cycle 15 and Unit 2 Cycle 16. For Unit 2 Cycle 17, only two Westinghouse LFAs (2TW02 and 2TW03) were inserted along the core periphery to gather additional data on grid-to-rod fretting resistance. The remaining LFAs (four FANP and two Westinghouse were discharged to the spent fuel pool).

For Unit 1 Cycle 19, two Westinghouse LFAs and two FANP LFAs were loaded in symmetric locations to evaluate the cladding at high pin burnups (up to 70 GWD/MTU).

3.7.4 CHRONOLOGY OF FUEL EXPERIENCE

The following summaries provide a brief synopsis of major fuel performance-related changes and experiments associated with each cycle.

3.7.4.1 Unit 1

a. Cycle 1

Three assemblies of Batch B fuel designated as experimental assemblies were inserted for the purpose of providing fuel design and performance data, and were the only assemblies inserted in Cycle 1 which were designed to be reconstitutable. One was scheduled to be irradiated for one cycle (BT01), one for two cycles (BT02), and one for three cycles (BT03). Experimental designs for fuel pellets, BPRs, and plenum springs were incorporated, as well as test samples of zirconium alloys and non-fueled rods to test cladding design/materials. The test assemblies are described in Section 3.7.3.2; Reference 1 gives a detailed survey of the fabrication and characterization of the test assemblies.

b. Cycle 2

The design of Batch D fuel was modified from the original design used in Batches A, B, and C. Some design changes were made to allow for the reconstitution of all assemblies in order to permit replacement of failed rods

or to provide for removal of rods for post irradiation examination. Design changes were also made to improve fuel performance. These changes included:

1. Changes in pellet shape [chamfering and reduced length/diameter (L/D) ratio];
2. Increased pellet density;
3. Increased clad thickness;
4. Decrease in helium fill gas pressure; and,
5. Decrease in pellet-clad gap size.

The density was increased to increase the stability of the fuel to densification, thereby improving gap conductance. The gap conductance was also enhanced by the reduction in gap size. The increased clad thickness and changes in pellet shape were made in order to reduce the susceptibility of fuel to failure by PCI/Corrosion cracking. The fill pressure could be reduced due to improved collapse resistance caused by thicker clad and a more stable fuel pellet design. The design changes are listed in Tables 3.3-1 and 3.3-2. The fuel assembly modifications were made, as stated above, to allow reconstitution, as well as to accommodate flow forces. Changes made to permit reconstitution were to the upper-end-fitting-to-guide-tube joint, anti-rotation device, and fuel rod upper and lower end caps. In addition, an Inconel bottom spacer grid replaced the mechanical retention grid.

Three test assemblies of a new design were incorporated into Unit 1 Cycle 2. The design of these three Batch D assemblies (D042, D047, D048), as well as an analysis of their impact on operations under normal and transient conditions, is contained in Reference 2. In addition to these three test assemblies, the two-and three-cycle Batch B test assemblies remained in the core for Cycle 2.

A detailed inspection program was undertaken to determine the extent and effect of bowing and hydriding of BPRs, which were present in Batch B fuel and 28 Batch C assemblies. This inspection was the result of: 1) the finding of a BPR end cap which had become detached from its BPR, and 2) the finding of several BPRs whose cladding had failed due to hydriding (Reference 3).

c. Cycle 3

The design of Batch E fuel was identical to standard Batch D design used in Cycle 2 with the exception of a 40 psi fill gas pressure reduction, and a pellet dish depth reduction of .002", increasing stack density to 10.046 g/cc. The increased density served to enhance fuel stability. The reduction in fill gas pressure reduced fuel temperature by accelerating clad creepdown and improving heat transfer. The resulting reduced fuel internal gas pressures remained conservative relative to predicted time for clad collapse.

Only one assembly containing BPRs (Batch B test assembly) remained in the core for Cycle 3. All three Batch D test assemblies remained in the core for their second cycle of irradiation.

A problem with CEA guide tube wear was noted prior to Cycle 3. This wear was the product of CEA vibration in the guide tubes (Reference 4). For Cycle 3 it was determined that no assemblies which had been under a CEA in a previous cycle would be inserted in a CEA location for Cycle 3 with the

exception of BT03 in the core center location. In addition, all assemblies which were to be placed in CEA locations would have sleeved guide tubes (Reference 4). The core reload pattern was modified in order to ensure that CEAs were in previously unworn guide tubes. This resulted in eight Batch C assemblies being discharged and replaced with eight Batch A assemblies.

All part length control rods were replaced (for preservation of pressure drops, flow rates, etc.) with CEA plugs (Reference 5).

d. Cycle 4

There were no design changes to the fuel pellets used in Batch F fuel. Batch F fuel assemblies were modified from previous design in that the holddown plate in the upper end fitting was thickened slightly, and the cross bracing that connects the lower end fitting posts was thickened and raised slightly from the lowermost surface of the fuel assembly.

As part of the continuing effort to mitigate guide tube wear, all previously burned fuel assemblies scheduled to be placed in CEA locations were sleeved (Reference 4). The only exception to this was test bundle BT03, which was in the center of the core for Cycles 3 and 4. This location is typically a low-wear location, and the degree of wear found during inspection after Cycle 3 was acceptable for placement under the center CEA for Cycle 4. Of the 72 new Batch F assemblies, 24 were placed under CEAs and were sleeved. Of the 48 F assemblies not under CEAs, 16 assemblies (unsleeved) had modified guide tubes (Reference 7) and 32 assemblies (unsleeved) had standard guide tubes.

At the EOC 3, test bundle BT03 was discharged, disassembled, and inspected. Six pins were removed to go to hot cell for examination and pins from the BT02 two-cycle assembly were inserted in their place. BT03 was then placed back in the center of the core. The three D test assemblies were retained in the core for a third cycle.

In addition to these, a new test fuel assembly was introduced into the Cycle 4 core. The Batch F SCOUT bundle was designed as a high burnup demonstration to provide information for the extended cycle/high burnup program (Reference 8).

e. Cycle 5

Batch G assemblies introduced into the Unit 1 core were composed of higher enriched fuel (40 assemblies at 3.65 wt% and 52 assemblies at 3.03 wt%) for extended cycle length/extended burnup. The 52 assemblies of lower enriched fuel contained 8 BPRs per assembly. These poison rods were of an improved design to eliminate the hydriding induced failure found in the earlier design, as well as to mitigate poison rod growth and bowing. The BPR changes included reduced pellet moisture limit and improved manufacturing to lessen moisture ingress. To reduce the likelihood of PCI which leads to axial growth/rod wall perforation, several other BPR design modifications were made, including: 1) increased pellet/clad gap, 2) chamfered pellets, 3) increased rod pressurization, and 4) reduced plenum spring preload.

Besides the addition of BPRs and the higher enrichment, no design changes were initiated with standard Batch G fuel.

As in previous cycles, assemblies placed in CEA locations had modified guide tubes to mitigate wear (with the exception of test bundle D047 in the core center location). Thirty-two of those assemblies were of the modified design while the rest were sleeved. Sixteen of the modified assemblies were Batch F assemblies placed in single CEA locations. The remaining 16 were Batch G assemblies placed in dual CEA locations.

In an effort to expand the database on material properties of irradiated Inconel 625 CEA cladding, three empty CEA tubes were placed in the center guide tubes of assemblies placed in high flux areas (Reference 9). Test bundle D047, placed in the center of the core, was left in the core for its fourth cycle of irradiation. It served as a carrier for 14 test pins; 13 pins from test bundle BT03, of which 8 were 4-cycle rods and 5 were 3-cycle rods, and 1 2-cycle non-fueled rod taken from BT02.

The Batch F SCOUT assembly remained in the core for its second cycle of irradiation. Introduced into the core for the first time in Cycle 5 were four lead assemblies called PROTOTYPE. They contained rods similar in design to SCOUT but greater in number in order to provide a sound statistical database for proper evaluation of fuel performance (Reference 10).

f. Cycle 6

Batch H assemblies introduced into Unit 1 core were composed of higher enriched fuel (40 assemblies at 4.00 wt% and 32 assemblies at 3.55 wt%) for extended cycle length/extended burnup. The 32 low enriched Batch H assemblies contained 8 BPRs per assembly. These BPRs were of similar design to those utilized in the low enriched Unit 1 Batch G fuel.

An additional design change introduced with Batch H fuel was a decrease in the overall length of the fuel rods of .200". This decrease yields additional shoulder gap clearance allowing for increased rod growth expected as the fuel is taken to higher burnups.

As in previous cycles, assemblies placed in CEA locations had modified guide tubes or were sleeved to mitigate wear. Four assemblies in CEA locations for Cycle 6 had modified guide tubes, while the remainder were sleeved.

Test assembly 1D042, which had been irradiated in Cycles 2, 3, and 4, was returned to the core for a fourth and final cycle of irradiation in Cycle 6. The Batch F SCOUT assembly remained in the core for its third cycle of irradiation. The four Batch G PROTOTYPE assemblies remained in the core for their second cycle of irradiation.

g. Cycle 7

Sixty-four new Batch J fuel assemblies were introduced into the Unit 1 core for Cycle 7. Of those, 48 were high enriched (4.05 wt%) and 16 were low enriched (3.40 wt%). None of the Batch J fuel contained BPRs. The mechanical design of Unit 1 Batch J fuel was identical to that of the Batch H fuel introduced in Cycle 6.

The SCOUT assembly remained in the core for its fourth cycle of irradiation during Cycle 7. At the EOC 6, two fuel rods were removed and replaced with SS rods.

The four PROTOTYPE assemblies remained in the core for their third cycle of irradiation during Cycle 7. At the EOC 6, two fuel rods were removed from one PROTOTYPE assembly, and replaced with SS rods.

Several Batch 1G and 1F fuel assemblies (remaining in the core for a third cycle of irradiation) were modified to allow for fuel rod growth. The modification involved the installation of a spacer shim which effectively raised the flow plate by .285". Modified upper end fitting corner posts were then installed to ensure compatibility between the upper end fitting and guide tubes. This modification is required when spacer shims are added.

Three inconel test specimens were placed in the core for irradiation during Unit 1 Cycle 5. Two of these remained in the core during Cycle 6. One specimen was then removed, leaving one specimen in the core for a third and final cycle of irradiation during Cycle 7.

h. Cycle 8

Seventy-two new Batch K fuel assemblies were introduced into the Unit 1 core for Cycle 8. Of those, 48 were high enriched (4.05 wt%) and 24 were low enriched (3.40 wt%). None of the Batch K fuel contained BPRs.

The mechanical design of the Batch K reload fuel was identical to that of Batch J, with the exception of the design features noted below:

1. The height of the lower end fitting is shorter. This reduction is achieved by shortening the legs of the lower end fitting assembly.
2. The overall lengths of the guide tubes are increased to compensate for the shorter lower end fitting described in 1. This increase is achieved by increasing the length of the buffer region, i.e., tapered region. The combination of this shorter lower end fitting and the longer guide tubes maintains the same overall assembly length as that of the Batch J fuel.
3. The elevations of the Inconel grid and the uppermost Zircaloy grid are changed to maintain their same relative elevations with respect to fuel rods as those of the reference cycle fuel design.

The changes described above were analyzed and found to have no significant adverse effect on the performance of the Batch K fuel relative to that of the Batch J fuel. These changes will result in improved fuel performance by increasing the shoulder gap from 1.400" to 1.775".

The SCOUT demonstration assembly remained in the core for its fifth cycle of irradiation in Cycle 8. The four PROTOTYPE assemblies remained in the Unit 1 core for their fourth cycle of irradiation.

As described in Section 3.3.2.4, nine CEAs were replaced and the configuration of two CEA banks were changed for Cycle 8 to increase net available scram worth.

CEA plugs were removed for Cycle 8 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling operation.

The phenomena of interpellet gap formation and clad collapse in modern PWR fuel rods was reassessed. It was concluded that the minimum time to clad collapse is significantly greater than its expected life and the

augmentation factor associated with interpellet gaps is insignificant compared with the uncertainties in the safety analysis. Therefore, the cycle-specific clad collapse analysis is not necessary and the augmentation factor associated with interpellet gaps is removed from the Technical Specifications.

i. Cycle 9

Fifty-two new Batch L fuel assemblies were introduced into the Unit 1 Cycle 9 core. Of those, 40 were high enriched (4.05 wt%) and 12 were low enriched (3.40 wt%). None of the Batch L fuel contained BPRs. The mechanical design of Batch L fuel is identical to that of the Batch K fuel introduced in Cycle 8.

The four Batch G PROTOTYPE assemblies remained in the core for their fifth cycle of irradiation. One SS rod (inserted at EOC 6) was replaced with a test fuel rod from the SCOUT assembly (which was discharged at EOC 8).

One Batch H assembly was kept in the core for a fourth cycle of irradiation to obtain high burnup data. It resides in the center core location.

Six fuel rods (three failed, one damaged and two for future inspection) were replaced by SS rods prior to Cycle 9.

Sixty-eight CEAs (all CEAs not replaced at EOC 7) were replaced with new CEAs. These CEAs are similar in design to original CEAs with the following notable exceptions:

1. Ag-In-Cd slug in the tip of the center finger
2. Reconstitutable corner fingers.

j. Cycle 10

Ninety-six new Batch M assemblies were introduced into the Unit 1 core for Cycle 10, the first Unit 1 24-month fuel cycle. It is also the first Unit 1 cycle to use low-leakage fuel loading pattern. Ninety-two of the assemblies were supplied by CE with 4.08 wt% U-235 enrichment. Of these, 16 assemblies had no BPRs and 76 assemblies had 12 B_4C BPRs. The remaining four assemblies (Batch MX) are the ANF demonstration assemblies which have 3.85 wt% U-235 average assembly enrichment and contain 12 gadolinium, fuel bearing (natural uranium) BPRs.

The mechanical design of the CE supplied Batch M reload fuel is identical to that of the Batch L fuel except some Batch M assemblies contain BPRs. The design of these poison rods is the same as that of the poison rods in Batch 2K assemblies from Unit 2 Cycle 8. The only changes in BPR design concern the use of higher poison loadings and hollow spacers rather than solid spacers.

Four damaged fuel rods were replaced with SS dummy rods in three Batch L assemblies prior to Cycle 10.

To support 24-month cycle operation, the very weak center CEA (all Al_2O_3 fingers) installed as part of the CEA bank configuration prior to Cycle 8 were replaced with a weak CEA (center finger B_4C, other four Al_2O_3).

k. Cycle 11

Eighty-four new Batch N fuel assemblies were introduced into the Unit 1 Cycle 11 core. This includes 12 assemblies with no BPRs, 20 assemblies with 4 BPRs, and 52 assemblies with 8 BPRs. All Batch N assemblies were enriched with 4.20 wt% U-235. Guide Tube Flux Suppressors were placed in selected fuel assemblies near the periphery of the core to reduce the fluence at the critical vessel weld. The Batch N fuel employs the GUARDIAN™ debris-resistant fuel design. The GUARDIAN™ design includes a new grid and fuel pin design. The four Batch MX ANF fuel assemblies were returned to the Unit 1 Cycle 11 core for their second cycle of irradiation.

Changes incorporated by the GUARDIAN™ fuel design include the following: the length of the Zircaloy lower end cap was increased to provide a solid Zircaloy region in the area where debris is to be trapped. The overall length of the fuel and BPRs was increased, while the length of the plenum regions was reduced. This was done to compensate for the increase in the length of the lower end cap. The guide tube length was increased to maintain the same shoulder gap. The position of the active fuel region and burnable poison region was raised due to the increase in height of the lower end cap. The height of the lower end fitting was decreased to keep the overall length of the bundle unchanged. This was accomplished by decreasing the compression region for the hold-down spring without a change in dimension of the spring.

The Zircaloy spacer grids were redesigned by increasing the size of the outer pin cell through enlargement of the outside envelope of the spacer grid assembly. This allows fuel rods located along the periphery of the fuel bundle to receive more coolant flow when in contact with adjacent bundles.

Fourteen fuel rods were replaced with SS dummy rods during reconstitution, for a total of 16 SS rods reinserted into Unit 1 Cycle 11.

The weak center CEA (center finger B_4C, others Al_2O_3) was replaced with a similar CEA utilizing a SS slug in the bottom of each weak finger. This replaced the Zircaloy slug which was subject to hydriding.

I. Cycle 12

Eighty-eight new Batch P fuel assemblies with 4.3 wt% U-235 were introduced into the Unit 1 Cycle 12 core. This includes 16 assemblies with no burnable absorbers, 12 assemblies with 20 erbium pins, 8 assemblies with 44 erbium pins, and 52 assemblies with 60 erbium pins. Guide tube flux suppressors remained in select peripheral assemblies to reduce the fluence to the critical vessel weld. The four Batch MX ANF fuel assemblies were returned to the Unit 1 Cycle 12 core for their third cycle or irradiation.

The Batch 1P GUARDIAN™ fuel assembly employs a design change which involves the replacement of the top TIG-welded Zircaloy grid with a laser weld Zircaloy grid. This design change introduces a backup arch in each grid cell in addition to the existing design with backup arches placed only in peripheral cell locations.

Three test capsule assemblies were placed in the outer guide tubes of select once-burned assemblies for Cycle 12. The capsules contain test specimens of an advanced spacer grid spring design.

m. Cycle 13

There was a design change to the perimeter spring design in the GUARDIAN™ grid to reduce protrusion of the spring beyond the outer edge of the grid.

Four test capsule assemblies were placed in the outer guide tubes of four separate fresh fuel assemblies for Cycle 13. Two of the capsules are from Cycle 12, and two are new. The capsules contain test specimens of advanced spacer grid spring designs.

Batch 1R erbium rods have cutback regions of 10.5” at the top and 12.0” at the bottom.

n. Cycle 14

The Batch 1S erbium rods have cutback regions of 10.5” at the top and 12.0” at the bottom.

Two test capsules were placed in the outer guide tubes of two separate fuel assemblies in Unit 1 Cycle 14. Both of these capsules were from Unit 1 Cycle 13. The capsules contain test specimens of advanced spacer grid spring designs.

o. Cycle 15

The Batch 1T fresh fuel assemblies represent the first full batch implementation of the VAP fuel assembly design. The VAP feature has been tested at Calvert Cliffs in lead fuel assembly batches 1RT and 2NT. Value Added Pellets’ diameter is 0.0045” larger than a standard pellet. The VAP clad wall thickness is 0.026” vs 0.028” for standard fuel. The Batch 1T erbium rods have cutback regions of 10.5” at the top of the fuel column and 14.0” at the bottom of the fuel column.

Unit 1 Cycle 15 is the first Calvert Cliffs reload designed with the ENDF/B-VI cross-section library in lieu of traditional ENDF/B-IV cross-sections.

Test capsules TCA-3 and TCA-5 were discharged to the spent fuel pool prior to the startup of Unit 1 Cycle 15.

Lead fuel assemblies 1RT1 and 1RT3 were returned to the core for a third cycle of irradiation. Lead fuel assemblies 1RT2 and 1RT4 were discharged to the spent fuel pool at the end of Unit 1 Cycle 14.

During the refueling outage, inspections were performed on the Batch 1R and 1RT fuel assemblies. Higher than expected grid-to-rod fretting was observed in certain fuel assemblies that had resided on the core periphery near the core shroud. Grid-to-rod fretting is an ongoing historical phenomena observed at Calvert Cliffs. To minimize grid to rod fretting during Unit 1 Cycle 15, several compensatory actions were implemented. First of all, the fuel management pattern was changed to rotate 1RT1 by 180° so that it would not have the same face against the core shroud for two consecutive cycles. Next, several fuel rods were rotated to present a fresh unworn surface to the rod support features. Finally, one heavily worn pin in 1R222 and two heavily worn pins in 1RT1 were replaced with stainless steel pins. These actions will minimize grid-to-rod fretting for Unit 1 Cycle 15. The grid-to-rod fretting wear that is expected to occur during Unit 1 Cycle 15 will

occur in low power peripheral fuel rods that will not contribute significant activity to the coolant if they were to fail. The number of grid-to-rod fretting failures likely to be experienced in Cycle 15 will not result in the RCS coolant activity approaching the Technical Specification limit. The impact of the population of fuel rods exhibiting significant wear such that integrity under transient and accident conditions could be challenged is small enough that the coolant activity impact is bounded by previously analyzed levels.

p. Cycle 16

Due to the merger of Westinghouse and CE, the CE fuel fabrication facility at Hermatite was permanently shut down. Before it was closed, all of the Batch 1V erbium rods were built at Hermatite. The remaining non-erbium rods were built at the Westinghouse Columbia manufacturing facility.

The integration of the basic CE fuel designs into the Columbia fabrication process necessitated numerous internal rod design changes including TIG welding of end caps, elimination of the lower alumina spacer, redesign of the plenum spring, etc.

The merger of Westinghouse and CE resulted in the Westinghouse standard cladding material (ZIRLO) becoming available to Calvert Cliffs. Since ZIRLO has better water side corrosion properties than the CE standard OPTIN cladding material, Calvert Cliffs elected to phase in ZIRLO cladding. ZIRLO was only available in time to support manufacturing of the non-erbium pins. The erbium pins for Unit 1 Cycle 16 still use the OPTIN cladding material.

Unit 1 Cycle 16 is the first full batch implementation of the advanced grid design known as Turbo. This grid design was tested in the Batch 1RT LFAs. The Turbo grid features include mixing vanes (at five of the eight spacer grid locations) and a new rod retention device known as I-springs (at all eight spacer grid locations).

Batch 1V will be the second full batch of VAP fuel assemblies for Unit 1.

Some of the Batch 1V fuel assemblies will use a 60 erbium pin lattice pattern. This is the first use of the 60-pin pattern with VAPs. The 60 pin pattern was previously used at Calvert Cliffs in Batch 1P with standard fuel pellets.

q. Cycle 17

Eighty-eight fresh assemblies were installed for Unit 1 Cycle 17 (batch designation 1W).

Batch 1W is the third batch of VAP for Unit 1. Erbia remains the burnable absorber, the Erbia fuel pins have cutback regions of 10.5 inches at the top of the rod and 12.0 inches at the bottom.

Batch 1W is the second batch of the Turbo fuel assembly design for Unit 1. All but four assemblies utilized the same design and the same grid cage design as the U2C15 Westinghouse LFAs, see Section 3.7.3.13. The four different assemblies do not have the increased backup-arch length and have been given a unique sub-batch identifier. All of the fuel was manufactured by Westinghouse at their Columbia, SC facility. The cladding material for all fresh fuel is ZIRLO™.

As Unit 1 contains approximately 85% of the Turbo fuel assembly design, transient analyses were updated to utilize the ABB-TV CHF Correlation. Since the Turbo fuel has a non-mixing vane lower axial section and an upper section with mixing grids, both the ABB-NV and the ABB-TV CHF correlations are applied in the safety analysis.

r. Cycle 18

Ninety-seven fresh assemblies were installed for Unit 1 Cycle 18 (batch designation 1X). This reload incorporated the "T" pattern and contains no fresh fuel on the core periphery.

Beginning with Cycle 18, ZrB_2 (IFBA) has replaced erbia as the burnable absorber. Batch 1X fuel assemblies, with the exception of single 1X7 assembly, also contain axial blankets, which consist of a lower-enrichment at the top and bottom 6" of the fuel. The axial blankets in the fuel rods that contain IFBA consist of annular pellets to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

Also beginning with Cycle 18 is the addition of radial enrichment zoning. Batch 1X assemblies, with the exception of single 1X7 assembly, contain fuel rods of two different enrichments, with lower enriched rods placed on the assembly corners and next to guide tube locations.

Cycle 18 also contains a single assembly, sub-batch 1X7, which is a single-enrichment assembly at 2.0 wt% U-235 and does not contain either IFBA rods or axial blankets. The assembly has been placed in the center of the core in lieu of a twice burned assembly. This is the first reload in which a low-enriched fresh assembly is used in this location.

During the refueling outage prior to the startup of Cycle 18, inspections were performed on several reinsert Batch 1W assemblies due to indications of pin failures during their first duty cycle. In three assemblies, a total of four fuel pins were replaced with other fuel pins (from the parent assemblies) and a total of four stainless rods were inserted.

s. Cycle 19

Ninety-six fresh assemblies were loaded for Unit 1 Cycle 19 with a batch designation of 1Z. Cycle 19 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch 1Z consists of sub-batches with two-enrichment and three-enrichment radial zoning. Batch 1Z retains the use of IFBA as a burnable

absorber and the use of axial blankets. Annular pellets are used in the blanket regions of rods that contain IFBA to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

Two Westinghouse LFAs and two FANP LFAs were loaded in symmetric locations for a third cycle of irradiation of up to 70 GWD/MTU.

During the 2008 refueling outage prior to U1C19 startup, fuel inspections identified failed fuel pins in three once-burned (Batch 1X) assemblies slated for reinsertion into Cycle 19. Two of the failed assemblies were replaced with substitute assemblies and were not reinserted into the core. An inert stainless steel pin was inserted into the third failed assembly, allowing it to be used in Cycle 19.

t. Cycle 20

Ninety-two fresh assemblies were loaded for Unit 1 Cycle 20, eighty-eight with a batch designation of AA, and four with a batch designation of 2X7. Cycle 20 included the “T” pattern and contains no fresh fuel on the core periphery.

Batch AA consists of sub-batches with two-enrichment and three-enrichment radial zoning. Batch AA retains the use of IFBA as a burnable absorber and the use of axial blankets. Annular pellets are used in the blanket regions of rods and contain IFBA to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

Batch 2X7 was originally purchased as spare fuel for Unit 2 Cycle 18. These assemblies were not used in U2C18 and are fresh for U1C20. Batch 2X7 consists of only one enrichment and does not contain any IFBA as a burnable absorber.

The rated thermal power was raised from 2700 MWt to 2737 MWt during this cycle and was based on flow measurement uncertainty capture.

u. Cycle 21

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 1 Cycle 21 with a batch designation of AB. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the first implementation of AREVA fuel in Calvert Cliffs Unit 1. Cycle 21 included the “T” pattern and contains no fresh fuel on the core periphery.

Batch AB consists of sub-batches with two-enrichment radial zoning. Batch AB uses Gd_2O_3 as a burnable absorber. Six inch low-enriched blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 1 Cycle 21 was designed using the AREVA physics code package (CASMO/PRISM).

v. Cycle 22

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 1 Cycle 22 with a batch designation of AC. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the second implementation of AREVA fuel in Calvert Cliffs Unit 1. Cycle 22 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch AC consists of sub-batches with two-enrichment radial zoning. Batch AC uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 1 Cycle 22 was designed using the AREVA physics code package (CASMO/PRISM). Unit 1 Cycle 22 is the second Calvert Cliffs reload to use the HTP critical heat flux correlation for AREVA fuel.

w. Cycle 23

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 1 Cycle 23 with a batch designation of AD. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the third implementation of AREVA fuel in Calvert Cliffs Unit 1. Cycle 23 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch AD consists of sub-batches with two-enrichment radial zoning. Batch AD uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 1 Cycle 23 was designed using the AREVA physics code package (CASMO/PRISM). Unit 1 Cycle 23 is the third Unit 1 reload to use the HTP critical heat flux correlation for AREVA fuel.

3.7.4.2 <u>Unit 2</u>

a. Cycle 1

Except for differences noted earlier in this chapter, the design of Unit 2 Cycle 1 fuel was identical to that of Unit 1 Cycle 1 fuel. BPRs incorporated into Cycle 1 fuel were of the improved design.

b. Cycle 2

Batch D fuel was identical in design to Batch E fuel from Unit 1. Design differences from original core load fuel were:

1. A 40 psi reduction in fill gas pressure,
2. A reduction in pellet dish depth of .002", increasing stack density to 10.046 g/cc.

To mitigate guide tube wear, all assemblies to be placed under CEAs were modified. Sixteen Batch D low enrichment assemblies had modified guide tubes. Of the 16, 4 were put in non-CEA locations, 4 were placed under single CEAs and the remaining 8 were placed as 4 pairs under dual CEAs (Reference 4). The remainder of the assemblies to be placed under CEA locations were sleeved (Reference 4).

All PLCEAs were replaced with CEA plugs (Reference 11).

c. Cycle 3

The mechanical design of new Batch E reload fuel was identical to that of Batch D fuel for the Unit 1 Cycle 2 reload. All assemblies placed under CEAs were sleeved to mitigate guide tube wear and, in addition, all Batch E fuel assemblies were sleeved prior to insertion. None of the modified assemblies in Batch D were used in CEA locations for Cycle 3.

A prototype CEA (Reference 12) was introduced as part of CEA Group 5. The design of the prototype involved a change in cladding material from Inconel to SS, as well as a change to reconstitutable poison rods. The poison rods themselves were modified to replace the Ag/In/Cd tips with B_4C for economic as well as material availability reasons.

d. Cycle 4

The mechanical design of Batch F reload fuel was identical to that of Batch G fuel used in Unit 1 Cycle 5 reload. The enrichment was increased from the previous cycle to accommodate the extended cycle/extended burnup program. The lower enriched Batch F assemblies each contained eight BPRs of the improved design. All assemblies placed in CEA locations contained sleeves to mitigate guide tube wear.

e. Cycle 5

The mechanical design of Batch G reload fuel was identical to that of the Batch H fuel used in Unit 1 Cycle 6. The overall rod length was reduced by .200" from that of Unit 2 Batch F fuel. This decrease yields additional shoulder gap clearance allowing for increased rod growth expected as the fuel is taken to higher burnups. Fuel enrichments were increased over Unit 2 Batch F enrichments to accommodate higher burnup/extended cycles. Batch G was comprised of 48 high-enriched (4.0 wt% U-235) and 28 low-enriched (3.55 wt% U-235) assemblies. The lower enriched assemblies each contained eight BPRs. All assemblies placed in CEA locations contained sleeves to mitigate guide tube wear problems.

f. Cycle 6

Seventy-two new Batch H assemblies were introduced into the Unit 2 core for Cycle 6. Of those, 48 were high-enriched (4.05 wt%) and 24 were low-enriched (3.40 wt%). None of the Batch H fuel contained BPRs. The mechanical design of Unit 2 Batch H fuel was identical to that of the Batch J fuel introduced in Unit 1 Cycle 7 with the following exception. A 0.2" spacer shim was installed between the upper end of the guide tube and the upper end fitting to provide more space for rod growth. The upper end fitting design was then modified to make it compatible.

Several of the high-enriched Batch 2F assemblies (remaining in the core for a third cycle of irradiation) required field installation of .285" spacer shims to allow for rod growth during Cycle 6. Four failed fuel pins from Batch 2G fuel assemblies (once burned) were identified and replaced with SS dummy pins prior to Cycle 6.

The prototype CEA-X remained in the core center location for its fourth cycle of irradiation in Cycle 6.

g. Cycle 7

Sixty new Batch J assemblies were introduced into the Unit 2 core for Cycle 7. Of those, 40 were high-enriched (4.05 wt%) and 20 were low-enriched (3.40 wt%). None of the Batch J fuel contained BPRs.

The mechanical design of Batch J reload fuel was identical to that of Batch K fuel used in Unit 1 Cycle 8.

The reassessment of the phenomena of interpellet gap formation and clad collapse in modern PWR fuels led to the conclusion that the minimum time to clad collapse is significantly greater than its expected life, and the augmentation factor associated with interpellet gaps is insignificant. Therefore, the cycle-specific clad collapse analysis and the augmentation factor associated with interpellet gaps were eliminated.

As described in Section 3.3.2.4, the composition of eight CEAs and the configuration of two CEA banks were changed. The prototype CEA-X was moved to another core location as part of the CEA bank reconfiguration and remained in the core for its fifth cycle of irradiation.

The eight CEA plugs were removed to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling operation.

h. Cycle 8

Eighty-eight new Batch K assemblies were introduced into the Unit 2 core for Cycle 8, the first 24-month cycle. It is also the first cycle to use low-leakage fuel loading pattern. All 88 were enriched to 4.08 wt%. Of these, 16 assemblies had no BPRs, 28 assemblies had 12 BPRs and 44 assemblies had 8 BPRs.

The mechanical design of Batch K reload fuel is identical to that of Batch J fuel except some Batch K fuel assemblies contain BPRs. The design of these poison rods is essentially the same as that of the poison rods in Batch 2G assemblies from Cycle 5. The only changes in BPR design concern the use of higher poison loadings and hollow spacers rather than solid spacers.

Nine fuel rods were replaced with SS dummy rods prior to Cycle 8.

The prototype CEA-X was discharged at the EOC 8 after its sixth cycle of irradiation.

To support 24-month cycle operation, the very weak center CEA (five Al_2O_3 fingers), installed as part of the CEA bank reconfiguration prior to Cycle 7, was replaced with a weak CEA (center finger B_4C, other Al_2O_3).

i. Cycle 9

Ninety-Two new Batch L assemblies were introduced into the Unit 2 Core for Cycle 9: 16 unshimmed Batch L assemblies, 20 4-shimmed (B_4C) Batch LX assemblies, 24 8-shimmed (B_4C) Batch L/assemblies, 28 12-shimmed (B_4C) Batch L* assemblies all at 4.30 wt% U-235 enrichment, and 4 Batch LE Erbium demonstration assemblies.

The Erbium demonstration assemblies are included in Cycle 9 in order to determine their incore characteristics during a 24-month cycle. These assemblies contain Erbium as the burnable poison material. Each assembly consists of 80 standard pins at 4.30 wt% U-235, 52 standard pins at 3.40 wt% U-235, and 44 Erbium bearing pins at 3.40 wt% U-235. This configuration results in an assembly average U-235 enrichment of 3.81 wt%.

The mechanical design for the Batch L reload fuel is essentially identical to that of the Batch K fuel except as noted below.

The lower end fitting of the Batch L fuel is essentially the same as that of the previous design except for the configuration of the flow holes. In the Batch L design, a 3x3 array of small flow holes has replaced each of the large flow holes. Also, wherever possible, additional small holes were added to the Batch L design to minimize the increase in the pressure drop of the small hole lower end fitting design, relative to the previous design. This design change was made to improve the debris resistance of the Batch L reload fuel.

The fuel rod plenum spring in the Batch L fuel has been redesigned to minimize the amount of rod internal void volume that it occupies.

The overall length of the Batch L poison rods was increased such that the poison rod length is now the same as the fuel rod length.

The size and quantity of the crimp holes in the upper end of the guide tubes were modified for the Batch L fuel. This change allows the upper end fitting posts to be re-used if an assembly must be reconstituted.

Seven fuel rods were replaced with SS dummy rods prior to Cycle 9.

The weak center CEA (center finger B_4C, other Al_2O_3), installed prior to Cycle 8, was replaced with a similar CEA utilizing a SS slug in the bottom of each weak finger. The slug was used instead of Zircaloy, which in this application was found to be subject to hydriding.

j. Cycle 10

Eighty-eight new Batch 2M fuel assemblies were introduced into the Unit 2 core for Cycle 10. One-hundred twenty-four Batch 2K and 2L assemblies were retained from Cycle 9. One reinserted Batch 2H* assembly which had been discharged from Unit 2 Cycle 7 and four reinserted Batch 2J* assemblies which had been discharged at the end of Unit 2 Cycle 8 make up the remainder of the core. Four erbium demonstration assemblies (2LE) are carried over from Unit 2 Cycle 9 from a second cycle of irradiation.

The Batch 2M fuel employs the debris resistant GUARDIAN™ design as described in Section 3.7.4.1.k.

Eight fuel rods were replaced by SS replacement rods in Batch 2K during the refueling outage prior to Cycle 10.

The Batch 2M burnable absorber pins consist of a 115.7" central region containing the burnable material B_4C with two 10.5" cutback regions containing Al_2O_3, one at each end of the stack. This change is being made to enhance thermal margin by lowering the axial peak at BOC.

k. Cycle 11

The following changes were made to the Unit 1 Batch P fuel bundle assembly design to create the Unit 2 Batch N standard fuel bundle assembly design:

The TIG welded wavy strip GUARDIAN™ Inconel bottom spacer grid assembly has been replaced by a laser welded straight strip GUARDIAN™ spacer grid assembly. In conjunction with this change, the fuel rod lower end cap was redesigned for compatibility with the new spacer grid and to facilitate manufacturing. The new lower end cap has one long taper, rather than a taper, a flat and then another taper at the bottom of the cap.

The TIG welded wavy strip intermediate Zircaloy spacer grid assemblies have been replaced by laser welded wavy strip intermediate Zircaloy spacer grid assemblies.

Batch 2N erbium rods have a 12” cutback in neutron absorbing material.

l. Cycle 12

The following changes were made to the Unit 1 Cycle 13 (Batch 1R) fuel assembly design to create the Unit 2 Cycle 12 (Batch 2P) standard fuel bundle assembly design.

The perimeter strips have small guide holes to match pins on the grid assembly weld fixture. This is to ensure more consistent alignment of the grid strips within the fixture during welding.

Batch 2P erbium rods have cutbacks in the neutron absorbing material of 14.0" at the bottom and 10.5" at the top.

m. Cycle 13

Ninety-two fresh Batch 2R fuel assemblies were introduced into the Unit 2 Cycle 13 core. The Batch 2R erbium rods have a symmetric cutback region of 12.0” at the top and bottom of the fuel column.

n. Cycle 14

Ninety-two fresh Batch 2S VAP fuel assemblies were introduced into the Unit 2 Cycle 14 core. The Batch 2S VAP fuel assemblies represent the first full batch of VAP assemblies in Unit 2. Value Added Pellets' diameter are 0.0045" larger than a standard pellet. The VAP clad wall thickness is 0.026" vs 0.028" for standard fuel. The Batch 2S erbium rods have a symmetric cutback region of 12.0" at the top and bottom of the fuel column.

Unit 2 Cycle 14 was designed with the ENDF/B-VI cross-section library in lieu of the traditional ENDF/B-IV cross-sections.

Reconstituted LFA 1RT4 was inserted into the Unit 2 Cycle 14 core. The original LFA 1RT4 was reconstituted with fuel rods from LFA 1RT2, after Unit 1 Cycle 14. The original LFA 1RT4 assembly was reconstituted because the corrosion performance of Anikuloy, Zircaloy-2P, and Zirconium Alloy C claddings were not better than the standard OPTIN cladding. During the reconstitution, 1RT4 fuel rods with Anikuloy, Zircaloy-2P, and Zirconium Alloy C Claddings were replaced with LFA 1RT2 fuel rods with OPTIN or Zirconium Alloy E claddings. The reconstituted LFA 1RT4 contains fuel rods with OPTIN, Zirconium Alloy E, and Zircaloy-4F claddings.

Unit 2 Cycle 14 is the first Calvert Cliffs reload designed with ABB-NV critical heat flux correlation in lieu of the traditional CE-1 correlation.

o. Cycle 15

Ninety-two fresh Batch 2T VAP fuel assemblies were introduced into the Unit 2 Cycle 15 core. The Batch 2T VAP fuel assemblies represent the second full batch of VAP for Unit 2. The Batch 2T erbium rods had an asymmetric cutback region of 10.5 inches at the top of the rod and 14.0 inches at the bottom. Included in the 92 fresh assemblies are 8 LFAs: 4 from Westinghouse and 4 from FANP. These LFAs are further described in Section 3.7.3.13.

Due to the merger of Westinghouse and CE, the CE fuel fabrication facility at Hermatite was permanently shut down. All of the Batch 2T rods were built at the Westinghouse Columbia manufacturing facility.

The integration of the basic CE fuel designs into the Columbia fabrication process necessitated numerous internal rod design changes, including TIG welding of end caps, elimination of the lower alumina spacer, redesign of the plenum spring, etc.

The merger of Westinghouse and CE resulted in the Westinghouse standard cladding material (ZIRLO) becoming available to Calvert Cliffs. Since ZIRLO has better water side corrosion properties than the CE standard OPTIN cladding material, Calvert Cliffs elected to utilize the ZIRLO cladding.

Unit 2 Cycle 15 is the second full batch implementation of the advanced grid design known as Turbo at Calvert Cliffs. The first full batch of Turbo was implemented on Unit 1 Cycle 16. This grid design was tested in the Batch 1RT LFAs. The Turbo grid features include mixing vanes (at five of the eight spacer grid locations) and a new rod retention device known as I-springs (at all eight spacer grid locations).

Batch 2T will be the second full batch of VAP fuel assemblies for Unit 2.

p. Cycle 16

Ninety-two fresh assemblies were installed for Unit 2 Cycle 16 (batch designation 2V). This reload incorporated the “T” pattern and contains no fresh fuel on the core periphery.

Beginning with Cycle 16, ZrB_2 (Integral Fuel Burnable Absorber or IFBA) has replaced erbia as the burnable absorber. Batch 2V fuel assemblies also contain axial blankets, which consist of a lower-enrichment at the top and bottom 6" of the fuel. The axial blankets in fuel rods that contain IFBA consist of annular pellets to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

Also beginning with Cycle 16 is the addition of radial enrichment zoning. Batch 2V assemblies contain fuel rods of two different enrichments, with lower enriched rods placed on the assembly corners and next to guide tube locations.

Sixty-four CEAs were replaced for Cycle 16. All Unit 2 CEAs now have a 12" Ag-In-Cd slug to increase life expectancy.

q. Cycle 17

Ninety-six fresh assemblies were installed for Unit 2 Cycle 17 (batch designation 2W). This reload continues use of the "T" pattern and contains no fresh fuel on the core periphery.

As with Batch 2V, Batch 2W fuel assemblies also contain axial blankets, which consist of a lower-enrichment at the top and bottom 6" of the fuel. The axial blankets in the fuel rods that contain IFBA consist of annular pellets to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

Some subbatches (W1 and W2) use, like Batch 2V, two radial enrichment zones. But, beginning with Cycle 17, other subbatches (W3 through W6) use three radial enrichment zones.

Two Westinghouse LFAs (2TW02 and 2TW03) were reinserted into Unit 2 Cycle 17 for a third cycle of irradiation. The other two Batch 2TW and four Batch 2TF LFAs were discharged to the spent fuel pool.

r. Cycle 18

Ninety-six fresh assemblies were loaded for Unit 2 Cycle 18 with a batch designation of 2X. Cycle 18 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch 2X consists of sub-batches with two-enrichment and three-enrichment radial zoning. Batch 2X retains the use of IFBA as a burnable absorber and the use of axial blankets. Annular pellets are used in the blanket regions of rods that contain IFBA to provide extra plenum volume to accommodate increased helium gas production associated with the ZrB_2.

During the 2009 refueling outage prior to U2C18 startup, fuel inspections indicated failed fuel pins in assembly 2V109, which was replaced with 2V103. Additionally, assembly 2V011 was flagged as suspect (but later cleared) and was replaced with 2V101. Lastly, assembly 2W508 was found with an elevated CEA, and was replaced with 1X411 (a once-burned assembly from Unit 1).

s. Cycle 19

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 2 Cycle 19 with a batch designation of 2Z. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the first implementation of AREVA fuel in Calvert Cliffs Unit 2. Cycle 19 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch 2Z consists of sub-batches with two-enrichment radial zoning. Batch 2Z uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 2 Cycle 19 was designed using the AREVA physics code package (CASMO/PRISM). Unit 2 Cycle 19 is the first Calvert Cliffs reload to use the HTP critical heat flux correlation for AREVA fuel.

t. Cycle 20

One-hundred fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 2 Cycle 20 with a batch designation of BA. Four assemblies from batch 2Z were discharged to the pool for contingency planning for failed fuel assemblies. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the second implementation of AREVA fuel in Calvert Cliffs Unit 2. Cycle 20 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch BA consists of sub-batches with two-enrichment radial zoning. Batch BA uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low-enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 2 Cycle 20 was designed using the AREVA physics code package (CASMO/PRISM). Unit 2 Cycle 20 is the second Calvert Cliffs reload to use the HTP critical heat flux correlation for AREVA fuel.

u. Cycle 21

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 2 Cycle 21 with a batch designation of BB. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the third implementation of AREVA fuel in Calvert Cliffs Unit 2. Cycle 21 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch BB consists of sub-batches with two-enrichment radial zoning. Batch BB uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch low enriched axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 2 Cycle 21 was designed using the AREVA physics code package (CASMO/PRISM). Unit 2 Cycle 21 is the third reload to use the HTP critical heat flux correlation for AREVA fuel.

w. Cycle 22

Ninety-six fresh AREVA Advanced CE HTP™ fuel assemblies were loaded for Unit 2 Cycle 22 with a batch designation of BC. The AREVA Advanced CE HTP™ fuel design employs the FUELGUARD™ lower tie plate, MONOBLOC™ guide tubes, and HTP™ spacer grids. Zr-4 was used for the guide tubes and spacer grids. M5® was used as the fuel cladding material. This is the fourth implementation of AREVA fuel in Calvert Cliffs Unit 2. Cycle 22 included the "T" pattern and contains no fresh fuel on the core periphery.

Batch BC consists of sub-batches with two-enrichment radial zoning. Batch BC uses Gd_2O_3 as a burnable absorber. Six inch low-enriched axial blankets are used on the top and bottom of non-gadolina-bearing fuel rods. Twelve inch, low enriched, axial blankets are used on the top and bottom of gadolina-bearing fuel rods.

Unit 2 Cycle 22 was designed using the AREVA physics code package (CASMO/PRISM). Unit 2 Cycle 22 is the fourth Unit 2 reload to use the HTP critical heat flux correlation for AREVA fuel.

3.7.5 REFERENCES

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3. CEN-48(B)-P, "Calvert Cliffs Unit 1 Fuel Inspection Report," February 1977
4. CEN-83(B)-P and Amendment 1 to same, "Calvert Cliffs Unit 1 Reactor Operation with Modified CEA Guide Tubes," February 1978
5. A.E. Lundvall, Jr. to E.G. Case, letter, "Supplement to Cycle 3 Reload Application," March 17, 1978
6. CEN-101(B)-P, "Calvert Cliffs Unit 2 Cycle 2 Reload Submittal Amendment," August 28, 1978
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8. CE-NPSD-169-P, "A Summary of the Design, Fabrication and Characterization of the SCOUT High Burnup Demonstration Assembly," February 1982
9. Letter from A.E. Lundvall, Jr. to R.A. Clark, "Amendment to Operating License DPR-53, Fifth Cycle License Application," September 22, 1980
10. Letter from A.E. Lundvall, Jr. to R.A. Clark, "Responses to NRC Staff Questions," November 20, 1980
11. Letter from A.E. Lundvall, Jr. to R.W. Reid, "Application for Cycle 2 Reload for CCNPP-2," July 26, 1978
12. Letter from A.E. Lundvall, Jr. to R.W. Reid, "Supplement to Application for Cycle 4 Reload - B_4C Type CEA Design," March 5, 1979
13. CENPD-187-P-A, "Method of Analyzing Creep Collapse of Oval Cladding, CEPAN," March 1976

14. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983
15. CENPD-198-P, "Zircaloy Growth - In Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies," December 1975
16. Deleted
17. CENPD-139-P-A, "Fuel Evaluation Model, FATES," July 1974
18. Technical Report on Densification of Light Water Reactor Fuels, Regulatory Staff, U. S. Atomic Energy Commission, November 1972
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20. CE-NPSD-72, "Examination of Calvert Cliffs 1 Test Fuel Assemblies at EOCs 1 and 2," September 1978
21. CE-NPSD-87, "Examination of Calvert Cliffs 1 Test Fuel Assembly After Cycle 3," September 1979
22. CE-NPSD-146, "Examination of Calvert Cliffs 1 Test Fuel Assembly After Cycle 4," October 1981
23. CE-NPSD-75, "Gas Release and Microstructural Evaluation of One and Two Cycle Fuel Rods from Calvert Cliffs 1," March 1979
24. CE-NPSD-119, "Gas Release and Microstructural Evaluation of Three-Cycle Fuel Rods from Calvert Cliffs-1," December 1980
25. "Densification, Swelling and Microstructures of LWR Fuels Through Extended Burnup," ANS Topical Meeting on LWR Extended Burnup - Fuel Performance and Utilization, Vol. 1, pgs. 4-39, April 4-8, 1982
26. CEN-182(B), "Statistical Approach to Analyzing Creep Collapse of Oval Fuel Rod Cladding Using CEPAN," September 1981
27. CEN-161(B)-P, "Improvements to Fuel Evaluation Model (FATES3)," July 1981
28. Letter from A.E. Lundvall, Jr. (BGE) to J.R. Miller (NRC), Docket Nos. 50-317 & 50-318, "Request for Amendment (Clad/Collapse Augmentation Factors)," December 31, 1984
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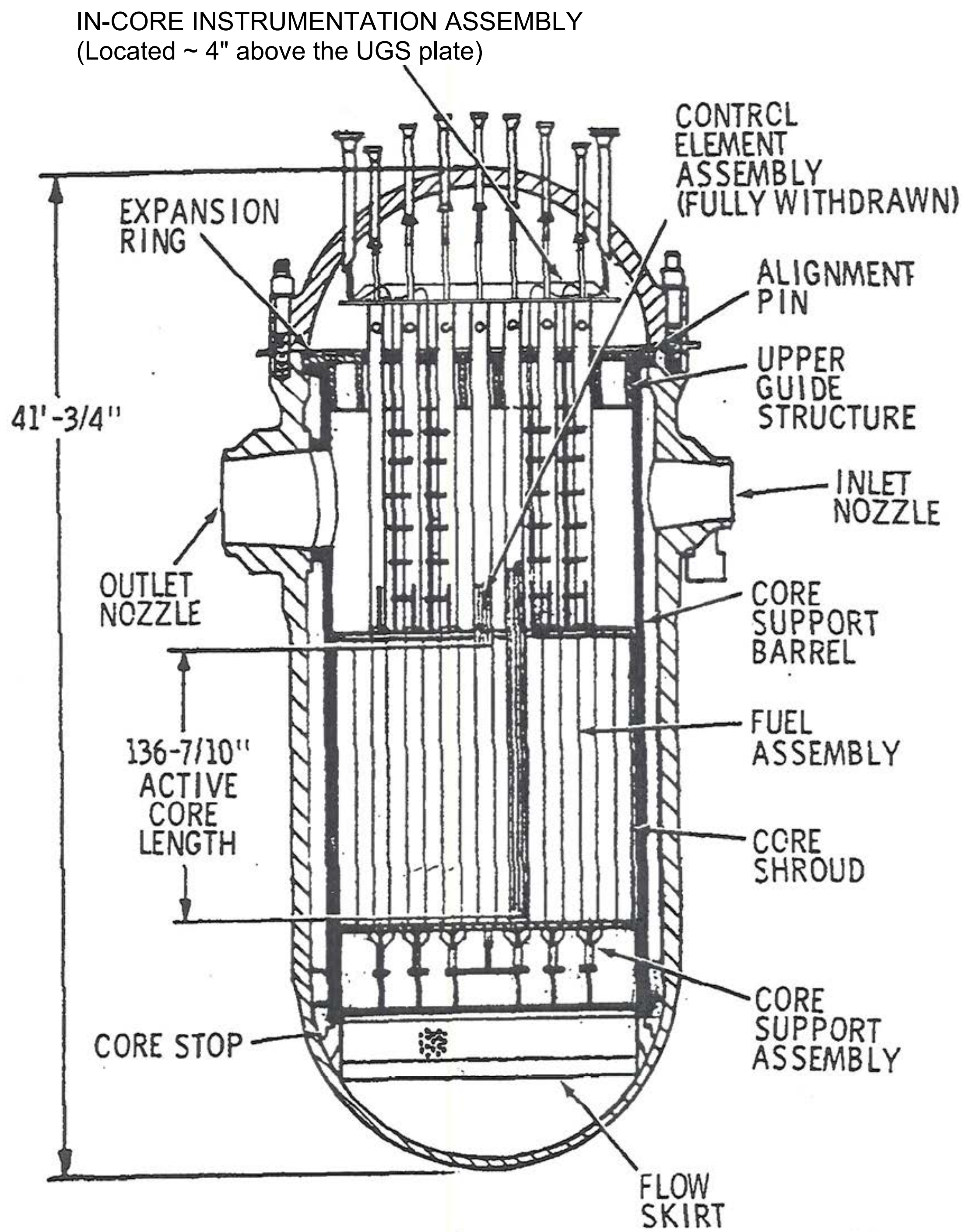
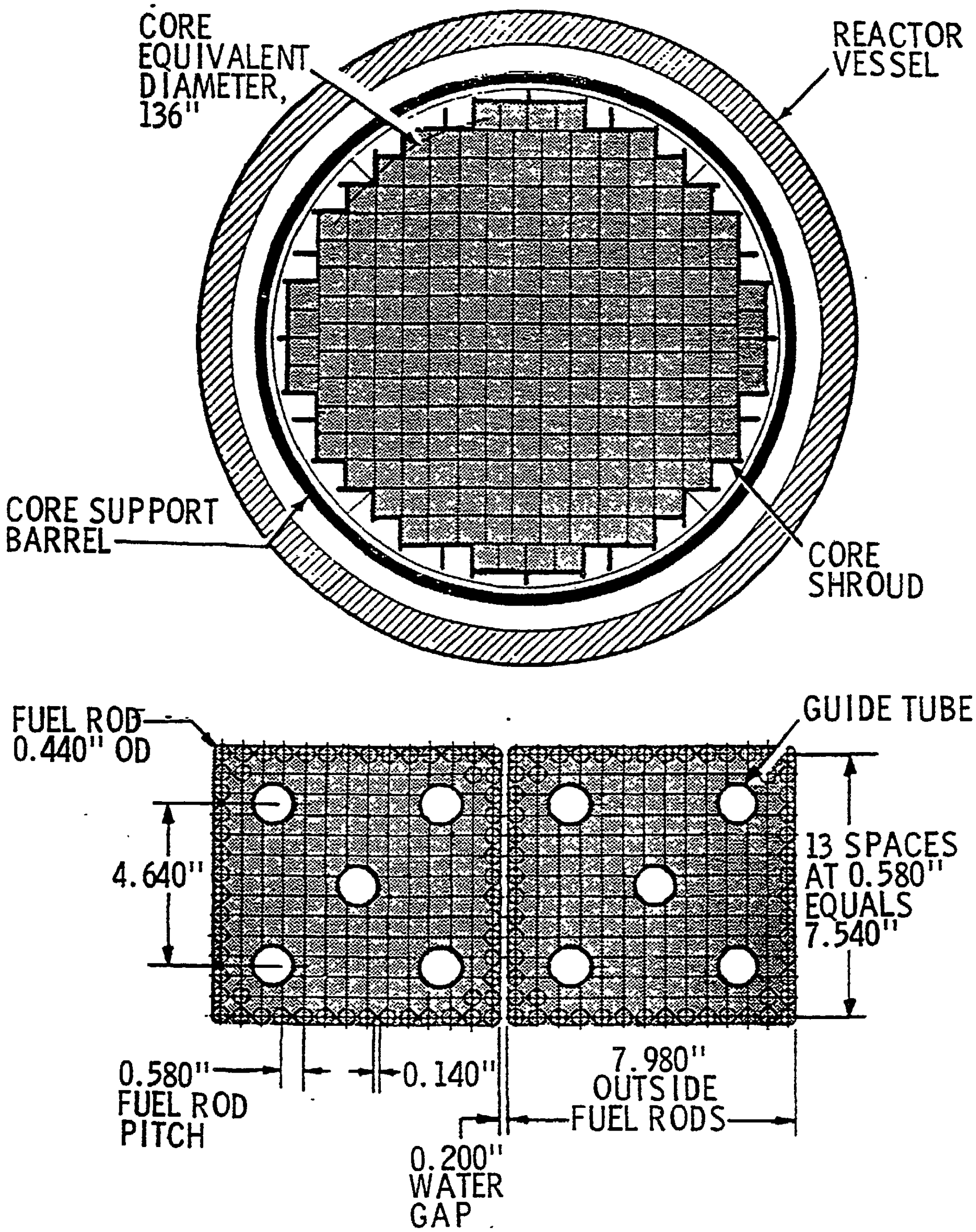
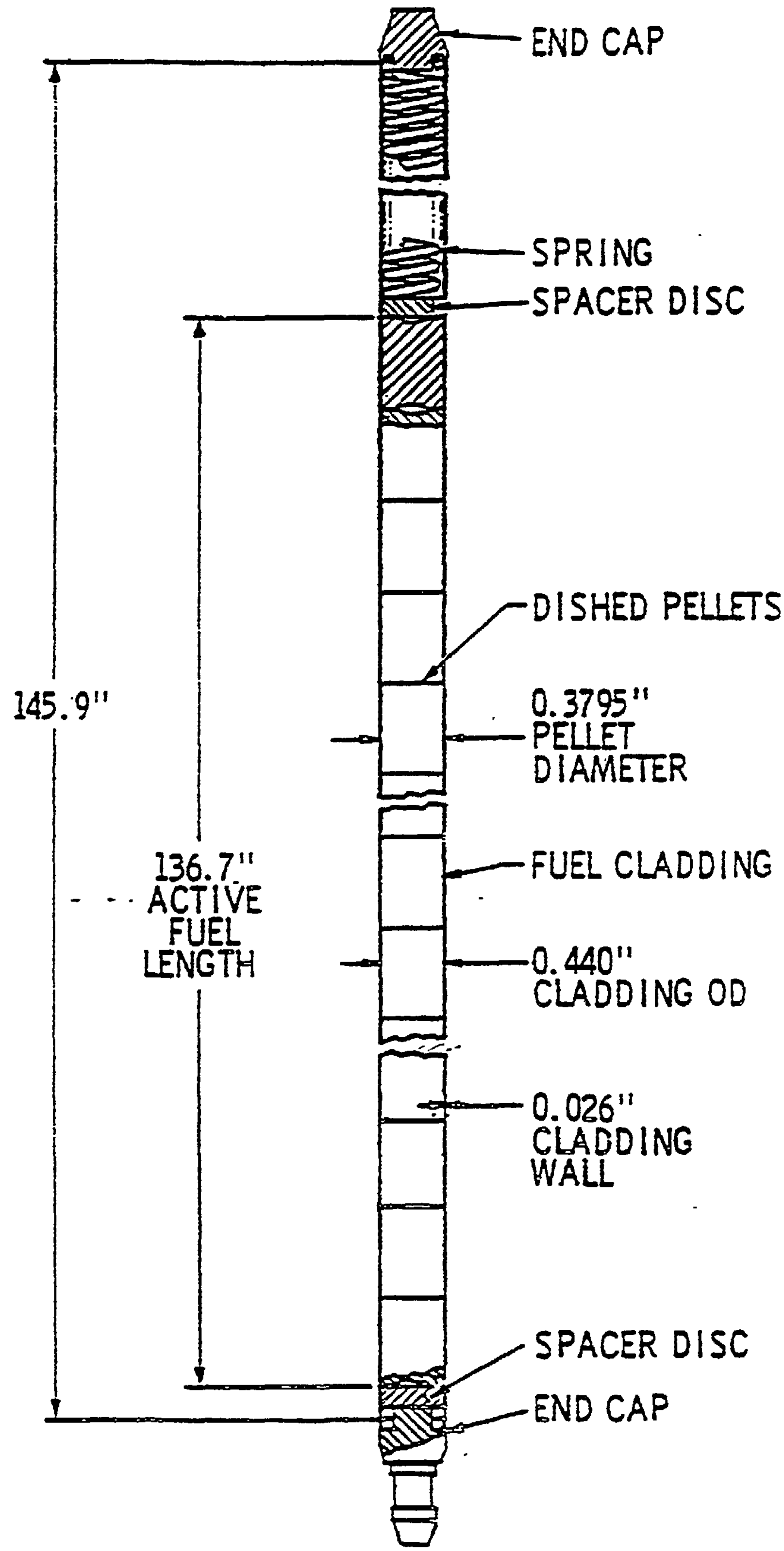


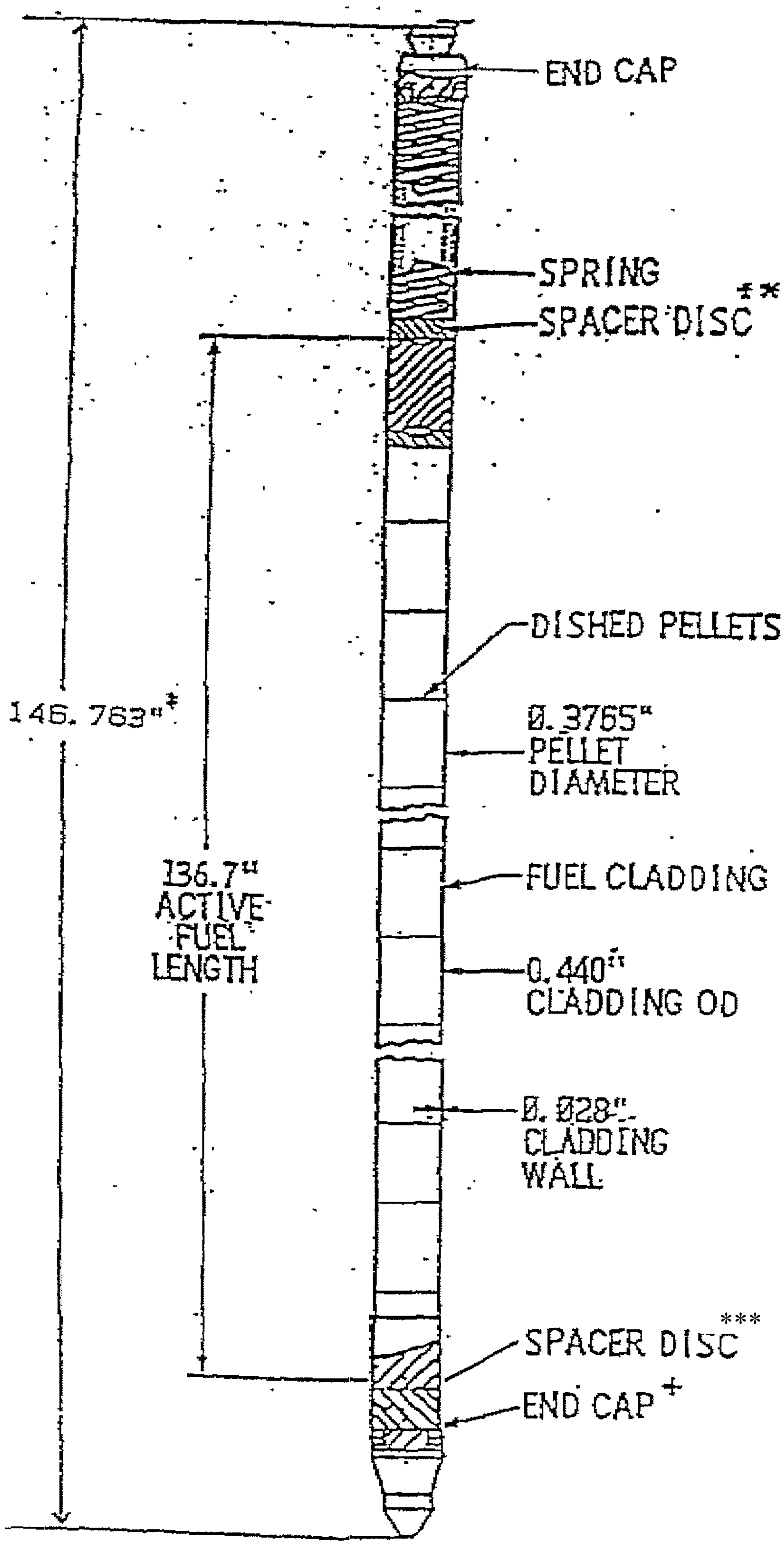
Calvert Cliffs Nuclear Power Plant	REACTOR VERTICAL ARRANGEMENT	Figure 3.1-1 Revision 39



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Reactor Core Cross-Section	Figure 3.3-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FIRST CYCLE FUEL ROD	Figure 3.3-2



The AREVA HTP fuel assemblies use a 0.0265" cladding wall, a 0.3805" pellet diameter, and a rod length on 146.67"

Value Added Pellet fuel assemblies use a 0.026” cladding wall, a 0.3810” pellet diameter, and a rod length of 146.903” (Hematite) and 146.955” (Columbia)

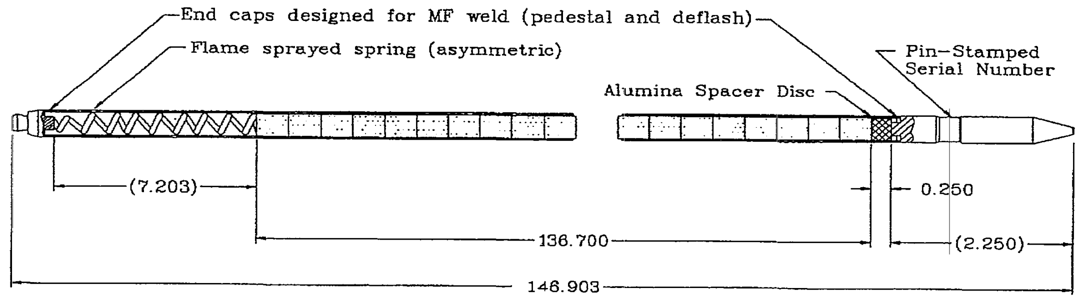
Beginning with Unit 2 Cycle 11 and Unit 1 Cycle 13, the lower end cap was redesigned to one taper, rather than the taper – flat – taper design as shown.

* The GUARDIAN™ fuel design has a total length of 147.229"

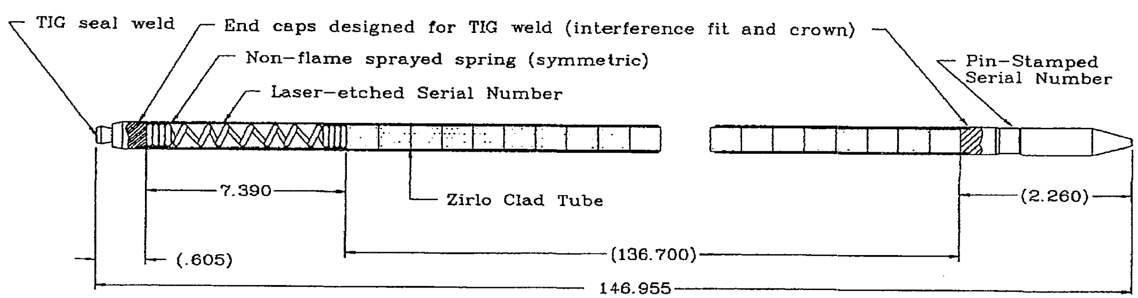
** Upper spacer disc removed beginning with Unit 1 Cycle 12 and Unit 2 Cycle 11.

*** Lower spacer disc removed for rods manufactured at Columbia.

Calvert Cliffs Nuclear Power Plant	FUEL ROD	Figure 3.3-3 Revision 43



Typical Rod from Hematite(1)



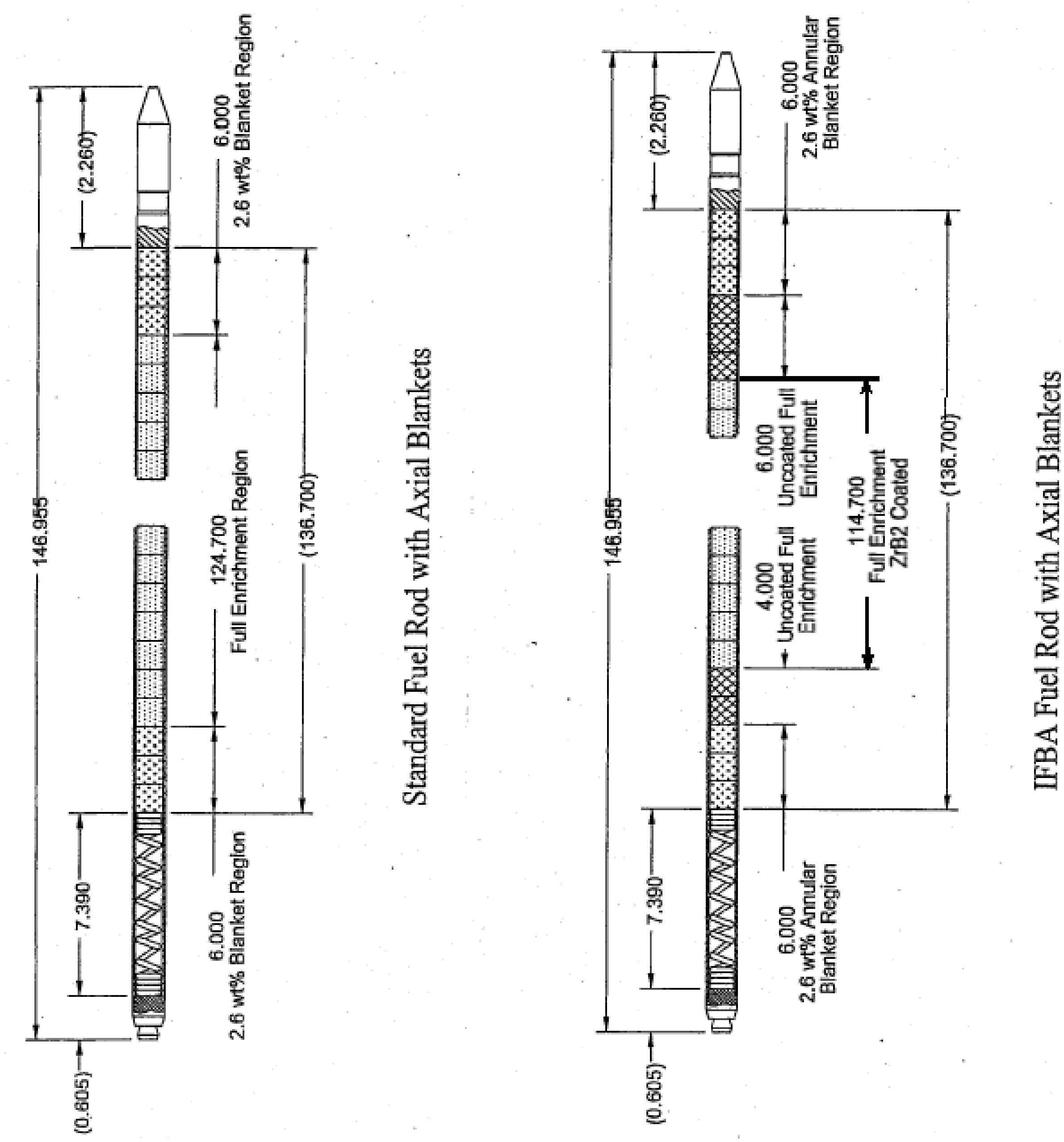
Typical Rod from Columbia(1)

(1) Rod dimensions are nominal values. Batch specific values may vary slightly due to manufacturing practices and tolerances. Significant details for lead fuel assemblies may vary and are described in Section 3.7.

Calvert Cliffs
Nuclear Power Plant

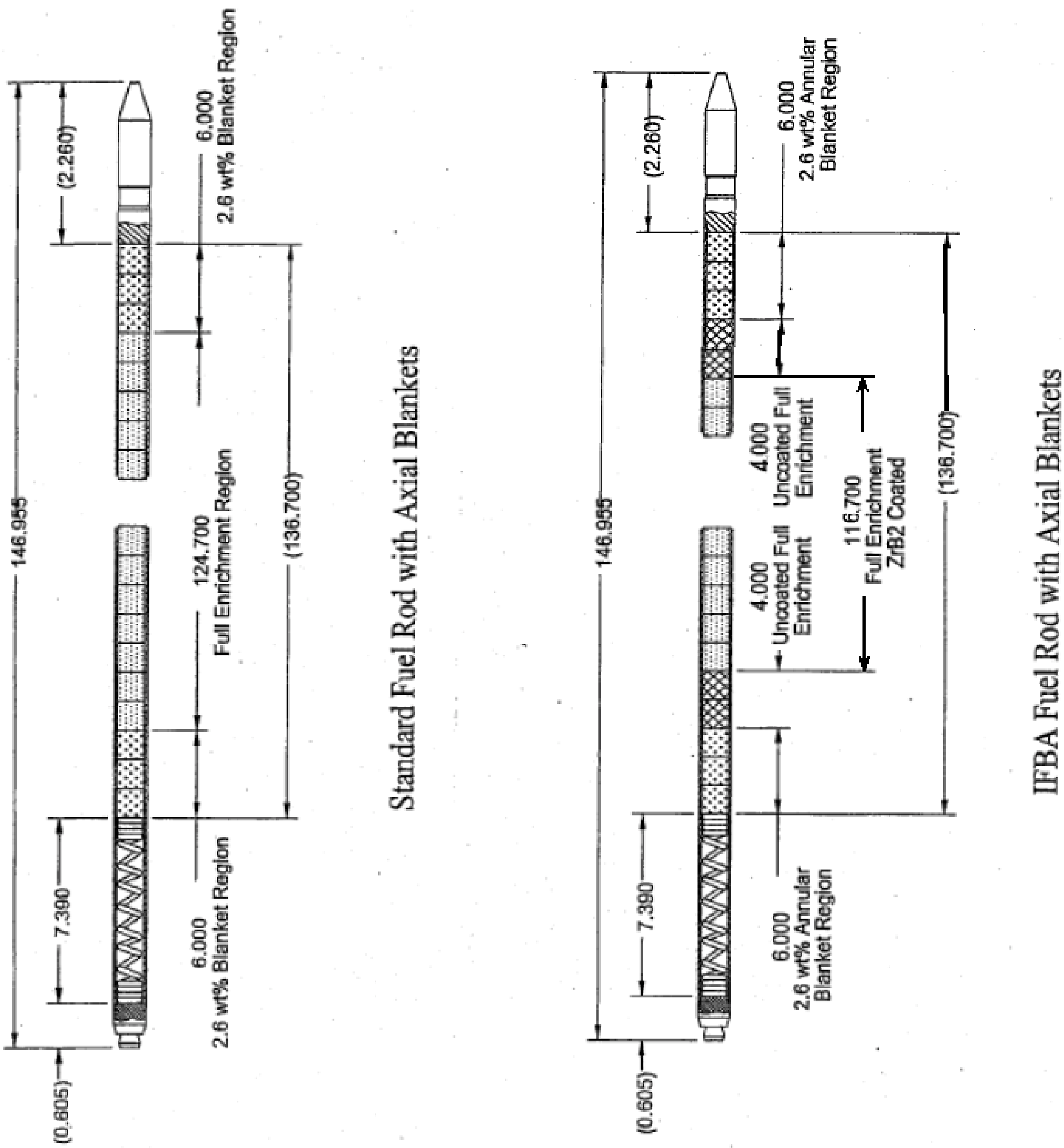
FUEL ROD ASSEMBLY
(Westinghouse)

Figure 3.3-3A
Revision 43



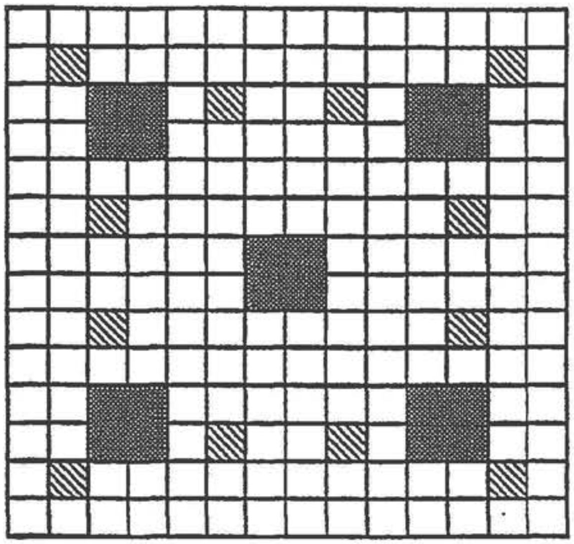
All dimensions are in inches.

Calvert Cliffs Nuclear Power Plant	FUEL ROD DESIGN (UNIT 2 CYCLE 16)	Figure 3.3-3B Revision 40



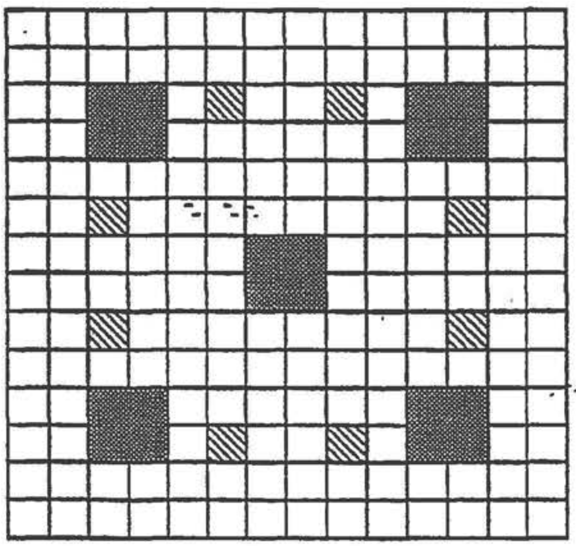
All dimensions are in inches.

Calvert Cliffs Nuclear Power Plant	FUEL ROD DESIGN (UNIT 1 CYCLES 18, 19, & 20 AND UNIT 2 CYCLES 17 & 18)	Figure 3.3-3C Revision 42



12 POISON ROD ASSEMBLY

BUNDLES	
2K*	1M*
2L*	1MX
2F*	
2M3	



8 POISON ROD ASSEMBLY

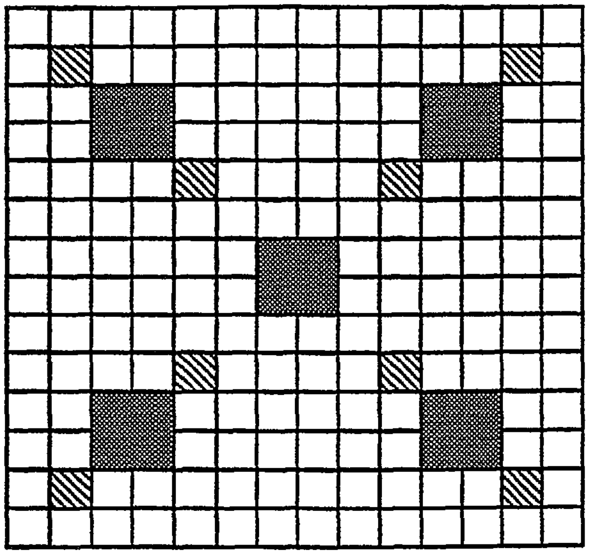
BUNDLES
1G/
2F/

Fuel Rod Location

Poison Rod Location

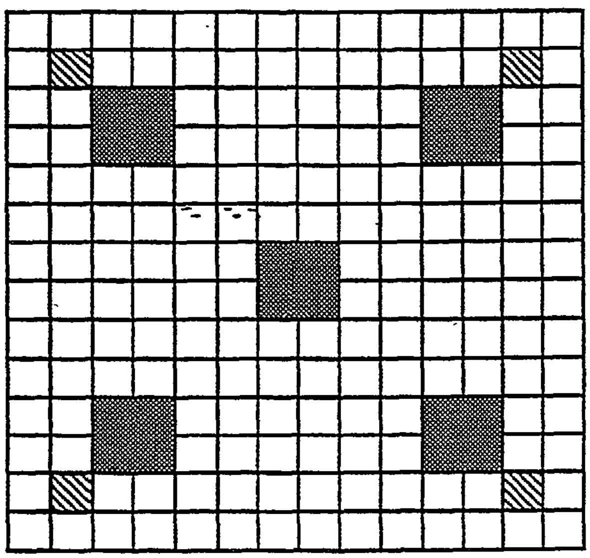
Guide Tube Location

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION	Figure 3.3-4 Sheet 1 Rev. 15



8 POISON ROD ASSEMBLY

BUNDLES	
1H/	2M2
2G/	
2K/	
2L/	
1N/	



4 POISON ROD ASSEMBLY

BUNDLES
2LX
1NX
2M1

Fuel Rod Location

Poison Rod Location

Guide Tube Location

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION	Figure 3.3-4 Sheet 2 Rev. 15

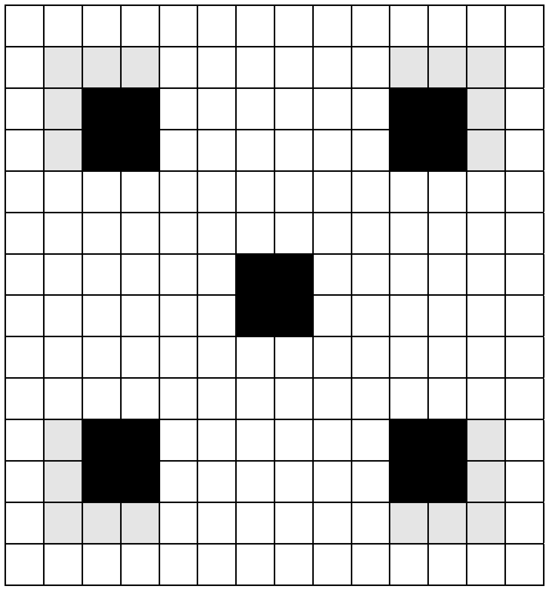
ERBIUM DEMONSTRATION ASSEMBLY

Z	Z	Z	Z	Z					Z	Z	Z	Z	Z
Z	E	E	E	Z					Z	E	E	E	Z
Z	E	Guide Tube		E					E	Guide Tube		E	Z
Z	E			E					E			E	Z
Z	Z	E	E	Z					Z	E	E	Z	Z
					Z	E	E	Z					
					E	Guide Tube		E					
					E			E					
					Z	E	E	Z					
Z	Z	E	E	Z					Z	E	E	Z	Z
Z	E	Guide Tube		E					E	Guide Tube		E	Z
Z	E			E					E			E	Z
Z	E	E	E	Z					Z	E	E	E	Z
Z	Z	Z	Z	Z					Z	Z	Z	Z	Z

Z 3.40 w/o U235 fuel rod
E 3.40 w/o U235 w/ 0.90 w/o Er_2O_3 fuel rod

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION	Figure 3.3-4 Sheet 3 Rev. 18

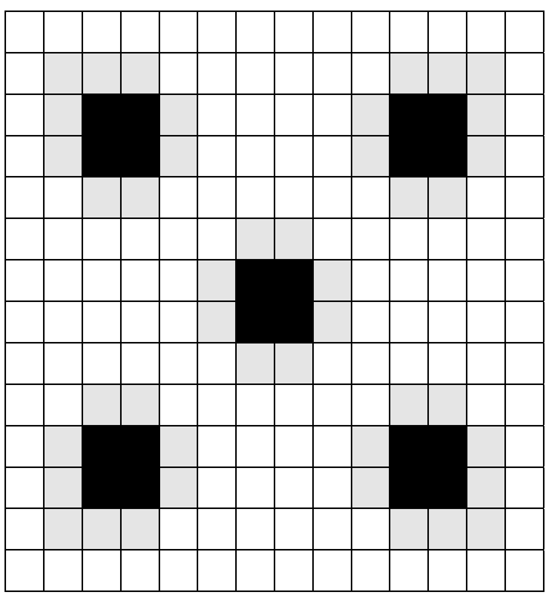
20 Pin Erbium Assembly



Bundles

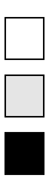
1P1	1R0	1S1		
2N2	2P1	2R1	1T1	2S1
2T1	1W1			

44 Pin Erbium Assembly



Bundles

1P2	1R1	1RT	1S2	
2N4	2NT	2P2	1R2	1T2
2S2	1V1	2T2	1W2	



Calvert Cliffs
Nuclear Power Plant

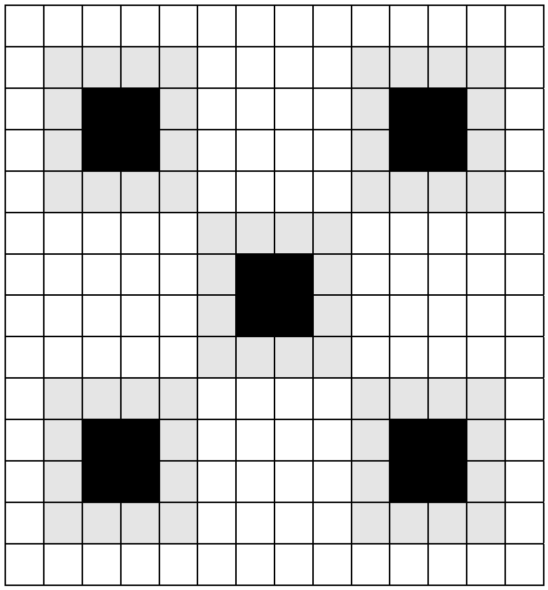
BURNABLE POISON ROD LOCATION

Sheet 4

Figure 3.3-4

Revision 35

60 Pin Erbium Assembly

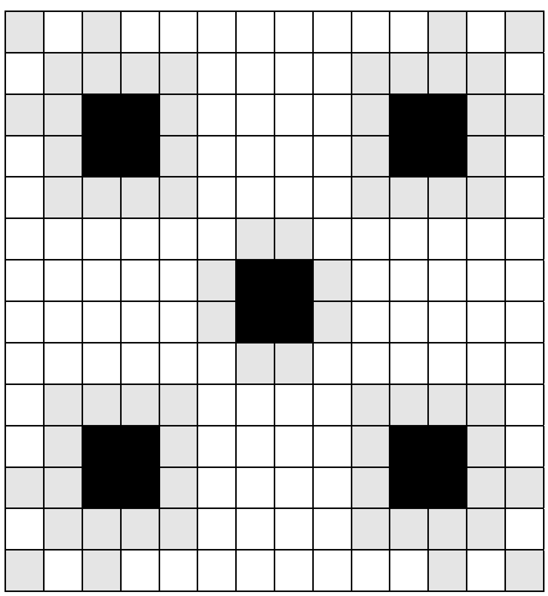


Bundles

1P3 1V2

1W3 1W4

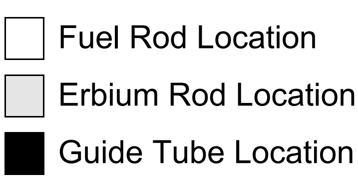
68 Pin Erbium Assembly



Bundles

1R2 1S3

2N6 2R3 2S3 2T3



Calvert Cliffs
Nuclear Power Plant

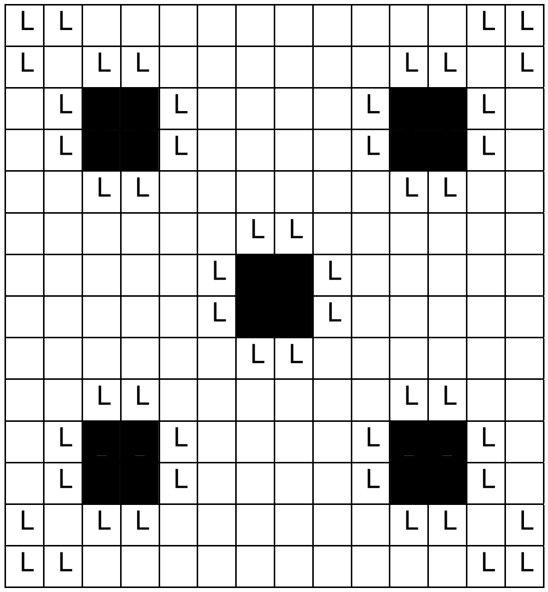
BURNABLE POISON ROD LOCATION

Sheet 5

Figure 3.3-4

Revision 35

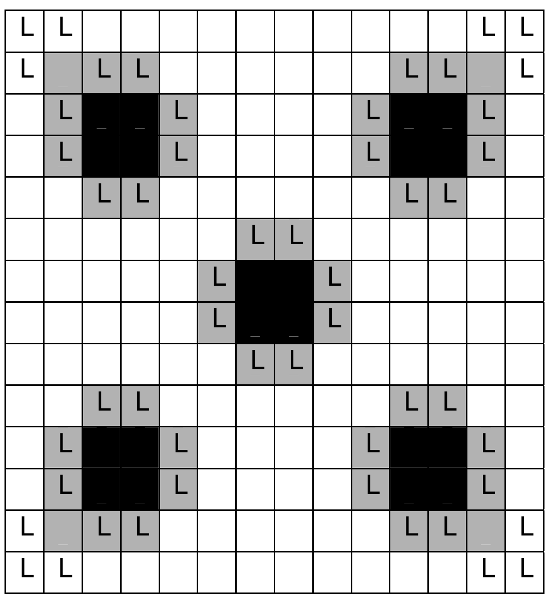
Radial Zoned Assembly with No IFBA Pins



Bundles

2V0, 1Z1

44 IFBA Pin Radial Zoned Assembly



Bundles

2V1, 1X1, 1Z3

L Low Enriched Rod Location with No IFBA

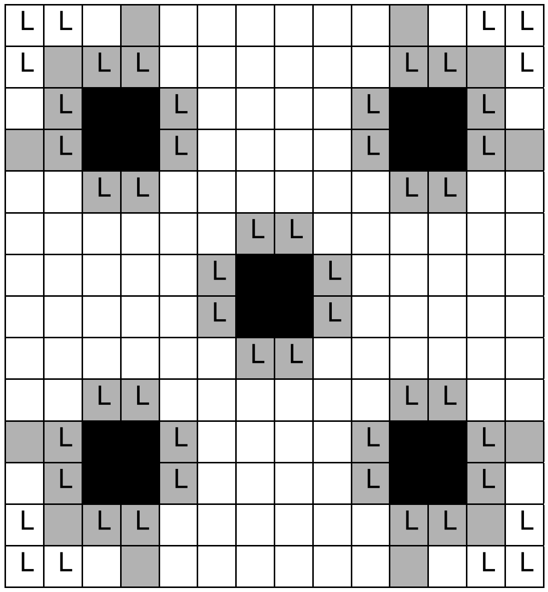
High Enriched Rod Location with No IFBA

L IFBA Low Enriched Rod Location

IFBA High Enriched Rod Location

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 6	Figure 3.3-4 Revision 40

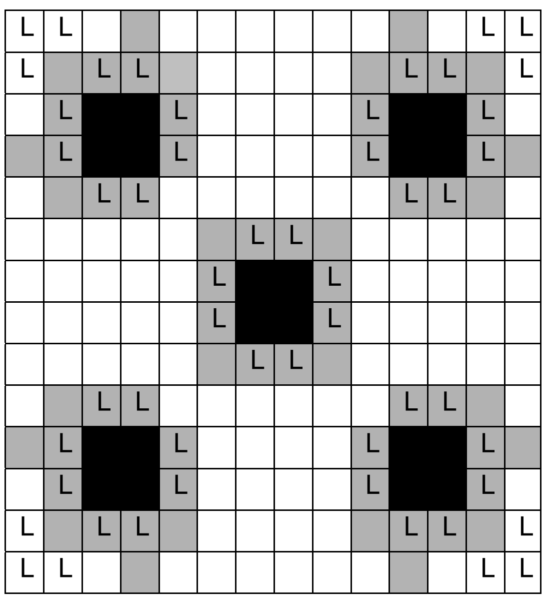
52 IFBA Pin Radial Zoned Assembly



Bundles

2V2	2X2
1X2	AA2

64 IFBA Pin Radial Zoned Assembly



Bundles

2V3	2W2
1X3	

L Low Enriched Rod Location with No IFBA

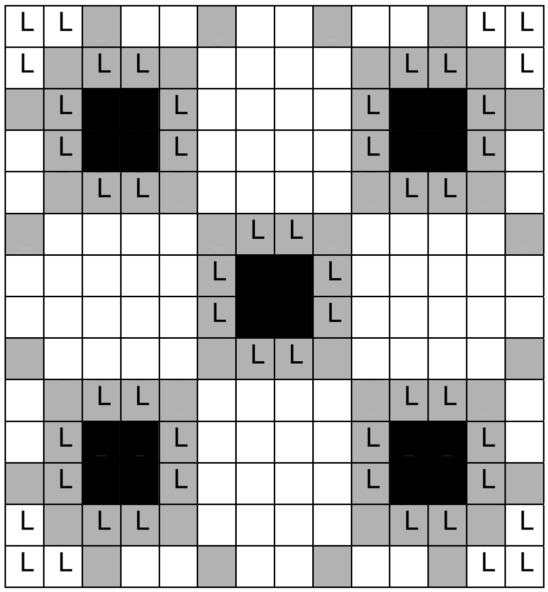
High Enriched Rod Location with No IFBA

L IFBA Low Enriched Rod Location

IFBA High Enriched Rod Location

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 7	Figure 3.3-4 Revision 42

76 IFBA Pin Radial Zoned Assembly

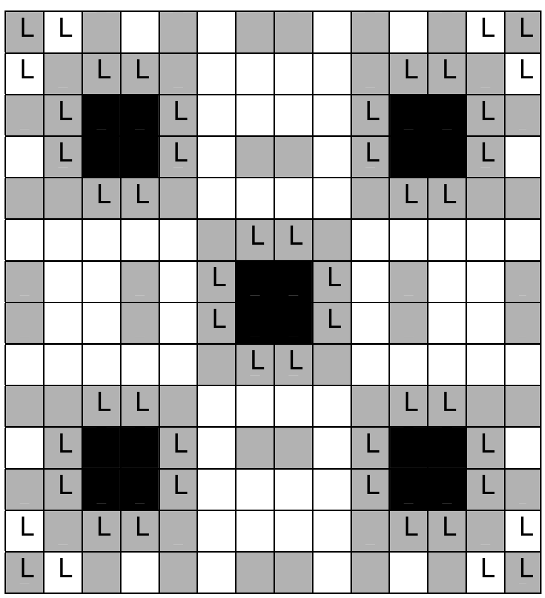


Bundles

2V4

1X4

96 IFBA Pin Radial Zoned Assembly



Bundles

2V5

1X5

L Low Enriched Rod Location with No IFBA

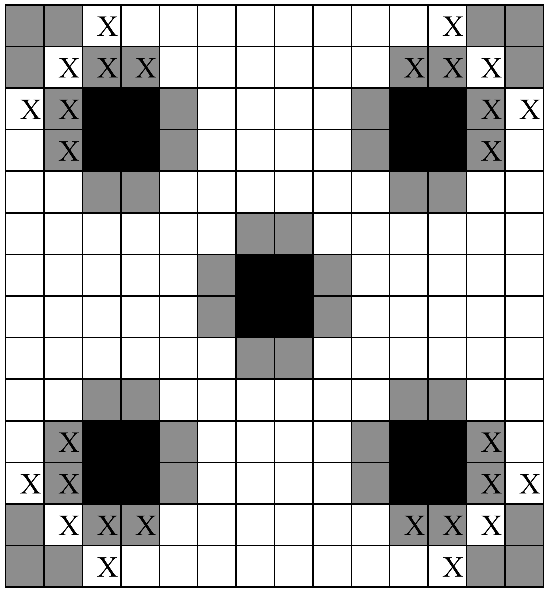
High Enriched Rod Location with No IFBA

L IFBA Low Enriched Rod Location

IFBA High Enriched Rod Location

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 8	Figure 3.3-4 Revision 38

28 IFBA Pin Radial Zoned Assembly



Bundles

2W1, 1Z2, 2X1, AA1

High Enriched Fuel Rod

Low Enriched Fuel Rod

X High Enriched ZrB_2 Rod

X Low Enriched ZrB_2 Rod

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 9	Figure 3.3-4 Revision 42

52 ZrB_2 Rods

3	3	2	5	2	2	1	1	2	2	5	2	3	3
3	5	6	6	2	1	1	1	1	2	6	6	5	3
2	6			6	1	1	1	1	6			6	2
5	6			6	2	1	1	2	6			6	5
2	2	6	6	2	2	2	2	2	2	6	6	2	2
2	1	1	2	2	2	6	6	2	2	2	1	1	2
1	1	1	1	2	6			6	2	1	1	1	1
1	1	1	1	2	6			6	2	1	1	1	1
2	1	1	2	2	2	6	6	2	2	2	1	1	2
2	2	6	6	2	2	2	2	2	2	6	6	2	2
5	6			6	2	1	1	2	6			6	5
2	6			6	1	1	1	1	6			6	2
3	5	6	6	2	1	1	1	1	2	6	6	5	3
3	3	2	5	2	2	1	1	2	2	5	2	3	3

Legend:
1. High Enriched Fuel
2. Medium Enriched Fuel
3. Low Enriched Fuel
4. High Enriched with ZrB_2
5. Medium Enriched with ZrB_2
6. Low Enriched with ZrB_2

Bundles

2W3

64 ZrB_2 Rods

3	3	2	5	2	2	1	1	2	2	5	2	3	3
3	5	6	6	5	1	1	1	1	5	6	6	5	3
2	6			6	1	1	1	1	6			6	2
5	6			6	2	1	1	2	6			6	5
2	5	6	6	2	2	2	2	2	2	6	6	5	2
2	1	1	2	2	5	6	6	5	2	2	1	1	2
1	1	1	1	2	6			6	2	1	1	1	1
1	1	1	1	2	6			6	2	1	1	1	1
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2	5	6	6	2	2	2	2	2	2	6	6	5	2
5	6			6	2	1	1	2	6			6	5
2	6			6	1	1	1	1	6			6	2
3	5	6	6	5	1	1	1	1	5	6	6	5	3
3	3	2	5	2	2	1	1	2	2	5	2	3	3

Bundles

2W4, 1Z4, 2X3, 2X6, AA3

76 ZrB_2 Rods

3	3	5	2	2	5	1	1	5	2	2	5	3	3
3	5	6	6	5	1	1	1	1	5	6	6	5	3
5	6			6	1	1	1	1	6			6	5
2	6			6	2	1	1	2	6			6	2
2	5	6	6	5	2	2	2	2	5	6	6	5	2
5	1	1	2	2	5	6	6	5	2	2	1	1	5
1	1	1	1	2	6			6	2	1	1	1	1
1	1	1	1	2	6			6	2	1	1	1	1
5	1	1	2	2	5	6	6	5	2	2	1	1	5
2	5	6	6	5	2	2	2	2	5	6	6	5	2
2	6			6	2	1	1	2	6			6	2
5	6			6	1	1	1	1	6			6	5
3	5	6	6	5	1	1	1	1	5	6	6	5	3
3	3	5	2	2	5	1	1	5	2	2	5	3	3

<u>Legend</u>:

1. High Enriched Fuel
2. Medium Enriched Fuel
3. Low Enriched Fuel
4. High Enriched with ZrB_2
5. Medium Enriched with ZrB_2
6. Low Enriched with ZrB_2

<u>Bundles</u>

2W5, 1Z5, 2X4, AA4

96 ZrB_2 Rods

6	3	5	2	5	2	4	4	2	5	2	5	3	6
3	5	6	6	5	1	1	1	1	5	6	6	5	3
5	6			6	1	1	1	1	6			6	5
2	6			6	2	1	1	2	6			6	2
5	5	6	6	5	5	2	2	5	5	6	6	5	5
2	1	1	2	5	5	6	6	5	5	2	1	1	2
4	1	1	1	2	6			6	2	1	1	1	4
4	1	1	1	2	6			6	2	1	1	1	4
2	1	1	2	5	5	6	6	5	5	2	1	1	2
5	5	6	6	5	5	2	2	5	5	6	6	5	5
2	6			6	2	1	1	2	6			6	2
5	6			6	1	1	1	1	6			6	5
3	5	6	6	5	1	1	1	1	5	6	6	5	3
6	3	5	2	5	2	4	4	2	5	2	5	3	6

<u>Bundles</u>

2W6, 1Z6, 2X5, AA5

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 11	Figure 3.3-4 Revision 42

0 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	H	L	L	H	H	H	H	H	H	L	L	H	L
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
L	H	L	L	H	H	H	H	H	H	L	L	H	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles

2Z1

4 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	2	L	L	H	H	H	H	H	H	L	L	2	L
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
L	2	L	L	H	H	H	H	H	H	L	L	2	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

2Z2

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	2	L	L	H	H	H	H	H	H	L	L	2	L
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
L	2	L	L	H	H	H	H	H	H	L	L	2	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles

2Z3

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	8	L	L	H	H	H	H	H	H	L	L	8	L
H	L			L	8	H	H	8	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	8	H	H	8	L			L	H
L	8	L	L	H	H	H	H	H	H	L	L	8	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

2Z4

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 13

Figure 3.3-4

Revision 43

12 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	8	L	L	H	H	H	H	H	H	L	L	8	L
H	L			L	H	H	H	H	L			L	H
H	L			L	8	H	H	8	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	8	H	H	L	L	H	H	8	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	8	H	H	L	L	H	H	8	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	8	H	H	8	L			L	H
H	L			L	H	H	H	H	L			L	H
L	8	L	L	H	H	H	H	H	H	L	L	8	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles

2Z5

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 14

Figure 3.3-4

Revision 43

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	8	H	H	8	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	8	H	H	8	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles

AB1

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

AB2

12 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	H	H	H	H	L			L	H
H	L			L	4	H	H	4	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	4	H	H	L	L	H	H	4	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	4	H	H	L	L	H	H	4	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	4	H	H	4	L			L	H
H	L			L	H	H	H	H	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

- H High Enriched Fuel
- L Low Enriched Fuel
- 4 4 w/o Gd_2O_3

Bundles

AB3

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 16

Figure 3.3-4

Revision 45

No Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	H	L	L	H	H	H	H	H	H	L	L	H	L
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
L	H	L	L	H	H	H	H	H	H	L	L	H	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles

BA1

4 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

BA2

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 17

Figure 3.3-4

Revision 46

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	2	L	L	H	H	H	H	H	H	L	L	2	L
H	L			L	4	H	H	4	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	4	H	H	H	H	4	L	L	H	H
H	H	4	H	H	H	L	L	H	H	H	4	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	4	H	H	H	L	L	H	H	H	4	H	H
H	H	L	L	4	H	H	H	H	4	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	4	H	H	4	L			L	H
L	2	L	L	H	H	H	H	H	H	L	L	2	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles

BA3

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

BA4

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 18

Figure 3.3-4

Revision 46

20 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	6	L	L	H	2	H	H	2	H	L	L	6	L
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	2	H	6	H	H	L	L	H	H	6	H	2	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	2	H	6	H	H	L	L	H	H	6	H	2	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
L	6	L	L	H	2	H	H	2	H	L	L	6	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

<u>Legend</u>:

H High Enriched Fuel
L Low Enriched Fuel
2 2 w/o Gd_2O_3
4 4 w/o Gd_2O_3
6 6 w/o Gd_2O_3
8 8 w/o Gd_2O_3

<u>Bundles</u>

BA5

12 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	H	H	H	H	L			L	H
H	L			L	8	H	H	8	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	8	H	H	L	L	H	H	8	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	8	H	H	L	L	H	H	8	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	8	H	H	8	L			L	H
H	L			L	H	H	H	H	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

<u>Bundles</u>

BA6

Calvert Cliffs Nuclear Power Plant

BURNABLE POISON ROD LOCATION

Sheet 19

Figure 3.3-4

Revision 46

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	6	H	H	H	L	L	H	H	H	6	H	H
H	H	L	L	6	H	H	H	H	6	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles

BA7

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 20	Figure 3.3-4 Revision 46

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	8	H	H	8	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	8	H	H	H	L	L	H	H	H	8	H	H
H	H	L	L	8	H	H	H	H	8	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	8	H	H	8	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles

AC1

12 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	6	L	L	H	H	H	H	H	H	L	L	6	L
H	L			L	H	H	H	H	L			L	H
H	L			L	6	H	H	6	L			L	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	H	H	6	H	H	L	L	H	H	6	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	6	H	H	L	L	H	H	6	H	H	H
H	H	L	L	H	H	H	H	H	H	L	L	H	H
H	L			L	6	H	H	6	L			L	H
H	L			L	H	H	H	H	L			L	H
L	6	L	L	H	H	H	H	H	H	L	L	6	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

Bundles

AC2

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 21	Figure 3.3-4 Revision 47

16 Gd_2O_3 Rods

L	L	H	H	H	H	H	H	H	H	H	H	L	L
L	4	L	L	H	H	H	H	H	H	L	L	4	L
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
H	H	L	L	4	H	H	H	H	4	L	L	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	L			L	H	H	H	H	H
H	H	H	H	H	H	L	L	H	H	H	H	H	H
H	H	L	L	4	H	H	H	H	4	L	L	H	H
H	L			L	H	H	H	H	L			L	H
H	L			L	H	H	H	H	L			L	H
L	4	L	L	H	H	H	H	H	H	L	L	4	L
L	L	H	H	H	H	H	H	H	H	H	H	L	L

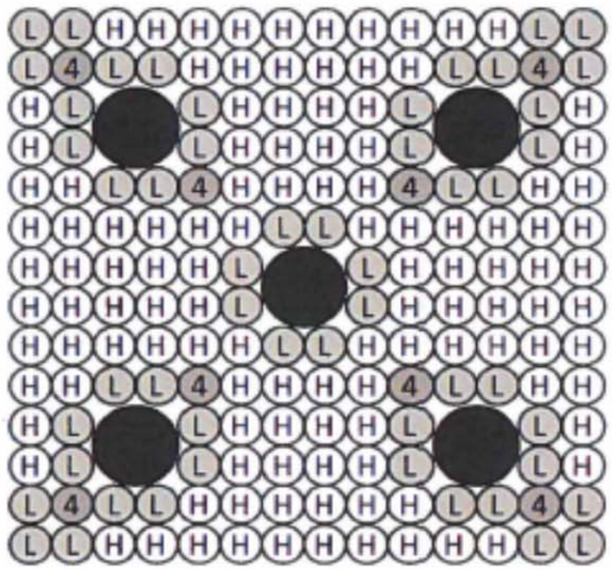
Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles

AC3

Bundles: **BB1**

8 Gd_2O_3 Rods

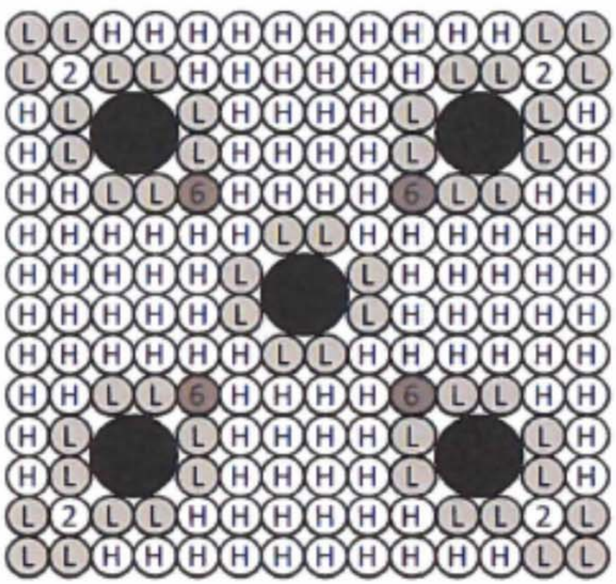


Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles: **BB2**

8 Gd_2O_3 Rods



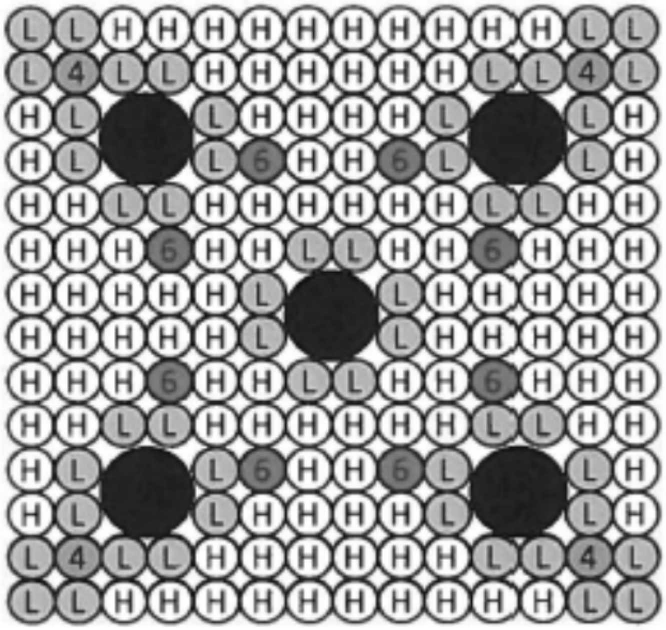
Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 23	Figure 3.3-4 Revision 48

Bundles: **BB3**

12 Gd_2O_3 Rods

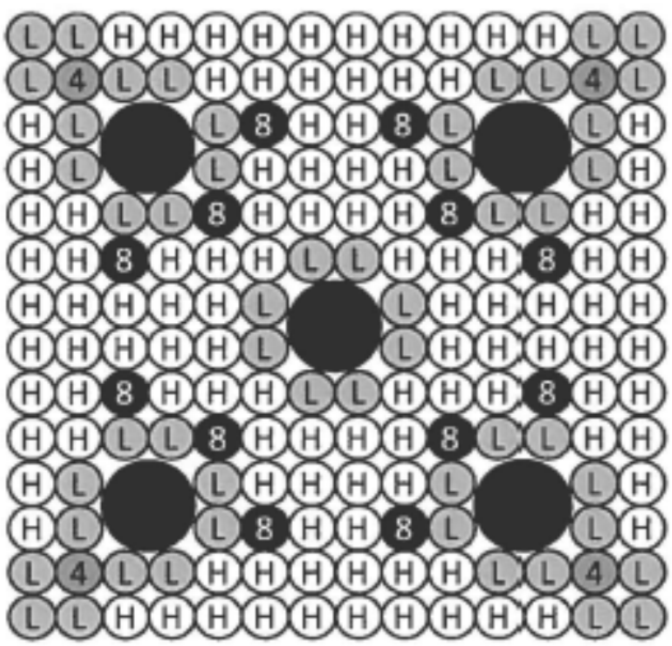


Legend:

H High Enriched Fuel
L Low Enriched Fuel
2 2 w/o Gd_2O_3
4 4 w/o Gd_2O_3
6 6 w/o Gd_2O_3
8 8 w/o Gd_2O_3

Bundles: **BB4**

16 Gd_2O_3 Rods



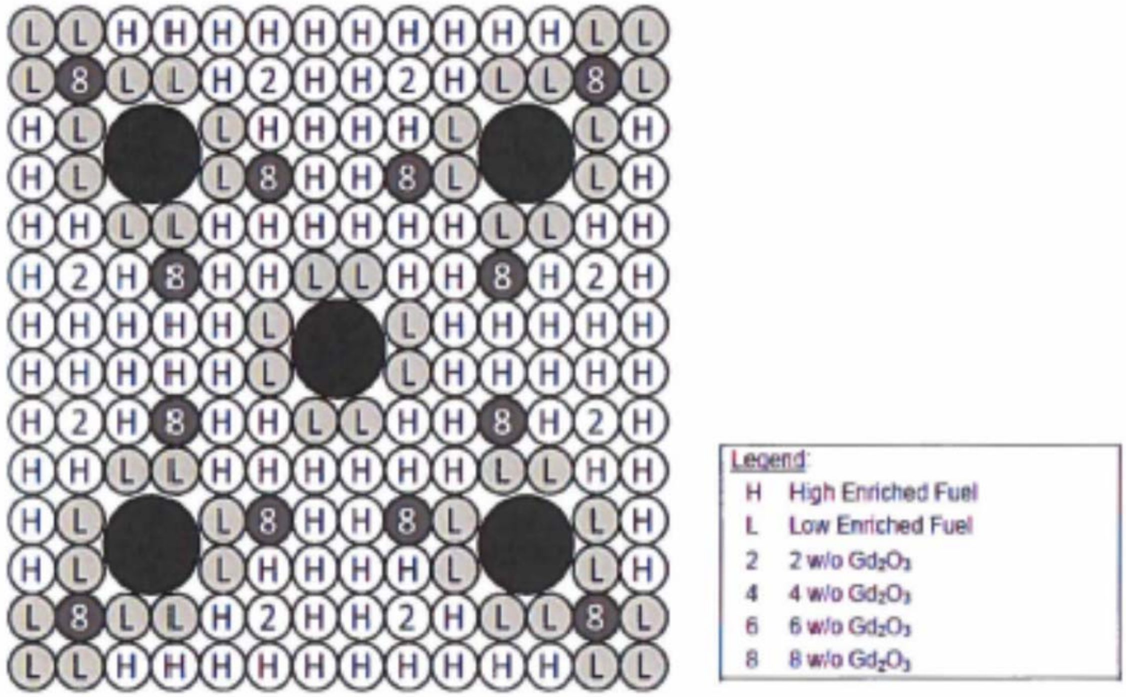
Legend:

H High Enriched Fuel
L Low Enriched Fuel
2 2 w/o Gd_2O_3
4 4 w/o Gd_2O_3
6 6 w/o Gd_2O_3
8 8 w/o Gd_2O_3

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION Sheet 24	Figure 3.3-4 Revision 48

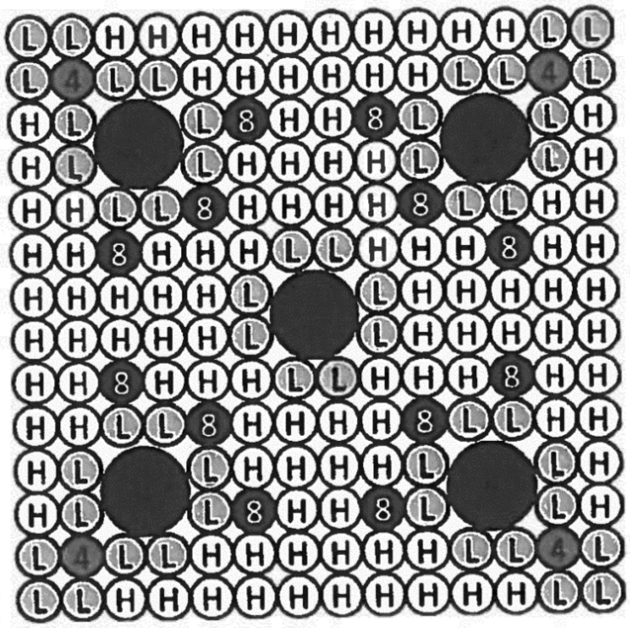
Bundles: BB5

20 Gd_2O_3 Rods



Bundles: AD1

16 Gd_2O_3 Rods

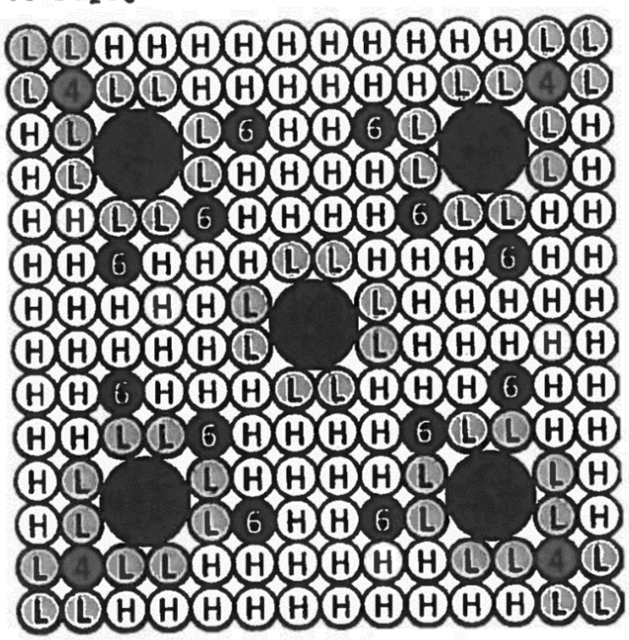


Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles: AD2

16 Gd_2O_3 Rods

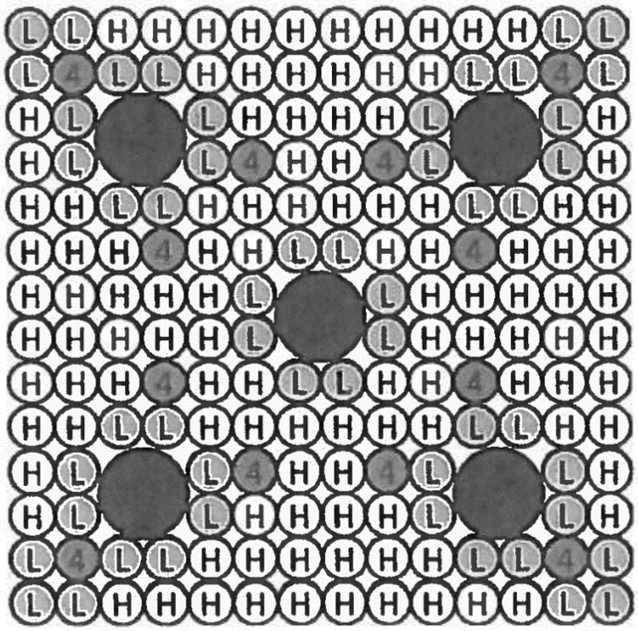


Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

Bundles: AD3

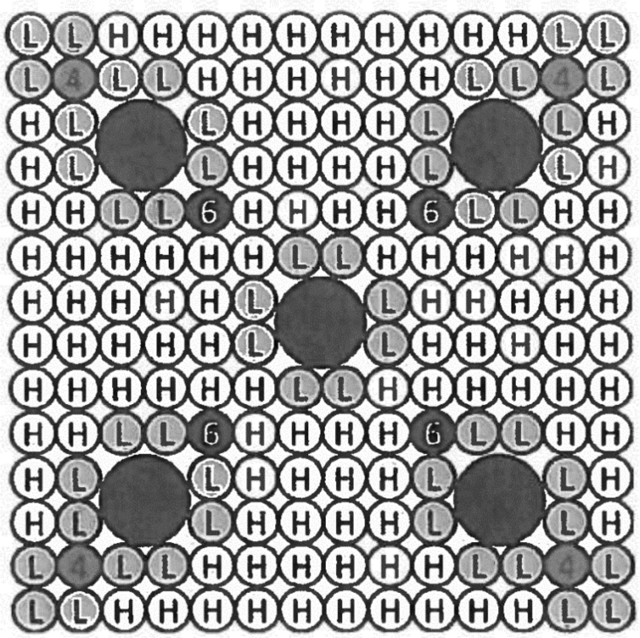
12 Gd_2O_3 Rods



Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles: AD4

8 Gd_2O_3 Rods

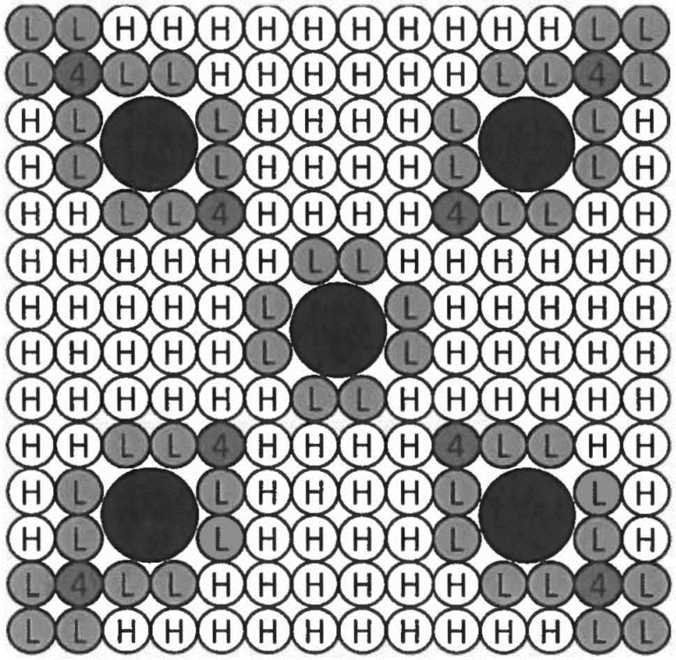


Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION SHEET 27	Figure 3-3-4 Revision 49

Bundles: BC1

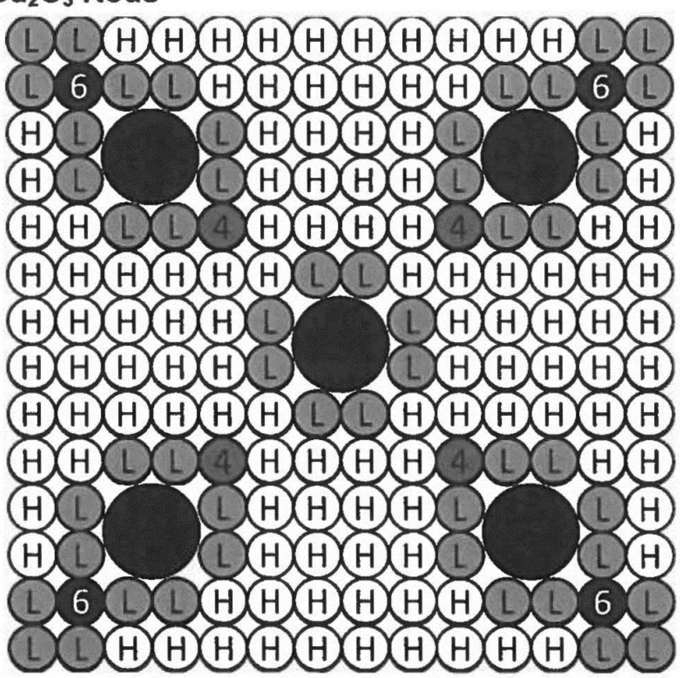
8 Gd_2O_3 Rods



Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles: BC2

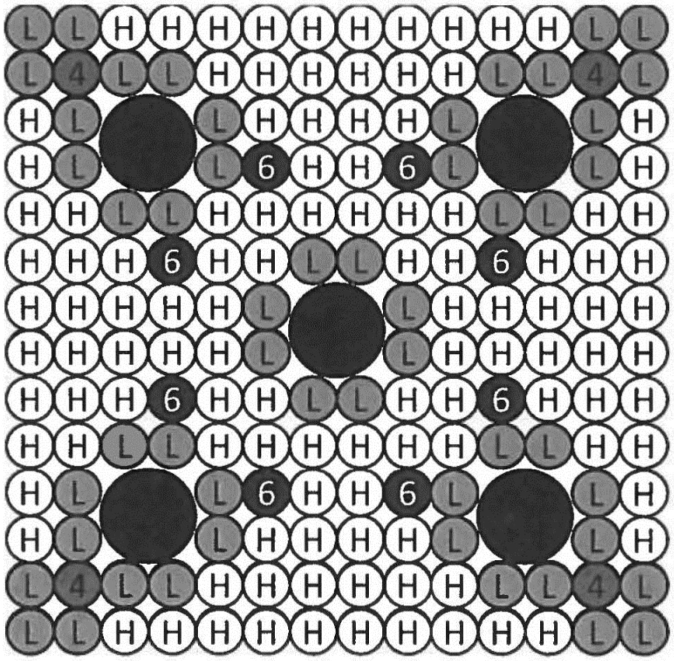
8 Gd_2O_3 Rods



Legend:
- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles: **BC3**

12 Gd_2O_3 Rods

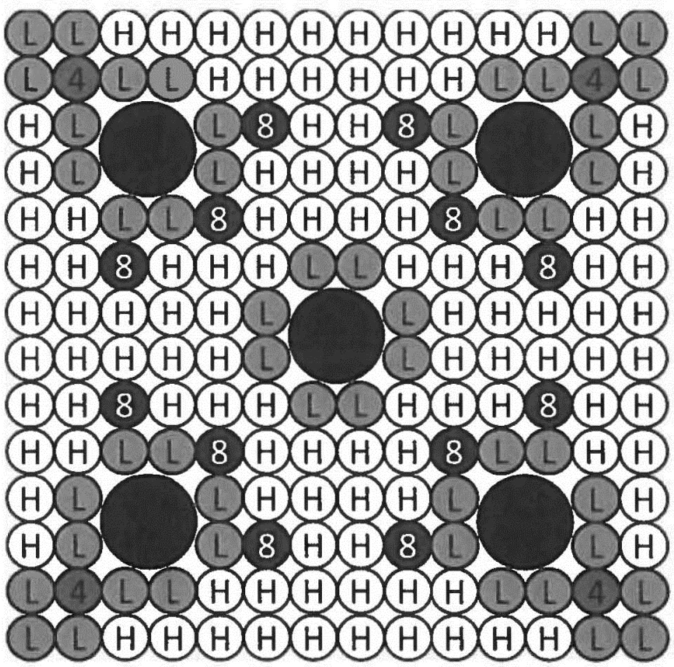


Legend:

- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Bundles: **BC4**

16 Gd_2O_3 Rods



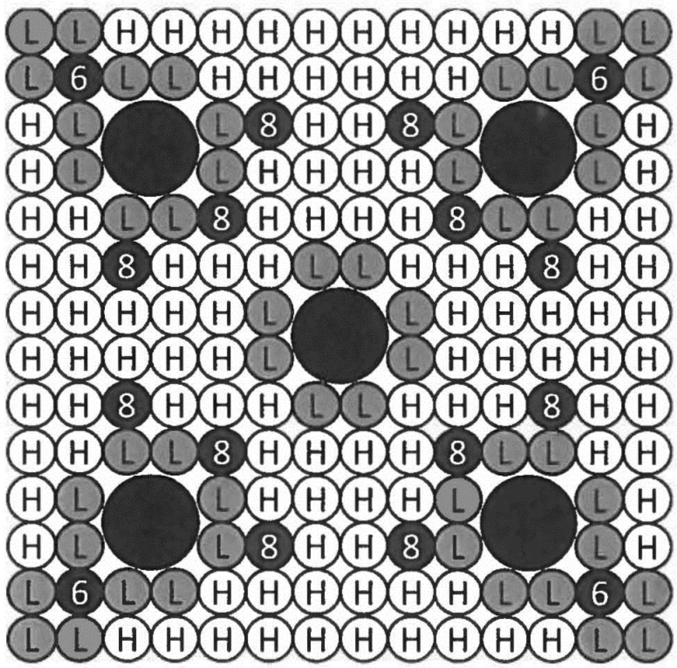
Legend:

- H High Enriched Fuel
- L Low Enriched Fuel
- 2 2 w/o Gd_2O_3
- 4 4 w/o Gd_2O_3
- 6 6 w/o Gd_2O_3
- 8 8 w/o Gd_2O_3

Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION SHEET 29	Figure 3-3-4 Revision 49

Bundles: BC5

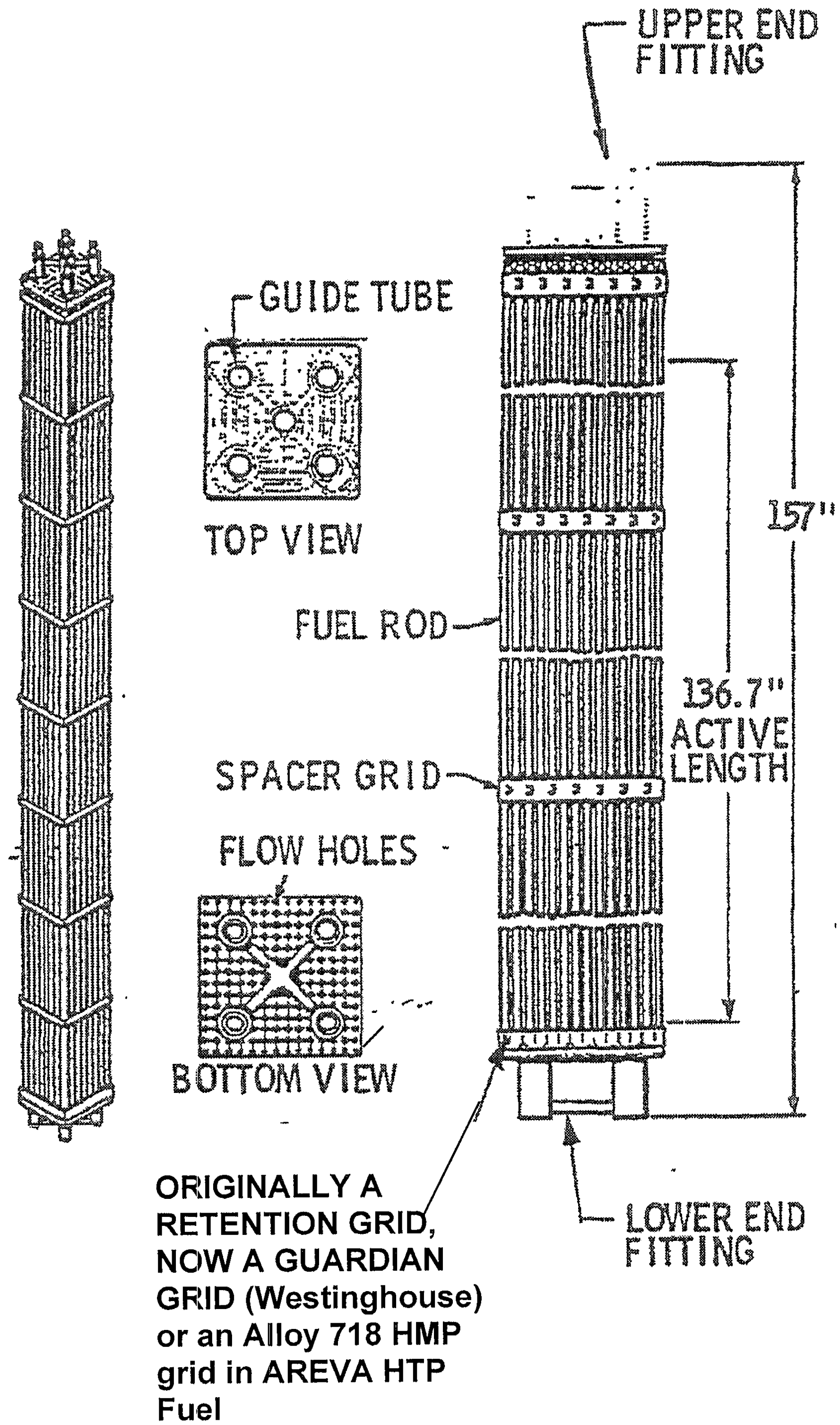
16 Gd_2O_3 Rods



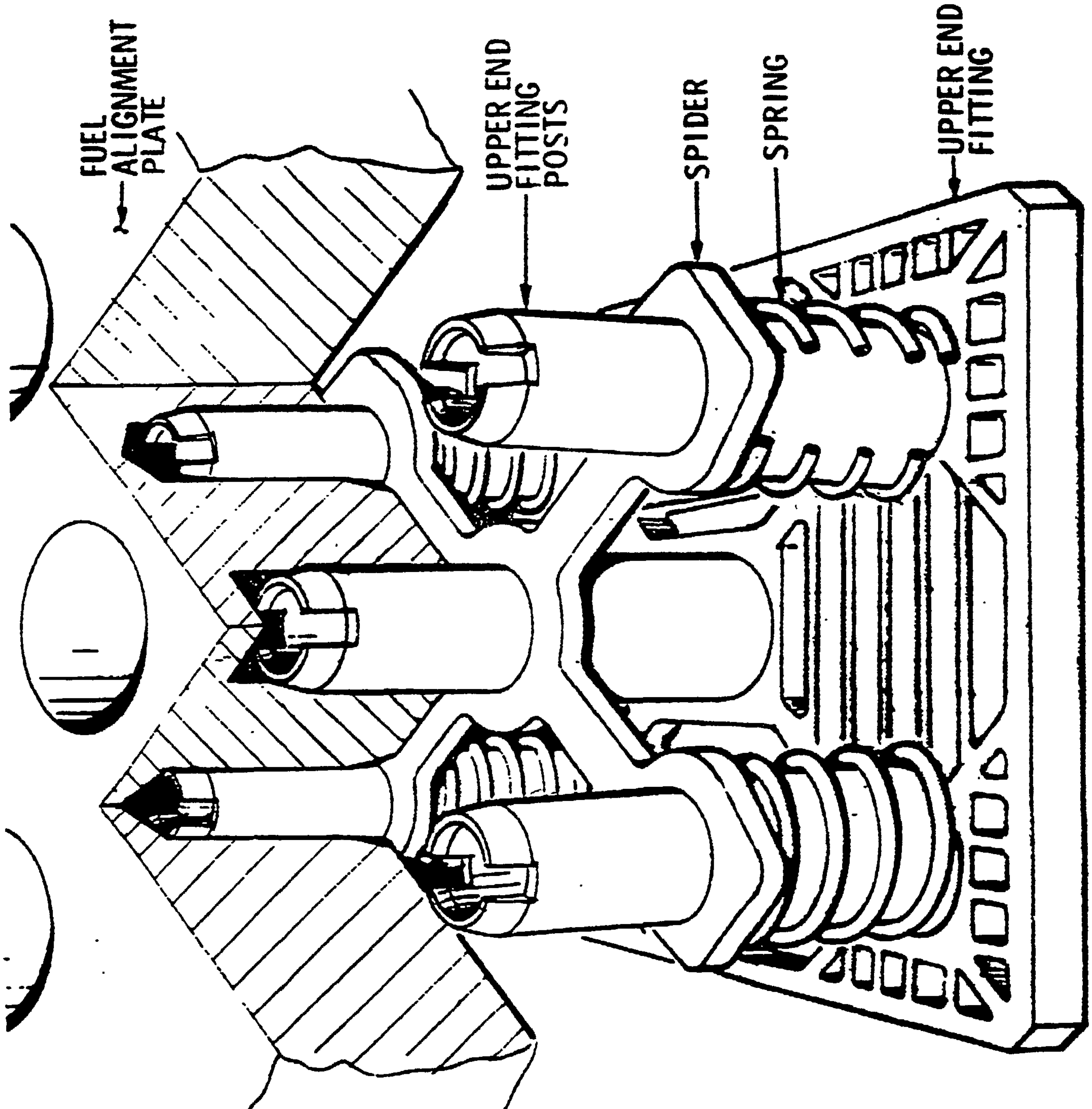
Legend:

H	High Enriched Fuel
L	Low Enriched Fuel
2	2 w/o Gd_2O_3
4	4 w/o Gd_2O_3
6	6 w/o Gd_2O_3
8	8 w/o Gd_2O_3

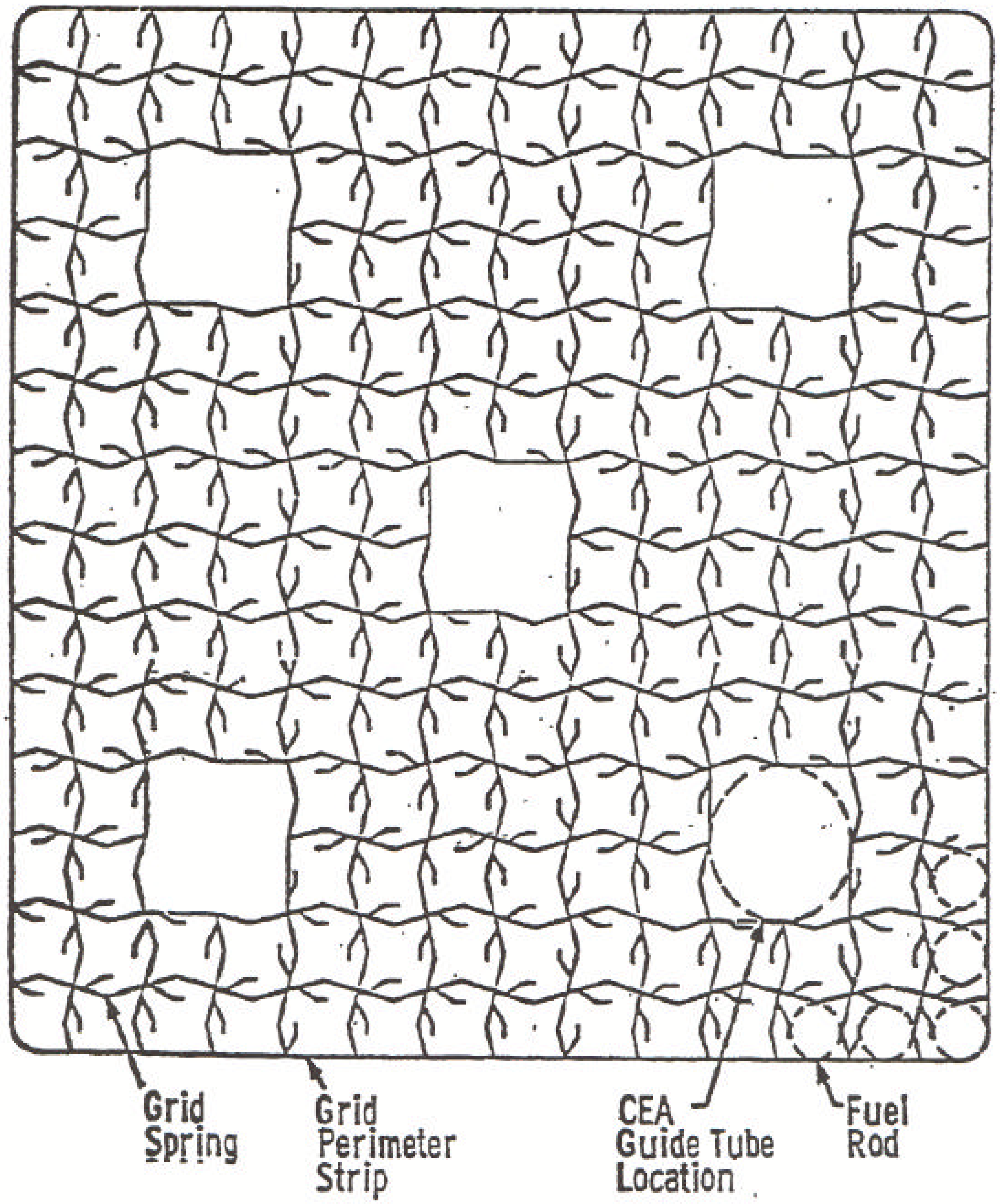
Calvert Cliffs Nuclear Power Plant	BURNABLE POISON ROD LOCATION SHEET 30	Figure 3-3-4 Revision 49



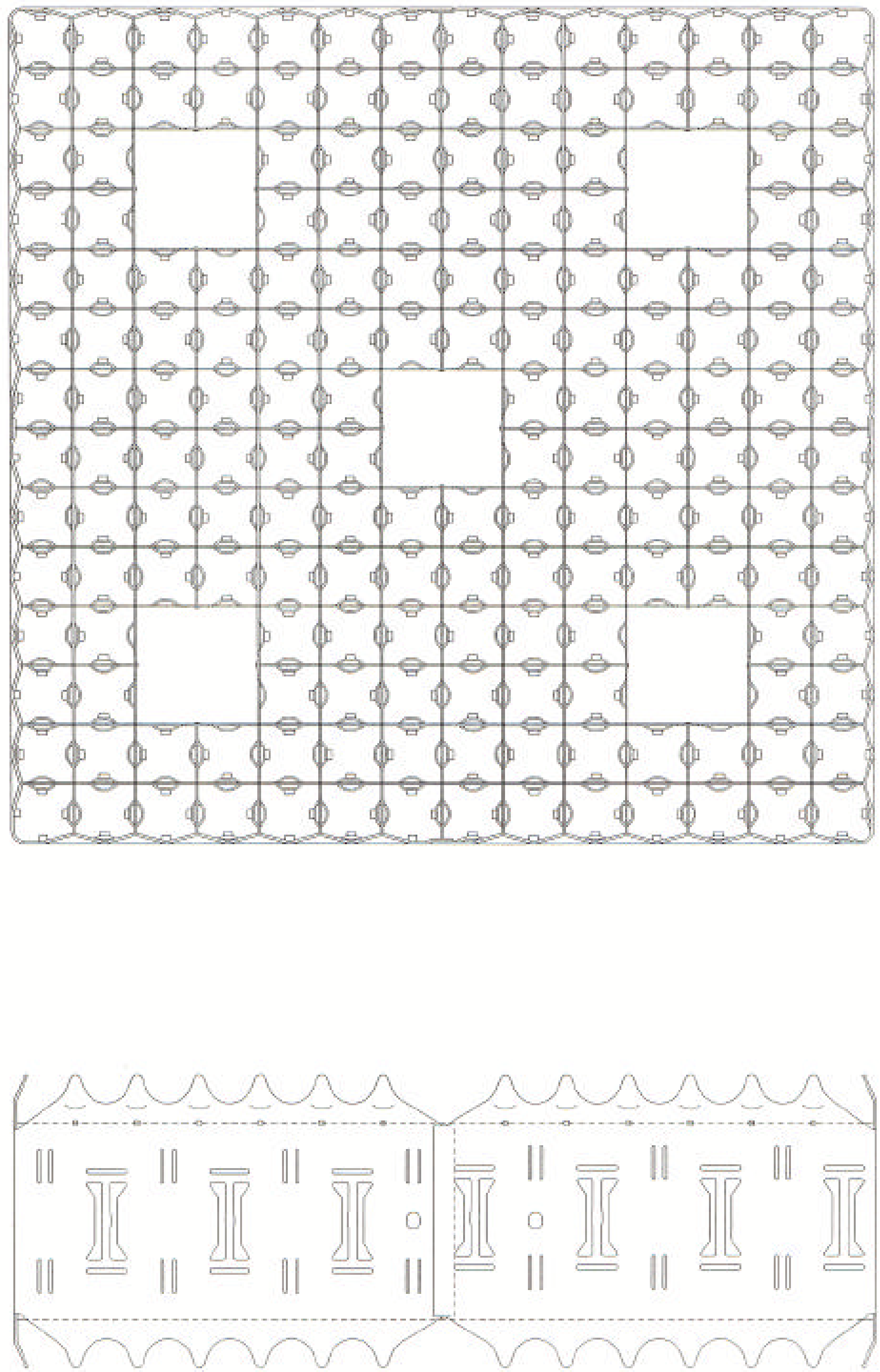
Calvert Cliffs Nuclear Power Plant	FUEL ASSEMBLY	Figure 3.3-5 Revision 43



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Fuel Assembly Hold Down	Figure 3.3-6



Calvert Cliffs Nuclear Power Plant	CANTILEVER TAB FUEL SPACER GRID	Figure 3.3-7 Revision 32

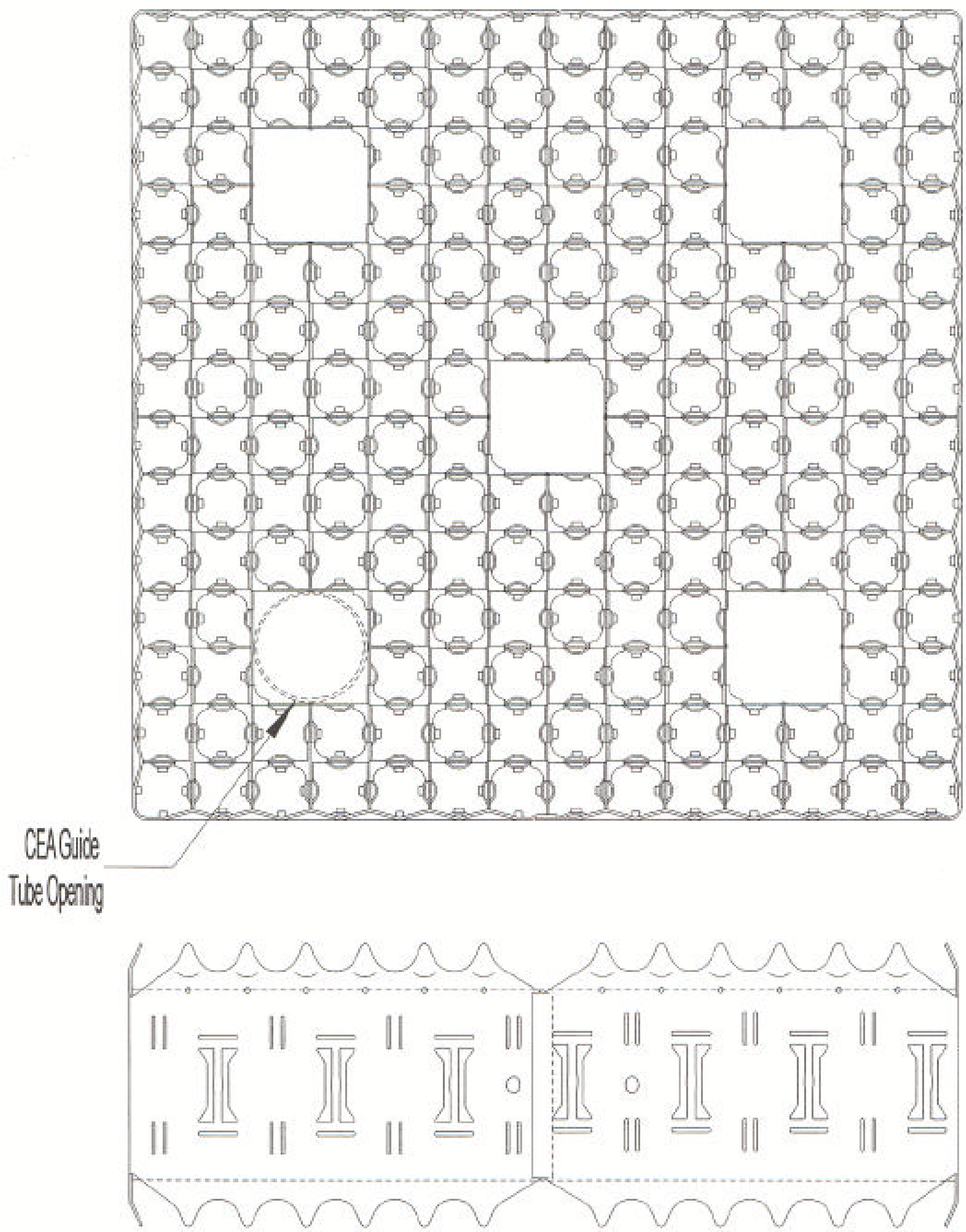


Calvert Cliffs
Nuclear Power Plant

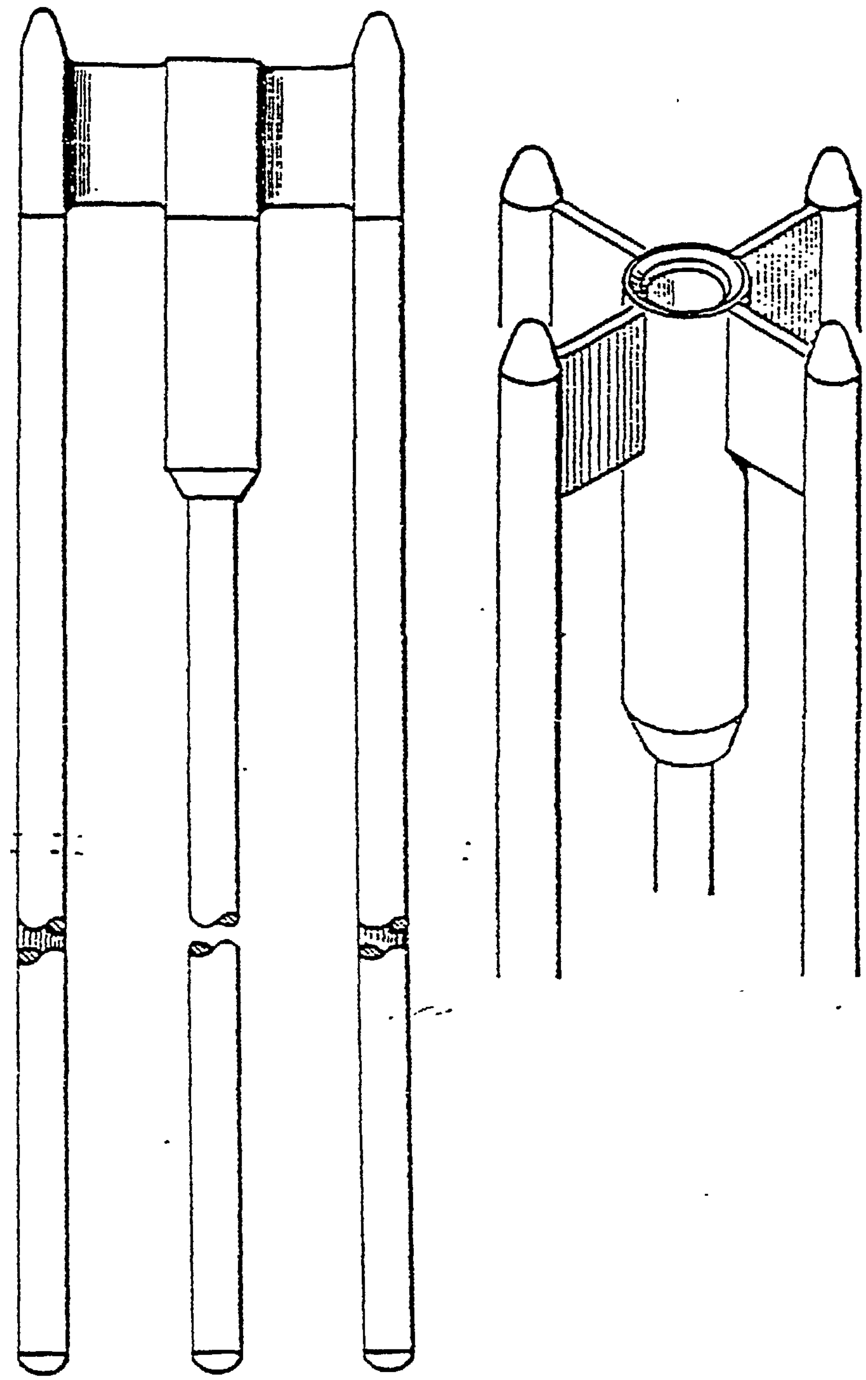
I-SPRING UNVANED SPACER GRID (TURBO)

Figure 3.3-7A

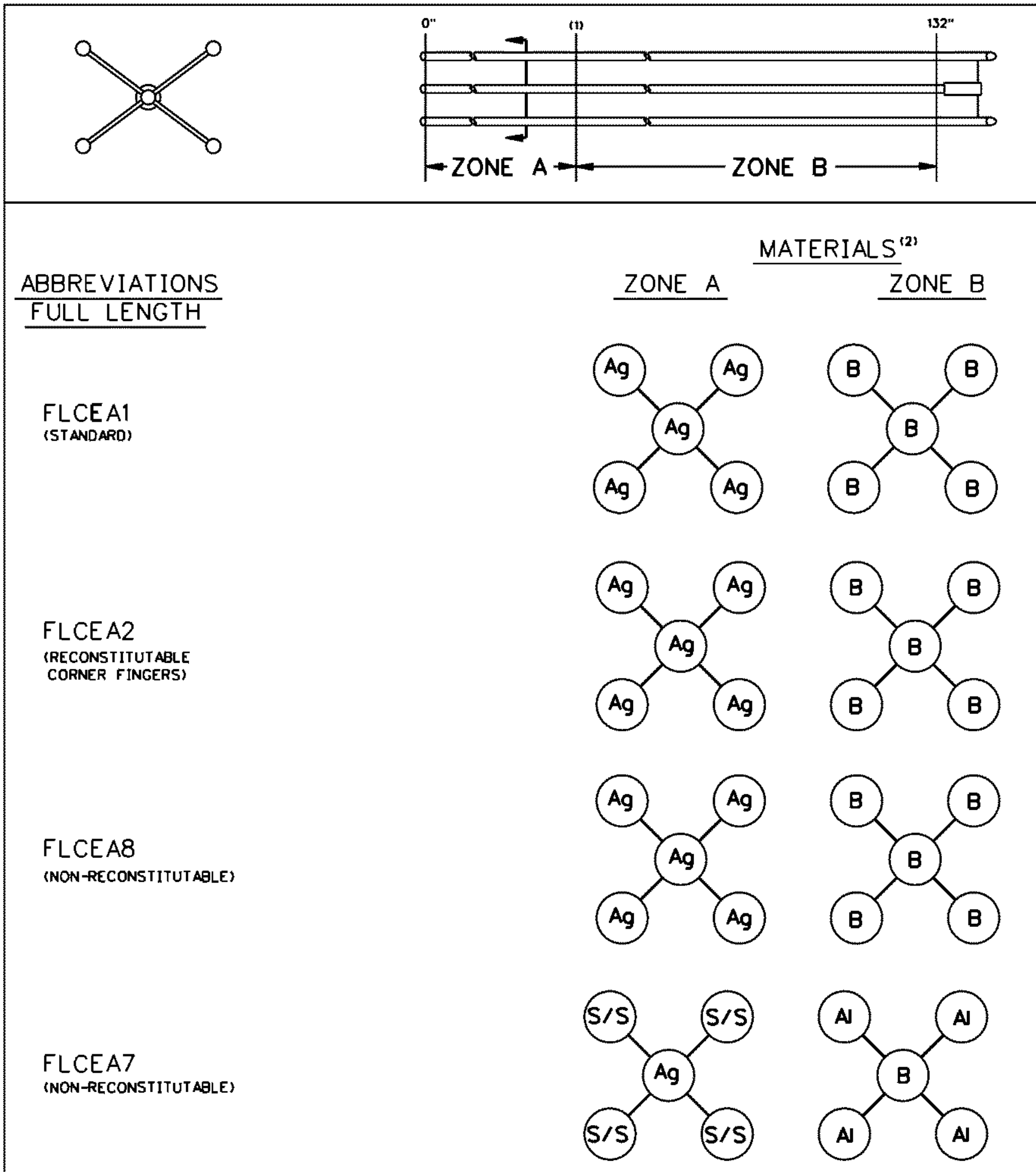
Revision 32



Calvert Cliffs Nuclear Power Plant	I-SPRING VANED SPACER GRID (TURBO)	Figure 3.3-7B Revision 32



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CONTROL ELEMENT ASSEMBLY (CEA)	FIGURE 3.3-8 Rev. 0 1/82



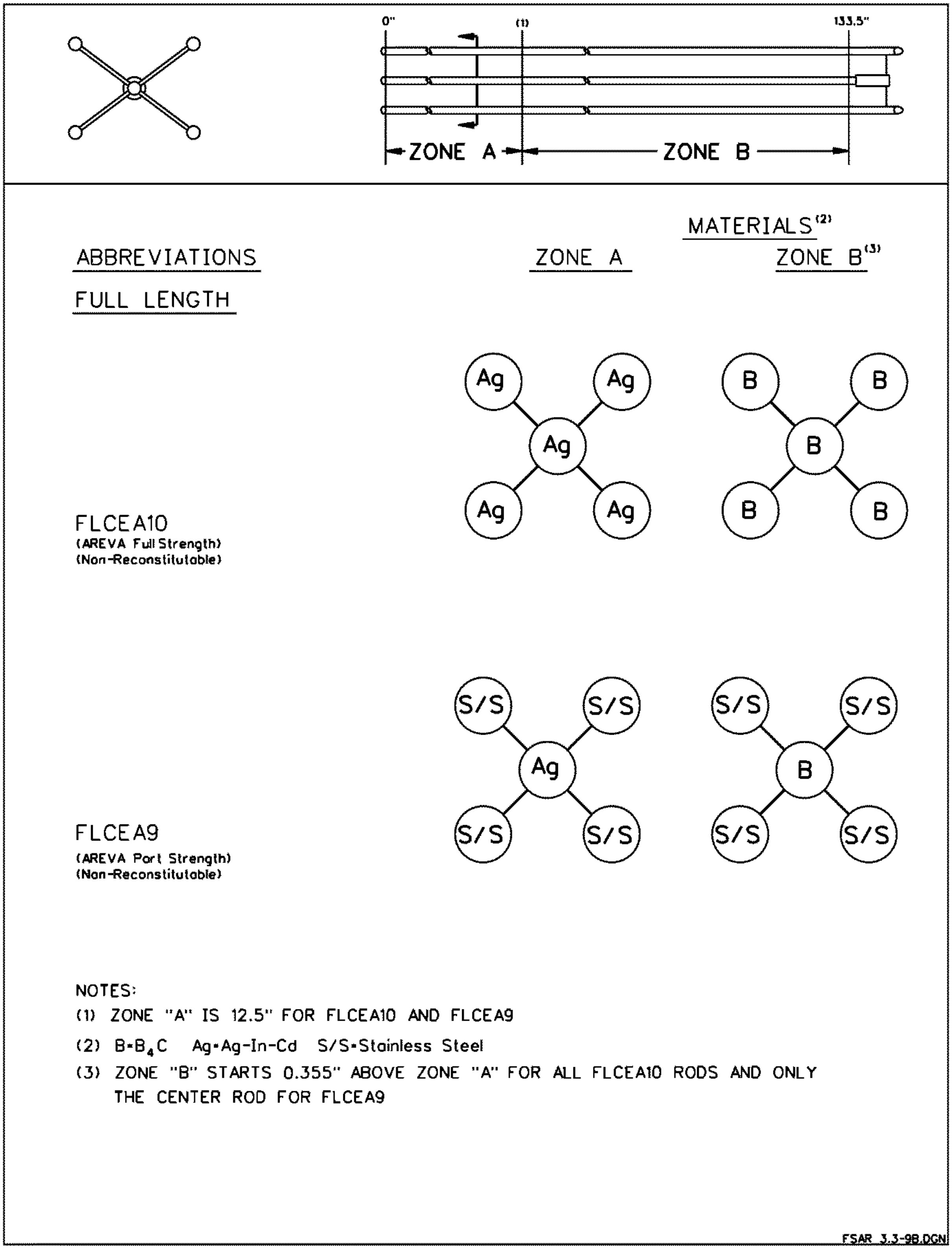
NOTES:

(1) ZONE "A" IS 8" FOR FLCEA1, FLCEA2, AND FLCEA7, AND 12" FOR FLCEA8.

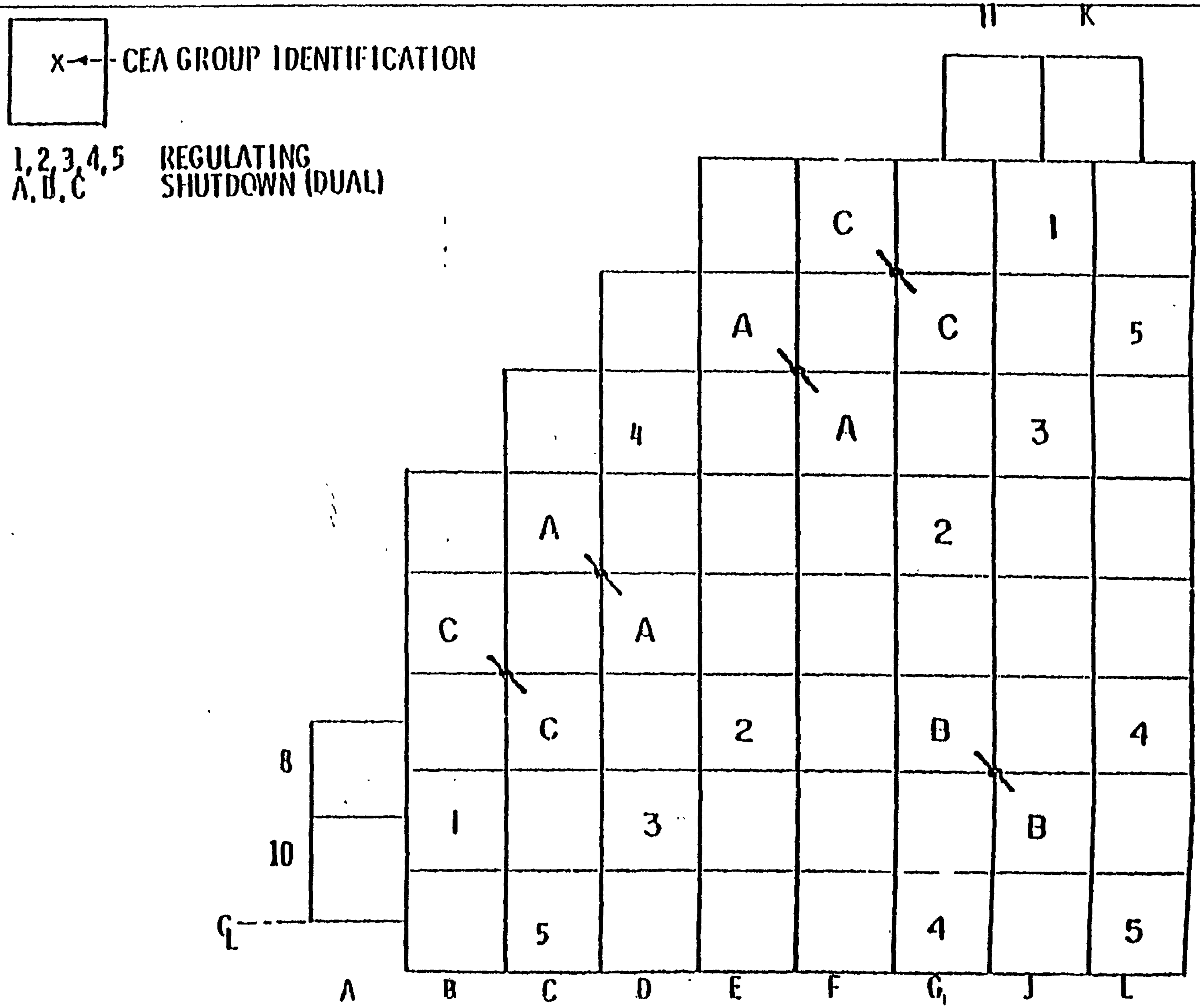
(2) B=B_4C Ag=Ag-In-Cd Al=Al_2O_3 S/S=Stainless Steel

FSAR 3.3-9A.DGN

Calvert Cliffs Nuclear Power Plant	WESTINGHOUSE/ABB-CE CONTROL ELEMENT ASSEMBLIES	Figure 3.3-9A Revision 49



Calvert Cliffs Nuclear Power Plant	AREVA CONTROL ELEMENT ASSEMBLIES	Figure 3.3-9B Revision 49

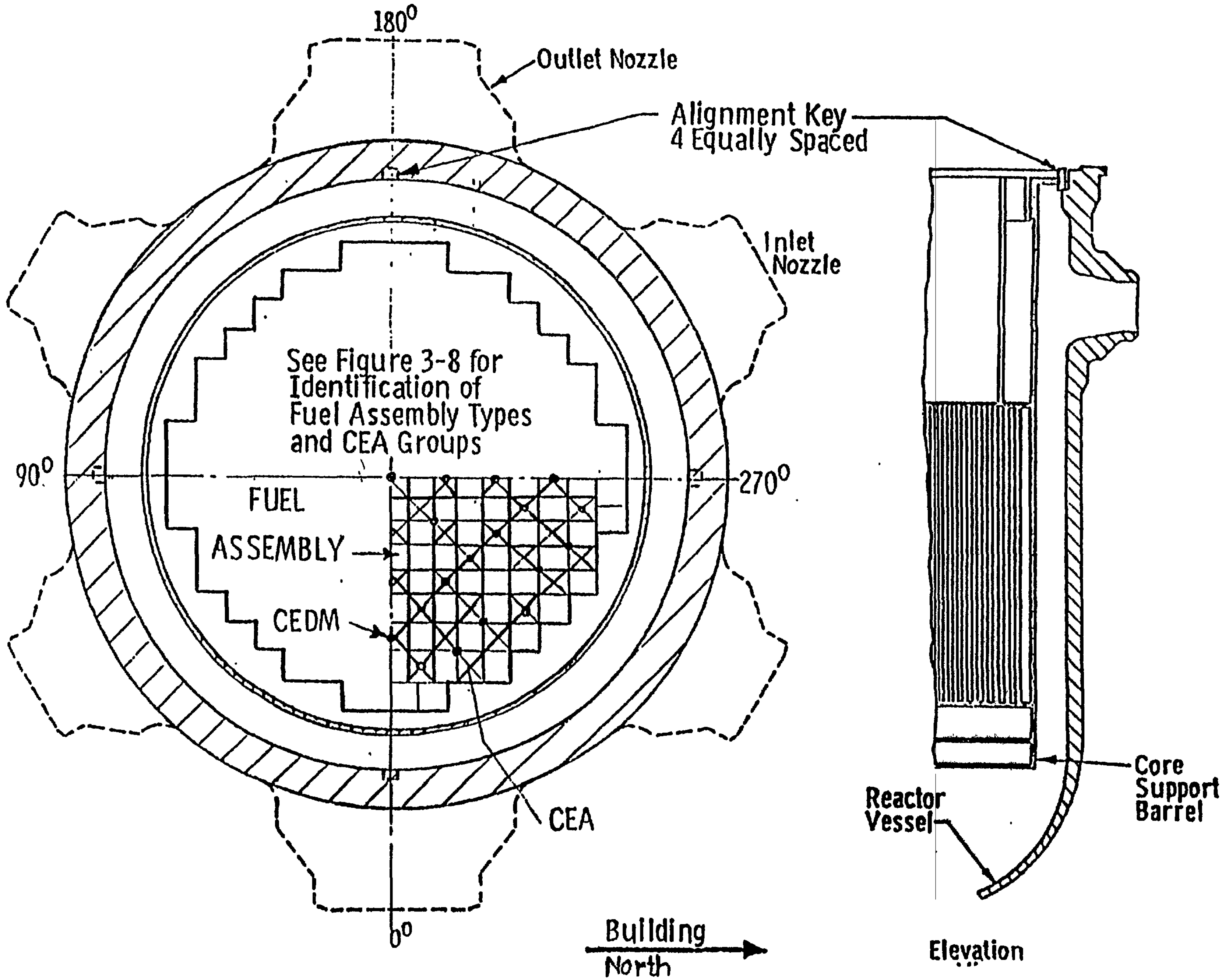


BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

CEA GROUP IDENTIFICATION

REV. 5

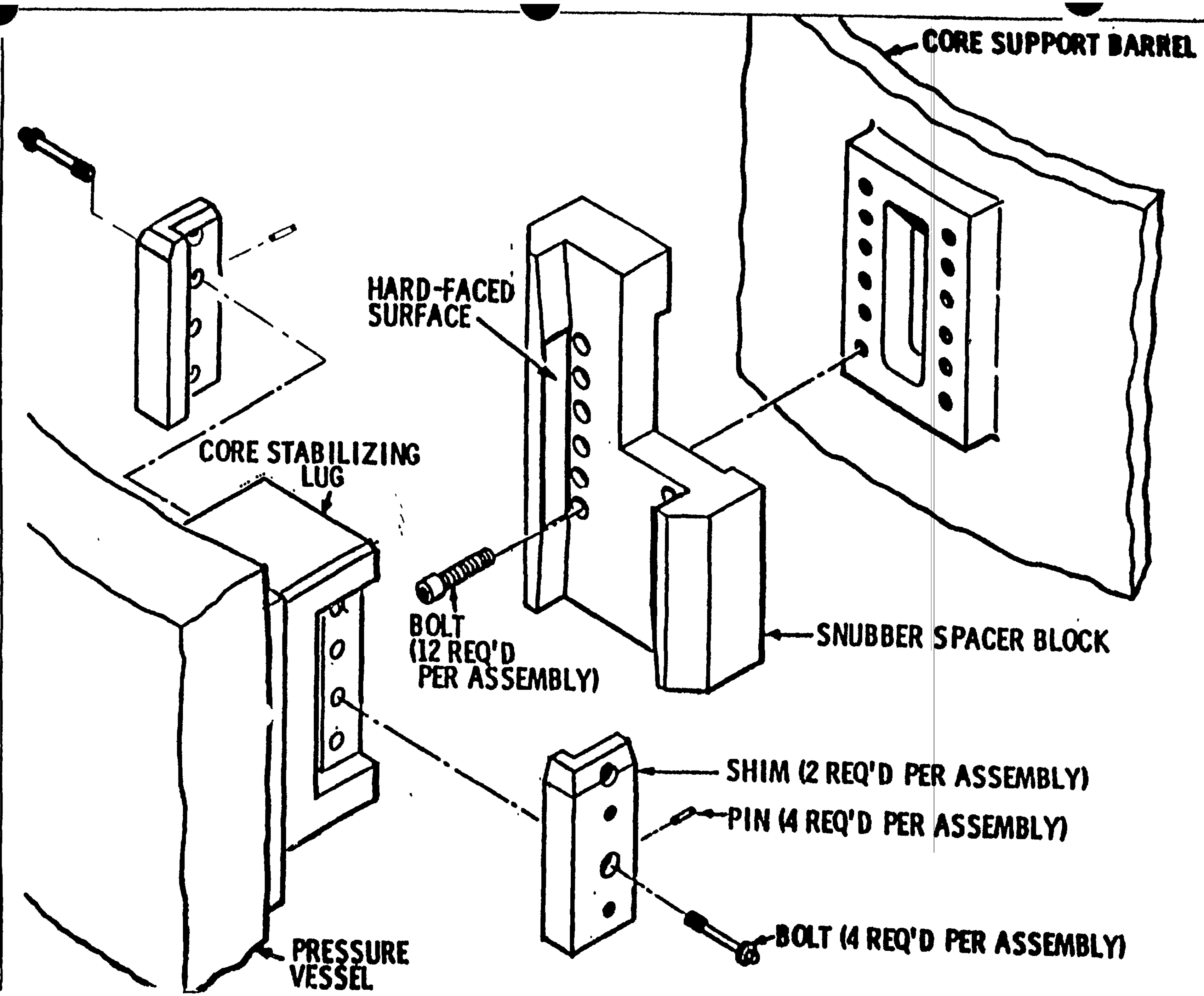
Figure 3.3-18



BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

Core Orientation

Figure
3.3-11

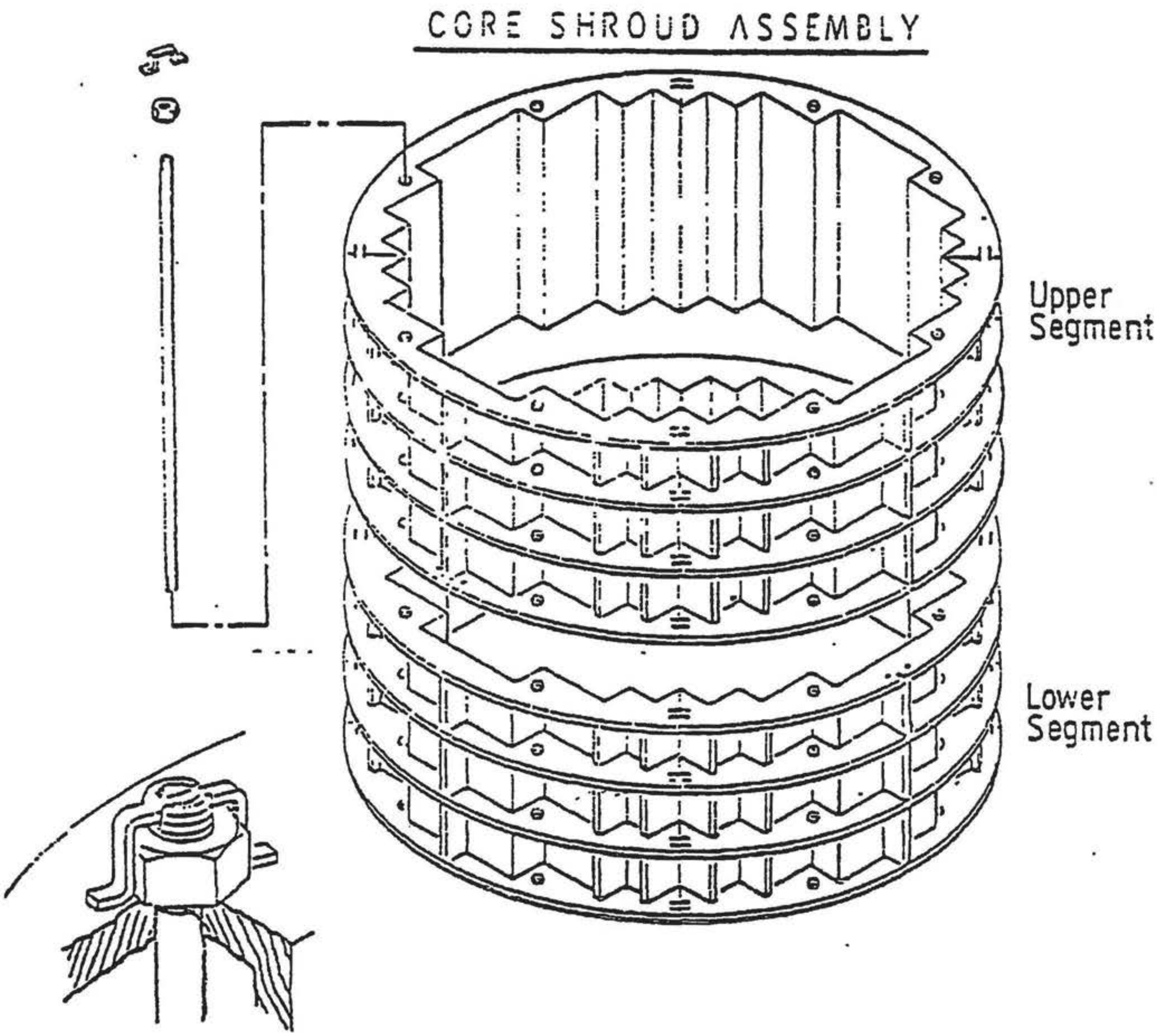


BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

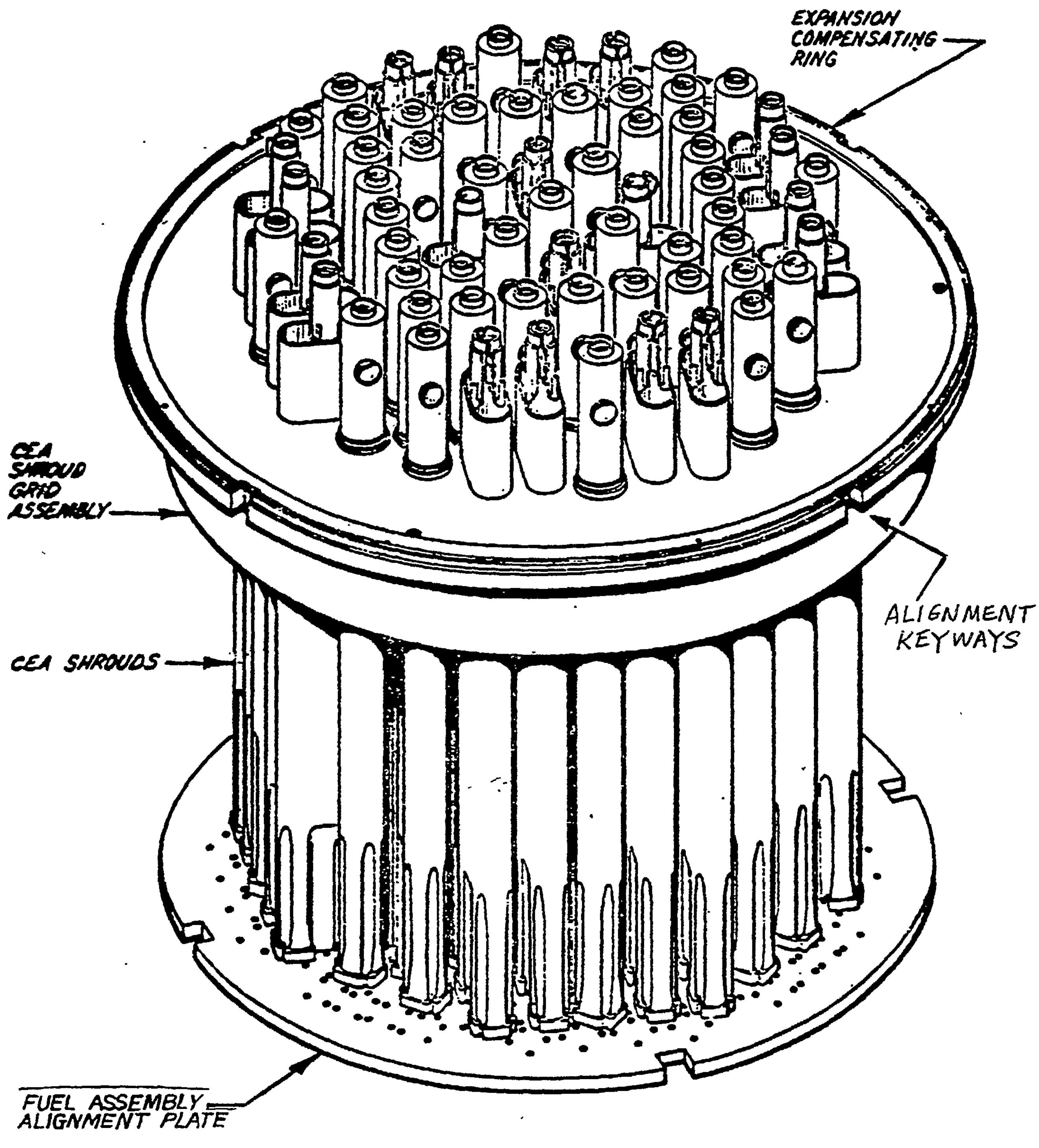
Pressure Vessel - Core Support Barrel Snubber Assembly

REV.11 1/91

Figure 3.3-12

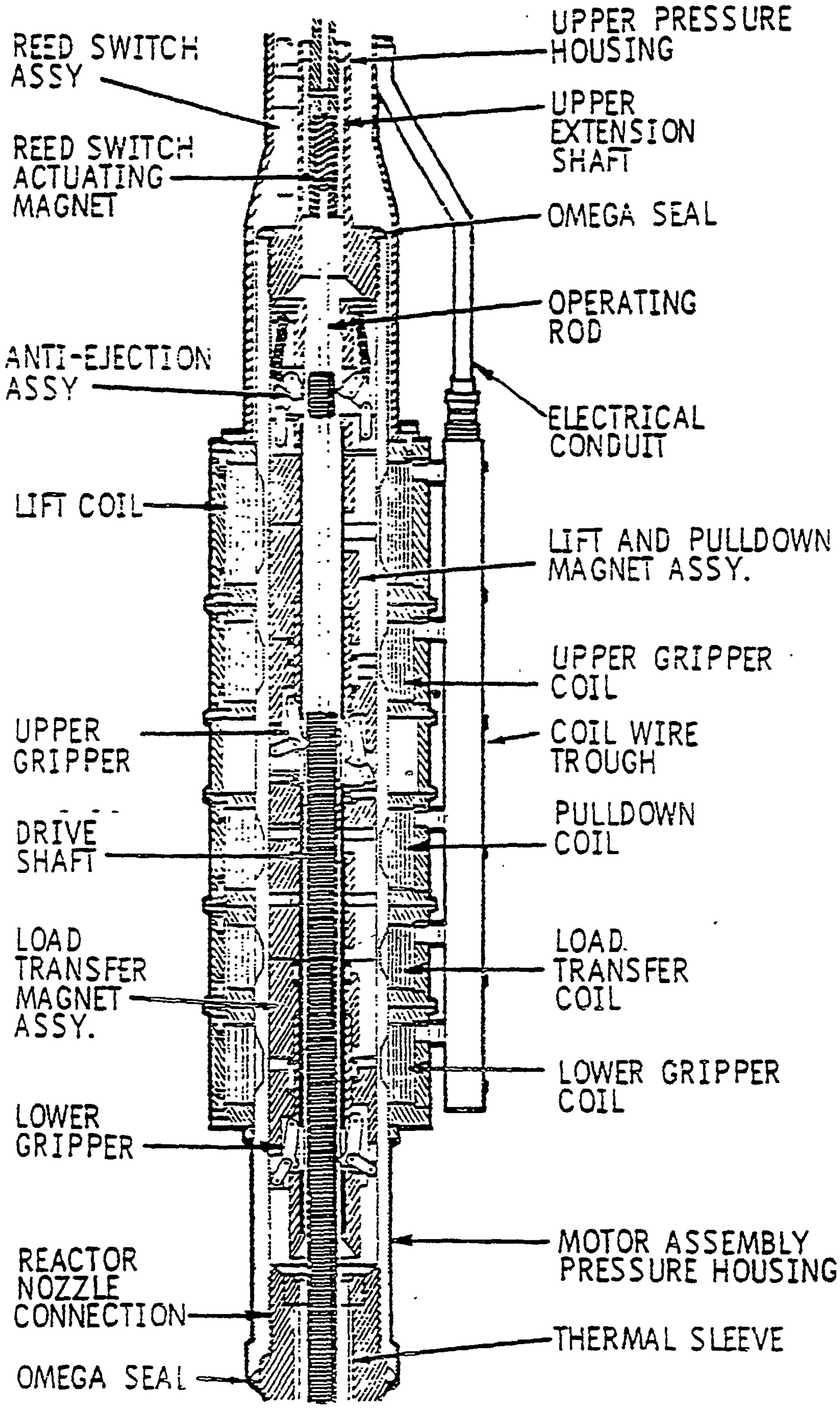


BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CORE SHROUD ASSEMBLY	Figure 3.3-13



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	UPPER GUIDE STRUCTURE ASSEMBLY	Figure 3.3-14

Revision 21



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CONTROL ELEMENT DRIVE MECHANISM (MAGNETIC JACK)	Figure 3.3-15

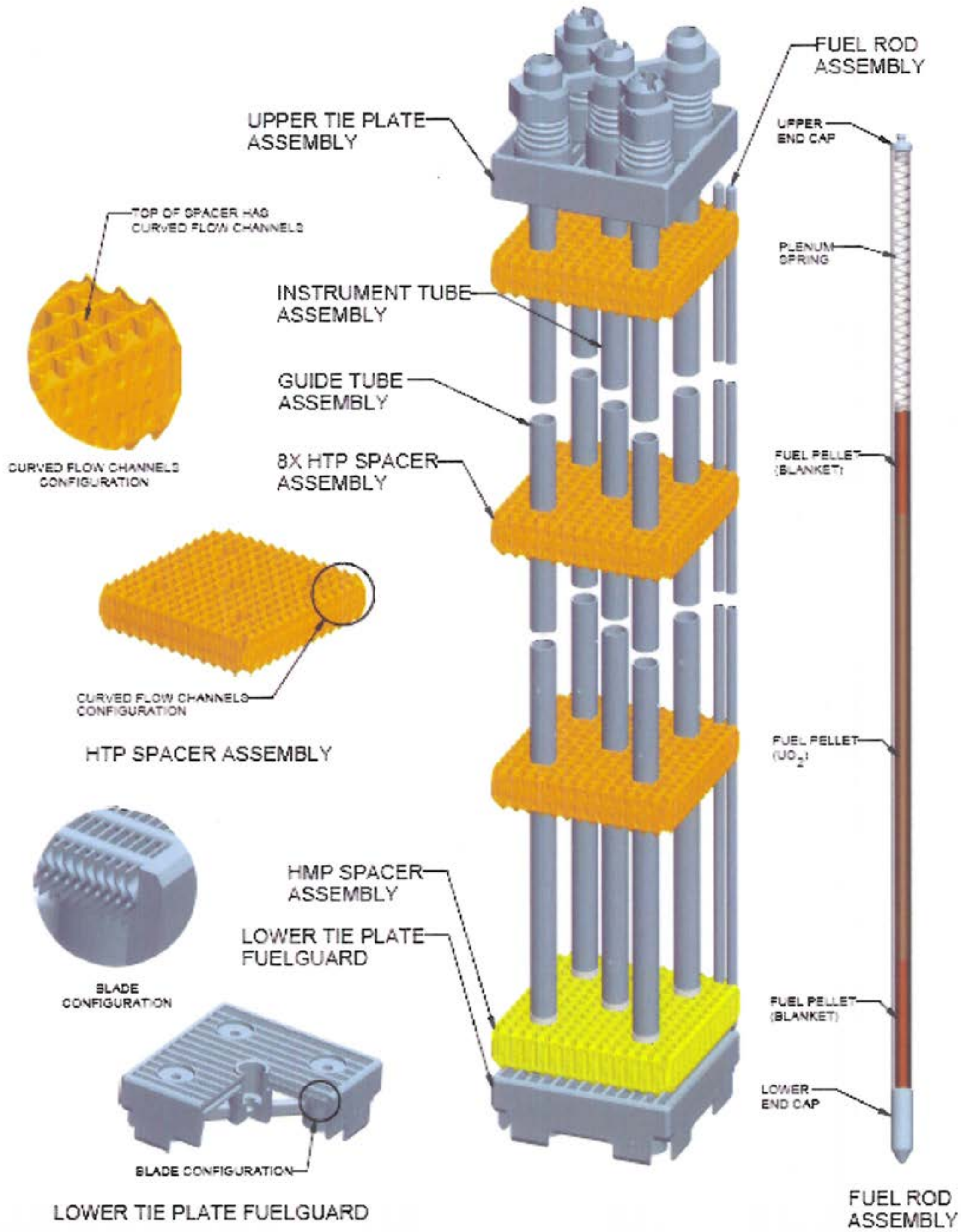


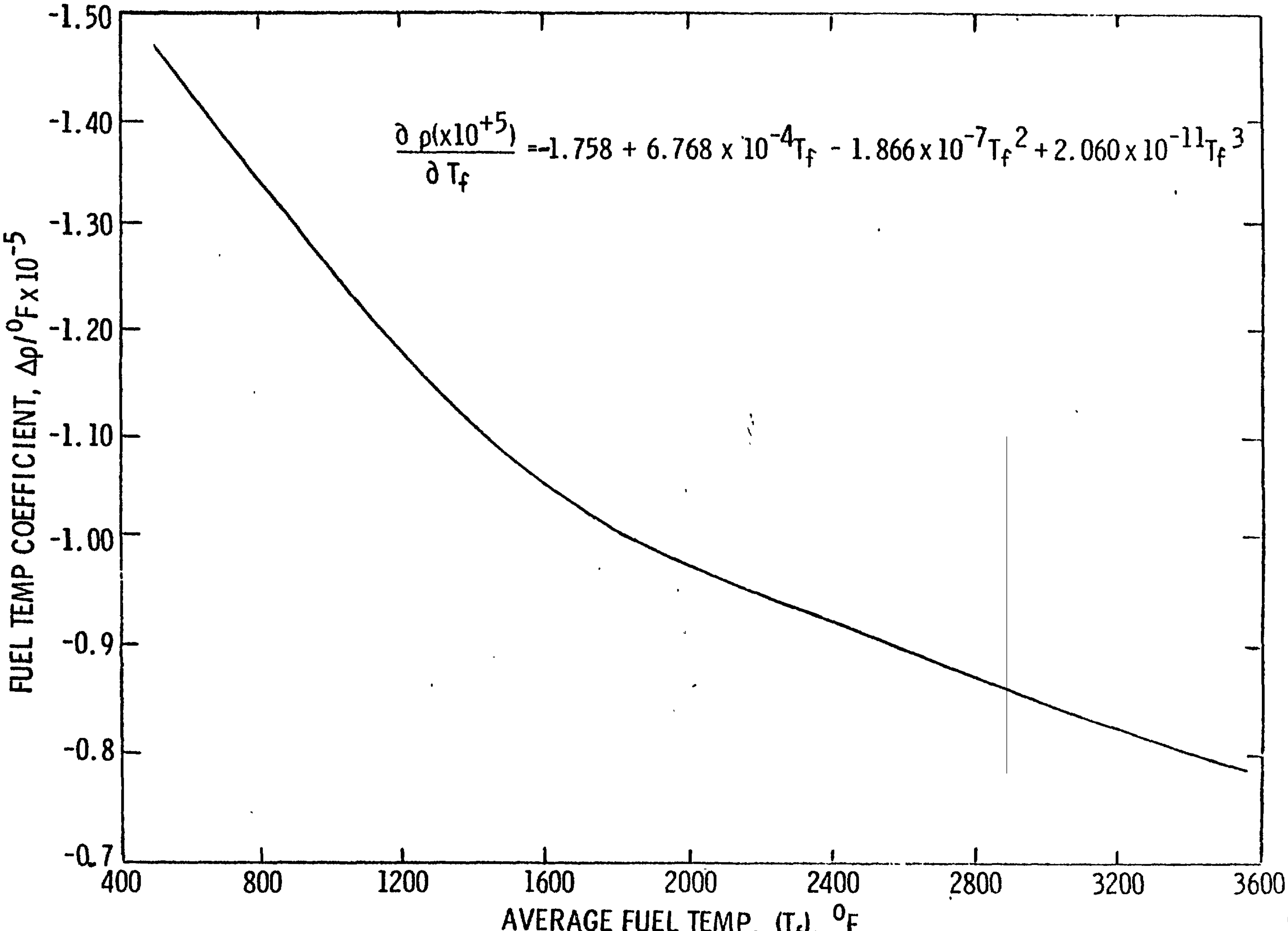
Figure 3.3-16 AREVA HTP Fuel Assembly, Fuel Rod, and Spacer Grids

Calvert Cliffs Nuclear Power Plant	AREVA HTP FUEL ASSEMBLY, FUEL ROD, AND SPACER GRIDS	Figure 3.3-16 Revision 43

BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

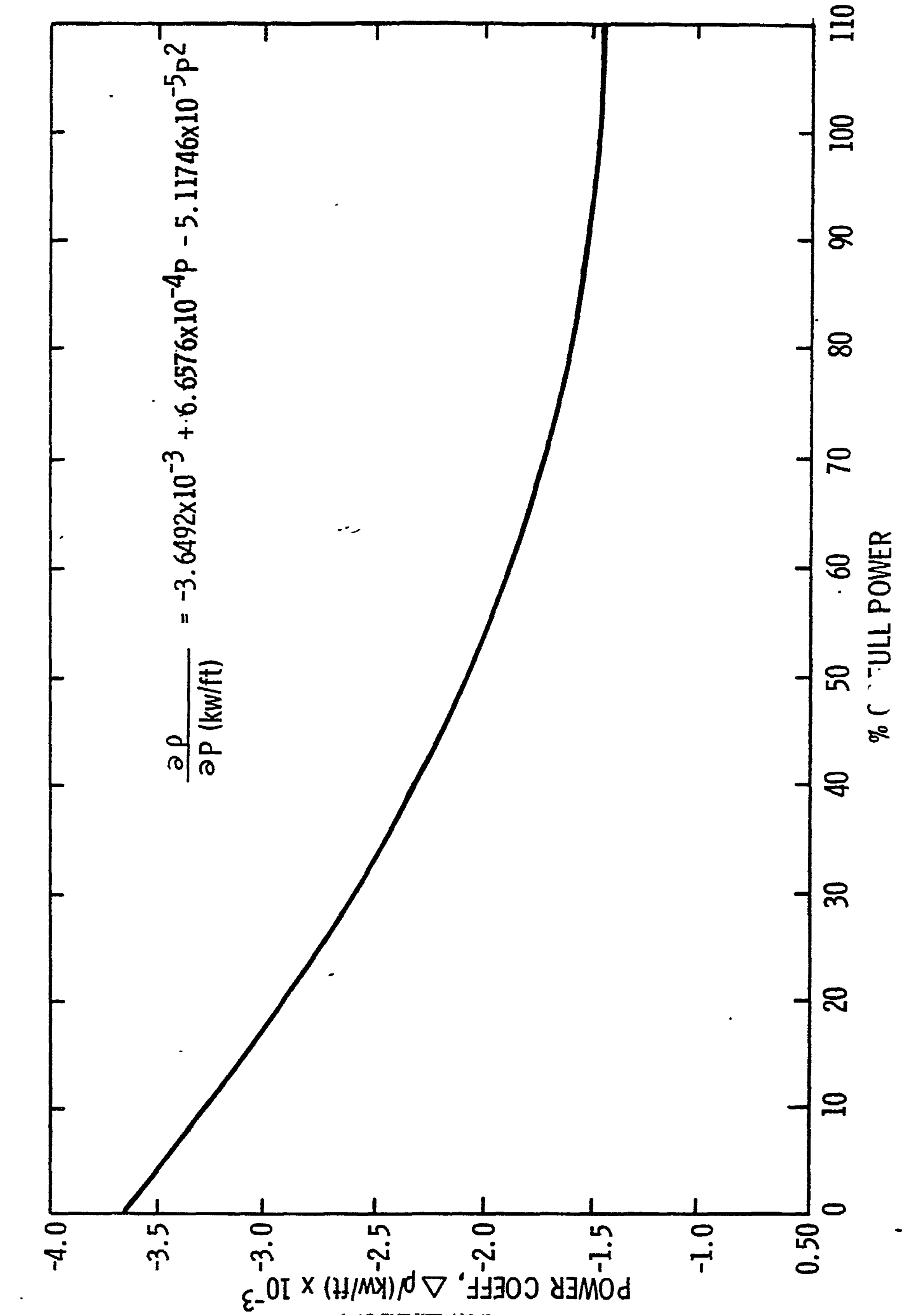
CYCLE 1 FUEL TEMPERATURE COEFFICIENT VS AVERAGE FUEL TEMPERATURE

Figure 3.4-1



3.4-1 CYCLE 1 FUEL TEMPERATURE COEFFICIENT VS AVERAGE FUEL TEMPERATURE

3.4-2 CYCLE 1 POWER COEFFICIENT VS PERCENT OF FULL POWER (BEGINNING OF FIRST CYCLE)



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CYCLE 1 POWER COEFFICIENT VS PERCENT OF FULL POWER (BEGINNING OF FIRST CYCLE)	Figure 3.4-2

Rev.0

BATCH:

A 69 ASSY's AT 0 SHIMS
B+ 80 ASSY's AT 12 SHIMS
C 40 ASSY's AT 0 SHIMS
C• 12 ASSY's AT 12 SHIMS (LO. CONC.)
C+ 16 ASSY's AT 12 SHIMS (HI. CONC.)

	A	B	C	D	E	F	G	H	J	K	L
1								C		C	
2					C	C	C+		C+		B+
3				C	C•	B+	A		B+		A
4			C	C•	B+	A	B+		A		B+
5		C	C•	B+	A	B+	A		B+		A
6		C	B+	A	B+	A	B+		A		B+
7		C+	A	B+	A	B+	A		B+		A
8	C										
9		C+	B+	A	B+	A	B+		A		B+
10	C										
11		B+	A	B+	A	B+	A		B+		A

3.4-3 FIRST CYCLE FUEL ASSEMBLY IDENTIFICATION BOTH UNITS

BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

FIRST CYCLE
FUEL ASSEMBLY IDENTIFICATION
BOTH UNITS

FIGURE
3.4-3
Rev. 0 1/82

X	X - Box Number
Y	Y - Batch

							1 AB3	2 AC1
				3 AB3	4 AC1	5 AC1	6 AD4	7 AD4
			8 AB3	9 AD4	10 AD3	11 AD3	12 AC3	13 AC3
		14 AB3	15 AD4	16 AC3	17 AD1	18 AC1	19 AD1	20 AC3
	21 AB3	22 AD4	23 AC3	24 AD1	25 AC2	26 AD2	27 AC1	28 AD1
	29 AC1	30 AD3	31 AD1	32 AC2	33 AD1	34 AC1	35 AD2	36 AC1
	37 AC1	38 AD3	39 AC1	40 AD2	41 AC1	42 AD1	43 AC2	44 AD1
45 AB3	46 AD4	47 AC3	48 AD1	49 AC1	50 AD2	51 AC2	52 AD2	53 AC1
54 AC1	55 AD4	56 AC3	57 AC3	58 AD1	59 AC1	60 AD1	61 AC1	62 BA5

X Y	X - Box Number Y - Batch							

						1 BA2	2 BB4	
				3 BA1	4 BB5	5 BB5	6 BC2	7 BC3
			8 BA1	9 BC1	10 BC3	11 BC3	12 BB1	13 BB1
		14 BA3	15 BC1	16 BB2	17 BC5	18 BB4	19 BC4	20 BB4
	21 BA1	22 BC1	23 BB2	24 BC5	25 BB3	26 BC4	27 BB5	28 BC5
	29 BB5	30 BC3	31 BC5	32 BB3	33 BC5	34 BB3	35 BC5	36 2Z3
	37 BB5	38 BC3	39 BB4	40 BC4	41 BB3	42 BC5	43 BB4	44 BC5
45 BA2	46 BC2	47 BB1	48 BC4	49 BB5	50 BC5	51 BB4	52 BC4	53 BB3
54 BB4	55 BC3	56 BB1	57 BB4	58 BC5	59 2Z3	60 BC5	61 BB3	62 BA5

Calvert Cliffs Nuclear Power Plant	UNIT 2 QUARTER-CORE ASSEMBLY MAP	Figure 3.4-5 Revision 49

X
0.00 — BOX PEAKING FACTOR
0.00 — ENTHALPY RISE FACTOR
0.00 — PIN PEAKING FACTOR

X — ENRICHMENT TYPE

MAXIMA

BOX PEAKING FACTOR 1.29
ENTHALPY RISE FACTOR 1.36
PIN PEAKING FACTOR 1.39

	H	K
1	C 0.57 0.94 0.96	C 0.75 1.03 1.05

	A
8	C 0.57 0.94 0.96
10	C 0.75 1.03 1.05

	B		D	E	F	G	J	L
2				C 0.54 0.90 0.92	C 0.76 1.05 1.08	C+ 0.91 1.19 1.19	C+ 1.05 1.16 1.19	B+ 0.92 1.04 1.06
3			C 0.63 0.98 1.00	C• 0.94 1.11 1.13	B+ 0.92 1.05 1.07	A 0.99 1.06 1.09	B+ 1.03 1.13 1.15	A 1.02 1.10 1.13
4		C 0.63 0.98 1.00	C• 0.98 1.13 1.16	B+ 0.95 1.07 1.09	A 1.01 1.08 1.12	B+ 1.08 1.18 1.20	A 1.09 1.16 1.20	B+ 1.12 1.20 1.23
5	C 0.54 0.90 0.92	C• 0.94 1.11 1.13	B+ 0.95 1.07 1.09	A 1.02 1.09 1.13	B+ 1.10 1.20 1.22	A 1.12 1.20 1.23	B+ 1.17 1.26 1.28	A 1.16 1.24 1.27
6	C 0.76 1.05 1.08	B+ 0.92 1.05 1.07	A 1.01 1.08 1.12	B+ 1.10 1.20 1.22	A 1.13 1.21 1.25	B+ 1.19 1.29 1.31	A 1.20 1.28 1.32	B+ 1.23 1.31 1.34
7	C+ 0.91 1.19 1.19	A 0.99 1.06 1.09	B+ 1.08 1.18 1.20	A 1.12 1.20 1.23	B+ 1.19 1.29 1.31	A 1.21 1.29 1.33	B+ 1.26 1.34 1.37	A 1.24 1.32 1.36
9	C+ 1.05 1.16 1.19	B+ 1.03 1.13 1.15	A 1.09 1.16 1.20	B+ 1.17 1.26 1.28	A 1.20 1.28 1.32	B+ 1.26 1.34 1.37	A 1.26 1.34 1.38	B+ 1.29 1.36 1.39
11	B+ 0.92 1.04 1.06	A 1.02 1.10 1.13	B+ 1.12 1.20 1.23	A 1.16 1.24 1.27	B+ 1.23 1.31 1.34	A 1.24 1.32 1.36	B+ 1.29 1.36 1.39	A 1.27 1.35 1.39

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BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

CYCLE 1 CORE POWER DISTRIBUTION 2560 MWT (BEGINNING-OF-LIFE) NO XENON

Figure 3.4-6
Rev. 0

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 AB3 0.241	2 AC1 0.437
				3 AB3 0.249	4 AC1 0.475	5 AC1 0.657	6 AD4 1.125	7 AD4 1.239
			8 AB3 0.326	9 AD4 0.946	10 AD3 1.177	11 AD3 1.290	12 AC3 1.199	13 AC3 1.219
		14 AB3 0.325	15 AD4 0.985	16 AC3 1.098	17 AD1 1.275	18 AC1 1.128	19 AD1 1.309 X	20 AC3 1.205
	21 AB3 0.247	22 AD4 0.943	23 AC3 1.098	24 AD1 1.268	25 AC2 1.152	26 AD2 1.287	27 AC1 1.103	28 AD1 1.279
	29 AC1 0.473	30 AD3 1.172	31 AD1 1.272	32 AC2 1.151	33 AD1 1.274	34 AC1 1.091	35 AD2 1.260	36 AC1 1.078
45 AB3 0.237	37 AC1 0.652	38 AD3 1.282	39 AC1 1.124	40 AD2 1.284	41 AC1 1.092	42 AD1 1.246	43 AC2 1.097	44 AD1 1.224
54 AC1 0.436	46 AD4 1.115	47 AC3 1.184	48 AD1 1.302	49 AC1 1.102	50 AD2 1.262	51 AC2 1.099	52 AD2 1.217	53 AC1 1.015
	55 AD4 1.241	56 AC3 1.219	57 AC3 1.207	58 AD1 1.281	59 AC1 1.081	60 AD1 1.225	61 AC1 1.016	62 BA5 0.937

Long N-1 Burnup (21.550 GWd/MTU)

Note: X = Maximum Fr Value = 1.561

Calvert Cliffs Nuclear Power Plant	UNIT 1 ASSEMBLY RELATIVE POWER DENSITY AT BOC, HFP, ARO, EQUILIBRIUM XENON	Figure 3.4-7 Revision 49

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 BA2 0.233	2 BB4 0.413
				3 BA1 0.246	4 BB5 0.463	5 BB5 0.640	6 BC2 1.074	7 BC3 1.119
			8 BA1 0.326	9 BC1 0.927	10 BC3 1.133	11 BC3 1.244	12 BB1 1.175	13 BB1 1.165
		14 BA3 0.304	15 BC1 0.971	16 BB2 1.084	17 BC5 1.258	18 BB4 1.122	19 BC4 1.270	20 BB4 1.095
	21 BA1 0.244	22 BC1 0.917	23 BB2 1.078	24 BC5 1.257	25 BB3 1.165	26 BC4 1.293	27 BB5 1.129	28 BC5 1.285
	29 BB5 0.462	30 BC3 1.131	31 BC5 1.258	32 BB3 1.166	33 BC5 1.292	34 BB3 1.158	35 BC5 1.293	36 2Z3 1.201
45 BA2 0.233	37 BB5 0.643	38 BC3 1.248	39 BB4 1.126	40 BC4 1.295	41 BB3 1.158	42 BC5 1.279	43 BB4 1.125	44 BC5 1.293
54 BB4 0.414	46 BC2 1.082	47 BB1 1.186	48 BC4 1.275	49 BB5 1.130	50 BC5 1.293 X	51 BB4 1.125	52 BC4 1.292	53 BB3 1.189
	55 BC3 1.119	56 BB1 1.165	57 BB4 1.095	58 BC5 1.285	59 2Z3 1.201	60 BC5 1.292	61 BB3 1.188	62 BA5 1.082

Long N-1 Burnup (21.535 GWd/MTU)

Note: X = Maximum Fr Value = 1.541

Calvert Cliffs Nuclear Power Plant	UNIT 2 ASSEMBLY RELATIVE POWER DENSITY AT BOC, HFP, ARO, EQUILIBRIUM XENON	Figure 3.4-8 Revision 49

BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

CYCLE 1 CORE POWER DISTRIBUTION
2560 MWT
1000 MWD/MTU, EQUILIBRIUM XENON

FIGURE
3.4-9

Rev. 0 1/82

Key (each box):

- X — ENRICHMENT TYPE
- 0.00 — BOX PEAKING FACTOR
- 0.00 — ENTHALPY RISE FACTOR
- 0.00 — PIN PEAKING FACTOR

MAXIMA

BOX PEAKING FACTOR	1.31
ENTHALPY RISE FACTOR	1.38
PIN PEAKING FACTOR	1.41

Row	A	B	C	D	E	F	G	J	L	H	K
1										C 0.54 0.90 0.92	C 0.71 0.98 1.00
2					C 0.53 0.87 0.89	C 0.74 1.02 1.05	C+ 0.89 1.17 1.18	C+ 1.03 1.15 1.18	B+ 0.91 1.03 1.05		
3				C 0.61 0.96 0.98	C● 0.93 1.10 1.13	B+ 0.91 1.06 1.07	A 0.98 1.06 1.10	B+ 1.03 1.14 1.16	A 1.02 1.10 1.13		
4			C 0.61 0.96 0.98	C● 0.96 1.13 1.16	B+ 0.95 1.08 1.10	A 1.02 1.09 1.13	B+ 1.09 1.19 1.21	A 1.10 1.17 1.21	B+ 1.13 1.22 1.24		
5		C 0.53 0.87 0.89	C● 0.93 1.10 1.13	B+ 0.95 1.08 1.10	A 1.03 1.10 1.14	B+ 1.11 1.22 1.24	A 1.14 1.21 1.25	B+ 1.19 1.28 1.30	A 1.17 1.25 1.29		
6		C 0.74 1.02 1.05	B+ 0.91 1.06 1.07	A 1.02 1.09 1.13	B+ 1.11 1.22 1.24	A 1.15 1.22 1.26	B+ 1.21 1.31 1.33	A 1.21 1.29 1.33	B+ 1.25 1.33 1.36		
7		C+ 0.89 1.17 1.18	A 0.98 1.06 1.10	B+ 1.09 1.19 1.21	A 1.14 1.21 1.25	B+ 1.21 1.31 1.33	A 1.23 1.30 1.34	B+ 1.28 1.36 1.39	A 1.26 1.33 1.37		
8	C 0.54 0.90 0.92										
9		C+ 1.03 1.15 1.18	B+ 1.03 1.14 1.16	A 1.10 1.17 1.21	B+ 1.19 1.28 1.30	A 1.21 1.29 1.33	B+ 1.28 1.36 1.39	A 1.27 1.34 1.39	B+ 1.31 1.38 1.41		
10	C 0.71 0.98 1.00										
11		B+ 0.91 1.03 1.05	A 1.02 1.10 1.13	B+ 1.13 1.22 1.24	A 1.17 1.25 1.29	B+ 1.25 1.33 1.36	A 1.26 1.33 1.37	B+ 1.31 1.38 1.41	A 1.29 1.36 1.40		

CL

3.4-9 CYCLE 1 CORE POWER DISTRIBUTION, 2560 MWT, 1000 MWD/MTU, EQUILIBRIUM XENON

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 AB3 0.230	2 AC1 0.395
				3 AB3 0.273	4 AC1 0.496	5 AC1 0.624	6 AD4 0.997	7 AD4 1.065
			8 AB3 0.340	9 AD4 0.976	10 AD3 1.256	11 AD3 1.292	12 AC3 1.023	13 AC3 0.999
		14 AB3 0.339	15 AD4 0.978	16 AC3 1.049	17 AD1 1.351	18 AC1 1.075	19 AD1 1.293	20 AC3 1.062
	21 AB3 0.271	22 AD4 0.975	23 AC3 1.050	24 AD1 1.320	25 AC2 1.120	26 AD2 1.401	27 AC1 1.072	28 AD1 1.330
	29 AC1 0.496	30 AD3 1.255	31 AD1 1.350	32 AC2 1.119	33 AD1 1.364	34 AC1 1.095	35 AD2 1.404	36 AC1 1.085
45 AB3 0.227	37 AC1 0.622	38 AD3 1.289	39 AC1 1.074	40 AD2 1.400	41 AC1 1.095	42 AD1 1.364	43 AC2 1.113	44 AD1 1.346
54 AC1 0.394	46 AD4 0.992	47 AC3 1.016	48 AD1 1.289	49 AC1 1.072	50 AD2 1.405 X	51 AC2 1.115	52 AD2 1.362	53 AC1 1.024
	55 AD4 1.065	56 AC3 0.998	57 AC3 1.063	58 AD1 1.330	59 AC1 1.086	60 AD1 1.347	61 AC1 1.025	62 BA5 0.930

Long N-1 Burnup (21.550 GWd/MTU)

Note: X = Maximum Fr Value = 1.512

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 BA2 0.239	2 BB4 0.410
				3 BA1 0.282	4 BB5 0.500	5 BB5 0.634	6 BC2 1.022	7 BC3 1.075
			8 BA1 0.358	9 BC1 1.005	10 BC3 1.236	11 BC3 1.276	12 BB1 1.047	13 BB1 1.009
		14 BA3 0.336	15 BC1 1.016	16 BB2 1.070	17 BC5 1.359 X	18 BB4 1.078	19 BC4 1.280	20 BB4 0.997
	21 BA1 0.279	22 BC1 0.997	23 BB2 1.066	24 BC5 1.336	25 BB3 1.131	26 BC4 1.353	27 BB5 1.072	28 BC5 1.319
	29 BB5 0.499	30 BC3 1.234	31 BC5 1.358	32 BB3 1.131	33 BC5 1.361	34 BB3 1.112	35 BC5 1.345	36 2Z3 1.143
45 BA2 0.239	37 BB5 0.635	38 BC3 1.277	39 BB4 1.080	40 BC4 1.354	41 BB3 1.112	42 BC5 1.333	43 BB4 1.067	44 BC5 1.325
54 BB4 0.410	46 BC2 1.026	47 BB1 1.053	48 BC4 1.282	49 BB5 1.073	50 BC5 1.345	51 BB4 1.067	52 BC4 1.297	53 BB3 1.072
	55 BC3 1.075	56 BB1 1.009	57 BB4 0.997	58 BC5 1.319	59 2Z3 1.143	60 BC5 1.325	61 BB3 1.071	62 BA5 0.950

Long N-1 Burnup (21.535 GWd/MTU)

Note: X = Maximum Fr Value = 1.512

X — ENRICHMENT TYPE
0.00 — BOX PEAKING FACTOR
0.00 — ENTHALPY RISE FACTOR
0.00 — PIN PEAKING FACTOR

MAXIMA

BOX PEAKING FACTOR 1.17
ENTHALPY RISE FACTOR 1.29
PIN PEAKING FACTOR 1.32

	H	K	
	C 0.57 0.94 0.95	C 0.73 1.02 1.03	1

	A	B		D	E	F	G	J		
					C 0.55 0.91 0.92	C 0.76 1.05 1.06	C+ 0.97 1.22 1.26	C+ 1.14 1.29 1.33	B+ 1.05 1.17 1.20	2
				C 0.65 1.02 1.03	C• 1.01 1.21 1.25	B+ 1.02 1.14 1.17	A 1.01 1.08 1.10	B+ 1.14 1.20 1.24	A 1.06 1.10 1.13	3
			C 0.65 1.02 1.03	C• 1.05 1.24 1.29	B+ 1.08 1.17 1.21	A 1.03 1.09 1.11	B+ 1.14 1.20 1.24	A 1.07 1.10 1.13	B+ 1.17 1.21 1.25	4
		C 0.55 0.91 0.92	C• 1.01 1.21 1.25	B+ 1.08 1.17 1.21	A 1.05 1.09 1.12	B+ 1.15 1.20 1.24	A 1.06 1.10 1.12	B+ 1.16 1.21 1.25	A 1.07 1.10 1.13	5
		C 0.76 1.05 1.06	B+ 1.02 1.14 1.17	A 1.03 1.09 1.11	B+ 1.15 1.20 1.24	A 1.06 1.10 1.12	B+ 1.16 1.20 1.24	A 1.06 1.10 1.12	B+ 1.16 1.20 1.24	6
8	C 0.57 0.94 0.95	C+ 0.97 1.22 1.26	A 1.01 1.08 1.10	B+ 1.14 1.20 1.24	A 1.06 1.10 1.12	B+ 1.16 1.20 1.24	A 1.06 1.09 1.12	B+ 1.15 1.20 1.23	A 1.05 1.09 1.11	7
10	C 0.73 1.02 1.03	C+ 1.14 1.29 1.33	B+ 1.14 1.20 1.24	A 1.07 1.10 1.13	B+ 1.16 1.21 1.25	A 1.06 1.10 1.12	B+ 1.15 1.20 1.23	A 1.04 1.08 1.10	B+ 1.14 1.18 1.22	9
℄		B+ 1.05 1.17 1.20	A 1.06 1.10 1.13	B+ 1.17 1.21 1.25	A 1.07 1.10 1.13	B+ 1.16 1.20 1.24	A 1.05 1.09 1.11	B+ 1.14 1.18 1.22	A 1.04 1.07 1.10	11

BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

CYCLE 1 CORE POWER DISTRIBUTION 2560 MWT
END-OF-CYCLE, EQUILIBRIUM XENON

Figure 3.4-12
Rev. 0

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 AB3 0.321	2 AC1 0.523
				3 AB3 0.350	4 AC1 0.577	5 AC1 0.709	6 AD4 1.104	7 AD4 1.167
			8 AB3 0.428	9 AD4 1.046	10 AD3 1.217	11 AD3 1.244	12 AC3 1.044	13 AC3 1.041
		14 AB3 0.428	15 AD4 1.063	16 AC3 1.057	17 AD1 1.312	18 AC1 1.036	19 AD1 1.287	20 AC3 1.059
	21 AB3 0.348	22 AD4 1.046	23 AC3 1.058	24 AD1 1.305	25 AC2 1.068	26 AD2 1.291	27 AC1 1.024	28 AD1 1.292
	29 AC1 0.577	30 AD3 1.218	31 AD1 1.313 X	32 AC2 1.068	33 AD1 1.297	34 AC1 1.026	35 AD2 1.284	36 AC1 1.022
45 AB3 0.318	37 AC1 0.709	38 AD3 1.244	39 AC1 1.036	40 AD2 1.292	41 AC1 1.026	42 AD1 1.286	43 AC2 1.039	44 AD1 1.280
54 AC1 0.523	46 AD4 1.102	47 AC3 1.041	48 AD1 1.286	49 AC1 1.025	50 AD2 1.284	51 AC2 1.040	52 AD2 1.263	53 AC1 0.980
	55 AD4 1.167	56 AC3 1.040	57 AC3 1.059	58 AD1 1.292	59 AC1 1.022	60 AD1 1.279	61 AC1 0.981	62 BA5 0.916

Long N-1 Burnup (21.550 GWd/MTU)

Note: X = Maximum Fr Value = 1.389

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 BA2 0.324	2 BB4 0.526
				3 BA1 0.349	4 BB5 0.575	5 BB5 0.711	6 BC2 1.100	7 BC3 1.166
			8 BA1 0.425	9 BC1 1.030	10 BC3 1.210	11 BC3 1.241	12 BB1 1.055	13 BB1 1.035
		14 BA3 0.402	15 BC1 1.037	16 BB2 1.043	17 BC5 1.307 X	18 BB4 1.040	19 BC4 1.275	20 BB4 1.001
	21 BA1 0.347	22 BC1 1.025	23 BB2 1.041	24 BC5 1.293	25 BB3 1.073	26 BC4 1.296	27 BB5 1.037	28 BC5 1.289
	29 BB5 0.574	30 BC3 1.209	31 BC5 1.307 X	32 BB3 1.072	33 BC5 1.296	34 BB3 1.056	35 BC5 1.291	36 2Z3 1.085
45 BA2 0.323	37 BB5 0.711	38 BC3 1.241	39 BB4 1.041	40 BC4 1.296	41 BB3 1.056	42 BC5 1.284	43 BB4 1.027	44 BC5 1.281
54 BB4 0.526	46 BC2 1.101	47 BB1 1.057	48 BC4 1.276	49 BB5 1.037	50 BC5 1.291	51 BB4 1.027	52 BC4 1.264	53 BB3 1.036
	55 BC3 1.166	56 BB1 1.035	57 BB4 1.001	58 BC5 1.289	59 2Z3 1.085	60 BC5 1.281	61 BB3 1.036	62 BA5 0.940

Long N-1 Burnup (21.535 GWd/MTU)

Note: X = Maximum Fr Value = 1.381

Calvert Cliffs Nuclear Power Plant	UNIT 2 ASSEMBLY RELATIVE POWER DENSITY AT EOC, HFP, ARO, EQUILIBRIUM XENON	Figure 3.4-14 Revision 49

BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

CORE POWER DISTRIBUTION - CEA GROUP 5
BEGINNING OF FIRST CYCLE
NO XENON

FIGURE
3.4-15

Rev. 0 1/82

X — ENRICHMENT TYPE
0.00 — BOX PEAKING FACTOR
0.00 — ENTHALPY RISE FACTOR
0.00 — PIN PEAKING FACTOR

MAXIMA

BOX PEAKING FACTOR	1.30
ENTHALPY RISE FACTOR	1.39
PIN PEAKING FACTOR	1.42

Row	H	K
1	C 0.56 0.93 0.95	C 0.73 0.99 1.01

Row	A	B	C	D	E	F	G	J	I
2					C 0.55 0.92 0.94	C 0.79 1.09 1.12	C+ 0.92 1.19 1.20	C+ 1.01 1.12 1.15	B+ 0.84 0.90 0.94
3				C 0.57 0.89 0.91	C• 0.94 1.12 1.15	B+ 0.95 1.10 1.12	A 1.01 1.09 1.12	B+ 0.95 1.09 1.11	A 0.70 0.75 0.76
4			C 0.57 0.89 0.91	C• 0.73 0.92 1.00	B+ 0.93 1.11 1.12	A 1.06 1.15 1.19	B+ 1.13 1.24 1.26	A 1.07 1.17 1.21	B+ 1.03 1.20 1.22
5		C 0.55 0.92 0.94	C• 0.94 1.12 1.15	B+ 0.93 1.11 1.12	A 1.06 1.16 1.20	B+ 1.18 1.31 1.33	A 1.21 1.30 1.34	B+ 1.23 1.34 1.37	A 1.20 1.30 1.33
6		C 0.79 1.09 1.12	B+ 0.95 1.10 1.12	A 1.06 1.15 1.19	B+ 1.18 1.31 1.33	A 1.24 1.33 1.37	B+ 1.30 1.39 1.41	A 1.28 1.36 1.41	B+ 1.30 1.37 1.41
7		C+ 0.92 1.19 1.20	A 1.01 1.09 1.12	B+ 1.13 1.24 1.26	A 1.21 1.30 1.34	B+ 1.30 1.39 1.41	A 1.29 1.37 1.42	B+ 1.30 1.38 1.41	A 1.25 1.34 1.38
8	C 0.56 0.93 0.95								
9		C+ 1.01 1.12 1.15	B+ 0.95 1.09 1.11	A 1.07 1.17 1.21	B+ 1.23 1.34 1.37	A 1.28 1.36 1.41	B+ 1.29 1.38 1.41	A 1.17 1.27 1.31	B+ 1.08 1.26 1.28
10	C 0.73 0.99 1.01								
11		B+ 0.84 0.90 0.94	A 0.70 0.75 0.76	B+ 1.03 1.20 1.22	A 1.20 1.30 1.33	B+ 1.30 1.37 1.41	A 1.25 1.34 1.38	B+ 1.08 1.26 1.28	A 0.63 0.74 0.78

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3.4-15 CORE POWER DISTRIBUTION - CEA GROUP 5 BEGINNING OF FIRST CYCLE, NO XENON

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 AB3 0.231	2 AC1 0.413
				3 AB3 0.254	4 AC1 0.478	5 AC1 0.644	6 AD4 1.069	7 AD4 1.152
			8 AB3 0.335	9 AD4 0.967	10 AD3 1.190	11 AD3 1.278	12 AC3 1.139	13 AC3 1.054
		14 AB3 0.335	15 AD4 1.013	16 AC3 1.125	17 AD1 1.297	18 AC1 1.131	19 AD1 1.282	20 AC3 1.156
	21 AB3 0.251	22 AD4 0.964	23 AC3 1.125	24 AD1 1.300	25 AC2 1.178	26 AD2 1.307	27 AC1 1.108	28 AD1 1.280
	29 AC1 0.476	30 AD3 1.186	31 AD1 1.294	32 AC2 1.177	33 AD1 1.306 X	34 AC1 1.117	35 AD2 1.286	36 AC1 1.099
45 AB3 0.227	37 AC1 0.639	38 AD3 1.270	39 AC1 1.127	40 AD2 1.305	41 AC1 1.117	42 AD1 1.279	43 AC2 1.125	44 AD1 1.254
54 AC1 0.412	46 AD4 1.060	47 AC3 1.125	48 AD1 1.276	49 AC1 1.108	50 AD2 1.288	51 AC2 1.127	52 AD2 1.247	53 AC1 1.036
	55 AD4 1.154	56 AC3 1.055	57 AC3 1.159	58 AD1 1.282	59 AC1 1.102	60 AD1 1.256	61 AC1 1.038	62 BA5 0.916

Long N-1 Burnup (21.550 GWd/MTU)

Note: X = Maximum Fr Value = 1.554

Calvert Cliffs Nuclear Power Plant	UNIT 1 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED TO PDIL AT BOC, HFP, EQUILIBRIUM XENON	Figure 3.4-16 Revision 49

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 BA2 0.223	2 BB4 0.390
				3 BA1 0.251	4 BB5 0.465	5 BB5 0.627	6 BC2 1.021	7 BC3 1.039
			8 BA1 0.335	9 BC1 0.947	10 BC3 1.145	11 BC3 1.233	12 BB1 1.117	13 BB1 1.007
		14 BA3 0.312	15 BC1 0.997	16 BB2 1.109	17 BC5 1.279	18 BB4 1.125	19 BC4 1.247	20 BB4 1.053
	21 BA1 0.248	22 BC1 0.937	23 BB2 1.104	24 BC5 1.288	25 BB3 1.190	26 BC4 1.313	27 BB5 1.137	28 BC5 1.288
	29 BB5 0.464	30 BC3 1.143	31 BC5 1.278	32 BB3 1.190	33 BC5 1.322 X	34 BB3 1.184	35 BC5 1.319	36 2Z3 1.223
45 BA2 0.223	37 BB5 0.629	38 BC3 1.235	39 BB4 1.128	40 BC4 1.315	41 BB3 1.184	42 BC5 1.310	43 BB4 1.151	44 BC5 1.322
54 BB4 0.391	46 BC2 1.028	47 BB1 1.127	48 BC4 1.251	49 BB5 1.138	50 BC5 1.319	51 BB4 1.151	52 BC4 1.319	53 BB3 1.210
	55 BC3 1.039	56 BB1 1.007	57 BB4 1.052	58 BC5 1.288	59 2Z3 1.223	60 BC5 1.322	61 BB3 1.209	62 BA5 1.054

Long N-1 Burnup (21.535 GWd/MTU)

Note: X = Maximum Fr Value = 1.574

Calvert Cliffs Nuclear Power Plant	UNIT 2 ASSEMBLY RELATIVE POWER DENSITY AT BANK 5 INSERTED TO PDIL AT BOC, HFP, EQUILIBRIUM XENON	Figure 3.4-17 Revision 49

BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

CORE POWER DISTRIBUTION - CEA GROUP 5
END-OF-CYCLE 1, EQUILIBRIUM

Figure 3.4-18
Rev. 5

X	
0.00	BOX PEAKING FACTOR
0.00	ENTHALPY RISE FACTOR
0.00	PIN PEAKING FACTOR

X = ENRICHMENT TYPE

MAXIMA

BOX PEAKING FACTOR	1.27
ENTHALPY RISE FACTOR	1.33
PIN PEAKING FACTOR	1.37

	H	K
1	C 0.57 0.92 0.93	C 0.71 0.97 0.98

	B		D	E	F	G	J	
2				C 0.56 0.93 0.94	C 0.78 1.08 1.10	C+ 0.97 1.22 1.26	C+ 1.09 1.22 1.26	B+ 0.96 1.03 1.06
3			C 0.58 0.91 0.92	C• 1.00 1.21 1.26	B+ 1.05 1.19 1.22	A 1.03 1.10 1.12	B+ 1.05 1.17 1.20	A 0.73 0.85 0.87
4		C 0.58 0.91 0.92	C• 0.77 0.98 1.10	B+ 1.05 1.20 1.24	A 1.08 1.16 1.18	B+ 1.19 1.27 1.31	A 1.05 1.14 1.16	B+ 1.07 1.20 1.24
5	C 0.56 0.93 0.94	C• 1.00 1.21 1.26	B+ 1.05 1.20 1.24	A 1.08 1.17 1.19	B+ 1.24 1.31 1.36	A 1.15 1.20 1.23	B+ 1.23 1.30 1.34	A 1.11 1.17 1.19
6	C 0.78 1.08 1.10	B+ 1.05 1.19 1.22	A 1.08 1.16 1.18	B+ 1.24 1.31 1.36	A 1.17 1.21 1.24	B+ 1.27 1.33 1.37	A 1.14 1.19 1.22	B+ 1.24 1.29 1.33
7	C+ 0.97 1.22 1.26	A 1.03 1.10 1.12	B+ 1.19 1.27 1.31	A 1.15 1.20 1.23	B+ 1.27 1.33 1.37	A 1.14 1.20 1.23	B+ 1.20 1.29 1.33	A 1.08 1.15 1.17
9	C+ 1.09 1.22 1.26	B+ 1.05 1.17 1.20	A 1.05 1.14 1.16	B+ 1.23 1.30 1.34	A 1.14 1.19 1.22	B+ 1.20 1.29 1.33	A 1.00 1.10 1.13	B+ 0.99 1.14 1.17
11	B+ 0.96 1.03 1.06	A 0.73 0.85 0.87	B+ 1.07 1.20 1.24	A 1.11 1.17 1.19	B+ 1.24 1.29 1.33	A 1.08 1.15 1.17	B+ 0.99 1.14 1.17	A 0.55 0.68 0.74

	A
8	C 0.57 0.92 0.93
10	C 0.71 0.97 0.98

CL

3.4-18 CORE POWER DISTRIBUTION - CEA GROUP 5 END-OF-CYCLE 1, EQUILIBRIUM

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 AB3 0.309	2 AC1 0.498
				3 AB3 0.356	4 AC1 0.581	5 AC1 0.697	6 AD4 1.052	7 AD4 1.086
			8 AB3 0.440	9 AD4 1.068	10 AD3 1.229	11 AD3 1.233	12 AC3 0.993	13 AC3 0.892
		14 AB3 0.439	15 AD4 1.090	16 AC3 1.080	17 AD1 1.332	18 AC1 1.039	19 AD1 1.262	20 AC3 1.018
	21 AB3 0.355	22 AD4 1.068	23 AC3 1.081	24 AD1 1.335 X	25 AC2 1.089	26 AD2 1.310	27 AC1 1.030	28 AD1 1.294
	29 AC1 0.582	30 AD3 1.230	31 AD1 1.333	32 AC2 1.090	33 AD1 1.327	34 AC1 1.048	35 AD2 1.308	36 AC1 1.040
45 AB3 0.307	37 AC1 0.697	38 AD3 1.234	39 AC1 1.039	40 AD2 1.311	41 AC1 1.049	42 AD1 1.317	43 AC2 1.063	44 AD1 1.308
54 AC1 0.497	46 AD4 1.051	47 AC3 0.990	48 AD1 1.262	49 AC1 1.030	50 AD2 1.308	51 AC2 1.064	52 AD2 1.289	53 AC1 0.998
	55 AD4 1.086	56 AC3 0.891	57 AC3 1.018	58 AD1 1.294	59 AC1 1.040	60 AD1 1.308	61 AC1 0.998	62 BA5 0.889

Long N-1 Burnup (21.550 GWd/MTU)

Note: X = Maximum Fr Value = 1.412

Calvert Cliffs Nuclear Power Plant	UNIT 1 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED TO PDIL AT EOC, HFP, EQUILIBRIUM XENON	Figure 3.4-19 Revision 49

X Y XXX	X - Box Number Y - Batch XXX - Assembly Relative Power Density

							1 BA2 0.312	2 BB4 0.501
				3 BA1 0.356	4 BB5 0.579	5 BB5 0.699	6 BC2 1.048	7 BC3 1.085
			8 BA1 0.436	9 BC1 1.051	10 BC3 1.222	11 BC3 1.231	12 BB1 1.003	13 BB1 0.887
		14 BA3 0.413	15 BC1 1.063	16 BB2 1.066	17 BC5 1.328 X	18 BB4 1.043	19 BC4 1.252	20 BB4 0.962
	21 BA1 0.354	22 BC1 1.046	23 BB2 1.064	24 BC5 1.323	25 BB3 1.094	26 BC4 1.315	27 BB5 1.044	28 BC5 1.292
	29 BB5 0.578	30 BC3 1.221	31 BC5 1.327 X	32 BB3 1.094	33 BC5 1.325	34 BB3 1.079	35 BC5 1.316	36 2Z3 1.104
45 BA2 0.311	37 BB5 0.699	38 BC3 1.230	39 BB4 1.044	40 BC4 1.315	41 BB3 1.079	42 BC5 1.315	43 BB4 1.050	44 BC5 1.309
54 BB4 0.501	46 BC2 1.049	47 BB1 1.005	48 BC4 1.252	49 BB5 1.043	50 BC5 1.315	51 BB4 1.050	52 BC4 1.290	53 BB3 1.053
	55 BC3 1.085	56 BB1 0.887	57 BB4 0.962	58 BC5 1.292	59 2Z3 1.104	60 BC5 1.309	61 BB3 1.053	62 BA5 0.912

Long N-1 Burnup (21.535 GWd/MTU)

Note: X = Maximum Fr Value = 1.401

Calvert Cliffs Nuclear Power Plant	UNIT 2 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED TO PDIL AT EOC, HFP, EQUILIBRIUM XENON	Figure 3.4-20 Revision 49

BALTIMORE GAS & ELECTRIC CO.
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Nuclear Power Plant

CORE POWER DISTRIBUTION - PART LENGTH CEA (P-1), BEGINNING OF FIRST CYCLE, NO XENON

Figure 3.4-21

X — ENRICHMENT TYPE
0.00 — BOX PEAKING FACTOR
0.00 — ENTHALPY RISE FACTOR
0.00 — PIN PEAKING FACTOR

MAXIMA

BOX PEAKING FACTOR 1.23
ENTHALPY RISE FACTOR 1.31
PIN PEAKING FACTOR 1.35

	A	B	C	D	E	F	G	H	J	K	L
1								C 0.63 1.04 1.07		C 0.83 1.13 1.16	
2					C 0.63 1.03 1.06	C 0.86 1.18 1.21	C+ 1.00 1.29 1.30		C+ 1.14 1.23 1.27		B+ 0.99 1.09 1.11
3				C 0.74 1.14 1.17	C• 1.09 1.26 1.29	B+ 1.03 1.15 1.17	A 1.06 1.13 1.16		B+ 1.04 1.12 1.15		A 1.01 1.08 1.11
4			C 0.74 1.14 1.17	C• 1.14 1.30 1.33	B+ 1.08 1.19 1.21	A 1.10 1.17 1.21	B+ 1.09 1.18 1.20		A 0.98 1.06 1.09		B+ 0.90 1.04 1.06
5		C 0.63 1.03 1.06	C• 1.09 1.26 1.29	B+ 1.08 1.19 1.21	A 1.13 1.20 1.24	B+ 1.15 1.23 1.25	A 1.08 1.16 1.20		B+ 0.93 1.10 1.11		A 0.55 0.64 0.69
6		C 0.86 1.18 1.21	B+ 1.03 1.15 1.17	A 1.10 1.17 1.21	B+ 1.15 1.23 1.25	A 1.14 1.22 1.25	B+ 1.14 1.22 1.25		A 1.03 1.12 1.16		B+ 0.96 1.13 1.15
7		C+ 1.00 1.29 1.30	A 1.06 1.13 1.16	B+ 1.09 1.18 1.20	A 1.08 1.16 1.20	B+ 1.14 1.22 1.25	A 1.15 1.22 1.26		B+ 1.17 1.26 1.28		A 1.14 1.22 1.26
8	C 0.63 1.04 1.07										
9		C+ 1.14 1.23 1.27	B+ 1.04 1.12 1.15	A 0.98 1.06 1.09	B+ 0.93 1.10 1.11	A 1.03 1.12 1.16	B+ 1.17 1.26 1.28		A 1.20 1.28 1.32		B+ 1.23 1.31 1.34
10	C 0.83 1.13 1.16										
11		B+ 0.99 1.09 1.11	A 1.01 1.08 1.11	B+ 0.90 1.04 1.06	A 0.55 0.64 0.69	B+ 0.96 1.13 1.15	A 1.14 1.22 1.26		B+ 1.23 1.31 1.34		A 1.23 1.31 1.35

C_L

3.4-22 CORE POWER DISTRIBUTION - PART LENGTH CEA (P-1), BEGINNING OF FIRST CYCLE, EQUILIBRIUM XENON

X	
0.00	BOX PEAKING FACTOR
0.00	ENTHALPY RISE FACTOR
0.00	PIN PEAKING FACTOR

X = ENRICHMENT TYPE

MAXIMA

BOX PEAKING FACTOR	1.23
ENTHALPY RISE FACTOR	1.42
PIN PEAKING FACTOR	1.47

	H	K
1	C 0.64 1.03 1.04	C 0.81 1.11 1.12

	A	B	C	D	E	F	G	J	K
2					C 0.63 1.04 1.05	C 0.85 1.16 1.18	C+ 1.06 1.32 1.36	C+ 1.23 1.36 1.40	B+ 1.12 1.22 1.26
3				C 0.76 1.18 1.19	C● 1.16 1.37 1.42	B+ 1.14 1.25 1.28	A 1.08 1.13 1.15	B+ 1.16 1.23 1.27	A 1.06 1.12 1.15
4			C 0.76 1.18 1.19	C● 1.22 1.42 1.47	B+ 1.22 1.30 1.34	A 1.12 1.18 1.20	B+ 1.16 1.24 1.28	A 0.97 1.07 1.10	B+ 0.96 1.12 1.14
5		C 0.63 1.04 1.05	C● 1.16 1.37 1.42	B+ 1.22 1.30 1.34	A 1.15 1.19 1.22	B+ 1.21 1.29 1.33	A 1.03 1.11 1.13	B+ 0.93 1.09 1.12	A 0.51 0.65 0.71
6		C 0.85 1.16 1.18	B+ 1.14 1.25 1.28	A 1.12 1.18 1.20	B+ 1.21 1.29 1.33	A 1.07 1.14 1.17	B+ 1.09 1.18 1.22	A 0.90 0.98 1.00	B+ 0.88 1.00 1.03
7		C+ 1.06 1.32 1.36	A 1.08 1.13 1.15	B+ 1.16 1.24 1.28	A 1.03 1.11 1.13	B+ 1.09 1.18 1.22	A 0.98 1.04 1.06	B+ 1.04 1.09 1.13	A 0.93 0.98 1.00
8	C 0.64 1.03 1.04								
9		C+ 1.23 1.35 1.40	B+ 1.16 1.23 1.27	A 0.97 1.07 1.10	B+ 0.93 1.09 1.12	A 0.90 0.98 1.00	B+ 1.04 1.09 1.13	A 0.96 1.00 1.02	B+ 1.05 1.10 1.14
10	C 0.81 1.11 1.12								
11		B+ 1.12 1.22 1.26	A 1.06 1.12 1.15	B+ 0.96 1.12 1.14	A 0.51 0.65 0.71	B+ 0.88 1.00 1.03	A 0.93 0.98 1.00	B+ 1.05 1.10 1.14	A 0.97 1.00 1.02

℄

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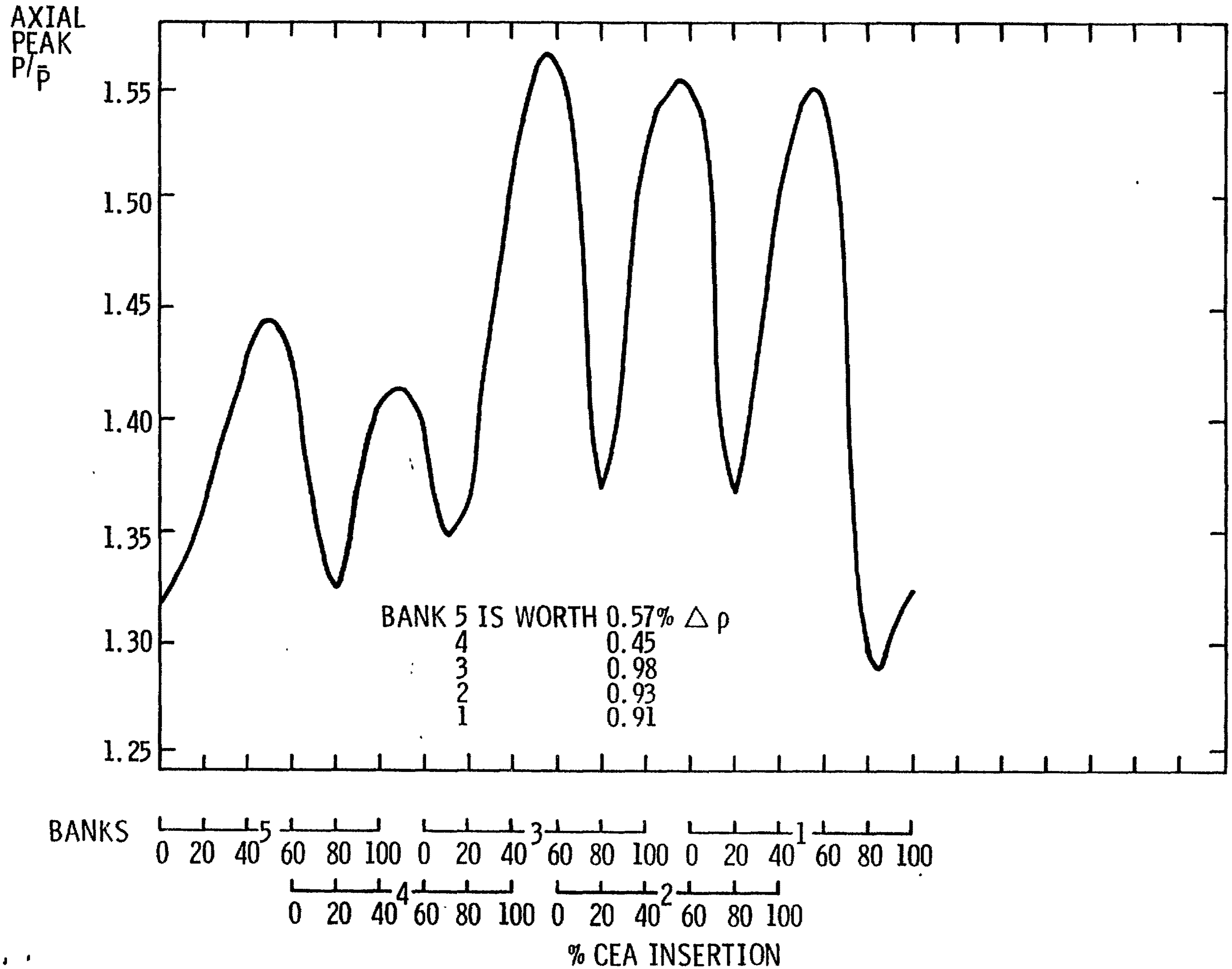
Calvert Cliffs Nuclear Power Plant

CORE POWER DISTRIBUTION - PART LENGTH CEA (P-1), BEGINNING OF FIRST CYCLE, EQUILIBRIUM XENON

Figure 3.4-22



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AXIAL PEAK vs % CEA INSERTION
(BEGINNING OF FIRST CYCLE)

Figure 3.4-23

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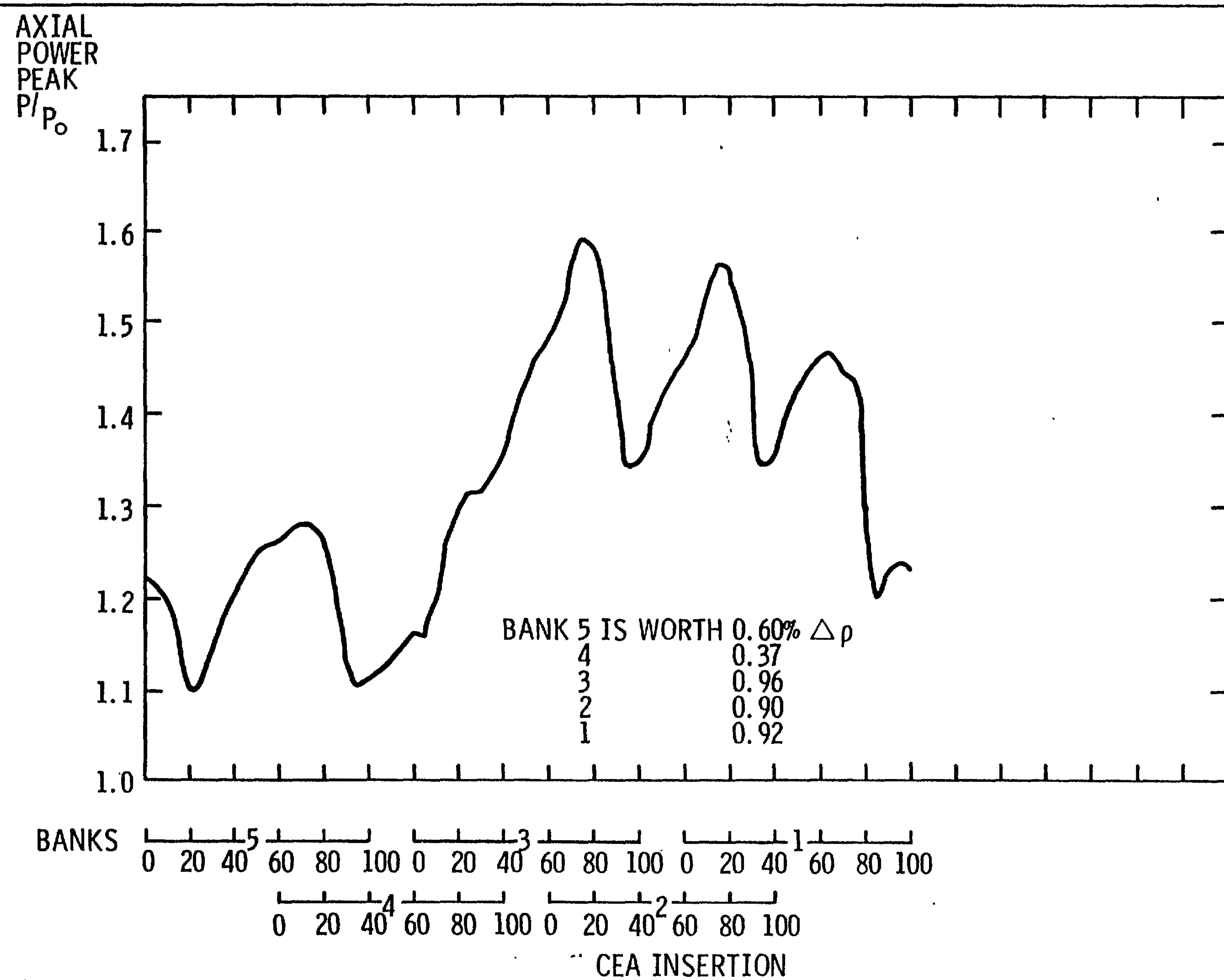
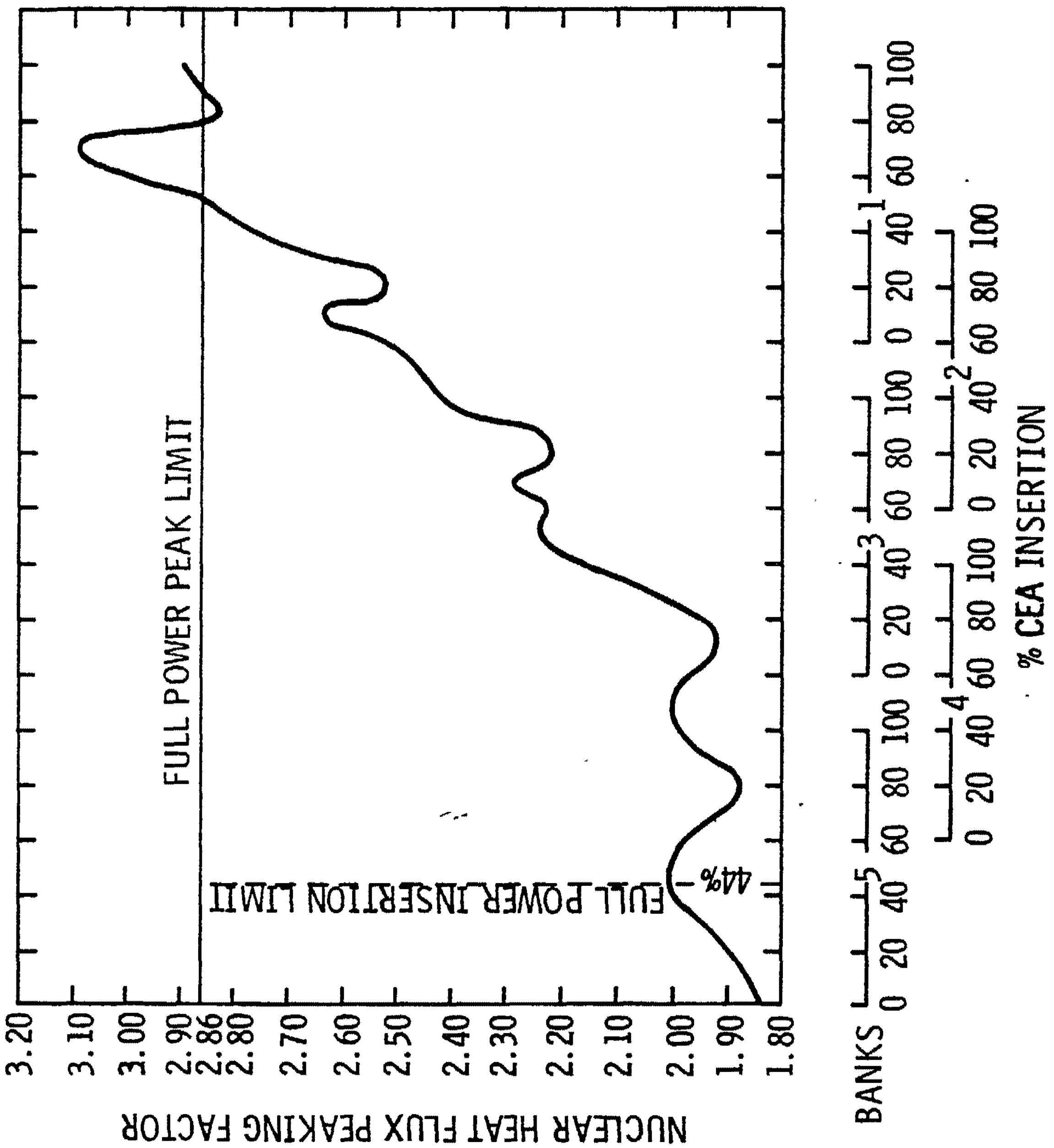
AXIAL PEAK vs CEA INSERTION WITH PART LENGTH CEA'S (END OF FIRST CYCLE)

Figure 3.4-24



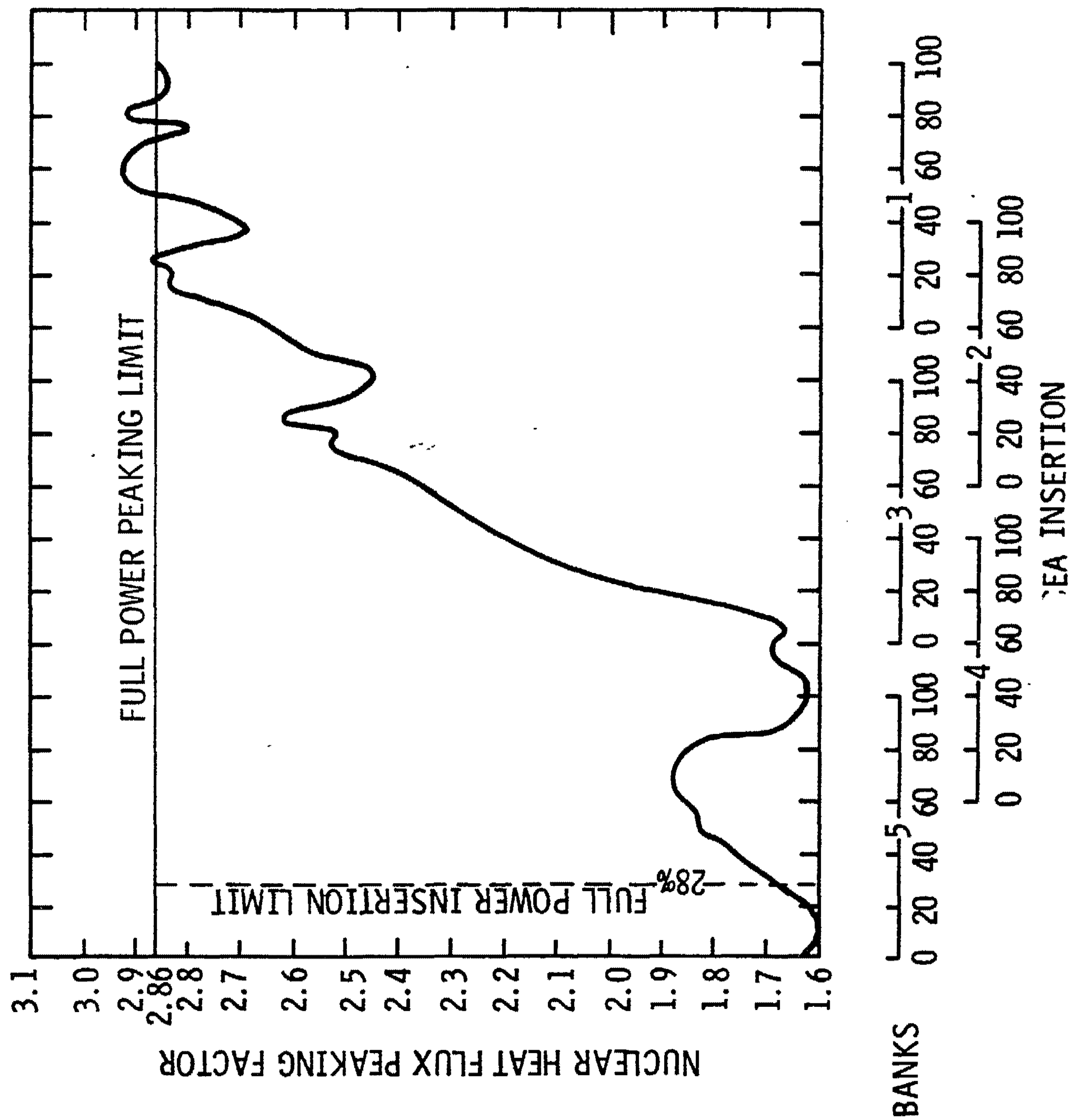
3.4-24 AXIAL PEAK VS CEA INSERTION WITH PART LENGTH CEAs (END OF FIRST CYCLE)



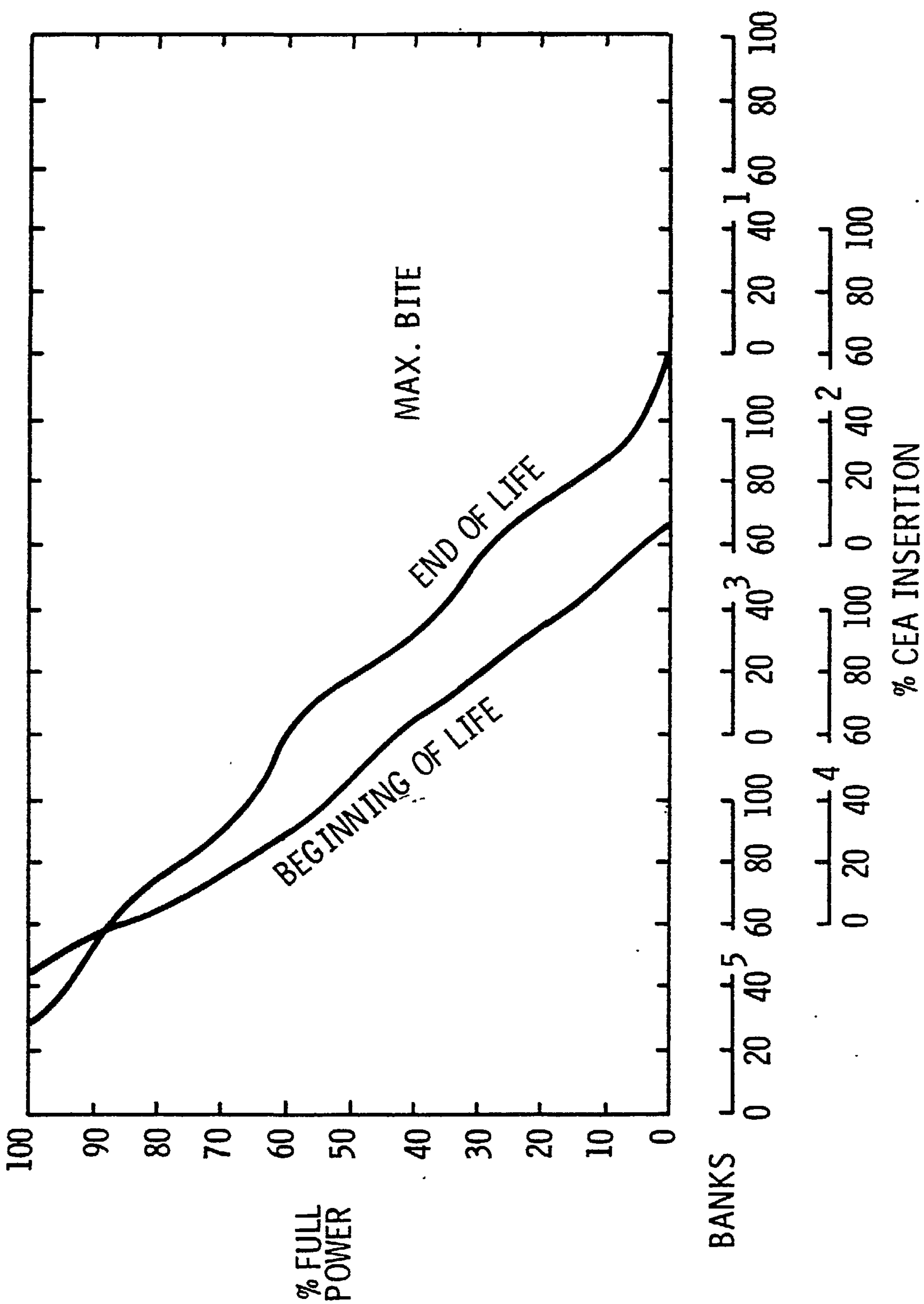
BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

NUCLEAR HEAT FLUX PEAK vs CEA INSERTION (BEGINNING OF FIRST CYCLE)

Figure 3.4-25



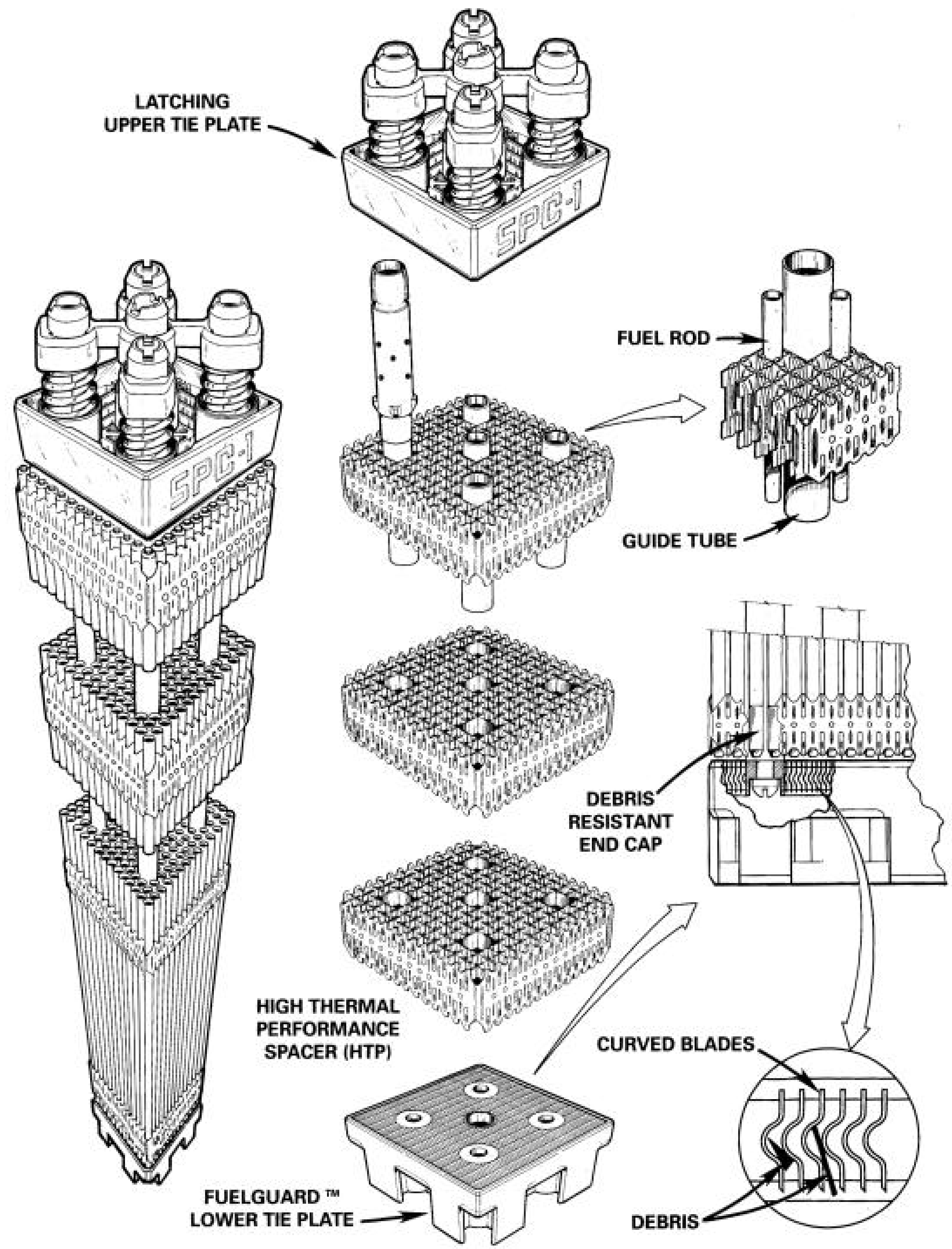
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	NUCLEAR HEAT FLUX PEAK vs CEA INSERTION WITH PART LENGTH CEA'S (END OF FIRST CYCLE)	Figure 3.4-26



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FIRST CYCLE POWER DEPENDENT CEA INSERTION LIMITS	Figure 3.4-27

Rev.0

14x14 PWR FUEL BUNDLE



Calvert Cliffs Nuclear Power Plant	Framatome Lead Fuel Assembly	Figure 3.7-1 Revision 33

CHAPTER 4
REACTOR COOLANT AND ASSOCIATED SYSTEMS

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REACTOR COOLANT AND ASSOCIATED SYSTEMS

LIST OF ACRONYMS

ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
BGE	Baltimore Gas and Electric Company
BWC	Babcock & Wilcox, Canada
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CCNPP	Calvert Cliffs Nuclear Power Plant
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
GMA	Gas Metal Arc
GTA	Gas Tungsten Arc
HAZ	Heat Affected Zone
HPSI	High Pressure Safety Injection
HSST	Heavy Section Steel Technology
LOCA	Loss-of-Coolant Accident
LTOP	Low Temperature Overpressure Protection
MPT	Minimum Pressurization Temperature
NDTT	Nil Ductility Transition Temperature
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OSG	Original Steam Generator
P-T	Pressure-Temperature
PORV	Power-Operated Relief Valve
PTS	Pressurized Thermal Shock
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RSG	Replacement Steam Generator
RVCH	Reactor Vessel Closure Head
SA	Submerged Arc
SCC	Stress Corrosion Cracking
SG	Steam Generator
SMA	Shielded Metal Arc
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake

4.0 REACTOR COOLANT AND ASSOCIATED SYSTEMS

4.1 REACTOR COOLANT SYSTEM

4.1.1 DESIGN BASIS

The Reactor Coolant System (RCS) initially operated at a power level of 2570 MWt. Due to the conservative design of the RCS, it is capable of attaining higher power levels. Rated thermal power was upgraded first to 2700 MWt, and then increased once again to 2737 MWe as part of a measurement uncertainty recapture modification. The major systems and components that bear significantly on the acceptability of the site have been evaluated for operation at a core power level of 2737 MWt. The principal design parameters for the RCS after the approval for 2737 MWt operation are listed in Table 4-1. The design parameters for each of the major components are given in Section 4.1.3 for each individual component. The RCS is designated a Category 1 system for seismic design and is designed to the criteria for load combinations and stress, which are presented in Table 4-8.

The number of seismic loading cycles used for design was based upon the occurrence of 40 full cycles, with significant motion peaks, during one seismic event and the assumption that 5 seismic events may occur during the life of the plant. These design loads were calculated for longitudinal and circumferential breaks occurring at any point in the geometry of the RCS without restriction. The worst case was selected for each geometry and for each direction.

The system design temperature and pressure are conservatively established and exceed the combined normal operating value and the change due to anticipated operating transients. They include the effects of instrument error and the response characteristics of the control system. The change due to the anticipated transients also considers the effect of reactor core thermal lag, coolant transport time, system pressure drop and the characteristics of the safety and relief valves.

The following design cyclic transients, which include conservative estimates of the operational requirements for the components discussed in Section 4.1.3, were used in the fatigue analyses required by the applicable codes listed in Table 4-9. The design cyclic transients for the operational requirements of the steam generator (SG) and pressurizer are discussed in Sections 4.1.3.2 and 4.1.3.5, respectively:

a. 500 [300 for replacement reactor vessel closure head (RVCH)] heatup and cooldown cycles during the system 40-year design life at a heating and cooling rate of 100°F/hr between 70°F and 532°F;

b. 15,000 (11,000 for replacement RVCH) power change cycles over the range of 15% to 100% of full load with a ramp load change of 5% of full load per minute increasing and decreasing;

c. 2,000 cycles of 10% of full load step power changes, increasing from 10% to 90% of full power and decreasing from 100% to 20% of full load;

d. 10 cycles of hydrostatic testing the RCS at 3125 psia and a temperature at least 60°F above the Nil Ductility Transition Temperature (NDTT) of the component having the highest NDTT;

e. 320 cycles of leak testing at 2500 psia and at a temperature at least 60°F greater than the NDTT of the component having the highest NDTT;

f. 10^6 cycles of normal variations of ±100 psi and ±6°F at operating temperature and pressure; and

g. 400 reactor trips from 100% power.

In addition to the above list of normal design transients, the following abnormal transients were also considered when arriving at a satisfactory usage factor as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III; however, these transients were not used to form the basis for the code design of the components.

a. 40 cycles of loss of turbine load from 100% power without a direct reactor trip [replacement steam generators (RSGs) used 50 cycles];
b. 40 cycles of total loss of reactor coolant flow when at 100% power; and
c. 5 cycles of loss of secondary system pressure.

4.1.1.1 Effect of Steam Generator Replacement on Structural Design Basis

The original steam generators (OSGs) provided by Combustion Engineering, Inc. (CE) have been replaced by SGs designed and fabricated by Babcock & Wilcox, Canada (BWC). The RSGs are designed to be "form, fit, and function" replacements of the OSGs, with only minor differences in design and functionality. The RSGs consist of a combination of retained OSG components and replacement components. The OSG steam drum shell and the upper half of the transition cone shell are retained while the rest of the pressure boundary and all of the internals are replaced. Reference 40 provides a comparison of the OSGs and RSGs for the factors that have the potential to affect the structural evaluation of the RCS loop. The differences are summarized below:

a. The weight of the RSG is 2% greater than that of the OSG.
b. The center of gravity of the RSG is 1% lower than that of the OSG.
c. The replacement secondary shell of the RSG is 35% thinner than that of the OSG.
d. The RSG replacement pressure boundary materials have 3.5% higher coefficients of thermal expansion than those of the OSG.
e. The RSG replacement pressure boundary materials have 3.5% lower Young's Modulii than those of the OSG.

Reference 40 has evaluated the effects of the differences between the RSGs and OSGs on the structural qualification of the RCS components, component supports, and piping for pressure, thermal, deadweight, seismic, and pipe rupture loading. It concluded the existing RCS design basis loading and analyses remain valid for all components.

4.1.2 SYSTEM DESCRIPTION

The function of the RCS is to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system. The RCS, which is entirely located within the containment, consists of two heat transfer loops connected in parallel across the reactor pressure vessel. Each loop contains one SG, two circulating pumps, connecting piping, and flow and temperature instrumentation. Coolant system pressure is maintained by a pressurizer connected to one of the loop hot legs.

The RCS is shown in Figure 4-1 (Unit 1) and Figure 4-17 (Unit 2). During operation, the four pumps circulate water through the reactor vessel where it serves as both coolant and moderator for the core. The heated water enters the two SGs, transferring heat to the secondary (steam) system, and then returns to the pumps to repeat the cycle.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations

caused by contraction or expansion of the reactor coolant. The pressurizer is located with its base at a higher elevation than the reactor coolant loop piping. This eliminates the need for a separate pressurizer drain, and ensures that the pressurizer is drained before maintenance operations. A vent is provided in the pressurizer vapor sample line for removal of non-condensable gases during natural circulation.

Overpressure protection is provided by two power-operated relief valves (PORVs) and two spring-loaded safety valves connected to the top of the pressurizer. Steam discharged from the valves is cooled and condensed by water in the quench tank. The reactor coolant vent lines from the reactor vessel and the pressurizer also discharge to the quench tank. In the unlikely event that the discharge exceeds the capacity of the quench tank, the tank is relieved to the containment via the quench tank rupture disc. The quench tank is located at a level lower than the pressurizer. This ensures that any PORV or pressurizer safety valve leakage from the pressurizer, or any discharge from these valves, drains to the quench tank.

The RCS and its associated controls are designed to accommodate plant step load changes of ±10% of full power and ramp changes of ±5% of full power per minute without reactor trip. The system will accept, without damage, a complete loss of load.

To maintain reactor coolant chemistry within the limits discussed in Section 4.1.4.2.3 and to control pressurizer level, a continuous but variable letdown flow from one loop upstream of the reactor coolant pump (RCP) is maintained. This bleed flow is controlled by pressurizer level. Constant coolant makeup is added by charging pumps in the Chemical and Volume Control System (CVCS).

An inlet nozzle on each of the four reactor vessel inlet pipes allows injection of borated water into the reactor vessel from the Safety Injection System in the event emergency core cooling is needed. During a normal plant shutdown, these nozzles are also used to supply shutdown cooling flow from the low pressure safety injection pumps. An outlet nozzle on one reactor vessel outlet pipe is used to remove shutdown cooling flow.

There are two normally closed, solenoid valves in each of the vent lines from the reactor vessel and the pressurizer. These valves fail closed.

Drains from the reactor coolant piping to the radioactive waste processing system are provided for draining the RCS for maintenance operations. A connection is also provided on the quench tank for draining it to the radioactive waste processing system following a relief-valve or safety-valve discharge.

The major RCS components are designed for a 40-year service life. To assure that this objective can be attained, strict quality assurance standards as outlined in Sections 4.1.5.5 and 4.1.5.6 were followed.

Protection provided the RCS against environmental factors such as fires, floods and missiles are described in Sections 1.0, 5.0, 9.0, and 11.0.

4.1.3 COMPONENT DESCRIPTION

4.1.3.1 Reactor Vessel

The reactor vessel (Figure 4-2) is supported vertically and horizontally by three pads welded to the underside of the reactor vessel nozzles. Each assembly consists of the following:

a. A support foot (SA-508 CL2) welded to a reactor coolant nozzle;

b. A socket [American Society for Testing and Materials (ASTM A283-67)] bolted to the support foot with Allenoy cap screws; and

c. A sliding bearing (ASTM B22, Alloy E) whose spherical crown fits into the socket and whose flat sliding surface rests on a base plate (AISI-4140).

Design parameters for the reactor vessel are given in Table 4-2.

The arrangement of the vessel supports allows radial growth of the reactor vessel due to thermal expansion while maintaining it centered and restrained from movement caused by seismic disturbances. Departure from levelness of not more than 0.005 inch per foot of flange diameter is maintained during construction to facilitate proper assembly of reactor internals. The supports were analyzed in accordance with ASME B&PV Code, Section III.

The vessel closure flange is a forged ring with a machined ledge on the inside surface to support the reactor internals and core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seal. The vessel closure contains 54 studs, 7" in diameter, with 8 threads per inch. The stud material is ASTM A540, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 40 ksi. Calculations show that 32 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or 4 adjacent studs is necessary before the closure will fail by "zippering" open.

Analysis has shown that two thirds of the reactor vessel head studs can be removed while in Mode 5 in preparation for refueling operations. Eighteen of the total 54 studs remain in place (every third stud), tensioned to a pre-load equal to a minimum of 75% of design pre-load, with RCS pressure equal to 500 psia. Two loading conditions are considered: (1) RCS temperature equal to the saturation temperature, 467°F, simulating the worst case (highest) temperature, which could occur as a result of a loss of shutdown cooling; and (2) RCS temperature equal to 200°F, which is the normal maximum temperature for Mode 5 operation. The analysis indicates that the reactor vessel O-rings remain in compression during both conditions (no leakage), and the stud stresses due to detensioning in Mode 5 are less than allowable. Calculated stresses within the flanges which arise from Mode 5 detensioning are bounded by the stresses from the original design. It was also determined that the minimum number of studs required to properly seat the vessel head in Mode 5 is 12, but 18 is used for conservatism. Also, the single LTOP shutdown cooling PORV setpoint will be used to ensure reactor vessel pressure is maintained ≤ 500 psia.

Six radial nozzles on a common plane are located just below the vessel closure flange. Extra thickness in this vessel-nozzle course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around each outlet nozzle on the inside diameter of the vessel wall provides a mating surface for the internal structure which guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to reduce core bypass leakage. A fixed hemispherical head is attached to the lower end of the shell. There are no penetrations in the lower head.

The removable top closure head is hemispherical. The head flange is drilled to match the vessel flange stud bolt locations. The 54 stud bolts are fitted with spherical washers located between the closure nuts and head flange to maintain

stud alignment during head flexing due to boltup. To ensure uniform loading of the closure seal the studs are hydraulically tensioned with a special tool.

Flange sealing is accomplished by a double-seal arrangement utilizing two silver-plated Ni-Cr-Fe alloy, self-energized O-rings. The space between the two rings is monitored to allow detection of any inner ring leakage. The control element drive mechanism (CEDM) nozzles (Ni-Cr-Fe alloy through the head, stainless steel flange) terminate with threaded and seal-welded flanges at the upper end. There are eight instrumentation nozzles that terminate with a threaded flange and swagelok type seal for the instruments. In addition to these nozzles there is a 3/4" vent connection.

The core is supported from an internally machined core support ledge. The CEDMs are supported by the nozzles in the reactor vessel head.

4.1.3.2 Steam Generator

Replacement Steam Generator

The nuclear steam supply system (NSSS) utilizes two SGs (Figures 4-3A and 4-3B) to transfer the heat generated in the RCS to the secondary system. The SG shell is constructed of carbon steel. Manways and handholes are provided for easy access to the SG internals. The design parameters for the SGs are given in Table 4-3.

The SG is a vertical U-tube heat exchanger. The SG operates with the reactor coolant in the tube side and the secondary fluid in the shell side.

Reactor coolant enters the SG through the inlet nozzle, flows through 3/4" OD U-tubes, and leaves through two outlet nozzles. A vertical divider plate in the lower head separates the inlet and outlet plenums. The plenums are stainless steel clad with a SB-168 Alloy 690 unclad divider plate, while the primary side of the tube sheet is UNS N06052 cladding. The vertical U-tubes are SB-163 Alloy 690. The tube-to-tube sheet joint is welded on the primary side. Tubes that have degraded may be repaired using NRC approved methods. Any gases that may be trapped in the SG U-tubes are swept out by the initial reactor coolant flow and released via the pressurizer, CEDM, or reactor vessel venting systems. The maximum number of plugged tubes is 847 (10 percent of total tubes) per SG. If more than 847 tubes per SG are plugged, RCS flow may decrease below the Technical Specification required design flow. Of the 847 tubes, up to 20 may be plugged as a result of an intentional removal of a piece of the tube for purposes such as material testing (Section 14.20.1). The maximum allowed difference between the number of tubes plugged in each SG is 750.

Feedwater enters the SG through the feedwater nozzle where it is distributed via a feedwater distribution ring. The SG feedwater distribution system incorporates an all welded SA-335 P22 schedule 80 piping design, a “gooseneck” feed system positioned between the thermal sleeve and the feedwater header, and top discharging hairpin bend J-tubes to minimize the risk of water hammer, thermal flow stratification, and erosion. The water exits through J-tubes on the top of the feedwater ring, then flows into the downcomer. The downcomer is an annular passage formed by the inner surface of the SG shell and the cylindrical shell wrapper, which encloses the U-tubes. At the bottom of the downcomer, the secondary water is directed upward past the vertical U-tubes where heat transfer from the primary side produces a water-steam mixture.

Upon exiting from the vertical U-tube heat transfer surface, the steam-water mixture enters the primary separators, which are centrifugal-type separators. The mixture enters the curved arms of the primary separator where a film of water develops on the inner wall of the return cylinder, and spirals downward for recirculation. Steam at greater than 90% quality exits the top of the primary separators into the inter-stage region, which distributes steam prior to the secondary separators. The secondary separators are also centrifugal-type separators. Once the mixture is through the secondary separators, the design maximum moisture carryover is 0.05% (by weight) for 2717 MWt Reactor Core Power plus reactor coolant pump heat. Field experience and laboratory testing has demonstrated the primary and secondary separators are insensitive to operational pressure fluctuations, steam flow imbalances, and wide range in water level fluctuations. The separators are designed to last for the life of the RSG without maintenance or periodic cleaning. The primary separators have large flow passages that preclude plugging even if deposition occurs. The secondary separator inlet body and outlet passages are also large. The skimmer and vent hole passages of the secondary separator are continuously swept by flow during operation. In the unlikely event of skimmer or vent hole plugging, cleaning by water lancing is possible.

The installation of the RSGs involved first cutting and removing the upper (steam drum) portion of each OSG. The steam drums were then refurbished and reinstalled atop the new RSG lower assemblies. The previous paragraph discusses design improvements in moisture separation equipment for the RSGs. Other design improvements include:

a. Use of improved SG tube materials and fabrication to increase tube reliability.
b. Use of improved tube support materials to minimize flow-assisted corrosion of the tube supports.
c. Design measures taken to optimize feedwater distribution within the SGs, minimize thermal stratification, and preclude water hammer.
d. Design improvements to the blowdown functions of the RSGs for enhanced SG water chemistry control.
e. Use of an integral steam flow restrictor to enhance the results of the main steam line break accident analysis.

In performing the “2-piece” replacement of the SGs, the code of record for the steam drums vessels is and will remain the ASME III 1965 Edition, with Winter 1967 Addenda. New items (RSG lower assembly and feedwater transition piece) were constructed to meet all requirements of ASME Section III, 1989, Edition, No Addenda.

Overpressure protection for the shell side of the SGs and the main steam line piping up to the inlet of the turbine stop valve is provided by 16 spring-loaded ASME code safety valves that discharge to atmosphere. The power-operated steam dump valves and steam bypass valves obviate opening of the main steam safety valves following turbine and reactor trip from full power. The Main Steam System, including the steam dump and bypass valves, and the main steam safety valves, is described in Section 10.1.

The SGs are mounted on bearing plates which allow lateral motion due to thermal expansion of the reactor coolant piping. Stops are provided to limit this motion in case of a coolant pipe rupture. The top of each unit is restrained from sudden lateral movement by suitable stops and hydraulic snubbers mounted rigidly to the

concrete structure. Each SG is supported by a lower support assembly consisting of:

a. A support skirt (SA-533, Type B, Class 1) welded to the high pressure head;
b. A sliding base (ASTM A27, Gr 70-40) bolted to the skirt with ASTM A193-97 bolts and ASTM A194, Gr 7 nuts; and
c. Four sockets (ASTM A283-67) are bolted to the underside of the base with Allenoy cap screws and a sliding bearing (ASTM B22, Alloy E) whose spherical crown fits into each socket and whose flat sliding surface rests on a base plate (AISI 4140).

The supports were analyzed in accordance with ASME B&PV Code, Section III. The upper SG supports consist of:

a. Eight clevices (SA-533, Gr. B, Class 1) welded to the upper shell;
b. A hydraulic snubber is attached to each clevice by means of a pin (ASTM A193, B7); and
c. Two keys (SA-533, Gr. B, Class 1) welded to the upper shell. The clevices and keys are analyzed to ASME B&PV Code, Section III. The pin and hydraulic snubbers are sized so that stresses will not exceed 90% of yield. These items are tested at rated load to confirm their adequacy.

The following design cyclic transients for the operational requirements of the SG were used in the fatigue analyses required by the applicable codes listed in Table 4-9 and in accordance with the load combinations noted in Table 4-8 (References 1 and 2):

a. 500 heatup and cooldown cycles during the vessels 40-year design life at a heating and cooling rate of 100°F/hr between 70°F and 532°F;
b. 15,000 power change cycles over the range of 15% to 100% of full power with ramp load change of 5% of full power per minute increasing and decreasing;
c. 2000 cycles of 10% of full load step power changes, increasing from 90% to 100% of full load and decreasing from 100% to 90% of full load;
d. 10,000 cycles of adding 600 gpm of 70°F feedwater with the plant in hot standby condition;
e. 10^6 cycles of normal primary side variations of ±100 psi and ±6°F at operating temperature and pressure;
f. 10^6 cycles of normal secondary side variations of ±40 psi at operating temperature and pressure;
g. 4,000 cycles of transient pressure differentials of 85 psi across the primary head divider head plate due to starting and stopping the primary coolant pumps;
h. 400 reactor trips from 100% power;
i. 40 cycles of total loss of reactor coolant flow when at 100% power;
j. 50 cycles of loss of turbine load from 100% power without a direct reactor trip;
k. 5 cycles of loss of secondary system pressure;
l. 8 cycles of loss of feedwater flow;
m. 320 cycles of leak testing at 2500 psia and 130°F;
n. 320 cycles of leak testing at 1015 psia and 150°F;

o. 10 cycles of hydrostatic testing the RCS at 3125 psia and 70°F for shop hydrostatic tests, and 150°F for field hydrostatic tests with the secondary side at atmospheric pressure. Any combination of shop or field hydrostatic tests can be used to obtain a total of 10 cycles.

p. 10 cycles of hydrostatic testing of the secondary side at 1269 psia and 70°F for shop hydrostatic tests and 175°F field hydrostatic tests with the primary side at atmospheric pressure. Any combination of shop or field hydrostatic test can be used to obtain a total of 10 cycles.

q. 40 cycles for testing the upper snubber lugs;

r. 100 cycles with SG level lower than top of feedwater header gooseneck for greater than 60 minutes with no feedwater flow; and

s. 200 pre-load cycles for primary manway, secondary manway, handhole, and inspection port covers.

In addition to the normal design transients listed above and those listed in Section 4.1.4.1, the following additional abnormal transient was also considered in arriving at a satisfactory usage factor as defined in ASME B&PV Code, Section III: 8 cycles of adding a maximum of 650 gpm of 32°F feedwater, with the SG secondary side dry and at 600°F. This transient, however, did not form a basis for the code design of the SGs.

The unit is capable of withstanding these conditions for the prescribed numbers of cycles, in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor as prescribed in ASME B&PV Code, Section III.

More specifically, the SG will be designed such that no damage to the equipment is caused by the frequency ranges of 14-15 cps and 70-75 cps. The lower frequency range is defined as a mechanical vibration induced by the RCP, and the upper frequency range is defined as a sinusoidal pressure variation of ±6 psi in the cold leg piping also induced by the RCP. It has been determined that there will be tube wall margin under the postulated condition of the largest design basis pipe break in the reactor coolant pressure boundary during reactor operation. Tubing design incorporates an acceptable wall thinning up to 44.5 percent. Therefore, providing excess material in the tube wall thickness has accommodated any degradation of tubes that may occur during the service life.

The SG has been designed to ensure that critical vibration frequencies will be well out of the range expected during normal operation and during abnormal conditions. The SG tubing and tubing supports are designed and fabricated with considerations having been given to both secondary-side flow-induced vibrations and RCP-induced vibrations. In addition, the heat transfer tubing and tube supports have been designed such that they shall not be structurally damaged under the loss of secondary pressure conditions that may produce a fluid velocity in the tube bundle four times design velocity.

It has been found that all tubes and tube sections that will experience forcing functions from cross flow and parallel flow have natural frequencies sufficiently different from the frequency of the forcing function that they will not experience damaging vibrations. The mechanical excitation frequency is sufficiently different from the lowest natural frequency for out-of-plane or lateral vibration in any tube span that critical vibration will not occur.

Even though some of the tube sections have a geometry such that they may vibrate near resonance with the 70 to 75 cps hydraulic pulse forcing function, the resulting stresses and deflections are negligible.

Tube denting has occurred on both units at Calvert Cliffs in the original CE SGs to a minor extent. Denting is a phenomenon characteristic to SGs with drilled carbon steel support plates. Volumetric expansion of the support plate material, when converted to magnetite, causes a squeezing action on the tubes. If the support plates are detached from the shroud, a large portion of the stresses are relieved. On original CE SGs 11 and 12, the attachment lugs for support plate numbers 9 and 10 were cut for this purpose. The RSGs do not have drilled plate tube supports, but instead use stainless steel lattice grids and fan bars. Therefore, the denting phenomena is not expected.

Replacement Steam Generator Structural Damping

Damping values are assigned to the RSG internals based on the recommendations of NRC Regulatory Guide (RG) 1.61 and ASME Code Case N-411.

With the exception of the U-tubes, the RSG internals are predominantly an assembly of welded and bolted steel structures. Per RG 1.61, they qualify for 2% to 4% damping for the operating basis earthquake (OBE) and 4% to 7% for the safe shutdown earthquake (SSE). Damping values of 2% for the OBE and 4% for the SSE are used for the internals.

The U-tubes are an assembly of small pipes. For OBE loading, per ASME Code Case N-411, 5% damping is used for frequencies below 10 Hz, 2% damping for frequencies above 20 Hz, and a linear relationship between these damping values is used for intermediate frequencies. The ASME Code Case N-411 damping is also used for the U-bend support assembly, since it is embedded within and supported by the U-tubes, and it is only significantly loaded when the U-tubes are in motion. Testing done at BWC has shown that higher structural damping is appropriate for the U-tubes and U-bend support assembly, as allowed by RG 1.61 when supported by documented tests data. Therefore, for SSE loading, 7% damping is used for the U-tubes and U-bend support assembly.

Steam Generator Tube Inspections

A program of inservice inspection of SG tubes using eddy current techniques is employed. The details regarding techniques, extent of inspections, frequency of inspections, and calibration and acceptance criteria are specified by the Technical Specifications and ASME B&PV Code, Section XI.

The baseline examination of the Unit 1 SG tubes was performed in June 1972. It was performed for the purpose of establishing typical eddy current signatures of new tubes in good condition. Approximately 8% of the tubes in each SG were examined. Thirty-eight tubes (in both generators) were examined over their entire lengths. For the remainder, examinations did not extend beyond the bends. No significant indications were noted.

Subsequently, new "baseline" examinations of the Unit 1 SG tubes were performed in November 1986, and Unit 2 in April 1987. They were performed to establish, based on current technology, eddy current signatures of tube conditions. All tubes in each SG were examined over their entire length. Relevant indications were noted and resolved by tube plugging or data analysis.

With the replacement of the original CE SGs with BWC RSGs, the preceding two paragraphs present historical information only. The baseline for the BWC RSGs was established by performing a 100% bobbin-coil preservice inspection for the SG tubes. In addition, rotating probe coil examinations were performed for the following:

a. dings, dents, manufacturing burnish marks, and anomalous signals over 5 Volts;
b. First three rows of U-bends above the upper support grid;
c. Two smallest radius rows of non-heat treated U-bends; and
d. 100% top of tubesheet (±3 inches of secondary side face of tubesheets).

Steam Generator Pressure/Temperature Limitation

The SG pressure and temperature are limited to ensure that the pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. The temperatures of both the primary and secondary coolants in the SGs are maintained above 80°F for Unit 1 and Unit 2 (Reference 41) when the pressure of either coolant in the SG is greater than 200 psig. These limitations are based on SG secondary side limitations and are sufficient to prevent brittle fracture.

4.1.3.3 Reactor Coolant Pumps

The reactor coolant is circulated by four vertical, single-suction, centrifugal-type pumps (Figure 4-4). The suction nozzles are in the bottom vertical position. The pressure containing components are designed and fabricated in accordance with ASME B&PV Code, Section III, Class A.

The pump impeller is keyed and locked to the shaft. A close clearance thermal barrier assembly is mounted above the water-lubricated bearing to retard heat flow from the pump to the seal cavity which is located above the thermal barrier. The thermal barrier assembly also tends to isolate the hot fluid in the pump from the cooler fluid above, and, in the event of a seal failure, serves as an additional barrier to reduce leakage from the pump. Each pump is equipped with replaceable casing wear rings. A water-lubricated bearing is located in the fluid between the impeller and thermal barrier to provide shaft support. Additional shaft support is provided by bearings in the electric motor, which is directly connected to the pump shaft by a rigid coupling.

The shaft seal assembly, located above the thermal barrier, consists of four face-type mechanical seals, three full-pressure seals mounted in tandem and a fourth low-pressure backup vapor seal designed to withstand operating system pressure with the pumps operating or stopped (Figures 4-4, 4-6, and 4-6A). The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system. A controlled bleedoff flow through the pump seals is maintained to cool the seals and to equalize the pressure drop across each seal. The controlled bleedoff flow is collected and processed by the CVCS. Any leakage past the vapor seal (the last mechanical seal) is piped to containment trench at Elevation 10'0". The seals are cooled by circulating the controlled leakage through a heat exchanger mounted integrally within the pump cover assembly; no damage would result in the event of pump operation without cooling water for at least five minutes. To reduce plant downtime and personnel exposure to radiation during seal maintenance, the seal system is contained in a cartridge which can be removed and replaced as a unit. The face seals can be replaced without draining the pump casing. The seal detail is shown in Figure 4-6.

The original Byron Jackson SU seals have been replaced with a state-of-the-art design under FCR 87-0074. The new seals are designed in accordance with ASME B&PV Code, Section III, 1983 Edition with Summer 1983 Addenda, and manufactured by Sulzer Bingham Pumps. The model (RCR875B-3V) incorporates several design features in addition to those listed above. These features include:

Flexible Stator - Accommodates large shaft tilt without disturbing the sealing faces or fluid film.

Rotating Support Ring - Isolates the sealing ring from adjacent metallic parts through the use of a narrow support nose and three nonsealing O-rings. This feature protects the sealing faces and fluid film from the effects of large temperature and pressure transients.

Material Selection - All materials were carefully chosen based on their properties for strength, corrosion, temperature and radiation resistance, chemical content and differential thermal expansion.

A motor-mounted flywheel reduces the rate of flow decay upon loss of pump power. The combined inertia of the pump motor and flywheel is 100,000 lbm/ft^2. Flow coastdown characteristics are discussed in Section 14.9.

The pump motor assembly includes motor bearing oil coolers, seal chamber, controls and instruments. Cooling water is provided from the Component Cooling System. A mechanism is provided on each pump to prevent reverse rotation. This mechanism is a circular rack with sprags.

The design parameters for the RCPs are given in Table 4-5. The CE report of November 1973 on RCP flow measurement is found in Appendix A at the end of this chapter.

The RCP and motor are supported by four support lugs welded to the volute. The pump is supported on spring hangers employed between the support lugs and the floor below. The lugs are analyzed to ASME B&PV Code, Section III. Movement in the horizontal plane to compensate for pipe thermal growth and contraction is permitted. Vertical movement is not restrained.

The pump is constructed of high alloy cast stainless steel parts to minimize corrosion. The mechanical seals consist of a rotating tungsten carbide ring riding over a carbon graphite stationary face. The design life of this seal arrangement is at least four years. Each seal is designed to accept a pressure drop equal to full operating system pressure, but normally operates at one-third this pressure drop.

The expected pump performance curve is shown in Figure 4-7. The air-cooled, self-ventilated pump motor is sized for continuous operation at flows resulting from four-pump operation or partial pump operation with 0.74 specific gravity water. The motor service factor is sufficient to allow continuous operation with 1.0 specific gravity water. The motors are designed to start and accelerate to speed under full load when 80% or more of their normal voltage is applied. The motors are contained within standard drip-proof enclosures and are equipped with electrical insulation suitable for a 0% to 100% humidity and radiation environment of 30 R/hr.

4.1.3.3.1 Reactor Coolant Pump Flywheel

The design requirements of the RCPs include a minimum inertia for the rotating assembly of 100,000 lbm/ft^2; to achieve this total, a flywheel with an inertia of 70,000 lbm/ft^2 has been incorporated.

The flywheel assembly consists of two discs bolted together and keyed to the shaft above the motor. The dimensions of the discs are:

	Upper Disc	Lower Disc
Outside Diameter, in.	75	65
Thickness, in.	8	5
Weight, ea, lb	10,123	4,750

The selection of material, machining and manufacturing operations, quality control and the rigorous acceptance criteria established to assure the integrity of the flywheel and to minimize operating stresses include the following:

a) The A533 vacuum-degassed flywheel material normally shows a 15 ft/lb longitudinal V-notch transition temperature of 40°F; the NDTT of the flywheel material is at least 30°F below the minimum motor operating temperature.

b) In flame cutting of the bore, at least l/2" of stock was left on the radius for machining;

c) There are no stress concentrations such as stencil or punch marks, or drilled or tapped holes within 8" of the edge of the flywheel bore;

d) Each flywheel plate was ultrasonically inspected to ASME B&PV Code, Section III, Paragraph N-322;

e) After balancing, the flywheel and motor assembly were tested at 120% of design speed. The maximum allowable vibration for acceptance of the assembly was 1.5 mils; and

f) A keyway fillet radius not less than 1/8" minimizes stress concentration factors.

An analysis to determine the rotational velocity which would cause flywheel failure if a crack at the flywheel bore were present indicates the following:

a) No failure would occur at the design speed;

b) At 100% overspeed a crack 15" on each side of the bore would be required; and

c) At 150% overspeed a crack 6" on each side of the bore would be required.

To assure that no deleterious effects occur after the flywheels are placed in service, the flywheels will be inspected periodically as part of the inservice inspection program.

The NDTT of the flywheel material is less than +10°F. The data for the eight heats used in the fabrication of the RCP flywheels is tabulated below:

Heat No.		V-notch at +10°F		
C7689	T	53	59	55
	L	115	110	104
C7174	T	53	70	61
	L	110	105	97
A6678	T	52	41	57
	L	86	57	103

Heat No.		V-notch at +70°F		
B3616	T	128	116	112
	L	114	118	116
D8392	T	110	112	110
	L	140	146	146

Heat No.		V-notch at +10°F		
A6735	T	27	43	52
	L	53	51	50
C7434	T	65	65	50
	L	78	81	116
C9220	T	51	32	43
	L	92	88	94

The Charpy V-notch upper shelf energy level was not specified. The data contained above is acceptable.

The fracture toughness of the material at 75°F is 140 ksi/in$^{1/2}$.

Flywheel material has been subjected to a 100% volumetric ultrasonic inspection from the flat surface, per ASME B&PV Code, Section III, Paragraph N-322. The acceptance standard is as follows:

a. Indication greater than 4" diameter is cause for rejection per ASTM A435; and

b. Indications of complete loss of back reflection greater than one crystal diameter and less than 4" diameter shall be tested by angle beam technique. Indications greater than a 3% notch is cause for rejection.

The flywheel has been subjected to a magnetic-particle or liquid-penetrant examination in accordance with ASME B&PV Code, Section III, Paragraph N-322 before assembly of flywheel plates. This inspection was done on both plate surfaces to a distance of 8" minimum beyond the final bore diameter and after machining in the bore. The acceptance standard for the inspection is as follows:

a. Indications of cracks and linear defects are subject to rejection. A linear defect is one in which the length is three times the width. The minimum length of defects to be considered linear shall be 3/16"; and

b. The principal stress is to be not greater than 50% of the yield point (based on transverse test specimens taken at 1/4t) of the flywheel material at normal operating speed, not considering keyway stress concentration factors.

The pump assembly is designed to be capable of operating up to 125% of normal operating speed for periods not exceeding 10 seconds in accordance with NEMA standards. In addition, each motor and flywheel is tested at 120% overspeed.

The calculated combined primary stresses in the flywheel at the normal operating speed is 7320 psi at 900 rpm. The interference fit of the flywheel in this assembly is negligible.

During normal plant operation, the RCPs are powered by station-generated power. Since the motor speed is directly proportional to the frequency of the power source, the maximum anticipated pump speed is 110% of nominal, which is equal to the overspeed trip setpoint of the turbine generator.

The estimated rotational speed that the flywheel would attain during a rupture in the discharge piping of the RCP is about 152% of rated speed (888 rpm) or 1350 rpm. A break in the suction piping causes the reactor coolant to flow through the pump opposite to the normal direction of flow decelerating the rotating assembly until it is brought to rest against the anti-reverse rotation device. The model used is described in CENPD-2, Appendix I, a proprietary report entitled, "Combustion Engineering Analytical Techniques for Evaluating Loss of Coolant Accidents."

Previous studies of the rotating limitations of RCPs have indicated that the first characteristic which will prevent excessive operating speeds is binding of the motor against the stator.

This limitation does not become effective for the Calvert Cliffs RCP motors during any mode of operation, including the worst-case pipe rupture, since it is estimated that approximately twice the speed calculated following a loss-of-coolant accident (LOCA) is required to produce rotor binding.

The RCPs are typical centrifugal volumetric flow machines. The pump response following a LOCA is predicted using generally accepted methods as described in CENPD-26. A spectrum of breaks in the RCP discharge line have been analyzed and the results follow a predictable pattern. Assuming loss of electrical power to the pump at the start of the LOCA, it is seen that the pumps initially lose speed because the volumetric flow through the pump is not sufficient to sustain the nominal speed of rotation. The volumetric flow increases during the transient, accelerating the pump to its maximum speed. The extent of the initial loss of speed varies with the break size. The larger the break size, the less the initial deceleration and the higher the maximum speed attained. As previously stated, for the double-ended discharge break, the maximum speed the pump attains is 152% of normal operating speed. The calculated torque imposed on the impeller follows the same trend as the speed, with the maximum value occurring following the double-ended discharge break.

The need for a disengaging device to prevent motor overspeed has been evaluated. In view of the fact that the maximum anticipated pump speed is well within the safe operating limits for all rotating parts, a means to disengage the motor from the pump is not necessary.

Analysis of the LOCA leading to pump overspeed incidents indicates no probability of significant structural damage to the pumps.

The calculational model used to obtain the bursting speeds is described as follows:

Outer radius of flywheel, r = 37.5 in.

Density of A 533 steel ρ = 0.283/386 lb sec^2/in^4

Angular velocity w = 2 πN/60 rad/sec

Poisson ratio μ = 0.3

The maximum stress, S, in a solid disk of uniform thickness (Timoshenko, Theory of Elasticity, p. 71)

$$S = \frac{3+\mu}{8}\rho w 2_r 2$$

S = 0.00465N2

If the entire plate was in a uniform stress field equal to this maximum value, the stress would be related to the toughness, K_{IC}, by the approximate formula,

$$K_{IC} = S\sqrt{\pi C}$$

where C is the distance from the crack tip to the center of the disk. If a conservative value of 100,000 psi $\sqrt{in}$ is assumed for the room temperature fracture toughness for this steel, a relationship between C and N can be determined.

$$N = \left(\frac{K_{IC}}{\sqrt{\pi C\ (0.00465)}}\right)^{1/2}$$

N = 3500 C $^{-1/4}$

4.1.3.4 Reactor Coolant Piping

The RCS piping consists of two loops which connect the SGs to the reactor vessel. Each loop can be considered to consist of a 42" ID "hot leg" pipe connecting the reactor vessel outlet to the SG inlet, and 30" ID piping connecting the SG outlets to the RCPs and the coolant pumps to the reactor vessel inlets. A 12" schedule 160 surge line connects one loop hot leg to the pressurizer. Design parameters for the reactor coolant piping are given in Table 4-6.

The reactor coolant piping was designed and fabricated in accordance with the rules and procedures of ANSI B31.7, Class I. The anticipated transients listed in Section 4.1.1 form the basis for the required fatigue analysis to ensure an adequate usage factor.

The reactor coolant piping is fabricated from SA516, GR70 carbon steel mill clad internally with type 304L stainless steel. A minimum clad thickness of 1/8" is maintained. The 12" surge line is fabricated from ASTM A351, Gr CF8M alloy steel.

Thermal sleeves are installed in the surge nozzle, charging nozzles and shutdown cooling inlet nozzle to reduce thermal shock effects from auxiliary systems. Clad sections of piping were fitted, where necessary, with safe ends for field welding to stainless steel components.

The piping was shop fabricated and shop welded into subassemblies to the greatest extent practicable to minimize the amount of field welding. Fabrication of piping and subassemblies was done by shop personnel experienced in making large heavy wall welds. Welding procedures and operations met the requirements of ASME B&PV Code, Section IX. All welds were 100% radiographed and liquid-penetrant tested and all reactor coolant piping penetrations were attached in accordance with the requirements of ANSI B31.7. Cleanliness standards consistent with nuclear service were maintained during fabrication and erection.

4.1.3.5 Pressurizer

The pressurizer maintains RCS operating pressure and compensates for changes in coolant volume during load changes. Table 4-7 gives design parameters for the pressurizer. The pressurizer is shown in Figure 4-8.

Pressure is maintained by controlling the temperature of the saturated liquid volume in the pressurizer. At full load nominal conditions, slightly more than one-half the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. A number of pressurizer heaters are operated continuously to offset the heat losses and the continuous minimum spray, thereby maintaining the steam and water in thermal equilibrium at the saturation temperature corresponding to the desired system pressure.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 4-9. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level causing the reactor system pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value and additional charging pumps in the CVCS are automatically started to add coolant to the system and restore pressurizer level.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program (Figure 4-9). The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer (in-surge), compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the RCP discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a power-dependent function. A high level error signal produced by an in-surge causes the letdown control valves to open, releasing coolant to the CVCS and restoring the pressurizer to the prescribed level.

Small pressure and coolant volume variations are accommodated by the steam volume which absorbs flow into the pressurizer, and by the water volume which allows flow out of the pressurizer. The total volume of the pressurizer is determined by consideration of the following factors:

a. Sufficient water volume is necessary to prevent draining the pressurizer as the result of a reactor trip or a loss-of-load incident. In order to preclude the initiation of safety injection and of automatic injection of concentrated boric acid by the charging pumps, the pressurizer is designed so that the minimum pressure observed during such transients is above the setpoint of the safety injection actuation signal;
b. The heaters should not be uncovered by the out-surge following load decreases; 10% step decrease, and 5% per minute ramp decrease;
c. The steam volume should be sufficient to yield acceptable pressure response to normal system volume changes during load change transients;
d. The water volume should be minimized to reduce the energy release and resultant containment pressure during a LOCA;
e. The steam volume should be sufficient to accept the reactor coolant in-surge resulting from loss of load or loss of feedwater without the water level reaching the safety and PORV nozzles; and
f. During load following transients, the total coolant volume change and associated charging and letdown flows should be kept as small as practical and be compatible with the capacities of the volume control tank, charging pumps, and letdown control valves in the CVCS.

The following design cyclic transients, which include conservative estimates of the operational requirements for the components discussed in Section 4.1.3, were used in the fatigue analyses required by the applicable codes listed in Table 4-9, and in accordance with the load combinations noted in Table 4-8:

a. 500 heatup and 500 cooldown[a] cycles during the system's 40-year design life at a heating rate of 100°F/hr and a cooling rate of 200°F/hr between the temperature limits of 70°F and 653°F;
b. 15,000 power change cycles (increasing load) over the range of 15% to 100% of full load with a ramp load change of 5% of full load per minute and 15,000 power change cycles (decreasing load) over the range of 100% to 15% of full load with a ramp load change of 5% of full load per minute;
c. 2,000 cycles of 10% of full load step power changes (increasing load) from 10% to 90% of full power, and 2,000 cycles of 10% of full load step power changes (decreasing load) from 100% to 20% of full load;
d. 10^6 cycles of normal variations[b] of ± 100 psi and ± 7°F;
e. 400 reactor trips from 100% power;
f. 320 cycles of leak testing[c] at 2500 psia and a temperature between 100°F and 400°F;
g. 10 cycles of hydrostatic testing the RCS at 3125 psia and a temperature between 100°F and 400°F;
h. 40 cycles of total loss of reactor coolant flow when at 100% power[d];
i. 40 cycles of loss of turbine load from 100% power with a direct reactor trip[d]; and
j. 5 cycles of loss of secondary system pressure[d].

NOTES:

[a] Cooldowns may include the introduction of 40°F spray water at the rate of 132 gpm with the temperature of the pressurizer and spray nozzle at 443°F and the pressure at 395 psia.

[b] During normal pressure and temperature variations, as the pressure increases above 2300 psia, the spray nozzle will be subjected to an instantaneous increase in spray flow from 1.5 gpm to 375 gpm, with the fluid temperature increasing from 500°F to 550°F in 7 seconds. Fluid temperature remains at 550°F and fluid flow at 375 gpm for 5 minutes. Then the flow instantaneously drops to 1.5 gpm and the fluid temperature decreases to 500°F at the rate of 2°F/minute.

[c] Performed in conjunction with heatup.

[d] These transients specify abnormal conditions. Their effect should be evaluated in conjunction with the other transients specified. The abnormal conditions do not form the basis for Code Design of the Vessel.

To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature. A representation of this relationship is shown in Figure 4-10. Actual response is determined by the calibration of the reactor regulating system. High or low water level error signals result in the control actions shown in Figure 4-11 and described above.

The pressurizer heaters are single-unit, direct-immersion heaters which protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown. Approximately 20% of the heaters are connected to proportional controllers which adjust the heat input as required to account for steady state losses and to maintain the desired steam pressure in the pressurizer. If pressure falls below the proportional band, all of the backup heaters are energized.

The remaining backup heaters are connected to on-off controllers. These heaters are normally deenergized but are turned on by a low pressurizer pressure signal or high level error signal. This latter feature is provided since load increases result in an insurge of relatively cold coolant into the pressurizer, decreasing the temperature of the water volume. The action of the CVCS in restoring the level results in a pressure undershoot below the desired operating pressure. To minimize the pressure undershoot, the backup heaters are energized earlier in the transient, contributing more heat to the water before the low pressure setting is reached. A low-low pressurizer level signal deenergizes all heaters to prevent heater burnout.

The pressurizer spray is supplied from each of the RCP discharges on one loop to the pressurizer spray nozzle. Automatic spray control valves control the amount of spray as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the PORVs during normal load following transients. A small continuous flow is maintained through the spray lines at all times to keep the spray lines and the surge line warm, reducing thermal shock during plant transients. This continuous flow also aids in keeping the chemistry and boric acid concentration of the pressurizer water equal to that of the coolant in the heat transfer loops.

The pressurizer spray system is designed for the transients listed for the pressurizer. In addition, operating experience has discovered the existence of thermal stratification in the pressurizer spray piping under various plant conditions. In particular, during plant cooldowns with the initiation of auxiliary spray, the pressurizer spray piping can see a large number of thermal shock transient cycles. See Reference 38 for further details on the conditions and allowed number of occurrences of these and other transients that affect the pressurizer spray piping.

An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the RCPs are shut down. The pressurizer specification requires the design to be suitable for 500 cycles of auxiliary spray at 40°F and 395 psia. These conditions represent a conservative maximum for analytical purposes. The results of the design analysis indicate an overall usage factor of less than 0.5 for the spray connections. The number of auxiliary spray occurrences during the lifetime of the plant is expected to be much less than 500. However, it is planned to record each operation of auxiliary spray during plant cooldown and review the total periodically. If the total approaches 500 during the plant lifetime, a detailed review of the fatigue history of the spray connections will be conducted and appropriate action taken.

The auxiliary spray system is designed for the transients listed for the pressurizer. In addition, the auxiliary spray system fatigue analysis considers other transients, including actuations of auxiliary spray at various RCS temperatures and plant operating modes. See Reference 39 for further details, including the allowed number of auxiliary spray actuations and the assumed operating conditions of these transients.

In the event of an abnormal transient which causes a sustained increase in pressurizer pressure, at a rate exceeding the control capacity of the spray, a high pressure trip level will be reached. This signal trips the reactor and opens the two PORVs. The steam discharged by the relief valves is piped to the quench tank where it is condensed. In accordance with ASME B&PV Code, Section III, the RCS is protected from overpressure by two spring-loaded safety valves. The discharge from the safety valves is also piped to the quench tank.

The pressurizer is supported by a cylindrical skirt (SA 516, Gr70) welded to the lower head. The skirt is analyzed to ASME B&PV Code, Section III. Since the pressurizer surge line has sufficient flexibility, no provisions are made for horizontal movement and the skirt is bolted rigidly to the floor.

The pressurizer was designed and fabricated in accordance with ASME B&PV Code, Section III, Class A. The pressurizer is constructed of A-533, Grade B, Class 1 steel plate. The interior surface of the cylindrical shell and upper head is clad with weld deposited stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell. Safe ends were provided on the pressurizer nozzles as required to facilitate field welds to the connecting piping.

In May 1989, leakage caused by primary water stress corrosion cracking (PWSCC) was discovered in the Ni-Cr-Fe Alloy 600 heater sleeves of the Unit 2 pressurizer. In 1990, modifications were completed on all of the 120 Calvert Cliffs Unit 2 pressurizer heater sleeves. One sleeve location was bored out for testing and plugged with a Ni-Cr-Fe Alloy 690 plug in a Ni-Cr-Fe Alloy 690 outer sleeve. Location N-3 was plugged in 2011. The remaining 118 heater sleeve locations

were replaced with a dual sleeve (inner and outer) design also using Alloy 690 material. The replacement outer sleeves were welded to the original Ni-Cr-Fe Alloy clad and welded to Ni-Cr-Fe Alloy weld pad buildups made on the outside of the pressurizer lower head. Inner sleeves were installed through the outer sleeves and welded to the outer sleeve to return the heater geometry to its original configuration.

The PWSCC-susceptible portions of the heater sleeves of Unit 1 were nickel-plated in 1994. The decision to do this was based on the PWSCC damage which occurred on the Calvert Cliffs Unit 2 pressurizer heater sleeves and other PWSCC-related Ni-Cr-Fe Alloy 600 failures at other plants. Plating was performed to eliminate exposure of the Ni-Cr-Fe heater sleeve Alloy 600 material to the primary reactor coolant, thus eliminating PWSCC occurrence. The nickel-plating process has been used successfully to reduce PWSCC failures on SG tubes in nuclear power plants in Europe.

Also in 1994, leakage due to PWSCC damage occurred on Unit 1 pressurizer heater sleeves at penetrations B-3 and FF-1. The lower portions of the sleeves were removed from the penetrations and Alloy 690 plugs were installed in their place. The plugs were attached by welding them to the pressurizer lower head outside surface using the SMAW half-bead technique.

Based on leakage which occurred in Unit 1 pressurizer heater at location CC-1, the heater was removed from service and a stainless steel plug was seal-welded at the bottom of the existing heater sleeve. Location B-1 was plugged in 1998.

In 2012 117 out of 120 Unit 1 pressurizer heater penetrations were modified. Three heater sleeve penetrations, FF-1, B-3, and B-1, are plugged and not modified. Location CC-1 was also modified and a new plug installed. Location V-1 was modified and a plug has been installed in place of a heater. The two lower level instrument nozzles were also modified. The heater modification utilizes the half-nozzle approach to modify the pressurizer heater sleeves. A portion of the existing heater sleeve is removed, and a new stainless steel lower sleeve inserted into the penetration. The ambient temperature temper bead (IDTB) welding technique, using the gas tungsten arc welding process with stainless steel filler metal attached the new sleeve to the bore ID, establishing a new pressure boundary weld. The IDTB weld is disassociated from the original heater sleeve. A heater stabilizer insert (or plug at location CC-1 and V-1) was then welded to the lower end of each sleeve.

For the two instrument nozzles located in the bottom head of the pressurizer, the IDTB welding process was used to install the replacement stainless steel nozzles. A socket adaptor was then welded to the lower end of each replacement nozzle.

In 2014 Unit 1 pressurizer heaters L3 and BB3 were electrically disconnected due to a grounding issue.

In 2016 Unit 1 pressurizer heater N4 was electrically disconnected due to a grounding issue.

4.1.3.6 Reactor Coolant Vent System

Vents were added to the reactor vessel and to the pressurizer head in response to the TMI Lessons Learned Report, NUREG 0737, Item II.B.1. These vents are intended to provide a means of releasing non-condensable gases from the RCS during natural circulation. The pressurizer vent line valves are used as a backup

to main and auxiliary spray to depressurize the RCS during a SG tube rupture. The original design of the Calvert Cliffs plant allowed venting of the RCS only during cold shutdown. The vent modifications provide electrically-operated solenoid valves, powered from emergency busses, that are operated from the Control Room. The reactor vessel and the pressurizer each have two of these valves in series, which fail closed (power-to-open). The reactor vessel vent line valves are installed in previously existing lines; the pressurizer vent line valves are installed in a line that was added as an additional branch off the pressurizer vapor sample line. The two vent lines join to a common line that leads to the quench tank. The common line contains a temperature element and alarm which is used for valve seat leak detection and flow indication. Detailed information on the solenoid vent valves is provided in Table 4-9A.

4.1.4 DESIGN EVALUATION

4.1.4.1 Codes

The codes adhered to and component classifications are listed in Table 4-9.

The impact properties of all materials which form a part of the pressure boundary meet the requirements of ASME B&PV Code, Section III, Paragraph N-330, at a temperature of 40°F or less. The replacement RVCH complies with 10 CFR Part 50, Appendix G, Fracture Toughness Requirements, and ASME B&PV Code, Section III, Paragraph NB-2300.

All cast valves and components which form the RCS boundary (Class I) were measured in the factory or at the field prior to plant operation to assure that their wall thickness was equal to or greater than the manufacturer's calculated minimum wall thickness.

4.1.4.2 Materials Compatibility

4.1.4.2.1 Materials Exposed To Coolant

Materials exposed to reactor coolant have shown satisfactory performance in operating reactor plants. A listing of materials is given in Table 4-10.

4.1.4.2.2 Insulation

Piping and equipment are insulated with a mass-type and reflective-type material compatible with the temperature and functions involved.

A removable metal reflective-type thermal insulation is provided on the flange stud area of the reactor vessel closure head to permit access to the head studs for removal and reinstallation of the head. Removable insulation is provided on weld areas of the RCS subject to inservice inspection. Nonremovable (reflective type) metal thermal insulation is provided on the reactor cavity wall.

The thickness of insulation is such that the exterior surface temperature is not higher than approximately 55°F above the maximum containment ambient (120°F). All insulation support attachments are attached prior to final stress relief.

4.1.4.2.3 Coolant Chemistry

Reactor coolant chemistry is maintained within the limits of the reactor coolant chemistry program which is based on the guidance provided by the reactor vendor and the Electric Power Research Institute. Control is accomplished by limiting the input of potentially deleterious materials to the reactor coolant and by operation of the CVCS. High purity demineralized water is used for makeup and strict controls are placed on the purity of chemical additives and ion exchange resins. The CVCS can be used to further purify the reactor coolant. Periodic sampling and analysis of the reactor coolant is performed to ensure that positive control of the reactor coolant is maintained. Approved surveillance and specification procedures require routine monitoring and control of parameters that affect RCS chemistry (i.e., chloride, fluoride, sulfate, lithium, and dissolved hydrogen).

Chloride:

Chloride, in combination with oxygen and high temperature, induces stress corrosion cracking (SCC) of austenitic stainless steels. Therefore, it is important to maintain RCS chloride concentrations below the threshold values of SCC.

Fluoride:

The fluoride concentration in the reactor coolant system is limited to prevent fluoride attack of Zircaloy-4 in non-boiling systems and prevent SCC of austenitic stainless steels.

Sulfate:

The sulfate concentration in the RCS is monitored as an indicator of water purity. Additionally, reduced-sulfur bearing species have been implicated in SCC of austenitic stainless steels and in primary side cracking of Alloy 600 during cold shutdown conditions. Sulfate is the fully oxidized form of the aggressive species. Therefore, the sulfate concentration in the RCS is limited.

Lithium:

The lithium concentration in the RCS is monitored and controlled. Lithium directly effects primary coolant pH. Primary coolant pH is controlled to maintain the integrity of the RCS and minimize out-of-core radiation levels.

Dissolved Hydrogen:

Dissolved hydrogen is added to the RCS to scavenge oxygen and suppress the radiolysis of water during power operations. Hydrazine may be used to scavenge oxygen when there is insufficient gamma flux and the RCS temperature is below 250°F.

4.1.4.3 Welding Procedures

Sensitization of stainless steel occurs when unstabilized 300 Series stainless material is held in the temperature range of 900-1400°F for sufficient time to form a continuous network of chromium carbide precipitates. Sensitization occurs after approximately 100 hours at 900°F, as compared to one hour at 1400°F. Stabilized

300 Series stainless material avoids continuity of chromium carbide precipitates in the grain boundaries by careful control of metal chemistry.

No furnace sensitized stainless steels are employed in the RCS pressure boundary. Sensitization is precluded from NSSSs through materials selection and control of all welding and heat treating procedures.

Major portions of the RCS boundary in CEs nuclear plants are formed by carbon steels and a high nickel base alloy. None of these materials is susceptible to furnace sensitization (a continuous network of iron-chromium grain boundary carbides) in the sense of unstabilized 300 Series stainless steels. All internal carbon steel surfaces are weld-deposit or roll-on clad with Inconel or stainless steel, to preclude excessive corrosion product release.

Internal surfaces of the reactor vessel pressurizer and SG primary head are overlaid with 308 weld deposited metal (309L/308L cladding for replacement RVCH). Weld metal composition is carefully controlled to overcome interface dilution and promote an austeno-ferritic duplex structure. Therefore, during the stress relief heat treatment (1150°F ± 25°F) required by the ASME code for the pressure vessel, a continuous network of chromium carbide precipitates is not formed in the 308 weld overlay even though this material has been subjected to a furnace heat treatment. The delta ferrite acts as a carbon sink and prevents continuity of carbide precipitates.

The primary head of the RSGs is clad with Stainless Steel Type 308L and 309L weld-deposited metal. Use of these materials is in compliance with NRC RG 1.44, "Control of the Use of Stainless Steels." Since the ferrite content of Stainless Steel Type 308L and 309L weld-deposited metal is 5% or more (RSG specification is 5-15%), these materials are exempt from corrosion testing (ASTM A262 Pr. A/E) for verification of freedom from intergranular attack after exposure to sensitizing temperatures in the range of 800-1500°F. Further, this level of ferrite content allows for preferential carbide precipitation at the ferrite-austenite interfaces, which in turn prevents carbide precipitation leading to sensitization of the austenite grain boundaries. In addition to the benefit of ferrite in preventing intergranular attack, the stainless steel cladding utilized is low carbon "L" grade (0.03% max [C]), which will reduce carbide precipitation, and therefore susceptibility to intergranular corrosion.

Extensive testing has confirmed that, properly formulated (a duplex structure), 308 weld deposited metal does not form a continuous carbide network within grain boundaries even following a typical vessel post weld heat treatment (viz, 1150°F for 20 hours). Hence, the material is immune to intergranular corrosion.

All other type 300 Series stainless steel used either is not subjected to a furnace sensitization heat treatment or, as is the case of cladding on the primary piping, is of type 304L (low carbon) composition and is not susceptible to the formation of continuous chromium carbide grain boundary networks.

The primary system coolant pump casing is CF8M (Cast 316) which is a duplex material. The casting is solution annealed after welding; hence, this component will not have a sensitized structure.

Because carbon steel piping is used in the RCS, carbon steel safe ends are required on the reactor vessel and SG large nozzles. Where small diameter solid stainless pipes are employed (or in the instance of welding the coolant pump

casing to carbon steel), an Inconel-18 weld deposit is built up on the nozzle prior to vessel post-weld heat treatment. Thereafter, an annealed stainless steel safe end is shop-welded to the Inconel-182 buildup using 182 filler metal.

In joining small diameter annealed solid stainless steel piping, as is used in the pressurizer surge line, charging pump lines and safety injection systems, some carbide precipitation will occur as a result of welding. However, the precipitation that occurs in the weld heat affected zone (HAZ) does not sensitize the material in the context of forming continuous grain boundary carbide precipitates. Typical samples from such welds pass the industry accepted standard for intergranular corrosion susceptibility (i.e., Strauss Test - ASTM A393). Metallographic examination of such welds reveals that only discontinuous grain boundary precipitates are present.

The following four welding processes are used to weld stainless steel in CEs NSSSs. Welding processes are performed in accordance with written procedures, as provided in the Quality Assurance Topical Report. Nitrogen will not be used in lieu of argon or helium gas as a purge gas in the welding process.

- Shielded Metal Arc (SMA)
- Gas Tungsten Arc (GTA)
- Gas Metal Arc (GMA)
- Submerged Arc (SA)

a. Shielded metal arc (SMA) is a process wherein coalescence is produced by heating with an electric arc drawn between a flux covered metal electrode and the work.
b. In the GTA, coalescence is produced by heating with an electric arc drawn between a tungsten electrode and the work. Filler metal, if required, is added by feeding a bare metal rod or wire into the weld pool. Shielding of the weld is obtained from an inert gas mixture.
c. With GMA, coalescence is effected by heating with an arc drawn between a continuous feed wire electrode and the work. Shielding of the weld is obtained from an externally supplied inert mixture.
d. Submerged arc (SA) produces coalescence by heating with an arc or arcs drawn between a bare metal (filler) electrode or electrodes and the work. The arc and weld are shielded by a blanket of granular fusible flux.

The procedures used in welding nozzles with CE manufacturing facilities are generally as follows. For nozzles with stainless steel safe ends, the safe ends are not attached until after final stress relief. The stainless steel safe end is welded to Inconel buttering on the alloy steel and the weld made using Inconel weld wire. With the above procedure, furnace sensitizing of stainless steel is precluded.

During manufacture of the core structures, various parts of the core structure are tested for sensitization using the Strauss Test (ASTM A393). Test specimens consist of: (1) mock ups of various welded joints, and (2) monitoring specimens included in any heat treatment of various components. None of the specimens tested in conjunction with fabrication of reactor vessel internals for previous CE plants have failed the Strauss Test.

The typical weld heat input with the above processes as used by CE to join 300 Series stainless steel varies from 6000 joules per inch (GTA) to 96,000 joules per inch (SA). To avoid weld HAZ sensitization, CE limits the interpass temperature on multipass welds in stainless steels to 350°F maximum. The combination of

normal heat input using the above welding procedures and control of interpass temperature assures minimum carbide precipitation in the weld HAZ. Samples from large welds have been examined in the laboratory and none have failed the Strauss Test.

In field welding operations, Bechtel uses welding procedures that limit heat input to the weld areas, and thus precludes the possibility of sensitization of austenitic stainless steels. Most of the welding employed is of the manual SMA process; a minor amount of GMA welding is also used. Neither one of these processes would be classified as an excessively high heat input welding procedure.

Further precautions employed to preclude field sensitization of austenitic stainless steels consist of:

- Preheat and interpass temperatures are limited to 350°F maximum.
- Controlled welding sequence is used to minimize heat input.
- The practice of block welding is prohibited.
- Post-weld heat treatment is prohibited on equipment and/or parts that are completely or partially fabricated of austenitic stainless steel.
- Application of heat to correct weld distortions resulting in dimensional deviations in equipment and/or parts fabricated of austenitic stainless steel is prohibited.

In preparing for and engineering the field welding requirements, close liaison was maintained between Bechtel and CE. Detailed welding parameters prepared by CE were submitted to Bechtel. Based on this information and its own welding practices, Bechtel prepared detailed welding procedures which were then submitted to CE for review and mutual concurrence and approval before they were adopted for use. Bechtel quality assurance procedures for field welding are discussed in Appendix 5B.

Nitrogen-enhanced stainless steel, type 304 or 316, was not used in the fabrication of pressure containing stainless steel parts of the reactor coolant pressure boundary or those load-bearing stainless steel members which are vital to the structural integrity of the reactor vessel core. The process of electroslag welding was not used in the fabrication of any components within the reactor coolant boundary. All B31.7 Class I and Class II piping welds in the system have been inspected using radiographic techniques as required by applicable codes. In addition, welds that fall within the scope of ASME B&PV Code, Section XI will be inspected as shown in Technical Specifications. The selection of welds to be inspected on the pre-service and subsequently on inservice are based on ASME B&PV Code, Section XI.

4.1.4.4 Seismic Design

The NSSS is designed to withstand the loads imposed by the maximum hypothetical accident and the maximum seismic disturbance without loss of functions required for reactor shutdown and emergency core cooling. The method of combining stresses produced by these simultaneous conditions is described in Section 4.1.1.

4.1.4.5 Prevention Of Brittle Fracture

Ferritic materials, such as carbon steels and low-alloy steels, undergo a ductile-to-brittle transition. The temperature at which this transition occurs is a function of many variables including material composition, processing, and neutron irradiation.

To ensure that brittle failure will not occur special attention must be given to materials selection, fabrication procedure, and operating procedures and the effect of irradiation upon the material properties. Gamma heating in the vessel wall accounts for only 20°F temperature rise at the outer vessel wall at the core midplane. A complete and thorough stress analysis of RCS components which are part of the pressure boundary establishes the stress distributions. These analyses include the effects of geometric shapes and surface stress concentrations. Following fabrication of the components, a post-weld heat treatment reduces the effect of residual stresses, the magnitudes of which would otherwise be unknown.

4.1.4.5.1 Material NDTT

The original materials that form the pressure boundary of the RCS have impact properties which meet the requirements of ASME B&PV Code, 1965, N67, Section III, Paragraph N-330, at 40°F or less, except the CEDMs which meet these requirements at 75°F or less. Materials for several replacement components were designed and fabricated to later Code editions. Refer to Table 4-9 for specific code requirements.

For the RSGs and the replacement RVCH, the RT_{NDT} for each pressure boundary plate, forging, and weld is equal to or less than 0°F.

4.1.4.5.2 NDTT Shift During Irradiation

Figure 4-12 presents the CE design curve for NDTT increase as a function of integrated neutron exposure (E>I Mev) for A-302-B and A 533 B steel irradiated at 550°F. This curve forms an outer envelope with respect to all known irradiation data available in April 1970. The data was compiled from the reference documents listed in Table 4-11. This design curve provides a conservative prediction for the change in NDTT with irradiation.

4.1.4.5.3 NDTT Determination

The reactor vessel is designed and fabricated in such a manner that significant operational limitations will not be imposed on the RCS resulting from shifts in reactor vessel NDTT. The vessel material monitoring program will be conducted within the guidelines of ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors." The pre-irradiated NDTT of the baseplate material has been established using drop weight tests in accordance with ASTM E208 and correlations made with Charpy impact specimen tests conducted in accordance with ASTM A370. This correlation, along with the Charpy impact specimens irradiated in the surveillance program, was used to monitor vessel material NDTTs.

For the pre-irradiated Charpy tests, a minimum of three specimens of each material were tested at any one temperature. Tests were performed at a sufficient number of different temperatures to establish the energy-temperature curve.

The test material used in establishing the unirradiated NDTT of the base metal was obtained from (1/4) T (where T is plate thickness) locations of sections of the plate used in the intermediate or lower shell courses. The thermal history of the plate from which the specimens were taken was representative of that of the shell plating. The impact properties at this

location were considered to be representative of the material through the plate. The properties at the (1/4) T location were used to establish the initial minimum operating temperature and form the basis for the predicted minimum operating temperature after irradiation.

The material toughness test requirements were as follows:

a. For the Reactor Vessel:

   Carbon and low-alloy steel materials which form a part of the pressure boundary shall meet the requirements of ASME B&PV Code, 1965, N67, Section III, Paragraph N-330 at a temperature of +40°F. Refer to Table 4-9 for specific code requirements. It shall be an objective that the materials meet this requirement at +10°F. Charpy tests shall be performed and the results used to plot a transition curve of impact values vs. temperature extending from fully brittle to fully ductile behavior. The actual NDTT of inlet and outlet nozzles, vessel and head flanges, and shell and head materials shall be determined by drop weight tests per ASTM E208. NDTT will be established by Charpy test. Drop weight tests will be conducted and the results used for information only. The replacement RVCH is made of two forgings - a flange and dome portion. Both forgings meet the ASME B&PV Code (1995 Edition, 1996 Addenda) Section III, NB 2300. The maximum RT_{NDT} is confirmed to be 0°F or less. This has been confirmed by drop weight testing per ASTM E208-91.

b. For the RSG and Pressurizer:

   It shall be an objective that impact properties of all ferritic steel materials forming a part of the pressure boundary shall meet the requirements of ASME B&PV Code, Section III, at a temperature of 0°F for the SG lower assembly, +10°F for the SG steam drum, and +10°F for the pressurizer shell. For the SG steam drum and pressurizer, alternate higher temperature levels up to 40°F may be used only if the material fails at +10°F. Such higher temperature levels, if applicable, shall be determined and documented.

c. For the Reactor Coolant Piping:

   Materials used to fabricate the pipe and fittings shall be specified, examined and tested to satisfy, as a minimum, the requirements of Chapter I-III of ANSI Code for Pressure Piping B31.7, Class 1. Impact properties of carbon steel materials, including welds, shall have a minimum V-notch value of 20 ft/lb (average of three specimens) or 15 ft/lb (any individual specimen) at 40°F. It shall be a design objective that the materials meet this requirement at 10°F. Weld procedure qualifications and weld metal certifications records may serve to demonstrate impact properties of welds.

The maximum NDTT for the reactor vessel as obtained from drop weight tests is: Unit 1 = +10°F, Unit 2 = +30°F. Drop weight tests were conducted only on material used in the reactor vessel. The maximum RT_{NDT} for the replacement RVCH is 0°F or less.

The minimum upper-shelf C_v energy value for the strong direction of the material used in the reactor vessel is: Unit 1 = 101 ft/lb, Unit 2 = 108 ft/lb.

The upper-shelf C_v energy was not determined for the materials used in fabricating the SG, pressurizers, or reactor coolant piping. The data was not obtained for the weak direction in the material used to fabricate the reactor vessels.

The identification and location of the material relating to limiting values listed above is as follows:

a) For Maximum NDTT:

Unit 1 -	reactor vessel outlet nozzles	(forging)
	reactor vessel outlet nozzle extensions	(forging)
Unit 2 -	reactor vessel flange	(forging)

b) For Minimum upper-shelf C_v energy value:

Unit 1 -	reactor vessel outlet nozzle	(forging)
Unit 2 -	reactor vessel outlet nozzle	(forging)

4.1.4.5.4 Radiation Embrittlement of Reactor Pressure Vessel Materials and Fracture Toughness Requirements for Protection Against PTS Events

Neutron irradiation of reactor pressure vessel materials reduces the fracture toughness of these materials over time. The reactor pressure vessel fracture toughness is an important material property that must meet minimum requirements to ensure the reactor pressure vessels are able to withstand pressurized thermal shock (PTS) events.

10 CFR Part 50 Appendices G and H require monitoring changes in fracture toughness of reactor pressure vessel materials induced by neutron irradiation. Regulatory Guide 1.99 Revision 2 provides a method acceptable to the Nuclear Regulatory Commission for calculating the changes in the fracture toughness of the reactor pressure vessel materials. The current PTS rule contained in 10 CFR 50.61, Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events, incorporates the requirements of RG 1.99, Revision 2.

Both the PTS rule and RG 1.99 Revision 2 require the computation of a value for each reactor pressure vessel material representing the effect of neutron embrittlement on that material. This value is given in terms of the reference temperature (RT_{NDT}). The value from RG 1.99, Revision 2 is called the ART and is determined by the following equation:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

The Initial RT_{NDT} is the initial reference temperature for the unirradiated material as defined by ASME B&PV Code, Section III, Paragraph NB-2331. Tables 4-11A and 4-11B provide the initial RT_{NDT} and other material properties for the reactor vessel beltline.

The shift in the reference temperature caused by irradiation (ΔRT_{NDT}) is the product of a chemistry factor and a fluence factor. Regulatory Guide 1.99, Revision 2 provides two ways to determine the chemistry factor.

a. By using tabular values given in RG 1.99, Revision 2; or
b. By a calculation using a "Least Squares" fit to a minimum of two sets of credible surveillance data.

Margin is a quantity given in RG 1.99, Revision 2 that is to be added to yield conservative upper-bound values of the ART.

The PTS Rule provides the same methodology as RG 1.99, Revision 2, but uses slightly different terminology. The ART is equivalent to RT_{PTS} which is given by:

$$RT_{PTS} = \text{Initial } RT_{NDT} + \Delta RT_{PTS} + \text{Margin}$$

where ΔRT_{PTS} is computed identically to ΔRT_{NDT}.

Calvert Cliffs Nuclear Power Plant (CCNPP) has submitted projected values of RT_{PTS} for Calvert Cliffs Units 1 and 2 reactor vessel beltline materials, calculated in accordance with the procedures given in 10 CFR 50.61(b)(2) (References 1, 2, and 3).

Fluence values used in the final RT_{PTS} calculations are based on analyses performed by Babcock and Wilcox Nuclear Technologies (BWNT) in 1993 and 1994 as part of analyzing the 97° capsule removed from each reactor vessel in those years. Those analyses established the actual neutron fluence in the Unit 1 reactor vessel through the end of Cycle 10, and in the Unit 2 reactor vessel through the end of Cycle 9.

The Unit 1 fluence calculations were performed by ABB/Combustion Engineering under subcontract to BWNT, using the two-dimensional neutron transport code DOT-4 (Reference 4), with the 22-group energy structure of the CASK (Reference 5) transport cross-section library. The reaction cross sections were calculated using the SAND (Reference 6) computer code with the DOSDAM (Reference 7) cross-section library, which is based on ENDF/B-V data. The Unit 2 fluence calculations were performed by BWNT also using DOT-4 with the 47-group energy structure of the BUGLE (Reference 8) transport cross-section library. The reaction cross-sections were taken directly from ENDF/B-V.

The final neutron fluence projections for both Units 1 and 2 reflected the 24 month fuel cycle core pattern similar to that used in Unit 1 Cycle 11, and for Unit 1 incorporated the extensive flux reduction measures which were taken during Cycles 10, 11, and 12.

For each beltline material, CCNPP calculated RT_{PTS} values for the projected fluence at the end of the current 40-year license (2014 for Unit 1 and 2016 for Unit 2), and for the end of a renewed operating license (20 years beyond the current operating license).

The projected values of RT_{PTS} do not exceed the screening criteria for any of the Unit 1 or Unit 2 beltline materials. Note, however, that the projections for Unit 1 weld 2-203 A/B/C are based on surveillance data which was obtained from Duke Power Company's William B. McGuire Nuclear Generating Station, Unit 1. The Baltimore Gas and Electric Company November 29, 1993 submittal demonstrated the equivalence of the conditions between Calvert Cliffs Unit 1 and McGuire Unit 1. This now permits CCNPP to use the surveillance results obtained from McGuire Unit 1 to calculate a RT_{PTS} using the "least squares" method described above. Similar weld material has also been placed in a supplemental surveillance capsule which was placed in the Unit 1 reactor

vessel in a location left vacant when the first surveillance capsule was removed (263° position). Calvert Cliffs Nuclear Power Plant will continue to update the RT_{PTS} projections for Calvert Cliffs as new data becomes available.

4.1.4.6 Reactor Vessel Thermal Shock

An analysis of the thermal stresses produced in the reactor vessel wall due to the operation of the safety injection system has been performed. The analysis has been reported in a CE report "Thermal Shock Analysis on Reactor Vessels Due to Emergency Core Cooling System Operation," A-68-9-1, and was submitted for the record on Docket No. 50-309, Maine Yankee Atomic Power Station.

Experiments on the heat transfer coefficient involved instrumenting a two-foot square steel block six-inches thick, quenching it from a representative reactor vessel temperature and calculating a suitable average heat transfer coefficient from the test data. This work was reported in a CE report, "Experimented Determination of Limiting Heat Transfer Coefficients during the Quenching of Thick Steel Plates in Water," A-68-10-1, which was submitted to the AEC and is on file in the Public Document Room.

The stress near the tip of axial and circumferential vessel cracks of various depths has been determined by the finite element method. This work was reported by a CE report, "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel during Emergency Core Cooling," A-70-19-2, January 1970, and is part of the public record.

These reports substantiate the analytical conclusion that a vessel failure will not occur due to Emergency Core Cooling System (ECCS) operation. An acute crack, even if formed, will not propagate.

4.1.5 TESTS AND INSPECTIONS

4.1.5.1 General

Shop inspection and tests of all major components was performed at the vendor's plant prior to shipment. An inspection at the site was performed to assure that no damage has occurred in transit. Testing of the RCS was performed at the site upon completion of plant construction. These tests included hydrostatic tests of all fluid systems. A complete visual inspection of all welds and joints was performed prior to the installation of the insulation. All field welds were volumetrically and inspected in accordance with the requirements of the Codes applicable to the construction of the component.

A hot flow test of the reactor coolant loop up to full power operating pressure and temperature without the core installed was made. The system was checked for vibration and cleanliness. Auxiliary systems were checked for performance (Chapter 13).

4.1.5.2 Surveillance Program

The surveillance program was implemented to monitor the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials. This surveillance program complies with 10 CFR Part 50, Appendix H, and ASTM E185-70. Changes in the impact properties of the material will be evaluated by the comparison of pre-irradiation and post-irradiation Charpy impact test specimens. Changes in mechanical properties will be evaluated by the comparison of pre-irradiation and post-irradiation data from tensile test specimens.

Three metallurgically different materials representative of the pressure vessel will be investigated. These are base metal, weld metal and weld HAZ material. In addition to the materials from the reactor vessel, material from a standard heat of A533, which has been made available through the Heavy Section Steel Technology (HSST) Program, will also be used. This reference material has been fully processed and heat treated and will be used for Charpy impact specimens so that a comparison may be made between the irradiations in various operating power reactors and in experimental reactors. A complete record of the chemical analysis, fabrication history and mechanical properties of all surveillance test materials will be maintained.

The exposure locations and a summary of the specimens at each location is presented in Table 4-12. The pre-irradiation NDTT of each plate in the intermediate and lower vessel shell courses was determined from the drop weight tests and correlated with Charpy impact tests.

Base metal test specimens were fabricated from sections of the shell plate in either the intermediate or the lower shell course which exhibits the highest unirradiated NDTT. All material for base test specimens was cut from the same shell plate.

The material used for the base metal test specimens was adjacent to the test material used for ASME B&PV Code, Section III tests and was at least one plate thickness away from any quenched edge. This material was heat treated to a condition which is representative of the final heat treated condition of the base metal in the completed reactor vessel.

Weld metal and HAZ material was produced by welding together two plate sections from the intermediate or lower shell course of the reactor vessel. All HAZ test material was also fabricated from the plate which exhibits the highest unirradiated NDTT.

The material used for weld metal and HAZ test specimens was adjacent to the test material used for ASME B&PV Code, Section III tests, and was at least one plate thickness from any water-quenched edge. The procedures used for making the shell girth welds in the reactor vessel was followed in the preparation of the weld metal and HAZ test material. The procedures for inspection of the reactor vessel welds will be followed for inspection of the welds in the test materials. The welded plate was heat treated to a condition which is representative of the final heat treated condition of the completed reactor vessel.

The test specimens were contained in six irradiation capsule assemblies. The axial position of the capsules was bisected by the midplane of the core. The circumferential locations include the peak flux regions.

The location of the surveillance capsule assemblies is shown in Figure 4-13. A typical surveillance capsule assembly is shown in Figure 4-14. A typical Charpy impact compartment assembly is shown in Figure 4-15. A typical tensile monitor compartment assembly is shown in Figure 4-16.

Fission threshold detectors (U-238) were inserted into each surveillance capsule to measure the fast neutron flux. Threshold detectors of Ni, Ti, Fe, S, and Cu with known Co content were selected for this application to monitor the fast neutron exposure. Cobalt was included to monitor the thermal neutron exposure.

The selection of threshold detectors was based on the recommendations of ASTM E261, "Method for Measuring Neutron Flux by Radioactive Techniques." Activation of the specimen material was also analyzed to determine the amount of exposure.

The maximum temperature of the encapsulated specimens is monitored by including in the surveillance capsules small pieces of low-melting-point eutectic alloys or pure metals individually sealed in quartz tubes.

The temperature monitors provide an indication of the highest temperature to which the surveillance specimens were exposed but not the time-temperature history or the variance between the time-temperature history of different specimens. These factors, however, will affect the accuracy of the estimated vessel material NDTT to only a small extent.

The periodic analysis of the surveillance samples will permit the monitoring of the neutron radiation effects upon the vessel materials.

Test specimens removed from the surveillance capsules will be tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens will be compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material will then be made. This assessment of the NDTT shift is based on the temperature shift in the average Charpy curves, the average curves being considered representative of the material.

The integrated fast neutron dose (fluence) to the reactor vessel has been calculated using the methods described in Section 4.1.4.5.2. The predicted change in NDTT as a function of vessel fluence is shown in Figure 4-12.

All surveillance capsules were inserted into their designated holders during the final reactor assembly operation. Capsule withdrawal schedules are listed in Table 4-13A (Unit 1) and Table 4-13B (Unit 2). For capsule withdrawal history, refer to References 31, 32, and 33 in Section 4.1.6.

4.1.5.2.1 Supplemental Surveillance and Dosimetry

The 263° location left vacant by the withdrawal of the first surveillance capsule in Unit 1 was employed in a supplemental dosimetry program. The program consisted of installing one set of in-vessel and ex-vessel neutron dosimetry at azimuthally identical locations. This set of dosimetry capsules was installed at the beginning, and removed at the end, of Cycle 9.

During fabrication of the replacement in-vessel supplemental dosimetry capsule, installed in the 263° location in Unit 1 at the start of Cycle 10, an archival weld block comprised of Calvert Cliffs limiting material was located in Duke Power Company's McGuire Unit 1 Surveillance Program. Charpy impact specimens were fabricated from the McGuire archival weld block and from another reactor vessel weld with a similar weld chemistry and were inserted into the replacement capsule. Sufficient Charpy specimens were installed to provide two sets of data. The removal schedule for the in-vessel capsule with the Calvert Cliffs controlling material will be determined based on flux reduction plans and as a need dictates.

An ex-vessel dosimetry capsule was installed at the 263° location at the beginning of Cycle 10. This capsule was removed at the end of Cycle 10 in conjunction with removal of the azimuthally equivalent (not identical) 97° in-vessel surveillance capsule. At that time, it was determined that sufficient data had been obtained from this capsule, and replacement ex-vessel dosimetry was not installed at the end of Cycle 10. The supplemental capsule withdrawal schedule is given in Table 4-13C. Further information can be found in Reference 33 in Section 4.1.6.

4.1.5.3 Non-destructive Tests

Prior to and during fabrication of the reactor vessel, non-destructive tests based upon ASME B&PV Code, Section III were performed on all welds, forgings and plates as follows.

All full penetration pressure-retaining welds were 100% radiographed to the standards of ASME B&PV Code, Section III, Paragraph N-624. Other pressure-retaining welds such as used for the attachment of mechanism housings, vents and instrument housings to the reactor vessel head were inspected by liquid-penetrant tests of the root passes, each one-half inch of weld material or one-third of weld thickness (whichever is less), and the final surface. The full penetration welds on the replacement RVCH were 100% RT inspected per ASME B&PV Code, Section III, Division 1, NB-5000 and ASME B&PV Code, Section V, Article 2. The other pressure retaining welds were PT inspected progressively at 1/2” increments and the final weld surfaces were inspected.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75% of screen height, was based on interpretation of indications causing complete loss of back reflection. Rejection under shear wave inspection was based on indications exceeding in amplitude the indication from a calibration notch whose depth is 3% of the forging thickness, not exceeding 3/8" with a length of 1". All forgings for the replacement RVCH were UT inspected per the requirements of the ASME B&PV Code, Section III, NB-2542.

All forgings were also subjected to magnetic-particle examination or liquid-penetrant testing depending upon the material. Rejection was based on ASME B&PV Code, Section III, Paragraph N626.3 (NB-2545 for the replacement RVCH) for magnetic-particle and Paragraph N627.3 (NB-2546 for the replacement RVCH) for dye-penetrant testing.

Plates were ultrasonically tested using longitudinal ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME B&PV Code, with calibration so that the first back reflection is at least 50% of screen height, was based on defects causing complete loss of back reflection which could not be contained within a circle whose diameter is the greater of three inches or one-half the plate thickness. Two or more defects smaller than described above, which caused a complete loss of back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above.

Non-destructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as frequent calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict

adherence to the non-destructive testing requirements of ASME p87 Code, Section III.

The detection of flaws in irregular geometries was facilitated because most non-destructive testing of the materials was completed while the material was in its simplest form. Non-destructive inspection during fabrication was scheduled so that full penetration welds were capable of being radiographed to the extent required by ASME B&PV Code, Section III.

Each of the vessel studs received one ultrasonic test and one magnetic particle inspection during the manufacturing process.

The ultrasonic test was a radial longitudinal beam inspection, and a discontinuity which caused an indication with a height which exceeded 20% of the height of the adjusted first back reflection was cause for rejection. Any discontinuity which prevented the production of a first back reflection of 50% of the screen height was also cause for rejection.

The magnetic-particle inspection was performed on the finished studs. Linear axially-aligned defects whose lengths are greater than one-inch long and linear nonaxial defects were unacceptable.

Prior to and during fabrication of the components of the RCS, non-destructive testing based upon the requirements of ASME B&PV Code, Section III is used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C will be fabricated to the standards of ASME B&PV Code, Section III, Article 21.

Table 4-14 summarizes the component inspection program during fabrication and construction. Periodic tests and examinations of the RCS have been conducted after startup on a regular basis. For pre-operational and inservice structural surveillance of the RCS, refer to Section 4.0 of the Technical Specifications.

4.1.5.4 Additional Tests

During design and fabrication of the reactor vessel, additional operations beyond the requirements of ASME B&PV Code, Section III were performed by the vendor. Table 4-15 summarizes the additional tests by component.

During the design of the reactor vessel, detailed calculations were performed to assure that the final product would have adequate design margins. A detailed fatigue analysis of the vessel for all design conditions has been performed. In those areas which are not amenable to calculation, stress concentrations have been obtained through the use of photo-elastic models. In addition, CE has performed test programs for the determination and verification of analytical solutions to thermal stress problems. Also, fracture mechanics and brittle fracture evaluations have been performed.

All material used in the reactor vessel was carefully selected and precautions were taken by the vessel fabricator to insure that all material specifications were adhered to. To assure compliance, the quality control staff of CE reviewed the mill test reports and the fabricator's testing procedures.

All welding methods, materials, techniques, and inspections comply with ASME B&PV Code, Sections III and IX. Before fabrication was begun, detailed qualified welding procedures, including methods of joint preparation, together with certified

procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welder and wire by subjection to a complete and thorough testing program in order to insure maximum quality of welded joints.

During the manufacture of the reactor vessel, in addition to the areas covered by ASME B&PV Code, Section III, quality control by the vendor included:

a. preparation of detailed purchase specifications which included cooling rates for test samples;
b. requiring vacuum degassing for all ferritic plates and forgings;
c. specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication;
d. use of written instructions and manufacturing procedures which enabled continual review based on past and current manufacturing experiences;
e. performance of chemical analysis of welding electrodes, welding wire, and materials for automatic welding, thereby providing continuous control over welding materials;
f. the determination of NDTT through use of drop weight testing methods as well as Charpy impact tests; and
g. test programs on fabrication of plates up to 15" thick to provide information about material properties as thickness increases.

Longitudinal wave ultrasonic testing was performed on 100% of all plate material.

Cladding for the reactor vessel is a continuous integral surface of corrosion-resistant material, 5/16" nominal thickness. The detailed procedure used, i.e., type of weld rod, welding position, speed of welding, non-destructive testing requirements, etc., was in compliance with ASME B&PV Code. One hundred percent ultrasonic testing of the reactor vessel cladding has been performed.

Upon completion of all post-weld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with ASME B&PV Code, Section III, Paragraph N-618.

Surveillance of the quality control program was also carried out during the manufacture of the vessel by the Windsor quality control Section of CE and by the applicant with an independent consultant. This work included independent review of radiographs, magnetic-particle tests, ultrasonic tests, and dye-penetrant tests conducted during the manufacture of the vessel. A review of material certifications, and vendor manufacturing and testing procedures was also conducted. Manufacturers' records such as heat-treat logs, personnel qualification files and deviation files were also included in this review.

4.1.5.5 <u>In-Service Inspection</u>

Consideration for probable inservice inspection of the reactor coolant pressure boundary was made in the early stages of plant design. During design development of the containment internal structures, equipment arrangement and system piping design, provisions were made to facilitate in-service inspection in accordance with ASME B&PV Code, Section XI.

Design improvements have been made to the reactor vessel to facilitate inservice inspection. Additional room has been provided between the nozzle piping surrounding concrete to allow inspection of the piping.

Provisions have been made for access to perform the inspections. The general scheme of access is as follows:

a. Closure Head

   Head to Flange Weld, Cladding, and Nozzle Welds - Accessible for inspection with head in laydown position.

b. Reactor Vessel

   Vessel to Flange Weld - Accessible for inspection with head removed.

   Vessel to Nozzle Welds, Longitudinal Welds, Circumferential Welds and Cladding. Available for inspection from the inside with the core barrel removed. Partial inspection of these welds is also possible from the outside.

   Bottom Head Welds - Available for inspection from the inside with core barrel removed. Sufficient room for outside inspection using remote equipment.

c. Reactor Coolant Piping - Removable insulation allows access to butt welds and the required adjacent sections longitudinal welds.

d. Steam Generators and Pressurizer - Removable insulation provides access to welds from the outside. A remote inspection device is used to inspect the cavity which extends upward from the bottom of each SG. Cladding is accessible by removing man ways.

e. Reactor Coolant Pump Casings - Alternate examinations to those required by Section XI have been approved by the NRC as follows:

   1. The pump interior will be inspected to the extent practical should the pump be disassembled for any other reason.
   2. The RCPs shall be hydrostatically tested per the requirements of ASME Code Section XI.
   3. A surface examination of one RCP in each unit shall be performed on the exterior casing weld surface areas by the liquid penetrant method once per interval.
   4. A visual examination of one RCP in each unit shall be performed on the exterior pump case surfaces once per interval.

   Other components are generally arranged with sufficient surrounding space and removable insulation to allow inspection.

Calvert Cliffs Nuclear Power Plant retained CE and Southwest Research Institute to perform the pre-service baseline inspection of the Calvert Cliffs systems in accordance with ASME B&PV Code, Section XI.

It is planned to inspect the bulk of the reactor vessels and nozzles from the inside using a remote inspection device. This device is capable of volumetrically inspecting the longitudinal welds, circumferential welds, vessel to flange weld, nozzle to vessel welds, nozzle to transition piece welds and transition piece to pipe welds. The closure head to flange weld will be inspected.

In some areas, alternatives to the above inspection methods may be employed if they are feasible and considered desirable.

4.1.5.6 Pre-operational Vibration Test Program

An analysis has been performed and the results show that the natural frequency of the Reactor Coolant Loops is between 4 and 5 cps. The RCPs will impose frequencies of 14-15 cps from the shaft rotation and 70-75 cps from the impeller. Therefore, no vibrational problem is anticipated.

Piping and components within the reactor coolant boundary were observed by start-up engineers during testing and startup. The RCPs are equipped with vibration switches to alarm when vibration exceeds 0.0015" double amplitude. The remainder of the systems will be visually inspected. Any visible vibration will be evaluated to determine the resultant stress levels. Should any vibration which results in unacceptable stress levels occur, the condition causing them will be corrected.

The transient conditions under which the systems were inspected for vibrations are as follows:

a. Starting and stopping each RCP under full operating pressure. Running all combinations of RCPs.
b. Starting and stopping pressurizer spray flow.
c. Starting and stopping charging flow. Starting and stopping letdown flow.
d. Starting and stopping safety injection flow with the RCS vented to the quench tank. Open and close the motor operated valves in the path from the safety injection pumps to the coolant loops with the pumps operating. Open and close the motor operated valves isolating the safety injection tanks from the coolant loops.
e. Starting and stopping feedwater flow from the main feed pumps and the auxiliary feed pumps.
f. Starting and stopping steam flow through the turbine stop valves, through the turbine bypass valves and through the atmospheric steam dumps.

4.1.5.7 Boric Acid Corrosion Monitoring Program

A program has been implemented at Calvert Cliffs to monitor for boric acid leakage onto carbon steel components. The program consists of systematic measures to ensure that boric acid corrosion does not lead to an increase in the probability of abnormal leakage, rapidly propagating failure, or gross rupture of the reactor coolant pressure boundary.

Carbon steel components that could be subject to a boric acid environment are examined on a refueling outage basis.

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32. Letter from A. W. Dromerick (NRC) to C. H. Cruse (BGE), dated December 15, 1997, Revision to the Reactor Vessel Surveillance Capsule Withdrawal Schedule for CCNPP

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34. Perrin, J.S. et al., "Calvert Cliffs Unit No. 1 Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule 263," Battelle Columbus Laboratories, December 1980

35. Lowe, A.L., Jr., et al., "Analysis of Capsule 97° Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant Unit No. 1," B&W Nuclear Service Company, BAW-2160, June 1993

36. Norris, E.B., "Reactor Vessel Material Surveillance Program for Calvert Cliffs Unit 2 Analysis of 263° Capsule, " Southwest Research Institute, SwRI-7524, September 1985

37. Lowe, A.L., Jr., et al., "Analysis of Capsule 97° Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant Unit No. 2," B&W Nuclear Service Company, BAW-2199, February 1994

38. Bechtel Specification 6750-M-0310F, "Design Specification for Piping, Valves, and Associated Equipment of the Pressurizer Spray System"

39. Bechtel Specification 6750-M-0310D, "Design Specification for Piping, Valves, and Associated Equipment of the Charging System and Auxiliary Spray System"

40. Babcock & Wilcox, Canada Letter Report BWC-LR-2000-0006, Revision 3, “Reactor Coolant System Structural Evaluation”

41. Westinghouse Letter No. LTR-SGDA-02-76, Revision 0, March 15, 2002, "Review of Calvert Cliffs Steam Generator Manufacturing Records for RT of Secondary Side Head, Shell and Cone Plates"

TABLE 4-1

PRINCIPAL DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM

Design Thermal Power,	
MWt	2737
Btu/hr	9.34×10^9
Design Pressure, psia	2500
Design Temperature (Except Pressurizer), °F	650
Coolant Flow Rate through core, Minimum, gpm	370,000
Cold Leg Temperature, Operating, °F	548
Average Temperature, Maximum, °F	577
Hot Leg Temperature, Maximum, °F	604
Normal Operating Pressure, psia	2250
System Volume, ft^3 (Without Pressurizer)	9576[a]
Pressurizer Water Volume, ft^3	800
Pressurizer Steam Volume, ft^3	700

[a] With no plugged SG tubes.

TABLE 4-2

REACTOR VESSEL PARAMETERS

Design Pressure, psia	2500
Design Temperature, °F	650
Nozzles	
Inlet (4 ea), ID, in.	30
Outlet (2 ea), ID, in.	42
CEDM (61), ID, nominal in.	Inlet 2 8/10, Mid 2 8/10, Out 2 4/10
Instrumentation (8), nominal, in.	Inlet 4 3/4, Out 4 6/10
Dimensions	
Inside Diameter, nominal, in.	172
Overall Height, Including CEDM Nozzles, in.	503 3/4
Height, Vessel Without Head, in.	408 9/16
Wall Thickness, minimum, in.	8 5/8
Upper Head Thickness, minimum, in.	7 1/2 min
Lower Head Thickness, minimum, in.	4 3/8
Cladding Thickness, nominal, in.	5/l6 (0.2 min for replacement RVCH)
Material	
Shell	SA-533 Grade B, Class I Steel
Forgings	A-508 Class 2 (SA-508 Gr3 Cl 1 for the replacement RVCH)
Cladding	Stainless Steel[a]
CEDM Nozzles	Ni-Cr-Fe Alloy
Instrumentation Nozzles	Ni-Cr-Fe Alloy
Dry Weights	
Head, lb	145,900
Vessel, without flow skirt, lb	671,400
Studs, Nuts, and Washers, lb	38,500
TOTAL	855,800

[a] Weld deposited type 308-309 stainless steel composition with type 308 in contact with coolant, 308L/309L for the replacement RVCH.

TABLE 4-3

STEAM GENERATOR PARAMETERS

Number	2
Type	Vertical U-Tube
Number of Tubes	8471
Tube Outside Diameter, in.	0.750
Nozzles, Ports, and Manways	
Primary Inlet Nozzle (1 ea), ID, in.	42
Primary Outlet Nozzle (2 ea), ID, in.	30
Steam Nozzle (1 ea), ID, in.	34
Feedwater Nozzle (1 ea), nominal, in.	16
Instrument Nozzles (14 ea), nominal, in.	1
Secondary Inspection Ports (12 ea), nominal, in.	2.5
Primary Manways (2 ea), ID, in.	18
Secondary Manways (2 ea), ID, in.	16
Secondary Handhole (6 ea), ID, in.	8
Bottom Blowdown (2 ea), nominal, in.	2
Surface Blowdown (1 ea), nominal, in.	2
Auxiliary Feedwater (1 ea) nominal, in.	4
Water Level Nozzle (12 ea), nominal, in.	1
Recirculation Nozzle (1 ea), nominal, in.	3
Thermowell Nozzle (1 ea), nominal, in.	1 NPT
Primary Side Design	
Design Pressure, psia/psig	2500/2485
Design Temperature, °F	650
Design Thermal Power (NSSS), MWt	2737
Coolant Flow (Each), lb/hr	$61x10^6$
Normal Operating Pressure, psia/psig	2250/2235
Coolant Volume, each, ft^3, cold conditions	1641.3
Coolant Volume, each, ft^3, hot conditions	1662.4
Secondary Side Design	
Design Pressure, psia/psig	1015/1000
Design Temperature, °F	550
Normal Operating Steam Pressure, Full Load, psia/psig	888/873[(a)]
Normal Operating Steam Temperature, Full Load, °F	530.4
Blowdown Flow, Full Power Maximum, Both SGs, lb/hr	150,000
Steam Flow (Each), lb/hr	$6.005x10^6/6.015x10^6$
Feedwater Temperature, °F	433.55[(c)]
Dimensions	
Overall Height, Including Support Skirt, in.	749
Upper Shell Outside Diameter, in.	239 - 3/4
Lower Shell Outside Diameter, in.	164 – 5/16
Dry Weight, tons	521.6
Flooded Weight, tons	824.5
Operating Weight, tons	630.9

(a) With no plugged SG tubes.

(c) Feedwater temperature will decrease if feedwater heaters are bypassed in accordance with operating instructions.

TABLE 4-5

REACTOR COOLANT PUMP PARAMETERS

Number	4
Type	Vertical, Limited Leakage Centrifugal
Shaft Seals	Mechanical (4)
Stationary Face	Carbon Granite
Rotating Support Ring	Tungsten Carbide
Rotating Seal Ring	Tungsten Carbide
Design Pressure, psia	2,500
Design Temperature, °F	650
Normal Operating Pressure, psia	2,250
Normal Operating Temperature, °F	548
Design Flow, gpm	81,200
Total Dynamic Head, ft	243
Maximum Flow (one-pump Operating), gpm	120,000
Dry Weight, lb	141,000
Flooded Weight, lb	148,000
Reactor Coolant Volume, ft^3	112
Motor	
Voltage, volts	13,200
Frequency, hz	60
Phases	3
Horsepower/Speed, Hot, hp/rpm	4500/900
Horsepower/Speed, Cold, hp/rpm	6000/900
Instrumentation	
Seal Temperature Detectors	1
Pump Casing Pressure Taps	2
Seal Pressure Taps	3
Seal Pressure Detectors[a]	2
Controlled Bleedoff Flow Detectors	1
Controlled Bleedoff Temperature Detectors	1
Motor Oil Level Detectors	2
Motor Bearing Temperature Detectors	4
Motor Stator Temperature Detectors	2
Vibration Detector	8
Oil Lift Pressure Detector	2
Lubrication Oil Temperature	2
Total Seal Assembly Leakage (Normal and Standby Operation)	
Three Pressure Seals Operating, gpm	1.50
Two Pressure Seals Operating, gpm	1.84
One Pressure Seal Operating, gpm	2.60

[a] Lower seal pressure transmitter and sensing line were removed and capped under FCR 82-190.

TABLE 4-6

REACTOR COOLANT PIPING PARAMETERS

Number of loops	2
Flow per loop, lb/hr	$61x10^6$
Pipe Size	
Reactor outlet, ID, in.	42
Reactor inlet, ID, in.	30
Surge line, nominal, in.	12
Design Pressure, psia	2500
Design Temperature, °F	650
Velocity Hot leg, ft/sec	42
Velocity Cold leg, ft/sec	37

TABLE 4-7

PRESSURIZER PARAMETERS

Design Pressure, psia	2,500
Design Temperature, °F	700
Normal Operating Pressure, psia	2250
Normal Operating Temperature, °F	653
Internal Free Volume, ft^3	1500
Normal Operating Water Volume, ft^3	600-800
Normal Steam Volume, Full Power, ft^3	700-900
Installed Heater Capacity, kW	1400-Unit 1,[c] 1475-Unit 2[a]
Spray Flow, Maximum, gpm	375
Spray Flow, Continuous, gpm	1.5
Nozzles	
Surge Line (1 ea) nominal, in.	12
Safety and Relief Valves (2) ID, in.	4
Spray (1 ea) nominal, in.	4
Heaters (112-Unit 1[c], 118-Unit 2[a]) OD, in.	0.855 (Unit 1)[d][e] 0.875 (Unit 2)[e]
Instrument, Level (4 ea) nominal, in.	1
Temperature (1 ea) nominal, in.	1
Pressure (2 ea) nominal, in.	1
Materials	
Vessel	A-533, Gr B, Class 1
Cladding	Stainless Steel[b] and Ni-Cr-Fe Alloy
Dimensions	
Overall Length, in.	441 3/8
Outside Diameter, in.	106 1/2
Inside Diameter, in.	95 9/16
Cladding Thickness, in. (minimum)	1/8
Dry Weight, Including Heaters, lb	206,000
Flooded Weight, Including Heaters, lb	300,000

(a) Penetration H-3 was plugged during Unit 2 pressurizer heater sleeves replacement in 1989-90; Location N-3 was plugged in 2011. Any further reduction in Unit 2 pressurizer heater capacity shall be compared against the original (120) heater capacity.

(b) Weld-deposited type 308-309 stainless steel composition with type 308 in contact with coolant.

(c) Unit 1 pressurizer heater sleeve at B-3, FF-1 and CC-1 were plugged in 1994; Location B-1 was plugged in 1998 and location V-1 was plugged in 2012. Locations L3 and BB3 were electrically disconnected in 2014. Location N4 was electrically disconnected in 2016. Any further reduction in Unit pressurizer heater capacity shall be compared against the original (120) heater capacity.

(d) Smaller diameter heaters for Unit 1 to allow for nickel plating of heater sleeves.

(e) Tolerances for these dimensions are proprietary.

TABLE 4-8

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS

LOADING COMBINATIONS	PRIMARY STRESS LIMITS Vessels[d]	Piping	Supports
1. Design Loading + Operating Basis Earthquake	$P_M \leq S_M$ $P_B + P_L \leq 1.5\ S_M$	$P_M \leq 1.2\ S_h$ $P_B + P_M \leq 1.2\ S_h$	Working Stress
2. Normal Operating Loadings + Safe Shutdown Earthquake	$P_M \leq S_D$ $P_B \leq 1.5\left[1-\frac{(P_M)^2}{(S_D)^2}\right]S_D$ (b)	$P_M \leq S_D$ $P_B \leq \frac{4}{\pi}S_D Cos\left(\frac{\pi}{s}\bullet\frac{P_M}{S_D}\right)$ (c)	Within Yield
3. Normal Operating Loadings + Pipe Rupture + Safe Shutdown Earthquake	$P_M \leq S_L$ $P_B \leq 1.5\left[1-\frac{(P_M)^2}{(S_L)^2}\right]S_L$ (b)	$P_M \leq S_L$ $P_B \leq \frac{4}{\pi}S_L Cos\left(\frac{\pi}{s}\bullet\frac{P_M}{S_L}\right)$ (a),(c)	Deflection of supports limited to maintain supported equipment within limits shown in columns (1) and (2)

NOTES:

(a) These stress criteria are not applied to the piping run within which a pipe break is considered to have occurred.

(b) For loading combinations 2 and 3, stress limits for vessel, with the symbol P_M changed to P_L, should also be used in evaluating the effects of local loads imposed on vessels and/or piping.

(c) The tabulated limits for piping are based on a minimum "shape factor." These limits may be modified to incorporate the shape factor of the particular piping being analyzed.

(d) The above criteria does not apply to the replacement RVCH. Refer to Table 4-9 for the Design Code for the replacement RVCH and CEDMs.

TABLE 4-8

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS

Legend:

P_M = Calculated Primary Membrane Stress
P_B = Calculated Primary Bending Stress
P_L = Calculated Primary Local Membrane Stress
S_M = Tabulated Allowable Stress Limit at Temperature from ASME B&PV Code, Section III or ANSI B31.7
S_Y = Tabulated Yield at Temperature, ASME B&PV Code, Section III
S_D = Design Stress
= S_Y (for ferritic steels)
= $1.2S_M$ (for austenitic steels)
S_L = $S_Y + 1/3\ (S_u - S_Y)$
S_u = Tensile Strength of Material at Temperature

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lb/in^2.

Material	S_Y (a)	S_u	S_D	S_L
A 106B	25.4	60.0 (b)	25.4	36.9
SA 533B	41.4	80.0 (b)	41.4	54.3
304 SS	17.0	54.0 (c)	18.35	29.3
316 SS	18.5	58.2 (c)	22.2	31.7

a. From ASME B&PV Code, Section III, at 650°F.
b. Minimum value at room temperature which is approximately the same at 650°F for ferritic materials.
c. Estimated.

TABLE 4-9

REACTOR COOLANT SYSTEM CODE REQUIREMENTS

COMPONENT	CODE
Reactor Vessel	ASME III, Class A[(a)(h)]
RSG	ASME III, Class A[(a)(g)] Code Cases 1401, N-20-4, N-411-1, N-474-1, 2142-1, 2143-I, 1332-2, 1332-4, 1359-1
Pressurizer	ASME III, Class A[(a)]
Coolant Pumps	ASME III, Class A[(a)]
Seal Assemblies	ASME III, Class 1[(f)]
Quench Tank	ASME III, Class C[(c)]
Pressurizer Safety and Power Operated Relief Valves	ASME III[(b)]
Piping	ASME III[(a)(d)] USAS B 31.7 Class I[(e)]

NOTES:

(a) The latest editions, addenda and rulings in effect through the winter of 1967 were used for ASME B&PV Code.

(b) The latest editions, addenda and rulings in effect through winter of 1968 were used for ASME B&PV Code.

(c) Code effective date is May 29, 1969. The requirements of Paragraph UW-2(a) of Section VIII apply.

(d) The summer 1969 addenda was added. Code Case N-1401 is included.

(e) Code Cases 83 and 1477 are included.

(f) Replacement Sulzer Bingham RCP seals are designed in accordance with ASME B&PV Code, Section III, 1983 Edition with Summer 1983 Addenda.

(g) ASME B&PV Code 1989 Edition, no addenda is applicable to the RSG lower assembly part.

(h) The replacement RVCH was designed and fabricated to the 1995 Edition, 1996 Addenda of the ASME B&PV Code Section III. The CEDMs were designed and fabricated to the 1998 Edition through 2000 Addenda of Section III.

TABLE 4-9A

REACTOR COOLANT VENT VALVES

Valve Type:	Solenoid operated globe valve, energize to open
Size:	3/4" socket weld ends
Code:	ASME B&PV Code Section III, 1977 Edition, Winter 1977 Addenda, Class 1
Rating:	2173 ANSI B16.34
Seismic:	Seismic Category I
Material:	SA182F, 316L
Design pressure:	2485 psig
Design temperature:	700°F

TABLE 4-10

MATERIALS EXPOSED TO COOLANT

Reactor	
Vessel Cladding	Weld Deposited Type 308 SS (308L/309L for the replacement RVCH)
Vessel Internals	304 SS and Ni-Cr-Fe Alloy
Fuel Cladding	Zircaloy-4, Zircaloy-2P, M5®
Control Element Drive Mechanisms	Ni-Cr-Fe
Piping (excluding surge line)	Austenitic Stainless Steel Cladding Type 304L
Piping (at Hot Leg instrument nozzles)	SA-516, Gr 70 Carbon Steel
Surge Line	ASTM 351, Gr CF8M Alloy Steel
SG	
Bottom Head Cladding	Weld Deposited ER309L ER308L
Tube Sheet Cladding	Weld Deposited UNS N06052 (Code Case 2142-1)
Tubes	SB-163 Alloy 690 (Code Case N-20-4)
Divider Plate	SB-168 UNS N06690
Pumps	
Casing	Austenitic Stainless Steel, Type 3l6
Internals	Austenitic Stainless Steel, Type 316 & 304
Pressurizer	
Cladding	Weld Deposited Stainless Steel Type 308 and Ni-Cr-Fe Alloy
Upper Portion Heater	Electrodeposited Pure Nickel
Sleeve Bores [(a)(c)]	Plating
Upper Level Instrument Nozzle Base	SA-533, Gr B, Cl 1[(b)]
Heater Sleeves [(c)]	SA-213, Grade TP316/TP316L
Lower Level Instrument Nozzle [(c)]	SA-479, Grade TP316/TP316L

(a) Lower portions of the heater sleeves at B-3 and FF-1 are removed from their penetrations leaving a portion of the SA-533, Gr B, Cl 1 base material exposed to reactor coolant.

(b) Repairs to instrument nozzle for 2LT110X resulted in a small portion of the penetration bore base material being exposed to reactor coolant.

(c) The half-nozzle modification of the Unit 1 heater sleeves and lower level instrument nozzle results in a small portion of each penetration bore base material being exposed to reactor coolant.

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE ($n/cm^2 x10^{19}$)	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%)			REFERENCE
				P	S	Cu	
1	A302B Plate	0.2	50	0.015	0.021	0.22	For Data Points 1-12 (Reference 9)
2	A302B Plate	1.1	140	0.015	0.021	0.22	
3	A302B Plate	1.5	155	0.015	0.021	0.22	
4	A302B Plate	1.7	140	0.015	0.021	0.22	
5	A302B Plate	2.1	150	0.015	0.021	0.22	
6	A302B Plate	2.3	160	0.015	0.021	0.22	
7	A302B Plate	3.0	155	0.015	0.021	0.22	
8	A302B Plate	3.1	160	0.015	0.021	0.22	
9	A302B Plate	3.1	170	0.015	0.021	0.22	
10	A302B Plate	3.1	155	0.015	0.021	0.22	
11	A302B Plate	3.4	180	0.015	0.021	0.22	
12	A302B Plate	4.8	195	0.015	0.021	0.22	
13	A302B Plate	0.5	75	0.018	0.018	NA[a]	For Data Points 13-18 (Reference 10)
14	A302B Plate	1.0	120	0.018	0.018	NA[a]	
15	A302B Plate	0.7	100	0.018	0.018	NA[a]	
16	A302B Plate	1.5	140	0.018	0.018	NA[a]	
17	A302B Plate	4.8	155	0.018	0.018	NA[a]	
18	A302B Plate	17.5	225	0.018	0.018	NA[a]	
19	A302B Plate	3.0	205	0.012	0.025	0.20	For Data Points 19-22 (Reference 11)
20	A302B Plate	3.0	135	0.009	0.024	NA	
21	A302B Plate	3.0	140	0.009	0.024	NA	
22	A302B Plate	3.0	120	0.009	0.024	NA	
23	A302B Plate	0.5	65	0.009	0.024	NA	For Data Points 23-24 (Reference 12)
24	A302B Plate	3.1	165	0.009	0.024	NA	
25	A302B Plate	3.4	170	Nominal A302B Composition[b]			For Data Point 25 (Reference 13)
26	A302B Plate	3.8	160	Nominal A302B Composition[b]			For Data Point 26 (Reference 14)
27	A302B Plate	2.9	195	Nominal A302B Composition[b]			For Data Points 27-28 (Reference 15)
28	A302B Plate	1.1	130				
29	A302B Plate	2.1	85	0.007	0.018	0.11	For Data Points 29-31 (Reference 16)
30	A302B Plate	2.1	30	0.007	0.018	0.11	
31	A302B Plate	2.1	70	0.007	0.018	0.11	

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE $(n/cm^2 x10^{19})$	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%)			REFERENCE
				P	S	Cu	
32	A533B Plate	2.3	120	0.009	0.022	0.14	For Data Points 32-42 (Reference 17)
33	A533B Plate	2.3	95	0.010	0.023	0.14	
34	A533B Plate	1.7	190	0.010	0.017	0.19	
35	A533B Plate	0.2	0	0.008	0.015	0.09	
36	A533B Plate	2.0	80	0.008	0.015	0.09	
37[c]	A533B Plate	2.0	90	0.008	0.015	0.09	
38	A533B Plate	0.5	35	0.008	0.015	0.09	
39	A533B Plate	2.0	75	0.008	0.015	0.09	
40	A533B Plate	1.7	70	0.008	0.015	0.12	
41	A533B Plate	1.7	85	0.008	0.019	0.11	
42	A533B Plate	1.8	50	0.008	0.018	0.12	
43	A533B Plate	0.5	0	0.008	0.014	0.09	For Data Points 43-44 (Reference 18)
44	A533B Plate	2.4	85	0.008	0.014	0.09	
45	A533B Plate	2.5	60	0.003	0.014	0.09	For Data Point 45 (Reference 19)
46[d]	A533B Plate	3-4	215	0.012	0.016	0.25	For Data Points 46-47 (Reference 20)
47[d]	A533B Plate	3-4	255	0.012	0.016	0.25	
48	A533B Submerged Arc Weld	1.7	200	0.015	0.011	0.22	For Data Points 48-49 (Reference 21)
49	A533 Electroslag Weld	1.8	165	0.008	0.014	0.19	
50	A533B Weld	0.5	0	NA	NA	0.09	For Data Points 50-53 (Reference 22)
51	A533B Weld	2.4	90	NA	NA	0.09	
52	A533B Weld	0.5	105	0.010	0.014	0.14	
53	A533B Weld	2.4	210	0.010	0.014	0.14	
54	A533B Electroslag Weld	2.5	100	0.002	0.012	0.09	For Data Point 54 (Reference 23)
55	A533B Weld	3-4	260	NA	NA	NA	For Data Point 55 (Reference 24)
56	A533B Submerged Arc HAZ	1.7	145	From Data Points 40 or 41			For Data Point 56 (Reference 25)
57	A533B Submerged Arc HAZ	3-4	115	From Data Point 55			For Data Point 57 (Reference 26)
58	A533B Plate	1.0	101	0.012	0.018	NA	For Data Points 58-60 (Reference 27)
59	A533B Plate	1.0	126	0.012	0.018	NA	
60	A533B Plate	1.0	70	0.012	0.018	NA	

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE ($n/cm^2 x 10^{19}$)	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%)			REFERENCE
				P	S	Cu	
61	A533B Plate	5.0	80	0.012	0.25	NA	For Data Points 61-64 (Reference 28)
62	A533B Plate	4.0	135	0.012	0.25	NA	
63	A533B Plate	1.0	85	0.012	0.25	NA	
64	A533B Submerged Arc Weld	3.5	256	0.019	0.13	0.22	
65	A533B Submerged Arc Weld	2.8	65	0.009	NA	0.03	For Data Points 65-66 (Reference 29)
66	A533B Submerged Arc Weld	2.8	40	0.009	NA	0.03	
67	A533B Submerged Arc Weld	.47	70	0.012	0.018	NA	For Data Points 67-69 (Reference 30)
68	A533B Submerged Arc Weld	.94	95	0.012	0.018	NA	
69	A533B Submerged Arc Weld	1.05	130	0.012	0.018	NA	

NOTES:

(a) Analysis not available.

(b) Specific analysis of material not available.

(c) Transverse specimens.

(d) Exact fluence not reported.

TABLE 4-11A

CALVERT CLIFFS UNIT 1 REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

		WELD						
ID	**LOCATION**	**WIRE SPEC. (Heat No.)**	**FLUX TYPE. (Lot No.)**	**PLATE HEAT NO.**	**Cu (%)**	**Ni (%)**	**INITIAL RT_{NDT} (°F)**	**INITIAL UPPER SHELF ENERGY (ft/lb)**
2-203-A,B,C	Intermediate Shell Axial Welds	MIL B-4 Mod. (20291, 12008)	Linde 1092 (3833)	---	0.22	0.83	-50.0	110.0
3-203-A,B,C	Lower Shell Axial Welds	MIL B-4 Mod. (21935)	Linde 1092 (3869)	---	0.17	0.72	-56.0	109.0
9-203	Lower to Intermediate Girth Weld	MIL B-4 (33A277)	Linde 0091 (3922)	---	0.23	0.16	-80.0	160.0
D-7206-1	Intermediate Shell Plate			C4351-2	0.11	0.55	20.0[a]	90.0[a]
D-7206-2	Intermediate Shell Plate	---	---	C4441-2	0.12	0.64	-30.0[a]	81.0[a]
D-7206-3	Intermediate Shell Plate			C4441-1	0.12	0.64	10.0	112.0
D-7207-1	Lower Shell Plate			C4420-1	0.13	0.54	10.0[a]	77.0[a]
D-7207-2	Lower Shell Plate	---	---	B8489-2	0.11	0.56	-10.0[a]	90.0[a]
D-7207-3	Lower Shell Plate			B8489-1	0.11	0.53	-20.0[a]	81.0[a]

All values, except those for Cu and Ni, are from the Comprehensive Reactor Vessel Surveillance Program, Revision 2. The Cu and Ni values can be found in the 1995 PTS submittal.

[a] These values have been corrected for the transverse charpy direction in accordance with NRC Branch Technical Position MTEB 5-2.

TABLE 4-11B

CALVERT CLIFFS UNIT 2 REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

ID	LOCATION	WELD WIRE SPEC. (Heat No.)	WELD FLUX TYPE. (Lot No.)	PLATE HEAT NO.	Cu (%)	Ni (%)	INITIAL RT_{NDT} (°F)	INITIAL UPPER SHELF ENERGY (ft/lb)
2-203-A,B,C	Intermediate Shell Axial Welds	MIL B-4 (A8746)	Linde 124 (3878)	---	0.16	0.10	-56.0	83.5
3-203-A,B,C	Lower Shell Axial Welds	MIL B-4 (33A277)	Linde 0091 (3922)	---	0.23	0.16	-80.0	160.0
9-203	Lower to Intermediate Girth Weld	MIL B-4 (10137)	Linde 0091 (3999)	---	0.21	0.06	-60.0	140.0
D-8906-1	Intermediate Shell Plate			A4463-1	0.15	0.56	10.0[a]	77.0[a]
D-8906-2	Intermediate Shell Plate	---	---	B9427-2	0.11	0.56	10.0[a]	74.0[a]
D-8906-3	Intermediate Shell Plate			A4463-2	0.14	0.55	5.0[a]	75.0[a]
D-8907-1	Lower Shell Plate			C5804-1	0.15	0.60	-8.0[a]	83.0[a]
D-8907-2	Lower Shell Plate	---	---	C5286-1	0.14	0.66	20.0	115.0
D-8907-3	Lower Shell Plate			C5803-3	0.11	0.74	-16.0[a]	84.5[a]

All values, except those for Cu and Ni, are from the Comprehensive Reactor Vessel Surveillance Program, Revision 2. The Cu and Ni values can be found in the 1995 PTS submittal.

[a] These values have been corrected for the transverse charpy direction in accordance with NRC Branch Technical Position MTEB 5-2.

TABLE 4-12

SUMMARY OF SPECIMENS PROVIDED FOR EACH EXPOSURE LOCATION

CAPSULE LOCATION ON VESSEL WALL	BASE METAL IMPACT		BASE METAL TENSILE	WELD METAL IMPACT	WELD METAL TENSILE	HAZ IMPACT	HAZ TENSILE	REFERENCE IMPACT[c]	TOTAL SPECIMENS IMPACT	TOTAL SPECIMENS TENSILE
	L[a]	T[b]								
83°	12	12	3	12	3	12	3	-	48	9
97°	12	12	3	12	3	12	3	-	48	9
104°	12	-	3	12	3	12	3	12	48	9
263°	12	-	3	12	3	12	3	12	48	9
277°	12	12	3	12	3	12	3	-	48	9
284°	12	12	3	12	3	12	3	-	48	9
	72	48	18	72	18	72	18	24	288	54

(a) L = Longitudinal

(b) T = Transverse

(c) Reference material correlation monitors

TABLE 4-13A

UNIT 1 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

CAPSULE AZIMUTHAL POSITION	TARGET FAST NEUTRON FLUENCE ($x10^{19}n/cm^2$)	TARGET FLUENCE EXPECTED AT END OF CYCLE	PROJECTED END OF CYCLE DATE
263°	0.62[a]	3	Withdrawn, 1979
97°	2.64[b]	10	Withdrawn, 1992
284°	3.08[c]	19	Withdrawn, 2010
83°	5.28[d]	24	2020
277°	6.62[e]	30	2032
104°	STANDBY		

[a] Actual capsule fluence (Reference 34).

[b] Actual capsule fluence (Reference 35).

[c] Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel ¼-T location at end of extended life.

[d] Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel inner wall location at end of extended life.

[e] Withdrawal criteria - Not less than once or greater than twice the peak end of extended life vessel fluence at the vessel inner wall ($5.28x10^{19}$<fluence in n/cm^2<$10.56x10^{19}$). Note: This capsule also satisfies the requirement in the NRC safety evaluation report for Calvert Cliffs license renewal, that one capsule containing dosimetry is to be removed during the final 5 years of the extended license.

TABLE 4-13B

UNIT 2 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

CAPSULE AZIMUTHAL POSITION	TARGET FAST NEUTRON FLUENCE ($x10^{19}n/cm^2$)	TARGET FLUENCE EXPECTED AT END OF CYCLE	PROJECTED END OF CYCLE DATE
263°	0.806[a]	4	Withdrawn, 1982
97°	1.85[b]	9	Withdrawn, 1993
104°	3.27[c]	18	Withdrawn, 2011
83°	6.21[d]	25	2025
277°	7.50[e]	29	2033
284°	STANDBY		

[a] Actual capsule fluence (Reference 36).

[b] Actual capsule fluence (Reference 37).

[c] Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel ¼-T location at the end of extended life.

[d] Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel inner wall location at the end of extended life.

[e] Withdrawal criteria - Not less than once or greater than twice the peak end of extended life vessel fluence at the vessel inner wall ($6.21x10^{19}$<fluence in n/cm^2<$12.42x10^{19}$). Note: This capsule also satisfies the requirement in the safety evaluation report for Calvert Cliffs license renewal, that one capsule containing dosimetry is to be removed during the final 5 years of the extended license.

TABLE 4-13C

REACTOR VESSEL SURVEILLANCE PROGRAM UNIT 1 SUPPLEMENTAL CAPSULE REMOVAL SCHEDULE

CAPSULE IDENTIFICATION	PROJECTED END OF CYCLE DATE
S1	2018
S2	2032

TABLE 4-14

RCS QUALITY ASSURANCE PROGRAM

a. Reactor Vessel	
Forgings	
Flanges	UT,MT
Studs	UT,MT
Cladding	UT,PT
Nozzles	UT,MT
Plates	UT,MT
Cladding	UT,PT
Welds	
Main Seams	RT,MT
CEDM Head Nozzle Connection	PT,RT[d],UT[d]
Instrumentation Nozzles	PT,RT[d],UT[d]
Main Nozzles to Shell	RT,MT
Cladding	UT,PT
Nozzle Safe Ends	RT,PT
Vessel Support Buildup	UT,MT
All Welds - After Hydrostatic Test	MT,PT & UT of all J-welds[d]
Replacement RVCH Vent	PT,UT[d]
b. RSG	
Tube Sheet	
Forging	UT,MT
Cladding	UT,PT
Weld Buildup	UT,PT
Primary Head	
Forging	UT,MT
Cladding	UT,PT
Secondary Shell and Head	
Plates and Forgings	UT,MT
Tubes	UT,ET
Nozzles (Forgings)	UT,MT
Studs (>2”)	UT,MT
Studs (≤2”)	MT
Welds	
Shell, Longitudinal[a]	RT,MT
Shell, Circumferential[c]	RT,MT,UT
Cladding	UT,PT
Nozzles to Shell[a]	RT,MT
Tube-to-Tube Sheet	PT
Instrument Connections	MT,RT,PT
Temporary Attachments After Removal	MT
All Welds – After Hydrostatic Test	MT or PT
Nozzle Safe Ends[a]	RT,(MT or PT)
Level Nozzles[b]	MT,RT,PT
Vessel Support Buildup	UT,MT
Girth Weld (Transition Weld)	RT (PT or MT)

TABLE 4-14

RCS QUALITY ASSURANCE PROGRAM

c.	<u>Pressurizer</u>	
	Heads	
	Plates	UT,MT
	Cladding	UT,PT
	Shell	
	Plates	UT,MT
	Cladding	UT,PT
	Heaters	
	Tubing	UT,PT
	Centering of Elements	RT
	Nozzles	UT,MT
	Studs	UT,MT
	Welds	
	Shell, Longitudinal	RT,MT
	Shell, circumferential	RT,MT
	Cladding	UT,PT
	Nozzles	RT,MT
	Nozzle Safe Ends	RT,PT
	Instrument Connections	PT
	Support Skirt	RT,MT
	Temporary Attachments After Removal	MT
	All Welds After Hydrostatic Test	MT
	Heater Assembly	RT,PT
d.	<u>Pumps</u>	
	Castings	RT,PT
	Forgings	UT,PT
	Welds	
	Circumferential	RT,PT
	Instrument Connections	PT
	All Welds After Hydrostatic Test	PT
e.	Piping	
	Fittings	RT,PT
	Pipe	RT,PT
	Nozzles	RT,PT
	Welds	
	Circumferential	RT,PT
	Nozzle to Run Pipe	RT,PT
	Instrument Connections	PT
	Cladding	UT,PT

RT - Radiographic   PT - Dye Penetrant   ET - Eddy Current
UT - Ultrasonic   MT - Magnetic Particle   GT - Gas Leak Test

(a) These examinations only apply to the steam drum.

(b) RT and PT are not performed for steam drum level nozzles.

(c) UT is not performed for steam drum circumferential welds.

(d) Applicable to replacement RVCH.

TABLE 4-15

RCS INSPECTION CE REQUIREMENTS[a]

REACTOR VESSEL	**CE REQUIREMENTS**	**CODE REQUIREMENT**
Ultrasonic Testing (UT)	1. UT of Weld Clad for bond	1. None
Dye-Penetrant	1. PT Test Root each 1/2 in. and Final Layer of Welds for Partial Penetration Welds to Control Element Driver Mechanism Head Adapters and Instrument Tube Connections	1. PT Test of each 1/3 weld throat or 1/2" which-ever is lesser N-462.4 (d)(l)
Replacement Steam Generator		
Ultrasonic Test	1. UT for Defects in Tube Sheet Clad	1. None
	2. UT of Weld Clad for bond	2. None
Pressurizer		
Ultrasonic Testing (UT)	1. UT clad for bond	1. None
Radiography (RT)	1. Radiograph Heaters to Check Heater Wire Positioning	1. None

[a] Replacement steam generators are manufactured by BWC.

4.2 OVERPRESSURE PROTECTION SYSTEM

4.2.1 HIGH TEMPERATURE OVERPRESSURE

4.2.1.1 Design Basis

The RCS is structurally designed for operation at 2500 psia and 650°F (pressurizer 700°F). Operation of the system at 2250 psia nominal and 600°F will result in material stresses 90% of design values. An evaluation of the effect of the RSGs on the structural analysis was performed by BWC and Framatome Technologies, Inc. Detailed structural analyses have been performed by the component vendors and reviewed independently by CE for all portions of the system. Welding materials used have physical properties superior to the materials which they join. Inspection procedures and tests specified and independently reviewed by CE were carried out to ensure that pressure-containing components have the maximum integrity obtainable with present code-approved inspection techniques.

The RCS is protected against overpressure by two ASME B&PV Code approved safety valves which limit system pressure to a maximum of 110% of design and by two solenoid-operated relief valves. These valves are described in Section 4.2.1.2.

Portions of the piping for the Code approved safety valves are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the relief valve system fatigue analysis considers 100 events where the relief valves are operated. See Reference 1 for further details.

4.2.1.2 System Description

Valves

Parameters for the actuator-operated pressurizer spray valves are given in Table 4-16. The actuator-operated relief valve isolation stop valve parameters are given in Table 4-17. The position of each valve on loss of actuating signal (failure position) is selected to ensure safe operation of the system and plant. System redundancy is considered when specifying the failure position of any given valve. Valve position indication is provided at the main control panel where considered necessary to ensure safe operation of the plant.

Manually operated valves in the RCS have backseats to limit stem leakage when in the open position. Globe valves are installed with flow entering the valve under the seat. This arrangement will reduce stem leakage during normal operation or when closed.

The two augmented quality PORVs relieve sufficient pressurizer steam during abnormal transients to prevent opening of the RCS safety valves. The relief valves are actuated by the high RCS pressure trip signal. Parameters for these valves are given in Table 4-18.

The valves are solenoid-operated power relief valves. The two half-capacity valves are located in parallel pipes which are connected to the two pressurizer safety and relief valve nozzles on the inlet side, and to the relief line piping to the quench tank on the outlet side. A motor-actuated isolation valve is provided upstream of each of the relief valves to permit isolating the valve for maintenance or in case of valve failure.

The PORVs, block valves, and associated control systems are classified as augmented quality. The main control board wiring for the isolation motor-operated valves is not required to meet the electrical separation criteria for safety related circuits. The wiring configuration is in accordance with the guidance issued in NRC Generic Letter 90-06.

The capacity of the PORVs is sufficient to pass the maximum steam surge associated with a continuous CEA withdrawal incident starting from low power. Assuming that a reactor trip is effected on a high-pressure signal, the capacity of the PORVs is sufficient so that the safety valves do not open. The relief valve capacity is also large enough so that the safety valves do not open during a loss-of-load incident from full power. This assumes normal operation of the pressurizer spray system, and reactor trip on high pressure.

Two safety valves located on the pressurizer provide overpressure protection for the RCS. They are totally enclosed, back pressure compensated, spring-loaded safety valves meeting ASME B&PV Code requirements. The stress analysis on these valves included the effects of sudden opening of these valves, and support and restraint location were selected on this basis. Parameters for these valves are given in Table 4-19.

The safety valves pass sufficient pressurizer steam to limit the primary system pressure to 110% of design (2,750 psia) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 2737 MWt. The reactor is assumed to trip on a high RCS pressure signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the steam system safety valves. Conservative values for all system parameters, delay times, and core moderator coefficient are assumed. A safety valve technical summary report is required by ASME B&PV Code, Section III, 1968 Edition (Summer 1969 Addendum). The effect of the RSGs on the ability of the safety valves to limit the pressure to 110% of design was evaluated. It was determined that this criterion is still met with the RSGs.

Forged and stainless steel valves for use within the Reactor Coolant Boundary have been supplied by Velan Engineering of Montreal, Quebec. Velan has supplied valves to numerous nuclear projects in the United States. Examples include Yankee Rowe, Connecticut Yankee, Palisades, Fort Calhoun and Maine Yankee. No other pressure boundary components were designed or fabricated outside of the United States.

The following steps are taken during fabrication to ensure foreign procured components are acceptable. ASTM materials are specified and mill tests report are submitted to verify material. Non-destructive tests, ultrasonic and liquid penetrant, are performed on the pressure containing parts. Hydrostatic shell pressure and leak tests are performed to MSS-SP-61. Manufacturing sequence plans, non-destructive test technique procedures and testing procedures are all submitted by the vendor to the purchaser for review and approval prior to use. All tests are witnessed by quality control representatives of the purchaser in addition to the vendor's own inspection force.

Quench Tank

The quench tank is designed to prevent the discharge of the pressurizer relief or safety valves from being discharged to the containment. The steam discharged into the quench tank from the pressurizer is discharged under water by a sparger

to enhance condensation. The normal quench tank water volume of 135 ft^3 is sufficient to condense the steam released from the pressurizer safety and relief valves. The steam released as a result of the uncontrolled rod withdrawal is based on no coolant letdown or pressurizer spray.

The water temperature rise in the quench tank is limited to 160°F, assuming a maximum initial water temperature of 120°F. The gas volume in the tank is sufficient to limit the tank pressure after the above steam release to approximately 85 psia. The quench tank is equipped with a demineralized water supply to cool the tank after a steam discharge into it.

The quench tank can condense the steam discharged during a loss-of-load incident as described in Section 14.5 without exceeding the rupture disc setpoint of 100 psig, assuming normal closing of the safety valves at the end of the incident. It is not designed to accept a continuous uncontrolled safety or relief valve discharge. The rupture disc vents to the containment atmosphere. The quench tank parameters are given in Table 4-21.

The tank normally contains demineralized water under a nitrogen overpressure. The sparger, spray header, nozzles and rupture disc fittings are stainless steel. The tank is designed and fabricated in accordance with ASME B&PV Code, Section III, Class C.

4.2.2 LOW TEMPERATURE OVERPRESSURE PROTECTION

Low temperature overpressure protection is provided at Calvert Cliffs by a combination of administrative controls and hardware provisions. The hardware provisions include the incorporation of a multiple setpoint capability in the PORV control circuitry and enabling the low temperature pressure setpoint of PORVs during low temperature operations. A microprocessor-based control unit provides either a variable setpoint that varies as a function of RCS temperature or a fixed setpoint, depending on plant conditions. Although the PORVs are the primary means of protection, it is desirable to avoid challenging them. Therefore, maintenance of administrative controls is integral to overpressure protection. Disabling components when unnecessary for plant operation will prevent their inadvertent actuation and therefore minimize their potential for causing overpressurization.

Operator action is also used to mitigate LTOP events. However, because operator action cannot always be assumed, and because possible equipment malfunctions must be considered, additional controls have been put in place to ensure adequate protection exists for all postulated events. Analyses have been performed which demonstrate that a combination of administrative controls and hardware modifications provide this protection. In general, this protection includes the following:

- Procedural precautions and controls;
- Disabling of non-essential components whenever LTOP is required [below Minimum Pressurization Temperature (MPT) enable temperature and RCS not vented];
- Maintenance of a non-solid system whenever practical; and
- Use of the variable or fixed setpoint in the PORV control logic.

<u>Design Criteria</u>

The basic criteria to be satisfied in determining the adequacy of overpressure protection is that no single equipment failure or operator error shall result in violation of the pressure-temperature (P-T) limits.

Design Events

Overpressurization analyses were performed as follows:

- The worst-case overpressurization scenarios were identified for both mass and energy addition events; and
- The effectiveness of the PORV to terminate an overpressurization event was evaluated.

RCS Mass Addition Analysis

The following mass addition events were postulated:

- Inadvertent high pressure safety injection (HPSI) pump start;
- Inadvertent HPSI and charging pump start; and
- Inadvertent mismatch of charging and letdown flow.

RCS Energy Addition Analysis

The following energy addition events were postulated:

- RCS expansion following loss of shutdown cooling, including SG heat addition;
- Inadvertent pressurizer heater actuation; and
- Energy addition from the SG secondary side to the RCS due to a start of an RCP when the SGs are at a higher temperature than the reactor vessel inventory.

Energy additions which are constant with time include inadvertent pressurizer heater actuation and decay heat addition. Also, all letdown flow paths which could mitigate or terminate a particular overpressurization event were assumed isolated. Hand calculations were sufficient to model the resulting transients.

The design events are:

- An RCP start with hot SGs; and
- An inadvertent HPSI actuation with concurrent charging.

Any measures which will prevent or mitigate the design events are sufficient for any of the less severe incidents.

A single PORV and the administrative controls will provide satisfactory control of all transients. Overpressurization due to the spurious actuation of full flow from a HPSI pump will be precluded at and below the MPT enable temperature by disabling two HPSI pumps, placing the third in pull-to-lock, and by throttling the third pump when used to add mass to the RCS. Lifting of the PORV on an RCP start will be precluded by placing administrative limitations on initial pressurizer pressure, secondary-to-primary temperature ΔT, and pressurizer level.

Operator Action

In each of the transient analyses, operator action was not credited for the first 10 minutes. The pressure alarms, in addition to other plant condition indications, will make the operator aware of the transient.

Single Failure

A single failure is considered in the overpressure mitigation system response to an initiating event.

The sensing/actuating/relieving system consists of two redundant and independent trains.

For the operational energy addition transient following an RCP start with a hot SG, the PORV setpoint will not be challenged for at least 10 minutes if specified initial conditions for the pump start are satisfied. In this case, failure of a PORV cannot result in overpressurization since the valve setpoint is not challenged. Failure to satisfy one of the initial conditions may result in opening one or both PORVs. In the case with a water solid system, pressure could exceed the Appendix G limits following an RCP start.

For the mass addition design basis event [one HPSI pump actuation], a single PORV and a pressurizer steam volume provides protection provided that two of the three HPSI pumps are disabled and the remaining pump's flow is throttled. If the single failure is a failure to throttle the HPSI pump while adding mass through one HPSI loop motor-operated valve, then two PORVs are capable to maintain the pressurization below Appendix G limits.

Pressure-Temperature Limits

The technical specification P-T limits, from which the heatup and cooldown curves were derived, were calculated per the requirements of 10 CFR Part 50, Appendix G as supplemented by ASME B&PV Code, 1986 Edition, Section III, Appendix G. Pressure-Temperature limits were calculated using adjusted reference temperatures (ARTs) developed from the guidance of RG 1.99, Revision 2. In addition, these P-T limits were corrected for pressure drops and for pressure and temperature instrument uncertainties.

The low temperature PORV pressure lift setpoint is based on protecting the most restrictive pressure of both the heatup and cooldown curves.

The LTOP enable temperature (MPT enable) has been developed using the guidance found in Nuclear Regulatory Commission Standard Review Plan (SRP) 5.2.2, Revision 2. This SRP defines MPT enable as "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4 T or 3/4 T) that is controlling the Appendix G limit calculations." MPT enable temperature was calculated accordingly by using specific heatup transients with changing thermal rates to accurately determine stress distributions.

Seismic and IEEE-279 Design Criteria

The PORV installation meets seismic criteria consistent with the basic objective of preventing a LOCA pathway. In addition, LTOP mitigating system equipment is designed such that their failure will not degrade the performance of other safety-related equipment.

In addition, the intent of IEEE-279 criteria is met for the reliability and effectiveness of the mitigating system in that a single failure which initiates an overpressurization event does not disable the mitigating system.

Power is supplied to the PORVs from vital supplies. Cable raceways for this equipment are supported to withstand a seismic event.

The low temperature overpressure protection (LTOP) controls are also used to prevent and mitigate overpressure events in the shutdown cooling system (Section 9.2.6).

Testability

The LTOP system is designed to be tested with a frequency that will ensure the system is operable when needed.

4.2.3 REFERENCE

1. Bechtel Specification 6750-M-0310E, "Design Specification for Piping, Valves, and Associated Equipment of the Pressurizer Relief System"

TABLE 4-16

ACTUATOR-OPERATED THROTTLING VALVE PARAMETERS

Service - Pressurizer Spray Valves	
Design Temperature, °F	650
Design Pressure, psia	2,500
Flow, gpm	375
Pressure Drop, psi	8.5 - 40
Failure Position	Fail Closed

TABLE 4-17

ACTUATOR-OPERATED STOP VALVE PARAMETERS

Service - Pressurizer Power-Operated Relief Isolation

Design Temperature, °F	675
Design Pressure, psia	2,500
Actuator	Electric Motor
Failure Position	As Is
ANSI Class	1,703 lb

TABLE 4-18

PRESSURIZER POWER-OPERATED RELIEF VALVE PARAMETERS

Design Pressure, psia	2,500
Design Temperature, °F	700
Fluid	Saturated Steam, 0.1% (wt) Boric Acid
Number	2
Capacity, lb/hr min. Each	153,000
Type	Solenoid Operated
Set Pressure, psig	2,385
Failure Position	Fail Closed

TABLE 4-19

PRESSURIZER SAFETY VALVE PARAMETERS

Design Pressure, psia	2,500
Design Temperature °F	700
Fluid	Saturated Steam, 0.1% (wt) Boric Acid
Set Pressure	
RC-200, psig	2,485
RC-201, psig	2,510
Capacity, lb/hr, at set pressure	
RC-200	296,068[a]
RC-201	299,065
Type	Spring loaded-balanced bellows, enclosed bonnet
Accumulation, %	3
Back Pressure Compensation	Yes

(a) Rated capacity is based on valve area at accumulation pressure (set pressure plus 3%).

TABLE 4-21

QUENCH TANK PARAMETERS

Design Pressure, psig	100
Design Temperature, °F	350
Normal Operating Pressure, psig	3
Normal Operating Temperature, °F	120
Internal Volume, ft^3	217
Normal Water Volume, ft^3	135
Normal Gas Volume, ft^3	82
Blanket Gas	Nitrogen
Manway (1 ea.) in.	16
Nozzles	
Pressurizer discharge (1 ea) nominal, in.	10
Demineralized water (1 ea) in.	2
Rupture Disc (1 ea) in.	18
Drain (1 ea) in.	2
Temp. Instrument (1 ea) in.	1
Level Instrument (2 ea) in.	1/2
Vent (1 ea) in.	1 ½
Vessel Material	ASTM A240, TP 304
Dimensions	
Overall Length, in.	144 3/8
Outside Diameter, in.	60
Dry Weight, lb	4600
Flooded Weight, lb	18,120

4.3 LEAK DETECTION SYSTEM

4.3.1 DESIGN BASIS

Three methods are provided to alert the operator of the presence of leakage from the RCS in a timely manner. The timed response allows detection and isolation of the leak to ensure the radiation emitted as result of the leak does not exceed acceptable limits. The systems which comprise the leakage detection system are, the containment sump level alarm system, a containment atmosphere particulate radioactivity monitoring system, and a containment atmosphere gaseous radioactivity monitoring system. In determining the acceptability of this leakage detection system, the system was compared to the guidance given in Regulatory Guide 1.45 (May 1973). The system, as described below, meets the intent of the guidance provided by Regulatory Guide 1.45. However, it must be noted that the three systems which comprise the leakage detection system are not equally sensitive to RCS leakage. The Regulatory Guide addresses the fact that when there is little radioactivity in the RCS, the radiation monitors are less effective in determining leaks. The design basis for the gaseous and particulate monitors is based on an assumption of 1% failed fuel and the resultant activity level in the RCS. As shown below, the time to detect a small leak varies with the RCS radioactivity concentration. Given the low concentration of radionuclides in the RCS during a normal operating cycle, it is unlikely that the radiation monitors would detect small leaks. Under normal operating conditions, there are two other diverse systems that can detect leaks in the RCS: the containment sump level alarm system and the performance of a water inventory balance.

During normal operations, a reactor coolant inventory analysis (commonly called a water balance inventory) is performed at least every 72 hours as required by Technical Specifications. This analysis uses changes in pressurizer level, changes in volume control tank level, changes in RCS average temperature and RCS makeup as inputs.

There is no practical analytical method available by which a leak rate can be correlated to crack size. In addition, due to the magnitude of more probable RCS leakage sources (i.e., packing and seals), RCS leakage is not relied upon for assurance of RCS integrity. Assurance of RCS integrity is warranted by the conservatism of the design and the operating restrictions that limit loadings on the RCS components. Periodic in-service inspections will, in addition, verify the integrity of the RCS boundary and ensure that degradation of the boundary has not occurred.

4.3.2 SYSTEM DESCRIPTION

The leakage detection system consists of three diverse systems, the containment sump level alarm system, a containment atmosphere particulate radioactivity monitoring system, and a containment atmosphere gaseous radioactivity monitoring system.

The containment normal sump level detection and alarm system consists of instrumentation that senses and indicates sump level, and provides an alarm. A sensor in the sump sends a signal to the level indicator/alarm in the Control Room. Should sump level exceed a predetermined value, the indicator/alarm provides an alarm to alert operators of the condition. The water collected in the sump is then processed by the Miscellaneous Waste Processing System.

The Containment Atmosphere Particulate and Gaseous Radioactivity Monitors consist of off-line detectors, sample pumps, pump controls and signal conditioning, recording, indication and alarm functions. The off-line sample pumps take suction from the containment ventilation system - this provides a representative sample of the containment atmosphere - and deliver the sample to the monitors. The particulate monitor is a fixed filter, scintillation type detector. The filter is removed periodically and analyzed for identification of any radioactive isotopes captured by the filter. The detector continuously

monitors the radiation level of the filter and provides a signal which is conditioned so it can be read by the indicating and recording electronics in the Control Room. The signal is sent to the recorders for trending purposes. Should the samples collected by the filter exceed predetermined limits, the radiation monitoring electronics also provide an alarm to alert operators of the condition. After the particulates are collected by the filters, the cleaned atmosphere sample is sent to the scintillation detector gaseous monitor where the radiation level is continuously monitored. The signal from the gaseous monitor is processed in a manner similar to the particulate monitor described above.

During plant operation, Technical Specification periodic surveillance of the leakage detection system are performed.

4.3.3 DESIGN ANALYSIS

The response of the containment sump level alarm is highly variable depending upon the location of the leak, how much vapor condenses and where it condenses. If the leaking vapor condenses on the containment air coolers, the water will drain by gravity to the containment normal sump. If leaking water condenses in other locations and drains to the normal sump, a total amount of approximately 49 gallons will cause a sump level alarm.

The sensitivity of the containment gaseous monitor is at least $3x10^{-6}$ μCi/cc for Xe-133. The response time of this monitor will vary greatly with radiation level in the RCS, the location of the leak, and whether the leakage occurs from a line coming from the RCS or one returning to the RCS from the CVCS. As discussed before, when there is little radioactivity in the RCS, the radiation monitors are much less effective is detecting leaks. The values given in the table below represent the time required to detect a leak with the gaseous radiation monitor. The time provided in this table represents the time needed to leak a sufficient amount of reactor coolant at 0.1 gpm, the time required to obtain sufficient mixing of the Xe-133 in the containment atmosphere, and the time required to obtain $3x10^{-6}$ μCi/cc of Xe-133 in the portion of the containment volume directly affected by the containment ventilation system.

Time in Minutes	Percent Failed Fuel
61.1	1.0
75.1	0.1
208.6	0.01
1743.0	0.001

The minimum setting for the level of radioactivity the Containment Atmosphere Particulate Monitor can detect is based on background radiation and operating experience. Operating experience is a factor because there are many assumptions (e.g., particulate size, distribution, transport, and deposition) that must be made in order to quantify a setpoint. Since the setpoint is a variable factor, the monitor is only able to provide operators with the ability to observe and trend leaks of radioactivity inside containment. With no fuel leakage present, the alarm for the Containment Atmosphere Particulate Monitor is set as low as possible and still avoid nuisance alarms caused by varying background levels.

4.3.4 IDENTIFIED VERSUS UNIDENTIFIED LEAKAGE

In accordance with Regulatory Guide 1.45, it is important to differentiate between identified and unidentified leakage from the RCS. This permits the development and acceptance of leakage detection systems that can detect leakage for these different conditions. The discussion below addresses the basis for identified and unidentified leakage as it impacts the Technical Specification limits and the capabilities of the leakage detection systems within the confines of Regulatory Guide 1.45.

The maximum allowable leak rate for the RCS is defined in Technical Specifications as 1 gpm for unidentified leakage and 10 gpm for identified leakage. The basis for the 1 gpm leak rate is that this rate can be readily detected and that appropriate action can be taken well prior to the time it constitutes a public hazard. The basis for the 10 gpm leak rate is that it is well within the 44 gpm capacity of one charging pump. Therefore, the leakage rate is based on the ability of one charging pump to replace reactor coolant leakage and still maintain a reasonable margin to the 10 gpm allowed limit such that repairs may be made without the necessity for shutdown.

The consensus of opinion by operating plants is that there would normally be some minimum quantity of unidentifiable RCS leakage. Significantly greater leakage than this reported minimum would be considered abnormal and would require an immediate search for its source. After a source of leakage is identified, it would be eliminated as soon as repairs could be made. The normal total unidentified leakage for each Calvert Cliffs unit is approximately 0.4 gpm. The typical sources of leakage are valve stem packing, valve bonnet gaskets and other gasketed mechanical joints.

Instrumentation is provided in the Control Room to provide means to identify the general location of the leak. These objectives are accomplished by either or both of the following:

a. Leakage from the RCS to the containment may be indicated by: increased pressure and temperature in containment, increased airborne radioactivity, increased level in the containment sump, decreasing pressure and level in the pressurizer, increased make-up flow from the CVCS, increased level in the reactor coolant drain tank, or increased humidity in the containment.
b. Leakage from components in the RCS can also be identified for the following components: relief and safety valves, reactor vessel head closure seal, reactor coolant pump seals, steam generator tubes or tubesheet and miscellaneous RCS valve stem leakage.

4.3.5 OTHER CONSIDERATIONS

Other methods and instrumentation that may be used, but are not credited, as part of the leakage detection system are contained in this section.

Leaks from the RCS to the containment may be indicated by the following instrumentation channels in the Control Room:

Pressurizer pressure indication and alarm
Pressurizer level indication and alarm
Containment pressure indication
Containment temperature
Containment humidity indicators
Reactor coolant drain tank level indication
Reactor coolant make-up water flow integrators

The sensors and indicators discussed above are factory calibrated prior to shipping to the plant. Calibration (sensitivity) was rechecked at the site and was verified during start-up testing. During plant operation, periodic surveillance was conducted as described in the Technical Specifications.

Leakage from components in the RCS can be identified and located utilizing one or more of the following:

a. Relief and Safety Valves
    1. Increased temperature of piping downstream of the valves

2. Increased pressure in the quench tank
3. Increased temperature in the quench tank
4. Increased level in the quench tank
5. Increased level on the acoustic flow monitors

b. Reactor Vessel Head Closure Seal
   1. High pressure in the area between the double O-ring seal

c. Reactor Coolant Pump Seals
   1. Increased seal pressure
   2. Increased seal temperature
   3. Increased level in the containment sump
   4. Increased temperature of the component cooling water from the RCPs
   5. Increased level in component cooling head tanks
   6. Increased radiation monitor reading in component cooling water system
   7. Increased controlled bleed off flow

d. Steam Generator Tubes or Tube Sheet
   1. Increased activity indicated by the condenser off-gas radiation monitor
   2. Increased activity indicated by analysis of steam generator water samples
   3. Increased N-16 activity indicated by the main steam line N-16 radiation

e. Miscellaneous RCS Valve Stem Leakage
   1. Increased level in the reactor coolant drain tank

The response of the containment pressure, temperature, humidity and sump level instruments is highly variable, depending upon the location of the leak, how much water condenses and where it condenses. However, some estimates of sensitivity can be made.

a. If the leaking water all vaporizes and does not condense: A total amount of 106 gallons (1.06 gpm for 100 minutes, 10.6 gpm for 10 minutes, etc.) would have the following effects:
   1. Increase relative humidity by 10% which is noticeable on the humidity indicator.
   2. Result in no significant increase in containment temperature.
   3. Result in no significant increase in containment pressure.

b. If the leaking water condenses in the containment coolers: The water drains by gravity to the normal sump. Upon receipt of the alarm, the operator will open drain valves and the water will drain by gravity to the ECCS room sump.
c. If the leaking water condenses and drains to the normal sump: A total amount of approximately 49 gallons (0.49 gpm for 100 minutes, 4.9 gpm for 10 minutes, etc.) will cause a sump level alarm.
d. If the leaking water drains to the reactor coolant drain tank: the operator in the Control Room is made aware of this by an increasing level indication and eventually a high level alarm which requires the operator to operate the reactor coolant drain tank pump. Since normal and expected leakage, such as valve stem leakage, drains to this tank, an increase in leak rate would be detected as a change in frequency of pumping. A ten gallon per minute leak rate would result in approximately one additional pumping each 75 minutes, which will be noticeable by the operator, within one to two hours.

Pressurizer level sensitivity to leakage depends upon operating conditions. During steady state power operation leaks are indicated as follows:

a. Leak rate less than 11 gpm.

   Leakage in this range is not detectable using pressurizer level, as the level is maintained automatically by the charging pumps within 4.1" of programmed level.

b. Leak Rate 11-55 gpm.

   Leakage in this range is indicated by the first backup charging pump cycling on and off to maintain level between -4.1" and -9.3".
   Time response for a 33 gpm leak:
   5.57 minutes to start the first backup pump.

c. Leak rate 55-99 gpm.

   Leakage in this range is indicated by the second backup charging pump cycling on and off to maintain level between -6.5" and -13.5".
   Time response for a 77 gpm leak:
   2.02 minutes to start the first backup pump.
   4.84 minutes to start the second backup pump.

d. Leak rate greater than 99 gpm.

   Leakage in this range is indicated by all three charging pumps running while level continues to decrease, and a low level alarm.
   Time response for a 121 gpm leak:
   1.23 minutes to start the first backup pump.
   2.17 minutes to start the second backup pump.
   3.18 minutes to low level alarm.

Pressurizer pressure instrumentation will only respond to very large leaks.

4.4 LOOSE PARTS DETECTION SYSTEM

4.4.1 DESIGN BASIS

The loose parts monitoring system monitors the RCS for internal loose parts. The system is designed to detect a loose part striking the internal surface of RCS components with an energy level of one-half foot pound or more.

4.4.2 SYSTEM DESCRIPTION

The loose parts monitoring system consists of transducers, preamplifiers, amplifiers and an analyzer to record the occurrence of a loose part within the RCS. Eight piezoelectric accelerometers are attached to the RCS boundary, where loose parts are most likely to become entrapped, as follows:

- 2 on the reactor vessel lower head, diametrically opposed
- 2 on the reactor vessel studs, diametrically opposed
- 2 on the primary head of each SG.

In addition, the RSGs are provided with two accelerometer attachment locations on the lower secondary shell adjacent to the tubesheet on the SG. These accelerometer attachment locations are provided for future use.

The signals from the transducers are amplified inside the Containment and are then directed to the data acquisition and analysis system in the Control Room. Signals that exceed a fixed and floating setpoint value actuate an alarm. Additional information may be obtained from an analysis module that records and analyzes the signals from all eight channels. The audio output of any one of the eight channels can also be monitored through the use of a loudspeaker and a selector switch.

4.4.3 DESIGN ANALYSIS

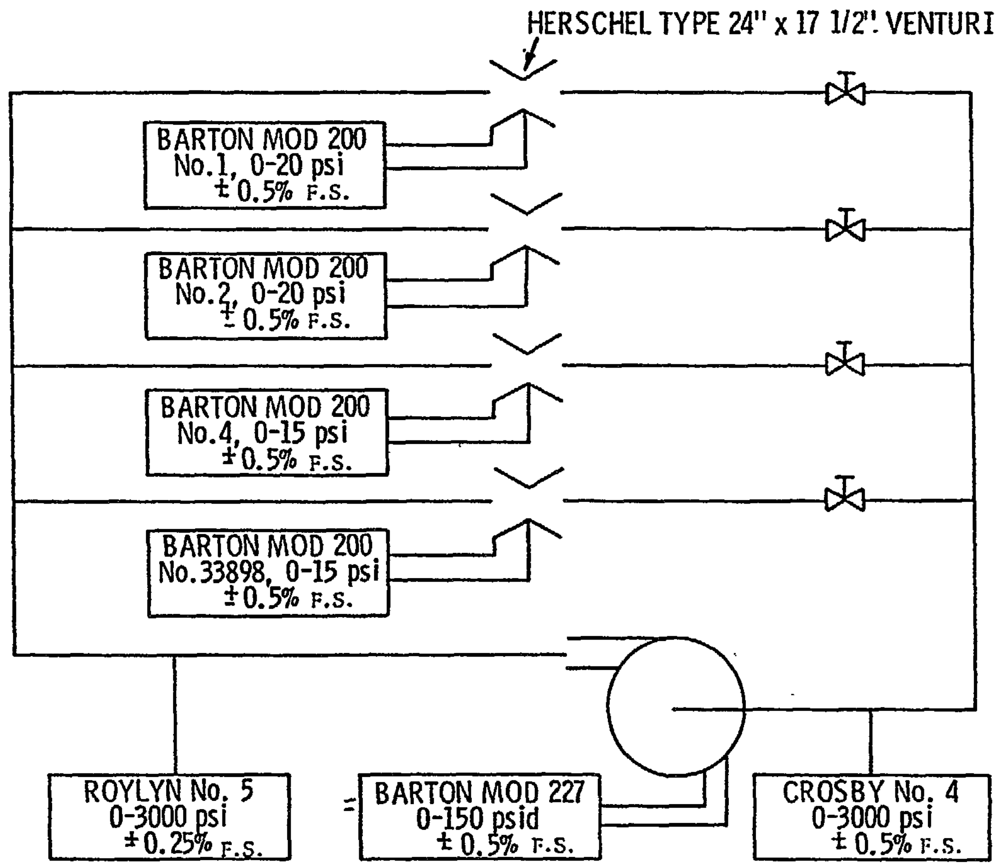
Experience in laboratories and on operating plants has shown that the signal-to-noise ratio resulting from a loose part with an energy level of one-half foot-pound or more is of sufficient magnitude that the loose part will be detected by the loose parts monitoring system.

4.5 COASTDOWN OPERATION AT END OF CYCLE

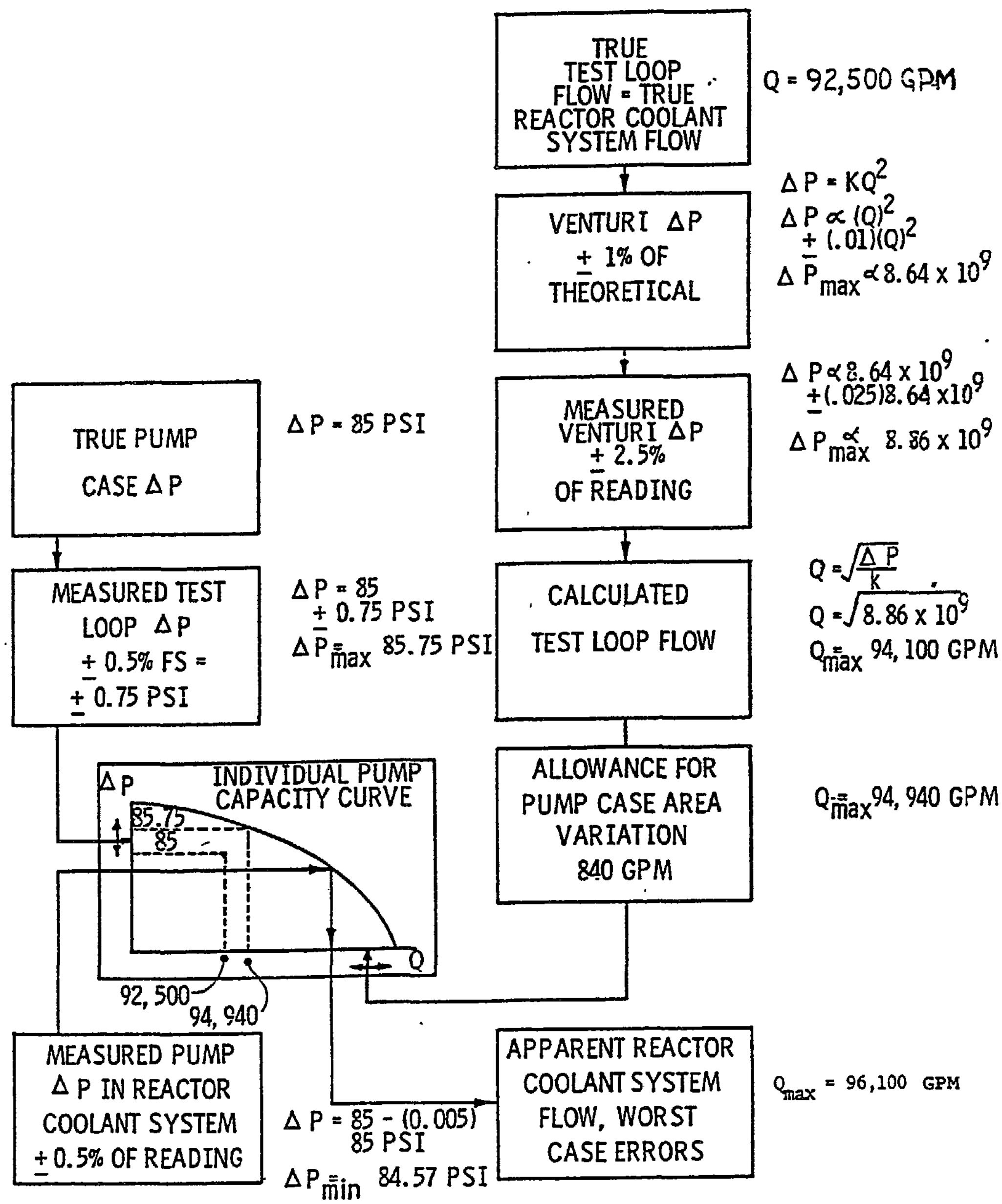
Toward the very end of a fuel cycle, the reactor core may reach a point at which it no longer has sufficient nuclear fuel to allow full power operation under normal operating conditions. Operation beyond that point is called coastdown operation. During coastdown operation, the reactor thermal power gradually decreases. However, it is possible to minimize (or even delay the start of) the loss of thermal power during the coastdown operation by reducing the RCS inlet temperature. Reducing the RCS inlet temperature at end of cycle adds positive reactivity to the core by taking advantage of the negative moderator temperature coefficient.

The reload safety analyses support both a RCS bulk inlet temperature coastdown and a thermal power coastdown. The inlet temperature may be reduced to 537°F. As long as the inlet temperature remains ≥ 537°F, the thermal power level shall be ≤ 100% but ≥ the inlet temperature/power program limit. To operate below 537°F or below 31.25% power, the plant must be on the inlet temperature/power program shown in Figure 4-9. The coastdown must end when the burnup reaches the cycle specific burnup limit.

Figure 4-18 shows the allowed combination of thermal power and RCS inlet temperature during coastdown operation.

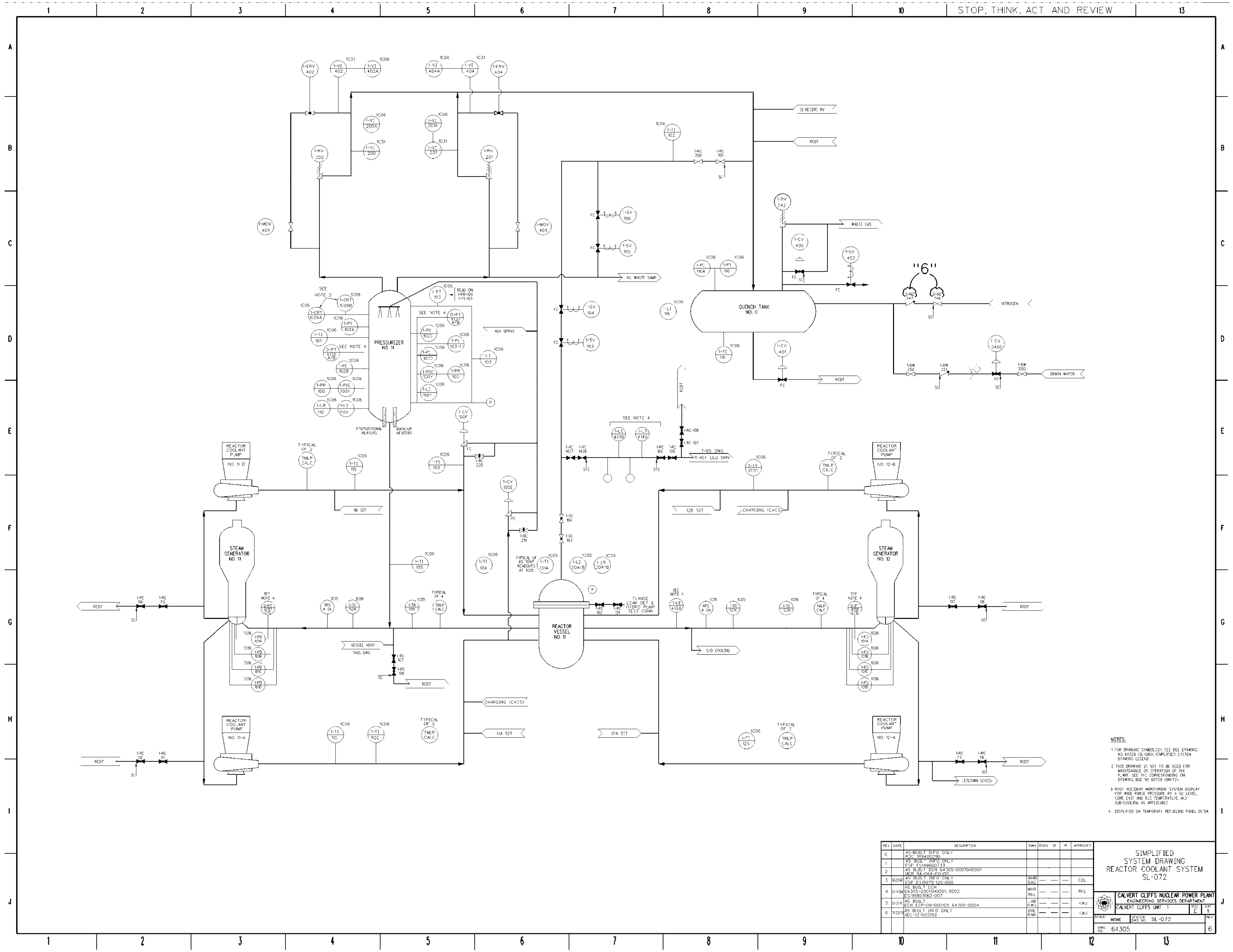


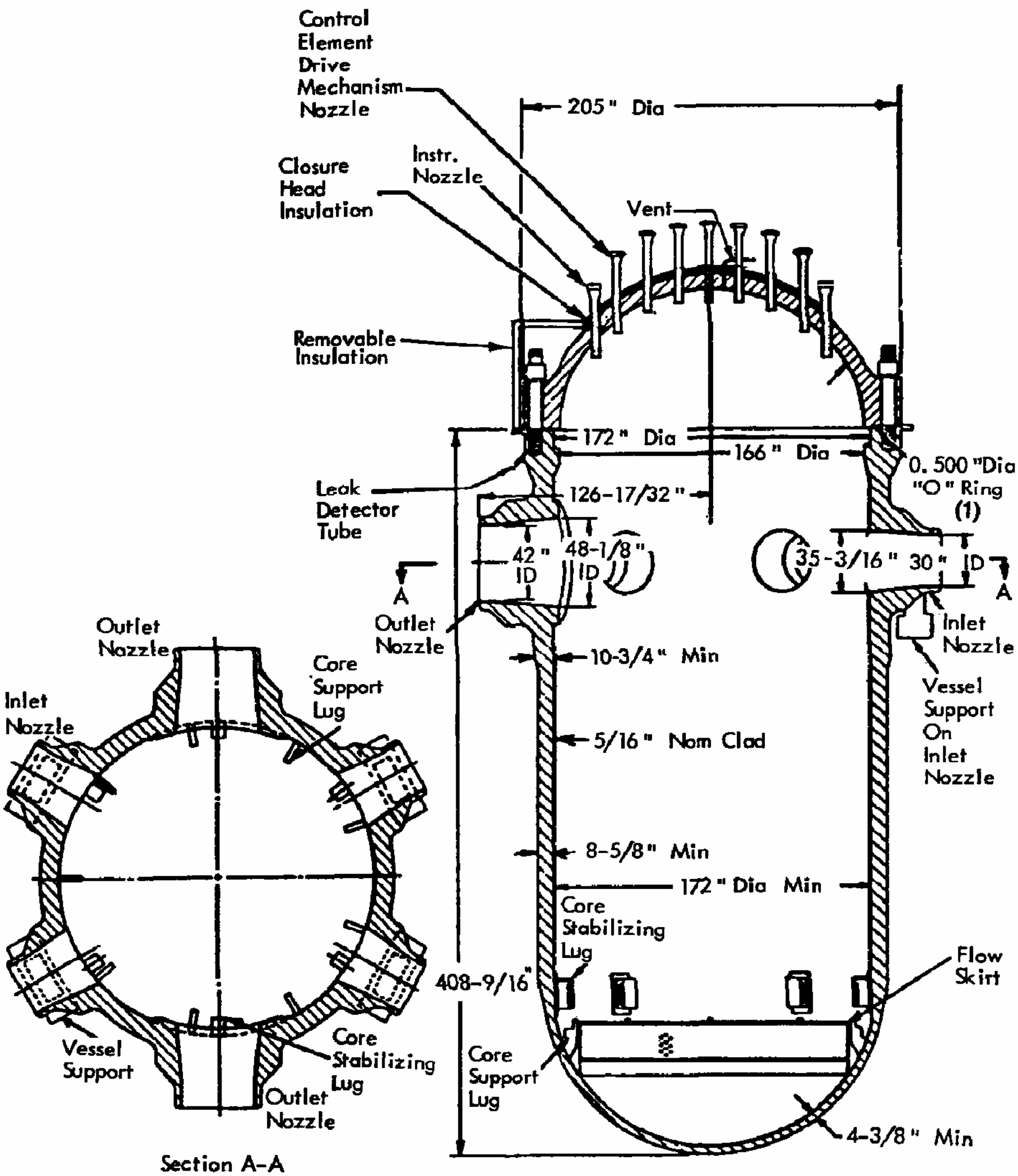
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	REACTOR COOLANT PUMP FLOW TEST INSTRUMENTATION	Figure A Rev. 18



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	ERROR IN ESTABLISHMENT OF FLOW	**Figure B** **Rev. 18**

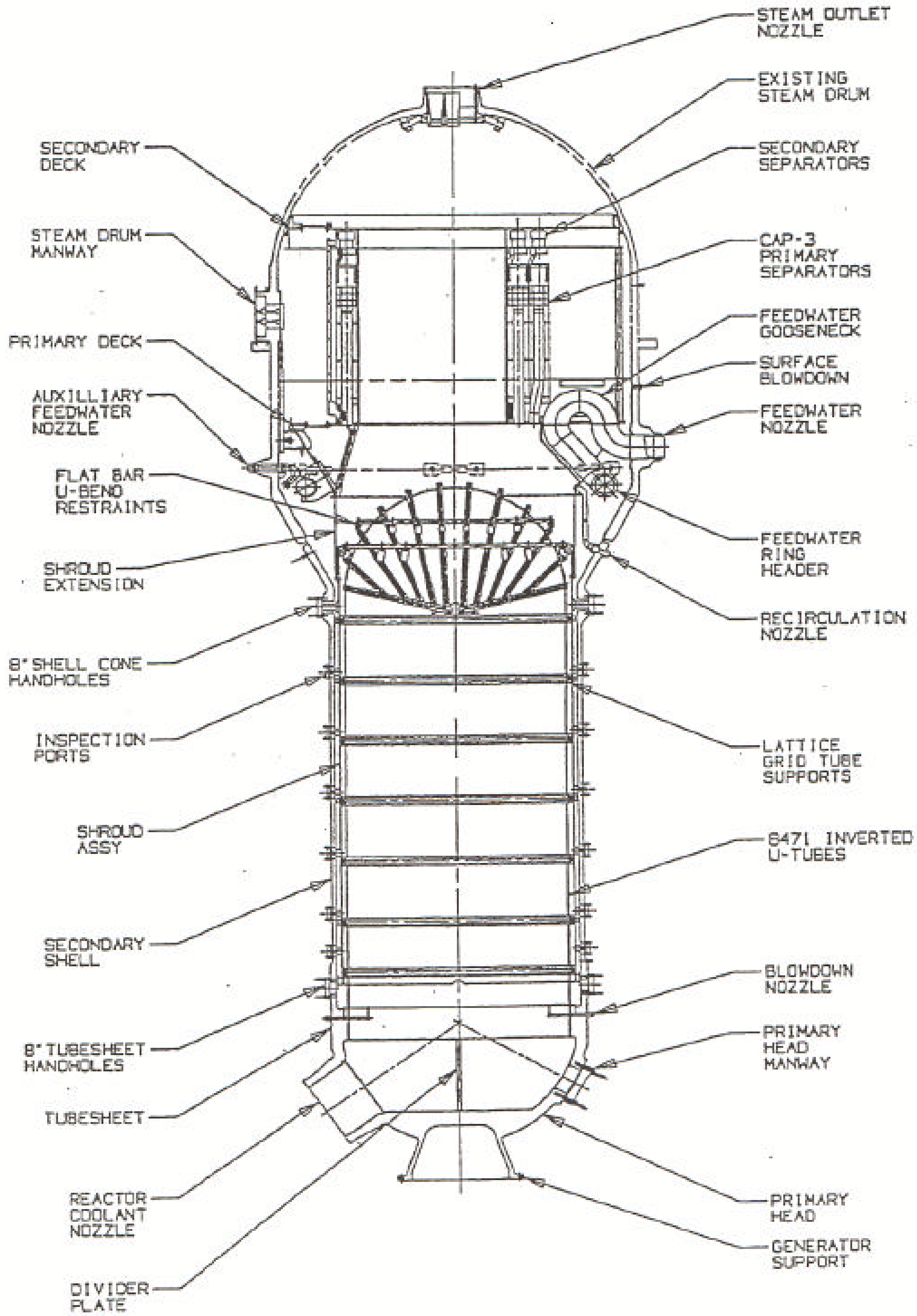
FIGURE 4-1 REACTOR COOLANT SYSTEM – UNIT 1



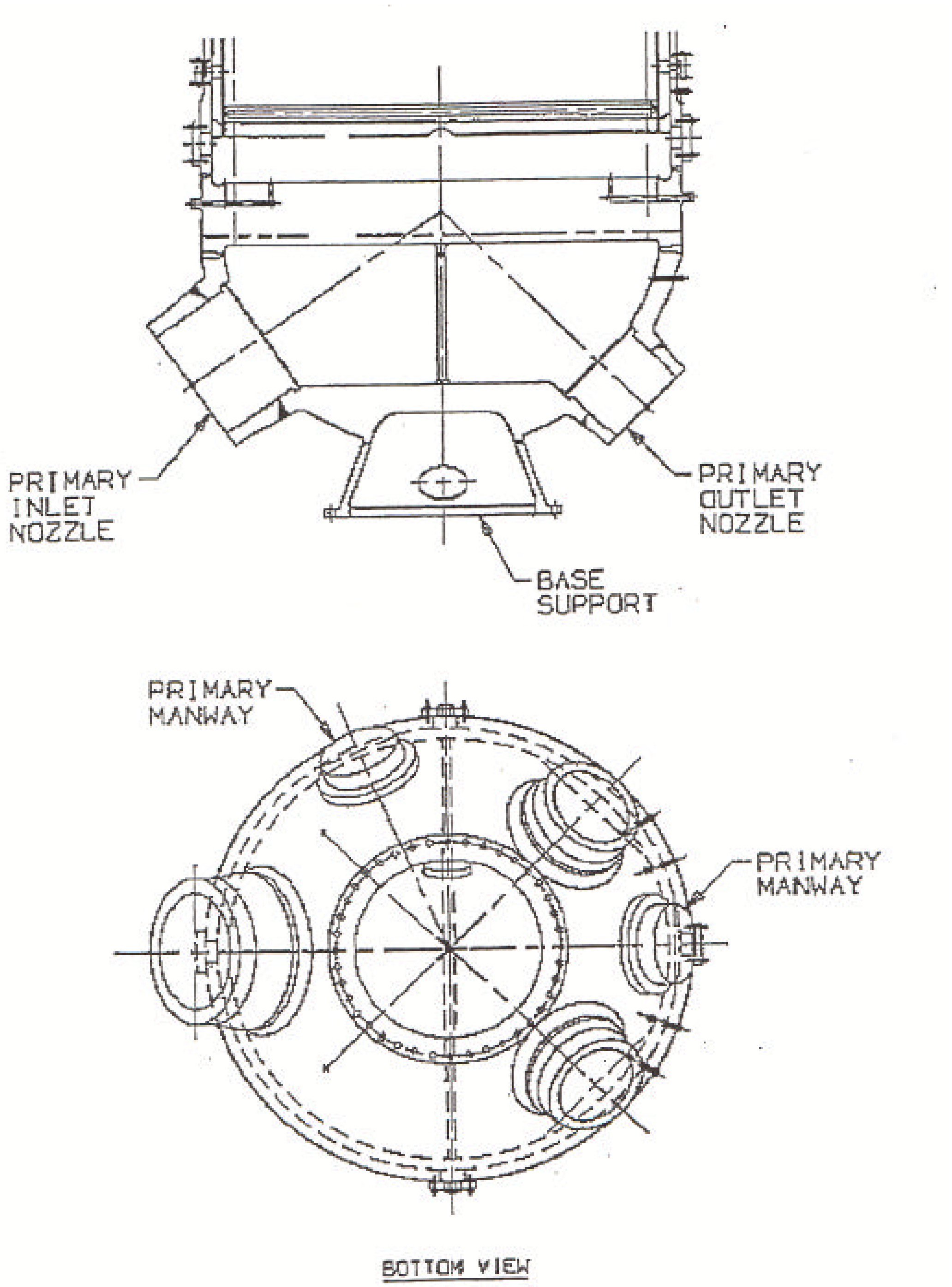


(1) The actual dimension depends on manufacturer's design and manufacturing tolerances.

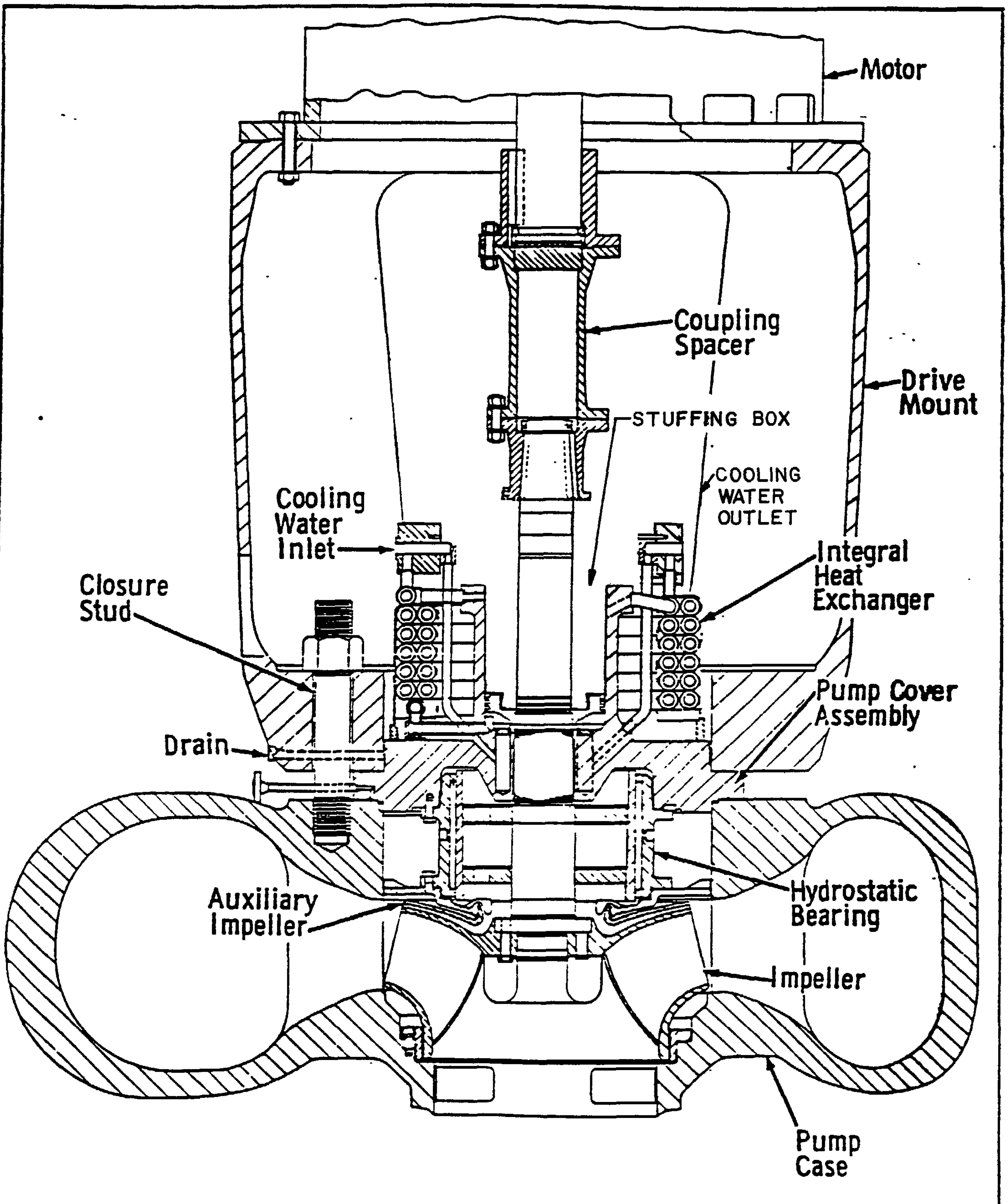
Calvert Cliffs Nuclear Power Plant	REACTOR VESSEL	Figure 4-2 Rev. 38



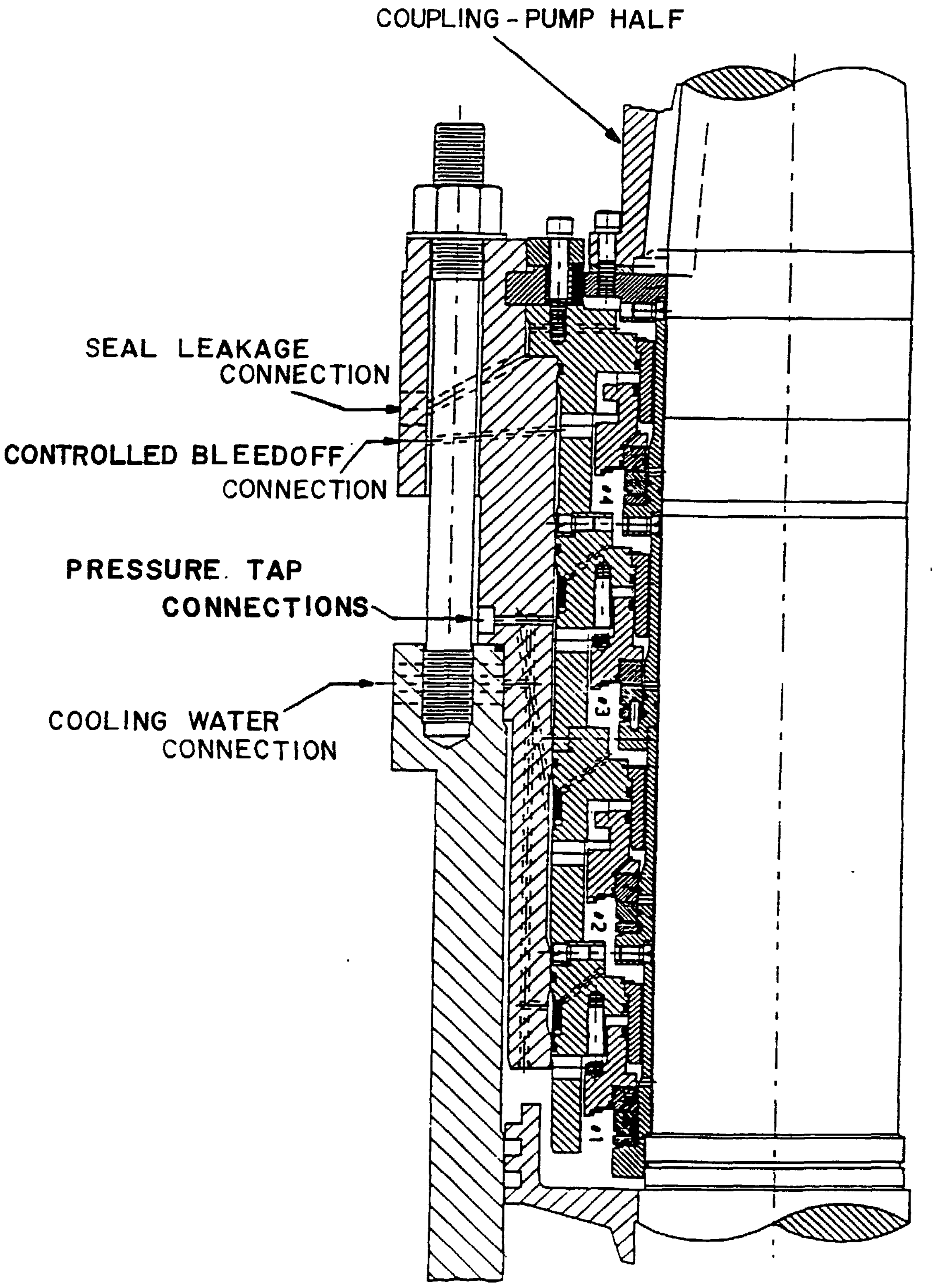
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR PROFILE VIEW	Figure 4-3A Revision 33



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR PRIMARY HEAD DESIGN	Figure 4-3B Revision 33

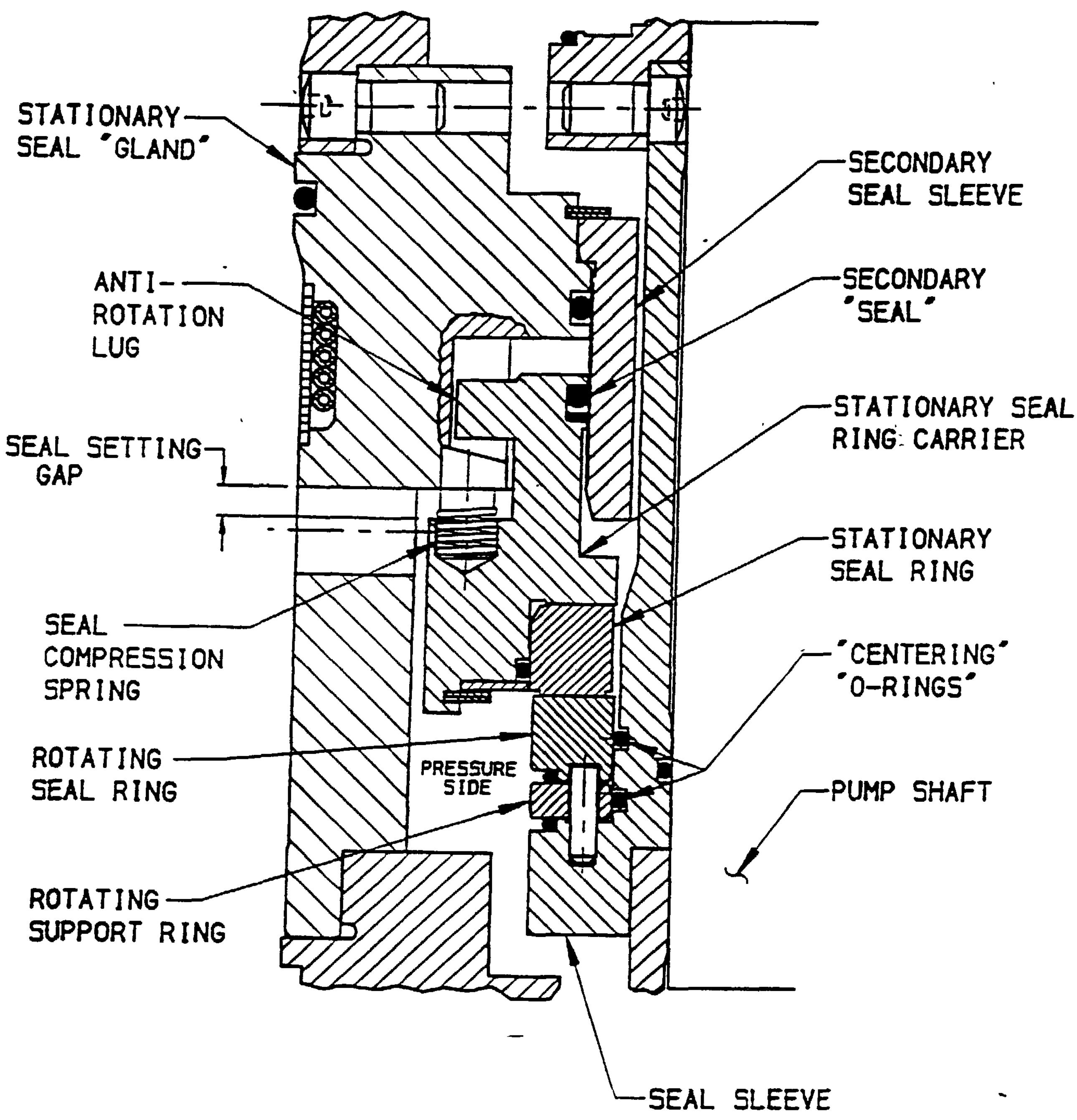


BALTIMORE GAS & ELECTRIC Co. Calvert Cliffs Nuclear Power Plant	Reactor Coolant Pump (FOR SEAL AREA SEE FIG. 4-6 AND FIG.4.6A) REV.11 1/91	Figure 4-4



BALTIMORE GAS & ELECTRIC Co. Calvert Cliffs Nuclear Power Plant	Reactor Coolant Pump - Seal Area	Figure 4-6
	REV.11 1/91	

BALANCED STATOR™ SEAL DESIGN FEATURES



BALTIMORE GAS & ELECTRIC Co. Calvert Cliffs Nuclear Power Plant	Reactor Coolant Pump - Seal Area REV. 11 1/91	Figure 4-6A

BALTIMORE GAS & ELECTRIC CO.
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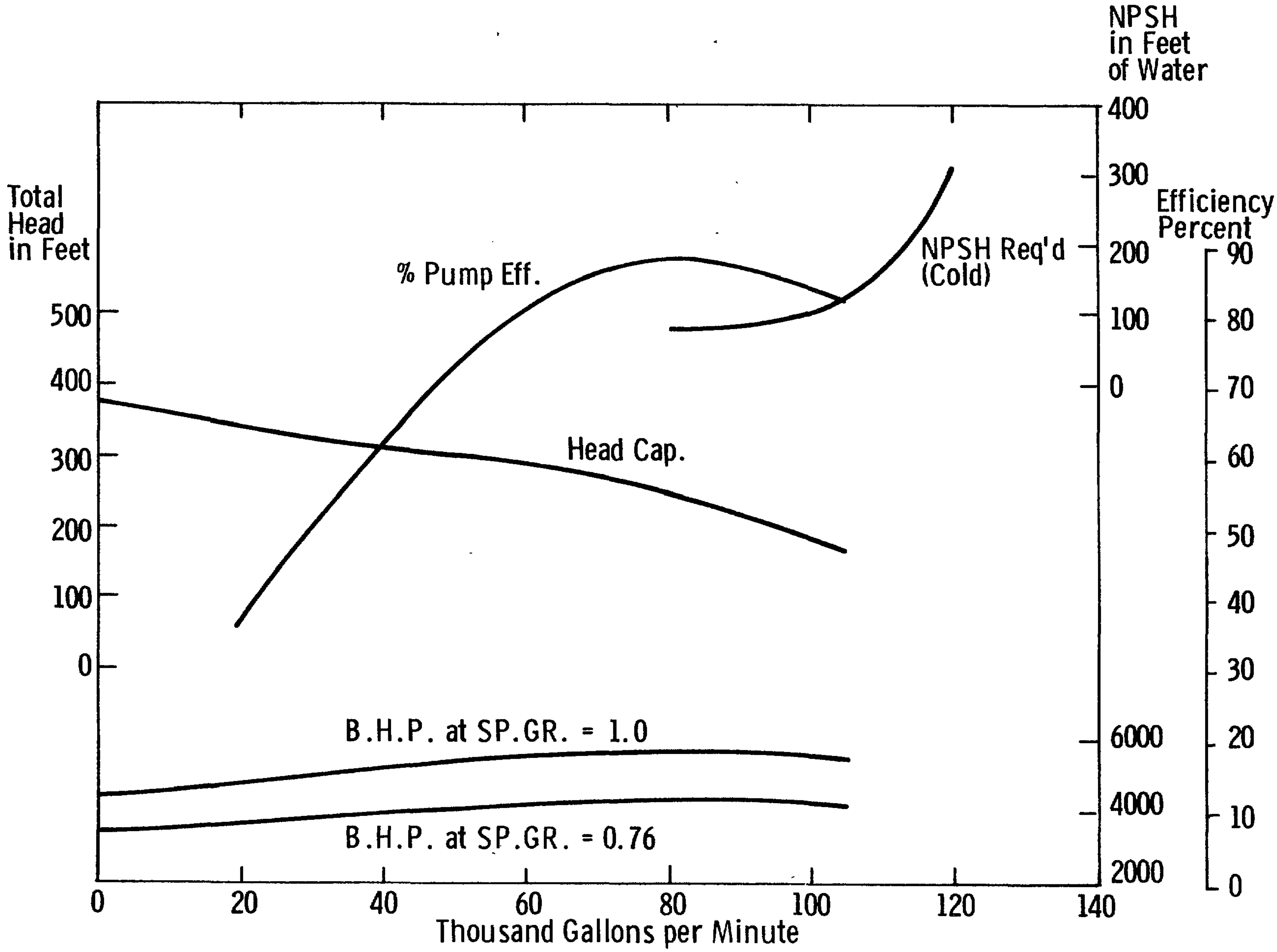
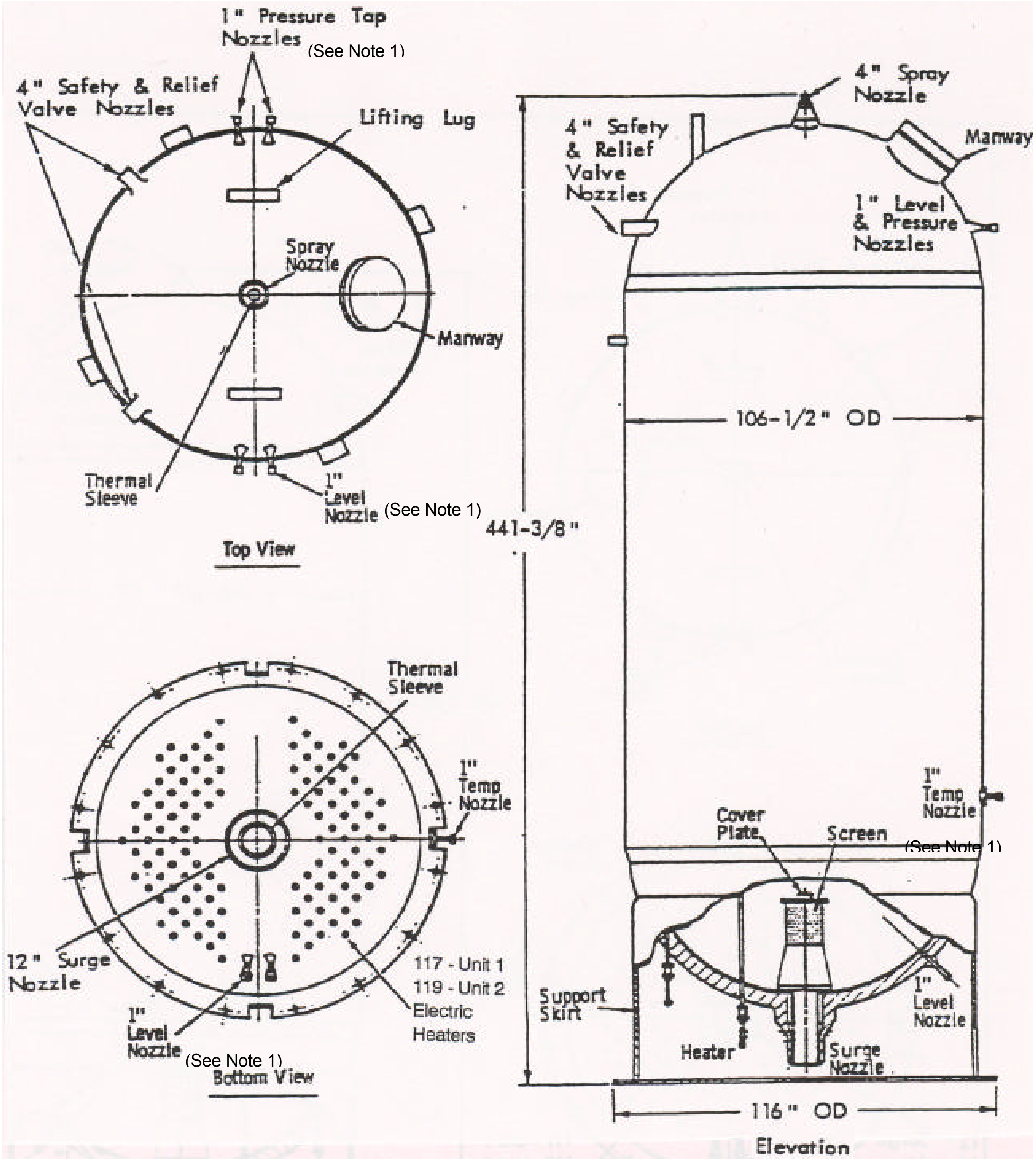
Nuclear Power Plant



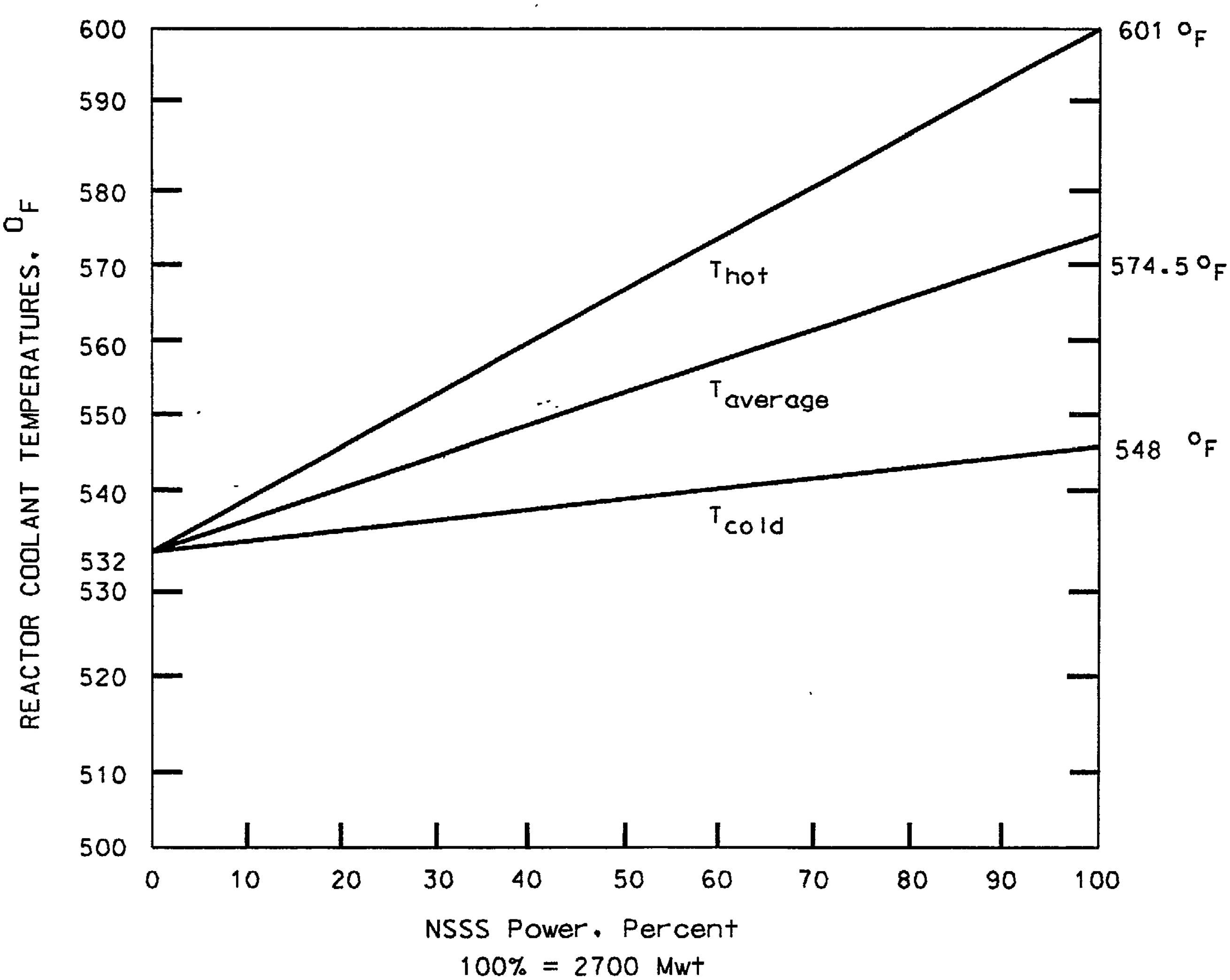
Reactor Coolant Pump Performance

Figure 4-7



Note 1: Mechanical nozzle seal assemblies may be installed on the 1" instrument nozzles in place of the "J-Groove" welds. There are restrictions for use on leaking nozzles. See Safety Evaluation for NRC Code Relief Request approval dated August 7, 2002.

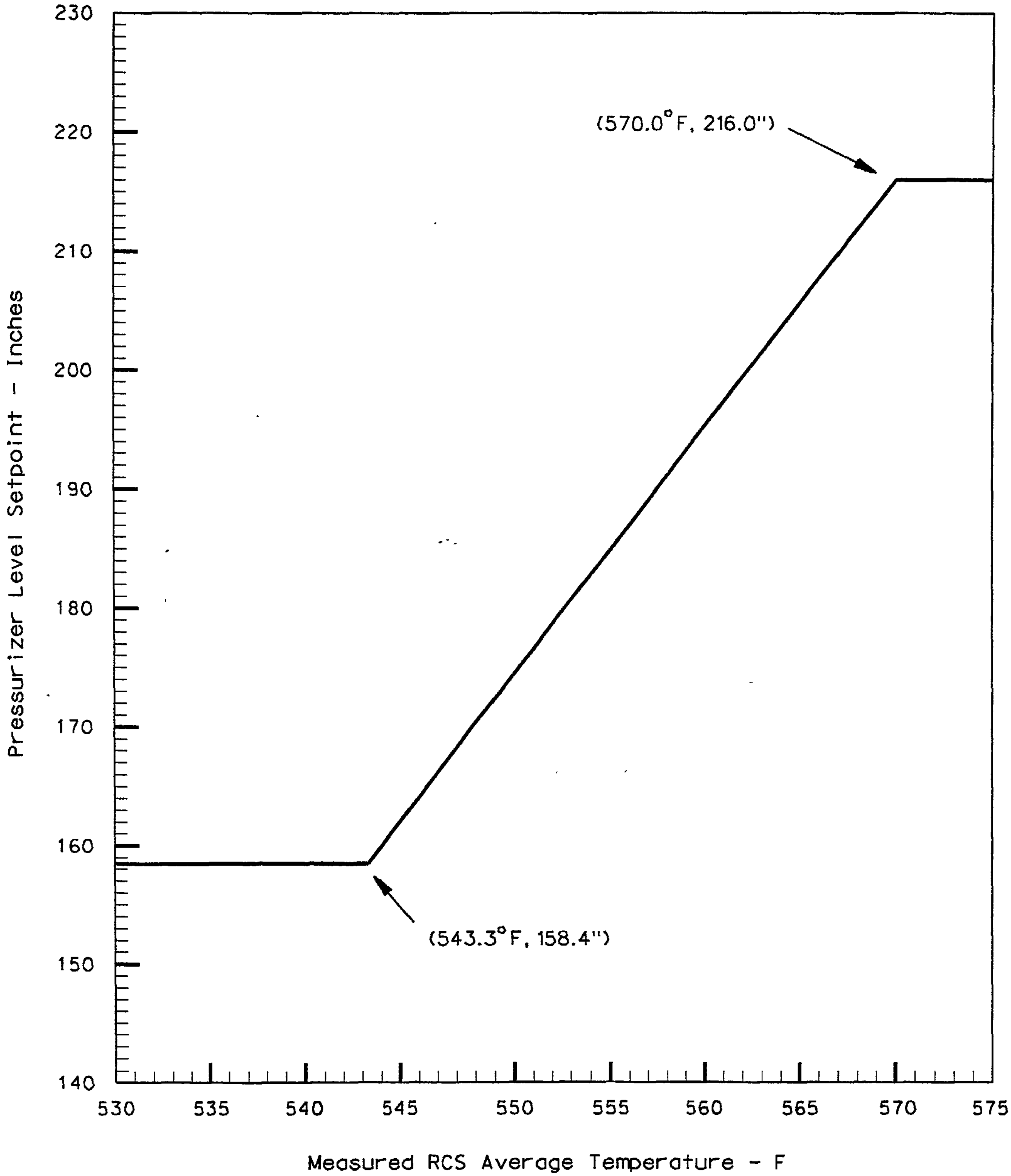
Calvert Cliffs Nuclear Power Plant	PRESSURIZER	Figure 4-8 Revision 33



NOTE:

THIS FIGURE SHOWS MAXIMUM T_{hot} AND $T_{average}$ ASSUMING MINIMUM RCS FLOW.

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	TEMPERATURE CONTROL PROGRAM	REV. 24	Figure 4-9

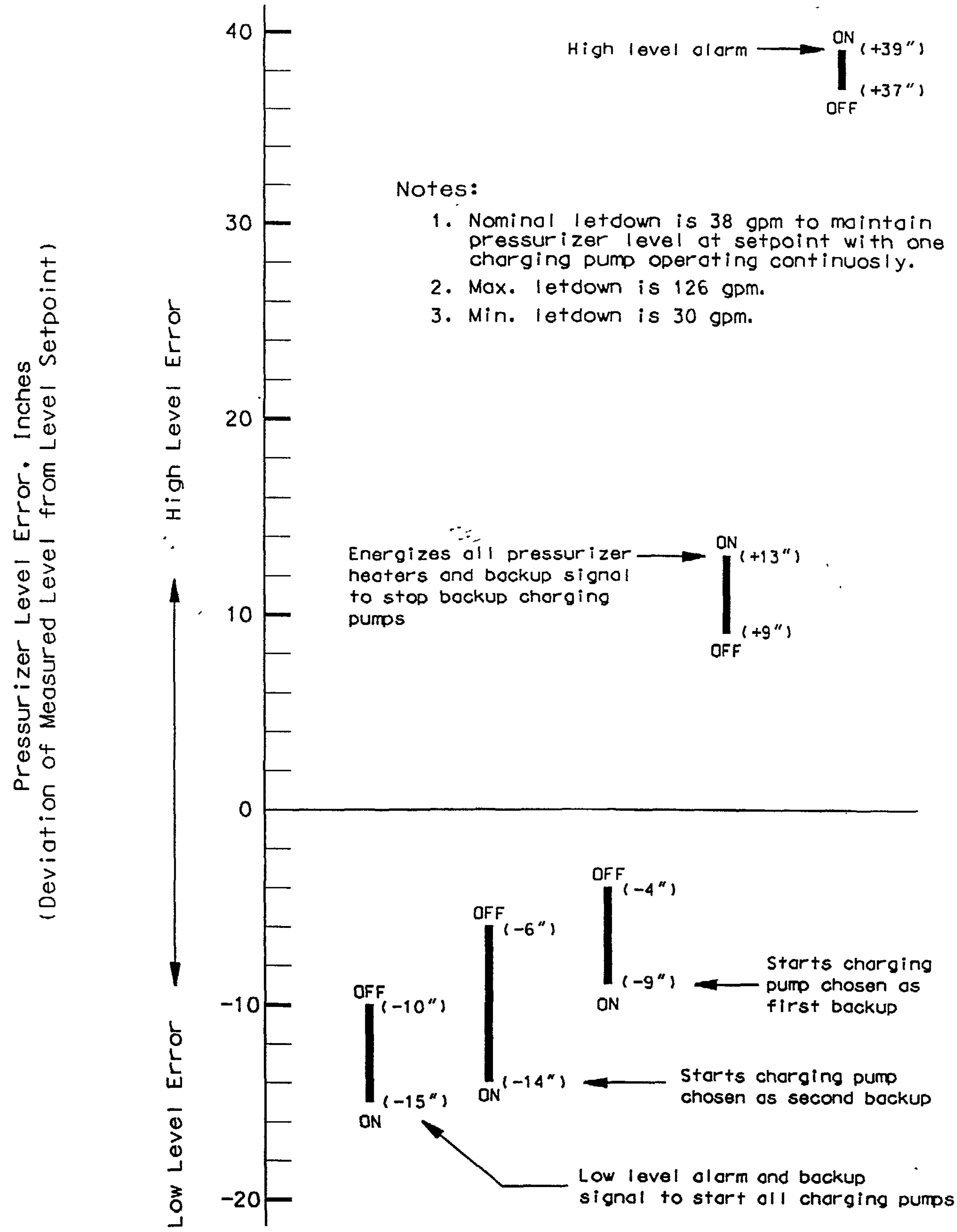


NOTE:

THE PRESSURIZER LEVEL SET POINT PROGRAM MAY BE ADJUSTED FOR THE INCREASE IN Tave CAUSED BY SG TUBE PLUGGING.

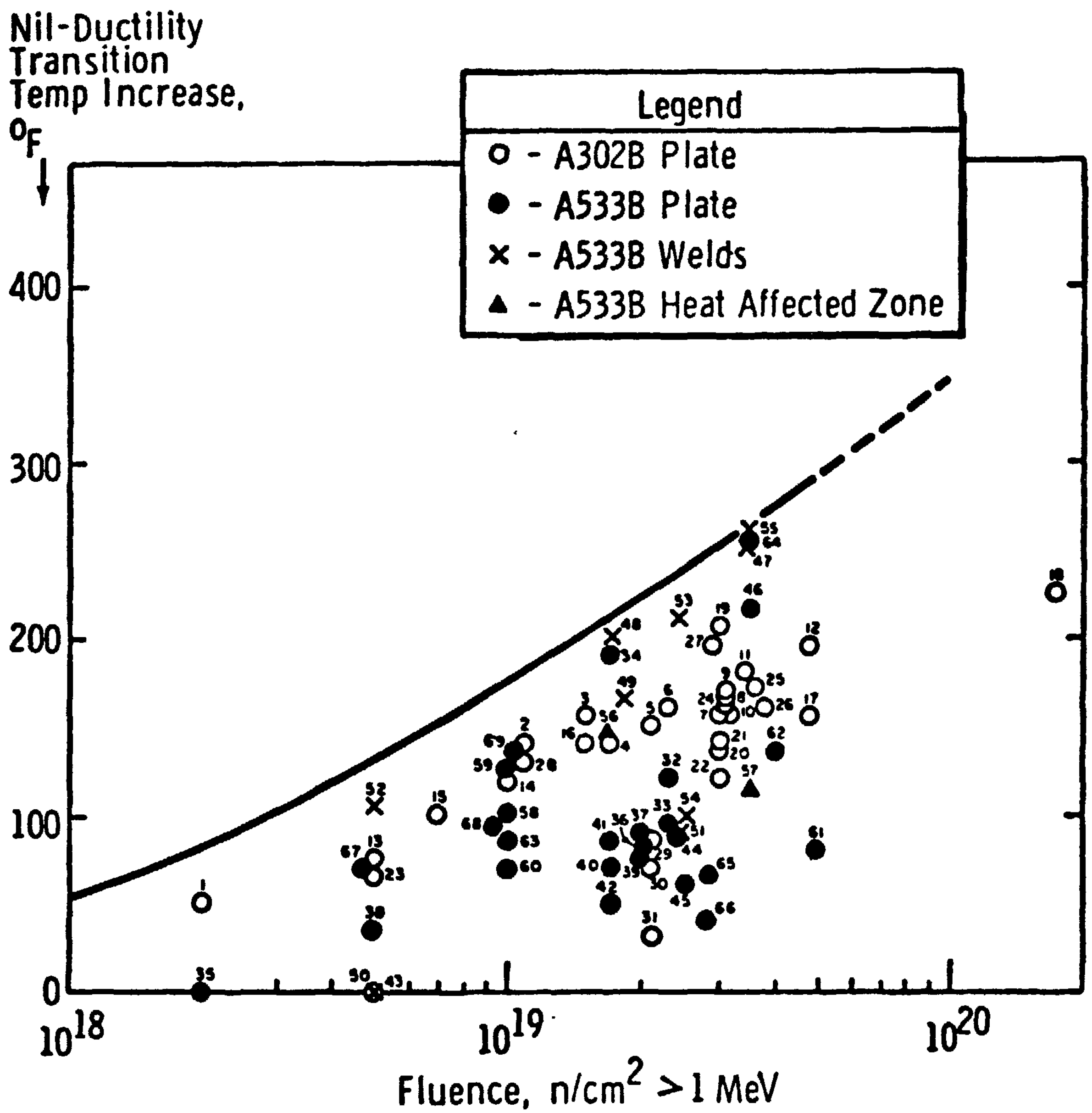
REV. 24

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Pressurizer Level Set Point Program	Figure 4-10

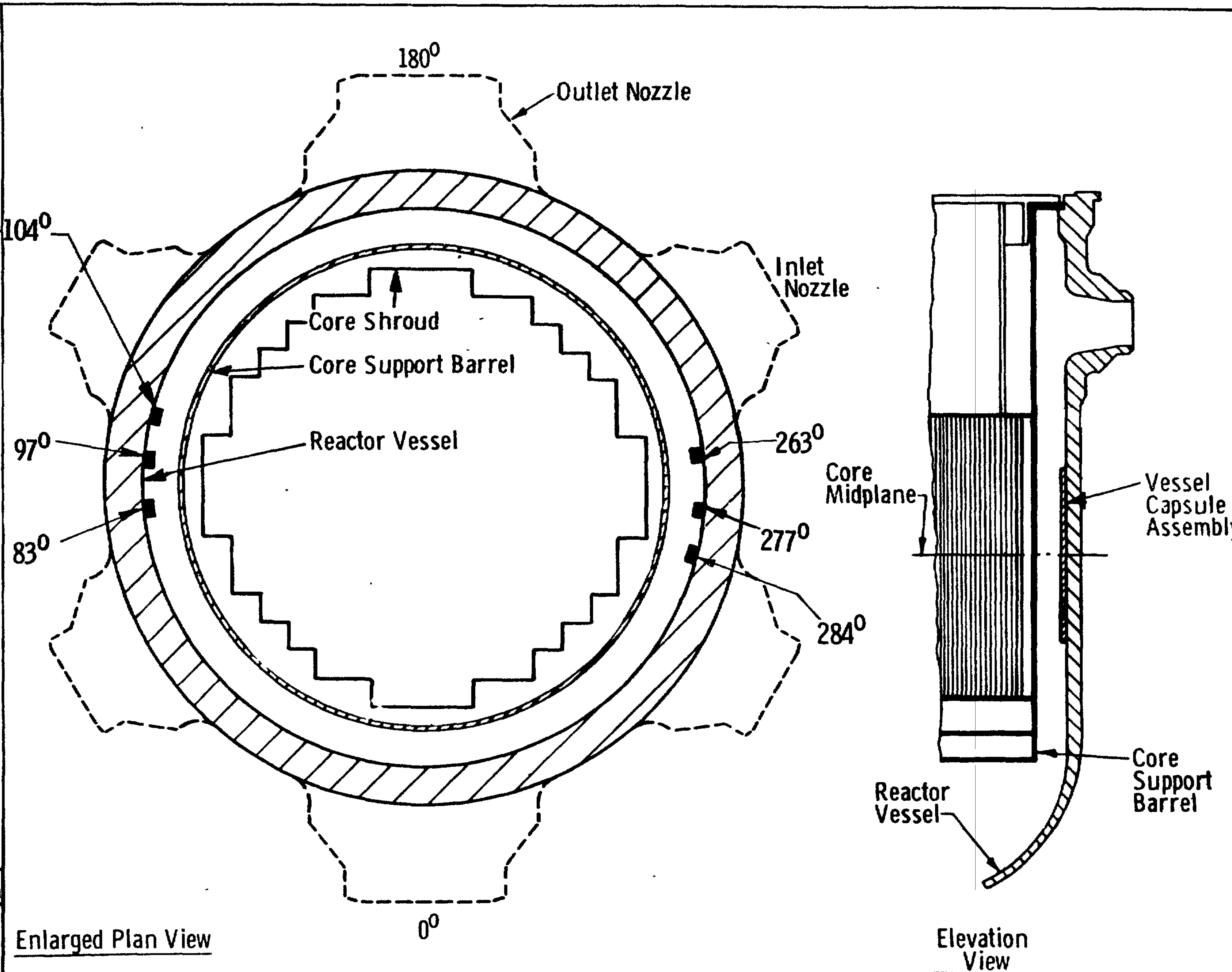


Revision 24

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Pressurizer Level Control Program	Figure 4-11



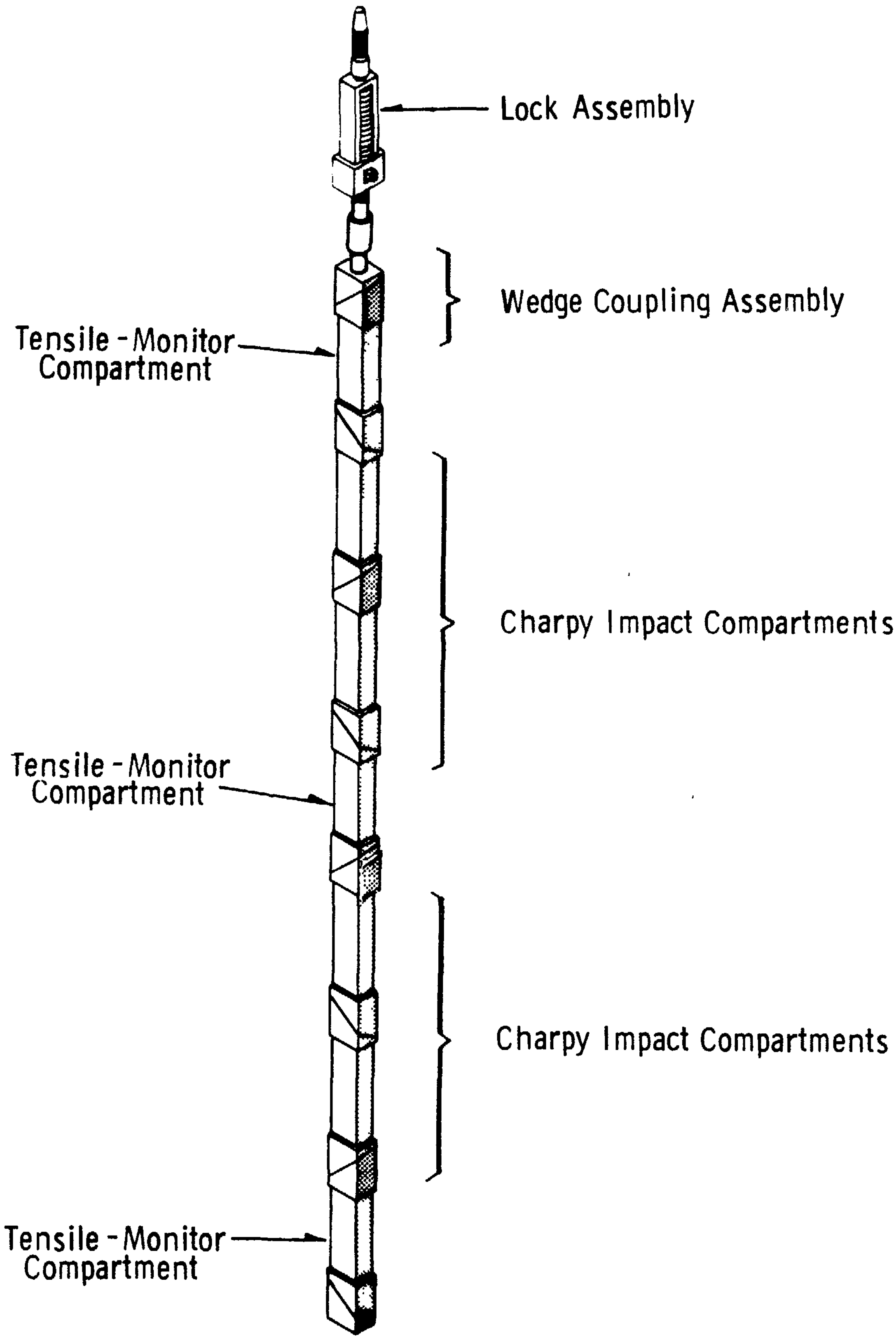
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	C-E Design Curve of NDTT Increase (550^0F Irradiation)	Figure 4-12



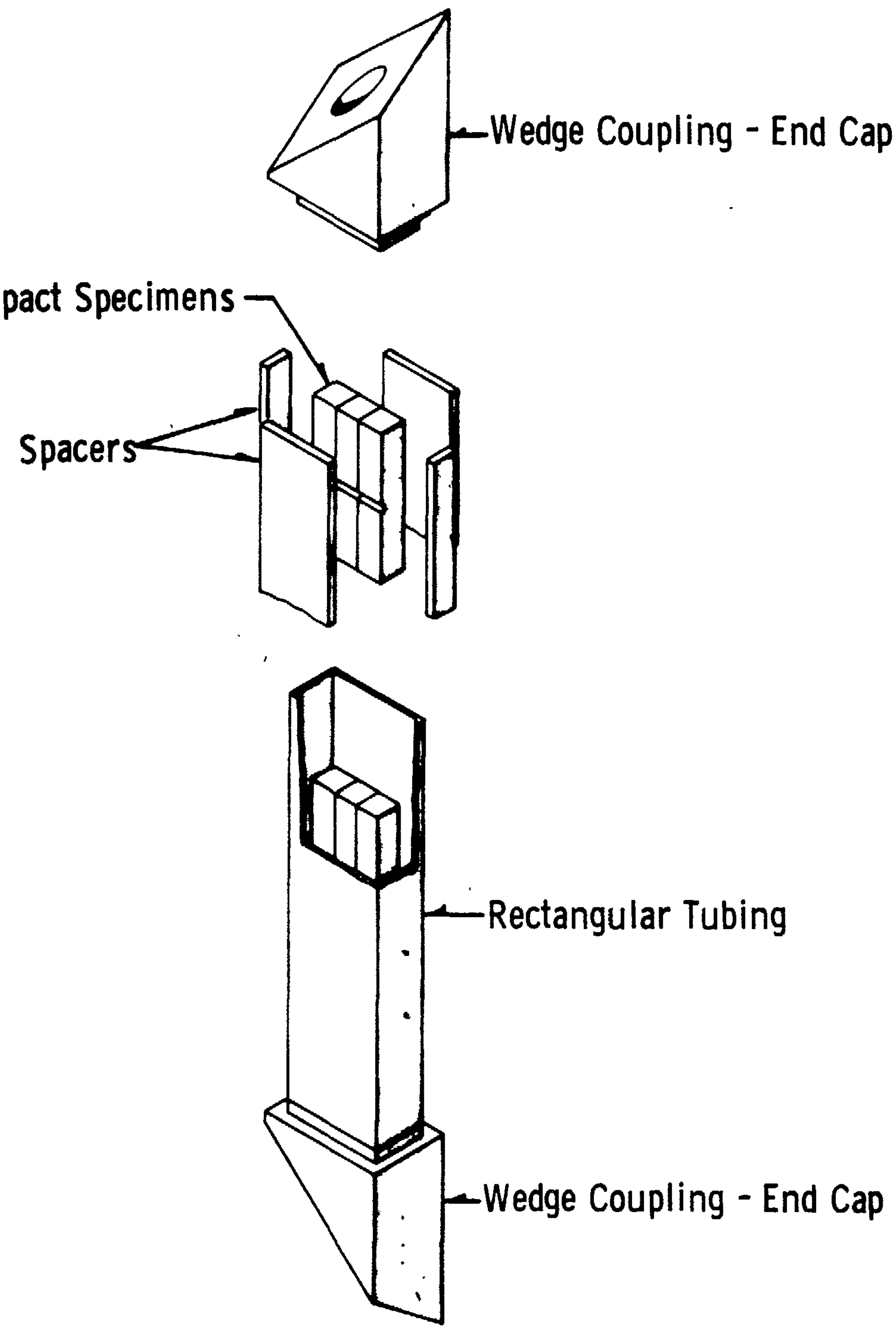
BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

Location of Surveillance Capsule Assemblies

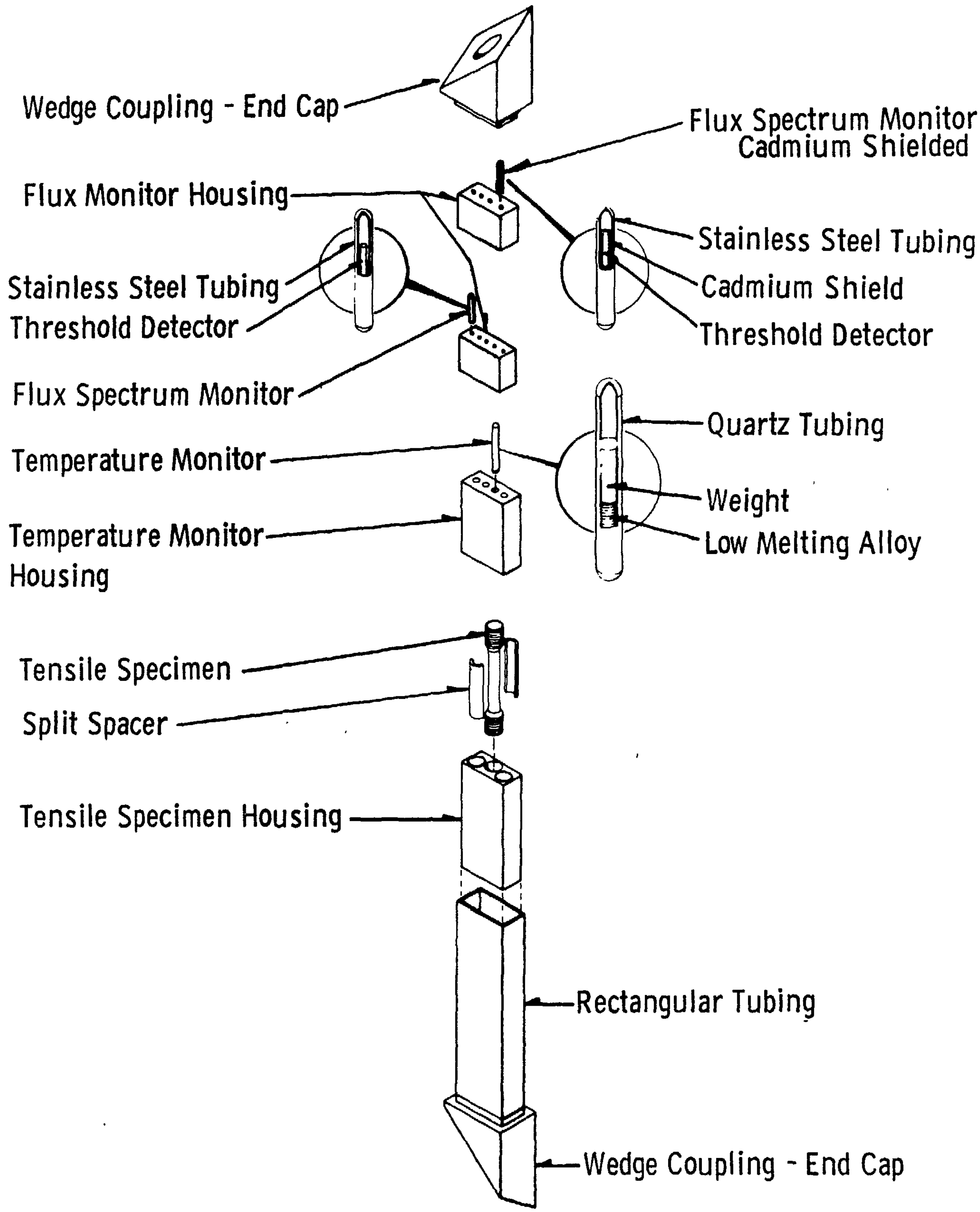
Figure 4-13



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Typical Surveillance Capsule Assembly	Figure 4-14

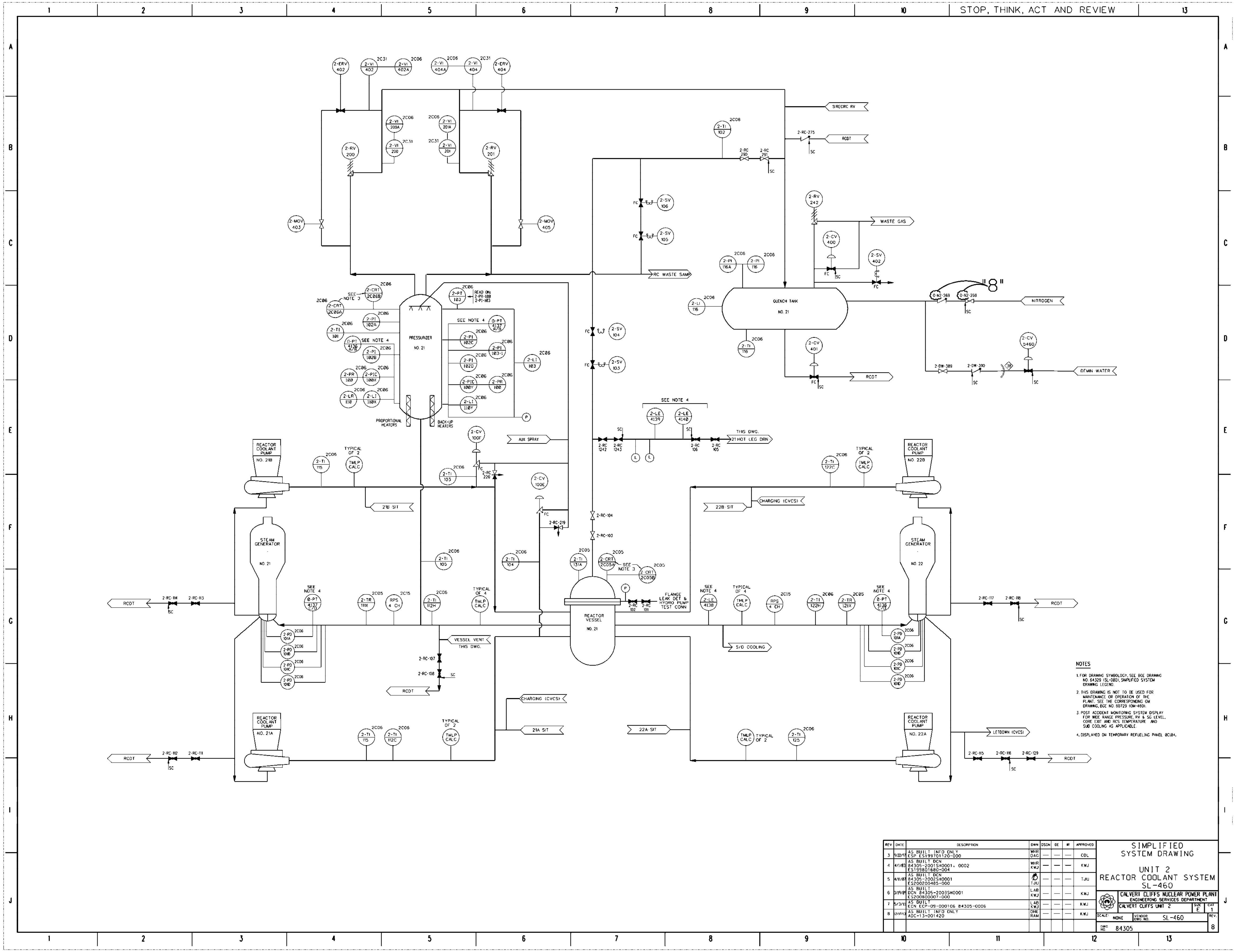


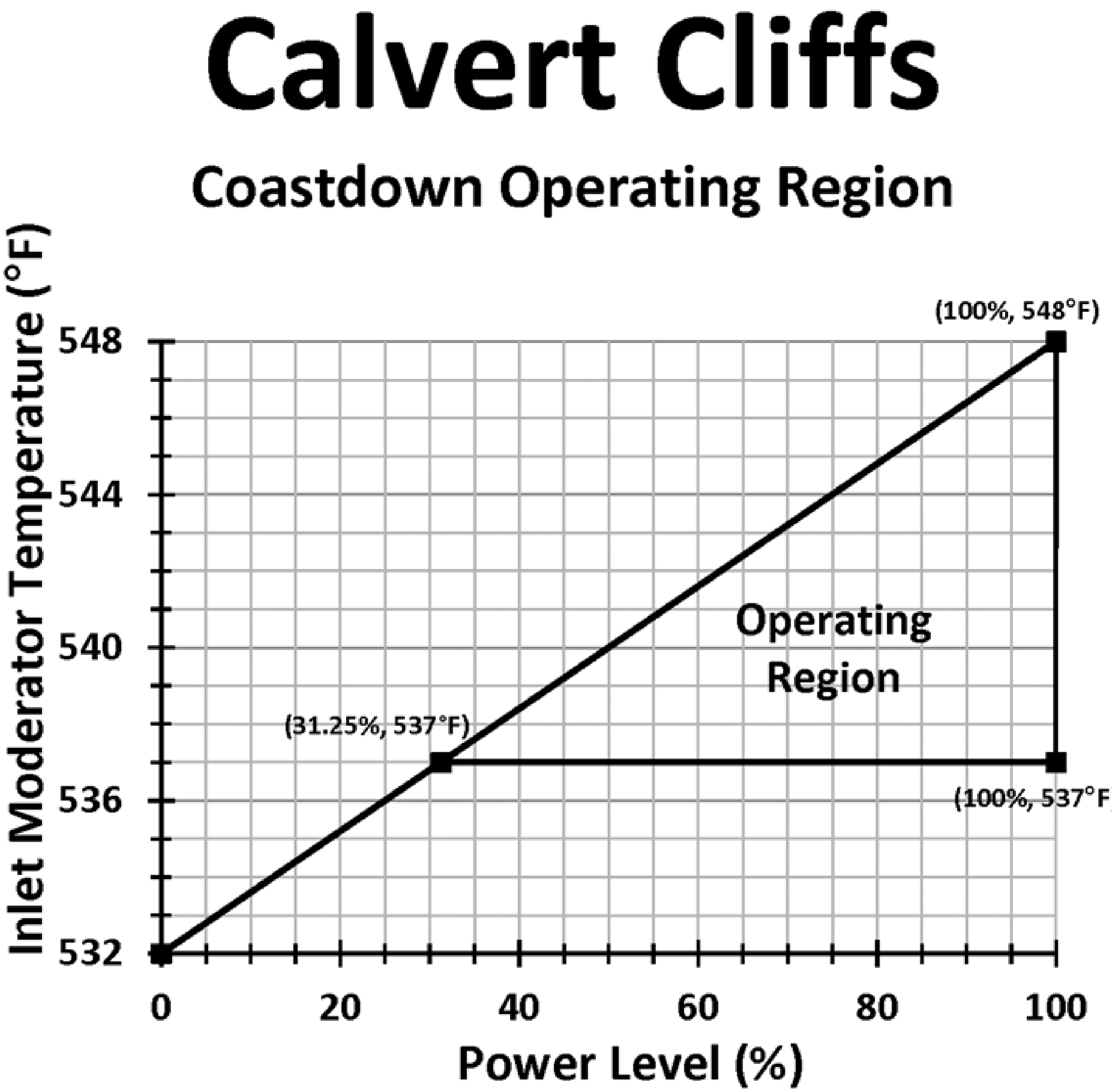
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Typical Charpy Impact Compartment Assembly	Figure 4-15



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Typical Tensile-Monitor Compartment Assembly	Figure 4-16

FIGURE 4-17 REACTOR COOLANT SYSTEM – UNIT 2





Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS COASTDOWN OPERATING REGION	Figure 4-18 Revision 49

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STRUCTURES

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(NOTE: Appendices 5A, 5B, 5C, 5D, and 5E are referenced here, but located in UFSAR Volume III.)

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LIST OF ACRONYMS

ACI	American Concrete Institute
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASA	American Standards Association
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
BBRV	
FSAR	Final Safety Analysis Report
HVAC	Heating, Ventilation, and Air Conditioning
LOCA	Loss-of-Coolant Accident
NDTT	Nil-Ductility Transition Temperature
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
RCS	Reactor Coolant System
SBO	Station Blackout
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
UBC	Uniform Building Code
USI	Unresolved Safety Issue

5.0 STRUCTURES

5.1 CONTAINMENT STRUCTURE

5.1.1 DESIGN BASIS

General plans at various elevations and sections through the Containment Structure interior are shown in Figures 1-6 through 1-15 and show the general arrangement of various equipment such as reactor, steam generators, pressurizer and reactor coolant pumps. The support and anchorage details of these components are shown in Figures 5-11 through 5-15. At an Elevation 177'0", a landing platform, to the ladder on the vent stack provides an access route to the top of the Containment Structure. A steel ladder, provided inside the Containment Structure provides an access up to the polar crane. The Containment Structure is a Seismic Category I structure and is designed for all loading combinations described in Section 5A.3.

The Containment Structure completely encloses the Reactor Coolant System (RCS) to minimize release of radioactive material to the environment should a serious failure of the RCS occur. The structure provides adequate biological shielding for both normal operation and accident situations. The Containment Structure is designed for maximum of 0.16%/day leakage by weight of the original content of air at a design pressure of 50 psig and a concrete surface temperature of 276°F.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss-of-coolant accident (LOCA) with no loss of integrity. In this event, the total energy contained in the water of the RCS is assumed to be released into the Containment Structure through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks.

Energy is available for release into the Containment Structure from the following sources:

RCS Stored Heat
Reactor Stored Heat
Reactor Decay Heat
Metal-Water Reactions

The energy release and the containment pressure transient curves are shown in Section 14.20.

The design of the engineered safety features systems and their operation is discussed more fully in Chapter 6; only their relation to the basis of Containment Structure design is discussed below. The engineered safety features systems are provided to limit the consequences of an accident. Their energy removal capabilities limit the internal pressure so that Containment Structure design limits are not exceeded and the potential for release of fission products is minimized.

The safety injection systems inject borated water into the reactor vessel to remove core decay heat and to minimize metal-water reactions and the associated release of heat and fission products. Flashed primary coolant, RCS sensible heat, and core decay heat transferred to the Containment Structure are removed by the Containment Cooling System which is comprised of two subsystems; the containment spray subsystem and the air recirculation subsystem.

The containment spray subsystem reduces pressure in the containment by condensing the Containment Structure steam and removing heat from the containment atmosphere by recirculation of the spray water through the shutdown cooling heat exchangers.

The air recirculation subsystem reduces pressure and removes heat directly from the Containment Structure atmosphere to the Service Water System with recirculating fans and cooling coils.

5.1.2 DESIGN CRITERIA

5.1.2.1 General Description

The Containment Structure houses the RCS. Its purpose is to contain any accidental release of radioactivity from the RCS. It is designated as a Seismic Category I structure.

The basic design criteria are that the integrity of the liner plate be guaranteed under all loading conditions and the structure shall have a low-strain elastic response such that its behavior will be predictable under all design loadings.

The structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab as shown in Figure 5-1. The entire interior surface of the structure was lined with a 1/4" thick welded American Society for Testing and Materials (ASTM) A36 steel plate to assure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the Containment Structure wall through welded steel penetrations as shown in Figures 5-2 and 5-3. The penetrations and access openings were designed, fabricated, inspected, and installed in accordance with the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Section III, Class B.

Principal dimensions of the Containment Structure are:

Inside Diameter	130'
Inside Height (including Dome)	181-2/3'
Vertical Wall Thickness	3-3/4'
Dome thickness	3-1/4'
Foundation Slab Thickness	10'
Liner Plate Thickness	1/4"
Internal Free Volume	1,989,000 ft^3

The Containment Structure is shown in Figures 1-6 through 1-16.

In the concept of a post-tensioned Containment Structure, the internal pressure load is balanced by the application of an opposing external force on the structure. Sufficient post-tensioning was used on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the LOCA pressure. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress. Nominal, bonded reinforcing steel was also provided to distribute strains due to shrinkage and temperature. Additional bonded reinforcing steel was used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning was not required to exert an external pressure for this portion of the structure.

The post-tensioning system consists of:

a. Three groups of 68 dome tendons oriented at 60° to each other for a total of 204 tendons anchored at the vertical face of the dome ring girder.
b. Two hundred four vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.
c. Six groups of 78 hoop tendons, each enclosing 120° of arc, for a total of 468 tendons anchored at the 6 vertical buttresses.

Each tendon consists of approximately 90 1/4" diameter wires with button-headed BBRV-type anchorages, furnished by the Prescon Corporation. The tendons are housed in spiral wrapped, corrugated, thin-wall, carbon steel sheathing. After fabrication, the tendon was shop dipped in a petrolatum corrosion protection material, bagged and shipped. After installation, the tendon sheathing was filled with a corrosion preventive grease (Viconorust 2090P). The ends of all tendons were covered with pressure-tight, grease-filled caps for corrosion protection. All the vertical tendons for each Unit received new corrosion preventive grease between 1997 and the end of 2002. Some original vertical tendons for each Unit were restressed or replaced with a new tendon between 2001 and 2002. See Appendix 5E for details.

American Society for Testing and Materials A615, Grade 60 reinforcing steel, mechanically spliced as needed with B- and T-series CADWELDS, was used throughout the foundation slab and around the large penetrations. The same type of steel was used for the bonded reinforcing throughout the cylinder and dome as crack control reinforcing and at areas of discontinuities to provide an additional margin of elastic strain capability.

The 1/4"-thick liner plate was attached to the concrete by means of an angle grid system stitch welded to the liner plate and embedded in the concrete. The details of the anchoring system are provided in Figure 5-1. The frequent anchoring is designed to prevent significant distortion of the liner plate during accident conditions and to insure that the liner maintains its leak-tight integrity. The design of the liner anchoring system also considers the various erection tolerances and their effect on its performance. The liner plate was protected from corrosion on the inside with 3 mils of inorganic zinc primer topped with 6 mils of an organic epoxy up to Elevation 75'0", and 3 mils of an inorganic topcoat above that elevation. There is no paint on the side in contact with concrete.

The aggregate used in the structure produced an excellent high-strength, dense, sound concrete. The 28-day design strengths were 5000 psi for the shell and 4000 psi for the foundation slab.

Personnel and equipment access to the structure is provided by a two-door personnel lock with double seals on both doors and by a 19'0" clear diameter, double gasketed, single-door equipment hatch as shown in Figure 5-3. A two-door emergency personnel escape lock is also provided. These locks and hatch were designed and fabricated from A516, Grade 70 firebox quality steel made to A300 specification and Charpy V-notch impact tested to 0°F in accordance with the ASME, B&PV Code, Section III. All piping penetrations furnished adhered to the same requirements.

A containment outage door on the exterior of each Containment Structure at the equipment hatch opening serves as a substitute for the equipment hatch when setting containment closure during Modes 5 and 6 conditions. The containment

outage door was designed, fabricated, examined, inspected and tested to the requirements of the 1995 Edition of ASME Section VIII, Rules for Construction of Pressure Vessels, Division 1. However, the containment outage door cannot be credited for severe weather. If Emergency Response Plan Implementation Procedure 3.0, Preparing for Severe Weather, is implemented and requires containment closure to be established, then the equipment hatch must be used.

Structural brackets provided for the Containment Structure polar crane runway were fabricated from ASTM A36 steel shapes and ASTM A516, Grade 70 insert plates (Figure 5-1). Like the penetration assemblies, structural brackets and thickened plates were shop fabricated, stress relieved and shipped to the job site for welding to the 1/4" liner plate.

The strength of the Containment Structure at working stress and overall yielding was compared to various loading combinations to assure safety. The Containment Structure was examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to assure proper performance. The structure was designed and constructed in accordance with design criteria based upon American Concrete Institute (ACI) 318-63, ACI 301-66, and ASME, B&PV Code, Sections III, VIII, and IX to meet the performance and strength requirements prior to prestressing, at transfer of prestress, under sustained prestress, at design loads, and at yield loads.

The structure was originally analyzed using Bechtel's Finite Element Program for Cracking Analysis CE 316-4, for individual and various combinations of loading cases of dead load, live load, prestress, temperature and pressure. The computer output included direct stresses, shear stresses, principal stresses and displacements of each nodal point. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

Stress plots which showed the total stresses from appropriate combinations of loading cases were made and areas of high stress were identified. The modulus of elasticity was corrected to account for the nonlinear stress-strain relationship at high compression in these areas and stresses were recomputed.

In order to consider creep deformation, the modulus of elasticity of concrete under sustained loads such as dead load and prestress was differentiated from the modulus of elasticity of concrete under instantaneous loads such as internal pressure and earthquake loads.

The forces and shears were added over the cross-section and the total moment, axial force and shear were determined. From these values, the straight-line elastic stresses were computed and compared to the allowable values. The ACI 318-63 design methods and allowable stresses were used for concrete and prestressed and unprestressed reinforcing steel except as noted in the design criteria.

It is the intent of the criteria to provide a structure of unquestionable integrity that will meet the postulated design conditions with a low strain elastic response. The Calvert Cliffs Containment Structure meets these criteria because: (See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.)

a. The design criteria are in general based on the proven stress and strain to meet the ACI or ASME codes. Departures from or additions to these codes have been made in the following manner:

   1. The environmental conditions of severity of load cycling, weather, corrosion conditions, maintenance, and inspection for this structure have been compared and evaluated with those for code structures to determine the appropriateness of the modifications.
   2. The consultant firm of T.Y. Lin, Kulka, Yang and Associate was retained on earlier projects to assist in the development of design criteria. In addition to assisting with the criteria submitted in the Preliminary Safety Analysis Report, they were involved in the review of design methods to assure that the criteria were implemented as intended.
   3. Dr. Alan H. Mattock of the University of Washington was retained on earlier projects to assist in developing the proper design criteria for combined shear, bending and axial load.
   4. All criteria, specifications and details relating to liner plate and penetrations and corrosion protection have been referred to Bechtel's Metallurgy and Quality Control Department. This department maintains a staff to advise the corporation on problems of welding, quality control, metallurgy and corrosion protection.
   5. The design of the Calvert Cliffs Containment Structure was continually reviewed as the criteria were revised for successive license applications.

b. The primary membrane integrity of the structure is provided by the unbonded post-tensioning tendons, each one of which is stressed to 80% of ultimate strength during installation and performs at approximately 50 to 60% during the life of the structure. Thus, the main strength elements are individually proof-tested prior to operation of the plant.

c. Eight-hundred-seventy-six such post-tensioning elements have been provided, 204 in the dome, and 204 vertical and 468 hoop tendons in the cylinder. Any three adjacent tendons in any of these groups can be lost without significantly affecting the strength of the structure due to the load redistribution capabilities of the shell structure. The bonded reinforcing steel provides for crack control assures that this redistribution capability exists.

d. The unbonded tendons are continuous from anchorage to anchorage, being deflected around penetrations and isolated from secondary strains of the shell. Thus, the membrane integrity of the shell can be assured regardless of conditions of high local strains.

e. The unbonded tendons exist in the structure at a slightly ever-decreasing stress due to relaxation of the tendon and creep of the concrete and, even during pressurization, are subject to a stress change of very small magnitude (2% to 3% of ultimate strength). Thus, the main structural system is never subjected to large changes in load, even during accident conditions.

f. The concrete portion of the structure, similar to the tendons, was subject to the highest state of stress during the initial post-tensioning. During pressurization, it is subject to a large change in load (or state of stress) but the change is, in general, a decrease in load. The large membrane compressive forces are diminished and/or replaced by relatively small radial pressures and stresses.

g. The deformations of the structure during plant operation, or due to accident conditions, are relatively minor due to the low-strain behavior of the concrete. The largest deformations occurred at the time of initial post-tensioning and shortly thereafter, prior to operation. This low strain behavior and the inherent strength of the structure permit the anchoring of all piping penetrating the structure directly to the shell. Such details (Figure 5-2) eliminate the use of expansion bellow seals and significantly reduce the likelihood of leaks developing at the penetrations. The exception to this is when the fuel transfer tube is in use, requiring use of the transfer tube bellows (Section 5.1.4.4.d).

5.1.2.2 Loads

Prior to prestressing, the structure was designed as a conventionally reinforced concrete structure. It is designed for dead load, live loads and a reduced-wind load. Allowable stresses are computed in accordance with ACI 318-63.

Loads at Transfer of Prestress

See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

The Containment Structure is checked for prestress loads and the stresses compared with those allowed by ACI 318-63 with the following exceptions: ACI 318-63, Chapter 26, allows concrete stress of 0.60 f'_{ci} at initial transfer. In order to limit creep deformations, the membrane compression stress is limited to 0.30 f'_{ci} whereas in combination with flexural compression the maximum allowable stress will be limited to 0.60 f'_{ci} per ACI 318-63.

For local stress concentrations with nonlinear stress distribution as predicted by the finite element analysis, 0.75 f'_{ci} is permitted when local bonded reinforcing is included to distribute and control the localized strains. These high local stresses are present in every structure but they are seldom identified because of simplifications made in design analysis. These high stresses are allowed because they occur in a very small percentage of the cross-section, are confined by material at lower stress and would have to be considerably greater than the values allowed before significant local plastic yielding would result. Bonded reinforcing was added to distribute and control these local strains.

Membrane tension and flexural tension are permitted provided they do not jeopardize the integrity of the liner plate. Membrane tension is permitted to occur during the post-tensioning sequence but will be limited to $\sqrt{f'_{ci}}$. When there is flexural tension but no membrane tension, the section is designed in accordance with the ACI Code, Section 2605(a). The stress in the liner plate due to combined membrane tension and flexural tension is limited to 0.5 f_y.

Shear criteria are in accordance with the ACI 318-63 Code, Chapter 26, as modified by the equations in the structural yielding subsection of this section using a load factor of 1.5 for shear loads.

Loads Under Sustained Prestress

See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

The conditions for design and the allowable stresses for this case are the same as above except that the allowable tensile stress in unprestressed reinforcing is

limited to 0.5 f_y. ACI 318-63 limits the concrete compression to 0.45 f'_{ci} for sustained prestress load. Values of 0.30 f'_{ci} and 0.60 f'_{ci} are used as described above, which bracket the ACI allowable value. However, with these same limits for concrete stress at transfer of prestress, the stresses under sustained load are reduced due to creep, shrinkage, relaxation, and possible tendon wire breakage. See Appendix 5E for a discussion on possible tendon wire breakage.

<u>At Design Loads</u>

This loading case is the basic "working stress" design. The Containment Structure is designed for the following loading cases:

a. D + F + L (Construction case)

b. D + F + L + T_o + E (Operating case)

c. D + F + L + P + T_A (Design incident case)

d. D + F + L + 1.15P (Test case)

e. D + F + L + T_s + E (Prolonged shutdown case)

D = Dead Load
L = Appropriate Live Load
F = Appropriate Prestressing Load
P = Design Pressure
T_o = Thermal Loads Due to Operating Temperature
T_A = Thermal Loads Corresponding to Pressure P
E = Operating Basis Earthquake (OBE) of 0.08g
T_s = Thermal Loads Due to Transient Wall Temperatures Over a Prolonged Shutdown
(20°F outside face, 50°F inside face)

Sufficient prestressing is provided in the cylindrical and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking is permitted but is controlled by bonded reinforcing steel.

Under the design loads, the same performance limits given for loads at transfer of prestress apply with the following exceptions:

a. If the net membrane compression is below 100 psi, it is neglected and a cracked section is assumed in the computation of flexural bonded reinforcing steel. The allowable tensile stress in bonded reinforcing is 0.5 f_y.

b. When the maximum flexural stress does not exceed $6\sqrt{f'_c}$ and the extent of the tension zone is not more than 1/3 the depth of the section, bonded reinforcing steel is provided to carry the entire tension in the tension block. Otherwise, the bonded reinforcing steel is designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel is 0.5 f_y minus the stress in reinforcing due to the thermal gradient as determined in accordance with the method of ACI-505.

c. The problem of shear and diagonal tension in a prestressed concrete structure should be considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined

membrane tension and membrane shear is considered in the next subsection.

Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). The present provisions of ACI 318-63, Chapter 26 for shear are adequate for design purposes with proper modifications as discussed in the next subsection using a load factor 1.5 for shear loads.

Crack control in the concrete is accomplished by adhering to the ACI-ASCE Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute and are as follows:

0.25 percent reinforcing shall be provided at the tension face for small members

0.20 percent for medium size members

0.15 percent for large members

A minimum of 0.20% bonded steel reinforcing is provided in two perpendicular directions on the exterior faces of the wall and dome for proper crack control.

The liner plate is attached on the inside faces of the wall and dome. Since, in general, there is no tensile stress due to temperature on the inside faces, bonded reinforcing steel is not necessary there.

Loads Necessary to Cause Structural Yielding

The structure is checked for the factored loads and load combinations that will cause structural yielding.

The load factors are the ratio by which loads will be multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, low-strain behavior. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The final design of the structure satisfies the load combinations and factors shown in Appendix 5A.

The load combinations, considering load factors referenced above, are less than the yield strength of the structure. The yield strength of the structure is defined as the upper limit of elastic behavior of the effective load carrying structural materials. For steels (both prestressed and unprestressed) this limit is taken to be the guaranteed minimum yield given in the appropriate ASTM specification. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 and the 28-day ultimate compressive strength for concrete in flexure (f_c'). The ultimate strength assumptions of the ACI Code for concrete beams in flexure are not allowed; that is, the concrete stress is not allowed to go beyond yield.

The maximum strain due to secondary moments, membrane loads and local loads exclusive of thermal loads is limited to that corresponding to the ultimate stress divided by the modulus of elasticity $(f_c'|E_c)$ and a straight-line distribution from there to the neutral axis assumed.

For the loads combined with thermal loads the peak strain is limited to 0.003 in./in. For concrete membrane compression, the yield strength is assumed to be 0.85 f_c' to allow for local irregularities, in accordance with the ACI approach. The reinforcing steel forming part of the load carrying system is allowed to go to, but not to exceed, yield as is allowed for ACI ultimate strength design.

A further definition of yielding is the deformation of the structure which causes strains in the steel liner plate to exceed 0.005 in./in. The yielding of unprestressed reinforcing steel is allowed, either in tension or compression, if the above restrictions are not violated. Yielding of the prestressed tendons is not allowed under any circumstances.

Principal concrete tension due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3\sqrt{f_c'}$. Principal concrete tension due to combined membrane tension, membrane shear, and flexural tension due to bending moments or thermal gradients is limited to $6\sqrt{f_c'}$. When the principal concrete tension exceeds the limit of $6\sqrt{f_c'}$, bonded reinforcing steel is provided in the following manner:

a. Thermal Flexural Tension - Bonded reinforcing steel is provided in accordance with the methods of ACI-505. The minimum area of steel provided is 0.20% in each direction.

b. Bending Moment Tension - Sufficient bonded reinforcing steel is provided to resist the bending moment on the basis of cracked section theory using the yield stresses stated above with the following exception: When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel is f_y minus the stress in reinforcing due to the thermal gradient, as determined in accordance with the methods of ACI-505.

Shear stress limits and shear reinforcing for radial shear are in accordance with ACI 318-63, Chapter 26 with the following exceptions:

Formula 26-12 of the Code was replaced by:

$$V_{ci} = Kb'd\sqrt{f_c'} + M_{cr}\left[\frac{V}{M^1}\right] + V_1$$

where

$$K = \left[1.75 - \frac{0.036}{np'} + 4.0np'\right]$$

but not less than 0.6 for p' ≥ 0.003.

For p' >0.003, the value of K shall be zero.

$$M_{cr} = \frac{I}{Y}\left[6\sqrt{f_c'} + f_{pe} + f_n + f_i\right]$$

f_{pe} = Compressive stress in concrete due to prestress applied normal to the cross-section after all losses (including the stress due to any secondary moment) at the extreme fiber of the section at which tension stresses are caused by live loads.

f_n = Stress due to axial applied loads (f_n shall be negative for tension stress and positive for compression stress).

f_i = Stress due to initial loads at the extreme fiber of a section at which tension stresses are caused by applied loads including the stress due to any secondary moment (f_i shall be negative for tension stress and positive for compression stress).

n = $\frac{475}{\sqrt{f_c'}}$, constant in value of K above

p' = $\frac{As'}{bd}$ ratio of compression steel area to area concrete

V = Shear at the section under consideration due to the applied loads.

M' = Moment at a distance d/2 from the section under consideration, measured in the direction of decreasing moment, due to applied loads.

V_i = Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads).

The lower limit placed by ACI 318-63 on V_{ci} of 1.7 b'd $\sqrt{f_c'}$ is not applied.

Formula 26 -13 of the Code was replaced by:

$$V_{cw} = 3.5b'd(f_c')^{1/2}\left[1 + \frac{f_{pc} + f_n}{3.5\sqrt{f_c'}}\right]^{1/2} + V_p$$

Where V_p = radial shear component of effective prestress due to tendon curvature at the section considered, and the term f_n is as defined above. All other notations are in accordance with ACI 318-63, Chapter 26. It should be noted that this formula is based on the tests and work done by Dr. A. H. Mattock of the University of Washington, and has been included in ACI 318-77, Section 11.5.2.

When the above-mentioned equations show that allowable shear in concrete is zero, radial horizontal shear ties are provided to resist all the calculated shear.

<u>Other Design Loads</u>

The Containment Structure shell is designed for the following loads:

a. Dead load
b. Prestress forces
c. Live load including allowances for piping, ductwork and cable trays
d. Wind, including tornado
e. Earthquake
f. Thermal expansion of pipes attached to the Containment Structure wall
g. Uplift due to buoyant forces

Transients resulting from the LOCA and other lesser incidents are presented in Chapter 14 and serve as the basis for the Containment Structure design pressure of 50 psig and a design concrete surface temperature of 276°F.

The external design pressure of the Containment Structure shell is 3 psig. This value is approximately 0.5 psig beyond the maximum external pressure that could be developed if the Containment Structure were sealed during a period of low barometric pressure and high temperature and, subsequently, the Containment Structure atmosphere was cooled with a concurrent rise in barometric pressure. Vacuum breakers are not provided.

5.1.2.3 Equipment Supports

a. Reactor Vessel Supports

1. Restrain the vessel to maintain the integrity of emergency core cooling systems and to prevent the rupture of additional primary pipes should LOCA occur due to single pipe rupture;
2. Permit slow radial thermal expansion of the vessel under normal operation; and
3. Restrain the vessel against seismic and LOCA jet forces.

b. Steam Generator Supports

1. Restrain the vessel to prevent simultaneous rupture of the primary coolant pipe, and the steam or the feedwater pipes;
2. Permit slow thermal growth of the loops and the vessel; and
3. Restrain all motion under seismic or LOCA loads.

Calvert Cliffs Nuclear Power Plant Units 1 and 2 are approved for leak-before-break based on References 6 and 8, and compliance with Regulatory Guide 1.45 for leak detection as documented in UFSAR Section 4.3.1. As a result of the application of leak-before-break, the mechanical/structural loads associated with the dynamic effects of a large break LOCA in the RCS hot leg or cold legs are no longer considered part of the plant design basis. Accordingly, leak-before-break is credited in the design of the steam generator sliding base supports for the replacement steam generators.

c. Pressurizer Support

1. Support normal operating loads; and
2. Restrain the vessel under seismic loads.

d. Main Coolant Pumps Supports

1. Permit slow thermal movements of the pump; and
2. Restrain the pump under seismic loads.

e. Safety Injection Tank Support

1. Support normal operating loads; and
2. Restrain the tank under seismic loads.

Materials for Equipment Supports

a. Reactor Vessel

1.	Plate material	ASTM A302, Gr B
2.	Structural shapes	ASTM A441
3.	Anchor bolts	ASTM A354, Gr BC
4.	Welding electrodes	ASTM A233, E 7018

b. Steam Generator

1.	Plate material	ASTM A302, Gr B
2.	Anchor bolts	ASTM A490
3.	Welding electrodes	ASTM A233, E 7018

c. Pressurizer

1.	Anchor bolts	ASTM A354, Gr BC

d. Main Coolant Pump

1.	Structural shapes	ASTM A36
2.	Machine bolts	ASTM A325
3.	Steel plate (S. hangers)	ASTM A515, Gr 65
4.	Pipe section (S. hangers)	ASTM A53
5.	Machined rod (S. hangers)	AISI C-1015
6.	Spring coils (S. hangers)	AISI 4161 H

e. Safety Injection Tank

1.	Anchor bolts	ASTM A354, Gr BC

Allowable Stresses and Strain for Equipment Supports Excluding Supports for Piping and Vessels:

a. Stresses under normal operating loads and the OBE are within the applicable code allowable limits.

b. Stresses under combined normal, Safe Shutdown Earthquake (SSE) and accident loads are kept below the yield strength of the material.

c. Allowable local and average strains are determined by the maximum permissible deformation for the support.

5.1.3 CONTAINMENT STRUCTURE DESIGN ANALYSIS

The analysis for the Containment Structure shell falls into two parts, axisymmetric and non-axisymmetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loading cases of dead load, live load, temperature, prestress and pressure, as described in Section 5.1.3.1. The axisymmetric finite element representation of the Containment Structure shell does not include the buttresses, penetrations, brackets, and anchors. These items of configuration, the lateral loads due to earthquakes or winds, and various concentrated loads are considered in the non-axisymmetric analysis described in Section 5.1.3.2.

This section includes only analytical techniques, references and design philosophy. The results of these analyses are shown in Section 5.1.4. The design criteria and analysis have been reviewed by Bechtel's consultants, T.Y. Lin, Kulka, Yang and Associate.

5.1.3.1 Axisymmetric Techniques

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analysis of frames and trusses can be considered an example of the finite element method. In the application of the method to an axisymmetric solid structure such as the Containment Structure shell, the continuous structure is replaced by a system of rings of quadrilateral cross-section which are interconnected along circumferential joints. Based on energy principles, sets of work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are the

unknowns. The results of the solution of this set of equations are the deformations of the structure under the given loading conditions. For the output, the stresses are computed knowing the strain and stiffness of each element.

The original finite element mesh used to describe the structure is shown in Figure 5-4 (see Sheets 1 and 2). See Appendix 5E for an evaluation that reduced the original containment minimum design prestress. The upper and lower portions of the structure were analyzed independently to permit the use of a greater number of elements for those areas of the structure of major concern, e.g., the ring girder area and the base of the cylinder. The finite element mesh of the base slab was extended down into the foundation to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab. The tendon access gallery was designed as a separate structure.

The finite element mesh for the Containment Structure does not include the interior structures. The interior structures were included in the finite element input as a lumped mass. The finite element analysis produces stresses due to axisymmetric loads. The stresses from interior structure loads and earthquake loads are superimposed on the finite element stresses. The final summation of all stresses was used to design the base slab, exterior shell and dome. The use of Bechtel's finite element computer program, CE 316-4, permitted an accurate estimate of the stress pattern at various locations of the structure. The major benefit of the program is the capability to predict shears, normal forces and moments due to internal restraint and the interaction of the foundation base slab with the subgrade. The forces and moments were applied to all directions. The following material properties were used in the program for the various loading conditions:

$E_{concrete'}$ Foundation	$3.64x10^6$ psi
$E_{concrete'}$ Shell	$4.07x10^6$ psi
$_{concrete}$ (Poisson's Ratio)	0.17
$_{concrete}$ (Coefficient of Expansion)	$0.55x10^{-5}$
E_{liner}	$29x10^6$ psi
$F_{y\ liner}$	34,000 psi
E_{soil} (Construction and Operating Case, Figure 5-4, Sheet 1)	
1st Layer	E = 6,200 psi
2nd Layer	E = 9,600 psi
3rd Layer	E = 12,000 psi
E_{soil} (Testing and Accident Case)	
1st Layer	E = 9,600 psi
2nd Layer	E = 14,400 psi
3rd Layer	E = 18,000 psi
E_{soil} (Factored Load-Yield Stress)	
1st Layer	E = 10,000 psi
2nd Layer	E = 20,000 psi
3rd Layer	E = 30,000 psi

In arriving at the above-tabulated values of E, the effect of creep is included by using the following equation for long-term loads such as thermal load, dead load and prestress:

$$E_{cs} = E_{ci} [\xi_i/(\xi_s + \xi_i)],$$

where:

E_{cs} = sustained modulus of elasticity of concrete,

E_{ci} = instantaneous modulus of elasticity of concrete,

ξ_i = instantaneous strain, in./in. per psi, and

ξ_s = creep strain, in./in. per psi

The thermal gradients used in the design are shown in Figure 5-5. The design pressure and concrete surface temperature of 50 psig and 276°F became 75 psig and 276°F at factored conditions.

The compressive stress and strain level is the highest (after the LOCA when the temperature is still relatively high, 200°F, and the pressure is dropping rapidly) at the inside face of the concrete at the edge of openings and also near the liner plate anchors. Neither concentration is a result of what may be considered a real load. In the case of an opening, the real stress is a result of prestress, reduced pressure and dead load. Applying stress concentration factors to these stresses maintains the concrete stress essentially in the elastic range. When the strain and resulting stress from the thermal gradient are also multiplied by a stress concentration factor, the total strain and resulting stress will be above the linear stress range determined by a uniaxial compression test. The relatively high stress level is not of real concern due to the following:

a. The concrete affected is completely surrounded by either other concrete or the penetration nozzle and liner reinforcing plate. This confinement puts the concrete in triaxial compression and gives it the ability to resist forces far in excess of that indicated by a uniaxial compression test.

b. The high state of stress and strain exist at a very localized area and have no effect on the overall containment integrity.

However, to be conservative, reinforcing steel was placed in these areas. The penetration nozzle will also function as compressive reinforcement.

The concrete under the liner plate anchors has some limited yielding in order to get the necessary stress distribution required to resist the liner plate self-relieving loads.

By criteria, yielding was only permitted in the design of the liner plate and of the bonded reinforcing steel for the Containment Structure. Subsequent design analyses, as tabulated in Table 5-1, indicate that the stresses in the liner plate and bonded reinforcing steel will not exceed the allowable yield stresses.

The thermal loads are a result of the temperature gradient across the structure wall. In the finite element analysis, when temperatures are given at every nodal point, stresses are obtained at the center of each element.

The liner plate was handled as an integral part of the structure and was included in the finite element mesh of the Containment Structure, but having different material properties (Figure 5-4, Sheet 1).

Under the LOCA condition or factored load condition, cracking of the concrete at the outside face would be expected. The value of the sustained modulus of elasticity of concrete, E_{cs}, was used in ACI Code 505-54 to find the stresses in concrete, reinforcing steel and liner plate from the predicted design incident thermal loads and factored incident loads.

The method of determining stresses in the concrete and reinforcement required the evaluation of the stress blocks of the cross-section being analyzed.

Stress values were taken from the computer output in the case of axisymmetric loading and from analytical solutions in case of non-axisymmetric loading. Both computations were based on homogeneous materials; therefore, some adjustment was necessary to evaluate the true stress-strain conditions when cracks develop in the tensile zone of the concrete.

An equilibrium equation was written considering the tension force in the reinforcement, the compressive force in the concrete and the axial force acting on the section. In this manner, the neutral axis was shifted from the position defined by the computer analyses to a position which is a function of the amount of reinforcement, the modulus ratio, and the acting axial forces.

The thermal stresses in the containment wall are comparable to those developed in a reinforced concrete slab which is restrained from rotation. The temperature varies linearly across the slab. The concrete will crack in tension and the neutral axis will be shifted toward the compressive extreme fiber. The cracking will reduce the compression at the extreme fiber and increase the tensile stress in reinforcing steel.

The following analysis is based on the equilibrium of normal forces; therefore, any normal force acting on the section must be added to the normal forces resulting from the stress diagram. The effects of Poisson's ratio are considered assuming the reinforcement to be identical in both directions.

Stress-strain relationship in compressed region of concrete:

$$E_c \Sigma_x = \sigma_x - V_c \sigma_y \quad (1)$$

$$E_c \Sigma_y = - V_c \sigma_x + \sigma_y \quad (2)$$

From the above equations (1) and (2):

$$\sigma_x = E_c \frac{\Sigma_x + \Sigma_y V_c}{1 - V_c^2} \quad (3)$$

$$\sigma_\gamma = E_c \frac{\Sigma_\gamma + \Sigma_x V_c}{1 - V_c^2} \quad (4)$$

Substituting,

$\sigma_x = \sigma_y = \sigma_c$ and $\Sigma_x = \Sigma_y = \Sigma_c$ into equations (3) and (4)

$$\sigma_c = E_c \Sigma_c \frac{1}{1 - v_c} = 1.205 E_c \Sigma_c \text{ (if } v_c = 0.17\text{)}$$

The reinforcement is acting in one direction, independently from the reinforcement in the perpendicular direction.

Example: If $E_c = 4.07\text{x}10^6$ and $E_s = 29\text{x}10^6$

$$n_R = \frac{29}{1.205\ x\ 4.07} = 5.9$$

The liner plate is acting in two directions, similar to the concrete except for the difference caused by the Poisson's ratios and elastic modulus:

$$\sigma_L = E_s \Sigma_s \frac{1}{1 - v_L} = 1.35 E_s \Sigma_s, v_L = 0.26$$
$$n_L = \frac{1.35\ x\ 29}{1.205\ x\ 4.07} = 7.98, v_c = 0.17$$

The concrete and reinforcement stresses, due to a moment caused by a loading other than thermal, are calculated by conventional methods. The analysis assumes homogeneous concrete sections. Those concrete and reinforcing steel stresses are then added to the thermal stresses as obtained by the method described above.

Notation:

- E_c Modulus of elasticity of concrete.
- E_s Modulus of elasticity of steel.
- n_L Modular ratio of liner plate-concrete.
- n_R Modular ratio of reinforcement-concrete.
- Σ_c Concrete strain.
- Σ_s Steel strain.
- Σ_x Concrete strain in the X direction.
- Σ_y Concrete strain in the Y direction.
- v_c Poisson's ratio of concrete.
- v_L Poisson's ratio of liner plate.
- σ_c Stress in concrete.
- σ_L Stress in liner plate.
- σ_x Stress in concrete in the X direction.
- σ_y Stress in concrete in the Y direction.

5.1.3.2 Non-axisymmetric Techniques

The non-axisymmetric aspects of configuration or loading required various methods of analysis. The descriptions of the methods used, as applied to different parts of the containment, are given below.

Buttresses

The buttresses and tendon anchorage zones are defined as Seismic Category I elements and were designed in accordance with the general design criteria for the Containment Structure and with the applicable provisions of ACI 318-63, Chapter 26.

The buttresses were analyzed for two effects, non-axisymmetric and anchorage zone stresses.

At each buttress, two out of three hoop tendons are spliced by being mutually anchored on the opposite faces of the buttresses; the third tendon is continuous through the buttress. (The anchors are located on 21-3/4" centers, and are

staggered every 7-1/4" to the opposite face of the buttress. Combined with the continuous tendon, this results in the hoop tendons being positioned at every 7-1/4" along the vertical cross-section of the wall.) Between the opposite anchorages, the compressive force exerted by the spliced tendon is twice as much as elsewhere. This value, combined with the effect of the tendon which is not spliced, will be 1.5 times the prestressing force acting outside of the buttresses. The cross-sectional area at the buttress is about 1.5 times that of the wall, thus the hoop stresses, as well as the hoop strains and radial displacements, can be considered as being nearly constant all around the structure.

The vertical stresses and strains, caused by the vertical post-tensioning, become constant at a short distance away from the anchorages because of the stiffness of the cylindrical shell. The stresses and strains remain nearly axisymmetric despite the presence of the buttresses. The effect of the buttresses on the overall analysis is negligible when the structure is under dead load or prestressing loads.

When an increasing internal pressure acts upon the structure, combined with a thermal gradient such as at the design incident condition, the resultant forces are axisymmetric. The stiffness variation caused by the buttresses will decrease as the concrete develops cracks. The structure will then tend to shape itself to follow the direction of the acting axisymmetric resultant forces even more closely. Thus, the buttress effect is more axisymmetric at yield loads (which include factored pressure) than at design loads including pressure. This fact, combined with the design provision that alternate horizontal tendons terminate in a single buttress, indicates that the buttresses will not reduce the margins of safety available in the structure.

The analysis of the anchorage zone stresses at the buttresses has been determined to be the most critical of all the various types of anchorage areas of the shell. The local stress distribution in the immediate vicinity of the bearing plates has been derived by the following two analysis procedures:

a. The Guyon equivalent prism method: This method is based on experimental photoelastic results as well as on equilibrium considerations of homogeneous and continuous media. It should be noted that the relative bearing plate dimensions are considered.

b. In order to include biaxial stress effects, use has been made of the experimental test results presented by S. J. Taylor at the March 1967 London Conference of the Institution of Civil Engineers (Group H, Paper 49). This paper compares test results with most of the currently used approaches (such as Guyon's equivalent prism method). It also investigates the effect of the rigid trumpet welded to the bearing plate.

The Guyon method yields the following results:

Maximum compressive stress under the bearing plate,

σ_c = -2400 psi

Maximum tensile stress in spalling zone,

$\sigma_{spalling}$ = +2400 psi = $-\sigma_c$

Maximum tensile stress in bursting zones,

$\sigma_{maximum\ bursting}$ = (0.04) x avg. stress = +96 psi

S. J. Taylor's experimental results indicate that the anchor plate will give rise to a similar stress distribution pattern as Guyon's method; the main difference lies in

the fact that the central bursting zone has a tensile stress peak of twice Guyon's value:

$\sigma_{\text{maximum bursting}}$ = +192 psi

A state of biaxial tension in the concrete will appear on the outside face under the loading case 1.05D + 1.5P + 1.0T_A + 1.0F. The superposition of the corresponding state of stress with the local anchor stresses reduces the load carrying capacity of the anchorage unit and causes a reduction in the maximum tensile strain to cracking.

On the other hand, the uniform compressive state of stress (vertical prestress) applied to the anchorage zone increases the load carrying capacity of the anchorage unit, with the maximum tensile strain to cracking being increased.

The design of the buttress anchor zones considered such additional vertical stresses, leading to a state of pseudo biaxial stress, the second direction being radial through the thickness.

For the above-mentioned case, i.e., 1.05D + 1.5P + 1.0T_A + 1.0F, the averaged vertical (meridional) stress component is:

$f_a \simeq$ + 400 psi

The compressive bearing plate stress at 10" depth below the bearing plate is:

fc $\simeq$ - 1500 psi

Thus, the two values introduced in the biaxial stress envelopes proposed in S. J. Taylor's article are:

f_c/f_c' = 1500/5000 = 0.3
$f_c//f_c'$ = 400/5000 = 0.08

These values show that failure could occur if vertical reinforcing was not provided. In fact, the maximum allowable vertical average tensile stress according to Taylor's interaction curve is f_a/f_c' = 0.03, therefore f_a = +150 psi.

The three-dimensional stress distribution in the anchor zones was analyzed in sufficient detail to permit the rational evaluation of stress concentrations. A conical wedge segment was used as the basic design element and the radial splitting tension was determined as a tangential distribution function. The summation of splitting stresses through the entire volume of the lead-in zone established the value of the splitting force. This force is a function of the a/b ratio and the cone angle and/or, a/b and h. Several different combinations of the values were analyzed and the most critical values selected. A system analysis for the vertical splitting force was carried out based on statics. The magnitude of vertical and spalling forces were also determined.

The most unfavorable loads and load combinations were considered in the analysis of the anchorage zone. Stresses based on transient thermal gradients were used in all cases where the use of a steady state gradient underestimated the stresses and strains and were superimposed on the bursting stresses determined from the triaxial stress calculations. The computed stresses are less than the ACI allowable values. The design of the concrete reinforcement is based on this conservative analysis to provide a margin of safety similar to the other components of the reactor building structure and to control cracking in the

anchorage zone. As a result, there is no danger of delayed rupture of the concrete under sustained load due to local overstress and microcracking.

The reinforcing details, including the method for anchoring and splicing the reinforcing, are shown on Figure 5-1.

The reinforcement required has been designed primarily to resist tensile forces, and has been located such that it will efficiently do so. The reinforcement was provided for load cases which create the maximum tensile forces and for other load cases the relevant shear forces or stresses were superimposed.

The amount of reinforcing steel was computed manually for all Seismic Category I structures, except the exterior shell and the base slab of the Containment Structure, using conventional reinforced concrete design methods for "Working Stress Design" and "Ultimate Strength Design" depending on the governing load combination.

The seismic analysis was conducted as described in the following subsection, providing values for lateral accelerations, shears, moments and displacements at specified locations. The lateral acceleration values are applied to the axisymmetric Containment Structure as non-axisymmetric static loading in Bechtel's Analysis of Axisymmetric Shell and Solid Subject to Non-Axisymmetric Static Loading, Dynamic Loading or Base Acceleration Program, CE 771. The Containment Structure is idealized as an assemblage of a series of discretized elements. Resultant shear forces, longitudinal, circular and cross moments are calculated from stresses obtained as an output from program CE 771 at 15° increments, from 0° to 180° and at 270°. The combined shear forces, normal forces, and moments are applied to the specified section and, using crack section analysis and compressive and tensile stresses in concrete, liner plate interior and exterior reinforcement are determined. Allowable working case stress and yield case stress determine the required area of reinforcement in the specified section.

The possibility of the concrete breaking along shear planes was considered at the intersection of (1) the buttress with the cylinder and (2) the cylinder with the base slab.

a. Buttress - Cylinder Intersection

   An increase in the compression force at the buttress corresponds to an increase in the concrete area of the same magnitude.

b. Cylinder - Base Slab Intersection

   An analysis for the most critical radial shear conditions was performed. The difference in shear stiffness between the shell and the buttress and the remainder of the shell was included as a shear amplification factor. The reinforcing required was less then the reinforcing provided.

The possibility of concrete breaking along a shear plane is excluded by providing ample reinforcing. In other locations, breakage along the shear plane has been excluded by the opposition of prestressing and anchor forces.

For this reason, special anchorage zone reinforcing is based on the following considerations:

a. Full-scale load tests and final designs of similar anchorages.

b. The post-tensioning supplier's recommendations of anchorage reinforcing requirements.

c. Review of the final details of the combined reinforcing on earlier projects by the consulting firm of T.Y. Lin, Kulka, Yang and Associate.

Seismic or Wind Loading

Seismic loading of the structure is higher in all cases than that of tornado or wind loading. The seismic analysis was conducted in the following manner:

The loads on the Containment Structure caused by earthquake were determined by a dynamic analysis of the structure. The dynamic analysis was made on an idealized structural model of lumped masses and weightless elastic columns acting as springs.

The analysis consists of three steps:

(1) The determination of the natural frequencies of the structure and its mode shapes, (2) the response of these modes to the earthquake by the response spectrum technique, and (3) combining modal responses to obtain structural response.

The natural frequencies and mode shapes were computed using the matrix equation of motion shown below for a lumped mass system. The form of the equation is:

$$\omega_n{}^2[m]\{\phi_n\} = [K]\{\phi_n\}, \text{ where}$$

[K] = matrix of stiffness coefficients including the combined effects of shear and flexure in the structure and the rotation, and horizontal translation of the base slab on soil.

[m] = matrix of concentrated masses.

$\{\phi_n\}$ = matrix of mode shapes

ω_n = angular frequency of vibration in the n-th mode.

The results of this computation are the several values of ω_n for n and mode shapes ϕ_n = 1, 2, 3,….p, where p is the total number of degrees of freedom (i.e., lumped masses) in the idealized model.

The response of the structure to the specified earthquake was then computed by the response spectrum technique as follows:

a. Using mode frequencies and respective damping values, a response acceleration, S_n, is read from the spectrum curves for each mode.

The modal acceleration, A_n, is given by:

$$A_n = S_n \frac{\sum_{i=1}^{P} \emptyset_{in} m_i}{\sum_{i=1}^{P} \emptyset_{in}^2 m_i}$$

The acceleration per point i and per mode n is given by:

$A_{in} = \phi_{in} A_n$

The inertial force per point per mode, $F_{in'}$ is given by:

$F_{in} = m_i A_n$

where m_i is the mass lumped at point i.

b. Using the inertial force per point per mode, F_{in}, shears and moments are computed per point per mode. The "Root Mean Square" method is used for combining modal responses (shears, moments, stresses, deflections, and/or accelerations) in the response spectrum modal analysis of Seismic Category I structures. In this method, responses are combined using the square root of the sum of the squares of responses of each mode. This method was for combining all predominant modal responses including closely spaced modal frequencies. We have examined the effect of adding closely spaced modes linearly, and find that allowable stresses have not been exceeded.

c. Seismic and wind shears are transferred across construction joints either by friction, by bond, by shear keys or by a combination of these.

<u>Large Openings</u> (Equipment Hatch and Personnel Lock Opening)

The primary loads considered in the design of the equipment hatch and personnel lock opening, as in any other part of the structure, were dead load, prestress, pressure, earthquake, and thermal loads. The secondary loads considered were the following effects caused by the above primary loads:

The deflection of tendons around the opening;

The curvature of the shell at the opening; and

The thickening around the opening.

The primary loads listed are mainly membrane loads with the exception of the thermal loads. In addition to membrane loads, incident pressure also produces punching shear around the edge of the opening. The magnitudes of these loads for design purposes were the magnitudes at the center of the opening. These are fairly simple to establish knowing the values of hoop and vertical prestressing, incident pressure, and the geometry and location of the opening.

Secondary loads were computed by the following methods:

a. The membrane stress concentration factors and effect of the deflection of the tendons around the equipment hatch were analyzed for a flat plate by the finite element method. The stresses computed by conventional stress concentration factors, compared with those values found from the above-mentioned finite element computer program, demonstrated that the deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis and does not include the effect of the curvature of the shell. However, it gives assurance of the correctness of the assumed membrane stress pattern caused by the prestressing around the opening.

b. With the help of Reference 1, stress resultants around the large opening were found for various loading cases. Comparison of the results found from this reference with the results of a flat plate of uniform thickness with a circular hole, showed the effect of the cylindrical curvature on stress concentrations around the opening.

Normal shear forces (relative to the opening) were modified to account for the effect of twisting moments (Reference 1). These modified shear forces are called Kirschoff's shear forces. Horizontal wall ties were provided to resist a portion of these shear forces.

c. The effect of the thickening on the outside face around the large opening was investigated using several methods. Reference 2 was used to evaluate the effect of thickening on the stress concentration factors for membrane stress. A separate axisymmetric finite element computer analysis, for a flat plate with thickening on the outside face, was prepared to handle both axisymmetric and non-axisymmetric loads. This program predicts the effect of the concentration of hoop tendons with respect to the Containment Structure at the top and bottom of the opening.

For the analysis of the thermal stresses around the opening, the same method was used as for the other loadings. At the edge of the opening, a uniformly distributed moment, equal but opposite to the thermal moment existing on the rest of the shell, was applied and evaluated using the methods of the preceding Reference 1. The stresses were then superimposed on the stresses calculated for the other loads.

In the case of LOCA temperature, after the incident pressure has already decreased, very little or no tension develops on the outside, so thermal strains will exist without the relieving effect of the cracks. However, the liner plate will reach a high strain level and so will the concrete at the inside corner of the penetrations, thereby relieving the very high stresses, but still carrying a high moment in the state of redistribution stresses.

In the case of 1.5P (prestress fully neutralized) + $1.0T_A$ (accident temperature), the cracked concrete, with highly strained tension reinforcement, constitutes a shell with stiffness decreased but still essentially constant in all directions. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress. In order to control the increased hoop moment around the opening, the hoop reinforcement is about twice that of the radial reinforcement (Figure 5-3).

The equipment hatch opening was thus thickened for the following reasons:

1. To reduce the predicted high membrane stresses around the opening;
2. To facilitate tendon placement;
3. To facilitate steel reinforcing placement; and
4. To compensate for the reduction in the overall shell stiffness due to the opening.

Since the resultant forces on any part of the containment exterior structure produce compressive stresses in the concrete, and sufficient unprestressed reinforcement is provided to control any local cracking, no significant cracks are expected. In the absence of any significant cracks, it is believed that the concrete will provide adequate corrosion protection for the liner plate. Cathodic protection for the Containment Structure is described in Section 5.1.7. Thus, no surveillance measures are necessary to detect the corrosion of the liner plate in the Containment Structures.

The working stress method, i.e., elastic analysis, was applied to both load combinations for design loads, as well as for yield loads, using the analytical

procedures described above. The only difference is the higher allowable stresses under yield conditions. The various factored load combinations and capacity reduction factors are specified in Appendix 5A and were used for the yield load combinations using the working stress design method. The design assumption of straight line variation of stresses was maintained under yield conditions.

The governing design condition for the edges of the equipment hatch opening at the outside face is the LOCA. Under this condition, approximately 60% of the total bonded reinforcing steel needed at the edge of the opening at the outside face, is required to resist the thermal load.

Excluding the thermal load, the remaining stress, equivalent to approximately 40% of the total stress including thermal, at the edge of the outside face, is the sum of the following stresses:

Normal stresses, resulting from membrane forces, including the effect of thickening, contribute approximately minus 45% (minus 18% of the total).

Flexural stresses, resulting from the moments caused by thickening on the outside face, contribute approximately 155% (62% of the total).

Normal and flexural stresses, resulting from membrane forces and moments caused by the effect of cylindrical curvature, contribute approximately minus 10% (minus 4% of total).

Penetrations

Analysis of the Containment Structure penetrations was divided into three parts: (1) the concrete shell, (2) the liner plate reinforcement and closure to the pipe, and (3) the thermal gradients and protection requirements at the high-temperature penetrations. The three parts will be discussed separately.

a. Concrete Shell

In general, special design consideration was given to all openings in the Containment Structure. Analysis of the various openings has indicated that the degree of attention required depends upon the penetration size. Small penetrations are considered to be those with a diameter smaller that 2-1/2 times the shell thickness, i.e., approximately 8' in diameter or less. For openings of 8' diameter or less, the curvature effect of the shell is negligible (Reference 1). In general, the typical concrete wall thickness has been found to be capable of taking the imposed stresses using bonded reinforcement, and the thickness is increased only as required to provide space requirements for radially deflected tendons. The induced stresses, due to normal thermal gradients and postulated rupture conditions, distribute rapidly and are of a minor nature compared to the numerous loading conditions for which the shell must be designed. The small penetrations are analyzed as holes in a flat plate. Applied piping restraint loads due to thermal expansion or accident forces are assumed to distribute in the cylinder as stated in Reference 3. Typical details associated with these opening are indicated in Figure 5-2.

b. Liner Plate Closure

The stress concentrations around openings in the liner plate were calculated using the theory of elasticity. The stress concentrations were then reduced by the use of a thickened plate around the opening. In the case of a penetration with no appreciable external load, shear connectors were used to maintain deformation compatibility between the liner plate

and the concrete. Inward displacement of the liner plate at the penetration was also controlled by shear connectors.

In case significant external operating loads are imposed upon the pipe penetration, the stress level from the external loads is limited to the design stress intensity values, S_m, given in the ASME, B&PV Code, Section III, Article 4. The stress level in the shear connectors from external loads is in accordance with American Institute of Steel Construction (AISC) Code for A-36 steel.

The combining of stresses from all effects was performed using the methods outlined in the ASME, B&PV Code, Section III, Article 4, Figure N-414. Figure 5-9 shows a typical penetration and the applied loads.

Stresses due to the effects of pipe loads, pressure loads, dead loads, and earthquake loads were calculated and the stress intensity was kept below S_m.

The stresses from the remaining effects were combined with the above-calculated stresses and the resultant stress intensity kept below S_a.

c. Thermal Gradient

The only high temperature pipes penetrating the Containment Structure shell are the main steam and feedwater pipes, steam generator blowdown line, and the reactor coolant letdown and sampling lines. Cooling was provided to maintain the temperature in these penetrations below 150°F.

Liner Plate

The primary purpose of the liner plate (including welds) is to provide leak tightness integrity to the post-tensioned concrete containment. Structural integrity of the structure is provided by the post-tensioned concrete and not by the liner plate.

The design, construction, inspection, and testing of the liner plate, which acts as a membrane and is not a pressure vessel, was not covered by any recognized code or specification. All components of the liner that must resist the full design pressure, such as penetrations, were selected to meet the requirements of ASME, B&PV Code, Section III, Nuclear Vessels, Paragraph N-1211. ASTM A516, Grade 60 or 70 made to ASTM A300 is a steel which meets these requirements and thus was used as a plate material for penetrations. This material has excellent weldability characteristics.

There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure, even though the liner may provide assistance in order to maintain deformation compatibility.

Loads are transmitted to the liner plate through the anchorage system and through direct contact with the concrete and vice versa. At times, loads may also be transmitted by bond and/or friction with the concrete. These loads cause, or are caused by, liner strain. The liner was designed to withstand the predicted strains.

Possible cracking of concrete has been considered and reinforcing steel is provided to control the width and spacing of the cracks. In addition, the design is such that total structural deformation remains small during the loading conditions, and that any cracking will be orders of magnitude less than that sustained in the repeated attempts to fail the prestressed concrete reactor vessel "Model 1," and

even smaller than the concrete strains of overpressure tests of "Model 2" (both at Gulf-General Atomic) (References 4 and 5).

As described, the consequences of concrete cracking on structural integrity are limited by the bonded reinforcing and unbonded tendons that have been provided. The effect of concrete cracking on the liner plate has also been considered. The anchor spacing and other features are such that the liner will sustain orders of magnitude of strain less than did the liner of Model 1 at Gulf-General Atomic (Reference 4) without tensile failure.

Liner Plate Anchors

The liner plate anchors were designed to preclude failure when subjected to the worst possible loading combinations. The anchors were also designed such that, in the event of a missing or failed anchor, the total integrity of the anchorage system would not be jeopardized.

The following loading conditions were considered in the design of the anchorage system:

- Prestress
- Internal Pressure
- Shrinkage and Creep of Concrete
- Thermal Gradients
- Dead Load
- Earthquake Loading
- Wind or Tornado Loadings
- Vacuum

The following factors were also considered in the design of the anchorage system:

- Initial inward curvature of the inner plate between anchors due to fabrication and erection inaccuracies.
- Variation of anchor spacing.
- Misalignment of liner plate seams.
- Variation of plate thickness.
- Variation of liner plate material yield strength.
- Variation of Poisson's ratio for liner plate material.
- Cracking of concrete in anchor zone.
- Variation of the anchor stiffness.

The anchorage system satisfies the following conditions:

- The system has sufficient strength and ductility, with energy absorbing capability sufficient to restrain the maximum force and displacement resulting from the condition where a panel with initial outward curvature is adjacent to a panel with initial inward curvature.
- The system has sufficient flexural strength to resist the bending moment which would result from the above condition.
- The system has sufficient strength to resist any radial pull-out forces.

Liner Supports

In designing for structural bracket loads applied parallel to the plane of the liner plate, or loads transferred through the thickness of the liner plate, the following criteria and methods have been used:

The liner plate was thickened to reduce the predicted stress level. The thickened plate, with the corresponding thicker weld attaching the bracket to the plate, will also reduce the probability of the occurrence of a leak at this location.

For tensile loads applied perpendicular to the plane of the liner plate, sufficient anchorage is provided.

The allowable stress in the perpendicular direction was calculated using the allowable predicted strain in that direction together with the predicted stresses in the plane of the liner plate.

In setting the above criteria, the reduced strength and strain capability of the material, perpendicular to the direction of rolling, was also considered. In this case, the major stress is normal to the plane of the thickened liner plate. The allowable stresses were reduced to 75% of the allowable stress calculated above.

5.1.4 IMPLEMENTATION OF CRITERIA

See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

This section documents the manner in which the design criteria were met by the designer.

Section 5.1.4.1 discusses original isostress plots and tabulations of predicted stresses for various materials. The isostress plots of the homogeneous cracked concrete structure indicate the general stress pattern for the structure as a whole, under various loading conditions. More specific documentation is made of the predicted stresses for all materials in the structure. In these tabulations, the predicted stresses are compared with the allowable to permit an easy comparison and evaluation of the adequacy of the design.

Sections 5.1.4.3 and 5.1.4.4 illustrate the actual details used in the design to implement the criteria.

5.1.4.1 Results of Analysis

See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

The isostress plots, Figures 5-6 and 5-7, show the three principal stresses and the direction of the principal stresses normal to the hoop direction. The principal stresses are the most significant information about the behavior of the structure under various conditions and were a valuable aid for the final design.

The plots were prepared by a cathode-ray tube plotter. The data for plotting were taken from the stress output of the finite element computer program of the following load cases:

D + F + L (Construction Case)

D + F + L + 1.15P (Test Case)

$D + F + L + P + T_A$ (Design Incident Case)

$\frac{1}{\phi}(1.05\,D + F + 1.5P + T_A)$ (Factored Load Case)

The above axisymmetric loading conditions have been found to be governing in the design since they result in the highest stresses at various locations in the structure.

The containment stress analysis results for structural concrete and liner plate, including shear stresses, are shown here.

5.1.4.2 Prestress Losses

See Appendix 5E for a discussion on possible prestress losses due to tendon wire breakage.

In accordance with ACI Code 318-63, the original design provided for prestress losses caused by the following effects:

a. Seating of anchorage;
b. Elastic shortening of concrete;
c. Creep of concrete;
d. Shrinkage of concrete;
e. Relaxation of prestressing steel stress; and
f. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses can be predicted with sufficient accuracy.

In this case, the environment of the prestress system and concrete is not appreciably different from that found in numerous bridge and building applications. Considerable research has been done to evaluate the above items and is available to designers in assigning the allowances. Building code authorities consider it acceptable practice to develop permanent designs based on these allowances.

The following categories and values of prestress losses have been considered in the design:

Type of Loss[a]	Assumed Value
Seating of Anchorage	None
Elastic Shortening of Concrete	2 ksi
Shrinkage and Creep of Concrete for	
Dome tendons (test data)	146.7×10^{-6} in./in.
Hoop tendons (test data)	248×10^{-6} in./in.
Vertical tendons (test data)	137.5×10^{-6} in./in.
Relaxation of Prestressing Steel(b)	9.5% of 0.70 f_s' = 15.96 ksi
Frictional Loss	K = 0.0003, μ = 0.158

(a) See Appendix 5E for a discussion on possible prestress losses due to tendon wire breakage.

(b) See Appendix 5E for a discussion on the relaxation value of the prestressing steel used for vertical tendons replaced in 2001 through 2002 on both Units.

There is no allowance for the seating of the BBRV anchor since no slippage occurs in the anchor during transfer of the tendon load into the structure. Sample lift-off readings will be taken to confirm that any seating loss is negligible.

The loss of tendon stress due to elastic shortening was based on the change in the initial tendon relative to the last tendon stressed.

The value used for shrinkage and creep loss represents only that which could occur after stressing. In general, since the concrete is well aged at the time of stress, little shrinkage and creep is left to occur and add to prestress loss.

The value of relaxation loss is based on the information furnished by the tendon system vendor, The Prescon Corporation. See Appendix 5E for a discussion on the relaxation value of the prestressing steel used for vertical tendons replaced in 2001 through 2002 on both Units.

Frictional loss parameters for unintentional curvature (K) and intentional curvature (μ) are based on full-scale friction test data. This data indicates actual values of K = 0.0003 and μ = 0.125 versus the design values of K = 0.0003 and μ = 0.158.

Assuming that the jacking stress for the tendons is 0.80 f_s' or 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at the end of 40 years for a typical dome, hoop, and vertical tendon. See Appendix 5E for a discussion on the relaxation value of the prestressing steel used for vertical tendons replaced in 2002 through 2002 on both Units, and a discussion on the final effective prestress at the end of 60 nominal years due to license extension.

	Dome[c] (ksi)	Hoop[c] (ksi)	Vertical[c] (ksi)
Jacking Stress	192	192	192
Friction Loss	20.86	29.6(a)	21.7
Seating Loss	0	0	0
	171.14	162.4	170.3
Elastic Loss	2.0	2.0	2.0
Creep and Shrinkage Loss	4.26	7.19	3.98
Relaxation Loss	15.96	15.96	15.96
Final Effective Prestress(b)	148.92	137.25	148.36

(a) Average of crossing tendons.

(b) This force does not include the effect of pressurization, which increases the prestress force.

(c) Losses shown are for original nominal 40-year license. See Appendix 5E for a discussion of losses at the end of 60 nominal years due to the license extension.

To provide assurance of achievement of the desired level of final effective prestress and that ACI 318-63 requirements were met, a written procedure was prepared for guidance of post-tensioning work. The procedure provided nominal values for end anchor forces in terms of pressure gauge readings for calibrated jack-gauge combinations. Force measurements were made at the end anchor since that is the only practical location of such measurements.

The procedure required the measured temporary jacking force for a single tendon to approach, but not exceed 850 kips (0.8 f_s'). Thus the limits set by ACI 318-63 2606(a)1, and of the prestressing system supplier, were observed. Additionally, benefits were obtained by in-place testing of the tendon to provide final assurance that the force capability exceeded that required by design. During the increase in force, measurements were required of elongation changes and force changes in order to allow documentation of compliance with ACI 318-63 2621(a). The procedure required that the prestressing steel be installed in the sheath before stressing for a sufficient time period that the temperatures of the prestressing steel and concrete reach essential equilibrium, to establish conformance with ACI 318-63 2621(e). The jacking force of 0.8 f_s' further provided for a means of equalizing the force in individual wires of a tendon to establish compliance with ACI 318-63 2621(b). The procedures required compliance with ACI 318-63 such that, if broken wires resulted from the post-tensioning sequence, compliance with Section 2621(d) was documented. Each of the above procedures contributed to assurance that the desired level of final effective prestress would be achieved.

Paragraph 2606(a)2 of ACI 318-63 refers to "tendons" rather than to an individual tendon. Further, the paragraph does not refer to the location to be considered for the determination of 0.8 f_s', for example, "temporary jacking force" referred in paragraph 2606(a)1.

Two interpretations were required; both had to consider the effect of the resultant actions on both the prestressing system and structure.

The first interpretation was that the location for measurement of the seating force, used in calculating the percentage of f_s', was at the end anchor and just subsequent to the measurement of the "temporary jacking force" referred in ACI 2606(a)1. The advantages of this location are several. One is that it is a practical one and thus the possibility for achieving valid measurements is greater. The second is that it is the same location used for measuring the "temporary jacking force" and measurements could be made without the added complexity of additional measuring devices. The third advantage is that measurements at this location provide assurance that the calculated percentage of f_s' at seating does not anywhere exceed the maximum percentage of f_s' to which that tendon has been subjected.

Several possible cases were considered for the second interpretation so as to allow anchoring of an individual tendon without exceeding the requirement stated for "tendons" collectively in ACI 318-63 2606(a)2. One such case assumed that the anchoring force for the typical tendon was that for a tendon anchored midway through the prestressing sequence. It further assumed that the losses to be assumed were one-half of the sum of elastic losses, and of the creep, shrinkage, and relaxation predicted to occur during the entire prestressing sequence. This interpretation, however, was considered to be neither practical nor enforceable, since it resulted in changing the seating forces as the actual (as compared to the schedule) time length of the prestressing period was dictated by weather and manpower availability.

In another case, the stressing is done by jacking each tendon to the required force of 850 kips (0.8 f_s'), and placing shims of predetermined thickness, corresponding to the calculated elongation, between the bearing plate assembly and the washer. Proper tendon stress is assured by comparing the jack pressure and tendon elongation with previously calculated values.

The case adopted was to seat each tendon with a measured "pressure" reading for the jack, at "lift-off" of the end anchor, of 775 kips (between 0.72 and 0.73 f_s',). This procedure had several advantages.

One advantage was that the force on the containment and the tendon was within the bounds of those for which it had been tested and resulted in no known detrimental effects. The second advantage was that the stressing procedure was simplified, since the stressing crews did not have to accommodate a large number of different anchoring force requirements. The third advantage was that, at the completion of stressing the last tendon, the expected losses were such that the average percentage of f_s' at the end anchors of the tendons would be less than 0.7 f_s' thus establishing compliance with ACI 318-63 2606(a)1 and 2. The fourth advantage was that the percentage loss of prestressing force was less than would be the case if the tendons were anchored in such a manner that the calculated percentage of f_s' nowhere exceeded 0.7 f_s'.

The latter advantage deserves special mention since it plays a strong role in assuring that the final effective prestress equaled or exceeded the desired value. For example, if the percentage of f_s' at the anchorage of the tendons were 0.1 f_s', creep and shrinkage of concrete could result in the loss of almost all of the prestressing force. Assuming that the total losses due to creep, shrinkage and elastic shortening equal 0.1 f_s', then the final effective prestress would be 20% of an initial prestress equivalent to 0.5 f_s'. If the initial prestress was equivalent to 0.7 f_s', the final effective prestress, neglecting relaxation for the moment, would be about 86% of the initial prestress. Clearly, the assurance (that the concrete creep and shrinkage losses have been properly accounted for) increases as the percentage of f_s' for the anchored tendon(s) increases. However, this design was committed to meeting the ACI 318-63 requirement and the anchorage force for the tendons was kept at or below 0.7 f_s' in accordance with the interpretation described.

5.1.4.3 Liner Plate

The following design bases were applied to the containment liner so that it meets the specified leak-rate under LOCA conditions:

a. The liner is protected against damage by missiles. (Section 5.1.5.3)
b. The liner plate strains are limited to allowable values that have been shown to result in leak-tight vessels or pressure piping.
c. The liner plate is prevented from development of significant distortion.
d. All discontinuities and openings are properly anchored to accommodate the forces exerted by the restrained liner plate, and careful attention is paid to details of corners and connections to minimize the effects of discontinuities.

Pressure vessels, pressure piping, high pressure hydraulic tubing, and similar containers are made by cold forming, drawing, and dishing operations, where strains may approach the elongation capacity of the material. (For mild steel at failure, this elongation varies from 15 to 30%.) These forming operations result in high strains both in tension and compression. Vessels and piping components manufactured by these methods have a history of high leak-tight integrity, proving that subjecting the steel material to high strain does not affect its leak-tight integrity.

The best basis for establishing allowable liner plate strains is considered to be that portion of the ASME, B&PV Code, Section III, Nuclear Vessels, Article 4.

Specifically, the following sections have been adopted as guides in establishing allowable strain limits:

Paragraph N-412(m)	Thermal Stress
Paragraph N-414.5	Peak Stress Intensity
Table N-413	Classification of Stresses for Some Typical Cases
Figure N-414	Stress Categories and Limits of Stress Intensity
Figure N-415(A)	Design Fatigue Curves
Paragraph N-412(n)	Operational Cycle
Paragraph N-415.1	Vessels Not Requiring Analysis for Cyclic Operation

American Society of Mechanical Engineers design codes require that the liner material be prevented from experiencing significant distortion due to the thermal load and that the stresses be considered from a fatigue standpoint [Paragraph N-412(m)(2)]. The following fatigue loads were considered in the design of the liner plate:

a. Thermal cycling due to annual outdoor temperature variations. Daily temperature variations do not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner plate. The number of cycles for this loading is 40 cycles for the plant life of 40 years.

b. Thermal cycling due to interior temperature variations during the startup and shutdown of the reactor system. The number of cycles for this loading was assumed to be 500.

c. Thermal cycling due to the LOCA was taken very conservatively to occur only once during plant life. Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the concrete. The attachment sleeve was designed in accordance with ASME, B&PV Code, Section III fatigue considerations. All penetrations were reviewed for a conservative number of cycles to be expected during plant life.

Thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME, B&PV Code. The allowable stress in Figure N-415(A) is for alternating stress intensity for carbon steels and temperatures not exceeding 700°F. In addition, the ASME code further requires that significant distortion of the material be prevented.

In accordance with ASME, B&PV Code, Paragraph 412(m)(2), the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F. It also satisfies the requirements for limiting strains on the basis of fatigue consideration. A typical section showing the anchors is included in Figure 5-1.

American Society of Mechanical Engineers, B&PV Code, Paragraph 412(n), Figure N-415(A), has been developed as a result of research, industry experience, and the proven performance of code vessels. Because of the conservative factors it contains on both stress intensity and stress cycles, and its being a part of a recognized design code, Figure N-415(A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415(A) does not extend below ten cycles, ten cycles was used for the LOCA instead of one cycle mentioned above.

Establishing an allowable strain based on ten significant thermal cycles of the LOCA condition would permit an allowable strain [from Figure N-415(A)] of approximately 2%. Maximum allowable tensile or compressive strain has been conservatively set at 0.5% (compared to 2% shown above). The maximum predicted strain in the liner plate during LOCA conditions has been found to be 0.25% compression.

At the design LOCA condition, there will be no tensile stress anywhere in the liner plate membrane. This is true both at the time of initial pressure release and under any later pressure and temperature condition. The purpose of specifying a non-destructive examination temperature requirement is to provide protection against a brittle fracture or cleavage mode of failure. However, this type of failure is precluded by the absence of tensile stresses.

No allowable compressive strain value has been set for the test condition because the value will be less than that experienced under the LOCA condition. The maximum allowable tensile strain will be 0.2% under test conditions; the predicted value is much smaller.

The maximum compressive strains are caused by LOCA pressure, thermal loading prestress, shrinkage, and creep. The maximum calculated strains do not exceed 0.0025 in./in. and the liner plate will always remain in a stable condition.

The stability of the liner plate has been studied for the loading cases and deformations to which it may reasonably be subjected. The critical loading cases that were considered included the LOCA condition and the operating condition during the winter.

Two separate solutions to the plate stability problem were made:

a. As a compressed panel under biaxial compression, assuming that the channel and angle stiffeners are rigid in their attachment to the prestressed concrete Containment Structure and the liner.

b. As a compressed panel under biaxial compression, assuming the panel to be a portion of a large cylinder with a flexible stiffener system.

Figure 5-1 illustrates the actual physical configuration of the stiffening system for the liner plate. The channels function as horizontal stiffeners and the angles as vertical stiffeners.

For the solution, an initial deflected form for the liner plate is expressed in terms of a Fourier Series of the form:

$$\Delta = \sum_{n=0}^{\infty} \sum_{m=i}^{\infty} \Delta_{mn} \bullet \cos m\phi \bullet sin \frac{n\pi x}{\ell}$$

where: ϕ defines the central angle in a plan view of the cylinder from a point on the circumference where there is zero radial deflection to the point on the circumference where there is maximum radial deflection; x defines the radial deflection and ℓ defines the unsupported length in the vertical direction.

The stability analysis based on the above assumptions indicates that the critical buckling stress of the liner will be approximately 29,000 psi.

Under normal operating conditions, the maximum compressive stress in the liner will be approximately 23,000 psi. This reflects a buckling margin of 20%. Under LOCA conditions the expected stress in the liner plate will be 35,000 psi. Under this stress the overall structural stability of the liner plate can be maintained.

Also of concern is the nature of the state of stress and its behavior at the point of attachment between the stiffeners and the liner plate. Special tests have been conducted on simulated models of the liner plate and vertical stiffener assembly to determine the shear capacity of the angle and vertical stiffener assembly in order to determine the shear capacity of the angle anchorage. In the tests, two different configurations of support were used for the simulated continuous anchor. One case simulates the expected condition that will exist in the Containment Structure. A second case simulates the condition that might exist at an isolated location if the concrete were not in continuous contact with the anchor. Being guided in the proportioning of the liner plate stiffeners by the more critical values of shear transfer for the case of non-continuous contact will, in general, result in a margin of safety for progressive failure of anchors of approximately 2.7. The weld configuration shown in Figure 5-1 is felt to be adequate to transfer all loads that are considered in the design of the structure between the liner plate and the stiffener-anchors.

The conservative design approach of the stiffening system used in the liner plate to prevent significant distortions at accident conditions, and the stringent welding and weld inspection requirements ensure that the leak tightness of the liner plate at the LOCA condition will not change from that at the test condition.

In isolated areas the liner plate may have initial inward curvature due to construction. The anchors are designed to resist the forces and moments induced when a section of the liner plate between anchors has initial inward curvature.

The liner plate is anchored at all discontinuities to eliminate excessive strains. The forces in the liner plate at the discontinuities were evaluated and the anchors designed to resist these forces.

The liner adjacent to the penetrations is backed up by concrete. Refer to Figure 5-2 for a description of the anchoring arrangement at discontinuities and typical details.

The containment penetrations behave, partly or fully depending on their size, as elements of a pressure vessel. The sizes range from the small closure pieces for small pipe penetrations to larger components such as air locks and the equipment hatch. Those portions that are not backed up by concrete (the penetrations themselves) are treated as a pressure vessel and comply with ASME, B&PV Code, Section III, Nuclear Vessels, Subsection B, and are designed in accordance with the requirements of Section VIII, Paragraphs UG-27 through UG-33 for the service conditions outlined in Section 5.1.

The pressure part to thickened liner plate transition areas are designed and reinforced in accordance with the requirements of Paragraphs UG-36 through UG-41 as a welded assembly conforming to ASME, B&PV Code, Section VIII, Paragraph UW-13.

The air locks to thickened liner attachment rings were analyzed by finite-element methods; the membrane meridional, membrane hoop and the average radial stresses were investigated and the adequacy of design was verified.

The liner plate is considered as a composite part of the containment shell and as such, it is investigated by utilizing the strain and deformation data obtained by the finite-element analysis of the containment shell as tabulated in Table 5-1. See Appendix 5E for later tables associated with an evaluation to reduce the original amount of required containment prestress. The ASME B&PV Code is used only as a guideline to establish allowable strain limits, and not to verify the structural and leaktight adequacy of the liner.

In general, where the liner is firmly anchored and bonded to the concrete and maintains a strain compatibility with the concrete shell, the thickening of the liner will neither reduce nor increase the stresses. However at the openings, and particularly in the transition boundaries where the pressure elements of the penetrations are welded to the liner, the increase in the thickness of the steel plate will:

a. reduce the possible local stress concentrations, both in the steel plate and the adjacent concrete;

b. increase the stiffness of the plate and hence its buckling limit; and

c. increase the leaktight integrity of the transition welds.

At all penetrations the liner plate is thickened to reduce stress concentrations in accordance with the ASME, B&PV Code 1968, Section III, Nuclear Vessels. The thickened portion of the liner plate is anchored to the concrete by use of anchor studs all around the penetrations. For details of the penetrations see Figure 5-9. The sleeves, pipe cap and all welds associated with the penetrations are designed to resist all loads previously mentioned and also the prestress forces and internal design pressure.

At each location where a load is to be delivered to the walls, slab, or dome of the Containment Structure, an insert plate is provided to transmit the load through the liner. The insert plate is thickened and stiffened as required to deliver the load and to reduce stress concentrations in the liner. The insert plate is anchored to the concrete by appropriate anchors and shear connections. Typical examples of such insert plates are the polar crane brackets, and the floor beam brackets at the operating deck; typical details are shown in Figure 5-1.

5.1.4.4 Penetrations

All penetrations are pressure resistant, leak-tight, welded assemblies designed, fabricated, and tested in accordance with the ASME, B&PV Code, Section III, Nuclear Vessel Code.

Types of Penetrations

a. Electrical Penetrations

Two types of electrical penetration assemblies are used - canister and unitized header. All electrical penetration assemblies are fabricated and tested in accordance with the ASME, B&PV Code, Section III, Nuclear Vessel Code. The canister type is inserted in a nozzle of suitable diameter integral with the Containment Structure and field welded on the inside end. The unitized header type is welded to the nozzle on the outside end. All penetration assemblies are provided with a means to pressurize for monitoring of leakage. Any abnormal depressurization of an assembly is annunciated locally and in the Control Room.

There are three different types of electrical penetrations, used as follows:

1. Type 1 - 15 kV medium voltage power penetration canisters. These are 30" diameter cylinders constructed from steel plate (SA516, GR 70, 0.652" thick). Stainless steel header plates are then welded into each end containing epoxy bushings welded into the header plates.
2. Type 2 - Low voltage power and control penetration:
   a. Canister design: These are constructed of 10" diameter schedule 40 seamless steel pipe (SA 106B). Conductors are sealed with glass hermetics either direct fired into header plates or as assemblies welded into header plates.
   b. Unitized Header Design: Header is constructed of stainless steel, Type 304 (SA 240). Feedthroughs consist of conductors with resilient seal, are mounted and sealed to a header.
3. Type 3 - Instrumentation, thermocouple and coaxial penetrations:
   a. Canister design: See type 2.a above.
   b. Unitized Header design: See type 2.b above.

Figure 5-2 shows the different types of electrical penetration assemblies.

b. Piping Penetrations

Single barrier piping penetrations are provided for all piping passing through the containment walls. The closure of the pipe to the liner plate is accomplished with a pipe cap welded to the pipe and to the liner plate reinforcement. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to reduce the concrete temperature to 150°F. Figure 5-2 shows typical hot and cold pipe penetrations. The modes of containment isolation are covered in Section 5.2.

The anchorage of penetration closure connecting pipes to the containment wall were designed as Seismic Category I structures to resist all forces and moments caused by a postulated pipe rupture. The design conditions include the maximum pipe reactions and pipe rupture forces.

The following design basis for typical piping penetrations was used to ensure the integrity of the liner penetration junction at the piping.

1. The penetration assembly, consisting of pipe cap and the assembly welds and welds to the liner plate, utilizes full penetration welds. The assembly is anchored into the wall concrete and designed to accommodate all forces and moments due to pipe rupture and thermal expansion.
2. The design basis is that the pipe penetration is the strongest point in the system when a pipe break is postulated. Pipe stops, increased pipe thickness or other means are used to attain this. Part of this basis also is that the operation of closure valves will not be impaired by any postulated pipe break.

c. Large Penetrations (Equipment and Personnel Access Hatches)

An equipment hatch opening, 19' in diameter, is provided as shown on Figure 5-3. The equipment hatch opening is covered on the inside of the Containment by a dished hatch, fabricated from welded steel, furnished with a double gasketed flange and bolted in place. The equipment hatch opening also has an outage use door at the exterior opening. The containment outage door is comprised of three assemblies: (1) a transition

ring assembly welded directly to the equipment hatch nozzle steel liner plate; (2) a door frame assembly attached to the transition ring by means of bolts and locating pins that support all dead weight and seismic loads; and (3) a hinged door assembly. The frame of the containment outage door is equipped with fittings or penetrations that will allow electricity, compressed air, water, etc., to be run through the equipment hatch opening without going through the door. Flanges, cable penetrations, and isolation valves at these penetrations will be designed to withstand containment pressure resulting from postulated boiling in the RCS, as might occur during a loss of shutdown cooling. The design load to pass through the containment outage door is a reactor coolant pump motor on its transport trailer. For larger equipment, like replacement steam generator components, the bolted-on door frame assembly and door assembly may be removed from the transition ring assembly to restore the equipment hatch opening to its full diameter. Internal lugs welded to the nozzle liner plate and designed to carry the reactions due to the axial pressure loads and seismic loads to the nozzle liner plate may be removed and replaced when the door is restored.

Two personnel locks are provided. One of these is for emergency access only. Each personnel lock is a double door, welded-steel assembly. A quick-acting equalizing valve connects the personnel lock with the interior and exterior of the Containment Structure for the purposes of equalizing pressure in the two systems when entering or leaving. Typical details of the personnel access lock are shown in Figure 5-3.

The two doors in each personnel lock are interlocked to prevent both from being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. Remote indicating lights and annunciators situated in the Control Room indicate the operational status of the door. Provision is made to permit bypassing the door interlocking system to allow doors to be left open during plant cold shutdown. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior.

d. Special Penetrations

1. Refueling Tube Penetration

A refueling tube penetration is provided for fuel movement between the refueling pool in the Containment Structure and the spent fuel pool in the Auxiliary Building. The penetration consists of a 36" stainless steel pipe installed inside a 42" pipe sleeve. The inner pipe acts as the refueling tube and is fitted with a gate valve in the spent fuel pool and an encapsulating pipe sleeve which is welded to the refueling pool liner and sealed off from the containment with a testable double O-ring blind flange in the refueling pool. This arrangement prevents leakage through the refueling tube in the event of a LOCA. The 42" pipe sleeve is welded to the containment liner.

Bellows expansion joints are provided on the transfer tube to compensate for any differential movement between the tube and the building structures. Figure 9-14 is a drawing of the refueling tube installation. The design basis for each expansion joint is listed below:

AUX. BUILDING FUEL POOL EXPANSION JOINT	CONTAINMENT REFUELING POOL EXPANSION JOINT
External design pressure 28 psig	External design pressure 29 psig
Internal design pressure 28 psig	Internal design pressure 29 psig
Design temperature 150°F	Design temperature 273°F
Lateral movement 1-1/2"	Lateral movement 1/2"
Axial movement 1-1/2" (expansion or contraction)	

Displacements were selected to accommodate the maximum differential building settlements. The expansion joint in the refueling pool may be visually inspected from the outside with a periscope when the refueling pool is empty. Repair would probably involve removing the 54" OD pipe that forms the containment boundary. The outside of the expansion joint in the spent fuel pool may be visually inspected by draining the pool or by remote means. A test connection on the bellows provides a means of testing bellows integrity. Repair would require draining the spent fuel pool.

2. Containment Supply and Exhaust Purge Penetrations

   The ventilation system purge penetrations are equipped with a testable double O-ring blind flange in the penetration room and a tight-seating butterfly valve inside containment used for isolation purposes. The blind flange is used to provide containment integrity during Modes 1-4. The valve is manually opened for containment purging in Modes 5 and 6 as described in Section 9.8.2.2.

3. Containment Vent Penetration

   This system is equipped with two valves to be used for isolation purposes. These valves are opened by a handswitch in the Control Room to vent containment pressure during power operation. Although control of hydrogen in Containment following an accident is not required, this penetration may be used as a hydrogen purge.

Design of Penetrations

a. Design Basis

   Penetrations conform to the applicable sections of American Standards Association (ASA) N6.2-1965, "Safety Standard for the Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors" which has since been withdrawn. All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME, B&PV Code, Section III, Code for Nuclear Vessels. All future penetrations will conform to ASME, B&PV Code, Section III, Division 1 for fluted head analysis, and Division 2 for those integrated with steel in concrete openings.

Each line which penetrates the containment and contains high-pressure or high-temperature fluids (steam, feedwater, and steam generator blowdown) passes through a structural steel sleeve mounted on the containment wall. This sleeve acts as a positive stop to prevent whipping associated with fracture of a line containing high internal energy and thereby prevents damage to the penetration and breaching of the containment.

Further protection of each line, necessary to preclude pipe rupture between the penetration and the first valve, is accomplished by shortening the exposed length of pipe and installing the first valve as close as possible to the internal or external wall of the structure, dependent upon valve operating and maintenance clearances. Design bases which apply to the provision of automatic and manual isolation valves in the penetration lines are contained in Section 5.2.

All penetrations, except the equipment hatch and emergency personnel lock, are inside the Auxiliary Building; therefore, the temperature of Auxiliary Building penetration material will not fall below 30°F. Using the assumption of +60°F temperature inside the containment and outside air temperature of +20°F, the temperature of the equipment hatch and emergency personnel door will not be below +30°F, during startup, operation, or cooldown of the reactor. An investigation was made to determine necessity and/or feasibility of protecting the exterior surface of these penetrations. None was required. Two temperature indicators are provided to monitor the temperature inside the containment. A technical specification is not considered to be necessary for the above reasons.

b. High-Temperature Penetrations

The main high-temperature piping consists of two penetrations for feedwater, two penetrations for main steam, two penetrations for steam generator blowdown, one for the reactor coolant letdown line, and one for the reactor coolant sampling line. These have a maximum operating temperature range between 435°F and 653°F. Thermal insulation is provided on the outside diameter of each line and separate coolant circulation, with instrumentation suitable for flow monitoring, is provided in the air gap between the insulation and the penetration liner sleeve. The combination of insulation and coolant circulation is designed to restrict the maximum temperature in the concrete to 150°F.

For the condition of loss of penetration coolant circulation, the maximum steady state temperature in the concrete will be 300°F at the penetration surface and decreases to 120°F at a maximum radial depth of 48" in the containment wall. Actual peak temperatures in the penetrations resulting from LOCA conditions are expected to subside within six hours. A maximum temperature of 390°F may be tolerated for 120 days (Reference 5) without appreciable loss of concrete strength.

The basis for limiting strains in the penetration steel is the ASME, B&PV Code, Section III, Nuclear Vessels, Article 4, 1965, and therefore, the penetration structural and leak-tightness integrity will be maintained. Local heating of the concrete immediately around the penetration will develop compressive stress in the concrete adjacent to the penetration and a negligible amount of tensile stress over a large area. The mild steel

reinforcing added around penetrations will distribute local compressive stresses for overall structural integrity.

c. Penetration Materials

The material for the penetrations, including the personnel and equipment access hatches together with the mechanical and electrical penetrations, is carbon steel and conforms with the requirements of the ASME, B&PV Code, Nuclear Vessel Code. As required by the Nuclear Vessel Code, the penetration materials which form the pressure boundary meet the necessary Charpy V-notch impact values at a temperature 30°F below the lowest service temperature.

1. Piping Penetration Materials

Materials specifications are listed below:

Piping Penetration Material	Specification
Penetration Sleeve	ASTM A155
Penetration Reinforcing Rings	ASTM A516
Penetration Sleeve Reinforcing	ASTM A516
Bar Anchoring Rings and Plates	ASTM A516
Rolled Shapes (nonpressure boundaries)	ASTM A36

2. Electrical Penetration Materials

The penetration sleeves to accommodate the electrical penetration assemblies are Schedule 40 carbon steel pipe, except where otherwise noted.

3. Access Penetration Materials

The equipment hatch and personnel access locks materials are all ASTM A516 made to ASTM A300.

Installation of Penetrations

The qualification of welding procedures and welders is in accordance with Section IX, "Welding Qualifications" of the ASME, B&PV Code. The repair of defective welds is in accordance with paragraph N-528 of Section III, "Nuclear Vessels" and Section VIII of the Code.

Testability of Penetrations

Only the following penetrations are ASME, B&PV Code, Section III, Class B penetrations as classified in the Atomic Energy Commission (AEC) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," December 15, 1966. As Class B penetrations, they are subject to individual periodic leak-rate tests separate from the integrated Class A tests.

a. Equipment Hatch
b. Personnel Lock
c. Emergency Personnel Escape Lock
d. Refueling Tube
e. Purge Line Inlet and Outlet

5.1.5 INTERIOR STRUCTURE

For the original compartment designs, the occurrence of a LOCA was postulated to result from the rupture of the RCS piping, including the main loop piping, either within the reactor cavity or the SG compartments of the containment. For the current compartment designs, breaks are not postulated to occur in the main loop piping based on the Leak-Before-Break Evaluation (References 6 and 7). The design of the Babcock & Wilcox, Canada replacement steam generators also incorporates the application of leak-before-break, which included dynamic analysis of steam generator internals (U-tubes, divider plate, lattice grids, and shroud) for the effects of the most limiting postulated break (12" pressurizer surge line break). American Society of Mechanical Engineers Code-reconciled design and methods were used to install the replacement steam generators to ensure the equivalency of the replacement steam generators-to-RCS piping connections to the original steam generator–to-RCS piping connections, and thus the continued validity of the assumptions made in References 6 and 7. Since only main loop piping is present in the reactor cavity, no LOCA is currently postulated to occur in this compartment. However, smaller bore piping connected to the RCS is present in both steam generator compartments and the pressurizer compartment. The following discussion regarding a LOCA in the reactor compartment is provided as historical design information only.

5.1.5.1 Design Basis

Design of the containment interior structures evolves around four basic systems: The primary coolant system, the turbine steam system, the engineered safety system, and the fuel handling system.

The structures which house or support the basic systems are designed to sustain the factored loads of Appendix 5A.

The design bases applied are:

a. The structures will sustain all operating dead and live loads, thermal loads, and design seismic loads, without exceeding code allowable stresses.

b. Loads and deformations resulting from a LOCA failure and its associated effects in any one of the basic systems will be sustained and restricted such that propagation of the failure to any other system is prohibited. In addition, a failure in one loop of the Nuclear Steam Supply System will be restricted such that propagation of the failure to the other loop is prohibited.

Loss-of-coolant accident loads and associated effects include:

a. Thrust loads resulting from rapid mass release from a pipe break in any system;

b. Pressure buildup in locally confined areas such as the reactor cavity or the secondary shielding compartments; and

c. Jet forces resulting from the impingement of the escaping mass upon adjacent structures. The following method was employed to compute the jet forces:

    1. Rupture may occur anywhere on portions of the main coolant pipes that are near to the structure or component under investigation. The jet force due to a double-ended break acts along the pipe axis, which may change its direction. The jet force due to a slot break may act in any radial direction normal to the pipe axis.

2. Both the double-ended and the slot break expose an area equal to the inside cross-sectional area of the pipe ruptured. The length of a slot break is equal to twice the diameter of the ruptured pipe.

3. The magnitude of the jet force is computed by the following equation, and does not change with distance to target:

   F = 2.0 x Operating Pressure x Inside Cross-Sectional Area of Pipe

4. The area of the jet plume diverges at a half angle of 10° from the pipe opening. The jet force is evenly distributed to the area of plume at any distance to obtain the impinging pressure on the structure.

5. A spectrum of all possible breaks are considered for a particular structure, and the ones that produce maximum stresses in the structure are used for design.

d. Erosion effects of jet spray.

e. Pipe whipping following a break in the pipe.

f. Rapid rise in ambient temperature to 276°F and accompanying rise in ambient pressure to 50 psig. All Containment Structures have been evaluated for the revised maximum vapor temperature referenced in Section 14.20.

g. Missiles as described in Section 5.1.5.3.

The containment interior structures were analyzed by utilizing a lumped parameter model which treated the exterior shell, base slab, foundation media and the interior structures as a coupled system, and included details of the interior structures. The ground response spectra, as shown in Section 2.6.3, were used to determine the shears, moments, forces and displacements in the various structural, components, expected to be induced by the SSE and OBEs. The method employed is basically the analysis through the response of the normal modes to spectral accelerations, and is outlined in Section 5.1.3.2 as part of the "Containment Structure Design Analysis." Seismic induced shears, moments and forces are then multiplied with the load factors for Seismic Category I structures as indicated in Section 5A.3.1. When checking each component, particular attention has been paid to tall and slender, or tall and cantilevered portions for possible further amplification. Such components were investigated by utilizing the floor response spectra.

Seismic analysis for the interior structures was done using procedures outlined in Section 5.1.3.2.

Where concrete structures such as the primary shield wall are subjected to sustained internal heat buildup, mechanical cooling devices are included to keep the internal temperatures below 150°F. Localized concrete temperatures up to 200°F are acceptable.

The total heat generation rate in the primary shield includes contributions from the following radiation sources: fission neutrons and gammas; core secondary gammas; secondary gammas from the core shroud, core support barrel, the reactor vessel, and the intermediate water regions; and secondary gammas in the concrete primary shield. Secondary gamma sources due to neutron scattering and capture were based on neutron distributions calculated using methods described in Chapter 3. The computer program QAD-P5 incorporates a point-kernel numerical integration method for the gamma radiation and a point-kernel moments method for each of the source contributions listed above and the heating rates due to the several contributions summed. The temperature gradient of the primary shield wall resulting from radiation-generated heat is combined with the

temperature gradient resulting from convective heat from the reactor to form a combined temperature gradient in the primary shield wall. The combined gradient was used in the thermal stress analysis and the resulting stresses in the wall were determined by using the design methods established by ACI 505-54. These stresses were combined with the stresses resulting from the analysis of loading combinations as described in Section 5.1.5.1. The critical stresses were then used to evaluate required reinforcing in both directions. All stresses are within the allowable as established Appendix 5A.

5.1.5.2 Design Loads and Materials

a. Design Loads:

The reactor cavity wall, the steam generator compartment walls, and containment interior structures are all Seismic Category I structures, the final design of which satisfied the most severe of the load combination equations of Appendix 5A.

b. Materials:

The following materials have been used in the construction of the Containment Structure interior:

Concrete	f_c'= 5000 psi @ 28 days
Rebar	A615, Gr 60
Plate Steel	A441 & A533
Structural Steel	A36 & A441
Anchor Bolts	A325, A354 & A490

5.1.5.3 Missile Protection Inside Containment

High pressure RCS components which could be a source of missiles are suitably screened, either by the concrete shield wall enclosing the reactor coolant loops, by the concrete operating floor or by special missile shields, to block any passage of missiles to the containment walls. Potential missile sources are oriented so that the missile will be intercepted by the shields and structures provided. A shield is provided over the control rod drive mechanism to block any missiles generated from postulated fracture of the nozzles.

All internally-generated missiles inside the Containment Structure are listed in Table 5-2. For all other internally-generated missiles outside the Containment Structure, see Section 5.3.1.1. All Seismic Category I structures or parts of structures have been checked for impact of missiles. The modified Petry formula was used for checking the effects of missiles on the structure at the point of impact. Even under extreme assumptions, missiles do not penetrate the concrete barriers completely.

Missile protection inside the Containment Structure is provided to comply with the following criteria:

a. The Containment Structure and liner are protected from loss-of-function due to damage by such missiles as might be generated in a LOCA for break sizes up to and including the double-ended severance of a reactor coolant pipe.
b. The engineered safety features and components required to maintain containment integrity are protected against loss-of-function due to damage by such missiles.

Missile protection necessary to meet the above requirements was implemented using the following methods:

a. Components of the RCS were examined to identify and to classify missiles according to size, shape, and kinetic energy for purpose of analyzing their effects.

b. Missile velocities were calculated considering both fluid and mechanical driving forces which can act during missile generation.

c. The RCS is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with the double-ended rupture of a reactor coolant pipe and designed to stop missiles.

d. The structural design of the missile shielding takes into account impact loads and is based upon the state of the art of missile penetration calculational techniques.

The types of missiles for which protection is provided are:

a. Valve stems;

b. Valve bonnets;

c. Instrument thimbles; and

d. Various types and sizes of nuts and bolts.

Protection is not provided for certain types of missiles for which postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, and conservative design as applied to the particular component. Included in this category are missiles caused by massive, rapid failure of the reactor vessel, steam generator, pressurizer, and main coolant pump flywheels and casings.

It is not expected that the polar crane will become or generate an internal missile. The crane is anchored with seismic lugs and the crane and its components are designed for the OBE and SSE. Additionally, design features and administrative controls are in place to minimize the probability of a load drop from the polar crane. A description of our means of controlling heavy loads is presented in Section 5.7.

5.1.5.4 <u>Thermal Gradients</u>

Ventilation and cooling systems maintain thermal gradients at a level low enough to have very small structural effects on the concrete walls. These effects are, nevertheless, considered in the design.

<u>Reactor Cavity Wall</u>

Energy is deposited in the concrete of the reactor cavity wall by nuclear radiation emanating from the surface of the reactor vessel. In the equilibrium condition, assuming no axial leakage, all the deposited energy is removed either by the air-cooling system at the inner surface of the cavity wall or by convection from the outer surface. In order to find the temperature distribution within the concrete, an analytical expression for the energy deposition distribution is first derived. This expression is substituted in the heat diffusion equation which is solved for temperature using the temperature boundary conditions at the inner and outer face of the concrete.

Steam Generator Compartments

The ventilation system provided eliminates temperature gradients across the secondary shield walls, across the refueling pool walls and across the operating floor.

5.1.5.5 Differential Pressures

Generally, the occurrence of a LOCA is postulated to result from a rupture of the primary system piping either within the reactor cavity or the steam generator compartments of the containment. A pipe rupture of this type, within a compartment, results in the expulsion of high enthalpy water that flows out of the ruptured pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-air-water mixture flows through openings into the main containment. The maximum pressure differential will depend on the number and shape of the openings leading between the compartments, the volume of each compartment, and the blowdown rate from the broken pipe.

Differential pressure analyses are made with the Bechtel computer program COPRA, which calculates the transient pressure responses of two containment compartments during a LOCA. COPRA calculates a mass and energy balance of the two-phase, two-component steam-water-air mixture as primary coolant enters the compartment during the LOCA and exits through vents and openings into the main containment building. There is no provision in the code for heat transfer to structures or for engineered safety features. These options generally have a negligible effect on compartment pressures for the short time following the rupture within which peak differential pressures occur.

Because of this short time interval between the LOCA and the peak compartment pressure differential, each case analyzed is divided into very small time intervals, generally equal to or less than 0.1 msec. The thermodynamic equations are solved for each time advancement.

The steam, water and air throughout each compartment are in thermal equilibrium at all times. Water, steam, and air entering a compartment are mixed homogeneously and instantaneously; no accumulation of water occurs on the walls or in the sump.

A COPRA Summary Document which lists and discusses the program assumptions, is contained in Reference 9. This document is on file with the Nuclear Regulatory Commission (NRC).

The discussion of the results of the analyses performed for each interior containment compartment is contained in FSAR [Final Safety Analysis Report] Section 14.16.5, as revised by FSAR Amendment No. 29, Revision 18 (currently Updated Final Safety Analysis Report Section 14.20).

The compartment walls have been designed for jet impingement forces and differential pressures, both assumed to occur simultaneously with the SSE.

This design has the following conservative considerations:

a. A dynamic load factor of two is applied to the maximum jet load. (The jet load is based on sudden severance of the primary loop piping nearest to the wall.)
b. The differential pressure and jet impingements are considered to be in phase.

As dictated by design, any transient response of the containment compartment walls is expected to contain the most severe accident postulated.

Reactor Cavity

There are two types of openings for flow out of the reactor cavity into the main containment volume:

a. Openings around the main coolant pipes - 40.0 ft^2
b. Openings around the reactor vessel seal ledge - 173 ft^2

The current plant design includes a permanent refueling pool seal that further restricts the opening around the reactor vessel. This analysis is provided as historical design information only (Reference 7).

The areas around the main coolant pipes extend along the annular space between the pipes and the walls of the pipe tunnels. A calculated nozzle flow coefficient of 0.61, based on the geometry of these flow spaces, is utilized. The opening around the refueling seal ledge is treated as an orifice. Flow coefficients for orifice geometries are supplied automatically by the COPRA computer code.

Immediately after a pipe rupture, the volume available to the expanding steam within the reactor cavity is the free volume below the seal ledge.

Both guillotine and slot primary pipe failures have been considered:

a. For the guillotine break, the coolant pipes are partially restrained by the tunnel walls. Restraining the primary pipes in this manner gives a flow area of 0.61 ft^2 for failure of the 42" line, which results in a cavity peak differential pressure of 23 psi.
b. The slot longitudinal break has a length of twice the inside pipe diameter and an area equal to the pipe cross-sectional area. The flow from the part of the break within the pipe tunnel is divided between the reactor cavity and the steam generator compartment, while the flow from the part of the break that is between the primary shield wall and the nozzle of the vessel enters the reactor cavity only. The total flow of mass and energy into the reactor cavity is approximately 27% of the flow from a 42" single-ended rupture. As the reactor cavity pressure builds up, the flow leaving the pipe within the pipe tunnel will enter the steam generator compartment; however, credit for this effect is conservatively ignored to simplify the problem. The peak differential pressure resulting from the slot failure is 31 psi.

The following table summarizes the input parameters used for the computer analysis of reactor cavity pressurization:

Main containment free volume	2.0x106 ft^3
Reactor cavity free volume	5.135×10^3 ft^3
Nozzle-type relief area (around piping)	40.0 ft^2
Nozzle coefficient	0.61
Orifice-type relief area (around seal ledge)	173.0 ft^2
Initial containment temperature	110°F
Initial containment pressure	14.7 psia
Initial containment humidity	50%

Note that this analysis was redone for the neutron shield with no significant differences.

Steam Generator Compartments

For the steam generator compartments, only the double-ended guillotine break in the hot leg is considered since it provides the largest rate of energy and mass release. The relief areas are divided into three classes: long, sharp-edged nozzles, sharp-edged orifices and well-rounded orifices. An orifice coefficient of 0.61 is applied to the long nozzles and a coefficient of 0.97 is applied to the well-rounded orifices; coefficients for sharp-edged orifices are supplied by the computer. The compartment volumes and total flow areas are listed below:

Compartment	Volume	Long Sharp -edged Nozzles	Sharp -edged Openings	Well Rounded Openings	P
East	[(a)] 53,980 ft^3	356 ft^2	770 ft^2	1,072 ft^2	16.4 psi
West	51,500 ft^3	182 ft^2	414 ft^2	1,072 ft^2	18.4 psi

(a) East volume is greater than west because east includes pressurizer as well as one steam generator and two reactor coolant pumps.

The initial conditions and the main compartment volume are the same as those used in the reactor cavity.

5.1.5.6 Design Bases Temperature and Pressure Differentials

Safety factors were not determined or established for pressure and temperature acting alone. The design and analysis methods as applied to the interior structures, accounted for and investigated the effects of temperature and pressure in conjunction with the other loads occurring in those conditions as stated in Appendix 5A. The associated individual safety margins can best be examined by: discussing the contribution of these loads to the total stress intensities, or to the total ultimate load carrying capacity of the structure or component under the two main categories of design loading conditions.

The two main design loadings conditions and the serviceability required of the interior structures in each case are that:

a. They provide support during plant operation and prevent the occurrence of a LOCA from:

   1. Normal operating loads (including dead loads)
   2. Design live loads
   3. Hydrostatic loads
   4. Thermal loads
   5. OBE seismic loads

b. They mitigate its consequences, should a LOCA occur, by protecting the containment and all engineered safety features systems from:

   1. Blowdown forces (including jet thrust and impingement loads)
   2. Whipping pipes
   3. Missiles
   4. Differential pressures

5. SSE seismic loads
6. Loads a.1, and others, if applicable

The safety factor of the structure is the ratio of allowable stresses to the computed stresses in case (a) above, and the ratio of the ultimate load resisting capacity of the structure to the total applied loads in case (b).

In general, this ratio is not the same for all components of the interior structure and varies with both the modes of stress or loading and the hazard associated with the structure or component thereof (i.e., some of the components that were governed by radiological shielding requirements have capacities much higher than the design loads; uniaxial, flexural normal, shearing torsional, tensile and compressive stresses were treated differently). Further, the loads above were combined per Appendix 5A load combination equations which have an inherent factor of safety defined by their load coefficients. No credit is taken for this safety factor.

For the operating loads, the minimum safety margin is as determined from ACI 318-63 code for reinforced concrete, and AISC Specification, 1963, for structural steel. The contribution of the operating temperatures is negligible in all components except the reactor cavity wall, for which additional reinforcing steel was furnished for the thermal stresses in accordance with the requirements of ACI 505-59. The accident condition controlled the design of most of the interior structures, including the reactor cavity wall, the secondary shield walls which house the RCS, and all major vessel support structures. Where stresses due to blowdown forces were in the same direction and thus combined with the stresses due to other loads including temperature and pressure, the share of the former ranged from 60 to 80%, thus resulting in an increase of 2.5 to 5 times the inherent safety factor for the latter loads acting alone. Where temperature and pressure stresses acted alone, the safety factors inherent in the load combination equations were increased by 3.0 and 1.5 in the operating and accident cases respectively, for both reinforcing and structural steel. In all cases, the safety margin for concrete stresses were kept higher than steel in order to preclude brittle modes of failure.

The calculated differential pressures and temperature gradients, and those that have been used as the design bases of the major interior structures are listed below:

	ΔP (Design)	ΔP (Calculated)	ΔT (Design)	ΔT (Calculated)
East Steam Generator Compartment Walls	28 psi	16.4 psi	None	None
West Steam Generator Compartment Walls	28 psi	18.4 psi	None	None
Reactor Cavity Wall	96 psi	31.0 psi[a]	30°F/7'	6.7°F/7'

[a] The current plant design includes a permanent refueling pool seal which further restricts the opening around the reactor vessel. This analysis result is provided as historical design information only (Reference 7).

5.1.6 LIGHTNING PROTECTION OF CONTAINMENT

Lightning protection is provided over the dome of the containment vessel and consists of the following:

One 1/2"x24" copper air terminal is placed in the center of the dome. Eight equally spaced 1/2"x25" air terminals are installed on a 20'-diameter circumference. These are connected to closed loop of 2/0 AWG cable. Four cables of like size connect the loop with the center air terminal. Another 16 equally spaced 1/2"x24" air terminals are installed on 40' circumference. These are connected to a closed loop of 2/0 AWG cable. Another 20 equally spaced 1/2"x24" air terminals are installed on a 69'3" radius circumference.

These are connected to a closed loop of 2/0 AWG cable. Four equally spaced 1/2"x24" air terminals connect the three loops. Four 2/0 AWG down conductors connect the outer loop to the ground grid. Ground guards are provided to protect down conductors to an approximate height of 8' above grade.

5.1.7 CATHODIC PROTECTION

The containment cathodic protection system and related environmental or galvanic corrosion protection schemes are explained in the following discussion.

Heavy waterproofing (40 mils thickness) membrane is used underneath the containment base slab, within the 4"-thick mud mat and around the cylindrical shell up to finish grade. This membrane is protected to full height by 1/2" (minimum) thick asphalt-impregnated fiber board prior to backfilling. Remotely-located shallow anodes are used in cathodic protection system. Anodes placed approximately 100' from the membrane could protect the structure regardless of the total cross-section of apertures and would provide protection as required for pipes and tank bottoms. Zinc and/or copper sulphate permanent reference electrodes are located in the soil to enable measurement of protective gradients around the foundation. This scheme is based on the concept that the containment building foundation steel is formed with an inhibitive concrete cover, a heavy waterproofing membrane and a relatively low chloride/oxygen environment. Steel embedded in or covered by concrete in the containment sub-structure will be protected by the waterproof membrane. The system is conservatively designed for a 40-year life, derating manufacturer's recommendations for anodes by approximately 50%.

The surface of the exposed liner plate was protected by an initial cleaning by wire brushing or sandblasting, and then coated with Amercoat zinc paint or its equivalent at the fabrication plant. After erection, all exposed areas adjacent to field welds were cleaned or reprimed, and the entire exposed face was painted with a finish coat. The outside of the liner is protected by a minimum of 3-1/4' of prestressed concrete, which is cast against the liner offering a very high degree of corrosion protection.

The ACI Building Code Requirements for Reinforced Concrete (ACI 318-63), Section 808, Concrete Protection For Reinforcing, have proved to be effective for preventing chemical corrosion of concrete reinforcement. The reinforcing concrete protection exceeds Section 808 requirements by 50% or more. Tendon sheathing has a minimum of 11" of concrete cover at any location in the Containment Structure.

5.1.8 LEAKAGE MONITORING SYSTEM

No continuous leakage monitoring system is provided.

The barrier to leakage in the Containment Structure is the 1/4" steel liner plate. All penetrations were continuously welded to the liner plate before the concrete in which they are embedded was placed. These penetrations, shown on Figures 5-2

and 5-3, became an integral part of the liner and were so designed, installed, and tested.

The steel liner plate is securely attached to and is an integral part of, the prestressed concrete Containment Structure which is conservatively designed and rigorously analyzed for the extreme loading conditions of the postulated LOCA, as well as for all other types of anticipated loading conditions. Thorough controls were maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetrations, and during construction of the entire Containment Structure.

The comprehensive program for preoperational testing, inspection, and post-operational surveillance is described in detail in Section 5.5 and is summarized in the following paragraphs.

During construction, the entire length of every seam weld in the liner plate was leak tested. Individual penetration assemblies were shop tested. Welded connections between penetration assemblies and the liner plate were individually leak tested after installation. Following completion of construction, the entire Containment Structure, the liner and all its penetrations were tested at 115% of the design pressure to establish structural integrity. The initial leak rate test of the entire structure was conducted at 50 and 100% of the calculated peak pressure to demonstrate vapor tightness, and to establish a reference for periodic leak testing for the life of the plant.

Penetrations such as the permanent equipment hatch opening and personnel access locks cannot be opened except by deliberate action. The personnel access locks are interlocked and provided with alarm devices so that the Containment Structure cannot be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate welds is not possible, means are provided so that they can be tested for leakage. The liner plate is protected against corrosion by suitable coatings and by cathodic protection. Walls and floors for biological and missile shielding also provide protection for the liner plate.

Once the adequacy of the liner plate was established initially, there is no reason during the life of the plant to anticipate progressive deterioration which would reduce the effectiveness of the liner as a vapor barrier. Inside the Containment Structure, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by prestressed concrete which is resistant to all weather conditions.

Inspection on a periodic basis, as necessary, may be conducted in all accessible areas during full power operation. Biological shielding is provided to reduce direct radiation from the RCS to acceptable limits. A visual inspection of the Containment Structure, from the inside and outside, will be conducted during regularly scheduled unit shutdowns.

All penetrations, except the following, are located in groups, and a penetration room is located at each group.

a. Equipment Hatch Opening
b. Personnel Access Lock
c. Emergency Personnel Access Lock
d. Refueling Tube

e. Purge Line Inlet and Outlet

f. Containment Emergency Sump

Any leakage that might occur from these penetrations will be collected and exhausted through vents or drains. In this manner, leakage which might occur from these groups of penetrations will be isolated from leakage which might occur through the Containment Structure, itself.

Provisions are made such that these penetrations may be pressure tested for leakage during normal operation.

The containment normal sump penetration is another exception that is not located in a penetration room. However, there are no provisions made to collect and exhaust leakage from this penetration or to enable pressure testing during normal operation. This penetration is further described in Section 5.2.2.

Within the penetration rooms, provisions are made such that individual penetration assemblies with resilient seals may be pressure tested for leakage during normal operation. This degree of control over leakage through penetrations greatly reduces the probability of undetected leakage for the Containment Structure as a whole.

A considerable background of operating experience is being accumulated on containment and penetrations. Full advantage of this knowledge is taken in all phases of design, fabrication, installation, inspection, testing, and operation.

For the foregoing reasons, it has been concluded that a continuous leakage monitoring system is unnecessary. Since there is no such system provided, there can be no misoperation or malfunction, which in itself might constitute a hazard. The steel-lined Containment Structure is self-sufficient, and other than valves and hatch doors, there are no operating parts. The containment boundary is extended only by specific penetrations which are further described and tabulated in Section 5.2, "Isolation System."

5.1.9 REFERENCES

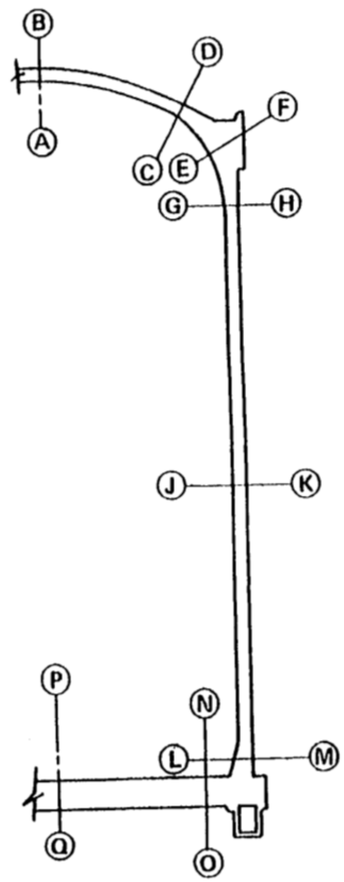
1. A.C. Eringen and A.K. Naghdi, "State of Stress in a Circular Cylindrical Shell with a Circular Hole"
2. Levy, Samuel, A.E. McPherson, and F.C. Smith, "Reinforcement of a Small Circular Hole in a Plane Sheet Under Tension," Journal of Applied Mechanics, June 1948
3. K.R. Wickman, A.G. Hopper, and J.L. Mershon, "Local Stress in Spherical and Cylindrical Shells due to External Loadings," Welding Research Council Bulletin No. 107, August 1965
4. HTGR and Laboratory Staff, Prestressed Concrete Reactor Vessel, Model 1, GA7097
5. Advanced HTGR Staff, Prestressed Concrete Reactor Vessel, Model 2, (GA7150)
6. CEN-367-A, “Leak Before Break Evaluation of Primary Coolant Piping in Combustion Engineering Designed NSSS,” February 1991
7. Letter from Mr. D. G. McDonald (NRC) to Mr. R. E. Denton (BGE), dated February 3, 1994, Installation of a Neutron Shield/Pool Seal at the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M87176 and M87177)

8. Letter from NRC to Mr. E. C. Sterling, III, Chairman CEOG, dated October 30, 1990, Acceptance for Referencing Topical Report CEN-367, “Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems”

9. Containment Pressure Analysis, NS731 TN, December 1968

TABLE 5-1

STRESS ANALYSIS RESULTS[8]



KEY ELEVATION
(Showing location of reference sections)

LOCATION	STRUCTURAL DATA: CONCRETE		STRUCTURAL DATA: REINFORCING STEEL	
	f_c' - psi	t - in	P_m - %	P_h - %
A	5000	39	.260	.094
B	5000	39	.619	.214
C	5000	55.6	.159	.150
D	5000	55.6	.253	.232
E	5000	138.12	.060	.060
F	5000	138.12	.286	.112
G	5000	45	.316	.185
H	5000	45	.741	.289
J	5000	45	.235	.235
K	5000	45	.235	.235
L	5000	70.2	.315	.247
M	5000	70.2	.570	.247
N	4000	120	.223	.185
O	4000	120	.447	.479
P	4000	120	.286	.222
Q	4000	120	1.144	.514

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

ALLOWABLE STRESSES

WORKING STRESS DESIGN	YIELD STRESS DESIGN
SHELL CONCRETE:	
f_a = 1500 psi	$f_a = \phi_a (f_c)$ = (0.85) (5000) = 4,250 psi
f_{ce} = 3000 psi	$f_{ce} = \phi_{ce} (f_c)$ = (0.90) (5000) = 4,500 psi
BASE CONCRETE:	
f_{ce} = 1800 psi	$f_a = \phi_a (f_c)$ = (0.85) (4000) = 3400 psi
	$f_{ce} = \phi_{ce} (f_c)$= (0.90) (4000) = 3600 psi
STEEL: A615, GR 60	
f_s = 30,000 psi	$f_s = \phi (f_y)$ = (0.90) (60,000) = 54,000 psi

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

D + F + L (Stresses in psi) Construction Case I

	SECTION	MERIDIONAL σ_e OUTSIDE	MERIDIONAL σ_e INSIDE	MERIDIONAL σ_a AXIAL	HOOP σ_e OUTSIDE	HOOP σ_e INSIDE	HOOP σ_a AXIAL	SHEAR τ	SHEAR v_{ci}	SHEAR v_{cw}
Shell										
	A - B	-827	-1630	-1256	-814	-1609	-1207	-23	105	687
	C - D	+243	-1630	-609	+77	-734	-250	+32	45	452
	E - F	-248	-784	-464	+136	-863	-330	+15	43	360
	G - H	-1	-26	-618	-55	-1154	-613	-40	104	513
	J - K	-1	-8	-664	-575	-1645	-1120	-8	69	427
	L - M	2	-27	-499	-481	-645	-583	6	98	359
Base										
	N - O	113	-251	-41	565	-426	-29	32	138	157
	P - Q	560	-971	-69	565	-470	-12	44	135	123

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

			CONCRETE							
			COMPUTED					COMPUTED VS. ALLOWABLE		
SECTION		**LOAD CASE**	σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_a}$	$\frac{\tau}{v}$
	II	D+F+L+1.15P	-997	-935	-591	-542	-6	0.332	0.394	0.017
	III	D+F+L+T_O+E	-2141	-2363	-1307	-1324	20	0.788	0.883	0.056
A-B	IV	D+F+L+T_A+P	-2990	-2985	-545	-510	7	0.993	0.363	0.020
	V	1.05D+F+1.5P+ T_A	-2610	-2213	-215	-61	2	0.580	0.051	0.003
	VI	1.05D+F+1.25P+ T_A+1.25E	-2048	-2205	-652	-519	24	0.490	0.153	0.040
	VII	D+F+P+ T_A+E'	-2247	-2392	-804	-732	17	0.532	0.189	0.030
	II	D+F+L+1.15P	-814	-573	-240	50	68	0.271	0.160	0.200
	III	D+F+L+T_O+E	-2210	-1367	-645	-293	65	0.737	0.430	0.180
C-D	IV	D+F+L+T_A+P	-2820	-2539	-224	-35	83	0.940	0.149	0.230
	V	1.05D+F+1.5P+ T_A	-1970	-2058	-94	28	128	0.457	0.022	0.220
	VI	1.05D+F+1.25P+ T_A+1.25E	-2444	-2183	-145	25	171	0.543	0.034	0.280
	VII	D+F+P+ T_A+E'	-2848	-2564	-226	-39	125	0.633	0.053	0.210
	II	D+F+L+1.15P	-357	-709	-293	-235	15	0.236	0.195	0.040
	III	D+F+L+T_O+E	-232	-1409	-471	-293	15	0.469	0.314	0.040
E-F	IV	D+F+L+ T_A+P	-346	-2984	-269	-133	7	0.995	0.179	0.020
	V	1.05D+F+1.5P+ T_A	-451	-2626	-197	42	23	0.584	0.046	0.040
	VI	1.05D+F+1.25P+ T_A+1.25E	-374	-2843	-231	-25	20	0.632	0.054	0.033
	VII	D+F+P+ T_A+E'	-349	-3014	-272	-135	11	0.670	0.064	0.018

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

			REINFORCING STEEL			
			COMPUTED		COMPUTED VS. ALLOWABLE	
SECTION		**LOAD CASE**	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
	II	D+F+L+1.15P	-269	-16	0.009	0.001
	III	D+F+L+T_O+E	2098	694	0.070	0.023
A-B	IV	D+F+L+T_A+P	+3590	4253	0.120	0.142
	V	1.05D+F+1.5P+T_A	4760	4512	0.088	0.084
	VI	1.05D+F+1.25P+T_A+1.25E	14440	28664	0.267	0.531
	VII	D+F+P+T_A+E'	9888	16483	0.183	0.305
	II	D+F+L+1.15P	1750	4786	0.058	0.160
	III	D+F+L+T_O+E	6282	10485	0.209	0.349
C-D	IV	D+F+L+T_A+P	4040	15967	0.134	0.532
	V	1.05D+F+1.5P+T_A	147	19229	0.003	0.356
	VI	1.05D+F+1.25P+T_A+1.25E	1475	18794	0.027	0.348
	VII	D+F+P+T_A+E'	4084	16127	0.076	0.298
	II	D+F+L+1.15P	124	2148	0.004	0.072
	III	D+F+L+T_O+E	-276	5701	0.01	0.190
E-F	IV	D+F+L+T_A+P	1110	8667	0.037	0.289
	V	1.05D+F+1.5P+T_A	4080	10259	0.076	0.190
	VI	1.05D+F+1.25P+T_A+1.25E	2343	10036	0.0434	0.186
	VII	D+F+P+T_A+E'	1121	8754	0.021	0.162

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

			CONCRETE							
			COMPUTED					COMPUTED VS. ALLOWABLE		
SECTION		**LOAD CASE**	σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_a}$	$\frac{\tau}{v}$
	II	D+F+L+1.15P	-69	-746	-154	-224	-13	0.249	0.149	0.037
	III	D+F+L+T_O+E	-1415	-1770	-683	-793	61	0.590	0.529	0.170
G-H	IV	D+F+L+T_A+P	-101	-2827	-68	-195	-11	0.942	0.130	0.310
	V	1.05D+F+1.5P+T_A	-96	-1766	114	23	-66	0.392	0.027	0.110
	VI	1.05D+F+1.25P+T_A+1.25E	-1453	-1496	-162	-221	-20	0.332	0.052	0.033
	VII	D+F+P+T_A+E'	-1529	-1683	-194	-364	1	0.374	0.086	0.002
	II	D+F+L+1.15P	-60	-653	-201	-133	1	0.218	0.134	0.003
	III	D+F+L+T_O+E	-1821	-2244	-732	-1209	-9	0.747	0.806	0.250
J-K	IV	D+F+L+T_A+P	-67	-2629	-111	-99	-2	0.876	0.074	0.006
	V	1.05D+F+1.5P+T_A	-96	-2828	169	420	2	0.628	0.099	0.003
	VI	1.05D+F+1.25P+T_A+1.25E	-1488	-957	-209	-75	5	0.331	0.049	0.008
	VII	D+F+P+T_A+E'	-1697	-1669	-311	-447	-3	0.377	0.105	0.005
	II	D+F+L+1.15P	+10	-848	-277	-718	1	0.283	0.479	0.003
	III	D+F+L+T_O+E	-621	-2012	-494	-635	-2	0.671	0.423	0.006
L-M	IV	D+F+L+T_A+P	-29	-2914	-254	-731	147	0.971	0.487	0.415
	V	1.05D+F+1.5P+T_A	-84	-2973	-263	-465	163	0.661	0.109	0.271
	VI	1.05D+F+1.25P+T_A+1.25E	-1633	-3510	-197	-914	14	0.780	0.215	0.023
	VII	D+F+P+T_A+E'	-1427	-3360	-296	-870	-6	0.745	0.205	0.010

TABLE 5-1

STRESS ANALYSIS RESULTS(8)

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

			REINFORCING STEEL			
			COMPUTED		COMPUTED VS. ALLOWABLE	
SECTION		**LOAD CASE**	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
	II	D+F+L+1.15P	2884	2284	0.096	0.076
	III	D+F+L+T_O+E	2616	3332	0.087	0.111
G-H	IV	D+F+L+T_A+P	16649	10293	0.555	0.343
	V	1.05D+F+1.5P+T_A	7381	15437	0.136	0.285
	VI	1.05D+F+1.25P+T_A+1.25E	23724	28024	0.440	0.520
	VII	D+F+P+T_A+E'	23595	20489	0.437	0.379
	II	D+F+L+1.15P	2368	2882	0.080	0.096
	III	D+F+L+T_O+E	5283	351	0.176	0.012
J-K	IV	D+F+L+T_A+P	11580	11410	0.386	0.380
	V	1.05D+F+1.5P+T_A	8720	10523	0.161	0.195
	VI	1.05D+F+1.25P+T_A+1.25E	29043	26964	0.537	0.499
	VII	D+F+P+T_A+E'	25474	14396	0.471	0.266
	II	D+F+L+1.15P	22007	-5482	0.734	0.183
	III	D+F+L+T_O+E	3366	1812	0.112	0.0604
L-M	IV	D+F+L+T_A+P	11216	3168	0.374	0.105
	V	1.05D+F+1.5P+T_A	13369	3063	0.247	0.057
	VI	1.05D+F+1.25P+T_A+1.25E	19995	9888	0.370	0.183
	VII	D+F+P+T_A+E'	10770	9651	0.199	0.178

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

SECTION		LOAD CASE	CONCRETE COMPUTED σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	COMPUTED VS. ALLOWABLE $\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_a}$	$\frac{\tau}{v}$
	II	D+F+L+1.15P	69	-362	68	76	127	0.201		0.600
	III	D+F+L+T_O+E	-911	-1204	-64	-17	48	0.669		0.220
N-O	IV	D+F+L+T_A+P	-207	-594	26	58	114	0.330	Note (a)	0.540
	V	1.05D+F+1.5P+T_A	7	-704	20	86	149	0.196		0.413
	VI	1.05D+F+1.25P+T_A+1.25E	-861	-1203	54	52	164	0.334		0.454
	VII	D+F+P+T_A+E'	-1098	-1198	25	52	154	0.333		0.427
	II	D+F+L+1.15P	-1338	-473	56	113	74	0.743		0.350
	III	D+F+L+T_O+E	-1562	-1540	-43	-63	43	0.868		0.202
P-Q	IV	D+F+L+T_A+P	-1697	-599	-45	121	65	0.943	Note (a)	0.306
	V	1.05D+F+1.5P+T_A	-2253	-829	-31	68	135	0.626		0.374
	VI	1.05D+F+1.25P+T_A+1.25E	-2115	-1086	-15	75	74	0.588		0.205
	VII	D+F+P+T_A+E'	-1621	-1251	-42	46	79	0.450		0.220

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

SECTION		LOAD CASE	REINFORCING STEEL COMPUTED σ_m	REINFORCING STEEL COMPUTED σ_h	COMPUTED VS. ALLOWABLE $\frac{\sigma_m}{f_s}$	COMPUTED VS. ALLOWABLE $\frac{\sigma_h}{f_s}$
	II	D+F+L+1.15P	256	20957	0.009	0.700
	III	$D+F+L+T_O+E$	19168	26123	0.640	0.871
N-O	IV	$D+F+L+T_A+P$	13172	26362	0.440	0.878
	V	$1.05D+F+1.5P+T_A$	16596	35106	0.307	0.650
	VI	$1.05D+F+1.25P+T_A+1.25E$	30315	41218	0.561	0.763
	VII	$D+F+P+T_A+E'$	29196	40450	0.542	0.749
	II	D+F+L+1.15P	26933	24818	0.897	0.827
	III	$D+F+L+T_O+E$	24970	30500	0.832	1.000
P-Q	IV	$D+F+L+T_A+P$	26824	30012	0.894	1.000
	V	$1.05D+F+1.5P+T_A$	36983	42126	0.684	0.780
	VI	$1.05D+F+1.25P+T_A+1.25E$	35862	40468	0.664	0.749
	VII	$D+F+P+T_A+E'$	29984	41033	0.555	0.759

NOTE [a]: $f_a = 0.3\, f_c'$ not applicable to base slab.

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

		CONCRETE			REINFORCING STEEL	
SECTION	**LOAD CASE**	σ_{em}	σ_{eh}	τ	σ_m	σ_h
A-B	$D + F + L + T_o$	-1860	-1845	-18	-190	111
	$D + F + L + T_o + E$	-2141	-2363	20	2098	694
	$D + F + L + T_s$	-1400	-1374	-23	-4470	-4419
	$D + F + L + T_s + E$	-1466	-1526	-10	-10516	-10246
	$1.05D + F + L + 1.25P + T_A$	-653	-465	23	3700	4198
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-2048	-2205	24	11731	25506
	$D + F + L + P + T_A$	-897	-794	7	3590	4253
	$D + F + L + P + T_A + E'$	-2247	-2393	17	9889	16483
C-D	$D + F + L + T_o$	7	1	43	-12,200	-6074
	$D + E + L + T_o + E$	-2281	-1393	65	8139	7912
	$D + F + L + T_s$	137	220	32	-7880	-1838
	$D + F + L + T_s + E$	-2283	-1392	-4	8101	7903
	$1.05D + F + L + 1.25P + T_A$	7	1	114	-4040	7353
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-2006	-946	171	14042	10321
	$D + F + L + P + T_A$	8	1	83	-4340	-1033
	$D + F + L + P + T_A + E'$	-1545	-1209	125	9390	13818
E-F	$D + L + F + T_o$	-242	-1266	10	-1396	5701
	$D + L + F + T_o + E$	-836	-1542	15	-728	10234
	$D + F + L + T_s$	-225	-681	15	-993	1337
	$D + F + L + T_s + E$	-732	-723	30	-1135	901
	$1.05D + F + L + 1.25P + T_A$	321	-653	13	-1052	7427
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-193	-2132	20	-750	52514
	$D + F + L + P + T_A$	262	-711	7	-809	9653
	$D + F + L + P + T_A + E'$	-336	-2012	11	-1691	49431

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

SECTION	LOAD CASE	CONCRETE			REINFORCING STEEL	
		σ_{em}	σ_{eh}	τ	σ_m	σ_h
G-H	$D + F + L + T_o$	-1321	-1522	41	2622	3475
	$D + F + L + T_o + E$	-1415	-1770	61	2616	3332
	$D + F + L + T_s$	-816	-1022	-40	-1187	-1466
	$D + F + L + T_s + E$	-783	-1095	18	4114	4984
	$1.05D + F + L + 1.25P + T_A$	-2584	-2440	-32	14927	12730
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-1452	-1495	-20	23724	28024
	$D + F + L + P + T_A$	-2660	-2712	-11	14972	-10960
	$D + F + L + P + T_A + E'$	-1529	-1682	1	23595	20489
J-K	$D + F + L + T_o$	-1595	-2115	-6	4228	476
	$D + E + L + T_o + E$	-1821	-2244	-9	5283	351
	$D + F + L + T_s$	-927	-1407	-8	-511	-3931
	$D + F + L + T_s + E$	-990	-1453	29	3960	-8747
	$1.05D + F + L + 1.25P + T_A$	-1029	-1648	57	-511	17727
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-989	-957	5	3960	26965
	$D + L + F + T_A$	-2655	-2628	-2	11580	11410
	$D + L + F + T_A + E'$	-1696	-1669	-3	25473	14396
L-M	$D + F + L + T_o$	-304	392	-15	-2422	3821
	$D + F + L + T_o + E$	-621	-2012	-2	3366	1812
	$D + F + L + T_s$	-1039	-614	6	961	-3751
	$D + F + L + T_s + E$	-1117	-788	-16	833	-10188
	$1.05D + F + L + 1.25P + T_A$	-2005	-2980	165	37136	2914
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-1633	-3510	14	19995	9888
	$D + F + L + P + T_A$	-1845	-2914	147	28111	3168
	$D + F + L + P + T_A + E'$	-1427	-3360	-6	10770	9651

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

CONTAINMENT STRUCTURE - SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES (psi)

SECTION	LOAD CASE	CONCRETE σ_{em}	CONCRETE σ_{eh}	CONCRETE τ	REINFORCING STEEL σ_m	REINFORCING STEEL σ_h
N-O	$D + F + L + T_o$	-708	-696	24	12251	15290
	$D + F + L + T_o + E$	-910	-1070	48	19168	26123
	$D + F + L + T_s$	-537	-492	32	8742	5101
	$D + F + L + T_s + E$	-930	-1080	33	20719	27297
	$1.05D + F + L + 1.25P + T_A$	-484	-728	98	22724	30046
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-861	-1203	164	30315	41218
	$D + F + L + P + T_A$	-763	-743	114	25316	28528
	$D + F + L + P + T_A + E'$	-1098	-1198	154	29196	40450
P-Q	$D + F + L + T_o$	-1251	-688	22	17161	15582
	$D + F + L + T_o + E$	-1562	1355	43	24970	30500
	$D + F + L + T_s$	-911	-446	44	8506	3814
	$D + F + L + T_s + E$	-1165	-630	60	17945	13720
	$1.05D + F + L + 1.25P + T_A$	-1821	-670	44	32542	30141
	$1.05D + F + L + 1.25P + T_A + 1.25E$	-2115	-1086	74	35862	40468
	$D + F + L + P + T_A$	-1697	-599	65	26824	30012
	$D + F + L + P + T_A + E'$	-1621	-1251	79	29984	41033

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

NOTES:

1. Loading Cases I, II, III, & IV are working stress analysis whereas Loading Cases V, VI, & VII are yield stress analysis.
2. For notation see next page.
3. All concrete extreme fiber stresses sc are for the inside surface. Outside surfaces stresses are indicated by ( ). The stresses listed are the controlling stresses for that section.
4. Computed vs. allowable ratios for Cases V, VI, and VII include appropriate ϕ factors, e.g.,f $\sigma_a/f_a = \sigma_a/\phi_a(f_c)$.
5. Allowable shear stresses include stirrups whenever applicable.
6. Deviations in allowable stresses are in accordance with Section 5.1.2.2.
7. Reinforcing steel is Type A615, GR 60.
8. See Appendix 5E for later tables associated with an evaluation to reduce the original amount of required containment prestress.

NOTATION

D	Dead load
F	Prestress
P	LOCA pressure load
E	Earthquake (OBE)
E'	Earthquake (SSE)
T_A	Accident temperature
f_c	Concrete compressive strength at age of 28 days
f_y	Steel re-bar yield stress
f_a	Allowable concrete axial stress
f_{ce}	Allowable concrete axial & flexure stress
y	Allowable concrete shear stress including stirrups if applicable
f_s	Allowable steel stress
σ_a	Nominal membrane stress
σ_e	Combined axial & flexure nominal stress
τ	Actual shear stress
h	Subscript indicating hoop direction
m	Subscript indicating meridional direction
p_h	Hoop steel percentage
p_m	Meridional steel percentage

TABLE 5-1

STRESS ANALYSIS RESULTS[8]

+	Tensile stresses
-	Compressive stresses
	Cracked section analysis
ν_{ci}	Section 5.1.2.2 (loads necessary to cause structural yielding)
ν_{cw}	Section 5.1.2.2 (loads necessary to cause structural yielding)

TABLE 5-2

ASSUMED INTERNALLY GENERATED MISSILES IN CONTAINMENT STRUCTURE

	ITEM AND LOCATION	WEIGHT	MASS/ AREA lb/in^2	SOURCE [b,c]	SHAPE	IMPACT VELOCITY ft/sec	POINT OF IMPACT
I.	REACTOR VESSEL						
	a) Closure Head Nut	116	.0034	1	Cylindrical	35.5	Overhead Shielding Slab
	b) Closure Head Nut and Stud	710	.0477	1	Cylindrical	21.2	Overhead Shielding Slab
	c) Instrument Assembly	335	.0445[a]	2	Rod	49.4	Overhead Shielding Slab
	d) Instrument	165	.0219[a]	2	Flat Rod	211	Overhead Shielding Slab
II.	STEAM GENERATOR						
	a) Primary Manway Cover	1000	6.57[a]	2	Disk	156.6	Steam Generator Walls
	b) Primary Manway Cover Stud and Nut	24	7.64	1	Rod	33	Steam Generator Walls
	c) Secondary Handhole Cover	150	3.76[a]	2	Disk	181	Steam Generator Walls
	d) Secondary Handhole Cover Stud and Nut	4	4.04	1	Rod	17.9	Steam Generator Walls
	e) Secondary Manway Stud	4.67	.0098	1	Rod	506.7	Pressurizer Shield Wall
	f) Inspection Port Cover	30	1.72	2	Disk	668	Steam Generator Walls
	g) Inspection Port Cover Stud and Nut	1.33	3.02	1	Rod	1295	Steam Generator Walls
III.	PRESSURIZER						
	a) Safety Valve & Flange	550/800	.4540	2	Cylindrical	97.2/80.6	Pressurizer Shield Wall
	b) Valve Flange Bolt	3.75	.0079	1	Rod	16.5	Pressurizer Shield Wall
	c) Relief Valve & Flange	150	.0311[a]	2	Cylindrical	150	Pressurizer Shield Wall
	d) Manway Cover	680	.0150[a]	2	Disk	369	Pressurizer Shield Wall
	e) Upper Temp. Element	2.75	.0052[a]	2	Rod	81.8	Pressurizer Shield Wall
	f) Lower Temp. Element	3	.0056[a]	2	Rod	80.2	Pressurizer Shield Wall
	g) Safety Valve	200	.1648	2	Cylindrical	158	Pressurizer Shield Wall
IV.	CONTROL ROD DRIVER	1100	.0365	2	Cylindrical	58	Overhead Missile Shield
V.	MAIN PUMP & PIPE						
	a) Temp. Nozzle & Rtd	11.1	.0209[a]	2	Cylindrical	81.3	Secondary Shield Wall
VI.	SURGE & SPRAY LINES						
	a) Thermal Well & Rtd	1.75	.0033[a]	2	Cylindrical	105	Secondary Shield Wall

TABLE 5-2

ASSUMED INTERNALLY GENERATED MISSILES IN CONTAINMENT STRUCTURE

ITEM AND LOCATION	WEIGHT	MASS/ AREA lb/in^2	SOURCE (b,c)	SHAPE	IMPACT VELOCITY ft/sec	POINT OF IMPACT
VII. SHUTDOWN COOLING						
a) Line Valve Stem	85	.0450	2	Cylindrical	50.4	Secondary Shield Wall

[a] Cross-section used is projected edge area (dia x thk) all other areas are minimum projected area.
[b] Stored mechanical strain energy converted to kinetic energy by separation of stud.
[c] Hydrostatic pressure force converted to fluid jet force by separation of mechanical restraint.

5.2 ISOLATION SYSTEM

5.2.1 DESIGN BASIS

The general design basis governing isolation valve requirements is that the leakage through all fluid penetrations not serving engineered safety feature systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Containment Structure, and various types of isolation valves. These double barriers support containment integrity requirements in Modes 1-4. During refueling operations, containment closure is established. As required by Reference 1, containment closure is also established during reduced inventory operations. These containment integrity and closure requirements are defined in the Technical Specifications.

Containment Structure isolation (Containment Isolation Signal and Safety Injection Actuation Signal) is initiated at a setpoint between the Technical Specification high pressure value and a value that is based on the minimum pressure compatible with normal operating conditions. Valves which isolate penetrations that are directly open to the Containment Structure, such as the purge valves and sump drain valves, will also be automatically closed on an isolation signal.

As required by Three Mile Island Action Item II.E.4.2, Containment Isolation Dependability, non-essential fluid systems shall be automatically isolated. Essential and non-essential fluid systems were defined in Reference 2 and approved in Reference 3. Essential fluid systems are those that are actively required during the early stages of an accident to control and mitigate the consequences of an accident. The essential fluid systems are: high and low pressure safety injection (penetrations 3, 4, 5, 6), containment spray (penetrations 9, 10), service water to the containment air coolers (penetrations 25, 26, 27, 28), auxiliary feedwater (penetrations 21, 22), Reactor Coolant System charging (penetration 2B), sump recirculation (penetrations 11, 12), service water outlet (penetrations 29, 30, 31, 32), main steam lines (penetrations 35, 36) and containment pressure instrument lines (penetrations 72, 77, 78, 83). These essential fluid systems are not automatically isolated by safety injection actuation signal or containment isolation signal. The remaining fluid lines are considered non-essential. These penetrations are normally locked closed, close on safety injection actuation signal, containment isolation signal, steam generator isolation signal, or containment radiation signal or close via check valves because of a loss of pipe integrity. Three containment penetrations are not classified as either essential or non-essential. They are instrument air (penetration 19A), component cooling water to the reactor coolant pumps (penetration 16), component cooling water from the reactor coolant pumps (penetration 18). These three penetrations are isolated on a containment isolation signal.

Fluid penetrations serving engineered safety feature systems also meet the design basis double barrier criterion, but do not close upon a containment isolation signal.

All containment isolation valves are designed to ensure leak-tightness and reliability of operation. Containment isolation globe, check, and gate valves meet the requirements of MSS-SP-61 and containment isolation butterfly valves meet the requirements of American Water Works Association C-504. Required valve closing times are achieved by appropriate selection of valve, operator type, and operator size.

Upon receiving the appropriate Engineered Safety Features Actuation Signal as described in Chapter 7, all isolation valves required to isolate the containment from the surrounding environment and other systems within the plant close automatically, if not already closed.

With the exception of the containment sample solenoid valves associated with the Post-Accident Monitoring System, the containment isolation valves are provided with handswitches, located in the Control Room, for normal and backup control during an emergency. The normally closed and deenergized containment sample solenoid valves, associated with the Post-Accident Monitoring System are key operated from 1C101 and 1C102 on the 45' Elevation of the Auxiliary Building.

In most systems, standard valve operators are sufficient to close their respective valves in an acceptable time limit as determined by the size of the line and the system it is serving.

Isolation valves of other systems are sized and actuating times determined, depending on the amount of radioactivity those systems could release from containment in the event of a LOCA. Automatic actuation of the isolation valves not required by the engineered safety features will ensure a rapid closure independent of the reactor operator.

All system isolation valves are located either within the Containment Structure or within the adjoining penetration rooms. In order to provide protection from missile damage, all isolation valves located within the Containment Structure are backed up by a valve located outside the containment.

The two penetration rooms are located approximately 180° apart to insure adequate separation of redundant piping and valving. Both penetration rooms are within the Auxiliary Building which is a Seismic Category I structure up to Elevation 69'0" (Section 5A.2.1), designed to protect against potential horizontal tornado missiles and turbine missiles by virtue of a 2'6"-thick reinforced concrete roof and walls and a 3'0"-thick concrete wall between the Auxiliary and Turbine Buildings.

5.2.2 SYSTEM DESIGN

The fluid penetrations which require isolation after an incident may be classed as follows:

Type I. Each line connecting directly to the RCS has two Containment Structure isolation valves. One valve is external, and the other is internal to the Containment Structure. These valves may be either a check valve and a remotely-operated valve, or two remotely-operated valves, depending upon the direction of normal flow. (Except Penetration No. 2A which is described in Figure 5-10 and Table 5-3.)

Type II. Each line connecting directly to the Containment Structure atmosphere has two isolation valves. At least one valve is external, and the other may be internal or external to the Containment Structure. These valves may be either a check valve and a remotely-operated valve or two remotely-operated valves, depending upon the direction of normal flow. The containment sump recirculation line has one remotely-operated valve, which is external to the Containment Structure. The valve is located as close to the containment as valve size will permit.

Type III. Each line not directly connected to the RCS or not open to the Containment Structure atmosphere has at least one valve, either a check valve or a remotely-operated valve. The valve is located external to the Containment Structure.

Type IV. Lines which penetrate the Containment Structure and are connected to either the Containment Structure atmosphere or the RCS, but which are never opened during reactor operation (except as permitted under Footnote “A” of Table 5-3), have valves with provisions for locking in a closed position.

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Containment Structure isolation valve arrangement fits into its representative system. Each valve may be tested periodically either during normal operation or during shutdown conditions to insure its operability when needed.

For convenience, each different valve arrangement is shown in Figure 5-10. Figure 5-10 identifies the containment isolation valves for all containment penetrations, including those exempted from local leak rate testing. Figure 5-10 defines the containment isolation valves for Technical Specification 3.6.3. The symbols on these figures are identified on Figure 9-1. Figure 5-10 lists the mode of actuation, the type of valve, its position during normal plant operation (Mode 1), and the position of each containment isolation valve under Containment Structure isolation conditions. The valves shown are for Unit 1. Some manual valves and check valves on Unit 2 have different equipment identification numbers. The specific system penetrations to which each of these arrangements is applied are also presented. Table 5-3 identifies the penetrations and associated containment isolation valves which are subject to Type C local leak rate testing. The table gives signals and closure times for each containment isolation valve.

The failure characteristics of the isolation valves are presented with the respective system evaluation of which the valve is a part, e.g., containment spray system.

There is sufficient redundancy in the instrumentation circuits of the engineered safety features protective system to minimize the possibility of inadvertent tripping of the isolation system. Further discussion of this redundancy and the instrumentation signals which trip the isolation system is presented in Chapter 7.

Those containment penetrations which communicate directly between the containment atmosphere and the outside environment are provided with normally closed isolation valves and/or blind flanges.

Some lines which penetrate the containment are not open to the containment atmosphere. Where these lines are located between the secondary shield wall (missile barrier) and the containment shell, they are considered to be "closed systems," not subject to rupture following a LOCA. The main steam lines, the feedwater lines, and the service water lines which provide cooling water to the coolers in the ventilation air handling units all fall within this category. The portions of the main steam lines and the feedwater lines located inside the secondary shield wall also fall within this category since the portions of these lines that are inside the secondary shield are protected from missiles by the concrete floor within the containment at Elevation 69'0" (refueling level floor).

Any leakage through these closed lines would have been detected as part of the preoperational integrated leak test of the containment.

Some lines which penetrate the containment are part of the closed piping systems located outside the Containment Structure and, consequently, are not supplied with remotely-operated isolation valves outside the containment. The following lines fall within this category:

a. The reactor coolant pump cooling water supply lines
b. The shutdown coolant inlet line

These lines are in systems which are normally water filled and normally operate at positive pressures. Any significant leakage from these lines will be detected during plant operation.

Penetration 1B, the containment vent header to the waste gas surge tank, is not connected directly to the RCS nor open to the containment atmosphere and is classified as a Type III penetration. However, two containment isolation valves are provided outside of the containment structure. Both of the valves are periodically local leak rate tested.

Penetration 8, the containment sump normal drain, is imbedded in the containment base slab. Although this is a Type II penetration, it would be highly impractical to locate one of the automatic valves inside the containment. These two automatic valves are located in a pipe tunnel in the Auxiliary Building. The first valve is located as close as practicable to the containment, and both valves are remote from any source of external damage.

Penetrations 9 and 10, the containment spray lines, do not have remotely-operated valves, but have two check valves for each penetration. Although classified as Type II penetrations, they are required to be open following an incident to allow the containment spray system to operate.

If a pipe near the inside containment wall at penetration 3, 4, or 5 were to fail during a LOCA, there would be no adverse consequences. Water from the safety injection system is continually pumped through these penetrations during the LOCA, thus preventing any release of fission products or contamination to the site boundary.

Penetration 20 consists of three individual lines (20A, 20B, 20C) in one penetration assembly, each with a check valve outside the containment. Penetrations 20B and 20C are also provided with a check valve inside containment. These lines are not directly connected to the RCS. They are connected to vessels which, in turn, may be isolated by valving from the RCS.

Penetration 39, which is connected to the RCS, is a Type IV penetration. The containment isolation valves outside the Containment Structure are normally closed and locked closed during reactor operation. Inside the Containment Structure between Penetration 39 and the connection to the RCS, there are remotely-operated valves which are normally in the closed position and serve as an additional barrier to prevent a potential leakage path to the environment (Figure 5-10, Sheet 26).

Containment penetrations 62 and 64 were originally designated for the plant heating system, but the system was never used to heat containment. Therefore, the plant heating containment subsystem was retired, and the penetration piping was cut, capped, and welded inside containment. The caps and their associated welds provide the only necessary containment barrier, and isolation valves exterior to the Containment Structure are not required.

5.2.3 TESTING AND INSPECTION

All isolation valves were tested by the manufacturer for leak-tightness and reliability of operation prior to delivery. Valves were subjected to the manufacturer's standard tests on an individual basis to insure reliability. An acceptable leak-rate was determined and specified for each valve purchased.

Each valve was tested after installation to insure its leak-tightness. The valve operators specified for these valves have a proven record of a number of years of reliability in respect to method of operation and material used. Throughout plant life, these valves will be tested periodically. Those which cannot be tested during operation (those which must remain open or closed), will be tested during the scheduled shutdowns and plant outages.

5.2.4 REFERENCES

1. Generic Letter 88-17, Loss of Decay Heat Removal, 10 CFR 50.54(f), dated October 17, 1988
2. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated November 20, 1979, Follow-up Actions Resulting from TMI-2 Incident (Lessons Learned Short Term)
3. Letter from R. W. Reid (NRC) to A. E. Lundvall, Jr. (BGE), dated April 7, 1980, Staff Evaluation of the Implementation of Category "A" Lessons Learned Requirements

TABLE 5-3

CONTAINMENT ISOLATION VALVES SUBJECT TO TYPE C LOCAL LEAK RATE TESTING

PENETRATION NO.	TYPE	ISOLATION CHANNELS	UNIT	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (Seconds)
1A	I	SIAS A	1,2	PS-5465-CV	Reactor Coolant and Pressurizer	≤ 7
		SIAS A	1,2	PS-5466-CV	Sampling	≤ 7
		SIAS A	1,2	PS-5467-CV		≤ 7
		SIAS B	1,2	PS-5464-CV		≤ 7
1B	III	SIAS A	1,2	WGS-2180-CV	Containment Vent Header to Waste Gas	≤ 10
		SIAS B	1,2	WGS-2181-CV		≤ 10
1C	I	SIAS A	1,2	CVC-506-CV	Reactor Coolant Pump Seals Controlled	≤ 7
		SIAS B	1,2	CVC-505-CV	Bleedoff	≤ 7
1D	IV	NA	1,2	PS-6529-SV[a]	Post Accident Sampling Liquid Return to Reactor Coolant Drain Tank	NA
2A	I	SIAS A	1,2	CVC-515-CV	Letdown Line	≤ 13
		SIAS B	1,2	CVC-516-CV		≤ 13
		NA	1,2	CVC-105		NA
		NA	1,2	CVC-103		NA
2B	I	NA	1,2	CVC-517-CV[b]	Charging Line	NA
		NA	1,2	CVC-518-CV[b]		NA
		NA	1,2	CVC-519-CV[b]		NA
		NA	1,2	CVC-435-RV[b]		NA
		NA	1,2	CVC-184[b]		NA
7A	IV	NA	1,2	Blind Flange	Integrated Leak Rate Testing	NA
		NA	1,2	ILRT-1		NA
7B	IV	NA	1,2	Blind Flange	Integrated Leak Rate Testing	NA
		NA	1,2	ILRT-2		NA
8	II	SIAS A	1,2	EAD-5462-MOV[b]	Containment Normal Sump	≤ 13
		SIAS B	1,2	EAD-5463-MOV[b]		≤ 13
13	IV	CRS A	1	CPA-1410-CV[g]	Purge Air Inlet	≤ 15[g]
			1	Blind Flange[g]		NA
		CRS A	2	CPA-1410-CV[g]	Purge Air Inlet	≤ 15[g]
			2	Blind Flange[g]		NA

TABLE 5-3

CONTAINMENT ISOLATION VALVES SUBJECT TO TYPE C LOCAL LEAK RATE TESTING

PENETRATION NO.	TYPE	ISOLATION CHANNELS	UNIT	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (Seconds)
14	IV	CRS A	1	CPA-1412-CV[g]	Purge Air Outlet	≤ 15[g]
			1	Blind Flange[g]		NA
		CRS A	2	CPA-1412-CV[g]	Purge Air Outlet	≤ 15[g]
			2	Blind Flange[g]		NA
15	II	SIAS A	1,2	RE-5291-CV	Purge Air Monitor	≤ 7
		SIAS B	1,2	RE-5292-CV		≤ 7
16	III	CIS A	1,2	CC-3832-CV	Component Cooling Water Inlet	≤ 18
18	III	CIS B	1,2	CC-3833-CV	Component Cooling Water Outlet	≤ 18
19A	III	NA	1	IA-337	Instrument Air	NA
		NA	2	IA-175		NA
		CIS A	1,2	IA-2080-MOV		≤ 13
19B	IV	NA	1	PA-1040[a]	Plant Air	NA
		NA	2	PA-137[a]		NA
		NA	1,2	PA-1044[a]		NA
20A	III	NA	1	N2-344	Nitrogen Supply	NA
		NA	2	N2-347		NA
		NA	1,2	N2-612-CV[a]		NA
		NA	1,2	N2-622-CV[a]		NA
		NA	1,2	N2-632-CV[a]		NA
		NA	1,2	N2-642-CV[a]		NA
20B	III	NA	1	N2-389	Nitrogen Supply	NA
		NA	1	N2-345		NA
		NA	2	N2-348		NA
		NA	2	N2-395		NA
20C	III	NA	1	N2-346	Nitrogen Supply	NA
		NA	1	N2-392		NA
		NA	2	N2-349		NA
		NA	2	N2-398		NA
23	III	SIAS A	1,2	RCW-4260-CV	Reactor Coolant Drain Tank Drains	≤ 10
24	III	SIAS B	1,2	PS-6531-SV	Oxygen Sample Line	≤ 7

TABLE 5-3

CONTAINMENT ISOLATION VALVES SUBJECT TO TYPE C LOCAL LEAK RATE TESTING

PENETRATION NO.	TYPE	ISOLATION CHANNELS	UNIT	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (Seconds)
37	II	NA	1	PSW-1019	Plant Water	NA
		NA	1	PSW-1008		NA
		NA	2	PSW-1020		NA
		NA	2	PSW-1009		NA
38	III	NA	1,2	DW-5460-CV[a]	Demineralized Water	NA
39	IV	NA	1,2	SI-463	Safety Injection Tank Test Line	NA
		NA	1,2	SI-455		NA
44	III	NA	1	FP-141-A	Fire Protection	NA
		NA	1	FP-141-B		NA
		NA	2	FP-145-A		NA
		NA	2	FP-145-B		NA
47A	II	NA	1,2	PS-6540A-SV[a]	Hydrogen Sample Outlet	NA
		NA	1,2	PS-6507A-SV[a]		NA
47B	II	NA	1,2	PS-6540E-SV[a]	Hydrogen Sample Outlet	NA
		NA	1,2	PS-6507E-SV[a]		NA
47C	II	NA	1,2	PS-6540F-SV[a]	Hydrogen Sample Outlet	NA
		NA	1,2	PS-6507F-SV[a]		NA
47D	II	NA	1,2	PS-6540G-Sa[a]	Hydrogen Sample Return	NA
		NA	1,2	PS-6507G-Va[a]		NA
48A	II	SIAS B	1	HP-6900-MOV[f]	Containment Vent Isolation	≤ 15
		SIAS A	2			
		SIAS A	1	HP-6901-MOV[f]		≤ 15
		SIAS B	2			
48B	II	NA	1,2	HP-104	Hydrogen Purge Inlet	NA
		NA	1,2	HP-6903-MOV		NA
49A	II	NA	1,2	PS-6540B-SV[a]	Hydrogen Sample	NA
		NA	1,2	PS-6507B-SV[a]		NA
49B	II	NA	1,2	PS-6540C-SV[a]	Hydrogen Sample	NA
		NA	1,2	PS-6507C-SV[a]		NA

TABLE 5-3

CONTAINMENT ISOLATION VALVES SUBJECT TO TYPE C LOCAL LEAK RATE TESTING

PENETRATION NO.	TYPE	ISOLATION CHANNELS	UNIT	ISOLATION VALVE IDENTIFICATION NO.	FUNCTION	ISOLATION TIME (Seconds)
49C	II	NA	1,2	PS-6540D-SV[a]	Hydrogen Sample	NA
		NA	1,2	PS-6507D-SV[a]		NA
50	IV	NA	1,2	Blind Flange	Integrated Leak Rate Testing	NA
		NA	1,2	Blind Flange		NA
59	IV	NA	1	SFP-170	Refueling Pool Inlet	NA
			NA	1	SFP-171	NA
		NA	2	SFP-178		NA
		NA	2	SFP-179		NA
60	IV	NA	1,2	ES-144	Steam to Reactor Head Laydown	NA
		NA	1,2	ES-142		NA
61	IV	NA	1	SFP-176	Refueling Pool Outlet	NA
		NA	1	SFP-174		NA
		NA	1	SFP-172		NA
		NA	1	SFP-189		NA
		NA	2	SFP-184		NA
		NA	2	SFP-182		NA
		NA	2	SFP-180		NA
		NA	2	SFP-186		NA
84	IV	NA	1, 2	Blind Flange	Integrated Leak Rate Testing 12” Vent	NA
		NA	1, 2	Blind Flange		NA

[a] May be open on an intermittent basis under administrative control. An evaluation under the rules of 10 CFR 50.59 shall be performed if additional valves are identified that could be opened on an intermittent basis under administrative control. This evaluation shall include, but not be limited to, actions necessary to place the valve in its post-accident condition. These actions could include the need to station an operator at the valve controls who is in constant communication with the Control Room; instructions to this operator to close the valve in an accident situation; or evaluating the environmental conditions to assure that an operator would have access to close the valve and that this action will prevent the release of radioactivity outside the containment. Plant procedures would have to be revised to include any additional administrative controls deemed necessary to allow safe opening of these valves.

TABLE 5-3

CONTAINMENT ISOLATION VALVES SUBJECT TO TYPE C LOCAL LEAK RATE TESTING

[b] This valve is not tested as part of the Local Leak Rate Test Program. This valve is located in a water filled penetration and does not constitute a potential primary containment atmospheric leak path.

[d] Deleted.

[f] Containment vent isolation valves shall be opened for containment pressure control, airborne radioactivity control, and surveillance testing purposes only.

[g] Containment Purge Inlet and Outlet Penetrations are closed by a blind flange during Modes 1-4. During Modes 5-6, closure is provided by CV-1410 and CV-1412. Closure times for these CVs are only applicable during Mode 6 when the valves are required to be operable and they are open.

5.3 EXTERNAL MISSILES, SNUBBERS, AND WATERTIGHT DOORS

5.3.1 EXTERNAL MISSILES

5.3.1.1 Containment Structure Design Missiles

Turbine Generator Produced Missiles

Both of the turbine generator suppliers made a study of failure of rotating elements of steam turbines and generators. The postulated types of failures are: (a) failure of rotating components operating at or near normal operating speed and, (b) failure of components that control admission of steam to the turbine resulting in destructive shaft rotational speed.

Failure at or Near Operating Speed

All of the known turbine and generator rotor failures at or near rated speed resulted from the combination of severe strain concentrations in relatively brittle materials. New alloys and processes have been developed and adopted to minimize the probability of brittle fracture in rotors, wheels, and shafts. Careful control of chemistry and detailed heat treating cycles have greatly improved the mechanical properties of all of these components. Transition temperatures [the temperature at which the character of the fracture in the steel changes from brittle to ductile, often identified as nil-ductility transition temperature (NDTT)] have been reduced on the low temperature wheel and rotor applications for nuclear units to well below startup temperatures. Improved steel mill practices in vacuum pouring and alloy addition have resulted in forgings which are much more uniform and defect free than ever before. More comprehensive vendor and manufacturer tests involving improved ultrasonic and magnetic particle testing techniques are better able to discover surface and internal defects than in the past. Laboratory investigation has revealed some of the basic relationships between structure strength, material strength, NDTT and defect size, and location, so that the reliability of the rotor as a structure has been significantly improved.

New starting and loading instructions have been developed to reduce the severity of surface and bore thermal stress cycles incurred during service. The new practices include:

- Better temperature sensors;
- Better control devices for acceleration and loading; and
- Better guidance for station operators in the control speed, acceleration, and loading rates to minimize rotor stresses.

New rotor designs have also contributed to the reduced likelihood of a turbine missile. The original Unit 1 low-pressure turbine rotors were conventional rotors with shrunk-on wheels and axial keyways. These built-up rotors were replaced with ones manufactured from monoblock forgings in the 2004 Unit 1 refueling outage. The new Unit 1 rotors are not susceptible to the keyway stress-corrosion cracking mechanism. The result of this is that brittle fracture failure mode (and turbine missile probability) at near-rated speed is essentially eliminated for Unit 1. Therefore, the only credible Unit 1 turbine missile scenario is a ductile failure of the limiting rotor due to a significant over-speed event (greater than 170% of rated speed).

Failure at Destructive Shaft Rotational Speeds

Improvement of rotor quality discussed above, while reducing the chance of failures at operating speed, tend to increase the hazard level associated with

unlimited overspeed because of higher bursting speed. Therefore, turbine overspeed protection systems have been evaluated as follows:

UNIT 1

The Unit 1 main turbine has a GE Speedtronic Mark VI control system. This system is triple modular redundant to ensure a high level of reliability. This system interfaces with the turbine stop valves, control valves, and combined intermediate valves to control the unit.

Overspeed protection is provided at three levels: control, primary, and emergency. Control protection comes through closed loop speed control using the turbine control valves. There are two sets of three magnetic speed pick-ups; one set is for primary protection and the other set is for emergency protection. Primary over-speed protection is provided by a set of three controllers that provide 2-of-3 voting for the turbine trip. The emergency overspeed protection is provided by an independent triple redundant system which also trips the turbine. Three independent speed signals are used permitting speed control with one of the signals failed. The overspeed protection systems will secure steam to the turbine as follows:

a. Main and secondary steam inlets have the following valves in series:

   Stop valves - actuated by the hydraulic fluid trip system via the primary and emergency overspeed protection systems, providing two levels of protection, each of which is triple redundant.

   Control valves - controlled by the speed-load control unit and tripped closed by the primary and emergency overspeed protection systems, providing three levels of redundancy for closure.

   Combined intermediate valves in cross-around systems - actuated by the speed load control unit and tripped closed by the primary and emergency overspeed protection systems.

b. Uncontrolled Extraction Lines to Feedwater Heaters

   Positive closing nonreturn valves are provided in extraction lines which have sufficient stored energy to cause a dangerous overspeed condition on a turbine trip. The valves close on a turbine trip via the extraction air relay dump valve which is actuated by the hydraulic fluid trip system. The valves are designed for local or remote manual periodic tests to assure proper operation. The station piping, heater, and check valve systems were reviewed during the design stages to assure that the entrained steam cannot overspeed the unit beyond safe limits.

UNIT 2

a. Main and secondary steam inlets have the following valves in series:

   Governor valves - controlled by the speed governor and tripped closed by emergency governor and backup overspeed trip, thus providing three levels of control redundancy.

   Throttle valve - actuated by the emergency governor and backup overspeed trip, thus providing two levels of control redundancy.

Reheat stop and intercept valves in cross-around systems - these are actuated by the speed governor, emergency governor and backup overspeed trips.

The speed sensing devices for the governor and emergency governor are separate from each other, thus providing two independent lines of defense.

b. Uncontrolled Extraction Lines to Feedwater Heaters

Positive closing nonreturn valves are provided in extraction lines which have sufficient stored energy to cause an overspeed condition on a turbine trip. These valves are designed for local or remote-manual periodic tests to assure proper operation. The station piping, heater and check valve systems were reviewed to assure that the entrained steam cannot overspeed the unit beyond safe limits.

Special field tests are made of new components to obtain design information and to confirm proper operation. These include the capability of controls to prevent excessive overspeed on loss of load.

Careful analysis of all past failures has led to design, inspection, and testing procedures to substantially eliminate destructive overspeed as a possible cause of failure in modern design units.

Missile Protection

The NRC-preferred method of protecting against turbine missiles is to ensure that turbine missile generation probability, P1, is maintained at a value of less than 10^{-5} per year (Reference 14). While this method has been used for the Unit 2 turbine, it was not a viable method for Unit 1 until the replacement of the low pressure turbine rotors during the 2004 Unit 1 refueling outage.

Maintaining a low value for P1 is accomplished by performing regular, vendor-approved maintenance, and testing of the turbine control and overspeed protection systems. The test intervals have a direct effect on overspeed control system failures and, therefore, turbine missile generator probability, P1.

As mentioned previously, since the new Unit 1 monoblock rotors have no credible failure mode at near-rated speed, the only credible turbine missile scenario for Unit 1 is a significant overspeed event. It follows therefore that the Unit 1 turbine missile probability is essentially the same as the probability of a significant overspeed.

The Unit 1 overspeed probability is calculated by General Electric in Reference 16 as less than 3×10^{-6} per year. This is based on the Mark VI turbine control system, monoblock LP turbine rotors, and maintaining the current (extended) valve test intervals. General Electric approved maintenance practices are assumed as well. Per Reference 14, maintaining P1 at less than 10^{-5} per year is an acceptable method of managing turbine missile risk. Therefore, the turbine missile risk from Unit 1 is acceptably low for all systems, structures, and components, and no further analysis is required.

For the Westinghouse turbine generator, the guidance in Reference 10 was used. Reference 10 guidance is the same as that used for the General Electric turbine where turbine missile risk can be effectively managed by maintaining the turbine missile generation probability, P1, less than 10^{-5} for unfavorably oriented turbine generators.

The missile generation probability for the Unit 2 (Westinghouse) turbine is calculated by Reference 11. In Reference 11, Westinghouse performed an evaluation of the probability of generating turbine missiles as a direct function of the testing frequency for the turbine governor valves and throttle valves. The report focuses on the probability of turbine missile ejection due to destructive overspeed (runaway speed in excess of approximately 180%). The turbine missile ejection frequencies in Reference 11 were calculated following the same basic methodology as is described in Reference 12. In a supplemental safety evaluation (Reference 13), issued to Westinghouse, the NRC staff accepted the Reference 12 methodology for use in the determination of the probability of turbine missile generation. The turbine generator failure rates used in Reference 11 for turbine governor and throttle valves were based on plant operating experience over a data collection period from 1990 through and including 1995. This time period provided failure rates based on current valve design and maintenance practices while retaining adequate time for rare events to occur. Westinghouse added an allowance to cover any model uncertainties and to account for the probability of missile ejection from design and intermediate overspeed events. The destructive overspeed model was constructed assuming that a loss of load or system separation occurred. The frequency of system separation was calculated to be 0.29 per year; however, a more conservative value of 0.4 per year was used in the Westinghouse analysis. The conditional probability of missile ejection (e.g., the probability of valve failures) was then multiplied by the frequency of system separation to obtain the probability of missile ejection per year from destructive overspeed. The probability of turbine missile ejection due to destructive overspeed was calculated for turbine valve test intervals of one week, one month, three months, six months, and twelve months.

Values for P1 are given in Reference 11 for various valve test intervals and are below 10^{-5}. Maintaining an initial small value of the probability of a turbine failure as discussed above simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained. In addition, maintaining P1 at a low value is the NRC preferred method for controlling turbine missile risk per References 10 and 13. By focusing on the missile generation probability, we avoid the numerous modeling approximations that often must be made to incorporate interactions of missiles with obstacles, their trajectories as they deflect off barriers, and the identification and location of safety-related targets.

The analysis for turbine missiles from the Unit 2 turbine is based on the current missile generation probabilities provided by Westinghouse and our current testing interval. The analysis shows that we meet current acceptance criteria.

<u>Tornado Produced Missiles</u>

For an analysis of horizontal missiles created by a tornado having maximum wind speeds of 300 mph, two horizontal missiles were considered. One is a horizontal missile equivalent to a 12' plank with a 12"x4" cross-section, traveling end-on at 300 mph; the second is a 4000 lb automobile traveling at a speed of 50 mph at no more than 25' above the ground.

For the wood horizontal missile, calculations based on energy principles indicate that, because the impact pressure exceeds the ultimate compressive strength of wood by a factor of about four, the wood would crush due to impact. However, this could cause a secondary source of missiles if the impact force is sufficiently large to cause spalling of the free (inside) face. The compressive shock wave which

propagates inward from the impact area generates a tensile pulse which, if it is large enough, will cause spalling of concrete as it moves back from the free (inside) surface. This spalled piece moves off with some velocity due to energy trapped in the material. Successive pieces will spall until a plane is reached where the tensile pulse becomes smaller than the tensile strength of the concrete. From the effects of impact of the wood plank, this plane in a conventionally reinforced concrete section would be located approximately 3" from the free (inside) surface. However, since the Containment Structure is prestressed, there will be residual compression in the free face, as the tensile pulse moves out and spalling will not occur. Calculations indicate that, in the impact area, a 2" or 3"-deep crushing of concrete should be expected as result of excessive bearing stress due to impact.

For the automobile missile, using the same methods as in the original turbine failure analysis, the calculated depth of penetration is 1/4" and, for all practical purposes, the effect of impact on the Containment Structure is negligible.

From the above, it can be seen that the horizontal tornado-generated missiles neither penetrate the Containment Structure wall nor endanger the structural integrity of the Containment Structure or any components of the RCS.

Removable Slabs, Blocks and Partitions

It is improbable that removable blocks which are used only in the waste processing area in the Auxiliary Building would break loose and become missiles, even during a DBE. These blocks are self-locking and contain staggered horizontal and vertical joints. All removable concrete slabs are located in the Auxiliary Building. These slabs are placed over low pressure radwaste equipment, such as filters and demineralizers, and weigh approximately 4000 lbs. It is unlikely that a slab could receive a seismic acceleration in the upward direction sufficient to cause the slab to become a missile.

5.3.1.2 Other Structures Design Missiles

Unit 1 Turbine Missile Analysis

The Unit 1 turbine (General Electric) has a missile generation probability (P1) of less than 10^{-5} per year. Therefore, per Reference 14, as long as P1 is maintained less than 10^{-5} per year, the Unit 1 turbine presents an acceptably low risk and no further analysis of missile risk from the Unit 1 turbine is necessary. This also applies to the turbine missile risk for all equipment, including the safety-related Diesel Generator Building.

Unit 2 Turbine Missile Analysis

The Unit 2 turbine (Westinghouse) has a missile generation probability (P1) of less than 10^{-5} per year. Therefore, per Reference 10, as long as P1 is maintained less than 10^{-5} per year, the Unit 2 turbine presents an acceptably low risk and no further detailed analysis of missile risk from the Unit 2 turbine is necessary.

Conclusion

Based on the above discussion, both Units' turbine generators have an acceptably low probability of generating a missile, and Units 1 and 2 are adequately protected against turbine missiles.

5.3.2 SNUBBERS

All safety-related snubbers must be capable of performing their specified function to ensure that the structural integrity of the Reactor Coolant System and all other safety-

related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this program are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based on maintenance of a constant level of snubber protection to systems. Therefore, inspection intervals vary inversely with the observed snubber failures. These intervals are determined by the number of inoperable snubbers found during the previous inspection, the total population or category size, and the previous inspection interval.

Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection. The decision shall be used in determining the next inspection interval for that category. Inspections performed before an interval has elapsed may be used as new reference points in determining the next interval. However, the results of such early inspections (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of snubber rejection is clearly established, remedied and verified by inservice functional testing, that snubber and any other snubbers that may be generically susceptible, may be exempted from being counted as inoperable. Generically susceptible snubbers are those that are: (1) of a specific make or model; (2) of the same design; and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These snubber installation characteristics shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

A snubber is considered inoperable if it fails to satisfy the acceptance criteria of the visual inspection. When a snubber is found inoperable, a determination of the snubber mode of failure is made. In addition, an engineering evaluation is performed to determine if the supported component or system, or any safety-related component or system has been adversely affected by the inoperability of the snubber. Operation may continue indefinitely if an engineering review and evaluation can document within 12 or 72 hours, depending on applicability of LCO 3.0.8, that the equipment to which the snubber is connected can perform its required safety functions with the snubber inoperable. If the review and evaluation cannot justify that the supported equipment will perform its required functions, the system must be declared inoperable and the applicable action requirements met.

The surveillance program allows inspection intervals to be compatible with a 24-month fuel cycle, up to and including an increase to every other refueling outage. To provide assurance of snubber functional reliability, a representative sample (10%) of the installed snubbers of each type [e.g., small bore (< 8") and large bore (> 8")] will be functionally tested during plant shutdowns or at refueling intervals. Observed failures of these sample snubbers shall require functional testing of additional units (5% for each failure or until every snubber has been functionally tested).

The service life of a snubber is determined by reviewing manufacturer information, snubber service conditions, and associated installation and maintenance records (newly installed snubber, seal replaced, in high radiation area, in high temperature area, etc.) The requirement to monitor snubber service life ensures the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The service life program is designed to uniquely reflect the conditions at Calvert Cliffs. The criteria for evaluating service life is be determined, and documented, by the licensee.

Records provide statistical bases for future determination of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

5.3.3 WATERTIGHT DOORS

Watertight doors are provided in various locations to ensure the protection of safety-related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms. While the plant is in operating Modes 1 through 4, the following watertight doors are determined to be closed at least once per 12 hours, except when the door is being used for normal entry and exit:

a. ECCS Pump Room Doors (4),
b. Service Water Pump Room to Heater Bay Doors (2),
c. Auxiliary Feed Pump Room to Heater Bay Doors (2),
d. Emergency Escape Hatch, Service Water Pump Room from Penetration Room,
e. Main Steam Piping Area from Piping Penetration Room Door,
f. Passage to Main Steam Piping Area Door,
g. Warehouse to Intake Structure Door, (Elevation 12'), and
h. Intake Structure Door from Outside.

5.3.4 REFERENCES

1. Deleted
2. Deleted
3. Deleted
4. Deleted
5. Deleted
6. Deleted
7. Deleted
8. Deleted
9. Deleted
10. Letter from C. E. Rossi (NRC) to J. A. Martin (Westinghouse Electric Corporation), Safety Evaluation Report, dated February 2, 1987, Approval for Referencing of Licensing Topical Reports: March 1974 Report; WSTG-2-P, May 1981; and WSTG-3-P, July 1984
11. Westinghouse Report WCAP-14732, dated June 1997, "Probabilistic Analysis of Reduction in Turbine Valve Test Frequency for Nuclear Plants with Westinghouse BB-296 Turbines with Steam Chests"
12. Westinghouse Report WCAP-11525, dated June 1987, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency"
13. Letter from NRC to the Chairman of the Turbine Valve Test Frequency Evaluation Subgroup, Mr. D. M. Musolf, Manager, Nuclear Support Services, Northern States Power Company, dated November 2, 1989
14. NUREG-1048 Supplement No. 6, Safety Evaluation Report related to the Operation of Hope Creek Generation Station, July 1986
15. Deleted

16. Turbine Missile Analysis Statement, Constellation Nuclear, Calvert Cliffs Unit 1, TB.# 170X413, from David B. Troischt, GE Steam Turbine Technology, September 23, 2005

5.4 SYSTEM DESIGN EVALUATION

This system is not dependent on the operation of a continuous leakage monitoring system for the entire containment, or on a continuous leakage surveillance system for containment penetrations and seals, since neither of these systems is being furnished. Therefore, no analysis of the capability of these systems is necessary.

The penetration room ventilation system provides a partial double containment system and is an additional engineered safety feature.

A full evaluation of the containment system is included in Section 5.1.8, Leakage Monitoring System. The Containment Structure, with its associated engineered safety systems, prevents uncontrolled release of radioactivity to the plant and surrounding areas during normal operating and accident conditions.

5.5 TESTS AND INSPECTION

The quality of both the materials and construction of the Containment Structure was assured by a continuous program of testing and inspection.

Qualified field supervisory personnel and inspectors were assigned to the project to carry out the work in accordance with the specifications and drawings. Project design personnel made frequent visits to the job site to coordinate the construction with the design. Inspectors were experienced and thoroughly familiar with the type of work to be inspected, particularly in the field of prestressed concrete. The inspector was given complete access to the work to perform such examinations as were necessary to satisfy himself that the standards set forth in the applicable codes and specifications were met. Where material did not satisfy the standards, he had the authority to stop work until the necessary alterations were made. Appropriate inspection records were maintained.

5.5.1 PREOPERATIONAL TESTING AND INSPECTION

5.5.1.1 During Construction

Test, code, and cleanliness requirements accompanied each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers were enumerated in the specifications together with the requirements, if any, for test witnessing by Bechtel inspectors. Fabrication and cleanliness standards, including final cleaning and sealing, were described together with shipping procedures. Standards and tests were specified in accordance with applicable regulations, recognized technical society codes, and current industrial practices. Inspection was performed in the shops of vendors and subcontractors as necessary to verify compliance with the specifications.

The following codes and practices were used to establish standards of construction procedures:

ACI 301	-	Specification for Structural Concrete for Buildings
ACI 318	-	Building Code Requirements for Reinforced Concrete
ACI 306	-	Recommended Practice for Cold Weather Concreting
ACI 347	-	Recommended Practice for Concrete Formwork
ACI 605	-	Recommended Practice for Hot Weather Concreting
ACI 613	-	Recommended Practice for Selecting Proportions for Concrete
ACI 614	-	Recommended Practice for Measuring, Mixing, and Placing Concrete
ACI 315	-	Manual of Standard Practice for Detailing Reinforced Concrete Structures
Part UW	-	Requirements for Unfired Pressure Vessels Fabricated by Welding of Section VIII of the ASME, B&PV Code
AISC	-	Steel Manual, Code of Standard Practice
ACI	-	Manual of Concrete Inspection
PCI	-	Inspection Manual
AWS	-	Code for Welding in Building Construction (D 1.0-66 and D 2.0-66)

Dimensional tolerances for construction, unless stated otherwise, conform to AISC Code of Standard Practice for erection of steel, and to ACI 301-66 and ACI 318-63 for placing of concrete.

Concrete

Concrete work was accomplished basically in accordance with ACI 318-63, "Building Code Requirements for Reinforced Concrete" and ACI-301, "Specifications for Structural Concrete for Buildings." Other codes and specifications are listed above. Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, and water. Admixtures were added to improve the quality and workability of the plastic concrete during placement and to retard the set of the concrete. Maximum practical size aggregate, water reducing additives, and a low slump of 2 or 3" were used to minimize shrinkage and creep. Aggregates conformed to "Standard Specifications for Concrete Aggregate," ASTM Designation C33.

Acceptability of aggregates was based on the following ASTM tests. These tests were performed by a qualified commercial testing laboratory.

Test	
L. A. Abrasion	ASTM C131
Clay Lumps Natural Aggregate	ASTM C142
Material Finer No. 200 Sieve	ASTM C117
Mortar Making Properties	ASTM C87
Organic Impurities	ASTM C40
Potential Reactivity (Chemical)	ASTM C289
Sieve Analysis	ASTM C136
Soundness	ASTM C88
Specific Gravity and Absorption	ASTM C127
Specific Gravity and Absorption	ASTM C128
Petrographic	ASTM C295

Cement was Type II low alkali cement as specified in "Standard Specification for Portland Cement," ASTM Designation C150, and was tested to comply with ASTM C114. Fly ash was not used in the concrete for the Containment Structure, or in any other concrete on the project.

Water used in concrete was clean and free from deleterious amounts of acid, alkali, salts, oil, sediment, or organic matter. Water used in concrete mixing was sampled and analyzed by a qualified testing laboratory to assure conformance with specification.

The water-reducing agent, Placewell LS, was selected as the one providing shrinkage similar to that prescribed by ASTM C494, "Specifications for Chemical Admixtures for Concrete." Admixtures containing chlorides were not used.

Concrete mixes were designed in accordance with ACI 613, using materials qualified and accepted for this work. Only mixes meeting the design requirements specified for Containment Structure concrete were used. Trial mixes were tested in accordance with applicable ASTM Codes as indicated below:

Test	
Making and curing cylinder in Laboratory	ASTM C192
Air Content	ASTM C231

Slump	ASTM C143
Compressive Strength Tests	ASTM C39

The concrete had a design compressive strength of 5000 psi at 28 days for the containment wall and dome, and 4000 psi at 28 days for the containment base slab.

Concrete strength, slump, and temperature tests were performed. The purpose of the tests was to ascertain conformance to specifications. The basis for the inspection procedures was the ACI Manual of Concrete Inspection with modifications as set forth in construction specifications for this application.

Test cylinders were cast from the mix selected for construction and the following concrete properties were determined:

Uniaxial creep	ASTM C512
Modulus of Elasticity and Poisson's Ratio	ASTM C469
Autogenous Shrinkage	ASTM C342
Thermal Diffusivity	ASTM C34 and CRD-C36-63
Thermal Coefficient of Expansion	ASTM C342 and CRD-C124-62
Compressive Strength	ASTM C39

An independent laboratory tested the concrete mixes. To maintain the quality of the mix used in the structure, the workability and other characteristics of the mixes were ascertained before placement. A small concrete-control laboratory was set up close to the batch plant. A batch plant inspector was assigned, and testing, as shown below, was performed. Field control was accomplished basically in accordance with the ACI Manual of Concrete Inspection as reported by Committee 611.

Aggregate testing was carried out as follows:

a. Sand Sample for Gradation (ASTM C33 Fine Agg)
b. Organic Test on Sand (ASTM C40)
c. 3/4" Sample for Gradation (ASTM C33 Size No. 67)
d. 1-1/2" Sample for Gradation (ASTM C33 Size No. 4)
e. Check for Proportion of Flat and Elongated Particles

Concrete samples were taken from the mix according to ASTM C172, "Sampling Fresh Concrete." From these samples, cylinders for compression testing were made in accordance with ASTM C31, "Tentative Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field."

Samples were taken at the point of truck discharge. In addition, a minimum of five cylinders were taken at the pipe discharge for each 1000 c.y. for each class of concrete placed in Seismic Category I structures.

Slump, air content, and temperature measurements were taken when cylinders were cast. Slump tests were performed in accordance with ASTM C143, "Standard Method of Test for Slump of Portland Cement Concrete." Air content tests were performed in accordance with ASTM C231, "Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive

strength tests were made in accordance with ASTM C39, "Method of Test for Compressive Strength of Molded Concrete Cylinders." Evaluation of compression tests was in accordance with ACI 214-65.

The inspection and testing of cement, in addition to the tests required by the cement manufacturers, included the following:

Chemical Analysis	ASTM C114
Fineness of Portland Cement	ASTM C115
Autoclave Expansion	ASTM C151
Time of Set	ASTM C191
Compressive Strength	ASTM C109
Tensile Strength	ASTM C190

The purpose of the above tests was to ascertain conformance with ASTM Specification C150. In addition, tests ASTM C191 and ASTM C109 were repeated periodically during construction to check storage environmental effects on cement characteristics. These tests supplemented visual inspection of material storage procedures.

Initial Containment Prestressing After Construction

See Appendix 5E for a discussion on the stressing and restressing of new replacement vertical tendons and original construction tendons, respectively, between 2001 and 2002 on both Units.

Testing and inspection of all prestressing materials and special installation equipment is described in Appendix 5B. Full-time supervision of the prestressing operation was provided. The BBRV post-tensioning system furnished by the Prescon Corporation was used.

Each tendon consists of 90 1/4"-diameter wires conforming with ASTM A421-65T, and 2 anchor heads and 2 sets of shims conforming with ASTM A6-66. The tendon sheathing system consists of spirally-wound carbon steel tubing connecting to a trumplate (bearing plate and trumpet) at each end. The bearing plates were fabricated from steel plate conforming with ASTM A6-66 and the trumpets from AISI C1010-C1020 material.

Tendons were delivered to the site coated with a rust preventive and specially covered. Each tendon came precut to exact length, with one end unfinished and the other end shop button-headed and threaded through the stressing washer.

Tendons were fabricated by the Prescon Corporation at their Mauldin, South Carolina, plant and shipped to the Calvert Cliffs job site. During combing and twisting operation, the hoop and dome tendons were banded every 8' to 10' with steel banding and twisted one complete 360° turn per each 40' of tendon length by an automatic twister. During tendon fabrication, a wire sampling method was used to ensure that all wires met minimum tensile strength specifications. The wires were shipped in coils, each weighing 800 to 1200 lbs. Two samples per coil were used for wire-sampling inspection to compare their actual breaking strength with minimum wire breaking strength. A 3' minimum wire sample was taken from the start end of each coil and placed in an appropriate storage tube in the wire sample cart for delivery to the tensile testing machine. After the sample broke, a reading was taken from the maximum load pointer and recorded on a wire inspection record under the minimum breaking strength column. If the sample failed, a

second supporting inspection was conducted. Wire strength acceptance was based on the final results of two out of three samples if the first sample failed. If the actual breaking strength was greater than the minimum breaking strength, the wire was recorded as being acceptable. This wire sampling operation verified the initial strength of the wires.

One unit of tendon stressing equipment consisted of a 500-ton ram, a jack base (which is bolted on the ram), a pull rod, a nut, 2 hydraulic hoses, and a hydraulic pump with 440 Volt electric power. The jack base was designed to rotate with ease by inserting a rod into one of the eight holes in the base and turning it. This simplified the dome or hoop stressing operation that required many orientations for the jack base. During tendon stressing, the pull rod could rotate a maximum of 90° due to the twist applied to the hoop and dome tendons during fabrication. This pull rod could freely rotate while stressing the tendon without exerting any twisting moments on the anchor.

It is conceivable, however, that the torque remaining after the tendon had seated, would produce insignificant stress in the anchor. The primary bearing and shear stresses (due to twisting), produced during post-tensioning and which were transferred to the concrete, were well below allowable design value. Only the dome and hoop tendons were twisted when fabricated to give helical shape to the wires and equalize their lengths. (See Appendix 5E for a discussion on new vertical tendons installed between 2001 and 2002. These new vertical tendons were twisted when fabricated.) The average maximum eccentricity, (1/2" over the total tendon length) that may exist due to erection inaccuracies, in our judgment, produced no significant increase in compressive stress in the anchor material or concrete.

The tendon installation prestressing procedure was carried out as follows:

a. To assure a clear passage for the tendons, a "sheathing rabbit" was run through the sheathing prior to, during, and following placement of the concrete.

b. Tendons were uncoiled and pulled through the sheathing unfinished end first.

c. The unfinished end of the tendons was pulled out with enough length exposed so that field attachment of the stressing washer and button-heading could be performed. To allow this operation, trumpets on the opposite end have a larger diameter to permit pulling in the shop finished ends with their stressing washers.

d. The stressing washers were attached and the tendon wires button-headed.

e. The shop finished end of the tendon was pulled back and the stressing jack attached.

f. The post-tensioning was done by jacking to the permissible overstressing force to compensate for friction and placing the shims (as required) to lengths corresponding to the calculated elongation. Proper tendon stress was achieved by comparing both jack pressure and tendon elongation against previously calculated values. The vertical tendons were prestressed from either one or both ends, while the horizontal and dome tendons were prestressed from both ends.

g. The grease caps were bolted onto anchorages at both ends and made ready for pumping the tendon sheathing filler material. See Appendix 5E for a discussion on new vertical tendon grease caps installed between 2000 and 2002 on both Units.

h. The tendon sheaths and grease caps were filled with sheathing filler and sealed. The sheathing filler material had limitations specified for deleterious water soluble salts.

During installation of the Unit 1, post-tensioning system two vertical and three horizontal tendons were not installed. These missing tendons are addressed in Section 3.1.4 of the Final Prestressing Report for Unit 1, November 1973. Similarly, one horizontal tendon was abandoned during construction of Unit 2 containment, and is addressed in Section 3.1.4 of the Prestressing Report for Unit 2, June 1977.

Reinforcing Steel

Reinforcing steel in the base slab of the Containment Structure and around penetrations in the cylinder was of the deformed billet steel bars conforming to ASTM Designation A615-68, Grade 60. This steel had a minimum elongation of 7% in an 8" specimen. Deformed billet steel bars conforming to ASTM A615, Grade 40 or Grade 60, were used in the cylinder wall and the domed roof to control shrinkage and tensile cracks. The Grade 40 steel had a minimum yield strength of 40,000 psi and a minimum tensile strength of 70,000 psi; the Grade 60 steel had a minimum yield strength of 60,000 psi and a minimum tensile strength of 90,000 psi.

Mill test reports were obtained from the reinforcing steel and "Cadweld" suppliers for each heat of steel to show proof that the reinforcing steel and mechanical splice sleeves had the specified composition, strength, and ductility.

Welding of reinforcing steel, if required, was performed by qualified welders in accordance with AWS D12.1, "Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction." For the filling of blockouts in the Auxiliary Building, reinforcing steel was welded using an angle splice as shown in Figure 5-16. The design criteria and quality control is described in Section 5B.1.

Reinforcing steel had not been welded at anytime in the Containment Structure. Number 14S and 18S reinforcing steel was spliced by the Cadweld Process. The design criteria and quality control for Cadweld is described in Section 5B.3.

All reinforcing steel was user-tested in accordance with ASTM specifications. Tests include one tension and one bend test per heat for each diameter bar except that no bend tests were performed on #14 and #18 bars. High strength bars were clearly identified prior to shipment to prevent any possibility of mix-up with lower strength reinforcing bars.

Visual inspection of fabricated reinforcement was performed to ascertain dimensional conformance with specifications and drawings. Visual inspection of in-place reinforcement was performed by a placing inspector to assure dimensional and location conformance with drawings and specifications.

Liner Plate

The Containment Structure is lined with a welded steel plate 1/4" thick conforming to ASTM A36 to ensure low leakage. This steel had a minimum yield strength of 36,000 psi and a minimum elongation in an 8" specimen of 20%. Structural steel shapes, bars, and backing strips used in fabrication of the liner also conformed to ASTM A36.

The A-36 material was chosen on the basis that it has sufficient strength as well as ductility to resist the expected stresses from design basis loading and at the same time preserve the required leak tightness of the containment. In addition, A-36 steel is readily weldable by all of the commercially available arc and gas welding processes.

The crane bracket together with the thickened liner plate was a shop fabricated assembly, and all welds have been spot radiographed and magnetic particle inspected. These welds were not considered working welds, since all applied loads were transferred to the concrete and not the liner plate.

The liner plate was designed to function only as a leaktight membrane. It does not serve as a structural member to resist the tension loads from internally applied pressure which may result from any credible accident.

Structural integrity of the containment is maintained by the prestressed, post-tensioned concrete. Since the principal applied stress to the liner plate membrane is in compression and no significant applied tension stresses were expected from internal pressure loading, there was no need to apply special NDTT requirements to the liner plate material. On the other hand, all material for containment parts which must resist applied internal pressure stresses, such as penetrations, was impact tested in accordance with the requirements of ASME, B&PV Code, Section III, Nuclear Vessels, Paragraph N-1211.

A fundamental requirement for fabrication and erection of the liner plate was that all welding procedures and welding operators be qualified by tests as specified in ASME, B&PV Code, Section IX. This code required testing of welded transverse root and face bend samples in order to verify adequate weld metal ductility. Specifically, Section IX of the Code required that transverse root and face bend samples be capable of being bent cold 180° to an inside radius equal to twice the thickness of the test sample. Satisfactory completion of these bend tests was accepted as adequate evidence of required weld metal and plate material compatibility.

Mill test reports were obtained for the liner plate material. The plate was visually checked for thickness, possible laminations, and pitting.

Steel plate was tested at the mill in full conformance to the applicable ASTM Specifications. Certified mill test reports were supplied for review and approval by the design group in the project engineer's office.

There was impact testing done on the liner plate material. The purpose of impact testing is to provide protection against brittle failure. The possibility of a brittle fracture of the liner plate is precluded because at the design accident pressure condition, there will be no significant tensile stress anywhere in the liner plate since the principal applied stress is compression. This is true whether there is instantaneous release of pressure or there is some time lag in temperature load application.

Welding inspection conformed to the quality control inspection procedure described in detail by Appendix 5B.

All of the welding was visually examined by a technician responsible for welding quality control. The basis for visual quality of welds was as follows:

Each weld was uniform in width and size throughout its full length. Each layer of welding was smooth and free of slag, cracks, pinholes, and undercut, and was completely fused to the adjacent weld beads and base metal. In addition, the cover pass was free of coarse ripples, irregular surface, nonuniform bead pattern, high crown, and deep ridges or valleys between beads. Peening of welds was not permitted.

Butt welds were of multipass construction, slightly convex, of uniform height, and had full penetrations.

Fillet welds were of the specified size, with full throat and legs of uniform length.

All welding covered by concrete or otherwise inaccessible after construction was vacuum box soap bubble tested. In this test a leak detector solution was applied to the weld. A vacuum box containing a window was then placed over the area to be tested, and was evacuated to produce at least a 5 psi pressure differential. Leaks were indicated by the appearance of bubbles which were observed through the window in the vacuum box. Welds which were inaccessible for soap bubble testing due to physical limitations or configurations were liquid penetrant inspected.

Radiography was not recognized as an effective method for examining welds to assure leak tightness. Therefore, the only benefit that could be expected from radiography in connection with obtaining leak-tight welds was an aid to quality control. Random radiography of each welder's work provided verification that the welding was or was not under control and being done in accordance with the previously established and qualified procedures. In addition, employing random radiography to inspect each welder's work had been demonstrated by past experience to have a positive psychological effect on improving overall welding workmanship.

Radiographic techniques were in accordance with ASME, B&PV Code, Section VIII, Paragraph UW-51. At least one 12" spot radiograph was taken in the first 10' of welding completed in the flat, vertical, horizontal, and overhead positions by each welder. Thereafter, approximately 10% of the welding was spot examined on a random basis using 12" film.

Dye penetrant and magnetic particle inspections were also used as an aid to quality control. The field welding inspectors used dye penetrant or magnetic particle inspection to closely examine welds judged to be of questionable quality of the basis of the initial visual inspection. Also, dye penetrant inspection was used to confirm the complete removal of all defects from areas which had been prepared for repair welding. Dye penetrant or magnetic particle inspection of liner plate welds were in accordance with ASME, B&PV Code, Section VIII.

The welds for each section of base slab liner plate were vacuum box soap bubble tested immediately upon installation. After successfully passing this leakage test, they were covered with test channels and the particular welds associated with that section of liner plate were pressure tested. Any repairs were carried out utilizing the same high standards and control exercised in the initial construction.

A testing pipe was provided for each continuous segment of the bottom liner plate leak chase channels (equivalent to containment weld channels). The tops of the pipes were above the cover slab and were sealed with caps. These pipes were initially used to test the leak tightness of the bottom liner and can also be used at a later date, if so required.

5.5.1.2 Structural Test at Completion of Initial Construction

The purpose of instrumenting and testing prestressed concrete Containment Structure is to provide a means for comparing the actual response of the structure to the loads induced both during post-tensioning and pressure testing with the predictions of the design calculations. If the response is as predicted, the design techniques are assumed to have been verified.

The Containment Structure was pressurized to 115% of design pressure for one hour following completion of construction to establish the structural integrity of the building. The structural integrity test was conducted in accordance with a written procedure. Personnel access limitations included in the written procedures designated areas of limited access during specific periods of the test.

The test objectives were:

a. To provide direct verification that the structural integrity as a whole is equal to or greater than that necessary to sustain the forces imposed by (a) the structural test at 115% of the design pressure and (b) the post tensioning sequence.
b. The in-place tendons (the major strength elements) have a strength of at least 80% of guaranteed ultimate tensile strength and that the concrete has the strength needed to sustain a strain range from high initial average concrete compression when unpressurized to low average concrete compression when pressurized.

A quality assurance program was instituted as described in Appendix 5B. In addition, each individual tendon was tensioned in place to 80% of the guaranteed ultimate tensile strength and then anchored at a lower load that is still in excess of those predicted to exist at test pressure levels. During pressurization of the structure, the structure's response was observed at selected pressure levels with the highest being 115% the design pressure. An indication that the structure is capable of withstanding internal pressure resulted from these tests. The strain measuring program is described earlier. Individual test values which fall outside the predicted ranges will not be considered as necessarily indicative of a lack of adequate structural integrity.

The Calvert Cliffs Units were very similar to the Turkey Point, Oconee, Point Beach, and Palisades structures, differing only in being somewhat larger in diameter. The design and construction are the same. The structures for both Turkey Point and Palisades are completely instrumented. The Turkey Point instruments provide approximately 400 strain measurements at 55 locations throughout the structure and liner. In addition, about 25 optical measurements of structural deformation are made. The Palisades instrumentation is comparable. This amount of data will permit a detailed comparison between design calculations and observed response. The basic structural design and the accuracy of the calculation procedures used by Bechtel was, therefore, verified by these tests. This verification was applicable to the Calvert Cliffs design calculations.

Since the detailed confirmation of the design techniques is available, instrumentation of the Calvert Cliffs structure is not required and no additional confirmation of design techniques is necessary. For these reasons, no provisions for strain gauge instrumentation of the structural members of the Calvert Cliffs Containment Structure are made.

Prior to reactor fuel loading and operation, the integrity of the Containment Structure was demonstrated by a pressure proof test. The post-tensioning and pressure tests permitted verification that the structural response due to the induced loads is consistent with the predicted behavior. This was accomplished by measuring deflection of Containment Structure using taut wires.

The measurement technique required stretching taut wires across the Containment Structure at appropriate elevations and azimuths and around the equipment hatch openings. These displacements were correlated with measurements made on Turkey Point, Oconee, and Point Beach I Containment Structures for verification of structural behavior.

In analyzing the structures to obtain the calculated displacement, the most probable values of material constants were used rather than the highly conservative design values. For example, values of the elastic modulus for concrete were predicted to provide an estimate of its most probable value at the time of the test.

The use of only two meridians for taking measurements during pressure testing is justified as follows:

a. It represents the true cross-section of the cylindrical shell where uniform wall thickness and buttress (thickened wall) sections exist. Other discontinuity areas, such as the equipment hatch, are individually checked for strain measurements.
b. Analytical methods are based on an assumption that the structure is axisymmetric and the material properties assumed for calculation purposes are idealized for derivation of the theories of elasticity. The basic method of analysis is Bechtel's Finite Element Program, CE 316-4 as explained in Section 5.1.3.1. This analysis furnished the predicted strain for this test, assuming the actual structure was perfectly cylindrical with no discontinuities such as buttresses or penetrations and that there were no deviations from axisymmetry of applied forces.
c. The correctness of the predicted strains versus measured strain will not significantly differ by increasing the number of measurements at more than two meridians because the basic assumptions as mentioned in b would be identical.
d. Tests of Containment Structures with similar configuration have demonstrated that the predicted and measured strain values are in good agreement. The applied test procedure and selected points for strain measurements were identical for all tests.

Nevertheless, the Calvert Cliffs Nuclear Power Plant test procedure included additional points to the extent possible to obtain measurements as described in AEC Safety Guide 18, Structural Acceptance Test for Concrete Primary Reactor Containments.

From the previous experience and analytical assumptions, it was expected that agreement would have been between test results and analytical predictions in the following range:

Cylinder at equator	15%
Dome	15%
Bottom slab	25%
Bottom slab - Wall junction	25%

Dome - Wall junction	20%
Around opening	30%
Localized stress concentration	100%

If the measured strains had fallen noticeably beyond the above-mentioned ranges of error, a review and investigation would have been made to determine the cause of such discrepancies.

5.5.1.3 Initial Leakage Test

At the time of the initial leakage test, the design leak-rate was 0.20% by weight of the contained atmosphere in 24 hrs at 50 psig. It has been demonstrated that, with good quality during erection, this is a reasonable requirement. The purpose of these tests is to ensure that leakage through the Containment Structure and associated systems is held below the design leakage rate (Reference 1).

Initial leak-rate tests of the Containment Structure and its penetrations were conducted at pressures of 50 and 100% of the calculated peak pressure, maintaining each pressure for a sufficient length of time to establish the leak-rate. Values of Containment Structure ambient dry bulb temperature and relative humidity were recorded during the test period for correction of data as required.

The preservice leak-rate test equipment consisted of bottled air or nitrogen, pressure regulator and pressure, temperature and flow indicator. Each part's measuring range and accuracy were as follows:

a. Pressure Regulator

   Range: 2000 psig to 50 psig

b. Pressure Indicator

   Pressure gauge: Readout unit, calibration accuracy of 0.015% of reading, readout 100,000 counts = full scale

   Range: 0 psia to 100 psia

   Minimum graduation: 0.1 psia

   Accuracy: 0.1% of full scale

   Repeatability: 0.03% of full scale

   Sensitivity: 0.01% of full scale

c. Temperature Indicator

   Range: 0°F to 125°F

   Accuracy: 0.5°F

   Readability: 0.5°F

d. Flow Indicator (Rotameter)

   Range (dual scale): 87-875 cc/min
   23-230 cc/min

   (air at 70°F and 50 psig)

   Accuracy: ± 2% of maximum flow

The test established the capability of the Containment Structure to contain the pressure for which it was designed at a leak-rate not exceeding that specified in the license application. These data were plotted to establish initial relationships

between internal pressure, leak-rate, external pressure, temperature, relative humidity.

5.5.2 POST-OPERATIONAL SURVEILLANCE

5.5.2.1 Leakage Monitoring

The reactor containment and other equipment subjected to containment test conditions are designed to allow periodic leakage rate testing at containment design pressure in compliance with AEC General Design Criteria 52, published in the Federal Register on February 20, 1971. Frequency of the periodic leakage rate test is explained in the Containment Leakage Rate Testing Program.

Periodic leakage-rate tests of the Containment Structure will be conducted to verify its continued leak-tight integrity. The post-operational leakage-rate tests are conducted at an internal pressure between 96% of the peak containment accident pressure and 100% of the containment design pressure. The acceptable leakage rate for the test pressure used is given in the Technical Specifications and in the Containment Leakage Rate Testing Program.

The temporary hatch cover plate on the emergency personnel lock shall be seal-tested prior to use during movement of irradiated fuel within the containment.

Periodically, in accordance with the Containment Leakage Rate Testing Program, a visual inspection of the exposed accessible interior and exterior surfaces of the containment, including the liner plate will be conducted to assure that no corrosion or other visually apparent deterioration has occurred.

The basic steps in conducting leakage-rate tests include the following:

a. Measurements of absolute pressure, temperature and moisture content within the Containment Structure.
b. Verification of the integrated leakage-rate measurement system by the use of precise measurements of a flow causing a change in the weight of air in the containment that is approximately equal to the measured or permissible 24-hr leak.
c. Maintaining pressure between 47.4 and 50 psig for the length of time required by the integrated leakage rate test procedures.
d. Controlling containment temperature between 50°F and 120°F.
e. Obtaining measurement accuracy tolerances within 95% confidence limits, such that the calculated leakage-rate plus the accuracy tolerance is less than the permissible leakage-rate at the appropriate test conditions.

Formulas used in computing the integrated leakage-rate are based on the formulas found in American National Standards Institute/American Nuclear Society 56.8 - 1994, “Containment System Leakage Testing Requirements.” The Type A (primary containment overall leakage), Type B (local leakage at penetrations), and Type C (isolation valves) tests for both pre-service and inservice are discussed in Sections 5.1.8, 5.2.3, and 5.5, the Technical Specifications and the Containment Leakage Rate Testing Program, which include acceptance criteria, corrective action to meet the acceptance criteria, test frequency and duration and requirements for reporting test results.

It was expected that the inservice leakage-rate test equipment will be similar to that for the preservice tests.

5.5.2.2 <u>Surveillance of Structural Integrity</u>

See Appendix 5E for additional surveillances associated with the long-term corrective action plan for addressing vertical tendon corrosion discovered in 1997.

The primary objective of the program for Inservice Inspection of the Containment Structure concrete, tendons, and liner during the lifetime of the plant is to ensure the strength and reliability of the post-tensioning steel and other major components such as stressing washers, shims, and bearing plates. The condition of the containments is monitored by a combination of physical testing and visual examinations performed on a regular schedule as called for by ASME Section XI, Subsections IWE/IWL (ASME XI) and 10 CFR 50.55a as it pertains to the Containment Structures.

During construction, 3 tendons of each type, hoop, dome, and vertical, were constructed with 93 wires in lieu of the standard 90 to provide designated surveillance tendons. However, the current ASME XI program requires that a random selection be made with the number examined specified as a percentage of the total population, with a minimum and maximum, of each type of tendon. This percentage varies with plant age and previous surveillance results. Thus, the original surveillance tendons are now a part of the general population. The Code also requires the designation of a common tendon of each type that is examined at each surveillance.

Under the Regulatory Guide 1.35 testing program that followed the initial tendon surveillance program, the Unit 2 Containment did not undergo tendon lift-off testing. The current ASME XI program now requires that both units undergo comparable inspections with an allowance to shift some examinations between units based on the similarity and timing of their construction. Thus, Unit 2 is now subject to the same tendon surveillance requirements as Unit 1. The selection criteria and examination frequencies as specified in ASME XI.

Because the tendons were initially strength-tested, the inservice inspection program is conducted to monitor the tendons for corrosion and to verify that the force applied by the tendons meets design assumptions. To achieve those goals, the random selection of tendons is visually examined and the corrosion protection grease is sampled. At alternating surveillances, the tendons are force checked via lift-off testing and one tendon of each type is detensioned and a wire is removed for tensile testing. Those tendons are restressed appropriately immediately after the wire is removed.

The lift-off values are compared to predicted values to determine whether the force required at the end-of-plant life will be met.

Any components or values not meeting the acceptance criteria of ASME XI requires scope expansion and/or engineering evaluation.

Visual examinations are conducted over the entire exterior of the containments in accordance with the ASME XI and 10 CFR 50.55a. These examinations are timed and designed to detect any degradation mechanism before it can affect the structural integrity.

Accessible portions of the interior steel liner are visually examined once per inservice inspection period. This examination is to detect any abnormality that could affect the leak tightness of the liner. When necessary, the visual examinations are supplemented with other methods.

Since the Unit 2 containment is a duplicate of Unit 1 design, tendon surveillance has been limited to visual inspection without dismantling load bearing components or the anchorage. End anchorages, adjacent concrete surfaces, and the liner plate are inspected.

5.5.3 REFERENCES

1. Bechtel Corporation, Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants, BN-TOP-1, Rev. 1, November 1, 1972

5.6 OTHER STRUCTURES

5.6.1 AUXILIARY BUILDING

5.6.1.1 General Description

General plans at various elevation and sections through the Auxiliary Building are shown in Figures 1-5 through 1-16 in Chapter 1.

The Auxiliary Building is primarily a reinforced concrete structure and the mat foundation supports a structural steel and reinforced concrete frame which consists mainly of reinforced concrete walls and floors. On the top structure and over the fuel handling area is a secondary steel frame structure with missile resistant concrete walls and roof, which houses the Spent Fuel Cask Handling Crane.

Facilities related to the Nuclear Steam Supply System which are located in the Auxiliary Building include:

a. New and spent fuel handling, storage and shipment (spent fuel pool, bottom at el. 30'0" and new fuel racks at el. 69'0")
b. Control Room (Elevation 45'0")
c. Waste Processing System
d. Chemical Addition and Sampling System
e. Component Cooling System
f. Part of the Containment Structure Spray System
g. High and Low Pressure Injection Systems
h. Spent Fuel Cooling System
i. Electrical Distribution System
j. Chemical and Volume Control System

During the week of May 8, 1969, part of the concrete for the Auxiliary Building walls was pumped through aluminum pipe (Figure 5-17). During the week of August 15, 1969, 20% of concrete pour No. C-5.5a, i.e., for the base slab, Containment Structure, Unit 1, was also pumped through aluminum pipe. The area to which the concrete was pumped is shown in Figures 5-18 and 5-19.

Upon discovery, extensive tests were performed on the concrete, initially with the "Swiss Hammer," which indicated the average strength of the concrete to be above 5000 psi, for walls in the Auxiliary Building. The design strength of the concrete for the walls of interest in the Auxiliary Building and for the base slab in the Containment Structure is 4000 psi. Five cores were taken from the above walls for further testing. These tests indicated the minimum strength of concrete to be 4727 psi and maximum strength 5583 psi.

Since the actual calculated stresses in the concrete are well below allowable stresses, and since the tests on concrete pumped through aluminum pipe in the Auxiliary Building indicate that the strength of the concrete is well above the design strength of 4000 psi, it is evident that the concrete pumped through aluminum pipe, in the base slab of the Containment Structure Unit 1, and in the Auxiliary Building wall, is structurally adequate.

5.6.1.2 Design

The areas of the building housing the above facilities have been designed for the loads and conditions as shown in Table 5-6, the reinforced concrete design being in accordance with ACI 318-63 and the structural steel with AISC.

The spent fuel storage racks, located in the spent fuel pools, are designed for seismic loadings by considering the appropriate spectral acceleration for a 2% damping. The racks support the fuel element assemblies at the bottom, maintaining a center to center distance of 10.09". A 3/16", solid stainless steel liner plate was used on the inside face of both pools for leak tightness, and all of the field welds have leak-test channels welded to the outer side of the liner plates. The channels are grouped into 10 zones, each with its own detector pipe to localize leaks in the liner seams. Access from one pool to the other is provided by an opening in the central dividing wall, which may be closed by the use of stainless steel bulkhead gates.

5.6.1.3 Tornado Protection In Spent Fuel Area

In addition to all other loads including OBE and SSE, the steel-framed structure over the spent fuel pool is designed to resist tornadoes and horizontal missiles without partial or complete collapse, except for the west wall. A study indicates that the possibility of horizontal tornado missiles impacting the spent fuel pool from the west side is remote. Two foot thick concrete for missile protection is provided in the roof and the north, east, and south walls.

Since the steel-framed structure is designed to resist tornadoes with all other load combinations as listed in Table 5-6 and the cask handling crane is supported by the steel-framed structure, the cask handling crane will not be damaged during the tornado loading.

5.6.1.4 Fuel Pool Floor Design Criteria

The following design criteria were used in the analysis of the fuel pool floor to withstand the effect of a fuel cask drop.

a.	Weight of the Cask in air	50 kips
b.	Length of the Cask	16' 1-1/2"
c.	Diameter of the Cask	2'0"
d.	Distance Traveled (During Accidental Drop)	
	1. Free Fall in air	3'6"
	2. In water	39'0"
e.	Water density	62.4 lbs/ft^3
f.	Strength of Concrete ($f_c{}'$) (based on cylinder tests)	4150 psi
g.	Modulus of Elasticity (Ec) ($3.1x10^6$ psi)	$4.46x10^5$ k.s.f.
h.	Maximum Acceleration	290 g's

Determining the striking velocity at collision involves evaluating the velocity change due to missile weight, geometry of the missile, the buoyant force, and the drag force. In the above determination, the drag coefficient (Cd) was assumed to be unity.

In the evaluation of a structure due to missile impact, both the penetration criteria and structural response characteristics must be reviewed. The penetration of the cask missile into the concrete pool floor was evaluated by the use of the Modified Petry Formula. The Ballistic Research formula was used to check against perforation. The dynamic response was evaluated by calculating the ductility ratio of the elastic-plastic system response to a triangular pulse impact force-time history.

5.6.1.5 Conclusion of The Cask Drop Analysis

Analysis of the floor of the fuel pool indicates that the spent fuel pool bottom is capable of safely withstanding the impact of the accidental drop of the cask. Cracks will be developed by diagonal tension near the support but they will be of microscopic type considering the shear stresses. The structural integrity of fuel pool bottom will not be impaired.

In 1980, a reanalysis of this accidental drop using more conservative parameters also indicated that the truck cask drop did not impair the structural integrity of the fuel pool bottom.

In 1992, the spent fuel cask handling crane was upgraded to single-failure proof to allow use of the NUHOMS transfer cask for transfer of spent fuel from the spent fuel pool to the Independent Spent Fuel Storage Installation dry storage facility. The single-failure-proof crane, designed to meet requirements of NUREG 0554, provides the primary plant protection against a cask drop accident.

In addition to the crane upgrade, an energy absorbing cask platform was installed in the cask pit area of Unit 1 for further protection against crane drive train failure. The structural integrity of the fuel pool bottom will not be impaired in the unlikely event of a crane drive train failure in the cask pit area.

5.6.1.6 Thermal Stresses in The Spent Fuel Pool Walls

The maximum thermal stresses developed in the spent fuel pool walls under the most adverse conditions will be in the range of 950 psi, compressive, and in the range of 7500 psi, tension, in the reinforcing steel. Reinforcing steel is provided in the floor and both faces of the fuel pool walls to accommodate these thermal stresses.

5.6.2 INTAKE STRUCTURE

5.6.2.1 General Description

The intake structure is situated to the east of the main plant and is primarily a reinforced concrete structure, founded on a slab varying in Elevation from -26'0" to -14'3". It houses 12 circulating water pumps supplying water from the Chesapeake Bay to the condensers, located under the turbine generators, and to 6 saltwater pumps. To protect the condensers from foreign bodies present in the Bay water, trash racks and traveling water screens are provided. Vertical guides are provided down the sides of each intake channel to receive stop-logs. Running the full length of the structure is a gantry crane having a lifting capacity of 35 tons. Design features and administrative controls are in place to minimize the probability of a load drop from the gantry crane. A description of our means of controlling heavy loads is presented in Section 5.7.

Water from the traveling water screens flows through a separate fish collection and holding structure/facility constructed to allow for environmental aquatic studies.

5.6.2.2 Design

The Intake Structure has been designed for the loads and conditions as shown in Table 5-7. The reinforced concrete structure is designed in accordance with ACI 318-63 and the structural steel with AISC. All of the structural concrete is a 4000 psi mix at 28 days, and the reinforcement is in accordance with ASTM A615, Grade 60. The total length of the structure is divided into three sections above the base slab by two expansion joints. The high level roof at Elevation 28'6" is comprised of a reinforced concrete slab supported on a structural steel frame. Within this roof are access covers to each of the saltwater and circulating water pumps. The concrete structure design conforms to Seismic Category I requirements.

To prevent wave runup overtopping of the pump room wall, an adverse slope was constructed at the top 5' of the structure (Section 2.8.3.6).

[redacted]

[redacted]

5.6.3 TURBINE BUILDING

5.6.3.1 General Description

The building, comprising the turbine generator bays and the heater bays, is an integrated steel structure, with metal siding, supported on reinforced concrete foundations. The circulating water intake and discharge conduits are incorporated into the spread footings. Turbine generator Units No. 1 and 2 are separated by expansion joints from their respective superstructures. Also, Unit Nos. 1 and 2 Turbine Building Superstructures are separated by an expansion joint.

5.6.3.2 Design

The building is a Seismic Category II structure designed as described in Appendix 5A, except that the auxiliary feedwater pump enclosure is Seismic Category I. All of the structural steel columns, beams, and roof trusses of the building have been designed as independent members and in accordance with the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1963 Edition.

Two bridge cranes, a 200-ton unit and a 40-ton radio-controlled unit, are located in the turbine generator section of the building.

The two turbine generators are mounted on their own concrete pedestals which project up through the building to the operating deck at Elevation 45.0'.

5.6.4 SERVICE BUILDING

5.6.4.1 General Description

The building is situated between the Turbine Building and the intake structure and accommodates a warehouse, lube oil room and water treatment area at Elevation +12.0’ and office space and a machine shop at Elevation +45.0’. It is primarily a structural steel frame supporting a reinforced concrete floor slab. At Elevation 45',

part of the warehouse is covered by a roadway and parking area. The structural steel columns are supported by reinforced concrete piers.

5.6.4.2 Design

The structural steel members are designed as a continuous frame across the width of the building in an east-west direction, all being fabricated from ASTM A36 steel. The floors are made with 3000 psi concrete and ASTM Grade 40 reinforcement.

The entire building falls into a Seismic Category II category for design as described in Appendix 5A.

5.6.5 SAFETY-RELATED DIESEL GENERATOR BUILDING

5.6.5.1 General Description

The safety-related Diesel Generator Building is a three-story, reinforced concrete structure approximately 75' by 97' in plan, excluding the east end entry ports. The corners of the building are beveled 15' to increase the available area on the east and west wall elevations. This increased area will allow wall surfaces to function as baffle areas for building ventilation and radiator cooling and as structural shear walls. Exhaust ducts were installed over the east baffle areas in order to direct radiator exhaust vertically. This prevents design winds (100 mph) from interfering with radiator fan performance. The building is about 60' in height and is supported on mat foundations at grade with a partial basement in the area of the diesel generator pedestal. In addition, a one-story structure is provided on the east side of the building as missile protection for the main building entry and diesel generator area exhaust louver.

Additional safety-related Diesel Generator Building description is provided in Reference 1.

5.6.5.2 Design

The safety-related Diesel Generator Building structure is classified as Seismic Category I.

The safety-related Diesel Generator Building structure is designed for wind loadings resulting from 100 mph winds at 33' above the ground. The vertical velocity profiles and gust factors used in the design of the Diesel Generator Building are in accordance with the Standard Building Code, Subsection 1205.2 and American Society of Civil Engineers 7-88.

The safety-related Diesel Generator Building structure is designed for tornado-induced forces in accordance with Regulatory Guide 1.76, Revision 1, Standard Review Plan (SRP) Section 3.8.4 and "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," BC-TOP-3-A (Reference 2).

The tornado loadings used in the design are simultaneously applied as follows:

a. The velocity components on the safety-related Diesel Generator Building are applied as a funnel of wind traveling at 70 mph with a maximum tangential velocity of 290 mph. Therefore, the maximum effective wind velocity on any structure is 360 mph.
b. The tornado-induced pressure differential is applied as a 3 psi positive pressure within the safety-related Diesel Generator Building occurring in 1.5 seconds (2 psi per second) followed by a calm for two seconds and

then a repressurization to atmospheric pressure at a rate of 2 psi per second.

c. The safety-related Diesel Generator Building is designed for missile impingement as shown in Table 5-8. A vertical velocity of 70% of the postulated horizontal velocity shall be considered acceptable for all missiles except for steel rods. A steel rod missile's vertical velocity is assumed to be equal to the horizontal velocity.

The safety-related Diesel Generator Building roof load was conservatively estimated using the Probable Maximum Snowfall saturated with residual rainfall. Therefore, based on the absence of parapets on the east and west sides of the Diesel Generator Building roof and a roof slope of 2%, the design roof loading due to the 100-year snow pack and the Probable Maximum Snowfall with residual rainfall was estimated as 70 psf.

The concrete structures and components are designed in accordance with the strength design methods of ACI-318 and ACI-349, using the load combinations specified in Reference 1. The structural steel design is based upon the AISC "Specification for Structural Steel Buildings, Allowable Stress Design and Plastic Design." Welding of structural steel was performed in accordance with the Structural Welding Code AWS D.1.1.

Regulatory Guide 1.60, Revision 1, design response spectra, anchored at the high frequency end at 0.15 g and 0.10 g for the SSE horizontal and vertical design motions, respectively, and at 0.080 g and 0.053 g for the OBE horizontal and vertical design motions, respectively, have been adopted for the design of the safety-related Diesel Generator Building. The peak horizontal ground acceleration for the OBE and SSE are 0.08 g and 0.15 g, respectively. The maximum vertical acceleration is defined as two-thirds of the maximum ground acceleration. The three synthetic time histories, developed for each of the three components of the earthquake, comply with the SRP acceptance criteria for total duration, strong motion duration, power spectral density enveloping, and response spectra enveloping.

The damping values used in the seismic analysis of the Diesel Generator Building and associated structures, systems, and components are provided in Table 5-9. Damping values given in Table 5-9 are in compliance with Regulatory Guide 1.61 with the following exceptions:

a. When cable trays are loaded to 100% of their maximum capacity, their damping value is 15%.

b. A damping value of 7% of critical is assumed for conduit at 100% maximum capacity.

These values were justified based on the testing of cable tray and conduit systems. Testing of these systems (Reference 3), demonstrated that a substantial amount of energy was dissipated by friction between cables and by friction between cables and the cable tray.

c. In lieu of damping values for piping identified in Table 5-9, ASME, B&PV Code, Section III, Division I, Code Class 3 piping systems, ASME Code Case N-411 damping values may be used. This method, which is preferred in most cases, permits the use of Code Case N-411 variable damping with a three dimensional spectra input (two horizontal and one vertical earthquake responses combined by the SRSS method). If this method is adopted, the spectra curves are enveloped curves with 15%

peak broadening. In addition, the modal combinations specified in Regulatory Guide 1.92, Revision 1, will also be used. Furthermore, the zero period acceleration effects shall be combined with inertia effects by the SRSS or CMSRS method. Use of the multiple zone response spectra is prohibited when Code Case N-411 damping is used. The use of Code Case N-411 damping was restricted as specified in Regulatory Guide 1.84 (Reference 4).

The loads considered during the design of the safety-related Diesel Generator Building are normal loads, severe environmental loads, extreme environmental loads and abnormal loads. Cumulative loads from the heating, ventilation and air conditioning system, conduit, cable trays, and pipe supports are considered in the design of major structural components.

Normal Loads

Normal loads are those loads encountered during normal plant operation and shutdown. These loads include:

Dead Loads or their related internal moments and forces, including any permanent equipment loads. Vertical and lateral fluid loads imposed upon the fuel oil tank enclosure, due to a tank rupture, shall be treated as a dead load.

Required section strength to resist design loads based on the strength design methods of ACI 318 and ACI 349.

Required section strength based on elastic design methods and allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

Live loads or their related internal moments and forces, including any movable equipment loads or other loads which vary with intensity and occurrence, such as soil pressure. Seismic or tornado load cases containing L substitute the operating live load term L_o, which is defined as the live load expected during normal plant operations. The term L_o is taken as 0.25 L or the actual weight of the equipment spread on the floor, whichever is larger. In laydown areas, the actual weight of the equipment spread out on the floor shall be considered as L_o. Snow loads were also considered as live loads in the design of the Diesel Generator Building. No reduction in live load (L_o) is taken for snow loads.

Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

Severe Environmental Loads

Severe environmental loads are those loads infrequently encountered during plant life. Included in this category are:

Loads generated by the OBE. These include associated hydrodynamic and dynamic incremental soil pressures.

Loads generated by the design wind specified for the structure.

Extreme Environmental Loads

Extreme environmental loads are those loads that are credible but highly improbable. These loads include:

Loads generated by diesel engine/generator set missiles.

Loads generated by the safe shutdown earthquake.

Loads generated by the design tornado. Tornado loads include loads due to the tornado wind pressure (W_w), the tornado-created differential pressure (W_p), and tornado-generated missiles (W_m).

Abnormal Loads

Abnormal loads are those loads generated by a postulated high- or moderate-energy pipe break accident within the building. High- and moderate-energy pipe breaks are defined by Branch Technical Position SPLB 3-1, SRP 3.6.1 and SRP 3.6.2. The Diesel Generator Building structure provides physical separation of high- and moderate-energy systems and components of the redundant diesel generator train.

5.6.6 STATION BLACKOUT DIESEL GENERATOR BUILDING

5.6.6.1 General Description

The Station Blackout (SBO) Diesel Generator Building is a two-story structure with a basement and a penthouse. The building measures approximately 70’ by 75’ and is about 40’ high. Reinforced concrete is used for the foundation mat, diesel generator pedestal, and basement walls. Structural steel is used for the columns, floor beams, and other supporting members. A composite slab is used for the floor framing. Metal decking is used for the roof, and insulated metal siding is used for exterior walls.

5.6.6.2 Design

A requirement by NUMARC 87-00 is that alternate AC power systems and components and required subsystems must be located within a structure meeting, as a minimum, the Uniform Building Code (UBC) requirements. The UBC equivalent design code of record for the plant area is the Standard Building Code. As required by the Standard Building Code, likely weather-related external events are considered in the design of the SBO Diesel Building. The SBO building is designed with roof metal decking and metal wall siding that will deform or blow out to reduce the potential for high-pressure buildup due to a sudden pressure drop as a result of a tornado. Miscellaneous equipment related to the SBO Diesel Generator mounted outdoors, or on the roof of the SBO Diesel Generator Building, which could exceed the parameters for a Spectrum II tornado missile (as defined by SRP 3.5.1.4, Revision 2) is anchored to resist tornado wind loads. By these means, the potential for large missiles is greatly mitigated, and no tornado missile originating from the SBO Diesel Building will threaten the integrity of the adjacent safety-related Diesel Generator Building.

The acceptance criteria of SRP Section 3.7.2 require interfaces between Category I and non-Category I structures to be designed for dynamic loads and displacement produced by both the Category I and non-Category I structures. In order to prevent a structural failure and impact on the adjacent Safety-related Diesel Generator Building, the main girders, columns, and bracing for the SBO Diesel Generator Building have been analyzed to demonstrate that the building will not collapse under design basis earthquake loads. The seismic displacements of

both the Category I Diesel Generator Building and the SBO Diesel Generator Building were calculated, and a seismic gap was sized to prevent structure-to-structure interaction between the buildings. Additional information regarding the design of the SBO Diesel Generator Building is provided in Reference 5.

5.6.7 REFERENCES

1. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated December 18, 1992, Emergency Diesel Generator Project-Civil Engineering Design Report
2. Topical Report, BC-TOP-3A, Revision 3, Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Bechtel Power Corporation, San Francisco, California, August 1974
3. Anco Engineers, Inc., Cable Tray and Cable Raceway Seismic Test Program, Release 4 (Report 1053-21.1-4), dated December 15, 1978
4. Regulatory Guide 1.84, Design and Fabrication Code Case Acceptability – ASME Section III, Division 1, Revision 30, October 1994
5. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated March 7, 1994, AAC Power Source Design Report

TABLE 5-6

AUXILIARY BUILDING LOADS AND CONDITIONS

AREA	CLASS
Control Room	A.B.C.D.E.
Cable Spreading Room	A.B.C.D.E.
Electrical Equipment Rooms	A.B.C.D.E.
Spent Fuel Pool	A.D.
Spent Fuel Storage Racks	D.F.
Spent Fuel Cask Handling Crane	A.B.D.
Piping Penetration Room Frames	A.D.
Hot Machine Shop	A.D.

Key

A. All normal dead, equipment, live, and wind loads due to 90 mph wind.

B. Normal dead, equipment, live, and tornado wind load due to 300 mph wind.

C. Horizontal tornado missiles of (1) a 4000 lb automobile traveling at 50 mph up to a height of 25' in the air and (2) a wooden plank 4"x12"x12' long traveling end on at a velocity of 300 mph at any height.

D. Normal dead and equipment loads, plus SSE loads.

E. Turbine generator missile.

F. Normal dead load, live load, and equipment loads.

TABLE 5-7

INTAKE STRUCTURE LOADS AND CONDITIONS

CLASSIFICATION

	Seismic Category
Circulating Water Pumps	II
Saltwater Pumps	I
Concrete Structure	I

LOADING CONDITIONS

Area Description	Loading Designation
Floors - Slab @ E1 +10'0"	D.E.F.
Slab @ E1 +28'6"	D.E.
Well Walls	D.F.
Pump Room Walls	A.B.C.D.F.

Key

A. All normal dead, equipment, live, and wind loads due to a 90 mph wind.

B. Normal dead, equipment, live, and tornado wind load due to 300 mph wind.

C. Horizontal tornado missiles of (1) 4000 lb automobile traveling at 50 mph up to a height of 25' in the air, and (2) wooden plank 4"x12"x12' long traveling end on with a velocity of 300 mph at any height.

D. Normal dead and equipment loads plus SSE loads.

E. Normal dead load, live load and equipment loads.

F. Live loads from wave run-up.

TABLE 5-8

DESIGN BASIS TORNADO MISSILES

MISSILE	MASS (Kg)	DIMENSIONS (m)	VELOCITY (m/s)
Wood Plank	52	0.092 x 0.289 x 3.66	83
6” Schedule 40 pipe	130	0.168D x 4.58	52
1” Steel Rod	4	0.0254D x 0.915	51
Utility Pole	510	0.343D x 10.68	55
12” Schedule 40 pipe	340	0.32D x 4.58	47
Automobile	1,810	5 x 2 x 1.3	59

TABLE 5-9

DAMPING VALUES FOR THE 1A DIESEL GENERATOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS

(Percent of Critical Damping)[a]

STRUCTURE OR COMPONENT	OBE	SSE
Equipment and large-diameter piping systems[b][c] (pipe diameter > 12 inches)	2	3
Small-diameter piping systems[c] (pipe diameter ≤ 12 inches)	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Reinforced concrete structures	4	7
Electrical cable trays, conduits and supports	15 for Cable Tray 7 for Conduit	15 for Cable Tray 7 for Conduit
Sloshing fluid[d]	0.5	0.5

(a) Damping values for foundation material and for foundation structure interaction analysis are not included in this table.

(b) This includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, the values for small diameter piping are used.

(c) In lieu of these values, for ASME B&PV Code, Section III, Division I, Code Class 3 piping systems, the damping values provided in the Civil Design Report for the Emergency Diesel Generator Project may be used per Code Case N-411.

(d) These damping values are applicable to the horizontal sloshing of oil in the fuel oil storage tank.

5.7 CONTROL OF HEAVY LOADS

The NRC issued NUREG-0612, "Control of Heavy Loads at Nuclear Power Facilities" in July 1980 to document the results of Unresolved Safety Issue (USI) A-36. NUREG-0612 provided a set of guidelines intended to minimize the possibility of load drops on safe shutdown or decay heat removal systems. In response to Generic Letters dated December 22, 1980 (Un-numbered), and February 3, 1981 (Generic Letter 81-07), Baltimore Gas and Electric Company submitted a two-phase report reviewing provisions for handling and control of heavy loads at Calvert Cliffs, and evaluating these provisions with respect to the guidelines of NUREG-0612. The NRC accepted our Phase I evaluation (Reference 1). In a safety evaluation report on the implementation of Generic Letter 85-11, the NRC declined to review Phase II responses and released all the respondents from any commitments made in them. However, the NRC stated that while not a requirement, they encourage implementation of actions identified in Phase II. As a result, Calvert Cliffs considered the implementation of these actions to be voluntary.

Overhead load handling systems which handle heavy loads (at Calvert Cliffs - loads in excess of 1600 pounds) in the vicinity of the reactor vessel or near spent fuel in the spent fuel pool are subject to the general guidelines of NUREG-0612. Overhead load handling systems that handle any weight in areas where a load drop may damage safe shutdown or decay heat removal systems are subject to the guidelines of NUREG-0612. These guidelines include:

a. Definition of safe load paths
b. Development of load handling procedures
c. Qualifications, training, and specified conduct of operators
d. Special lifting devices should satisfy the guidelines of American National Standards Institute (ANSI) N14.6-1978
e. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9
f. Periodic inspection and testing of cranes
g. Design of cranes to ANSI B30.2 or CMAA-70

The Calvert Cliffs equipment identified as meeting the criteria of NUREG-0612 are listed in Table 5-10. The remainder of the overhead load handling equipment existing at the time was excluded from NUREG-0612 by the Phase II submittal. The Spent Fuel Cask Handling Crane was designated "single failure proof" using NUREG-0612 and NUREG-0554 criteria. The change in status of the Spent Fuel Cask Handling Crane was acknowledged by the NRC via license amendment (Reference 2). These amendments allow the movement of heavy loads by the Spent Fuel Cask Handling Crane over the spent fuel pool. More information on this crane can be found in Section 9.7.2. The two monorails installed to handle the containment purge system blind flanges, the Containment Auxiliary Crane, and the four containment roof (exterior) jib cranes were evaluated as meeting the criteria of NUREG-0612. Table 5-10 states whether each piece of load handling equipment is subject to NUREG-0612 guidance and what the exemptions are based on.

The NRC accepted our method of compliance with NUREG-0612 (Reference 3). A Phase II report was submitted, but the NRC announced a blanket acceptance of Phase II via Generic Letter 85-11 (Reference 4). Based upon our method of compliance with the seven general guidelines of NUREG-0612, fuel damage events resulting from heavy loads incidents will result in offsite doses, due to the release of gap activity, of less than one-fourth of the 10 CFR Part 100 limits. Calvert Cliffs Nuclear Power Plant maintains various controlling procedures providing guidance on load path and lift height restrictions for those cranes subject to compliance with NUREG-0612 and Reference 10.

Restrictions on movement of heavy loads:

a. When minimum electrical conditions as defined in Technical Specifications are not met, movement of heavy loads over irradiated fuel is prohibited during shutdown conditions.

b. Operations involving movement of recently irradiated fuel in or movement of loads over recently irradiated fuel on a spacer in the spent fuel pool, other than with the single-failure-proof spent fuel cask handling crane, require at least an operable charcoal absorber bank, an operable exhaust fan and an operable high efficiency particulate air filter in the spent fuel pool ventilation system. Additionally, these operations also require that the spent fuel pool water level must be at or above 21 1/2' above the irradiated fuel assemblies.

c. When either unit is shut down, the fuel storage oil tank for the emergency diesel generator supporting the shut down unit must be operable for movement of heavy loads over irradiated fuel assemblies.

On December 1, 2008, the NRC issued Regulatory Issue Summary 2008-28 (Reference 10), which endorsed Nuclear Energy Institute (NEI) 08-05. The NEI guidance addressed a concern about lifting the reactor vessel head. An Engineering Evaluation was performed to document a reactor vessel head drop analysis that used the NEI guidance. The analysis concluded that the reactor vessels in both Units, attached piping, and support systems are capable of withstanding the impact loads of a hypothetical reactor vessel head drop concentrically onto the reactor vessel flange from a height of 29'. All stresses in steel components would be well within allowable limits; however, there would be significant damage to the concrete beneath the reactor vessel nozzle supports. Due to the damage in the concrete, the reactor vessel would come to rest fully supported by the RCS piping.

Procedures are used to control the lift and replacement of the reactor pressure vessel head. These procedures establish limits on load height, load weight, and medium present under the load.

Heavy load equipment is inspected to:

Code		Equipment
B30.2-1976		Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist) (Reference 5)
B30.9-1971		Slings (Reference 5)
B30.11-1973		Monorails, Underhung Cranes and Jib Cranes (Reference 6)
B30.16		Overhead Hoists (underhung) and Chain Falls (Reference 5)
	-1987	Same equipment in the Societe Alsacienne De Constructions Mecaniques De Mulhouse Emergency Diesel Generator Building (Reference 7)
CMAA-70-1975		Electric Overhead Traveling Cranes (Reference 8)
	-1983	Same equipment in the Independent Spent Fuel Storage Installation (Reference 9)
B30.22-2000		Articulating Boom Cranes

5.7.1 REFERENCES

1. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated May 27, 1983, Evaluation of Phase I of Control of Heavy Loads

2. Letter from D. G. McDonald, Jr. (NRC) to G. C. Creel (BGE), dated January 17, 1992, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant

3. Letter from S. A. McNeil (NRC) to G. C. Creel (BGE), dated August 7, 1989, Supplement to Phase I Safety Evaluation of the Control of Heavy Loads

4. Generic Letter 85-11, NRC to Licensees, dated June 28, 1985, Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612

5. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated January 4, 1982, Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318, Control of Heavy Loads

6. Letter from R. F. Ash (BGE) to D. G. Eisenhut (NRC), dated August 2, 1982, Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318, Control of Heavy Loads

7. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated July 20, 1993, Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2; Docket Nos. 50-317 & 50, Emergency Diesel Generator Project – SACM Diesel Generator and Mechanical Systems Design Report

8. Letter from A. E. Lundvall, Jr. (BGE) to D. G. Eisenhut (NRC), dated March 1, 1982, Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318, Control of Heavy Loads

9. Letter from G. C. Creel (BGE) to NRC, Director, Division of Industrial and Medical Nuclear Safety Office of Nuclear Material Safety and Safeguards, dated July 20, 1993, Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) Application

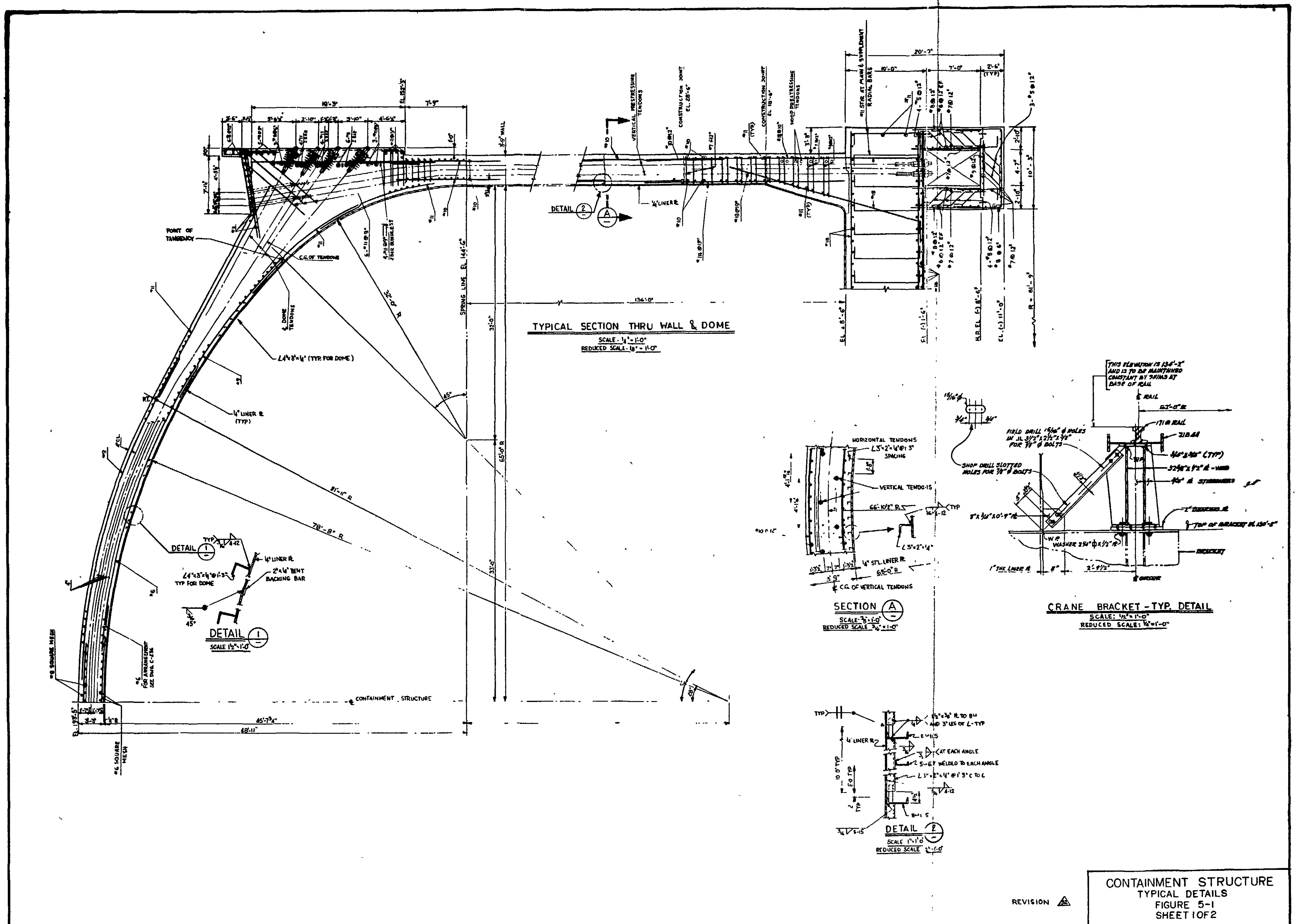
10. NRC Regulatory Issue Summary 2008-28, dated December 1, 2008, Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts

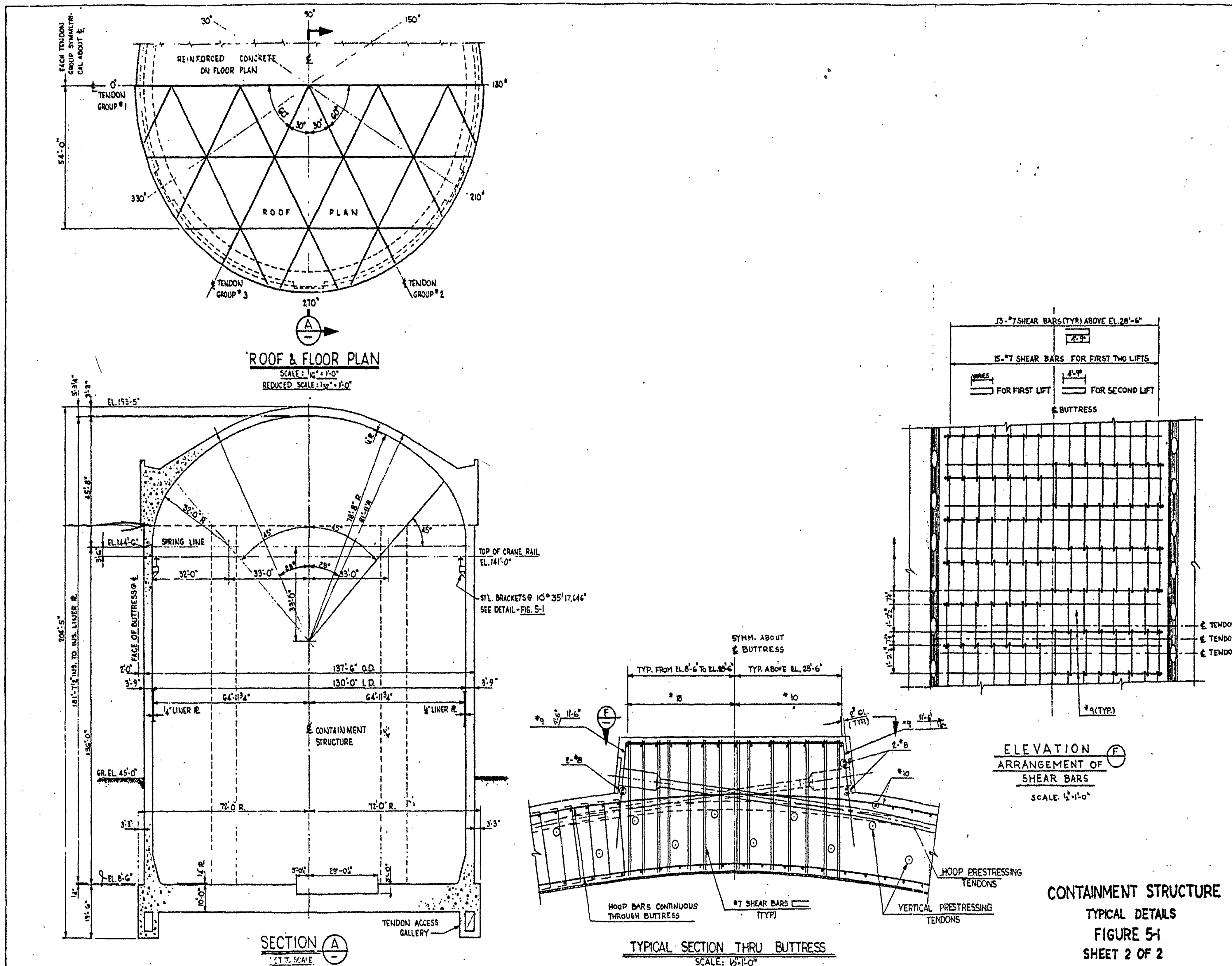
TABLE 5-10

EQUIPMENT MEETING NUREG-0612 CRITERIA

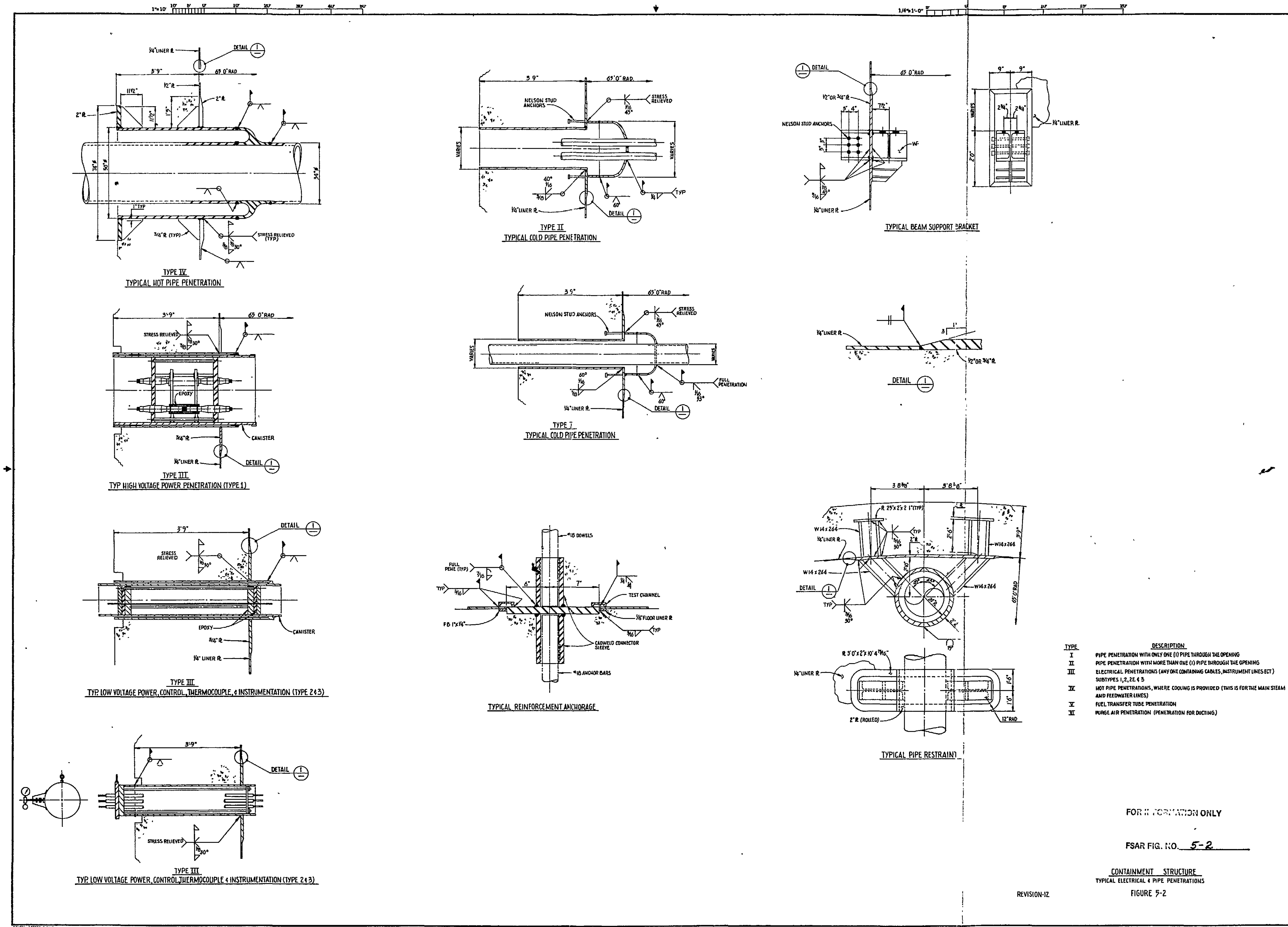
LIFTING EQUIPMENT	NUREG-0612	REASON FOR EXCLUSION FROM NUREG-0612
Polar Crane	Y	
Intake Structure Semi-Gantry Crane	Y	
Transfer Machine Jib-Crane	Y	
Purge Flange Monorail (2)	Y	
Spent Fuel Cask Crane	Y	
Containment Roof (Exterior) Jib Crane (4)[a]	Y	
Containment Auxiliary Crane	Y	
Turbine Building Main Crane	N	Sufficient separation
Turbine Building Auxiliary Crane	N	Sufficient separation
Filter Cask Monorail	N	No floor penetration
Solid Waste Disposal Trolley	N	No safe shutdown or decay heat removal systems endangered
Diesel Generator Room Monorail	N	Sufficient separation
Main Steam Room Monorail	N	Sufficient separation; No safe shutdown or decay heat removal systems endangered
Main Steam (MSIV) Room Access Hoist	N	Sufficient separation; No safe shutdown or decay heat removal systems endangered
Machine Shop Monorail	N	Sufficient separation
Containment Equipment Hatch Hoist	N	Sufficient separation
Component Cooling Water Room Hoist	N	No floor penetration; No safe shutdown or decay heat removal systems endangered
Switchgear Room Monorail Hoist	N	Sufficient separation
Chlorine House Monorail	N	Sufficient separation
Condensate Demineralizer Area Monorail	N	Sufficient separation
Condenser Waterbox Removal Monorail	N	Sufficient separation
Vertical Lifting Rail	N	Sufficient separation
Hot Machine Shop Crane	N	Sufficient separation
Decontamination Room Hoist	N	Sufficient separation

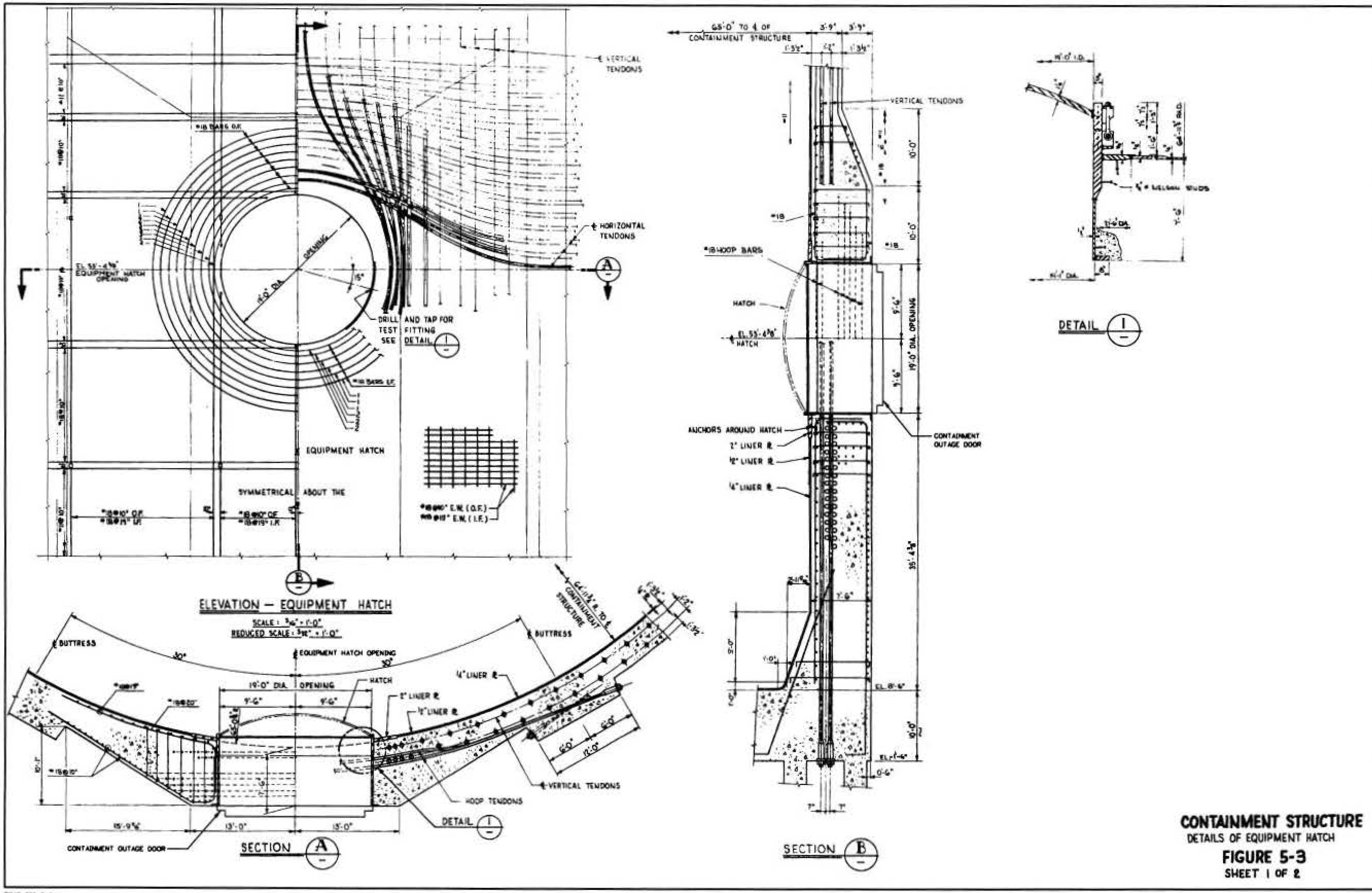
[a] These cranes have been classified as Augmented Quality and meet the criteria of NUREG-0612. These cranes are unique in that they do not have their own motorized hoisting systems. As such, they are inspected prior to use in lieu of the codes specified in Section 5.7.


TYPICAL SECTION THRU WALL & DOME
SCALE - 1/8" = 1'-0"
SECTION (A)
DETAIL (1)
DETAIL (2)
CRANE BRACKET - TYP. DETAIL
REVISION
CONTAINMENT STRUCTURE
TYPICAL DETAILS
FIGURE 5-1
SHEET 1 OF 2

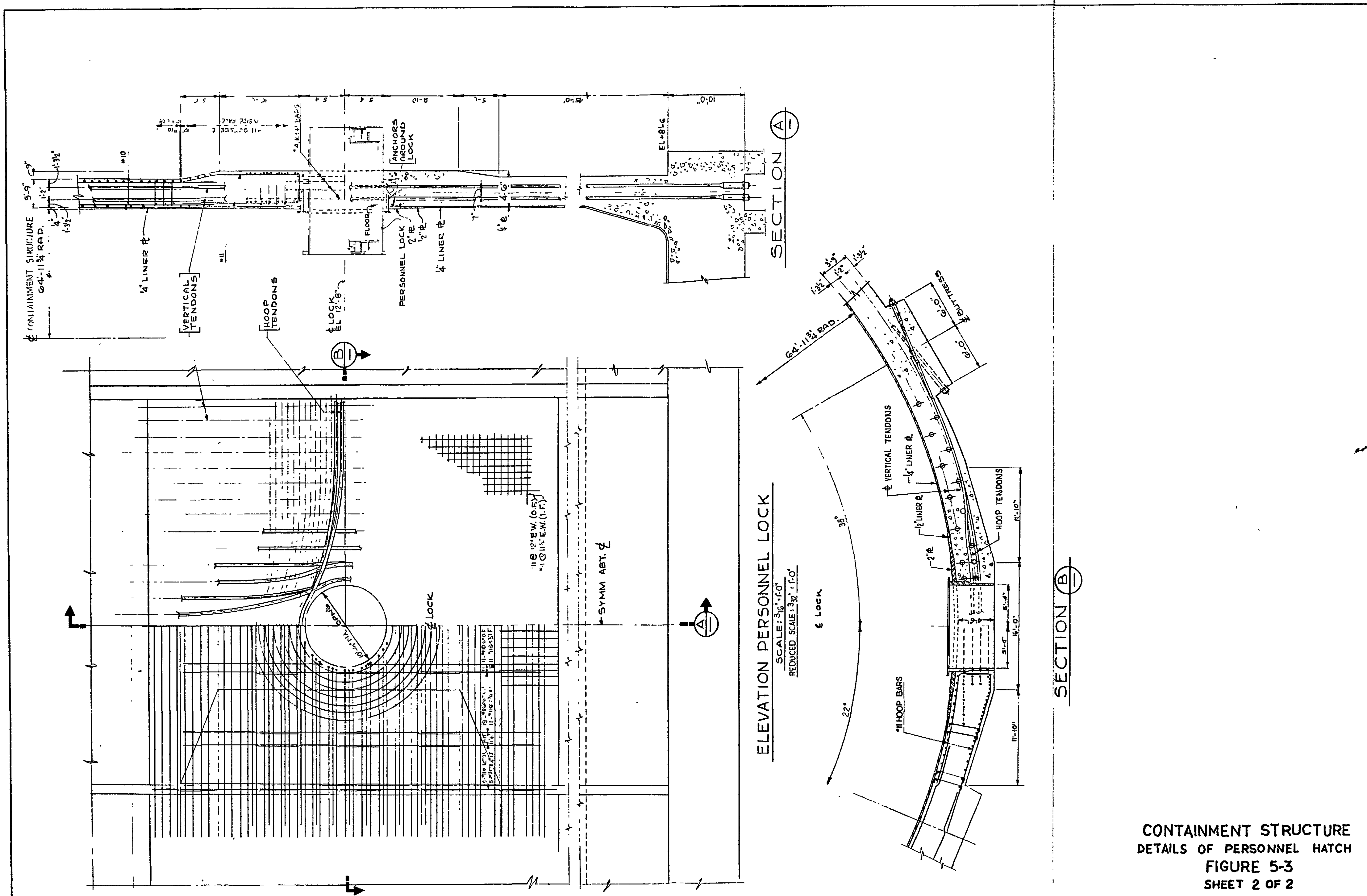

ROOF & FLOOR PLAN
SCALE: 1/16" = 1'-0"
REDUCED SCALE: 1/32" = 1'-0"
REINFORCED CONCRETE ON FLOOR PLAN
ROOF PLAN
TENDON GROUP #1
TENDON GROUP #2
TENDON GROUP #3
EACH TENDON GROUP SYMMETRICAL ABOUT ℄
SECTION A
SPRING LINE
TOP OF CRANE RAIL EL. 141'-0"
STL. BRACKETS @ 10°35'17.646" SEE DETAIL - FIG. 5-1
137'-6" O.D.
130'-0" I.D.
1/4" LINER PL
℄ CONTAINMENT STRUCTURE
FACE OF BUTTRESS ℄
GR. EL. 45'-0"
TENDON ACCESS GALLERY
TYPICAL SECTION THRU BUTTRESS
SCALE: 1/2"=1'-0"
SYMM. ABOUT ℄ BUTTRESS
HOOP BARS CONTINUOUS THROUGH BUTTRESS
#7 SHEAR BARS (TYP)
HOOP PRESTRESSING TENDONS
VERTICAL PRESTRESSING TENDONS
ELEVATION F
ARRANGEMENT OF SHEAR BARS
FOR FIRST LIFT
FOR SECOND LIFT
℄ BUTTRESS
℄ TENDON
#9 (TYP.)
CONTAINMENT STRUCTURE
TYPICAL DETAILS
FIGURE 5-1
SHEET 2 OF 2

Revision 33

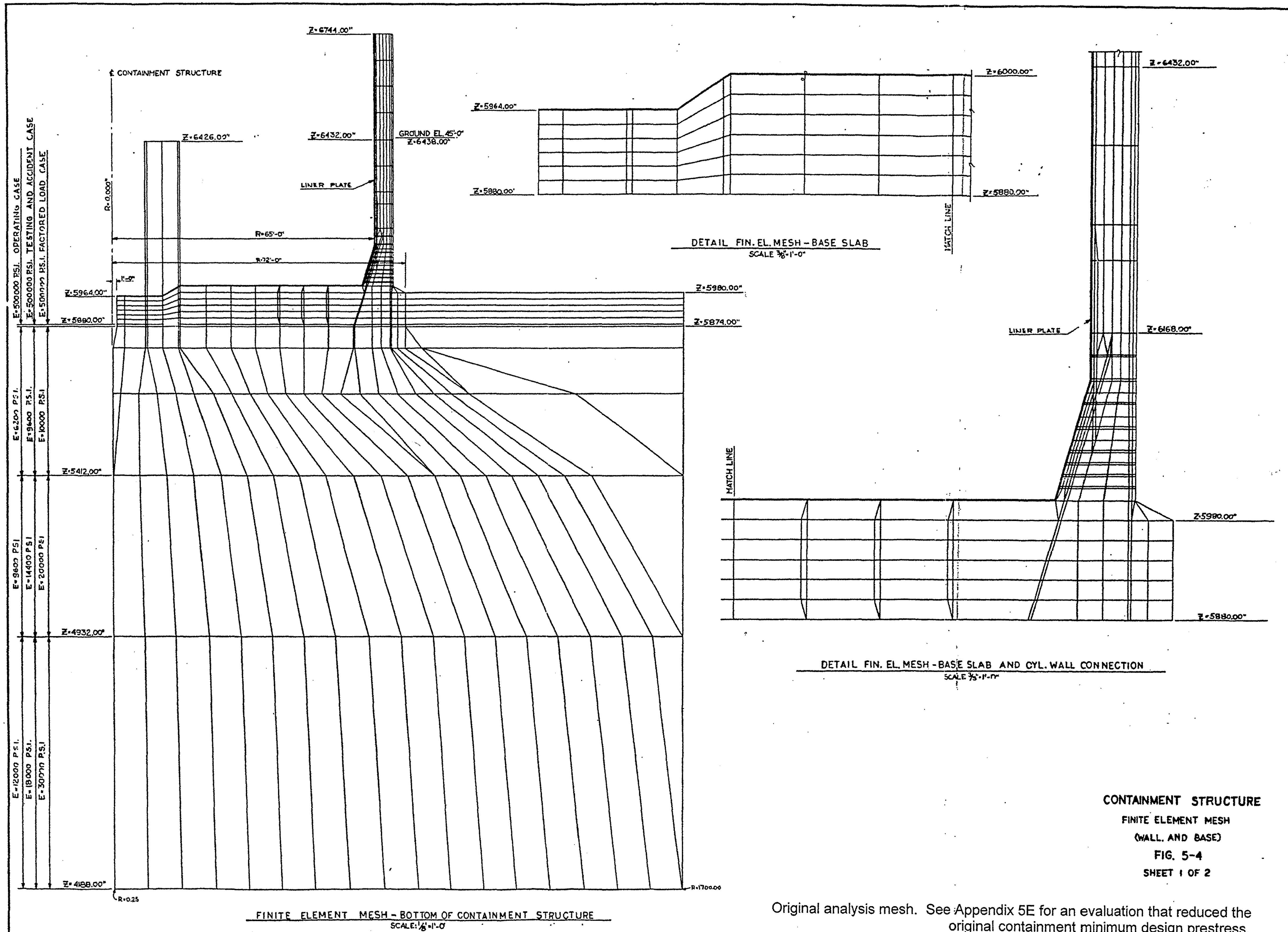

TYPE IV
TYPICAL HOT PIPE PENETRATION
TYPE II
TYPICAL COLD PIPE PENETRATION
TYPICAL BEAM SUPPORT BRACKET
TYPE III
TYP HIGH VOLTAGE POWER PENETRATION (TYPE 1)
TYPE I
TYPICAL COLD PIPE PENETRATION
DETAIL
TYPE III
TYP. LOW VOLTAGE POWER, CONTROL, THERMOCOUPLE, & INSTRUMENTATION (TYPE 2 & 3)
TYPICAL REINFORCEMENT ANCHORAGE
TYPICAL PIPE RESTRAINT
TYPE III
TYP. LOW VOLTAGE POWER, CONTROL, THERMOCOUPLE & INSTRUMENTATION (TYPE 2 & 3)
TYPE DESCRIPTION
I PIPE PENETRATION WITH ONLY ONE (1) PIPE THROUGH THE OPENING
II PIPE PENETRATION WITH MORE THAN ONE (1) PIPE THROUGH THE OPENING
III ELECTRICAL PENETRATIONS (ANY ONE CONTAINING CABLES, INSTRUMENT LINES ECT) SUBTYPES 1, 2, 2E & 3
IV HOT PIPE PENETRATIONS, WHERE COOLING IS PROVIDED (THIS IS FOR THE MAIN STEAM AND FEEDWATER LINES)
V FUEL TRANSFER TUBE PENETRATION
VI PURGE AIR PENETRATION (PENETRATION FOR DUCTING)
FSAR FIG. NO. 5-2
CONTAINMENT STRUCTURE
TYPICAL ELECTRICAL & PIPE PENETRATIONS
REVISION-12
FIGURE 5-2



FSAR FIG. 5-3
74940.DGN



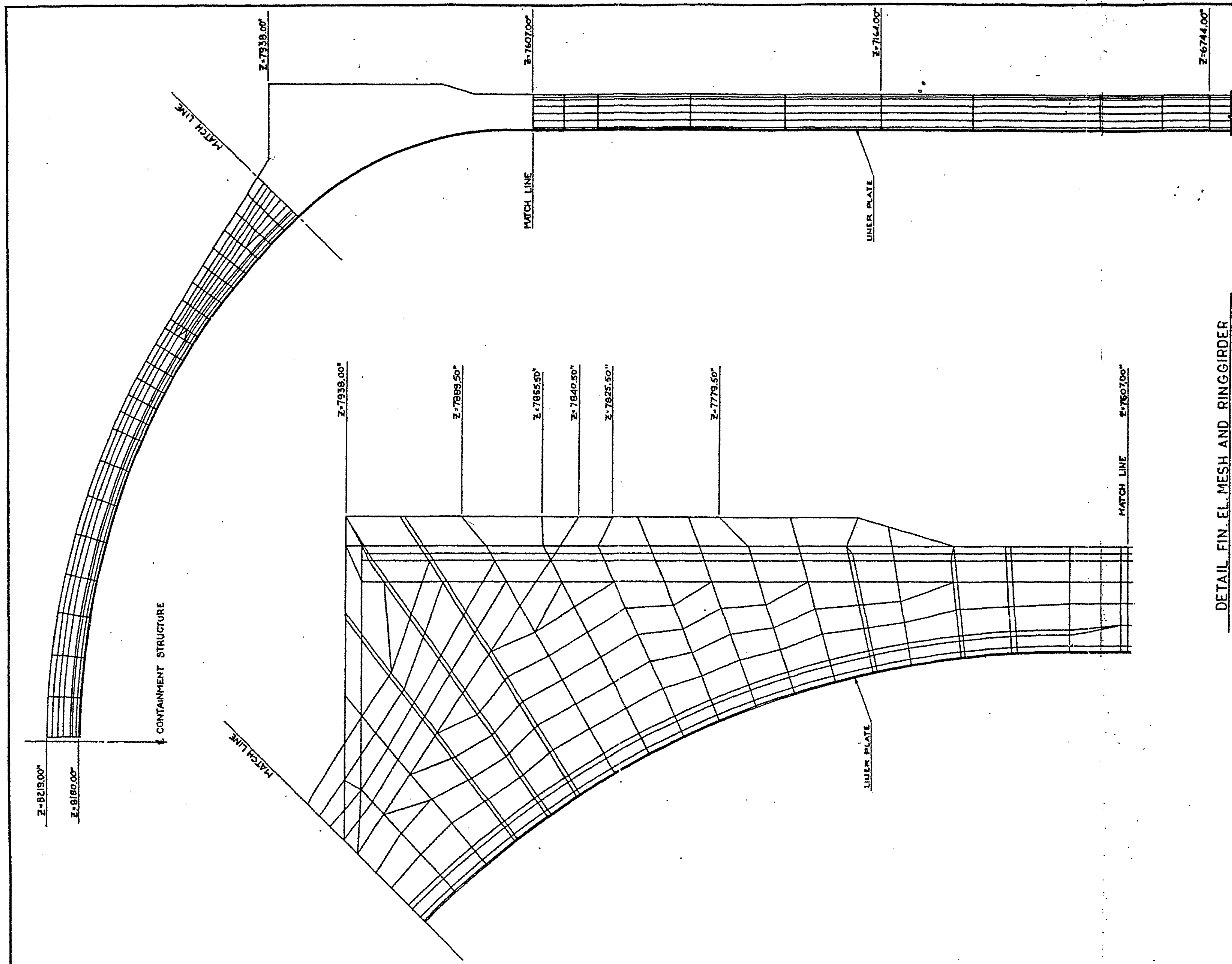
CONTAINMENT STRUCTURE
DETAILS OF PERSONNEL HATCH
FIGURE 5-3
SHEET 2 OF 2



CONTAINMENT STRUCTURE
FINITE ELEMENT MESH
(WALL AND BASE)
FIG. 5-4
SHEET 1 OF 2

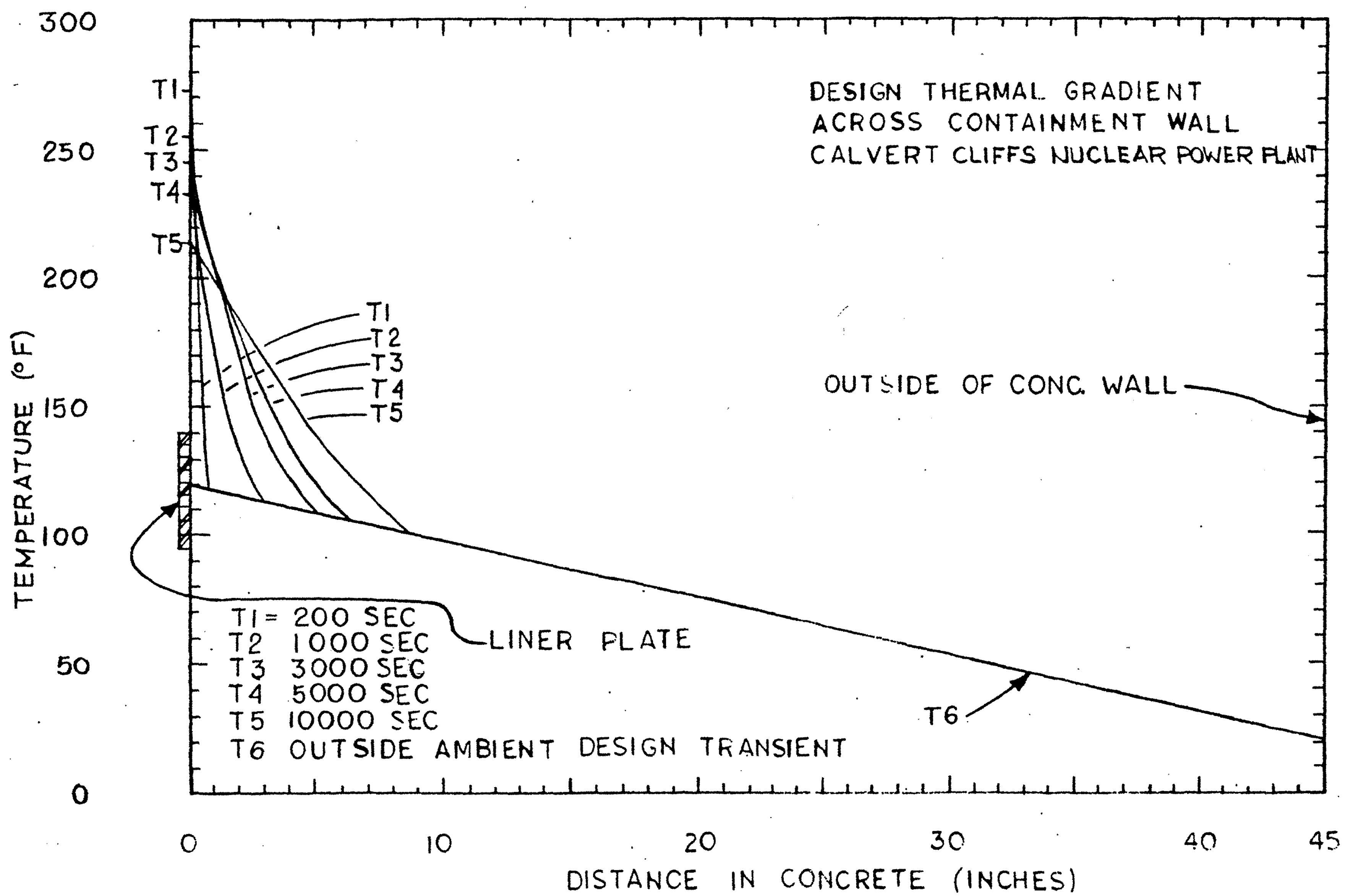
Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

Revision 33



CONTAINMENT STRUCTURE
FINITE ELEMENT MESH
(WALL AND DOME)
FIG. 5-4
SHEET 2 OF 2

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.



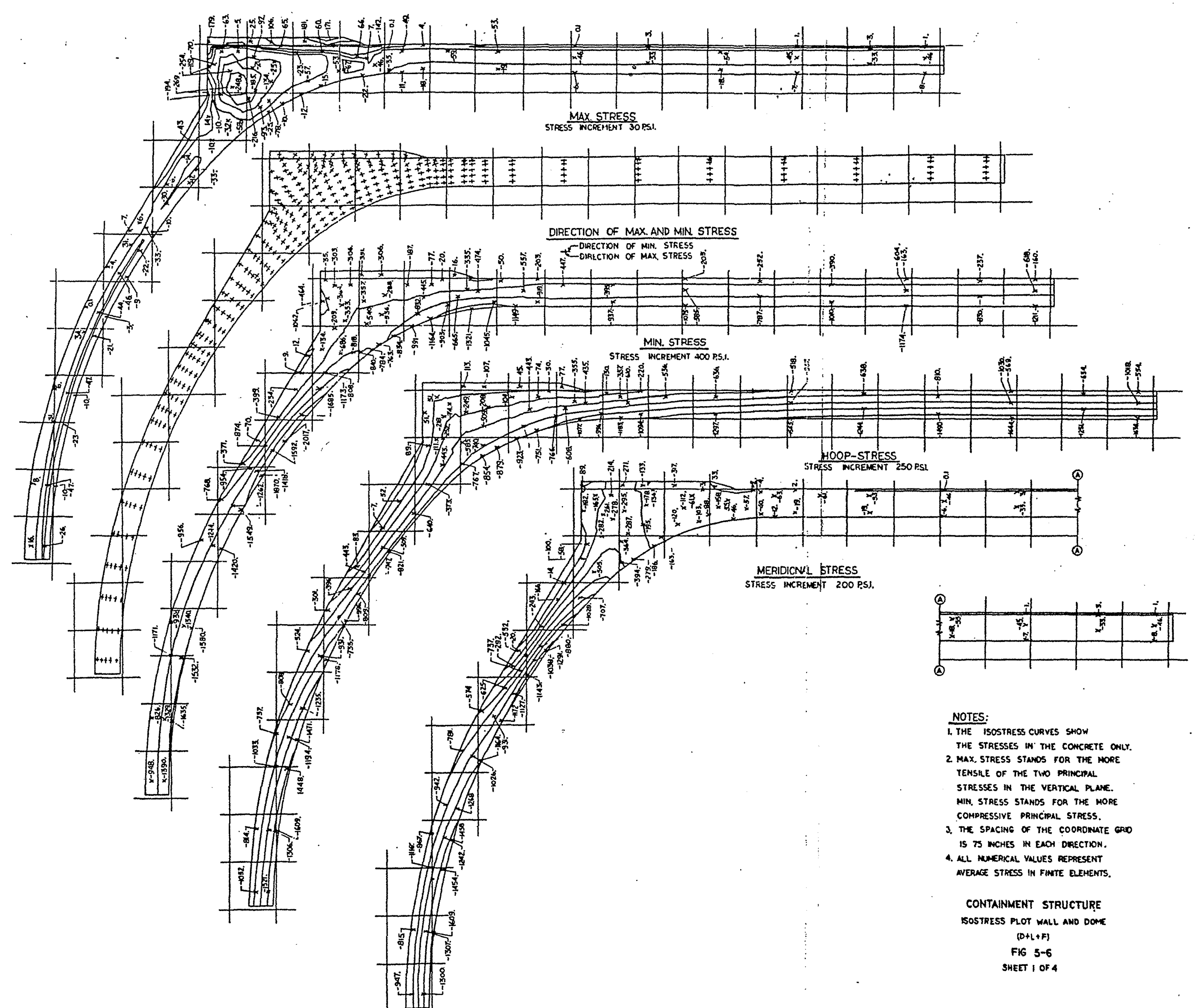
CONTAINMENT STRUCTURE

THERMAL GRADIENT

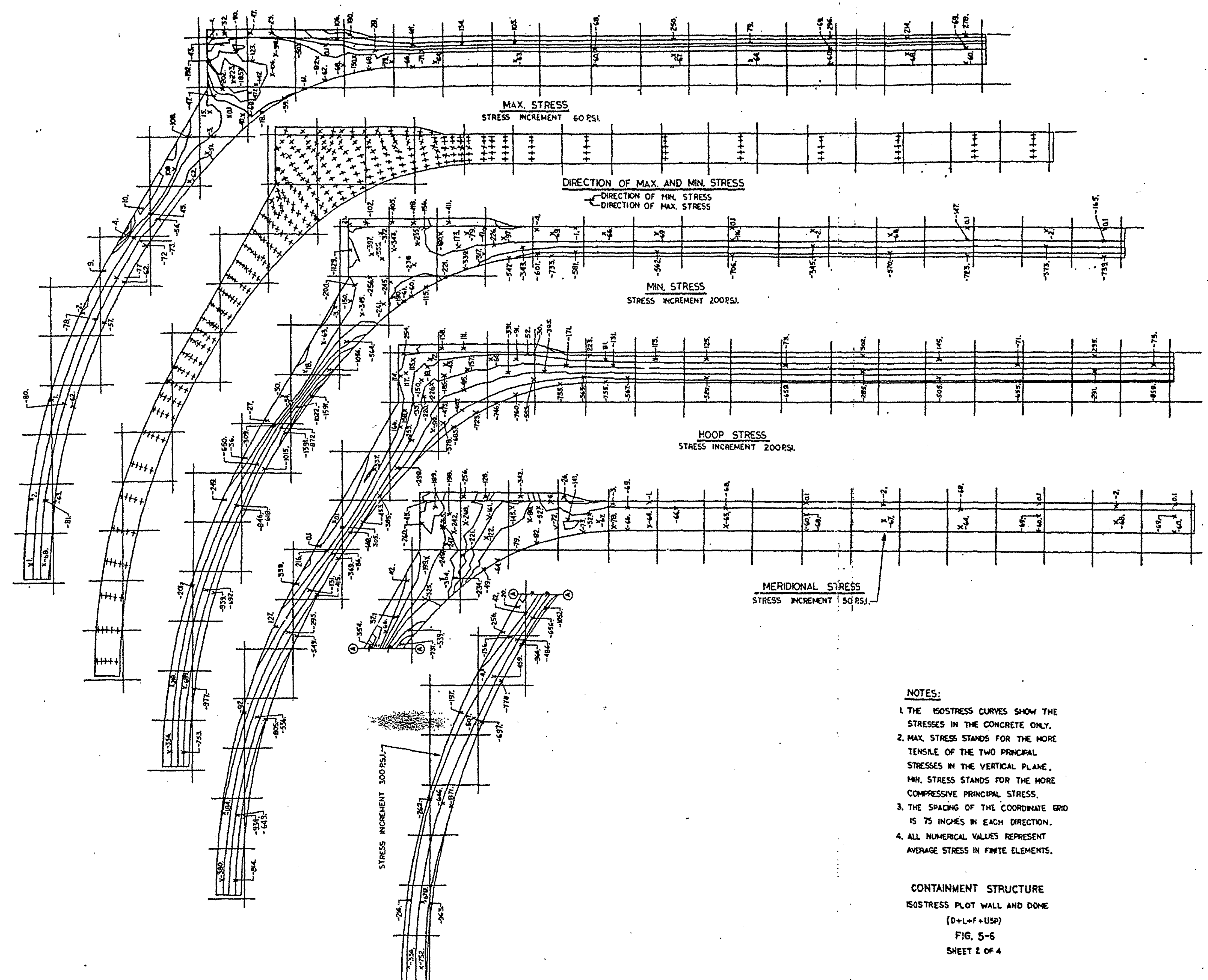
FIG. 5-5

See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

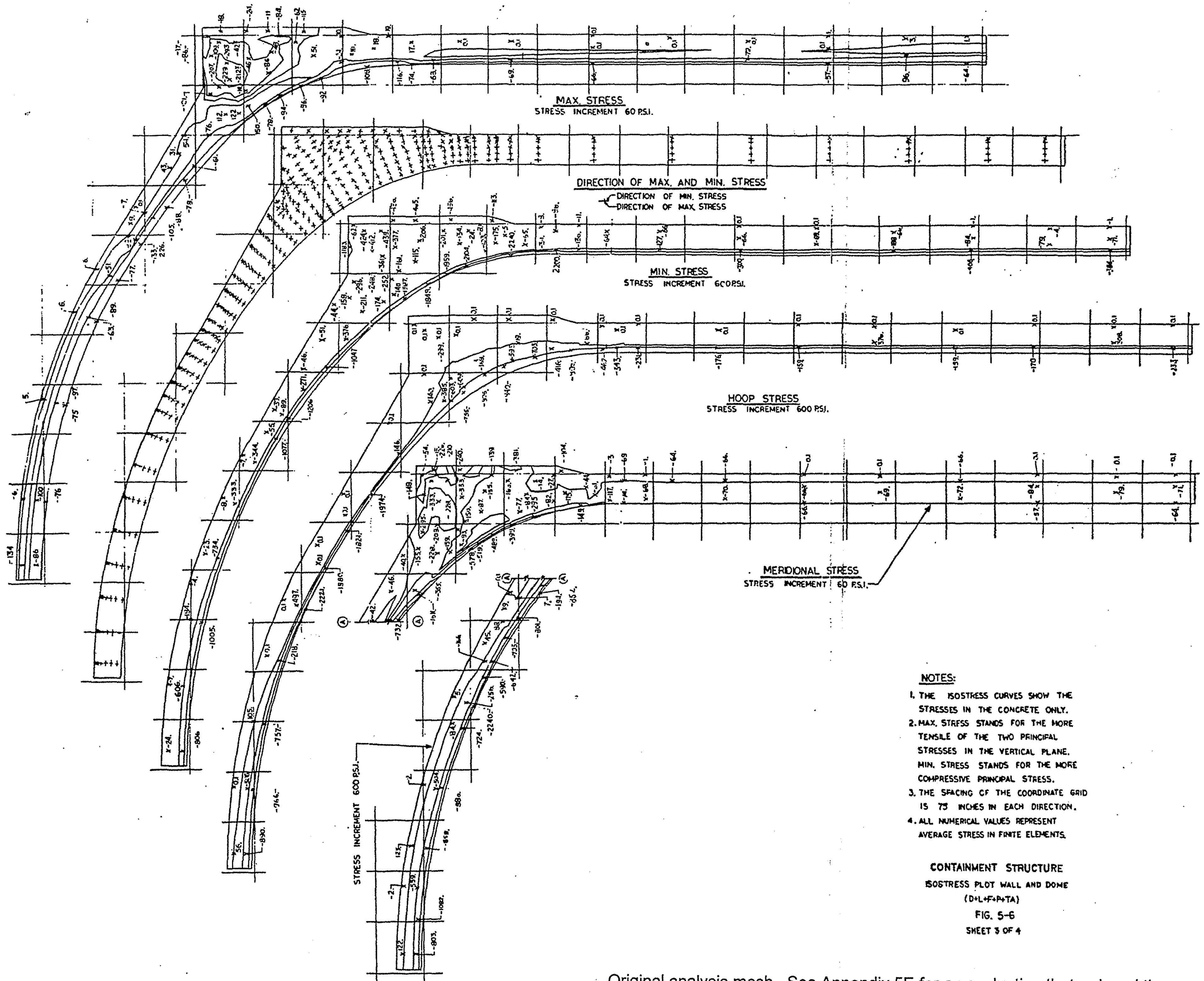
Revision 33



Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.



Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

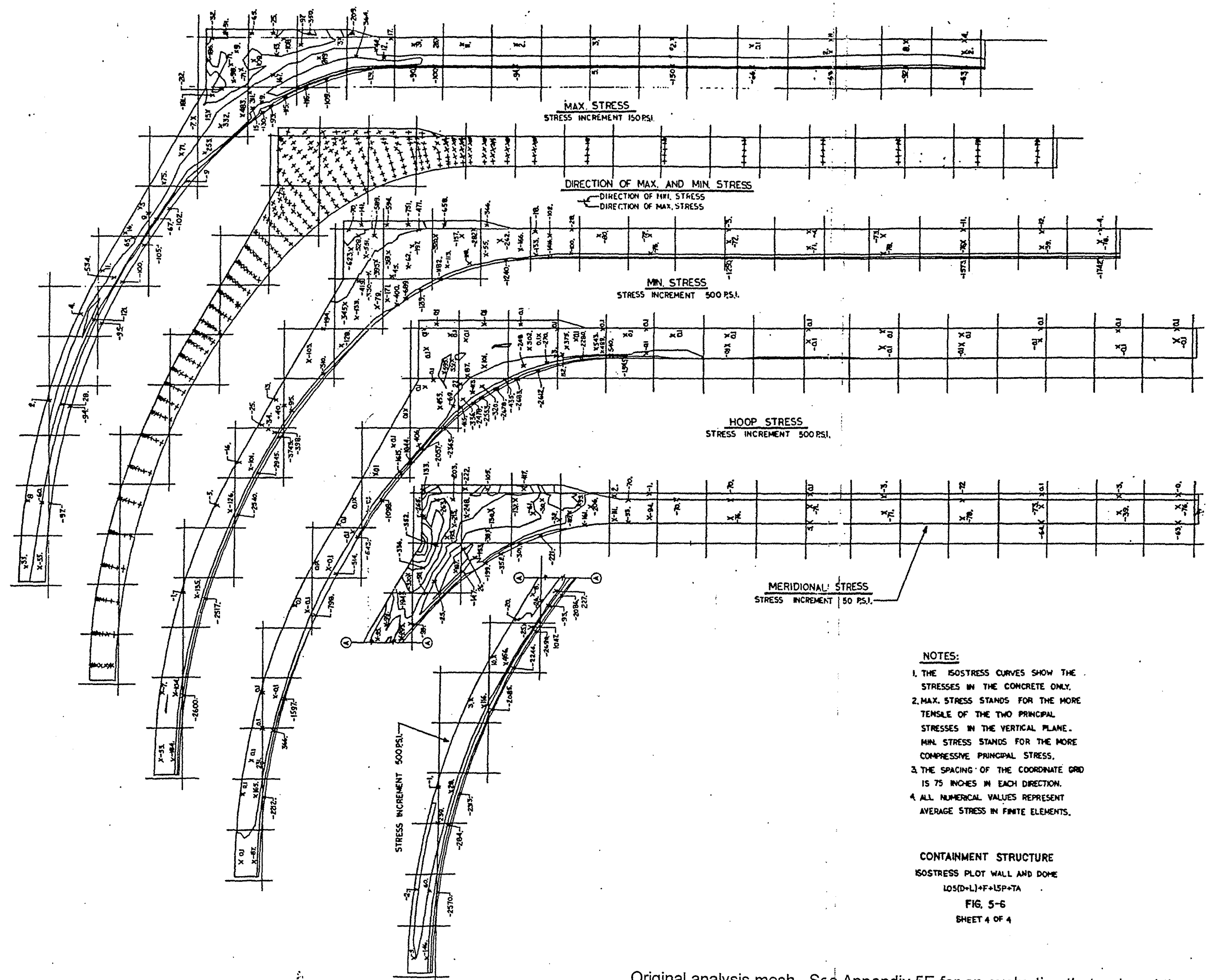


NOTES:

1. THE ISOSTRESS CURVES SHOW THE STRESSES IN THE CONCRETE ONLY.
2. MAX. STRESS STANDS FOR THE MORE TENSILE OF THE TWO PRINCIPAL STRESSES IN THE VERTICAL PLANE. MIN. STRESS STANDS FOR THE MORE COMPRESSIVE PRINCIPAL STRESS.
3. THE SPACING OF THE COORDINATE GRID IS 75 INCHES IN EACH DIRECTION.
4. ALL NUMERICAL VALUES REPRESENT AVERAGE STRESS IN FINITE ELEMENTS.

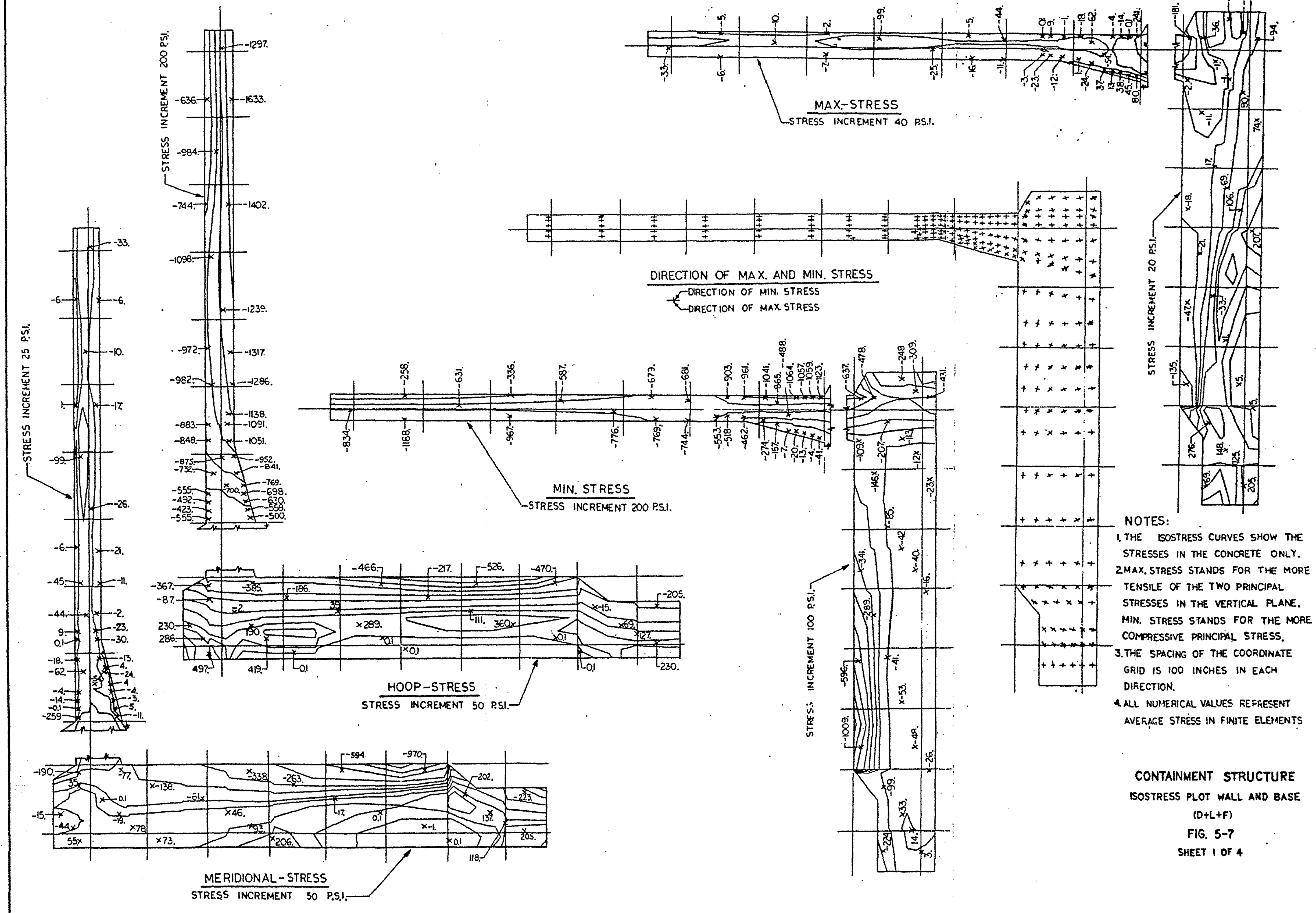
CONTAINMENT STRUCTURE
ISOSTRESS PLOT WALL AND DOME
(D+L+F+P+TA)
FIG. 5-6
SHEET 3 OF 4

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.



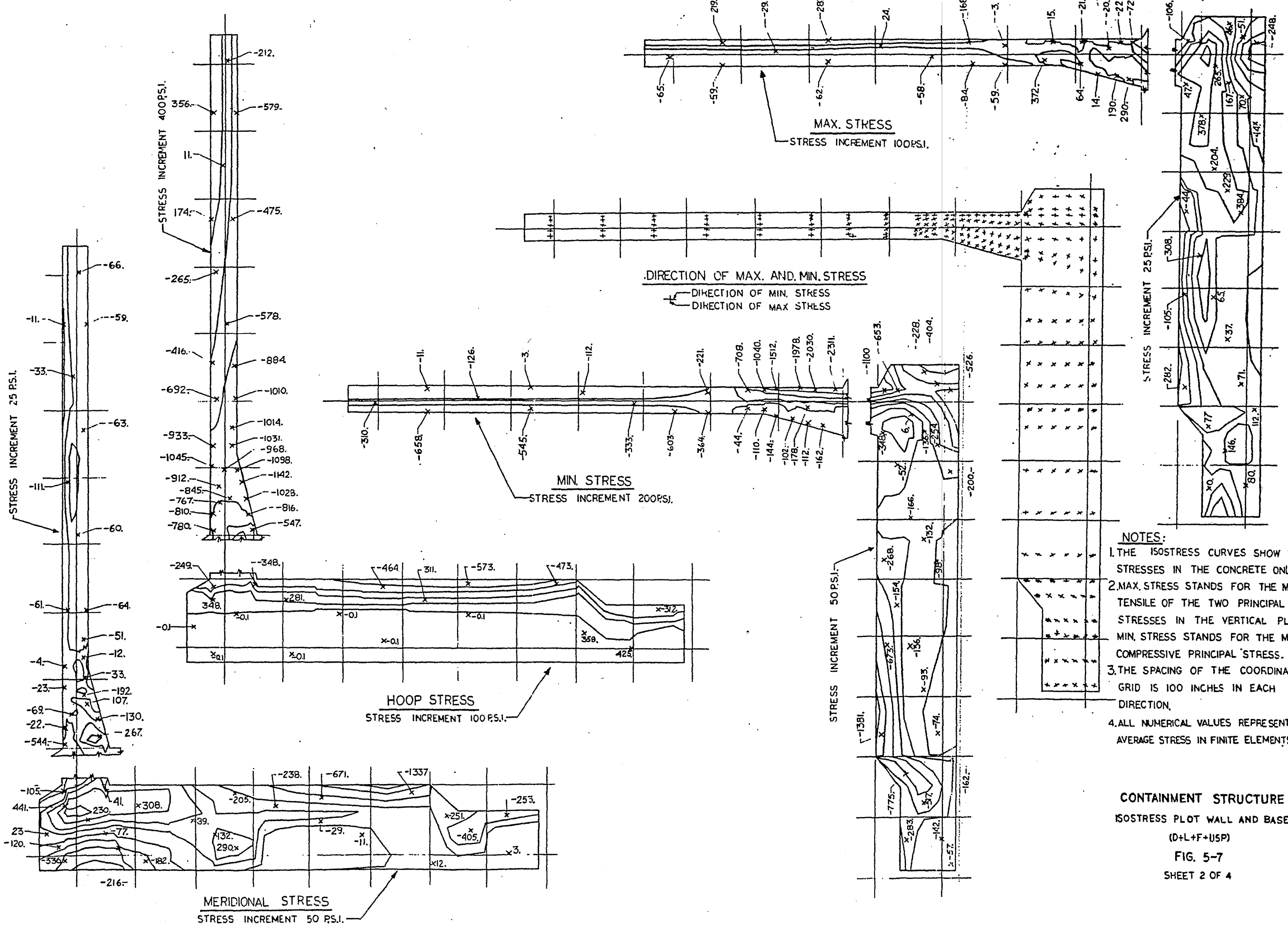
Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.



Revision 33

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

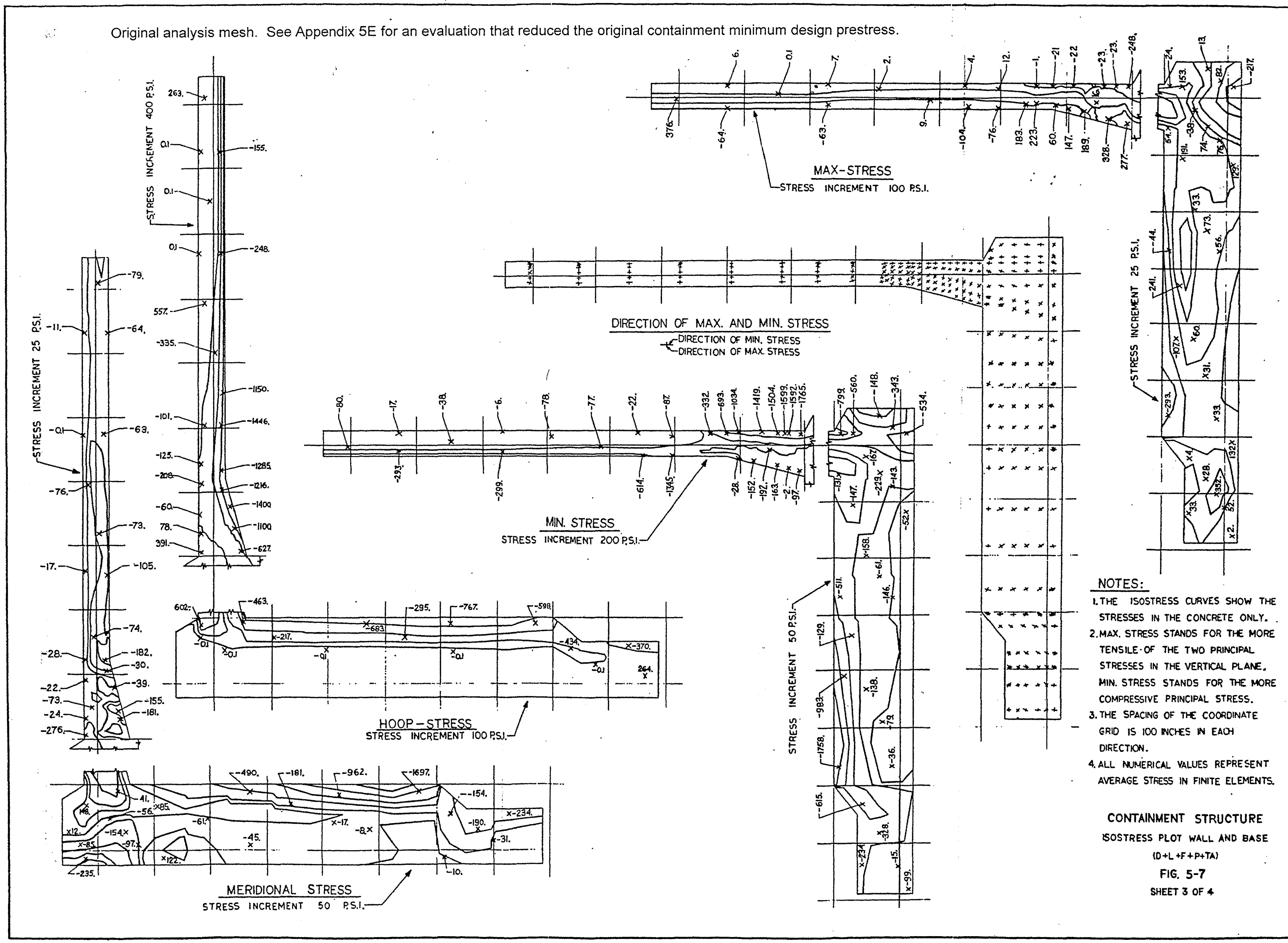


NOTES:

1. THE ISOSTRESS CURVES SHOW THE STRESSES IN THE CONCRETE ONLY.
2. MAX. STRESS STANDS FOR THE MORE TENSILE OF THE TWO PRINCIPAL STRESSES IN THE VERTICAL PLANE. MIN. STRESS STANDS FOR THE MORE COMPRESSIVE PRINCIPAL STRESS.
3. THE SPACING OF THE COORDINATE GRID IS 100 INCHES IN EACH DIRECTION.
4. ALL NUMERICAL VALUES REPRESENT AVERAGE STRESS IN FINITE ELEMENTS.

CONTAINMENT STRUCTURE
ISOSTRESS PLOT WALL AND BASE
(D+L+F+1.15P)
FIG. 5-7
SHEET 2 OF 4

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

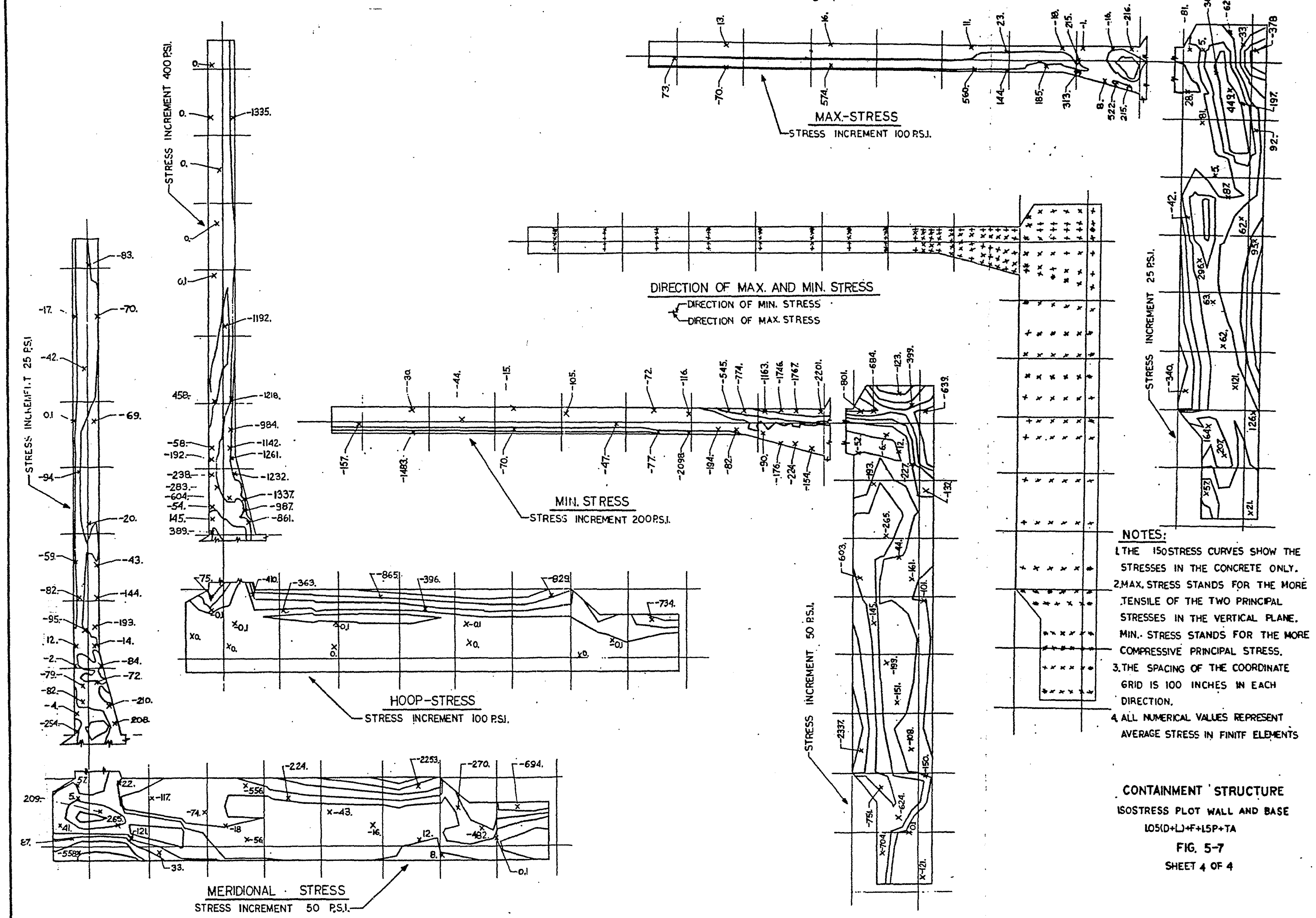


NOTES:

1. THE ISOSTRESS CURVES SHOW THE STRESSES IN THE CONCRETE ONLY.
2. MAX. STRESS STANDS FOR THE MORE TENSILE OF THE TWO PRINCIPAL STRESSES IN THE VERTICAL PLANE. MIN. STRESS STANDS FOR THE MORE COMPRESSIVE PRINCIPAL STRESS.
3. THE SPACING OF THE COORDINATE GRID IS 100 INCHES IN EACH DIRECTION.
4. ALL NUMERICAL VALUES REPRESENT AVERAGE STRESS IN FINITE ELEMENTS.

CONTAINMENT STRUCTURE
ISOSTRESS PLOT WALL AND BASE
(D+L+F+P+TA)
FIG. 5-7
SHEET 3 OF 4

Original analysis mesh. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.



NOTES:

1. THE ISOSTRESS CURVES SHOW THE STRESSES IN THE CONCRETE ONLY.
2. MAX. STRESS STANDS FOR THE MORE TENSILE OF THE TWO PRINCIPAL STRESSES IN THE VERTICAL PLANE. MIN. STRESS STANDS FOR THE MORE COMPRESSIVE PRINCIPAL STRESS.
3. THE SPACING OF THE COORDINATE GRID IS 100 INCHES IN EACH DIRECTION.
4. ALL NUMERICAL VALUES REPRESENT AVERAGE STRESS IN FINITE ELEMENTS

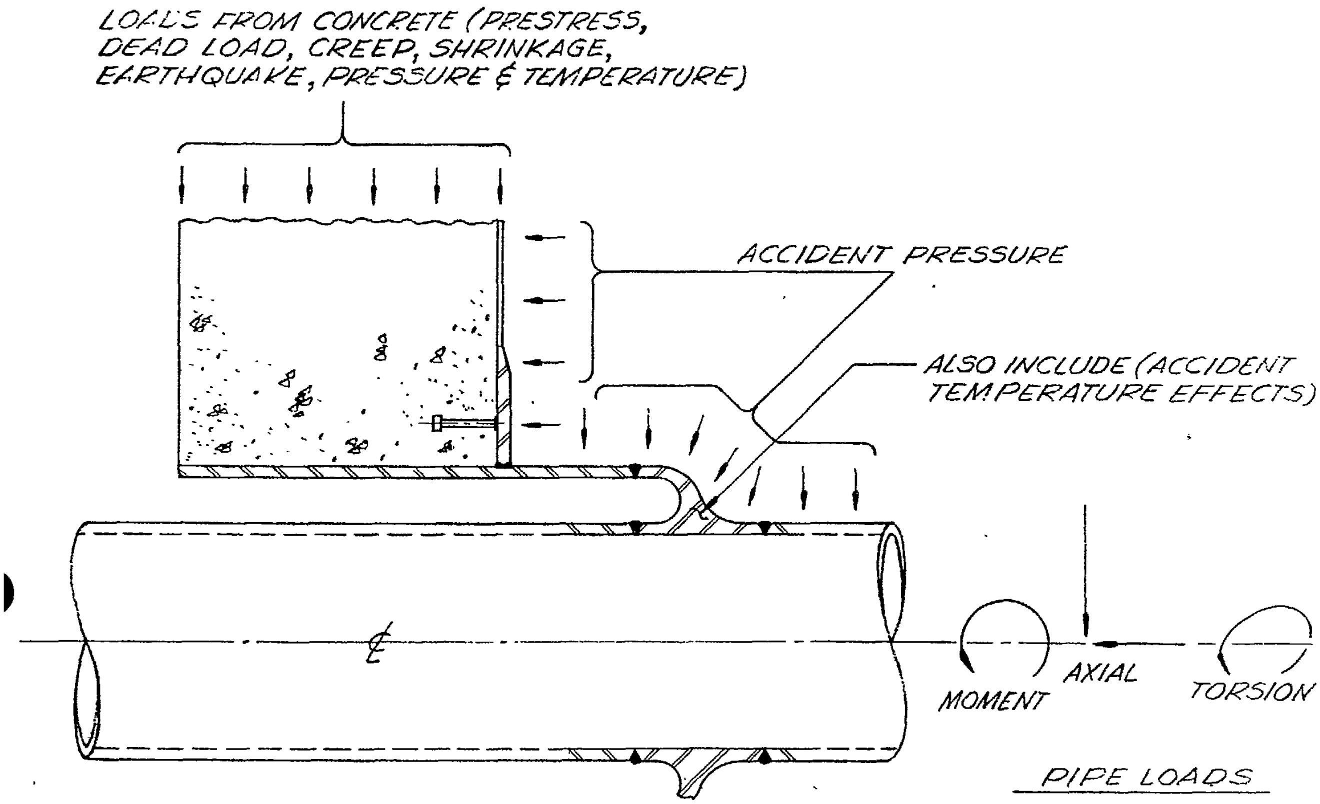
CONTAINMENT STRUCTURE

ISOSTRESS PLOT WALL AND BASE

1.05(D+L)+F+1.5P+TA

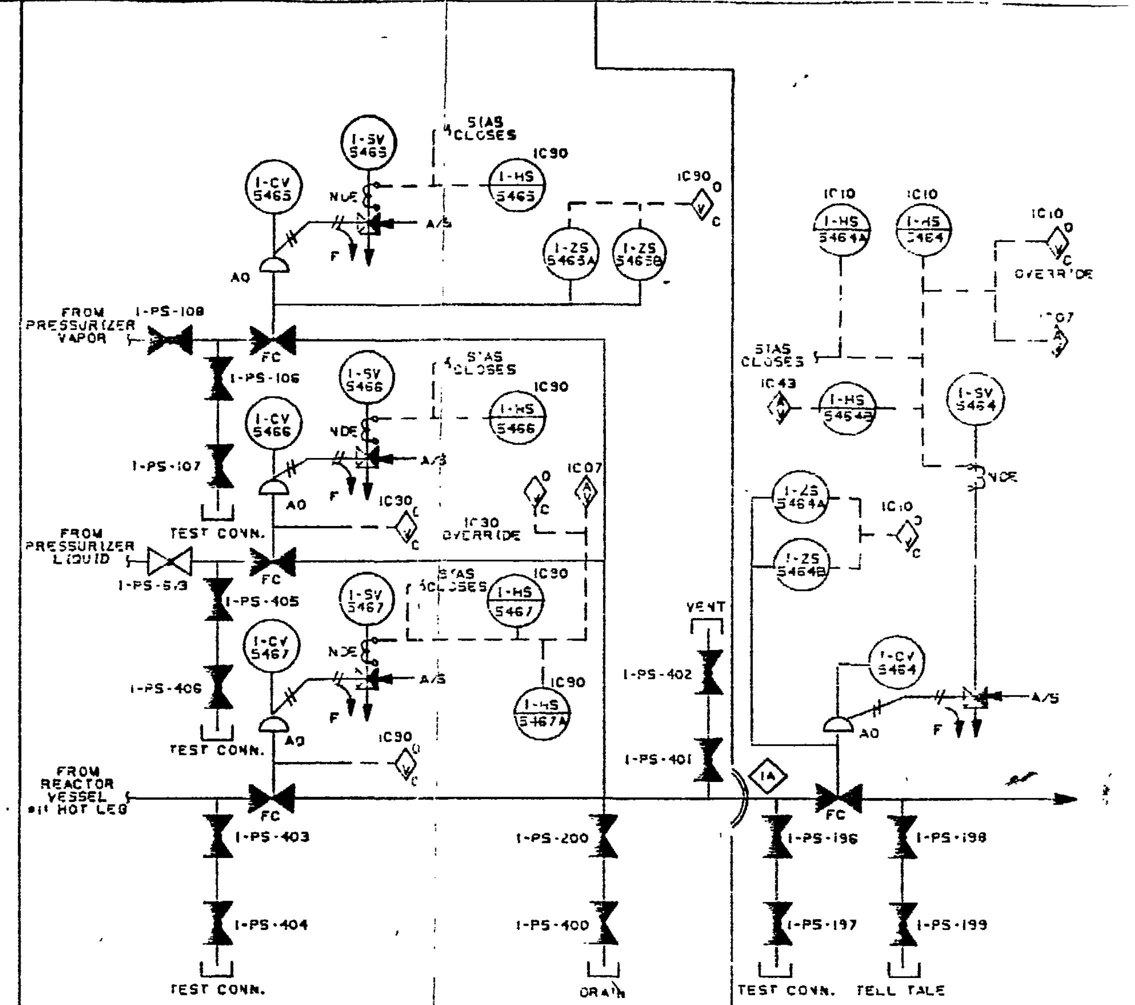
FIG. 5-7

SHEET 4 OF 4



CONTAINMENT STRUCTURE
PENETRATION LOADS

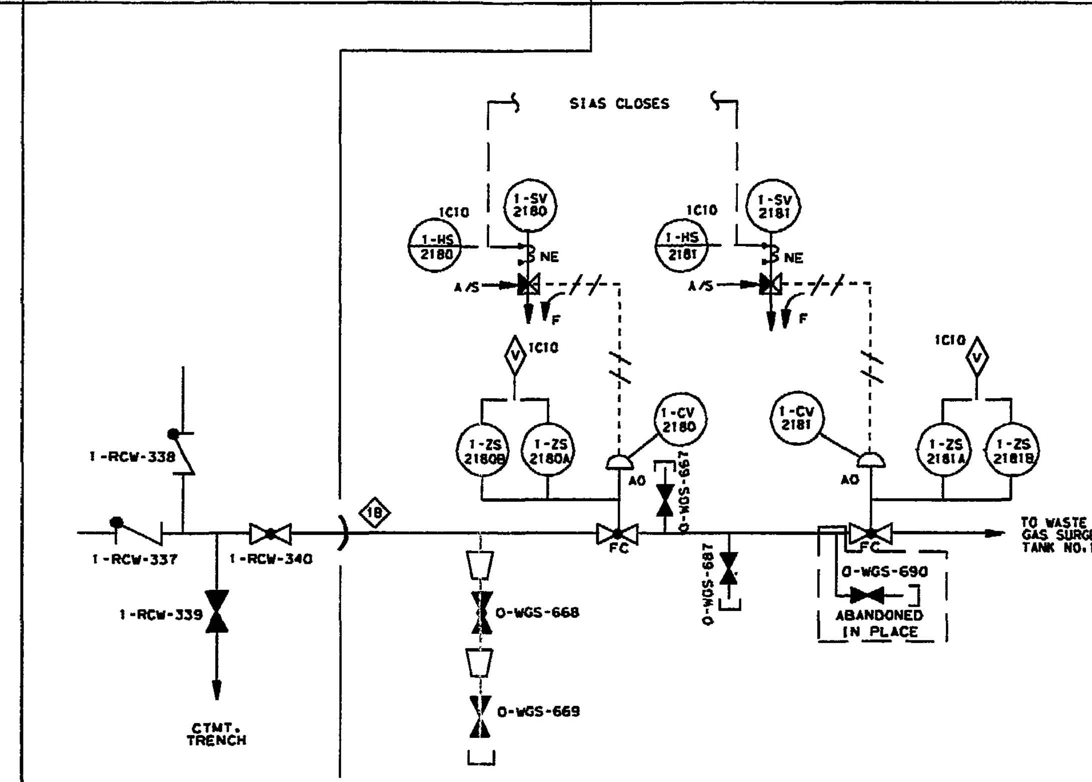
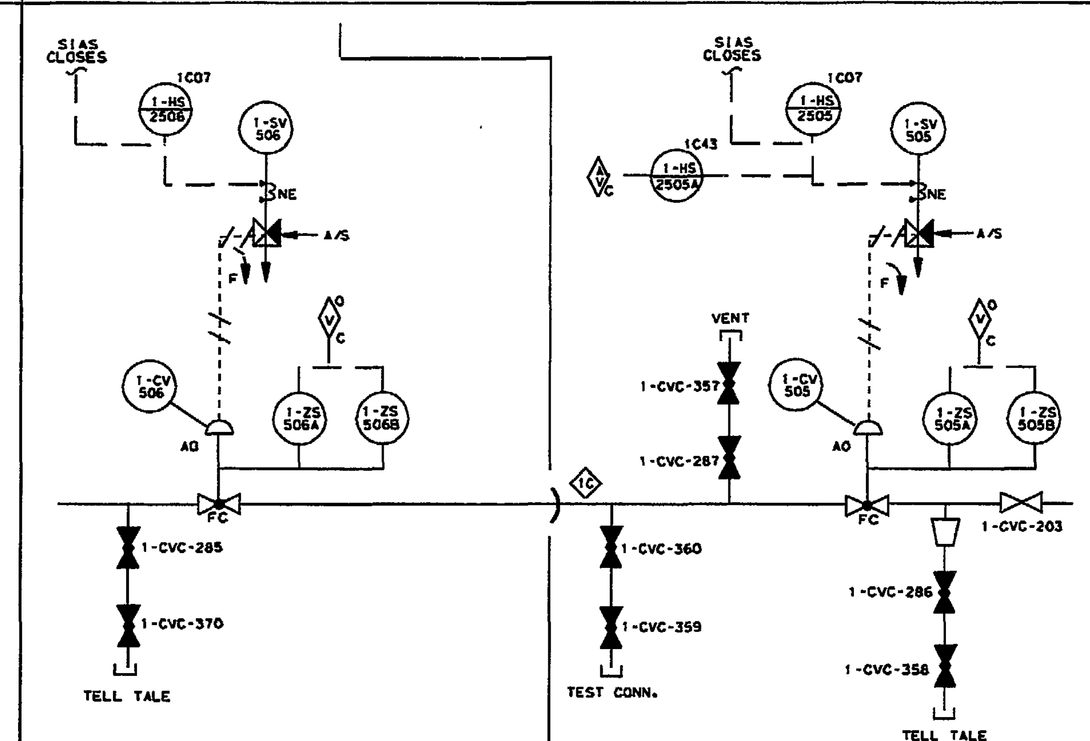
FIGURE 5-9

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
1A	REACTOR COOLANT AND PRESSURIZER SAMPLING	21	I	19	WEST PIPING PENETRATION RM.	3/4"	OM-66 SH.1 OF 3	CV-5464 CV-5465 CV-5466 CV-5467	CLOSED CLOSED CLOSED CLOSED	

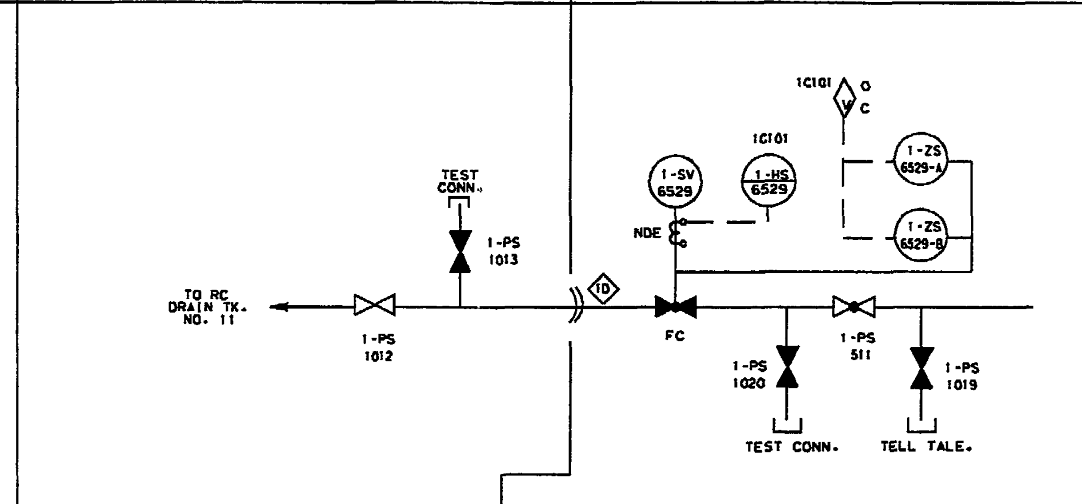
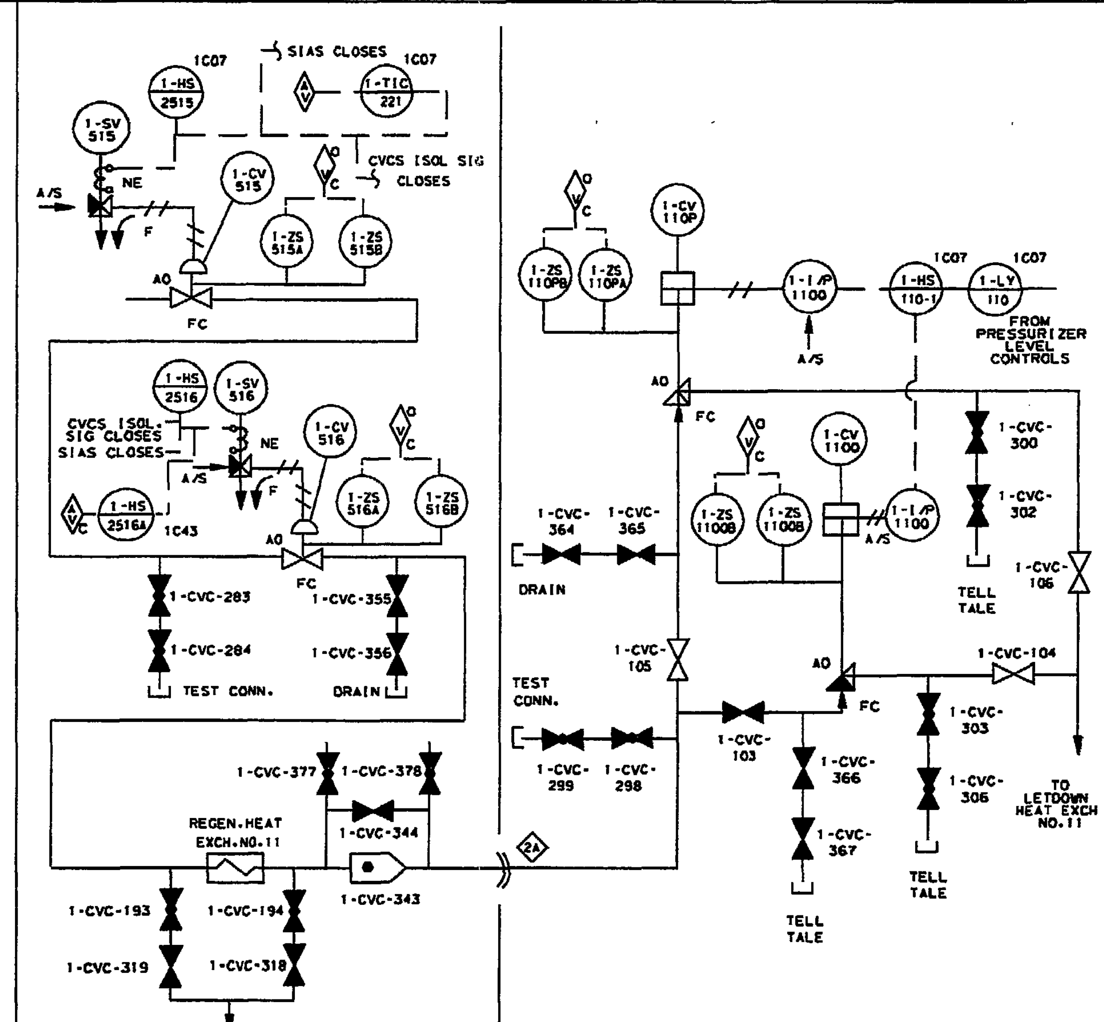
NOTES

1. INFORMATION ON ALL SHEETS SUCH AS DWG.NOS., ISOLATION VALVE NOS. AND THE DIAGRAM ARE GIVEN FOR UNIT I ONLY. REFER TO THE APPROPRIATE DOCUMENTS FOR SIMILAR INFORMATION ON UNIT 2.
2. THE DIAGRAMS ARE TAKEN FROM THE REFERENCED OM DRAWINGS. ALL VALVES ARE SHOWN IN THEIR POSITION DURING NORMAL PLANT OPERATION (MODE 1) AS SHOWN ON THE OM DRAWINGS.
3. SYMBOLS ON THE DIAGRAMS ARE IDENTIFIED ON FSAR FIG. 9-1. ALL SOLENOID VALVES ARE SHOWN IN THEIR DEENERGIZED POSITION PER FSAR FIG. 9-1.

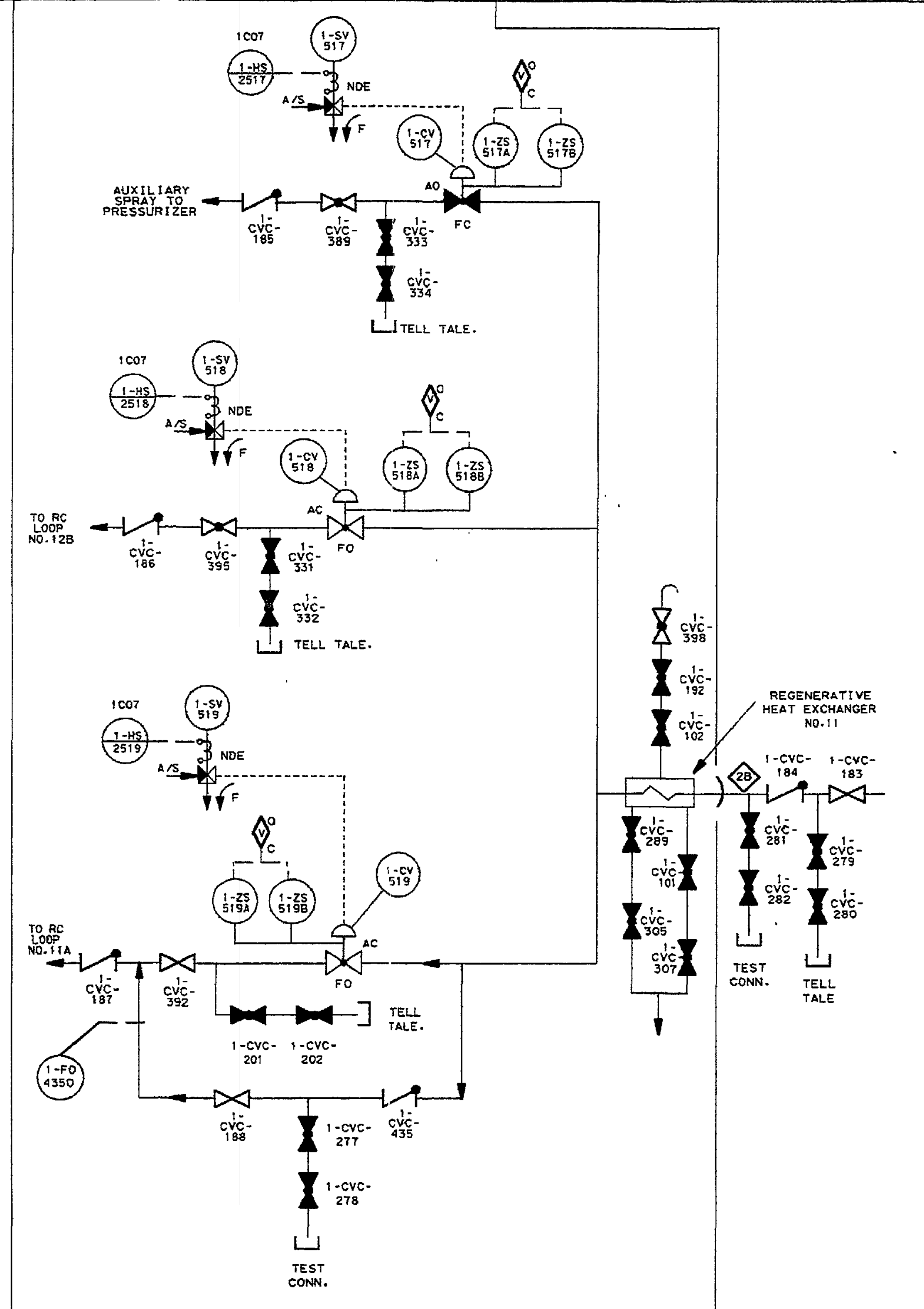
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 1

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
1B	RC DRAIN TANK NO.11 VENT HEADER TO WASTE GAS SURGE TANK	36	III	12	WEST PIPING PENETRATION RM.	2"	OM-078 SH0001 OM-077 SH0001	CV-2180 CV-2181	CLOSED CLOSED	
1C	REACTOR COOLANT PUMP SEALS CONTROLLED BLEED OFF	45	I	6	WEST PIPING PENETRATION RM.	3/4"	OM-073 SH0002	CV-505 CV-506	CLOSED CLOSED	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 2 | REVISION: 21

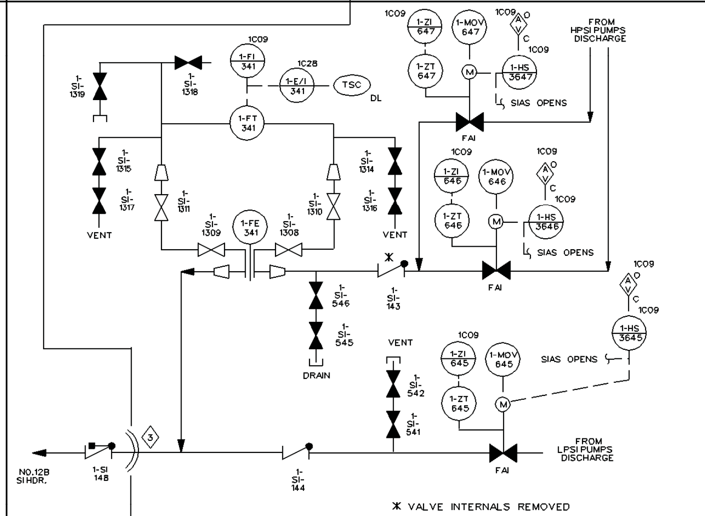
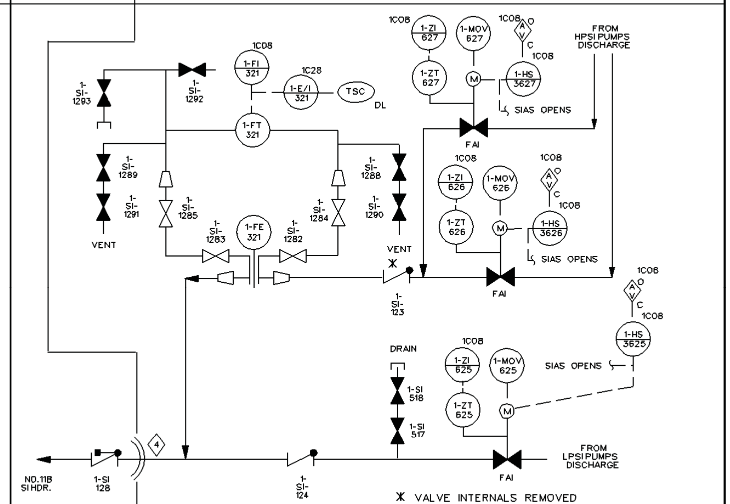
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
1D	POST-ACCIDENT SAMPLING LIQUID RETURN TO RC DRAIN TANK	--	IV	37	WEST PIPING PENETRATION RM.	1/4"	OM-066 SH0003	SV-6529	LOCKED SHUT	
2A	LETDOWN LINE TO PURIFICATION DEMIN. NOTE 1: EITHER CVC-103 OR CVC-105 IS NORMALLY OPEN. THE SECOND VALVE IS NORMALLY CLOSED.	2	I	7	WEST PIPING PENETRATION RM.	2"	OM-073 SH0002 SH0003	CV-515 CV-516 CVC-103 CVC-105	CLOSED CLOSED SEE NOTE 1	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 3 | REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM
2B	REACTOR COOLANT CHARGING LINE	3	I	9	WEST PIPING PENETRATION RM.	2"	OM-73 SH.2 OF 3	CVC-184 CVC-435 CV-519 CV-518 CV-517	----- ----- OPEN OPEN CLOSED	

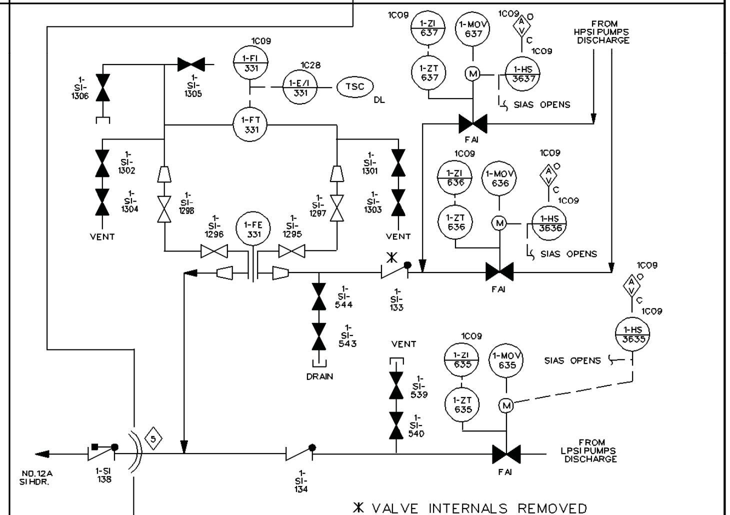
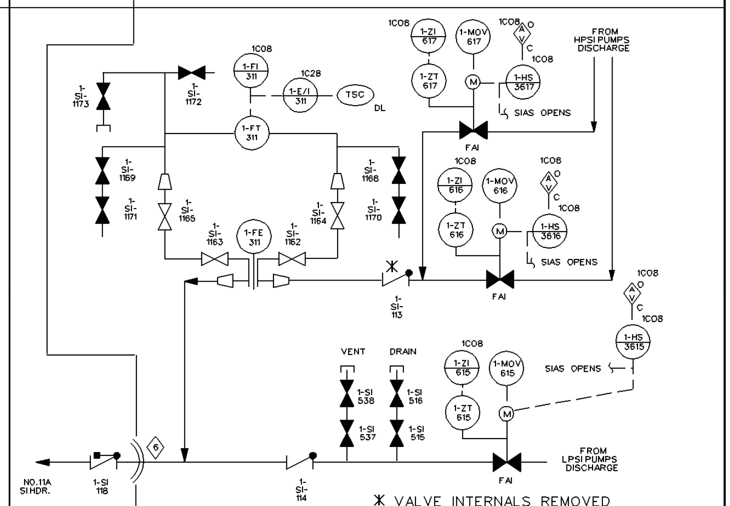
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 4 | REVISION: 17

FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 05)

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
3	SAFETY INJECTION LOW & HIGH PRESSURE (SI TK.NO.12B DISCH)	6	I	10	WEST PIPING PENETRATION RM.	6"	OM-74 SH.1 OF 3 SH.2 OF 3	SI-148 MOV-645 MOV-646 MOV-647	- - - OPEN OPEN OPEN	
4	SAFETY INJECTION LOW & HIGH PRESSURE (SI TK. NO. 11B DISCH)	7	I	10	EAST PIPING PENETRATION RM.	6"	OM-74 SH.1 OF 3 SH.2 OF 3	SI-128 MOV-625 MOV-626 MOV-627	- - - OPEN OPEN OPEN	

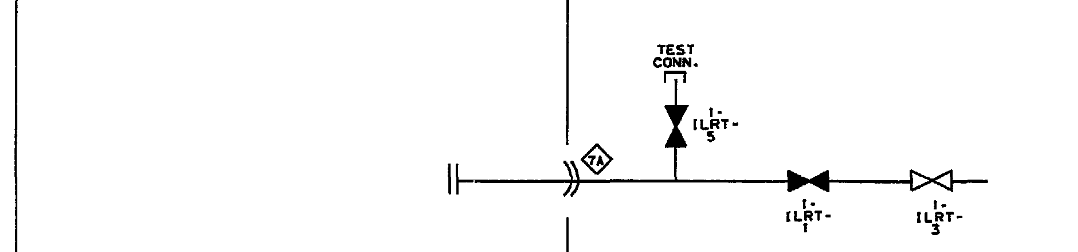
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 5 | REVISION: 2

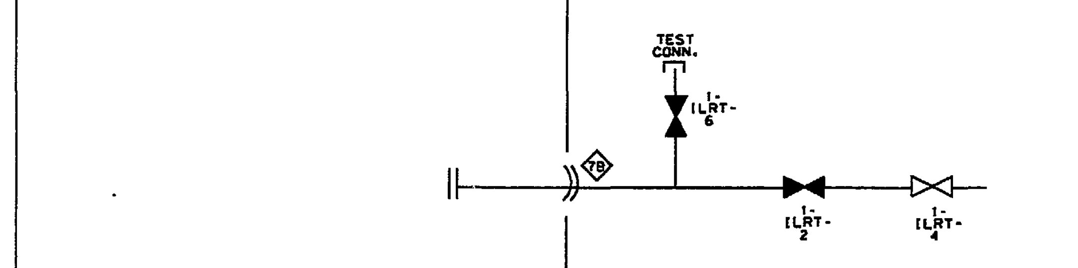
FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 06)

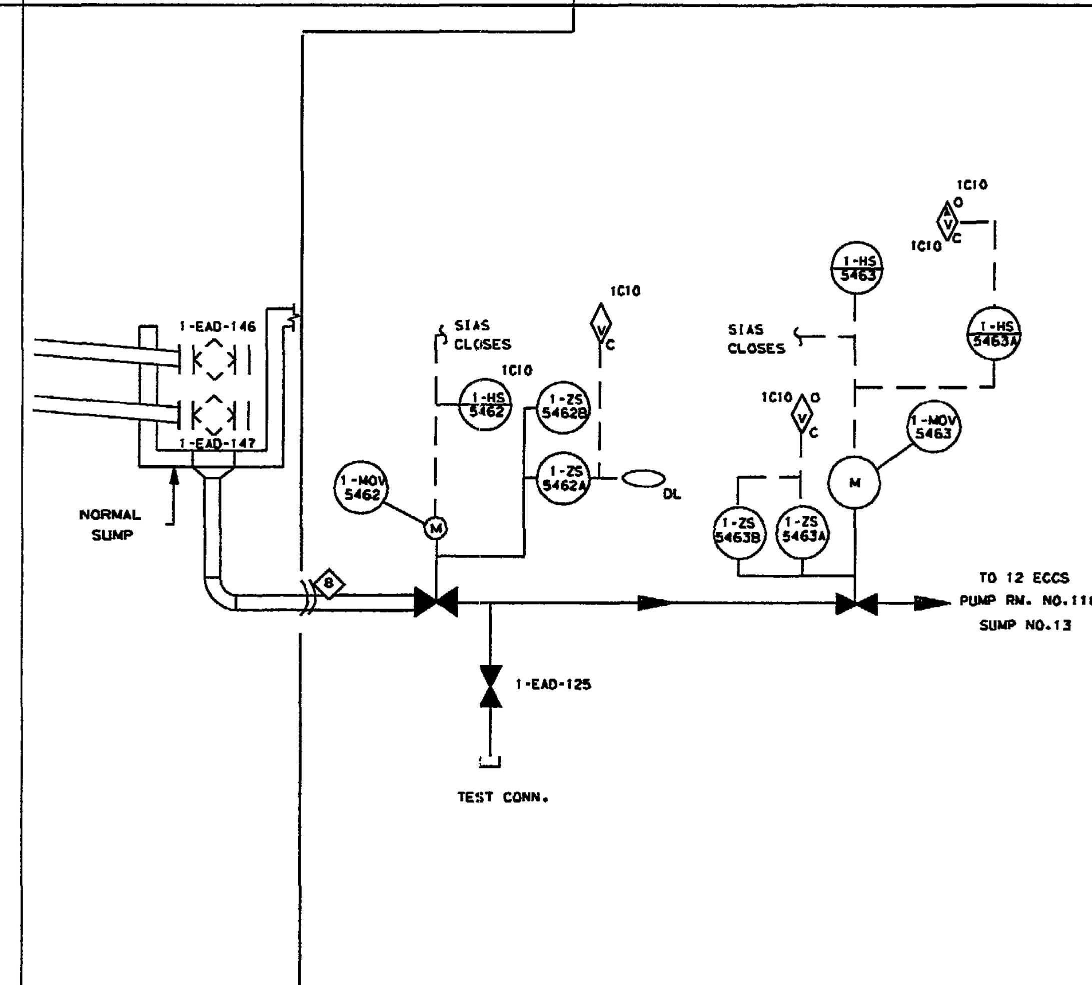
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
5	SAFETY INJECTION LOW & HIGH PRESSURE (SI TK.NO.12A DISCH)	8	I	10	WEST PIPING PENETRATION RM.	6"	OM-74 SH.1 OF 3 SH.2 OF 3	SI-138 MOV-635 MOV-636 MOV-637	- - - OPEN OPEN OPEN	
6	SAFETY INJECTION LOW & HIGH PRESSURE (SI TK. NO. 11A DISCH)	9	I	10	EAST PIPING PENETRATION RM.	6"	OM-74 SH.1 OF 3 SH.2 OF 3	SI-118 MOV-615 MOV-616 MOV-617	- - - OPEN OPEN OPEN	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 6	REVISION: 2

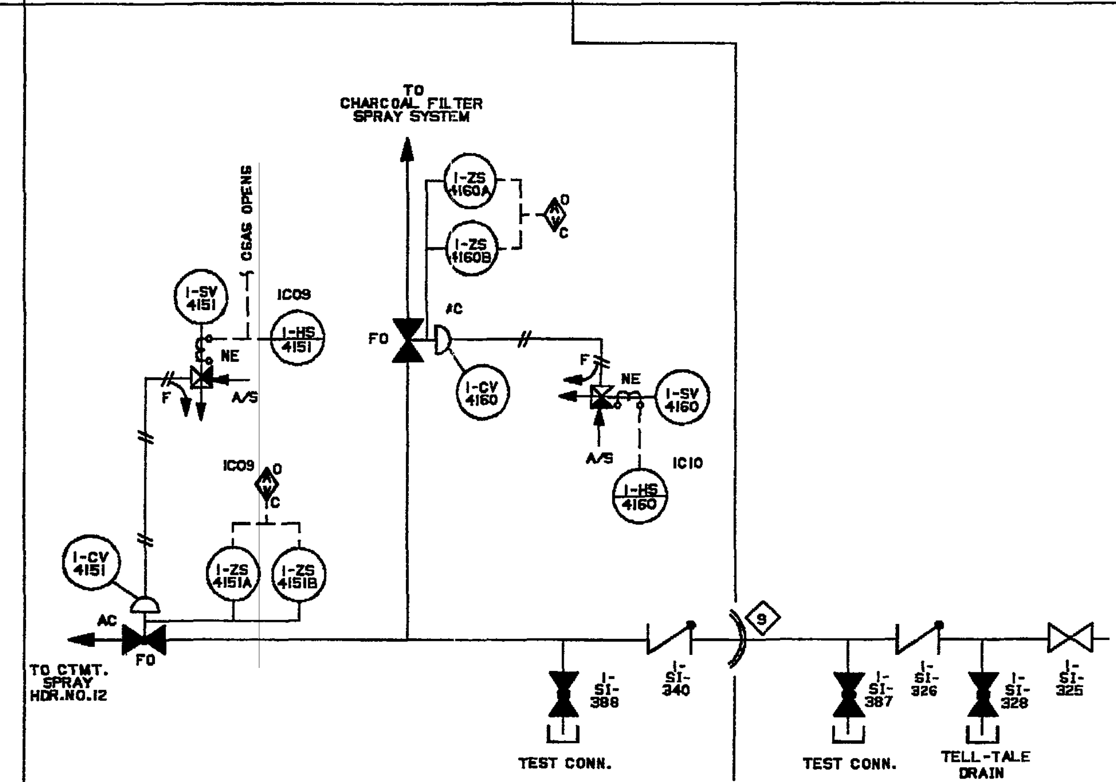
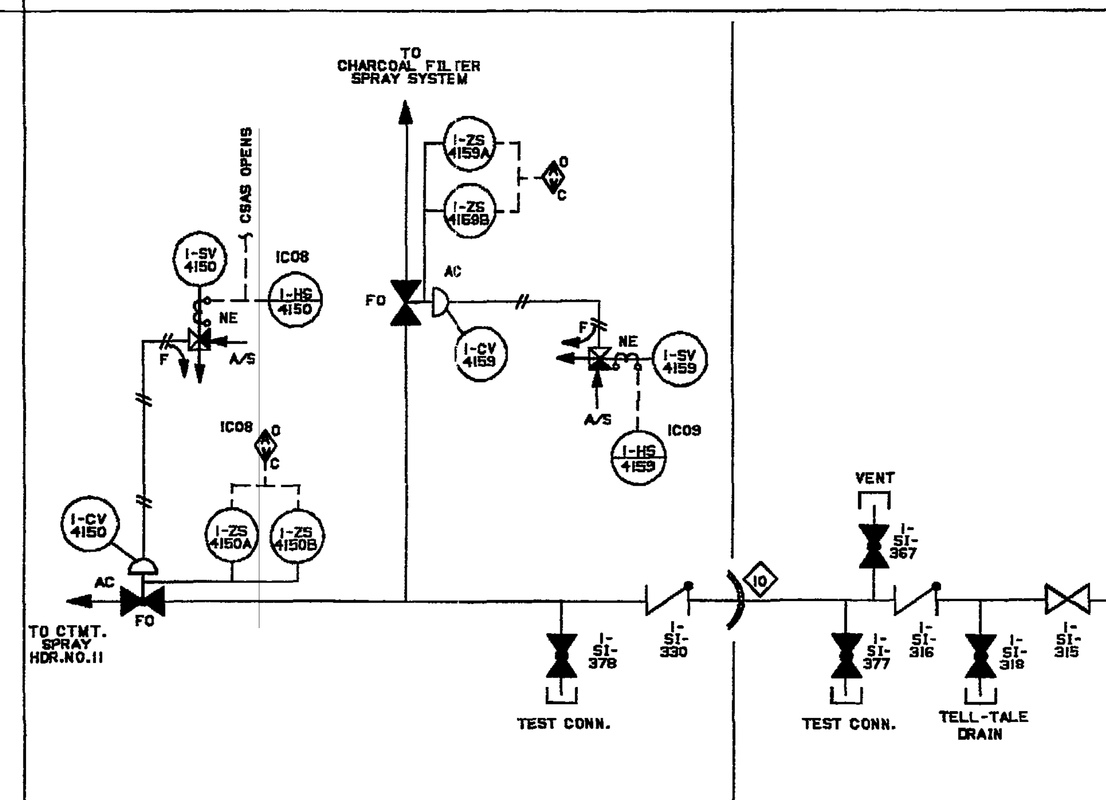
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
7A	ILRT PENETRATION	--	IV	45	WEST PIPING PENETRATION ROOM	3/4"	OM-065 SH0002	ILRT-1 BLIND FLANGE	CLOSED CLOSED	See diagram 7A below
7B	ILRT PENETRATION	--	IV	45	WEST PIPING PENETRATION ROOM	3/4"	OM-065 SH0002	ILRT-2 BLIND FLANGE	CLOSED CLOSED	See diagram 7B below
8	CONTAINMENT NORMAL SUMP TO MISC. WASTE DRAIN TK.	14	II	13	CONTAINMENT RECIRC. PIPE TUNNEL	4"	OM-076 SH0001 SH0004	MOV-5463 MOV-5462	CLOSED CLOSED	See diagram 8 below



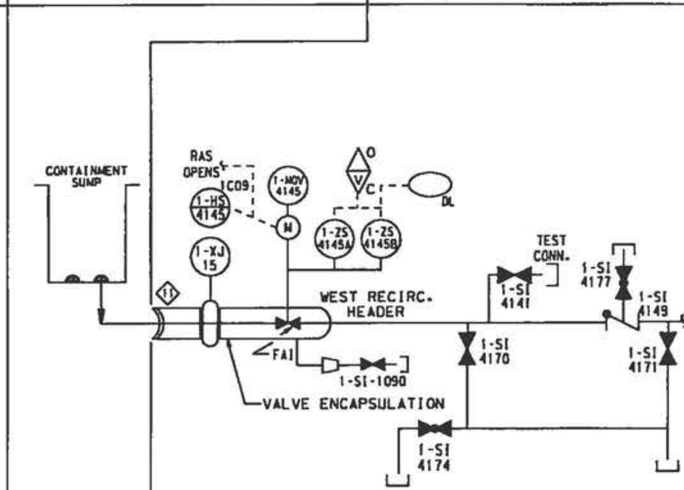
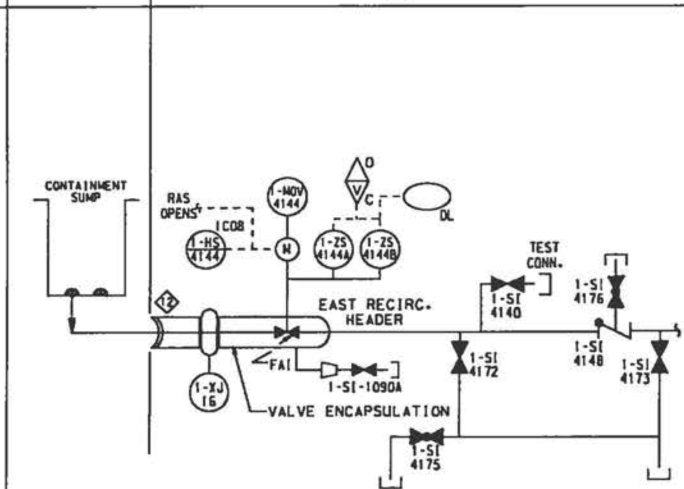




CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 7 | REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
9	CONTAINMENT SPRAY WATER (NO.12 CS HEADER)	4	II	17	WEST PIPING PENETRATION ROOM	8"	OM-74 SH.3 OF 3 OM-52	SI-326 SI-340	----- -----	
10	CONTAINMENT SPRAY WATER (NO.11 CS HEADER)	5	II	17	EAST PIPING PENETRATION ROOM	8"	OM-74 SH.3 OF 3 OM-52	SI-316 SI-330	----- -----	

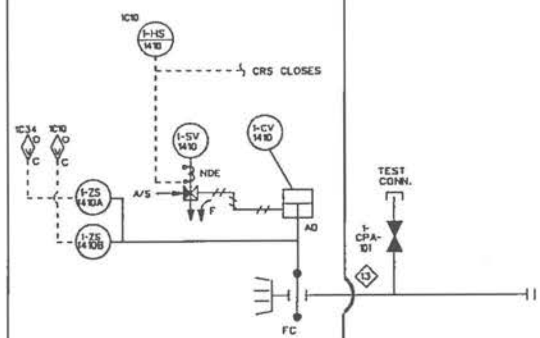
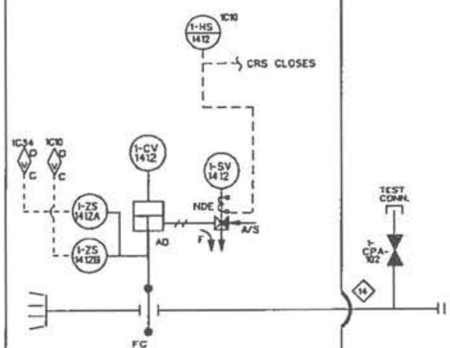
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 8 | REV: 15

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE	DIAGRAM: OUTSIDE CTMT. STRUCTURE
11	CONTAINMENT STRUCTURE SUMP WEST RECIRC.HEADER	12	11	16	CONTAINMENT RECIRC.PIPE TUNNEL	24"	OM-074 SH0003	MOV-4145	OPEN		
12	CONTAINMENT STRUCTURE SUMP EAST RECIRC.HEADER	13	11	36	CONTAINMENT RECIRC.PIPE TUNNEL	24"	OM-074 SH0003	MOV-4144	OPEN		

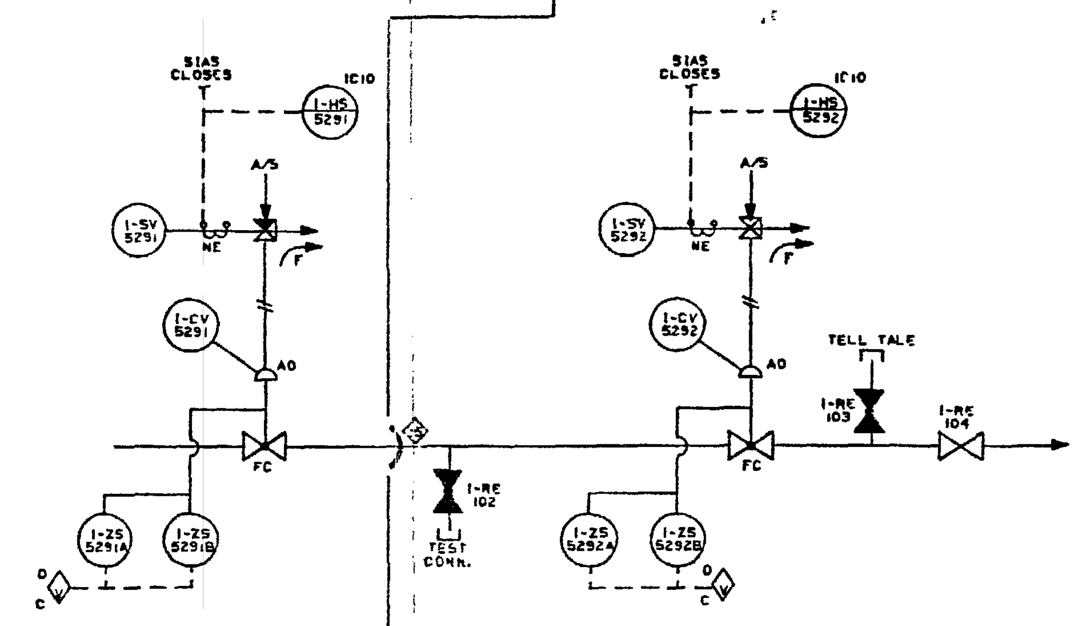
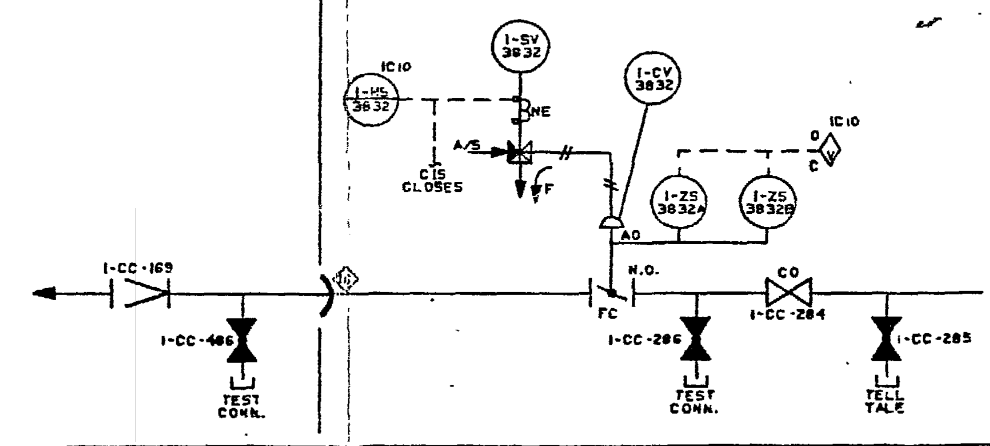
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 9 | REV:32 | DATE:11/12/02

80073.DGN FSAR 5-10SH0009

FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 10)

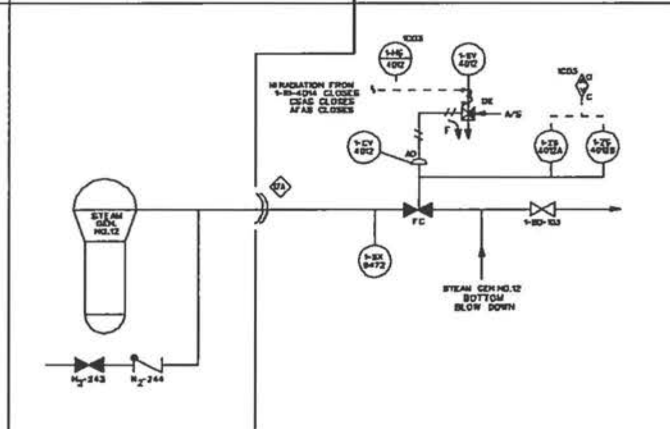
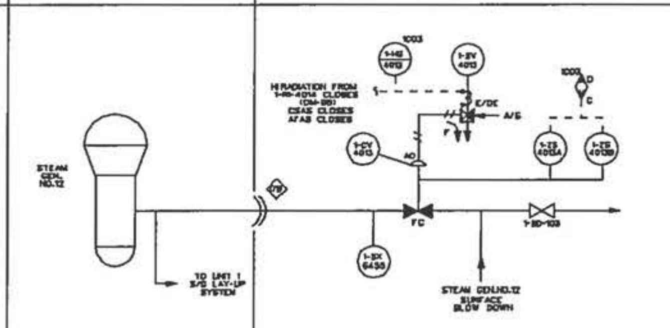
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM INSIDE CTMT. STRUCTURE	DIAGRAM OUTSIDE CTMT. STRUCTURE
13	PURGE AIR INLET	40	IV		EAST PIPING PENETRATION ROOM	48"	OM-065 SH0001	BLIND FLANGE	INSTALLED		
14	PURGE AIR OUTLET	41	IV		EAST PIPING PENETRATION ROOM	48"	OM-065 SH0001	BLIND FLANGE	INSTALLED		

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 10 | REVISION: 35

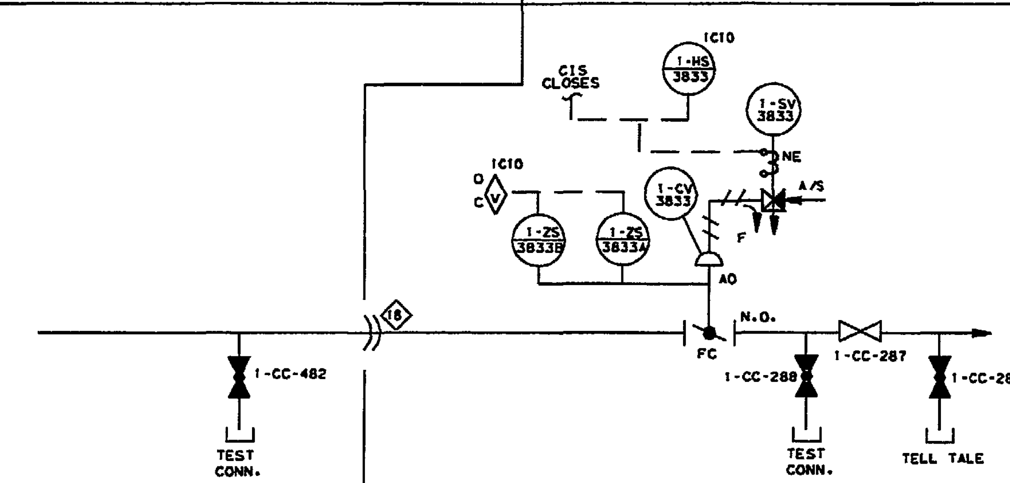
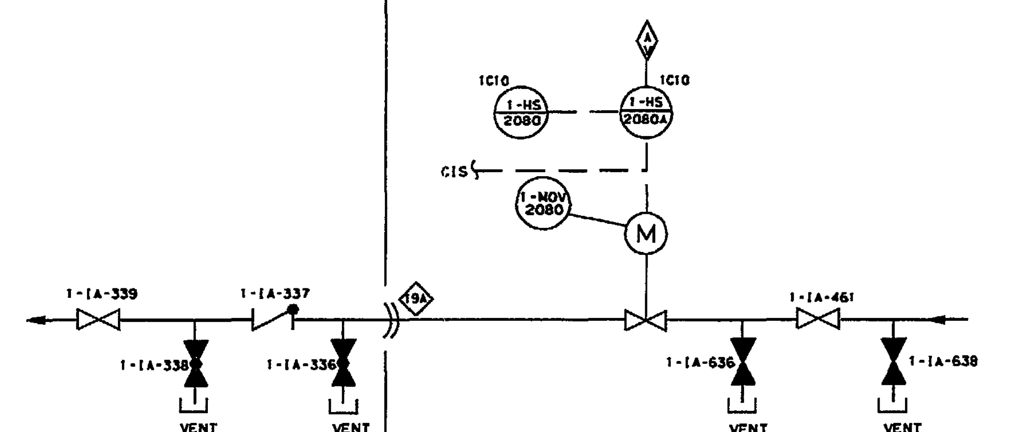
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
15	CONTAINMENT ATMOS. AND PURGE AIR MONITOR	42	II	5	EAST PIPING PENETRATION ROOM	1"	OM-98 SH.1 OF 2	CV-5292 CV-5291	CLOSED CLOSED	
16	COMPONENT COOLING WATER INLET TO REACTOR COOLANT PUMPS	24	III	25	EAST PIPING PENETRATION ROOM	10"	OM-51 SH.2 OF 3	CV-3832	CLOSED	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.11

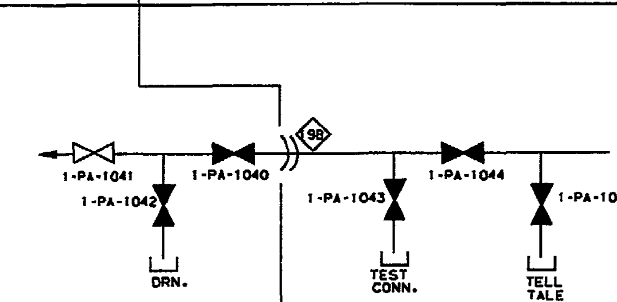
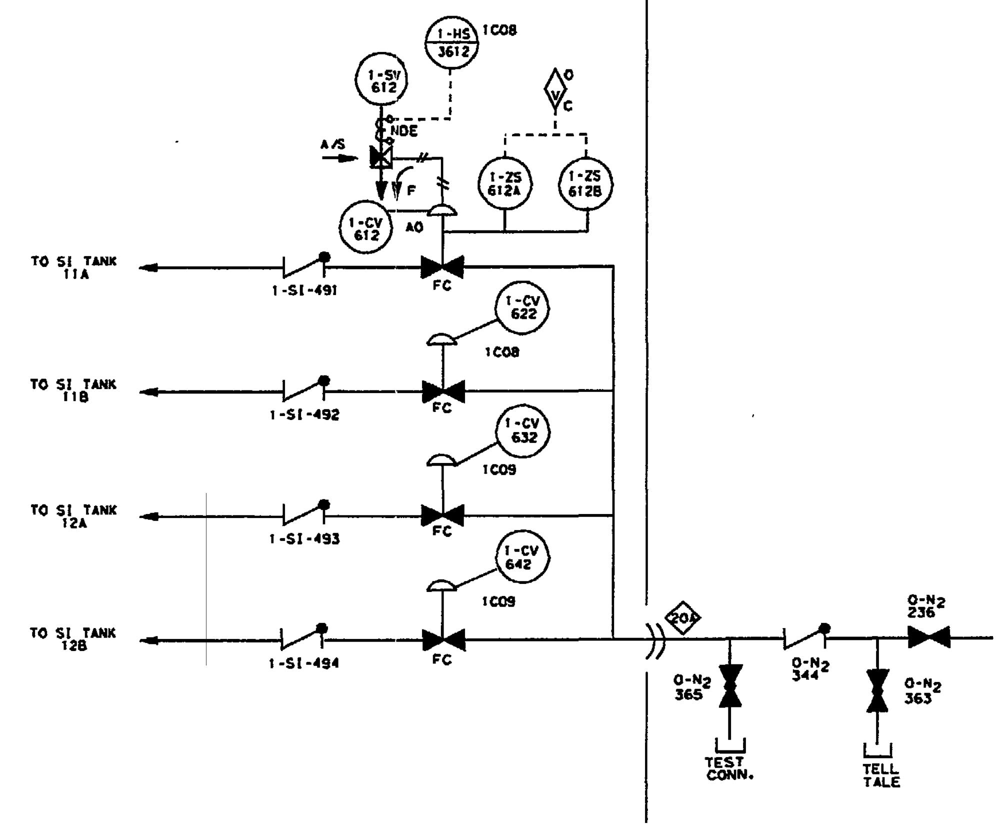
FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 12)

<table>
<tr><th rowspan="2">PENE. NO.</th><th rowspan="2">SERVICE</th><th rowspan="2">ORIG. FSAR PENE. NO.</th><th rowspan="2">PENE. TYPE</th><th rowspan="2">VALVE ARRGT.</th><th rowspan="2">PENETRATION LOCATION</th><th rowspan="2">PENE. LINE SIZE</th><th rowspan="2">DWG. NUMBER</th><th rowspan="2">ISOLATION VALVE(S)</th><th rowspan="2">POST-INCIDENT POSITION</th><th colspan="2">DIAGRAM</th></tr>
<tr><th>INSIDE CTMT. STRUCTURE</th><th>OUTSIDE CTMT. STRUCTURE</th></tr>
<tr><td>17A</td><td>STEAM GEN. NO.12.
SURFACE BLOWDOWN</td><td>22</td><td>III</td><td>14</td><td>EAST PIPING PENETRATION ROOM</td><td>2"</td><td>OM-35
SH.1 OF 3
OM-464</td><td>CV-4012</td><td>CLOSED</td><td colspan="2">
</td></tr>
<tr><td>17B</td><td>STEAM GEN. NO.12.
BOTTOM BLOWDOWN</td><td>23</td><td>III</td><td>14</td><td>EAST PIPING PENETRATION ROOM</td><td>2"</td><td>OM-35
SH.1 OF 3
OM-464</td><td>CV-4013</td><td>CLOSED</td><td colspan="2">
</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td colspan="2"></td></tr>
</table>

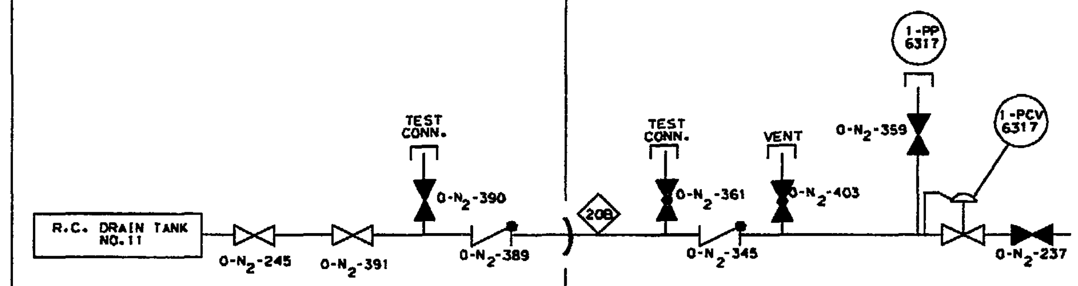
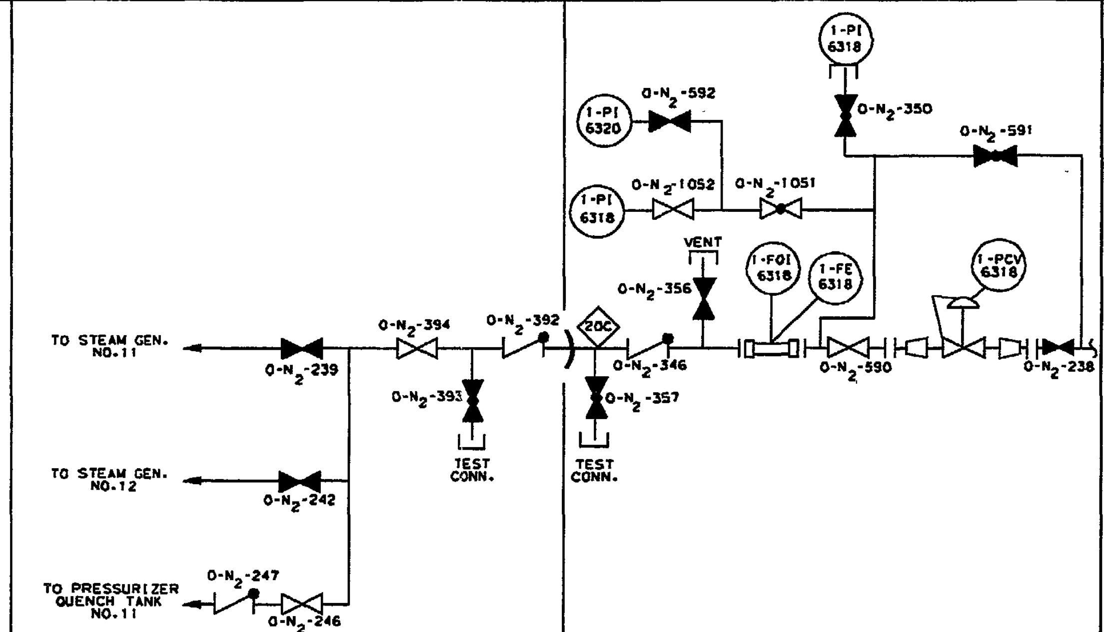
22571.DGN	FSAR-12	CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 12	REVISION: 42

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
18	COMPONENT COOLING WATER OUTLET FROM REACTOR COOLANT PUMPS	29	III	2	EAST PIPING PENETRATION ROOM	10"	OM-051 SH0002	CV-3833	CLOSED	
19A	INSTRUMENT AIR	38	III	31	EAST PIPING PENETRATION ROOM	2"	OM-053 SH0001 SH0003	MOV-2080 IA-337	CLOSED ----	

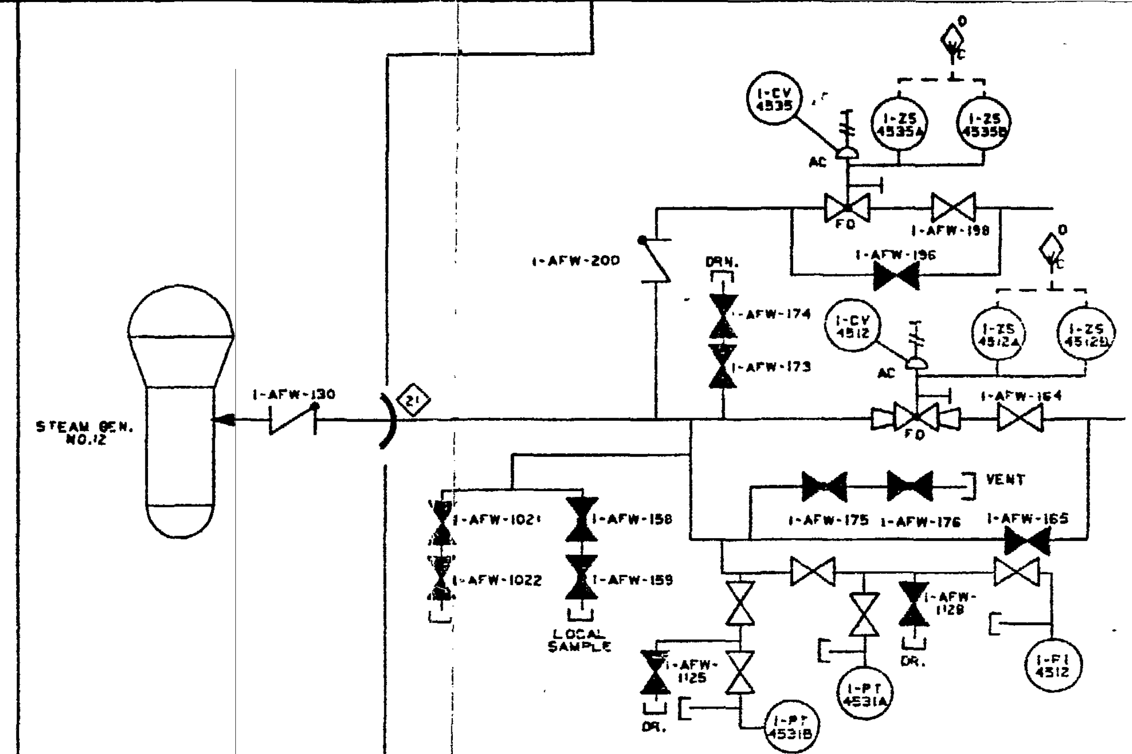
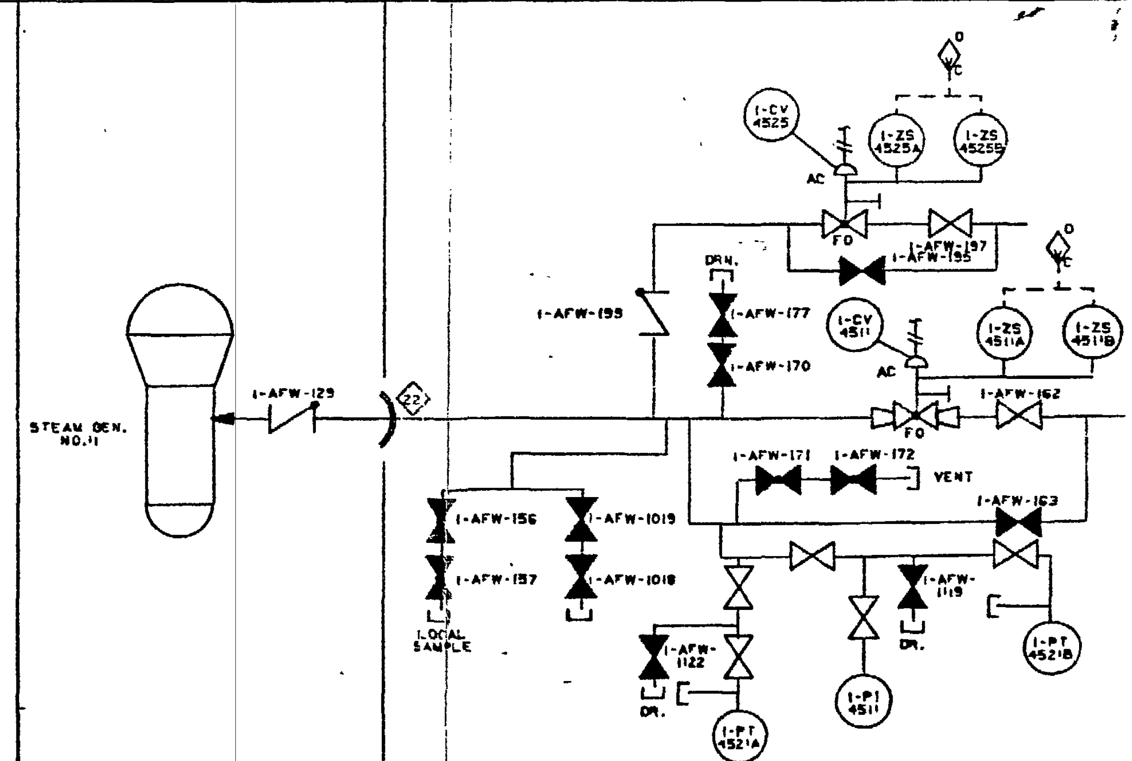
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO.13	REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
19B	PLANT AIR	39	IV	3	EAST PIPING PENETRATION ROOM	2"	OM-479 SH0002	PA-1044 PA-1040	LOCKED SHUT LOCKED SHUT	
20A	N_2 SUPPLY TO SAFETY INJECTION TANKS 11A,11B,12A & 12B	34	III	38	EAST PIPING PENETRATION ROOM	1"	OM-068 OM-074 SH0002	N_2-344 CV-612 CV-622 CV-632 CV-642	-- CLOSED CLOSED CLOSED CLOSED	

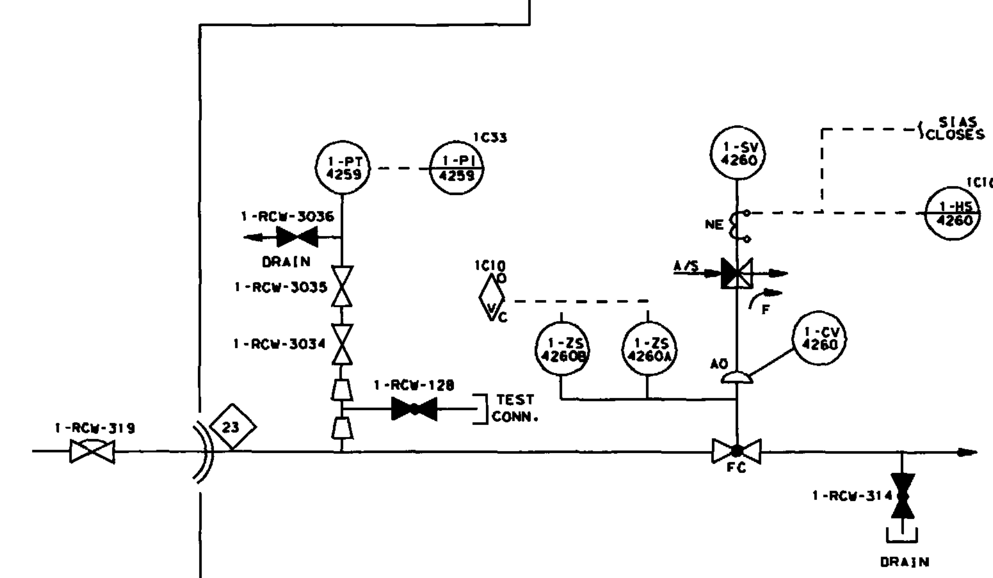
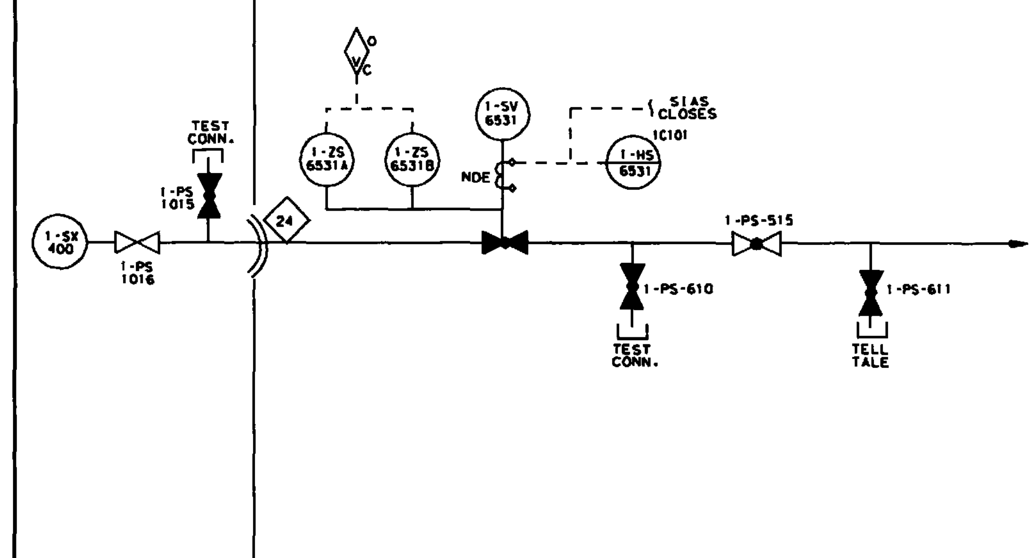
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.14 | REVISION: 21

	PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE	DIAGRAM: OUTSIDE CTMT. STRUCTURE
	20B	NITROGEN SUPPLY TO RC DRAIN TANK	34	III	18	EAST PIPING PENETRATION ROOM	1"	OM-068 OM-077 SH0001	N_2-345 N_2-389	-- --		
	20C	NITROGEN SUPPLY TO STEAM GENERATORS NO.11 & 12 AND TO PRESS. QUENCH TANK	34	III	39	EAST PIPING PENETRATION ROOM	1"	OM-068 OM-035 SH0001 OM-072 SH0001	N_2-346 N_2-392	-- --		

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.15 | REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
21	AUX FEEDWATER TO STEAM GENERATOR NO.12	17	III	15	EAST PIPING PENETRATION ROOM	4"	OM-800	AFW-200 CV-4512 AFW-165	-- OPEN LOCKED SHUT	
22	AUX FEEDWATER TO STEAM GENERATOR NO.11	18	III	15	EAST PIPING PENETRATION ROOM	4"	OM-800	AFW-199 CV-4511 AFW-163	-- OPEN LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.16

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
23	DRAINS FROM REACTOR COOLANT SYSTEM DRAIN TANK	35	III	24	WEST PIPING PENETRATION ROOM	2"	OM-077 SH0001	CV-4260	CLOSED	
24	OXYGEN SAMPLE LINE (FROM PRZR. QUENCH TANK NO.11)	—	III	37	WEST PIPING PENETRATION ROOM	¼"	OM-463 SH0001	SV-6531	LOCKED SHUT	

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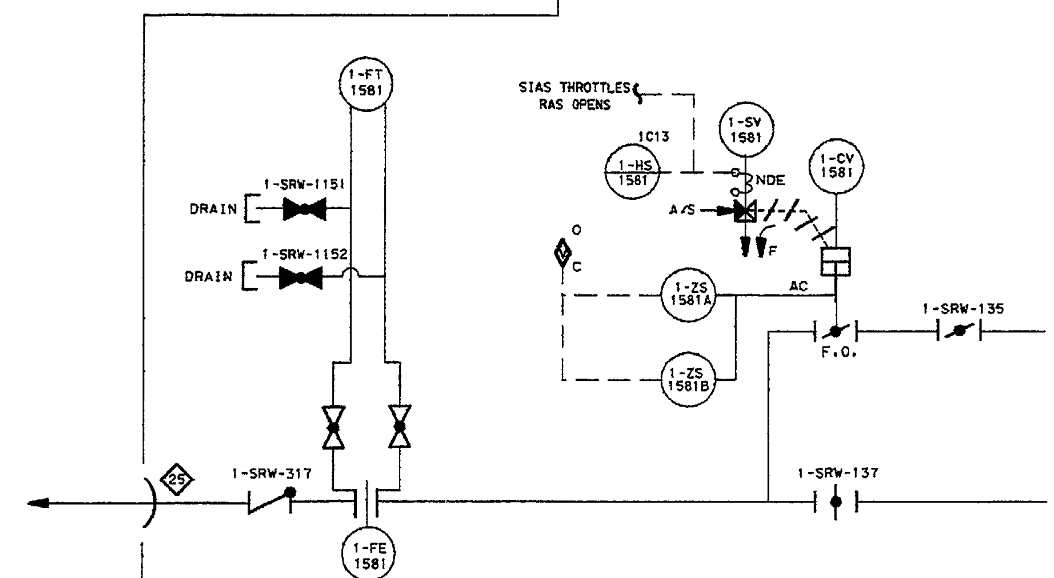
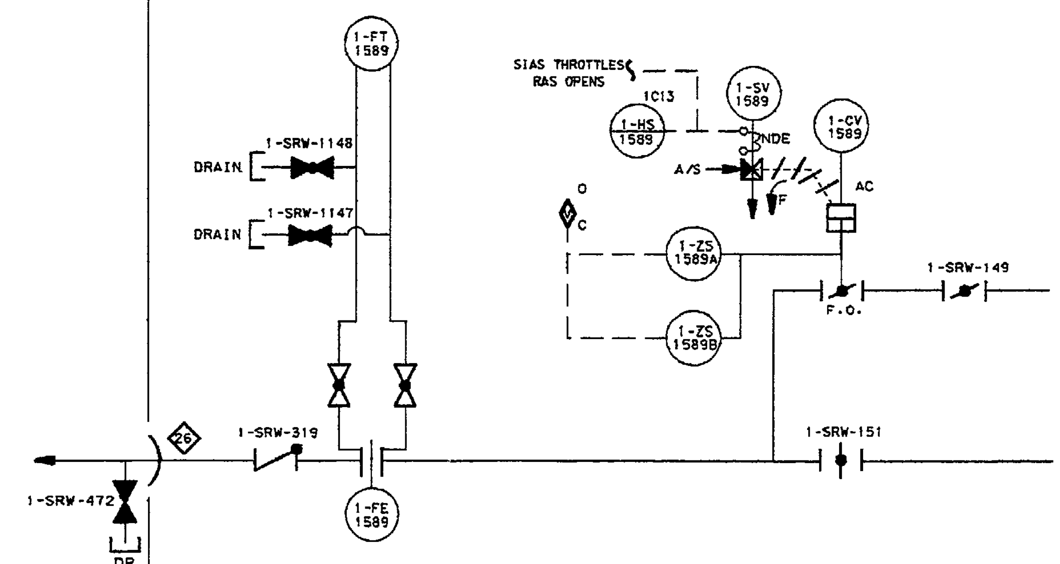
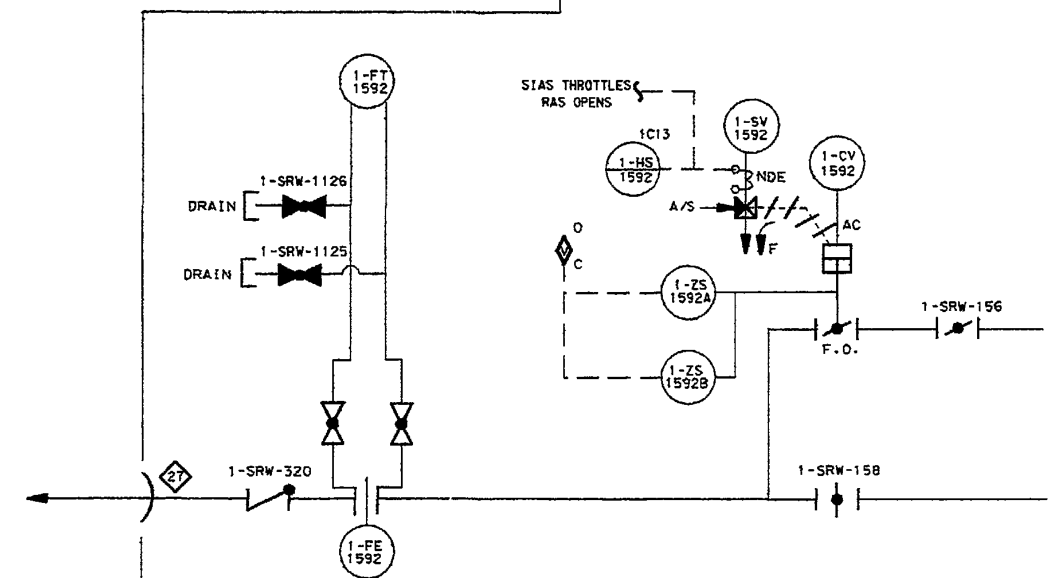
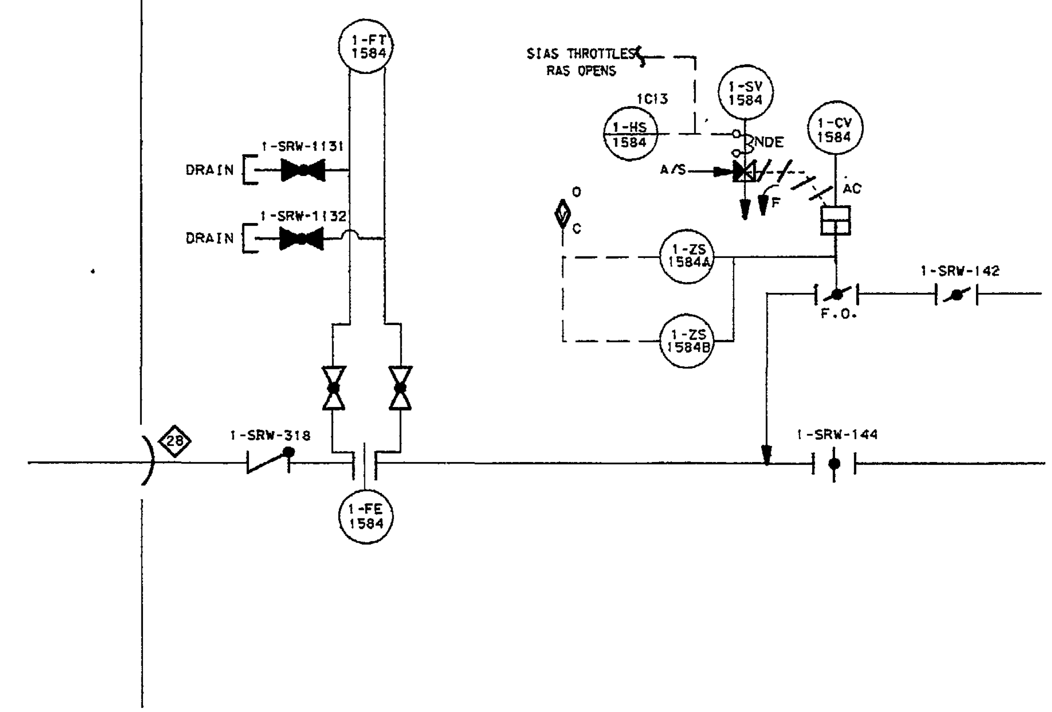
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.17 | REVISION: 26

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
25	SERVICE WATER INLET TO CONTAINMENT COOLING UNIT NO.11	25	III	26	EAST PIPING PENETRATION ROOM	8"	OM-46 SH.2 OF 2	SRW-317	--	
26	SERVICE WATER INLET TO CONTAINMENT COOLING UNIT NO.13	26	III	26	EAST PIPING PENETRATION ROOM	8"	OM-46 SH.2 OF 2	SRW-319	--	

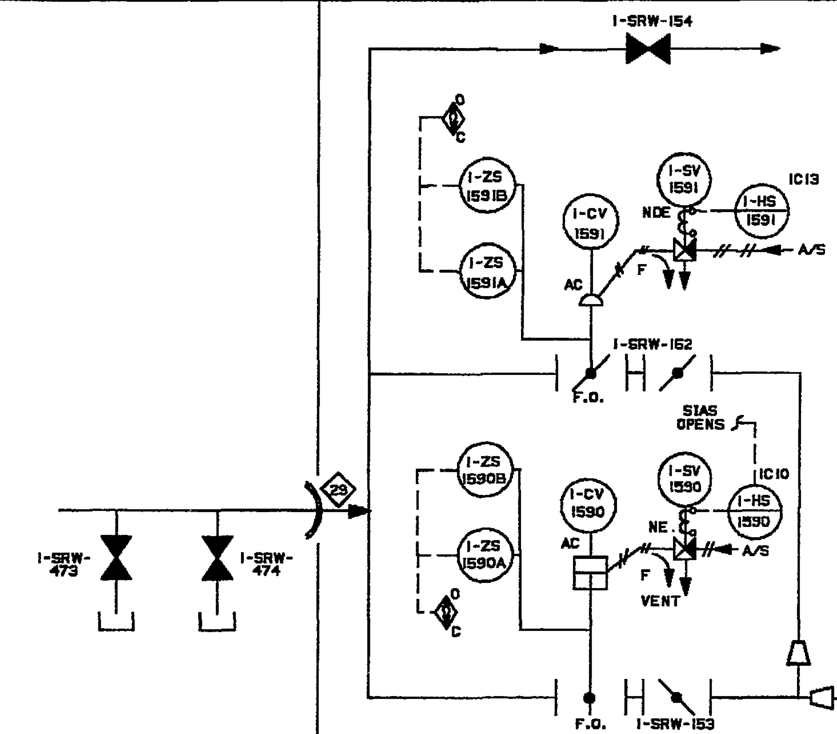
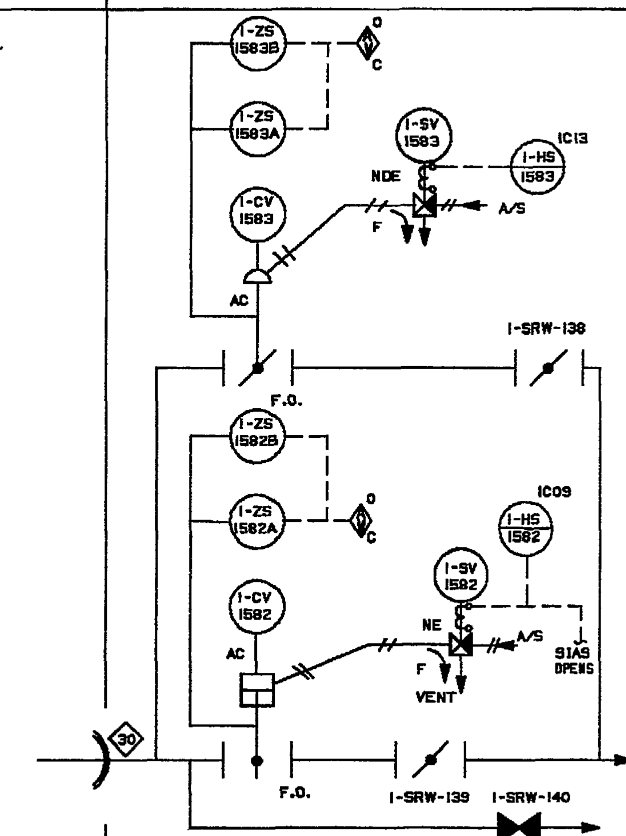
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO.18	REVISION: 17

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM	
										INSIDE CTMT. STRUCTURE	OUTSIDE CTMT. STRUCTURE
27	SERVICE WATER INLET TO CONTAINMENT COOLING UNIT NO.14	27	III	26	WEST PIPING PENETRATION ROOM	8"	OM-46 SH.2 OF 2	SRW-320	--		
28	SERVICE WATER INLET TO CONTAINMENT COOLING UNIT NO.12	28	III	26	WEST PIPING PENETRATION ROOM	8"	OM-46 SH.2 OF 2	SRW-318	--		

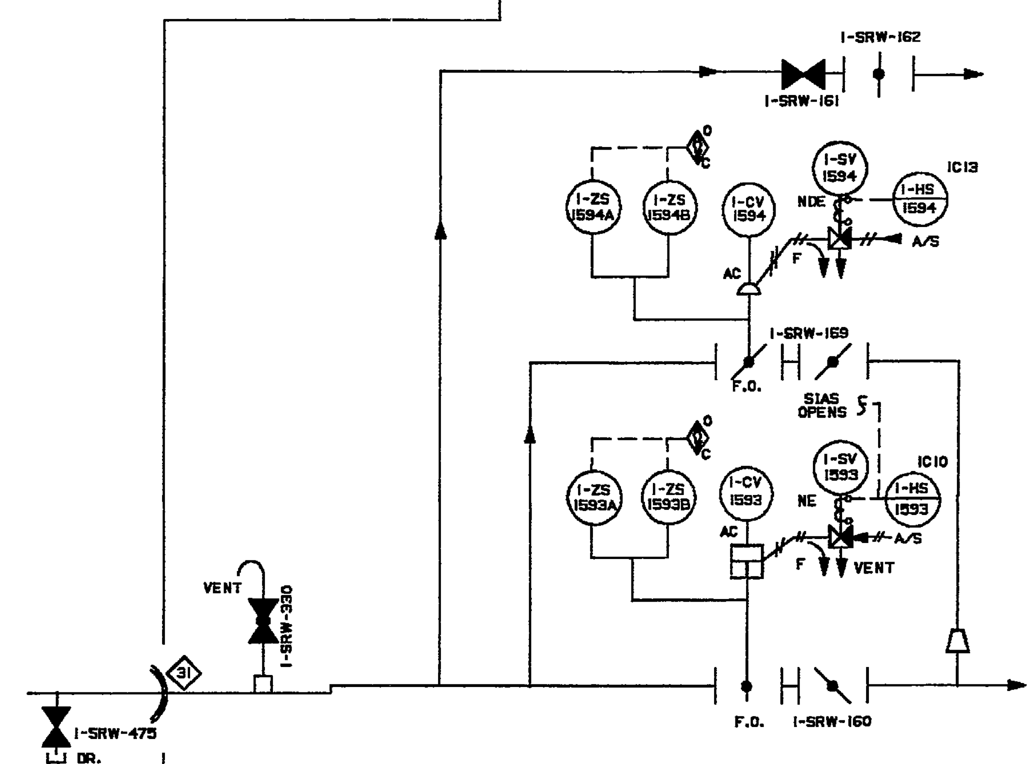
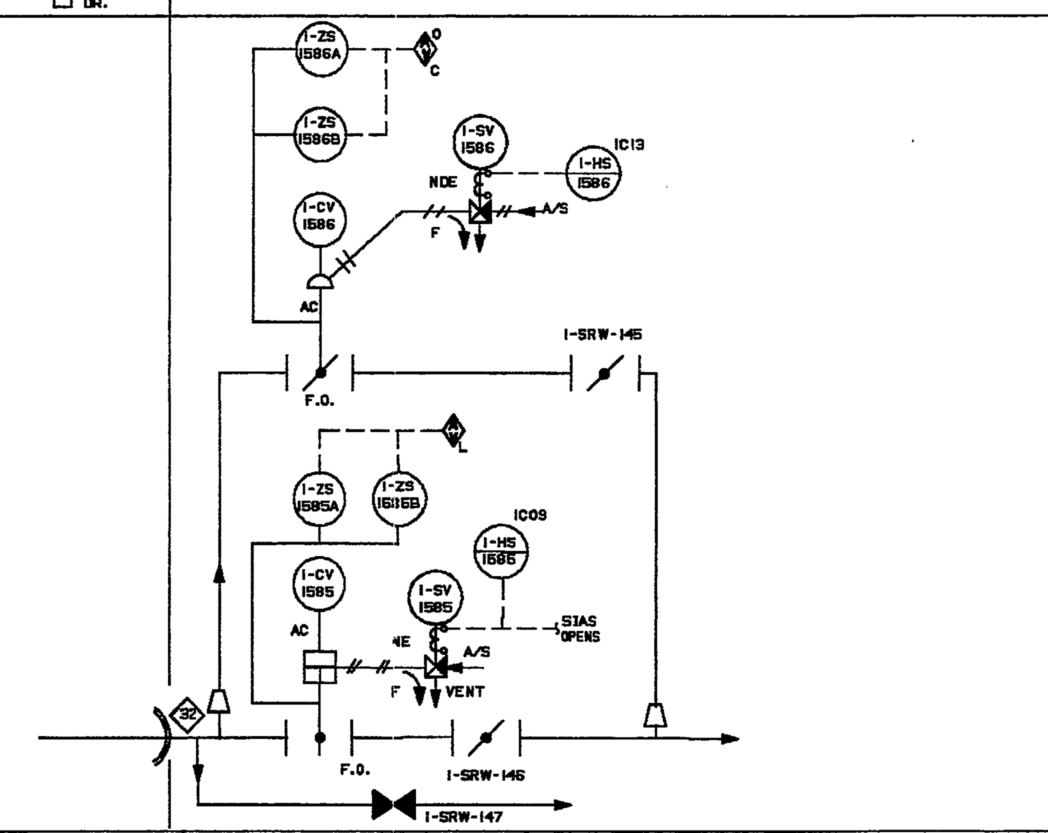




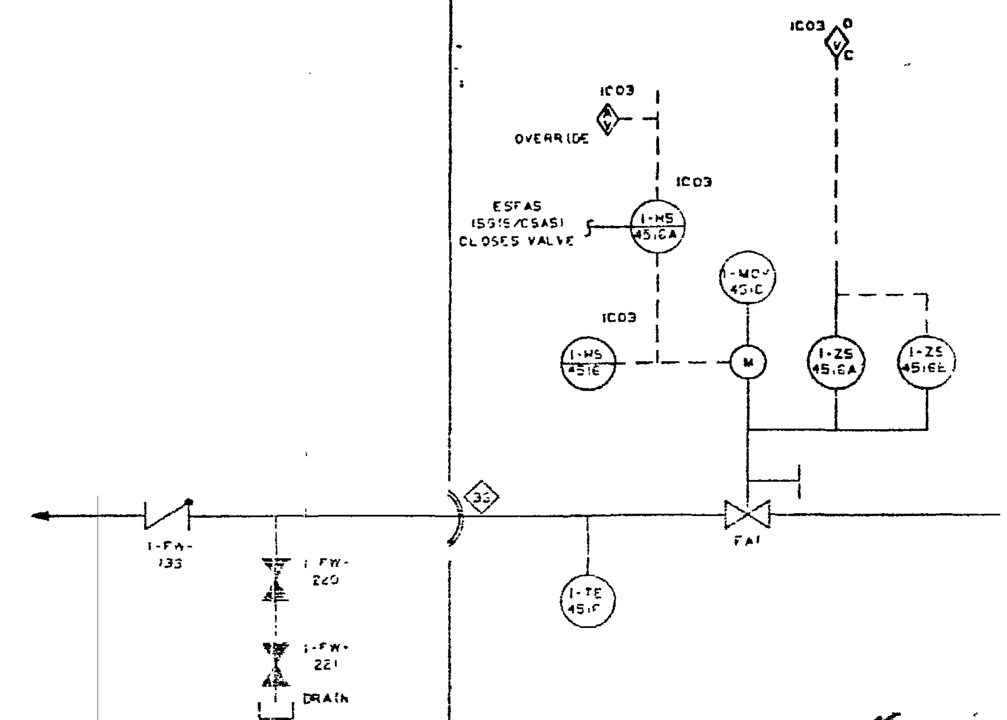
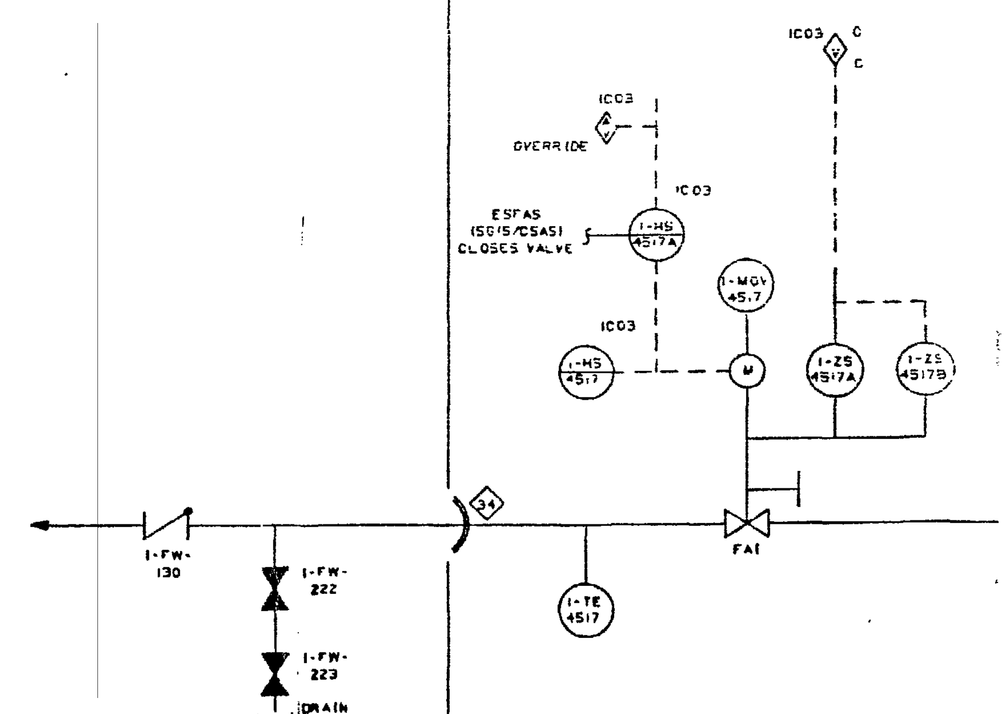
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.19 | REVISION: 17

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
29	SERVICE WATER RETURN CONTAINMENT COOLING UNIT#13	30	III	22	EAST PIPING PENETRATION RM.	8"	OM-46 SH.2 OF 2	CV-1590 CV-1591 SRW-154	OPEN OPEN LOCKED SHUT	
30	SERVICE WATER RETURN CONTAINMENT COOLING UNIT#11	31	III	22	EAST PIPING PENETRATION RM.	8"	OM-46 SH.2 OF 2	CV-1582 CV-1583 SRW-140	OPEN OPEN LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 20 | REV: 15

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
31	SERVICE WATER RETURN CONTAINMENT COOLING UNIT#14	32	III	22	WEST PIPING PENETRATION RM.	8"	OM-46 SH.2 OF 2	CV-1594 CV-1593 SRW-161	OPEN OPEN LOCKED SHUT	
32	SERVICE WATER RETURN CONTAINMENT COOLING UNIT#12	33	III	22	WEST PIPING PENETRATION RM.	8"	OM-46 SH.2 OF 2	CV-1586 CV-1585 SRW-147	OPEN OPEN LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 21	REV: 15

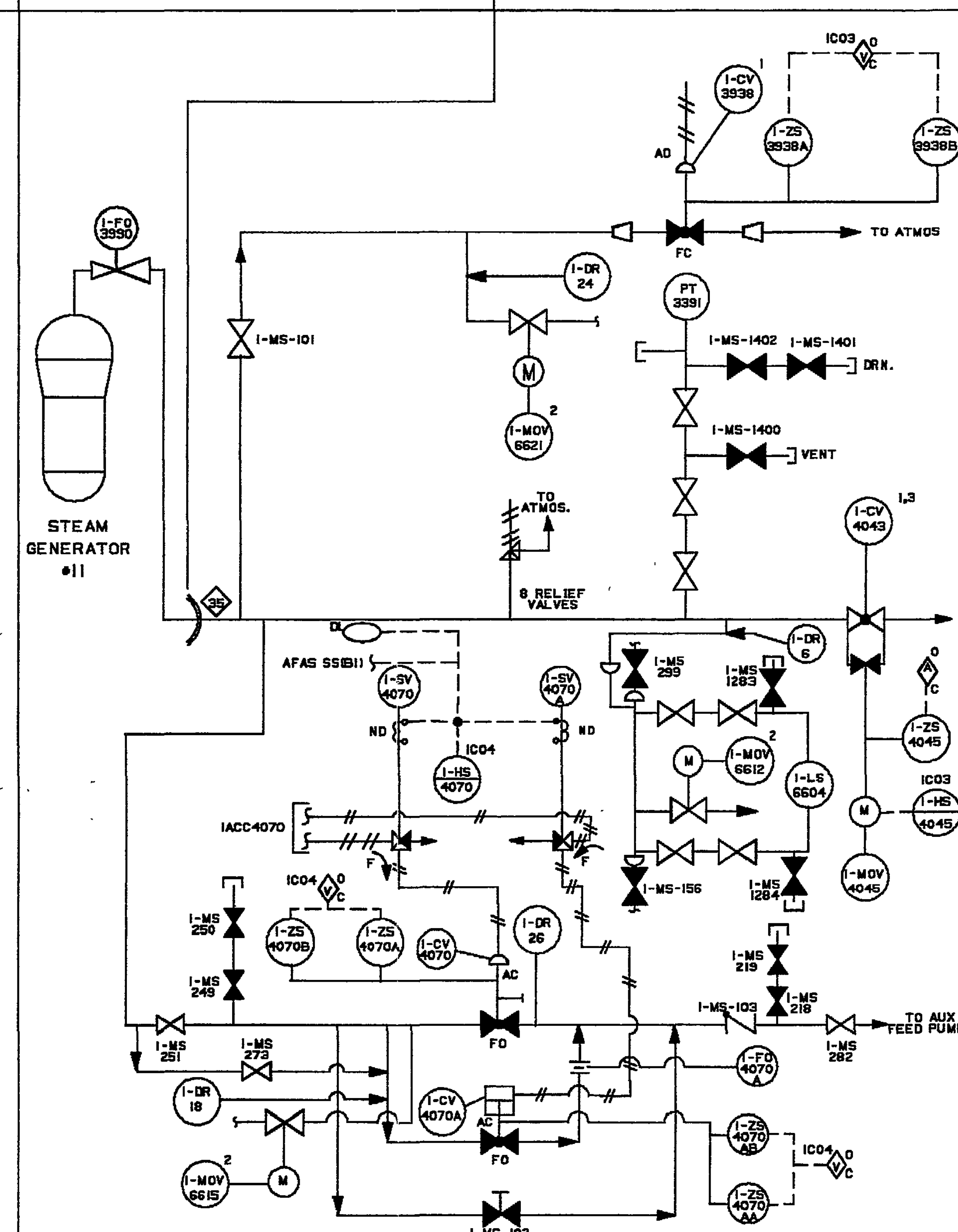
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM INSIDE CTMT. STRUCTURE	DIAGRAM OUTSIDE CTMT. STRUCTURE
33	MAIN FEEDWATER TO #11 STEAM GENERATOR	15	III	32	MAIN STEAM PENETRATION RM.	16"	OM-39 SH.4 OF 4	MOV-4516	CLOSED		
34	MAIN FEEDWATER TO #12 STEAM GENERATOR	15	III	32	MAIN STEAM PENETRATION RM.	16"	OM-39 SH.4 OF 4	MOV-4517	CLOSED		

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 22

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION
35	MAIN STEAM FROM #11 STEAM GENERATOR	19	III	23	MAIN STEAM PENETRATION ROOM	34"	OM-35 SH.1 OF 3	CV-3938	CLOSED
							OM-114 SH.1 OF 2	MOV-6621	CLOSED
								CV-4043	CLOSED
							OM-800	MOV-4045	CLOSED
								MOV-6612	CLOSED
								CV-4070	OPEN/ CLOSED
								MS-102	CLOSED
								MOV-6615	CLOSED
								CV-4070A	OPEN/ CLOSED

DIAGRAM

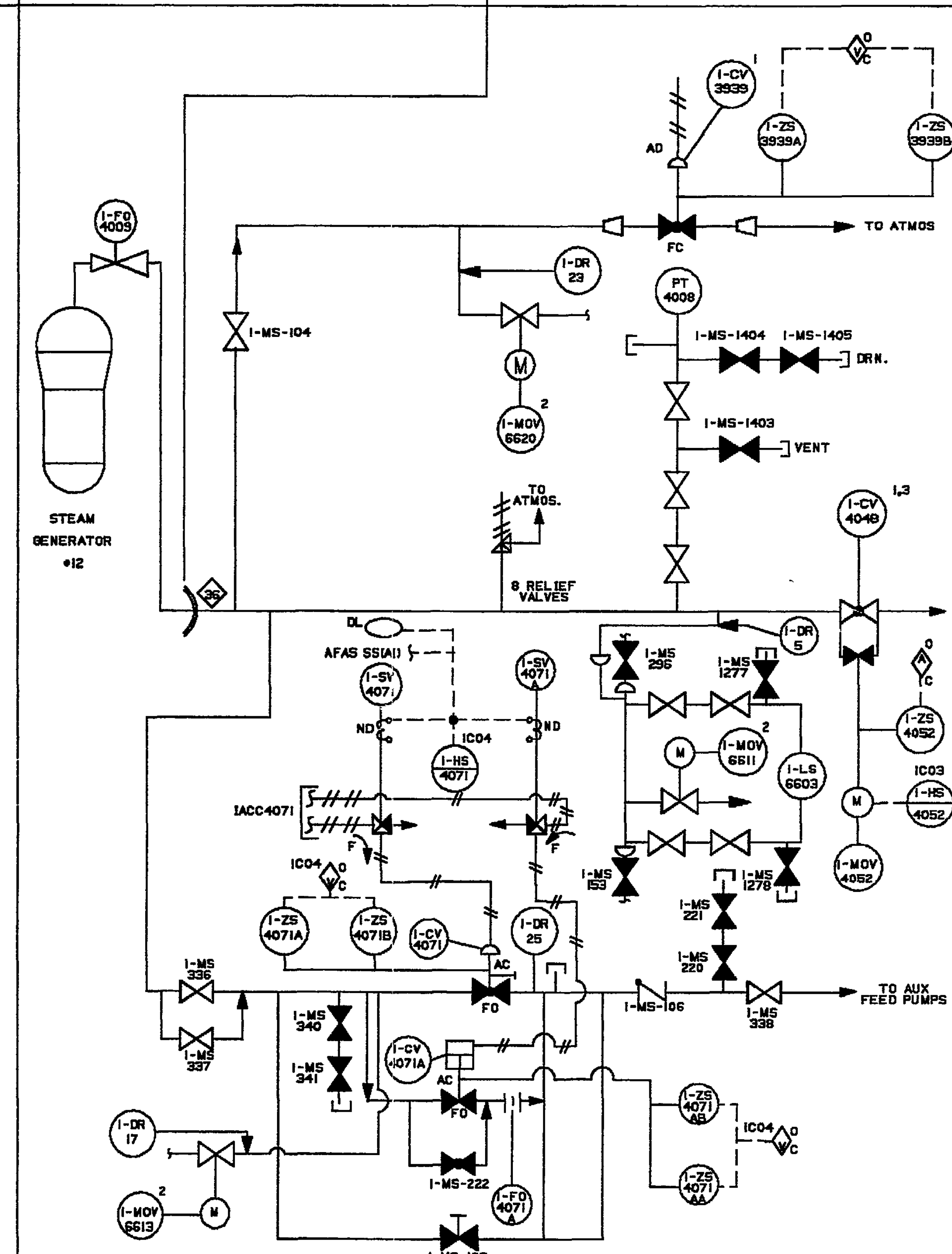
INSIDE CTMT. STRUCTURE | OUTSIDE CTMT. STRUCTURE



NOTES:

1. FOR CONTROL DETAILS, SEE DWG. OM-35, SH.1 OF 3.
2. FOR CONTROL DETAILS, SEE DWG. OM-114, SH.1 OF 2.
3. FOR CONTROL DETAILS, SEE DWG. OM-480.

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 23	REV: 15

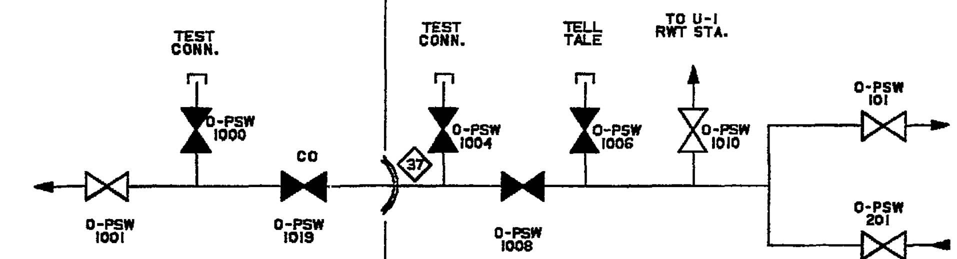
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM
36	MAIN STEAM FROM #12 STEAM GENERATOR	20	III	23	MAIN STEAM PENETRATION ROOM	34"	OM-35 SH.1 OF 3	CV-3939	CLOSED	
							OM-114 SH.1 OF 2	MOV-6620	CLOSED	
							OM-800	CV-4048	CLOSED	
								MOV-4052	CLOSED	
								MOV-6611	CLOSED	
								CV-4071	OPEN/ CLOSED	
								MS-222	CLOSED	
								MS-105	CLOSED	
								MOV-6613	CLOSED	
								CV-4071A	OPEN/ CLOSED	

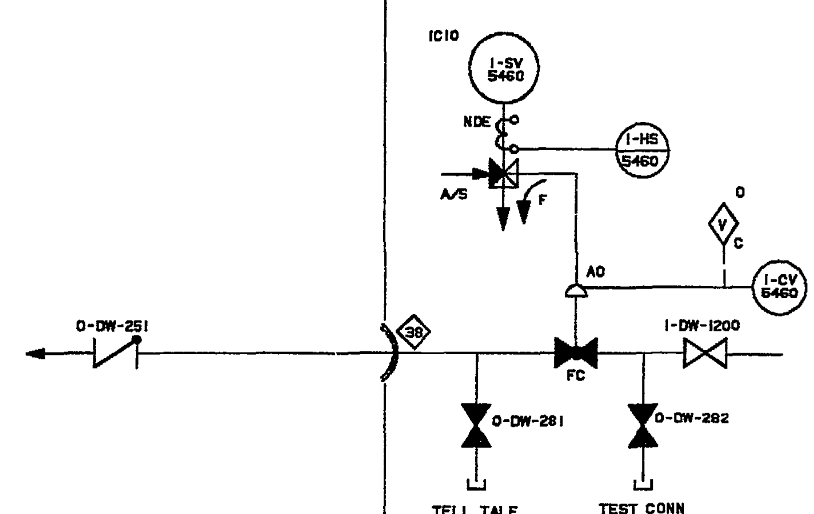
NOTES:

1. FOR CONTROL DETAILS, SEE DWG. OM-35, SH.1 OF 3.
2. FOR CONTROL DETAILS, SEE DWG. OM-114, SH.1 OF 2.
3. FOR CONTROL DETAILS, SEE DWG. OM-480.

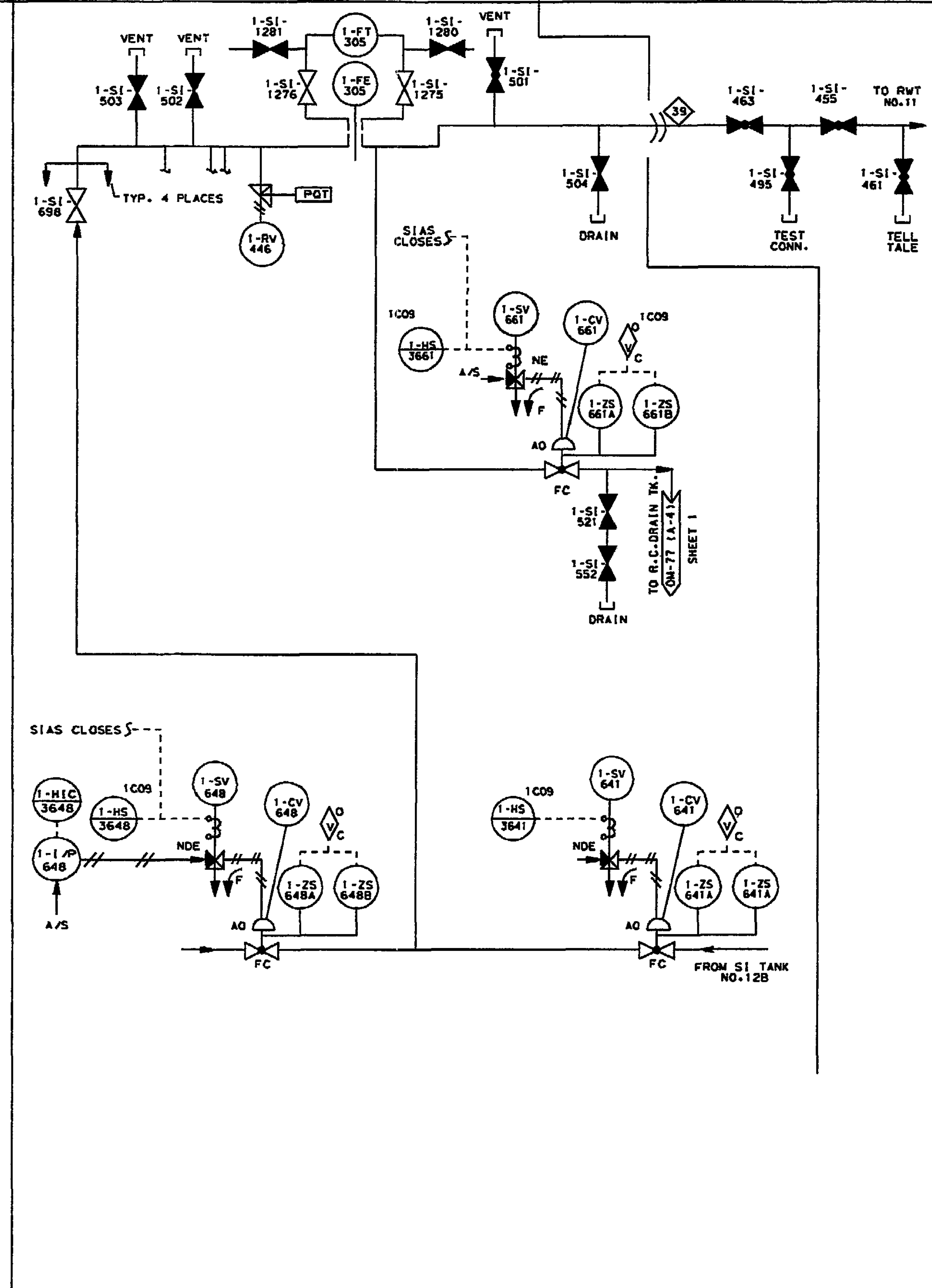
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 24	REV: 15

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM INSIDE CTMT. STRUCTURE	DIAGRAM OUTSIDE CTMT. STRUCTURE
37	PLANT SERVICE WATER	--	II	43	WEST PIPING PENETRATION ROOM	3"	OM-479 SH.1 OF 2 2 OF 2	PSW-1008 PSW-1019	CLOSED CLOSED		
38	DEMINERALIZED WATER TO QUENCH TANK	I	III	I	EAST PIPING PENETRATION ROOM	2"	OM-72 SH.1 OF 2	CV-5460	CLOSED		





CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 25	REV: 15

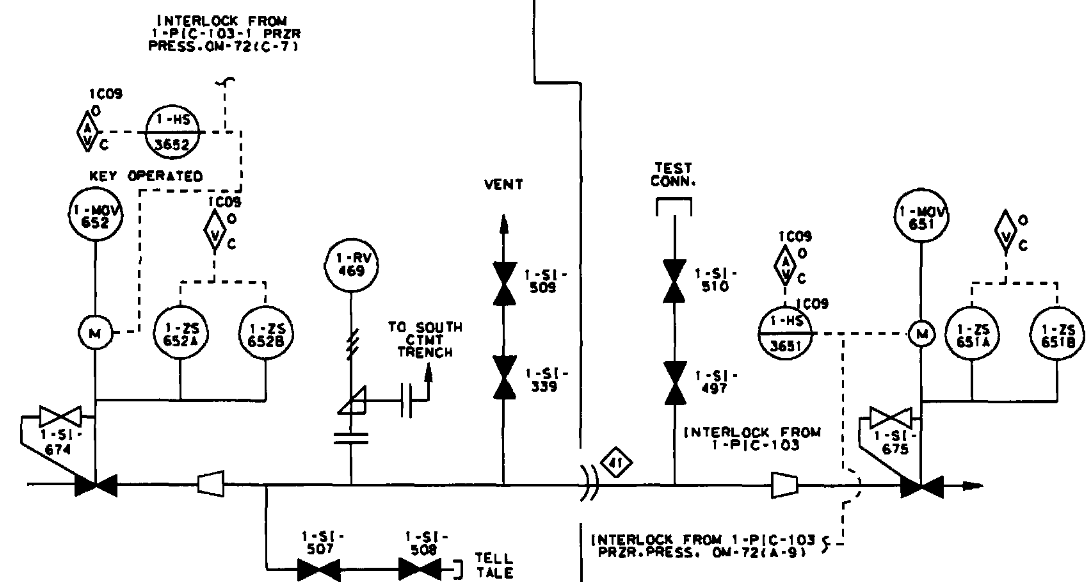
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE	DIAGRAM: OUTSIDE CTMT. STRUCTURE
39	SAFETY INJECTION TANK TEST LINE	11	IV	20	WEST PIPING PENETRATION ROOM	2"	OM-074 SH0002	SI-463 SI-455	LOCKED SHUT LOCKED SHUT		

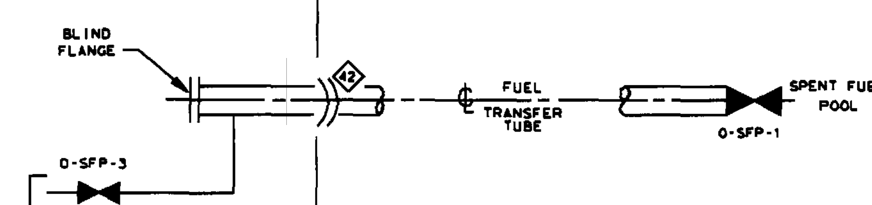
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 26 | REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION
41	REACTOR COOLANT SHUTDOWN COOLING	10	IV	11	EAST PIPING PENETRATION ROOM	14"	OM-074 SH0002	MOV-651 MOV-652	LOCKED SHUT LOCKED SHUT
42	FUEL TRANSFER TUBE	43	IV	8	SPENT FUEL POOL	36"	OM-058	SFP-1	CLOSED

DIAGRAM

INSIDE CTMT. STRUCTURE | OUTSIDE CTMT. STRUCTURE





*SFP-1 IS NOT SUBJECT TO TYPE "C" TESTING.

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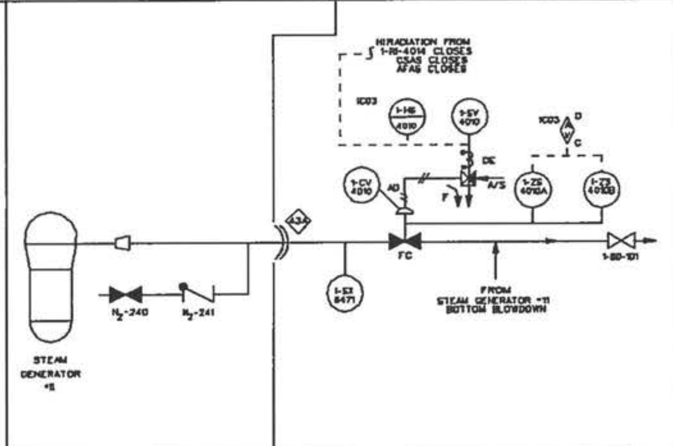
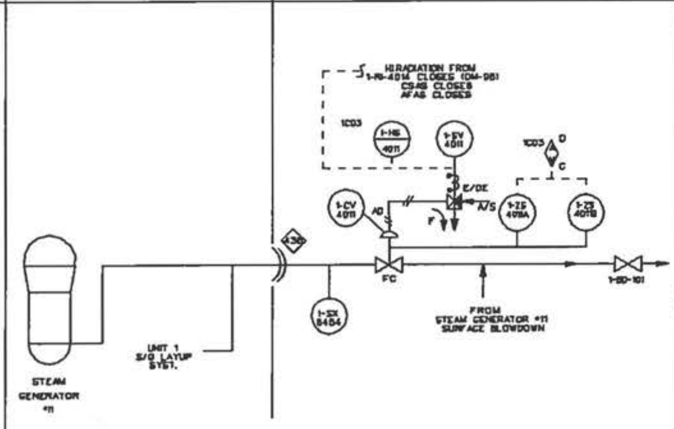
10/19/99

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 27	REVISION: 26

FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 28)

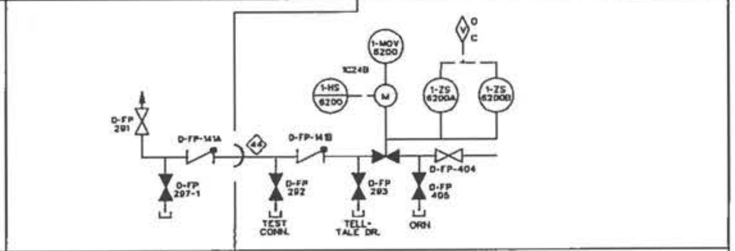
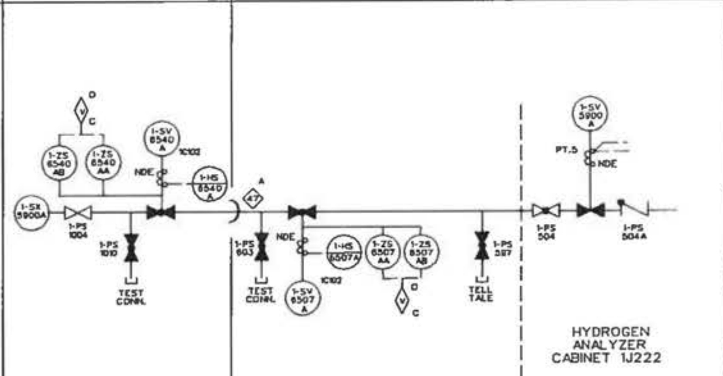
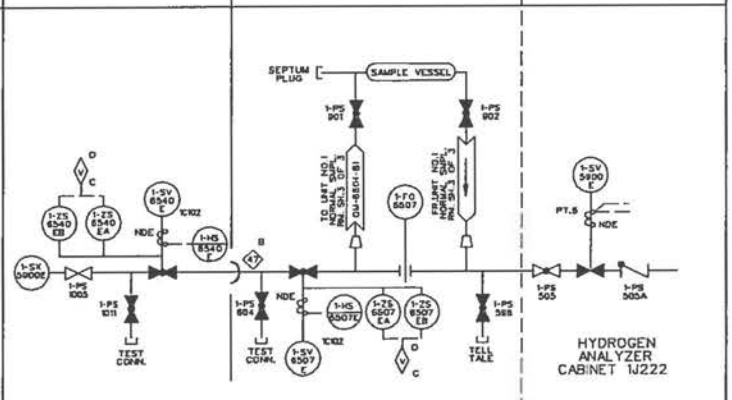
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
43A	STEAM GENERATOR #11 SURFACE BLOWDOWN	47	III	14	EAST PIPING PENETRATION ROOM	2"	OM-35 SH.1 OF 3 OM-464	CV-4010	CLOSED	see diagram 43A below
43B	STEAM GENERATOR #11 BOTTOM BLOWDOWN		III	14	EAST PIPING PENETRATION ROOM	2"	OM-35 SH.1 OF 3 OM-464	CV-4011	CLOSED	see diagram 43B below



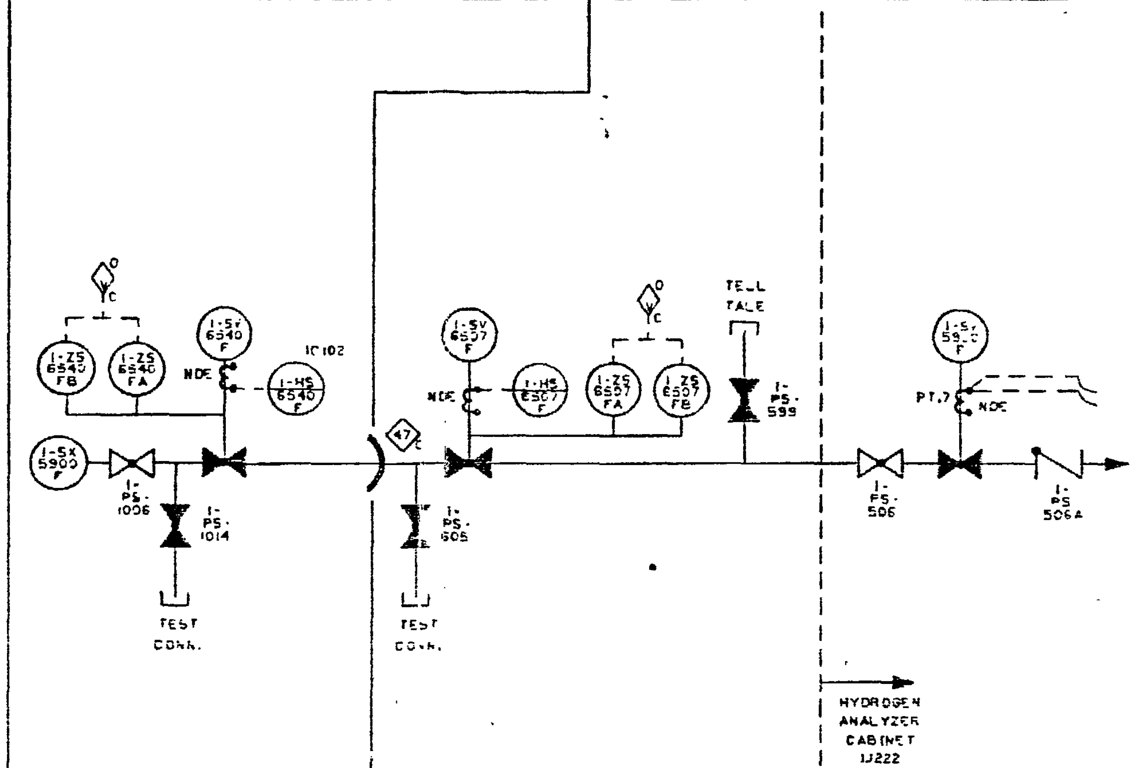
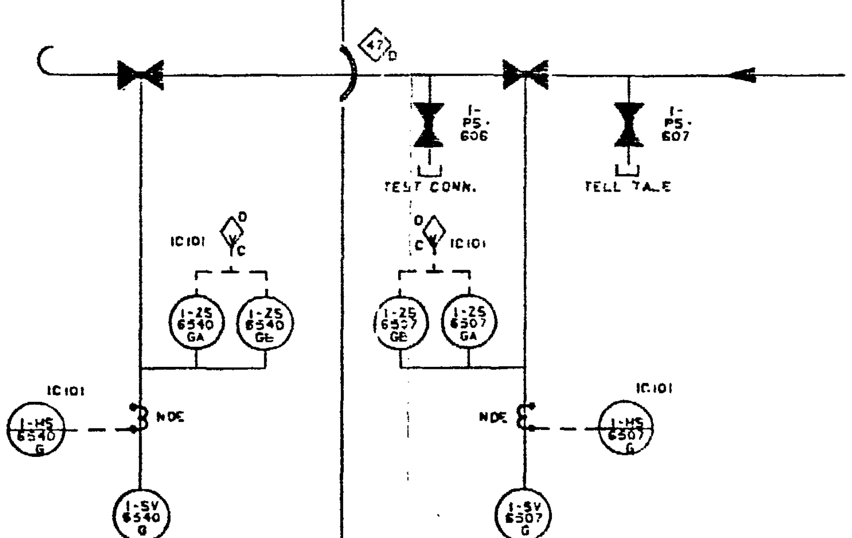


22589.DGN	FSAR-28	CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 28	REVISION: 42

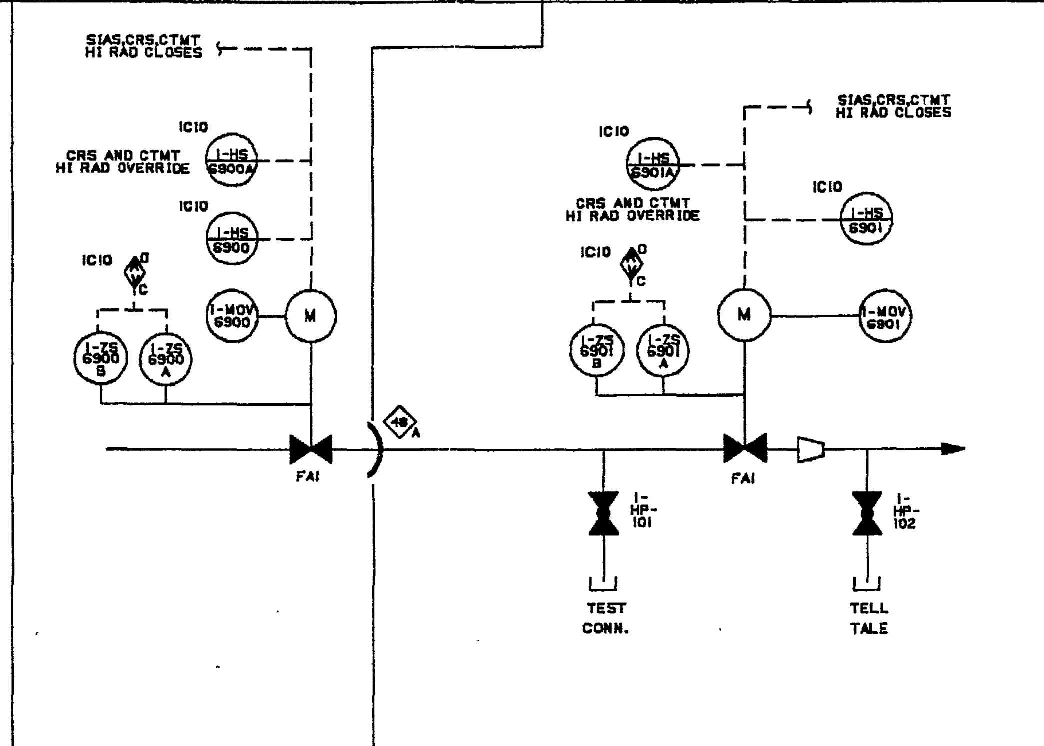
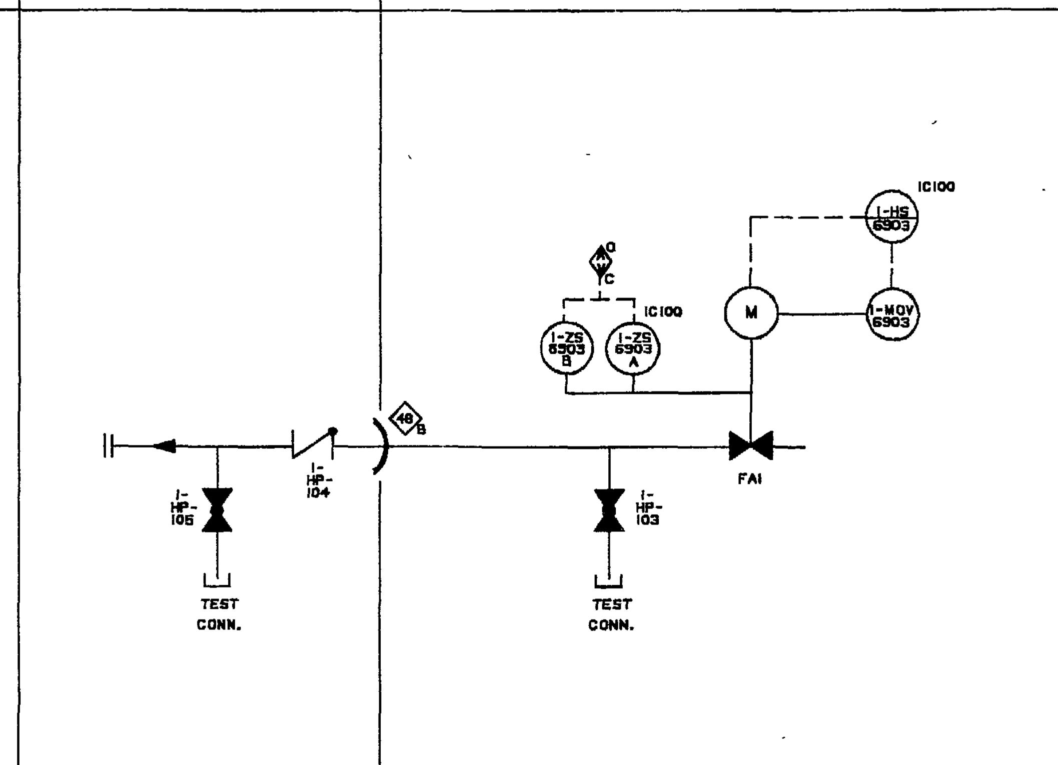
FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 29)

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
44	FIRE PROTECTION	44	III	21	EAST PIPING PENETRATION ROOM	6"	OM-56 SH.2 OF 6	FP-141B FP-141A	--	
47A	HYDROGEN SAMPLING N. OF PRI. SHIELDING	--	II	34	WEST PIPING PENETRATION ROOM	1/4"	OM-463 SH.1 OF 2	SV-6507A SV-6540A	LOCKED SHUT	
47B	HYDROGEN SAMPLING W. ELEV. AT 135 FT.	--	II	34	WEST PIPING PENETRATION ROOM	1/4"	OM-463 SH.1 OF 2	SV-6507E SV-6540E	LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 29 | REVISION: 36

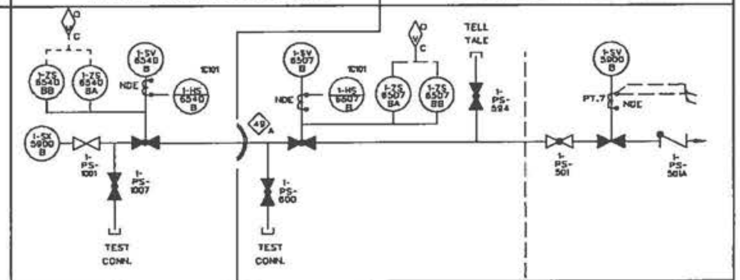
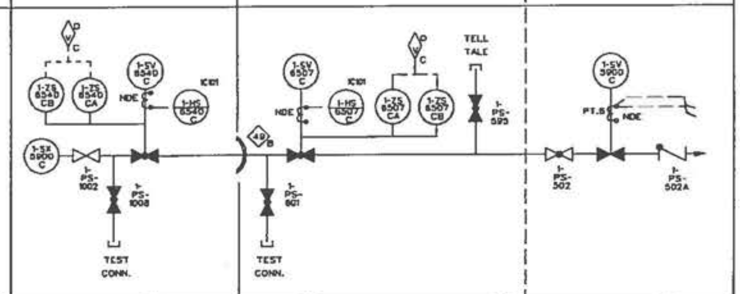
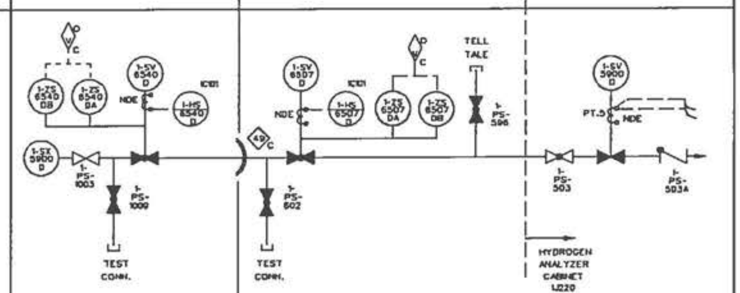
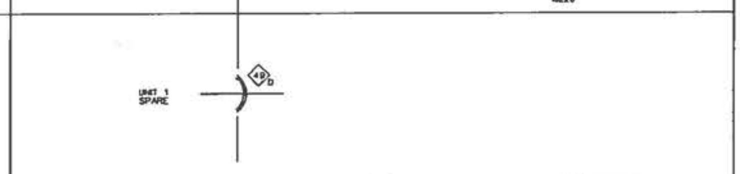
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
47C	HYDROGEN SAMPLING DOME ELEV.AT 189 FT.	- -	II	34	WEST PIPING PENETRATION ROOM	1/4"	OM-463 SH.1 OF 2	SV-6540F SV-6507F	LOCKED SHUT LOCKED SHUT	
47D	HYDROGEN SAMPLE RETURN TO CTMT.	- -	II	44	WEST PIPING PENETRATION ROOM	1/4"	OM-463 SH.2 OF 2	SV-6540G SV-6507G	LOCKED SHUT LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 30

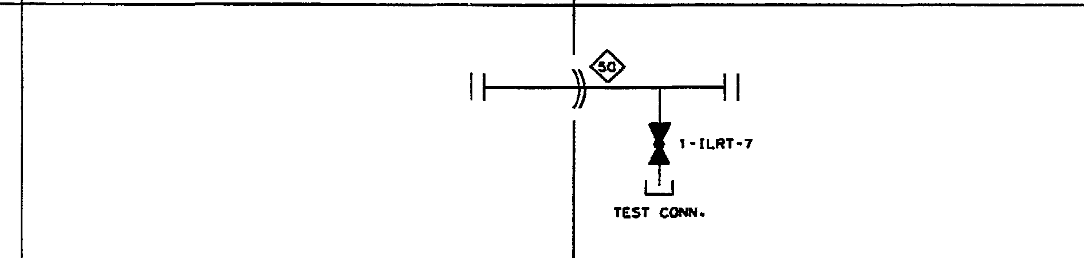
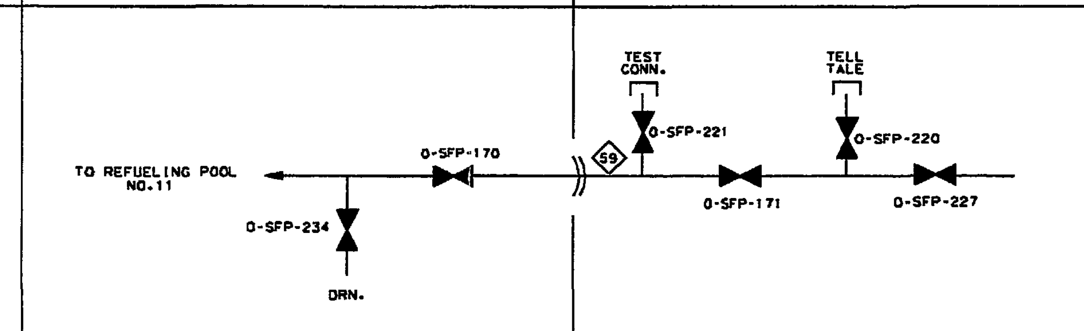
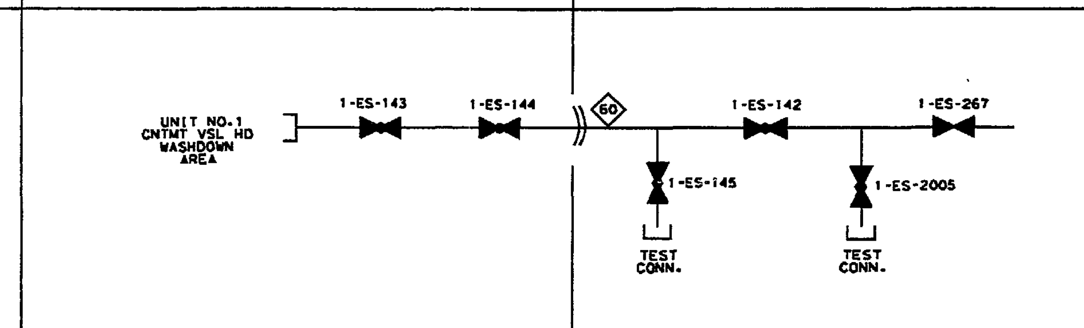
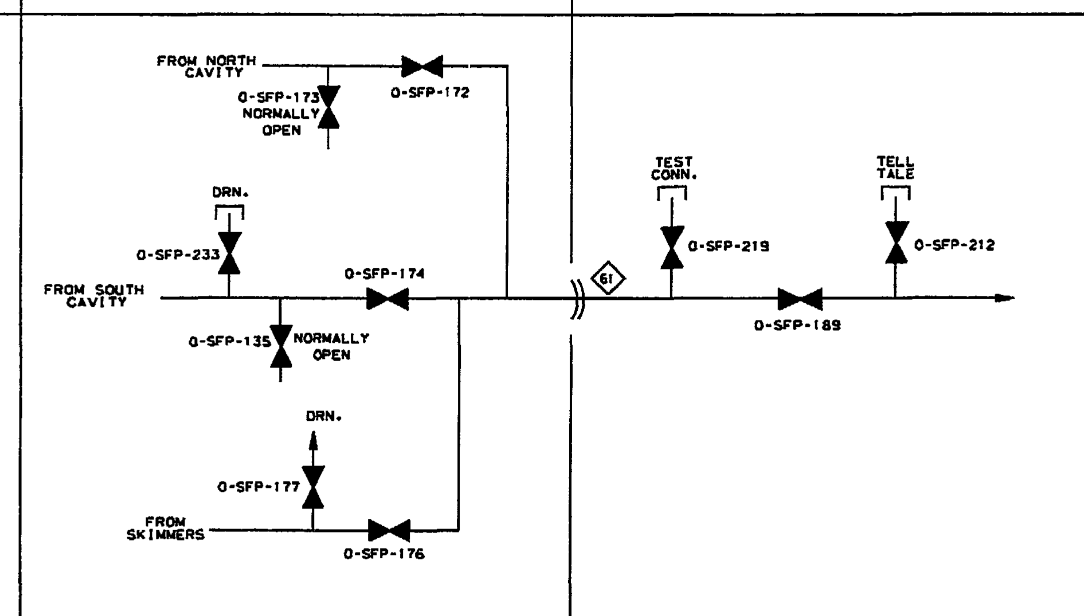
PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
48A	CONTAINMENT HYDROGEN PURGE OUTLET	--	II	35	EAST PIPING PENETRATION ROOM	4"	OM-65 SH.2 OF 4	MOV-6900 MOV-6901	CLOSED CLOSED	
48B	CONTAINMENT HYDROGEN PURGE INLET	--	II	41	EAST PIPING PENETRATION ROOM	4"	OM-65 SH.2 OF 4	HP-104 MOV-6903	-- CLOSED	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO. 31	REV:13	DATE:1/92

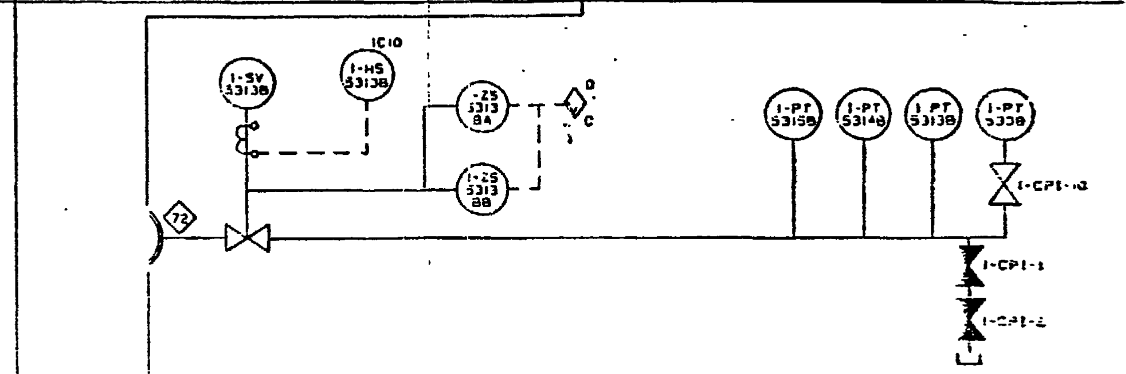
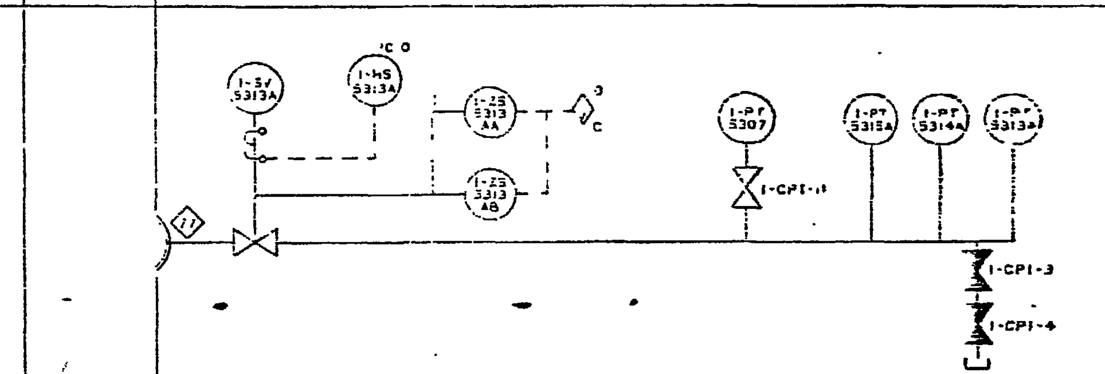
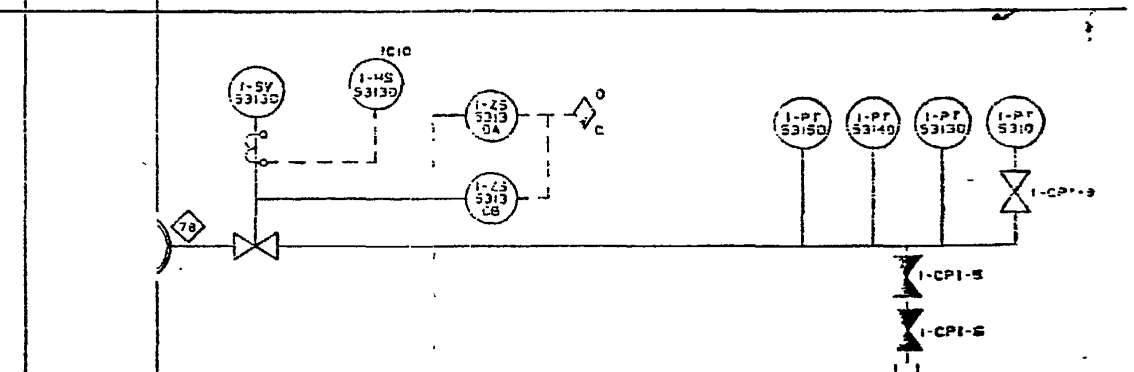
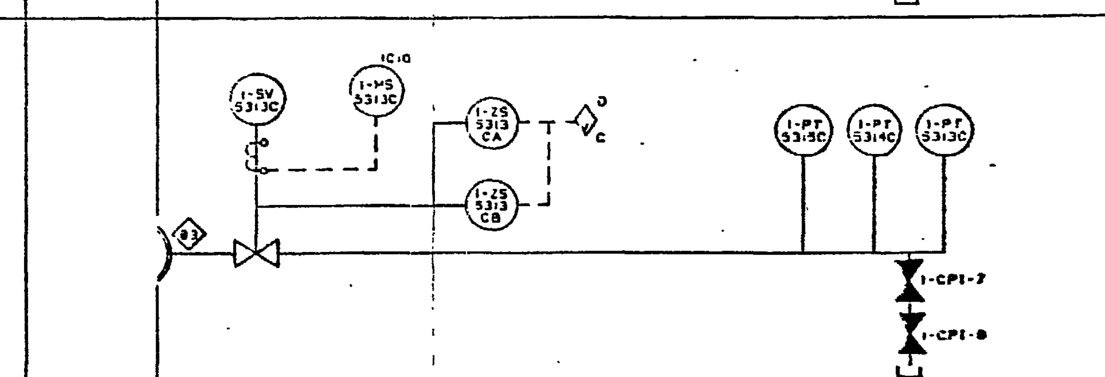
FIGURE 5-10 CONTAINMENT STRUCTURE – ISOLATION VALVE ARRANGEMENT (Sheet 32)

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
49A	HYDROGEN SAMPLE SOUTH OF PRIMARY SHIELD	---	II	34	WEST PIPING PENETRATION RM.	1/4"	OM-463 SH.1 OF 2	SV-6507B SV-6540B	LOCKED SHUT	
49B	HYDROGEN SAMPLE PRESSURIZER COMPARTMENT	---	II	34	WEST PIPING PENETRATION RM.	1/4"	OM-463 SH.1 OF 2	SV-6507C SV-6540C	LOCKED SHUT	
49C	HYDROGEN SAMPLE EAST ELEVATION AT 135 FT.	---	II	34	WEST PIPING PENETRATION RM.	1/4"	OM-463 SH.1 OF 2	SV-6507D SV-6540D	LOCKED SHUT	
49D	SPARE	---	II	47	WEST PIPING PENETRATION RM.	1/4"	OM-463 SH.1 OF 2			

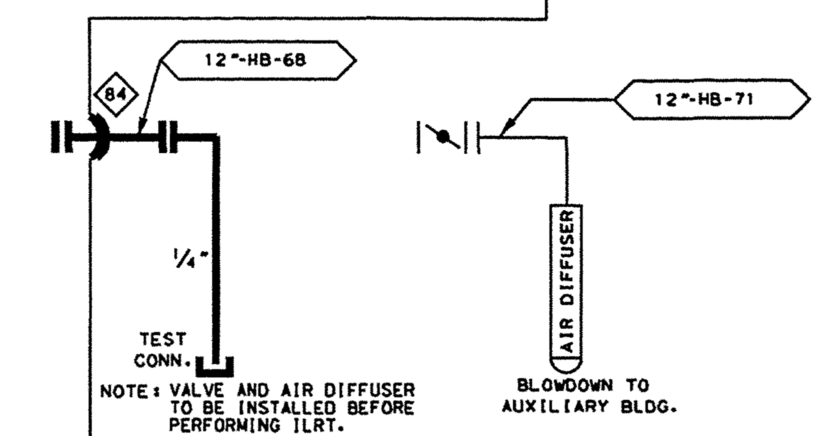
CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO. 32 | REVISION: 36

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
50	ILRT PENETRATION	---	IV	46	EAST PIPING PENETRATION RM.	6"	OM-065 SH0002	BLIND FLANGE BLIND FLANGE	CLOSED CLOSED	
59	REFUELING POOL RECYCLE INLET	---	IV	30	WEST PIPING PENETRATION RM.	8"	OM-058	SFP-171 SFP-170	LOCKED SHUT LOCKED SHUT	
60	STEAM TO REACTOR HEAD WASHDOWN AREA	---	IV	42	WEST PIPING PENETRATION RM.	1"	OM-077 SH0003	ES-142 ES-144	LOCKED SHUT LOCKED SHUT	
61	REFUELING POOL OUTLET TO FUEL POOL COOLING PIPE	46	IV	27	WEST PIPING PENETRATION RM.	8"	OM-058	SFP-189 SFP-172 SFP-174 SFP-176	LOCKED SHUT LOCKED SHUT LOCKED SHUT LOCKED SHUT	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.33 | REVISION: 21

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM (INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE)
72	CONTAINMENT PRESSURE MONITOR INSTR.	---	INSTRU.	33	EAST ELECTRICAL PENETRATION RM.	3/4"	OM-65 SH.2 OF 4	SV-5313B	OPEN	
77	CONTAINMENT PRESSURE MONITOR INSTR.	---	INSTRU.	33	EAST ELECTRICAL PENETRATION RM.	3/4"	OM-65 SH.2 OF 4	SV-5313A	OPEN	
78	CONTAINMENT PRESSURE MONITOR INSTR.	---	INSTRU.	33	WEST ELECTRICAL PENETRATION RM.	3/4"	OM-65 SH.2 OF 4	SV-5313D	OPEN	
83	CONTAINMENT PRESSURE MONITOR INSTR.	---	INSTRU.	40	WEST ELECTRICAL PENETRATION RM.	3/4"	OM-65 SH.2 OF 4	SV-5313C	OPEN	

CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT | FIGURE 5-10 | SHEET NO.35

PENE. NO.	SERVICE	ORIG. FSAR PENE. NO.	PENE. TYPE	VALVE ARRGT.	PENETRATION LOCATION	PENE. LINE SIZE	DWG. NUMBER	ISOLATION VALVE(S)	POST-INCIDENT POSITION	DIAGRAM: INSIDE CTMT. STRUCTURE / OUTSIDE CTMT. STRUCTURE
84	ILRT 12" VENT PENETRATION	---	IV		EAST ELECTRICAL PENETRATION ROOM	12"	OM-065 SH0002	BLIND FLG. BLIND FLG.	CLOSED CLOSED	

FSAR-36 74853.DGN	CONTAINMENT STRUCTURE ISOLATION VALVE ARRANGEMENT	FIGURE 5-10	SHEET NO.36	REV:31	DATE:10/01

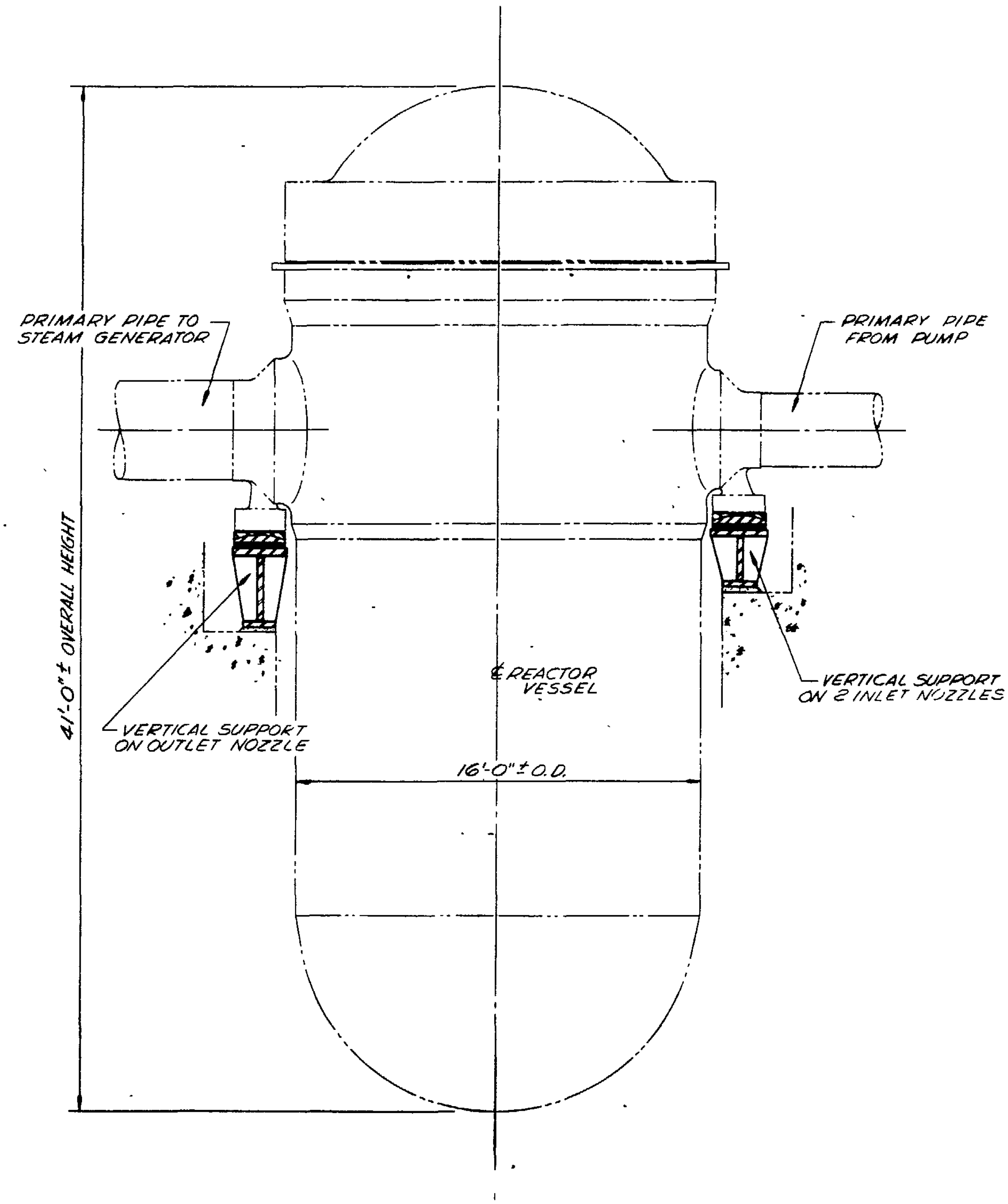


FIG. 5-11 ~ REACTOR VESSEL
Sheet 1 of 2

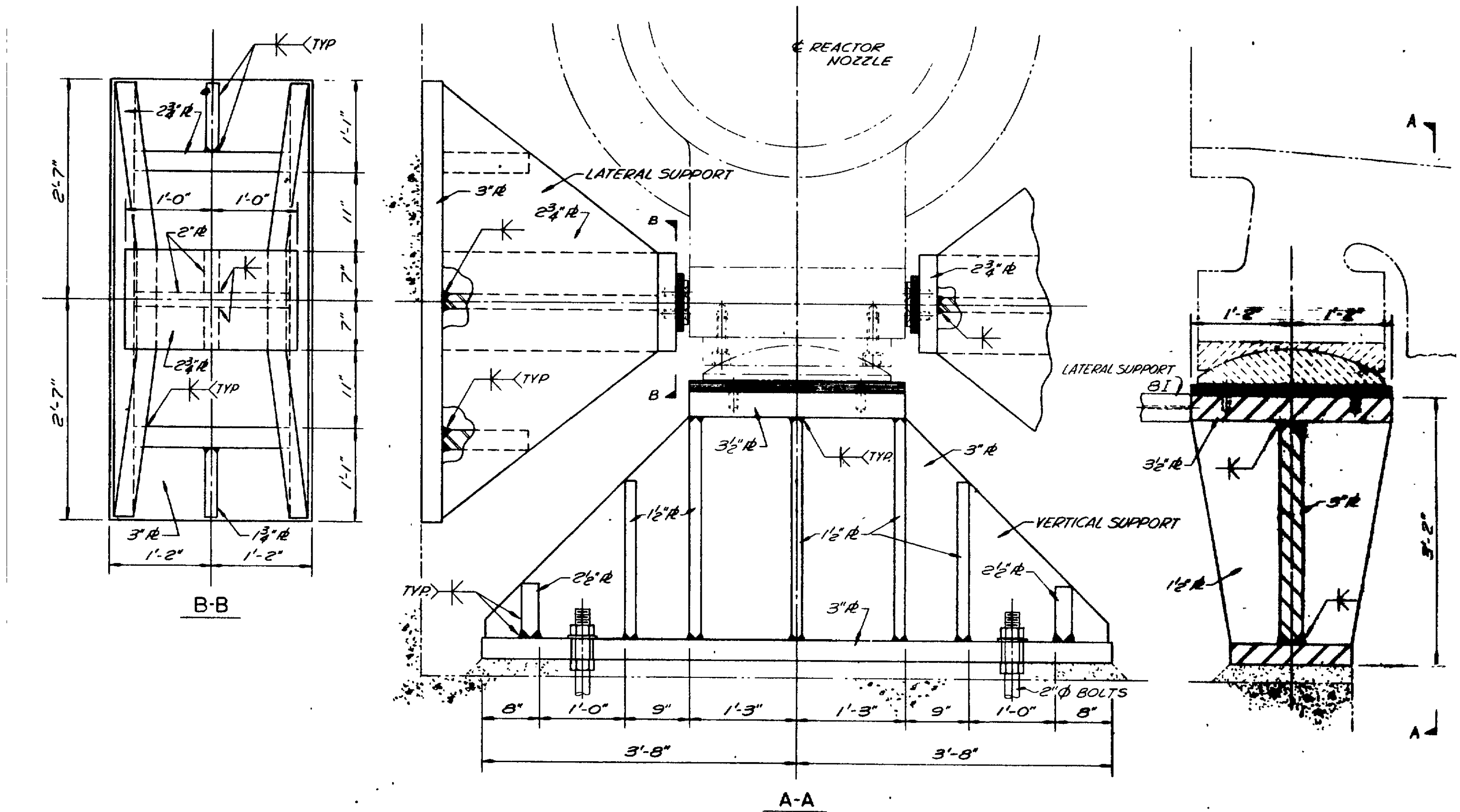


FIG. 5-11 ~ REACTOR VESSEL SUPPORTS
Sheet 2 of 2

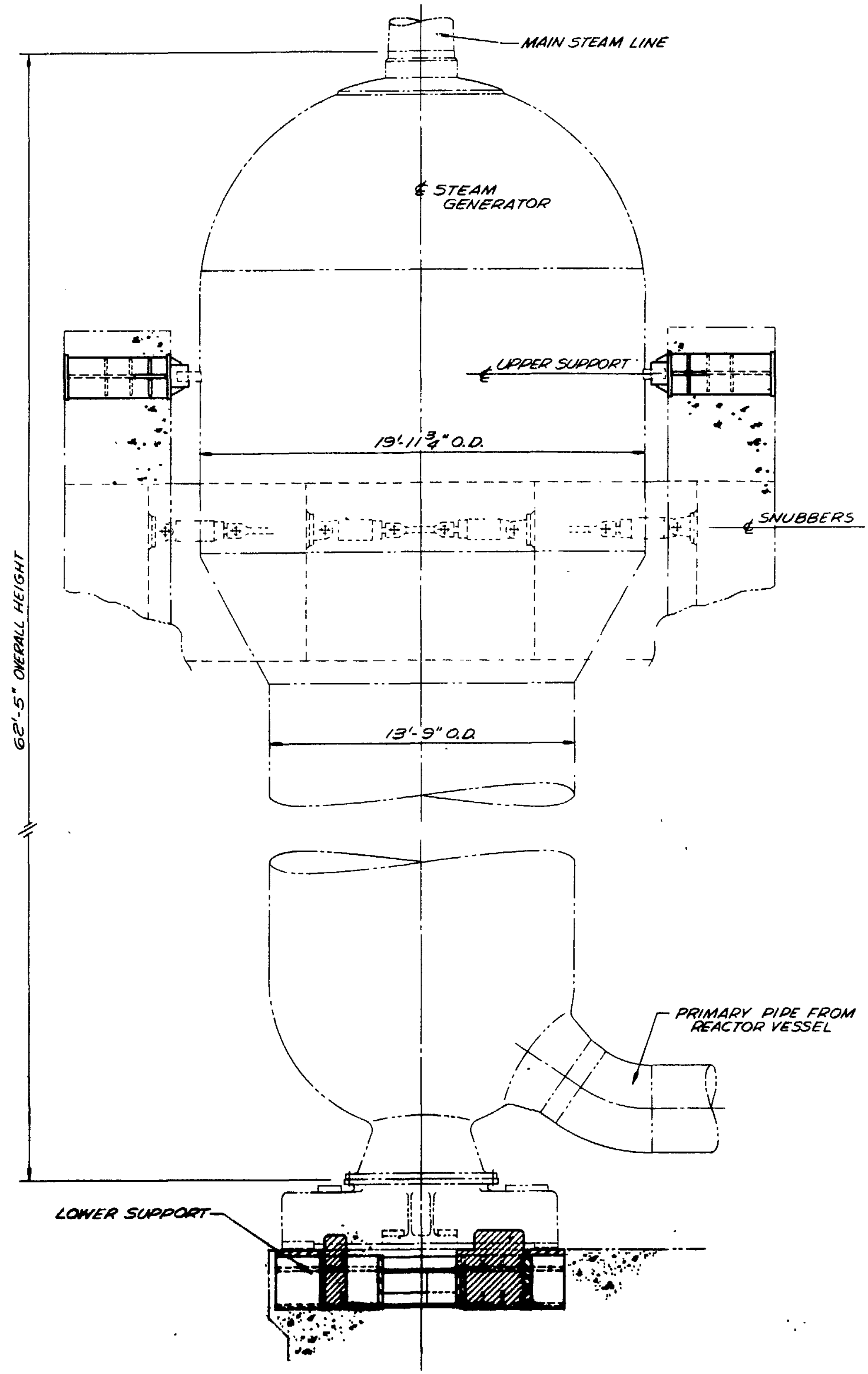
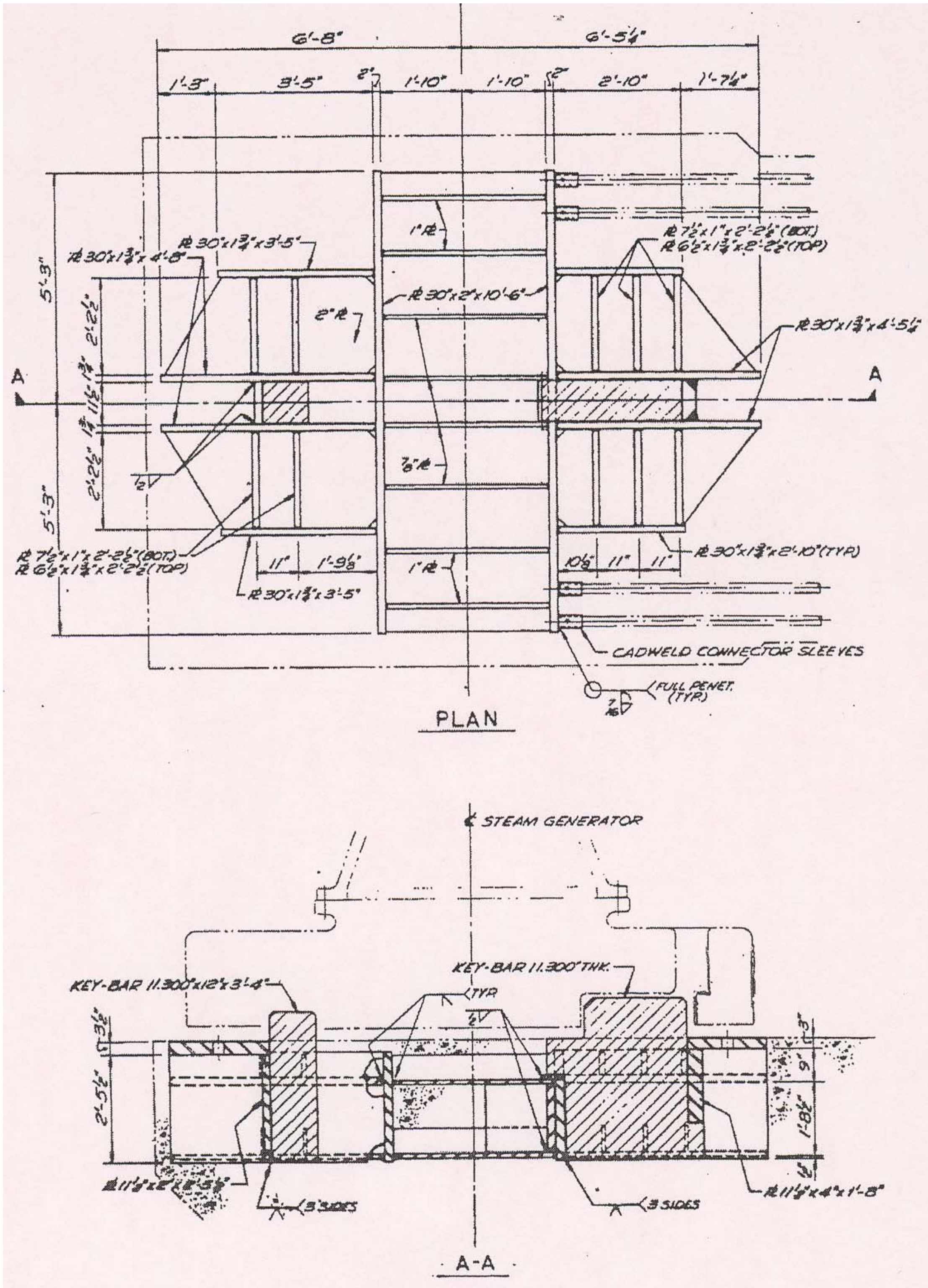


FIG. 5-12 ~ STEAM GENERATOR
Sheet 1 of 3

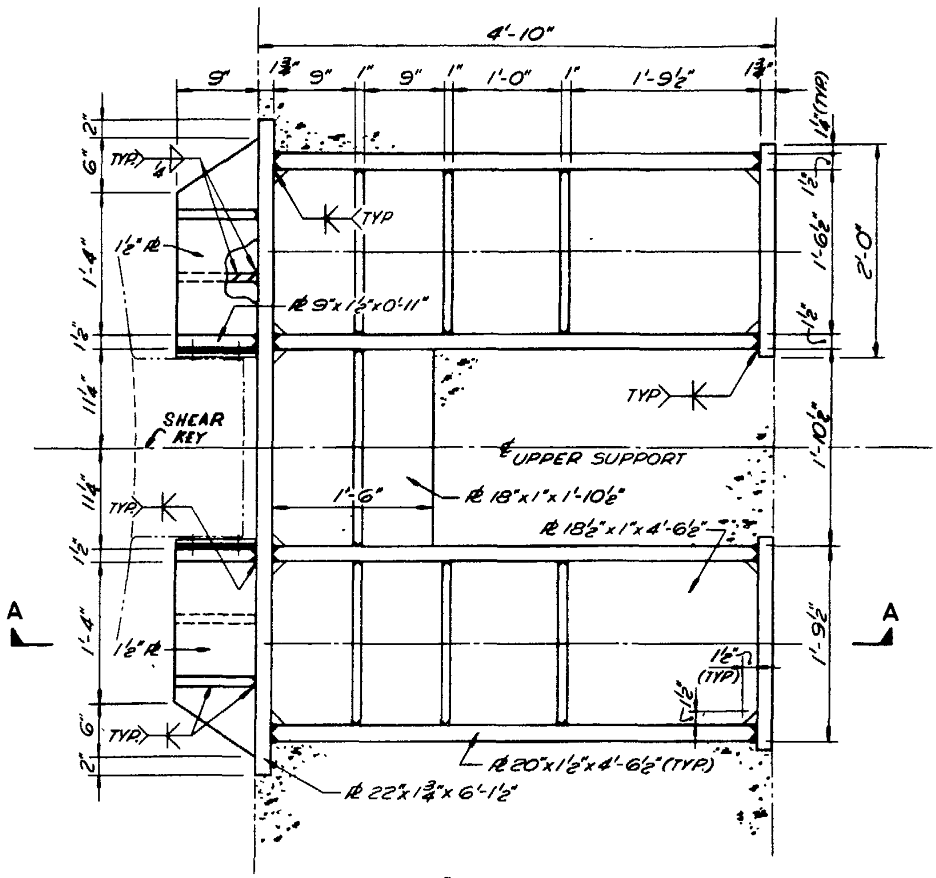


Calvert Cliffs Nuclear Power Plant

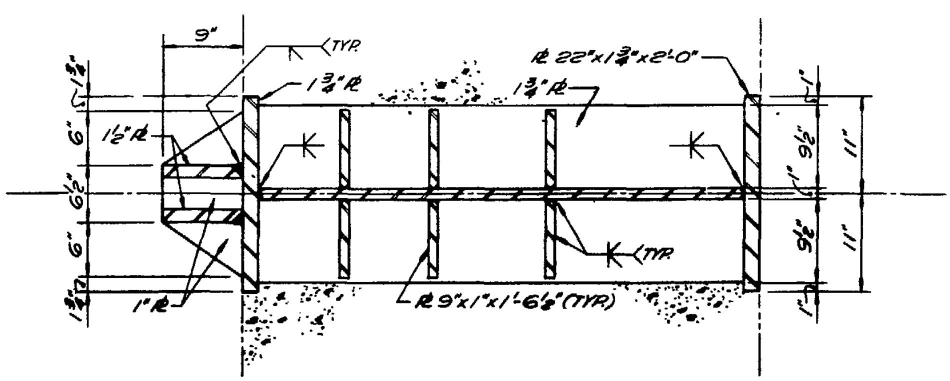
STEAM GENERATOR LOWER SUPPORT (Sheet 2 of 3)

Figure 5-12

Revision 33



PLAN



A-A

FIG. 5-12 ~ STEAM GENERATOR UPPER SUPPORT
Sheet 3 of 3

FIGURE 5-13 MAIN COOLANT PUMP SUPPORTS

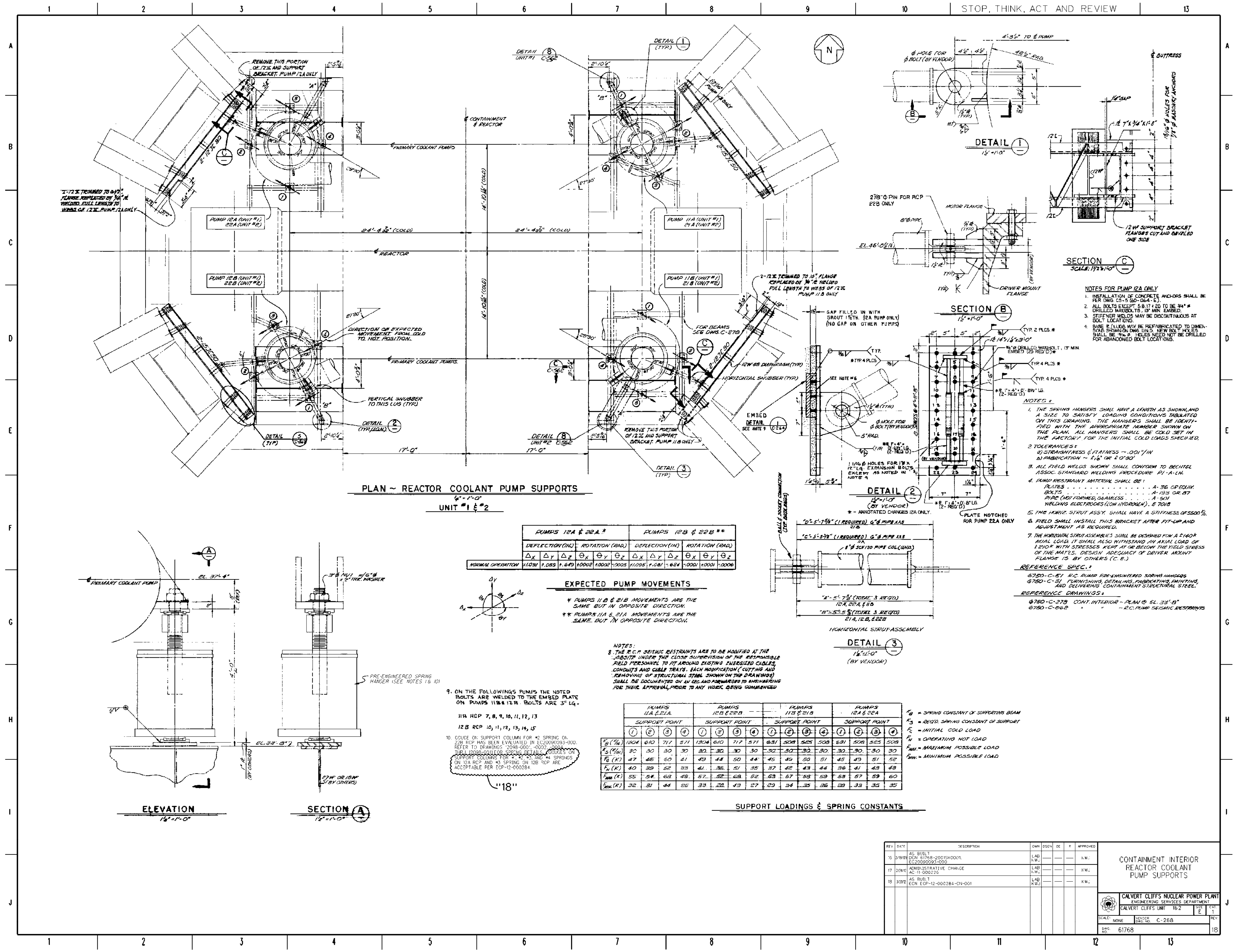
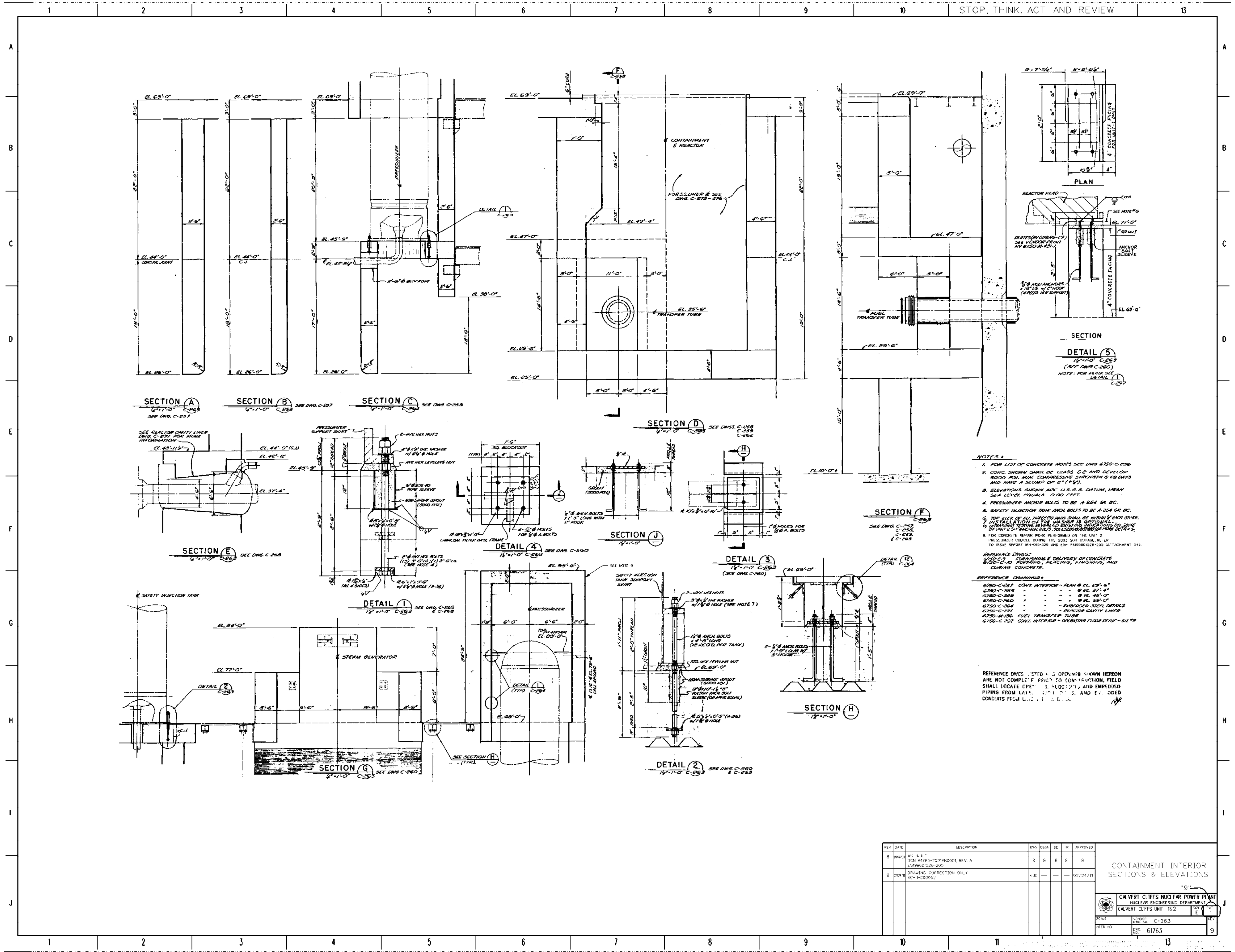
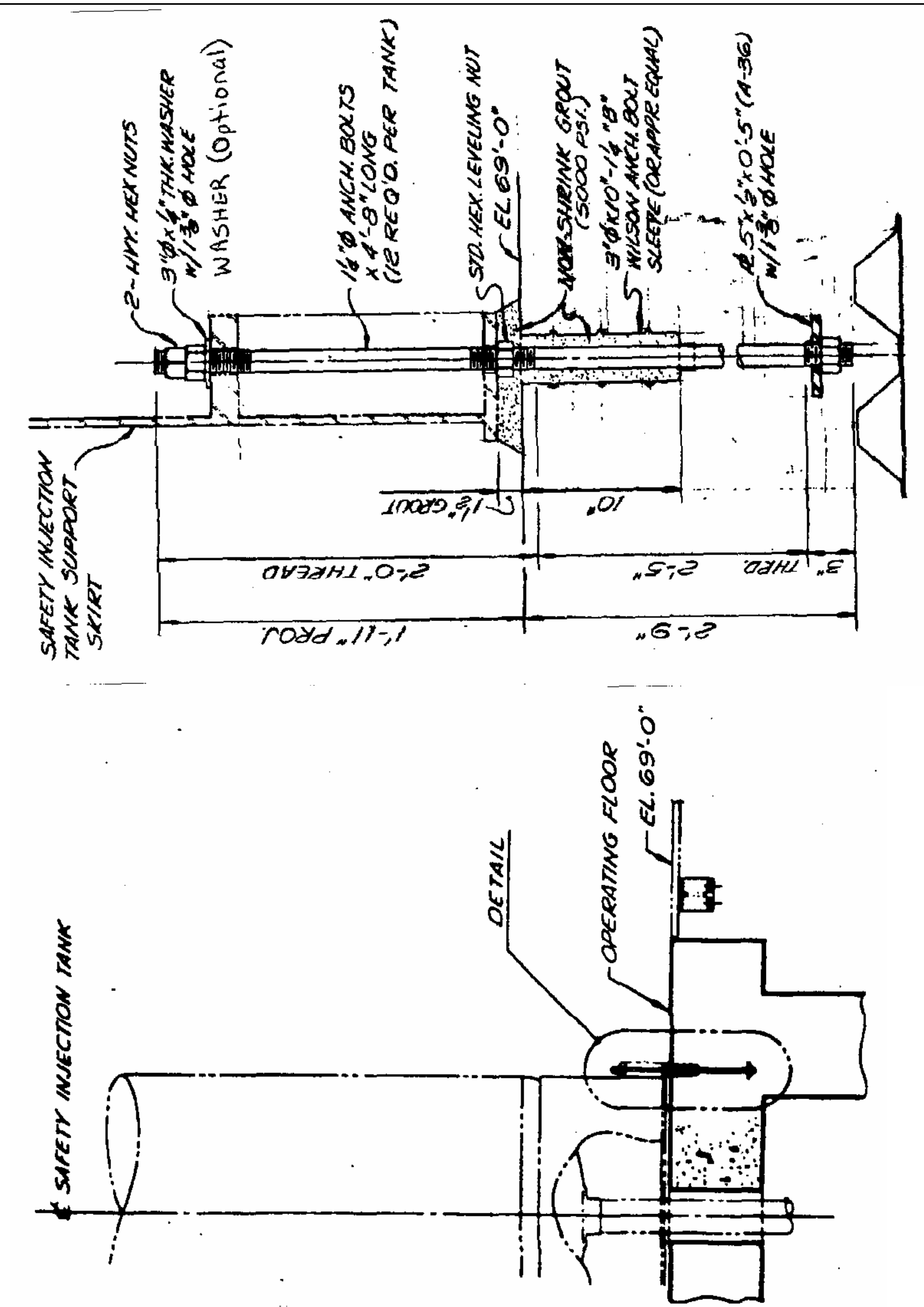


FIGURE 5-14 PRESSURIZER SUPPPORT

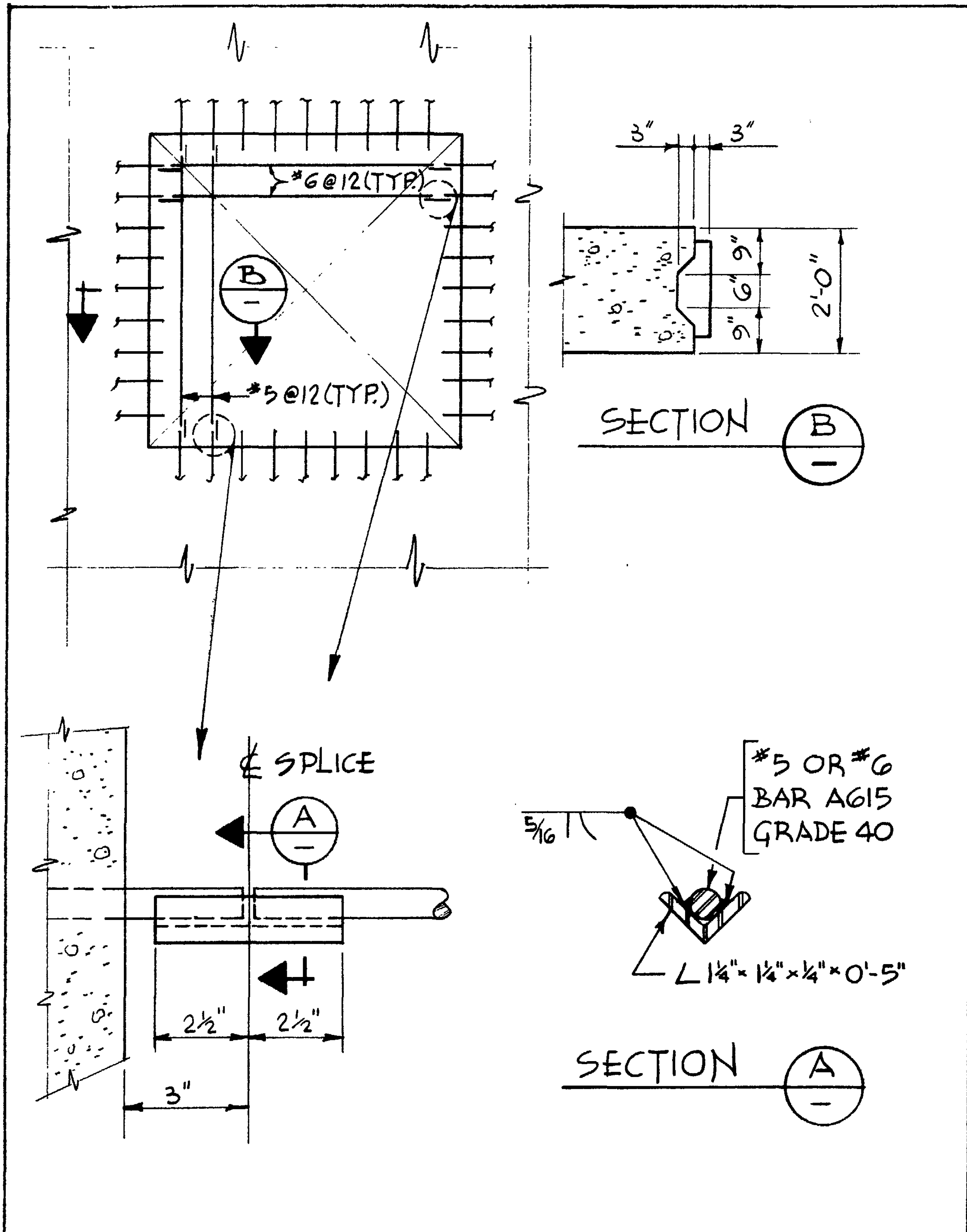




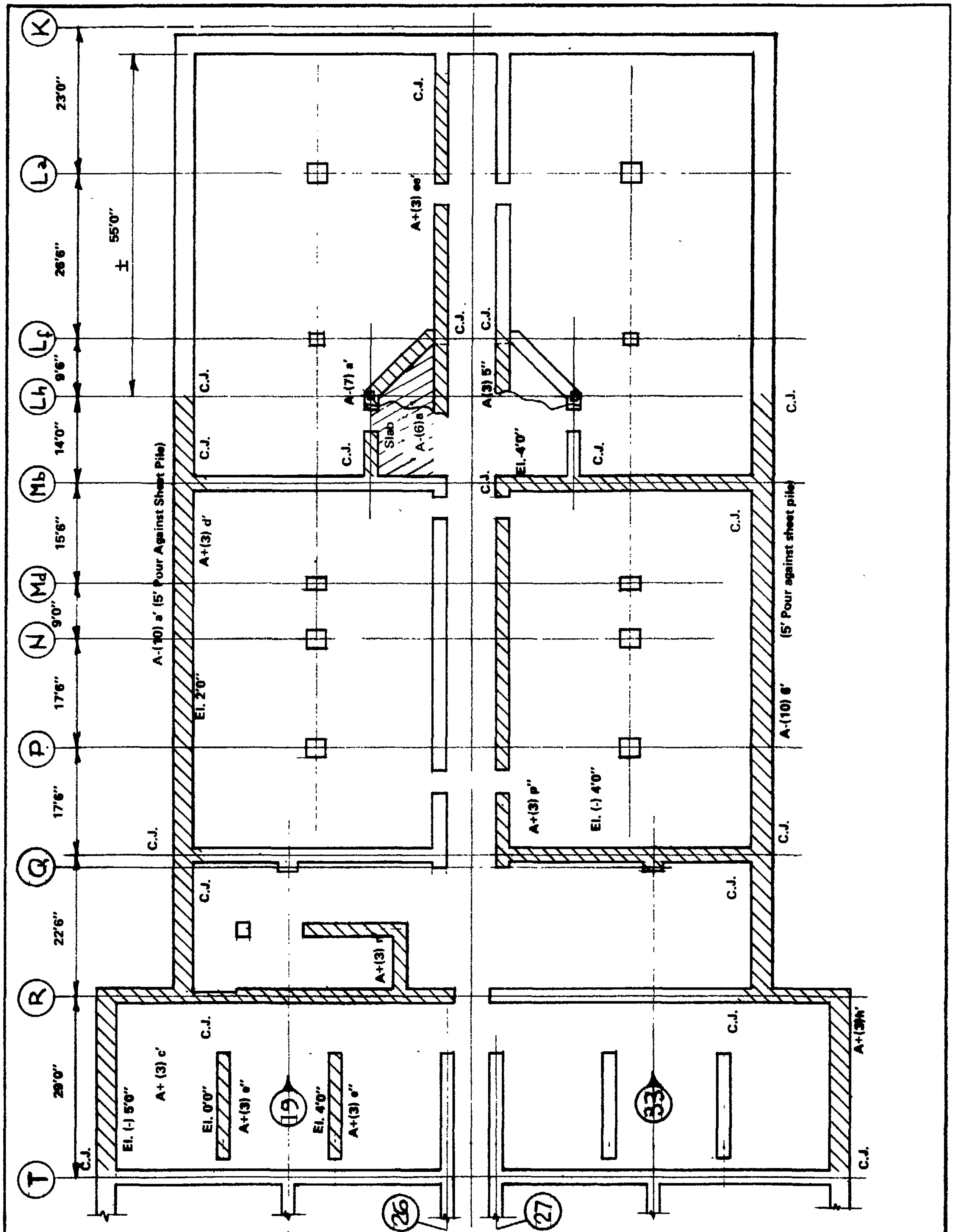
NOTE: Ultrasonic Testing revealed existing indications on some of the Unit 2 SIT anchor bolts. See ES200100357-000 for more details.

61-763-E

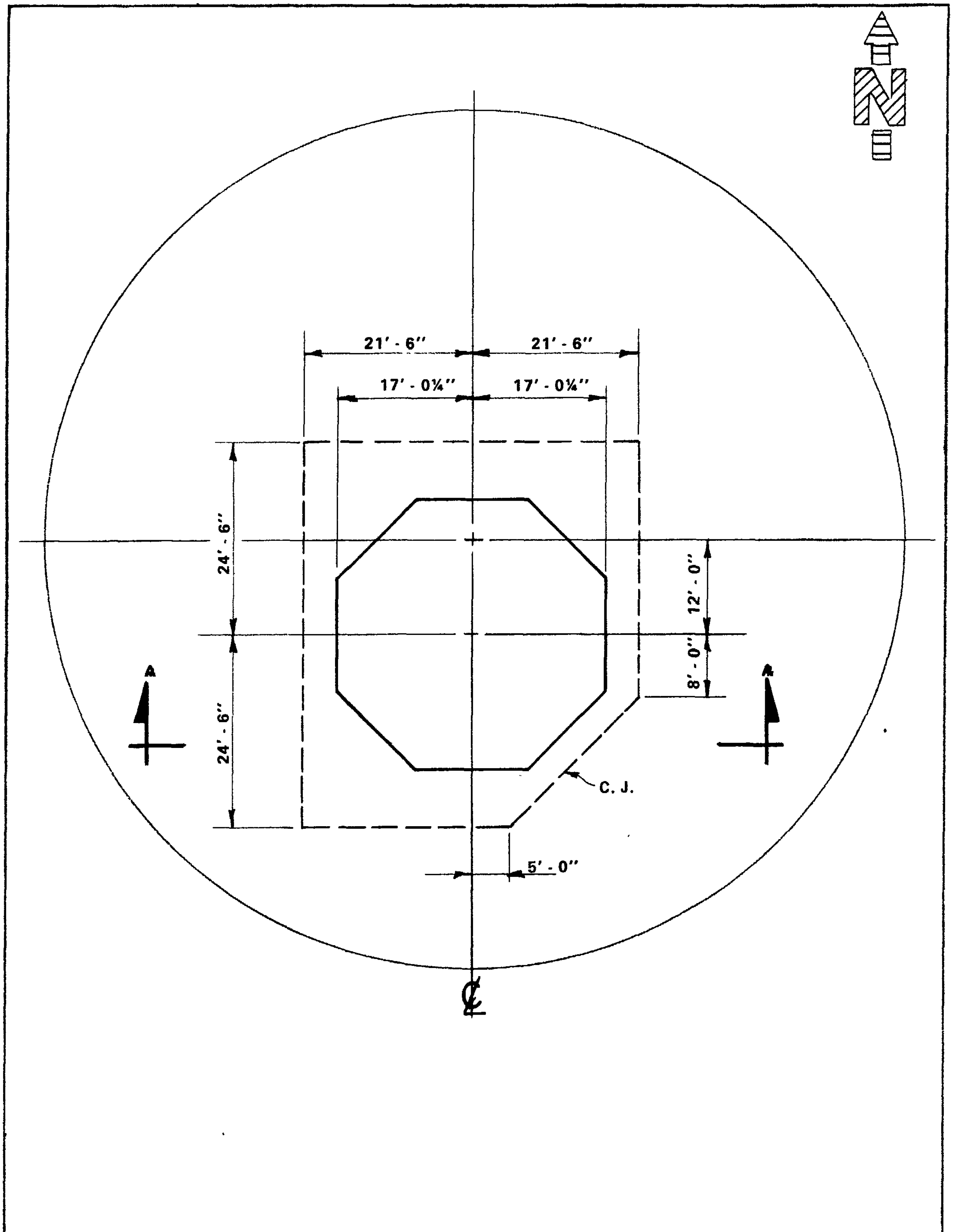
Calvert Cliffs Nuclear Power Plant	SAFETY INJECTION TANK SUPPORT	Figure 5-15 Revision 38



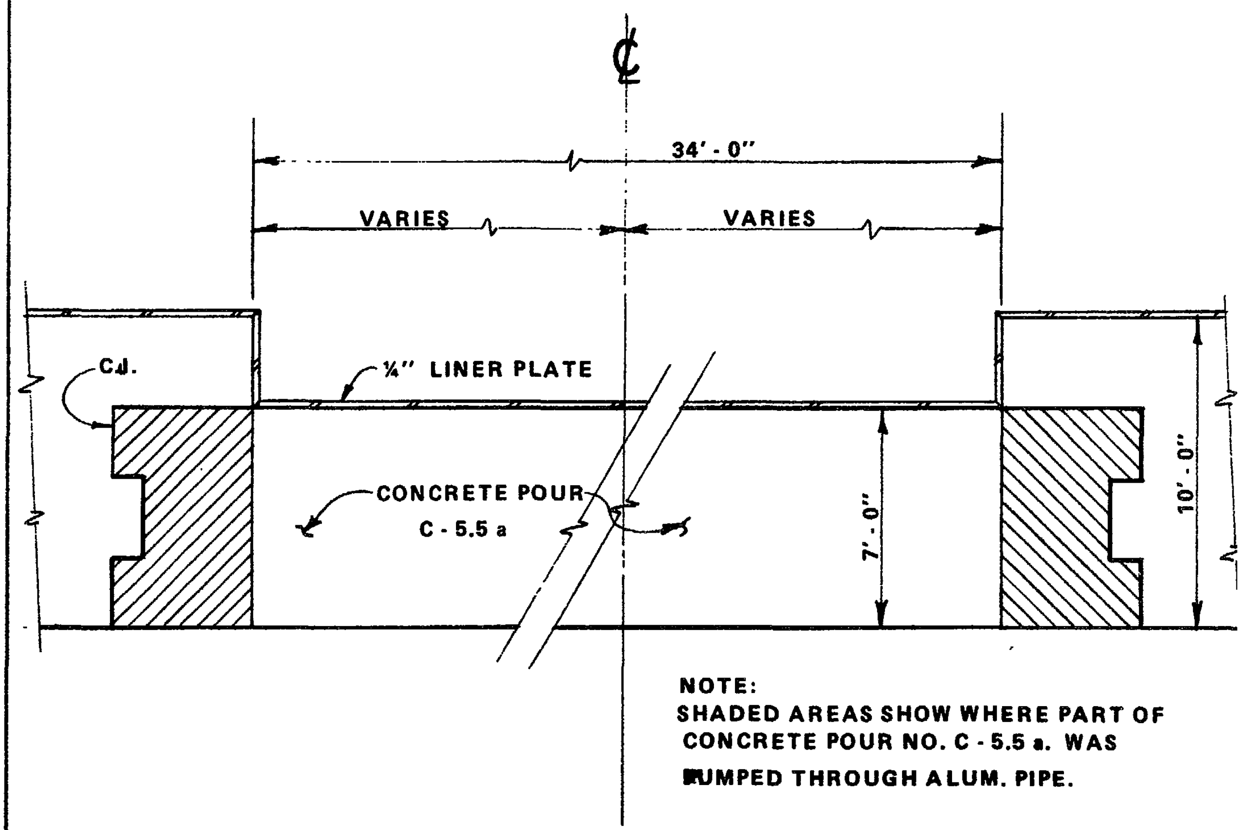
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	DETAIL OF EQUIPMENT BLOCKOUT SHOWING WELDED FABRICATION OF REINFORCEMENT	FIGURE 5-16



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	AUX. BLDG. POURS PUMPED WITH ALUM. PIPE	FIGURE 5-17



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CONTAINMENT STRUCTURE UNIT NO. 1	FIGURE 5-18



NOTE:
SHADED AREAS SHOW WHERE PART OF CONCRETE POUR NO. C - 5.5 a. WAS PUMPED THROUGH ALUM. PIPE.

SECTION A-A

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BASE SLAB. CONTAINMENT STRUCTURE UNIT NO. 1	FIGURE 5-19

APPENDIX 5A
STRUCTURAL DESIGN BASIS

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STRUCTURAL DESIGN BASIS

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APPENDIX 5A
STRUCTURAL DESIGN BASIS

LIST OF ACRONYMS

ADV	Atmospheric Dump Valve
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combustion Engineering, Inc.
EDG	Emergency Diesel Generator
FOST	Fuel Oil Storage Tank
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss-of-Coolant Accident
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
RCS	Reactor Coolant System
SG	Steam Generator
SSE	Safe Shutdown Earthquake
UFSAR	Updated Final Safety Analysis Report
USI	Unresolved Safety Issue

APPENDIX 5A

5A.0 STRUCTURAL DESIGN BASIS

5A.1 GENERAL

The design basis for structures for normal operating conditions are governed by the applicable building design codes. The design bases for specific systems and equipment are stated in the appropriate Updated Final Safety Analysis Report (UFSAR) section. The design basis for the maximum loss-of-coolant accident (LOCA) and seismic conditions is that there be no loss of function if that function is related to public safety. The method used to determine the seismic response resulting from the Operating Basis (OBE) and Safe Shutdown Earthquake (SSE) is described in Sections 2.6.5 and 5.1.3.2.b.

5A.1.1 RESPONSIBLE DESIGN ORGANIZATIONS

Calvert Cliffs Nuclear Power Plant (CCNPP), as the applicant, has the ultimate responsibility for the design and construction of Calvert Cliffs Nuclear Power Plant, Units 1 and 2. Calvert Cliffs Nuclear Power Plant utilizes its experienced staff to perform project management, engineering review, construction coordination and quality assurance functions.

Combustion Engineering, Inc. (CE), as the Nuclear Steam Supply System (NSSS) supplier, has supplied the components of the Reactor Coolant System, the Chemical and Volume Control System and the Safety Injection System. Babcock & Wilcox, Canada is the supplier of the replacement steam generators. As the suppliers, they are responsible for the seismic design of their components in a manner consistent with the design criteria for the project.

Bechtel Associates (Bechtel), as the Architect-Engineer for CCNPP, was responsible for developing the seismic criteria and the design of Category I structures and for the approval of all other Category I equipment. Once the seismic criteria for the project were established by Bechtel with the assistance of Dames and Moore, soil consultants, Bechtel ensured that these criteria were implemented in the design of Category I structures.

Bechtel Associates also developed response spectra curves and all other requirements necessary for the design of all Category I equipment including the NSSS. This information is given to all suppliers of Category I equipment including CE who has implemented these seismic criteria in their design. All interdisciplinary exchanges, between Bechtel, CE, CCNPP and any vendor supplying Category I equipment, were documented with memoranda or conference notes. Exchanges of letters, specifications and drawings in accordance with defined procedures are used to maintain uniform design throughout the plant.

In 2001 Stevenson and Associates developed in-structure acceleration time-histories and seismic response spectra to be utilized for the design and evaluation of Category I equipment within the Containment, including the NSSS.

5A.1.1.1 Design Control

Design control is effected by successive levels of review of seismic criteria, calculations, and seismic sections of specifications and drawings. At Bechtel, these levels are the Responsible Engineer, Group Supervisor, and Chief Engineer with final approval by the Project Engineer. These reviews also cover seismic requirements placed on suppliers, such as CE, where reviews are performed by the Specialty Group and Project Manager.

5A.1.2 SPECIFIC REQUIREMENTS FOR SAFETY-RELATED PURCHASES

Both the NSSS supplier and the Architect Engineer included requirements for seismic design in specifications for Category I equipment. Combustion Engineering, Inc. is required to design the NSSS to withstand the load imposed by the maximum hypothetical accident, and by the maximum seismic disturbance without loss of functions required for reactor shutdown and emergency core cooling.

Definitions in typical seismic specifications for Category I equipment:

a. The OBE: has a maximum horizontal ground acceleration of 0.08 g and a maximum vertical ground acceleration of 0.053 g, acting simultaneously.

b. The SSE: has a maximum horizontal ground acceleration of 0.15 g and a maximum vertical ground acceleration of 0.10 g, acting simultaneously. These seismic acceleration levels were established to provide an appropriate margin of safety for withstanding stresses greater than those recorded and reflect uncertainties about the historical data and their suitability for design basis.

c. All Category I systems, equipment and components shall be designed to withstand the appropriate seismic load combined with other applicable loads without loss of function. The analysis of the dynamic loads on Category I systems is accomplished by using the Response-spectrum Method as outlined in Bechtel's seismic specification.

All vendors supplying Category I equipment or systems, are required to submit copies of their dynamic analyses or dynamic test results, based on seismic criteria, for approval.

5A.2 CLASSES OF STRUCTURES, SYSTEMS, AND EQUIPMENT

5A.2.1 SEISMIC CATEGORY I

Throughout the context of this appendix, Category I shall mean Seismic Category I structures, systems, and equipment. Category I structures, systems, and equipment are those whose failure could cause uncontrolled release of radioactivity or those essential for immediate and long-term operation following a LOCA. When a system as a whole is referred to as Category I, positions not associated with loss-of-function of the system may be designated as Category II.

5A.2.1.1 Typical Category I Structures

a. Containment structure shell
b. The Auxiliary Building below Elevation 69'0"
c. Enclosures for the critical service water pumps, critical saltwater pumps, and auxiliary feedwater pumps and for auxiliary feedwater valves and piping header in the tank farm
d. Foundations for Category I system components

The analysis of Category I equipment, systems and components located at various levels of the structures, were performed by modal analysis floor response spectra method.

The floor response spectra curves were generated by subjecting the building model to the selected base excitation, then determining the maximum response of a single degree of freedom system, of varying natural period of vibration and for several values of damping, at each floor elevation.

5A.2.1.2 Typical Category I Equipment and Systems

a. Reactor vessel and internals including control rods and control rod drives
b. Other primary coolant system components (steam generators, pressurizer, pumps, etc.) and piping, including vent and drain piping inside the Containment
c. Containment penetrations up to and including the first isolation valve outside the Containment
d. Atmospheric dump and main steam safety valves and associated piping from main steam headers
e. Penetration room ventilation ducting
f. Spent fuel storage racks
g. Auxiliary feedwater pump, condensate storage tank, and associated piping
h. Main emergency generator including fuel supply
i. Control boards, switchgear, load centers, batteries, and cable runs serving Category I equipment
j. Unit 1 and 2 cable spreading room cable tray support systems are qualified by analysis and tests per Section 5A.3.1.5.11 and IEEE 344 1971/Later
k. Critical Service Water System
l. Critical Saltwater System
m. Containment Spray System, including refueling water tank
n. Containment structure air cooling system
o. Low-Pressure Safety Injection and Shutdown Cooling System
p. High-Pressure Safety Injection System

q. Chemical and Volume Control System
r. Safety injection tanks and piping
s. Spent fuel pool purification system
t. All equipment in the radioactive waste processing systems except the reactor coolant and miscellaneous waste evaporators (the evaporators are no longer used), the spent resin metering tank, the miscellaneous waste monitor tank and associated piping and the radioactive waste processing skid. UFSAR Section 14.23 describes the supporting analysis for the waste processing system components that are postulated to fail due to a seismic event. Although failure of the liquid waste processing system components downstream of 1/2CV-4260 have been analyzed for dose consequences in UFSAR Section 14.23, the effects of flooding on components located outside of Auxiliary Building Room 418 have not been analyzed. As a result, the liquid waste processing system, exclusive of the components listed above, will be maintained Seismic Category I.)

Calvert Cliffs has no Category I tunnels or underground cells. For the design methods and actual arrangement of Category I underground piping, refer to Section 5A.3.2.1.

Category I underground cabling supplies power to the saltwater pumps. These cables are located in reinforced concrete conduit duct banks extending from the Category I steam generator auxiliary feed pump room to the Category I intake structure. Both ends of the duct bank penetrate Category I structure wall, where the cables rise in conduit and terminate at the switchgear. It is not considered credible that service continuity to the saltwater pumps would be interrupted due to slight relative movements between the duct bank and the Category I building connections. A Category I buried duct bank runs between the new Diesel Generator Building and the Auxiliary Building for the electrical distribution for Diesel Generator 1A. Portions of this buried duct bank are also common to the Station Blackout Diesel.

Where Category I structures are directly connected to Category II items such as equipment and piping systems, Category II systems are restrained in such a way that damage or excessive movement of Category II items will not adversely affect the Category I structures or equipment.

5A.2.2 SEISMIC CATEGORY II

Category II structures, systems, and equipment are those whose failure would not result in the release of radioactivity and would not prevent reactor shutdown. The failure of Category II structures, systems, and equipment may interrupt power generation.

All Category II structures are located at a sufficient distance away from Category I structures such that the failure or excessive movement of Category II structures will not cause the failure of the Seismic Category I structures.

5A.3 DESIGN BASES

5A.3.1 CATEGORY I STRUCTURE DESIGN

5A.3.1.1 Time History

The time-history analysis was utilized in the seismic analysis of Category I structures to construct the floor response-spectrum curves. These curves were used as input in the evaluation of equipment and piping systems. The response-spectrum curves were generated utilizing El Centro, California, 1940 earthquake. Horizontal (East-West) components were scaled down to the 8% gravity maximum acceleration specified for the OBE. The vertical spectrum response component was scaled down to the 5.3% gravity maximum acceleration specified for the OBE.

The time-history response at the mass points consists, in general, of a superposition of the time-varying responses admitted through narrow frequency bands whose central frequencies are the natural frequencies of the system. The ground spectra curves of El Centro E-W component do envelope the design response spectra with a considerable margin at or near the natural frequencies of all Category I structures, thus assuring conservatism of the earthquake time history.

A parametric study was performed to investigate the effects of the variations in the basic time-history input. The El Centro E-W time-history accelerograph was modified by passing it successively through arbitrary filters, so that the new accelerograph produced a smooth ground spectra which completely enveloped the 2% design spectra reducing the valleys and peaks. This modified time history was then applied to the structures and new floor response curves obtained. The resulting curves were compared to those produced by the actual earthquake and found to be very similar in shape, but lower in magnitude at all frequencies. It was concluded that the use of the original earthquake time history was more conservative.

The fact that the original earthquake time-history responses are higher than those of design response spectra attests to the above conclusion.

The Containment Structure seismic analysis results obtained by both the time-history and the response-spectrum methods are provided in Tables 5A-2, 5A-3, and 5A-4. Table 5A-1 identifies the location of the structural model points for which responses are tabulated. The results of the response-spectrum method for the 8% gravity OBE, and for the 15% gravity SSE, are presented in Tables 5A-2 and 5A-3, respectively. Table 5A-2 presents the responses to El Centro E-W component that has been linearly scaled down to have an 8% gravity maximum acceleration, and used as the operating basis time history. The tabulated responses indicate that:

a. The relative response of structural points with respect to each other compare well for both methods.
b. The results of the 8% gravity time history are nearly equivalent to the results of the frequency response method for 15% gravity, thus reflecting the conservatism in use of the El Centro earthquake.

5A.3.1.2 Ground Spectra Curves

The ground spectra curves of the El Centro E-W component are shown in Figures 5A-1 through 5A-4. These curves are drawn over the design response spectra curves for comparison and are considerably above the spectra in the damping range of 2 to 5% of the critical, reflecting a high safety margin.

Computations of the spectrum curves from the digitalized time-history records were accomplished utilizing Bechtel Computer Program CE-791. This program is basically a fourth order Runge-Kutta stepwise numerical solution of the equation of motion and calculates, prints, and plots the absolute values of maximum displacements, velocities, and spectral accelerations for a total of 124 characteristic frequencies ranging from 0.1 to 25.0 CPS, for a given critical damping value.

5A.3.1.3 Confidence Limits

The confidence limits of the fourth order Runge-Kutta algorithm exceed the round-off error encountered in the single-precision accuracy of GE-635 (a digital electronic computer used by Bechtel) for the above frequency range and for the 0.01 second time interval on which the digital accelerogram is based. The frequency interval, in CPS, is 0.05 between frequencies 0.1 and 1.0, 0.10 between frequencies 1.00 and 10.0, and 1.00 between frequencies 10.0 and 25.0, thus describing the spectrum with 124 coordinates. These frequency intervals produce spectra which compare well with spectra produced by analog computers for all damping values, except 0.0% of critical for which no interval can be defined with certainty.

5A.3.1.4 Amplified Response Loading for Structure and Floors

The response of the structure and floors to the SSE and OBE was computed using the response-spectrum technique. This method, described in Section 5.1.3.2.b, computes the horizontal response in the direction that will give the maximum stresses.

For the response in vertical direction, the structure was reduced into a one-degree-of-freedom system. The responses (shears, moments, and inertia forces) in both vertical and horizontal directions were combined to produce maximum loading.

Since the frequencies of a structure cannot be precisely determined because of material property variations, lumping of masses, idealization of stiffness (both structure and foundation), and nonlinear response characteristics of actual structures, the spectrum response curves obtained for the above-ground elevations in structures were modified by a shifting of the peak response by an approximate amount of 10% on both sides of the original curve. This was done to ensure against any uncertainties of the important variables in the structural model.

A multi-mass and multi-degree of freedom analysis was the original intent for determining the response of Category I structures in the vertical direction. However, the thick concrete members of the structures produced a large extensional stiffness in the vertical direction with respect to the relatively lower soil stiffness and, hence, resulted in a singular flexibility matrix. An investigation showed that the vertical degrees of freedom were constrained, that is, they moved approximately the same amount at the same time. Therefore, the constrained vertical degrees of freedom were reduced into one independent vertical degree of freedom for the response spectra generation.

The natural frequencies of the structure and the location of the amplified spectral values were subjected to some error due to assumptions made in the development of the analytical model, the possible variation in the building mass, and the stiffness calculated for the analytical model. The spectrum response curves were modified by shifting a peak response by an approximate amount of 10% on both

sides of the original curve. This was done to ensure against any uncertainties of the important variables.

5A.3.1.5 Amplified Response Loading for Equipment and Components

The spectrum response curves, for Category I equipment and components, were generated using the time-history technique of seismic analysis. These curves were generated for two horizontal directions (North-South and East-West) and for the vertical direction, using various damping values at designated floor elevations (see Table 5A-8).

The seismic analyses are based on elastic and linear behavior of all components involved, and as such, does not include any gradual or accidental deterioration of the structure. The blowdown forces associated with a concurrent LOCA are computed separately and combined with the seismic loads.

The following was included in the analysis of all Category I equipment design:

a. Applicable response curve information from the specification concerning equipment location, and the appropriate damping value (see Table 5A-8).
b. Evaluation of the natural frequency(s) in both horizontal and vertical directions. The horizontal frequency(s) shall be computed in the direction that will give the highest stresses.
c. Utilizing the natural frequency(s), enter the applicable spectrum response curve to ascertain the comparable response acceleration.
d. For equipment and systems modeled as multi-degree systems, the acceleration per mode shall be combined by the normal mode method. The effect of adding closely spaced modes linearly has been investigated. It has been found that combining stress components by this method does not exceed the allowable stress levels.
e. The horizontal and vertical seismic forces are equal to the mass of the equipment times the respective spectrum acceleration.
f. Horizontal and vertical forces are applied simultaneously at the center of gravity of the equipment, or at lumped mass point.
g. Seismic stresses shall be computed and combined with all other stresses that might exist in critical components.
h. Stresses from the OBE, when combined with normal operating stresses, shall be kept at or below the applicable code allowable stresses.
i. Stresses from the SSE, when combined with normal operating stress, shall be kept at or below the applicable code allowable stresses, and shall be kept at or below yield strength of the material provided that no loss of function can occur.
j. For equipment, where analysis is not reliable, vibration tests shall be employed to verify the seismic adequacy of the equipment. The test procedure shall be submitted for approval.

5A.3.1.6 Amplified Response Loading for Piping and Instrumentation

A multi-mass response-spectrum, modal analysis method was employed in the seismic analysis of Category I piping, support systems and instrumentation. American Society of Mechanical Engineers (ASME) Code Case N-411 may be used to take advantage of the flexibility in piping systems (Section 5A.3.2.2). The natural frequencies, mode shapes, and the maximum response accelerations were determined using the appropriate response-spectrum curves in the horizontal and vertical direction. The response-spectrum curves are generated using the time-

history of the floor, which includes the seismic response of the building. The horizontal and vertical seismic forces were applied simultaneously. Shear stresses, moments, and deflections were determined for the piping system and restraints. The load and stresses due to seismic loadings were assumed to be acting simultaneously with operating weights and longitudinal pressure loads.

5A.3.1.7 Normal Operation

For loads to be encountered during normal plant operation (including OBE loads), Category I structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

5A.3.1.8 Loss-of-Coolant Accident, Seismic and Tornado Loads

The Category I structures are in general proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, LOCA loads, seismic and tornado loads. The upper limit of elastic behavior, considered to be the yield (Y) for reinforced concrete structures, is considered to be the ultimate resisting capacity as calculated from the "Ultimate Strength Design" portion of the ACI-318-63 code when ϕ is taken as unity. Reinforced concrete structures are designed for ductile behavior whenever possible, i.e., with steel stresses controlling the design. For structural steel, the allowable stresses are per the American Institute of Steel Construction Manual, 6th Edition.

The load factors and load combinations represent the consensus of a group of individual engineers and consultants who are experienced in both structural and nuclear power plant design. Additionally, their judgments have been influenced by current and past practice, by the degree of conservativeness inherent in the basic loads, and particularly by the probabilities of coincident occurrences in the case of incident, wind, and seismic loads.

Factored load equations used for the design of Category I structures, except the Containment Structure, are in accordance with the equations presented in the ACI-318-63 code. The following discussion will explain the justification of individual load factors:

LOAD FACTORS

a. Dead Load -- Dead load in a large structure such as this is easily identified and its effect can be accurately determined at each point in the vessel. For combination with accident and seismic or wind loads, a load factor representing a tolerance of 5% was chosen to account for inaccuracies. The ACI code allows a tolerance of +25% and -10%, but was written to cover a variety of conditions where weights and configurations of materials in and on the structure may not be clearly defined and are subject to change during the life of the structure.
b. Live Load -- The live load in combination with accident, seismic, and wind loads produces a very small portion of the stress at any point. It is extremely unlikely that the full live load would be present over a large area at the time of an unusual occurrence. Therefore, a low load factor is felt to be justified, and live load will be considered together with dead load at a load factor of 1.05.
c. Seismic -- The design earthquake is considered to be the strongest probable earthquake which could occur during the life of the plant. In addition, a maximum earthquake which could occur at the site is also considered in design. Category I structures are designed so that no loss of function would result. Consequently, the probability of a LOCA is very

small. For this reason the two events, seismic and LOCA, are considered together, but at much lower load factors than those applied separately. The earthquake load factors of 1.25 and 1.0 are conservative for the design and maximum earthquakes combination with the factored LOCA.

d. Winds -- Loads are determined from the design tornado wind speed. The Containment Structure is designed for this extreme wind and it is inconceivable that it would cause a LOCA. Therefore, wind loads will not be considered with incident loads, but a factor of 1.25 will be applied to the tornado load to provide assurance of the structure performing satisfactorily.

e. Loss-of-Coolant Accident -- The design pressure and temperature are based on the operation of partial safeguards equipment using emergency diesel power.

   European practice has been to use a load factor of 1.5 on the design pressure (Reference 1). This factor is reasonable and has been adopted for this design. The probabilities of a LOCA occurring simultaneously with a maximum wind or seismic disturbance are very small; therefore, a reduced load factor of 1.25 is used.

   In all cases the design temperature is defined as that corresponding to the factored pressure of 1.5P, and will be somewhat higher than that temperature at P. A temperature factor of 1.5 is unrealistic since this could only occur with a pressure much greater than that of 1.5P.

The ϕ factors are provided to allow for variations in materials and workmanship. In ACI Code 318-63, ϕ varies with the type of stress or member considered; that is, with flexure, bond or shear stress, or compression.

The ϕ factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength assuming materials are as strong as specified, sizes are as shown on the drawings, the workmanship is excellent, and the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables as indicated above by specifying appropriate ϕ values. These values are larger for concrete flexure because the variability of steel is less than that of concrete and the concrete in compression has a fail-safe mode of behavior; that is, material understrength without failure. The values for columns are lower (favoring the toughness of spiral columns over tied columns) because columns fail in compression where concrete strength is critical. It is possible that the analysis might not combine the worst combination of axial load and moment. Since the column member is critical in the gross collapse of the structure, a lower value is used.

The additional ϕ values used represent the best judgment of how much understrength should be assigned to each material and condition not covered directly by the ACI code, and have not been selected based on material quality in relation to the existing ϕ factors.

Conventional concrete design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI used ϕ = 0.90 to reinforcing steel which now includes axial tension. The code recognizes the possibility of reduced bond of bars at the laps by specifying a ϕ of 0.85.

Mechanical and welded splices will develop at least 125% of the yield strength of the reinforcing steel. Therefore, ϕ = 0.85 is recommended for this type of splice.

The only significantly new value introduced is ϕ = 0.95 for prestressed tendons in direct tension. The higher ϕ value has been allowed because: (1) during installation, the tendons are each jacked to about 94% of their yield strength so, in effect, each tendon has been proof tested; and (2) the method of manufacturing prestressing steel (cold drawing and stress relieving) ensures a higher quality product than conventional reinforcing steel.

The final design of Category I structures (except the Containment Structure) satisfied the most severe of the following load combination equations.

$S \geq D + L + T + E$
$Y \geq 1/\phi\ (1.25D + 1.0R + 1.25E)$
$Y \geq 1/\phi\ (1.25D + 1.25T + 1.25E)$
$Y \geq 1/\phi\ (1.0D + 1.0R + 1.0E')$
$Y \geq 1/\phi\ (1.0D + 1.0T + 1.0E')$

The final design of the Containment Structure satisfies the following load combinations and factors (factored load cases):

$Y \geq 1/\phi\ (1.05D + 1.5P + 1.0T_A + 1.0F)$
$Y \geq 1/\phi\ (1.05D + 1.25P + 1.0T_A + 1.25H + 1.25E + 1.0F)$
$Y \geq 1/\phi\ (1.05D + 1.25H + 1.0R + 1.0F + 1.25E + 1.0T_o)$
$Y \geq 1/\phi\ (1.05D + 1.25H + 1.0F + 1.25W + 1.0T_o)$
$Y \geq 1/\phi\ (1.0D + 1.0P + 1.0T_A + 1.0H + 1.0E' + 1.0F)$
$Y \geq 1/\phi\ (1.0D + 1.0H + 1.0R + 1.0E' + 1.0F + 1.0T_o)$

(Wind, W, is to replace earthquake, E, in the above formula where wind stresses control)

(0.90 D is used where dead load reduces the critical stress in the first two equations).

Limiting yield allowables and allowable erosion of barriers under all load conditions, including the SSE (E') and jet or missile forces, respectively; is acceptable; provided deflections are also checked to ensure the affected Category I systems and equipment do not suffer loss of function. The Containment Structure must also retain its required leak-tight integrity under LOCA loadings.

S = Strength of structures with stresses ≤ their applicable code allowable value.

Y = Strength of structure with stresses not exceeding their yield value. For structural steel, it is the minimum specified yield strength. For reinforced concrete, it is the reinforcement yield strength. All accident loads, R, are considered purely dynamic and an increase in the static yield strength of the material have been given consideration as recommended in the American Society of Civil Engineers (ASCE) manual #42.

D = Dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. In addition, a portion of "live load" is added when such load is

expected to be present when the plant is operating. An allowance is also made for future permanent loads.

L = Live Loads

R = Forces on structure due to a rupture of any one primary coolant pipe including the following:

Reaction forces transmitted to the major component support structures.

Jet impingement force on the structural component that can become a target to the ensuing jet.

Differential pressures that can develop across structural compartments.

H and T = force on structure due to thermal expansion of restrained pipes under operating conditions.

E = OBE load from horizontal ground acceleration of 0.08 g.

E' = SSE load from horizontal ground acceleration of 0.15 g.

W = Tornado wind load.

P = LOCA pressure load.

F = Final prestress load.

T_A = Thermal load due to the incident temperature gradient through the wall and expansion of the liner. It is based on a temperature corresponding to the factored LOCA pressure.

T_o = Thermal load due to the normal operating temperature gradient through the walls.

ϕ = Yield capacity reduction factor as follows:

0.90 for reinforced concrete in flexure.
0.85 for tension, shear, bond, and anchorage in reinforced concrete.
0.75 for spirally reinforced concrete compression members.
0.70 for tied compression members.
0.90 for fabricated structural steel.
0.90 for unprestressed reinforcing steel in direct tension.
0.95 for prestressed tendons in direct tension.

The Containment Structure and engineered safety system components are protected by barriers from all credible missiles that might be generated from the primary system during a LOCA.

The final design of the missile barrier and equipment support structures inside the Containment were reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe reactions and earthquake loads without loss of function. The deflections or deformations of structures and supports were checked to assure that the functions of the Containment and Engineered Safety Features are not impaired.

SAFETY FACTORS

The safety factor that would result from the factored load equations of ACI 318-63 code are based on the ratio of live load to dead load. This factor varies from 1.67 to 1.93 for a ratio of live load to dead load factor of 4. The safety factor that is actually provided from the use of the factored load equations used for the design of Containment Structure depends on the ratio of various loading combinations to

the dead load. The safety factors are well within the range specified in ACI 318-63 code, to make failures very unlikely.

5A.3.1.9 Tornado Forces

All Category I structures and critical components of Category I structures are designed to resist a lateral force caused by a tornado having a velocity of 300 mph and a forward progression of 60 mph. There are no removable shielding blocks located on the external boundary of any Category I structure.

Category I structures are designed to resist the effects of a tornado. These structures are analyzed for tornado loading (not coincident with LOCA or earthquake) on the following basis:

a. Differential bursting pressure between the inside and outside of the Containment Structure is assumed to be 3 psi positive pressure.

   For the safety related Diesel Generator Building, the tornado-induced pressure differential is applied as a 3 psi positive pressure within the Diesel Generator Buildings occurring in 1.5 seconds (2 psi per second) followed by a calm for two seconds and then a repressurization to atmospheric pressure at a rate of 2 psi per second.

b. Lateral force is assumed as the force caused by a tornado funnel having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph. The applicable portions of wind design methods described in ASCE Paper 3269 are used, particularly for shape factors. The provisions for gust factors and variation of wind velocity with height do not apply.

   For the safety-related Diesel Generator Building, the velocity components are applied as a funnel of wind traveling at 70 mph with a maximum tangential velocity of 290 mph (giving a total effective wind velocity of 360 mph).

c. Torsion of the Containment Structure is computed from the drag on a cylinder resulting from a 300 mph rotary wind centered over the structure.

d. A tornado-driven horizontal (i.e., no vertical velocity component) missile equivalent to a 4000 pound automobile flying horizontally through the air at 50 mph and at not more than 25' above the ground or a 4"x12"x12'-long piece of wood traveling end-on at 300 mph at any height.

   At the time of the original design and licensing of the plant, design-criteria for non-horizontal missiles (i.e., missiles with a vertical velocity component) did not exist and the site is not committed to any specific criteria for vertical missiles; with one exception being the safety-related Diesel Generator Building. Changes to the plant design will provide a similar level of protection as existed in the original licensed design. Since the safety-related Diesel Generator Building was constructed subsequent to the original design and licensing design of the plant, its design criteria is specifically mentioned below.

   The safety-related Diesel Generator Building is designed for missile impingement for the missiles given in Table 5-8. A vertical velocity of 70 percent of the postulated horizontal velocity is used for all missiles except for steel rods. A steel rod missile's vertical velocity is assumed to be equal to the horizontal velocity.

e. All exposed Containment penetrations are designed for tornado forces.

f. The possible increase of the tornado loading on the Containment Structure due to the funneling effect of the tornado wind blowing between the two Containment Structures has also been considered.

Except for local crushing at the missile impact area, the allowable stresses to resist the effects of tornadoes are 90% of the yield of the reinforcing steel and 85% of the ultimate strength of the concrete.

Tornado-generated missile protection is not required for systems designed to meet the performance standards of draft General Design Criteria 2 if the resultant aggregated probability of exposures in excess of 10 CFR Part 100 guidelines is less than 10^{-6} per year per unit. The aggregate probability includes reasonable qualitative arguments, or conservative assumptions, such that the realistic probability can be shown to be lower than the calculated value.

Probabilistic Evaluation Techniques were used to determine the probability of exposures in excess of 10 CFR Part 100 guidelines due to tornado missiles. The evaluation was performed for the Emergency Diesel Generator (EDG) Nos. 1B, 2A, and 2B engine air intake filter and exhaust piping and muffler, EDG No. 1A exhaust ducts, the 11, 12, 21, and 22 auxiliary feedwater (AFW) turbine exhaust piping, main steam safety valve (MSSV), and atmospheric dump valve (ADV) vent stacks, service water head tanks and sections of service water piping in the Turbine Building, saltwater pumps and piping in the intake structure, and the 21 Fuel Oil Storage Tank (FOST) vent/flame arrestor on top of the 21 FOST Building. The evaluation determined that the aggregate probability of exposures in excess of 10 CFR Part 100 guidelines is less than 5E-07 per unit per year (Table 5A-5). The key parameters in the evaluation which could be affected by site activities are the total missile population, modifications that significantly change the size of the component target areas, and changes to plant equipment, procedures, or practices which would affect the Probabilistic Risk Assessment. This analysis conservatively assumed that:

1. All loss of offsite power is not recoverable following a tornado.
2. Tornado missile damage is not recoverable.
3. Guaranteed failure of the component is assumed to occur from a tornado missile strike to the exposed components for the Saltwater System, Service Water System, turbine-driven AFW system, and the 21 FOST.
4. A tornado missile strike to any part of the array of 16 MSSV and 2 ADV vent stacks has a 1 in 10 chance crimping enough stacks to fail the steam generator decay heat removal function.
5. Tornado Point strike frequency is based on NUREG/CR-4661, Revision 2. (This frequency includes relatively recent tornado events and is higher than the tornado point strike frequency previously used for this analysis.) This value is conservative because it includes tornadoes of all intensities, including the smaller and more frequent tornadoes which generally do not produce significant tornado missiles.
6. Core damage will result in exposures in excess of 10 CFR Part 100 guidelines. This assumption is conservative in that it assumes the containment function is guaranteed to fail.

5A.3.1.10 Seismic Forces (E and E')

Atomic Energy Commission publication TID 7024, "Nuclear Reactors and Earthquakes," is used as the basic design guide for seismic analysis. All Category I structures are designed for the loading combinations described in Chapter 5.

The "OBE" used for this plant is a maximum ground acceleration of 0.08 g horizontally and 0.053 g vertically, acting simultaneously. The "SSE" is a ground acceleration of 0.15 g horizontally and 0.10 g vertically, acting simultaneously. The maximum occurring vertical and horizontal accelerations are added directly with stresses developed from other load conditions.

Seismic loads on structures, systems, and equipment are determined by realistic evaluation of dynamic properties and the accelerations from the acceleration spectrum curves in Chapter 2.

Containment Structure Exterior:

The summary of stresses at critical locations in the Containment Structure exterior are listed in Table 5-1. Table 5-1 is the original analysis. See Appendix 5E for later tables associated with an evaluation that reduced the original containment minimum design prestress. The contributions of seismic stresses to total stresses in the Containment Structure exterior are shown in Table 5-1. The summary of stresses in the Containment Structure for different loading combinations such as dead loads, live loads, prestress pressure, temperature are listed in the table with and without seismic stresses. The technique used to combine seismic stresses with other stresses is described in Section 5.1.3.2. It can be seen from Table 5-1 that the contribution of seismic stresses to the total stresses in the Containment Structure varies widely; but, the total stresses including the seismic stresses are well within the allowable stresses. (Table 5-1 is the original analysis. See Appendix 5E for later tables associated with an evaluation that reduced the original containment minimum design prestress.)

Containment Structure Interior:

The maximum contribution of the seismic stresses to the total stresses for the Containment Structure interior is about 66% in the reactor cavity wall during an OBE, but the design of the reactor cavity wall is governed by an accident loading condition for which the contribution of seismic stresses during SSE is only 26% of the total stresses.

Auxiliary Building and Intake Structure:

In other Category I structures such as the Auxiliary Building and the Intake Structure, the contribution of seismic stresses to the total stresses varies widely, but under no conditions do the total stresses, including the seismic stresses, in these structures or their components, exceed the allowable stresses for concrete, reinforcement or structural steel, as defined in Chapter 5.

5A.3.1.11 Torsional Modes of Vibration

Torsional modes were not considered in the seismic modal analysis of Category I structures, but were calculated separately. In symmetrical Category I structures, torsional modes were not present. For non-symmetrical Category I structures, torsional moments were calculated using the following method:

a. The center of twist along with the center of gravity was obtained during the calculation for modal properties. The torsional moments were calculated

as the product of resultant shear force and distance between the center of gravity and the center of twist.

b. Torsional shears were added to the shears due to lateral loading. However, when they acted in the opposite direction to the lateral shears, the greater shear values were used without reduction.

During an OBE and SSE the torsional modes of vibration of a structure may result from: (1) the unsymmetrical plan or elevation of the structure or the irregular arrangement of walls leading to a discord between the center of gravity and that of rigidity, and (2) the earthquake itself arising from rotational characteristics of the earthquake wave. The torsional loads in symmetrical structures are a result of the latter case, and Newmark's paper deals mainly with this type of torsion. Generally the flexural and torsional modes due to the first cause tend to be coupled, while those due to the second cause are not coupled. From the number of papers published to date (April 25, 1972), it can be stated that torsional loads are of significance to conventional multi-storied high-rise buildings with open moment-resisting frame due to their low torsional resistance.

Containment and Auxiliary Building

The structures considered to be Category I are the Containment Structure, Auxiliary Building and intake structure. The Containment Structure and the Auxiliary Building are respectively of closed circular and rectangular sections with concrete wall thicknesses of 3'9" for the former and 2' or more for the latter. The torsional resistance of these structures is considerably higher than that in any conventional structure, especially in the case of the Containment.

Intake Structure

The intake structure consists of concrete walls with thicknesses of 2' or more, but is not of closed sections. Based on theory of strength of materials, the torsional resistance of open sections is less than that of closed sections. In view of this, an analysis of the torsional effects on the intake structure based on the method as described has been made and the increase in shear stress is found to be in the range of 5 to 10%. The horizontal force to cause the torsional shear is that obtained from the uncoupled flexural vibration analysis. It has been found that rectangular buildings with either central cores or peripheral shear walls as resistive elements tend to have relatively uncoupled modes, while a smooth, even distribution of columns can result in strong modal coupling. In the light of such finding, it is believed that the horizontal force thus used in computing the torsional shear is reasonably accurate.

On the basis of the results of the torsional analysis of the intake structure, it was further concluded that the effects of the torsional loads on the Containment Structure and the Auxiliary Building should be of lesser significance and, therefore, no torsional analysis was made on these two structures.

Safety-Related Diesel Generator Building

Torsional effects for the safety-related Diesel Generator Building are accounted for directly in the seismic analysis through the incorporation of torsional degrees of freedom into the models which represent the Diesel Generator Building enclosure, the diesel generator pedestal and the fuel oil storage tank. In addition, torsional effects are represented in the building enclosure model by eccentrically located masses and by offsetting the beam elements representing the stiffness of the various levels in each Diesel Generator Building. Accidental torsion is accounted

for by increasing the mass eccentricities by 5% of the maximum lateral building dimension.

5A.3.1.12 Foundation Isolation Joints

The foundations of the main structures are closely spaced and foundation isolation joints have been provided. Because the maximum separation of foundation is small in comparison with the length of the earthquake wave, it is unlikely that the foundations will be displaced, relative to one another, sufficiently to damage each other.

The differential movement of adjacent structures due to seismic motion was evaluated. The size of the isolation joint is based on the anticipated horizontal movement of the foundation during OBE and SSE. The isolation joint is filled with compressible material to reduce the influence of the foundations of the main structures on each other. Flexible joints are provided between the Containments and Auxiliary Building to serve as water stops and partial air leakage barriers.

5A.3.1.13 Soil/Foundation Interaction

Interaction between soil and the foundation is included in the seismic analysis in the form of soil spring constants. As shown in Figure 5-4 (Sheets 1 and 2), soil below the base slab of the Containment structure was included in the Finite Element Mesh and used in Bechtel's Finite Element Method Analysis computer program, CE 316-4, to compute the stresses generated in the soil and base slab. Figure 5-4 is based on the original analysis. See Appendix 5E for an evaluation that reduced the original containment minimum design prestress.

Stresses generated in the soil were included in the computation for the soil bearing pressure.

The total stresses in the soil, including those during the OBE and the SSE, were well below allowable bearing capacity of the soil. The total stress in the base slab of the Category I structure was also below allowable design values.

The effect of the soil on the sides of the walls below finish grade was negligible and was not included in the analysis. However, the walls below finish grade were designed for the dynamic earth pressure as well as static earth pressure (Section 2.7.6.4).

5A.3.1.14 Design Code References

The design and checking of the design have been made in accordance with the provisions indicated in the ACI Code and Commentary 318-63, Section 2603(a), 2603(b) and ACI Committee 334 (Concrete Shell Structures Practice and Commentary), Section 202(d), 202(e) and Commentary Part 4, except as modified in Updated Final Safety Analysis Report Sections 5.1.2.3 through 5.1.2.6 and Section 5.1.3.2.

5A.3.2 SEISMIC CATEGORY I SYSTEMS AND EQUIPMENT DESIGN

Seismic Category I systems and equipment, including pipe, are designed to meet the load combinations and stresses as stated in Table 4-8 for Nuclear Class 1, and Table 5A-6 for Nuclear Class 2 and 3, and non-class. Seismic Category I systems and equipment are bolted down rigidly to supports or braced (as in the case of cable tray supports) to resist seismic and tornado forces. The NSSS contractor is taking exception to this support approach and the individual supports were designed based on the criteria outlined in Sections 5.1.1 and 5.1.2.3. There are no significant gaps between the equipment and

their supports or restraints. Any small gap will not cause significant impact forces on the equipment, restraints or the structures. Therefore, small gaps between the equipment and supports or restraints are not significant in the consideration of the seismic analysis.

Deformations in support structures will limit strains in piping systems to the criteria stated in Tables 4-8 and 5A-6 for those systems essential to safe shutdown of the plant following a LOCA.

The Containment penetration assemblies are designed to accommodate the forces and moments due to pipe rupture. Guides, pipe stops, increased pipe thicknesses or other means are provided to make the penetration the strongest part of the system.

The mathematical models employed in dynamic (seismic) analysis of the Reactor Coolant System components were formulated using lumped parameter modeling techniques. A single composite model was employed in the analysis of the couple components, which included the reactor vessel assembly, the two steam generators (SGs), the four reactor coolant pumps and the reactor coolant piping. The total mass and related stiffness of each of the coupled components was included in the model. Sufficient mass points were included in the model and, at each mass point, translational dynamic degrees of freedom retained, so dynamic analysis includes the combined vertical, torsional and horizontal response of the system due to seismic excitations.

A separate multi-mass model was employed in the seismic analysis of the pressurizer.

5A.3.2.1 Piping

For the design of Seismic Category I piping and equipment, coefficients were based on the floor response-spectrum curves. These curves were generated using the time-history technique for both horizontal and vertical direction, for various damping values, and at designated floor elevations in the Category I structures. This method is based on a dynamic analysis of multi-degree-of-freedom system. Code Case N-411 of the ASME Boiler and Pressure Vessel (B&PV) Code may be used to take advantage of the flexibility in piping systems (Section 5A.3.2.2).

Buried Pipes

All Category I buried pipes are designed for bending stresses due to ground motion. At the joints, where direction of pipe changes, a cushion of compressible material is provided to accommodate any rotation of the pipe joint.

Above-Ground Pipes

Piping systems are anchored and restrained to floors and walls of buildings. The relative seismic displacements between buildings, between floors in buildings, and between major components are applied to the piping, anchors and restraints. When appropriate, seismic movements are considered to be out of phase between structures and/or major components, thereby evaluating the piping systems for the maximum hypothetical relative displacements. The resulting stresses are classified as secondary and are combined with other secondary stresses. The sum of secondary stresses is held within the limits of the applicable piping code.

Local stress analysis of welded attachments using Code Case N-392-1 (12-11-89) is documented in Table 5A-7 as required by Regulatory Guide 1.84.

5A.3.2.2 Routing of Seismic Category I Piping

The routing of Category I piping is typically confined within and/or attached to Category I structures, such as the Containment Structure or the Auxiliary Building. Also, some Category I piping is routed underground between Category I system components, such as the Fuel Oil Storage Tank, and a Category I structure or component. In addition, classification reevaluations and upgrades have occurred. As a result, a few of the upgraded Category I piping segments, such as the Saltwater Ram's Heads, are located in Category II structures, such as the Turbine Building. In each case the upgraded components and the relocations have been evaluated and found to be acceptable (as-is or with already completed modifications) to perform their required safety functions.

Category II piping such as instrument and plant air, plant heating system water, nitrogen, wash water service, fire protection, and roof drain lines are primarily 2" and smaller piping. The 2" and smaller Category II pipe runs which are routed in close proximity of Category I piping do not have the potential to inflict damage on the Category I piping. Physical separation of larger Category II piping is routed such that its failure would not pose a hazard. Where larger Category II piping whose rupture could pose a hazard is routed near Category I piping, adequate pipe restraints are provided to preclude the possibility of pipe whip damage to the Category I piping.

Category I piping was designed in accordance with B31.1 1967, Power Piping, or B31.7 1969, Nuclear Power Piping. Exceptions are noted in relevant sections of the UFSAR for specific systems and components. Effective August 6, 1985, ASME Code Case N-411, "Alternative Damping Values for Seismic Analysis of Piping, Section III Div. 1 Class 1, 2 and 3 Construction," may be used for new analyses or for reconciliation work on new or existing systems (Reference 2). This case takes advantage of piping system flexibility. See the provisions in the NRC letter dated August 6, 1985, when using this code case. All Category I piping, with the exception of 2" and smaller B31.1 and B31.7 Nuclear Class 2 and 3 piping, was originally designed by Bechtel Power Corporation and included the location of restraints and supports, and determination of loads. The building structure connections were checked by the structural engineering group. The piping support contractor was given all necessary information to design and locate pipe supports, and indicates the location of the supports on Bechtel's piping fabrication isometric erection sketch. These drawings, as well as the support design drawings and field installation were checked by Bechtel Engineering. For 2" and smaller Category I piping, a Bechtel field installation manual was provided so that field engineers could properly design and locate pipe supports and restraints. When Bechtel field engineers had completed their design, drawings were submitted to Bechtel engineering for review. The field did not locate any of the seismic supports or restraints for Category I system equipment or components. This work was done at the CE and Bechtel engineering offices.

5A.3.2.3 Equipment, Personnel, and Escape Locks

The equipment, personnel and escape locks are Category I equipment and are designed for the following accelerations: (OBE)

Lock	Vertical Acceleration	Horizontal Acceleration
Equipment Lock	0.07 g	0.11 g
Personnel Lock	0.08 g	0.12 g
Escape Lock	0.07 g	0.10 g

The acceleration values are multiplied by the normal operating weight of the lock or parts of the lock to obtain the horizontal and vertical components of the earthquake force. Both components are considered acting simultaneously with normal operating loads without exceeding code allowable, at a temperature of 120°F.

The earthquake forces due to the SSE are obtained by multiplying the accelerations above by 1.90. The locks are designed to withstand the simultaneous action of SSE components and accident loads as stated in Chapter 5, at a temperature of 276°F, without exceeding material yield stress nor loss-of-lock function.

5A.3.2.4 Seismic Verification of Equipment

The Unresolved Safety Issue (USI) A-46 methodology, as embodied in the Generic Implementation Procedure (Reference 3), and clarified by the NRC (Reference 4), may be used as an alternate method, in addition to existing methods, for verifying the seismic adequacy of mechanical and electrical equipment for which seismic verification is required. The A-46 methodology applies to new and replacement equipment, and existing equipment which has been walked down according to the Generic Implementation Procedure. The A-46 methodology applies to all 20 classes of equipment, tanks and heat exchangers, and cable and conduit raceways covered by the Generic Implementation Procedure, except for the following:

1. Auxiliary Feedwater Actuation System
2. Portions of the Engineered Safety Features Actuation System installed per Facility Change Request 87-0087
3. Regulatory Guide 1.97 Category I (PAMI) Instrumentation
4. All safety-related items/equipment for EDG No. 1A and its associated building.

In the case of tanks and heat exchangers, the A-46 methodology may be used only for existing tanks and heat exchangers, not for new installations.

5A.3.3 CATEGORY II

5A.3.3.1 Structure Design

Category II structures are designed in accordance with design methods of accepted codes and standards insofar as they are applicable. Wind design (25 psf zone) is in accordance with the Uniform Building Code, with a one-third increase in the allowable stresses. Seismic design is in accordance with the Uniform Building Code. Seismic forces are based on Seismic Probability Zone 3 multiplied by a ratio of 0.08/0.30. A one-third increase in allowable stresses is not allowed.

5A.3.3.2 Systems and Equipment Design

Category II systems and equipment are designed in accordance with design methods of accepted codes and standards. Wind loads (25 psf zone) and seismic loads, where applicable, conform to the requirements of the Uniform Building Code as stated in Section 5A.3.3.1.

5A.3.4 COUPLED CONTAINMENT, CONTAINMENT INTERNAL STRUCTURE, AND REACTOR COOLANT SYSTEM (RCS) SEISMIC ANALYSIS

Seismic response spectra were developed for the Containment Building shell, Containment Building internal structure, and RCS attachment nozzles and connection points using a coupled Containment Building, containment internal structure/RCS model.

Seismic analyses were performed using "state-of-the-art" methodologies, which included the following:

a. Seismic ground motion based on Regulatory Guide 1.60, Revision 1 recommendations.
b. Structural damping based on Regulatory Guide 1.61, Revision 0 recommendations.
c. Guidance as provided in NUREG-0800 (Standard Review Plan) Sections 3.7.1 and 3.7.2 and NUREG/CR-5347.
d. Development of floor response spectra based on Regulatory Guide 1.84, Revision 31, Regulatory Guide 1.122, Revision 1, and ASCE 4-86 guidance.
e. Three-dimensional (3-D) representation of the structures and the RCS.
f. Soil structure interaction analysis.

Because of the axial symmetry of the containment shell, the existing two-dimensional (2-D) model is the same as the 3-D model. Thus, there is no difference between analysis results obtained during the 2-D model versus the 3-D model.

Although the containment shell is considered symmetrical, the interior structure of the Containment exhibits some degree of asymmetry, which affects the overall dynamic properties, resulting in added torsional response. A revised 3-D model was generated for the containment internal structure.

Response spectra were generated at 14 different locations, including the reactor vessel nozzle, SG supports, containment shell, basemat, 45-foot and 69-foot floor elevations.

The response spectra are generated for two earthquake levels, OBE and SSE. Operating basis earthquake response spectra are generated for constant damping values of 1%, 2%, 3%, 4%, 5%, and for the variable damping of Code Case N-411. Safe shutdown earthquake response spectra are generated for constant damping values of 2%, 3%, 4%, 5%, 7%, and for the variable damping of Code Case N-411.

5A.3.4.1 <u>Methodology</u>

<u>Seismic Ground Motion</u>:

The basic seismic input to the building structure was the Regulatory Guide 1.60 ground motion spectra in two horizontal and one vertical direction. The horizontal spectra were normalized to 0.15g for SSE and 0.08g for OBE in both horizontal directions, and to 2/3 of the horizontal acceleration in the corresponding vertical direction as specified in Section 2.6. Uncorrelated acceleration time history functions based on enveloping the 2%, 5%, and 7% shaped Regulatory Guide 1.60 spectra were generated for each of the three directions.

<u>Structural Damping</u>:

The damping values used in the coupled Containment Building (including internal structure)/RCS analysis were in compliance with Regulatory Guide 1.61 and are as follows:

	OBE	SSE
Containment Structure – shell	2%	5%
Containment Structure - internal structures	4%	7%
RCS	2%	3%

Development of 3-D Model:

For this coupled Containment Building, containment internal structure/RCS seismic analysis, the original 2-D model of the Containment Building was revised to create a 3-D stick model. The stiffness and mass properties of the new 3-D model were developed based on information contained in design basis calculations and drawings. The shear center and mass center for each floor elevation were based on design drawings showing the structural details of the walls and floors of the internal structure of the Containment Building. This model includes the 3-D representation of the RCS attached at the appropriate elevations.

The new 3-D models are described as follows:

1. Containment shell stick model – Since the containment shell is axisymmetric, the 3-D containment shell model is the same as the original 2-D model.
2. Containment internal structure stick model – This dynamic model of the internal structures is a multi-branch 3-D stick model. The first stick represents the primary shield walls only. The second stick represents the secondary shield walls only without the SG pedestals. The secondary shield walls stick splits into three branches above Elevation 69'-0". Two branches correspond to the SG boxes and the third branch corresponds to the pressurizer box. The third stick includes two branches modeling the SG pedestals. This 3-D model is based on the actual wall and floor stiffness and masses, allowing for differences between centers of mass and shear centers at each major floor elevation. This modeling captures the torsional response of the containment internal structure.
3. RCS stick model – This is a 3-D stick representation of the RCS incorporating SG dynamic properties. A composite 3-D lumped-mass ANSYS model of the reactor vessel, two SGs, four reactor coolant pumps, and main coolant loop piping is included. In addition, representations of the reactor vessel and SG assemblies used in this model include sufficient detail of the reactor internals and replacement SGs internals to account for possible dynamic interaction between those internal components and the RCS. The RCS stick model is coupled to the reactor building internal structure stick model at appropriate support or restrained elevations. The number of masses and dynamic degrees of freedom are consistent with NRC guidance.

Seismic Analysis:

The seismic analysis is performed using the program SUPER SASSI/PC. The SASSI code uses the substructuring method for analysis, including soil structure interaction.

The following analyses were performed:

1. A fixed-base frequency analysis to determine the fundamental frequencies and associated mode shapes of the 3-D containment stick model, including the RCS model, without the effects of the supporting soil. The frequency results were used as a basis for initial specification of frequencies to perform the soil structure interaction analysis.

2. Time-history analyses, including soil structure interaction, for both OBE and SSE using the Regulatory Guide 1.60 as control motion and Regulatory Guide 1.61 damping values for the structures. The model of the Containment Building was subjected to excitation in the three orthogonal directions of the N–S, E-W, and the vertical direction applied separately. Soil properties for input to SUPER SASSI were determined using the computer program SHAKE91.

From the time-history analyses, time histories of response are generated at the points of interest for the containment structure and containment internal structure. Time histories for these points for each of the three earthquake directions are added algebraically at each time point. These in-structure time history results are then used to generate in-structure response spectra with the computer program SPECTRA, at all points that floor response spectra were originally provided, plus the upper feedwater nozzles of the SGs, for the two earthquake levels of OBE and SSE. The resulting spectra for the three soil cases (best estimate, lower bound, and upper bound properties) were enveloped and then peak broadened. In accordance with Regulatory Guide 1.122 and ASCE 4-86, a broadening factor of ± 15% was used to account for the effects of uncertainties. Additionally, prior to broadening, the peaks were reduced 15%, as directed by ASCE 4-86, in order to prevent the introduction of considerable conservatism within the broadened peak region.

At the base node for the primary shield wall, for each earthquake level of OBE and SSE, three translational and three rotational time histories of acceleration response corresponding to the three orthogonal directions are generated. Three sets of time histories are developed corresponding to the three soil cases (i.e., best estimate, lower bound, and upper bound). These time histories are then used in the Westinghouse detailed RCS model.

5A.3.4.2 Results

The 3-D Containment Building, containment internal structure/RCS seismic analysis generated three sets of time histories at the base node that represent the three soil cases. These time histories are utilized for analyses of the RCS. Also generated were numerous floor response spectra. The peaks of these response spectra are 50-60% lower than the response spectra from the original 2-D containment model. However, some of the areas on either side of the peaks contain a somewhat higher response. The damping values to be utilized for the new time histories and floor response spectra are shown in Table 5A-9.

5A.3.5 REFERENCES

1. T.C. Waters and N.T. Barret, "Prestressed Concrete Pressure Vessels for Nuclear Reactors," J. Brit. Nuclear Society 2, 1963

2. Letter from H. R. Denton (NRC) to A. E. Lundvall, Jr. (BGE), dated August 6, 1985, Use of ASME Code Case N-411

3. Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 2, Corrected February 14, 1992, Seismic Qualification Utility Group (SQUG)

4. Letter from J. G. Partlow (NRC) to G. C. Creel (BGE), dated May 22, 1992, Supplement No. 1 to Generic Letter 87-02 that Transmits Supplemental Safety Evaluation Report No. 2 on SQUG Generic Implementation Procedure, Revision 2, As Corrected on February 14, 1992

TABLE 5A-1

CONTAINMENT SEISMIC ANALYSIS MODEL POINTS

POINT	DESCRIPTION	ELEVATION OF POINT
1	East SG Main Steam Nozzle	EL 87'6"
2	East SG to Snubbers Attachment	EL 75'0"
3	East SG	EL 55'0"
4	East SG Cold Nozzle	EL 37'6"
5	East SG Snubbers to Concrete Attachment	EL 75'0"
6	East Portion of Operating Slab	EL 67'6"
7	East SG Bottom Support Anchorage	EL 26'0"
8	Interior Concrete Structures	EL 44'0"
9	Reactor Vessel Nozzles	EL 37'6"
10	Reactor Vessel Supports	EL 28'0"
11	Primary Shield Wall	EL 18'6"
12	Bottom Slab	EL 8'6"
13	West SG Snubbers to Concrete Attachment	EL 75'0"
14	West Portion of the Operating Slab	EL 67'6"
15	West SG Bottom Support Anchorage	EL 26'0"
16	West SG Main Steam Nozzle	EL 87'6"
17	West SG to Snubbers Attachment	EL 75'0"
18	West SG	EL 55'0"
19	West SG Cold Nozzle	EL 37'6"
20	Containment Shell	EL 163'0"
21	Containment Shell	EL 144'6"
22	Containment Shell	EL 127'0"
23	Containment Shell	EL 108'0"
24	Containment Shell	EL 90'0"
25	Containment Shell	EL 71'0"
26	Containment Shell	EL 50'0"
27	Containment Shell	EL 38'0"
28	Containment Shell	EL 26'0"

TABLE 5A-2

CONTAINMENT RESPONSES BY RESPONSE-SPECTRUM TECHNIQUE FOR OBE

GROUND RESPONSE SPECTRA FOR A MAX HORIZONTAL ACCEL. OF 8% GRAVITY
RESULTS OF SPECTRUM RESPONSE TECHNIQUE
SQUARE ROOT OF THE SUM OF THE SQUARES

POINT	INERTIAL FORCE kips	ACCELERATION g	DISPLACEMENT feet
1	0.32025E 02	0.18838E 00	0.42611E-01
2	0.74922E 02	0.16951E 00	0.37214E-01
3	0.74056E 02	0.14521E 00	0.28714E-01
4	0.10632E 03	0.13203E 00	0.21545E-01
5	0.73381E 02	0.15798E 00	0.36830E-01
6	0.50756E 03	0.14677E 00	0.33584E-01
7	0.12000E 03	0.13487E 00	0.17282E-01
8	0.11608E 04	0.12826E 00	0.24061E-01
9	0.31474E 03	0.12941E 00	0.21693E-01
10	0.25159E 03	0.13154E 00	0.18321E-01
11	0.13270E 03	0.13502E 00	0.15379E-01
12	0.41119E 04	0.14175E 00	0.12919E-01
13	0.76053E 02	0.16373E 00	0.37036E-01
14	0.55810E 03	0.15071E 00	0.33738E-01
15	0.11879E 03	0.13350E 00	0.17168E-01
16	0.33569E 02	0.19746E 00	0.42900E-01
17	0.77786E 02	0.17599E 00	0.37441E-01
18	0.75445E 02	0.14793E 00	0.28839E-01
19	0.10657E 03	0.13234E 00	0.21565E-01
20	0.12940E 04	0.14220E 00	0.91545E-01
21	0.12917E 04	0.11566E 00	0.81372E-01
22	0.39734E 03	0.92098E-01	0.71656E-01
23	0.31449E 03	0.71909E-01	0.61140E-01
24	0.27536E 03	0.62963E-01	0.51247E-01
25	0.32372E 03	0.68468E-01	0.40949E-01
26	0.34375E 03	0.88127E-01	0.29918E-01
27	0.29112E 03	0.10262E-00	0.23957E-01
28	0.53101E 03	0.11822E-00	0.18528E-01

TABLE 5A-3

CONTAINMENT RESPONSES BY RESPONSE-SPECTRUM TECHNIQUE FOR SSE

GROUND RESPONSE SPECTRA FOR A MAX HORIZONTAL ACCEL. OF 15% GRAVITY
RESULTS OF SPECTRUM RESPONSE TECHNIQUE
SQUARE ROOT OF THE SUM OF THE SQUARES

POINT	INERTIAL FORCE kips	ACCELERATION g	DISPLACEMENT feet
1	0.47902E 02	0.28178E 00	0.72303E-01
2	0.11243E 03	0.25437E 00	0.63029E-01
3	0.11158E 03	0.21878E 00	0.48381E-01
4	0.16016E 03	0.19888E 00	0.35937E-01
5	0.11044E 03	0.23776E 00	0.62445E-01
6	0.76546E 03	0.22135E 00	0.56879E-01
7	0.18036E 03	0.20270E 00	0.28302E-01
8	0.17517E 04	0.19354E 00	0.40419E-01
9	0.47427E 03	0.19500E 00	0.36239E-01
10	0.37838E 03	0.19784E 00	0.30227E-01
11	0.19927E 03	0.20276E 00	0.24861E-01
12	0.61672E 04	0.21261E 00	0.20134E-01
13	0.11430E 03	0.24607E 00	0.62761E-01
14	0.84083E 03	0.22706E 00	0.57113E-01
15	0.17854E 03	0.20066E 00	0.28125E-01
16	0.50122E 02	0.29484E 00	0.72745E-01
17	0.11656E 03	0.26372E 00	0.63377E-01
18	0.11361E 03	0.22276E 00	0.48572E-01
19	0.16053E 03	0.19934E 00	0.35968E-01
20	0.20993E 04	0.23069E 00	0.15681E 00
21	0.21253E 04	0.19029E 00	0.13943E 00
22	0.66603E 03	0.15438E 00	0.12281E 00
23	0.53725E 03	0.12285E 00	0.10481E 00
24	0.46568E 03	0.10648E 00	0.87843E-01
25	0.51936E 03	0.10985E 00	0.70126E-01
26	0.52673E 03	0.13504E 00	0.51016E-01
27	0.44071E 03	0.15535E 00	0.40561E-01
28	0.79903E 03	0.17789E 00	0.30840E-01

TABLE 5A-4

CONTAINMENT RESPONSES BY TIME HISTORY METHOD FOR OBE

					MAXIMUM TIME HISTORY VALUE			
POINT	RELATIVE DISPLACEMENT feet	TIME second	RELATIVE VELOCITY feet/sec	TIME second	RELATIVE ACCELERATION g	TIME second	ABSOLUTE ACCELERATION g	TIME second
1	0.57680E-01	11.790	0.47323E 00	11.560	-0.27655E 00	4.100	-0.31580E 00	4.100
2	-0.50162E-01	12.340	0.41154E 00	11.560	-0.24317E 00	4.100	-0.28241E 00	4.100
3	-0.38167E-01	12.340	-0.32876E 00	6.050	-0.19179E 00	4.100	-0.23403E 00	4.090
4	0.27781E-01	5.670	-0.28418E 00	6.050	0.15859E 00	6.130	-0.19794E 00	4.080
5	-0.49854E-01	12.330	0.39791E 00	11.560	-0.22220E 00	4.100	-0.26458E 00	4.090
6	-0.45353E-01	12.330	0.35948E 00	11.560	-0.20160E 00	4.090	-0.24403E 00	4.090
7	0.23268E-01	5.670	-0.26505E 00	6.050	0.15375E 00	6.130	-0.19224E 00	4.080
8	-0.31748E-01	12.340	-0.29188E 00	6.050	0.15832E 00	6.130	-0.19701E 00	4.080
9	-0.28062E-01	12.340	-0.28193E 00	6.050	0.15599E 00	6.130	-0.19448E 00	4.080
10	0.24335E-01	5.670	-0.26695E 00	6.050	0.15250E 00	6.130	-0.19083E 00	4.080
11	0.20990E-01	5.670	-0.25350E 00	6.050	0.15020E 00	6.130	-0.18880E 00	4.080
12	0.17374E-01	5.670	0.24315E 00	5.580	0.15282E 00	5.500	-0.19016E 00	4.080
13	-0.50036E-01	12.340	0.40461E 00	11.560	-0.23077E 00	4.100	-0.27223E 00	4.090
14	-0.45491E-01	12.340	0.36433E 00	11.560	-0.20717E 00	4.090	-0.24959E 00	4.090
15	0.23099E-01	5.670	-0.26280E 00	6.050	0.15229E 00	6.130	-0.19071E 00	4.080
16	-0.57940E-01	12.340	0.48288E 00	11.560	-0.28807E 00	4.100	-0.32731E 00	4.100
17	-0.50365E-01	12.340	0.41881E 00	11.560	-0.25184E 00	4.100	-0.29109E 00	4.100
18	-0.38275E-01	12.340	-0.33140E 00	6.050	-0.19602E 00	4.100	-0.23776E 00	4.090
19	0.27618E-01	5.670	-0.28459E 00	6.050	0.15892E 00	6.130	-0.19835E 00	4.080
20	0.13972E-00	11.800	-0.81571E 00	12.020	-0.20422E 00	11.810	0.23952E 00	24.830
21	0.12381E-00	11.800	-0.72925E 00	12.020	-0.17635E 00	11.820	0.21776E 00	1.870
22	0.10856E-00	11.800	-0.64666E 00	12.020	-0.15072E 00	11.830	0.20146E 00	1.870
23	0.91971E-01	11.800	-0.55685E 00	12.020	-0.12578E 00	11.850	0.18373E 00	1.870
24	0.76249E-01	11.800	-0.47181E 00	12.030	-0.10575E 00	11.860	0.16690E 00	1.870
25	0.59671E-01	11.800	-0.38238E 00	12.030	-0.10468E 00	5.690	0.14913E 00	1.870
26	-0.41472E-01	12.330	-0.28361E 00	12.030	-0.12066E 00	5.690	0.15232E 00	25.310
27	-0.32545E-01	12.330	-0.25791E 00	6.050	-0.12941E 00	5.690	-0.16089E 00	4.070
28	0.23989E-01	5.670	-0.25166E 00	6.050	0.13807E 00	6.130	-0.17276E 00	4.070

TABLE 5A-5

ASSESSMENT OF PROBABILITY OF EXPOSURES IN EXCESS OF 10 CFR PART 100 FOR EQUIPMENT LOCATIONS WITHOUT TORNADO-GENERATED MISSILE RESISTANT BARRIERS

NUMBER	DESCRIPTION	LOCATION	PROBABILITY OF EXPOSURE IN EXCESS OF 10 CFR PART 100 GUIDELINES PER YEAR PER UNIT
1	EDG Nos. 1B, 2A, and 2B engine intake air filter and exhaust piping and muffler	Exposed components located on the roof of the Auxiliary Building	< 5E-08
2	AFW turbine exhaust piping	Portion of piping running from the floor of the 27' in the Turbine Building out through the roof of the Auxiliary Building on the associated Unit.	< 1E-08
3	MSSV and ADV vent stacks	Portion of vent stack from floor of the 69' Elevation in the main plant exhaust equipment room out through the roof of the Auxiliary Building on the associated Unit.	< 5E-08
4	SRW head tanks and exposed piping	SRW head tanks in the main plant exhaust equipment room (69' Elevation) and exposed SRW piping on the main generators in the Turbine Building (45' Elevation). For Unit 1 only, the SRW piping to the condensate booster pumps in the Unit 1 Turbine Building 12' Elevation.	< 1E-07
5	Saltwater pumps and piping	Below the intake structure roof pump access hatches on each unit.	< 1E-07
6	21 FOST vent	On roof of 21 FOST Building	<1E-09
7	EDG No. 1A exhaust ducts	Attached to the safety-related Diesel Generator Building structure	<1E-07
Aggregate Probability (per Section 5A.3.1.9, acceptable if less than < 1E-06.)			< 5E-07

TABLE 5A-6

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS FOR NUCLEAR CLASS 2 AND 3 PIPING

LOADING COMBINATIONS	PRIMARY STRESS LIMITS Vessels[d]	Piping	Supports
1. Design Loading + OBE	$P_M \le S_M$	$P_M \le 1.2\ S_h$	Working Stress
	$P_B \le P_L \le 1.5\ S_M$	$P_B + P_M \le 1.2\ S_h$	
2. Normal Operating Loadings + Safe Shutdown + Earthquake	$P_M \le S_D$	$P_M \le S_D$	Within Yield
	$P_B \le 1.5\left[1 - \frac{(P_M)^2}{(S_D)^2}\right] S_D$ [(b)]	$P_B \le \frac{4}{\pi} S_D Cos\left(\frac{\pi}{2} \bullet \frac{P_M}{S_D}\right)$ [(c)]	
3. Normal Operating Loadings + Pipe Rupture + Safe Shutdown Earthquake	$P_M \le S_L$	$P_M \le S_L$	Deflection of supports limited to maintain supported equipment within limits shown in columns (1) and (2)
	$P_B \le 1.5\left[1 - \frac{(P_M)^2}{(S_L)^2}\right] S_L$ [(b)]	$P_B \le \frac{4}{\pi} S_L Cos\left(\frac{\pi}{2} \bullet \frac{P_M}{S_L}\right)$ [(a),(c)]	

(a) These stress criteria are not applied to the piping run within which a pipe break is considered to have occurred.

(b) For loading combinations 2 and 3, stress limits for vessel, with symbol P_M changed to P_L, should also be used in evaluating the effects of local loads imposed on vessels and/or piping.

(c) The tabulated limits for piping are based on a minimum "shape factor." These limits may be modified to incorporate the shape factor of the particular piping being analyzed.

TABLE 5A-6

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS FOR NUCLEAR CLASS 2 AND 3 PIPING

Legend:

P_M	=	Calculated Primary Membrane Stress
P_B	=	Calculated Primary Bending Stress
P_L	=	Calculated Primary Local Membrane Stress
S_M	=	Tabulated Allowable Stress Limit at Temperature from ASME Boiler and Pressure Vessel Code, Section III or ANSI B31.7
S_Y	=	Tabulated Yield at Temperature, ASME B&PV Code, Section III
S_D	=	Design Stress
	=	S_Y (for ferritic steels)
	=	1.2 S_M (for austenitic steels)
S_L	=	$S_Y + 1/3 (S_u - S_Y)$
S_u	=	Tensile Strength of Material at Temperature
S_h	=	Tabulated Yield at Temperature, from U.S.A. Standard B31.1

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits Units are 10^3 lbs/in^2. These values are illustrative and not necessarily those to be used in design.

Material	**S_y[1]**	**S_u**	**S_D**	**S_L**
A-106B	24.5	60.0 [2]	25.4	36.9
SA-533B	41.4	80.0 [2]	41.4	54.3
304 SS	17.0	54.0 [3]	18.35	29.3
316 SS	18.5	58.2 [3]	22.2	31.7

(1) From ASME B&PV Code, Section III, at 650°F.

(2) Minimum value at room temperature which is approximately the same at 650°F for ferritic materials.

(3) Estimated.

TABLE 5A-7

USE OF CODE CASE N-392-1 (12-11-89) – EVALUATION OF THE DESIGN OF HOLLOW CIRCULAR CROSS-SECTION WELDED ATTACHMENTS ON CLASS 2 AND 3 PIPING

PIPING CLASS	DESCRIPTION	DRAWING NO.	LOCATION (SUPPORT)	METHOD OF ATTACHMENT
GC-8-1006	1-RV-200 Discharge Piping	91-098 Sh 0002	R-9	Welded
GC-8-1003	1-RV-200 Discharge Piping	91-098 Sh 0002	A-1	Welded
GC-8-1006	1-RV-200 Discharge Piping	91-098 Sh 0002	H-2	Welded
GC-8-1006	1-RV-200 Discharge Piping	91-098 Sh 0002	H-1	Welded
GC-8-1013	1-RV-201 Discharge Piping	91-098 Sh 0001	R-11	Welded
GC-8-1013	1-RV-201 Discharge Piping	91-098 Sh 0001	R-15	Welded
GC-8-1013	1-RV-201 Discharge Piping	91-098 Sh 0001	R-16	Welded
GC-8-1013	1-RV-201 Discharge Piping	91-098 Sh 0001	A-2	Welded

NOTE: Per Regulatory Guide 1.84, Revision 31, May 1999, Design and Fabrication Code Case Acceptability, ASME Section III, Division 1, Code Case N-392-1 is acceptable subject to the following conditions in addition to those conditions specified in the Code Case: Applicants should identify in their Safety Analysis Report: (1) the method of lug attachment, (2) the piping system involved, and (3) the location in the system where the Case is to be applied. Accordingly, this table should be updated as necessary to use the Code Case.

TABLE 5A-8

DAMPING VALUES FOR ALL[1] SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS

(Percent of Critical Damping)

STRUCTURE OR COMPONENT	OBE	SSE
Welded steel plate assemblies	1	1
Welded steel framed structures	2	2
Bolted or riveted steel framed structures	2.5	2.5
Reinforced concrete equipment supports	2	3
Reinforced concrete frame and buildings	3	5
Prestressed concrete structures	2	5
Steel piping[2]	0.5	0.5
Soil	2	3
Rocking motion for prestressed concrete structures	5	7
Rocking motion for reinforced concrete structures	5	7

(1) See Table 5-9 for Damping Values of the 1A Diesel Generator Structures, Systems, and Components.

(2) In lieu of these values, ASME Code Case N-411 may be used as described in Section 5A.3.1.6.

TABLE 5A-9

DAMPING VALUES FOR CONTAINMENT SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS (FOR USE WITH 3-D CONTAINMENT MODEL GENERATED TIME HISTORIES AND RESPONSE SPECTRA)

(Percent of Critical Damping)[1]

STRUCTURE OR COMPONENT	OBE[2]	SSE
Equipment and large-diameter piping systems[3][4], pipe diameter > 12”	2	3
Small-diameter piping systems[4], pipe diameter ≤ 12”	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

(1) Damping values higher than the ones delineated above within the Table may be used in a dynamic seismic analysis if documented test data exists to support the use of higher values.

(2) In the dynamic analysis of active components, as defined in Regulatory Guide 1.48, these values should also be used for SSE.

(3) Includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, use values for small-diameter piping.

(4) In lieu of these values, ASME Code Case N-411 may be used as described in Section 5A.3.1.6.

5A.4 LOADINGS COMMON TO ALL STRUCTURES

Ice or Snow Loading - a uniformly distributed live load of 30 psf on all roofs provides for any anticipated snow and/or ice loading.

Temperature - The plant is designed for a temperature range of 20°F to 90°F.

5A.5 FLOODING

Finished grade of the plant site is at Elevation 45', therefore no loads due to floods or inundation are considered.

The saltwater cooling system pump motors, located in the Intake Structure, are protected against the maximum hypothetical hurricane tide and storm surges including wave action. Maximum design wave runup is 28.5' above mean sea level.

The roof and roof hatches of the intake structure are designed for live load (250 psf), dead load (150 psf), tornado uplift (100 psf), Probably Maximum Hurricane waves (250 psf) and seismic load of 10% of the dead load acting downwards.

For all major structures below finish grades, a heavy waterproofing membrane of 40 mils thickness is provided at the exposed face of the exterior walls and below the base slab. Rubber waterstops are also provided at all construction joints up to grade elevation. Subsurface drains are provided to lower the elevation of ground water around the plant. All of these provisions are made to eliminate any possibility of flooding, by ground water infiltration, of equipment located below the elevation of highest flood water level.

5A.6 MASONRY WALL DESIGN

NRC Bulletin 80-11 required licensees to identify plant masonry walls and their intended functions. Licensees were also required to present reevaluation criteria for the masonry walls with analyses to justify those criteria. If modifications were proposed, licensees were to state the methods and schedules for the modifications.

In response to the bulletin, BGE provided the NRC with a description of the status of masonry walls at Calvert Cliffs Nuclear Power Plant. A total of 147 safety-related walls were initially identified. All walls subject to reevaluation were in the Auxiliary Building.

The masonry construction at Calvert Cliffs Units 1 and 2 consists of single- and double-wythe walls of the running bond type whose functions include partition, shielding, blockout, bearing, and filler. Both reinforced and unreinforced walls were built in the plant. Vertical reinforcement was provided by grouting reinforcing bars in vertical cells and "Dur-o-Wall" was used for horizontal reinforcement.

The masonry walls were reevaluated using the following criteria:

a. The design allowables are based on ACI 531-79.

b. The working stress design method and the energy balance technique were used in the analysis. Out of 147 safety-related walls, 22 were qualified by the energy balance technique. Based on a subsequent review, four of the 22 walls were reclassified as non-safety-related walls because failure of these walls would not have any impact on safety-related equipment.

c. Loads and load combinations were consistent with the other parts of this appendix.

d. Critical damping values of 4% for OBE and 7% for SSE were used for vertically reinforced walls which were assumed to crack under seismic conditions. A damping value of 2% was used for walls that were assumed not to crack.

e. The typical analytical procedure is summarized below:

    1. Determine wall boundary conditions

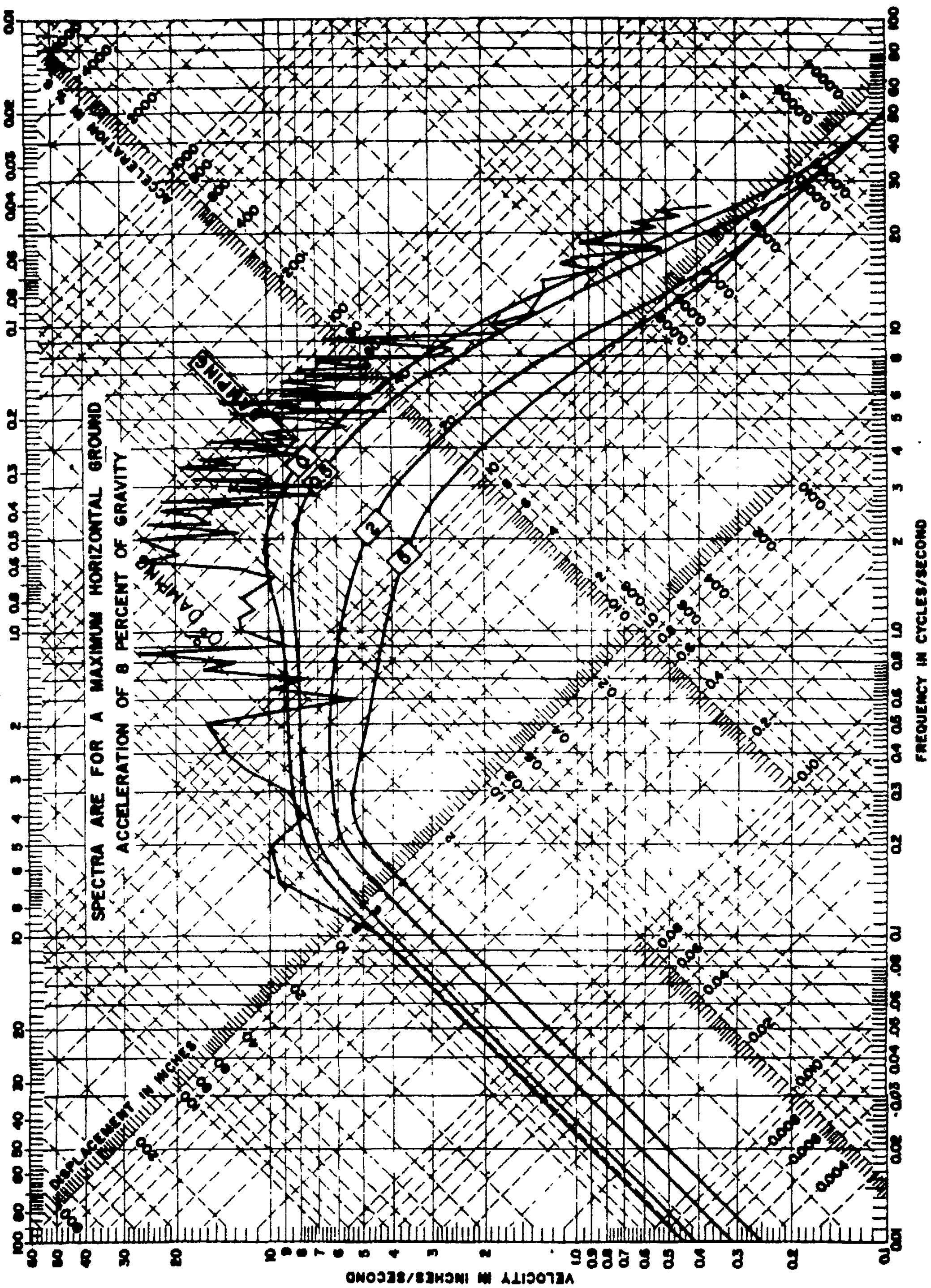
    2. Using a one-way beam model and the floor response spectrum, determine the responses of the first three modes and combine them by the square root of the sum of the squares method.

    3. Compare computed stresses with the allowable values in ACI 531-79.

Because arching action had been used to qualify one of the original 147 walls, it was modified to bring it within elastic requirements. All other walls satisfied the reevaluation criteria and no other modifications were proposed.

All but two of the walls were qualified by the elastic criteria (consistent with NRC acceptance criteria) when the existing conservatism in the masonry wall analysis was accounted for. The remaining two walls were also qualified by the elastic criteria using a "plate" analysis approach rather than "beam" analysis approach.

The NRC concluded there is reasonable assurance that all safety-related masonry walls will withstand the specified design load conditions without impairment of either wall integrity or the performance of the required function.

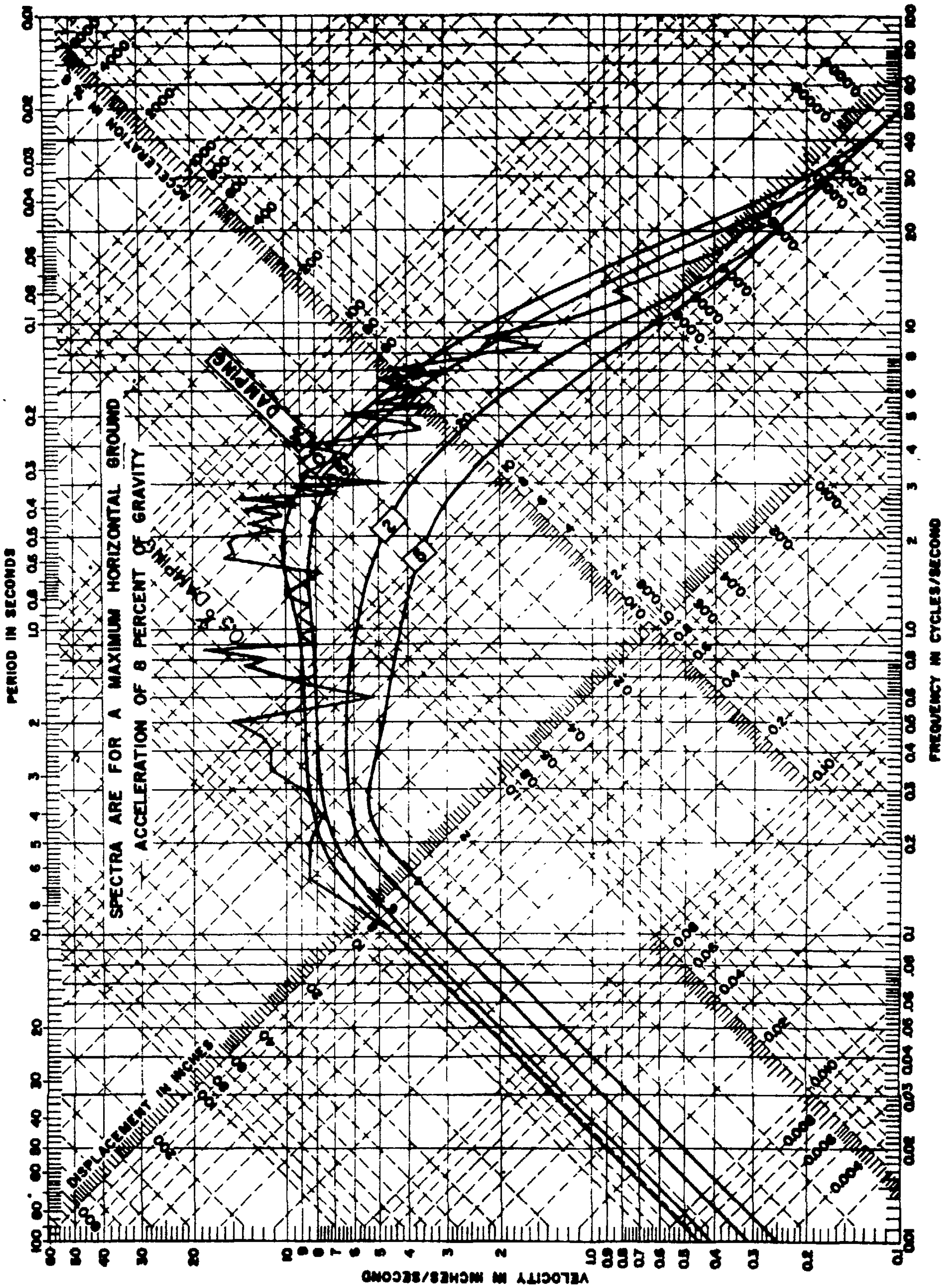


RESPONSE SPECTRA

OPERATING BASIS EARTHQUAKE

FIGURE 5A-1

FIGURE 5A-2 GROUND RESPONSE SPECTRUM OF EL-CENTRO QUAKE, E-W COMPONENT FOR 0.5% DAMPING



RESPONSE SPECTRA

OPERATING BASIS EARTHQUAKE

DAMES & MOORE

FIGURE 5A-2

FIGURE 5A-3 GROUND RESPONSE SPECTRUM OF EL-CENTRO QUAKE, E-W COMPONENT FOR 2.0% DAMPING

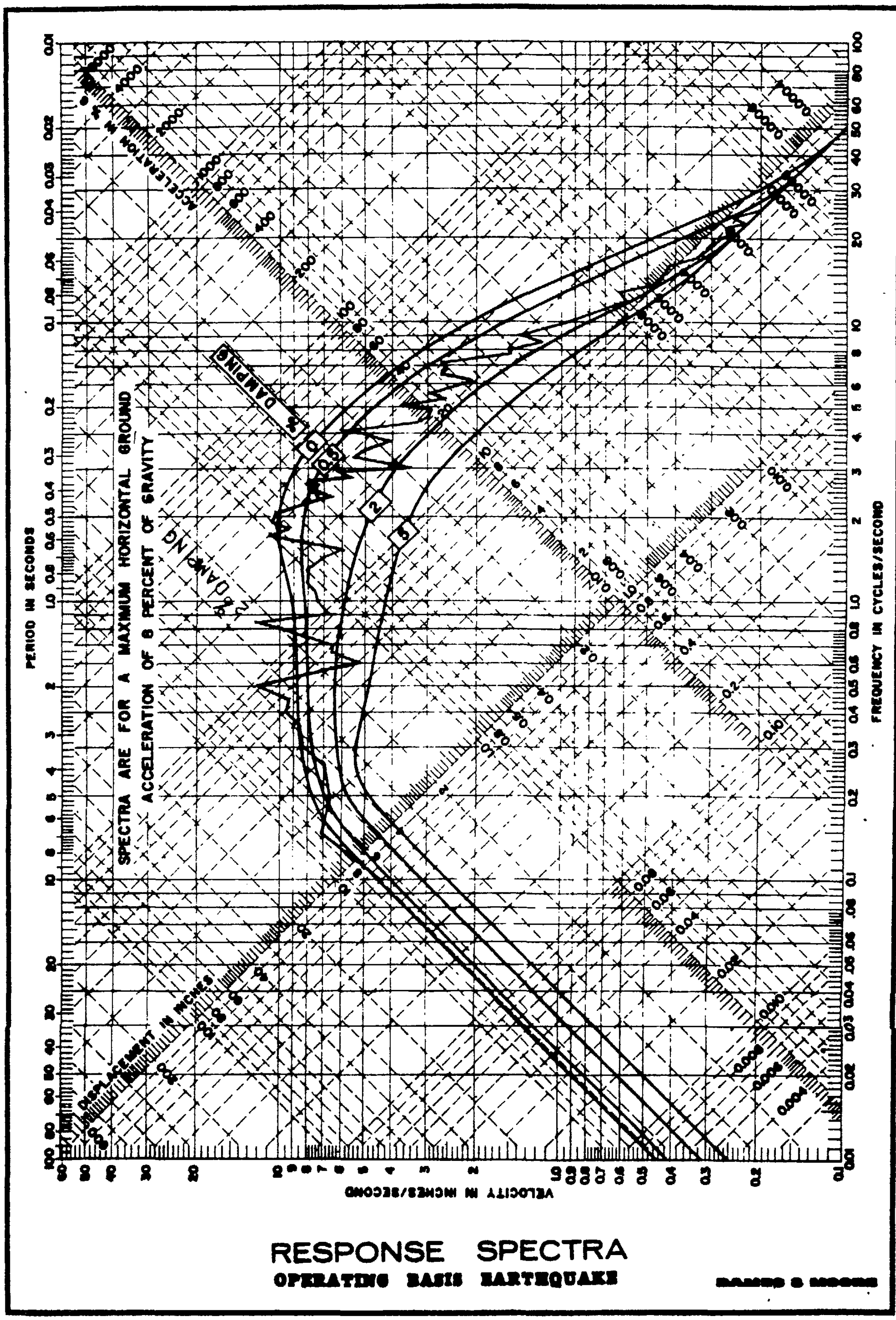


FIGURE 5A-3

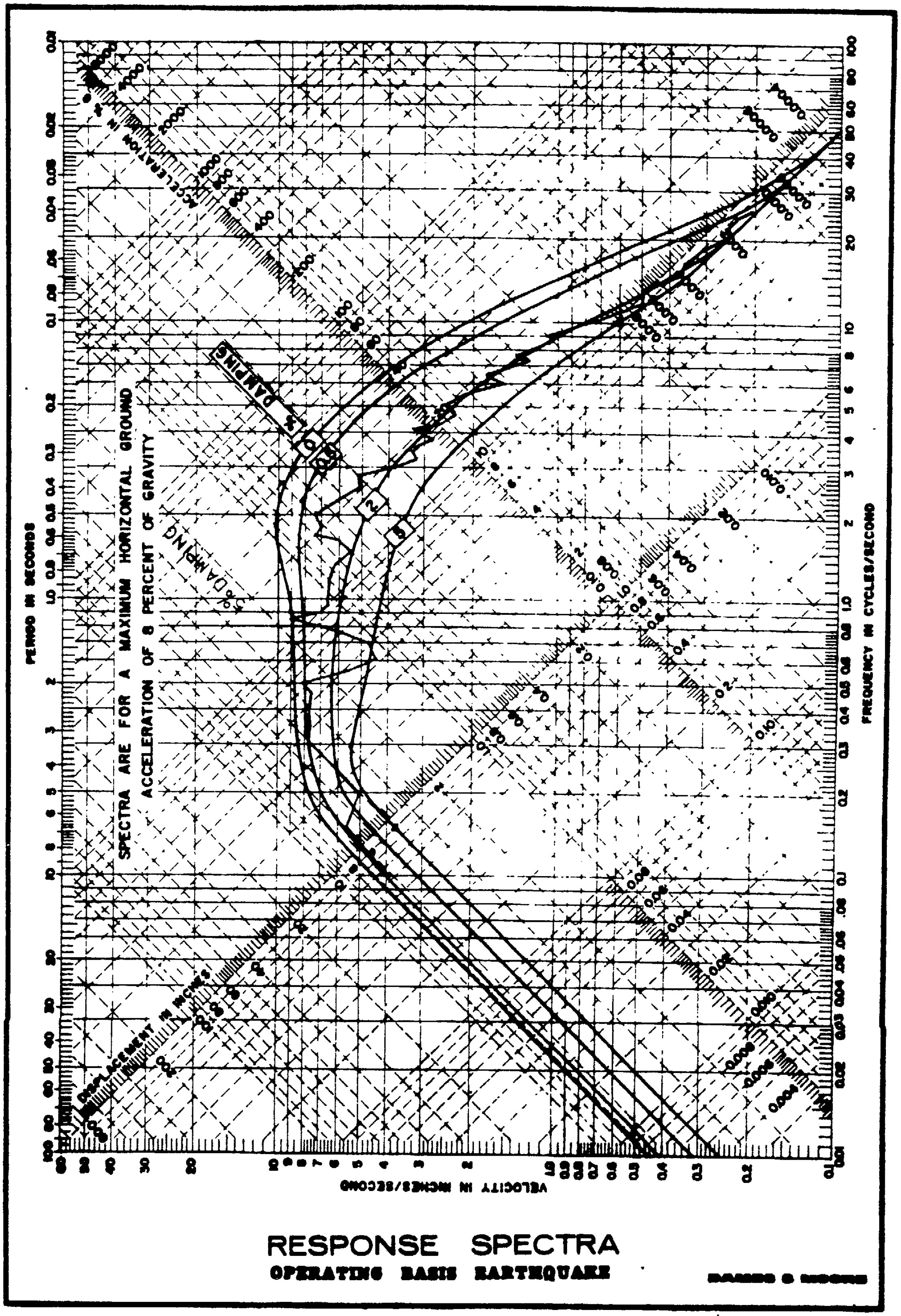


FIGURE 5A-4 GROUND RESPONSE SPECTRUM OF EL-CENTRO QUAKE, E-W COMPONENT FOR 5 0 % DAMPING

FIGURE 5A-4

APPENDIX 5B

5B QUALITY CONTROLS

Appendix 5B, Quality Controls, is historical information about the construction phase of the plant. Appendix 5B has been removed from the Safety Analysis Report and has been sent to Plant History. An image of the contents of the appendix can be accessed in the NORMs Records System under Document ID (DOC ID) “Appendix-5B.”

APPENDIX 5C

5C STUDY OF THE EFFECTS OF MISLOCATED VERTICAL TENDONS

Appendix 5C, Mislocated Tendon Study, is historical information about the construction phase of the plant. Appendix 5C has been removed from the Safety Analysis Report and has been sent to Plant History. An image of the contents of the appendix can be accessed in the NORMs Records System under Document ID (DOC ID) “Appendix-5C.”

APPENDIX 5D

5D STUDY OF UPPER VERTICAL TENDON BEARING PLATES

Appendix 5D, Study of Upper Vertical Tendon Bearing Plates, is historical information about the construction phase of the plant. Appendix 5D has been removed from the Safety Analysis Report and has been sent to Plant History. An image of the contents of the appendix can be accessed in the NORMs Records System under Document ID (DOC ID) "Appendix-5D."

APPENDIX 5E
REDUCTION IN CONTAINMENT PRESTRESS AND LONG-TERM CORRECTIVE ACTIONS FOR VERTICAL TENDON CORROSION

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LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
CCNPP	Calvert Cliffs Nuclear Power Plant
ISI	Inservice Inspection
NRC	Nuclear Regulatory Commission
PSC	Precision Surveillance Corporation

APPENDIX 5E

5E.0 REDUCTION IN CONTAINMENT PRESTRESS AND LONG-TERM CORRECTIVE ACTIONS FOR VERTICAL TENDON CORROSION

5E.1 REDUCTION IN CONTAINMENT PRESTRESS

5E.1.1 DESIGN BASIS

The design basis for the Containment Structure is described in Section 5.1.1.

5E.1.2 DESIGN CRITERIA FOR PRESTRESS

In the concept of a post-tensioned Containment Structure, the internal pressure load is balanced by the application of an opposing external force on the structure. Sufficient post-tensioning was applied to the containment cylinder and dome to more than balance the internal pressure. Therefore, a margin of external pressure exists beyond that required to resist the design basis loss-of-coolant accident pressure.

Nominal, bonded reinforcing steel was also provided to distribute strains due to shrinkage and temperature. Additional bonded reinforcing steel was used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning was not required to exert an external pressure for this portion of the structure. The post-tensioning system is described in Sections 5.1.2.1, 5.1.4.2, and 5.5.1. Design load combinations are provided in Sections 5.1.2.2 and 5A.3.1.8.

5E.1.3 DESIGN REANALYSIS

As a result of some hoop tendon lift-off values being lower than expected during the third-year tendon surveillance, as required per Reference 1 at the time, a reanalysis of the Containment was performed between 1977 and 1979 to reduce the minimum required prestress. The third year lift-off force values for hoop tendons indicated that there was sufficient post-tensioning to meet design requirements, but that the losses appeared to be more accelerated than originally expected. The results of the reanalysis were used to develop minimum tendon force requirements for continual tendon surveillance.

The basic approach in the reanalysis was to take advantage of conservatism existing in the initial prestressing system and the conventional reinforcing with respect to the allowable stresses provided in Table 5-1. The reanalysis was done to more accurately reflect the expected results of future surveillances without reducing the original intended margins of the design. The strict requirements of the reanalysis were to assure that all the original design criteria was maintained.

The basic approach of the reanalysis was to reduce the prestress on all three major tendon groups by a uniform percentage of 9%. The original containment vertical prestress level of 1.32P would be reduced to 1.2P. Specific load cases, as identified in Sections 5.1.2.2 and 5A.3.1.8 were reanalyzed and those components of the Containment Structure potentially affected by the reduced prestress were reevaluated. The approach was satisfactory for both the hoop and dome tendon groups with all the design criteria satisfactorily met or exceeded. However, the reinforcing steel in the shell/base slab interface was slightly overstressed for the structural integrity test (Containment) working stress design load case. To overcome this localized overstress, the vertical tendon group prestressing level was only reduced to 1.29P. Therefore, all the original design criteria, as outlined in Sections 5.1.2 and 5A.3.1, were completely satisfied, assuring sufficient prestress to be available at the end of the nominal 60-year design life. Those components

of the Containment Structure not affected by a reduced prestress level were not reevaluated.

Since the only change made in the 1977-1979 reanalysis was to reduce the prestress level, the reanalysis concentrates on the portion of the analysis/design that was affected by prestress. Otherwise, the original analysis calculations and results remain valid.

The 1977-1979 reanalysis was performed in two parts. One part addresses the base/shell haunch issue and used separate models for the base slab, and the haunch that were analyzed manually using classical plate and shell theory and finite difference methods. Compatibility relationships were used to establish continuity at the slab/shell boundary.

The second part used a finite element analysis (FINEL CE-316) to model the Containment and obtain force and moments as output. Seismic loads were recomputed using a finite element model axisymmetric shells and solids (CE-771), which provided a more exact distribution of seismic loads, compared to those provided in the original seismic analysis. Results of the analysis were post-processed to convert the force and moments on the concrete and reinforcing steel into stresses.

5E.1.3.1 Summary of Calvert Cliffs Containment Reanalysis

The Calvert Cliffs Containment was reanalyzed to check the effect of reduced prestressing forces. The following table illustrates the original design and reanalysis prestressing forces.

	Containment Original Prestress Forces	Containment Reanalysis Prestress Forces	Reduction
HOOP	630 K/Ft	573.16 K/Ft	9%
VERTICAL	300 K/Ft	294.2 K/Ft	2%
DOME	360 K/Ft	327.52 K/Ft	9%

The reanalysis consisted of an elastic finite element analysis of the upper containment shell and dome. The lower part of the shell and base slabs were analyzed using “cracked” section properties in order to incorporate the proper redistribution of stresses. The reanalysis considered applicable dead, thermal, and pressure loadings in addition to the revised prestress forces. The results of the original seismic analysis were used in the reanalysis, as described above. The stresses derived from the reanalysis were checked against the Table 5-1 set of allowables. There is no significant overstressing in either the concrete or the reinforcing steel. The stresses checked include those in the meridional, hoop, and radial directions. The containment stresses from the reanalysis are tabulated in Table 5E-1. The location key, allowable stresses, and general notes are provided in Table 5-1.

5E.1.4 REFERENCES

1. Nuclear Regulatory Commission Regulatory Guide 1.35, Revision 2, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures

TABLE 5E-1

STRESS ANALYSIS RESULTS

CONTAINMENT STRUCTURE – SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES

REINFORCING STEEL

			COMPUTED (psi)		COMPUTED vs. ALLOWABLE	
SECTION		**LOAD CASE**	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
	II	D+F+L+1.15P	C	C	---	---
	III	D+F+L+T_O+E	C	7900	---	0.26
A-B	IV	D+F+L+T_A+P	10200	9900	0.34	0.33
	V	1.05D+F+1.5P+T_A	3300	7000	0.06	0.13
	VI	1.05D+F+1.25P+T_A+1.25E	2900	6900	0.05	0.13
	VII	D+F+P+T_A+E'	2200	7500	0.04	0.14
	II	D+F+L+1.15P	C	C	---	---
	III	D+F+L+T_O+E	8800	15900	0.29	0.53
C-D	IV	D+F+L+T_A+P	13900	20800	0.46	0.69
	V	1.05D+F+1.5P+T_A	C	21500	---	0.40
	VI	1.05D+F+1.25P+T_A+1.25E	8100	21200	0.15	0.39
	VII	D+F+P+T_A+E'	17800	20800	0.33	0.39
	II	D+F+L+1.15P	C	C	---	---
	III	D+F+L+T_O+E	11200	12000	0.37	0.40
E-F	IV	D+F+L+T_A+P	C	5600	---	0.19
	V	1.05D+F+1.5P+T_A	C	2000	---	0.04
	VI	1.05D+F+1.25P+T_A+1.25E	C	3100	---	0.06
	VII	D+F+P+T_A+E'	C	5200	---	0.10
	II	D+F+L+1.15P	C	C	---	---
	III	D+F+L+T_O+E	14900	12100	0.50	0.40
G-H	IV	D+F+L+T_A+P	27800	22400	0.93	0.75
	V	1.05D+F+1.5P+T_A	35900	30100	0.66	0.56
	VI	1.05D+F+1.25P+T_A+1.25E	29200	27800	0.54	0.51
	VII	D+F+P+T_A+E'	24700	24500	0.46	0.45
	II	D+F+L+1.15P	C	C	---	---
	III	D+F+L+T_O+E	8500	10300	0.28	0.34
J-K	IV	D+F+L+T_A+P	24200	26800	0.81	0.89
	V	1.05D+F+1.5P+T_A	36400	55600	0.67	1.03
	VI	1.05D+F+1.25P+T_A+1.25E	31400	36600	0.58	0.68
	VII	D+F+P+T_A+E'	25900	27500	0.48	0.51

TABLE 5E-1

STRESS ANALYSIS RESULTS

CONTAINMENT STRUCTURE – SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES

REINFORCING STEEL

			COMPUTED (psi)		COMPUTED vs. ALLOWABLE	
SECTION		**LOAD CASE**	σ_m	σ_h	$\frac{\sigma_m}{f_s}$	$\frac{\sigma_h}{f_s}$
	II	D+F+L+1.15P	9300	C	0.31	---
	III	D+F+L+T_O+E	15700	27900	0.52	0.93
L-M	IV	D+F+L+T_A+P	C	C	---	---
	V	1.05D+F+1.5P+T_A	16000	C	0.29	---
	VI	1.05D+F+1.25P+T_A+1.25E	400	C	0.01	---
	VII	D+F+P+T_A+E'	C	C	---	---
	II	D+F+L+1.15P	20800	25500	0.69	0.85
	III	D+F+L+T_O+E	9500	C	0.32	---
N-O	IV	D+F+L+T_A+P	18400	24200	0.61	0.81
	V	1.05D+F+1.5P+T_A	39300	44300	0.73	0.82
	VI	1.05D+F+1.25P+T_A+1.25E	25900	40500	0.48	0.75
	VII	D+F+P+T_A+E'	20800	32500	0.39	0.60
	II	D+F+L+1.15P	19800	25600	0.66	0.85
	III	D+F+L+T_O+E	8100	17200	0.27	0.57
P-Q	IV	D+F+L+T_A+P	24400	24300	0.81	0.81
	V	1.05D+F+1.5P+T_A	37600	41600	0.70	0.77
	VI	1.05D+F+1.25P+T_A+1.25E	39900	46300	0.74	0.86
	VII	D+F+P+T_A+E'	32900	39200	0.61	0.73

TABLE 5E-1
STRESS ANALYSIS RESULTS
CONTAINMENT STRUCTURE – SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES
CONCRETE

			COMPUTED (psi)					COMPUTED vs. ALLOWABLE		
SECTION		**LOAD CASE**	σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_s}$	$\frac{\tau}{v}$
	II	D+F+L+1.15P	-890	-790	-840	-790	12	0.30	0.56	0.03
	III	D+F+L+T_O+E	-2630	-2810	-1520	-1460	24	0.94	1.01	0.07
A-B	IV	D+F+L+T_A+P	-2770	-2670	-930	-880	27	0.92	0.62	0.07
	V	1.05D+F+1.5P+T_A	-2830	-2590	-640	-540	24	0.63	0.15	0.06
	VI	1.05D+F+1.25P+T_A+1.25E	-3140	-3140	-780	-730	21	0.70	0.18	0.06
	VII	D+F+P+T_A+E'	-3360	-3470	-930	-880	18	0.77	0.22	0.05
	II	D+F+L+1.15P	-480	-300	-430	-240	167	0.16	0.29	0.39
	III	D+F+L+T_O+E	-2660	-1480	-840	-310	131	0.89	0.56	0.31
C-D	IV	D+F+L+T_A+P	-2190	-2410	-480	-250	141	0.80	0.32	0.33
	V	1.05D+F+1.5P+T_A	-2010	-2040	-310	-210	99	0.45	0.07	0.23
	VI	1.05D+F+1.25P+T_A+1.25E	-2670	-2200	-400	-230	94	0.59	0.09	0.22
	VII	D+F+P+T_A+E'	-3270	-2410	-480	-250	94	0.73	0.11	0.22
	II	D+F+L+1.15P	-590	-290	-310	-270	116	0.20	0.21	0.39
	III	D+F+L+T_O+E	-1130	-1270	-480	-310	134	0.42	0.32	0.44
E-F	IV	D+F+L+T_A+P	-2110	-2890	-330	-270	128	0.96	0.22	0.42
	V	1.05D+F+1.5P+T_A	-1870	-2790	-260	-260	88	0.62	0.06	0.30
	VI	1.05D+F+1.25P+T_A+1.25E	-1810	-2820	-300	-260	97	0.63	0.07	0.32
	VII	D+F+P+T_A+E'	-2110	-2870	-340	-270	98	0.64	0.08	0.32
	II	D+F+L+1.15P	-160	-260	-130	-240	30	0.09	0.16	0.18
	III	D+F+L+T_O+E	-2170	-1870	-640	-610	123	0.72	0.42	0.74
G-H	IV	D+F+L+T_A+P	-1570	-1940	-240	-300	40	0.65	0.20	0.24
	V	1.05D+F+1.5P+T_A	T	-530	-40	-120	13	0.12	0.03	0.08
	VI	1.05D+F+1.25P+T_A+1.25E	-630	-930	-140	-170	19	0.21	0.04	0.11
	VII	D+F+P+T_A+E'	-1430	-1540	-240	-250	35	0.34	0.06	0.21
	II	D+F+L+1.15P	-320	-110	-220	-90	21	0.11	0.15	0.15
	III	D+F+L+T_O+E	-1860	-2390	-710	-1060	40	0.80	0.71	0.29
J-K	IV	D+F+L+T_A+P	-2240	-1140	-330	-210	97	0.75	0.22	0.70
	V	1.05D+F+1.5P+T_A	-740	T	-140	T	102	0.16	0.03	0.73
	VI	1.05D+F+1.25P+T_A+1.25E	-1300	T	-210	T	108	0.29	0.05	0.78
	VII	D+F+P+T_A+E'	-1990	-1040	-300	-200	97	0.44	0.07	0.70

TABLE 5E-1

STRESS ANALYSIS RESULTS

CONTAINMENT STRUCTURE – SUMMARY OF CONCRETE AND REINFORCING STEEL STRESSES

CONCRETE

			COMPUTED (psi)					COMPUTED vs. ALLOWABLE		
SECTION		**LOAD CASE**	σ_{em}	σ_{eh}	σ_{am}	σ_{ah}	τ	$\frac{\sigma_e}{f_{ce}}$	$\frac{\sigma_a}{f_s}$	$\frac{\tau}{v}$
	II	D+F+L+1.15P	-1100	-900	-250	-730	129	0.37	0.49	0.47
	III	D+F+L+T_O+E	-2790	-260	-680	T	80	0.93	0.45	0.44
L-M	IV	D+F+L+T_A+P	-800	-2410	-310	-650	144	0.80	0.43	0.45
	V	1.05D+F+1.5P+T_A	-1000	-3350	-130	-1700	161	0.74	0.40	0.34
	VI	1.05D+F+1.25P+T_A+1.25E	-20	-3050	-220	-960	258	0.68	0.23	0.54
	VII	D+F+P+T_A+E'	-800	-2600	-310	-810	212	0.58	0.19	0.44
	II	D+F+L+1.15P	T	-540	T	T	72	0.30	(a)	0.46
	III	D+F+L+T_O+E	-480	-20	-180	-150	27	0.27	(a)	0.17
N-O	IV	D+F+L+T_A+P	T	-640	-50	-30	113	0.36	(a)	0.72
	V	1.05D+F+1.5P+T_A	-180	-1440	-50	-20	167	0.40	(a)	0.70
	VI	1.05D+F+1.25P+T_A+1.25E	T	-430	-10	T	170	0.12	(a)	0.72
	VII	D+F+P+T_A+E'	T	-780	-40	-20	150	0.22	(a)	0.63
	II	D+F+L+1.15P	-540	-660	T	T	4	0.37	(a)	0.07
	III	D+F+L+T_O+E	-590	-990	-270	-100	47	0.55	(a)	0.83
P-Q	IV	D+F+L+T_A+P	-950	-980	-170	T	17	0.54	(a)	0.30
	V	1.05D+F+1.5P+T_A	-1850	-1810	-170	T	11	0.51	(a)	0.13
	VI	1.05D+F+1.25P+T_A+1.25E	-850	-1540	-110	T	48	0.43	(a)	0.56
	VII	D+F+P+T_A+E'	-980	-1840	-130	T	58	0.51	(a)	0.68

(a) $f_a = 0.3\ f_c'$

5E.2 LONG-TERM CORRECTIVE ACTIONS FOR VERTICAL TENDON CORROSION

5E.2.1 DISCOVERY OF CORROSION

During performance of the 20-year (1997) Technical Specification and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference 1) tendon surveillance on Unit 1, conditions that did not meet the acceptance standards were found on the Containment Structure. Conditions that did not meet the acceptance standards were found in all three containment tendon populations, i.e., hoop, dome, and vertical tendons. The abnormal conditions found on the hoop and dome tendons were considered minor enough that the acceptability of the concrete containment was not affected. The unacceptable conditions were found in the vertical tendon population. Several of the vertical tendons selected for surveillance were found to contain broken and corroded wires at their top ends, below the stressing washer. The discovery of broken wires in these tendons initiated an expansion of the Unit 1 vertical tendon inspection scope to perform visual inspections and lift-off testing on all Unit 1 vertical tendons. Subsequently, broken and corroded wires were found throughout the Unit 1 vertical tendon population at the top ends of the tendons. Following completion of the Unit 1 surveillance, the 20-year surveillance of the Unit 2 tendons was conducted. Although Unit 2 was only required to perform visual inspections, it was decided to also perform lift-off testing of all the vertical tendons in order to facilitate inspection of the tendon wires in the region of concern below the upper (top) stressing washer. Abnormal conditions very similar to Unit 1 were found on the Unit 2 vertical tendons. Precision Surveillance Corporation (PSC), performed all the inspection work. Precision Surveillance Corporation wrote a non-conformance report for every abnormally degraded condition that did not meet the acceptance standards of IWL-3220 and Reference 2. The original tendon post-tensioning system is described in Sections 5.1.2 and 5.1.4.

5E.2.2 CAUSE OF VERTICAL TENDON CORROSION

During the 20-year tendon surveillance on the Unit 1 and Unit 2 Containments, corrosion and broken wires were discovered on some vertical tendons. Reference 3 and Reference 4 evaluated the findings and determined the cause of the corrosion and wire breaks. The evaluation concluded that the tendon wire failures and corrosion problems resulted from a combination of water and moist air intrusion into the vertical tendon end caps (grease cans), and inadequate initial grease coverage of wires in the area just under the top stressing washer. To address issues identified in the evaluation, short-term corrective actions were taken. These actions included spraying hot grease under the top stressing washer, reorienting the stressing shims to leave a gap between the shims to allow a vent path to help eliminate voids, re-greasing non-corroded vertical tendons, and resealing around the original tendon can all-thread penetrations with caulking. Additional inspections were performed in 1999 and 2000 to verify the assumptions that were considered in evaluation and to provide additional data to help develop a long-term corrective action plan.

5E.2.3 LONG-TERM CORRECTIVE ACTIONS

The goal of the long-term corrective action plan is to ensure that the Containments meet their design basis requirements until plant end-of-life. As one part of the long-term corrective action plan, all the vertical tendons have been re-greased with new corrosion inhibiting grease (Visconorust 2090-P4). The non-corroded vertical tendons were re-greased in 2000, and the tendons with less severe corrosion were re-greased during 2001. The remaining vertical tendon population was replaced in 2001 and 2002, and had new grease put in place at that time. In addition, all of the vertical tendons had a redesigned pressure-tight, grease-filled cap installed at the upper-bearing plate to prevent water intrusion. The bottom grease cap for every vertical tendon was also replaced with a new redesigned pressure-tight grease cap. The redesigned grease cap has a flange that

is attached by studs and nuts to the tendon bearing plates utilizing existing taps in the plates.

As mentioned in the above paragraph, another part of the long-term corrective action plan involved the replacement of a portion of the corroded vertical tendon population. Preliminary evaluations using a wire breakage predicting model had shown that without vertical tendon replacement, neither Containment would meet its design basis at plant end-of-life due to prestress loss from predicted future wire breakage. To determine which vertical tendons were to be replaced, a final vertical tendon future wire breakage prediction model and selection criteria were developed. The wire breakage prediction model is described below first, followed by the vertical tendon replacement selection criteria.

5E.2.3.1 Future Wire Breakage Prediction Model (a.k.a. Weibull Model)

As discussed previously, corrosion (abnormal degradation) at some of the top ends of the Units 1 and 2 containment vertical tendons was found during the 20-year 1997 tendon surveillance. To determine the acceptability of the Containments without repairing the vertical tendon wires, a model was developed to predict how the degradation of the tendon wires would affect the wires over the plant life. Wire degradation is predicted to lead to additional future wire breakage, as was found during the 1997 inspections.

Reference 5 was used to determine the acceptability of the Containments without repairing or replacing the degraded vertical tendons. The objective of the report was to develop models for the future failures of tendon wires. The models could then be used to assess how long the vertical tendons would continue to meet structural integrity requirements, with the assumption that the short-term corrective actions and long-term corrective actions were not fully effective in stopping further corrosion degradation. The models were developed using a Unit 2 timeline. The Unit 1 tendons have been tensioned longer than those of Unit 2. However, the inspections of 1997, 1999, and 2000 indicated that the conditions of the tendons for the two units were comparable at the time. Therefore, the models are applicable and conservative for either unit on a calendar time basis, without adjustment for the longer service time under tension of the Unit 1 tendons. The models described in the report use the additional data derived from the 1999 and 2000 inspections to extend the postulated period of validity to the plant end-of-life. The model for ductile/general corrosion wire failures also uses the observed wire failures during lift-off tests to develop an additional data point for failures from that degradation mode. The report presents separate models for the two mechanisms of degradation: (1) hydrogen-induced cracking; and (2) ductile/general corrosion failures. The predictions for the two degradation mechanisms are then combined to obtain the total predicted wire failures.

Because of the conservative assumptions used to develop the combined model, it is anticipated that the model predictions will be bounding for observed behavior for the remaining extended operating license of the Units. Although the model developed in the report is expected to be a conservative upper-bound estimate of what will actually occur, it is based on minimal data and plausible assumptions. The conservatism of the model will be validated by future enhanced tendon inspections. The report shows that, conservatively, 2,714 wires could break on each Unit due to abnormal wire degradation by the end of the plant operating licenses. The report shows the spread of the number of future wire breaks in individual tendons throughout the tendon population. This range is from a predicted maximum of 86 wire breaks in one tendon, to a minimum of 1 wire break in 48 different tendons.

In order to apply the statistical model predictions to all of the original vertical tendons for each Unit, all of the vertical tendons were first grouped by as-found corrosion level in 1997. The various corrosion levels were determined during the visual examination performed on the tendon wire surfaces behind the shim stacks at the top of each vertical tendon in 1997 and 1998. This is the area defined by References 3 and 4 as the area susceptible to wire breakage. Once the individual vertical tendons were ranked by corrosion level, a list of tendons by corrosion level group was generated ranging from "extreme corrosion" to "no corrosion." An average predicted number of wire failures was then calculated for each corrosion level group. This average number was then assigned to each tendon in a particular corrosion level group to represent future wire breaks. This approach of assigning an average number of future wire breaks to a corrosion level group was conservative in that, as vertical tendons were selected for replacement, less future wire breaks would be removed from the predicted total. By taking this approach, more corroded tendons would be required to be replaced to achieve acceptability of the Containments.

5E.2.3.2 Selection Criteria for Corroded Tendons to be Replaced

To determine which vertical tendons would be replaced, a selection criterion was developed, as described below:

1. Replace all tendons that had two or more broken wires. Most of the additional broken wires discovered in 1999 were in tendons with two or more previous broken wires from 1997. Therefore, these tendons appeared to be the most likely to have future broken wires. Note: There was one exception to this criterion. Four buttonheads were found missing on the bottom of Unit 2 tendon 61V27 in 2001, and were not in the scope of Reference 4. Therefore, this tendon was not replaced.
2. Replace corroded tendons demonstrating lower lift-off forces. This applies to all tendons that were classified as having extreme or heavy corrosion and had a lift-off force of less than 649 kips in 1997. The small additional strain imparted by lift-off testing has the potential to cause additional wire breaks, as occurred in 1997. Corroded tendons with lower than predicted lift-off forces will be replaced to eliminate the possibility of premature wire breakage during future lift-off testing. Furthermore, replacing severely corroded tendons with low lift-off forces would prevent potential prestress losses associated with wire breakage from the restressing of these tendons. Restressing tendons increases the strain more than lift-off testing, and could potentially result in an even greater number of wire breaks.
3. Replace corroded tendons to ensure uniform distribution of prestress. The third criteria was specific to Unit 1 since it has two tendons that were not originally installed and, therefore, has two areas with low prestress force distribution. Calvert Cliffs replaced all the tendons that had extreme or heavy corrosion near the two empty tendon sheaths.
4. Replace corroded tendons to ensure uniform prestress force distribution after accounting for prestress losses from statistical Weibull model. This criteria ensures the loss of prestress that would result from the conservative prediction of wire breakage, would not violate design criteria described in Sections 5E.1.2 and 5E.1.3 at plant end-of-life. Calvert Cliffs applied the statistical model wire breaking predictions to all of the remaining original tendons that were not replaced under the first three criteria. This last criteria identified the areas around the Containments that, if the predicted wire breaks occurred, had the potential of driving the

distribution of vertical prestress force below the minimum design requirements. Once those areas were identified, appropriate corroded tendons were selected for replacement until the distribution of vertical prestress force exceeded the minimum design requirements at plant end-of-life.

Once a vertical tendon was replaced, the future number of broken wires in the new tendon is assumed to be zero. The number of future wire breaks in non-replaced tendons is the average wire breaks for that corrosion level group. After tendon replacement on each Unit, the conservative predicted number of wire breaks at plant end-of-life drops to 1,195 for Unit 1 and 1,228 for Unit 2.

It was determined that 47 tendons on Unit 1 and 46 tendons on Unit 2 were the most cost-effective number of tendons to replace on each Unit that would provide the most uniform circumferential vertical prestress at plant end-of-life. It was also determined to restress 20 original vertical tendons on Unit 1, and 30 original vertical tendons on Unit 2, that had exhibited low lift-offs in 1997.

5E.2.3.3 Stressing Sequence of Tendons Replaced and Restressed

The process of replacing and restressing vertical tendons on each Containment was done at full power. Therefore, to avoid operability issues during the work process, the number of tendons destressed at any one time, and the sequence in which tendons were destressed for removal, was critical to keeping the Containments within their design basis. Figures 5E-1 and 5E-2 show the final stressing sequence and individual vertical tendons replaced or restressed in 2001 and 2002 for Unit 1 and 2, respectively.

5E.2.3.4 Tendon Bearing Plate Concrete Void Repairs

While performing lift-off testing on all the vertical tendons, two Unit 1 bearing plates depressed during the testing. The concrete under these bearing plates had been previously repaired as part of the tendon bearing plate study discussed in Appendix 5D. However, the repairs made to these bearing plates were not adequate to prevent bearing plate flexure. It was decided to remove these bearing plates and perform additional concrete void repairs with grout. The vertical bearing plates that received grout repairs in 2001, type of grout used, and repair method are shown on Figure 5E-1 for Unit 1.

5E.2.4 ACCEPTABILITY OF CONCRETE CONTAINMENTS

Table 5E-2 provides a summary of vertical prestress conditions for both Units in 2002 for the original design and with corrective actions. Table 5E-2 also provides the predicted vertical prestress conditions in 2034 for Unit 1, and 2036 for Unit 2. The table is intended to provide a comparison of required and predicted vertical gross prestress, and a comparison of required and predicted mean average force per tendon sheath distribution.

Since not all the vertical tendons on each Unit exhibiting corrosion have been replaced, it should be noted that any corrosion on the original .25-inch diameter tendon wires could potentially reduce the effective cross-sectional area of the wire. A reduced effective cross-sectional area at the point of corrosion will cause the unit stress in the wire to increase. During initial stressing of the original tendons, the wires were left at a seating stress between 0.7 $f_s{}'$ (168 ksi) and 0.73 $f_s{}'$ (175.2 ksi) (Section 5.1.4.2). Over time, the original tendon wires have relaxed, reducing the stress in the wires as a percentage of $f_s{}'$. Therefore, the wire stress in the corroded areas should still be below the wire material minimum yield point of

0.8 f_s' (192 ksi). For the wires with severe corrosion that do become stressed beyond the wire material yield point and ultimately break, the total number has already been enveloped in the Containment acceptability evaluation.

5E.2.5 ENHANCED VERTICAL TENDON INSPECTIONS

The future inspection of the vertical tendons is a two-tiered approach. First, ASME Section XI code inspections will be performed as required by the NRC-mandated ASME Boiler and Pressure Vessel Code (Reference 1). Lift-off testing will be conducted on the replacement tendons as required by the ASME Code. Second, enhanced inspections will be performed to examine the tendons for potential wire breaks. To monitor future changes in the conditions of all tendons, a database has been created to catalog the complete scope of all tendon inspection and repair activities.

The goal of enhanced inspections is to ensure that the Weibull Model bounds existing field conditions. To accomplish the enhanced inspections, the anchorhead/buttonhead region is required to be examined to determine if any wire breaks have occurred in the area under the vertical tendon top-stressing washers.

By the end of 2005 and 2007, Calvert Cliffs Nuclear Power Plant (CCNPP) will perform an inspection for wire breakage on 100% of the original vertical tendons. Unit 1 has 155 remaining original vertical tendons. Unit 2 has 158 remaining original vertical tendons. The purpose of these inspections will be to determine the number of failed (i.e., protruding) buttonheads at the top end of the vertical tendons. The resulting total number of protruding buttonheads will be compared to the number of predicted wire breaks for future specific years. If the total number of actual wire breaks is less than the number predicted for that unit in that year, and no tendon has more than the average number of failed wires in their corrosion group, then the actual condition of the containment vertical tendons are within the bounding conditions predicted by the statistical Weibull Model (Reference 5). If the number of failed wires in a tendon exceeds the average for that corrosion level group, the number will be compared to the average for that group at plant end-of-life. If a tendon's actual number of wire breaks exceeds the plant end-of-life predicted failure numbers, then an engineering evaluation will be performed.

In 2007, following the results of the ASME Code and enhanced inspections, CCNPP will assess the need to continue with enhanced inspections. This assessment will determine if the ASME Code inspections alone would provide adequate information to validate the statistical Weibull Model. If the model continues to bound field conditions, but more of a sample is required than that provided by the ASME Code surveillance population, the enhanced inspection frequency will be changed to a five-year span and then completed concurrently with the ASME Code inspections.

5E.2.6 CONTAINMENT PRESSURE TEST

Reference 1, Article IWL-5000, provides requirements for pressure-testing concrete containments following repair or replacement activities. The concrete repairs to the Unit 1 Containment associated with the discovery of corrosion on the containment vertical tendons in 1997 only involved the removal of vertical tendon top bearing plates and the filling of voids with grout. These repairs were outside the outermost layer of structural reinforcing steel in the ring girder. The repairs and replacements to the containment vertical tendon system involved the exchange of post-tensioning tendons, tendon anchorage hardware, shims, and corrosion protection medium for both Containments. In accordance with the ASME Code, by performing these types of repairs and replacements only, no additional containment pressure tests were required to demonstrate containment structural integrity upon completion. The conclusions of the containment structural tests

to 115% of design pressure following original Units 1 and 2 construction remain valid as discussed in Section 5.5.1.2.

5E.2.7 REFERENCES

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, 1992 Edition through the 1992 Addenda, Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants"

2. NRC Regulatory Guide 1.35, Revision 2, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures"

3. Calvert Cliffs Unit 1 20-Year Containment Tendon Surveillance Engineering Evaluation, October 28, 1997

4. CCNPP Root Cause Analysis Report, RCAR-9808, Root Cause Investigation of Containment Tendon Wire Corrosion and Failure, March 26, 1998

5. Dominion Engineering, Inc. Report, R-3648-00-01, Updated Model for Containment Structure Vertical Tendon Degradation for Calvert Cliffs 1 and 2, December 14, 2000

TABLE 5E-2

PRESTRESS COMPARISON IN VERTICAL TENDON SYSTEM

(Tendon relaxation losses for new tendons installed in 2001 and 2002 based on actual wire test Reports)

(Values in kips)

		Unit 1(1)				**Unit 2**			
		Gross Prestress	Mean Average Force per Tendon Sheath	Gross Margin	Mean Average Force per Tendon Sheath Margin	Gross Prestress	Mean Average Force per Tendon Sheath	Gross Margin	Mean Average Force per Tendon Sheath Margin
1.	Design Basis	123,620	606	N/A	N/A	123,620	606	N/A	N/A
2.	Predicted Prestress in 2002. No replacements, no restressing, no wire breaks. Original tendon design.(2)	130,694	641	7,074	35	133,416	654	9,796	48
3.	Predicted Prestress after 2002 Corrective Actions. Includes replacements, includes restressing, no wire breaks.(3)	143,880	705	20,260	99	143,373	702	19,753	96
4.	Predicted Prestress at 2034 and 2036. No replacements, no restressing, no wire breaks. Original tendon design.(4)	129,280	634	5,660	28	132,192	648	8,572	42
5.	Predicted Prestress at 2034 and 2036. Includes replacements, includes restressing, includes wire breaks.(5)	132,334	648	8,714	42	131,353	643	7,733	37

TABLE 5E-2

PRESTRESS COMPARISON IN VERTICAL TENDON SYSTEM

Note: All values in Table 5E-2 have been conservatively rounded down except the line 2 Unit 1 Mean Average Force per Tendon Sheath value, which has been rounded up.

(1) Unit 1 has only 202 vertical tendons, but both Units have 204 sheath locations.

(2) The respective data considers the prestress losses associated with concrete creep/shrinkage and wire relaxation only, and does not consider potential wire breaks.

(3) Comparing Lines 2 and 3 show the predicted margin after the planned corrective actions.

(4) These values can be considered the “original design” margins expected at the end of the operating licenses in 2034 and 2036, if tendon corrosion and wire breakage had never occurred.

(5) Denotes the predicted prestress values following the 2001 and 2002 corrective actions of restressing and replacing tendons, while accounting for prestress losses associated with concrete creep/shrinkage and wire relaxation and a conservative amount of predicted wire breaks.

5E.3 EVALUATION OF PLANT OPERATING LICENSE EXTENSION ON TENDON SYSTEM

5E.3.1 TENDON POPULATIONS

Following original construction of the Units 1 and 2 Containments, there were only three distinct tendon populations: vertical, hoop, and dome. Inservice inspection (ISI) surveillance criteria for tendons during the early years of plant operation were based on Reference 1. Since the two Containments were identical in design, Reference 1 only required that lift-off testing surveillance be performed on one Unit for each tendon population. Tendon prestress losses in each population were generically calculated based on a nominal plant operating life of 40-years. In 2000, Calvert Cliffs received an extension of 20-years to its original operating license. With license extension, the prestress losses in each tendon population would require extrapolation from a 40-year nominal period to a 60-year nominal period. However, with the introduction of different NRC-mandated ISI criteria in 1996, and the discovery of corrosion in the vertical tendon population in 1997, simple extrapolation to 60 years of the existing prestress losses was not possible. Prestress losses for each tendon population on each Containment would have to be developed.

Part of the long-term corrective action plan was the replacement and restressing of vertical tendons on both Containments. The replacement vertical tendons have a different relaxation percentage than the original tendons, and a different effective prestress loss time period. The restressed original vertical tendons also have different relaxation characteristics from the original vertical tendons that were not restressed, and a different effective prestress loss time period.

These changes resulted in tendon prestress losses being developed for three vertical tendon sub-populations, the horizontal tendon population, and the dome tendon population for each Containment unit. Tables 5E-3 and 5E-4 show the tendon populations for each Unit and their effective life.

5E.3.2 ORIGINAL TENDON PRESTRESS LOSSES

The Unit 1 original vertical tendons, hoop tendons, and dome tendons already had 40-year average prestress losses calculated, which had been converted into average tendon force versus time curves to be used in evaluating ISI surveillance lift-off force data. For these three original tendon population curves, the expected tendon force curves only were required to be extended to cover the new nominal 60-year plant end-of-life time period due to license extension.

To develop prestress losses for the Unit 2 original vertical, hoop, and dome tendons, data from the original Unit 2 prestressing report was used. Since the prestress losses and conversion into a tendon force versus time curve represents the average tendon in a population, the average seating stress for the original three populations was taken from the original stressing report. The expected prestress losses at the end of 40 years were then determined based on the original loss values for elastic shortening of concrete, creep and shrinkage of concrete, and relaxation of tendon wire steel as provided in Section 5.1.4.2. Once a loss equation was developed to fit the 40-year values, the equation was used to extend the expected tendon force curves to cover the tendon initial stressing to plant end-of-life time period (60 nominal years).

Special consideration is required for the original Units 1 and 2 tendons that were restressed. These tendons were initially stressed in the early 1970s, held under sustained tension, and then retensioned in 2002. As described in Reference 2, tendons that are restressed will reinitiate relaxation losses as if being stressed for the first time, although not to the same extent. Reference 2 documents an acceptable restress factor of 0.65. To

apply the factor of 0.65 to restressed tendons, the tendon wire relaxation for a given time period is determined for virgin, unstressed wire, and then multiplied by the 0.65 factor.

The average date for restressing the Unit 1 vertical tendons was August 2002. The average date for initial stressing of the Unit 1 vertical tendons was March 1973. Therefore, the 2002 restress corresponds to year 29.5 of the tendon life. The end-of-plant operating life corresponds to year 61.5 of the tendon life. The new tendon force versus time curves based on prestress losses for restressed tendons have an initial stressing date of August 2002. The amount of concrete creep/shrinkage losses that are expected to occur between years 29.5 and 61.5 of the original tendon life were able to be determined since the creep/shrinkage loss is a function of time from the point that the Containment was originally stressed. No new concrete elastic shortening losses were considered. For tendons that are restressed or newly installed, there will not be an appreciable concrete elastic loss since only a few tendons were permitted to be detensioned at any given time. Since the vast majority of tendons remained in tension at all times, the concrete Containments remained under compressive stress and retained the associated elastic deformations from the original stressing in the early 1970s. Equations were developed to determine the amount of average prestress losses due to containment concrete creep and shrinkage between years 29.5 and 61.5, and the amount of prestress losses due to wire relaxation for 32 years after applying a 65% factor. Once the 32-year prestress losses were determined, another equation was fitted to the beginning average restressing force and final average calculated tendon force at the end-of-life. From this equation, tendon force versus time curves were developed to evaluate future tendon surveillance lift-off force test results.

The same approach was used for the Unit 2 restressed vertical tendons in developing final force versus time curves. The only differences were the average tendon remaining life and the average initial restressing force used in the approach.

5E.3.3 NEW TENDON PRESTRESS LOSSES

A total of 47 (Unit 1) and 46 (Unit 2) new vertical tendons were installed during 2001 and 2002. Each tendon consisted of 90-1/4" diameter wires with buttonheaded anchorages, just like the original tendon design. The only significant fabrication change for all the new tendons was that the upper stressing washers all utilized the Prescon 93 hole design. The three empty holes were to provide a vent path during the tendon greasing process to prevent formation of a grease void below the stressing washer. The new tendon wire is per American Society for Testing and Materials A421 Type BA. The relaxation of the prestressing wire steel was specified to be less than 4% for a design life of 40 years. Three 1000-hour relaxation tests were performed on sample wires. Test data was plotted and, using regression analysis, a best fit line was plotted. Each line was projected out to 40 years with the expected relaxation losses at 40 years ranging from 1.5% to 2.2%. An average relaxation value of 2.0% at 40 years was used in calculating the average tendon prestress loss and developing the tendon force versus time curves for the new tendons.

The new tendons will have a total life of 32 years. The tendon force versus time curves based on prestress losses for new tendons have an initial stressing date of August 2002. The amount of concrete creep/shrinkage losses that are expected to occur between years 29.5 and 61.5 of the original tendon life were determined since the creep/shrinkage loss is a function of time from the point that the Containment was originally stressed. Steel relaxation, however, is initiated at the time of stressing of the new tendons. No new concrete elastic shortening losses were considered. For tendons that are newly installed, there will not be an appreciable concrete elastic loss since only a few tendons were permitted to be detensioned at any given time. Since the vast majority of tendons remained in tension at any given time, the concrete Containments remained under compressive stress and retained the associated elastic deformations from stressing in the

early 1970s. Equations were developed to determine the amount of average prestress loss due to containment concrete creep and shrinkage between years 29.5 and 61.5, and the amount of prestress loss due to 2% (over 40 years) wire relaxation for 32 years only. Once the 32-year prestress losses were determined, another equation was fitted to the average new tendon initial stressing force and final average calculated tendon force at end-of-life. From this equation, tendon force versus time curves were developed to evaluate future tendon surveillance lift-off force test results. Since the tendon force versus time curves are used to evaluate tendon ISI surveillance lift-off data, the actual average wire relaxation test result of 2% was used in developing the curves instead of the maximum specified wire 40-year relaxation value of 4%.

The same approach was used for the Unit 2 new vertical tendons in developing final tendon force versus time curves. The only differences were the average tendon remaining life and the average initial new tendon stressing force.

5E.3.4 REFERENCES

1. NRC Regulatory Guide 1.35, Revision 2, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures"
2. Report From Ginna Nuclear Power Station, GAI Report No. 2499, Containment Vessel Tendons – Stress Relaxation Properties of Retensioned Wires, December 1, 1983

TABLE 5E-3

UNIT 1 TENDON POPULATIONS AND EFFECTIVE LIVES

UNIT 1

		Vertical Original	Vertical Restressed	Vertical New	Hoop	Dome
A	60-Year Plant License Expiration	7/31/2034	7/31/2034	7/31/2034	7/31/2034	7/31/2034
B	Average Stressing Date (Take as middle of month)	3/1973[a]	8/18/2002	8/20/2002	11/1971[b]	11/1971[b]
C	Years from Stressing Until Plant End-of-Life: A-B (Rounded to 1/4 year)	61.5	32	32	62.75	62.75

[a] Per the original Unit 1 prestressing report, nearly the entire population of Unit 1 vertical tendons was detensioned and repairs were made to the anchor bearing plates after initial stressing. The repairs and subsequent restressing of the tendons occurred between January and April 1973. Average date taken was March 1973.

[b] Per the original Unit 1 prestressing report, the Unit 1 tendons were initially stressed over a period of nine months between June 1971 and March 1972. Average date taken was November 1971.

TABLE 5E-4

UNIT 2 TENDON POPULATIONS AND EFFECTIVE LIVES

UNIT 2

		Vertical Original	Vertical Restressed	Vertical New	Hoop	Dome
A	60-Year Plant License Expiration	8/13/2036	8/13/2036	8/13/2036	8/13/2036	8/13/2036
B	Average Stressing Date (Take as middle of month)	6/1973[a]	5/2002	6/2002	10/1972[b]	10/1972[b]
C	Years from Stressing Until Plant End-of-Life: A-B (Rounded to 1/4 year)	63.25	34.25	34.25	63.75	63.75

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[a] Per the original Unit 2 prestressing report, nearly the entire population of Unit 2 vertical tendons was detensioned and repairs were made to the anchor bearing plates after initial stressing. The repairs and subsequent restressing of the tendons occurred between April and September 1973. Average date taken was June 1973.

[b] Per the original Unit 2 prestressing report, the Unit 2 tendons were initially stressed over a period of eleven months between May 1972 and April 1973. Average date taken was October 1972.

5E.4 VERTICAL TENDON PRESTRESS LOSSES DUE TO FUTURE WIRE BREAKAGE

In accordance with American Concrete Institute Code 318-63, the prestress system has been designed for prestress losses from the following effects:

a. Seating of anchorage;
b. Elastic shortening of concrete;
c. Creep of concrete;
d. Shrinkage of concrete;
e. Relaxation of prestressing steel stress; and
f. Frictional loss due to intended or unintended curvature in the tendons

Due to the discovery of corrosion at the top end of vertical tendons on both Units, the prestress system has also been designed for possible vertical tendon individual wire breaks. It is anticipated that over the remaining operating life of both Units, wire breakage will occur in the remaining original vertical tendon population. Unit 1 will meet its design basis with up to 1,195 additional broken vertical tendon wires, and Unit 2 will meet its design basis with up to 1,228 additional broken vertical tendon wires.

The tendon force versus time curves discussed in Section 5E.3 for the three vertical sub-populations, hoop population, and dome population on each Unit are based on 90-1/4" diameter wire tendons. Should individual wire breakage be discovered in tendons during future inspections, the tendon force versus time curves will be proportioned-down by the ratio of unbroken wires to 90. The difference between the proportioned down tendon force versus time curve, and the 90-wire tendon curve, will represent the prestress loss due to wire breakage at that period of tendon life.

The prestress losses caused by the effects listed in a through f, above, occur for the most part uniformly throughout the Containment Structures. The distribution of possible future vertical tendon wire breakage is predicted to be skewed heavily toward the more severely corroded remaining vertical tendons. This was a critical factor in the evaluation to support 1,195 (Unit 1) and 1,228 (Unit 2) wire breaks.

Therefore, in addition to monitoring the loss of prestress due to the gross number of future wire breaks, the localized area effects of prestress losses will also be monitored as part of the enhanced inspections described in Section 5E.2.5.

5E.5 CONCLUSION

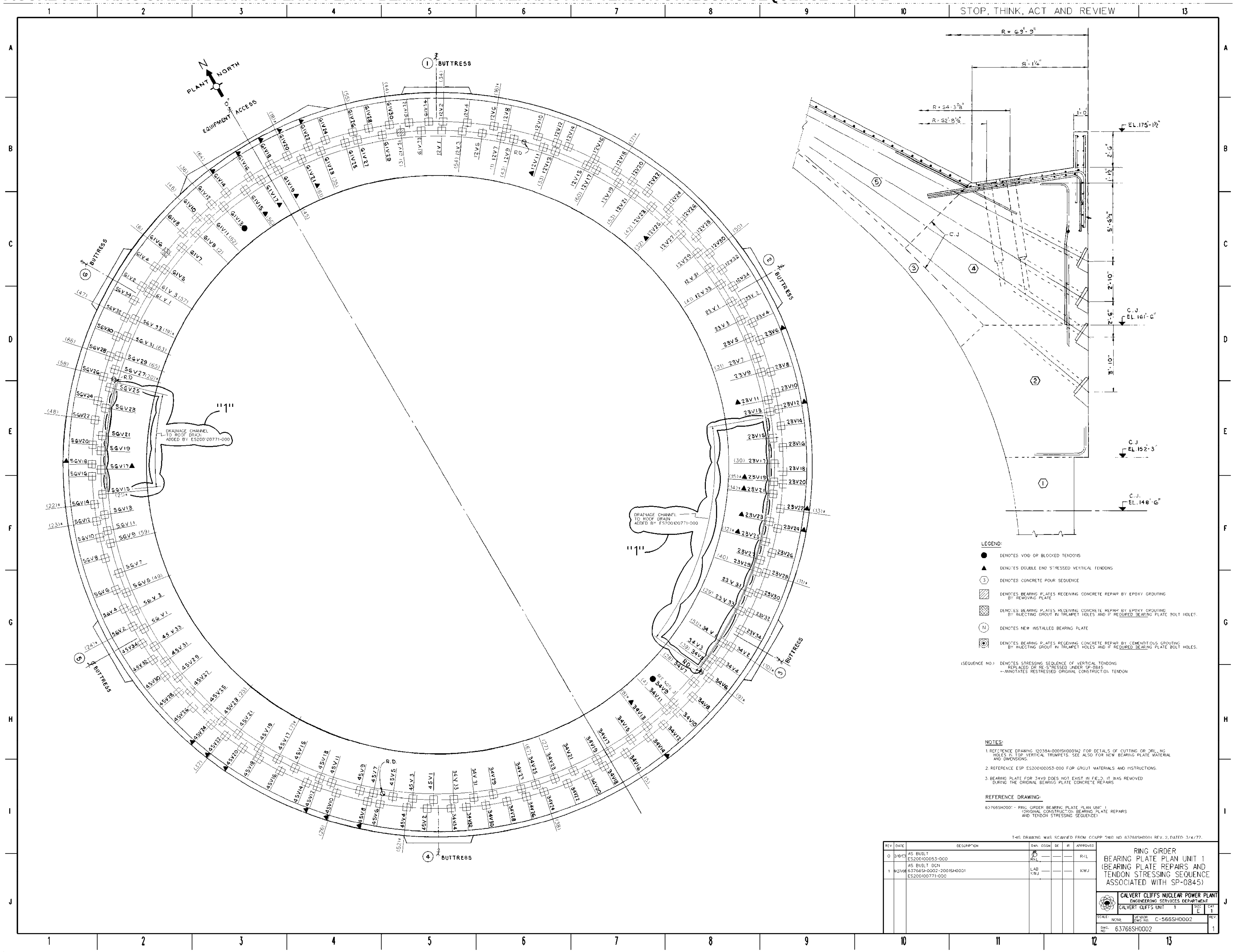
An engineering evaluation has demonstrated the acceptability of the CCNPP Units 1 and 2 concrete Containments with degraded conditions found during the 20-year (1997) containment tendon surveillance. The most severe conditions found were in the vertical tendon population. Some Units 1 and 2 vertical tendons, which have corrosion on individual wires below their upper (top) stressing washer, have not been replaced or repaired. The majority of the tendons exhibiting the greatest corrosion levels were replaced. In addition, vertical tendons that had exhibited lower than expected lift-off forces in 1997, and were not replaced, were restressed. The Containments were also demonstrated acceptable after considering the prediction that some corroded wires in tendons not replaced will break over the operating life of the two Containments. The Units 1 and 2 Containments will have sufficient vertical tendon prestress at the end of their operating licenses to meet the minimum design basis requirement of 123,620 kips gross vertical prestress, and 606 kips mean average force per tendon sheath.

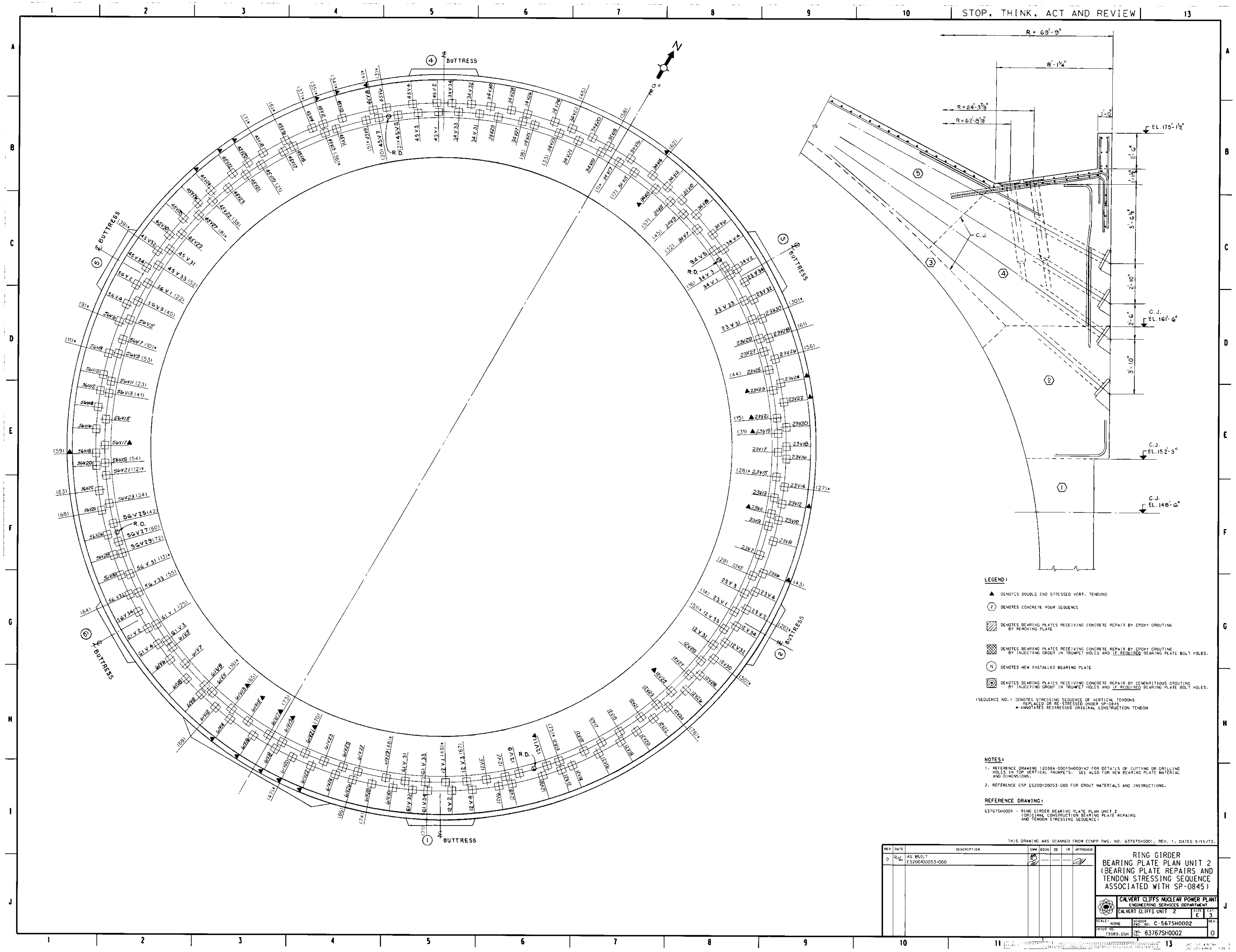
When its operating license expires July 31, 2034, the Unit 1 vertical tendon prestress will have appreciable margin above the two minimum design values. The 47 new tendons and 20 restressed tendons were seated to at least 742 and 725 kips, respectively. In addition, the design basis vertical prestress can be achieved with up to 1,195 additional wires breaking in the original non-replaced vertical tendon population between 2002 and July 31, 2034.

When its operating license expires August 13, 2036, the Unit 2 vertical tendon prestress will have appreciable margin above the two minimum design values. The 46 new tendons and 30 restressed tendons were seated to at least 742 and 725 kips, respectively. In addition, the design basis vertical prestress can be achieved with up to 1,228 additional wires breaking in the original non-replaced vertical tendon population between 2002 and August 13, 2036.

Table 5E-2 of Section 5E.2.4 shows that the Unit 1 Containment will have greater vertical tendon prestress at the end of its operating license after tendon replacement, tendon restressing, and predicted wire breaks, than it would have under the original tendon system design. However, the table also shows that the Unit 2 Containment will have slightly less vertical tendon prestress at the end of its operating license after tendon replacement, tendon restressing, and predicted wire breaks, than it would have under the original tendon system design. Although there is potentially less design margin above the minimum requirements, there is still sufficient margin. There was conservatism used in the development of the new calculated end-of-life vertical prestress. In addition, CCNPP expects that future tendon inspection data will verify that there will be far less actual wire breaks than currently assumed, allowing the predicted margin at the end of the extended operating licenses to approach (in Unit 2's case) or far exceed (in Unit 1's case) the original design margin.

FIGURE 5E-1 RING GIRDER BEARING PLANT PLAN – BEARING PLATE REPAIRS AND TENDON STRESSING SEQUENCE – UNIT 1




STOP, THINK, ACT AND REVIEW
BUTTRESS
R = 69'-9"
8'-1¼"
EL. 175'-1½"
C.J.
EL. 161'-6"
EL. 152'-3"
EL. 148'-6"
LEGEND:
DENOTES DOUBLE END STRESSED VERT. TENDONS
DENOTES CONCRETE POUR SEQUENCE
DENOTES BEARING PLATES RECEIVING CONCRETE REPAIR BY EPOXY GROUTING BY REMOVING PLATE
DENOTES BEARING PLATES RECEIVING CONCRETE REPAIR BY EPOXY GROUTING BY INJECTING GROUT IN TRUMPET HOLES AND IF REQUIRED BEARING PLATE BOLT HOLES.
DENOTES NEW INSTALLED BEARING PLATE
DENOTES BEARING PLATES RECEIVING CONCRETE REPAIR BY CEMENTITIOUS GROUTING BY INJECTING GROUT IN TRUMPET HOLES AND IF REQUIRED BEARING PLATE BOLT HOLES.
(SEQUENCE NO.) DENOTES STRESSING SEQUENCE OF VERTICAL TENDONS REPLACED OR RE-STRESSED UNDER SP-0845
*-ANNOTATES RESTRESSED ORIGINAL CONSTRUCTION TENDON
NOTES:
1. REFERENCE DRAWING 12038A-0001SH0001A2 FOR DETAILS OF CUTTING OR DRILLING HOLES IN TOP VERTICAL TRUMPETS. SEE ALSO FOR NEW BEARING PLATE MATERIAL AND DIMENSIONS.
2. REFERENCE ESP ES200100053-000 FOR GROUT MATERIALS AND INSTRUCTIONS.
REFERENCE DRAWING:
63767SH0001 - RING GIRDER BEARING PLATE PLAN UNIT 2 (ORIGINAL CONSTRUCTION BEARING PLATE REPAIRS AND TENDON STRESSING SEQUENCE)
THIS DRAWING WAS SCANNED FROM CCNPP DWG. NO. 63767SH0001, REV. 1, DATED 5/15/73.
REV DATE DESCRIPTION DWN DSGN DE IR APPROVED
0 AS BUILT ES200100053-000
RING GIRDER BEARING PLATE PLAN UNIT 2 (BEARING PLATE REPAIRS AND TENDON STRESSING SEQUENCE ASSOCIATED WITH SP-0845)
CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 2
SIZE E
CAT. 3
SCALE: NONE
VENDOR DWG. NO. C-567SH0002
INTER NO. 73089.DGN
DWG. NO. 63767SH0002
REV. 0

CHAPTER 6
ENGINEERED SAFETY FEATURES

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LIST OF ACRONYMS

AEC	Atomic Energy Commission
ANSI	American National Standards Institute
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWWA	American Water Works Association
BGE	Baltimore Gas and Electric Company
CE	Combustion Engineering, Inc.
CIS	Containment Isolation Signal
CRS	Containment Radiation Signal
CSAS	Containment Spray Actuation Signal
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DOP	Dioctyl-Phthalate
DRF	Dose-Reduction Factor
ECCS	Emergency Core Cooling System
EPR	Ethylene-Propylene-Rubber
ESF	Engineered Safety Feature
FCR	Facility Change Request
HEPA	High Efficiency Particulate Air
HPSI	High-Pressure Safety Injection
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss-of-Coolant Accident
LPSI	Low-Pressure Safety Injection
MWPS	Miscellaneous Waste Processing System
MOV	Motor-Operated Valve
NBS	National Bureau of Standards
NDE	Non-Destructive Examination
NEMA	National Electrical Manufacturers Association
NPSH	Net Positive Suction Head
ORNL	Oak Ridge National Laboratory
QC	Quality Control
RAS	Recirculation Actuation Signal
RCS	Reactor Coolant System
RWT	Refueling Water Tank
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SRW	Service Water
STB	Sodium Tetraborate Decahydrate
TEMA	Tubular Exchanger Manufacturers Association
UFSAR	Updated Final Safety Analysis Report
XLP	Cross-Linked Polyethylene

6.0 ENGINEERED SAFETY FEATURES

6.1 GENERAL

Several systems in Calvert Cliffs Units 1 and 2 are designated engineered safety features (ESF). The function of these systems is to localize, control, terminate, and otherwise mitigate the consequences of an accident and thereby enhance the protection of the public and plant personnel against the accidental release of fission products.

The systems function to cool the core, limit the magnitude and duration of a pressure transient within containment following a loss-of-coolant accident (LOCA), provide long-term post-accident cooling, and reduce the airborne radioactivity concentration in containment.

The systems defined as ESF are as follows:

a. The Safety Injection (SI) System [including as subsystems the safety injection tanks (SITs), high-pressure safety injection (HPSI) pumps and low-pressure safety injection (LPSI) pumps]:

   The SI system injects borated water into the Reactor Coolant System (RCS). The system supplies emergency core cooling to limit fuel rod damage and fission product release, and ensure adequate shutdown margin regardless of temperature. The injection system also supplies continuous long-term post-accident cooling of the core by recirculation of borated water from the containment sump. Portions of the SI system may be used to provide long-term cooling flush;

b. Containment cooling systems:

   1. The Containment Spray System:

      Removes heat by spraying cool, borated water through the containment atmosphere. Heat is transferred to the Component Cooling System through the shutdown cooling heat exchangers;

   2. The Containment Air Recirculation and Cooling System:

      Removes heat by circulating the post-accident containment atmosphere over coils cooled by service water (SRW);

c. Containment Penetration Room Ventilation System:

   Operation of the penetration room exhaust system ensures that radioactive materials discharging from the containment atmosphere following a LOCA are filtered prior to reaching the environment. High efficiency particulate and charcoal filters remove radioactivity associated with aerosols and the radioiodines;

d. Containment Iodine Removal System:

   Recycles the post-accident containment atmosphere through charcoal filters to remove radioiodines.

The essential supporting systems include the control systems, the normal and emergency power systems and the component cooling, SRW and saltwater systems. These systems are described in Chapters 7, 8 and 9, respectively.

6.2 DESIGN BASIS

The ESF system components are procured to detailed engineering specifications and tested to codes listed in the following sections. The double-ended rupture of the largest reactor coolant pipe is designated as the design basis accident (DBA), since the forces and thermal phenomena affecting the core are the most severe. The maximum post-accident containment temperature, pressure and humidity in the Containment will ordinarily be developed by the loss of coolant that occurs with a somewhat smaller break. A discussion of the analyses performed for a spectrum of break sizes, to determine these limiting service conditions, is presented in Sections 14.17 and 14.20.

All components of the ESF system and associated critical instrumentation are designed to operate in the most severe environment to which they could be exposed in the event of a LOCA. These design requirements are of primary significance for those portions of the ESF system which are located inside the Containment Structure, including piping, valves, filters, cooling coils, instrumentation, electrical wiring and motors. This equipment is designed to operate in the containment design atmosphere of 50 psig, 276°F (maximum concrete temperature), 100% relative humidity and 10^8 rads in the year following the accident. NOTE: All safety-related equipment in the containment have been evaluated for the revised maximum vapor temperature contained in Section 14.20.

Portions of the Safety Injection System are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the Safety Injection System fatigue analysis considers the injection of cool safety injection flow into the RCS at various RCS pressures and temperatures. See Reference 1 for further details, including the allowed number of injection transients and the assumed operating conditions of these transients.

Sections 14.13, 14.14, 14.17, and 14.26 postulate those accidents that could result in the combined radiation, high temperature, pressure, and humidity environment. Other accidents discussed in Section 14.0 do not require safety-related electrical and mechanical equipment and components to function in an environment other than the normal environment.

For those accidents not resulting in a combined high radiation, temperature, pressure and humidity environment, the electrical and mechanical systems which must function are discussed in the appropriate subsections of 14.0 and the systems are further delineated in the applicable sections of the Updated Final Safety Analysis Report (UFSAR).

For a list of all equipment and components, including those outside of the containment and not exposed to the LOCA environment, refer to Figure 7-10 (Engineered Safety Features Actuation System Logic Diagram).

The solenoids and motor operators for containment isolation valves within the containment which are required to operate following a LOCA will be qualified for maximum LOCA environmental conditions.

The forces generated by the maximum hypothetical earthquake, combined with the rupture of a reactor coolant pipe, are considered in the design of the ESF. The design ensures the functional capability of the system will be retained. Vessels which are connected to the ESF systems are supported and restrained to allow controlled movement during this load condition, and piping is designed to accept these imposed movements. Engineering calculations of the flexibility of the systems have been performed to verify that the piping can accept these additional vessel movements and still remain within code allowable limits of stress. Flexibility calculations have been performed according to the Code of Pressure Piping, American National Standards Institute (ANSI) B31.7. These systems also have been designed to minimize the effects of water hammer through pipe layout, support selection and support location. Supports

and hydraulic snubbers are designed to withstand the loads which could result from the quick closure of valves.

6.2.1 REFERENCES

1. Bechtel Specification 6750-M-0310B, "Design Specification for Piping, Valves, and Associated Equipment of the Safety Injection System"

6.3 SAFETY INJECTION SYSTEM

6.3.1 SYSTEM DESCRIPTION

The SI system is designed to supply emergency core cooling in the unlikely event of a LOCA. The system prevents fuel and cladding damage that would interfere with core cooling, and limit the cladding-water reaction to less than 1% for all breaks in the RCS piping up to and including the equivalent of a double-ended break in the largest coolant pipe, i.e., up to a flow area of 19.2 ft^2.

The safety injection water contains boron at a concentration required by technical specifications; consequently, the SI system also provides additional shutdown capability whenever the system is required to operate. This shutdown capability assists in maintaining the reactor subcritical following the rapid cooldown of the RCS caused by a rupture of a main steam line (Section 14.14). The reactivity provided by the SI system exceeds the minimum required for cold shutdown.

The SI system consists of high-pressure and low-pressure subsystems, shown on Figures 6-1 (Unit 1) and 6-10 (Unit 2).

The high-pressure subsystem is capable of delivering emergency coolant at a discharge pressure up to 1275 psia. Three HPSI pumps take suction from two independent suction headers. After the headers are initially supplied with at least 360,000 gallons of borated water from the Refueling Water Tank (RWT), a Recirculation Actuation Signal (RAS) occurs. The RAS shifts the suction of the headers from the RWT to the containment sump to recirculate the borated water.

The LPSI system utilizes four pressurized SITs and two LPSI pumps. Each of the two pumps is connected to one of the two independent suction headers which serve the high-pressure pumps. This assures an adequate supply of borated water.

Control valve CV-306, between the low-pressure pumps and the LPSI header, is a fail-open valve manufactured in accordance with the American Society of Mechanical Engineers (ASME) Code for Pumps and Valves for Nuclear Power. The material and welds were inspected, tested and documented to the same code. The normally locked open valve is not required to function during or subsequent to a LOCA.

The LPSI subsystem is designed to the performance capability required by Atomic Energy Commission (AEC) General Design Criterion 41. Criterion 41 requires that, as a minimum, each ESF shall provide its required safety function assuming a failure of a single active component. Failure of a pipe or valve body pressure boundary constitutes a passive failure. The system piping is fabricated and constructed in accordance with the ANSI B31.7 Code - Nuclear Power Piping. Materials and welds were inspected, tested and documented in accordance with the applicable class of B31.7.

A passive failure of such high quality components, inspected throughout plant life, is not considered credible over a short-term period.

During the recirculation mode (the earliest this is reached is about 30 minutes following the break), the LPSI subsystem is automatically shut down and the HPSI pumps are used to recirculate water from the containment sump. Thus, the LPSI subsystem functions over a short-term period for the transient.

The water from the SITs re-covers the core following a RCS blowdown to minimize core damage until the SI pumps can provide adequate water for reactor cooling. In the normal mode, there are two check valves, with a locked open, motor-operated valve (MOV) in series between the pressurized SIT and the RCS. The tanks are designed to passively

inject large quantities of borated water into the RCS immediately following a large pipe break. The water covers and cools the core, thereby limiting clad-melting and metal-water reaction (Section 14.17). The separate and independent tanks are each connected to one of four reactor vessel inlet pipes. The driving head for water injection is provided by nitrogen gas pressure within the tanks at a minimum pressure of 200 psig. As the RCS pressure falls below tank pressure, check valves open in the line connecting each tank to the system. The tanks operate as a passive stored-energy safety feature; no outside power or signal is required for their operation. A remotely-operated valve is provided to isolate the tanks during a normal depressurization of the RCS. The position of these motor control valves is displayed by the use of lights on the main control boards. Two lights are provided for each valve that indicate open or closed and are actuated by individual limit switches on the valve motor operator. A Control Room annunciator is actuated whenever the control switch is not in the open position. These valves are positioned and the position is checked as part of the startup check off. Once the valves are placed in the open position, a key is required to move the control switch and thus change the position of the valve. During normal operation, these valves are required to be open with power removed by maintaining the feeder breaker open. Furthermore, the system functions such that the four valves will be automatically opened whenever primary system pressure exceeds a preset value or a SI actuation signal (SIAS) is present. Two of the valves will be opened by signal from pressurizer pressure bistable PY103X and the other two by the redundant pressurizer pressure bistable PY103X-1. Safety Injection Actuation Signal Channel A opens two of the valves and Channel B opens the other two. A small drain valve controlled remotely from the Control Room is used to drain any inleakage from the RCS. The SITs are protected from overpressure by relief valves. Piping to each tank is arranged such that the operability of each tank can be demonstrated.

The tank size, gas/water fraction, gas pressure, and outlet pipe size for the SITs were selected to fulfill the following criteria:

a. The volume of water in the SITs is selected so that the contents of three of the four tanks injecting into the RCS following the worst case LOCA, will cover the top of the core;
b. The tank gas/water fractions, gas pressure, and outlet pipe size are selected to allow three of the four tanks to re-cover the core before significant clad melting or zirconium-water reaction can occur.

Safety injection is initiated either when the pressurizer pressure drops below the trip setpoint or when the containment pressure rises above the trip setpoint. Diversity of the SIAS is thus provided.

Upon initiation of SI, two of the three high-pressure and two LPSI pumps start and twelve SI line isolation valves open, injecting water from the RWT into the RCS. After sufficient water has been transferred from the tank, a continuous source of borated water is provided by recirculating containment sump water directly to the pump suction. Recirculation is automatically initiated by low water level in the RWT. In the event the automatic transfer fails to occur upon RAS, the operator can manually initiate recirculation using two buttons labeled Recirculation Manual Actuation Channel "A" ("B") on the main control board. As an additional backup, transfer to the recirculation mode may also be manually initiated by the Control Room operator via individual component handswitches.

Net Positive Suction Head (NPSH) is ensured for the HPSI pumps because the containment sump water level is 14'5" plant elevation at the time of RAS, and the centerline of the pump is at -12' plant elevation.
All manual valves that constitute barriers to the environment for isolation of the RWT are locked closed. Redundant, automatic motor-operated valves are provided on the safety

features pumps recirculation line back to the tank. These valves may receive a close signal upon a RAS. However, this signal is normally overridden to ensure minimum flow for SI pumps.

There are no interlocks associated with the containment sump recirculation line motor-operated valves. These valves are closed during normal operation. When the valves are stroked during periodic testing, the check valves located downstream of the sump recirculation line valves protect against dumping the RWT contents into the containment sump. The check valves are designed to seat against the head of water from the RWT. The check valves can be tested for seat leakage via a drain connection between the sump recirculation valves and the check valves.

The valves will be operated from the ESF panels in the Control Room by simulating a RAS. This signal will open a 24" motor-operated sump valve on one line. The valve position lights in the Control Room will verify that the valve has functioned properly. Upon verification that the valve has functioned properly, the operator will return the valve to its original position and test the remaining channel in the same manner.

The minimum time at which switchover to the recirculation mode could be required is 30 minutes. This period is based upon operation of two high-pressure pumps, both low-pressure pumps, and the containment spray pumps, all operating at or above design capacity. The maximum charging flow from the RWT is also considered. The system is designed to keep the core covered following initial SI. One high-pressure pump has sufficient capacity with complete spillage of the maximum flow leg to maintain core water level at the start of recirculation.

The automatic recirculation signal shuts down the LPSI pumps, sends an open signal to both recirculation line isolation valves, and a close signal to the minimum flow line isolation valves to signal isolation of the RWT. However, the minimum flow valves are normally locked out in the open position in the Control Room to prevent possible SI pump damage. The Control Room valve handswitch is actually locked out. The lock-out switch is turned to ON when the RWT reaches a low level before RAS, allowing the valves to automatically close on RAS. The HPSI pumps continue to operate to provide core cooling water.

A key-operated manual override of the RAS contact in the LPSI pumps is installed to allow the LPSI pumps to run regardless of RWT level during long-term cooling.

In the recirculation mode, the HPSI pumps take suction directly from the containment sump. At the discretion of the operator, a portion of the cooled water from the Containment Spray System may be diverted to the suction of the HPSI pumps. The spray diversion valves are not powered from the diesel generators and therefore are not available on loss of offsite power. This method of operation provides additional cooling and NPSH, but is not necessary to meet core cooling requirements.

The containment spray pumps are centrifugal pumps which discharge at the design flow rate against containment design pressure and system losses. When the system is switched over to the recirculation mode of operation, the pumps take suction from the Containment Building and therefore do not have to pump against the pressure in the containment. The reduced pumping requirements during recirculation cause the pump to operate at a higher capacity thus providing more spray flow. This excess spray flow can be diverted to the suction of the HPSI pumps without compromising the Containment Spray Systems effectiveness.

The design of the suction piping in the SI system conservatively assumes that the recirculated fluid is saturated at containment pressure and temperature conditions, therefore subcooled fluid at the SI pump inlet is not needed to meet NPSH requirements.

However, in the unlikely event of cavitation in the SI pump, the operator could divert a portion of the Containment Spray System water to the suction of the SI system pumps if offsite power is available. The operator can monitor the SI pumps for cavitation by observing SI system flow meters, pump ammeters, and pressure gauges for decreased and erratic readings.

The LPSI pumps are also used to supply coolant flow to remove heat from the reactor following reactor shutdown and to maintain a suitable temperature for refueling and maintenance operations. In this mode the system is designed to cool the RCS from 300°F to 130°F. The maximum coolant pressure during this cooldown is approximately 300 psig.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

6.3.2 SYSTEM COMPONENTS

6.3.2.1 High-Pressure Safety Injection Pumps

The HPSI pumps are sized to ensure that one high-pressure pump will keep the core covered at the start of recirculation, assuming complete spillage of the maximum flow leg.

The requirements for boron injection for the steam line break and the injection requirements for smaller break sizes are also considered in the sizing. The high-pressure pumps are designed for the thermal transient conditions of 40°F to 300°F in 5 to 10 seconds and 300°F to 40°F in 5 to 10 seconds.

The high-pressure pumps are seven-stage, horizontal, centrifugal units. Mechanical seals are used and are provided with leak-offs to the Waste Processing System which collects any leakage past the seals. The seals are cooled by water that is taken from the pump flow and is cooled by component cooling water flow in the seal cooler heat exchanger of each pump. With the heat exchangers in operation, the seals can operate continuously at a pump flow temperature of 300°F. If the heat exchanger function is lost post-RAS due to loss of component cooling water, the seals can operate at least 2 hours at a pump flow temperature of up to 250°F. This 2 hours will allow manual realignment of valves to allow the containment spray pumps to substitute for the HPSI pumps. The bearings of each HPSI pump are cooled by component cooling water flow to the bearing housings. If component cooling water flow is lost, the bearings can also operate at least 2 hours. The pump motor is capable of starting and accelerating the pump to full speed with 75% of rated voltage. The pumps are provided with drain and flushing connections to permit reduction of the radiation levels before maintenance. The pressure-containing parts of the pump are stainless steel with internals selected for compatibility with boric acid. The materials selected were analyzed to ensure that differential expansion during design transients can be accommodated with the clearances selected. The following inspections were performed on the HPSI pumps:

a. All surfaces of pressure-containing castings were liquid penetrant inspected in accordance with techniques and acceptance standards of ANSI B31.1, Paragraph 136.5.3(d);

b. Pressure containing parts were hydrostatically tested in accordance with American Petroleum Institute (API) Standard 610, Paragraph 34 except design pressure was used in lieu of maximum discharge pressure.

The pumps are provided with minimum flow protection to prevent damage resulting from operation against a closed discharge. These are nearly identical in design to those used for Palisades plant of Consumers Power Company (Docket No. 50-255) except for a slightly larger impeller. One Palisades pump was subjected to the following transient tests while operating at the design point:

a. Suction temperature increase from 70°F to 315°F in a range of 5 seconds;

b. Suction temperature decrease from 300°F to 70°F in a range of 10 seconds.

No adverse effects of the temperature transients upon pump performance were noted, nor was there any excessive vibration which would indicate any tendency of running parts to bind. When the tests were completed, the pump was disassembled and inspected. No abnormal wear was noted.

A full-scale hydraulic test was performed on each Calvert Cliffs pump assembly. All pump test setups, test procedures and instrumentation were in accordance with the Standards of the Hydraulic Institute and the ASME Power Test Code, PTC 8.2. This included verification of satisfactory operation at the stated NPSH. Figure 6-3 shows the pump performance obtained during the hydraulic testing. During the spring 1985 Unit 1 refueling outage extended HPSI pump flow testing was performed on HPSI Pump Nos. 11 and 13. This testing demonstrated that the pumps would operate satisfactorily in the pre-RAS condition with the minimum available system resistance. It is necessary in the post-RAS condition to throttle back HPSI flow to maintain adequate NPSH margin.

The high-pressure pump data are shown in Table 6-1.

The design temperature for the HPSI pumps is based upon the saturation temperature of the reactor coolant at the containment design pressure, about 300°F, plus a design tolerance of 50°F, yielding a design temperature of 350°F. The design pressure for the HPSI pumps is based upon the sum of the LPSI pump design pressure and the shutoff head of the HPSI pump.

6.3.2.2 Low-Pressure Safety Injection Pumps

The LPSI pumps are horizontal, single-stage, centrifugal units equipped with mechanical face seals backed up by a bushing, with a leak-off to collect the leakage past the seal. The seals are cooled by water that is taken from the pump flow and is cooled by component cooling water flow in the seal cooler heat exchanger of each pump. With the heat exchangers in operation, the seals can operate continuously at a pump flow temperature of 300°F. If the heat exchanger function is lost post-RAS due to loss of component cooling water, the seals can operate at least 2 hours at a pump flow temperature of up to 350°F. This 2 hours will allow manual realignment of valves to allow the containment spray pumps to substitute for the LPSI pumps. The bearings of each LPSI pump are cooled by component cooling water flow to the bearing housings. If component cooling water flow is lost, the bearings can also operate at least 2 hours. The pump motor is capable of starting and accelerating the pump to full speed with 75% of rated

voltage. The pumps are provided with drain and flushing connections to permit reduction of radiation levels before maintenance. The pressure-containing parts are fabricated from stainless steel; the internals are selected for compatibility with boric acid. The pumps are provided with minimum flow protection to prevent damage when starting against a closed system.

The pumps are of identical design to those used at the Palisades plant, Docket No. 50-255; one of these pumps was subjected to a suction temperature which was changed from 84°F to 351°F in 10 seconds while the pump was operating at the design point. No adverse effect upon pump performance was caused by this transient, nor was any excessive vibration detected that would indicate rubbing or binding. After the test was completed, the pump was disassembled and inspected. No abnormal wear was noted.

The following inspections were performed on the LPSI pumps:

a. All surfaces of pressure-containing castings were liquid penetrant inspected in accordance with techniques and acceptance standards of ANSI B31.1, Paragraph 136.5.3(d);
b. Pressure containing parts were hydrostatically tested in accordance with API Standard 610, Paragraph 34 except design pressure was used in lieu of maximum discharge pressure.

A full-scale hydraulic test of the pumps was performed. Figure 6-4 shows the pump performance obtained during the testing.

The design temperature for the LPSI pumps is based upon the temperature of the reactor coolant at the initiation of shutdown cooling, about 300°F nominal, plus a design tolerance of 50°F, yielding a total of 350°F. The design pressure for the low-pressure pumps is based upon the sum of the maximum pump suction pressure, which occurs at the initiation of shutdown cooling, and the pump shutoff head. This yields a nominal design pressure of 500 psig.

Table 6-2 contains the LPSI pump data.

6.3.2.3 Safety Injection Tanks

The four SITs are used to flood the core with borated water following a depressurization as a result of a LOCA. The tanks are sized to ensure that three of the four tanks will provide sufficient water to recover the core following a DBA. The tanks contain borated water at a sufficient boron concentration to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. The tanks are pressurized with nitrogen at 250 psig.

Level and pressure instrumentation is provided to monitor the availability of the tanks during plant operation. Verifying availability (boron concentration) consists of monitoring in-leakage and sampling when appropriate. Provisions have been made for sampling, filling, draining, venting and correcting boron concentration. The tanks are carbon steel internal clad with stainless steel. Design, construction and overpressure protection are in accordance with the ASME Code Section III, Class C.

Table 6-3 contains the SIT data.

6.3.2.4 Refueling Water Tank

The RWT is used to perform the following functions:

a. To provide borated water to the suction of the HPSI pumps, LPSI pumps and containment spray pumps for the initial operation of these pumps following a LOCA;

b. To provide a makeup water for the spent fuel pool;

c. To provide a storage for the contents of the refueling pool.

The tank is provided with connections for level control, sampling, filling, draining and venting. It is a flat bottom tank with a concave roof. The floor to shell weld is a double fillet; the shell is connected to the roof by a rolled angle. This angle is butt-welded to the shell and joined to the roof by a single lap weld. The tank is designed and fabricated to ASME Section III, Class C, and the floor welds, shell to bottom welds, and bottom course vertical welds were inspected by vacuum box. (ASME Code, Section III, Class C. "ND-3800, Design of Atmospheric and 0 to 15 psig Storage Tanks" follows very closely the API 650 Code. Due to the lack of shell weld definition in ASME Code, Section III, the definition of shell weld contained in API 650 is observed.)

In order to protect the tank contents from freezing in winter, a 60-gpm pump provides circulation through an external heat exchanger, heated by plant heating system hot water. Surveillance of the temperature of the tank contents is required by the Technical Specifications.

Periodic sampling of boron concentration and tank level surveillance are required by the Technical Specifications. A sample connection on the RWT is provided for these periodic samples which are taken to check the chemistry. The tank has an interconnecting line to the Chemical and Volume Control System (CVCS) which can be used to maintain the proper chemistry. The turbidity and radiological quality limits will be met by processing the water, while it is in the refueling pool or in the RWT, using the spent fuel pool purification system and/or portions of the CVCS.

The RWT is provided with both a wide range and narrow range level indicator. The narrow range instrument provides both a high and low level alarm. The wide range instrument provides only a low level alarm. The high level alarm is to alert the operators of an impending overflow of water from the RWT to the Miscellaneous Waste Processing System (MWPS). The low level alarms are used to assist the operator in monitoring for sufficient water inventory in the RWT. Redundant temperature instruments provide both high and low temperature alarms.

6.3.2.5 Shutdown Cooling Heat Exchangers

The shutdown cooling heat exchangers are used to remove decay and sensible heat during plant cooldowns, cold shutdowns and emergency containment spray operation. Additionally, the units are capable of achieving refueling temperature (≤140°F) within 36 hours following shutdown of an infinitely irradiated core, given a maximum component cooling water temperature of 120°F. The units are further specified to accept a 40°F to 276°F transient in 10 seconds when the containment spray pump suction is switched to the containment sump. The units are designed and constructed to the standards of ASME, Section III, Class C, and Tubular Exchanger Manufacturers Association (TEMA), Class R requirements. The units are of a U-tube design with two tube-side passes and one shell-side pass. The tubes are austenitic stainless steel and the shell is carbon steel.

The design temperature and pressure for the shutdown cooling heat exchangers are compatible with the design temperature and pressure for the LPSI pumps (Section 6.3.2.2).

Table 6-5 contains the shutdown cooling heat exchanger data.

6.3.2.6 Piping

The SI system piping is fabricated of austenitic stainless steel and conforms to the standards set forth in ANSI B31.7. Flexibility and seismic loading analyses have confirmed the adequacy of the system piping.

Safety Injection Piping

All piping is fabricated and constructed in accordance with the ANSI B31.7 Code, Nuclear Power Piping. Material and welds were inspected, tested and documented in accordance with the applicable class of B31.7. All valves are manufactured in accordance with the ASME Code for Pumps and Valves for Nuclear Power. The material and welds were inspected, tested and documented to the same code.

Recirculation Piping

Engineered safety features piping connected to the containment sump is an extension of reactor containment during the recirculation mode of core and containment cooling. The following items pertain to suction piping from the sump to the first isolation valve:

a. The piping has a nominal wall thickness of 3/8" which results in a maximum allowable pressure for the pipe minimum wall thickness of at least 12.5 times the maximum expected pressure of 50 psig;
b. All piping was designed, fabricated, tested and inspected in accordance with ANSI B31.7, Class II, including weld and material testing. The valves were manufactured, inspected and tested in accordance with the ASME Code for Pumps and Valves for Nuclear Power, Class II;
c. The recirculation lines from the containment out to and including the first isolation valve is enclosed in a pressure and leak-tight encapsulation barrier. This barrier will contain any possible leakage resulting from postulated pressure failure of the pipe or valve. The barrier is tightly attached to the exterior concrete of the containment. The encapsulation barrier is designed for 50 psig and is in accordance with the criteria for Category I structures;
d. Pipe material is Type 304 stainless steel.

The containment sump suctions are enclosed by particulate screens as described in Section 6.4.

6.3.3 SYSTEM OPERATION

Any condition which causes a low pressurizer pressure or high containment pressure will result in a SIAS. This signal will start two HPSI pumps, both LPSI pumps, open twelve SI system isolation valves and close the four check valve leak-off lines at the SITs. (Safety Injection Actuation Signal also performs some functions in the CVCS; Section 9.1.)

When RCS pressure falls below approximately 1275 psig, the HPSI pumps start delivering flow through both the high-pressure header and the auxiliary high-pressure header.

If reactor coolant pressure falls below approximately 200 psig, the passive pressurized SITs will start delivering flow into each cold leg along with the LPSI pumps.

The SI pumps initially draw borated water from the RWT. This tank has sufficient water volume to supply SI flow for at least 30 minutes assuming two high-pressure and two LPSI pumps and two containment spray pumps are running. The maximum charging flow from the RWT is also considered. When the RWT level reaches the RAS setpoint, a signal occurs which opens the isolation valves in the two lines from the containment sump and shuts down the LPSI pumps. At the operator's discretion, the use of the key-operated manual override of the RAS to the LPSI pumps is allowed as long as the minimum flow requirements of the LPSI pumps are met. The RWT suction valves remain open initially during the switch to the recirculation mode to preclude the loss of supply to a HPSI pump in the unlikely event the isolation valve in the containment sump line should experience delay in opening. Back flow through either RWT suction line is prevented by check valves. In addition, the operator will manually close the RWT suction valves after verifying the opening of the containment sump line valves. The mini-flow isolation valves will automatically close on RAS to prevent containment sump water from entering the RWT. However, the minimum flow valves are normally locked out in the open position in the Control Room to prevent possible SI pump damage. The Control Room Valve handswitch is actually locked out. The lock-out switch is turned to ON when the RWT reaches a low level before RAS, allowing the valves to automatically close on RAS. The earliest automatic recirculation would occur is 30 minutes assuming two HPSI, two LPSI, and two Containment Spray pumps are running. The maximum charging flow from the RWT is also considered. The recirculation mode can also be accomplished manually by the operator.

In the recirculation mode the HPSI pumps take suction from the containment sump. The SI flow spilling from the break in the RCS is cooled by mixing in the containment sump with the cooler containment spray water.

6.3.3.1 ECCS Long-Term Cooling Flush

During post-accident long-term cooling conditions for a large cold leg break, it is possible to have a boric acid reconcentration occur in the core area, due to the small core flow in effect. This condition may result in the crystallization of boric acid in the core and restriction of cooling flow. In order to prevent such an occurrence, two procedures have been developed. The operators will decide which method to use based on plant conditions.

Hot leg injection promotes flow through the core by establishing a flow path from the containment sump, through the LPSI system via the warm-up line, to the shutdown cooling suction line, and into the hot leg.

Pressurizer injection promotes flow through the core by diverting flow from one HPSI pump through the pressurizer auxiliary spray line and into the hot leg.

A minimum of 150 gpm injection flow (flow to the reactor coolant system hot side) is necessary to overcome the coolant boil-off rate and cause a net flushing flow downward through the core. The required injection flow must be provided within at least 11 hours after the accident.

The HPSI pumps are the preferred method of core flush. The LPSI pumps could be used for core flush only as a backup to the HPSI pumps when pressurizer injection is not possible.

The containment spray pumps can be aligned to provide core flush into the hot leg by realigning valves on the -10 foot level of the Auxiliary Building. This

realignment can be made after the containment spray pumps have completed the performance of their required post-LOCA function for spray service to reduce containment pressure. The containment spray pumps would only be used as a backup to the LPSI pumps.

6.3.4 DESIGN EVALUATION

The design bases and system requirements during a DBA are met with the operation of the SITs and one high-pressure and one LPSI pump, delivering rated flow and assuming complete spillage of the maximum flow leg through the break. During recirculation, one HPSI pump has sufficient capacity to maintain the water level in the reactor vessel above the core.

Ability to meet the core protection criteria is assured by the following features.

a. A high-capacity passive system (SITs) which requires no power source and will supply large quantities of borated water to rapidly recover the core after a major LOCA up to a break of the largest RCS pipe.

b. Low-pressure and high-pressure pumping and water storage systems with internal redundancy which will inject borated water. This capability provides core protection for RCS break sizes equal to and smaller than the largest line connected to the RCS (the 12" pressurizer surge lines or the shutdown cooling and SI lines). The pumping systems also provide borated water to keep the core covered and to continue cooling the core after the passive water supply has been exhausted. In addition, the high-pressure system will remove reactor core decay and sensible heat during long-term operation after the RCS rupture. Instrumentation and sampling provisions allow monitoring of the recirculated coolant.

c. Separated pump rooms and redundant pumping systems which will permit minimum safety features equipment to operate should pipe failure during long-term operation cause one pump room to flood.

d. Redundant onsite power supplies in the form of four emergency diesel generators, two dedicated to each unit, each of which has sufficient capacity for minimum safety features operation.

e. All active components which must function individually for the system to meet the performance criteria stated for core protection can be tested during normal reactor operation. Instrument sensors are tested for functioning at operating conditions. In addition, extensive shop and preoperational tests are performed to verify adequate component and system operation.

f. Most of the active components are located outside the containment where they are protected from accident-generated missiles and from post-accident environmental conditions. Those active components located inside the containment need only operate for a short time period after the accident.

g. The four injection lines are arranged such that movement of a ruptured reactor coolant pipe will not cause a subsequent failure of injection lines in non-ruptured loops. The maximum movement of the reactor coolant pipe at the injection nozzle in the non-ruptured loop will not damage the injection line.

h. The SI systems have been designed to meet the single failure criterion. This includes the fluid systems and the electrical control and instrument systems. All pumps and critical power-operated valves can be actuated from their respective switchgear or control centers. Instrumentation is also provided at locations other than the Control Room to ensure adequate control of the SI system if Control Room evacuation is required. Valve CV-306 in the LPSI system has instrument air removed from the valve operator to ensure the valve remains open under all required conditions.

i. All components, piping, cabling structures, power supplies, etc., in the SI support systems are designed to Category I seismic criteria.

   The effectiveness of the SI system to satisfy the criteria stated for core protection can be shown by the blowdown and refill transient curves following a LOCA. This analysis is presented in Section 14.17, LOCA.

6.3.5 SYSTEM RELIABILITY AND AVAILABILITY

The HPSI system is designed to minimize the amount of equipment which must operate when a SIAS is received. All valves not required to operate on initiation of SI are either isolated from the SI flow path or locked in the SI position during operation. Administrative controls ensure that the locked valves are in the correct position.

Three pumps are provided in the high-pressure system. Note that as described in Chapter 7, two HPSI pumps [11(21), 13(23)] are lined-up for automatic initiation, the third [12(22)] is in pull-to-lock. These pumps are located outside the containment. The pump rooms are in a controlled access area and are ventilated through charcoal filters to the plant vent. Floor drainage is collected and pumped to the MWPS.

The pump room location outside the containment is most favorable for extended operation and equipment life following a major LOCA. Temperature, humidity, and radiation levels will all be significantly lower, thus permitting the use of standard or more nearly standard equipment and components of proven performance and reliability. This location is also more rapidly accessible for service and inspection of the safety features systems components during plant operation and during the period of long-term cooling following the postulated LOCA.

The pumps are appropriately grouped, together with the pumps of the other ESF systems, in separate rooms. This arrangement permits access to, and operability of, those pumps required for minimum safety features operation.

With outside power available, the SIAS starts two high-pressure pumps and two low-pressure pumps. If outside power is not available, two high and two low-pressure pumps will be operated from the emergency diesel generators. Analyses of the LOCAs are performed assuming minimum ESF which includes only one high-pressure pump, one low-pressure pump and the four SITs (one spilling through the break).

Redundant flow paths are provided from the discharge of the HPSI pumps by independent HPSI headers. These headers, in turn, supply the four individual SI lines, one leading to each cold leg of the RCS.

Normal plant operating procedures include routine testing to ensure the operability of the pumps. The attention given to the selection of these pumps, the redundancy of power supplied, the design margins, and the fact that three pumps are installed assures a high degree of pump availability.

The HPSI valves are designed for 2485 psig. The power-operated SI isolation valves are located outside of the containment and are thus not subjected to the environmental conditions existing in the containment following any LOCA. The attention given to the selection of these valves, design margins, and the fact that eight HPSI valves are in parallel assure a high degree of valve availability. The main and auxiliary HPSI header supply valves are normally locked open.

Four SI valves are automatically opened from one emergency power bus on the initiation of SI; the remaining four HPSI valves are opened from the remaining emergency power

bus on receipt of a SIAS. There is a linear flow indicator for each safety injection line and a linear total flow indicator that provides the sum of all four high pressure safety injection flows. These indicators are located on Control Room panels 1/2C08 and 1/2C09. The valves are equipped with remote position indicators in the Control Room.

During recirculation, the HPSI pumps continue to operate, taking suction from the containment sump. The operator has the option of positioning valves so that the high-pressure pumps take partial suction from the shutdown cooling heat exchangers. These are supplied with sump water by the containment spray pumps or the LPSI pumps. The pump recirculation lines, the heat exchangers, the containment spray pumps and the recirculation suction headers are arranged to provide two independent flow paths. The LPSI pumps can also be used for recirculation, if necessary.

Fouling of heat transfer surfaces during the long-term post accident cooling mode is not expected to be a problem. The concentration of boric acid in the reactor vessel following blowdown is well below the solubility limit of 25 wt% at 200°F. Furthermore, boric acid solubility increases with increasing temperature and, consequently, precipitation onto hot surfaces is not expected. Means for core flushing are provided, should fouling of core heat transfer surfaces occur.

The LPSI system is also designed to minimize the amount of equipment which must operate when a SI signal is received. All valves not required to operate on initiation of SI are either isolated from the SI flow path or locked in the SI position during operation. Administrative controls ensure that the locked valves are in the correct position.

The two parallel pumps in the low-pressure system are appropriately grouped, together with the pumps of the other ESF systems, in separate rooms. This arrangement permits access to and operability of those pumps required for minimum safety features operation.

Normal plant operations, augmented by routine testing, ensure the operability of the pumps. The attention given to the selection of these pumps, the redundancy of power supplied, the design margins, and the fact that two pumps are supplied assures a high degree of pump availability.

The four SITs comprise a completely independent and redundant source of low-pressure injection water which requires no outside signal or source of power operation. The analysis in Chapter 14 shows that the core will be recovered quickly after a major break, before any clad melting occurs.

The power-operated LPSI valves are located outside of the containment and are thus not subjected to the environmental conditions existing in the containment following any LOCA. The attention given to the selection of these valves, design margins, and the fact that four LPSI valves are in parallel assure a high degree of valve availability. The SI valves are opened automatically on initiation of safety injection.

During recirculation, the low-pressure pumps are normally secured, but they may be arranged to take suction from the containment sump and discharge to the shutdown cooling heat exchangers or directly into the reactor vessel as backup either for the containment spray pumps or the HPSI pumps.

6.3.5.1 Tank Reliability

The tanks associated with the safety systems were ordered with quality control (QC) requirements consistent with the importance of these tanks to nuclear safety and plant reliability. Some of them were furnished under contract with Combustion

Engineering, Inc. (CE) and others were ordered according to Bechtel Associates' specifications.

For tanks ordered under the Bechtel specifications, the QC requirements are specified according to criteria set forth in Bechtel's Instruction BQC-201, which invokes certain parts of Bechtel's generic specification BQC-200 according to three levels or categories. The refueling water tanks, Condensate Storage Tank No. 12, radioactive waste tanks and spent fuel pool, were classified as Category 3, and as such, the following was required of tank manufacturers:

a. The contractor was required to have written procedures and instructions for control of all special fabrication and construction processes, such as welding, heat treating, cleaning, etc.
b. The contractor was required to have a system for assuring the control, identification and location of materials used in the finished item.
c. The vendor was required to provide copies of certain QC records and procedures such as welding procedures, non-destructive examination (NDE) results, etc.
d. The contractor was required to make available his QC program description document and the procedures, records and qualification governing the control of special processes noted in Items a, b, and c.
e. All manufacturers in this category were informed that they were subject to quality surveillance by Bechtel Associates, the customer or its agent as appropriate.

6.3.6 ROUTINE TESTS AND INSPECTIONS

Routine operational testing of major portions of the logic circuits, pumps and power-actuated valves in the SI system is described in Section 7.3.

The pumps are located outside the containment for access and to permit maintenance during normal plant operations. A recirculation line is provided on the discharge of each pump. Periodic testing will be performed by recirculating water back to the RWT.

Each SIT has two check valves in series between the tank nozzle and the RCS. These valves are tested periodically with other components of the system to assure their operability, in accordance with the Inservice Testing Program. Any one of the four SITs may be isolated for testing while shutdown.

Preoperational and periodic post-operational testing and inspections will be conducted to assure that the performance capability of the emergency core cooling system (ECCS), as installed, will conform to its design requirements during the life of the facility.

Preoperational tests of the SI system were performed after system installation prior to initial hydrostatic tests of the RCS. These preoperational tests were conducted in accordance with detailed test procedures which include testing of:

a. Safety Injection System automatic and remote operated control valves;
b. Low-pressure safety injection pumps;
c. High-pressure safety injection pumps;
d. Safety injection tanks;
e. Safety injection paths to the RCS; and,
f. Alarms and interlock circuitry.

The post-operational tests and inspections to be conducted on the SI system throughout plant life are described in the Technical Specifications.

Detailed test procedures for testing the SI system ensure proper operation of:

a. All SI pumps and their control circuits;

b. All actuator operated valves required to operate on receipt of SIAS and their control circuits;

c. The 12" SI check valves (SI-217, 227, 237, and 247);

d. The 12" SIT check valves (SI-215, 225, 235 and 245);

e. The 18" RWT discharge check valves (238M-1);

f. The 24" containment sump suction check valves (238M3-1);

g. 6" SI line check valves (207M3-1).

TABLE 6-1

HIGH-PRESSURE SAFETY INJECTION PUMP DATA SUMMARY

Quantity	3
Type	Seven-stage, Horizontal, Centrifugal
Motor Voltage	4000
Design Pressure, psig	1750
Design Temperature, °F	350
Design Flow (per pump), gpm	345
Design Head, ft	2500
Pumped Fluid	Reactor Coolant
Temperature of Pumped Fluid, °F	40-300
Shutoff Head, ft	2900
Maximum Flow, gpm	740
Head at Maximum Flow (one pump), ft	755
Material	Stainless Steel
Horsepower (motor)	400
Shaft Seal	Mechanical
Acceleration Time, seconds	4
Minimum Flow, gpm	30
NPSH Available pre-RAS (750 gpm)/post-RAS (620 gpm), ft	73.7/21.4
NPSH Required pre-RAS (750 gpm)/post-RAS (620 gpm), ft	40.0/19
Design Maximum Suction Pressure, psig	250

TABLE 6-2

LOW-PRESSURE SAFETY INJECTION PUMP DATA

Quantity	2
Type	Single Stage, Horizontal, Centrifugal
Motor Voltage	4000
Design Pressure, psig	500
Design Temperature, °F	350
Design Flow (per pump), gpm	3000
Design Head, ft	350
Pumped Fluid	Reactor Coolant
Temperature of Pumped Fluid, °F	40-300
Shutoff Head, ft	420
Maximum Flow, gpm	4500
Head at Maximum Flow (one pump), ft	235
Basic Material	Stainless Steel
Horsepower	400
Seals	Mechanical
Acceleration Time, seconds	4
NPSH Available (minimum), ft	30.0
NPSH Required at 3000 gpm, ft	12
Design Maximum Suction Pressure, psig	300
Minimum Flow, gpm	40

TABLE 6-3

SAFETY INJECTION TANK DATA

Quantity	4
Total Volume, ft^3	2000
Water Volume, ft^3	1113 (min.)
Design Pressure, psig	250
Operating Pressure, psig	200 (min.)
Design Temperature, °F	200
Operating Temperature, °F	120
Relief Valve Setpoint, psig	250

TABLE 6-5

SHUTDOWN COOLING HEAT EXCHANGER DATA

Quantity		2
Type		Shell and Tube
Codes		
	Tube Side	ASME Section III, Class C
	Shell Side	ASME Section III, Class C
Tube Side		
	Fluid	Reactor Coolant
	Design Pressure, psig	500
	Design Temperature, °F	450
	Pressure Loss, ($1.5x10^6$ lb/hr), psi	5
	Materials	Austenitic Stainless Steel
Shell Side		
	Fluid	Component Cooling Water
	Design Pressure, psig	150
	Design Temperature, °F	250
	Pressure Loss, ($2.41x10^6$ lb/hr), psi	10
	Materials	Carbon Steel

6.4 CONTAINMENT SPRAY SYSTEM

6.4.1 DESIGN BASIS

The function of the Containment Spray System is to limit the containment atmosphere pressure and temperature after a LOCA, and thus reduce the possibility of leakage of airborne radioactivity to the outside environment.

The original sizing of the containment air coolers for design and procurement was based on the heat removal capability, using three coolers, ($240x10^6$ Btu/hr) required to maintain the post-LOCA containment atmospheric pressure within the containment design pressure. Likewise, the original sizing of the containment spray system for design and procurement was based on the heat removal capability ($240x10^6$ Btu/hr) required to maintain the LOCA containment atmospheric pressure within the containment design pressure. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

6.4.2 SYSTEM DESCRIPTION

The Containment Spray System is shown in Figures 6-1 (Unit 1) and 6-10 (Unit 2). The system sprays cold borated water into the containment atmosphere and then recirculates and cools the water through the shutdown cooling heat exchangers. The Containment Spray System is redundant with the containment air cooling system and consists of two 50% capacity electric motor-driven pumps, two heat exchangers (the shutdown cooling system heat exchangers), two containment spray headers and nozzles, and all necessary piping, valves, instruments and accessories. The pumps discharge borated water from the RWT through the heat exchangers to the spray headers and nozzles located in the containment. The spray nozzles are arranged in the headers to give complete spray coverage at the containment horizontal cross-section area.

After an assumed LOCA, suction for the Containment Spray and SI Systems will be taken from the RWT. The water introduced into the containment in this manner will be intimately mixed with the water from the RCS. The resultant mixture will be used in the spray and injection systems only after the inventory of the RWT is nearly depleted.

The Technical Specification require a minimum inventory of 400,000 gallons be maintained in the RWT. Additional RWT data is provided in Table 6-4. The technical specification limit and RAS setpoint level have been established to ensure the RWT provides at least 360,000 gallons of usable water before the RAS is actuated. The capacity of the RCS is 77,800 gallons. The water in the RWT contains dissolved boric acid yielding a solution with a pH of approximately 5.0 at 80°F. If the reactor coolant has a pH of 10.0 due to an excess of LiOH produced from the radiolytic decomposition of boric acid and this reactor coolant is mixed with the water from the RWT, then a chemical acid-base reaction will occur, using up all the LiOH present, and the resulting solution will have a pH of 5.05. Therefore, the effect of the reactor coolant pH on the spray solution is controlled by the large volume of water from the RWT.

When the RWT level reaches the RAS setpoint, an RAS occurs which opens the containment sump discharge valves and continuation of containment spray is accomplished by automatic supply of the pump suction from the containment sump. The switchover is initiated, other than by operator action, on coincident low level signals from

level switches located in the RWT. Prior to discharge into the containment, the recirculated water is cooled by the shutdown cooling heat exchangers, using water from the component cooling water system.

The boric acid spray, with a pH of approximately 5, will come in contact with most surfaces in the containment including the equipment required for post-LOCA and, in some cases, on internal as well as external surfaces. To prevent chloride stress corrosion cracking of certain metals during operation of the ECCS, the spray water pH will be raised with sodium tetraborate (STB). The STB, located in the containment basement, is stored in baskets designed to allow dissolved STB to flow out. Each basket is constructed of stainless steel with 80 mesh screen sides, solid bottom and open top. The SI water will dissolve the chemical as it fills the containment. Mixing will be achieved as the solution is continuously recirculated and the final pH will be approximately 7.0. The minimum quantity of STB required to raise the pH to 7.0 has been calculated based on the boron concentration of the containment sump water following a LOCA (Reference 1).

All components and materials of the ESF located inside the containment have been designed to operate properly in the post-accident atmosphere. One of the major design considerations was the proper selection of components and materials which would not corrode or deteriorate in a manner that would hinder the operation of the ESF. All components were considered, including the transmitters, electrical wiring, equipment, valving and motors.

The electrical cables were subjected to environmental tests which proved that the cables remained in acceptable operating condition. The cables were flame-tested and subjected to environmental conditions to demonstrate the ability to operate successfully in the post-accident containment environment of radiation, temperature, pressure, boron concentration and the humidity as specified under LOCA conditions.

The flame propagation tests performed by BGE are summarized in IEEE published paper 71CP585-PWR, "Flame Propagation Tests on 600 Volt Control and Power Cables in Trays for Calvert Cliffs Nuclear Power Plant," presented at the IEEE Summer Power Meeting in Portland, Oregon, July 23, 1971. The major insulation selected for plant control, instrumentation and small power wiring was silicone-rubber (methyl-phenyl-vinyl base) with fiberglass and asbestos braid jackets. These cable makeups functioned satisfactorily both during and after exposure to severe oil fires. For large power cables, which are hard to ignite, the insulation consisted of cross-linked polyethylene (XLP) and ethylene-propylene-rubber (EPR) compounds which survived oil fires up to periods of 12 to 16 minutes.

The cables were also subjected to environmental tests performed by combinations of consulting laboratories and manufacturers' own test facilities. Cables (XLP, EPR, HTKerite and silicone-rubber) were actually subjected to the specified radiation and steam-boric acid exposures. All cables were affected in varying degrees both physically and in their insulating characteristics by the environmental exposures, but there was no significant degradation of any material's insulating ability after irradiation alone. The following table shows two critical parameters, tensile strength and elongation, before and after exposures.

	Tensile, psi		Elongation Percent	
Insulation	Original	After Exp.	Original	After Exp.
XLP	2403	2424	225	100
EPR	1050	600	375	80
HTK	800	670	380	220
Silicon Rubber	1260 to 1102	675 to 187	570 to 400	44 to 14

(All irradiation was by Cobalt-60 sources)

The tested insulations were all shown to be capable of satisfactory performance after exposure to the specified environmental conditions.

All valves in the containment are constructed of stainless steel or painted carbon steel, neither of which corrodes. Corrodible materials such as zinc and aluminum were strictly prohibited except where specifically approved. For example, aluminum was allowed to be used for the rotor of the motors which drive the containment cooler fans. These rotors are enclosed in the motor, and are further protected from the spray solution by two coats of epoxy. Examination of the motors, after operating in a simulated post-accident temperature, pressure, and spray environment, showed no degradation of any components. Galvanized surfaces were only approved for use in applications which would not interfere with the proper operation of any safety feature system, such as grating, ducting, and component casings. In general, all ESF were designed and/or tested to ensure that no harmful corrosion and/or deterioration would occur with pH potentials as high as 10.5.

Performance of the post-accident monitoring instrumentation will likewise not be harmfully affected by the post-accident environment. Identical components have been subjected to applicable temperature, pressure, boric acid, and radiation environmental tests, which proved that the instruments performed properly and did not deteriorate in any manner. All instrumentation has been located to reduce or eliminate the effects of post-LOCA environment. Instrumentation was located outside the containment or shielded within the containment to the maximum extent possible.

The original design capacity of the two containment spray pumps is such that they can limit the containment pressure to less than its design value following a LOCA without giving credit to the containment coolers. The containment pressure/temperature analysis with the containment spray system and the containment air coolers operating together post-LOCA in accordance with the Technical Specification requirements is provided in Section 14.20. A description of the components of the containment spray system is given in Table 6-6.

It is expected that the containment spray will be effective in removing fission products from the containment atmosphere. The method used for calculating the effectiveness of the spray is based on the methodologies detailed in References 3 through 5.

6.4.3 DESIGN EVALUATION

Separate suction headers from the refueling water storage tank are provided, one to each of the two separate and shielded rooms which house the pumps of the ESF systems. One pump room contains one spray pump, one LPSI pump and one HPSI pump. The other room contains one spray pump, one LPSI pump and two HPSI pumps. Separate headers, one to each of these pump rooms, are also provided from the containment sump.

Both suction recirculation headers from the emergency containment sump are completely enclosed by a structure consisting of a concrete curb and stainless steel plates reinforced by structural steel. Cassette type suction strainers filter the sump water (maximum opening is 1/16 inch diameter) which then feed into a stainless steel duct. This duct penetrates the concrete curb thus allowing the filtered water to reach the recirculation headers. This strainer design meets the requirements of Generic Letter 2004-02 (Reference 2). Drawing series 15960 provides the documents containing the details of the emergency containment sump strainer design.

This design constitutes a strong construction and will withstand severe shock and loading. With the wideness of the projected flow areas, it is very unlikely that the strainer will clog. Due to the extremely low flow velocity through the strainer box, the resulting pressure drop of the box construction will be negligible.

The system design is based on the spray water being heated to the temperature of the steam-air mixture within the containment. The nozzles will spray droplets with a mean diameter of approximately 700 microns with the spray system operating at design conditions and the containment at design pressure. In order that the spray droplets attain thermal equilibrium during the fall, adequate distance is provided between the spray nozzles and the highest obstruction in the containment.

The evaluation of post-incident containment pressure/temperature response is provided in Section 14.20.

6.4.4 SYSTEM RELIABILITY

The Containment Spray System has been designed to Seismic Category I criteria, including piping, valves, and containment spray pumps. The piping and valves have been purchased, designed, fabricated, inspected, cleaned, tested, and subject to material control in accordance with the quality assurance criteria presented in 10 CFR Part 50, Appendix B.

The containment spray pumps have been subjected to an extensive quality assurance program by which the manufacturer has controlled the material and manufacturing of these pumps. The pump castings were radiographed to assure their soundness and structural integrity. The manufacturer has performance-tested each containment spray pump and thereby demonstrated that the design requirements are satisfied. In addition, one spray pump was subjected to a thermal shock test, in which the temperature of the pumped fluid was raised from 100°F to 330°F in 10 seconds, to demonstrate that the pump will withstand the thermal shock that occurs when the recirculation mode begins after a LOCA. The manufacturer's shop has been audited to show that he can perform quality work in keeping with the requirements of 10 CFR Part 50, Appendix B.

All electrical equipment within the containment, which must function during and following a LOCA, has been qualified for the initial peak temperature and pressure environment and the subsequent long-term environment, including containment spray operation, as described in Section 14.20. Therefore, no equipment or component should be adversely affected by the containment spray operating during the post-accident period.

The containment spray pumps are initiated by SIAS (Section 7.3.2.2). To prevent an inadvertent actuation of containment spray in the case of an undesired trip of SIAS, the containment spray valves are opened only by a containment spray actuation signal (CSAS). In the event that a SIAS has been received without a CSAS, the containment spray pump will be activated with its flow directed back to the RWT via the mini-flow recirculation lines. Since the mini-flow isolation valves close on a RAS, it is necessary to secure the containment spray pumps prior to the RAS to avoid dead-heading the pumps.

However, inadvertent initiation of the spray system will not affect the safety of the plant, since all the instruments are drip proof or weatherproof and all motors are drip proof. All piping or equipment insulation which may come in contact with sprays is metallic jacketed, fiberglass cloth covered, or of the metal reflective type, in order to prevent large quantities of cold water from penetrating the insulation and coming in contact with hot piping; however a small amount of seepage or absorption will not present significant thermal shocking to any hot equipment. Additionally, for the sections of piping that are fiberglass cloth covered (i.e., not covered with metallic jacketing), the lines have been evaluated for the effects of water absorption on the piping stresses and support loads. These were determined to be acceptable for both static and seismic loads.

Inadvertent operation of the system is alarmed when the spray pumps are operated. Flow indication and valve position indication are also provided for the operator, so the situation would be quickly observed and remedial action taken.

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment.

The single failure characteristics of the Containment Spray System are given in Table 6-7.

6.4.5 TESTING

The spray pumps and heat exchangers are located outside the containment to permit access for periodic testing and maintenance during normal plant operation.

A recirculation line is provided on the discharge of each spray pump for testing, which can be accomplished by recirculating water back to the RWT. The recirculation line is sized to pass the minimum allowable pump flow of 50 gpm.

Figure 6-5 shows the containment spray pump characteristic curve. Also shown in this figure are the required NPSH, the pump efficiency and the brake horsepower curves. These curves are drawn from the manufacturer's performance test data. The head capacity of the pump at the design flow of 1400 gpm (1350 gpm spray + 50 gpm minimum recirculation) is greater than the design required head of 370' of water.

Each spray pump has been shop-tested at sufficient head capacity points to generate complete performance curves. NPSH requirements for the capacity range were verified by a suction pressure suppression test for each pump. A shop thermal transient test from ambient temperature to 300°F in 10 seconds was performed on one pump to assure the design is suitable for the switchover from the injection to the recirculation mode.

6.4.6 REFERENCES

1. Calculation CA06963, Mass of Sodium Tetraborate Decahydrate (STB) Buffer Required for Post LOCA Containment Building Sump pH Control
2. Generic Letter 2004-02, dated September 13, 2004, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors
3. SRP 6.5.2, Revision 2, December 1988, Containment Spray as a Fission Product Cleanup System
4. NUREG/CR-5966, June 1993, A Simplified Model of Aerosol Removal by Containment Sprays
5. Regulatory Guide 1.183, July 2000, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

TABLE 6-4

REFUELING WATER TANK DATA

Quantity	1
Total Volume, gal.	420,000
Water volume, gal.	400,000
Design pressure, psig	Atm.
Design temperature, °F	0
Material	Type 304 stainless steel
Seismic requirements	Category I

TABLE 6-6

CONTAINMENT SPRAY SYSTEM COMPONENT DESCRIPTION

CONTAINMENT SPRAY PUMPS

Quantity - 2

Type - Vertically split, horizontal centrifugal with mechanical seals backed up with an auxiliary gland.

Material - American Society for Testing and Materials (ASTM) A296, Gr CA-15[c][d]

Codes - Motor: National Electrical Manufacturers Association (NEMA); Pump: Standards of the Hydraulic Institute

	Mode of Operation	
	Injection (Pre-RAS)	Recirculation (Post-RAS)
Design Specification Flow[b] (each)	1,400 gpm[a]	1,350 gpm
Head	385'	390'
NPSH available	86'	23.2'
NPSH required	19.0'	18.5'
Transient temp.	40 - 300°F in 10 seconds	
Motor	200 hp, 3 phase, 60 Hz, 4000 Volts (A 300 hp motor may also be installed)	

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[a] Includes 50 gpm for minimum flow recirculation.

[b] The design specification flow rates are based on the values listed in Specification 6750-M-62, the actual flow may be different from this value. The NPSH required and Head are obtained from initial acceptance test data at the specified flow.

[c] Acceptable alternate material for pump cover is A-743, Gr CA6NM (ES200600162-000).

[d] Containment spray pump's casing and impellers were supplied per ASTM A351, Gr. CA-15.

PIPING, FITTINGS AND VALVES

A. Suction material - Type 304 stainless steel

Pipe Sizes	Wall Thickness
2" and smaller	Sch. 40S
2 1/2 through 12"	Sch. 10S
14" through 20"	0.250" nominal wall
24"	0.375" nominal wall

Design Pressure - 60 psig from Containment Sump/RWT to source isolation MOVs
- 200 psig from source isolation MOVs to pump suction

Design Temperature - 300°F

Construction	2 1/2" and larger	- Butt-welded except at flanged equipment
	2" and smaller	- Socket-welded except at screwed equipment
Valves	2 1/2" and larger	- Stainless steel, butt-welded, 150 lb.
	2" and smaller	- Stainless steel, socket-welded, 150 lb

Testing - As required by ANSI B31.7

Code - ANSI B31.7, Class II

TABLE 6-6

CONTAINMENT SPRAY SYSTEM COMPONENT DESCRIPTION

B. Discharge material (upstream of containment isolation valves) - Type 304 stainless steel

Pipe Sizes	Wall Thickness
8" and smaller	Sch. 40S
10" through 14"	0.250" nominal wall

Design Pressure - 500 psig
Design Temperature - 350°F

Valves	2 1/2" and larger	-	Stainless steel, butt-welded, 300 lbs.
	2" and smaller	-	Stainless steel, socket-welded, 600 lbs.
	Relief valve setpoint	-	500 psig (on tie to HPSI pump suction)
Testing		-	As required by ANSI B31.7
Code		-	ANSI B31.7, Class II
Spray Nozzles	Type	-	Hollow cone, centrifugal nozzle with vanes
	Material	-	Stainless Steel
	Number	-	90 nozzles per spray header (except the inner ring of Unit 1 which has 89)
	Pressure drop	-	40 psi at a nozzle flow of 15.2 gpm
	Spray droplet size	-	700 microns (mean)

TABLE 6-7

SINGLE FAILURE CHARACTERISTICS OF CONTAINMENT SPRAY SYSTEM

	COMPONENT	MALFUNCTION	COMMENTS AND CONSEQUENCES
1.	Check Valve in Spray Header Line	Sticks closed	Second header will supply 50% flow and the system will be supplemented by four containment air cooling units.
2.	Air-Operated Valve in Spray Header Line	Fails to open	Second header will supply 50% flow and the system will be supplemented by four containment air cooling units.
3.	Containment Spray Pump	Pump fails to start	Second spray pump will supply 50% flow and the system will be supplemented by four containment air cooling units.

6.5 CONTAINMENT AIR RECIRCULATION AND COOLING SYSTEM

6.5.1 DESIGN BASIS

The function of the Containment Air Recirculation and Cooling System is to remove heat from the containment atmosphere during normal plant operation. In the event of the occurrence of a LOCA, the system functions to limit the containment pressure rise to a level below the design value. In such an instance, the system also functions to reduce the leakage of airborne and gaseous radioactivity by providing a means of cooling the containment atmosphere.

The containment air recirculation and cooling system is independent of the SI and Containment Spray Systems. It is sized such that, following a LOCA, three of the four cooling units will limit the containment pressure to less than the containment design pressure even if the Containment Spray System does not operate.

The original sizing of the containment air coolers for design and procurement was based on the heat removal capability, using three coolers, (240×10^6 Btu/hr) required to maintain the post-LOCA containment atmospheric pressure within the containment design pressure. Likewise, the original sizing of the containment spray system for design and procurement was based on the heat removal capability (240×10^6 Btu/hr) required to maintain the LOCA containment atmospheric pressure within the containment design pressure. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Chapter 14.

All system components are designed to withstand Seismic Category I loadings.

6.5.2 SYSTEM DESCRIPTION

The containment air recirculation and cooling system (Figure 9-20A) includes four two speed cooling units located entirely within the containment. Service water is circulated through the air cooling coils.

Each coil house is equipped with 12 individual coils piped to supply and return manifolds which connect to the SRW System.

The SRW supply line for each cooler has an air-operated valve which is normally open and de-energized. Each of these valves is designed such that it would fail in the open position. A redundant line to the coolers is normally valved off by a local, manually-operated valve.

The SRW return line for each cooler has two air-operated stop valves and one local, manually-operated valve, all in parallel. The air-operated control valves can be operated from the Control Room. One control valve is used for normal cooling requirements and the other control valve opens automatically upon receipt of SIAS. The third, parallel, local, manually-operated valve is provided to permit passage of sufficient SRW in case the full flow SIAS-actuated valve should malfunction.

Air is drawn through the coils by a vane-axial fan driven by a direct coupled two-speed motor. Normal containment recirculation requirements are satisfied at high speed operation, whereas, after a LOCA, the low speed setting is used. All fan motors may be manually started or stopped from the Control Room. Upon occurrence of LOCA, all four fan motors start automatically upon receipt of SIAS.

Upon evacuation of the Control Room due to fire, control of all fan motors may be transferred to local control stations in the electrical penetration rooms. Upon selection of local operation, the ESF signals (SIAS and under-voltage) are overridden.

Performance data for the cooling units is given in Section 6.5.3. The materials of construction are listed in Table 6-8. The equipment is designed to withstand Seismic Category I accelerations and to operate in the LOCA environment.

Service water flow is shown in Figures 9-9 (Unit 1) and 9-27 (Unit 2).

6.5.3 SYSTEM OPERATION

a. Normal Operation

The number of operating coolers is temperature dependant. Three cooling units are normally in operation during the warmer months and two cooling units are normally in operation during the cooler months. Each unit is sized to remove in excess of one-third of the total normal cooling load. The maximum average temperature inside the Containment is limited to 120°F by operation of the three cooling units. The maximum expected SRW inlet temperature to the coolers is 95°F. During normal operation, the full-size SRW outlet valves, which are used following a LOCA, may be closed, while the smaller (4" diameter) valves are open. Occasionally, during extended periods of high outside temperature, all four coolers are used to limit the average containment temperature to 120°F. Service water flow to the containment air coolers may be supplemented by using the 8" full size SRW outlet valves.

Performance Data for Normal Operation

Total heat removal capacity	2.27×10^6 Btu/hr[a]
Motor horsepower	125 hp high speed (normal)
Air flow, each	110,000 cfm[a]
Fan horsepower, each	100 bhp
Fan speed	1,200 rpm
Cooling water flow, each	550 gpm[a]
Air temperature, inlet/outlet	120/99°F[a]
Water temperature, inlet/outlet	95/102.8°F[a]
Fan static pressure	3.2" H_20
Fan total pressure	4.2" H_20

[a] Cooler heat removal capacity is a function of SRW flow and temperature, fouling, air flow, and containment temperature and pressure.

b. Plant shutdown operation

During plant shutdown, i.e., Modes 4, 5, 6 and defueled, the cooling units operate as necessary, based on availability and containment conditions. The availability of the cooling units is directed by administrative controls.

c. Emergency Operation

Upon receipt of a SIAS, any idle cooling unit is automatically started on the low speed setting and, simultaneously, any running fan is switched from their normally operating high speed setting to low speed operation. The full flow (8" diameter) SRW outlet valves for each cooler are opened upon receipt of a SIAS. The SRW inlet valves move to a throttled position upon receipt of a SIAS, and return to the full open position upon receipt of a RAS.

With off-site power available under this mode of operation, the operating cooling unit fans are switched to low speed and the idle fan(s) are started on low speed as described above. If off-site power is not available, the associated emergency

diesel generators are started. Each emergency diesel generator supplies power to an independent safety-related bus. Each bus carries the load of two cooling units.

The evaluation of post-incident containment pressure/temperature response is provided in Section 14.20. These evaluations consider the actual heat removal capacity of the containment air coolers which is a function of SRW flow and temperature, fouling, air flow, and containment pressure and temperature.

With respect to long-term cooling after a LOCA, the cooling units are designed to operate for at least one year under air-steam mixture conditions of 5 psig and 160°F.

Performance Data for Emergency Post-LOCA Conditions

Motor horsepower	63 hp (low speed)
Fan horsepower (max.), each	33 bhp
Fan speed	600 rpm
Mixture temperature, inlet/outlet	275/270°F[(a)]
Water temperature, inlet/outlet	105/204°F[(a)]
Cooler, capacity at 273°F and 47 psig, each	90.45×10^6 Btu/hr[(a)]
Cooler mixture flow, each	55,000 cfm[(a)]
Maximum fin side pressure drop	0.5" H_20
Maximum tube side pressure drop	15.6 psi
Fan static pressure	1.2" H_20
Fan total pressure	1.445 in H_20
Water flow, each	1,900 gpm[(a)]

[(a)] Cooler heat duty will vary with flow, temperature, and humidity.

6.5.4 DESIGN EVALUATION

a. The coil capacity is based upon 95°F SRW inlet during normal operations and 105°F SRW inlet during LOCA. These values represent the maximum expected temperatures. The water velocity through the coils is 7.4 fps at 1,900 gpm flow.

b. Total effective face area in each cooler is 144 ft^2. With a normal operating air flow of 110,000 cfm, the velocity entering the coils is 765 fpm. With the emergency mixture flow of 55,000 cfm, the entering velocity is 382 fpm.

c. The fin spacing is 6 fins per inch of coiled length. With this pitch, water clogging of the coil fins is avoided.

d. A fouling factor of 0.0005 for the water side is included in the coil ratings. The water side of the cooling coil tubes is equipped with removable plugs on the return bends of the coils to permit cleaning in the field.

e. The cooler housing is designed to ensure no loss of function when subjected to a pressure differential of 2 psi.

f. Components are designed to be compatible for operation in an environment of borated water spray. The three- to four-hour short-time temperature exposure rating during a LOCA is about 280°F. The three- to four-hour short time humidity exposure rating is 100% relative humidity in a slightly acid atmosphere. Additionally, components which are considered susceptible to radiation damage, such as the gasket and motor, are designed to withstand a dose of 10^8 rad of gamma radiation. It has been calculated that these components will receive a dose of less than 5×10^7 rad during the year following a LOCA.

g. With respect to normal containment recirculation and cooling, the cooler assemblies are designed for a life of 40 years.

h. The condensate leaving the coils is conveyed over individual stainless steel drip pans to the sides of the coils out of the mixture flow stream. These pans cascade the liquid into the main sump of the housing from which it is drained via the Containment sump to one of the Auxiliary Building sumps, from which it is pumped to the Waste Processing System.

6.5.5 AVAILABILITY AND RELIABILITY

a. The cooling units are located outside the secondary shield. In this location they are protected from being flooded at post-accident conditions and they also are protected against credible missiles.

b. The original design heat removal capability of three of the four cooling units was to provide the same heat removal capability as the containment spray system. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20. The single failure characteristics for the cooling units are listed in Table 6-9.

c. Each fan discharge duct is provided with a fusible link door. These doors open at an abnormally high containment temperature such as would occur under a LOCA. This assures the free flow of the cooled air-stream mixture to the containment environment even if the ducts collapse during or following the LOCA.

d. The cooling units are designed to operate for the life of the plant. There are no belts or flexible couplings; the motor is directly connected to the fan wheel.

e. Upon loss of off-site power during a LOCA, the containment cooling fans are automatically sequenced onto the emergency diesel generator buses.

f. All associated system equipment, such as piping, valves, and instrumentation are also located outside of the secondary shielding to minimize the possibility of missile damage.

6.5.6 TESTS AND INSPECTIONS

a. The manufacturer has developed a computer program to size cooling coil units for saturated steam-air mixtures. This program was used to size the Calvert Cliffs cooling units. Tests performed on the coils manufactured for Palisades, Fort Calhoun, Three Mile Island No. 1, Kewaunee, and Oconee have confirmed the validity of the program.

   These tests were conducted with coils of material and configuration identical, except for shorter length, to those used in each specified large-scale containment system. Three of the coil tests were made with the air flow horizontal and the condensate drainage perpendicular to the air flow. Water-logging problems did not arise. The coil section drainage characteristics were identical to those which were predicted for the full-size units, and therefore provide assurance that the full-size coils will adequately drain condensate from the coil surfaces.

   The coil was tested at a pressure loading equal to a free velocity pressure of 500 fps to demonstrate the structural integrity of the coil. This loading test was performed to simulate a pressure wave which may occur during the initial phase of containment pressure buildup in the event of a LOCA. Upon examination, the coil showed only very minor deformation of some intermediate stiffeners; hence, the structural design of the coil was proven to be adequate.

b. A fan and motor have been tested to prove their ability to operate satisfactorily under conditions existing within the containment after a LOCA.

   Fans and motors also were tested by the fan manufacturer to assure the same characteristic performance curve for all fans.

c. Cooling unit performance can be tested with thermometers, manometers and a Pitot tube in the field at any time the containment is accessible.

d. The valves in the normal cooling water outlet lines (4") will be open during normal operation and the valves in the parallel emergency outlet lines (8") can be opened from the Control Room and the flow rate can be monitored at any time.

e. All equipment and associated components are arranged so that they can be inspected at any time the containment is accessible.

f. The containment air cooler blowdown door fusible links will be replaced every refueling outage to ensure that the links perform their design function.

TABLE 6-8

COOLING UNIT MATERIALS OF CONSTRUCTION

Tubes (seamless)	90/10 copper-nickel, ASTM B111-69, Alloy 706
Fins	ASTM B152
Headers	ASTM B466
Coil Frame	ASTM A525
Structure	ASTM A501, A36, and M1020
Motor	NEMA Class B, TEAO

TABLE 6-9

SINGLE FAILURE CHARACTERISTICS FOR COOLING UNITS

	COMPONENT	MALFUNCTION	COMMENTS AND CONSEQUENCES
1.	Unit Circulating Fan	Fails to operate	Three (or four) coolers and fans are normally operating. Fans may be tested for emergency mode of operation at any time.
2.	Cooler	Failure to tubes	Tube failure is considered unlikely during emergency operation since the tube water to air side ΔP is less than during normal service. If failure does occur, SRW will be spilled into the cooler since SRW pressure is above containment pressure. Tube leakage can be detected by indication of increased SRW flow to the cooler, decreased SRW head tank level, and the failed cooler can be isolated. Note: Passive failures are only considered post-RAS. In the event of tube failure associated with any one cooler after the LOCA, it is assumed, as an upper limit, that one subsystem of Service Water leaks into containment. The leak volume from one subsystem is approximately 16,000 gallons. Boron dilution, therefore, would be negligible, because the total volume of borated water in the containment structure is in excess of 400,000 gallons.
3.	SRW Emergency Outlet Valve	Fails to open	For those coolers in operation, the valve in the normal cooling water outlet line will be open. The normal operation valve will be open. If the emergency valve fails to open, the unit will operate at reduced heat removal capability. The Containment Spray System will supplement the heat removal capability of the cooling units.
4.	SRW Inlet Valve	Inadvertently left throttled	Valve status will be apparent from reduced flow, and the valve may be opened by operator action. If the valve fails to respond, the Containment Spray System will supplement the cooling requirements. Each cooler is provided with flow indication on the main control board.
		Fails to throttle	Does not adversely affect containment heat removal. May result in emergency diesel generator service water inlet temperatures in excess of 105°F under certain conditions with elevated bay water temperatures. Containment spray system remains available to supplement heat removal capability of the cooling units.

TABLE 6-9

SINGLE FAILURE CHARACTERISTICS FOR COOLING UNITS

COMPONENT	MALFUNCTION	COMMENTS AND CONSEQUENCES
	Inadvertently closed	Valve status will be apparent from lack of flow and position indication, and the valve may be opened by operator action. If the valve fails to respond, the Containment Spray System will supplement the cooling requirements. Each cooler is provided with flow indication in the Control Room. Note: Mechanical stops are installed on valves to limit stroke. The stops should prevent inadvertent full closure.

6.6 CONTAINMENT PENETRATION ROOM VENTILATION SYSTEM

6.6.1 DESIGN BASIS

The Containment Penetration Room Ventilation System is designed to collect and process containment penetration leakage, so as to reduce to a minimum the environmental radioactivity levels from post-accident containment leaks. See Section 14.24.3 for a discussion of the assumed operation of the system post-LOCA.

6.6.2 SYSTEM DESCRIPTION

The Containment Penetration Room Ventilation System is shown schematically in Figure 9-20A. Since experience has shown that containment leakage is more likely at penetrations such as electrical cables and air purging valves, rather than through the liner plates or weld joints (Reference 1), penetration rooms are built adjacent to the outside surface of each containment and enclose the areas around the majority of the penetrations. The only penetrations which do not pass through these areas are:

a. Two main steam lines;
b. Two main feedwater lines;
c. Equipment hatch;
d. Normal personnel access lock;
e. Emergency personnel access lock; and,
f. Refueling tube.

The main steam and feedwater lines are welded to the liner plate and, therefore, are not considered as a source of leakage. The equipment hatch and access lock openings can be tested during normal operation and are not considered sources of significant leakage. There are double seals at each of these three access openings. The refueling tube is valved on one end and blind flanged on the other end.

The principal function of this system is to control and minimize the release of radioactive materials from the containment to the environment during a post-accident period. Following a LOCA, a Containment Isolation Signal (CIS) will start both of the two full-size blowers. The penetration room exhaust ventilation system design basis is the Maximum Hypothetical Accident. The system is credited in the dose calculations with filtering the radioactive material released through the 4" containment vent line at the onset of a Maximum Hypothetical Accident as well as leakage through the penetrations. A gravity damper, which opens when the blower starts, is provided at the discharge of each blower to prevent recirculation through a failed or idle unit. The entire system is designed to operate under negative pressure up to the fan discharge.

To minimize the release of radioactive material to the environment, penetration room ventilation is continuously routed through a prefilter, an absolute high efficiency particulate air (HEPA) filter, and an activated charcoal filter, positioned in series. The use of these filters post-accident is described in Section 14.24.3.

In all cases, the flow rate from the penetration room will exceed the total maximum containment leakage rate. The containment purge equipment, if running, will be shut down by a containment radiation signal (CRS), and the valve in each purge line penetration will be closed. Refer to Section 14.24.3 for a discussion of relevant accident analysis assumptions.

6.6.3 SYSTEM COMPONENTS

The Containment Penetration Room Ventilation System is provided with two blowers and two filter assemblies. Both blowers, each of which is aligned with a filter assembly,

discharge through a single line to the unit vent. (Table 6-10) Power-operated dampers are provided for isolating each filter assembly from the penetration rooms. The filter assembly consists of a prefilter, a HEPA filter and an activated charcoal filter in series. The prefilter removes coarse airborne material and water droplets using pads of glass fiber, placed between perforated metal grids, as the filtering media.

The HEPA filter, which removes small airborne particles that pass through the prefilter, consists of two cells of fiber glass media mounted in a metal frame.

The activated charcoal filter removes methyl as well as elemental iodine contaminants resulting from a LOCA. It consists of six cells of activated charcoal having approximately 5 wt% impregnation of iodine compounds, and an ignition temperature of ≥ 680°F, held in place by stainless steel channel clamps and galvanized bolts.

As a means of checking the condition of the charcoal in each filter bank, one or more charcoal test trays, filled from the same principal batch of charcoal as the other trays, may be installed in lieu of regular trays in each filter bank. Since the test tray is a substitute for a regular tray, it experiences air flows at the same rate and angle as the other trays. This ensures that the samples taken from the test trays are representative of the charcoal in the entire bank. Trays of this type may be installed in any charcoal filter bank that is part of the iodine removal system in this power plant. For operator information, temperature and pressure monitoring is provided for all penetration rooms and an area radiation monitor is provided for the West Penetration Room. Differential pressure indicators are provided across the filters.

6.6.4 SYSTEM OPERATION

During normal operation, the system is held on standby with both blowers aligned with their respective filter assemblies. A CIS will start both of the blowers. The containment purge equipment, if running, will be shut down by a CRS, and the valve in each purge line will be closed.

All of the system components can be controlled from the Control Room.

6.6.5 DESIGN EVALUATION

The blower capacity of 2000 ± 200 cfm exceeds, in all cases, the total maximum Containment Building leakage rate. The blowers and the respective filters are aligned in a redundant manner to assure operation of one blower and its respective filter assembly, independent of a failure or malfunction of any of the active system components.

When the system is in operation, a negative pressure may be maintained in the penetration rooms and in the ducting up to the discharge of the blower. All components are designed to Seismic Category I requirements.

6.6.6 AVAILABILITY AND RELIABILITY

Redundancy of components, monitoring operation of the system from the Control Room and provision of proper instrumentation, assure proper response of the system when a LOCA does occur. Upon failure of the normal electrical power supply to the blowers, power is supplied from the emergency power source.

The system components and equipment are fully accessible during normal plant operation.

A single failure analysis for the main components of the system is given in Table 6-11.

6.6.7 TESTS AND INSPECTIONS

All equipment and associated components are arranged so that they can be inspected at any time the containment is accessible.

Testing of charcoal/HEPA filter units is established based on Technical Specification requirements. The Technical Specification specifies testing conditions based on the application of the particular filter unit.

6.6.8 REFERENCE

1. W.B. Cottrell and A.W. Savolainen, Editors, U. S. Reactor Containment Technology, ORNL-NSIC-5, Volume II, August 1965

TABLE 6-10

PENETRATION ROOM BLOWER DESCRIPTION

Type	Centrifugal
Quantity	2 in each unit
Capacity (cfm)	2000 at 8 1/2" w.g. (inch H_20)
Motor	5 hp, 460 Volt, 3 phase, 60 cycle
Codes	NEMA

TABLE 6-11

SINGLE FAILURE CHARACTERISTICS FOR CONTAINMENT PENETRATION ROOM VENTILATION SYSTEM

	COMPONENT	MALFUNCTION	COMMENTS AND CONSEQUENCES
1.	Blower	Fails to start	Spare blower is already operating.
2.	Blower	Fails during service	Alarm in Control Room will indicate loss of negative pressure, and spare blower is already in service.
3.	Blower valve	Fails to open	Spare blower is already operating.
4.	Filter valve	Fails to open	Failure not considered credible since one filter will always be lined up to operate when needed.
5.	Ductwork	Leakage	Leakage of unfiltered air may be inward since ductwork may be maintained at negative pressure.

6.7 <u>CONTAINMENT IODINE REMOVAL SYSTEM</u>

6.7.1 DESIGN BASIS

The containment iodine removal system is designed to collect, within the containment, the iodine released following a LOCA.

6.7.2 SYSTEM DESCRIPTION

Following a LOCA, SIAS automatically starts three 20,000 cfm recirculation filter units. Each unit has the capacity of 50% of the design air flow. These units consist of activated charcoal filters preceded by HEPA filters. A moisture separator is provided upstream of the particulate air filters to remove water droplets. An electric-driven induced-draft fan located at the end of the banks of filters pulls the containment atmosphere through these components and discharges vertically back into the containment (Figure 6-6).

The three containment charcoal filter units contain a total of approximately 7300 lbs. of Barnebey-Cheney #727 coconut shell charcoal (or equivalent) impregnated with 5 wt% iodine compounds. The flow velocity through each bed is 40 fpm and the corresponding residence time is .25 seconds. Filter testing is explained in Sections 6.6.7 and 6.7.7. Testing is performed to demonstrate that the installed charcoal adsorbers will perform satisfactorily in removing both elemental and organic iodides for design conditions or flow, temperature, and relative humidity.

Each of the recirculation filter units is provided with an emergency dousing system for the charcoal beds to dissipate the decay heat load in the event there is a significant rise in the charcoal bed temperature. During Modes 1, 2, 3, and 4, the dousing system is isolated by manual valves. An analysis (Reference 1) shows that maximum post-LOCA charcoal bed temperature will not cause iodine desorption or charcoal bed ignition. During Modes 5 and 6, the manual valves may be opened to allow the dousing system to be functional during iodine removal unit maintenance to provide fire protection if required.

This system is shown in diagram form on Figures 6-7 (Unit 1) and 6-11 (Unit 2).

6.7.3 SYSTEM COMPONENTS

a. Unit Housing

The unit housing is made of carbon steel and is capable of operating under the LOCA conditions of pressure and temperature. It is provided with service access doors, explosion-proof lights, a charcoal filter emergency dousing system, a monorail suitable for supporting an electric hoist to handle charcoal cells, test connections, connections for pressure gauges and floor drains. The housing is designed to be light-tight.

b. Moisture Separators

The moisture separators consist of a steel casing, two-and-one-half-pass vertical louvers, and filter element frames. The bottom of the unit is a sump equipped with drain connections. Elements are constructed of stainless steel wire mesh. In addition to acting as moisture separators, the filter elements also act as a prefilter for the HEPA filters and remove large particles and any fibrous material present in the air steam.

c. High Efficiency Particulate Air (HEPA) Filter

Each element of the HEPA filter measures 24"x24"x11 1/2" and is constructed by pleating a continuous sheet of waterproof fire retardant fiberglass mat into closely spaced pleats separated by aluminum inserts. The filter medium is treated with a

silicone-base water repellent to achieve wet strength characteristics. The filter medium and separators are encased in cadmium-plated carbon steel. The filter bank frame is galvanized steel.

d. Charcoal Filters

The material used in these filters is activated charcoal containing 5 wt% impregnation of iodine compounds and having an ignition temperature of ≥ 680°F. It is encased in perforated stainless steel beds. Each element is a standard manufactured size, has a frontal face size of approximately 8"x24", and contains two horizontal charcoal beds, each approximately 24"x24"x2" deep. The filter bank frame is galvanized steel.

As shown on Figures 6-7 (Unit 1) and 6-11 (Unit 2), each charcoal filter bank is equipped with an emergency cooling water dousing system. Provisions are made to collect the cooling water after it flows through the charcoal beds and onto the floor of the filter unit, and to drain it through a check valve into the Containment Structure. Each filter bank may contain a test tray which will be used to verify the efficiency of the charcoal bed periodically (Section 6.6.3).

e. Fan-Motor Unit

The fan is an internally-direct-driven, vane-axial type, equipped with an inlet and discharge adapter. The motor is a totally enclosed, air-over type and is provided with an insulation system capable of operating under LOCA conditions.

6.7.4 SYSTEM OPERATION

The containment iodine removal filter units are not in operation during normal reactor operation. However, following a LOCA, receipt of SIAS will automatically start all three filter unit fans. These units may also be started manually by the operator from the Control Room at any time.

During Modes 5 and 6, the charcoal bed emergency dousing system in each unit may be initiated manually, as needed.

6.7.5 DESIGN EVALUATION

a. The system consists of three recirculating filter units with a total capacity of 150% of design flow requirements. Each unit is protected from all expected missile sources by concrete structures.

b. The design life of the filter units is 40 years, the same as the design life of the plant. The units are not in operation during normal reactor operation.

c. The filter units are designed to operate under the maximum temperature and pressure conditions resulting from a LOCA (original calculations show an atmosphere composed of steam-air mixture at a pressure of 47.45 psig and a temperature of 274°F). The filter units have been evaluated for the revised maximum pressure and vapor temperature contained in Section 14.20. In addition to this, the units are designed to withstand conditions of ambient temperature and 57.5 psig for 24 hours during testing of the containment.

d. The units are designed to assure no loss of function when subjected to a pressure differential of 2 psi.

e. The filter units are classified as Seismic Category I equipment and are designed accordingly.

f. Filter unit components are designed to withstand a radiation level of 1 r/hr of principally gamma radiation with a 40 year cumulative dosage of $3.5x10^5$ r under normal environmental conditions.

g. The components are also designed to be capable of operation in an atmosphere of borated water spray and a maximum cumulative radiation dose of 10^8 r, occurring as a result of a LOCA.

h. Each element of the moisture separators is designed to have a nominal flow of 1500 cfm and a clean pressure drop of less than 1" of water.

i. Each element of the HEPA filters is designed to have a nominal flow rate of 1000 cfm and a clean pressure drop of less than 1" of water.

j. Each element of the charcoal filter units is designed for a nominal flow rate of 333.3 cfm and a clean pressure drop of less than 1" of water.

6.7.6 AVAILABILITY AND RELIABILITY

The Containment Iodine Removal System incorporates three filter units, each with a capability to handle 50% of the required air flow. The units are designed to operate without loss of function under the maximum temperature and pressure conditions resulting from a LOCA, as well as in the accompanying containment atmosphere environment, i.e., a steam-air mixture with borated water spray and radiation.

Because the system is not in use during normal reactor operation, special routine tests and inspections are incorporated into plant operating procedures. These periodic tests will be performed during plant operation to assure the availability of power to each of the electrical components (both visual and audible indicators are provided in the control panel for this test). During plant shutdown the condition of each filter bank will be determined by visual inspection and pressure differential gauge indication while the fans are in operation. In addition, the water line solenoid valves will be visually checked for proper operation during periods of plant shutdown.

Failure of the normal electrical power supply will automatically place the three fans on emergency electric power.

6.7.7 TESTS AND INSPECTIONS

See Section 6.6.7 for a current description of the tests and inspections performed for both systems.

6.7.8 REFERENCES

1. Nuclear Consulting Services, Inc. Report NUCON-6BG021/01, A Computer Analysis of the Iodine Decay Heat Generated in a Carbon Bed Following a Loss of Coolant Accident, January 19, 1990, and Supplement 1 July 25, 1990

6.8 HYDROGEN CONTROL SYSTEMS

6.8.1 GENERAL

The Calvert Cliffs containment has been designed to promote circulation of the contained air. Natural circulation is enhanced by large vent areas between compartments and by the use of gratings between elevations. A thorough review has been conducted to ascertain that no areas exist within which gases could be trapped. As a result of this review, design features such as the venting of the pressurizer compartment have been incorporated. Mechanical mixing of the air is achieved using the containment air recirculating and cooling system and the iodine removal system. Under LOCA conditions, hydrogen and radioisotopes will be rapidly distributed throughout the containment due to turbulent currents introduced by the escaping steam and water and natural convection due to the temperature differential between the sump water and the containment atmosphere. The air recirculation and Containment Spray Systems also promote mixing and minimize nonuniformities in the containment atmosphere.

On March 2, 2004, the Nuclear Regulatory Commission approved a license amendment which changed the definition of DBEs to exclude hydrogen generation in Containment as a consequence of the event. The amendment was based, in part, on the size of the Containment and free circulation of air within the Containment. Hydrogen recombiners are no longer needed and were removed from the licensing basis. However, hydrogen analyzers are required to be retained as non-safety-related equipment to evaluate events beyond the design basis.

6.8.2 DELETED

6.8.3 CONTAINMENT VENT SYSTEM

The Containment Vent System (Figure 9-20A) is used during power operations to vent the Containment to maintain containment pressure and airborne radioactivity within Technical Specification limits.

The vented air will be introduced to the penetration room exhaust system and will be passed through the penetration room exhaust system's HEPA and charcoal filters before being discharged to the environs. The system is equipped with a flow monitor and motor-operated valves.

The containment vent may be used to purge hydrogen from Containment if desired.

Upon receipt of a SIAS, CRS, or a high radiation signal, the MOVs close automatically. During post-accident conditions with the CRS or containment high-radiation signal present, the Containment Vent System can be operated by overriding the CRS or containment high-radiation closing signal using key-operated override handswitches on control panels 1C10 and 2C10.

6.9 AUXILIARY FEEDWATER PUMP ROOM EMERGENCY COOLING

6.9.1 DESIGN BASIS

The Auxiliary Feedwater Pump Room cooling system is designed to prevent the room air temperature from rising above 130°F so as to prevent failure of the air-cooled bearings of the pump during an emergency shutdown of the plant.

6.9.2 SYSTEM DESCRIPTION

If the "normal" operating mode of the cooling unit fails for any reason, the “emergency” mode of operation can be deployed. The emergency mode of operation relies on redundant fan coil units (FCUs), located in the mechanical equipment room at Elevation 5'0" of the Auxiliary Building, which can circulate and cool 3,300 cfm (nominal) of air from the AFW Pump Room through a system of connecting ductwork. The Service Water System supplies all of the cooling water required to the FCUs to cool the air. This system is shown in Figure 9-21.

6.9.3 SYSTEM COMPONENTS

a. Fans - The fans are made of carbon steel, direct-drive, SQA design 44 plug fans.

b. Motor and Drive - Each motor is of drip-proof/TEFC (totally enclosed, fan cooled) type with grease lubricated ball bearings.

c. Cooling Coils – The cooling coil tubes are copper-nickel and the fins are aluminum.

6.9.4 SYSTEM OPERATION

These fans are not in service during normal operation. Fans are controlled manually from a local station as well as automatically by temperature switches located in the AFW Pump rooms.

6.9.5 DESIGN EVALUATION

a. The system consists of redundant FCUs and ducting with automatic dampers located on the discharge opening of each fan and isolation dampers on the suction side of each fan.

b. Each FCU is furnished as a unit and meets Seismic Category I requirements.

c. All ducting and equipment is reinforced with Seismic Category I bracing.

6.9.6 AVAILABILITY AND RELIABILITY

All moving parts, such as fans or dampers, are redundant and each has the capacity to handle 100% of the load.

Because the system is not in use during normal plant operation, specific routine tests and inspections will be incorporated into the plant operating procedures. Periodic tests will be performed to ensure the availability of power and the operability of the components.

The FCU are arranged electrically so they can be operated using either off-site power or the emergency diesel generators. Each FCU can be powered from a separate emergency diesel generator.

6.9.7 TESTS AND INSPECTIONS

a. FCU, motors and dampers were tested in-place to ascertain their over-all performance.

b. Air and water flow rates were measured and certified before the system was accepted as being operable.

c. All equipment and associated components are arranged so that they can be inspected at any time.

6.10 ELECTRICAL HEAT TRACING SYSTEM

Electrical heat tracing is installed on all piping, valves, pumps and other line-mounted components that contain concentrated boric acid. Heat tracing is needed to maintain a 12% weight boric acid solution above the 135°F saturation temperature. The heat tracing system is designed to maintain 160°F; however, the operating temperature may be set lower. The thermal insulation is designed to limit the insulation surface temperature to 140°F based on an ambient air temperature of 80°F and component temperature of 160°F.

The electrical heat tracing system is designed such that a single failure will not cause loss of function and includes redundant heater elements, controls, and alarm functions. Each subsystem is supplied by separate emergency power sources.

Each subsystem is equipped with an independent alarm system. Functions which are alarmed locally and in the Control Room include high and low temperature and loss of power.

Power for heat tracing is considered as part of the load for the emergency diesel generator during the first eight hours of operation.

The pre-operational test of the heat tracing system verified that the design basis, as stated above, is adequate. This test program included verification of the following:

a. Proper functioning of all heating elements;
b. Alarm functions for loss of power to heating elements;
c. The temperature of the boric acid solution;
d. Manual transfer to the redundant subsystem from the local control panel; and,
e. All heat tracing controls function properly.

6.11 ECCS LONG-TERM COOLING FLUSH

During post-accident long-term cooling conditions for a large cold leg break, it is possible to have a boric acid reconcentration occur in the core area, due to the small core flow in effect. This condition may result in the crystallization of boric acid in the core and restriction of cooling flow. In order to prevent such an occurrence, two procedures have been developed. The operators will decide which method to use based on plant conditions.

Hot leg injection promotes flow through the core by establishing a flow path from the containment sump, through the LPSI system via the warm-up line, to the shutdown cooling suction line, and into the hot leg.

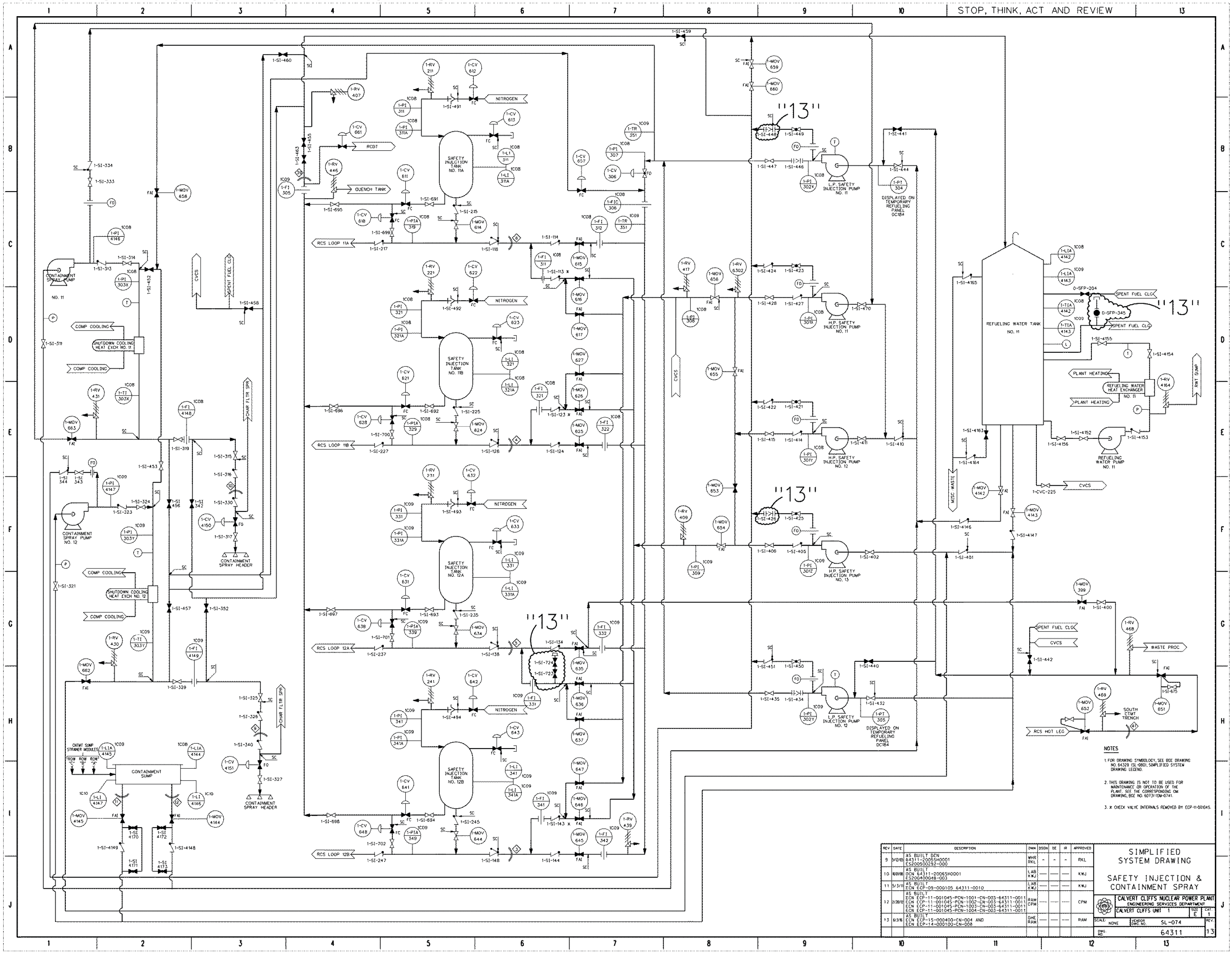
Pressurizer injection promotes flow through the core by diverting flow from one HPSI pump through the pressurizer auxiliary spray line and into the hot leg.

A minimum of 150 gpm injection flow (flow to the reactor coolant system hot side) is necessary to overcome the coolant boil-off rate and cause a net flushing flow downward through the core. The required injection flow must be provided within at least 11 hours after the accident.

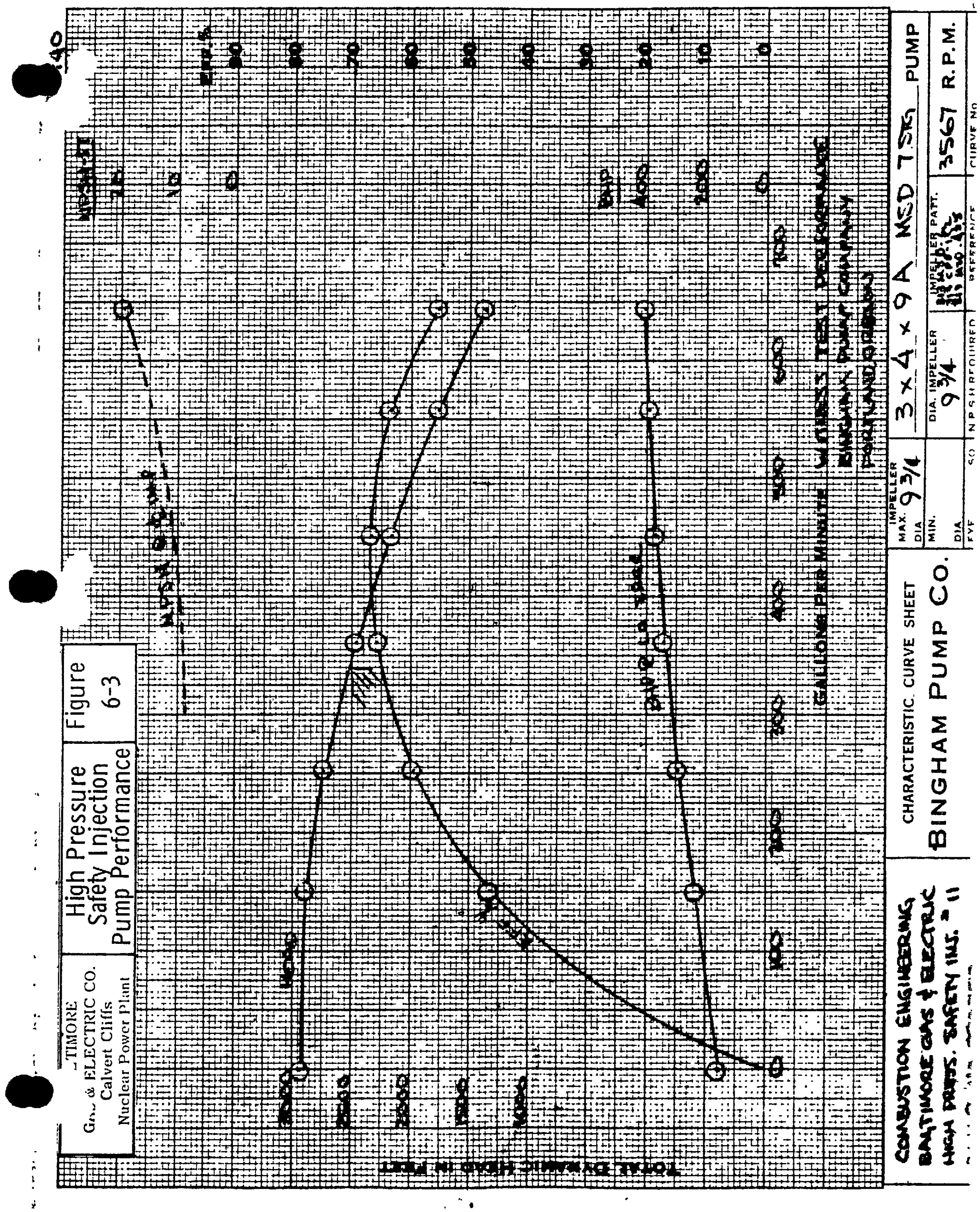
The HPSI pumps are the preferred method of core flush. The LPSI pumps could be used for core flush only as a backup to the HPSI pumps when pressurizer injection is not possible.

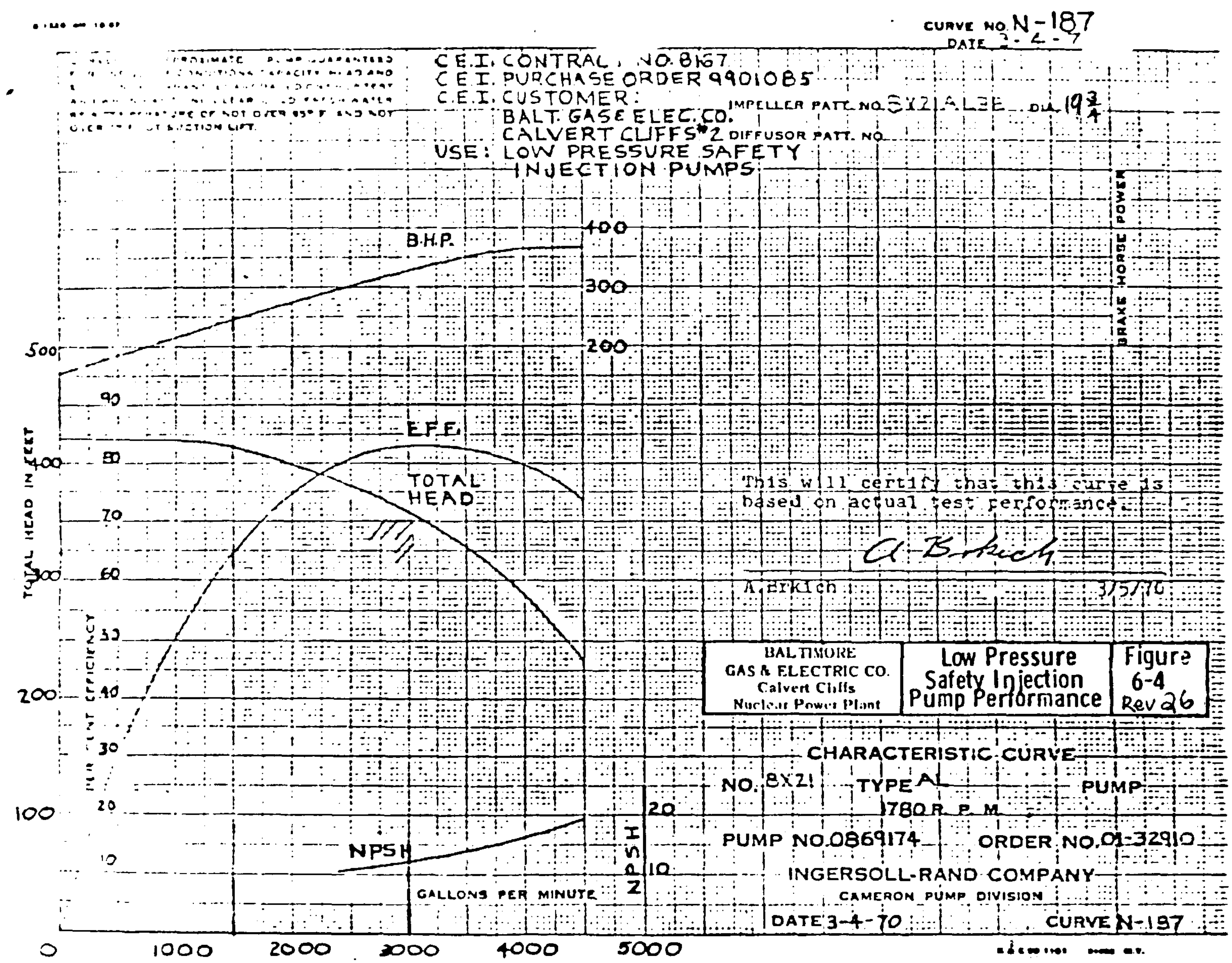
The containment spray pumps can be aligned to provide core flush into the hot leg by realigning valves on the -10 foot level of the Auxiliary Building. This realignment can be made after the containment spray pumps have completed the performance of their required post-LOCA function for spray service to reduce containment pressure. The containment spray pumps would only be used as a backup to the LPSI pumps.

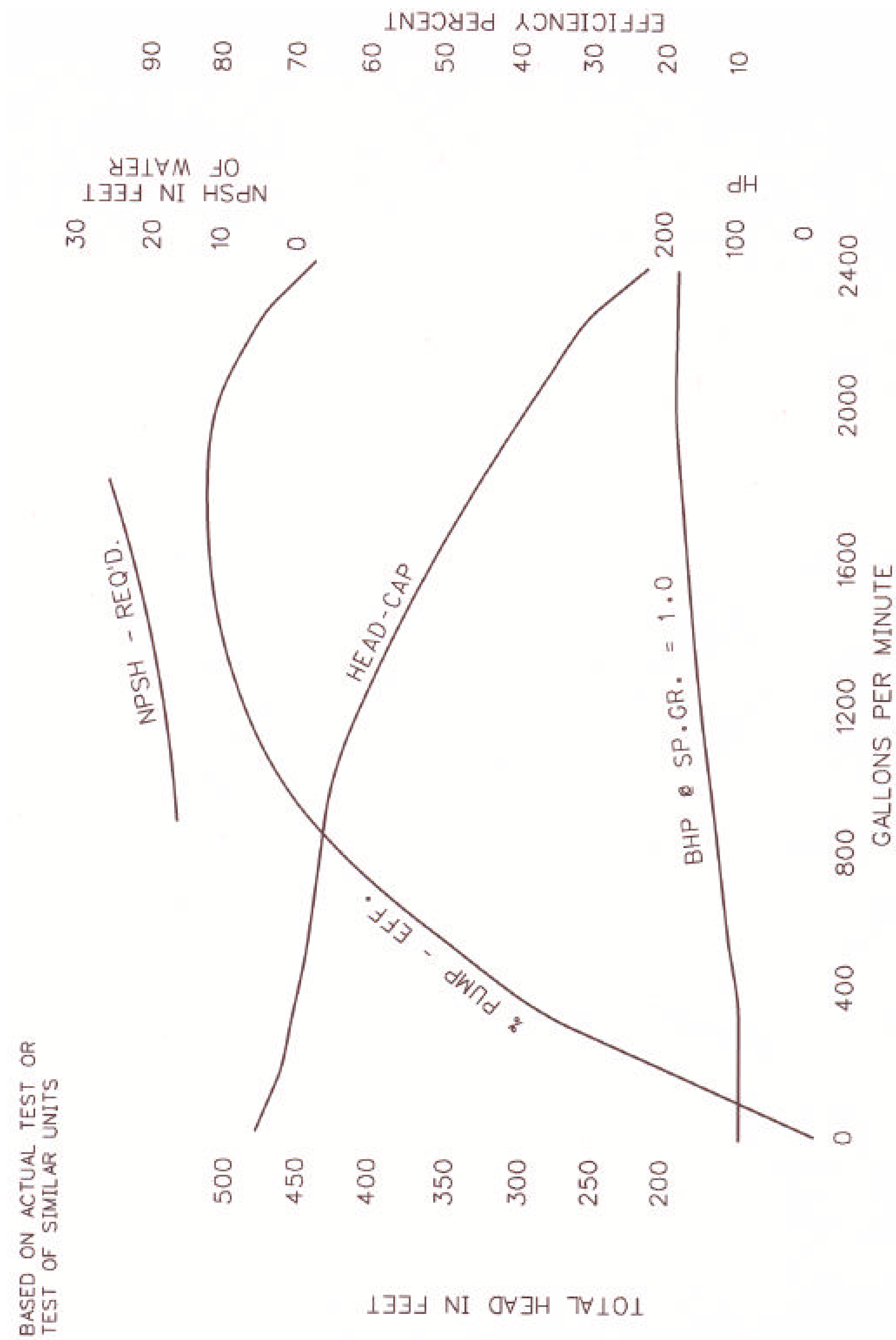
FIGURE 6-1 SAFETY INJECTION AND CONTAINMENT SPRAY – UNIT 1



Revision 49

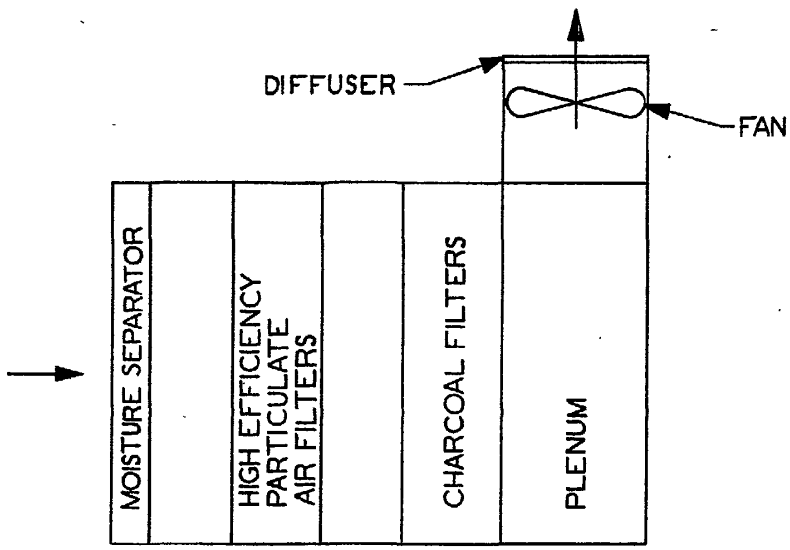

High Pressure Safety Injection Pump Performance
Figure 6-3
Calvert Cliffs Nuclear Power Plant
CHARACTERISTIC CURVE SHEET
BINGHAM PUMP CO.
3 x 4 x 9A MSD 7SG PUMP
3567 R.P.M.
GALLONS PER MINUTE
TOTAL DYNAMIC HEAD IN FEET


CURVE NO. N-187
C.E.I. CONTRACT NO. 8167
C.E.I. PURCHASE ORDER 9901085
C.E.I. CUSTOMER:
BALT. GAS & ELEC. CO.
CALVERT CLIFFS #2
USE: LOW PRESSURE SAFETY INJECTION PUMPS
IMPELLER PATT. NO.
DIFFUSOR PATT. NO.
B.H.P.
EFF.
TOTAL HEAD
NPSH
BRAKE HORSE POWER
TOTAL HEAD IN FEET
PER CENT EFFICIENCY
GALLONS PER MINUTE
400
300
200
500
90
80
70
60
50
40
30
20
10
100
0
1000
2000
3000
4000
5000
This will certify that this curve is based on actual test performance.
A. Brkich
3/5/70
BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant
Low Pressure Safety Injection Pump Performance
Figure 6-4
Rev 26
CHARACTERISTIC CURVE
NO. 8X21 TYPE AL PUMP
1780 R.P.M.
PUMP NO. 0869174 ORDER NO. 01-32910
INGERSOLL-RAND COMPANY
CAMERON PUMP DIVISION
DATE 3-4-70
CURVE N-187



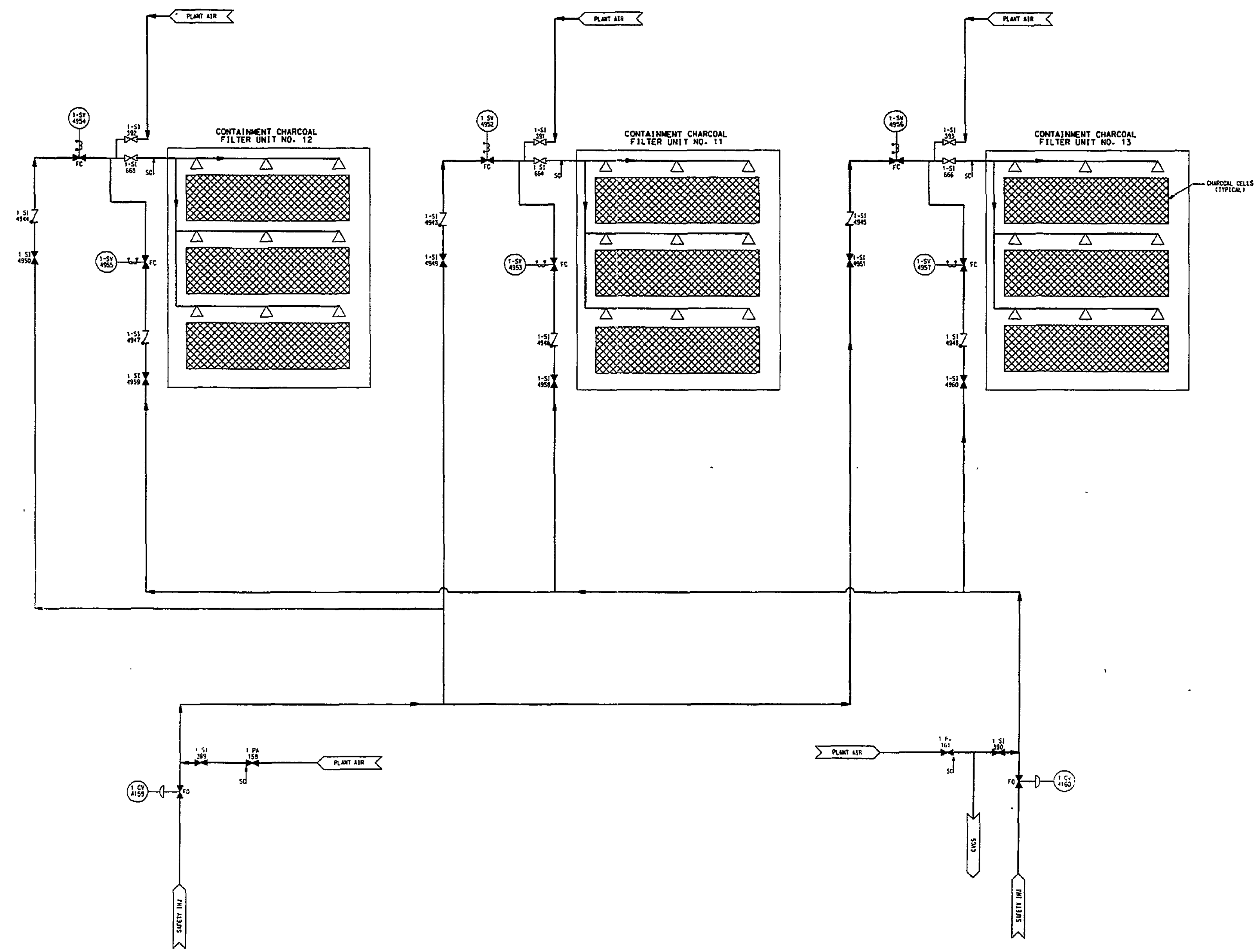
08110.DGN

Calvert Cliffs Nuclear Power Plant	CONTAINMENT SPRAY PUMPS	Figure 6-5 Revision 32



CONTAINMENT STRUCTURE
CHARCOAL FILTER UNIT ASS'Y
FIGURE 6-6

BALTIMORE GAS & ELECTRIC CO.
ELECTRIC ENGINEERING DEPARTMENT
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND



NOTES:

1 FOR DRAWING SYMBOLOGY, SEE BGE DRAWING NO 64329 (SL-060), SIMPLIFIED SYSTEM DRAWING LEGEND.

2. THIS DRAWING IS NOT TO BE USED FOR MAINTENANCE OR OPERATION OF THE PLANT SEE THE CORRESPONDING OM DRAWING, BGE NO 60711 (OM-52).

3. THIS SYSTEM IS DISABLED AT POWER IT IS AVAILABLE DURING OUTAGES ONLY, AND MUST BE MANUALLY INITIATED

CALVERT CLIFFS NUCLEAR POWER PLANT

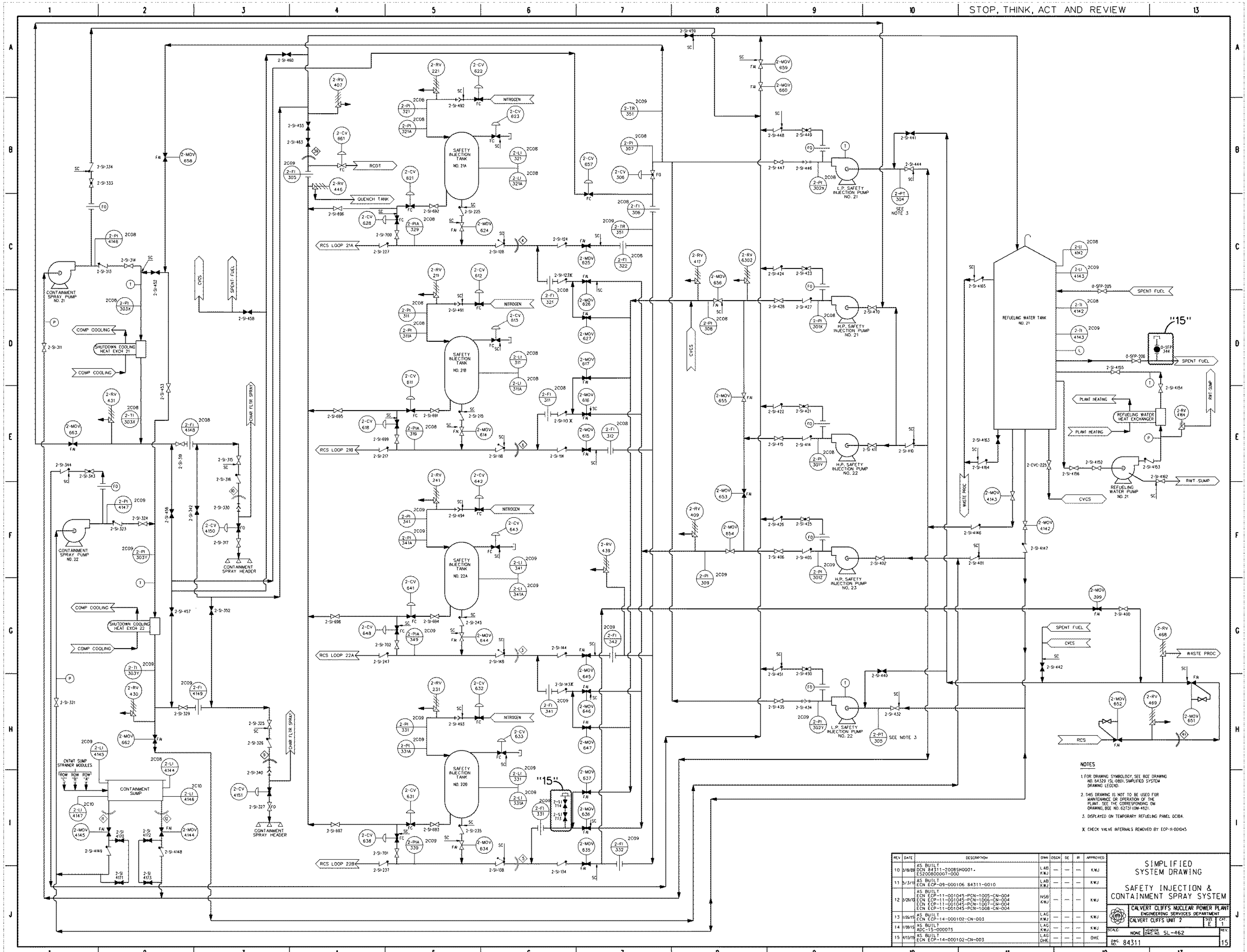
UFSAR FIGURE 6-7

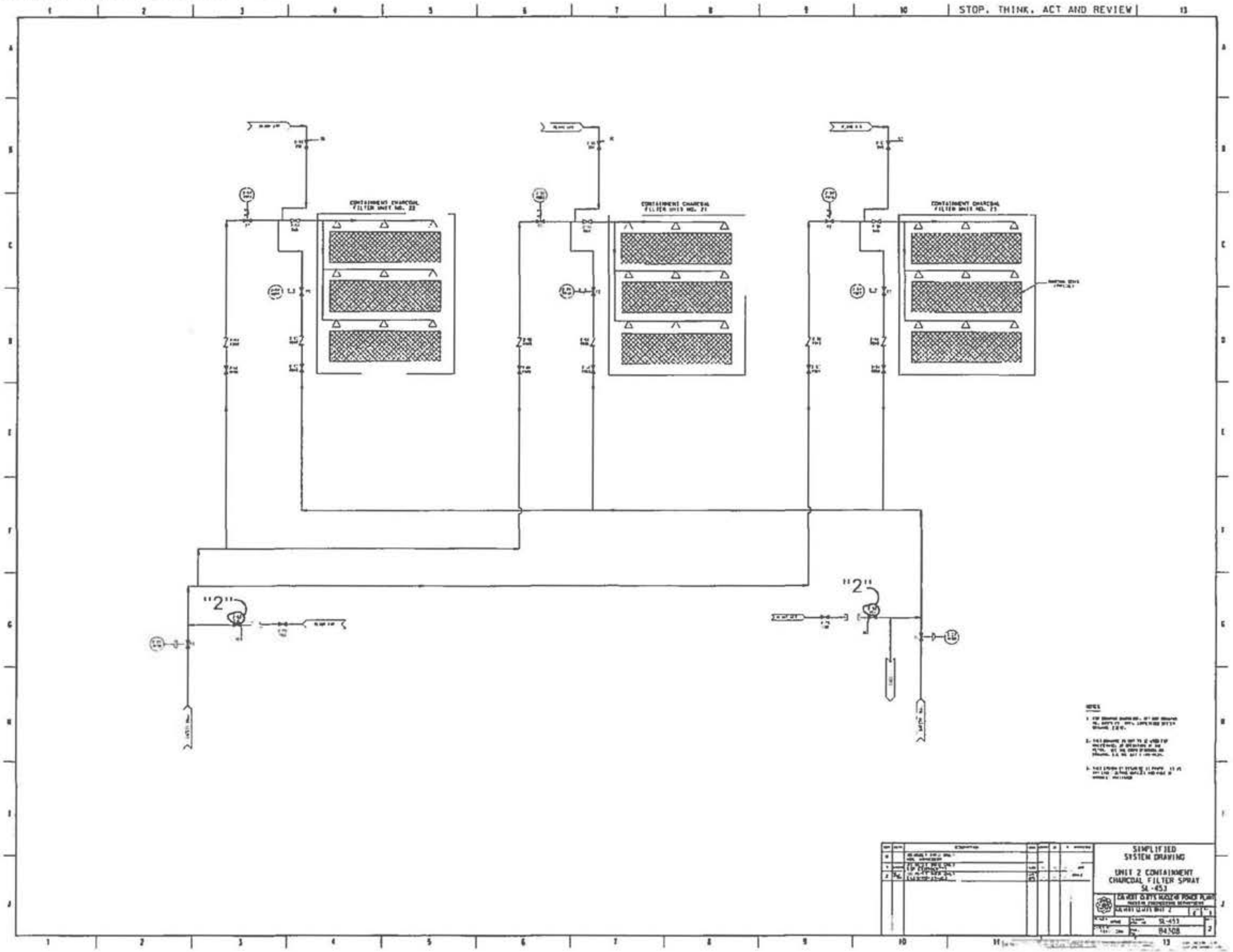
CONTAINMENT CHARCOAL FILTER SPRAY
UNIT 1

BGE DRAWING 64-308, REV 1

Revision 21

FIGURE 6-10 SAFETY INJECTION AND CONTAINMENT SPRAY - UNIT 2




STOP. THINK, ACT AND REVIEW
"2"
"2"
SIMPLIFIED SYSTEM DRAWING
UNIT 2 CONTAINMENT CHARCOAL FILTER SPRAY
SL-453
84308

CHAPTER 7
INSTRUMENTATION AND CONTROL

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LIST OF ACRONYMS

AFAS	Auxiliary Feedwater Actuation System
AFW	Auxiliary Feedwater
ATWS	Anticipated Transient Without Scram
BGE	Baltimore Gas and Electric Company
CCW	Component Cooling Water
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEDS	Control Element Drive System
CET	Core Exit Thermocouple
CFR	Code of Federal Regulations
CIS	Containment Isolation Signal
CMI	CEA Motion Inhibits
CRS	Containment Radiation Signal
CRT	Cathode Ray Tube
CSAS	Containment Spray Actuation Signal
CSF	Critical Safety Functions
CVCIS	Chemical and Volume Control Isolation Signal
d/p	Differential Pressure
DAS	Data Acquisition System
DEV	Deviation Alarm
DNB	Departure From Nucleate Boiling
DNBR	Departure From Nucleate Boiling Ratio
DSS	Diverse Scram System
DTT	Diverse Turbine Trip
ECCS	Emergency Core Cooling System
EHC	Electro-Hydraulic Control
ERGO	Excessive Regulatory Group Overlap
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
ETS	Emergency Trip Signal
FCR	Facility Change Request
HJTC	Heated Junction Thermocouple
HPSI	High Pressure Safety Injection
ICC	Inadequate Core Cooling
ICCI	Inadequate Core Cooling Instrumentation
ICI	Incore Instrumentation
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss-of-Coolant Accident
LOCI	Loss-of-Coolant Incident
LPSI	Low Pressure Safety Injection
MFIV	Main Feedwater Isolation Valve
MIRG	Shutdown Group Insertion Interlock
MISH	Regulating Group Withdrawal Interlock
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break

LIST OF ACRONYMS

MSR	Moisture Separator Reheaters
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OOS	Out-of-Sequence
PA	Public Address
PAM	Post-Accident Monitoring
PDIL	Power Dependent Insertion Limit
PORV	Power-Operated Relief Valve
Q-List	Quality-List
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protective System
RPSCIP	Reactor Protective System Calibration and Indication Panel
RTD	Resistance Temperature Detector
RVLMS	Reactor Vessel Level Monitoring System
RWT	Refueling Water Tank
SCMM	Subcooled Margin Monitor
SER	Safety Evaluation Report
SG	Steam Generator
SGIS	Steam Generator Isolation Signal
SIAS	Safety Injection Actuation Signal
SPDS	Safety Parameter Display System
SRW	Service Water
TM/LP	Thermal Margin/Low Pressure
TMR	Triple Modular Redundant
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report

7.0 INSTRUMENTATION AND CONTROL

7.1 INTRODUCTION

The plant systems are instrumented to provide information on plant conditions at selected locations, to protect equipment and personnel from undesirable conditions and to control the plant during startup, operation, and shutdown. The principal control station for the plant is in the Control Room.

The plant is started up and shut down under remote manual control. Annunciators, indicators, and recording devices will alert the operator and provide data on plant conditions.

Instrumentation and controls essential to plant safety are located in the Control Room. The instrumentation is arranged in groups on the control boards so that when corrective action is required, all pertinent indicators, recorders, and controllers are within easy reach of the operator. The control board is a duplex benchboard. Visible and audible alarms located on the superstructure over the main control board annunciate and identify abnormal operation conditions. Telephone systems provide both in-plant and external communication. The Control Room is a controlled temperature environment, kept well within the design ambient temperature requirements of the instruments. The computer room temperature and humidity are kept closely controlled.

To ensure reliability, components of established quality are selected and used in the instrumentation and control equipment. All protection systems that actuate reactor trip engineered safety features (ESFs), and auxiliary feedwater (AFW) components are designed to conform to the criteria of Institute of Electrical and Electronic Engineers (IEEE) 279 and those sections that are relevant from the Commission's proposed General Design Criteria, as published February 20, 1971. The Diverse Scram System (DSS) does not meet IEEE 279. The requirements for this system are established by 10 CFR 50.62 (Section 7.11). The protection instrumentation consists of four independent multiple channels to permit system testing without reducing the degree of protection provided. Reliable sources of electrical power are provided to ensure safe and reliable plant operation (Chapter 8).

The operation of the reactor within established limits is achieved by its inherent characteristics, instrumentation and control systems, and by operational procedures and administrative controls. Potential departures from these limits are audibly and visibly annunciated in the Control Room. A Reactor Protective System (RPS) is designed to protect the core and the Reactor Coolant System (RCS) pressure boundary and to initiate reactor trips.

The ESF instrumentation provides the equipment necessary to initiate the required safety features functions. This system also monitors the power sources acting to assure the availability of emergency power for operation of at least the minimum ESFs (Chapters 6 and 8). This system is provided with the necessary redundant circuitry and physical isolation so that a single failure within the system would not prevent the proper system action when required. This system is provided with test facilities and alarms to alert the operator when certain components trip, malfunction or are inoperable. The controls are designed to automatically provide the sequence of operations required to initiate ESF operation with or without off-site power available.

There are no RPS and ESFs instrumentation transmitters for which the trip setpoints are within 5% of the high or low end of the calibrated range, or within 5% of the overall instrument design range.

7.2 REACTOR PROTECTIVE SYSTEM

7.2.1 GENERAL

The RPS consists of sensors, amplifiers, logic, and other equipment necessary to monitor selected Nuclear Steam Supply System (NSSS) conditions and to effect reliable and rapid reactor shutdown if any one or a combination of conditions deviates from a preselected operating range. The system functions to protect the core and RCS pressure boundary.

7.2.2 DESIGN BASIS

The RPS is designed on the following bases to assure adequate protection for the core:

a. Instrumentation conforms to the provisions of the proposed IEEE, Criteria for Nuclear Power Plant Protection Systems (IEEE 279, August 1968).

b. No single component failure can prevent safety action.

c. Four independent measurement channels are provided for each parameter that can initiate safety action.

d. Channel independence is assured by separate connection of the sensors to the process systems and of the channels to vital instrument busses.

e. The four measurement channels provide trip signals to six independent logic matrices, arranged to effect a two-out-of-four coincidence logic having outputs to four independent trip paths.

f. A trip signal from any two-out-of-four protective channels causes a reactor trip.

g. When one of the four channels is taken out of service, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel.

h. The protective system AC power is supplied from four separate vital instrument busses.

i. Open circuiting, or loss of power supply for the channel logic, initiates an alarm and a channel trip.

j. The trip logic matrices assume the nonconducting state to provide a tripping function.

k. The RPS can be tested with the reactor in operation or shut down.

l. The manual trip system is independent of the automatic trip system.

m. Trip signals are preceded by pretrip alarms to alert the operator of undesirable operating conditions in cases where operator action can correct the abnormal condition and avoid a reactor trip.

n. The RPS components are independent of the control system.

o. All equipment, including panels, components and cables associated with the RPS, are marked with colored markers or nameplates in order to facilitate identification. The cabinets of the RPS are appropriately tagged A, B, C, and D, respectively, to distinguish between channels. Internal wiring in the RPS cabinets is not color coded. External to the RPS cabinets, the RPS uses color coded cable within the main control panels to ease identification of these channels. At termination points the incoming and outgoing cables of the RPS are appropriately tagged to identify the channel.

p. Electrical circuit isolation is provided between the RPS and the annunciators and plant computer.

q. The RPS is designed such that the de-energized state initiates a channel trip. This feature ensures that if channel continuity is lost, that channel will fail in a safe condition. The modules are not interlocked to prevent withdrawal but are designed such that withdrawal of one module causes a channel trip, associated channel trip annunciation and pretrip annunciation. Withdrawal of any other module of that

parameter will cause a full trip since the system is in the two-out-of-four trip mode. A unique key is available at the plant, allowing only one of the four channels of any one parameter to be bypassed at any time that the RPS is required to be operable. Strict administrative control ensures that this requirement is not violated. This bypass produces a two-out-of-three trip logic for the remaining three channels.

Generic Letter 83-28 requested response to certain generic concerns resulting from an incident at another plant in which the scram circuit breakers failed to open on receipt of an automatic reactor trip signal. There are four areas of concern:

Post-Trip Review

The Nuclear Regulatory Commission (NRC) asked for a description of the post-trip review to assure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

We responded that administrative controls exist that require a post-trip review to be conducted to determine the acceptability of the restart. Another review is also conducted to provide an in-depth analysis of the events relative to long-term plant operations.

Equipment Classification and Vendor Interface

We were tasked to identify all safety-related components necessary to trip the reactor. This identification could be in documents, procedures or information handling systems used to control safety-related plant activities. In addition, a program was required to be established and maintained to ensure that vendor information for safety-related components is complete, current, controlled and referenced or incorporated in plant instructions and procedures.

We have a program to identify, classify and treat components required for performance of reactor trip as safety-related. Vendor information comes from Combustion Engineering, Inc. (CE) the NSSS vendor, and the RPS is a part of the CE interface.

Post-Maintenance Testing

The objective of this requirement was to assure that post-maintenance operability testing of safety-related components in the RPS is conducted. This testing should also prove that the equipment is capable of performing its safety function before being returned to service.

We have post-maintenance testing procedures which are required to be performed when maintenance is performed on these components. Vendor guidance is incorporated in these procedures.

Reactor Protective System Reliability Improvements

Vendor-recommended reactor trip breaker modifications and associated RPS changes were required to be completed, a comprehensive program of preventive maintenance and surveillance testing was required for the reactor trip breakers, and the shunt trip attachment must activate automatically.

Vendor Recommendations for modifications of reactor trip components were reviewed and implemented. Since that time, the trip breakers have been replaced with NLI/Square D Masterpact type breakers. Preventive maintenance is performed on these breakers in accordance with the manufacturer's recommendations. We verify the response time of the undervoltage and shunt trip circuits.

7.2.3 SYSTEM DESCRIPTION

As shown in Figures 7-1 and 7-2, the RPS consists of four trip paths operating through the coincidence logic matrices to maintain power to, or remove it from, the control element drive mechanisms (CEDMs). Four independent measurement channels normally monitor each plant parameter which can initiate a reactor trip. Individual channel trips occur when the measurement reaches a preselected value. The channel trips are combined in six two-out-of-two logic matrices. Each two-out-of-two logic matrix provides trip signals to four one-out-of-six logic units, each of which causes a trip of the breakers in the AC supply to the CEDM power supplies. Each CEDM power supply source is separated into two branches.

Reference Figure 4-1 for RCS process instrumentation.

As shown in Figure 7-2, a two-out-of-four logic operating on undervoltage relays on the CEDM power supply lines is used to provide an auxiliary signal coincident with reactor trip. This signal is utilized to trip the turbine.

Reactor trip is accomplished by deenergizing the CEDM coils allowing the control element assemblies (CEAs) to drop into the core by gravity. The reactor trip allowable limits and pretrip limits are listed in Table 7-1.

Figure 7-2A shows the RPS interface logic.

Reactor trip is initiated by the following conditions:

7.2.3.1 <u>High Rate-of-Change of Power</u>

The rate-of-change of power high trip is used to trip the reactor when excore logarithmic power measured by the wide range logarithmic neutron flux monitors indicates an excessive rate of change. This trip functionally minimizes transients for events such as a boron dilution event, continuous CEA withdrawal, or CEA ejection from subcritical conditions. Because of this function, such events are assured of having much less severe consequences then events initiated from critical conditions. The rate-of-change of power is monitored at start-up by four wide-range channels, as shown in Figure 7-4. The channels cover a range of greater than ten decades.

7.2.3.2 <u>Variable Over Power Level</u>

A reactor trip in power level (Q) (Section 7.2.3.7) is provided to trip the reactor in the event of a CEA ejection incident, and to help prevent violation of the CEA position vs. power level assumed in the Thermal Margin and Axial Flux Offset trips. The high power trip setpoint can be set no more than a predetermined amount above the indicated plant power. Operator action is required to increase the setpoint as plant power is increased. The setpoint is automatically decreased as power decreases.

The variable setpoint and Q are compared in a bistable trip unit in each of the four safety channels. The high power trip is initiated by two-out-of-four coincidence logic from the four safety channels.

Figure 7-20 shows the operation of the system. If Q decreases, the setpoint Q_{TR} follows it, remaining above Q by a fixed, adjustable bias Q_b. If Q now increases, the setpoint remains at the minimum value of Q + Q_b last achieved until reset by the operator.

The system is capable of holding the setpoint Q_{TR} at the previous minimum of Q + Q_b indefinitely. This capability is achieved by storing Q_{TR} as a digital word.

The reset circuit is designed to apply a momentary signal to the appropriate terminal of the digital storage device when a pushbutton is pressed. This causes Q_{TR} to achieve the current value of Q + Q_b. The reset circuit is buffered to permit locating one of the pushbuttons outside of the RPS, another pushbutton is located on the RPS channel panel.

The signal Q_{TR} is limited so that, regardless of the logic described above, it cannot go above or below limits set by potentiometers.

Other circuits generate a pretrip limit for the bistable trip unit, as well as a contact closure to alert the operator when power increases after reaching a minimum. The pretrip alarm provides audible and visual annunciation in addition to CEA withdrawal prohibit signals.

Power level and Q - Q_{TR} are displayed on the main control board. Power Level is also taken to the Control Element Drive System (CEDS) for use in the power dependent insertion limit (PDIL) calculation.

The pretrip alarm signals are initiated by bistable trip units from the same channels which provide the reactor trip signals. The pretrip alarms provide audible and visual annunciation in addition to CEA withdrawal prohibit signals.

7.2.3.3 Low Reactor Coolant Flow

This reactor trip is provided to protect the core against Departure from Nucleate Boiling (DNB) in the event of a coolant flow decrease. The setpoint for this trip is low enough to allow continued operation during abnormal frequency transients down to 57.5 Hz on the 500 kV system.

The flow measurement signals are provided by summing the output of the differential pressure (d/p) transmitters across each steam generator (SG) to provide an indication of the total coolant flow through the reactor. A reactor trip is initiated by two-out-of-four coincidence logic from the four independent measuring channels when the flow function falls below the preset value, as shown in Figure 7-5.

Pretrip alarms are initiated if the coolant flow function approaches the minimum required for reactor operation. The zero power mode bypass switch allows this trip to be bypassed for subcritical testing of CEDMs. The trip bypass is automatically removed above 10^{-4}% power.

An indication of reactor flow rate is given by summing the d/p measurement across the SGs. This measurement is read out in the form of d/p in psi and not actual flow. The low-flow reactor trip is actuated directly by the summed d/p signal.

Both SG d/p signals are summed for all operating modes. The flow channels are also designated to monitor reverse flow.

The hot functional test program to determine low-flow trip setpoints consisted of two phases; first to demonstrate that the actual flow through the core was equal to or greater than the value stated in the Updated Final Safety Analysis Report (UFSAR); second, to establish the sum of the SG d/p at which the low-flow trip

would be set to meet the trip conditions listed in UFSAR, Chapter 14. These two phases are discussed in greater detail below.

PHASE I

The initial steady state reactor mass flow rates for four-pump, three-pump, two-pump same-loop and two-pump opposite-loop operation were determined during the power test program using the measured pump d/p and the pump characteristic curves. These flow rates, including the allowances made for measurement uncertainty, potential flow loss due to increased flow resistance with core life, and temperature effects were shown to be greater than or equal to the design flows used in the UFSAR Chapter 14 accident analysis.

The test instrumentation provided for Calvert Cliffs accurately measured the reverse flow which occurs with two-pump same-loop operation. The d/p across the pump was measured as the pump discharge pressure minus the pump suction pressure which is positive when the pump is operating. When a pump is not operating and reverse flow occurs through that pump, the pressure at the discharge side of the pump will still be higher than the pressure on the suction side, and a positive d/p occurs once again. The same pump d/p instrumentation which was calibrated to read the expected range of positive pump d/p was therefore used (along with the pump characteristic curves) to determine both forward and reverse flow. Accuracies on the d/p instrumentation were factored into the uncertainty assigned to the total reactor mass flow rate measurement.

The reactor mass flow rate was initially determined, as described above, by summing the measured mass flow rates through each primary coolant pump. In addition, a relationship was developed between the d/p across the SGs and the reactor mass flow rates for all pump combinations.

PHASE II

Having established the steady state adequacy of the flow for all allowable pump combinations, it was then necessary to develop protective system setpoints for all pump combinations which must be capable of tripping the reactor upon the loss of one or more pumps. The summed SG d/p values for a given pumping configuration are related to the reactor mass flow rate for any specified coastdown transient. The relationship was first calculated for each coastdown transient and then verified during the power test program. These relationships were used to generate low-flow trip calibration curves which in turn were used to determine the low-flow trip setpoints for the various pump configurations.

For the two-pump same-loop case in question, even though the SG d/p in the active loop is positive and the SG d/p in the inactive loop negative, the algebraic sum of the SG d/ps is related to the reactor mass flow rate during a pump coastdown transient by means of the two-pump same-loop low-flow trip calibration curve.

The SG d/p transmitters were calibrated such that the d/p for all pump combinations when the reactor is at operating temperature fell between the upper and lower 5% of their range.

7.2.3.4 Low Steam Generator Water Level

An abnormally low SG water level indicates a loss of SG secondary water inventory. If not corrected, this would result in a loss of capability for removal of heat from the RCS.

The low SG water level reactor trip protects against the loss of feedwater flow incident (Section 14.6). The trip allowable limit specified in Table 7-1 assures that sufficient water inventory will be in the SG at the time of trip to provide approximately 10 minutes before AFW is required for the removal of decay heat.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two downcomer level d/p transmitters, one on each SG. Audible and visual pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions (Figure 7-17).

7.2.3.5 Low Steam Generator Pressure

An abnormally high steam flow from one of the SGs, e.g., that which would occur as the result of a steam line break, would be accompanied by a marked decrease in steam pressure. To protect against an excessive rate of heat extraction from the SGs and subsequent cooldown of the reactor coolant following a steam line break, a reactor trip is initiated by low SG pressure. The low SG pressure trip provides protection for larger feedwater line breaks.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two pressure transmitters - one on each SG. Audible and visual pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions.

The reactor trip allowable limit specified in Table 7-1 is sufficiently below the full-load operating pressure so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow (Figures 7-17 and 7-18).

A bypass is provided for the low SG secondary pressure trip. Bypass is accomplished manually by means of a switch in each channel. The manual bypass is enabled only below a preset secondary pressure and is automatically removed above this setpoint.

The low SG pressure trip bypass is initiated manually by turning a switch to the BYPASS position. The bypass is removed, regardless of the manual switch position, if the auctioneered high of the SG pressures exceeds a predetermined setpoint. A latch feature ensures that the SG pressure will not remove the trip as it decreases. Figure 7-18 is a schematic of this circuit. Trip bypass is accomplished by energizing the "N" terminal of the trip unit with +15 Volts, through manually- and automatically-actuated contacts. Any open contact in this path will remove the bypass and allow trip. The automatic contact is actuated by relay K22, which is controlled through its own normally open (latch) contact by a bistable device set to operate at a predetermined pressure. To permit reset of K22, the latching contact is shunted when the manual switch is in the OFF position.

If the operator leaves the manual switch in the BYPASS position during plant heatup, the bistable contact will open when the highest SG pressure exceeds the setpoint. This will de-energize K22 and disconnect +15 Volts from the trip unit,

allowing trip. An annunciator is provided on the reactivity controls and protective system control board (1CO5) which indicates when the low SG channel trip is bypassed.

If the operator turns the manual switch to OFF, the path from +15 Volts to the trip unit will be interrupted by the manual contacts, and trip will be allowed.

For setting and testing the bistable device by use of the trip tester, a TEST SELECT switch is provided to disconnect the signal not being tested. This can only be done if the manual bypass switch is in the OFF position.

The contact testing system consists of two pushbuttons: one for auto bypass removal test and one for manual bypass removal test. The purpose of these tests is to check the status of the bypass circuit contacts. These tests do not alter or change the contacts from either an open or closed position.

Pressing the AUTO TEST pushbutton completes a path through the K22 contact to the light; therefore, an energized light indicates that bypass is allowed by the automatic removal circuit. Pressing the MAN TEST pushbutton similarly tests the manual contact. Pressing both pushbuttons energizes the light regardless of bypass status; this tests the light. The light is also energized for both the manual and automatic contacts closed, i.e., when the trip bypass is in effect.

7.2.3.6 High Pressurizer Pressure

A reactor trip for high pressurizer pressure with concurrent opening of the PORVs is provided to prevent excessive blowdown of the RCS by relief action through the pressurizer safety valves.

The trip signals are provided by four independent narrow-range pressure transducers measuring the pressurizer pressure.

A reactor trip is initiated by two-out-of-four coincidence logic from the four independent measuring channels if the pressurizer pressure exceeds 2400 psia. This signal also opens the power-operated relief valves (PORVs).

Pretrip alarms are initiated if the pressurizer pressure exceeds 2350 psia.

7.2.3.7 Thermal Margin/Low-Pressure Trip

A reactor trip is initiated whenever the RCS pressure signal drops below either a preset pressure or a computed value described below, whichever is higher. The computed value is derived using functions of the minimum permissible power to fuel design limit on Departure from Nucleate Boiling Ratio (DNBR) (P_{fdn}) and of the reactor inlet and outlet temperatures and nuclear instrument power. Output from the resistance temperature detectors (RTDs) in the hot and cold legs of each SG is used to generate a coolant differential temperature (ΔT) signal. This coolant ΔT signal is proportional to reactor power and is utilized as such. The ΔT power signal is compared to the nuclear instrument power signal (Figure 7-21) and the higher signal is modified by two functions which provide penalty factors based on the worst-case CEA position and the actual calculated Axial Shape Index. This signal feeds a setpoint calculator (Figure 7-6A).

In the setpoint calculator, a function generator produces a signal proportional to the minimum permissible P_{fdn}. The ratio of power and P_{fdn} is then combined with a

coolant temperature signal, a pressure signal representing an asymmetric SG loading (Figure 7-6B), and a reference signal (corresponding to the minimum pressure for trip) to define the pressure for reactor trip in order to assure that the minimum DNBR is not exceeded during anticipated operational occurrences.

The trip signal is initiated by a two-out-of-four coincidence logic from four independent safety channels, and audible and visual pretrip alarms are actuated to provide for annunciation on approach to reactor trip conditions. The pretrip action also initiates a CEA withdrawal prohibit. A block diagram of a thermal margin/low-pressure (TM/LP) trip channel is shown in Figure 7-6.

Figure 7-21 shows a block diagram of the thermal (ΔT) power calculation.

The calculation begins with the generation (by temperature transmitters) of currents representing the cold and hot leg temperatures in each loop. By forcing these currents through precision resistors and utilizing the resulting voltage drops, voltages representing cold leg temperatures (T_{c1} and T_{c2}) and hot leg temperature (T_h) are sent to the calculator. The latter signal is the average T_h for the two loops. The T_h temperature signal is also filtered by a lag module prior to the calculator to minimize the effect of momentary spikes and oscillations on the circuit.

In the calculator, the higher cold leg temperature signal is selected and subtracted from the hot leg temperature signal to determine the temperature rise. The calculator generates terms proportional to the first and second powers of the temperature rise and to the product of temperature rise and cold leg temperature. These three terms represent thermal power for four-pump operation and steady state conditions, accounting for coolant density, specific heat, and flow rate variations with temperature and power. To provide an adequate core power indication during mild transients, such as ramp load changes, a dynamic response term is added as shown. A bias term is added for calibration to adjust the output to zero at power. The sum of these terms represents the core power for four-pump operation under steady state or mild transient conditions.

The coefficient of the term proportional to the temperature rise (Kalpha) is set by the potentiometer labeled "Delta T Power Calibrate" on the Reactor Protective System Calibration and Indication Panel (RPSCIP) front panel. A plastic cover protects this potentiometer from accidental adjustment. This factor is adjusted to make the thermal power calculation agree with the plant calorimetric calculation.

The thermal power (B) is subtracted from the nuclear power (ϕ), generated by the NI Channel, and the difference is displayed on a meter with a range of -10% to +10% of full power. The meter has adjustable upper and lower setpoints. The contacts energize a local light when the deviation goes outside the range defined by the setpoints.

To make the nuclear power signal agree with the thermal power and/or the plant calorimetric calculation, a potentiometer labeled "Nuclear Power Calibrate" is provided on the RPSCIP front panel. This potentiometer adjusts the gain of the NI channel from 0.8 to 1.33. An auctioneering circuit selects the higher of nuclear power or thermal power for use in the remainder of the system. This auctioneered signal is called Q.

Regular checks are performed to ensure agreement between plant calorimetric and ΔT power; if required, adjustments are made at the "Delta T Power Calibrate" dial on the reactor protective calibration and indication panel.

The zero power mode bypass switch allows this trip to be bypassed for low power testing. The trip bypass is automatically removed above about 10^{-4}% power. An additional feature of this zero power mode bypass is to remove the ΔT power component of the power signal Q. This prevents RCS temperature channel range limits from causing the generation of incorrect ΔT power signals which would cause false trips on high power during low power testing or plant heat-up and cooldown. This bypass applies to the low flow and TM/LP trips. This circuit is similar to the low SG pressure bypass, except that the bypass is automatically removed by a contact from a bistable device located in the wide-range nuclear instrument drawer, which actuates at a preset power level. No latching feature is required in this circuit. The contact status testing system is identical to the system for the low SG pressure bypass. An annunciator is provided on the reactivity controls and protective system control board (1C05) which indicates when the zero power mode channel is bypassed.

7.2.3.8 Loss of Load

The loss-of-load trip is an equipment protective trip and is not required for reactor protection. (Section 14.5)

A loss-of-load trip above a preset power level is initiated by actuation of the turbine trip system. This trip is anticipatory in nature as it precedes the high-pressure trip.

The plant annunciator indicates on control board CO5 when this trip is bypassed. This inhibit is automatically removed at a predetermined setpoint.

7.2.3.9 High Containment Pressure

A trip is provided on high containment pressure in order to assure that the reactor is tripped prior to, or at least concurrent with, safety injection actuation.

Four pressure transmitters actuate trip units which are connected in a two-out-of-four coincidence logic to initiate the protective action if the containment pressure exceeds a preselected value.

The containment pressure transmitter sensing lines are the only instrument lines to which Safety Guide 11 is applicable and are designed in accordance with this Safety Guide.

The containment pressure transmitters are located outside the Containment Structure in the electrical penetration room. They are located as close as practical to the containment and installed using short connections between the containment penetrations and the instruments. The transmitters are designed as pressure retaining devices, whereby rupture of the sensing device would not release radioactivity to the environment, but would contain the radioactivity within the housing of the instrument. Each sensing line is provided with a solenoid-operated isolation valve which is located as close as possible to the containment penetration. The isolation valves are controlled from the Control Room and are provided with position switches for remote indication in the Control Room. The isolation valves and instrument lines, up to and including the pressure retaining parts of the instruments, are Seismic Category I.

7.2.3.10 Manual Trip

A manual reactor trip is provided to permit the operator to trip the reactor. The actuation of two adjacent pushbutton switches on the control panel causes interruption of the AC power to the CEDM power supplies. Two sets of trip pushbutton switches are provided. The manual trip function is testable during reactor operation. The pressing of these two buttons is required to effect a reactor scram; however, they do not need to be depressed simultaneously.

7.2.3.11 Axial Flux Offset Trip

A reactor trip is initiated as determined by signals from the power range safety channels whenever the axial flux shape approaches a preset value. The axial flux offset trip signals are initiated by two-out-of-four coincidence logic from the four power range safety instrumentation channels, with audible and visual pre-trip alarms actuated to provide for annunciation on approach to reactor trip conditions.

The trip setpoint is selected to ensure that the axial flux distribution does not result in conditions exceeding fuel damage limits.

7.2.3.12 Asymmetric Steam Generator Load

A reactor trip is initiated via the TM/LP trip channel as determined by combining the SG pressure signals within the TM/LP Trip Calculator, such that, if these pressures differ by more than a fixed amount in either direction, the calculated primary pressure trip setpoint is raised by putting a signal into the maximum selector function that selects the highest of: the asymmetric factor signal, or the calculated pressure P_{var}, or P_{min} (Figures 7-6A and 7-6B).

This trip functions to add additional safety margins in the event of a slow closure of one of the MSIVs.

7.2.4 SIGNAL GENERATION

Four instrument channels are used to generate the signals necessary to initiate the automatic reactor trip action. The signal cable routing and readout drawer locations are separated and isolated to provide channel independence.

7.2.4.1 High Rate-of-Change of Power

The wide-range logarithmic channels obtain signals from four detector channels. Each channel consists of a two-ganged fission chamber assembly. These assemblies are located in wells on the reactor cavity wall around the reactor. The outputs are amplified at amplifier assemblies located outside containment and carried to the signal processing drawer in the Control Room. A signal proportional to the logarithm of neutron flux over the range of 10^{-8}% to 200% of full power is obtained (Figure 7-7). This signal is then differentiated to obtain the rate-of-change of power.

7.2.4.2 High Power Level

The signal for each of the four power range safety channels is obtained from one of the four detector assemblies located on the reactor cavity wall around the reactor. Each assembly consists of two uncompensated ion chambers stacked vertically to monitor the full length of the core. The DC current signal from each ion chamber is fed directly to the Control Room drawer assembly. The ion chambers cover the range from 0.1% to 200% power (Figure 7-4).

7.2.4.3 Flow, Water Level, Pressure, and Thermal Margin

The flow, water level, pressure, and thermal margin trips are each actuated from signals generated by separate sets of transmitters. Flow is measured by monitoring the pressure difference between the hot leg piping and the SG outlet plenum. SG water level and pressure are monitored in each SG. The RCS pressure is measured in the pressurizer. Temperature measurements are taken from the reactor inlet and outlet piping in each loop and combined with coolant pressure to ensure adequate thermal margin.

Piping and connections for these transmitters are separated and isolated to provide independence. The output of each transmitter is an ungrounded current loop supplying signal receivers and bistable trip modules.

7.2.4.4 Axial Flux Offset

The signals for the axial flux offset trip are processed in the four power range safety channel drawer assemblies. Each channel receives signals from two vertically-stacked uncompensated ion chambers located in the reactor cavity. The two signals, representing flux in the upper half of the core and flux in the lower half of the core, are combined in each of the four drawer assemblies to generate four independent trip signals proportional to axial flux shape. The Axial Shape Index (Ye) from the excore detectors (power range and regulating channels) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The Axial Shape Index (YI) used for the trip and pretrip signals in the Reactor Protection System is the above value (Ye) modified by an appropriate multiple (A) and a constant (B) to determine the true core axial power distribution for that channel. This value of Axial Shape Index (YI) represents the excore detector equivalent of the peripheral Axial Shape Index determined by the incore detector system for a given excore channel.

7.2.5 LOGIC OPERATION

Refer to Figure 7-2 for the following discussion.

Each measurement channel which can initiate protective action operates a channel trip unit; each trip unit includes three sealed, electromagnetically-actuated reed relays and associated contacts. Four trip units are actuated for each trip condition, e.g., high reactor coolant pressure. The relays in each of these four trip units provide a separate trip path; the trip paths are designated channels A, B, C, and D.

The relays in each trip unit are numbered one, two, and three. The normally open contacts from the No. 1 relay group of Channel A are connected into a two-out-of-two logic matrix with Channel B relay contacts. (The normally open contacts are used for the logic ladders so that the relays are energized and the contacts closed under operating conditions.)

The No. 2 and No. 3 relay contacts are similarly connected into two other two-out-of-two logic matrices with Channel C and Channel D relay contacts.

With the No. 2 and No. 3 relay contacts of Channels B, C, and D similarly arranged in BC, BD, and CD combinations of two-out-of-two logic matrices, there are a total of six two-out-of-two logic matrices, forming a two-out-of-four coincidence logic with respect to the input channels.

At the output of each logic matrix is a set of four sealed, electromagnetically-actuated relays. These sets are designated the AB, AC, AD, BC, BD, and CD logic trips. The contacts from one relay of the logic trip set from each logic matrix output are placed in series with corresponding contacts from the remaining sets in each of the four trip paths. Each of these paths is the power supply line to a trip breaker control relay whose contacts provide actuation of undervoltage and shunt trips on the trip circuit breakers, thus interrupting the AC power to the CEDM power supplies. Deenergizing of any one trip breaker control relay interrupts (opens) one trip path and trips the two breakers controlled by that trip path. Deenergizing any set of four logic trip relays causes an interruption of all trip paths and a full trip. Each of the six logic trip matrices energizes one set-of-four logic trip relays.

If one of the trip units is to be removed for maintenance, the logic matrices may be changed from a two-out-of-four trip to a two-out-of-three trip by the operation of the logic bypass switch (shown on the output of the trip module, Figure 7-3). One key-operated switch is provided for each trip unit. Only one key is provided for the trips for any one variable to ensure that only one of a group of four could be bypassed at one time. The operation of the key-operated switch to bypass the trip function of a single bistable trip unit is indicated by a light on its face. This light meets the requirements of paragraph 4.13 in IEEE 279 in that it provides continuous indication of the bypass in the Control Room.

Where the trip is to be allowed only in selected power ranges, a neutron flux signal is utilized to inhibit the action of the trip units. A manually-actuated inhibit action may, under administrative control, be applied to the low reactor coolant flow, thermal margin and low SG pressure trips for zero power testing. The inhibits on reactor coolant flow and thermal margin are automatically removed above a preset power. The inhibit on SG pressure is automatically removed above a preset pressure. The high power rate-of-change trip is automatically inhibited below about 10^{-4}% power and above 15% power. Protective system criteria are met by this use of neutron flux signals to provide multiple independent inhibit or reset signals.

The CEDMs are separated into two groups. The CEDM power supplies in each group are supplied in parallel with three-phase AC power from the motor-generator sets. Two full capacity motor-generator sets are provided so that the loss of either set does not cause a release of the CEAs. Each power supply source is separated into two branches. Each side of each branch line passes through two trip circuit breakers (each actuated by a separate trip path) in series so that, although both sides of the branch lines must be deenergized to release the CEAs, there are two separate means of interrupting each side of the line. This arrangement provides means for the testing of the protective system.

7.2.6 TESTING

Since operation of the protective system will be infrequent, the system is periodically and routinely tested to verify its operability. A complete channel can be individually tested without initiating a reactor trip or violating the single failure criterion, and without inhibiting the operation of the RPS.

The RPS is capable of being checked from the trip unit input through the power supply circuit breakers of the CEDMs. The majority of the components in the protective system can be tested during reactor operation. The remainder of the components can be checked by comparison with similar channels or channels that involve related information. These components, which are not tested during reactor operation, will be tested during scheduled reactor shutdown to assure that they are capable of performing the necessary functions. Minimum frequencies for checks, calibration, and testing of the RPS

instrumentation are given in the Technical Specifications. Overlap in checking and testing is provided to assure that the entire channel is functional. The use of individual trip and ground detection lights, in conjunction with those provided at the supply bus, assure that possible grounds or shorts to another source of voltage can be detected.

During reactor operation, the measuring channels are checked by comparing the outputs of similar channels and cross-checking with related measurements. The trip units are tested by inserting a voltmeter in the circuit, noting the signal level, initiating a test input and noting signal level required to effect trip action. This provides the necessary overlap in the testing process and also enables the test to establish that the trip can be effected within the required tolerances. The test signal is provided by a test signal generator which is connected to the trip module at the signal input terminals. With the test signal generator connected, the desired signal is selected and then inserted into the trip unit by depressing the manual test switch. The test circuit permits various rates of change of signal input to be used. Trip action (opening) of each of the trip unit relays is indicated by individual lights on the front of the trip unit. The pretrip alarm action is indicated by a separate light.

The sets of logic trip relays at the output of each logic matrix are tested one at a time. The test circuits in the logic permit only one logic ladder to be opened and one set of relays to be held at a time; the application of hold power to one set denies the power source to the other sets. In testing a logic trip set (e.g., AB), a holding current is initiated in the test coils of the logic trip relays by turning the matrix relay trip test switch to "off" and depressing the matrix logic AB test pushbutton switch. Operation of the matrix trip test switch initiates a deenergizing current in the test coils of a parallel pair of trip unit relays. With the ladder logic relay contacts open, the logic trip relays may be deenergized one at a time (by rotating the matrix relay trip test switch) to open the associated trip breakers. Indicator lights on the trip status panel provide verification that coil operation and trip breaker actuation conditions have occurred.

Sensor responses were measured initially during factory acceptance tests. During plant startup testing, the response times from an input signal to the protection system trip units were verified through the opening of the trip circuit breakers.

The capability does not exist within the protection system to verify response times of trip parameters during normal plant operation; however, these tests can be performed during refueling periods.

The response times given in Table 7-2 provide one of the bases for operability determination of the RPS instrumentation referred to in the Technical Specifications.

Periodic testing can be carried out from the Control Room to ensure the continuity of the measurement loop. A supplementary signal is introduced into the measurement loop that is bypassed and the response to this signal is indicated on a meter in the protection system. This proves the continuity of the loop.

The overall loop response is designed to be less than those times used for safety analysis (Section 14.1).

7.2.7 SYSTEM EVALUATION

The RPS was manufactured under strict engineering and quality control specifications. These specifications require that the equipment be inspected for workmanship, proper materials, and channel separation as required by IEEE 279. Furthermore, all intra- and inter-connection wiring was tested for continuity and an insulation test was performed between each conductor and chassis ground, and between each individual pair of

connectors. An operational test was performed on the system during which time input signals were simulated to ensure that the protective system is capable of producing the proper trip signals. The system was packaged for shipment in accordance with specifications. The marking and packaging were inspected for compliance with the specification. All the above-mentioned tests were documented by the manufacturer. The quality assurance program described in Appendix 1A was applicable to the RPS during the construction phase.

The RPS is designed to limit reactor power and coolant conditions to levels within the design capability of the reactor core. Instrument performance characteristics, response time, and accuracy are selected for compatibility with, and adequacy for, the particular function. Trip setpoints are established by analysis of system parameters. Factors such as instrument inaccuracies, bistable trip times, CEA travel times, valve travel time, circuit breaker trip times, and pump starting times are considered in establishing the margin between the trip setpoints and the safety limits. The time response of the sensors and protective systems are evaluated for abnormal conditions. Since all uncertain factors are considered as cumulative for the derivation of these times, the actual response time may be more rapid. However, even at the maximum times which are added to the CEA drop time, the system provides conservative protection.

The wiring in the protective system is grouped so that no single fault or failure, including either an open or shorted circuit, will negate protective system operation. Signal conductors are protected and routed independently.

Loss of, or damage to, any one path will not prevent the protective action. Sensors are piped so that blockage or failure of any one connection does not prevent protective system action. The process transducers located in the Containment Structure are specified and rated for the intended service. Those components, which must operate in the loss-of-coolant accident (LOCA) environment, are rated for the LOCA temperature, pressure, and humidity conditions. Results of type test are used to verify these ratings. In the Control Room the nuclear instrumentation and protective system trip paths are located in four compartments. Mechanical and thermal barriers between these compartments reduce the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltage does not cause channel malfunction. Where RPS signals feed annunciators, data loggers, or computers, buffering by isolation amplifiers (or equal) is used to ensure circuit isolation. In instances where the RPS is feeding the annunciators, isolation is ensured through the use of relay contacts. When redundant channels supply the computer, isolation amplifiers are used.

The protective system is designed and arranged to be able to perform its function with a single failure of any component. Some of the faults and their effects are described below.

In the analog portion of the system:

a. A loss of signal in a measurement channel initiates channel trip action for all trips except high rate-of-change of power, high pressurizer pressure, high power level, and high containment pressure.
b. Shorting of the signal leads to each other has the same effect as a loss of signal. Shorting a lead to a voltage source has no effect since the signal circuit is ungrounded. Periodic testing includes checks for possible grounds or applications of potential to the signal circuit.
c. Open circuit of the signal leads has the same effect as a loss of signal.

d. Single grounds of the signal circuit have no effect. Periodic checking of the system will assure that the circuit remains ungrounded.

In the logic portion of the circuit:

a. Inadvertent operation of the relay contacts in the matrices will be identified by the indicating lights.
b. Shorting of pairs of contacts in the matrices will prevent the trip relay set from being released. Such shorts are detectable in the testing process by observing that the trip relay sets cannot be dropped out. Testing is accomplished by successive opening of the logic matrix contact pairs.
c. Shorting of the matrices to an external voltage has no effect since they are ungrounded. The testing process will indicate accidental application of potential to a matrix. Equipment is provided to detect grounds on the matrices.
d. The logic matrices will each be supplied by two power sources. Loss of a single power source has no effect on operation. Loss of power to a logic matrix initiates a trip condition.
e. Failure of a logic trip relay set to actuate has no effect since there are six sets in series in the trip action and any one set initiating trip action will cause the action to be completed.
f. The failure of one trip breaker control relay in a trip breaker circuit has no effect since there are two trip breakers in series, either of which will provide the necessary action.
g. Single grounds in the trip breaker control relay circuits have no effect since the circuit is ungrounded. Ground detectors on each 125 Volt DC bus also indicate an accidental ground.
h. The AC circuit supplying power to the trip breaker control relay coil is fed from an isolation transformer. The circuit has a local ground detection system. Each of the four trip paths are fed from a separate 120 Volt AC vital instrument bus.
i. The CEDM power supply circuits operate ungrounded so that single grounds have no effect. The CEDMs are supplied in two groups by separate pairs of power supplies to further reduce the possibility of a CEA being improperly held. The CEDM load requirements are such that the application of any other local available voltage would not prevent CEA release.

The locations of the sensors and the points at which the sensing lines are connected to the process loop have been selected to provide physical separation of the channels, thereby precluding a situation in which a single event could remove or negate a protective function. Process transmitters located inside the containment and required for short-term operation following a LOCA are qualified for the intended service in the LOCA environment. The routing of cables from these cabinets is arranged so that the cables are separated from each other and from power cabling to minimize the likelihood of common event failures. This includes separation at the containment penetration areas. In the Control Room, the four nuclear instrumentation and protective system trip channels are located in individual compartments. Mechanical and thermal barriers between these compartments minimize the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltages do not cause channel malfunction.

The RPS is designed as a Class I system. The specifications for the RPS components incorporate the applicable seismic requirements for each component, including spectrum response curves for the specific component location generated by the time-history method.

These components are qualified by either of the two following methods:

In most cases, the supplier is required to qualify his equipment by calculation or testing, or a combination of both. This qualification is formally documented and submitted for approval.

In other cases, tests or calculations are performed by independent consultants or laboratories who submit a formal report. Acceptance of the equipment from the supplier is contingent upon the proof of suitability as established by the results of those tests or calculations.

The choice of an analytical or experimental qualification procedure is determined by the size, shape, and structural or functional simplicity of the equipment in accordance with the criteria outlined in IEEE 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations." Racks, panels, or other supporting structures are generally qualified by analysis, while bistable trip units and other modules are generally qualified through testing. Tests and calculations are performed following the guidelines of IEEE 344.

Type testing was used for RPS panels, racks, and equipment qualification in accordance with IEEE 323, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations." The results of type tests was submitted by Combustion Engineering, Inc. (CE) in the form of a topical report covering a series of plants. The submission date for this report was September 1972.

Radiation design criteria for RPS components located within normally radioactive areas are specified at a gamma level of 1 rad/hr for 40 years, except for the main coolant RTDs, which were specified at 10 rad/hr for 40 years. Protective system equipment not located in normally radioactive areas or located in areas of very low activity has been specified accordingly. Periodic tests and calibration will allow detection of gradual equipment deterioration and will assure capability of the system to operate as required by the original design basis since the interval between such tests will be short compared to the time required for significant deterioration.

All other material and equipment associated with safety-related systems have been specified to be suitable for the appropriate 40-year integrated dose.

There are no RPS instrumentation transmitters for which the trip setpoints are within 5% of the high or low end of the calibrated range, or within 5% of the overall instrument design range.

7.2.8 POWER SUPPLY

The power for the protective system is supplied from four separate and independent vital 120 Volt AC busses. Each vital bus is supplied from a separate battery system through a dual inverter. During normal operation, the battery chargers maintain a floating charge on each battery while at the same time, supplying power to the vital inverters. Upon loss of auxiliary AC power, the batteries provide the power for inverter operation. In the event of loss of one battery supply, only the protective channel associated with the battery goes into a trip condition. Each preferred bus also has provision for connection to an inverter backup bus to permit servicing of the inverters.

The distribution circuits from the preferred busses are provided with fuses properly coordinated with upstream fuses and circuit breaker protection to assure that individual circuit faults are isolated.

TABLE 7-1

REACTOR TRIP ALLOWABLE LIMITS AND PRETRIP LIMITS

NO.	REACTOR TRIP	PRETRIP ALARM LIMIT	PRETRIP ALARM LIMIT
1	High Power Level	≤ 8% Above Measured Power Q	≤ 10% Above Measured Power Q
	4-Pump Operation	≤ 104.5%	≤ 107%
2	High Rate-of-Change of Power[a]	≤ 1.5 decades/min	≤ 2.6 decades/min
3	Low Reactor Coolant Flow[b] 4-Pump Operation	≥ 94%	≥ 92%
4	Low SG Water Level (Auctioneered low of SG #1, SG #2)	≥ 32" below normal water level	≥ 50" below normal water level
5	Low SG Pressure[c] (Auctioneered low of SG #1, SG #2)	≥ 735 psia	≥ 685 psia
6	High Pressurizer Pressure	≤ 2350 psia	≤ 2400 psia
7	Thermal Margin/Low-Pressure[b]	Variable, 50 psia above Trip Allowable Limit	Variable but not below 1875 psia
8	Loss of Load[d]	N.A.	N.A.
9	High Containment Pressure	≤ 3 psig	≤ 4 psig
10	Axial Flux Offset[d]	[e]	[e]
11	Manual Trip	N.A.	N.A.
12	Thermal Margin/SG Pressure Differential Hi	≤ 100 psid	≤ 135 psid

[a] Inhibited above 15% and below 10^{-4}% power.

[b] Manual inhibit permitted below about 10^{-4}% power: automatically removed above 10^{-4}% power.

[c] Manual inhibit permitted below 785 psia: automatically removed above 785 psia.

[d] Inhibited below 15% power.

[e] Trip and pretrip setpoints are a function of power level.

TABLE 7-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

(BOTH UNITS)

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Level – High	≤ 0.40 seconds[(a),(b)] and ≤ 12.0 seconds, cold leg[(c)] and ≥ 10.0 seconds and ≤ 60.0 seconds, hot leg[(c),(d)]
3.	Reactor Coolant Flow – Low	≤ 0.50 seconds
4.	Pressurizer Pressure – High	≤ 0.90 seconds
5.	Containment Pressure – High	≤ 0.90 seconds
6.	Steam Generator Pressure – Low	≤ 0.90 seconds
7.	Steam Generator Water Level – Low	≤ 0.90 seconds
8.	Axial Flux Offset	≤ 0.40 seconds[(a),(b)] and ≤ 12.0 seconds, cold leg[(c)] and ≥ 10.0 seconds and ≤ 60.0 seconds, hot leg[(c),(d]
9.	a. Thermal Margin/Low Pressure	≤ 0.90 seconds[(a),(b)] and ≤ 12.0 seconds, cold leg[(c)] and ≥ 10.0 seconds and ≤ 60.0 seconds, hot leg[(c),(d)]
	b. Steam Generator Pressure Difference – High	≤ 0.90 seconds
10.	Loss of Load	Not Applicable
11.	Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

NOTE: The response times given in this table provide one of the bases for operability determination of the RPS instrumentation referred to in the Technical Specifications.

(a) Neutron detectors are exempt from Response Time testing. Response Time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

(b) Response Time does not include contribution of RTDs.

(c) Calculator input response time only. This value is equivalent to the time interval required for the T-Cold RTDs and the T-Hot RTDs and T-Hot lag modules' outputs to achieve 63.2% of their total change when subjected to a step change in RTD temperature.

(d) T-Hot response time of ≥ 10 seconds is input to UFSAR, Chapter 14, Safety Analyses. T-Hot response time of ≤ 60 seconds ensures that plant operation is not impacted by a delayed RTD response.

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS

The Engineered Safety Features Actuation Systems (ESFAS) initiate the start of equipment which protects the public and plant personnel from the accidental release of radioactive fission products in the unlikely event of a loss-of-coolant, main steam line break (MSLB), or loss of feedwater incident. The safety features function to localize, control, mitigate, and terminate such incidents in order to minimize radiation exposure levels for the general public. The ESFAS was supplied by Vitro Laboratories, Division of Vitro Corporation of America.

Additional features were provided to the ESFAS with components supplied by Vitro Corporation. These features are maintenance bypass switches and bypass module, isolation module fault indication, and auctioneered 15 Volt DC power supplies for the logic modules, which are sequenced with the actuation relays' 28 Volt DC power supply. They were installed to minimize the potential for inadvertent actuations during maintenance and test activities.

The ESFAS provides independent (from RPS) actuation for the Auxiliary Feedwater Actuation System (AFAS). Implementation of Diverse Scram System (DSS) provides independent (from RPS) actuation of Diverse Turbine Trip (DTT). Implementation of DSS provides independent (from RPS) actuation of reactor trip. This satisfies the 10 CFR 50.62 requirements for mitigation of Anticipated Transient Without Scram (ATWS) events. (Section 7.10, 7.11)

7.3.1 DESIGN BASIS

7.3.1.1 Conformance to Standards

The design of the ESFAS and component parts was based on the applicable requirements of IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." Maximum consideration has been given to the following criteria consistent with the objectives of this document:

a. Single Failure

Any single failure within the protection system will not prevent proper protection system action when required.

b. Quality of Components and Modules

Components and modules used in the manufacture of the actuation systems exhibit a quality consistent with the nuclear power plant 40-year design life objective and with minimum maintenance requirements and low failure rates.

c. Channel Independence

The actuation systems include four redundant sensor subsystems and two redundant actuation subsystems. Independence has been provided between redundant subsystems or channels to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction. Independence has been obtained by:

1. Electrical Isolation

Electrical isolation has been provided between redundant channels, between sensor and actuation subsystems and between the ESFAS and ancillary equipment. Where electrical isolation is provided, an application of short circuit, open wire, ground, or potential does not

inhibit a protective action as a result of the failure of the redundant system (NOTE 1).

2. Physical Separation

   Physical separation has been maintained between redundant sensor subsystems, between sensor and actuation subsystems, and between redundant actuation subsystems by providing separate and isolated cabinets for each of the four sensor subsystems and each of the two actuation subsystems. Each of the four containment pressure sensor channels has a different containment penetration. A minimum of 3' is provided for each sensor channel (NOTE 1).

NOTE 1: The Containment Spray Actuation Signal (CSAS)/Steam Generator Isolation Signal (SGIS) trip of the main feedwater isolation valves (MFIVs), main feedwater (MFW) pump, Condensate Booster Pump, and Heater Drain Pump does not meet physical separation criteria. Electrical isolation is provided by relay contacts, 480 to 120 Volt control transformers, and fuses.

3. System Repair

   The system has been designed such that routine servicing and preventative maintenance can be performed without interfering with normal plant operation and without loss of system function availability. Performance of these operations does not result in a simultaneous unavailability of both actuation subsystems (Section 7.3.1.2). The system is mechanically and electrically divided into subunits or modules based on the following considerations:

   a) Standardization of subunits
   b) Minimization of interconnections and interwiring
   c) Interchangeability of subunits

   The subunits include associated equipment such as indicating lights, pushbuttons, potentiometers, and selector switches.

7.3.1.2 <u>Security and Annunciation</u>

The ESFAS is designed to provide annunciation and indication of module withdrawal or loss of power. Withdrawal or loss of power to a sensor module results in a trip signal to its associated two-out-of-four logic matrices. Sensor modules are not interlocked to prevent withdrawal of more than one module; however, withdrawal of two sensor modules of a common actuation signal will result in a trip of the associated actuation channels.

A maintenance bypass capability is provided for each sensor module. When a sensor module is placed in the bypass mode with its keylock switch, local indication and a Control Room alarm annunciate the bypass condition.

The doors of each ESFAS cabinet are equipped with a lock; one key fits all doors. Contacts are provided for annunciation of an unlocked or open sensor cabinet door. A withdrawal of an actuation logic module will not result in a trip of that channel.

7.3.1.3 Seismic Requirements

The ESFAS is classified as Seismic Category I and are designed to withstand all simultaneous horizontal and vertical accelerations resulting from the Safe Shutdown Earthquake without loss of functions.

The specifications for the ESFs and the emergency power system components incorporate the applicable seismic requirements for each component, including spectrum response curves for the specific component location generated by the time-history method.

These components are qualified by either of the following methods:

In most cases, the supplier is required to qualify his equipment by calculation or testing, or a combination of both. This qualification is formally documented and submitted for approval.

In other cases, tests or calculations are performed by independent consultants or laboratories who submit a formal report. Acceptance of the equipment from the supplier is contingent upon the proof of suitability as established by the results of those tests or calculations.

The choice of an analytical or experimental qualification procedure is determined by the size, shape, and structural or functional simplicity of the equipment in accordance with criteria equivalent to or exceeding that outlined in IEEE 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations (1971)." Racks, panels, or other supporting structures are generally qualified by analysis, while bistable trip units and other modules are generally qualified through testing. Tests and calculations are performed following criteria equivalent to or exceeding that outlined in the guidelines of IEEE 344-1971.

The new components supplied by Vitro Corporation for Facility Change Request (FCR) 87-87 are qualified to the 1987 version of IEEE 344. These components are very similar to the original equipment and were qualified by analysis based on similarity.

Equipment in the ESFs and the emergency power system requiring dynamic test is listed in Table 7-3.

All dynamic test results were in conformance to the seismic specifications indicating no failure or no loss of functions.

7.3.1.4 Environmental Requirements

All components which must operate in a LOCA environment were type-tested at the expected temperature, pressure, and humidity.

The components installed by FCR 87-87 meet the requirements of IEEE 323, 1983, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations" for the mild environments of the cable spreading rooms.

7.3.2 SYSTEM DESCRIPTION

The ESFAS is shown on Figures 7-10 and 7-22. The actuation system is divided into four sensor subsystems (sensor channels ZD, ZE, ZF, and ZG), two actuation subsystems (actuation channels ZA and ZB), and two logic subsystems for sequential loading of the diesel generators.

The cabinets of the ESFAS are tagged ZA, ZB, ZD, ZE, ZF, and ZG, respectively, to distinguish between channels. The ESFAS and emergency bus switchgear are distinguished from non-safety-related equipment by the use of red nameplates. At termination points, the incoming and outgoing cables of the ESFs are tagged to identify the channel, and additionally, the tag attachment devices are color coded to distinguish between channels.

7.3.2.1 Sensor Subsystems

The sensor subsystems monitor redundant and independent process variables and trip when the variables reach unsatisfactory levels. Physical locations of the sensors are shown on instrument location drawings prepared from plant general arrangement drawings. Each of the sensor subsystems consists of one sensor channel of the following process variables:

a. Containment Pressure - one each for Safety Injection Actuation Signal (SIAS), CSAS, and Containment Isolation Signal (CIS)
b. Pressurizer Pressure
c. Containment Radiation
d. Refueling Water Tank (RWT) Level
e. SG Pressure
f. SG Level (one for each SG)
g. Undervoltage
h. West Penetration Room Pressure and Letdown Heat-Exchanger Room Pressure

All process variables provide analog signals with the exception of the RWT level which produces a digital signal by actuation of level switches. Each analog sensor channel includes an indicator located at the actuation system cabinets for monitoring of the process variable.

In addition to the digital signals provided for the ESFs for the RWT subsystems, the RWT is equipped with level transmitters which provide analog signals to indicators mounted on the ESFs control board in the Control Room. The two indicators are separated by fire barriers and all electrical wiring to them is routed in separate paths. One indicator provides full (wide) range indication of tank level, and the other indicator provides narrow range indication of the normal water level. The full range indicator provides only a low level alarm. The narrow range indicator provides both a high level and a low level alarm. The high level alarm is to alert the operators of an impending overflow of water from the RWT to the Miscellaneous Waste Processing System. The low level alarms are used to assist the operator in monitoring for sufficient water inventory in the RWT. The alarms are located in the Control Room on the ESFs control board.

The RWT level switches are located within a room of the Auxiliary Building. This room is heated which will, in conjunction with the tank heating system, prevent freeze-up of the switches and will shelter the switches from rain, wind, etc.

Damage from missiles is extremely unlikely as only the tank heating pump, tank heating heat exchanger, and a room heater are located near the switches. To minimize damage in the unlikely event of a missile, the switches are physically separated. This physical separation, coupled with the curvature of the tank, will minimize the possibility of damage to more than one switch due to a missile of credible size.

Loss of one level switch will not adversely affect operation of the ESFs (Recirculation Actuation) since loss of the signal will not initiate or prevent actuation of the affected channel in the two-out-of-four logic network.

Recirculation is normally actuated automatically. Redundant level transmitters, alarms, and indicators are provided in the Control Room, to provide the operator with the necessary information to actuate the system.

The locating of the level switches within the Auxiliary Building, in conjunction with the guard force, provides protection from vandalism.

7.3.2.2 Actuation Subsystems

Two redundant and independent actuation subsystems monitor the sensor subsystem trip outputs and, by means of coincidence logics, determine whether a protective action is required. Each actuation subsystem initiates independent and redundant equipment. Either the A channel or the B channel controls sufficient equipment to protect the public in case of a LOCA, MSLB, or loss of feedwater.

Particular sensor and actuation channels are arranged to produce signals to initiate equipment consistent with the type of protective action required. These signals are designated:

a. SIAS
b. CSAS
c. CIS
d. Containment Radiation Signal (CRS)
e. Recirculation Actuation Signal (RAS)
f. SGIS
g. AFAS (The independent (from RPS) initiation of the AFW system satisfies the Diverse AFAS requirement for mitigation of ATWS events as required by 10 CFR 50.62)
h. Chemical Volume Control Isolation Signal (CVCIS)
i. DSS (see Section 7.11)
j. DTT [DSS (Section 7.11) provides an independent turbine trip from RPS, which satisfies the requirement for mitigation of ATWS events as required by 10 CFR 50.62]

The actuation channels of safety injection, containment spray, and containment isolation are subdivided into multiple actuation subchannels. The number of pieces of equipment initiated by a single actuation subchannel has therefore been reduced, allowing convenience and flexibility of periodic actuation system and equipment tests.

The response times given in Table 7-4 provide one of the bases for operability determination of the ESFAS instrumentation referred to in the Technical Specifications.

Safety Injection Actuation Signal - Providing signal inputs to SIAS are four independent pressurized pressure transmitters and four independent containment pressure transmitters (Figure 7-9). The containment pressure transmitters are separate from those used for initiation of the CSAS and the containment isolation actuation signal. Actuation occurs as a result of either two-out-of-four pressurizer

pressure sensor channel trip signals, two-out-of-four containment pressure sensor channel trip signals, or manual initiation from the Control Room. Each of the two independent SIASs from the two redundant actuation subsystems initiates the following:

SIAS ACTUATION SUBCHANNEL NO.			ACTION
1		a)	Opens Loop 11A(21B) Low Pressure Safety Injection (LPSI) Valve, 1(2)-MOV-615
		b)	Opens Loop 11B(21A) LPSI Valve, 1(2)-MOV-625
		c)	Opens Loop 12A(22B) LPSI Valve, 1(2)-MOV-635
		d)	Opens Loop 12B(22A) LPSI Valve, 1(2)-MOV-645
		e)	Starts Saltwater System Air Compressor Nos. 11 & 12, (21 & 22)
		f)	CR HVAC Temperature Control Bypass, Dampers Positioned in Recirculation, Post-Loss-of-Coolant Incident (LOCI) Filtration Starts, CR Lavatory/Kitchen Exhaust Fan Secures
	(d)	g)	Starts Service Water (SRW) Pumps No. 11, 12, & 13 (21, 22 ,& 23)
2		a)	Opens Loop 11A(21B) High Pressure Safety Injection (HPSI) Valve, 1(2)-MOV-616
		b)	Opens Loop 11B(21A) HPSI Valve, 1(2)-MOV-626
		c)	Opens Loop 12A(22B) HPSI Valve, 1(2)-MOV-636
		d)	Opens Loop 12B(22A) HPSI Valve, 1(2)-MOV-646
		e)	Opens Loop 11A(21B) Auxiliary HPSI Valve, 1(2)-MOV-617
		f)	Opens Loop 11B(21A) Auxiliary HPSI Valve, 1(2)-MOV-627
		g)	Opens Loop 12A(22B) Auxiliary HPSI Valve, 1(2)-MOV-637
		h)	Opens Loop 12B(22A) Auxiliary HPSI Valve, 1(2)-MOV-647
	(j)	i)	Starts HPSI Pump No. 11, 12, & 13 (21, 22, & 23)
3	(k)	a)	Throttles Containment Air Cooler 11(21) Service Water Inlet Valve 1(2)-CV-1581
	(k)	b)	Throttles Containment Air Cooler 12(22) Service Water Inlet Valve 1(2)-CV-1584
	(k)	c)	Throttles Containment Air Cooler 13(23) Service Water Inlet Valve 1(2)-CV-1589
	(k)	d)	Throttles Containment Air Cooler 14(24) Service Water Inlet Valve 1(2)-CV-1592
	(d)	e)	Starts Saltwater Pumps No. 11, 12, & 13 (21, 22, & 23)
4	(f)	a)	Opens Boric Acid Storage Tank No. 11(21) Gravity Feed Valve, 1(2)-MOV-509
	(f)	b)	Closes Boric Acid Storage Tank No. 11(21) Recirculating Valve, 1(2)-CV-510
	(f)	c)	Closes Boric Acid Storage Tank No. 12(22) Recirculating Valve, 1(2)-CV-511
	(f)	d)	Opens Boric Acid Direct Make-up Valve, 1(2)-MOV-514
		e)	Closes Containment Waste Gas Header Vent Valve, 1(2)-CV-2180
	(h)	f)	Closes Reactor Coolant Loop Hot Leg Sample Valve, 1(2)-CV-5467
	(b)	g)	Closes Safety Injection Tank Bleedoff Valve, 1(2)-CV-661

SIAS ACTUATION SUBCHANNEL NO.		ACTION
	[h] h)	Closes Reactor Coolant Sample Containment Isolation Valve, 1(2)-CV-5464
	i)	Closes Safety Injection Loop No. 11A(21B) Leakage Check Valve, 1(2)-CV-618
	j)	Closes Safety Injection Loop No. 11B(21A) Leakage Check Valve, 1(2)-CV-628
	k)	Closes Safety Injection Loop No. 12A(22B) Leakage Check Valve, 1(2)-CV-638
	l)	Closes Safety Injection Loop No. 12B(22A) Leakage Check Valve, 1(2)-CV-648
	[c] m)	Closes Volume Control Tank Make-up Stop Valve, 1(2)-CV-512
	n)	Opens Pressurizer Back-up Heater Bank No. 1, Breaker 52-1127(52-2127)
	o)	Opens Pressurizer Back-up Heater Bank No. 3, Breaker 52-1427(52-2427)
	p)	Opens Containment Cooler No. 11(21) SRW Out Valve 1(2)-CV-1582
	q)	Opens Containment Cooler No. 12(22) SRW Out Valve 1(2)-CV-1585
	r)	Opens Containment Cooler No. 13(23) SRW Out Valve 1(2)-CV-1590
	s)	Opens Containment Cooler No. 14(24) SRW Out Valve 1(2)-CV-1593
5	a)	Closes Turbine Building SRW Isolation Valve 1(2)-CV-1600
	b)	Closes Turbine Lube Oil & EHC Oil Cooler Water Isolation Valve, 1(2)-CV-1637
	c)	Closes RCP Seals Bleedoff Containment Isolation Valve, 1(2)-CV-506
	d)	Closes RCP Seals Bleedoff Containment Isolation Valve, 1(2)-CV-505
	[c] e)	Closes Volume Control Tank Outlet Valve, 1(2)-MOV-501
	f)	Closes Turbine Building SRW Isolation Valve, 1(2)-CV-1638
	g)	Closes Turbine Lube Oil & EHC Oil Cooler Water Isolation Valve, 1(2)-CV-1639
	[e] h)	Closes Loop 12A(22A) Letdown Line Containment Isolation Valve, 1(2)-CV-515
	[e] i)	Closes Loop 12A(22A) Letdown Line Containment Isolation Valve, 1(2)-CV-516
6	[f] a)	Opens Boric Acid Storage Tank No. 12(22) Gravity Feed Valve, 1(2)-MOV-508
	b)	Starts Boric Acid Pump Nos. 11 & 12 (21 & 22)
	[a] c)	Starts Charging Pump Nos. 11, 12 & 13 (21, 22, & 23)
7	[d] a)	Starts Component Cooling (CC) Pump Nos. 11, 12, & 13 (21, 22, & 23)
	b)	Opens CC Shutdown Cooling Heat Exchanger No. 11(21) Out Valve, 1(2)-CV-3828
	c)	Opens CC Shutdown Cooling Heat Exchanger No. 12(22) Out Valve, 1(2)-CV-3830

SIAS ACTUATION SUBCHANNEL NO.			ACTION
8		a)	Close Signal to Circulating Water Pump Room Air Cooler(s) Saltwater Isolation Valves Circuits, 1(2)-MOV-5250 & 1(2)-MOV-5251 (valves are spared in place and no longer required to function upon receipt of a SIAS signal)
		b)	Starts Containment Cooler Nos. 11, 12, 13, & 14 (21, 22, 23, & 24)
		c)	Starts Containment Spray Pump Nos. 11 & 12 (21 & 22)
	(a)	d)	Starts Containment Charcoal Filter Units Nos. 11, 12, & 13 (21, 22, & 23)
		e)	Closes Liquid Waste Evaporator No. 11(21) Component Cooling Isolation Valve, 1(2)-CV-3840
		f)	Closes Liquid Waste Evaporator No. 11(21) Component Cooling Isolation Valve, 1(2)-CV-3842
		g)	Starts LPSI Pumps No. 11 & 12 (21 & 22)
9		a)	Closes Containment Normal Sump Drain Isolation Valve, 1(2)-MOV-5462
		b)	Closes Containment Waste Gas Header Vent Valve, 1(2)-CV-2181
		c)	Closes Containment Normal Sump Drain Isolation Valve, 1(2)-MOV-5463
		d)	Stops Containment Purge Air Sample Isolation Valve, 1(2)-CV-5291
		e)	Closes Containment Purge Air Sample Isolation Valve, 1(2)-CV-5292
	(h)	f)	Closes Pressurizer No. 11(21) Vapor Sample Valve, 1(2)-CV-5465
	(h)	g)	Closes Pressurizer No. 11(21) Liquid Sample Valve, 1(2)-CV-5466
	(b)	h)	Closes Reactor Coolant Drain Tank Pump No. 11(21) Discharge Containment Isolation Valve, 1(2)-CV-4260
		i)	Closes Pressurizer Quench Tank O_2 Sample Valve, 1(2)SV-6531
		j)	Closes H_2 Purge Exhaust Valves, 1(2)-MOV-6900 & 1(2)-MOV-6901
10		a)	Opens Safety Injection Tank No. 11A(21B) Isolation Valve, 1(2)-MOV-614
		b)	Opens Safety Injection Tank No. 11B(21A) Isolation Valve, 1(2)-MOV-624
		c)	Opens Safety Injection Tank No. 12A(22B) Isolation Valve, 1(2)-MOV-634
		d)	Opens Safety Injection Tank No. 12B(22A) Isolation Valve, 1(2)-MOV-644
		e)	Starts Diesel Generator(s) No. 1A & 1B (2A & 2B)
		f)	Opens 480 V Unit Bus 17 Feeder Breaker 52-1702
		g)	Opens 480 V Unit Bus 17 Tie Breaker 52-1704
		h)	Starts Diesel Generator 1A Building Supply Fan F-10
		i)	Opens Diesel Generator 1A Essential Lighting Transformer Breaker 52-12308

SIAS ACTUATION SUBCHANNEL NO.		ACTION
	j)	Opens Diesel Generator 0C Feeder Breaker 152-1106 (152-2106)
	k)	Opens Diesel Generator 0C Feeder Breaker 152-1406 (152-2406)

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[a] Same as [d] except the success of circuit breaker closure is dependent upon the position of the isolating disconnect switch only.

[b] This action is not duplicated by the redundant actuation subsystem.

[c] The volume control tank discharge valve is closed by actuation Subsystem A. The make-up valve is closed by actuation Subsystem B.

[d] Where three 100% capacity pumps are provided, one pump is exclusively connected to ESFs electrical Bus No. 11(21) and started by actuation Subsystem A. Another pump is exclusively connected to the redundant ESFs Bus No. 14(24) and started by actuation Subsystem B. The third pump can be arranged electrically by movement of circuit breaker position and/or operation of disconnect switches for operation from either of the two independent ESFs electrical busses. Each actuation subsystem initiates a starting signal to the third pump which attempts closure of the third pump's circuit breaker associated with each subsystem. The success of circuit breaker closure is dependent upon failure of circuit breaker closure of the other pump associated with the same subsystem, and position of the isolating disconnect switch (attachment to Subsystem A or to Subsystem B).

[e] These valves cannot be closed for test during normal plant operation.

[f] Gravity valves are opened only by actuation Subsystem A. Feed valves are opened only by actuation Subsystem B. The gravity system and the feed system provide two redundant boric acid injection functions.

[h] Hot leg sample, pressurizer vapor sample, and pressurizer liquid sample valves are closed only by actuation Subsystem A. The redundant isolation function is provided by the main sample line valve closure by actuation system B. A key-locked, manual override of the SIAS signal to these valves is provided in the Control Room for the purpose of allowing post-accident sampling of the RCS. The override is annunciated in the Control Room via the plant annunciator system.

[j] Pump No. 11(21) is exclusively connected to Bus No. 11(21) and started by actuation Subsystem A. Pump No. 12(22), in pull-to-lock, is exclusively connected to Bus No. 14(24) and actuation Subsystem B. Pump No. 13(23) is connected to Bus No. 14(24) and started by actuation Subsystem B.

[k] The valves move to a throttled position upon a SIAS, and return to the full open position upon receipt of a RAS.

In addition to the capability for manual initiation of the actuation signal from the Control Room, each of the above listed actions may be individually initiated in the Control Room by appropriate control switch operation.

A safety injection block is provided to permit shutdown depressurization of the RCS without initiating safety injection. Block is accomplished manually. This process is under strict administrative control with block indicated by a local light and annunciated on the station annunciator system. It will not be possible to block above a present pressure and, if the system is blocked and pressure rises above this point, the block is automatically removed. The block circuit is designed to conform to the single failure criterion as specified in IEEE 279.

Wide range pressure and level transmitters are also provided to furnish pressurizer pressure and level measurements during reactor warm-up, when the reactor is shut down, or following a LOCA. These measurements will permit knowledgeable manual control of the safety injection pumps and valves.

The control switches for the manual control of those valves which isolate systems from the containment are wired so that the control switches must all be in their appropriate ESF position prior to allowing the ESF isolation signal to the valves to be reset to a non-isolation position. Upon loss of control air or loss of control circuit power, the ESF isolation valves automatically assume the operating position for the LOCA condition. Where two isolation control valves are provided for a single containment penetration, each valve is controlled by a separate actuation subsystem. Where one isolation valve is available, a single actuation subsystem initiates closure of the valve.

Containment Spray Actuation Signal - To provide the CSAS, four independent containment pressure transmitters are utilized. The pressure transmitters are separate from those used for initiation of the SIAS and the containment isolation actuation signal. Actuation occurs as a result of either two-out-of-four containment pressure sensor channel trip signals or manual initiation from the Control Room. Each of the two independent CSASs from the two redundant actuation subsystems initiates the following:

CSAS ACTUATION SUBCHANNEL NO.		**ACTION**
1		
2	a)	Opens Containment Spray Header No. 11(21) Isolation Valve, 1(2)-CV-4150
	b)	Opens Containment Spray Header No. 12(22) Isolation Valve, 1(2)-CV-4151
	c)	Closes SG Blowdown Service Water Isolation Valve, 1(2)-CV-1640
	d)	Closes SG 11(21) Top Bld Cont Valve, 1(2)-CV-4010
	e)	Closes SG 11(21) Bottom Bld Cont Valve, 1(2)-CV-4011
	f)	Closes SG 12(22) Top Bld Cont Valve, 1(2)-CV-4012
	g)	Closes SG 12(22) Bottom Bld Cont Valve, 1(2)-CV-4013
	h)	Closes Spent Fuel Pool Cooler No. 11 Service Water Out Valve, 1-CV-1596
	i)	Closes Spent Fuel Pool Cooler No. 11 Service Water In Valve, 1-CV-1597
	j)	Closes Spent Fuel Pool Cooler No. 12 Service Water Out Valve, 2-CV-1598
	k)	Closes Spent Fuel Pool Cooler No. 12 Service Water In Valve, 2-CV-1599
3	a)	Closes SG 11(21) Feedwater Isolation 1(2)-MOV-4516
	b)	Closes SG 12(22) Feedwater Isolation. 1(2)-MOV-4517
	c)	Closes MSIV 12(22) 1(2)-CV-4048
	d)	Closes MSIV 11(21) 1(2)-CV-4043
	e)	Stops Heater Drain Pump Nos. 11, 12, 21, 22
	f)	Stops Main Feedwater Pumps No. 11, 12, 21, 22
	g)	Stops Condensate Booster Pumps No. 11, 12, 13, 21, 22, & 23

NOTE: Actions 3a, b, e, f, and g may be overridden by a key-operated alarmed switch.

In addition to the capability for manual initiation of the actuation signal from the Control Room, each of the above-listed actions may be individually initiated in the Control Room by appropriate control switch operation.

To prevent an inadvertent Containment Spray System actuation in the case of an undesired trip of the CSAS, the containment spray pumps are started by SIAS, while the containment spray valves are opened by CSAS. These valves are fail-safe, i.e., upon loss of power, the valves open. While the containment spray valves are open, containment isolation is maintained by backup check valves in the spray system piping.

Containment Isolation Signal - The CIS is produced by either two-out-of-four containment pressure sensor channel trip signals or manual initiation from the Control Room. Containment pressure is monitored by four independent pressure transmitters which are separate from those utilized for SIAS and CSAS. Each of the two independent CISs from the two redundant actuation subsystems initiates the following:

CIS ACTUATION SUBCHANNEL NO.		ACTION
1		
3	[c] a)	Starts Penetration Room Exhaust Fans Nos. 11 & 12 (21 & 22)
	b)	Deenergizes Penetration Room Filter 11(21) Damper Solenoid Valve 1(2)-SV-5285 which opens Isolation Dampers 1(2)-PO-5285 & 1(2)-PO-5286
	[c] c)	Deenergizes Penetration Room Filter 12(22) Damper Solenoid Valve 1(2)-SV-5287 which opens Isolation Dampers 1(2)-PO-5287 & 1(2)-PO-5288
5	[a] a)	Closes Instrument Air Containment Isolation Valve, 1(2)-MOV-2080
	[b] b)	Closes RCP CCW Containment Isolation In Valve, 1(2)-CV-3832
	[b] c)	Closes RCP CCW Containment Isolation Out Valve, 1(2)-CV-3833

[a] A key-operated, manual override of CIS is provided to this valve.

[b] These valves cannot be closed for test during normal plant operation.

[c] Solenoid valves are normally deenergized and the dampers are normally opened using the handswitch.

In addition to the capability for manual initiation of the actuation signals from the Control Room, each of the above-listed actions may be individually initiated by appropriate control switch operation. Upon loss of control air or loss of control circuit power, the ESF isolation valves automatically assume the operating position for the LOCA condition. Where two isolation control valves are provided for a single containment penetration, each valve is controlled by a separate actuation subsystem. Where one isolation valve is available, a single actuation subsystem initiates closure of the valve.

The control switches for the manual control of those valves which isolate systems from the containment are wired so that the control switches must all be in the

appropriate ESF position prior to allowing the ESF isolation signal to the valves to be reset to a non-isolation position.

Containment Radiation Signal - The CRS is provided in order to limit release of radioactive fission products during refueling and maintenance periods when containment integrity is breached. Four independent radiation detectors located within the containment are provided and, upon coincidence of two-out-of-four trip signals or manual initiation from the Control Room, actuation occurs. The CRS (also, SIAS) isolates and secures the containment purge system. Each of the two independent signals from the two redundant actuation subsystems initiates the following:

CRS ACTUATION SUBCHANNEL NO.		**ACTION**
1	[a] a)	Closes Containment Purge Air Supply Isolation Valve, 1(2)-CV-1410
	[a] b)	Closes Containment Purge Air Exhaust Isolation Valve, 1(2)-CV-1412
	[a] c)	Stops Containment Purge Air Exhaust Fan No. 11(21)
	[a] d)	Stops Containment Purge Air Supply Fan No. 11(21)
	e)	Closes H_2 Purge Exhaust Valves, 1(2)-MOV-6900 & 1(2)-MOV-6901

[a] This action is not duplicated by the redundant subsystem.

In addition to the capability for manual initiation of the actuation signals from the Control Room, each of the above actions may be individually initiated in the Control Room by appropriate control switch operation.

The control switches for the manual control of those valves which isolate systems from the containment are wired so that the control switches must all be in the appropriate ESF position prior to allowing the ESF isolation signal to the valves to be reset to a non-isolation position. Upon loss of control air or loss of control circuit power, the ESF isolation valves automatically assume the operating position for the LOCA condition. Where two isolation control valves are provided for a single containment penetration, each valve is controlled by a separate actuation subsystem. Where one isolation valve is available, a single actuation subsystem initiates closure of the valve.

Recirculation Actuation Signal - Four independent RWT level switches provide digital inputs to the RAS. Upon coincidence of two-out-of-four low water tank level trip signals, or manual initiation from the Control Room, the RAS is generated. Each of the two independent signals from the two redundant actuation subsystems initiates the following:

RAS ACTUATION SUBCHANNEL NO.		**ACTION**
1		
	f)	Opens Containment Sump Discharge Valve, 1(2)-MOV-4144
	g)	Opens Containment Sump Discharge Valve, 1(2)-MOV-4145
	[a] h)	Closes Containment Spray & Safety Injection Pumps Recirculating Valve, 1(2)-MOV-659

RAS ACTUATION SUBCHANNEL NO.		ACTION
[a]	i)	Closes Containment Spray & Safety Injection Pumps Recirculating Valve, 1(2)-MOV-660
	j)	Stops LPSI Pumps No. 11 & 12 (21 & 22)
	k)	Allows Containment Air Cooler 11(21) Service Water Inlet Valve 1(2)-CV-1581 to return to full open
	l)	Allows Containment Air Cooler 12(22) Service Water Inlet Valve 1(2)-CV-1584 to return to full open
	m)	Allows Containment Air Cooler 13(23) Service Water Inlet Valve 1(2)-CV-1589 to return to full open
	n)	Allows Containment Air Cooler 14(24) Service Water Inlet Valve 1(2)-CV-1592 to return to full open

[a] A handswitch provides manual override of RAS to this valve. Switch is normally in lockout position with valve open. Switch is turned to ON at low RWT level to permit valve closure on RAS. Receipt of RAS when valve is in LOCKOUT and OPEN results in Control Room alarm. Lockout is provided to ensure minimum flow for safety injection pumps.

In addition to the capability for manual initiation of the actuation signal from the Control Room, each of the above actions may be individually initiated in the Control Room by appropriate control switch operation.

The RWT discharge valve is normally open and upon loss of control power the valve remains in the open position. Valve failure in the open position during the recirculation mode of operation does not affect pump suction since check valves are provided to maintain containment integrity and prevent reverse flow during the recirculation mode. The containment sump discharge valve is normally closed and upon loss of control power, the valve remains in the closed position.

Steam Generator Isolation Signal - Each SG is equipped with four independent pressure transmitters, the outputs of which are monitored by four independent bistables. Upon low SG pressure (which would occur as a result of steam line break) and two-out-of-four coincidence bistable trip signals, SG isolation signal is generated. The output of each independent isolation signal initiates closure of both steam line isolation valves and both main feed header isolation valves. In addition, the output of each independent isolation signal stops both heater drain pumps, both main feedwater pumps, and all three condensate booster pumps. This action reduces main feedwater flow to the steam generators while the main feed heater isolation valves are closing. The isolation signal may also be initiated by control switch operation in the Control Room.

A bypass is provided for shutdown depressurization and is accomplished manually by means of a momentary key operator switch in the Control Room. The bypass is enabled only below a preset pressure and is automatically removed above this pressure.

The control switches for the manual control of those valves which isolate systems from the containment are wired so that the control switches must all be in the appropriate ESF position prior to allowing the ESF isolation signal to the valves to be reset to a non-isolation position. Upon loss of control air or loss of control circuit power, the ESF isolation valves automatically assume the operating position for the LOCA condition. Where two isolation control valves are provided for a

single containment penetration, each valve is controlled by a separate actuation subsystem. Where one isolation valve is available, a single actuation subsystem initiates closure of the valve.

SGIS ACTUATION SUBCHANNEL NO.		ACTION
1	a)	Closes SG No. 11(21) Isolation Valve, 1(2)-MOV-4516
	b)	Closes SG No. 12(22) Isolation Valve, 1(2)-MOV-4517
	c)	Closes MSIV No. 11(21), 1(2)-CV-4043
	d)	Closes MSIV No. 12(22), 1(2)-CV-4048
	e)	Stops Heater Drain Pump Nos. 11, 12, 21, 22
	f)	Stops Main Feedwater Pump Nos. 11, 12, 21, 22
	g)	Stops Condensate Booster Pump Nos. 11, 12, 13, 21, 22, & 23

NOTE: Actions a, b, e, f, and g may be overridden by a key-operated alarm switch.

7.3.2.3 Undervoltage, Blocking, and Sequencing

Emergency diesel generators are provided for supplying power to ESFs in case of loss of the normal auxiliary system power. Undervoltage, blocking, and sequencing are required for sequential loading of the diesel generator and are a part of the engineered safety features actuation subsystem. Undervoltage, blocking, and sequencing signals associated with Bus No. 11 are a part of actuation Channel A. Those signals associated with Bus No. 14 are a part of actuation Channel B. Sequential actuation system blocking initially blocks, then unblocks in programmed steps, some of the actuation subchannels of safety injection, containment spray, and RASs when normal auxiliary power sources are unavailable and the loading of diesel generators is necessary. The undervoltage system contains multiple redundant undervoltage relays designated 127/B which are located at the 4160 Volt switchgear. The trip outputs are delayed, and then, by means of coincidence logic, it is determined whether action is required. The undervoltage system initiates the starting of the diesel generators and provides multiple contact outputs for load shedding.

Upon degradation of the normal auxiliary system voltage without a LOCA, the undervoltage system initiates the starting of diesel generators and load sheds the bus. The shutdown sequencer then automatically initiates the starting of the SRW pumps, the saltwater pumps, the instrument air compressor, Control Room air conditioning compressors, switchgear room air conditioning compressors, motor-driven auxiliary feedwater pumps, and 72' computer room air conditioning units.

7.3.3 SYSTEM SURVEILLANCE

7.3.3.1 Remote Annunciation

a. Tripped sensor channel
b. Tripped actuation channel
c. Permission to block pressurizer pressure
d. Blocked pressurizer pressure
e. Permission to block SG pressure
f. Blocked SG pressure
g. Tripped undervoltage relay
h. Tripped two-out-of-four undervoltage matrix

i. Blocked step of sequential actuation system blocking
j. Loss of power supply
k. Sequencer initiated
l. Sensor cabinet in maintenance bypass

7.3.3.2 Local Sensor Channel Surveillance

Each module or subunit is equipped with indicating lights. Typical functions to be indicated are:

a. Bistable trip
b. Power supply available
c. Cabinet fan failure
d. Sensor module in maintenance bypass
e. Isolation module fault or trip signal

All indicating lights are visible with the cabinet doors closed. Each indicating light is a push-to-test to check the lamp, except the indicating lights mounted with the maintenance bypass switches.

7.3.3.3 Local Actuation Channel Surveillance

Each module or subunit is equipped with indicating lights. Typical functions indicated are:

a. Tripped actuation subchannel
b. Blocked step of sequential actuation system blocking
c. Power supply available
d. Cabinet fan failure
e. Sequencer tripped
f. Undervoltage relay tripped
g. Tripped two-out-of-four undervoltage
h. Pressurizer pressure blocked
i. SG pressure blocked

7.3.4 ELECTRICAL POWER SUPPLY

The four redundant 118 Volt, 60 Hz, vital sources of supply (Section 8.3.5) are utilized by the ESFAS. Two vital sources provide power for a sensor subsystem and an actuation subsystem. The remaining two sensor subsystems receive power from the remaining vital sources. Physical and electrical isolation is maintained between the various redundant power supplies. Short circuit protection is provided at each system cabinet, and a trip of the protective device is indicated locally and annunciated in the Control Room.

7.3.5 SYSTEM EVALUATION

The ESFs initiation, control, and power supply systems were designed in accordance with Proposed IEEE Criteria No. 279, dated August 1968, so that no single fault in components, units, channels, or sensors will prevent ESFs operation.

The wiring is installed so that no single fault or failure, including either an open or shorted circuit, will negate minimum ESFs operation. Wiring for redundant circuits is protected and routed so that damage to any one path will not prevent minimum ESFs action. Sensors are piped so that blockage or failure of any one connection does not prevent ESFs operation.

The detailed design incorporates the following characteristics in order to counteract faults resulting in loss of power:

a. All redundant components are powered from separate busses;
b. The reliability of the 125 Volt DC and 120 Volt AC vital power busses used, as discussed in detail in Chapter 8;
c. The reliability of the 4160 Volt and 480 Volt systems, as discussed in Chapter 8;
d. The starting and loading of diesel generators, as described in Chapter 8; and,
e. Whenever practical, components of the ESFs system assume on loss of power the position called for under emergency conditions.

There are no ESFs instrumentation transmitters for which the trip setpoints are within 5% of the high or low end of the calibrated range or within 5% of the overall instrument design range.

The following features of the ESFAS cabinets reduce the potential for inadvertent actuation signals.

a. Maintenance Bypass - Each sensor cabinet has a lower front panel installed which contains 20 keylock/indicating light sets (some are spare) along with one bypass module. Each sensor module has its own switch and light set which, when placed in the bypass position, turns on the switch light, a light on the bypass module, and a remote alarm in the Control Room.

   This bypass feature essentially disconnects a sensor module bistable from its isolation module and provides a continuous on (no trip) signal to an isolation module. The isolation module changes the coincidence logic from two-out-of-four to two-out-of-three (three-out-of-three for the pressurizer and steam generator block permissive) because the bypassed channel's functions logic input cannot change to a trip signal input.

   The maintenance bypass allows maintenance or testing of a sensor module or its input loop without creating a sensor module trip signal to the coincidence logic. A trip signal input to the bypassed module is indicated by a trip light on the bypassed module.

b. Isolation Module Fault Indication - Each isolation module contains 15 subchannel optical isolators, each having a voltage comparator to detect an optical isolator output of 3 Volt DC or greater, which is possible with an actual trip signal or a faulty isolator. If the isolator output is greater than 3 Volt DC, the indicating light enables personnel to determine if an optical isolator is faulty.
c. Actuation Channel Sequenced Power Supplies - The actuation channels each contain a 28 Volt DC power supply for the actuation relays and redundant 15 Volt DC power supplies for the logic modules. The 28 Volt DC power supply is interlocked with the 15 Volt DC power supplies to ensure that the logic modules are energized before the relays, or deenergized after the relays. This prevents having power available to the relays with the logic modules in a potentially unstable state which could result in inadvertent equipment actuation. Using redundant 15 Volt DC power supplies accomplishes the same function in the event of a failure of one power supply.

7.3.6 MANUAL TESTING FEATURES

7.3.6.1 Bistable Trip Test

Each bistable has built-in provisions for testing bistable operation. An adjustable voltage source is applied when a local pushbutton is depressed. While initiating the test, the process variable input to the bistable is not interrupted. Local indicating lights and Control Room annunicators verify proper operation.

7.3.6.2 Testing of Refueling Water Tank Level Switches

The water level in the float chamber is lowered manually for each channel individually. Local indicating lights and Control Room annunciators verify proper operation.

7.3.6.3 Actuation Channel Trip Test

Each coincidence two-out-of-four matrix includes a local independent pushbutton. The test with simultaneous presence of a sensor channel trip causes an output of the associated coincidence matrix and trips the actuation channel logic.

A light is provided within the matrix for indication of each bistable trip test. The test pushbutton is initiated with each sensor channel individually, and in each case actuation channel output is observed both by remote annunciation in the Control Room and local indication. The combination of the testing procedure and the arrangement of the matrix provides for overlapping and ensures that a protective action will occur if any combination of two sensor channels simultaneously trip. For further information see the two-out-of-four matrix expansion as shown on Figures 7-10 and 7-22.

Except in the cases of actuation subchannels SIAS No. 5, CIS No. 5, CSAS No. 3, and SGIS No. 1, the actuation channel trip test as described above can be performed during normal plant operation for each actuation subchannel on an individual basis. The performance of the equipment associated with each subchannel is observed to assure proper operation. Equipment associated with subchannels SIAS No. 5, CIS No. 5, and SGIS No. 1 is to be tested during plant shutdown (Section 7.3.7).

7.3.6.4 Undervoltage, Blocking, and Sequencing Tests

Each undervoltage sensor channel may be tested individually by the interruption of the potential transformer connection to the undervoltage relays. This is accomplished by depressing a pushbutton in the actuation system cabinets. Proper operation is verified by local indicating lights and Control Room annunciators. Each coincidence undervoltage two-out-of-four matrix and the associated function may be tested in the same manner as described in Section 7.3.6.3. Four undervoltage matrices are provided, three of which can be tested on an individual basis during normal plant operation. Proper function is verified by actual opening of the appropriate circuit breakers and starting of the diesel generators. The matrix, which opens the 4160 Volt feed circuit breakers (through which the normal auxiliary power source feeds the 4160 Volt ESFs bus), can be tested during plant shutdown. The two matrices which shed various loads are not tested during normal operation in order to minimize system transients and risk.

Sequential actuation system blocking can be tested by depression of local pushbuttons simulating open 4160 Volt feeder circuit breakers and by initiation of the appropriate undervoltage matrix. The LOCI Sequencer is tested in conjunction

with the sequential actuation system blocking and proper operation is observed by local indicating lights and Control Room annunciators.

In order to initiate the LOCI Sequencer, the diesel generator circuit breaker may be closed after synchronization to the normal auxiliary power supply, or circuit breaker closure may be simulated by depressing the test pushbutton located on the ESFs cabinet. After initiation of SIAS Subchannel No. 10, the LOCI Sequencer is started. The shutdown sequencer is initiated when the diesel generator circuit breaker closes without SIAS and can be tested at any time. Proper starting of motors can be verified to assure operability.

7.3.7 PORTIONS NOT TESTED AT POWER

Those portions of the ESFAS that cannot be completely tested with the reactor at power are listed below:

	Unit 1	Unit 2
RCP	1-CV-505	2-CV-505
Seal Bleedoff Isolation Valves	1-CV-506	2-CV-506
SRW Isolation Valves	1-CV-1600	2-CV-1600
SRW Isolation Valves	1-CV-1637	2-CV-1637
SRW Isolation Valves	1-CV-1638	2-CV-1638
SRW Isolation Valves	1-CV-1639	2-CV-1639
Volume Control Tank Discharge Valves	1-MOV-501	2-MOV-501
Letdown Stop Valves	1-CV-515	2-CV-515
Letdown Stop Valves	1-CV-516	2-CV-516
CCW to Reactor Coolant Pumps	1-CV-3832	2-CV-3832
CCW from RCPs	1-CV-3833	2-CV-3833
MSIVs	1-CV-4043	2-CV-4043
MSIVs	1-CV-4048	2-CV-4048
Feedwater Isolation Valves	1-MOV-4516	2-MOV-4516
Feedwater Isolation Valves	1-MOV-4517	2-MOV-4517
Instrument Air Containment Isolation Valves	1-MOV-2080	2-MOV-2080
Heater Drain Pumps	11, 12	21, 22
Main Feedwater Pump	11, 12	21, 22
Condensate Booster Pumps	11, 12, 13	21, 22, 23
Mini Flow Return to RWT	1-MOV-659 1-MOV-660	2-MOV-659 2-MOV-660

The SRW Isolation Valves, 1(2) CV-1600, 1(2) CV-1637, 1(2) CV-1638, and 1(2) CV-1639, supply cooling water to the main turbine auxiliary systems (i.e., turbine lube oil coolers, electrohydraulic control (EHC) oil coolers, generator exciter coolers, generator hydrogen and stator liquid coolers, etc.). The turbine manufacturer recommends against securing cooling water to the turbine auxiliary systems while the unit is on line. The SRW isolation valves can be tested prior to putting the unit on line or during the periodic valve tightness tests for the turbine main stop and control valves.

The volume control tank discharge valve and the letdown stop valves cannot be exercised during normal operation because of the thermal transient imposed on the regenerative heat exchanger. These valves can be tested when the RCS is cooled down for refueling.

The CCW control valves cannot be periodically cycled during normal operation as this would impose undue transients to the RCP seals. These valves can be tested when the reactor is cooled down and pumps secured.

Though the MSIVs cannot be fully tested during normal operation, provisions have been incorporated which allows partial stroking of the valves from the Control Room. These valves can be fully tested when the reactor is shutdown.

The safety injection tank isolation valves cannot be tested during reactor operation since the valves must be open at all times while the plant is in Modes 1 through 3. The valves can be tested when the reactor is shutdown and cooled down.

	Unit 1	Unit 2
Safety Injection Tank	1-MOV-614	2-MOV-614
Isolation Valves	1-MOV-624	2-MOV-624
	1-MOV-634	2-MOV-634
	1-MOV-644	2-MOV-644

The RCP seal bleedoff isolation valves cannot be tested at power since the pumps must be shut off during the test. The valves can be tested when the reactor is shut down.

The feedwater isolation valves, heater drain pump, main feedwater pumps, and condensate booster pumps cannot be tested at power. The reactor must be shut down to test these components.

Instrument Air Containment Isolation valves cannot be tested during reactor operation since the valves must be open to supply air to components necessary to plant operation in Modes 1 through 3. The valves can be tested when the reactor is shutdown.

Miniflow isolation valves (MOV-659, MOV-660) to RWT cannot be tested during reactor operation since the valves are to remain open with the power removed during Modes 1, 2, and 3 with the pressurizer pressure ≥ 1750 psia. The valves can be tested when the reactor is shutdown.

TABLE 7-3

LIST OF CLASS I (SEISMIC) EQUIPMENT SUBJECTED TO DYNAMIC LOAD TEST

EQUIPMENT SPECIFICATION NUMBER	DESCRIPTION
E- 6	480 Volts Load Centers
E- 7	480 Volts Motor Control Centers
E- 19	Station Control Battery
E- 20	Station Control Battery Chargers
E- 21	Inverters for Vital AC System
E- 27	125 Volts DC Busses, Unit Control Panels
E- 96	4 kV Diesel Generator Isolating Switches
M-191	Controls for Class I (Seismic) HVAC System
M-237	Radiation Monitoring System
M-350	Miscellaneous Panel Mounted Indicators
M-363	Level Switches
M-369	Control System and Field Transmitters
M-370	ESFAS

TABLE 7-4

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
1.	Manual		
	a.	SIAS	
		Safety Injection [Emergency Core Cooling System (ECCS)]	Not Applicable
	b.	CSAS	
		Containment Spray	Not Applicable
	c.	CIS	
		Containment Isolation	Not Applicable
	d.	RAS	
		Containment Sump Recirculation	Not Applicable
	e.	AFAS	
		Auxiliary Feedwater Initiation	Not Applicable
2.	Pressurizer Pressure-Low		
	a.	Safety Injection (ECCS)	≤ 30[a]/30[b]
		(Low Pressure Safety Injection Flow	≤ 45)
3.	Containment Pressure-High		
	a.	Safety Injection (ECCS)	≤ 30[a]/30[b]
		(Low Pressure Safety Injection Flow	≤ 45)
	b.	Containment Isolation	≤ 30
	c.	Containment Fan Coolers	≤ 35[a]/10[b]
4.	Containment Pressure-High		
	a.	Containment Spray Isolation Valve	≤ 60.9[a]/60.9[b]
	b.	Containment Spray Pump	≤ 28.9 [a]/18.9 [b]
5.	Containment Radiation-High		
	a.	Containment Purge Valves Isolation	≤ 15
6.	Steam Generator Pressure-Low		
	a.	Main Steam Isolation	≤ 6.9[e]
	b.	Feedwater Isolation	≤ 65
7.	Refueling Water Tank-Low		
	a.	Containment Sump Recirculation	≤ 150
8.	Loss of Power		
	a.	4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≤ 2.2[c]
	b.	4.16 kV Emergency Bus Undervoltage (Steady State Undervoltage)	≤ 104.5
	c.	4.16 kV Emergency Bus Undervoltage (Transient Undervoltage)	≤ 8.4[c]
9.	Steam Generator Level-Low		
	a.	Steam Driven AFW Pump	≤ 180
	b.	Motor Driven AFW Pump	≤ 180
	c.	Isolate Steam Generator Blowdown	≤ 35
10.	Steam Generator ΔP – High		
	a.	Auxiliary Feedwater Isolation	≤ 20.0

TABLE 7-4

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	**RESPONSE TIME IN SECONDS**
11. Penetration Room Pressure – High	
a. CVCS Isolation	≤ 9

[a] Diesel generator starting and sequence loading delays included.

[b] Diesel generator starting and sequence loading delays not included. Offsite power available.

[c] Response time measured from the incident of the undervoltage condition to the diesel generator start signal.

[e] Response time accounts for isolation under accident steam flow conditions. In accordance with Section 14.14.2, "Discussion of Main Steam Isolation Valve Testing," ESF response time under no-flow test conditions is ≤ 6.1 seconds.

7.4 REGULATING SYSTEMS

7.4.1 REACTOR REGULATING SYSTEM

7.4.1.1 General

The Reactor Regulating System (Figure 7-11) provides control signals which are used to provide a steam dump program and a pressurizer level setpoint program.

7.4.1.2 System Description

A block diagram of the Reactor Regulating System is shown in Figure 7-11. The system consists of:

a. Steam Dump Program Function Generator;
b. Level Setpoint Function Generator;
c. T_{ref} Function Generator;
d. T_{avg} - T_{ref} Summer;
e. T_{avg} - T_{ref} Stability Compensation Unit;
f. Turbine Power - Reactor Power Stability Compensation Unit; and,
g. Pressurizer Pressure - Pressurizer Pressure Setpoint - Stability Compensation Unit.

The system includes the following inputs to each channel:

a. Loop 1 T_{hot}, Loop 2 T_{hot};
b. Loop 1 T_{cold}, Loop 2 T_{cold};
c. Pressurizer pressure (disconnected);
d. Neutron flux; and,
e. First-stage turbine pressure.

The system develops the following outputs from each channel:

a. T_{ref} and T_{avg} signals to recorders;
b. Automatic CEA withdrawal prohibit signals to the CEDM control systems (disconnected); and,
c. Deviation alarms for T_{avg} - T_{ref}

The Reactor Regulating System is used to provide a signal for pressurizer level setpoint, steam dump demand, and steam dump quick-opening. The operator has the ability to select between systems x or y with a selector switch. Each system is separate and independent of the other. The system functions to give controlling signals as the input parameters change. With a change in power level, the turbine first stage pressure will increase linearly with load. In each channel, a temperature programmer establishes the desired reactor coolant average temperature (T_{ref}), based on a power reference signal from first-stage turbine pressure. This T_{ref} signal is summed with the T_{avg} signal to provide a signal which represents the error between the actual temperature and the programmed temperature (E_t). If this deviation between T_{avg} and T_{ref} should become too high, an alarm will be annunciated. This T_{avg} signal is used to provide a programmed level set-point for the pressurizer. The operational level of the pressurizer is programmed to increase with an increase in T_{avg}. This is done to accommodate plant load changes and transients in order to minimize changes in RCS volume.

The T_{avg} signal is also used to provide an analog output for steam dump demand and quick-opening (Figure 7-15). As T_{avg} increases, the signal to the steam dump

control valve increases. The signal is proportional to the quantity (T_{avg}-532°F). Should reactor power as determined by T_{avg} be in excess of predetermined power level as determined by T_{avg} prior to a trip, the steam dump quick-opening override bistable will cause quick opening of the steam dump and bypass valves at the time of the trip.

The operator may assume remote manual control of the atmospheric steam dumps from the alternate shutdown panel by aligning four hand transfer valves.

7.4.2 CONTROL ELEMENT DRIVE SYSTEM

7.4.2.1 Design Basis

The reactor is controlled by reactivity adjustments with CEAs and with boric acid dissolved in the reactor coolant. Rapid changes in reactivity are compensated for or initiated by CEA movement. Long-term variations in reactivity due to fuel burnup and fission product concentration changes are controlled by adjusting the boric acid concentration. Since this rate of addition produces slow changes in reactor power level, operator action suffices to control the boron concentration change. The CEA groups provide a hot shutdown margin of at least 1% reactivity, even if the most reactive CEA is stuck out of the core. Interlocks require that the shutdown CEA group is within a predetermined permitted range before regulating groups can be withdrawn. An alarm and a motion inhibit are provided when further insertion of the regulating group of CEAs would reduce the amount of effective shutdown reactivity in the CEAs below specified limits.

Control element assembly movement is effected by the CEDMs (Chapter 3.0). The CEDS transmits signals from the CEDS control panel to the coil power programmers, which develop the pulses for magnetic jack operation.

Control element assembly withdrawal will be prevented when a high power pretrip, high rate-of-change-of-power pretrip or thermal margin/low-pressure pretrip is present. All CEA motion is inhibited when CEA malposition limits are reached.

7.4.2.2 System Description

A block diagram of the CEDS is shown in Figure 7-16.

The CEDS control panel is a selection panel. Three types of selections are made by this panel; control mode, CEA group, and the individual CEA within each group. All selections are made by pressing the appropriate pushbutton switch. Upon selection, the switch will light and remain lit and selected until another selection within its scope is made. To operate, one selects one mode, one group, or one individual CEA if in manual individual mode. If not in manual individual then no individual CEA selection is made and only the group selection need be made. Electrical interlocks are incorporated in each of these scopes of selection. This permits only one selection to be made in each scope. A new selection within any scope automatically cancels the previous selection.

There are four different operable modes of control of CEAs. Three of these modes, manual individual, manual group, and OFF apply to both types of CEAs.

All four operable modes, manual individual, manual group, manual sequential, and off, apply only to the regulating CEAs. For all cases manual control is used. Automatic control of the regulating CEAs by the Reactor Regulating System is not used.

The upper and lower CEA group stops for the regulating and shutdown groups are provided by the CEA supervisory function of the plant computer to prevent the reactor from reaching undersirable conditions.

The CEDS contains design features that ensure the following actions:

a. Insertion of the regulating CEAs within a predetermined permitted range before the shutdown groups are inserted;
b. Simultaneous withdrawal of no more than two groups of CEAs;
c. Proper sequential withdrawal of CEAs; and,
d. Withdrawal of the shutdown CEAs to a predetermined permitted range before the regulating groups are withdrawn.

Three lines of defense are utilized to ensure that safety limits are not exceeded. First, the reactor is operated under strict administrative controls which dictate the proper CEA movement. Second, alarms are provided to warn the operator if CEA movement is improper. The third line of defense consists of the functions and design features described below.

The plant computer CEA supervisory function operates CEA permissive contacts that feed the CEDM logic system. This design feature determines which group or groups of regulating CEAs will be moved in the manual sequential mode. The computer also generates alarms if its logic detects regulating CEA groups out-of-sequence (OOS), regulating group overlap violation, deviation of individual CEA positions within their group, or violation of pre-power and PDILs.

The setpoints of these alarms are chosen such that the operator is given sufficient time to take corrective action before a safety limit is reached without alarming for normally-occurring conditions.

If an equipment failure or operator error should cause any of these alarm conditions to be reached, another alarm is also received from the reed switch position indication system, described in Section 7.5.3.3. This second system, diverse in nature from the plant computer CEA supervisory function, provides the operator with continuous indication of the position of each individual CEA and provides alarms redundant to those supplied by the plant computer. In addition, this system provides actuation signals to the CEA motion inhibit circuitry which stops all manual CEA motion when an alarm setpoint is reached.

The actuator signals are computed from CEA position information from the reed switch position transducers for each CEA and a reactor power signal (for PDIL only). The actuation signals are contact openings that fail open upon loss of power. The actuation signals are "anded" with mode OFF and the composite signal is sent to the lift coil power switch for each CEA from contact multiplying relays. This signal to the lift coil power switch will stop any power from being applied to the lift coil, so that regardless of CEA control system motion demand, CEA motion is inhibited.

The CEA misalignment, group sequencing, group overlap, and total insertion that can occur at the alarm and motion inhibit limits discussed above are factored into the thermal margin and axial flux offset trips. Furthermore, no single failure of the CEDS, other than a CEA drop incident, will cause any of these CEA limits to be violated. As a result, no anticipated operational occurrence other than a CEA drop incident can result in any of these CEA limits being violated. In the case of the

CEA drop, the analysis presented in Section 14.11 demonstrates that the resultant transient does not result in a violation of the fuel design limits and does not require a reactor trip.

Occasional operational failures of CEA position indication channels will be dealt with under the guidance of Generic Letter 91-18.

Additional prohibits are also provided to prohibit regulating group withdrawal and prevent the reactor from reaching undesirable conditions. These interlocks are summarized in Table 7-5.

Control element assembly speed is a function of the coil power programmer cycle speed and the CEA control system group speed setting.

The coil power programmer that sequences power signals to the magnetic jack coils of an individual CEA has an upper operating setting of 30" per minute for regulating CEAs, and 20" per minute for shutdown CEAs. This maximum speed would result only from a continuous demand for withdrawal from the CEA control system.

A continuous withdrawal signal from the CEA control system would result only during abnormal operating circumstances. The average speed of CEAs within the group is determined by the speed setting of the group programmer in the CEA control system. The upper bound of the speed setting of the group programmer is 50" per minute, but the normal operating speed setting of the group programmer is the same as that of the individual CEA coil power programmer. Maximum CEA speed is determined by the setting of the individual CEA coil power programmer, and this speed cannot be increased by a setting of the group programmer.

7.4.2.3 System Operation

The CEAs are divided into the following groups:

a. Shutdown: three groups; and,
b. Regulating: five groups;

Each CEA remains stationary except when a raise or lower signal is present. In response to a signal, the regulating CEAs move at a speed of 30" per minute. The shutdown CEAs move at a speed of 20" per minute.

The CEA position setpoints are shown on Figure 7-12.

The shutdown CEAs may be moved in the manual control mode only, with either individual or group movement. Movement of more than one shutdown group at any time is not possible. The shutdown CEAs must be withdrawn to within a predetermined permissible range of the upper limit before regulating group withdrawal is possible. A limit prevents group insertion of shutdown CEAs unless all regulating CEAs are within a predetermined permissible range of the lower limit.

Regulating CEAs may be moved in manual control by manual group or sequential group movement. Individual CEAs may be moved in manual control. Sequential group movement functions such that when the moving group reaches a programmed low (high) position, the next group begins inserting (withdrawing); the initial group stops upon reaching its lower (upper) limit. This procedure, applied successively to all regulating groups, allows a smooth and continuous rate-of-change of reactivity.

Under sequential group control, when the regulating groups reach the "prepower-dependent" insertion alarm point, this condition is annunciated. If sequential group insertion is continued, a "power-dependent" alarm point limit is reached, and a second alarm is initiated. These two programmed limits may be adjusted during the life of the plant and are provided to aid the operator in assuring adequate shutdown margin, limiting the reactivity worth of an ejected CEA, and maintaining the radial peaking within the limits which are factored into the monitoring and protective systems.

All CEAs are prevented from being withdrawn if either a high power pretrip, rate-of-change of power pretrip or thermal margin/low-pressure pretrip condition exists. There is provision for manually bypassing the CEA motion inhibit circuitry at the CEDS operator's console. The manual bypass requires a minimum of two operator actions to accomplish the bypass. The group to be bypassed must be manually selected by the operator by depressing the pushbutton for that group. In addition to selecting the group to be bypassed, the operator must also depress the "CEA motion inhibit bypass" button to accomplish the bypass function. This pushbutton must be held depressed by the operator during the bypass operation, as release of this pushbutton will cause the bypass to be removed. Actuation of the "CEA motion inhibit bypass" pushbutton will cause an alarm at the main control board, thereby alerting the operator of the bypass.

7.4.3 REACTOR COOLANT PRESSURE REGULATING SYSTEM

7.4.3.1 Design Basis

The reactor coolant pressure regulating system maintains system pressure within specified limits by the use of pressurizer heaters and spray valves. Pressurizer pressure sensors provide input to the system.

A high pressurizer pressure functions to open the pressurizer spray valves on a proportional basis, thereby reducing pressure. A low pressurizer pressure functions to energize heaters on a proportional or group basis to increase pressure. A high pressurizer level energizes the backup heaters in anticipation of a low-pressure transient; a low pressurizer water level deenergizes all heaters, for heater protection.

Two channels of control are provided and the controlling channel is selected by a switch. Manual control of the heaters and spray may be selected at any time.

7.4.3.2 System Design and Operation

Two independent pressure channels provide suppressed range (1500 to 2500 psia) signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function. During normal operation, a small group of heaters is proportionally controlled to maintain operating pressure. If the pressure falls below the proportional band, all of the backup heaters are energized. Above the normal operating range, the spray valves are proportionally opened to increase the spray flow rate as pressure rises. A small, continuous-spray flow is maintained through the spray lines at all times to keep the lines warm, to reduce thermal shock as the control valves open, and to ensure that the boric acid concentration in the coolant loops and pressurizer is in equilibrium.

Outputs from the two pressure control channels are recorded in the Control Room and provide independent high and low alarms.

The control and alarm pressure setpoints are shown in Figure 7-13.

7.4.3.3 System Evaluation

Two independent channels are available for automatically regulating the pressurizer heaters and spray valves. Either channel may be used to control the pressure in the system, and the output from both channels is recorded in the Control Room. Independent high- and low-pressure alarms are provided. Control of two banks of back-up heaters can be assumed at the alternate shutdown panel.

7.4.4 PRESSURIZER LEVEL REGULATING SYSTEM

7.4.4.1 Design Basis

Pressurizer level is maintained by the action of the chemical and volume control system (Section 9.1). The level setpoint is programmed as a function of coolant average temperature (T_{avg}). A low pressurizer level signal functions to reduce letdown flow proportionally and to start the nonoperating charging pumps. A high level indication functions to increase letdown flow proportionally by opening the letdown control valves and stopping all but one charging pump. There are two independent automatic control channels with channel selection by means of a manual control switch. Automatic control is normally used during operation, but manual control may be utilized at any time.

7.4.4.2 System Design and Operation

The operating level of the pressurizer is programmed as a function of power to accommodate plant load changes and transients to minimize the changes in RCS volume (Figure 4-10).

The level programmer establishes a program level which is directly proportional to coolant average temperature over the operating range of T_{avg}. The average temperature signal used by the level programmer is the signal used by the Reactor Regulating System.

The level controller compares the measured and programmed level signals and generates a proportional signal for regulating the letdown control valves. In addition, the level controller functions to start or stop additional charging pumps at low or high level setpoints. The outputs of either of two automatic control channels may be selected by the operator for level control in addition to manual control.

Two independent level channels provide pressurizer level signals for two specific functions:

a. A low level signal from either channel deenergizes all heaters; and,
b. A high level signal from the controlling channel energizes the backup heaters.

7.4.4.3 System Evaluation

Two separate and redundant level control systems are provided. The controllers are located in the Control Room. Both automatic and manual control of level is provided. Three charging pumps and two letdown control valves provide redundant means of increasing or decreasing reactor coolant inventory. The variable pressurizer level control program maintains the proper coolant inventory by means of discharge or addition as required during plant load changes.

7.4.5 FEEDWATER REGULATING SYSTEM

7.4.5.1 Design Basis

The feedwater regulating system maintains SG downcomer level within acceptable limits by regulating the speed of the feedwater pumps and positioning the feedwater regulating valves, which control the feedwater to each SG (Figure 7-14A and Figure 7-14B). Steam flow, feedwater flow, and downcomer level are used in a three-element control configuration on each SG to maintain the desired level during steady state and transient operation above approximately 18% of main feedwater flow.

Manual control of feedwater flow may be selected by the operator at any time. In the event of a reactor trip, feedwater flow is automatically ramped down to approximately 3.8% of full load feedwater flow; this is approximately the flow required after a trip from full power to remove decay heat such that AFAS should not actuate and the time to RCS cooldown, without operator action, is maximized. As downcomer level approaches its setpoint, automatic control of feedwater flow will be resumed to maintain downcomer level.

During abnormal conditions when a feedwater regulating valve malfunctions, the valve may be mechanically pinned to the manual operator in order to support safe operation or a controlled shutdown of the unit. In this case it may be necessary to manually secure main feedwater flow after a reactor or turbine trip if feedwater flow is excessive.

Following the identification of feedwater deficiencies in original industry MSLB accident analysis, the NRC issued IE Bulletin 80-04. This bulletin required all licensees of PWRs to review the containment pressure response analysis to determine if the potential for containment overpressure for MSLB inside containment included the impact of runout flow from the AFW System and the impact of other energy sources, such as continuation of feedwater or condensate flow. As a result of this bulletin, we installed a system where SGIS or CSAS will produce a trip of the main feedwater pumps, the heater drain pumps, and the condensate booster pumps, as well as closure of the main steam isolation and main feed isolation valves. This feature limits the peak containment pressure in the event of a steam line break inside containment by ensuring a prompt reduction in main feedwater flow to the affected SG.

Below approximately 18% of main feedwater flow, the feedwater controller automatically maintains SG downcomer level using the following inputs: SG level, wide range level and feedwater temperature. The feedwater controller uses these inputs to control the position of the feedwater main and bypass valve with the feedwater pumps operating to maintain a differential pressure across the feedwater regulating valves.

The low/high power mode is based upon feedwater flow being greater than a predefined setpoint which will be above the point where reliable steam and feedwater flow signals are available for use by the system.

7.4.5.2 System Design and Operation

A block diagram of the feedwater control system for each unit is shown in Figure 7-14A and Figure 7-14B.

The low/high power mode is based upon the feedwater flow being greater than a predefined setpoint which will be above the point where reliable steam and feedwater flow signals are available for use by the system.

Automatic Mode

The two SGs are operated in parallel. Each SG has a three-element controller using feedwater flow, steam flow, and downcomer level as inputs for level control above approximately 18% of main feedwater flow. The speed of the main feed pump turbines is controlled by pump speed demand, which is a function of main feedwater valve position demand, and is controlled at a speed which produces a relatively constant differential pressure across the main feedwater valve. Below approximately 18% of main feedwater flow, the position of the bypass feedwater valve is maintained using SG level, wide range SG level, and feedwater temperature. The speed of the main feed pump turbines is fixed at a minimum speed. At approximately 18% of main feedwater flow, the control is automatically transferred from low power control to the high power three-element control.

Manual speed control of each of the feedwater pump turbines can be accomplished from the Control Room.

Upon reactor trip, the main feedwater control valves are automatically closed and the feedwater bypass valves are automatically opened to approximately 3.8% of main feedwater valve flow, while the feedwater pumps are run back to the minimum speed.

Manual Mode

Manual control of the feedwater regulating system may be selected at any power level. When in manual control, the operator in the Control Room can:

a. Position each feedwater regulator control valve
b. Open or close each feedwater stop valve
c. Position each feedwater bypass regulating valve
d. Control speed of feedwater pumps.

Auxiliary Feedwater Flow

The operator can at any time control operation of the AFW pumps and position each AFW regulating valve. Remote control of AFW is available if control air is available, whereas if control air is lost, manual control would be accomplished at the turbine drive governor or at the regulating valve. Control of AFW may be assumed at the alternate shutdown panel provided air is available. Air from accumulators allows operation of the regulating valves for two hours after loss of instrument air.

7.4.5.3 System Evaluation

Conventional three-element feedwater control is used with fail-as-is feedwater control valves. Manual override of the automatic control is always available. Remote manual bypass valves and manual feedwater stop valves provide backup for feedwater valve failure.

7.4.6 STEAM DUMP AND TURBINE BYPASS SYSTEM

7.4.6.1 Design Basis

The steam dump system is designed to provide a means of dissipating excess NSSS stored energy and sensible heat following a turbine trip without lifting the safety valves. Steam is discharged from the main steam lines to the atmosphere via the steam dump valves and to the condenser via the turbine bypass valves. The steam dump and bypass valves are sized to prevent opening of the SG safety valves following a turbine trip at full load. The steam flow is regulated by the dump

and bypass valves in response to T_{avg} and secondary pressure signals, respectively.

A block diagram of the steam dump and bypass system is shown in Figure 7-15.

Inputs to the system are T_{avg}, turbine trip signal, condenser vacuum, and main steam line pressure.

7.4.6.2 System Design and Operation

Steam Dump Demand Signal

The steam dump demand program is shown in Figure 7-15, Sheet 3. The purpose of this program is to modulate the steam dump valves as function of T_{avg}.

At T_{α}, the steam dump valves are fully closed.

At T_{Υ}, steam dump valves are fully open.

A hysteresis is included in the program which requires T_{avg} to reach T_{β} in order to reopen the steam dump valves. This minimizes valve damage that would result from the steam dump valves operating too close to the shut seat.

Pressure Control Signal

The pressure control program is shown in Figure 7-15, Sheet 3. The purpose of the program is to modulate the turbine bypass valves as a function of main steam header pressure. The controller normally operates in automatic, controlling at 900 psi. During turbine valve testing, and following detection of a turbine valve malfunction, the pressure setpoint is adjusted to just above SG pressure, and than returned to 900 psia after completion. During initial turbine loading, the controller is set to modulate. The controller setpoint can be adjusted in automatic as conditions may require, or it can be operated in manual. The turbine bypass valves operate sequentially as controller output increases. They are also interlocked with condenser vacuum such that they will remain shut on loss of vacuum.

Steam Dump Quick Opening

Should a turbine trip occur when the plant is operating at a high temperature and power, the stored energy in the primary coolant system may be great enough that the steam dump area demand signal alone cannot open the dump valves fast enough to prevent lifting of the safety valves. Assuming condenser vacuum was not lost during the turbine trip, the RRS provides a contact closure output to the steam dump and bypass control system to quick open the valves following a turbine trip when T_{avg} is greater than a predetermined value (T_{Υ}). If the main condenser is unavailable due to loss of condenser vacuum, only the steam dump valves open following the turbine trip. When T_{avg} is reduced to $< T_{\Upsilon}$, the "quick open" signal is removed. Assuming the main condenser is available, both the turbine bypass and steam dump valves will continue to dump steam until T_{avg} is reduced to (T_{α}), at which point the steam dump valves close. A portion of the turbine bypass valves will modulate based on steam header pressure, as long as steam header pressure is greater than 900 psia. If T_{avg} increases to T_{β} the steam dump valves will again open. If the main condenser is unavailable (turbine bypass valves closed), the steam dump valves will control T_{avg} between T_{α} and T_{β}.

7.4.6.3 System Evaluation

The steam dump valves can be operated from either the Control Room or, aligning four hand transfer valves from the auxiliary shutdown panel. Automatic or manual control is provided for both turbine bypass and steam dump valves at their respective Control Room stations, and manual control of the steam dumps is provided at the auxiliary shutdown panel.

The total capacity of the system is sufficient to prevent lifting of the secondary steam safety valves following a simultaneous reactor-turbine trip at full power. The respective capacities of the steam dump and turbine bypass valves are 5% and 40% of total system steam flow at full power. The capacity of the dump valves is sufficient for plant cooldown in the event the condenser is not available.

Excessive cooldown of the RCS by the dump valves, when in automatic control is prevented by a narrow-range temperature signal. This signal has a minimum output corresponding to 535°F, at which point the dump flow will be zero.

The turbine bypass system will limit the maximum steam pressure to approximately 905 psia during hot standby when the condenser is available.

Loss of power in the steam dump and bypass system will cause the valves to fail closed or remain closed. Depending on the failure, a component failure in the system will cause valves to either fail closed, remain closed, fail open, or remain open. The safety analyses for the excess load and loss of load events did not take credit for this system (Sections 14.4 and 14.5, respectively).

The steam dump and bypass system is powered from 120 Volt AC instrument Busses No. 11 and/or No. 12. Solenoids in the system are powered from 125 Volt DC Bus No. 11.

7.4.7 TURBINE GENERATOR CONTROL SYSTEM

7.4.7.1 Unit 1 Turbine Control System (General Electric)

Unit 1 turbine generator control system (or electrohydraulic control system) is designed to control steam flow to the turbine.

The control system consists of the following:

A. Steam Control Valves

Four main stop valves are located in the main steam piping ahead of the control valves. The four control valves are mounted in a line on a common valve chest. The valve chest is separated from the turbine, and individual steam leads from the valve chest are provided for each control valve to inlet bowl sections of the high-pressure turbine. Steam leaving the high-pressure turbine enters the moisture separator reheaters (MSRs). Six combined reheat valves are provided downstream of the MSRs, one in each line supplying steam to the low-pressure turbines. The combined valves include an intercept valve and an intermediate stop valve.

The valve functions are described as follows:

Main Stop Valves: (four valves)

The main stop valves' primary function is to quickly shut off steam flow to the turbine under emergency conditions. One valve is provided with a

bypass for slow warming and for pressurizing below the seat of the stop valves for opening.

Control Valves: (four valves)

The control valves control flow to the high-pressure turbine during normal operation.

Intercept Valves: (six valves, two per low pressure turbine)

The intercept valves are part of the combined reheat valves. These valves control steam flow from the moisture separate-reheaters to the three low pressure turbines.

Intermediate Stop Valves: (six valves, two per low pressure turbine)

The intermediate stop valves shut off steam flow from the MSRs to the three low pressure turbines under emergency conditions.

B. Hydraulic Trip System

The hydraulic trip system consists of duplex electrical trip devices ETD-A and ETD-B, either of which can dump the emergency trip oil to drain. ETD-A and ETD-B are identical hydraulic trip manifold solenoid valve assemblies. Geared and ganged inlet and outlet isolation valves allow either ETD-A or ETD-B to be isolated for service on line. However, both are in service during normal conditions. Each ETD has 3 solenoid valves connected in a 2 out of 3 (2/3) arrangement to allow for redundant operation with the testing of each solenoid valve one at a time. There are position indication switches, lights and pressure transmitters which provide the control system with diagnostic information for the ETDs. The ETDs receive high pressure hydraulic oil from the hydraulic power unit and puts out an emergency trip signal (ETS) pressure.

The ETS pressure is high when the trip system is reset and low (drain pressure) when the system is tripped causing the main stop, control, intercept and intermediate stop valves to close.

C. High-pressure Fluid System

The hydraulic power unit supplies high-pressure fluid directly to the control pacs on the steam valves for opening and closing the valves, and to the trip devices in the trip and overspeed protection circuits.

D. Electrical Control System

The electrical control system is a triple modular redundant (TMR) computer based system with triple voting I/O for all controlling devices.

1. Speed Control

The speed control produces the speed error signal that is determined by comparing the actual speed with the desired speed of the turbine at steady-state conditions, or the ramped desired speed during startup. Three speed probes and input circuits provide TMR redundancy for speed control. The system can continue to run with one of the probes in a failed state. During normal operation at rated speed, the speed error signal is essentially zero, regardless of load. The speed error signal continues to input to the load control unit.

2. Load Control Unit

   The prime purpose of the load control unit is to develop output signals to which steam flow for the control valves and intercept valves may be proportioned. These outputs are based on a proper combination of the speed error and load reference signal biased with the first stage pressure signal. The load reference signal is the desired load at rated speed and rated steam conditions and speed error signal is inputted from the speed control unit. When the generator is not on line, the load reference is, in effect, a speed vernier adjustment; it is, therefore, used for synchronizing the turbine.

3. Control Valve and Intercept Valve Flow Control Units

   The purpose of the valve flow control units is to produce the steam flows that are commanded by the load control unit. Due to the appreciable non-linear steam flow characteristic of the steam valve, the Mark VI turbine control software applies flow lift compensation curves.

4. Electrical Alarm and Trip System

   The electrical alarm and trip system provides the following electrical signals:

   a) Trip and reset signals to the hydraulic trip system;
   b) Alarm information to the Mark VI operator displays and to the main control room annunciators;
   c) Sequence of Event information to the Mark VI data logger.

Any trip signal will de-energize the six primary trip relays which will initiate the following redundant trip actions:

1. de-energizing the three voting solenoid valves of the ETD-A;
2. de-energizing the three voting solenoid valves of the ETD-B.

Either one or both ETD-A or ETD-B will trip the system depending on the position of the geared isolation valve. The following is a list of Unit 1 turbine trips (Figure 8-7, Sheet 2):

1. Front Standard trip buttons at turbine front standard
2. Master trip buttons on 1C02
3. High exhaust hood temperature
4. Exhaust hood low vacuum
5. Hydraulic pressure low trip
6. Bearing oil low pressure
7. Main shaft oil pump low pressure
8. MSR and MSR heater drain tank high level (11 or 12)
9. Loss of stator coolant
10. Axial thrust
11. Feedwater heater high level
12. SG high level (11 or 12) or reactor trip bus under-voltage
13. Emergency acceleration/deceleration
14. Emergency overspeed

15. Emergency speed compared to Primary speed difference
16. Primary overspeed
17. Primary acceleration/deceleration
18. Unit protection
19. Power load unbalance

Primary Overspeed Trip

Primary overspeed is provided by a TMR controller with 3 magnetic speed probes. If speed exceeds 110%, the six primary trip relays are de-energized to initiate a trip of ETD-A and ETD-B.

Emergency Overspeed Trip

A separate TMR turbine protection module with 3 magnetic speed probes is used to provide emergency overspeed protection. If speed exceeds 111.5%, the six emergency trip relays are de-energized to initiate a (redundant) trip of ETD-A and ETD-B similar to the primary trip.

A reactor trip results from a turbine generator trip only when the system is operating above 15% of full power.

7.4.7.2 Unit 2 Turbine Control System (Westinghouse)

Unit 2 turbine generator control system or electrohydraulic governor control system is designed to control steam flow to the turbine.

The electrohydraulic system contains:

A. Steam Control Valves

Steam enters at both ends of the steam chest through throttle valves to two individually-controlled governor valves. There are two steam chests, identical in construction, controlling steam to the high-pressure turbine, one being located on each side of the unit. Steam flows through the high-pressure turbine blading and then through the crossunder piping to the MSRs. Crossover pipes return the steam through the reheat stop and interceptor valves to the three low pressure turbines.

The valve functions are described as follows:

1. Throttle Valves: (four valves, two per steam chest)

The throttle valves control the steam flow to the high-pressure turbine during wide-range speed control. The governor valves remain open during control by the throttle valves unless there exists excessive speed error. Both valves will close when actuated by the trip system.

2. Governor Valves: (four valves, two per steam chest)

Prior to synchronizing, a transfer is made from throttle valve to governor valve control. Speed and/or load is then controlled only by the governor valves. The throttle valves are wide open in this mode of operation. Both valves will close when actuated by the trip system.

3. Reheat Stop Valves: (eight valves, two per MSR)

The reheat stop valves are provided in each steam line between the MSR and the interceptor valve. Its purpose is to provide an additional safety

device to prevent overspeeding of the turbine, should the interceptor valve fail to close when the overspeed trip mechanism operates.

4. Interceptor Valve: (six valves, two per low pressure turbine)

The interceptor valves are provided in each reheat steam line to limit flow of steam from the MSR to the low pressure turbine after a load rejection.

B. Hydraulic Trip System

The hydraulic trip system consists of two trip oil headers, the auto-stop trip oil header and the high-pressure fluid emergency trip oil header. The emergency trip valve is the interface between these headers. The functioning of the mechanical overspeed trip mechanism or protective trip device results in a decay of the auto-stop oil pressure. This pressure, acting on the diaphragm of the emergency trip valve, is overcome by the diaphragm's spring pressure allowing the valve to open. This allows all fluid in the high-pressure trip header to drain, causing the throttle, governor, reheat stop and intercept to close to shut down the turbine.

The protective trip devices include a low vacuum tripping device, a low bearing oil pressure trip, a thrust bearing trip and remote-controlled solenoid actuated trips (Figure 8-12, Sheet 2).

The protective trip devices trips are as follows:

1. Low Condenser Vacuum (2-PV-8259);
2. Thrust Bearing Oil Pressure (2-PV-8257);
3. Low Bearing Oil Pressure (2-PV-8258);
4. Master Trip Solenoids (2SV 8250, 2SV 8251).

The Master Trip Solenoid requires 125 Volt DC power to be available and energizes on the following conditions:

1. deleted
2. Manual Trip Switch (2HS 8250);
3. deleted
4. Unit Protection Trips;
5. Feedwater heater high level (2LS 1446, 2LS 1447, 2LS 1452, 2LS 1453, 2LS 1448, 2LS 1449, 2LS 1454, 2LS 1455, 2LS 1450, 2LS 1451, 2LS 1456, 2LS 1457);
6. SG Nos. 21, 22 high level;
7. Reactor Trip Bus Undervoltage;
8. Overspeed or loss of 125 Volt DC; and
9. Load rejection.

A reactor trip results from a turbine generator trip only when the system 48
above 15% power level.

In addition to the mechanical overspeed trip mechanism and the protective trip devices, there exists an emergency trip control block in the high-pressure fluid emergency trip oil header containing three solenoid valves. The three solenoids (2SV 8235, 8236, 8237) are arranged in parallel on the control block and are identified as to their function.

The main turbine electrohydraulic fluid emergency trip solenoid (2SV 8235) opens on auto stop low pressure at 45 psig. This drains the high-pressure trip fluid to close the governor, throttle, reheat stop and interceptor valves.

The main turbine overspeed protection control solenoids (2SV 8236, 2SV 8237) are energized to open if the turbine speed exceeds 103% of rated speed. This will dump the high-pressure trip fluid from the governor and interceptor valves causing the valves to close. The closing of the governor valves and interceptor valves in response to a load loss impulse from the acceleration responsive auxiliary governor (103% of rated speed) will cause the turbine speed to decrease after the entrapped steam has been used. When the turbine speed is reduced slightly, the auxiliary governor will be de-energized and the solenoid valve will close returning the unit to load control.

C. High-pressure Fluid Control System

The function of the high-pressure fluid control system is to provide a motive force which positions the turbine steam valves in response to electronic commands from the controller, acting through the servo-actuators. The system is so arranged that one pump and one set of the various control components function while the duplicate set serves as a stand-by system.

The emergency trip fluid circuits are charged from the high-pressure fluid manifold through individual actuation mechanisms.

D. Solid State Controller

The solid state controller provides control signals to the turbine valve actuating servos for turbine speed and load control.

Wide-range speed control is used during the phase of operation from turning gear to rated speed. The speed controller receives a continuous turbine speed signal from a variable reluctance transducer mounted at the turbine shaft. A speed reference signal and the actual speed signal are algebraically added to produce a speed error. Initial speed control is by the throttle valve only with the governor valve wide open. The speed error signal is transmitted to the throttle valve servo amplifier for throttle valve position control. Prior to synchronizing, a transfer is made from throttle valve control to governor valve control. The throttle valves are then positioned wide open and speed is controlled only by the governor valves.

Once the turbine has reached synchronous speed, and the main generator breaker is closed, the control system changes from a speed controller to a load controller. The speed reference output is switched to a load summing junction and the speed reference is replaced by a DC reference signal equal to synchronous speed.

The system sums the load reference and the speed error signal and sends this signal to position the governor valves. For linear load response, the impulse pressure feedback is also switched to the load summing junction.

TABLE 7-5

PROHIBITS ON CEAs (MANUAL CONTROL MODE)

<u>WITHDRAWAL PROHIBIT CONDITION</u>

Pretrip Overpower

Dropped CEA

High Startup Rate Pretrip (Between 10^{-4}% and 15% power)

Thermal Margin/Low Pressure Pretrip

<u>MOTION INHIBIT CONDITION</u>

Regulating group OOS

Individual CEA deviation

Excessive regulating group overlap

PDIL alarm

Withdrawal of regulating groups prior to withdrawal of all shutdown CEAs to a predetermined permissible range

Insertion of shutdown groups prior to insertion of all regulating CEAs to a predetermined permissible range

<u>INPUT SIGNAL SOURCES FOR THE WITHDRAWAL PROHIBITS</u>

RPS:	Overpower Pretrip High Startup Rate Pretrip Thermal Margin/Low Pressure Pretrip
Reed Switch Transmitter:	Dropped CEA

<u>INPUT SIGNAL SOURCE FOR CEA MOTION INHIBIT</u>

Reed switch CEA Position Indication System

7.5 INSTRUMENTATION SYSTEMS

7.5.1 PROCESS INSTRUMENTATION

7.5.1.1 Design Basis

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in the RCS, secondary system, and auxiliary systems. Process variables required for startup, operation, and shutdown of the plant are indicated, recorded, and controlled from the Control Room. Other instrumentation used less frequently or which requires a minimum of operator action is located near the equipment. Alternate indicators and controls are located in places other than the Control Room to allow reactor shutdown should the Control Room have to be evacuated.

The containment pressure transmitter sensing lines are the only instrument lines to which Safety Guide 11, "Instrument Lines Penetrating Primary Reactor Containment," was applicable, and are designed accordingly.

The containment pressure transmitters are located outside the containment structure in the electrical penetration room. They are located as close as practical to the containment and installed using short connections between the containment penetrations and the instruments. The transmitters are designed as pressure retaining devices, whereby rupture of the sensing device would not release radioactivity to the environment, but would contain the radioactivity within the housing of the instrument. Each sensing line is provided with a solenoid-operated isolation valve which is located as close as possible to the containment penetration. The isolation valves are controlled from the Control Room and are provided with position switches for remote indication in the Control Room. The isolation valves and instrument lines, up to and including the pressure retaining parts of the instruments, are Seismic Category I.

Provisions have been made for periodic visual in-service inspection of the isolation valves, pressure sensing lines, sample lines, and the pressure transmitters.

Four independent measurement channels are provided to monitor each process parameter required for the RPS. Redundant channels are provided for ESFs action to meet the single-failure criterion.

Two channels are provided to monitor parameters required for critical control functions. These channels and associated sensors are independent of the RPS.

7.5.1.2 System Description

The process instrumentation described below is associated with the RPS, reactor control, or reactor plant controls (Figure 4-1).

Temperature

The temperature measurements are made with precision RTDs, which provide a signal to the remote temperature indicating control and safety devices.

The following is a brief description of each of the temperature measurement channels:

a. Hot leg temperature: Each hot leg contains five temperature measurement channels. Four of these channels provide a hot leg temperature signal to the TM/LP trip circuits and the sub-cooled margin monitors (SCMM). The other hot leg temperature measurement channel provides a signal to the

loop T_{avg} computer in the Reactor Regulating System. The five hot leg temperatures are indicated on the control panel.

b. Cold leg temperature: Each cold leg branch contains three temperature measurement channels. Two of the channels in each branch provide a cold leg temperature signal to the TM/LP trip circuits and the SCMM. These channels also provide cold leg temperature indication on the control panel. The third cold leg temperature measurement channel in one branch provides a signal to the loop T_{avg} computers. This channel also provides a high alarm and a signal to an automatic CEA withdrawal prohibit. The third channel in the other branch is recorded on the control panel.

c. Loop average temperature: Each of the two Reactor Regulating System channels receives a hot leg and cold leg temperature from each loop (as mentioned above). The T_{avg} summer receives input hot and cold leg temperatures from any combination of the loops and provides an average temperature output to the Reactor Regulating System and to a recorder. The temperature recorders are equipped with two pens. One pen records the average temperature and the other pen records the programmed reference temperature signal (T_{ref}) corresponding to turbine load (first stage pressure). Redundant hot and cold leg temperatures for each loop are displayed at the alternate shutdown panel.

Redundant hot and cold leg temperatures for each loop are displayed at the alternate shutdown panel.

Pressure

Pressure is measured by electromechanical pressure transmitters. The transmitter produces a DC output that is proportional to the pressure sensed by the instrument. The DC outputs are used to provide signals to the remote pressure indicating, control, and safety devices.

The following is a brief description of each of the pressure measurement channels:

a. Pressurizer pressure (protective action): Four pressurizer pressure transmitters provide independent pressure signals over a 1000 psi range. These four independent pressure channels provide the signals for the RPS high-pressure trip, the variable TM/LP trip, and the DSS high-pressure trip. The channels also provide the low-low pressure signal to initiate safety injection. All four pressure channels are indicated in the Control Room and high, low, and low-low alarms are annunciated. Figure 7-3 is a functional diagram of one of these channels.

b. Pressurizer pressure (control action): Two independent pressure channels provide narrow range signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function. Outputs from the two pressure control channels are recorded in the Control Room and provide independent high and low alarms.

c. Pressurizer pressure (miscellaneous channels): Two pressure channels, electrically independent of the safety and control channels, but sharing pressure taps with the safety channels, provide inputs to the Control Room, and generate the power interlocks (blocking) for the shutdown cooling isolation valves on high pressurizer pressure.

d. Pressurizer pressure (Wide Range): Two pressure channels, electrically independent of the safety and control channels, but sharing pressure

sensing lines with the safety channels, provide inputs to the SCMMs, the auxiliary shutdown panels, and the Control Room.

Pressurizer Level

Level is sensed by level transmitters which measure the pressure difference between a reference column of water and the pressurizer water level. This pressure difference is converted to a DC signal proportional to the level of water in the pressurizer. The DC output of the level transmitters provides signals to the remote level indicating and control devices.

Two independent pressurizer level transmitters provide signals to the chemical and volume control charging and letdown system. In addition, signals are provided for pressurizer heater override control. These level transmitters are calibrated for steam and water densities existing at normal pressurizer operating conditions.

The two pressurizer level control channels each provide a signal for level recorders in the Control Room. These recorders are two-pen recorders, with one pen recording actual level as sensed by the level control channel and the other pen recording the programmed level setpoint signal from the Reactor Regulating System.

The pressurizer pressure level channels provide read-out in the Control Room and on the auxiliary shutdown panel.

Flow

An indication of reactor coolant flow is obtained from measurement of pressure drop between the hot leg piping and the outlet plenum of each SG. The pressure drop is sensed by d/p transmitters which convert the pressure difference to DC output. The DC output provides a signal to the remote flow indicating and safety devices.

Four independent d/p transmitters are provided in each reactor coolant loop to measure the pressure drop across the SGs. The outputs of corresponding transmitters in each loop are summed by pairs to provide four independent signals representative of flow through the reactor core. These signals are indicated and supplied to the RPS for loss-of-flow determination. The d/p sensed by each transmitter is indicated in the Control Room.

7.5.2 NUCLEAR INSTRUMENTATION

7.5.2.1 Design Basis

The nuclear instrumentation monitors neutron flux over a range greater than ten decades with four independent and redundant channels. These channels generate a startup rate and provide the RPS with four independent signals. Neutron flux is also monitored over the power range to provide four redundant signals, proportional to reactor power, to the RPS. Furthermore, nuclear instrumentation monitors the power range with two additional channels for control of the reactor. These two channels are completely independent of those mentioned above. In addition, an independent excore wide-range neutron monitoring channel is provided to indicate neutron flux level on the auxiliary shutdown panel.

7.5.2.2 System Description

Installation

The nuclear instrumentation signal processing equipment associated with the RPS is located in the RPS cabinet in the Control Room. Four cabinets designated as A, B, C, and D each house one channel of the protective system. Each cabinet contains one power range safety channel and one wide-range logarithmic channel. Mechanical and thermal barriers between the cabinets reduce the possibility of common event failure. The detector cables are routed separately from each other. This includes separation at the containment penetration areas. The signal processing equipment for the independent wide-range channel display on the auxiliary shutdown panel is located in the switchgear room at Elevation 45'0" in the Auxiliary Building.

Functional Description

Ten channels of instrumentation are provided to monitor neutron flux. The system consists of wide-range logarithmic channels, power range safety, and power range control channels. Each channel is complete with separate detectors, power supplies, amplifiers, and bistables to provide independent operation. The operating capability of the ten monitoring channels is greater than ten decades of neutron flux and is adequate to monitor reactor power from shutdown through startup to 200% of full power.

Four wide-range logarithmic channels monitor flux from source level to above full power. The flux signals, obtained from fission chambers, are amplified and transmitted to the power and rate-of-change-of-power amplifiers located in the Control Room. Audible count rate signals are available in the Control Room. In addition to the information on reactor flux, these channels provide a rate-of-change-of-power signal to the RPS for reactor trip and to the CEDM control system for CEA withdrawal prohibit.

Four channels are designated as power range safety channels and provide signal outputs to the RPS. These channels operate from 0.1 to 200% of full power. Power level signals from these channels are supplied to the protective system. These four channels contain detectors composed of dual section ion chambers which monitor the full axial length of the reactor core at four circumferential positions equally spaced around the core. This arrangement enables detection of power tilts and imbalance.

Two separate power range control channels, similar to the power range safety channels, provide reactor power signals to the Reactor Regulating System. The channel output is a signal directly proportional to reactor power from 0.1 to 200%. The power signal is combined with the average coolant temperature, first-stage turbine pressure, and pressurizer pressure signals as the control parameters to the Reactor Regulating System. Additional power range control channel functions are discussed in Section 7.5.2.6.

The gain of each channel is adjustable to provide a means for calibrating the output against a plant heat balance. Each channel provides a power reference signal to one of the independent Reactor Regulating System channels.

The independent wide-range channel for the auxiliary shutdown panel continuously monitors neutron flux outside the reactor vessel and provides an indication of neutron flux over a range from 10^{-1} CPS to 200% power.

7.5.2.3 Design Criteria

The system associated with the RPS was designed in accordance with the criteria of IEEE 279, August 1968. In areas not covered or specifically identified by the criteria, the following criteria are used:

a. The nuclear instrumentation sensors are located so as to detect representative core flux conditions;
b. Four independent channels are used in each flux range;
c. The channel ranges overlap sufficiently to ensure that the flux is continually monitored from source range to 200% of rated power;
d. Power is supplied to the system from four separate AC busses. Loss of one bus trips one safety channel and one wide-range logarithmic channel;
e. Loss of power to channel logic results in a channel trip;
f. All channel outputs are buffered so that accidental connection to 120 Volts AC, or to channel supply voltage, or shorting individual outputs has no effect on any of the other outputs.

The independent wide-range channel for the auxiliary shutdown panel is designed in accordance with the requirements of 10 CFR Part 50, Appendix R. In addition, the electrical installation in the containment is seismically designed.

7.5.2.4 Wide Range Nuclear Instrumentation Channel Description

The wide-range logarithmic channels combine conventional pulse counting and mean square variation techniques to monitor power from source range to 200% of full power. The lowest decades of power indication utilize the pulse signal from the ganged fission chambers only. Power information is presented in terms of counts per second. The scale indication lights indicate the range change from source range to wide-range. The significant power information above this point is presented as a percent of full power. The two neutron flux indicating ranges overlap by approximately two decades. Pulses from the fission chambers are counted by pulse counting circuitry. After approximately five decades of counting, the counting circuitry saturates; further power level information is obtained through detection of the mean square variation of the input count rate.

Two redundant reactor cavity cooling fans (Section 9.8.2.2) cool the fission chambers. If a total loss of air flow occurred, the temperature rise in these neutron detectors would be very slow. Typically, the temperature of the closest detectors would rise from 120°F to 160°F in 30 minutes while the temperature of the farthest detectors would rise from 120°F to 138°F in 30 minutes. The fission counter assemblies have been designed to operate satisfactorily at a temperature of 300°F with no appreciable error. Therefore, ample time is available in the event of loss of air flow to take corrective action. This information is also applicable to the power range uncompensated ion chambers.

Two fission chamber detector elements within each assembly provide high sensitivity while operating in the gamma flux encountered following reactor shutdown. System reliability is enhanced through use of integral triaxial detector cables with mineral insulated insulation within the high neutron flux region and out-of-containment amplifiers. The outputs of each fission counter assembly are fed separately to an initial preamplification stage in the amplifier. The pulses are then combined, amplified, and transmitted to the signal processing drawer in the Control Room.

The high frequency pulse signal components pass through a conventional log count rate circuit utilizing a pulse shaping circuit, diode log pump circuit, and an active differentiating circuit to provide a rate-of-change signal.

A single fission chamber signal provides low frequency components which are separated by a bandpass amplifier. The AC output of the bandpass amplifier is (in accordance with Campbell's Theorem) proportional to the square root of the average pulse rate. This signal is rectified, filtered and applied to a logarithmic amplifier. As the lower portion of the output of the logarithmic amplifier is affected by gamma and alpha background, noise, imperfect rectification, and lack of pulse overlap (Campbell's Theorem applies only when pulse overlap is achieved), this portion of the response is cut off by a biased diode.

By summing the two signals, a DC signal proportional to the logarithm of neutron flux over the range of approximately 10^{-8}% full power to 200% full power is obtained. Log count rate information is presented from source level to around 10^{-7}% of full power. This signal is indicated on the main control board.

The log power level signal is differentiated to provide rate-of-change of power information from -1 to +7 decades/minute. The rate signal feeds a front panel meter, the RPS, and an indicator on the main control board.

Channel test and calibration is accomplished by internally-generated test signals. Pulse rates controlled by a ceramic resonator check the counting portion of the circuitry. The mean square portion of the circuitry is checked by inserting current signals of calibrated amplitude. During calibration, a full scale output signal is substituted for the rate signal feeding the RPS.

Bistable circuits are used in each wide-range log channel. These bistables perform the following functions. They initiate an alarm on decrease of detector voltage, remove the zero power mode bypass, disable the rate-of-change output to the RPS, switch the indicating lights on the control panel to indicate the appropriate indication scale (percent power or counts per second), annunciate when the start-up-rate trip is enabled, and enable the PDIL circuitry.

Each of the four wide-range nuclear instrumentation channels is capable of providing a pulse signal to an independent shutdown monitor which provides alarm capability in the Control Room to identify the approach to criticality during the postulated Boron Dilution Event (Section 14.3.2). The monitor establishes the count rate at a fixed multiple of the lowest measured count rate and alarms when the setpoint is exceeded while minimizing unwanted background counts. This provides high reliability, automatic count-rate monitoring during shutdown and refueling mode operations.

The wide-range logarithmic channels provide the signals that automatically remove the low-flow and TM/LP trip bypasses above 10^{-4}% full power.

The zero power manually-actuated bypass allows CEA drop testing, or CEA withdrawal for other tests, during shutdown. The trips bypassed are low-flow and thermal margin/low pressure. These trips are automatically reset above 10^{-4}% full power.

The high rate-of-change of power trip bypass at above 15% full power is initiated by a bistable in the power range safety channel.

The high rate-of-change of power pretrip is enabled above 10^{-4}% of full power. The pretrip also initiates CEA withdrawal prohibit.

7.5.2.5 Power Range Safety Channel Description

The four power range channels are capable of measuring flux linearly over the range of 0.1% to 200% of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of 12'. The DC signal from each of the ion chambers is fed directly to the Control Room drawer without preamplification. Integral shielded cable is used in the region of high neutron and gamma flux.

The signal from each ion chamber is fed to an independent amplifier. Within each channel the outputs of the two amplifiers are indicated, compared, and summed. The range of this indication is 0.1 to 200% full subchannel power. The individual amplifier output is indicated on the amplifier drawer. This output is subtracted from the output of the other amplifier in the same channel to provide for deviation signal. The summer output of the two amplifiers feed bistables, an indicator, and the RPS.

Each power range safety channel contains 2 bistables. These bistables are responsible for initiating the ability to enable and disable Axial Power Distribution, Loss of Load, and Start-up Rate trips within a predetermined setpoint value.

Channel calibration and test is accomplished by an internal source, which checks amplifier gain and linearity.

7.5.2.6 Power Range Control Channel Description

The power range control channels are similar to the power range safety channels. They are located in the Reactor Regulating System panels in the Control Room. The two power range control channels are capable to measuring flux linearly over the range of 0.1% to 200% of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of 12'. The DC signal from each of the ion chambers is fed directly to the Control Room drawer without preamplification. Integral shielded cable is used in the region of high neutron and gamma flux.

The signal from each ion chamber is fed to an independent amplifier. Within each channel, the outputs of the two amplifiers are indicated, compared, and summed. The range of this indication is 0.1% to 200% full subchannel power. The ion chamber outputs are displayed on indicators on the amplifier drawer. The compared output is derived by subtracting the output of one amplifier from the other amplifier in the same channel. The summer output is derived from either the average of the two amplifier outputs, or the value of either individual amplifier. The output for the power range control channels feed the Plant Computer, Feedwater Control System, NIS Power Range Control Recorder, Internal Vibration Monitoring System, Power Ratio Calculator, and the Reactor Control Unit Calculator.

Channel calibration and test is accomplished by an internal source that checks amplifier gain and linearity.

7.5.2.7 Independent Wide Range Channel Description

The Alternate Shutdown Panel (C43) is provided with two seismically-mounted indicators on the auxiliary shutdown panel. They have the ability to select one of two neutron flux monitoring signals from the RPSs wide-range nuclear instrumentation via optical isolation assemblies.

The associated selector switch and signal processor are seismically-mounted in the switchgear room, Elevation 45' in the Auxiliary Building. This installation is in accordance with the design requirements of 10 CFR Part 50, Appendix R.

7.5.3 CEA POSITION INSTRUMENTATION

7.5.3.1 Design Basis

The principal purpose of the CEA position indication system is to provide the operator with reliable, comprehensible, and timely information on CEA position.

The following bases are used in the CEA position indication system design:

a. Position readouts of all CEAs may be obtained;
b. Continuous position readouts of any selected CEA in a group are available;
c. A means of alerting the operator to deviation of CEAs within a group is provided;
d. A permanent record may be made of the position of any or all CEAs. A record is made automatically at time of CEA deviation or at predetermined intervals. The operator may obtain a record at any other desired time;
e. Separate "full-in" and "full-out" indication is provided for each CEA;
f. The position information is presented to the operator: (1) in a form easy to compare and interpret, and (2) in a compact location convenient to the CEA Controls;
g. Redundant and independent means of indicating CEA position is provided.

7.5.3.2 Pulse Counting Position Indication System Description

The pulse counting CEA position indication system infers the position of each CEA by maintaining a record of the raise and lower control pulses sent to each magnetic jack mechanism. This system provides a position indication accuracy of within 1-3/4". This system is incorporated in the plant computer which feeds control board digital displays. On these displays, there is one position indication for each CEA group. The selection of any CEA for individual movement also selects that CEA for continuous digital display. Position information is available via plant computer archive for viewing historical position information. The position of each CEA is periodically logged for permanent record purposes. A printout is available, on operator demand, of the position of all CEAs or of those CEAs within a given group. The plant computer also provides deviation information. If the deviation in position between the highest and the lowest CEA in any group exceeds setpoints of 3.75” and 7.5", the computer provides an alarm and initiates a log of the actual positions of all CEAs within the group. This group deviation alarm (DEV) setpoint was analyzed for regulating CEAs for its effects on the power distribution. The results showed the power distribution at 7.5" to be well within design limits (Section 3.4.6). The plant computer provides position information for CEA group position alarms.

7.5.3.3 Reed Switch Position Indication System Description

The reed switch CEA position indication system utilizes a series of magnetically-actuated reed switches, spaced at 1.5" intervals along the CEA housing and arranged with-precision resistors in a voltage divider network, to provide voltage signals proportional to CEA position. This system is required to have a position indication accuracy of within ± 3.0". These signals are displayed in bar chart form by a cathode ray tube (CRT) on the main control board. A logic package associated with the CRT provides redundant alarm functions. A backup readout is provided which can be utilized to read the output of any reed switch voltage divider.

The CRT logic package generates and displays the power-dependent and pre-power dependent insertion alarm limits. The first cycle alarm limit is a straight line approximation to the curve shown in Figure 3.4-27, with the alarm occurring at an insertion less than that shown in the figure.

Additional alarms generated by this system are Group Deviation (DEV), Regulating Groups OOS, Shutdown Group Insertion Interlock (MIRG), Excessive Regulatory Group Overlap (ERGO), and Regulating Group Withdrawal Interlock (MISH).

The above provide contact openings on alarm conditions which are used to generate two different CEA motion inhibits (CMI) at the CEDS as discussed in Section 7.4.2. A CMI is generated for the shutdown CEAs should any one combination of DEV, OOS, PDIL or MIRG alarm conditions occur. A CMI is generated for the regulating CEAs should any one or combination of DEV, OOS, PDIL, ERGO, or MISH alarm conditions occur.

The reed switch system is electrically-and mechanically-isolated from the pulse counting position system.

7.5.3.4 Additional CEA Position Indication

A group of 57 light displays, arranged in a shape corresponding to the CEA distribution, is located on the main control board. Each display, which represents one CEDM, contains four colored lights providing the information listed in Table 7-6.

All lights are actuated by reed switches with the exception of the blue exercise limit lights associated with the shutdown CEAs. The blue lights are actuated by the computer.

7.5.4 INCORE INSTRUMENTATION

7.5.4.1 Design Basis

The primary function of the incore instrumentation is to provide measured data, which may be used in evaluating the gross core power distribution in the reactor core, as an aid to reactor operations. This data may be used to evaluate thermal margins and to estimate local fuel burnup. Section 7.5.9.3 describes the Core Exit Thermocouple (CET) design bases. No credit is taken for this system in the accident analysis of Chapter 14.

The bases for the design of the incore monitoring system are as follows:

a. Detector assemblies are installed in the reactor core at selected locations to measure core neutron flux and coolant temperature information during reactor operation in the power range;

b. Flux detectors of the self-powered type, with proven capabilities for incore service, are used;
c. The information obtained from the detector assemblies may be used for fuel management purposes and to assess the core performance. It will not be relied on for automatic protective or control functions;
d. The output signal of the flux detectors will be adjusted for changes in sensitivity due to emitter material burnup and for undesirable background signals;
e. Each detector assembly is comprised of four local neutron flux detectors stacked vertically for axial monitoring, and one thermocouple at the assembly outlet.

Axial spacing of the detectors in each assembly and radial spacing of the assemblies permit an evaluation of the gross core power distribution through the use of an incore analysis computer program.

7.5.4.2 System Description

The incore instrumentation system consists of 35 fixed incore detector assemblies inserted into selected fuel assemblies. Each assembly contains four 40 cm long rhodium detectors, a background wire, and one thermocouple. Rhodium detector outputs are fed via the Data Acquisition System (DAS) to the plant computer in the Control Room for processing and logging. Thermocouple outputs are fed through the CET System described in Section 7.5.9.3.

Assemblies are inserted into the core through instrumentation nozzles in the top closure head of the reactor vessel. Each assembly is guided into position in the center of the fuel assembly via a fixed guide tube and instrument thimble assembly. For both Units 1 and 2, a swagelok-type seal forms a pressure boundary for each assembly at the instrument nozzle.

The neutron detectors produce a current proportional to neutron flux by a neutron-beta reaction in the detector wire. The emitter, which is the central conductor in the coaxial detector, is made of rhodium and has a high thermal neutron capture cross section. The maximum useful life of the rhodium detectors is based on detector accuracy, after which the detector assemblies will be replaced by new units.

The incore detectors are not required to be operable during power operation. However, periodic recalibration of the excore nuclear instrumentation will require use of the incore detectors.

The data from the thermocouples and detectors are read out by the plant computer which scans all assemblies, processes, and logs the data. The computer periodically computes integrated flux at each detector to update detector sensitivity factors to compensate for detector burnout.

7.5.4.3 Incore Instrumentation Requirements for Monitoring Technical Specification Limits

On July 16, 1993, the NRC issued a Final Policy Statement on Technical Specification improvements for nuclear power reactors. The Final Policy Statement contains four criteria which can be used to determine which constraints on the design and operation of nuclear power plants are appropriate for inclusion in the plant's Technical Specifications. The ICI System does not meet any of those four criteria. Therefore, on November 3, 1993, Baltimore Gas and Electric

Company (BGE) requested the elimination of Technical Specification 3.3.3.2 and the relocation of the Technical Specification limitations on the use of the ICI System to the Calvert Cliffs UFSAR.

The NRC approved BGE's request, stating that in order to change the requirements concerning the number and location of functional detectors a successful 50.59 with a rigorous evaluation and justification is required (Reference 1). The reduction of the number of ICIs from 45 to 35 was accomplished via such a 50.59 evaluation for both Unit 1 and Unit 2. The following considerations must be included in the evaluation:

a. How an inadvertent loading of a fuel assembly into an improper location will be detected;
b. How the validity of the tilt estimates will be ensured;
c. How adequate core coverage will be maintained;
d. Why the uncertainties are adequate to guarantee that measured peak linear heat rates, peak pin powers, radial peaking factors, and azimuthal power tilts will meet Technical Specification limits; and
e. The number of operable detectors must be at least 75% of the total number of detectors prior to the start of a new cycle.

The following definitions apply to this section:

a. An operable incore detector segment consists of an operable rhodium detector constituting one of the segments in a fixed detector string.
b. An operable incore detector location consists of a string in which at least three of the four incore detector segments are operable.
c. An operable quadrant symmetric incore detector segment group consists of a minimum of three operable rhodium incore detector segments in 90° symmetric fuel assemblies.
d. An axial elevation refers to any axial plane of core height that contains an incore detector segment.

The following uncertainties apply to this section:

a. A F_r^T measurement uncertainty factor of 1.06;
b. A linear heat rate measurement uncertainty factor of 1.07;
c. An engineering uncertainty factor of 1.03;
d. For measured thermal power less than or equal to 50%, but greater than 20% of rated full core power, a thermal power measurement uncertainty factor of 1.035; and
e. For measured thermal power greater than 50% of rated full core power, a thermal power measurement uncertainty factor of 1.020.

The Incore Detector System is demonstrated operable by performing the verification tests in Technical Requirements Manual Section 15.3.3. For various functions of the Incore Detector System, the requirements are presented below:

a. For base functionality of the Incore Detector System,
    1. At least one detector segment in each core quadrant at each of the four axial elevations must be operable.

b. For monitoring the azimuthal power tilt with the Incore Detector System,

   1. At least two quadrant symmetric detector segment groups must be operable at each of the four axial elevations in the outer 184 fuel assemblies.

c. For recalibration of the Excore Neutron Flux Detection System,

   1. At least 75% of all incore detector segments operable.
   2. A minimum of nine incore detector segments must be operable at each of the four axial elevations.
   3. A minimum of two detector segments in the inner 109 fuel assemblies must be operable at each of the four axial elevations.
   4. A minimum of two detector segments in the outer 108 fuel assemblies must be operable at each of the four axial elevations.

d. For monitoring total integrated radial peaking factor, or the linear heat rate,

   1. At least 75% of all incore detector string locations must be operable (i.e., at least three of four segments are considered operable).
   2. A minimum of nine incore detector segments must be operable at each of the four axial elevations.
   3. A minimum of two detector segments in the inner 109 fuel assemblies must be operable at each of the four axial elevations.
   4. A minimum of two detector segments in the outer 108 fuel assemblies must be operable at each of the four axial elevations.
   5. All 5x5 arrays of fuel assemblies that contain 25 fuel assemblies must contain at least one operable detector segment on any axial level.

e. For post-refueling startup testing and power ascension,

   1. Meet the requirements of "b" and "d.1" through "d.4" above for azimuthal power tilt monitoring and for monitoring total integrated radial peaking factor, or linear heat rate.

   AND

   2. Meet either of Criterion II **OR** IV for detection of core misload:
      (a) Criterion I
         (1) Deleted.
      (b) Criterion II
         (1) Deleted.
         (2) All 5x5 arrays of fuel assemblies that contain 25 fuel assemblies must contain at least one operable detector segment on any axial level.
         (3) Any ICI availability requirements established in the applicable misload detection analysis that are not specified in Technical Requirements Manual 15.3.3 shall be met.
      (c) Criterion III
         (1) Deleted.
      (d) Criterion IV
         (1) Perform an evaluation of the ability of the Incore Detector System to detect core power symmetry with the actual

operable incore detector pattern prior to exceeding 30% power.

(2) Implement symmetry checks as identified in the evaluation.

(3) Implement penalties on the total integrated radial peaking factor, and linear heat rate as identified in the evaluation.

The Incore Detector System may be used for monitoring the core power distribution by verifying that the incore detector local power density alarms are adjusted to satisfy the requirements of the core power distribution map, which shall be updated at least once every 31 days of accumulated operation in Mode 1 and have their alarm setpoint adjusted to less than or equal to the limits when the uncertainties are appropriately included in the setting of the alarms.

7.5.5 COMPUTER SYSTEMS

7.5.5.1 Data Acquisition System

The DAS is a two-channel system consisting of multiplexers and data concentrators. The multiplexers receive digital and analog process inputs from plant systems. The data concentrators convert this information into engineering units capable of being used by the computer systems which receive their input from the DAS. Additionally, the DAS processes plant data for sequence of events contacts and serves as the Class 1E isolation between safety-related process loops and the non-safety-related plant computers. The DAS has two channels to provide redundancy where required.

A Plant Data Network is provided for network connectivity to non-safety-related plant instruments, controls, and information systems. The Plant Data Network provides the first three layers of the Open Systems Interconnection model using network switches. Core switches provide layer 3, 2, and 1 services, while edge switches provide layer 2 and 1 services only. Uplinks are provided for connecting edge switches to the core switches. The Plant Data Network has two channels to provide redundancy. Plant applications are hosted by various input/output devices, programmable logic controllers, servers, workstations, and miscellaneous devices that are considered external to the Plant Data Network.

7.5.5.2 Plant Computer

A process computer is provided to assist the Control Room operators in the safe and efficient operation of each unit. The plant computer performs the following major functions:

a. receive and process data from the DAS and other acquisition interfaces;

b. store processed data for logging and trending purposes;

c. provide alarms on printers, displays and annunciators;

d. provide status information on monitors;

e. provide plant performance information such as calculating power distribution, burnup, and thermal margin, and various secondary plant efficiency assessment reports;

f. CEA regulating group position indication on C05, input to the CEA drive system, as described in Section 7.4.2 and 7.5.3.1; and,

g. provide input of selected data to the Technical Support Center (TSC).

7.5.5.3 Safety Parameter Display System

The Safety Parameter Display System (SPDS) software application is a subsystem of the plant computer. The function of the SPDS is to provide a display of critical plant parameters to Control Room personnel to aid in rapidly determining the safety status of the plant. The SPDS meets the requirements of NUREG-0737, Supplement 1, (TMI Action Plan Item I.D.2), taking into account the information provided in NUREG-1342.

The SPDS performs its function by presenting graphic displays of selected parameters from which an evaluation can be made to determine if critical safety functions (CSF) are being met. Critical safety functions are those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public and plant personnel. The CSFs monitored are:

a. reactivity control,
b. RCS pressure and inventory,
c. core/RCS heat removal,
d. containment environment,
e. containment isolation,
f. radiation control, and
g. vital auxiliaries.

This information is arranged into display pages which are available on monitors in the Control Room, Shift Manager's office, and the TSC.

7.5.5.4 Technical Support Center Computer

The function of the Technical Support Center Computer is implemented by the Plant Process Computer historian server. This server has been designed to satisfy the requirements of NUREG-0696. The TSC computer receives input from the plant computers on both Units. Its function is to provide selected plant status information to support the staff assigned to the TSC during designated times. This information is available on workstations. The TSC computer enables the support staff to monitor and assess the status of the plant and assist the Control Room operators in analyzing events and safely stabilizing the plant.

7.5.6 RADIOACTIVITY MONITORING

The radiation monitoring systems are designed to detect, indicate, record, and on high levels provide radiation alarms throughout the plant. The system is divided into three subsystems: area radiation monitoring for personnel protection and indication of abnormal conditions in various areas of the plant; liquid process monitoring for personnel protection and indication of abnormal conditions in the process systems; and, gaseous monitoring for personnel protection and indication of airborne radioactivity.

For detailed information of the Radiation Monitoring System see Chapter 11.

7.5.7 SEISMIC INSTRUMENTATION

A seismic acceleration monitoring system that will automatically detect and record the seismic activity acceleration response of important features of the nuclear power plant has been engineered to ensure complete fulfillment of Nuclear Regulatory Commission Safety Guide 12.

Strong motion, triaxial transducers monitor the seismic response of selected Seismic Category I structures and the vibratory ground motion of the plant site. Points to be monitored include: Unit 1 Containment basement Elevation 10'0"; Unit 1 Containment Building floor at Elevation 69'0"; and, Auxiliary Building basement slab, Intake Structure, and free field. Each instrument listed below will be operable at all times.

Instruments and Sensor Locations			Measurement Range
1.	Triaxial Time-History Strong Motion Accelographs		
	a.	0-YE-001 Unit 1 Containment Base	0-1g
	b.	0-YE-002 Unit 1 Containment 69’	0-1g
	c.	0-YE-003 Auxiliary Building Base	0-1g
	d.	0-YE-004 Intake Structure	0-1g
	e.	0-YE-005 Free Field	0-1g
2.	Seismic Acceleration Recorder		
	a.	0-YRC-001 Control Room	NA
	b.	0-YRC-002 Control Room	NA
	c.	0-YRC-003 Control Room	NA
3.	Seismic Computer		
	a.	0-CPU-1C26B Control Room	NA
	b.	0-CRT-1C26B Control Room	NA

A computer with multiple recorders located in the Control Room provides the time-history records necessary to evaluate the frequency response of the selected Seismic Category I structures. The recorder units indicate to the Control Room operator if predetermined values of seismic acceleration have been exceeded.

When one of the recorders senses a seismic event, an interconnect (RS-232 Cables) network will cause all of the recorders to trigger and record data. Once recording is completed the seismic computer retrieves all of the seismic event data files from the recorders, associates the events together, and performs automatic analyses on the data. The system operates continuously for the full duration of an earthquake. The system remains in operation for a set time beyond the last detection of a seismic signal of triggering intensity.

If the system is activated during a seismic event, the instruments need to be restored to an operable status within 24 hours. Data from the activated instruments will be retrieved by computer and CAV (Cumulative Absolute Velocity) calculated automatically and analyzed to determine the magnitude of the vibratory ground motion. A report will be provided to the NRC describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

7.5.8 POST-ACCIDENT MONITORING INSTRUMENTATION

Generic Letter 82-33 requested a report to the NRC to describe how the post-accident monitoring system for this plant meets the guidelines of Regulatory Guide 1.97. This section summarizes the BGE response to the request.

Instrumentation meeting the applicable portions of 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 13, “Instrumentation and Control,” Criterion 19, “Control Room,” and Criterion 64, “Monitoring Radioactive Releases” is installed in the plant. Variables and systems can be monitored under accident conditions. Included is the instrumentation required for the operators to take the plant to hot shutdown from outside the Control Room and to monitor for radiation released following a postulated accident.

Certain instrumentation in the Control Room and on the Auxiliary Safe Shutdown Panels used for normal plant operations is designated for post-accident monitoring (PAM) use. Instrumentation defined as PAM1 and PAM2, with the exception of switches and indicating lights, in the Calvert Cliffs Quality List (Q-List) is marked with colored tape to help operators recognize it as PAM instrumentation.

To determine which instrumentation is designated for PAM use, a list of instrumentation was prepared based on the guidelines of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3. The guidelines establish the variables to be monitored and the design criteria for the selected equipment. Depending upon the importance of the variable (category), the design criteria ranges from fully qualified, Class 1E to highly reliable, commercial grade installation.

The list of instrumentation was then evaluated against the specific function and safety significance of the variables as defined in the UFSAR and the Emergency Operating Procedures. Where deviation from the guidelines resulted, justification was prepared to account for plant-specific needs and acceptability, or the equipment was upgraded to meet the criteria of Regulatory Guide 1.97. The evaluation performed for Regulatory Guide 1.97 resulted in a list of variables which are classified by category.

A summary of the PAM instrumentation is given here. Indicators listed may be one of several in a loop. The Q-List contains the complete list.

a. Category 1 variables. They are considered key variables that most directly indicate the accomplishment of a safety function, the operation of a safety system or radioactive materials release.

	VARIABLE	INDICATOR
1.	Pressurizer level	LI-110X, Y, X-1, Y-1
2.	Pressurizer pressure	PI-105B, AA PI-103 PI-103-1 1CRT1C06A, B (Unit 1 only) 2CRT2C06A, B (Unit 2 only)
3.	SG level	LI-1114A, B, D LI-1124A, B, D 1CRT1C04A, B (Unit 1 only) 2CRT2C04A, B (Unit 2 only)
4.	SG pressure	PI-1013A, B, C, D, AA, BB PI-1023A, B, C, D, AA, BB
5.	Condensate Storage Tank level	LI-5610, A, 5611, A LIA-5610, 5611

	VARIABLE	INDICATOR
6.	Containment Sump level (WR)	LI-4146, 4147
7.	RCS temperature, T_{hot}	TI-112H, HA, HB, 122H, HA, HB 1CRT1C06A, B (Unit 1 only) 2CRT2C06A, B (Unit 2 only)
8.	RCS temperature, T_{cold}	TI-112C, CA, CB, 122C, CA, CB 1CRT1C06A, B (Unit 1 only) 2CRT2C06A, B (Unit 2 only)
9.	RCS Subcooled Margin (°F)	1CRT1C05A, B (Unit 1 only) 2CRT2C05A, B (Unit 2 only)
10.	Reactor Vessel level	1CRT1C05A, B (Unit 1 only) 2CRT2C05A, B (Unit 2 only)
11.	Core Exit temperature	1CRT1C05A, B (Unit 1 only) 2CRT2C05A, B (Unit 2 only)
12.	Neutron flux	JI-001, 002, 003, 004 JKI-001, 002, 003, 004
13.	Containment pressure	PI-5307, 5308, 5310
14.	Containment Isolation Valve position	ZL-505, 506, 515, 516, 2080, 2180, 2181, 3832, 3833, 4260, 5291, 5292, 6900, 6901
15.	Containment Area Radiation	RI-5317A, B

b. Category 2 variables. They provide system status information.

	VARIABLE	INDICATOR
1.	Pressurizer Heater status	II-100-1, -2
2.	Safety Relief Valve position/Flow	VI-200, A, 201, A, 402, A, 404, A
3.	RWT level	LIA-4143
4.	Auxiliary Feedwater flow	FI-4509A, B, 4510A, B, 4524A, B, 4534A, B FIC-4511A, 4512A, 4525A, 4535A
5.	HPSI flow	FI-311, 321, 331, 341, 351
6.	LPSI flow	FI-312, 322, 332, 342, FIC-306
7.	Decay Heat Removal flow (Shutdown Cooling)	same as b.6 (LPSI)
8.	Decay Heat Removal temperature (Shutdown Cooling)	TR-351
9.	Containment temperature	TI-5309
10.	Containment Heat Removal (cooling water flow to containment coolers)	FI-1581, 1584, 1589, 1592
11.	Containment Spray flow	FI-4148, 4149
12.	CCW to ESF temperature	TIA-3824, 3826
13.	CCW to ESF flow	PI-3814, 3816
14.	Main Vent radiation/flow	RIC-5415 RR-5420
15.	SG Relief Valve and Atmospheric Dump radiation	RIC-5421, 5422
16.	Boric Acid Charging flow	ZL224X, Y, Z, ZA
17.	Status of Standby Power (Class 1E distribution systems)	

<u>4 kV Busses</u>

1II1115A	2II2115A	1II1401A	2II2401A
1II1115B	2II2115B	1II1401B	2II2401B
1II1115C	2II2115C	1II1401C	2II2401C
1II1101AA	2II2101AA	1II1414A	2II2414A

1II1101AB	2II2101AB	1II1414B	2II2414B
		1II1414C	2II2414C
		1EI414	2EI424
1II1101AC	2II2101AC	1II1702A	
1EI411	2EI421	1II1702B	
		1II1702C	
		1EI1702	

Diesel Generators

1II1403A	2II2104A	1II1701A	2II2403A
1II1403B	2II2104B	1II1701B	2II2403B
1II1403C	2II2104C	1II1701C	2II2403C
1JI1403A	2JI2104A	1JI1701A	2JI2403A
1JI1403B	2JI2104B	1JI1701B	2JI2403B
1EI1422	2EI2122	1EI1701	2EI2422
1SI1401	2SI2101	1SI1701	2SI2401

480 Volt Busses

1II1112A	2II2112A	1II1413A	2II2413A
1II1112B	2II2112B	1II1413B	2II2413B
1II1112C	2II2112C	1II1413C	2II2413C
1EI511A	2EI521A	1EI514B	2EI524B
1II1113A	2II2113A	1II1703A	
1II1113B	2II2113B	1II1703B	
1II1113C	2II2113C	1II1703C	
1EI511B	2EI521B	1EI710	
1II1412A	2II2412A		
1II1412B	2II2412B		
1II1412C	2II2412C		
1EI514A	2EI524A		

120 Volt AC Vital Busses

1EI1911	2EI1921	1EI1913	2EI1923
1EI1912	2EI1922	1EI1914	2EI1924

125 Volt DC Vital Busses

1II201	2II205	1EI1401
1EI211	2EI221	1II1401
1II211	2II213	
1II223	2II221	
1II202	2II204	
1EI212	2EI222	
1II212	2II214	
1II224	2II222	

Individual 4 kW Load Ammeters

1II1570	2II1570	1II302X	2II302X
1II1571	2II1571	1II302Y	2II302Y
1II1572A	2II1572A	1II4146	2II4146
1II1572B	2II1572B	1II4147	2II4147
1II301X	2II301X	1II4540	2II4540
1II301Y	2II301Y	1II5199	2II5199
1II301Z1	2II301Z1	1II5200	2II5200
1II301Z2	2II301Z2	1II5201A	2II5201A
		1II5201B	2II5201B

Individual 480 Volt Load Ammeters

1II224X	2II224X	1II5300	2II5300
1II224Y	2II224Y	1II5301	2II5301
1II224Z1	2II224Z1	1II5302	2II5302
1II224Z2	2II224Z2		
1II3813	2II3813		
1II3815	2II3815		
1II3817A	2II3817A		
1II3817B	2II3817B		
1II5299	2II5299		

c. Category 3 variables. They provide backup and diagnostic information.

	VARIABLE	INDICATOR
1.	Pressurizer temperature	TI-101, 105
2.	Pressurizer Spray temperature	TI-229
3.	SG Steam flow	FR-1011/1111, FR-1021/1121
4.	Safety Injection Tank level	LI-311, 321, 331, 341
5.	Safety Injection Tank pressure	PI-311, 321, 331, 341
6.	Safety Injection Tank Isolation Valve position	ZL-3614, 3624, 3634, 3644
7.	Quench Tank level	LI-116
8.	Quench Tank pressure	PI-116
9.	Quench Tank temperature	TI-116
10.	Volume Control Tank level	LI-226
11.	CVCS Make-up Water flow	FIC-210X
12.	CVCS Make-up Boric Acid flow	FIC-210Y
13.	CVCS Letdown flow	FIA-202
14.	Containment Sump level (NR)	LI-4144, 4145
15.	High Radiation Liquid Tank level (waste receiver, monitor tanks)	LI-4279, 4280, 4281, 4282
16.	Radioactive Gas Tank pressure (waste gas system)	PI-2188, 2189, 2190
17.	RCS Boron concentration	Grab Sample Equipment
18.	RCP status	II-151, 161, 171, 181
19.	MFW flow	FI-1111, 1121
20.	Boric Acid Charging flow	FIA-212
21.	CEA position	HS-5500, 5501 XL-5510 through 5531 ZL-5500, 5501, 5501A, 5511, 5513, 5515, 5517, 5519, 5521, 5523, 5525, 5527, 5529, 5531
22.	RCS radiation	RE-202 RI-202
23.	RCS radiation - gamma	Grab sample equipment
24.	Area radiation	RR-11, 21 R-7004, 7005, 7006, 7010 through 7027
25.	Plant Release Points radiation	R-5321, 5415, 1-RR-11, 2-RR-21

7.5.9 INADEQUATE CORE COOLING INSTRUMENTATION

The SCMM, Reactor Vessel Level Monitoring System (RVLMS), and CET comprise the Inadequate Core Cooling Instrumentation (ICCI) as required by Item II.F.2 of NUREG-0737, the post TMI-2 Action Plan. The ICCI supplements existing instrumentation in order to provide an unambiguous, easy to interpret indication of

inadequate core cooling (ICC). The function of the ICCI is to enhance the ability of the plant operator to diagnose the approach to and recovery from ICC.

The bases for the design of the ICCI are as follows:

a. Provides an unambiguous indication of ICC.
   - Indicates the existence of ICC caused by various phenomena.
   - Does not erroneously indicate ICC because of the presence of an unrelated phenomena.
b. Gives advance warning of the approach to ICC.
c. Covers the full range from normal operation to complete core uncovery.
d. Core exit thermocouples meet requirements of Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," of NUREG-0737, II.F.2.
e. Types and locations of displays and alarms used, determined by human-factors analysis.
f. Provides 99% availability with respect to functional capability for liquid-level display.
g. Designed in accordance with guidance given in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

The ICCI is integrated into a safety-related, Class 1E, seismically-qualified, redundant, data acquisition, processing, and display system referred to as the PAM System. Each channel has two RCS cold leg temperature inputs, two hot leg temperature inputs, a single wide-range pressurizer pressure input, eight reactor vessel level inputs, two SG level inputs, and channel-specific CET data. All PAM System data (on a channel basis) is available on any similar channel touchscreen flat panel display in the event of a display failure. Facility group ZA and ZB displays are provided on each control room panel. Specific default displays provide digital and pictorial representation of the data and trend chart data. The PAM System is connected to the Plant Data Network and provides data via Ethernet to a data storage system. Maintenance and testing of the PAM System may be performed at the maintenance and test panels. There are two panels per channel.

7.5.9.1 Subcooled Margin Monitor

Subcooled margin calculations are performed by the PAM System and displayed in the Control Room. The PAM System calculates the temperature saturation margin using RCS and CET temperatures with the wide-range pressurizer pressure and displays the calculated digital value in degrees F on the PAM System flat panel displays. Pressure saturation margin is also calculated by the PAM System and that data is available via operator selectable displays. Subcooled margin data is provided by the PAM System flat panel displays. The PAM System provides a low subcooling alarm that is annunciated in the Control Room.

7.5.9.2 Reactor Vessel Level Monitoring System

The RVLMS is based on the CE Heated Junction Thermocouple (HJTC) system. The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the fuel alignment plate to the reactor vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples. The probe assemblies are of the "Full Length Probe" design, housed within a separator tube outside a CEA shroud. Of the eight HJTC sensor assemblies in each probe assembly, two are located above the CEA shroud in the

upper area of the reactor vessel closure head, and one more is located near the top of the CEA shroud. Three other HJTC sensors are located near the top, centerline, and bottom of the hot leg nozzle elevations, respectively. A seventh sensor is located approximately equidistant between the bottom hot leg area sensor and the lowest sensor, which is located less than 1' from the top of the fuel alignment plate. With at least four operable sensors in a probe, the RVLMS is capable of providing the plant operator with the information needed to assess void formation in the reactor vessel head region and the trend of liquid level in the reactor vessel plenum. The four sensors in a probe that are required for an operable RVLMS channel are: One of the upper three (vessel head region) and three of the lower five (plenum region) sensors.

The PAM System processes HJTC probe input data, provides probe heater output data, and displays reactor vessel level information in the Control Room. The SPDS provides the primary HJTC display. The PAM System flat panel displays are backup displays and they are seismically-qualified, Class 1E. The default reactor vessel level displays in the Control Room also display subcooling margin and CET data. Coolant low level and system malfunctions alarms are provided on the control board.

7.5.9.3 Core Exit Thermocouple

The reactor incore instrumentation system consists of 35 fixed incore detectors inserted into selected instrumentation assemblies. Each assembly contains four rhodium self-powered neutron detectors, a background wire, and one CET. The CET system is composed of two channels. Each of the two channels is powered from a separate reliable Class 1E power source. The cabling and connections for the CET system are channelized, Class 1E safety grade.

The PAM System processes CET data. The PAM System consists of redundant data acquisition, processing, and display instrumentation channel. Each channel is powered from a separate reliable Class 1E power source. The primary display for the CET data is the SPDS. The PAM System is the back-up system that displays the CET data on flat panel displays mounted in the Control Room. The operator may select specific thermocouple data via the touchscreen flat panel displays.

The operability requirements for the CETs are (Reference 2):

a. A minimum of two electrically independent CET channels, each containing at least two environmentally qualified CETs per core quadrant.
b. The distribution of CETs in a channel must be such that each channel contains at least one operable CET near the center of the core and one operable CET near the perimeter of the core (either peripheral or shroud peripheral).
c. A CET's operability is based on a comparison of the CET temperature indication with the hot leg RTD temperature indication. Different criteria have been specified for CET groups (interior, peripheral, or shroud peripheral) to account for the core radial power distribution.

7.5.9.4 Reduced Reactor Coolant System Inventory and Mid-loop Operations Instrumentation

To meet the requirements of Generic Letter 88-17, reactor vessel temperature and level monitoring capabilities are provided for reduced RCS inventory and mid-loop operations. A reduced RCS inventory condition exists when the reactor coolant level is lower than 3' below the flange. A mid-loop condition exists when the

reactor coolant level in the reactor vessel is below the top of the flow area of the hot leg piping at the junction with the reactor vessel.

Temperature of the reactor coolant in the reactor vessel is required to be monitored following the removal of the control rod drive mechanism platform and prior to the removal of the reactor vessel head during reduced RCS inventory and mid-loop operations. The temperature is measured through the select jumpering of a minimum of 2 of the 35 existing CETs. One CET is jumpered on electrical facility ZA and one on electrical facility ZB.

The CETs are displayed on the PAM System flat panel displays mounted on control room panel 1(2)C05. Core exit thermocouple inputs not being used at this time have to be disabled via a maintenance and test panel. High temperature and system malfunction alarms are provided in the Control Room via the main annunciators. The high temperature alarm setpoint is adjustable via a maintenance test panel. Core exit thermocouple real time and trend data is provided by the PAM System flat panel displays.

Removal of the reactor vessel head terminates the requirement to monitor the reactor coolant temperature via the CETs.

The reactor coolant level in the reactor vessel during reduced RCS inventory and mid-loop operations is monitored using narrow- and wide-range instrumentation, and the Mansell Level Monitoring System. The narrow-range instrumentation is a Westinghouse ultrasonic level transmitter installed on the bottom of the shutdown cooling suction side of Hot Leg #12 and Hot Leg #22 for Unit 1 and Unit 2, respectively. The wide-range instrumentation comprises a 3" diameter level stillwell with a continuous level probe and a separate sight gauge. This wide-range instrumentation is tied into the refueling level pressure transmitter tubing and pressure referenced to the reactor head vent via the reactor head vent tubing. The Mansell Level Monitoring System consists of two independent channels. It uses inputs from absolute pressure transducers installed at the pressurizer vents and the RCS hot leg instrument nozzles to calculate RCS levels. The computer based system provides operator indication, alarm (audible and visual), and trending capabilities on 0C184 for the narrow- and wide-range instrument loops, as well as the Mansell Level Monitoring System.

Also, for the reduced RCS inventory and mid-loop operations, suction pressure instrument loops are provided for each LPSI pump. Each instrument loop consists of a transmitter, bistable located in panel 1(2)C10, an indicator and a trend recorder (pen) mounted on the refueling cart 0C184. The bistable provides a signal to actuate an alarm window on panel 1(2)C09 on a low suction pressure condition. The transmitter/bistable/alarm portion of each instrument is continuously in operation. Each loop's indicator and recorder (pen) is placed into operation by installing cable between panels 1(2)C10 and 0C184 when the refueling cart is placed into service.

7.5.10 REFERENCES

1. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), dated August 24, 1994, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant (License Amendment Nos. 191 and 168)

2. Letter from D. G. McDonald, Jr. (NRC) to G. C. Creel (BGE), dated October 12, 1990, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant

TABLE 7-6

CEA POSITION LIGHT MATRIX

LIGHT COLOR	REGULATING	SHUTDOWN CEAs
Red	Upper electrical limit	Upper electrical limit
Green	Lower electrical limit	Lower electrical limit
Amber	Dropped CEA	Dropped CEA
White	Between upper and lower limits	
Blue		Below exercise limit

7.6 OPERATING CONTROL STATIONS

7.6.1 GENERAL LAYOUT

The operating control stations consist of: (a) the Control Room, used for plant control during startup, normal operation, shutdown, and emergency operation; (b) an auxiliary control station used for the waste processing systems (common to both Units 1 and 2); and, (c) various local control stations for miscellaneous noncritical systems.

7.6.2 CONTROL ROOM

The Control Room, which is accessible from both the Auxiliary and Turbine Buildings, houses benchboard control boards, the combined benchboard-vertical control boards and miscellaneous vertical boards for both units. All control boards are designed to Seismic Category I requirements. For further information concerning seismic loading and design, refer to Figures 2.6-4 and 2.6-5, Response Spectra - Operating Basis (OBE) and Design Basis Earthquake, respectively.

The Control Room can be occupied under all credible incident conditions. It has two separate air conditioning units, two particulate, absolute, and charcoal filter unit assemblies with dampers and fans, and an airborne radioactivity detector in the return line. Filter unit dampers and fans, which act to automatically shunt a portion of the Control Room air through the filter unit assemblies, actuate upon sensing a high airborne radioactivity level or in response to a SIAS initiation from either unit. The filter unit dampers and fans can also be remotely actuated from the Control Room.

None of the materials used in the construction of the Control Room will support combustion, and electrical wiring is flame resistant. Also, portable CO_2 fire extinguishers are placed in readily-accessible stations in the Control Room, and respiratory protective equipment is available to the operators at all times.

7.6.3 RADIOACTIVE WASTE DISPOSAL SYSTEM CONTROL PANELS

The waste-processing local control panel, located in the Auxiliary Building, provides the controls, instrumentation, and alarms required to initiate, operate, and monitor the waste-processing systems. Critical indications are duplicated in the Control Room. All alarms are annunciated at their local panel with a master alarm provided in the Control Room.

7.6.4 MISCELLANEOUS LOCAL CONTROL PANELS

Local control panels for noncritical systems are located throughout the plant. Each panel contains the indications, controls, and alarms required for safe operation of the system. The various systems are provided with local alarms and a common alarm on the control board to alert the operator of any abnormal conditions within each of the systems.

7.6.5 FEATURES WHICH ENHANCE SAFE OPERATION

In order to maintain channel separation, the control boards contain fire barriers that separate the ESF and control channels, thus preventing loss of all protection due to a single fault.

The plant annunciator system is located across the top of the main control boards, providing visual and audible indication of abnormal conditions which require operator action.

7.6.6 REMOTE SHUTDOWN CAPABILITY

Numerous design features are provided to maintain Control Room accessibility. However, in the event the operator is forced to abandon the Control Room, emergency procedures

require that the reactor shall first be tripped. During this condition, local controls and the Auxiliary Shutdown Panel, located in the 45' Elevation switchgear rooms (immediately adjacent to the Control Room), provide the instrumentation and controls necessary to safely bring the plant to the hot shutdown condition.

The Auxiliary Shutdown Panel, as supplemented by local control panels, provides the remote shutdown capability required by 10 CFR 50.48 and 10 CFR 50, Appendix R. No automatic safety features are actuated from remote shutdown monitoring instrumentation. Electrical isolation devices are installed such that a fire at the Auxiliary Shutdown Panel will not prevent shutdown of the plant from the Control Room, and vice versa.

Locally available instrumentation and the instrumentation available at the Auxiliary Shutdown Panel (1C43) ensure it will be possible to perform the following functions, and monitor their effectiveness, from areas external to the Control Room.

a. Insert the CEAs and trip the turbine generator;

b. Borate the RCS; and,

c. Remove reactor decay heat following a reactor trip.

These instruments include:

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE
Wide Range Neutron Flux	1C43	0.1 cps-200% power
Reactor Trip Breaker Indication	Cable Spreading Room	OPEN-CLOSE
Reactor Coolant Cold Leg Temperature	1C43	212-705°F
Pressurizer Pressure	1C43	0-4000 psia
Pressurizer Level	1C43	0-360 inches
Steam Generator Pressure	1C43	0-1200 psig
Steam Generator Level	1C43	-401 to +63.5 inches

Additional communications capability is provided at the control stations for use in accomplishing these functions.

It is possible to maintain the plant in a hot shutdown condition from these alternate locations until access is permitted back into the Control Room. If required, the plant can then be placed in a cold shutdown condition from the Control Room.

Instrumentation required by the reactor operator to monitor key safety parameters from outside the Control Room is specified in the Technical Specifications.

7.7 CONTROL ROOM ANNUNCIATION

Visual and audible alarm units are incorporated into the control boards to warn the operator if limiting conditions are approached or off-normal conditions exist for any system. These visual and audible units are supplied in two stages: (1) annunciator enclosures supplied as part of the original installation; and (2) status alarm panels.

Status alarm panels are supplied in the same area as the existing annunciator panels to provide for additional alarm points. The panels furnish annunciation for one or more systems. A "summary alarm" window is included in the existing annunciator for each one of the systems contained in the status panels. The "summary alarm" has "reflash" capability to warn of additional incoming alarms on the status panel and does not clear until all status alarms are cleared. These status panels are mounted on the associated system panel or immediately adjacent to it.

A list of all annunciator windows, legends, and alarm initiating devices is maintained in controlled BGE drawings.

7.8 COMMUNICATIONS

7.8.1 DESIGN BASIS

A communication system with multiple redundancy has been provided to ensure availability and ease of operation.

7.8.2 COMMUNICATION SYSTEM DESCRIPTION

The communication system consists of six subsystems:

a. Plant Public Address (PA);
b. Commercial Telephone;
c. Sound-powered phones for plant use;
d. Sound-powered phones for emergency use;
e. Microwave system; and,
f. Radio telephone system.

7.8.2.1 Public Address System

The primary plant PA system utilizes the site-installed Northern Telecom administrative telephone system. The site is divided into five zones. Each zone can be accessed individually from any telephone on the site. An "ALL CALL" is available on certain site telephones that allows all five zones to be accessed simultaneously. Priority paging is also available on certain telephones that allows any individual zone or "ALL CALL" to be accessed while also overriding any non-priority page in progress. The Control Room has telephones that are able to access all five zones simultaneously and override any non-priority or priority page in progress. The plant emergency alarms are also generated by this equipment. The primary plant PA system is powered by diesel-backed instrument bus feeder 2Y1081 via 1X61 and 1P61.

7.8.2.2 Commercial Telephones

The site administrative telephone system is located in the North Service Building. This is a stand-alone system with alternative call routing that allows the site to be independent of the C&P Prince Frederick public exchange. This telephone system allows an individual to place a call within the plant, throughout BGE, or outside the company. The system is capable of routing calls throughout the Prince Frederick, Baltimore, or Annapolis public exchanges on C&P-provided facilities. The telephone system can also route calls to the public exchanges via the BGE microwave facilities into Baltimore. The Control Room has the capability of activating all remote links simultaneously in order to complete notification requirements for all agencies identified in the Emergency Response Plan.

The administrative telephone system receives its power from Motor Control Center MCC-101AT. Power is, therefore, available from either the main generator or the emergency diesel generator. The telephone system also has an emergency battery backup located in the North Service Building to ensure a communication system independent of the plant status.

7.8.2.3 Sound-Powered Phone (Plant Use)

The sound-powered phone system is set up in portions of the plant where unbroken communications are needed for certain operations or maintenance. The system consists of a hard wired network with covered jacks at various stations. Phones, headsets, and handsets with extension cords are taken to these stations for remote operations and control communications.

7.8.2.4 Sound-Powered Phone (Emergency Use)

A backup system, completely redundant and maintaining physical separation from the first provides communications capability between the Control Room and areas of the plant, including the interior of containment in the event of loss of normal communications during a fire.

7.8.2.5 Microwave Communication System

Automatic ringdown phones are located in the Control Room and the 500 kV switchyard and are connected to the microwave system. The signals are sent to the antenna in the 500 kV switchyard and are transmitted via microwave radio propagation to the electric system load dispatcher at the Electric Operations Building in Baltimore.

The microwave system also relays radio telephone communications to radio-based stations and microwave receivers off site. Some of the normal telephone traffic between Calvert Cliffs and Baltimore is handled by the microwave system. There is one microwave channel coming into Calvert Cliffs from the Electric Operations Building, which is used by the load dispatcher to contact all BGE generating stations simultaneously. This "ALL CALL" channel does not carry voice communications away from the plant.

7.8.2.6 Radio Telephone System (Plant Use)

The radio telephone system is a system of base stations and repeaters at the plant linked to the Emergency Operations Facility and base stations remote from the plant. The microwave communication system links the onsite and remote equipment. A primary radio system capable of plant wide radio communications, including communications within each containment, is installed. Through use of repeaters, outside antennas and an indoor continuous antenna, communications among control consoles, hand-held portables and nearby mobile units is possible.

The Emergency Operations Facility and, state and local emergency response agencies are included in communications capabilities.

A single channel 150 MHz system is capable of direct radio contact with the Electric Operations Building. This capability is provided to ensure offsite communications in the unlikely event of a simultaneous commercial telephone line and microwave system failure.

7.8.3 RELIABILITY AND TESTING

Systems of the types described above are conventional and have a history of successful operation at existing BGE Plants. Most of these systems are in routine use and this will assure their availability. Those systems not frequently used are to be tested at periodic intervals to assure operability when required.

7.9 DELETED

7.10 AUXILIARY FEEDWATER ACTUATION SYSTEM

The AFAS starts the AFW pumps upon detection of very low level in either SG and it blocks AFW to a ruptured generator. Detection of very low level on any two of the four wide-range SG level signal channels on either SG will produce, after a time delay, signals to open the AFW pump turbine-driven steam supply valves, close the SG blowdown valves, and close the breaker of the motor-driven AFW pump. The AFW system valves are aligned to allow flow on start of the pumps. The time delay of the auto-start actuation signal is required to ensure that valid wide-range SG level signals exist. Spurious AFAS actuations can develop during reactor trip from high power due to the power dependent decalibration phenomena of the wide-range SG level indication system. A SG is identified as ruptured if ΔP exceeds a preset value on any two of the four SG pressure signal channels. Signals are generated to close the redundant AFW block valves in each flow leg which feeds the ruptured generator. The auto-start signals are sealed-in while the ruptured isolation signals are not. The independent (from RPS) initiation of the AFW system satisfies the Diverse AFAS requirement for mitigation of ATWS events, 10 CFR 50.62.

7.10.1 DESIGN BASIS

7.10.1.1 Conformance to Standards

The design of the auxiliary feedwater actuation systems and component parts was based on the applicable requirements of IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." Maximum consideration has been given to the following criteria consistent with the objectives of this document:

a. Single Failure

Any single failure within the protection system will not prevent proper protection system action when required.

b. Quality of Components and Modules

Components and modules used in the manufacture of the actuation systems exhibit a quality consistent with the nuclear power plant 40-year design life objective and with minimum maintenance requirements and low failure rates.

c. Channel Independence

The actuation systems include four redundant sensor subsystems and two redundant actuation subsystems. Independence has been provided between redundant subsystems or channels to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction. Independence has been obtained by:

1. Electrical Isolation

Electrical isolation has been provided between redundant channels, between sensor and actuation subsystems and between the auxiliary feedwater actuation system and ancillary equipment. Where electrical isolation is provided, an application of short circuit, open wire, ground, or potential does not inhibit a protective action as a result of the failure of the redundant system.

2. Physical Separation

   Physical separation has been maintained between redundant sensor subsystems, between sensor and actuation subsystems, and between redundant actuation subsystems by providing separate and isolated cabinets for each of the four sensor subsystems, and each of the two actuation subsystems. A minimum clearance of 3' is provided for each sensing point and its associated transmitter.

3. System Repair

   The system has been designed such that routine servicing and preventive maintenance can be performed without interface to normal plant operation or without loss of system function availability. Performance of these operations does not result in a simultaneous unavailability of both actuation subsystems. The system is mechanically and electrically divided into subunits or modules based on the following considerations:

   a) Standardization of subunits
   b) Minimization of interconnections and interwiring
   c) Interchangeability of subunits

   The subunits include associated equipment, such as indicating lights, pushbuttons, potentiometers, and selector switches.

7.10.1.2 Security and Annunciation

The ESFAS is designed to provide annunciation and indication of module withdrawal or loss of power. Withdrawal or loss of power to, a sensor module results in a trip signal to its associated two-out-of-four logic matrices. Sensor modules are not interlocked to prevent withdrawal of more than one module; however, withdrawal of two sensor modules of a common actuation signal will result in a trip of the associated actuation channels.

The doors of each cabinet are equipped with a lock; one key fits all doors. Contacts are provided for annunciation of an open door. A withdrawal of an actuation logic module will not result in a trip of that channel.

7.10.1.3 Seismic and Environmental Requirements

The auxiliary feedwater actuation systems are classified as Seismic Category I and are designed to withstand all simultaneous horizontal and vertical accelerations resulting from a Safe Shutdown Earthquake without loss of function.

The specifications for AFAS and emergency power system components incorporate the applicable seismic requirements for each component, including spectrum response curves for specific component location generated by the time-history method.

These components are qualified by either of the following methods:

In most cases, the supplier is required to qualify his equipment by calculation or testing, or a combination of both. This qualification is formally documented and submitted for approval.

In other cases, tests or calculations are performed by independent consultants or laboratories who submit a formal report. Acceptance of the equipment from the

supplier is contingent upon the proof of suitability as established by the results of those tests or calculations.

The choice of an analytical or experimental qualification procedure is determined by the size, shape, and structural or functional simplicity of the equipment in accordance with the criteria outlined in IEEE 344 "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations (1975)." Racks, panels, or other supporting structures are generally qualified by analysis, while bistable trip units and other modules are generally qualified through testing. Tests and calculations are performed following the guidelines of IEEE 344.

Analysis and/or tests have verified that the safety-related characteristics of components are maintained when those components are subjected to the worst-case environment postulated for the component location.

7.10.2 SYSTEM DESCRIPTION

The Auxiliary Feedwater Actuation System is shown on Figures 7-24A, B, C and D. The actuation system is divided into four sensor subsystems (sensor channels ZD, ZE, ZF and ZG), and two actuation subsystems (actuation channels ZA and ZB).

The cabinets of the Auxiliary Feedwater Actuation System are appropriately tagged ZA, ZB, ZD, ZE, ZF and ZG, respectively, to distinguish between channels.

7.10.2.1 Sensor Subsystems

The sensor subsystems monitor redundant and independent process variables and trip when the variables reach unsatisfactory levels. Physical locations of the sensors are shown on the instrument location drawings prepared from the plant general arrangement drawings. Each of the sensor subsystems consists of one sensor channel of the following process variables:

a. Wide Range SG Level (SG 11)
b. Wide Range SG Level (SG 12)
c. SG Pressure (SG 11)
d. SG Pressure (SG 12)

7.10.2.2 Actuation Subsystems

The two redundant and independent actuation subsystems monitor isolated sensor subsystem trip outputs and, by means of coincidence logics, determine whether a protective action is required. On detection of very low level in either SG on at least two of the four sensor channels, auto-start of AFW takes place and the SG blowdown valves close. One AFW Pump Turbine Drive steam supply valve is opened by each actuation channel. One actuation channel also initiates closure of the breaker of the motor-driven AFW pump. The auto-start signal is sealed in.

There is a bypass valve to each turbine steam supply valve. To avoid turbine overspeed on startup, initial steam flow is controlled by first opening the small by-pass valve, and then the main supply valve after an appropriate time delay.

It should be noted that at steaming rates corresponding to greater than 10% power, the velocity head in the vicinity of the wide-range level transmitter variable leg SG tap induces a significant offset in the transmitter output. The indicated level reads lower than actual. The magnitude of this effect is a function of power level. This phenomenon is factored into the safety analyses.

On detection of a d/p between the SGs in excess of the setpoint value on at least two of the four sensor channels, signals are generated to close redundant block valves in the AFW flow paths to the low pressure SG. Each flow path contains an "A" and a "B" channel block valve. This feature ensures that AFW flow to a ruptured SG is terminated in order to prevent return to power which could be caused by excessive cooldown of the primary system. The AFW block signal is not sealed-in, but a maximum hysteresis has been set in such that if the d/p approaches zero after initial AFW isolation the AFW block signal will reset. Seal-in of the block signal is not possible due to the scenario of remote probability described below in which first one and then the other SG is identified as ruptured by d/p.

A d/p first in one direction and then in the other occurs with MSLB just downstream of one of the MSIVs and failure of the other MSIV to close.

Performance of the AFW System is factored into the analyses for MSLB, excess load, feed line break, and loss of main feed water. The minimum auto-initiation setpoint, the minimum AFW flow controller setpoint, and the maximum system response times ensure that sufficient water is delivered to the SGs during the first 10 minutes following a feed line break or loss of main feed water.

The maximum auto-initiated setpoint, the maximum AFW flow controller setpoint, the minimum response times, and the AFW block valves ensure that:

a. The reactor does not return to power following an MSLB, and
b. The probability of initiating SIAS or draining the pressurizer in the first 10 minutes after initiation of AFW is very small.

7.10.3 SYSTEM SURVEILLANCE

7.10.3.1 Remote Annunciation

a. Tripped Sensor Channel
b. AFAS "A" Actuated
c. AFAS "B" Actuated
d. Steam Line 11 Rupture
e. Steam Line 12 Rupture
f. Motor System Line-up Improper
g. Turbine System Line-up Improper
h. SG 11 Line-up Improper
i. SG 12 Line-up Improper
j. AFW 11 Flow to Break
k. AFW 12 Flow to Break
l. Turbine System No Flow
m. Motor System No Flow
n. Air Accumulator Low Pressure
o. AFW System Suction Pressure Low
p. Excess Flow
q. Door Open/Module Withdrawn
r. Loss of Power Supply
s. Sensor Channel Bypassed

7.10.3.2 Local Sensor Channel Surveillance

Each module or subunit is equipped with its associated indicating lights. Typical functions to be indicated are:

a. Bistable trip
b. Power supply available

All indicating lights are visible with the cabinet doors closed. Features are provided for manually checking bulb function.

7.10.3.3 Local Actuation Channel Surveillance

Each module or subunit is equipped with its associated indicating lights. Typical functions indicated are:

a. Tripped actuation subchannel
b. Power supply available

7.10.4 ELECTRICAL POWER SUPPLY

The four redundant 118 Volt, 60 Hz, vital sources of supply (Section 8.3.5) are utilized by the AFW actuation systems. Two vital sources provide power for a sensor subsystem and an actuation subsystem. The remaining two sensor subsystems receive power from the remaining vital sources. Physical and electrical isolation is maintained between the various redundant power supplies. Short circuit protection is provided at each system cabinet, and a trip of the protective device is indicated locally and annunciated in the Control Room.

7.10.5 SYSTEM EVALUATION

The AFW initiation, control, and power supply systems were designed in accordance with the Proposed IEEE Criteria No. 279, so that no single fault in components, units, channels or sensors will prevent ESFs operation.

The wiring is installed so that no single fault or failure, including either an open or shorted circuit, will negate minimum AFAS operation. Wiring for redundant circuits is protected and routed so that damage to any one path will not prevent minimum AFAS action. Sensors are piped so that blockage or failure of any one connection does not prevent AFAS operation.

The detailed design incorporates the following characteristics in order to counteract faults resulting in loss of power:

a. All redundant components are powered from separate busses;
b. The reliability of the 125 Volt DC and 120 Volt AC vital power busses used, as discussed in detail in Chapter 8;
c. Loss of power to a sensor channel results in bistable trip signals. Loss of power to an actuation channel will not produce actuation signals.

There are no AFAS instrumentation transmitters for which the trip setpoints are within 5% of the high or low end of the calibrated range, or within 5% of the overall instrument design range.

7.10.6 MANUAL TESTING FEATURES

7.10.6.1 Bistable Trip Test

Each bistable has built-in provisions for testing bistable operation. An adjustable voltage source is applied when a local pushbutton is depressed. While initiating

the test, the process variable input to the bistable is not interrupted. Local indicating lights and Control Room annunciators verify proper operation.

7.10.6.2 Actuation Channel Trip Test

Each coincidence two-out-of-four matrix includes a local independent pushbutton. The test with simultaneous presence of a sensor channel trip causes an output of the associated coincidence matrix and trips the actuation channel logic.

An unbypassed bistable trip signal will indicate as an input to the logic module. This provides a means of verifying that the bistable trip signal reaches the two-out-of-four logic module. Testing during refueling outages verifies total system operation.

7.11 ANTICIPATED TRANSIENT WITHOUT SCRAM

Anticipated Transient Without Scram is an anticipated operational occurrence followed by the failure of the reactor trip portion of the protection system. This protection system automatically initiates the operation of systems, including the reactivity control systems, which assure that specified fuel design limits are not exceeded as a result of anticipated operational occurrences. Some examples of these occurrences are: loss of power to all reactor coolant pumps in a unit, loss of load, and loss of offsite power (Section 14.1.1.1).

Protection against ATWS events is comprised of three elements: DSS, DTT, and a diverse AFAS. These are all requirements of 10 CFR 50.62. The first two, DSS and DTT, are discussed in this section; AFAS is discussed in Section 7.10.

7.11.1 DIVERSE SCRAM SYSTEM

The purpose of the DSS is to provide reactor trip capability that senses high pressurizer pressure and will function separately from the primary reactor trip system (Section 7.3).

7.11.1.1 Design Basis

The DSS provides diversity from the existing RPS, electrical independence from sensor output to the final actuation device, isolation of non-safety-related from safety-related circuits, testability at power, environmental qualification for anticipated operational occurrences, and a design to prevent against inadvertent actuation and challenges to other safety systems.

a. Facility electrical separation is maintained through the use of existing circuitry in the ESFAS sensor, logic, and relay cabinets, which provide physical separation for four sensor channels and two logic and relay channels.
b. Seismic concerns are met by utilizing existing equipment already in place within the ESFAS sensor, logic, and relay cabinets.
c. At-power testing is accomplished with the use of a bypass contactor in parallel to the existing CEDM motor-generator load contactor.
d. Redundancy is not required by 10 CFR 50.62. Both channels of the DSS must trip, opening both of the CEDM motor-generator load contactors in order to cause a reactor trip.
e. Diversity is provided between the RPS and the DSS through the use of equipment supplied by different manufacturers or designed differently to provide the same function. The DSS interrupts power to the CEDM power supplies by opening the motor-generator load contactors, while the RPS uses the reactor trip switchgear to interrupt CEDM power. The same sensors are used by the DSS and the RPS for pressurizer pressure which is acceptable by 10 CFR 50.62. Sensor output for DSS is made diverse from RPS, as specified by 10 CFR 50.62, through the use of an electronic isolator.
f. The DSS is powered by inverter feed which are AC power sources that also supply the RPS. The use of common power supplies is acceptable for the DSS and the RPS sensors as they are not within the scope of the ATWS rule, 10 CFR 50.62. The DSS power supplies for each of the four DSS protection channels are independently breakered and fused from a different vital bus. This isolates the DSS power supplies from each of the vital buses in order to prevent common mode failures.
g. Environmental qualification to accident conditions is provided for DSS equipment installed in the ESFAS cabinets.

7.11.1.2 System Description

The DSS is a four channel sensor system which through two-out-of-four logic inputs to two actuation channels. Each actuation channel opens one of the two load contactors on each CEDM motor-generator. Both load contactors on each CEDM motor-generator must open to cause a reactor trip.

The four sensor channels consist of pressurizer pressure sensors (PT-102A, B, C, D) and associated circuits. The output of the sensors, through isolators, provides pressure signals to four high-trip bistables in the ESFAS sensor cabinets. Each bistable provides channel trip annunciation, input to a two-out-of-four logic module in channel "A" of the ESFAS cabinet and input to a two-out-of-four logic module in channel "B" of the ESFAS cabinet. The logic modules energize a relay in each of the ESFAS relay cabinets to open the CEDM motor-generator load contactors. Both channels must actuate to initiate a reactor trip.

At-power testing is provided through the use of a bypass contactor for the channel in test. The bypass contactor is in parallel with the load contactor and prevents the loss of output when the load contactor opens during testing. Due to the fact that the DSS is not available while the system is in bypass, administrative control will limit the time that the system may remain in bypass.

Annunciation is provided on 1(2)C05 for both "DIVERSE SCRAM SYSTEM TRIP" and "DSS LOAD CONTACTOR BYPASSED."

7.11.2 DIVERSE TURBINE TRIP

Main turbine trip circuitry consists of four safety-related instrument control channels which sense CEDM power bus undervoltage. The DSS provides a diverse means of deenergizing the CEDM power bus. This satisfies the ATWS requirement for diverse means of main turbine trip.

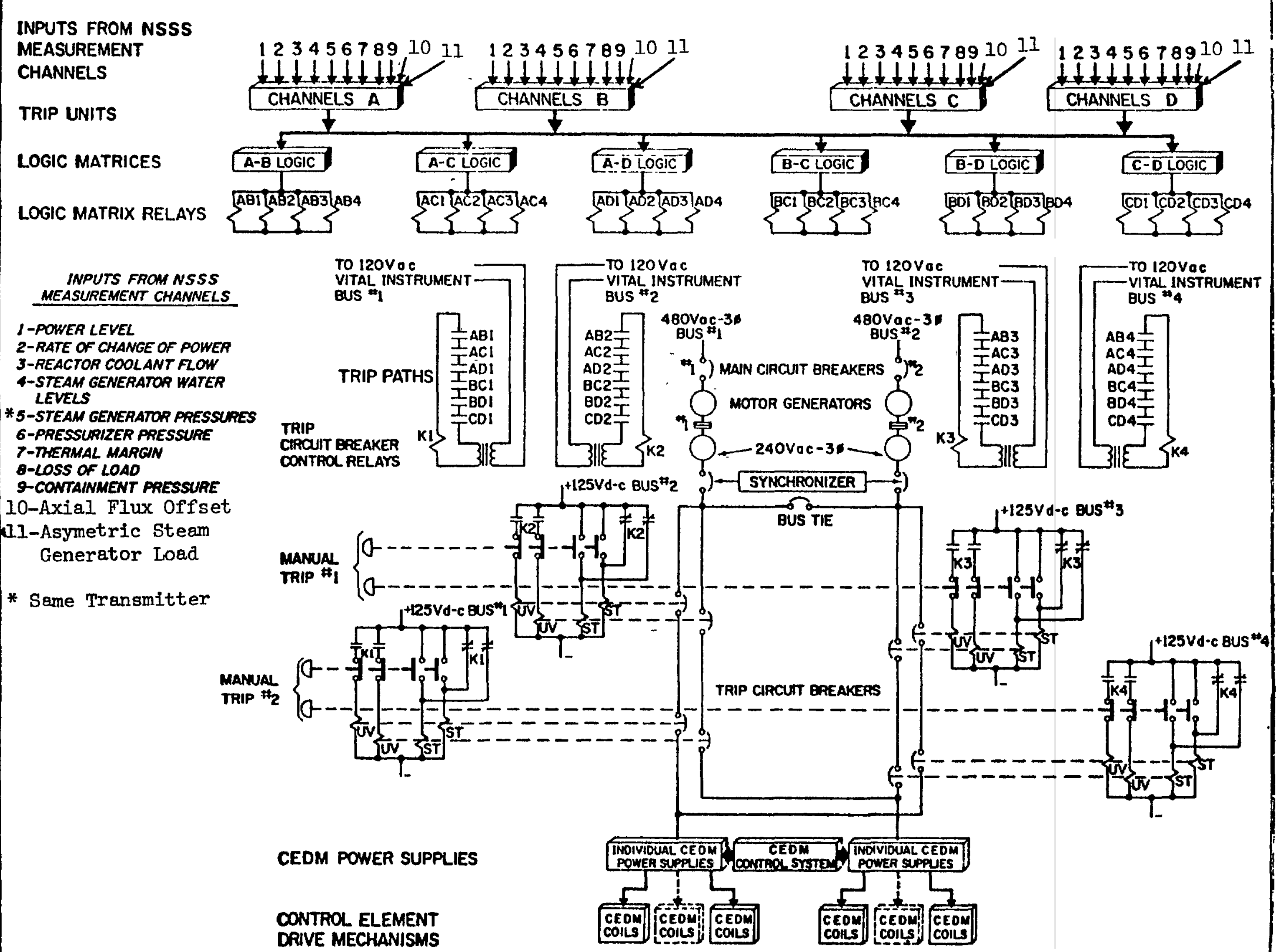
7.12 ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY

Equipment used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability for the time it is required to operate, under all service conditions postulated to occur during its installed life. This requirement, which is embodied in General Design Criteria 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment are contained in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," NUREG-0588, "Interim Staff Position of Environmental Qualification of Safety-Related Electrical Equipment" and "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines).

On January 14, 1980, Nuclear Regulatory Commission (NRC) issued Inspection and Enforcement Bulletin (IEB) 79-01B which included the DOR Guidelines and NUREG-0588 (for comment version). Subsequently, on May 23, 1980, Commission Memorandum and Order CLI-80-21 was issued and stated that the DOR Guidelines and portions of NUREG-0588 (for comment version) form the requirements that Calvert Cliffs must meet regarding environmental qualification of safety-related electrical equipment. Supplements to IEB 79-01B, issued on February 29, September 30, and October 24, 1980, NUREG-0588, Revision 1, dated July 1981 and NRC Generic Letter 82-09, dated April 20, 1982 provide further clarification and definition of the NRC's requirements.

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR Part 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for Calvert Cliffs may be qualified to the criteria specified in either the DOR Guidelines or NUREG-0588, (for comment version) except for replacement equipment. Replacement equipment installed subsequent to February 22, 1983 must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, Revision 1, unless there are sound reasons to the contrary.

The Calvert Cliffs Environmental Qualification Program complies with these requirements.

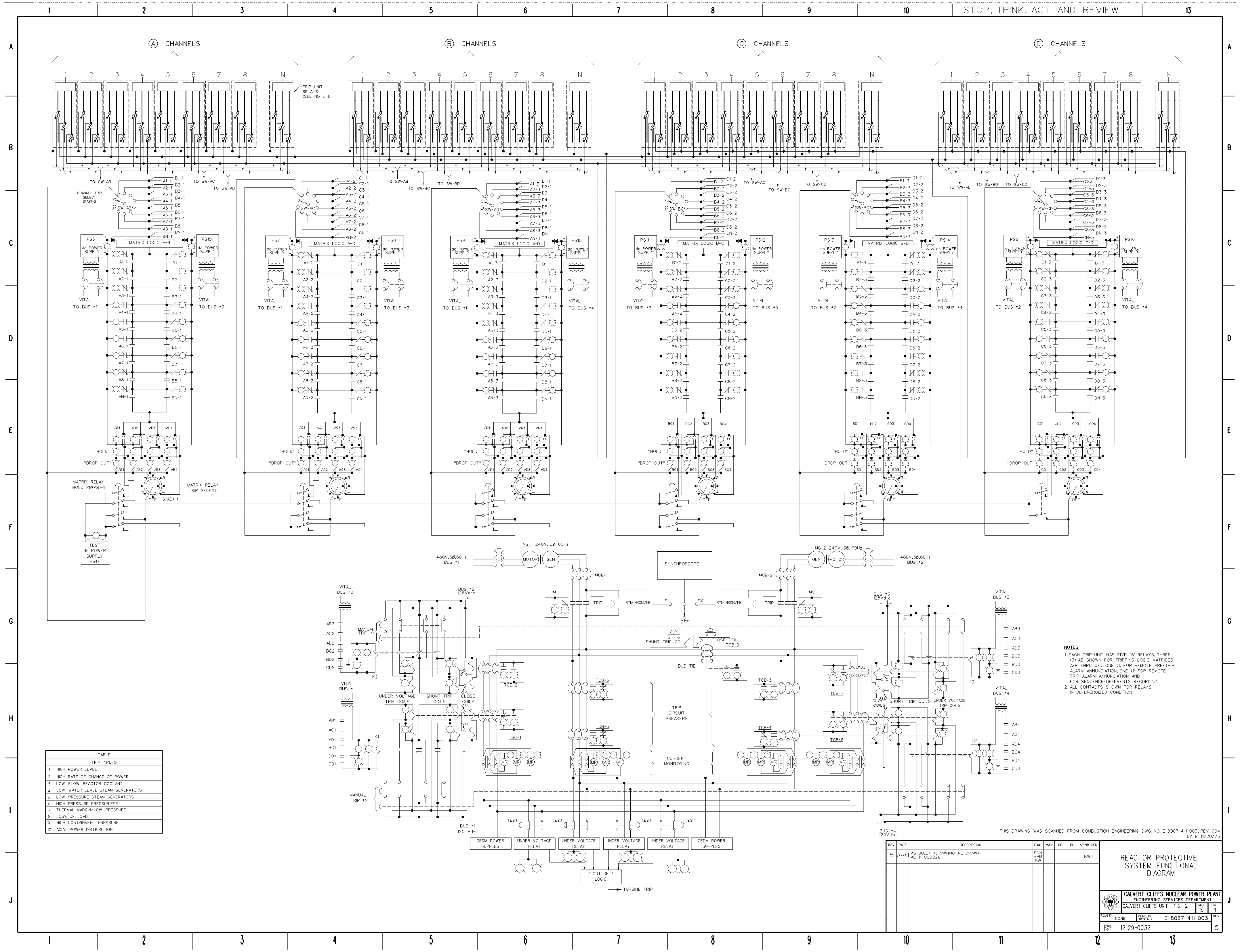


BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

Reactor Protective System - Block Diagram

Figure 7-1

FIGURE 7-2 REACTOR PROTECTIVE SYSTEM - FUNCTION DIAGRAM



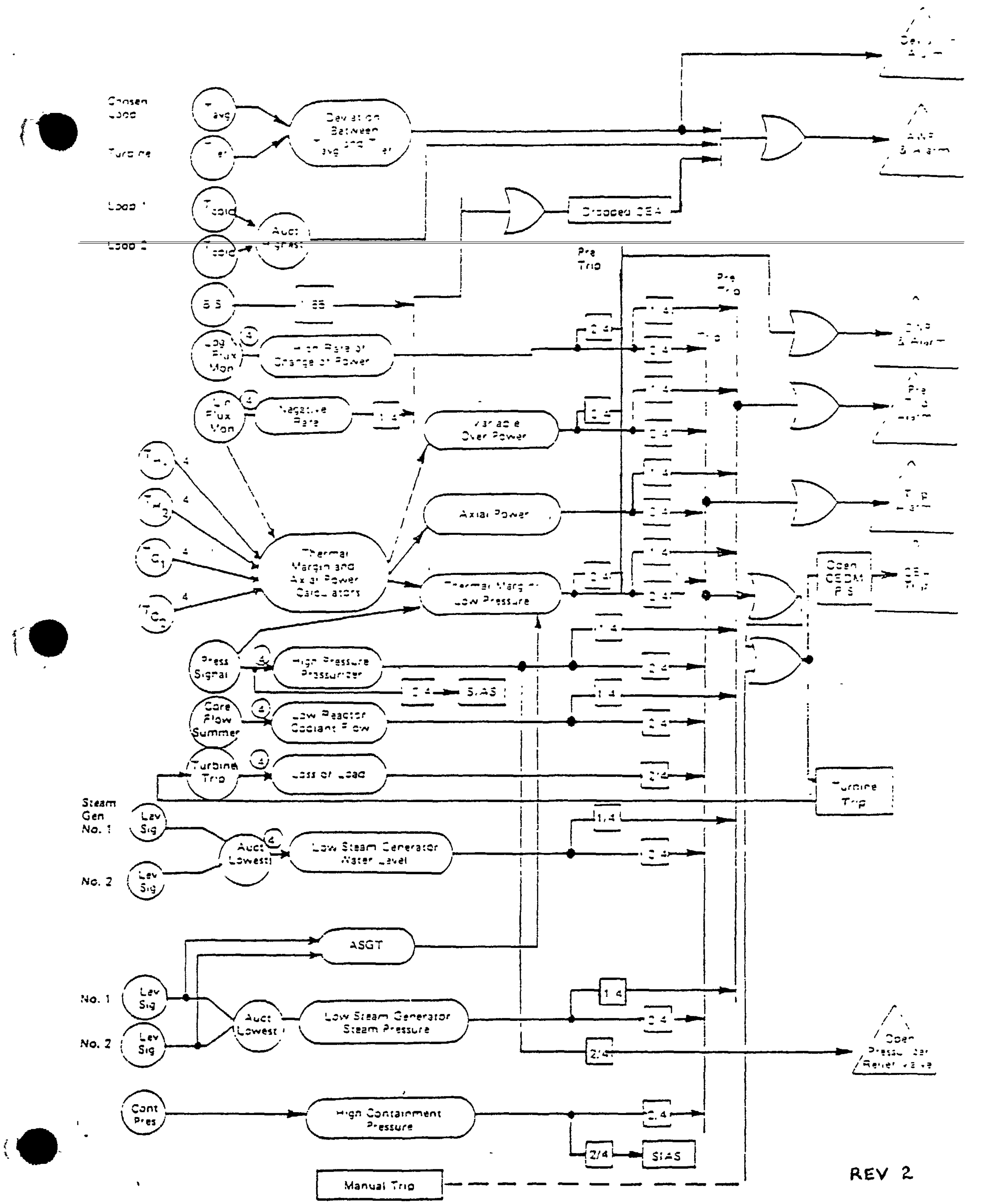
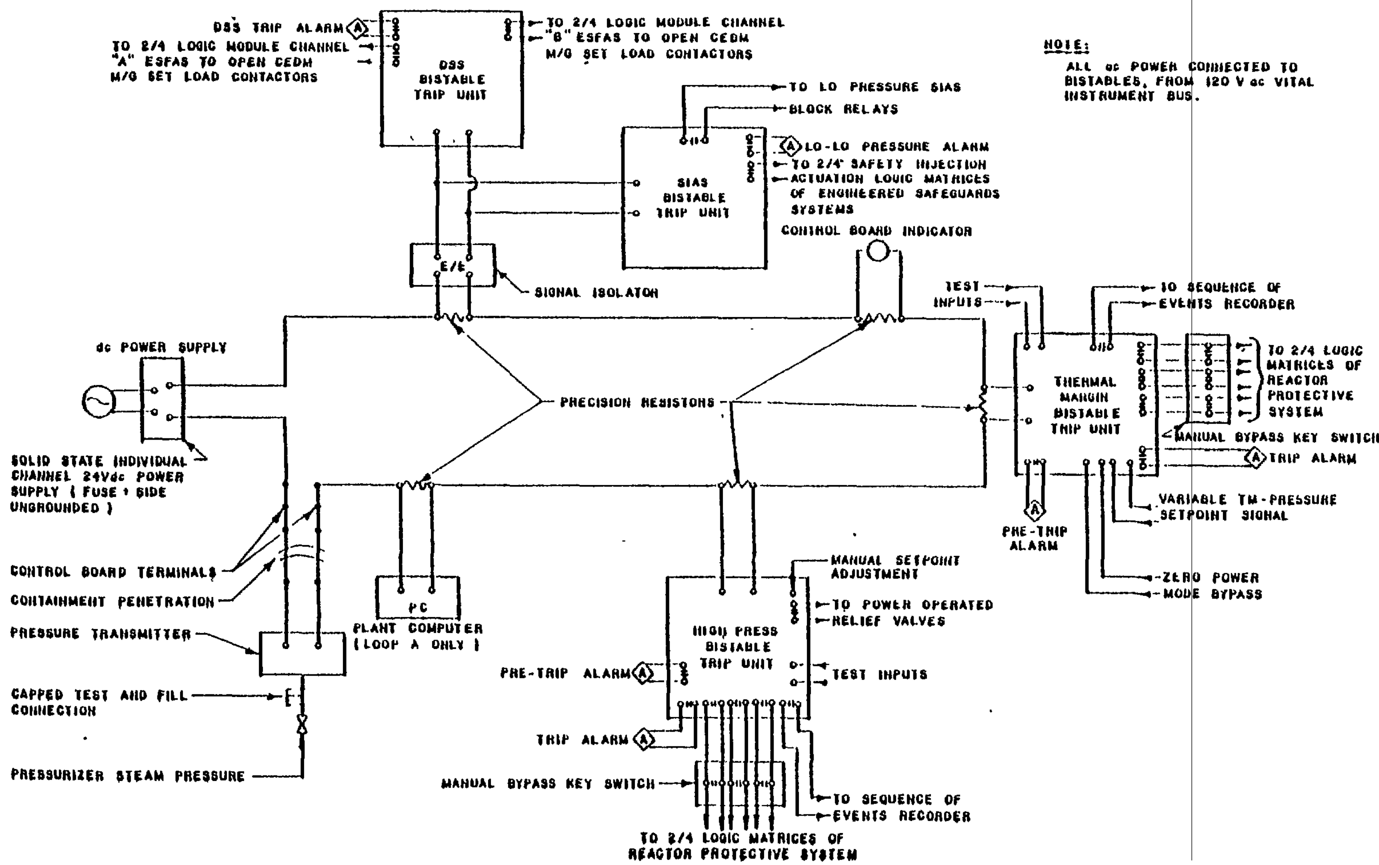


Figure 7-2A REACTOR PROTECTIVE SYSTEM INTERFACE LOGIC DIAGRAM

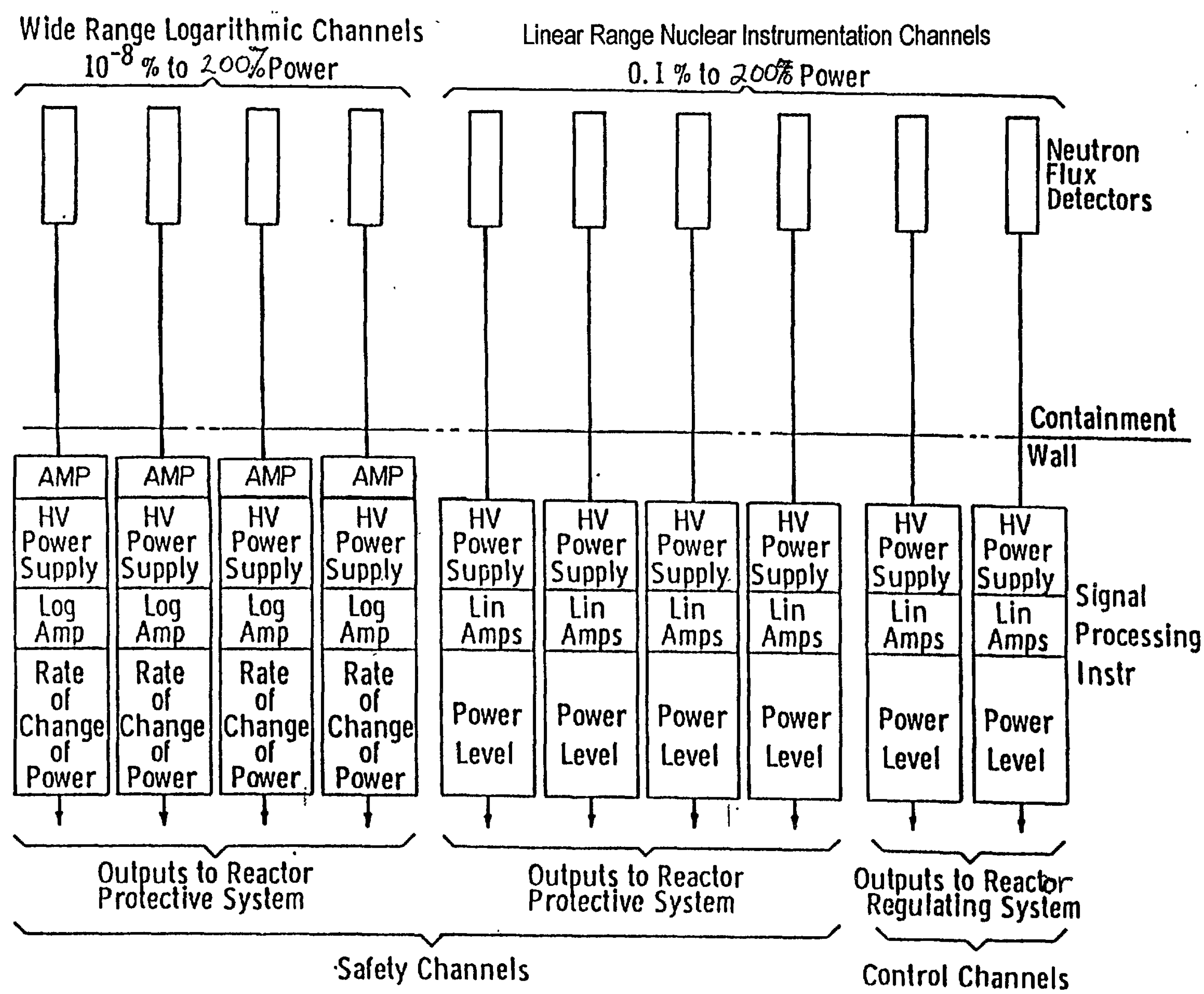


BALTIMORE
GAS & ELECTRIC CO.
Calvert Cliffs
Nuclear Power Plant

Pressurizer Pressure Measurement Channel
Functional Diagram

Figure
7-3

Rev. 15 12/93



BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

Ex-Core Nuclear Instrumentation

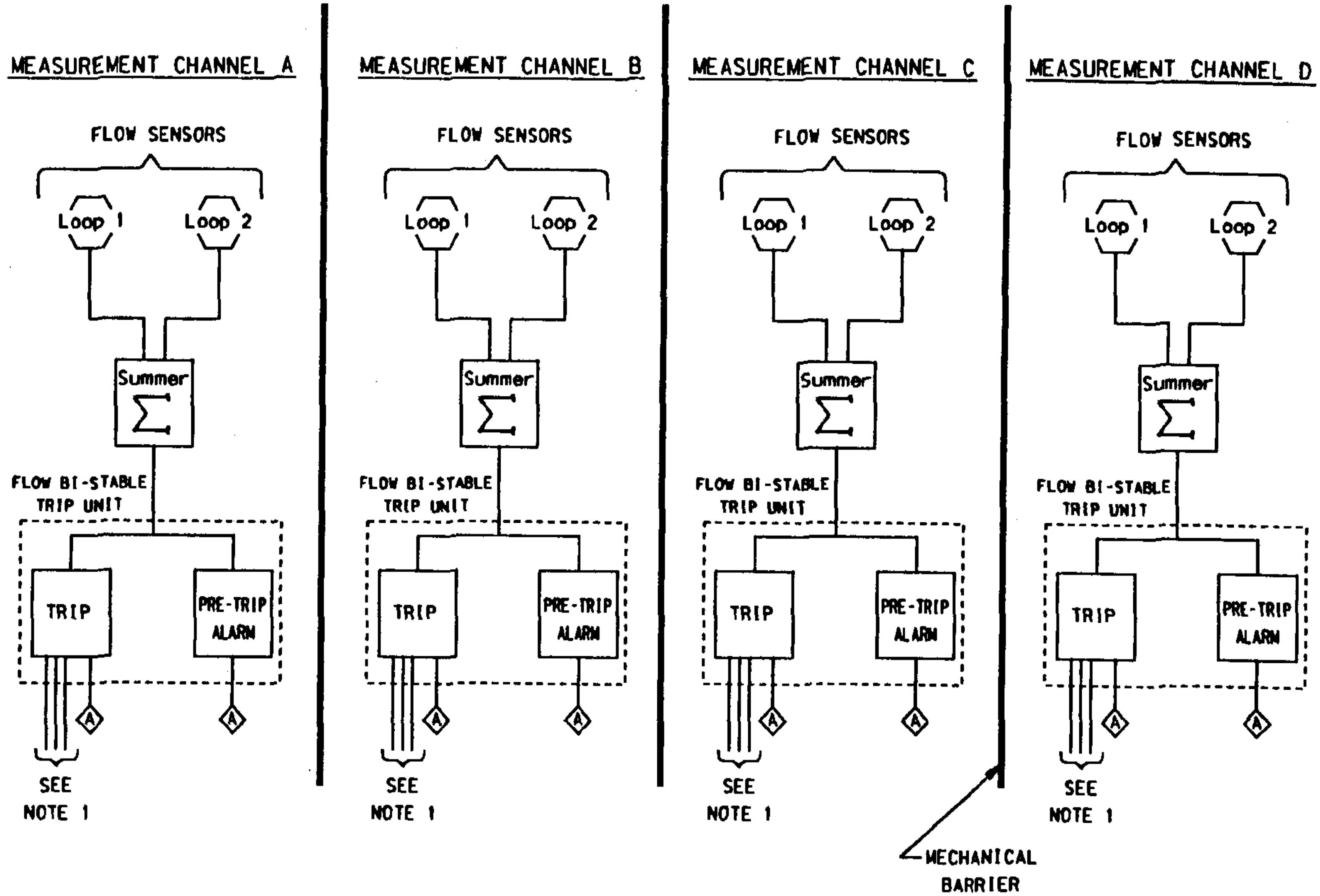
Figure 7-4
Rev. 20

Calvert Cliffs Nuclear Power Plant

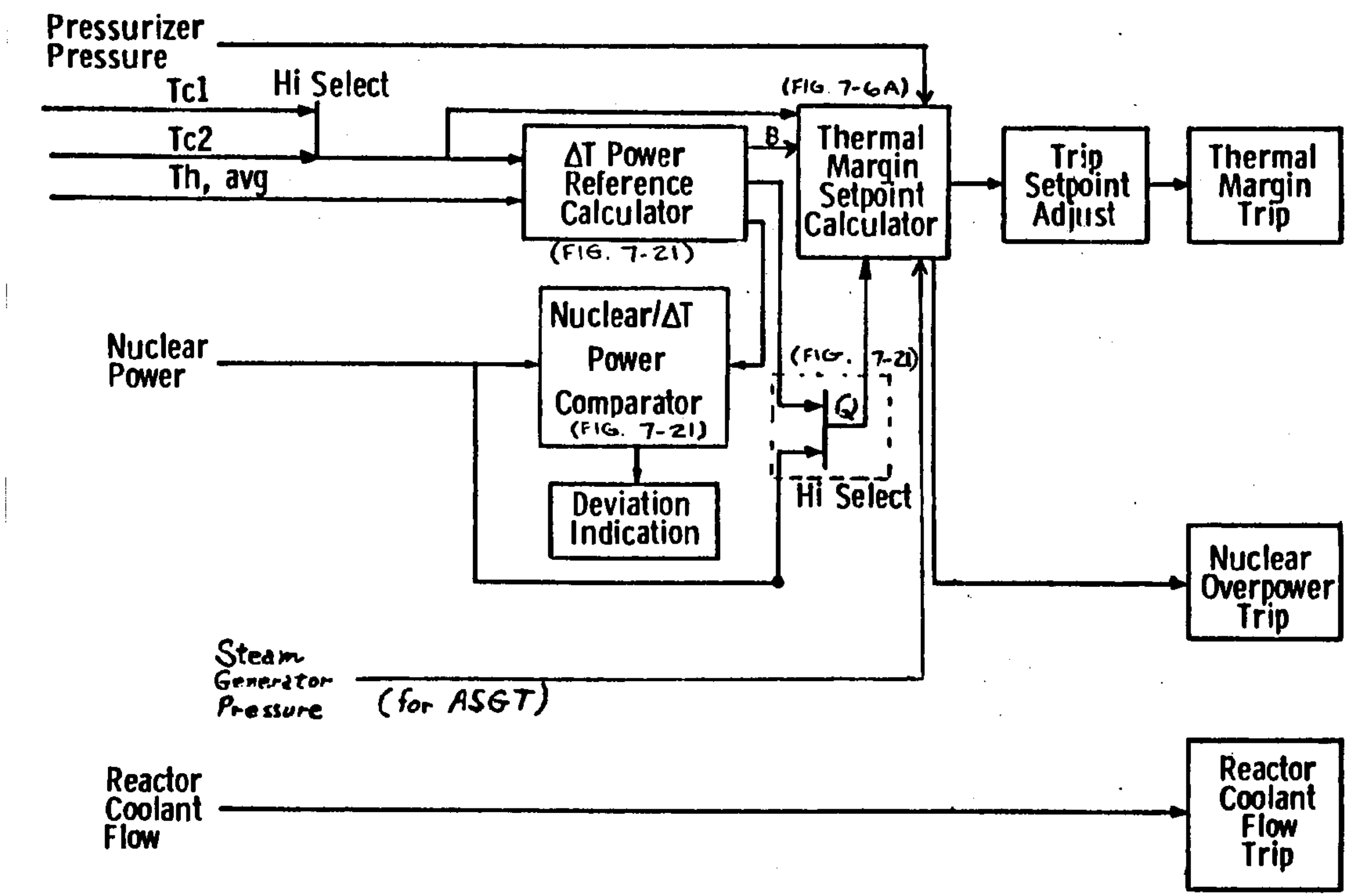
LOW FLOW PROTECTIVE SYSTEM – FUNCTIONAL DIAGRAM

Figure 7-5

Rev. 31



Note 1: To 2 out of 4 Logic Matrices Reactor Protective System Functional Diagram



Calvert Cliffs Nuclear Power Plant

THERMAL MARGIN/LOW PRESSURE TRIP CHANNEL - BLOCK DIAGRAM

Figure 7-6

Rev. 31

Calvert Cliffs
Nuclear Power Plant

THERMAL MARGIN/LOW PRESSURE TRIP
FUNCTION DIAGRAM

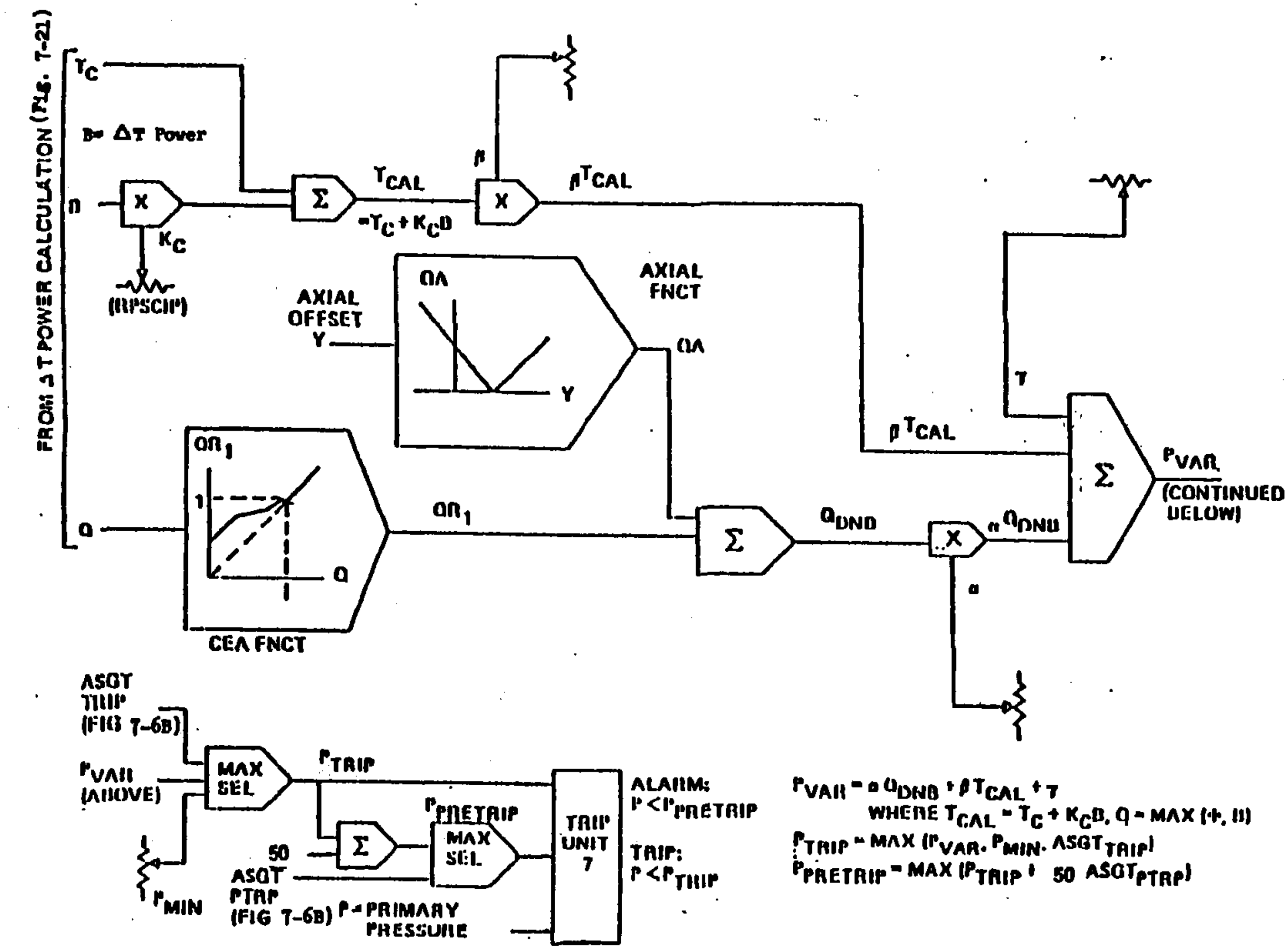


Figure 7-6A
Rev. 31

IF $|\Delta PSG| > ASGT_{TRIP}$ SET, $ASGT_{TRIP}$ = 2500 PSI. ELSE $ASGT_{TRIP}$ = 0 PSI

IF $|\Delta PSG| > ASGT_{PRETRIP}$, $ASGT_{PTRP}$ = 2500 PSI. ELSE $ASGT_{PRETRIP}$ - 0 PSI

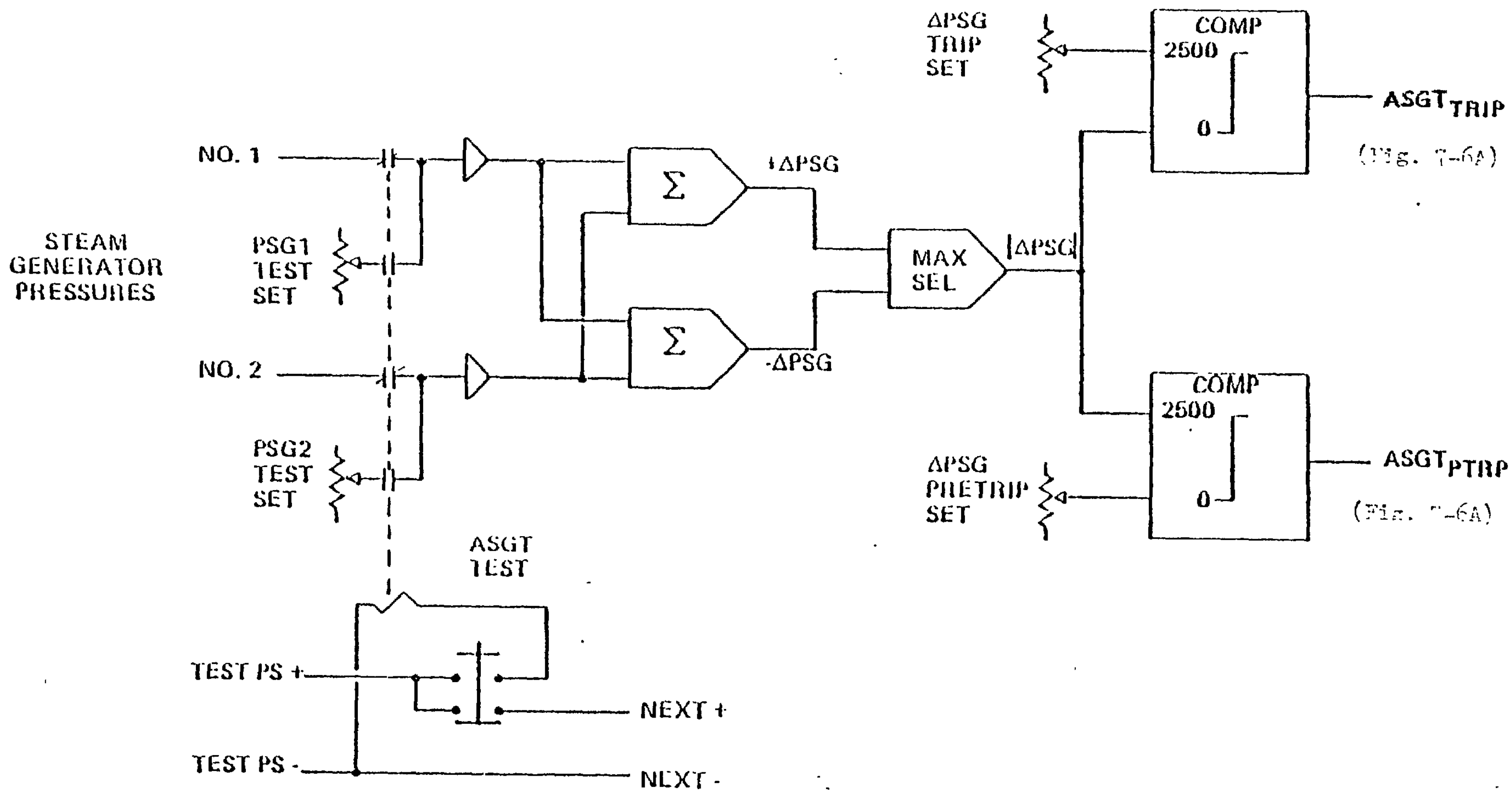
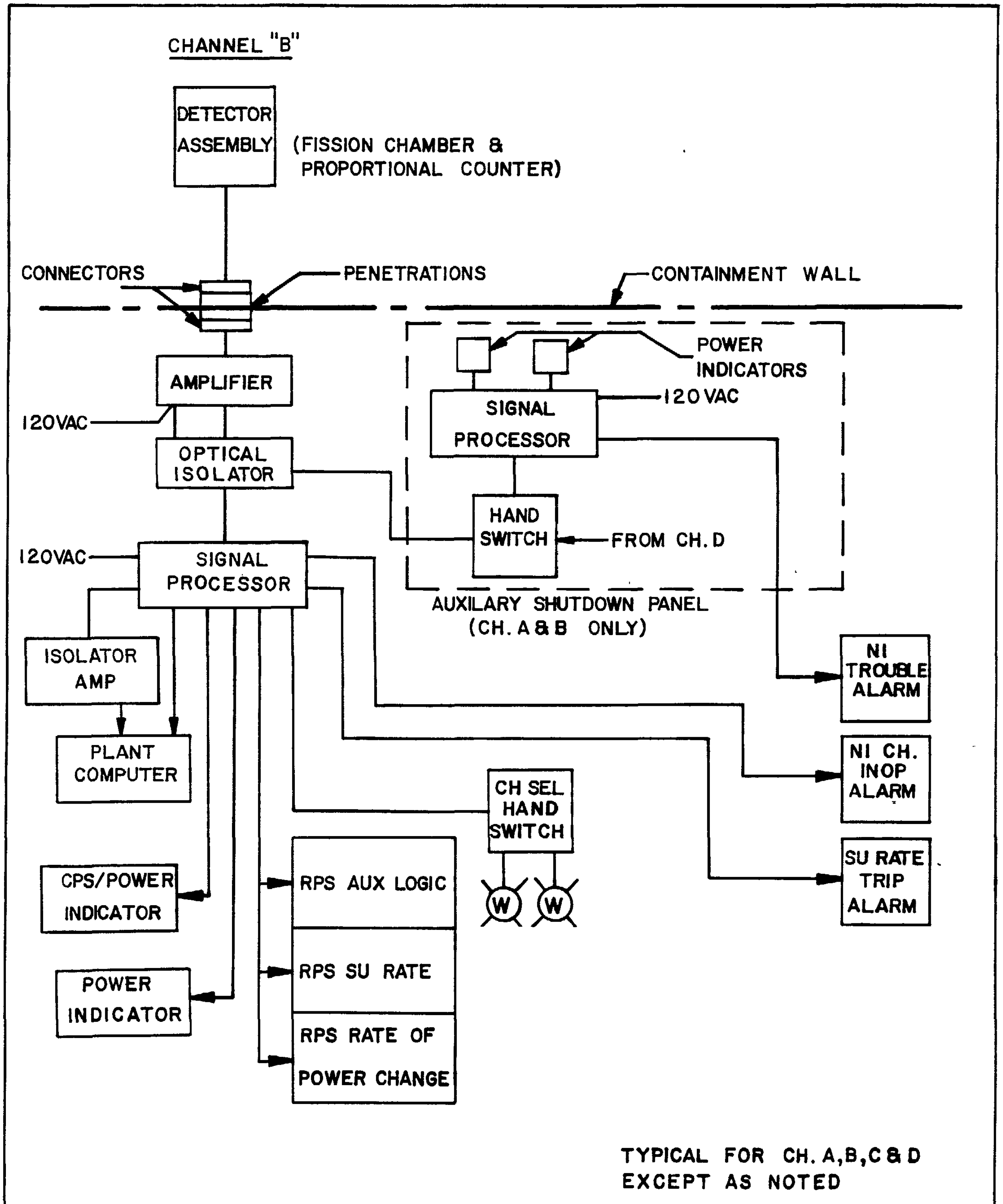
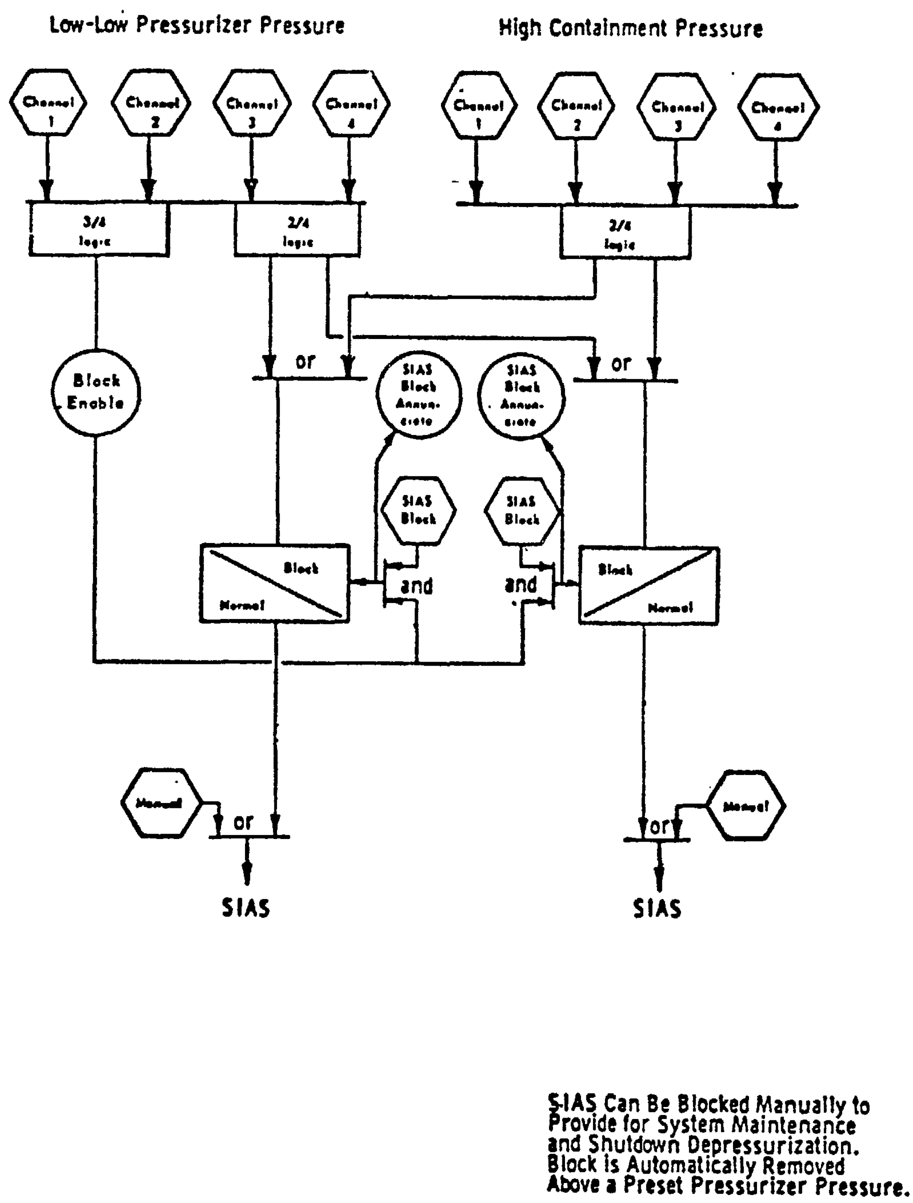


FIG. 7-6B Asymetric Steam Generator Trip
(ASGT) - Functional Diagram



BALTIMORE GAS & ELECTRIC CO. CALVERT CLIFFS NUCLEAR POWER PLANT	NEUTRON FLUX MONITORING SYSTEM STARTUP AND LOGARITHMIC RANGE CHANNELS	FIGURE 7-7

REV. 11 1/91



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Safety Injection Actuation Signal	Figure 7-9

FIGURE 7-10 LOGIC DIAGRAM - ENGINEERED SAFETY FEATURES ACTUATION SYSTEM UNIT 1 (Sheet 1)

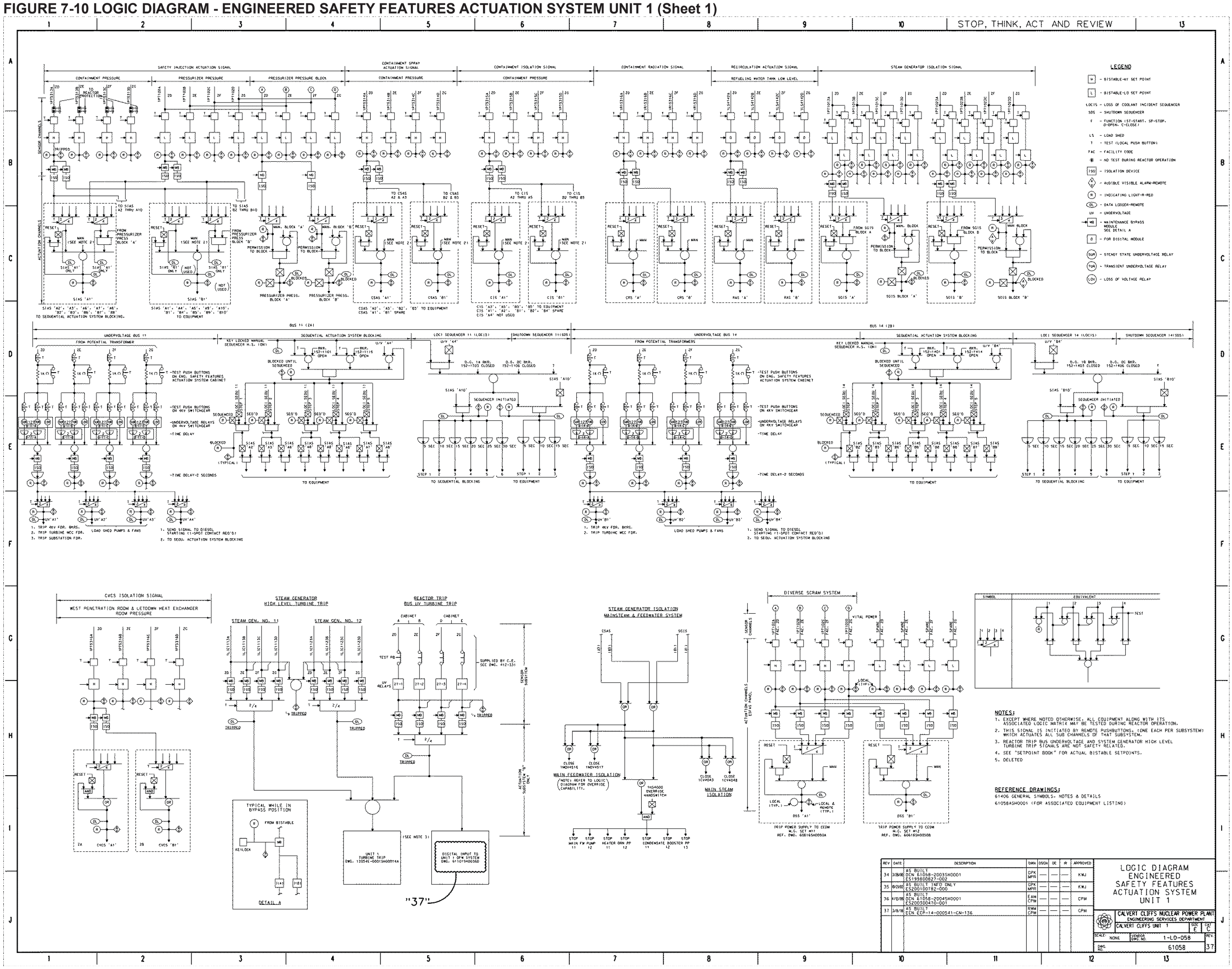
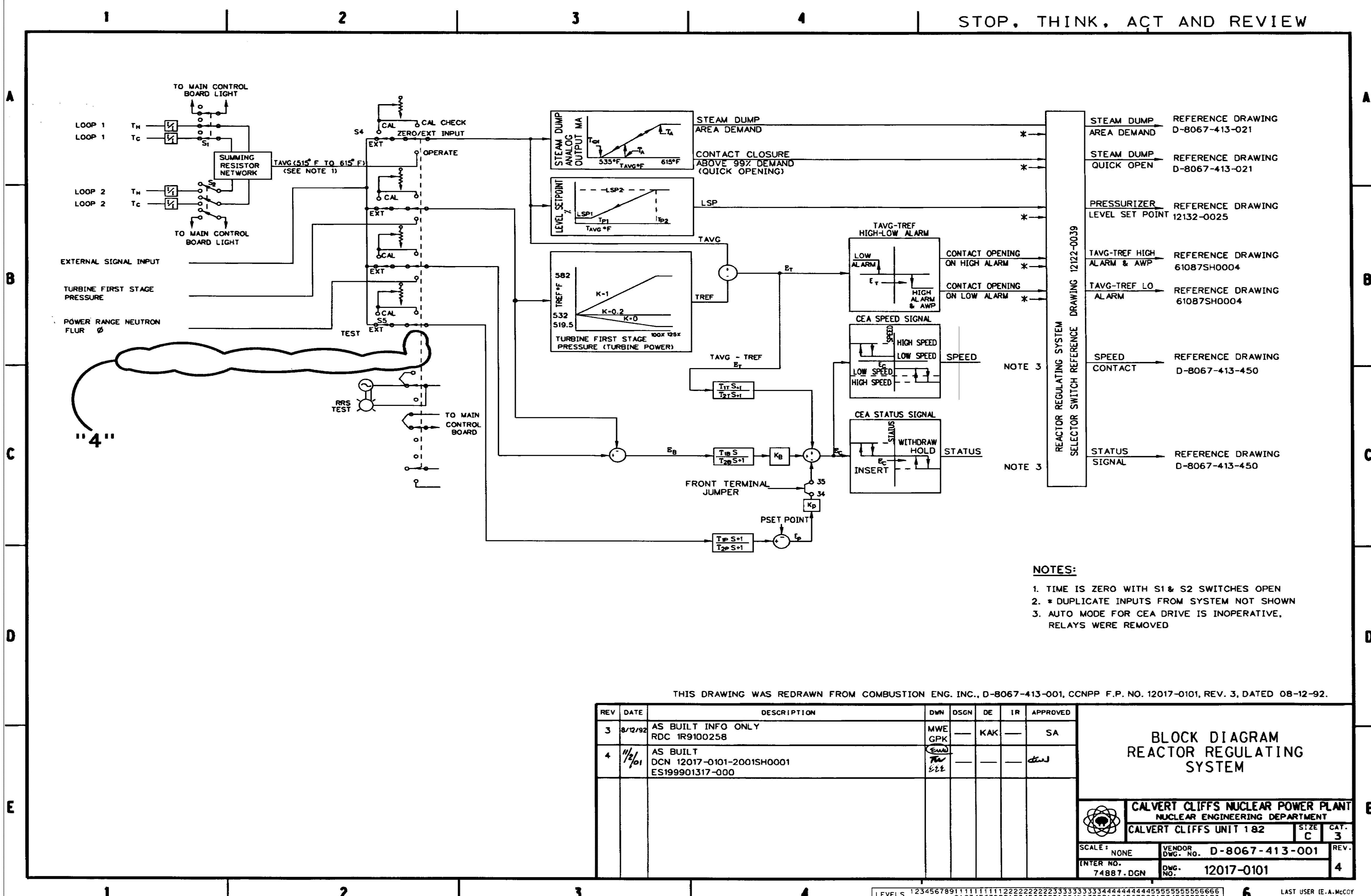


FIGURE 7-10 LOGIC DIAGRAM – ENGINEERED SAFETY FEATURES ACTUATION SYSTEM – UNIT 1 (Sheet 2)



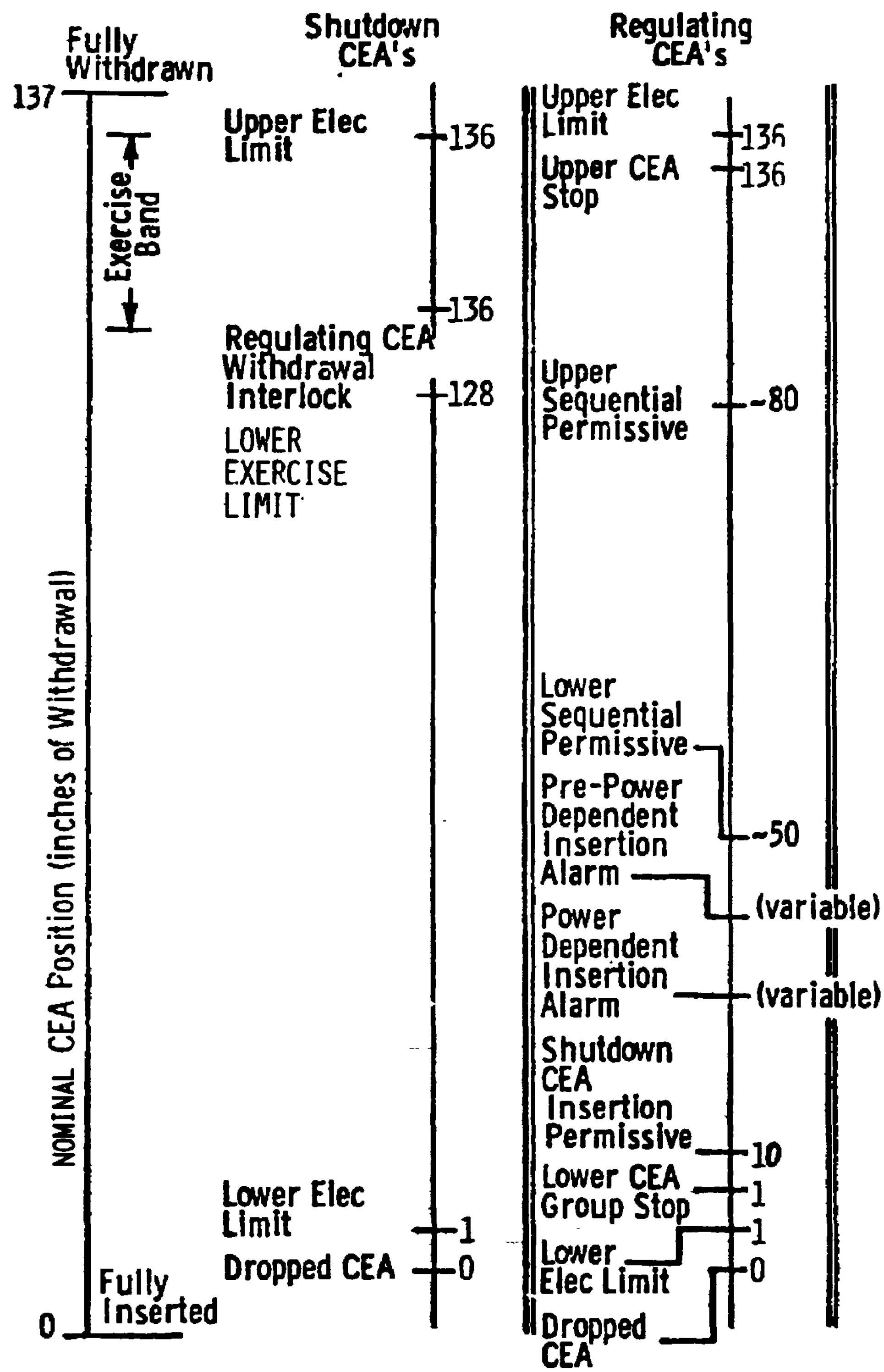
NOTES:
1. TIME IS ZERO WITH S1 & S2 SWITCHES OPEN
2. * DUPLICATE INPUTS FROM SYSTEM NOT SHOWN
3. AUTO MODE FOR CEA DRIVE IS INOPERATIVE, RELAYS WERE REMOVED

THIS DRAWING WAS REDRAWN FROM COMBUSTION ENG. INC., D-8067-413-001, CCNPP F.P. NO. 12017-0101, REV. 3, DATED 08-12-92.

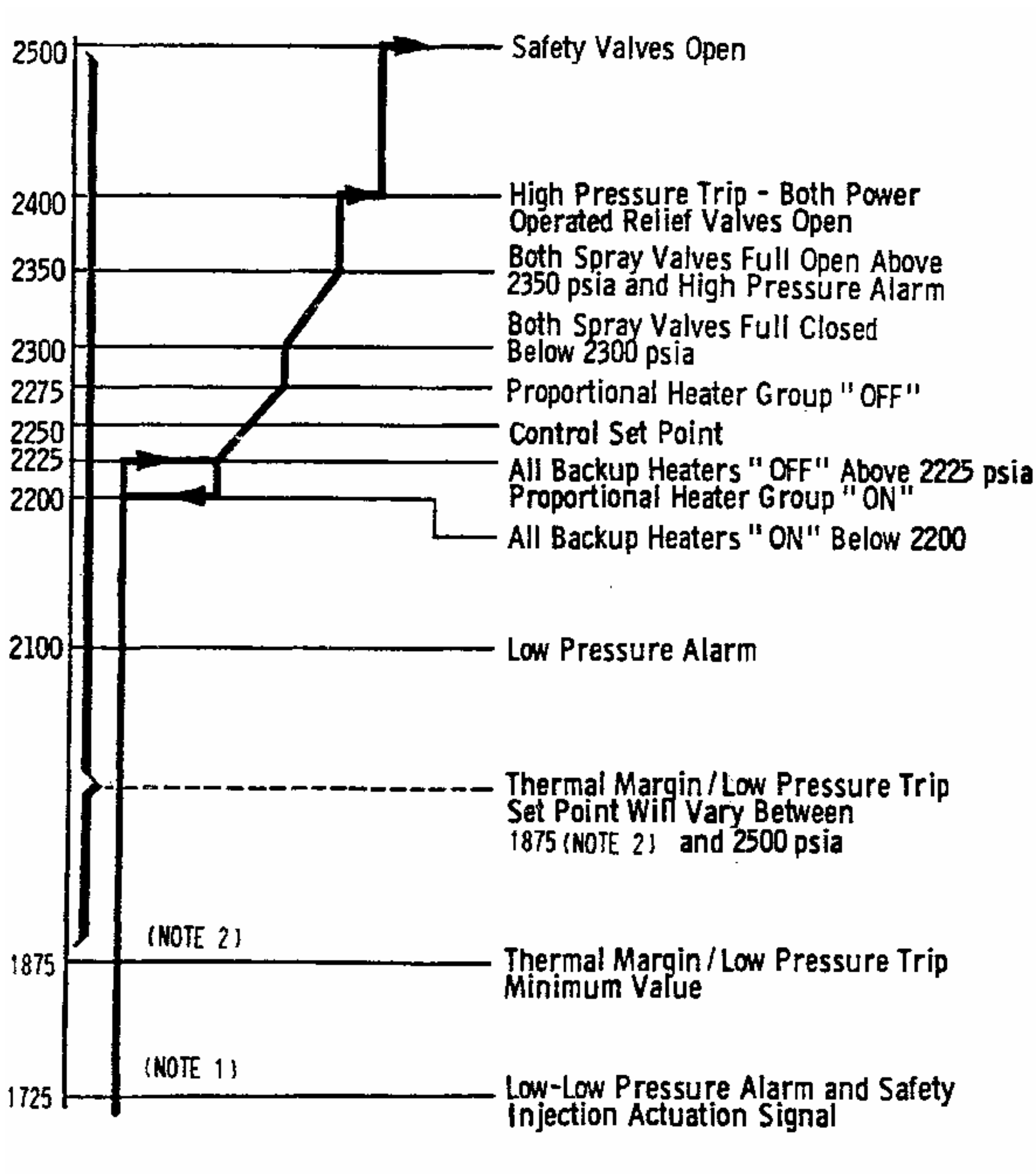
REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
3	8/12/92	AS BUILT INFO ONLY RDC 1R9100258	MWE GPK	—	KAK	—	SA
4	11/2/01	AS BUILT DCN 12017-0101-2001SH0001 ES199901317-000		—	—	—	

BLOCK DIAGRAM
REACTOR REGULATING SYSTEM

CALVERT CLIFFS NUCLEAR POWER PLANT
NUCLEAR ENGINEERING DEPARTMENT
CALVERT CLIFFS UNIT 1 & 2
SIZE C CAT. 3
SCALE: NONE
VENDOR DWG. NO. D-8067-413-001
REV. 4
INTER NO. 74887.DGN
DWG. NO. 12017-0101



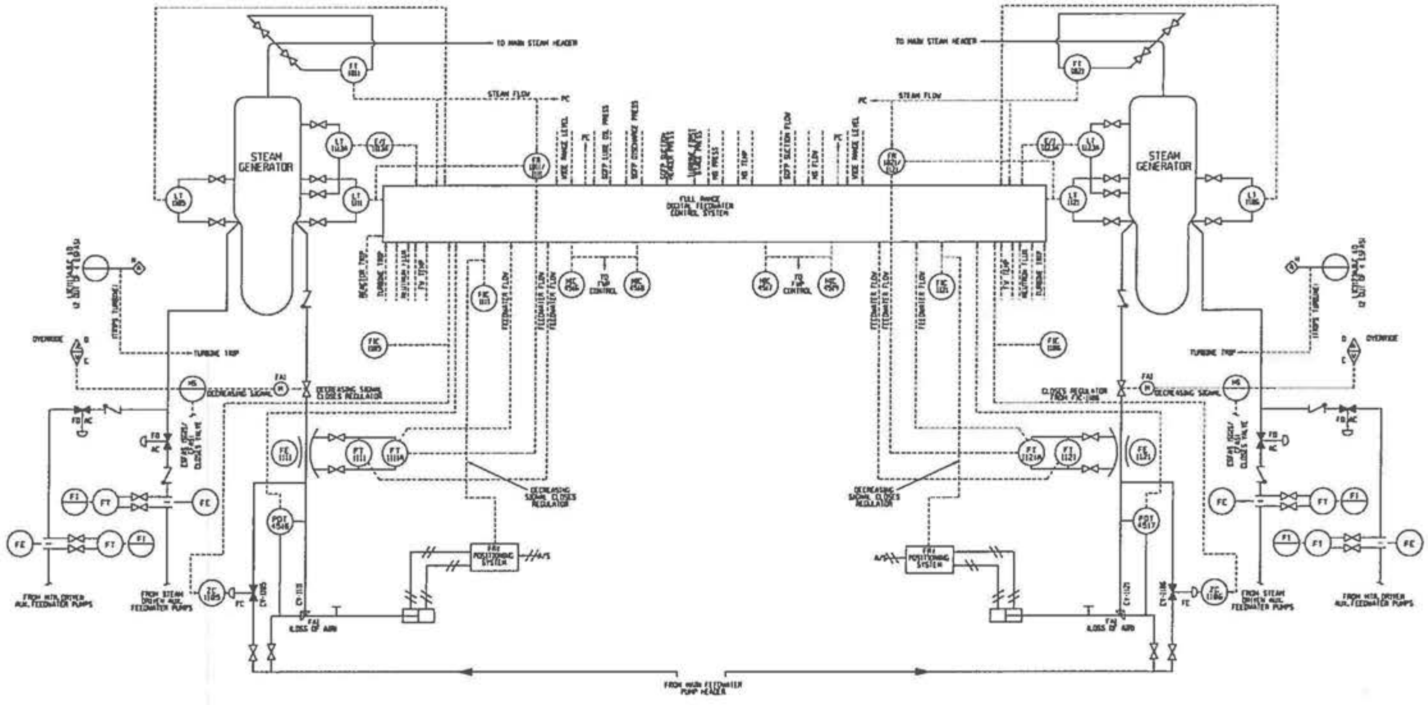
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CEA POSITION SETPOINTS	REV. 26	Figure 7-12



NOTE 1: Technical Specification 3.3-4, Table 3.3.4-1

NOTE 2: UFSAR Table 7-1

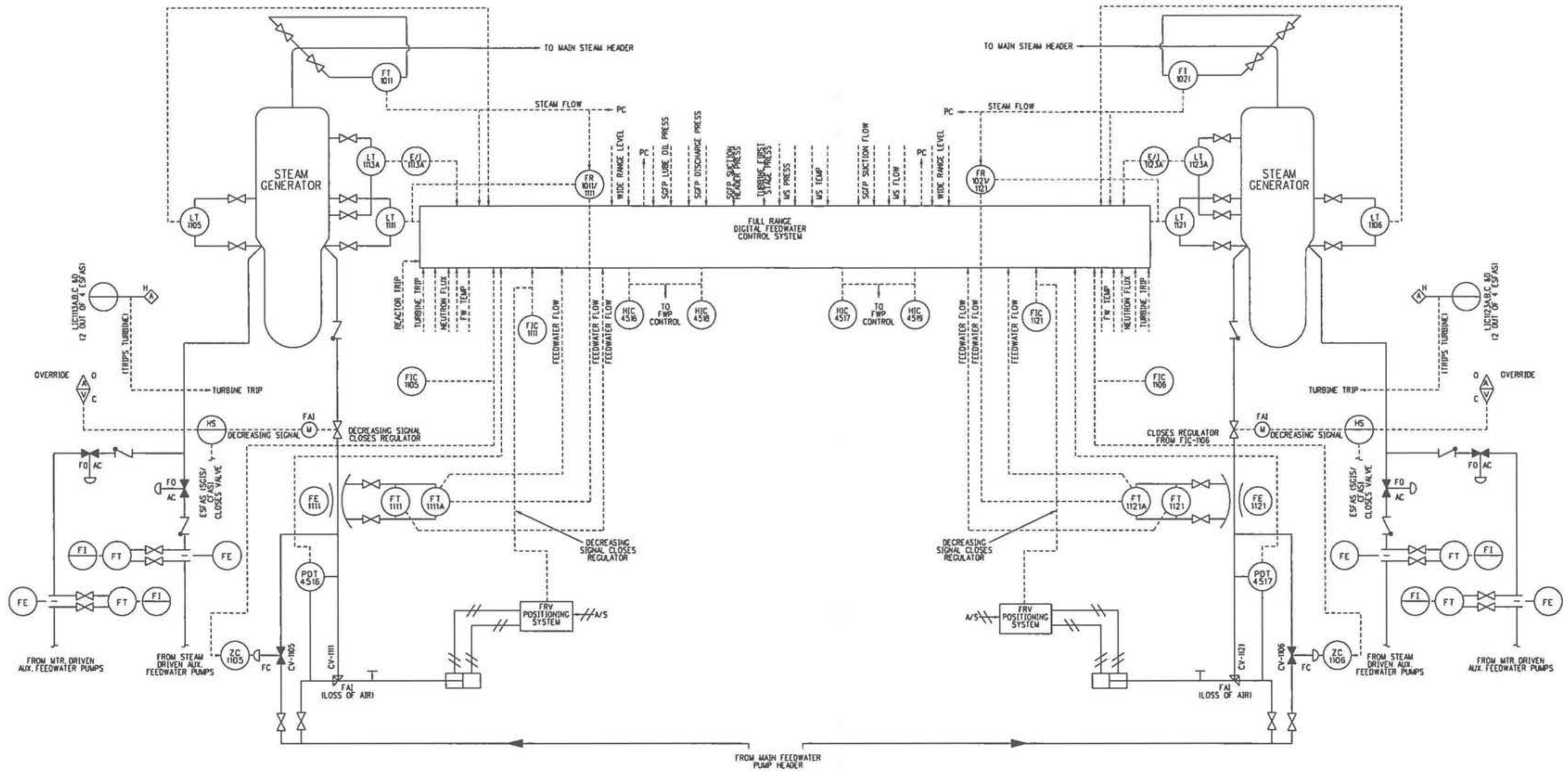
FIGURE 7-14A FEED WATER CONTROL SYSTEM - BLOCK DIAGRAM, UNIT 1



FOR AFW DETAILS SEE

CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 7-14A
FEEDWATER CONTROL SYSTEM BLOCK DIAGRAM UNIT 1
BGE DRAWING N/A

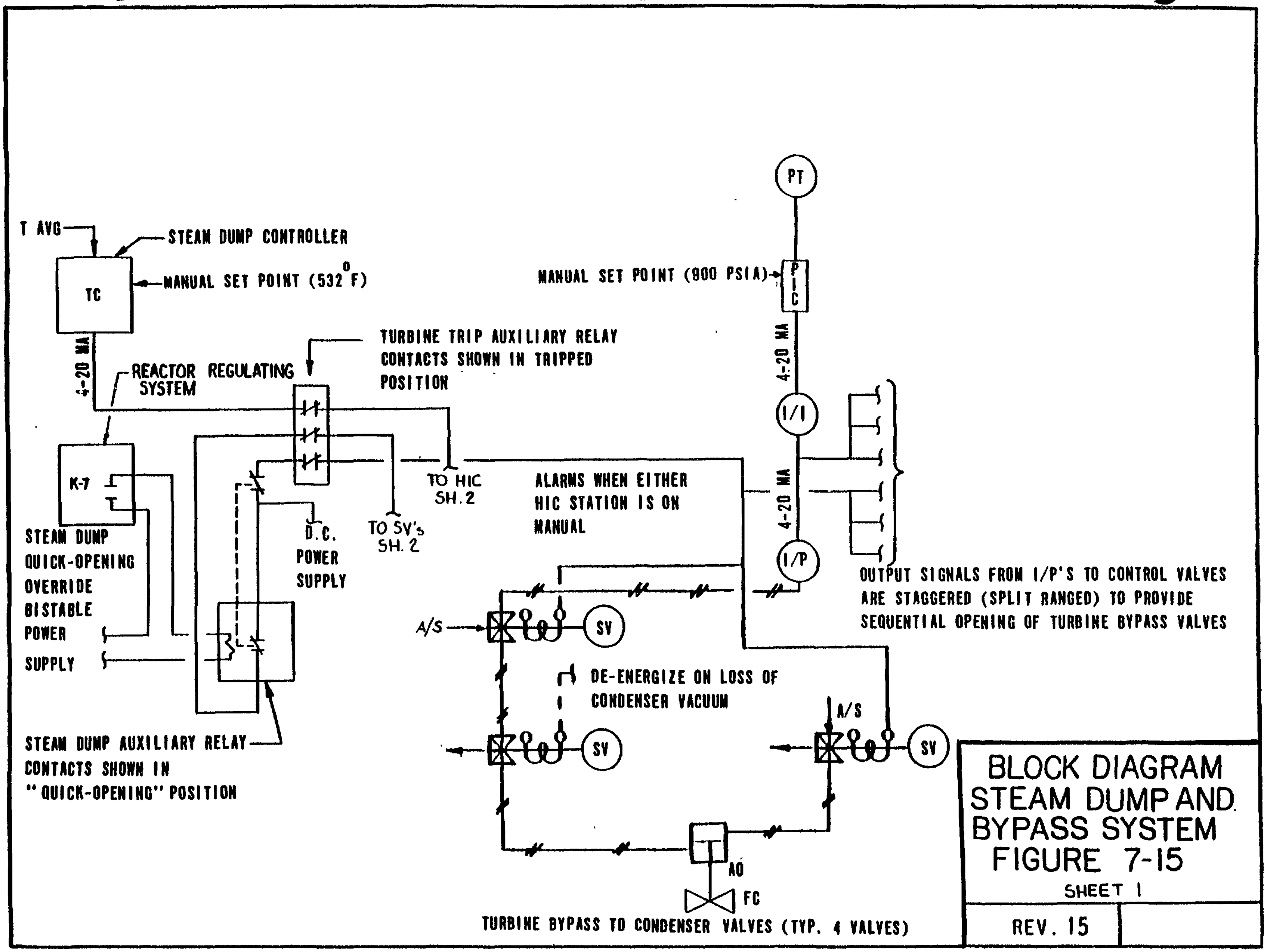
FIGURE 7-14B FEED WATER CONTROL SYSTEM- BLOCK DIAGRAM, UNIT 2



FOR AFW DETAILS SEE 62583 (OM-801)

CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 7-14B
FEEDWATER CONTROL SYSTEM BLOCK DIAGRAM UNIT 2
BGE DRAWING N/A

REVISION 38


T AVG
STEAM DUMP CONTROLLER
TC
MANUAL SET POINT (532 °F)
4-20 MA
REACTOR REGULATING SYSTEM
K-7
TURBINE TRIP AUXILIARY RELAY CONTACTS SHOWN IN TRIPPED POSITION
TO HIC SH. 2
TO SV's SH. 2
D.C. POWER SUPPLY
STEAM DUMP QUICK-OPENING OVERRIDE BISTABLE POWER SUPPLY
STEAM DUMP AUXILIARY RELAY CONTACTS SHOWN IN "QUICK-OPENING" POSITION
PT
MANUAL SET POINT (900 PSIA)
PIC
4-20 MA
I/I
4-20 MA
I/P
ALARMS WHEN EITHER HIC STATION IS ON MANUAL
OUTPUT SIGNALS FROM I/P'S TO CONTROL VALVES ARE STAGGERED (SPLIT RANGED) TO PROVIDE SEQUENTIAL OPENING OF TURBINE BYPASS VALVES
A/S
SV
DE-ENERGIZE ON LOSS OF CONDENSER VACUUM
SV
A/S
SV
AO
FC
TURBINE BYPASS TO CONDENSER VALVES (TYP. 4 VALVES)
BLOCK DIAGRAM
STEAM DUMP AND
BYPASS SYSTEM
FIGURE 7-15
SHEET 1
REV. 15

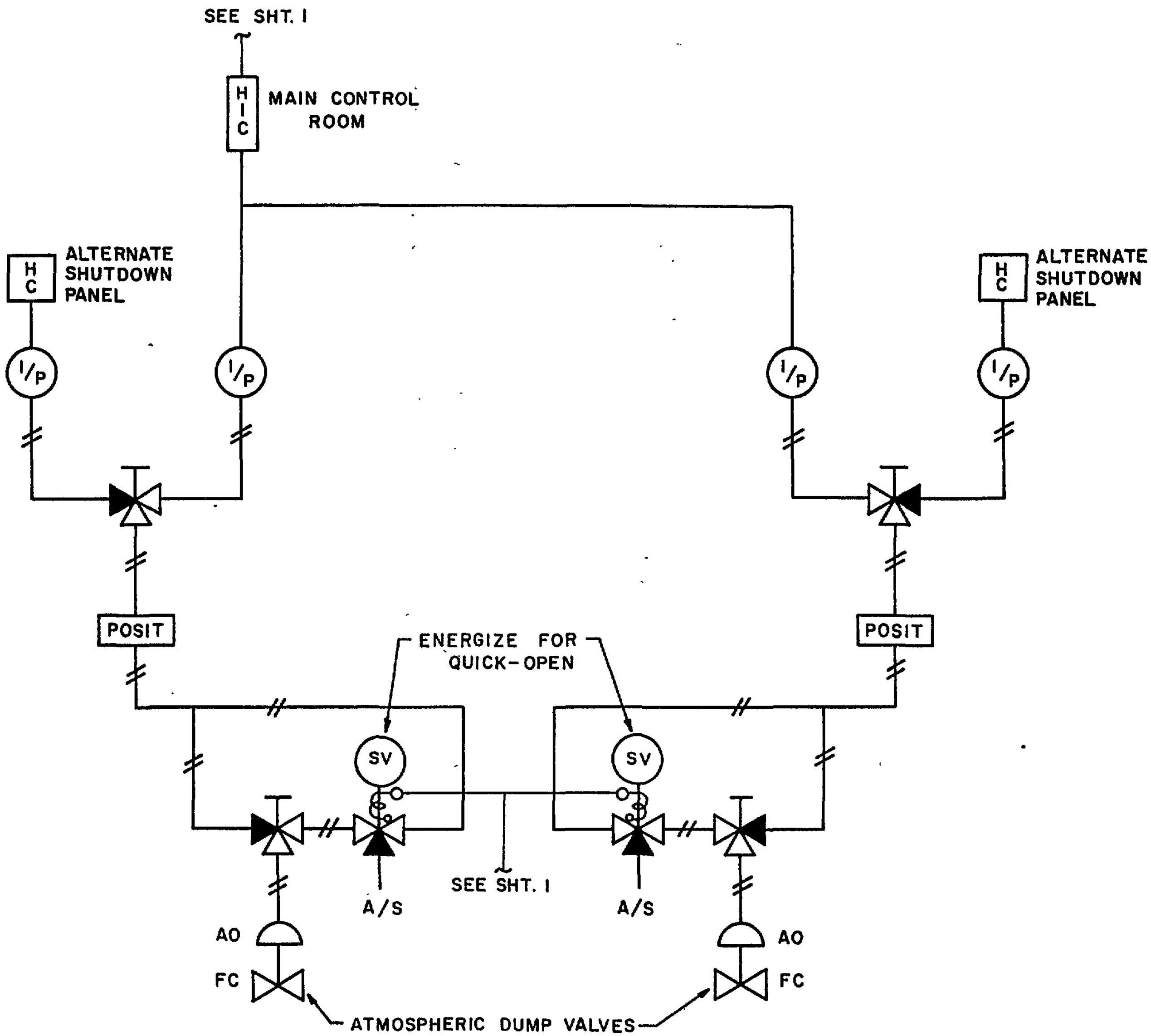
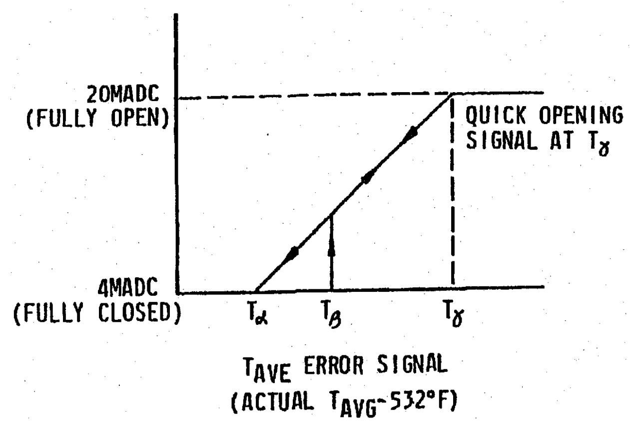


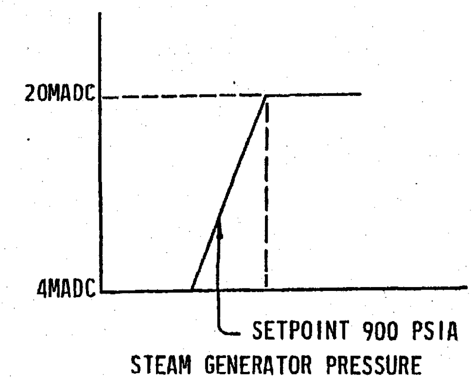
FIGURE 7-15 SHT. 2
REV. 3

NOTE:
1. QUICK OPENING BOOSTER AIR SOLENOID VALVES ENERGIZED BY OVERRIDE BISTABLE WHEN OUTPUT SIGNAL > 19.80MA, DE-ENERGIZED WHEN OUTPUT SIGNAL < 19.64MA

DUMP VALVE POSITION SIGNAL



STEAM DUMP CONTROLLER PROGRAM (ADV)



PRESSURE CONTROLLER PROGRAM (TBV)

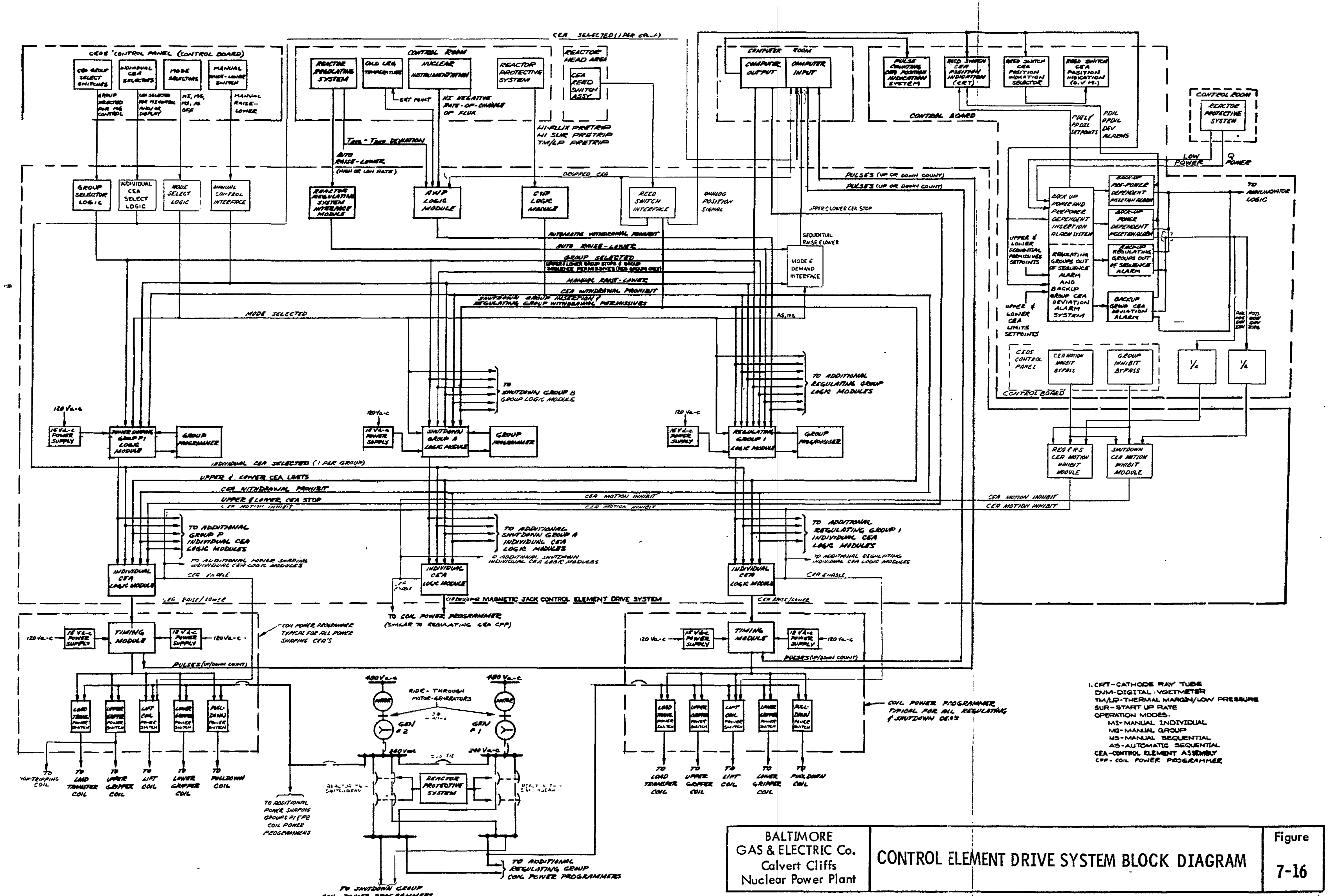
Calvert Cliffs Nuclear Power Plant

PROGRAM OF STEAM DUMP BYPASS VALVES & QUICK OPENING OVERRIDE

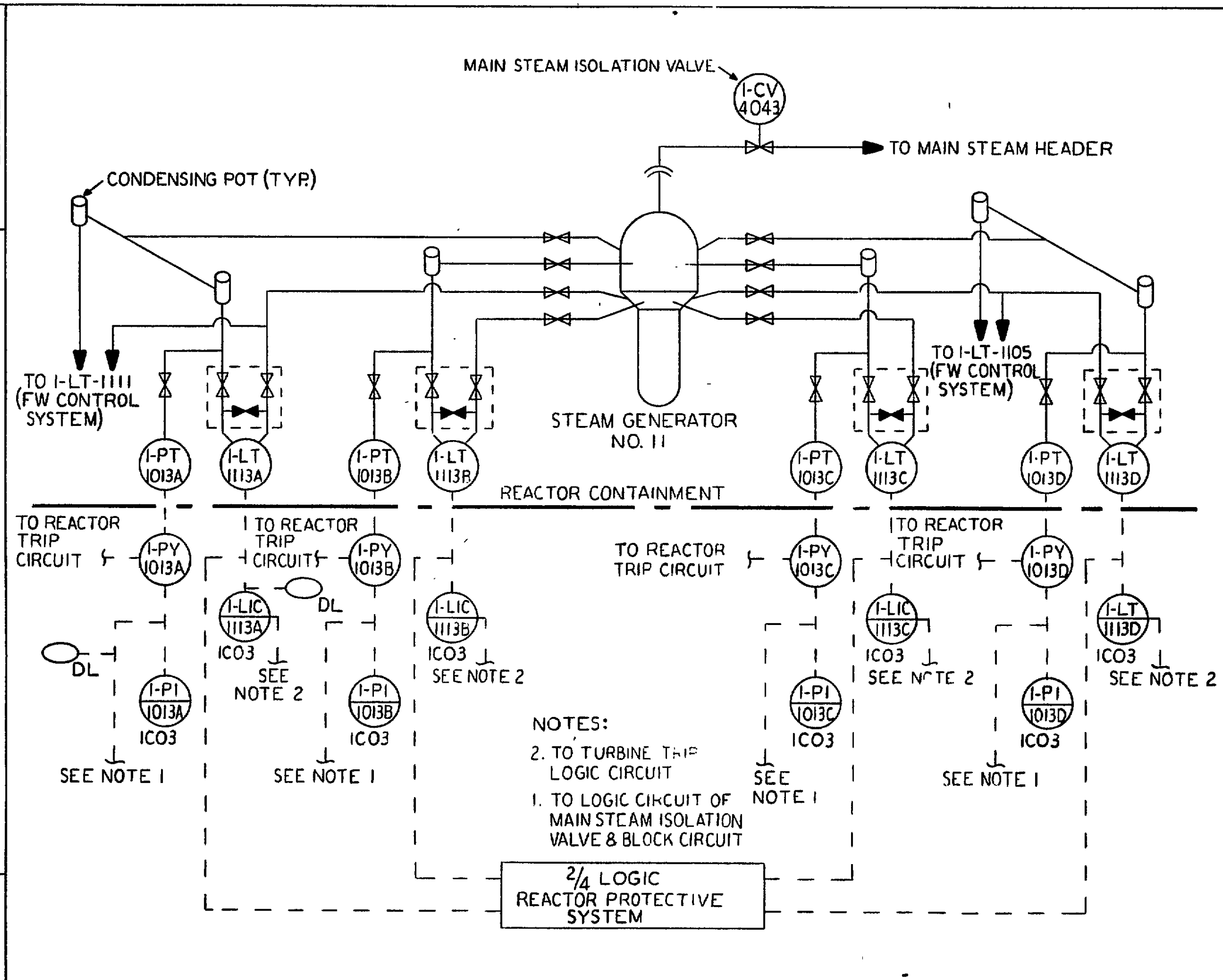
Sheet 3

Figure 7-15

Revision 37


MAGNETIC JACK CONTROL ELEMENT DRIVE SYSTEM
1. CRT-CATHODE RAY TUBE
DVM-DIGITAL VOLTMETER
TM/LP-THERMAL MARGIN/LOW PRESSURE
SUR-START UP RATE
OPERATION MODES-
MI-MANUAL INDIVIDUAL
MG-MANUAL GROUP
MS-MANUAL SEQUENTIAL
AS-AUTOMATIC SEQUENTIAL
CEA-CONTROL ELEMENT ASSEMBLY
CPP-COIL POWER PROGRAMMER
BALTIMORE GAS & ELECTRIC Co.
Calvert Cliffs
Nuclear Power Plant
CONTROL ELEMENT DRIVE SYSTEM BLOCK DIAGRAM
Figure
7-16

7-17 STEAM GENERATOR PROTECTIVE SYSTEM MEASUREMENT CHANNEL



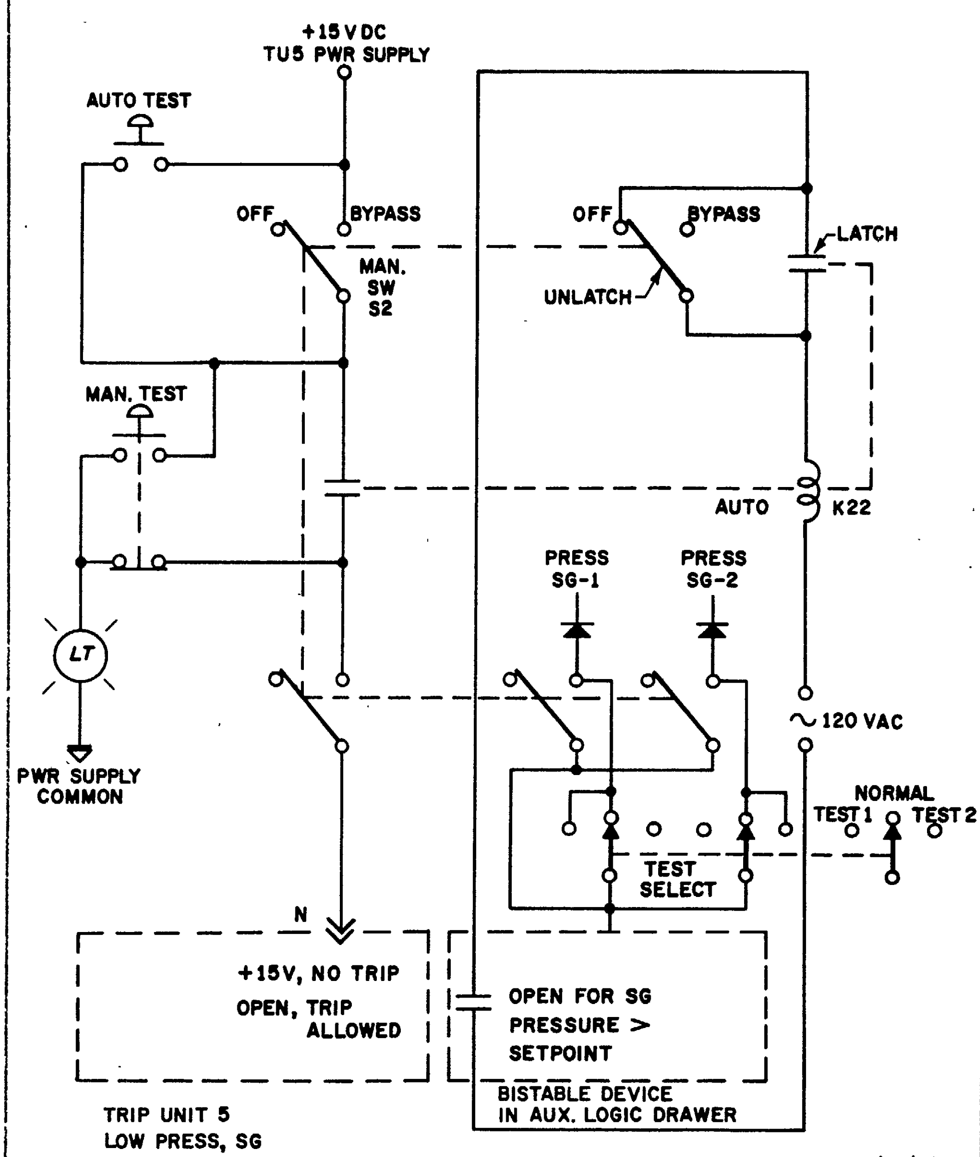
BALTIMORE GAS & ELECTRIC CO.
CALVERT CLIFFS NUCLEAR POWER PLANT

STEAM GENERATOR PROTECTIVE SYSTEM MEASUREMENT CHANNEL

FIGURE 7-17

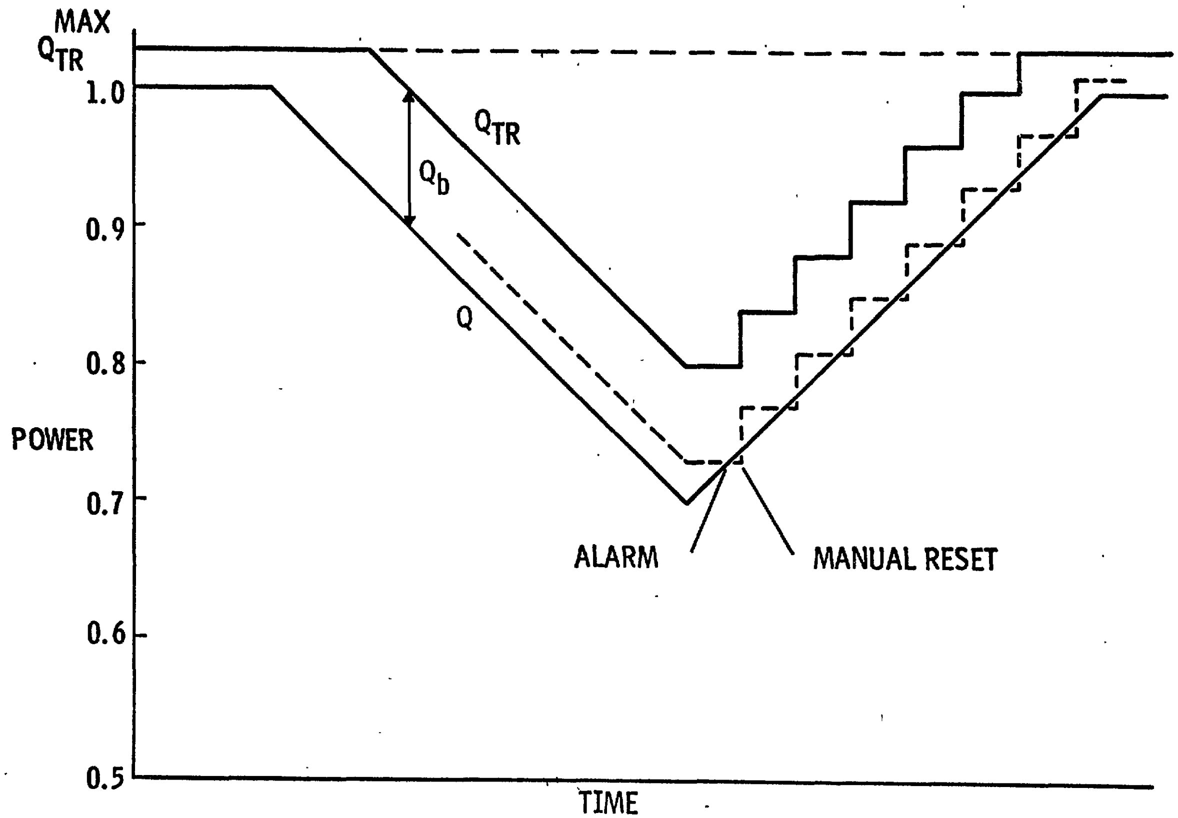
Rev. 1/28/72

Rev.0



Rev. 1/28/72

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Schematic Low Steam Generator Pressure Bypass	Figure 7-18



BALTIMORE
GAS & ELECTRIC Co.
Calvert Cliffs
Nuclear Power Plant

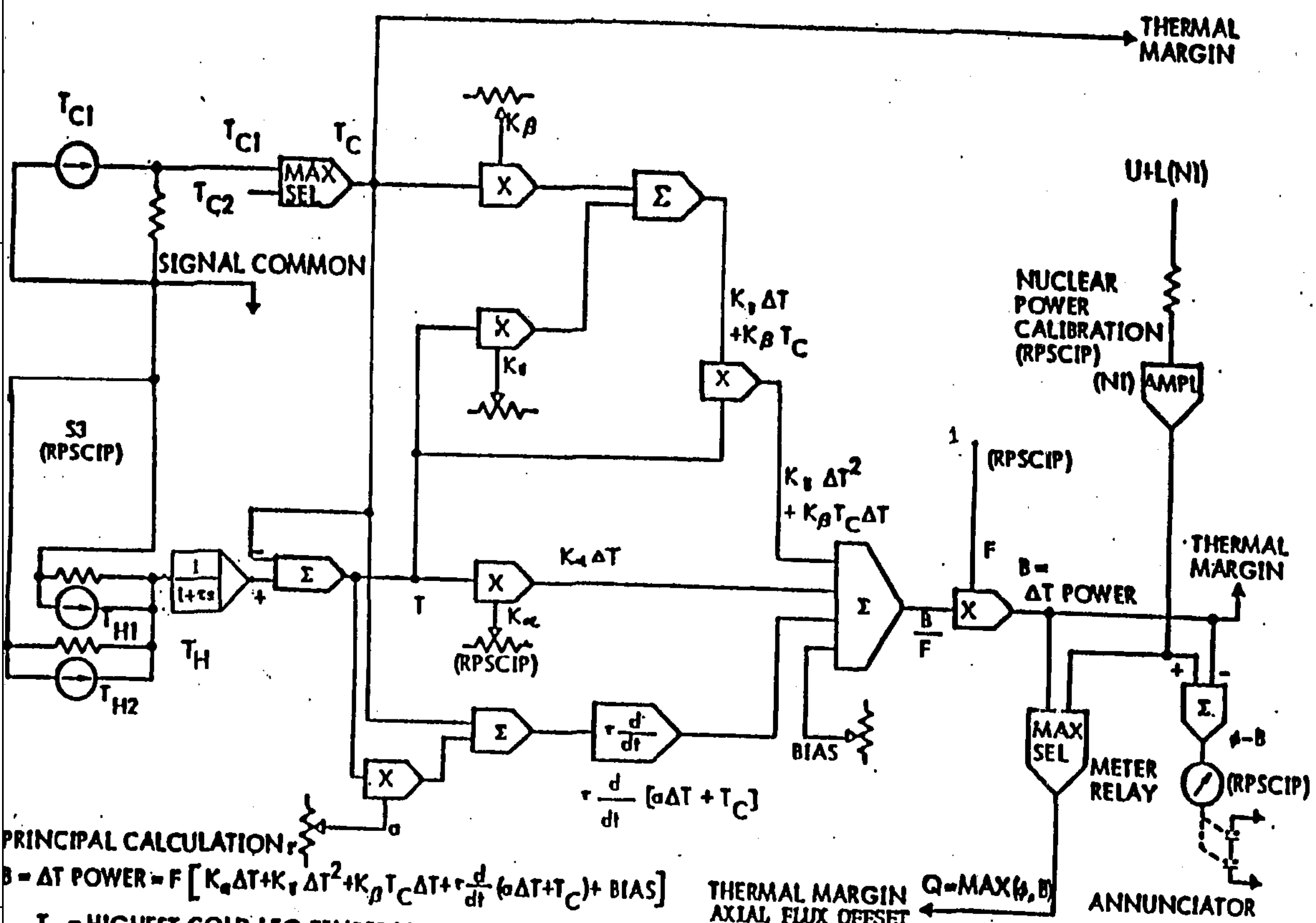
VARIABLE HIGH POWER TRIP OPERATION

Rev. 9/25/73

Figure
7-20

Calvert Cliffs
Nuclear Power Plant

ΔT POWER CALCULATION



PRINCIPAL CALCULATION

$B = \Delta T\ POWER = F\left[K_\alpha \Delta T + K_\gamma \Delta T^2 + K_\beta T_C \Delta T + \tau \frac{d}{dt}(\alpha\Delta T + T_C) + BIAS\right]$

T_C = HIGHEST COLD LEG TEMPERATURE

T_H = AVERAGE OR ACTIVE LOOP HOT LEG TEMPERATURE

$\Delta T = T_H - T_C$

Figure 7-21
Rev. 31

FIGURE 7-22 LOGIC DIAGRAM - ENGINEERED SAFETY FEATURES ACTUATION SYSTEM UNIT 2 (Sheet 1)

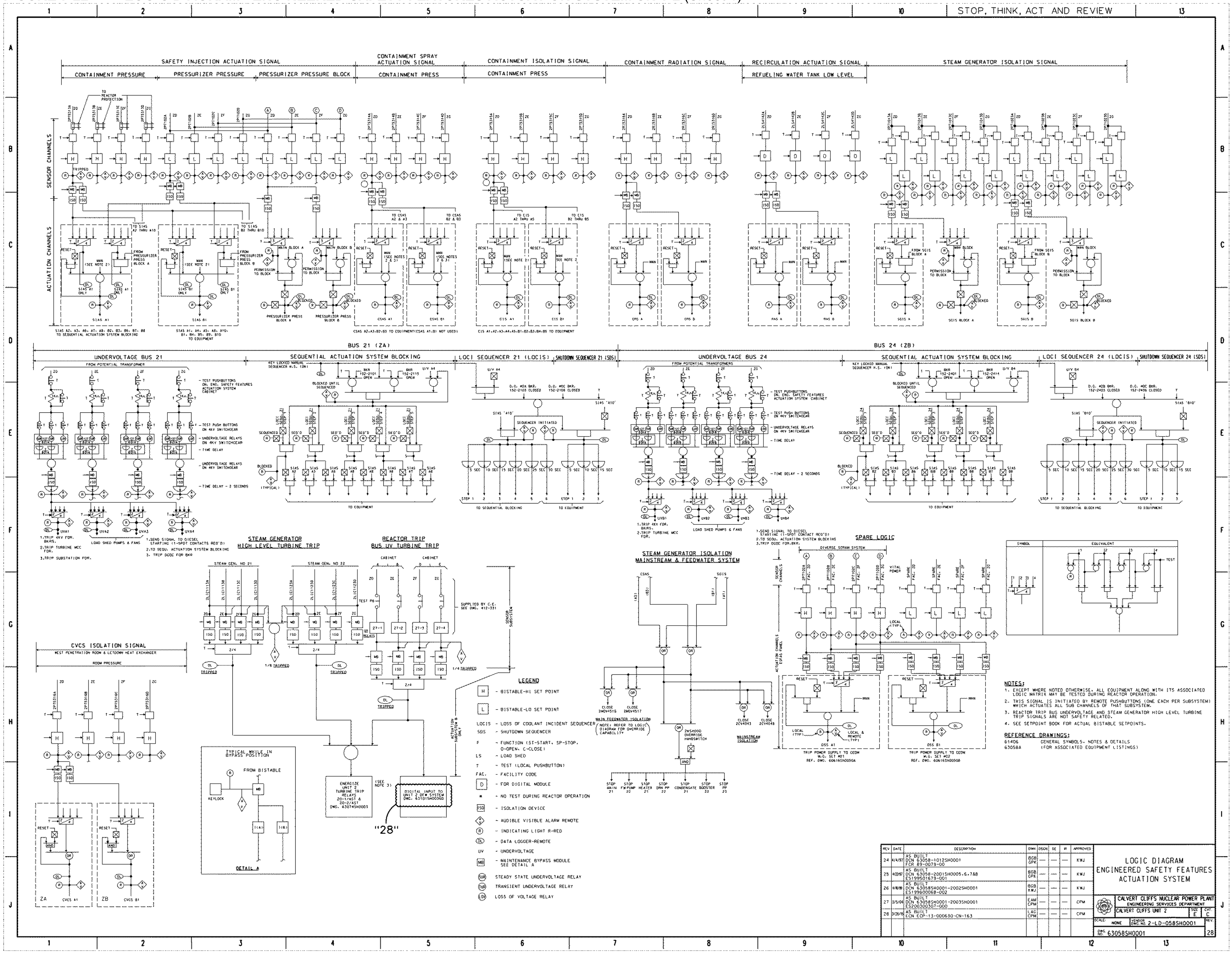
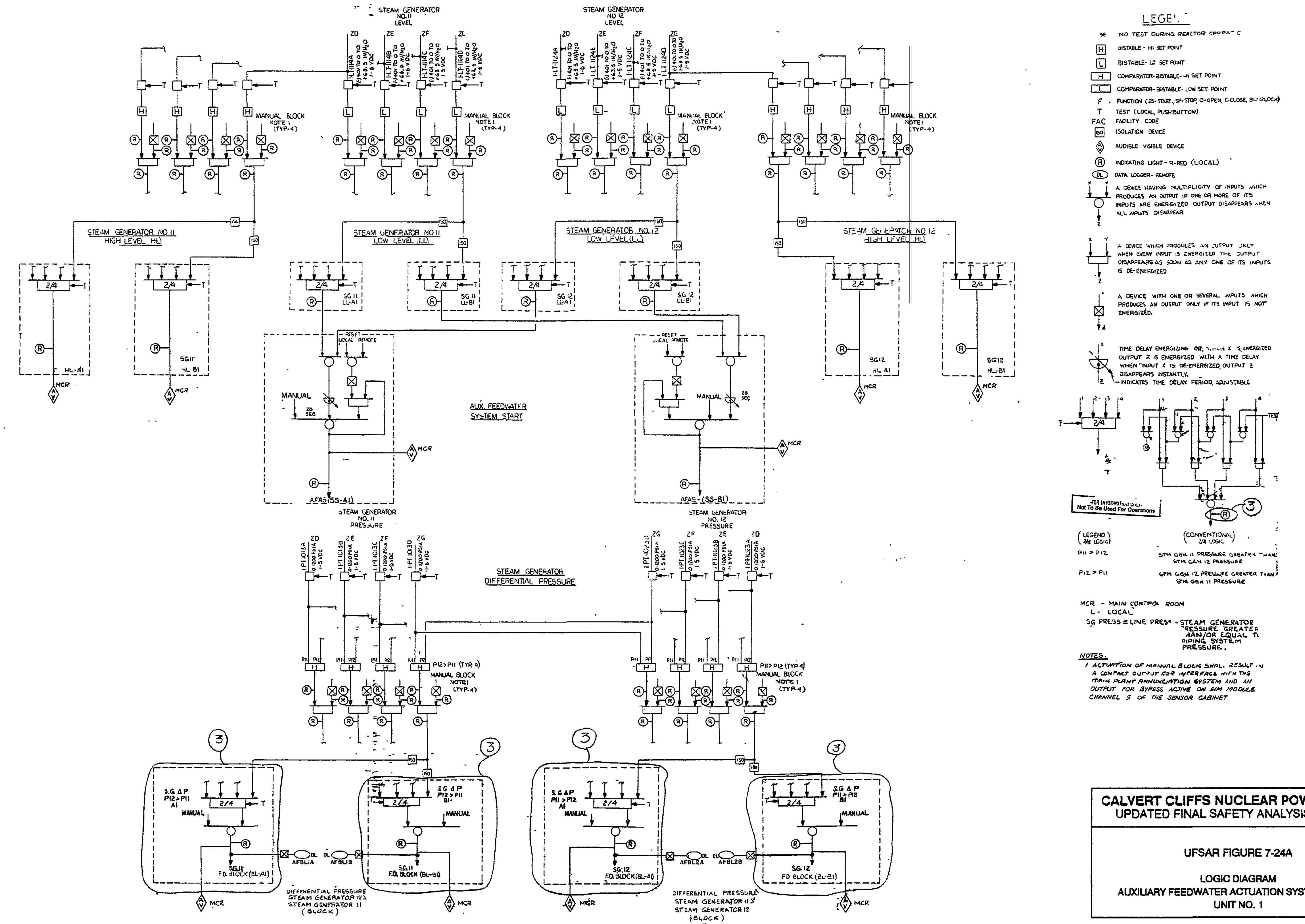


FIGURE 7-22 LOGIC DIAGRAM - ENGINEERED SAFETY FEATURES ACTUATION SYSTEM UNIT 2 (Sheet 2)



CALVERT CLIFFS NUCLEAR POWER PLANT
UPDATED FINAL SAFETY ANALYSIS REPORT

UFSAR FIGURE 7-24A

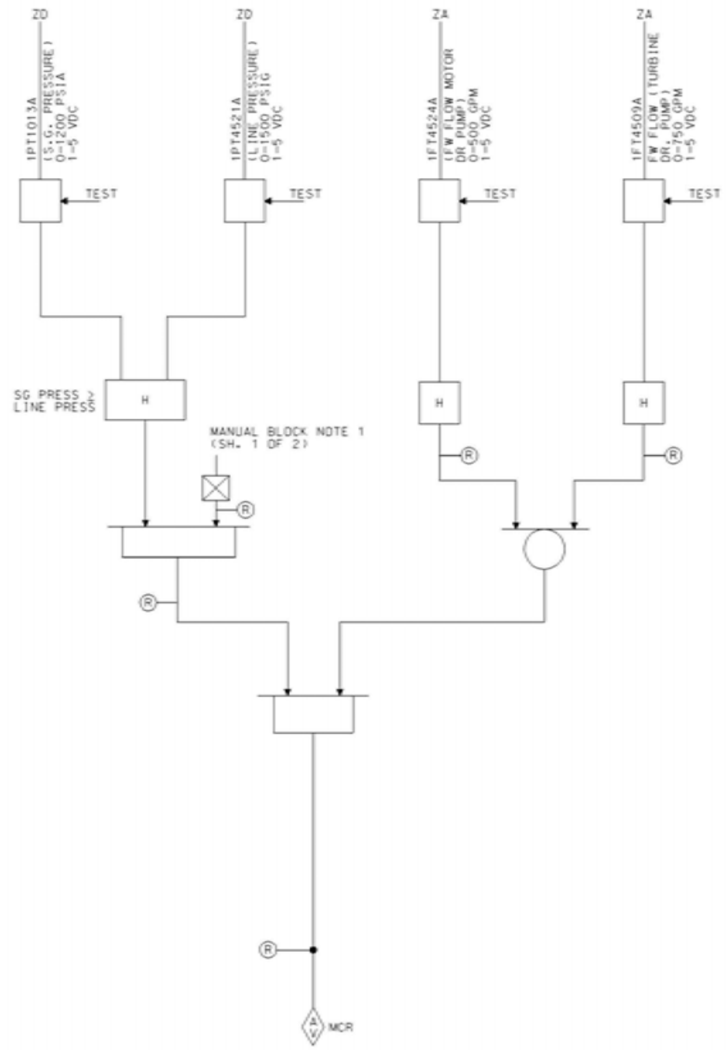
LOGIC DIAGRAM
AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS)
UNIT NO. 1

Rev. 13 1/92

BALTIMORE GAS & ELECTRIC COMPANY DRAWING NO. 61-060 SH. 1 (1-LD-60) REVISION 3

FIGURE 7-24B LOGIC DIAGRAM – AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) UNIT NO. 1

STOP, THINK, ACT AND REVIEW



LOSS OF AUXILIARY FEEDWATER FLOW ALARM S.G. 11 CHANNEL A (SEE TABLE BELOW FOR OTHER CHANNELS)

PIPE RUPTURE LOGIC

TAG NO.	DESCRIPTION	SCHEMATIC DIAGRAM	SOLENOID VALVE	FACILITY	AFAS (1) START (SS) A1	AFAS (1) START (SS) B1	AFAS (1) START (SS) F	AFAS (1) BLOCK (BL) A1	AFAS (1) BLOCK (BL) B1	AFAS (1) BLOCK (BL) F
1CV4521	SG11 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	61079SH0057A	1SV4521	ZA				X		C
1CV4520	SG11 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	61079SH0057A	1SV4520	ZB					X	C
1CV4523	SG11 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	61079SH0057B	1SV4523	ZA				X		C
1CV4522	SG11 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	61079SH0057B	1SV4522	ZB					X	C
1CV4531	SG12 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	61079SH0057A	1SV4531	ZA				X		C
1CV4530	SG12 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	61079SH0057A	1SV4530	ZB					X	C
1CV4533	SG12 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	61079SH0057B	1SV4533	ZA				X		C
1CV4532	SG12 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	61079SH0057B	1SV4532	ZB					X	C
1CV4070	AFW TURBINE STEAM ADMISSION VALVE SG11	61079SH0055A	1SV4070	ZB		X	O			
1CV4071	AFW TURBINE STEAM ADMISSION VALVE SG12	61079SH0055B	1SV4071	ZA	X		O			
1CV4070A	AFW TURBINE STEAM ADMISSION BYPASS VALVE SG11	61079SH0055A	1SV4070A	ZB		X	O			
1CV4071A	AFW TURBINE STEAM ADMISSION BYPASS VALVE SG12	61079SH0055B	1SV4071A	ZA	X		O			
PP-3	AFW MOTOR-DRIVEN PUMP	61079SH0054	N/A	ZA	X		SS			
1CV4010	SG11 SURFACE BLOWDOWN CONTAINMENT ISOLATION VALVE	61076SH0014K	1SV4010	ZA	X		C			
1CV4011	SG11 BOTTOM BLOWDOWN CONTAINMENT ISOLATION VALVE	61076SH0014L	1SV4011	ZA	X		C			
1CV4012	SG12 SURFACE BLOWDOWN CONTAINMENT ISOLATION VALVE	61076SH0014M	1SV4012	ZB		X	C			
1CV4013	SG12 BOTTOM BLOWDOWN CONTAINMENT ISOLATION VALVE	61076SH0014N	1SV4013	ZB		X	C			

"6"

AFAS ACTUATED DEVICES

NOTE:
1. REFER TO SHEET NO.1 FOR GENERAL NOTES & LEGEND

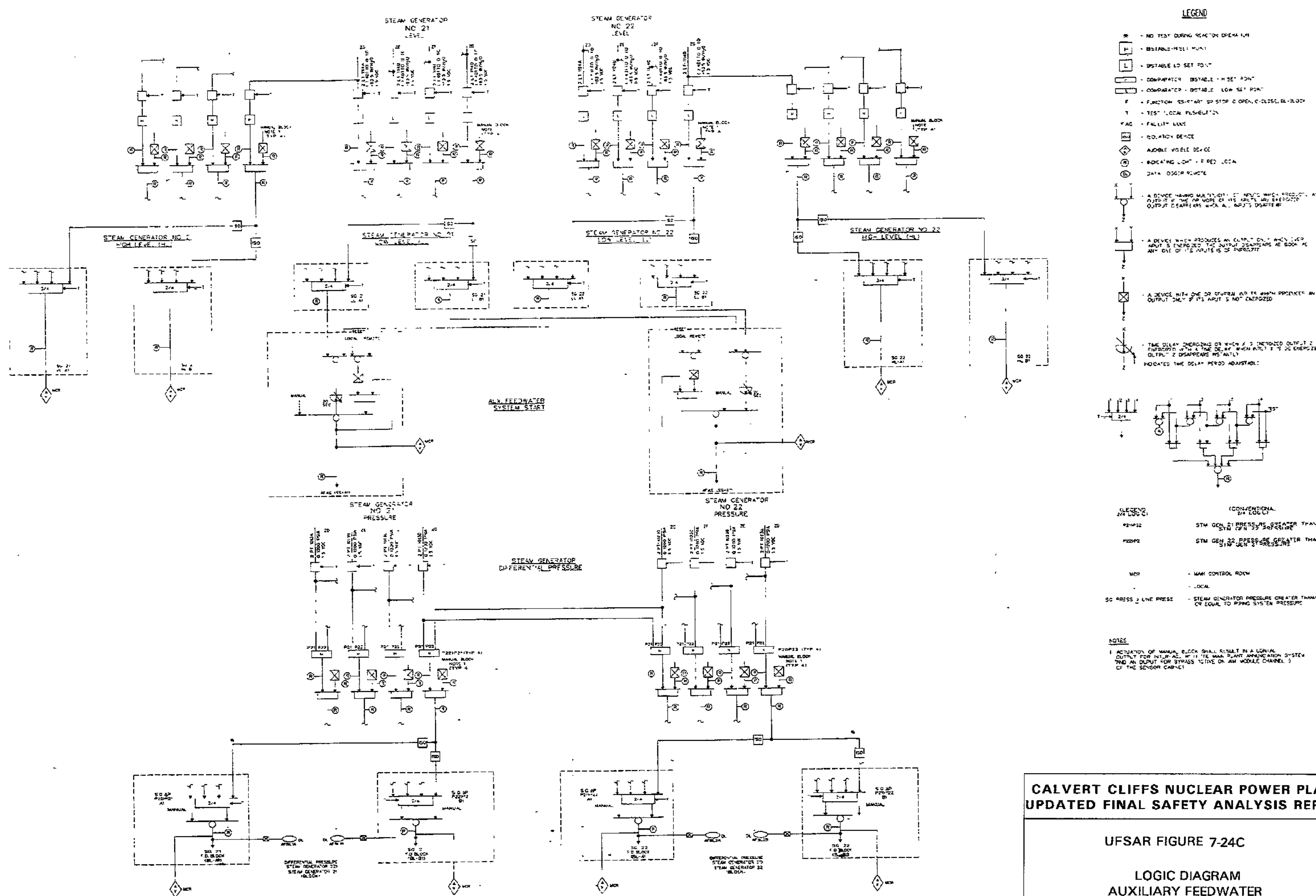
FACILITY	STEAM GENERATOR	PRESSURE TRANSMITTER	PRESSURE TRANSMITTER	FACILITY	FLOW TRANSMITTER	FLOW TRANSMITTER	ALARM
ZD	11, CH. A	1PT1013A	1PT4521A	ZA	1FT4524A	1FT4509A	MCR
ZE	11, CH. B	1PT1013B	1PT4521B	ZB	1FT4524B	1FT4509B	
ZD	12, CH. A	1PT1023A	1PT4531A	ZA	1FT4534A	1FT4510A	MCR
ZE	12, CH. B	1PT1023B	1PT4531B	ZB	1FT4534B	1FT4510B	

THIS DRAWING WAS IN THE SYSTEM DRAWN BY CADD-CAM, REV. 4, DATED 09-25-92.

REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
5	4/23/02	AS BUILT DCN 61060SH0002-2001SH0001 ES200100849-000	EAM KWJ	—	—	—	KWJ
6	9/14/06	AS BUILT DCN 61060SH0002-2002SH0001 ES200500513-000	RWM CPM	—	—	—	CPM

LOGIC DIAGRAM
AUXILIARY FEEDWATER
ACTUATION SYSTEM (AFAS)
UNIT NO. 1

CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 1 SIZE D CAT. C
SCALE: NONE VENDOR DWG. NO. 1-LD-060SH0002 REV. 6
DWG. NO. 61060SH0002



BALTIMORE GAS & ELECTRIC COMPANY DRAWING NO. 62-621-E SH. 1 OF 2 (2-LD-60) REVISION 6

CALVERT CLIFFS NUCLEAR POWER PLANT
UPDATED FINAL SAFETY ANALYSIS REPORT

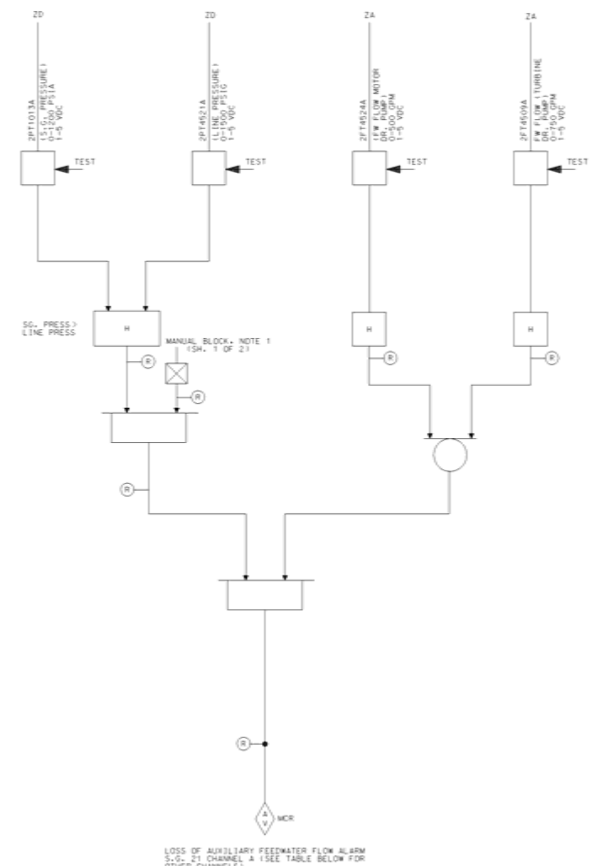
UFSAR FIGURE 7-24C

LOGIC DIAGRAM
AUXILIARY FEEDWATER
ACTUATION SYSTEM (AFAS)
UNIT NO. 2

Revision 15

FIGURE 7-24D LOGIC DIAGRAM – AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) UNIT NO. 2

STOP, THINK, ACT AND REVIEW



PIPE RUPTURE LOGIC

TAG NO.	DESCRIPTION	SCHEMATIC DIAGRAM	SOLENOID VALVE	FACILITY	AFAS (2) START (SS) A1	B1	F	AFAS (2) BLOCK (BL) A1	B1	F
2CV4521	SG21 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	63079SH0057	2SV4521	ZA				X		C
2CV4520	SG21 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	63079SH0057	2SV4520	ZB					X	C
2CV4523	SG21 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	63079SH0057	2SV4523	ZA				X		C
2CV4522	SG21 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	63079SH0057	2SV4522	ZB					X	C
2CV4531	SG22 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	63079SH0057	2SV4531	ZA				X		C
2CV4530	SG22 AFW BLOCK VALVE, TURBINE-DRIVEN PUMP	63079SH0057	2SV4530	ZB					X	C
2CV4533	SG22 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	63079SH0057	2SV4533	ZA				X		C
2CV4532	SG22 AFW BLOCK VALVE, MOTOR-DRIVEN PUMP	63079SH0057	2SV4532	ZB					X	C
2CV4070	AFW TURBINE STEAM ADMISSION VALVE S.G. 21	63079SH0055A	2SV4070	ZB		X	O			
2CV4071	AFW TURBINE STEAM ADMISSION VALVE S.G. 22	63079SH0055B	2SV4071	ZA	X		O			
2CV4070A	AFW TURBINE STEAM ADMISSION BYPASS VALVE S.G. 21	63079SH0055A	2SV4070A	ZB		X	O			
2CV4071A	AFW TURBINE STEAM ADMISSION BYPASS VALVE S.G. 22	63079SH0055B	2SV4071A	ZA	X		O			
PP23	AFW MOTOR-DRIVEN PUMP	63079SH0054	N/A	ZB		X	SS			
2CV4010	SG21 SURFACE BLOWDOWN CONTAINMENT ISOLATION VALVE	63076SH0014K	2SV4010	ZA	X		C			
2CV4011	SG21 BOTTOM BLOWDOWN CONTAINMENT ISOLATION VALVE	63076SH0014L	2SV4011	ZA	X		C			
2CV4012	SG22 SURFACE BLOWDOWN CONTAINMENT ISOLATION VALVE	63076SH0014M	2SV4012	ZB		X	C			
2CV4013	SG22 BOTTOM BLOWDOWN CONTAINMENT ISOLATION VALVE	63076SH0014N	2SV4013	ZB		X	C			

"7"

AFAS ACTUATED DEVICES

FACILITY	STEAM GENERATOR	PRESSURE TRANSMITTER	PRESSURE TRANSMITTER	FACILITY	FLOW TRANSMITTER	FLOW TRANSMITTER	ALARM
ZD	21, CH. A	2PT1013A	2PT4521A	ZA	2FT4524A	2FT4509A	MCR
ZE	21, CH. B	2PT1013B	2PT4521B	ZB	2FT4524B	2FT4509B	
ZD	22, CH. A	2PT1023A	2PT4531A	ZA	2FT4534A	2FT4510A	MCR
ZE	22, CH. B	2PT1023B	2PT4531B	ZB	2FT4534B	2FT4510B	

THIS DRAWING WAS ALREADY IN THE SYSTEM AT REV. 4, DATED 2/15/92.

REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
5	7/03/98	AS BUILT INFO ONLY ES199801067-000	MPR FWM	—	—	—	WHW
6	4/15/03	AS BUILT DCN 62621SH0002-2001SH0001 ES200100849-001	EAM KWJ	—	—	—	KWJ
7	3/25/07	AS BUILT DCN 62621SH0002-2002SH0001 ES200500513-000	EAM CPM	—	—	—	CPM

LOGIC DIAGRAM
AUXILIARY FEEDWATER
ACTUATION SYSTEM (AFAS)
UNIT NO. 2

CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 2 SIZE E CAT. 1
SCALE: NONE VENDOR DWG. NO. 2-LD-060SH0002 REV. 7
DWG. NO. 62621SH0002

CHAPTER 8
ELECTRICAL SYSTEMS

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LIST OF ACRONYMS

AAC	Alternate AC
ABB	Asea Brown Boveri, Inc.
CEA	Control Element Assembly
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FOST	Fuel Oil Storage Tank
FSAR	Final Safety Analysis Report
GE	General Electric
HT	High Temperature
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss-of-Coolant Accident
LOCI	Loss-of-Coolant Incident
LT	Low Temperature
MCC	Motor Control Center
PJM	Pennsylvania-New Jersey-Maryland
RCP	Reactor Coolant Pump
RPS	Reactor Protective System
SACM	Societe Alsacienne De Constructions Mecaniques De Mulhouse
SBC	Standard Building Code
SBO	Station Blackout
SIAS	Safety Injection Actuation Signal
SMECO	Southern Maryland Electric Cooperative
SSE	Safe Shutdown Earthquake

8.0 ELECTRICAL SYSTEMS

8.1 INTRODUCTION

The electrical systems include the equipment and systems necessary to generate power and deliver it to the high voltage system. They also include facilities for providing power to, and controlling the operation of, electrically-driven plant auxiliary equipment and instrumentation.

Essential instrumentation, including the Reactor Protective System (RPS) and the Engineered Safety Features (ESF) instrumentation, is fed from vital instrumentation busses to provide continuous monitoring and control. The plant batteries provide circuit breaker control, Control Room emergency lighting, vital instrumentation power, and operating power for certain other equipment.

8.1.1 DESIGN BASIS

The plant electrical systems are designed to ensure a continuous supply of electrical power to all essential plant equipment during normal operation and under accident conditions.

All electrical systems and components vital to plant safety, including the emergency diesel generators (EDGs), are designed as Class 1E so that their integrity is not impaired by the Safe Shutdown Earthquake (SSE), high winds, or disturbances on the external electrical system.

8.1.2 DESCRIPTION AND OPERATION

The plant electrical system is shown on Figure 8-1, Main Single Line Diagram.

In order to achieve maximum reliability and efficiency of operation of the electrical systems, the following criteria are employed:

a. The main generators, described in Section 10, feed electrical power at 25 kV and 22 kV for Units 1 and 2, respectively, through forced air cooled isolated phase busses to two main unit transformers installed per unit.

b. Plant auxiliary sources of power are the two 500 kV/14 kV plant service transformers, which are fed from separate 500 kV switchyard busses and a 13 kV line from the Southern Maryland Electric Cooperative (SMECO) system. Each 500 kV/14 kV plant service transformer is capable of supplying the total (two unit) plant auxiliary load. The 13 kV SMECO line is capable of supplying the power required to maintain both units in a safe shutdown condition. It may be substituted for one of the 500 kV/13 kV circuits as one of the two required, physically independent, offsite circuits.

c. The two plant service transformers feed six 13.8 kV 2MVA ± 10% voltage regulators which feed six 13.8 kV/4.16 - 4.16 kV service transformers, three of which are capable of supplying the total plant 4.16 kV auxiliary load.

d. The 13.8 kV system consists of multiple reactor coolant pump (RCP) busses, each of which can be fed from either of the two plant service transformers. Two 13.8 kV service busses (with tie breakers) are provided for distribution to the voltage regulators and unit service transformers.

e. The plant is split into two independent load groups, each with its own power supply, busses, transformers, loads, and 125 Volt DC control power (Figure 8-9).

f. A reserve 125 Volt DC system, capable of replacing any of the 125 Volt DC batteries, if required, is provided. The system consists of one battery, one battery charger, and associated DC switching equipment.

g. The 4.16 kV system is divided into several bus sections, each of which can be supplied from either of two unit service transformers fed from different plant

service transformers. The four 4.16 kV ESF busses can be supplied from the EDGs.

h. The plant has four safety-related EDGs, two dedicated to each unit. Any combination of two of the EDGs (one from each unit) is capable of supplying sufficient power for the operation of necessary ESF loads during accident conditions on one unit and shutdown loads of the alternate unit concurrent with a loss of offsite power and for the safe and orderly shutdown of both units under loss of offsite power conditions. The diesel generators start automatically on safety injection actuation signal (SIAS) or an undervoltage condition on the busses which supply vital loads, and are ready to accept loads within 10 seconds (Figure 8-6). A Station Blackout diesel generator can also be aligned to any of the four ESF busses.

i. All necessary ESF are duplicated and power supplies are so arranged that the failure to energize any one of the applicable busses, or the failure of one diesel generator to start, will not prevent the proper operation of the ESF systems.

j. The ESF electrical system has been designed to satisfy the single failure criterion as defined Institute of Electrical and Electronic Engineers (IEEE) 279, Section 4.2.

k. Four vital AC instrument busses per unit are provided for essential instrumentation and reactor protection circuits. Each vital bus is fed from a separate battery supply through a dual static inverter.

l. The design criteria for all electrical control cable and safety-related equipment power cable are that the cable shall not fail when subjected to associated accident conditions after the long-term, normal operating conditions.

m. Power cables in 13.8 kV service are HT Kerite Permashield insulated cables rated at 15 kV. Cables are single conductor shielded and provided with Kerite type FR fire resistant jackets.

n. Power cables in 4.16 kV service are HT or HV Kerite insulated cables rated at 5 kV. Cables are triplexed or single conductor, nonshielded, and provided with Kerite type NS neoprene or CSPE sheath jackets.

o. Control cables are of multiconductor construction with either cross-linked polyethylene, ethylene propylene rubber or silicon rubber insulation with jackets of Hypalon, neoprene or asbestos braid. Control cables are rated at 600 Volts. Low voltage instrumentation cables are of multiconductor construction with either cross-linked polyethylene, ethylene propylene rubber or silicon rubber insulation with jackets of Hypalon, neoprene or asbestos braid with voltage ratings suitable for the application. Specialty low voltage instrumentation cables are supplied by OEM with unique constructions and voltage ratings specific for their equipment. Control cables for use in underground ducts are insulated with cross-linked polyethylene, and jacketed with neoprene, hypalon or, polyvinyl chloride. Low voltage instrument cables have total coverage electrostatic shielding, or electrostatic shielding covering individual twisted pairs or triads.

p. The normal current rating of all insulated conductors is limited to the continuous value which does not cause excessive insulation deterioration from heating. Selection of conductor sizes is based on "Power Cable Ampacities," published by the Insulated Power Cable Engineers Association.

q. All cables, terminations and splices within the containment associated with safety-related equipment are qualified by being type tested for the loss-of-coolant accident (LOCA) environmental conditions.

r. The electrical systems have been designed in accordance with "IEEE Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," IEEE No. 308 - 1974.

s. Electrical penetration qualification tests were combined using the postulated worst combination of environmental conditions as described in Section 14.20. The

electrical penetrations were tested to verify leak integrity and also electrical integrity on those penetrations carrying ESF or reactor protective circuits. All materials used in the construction of electrical penetrations were qualified for radiation exposure of 10^8 rads either by materials manufacturer's or the penetration manufacturer's tests.

Electrical penetration assemblies were supplied by the Amphenol Corporation and the Conax Buffalo Corporation.

The Amphenol Type 1, 15 kV, medium voltage power prototype penetration canister was tested by enclosing the inside containment end of the canister in a tank and subjecting it to steam made with 1720 ppm borated water. The penetration was subjected to 275°F, at 41 psig for 15 minutes. This temperature was reached within 30 seconds. The next 45 minutes the penetration was at 260°F, at 33 psig. The following 23 hours were above 250°F, at 30 psig. Throughout the entire test the leak rate was monitored using helium and found to be within the required 1×10^{-6} standard cubic centimeters per second of dry helium.

The Amphenol Types 1, 2, and 3, low voltage power, control and instrumentation, thermocouple and coaxial penetration canisters were tested in a prototype canister containing two coaxial conductors and three or more of each other type of conductor.

The test was performed in the same manner as on the Amphenol Type 1, except that during the first 15 minutes the unit was subject to 279°F, at 44 psig, the next 45 minutes 265°F, at 35 psig, the next 23 hours were above 250°F, at 30 psig. The leak rate again was within 1×10^{-6} standard cubic centimeters per second of dry helium. During the test, the 480 Volt power conductors and the 120 Volt control and instrumentation conductors were energized at their operating voltage and there was no excessive leakage current. The power, control, and instrumentation conductors were also terminated in the manner they will be in the field in order to qualify the termination methods to be used inside containment.

All Amphenol prototype canister penetration assemblies successfully passed the environmental test as described above.

Conax Type 1, 2, and 4 penetration assemblies are header plate type and were designed, fabricated and prototype-tested to withstand Design Basis Event environmental conditions described in Section 14.20.

All Conax penetration assemblies successfully passed Design Basis Event environmental testing.

8.1.3 SHARED ELECTRICAL EQUIPMENT

The following electrical auxiliary system equipment is shared by Units 1 and 2.

a. Service Transformer P-13000-1 and P-13000-2
b. 13 kV Service Bus 11 and 21
c. 13 kV Service Bus 12 and 22
d. 500 kV Red Bus
e. 500 kV Black Bus
f. 125 Volt DC Plant Control Batteries 01, 11, 12, 21 and 22
g. 250 Volt DC Emergency Pump Battery 13 and 23

h. 125 Volt DC Busses 11, 12, 21, and 22

i. 250 Volt DC Bus 13

j. 0C Diesel Generator

k. 125 Volt DC Unit Control Panel 24

8.2 NETWORK INTERCONNECTION

8.2.1 DESIGN BASIS

The 500 kV switchyard is designed to be the interconnection point between the power plant and the 500 kV network. It is designed to function reliably under all conditions of power plant operation. It will furnish service startup power to the power plant, and reliably function and isolate trouble in the power system grid under power system normal and abnormal conditions (Figures 8-7 and 8-12).

Electrical power from the 500 kV network to the switchyard is supplied by three physically independent transmission lines designed and located to minimize the likelihood of their simultaneous failure under operating, postulated accident and postulated adverse environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system are also provided. The switchyard is designed with duplicate and redundant systems - e.g., two battery systems, two trip coils per breaker, two protective relay schemes, and two auxiliary AC supplies from plant emergency busses.

Load flow and stability studies were performed when the plant was licensed to indicate that the tripping of one or both fully loaded Calvert Cliffs generating units would not impair the ability of the system to supply plant service. These studies were made at the projected peak load conditions and also at minimum load conditions when the two Calvert Cliffs units were supplying the entire Baltimore system. Simultaneous loss of the units at either load period did not overstress the ability of the planned transmission grid, either from a thermal or voltage standpoint, to supply power to the area and the plant. In addition, some major transmission circuits were assumed to be out of service at the time. The spinning reserve policy of the Pennsylvania-New Jersey-Maryland (PJM) Interconnection, of which we are a member, is to maintain enough reserve capacity synchronized to the system to cover the largest single contingency in PJM.

High-speed clearing of faults and selective reclosing assure maximum availability of power and system grid stability. Transient stability under fault conditions has been verified by a digital computer study which included the interconnected systems and analyzed for various contingencies, including the failure of a 500 kV breaker to trip under a fault condition.

8.2.2 DESCRIPTION AND OPERATION OF SWITCHYARD

The switchyard is shown in Figure 8-2. Major switchyard components are described in Table 8-1.

The switchyard operates at 500 kV and the equipment is selected to have the capability of isolating system and switchyard faults with a minimum effect on the stability of the power system grid.

The switchyard is arranged in a breaker-and-one-half arrangement and has two bays consisting of three breakers each and one bay of two breakers with two main busses and connections to the generator main power transformers, the two plant service transformers, and three 500 kV lines to the 500 kV network. Each line has sufficient capacity to carry the entire output of both turbine generators.

The switchyard 500 kV power circuit breakers, the circuits from the switchyard to the generator main power transformers, and from the switchyard to the plant service transformers, are provided with disconnect switches or isolating links to permit isolating

any power circuit breaker or any circuit from the switchyard while allowing the 500 kV busses to remain tied together.

The 500 kV lines to the 500 kV network consist of three physically independent lines designed and located to minimize the likelihood of simultaneous failure under operating, postulated accident and postulated adverse environmental conditions.

Zone relaying is provided for the circuit from the switchyard to the generator main power transformers and for the two switchyard main busses. The main bus zones include the circuit from the switchyard to the plant service transformers. New relaying has been added to the circuit from the switchyard to the plant service transformers to monitor the circuit for the existence of an open phase condition. If an open phase condition is detected an alarm is initiated to the main Control Room annunciator system to alert the operators of the event.

Conservative margins have been allowed between the maximum expected fault duty and the rating of the equipment. Reliability is assured by the switchyard arrangement which utilizes a three bay breaker-and-one-half scheme. With this scheme, any breaker may be removed from service without affecting switchyard operation.

All circuits or portions of the busses and overhead lines have primary and backup relaying. The outgoing lines have two sets of high-speed relays. The circuit breakers have dual trip coils on separate isolated DC control circuits, and breaker failure relays to trip the adjacent breakers.

8.2.3 DESCRIPTION AND OPERATION OF SWITCHYARD CONTROL SYSTEM

The 480/277 - 120 Volt AC power is supplied from two 4.16 kV/480-277 Volt, 500 kVA, three-phase transformers which are located in the switchyard. The service transformers are supplied by two isolated 4.16 kV feeders from the plant. These 4.16 kV feeders are supplied from separated emergency 4.16 kV busses in the plant (4.16 kV Unit Busses 11 and 21). Each of the switchyard power transformers supplies the 60 Hertz power requirements for the switchyard. The AC load is divided between two low voltage manual changeover power panels to allow for loss of one station power transformer.

The Asea Brown Boveri, Inc. (ABB) Type 550 gas circuit breakers are installed in the 500 kV switchyard. The ABB Type 550 gas circuit breaker is a motor- and spring-operated breaker that uses SF6 gas as the interrupting medium.

The 125 Volt DC auxiliary power is supplied from two 59-cell batteries which are located in the switchyard. Each can supply the switchyard DC power requirements for eight hours without recharging. Two battery chargers are supplied to keep the batteries fully charged and, under normal conditions, to supply the 125 Volt DC power requirement. A battery switch and fuse are used to isolate the battery from the DC power panels in the event of a battery fault. The power panels can then be energized by their respective battery chargers. Each battery charger is fed from a separate 480 Volt, 60 Hertz power panel. The DC load is divided between two power panels such that the loss of one power panel will not disrupt the 125 Volt DC auxiliary power feeds necessary to maintain switchyard operation.

The monitoring of battery charger operation and battery voltage for each battery system is provided by individual alarms in the switchyard control house plus a general alarm monitored in the plant Control Room.

Normal Operation - The switchyard normally operates energized with all breakers closed. Opening and closing of the breakers can be accomplished locally in the switchyard control house or remotely from the plant Control Room. Indicating lamps in the plant Control Room indicate circuit breaker status.

Testing - The power circuit breakers may be removed from service and tested. Individual components and partial circuit tests may be carried out while the circuit breakers are carrying load.

The relays are supplied with test switches that will permit the removal from service of one relay of the two independent sets of protective relays for maintenance.

8.2.4 DESCRIPTION AND OPERATION OF SMECO LINE

8.2.4.1 Design Basis

The SMECO power source has the capability to supply the power necessary to maintain Unit 1 and Unit 2 in a safe shutdown condition. The SMECO system can be used to energize 13 kV Service Bus 23 that then can be used to supply either 13 kV Bus 11 or 21 as required. Once 13 kV Service Bus 23 is energized from SMECO, it could then be held in this "ready" state as required. The switchyard power source will then be used to operate both Units 1 and 2 auxiliary loads. Upon loss of the switchyard power source, the SMECO system could then be used to supply any two 4.16 kV ESF busses, one for each Unit (Section 8.3.2), through either 13 kV Service Bus 11 or 21.

A manual Engineered Safety Feature Actuation System (ESFAS) LOCI and shutdown sequencer actuation is provided in the Control Room to ensure that the SMECO system is loaded in an orderly manner to minimize system transients.

8.2.4.2 Description and Operation

The SMECO system is shown on Figures 7-10 Sh. 1, 7-22 Sh. 1, 8-1, and 8-9. The SMECO System is also described in Table 8-1A. It consists of a single, direct buried cable from the SMECO substation to 13 kV Service Bus 23 via a manual load break switch which can be used to supply warehouse power during normal operation. When the SMECO line is used as one of the independent offsite circuits, the warehouse feed will be disconnected and 13 kV Service Bus 23 and either Bus 11 or 21 will be energized. The SMECO system will have a capability at all times of 5000 kW. Electrical indication is provided in the Control Room for bus voltage, bus current and power usage.

TABLE 8-1

RATINGS AND CONSTRUCTION OF MAJOR SWITCHYARD COMPONENTS

Breakers	-	500 kV nominal
	-	3,000 A continuous
	-	36,370 MVA interrupting
Insulators	-	1,800 kV BIL
Main Bus	-	2,500 A
Bay Bus	-	2,500 A
Disconnect Switches	-	3,000 A continuous
	-	70,000 A momentary

TABLE 8-1A

RATINGS AND CONSTRUCTION OF SMECO SYSTEM

SMECO	-	5000 kW, 69/13.2 kV, regulated to 13.8kV within ± 5% voltage variation, 3-phase, 60 Hz, 328A continuous
Cable	-	500 MCM, N-TRIPLEX, Kerite, 15 kV direct burial (from SMECO STA to WHSE)
	-	750 MCM, CU TRIPLEX, Kerite (from WHSE to Bus 23)
Load Break Switch	-	13.8 kV nominal, 200 Amp, 300 MVA interrupting, manually operated

8.3 STATION DISTRIBUTION

8.3.1 13.8 kV SYSTEM

8.3.1.1 Design Basis

The 13.8 kV system is designed to function reliably and supply power to plant auxiliaries during normal operation and under accident conditions. The system supplies power to the RCPs directly and to the 4.16 kV system through voltage regulators and unit service transformers.

8.3.1.2 Description and Operation

The plant 13.8 kV system is shown on Figure 8-1, Electrical Main Single Line Diagram. A description can be found in Table 8-2.

The 13.8 kV system for the plant consists of two plant service transformers (P-13000-1 and P-13000-2), five service busses, six voltage regulators with associated transfer switches, and eight RCP busses. The capacities of the two plant service transformers and associated switchgear and cable are such that either one of the transformers can supply the total auxiliary load of the plant. Each RCP is attached to separate 13.8 kV busses which are fed from either of the two plant service transformers.

In the event of failure of one of the transformer supplies, the RCPs can be manually transferred to the alternate supply.

Service Busses 11 and 21 supply power to the unit service transformers, and are equipped with bus tie breakers.

The future capability of regulating plant auxiliary power distribution system voltage is provided by 13.8 kV voltage regulators. A voltage regulator with associated transfer switches are installed at the 13.8 kV level for each U-4000 transformer. The transfer switches are provided to bypass and isolate the voltage regulators during regulator maintenance. Six regulators and six transfer switches, three per unit, are located outdoors on the western side of the plant access road in the vicinity of the P-13000 Station Service Transformers. The voltage regulator units are provided with a deluge water spray system designed to the requirements of NFPA 15, Standard for Water Spray Fixed System for Fire Protection.

The unit switchgear for the RCPs is metal-clad with removable circuit breakers and is designed for indoor installation. The switchgear for Service Busses 11, 12, 21, 22 and 23 is also metal-clad with removable circuit breakers, but is designed for outdoor installation,

Relay protection, including open phase detection, ground connections, and structural safeguards are provided to assure adequate personnel protection and to prevent or limit equipment damage during system short circuits.

Operation - During normal operation, both plant service transformers are energized and share the total plant auxiliary load. All RCP motors for one unit are fed from one common plant service transformer and motors of the second unit from the other transformer.

Operation of all 13.8 kV equipment is effected and monitored in the Control Room, with the exception of site facility power feeds, which have local control and indicators but provide Control Room annunciation. Circuit breaker position is indicated by red and green lights. Typical functions annunciated are circuit

breaker trip, bus undervoltage, and motor overload. Electrical indication is provided in the Control Room for bus voltage, service transformer current and power, bus current, and motor current.

Testing - Portions of the system can be tested during normal operation. For example, various backup protective relays may be withdrawn from their cases for inspection, calibration, and test. While equipment is shut down, circuit breakers may be placed in the test position and the control circuit functionally tested.

8.3.2 4.16 kV SYSTEM

8.3.2.1 Design Basis

The 4.16 kV system is designed to function reliably and supply power during normal operation and under accident conditions. The system will supply power to the 4.16 kV auxiliary loads from the 13.8 kV system through the six unit service transformers. There are six 4.16 kV busses per unit, two of which supply power to the ESF. The ESF electrical system incorporates the two-channel concept, i.e., independent electrical controls and power systems supply redundant 4.16 kV ESF. The 4.16 kV ESF electrical system meets the single failure criterion defined in IEEE 279, Section 4.2, and is designed as a Class 1E system.

8.3.2.2 Description and Operation

Description - The plant 4.16 kV system is shown on Figures 8-1, 8-4, and 8-10 and in Table 8-3; it consists of seven unit service transformers (OX01, U-4000-11, U-4000-12, U-4000-13, U-4000-21, U-4000-22, and U-4000-23), fourteen 4.16 kV busses (07, 11, 12, 13, 14, 15, 16, 17, 21, 22, 23, 24, 25 and 26), the motor feeder circuits, and 480 Volt load center feeder circuits.

The 4.16 kV busses consist of metal-clad switchgear assemblies with draw out circuit breakers. Relay protection, including open phase detection for the 4.16 kV buses that feed the ESF loads, ground connections, and structural safeguards are provided to assure adequate personnel protection and to prevent or limit equipment damage during system short circuits. This equipment, except open phase detection associated relying is designed to function properly while subject to SSE accelerations.

Two of the 4.16 kV busses for each unit (11 and 14 for Unit 1, 21 and 24 for Unit 2) supply power to ESF. The two busses feed redundant equipment. Each of the two busses per unit can be supplied from separate EDGs. These busses are located in separate Seismic Category I rooms. Feeder cables from the EDGs and from ESF equipment are also located within Seismic Category I structures, and separation is maintained between the feeder cables of the two busses.

Operation - Whenever offsite power is available, the 4.16 kV system is supplied by the 13.8 kV system through the six unit service transformers. Each 4.16 kV bus can be fed from either of two 13.8 kV sources of auxiliary power through different unit service transformers. Normally, Busses 11, 12, and 13 are fed from unit Service Transformer U-4000-11, Bus 14 from U-4000-21, Bus 21 from U-4000-12, Busses 22, 23, and 24 from U-4000-22, Busses 15 and 16 from U-4000-13, and Busses 25 and 26 from U-4000-23. Transfers, if required, are performed manually.

The ESF busses are equipped with one set of undervoltage sensing relays, and upon receipt of a two-out-of-four logic signal, the diesel generators are energized to supply power (Section 8.4.1).

With the exception of the non-Class 1E feeders to the South Service Building, the Control Room Chilled Water System, and the Access Control Expansion Area, all 4.16 kV equipment can be operated from the Control Room. Breaker status is indicated in the Control Room by red and green lights. Typical functions annunciated are circuit breaker trip, motor overload, bus undervoltage and, for most breakers feeding Class 1E loads, blocked auto start due to lockout or discharged springs. Electrical parameters such as bus voltage, bus current, motor current, and transformer current and power are displayed in the Control Room.

Testing - The 4.16 kV ESF are designed to permit testing during normal plant operation.

8.3.3 480 VOLT SYSTEM

8.3.3.1 Design Basis

The 480 Volt system is designed to function reliably and supply power during normal operation and under accident conditions. The system will supply power to the 480 Volt auxiliary loads from the 4.16 kV system through the 4160/480 Volt unit service transformers. Four of the unit load centers and two motor control centers (MCCs) supply power to the ESF. The ESF electrical system incorporates the two-channel concept, i.e., independent electrical controls and power systems supply redundant 480 Volt ESF. The 480 Volt ESF electrical system meets the single failure criterion as defined in IEEE 279, Section 4.2, and is designed as a Class 1E system.

8.3.3.2 Description and Operation

Description - The 480 Volt system is shown on Figures 8-1, 8-3, and 8-11, Single Line, Meter and Relay Diagrams, 480 Volt Unit Busses. The system is also described in Table 8-4.

The 480 Volt system for the plant consists of double-ended unit load centers, single-ended unit load centers, and MCCs. Power for each load center bus is supplied from a separate 4.16 kV/480 Volt unit service transformer. The MCC feeders from the 480 Volt load center busses are arranged so that each end of a double-ended MCC is fed from a different 480 Volt load center bus.

The 480 Volt unit load centers consist of metal-clad switchgear with draw-out air circuit breakers. The MCCs are metal enclosed with removable starter and breaker combination modules. Relay protection, ground connections, and structural safeguards are provided to assure adequate personnel protection and to prevent or limit equipment damage during system short circuits. This equipment is designed to function properly while subjected to SSE accelerations.

Four of the 480 Volt unit load centers for each unit (11A, 11B, 14A, and 14B for Unit 1; 21A, 21B, 24A, and 24B for Unit 2) supply power to ESF. Busses 11A/B and busses 14A/B feed redundant ESF. The redundant busses are supplied from separate EDGs through the 4.16 kV/480 Volt unit service transformers. Similarly, two of the MCCs for each unit (MCC 104R and MCC 114R for Unit 1, MCC 204R and MCC 214R for Unit 2) supply power to redundant ESF. Each of the two busses per unit are supplied from separate EDGs via the 480 Volt unit load centers. Redundant ESF 480 Volt busses are located in separate Seismic Category I rooms. Feeder cables from ESF equipment are also located within Seismic Category I structures, and separation is maintained between the feeder cables of the two redundant systems.

Operation - During normal operation all incoming bus breakers and MCC feeder breakers are closed. The tie breakers between MCCs 104R-114R, 101AT-101BT, 204R-214R, and 201AT-201BT are normally open and are closed only for emergency or maintenance.

Key interlocks are provided to prevent simultaneous closure of the tie breakers and both MCC feeder breakers. The operation of the bus tie is a manual function only.

The 480 Volt ESF busses are supplied with power from the diesel generators through the 4.16 kV/480 Volt service transformers in case of failure of the preferred source of power to the 4.16 kV busses (Section 8.4.1).

Operation of ESF equipment may be controlled from the Control Room as may other essential equipment. The status of this equipment (breaker and starter position) is indicated by red and green lights in the Control Room. All equipment has local indication of breaker or starter status. Circuit breaker trip and motor overload are annunciated in the Control Room. Electrical indication for bus voltage and bus current is provided in the Control Room.

Testing - The 480 Volt ESF are designed to permit testing during normal plant operation.

8.3.4 CONTROL ELEMENT ASSEMBLY POWER SUPPLY

8.3.4.1 Design Basis

The control element assembly (CEA) power supply is designed as a stable, reliable power supply for the CEAs.

8.3.4.2 Description and Operation

The CEA power supply for one unit consists of two ride-through flywheel motor-generator power systems. The motor-generator sets have the capability, individually or in parallel, to hold all the control elements and sustain the motion of any element already being stepped during a one second transient in the station service voltage. Each motor-generator set is connected to a different 480 Volt load center.

8.3.5 125 VOLT DC AND VITAL 120 VOLT AC SYSTEMS

8.3.5.1 Design Basis

The 125 Volt DC and vital AC systems are designed to furnish continuous power to the plant vital instrumentation and control systems regardless of auxiliary electrical system condition. The reliability of the system is increased by redundancy of vital equipment and circuits.

8.3.5.2 Description and Operation

Description - The 125 Volt DC and 120 Volt vital AC systems are shown on Figure 8-5, Single Line Diagram (Section 8.4.3). System data can also be found in Table 8-5.

The 125 Volt DC and 120 Volt vital AC systems for the plant are divided into four independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus, multiple DC unit control panels, and two dual inverters. Each inverter has an associated vital AC distribution panelboard.

Power to the DC bus, DC unit control panels, and dual inverters is supplied by the station batteries (Table 8-10) and/or the battery chargers.

Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system.

A reserve 125 Volt DC system for the plant is completely independent and isolated from all four separation groups, yet is capable of replacing any of the 125 Volt DC batteries. This system consists of one battery, one battery charger and the associated DC switching equipment. Only the battery may be transferred for replacement duty.

As shown on Figure 8-9, the 125 Volt DC Busses 11 and 22 are a part of Load Group A, and 125 Volt DC Busses 12 and 21 are a part of Load Group B. The 125 Volt DC Bus 11 provides control power for equipment associated with Load Group A for both units. The 125 Volt DC Bus 21 provides control power for equipment associated with Load Group B for both units. The 125 Volt DC Busses 12 and 22 are used to supply power to the computer inverters, Control Room emergency lighting, and two channels of the 120 Volt vital AC system. The 125 Volt DC Bus 01 is not associated with a load group except while connected to one of the DC busses.

There is one battery charger fed from Unit 1 and another battery charger fed from Unit 2 connected to each 125 Volt DC bus. The AC power for both battery chargers per bus is obtained from the same load group. The reserve battery is connected to its own charger when it is not connected to a safety-related 125 Volt DC bus.

The 120 Volt vital AC system provided for each unit has four separate distribution panelboards which provide power for the four RPS channels and the four ESF and auxiliary feedwater actuation systems channels. Each panelboard is supplied by a dual inverter with its own DC feeder from a separate battery. Each dual inverter has two built-in independent inverters, one to serve as primary and the other as backup. In the dual inverter, 120 Volt AC power output can be manually switched from primary inverter to backup inverter. Each 120 Volt AC distribution panelboard can be manually switched from the dual inverter to a 120 Volt AC bus fed from an ESF MCC through a regulating transformer.

The 125 Volt DC system and the 120 Volt vital AC system are ungrounded and equipped with ground detectors.

Each of the four 125 Volt DC power sources is equipped with the following instrumentation in the Control Room to enable continual operator assessment of 125 Volt DC power source condition.

a. DC bus undervoltage alarm
b. Battery current indication
c. Charger current indication
d. Charger malfunction alarm (including input AC undervoltage, output DC undervoltage and output DC overvoltage)
e. DC bus voltage indication, and
f. DC ground indication

The undervoltage relay features an extra high dropout characteristic and is designed specifically to monitor the charging supply for a station battery and sound an alarm if this supply fails.

In addition to the above instrumentation, continuous monitoring of the battery's connection to the bus and of battery circuit continuity is provided.

The 125 Volt DC system has been designed to function properly while subjected to SSE accelerations.

Operation - During normal operation, all battery chargers are energized and maintain a constant voltage to supply the batteries with sufficient current to keep them fully charged and maintain the steady state load of DC instruments, control circuits, and inverters. In the event of loss of auxiliary system power, the batteries will continue to supply the required DC and vital AC equipment. When AC power is regained from the diesel generators, the battery chargers will be re-energized and resume normal operation. The batteries are sized to supply the anticipated DC and vital AC load, without support from battery chargers, for a period of two hours (Section 8.4.3).

When the reserve battery system is not in use, the spare battery charger is energized to maintain a constant voltage to supply the battery with sufficient current to keep it fully charged. If the battery is connected to one of the 125 Volt DC busses, its charger is disconnected before the battery is connected to the bus. The reserve battery is kept then fully charged by the chargers on the associated DC bus. During the battery transfer, the DC bus voltage is maintained by these chargers.

8.3.6 250 VOLT DC EMERGENCY PUMP SYSTEM

8.3.6.1 Design Basis

The 250 Volt DC emergency pump system is designed to supply power to the various plant backup lube oil and seal oil emergency pumps in case of loss of auxiliary AC power or failure of the normal AC pumps. There are no loads connected to the 250 Volt DC bus that are related to the functioning of ESF.

8.3.6.2 Description and Operation

Description - The 250 Volt DC emergency pump system is shown on Single Line Diagram, Figure 8-5. Additional data can be found in Table 8-6.

The single 250 Volt DC emergency pump system for the plant consists of one MCC, two battery chargers, and two batteries. Only one battery is connected to the MCC. The backup battery will be used when the first battery is out-of-service. The battery chargers are sized such that in combination they are capable of supplying the continuous load of the largest connected motor. Each battery charger is fed from separate ESF 480 Volt load centers (one from Unit 1 and one from Unit 2).

Each battery consists of lead-acid cells electrically-connected in series to establish a nominal 250 Volt supply. The grid structure is of the lead-calcium design. Each battery is sized to supply the total connected load for one hour without support from battery chargers. In addition, they each are of sufficient rating to start the two largest motors simultaneously while all other motors are operating at full load without allowing the battery voltage to fall below 210 Volts.

The 250 Volt DC emergency pump system is ungrounded and equipped with ground detectors. Battery charger current and 250 Volt DC bus voltage are displayed in the Control Room. Typical functions annunciated in the Control Room are 250 Volt DC bus undervoltage and battery charger undervoltage.

Operation - During normal operation, both battery chargers are energized maintaining a constant voltage such that they supply the battery connected to the MCC with sufficient current to keep it fully charged. In case of loss of auxiliary system power or failure of a normal AC pump, the connected battery will supply the necessary power for backup pump operation. After availability of AC power from the diesel generators, the battery chargers can be manually energized to resume normal operation.

8.3.7 INSTRUMENT AC SYSTEM

8.3.7.1 Design Basis

The 208-120 Volt instrument AC system is designed to furnish power to all plant instruments other than those supplied from the DC and the vital AC systems. In addition, it is utilized as a backup supply of power for the computer and the preferred source of power for the public address system.

8.3.7.2 Description and Operation

Description - The instrument AC system for each unit is divided into two panelboard sections. Each section is supplied by a single three-phase transformer connected to an ESF MCC. In case of loss of normal auxiliary power, the transformers will automatically be energized by the EDGs. A manually operated bus tie switch has been provided between the two sections.

Operation - During normal operation, the bus tie switch is open and both transformers are energized and supply their associated panelboards. Should the power from one transformer source be interrupted, the bus tie is closed and both panelboard sections are fed from a single transformer.

TABLE 8-2

RATINGS AND CONSTRUCTION OF 13.8 kV SYSTEM COMPONENTS

Plant Service Transformers P-13000-1 and P-13000-2 2-winding, Wye/Wye	-	500/14 kV, 3-phase, 60 Hz, 60/80/100 MVA OA/FA/FOA
Voltage Regulators	-	13.8 kV, 3-phase, 2 MVA ± 10% voltage regulation
Transfer Switches	-	15 kV, 1200 A
Service Busses	-	3000 A continuous rating
Unit Busses	-	1200 A continuous rating
Circuit Breakers	-	1000 MVA interrupting rating, 3000 A and 1200 A continuous rating

TABLE 8-3

RATINGS AND CONSTRUCTION OF 4.16 kV SYSTEM COMPONENTS

Unit Service Transformers	-	13.8 kV/4.16-4.16 kV, 3-phase, 60 Hz 12/16/20 MVA, OA/FA/FOA
Bus	-	2000 A continuous rating
Incoming Breakers on Busses 12, 13, 15, 16, 22, 23, 25 & 26	-	2000 A continuous, 250 MVA interrupting
Feeder Breakers and Incoming Breakers on Busses 11, 14, 21 & 24	-	1200 A continuous, 250 MVA interrupting

TABLE 8-4

RATINGS AND CONSTRUCTION OF 480 VOLT SYSTEM COMPONENTS

480 Volt Unit Load Centers		
Transformers	-	1000 kVA OA or AA, 3-phase, 60 Hz, 4160/480 Volts
Bus	-	1600 A continuous
Breakers, Metal-clad	-	25,000 A rms symmetrical minimum interrupting rating
480 Volt MCCs		
Horizontal Bus	-	600 A continuous, 25,000 A rms symmetrical
Vertical Bus	-	300 A continuous, 25,000 A rms symmetrical
Breakers, Molded Case	-	25,000 A rms symmetrical minimum interrupting rating

TABLE 8-5

RATINGS AND CONSTRUCTION OF 125 VOLT DC AND VITAL 120 VOLT AC SYSTEM COMPONENTS

125 Volt DC Bus	-	1200 A continuous, 10,000 A momentary
125 Volt Unit Control Panels	-	600 A continuous, 10,000 A momentary
125 Volt Bus Fuse Switches	-	10,000 A interrupting at 125 Volt DC
Battery Chargers	-	500 A continuous output
Dual Inverters	-	7.5 kVA, 0.7 P.F., 120 Volt AC continuous output, 60 Hz, single phase
Vital Instrumentation Bus Fuse Switches	-	10,000 A interrupting at 120 Volt AC
125 Volt Unit Control Panel Fuse Switches	-	10,000 A interrupting at 125 Volt DC
Battery Fuse	-	1,600 A continuous rating

TABLE 8-6

RATINGS AND CONSTRUCTION OF 250 VOLT DC EMERGENCY PUMP SYSTEM COMPONENTS

250 Volt DC Bus	-	1200 A continuous, 20,000 A momentary
250 Volt Fuse Switches	-	200,000 A interrupting at 250 Volt DC
Battery Chargers	-	150 A output
Battery Fuse	-	1200 A continuous rating
Batteries	-	Lead Calcium, 250 Volt, nominal 1950 A.H., 8-hour discharge rate at 77°F, minimum capacity is based on the following load cycle: First minute 585 A Next 4 minutes 270 A Next 1 minute 1138 A Next 54 minutes 661 A

TABLE 8-10

RATINGS AND CONSTRUCTION OF STATION BATTERIES

Reserve Battery - Cells are lead acid with lead calcium grid structure, 125 Volt, nominal 1500 Amp-hour, 8-hour discharge rate at 77°F.

Control Battery - Cells are lead acid with lead-calcium grid structure, 125 Volt, with nominal capacities of 1500 Amp-hour or 1950 Amp-hour, 8-hour discharge rate at 77°F, minimum capacity based on following load cycle:

BATTERY NO.	Two-Hour Accident Load Cycle TIME INTERVAL	AMPERES
11	First minute	512
	Second minute	451
	Next 117 minutes	250
	Last minute	325
12	First minute	291
	Second minute	272
	Next 117 minutes	286
	Last minute	319
21	First minute	473
	Second minute	394
	Next 117 minutes	225
	Last minute	319
22	First minute	317
	Second minute	299
	Next 117 minutes	311
	Last minute	341
01	Any of the above[a]	

[a] Battery 01 can be a replacement for any of the other four batteries and must be able to handle any of the load cycles shown.

The load cycle delineated above is based on anticipated breaker operations required during an accident on Unit 1 and a simultaneous undervoltage on Units 1 and 2. The load consists primarily of Control Room emergency lighting, vital bus dual inverters, plant computer inverters, DC-operated controls and instruments. The batteries are sized to carry the above loads for the duration stated after a loss of AC power.

Actual load cycle testing is performed using the two-hour scenario. Battery 01 is tested using a unique load cycle incorporating the largest discharge of any of the Stations Batteries' time intervals.

8.4 EMERGENCY POWER SOURCES

The emergency power sources are designed to furnish onsite power (upon a loss of normal supplies of power) to reliably shut down the plant and maintain it in a safe shutdown condition under all conditions, including accidents. The emergency power sources are part of the ESF electrical system and are designed as Class 1E systems. The diesel generator sets selected for use as standby power supplies have the capability to: (1) power the ESF in rapid succession, and (2) supply continuously the sum of the loads needed to be powered at any one time.

The EDGs are designed to provide a dependable onsite power source capable of starting and supplying the essential loads necessary to safely shut down the plant and maintain it in a safe shutdown condition under all conditions. Four diesel generators are provided for the plant although each Unit requires only one diesel generator to supply the minimum power requirements for its ESF equipment.

If one of the two diesel generators (per Unit) should fail to start or carry load, the system continues to provide an electrically independent channel of emergency power to the Unit. Reliability is increased by the adoption of the two-channel concept, i.e., independent electrical controls and sources supply redundant AC and DC ESF equipment.

Diesel Generator 1A is connected to 4.16 kV Bus 11, 1B is connected to 4.16 kV Bus 14, 2A is connected to 4.16 kV Bus 21, and 2B is connected to 4.16 kV Bus 24 as shown on Figure 8-1.

The EDGs are designed to reach rated speed and voltage and to start accepting load within 10 seconds after the receipt of a starting signal. The diesel generators and their auxiliaries are designed to withstand Seismic Category I accelerations and are installed within Seismic Category I structures.

The Station Blackout (SBO) diesel generator is designed to provide a power source capable of starting and supplying the essential loads necessary to safely shutdown one unit and maintain it in a safe shutdown condition during a SBO event. The SBO diesel generator has the ability to supply any of the four ESF busses.

The SBO diesel is started manually. The diesel is loaded onto a bus when it is determined that the EDG dedicated to that bus is not available to supply the plant loads. The SBO diesel is capable of supplying the same emergency plant loads as the EDGs.

The predicted accident loads for large break LOCA, small break LOCA, and main steam line break are less than 3000 kW.

The diesel generators are started by either a 4.16 kV bus undervoltage or SIAS; however, in the latter case, actual transfer to the bus is not made until the preferred source of power is actually lost. When all four diesel generators are available, the design provides two independently-capable and concurrently-operating systems for safety injection, containment spray, and miscellaneous 480 Volt auxiliary devices for the unit incurring the accident. In addition, the design provides power to operate two sets of equipment for shutting down the non-accident unit; including, for example, two saltwater pumps, two service water pumps, two auxiliary feedwater pumps, containment cooling fans, and emergency turbine auxiliaries.

The independence and redundancy of the auxiliary power system features that initiate and control the connection of diesel generators to the AC emergency busses are described as follows:

a. Each of the redundant 4 kV emergency busses is equipped with one set of four redundant and independent undervoltage relays. This set of relays has three elements to sense various undervoltage conditions. The first set of elements are set to provide a

two-out-of-four undervoltage signal upon loss of bus voltage. The second set of elements are set to provide a two-out-of-four undervoltage signal on a transient bus undervoltage. The third set of elements are set to provide a two-out-of-four undervoltage signal on a steady state bus undervoltage. Upon coincidence of two-out-of-four, the preferred supply circuit breakers of the bus with which the undervoltage relays are associated are tripped. The signal from the undervoltage relays of a particular load group acts on circuit breakers of the same load group only.

b. The coincidence of two-out-of-four undervoltage signals from the four relays also initiates starting of the diesel generator which is aligned to the bus with which the undervoltage relays are associated. Again, independence is assured since Load Group A relays initiate starting of a diesel generator redundant to that initiated by Load Group B relays.

c. The SIAS is also capable of initiating diesel generator start. As in the case of undervoltage signal, independence is assured since Load Group A SIAS initiates starting of a diesel generator redundant to that initiated by SIAS Load Group B.

d. For each emergency bus, each control circuit which functions to connect an EDG to the 4 kV emergency busses is exclusively associated with a single diesel generator only. That is, there are four controlling circuits, one for each of the diesel generators. Each of these circuits is physically isolated from, and independent of, the others.

The above system features are designed in accordance with General Design Criterion 18 (February 20, 1971) to permit periodic inspection and testing.

During accident conditions accompanied by simultaneous loss of offsite power, the loss-of-coolant incident (LOCI) sequencers will start automatically to load the diesel generators sequentially. Similarly, the shutdown sequencer for the non-accident unit will start automatically. The LOCI sequencers initially block the SIAS and the containment spray actuation signal to the equipment to be sequenced and then unblock in programmed steps, as shown in Table 8-7. The sequencing is performed so that essential loads are started within the time limits of the appropriate safety analyses.

8.4.1 FAIRBANKS MORSE EMERGENCY DIESEL GENERATORS

8.4.1.1 Description

Three of the four emergency power sources consist of 4.16 kV, three-phase, 60-cycle Fairbanks Morse diesel generators with nominal continuous ratings given below. All generator sets are physically separated and electrically isolated from each other.

The three Fairbanks Morse diesel generators have been upgraded and are rated as follows:

3000 kW	Continuous	(consumes approximately 3.87 gpm fuel oil)
3300 kW	2000 hour	
3500 kW	200 hour	(consumes approximately 4.454 gpm fuel oil)
3600 kW	30 minutes	

At no time during the loading sequence will the frequency and voltage decrease to less than 95% of normal and 75% of normal, respectively. During recovery from transients caused by step load increases or resulting from the disconnection of the largest single load, the speed of the diesel generator will not exceed nominal speed plus 75% of the difference between nominal speed (900 RPM) and 115% of nominal speed. Voltage is restored to within 10% of nominal in less than 40% of each load sequence time interval. Frequency is restored to within 2% of nominal

in less that 60% of each load sequence time interval. The nominal values of speed, voltage and frequency are defined in the Technical Specifications.

8.4.1.2 Auxiliary Systems

Separate MCCs are provided for each diesel generator. Each MCC supplies the engine room ventilation fan, fuel oil transfer pump, engine standby warming systems, and air compressor.

Starting Air

The three Fairbanks Morse diesel generators share a starting air supply system which includes two redundant subsystems (Figures 8-8A, 8-8B, and 8-8C). There are two redundant air supply headers to which the two redundant air receivers per diesel generator are connected. Each diesel generator includes one electric motor-driven air compressor. The air compressor associated with Diesel Generator 1B can also be driven by a small diesel engine.

Warming System

Each diesel generator is also equipped with a standby warming system which automatically maintains the engine cooling water and lubricating oil temperature at satisfactory levels (Figures 8-8A, 8-8B, and 8-8C).

Fuel Oil

The fuel oil system for the three Fairbanks Morse EDGs consists of two (No. 11 and No. 21) above-ground fuel oil storage tanks (FOSTs), three fuel oil transfer pumps rated at 10 gpm each, and three fuel oil day tanks having a maximum capacity of approximately 485 gallons each. Each FOST is sized to hold approximately 107,000 gallons of usable fuel oil. The FOSTs are redundant, with the exception of the concrete enclosure around No. 21 FOST and the elevation of the internal standpipes. Redundant diesel supply headers interconnect the two independent tanks and manual valves are positioned such that normally each tank supplies a different header. A manual valve in each supply header ensures a failure of No. 11 FOST will not drain No. 21 FOST. The associated piping for the EDGs is designed for Seismic Category I accelerations. However, the transfer piping for the SBO diesel day tank (from No. 11 FOST) is only designated Seismic Category I from the tank to the first seismic anchor downstream of the tie-in isolation valve. The Seismic Category I FOSTs provide fuel oil for operation of the EDGs, auxiliary heating boiler, diesel-driven fire pump, and the SBO EDG. Internal standpipes provide fuel oil to the auxiliary heating boilers and the diesel-driven fire pump. The volume of oil below the standpipes is reserved exclusively for the Fairbanks Morse EDGs (in No. 11 FOST, the volume of fuel oil below the standpipe is also used by the SBO EDG). The standpipe in No. 21 FOST has a height of 11’, while the one in No. 11 FOST has a height of 7’6”.

There are two principal design criteria for the FOSTs: (1) design basis accident requirements, and (2) requirements for protection against external phenomena such as earthquakes and tornadoes. A tornado/missile event is not assumed to occur simultaneously with a design basis accident (LOCA).

The design of the EDG fuel oil system is based on a fuel oil capacity of seven days following a design basis accident. Specifically, IEEE-308 requires that, for multi-unit stations, sufficient fuel oil be available to run one EDG powering one unit under accident conditions (3,500 kW) and one EDG powering the opposite unit under normal shutdown conditions (3,000 kW) for seven days (or the time required to replenish fuel oil from an offsite source following a design basis event, whichever is longer). The specific emergency diesel generator fuel oil volumes (equivalent to duration based requirements) contained in the fuel oil storage tanks

referenced in Technical Specification 3.8.3 and the day tanks referenced in Technical Specification 3.8.1 are calculated using the method provided in American National Standards Institute N195-1976, Section 5.4, as endorsed by Regulatory Guide 1.137, Revision 1, Section C.I.c. The fuel oil calculation is based on applying the conservative assumption that the emergency diesel generator is operated continuously at rated capacity. This is one of two approved methods specified in Regulatory Guide 1.137, Revision 1. FOST No. 21 contains a volume of fuel oil in excess of that needed to satisfy this requirement. The minimum required volume under design basis accident conditions (LOCA) and tornado/missile conditions is in the Technical Specifications and is controlled administratively. The standpipe is not a requirement of IEEE-308.

Although protection against earthquakes was an original design criterion for the FOSTs, protection against tornadoes was not. In 1972, the decision was made to protect No. 21 FOST from tornadoes and horizontal tornado missiles by adding a Seismic Category I concrete enclosure. Bursting pressures are relieved by baffled, missile proof vents. This structure will also withstand the impact of a transmission tower falling on it without damage to the FOST. The enclosure also acts as a dike for No. 21 FOST with fuel being supplied by way of a non-safety-related line. In the event of a tornado, only No. 21 FOST is credited. It is assumed that one Fairbanks Morse diesel on each unit would be loaded to 3,000 kW. Thus, the minimum fuel oil requirement is somewhat less than that required under design basis accident conditions. The minimum Technical Specification volume maintained for these conditions in No. 21 FOST is the same as that required under design basis accident conditions.

The fuel oil volume in each day tank is normally maintained by automatic cycling of the fuel oil transfer pump. Operation of the transfer pump is automatic and is controlled by pump start and stop level switches connected to the day tank. High and low day tank level alarms are also provided to warn of abnormal conditions. Although not originally designed to American National Standards Institute N195-1976 (Fuel Oil Systems for Standby Diesel Generators), Calvert Cliffs has adopted the day tank minimum volume recommendations of this standard. The minimum volume required by the Technical Specifications permits at least one hour to correct minor problems in the fuel oil transfer system assuming an EDG load of 3,500 kW.

Additional design and quality control requirements for the storage tanks are given in Section 10.2.4.1.

An additive is used to inhibit deterioration of the fuel oil within the outdoor storage tanks. Samples are taken from the incoming fuel oil and analyzed for water and sediment, specific gravity and viscosity prior to adding the fuel to the storage tanks. Additional samples of the incoming fuel are analyzed for flash point, sulfur, ash, and Btu per gallon, as required, to insure the quality of the fuel. Samples are obtained from the FOSTs every three months and analyzed for oxidation/deterioration products.

<u>Instrumentation and Control</u>

Each diesel generator is equipped with various local and Control Room alarms, as indicated on Figure 8-6. Electrical instruments are provided in the Control Room and at the diesel generator for surveillance of generator voltage, frequency, power, and reactive volt-amperes.

Physical separation and electrical isolation are maintained between the redundant generator control circuits. The 125 Volt DC control power for Diesel Generator 1B is provided by Battery 12. Diesel Generator 1B is a part of and supplies power to

Load Group B. The 125 Volt DC control power for Diesel Generator 2A is provided by Battery 11. Diesel Generator 2A is a part of and supplies power to Load Group A. The 125 Volt DC control power for Diesel Generator 2B is provided by Battery 21. Diesel Generator 2B is a part of and supplies power to Load Group B.

Equipment is provided in the Control Room for each generator, for remote manual starting, remote stopping, remote synchronization, governor and voltage regulation, governor and voltage droop selection, and automatic or manual voltage regulator selection. Equipment is provided locally at each diesel generator for restricted manual starting in case of Control Room emergency, manual starting during routine diesel generator testing or maintenance, manual stopping, governor and voltage regulation, automatic or manual regulator selection, exciter field removal and reset, and remote and automatic or local manual control selection.

Protective Functions

Relay protection, ground connections, structural safeguards, and other protection systems are provided to assure adequate personnel protection and to prevent or limit rapid equipment deterioration during system short circuits or mechanical component failures. Special underfrequency protection is provided for safely separating the diesel generator from the preferred power supply (when previously synchronized to it) without damage to, or shutdown of, the diesel generator.

The following protective functions are provided for each Fairbanks Morse EDG:

a. Start failure relay
b. Engine overspeed
c. High jacket coolant temperature (2/3 logic)
d. Low jacket coolant pressure (2/3 logic)
e. Low lube oil pressure (2/3 logic)
f. High crankcase pressure (2/3 logic)
g. Loss of field
h. Generator differential
i. Generator ground overcurrent

Reverse power, loss of field, and underfrequency protection are provided but are made permissive to trip only upon diesel generator synchronization to the normal auxiliary power supply.

Time overcurrent with voltage restraint protective relays are provided on the diesel generator feeder breakers. These relays will trip the feeder breaker on overcurrent, but will not shut down the diesel generator.

The only protective devices that are retained during a SIAS or undervoltage (UV) are:

a. Overspeed
b. Lube-oil pressure low
c. Generator differential overcurrent
d. Generator ground overcurrent

These devices prevent a rapid destruction of the Fairbanks Morse diesel generators and are therefore the only shutdown functions permitted during SIAS and UV. The pressure shutdown function initiates upon coincidence of two-out-of-three logic to provide additional reliability. Each device, when actuated, initiates

an annunciator in the Control Room. Protection of the diesel generator unit from excessive overspeed, which can result from loss of load, is afforded by the operation of a diesel generator trip.

Service Water

The Service Water System removes the heat from the diesel generator heat exchangers. After operating at full load with a jacket temperature of 185°F, the diesel can continue to operate for one minute without service water cooling before the jacket temperature reaches 200-205°F and the diesel is automatically shut down. With an initial jacket temperature of 140-145°F, the diesel generator can operate three minutes before tripping.

8.4.2 SACM EMERGENCY DIESEL GENERATOR

8.4.2.1 Description

One of the four emergency power sources is the 4.16 kV, three-phase, 60-cycle tandem-engine Societe Alsacienne De Constructions Mecaniques De Mulhouse (SACM) diesel generator which has a nominal continuous rating of 5400 kW. The generator set for the SACM diesel, like those for the Fairbanks Morse diesel generators is physically separated and electrically isolated from the other generator sets.

The SACM diesel is rated as follows:

5400 kW	Continuous
5940 kW	2 hour
1620 kW	Minimum continuous load (Diesel may be operated at lower load for up to 7 days)

During loading of the safety-related SACM diesel generator, the frequency and voltage at the diesel generator terminals will not decrease to less than 95% and 82%, respectively. The diesel generator has the capability of starting the largest single motor with all other sequenced loads running. During recovery from transients caused by step loading or disconnection of full load, the safety-related SACM diesel generator's speed will not exceed 75% of the difference between nominal and the overspeed trip setpoint or 15% above nominal speed, whichever is lower. In accordance with the recommendations of Regulatory Guide 1.9, Revision 3 (Reference 1), frequency is restored to within 2% of nominal within 60% of each load sequence time interval, and voltage is restored to within 10% of nominal within 60% of each load sequence time interval.

The SACM diesel generator is a redundant, standby onsite power source installed in a separate and independent Category I building. The physical separation requirements of Regulatory Guide 1.75 and IEEE 384-1981 have been satisfied.

Additional details concerning the SACM diesel can be found in the EDG Project SACM Diesel Generator and Mechanical Systems Design Report (Reference 2).

8.4.2.2 Auxiliary Systems

Starting Air

The starting air system (Figure 8-8E) for the tandem SACM diesel generator consists of skid-mounted subsystems which include two pairs of redundant air receivers and two redundant air compressors complete with air dryers. This system supplies pressurized air to the engine mounted starting air distributors.

Only one air receiver per engine is required to be operational in order to successfully start the diesel engines within 10 seconds. The two air systems can be cross connected to recharge all four air receivers from one compressor. The diesel generator starting air system initiates an engine start such that the generator attains the rated frequency and voltage within 10 seconds of receipt of the start signal. Portions of the starting air system which are required to start the diesel generator are designed to remain functional during and after a safe shutdown earthquake. Active components of the diesel generator starting air system are capable of being tested in accordance with 10 CFR Part 50, Appendix A, General Design Criterion 18.

<u>Cooling Water Systems</u>

Each diesel engine of the tandem-driven generator is provided with independent high temperature (HT) and low temperature (LT) closed loop cooling system, which together make up the cooling water system for the SACM EDG. The HT system provides cooling flow to the engine block and turbochargers, while the LT system provides cooling flow to the combustion air coolers and the lube oil heat exchanger. Both systems consist of an engine-driven pump, expansion tank, and thermostatic control valve. Each engine has a radiator with separate HT and LT sections to dissipate heat to the outdoor ambient air.

<u>Warming System</u>

In order to reduce the thermal stress and wear while starting the diesel generator, the HT cooling system is provided with a preheating loop (keep warm system). When the diesel generator is in standby mode, the HT cooling water system motor-driven pump circulates coolant through an electric heater and supplies the warmed coolant to the engine water jacket and the lube oil standby heat exchanger. Startup of the diesel engine de-energizes the keep warm pump and the electric coolant heater.

<u>Fuel Oil</u>

The fuel oil system (Figure 8-8D) for the SACM, safety-related diesel generator consists of a FOST, a recirculation loop and pump, two redundant motor-driven fuel oil transfer pumps, two redundant transfer filters, a common day tank, a leakage tank, a common dirty fuel oil tank, a duplex filter, and an engine-driven fuel oil pump with an AC motor-driven backup fuel oil pump. Each fuel oil transfer pump is capable of meeting the diesel generator's fuel needs at 100% rated load. The common day tank may be partially drained during surveillance testing to verify proper transfer pump operation.

The FOST for the SACM diesel is located in an isolated sector within the Safety-related Diesel Generator Building. The enclosure is large enough to hold the contents of both the FOST and the fuel oil day tank should a rupture occur. The enclosure, as well as the Diesel Generator Building exterior walls, has a three-hour fire rating. Portions of the diesel generator fuel oil system which are required for operation of the diesel generator are designed to remain functional during and after a safe shutdown earthquake.

The design of the FOST allows for replenishment of fuel oil without interrupting operation of the diesel generator, and two tanker fill connections for the FOST (one outside and one inside the Diesel Generator Building) are provided. The diesel fuel oil system provides onsite storage and delivery of fuel oil for operation at 100% continuous rated load for seven days, assuming the loss of all offsite power sources, as required by American Nuclear Society 59.51-1989.

The specific emergency diesel generator fuel oil volumes (equivalent to duration based requirements) contained in the fuel oil storage tanks referenced in Technical Specification 3.8.3 and the day tanks referenced in Technical Specification 3.8.1 are calculated using the method provided in American National Standards Institute N195-1976, Section 5.4, as endorsed by Regulatory Guide 1.137, Revision 1, Section C.l.c. The fuel oil calculation is based on applying the conservative assumption that the emergency diesel generator is operated continuously at rated capacity. This is one of two approved methods specified in Regulatory Guide 1.137, Revision 1.

Instrument and Control

The SACM diesel generator is equipped with various engine-mounted instruments, control panels located both in the Control Room and the Diesel Generator Building Control Room, and two engine auxiliaries desks. The auxiliaries desks include gauges to indicate engine temperatures, pressures, rack position, and engine speed (in RPM).

The 125 Volt DC power for the Safety-related Diesel Generator Building and the SACM diesel generator auxiliaries is supplied by a dedicated 125 Volt DC power system. This system consists of a 125 Volt DC battery, battery charger, a safety-related distribution panel, a non-safety-related distribution panel, and associated 125 Volt DC instrumentation. This system provides a reliable source of continuous power for control and instrumentation in the Diesel Generator Building. In accordance with the recommendations of Safety Guide 6, the 125 Volt DC power system is independent and supports only the electrical load group associated with its diesel generator.

Protective Functions

The following protective functions will be provided for the SACM EDG under normal operating conditions. Protective functions which are designed to prevent rapid destruction of the SACM EDG, and are retained during SIAS and UV, are indicated below:

	PROTECTIVE FUNCTION	RETAINED/BYPASSED DURING SIAS & UV
a.	Engine Overspeed	Retained
b.	Lube Oil Pressure (Lo-Lo)	Retained
c.	Lube Oil Temperature (Hi-Hi)	Bypassed
d.	HT Coolant Temperature (Hi)	Bypassed
e.	HT Coolant Temperature (Hi-Hi)	Bypassed
f.	HT Coolant Pressure (Lo)	Bypassed
g.	LT Coolant Pressure (Lo)	Bypassed
h.	Generator Bearing Temperature (Hi-Hi)	Bypassed
i.	High Crankcase Pressure	Bypassed
j.	Cranking Time Exceeded	[a]
k.	Generator Ground Overcurrent	Retained
l.	Generator Differential Current	Retained
m.	Generator Overvoltage	Bypassed
n.	Generator Voltage Controlled Overcurrent	Bypassed
o.	Excitation Faults	Bypassed

[a] A "Cranking time exceeded" signal is provided to block a start signal in all modes when cranking time for the diesel engine is exceeded.

In addition, the following functions also protect the SACM EDG under normal operating conditions. However, during SIAS or undervoltage conditions, these

functions will shift the governor to hydraulic governor control, which can result in EDG operation at frequencies beyond the analyzed range.

a. Load sharing control failure
b. Linear variable differential transformer failure
c. Electronic Governor 24 VDC power supply failure

Reverse power, loss of field, and underfrequency protection are provided for protection of the diesel during parallel operation with the plant auxiliary power distribution system.

The diesel generator and the auxiliary equipment essential for operation in order to safely shut down the reactor or for accident mitigation following a design basis accident, are considered safety-related. Physical identification and methods used to readily distinguish between redundant Class 1E systems and non-Class 1E systems are consistent with those described in Section 8.5.

Loading of the diesel generator is accomplished automatically as described by Table 8-7 with the addition of the 480 Volt loads associated with the Diesel Generator Building auxiliaries.

8.4.3 STATION CONTROL BATTERIES

8.4.3.1 Design Basis

The batteries are designed to furnish a reliable and continuous supply of power to the 125 Volt DC and 120 Volt AC vital systems. They are also used for supplying Control Room emergency lighting and the plant computers.

8.4.3.2 Description and Operation

Description - The station control battery system for the plant consists of four operational and one reserve battery each nominally rated at 125 Volt DC, as shown in Figure 8-5 and Table 8-10. Each cell is of a sealed lead-acid type, assembled in shock-absorbing, clear plastic container, with covers bonded in place to form a leak-proof seal. The batteries are mounted on corrosion-resistant, earthquake-proof racks suitable for use during SSE accelerations. The batteries are located in a Seismic Category I structure and each battery is located in a separate room. Battery capacity calculations verify that each of the four 125 VDC Class 1E batteries have capacity to carry SBO loads for at least one hour. This is sufficient for SBO since the AAC power source (0C Diesel Generator) will be available within one hour to supply the battery chargers.

The battery rooms share a ventilation system consisting of one supply duct, one exhaust duct, one battery room supply fan, and one battery room exhaust fan. The two fans are associated with redundant load groups. Upon loss of either fan, sufficient ventilation is provided by the remaining fan to preclude the possibility of hydrogen accumulation within the battery rooms.

Operation - The associated battery chargers maintain a floating charge on each battery, and are capable of supplying an equalizing charge when necessary. During normal operation, the chargers supply the power required by all the 125 Volt DC loads. Upon loss of auxiliary AC power, the entire DC load is drawn from the batteries. After availability of AC power from the diesel generators, the battery chargers will be energized and resume normal operation.

The 125 Volt DC power for the Safety-related Diesel Generator Building and the SACM diesel generator auxiliaries is supplied by a dedicated 125 Volt DC power

system for each diesel/building. This system provides a reliable source of continuous power for control and instrumentation in each Diesel Generator Building. In accordance with the recommendations of Safety Guide 6, the 125 Volt DC power system is independent and supports only the electrical load group associated with its diesel generator. The 125 Volt DC power system consists of a 125 Volt DC battery, battery charger, a safety-related distribution panel, a non-safety-related distribution panel and associated 125 Volt DC instrumentation. The Electrical Engineering Design Report, Diesel Generator Project (Part A) (Reference 3), provides additional design data concerning the 125 Volt DC system for the safety-related Diesel Generator Building and the SACM diesel generator auxiliaries.

8.4.4 TURBINE-GENERATOR COASTDOWN (Unit 2 only)

The turbine-generator coastdown circuits are designed to utilize the kinetic energy of the turbine generator for maintaining primary coolant flow for 20 seconds following a reactor or turbine trip. These trips are associated with the unavailability of system grid power to the plant service transformer feeding the RCPs of the affected reactor unit (Figure 8-7). During the coastdown period, excitation control equipment will function automatically to maintain a constant ratio of volts per Hertz for the protection of transformers and motors. At the outset of coastdown, all plant auxiliaries, with the exception of RCPs, are separated automatically from the coastdown supply. At the conclusion of coastdown, signaled by either 80% generator voltage or 20 seconds elapsed time, the switchyard circuit breaker through which coastdown takes place is opened automatically and the generator field circuit breakers are tripped.

8.4.5 STATION BLACKOUT DIESEL GENERATOR

8.4.5.1 Description

The SBO diesel generator was added as part of the Calvert Cliffs Nuclear Power Plant response to 10 CFR 50.63, “Loss of All Alternating Current Power.” Guidelines set forth in Regulatory Guide 1.155 were used for quality assurance activities and specifications for new non-safety-related equipment used to address 10 CFR 50.63 and not already addressed by Appendix A and Appendix R of 10 CFR Part 50.

The SBO diesel generator is a 4.16 kV, three-phase, 60-cycle tandem-engine SACM diesel generator, similar to the safety-related SACM diesel generator, with a nominal continuous rating of 5400 kW. The SBO diesel generator, and its supporting systems and associated switchgear are housed in the SBO Diesel Generator Building.

The SBO diesel generator is electrically isolated from the ESF buses by two breakers (one Class 1E and one non-Class 1E) in series and a Class 1E disconnect switch, and no provisions exist for automatically connecting redundant safety features busses to the SBO diesel generator. The design of power connections from the SBO diesel generator allow for manual alignment to any one safety-related train in either unit via a Class 1E ESF bus. Manual switching capability is provided through Class 1E disconnect switches and Class 1E breakers.

Loading of the diesel generator is accomplished automatically as described by Table 8-7 with the addition of the 480 Volt loads associated with the SBO Diesel Generator Building auxiliaries.

At no time during the loading sequence will the frequency and voltage at the diesel generator terminals decrease to less than 95% of 60 Hz and 82% of 4.16 kV.

During recovery from transients caused by step load increases, including initial step loads, or resulting from the disconnection of full load, the speed of the diesel generator will not exceed 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is lower. These load accepting characteristics meet or exceed the design considerations recommended in Regulatory Guide 1.9, Revision 3 (Reference 1).

NUMARC 87-00 gives guidance pertaining to the design requirements for the SBO diesel generator for use as an alternate AC (AAC) power source. The following considerations have been included in the SBO diesel generator design. Calvert Cliffs' method of satisfying the design criteria follows each consideration.

a. The new alternate AC (AAC) system and its components need not be designed to meet Class 1E or safety-related requirements.

   The AAC diesel generator is classified as augmented quality. The AAC diesel generator and diesel generator 1A were purchased under one safety-related equipment specification. The non-SACM components (e.g., structures and piping) associated with the AAC diesel generator are not qualified or certified to industry and regulatory requirements applicable to a Class 1E diesel generator. Appropriate performance requirements are imposed on critical AAC diesel generator equipment in order to ensure the design basis requirements of the AAC diesel generator are maintained.

b. Unless otherwise provided in the NUMARC 87-00 criteria, the new AAC system need not be protected against the effects of:

   (a) failure or misoperation of mechanical equipment, including: (i) fire, (ii) pipe whip, (iii) jet impingement, (iv) water spray, (v) flooding from a pipe break, (vi) radiation, pressurization, elevated temperature or humidity caused by a pipe break, and (vii) missiles resulting from the failure of rotating equipment or high energy systems; or (b) seismic events.

   Hazards internal to the SBO Diesel Generator Building are not evaluated for their potential to cause failure during an SBO event. As required by the Standard Building Code (SBC), likely weather-related external events are considered in the design of the SBO power system and the structures that house AAC diesel generator equipment. The AAC power system and associated structures are not designed to remain functional during or following design basis seismic events.

c. Components and subsystems shall be protected against the effects of likely (i.e., not tornadoes or hurricanes) weather-related events that may initiate the loss of off-site power event.

   As a minimum, the SBO Diesel Generator Building is designed to withstand the weather-related loads addressed in the local SBC. Electrical cabling between the SBO Diesel Generator Building and the Auxiliary Building is housed in a buried concrete ductbank. In order to run AAC diesel generator cabling into the Auxiliary Building, a raceway is routed up the wall of the Auxiliary Building and across the roof of the Auxiliary Building and Turbine Building. As a minimum, this raceway is designed to meet the requirements of the SBC for weather related events.

d. Physical separation of the new AAC equipment from safety-related components or equipment shall conform with the separation criteria applicable for the unit’s licensing basis.

Some systems (e.g., lube oil drain system) are common to both the SBO Diesel Generator Building and the adjacent Category I Diesel Generator Building. Safety-related, Category I missile barriers are installed in the SBO Diesel Generator Building at wall penetrations (above grade) between the buildings for these common systems. Additionally, some systems in the power block serve equipment in the SBO Diesel Generator Building (e.g., fire protection and demineralized water systems). These systems are connected in such a way as to avoid any adverse effect on safety-related components and equipment. Refer to Section 8.5 for a discussion of electrical separation and group designations.

e. Failure of the new AAC diesel generator components shall not adversely affect Class 1E AC power systems.

   Mechanical systems common to the SBO Diesel Generator Building and the Category I Diesel Generator Building have been evaluated to ensure failure of the AAC diesel generator components do not adversely affect the operability of safety-related structures, systems, and components. In order to preclude failure and impact on the adjacent safety-related Diesel

   Generator Building, the main girders, columns, and bracing for the SBO Diesel Generator Building have been analyzed for safe shutdown earthquake loads. Miscellaneous AAC related equipment mounted outdoors or on the building roof, which could exceed the parameters for a Spectrum II tornado missile (as defined by Standard Review Plan 3.5.1.4, Revision 2) are anchored to resist tornado wind loads.

f. Electrical isolation of AAC power shall be provided through an appropriate isolation device.

   The AAC diesel generator is isolated from the Class 1E emergency busses by two circuit breakers in series. One breaker is Class 1E and the second breaker is non-Class 1E. Additionally, four disconnect switches (one for each emergency bus) provide isolation. The Class 1E circuit breakers serve as the interface between the AAC diesel generator and the 4.16 kV distribution system. The 480 VAC distribution system for the SBO diesel generator is connected to the Class 1E 480 VAC system in the Category I Diesel Generator Building. Devices which electrically isolate the AAC electrical systems from the Class 1E systems in the Category I Diesel Generator Building are designed in accordance with Regulatory Guide 1.75, Revision 2.

g. The AAC power source shall not normally be directly connected to the on-site emergency AC power system for the unit affected by the blackout. In addition the AAC system shall not be capable of automatic loading of shutdown equipment from the blacked-out unit unless licensed with such capability.

   The AAC diesel generator is connected to the onsite electrical distribution system through a Class 1E breaker, a non-Class-1E breaker, and a Class 1E disconnect switch, all of which are normally open. The AAC diesel generator is capable of powering a single safety-related train of equipment on one unit. Operator action is required to isolate the safety-related diesel generator dedicated to the emergency bus. The AAC diesel generator is then started manually, connected to the emergency bus, and automatically loaded using the load sequencer.

h. There shall be minimal potential for common cause failure of the new AAC diesel generator. The following system features provide assurance that the minimal potential for common cause failure has been adequately addressed.

1. The new AAC diesel generator power system shall be equipped with a DC power source that is electrically independent from the blacked-out unit's preferred and Class 1E power system.

   A separate battery system is provided for the AAC diesel generator. The battery is sized in accordance with IEEE 485-1983 to supply its design basis loads (e.g., diesel generator field flashing and instrumentation) for a four hour coping duration. During normal operation, the battery charger is energized from the 480 VAC motor control center located in the SBO Diesel Generator Building. The motor control center in the SBO Diesel Generator Building is normally energized by an offsite power source. In case maintenance is being performed, an alternate 480 VAC feed from the Class 1E 480 VAC unit substation bus (located in the Category I Diesel Generator Building) is manually aligned to energize the SBO Diesel Generator Building's 480 VAC distribution system. Loss of the normal or alternate source of 480 VAC will not interrupt power from the AAC diesel generator battery to the AAC diesel generator DC distribution system. Thus, loss of either 480 VAC feed will not create the possibility of common cause failure of the preferred Class 1E power system and the AAC power source.

2. The AAC power system shall be equipped with an air start system that is independent of the preferred and the blacked out unit's preferred and Class 1E power supply.

   The AAC diesel generator is equipped with its own starting air system consisting of four air receivers that supply pressurized air to start the AAC diesel generator. The four air receivers of the starting air system are supplied air from an air compressor located in the SBO Diesel Generator Building. This air compressor is energized from the SBO Diesel Generator Building's 480 VAC MCC and is not required to operate during the SBO event.

3. The AAC power supply shall be provided with a fuel oil supply that is separate from the fuel oil supply for the on-site emergency AC power system. A separate day tank supplied from a common storage tank is acceptable provided the fuel oil is sampled and analyzed consistent with applicable standards prior to transfer to the day tank.

   The AAC diesel generator is provided with two fuel oil day tanks, connected in series, that have a combined capacity sufficient to allow AAC diesel generator operation at 100% nominal load, without fuel transfer to the day tanks, for the design basis SBO coping period of four hours. Replenishment of the fuel oil day tanks is accomplished using the existing No. 11 FOST. The non-safety-related piping will be isolated from the safety-related FOST by a normally closed safety-related manual valve. Fuel oil in the No. 11 FOST is sampled prior to transferring fuel oil to the fuel oil day tanks.

   The oil in the SBO fuel oil day tanks is sampled and analyzed for the same characteristics and parameters and on the same frequency as is currently performed for EDG fuel oil.

4. If the AAC power source is an identical machine to the emergency on-site AC power source, active failures of the emergency AC power source shall be evaluated for applicability and corrective action taken to reduce subsequent failures.

   The AAC diesel generator is procured from the same manufacturer and is of the same basic design as Diesel Generator 1A. Any corrective actions identified for one SACM diesel generator are reviewed for incorporation on the other SACM diesel generator.

5. No single point vulnerability shall exist whereby a likely weather-related event or single active failure could disable any portion of the on-site emergency power sources or the preferred power sources, and simultaneously fail the new AAC power source(s).

   Protection for AAC diesel generator structures, systems, and components from likely weather-related events is provided by: (1) housing the AAC diesel generator in a structure designed to meet SBC requirements, and (2) routing cabling from the AAC diesel generator to the plant through ductbanks described above. The design of the AAC power source precludes a single active failure from disabling any portion of the existing power sources listed above and simultaneously fail the new AAC power source.

6. The AAC power system shall be capable of operating during and after a station blackout without any support systems powered from the preferred power supply, or the blacked-out unit's Class 1E power sources affected by the event.

   Mechanical systems and electrical distribution systems for the AAC diesel generator are self-supporting and do not require support from the blacked-out unit's buses or from existing on-site electrical power sources in order to cope with an SBO event. Alternate AC diesel generator auxiliary systems are provided with adequate amounts of starting air, fuel oil, lube oil, and cooling water in order to meet the requirements of the AAC diesel generator during the SBO event.

7. The portions of the AAC power system subjected to maintenance activities shall be tested prior to returning the AAC power system to service.

   The AAC power system is designed for periodic maintenance and testing through the Class 1E engineered safety features buses. Procedures have been established to ensure portions of the AAC power system subjected to maintenance activities shall be used prior to returning the AAC power system to service.

i. The AAC power system shall be sized to carry the required shutdown loads for four hours and be capable of maintaining voltage and frequency within the limits consistent with established industry standards that will not degrade the performance of any shutdown system or component.

   The AAC diesel generator is sized to accommodate the largest loading on any of the four emergency busses for at least a four hour duration, and is designed to maintain voltage and frequency within the limits specified by IEEE 387-1984.

j. The AAC power source shall be started and loaded at least every three months following the manufacturer's recommendations or using plant-developed procedures. A timed start and rated load capacity test shall be performed at least once each refueling outage.

   Procedures have been established that start and load the AAC diesel generator in a manner consistent with its functions as an alternate AC power source at intervals not longer than three months. In addition, a timed start and a rated load capacity test of the AAC diesel generator will be performed once every refueling outage interval for either Unit 1 or Unit 2.

k. Alternate AC system surveillance and maintenance activities shall be implemented considering the manufacturer's recommendations or in accordance with plant-developed procedures.

l. The AAC system shall be demonstrated by initial test to be capable of powering the required shutdown equipment within one hour of a station blackout event.

   Acceptance testing for the AAC diesel generator demonstrates the capability to power the required shutdown equipment within one hour.

m. The AAC system should be maintained at a target reliability of 0.95 per demand.

   The AAC diesel generator is maintained as a standby system. The diesel generator design features, periodic testing, and maintenance programs are designed to maintain system reliability at the target of 0.975 per demand.

Additional design data concerning the SBO diesel generator can be found in the Alternate AC Power Source Design Report (Reference 4).

8.4.5.2 Auxiliary Systems

Starting Air

The starting air system (Figure 8-8G) for the SBO diesel generator consists of four air receivers and an air compressor complete with air dryers. This system supplies pressurized air to the engine mounted starting air distributors.

Cooling Water Systems

Each diesel engine of the tandem-driven generator is provided with independent HT and LT closed loop cooling system, which together make up the cooling water system for the SACM diesel generator (Figure 8-8G). The HT system provides cooling flow to the engine block and turbochargers, while the LT system provides cooling flow to the combustion air coolers and the lube oil heat exchanger. Both systems consist of an engine-driven pump, expansion tank, and thermostatic control valve. Each engine has a radiator, with separate HT and LT sections to dissipate heat to the outdoor ambient air.

Warming System

In order to reduce the thermal stress and wear while starting the diesel generator, the HT cooling system is provided with a preheating loop (keep warm system). When the diesel generator is in standby mode, the HT cooling water system motor-driven pump circulates coolant through an electric heater and supplies the warmed coolant to the engine water jacket and the lube oil standby heat

exchanger. Startup of the diesel engine de-energizes the keep warm pump and the electric coolant heater.

Fuel Oil

The fuel oil system for the SBO diesel generator (Figure 8-8F) consists of a day tank, an auxiliary day tank, a fuel oil transfer filter, and a 100% fuel oil transfer pump. The day tank and auxiliary day tank are connected in series and have a combined capacity sufficient to allow operation, at rated load, for the four-hour coping duration, without transfer of fuel to the day tanks. Fuel Oil Storage Tank No. 11 supplies fuel oil to the day tanks via non-safety-related piping and a normally closed safety-related manual valve. Fuel oil transfer from No. 11 FOST to the day tanks is under operator control.

Instrument and Control

The SBO diesel generator is equipped with various engine-mounted instruments, control panels located both in the Control Room and the SBO Diesel Generator Building control room, and two engine auxiliaries desks. The auxiliaries desks include gauges to indicate engine temperatures, pressures, rack position, and engine speed (in RPM).

125 Volt DC power for the SBO Diesel Generator Building and SBO diesel generator auxiliaries is supplied by a dedicated 125 Volt DC power system. This system provides a reliable source of continuous power for control and instrumentation in the SBO Diesel Generator Building. The 125 Volt DC power system consists of a 125 Volt DC battery, battery charger, and associated 125 Volt DC instrumentation.

Protective Functions

The following protective functions are provided for the SBO diesel generator under normal operating conditions. Protective functions which are retained during emergency manual switch operation and which are designed to prevent rapid destruction of the diesel generator are presented below:

	PROTECTIVE FUNCTION	**RETAINED/BYPASSED DURING EMERGENCY OPERATION**
a.	Engine Overspeed	Retained
b.	Lube Oil Pressure (Lo-Lo)	Retained
c.	Lube Oil Temperature (Hi-Hi)	Bypassed
d.	HT Coolant Temperature (Hi)	Bypassed
e.	HT Coolant Temperature (Hi-Hi)	Bypassed
f.	HT Coolant Pressure (Lo)	Bypassed
g.	LT Coolant Pressure (Lo)	Bypassed
h.	Generator Bearing Temperature (Hi-Hi)	Bypassed
i.	High Crankcase Pressure	Bypassed
j.	Cranking Time Exceeded	(a)
k.	Generator Ground Overcurrent	Retained
l.	Generator Differential Current	Retained
m.	Generator Overvoltage	Bypassed
n.	Generator Voltage Controlled Overcurrent	Bypassed

	PROTECTIVE FUNCTION	RETAINED/BYPASSED DURING EMERGENCY OPERATION
o.	Excitation Faults	Bypassed

[a] A "cranking time exceeded" signal is provided to block a start signal in all modes when cranking time for the diesel engine is exceeded.

In addition, the following functions also protect the SBO diesel generator under normal operating conditions. However, during emergency manual switch operation, these functions will shift the governor to hydraulic governor control, which can result in EDG operation at frequencies beyond the analyzed range.

a. Load sharing control failure
b. Linear variable differential transformer failure
c. Electronic Governor 24 VDC power supply failure

Reverse power, loss of field, and underfrequency protection are provided for protection of the diesel during parallel operation with the plant auxiliary power distribution system.

Protective functions which are retained during an emergency manual switch operation and which are designed to prevent rapid destruction of the diesel generator are as follows:

Engine overspeed
Lube oil pressure, Lo Lo
Generator ground over current
Generator differential current

8.4.6 REFERENCES

1. Regulatory Guide 1.9, Revision 3, July 1993, Selection, Design, and Qualifications of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants
2. Letter from R. E. Denton (BGE) to Document Control Desk (NRC), dated July 20, 1993, Emergency Diesel Generator Project - SACM Diesel Generator and Mechanical Systems Design Report
3. Letter from R. E. Denton (BGE) to Document Control Desk (NRC), dated July 26, 1993, Emergency Diesel Generator Project - Electrical Engineering Design Report
4. Letter from R. E. Denton (BGE) to Document Control Desk (NRC), dated March 7, 1994, Alternate AC Power Source Design Report

TABLE 8-7

LOAD SEQUENCING

SEQUENCER STEP NO.	TIME (Seconds)	SERVICE	EQUIPMENT NUMBER FED BY EACH 4 kV BUS 1ZA BUS 11	1ZB BUS 14	2ZA BUS 21	2ZB BUS 24
-	0(1)(3)	Reactor Motor Control Centers	114	104	214	204
		Turbine Bearing Oil Pump *(2)	-	-	21	-
		1E Battery Chargers	11 & 14	12 & 13	22 & 23	21 & 24
		Transformer for 208/120 Volt Instrumentation Busses	11	12	21	22
		Penetration Room Exhaust Fan	11	12	21	22
		Diesel Generator Room Exhaust Fan	-	1B	2A	2B
		Control Room HVAC Fans	11	-	-	12
		Control Room Air Conditioning Condenser Fans *	11	-	-	12
		Saltwater System Air Compressor	11	12	21	22
		Motor-Operated Valves	various	various	various	various
		Emergency Core Cooling System Pump Room Air Coolers	11	12	21	22
		Emergency Core Cooling System Pump Room Exhaust Fans	11	12	21	22
		Boric Acid Storage Tank Heaters *	two	two	two	two
		Heat Tracing System *	11	12	21	22
		Diesel Building 1A and Auxiliaries	1A	-	-	-
		Switchgear Room HVAC Fans	11	12	21	22
		1E Battery Room Fans	one Exhaust fan and one redundant Supply fan			
		Service Water Pump	11	12	21	22
		Containment Vent Isolation	6900	6901	6900	6901
1	5	High Pressure Injection Pump(6)	11	13	21	23
		High Pressure Injection Pumps Motor-Operated Valves	various	various	various	various
2	10	Charging Pumps	11 & 13	12 & 13	21 & 23	22 & 23
		Boric Acid Pump	11	12	21	22
		Boric Acid Motor-Operated Valve	508	-	508	-
		Saltwater Pump	11	12	21	22

TABLE 8-7

LOAD SEQUENCING

SEQUENCER STEP NO.	TIME (Seconds)	SERVICE	EQUIPMENT NUMBER FED BY EACH 4 kV BUS 1ZA BUS 11	1ZB BUS 14	2ZA BUS 21	2ZB BUS 24
3	15	Containment Air Coolers	11 & 12	13 & 14	21 & 22	23 & 24
		Containment Spray Pump	11	12	21	22
4	20	Component Cooling Pump	11	12	21	22
		Containment Filter Units	11 & 13	12 & 13	21 & 23	22 & 23
5	25	Low Pressure Injection Pump	11	12	21	22
6	30	Control Room Air Conditioning Compressor(7)	11	-	-	12
		Switchgear Room Air Conditioning Compressor *	11	12	21	22
6A	45	Auxiliary Feed Water Pump	13	-	-	23
6B	40	Computer Room HVAC Unit *	-	11	12	-

NOTES:

(1) At time 0 seconds, the generator breaker is closed and the loads listed for the 0-second time step are energized independent of sequencer action.

(2) The loads identified with * are process controlled. The load feeder breaker will be closed at the time listed but the equipment will not run until called for by the process signal.

(3) There are additional minor loads energized at time 0 not shown in table.

(4) Low voltage equipment is indirectly fed by 4 kV Busses through step-down transformers and low voltage busses.

(6) HPSI Pumps 12 and 22 are normally in pull-to-lock and will not start.

(7) The Control Room air conditioning compressor is normally process controlled. However, during load sequencing, the compressor is forced to start within a certain amount of time and then run continuously until after the auxiliary feedwater pump has been sequenced and started. The Control Room air conditioning compressor control then automatically reverts back to process control.

8.5 SEPARATION CRITERIA

8.5.1 DESIGN BASIS

Channels that provide signals for the same plant protective function are independent and physically separated to assure that the minimum availability required during any design basis event is met.

8.5.2 CHANNEL IDENTIFICATION

Each circuit (scheme) and raceway is given a unique identification and each is additionally coded as follows:

a. The channel or load group designation,

b. Whether the circuit or raceway is associated with safety-related equipment,

c. Whether the circuit is associated with 125 Volt DC or 120 Volt vital AC panel feeders and non-safety-related equipment.

The facility code designations are classified according to their association and redundancy with respect to one another. This classification has resulted in the seven separation groups shown here with the associated facility codes. The facility codes are assigned on the basis of the accompanying descriptions:

SEPARATION GROUP 1

A - A non-safety-related scheme or raceway, Channel A.

DA - A 125 Volt DC or 120 Volt vital AC control feed to a non-safety-related item associated with Battery No. 11.

ZA - A safety-related instrumentation, control, or power scheme or raceway, Channel A.

ZD - One channel of a four-channel safety-related instrumentation channel or raceway, Channel D.

D - One channel of a four-channel non-safety-related instrumentation channel or raceway, Channel D.

SEPARATION GROUP 2

B - A non-safety-related scheme or raceway, Channel B.

DB - A 125 Volt DC or 120 Volt vital AC control feed to a non-safety-related item associated with Battery No. 21.

ZB - A safety-related instrumentation, control, or power scheme or raceway, Channel B.

ZE - One channel of a four-channel safety-related instrumentation channel or raceway, Channel E.

E - One channel of a four-channel non-safety-related instrumentation channel or raceway, Channel E.

SEPARATION GROUP 3

ZC - A safety-related control or power scheme or raceway associated with Battery No. 12 and shared items; e.g., all third-pump power circuits and Diesel Generator No. 1B.

DC - A 125 Volt DC or 120 Volt vital AC control feed to a non-safety-related item associated with Battery No. 12.

ZF - One channel of a four-channel safety-related instrumentation channel or raceway, Channel F.

F - One channel of a four-channel non-safety-related instrumentation channel or raceway, Channel F.

SEPARATION GROUP 4

ZH - A safety-related control scheme or raceway associated with Battery No. 22.

DH - A 125 Volt DC or 120 Volt vital AC control feed to a non-safety-related item associated with Battery No. 22.

ZG - One channel of a four-channel safety-related instrumentation channel or raceway, Channel G.

G - One channel of a four-channel non-safety-related instrumentation channel or raceway, Channel G.

SEPARATION GROUP 5

A - A non-safety-related scheme or raceway, Channel A.

B - A non-safety-related scheme or raceway, Channel B.

SEPARATION GROUP 6

ZJ - A safety-related control scheme or raceway associated with Battery No. 01, which is capable of assuming any of the first four separation groups, one at a time. Cables from another group may not be routed with Separation Group 6.

SEPARATION GROUP 7

K - An Augmented Quality-Station Blackout instrumentation, control, or power scheme or raceway related to Diesel Generator 0C or its dedicated battery (Battery No. 15).

A - A non-safety-related scheme or raceway, Channel A.

The facility code facilitates and ensures the maintenance of channel separation in the routing of cables and the connection of control boards and panels. All cables and raceways are physically labeled with the appropriate facility code for positive identification. Cable routing is checked and confirmed visually at the time of the cable pull.

8.5.3 CABLE ROUTING

The following principles apply for the routing of cables throughout the plant:

a. Cables with a facility code preceded by Z of a particular separation group are routed only in safety-related raceways of the same separation group.

b. Non-safety-related cables (A or B facility code) may be routed in safety-related raceways, but cannot be routed in safety-related trays of more than one safety separation group (i.e., Groups 1 or 2 respectively).

c. Non-safety-related cables (A or B facility code) of different non-safety facility codes can be routed together in non-safety-related raceways.

d. Control feeders (DA, DB, DC and DH facility code) from redundant 125 Volt DC unit control or 120 Volt vital AC control panels to non-safety-related equipment are routed separately to maintain independent battery and inverter emergency power sources.

e. Protective system instrumentation cables (ZD, ZE, ZF, and ZG facility code) are routed solely in "instrumentation only" raceways of the same separation group.

f. Control and instrumentation cables with K facility code are related to Diesel Generator 0C or its dedicated battery and are, therefore, routed separately.

g. The non-safety-related low voltage power, control, and instrumentation circuits related to the Station Blackout Diesel Generator Building are assigned a Facility Code A. These Facility Code A circuits will be routed in Separation Group 7 with the following exceptions:

   In the Auxiliary Building, these control and instrumentation circuits may be routed in either Separation Group 7 or Facility Code ZA or A raceway. However, once these circuits have been moved into Facility Code ZA or A to use existing raceway to enter equipment, they may not move back to Separation Group 7.

   In the Safety-related Diesel Generator Building, these circuits may be routed in the dedicated Facility Code A raceway.

h. The power cables from Diesel Generator 0C to the four safety-related emergency buses may be compatible with either safety-related Separation Group 1 or 2, but not both at the same time.

i. Cables are separated into four groups according to voltage classification and function as follows:

   1. Medium voltage power cables
   2. Low voltage load center power cables
   3. Low voltage power and control cables
   4. Instrumentation cables

8.5.4 CONTROL BOARDS AND OTHER PANELS

Within the control boards and other panels associated with Class 1E (in reference to electrical separation, post accident monitoring category #1 [PAM 1] circuits are included as Class 1E) systems, circuits and instruments are independent and physically separated by a distance of 6". Where physical separation is impracticable, conduit, metal barriers and fire retardant barriers are used to maintain independence.

Single-control devices to which redundant circuits are connected are avoided wherever practicable. Where single devices are unavoidable, electrical isolation is provided. Devices that provide electrical isolation include relays, isolation amplifiers, solid-state optical couplers, and resistors in instrumentation current loops across which isolated voltage signals are obtained. In the case of third-pump circuit-breaker control switches, the redundant circuits are in separate conduit and connect to the switch at separate locations. The connections are separated by an empty stage of the switch. Additional protection is obtained by the automatic disconnection of DC control power from the unused circuit. Therefore, both redundant circuits at the switch are not energized simultaneously.

With reference to the facility code designations, the separation criteria within control boards and other panels associated with Class 1E systems can be tabulated as follows to indicate compatibility of differently designated cables and devices:

SEPARATION GROUP 1 - ZA, ZD, DA, D, A
SEPARATION GROUP 2 - ZB, ZE, DB, E, B
SEPARATION GROUP 3 - ZC, ZF, DC, F
SEPARATION GROUP 4 - ZH, ZG, DH, G
SEPARATION GROUP 5 - A, B
SEPARATION GROUP 6 - ZJ
SEPARATION GROUP 7 - K, A

In the case of non-Class 1E A or B circuits in the control boards and other panels, the above criteria are modified to permit A and B association with all of the separation groups. Four-channel non-Class 1E circuits are permitted to associate with each other.

8.5.5 RACEWAYS

Separation and independence is maintained between cable trays of different separation groups throughout the plant, including the containment, the penetration rooms, cable spreading rooms, and other congested or hostile areas. The criteria for separating are as follows:

a. A minimum of 3' horizontal separation is maintained or physical fire barriers are installed between redundant cable trays. Where a barrier is required, it extends to a minimum of 1' above and below the cable tray or to the ceiling or floor, or it completely encloses each cable tray of one separation group.

b. Where the vertical stacking of redundant cable trays is unavoidable, a minimum spacing of 5' is maintained, or horizontal fire barriers are installed between trays, or each cable tray of one separation group is completely enclosed with a fire barrier.

c. In the case of the crossover of one redundant tray to another, a minimum of 9" vertical separation is maintained and fire barriers are installed on both top and bottom of one tray to extend 2' from the crossover. In the protected cable spreading room where arrangements preclude maintaining separation as outlined above, fire barriers are installed on the top and bottom of both redundant trays. These barriers, used in conjunction with flame-retardant cables, ensure that a fire in the cable trays, (in the cable spreading room) caused by a cable fault, will not render safety-related cables in a redundant tray inoperable.

d. The arrangement and/or installation of protective barriers precludes the possibility that a locally-generated force or missile will destroy redundant systems. For example, in rooms having heavy rotating machinery or high-energy piping:

   1. Redundant cable trays are maintained 20' apart, or
   2. One of the redundant trays must be 20' or more from the missile source or high-energy pipe, or
   3. A 6"-thick reinforced concrete wall isolates one tray from its redundant tray, or
   4. A steel barrier isolates the trays from the heavy rotating machinery or high-energy piping.

e. Within missile-endangered areas, a minimum horizontal separation of 20' is maintained, or a protective wall, ceiling, or floor of 6"-reinforced concrete provides isolation between redundant switchgear and between other redundant electrical equipment.

f. Where routing is unavoidable through areas with potential accumulation of large quantities of oil or other combustible fluids as a result of leakage or rupture of lube oil or cooling systems, a single separation group only is routed through this area and the cables are protected from dripping oil by conduit or covered tray.

g. Where it is necessary that cables of redundant systems approach the same or adjacent control panels with less than 3' separation, one system is installed in steel conduit or wireway.

h. Isolation between redundant circuits is considered to be adequate where physical separation is less than that indicated above, and when one of the circuits is routed in steel conduit or wireway.

i. The worst credible incident is a cable tray insulation fire in the cable spreading room caused by an electrical fault. There are no 208 Volt, 480 Volt or other high-voltage or high-fault cables in the trays of the cable spreading room. The highest

fault current in the trays is less than 1,000 Amp or less than an equivalent energy of 360,000 ampere^2 x seconds.

8.5.6 PENETRATION ROOMS

Two separate penetration rooms are provided for all cables that must pass through the containment wall. The East Penetration Room is divided such that there are, to one side of the division, the Separation Group 1 penetrations and to the other side of the division, the Separation Group 2 penetrations. The horizontal separation between redundant penetrations and associated cable trays is 3'. The West Penetration Room is similarly divided except for the addition of Separation Group 3 to the same side as Separation Group 2 and Separation Group 4 to the same side as Separation Group 1. Vertical and horizontal separation between redundant penetrations will be a minimum of 3'. Cable tray separation criteria, as described earlier, are applicable for the penetration rooms. Power cable penetrations of high-energy levels are located above those of low-energy level circuits.

FIGURE 8-1 ELECTRICAL MAIN SINGLE LINE DIAGRAM

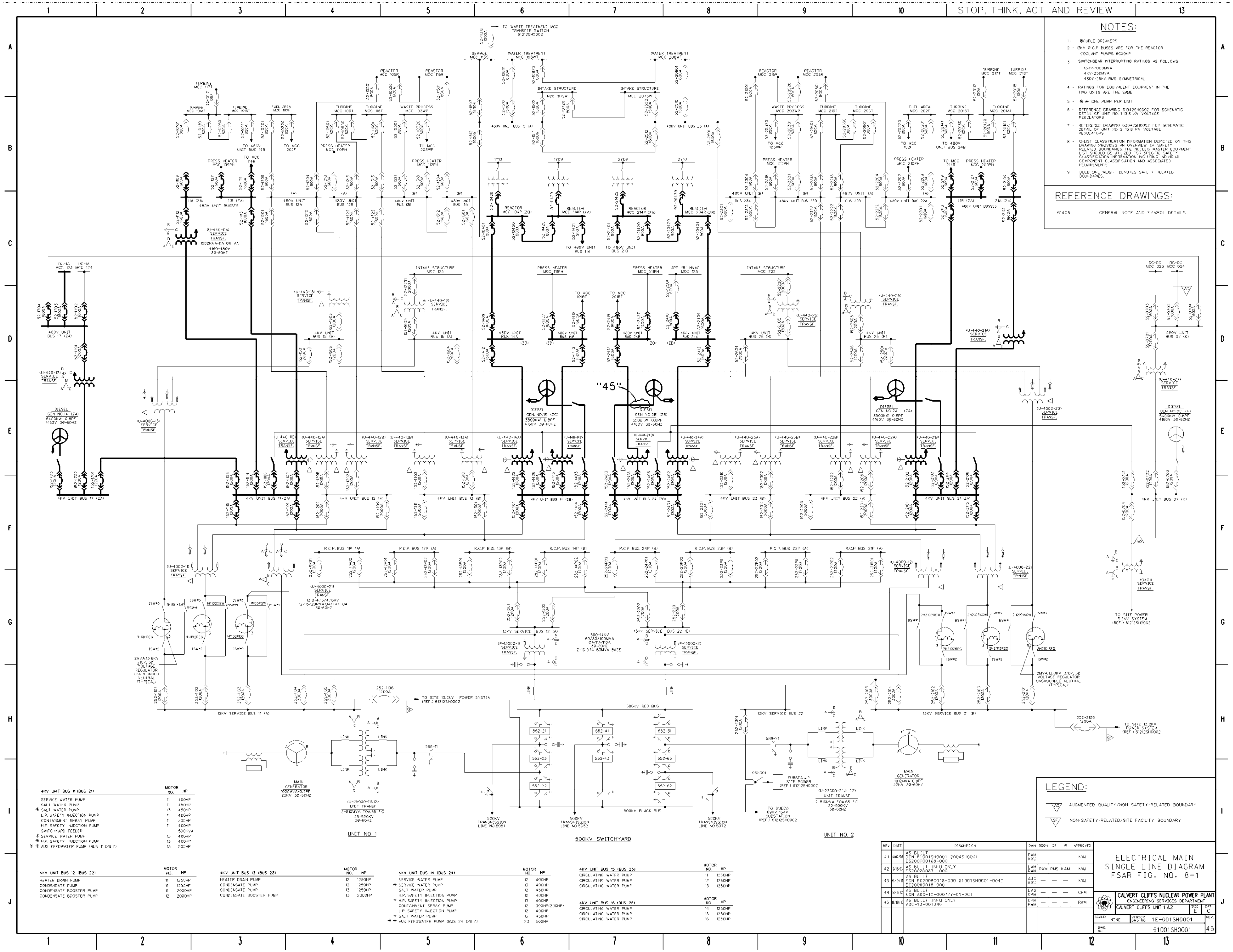


FIGURE 8-2 SINGLE LINE METER AND RELAY DIAGRAM 500 kV SWITCHYARD

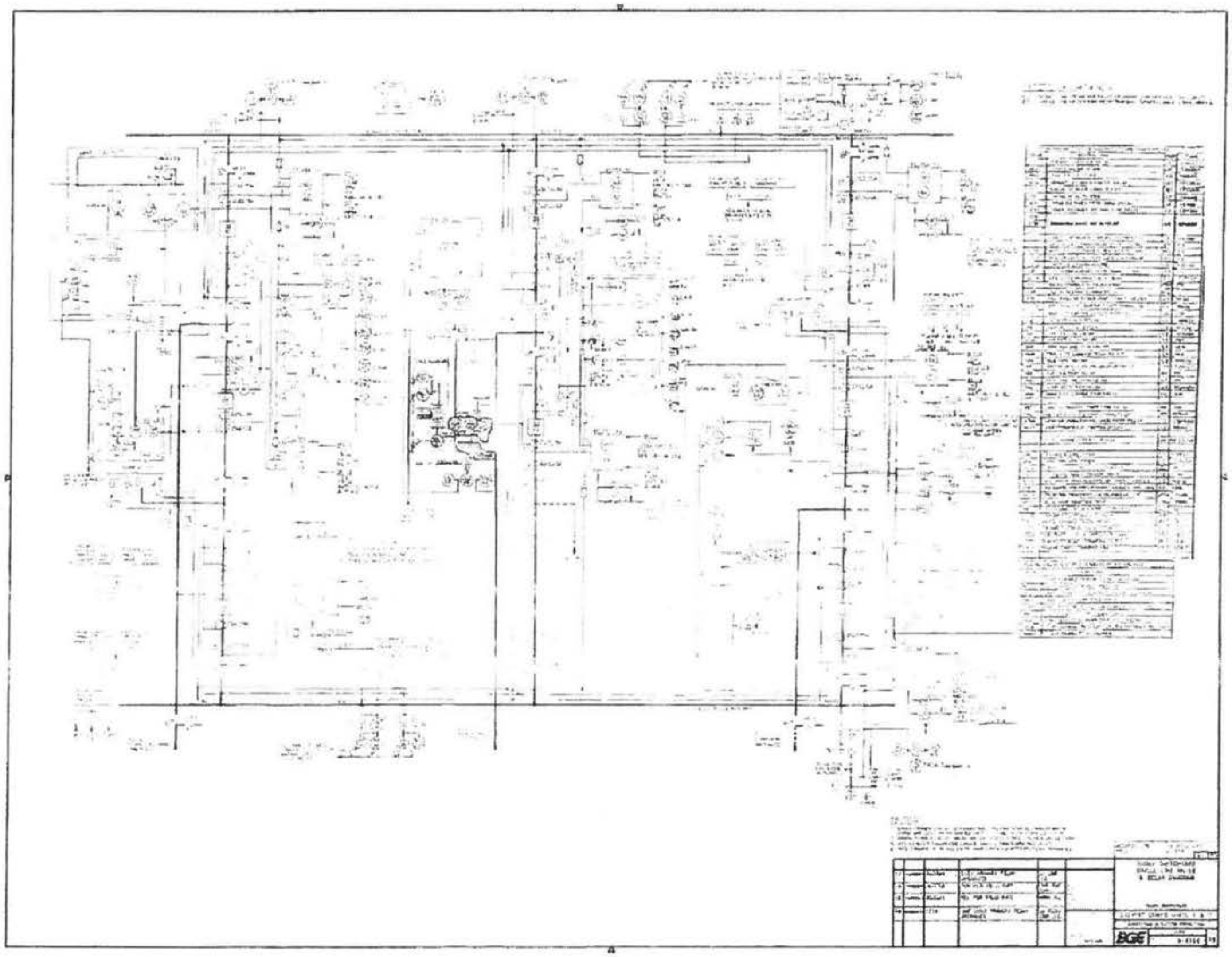
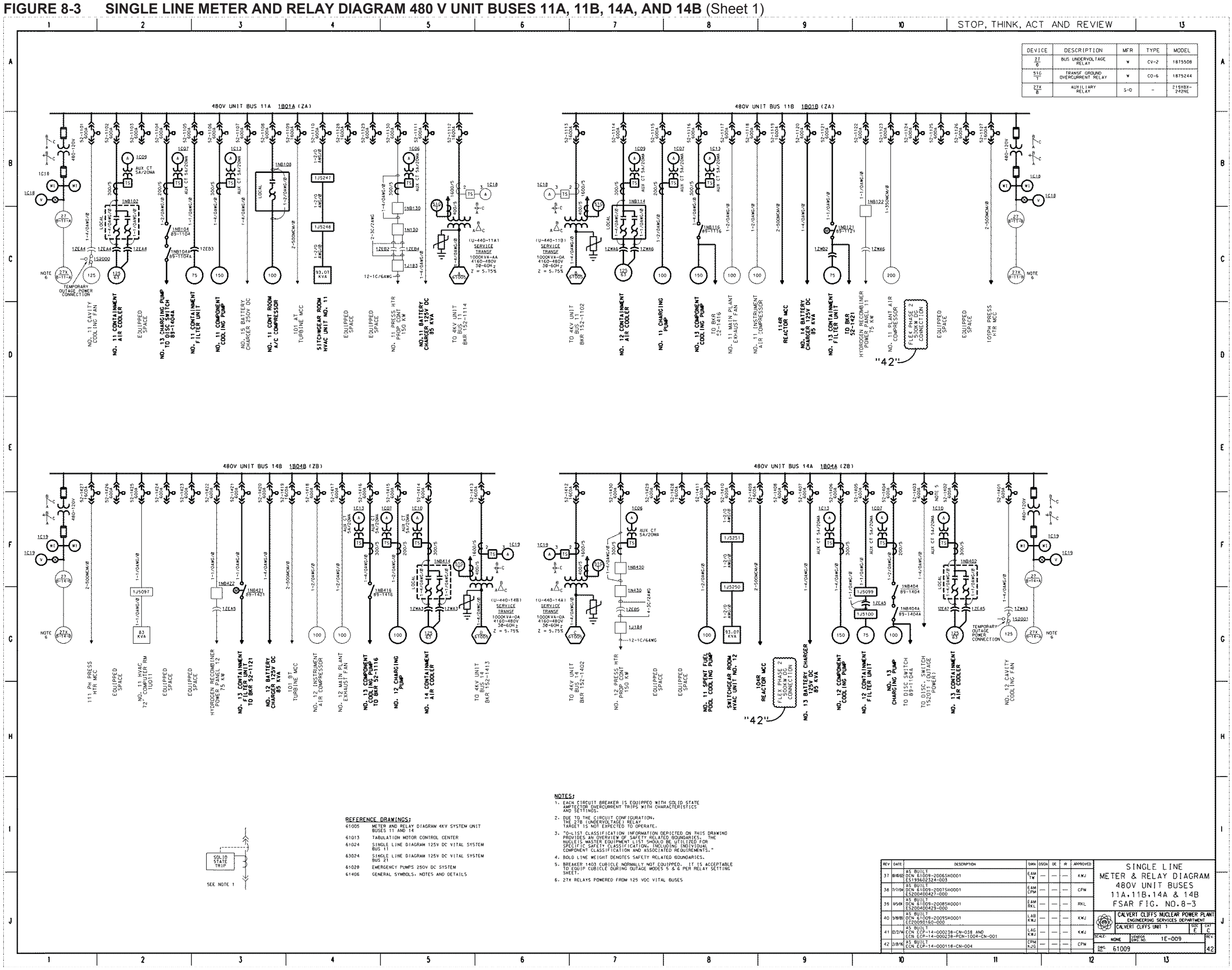
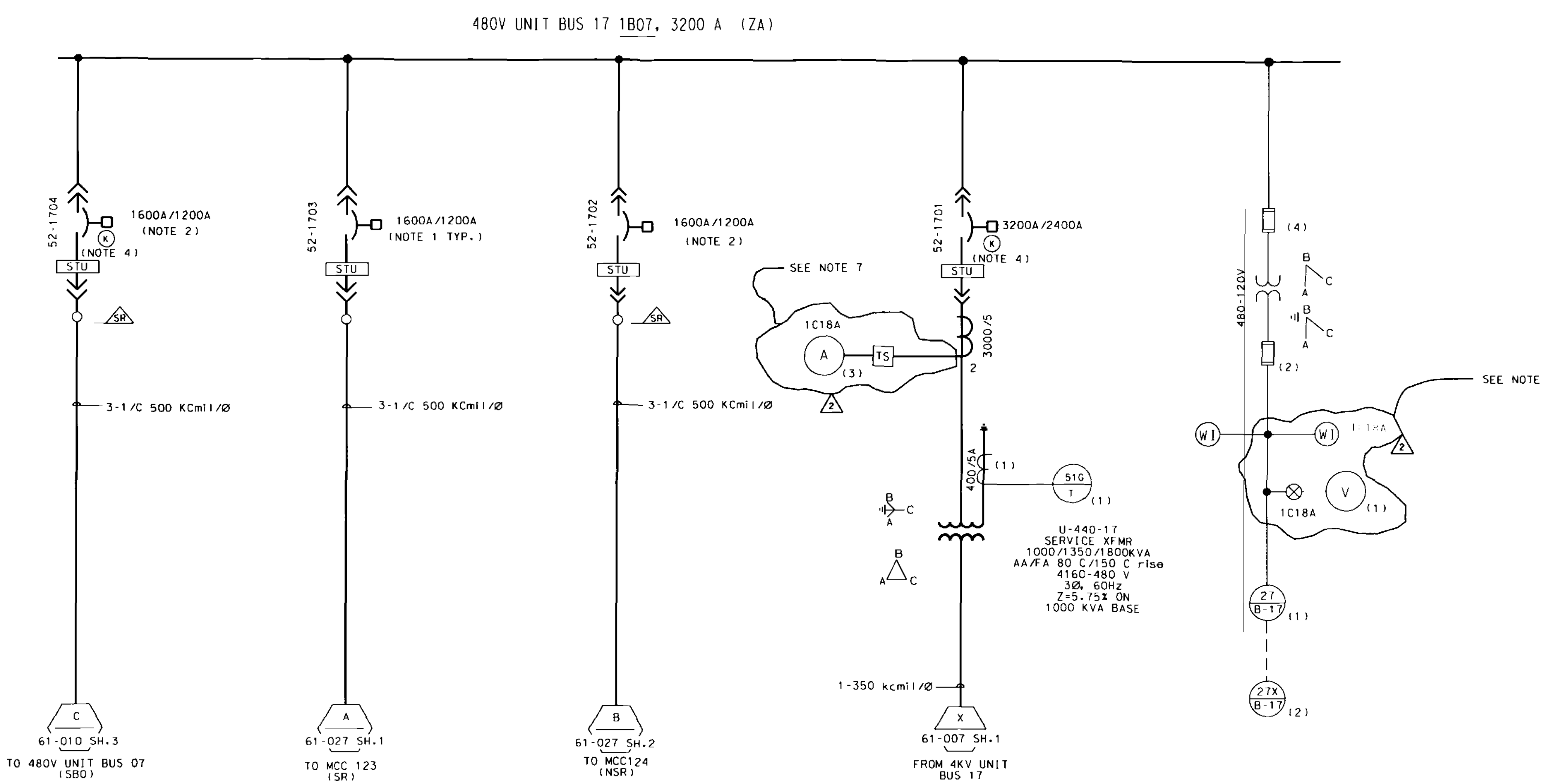


FIGURE 8-3 SINGLE LINE METER AND RELAY DIAGRAM 480 V UNIT BUSES 11A, 11B, 14A, AND 14B (Sheet 1)





DEVICE	DESCRIPTION	MFR	TYPE	MODEL	FUNCTION
27/B-17	BUS UNDERVOLTAGE RELAY	W	CV 2	1875508	OPERATE 27X/B-17
51G/T	XFMR GROUND OVER CURRENT RELAY	W	CO-6	264C898A05	TRIP BKR [52 1702]
STU	STATIC TRIP UNIT	*	AMPTECTOR I-A W/LSG UNIT	6998D02G04	TRIP BKR
27X/B-17	AUXILIARY RELAY	GE	HFA100	12HFA151A2H	ALARM

NOTES :

1. EACH BREAKER IS EQUIPPED WITH SOLID STATE OVERCURRENT TRIP UNIT (STU) WITH LONG AND SHORT TIME DELAY WITH INTEGRAL GROUND FAULT PROTECTION CHARACTERISTICS. BREAKER RATINGS ARE SHOWN AS: FRAME SIZE/SENSOR RATING.
2. BREAKER TRIPS ON ESFAS SIGNALS
3. NUMBERS IN PARENTHESIS ( ) DENOTE THE QUANTITY OF DEVICES REQUIRED.
4. MECHANICAL (KIRK KEY) INTERLOCKS ARE PROVIDED SUCH THAT EITHER NORMAL FEED BREAKER 52-1701 OR 52-0701 MUST BE OPENED BEFORE ALTERNATE FEED BREAKERS 52-1704 AND 52-0704 CAN BE CLOSED.
5. ALL EQUIPMENT ON THIS DRAWING IS SAFETY RELATED UNLESS NOTED OTHERWISE.
6. ALARM INPUTS ARE NON SAFETY RELATED
7. THE PORTION OF THE WORK IDENTIFIED BY THE BUBBLE IS ON HOLD. THE HOLD WILL BE REMOVED AND THIS PORTION OF THE DRAWING WILL BE RELEASED FOR TURNOVER TO THE PLANT AFTER THE 1996 UNIT OUTAGE BY A DCN IN FCR SUBMITTAL Y.

SOLID STATE TRIP

LEGEND:

- VOLTMETER SWITCH
- TS TEST SWITCH
- WI INDICATING LIGHT
- K KIRK KEY INTERLOCK
- SR INDICATES SAFETY RELATED/NON SAFETY RELATED BOUNDARY

REFERENCE DRAWINGS :

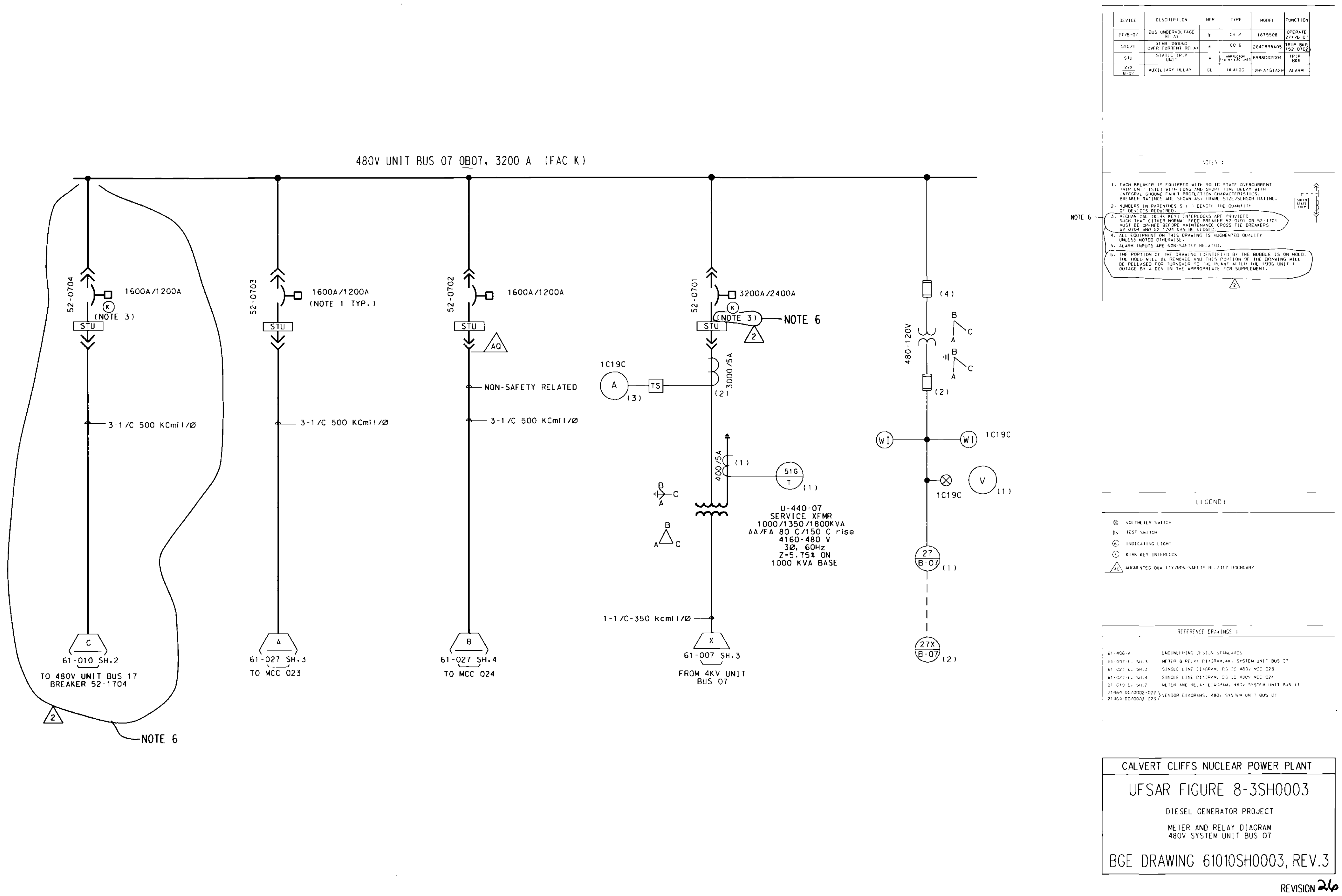
61-406-A	ENGINEERING DESIGN STANDARDS
61-007-E, SH.1	METER & RELAY DIAGRAM,4K. SYSTEM UNIT BUS 17
61-027-E, SH.1	SINGLE LINE DIAGRAM, DG 1A 480V MCC 123
61-027-E, SH.2	SINGLE LINE DIAGRAM, DG 1A 480V MCC 124
61-010-E, SH.3	METER AND RELAY DIAGRAM, 480V SYSTEM UNIT BUS 07
21464-DG70002-009	LAYOUT DIAGRAM, 480V SYSTEM UNIT BUS 17
21464-DG70002-005	SINGLE LINE DIAGRAM, 480V SYSTEM UNIT BUS 17

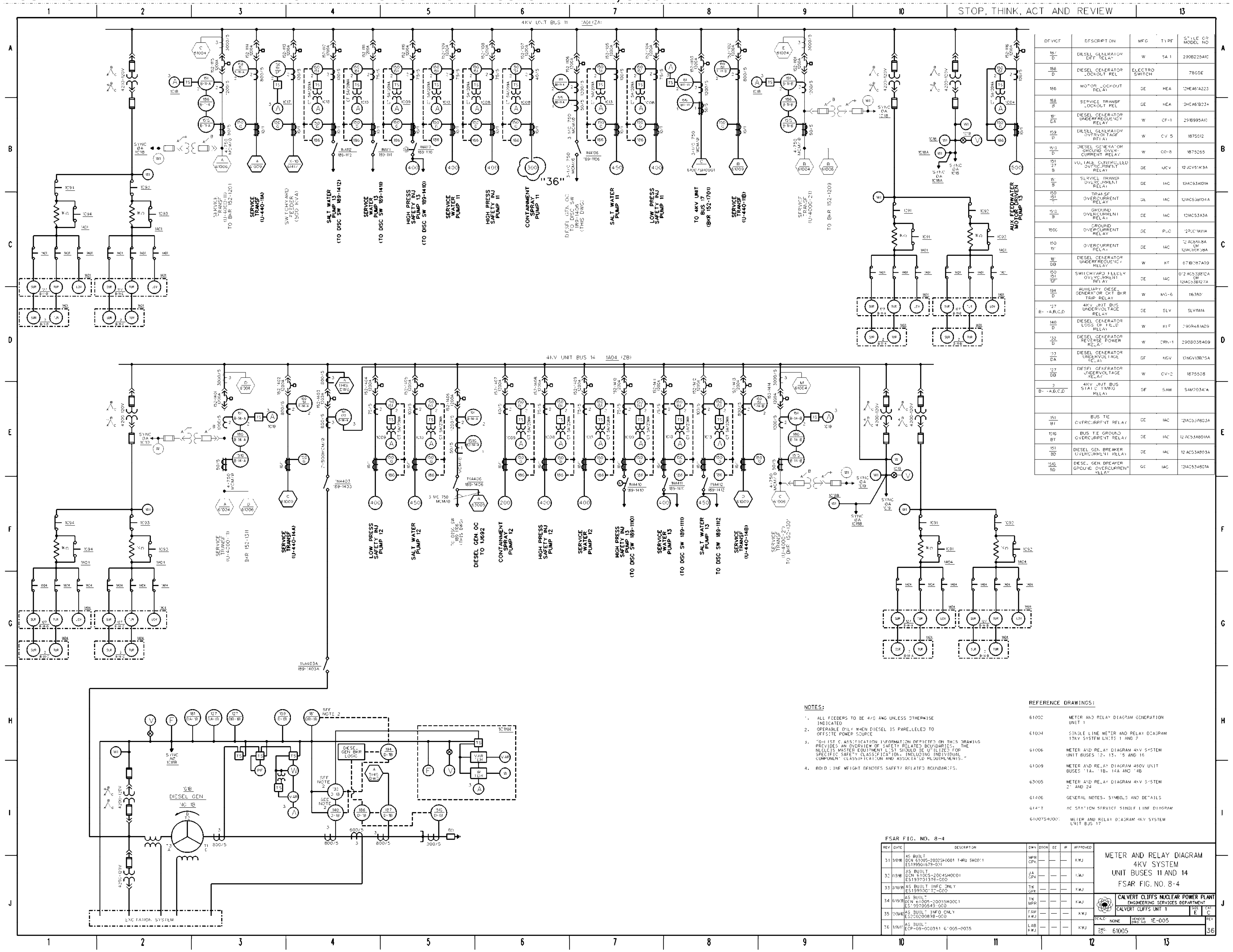
CALVERT CLIFFS NUCLEAR POWER PLANT

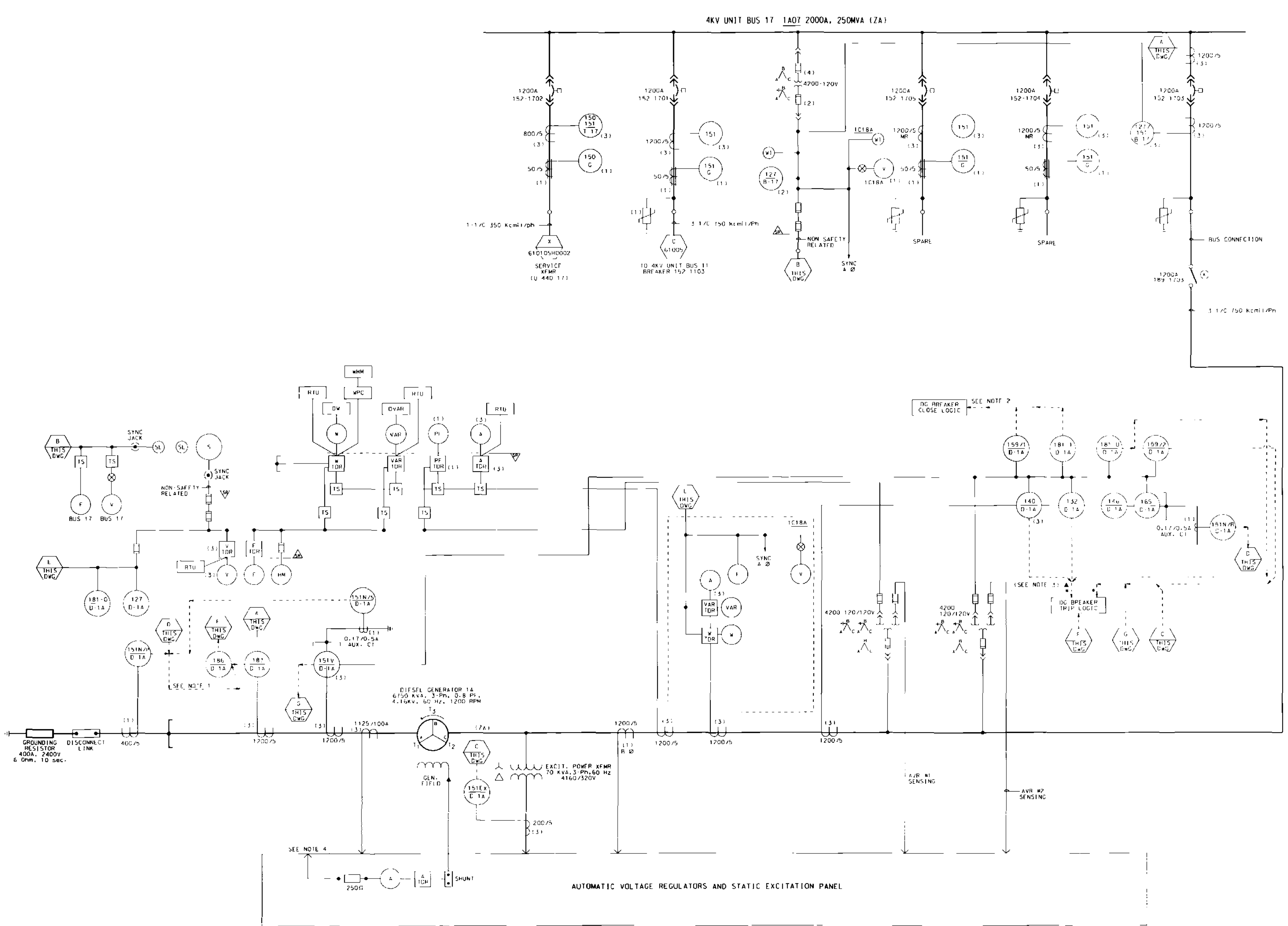
UFSAR FIGURE 8-3SH0002

DIESEL GENERATOR PROJECT
METER AND RELAY DIAGRAM
480V SYSTEM UNIT BUS 17

BGE DRAWING 61010SH0002, REV.3


480V UNIT BUS 07 0B07, 3200 A (FAC K)
52-0704
1600A/1200A
(NOTE 3)
STU
3-1/C 500 KCmil/Ø
61-010 SH.2
TO 480V UNIT BUS 17 BREAKER 52-1704
NOTE 6
52-0703
1600A/1200A
(NOTE 1 TYP.)
61-027 SH.3
TO MCC 023
52-0702
1600A/1200A
AQ
NON-SAFETY RELATED
61-027 SH.4
TO MCC 024
52-0701
3200A/2400A
NOTE 6
1C19C
3000/5A
400/5A
51G
U-440-07
SERVICE XFMR
1000/1350/1800KVA
AA/FA 80 C/150 C rise
4160-480 V
3Ø, 60Hz
Z=5.75% ON
1000 KVA BASE
1-1/C-350 kcmil/Ø
61-007 SH.3
FROM 4KV UNIT BUS 07
480-120V
27 B-07
27X B-07
NOTES:
LEGEND:
REFERENCE DRAWINGS:
CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 8-3SH0003
DIESEL GENERATOR PROJECT
METER AND RELAY DIAGRAM
480V SYSTEM UNIT BUS 07
BGE DRAWING 61010SH0003, REV.3
REVISION 26


STOP, THINK, ACT AND REVIEW
4KV UNIT BUS 11
4KV UNIT BUS 14
DIESEL GEN
DEVICE | DESCRIPTION | MFG | TYPE | STYLE OR MODEL NO
187/D | DIESEL GENERATOR DIFF RELAY | W | SA 1 | 290B225A1C
186/D | DIESEL GENERATOR LOCKOUT REL | ELECTRO SWITCH | | 7805E
186 | MOTOR LOCKOUT RELAY | GE | HEA | 12HEA61A223
186/B | SERVICE TRANSF LOCKOUT REL | GE | HEA | 12HEA61B234
181/DA | DIESEL GENERATOR UNDERFREQUENCY RELAY | W | CF-1 | 291B995A1C
159/D | DIESEL GENERATOR OVERVOLTAGE RELAY | W | CV 5 | 1875612
151G/D | DIESEL GENERATOR GROUND OVER-CURRENT RELAY | W | CO-8 | 1875265
151/127/B | VOLTAGE CONTROLLED OVERCURRENT RELAY | GE | IJCV | 12IJCV51A13A
151/B | SERVICE TRANSF OVERCURRENT RELAY | GE | IAC | 12IAC53A101A
150/151/T | TRANSF OVERCURRENT RELAY | GE | IAC | 12IAC53B104A
151G/B | GROUND OVERCURRENT RELAY | GE | IAC | 12IAC53A3A
150G | GROUND OVERCURRENT RELAY | GE | PJC | 12PJC11AV1A
150/151 | OVERCURRENT RELAY | GE | IAC | 12IAC66K8A OR 12IAC66K5BA
181/DB | DIESEL GENERATOR UNDERFREQUENCY RELAY | W | KF | 671B287A09
150/151/SF | SWITCHYARD FEEDER OVERCURRENT RELAY | GE | IAC | 012IAC53B812A OR 12IAC53B127A
194/D | AUXILIARY DIESEL GENERATOR CKT BKR TRIP RELAY | W | MG-6 | 1163801
127/B--A,B,C,D | 4KV UNIT BUS UNDERVOLTAGE RELAY | GE | SLV | SLV11A1A
140/D | DIESEL GENERATOR LOSS OF FIELD RELAY | W | KLF | 290B481A09
132/D | DIESEL GENERATOR REVERSE POWER RELAY | W | CRN-1 | 290B038A09
127/DA | DIESEL GENERATOR UNDERVOLTAGE RELAY | GE | NGV | 12NGV13B25A
127/DB | DIESEL GENERATOR UNDERVOLTAGE RELAY | W | CV-2 | 1875508
2/B--A,B,C,D | 4KV UNIT BUS STATIC TIMING RELAY | GE | SAM | SAM203A1A
151/BT | BUS TIE OVERCURRENT RELAY | GE | IAC | 12IAC53A803A
151G/BT | BUS TIE GROUND OVERCURRENT RELAY | GE | IAC | 12IAC53A801AA
151/BD | DIESEL GEN. BREAKER OVERCURRENT RELAY | GE | IAC | 12IAC53A803A
151G/BD | DIESEL GEN. BREAKER GROUND OVERCURRENT RELAY | GE | IAC | 12IAC53A801A
NOTES:
1. ALL FEEDERS TO BE 4/0 AWG UNLESS OTHERWISE INDICATED
2. OPERABLE ONLY WHEN DIESEL IS PARELLELED TO OFFSITE POWER SOURCE
3. "Q-LIST CLASSIFICATION INFORMATION DEPICTED ON THIS DRAWING PROVIDES AN OVERVIEW OF SAFETY RELATED BOUNDARIES. THE NUCLEIS MASTER EQUIPMENT LIST SHOULD BE UTILIZED FOR SPECIFIC SAFETY CLASSIFICATION, INCLUDING INDIVIDUAL COMPONENT CLASSIFICATION AND ASSOCIATED REQUIREMENTS."
4. BOLD LINE WEIGHT DENOTES SAFETY RELATED BOUNDARIES.
REFERENCE DRAWINGS:
61002 METER AND RELAY DIAGRAM GENERATION UNIT 1
61004 SINGLE LINE METER AND RELAY DIAGRAM 13KV SYSTEM UNITS 1 AND 2
61006 METER AND RELAY DIAGRAM 4KV SYSTEM UNIT BUSES 12, 13, 15 AND 16
61009 METER AND RELAY DIAGRAM 480V UNIT BUSES 11A, 11B, 14A AND 14B
63005 METER AND RELAY DIAGRAM 4KV SYSTEM 21 AND 24
61406 GENERAL NOTES, SYMBOLS AND DETAILS
61417 AC STATION SERVICE SINGLE LINE DIAGRAM
61007SH0001 METER AND RELAY DIAGRAM 4KV SYSTEM UNIT BUS 17
FSAR FIG. NO. 8-4
REV | DATE | DESCRIPTION | DWN | DSCN | DE | IR | APPROVED
31 | 5/12/98 | AS BUILT DCN 61005-2002SH0001 THRU SH0011 ES199501679-001 | MPR GPK | — | — | — | KWJ
32 | 1/3/98 | AS BUILT DCN 61005-2004SH0001 ES199701376-000 | JA GPK | — | — | — | KWJ
33 | 2/10/99 | AS BUILT INFO ONLY ES199900170-000 | TW GPK | — | — | — | KWJ
34 | 6/19/00 | AS BUILT DCN 61005-2003SH0001 ES199700549-000 | TW MPR | — | — | — | KWJ
35 | 12/30/02 | AS BUILT INFO ONLY ES200200898-000 | FAM KWJ | — | — | — | KWJ
36 | 1/26/11 | AS BUILT ECP-09-000391 61005-0035 | LAB KWJ | — | — | — | KWJ
METER AND RELAY DIAGRAM
4KV SYSTEM
UNIT BUSES 11 AND 14
FSAR FIG. NO. 8-4
CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 1
SIZE E
CAT. C
SCALE: NONE
VENDOR DWG NO 1E-005
DWG. NO. 61005
REV 36



DEVICE	DESCRIPTION	MFG TYPE	STYLE OR MODEL NO.	FUNCTION	LOCATION
187 D-1A	DIESEL GENERATOR DIFF RELAY	ABB/W SA1	290B225A10	OPERATE RELAY 186/D-1A	1C188
186 D-1A	DIESEL GENERATOR LOCKOUT RELAY (HAND RESET)	ELECT.SW 24LOR	7802D	EMERG. SHUT DOWN & DG BKR TRIP	1C188
181-U D-1A	DG UNDERFREQUENCY RELAY	ABB/W KF	671B287A15B	TRIP DG BKR 1703	1C188
181-O D-1A	DG OVERFREQUENCY RELAY	ABB CF1	291B995A09	ALARM	1C188
181-1 D-1A	DIESEL GENERATOR CORRECT SET FREQUENCY RELAY	ABB/W KF	671B287A15B	DG BKR 1703 CLOSE LOGIC	1C188
165 D-1A	DG GOVERNOR CONTROL	WOODWARD	2301A	LOAD SHARING AND SPEED CONTROL	1C188
159/1 D-1A	DIESEL GENERATOR CORRECT SET VOLTAGE RELAY	GE NGV	12NGV13B30A	DG BKR 1703 CLOSE LOGIC	1C188
159/2 D-1A	DIESEL GENERATOR OVERVOLTAGE RELAY	ABB/W CV-5	1975512	NORM SHUT DOWN & DG BKR 1703 TRIP	1C188
151	OVERCURRENT RELAY	GE IAC	121AC53A803A	TRIP ASSOCIATED BKR	1A07
151V D-1A	DG VOLTAGE CONTROLLED OVERCURRENT RELAY	ABB/W COV7	1878511	NORM SHUT DOWN & DG BKR 1703 TRIP	1C188
151N/P D-1A	DG GROUND OVERCURRENT RELAY	ABB/W CO8	288B717A13	OPERATE RELAY 186/D-1A	1C188
151N/B D-1A	DIESEL GENERATOR NEUTRAL OVERCURRENT RELAY	ABB/W CO8	288B717A13	OPERATE RELAY 186/D-1A	1C188
151N/S D-1A	DIESEL GENERATOR NEUTRAL OVERCURRENT RELAY	ABB/W CO8	288B717A13	OPERATE RELAY 186/D-1A	1C188
151EX D-1A	DG EXCITATION XFMR OVERCURRENT RELAY	GE PJC	PJC32J	TRIP DG-BKR 1703	1C188
151G	GROUND TIME OVERCURRENT RELAY	GE IAC	121AC53A801A	TRIP ASSOCIATED BKR	1A07
150/151 T-17	SERVICE TRANSFORMER OVERCURRENT RELAY	GE IAC	121AC53B811A	TRIP BKR 1702	1A07
150G	SERVICE TRANSFORMER GROUND INST. OVERCURRENT RELAY	GE PJC	12PJC11AV1A	TRIP BKR 1702	1A07
146 D-1A	DG NEGATIVE SEQUENCE RELAY	ABB/W COQ	3499A08A11	ALARM	1C188
140 D-1A	DG LOSS OF FIELD RELAY	ABB/W KLF	290B481A09	TRIP DG BKR 1703	1C188
132 D-1A	DG REVERSE POWER RELAY	ABB/W CRN-1	290B038A10 OR 290B038A09	TRIP DG BKR 1703	1C188
127/151 B-17	VOLTAGE RESTRAINT OVERCURRENT RELAY	GE IFCV	121FCV51AD1A	TRIP DG BKR 1703	1A07
127 D-1A	DIESEL GENERATOR UNDERVOLTAGE RELAY	ABB/W CV-2	1875508	ALARM	1C188
127 B-17	BUS 17 UNDERVOLTAGE RELAY	GE NGV	12NGV13B25A	ALARM	1A07

NOTES :

1. GENERATOR GROUND OVERCURRENT TWO OUT OF THREE LOGIC
2. VOLTAGE & FREQUENCY CORRECT SET RELAYS
3. ✱ INDICATES NON-EMERGENCY INPUT TO DG BREAKER TRIP LOGIC
   ▲ INDICATES TRIP WHEN DG IN PARALLEL MODE
4. PROVISION FOR BG&E LOCAL MEASUREMENT
5. ALL EQUIPMENT ON THIS DRAWING IS SAFETY RELATED UNLESS NOTED OTHERWISE.
6. ALARM AND RTU INPUTS ARE NON SAFETY RELATED.
7. DELETED
8. DELETED
9. POWER FACTOR METER DISCONNECTED, RETIRED IN PLACE AND LABELED "NOT USED".

LEGEND:

- VOLTMETER SWITCH
- DW DIGITAL WATT METER
- DVAR DIGITAL VAR METER
- TDR TRANSDUCER
- WPC WATT PULSES COUNTER
- HM HOUR METER
- RTU REMOTE TERMINAL UNIT
- TS TEST SWITCH
- SL SYNCHROSCOPE LAMP
- INDICATING LIGHT
- SURGE ARRESTOR
- S SYNCHROSCOPE
- K KIRK KEY INTERLOCK
- SR INDICATES SAFETY RELATED/NON SAFETY RELATED BOUNDARY

REFERENCE DRAWINGS :

[illegible]	METER & RELAY DIAGRAM, 4KV SYSTEM UNIT BUSES 11 & 14
[illegible]	METER & RELAY DIAGRAM, 4KV SYSTEM UNIT BUS 17
[illegible]	SCHEMATIC DIAGRAM, PLANT SYNCHRONIZING 13KV & 4KV UNIT [illegible]
61406	ENGINEERING DESIGN STANDARDS
21464 616-016	SINGLE LINE DIAGRAM
21464 616-229	ACCESSORY COMPONENTS DATA SHEETS
21464-DG70004-002	FRONT PANEL LAYOUT
21464 DG70004 006	SINGLE LINE DIAGRAM BUS 17

CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 8-4SH0002

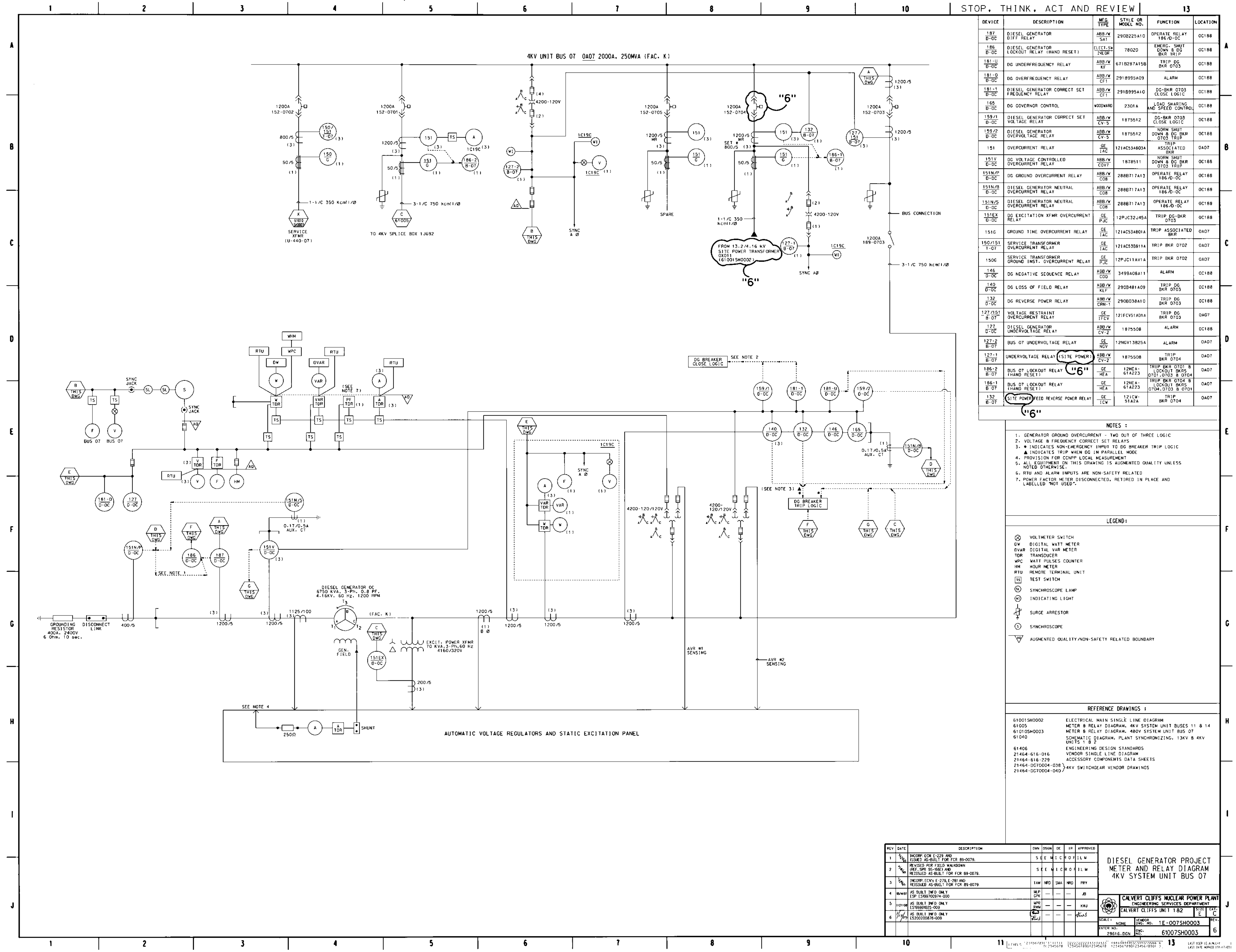
DIESEL GENERATOR PROJECT

METER AND RELAY DIAGRAM
4KV SYSTEM UNIT BUS 17

BGE DRAWING 61007SH0001, REV.6

REVISION 26

8-4 METER AND RELAY DIAGRAM 4 kV SYSTEM UNIT BUSSES 11 AND 14, Sheet 3



DEVICE	DESCRIPTION	MFG / TYPE	STYLE OR MODEL NO.	FUNCTION	LOCATION
187 / D-0C	DIESEL GENERATOR DIFF RELAY	ABB/W / SA1	290B225A10	OPERATE RELAY 186/D-0C	0C188
186 / D-0C	DIESEL GENERATOR LOCKOUT RELAY (HAND RESET)	ELECT.SW / 24LOR	7802D	EMERG. SHUT DOWN & DG BKR TRIP	0C188
181-U / D-0C	DG UNDERFREQUENCY RELAY	ABB/W / KF	671B287A15B	TRIP DG BKR 0703	0C188
181-O / D-0C	DG OVERFREQUENCY RELAY	ABB/W / CF1	291B995A09	ALARM	0C188
181-1 / D-0C	DIESEL GENERATOR CORRECT SET FREQUENCY RELAY	ABB/W / CF1	291B995A10	DG-BKR 0703 CLOSE LOGIC	0C188
165 / D-0C	DG GOVERNOR CONTROL	WOODWARD	2301A	LOAD SHARING AND SPEED CONTROL	0C188
159/1 / D-0C	DIESEL GENERATOR CORRECT SET VOLTAGE RELAY	ABB/W / CV-5	1875512	DG-BKR 0703 CLOSE LOGIC	0C188
159/2 / D-0C	DIESEL GENERATOR OVERVOLTAGE RELAY	ABB/W / CV-5	1875512	NORM SHUT DOWN & DG BKR 0703 TRIP	0C188
151	OVERCURRENT RELAY	GE / IAC	121AC53A803A	TRIP ASSOCIATED BKR	0A07
151V / D-0C	DG VOLTAGE CONTROLLED OVERCURRENT RELAY	ABB/W / COV7	1878511	NORM SHUT DOWN & DG BKR 0703 TRIP	0C188
151N/P / D-0C	DG GROUND OVERCURRENT RELAY	ABB/W / CO8	288B717A13	OPERATE RELAY 186/D-0C	0C188
151N/B / D-0C	DIESEL GENERATOR NEUTRAL OVERCURRENT RELAY	ABB/W / CO8	288B717A13	OPERATE RELAY 186/D-0C	0C188
151N/S / D-0C	DIESEL GENERATOR NEUTRAL OVERCURRENT RELAY	ABB/W / CO8	288B717A13	OPERATE RELAY 186/D-0C	0C188
151EX / D-0C	DG EXCITATION XFMR OVERCURRENT RELAY	GE / PJC	12PJC32J45A	TRIP DG-BKR 0703	0C188
151G	GROUND TIME OVERCURRENT RELAY	GE / IAC	121AC53A801A	TRIP ASSOCIATED BKR	0A07
150/151 / T-07	SERVICE TRANSFORMER OVERCURRENT RELAY	GE / IAC	121AC53B811A	TRIP BKR 0702	0A07
150G	SERVICE TRANSFORMER GROUND INST. OVERCURRENT RELAY	GE / PJC	12PJC11AV1A	TRIP BKR 0702	0A07
146 / D-0C	DG NEGATIVE SEQUENCE RELAY	ABB/W / COQ	3499A08A11	ALARM	0C188
140 / D-0C	DG LOSS OF FIELD RELAY	ABB/W / KLF	290B481A09	TRIP DG BKR 0703	0C188
132 / D-0C	DG REVERSE POWER RELAY	ABB/W / CRN-1	290B038A10	TRIP DG BKR 0703	0C188
127/151 / B-07	VOLTAGE RESTRAINT OVERCURRENT RELAY	GE / IFCV	12IFCV51AD1A	TRIP DG BKR 0703	0A07
127 / D-0C	DIESEL GENERATOR UNDERVOLTAGE RELAY	ABB/W / CV-2	1875508	ALARM	0C188
127-2 / B-07	BUS 07 UNDERVOLTAGE RELAY	GE / NGV	12NGV13B25A	ALARM	0A07
127-1 / B-07	UNDERVOLTAGE RELAY (SITE POWER) "6"	ABB/W / CV-2	1875508	TRIP BKR 0704	0A07
186-2 / B-07	BUS 07 LOCKOUT RELAY (HAND RESET)	GE / HEA	12HEA-61A223	TRIP BKR 0701 & LOCKOUT BKRS 0701,0703 & 0704	0A07
186-1 / B-07	BUS 07 LOCKOUT RELAY (HAND RESET)	GE / HEA	12HEA-61A223	TRIP BKR 0704 & LOCKOUT BKRS 0704,0703 & 0701	0A07
132 / B-07	SITE POWER FEED REVERSE POWER RELAY "6"	GE / ICW	12ICW-51A2A	TRIP BKR 0704	0A07

NOTES :

1. GENERATOR GROUND OVERCURRENT - TWO OUT OF THREE LOGIC
2. VOLTAGE & FREQUENCY CORRECT SET RELAYS
3. * INDICATES NON-EMERGENCY INPUT TO DG BREAKER TRIP LOGIC
   ▲ INDICATES TRIP WHEN DG IN PARALLEL MODE
4. PROVISION FOR CCNPP LOCAL MEASUREMENT
5. ALL EQUIPMENT ON THIS DRAWING IS AUGMENTED QUALITY UNLESS NOTED OTHERWISE.
6. RTU AND ALARM INPUTS ARE NON-SAFETY RELATED
7. POWER FACTOR METER DISCONNECTED, RETIRED IN PLACE AND LABELLED "NOT USED".

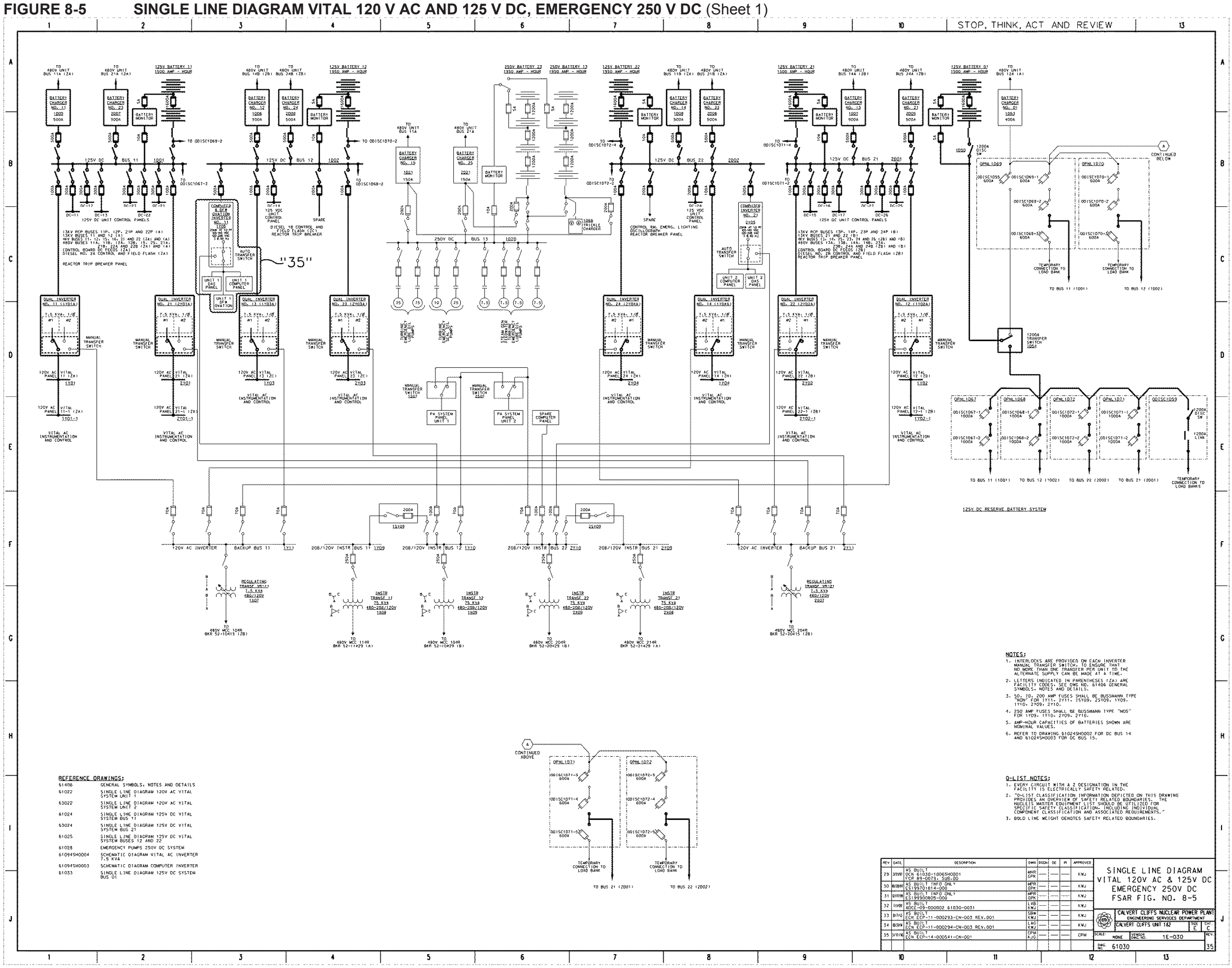
LEGEND:

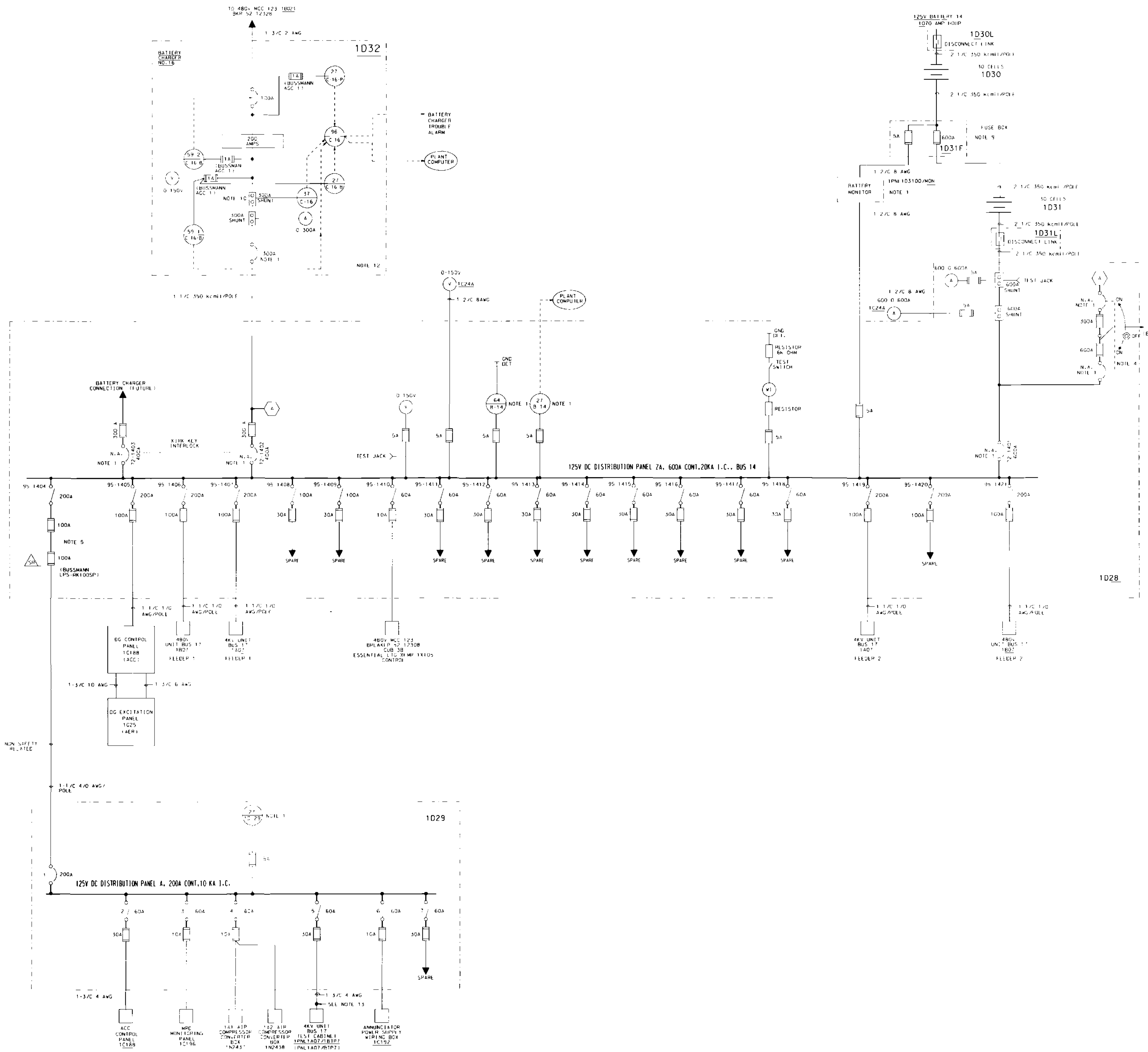
- ⊗ VOLTMETER SWITCH
- DW DIGITAL WATT METER
- DVAR DIGITAL VAR METER
- TDR TRANSDUCER
- WPC WATT PULSES COUNTER
- HM HOUR METER
- RTU REMOTE TERMINAL UNIT
- TS TEST SWITCH
- SL SYNCHROSCOPE LAMP
- WI INDICATING LIGHT
- SURGE ARRESTOR
- S SYNCHROSCOPE
- AQ AUGMENTED QUALITY/NON-SAFETY RELATED BOUNDARY

REFERENCE DRAWINGS :

61001SH0002	ELECTRICAL MAIN SINGLE LINE DIAGRAM
61005	METER & RELAY DIAGRAM, 4KV SYSTEM UNIT BUSES 11 & 14
61010SH0003	METER & RELAY DIAGRAM, 480V SYSTEM UNIT BUS 07
61040	SCHEMATIC DIAGRAM, PLANT SYNCHRONIZING, 13KV & 4KV UNITS 1 & 2
61406	ENGINEERING DESIGN STANDARDS
21464-616-016	VENDOR SINGLE LINE DIAGRAM
21464-616-229	ACCESSORY COMPONENTS DATA SHEETS
21464-DG70004-038, 21464-DG70004-040	4KV SWITCHGEAR VENDOR DRAWINGS

FIGURE 8-5 SINGLE LINE DIAGRAM VITAL 120 V AC AND 125 V DC, EMERGENCY 250 V DC (Sheet 1)





DEVICE TABLE

DEVICE	DESCRIPTION	FUNCTION	MFR & TYPE	MODEL
27/C-16-P	CHARGER AC UNDERVOLTAGE RELAY (VENDOR TAG ACF)	ACTIVATES 96/C-16	PCP	BY VENDOR
27/C-16-B	CHARGER DC UNDERVOLTAGE RELAY (VENDOR TAG DSL)	ACTIVATES 96/C-16	PCP	BY VENDOR
59-1/C-16-B	CHARGER DC OVERVOLTAGE RELAY (VENDOR TAG DSH)	ACTIVATES 96/C-16	PCP	BY VENDOR
27/B-14	BUS 14 UNDERVOLTAGE RELAY	ALARM/COMPUTER SIGNAL	GE,NGV	12NGV17C1
64/B-14	GROUND FAULT DETECTION RELAY	ALARM	EXIDE,FJ	EJ1215
27/1D-29	CONTROL PANEL 1D29 UNDERVOLTAGE RELAY	ALARM	ABB,27S	439U6345
96/C-16	CHARGER TROUBLE (SUMMATION) RELAY (VENDOR TAG K5)	ALARM/COMPUTER SIGNAL	MAGNECRAFT	219XBX237NE
59-2/C-16-B	HIGH VOLTAGE SHUT DOWN RELAY (VENDOR TAG HVS)	TRIPS CHARGER AC BREAKER	PCP	BY VENDOR
37/C-16	LOW DC CURRENT (VENDOR TAG CFA)	ACTIVATES 96/C-16	PCP	BY VENDOR

NOTES :

1. ALARM CONTACT FOR ABNORMAL CONDITION. ALL ALARM CONTACTS ARE USED FOR NON 1E ANNUNCIATIONS.
2. FUSE TYPES SHALL BE: 600A BATTERY MIDSPAN GOULD SHAWMUT A4J
   300A CHARGER OUTPUT GOULD SHAWMUT A4J
   300A & 600A DUMMY LOAD GOULD SHAWMUT A4J
   10A THRU 100A BRANCH CIRCUITS GOULD SHAWMUT OT
   5A GOULD SHAWMUT ATM
   (EXCEPT AS NOTED AND IN PANEL 1D29 WHERE FUSES ARE BUSSMANN NON)
3. MINIMUM FEEDER CABLE SIZE IS 1-2/C 8 AWG UNLESS OTHERWISE NOTED.
4. TWO 2-POLE, 600A MOLDED CASE SWITCHES WHICH ARE MECHANICALLY INTERLOCKED TO PREVENT BOTH SWITCHES FROM BEING CLOSED AT THE SAME TIME.
5. DIVERSE CLASS 1E FUSES (ISOLATION DEVICE).
6. DELETED
7. ALL SWITCHES ARE TWO POLE, SINGLE THROW UNLESS OTHERWISE NOTED.
8. ALL EQUIPMENT IS SAFETY RELATED UNLESS NOTED OTHERWISE.
9. FUSE BOX HAS TWO 600A FUSES AND TWO 5A FUSES. ONE OF EACH WILL BE UTILIZED AND THE REST WILL BE SPARED.
10. PROVISION FOR REMOTE AMMETER.
11. REV. A OF THIS DRAWING WAS ISSUED UNDER DRAWING NO.61-074-F,SH.1.
12. ALL METERS AND RELAYS IN THE BATTERY CHARGER ARE NON-1E EXCEPT 59-2/C-16-B.
13. DOWNSIZE CONDUCTORS OF FEEDER CABLE TO PANEL 1PNL1A07/1BTP7 TO #8 AWG IN ACCORDANCE WITH 61-406 SECT. XI.3 OR 226.
14. DELETED
15. DELETED

LEGEND :

A AMMETER
V VOLTMETER
N.A. NON AUTOMATIC MOLDED CASE SWITCH
SR SAFETY RELATED/NON-SAFETY RELATED BOUNDARY
WI WHITE INDICATING LIGHT NEON TYPE
● DENOTES A SPLICE

REFERENCE DRAWINGS :

61406	ENGINEERING DESIGN STANDARDS
61007SH0001	DIESEL GENERATOR PROJECT, SINGLE LINE & RELAY DIAGRAM 4KV SYSTEM UNIT BUS NO. 17
61010SH0002	DIESEL GENERATOR PROJECT SINGLE LINE & RELAY DIAGRAM 480V SYSTEM UNIT BUS NO. 17
61027SH0002	DIESEL GENERATOR PROJECT SINGLE LINE DIAGRAM DG AUXILIARY MCC 123 SR
21464 DG70008 004	BATTERY CHARGER CABINET OUTLINE
21464 DG70012 064	CLASS 1E DC PANEL 1D28 OUTLINE, SH 1 OF 2
21464 DG70012 065	CLASS 1E DC PANEL 1D28, OUTLINE, SH 2 OF 2
21464 DG70021 001	DC PANEL 1D29 OUTLINE, VENDOR DRAWING

CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 8-5SH0002

DIESEL GENERATOR PROJECT
SINGLE LINE DIAGRAM
DG1A 125V DC SYSTEM
BUS 14

BGE DRAWING 61024SH0002, REV.3

REVISION 26

FIGURE 8-5 DIESEL GENERATOR PROJECT SINGLE LINE DIAGRAM DG0C 124V DC SYSTEM BUS 15 (Sheet 3)

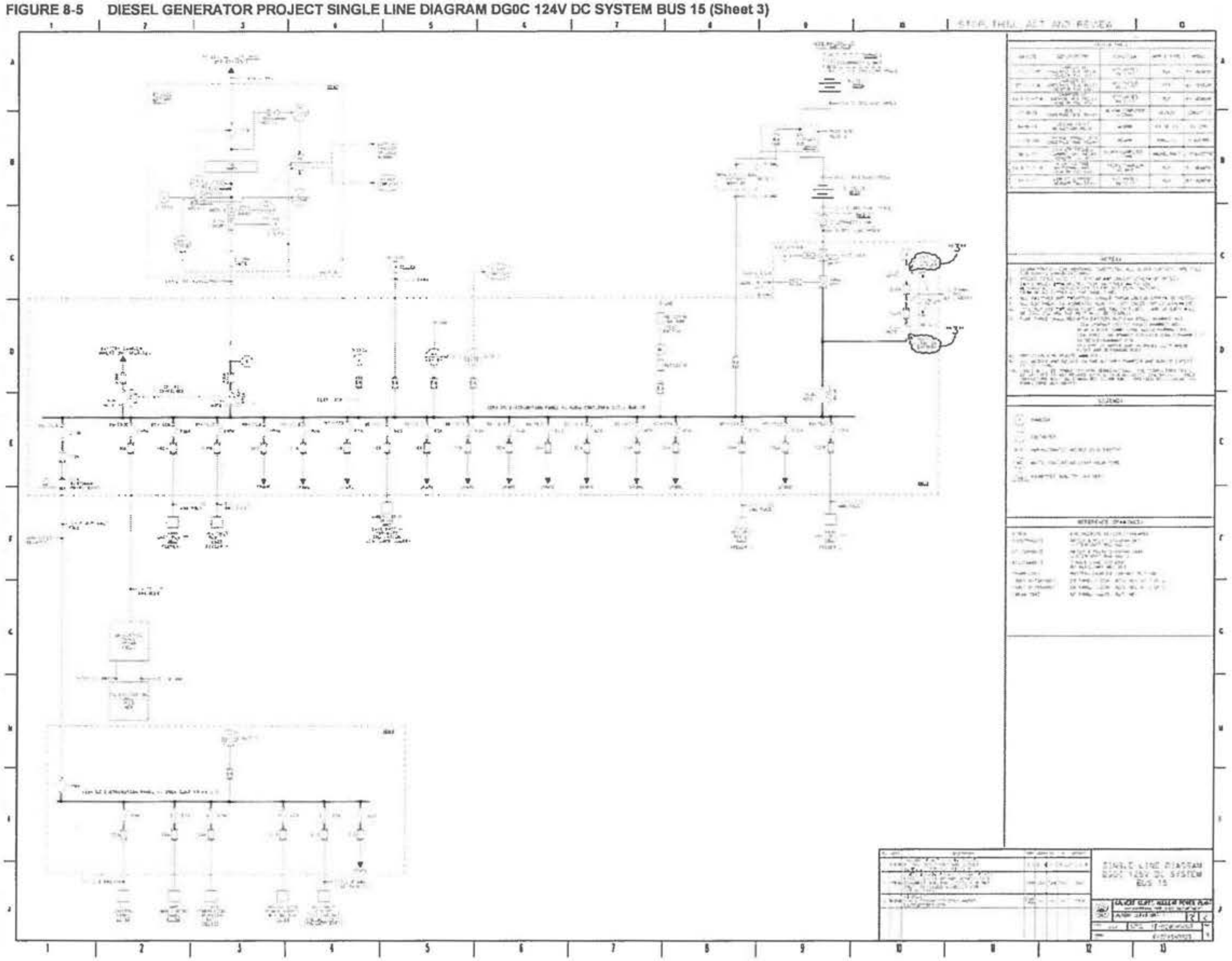


FIGURE 8-6 LOGIC DIAGRAM, DIESEL GENERATORS

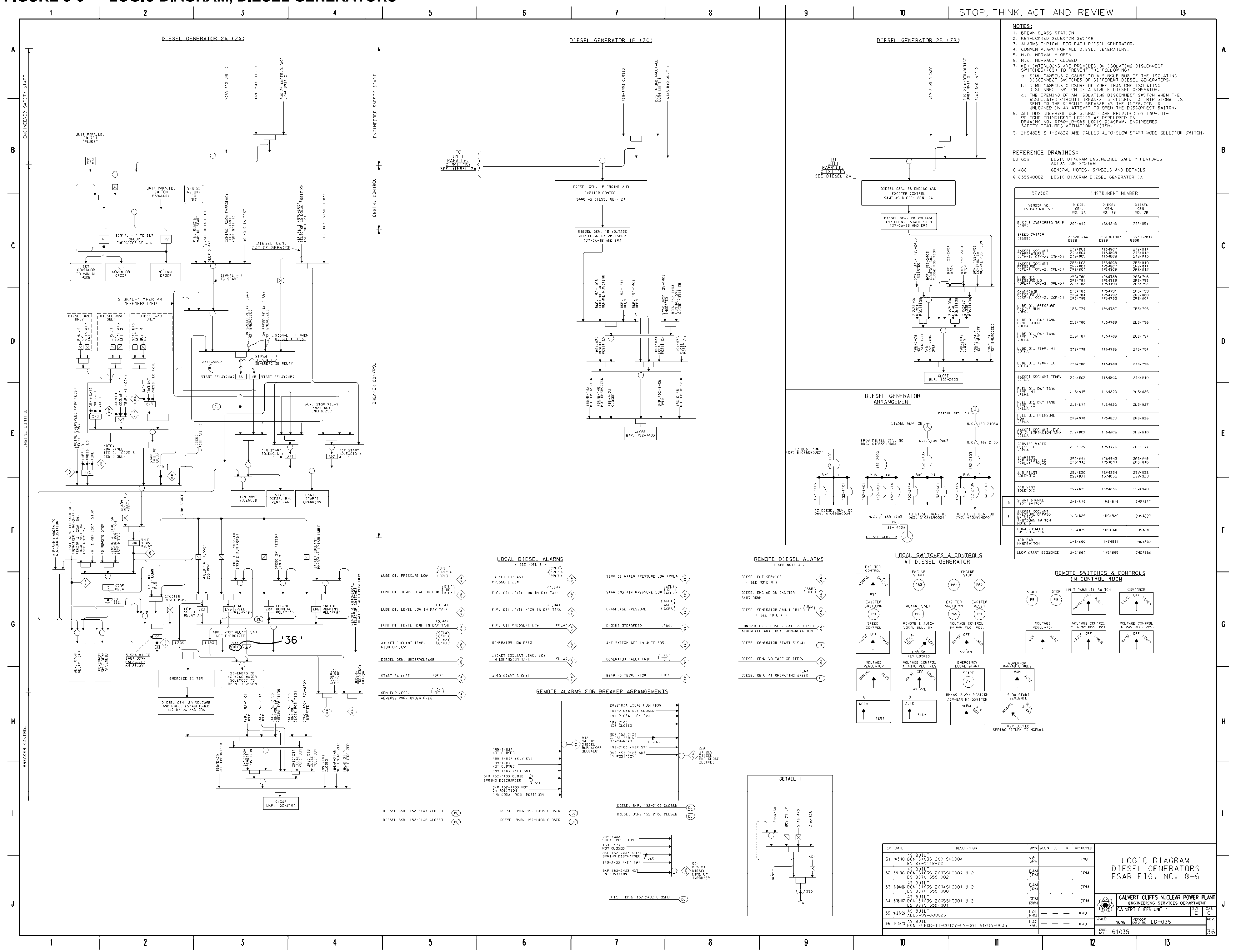


FIGURE 8-7 PLANT PROTECTION BLOCK DIAGRAM (Sheet 1)

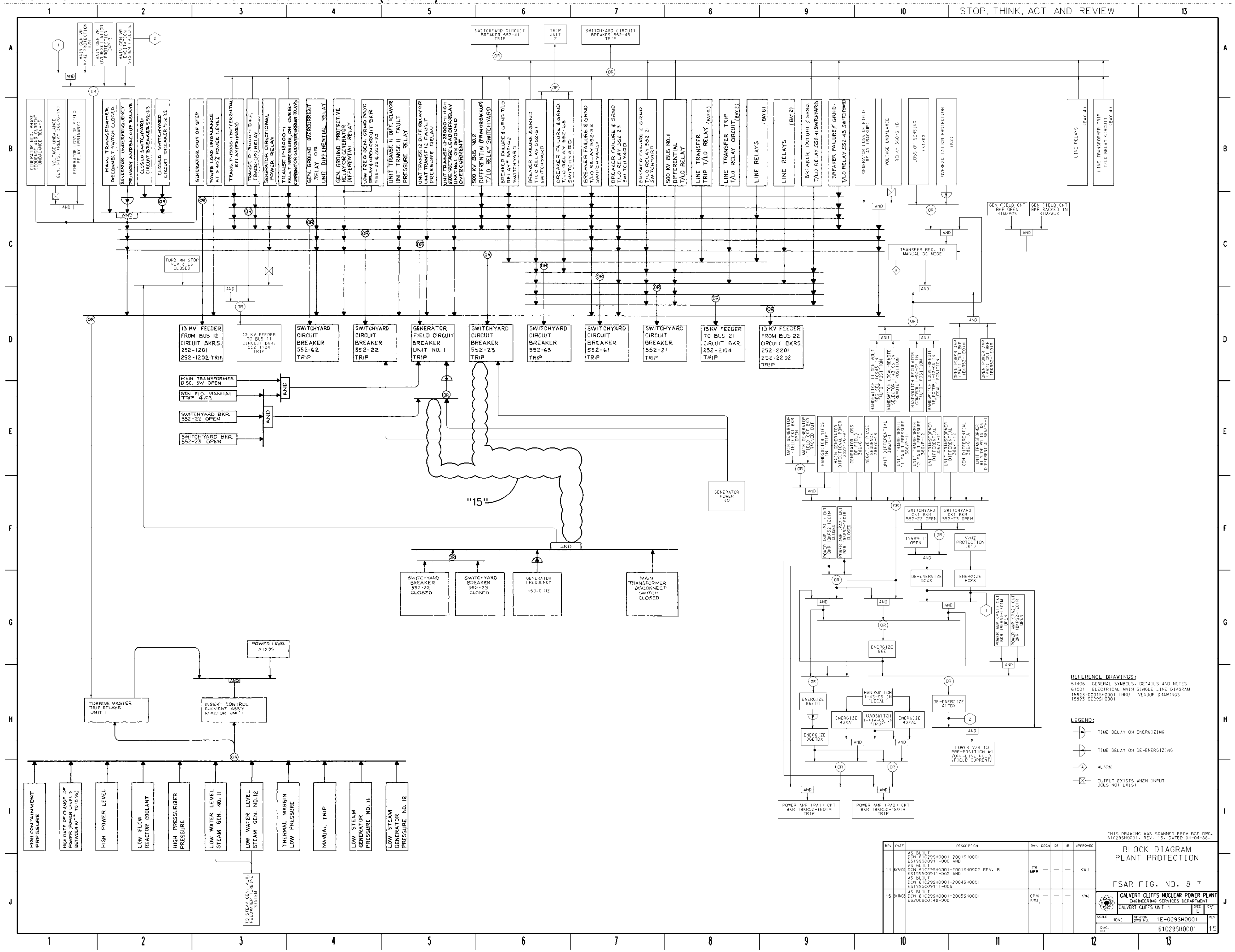


FIGURE 8-7 PLANT PROTECTION BLOCK DIAGRAM UNIT NO. 1 (Sheet 2)

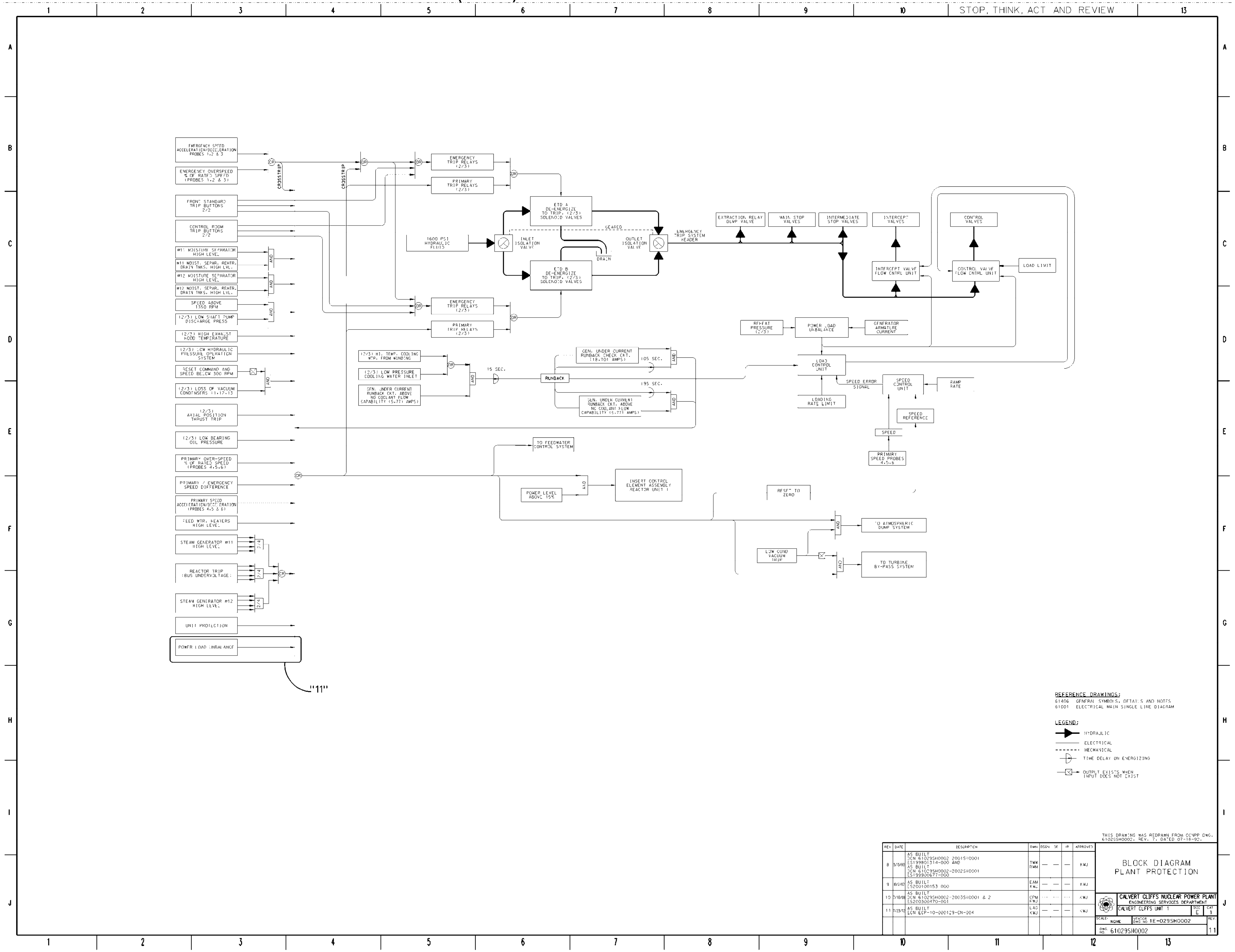


FIGURE 8-8A DIESEL NO. 2A – STARTING AIR, FUEL OIL, AND LUBE OIL

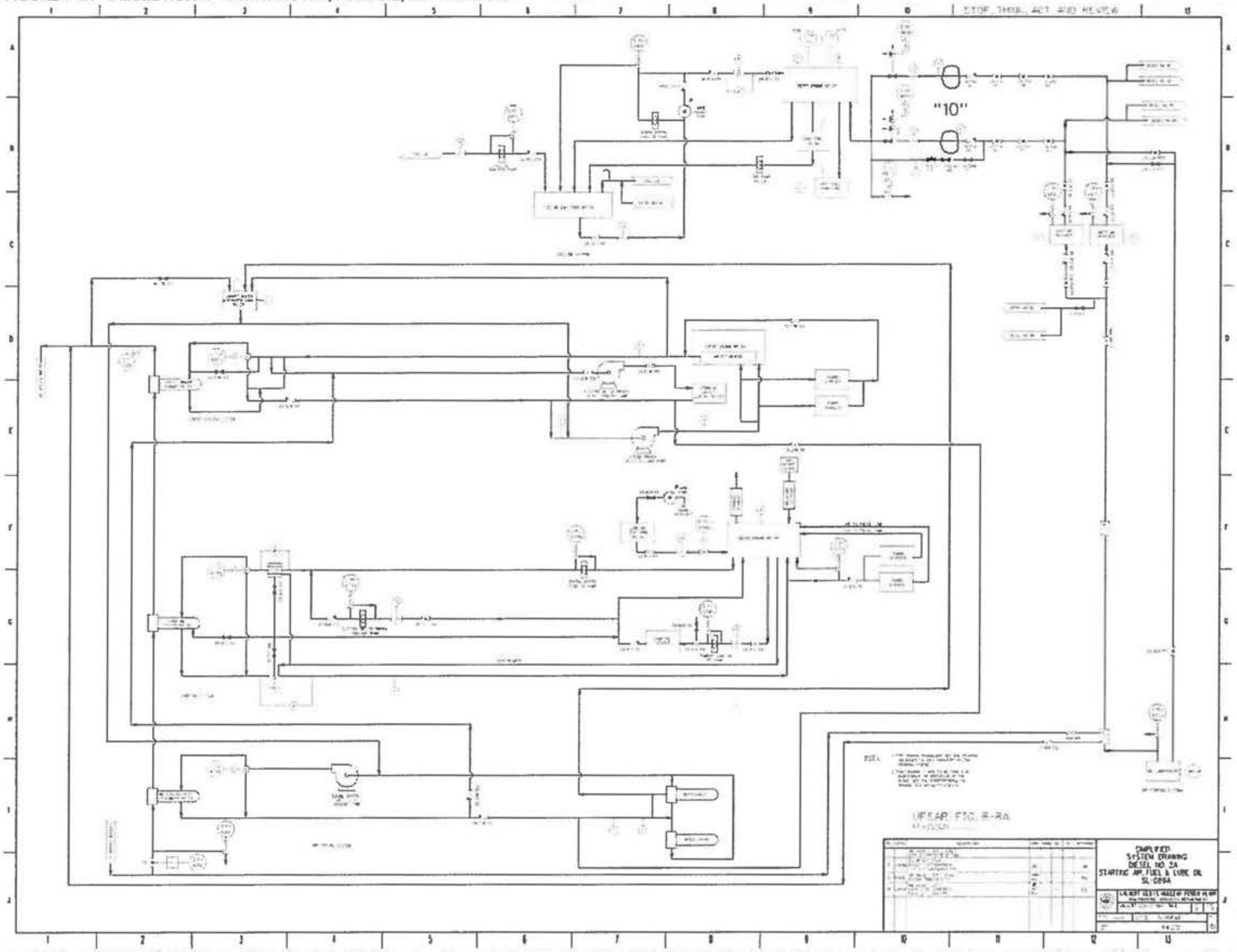


FIGURE 8-8B DIESEL NO. 1B - STARTING AIR, FUEL, AND LUBE OIL

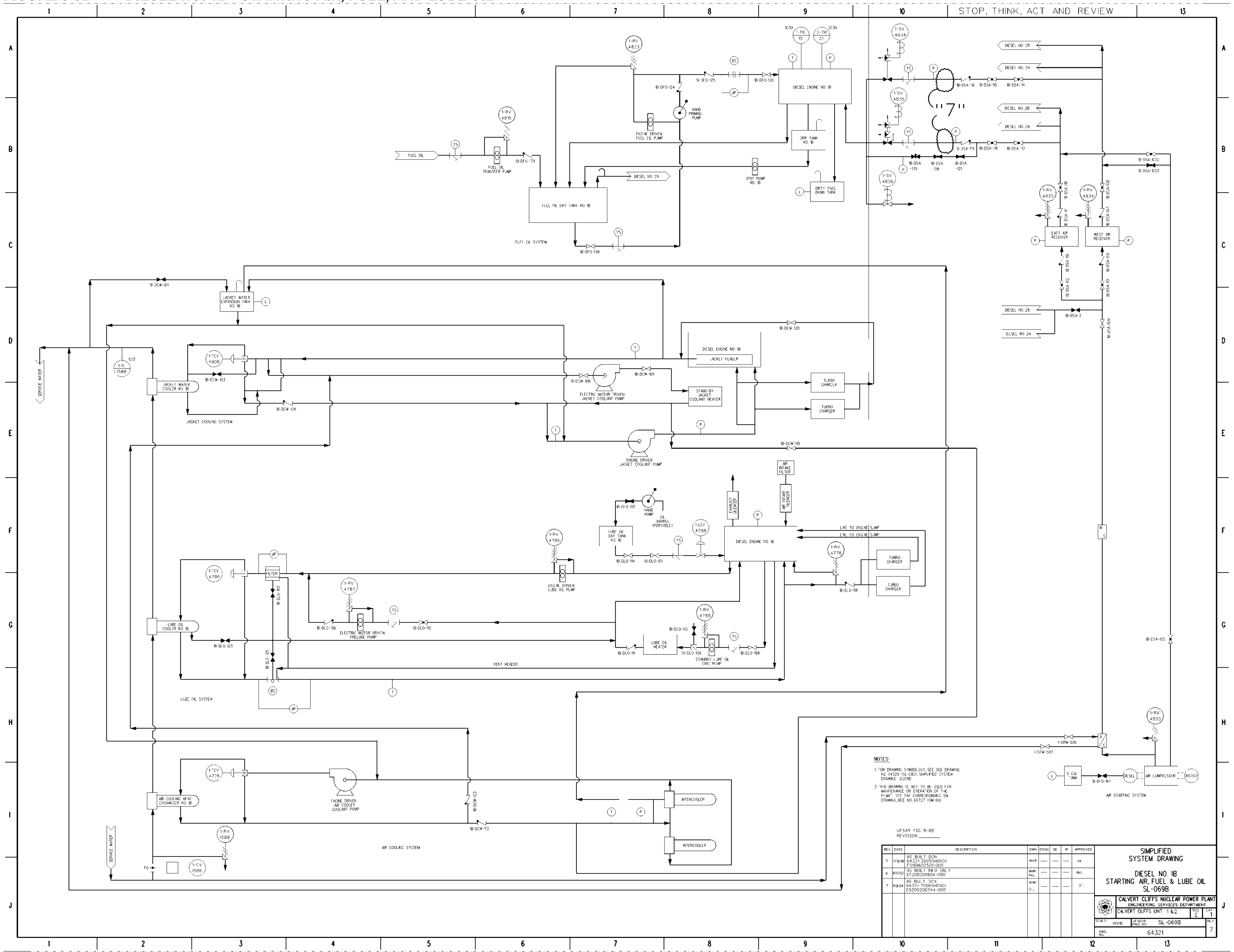
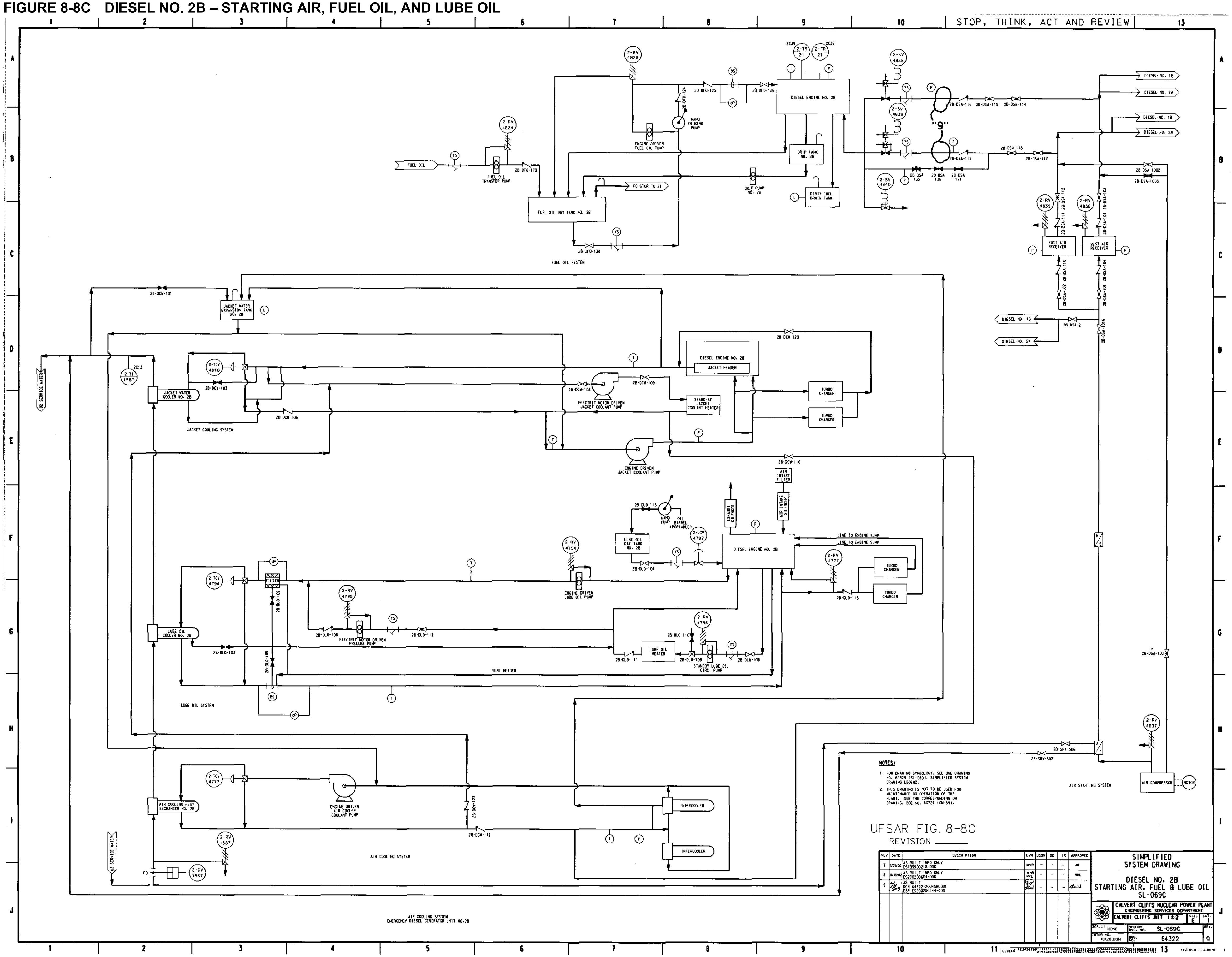
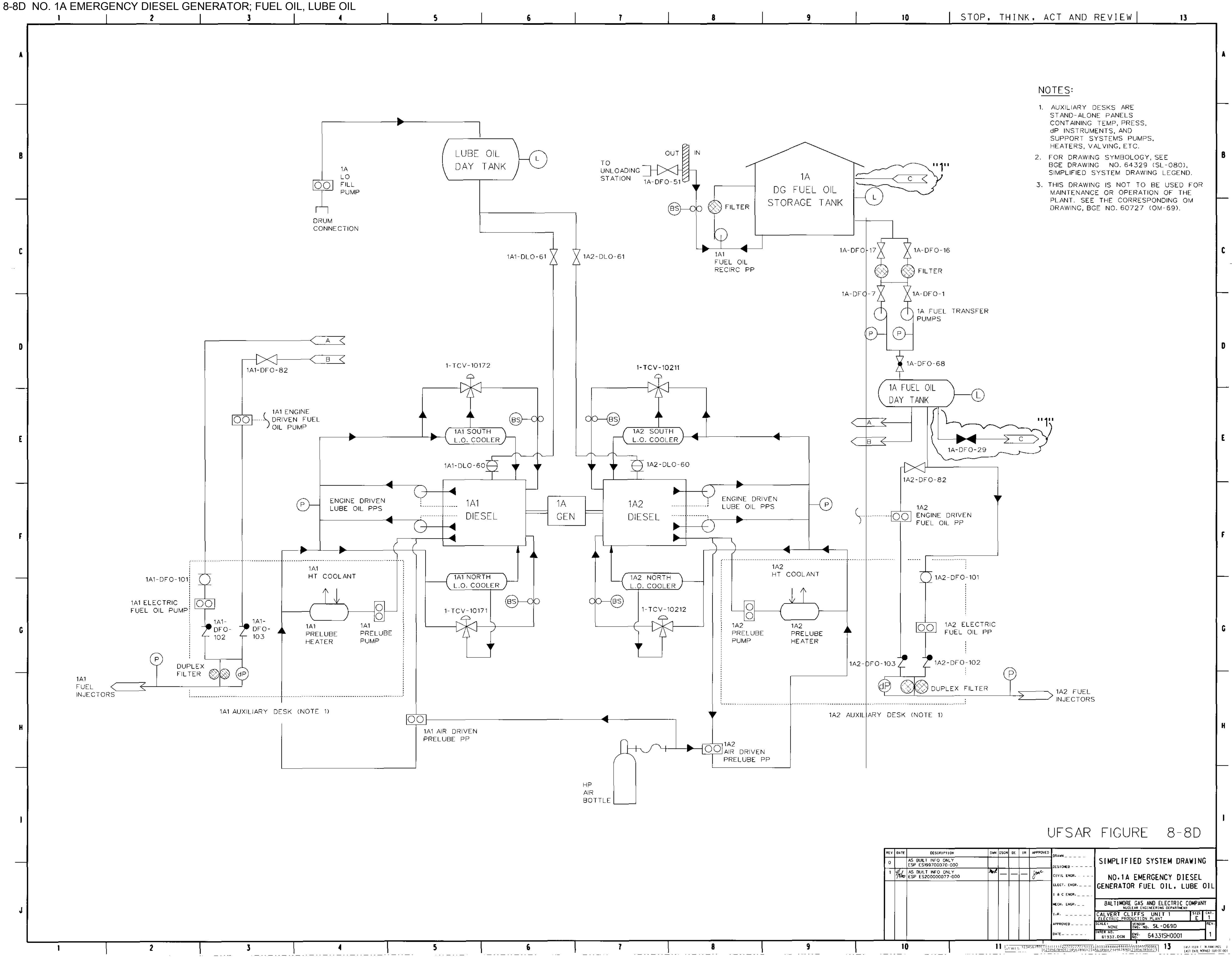


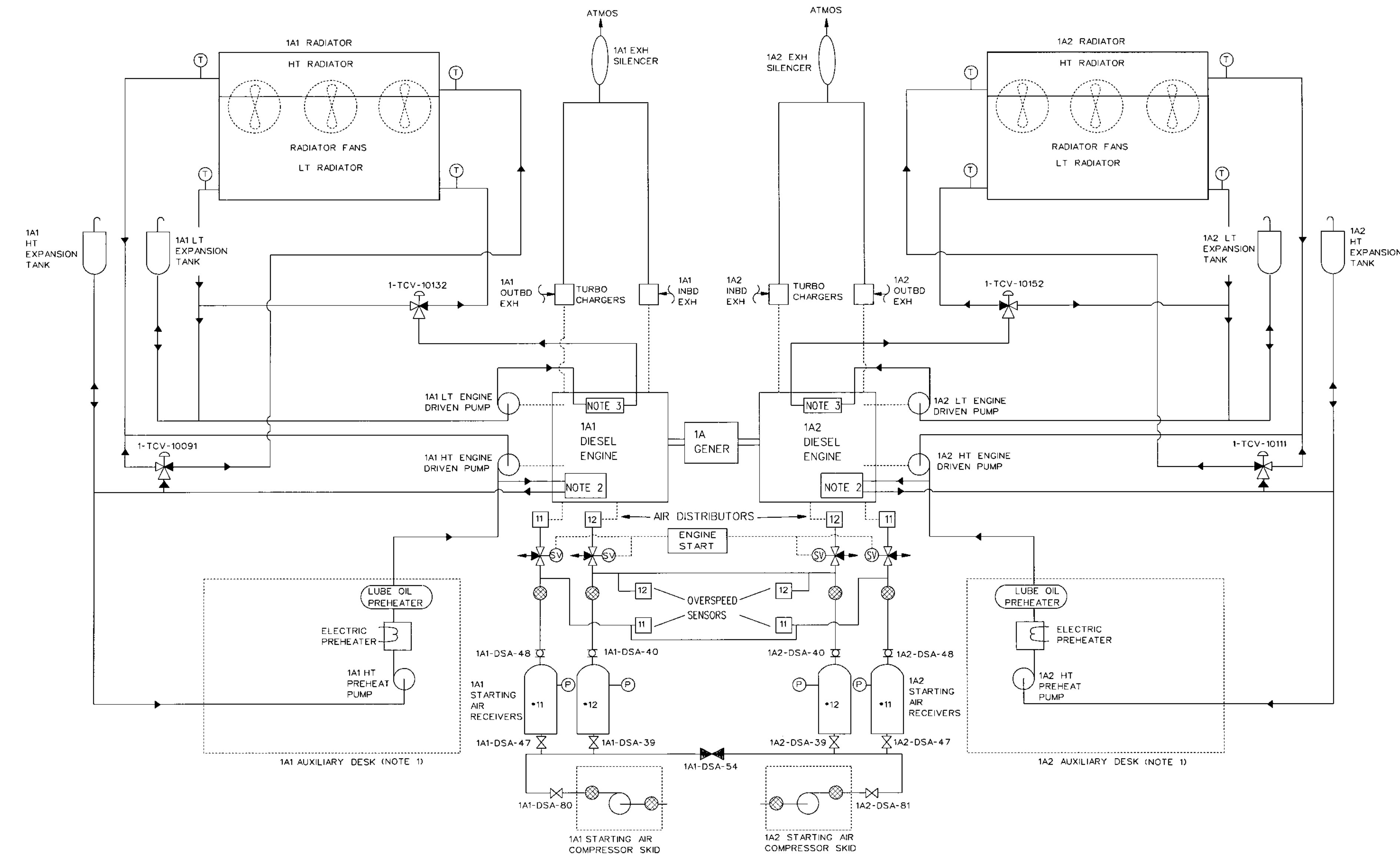
FIGURE 8-8C DIESEL NO. 2B – STARTING AIR, FUEL OIL, AND LUBE OIL



8-8D NO. 1A EMERGENCY DIESEL GENERATOR; FUEL OIL, LUBE OIL



STOP, THINK, ACT AND REVIEW



NOTES:

1. AUXILIARY DESKS ARE STAND-ALONE PANELS CONTAINING TEMP, PRESS & dP INSTRUMENTS AND SUPPORT SYSTEMS PUMPS, HEATERS, VALVING, ETC.
2. HT COOLANT SUPPLIES THE ENGINE BLOCK, TURBOCHARGER AND GOVERNOR OIL TEMPERATURE CONTROLLER.
3. LT COOLANT SUPPLIES THE COMBUSTION AIR INTERCOOLERS AND LUBE OIL COOLERS.
4. FOR DRAWING SYMBOLOGY, SEE BGE DRAWING NO. 64329 (SL-080), SIMPLIFIED SYSTEM DRAWING LEGEND.
5. THIS DRAWING IS NOT TO BE USED FOR MAINTENANCE OR OPERATION OF THE PLANT. SEE THE CORRESPONDING OM DRAWING, BGE NO. 60727 (OM-69).

UFSAR FIGURE 8-8E

REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
0		AS BUILT INFO ONLY ESP ES199700070-000	SF	—	—	—	

DRAWN S.FRASER
DESIGNED
CIVIL ENGR.
ELECT. ENGR.
I & C ENGR.
MECH. ENGR.
I.R.
APPROVED
DATE

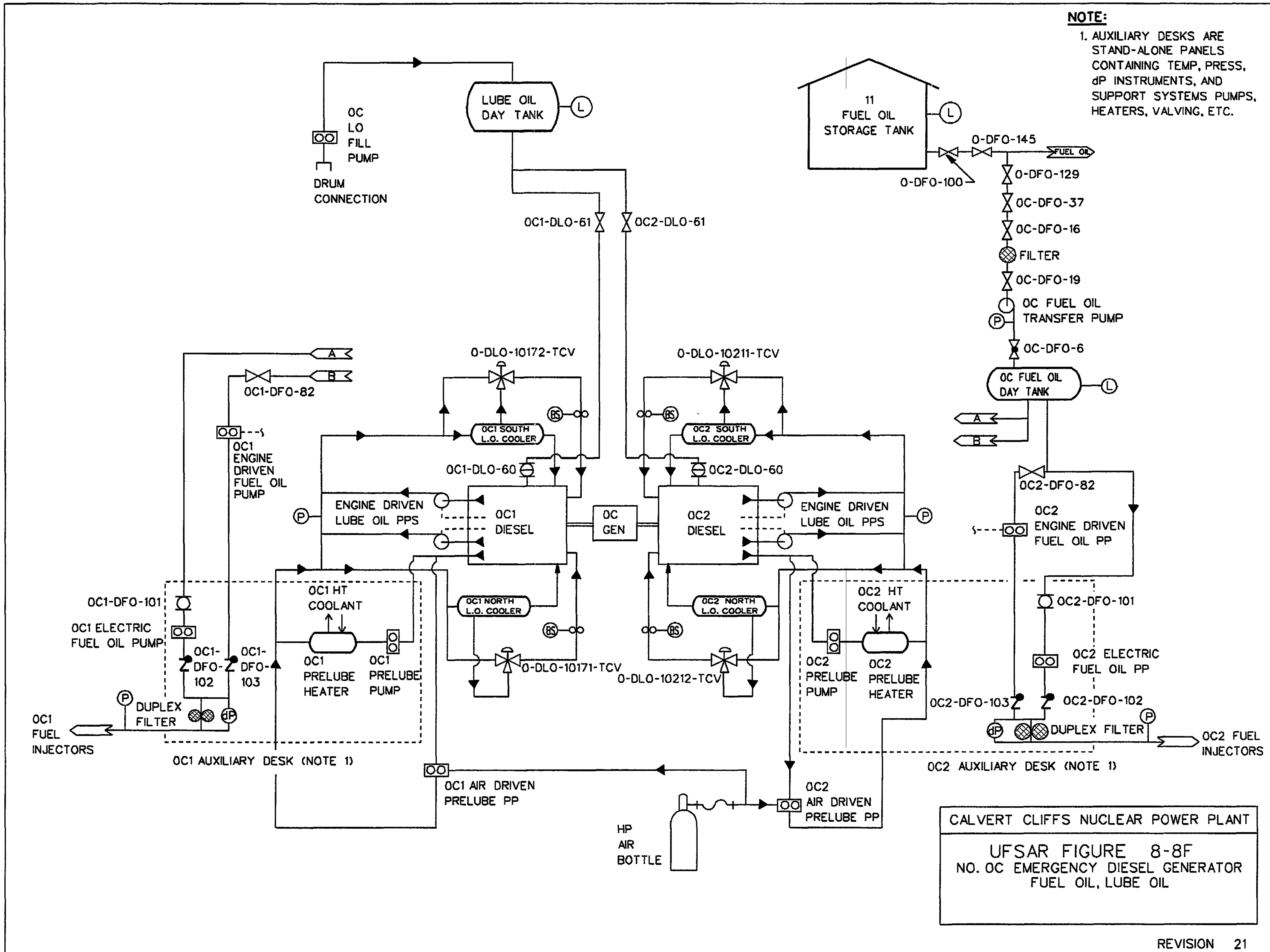
SIMPLIFIED SYSTEM DRAWING
NO.1A EMERGENCY DIESEL GENERATOR STARTING AIR, COOLING, LT COOLING

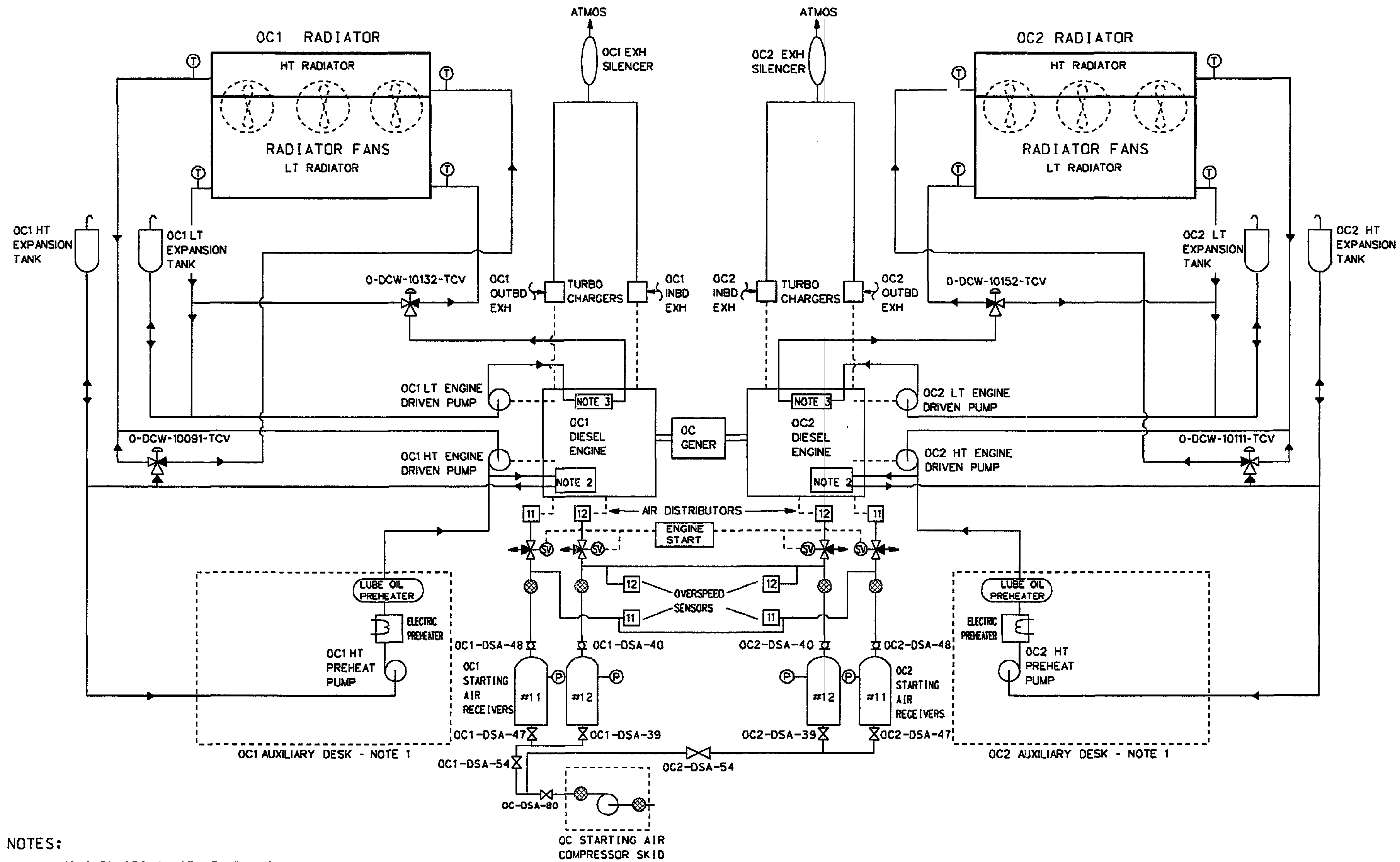
BALTIMORE GAS AND ELECTRIC COMPANY
NUCLEAR ENGINEERING DEPARTMENT

CALVERT CLIFFS UNIT 1
ELECTRIC PRODUCTION PLANT

SIZE E | CAT. 1
SCALE: NONE | VENDOR DWG. NO. SL-069E | REV. 0
INTER NO. 61941.DGN | DWG. NO. 64333SH0001

LAST USER IS.FRASER


NOTE:
1. AUXILIARY DESKS ARE STAND-ALONE PANELS CONTAINING TEMP, PRESS, dP INSTRUMENTS, AND SUPPORT SYSTEMS PUMPS, HEATERS, VALVING, ETC.
0C LO FILL PUMP
DRUM CONNECTION
LUBE OIL DAY TANK
11 FUEL OIL STORAGE TANK
0-DFO-145
FUEL OIL
0-DFO-100
0-DFO-129
0C-DFO-37
0C-DFO-16
FILTER
0C-DFO-19
0C FUEL OIL TRANSFER PUMP
0C-DFO-6
0C FUEL OIL DAY TANK
0C1-DLO-61
0C2-DLO-61
0-DLO-10172-TCV
0-DLO-10211-TCV
0C1-DFO-82
0C1 ENGINE DRIVEN FUEL OIL PUMP
0C1 SOUTH L.O. COOLER
0C2 SOUTH L.O. COOLER
0C1-DLO-60
0C2-DLO-60
ENGINE DRIVEN LUBE OIL PPS
0C1 DIESEL
0C GEN
0C2 DIESEL
ENGINE DRIVEN LUBE OIL PPS
0C2-DFO-82
0C2 ENGINE DRIVEN FUEL OIL PP
0C1-DFO-101
0C1 ELECTRIC FUEL OIL PUMP
0C1 HT COOLANT
0C1 NORTH L.O. COOLER
0C2 NORTH L.O. COOLER
0C2 HT COOLANT
0C2-DFO-101
0C1-DFO-102
0C1-DFO-103
0C1 PRELUBE HEATER
0C1 PRELUBE PUMP
0-DLO-10171-TCV
0-DLO-10212-TCV
0C2 PRELUBE PUMP
0C2 PRELUBE HEATER
0C2 ELECTRIC FUEL OIL PP
0C2-DFO-103
0C2-DFO-102
DUPLEX FILTER
0C1 FUEL INJECTORS
DUPLEX FILTER
0C2 FUEL INJECTORS
0C1 AUXILIARY DESK (NOTE 1)
0C2 AUXILIARY DESK (NOTE 1)
0C1 AIR DRIVEN PRELUBE PP
0C2 AIR DRIVEN PRELUBE PP
HP AIR BOTTLE
CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 8-8F
NO. 0C EMERGENCY DIESEL GENERATOR FUEL OIL, LUBE OIL
REVISION 21



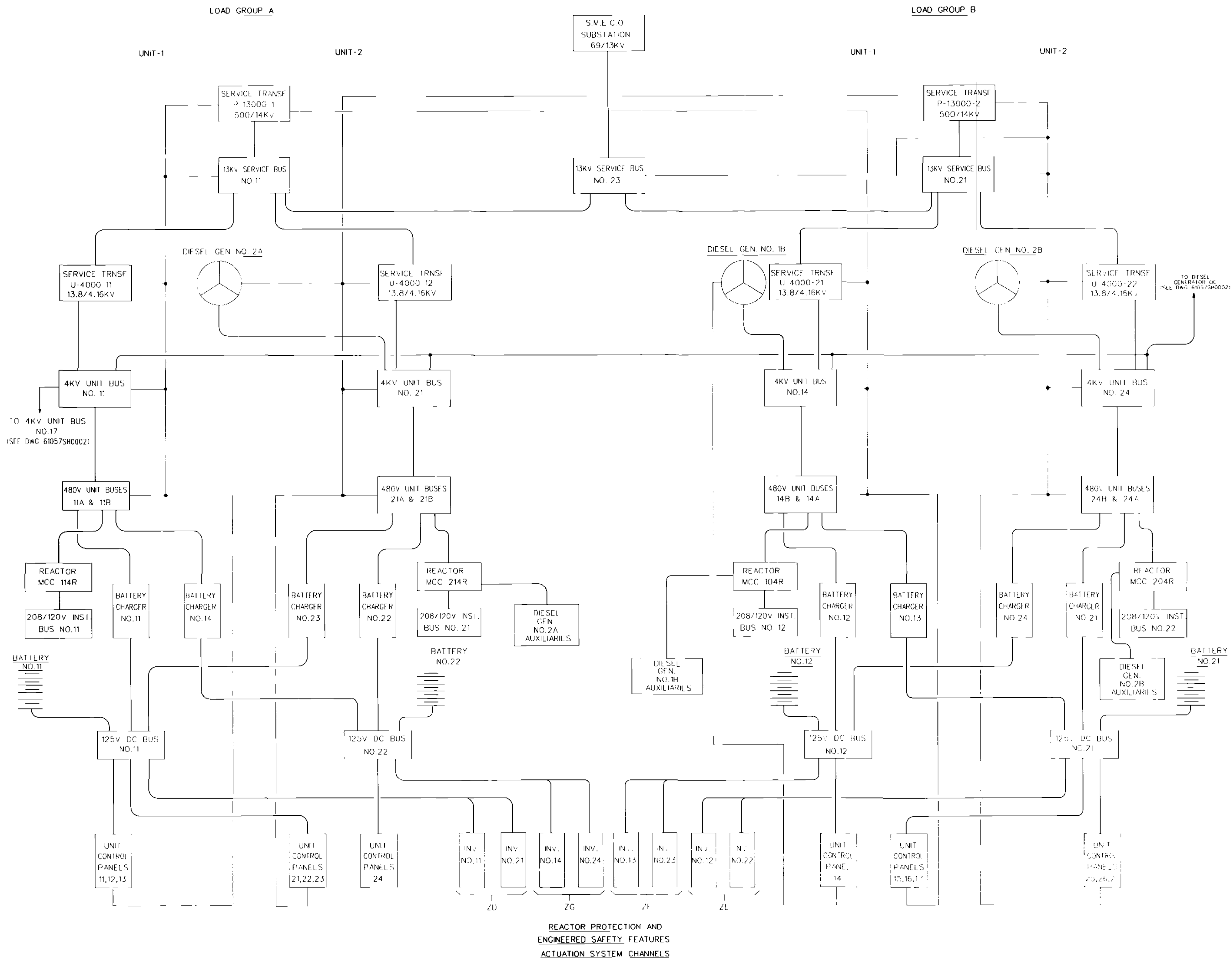
NOTES:

1. AUXILIARY DESKS ARE STAND-ALONE PANELS CONTAINING TEMP. PRESS & dP INSTRUMENTS AND SUPPORT SYSTEMS PUMPS. HEATERS. VALVING. ETC.
2. HT COOLANT SUPPLIES THE ENGINE BLOCK. TURBOCHARGER AND GOVERNOR OIL TEMPERATURE CONTROLLER.
3. LT COOLANT SUPPLIES THE COMBUSTION AIR INTERCOOLERS AND LUBE OIL COOLERS

CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 8-8G
NO. 0C EMERGENCY DIESEL GENERATOR STARTING AIR, HT COOLING, LT COOLING

REV. 21



NOTE:
DARK LINES ARE POWER FEEDERS
LIGHT LINES ARE DC CONTROL FEEDERS

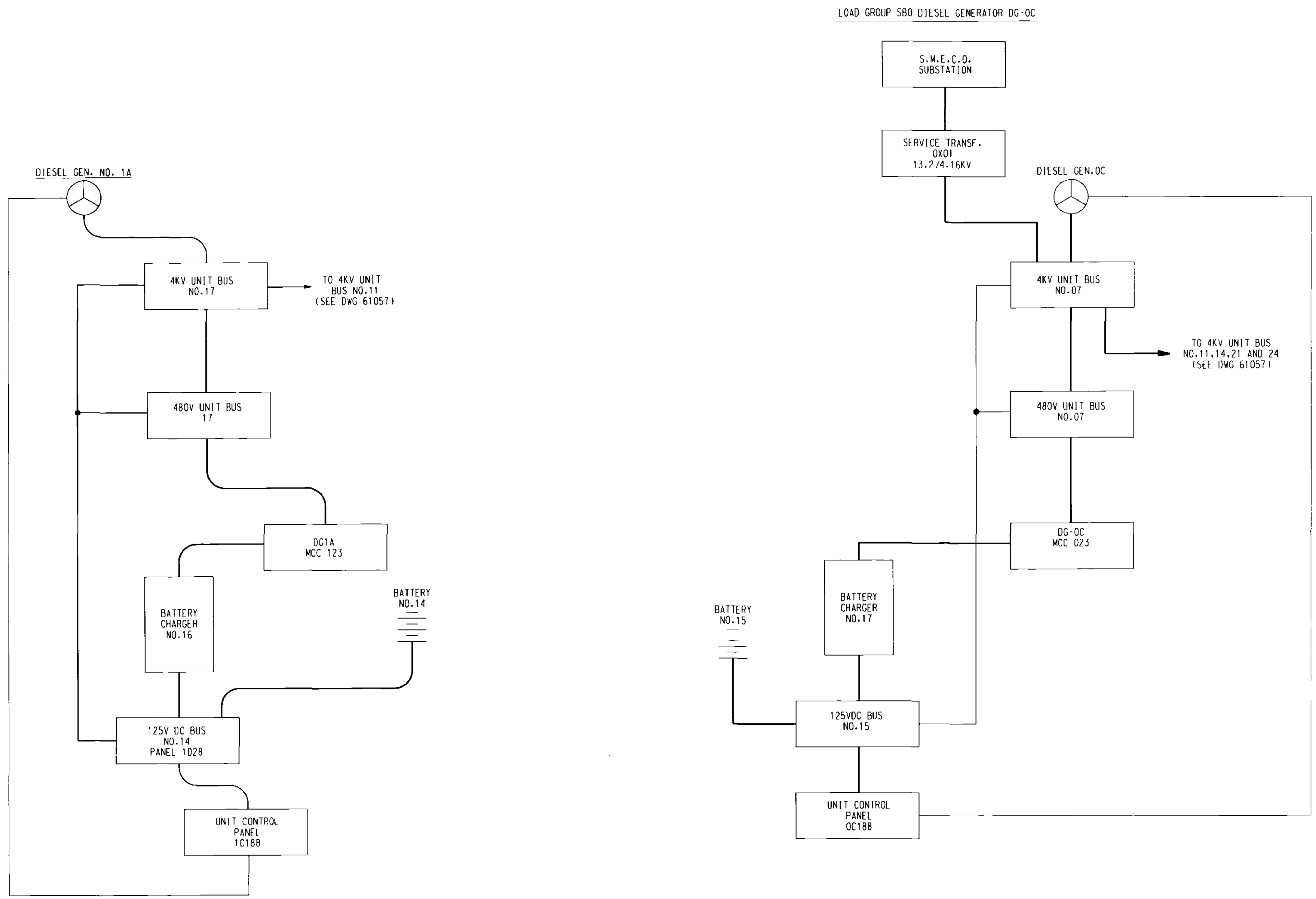
CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 8-9SH0001

BLOCK DIAGRAM
AUXILIARY SYSTEM
LOAD GROUPS
UNITS 1 & 2

BGE DRAWING 61057, REV.9

REVISION 26



NOTES :

1. DARK LINES ARE POWER FEEDERS. LIGHT LINES ARE DC CONROL FEEDERS.
2. DELETED
3. DELETED

REFERENCE DRAWINGS :

61057 BLOCK DIAGRAM AUXILIARY SYSTEM LOAD GROUPS, UNITS 1&2

CALVERT CLIFFS NUCLEAR POWER PLANT

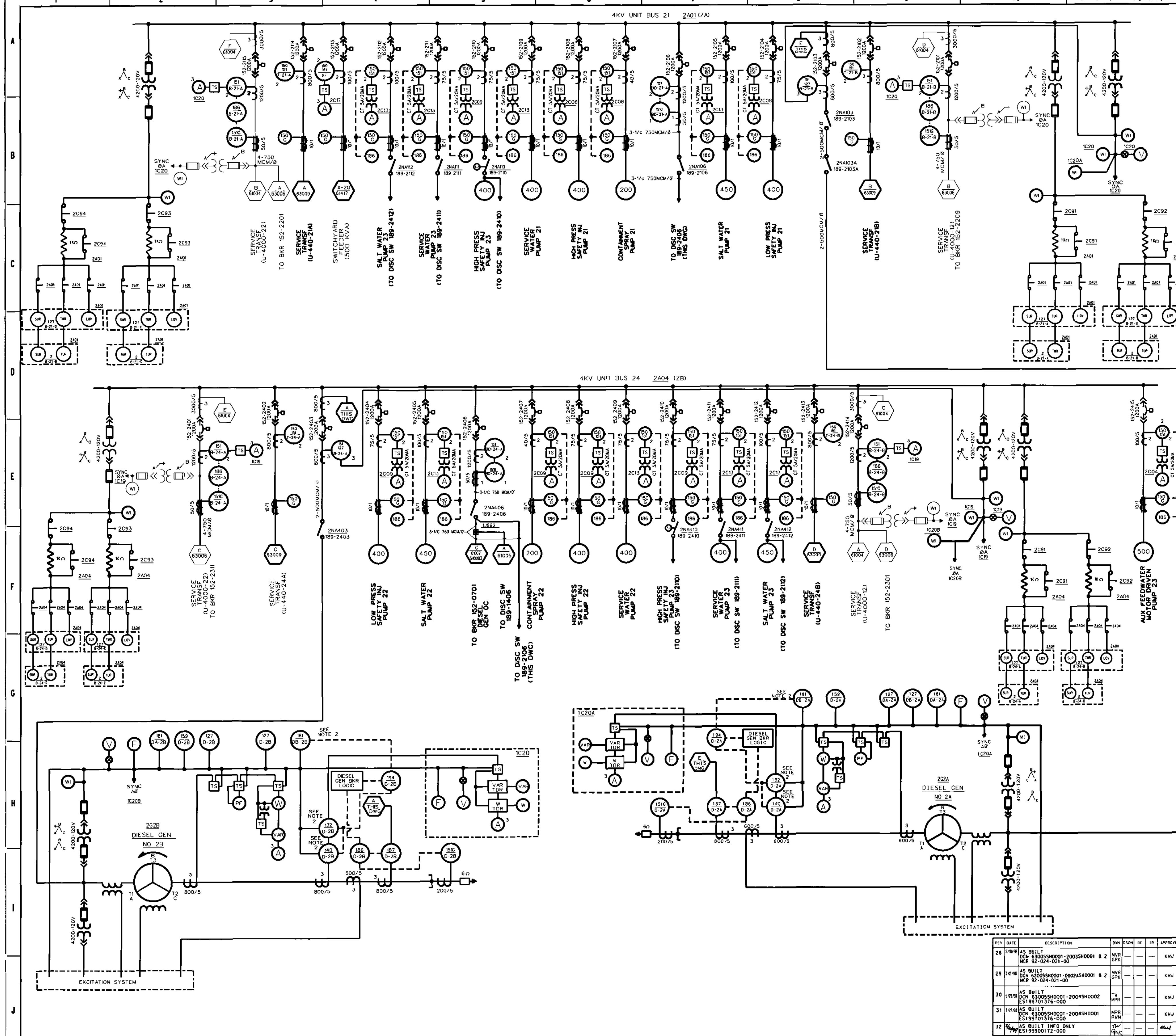
UFSAR FIGURE 8-9SH0002

BLOCK DIAGRAM AUXILIARY SYSTEM LOAD GROUPS DG-OC SBO & DG1A DIESEL GENERATORS

BGE DRAWING 61057SH0002, REV.5

REVISION 26

STOP. THINK. ACT AND REVIEW



DEVICE	DESCRIPTION	MFG	TYPE	STYLE OR MODEL NO
187/D	DIESEL GENERATOR DIFF RELAY	W	SA-1	290B225A10
186/D	DIESEL GENERATOR LOCKOUT REL	ELECTRO SWITCH		7805E
186	MOTOR LOCKOUT RELAY	GE	HEA	12HEA61A223
186/B	SERVICE TRANSF LOCKOUT REL	GE	HEA	12HEA61B234
181/DA	DIESEL GENERATOR UNDERFREQUENCY RELAY	W	CF-1	291B995A10
159/D	DIESEL GENERATOR OVERVOLTAGE RELAY	W	CV-5	1875512
151G/D	DIESEL GENERATOR GROUND OVER-CURRENT RELAY	W	CO-8	1875265
151 127/B	VOLTAGE CONTROLLED OVERCURRENT RELAY	GE	IJCV	12IJCV51A13A
151/B	SERVICE TRANSF OVERCURRENT RELAY	GE	IAC	12IAC53A101A
150 151/T	TRANSF OVERCURRENT RELAY	GE	IAC	12IAC53B104A
151G/B	GROUND OVERCURRENT RELAY	GE	IAC	12IAC53A3A
150G	GROUND OVERCURRENT RELAY	GE	PJC	12PJC11AV1A
150 151	OVERCURRENT RELAY	GE	IAC	12IAC66K8A OR 12IAC66K58A
181/DB	DIESEL GENERATOR UNDERFREQUENCY RELAY	W	KF	671B287A09
150 151/SF	SWITCHYARD FEEDER OVERCURRENT RELAY	GE	IAC	"32" Q12IAC53B812A OR 12IAC53B127A
194/D	AUXILIARY DIESEL GENERATOR CKT BKR TRIP RELAY	W	MG-6	1163801
127/B- -A,B,C,D	4KV UNIT BUS UNDERVOLTAGE RELAY	GE	SLV	SLV11A1A
140/D	DIESEL GENERATOR LOSS OF FIELD RELAY	W	KLF	290B481A09
132/D	DIESEL GENERATOR REVERSE POWER RELAY	W	CRN-1	290B038A09
127/DA	DIESEL GENERATOR UNDERVOLTAGE RELAY	GE	NGV	NGV 13B
127/DB	DIESEL GENERATOR UNDERVOLTAGE RELAY	W	CV-2	1875508
2/B- -A,B,C,D	4KV UNIT BUS STATIC TIMING RELAY	GE	SAM	SAM203A1A
151/BD	DIESEL GEN. BREAKER OVERCURRENT RELAY	GE	IAC	12IAC53A803A
151G/BD	DIESEL GEN. BREAKER GROUND OVERCURRENT RELAY	GE	IAC	12IAC53A801A

REFERENCE DRAWINGS:

63002 METER AND RELAY DIAGRAM GENERATION UNIT 2

61004 SINGLE LINE METER AND RELAY DIAGRAM 13KV SYSTEM UNITS 1 AND 2

63006 METER AND RELAY DIAGRAM 4KV SYSTEM UNIT BUSES 22, 23, 25 AND 26

63009 METER AND RELAY DIAGRAM 480V UNIT BUSES 21A, 21B, 24A AND 24B

61005 METER AND RELAY DIAGRAM 4KV SYSTEM UNIT BUSES 11 AND 14

61406 GENERAL NOTES, SYMBOLS AND DETAILS

61417 AC STATION SERVICE SINGLE LINE DIAGRAM

NOTES:

1 - ALL FEEDERS TO BE 4/0 AWG UNLESS OTHERWISE INDICATED

2 - OPERABLE ONLY WHEN DIESEL IS PARALLELED TO OFFSITE POWER SOURCE

3 - "Q-LIST CLASSIFICATION INFORMATION DEPICTED ON THIS DRAWING PROVIDES AN OVERVIEW OF SAFETY RELATED BOUNDARIES. THE NUCLEIS MASTER EQUIPMENT LIST SHOULD BE UTILIZED FOR SPECIFIC SAFETY CLASSIFICATION, INCLUDING INDIVIDUAL COMPONENT CLASSIFICATION AND ASSOCIATED REQUIREMENTS."

4 - BOLD LINE WEIGHT DENOTES SAFETY RELATED BOUNDARIES.

REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
28	2/18/98	AS BUILT DCN 63005SH0001-2003SH0001 & 2 MCR 92-024-021-00	MVR GPK	—	—	—	KWJ
29	3/2/98	AS BUILT DCN 63005SH0001-0002ASH0001 & 2 MCR 92-024-021-00	MVR GPK	—	—	—	KWJ
30	6/29/98	AS BUILT DCN 63005SH0001-2004SH0002 ES199701376-000	TW MPR	—	—	—	KWJ
31	7/22/98	AS BUILT DCN 63005SH0001-2004SH0001 ES199701376-000	MPR RWM	—	—	—	KWJ
32	2/10/99	AS BUILT INFO ONLY ES199900172-000	TW GPK	—	—	—	[illegible]

DRAWN — DESIGNED — CIVIL ENGR. — ELECT. ENGR. — I & C ENGR. — MECH. ENGR. — I.R. — APPROVED — DATE —

METER AND RELAY DIAGRAM
4KV SYSTEM
UNIT BUSES 21 AND 24
FSAR FIG. NO. 8-10

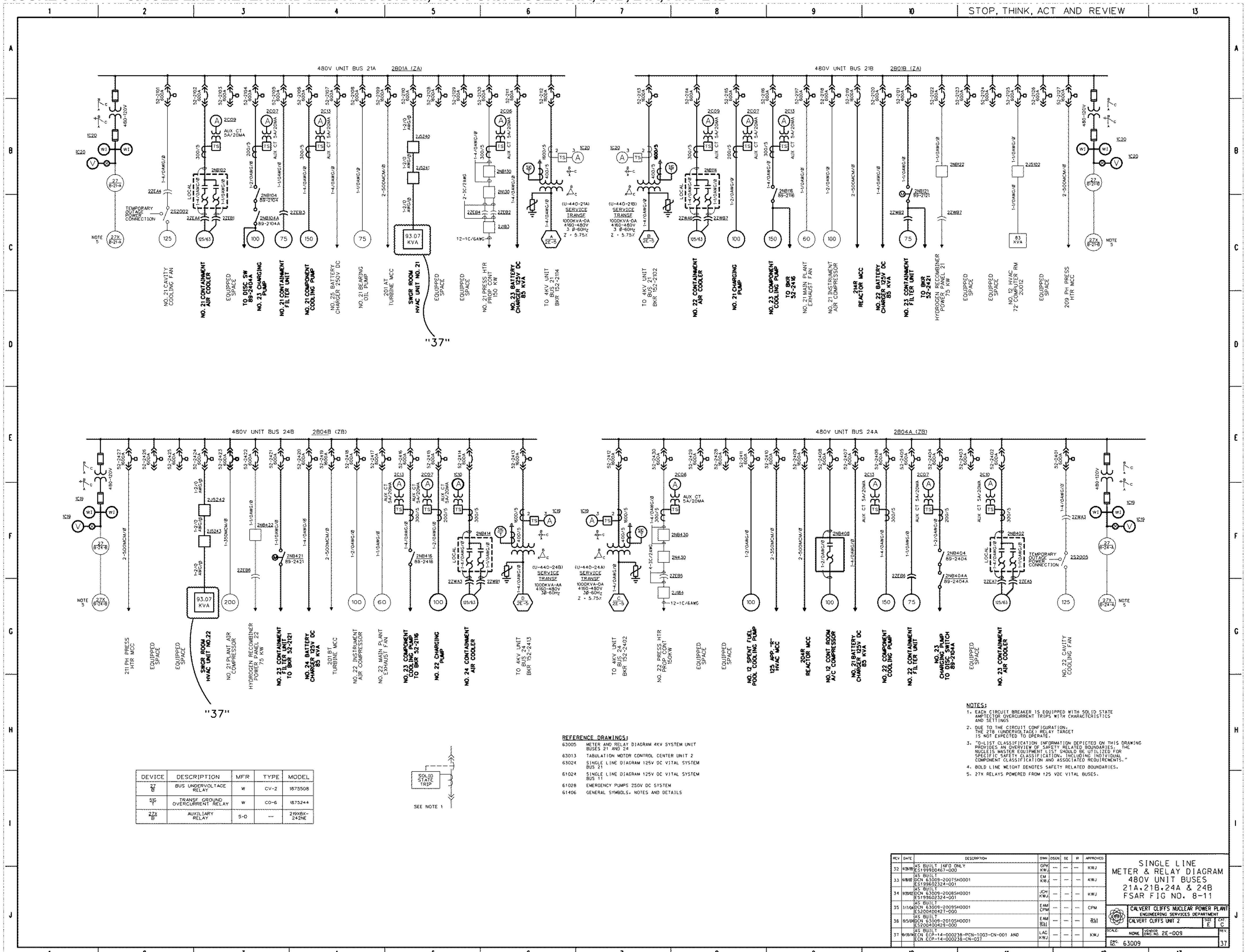
BALTIMORE GAS AND ELECTRIC COMPANY
NUCLEAR ENGINEERING DEPARTMENT

CALVERT CLIFFS UNIT 2
ELECTRIC PRODUCTION PLANT

SIZE E CAT. C
SCALE: NONE
VENDOR DWG. NO. 2E-005SH0001
INTER NO. 383-DGN
DWG. NO. 63005SH0001
REV. 32

LAST USER : T. WILLIAMS
LAST DATE WORKED : 02/09/99

FIGURE 8-11 SINGLE LINE METER AND RELAY DIAGRAM, 480 V UNIT BUSES 21A, 21B, 24A, AND 24B



DEVICE	DESCRIPTION	MFR	TYPE	MODEL
27/B	BUS UNDERVOLTAGE RELAY	W	CV-2	1875508
51G/Y	TRANSF GROUND OVERCURRENT RELAY	W	CO-6	1875244
27X/B	AUXILIARY RELAY	S-D	---	219XBX-242NE

FIGURE 8-12   PLANT PROTECTION BLOCK DIAGRAM, UNIT NO. 2 (Sheet 1)

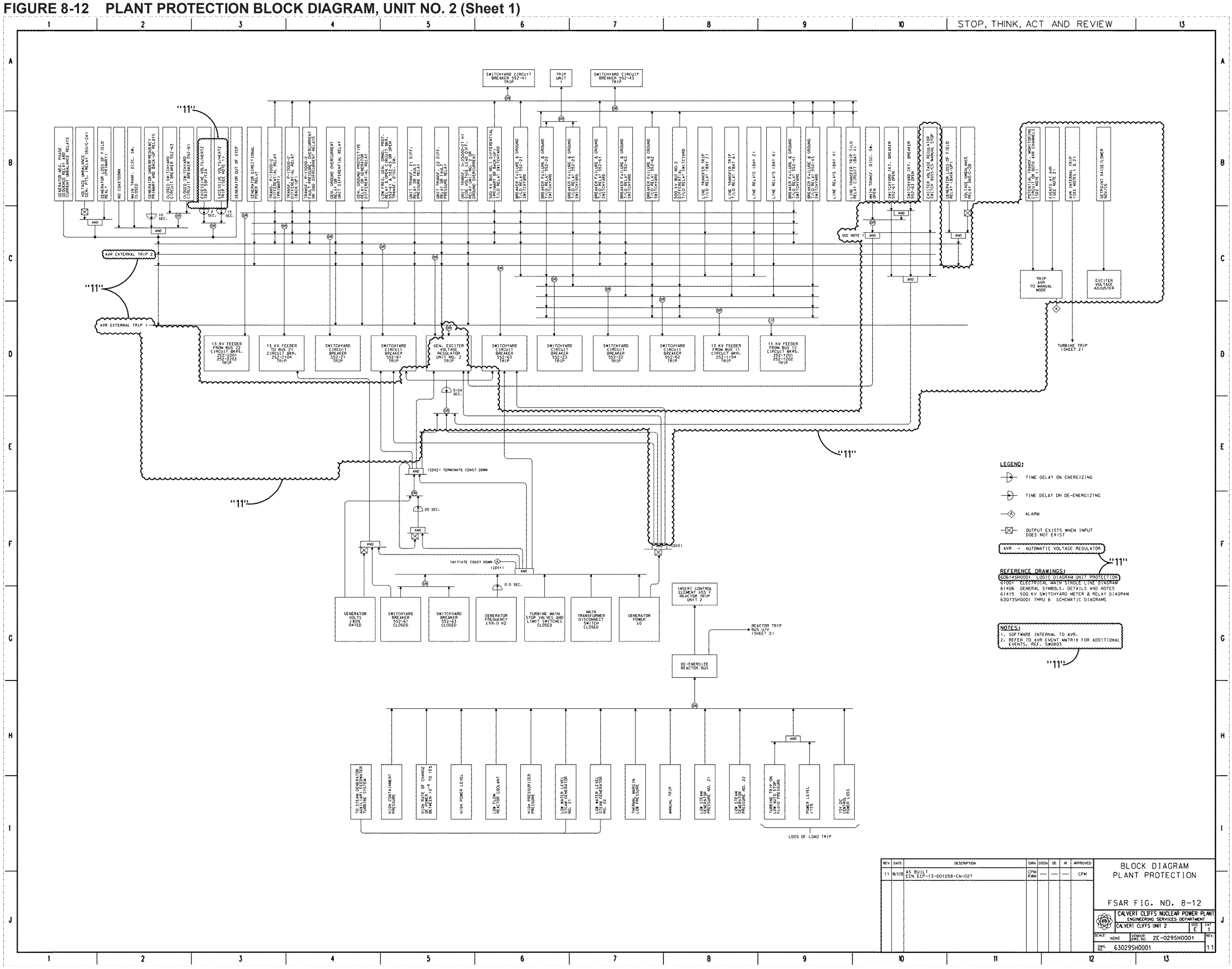
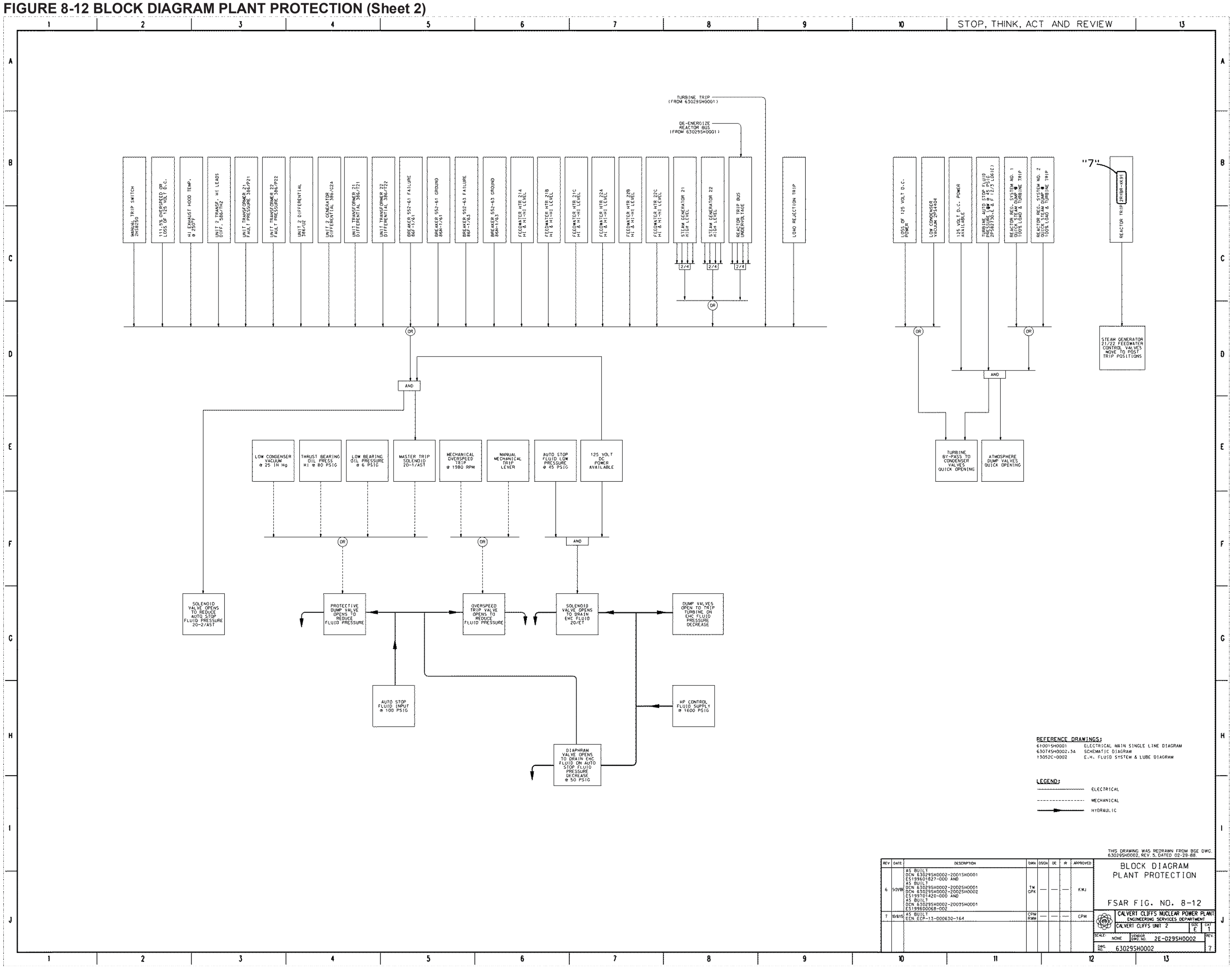


FIGURE 8-12 BLOCK DIAGRAM PLANT PROTECTION (Sheet 2)



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CHAPTER 9
AUXILIARY SYSTEMS

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LIST OF ACRONYMS

AFFF	Aqueous Film Forming Foam
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CC	Component Cooling
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FCR	Facility Change Request
FSAR	Final Safety Analysis Report
HEPA	High Efficiency Particulate Air
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilation, and Air Conditioning
ICI	Incore Instrumentation
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MOV	Motor-Operated Valve
MWPS	Miscellaneous Waste Processing System
NEMA	National Electrical Manufacturers Association
NEOP	Nuclear Engineering Operator Procedure
NFPA	National Fire Protection Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PASS	Post Accident Sampling System
QC	Quality Control
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTD	Resistance Temperature Detector
RWT	Refueling Water Tank
SDC	Shutdown Cooling
SFHM	Spent Fuel Handling Machine
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SRW	Service Water
SWAC	Saltwater Air Compressors

LIST OF ACRONYMS

TEMA	Tubular Exchanger Manufacturers Association
VAP	Value Added Pellet
VCT	Volume Control Tank
WPS	Waste Processing System
ZrB_2	Zirc Diboride

9.0 AUXILIARY SYSTEMS

A legend for the figures in Chapter 9 is located on Figure 9-1.

9.1 CHEMICAL AND VOLUME CONTROL SYSTEM

9.1.1 DESIGN BASIS

The Chemical and Volume Control System (CVCS) is designed to perform the following functions:

a. Maintain reactor coolant activity at the desired level by removing corrosion and fission products;

b. Inject chemicals into the Reactor Coolant System (RCS) to control coolant chemistry and minimize corrosion;

c. Control the reactor coolant volume by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature and other coolant losses or additions;

d. Provide means for transferring fluids to the radioactive Waste Processing System (WPS);

e. Inject concentrated boric acid into the RCS upon a safety injection actuation signal (SIAS);

f. Control the reactor coolant boric acid concentration;

g. Provide auxiliary pressurizer spray for operator control of RCS pressure during shutdown;

h. Provide a means for functionally testing the check valves which isolate the Safety Injection (SI) System from the RCS (Although this is a design function of the CVCS, these check valves are functionally tested in accordance with the Inservice Test Program.), and for hydrostatic and leak testing of the RCS;

i. Provide continuous on-line measurement of reactor coolant fission product activity.

Portions of the letdown system are American Society of Mechanical Engineers (ASME) Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the letdown system fatigue analysis considers 200 events where letdown flow is lost for an extended period of time. After the letdown piping has cooled to ambient temperature, a restart of letdown flow results in a rapid increase to RCS temperature. See Reference 1 for further details.

Portions of the charging system are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, b, c, d, e, and g of Section 4.1.1, the charging system fatigue analysis considers 200 loss of letdown events and 200 loss of charging events. Also, certain auxiliary spray transients must be considered for portions of the charging system. See Reference 2 for details on the assumed temperatures and sequence of events of these transients.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

9.1.2 SYSTEM DESCRIPTION

9.1.2.1 General

The CVCS is shown in Figures 9-3 (Unit 1) and 9-24 (Unit 2). Coolant normally flows through the CVCS, as shown in Figures 9-3 and 9-24. Coolant letdown from one reactor coolant loop cold leg first passes through the tube side of a regenerative heat exchanger where the temperature is reduced to approximately 232°F and then through the letdown control valves. The letdown control valves, which are modulated by the pressurizer level control system, control the letdown flow to maintain proper pressurizer level. The letdown coolant temperature is reduced to 120°F at the letdown heat exchanger downstream of the letdown control valves. This temperature is selected to prevent deterioration of the ion exchange resins downstream. Flashing of the hot liquid between the letdown control valves and the letdown heat exchanger is prevented by controlling back pressure with a pressure control valve downstream of the letdown heat exchanger.

The cooled letdown next passes through one of two purification filters which remove suspended solids from the letdown before it enters an ion exchanger. The purified flow from the ion exchanger is sprayed into the Volume Control Tank (VCT) after passing through a strainer.

Charging pumps take suction from the VCT to add makeup coolant to the RCS via the shell side of the regenerative heat exchanger.

A small bypass flow around the purification filters passes through a process radiation monitor (to measure coolant activity). The proper flow rate is obtained by throttling the process radiation monitor outlet valve. There is also an indicating alarm on the discharge to monitor flow and alarm on a low flow condition.

If the level in the VCT reaches the high level setpoint, the letdown flow is automatically diverted to the liquid WPS. If the level in the VCT reaches the low-level setpoint, makeup water, borated to the existing concentration of the RCS, can be automatically supplied to the VCT. During normal operation when the level in the VCT reaches the low-low level setpoint, the tank discharge valve shuts and the suction of the charging pump is automatically aligned to the Refueling Water Tank (RWT).

With the level in the normal control band, the VCT has sufficient capacity to accommodate the variation in water inventory of the RCS due to power level changes in excess of that accommodated by the pressurizer.

Boric acid required for makeup can be supplied from either boric acid batching or from the boron recovery system. The boron recovery system is described in Section 11.1.2. The concentrated boric acid is stored in two heated storage tanks. Two pumps are provided to transfer concentrated boric acid. The piping is arranged such that the boric acid may be mixed with demineralized water in a predetermined ratio prior to being introduced to the VCT.

Chemicals are introduced to the RCS directly from a chemical addition tank or via a chemical addition metering pump, both of which are connected to the charging pump suction header.

The RCS may be tested for leaks when the plant is shut down using a charging pump for pressurization. The system is also provided with connections for installing a hydrostatic test pump.

9.1.2.2 Volume Control

The CVCS automatically adjusts the volume of water in the RCS using a signal from level instrumentation located on the pressurizer. The system reduces the amount of fluid that must be transferred between the coolant system and the CVCS during power changes by employing a programmed pressurizer level setpoint which varies with reactor power level. The setpoint varies linearly with the average reactor coolant temperature. This linear relationship is shown in Figure 4-10 (Section 4.3). The control system compares the programmed level setpoint with the measured pressurizer water level. The resulting error signal is used to control the operation of the charging pumps and the letdown valves, as described below. The pressurizer level control program is shown in Figure 4-11.

The pressurizer level control program regulates the letdown flow by adjusting the letdown control valve, so that the RCP controlled bleed-off plus the letdown flow matches the input from the operating charging pump. When the equilibrium is disturbed by a power change or for any other reason, a decrease in level will start one or both standby charging pumps to restore level, and an increase in level will increase the letdown flow rate and initiate a backup signal to stop the two standby charging pumps.

The VCT coolant level can be automatically controlled. When the level in the tank reaches the high-level setpoint, the letdown flow is automatically diverted to the liquid waste processing system. If the makeup mode selector switch is in auto when the level in the tank reaches the low-level setpoint, makeup water is automatically supplied.

When the Control Room handswitches for the VCT outlet valve and RWT charging pump suction valve are in AUTO, and the level in the VCT reaches the low-low-level setpoint, the VCT outlet valve automatically closes and the RWT charging pump suction valve automatically opens, realigning the suction of the charging pumps to the RWT. On a loss of power to the level transmitter controlling this automatic action, the handswitch for the VCT outlet valve is placed in OPEN and the handswitch for the RWT charging pump suction valve is placed in CLOSE to reverse the automatic realignment that occurs. In this condition, VCT level can still be monitored in the Control Room from an indicator that is supplied from an independent power source.

The VCT can be vented to the WPS. The tank is normally operated with sufficient hydrogen partial pressure such that the RCS hydrogen concentration is consistent with plant chemistry requirements as discussed in Sections 4.1.4.2.3 and 9.1.2.3. However, other gases dissolved in the reactor coolant can leave solution when the letdown flow is sprayed into the VCT.

9.1.2.3 Chemical and Reactivity Control

The CVCS purifies and conditions the coolant by means of ion exchangers, filters, degasification and chemical additives. The purification ion exchangers contain a mixed resin bed which removes soluble impurities by ion exchange and suspended impurities by impaction of the particles on the surface of the resin beds.

Cartridge-type filters located upstream of the ion exchangers remove most of the suspended impurities to prevent clogging of the resin beds.

Dissolved gases may be removed from the coolant by venting the VCT and purging with nitrogen as required.

The reactor coolant is chemically conditioned to the typical conditions recommended in the EPRI PWR Primary Water Chemistry Guidelines:

a. Hydrazine scavenging to remove oxygen prior to exceeding 250°F;
b. Maintaining excess hydrogen concentration to control oxygen concentration and suppress radiolysis when the reactor is critical;
c. Chemical additives to control pH when the reactor is critical. As an exception to the Electric Power Research Institute Guidelines, the RCS lithium concentration may be as high as 5.33 ppm for approximately the first 4 effective full power days and 5.30 ppm for the remainder of the fuel cycle to optimize RCS pH.
d. Low levels of zinc acetate may be added to Units 1 and 2 for purposes of reducing dose and mitigating primary water stress corrosion cracking.
e. Hydrogen peroxide may be added at shutdown to promote dissolution of radiocobalt and scavenge hydrogen.

The chemical addition tank or chemical addition metering pump is used to feed chemicals to the charging pumps which inject the additives into the RCS. The reactor coolant makeup pumps can inject hydrazine into the makeup train to scavenge dissolved oxygen.

The CVCS is designed to prevent fission and corrosion product activities from exceeding the values given in Chapter 11 when operating with 1% failed fuel.

Reactivity Control

The boron concentration of the reactor coolant is controlled by the CVCS to:

1. Optimize the position of the control rods;
2. Compensate for reactivity changes caused by variations in the temperature of the coolant, and by burnup of the core;
3. Provide a margin of shutdown for maintenance, refueling or emergencies.

The system includes a batching tank for preparing boric acid solution, two tanks for storing the solution, and two pumps for supplying boric acid solution to the makeup system. Boric acid from the waste processing system is pumped to either the boric acid storage or batching tanks.

Normally, the CVCS adjusts the boric acid concentration of the coolant by feed and bleed. To change concentration, the makeup (feed) system supplies either demineralized water or concentrated boric acid to the VCT or directly to the charging pump suction header, and the letdown (bleed) stream is diverted to the WPS. Toward the end of a core cycle, an ion exchanger is used to deborate. This avoids the excessive quantity of waste produced due to the feed and bleed operations.

The system can add boric acid to the reactor coolant at a sufficient rate to override the maximum increase in reactivity due to cooldown and the decay of xenon in the reactor. The control element assemblies (CEAs) can decrease reactivity far more rapidly than the boron removal system can increase reactivity.

The charging pumps may be used to leak test the RCS at normal operating pressure when the plant is shut down. Leaks in the RCS may be detected while

the plant is at power by monitoring pressurizer level, VCT level, letdown flow, reactor coolant drain tank level, coolant temperature, and charging flow rate.

9.1.3 SYSTEM COMPONENTS

The major components of the CVCS and their functions are described in this section.

9.1.3.1 Description

Regenerative Heat Exchanger

The regenerative heat exchanger (Table 9-3) transfers heat from the letdown stream to the charging stream. Materials of construction are primarily austenitic stainless steel.

Letdown Control Valves

The letdown control valves (Table 9-4) regulate the reactor coolant flow from the regenerative heat exchanger as required by the pressurizer level regulating system. The valves reduce the pressure of the letdown fluid to about 460 psig. This value prevents flashing with about a 30 psi margin, even with minimum makeup flow (44 gpm charging) and maximum letdown flow (128 gpm) [Table 9-1, note [(a)]]. The letdown flow is nominally 38 gpm, for coolant purification, but will vary as the pressurizer water level changes. The valves are pneumatically-operated and fail closed. All parts in contact with reactor coolant are of austenitic stainless steel.

Letdown Heat Exchanger

The letdown heat exchanger (Table 9-5) cools the letdown stream in the tube side of the regenerative heat exchanger to a temperature suitable for entry into a purification ion exchanger. Component cooling system fluid is the cooling medium on the shell side of the letdown heat exchanger. Tube side materials of construction are primarily austenitic stainless steel; shell side materials of construction are primarily carbon steel [Table 9-1, note [(a)]].

Ion Exchangers

Three purification ion exchangers (Table 9-6) are available to purify and remove boron from the reactor coolant. These ion exchangers are identical in design and may be interchanged during operation. Each unit is designed to handle the maximum letdown flow of 128 gpm [Table 9-1, note [(a)]]. The vessels and resin retention element are of austenitic stainless steel construction.

Mixed bed resin is loaded into an Ion Exchanger to purify the reactor coolant by removing corrosion and fission products. Toward the end of core life, resin is loaded into one or more ion exchangers to reduce boron concentration in the reactor coolant. This method is preferable to using feed and bleed since it minimizes the volume of radioactive waste water produced.

Purification Filters

The purification filters (Table 9-7) remove suspended impurities from the reactor coolant. Each filter will accommodate maximum letdown flow of 128 gpm. The filter housings are austenitic stainless steel.

Volume Control Tank (VCT)

The VCT (Table 9-8) accumulates water from the RCS. The tank has sufficient capacity to accommodate the variation in water inventory of the RCS due to power

level changes in excess of that accommodated by the pressurizer. The tank provides a gas space where a partial pressure of hydrogen and nitrogen is maintained to control the hydrogen and nitrogen concentration in the reactor coolant. A vent to the WPS permits removal of hydrogen, nitrogen and gaseous fission products released from solution in the VCT. The tank is of austenitic stainless steel construction and provided with overpressure protection. Level controls divert coolant to the WPS on high level or operate coolant makeup valves on low level.

With respect to quality control (QC) the CVCS volume control tanks, purchased using Combustion Engineering, Inc. (CEs) generic specification WQC-11.1, Level II, required the following:

a. The manufacturer was required to maintain a quality assurance system acceptable to CE. This system included inspection and testing procedures, a manufacturing and QC plant, control of procedure revisions and control and submittal of documents and records.

b. The manufacturer was required to have written procedures which ensured the latest applicable drawings, specifications and instructions were used for fabrication, inspections and tests and ensured control over all measuring and testing equipment.

c. The manufacturer was responsible for assuring that all supplies and services procured from his suppliers (sub-contractors and vendors) conformed to the contract requirements.

d. The QC program of the manufacturer was required to ensure that raw material to be used in fabrication or processing products conformed to the applicable physical, chemical and all other technical requirements. The identification of all material was maintained throughout all operations by job number, lot number, heat number or any other suitable identification means and recorded on proper inspection records for each component.

e. The manufacturer was informed that all processing, testing and insertion operations taking place in the supplier's or subcontractor's facilities were subject to CE/Baltimore Gas and Electric Company (BGE) quality surveillance and verification.

<u>Charging Pumps</u>

Three positive displacement charging pumps (Table 9-9) supply makeup water to the RCS. The pumps return coolant to the RCS. On a SIAS, all three pumps are started and discharge concentrated boric acid into the RCS. All wetted parts, except seals, are of stainless steel and titanium. The charging pumps have a design flow of 44 gpm each.

<u>Concentrated Boric Acid Tanks</u>

Each of the two concentrated boric acid tanks (Table 9-10) stores enough concentrated boric acid solution to bring the reactor to a cold shutdown condition at any time during the core lifetime. The solution is either prepared in the boric acid batching tank and flows through the boric acid batching strainer before entering the storage tanks or is obtained from the boron recovery system. The combined capacity of the tanks will also be sufficient to bring the coolant to refueling concentration. The tanks have duplicate electric heaters to maintain a temperature above the saturation temperature of the concentrated solution, and sampling connections are used to verify that proper concentration is maintained. The tanks are constructed of stainless steel.

Boric Acid Pumps

The two boric acid pumps (Table 9-11) supply concentrated boric acid solution through the boric acid strainer (Table 9-11) to the makeup system where the boric acid may be diluted with demineralized water. On receipt of SIAS, these pumps line up with the charging pumps to permit direct introduction of concentrated boric acid into the RCS. Each is capable of supplying boric acid at the maximum demand conditions. Wetted parts of the pumps are stainless steel.

Process Radiation Monitor

The process radiation monitor (Table 9-13) continuously measures the activity of the reactor coolant and actuates an alarm in the Control Room if a predetermined activity level is reached. The sensor is a gross-gamma plus specific isotope (I-135) monitor; the system is designed to detect activity release from the fuel to the reactor coolant within five minutes of the event.

9.1.3.2 Codes and Standards

All components are designed, manufactured, tested and inspected according to applicable codes. The following code classifications apply to the CVCS components:

Regenerative Heat Exchanger	ASME III Class C[a]
Letdown Heat Exchanger	ASME III Class C
Deborating Purification Demineralizers	ASME III Class C
Purification Filters	ASME III Class C
Volume Control Tank	ASME III Class C
Boric Acid Storage Tanks	ASME III Class C

[a] The regenerative heat exchanger is built as a Class A vessel, but is stamped Class C.

9.1.3.3 Testing and Inspection

Each component is inspected and cleaned prior to installation into the system. Demineralized water will be used to flush each system.

Instruments will be calibrated during testing. Automatic controls will be tested for actuation at the proper setpoints. Alarm functions will be checked for operability and limits during preoperational testing. The relief valve setpoints will be checked.

The system will be operated and tested initially with regard to flow paths, flow capacity and mechanical operability. At least one pump of each type will be tested to demonstrate head and capacity.

Data will be taken periodically during normal plant operation to confirm heat transfer capabilities and purification efficiency.

9.1.4 SYSTEM OPERATION

9.1.4.1 Startup

During startup, the plant is brought from cold shutdown to hot standby at normal operating pressure and zero power temperature, before the reactor is brought critical. While the coolant is being heated, and until the pressurizer steam bubble is established, the charging pumps and letdown backpressure valve are used to maintain pressure in the RCS. After a steam bubble is established in the

pressurizer, the operator adjusts the pressurizer water level manually with the letdown backpressure and letdown control valves. The level controls of the VCT automatically divert the letdown flow to the WPS.

While the reactor is shut down, the VCT can be vented to the WPS. Prior to startup, the tank is purged with nitrogen to remove air. After purging is completed, the vent is secured and a nitrogen-hydrogen blanket is established in the tank. Any oxygen in the reactor coolant is normally removed by radiolytic recombination with excess hydrogen in the coolant. However, should the residual radiation from the core be insufficient to reduce the oxygen level, hydrazine can be added to scavenge the oxygen if the temperature is below 250°F.

Throughout startup, one purification filter is in service to reduce the activity of wastes entering the WPS. When the letdown temperature is stabilized at the desired RCS hot standby temperature, one or more purification ion exchangers are put into service as required.

Within limitations placed on the shutdown margin, the boric acid concentration may be reduced during heatup; however, the shutdown group of CEAs must be in the fully-withdrawn position before the operator may start reducing the concentration of boric acid in the RCS. The operator may inject a predetermined amount of demineralized makeup water by operating the system in the makeup controller "Dilute" mode. The concentration of boric acid in the reactor coolant is determined by chemical analysis.

9.1.4.2 Normal Operation

Normal operation includes operating the reactor both at hot standby and when it is generating power, with the RCS at normal operating pressure and temperature.

During normal operation:

a. Level instrumentation on the pressurizer automatically controls the volume of water in the reactor system by adjusting the letdown flow.
b. The VCT level is increased manually by the operator using makeup and automatically decreased by diversion to the WPS. Level can also be controlled by automatic makeup.
c. The hydrogen concentration and pH of the coolant are adjusted.
d. Changes in reactivity may be compensated for by adjusting the concentration of boric acid in the reactor coolant. Throughout most of the cycle, changes in boron concentration are effected by feed-and-bleed, discharging the excess coolant to the WPS. Late in cycle life, the dissolved boron in the reactor coolant is maintained at a very low concentration; at this time, feed-and-bleed generates excessive radioactive wastes; further reduction is accomplished by use of a purification ion exchanger with deborating resin. The makeup system may be operated in four modes:
    1. In the "Dilute" mode, a quantity of demineralized makeup water is selected and introduced into the VCT or directly to the charging pump suction header at a preset rate. When the integrating flowmeter indicates that the selected quantity of makeup water has been added, the flow is automatically terminated.
    2. In the "Borate" mode, a quantity of concentrated boric acid is selected and introduced at a preset rate as described above.
    3. In the "Manual" mode, the flows of the demineralized water and concentrated boric acid are set for any blend concentration between

demineralized makeup water and concentrated boric acid. This mode is primarily used to supply the VCT. It is also used for positive reactivity control during power operation.

4. In the "Automatic" mode, the flow rates of the demineralized water and concentrated boric acid are set to achieve the concentration present in the reactor coolant. The solution is automatically blended and introduced into the VCT according to signals received from the VCT level program.

e. The letdown flow is routed through one of the purification ion exchangers to reduce coolant activity resulting from soluble and insoluble corrosion and fission products.

9.1.4.3 Cooldown

Plant cooldown is accomplished by a series of operations which bring the reactor plant from hot standby condition at normal operating pressure and zero power temperature, to a cold shutdown.

Before the plant is cooled down, the VCT is vented to the WPS to reduce the activity and the hydrogen concentration in the reactor coolant. The operator may also increase the letdown flow rate to accelerate degasification, ion exchange and filtration of the reactor coolant. The operator increases the concentration of boric acid in the reactor coolant to ensure that the reactor has an adequate shutdown margin throughout its period of cooldown.

During cooldown, makeup water is introduced at the shutdown boric acid concentration. When the CVCS makeup system is in the automatic mode, a preset boric acid solution is automatically blended and introduced into the VCT upon demand from the VCT level program. The preset solution concentration corresponding to the desired shutdown concentration will have been previously determined and selected on the blender switch by the operator. During the cooldown, the charging pumps and letdown control valves are used to adjust and maintain the pressurizer water level. High charging flow results in a low level in the VCT which sounds an alarm. The operator then manually makes up fluid volume at the preselected shutdown boric acid concentration.

The estimated dissolved boron in the reactor coolant required to maintain cold shutdown conditions is shown in curves found in the Nuclear Engineering Operating Procedure (NEOP).

The total volume of both concentrated boric acid storage tanks is also sufficient to bring the RCS to refueling concentration.

A portion of the charging flow is used as an auxiliary spray to cool the pressurizer when the pressure of the RCS is below that required to operate the RCPs.

9.1.4.4 Safety Injection

Under event conditions, the charging pumps are used to inject concentrated boric acid into the RCS. Either the pressurizer level control system or SIAS will automatically start all charging pumps. The SIAS will also function to transfer the charging pump suction from the VCT to the discharge of the boric acid pump. If the boric acid pumps are not operable, boric acid flows by gravity from the concentrated boric acid tanks to the charging pump suction header. If the charging line inside the reactor containment building is inoperative, the line may be isolated outside the reactor containment, and the concentrated boric acid solution

may be injected by the charging pumps through the high-pressure safety injection (HPSI) piping.

9.1.5 DESIGN EVALUATION

To assure reliability, the design of the CVCS incorporates redundant critical components to reduce dependence upon any single critical component. Redundancy is provided as follows:

Component	Redundancy
Purification Demineralizer	Parallel Standby Unit
Purification Filters	Parallel Standby Unit
Charging Pump	Two Parallel Standby Units
Letdown Flow Control	Parallel Standby Valve
Letdown Backpressure Regulator	Parallel Standby Valve
Boric Acid Pump and Tank	Parallel Standby Unit

The charging and boric acid pumps are powered by the diesel generators if normal power sources are lost. One charging pump and one boric acid pump are supplied from each emergency bus. The third charging pump may be supplied from either emergency bus. Physical separation and barriers are provided between the power and control circuits for the redundant pumps.

Standby features are provided so that at least one charging pump is running after SIAS. If two diesel generators are available, both boric acid pumps will be running. The charging pumps and boric acid pumps may be controlled locally at their switchgear. Separate power supplies for pump power and separate control circuits assure that this system satisfies the single failure criterion.

The boric acid solution is stored in heated and insulated tanks and is piped in heat-traced and insulated lines to preclude precipitation of the boric acid. Two independent and redundant heating systems are provided for the boric acid tanks and lines. Low temperature alarms and automatic temperature controls are included in the heating system. If the boric acid pumps are not available, boric acid from the concentrated boric acid tanks may be gravity fed into the charging pump suction. If the charging line inside the reactor containment building is inoperative, the charging pump discharge may be routed via the SI system to inject concentrated boric acid into the RCS.

9.1.6 REFERENCES

1. Bechtel Specification 6750-M-0310C, "Design Specification for Piping, Valves, and Associated Equipment of the Letdown System"
2. Bechtel Specification 6750-M-0310D, "Design Specification for Piping, Valves, and Associated Equipment of the Charging System and Auxiliary Spray System"

TABLE 9-1

CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS

Normal Letdown and Purification Flow, gpm[a]	38
Normal Charging Flow, gpm	44
Reactor Coolant Pump Controlled Bleedoff (4 pumps), gpm[a]	6
Normal Letdown Temperature at Loop °F	548
Ion Exchanger Operating Temperature, °F	120

[a] The original design for normal letdown and purification flow rate was 40 gpm with 4 gpm of reactor coolant pump controlled bleedoff. (The maximum letdown and purification flow rate was 128 gpm with 4 gpm controlled bleedoff.) These flows combined to equal the normal charging flow rate of 44 gpm (maximum charging flow rate of 132 gpm). Facility Change Request (FCR) 87-0074 replaced the reactor coolant pump seal with a state-of-the-art design that requires 1.5 gpm (nominal) controlled bleedoff per pump for stable operation. The net effect of this FCR was to reduce the normal letdown and purification flow rate to 38 gpm (the maximum letdown and purification flow rate was reduced to 126 gpm) and increase the reactor coolant pump controlled bleedoff rate to 6 gpm (total for all four pumps).

TABLE 9-3

REGENERATIVE HEAT EXCHANGER

DESIGN PARAMETERS

Quantity	1
Type	Shell and Tube, Vertical
Code	ASME III, Class C[(a)]
Tube Side (Letdown) Fluid	Reactor Coolant, 1.5 wt% Boric Acid, Maximum
Design Pressure, psig	2485
Design Temperature, °F	650
Materials	Stainless Steel, Type 304
Pressure Loss at 63,500 lb/hr	99
Shell Side (Charging) Fluid	Reactor Coolant, 6.25 wt% Boric Acid, Maximum
Design Pressure, psia	3025
Design Temperature, °F	650
Materials	Stainless Steel, Type 304
Pressure Loss at 132 gpm, psi	70

(a) The regenerative heat exchanger is built as a Class A vessel, but is stamped Class C.

DESIGN OPERATING PARAMETERS - REGENERATIVE HEAT EXCHANGER

TUBE SIDE (LETDOWN)	NORMAL	MAXIMUM UNBALANCED CHARGING WITH HEAT TRANSFER	MAXIMUM PURIFICATION	MAXIMUM UNBALANCED LETDOWN
Flow – gpm [Table 9-1 note (a)]	38	30	126	126
Inlet Temp. - °F	548	548	548	548
Outlet Temp. - °F	232	143	350	433
Shell Side (Charging)				
Flow – gpm	44	132	132	44
Inlet Temp. - °F	120	120	120	120
Outlet Temp. - °F	415	220	324	475

TABLE 9-4

LETDOWN CONTROL VALVES

Quantity	2
Design Pressure, psia	2500
Design Temperature, °F	650
Flow, each	
Maximum, gpm	128
Minimum, gpm	29

TABLE 9-5

LETDOWN HEAT EXCHANGER

DESIGN PARAMETER

Quantity	1
Type	Shell and Tube, Horizontal
Code	ASME III, Class C
Tube Side (Letdown) Fluid	Reactor Coolant, 1.5 wt% Boric Acid, Maximum
Design Pressure, psig	650
Design Temperature, °F	550
Pressure Loss at 63,500 lb/hr, psi	52
Materials	Stainless Steel, Type 304
Shell Side (Cooling Water)	
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, °F	250
Materials	Carbon Steel
Design Flow, lb/hr	594,390

DESIGN OPERATING PARAMETERS

TUBE SIDE (LETDOWN)	NORMAL	MAXIMUM UNBALANCED CHARGING WITH WITH LETDOWN	MAXIMUM PURIFICATION	MAXIMUM UNBALANCED LETDOWN
Flow - gpm [Table 9-1 note (a)]	38	30	126	126
Inlet Temp. - °F	232	143	350	433
Outlet Temp. - °F	120	120	125	135
Shell Side (Cooling Water)				
Flow - gpm	157	21	1200[a]	1200[a]
Inlet Temp. - °F	95	65	95	95
Outlet Temp. - °F	122	128	118	127

[a] This flowrate represents design points selected during the design of the system and components. It does not indicate the normal operating point or minimum or maximum limitation of the system or components.

TABLE 9-6

ION EXCHANGERS

Quantity	3
Type	Flushable
Design Pressure, psig	200
Design Temperature, °F	250
Normal Operating Pressure, psig	60
Normal Operating Temperature, °F	120
Resin Volume, ft^3, each	36
Normal Flow, gpm	38
Maximum Flow, gpm	128 [Table 9-1, note (a)]
Code for Vessel	ASME III, Class C
Material	ASME SA 240, Type 304
Fluid, wt% Boric Acid, Maximum	1.5

TABLE 9-7

PURIFICATION FILTERS

Quantity	2
Type of Elements	Single Element Disposable Cartridge
Filter Rating, microns (absolute)	0.1 to 6.0, various, depending on plant conditions
Vessel Design Pressure, psig	200
Vessel Design Temperature, °F	250
Design Flow, gpm	128 (Table 9-1, note (a)]
Normal Flow, gpm	38
Code for Vessel	ASME III, Class C
Material	Austenitic Stainless Steel
Fluid, wt% Boric Acid, Maximum	1.5

TABLE 9-8

VOLUME CONTROL TANK

Quantity	1
Type	Vertical, Cylindrical
Design Pressure, Internal, psig	75
Design Pressure, External, psig	15
Design Temperature, °F	250
Operating Pressure Range, psig	0 to 65
Normal Operating Pressure, psig	25 to 50
Normal Operating Temperature, °F	120
Normal Spray Flow, gpm	38
Blanket Gas	Hydrogen and/or Nitrogen
Code	ASME III, Class C
Fluid, wt% Boric Acid, Maximum	12
Material	Austenitic Stainless Steel

TABLE 9-9
CHARGING PUMPS

Quantity	3
Type	Positive Displacement
Design Pressure, psig	2735
Design Temperature, °F	250
Capacity, gpm	44
Normal Discharge Pressure, psig	2311
Normal Suction Pressure, psig	50
Normal Temperature of Pumped Fluid, °F	120
Maximum Discharge Pressure (Short Term), psig	3010
Minimum NPSH, psia	9
Driver Rating, hp	100
Materials in Contact with Pumped Fluid	Stainless Steel, Titanium
Fluid, wt% Boric Acid, Maximum	12

TABLE 9-10

CONCENTRATED BORIC ACID PREPARATION AND STORAGE

CONCENTRATED BORIC ACID TANKS

Quantity	2
Internal Volume, ft^3	1270
Design Pressure, psig	15
Design Temperature, °F	200
Normal Operating Temperature, °F	150
Type Heater	Duplicate Electrical, Strap-on heaters
Fluid, wt% Boric Acid, Maximum	12
Material	Stainless Steel
Code	ASME III, Class C

BORIC ACID BATCHING STRAINER

Quantity	1
Type	Basket
Design Pressure, psig	150
Design Temperature, °F	200
Screen Size, US Mesh	80
Design Flow, gpm	50
Materials	Stainless Steel
Fluid, wt% Boric Acid, Maximum	12

BORIC ACID BATCHING TANK

Quantity	1
Useful Volume, ft^3	67
Design Pressure	Atmospheric
Design Temperature, °F	200
Normal Operating Temperature, °F	150
Type Heater	Electrical Immersion
Heater Capacity, min, kW	45
Fluid, wt% Boric Acid, Maximum	12
Material	Austenitic Stainless Steel

TABLE 9-11

BORIC ACID PUMPS AND STRAINER

PUMPS	
Quantity	2
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	250
Design Head, ft	231
Design Flow, gpm	143
Normal Operating Temperature, °F	150
NPSH Required, ft	20
Horsepower	25
Fluid, wt% Boric Acid, Maximum	12
Material in Contact With Liquid	Stainless Steel
STRAINER	
Quantity	1
Type	Basket
Screen Size US Mh	80
Design Pressure, psig	150
Design Temperature, °F	200
Design Flow, gpm	140
Materials	Austenitic Stainless Steel
Liquid, wt% Boric Acid, Maximum	12

TABLE 9-13

PROCESS RADIATION MONITOR

Quantity	1
Design Pressure, psig	200
Design Temperature, °F	250
Normal Operating Pressure, psig	80
Normal Operating Temperature, °F	120
Normal Flow Rate, gpm	0.5
Measurement Range, μCi/cc I-135	10^{-4} to 100[a]
Measurement Range, cpm	10 to 10^6

[a] Upper measurement range of 100 μCi/cc is based upon use of a collimator between detector and sample.

9.2 SHUTDOWN COOLING SYSTEM

9.2.1 DESIGN BASIS

The Shutdown Cooling (SDC) System is used to remove core decay heat and reactor coolant sensible heat during plant cooldowns and cold shutdowns. The system also cools the containment spray water during Containment Spray System (CSS) operation following a Recirculation Actuation Signal (RAS) and maintains RCS temperature during refueling operations. Additionally, the heat exchangers can be used to provide additional spent fuel pool cooling (SFPC) when the complete core is removed from the reactor vessel and temporarily stored in the spent fuel pool (SFP).

Portions of the SDC System piping are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the SDC System fatigue analysis considers 500 initiations of SDC. In this transient, 300°F water from the RCS is injected into the SDC piping which is at a higher temperature due to the residual effects of the RCS normal operating temperatures. See Reference 1 for further details.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

9.2.2 SYSTEM DESCRIPTION

The SDC system is shown schematically in Figure 9-5. The system uses portions of other systems, i.e., the RCS (Section 4.1) and the engineered safeguards (Sections 6.3 and 6.4).

In the SDC mode of operation, reactor coolant is circulated through the tube side of the SDC heat exchanger using the low-pressure safety injection (LPSI) pumps. The flow path from the pump discharge runs through normally locked-closed valve, SI 658, through the shutdown cooling heat exchangers, and through normally locked-closed valve, SI 657, to the LPSI header, and enters the RCS through the four safety injection nozzles. The circulating fluid flows through the core and is returned from the RCS through the SDC nozzle in the loop No. 2 reactor vessel outlet (hot leg) pipe. The coolant is returned to the suction of the LPSI pumps through normally locked-closed valves SI-651 and SI-652.

In Mode 5, 6, or defueled, a containment spray pump may be used to circulate reactor coolant through the tube side of the SDC heat exchangers. The appropriate valve lineup and plant operating conditions are specified in the Operating Procedures.

During CSS operation, prior to recirculation, RWT inventory passes through the tube side of the SDC heat exchanger via containment spray pumps for containment cooling purposes. After recirculation, the containment spray pumps switch suction from the RWT to the containment sump and sump water is circulated through the SDC heat exchangers.

In both the SDC mode of operation and during CSS operation, component cooling (CC) water flows through the shell side of the SDC heat exchangers. During shutdown cooling, CC cools reactor coolant and during containment spray operation after RAS, CC cools the containment sump fluid. Prior to RAS, CC is not needed for cooling purposes because RWT water does not need cooling. Also note that prior to RAS, CC is not cooled by Saltwater, so CC could provide no cooling.

Shutdown cooling and total low-pressure injection flow are measured by a flow element installed in the low-pressure injection header. Flow is indicated in the Control Room. The flow element also transmits a signal to a controller which will provide automatic flow control during SDC operation.

9.2.3 SYSTEM COMPONENTS

The SDC system is made up of portions of the SI system and the RCS. The principal characteristics of the major components in those systems are given in Sections 6.3 and 4.1, respectively.

Each component is inspected and cleaned prior to installation. Demineralized water is used to flush each system. Initially the system is operated and tested to verify that the flow path, flow, thermal capacity and mechanical operability meet the design requirements. Instruments are calibrated during testing. The automatic flow control is tested.

Periodic testing of the LPSI pumps, as described in Section 6.3.6, assures the availability of this equipment for shutdown cooling. Data can be taken during refueling operations to confirm heat transfer capacity.

9.2.4 SYSTEM OPERATION

During normal plant operation, there are no components of the system in operation. All components are on standby for possible emergency operation, as a part of the CSS and SI system. The SDC capability may be used during the early stages of plant startup to control the reactor coolant temperature. As the coolant temperature approaches 300°F and the pressure approaches 270 psig, this method of control must be discontinued and the system aligned for emergency operation.

Following reactor shutdown and cooldown, the system is operated in the shutdown mode for further cooling of the RCS when the coolant temperature falls below 300°F and the coolant pressure falls below 270 psig. At this time, the system must be manually realigned for shutdown cooling. Prior to placing the system in operation, the boron concentration is verified at various points in the system. During the early stages of shutdown cooling, the cooldown rate is controlled by limiting the flow through the tube side of the heat exchanger. Constant flow through the core is maintained by using valve SI 306 as a heat exchanger bypass valve.

During Mode 6 operation, the SDC system serves to remove decay heat and other residual heat from the RCS, provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchangers where its heat is transferred to the component cooling water system which is, in turn, cooled by the Saltwater System. Therefore, in Mode 6, an OPERABLE SDC loop requires the support of a functional component cooling and saltwater subsystem. In Mode 6 conditions where both SDC loops are required to be operable (water level < 23 feet above the tops of the irradiated fuel assemblies seated in the reactor vessel), only one functional component cooling and one saltwater subsystem is necessary, provided their heat removal capacity is sufficient.

9.2.5 SYSTEM PERFORMANCE

The SDC system is designed to reduce the temperature of the reactor coolant at the controlled cooldown rate from 300°F to refueling temperature (≤ 140°F) within 36 hours

after shutdown. This assumes both CC pumps, both CC heat exchangers, and both SDC heat exchangers are on line and CC reaches a maximum of 120°F. This further assumes that each LPSI is circulating 3000 gpm and each CC pump is circulating 2500 gpm through the SDC heat exchanger (Section 6.3.2.5). Cooldown will occur more or less rapidly depending on pump and heat exchanger availability and component cooling loads on line.

The SDC system is designed to cool containment spray flow in order to bring containment temperature down to 120°F within 30 days following an accident. This assumes minimum safeguards: one train of containment spray, one train of safety injection, and one train of containment air coolers. This also assumes a CC flow of 1800 gpm and a containment spray flow of 1250 gpm.

9.2.6 DESIGN EVALUATION, AVAILABILITY, AND RELIABILITY

During normal cooldown the system utilizes the LPSI pumps to circulate the reactor coolant through the two SDC heat exchangers, returning it to the RCS through the LPSI header. Cooldown rate is controlled by adjusting the flow through the heat exchangers. Both heat exchangers are required to achieve cooldown at the maximum design rate. One exchanger provides cooldown capability at a reduced rate.

Control valves which were originally equipped with two sets of packing and intermediate leakoff connections that discharged to the WPS were repacked with Chesterton packing. Valves that are repacked with Chesterton packing have one set of packing. These valves may have their leakoff lines removed and the valve leakoff connection plugged or tubing capped. Some manual valves have backseats to facilitate repacking.

All piping in the SDC system is austenitic stainless steel. The piping is welded except for flanged connections at the pumps and components, which can be removed for maintenance.

During plant operation, double valves with a relief valve between the two valves, isolate the suction of both LPSI pumps from the RCS. These two valves, SI-651 and 652, are key-locked closed at the control board during plant operation. Additionally, the valve between the SDC heat exchangers and the LPSI header, SI-657, is locked shut during plant operation, both locally and at the control board. The keys are kept under administrative control to ensure that these valves cannot be opened inadvertently during plant operation.

Pressurizer pressure instrument channels P-103 and P-103-1 each provide an open permissive interlock to the two LPSI pump suction isolation valves SI-651 and 652, respectively. These independent and redundant interlocks prevent opening of these valves whenever the RCS is already pressurized at or above the SDC System design pressure. The suction piping to the LPSI pumps is the SDC System component with the limiting design pressure rating.

During SDC System operation, a visual and audible alarm on the main control board is activated whenever either SI-651 or 652 are not fully closed and RCS pressure is above the SDC System design pressure. These two separate alarms are tested at each refueling outage to ensure reliability and are designed to alert the operator in the event of an alarm or control circuit power supply failure. These alarms, associated procedural controls and operator training ensures a high probability of achieving double isolation of the SDC System from the RCS when the RCS pressure is raised above the SDC System design pressure.

The suction isolation valves, SI-651 and SI-652, and associated control system design, therefore, provide two independent and redundant means for achieving and maintaining isolation of the SDC System from the RCS.

Overpressure protection of the SDC System is provided by relief valve RV-468, which is located on the SDC Return Header downstream of 1(2)-MOV-651. This valve is sized to protect the SDC flowpath from overpressure due to the simultaneous operation of three charging pumps while on SDC with the pressurizer in a solid condition.

Certain transients in the RCS, such as an inadvertent RCP or HPSI pump start, can cause a pressure transient that exceeds the capacity of RV-468. However, these transients are prevented or mitigated by the LTOP controls outlined in Section 4.2.2 and in the Technical Specifications. The LTOP controls are in place at all times when both SDC is in operation and a pressurization event is possible (e.g., until the RCS is vented to at least 8 square inches).

9.2.7 OPERATION AT REDUCED INVENTORY

Generic Letter 88-17, Loss of Decay Heat Removal, described concerns and recommended actions for operation of the RCS and the SDC System during reduced inventory conditions. Reduced inventory is defined by the generic letter as an RCS inventory which results in a reactor vessel level lower than 3' below the vessel head flange. Three key areas were addressed in the resolution of this issue: (1) Prevention of a loss of decay heat removal; (2) In-depth mitigation of a loss of decay heat removal; and (3) Providing a closed containment before the core uncovers if a loss of decay heat removal occurs. These three areas were addressed in responses to the recommendations made in the generic letter. The recommendations and their responses are provided below:

a. Provide reliable indication of parameters that describe the state of the RCS and the performance of systems normally used to cool the RCS for both normal and accident conditions. At a minimum, provide the following in the Control Room:

    1. Two independent RCS level indications.

       We have at least two independent, continuous level indications and audible alarms. (Section 7.5.9.4)

    2. At least two independent temperature measurements representative of the core exit whenever the reactor vessel head is located on top of the reactor vessel.

       We have two independent, continuous coolant temperature indications that are representative of the core exit conditions. (Section 7.5.9.4)

    3. The capability of continuously monitoring decay heat removal system performance whenever the system is used for cooling the RCS.

       We have instrumentation to monitor SDC pump suction pressure, discharge pressure, motor current, system flow and RCS level. (Section 7.5.9.4)

    4. Visible and audible indications of abnormal conditions in temperature, level and decay heat removal system performance.

       We have visible and audible indications in the Control Room for temperature, level and SDC performance. (Section 7.5.9.4)

b. Develop and implement procedures that cover reduced inventory operation and that provide an adequate basis for entry into a reduced inventory condition. Procedures should cover normal and off-normal operation of the Nuclear Steam

Supply System during times when cooling is normally provided by the SDC System.

There are procedures and administrative controls which cover normal and off-normal operation of the RCS, SDC, supporting systems and containment. Additionally, we have implemented controls that ensure the status of each containment penetration required for containment closure is known, and the time and method of closure has been addressed for those penetrations which are open. The definition of containment closure is consistent with Technical Specifications.

c. Ensure adequate equipment is operating, operable and/or available to provide cooling for the RCS. Adequate equipment must remain operable or available to mitigate a loss of SDC. In addition, adequate equipment must be provided for personnel communications for activities necessary to maintain the RCS in a stable and controlled condition.

   Normally, the SDC System provides cooling for the RCS. Technical Specifications require that the SDC System remain operable. The HPSI pump and one other means remain available to mitigate the consequences of a loss of SDC. The normal plant paging system provides communication capability onsite. Any page can be overridden by a Control Room page which is simultaneously broadcast to all zones.

d. Conduct analyses to supplement existing information and develop a basis for procedures, instrument and installation and response, and equipment/Nuclear Steam Supply System interactions and response. The analysis should encompass thermodynamic and physical conditions, and should emphasize complete understanding of Nuclear Steam Supply System behavior.

   Analyses have been performed and include: time to reach saturated conditions; peak pressurization of the RCS based on reactor vessel head vent paths; times to reach core uncovery for a variety of conditions, assuming no operator action; effects of steam generator nozzle dam installation; instrument uncertainties; and analyses of flow paths as a function of RCS back pressure. Containment response and airborne activity analyses were also performed. All calculations are dependent upon initial conditions, such as: RCS heat sinks; level; temperature; vent paths; and time after shutdown.

e. The Technical Specifications should not restrict the safety benefit of actions identified under Generic Letter 88-17.

   We reviewed our Technical Specifications and made the necessary changes.

   Subsequent to the original Generic Letter 88-17 responses, a containment outage door was installed at the exterior of the equipment hatch opening as an additional programmed enhancement to provide for closure during Mode 5 and 6 conditions. The containment outage door is designed to mitigate the offsite radiological consequences of a fuel handling incident and a loss of shutdown cooling incident. Because it can be opened and shut more quickly than the equipment hatch, the containment outage door increases the availability of the equipment hatch opening for access to the Containment during unit outages.

9.2.8 REFERENCES

1. Bechtel Specification 6750-M-0310A, "Design Specification for Piping, Valves, and Associated Equipment of the Shutdown Cooling System"

9.3 <u>CIRCULATING WATER SYSTEM</u>

9.3.1 DESIGN BASIS

The condensers for both of the electrical generating units have been designed such that the increase in temperature of the Chesapeake Bay water passing through them is not more than 10°F at maximum expected, not guaranteed, operating conditions. Under guaranteed operating conditions, i.e., that maximum operating condition at which both the reactor supplier and the turbine generator suppliers guarantee their equipment, the temperature increase of the Bay water is no more than 9.6°F. A test program allowing a temperature increase up to 12°F has been completed by the State of Maryland. Current limits allow use of a 12°F increase on a permanent basis.

Circulating water pumps and piping conduits are designed to fulfill the design basis requirements described above.

9.3.2 SYSTEM DESCRIPTION

The circulating water system (CWS) is shown in Figures 9-8 (Unit 1) and 9-26 (Unit 2). The intake and discharge are shown on Figures 1-3A and 1-3B.

9.3.2.1 <u>Circulating Water System</u>

The CWS incorporates design information developed from the model testing discussed in Section 2.8.2. The full width of the intake channel (which serves both units) is 560'. The Intake Structure houses a total of 24 circulating water screens, 12 for each unit, consisting of single-flow or dual-flow designs. The purpose of these screens is to prevent debris larger than 3/8" or 10 mm from passing into the circulating water pumps, condenser, saltwater pumps, and the saltwater-to-fresh-water heat exchangers. The Screen Wash Systems provide a high pressure spray to remove debris from the water screens. The screen wash systems consist of eight submersible screen wash pumps including two installed spares and two submersible trough wash pumps. The screen wash pumps serve four screens each and the trough wash pumps serve the trough for each unit.

The Circulating Water Chemical Addition System serves both the Unit 1 and Unit 2 Circulating Water Systems to minimize the marine fouling of piping and heat exchanger surfaces. This system has the ability to inject approved chemicals into each of the Circulating Water System intake and discharge conduits, as necessary.

The CWS has six vertical centrifugal pumps per unit. These pumps provide the motive force required to circulate bay water through the system and back into the bay.

9.3.2.2 <u>Condensers</u>

The condensers for each unit consist of three separate shells, each with the same capacity, to condense exhaust steam from the power generating turbine. The condensers are of the single-pass or once-through design with divided water boxes to permit one-half of each shell to be opened and manually cleaned during plant operation, if necessary.

Each condenser has approximately 49,500 tubes, each being 1-1/4" in diameter and 28' long. The tube material is austenitic stainless steel and titanium (Unit 1), and titanium (Unit 2).

Each condenser is equipped with a mechanical cleaning system utilizing small sponge rubber balls which are injected at the condenser inlet, passed through the tubes, collected at the condenser outlet, and returned for recycling.

A butterfly valve equipped with a perforated disc instead of a solid disc is installed in each circulating water pipe at the inlet to the condenser water boxes. It is possible to close this valve when its corresponding circulating water pump is shut down. Marine growth that may have been pumped against the condenser tube sheet should fall off and be caught by this strainer-type valve instead of falling further down and out of reach in the pipes. Conveniently located manhole doors can then be opened and the marine growth manually removed from the condenser.

Two temperature sensors are provided in each discharge pipe. These temperatures are monitored by the computer in the Control Room. Since each unit has six discharge pipes, circulating water discharge is monitored by twelve independent temperature readings prior to being discharged into the bay.

9.3.2.3 Bay Water Systems Discharges

At mean low tide level, the top of the discharge conduits is approximately 6 feet below the surface of the water. The entire discharge structure is composed of four separate conduits, two for each unit.

The effluent from the WPS may be discharged into any one or more of these four separate conduits, thus providing a means to ensure a maximum dilution of the effluent under all operating conditions. Discharge from the condensate system, the steam generator blowdown recovery system, the storm water system, the yard oil interceptor, and the auxiliary blowdown tank are also directed to the discharge conduits.

9.3.3 COMPONENTS

The component description for the CWS is contained in Table 9-15.

9.3.4 TESTING AND INSPECTION

Each component is inspected and cleaned prior to installation into the system.

Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability and limits during preoperational testing. The relief valve setpoints are checked.

The system was operated and tested initially with regard to flow paths, flow capacity and mechanical operability.

Data will be taken periodically during normal plant operation to confirm heat transfer capabilities.

9.3.5 SYSTEM RELIABILITY

The CWS is similar to other systems operating in conventional and nuclear power plants. The equipment in this system is designed to applicable codes and standards as listed in Table 9-15. Adequate redundancy, protective devices, and controls are provided to assure reliable and safe operation.

TABLE 9-15

CIRCULATING WATER SYSTEM COMPONENT DESCRIPTION

Traveling Water Screen	
Types	Vertical, single flow through and dual flow through
Quantity	12 (per unit)
Speed (ft/min)	10 (single flow through) 16.5/50 (dual flow through)
Temperature range (F)	0 – 100
Screen Wash Pumps	
Type	Submersible, vertical, centrifugal
Quantity	3 per unit and 2 spares
Capacity each (gpm)	1560
Head (ft)	220
Motor	150 hp, 460 Volt, 3 phase, 60 Hz, 1700 RPM
Codes	National Electrical Manufacturers Association (NEMA), Standards of the Hydraulic Institute, ASME Boiler and Pressure Vessel (B&PV) Codes, Section VIII ANSI B16.5
Trash Trough Wash Pumps	
Type	Submersible, vertical, centrifugal
Quantity	1 per unit
Capacity (gpm)	400
Head (ft)	25
Motor	6.2 hp, 460 Volt, 3 phase, 60 Hz, 1700 RPM
Code	NEMA; Standards of the Hydraulic Institute; ASME B&PV Codes, Section VIII, ANSI B16.5
Circulating Water Pumps	
Type	Vertical, dry pit
Quantity	6 (per unit)
Capacity each (gpm)	200,000
Head (ft)	19.5
Motor	Synchronous 1250 hp, 4160 Volt, 60 Hz, 3 phase, 150 RPM
Codes	NEMA, Standards of the Hydraulic Institute, ASME B&PV Code, Section VIII, Pressure Vessels, ANSI B16.5

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 DESIGN BASIS

The SFPC system is common to both units. The pool contains water with the proper dissolved concentration of boron and has the capacity to store 1830 fuel assemblies.

The SFPC system is designed to remove the maximum decay heat expected from 1613 fuel assemblies, not including a full core off-load. The maximum pool temperature in this case is 120°F. The system is also capable of being used in conjunction with the SDC system to remove the maximum expected decay heat load from 1830 fuel assemblies, including a full core discharge. The maximum SFP temperature in this case is 130°F.

The maximum decay heat load expected from 1613 fuel assemblies, not including a full core off-load, is a function of decay time. For a limiting decay time of 3.5 days, which results in an initial core alteration time of 3.0 days after reactor shutdown, the decay heat load is 22.33×10^6 Btu/hr. The fuel is assumed to have undergone steady-state burnup at 2738 MWt for an average of 1562.4 days for an 100 assembly batch reload. The total SFP decay heat load as a function of decay time is compared to the heat removal capacity from the two SFP heat exchangers as a function of SRW temperature to show what time after shutdown is acceptable for each SRW temperature condition to maintain the pool at a temperature of 120°F. A maximum SRW temperature of 65°F is required to support a minimum decay time of 3.5 days. In the event that one SFP cooling loop is lost, the remaining loop can remove the heat load while maintaining the pool temperature at 155°F.

The maximum decay heat rate for 1830 fuel assemblies stored in the SFP is a function of decay time. For a limiting decay time of 4.5 days, which results in an initial core alteration time of 3.0 days after reactor shutdown, the decay heat load is 45.96×10^6 Btu/hr based upon the following hypothetical sequence of events:

1. Eighty-four fuel assemblies are removed from Unit 1 after an average of 1860 days of reactor operation at 2738 MWt, and are replaced with fresh fuel. Unit 1 is then returned to full power.
2. Three-hundred-sixty-five days after the Unit 1 refueling, 84 fuel assemblies are removed from Unit 2 after an average of 1860 days of irradiation and are replaced with fresh fuel. Unit 2 is then returned to full power.
3. Three-hundred-sixty-five days after the Unit 2 refueling, 84 fuel assemblies are removed from Unit 1 after an average of 1860 days of irradiation and are replaced with fresh fuel. Unit 1 is then returned to full power.
4. This refueling cycle continues until the pool contains 1613 fuel assemblies at the end of a Unit 2 refueling. It has been conservatively assumed that the 67 oldest assemblies have been removed from the pool to allow for complete filling of the racks with newer fuel.
5. Unit 1 is then shutdown 60 days after the previous Unit 2 shutdown and the entire core is offloaded after a minimum of 4.5 days of decay. At this point, it is conservatively assumed that the fuel has completed its current cycle, and is therefore at maximum irradiation.

Upon completion of the last operation, the pool will contain 1830 fuel assemblies, with each discharge subjected to different periods of irradiation and decay, in accordance with the table below assuming the minimum decay time of 4.5 days:

	Number of Assemblies	Irradiation Period (Days)	Decay Period (Days)
a.	17	1860	6964.5
b.	84	1860	6599.5
c.	84	1860	6234.5
d.	84	1860	5869.5
e.	84	1860	5504.5
f.	84	1860	5139.5
g.	84	1860	4774.5
h.	84	1860	4409.5
i.	84	1860	4044.5
j.	84	1860	3679.5
k.	84	1860	3314.5
l.	84	1860	2949.5
m.	84	1860	2584.5
n.	84	1860	2219.5
o.	84	1860	1854.5
p.	84	1860	1489.5
q.	84	1860	1124.5
r.	84	1860	759.5
s.	84	1860	394.5
t.	84	1860	64.5
u.	217	1860	4.5

The total SFP decay heat load as a function of decay time is compared to the heat removal capacity from both loops of SFPC as a function of SRW temperature, supplemented with one loop of SDC to show what time after shutdown is acceptable for each SRW temperature condition to maintain the pool at a temperature at 130°F. A maximum SRW temperature of 75°F is required to support a minimum decay time of 4.5 days.

9.4.2 SYSTEM DESCRIPTION

The SFPC System shown in Table 9-16 and Figure 9-7 is a closed-loop system consisting of two half-capacity pumps and two half-capacity heat exchangers in parallel, a bypass filter that removes insoluble particulates, and a bypass demineralizer that removes soluble ions. The SFPC heat exchangers are cooled by service water (SRW).

Skimmers are provided in the SFP to remove accumulated dust from the pool. The clarity and purity of the water in the SFP, refueling pool, and the RWT are further maintained by passing a portion of the flow through the bypass filter and/or demineralizer. The SFP filter and demineralizer removes fission products from the cooling water in the event of a leaking fuel assembly.

Connections are provided for tie-in to the SDC system to provide for additional heat removal in the event that 1830 fuel assemblies are contained in the pool. When the pressure in the SDC system is greater than the design pressure of the SFPC system, the SFPC system is isolated from the SDC system via two manual isolation valves. Although not required by the design code, double valve isolation is provided at this system interface to meet the original FSAR design basis (FCR 90-87).

The entire SFPC system is tornado-protected and is located in a Seismic Category I structure. Borated makeup water comes from the RWT. Non-borated makeup water comes from the demineralized water system.

9.4.3 COMPONENTS

9.4.3.1 Functional Description

A description for the spent fuel pool cooling system is contained in Table 9-16.

9.4.3.2 Codes and Standards

The following codes and standards were used in the design of the SFPC System components:

Pump	Standards of: ASME (III, VIII, IX, PTC8.2), ASTM, NEMA, ANSI
Heat Exchanger	Standards of: Tubular Exchanger Manufacturers Association (TEMA), ASME (III, VIII, IX), ASTM, ANSI
Filter	ASME III C and ASME VIII paragraph UW-2(a)
Ion Exchanger	ASME III C and ASME VIII paragraph UW-2(a)
Valves, Piping, Fittings	ANSI B31.7 Class III

9.4.3.3 Tests and Inspections

Each component is cleaned and inspected before installation and the assembled systems flushed with demineralized water. The flow paths, flow capacity and mechanical operability are tested by operation. The head and capacity of the pumps are also tested.

Instruments are calibrated prior to tests. Alarm functions are checked for operability and limits during preoperational testing. During normal operation, periodic tests will be made to confirm design criteria.

9.4.4 SYSTEM OPERATION AND RELIABILITY

In the normal case (i.e., with no full-core off load), if one SFPC loop is lost, the remaining loop can remove decay heat while maintaining the pool temperature at 155°F. In the case of total loss of SFPC with 1613 fuel assemblies in the pool, it would take more than 8 hours to raise the pool temperature from 155°F to 210°F. The case of total loss of SFP cooling is only discussed to demonstrate the time available to take appropriate action in such an event to preclude boiling, and the resulting loss in pool water level. The design of the SFPC System and pool structural components (e.g., pool liner plate, SFPC piping and pumps) for total loss of cooling is not part of the system's design basis.

The most serious failure to the system is the loss of SFP water. This is avoided by routing all SFP piping connections above the water level and providing them with siphon breakers to prevent gravity drainage.

The SFP is designed to preclude the loss of structural integrity. Section 5.6.1 describes the analysis made to verify that the structural integrity cannot be impaired. Additional design and quality control requirements for the SFP are given in Section 6.3.5.1. However, if a leak from the SFP is postulated, the capabilities for controlling the leak are as follows:

Makeup water can be supplied indefinitely to the SFP at a rate of at least 150 gpm. It can usually be supplied at a greater rate for a period of many days, but this depends upon plant conditions. The makeup water flow path is as follows:

a. Source - Well water
b. Portable Makeup Demineralizers
   - Typical capacity 150 gpm or more
c. Demineralized Water Storage Tank
   - Storage capacity 350,000 gallons
d. Four Reactor Coolant Makeup Pumps (Normally run one per unit)
   - Capacity 165 gpm each, less the amount required for reactor coolant makeup
e. Two RWTs (One per unit)
   - Storage capacity 420,000 gallons
   - Required to have 400,000 gallons during operation
   - During refueling this water has been transferred to the refueling pool where it is also available for pumping if conditions permit
f. Two Spent Fuel Cooling Pumps (One per RWT)
   - Capacity 1390 gpm each
g. Spent Fuel Pool

The two halves of the SFP can be isolated from each other and 830 fuel assemblies, as a minimum, can be stored in the non-leaking half.

The four Emergency Core Cooling System (ECCS) equipment rooms on the lowest level of the Auxiliary Building (Figure 1-5) can be prevented from flooding by shutting their watertight doors. In addition, each ECCS pump room is also drained by an 80 gpm sump pump. The remainder of this level is drained by two sump pumps at a rate of 160 gpm. The sump pumps discharge to the Miscellaneous Waste Processing System (MWPS), which has storage capacity of 8000 gallons and can process 128 gpm.

TABLE 9-16

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESCRIPTION

Pump	
Type	Horizontal, centrifugal with mechanical seals
Number	2
Capacity (each)	1390 gpm
TDH	200 feet
Materials	
Casing	American Society for Testing and Materials (ASTM) A296, Gr CA-15 or ASTM A217, Gr CA-15
Stuffing Box Extension Assy. (Backhead)	ASTM A296, Gr CA-15, ASTM A217, Gr CA-15, ASTM A487 Gr CA-15, or ASTM A487 Gr CA6NM Class A
Motor	100 hp, 460 Volt, 60 Hz, 3 phase, 3550 RPM
Heat Exchanger	
Type	Horizontal counter flow Straight tube rolled and seal welded into tube sheets
Number	2 in parallel
Heat Transfer area (each)	1920 ft^2
Materials	
Shells	C.S. SA-285-C
Tubes	SS-304, SA-213
Tube Sheets	SS-304, SA-240
Shell side relief valve setpoint	150 psig
Fuel Pool Filter	
Type	Cartridge
Number	1
Design/Operating Flow	128/120 gpm
Design Pressure	175 psig
Design Temperature	250°F
Material	ASTM SA240, Type 304
Fuel Pool Demineralizer	
Type	Mixed bed, non-regenerable
Number	1
Design/Operating Flow	128/120 gpm
Design Pressure	200 psig
Design Temperature	250°F
Resin	Mixed (anion, cation)
Materials	ASTM SA240, Type 304

TABLE 9-16

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESCRIPTION

SFP Piping, Fittings, Valves

Material	Stainless Steel 304
Design Pressure	160 psig
Design Temperature	150°F/155°F[(a)]
Joints 2-1/2" and Larger	Butt-welded except at flanged equipment
Joints 2" and Smaller	Socket weld except at flanged equipment
Valves 2-1/2" and Larger	Stainless steel, butt weld-ends, 150 psi
Valves 2" and smaller	Stainless steel, socket weld ends, 150 psi
Relief valve setpoint	150 psig (on tube side of spent fuel pool cooling heat exchanger)
Butterflies 3" and larger	Rubber seated carbon steel lug type, 150 psi

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(a) Portions of the SFP Cooling System are designed for a maximum postulated temperature of 155°F [Section 9.4.4, Doc. No. 92-769(M601)].

9.5 COOLING WATER SYSTEMS - COMPONENT COOLING, SERVICE WATER, AND SALTWATER

9.5.1 DESIGN BASIS

The CC and SRW systems are designed to remove heat from the plant's various auxiliary systems. The Saltwater System provides the cooling medium for the CC and SRW heat exchangers, and the ECCS pump room air coolers. System components are rated for maximum duty requirements during normal and SDC, and are also capable of providing heat removal during a LOCA. The CC and SRW systems serve as an intermediate barrier between the various auxiliary systems and the saltwater system.

9.5.2 SYSTEM DESCRIPTIONS

9.5.2.1 Component Cooling System

Figures 9-6 (Unit 1) and 9-25 (Unit 2) shows the schematic diagram of the CC. The system for each unit consists of three motor-driven component cooling circulating pumps, two component cooling heat exchangers (Table 9-17), a head tank, associated valves, piping, instrumentation, and controls.

The component cooling heat exchangers are designed for a CC supply temperature of 95°F (a range of 70°F-95°F is acceptable during normal operating conditions), with a saltwater cooling supply temperature of 90°F, at normal operating conditions. Component cooling water may reach as high as 120°F during a LOCA and during plant cooldown and cold shutdowns.

The items cooled by CC include:

a. Letdown heat exchanger
b. Shutdown cooling heat exchangers
c. Miscellaneous waste processing heat exchanger (retired in place)
d. Waste gas compressor aftercoolers and jacket coolers
e. Control element drive mechanism (CEDM) coolers
f. RCP mechanical seals and lube oil coolers
g. LPSI pump seals and coolers
h. HPSI pump seals and coolers
i. Containment penetration cooling
j. Reactor support cooling
k. Steam generator lateral support cooling
l. Coolant waste evaporators
m. Reactor coolant and miscellaneous waste sampling system
n. Degasifier vacuum pump cooler
o. Post-accident sample system
p. Reactor coolant drain tank heat exchanger

During normal plant operation, one of the pumps and one of the heat exchangers are required for cooling service.

During normal plant cooldowns from 300°F to ≤ 140°F, two CCW pumps and two CCW heat exchangers are required to provide maximum reactor decay heat removal. During post-LOCA long-term core cooling two CCW pumps and two CCW heat exchangers provide the necessary cooling capacity to remove the decay heat from the two shutdown cooling heat exchangers.

The CCW heat exchangers are designed such that, given any single failure, the CC heat exchangers can remove sufficient reactor decay heat to ensure that the containment pressure and temperatures remain within acceptable values during post-LOCA long-term core cooling. Because the two CCW system trains are cross- connected, there are certain failures scenarios where CCW flow may be directed through a CCW heat exchanger which is not removing heat (e.g., the CCW heat exchanger has lost saltwater cooling flow). In these cases CCW system heat removal performance is enhanced by isolating CCW flow to the non-heat removing CCW heat exchanger and directing all CCW flow through the in-service CCW heat exchanger. Depending on the failure, it may be required to isolate the non-functioning CCW heat exchangers to ensure that the post-LOCA containment pressures and temperatures remain within acceptable values.

The CC pump motors are supplied from two separate 480 Volt engineered safety feature (ESF) busses, with the third motor having two breakers, one from each bus. If a loss of offsite power occurs, the pumps can be supplied by the Emergency Diesel Generators (EDGs). During normal shutdown cooling, two pumps are running with the third pump on standby. Low discharge header pressure is annunciated in the Control Room where the operator can start the third pump.

A head tank allows for expansion of the system water and provides sufficient net positive suction head (NPSH) for the component cooling circulating pumps. Makeup can be added to the system to maintain head tank level. The source of makeup water is the plant demineralized water system. Additional makeup capacity may be provided from the condensate system.

A chemical additive tank connected to the system permits maintenance of the proper corrosion inhibitor concentration in the CC.

The operation of each system is controlled and monitored in the Control Room with the following instrumentation:

a. Temperature indicators and high temperature alarms from the component cooling heat exchangers and RCP CC outlets;
b. Temperature indicators on the shutdown cooling heat exchangers;
c. Pressure indicators and low pressure alarms for each discharge header;
d. Level indicators and high-low level alarms for the head tank;
e. Handswitches and indicating lights for the pumps and remotely-operated control valves;
f. Radiation indicators and high radiation level alarm from the discharge side to the suction side of the CC pumps; and,
g. Low component cooling flow alarm to RCPs.

9.5.2.2 Service Water System

The SRW System as shown in Figures 9-9 (Unit 1) and 9-27 (Unit 2) is a closed system and uses plant demineralized water with a corrosion inhibitor added. The system removes heat from turbine plant components, blowdown recovery heat exchangers, containment cooling units, SFPC heat exchangers, AFW Pump Room Emergency Cooling Fan Coil Units, and Fairbanks Morse Emergency Diesel Generator heat exchangers.

The system has been divided into two subsystems in the Auxiliary Building to meet single failure criteria. Each subsystem has a head tank to maintain the subsystem's pressure and to allow for thermal expansion. Demineralized water makeup to the head tank is automatically controlled by level controllers. Additional makeup capacity may be provided from the condensate system.

Operating instructions provide the operators with procedures for aligning alternate sources of SRW make-up during accident or abnormal operating conditions. A cross-connection (via temporary hose) between the Saltwater and SRW Systems could be established if all non-seismic make-up water sources (demineralized water, condensate, or fire system) are unavailable.

The SRW additive tank is connected to both subsystems to allow chemical addition and control to prevent corrosion.

During normal operation, both subsystems are required and are independent to the degree necessary to assure the safe operation and shutdown of the plant-assuming a single failure. During the shutdown, operation of the SRW system is the same as normal operation, except that the heat loads are reduced.

During LOCA operation, each of the two subsystems for the two nuclear units will cool a maximum of two containment air coolers and one diesel generator. Although Unit 2 has identical heat loads and flow requirements for LOCA operations, Unit 1 subsystems do not have identical heat loads as Unit 1 has only one service water-cooled diesel generator. Service Water Subsystem 12 cools Diesel Generator 1B, and 1A is cooled from an independent cooling source located in the safety-related Diesel Generator Building. The original design heat removal capability of three of the four containment cooling units was to provide the same heat removal capability as the containment spray system. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20.

There are three SRW pumps in all. Each of two pumps is powered from a different ESFs 4 kV bus. The third pump is capable of being powered from either ESFs 4 kV bus. In the event that one bus is unavailable, a manual transfer capability to the operating bus is provided for this pump.

A low discharge header pressure will annunciate in the Control Room and the operator can then manually activate the standby pump.

The turbine plant components cooled by SRW include:

a. Generator isolated 3 phase bus duct coolers
b. Exciter air coolers
c. Generator hydrogen coolers
d. Stator liquid coolers (Unit 1 only)
e. Circ. Water System Priming Pump seal water coolers
f. Condenser vacuum pump seal water coolers
g. Feed pump turbine lube oil coolers
h. Condensate booster pump lube oil and seal water coolers
i. Instrument and plant air compressors and aftercoolers
j. Turbine lube oil cooler
k. Electro-hydraulic oil coolers

l. Turbine Building sample cooling system
m. Seal oil system coolers (Unit 2 only)
n. Auxiliary Feed Pump Room Air Cooler

Service water to the Turbine Building is not automatically isolated upon a seismic event. However, to ensure that this portion of the system will perform its pressure boundary function and provide continued cooling to the diesel generators during and after a seismic event, the entire Turbine Building SRW piping was walked down and evaluated for seismic adequacy. Consequently, a few small bore pipes and associated supports were modified to protect the non-safety-related portion of the system from potential seismically-induced spatial interaction with adjacent stationary structures and/or pipes and components.

Supply and return line redundancy is provided for containment cooling units, and diesel generators. Redundancy for SFPC is provided by cooling one SFP cooler from each unit.

Radiation monitors are installed in the SRW return header from the SFP coolers to detect possible in-leakage of radioactive liquids through the heat exchangers (Section 11.2.3.1).

9.5.2.3 <u>Saltwater System</u>

The Saltwater System has three pumps for each unit æ Nos. 11, 12, 13 in Unit 1, and Nos. 21, 22, 23 in Unit 2. The pumps provide the driving head to move saltwater from the intake structure, through the system and back to the circulating water discharge conduits (Figures 9-8 and 9-26). The system is designed such that each pump has sufficient head and capacity to provide cooling water for the SRW and CC Systems, as required by 10 CFR Part 50, Appendix A. The system also cools the ECCS pump room air coolers. The maximum recommended pump flow for each pump is 25,000 gpm. Although under most conditions this is not a limiting feature, when storm conditions consisting of the lowest expected tide of 4’0” below mean sea level and the lowest expected barometric pressure of 26.9” of mercury are considered, sufficient net positive suction head may not be available at flows above 25,000 gpm.

Power is supplied to Pumps No. 11 and 21 by 4 kV Busses No. 11 and 21, respectively, and Pumps No. 12 and 22 from 4kV Busses No. 14 and 24, respectively. Pumps No. 13 and 23 can receive power from either of the 4 kV busses in their respective units. (Figure 8-1) Pumps No. 11, 12, 21 and 22 start automatically on a SIAS or Shutdown Sequencer Signal. Pump No. 13(23) is aligned to back up Pump Nos. 11 or 12 (21 or 22). Pump No. 13(23) starts on a SIAS on shutdown sequencer signal when the backed up pump [Nos. 11 or 12 (21 or 22)] fails to start. A low discharge header pressure alarm will annunciate in the Control Room where the operator can manually activate the standby pump. The motors and controls for the saltwater pumps are located at or above Elevation +17’00” to protect them against flooding. The peak hypothetical tide and storm surge is 16.2’00” above mean low water. (Sections 7.3.2.2 and 7.3.2.3)

The Saltwater System consists of two subsystems in each unit. Each subsystem provides saltwater to two SRW heat exchanger, a CC heat exchanger and the ECCS pump room air cooler in order to transfer heat from those systems to the Chesapeake Bay. Seal water for the circulating water pumps is supplied by both subsystems. A self-cleaning strainer is installed upstream of each SRW heat exchanger.

During normal operation, both subsystems in each unit are in operation with one pump running on each header and a third pump in standby. If needed, the standby pumps can be lined-up to either supply header. Normally, the saltwater flow through the CC heat exchangers is throttled and the SRW heat exchanger saltwater valves are full open to provide sufficient cooling to the heat exchangers, while maintaining total subsystem flow below the maximum recommended value to prevent pump runout.

The operator has the option to reduce saltwater flow to the SRW heat exchangers by placing the SRW heat exchanger saltwater outlet valve flow controllers in automatic to maintain saltwater flow to each plate heat exchanger at a nominal value of 4550 gpm. At the design saltwater flow rates, the SRW heat exchangers can remove the accident heat load at saltwater inlet temperatures up to 90°F.

The saltwater pumps were originally designed for a nominal flow of 20,000 gpm with a minimum flow requirement of 10,000 gpm. To allow system operation in lower flow configurations, a saltwater bypass line exists around the SRW heat exchangers. The saltwater bypass valves are normally shut; however, they may be automatically throttled by a pressure controller to maintain saltwater header pressure within selected limits.

Operation following a LOCA has two phases æ before the RAS and after the RAS. One subsystem can satisfy the cooling requirements of both phases.

After a LOCA, but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler. Any flow established to the CC heat exchanger prior to the accident will continue during this phase. The minimum required saltwater flow is 4,000 gpm to each SRW heat exchanger, and 400 gpm to each ECCS pump room air cooler at 90°F. There is no required flow to the CC heat exchangers. The SRW heat exchanger saltwater outlet valves will remain full open or, if the outlet valves are in automatic, the saltwater flow controllers will continue to maintain flow at the same setpoint used during normal operation.

When an RAS occurs, the minimum required flow to each SRW heat exchanger remains at 4,000 gpm, and each ECCS pump room air cooler remains at 400 gpm with saltwater temperatures at 90°F. Flow is initiated or increased to the CC heat exchangers at a minimum required flow of 5,500 gpm each. The operator will throttle saltwater flow through the CC heat exchangers to maintain CC temperature. If in use to meet saltwater pump minimum flow requirements, the SRW heat exchanger bypass control valve is automatically throttled by the pressure controller to maintain the saltwater header pressure within the selected limits.

Should a piping rupture or blockage occur downstream of the heat exchangers and air coolers, an alternate flow path may be employed so the function of the components will not be impaired.

In an accident situation, control air for the throttling valves is supplied by two 64 scfm (Unit 1)/64 scfm (Unit 2), Seismic Category I air compressors. These designated air compressors are used because the Instrument Air System compressors which normally supply control air to the valves are not designated safety-related, and are not required to be operational after an accident. The compressors are normally not running, but will start automatically on receipt of a SIAS, or may be manually started from the Control Room. Upon evacuation of the

Control Room, remote manual control may be shifted to local manual control, and SIAS input to the compressors is overridden.

The throttling system meets all applicable requirements of IEEE 279.

The Saltwater Chemical Addition System serves both the Unit 1 and Unit 2 Saltwater Systems to minimize the marine fouling of piping and heat exchanger surfaces. This system has the ability to inject approved chemicals into each saltwater header, as necessary.

9.5.3 TESTING AND INSPECTION

Each component was inspected and cleaned prior to installation into the system.

Instruments were calibrated during testing. Automatic controls were tested for actuation at the proper setpoints. Alarm functions and limits were checked for operability during preoperational testing. The safety valves were set and checked.

Figures 9-6, 9-8, 9-9, 9-25, 9-26, and 9-27 show the CC, saltwater cooling, and SRW systems.

The pre-operational testing verified the following:

a. Pumps produce proper capacity and discharge head with one, two or more pumps.
b. System components receive proper flow for all modes of operation (i.e., normal, shutdown and LOCA).
c. Instrumentation and controls are functioning or responding properly.
d. Motor-operated (MOV) and control valves function.

Data is taken periodically during normal plant operation to confirm heat transfer capabilities.

9.5.4 RATINGS AND CONSTRUCTION OF COMPONENTS

Components of the cooling water system are described in Table 9-17.

9.5.5 SINGLE FAILURE ANALYSIS

The results of a single failure analysis (Table 9-17A) show that no single active failure at any time nor any single passive failure after recirculation from the containment sump will prevent the safety feature systems from fulfilling their design function.

The Nuclear Regulatory Commission (NRC) approved the application of a revised methodology for the evaluation of passive failures in moderate energy systems (Reference 1). The revised methodology assumes the passive failure to be a through-wall leakage crack of dimensions equal to one-half the pipe diameter in length, and one-half the wall thickness in width. The passive failure is postulated to occur in the largest pipe in the area to be evaluated, at least 24 hours after the initiating event. This methodology was specifically evaluated for a passive failure of the Saltwater System piping in the SRW Pump Room, but may be adopted for other moderate energy systems if supported by a similar analysis to that performed on the Saltwater System to ensure the validity of the revised methodology for those systems/subsystems. Systems evaluated by the revised methodology are annotated as such.

9.5.6 REFERENCES

1. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), dated February 24, 1995, Methodology for Postulating Passive Failure Pipe Breaks

TABLE 9-16A

HEAT EXCHANGER CONTROL VALVE POSITION

		NORMAL	LOCA BEFORE RECIRCULATION	LOCA DURING RECIRCULATION	ALTERNATE MODE
Heat Exchanger Discharge Control Valve					
Component Cooling	CV-5206	Throttle[a]	Throttle	Throttle	Closed
	CV-5208	Throttle[a]	Throttle	Throttle	Closed
	CV-5163	Open	Open	Open	Closed
	CV-5165	Closed	Closed	Closed	Open
	CV-5166	Closed	Closed	Closed	Open
Service Water	CV-5209	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5210	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5211	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5212	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5153	Open	Open	Open	Closed
	CV-5155	Closed	Closed	Closed	Open
	CV-5156	Closed	Closed	Closed	Open
Emergency Core Cooling	CV-5171	Closed[b]	Closed[b]	Closed[b]	Closed
	CV-5174	Open	Open	Open	Closed
	CV-5175	Open	Open	Open	Closed
	CV-5177	Closed	Closed	Closed	Open
	CV-5178	Closed	Closed	Closed	Open
Heat Exchanger Inlet Control Valve					
Component Cooling	CV-5160	Open[a]	Open	Open	Closed
	CV-5162	Open[a]	Open	Open	Open
Emergency Core Cooling	CV-5170	Closed[b][c]	Closed[b][c]	Closed[b][c]	Closed[c]
	CV-5173	Closed[b]	Closed[b]	Closed[b]	Closed[b]
Service Water	CV-5150	Open	Open	Open	Closed
	CV-5152	Open	Open	Open	Open
Heat Exchanger Bypass Control Valve					
Service Water	CV-5154	Closed[g]	Closed[g]	Closed[g]	Closed
	CV-5157	Closed[g]	Closed[g]	Closed[g]	Closed[g]
Saltwater Strainer Control Valve					
Diverter Valve	CV-5148	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5151	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5158	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5159	Open[d]	Open[d]	Open[d]	Open[d]
Flushing Valve	CV-5148A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5151A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5158A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5159A	Closed[e]	Closed[e]	Closed[e]	Closed[e]

TABLE 9-16A

HEAT EXCHANGER CONTROL VALVE POSITION

(a) Routinely only one CC heat exchanger is required for heat removal during normal operations. Saltwater flow to the other CC heat exchanger may be secured.

(b) ECCS pump room air cooler saltwater valves are automatically opened in order to regulate the ECCS pump room ambient temperature.

(c) A bypass line has been installed around this valve to allow for fluid expansion back into the saltwater header.

(d) Diverter valve is normally open; closes during strainer flush.

(e) Flushing valve is normally closed; opens during strainer flush.

(f) The SRW plate heat exchanger outlet valves are normally full open. They may be throttled and controlled by an FIC if the operator needs to reduce saltwater flow.

(g) Bypass valve is normally shut. It may be placed in automatic to assist in satisfying pump minimum flow requirements.

TABLE 9-16B

SALTWATER SYSTEM AIR COMPRESSORS

Type	Oil-less, Reciprocating Duplex (each SWAC has two compressor units mounted on a common air receiver tank)
No. of Stages	One
Quantity	Two
Design Capacity (scfm)	64
Design Pressure (psig)	100
Motor	Electric Motors, 10 hp each, 460 Volt, 3 phase, 60 Hz (two motors per SWAC)
Accessories	Air Receiver, Air-cooled Aftercooler, Automatic Condensate Trap
Seismic Requirements	Category I
Codes	Receiver - ASME Section VIII, Motor - NEMA

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

Component Cooling Pumps

Type	Centrifugal, horizontal, double volute, with mechanical seal
Quantity	3
Capacity each (gpm)[c]	5000
Head (feet)[c]	100
Material	
Case	ASTM A216-59T-WCB
Impeller	ASTM B145, Gr 4A ASTM B584 C83600, C87500, or C87600
Shaft	ASTM A276, Type 410 ASTM A276, Type 316 (ALT)
Motor	150 hp, 480 Volt, 60 Hz, 3 phase, 1750 RPM
Codes	Motor: NEMA Pump: Standards of the Hydraulic Institute, ASME VIII and IX

Component Cooling Heat Exchangers

Type	Horizontal, counterflow, straight tubes rolled into tubesheets
Quantity	2
Design duty each (Btu/hr)	$10.4x10^6$ (Normal)[a] $122x10^6$ (3.5 hrs after shutdown)[a] $31.2x10^6$ (27.5 hrs after shutdown) [a] $43.5x10^6$ (long term cooling following a LOCA) [a]
Heat transfer area, each (ft^2)	5860
Design pressure (psig)	Shell side: 150 Tube side: 50
Design temperature (°F)	Shell side: 200 Tube side: 200
Material	
Shell	Carbon steel ASTM A285, Gr C
Tubes	90-10 Cu-Ni ASTM B111
Tube Sheets	Aluminum bronze ASTM B171-67
Codes	ASME Section VIII, TEMA Class R

Head Tank

Type	Horizontal
Quantity	1
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Volume (gallons)	2550
Material	
Shell	ASTM A455A
Dished head	ASTM A455B
Code	ASME Section VIII

Additive Tank

Type	Vertical
Quantity	1
Design pressure (psig)	150
Design temperature (°F)	200
Volume (gallons)	75
Material	Carbon steel
Code	ASME Section VIII

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

Component Cooling Piping, Fittings, and Valves

Piping material	Carbon steel, seamless
Design pressure (psig)	150
Design temperature (°F)	180
Construction:	
2-1/2" and larger	
a. Gate and globe	Carbon steel, butt weld ends, ANSI 150 psi
b. Check and butterfly	Carbon steel, wafer type, ANSI 150 psi
2" and smaller	Carbon steel, socket weld ends, ANSI 600 psi
Codes	ANSI B31.1 except penetration piping. Penetration piping is designed and fabricated to ANSI B31.7, Class II

Service Water Heat Exchanger

Type	One pass plate and frame
Quantity	4
Capacity each (Btu/hr)	$18x10^6$ (normal operation)[a] $137x10^6$ (LOCA operation)[a,b]
Heat Transfer Area each (ft^2)	7704.4
Design Pressure (psig)	150
Design Temperature (°F)	300
Material	
Pressure Plates	Steel - SA516-70
Plates	Titanium - SB265 GR 1
Port Liners	Titanium - SB337 GR 2 (Saltwater) Stainless Steel - SA312-316 (SRW)
Gaskets	EPDM
Codes	ASME B&PV Code, Section VIII - Pressure Vessels, Section IX - Welding Qualifications, ASTM

Service Water Pumps

Type	Centrifugal, horizontal, double volute, with packed seal
Quantity	3
Capacity each (gpm)[c]	7,050
Head (ft)[c]	180
Material	
Case	ASTM A216, WCA
Impeller	ASTM B145-52, Gr-4A or B584 UNS # C83600 or B584 UNS # C87600
Shaft	ASTM A276, Type 410 or ASTM A276, Type 316
Motor	450 hp, 4000 Volt, 3 phase, 60 Hz 585 RPM
Codes	NEMA, Standards of the Hydraulic Institute, ASTM, ANSI B16.5

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

Service Water Head Tank	
Type	Vertical
Quantity	2
Design Pressure (psig)	15
Design Temperature (°F)	150
Volume (gallons)	2,350
Material	ASTM A455, Gr A
Code	ASME Section VIII
Service Water Additive Tank	
Type	Vertical
Quantity	1
Design Pressure (psig)	175
Design Temperature (°F)	200
Volume (gallons)	75
Material	ASTM A283, Gr C
Code	ASME Section VIII
Saltwater Pumps	
Type	Vertical, dry pit
Quantity	3
Design Capacity each (gpm)[c]	15,500
Design Head (ft)[c]	82
Material	
Volute	2% Ni ASTM A48, C1.35 or A439 Type D3
Impeller	ASTM B148, Gr 9D or UNS # C95500
Shaft	AISI-C1141 or ASTM A322, Gr 4140
Motor	450 hp, 4000 Volt, 3 phase, 60 Hz, 600 RPM (nominal)
Codes	Motor: NEMA Pump: Standards of Hydraulic Institute, ASME B&PV Code, Section VIII, Pressure Vessels and IX, Welding
Saltwater Strainers	
Type	Self-cleaning basket
Quantity	4
Design pressure (psig)	50
Design Temperature (°F)	100
Material	
Body	A416-60
Basket	A240 GR TP316
Code	ANSI B31.1

(a) Per vendor rating sheet; actual heat duty will vary with flow and temperature.

(b) Per accident analysis, no rating sheet available for LOCA; actual heat duty will vary with flow and temperature.

(c) These numbers, together, represent a single point on the pumps' performance curve.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ITEM	COMPONENT	NO. INSTALLED PER UNIT	NO. NEEDED FOR NORMAL OPERATION	MINIMUM NO. NEEDED FOR OPERATION FOLLOWING LOCA	DESIGN FUNCTION OF COMPONENT
Saltwater	Saltwater Pumps	3	2	1	Provide cooling water for SRW and component cooling heat exchangers and the ECCS pump room coolers.
Saltwater	Service Water Heat Exchangers	4	(a)	(b)	Provide cooling for turbine auxiliaries, SFP coolers, blowdown recovery system, containment coolers, and diesel generators.
Saltwater	Component Cooling Heat Exchangers	2	1	(b)	Provide cooling for reactor auxiliaries, HPSI pumps, LPSI pumps, and SDC heat exchangers.
Saltwater	ECCS Room Coolers	2	-	1	Maintain design temperature in ECCS rooms for long-term operation of the safety feature pumps.
Service Water	Service Water Pumps	3	2	1	Provides driving force for SRW system.
Service Water	Containment Coolers	4	(f)	(b)	Cools the containment.
Service Water	Diesel Generators	2(c)	-	1	Provides source of emergency on-site power.
Component Cooling	Component Cooling Water Pumps	3	(d)	1	Provides driving force for CC.
Component Cooling	Low Pressure Safety Injection Pumps	2	(d)	(e)	Provides safety injection water and SDC.
Component Cooling	High Pressure Safety Injection Pumps	3	-	1	Provides safety injection water.
Component Cooling	Shutdown Cooling Heat Exchanger	2	(d)	(b)	Provides cooling medium for spray water to remove heat from the containment following recirculation and SDC.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

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[a] Four SRW heat exchangers are needed.

If one saltwater subsystem is out for maintenance, the two subsystems of the SRW system may be cross-connected and the two remaining heat exchangers utilized to remove the heat load during normal operations. The two subsystems are physically separated during accident conditions; only the diesel generator connected to the operable SRW heat exchanger will be considered operable. Refer to Note (c).

[b] Each containment cooler was originally designed to remove 1/3 of the containment design heat load and each containment spray pump-shutdown heat exchanger was originally designed to remove 1/2 of the containment design heat load. The original design heat removal capability of three of the four cooling units was to provide the same heat removal capability as the containment spray system. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20. There are several combinations of equipment that could be utilized to remove heat from the containment. Each SIAS channel would actuate two containment coolers and one spray pump. For each shutdown cooling heat exchanger, one component cooling heat exchanger would be placed in service. If three containment coolers are utilized, at least three SRW heat exchangers need to be placed in service.

[c] Two diesel generators are installed per unit; however, only one diesel generator for Unit 1 and both diesel generators for Unit 2 are served by the SRW System.

[d] The LPSI pumps, the SDC heat exchangers and the CC System (i.e., CC heat exchangers and CC pumps) provide heat removal for a normal plant cooldown. One LPSI pump, one SDC heat exchanger, one CC pump, and one CC heat exchanger would provide cooldown at a slower rate. However, even at this slower rate, the plant is maintained in a safe condition (Sections 6.3.2.5, 9.5.2.1).

[e] One LPSI pump is required when suction is from the RWT and none is required when suction is switched to the containment sump during the recirculation mode of cooling.

[f] Three containment air coolers are normally in operation. Occasionally, during extended periods of high outside temperatures, all four coolers are used to limit average containment temperature to 120°F.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
Saltwater	9-8	Remotely actuated valves (CV-5150, 5152, 5153, 5155, 5156, 5160, 5162, 5163, 5165, 5166)	Fails to open or close, as applicable	These values are only operated to align the redundant discharge header. A failure of any valve would not impair the integrity of the system or prevent it from functioning.
Saltwater	9-8	Remotely actuated valves (CV-5209, 5210, 5211, 5212)	Fails open (fails to throttle)	Normally, these valves are full open. Should any one of these valves fail open while in automatic, saltwater flow to the associated PHE will be increased, improving the component's heat removal capability. The other PHE on the subsystem would continue to operate. Total system flow will increase or, if the saltwater bypass valve is in automatic, will be automatically adjusted by the bypass line CVs. The other saltwater subsystem would be unaffected by this failure and remain capable of removing the full design accident heat load.
Saltwater	9-8	Remotely actuated valves (CV-5154, 5157)	Fails closed (fails to throttle)	Normally, the saltwater bypass valves will be shut. However, should a bypass valve fail closed while in automatic, the PHE saltwater flow will increase or, if the FIC is in automatic, the outlet throttle CVs will maintain the flow through the PHEs at the setpoint. However, if only the SRW PHEs are in service, and the PHE saltwater outlet valves are in automatic, saltwater flow may drop below the minimum required flow for pump operation. The safety-related functions of the saltwater and SRW systems would not be immediately impacted. The operator can raise flow, if desired, by manually raising the FIC setpoint or remotely opening the PHE outlet valves, disabling the FIC. The other saltwater subsystem would be unaffected and capable of removing the full design accident heat load.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
Saltwater	9-8	Strainer	Basket clogs	The strainer is designed to flush automatically and on manual initiation. Should clogging occur, the affected strainer will eventually reach its dP limit and alarm setpoint. Heat exchanger saltwater flow will be maintained by the FIC, if in automatic, or will start to gradually decrease. Eventually the saltwater low flow alarm setpoint would be reached. A handhole allows quick inspection and manual cleaning. The affected strainer can be deenergized, allowing the unaffected strainer to resume automatic flushing. The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Saltwater	9-8	Strainer flushing valves (CV-5148A, 5151A, 5158A, 5159A)	Fails to cycle properly	If the valves fail to shut, the affected strainer would continue to flush and remain relatively clean. Condition would initiate system trouble alarm to alert operator. Without operator action, the interlock between the two subsystem strainers will prevent flushing of the unaffected strainer. As the strainer clogs, PHE saltwater flow will gradually decrease or, if in automatic, the FIC will compensate to maintain minimum flow to the heat exchanger. The operator can deenergize the failed strainer to allow the unaffected strainer to resume its automatic flushing sequence. Both PHEs will continue to remove their design basis heat load until the heat exchanger low flow setpoint is reached. If the valves fail to open, the operator will be alerted by the system trouble alarm. The affected strainer will gradually clog. Saltwater flow to the associated PHE will start to decrease. (Initially, the associated heat exchanger FIC will compensate, if in automatic.) The flushing circuit on the unaffected strainer would continue to function. Both PHEs will continue to remove their design basis heat load until the heat exchanger low flow setpoint is reached on the affected side.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
				The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Saltwater	9-8	Strainer diverter valves (CV-5148, 5151, 5158, 5159)	Fails to cycle properly	If the valve fails to shut during regeneration, the saltwater trouble alarm will be activated. This failure would lead to less effective flushes, probably resulting in an increased number of automatically-initiated flushes. This would eventually have the same effect as a flushing valve failing closed. If the valve fails to open during the flush cycle, the saltwater trouble alarm will be activated. The number of automatic strainer flushes would increase. Eventually this would have the same affect as a flushing valve failing closed. The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Service Water	9-9B	Valves No. 1, 3, 9, or 11	Fails to close on SIAS	Valves No. 2, 4, 10, and 12 are actuated by a redundant channel and would shut, isolating SRW as required.
Service Water	9-9B	Valves No. 5, 7	Fails to close on CSAS	Valves No. 6 and 8 are actuated by a redundant channel and would shut, isolating SRW as required.
Service Water	9-9B	Valve 27(28)	Fails to close on CSAS	Failure of valve 27(28) could render subsystem 11(21) inoperable. However, subsystem 12(22) would continue to provide the necessary cooling for Unit 1 (Unit 2).
Service Water	9-9B	Check Valves No. 17, 18, 19, 20, 21 or 22	Fails to close under reverse flow	Since in all cases two check valves are provided in series, the second valve would close providing isolation.

NOTE: As shown above, sufficient numbers of all other active components are supplied to provide sufficient redundancy for all modes of operation.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

PASSIVE FAILURE DURING CONTAINMENT SUMP RECIRCULATION

SYSTEM	FIGURE	LOCATION OF RUPTURE	CONSEQUENCES
Saltwater	9-8	Anywhere	Water is lost from one of the two subsystems. Either subsystem can provide all necessary cooling water. Double valves are provided whenever subsystems are tied together. Both of these valves are normally closed. Hence, a rupture of any one valve will not cause failure of both subsystems.
Service Water	9-9B	Valves No. 23, 24, 25, or 26	One subsystem from each unit would be drained and rendered inoperable. However, one subsystem in each unit would continue to operate. This is adequate to provide the necessary cooling for each unit. No single rupture in any location could cause the loss of both subsystems of a unit as two normally closed valves are provided where two subsystems are tied together.
Component Cooling	9-6	Anywhere	The entire system would be lost. The unit can still be maintained in a safe condition since the containment coolers would be utilized in lieu of the spray pumps/shutdown heat exchangers to cool the containment and one of the air cooled spray pumps would be manually aligned from outside the ECCS rooms for safety injection. The HPSI pumps can operate for a minimum of two hrs without cooling water and this is considered sufficient to realign the valves. Flow of one spray pump is sufficient to keep the core covered during the recirculation of the containment sump.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Intake Structure	High level alarm/Circulating Water Pumps Trip	Saltwater	The bottom of the Intake Structure is at Elevation 3'. The operator enters the Intake Structure from the Turbine Building at Elevation 12' and the saltwater pump motors are at Elevation 17'. It would take approximately 82 minutes for the water level to reach the motors and approximately 53 minutes to reach the entrance from the Turbine Building. This is sufficient time to allow shutting down one saltwater pump at a time until the leakage stops as visually determined by an operator in the Intake Structure.
Intake Structure	High level alarm/Circulating Pump Trip	Circulating Water	Before the saltwater pump motors would be flooded, the circulating water pump motors would flood and trip, eliminating the source of flooding. In addition, high level switches would trip the circulating water pump motors, eliminating the source of flooding.
Service Water Room	High level alarm in the room with normal service water head tank level	Saltwater[a]	Operators would have sufficient time to identify and isolate the break in the Saltwater System before safety-related equipment required to function would be affected by the break. For a saltwater line break that is limited to a single train, approximately 30 minutes is available to identify the affected train and isolate it. For a saltwater line break in the common portion of the SRW heat exchanger discharge piping, approximately 80 minutes is available to shift to overboard discharge after isolating the break.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Service Water Room	High level alarm in room and low level from either SRW level tank	Service Water	One subsystem would be drained. However, the other subsystem would continue to operate and is sufficient to provide all necessary SRW. The entire contents of one SRW subsystem would not flood out the SRW pumps and motors.
Component Cooling Room	High level alarm in room with normal head tank level	Saltwater	Since this room is open to the entire Elevation 5', flooding is not considered credible. A 6" curb is provided at the doorway to provide a room level indication. Flooding would be terminated by closing the remote manual valves.
Containment	Low level alarm from either SRW head tank	Service Water	In the event of a line break associated with any one containment air cooler after the LOCA, it is assumed, as an upper limit, that one subsystem of SRW leaks into containment. The leak volume from one subsystem is approximately 16,000 gallons. Boron dilution, therefore, would be negligible, because the total volume of borated water in the containment structure is in excess of 400,000 gallons.
Component Cooling Room	High level alarm in room with low head tank level	Component Cooling	Since this room is open to the entire Elevation 5', flooding is not considered credible. A 6" curb is provided at the doorway to provide a room level indication.
ECCS Room	High level alarm in room	Safety injection containment spray, containment cooling, or salt water	ECCS room is isolated. Each room is watertight and fully redundant to the other. Remote manual valves would be closed to prevent further flooding.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Condenser Pit	High level alarm in Condenser Pit	Circulating Water Expansion Joint	If the circulating water pumps were not tripped rapidly, the condenser pit would be flooded and would overflow into the Turbine Building. The Auxiliary Steam Generator Feed Pumps, auxiliary control panel, SRW Pumps and Intake Structure are protected by watertight doors. The maximum flood height from the expansion joint rupture is 15.6'. It would take approximately 45 minutes to reach the watertight door at elevation 12'-6".

[a] The passive failure is evaluated using the methodology approved by the NRC in the Safety Evaluation Report dated February 24, 1995. The passive failure is assumed to be a through wall leakage crack of dimensions equal to one-half the pipe diameter in length, and one-half the pipe wall thickness in width. The passive failure is assumed to occur in the largest pipe in the area to be evaluated, at least 24 hours after the initiating event.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

NOTE: Power cables to the saltwater pump motors could be submerged by the flooding under certain conditions. The following precautions have been taken to prevent this flooding from causing a failure of the saltwater pump motor cables:

1. These 5 kV Kerite HT and HV insulation type cables are suitable for submerged operation.
   a. The Kerite Company states that saltwater in contact with their 5 kV NS jacketed cables would cause no deleterious effects whatsoever.
   b. The Kerite Company has made numerous documented tests to prove the reliability of their Kerite HT and HV insulation cables in submerged applications. The reliability of this type of cable has also been proven through experience. The Kerite Company has been making cables with Kerite type insulation for over 100 years and has supplied many miles of this type cable for continuously submerged use.
   c. Baltimore Gas and Electric Company has used more than 100,000' of three-phase Kerite medium voltage (2.4 kV, 4 kV and 13 kV) cable for hundreds of circuits at 16 generating units. These cables have been installed for up to 31 years and over half of them are intermittently flooded by fresh or brackish water. This experience totaling nearly 1-1/2 million foot-years resulted in only one failure which was attributed to seven years of exposure to a concentrated caustic chemical powder. This 1951 cable had an asbestos fabric jacket instead of the neoprene type used since 1952.
2. To insure that the cables would not be damaged during the pulling operation, the maximum required pulling tension was calculated in 1970. These calculations showed that the maximum required tension would be less than one-third of the maximum allowable tension. Kerite engineers reviewed and concurred with these calculations. Calculations also indicated that the use of an approved pulling compound would reduce the maximum required pulling tension to less than one-sixth of the maximum allowable. Dynamometer checks during the actual cable pull measured the pulling tension at approximately one-seventh of the maximum allowable.
3. During cable installation visual spot checks were made to ascertain whether any damage was done during the cable pull. Cable was inspected after it was pulled into the Intake Structure pull box. No damage of any kind was detected.
4. To increase cable reliability, no cable splices were allowed in any of the cable runs.
5. The extensive experience which the Kerite Company and Baltimore Gas and Electric Company have had with this type cable indicates that there will be no deleterious effects due to aging during the life of the power plant.
6. The Kerite Company recommends the megger test for locating trouble without causing additional cable damage; also, the megohm readings will indicate trends toward insulation deterioration. In view of this recommendation, these saltwater pump motor feeders will be tested annually by a 2,500 Volt megger as a means of detecting any cable degradation.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

DESCRIPTION OF LEVEL SWITCHES USED IN TABLE 9-17A

INTAKE STRUCTURE

Four level switches per unit are used. The Unit 1 side of the Intake Structure is separated from the Unit 2 side by a wall three feet high. Level switches on the Unit 1 side trip the Unit 1 circulating water pumps, and level switches on the Unit 2 side trip the Unit 2 circulating water pumps. Level switches are located at a height of 3" above floor, are Seismic Category I, are manufactured to special quality control requirements, are waterproof, and are testable. Level switches feed a two-out-of-four logic system located in the service building, which is testable when the plant is shut down. The logic system provides two outputs, either of which will trip all of the affected unit's individual pump circuit breakers. The system provides redundancy but not separation of components for tripping the pumps. The system also actuates an alarm in the Control Room for each unit.

CONDENSER PIT

Two level switches per location are used. Level switches are waterproof, testable, and are designed to function under seismic acceleration. The level switches feed a one-out-of-two logic system located in the Cable Spreading Room. The logic system is Seismic Category I, is testable, and actuates an alarm in the Control Room.

COMPONENT COOLING ROOM, ECCS ROOM AND SERVICE WATER ROOM

Two level switches per location are used. Level switches are waterproof, testable, and Seismic Category I. The level switches feed a one-out-of-two logic system located in the Cable Spreading Room. The logic system is Seismic Category I, is testable and actuates an alarm in the Control Room.

9.6 SAMPLING SYSTEMS

9.6.1 DESIGN BASIS

The sampling systems are designed to permit the sampling of liquids, steam, and gases for radioactive and chemical control of the plant primary and secondary fluids.

9.6.2 SYSTEM DESCRIPTION

The sampling system consists of six subsystems; reactor coolant sampling, steam generator blowdown sampling, radioactive miscellaneous waste sampling, turbine plant sampling, gas analyzing sampling, and post-accident sampling systems (PASS). Figure 9-10 shows the reactor coolant, the steam generator blowdown, PASS and the waste process sample systems. Figure 9-30 shows the turbine plant sample system. Figure 9-11 shows the gas analyzing system.

9.6.2.1 Reactor Coolant Sampling

Each reactor coolant sampling system consists of one stainless steel sink enclosed inside a hood. The hood is ventilated by an individual blower through a high-efficiency filter and located inside the sample room (Auxiliary Building). Interlocking high-density concrete block shielding separates the hood from the rest of the sample room, which also contains the steam generator blowdown system. The reactor coolant hood is used to determine the chemical and radiochemical condition of the reactor coolant and related auxiliary systems. The hood contains piping, valves, coolers, instrumentation, and sample bombs necessary to take liquid and gaseous samples from various systems. Two samples from the pressurizer (liquid, vapor) and one from the reactor coolant hot leg system can be controlled by three handswitches located on the steam generator blowdown panel. Should any one of the remotely-operated sampling valves fail to close after a sample is taken, a second remotely-operated valve can be shut from the Control Room. These valves are also closed by SIAS. The remotely-operated valves are backed up by manually-operated valves at the reactor coolant sampling hood. High-pressure samples flow through metering valves in order to reduce their pressure. One high-temperature sample is cooled in a sample cooler supplied with CC. All analyses on these samples are performed in the laboratory located in the Auxiliary Building.

9.6.2.2 Post-Accident Sampling

If needed (see Section 1.8.1, Item II.B.3), post-accident samples can be obtained in Unit 1 or Unit 2 Nuclear Steam Supply System Sample Room in the 45' Auxiliary Building. The Unit 1 or Unit 2 Nuclear Steam Supply System Sample Room contains piping, valves, coolers, and instrumentation necessary to sample either Unit 1 or Unit 2 RCS via either the normal RCS sampling line, or Unit 1 or Unit 2 Containment sump via the LPSI system header. A grab sample is used to obtain a liquid sample from the RCS. In the Chemistry Lab, the grab sample is depressurized, degassed, and diluted as necessary to enable handling the sample without excessive radiation exposure. This grab sample capability can be used to obtain samples from the RCS or the SI system. Sample purge waste is sent to the Reactor Coolant Drain Tank of the Unit being sampled, or alternatively to the Unit 1 or Unit 2 VCT. There is a provision to analyze the dissolved gasses in the liquid sample as well as chloride and boron. The gases from the degassed coolant are vented to atmosphere via Unit 2 Plant Vent via the Chemistry Lab hoods.

9.6.2.3 Steam Generator Blowdown Sampling

Each steam generator blowdown sampling system consists of one conditioning rack-panel unit and one ventilating hood, and is located inside the same sample room as the reactor coolant hood.

The conditioning rack section of the steam generator blowdown system contains isolation valves, primary coolers, rod-in-tube devices, an isothermal bath and chiller. High pressure samples are passed through a pressure-reducing valve (rod-in-tube type) located downstream of the primary coolers and upstream of the isothermal bath. High temperature samples first pass through a primary cooler (supplied with CC) and then through the isothermal bath. All samples pass through the isothermal bath which is capable of maintaining each sample at 77°F at the coil outlet. The chiller is supplied with cooling water from the component cooling system. Sample outlets from the conditioning rack are connected to the hood.

The panel section of the steam generator blowdown system contains conductivity and pH monitors, three hand switches for pressurizer sample selection, chiller controls, and an annunciator. The pH and conductivity samples are continuously monitored and alarmed on high conductivity. In addition, pH and conductivity are trended on the computer-based display in the chemistry laboratory. High sample temperature (downstream of the isothermal bath) actuates a common alarm point. Any point alarming on the local annunciator will actuate a master alarm in the Control Room (trouble alarm).

The ventilating hood contains two stainless steel sinks and is ventilated by an individual blower through a high-efficiency filter. The ventilating hood is used to obtain samples for determining the chemical and radiochemical content of the steam generator blowdown system. The radioactive miscellaneous sample system is also located inside the steam generator blowdown hood for Unit 1. The steam generator blowdown part of the hood contains all piping, grab sample valves, instrumentation including pH cells and conductivity analyzers, and all equipment necessary for this system.

9.6.2.4 Radioactive Miscellaneous Waste Sampling

The radioactive miscellaneous waste sampling is located inside the ventilating hood for the steam generator blowdown (Unit 1) and is used to obtain samples from which the chemical and radiochemical content of miscellaneous waste is determined. This system is common to both units. All samples are low pressure and are cooled, as necessary, in sample coolers (supplied with CC). This part of the hood contains isolation valves, piping, valves, and instrumentation necessary for obtaining liquid samples from both units. The analyses of these samples are performed in the laboratory located in the Auxiliary Building.

9.6.2.5 Turbine Plant Sampling System

Each turbine plant sampling system is used to obtain samples for determining the chemical condition of the steam, feed, and condensate systems associated with the turbine plant. The system consists of one sampling station per unit (stainless steel sink and panel) and one mechanical chiller as a separate unit. These sampling systems are located in the Turbine Building. The sink contains the isolation valves, piping, instrumentation, coolers, and grab valves necessary to take samples from the steam, condensate, and feedwater systems. High-pressure samples pass through a pressure-reducing valve (rod-in-tube device). All samples pass through individual primary coolers supplied with SRW. Every sample then

passes through cooling coils immersed in the isothermal bath that maintains each sample at 77°F at the coil outlet. The mechanical chiller circulates chilled water in the isothermal bath and is supplied with SRW. Each sample is provided with one grab sample valve for taking liquid samples as necessary. The steam generator feed pump headers are continuously monitored and recorded for hydrazine, oxygen and pH, any of which can cause an alarm on the annunciator. All samples are continuously monitored for conductivity and an alarm occurs when an abnormal condition is reached. In addition, samples are trended on the computer-based display in the Chemistry Laboratory. The turbine plant system contains conductivity, pH, and oxygen recorders, oxygen analyzers, handswitches (to control the hotwell sample pumps and the chiller circulating pump), and an annunciator. The annunciator alarms on high conductivity, high pH, high oxygen, high hotwell temperature, and low hotwell sample pump discharge pressures. Any annunciator alarm will activate a master alarm in the Control Room.

9.6.2.6 Gas Analyzing System

Control of hydrogen in Containment during and following a Design Basis Event is no longer required. On March 2, 2004, the NRC issued a license amendment that allows removal of the hydrogen recombiners and hydrogen analyzers from the Technical Specifications. The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond Design Basis Event.

The gas analyzing system is used to determine the hydrogen concentration of six points inside the containment and of four samples from the reactor coolant waste tanks (receiver and monitor tanks), as well as the oxygen concentration of several samples from the reactor coolant and miscellaneous waste systems. The gas analyzing system is installed in the sample room located in the Auxiliary Building (Elevation-10') and consists of two hydrogen analyzer cabinets and separate manifolds for the isolation valves and sample selection solenoid valves and one oxygen analyzer cabinet with a manifold for the isolation valves. Two of the analyzer cabinets are for hydrogen measurement and include a sample pump, cooler, piping, valves, and instrumentation. Each hydrogen cabinet panel contains one hydrogen analyzer, one multipoint recorder for recording each measured sample, one programmer for random selection of individual readout, and alarm contacts for activation of a master alarm in the Control Room. The third analyzer cabinet is for oxygen grab sample measurement and includes a sample pump, cooler, piping, valves and sample syringe. An exhaust system on the oxygen analyzer cabinet purges any hydrogen that may leak into this cabinet.

The H_2 and 0_2 sample points are routed to the analyzer in accordance with the following table:

Sample Point	H_2 Analyzer 0-AE-6519	H_2 Analyzer 0-AE-6527	0_2 Grab Sample
Containment 1 - North of Primary Shield	No	Yes	No
Containment 1 - South of Primary Shield	Yes	No	No
Containment 1 – Pressurizer Compartment	Yes	No	No
Containment 1 - East at Elevation 135’	Yes	No	No
Containment 1 - West at Elevation 135’	No	Yes	No
Containment 1 - Dome at Elevation 189’[(b)]	No	Yes	No
Containment 2 - N	Yes	No	No
Containment 2 - S	No	Yes	No

Sample Point	H_2 Analyzer 0-AE-6519	H_2 Analyzer 0-AE-6527	0_2 Grab Sample
Containment 2 - Press	No	Yes	No
Containment 2 - E	No	Yes	No
Containment 2 - W	Yes	No	No
Containment 2 - Dome	Yes	No	No
RC Waste Rec Tank 11[a]	Yes	Yes	No
RC Waste Rec Tank 12[a]	Yes	Yes	No
RC Waste Mon Tank 11[a]	Yes	Yes	No
RC Waste Mon Tank 12[a]	Yes	Yes	No
Waste Gas Decay Tank 11	No	No	Yes
Waste Gas Decay Tank 12	No	No	Yes
Waste Gas Decay Tank 13	No	No	Yes
Waste Gas Surge Tank	No	No	Yes
Degasifier Accumulator 11	No	No	Yes
Degasifier Accumulator 21	No	No	Yes
Evaporators Discharge Gas Cooler	No	No	Yes
Przr. Quench Tank 11	No	No	Yes
Przr. Quench Tank 21	No	No	Yes
Misc. Waste Evap	No	No	Yes

(a) These samples would normally be routed to either analyzer.

(b) The 189' sample line in Unit 1 is inoperable because it is no longer seismically supported. For this reason it is not credited for post-accident sampling.

The six containment samples of the hydrogen analyzer cabinets and two samples of the oxygen analyzer cabinet can be controlled through remotely-operated solenoid valves.

To provide a post-accident containment air sampling capability, a sample vessel was placed into the sampling lines from containment 1 and 2 west at Elevation 135' to allow a syringe sample to be taken and analyzed in the laboratory. This sample vessel is located on the 45' Elevation of the Auxiliary Building.

9.6.3 SYSTEM RELIABILITY

All piping, tubing, fitting, and valves (exception listed under d and e below) in contact with fluids is 316 stainless steel and complies with the following codes:

a. ASME B&PV Code, Section III, Class 3 (Nuclear Power Plant Components) for the gas analyzing system.

b. ANSI B31.1 for turbine plant, steam generator blowdown, post-accident sampling, reactor coolant, and miscellaneous waste sampling systems. The exception to this is the normally-closed isolation valves located in the cabinets which constitute the boundary from ASME Section III piping to non-Class piping. These valves are listed below. (**NOTE**: The reactor coolant, the miscellaneous waste, and steam generator blowdown sampling systems, originally designed to meet Seismic Category I requirements, were downgraded to Category II via FCR 88-0074).

c. ASTM 450-68 which requires an eddy-current test for all tubing.

d. Pressure relief valves that are in contact with fluid shall be made of 304 or 316 stainless steel material.

e. Pressure reducing valves for the turbine plant and steam generator blowdown sample systems shall be constructed of Types 303, 304, or 316 stainless steel.

All applicable valves, piping, and coolers are designed to accept full steam pressure and temperature.

The gas analyzing system, the component cooling portion of the sample coolers in the reactor coolant hood, and the valves listed below are designed to meet Seismic Category I requirements. The reactor coolant, miscellaneous waste, and steam generator blowdown sampling systems (within the hoods and excluding those portions delineated above) are designed to meet Seismic Category II requirements. (**NOTE**: The reactor coolant, the miscellaneous waste, and the steam generator blowdown sampling systems, originally designed to meet Seismic Category I requirements, were downgraded to Category II via FCR 88-0074).

The following valves must be normally <u>closed</u> and will retain their current ASME Section III Seismic Category I classifications.

<u>Post-Accident Sampling</u>

1-PS-172	2-PS-172
1-PS-193	2-PS-193

<u>Miscellaneous Waste</u>

0-PS-226
0-PS-229

The following valves were Seismic Category I and designed in accordance with ANSI B31.1.

<u>Steam Generator Blowdown</u>

1-PS-126	2-PS-126
1-PS-128	2-PS-128
1-PS-129	2-PS-129
1-PS-137	2-PS-137
1-PS-139	2-PS-139
1-PS-140	2-PS-140

The turbine plant (Turbine Building) is designed to meet Seismic Category II requirements.

9.6.4 TESTING AND INSPECTION

Each component is inspected and cleaned prior to installation into the system. Instruments were calibrated during testing. Automatic controls were tested for actuation at the proper setpoints. Alarm functions were checked for operability and limits during preoperational testing period. The system will be operated and tested for flow, capacity, and mechanical operability.

9.7 FUEL AND REACTOR COMPONENT HANDLING EQUIPMENT

9.7.1 NEW FUEL STORAGE

New fuel is removed from its shipping container by the auxiliary hook of the Spent Fuel Cask Handling Crane and is transferred to the new fuel storage racks. These dry storage racks are for both units and are constructed to provide storage for two-thirds of a core (144 assemblies). New fuel, with a maximum enrichment of 5.0 wt% U-235, may be stored in the new fuel storage racks. For the Westinghouse standard fuel design, this results in a maximum effective multiplication factor of 0.89 at a water density of 1.0 gm/cc (full flood), and a multiplication factor of less than 0.89 for aqueous foam. Due to the large available margin, an uncertainty analysis was not performed since typical uncertainty analyses result in uncertainties of less than 3.0%. For the Westinghouse value added pellet (VAP) fuel design and AREVA fuel assemblies, this results in a maximum effective multiplication factor of less than 0.95, including all biases and uncertainties for full flood and aqueous foam conditions. If there is space in the SFP, new fuel may be stored in the Unit 1 SFP provided its wt% U-235 does not exceed the maximum enrichment allowed in the Unit 1 SFP. New fuel may be stored in the Unit 2 SFP provided that the enrichment-burnup and checkerboarding restrictions of Limiting Condition for Operation 3.7.17 are met.

Unless specified, the reactivity of any SFP or refueling pool system completely filled with VAP assemblies with axial blankets and with or without a Zirc Diboride (ZrB_2) coating is always less reactive than the design-basis VAP configuration. Thus, VAP assemblies with enrichment up to 5.0 w/o, with axial blankets, and with or without ZrB_2 can be safety stored in the new fuel storage racks.

9.7.2 SPENT FUEL STORAGE

9.7.2.1 Spent Fuel Pool Racks

The SFP is located outside the containment in the Auxiliary Building and provides underwater storage of spent fuel assemblies after their removal from the reactor vessel. The pool, designed in two halves, can accommodate 1830 assemblies and one spent fuel shipping cask. The Unit 1 half of the SFP contains storage racks in six 10x10, two 8x10, and one 7x10 array. The Unit 2 half of the SFP contains racks in ten 10x10 arrays. Control element assemblies removed from the core can be stored in the guide tubes within the fuel assemblies. The pool is constructed of reinforced concrete and lined with stainless steel. The pool was designed in accordance with Safety Guide No. 13, published March 10, 1971.

The spent fuel assemblies are placed in stainless steel storage racks consisting of vertical cells grouped in parallel rows with a center-to-center distance of 10-3/32" in both Units. Sandwiched between the inner and outer walls of each storage cell is a 6.5" wide sheet of B_4C poison material. Unit 1 storage racks use a B_4C composite material, carborundum, and Unit 2 racks use Boraflex. There is a coupon surveillance program to monitor the condition of the carborundum material. Boraflex is no longer credited as a neutron absorber due to degradation calculations (License Amendment No. 246); therefore, testing the Boraflex coupons is no longer necessary and has been eliminated. The top opening of the racks has angled lead-in guides which effectively block the spaces between the cavities, as well as guide the fuel assembly into the open tube.

Per Title 10 Code of Federal Regulations (CFR) 50.68, if no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with unborated water. If credit it taken for soluble boron, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel

assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical) at a 95% probability, 95% confidence level, if flooded with unborated water. In addition, the maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to 5 wt%.

For an infinite axial and radial array of Unit 1 storage cells of nominal dimensions with credit for the carborundum poison sheets and containing the maximum enrichment of 5.0 w/o Westinghouse VAP fuel at the worst case temperature of 40°F, the maximum unborated k_{eff} value of 0.986 is calculated with all biases and uncertainties, which is less than the 10 CFR 50.68 regulatory value of 1.0. The maximum k_{eff} value of 0.947 at a moderator boron concentration of 350 ppm with all biases and uncertainties is less than the 10 CFR 50.68 regulatory value of 0.95. Note that Westinghouse VAP fuel is more reactive than similarly enriched Westinghouse standard fuel and AREVA fuel, thus any analysis performed for Westinghouse VAP fuel conservatively bounds that for Westinghouse standard fuel and AREVA fuel.

Possible boraflex degradation in the Unit 2 SFP was documented based on calculations using the Racklife software package. Crediting burnup in lieu of boraflex assures that the 10 CFR 50.68 regulatory k_{eff} limit is maintained. The CCNPP SFP Rack Criticality Methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than the 10 CFR 50.68 regulatory limit with Westinghouse VAP and standard fuel, and AREVA fuel ranging in enrichment from 2.0 to 5.0 w/o with burnup credit and with partial credit for soluble boron in the Unit 2 SFP. The soluble boron credit will be limited to 350 ppm per the restrictions of the Unit 1 criticality analysis including all biases and uncertainties.

The burnups required to store fuel in the Unit 2 SFP crediting 350 pm of soluble boron including all biases and uncertainties are the following:

Enrichment (w/o)	Burnup (GWD/MTU)
2.0	6.00
2.5	13.75
3.0	20.50
3.5	27.00
4.0	32.75
4.5	38.25
5.0	43.75

Each assembly offloaded from either reactor or from an Independent Spent Fuel Storage Installation dry shielded canister must be evaluated against the above burnup restrictions to determine if it can be safely stored in the Unit 2 SFP. No similar restrictions exist on the Unit 1 SFP.

Several checkboard patterns were modeled in an effort to store more reactive fuel in the Unit 2 SFP. Note that only one pattern meets the requirements of 10 CFR 50.68. If credit is taken for soluble boron, that fuel assembly must be surrounded on all four adjacent faces by empty rack cells or other nonreactive materials (e.g., wall, water, …).

The SCALE 4.4 CSAS25 code module with the 44 group ENDF/B-V cross-section library was utilized to perform the KENO-Va Monte Carlo criticality calculations. The neutron adsorption of the stainless steel rack cells was credited in the criticality calculations. However, no credit for the U-234 and U-236 fuel was taken.

The analysis methods and neutron cross-section data are benchmarked by comparison with critical experiment data for similar configurations. The benchmarking process establishes a calculational bias and uncertainty of the mean with a one-sided tolerance factor of 95% probability at a 95% confidence level. The maximum k_{eff} value for the SFP is obtained by summing the calculated value, the calculational bias, the total uncertainty defined as a statistical combination of the calculational and mechanical uncertainties, and the burnup axial distribution bias. Mechanical and material uncertainties may be treated by assuming worst case conditions or by performing sensitivity studies and obtaining worst case uncertainties. Uncertainties may be combined statistically provided that they are independent variations.

The fuel design uncertainty analysis of the Unit 1 SFP consists of the following:

	Westinghouse	AREVA
Delta k_{eff} for 95/95 calculational uncertainty	0.00760	0.00760
Delta k_{eff} for stack height density	0.00417	0.00267
Delta k_{eff} for storage cell pitch	0.00575	0.00571
Delta k_{eff} for steel thickness	0.00569	0.00462
Delta k_{eff} for poison loading	0.00607	0.00513
Delta k_{eff} for eccentric positioning	0.00249	0.00243
Delta k_{eff} for fuel pellet diameter	0.00000	0.00148
Delta k_{eff} for water in gap	0.00000	0.00574
Total Delta k_{eff}	0.01355	0.01365

The Unit 1 SFP Biases included:

Delta k_{eff} for calculational methodology	0.00080	0.00080
Delta k_{eff} for poison loading	0.00466	0.00000
Total Delta k_{eff}	0.00546	0.00080

The Unit 1 SFP worst-case assumptions are the following:

Temperature	4°C (39.2°F)	4°C (39.2°F)
Fuel Clad Composition	Optin	M5®
Fuel Enrichment	5 wt%	5 wt%

A total Unit 1 bias and uncertainty value of 0.01901 was included in all Unit 1 SFP reactivity results for Westinghouse fuel, and a value of 0.01445 was included in all Unit 1 SFP reactivity results for AREVA fuel.

The fuel design uncertainty analysis of the Unit 2 SFP consists of the following:

	Westinghouse	AREVA
Delta k_{eff} for 95/95 calculational uncertainty	0.00760	0.00760
Delta k_{eff} for stack height density	0.00090	0.00394
Delta k_{eff} for storage cell pitch	0.00358	0.00711
Delta k_{eff} for fuel enrichment	0.00155	0.00696
Delta k_{eff} for steel thickness	0.01346	0.01353
Delta k_{eff} for eccentric positioning	0.00961	0.01380
Delta k_{eff} for fuel depletion	0.02089	0.01956
Delta k_{eff} for fuel pellet diameter	0.00000	0.00185
Delta k_{eff} for water in gap	0.00000	0.00269

	Westinghouse	AREVA
Delta k_{eff} for clad composition	0.00000	0.00252
Total Delta k_{eff}	0.02799	0.03075

The Unit 2 SFP Biases included:

	Westinghouse	AREVA
Delta k_{eff} for calculational methodology	0.00080	0.00080
Delta k_{eff} for axial burnup distribution	0.03250	0.03030
Total Delta k_{eff}	0.03330	0.03110

The Unit 2 SFP worst-case assumptions are the following:

	Westinghouse	AREVA
Temperature	68.33°C (155°F)	68°C
Fuel Clad Composition	Zirc4	M5®
Poison Loading	0.000 gm/cm^2	0.000 gm/cm^2

A total Unit 2 bias and uncertainty value of 0.06129 for Westinghouse fuel and 0.06185 for AREVA fuel was included in all Unit 2 SFP reactivity results, where burnup credit was assumed. If no burnup or burnup credit is assumed, a total Unit 2 bias and uncertainty value of 0.01944 for Westinghouse fuel and 0.02452 for AREVA fuel may be assumed.

A finite radial and axial model of the Unit 1 SFP of nominal dimensions containing the maximum enrichment of 5.0 w/o fuel at the worst case temperature of 40°F at a soluble boron concentration of 350 ppm including all biases and uncertainties was modeled with alternate and sequential assemblies in the row closest to the SFP wall on spacers to simulate the reconstitution/inspection process. Sufficient margin exists in going from a two to three dimensional model to counteract any increase in reactivity from raising a row of assemblies on spacers. In addition, there is no reactivity penalty between reconstituting an entire row of assemblies or alternate assemblies in a row. Since the boraflex is no longer credited in the Unit 2 SFP, raising assemblies on spacers has no reactivity effect.

Dropping an assembly of 5.0 w/o fuel onto the SFP racks was analyzed, even though it is not a credible accident. The double contingency principle was applied, which required two unlikely, independent, concurrent events to produce a criticality accident. The double contingency principle means that realistic conditions may be assumed. For example, if soluble boron is normally present in the SFP water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, total credit for the presence of soluble boron may be assumed in evaluating this accident condition. Per Technical Specifications, the normal SFP boron concentration is conservatively assumed to be 2000 ppm. Taking credit for 2000 ppm per the double contingency principle drops the k_{eff} value for this accident to well below the regulatory requirement.

The racks are designed to withstand all anticipated loadings. Structural deformations are limited to preclude any possibility of criticality. The racks are supported in such a manner as to preclude a reduction in separation space under either the Operating Basis or Safe Shutdown Earthquake. The racks themselves are designed not to collapse or bow under the force of a fuel assembly dropped into an empty cavity, or dropped horizontally across the top of the racks assuming no drag resistance from the water. The structure is fabricated of stainless steel and boron carbide sheets in both units and meets the requirements of Seismic Category I.

Spent fuel decay heat is removed by the fuel pool cooling system described in Section 9.4. The design of the racks allows for adequate convective cooling of stored fuel assemblies by natural circulation.

Monitoring and alarm instrumentation are provided at appropriate locations to assure that the decay heat from the spent fuel elements is being removed and to assure that proper radiation levels are maintained. Means will be provided to control unauthorized entry and to account for the flow of tools in and out of the area.

9.7.2.2 ICI Trash Container Rack

A four-cell ICI trash container rack is located in the lower portion of the refueling canal adjacent to the upender machine. The rack is positioned such that assemblies or cans are submerged to the same elevation as when vertically in place in the upender. Incore instrumentation trash containers or new/spent fuel assemblies may be temporarily placed in the rack to facilitate handling during refueling. The reactor vessel closure head guide studs are stored in the rack during normal plant operation. The guide studs align the reactor vessel head during refueling outages and do not interfere with the temporary handling of fuel assemblies or the ICI trash containers. The rack is stainless steel with four vertical storage positions on 24" centers. The ICI storage rack is designed to withstand all anticipated loadings and meets the requirements of Seismic Category I. Open frame construction allows for natural convective cooling.

The ICI rack design includes members which are located on the tops and sides of the racks to prevent both the inadvertent insertion of an assembly between already stored assemblies and the transportation of an assembly to a position directly adjacent to already stored assemblies. Sufficient distance is provided between the top of the active fuel and the top deck of the storage rack to preclude criticality in the event that a fuel assembly is dropped and lands in a horizontal position on top of active fuel. Angled lead-ins guide an assembly or trash can into the rack and prevent inadvertent insertion of an assembly between already stored assemblies. The criticality analysis supports the Westinghouse standard and VAP design and the AREVA design for 5.0 wt% U-235 enrichment for the ICI storage racks. Four fresh 5.0 wt% assemblies stored in the ICI racks with a fifth positioned at the minimum standoff distance, assuming no soluble boron, will maintain k_{eff} less than 0.95 including all biases and uncertainties. The k_{eff} will also remain less than 0.95, including all biases and uncertainties, after dropping a fresh 5.0 wt% assembly on an ICI storage rack filled with four fresh 5.0 wt% assemblies, assuming no soluble boron.

9.7.2.3 Spent Fuel Shipping Cask Pit

The spent fuel shipping cask pit is located on the Unit 1 side of the dividing wall in the pool. The floor of the pit is equipped with a stainless steel cask support platform upon which a shipping cask is set before being loaded with spent fuel bundles. Every cask used is designed such that spent fuel bundles can be placed in them while still maintaining the minimum water level above the fuel bundles. The cask cover is then placed on the cask and the unit is transferred to the cask wash down area by the Spent Fuel Cask Handling Crane. The wash down water is then piped to the MWPS. Means will be taken to assure that surface contamination is less than required by transportation regulations.

A cask platform/energy absorbing device is located in the cask pit area and provides two functions:

a. Elevates the pit floor surface so that the lifting trunnions on the NUHOMS transfer cask do not interfere with the cask seismic restraint.

b. Provides a second level of protection beyond that provided by the single-failure-proof crane in that the platform has the capability of absorbing the energy associated with a crane drive train failure.

The energy absorbing cask support platform, located inside the cask pit area, is comprised of a stainless steel shell Hexcel aluminum honeycomb material designed to meet the requirements of Seismic Category I and to protect the floor of the cask pit area by absorbing the impact of a cask due to drive train failure of the Spent Fuel Cask Handling Crane.

9.7.2.4 Spent Fuel Cask Handling Crane

Heavy loads (loads in excess of 1600 lbs) are prohibited from travel over spent fuel assemblies in the SFP unless such loads are handled by a single-failure proof device. The Spent Fuel Cask Handling Crane, which is designed in accordance with the "single-failure-proof" criteria of NUREG-0554 and NUREG-0612, is used to handle heavy loads in the SFP area. The maximum design rated load for the Spent Fuel Cask Handling Crane is 150/15 ton (150 ton for the main hoist and 15 ton for the auxiliary hoist). Its maximum critical load rating is 125/15 ton.

The Spent Fuel Cask Handling Crane is used to handle casks over the spent fuel pool and surrounding structures. The crane is single-failure-proof and has been designed in accordance with NUREG 0554.

9.7.2.5 SFP Purification

The SFP purification system consists of a demineralizer and filter. The demineralizer is not regenerated. When the demineralizer or filter is depleted, or as necessary, they are replaced. Additionally, cask movement meets all criteria of NUREG-0612.

The height of the filter transfer cask is 5'6" (including lifting rig). The monorail hoist hook Elevation is 57'0", 12' above the floor. Therefore, there will be adequate clearance between a cask and the floor.

The filter transfer cask and the SFP purification filter together weigh 6.30 tons. The monorail and hoist is rated for 7.5 tons.

9.7.2.6 SFP Ventilation

The spent fuel handling area ventilation system, shown in Figure 9-21, contains charcoal filters, which remove iodine and other radioactive particulates. The Auxiliary Building air is discharged to the plant vent which is constantly monitored.

9.7.2.7 Spent Fuel Handling Machine and New Fuel Elevator

The spent fuel handling machine is located above the SFP. It is a bridge and trolley arrangement, similar to the refueling machine, which rides on rails set in the concrete on each side of the pool. The handling machine is designed to pass over the dividing wall (separating the two halves of the pool) and to serve both halves of the pool. Latitude and longitude motors on the bridge and trolley position the machine over the specified rack location in the SFP.

The spent fuel handling machine serves several purposes, some of which are given here. One purpose is to transfer the spent fuel from the upending mechanism to a location in the SFP for decay, or to transfer new fuel to the upending machine. A second purpose is to take fuel from the new elevators and transfer it to a rack location in the SFP or to the fuel upending mechanism. A third function is to move fuel to and from the spent fuel inspection elevator, inspection/repair stations, and to move fuel between storage rack locations. A fourth function of the spent fuel handling machine is to transfer the decayed spent fuel to the shipping cask.

The spent fuel handling machine and new fuel elevators are designed to Seismic Category I requirements. The new fuel elevator is mounted on the west side of the Unit 1 SFP. The function of this elevator is to transport new fuel assemblies with a maximum enrichment of 5.0 wt% U-235 in the pool where the spent fuel handling machine is able to grapple and transfer the fuel to the desired location in the SFP. The fuel elevator on the west side of the Unit 2 SFP was modified to allow its use in inspection of fuel assemblies with a maximum enrichment of 5.0 wt% U-235. Two standoffs located on the new fuel elevator box assemblies ensure that a fuel assembly suspended from the spent fuel handling machine cannot be brought within eight inches of new fuel in the new fuel elevators. This is an added measure, along with existing interlocks and administrative controls, for the prevention of criticality. The spent fuel handling machine and fuel elevators are shown in Figures 1-9 and 1-13.

The Spent Fuel Handling Machine (SFHM) is capable of four modes of operation:

a. *Manual* mode allows movement of SFHM bridge, trolley, hoist, and grapple without system power available to the SFHM.
b. *Manual-Electric* mode allows SFHM bridge and/or trolley movements via joystick operations.
c. *Semi-Automatic* mode allows the operator to set "from" and "to" locations via the console, and the SFHM will automatically move the bridge and/or trolley per those settings.
d. *Automatic* mode allows SFHM bridge and/or trolley movements per a pre-determined file that contains a range of "from" and "to" locations.

Some of the safety features incorporated in this equipment are interlocks to prevent movement into the walls. These interlocks can be bypassed and restored in accordance with approved procedures. Additional safety features include limit switches to prevent the hoist from raising fuel above the point where adequate water for shielding is available. A redundant mechanical stop will prevent a fuel bundle from being raised above specified limits. This results in a maximum dose rate of 7 mrem/hr over the pool during refueling operations. The fuel grappling tool is designed so that the fuel bundle cannot be released accidentally. All motors are equipped with mechanical brakes or self-locking gears to prevent movement in case of a loss of power.

9.7.2.8 Spent Fuel Pool Platform

The SFP platform is a 16' long, 4' wide platform that fastens to the side of the SFP. It is designed such that when installed it will not interfere with the operation of the fuel handling machine. Removable railings are provided for personnel safety. The work platform is portable and can be located along the west wall of the north pool or the east wall of the south pool.

The original purpose of the work platform was to provide an efficient work site for the repairing of worn fuel assembly guide tubes (Section 3.6). The work platform overhang allows repairs to be made from a position directly over the fuel assemblies. The platform was first installed in the south pool in September 1978 in preparation for Unit 2's first refueling. It was moved to the north pool in March 1979 for Unit 1's third refueling. Since then the platform has been moved between the two pools as needed. Subsequently, its use has expanded to include eddy current tests, capsule exchanges, and fuel assembly reconstitutions.

9.7.2.9 Independent Spent Fuel Storage Installation

A detailed description of the Independent Spent Fuel Storage Installation and the transfer operations is discussed in the Independent Spent Fuel Storage Installation Safety Analysis Report.

9.7.2.10 Storage of Failed Fuel Rods in Encapsulation Tubes

Encapsulation tubes are a standard Asea Brown Boveri/Combustion Engineering device for storing failed fuel rods and for containing solid fission products. They are easily identifiable and retrievable. Encapsulation tubes safely store individual irradiated failed fuel rods in the SFP in the peripheral guide tubes of empty grid cages. A single encapsulation tube containing a damaged fuel rod can be stored in an ICI trash can, can be stored in an empty SFP rack space that is inaccessible to both the SFHM and the Auxiliary Building cask handling crane, can be laid temporarily atop the spent fuel pool storage racks with administrative restrictions on fuel movement in the laydown area, or can be placed at the bottom of an upender trench with the associated upender tagged out. Failed fuel rods in encapsulation tubes cannot be stored in the center guide tube of an empty grid cage, since an encapsulation tube from the center guide tube of a grid cage can become wedged in the grapple of the SFHM. Encapsulated fuel rods stored within the guide tubes of empty grid cages or stored in an ICI trash can or empty SFP rack space, are prohibited from extending above the spent fuel pool racks to avoid interfering with the SFHM and its load.

A criticality incident in the SFP will not occur. Storage of the encapsulation tubes in the peripheral guide tubes of empty grid cages or in an ICI trash can or empty SFP rack space will cause a decrease in maximum SFP reactivity due to a decrease in fissile inventory and will not create the possibility of inadvertent criticality. An encapsulated fuel rod placed temporarily atop the spent fuel pool storage racks or at the bottom of the upender trench will be decoupled in reactivity space from the assemblies stored within the rack. Undamaged fuel rods can only be stored in the encapsulation tubes in empty grid cages. This will ensure that the consequences of a fuel handling incident will be limited by the current analysis.

9.7.3 REACTOR COMPONENT HANDLING EQUIPMENT

The refueling equipment arrangement is shown in Figure 9-12.

9.7.3.1 Reactor Refueling Machine

The reactor refueling machine is shown in Figure 9-13.

The refueling machine is a traveling bridge and trolley which spans the refueling pool, and moves on rails located at Elevation 69'6" in the containment area. The bridge and trolley motions allow coordinate location of the fuel handling mast and hoist assembly over the fuel in the core. The hoist assembly contains a coupling device which, when rotated by the actuator mechanism, engages the fuel assembly to be removed. The hoist assembly is moved in a vertical direction by a

cable that is attached to the swivel top of the hoist assembly, and runs over a sheave on the hoist cable support to the drum of the hoist winch. After the fuel assembly is raised into the hoist and the hoist into the refueling machine mast, the refueling machine transports the fuel bundle to another location or to the upender.

The controls for the refueling machine are mounted on a console which is located on the refueling machine trolley. Coordinate location of the bridge and trolley is indicated at the console by digital readout devices, which are driven by encoders coupled to the guide rails through rack and pinion gears. A system of pointers and scales is provided as a backup for the remote positioning readout equipment. Manually-operated handwheels are provided for bridge, trolley and winch motions in the event of a power loss.

The Refueling Machine is capable of three modes of operation:

a. *Manual-Electric* modes allow Refueling Machine bridge and/or trolley movements via joystick operation.
b. *Semi-Automatic* mode allows the operator to set “from” and “to” locations via the console, and the Refueling Machine will automatically move the bridge and/or trolley per those settings.
c. *Automatic* mode allows Refueling Machine bridge and/or trolley movements per a pre-determined file that contains a range of “from” and “to” locations.

During withdrawal or insertion of a fuel assembly, the load on the hoist cable is monitored at the control console to ensure that movement is not being restricted. Variations from normal loads in excess of 10% will automatically stop the motion of the hoist winch mechanism. A zoned, mechanical interlock is provided which prevents opening of the fuel grapple and protects against inadvertent dropping of the fuel. A piston-operated spreader device is provided which spreads adjacent fuel assemblies within the core to provide unrestricted removal and insertion. This spreader is part of the mast assembly and is operated after grappling of the fuel assembly. The safety features of the refueling machine are:

a. An anti-collision device on the refueling machine mast which stops bridge and trolley motion. This device consists of a hoop and limit switches to protect the mast from hitting the vessel guide studs, structures within the refueling cavity or the walls of the refueling cavity;
b. Interlocks which restrict simultaneous operation of either the bridge and trolley or the hoist winch drive mechanism;
c. An interlock which prevents bridge and trolley motion when spreader device is actuated;
d. Interlocks that prevent bridge and trolley motion until the hoisting operation is complete;
e. Over and under load switches which stop fuel hoist motion;
f. Automatic bridge and trolley speed restriction zones over the reactor core;
g. Fuel hoist programmed speed restriction while the fuel bundle is within the core and upending machine;
h. An interlock which prevents positioning of the refueling machine over the tilting machine unless the hoist is at the up limit and the spreader is retracted.

9.7.3.2 Fuel Transfer System

Upending Machines

Two upending machines are provided for each unit, one in the Containment Structure refueling pool and the other in the SFP. Each consists of a structural steel support base from which is pivoted an upending straddle frame, which engages the two-pocket fuel carrier. When the carriage with its fuel carrier is in position within the upending frame, the pivots for the fuel carrier and the upending frame are coincident. Hydraulic cylinders attached to both the upending frame and the support base rotate the fuel carrier between a vertical and horizontal position, as required by the fuel transfer procedure.

Interlocks are provided to ensure the safe operation of this equipment by prohibiting the lowering of a fuel assembly unless the fuel carrier is vertical, by preventing inadvertent rotation of the tilting cylinders while a fuel assembly is being lowered, and by deactivating the cable drive so that a premature attempt to move the carriage through the transfer tube cannot be initiated.

Fuel assemblies of 5.0 wt% U-235 can be inserted into the upenders. A k_{eff} less than or equal to 0.95 with biases and uncertainties can be maintained with two fuel assemblies of 5.0 wt% U-235 inserted into the upender.

Transfer Carriage

During refueling periods, the transfer carriage transports one or two fuel assemblies with a maximum enrichment of 5.0 wt% U-235 between the refueling pool and the fuel storage area. Ten large wheels, five on each side, support the carriage and allow it to roll on tracks within the transfer tube. Track sections at both ends of the transfer tube are supported from the pool floor and permit the carriage to be properly positioned to the upending mechanisms. The carriage is driven by steel cables connected to the carriage and through sheaves to its driving winches mounted below the operating floor level. The fuel carrier is mounted on the carriage and is pivoted for tilting by the upending machines.

Transfer Tube and Isolation Valve

The fuel transfer tube shown on Figure 9-14 connects the refueling pool with the SFP. During reactor operation, the transfer tube is closed by an isolation valve outside the containment and a blind flange inside the containment (Figure 9-14A). The flange is subject to local leak rate testing (Type B). The isolation valve is not local leak rate tested. The tube is supported by a larger diameter pipe which, in turn, is sealed to the containment envelope. The two concentric tubes are sealed to each other with a bellows-type expansion joint.

Transfer Rails

This assembly contains the rails on which the transfer carriage rides when moving between the reactor cavity and SFP area. The rail supports are welded to the 36" diameter transfer tube. The rail assemblies are fabricated to a length which will allow them to be lowered for installation in the transfer tube. A gap is left in the track at the valve on the fuel storage side of the transfer tube to allow closing of the valve.

9.7.3.3 CEA Handling Tool

Unit 1:

The refueling machine auxiliary hoist, used in conjunction with the CEA handling tool, is used to exchange CEAs within the reactor core under normal conditions. The auxiliary hoist has sufficient capacity to hoist a CEA.

Unit 2:

The refueling machine auxiliary hoists, used in conjunction with the CEA handling tool, are used to exchange CEAs within the reactor core under normal conditions. Each auxiliary hoist has sufficient capacity to hoist a CEA.

The CEA handling tool is visually aligned by a licensed operator to verify that the tool is correctly positioned above the fuel assembly prior to grappling. The CEA handling tool has a rotary grapple which rides in a vertical channel section so that inadvertent release of a CEA is not possible. A load cell is used with the CEA handling tool to verify loading (unloading) and prevent hoisting a bound CEA. Administrative controls prevent translation during hoisting or vice versa.

The CEA handling tool can also be used in the SFP, where it is lifted by the single failure proof crane.

9.7.3.4 Reactor Vessel Head Lifting Rig

The reactor vessel head lifting rig is shown in Figure 9-16.

This lifting rig is composed of a three-part lifting frame (tripod) and three lift links. The lift links are attached to the outer shroud, which is part of the CEDM air cooling structure. The outer shroud is attached to the reactor vessel head. The lift links support a service structure that includes three hoists for handling the hydraulic stud tensioners, reactor vessel studs, washers, and nuts. The lift links and service structure provide support for the CEDM, Reed Switch Position Transmitter, ICI, and Reactor Vessel Level Monitoring System electrical cables. The tripod is removed prior to plant operation.

9.7.3.5 Reactor Internals Lifting Rig

The reactor internals lifting rig consists of three major subassemblies: (1) upper guide structure lift rig which includes an ICI hoist, (2) a core support barrel lifting rig, and (3) an upper clevis (tripod) assembly. The upper clevis assembly is common to the upper guide structure lifting rig and the core support barrel lifting rig. The upper clevis assembly is a tripod-shaped structure connecting the lifting rigs to the containment crane lifting hook.

The upper guide structure lifting rig is shown in Figure 9-17. This lifting rig consists of a delta spreader beam which supports three columns providing attachment points to the upper guide structure. Attachment to the upper guide structure is accomplished manually from the working platform by means of lifting bolt torque tools. The integral ICI hoist connects to an adapter which is manually attached to the ICI structure by utilizing an adapter torque tool. The ICI is then lifted by the crane hook.

A core support barrel lifting rig, shown in Figure 9-18, is provided to withdraw the core barrel from the vessel for inspection purposes. The lifting rig includes a spreader beam providing three attachment points. Attachment is accomplished manually from the refueling machine bridge by means of a lift bolt torque tool.

Correct positioning of either the upper guide structure lifting rig or the core support barrel lifting rig is assured by guide bushings attached to the rigs which mate to the reactor vessel guide pins.

A separate upper guide structure lift rig is provided for each of the two reactors. The core support barrel lifting rig is shared by Unit 1 and Unit 2.

9.7.3.6 Surveillance Capsule Retrieval Tool

A retrieval tool is provided during the refueling shutdown for manual removal of the irradiated capsule assemblies of the reactor vessel materials surveillance program described in Section 4.1.5.4.

A diagram of the surveillance capsule retrieval tool is shown in Figure 9-19. The tool is operated from a position on the carriage walkway of the refueling machine. Access to the capsule assembly is achieved by inserting the tool through 3" diameter retrieval holes in the core support barrel flange provided in each capsule assembly location. A female acme thread at the end of the retrieval tool is mated to the surveillance capsule lock assembly (Figure 4-14) by turning the retrieval tool handle. A compressed spring in the lock assembly exerts a high frictional force at the retrieval tool-lock assembly interface to prevent disengagement during retrieval.

The overall length of the tool is 45.5'. The tool consists of two parts to facilitate storage. The upper portion is a 2" diameter tube and handle. The lower portion of the tool is also a 2" tube with a 1" outer diameter at the connector end. A 3/4" diameter hole in the upper end of the tool permits the containment crane to assist with the retrieval procedure and prevents inadvertent dropping of the tool. The tool is made of aluminum and has a dry weight of 40 lbs.

9.7.4 DESIGN EVALUATION AND SYSTEM RELIABILITY

Underwater transfer of spent fuel provides ease and safety in handling operations. Water is an effective, transparent radiation shield and an efficient cooling medium for removal of decay heat. Basic provisions to ensure the safety of refueling operations are:

a. Gamma radiation levels in the containment and fuel storage areas are continuously monitored and recorded (Section 11.2.3). These monitors provide an audible alarm at the initiating detector and in the Control Room, indicating an unsafe condition. Continuous monitoring of reactor neutron flux, with indication in the Control Room, provides immediate indication and alarm of an abnormal core flux level during fuel loading and unloading operations.
b. Whenever fuel is added to the reactor core, the source range neutron flux (count rate) is recorded to verify the subcriticality of the core.
c. The design of the equipment places physical limits on the extent of fuel movement, thereby avoiding any possibility of raising fuel beyond a safe limit. Fuel storage rack spacing provides positive protection against criticality in the event of inadvertent flooding of the fuel storage area with fresh water. The design of the spent fuel storage pool is such that water cannot drain out of the pool by gravity.

Manually-operated handwheels are provided to allow refueling bridge, trolley and winch motion in the event of a power loss.

The fuel transfer carriage is longer than the fuel transfer tube, assuring that one end of the carriage is accessible at all times during the transfer operation. Operability of the refueling system is assured by functional testing prior to each refueling operation.

At least 10" is provided between the top of the active fuel and the top deck of the storage rack to preclude criticality in the event of a fuel assembly is dropped and lands in the horizontal position on the top deck. The design of the racks assures adequate convective cooling to a fuel assembly lying horizontally across the top of the racks.

9.8 PLANT VENTILATING SYSTEMS

9.8.1 DESIGN BASIS

The plant ventilating systems are designed to provide a suitable environment for equipment and personnel with a maximum amount of safety and operating convenience. Potentially contaminated areas are separated from clean areas. Airflow patterns originate in areas of potentially low contamination and progress toward areas of higher activity. Generally, negative pressures are maintained in potentially contaminated areas and positive pressures in clean areas. The ventilating systems in the containment, waste processing and fuel-handling areas are designed for containment of radioactive particles. The path of the discharge from all potentially contaminated areas is directed into the respective plant vent where the radioactivity level is monitored. The equipment in most critical systems is redundant in character; detailed descriptions are presented where this occurs. Basic temperature design criteria are listed in Table 9-18.

9.8.2 SYSTEM DESCRIPTION AND OPERATION

9.8.2.1 General

The plant ventilation systems discussed in this section are shown on Figures 9-20A, 9-20B, and 9-21, and listed in Table 9-18.

The containment cooling and filtering systems are discussed in Sections 6.5 and 6.7, respectively. The penetration room ventilation system is discussed in Section 6.6.

9.8.2.2 Containment Ventilation

Control Element Assembly Drive Mechanism Cooling System

In this system, air is drawn from the containment at a rate of 800 cfm and design temperature of 120°F, through the reactor head cooling shroud and into two cooling coils of the CEDM cooler, which is located on the missile shield above the reactor. From there, 100% redundant fans discharge the cooled air upward into the containment again. Four ducts connect the shroud to the cooler coil house. One pair of ducts directs air to one cooling coil and the other pair supplies air to the opposite coil. Cooling water at a design inlet temperature of 95°F is pumped through the water-air coils. A power-operated damper located between each fan and the coil house prevents short-circuiting of air around the cooler when only one fan is operating. The switch-over from one fan to the other is accomplished by remote-manual control from the Control Room. Tests have shown this cooling air is more than adequate to maintain coil temperatures below 350°F. The airflow and geometrical cooling shroud configuration simulated that of the on-site installation. Testing of the CEDMs included holding, insertion, withdrawal and tripping operation.

In no case will loss of cooling air prevent the CEDM from releasing the CEAs if a reactor trip is initiated. Tests, in a simulated operating environment, have shown that the CEDM is capable of dropping the CEA after four hours of operation in the hold mode without cooling air supply. These heat transfer tests were conducted on a full-size prototype in a hot autoclave simulating reactor operation conditions.

Containment Purge System

There is a separate, identical purge system for each containment. In each system, an air-handling unit, located in the Auxiliary Building, supplies filtered and tempered air to the containment through a supply duct.

One exhaust fan for each Containment Structure, located in the Auxiliary Building, draws air from the containment through an exhaust duct and high efficiency particulate air (HEPA) filters, and discharges it into the respective main plant exhaust plenum where the fans force it into the plant vent. The air-operated butterfly valves, which fail closed, are located in the supply and exhaust ducts inside of containment to provide containment closure when the reactor is in a shutdown condition. During reactor operation, the purge penetrations are closed by a blind flange in each penetration outside of containment.

When the reactor is shut down, the containment purge isolation valves are closed and the purge system fans are stopped by a containment radiation signal.

During normal operations of the containment purge ventilation system, negative pressure is maintained in the Containment. Alternate line-ups may result in some natural air circulation in and out of Containment. Administrative procedures are in place to monitor and ensure that the potential release of radioactive particles from the Containment, while the purge air supply and exhaust fans are secured, will remain within the Offsite Dose Calculation Manual limits.

Containment Vent System

This system is designed to operate as a containment vent during power operations. The system is utilized to control containment pressure and airborne radioactivity within specified Technical Specification limits.

Upon receipt of a SIAS, containment radiation signal, or a high-radiation signal, the inboard and outboard MOVs close.

Although control of hydrogen in Containment following an accident is not required, this penetration may be used as a hydrogen purge.

Pressurizer Compartment Cooling

A metal wall, designed to blow out at less than 5 psi, separates the pressurizer compartment from the RCP area so as to prevent entrance of hot air from the pump motors into the compartment. In order to prevent the concrete upon which the pressurizer rests or pipe-mounted electrical components from overheating, cooling air is supplied at two levels from the containment coolers. In addition, air is supplied from the containment to the upper extremes of this compartment. The air supplied for cooling pressurizes this compartment and then exits through the access opening at Elevation 81'0".

Cavity Cooling System

Two redundant fans supply air from the containment air cooler plenum through ducting to the reactor cavity distribution manifold where it is used to cool the neutron detectors, the primary shield penetrations and the primary shield. System performance is adjusted by manual balancing. High efficiency filters are installed to protect each branch which serves a neutron detector.

9.8.2.3 Auxiliary Building Ventilating Systems

Control Room

The Control Room (Elevation 45'0") and the Cable Spreading Room (Elevation 27'0") are incorporated into a single year-round air-conditioning system serving both Units 1 and 2. Therefore, the ambient temperature in the Control Room is expected to be the same as the ambient temperature in the Cable Spreading

Room. Air handling and refrigeration equipment are redundant. The Control Room and Cable Spreading Room areas have a third source of cooling, which is not safety-related, in the form of a water chiller supplying a second set of coils in the safety-related air handling systems.

In the event that both the non-safety-related chiller and the safety-related condensers are rendered inoperable by a tornado, a post-tornado mode of cooling the Control Room and cable spreading rooms is available. In this mode of cooling, the fresh air dampers are fully opened, the recirculation dampers are fully closed, and the exhaust damper is fully opened to allow Control Room and cable spreading room cooling using outdoor air only.

The Control Room ventilation system continuously operates in the recirculation mode. The ventilation system is not designed to maintain the ventilated areas pressurized to a positive 1/8" water gauge pressure. If airborne contamination is detected, a high radiation signal (control room recirculation signal) from the recirculation air monitor will start the post-LOCI filter fans which will open their associated gravity discharge dampers, and close the toilet area exhaust duct damper. The post-LOCI filter fan unit inlet dampers are already in the open position. A Unit 1 SIAS A1 and Unit 2 SIAS B1 initiation signal was installed to augment the control room recirculation signal actuation. The SIAS initiation also starts the post-LOCI filter fans and opens their associated gravity discharge dampers, and secures the control room lavatory exhaust fan. Each post-LOCI filter unit is designed to process 10,000 ± 10% cfm of circulated air through HEPA and charcoal filters.

A separate exhaust fan is provided for the lavatory but, during the post-incident period when the air flow is in the complete recirculation mode, the lavatory exhaust is cut off automatically as described above.

All equipment except for ducting is remotely located so as to minimize the fire hazard.

The air conditioning system in this area is divided into three supply and return duct systems: one for each Cable Spreading Room and one for the Control Room. A portion of supply and return air is also routed to the Control Room heating, ventilation, and air conditioning (HVAC) equipment room. Each supply and return branch contains an isolation damper. Smoke detectors are located in the return duct from each zone.

In the event of a fire in one zone, the smoke detector automatically closes the corresponding isolation dampers. The air conditioning system continues to serve the other two zones without interruption.

With the isolation dampers closed, smoke can be evacuated from the isolated zone by means of an auxiliary fan. This fan is selectively connected to the return duct of any zone by operating motorized dampers in the auxiliary duct system. Air from the outside is allowed to enter the supply duct of the isolated zone by operating motorized dampers and manually opening the roof mounted hatch and damper. The operating panel for the motorized dampers and smoke removal fan is located just outside the Control Room entrance in the heater bay area.

Control Room Habitability

In accordance with TMI Item III.D.3.4, "Control Room Habitability," BGE committed to ensure Control Room Operators were adequately protected against the effects

of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions. The NRC concluded that the Control Room habitability systems were acceptable, and that the systems provided safe, habitable conditions within the Control Room under both normal and accident radiation and toxic gas conditions, including LOCAs (Reference 1). This conclusion was predicated upon commitments to install a shield wall to prevent any streaming through the pipe chase into the Control Room from below, and to ensure sufficient self-contained breathing apparatus are available to the Control Room personnel to meet the requirements of Regulatory Guide 1.78.

Subsequently, BGE suspended the Control Room habitability thyroid dose calculation pending the issuance of revised source terms for the evaluation of Control Room habitability from the NRC. The NRC concurred with the adequacy of the associated interim compensatory measures (Reference 2) during this period.

Reference 3 approved the use of an alternative radiological source-term methodology for analyzing design basis accident radiological consequences. The analysis assumptions regarding operation of the Control Room ventilation system are discussed in Section 14. Associated plant modifications are discussed in the appropriate Updated Final Safety Analysis Report sections.

Battery Rooms

The battery rooms, located east of the Cable Spreading Room, are ventilated using air from the access control area. Separate supply and exhaust fans are utilized so as to maintain a negative pressure in these rooms with respect to the surrounding areas. The reserve battery room on Elevation 45'0" is also ventilated by the same supply and exhaust system.

Access Control Area

The access control area common to both plant units is located on Elevation 69'0" and 72'0", partly in the Auxiliary Building and partly in the Turbine Building. In 1984, the original access control area was renovated, and an expansion added in the Turbine Building portion. The access control area is now divided into "Clean" and "Controlled" zones. A separate HVAC system is provided for each zone.

The controlled zone includes hot laboratory, cold laboratory, frisker area, etc. No air is recirculated, all the air is conditioned, and a negative pressure is maintained in the controlled zone.

In the controlled zone, the air is supplied by Access Control HVAC RTU-1 and exhausted by access control exhaust Fans No. 11 and 12 through Unit 2 main exhaust plenum. Access Control HVAC RTU-1 provides filtered, tempered air to the controlled zone. It takes suction on outside air intake through redundant safety-related dampers and discharges air to rooms through duct work to the two gas bottle storage rooms, the hot laboratory, cold laboratory, counting room, clean room, frisker areas, clothing disposal area, and corridors 592 and 594. Note that the redundant safety-related dampers fail closed on a loss of power. A redundant set of radiation monitors will be installed which also allow the dampers to close.

The clean zone includes office, sign in/out, locker room, etc. Negative pressure is not necessarily maintained and the air is recirculated. The clean zone is heated, ventilated, and air conditioned by self-contained units. Access Control HVAC AHU-1 has safety-related dampers that fail closed on a loss of power. A

redundant set of radiation monitors will be installed which also allow the dampers to close.

Main Steam Line Penetration Areas

Heat released by the main steam and feedwater pipes requires that cooling be provided all year round. One cooling and ventilating system is provided for each unit. This system uses outside air as the cooling medium.

Fresh air is mixed with recirculated air as required and supplied through ducting from an air-handling unit located on the floor at Elevation 27'0". The main steam line penetration area for each containment is pressurized and the excess air flows out through the open safety vent to the roof at Elevation 91'6". A room thermostat controls the position of the mixing dampers, which are located upstream of dust-stop filters.

Switchgear Rooms

Redundant and Seismic Category I HVAC and refrigeration systems are provided for each unit, with the exception that the pneumatic tubing is not Seismic Category I. Failure of the tubing does not affect safe shutdown of the plant. The equipment room for both is located on Elevation 69'0" and serves switchgear rooms for both units at Elevation 27'0" and 45'0". These rooms require cooling the year around. An "air conditioning" system supplies filtered air for ventilation and cooling at all times. The HVAC units and refrigeration components are redundant, but the supply and return ducts are not. High temperature air alarms annunciate in the Control Room to signal failure of the HVAC system. One alarm per room provides redundancy since both rooms are supplied by the same HVAC unit. If the unit fails, both rooms will heat up and provide an alarm. The inlet dampers fail open so that the rooms cannot become isolated.

The normal design and operating temperature of the Switchgear Room is 104°F. If both refrigeration units for a Switchgear Room fail, the fans can be arranged to supply 100% outside air to these rooms. The effect of purging with outside air can be evaluated by the operator and appropriate action taken.

Fairbanks Morse Diesel Generator Rooms

The Fairbanks Morse diesel generators are housed in three separate rooms located at Elevation 45'0" in the Auxiliary Building. Heat output from each generator is sufficiently high that cooling must be provided for both summer and winter. The ventilation system for this area is designed to limit room temperature to 120°F in summer and a minimum of 60°F in winter. Outside air is used as the cooling medium. An air-handling unit and mixing box-damper arrangement proportion the outside and recirculated air according to room temperature. When the diesel is running, its room is pressurized and the excess air is forced out through a weatherproof exhaust opening over the outside door. Hot water unit heaters maintain a minimum temperature of 60°F when the diesel is shut down.

Waste Processing Area Ventilation

A common air supply system consisting of three 50% capacity air handling units positioned on the west side of the Auxiliary Building at Elevation 69’0” supplies tempered air for ventilation of the common waste processing area. A system of ductwork ensures a uniform distribution throughout this area.

Separate exhaust systems for Units 1 and 2 draw air from their respective waste processing areas by means of ductwork and force it through HEPA filters, after which it is discharged into the main exhaust plenums provided for each unit. From here, the main plant exhaust fans force the air past the radioactivity monitors and out through the exhaust stacks. These exhaust fans are 100% redundant, but the filters are not.

Emergency Core Cooling Pump Room Ventilation

The ECCS pump rooms for Units 1 and 2 are served by the common waste processing area ventilation supply system. The ECCS pump room exhausts may be directed through HEPA filters prior to emptying into the main plant vent. When the ECCS pumps are operated post-accident, air flow from the ECCS pump room area may be diverted through the charcoal filters by manual remote actuation in the Control Room. However, the operation of this system and the resultant effects on offsite dose calculations are not credited in the accident analysis. This system provides defense-in-depth only.

Fan-coil coolers are installed in each ECCS pump room to provide additional cooling, if necessary, during pump operation.

Spent Fuel Pool Ventilation

An air supply system consisting of two 50% capacity air handling units, located at Elevation 86'0", directly above the three supply units for the waste processing area, provides ventilation for the SFP area. Tempered outside air is supplied to one side of the SFP area at Elevation 69'0". A separate exhaust system picks up air through a manifold, located on the opposite side of the pool, draws it through HEPA filters, and feeds it into the main plant vent of Unit 1. During movement of recently irradiated fuel assemblies, this air may be manually diverted by dampers into charcoal filters after it leaves the HEPA filter bank. Unit heaters are used to maintain a minimum temperature of 60°F in the winter.

The SFP Ventilation System is capable of maintaining a negative pressure with respect to surrounding areas of the building. The limitations placed on this system by the Technical Specifications ensure that, in the event of a fuel handling accident, involving recently irradiated fuel, all radioactive material released from a recently irradiated fuel assembly will be discharged to the atmosphere through the main plant vent. The operation of this system is consistent with the assumptions of the accident analyses.

Auxiliary Feedwater Pump Room

There are "normal" and "emergency" cooling systems used to cool the room. Identical systems are used for both Unit 1 and 2 pump rooms. (Refer to Section 6.9 for a description of the emergency mode of operation.) During normal plant operation, one operable self-contained HVAC unit is capable of maintaining the temperature in this room at 90°F or below. Air for ventilation is drawn in through redundant quick-close dampers from the Turbine Building and is forced out through ducting into the mechanical equipment room at Elevation 5'0" of the Auxiliary Building.

Decontamination Room

This system is intermittent in operation and has by necessity been appended to the normal ventilating system of the Unit 2 side of the Auxiliary Building at Elevation 5'0". While the decontamination room exhaust fan is running, the

"normal" exhaust system from the tank rooms located at the west end of Unit 2 at this level, plus that from the decontamination room itself, is automatically interrupted by means of powered dampers.

By means of a fan located within this room, air is collected from three exhaust hoods which cover separate cleaning areas. It is drawn through a water-pad type scrubber and directed into the normal exhaust system of the Auxiliary Building at this level. Air is supplied by the normal ventilating system to the hallways and enters this room through the connecting passageways. Waste water from the scrubber is directed into the WPS.

Auxiliary Building Ventilation Charcoal Filters

Table 9-19 lists the Auxiliary Building charcoal filters, total flow rates and total charcoal weights. The charcoal is Barnebey-Cheney #727 (or equivalent) impregnated with 5 wt% iodine compounds. The flow velocity through the charcoal bed is 40 fpm in all cases and the corresponding residence time is 0.25 seconds. A typical charcoal filter module is 24-1/4"x25-3/4"x6-1/4". Each module is designed for an air flow of 333 cfm. Each filter housing contains sufficient modules for the total flow rates shown in Table 9-19. Filter testing is explained in Section 6.6.7.

Testing is performed to demonstrate that the installed charcoal adsorbers will perform satisfactorily in removing both elemental and organic iodides for design conditions of flow, temperature, and relative humidity. Periodic testing is conducted to ensure filter efficiencies credited in the accident analysis are maintained.

Diesel Generator Building HVAC System

The HVAC System for the safety-related Diesel Generator Building is divided into safety-related and non-safety-related portions. Two non-safety-related air handling units provide ventilation to the Diesel Generator Building. Air handling unit one (1A-AHU-1) provides ventilation to the Diesel Generator Building Control Room, Battery Room, 1E Switchgear Room, and non-1E Electrical Panel Room. Air handling unit two (1A-AHU-2) provides ventilation to the Maintenance Shop, hallway, Future Expansion Room, and Fuel Oil Storage Tank Room. The Diesel Generator Room is cooled by four safety-related fans when the emergency diesel is in operation and stand-by conditions. While the EDG is not in operation, a non-safety-related ventilation system provides cooling to the Diesel Generator Building Control Room, Battery Room, 1E Switchgear Room, and Non-1E Electrical Panel Room using a constant volume, direct expansion cooling air handler unit one (1A-AHU-1). These rooms are exhausted using a non-safety-related exhaust fan (1A-F-7), except for the battery room which uses a separate safety-related exhaust fan. However, during diesel generator operation, a safety-related supply and exhaust fan provides cooling using only outdoor ambient air. Both the non-safety-related air handling unit one and the safety-related supply and exhaust fan share a common section of the ductwork to supply and exhaust these rooms. Interlocks are provided to ensure that both the safety-related fan and the non-safety-related air handling unit do not operate at the same time. The HVAC system is designed to maintain the diesel generator room between 50º and 120ºF.

Station Blackout Diesel Generator Building HVAC System

The HVAC System for the Station Blackout Diesel Generator Building is designed to maintain the temperature in the Station Blackout Diesel Generator Building within the standards of manufacturers of equipment in the building. Four

augmented-quality fans, each thermostatically controlled, are provided to exhaust air from the Diesel Generator Room. The Station Blackout Diesel Generator Building HVAC System also includes two augmented-quality air handling units to provide ventilation to the building. Air handling unit one (0C-AHU-1) supplies and exhaust conditioned air to the Control Room. Air handling unit two (0C-AHU-2) provides cooling using outside ambient air to provide ventilation to the Fuel Tank Room, Switchgear Room, Diesel Generator Room, Battery Room, and Cable Spreading Area. The Cable Spreading Area, Diesel Generator Room and the Fuel Tank Room are exhausted using an augmented quality exhaust fan (0C-F-6). However, the Battery Room has a separate augmented quality exhaust fan (0C-F-5).

9.8.2.4 Turbine Building

When Units 1 and 2 are both in operation, the Turbine Building requires cooling all year round. Outside air is used for this purpose since it will normally be at least 15 degrees below the Turbine Building maximum design air temperature. Twelve fans, one in each vertical air shaft at Elevation 95'0", supply air from mixing boxes through ducting to all levels of the Turbine Building and heater bay area. The selection of fresh air or recirculated air, which is accomplished by means of dampers located in the mixing box, is manually controlled for each fan by a switch located on the operating floor level. The walls of the vertical shafts are louvered above the fan level, thus permitting them to serve as an intake plenum. All Turbine Building ventilating fans are manually controlled. Plant operations personnel determine the number in operation at any one time.

Exhaust dampers located near the center of each horizontal air shaft are blocked open except as follows.

Horizontal air shafts 3 and 4 (two in center) have two exhaust fans mounted in each shaft in place of the relief dampers to exhaust approximately 80% of the air supplied by the intake fans. The fans are manually controlled. The four exhaust dampers adjacent to these four exhaust fans are blocked shut to prevent exhaust flow back into the building.

A mechanical cooling system is installed to provide local cooling at the four heater drain pump motor locations. Four fan coil units, located on Elevation 27' above each pump motor, are supplied chilled water from two air-cooled water chillers located at the north end of the Turbine Building. Air from the fan units is directed downward over the pump motors by removable ductwork. The fans are manually controlled. The chillers have self-contained controls and are turned on by flow switches in the chilled water lines. The chilled water pump is manually controlled.

Hot water unit heaters maintain a minimum temperature of 60°F at the operating floor level during a shutdown period. Unit heater fans are controlled by thermostats located on the operating floor level.

9.8.2.5 Service Building

The administrative area of the service building is air-conditioned all year round. All administration area rooms have individual temperature control.

The warehouse and shop areas at Elevation 45'0" are ventilated all year round by a single makeup-air unit. The unit provides both mechanical cooling and electric heat for year round operation. Unit heaters supplement the makeup air unit to maintain 60°F minimum in the winter time. Room thermostats control the

operation of the unit heater fans. The lower levels of the service building are ventilated only with fresh air being tempered to 60°F. The two 250 Volt battery rooms (Elevations 12' and 35') are cooled with separate mechanical refrigeration cooling systems.

9.8.2.6 Intake Structure

The amount of heat generated by the saltwater and circulating water pump motors in the intake structure is such that a cooling system is necessary year round. Six air supply units consisting of weather louvers and filter modules capable of high moisture separation efficiency, and a supply fan are located on the six saltwater pump hatches. These supply units force fresh air into the intake structure. Room thermostats control the fans' motors. After absorbing the heat from the saltwater pump motors and the circulating water pump motors, the air exits the intake structure through exhaust vents located on the twelve circulating water pump hatches. These exhaust vents consist of weather louvers capable of high moisture separation efficiency. This system limits the intake structure ambient temperature to approximately 104°F when the outdoor ambient temperature is 95°F. A minimum amount of outside air for ventilation enters through ducts at six locations and is exhausted through ducts by two fans located in the east wall of the service building. Hot water unit heaters prevent freeze-up during periods of complete shut down.

If these air supply units were to fail, the increase in temperature could cause the circulating water pump motors to overheat. Therefore, an over-temperature light and alarm are incorporated into the control board located in the Control Room.

The six fan units in the Intake Structure are not required for continuous satisfactory operation of two saltwater pumps per unit. Natural air circulation, cooling effect from the Intake Structure walls and cooling from the saltwater piping will provide sufficient heat removal for the saltwater pump motors.

9.8.3 CODES AND STANDARDS

The work, equipment and materials conform to the requirements and recommendations of the following codes and standards, as applicable:

National Fire Protection Association Code, Pamphlet 90A
National Electric Code
American Society of Heating, Refrigerating and Air-Conditioning Engineers Guides
Air Moving and Conditioning Association, Inc.
American Society of Mechanical Engineers
Sheet Metal and Air-Conditioning Contractors National Association, Inc.
American Society of Testing Materials

9.8.4 TESTS AND INSPECTIONS

All equipment is accessible for inspection. Testing and performance-indicating instruments are built into each critical apparatus to increase the maintainability and reliability of these systems. Where redundant equipment is provided, it will be operated alternately to provide increased assurance of operability (as required by the Technical Specifications).

In addition, the filters (HEPA and charcoal) are subjected to testing similar to that of the Containment Iodine Removal System filters described in Section 6.7.

9.8.5 REFERENCES

1. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated September 3, 1982, Review of NUREG-0737 Item III.D.3.4, Control Room Habitability
2. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), dated June 22, 1995, Control Room Habitability Interim Analysis for Thyroid Dose
3. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), dated August 29, 2007, Amendment re: Implementation of Alternative Radiological Source Term

TABLE 9-18

PLANT VENTILATION SYSTEM DESIGN CONDITIONS

SYSTEM	TYPE SYSTEM[c]	SUMMER (°F) INSIDE	SUMMER (°F) OUTSIDE	WINTER (°F) INSIDE	WINTER (°F) OUTSIDE	MAIN PLANT VENT FLOW CAPABILITY UNIT 1 (cfm)	MAIN PLANT VENT FLOW CAPABILITY UNIT 2 (cfm)
Turbine Building	HV	110	95[i]	60	0		
Containment Cooling	AC	120	95[i]	60	0		
Pressurizer Compartment	AC	NA	NA	NA	NA		
Cavity Cooling	AC	NA	NA	NA	NA		
CEDM Cooling	AC	NA	NA	NA	NA		
Purge System[a]	V	NA	NA	60, 45[j]	0	50,000	50,000
Pipe Penetration Rooms[b]	V	130	NA	NA	NA	2,000	2,000
Auxiliary Building							
Auxiliary Feedwater Pump Room	HVAC	90	NA	NA	NA		
Control Room	HVAC	75	95[i]	75	0		
Cable Room	HVAC	90	95[i]	75	0		
Access Control Area						NA	13,900
Health Physicist	HVAC	75	95[i]	75	0		
Hot Laboratory	HVAC	75	95[i]	75	0		
Other Controlled Rooms	HV	NA	NA	75	0		
Clean Rooms	HV	NA	NA	75	0		
Main Steam Pen. Areas	V	160	95[i]	60	0		
Switchgear Rooms	HVAC	104	95[i]	104	0		
Diesel Generator Rooms	HV	120	95[i]	60	0		
Spent Fuel Pool	HV	110	95[i]	60	0	32,000	NA
Radwaste Area	HV	110	95[i]	60	0	49,500	49,500
ECCS Pump Room	HVAC	110	95[i]	60	0	3,000	3,000
Intake Structure	HVAC	104	95[i]	104	0		
Service Building							
Office Area	HVAC	75	95[i]	75	0		
Locker Room	HV	NA	95[i]	80	0		
Warehouse	HV	110	95[i]	60	0		
Shop	HV	110	95[i]	60	0		

TABLE 9-18

PLANT VENTILATION SYSTEM DESIGN CONDITIONS

SYSTEM	TYPE SYSTEM[c]	SUMMER (°F) INSIDE	SUMMER (°F) OUTSIDE	WINTER (°F) INSIDE	WINTER (°F) OUTSIDE	MAIN PLANT VENT FLOW CAPABILITY UNIT 1 (cfm)	MAIN PLANT VENT FLOW CAPABILITY UNIT 2 (cfm)
EDG 1A Building							
Battery Room	HVAC	104	95[i]	69	0		
1E Switchgear Room	HVAC	104	95[i]	50[e]	0		
EDG Building Control Room	HVAC	104	95[i]	50	0		
Non-1E Electrical Panel Room	HVAC,V	104	95[i]	50[e][h]	0		
EDG Fan Room	H,V	120	95[i]	50	0		
All Other Rooms Below 3rd Floor	HV	120[d]	95[f,i]	50[e]	0		
Third Floor	V	104[f]	95[i]	0	0		
						136,500	118,400

[a] In operation only when containment is occupied

[b] Operated intermittently as required and during LOCA

[c] H = Heating only
V = Ventilation only
AC = Air Conditioning (cooling) only

[d] The hallway, Maintenance Shop, and Future Expansion Room may reach a maximum temperature of 150°F when the diesel is in operation.

[e] Minimum temperature may be lower in the 1E Switchgear Room, Fuel Oil Storage Tank Room, hallway, Non-1E Panel Room, Maintenance Shop, and the Future Expansion Room during diesel operation under accident conditions concurrent with design basis winter temperatures.

[f] Downstream of the radiators on the third floor, the maximum design temperature may reach 140°F during diesel operation.

[g] Deleted.

[h] 1E Switchgear Room and Non-1E Electrical Panel Room may experience temperatures no lower than 32°F during accident conditions with design basis outside temperatures.

[i] These temperatures reflect the ventilation or AC system design temperature (95°F dry bulb), as recommended in the American Society of Heating Refrigeration and Air Conditioning Engineers Guide of regional design conditions.

[j] Applicable only when Unit 1(2) is in Mode 5 or 6. When defueled, applicable with an RCS vent path of at least 8 in^2 available.

TABLE 9-19

AUXILIARY BUILDING VENTILATION CHARCOAL FILTERS

FILTER NAME AND LOCATION	FLOW RATE (cfm)	APPROX. WT OF CHARCOAL (lbs)
Spent Fuel Pool #11	32,000	4,224
Penetration Room Exh. #11[a]	2,000	264
Penetration Room Exh. #12[a]	2,000	264
Penetration Room Exh. #21[a]	2,000	264
Penetration Room Exh. #22[a]	2,000	264
ECCS Pump Room Exh. #11	3,000	396
ECCS Pump Room Exh. #21	3,000	396
Post-LOCI #11 (Control Room)[a]	10,000	1,500
Post-LOCI #12 (Control Room)[a]	10,000	1,500

[a] These charcoal filter units are tested in accordance with Technical Specification 5.5.11, Ventilation Filter Testing Program.

9.9 CALVERT CLIFFS NUCLEAR POWER PLANT FIRE PROTECTION PROGRAM

9.9.1 INTRODUCTION

The Fire Protection Program at the Calvert Cliffs Nuclear Power Plant (CCNPP) is the integrated effort involving systems, structures, components, procedures, and personnel used to carry out all activities of fire protection, fire prevention, and to ensure safe shutdown following a fire event. The program provides the necessary controls to protect the health and safety of CCNPP workers and the general public, satisfy NRC and Insurer requirements, meet applicable State of Maryland codes and standards and safeguard Company assets by preventing fires and minimizing the consequences of any fire that may occur.

The Fire Protection Program has been developed in accordance with the documents listed in Section 9.9.12 (References 1 through 19).

9.9.2 DESIGN BASIS

The Fire Protection Program uses a Defense-In-Depth concept to achieve a high degree of fire safety at CCNPP. The Defense-In-Depth concept achieves the following objectives: prevents fires from starting; detects and quickly controls and suppresses those fires that occur, while limiting damage; and ensures design of plant safety systems and passive barriers such that fire will not prevent essential plant safety functions from being performed.

Elements of the fire protection program at CCNPP include the following:

a. Organization; Responsibility, Authority and Qualifications
b. Administrative Controls and Procedures
c. Fire Brigade; Composition, Responsibility, and Training
d. Fire and Emergency Response Activities
e. Fixed and Manual Fire Protection Features
f. Passive Fire Protection Features
g. Fire Protection Controls and Compensatory Measures
h. Safe Shutdown Capability
i. Fire Protection Quality Assurance Program

9.9.3 ORGANIZATION; RESPONSIBILITY, AUTHORITY, AND QUALIFICATIONS

The line of authority for those positions of responsibility for the Fire Protection Program are defined in an administrative procedure and the Quality Assurance Topical Report. Specific qualification requirements for the individuals responsible for the Fire Protection Program are also delineated in an administrative procedure.

9.9.4 ADMINISTRATIVE CONTROLS AND PROCEDURES

Administrative controls associated with the Fire Protection Program are provided through several administrative procedures and the Quality Assurance Topical Report. Administrative requirements provide controls for activities that could affect fire protection, including the following:

a. In-situ and transient combustibles;
b. Ignition sources;
c. Hot work activities;
d. Smoking;

e. Design, maintenance, and plant modification processes; and
f. Surveillance of fire protection systems and equipment.

9.9.5 FIRE BRIGADE; COMPOSITION, RESPONSIBILITY, AND TRAINING

Calvert Cliffs Nuclear Power Plant has a designated fire brigade of at least five members available on each shift. Members of the fire brigade do not include the minimum operations shift crew necessary for safe shutdown of both units. The fire brigade is trained and equipped to respond to fire-related emergencies at CCNPP. A mutual aid agreement has been established with off-site fire departments to provide assistance to the plant fire brigade on an as-needed basis.

The on-shift fire brigade will be staffed with a fire brigade leader and at least two fire brigade members that have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppression on nuclear safety performance criteria. The sufficient training and knowledge is permitted to be provided by an Operations Advisor dedicated to respond with the fire brigade.

Fire brigade members are required to be physically fit and adequately trained before they may be assigned to the brigade. Initial training and periodic refresher training and drills of the fire brigade members includes both classroom instruction and hands-on training. The requirements for becoming a member and maintaining membership on the fire brigade are contained in an administrative procedure. Training requirements for the fire brigade, including the type of training, training topics, and training attendance is also contained in an administrative procedure.

The fire brigade is provided with approved fire fighting protective equipment, including turnout gear and self-contained breathing apparatus. Additional fire-fighting equipment is available, such as: hoses, nozzles, smoke ejectors, foam-making equipment, and other specialized tools. Minimum quantities of fire-fighting equipment are identified by plant procedures. Plant procedures also provided for periodic inspection and testing of fire-fighting equipment.

9.9.6 FIRE AND EMERGENCY RESPONSE ACTIVITIES

In the event of a fire, response actions by designated plant personnel are defined in plant administrative procedures. Notification and instructions are given to all plant personnel over the plant public address system by Control Room operations personnel.

Fire-fighting strategies and tactics have been developed for the fire brigade for those plant areas which contain safe shutdown equipment and cabling. The fire brigade is provided with sufficient equipment to combat a fire anywhere in the plant. Mutual aid agreements have been established with off-site fire departments to provide assistance if requested by the plant Fire Brigade Leader.

Should a fire be of such magnitude as to require shutting down the reactor(s), site contingency plans have been established. Plant procedures have been developed to bring the plant safely to a cold shutdown condition in accordance with the requirements of 10 CFR Part 50, Appendix R, Sections III.G and III.L in the event of a severe fire in those areas where the loss of components would require the use of alternate safe shutdown methods. Safe shutdown in the event of a fire in other plant areas can be accomplished using other plant procedures.

9.9.7 FIXED AND MANUAL FIRE PROTECTION FEATURES

Both units are served by a common fire protection system. Fire protection is provided by means of the following systems:

a. Fire water supply
b. Fire pumps and distribution piping
c. Fixed water suppression systems
    1. Automatic deluge water spray
    2. Automatic pre-action sprinklers
    3. Manual pre-action sprinklers
    4. Automatic wet pipe sprinklers
    5. Manual wet pipe sprinklers
    6. Automatic dry pipe sprinklers
    7. Manual foam
d. Fixed gaseous suppression systems
    1. Halon 1301
e. Manual fire suppression systems
    1. Indoor hose stations and outdoor hydrants/hose cabinets
    2. Portable extinguishers
f. Fire detection and alarm systems
    1. Smoke detectors
    2. Heat detectors
    3. Flame detectors

Each of the fire suppression and detection systems were designed and installed following the guidance in the applicable National Fire Protection Association (NFPA) standards. The NFPA code of record for the plant fire protection systems is the edition which was in effect at the time the system was initially designed. Any significant deviation in the design or installation of a fire protection system from the code requirements is supported by an engineering evaluation that documents the acceptability of the non-code-compliant condition.

See Table 9-20 for a list of plant areas provided with fire suppression and detection systems. Figure 9-22 provides a simplified system drawing of the fire protection system.

9.9.7.1 <u>Fire Water Supply</u>

The fire protection water supply originates from three wells located on site. The water is supplied by three well water pumps to two 500,000 gallon capacity (pretreated) water storage tanks located at the Fire Pump House. The layout of the discharge piping from the tanks is such that a minimum of 300,000 gallons (each tank) is always available to the fire protection system. The 300,000 gallons provides a 2-hour supply for the largest demand suppression system (diesel generator rooms), plus 1,000 gpm for manual hose streams. The remaining 200,000 gallons (each tank) may be used for other services and is also available for fire protection system supply backup. The well pumps have the capacity to replenish the minimum required 300,000 gallons to one of the storage tanks within 8 hours.

The tanks can be isolated in the case of a pipe rupture, to ensure that at least one tank is available. The pipe interconnecting the two tanks is provided with a normally locked-closed valve to preclude the inadvertent drain down of both tanks. Each of the tanks is also equipped with low-level alarms (less than 303,000 gallons) which annunciate in the Control Room and locally.

9.9.7.2 Fire Pumps and Distribution Piping

Water for the plant fire suppression systems is supplied by two full-capacity fire pumps, each rated at 2500 gpm at 125 psig. One pump is electrically-driven and the other is diesel engine-driven. A jockey pump rated at 30 gpm at 129 psig is provided to maintain a pressure of 115 to 125 psig in the fire protection water system under normal no-use conditions. A makeup pump rated at 215 gpm at 125 psig takes suction from a plant service water main and discharges to the fire protection system to meet the intermittent usage of water for the purposes other than fire protection. The diesel engine-driven fire pump is supplied with 8 hours of fuel from a nominal 500 gallon fuel tank located in the Fire Pump House. All of the pumps, as well as their controllers, are listed for fire protection use by an independent testing laboratory (such as Underwriters Laboratory, Inc.).

The electrically-driven pump starts automatically on a low-header pressure of 95 psig with the diesel engine-driven pump being started at 85 psig. The diesel engine-driven pump is thus arranged to provide backup for the electrically-driven pump in case the latter does not start or does not maintain adequate pressure at the header. The diesel engine-driven pump also starts automatically if electric power is interrupted to the electrically-driven pump. The two fire pumps can also be started manually from the Control Room; however, they can only be stopped locally at the Fire Pump House.

The jockey pump is provided to automatically maintain pressure in the system, thus eliminating the need for the main fire pumps to maintain system pressure. Excessive pressure developed at the discharge side of the main fire pumps is relieved through pressure regulating valves. These valves, along with bypass lines on the wet pipe sprinkler system alarm check valves located in the lower elevations of the plant, prevent over-pressurizing the fire water distribution system.

The electrically-driven centrifugal makeup fire pump is located in a sprinkler area of the Unit 1 Turbine Building basement. An administrative procedure establishes control for use of the fire system for purposes other than fire-fighting by limiting use to a single 1-1/2” hose stream and use of the makeup pump. This restriction to a single 1-1/2” hose applies to all non-fire protection use of the fire protection water supply unless evaluated and approved by the site fire protection engineer and documented in a procedure. Emergency, non-fire use, of the fire system is considered separate from the intermittent use discussed here (Section 10.3.2).

The fire pumps and jockey pump are located in the Fire Pump House, an independent structure located north of the Turbine Building. The Fire Pump House is protected by an automatic sprinkler system. All working parts of the electric fire pump are enclosed in a sturdy drip-proof sheet-steel cabinet for protection from the sprinkler systems. In addition, the diesel fuel oil tank is provided with a dike, sized to contain the entire contents of the tank. Therefore, the layout and fire protection features installed in the pump house will provide protection for the electrically-driven fire pump from a fire involving fuel oil. The configuration of both of the main fire pumps located in the same area will not prevent the plant reaching a safe shutdown condition in the event of a fire in the Fire Pump House.

Each fire pump discharges into the yard main through a 12” diameter underground line. An isolation valve is provided between the two points at which the fire pump discharge lines connect to the yard main so that in the event of the failure of one of the pumps, the other pump is still available.

The fire yard main loop consists of 12“ cement-lined iron piping and completely surrounds the plant. Post indicator type valves are provided to sectionalize the main for testing and maintenance purposes. The yard main is cross-connected by distribution piping that is routed through the plant structures using carbon steel pipe. The distribution piping supplies the various fire protection systems and provides alternate paths for water flow should any portion of the fire main become disabled. All of the water-based suppression systems in the power block structures and the yard hydrants within the Protected Area of the plant are supplied by this fire protection water supply system. This system also supplies fire protection water to the warehouses.

A separate fire protection system and yard main encircles the Nuclear Security Facility/Nuclear Office Facility and a single cross-connection is provided to the main plant fire protection system through a normally locked-closed valve. Should the fire protection system inside the Protected Area become disabled, opening this valve permits these fire pumps and associated water supply to provide a back-up to the Protected Area fire protection water supply system. This system is not covered by the plant Quality Assurance program and no credit for its availability is assumed in the plant fire protection design basis.

9.9.7.3 Fixed Water Suppression Systems

Fixed water suppression systems consist of several different types of systems including deluge systems, pre-action, wet pipe, dry pipe sprinkler, manual, and foam systems. The systems are automatically actuated with the exception of the sprinkler systems protecting the main turbine bearings, the foam systems, and Cable Chase 1A, 1B, 2A and 2B.

Deluge water spray system piping is normally dry. These systems are automatically actuated by an associated heat detection system. These systems are installed to provide protection for equipment containing significant quantities of oil. In addition, the systems are provided with open head sprinklers, thus water flows from all of the sprinklers upon actuation of the system’s deluge valve. The main shutoff valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Pre-action sprinkler system piping is normally dry. These systems are automatically actuated by an associated heat detection system that allows water into the system piping. In addition, the systems are provided with fusible head sprinklers which operate only when exposed to high temperatures. These systems are installed to provide general area protection (except for the turbine bearing systems which are hazard specific). The main shutoff valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Wet pipe sprinkler system piping is normally water-filled. These systems are provided with fusible head sprinklers which operate only when exposed to high temperatures. These systems are installed to provide general area protection. The main shutoff valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Dry pipe sprinkler system piping is normally dry. These systems are provided with fusible head sprinklers which operate when exposed to high temperatures. These systems are installed to provide general area protection. The main shutoff valve

for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

The foam systems provide protection for the two outdoor fuel oil storage tanks. The foam system piping is normally dry. The systems are designed to be operated manually in the event of a fire. The foam concentrate storage tank is located on the west side of the plant.

9.9.7.4 Fixed Gaseous Suppression Systems

These systems are automatically actuated by detection systems located in the protected rooms. Upon actuation, the systems distribute Halon 1301 throughout the Protected Area via system piping. In addition, air flow in and out of the room is isolated prior to system discharge so that the Halon concentration is maintained within the protected room. Actuation of each of these systems is annunciated in the Control Room.

9.9.7.5 Manual Fire Suppression Systems

These systems consist of standpipes/hose stations supplied with water from the fire protection water supply system. The standpipes/hose stations are located throughout the plant in permanent structures. The typical standpipe/hose station system consists of two 2-1/2" hose connection outlets. Each of the standpipes/hose stations is also provided with a universal spanner wrench. The standpipes/hose stations are spaced at approximately 100' intervals, located on all building elevations, and arranged to reach all safety-related components in the plant. Standpipes/hose stations in the plant were installed to meet the requirements of a Class I system, as defined in NFPA 14.

Yard hydrants have been provided at intervals of approximately 200' to 300' around the exterior of the plant. Each hydrant hose cabinet is provided with appropriate fire fighting equipment for use by the plant fire brigade.

Portable fire extinguishers are provided at convenient and readily accessible locations throughout the plant. The extinguishing agents utilized are appropriate for the service requirements of the area. The portable fire extinguishers are located and installed following the guidance of NFPA 10.

9.9.7.6 Fire Alarm And Detection System

A fire detection system consisting of various types of smoke and fire detectors is provided. Smoke detection includes both ionization and photoelectric type detectors. A majority of the detectors provide an "alarm only" function, however, there are several smoke detector sub-system installations which also cause actuation of an associated fixed suppression system. A third type of smoke detector installed at CCNPP, beam type detectors, are installed in the Calvert Cliffs Independent Spent Fuel Storage Installation warehouse and weld shop. Detectors are typically provided in areas which contain safe shutdown and safety-related components, except in the Turbine Building, certain high radiation areas and exterior structures in the yard.

Heat detection consists of both spot-type detectors and line-type detectors. Spot-type detectors consist of one of three types: rate-of-rise, fixed temperature, or a combination of the two types (rate-compensated). The spot-type detectors installed in the plant are generally installed as part of a fixed suppression system. These detectors cause the suppression system to actuate as well as transmit an

alarm to the Control Room. The line-type detectors, which are installed in several cable trays in the containment buildings, provide an alarm-only function.

Infra-red type flame detectors are installed in several areas of the Auxiliary Building where smoke detection is not appropriate. The flame detectors provide an alarm-only function.

Each of the detectors automatically transmits an alarm to the Control Room. An audible-visual alarm system is provided in the Control Room with annunciator windows to warn of the occurrence of the following conditions: fire, fire-alarm system trouble (includes valve supervision), electrical fire pump operation, diesel fire pump operation, and fire pump trouble. In addition, there are annunciator windows to designate the affected area. An audible alarm which is distinctive from other Control Room alarms is also provided.

9.9.8 PASSIVE FIRE PROTECTION FEATURES

9.9.8.1 Fire Barriers

The plant fire barriers (i.e., walls, floors, and ceilings) provide separation for redundant safe shutdown equipment and/or cabling. These designated fire barriers have been established based upon the results of the electrical separation analysis performed for 10 CFR Part 50, Appendix R. The plant barriers, which have fire resistance ratings supported by fire testing, are designated as either 1-hour, 2-hour, or 3-hour fire-rated barriers.

9.9.8.2 Penetrations

Fire-rated penetrations at CCNPP consist of fire doors, fire dampers, or penetration seal assemblies. Each of the fire-rated doors and dampers are listed for the appropriate fire resistance rating by an independent testing laboratory.

Fire barrier seal designs for electrical and piping penetrations at CCNPP were subjected to fire testing at an independent testing laboratory. Where penetration seal configurations within the plant are not bounded by a fire test, an engineering evaluation has been performed to document acceptability. The penetration seals in barriers requiring a fire rating are of a commensurate fire rating as the barrier itself.

9.9.9 FIRE PROTECTION CONTROLS AND COMPENSATORY MEASURES

The fire protection program is designed to assure that adequate levels of protection are available at all times. This is accomplished by established controls and compensatory measures, and through surveillance and testing procedures, as well as requiring documenting of non-compliances when fire protection features are determined to be inoperable.

Fire protection equipment and systems are inspected and tested upon initial installation and periodically thereafter. The inspection and testing is conducted following the guidance of applicable NFPA Codes and Standards as well as recommendations and requirements of the insurance carrier and the NRC. Plant procedures mandate test frequencies and the testing process. Applicability, compensatory actions, testing requirements, and testing frequencies for those fire protection systems which protect safe shutdown and safety-related equipment are contained in the CCNPP Technical Requirements Manual. Plant procedures also identify compensatory actions to be taken when equipment required for Appendix R safe shutdown actions becomes inoperable.

9.9.10 SAFE SHUTDOWN CAPABILITY

Title 10 CFR 50.48, “Fire Protection,” identifies 10 CFR Part 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” as establishing the requirements for fire protection. This section also states that only Sections III.G, III.J, and III.O apply to plants operating prior to January 1, 1979 (both of the Calvert Cliffs units were operating prior to 1979). Section III.G, “Fire Protection of Safe Shutdown Capability,” requires consideration of Section III.L, “Alternate and Dedicated Shutdown Capability” if alternate or dedicated shutdown is necessary to comply with III.G. Section III.J, addresses “Emergency Lighting” and Section III.O addresses “Oil Collection System for Reactor Coolant Pumps.”

9.9.10.1 Fire Protection of Safe Shutdown Capability

Calvert Cliffs Nuclear Power Plant utilizes both redundant and alternate methods of shutdown, as defined by Appendix R. Detailed post-fire shutdown procedures have been developed for those rooms that require alternate shutdown capability. These procedures ensure that the post-fire shutdown of the reactor will achieve and maintain subcritical reactivity conditions, maintain reactor coolant inventory, achieve hot shutdown conditions using equipment free of fire damage, and achieve and maintain cold shutdown within 72 hours of the fire event.

Appendix R permits post-fire repair of components necessary to achieve cold shutdown. Post-fire shutdown procedures identify where repairs can be utilized to achieve cold shutdown. Components necessary to make these repairs are available on site.

A detailed analysis of fire effects in those rooms that contain equipment necessary to achieve and maintain both hot shutdown and cold shutdown has been performed and is contained within a plant document (References 4 and 5). This analysis considers the loss of the components within the room and identifies how to mitigate the loss of each component. This analysis also identifies the minimum list of equipment, the shutdown methodology, and specific manual actions necessary to mitigate the loss of components. This analysis identifies certain assumptions that form the basis for the analysis.

Exemption Requests from certain requirements of Appendix R for specific rooms and areas have been approved by the NRC Staff (References 14, and 16 through 19). In addition, Fire Protection Engineering Evaluations have been prepared, in accordance with Generic Letter 86-10 to address fire barrier issues. These evaluations are retained by the Fire Protection Engineer and are available for review.

An Alternate Shutdown Panel has been provided for each reactor unit in a room that is separate from the Control Room. This Alternate Shutdown Panel contains controls and instrumentation, including source range flux monitoring. This panel is free of fire effects, or can be made free of fire effects by operator actions, following a fire in the Control Room.

Fire area boundary information regarding fire barriers is available in the Fire Hazards Analysis (Reference 9) and supporting documentation.

9.9.10.2 Emergency Lighting

Emergency lighting is provided for those areas that require plant operators to perform actions while completing the post-fire shutdown procedures. Emergency lighting typically consists of 8 hour self-contained battery units located in or near

the area where needed. In certain cases, other methods of providing illumination for the operators are used. Additionally, portable lighting, redundant to the fixed lighting, is provided to the operators to ensure that adequate emergency lighting is available.

9.9.10.3 Oil Collection System for Reactor Coolant Pumps

Each of the RCPs are provided with an oil collection system. This system is designed to capture any oil that could leak from the RCP. The system then drains the oil away to a safe location where it will not be subjected to an ignition source.

9.9.11 FIRE PROTECTION QUALITY ASSURANCE PROGRAM

Design, procurement, installation, testing, inspection, maintenance, modification, non-conforming items, corrective actions, audits, and administrative controls for the Fire Protection Program shall be controlled by the quality assurance program documents. Activities related to fire protection are performed within the applicable provisions of the Quality Assurance Topical Report based on 10 CFR Part 50, Appendix B and in accordance with the quality assurance guidance in BTP 9.5-1, Appendix A and the NRC's guidance document, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

9.9.12 REFERENCES

1. Appendix A to BTP APCSB 9.5-1 - Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976
2. 10 CFR 50.48 - Fire Protection
3. 10 CFR Part 50, Appendix R - Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979
4. Interactive Cable Analysis for Calvert Cliffs Nuclear Power Plant - Unit 1
5. Interactive Cable Analysis for Calvert Cliffs Nuclear Power Plant - Unit 2
6. Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance
7. Generic Letter 86-10 - NRC Fire Protection Policy
8. Generic Letter 88-12 - Removal of Fire Protection Requirements From Technical Specifications
9. Fire Hazards Analysis Summary
10. Safety Evaluation Report for Fire Protection, dated September 14, 1979
11. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated October 2, 1980, First Supplement to the Fire Protection Safety Evaluation Report
12. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated March 18, 1982, Second Supplement to the Fire Protection Safety Evaluation Report
13. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated August 6, 1982, Request for BGE to Commit to Install Source Range Flux Monitoring at the Alternate Shutdown Panel
14. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated August 16, 1982, Calvert Cliffs Units 1 and 2 - Fire Protection Exemption Request

15. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated September 27, 1982, Third Supplement to the Fire Protection Safety Evaluation Report

16. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated April 21, 1983, Exception Request of March 4, 1983 - Fire Suppression Requirements of Section III.0.2 of Appendix R to 10 CFR Part 50 - Calvert Cliffs Nuclear Power Plant, Units 1 and 2

17. Letter from Mr. J. R. Miller (NRC) to Mr. A. E. Lundvall, Jr. (BGE), dated March 15, 1984, Calvert Cliffs Units 1 and 2 - Fire Protection Exemption Request

18. Letter from Mr. S. A. Varga (NRC) to Mr. G. C. Creel (BGE), dated August 22, 1990, Issuance of a Technical Exemption from the Requirement of 10 CFR Part 50, Appendix R, for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, TAC Nos. 76127 (Unit 1) and 76128 (Unit 2)

19. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated April 7, 1999, Exemption from the Requirements of 10 CFR Part 50, Appendix R, Section III.J, Emergency Lighting -- Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M99791 and M99792)

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

A legend of the abbreviations is provided at the end of the table.

A. FIRE PROTECTION COMPONENT DESCRIPTION

Fire Pump, Electrically-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	2,500 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	250 hp/460 Volts/3 phase/60 Hz
Codes	U.L. Label
	Motor: NEMA
	Pump: Standards of the Hydraulic Institute

Fire Pump, Diesel Engine-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	2500 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Engine	283 hp
Codes	Pump: Standards of the Hydraulic Institute

Fire System Jockey Pump

Type	Horizontal Centrifugal
Number	1
Capacity	30 gpm
Discharge Press.	129 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	7-1/2 hp/460 Volts/3 phase/60 Hz
Codes	Motor: NEMA
	Pumps: Standards of the Hydraulic Institute

Makeup Pump, Electrically-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	215 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	40 hp/460 Volts/3 phs/60 Hz
Codes	Motor: Underwriters
	Label, NEMA
	Pumps Standards of the Hydraulic Institute

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

Piping, Fittings and Valves

	Underground	Aboveground
Material	Cast Iron	Carbon Steel[a]
Design Pressure	150 psig	175 psig
Design Temperature	100°F	100°F
Construction	Mechanical Joint	Welded and Screwed
Valves	Cast Iron	Cast Iron[a][c]
	Mechanical Joint	Flanged[b]
	175 psi	175 psi[c]
	U. L. Label	U. L. Label

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
B. LOCATION OF DETECTION AND SUPPRESSION SYSTEMS			
AUXILIARY BUILDING			
100/103/104*/116	Corridors - Elevation 10’0”	WP	POC
101	No. 21 ECCS Pump Room, Unit 2	WP	POC
102	No. 22 ECCS Pump Room, Unit 2		POC
105	Charging Pump Room, Unit 2	WP	POC
106	Miscellaneous Waste Monitor Tank	WP	POC
107/109	Coolant Waste Monitor Tank		FL
108	Waste Processing Room		POC
110	Coolant Waste Receiver/Monitor Tank Pump Room		POC
111	Cryogenics - Waste Processing Control Room		POC
112/114	Coolant Waste Receiver Tank		FL
113	Miscellaneous Waste Monitor Tank Room	WP	POC
115	Charging Pump Room, Unit 1	WP	POC
118	No. 12 ECCS Pump Room, Unit 1		POC
119	No. 11 ECCS Pump Room, Unit 1	WP	POC
120	Containment Recirculation Pipe Tunnel, Unit 2	WP*	POC
122	Containment Recirculation Pipe Tunnel, Unit 1	WP*	POC
200/202/212/219	Corridors	WP	POC
201	Component Cooling Pump Room, Unit 2	WP	POC
203	East Piping Area, Unit 2	WP	POC
204	Radiation Exhaust Vent Equipment Room, Unit 2	WP	POC
205	SRW Pump Room, Unit 2	WP	FL, POC
206/310	East Piping Penetration Room, Unit 2	WP	FL, POC
207/208	Waste Gas Equipment Room		POC
209/210	Decontamination/RCP Rebuild Area	WP	POC
211/321	West Piping Penetration Room, Unit 2		FL, POC
213	Degasifer Pump Room, Unit 2		POC
214	VCT Room, Unit 2		POC
215	Boric Acid Tank & Pump Room, Unit 2	WP	POC
216	Reactor Coolant Makeup Pumps	WP	POC
216A	Reactor Coolant Makeup Pumps	WP	POC
217	Boric Acid Tank & Pump Room, Unit 1	WP	POC
218	VCT Room, Unit 1		POC
220	Degasifer Pump Room, Unit 1		POC
221/326	West Piping Penetration Room, Unit 1		FL, POC
222	RCP Seal Rebuild Room	WP	POC
223	Hot Machine Shop	WP	POC
224	East Piping Area, Unit 1	WP	POC
225	Radiation Exhaust Vent Equipment Room, Unit 1	WP	POC
226	SRW Pump Room, Unit 1	WP	FL, POC
227/316	East Piping Penetration Room, Unit 1	WP	FL, POC
228	Component Cooling Pump Room, Unit 1	WP	POC

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
300/303	Corridors		POC
301/304	Battery Rooms, Unit 1		POC
302/2C	Unit 2 Cable Spreading Room & Cable Chase 2C	H-1	HT, POC, DD
305/307	Battery Rooms, Unit 2		POC
306/1C	Unit 1 Cable Spreading Room & Cable Chase 1C	H-1	HT, POC, DD
308	Corridor		POC
309	Main Steam Piping Area, Unit 2	WP	POC
311	Switchgear Room, Unit 2, Elevation 27’	H-1	POC, DD
312	Purge Air Supply Room, Unit 2		POC
315	Main Steam Piping Area, Unit 1	WP	POC
317	Switchgear Room, Unit 1, Elevation 27’	H-1	POC, DD
318	Purge Air Supply Room, Unit 1		POC
319/325	West Passage & Vestibule		POC
320	Spent Fuel Cooling Pump Room		POC
322	Letdown Heat Exchanger Room, Unit 2		POC
323	27’0” Valve Alley and Filter Room		POC
324	Letdown Heat Exchanger Room, Unit 1		POC
1A	Cable Chase 1A	MW	POC
1B	Cable Chase 1B	MW	POC
2A	Cable Chase 2A	MW	POC
2B	Cable Chase 2B	MW	POC
405	Control Room		POC, DD
406	DAS Computer Room, Unit 2	H-1, H-2	POC
407	Switchgear Room, Unit 2, Elevation 45’	H-1	POC, DD
408	Piping Area, Unit 2	WP	POC
409	East Electrical Penetration Room, Unit 2	WP	POC
410	North/South Corridor		POC
414	West Electrical Penetration Room, Unit 2	WP	POC
416	Diesel Generator No. (2B)	PA	HT
417/418	Solid Waste Processing	WP	POC
413/419/424/425/426	Cask & Equipment Loading Area	WP	FL, POC
420	Reactor Coolant Waste Evaporator Room	WP*	POC
421	Diesel Generator No. (1B)	PA	HT
422	Diesel Generator No. (2A)	PA	HT
423	West Electrical Penetration Room, Unit 1	WP	POC
428	Piping Area, Unit 1	WP	POC
429	East Electrical Penetration Room, Unit 1	WP	POC
430	Switchgear Room, Unit 1, Elevation 45’	H-1	POC, DD
431	DAS Computer Room, Unit 1	H-1, H-2	POC
432	Technical Support Center Computer Room		POC
436	Technical Support Center		POC
437	Technical Support Center Annex		POC
439	Refueling Water Tank Pump Room, Unit 1		POC
440	Refueling Water Tank Pump Room, Unit 2		POC

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
442	Reserve Battery Room		POC
444	Central Alarm Station		POC
512	Control Room HVAC Equipment		POC
517	Horizontal Cable Chase, Unit 2		POC
518	Horizontal Cable Chase, Unit 1		POC
520	SFP Area Vent Equipment Room		POC
523	Corridor		POC
524	Main Plant Exhaust Equipment Room, Unit 1		POC
525	Containment Access, Unit 1		POC
526	Main Plant Exhaust Equipment Room, Unit 2		POC
527	Containment Access, Unit 2		POC
529	Electrical Room, Unit 1		POC
530/531/533	Spent Fuel Pool Area, Fan Room, New Fuel Area		FL, POC
532	Electrical Room, Unit 2		POC
536/537	Miscellaneous Waste Evaporator & Equipment Rooms		POC
586-597	Radiation Chemistry Area	WP	POC
TURBINE BUILDING			
Access Control	Access Control Area, Offices, Locker Rooms, and Hallways	WP	
559	Computer Room	PA	POC
603	Auxiliary Feedwater Pump Room, Unit 1	WP	POC
605	Auxiliary Feedwater Pump Room, Unit 2	WP	POC
Elevation 12'0"	Steam Generator Feed Pump		
	No. 11	DS	HT
	No. 12	DS	HT
	No. 21	DS	HT
	No. 22	DS	HT
Elevation 12'0"	H_2 Seal Oil Unit, Unit 1	DS	HT
	H_2 Seal Oil Unit, Unit 2	DS	HT
Elevation 45'0"	Unit 1, Turbine Generator Bearing		
	System No. 1	PA	HT
	System No. 2	PA	HT
	System No. 3	PA	HT
	System No. 4	PA	HT
	System No. 5	PA	HT
Elevation 45'0"	Unit 2, Turbine Generator Bearing		
	System No. 1	PA	HT
	System No. 2	PA	HT
	System No. 3	PA	HT
	System No. 4	PA	HT
	System No. 5	PA	HT
Elevation 27'0"	Turbine Building Mezzanine, Unit 1	WP	
	Turbine Building Mezzanine, Unit 2	WP	
Elevation 12'0"	Turbine Building Basement, Unit 1	WP	
	Turbine Building Basement, Unit 2	WP	

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
Elev. 12'0"	Auxiliary Boiler Room	WP	

CONTAINMENT BUILDINGS

UNIT	LOCATION	SUPPRESSION	DETECTION
Unit 1	RCP Bay East		HT
Unit 1	RCP Bay West		HT
Unit 1	East Electrical Penetration Area		PW
Unit 1	West Electrical Penetration Area		PW
Unit 2	RCP Bay East		HT
Unit 2	RCP Bay West		HT
Unit 2	East Electrical Penetration Area		PW
Unit 2	West Electrical Penetration Area		PW

INTAKE STRUCTURE

ROOM	DESCRIPTION	SUPPRESSION	DETECTION
IS Unit 1 and Unit 2	Circulating Water Pump and Saltwater Pump Area		POC

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

NORTH SERVICE BUILDING

ELEVATION	LOCATION	SUPPRESSION	DETECTION
Elevation 45'0"	Tool Room/Weld Shop	WP	POC
Elevation 45'0"	Office Area	WP	POC
Elevation 45'0"	Telephone Equipment Room	H-1	POC
Elevation 12'0"	Storage/General Area	WP	
Elevation 12'0"	Lube Oil Storage Room, Unit 1	WP	
Elevation 12'0"	Lube Oil Storage Room, Unit 2	WP	

1A EMERGENCY DIESEL GENERATOR BUILDING

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
Area 2	Diesel Generator Room and Trench Area	PA	HT
Area 3	EDG 1-E Switchgear Room	PA	POC
Area 4	EDG Control Room	PA	POC
Area 5	Fuel Oil Storage Tank Room	PA	HT
Area 6	General Area 66' and 80' Elevation	PA*	POC*, HT*
Area 7	Battery Room	PA	POC

0C EMERGENCY DIESEL GENERATOR BUILDING

ROOM/AREA	DESCRIPTION	SUPPRESSION	DETECTION
SB003	Battery Room		POC
SB102	Diesel Generator Area, Trench Area, and Cable Tray Area	PA	HT
SB103	Switchgear Room		POC
SB104	Control Room		POC
SB202	Fuel Oil Tank Room	PA	HT

OTHER BUILDINGS

BUILDING	DESCRIPTION	SUPPRESSION	DETECTION
Fire Pump House	Protected Area Fire Pump House	WP	
South Service Building	Office Area, Shop Area, Cafeteria	WP	
Unit 1 Equipment Hatch Building	Unit 1 Containment Equipment Hatch Access Building	DP	
Unit 2 Equipment Hatch Building	Unit 2 Containment Equipment Hatch Access Building	DP	
Material Processing Facility	Materials Processing Facility for Low Level Radioactive Materials	WP, DP	DD
Warehouse 1	Office Area and Storage Area	WP	
Warehouse 2	General Storage Warehouse	WP	
Warehouse 3	General Storage Warehouse	WP	

TABLE 9-20

SUMMARY OF FIRE PROTECTION SYSTEMS

TRANSFORMERS

TRANSFORMER	SUPPRESSION	DETECTION
U-25000-11	DS	PW
U-25000-12	DS	PW
U-22000-21	DS	PW
U-22000-22	DS	PW
P-13000-1	DS	HT
P-13000-2	DS	HT
U-4000-11	DS	HT
U-4000-12	DS	HT
U-4000-21	DS	HT
U-4000-22	DS	HT
U-4000-13	DS	HT
U-4000-23	DS	HT

13.8 kV VOLTAGE REGULATORS

VOLTAGE REGULATOR	SUPPRESSION	DETECTION
1H1101REG, 1H11O2REG, 1H1103REG	DS	HT
2H2101REG, 2H2102REG, 2H2103REG	DS	HT

NO. 2 FUEL OIL STORAGE TANKS

STORAGE TANK	SUPPRESSION	DETECTION
No. 11	MF	
No. 21	MF	

LEGEND

* - Partial area sprinkler coverage
** - Partial detection coverage
DS = Deluge Sprinkler
WP = Wet Pipe Sprinkler
MW = Manual Wet Pipe Sprinkler
DP = Dry Pipe Sprinkler
PA = Automatic Pre-action Sprinkler
MP = Manual Pre-action Sprinkler
MF = Manual Foam System
H-1 = Halon Room Flooding
H-2 = Halon Underfloor Flooding
POC = Product of Combustion Detector
FL = Flame Detector
HT = Heat Detector
DD = Duct Mounted Detector
PW = Protecto-Wire

[a] For 3-1/2" and smaller size fittings, and for 2" and smaller size valves, alternate materials can be used.

[b] 2" and smaller valves are screw type.

[c] For 2-1/2" and greater size gate valves, material may be Ductile Iron and have a pressure class of up to 250 psi.

9.10 COMPRESSED AIR SYSTEM

9.10.1 DESIGN BASIS

The Compressed Air System consists of the instrument air and plant air subsystems. The instrument air subsystem is designed to provide a reliable supply of dry and oil-free air for the pneumatic instruments and controls and pneumatically operated containment isolation valves. The plant air subsystem is designed to meet necessary service air requirements for plant maintenance and operation. The designs of each subsystem are based on an estimated instrument air requirement of 260 scfm and an estimated plant air requirement of 600 scfm. The instrument air subsystem compressor is sized for 450 scfm.

9.10.2 SYSTEM DESCRIPTION

The Compressed Air System is shown schematically on Figures 9-23 (Unit 1) and 9-28 (Unit 2). The Plant Water and Air Service System is shown in Figure 9-29.

The system incorporates two full-capacity, non-lubricated compressors for instrument air, each having a separate inlet filter aftercooler and moisture separator. The instrument air compressors then discharge to a single header which is connected to two air receivers. Both air receivers discharge to a compressed air outlet header which supplies instrument air to the air dryers and filter assembly. The compressed air header then divides into branch lines supplying the pretreatment and tank storage area, the Intake Structure, the service building, the water treatment area, the Turbine Building, the containment structure, and the Auxiliary Building.

An emergency back-up tie from the plant air header has been provided to automatically supply air to the instrument air system if the pressure to the instrument filter and dryer assembly falls below a preset value. Local controls are provided to prevent plant air use when this occurs. For the transition from normal to emergency service, air storage tanks provide an approximate 20-minute supply (Table 9-21).

Particle size, dew point, and oil hydrocarbons are controlled for instrument air supply in accordance with Instrument Society of America standards. Additionally, the Calvert Cliffs approach to controlling air quality was submitted to the NRC in response to Generic Letter 88-14.

One full-capacity plant air compressor with an inlet filter, and integral air coolers and moisture separators, discharges to the plant air receiver. The receiver outlet header is connected to the prefilter assembly, which is followed by an outlet header branching into two separate air headers, one to the instrument air dryers and filter assembly, and the other to the plant air pretreatment and storage tank area, the Intake Structure, the service building, the water treatment area, the Turbine Building, the Containment Structure, and the Auxiliary Building. A system cross-tie between Unit 1 and Unit 2 has been provided for the plant air headers. Additionally, each plant air system has a permanent connection for the installation of a portable air compressor to allow for maintenance of the compressors or SRW system during Modes 3, 4, 5, 6 and defueled. This connection may also be used in Modes 1 and 2 to provide a contingency backup to an operating plant air compressor should the other installed plant air compressor be unavailable.

9.10.3 SYSTEM COMPONENTS

Ratings and construction of system components are listed in Table 9-21.

9.10.4 SYSTEM OPERATION

A continuous supply of instrument air is provided to hold various pneumatically-operated valve actuators in the positions necessary for operating conditions. Normally, the plant air

compressor and one instrument air compressor will operate and the second instrument air compressor will be on automatic standby.

9.10.5 SYSTEM RELIABILITY

The power supply for the normal compressors is the normal distribution system and can be backed up by the EDG. Additional emergency air compressors, known as the saltwater air compressors (SWACs), provide redundant air supply to most safety-related components when the normal air compressors are lost. The SWACs (Table 9-16B) are seismically qualified, air-cooled, and oil-free. The instrument air portion of the compressed air system is primarily used for valve actuation and is not used in any reactor indication, control, or protective circuitry. These valve actuators are designed to fail in the safe position after loss of the instrument air supply. The design of the system and installed equipment redundancy ensure that total loss of instrument air supply is highly improbable. Concurrently, attention has been given to ensure that valve failures from loss of instrument air supply are consistent with the capability to maintain the plant in a safe condition and mitigate the consequences of any simultaneous incident or accident.

9.10.6 TESTS AND INSPECTIONS

Each component is inspected and cleaned prior to installation into the system. Instruments were calibrated during testing and automatic controls were tested for actuation at the proper setpoints. Alarm functions were checked for operability and limits during plant operational testing. The systems were operated and tested initially with regard to flow paths, flow capacity, and mechanical operability.

TABLE 9-21

COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION

A. INSTRUMENT AIR SYSTEM

<u>Air Compressor</u>	
Type	Vertical, non-lubricated reciprocating, two state Y-angle type
Quantity	2 (per unit)
Design capacity (scfm)	470 (each)
Discharge pressure (psig)	100
Motor	100 hp, 3 phase, 60 Hz, 460 Volt
Code	ASME Section VIII, NEMA
<u>Intake Filter – Silencer</u>	
Type	dry
Quantity	2 per Unit
Base size	8"
<u>Aftercooler and Moisture Separator</u>	
Type	Shell and tube
Quantity	2 (1 per compressor)
Code	TEMA Class C, ASME Section VIII
<u>Air Receiver</u>	
Type	Vertical
Quantity	2 (1 per compressor)
Design pressure (psig)	115
Actual volume (ft^3)	96
Code	ASME Section VIII
<u>Prefilters</u>	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	720
Filtration	99% removal of all liquids, oil, and water droplets
<u>Air Dryer</u>	
Type	Heatless
Desiccant	Activated alumina absorbent
Quantity	2 per unit
Capacity (scfm)	475 (Nos. 12 and 22), 700 (Nos. 11 and 21)
Outlet moisture content with saturated air inlet	-40°F dew point at 100 psig
<u>Afterfilters</u>	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	600
Filtration	100% removal of all particulates over 0.9 microns

TABLE 9-21

COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION

<u>Piping and Valves</u>	
Valves	150 psi ANSI for 2-1/2" and larger, 600 psi ANSI for 2" and smaller
Piping	Seamless ASTM A106, Grade B (2-1/2" through 24")
Code	ANSI B31.1 (ANSI B31.7 - penetration piping)
B. PLANT AIR SYSTEM	
<u>Air Compressor</u>	
Type	Centrifugal, two stage, with integral air coolers and moisture separators
Quantity	One per Unit
Design capacity (scfm)	600
Discharge pressure (psig)	100
Motor	200 hp, 3 phase, 60 Hz, 460 Volt
Code	NEMA
<u>Intake Filter Silencer</u>	
Type	Dry
Quantity	One per Unit
<u>Air Receiver</u>	
Type	Vertical
Quantity	1
Design pressure (psig)	115
Actual volume (ft^3)	96
Code	ASME Section VIII
<u>Prefilter</u>	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	720
Filtration	99% removal of all liquids, oil, and water droplets
<u>Piping and Valving</u>	
Valves	150 psi ANSI for 2-1/2" and larger, 600 psi ANSI for 2" and smaller
Piping	Seamless ASTM A106, Grade B (2-1/2" through 24")
Code	ANSI B31.1 (ANSI B31.7 - penetration piping)
C. INSTRUMENT BACKUP AIR SYSTEM	
<u>Storage Tank</u>	
Type	Vertical
Quantity	4
Capacity	300 ft^3
Design pressure (psig)	225
Code	ASME Section VIII
<u>Air Amplifier</u>	
Ratio	2:1

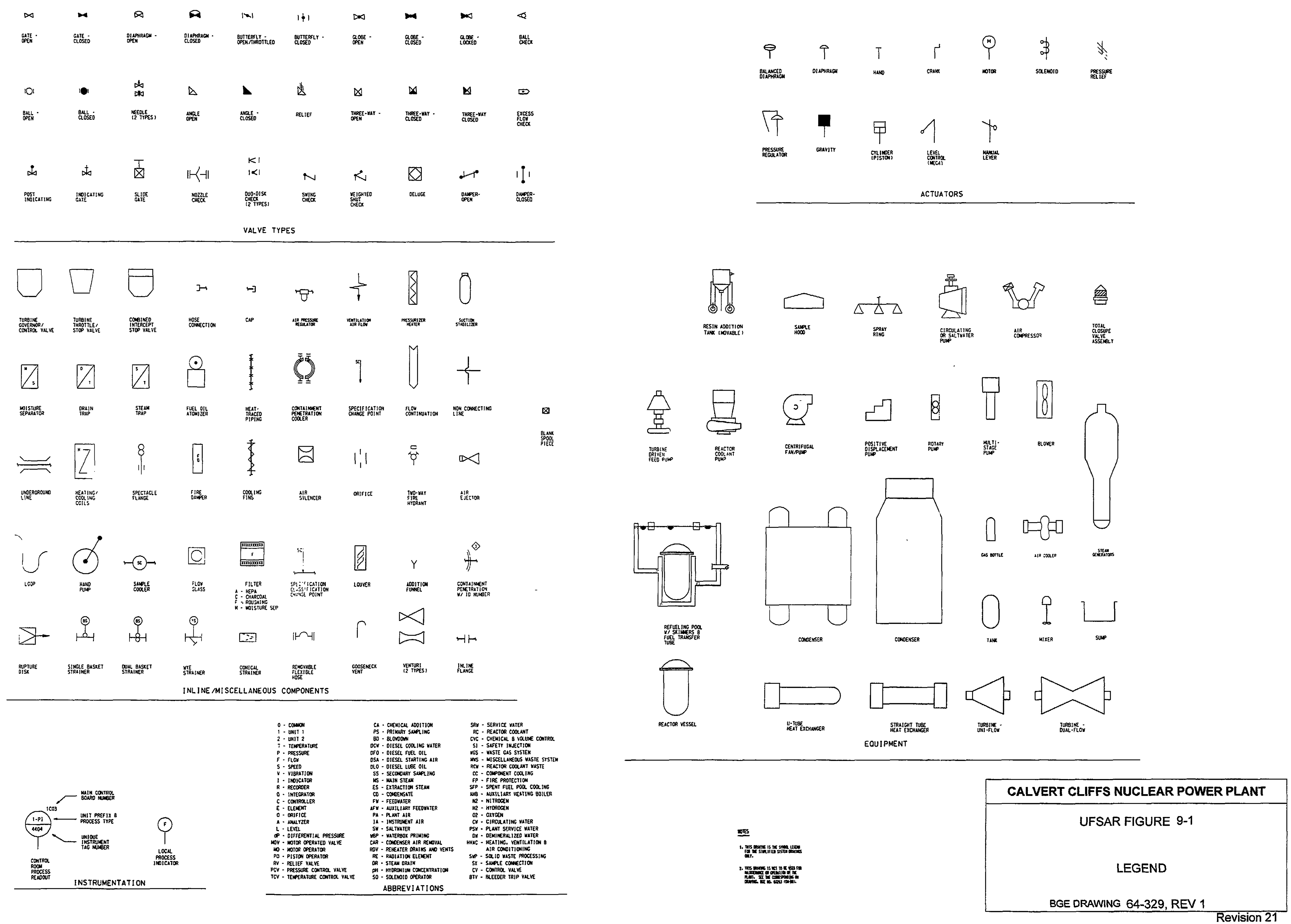
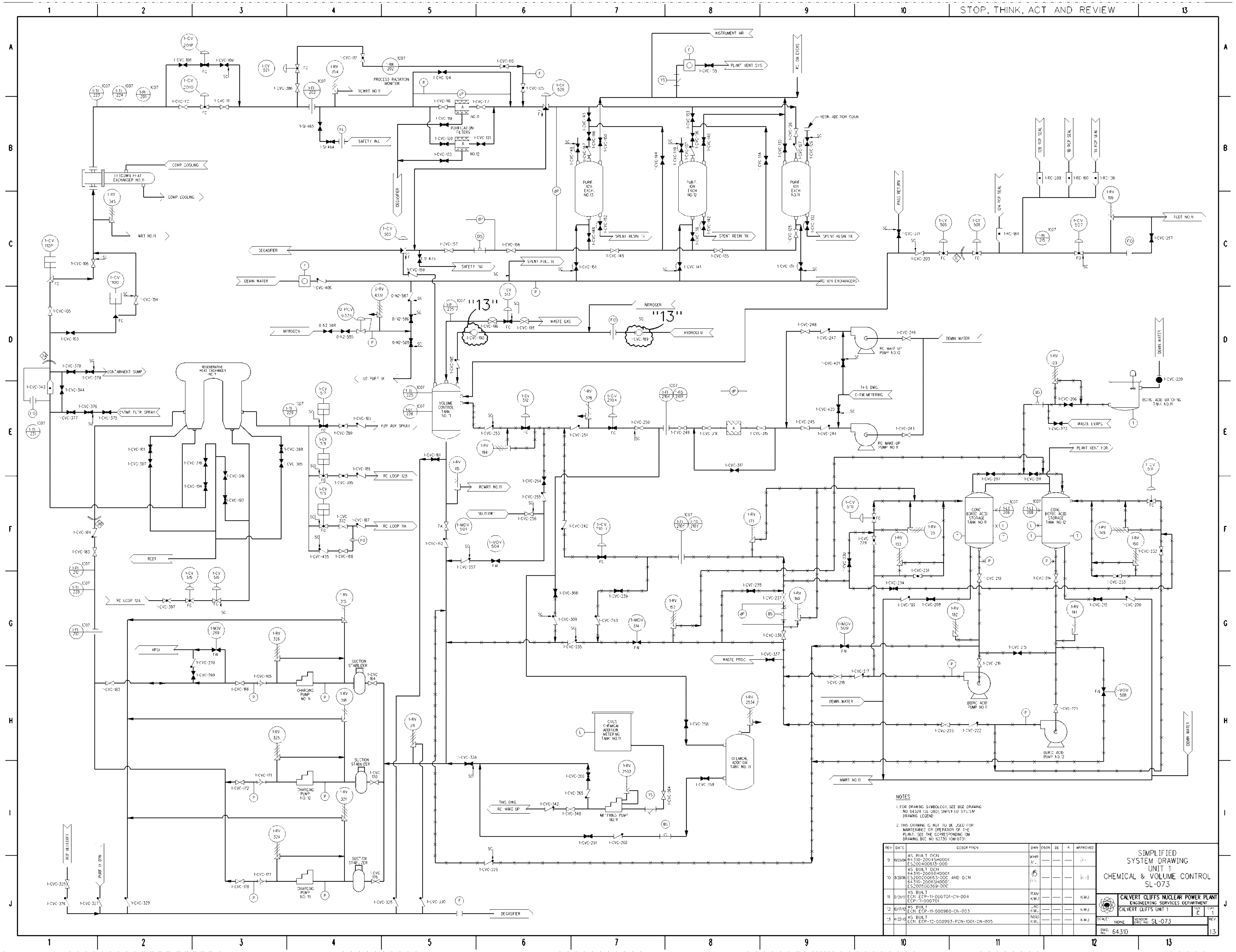
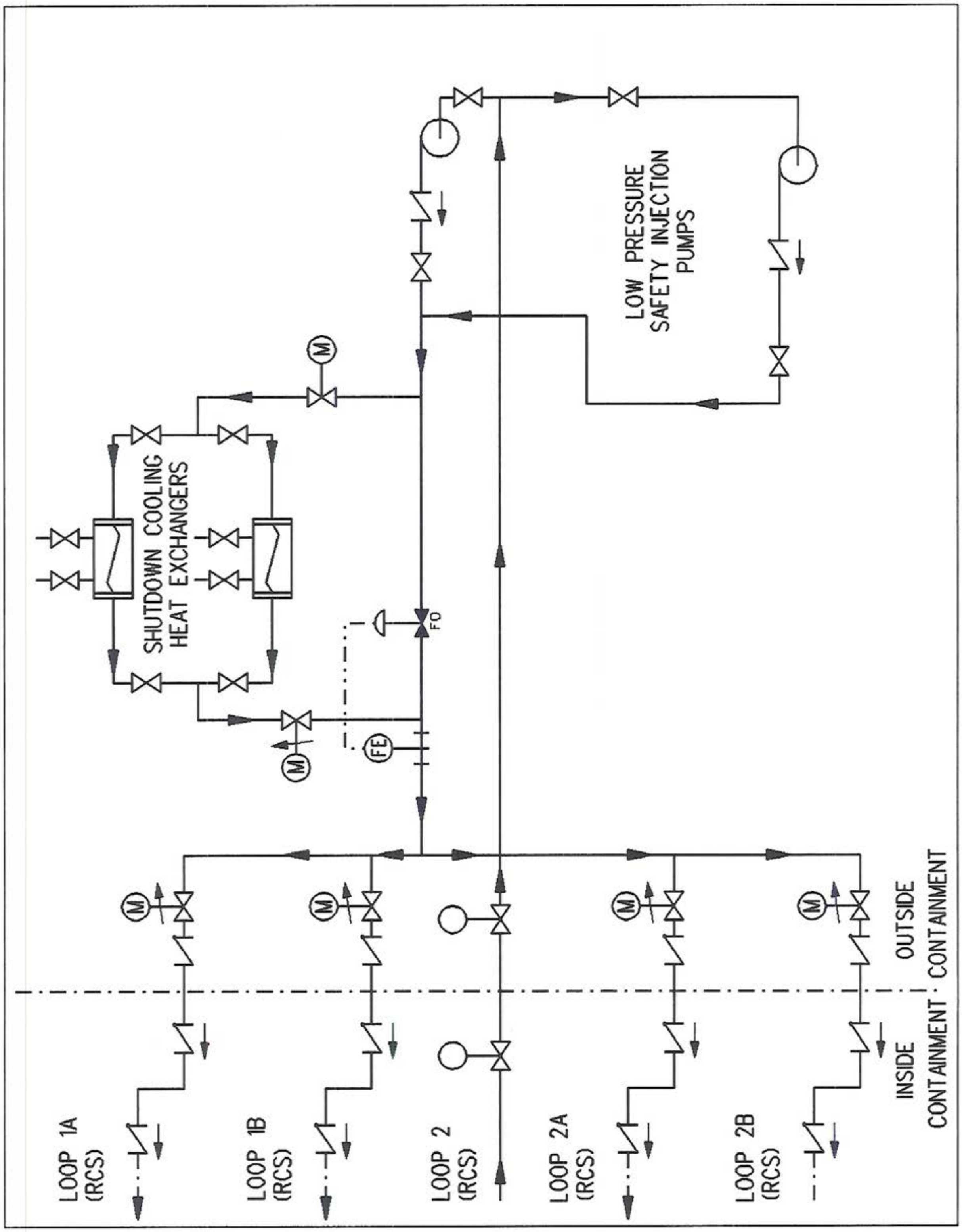

GATE - OPEN
GATE - CLOSED
DIAPHRAGM - OPEN
DIAPHRAGM - CLOSED
BUTTERFLY - OPEN/THROTTLED
BUTTERFLY - CLOSED
GLOBE - OPEN
GLOBE - CLOSED
GLOBE - LOCKED
BALL CHECK
BALL - OPEN
BALL - CLOSED
NEEDLE (2 TYPES)
ANGLE OPEN
ANGLE - CLOSED
RELIEF
THREE-WAY - OPEN
THREE-WAY - CLOSED
THREE-WAY CLOSED
EXCESS FLOW CHECK
POST INDICATING
INDICATING GATE
SLIDE GATE
NOZZLE CHECK
DUO-DISK CHECK (2 TYPES)
SWING CHECK
WEIGHTED SHUT CHECK
DELUGE
DAMPER-OPEN
DAMPER-CLOSED
VALVE TYPES
BALANCED DIAPHRAGM
DIAPHRAGM
HAND
CRANK
MOTOR
SOLENOID
PRESSURE RELIEF
PRESSURE REGULATOR
GRAVITY
CYLINDER (PISTON)
LEVEL CONTROL (MECH)
MANUAL LEVER
ACTUATORS
TURBINE GOVERNOR/ CONTROL VALVE
TURBINE THROTTLE/ STOP VALVE
COMBINED INTERCEPT STOP VALVE
HOSE CONNECTION
CAP
AIR PRESSURE REGULATOR
VENTILATION AIR FLOW
PRESSURIZER HEATER
SUCTION STABILIZER
MOISTURE SEPARATOR
DRAIN TRAP
STEAM TRAP
FUEL OIL ATOMIZER
HEAT-TRACED PIPING
CONTAINMENT PENETRATION COOLER
SPECIFICATION CHANGE POINT
FLOW CONTINUATION
NON CONNECTING LINE
BLANK SPOOL PIECE
UNDERGROUND LINE
HEATING/ COOLING COILS
SPECTACLE FLANGE
FIRE DAMPER
COOLING FINS
AIR SILENCER
ORIFICE
TWO-WAY FIRE HYDRANT
AIR EJECTOR
LOOP
HAND PUMP
SAMPLE COOLER
FLOW GLASS
FILTER
A - HEPA
C - CHARCOAL
F - ROUGHING
M - MOISTURE SEP
SPECIFICATION CLASSIFICATION CHANGE POINT
LOUVER
ADDITION FUNNEL
CONTAINMENT PENETRATION W/ ID NUMBER
RUPTURE DISK
SINGLE BASKET STRAINER
DUAL BASKET STRAINER
WYE STRAINER
CONICAL STRAINER
REMOVABLE FLEXIBLE HOSE
GOOSENECK VENT
VENTURI (2 TYPES)
INLINE FLANGE
INLINE/MISCELLANEOUS COMPONENTS
RESIN ADDITION TANK (MOVABLE)
SAMPLE HOOD
SPRAY RING
CIRCULATING OR SALTWATER PUMP
AIR COMPRESSOR
TOTAL CLOSURE VALVE ASSEMBLY
TURBINE DRIVEN FEED PUMP
REACTOR COOLANT PUMP
CENTRIFUGAL FAN/PUMP
POSITIVE DISPLACEMENT PUMP
ROTARY PUMP
MULTI-STAGE PUMP
BLOWER
REFUELING POOL W/ SKIMMERS & FUEL TRANSFER TUBE
CONDENSER
CONDENSER
GAS BOTTLE
AIR COOLER
STEAM GENERATORS
TANK
MIXER
SUMP
REACTOR VESSEL
U-TUBE HEAT EXCHANGER
STRAIGHT TUBE HEAT EXCHANGER
TURBINE - UNI-FLOW
TURBINE - DUAL-FLOW
EQUIPMENT
MAIN CONTROL BOARD NUMBER
1C03
1-PI
4404
UNIT PREFIX & PROCESS TYPE
UNIQUE INSTRUMENT TAG NUMBER
CONTROL ROOM PROCESS READOUT
F
LOCAL PROCESS INDICATOR
INSTRUMENTATION
0 - COMMON
1 - UNIT 1
2 - UNIT 2
T - TEMPERATURE
P - PRESSURE
F - FLOW
S - SPEED
V - VIBRATION
I - INDICATOR
R - RECORDER
Q - INTEGRATOR
C - CONTROLLER
E - ELEMENT
O - ORIFICE
A - ANALYZER
L - LEVEL
dP - DIFFERENTIAL PRESSURE
MOV - MOTOR OPERATED VALVE
MO - MOTOR OPERATOR
PO - PISTON OPERATOR
RV - RELIEF VALVE
PCV - PRESSURE CONTROL VALVE
TCV - TEMPERATURE CONTROL VALVE
CA - CHEMICAL ADDITION
PS - PRIMARY SAMPLING
BD - BLOWDOWN
DCW - DIESEL COOLING WATER
DFO - DIESEL FUEL OIL
DSA - DIESEL STARTING AIR
DLO - DIESEL LUBE OIL
SS - SECONDARY SAMPLING
MS - MAIN STEAM
ES - EXTRACTION STEAM
CD - CONDENSATE
FW - FEEDWATER
AFW - AUXILIARY FEEDWATER
PA - PLANT AIR
IA - INSTRUMENT AIR
SW - SALTWATER
WBP - WATERBOX PRIMING
CAR - CONDENSER AIR REMOVAL
RDV - REHEATER DRAINS AND VENTS
RE - RADIATION ELEMENT
DR - STEAM DRAIN
pH - HYDRONIUM CONCENTRATION
SO - SOLENOID OPERATOR
SRW - SERVICE WATER
RC - REACTOR COOLANT
CVC - CHEMICAL & VOLUME CONTROL
SI - SAFETY INJECTION
WGS - WASTE GAS SYSTEM
MWS - MISCELLANEOUS WASTE SYSTEM
RCW - REACTOR COOLANT WASTE
CC - COMPONENT COOLING
FP - FIRE PROTECTION
SFP - SPENT FUEL POOL COOLING
AHB - AUXILIARY HEATING BOILER
N2 - NITROGEN
H2 - HYDROGEN
O2 - OXYGEN
CW - CIRCULATING WATER
PSW - PLANT SERVICE WATER
DW - DEMINERALIZED WATER
HVAC - HEATING, VENTILATION & AIR CONDITIONING
SWP - SOLID WASTE PROCESSING
SX - SAMPLE CONNECTION
CV - CONTROL VALVE
BTV - BLEEDER TRIP VALVE
ABBREVIATIONS
NOTES
1. THIS DRAWING IS THE SYMBOL LEGEND FOR THE SIMPLIFIED SYSTEM DRAWINGS ONLY.
CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 9-1
LEGEND
BGE DRAWING 64-329, REV 1
Revision 21

FIGURE 9-3 CHEMICAL AND VOLUME CONTROL SYSTEM – UNIT 1





Calvert Cliffs Nuclear Power Plant	SHUTDOWN COOLING FLOW DIAGRAM	Figure 9-5 Revision 45

FIGURE 9-6 COMPONENT COOLING SYSTEM – UNIT 1

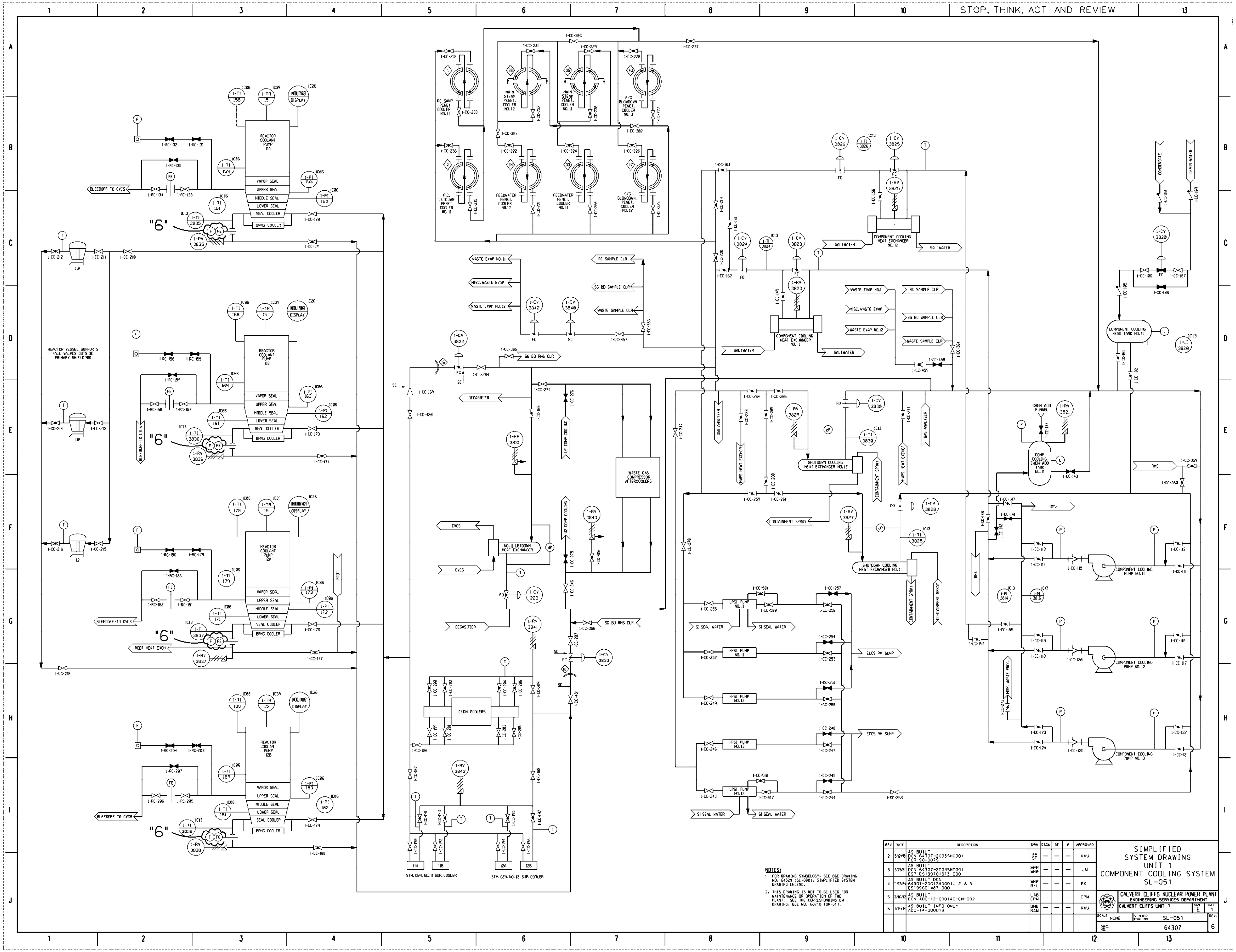


FIGURE 9-7 SPENT FUEL POOL COOLING

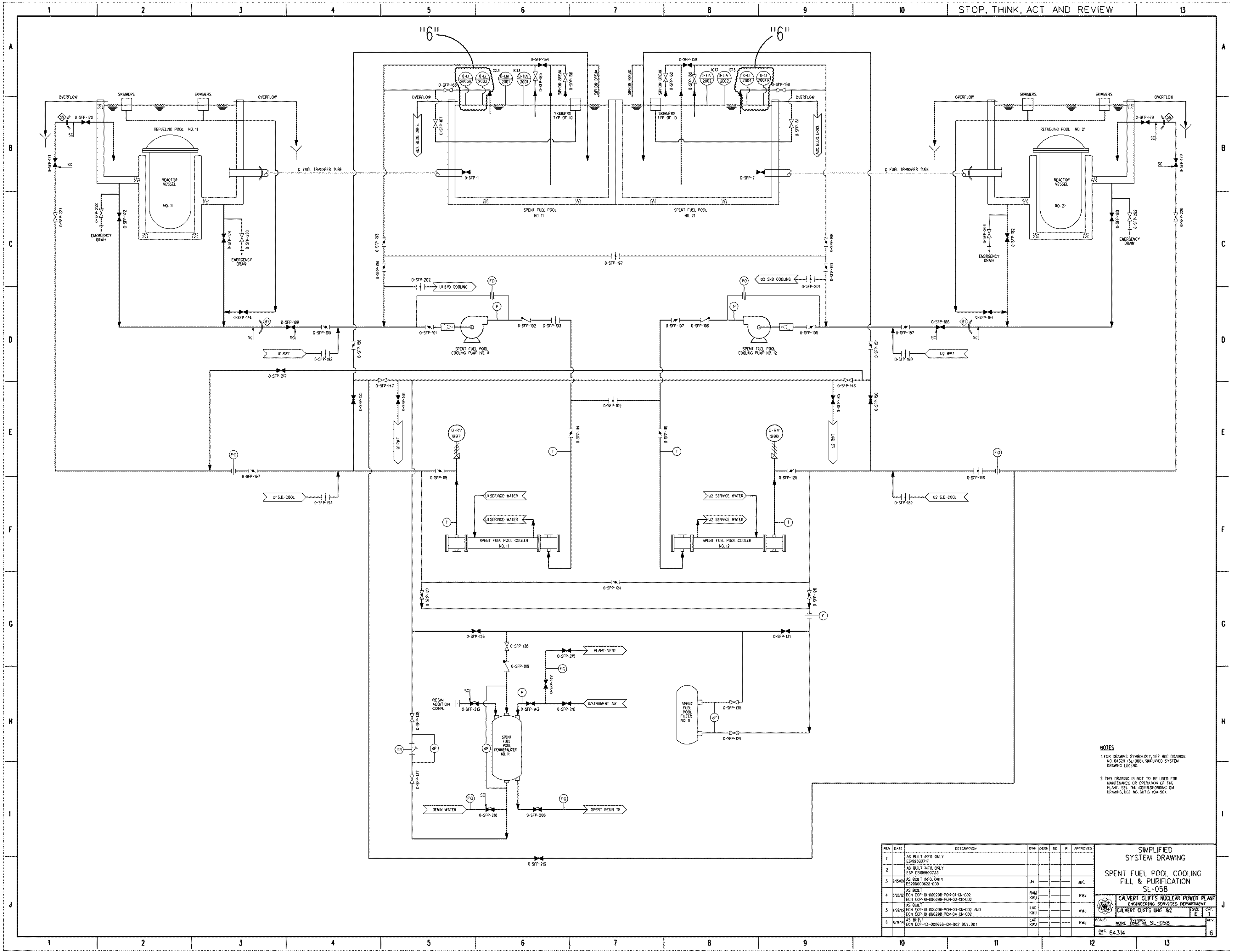


FIGURE 9-8 CIRCULATING AND SALTWATER COOLING SYSTEM – UNIT 1

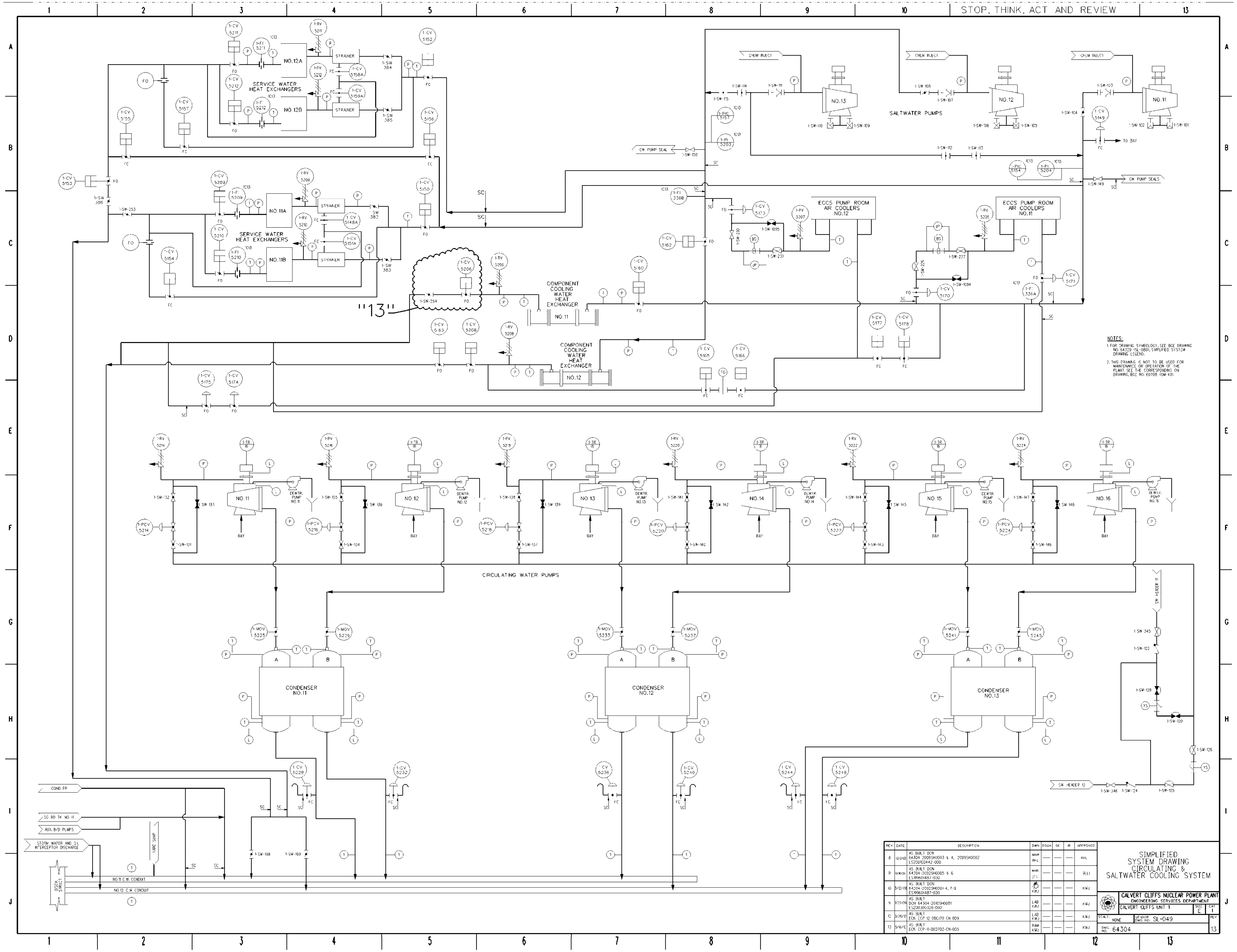


FIGURE 9-9 SERVICE WATER - UNIT 1

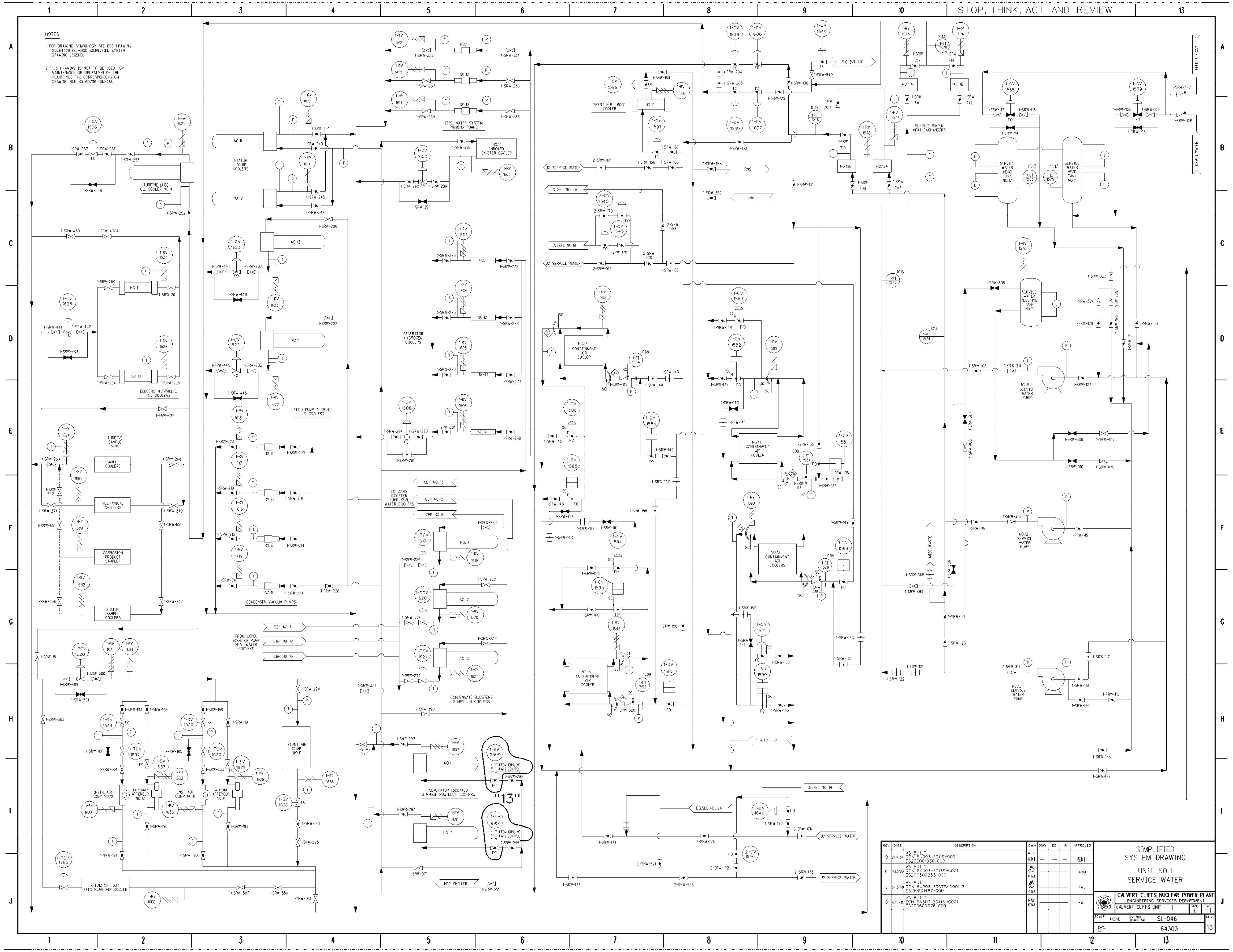


FIGURE 9-9B SERVICE WATER SYSTEM – CONTAINMENT AND AUXILIARY BUILDING UNITS 1 AND 2

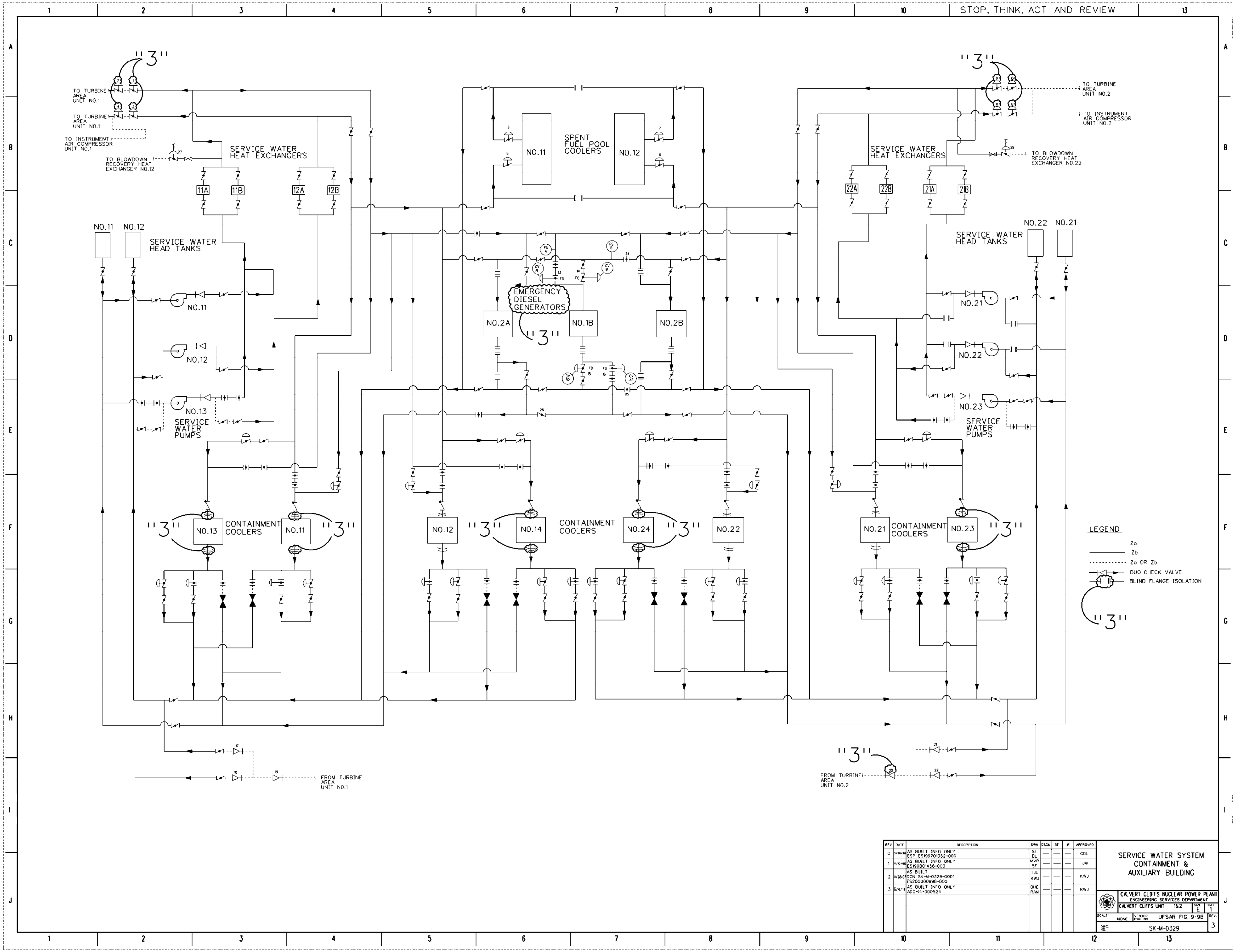
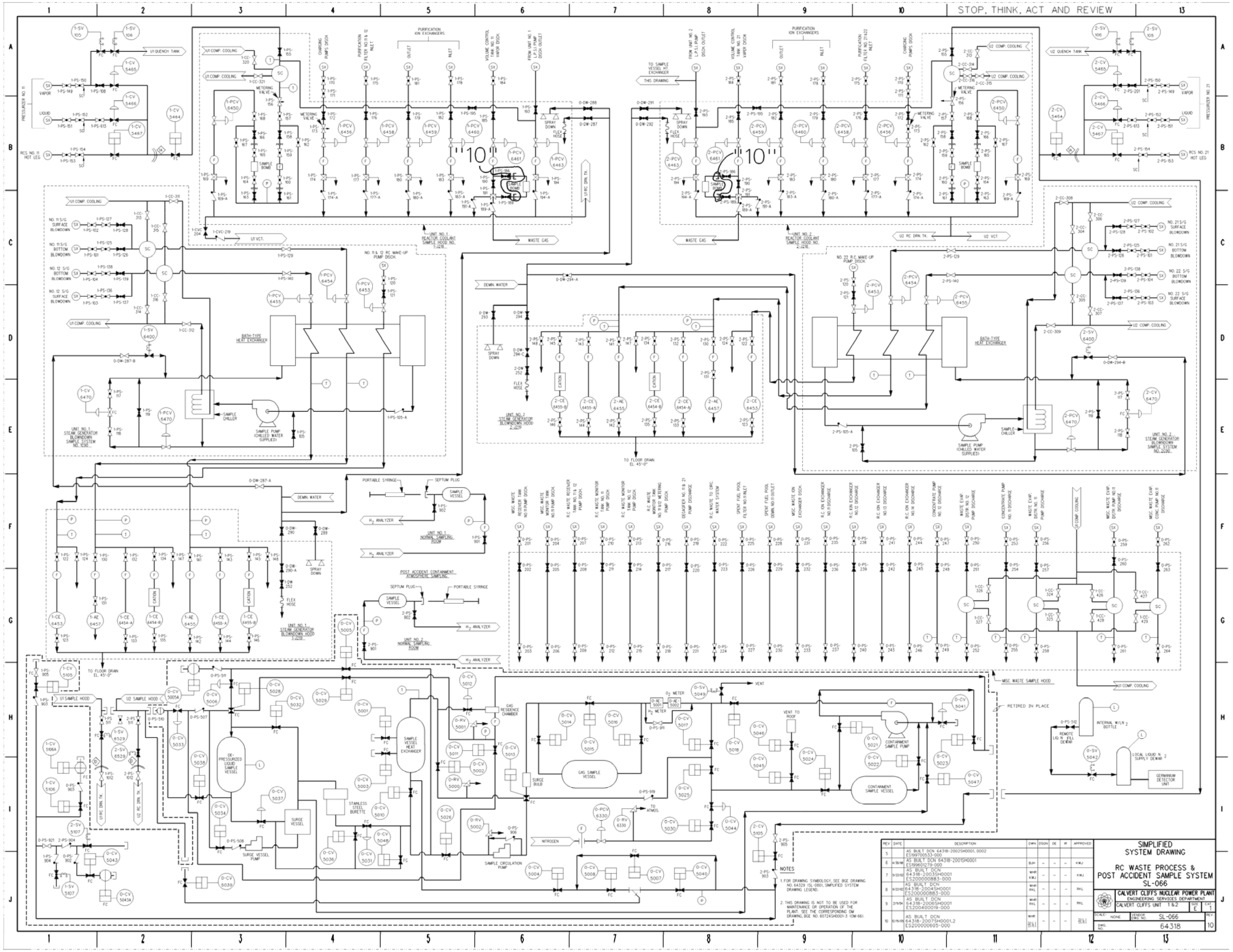
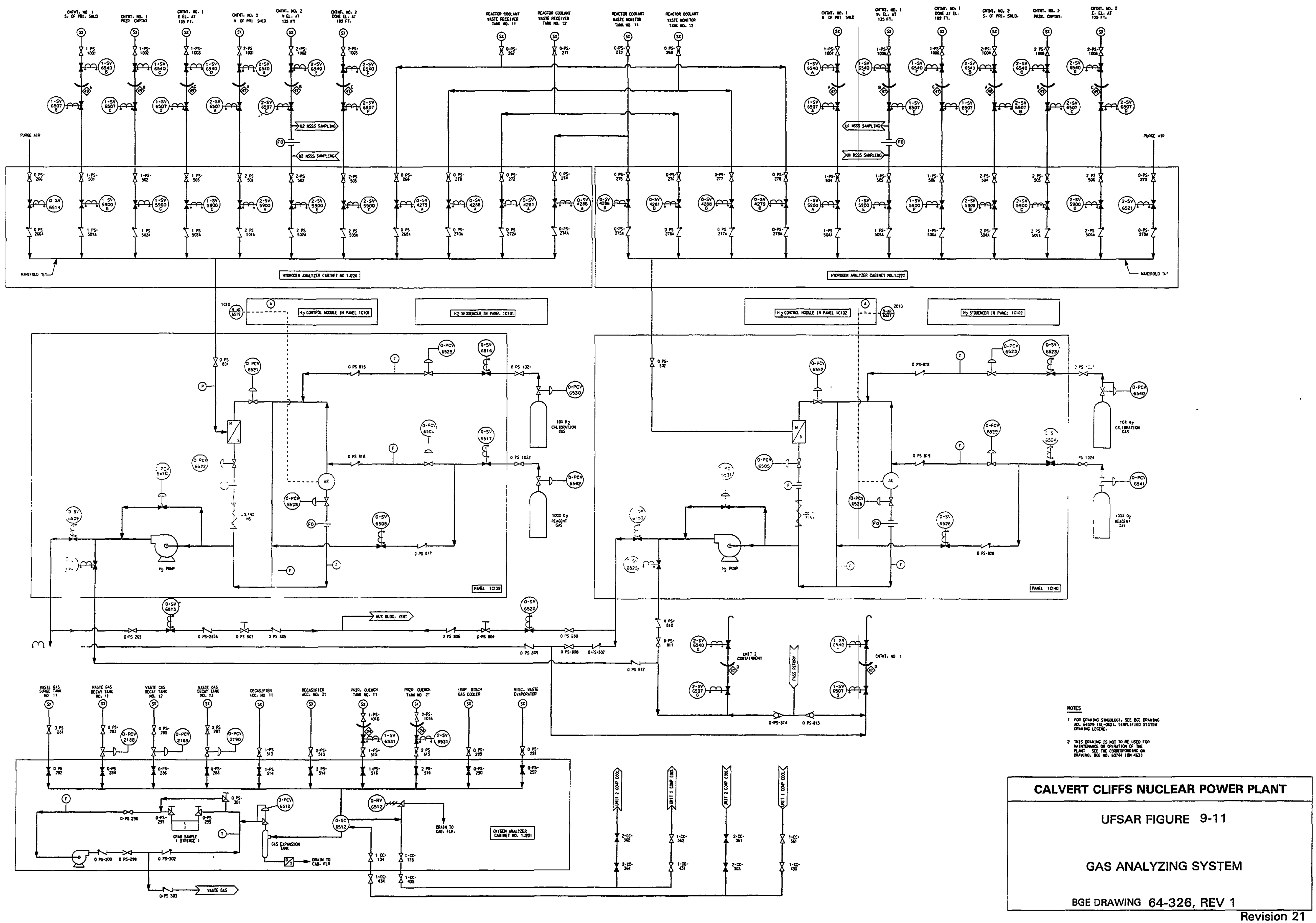
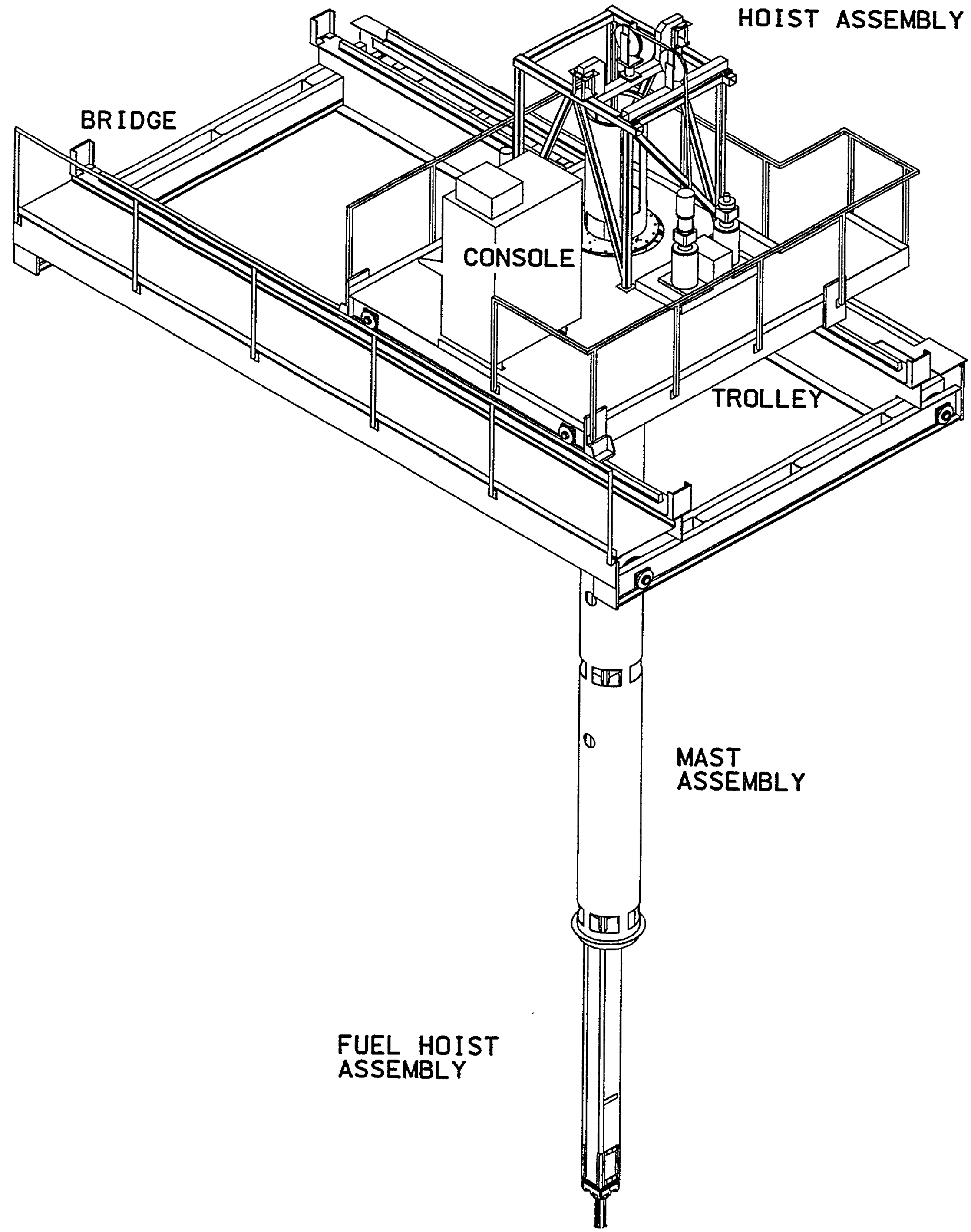


FIGURE 9-10 REACTOR COOLANT/WASTE PROCESSING AND POST ACCIDENT SAMPLING SYSTEMS




CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 9-11
GAS ANALYZING SYSTEM
BGE DRAWING 64-326, REV 1
Revision 21



Calvert Cliffs Nuclear Power Plant	REFUELING MACHINE	Figure 9-13 Rev. 27

FIGURE 9-14 FUEL TRANSFER TUBE – UNITS 1 OR 2

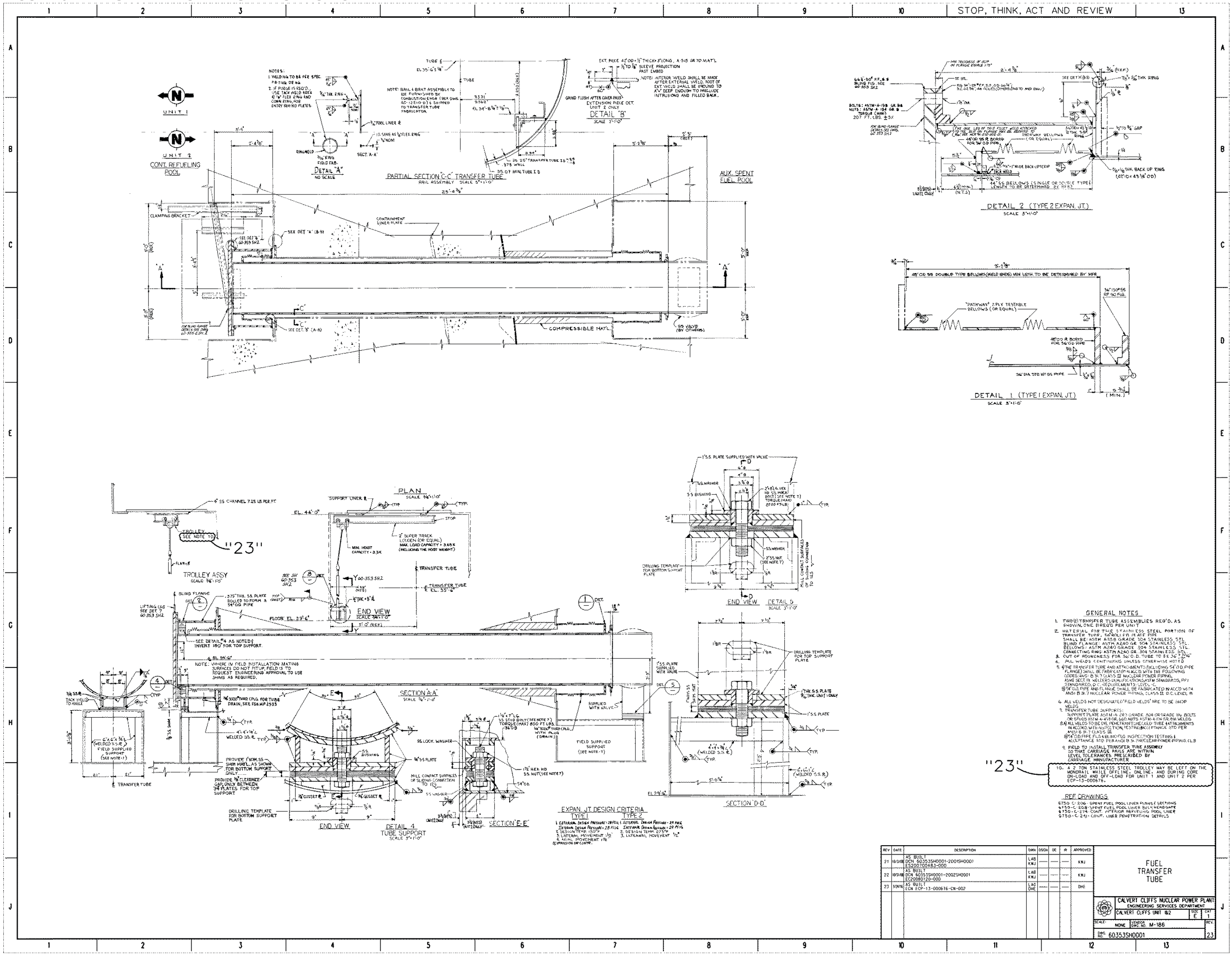
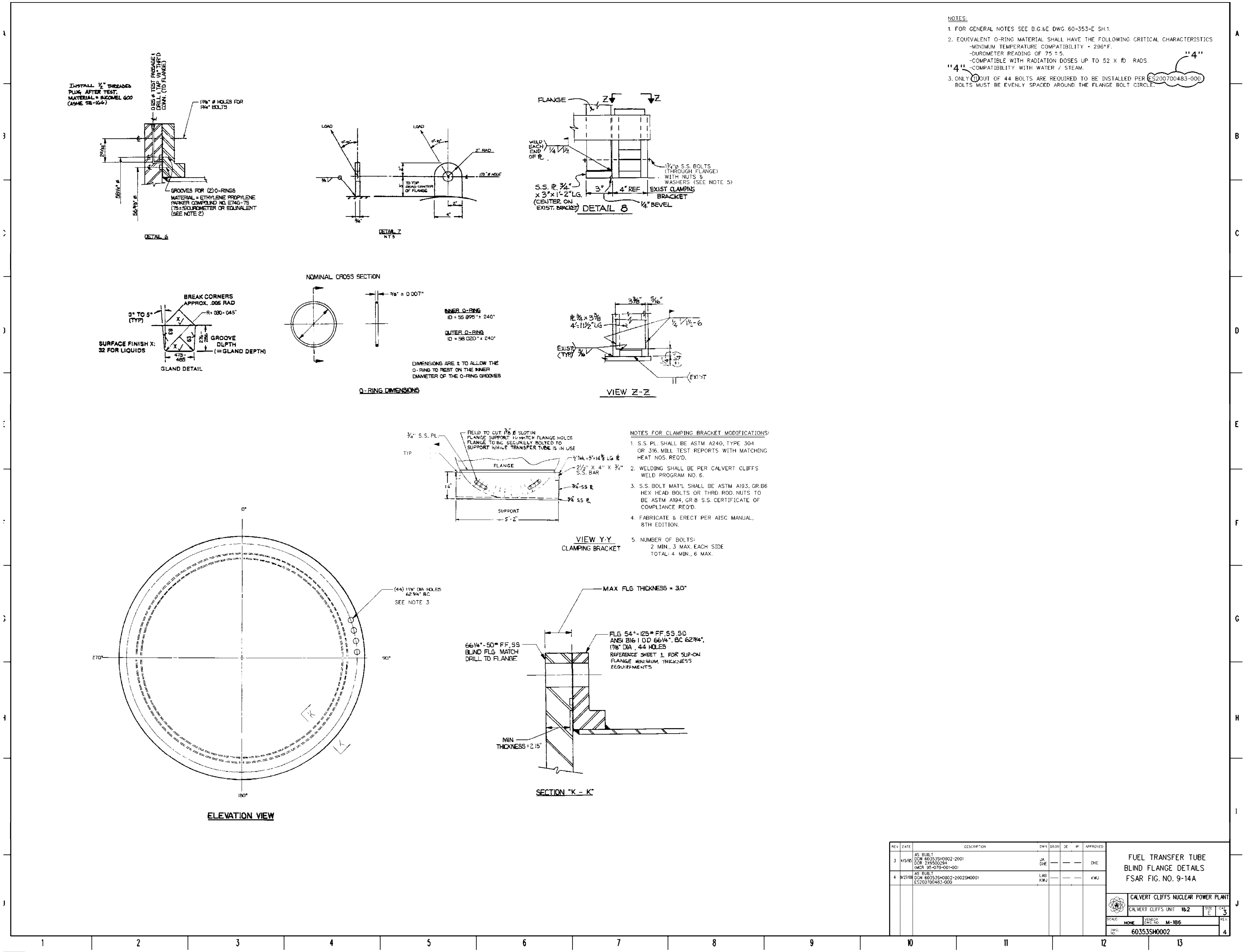
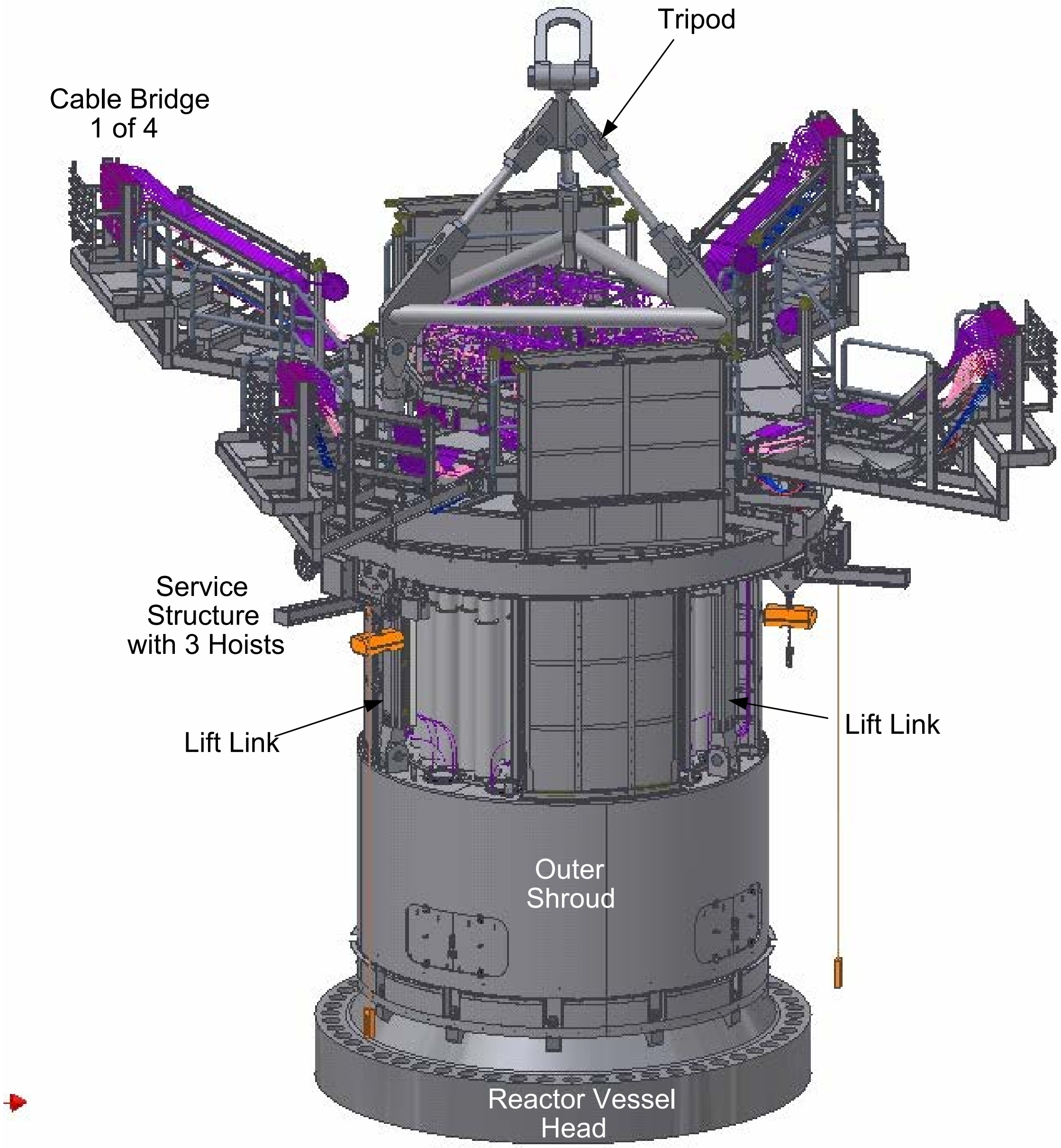
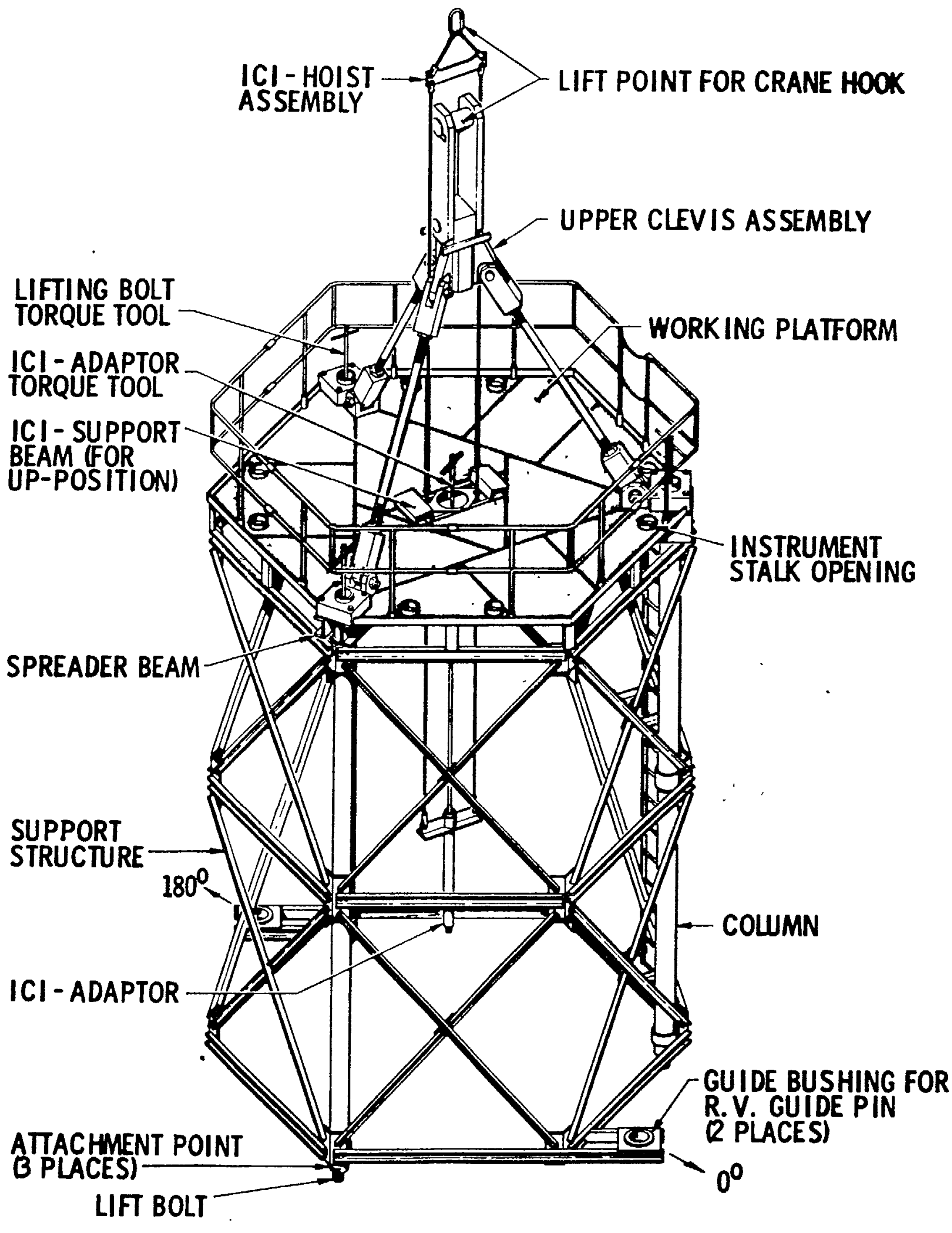


FIGURE 9-14A FUEL TRANSFER TUBE BLIND FLANGE DETAILS

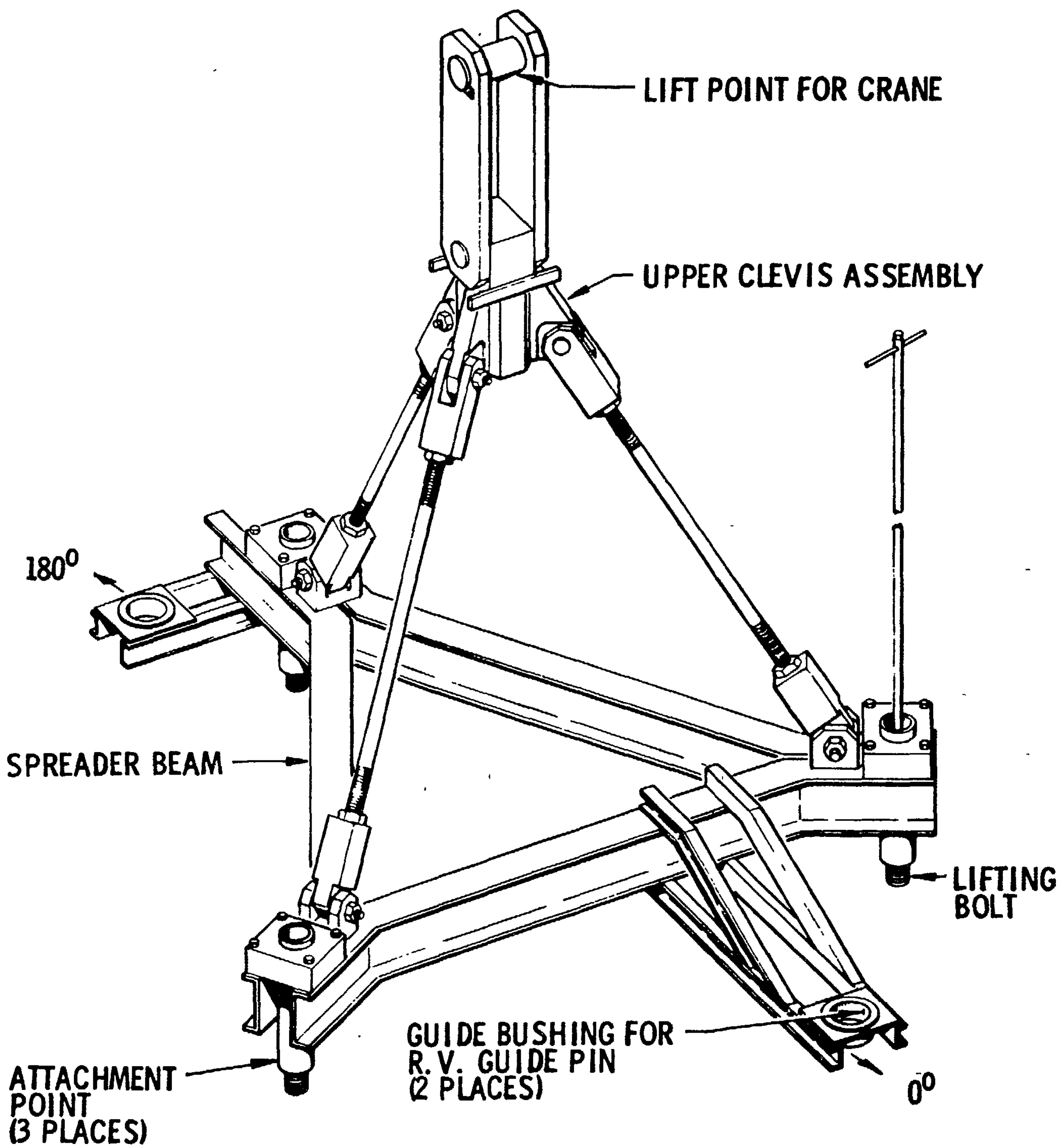




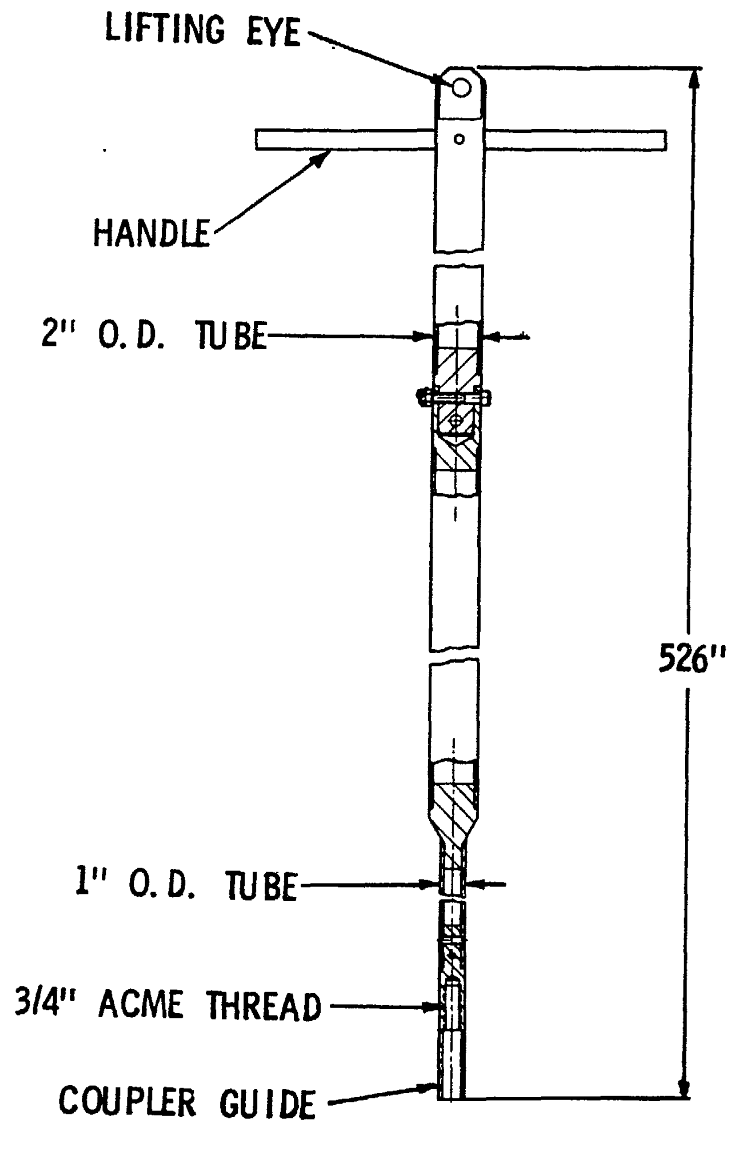
Calvert Cliffs Nuclear Power Plant	REPLACEMENT REACTOR VESSEL CLOSURE HEAD LIFTING RIG	Figure 9-16 Revision 39



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	UPPER GUIDE STRUCTURE LIFTING RIG	Figure 9-17



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CORE SUPPORT BARREL LIFTING RIG	Figure 9-18



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Surveillance Capsule Retrieval Tool	Figure 9-19

FIGURE 9-20A CONTAINMENT AND PENETRATION ROOM VENTILATION

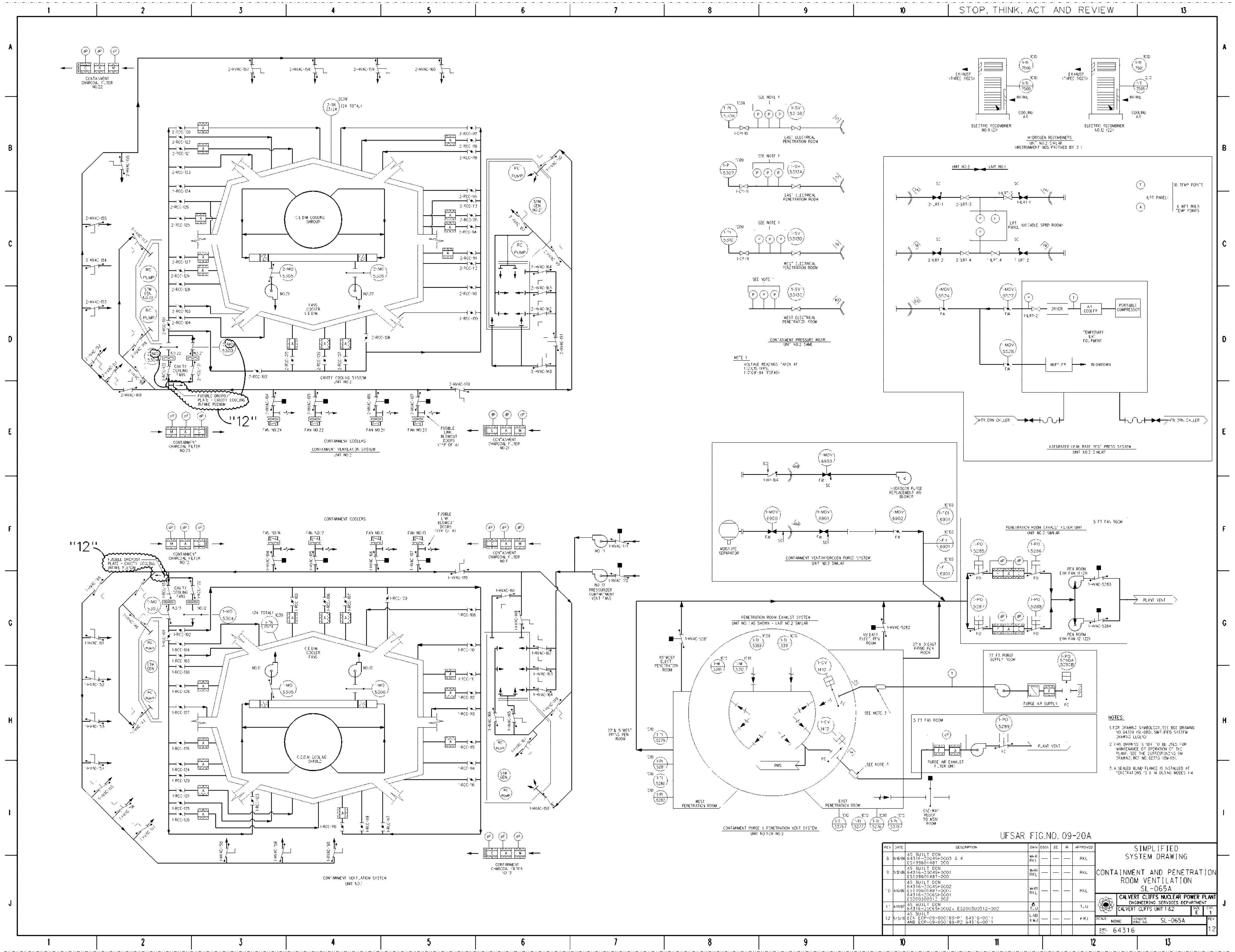
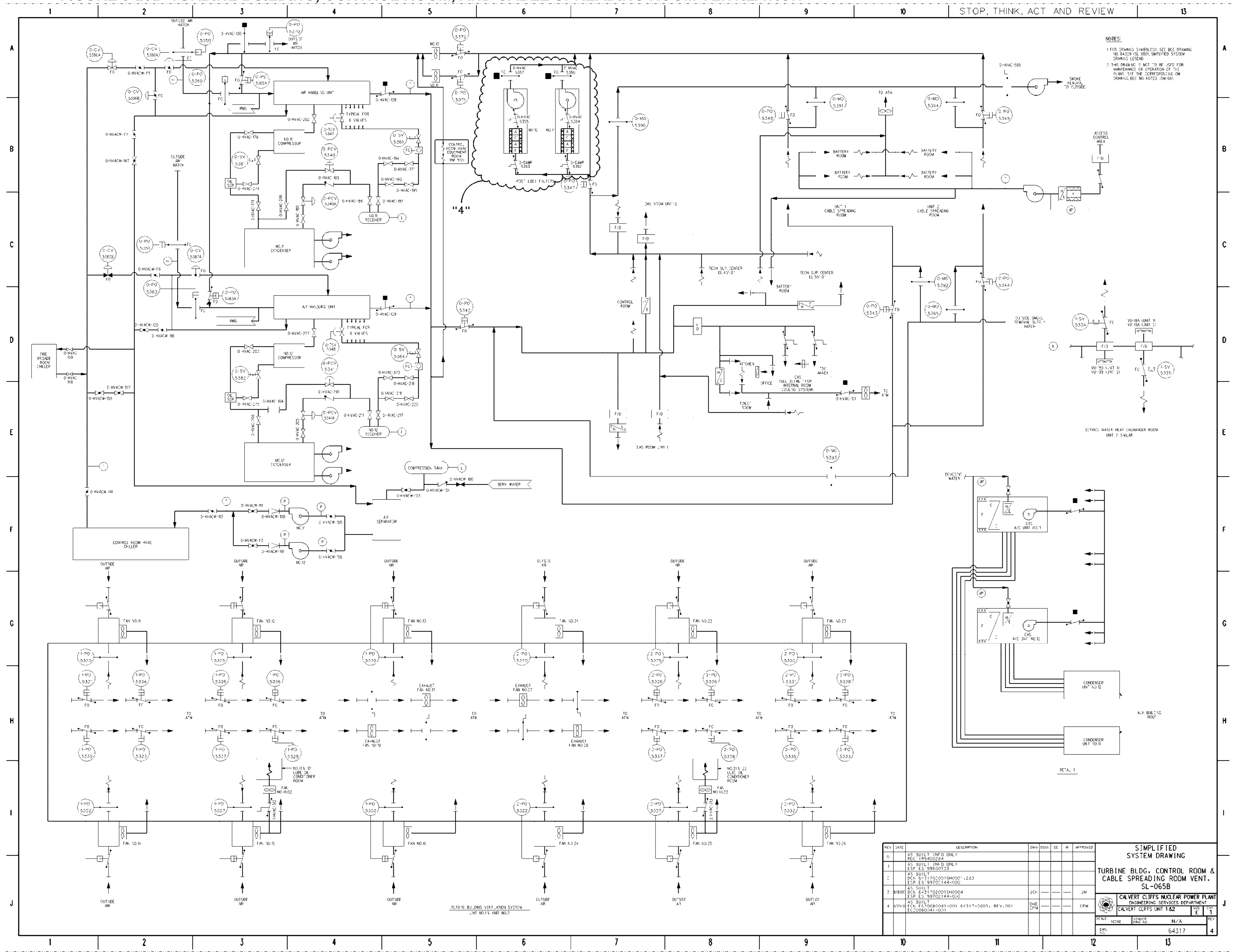


FIGURE 9-20B TURBINE BUILDING, CONTROL ROOM, AND CABLE SPREADING ROOM VENTILATION



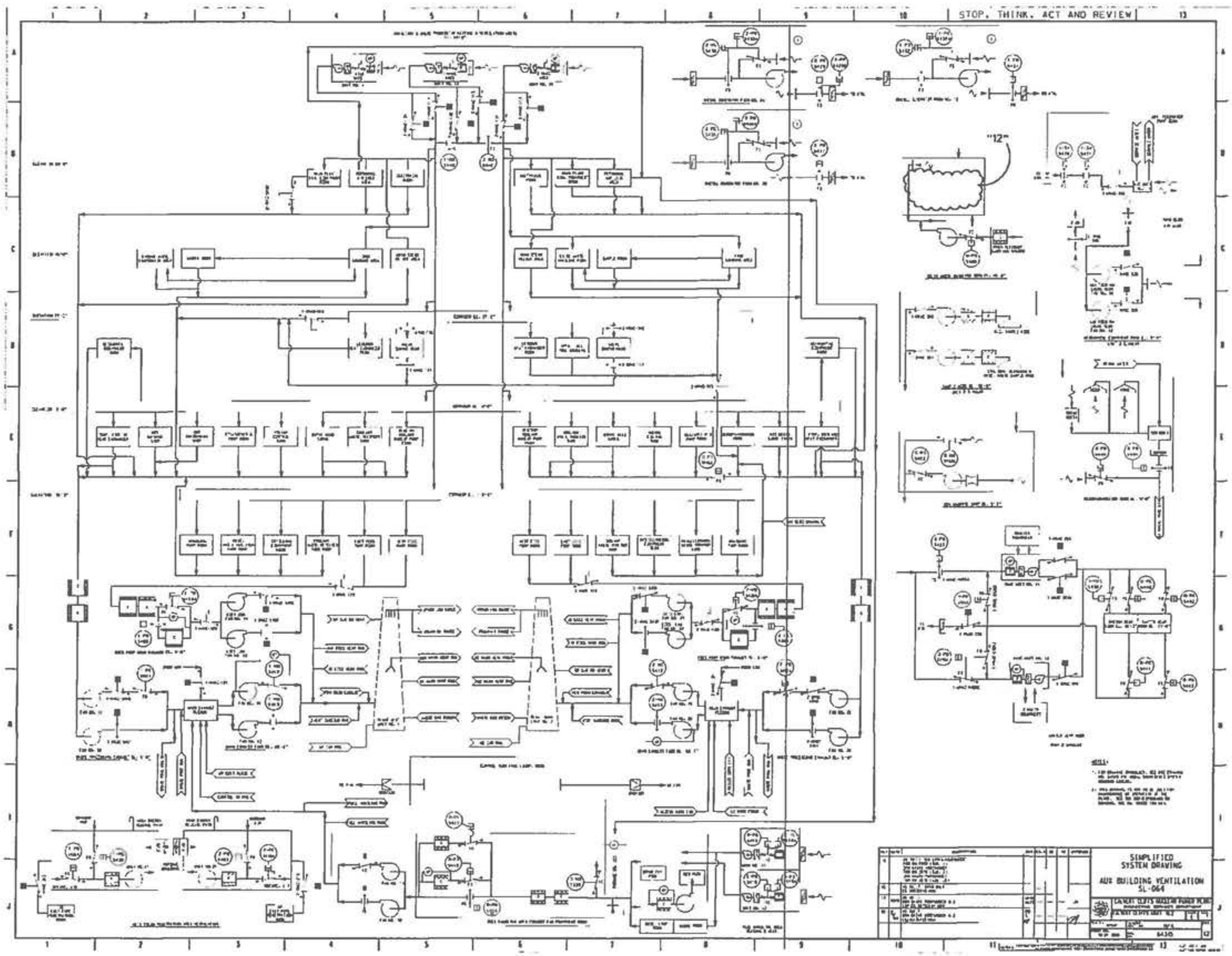

STOP, THINK, ACT AND REVIEW
"12"
SIMPLIFIED
SYSTEM DRAWING
AUX BUILDING VENTILATION
SL-064

FIGURE 9-22 PLANT FIRE PROTECTION

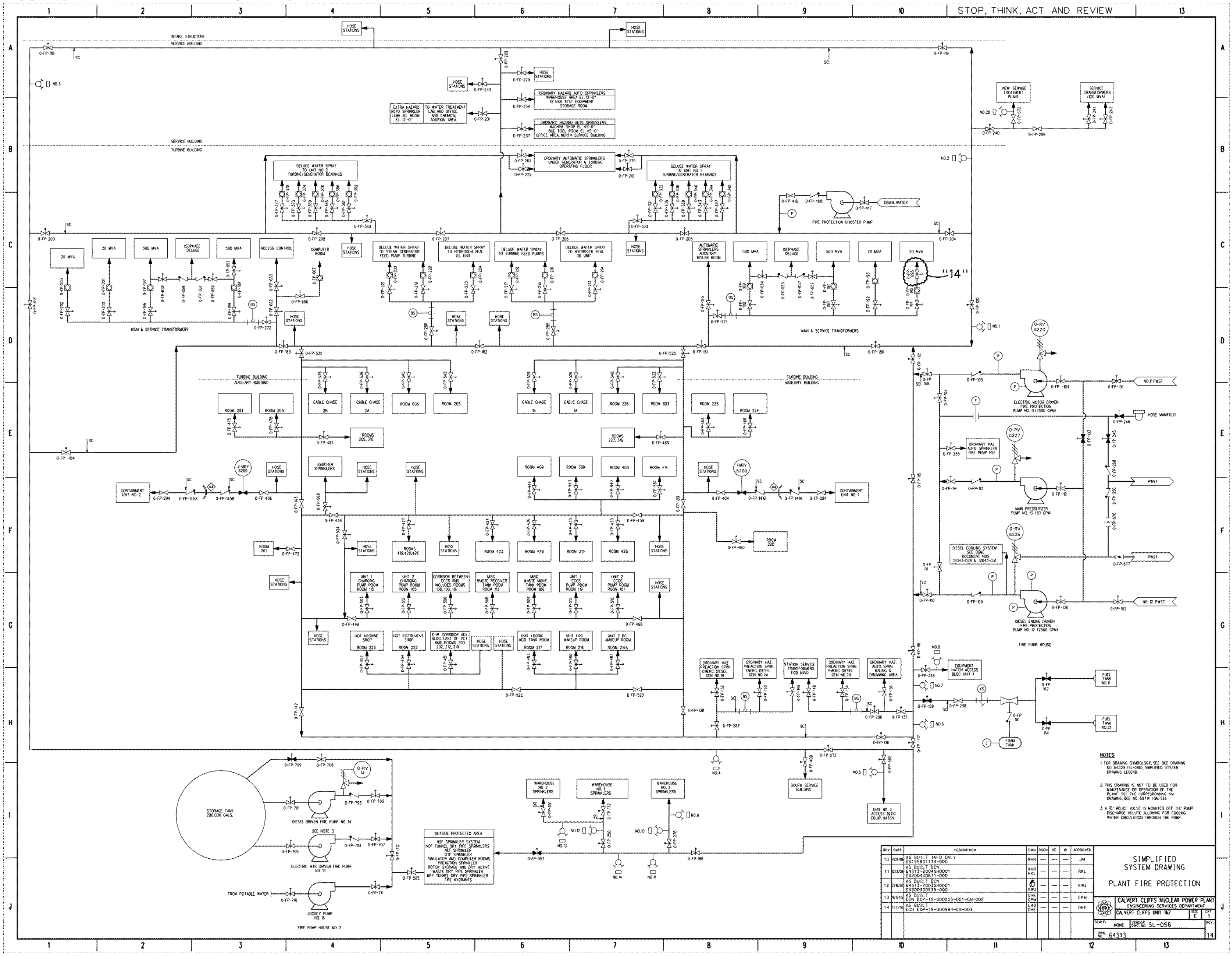


FIGURE 9-23 COMPRESSED AIR – UNIT 1

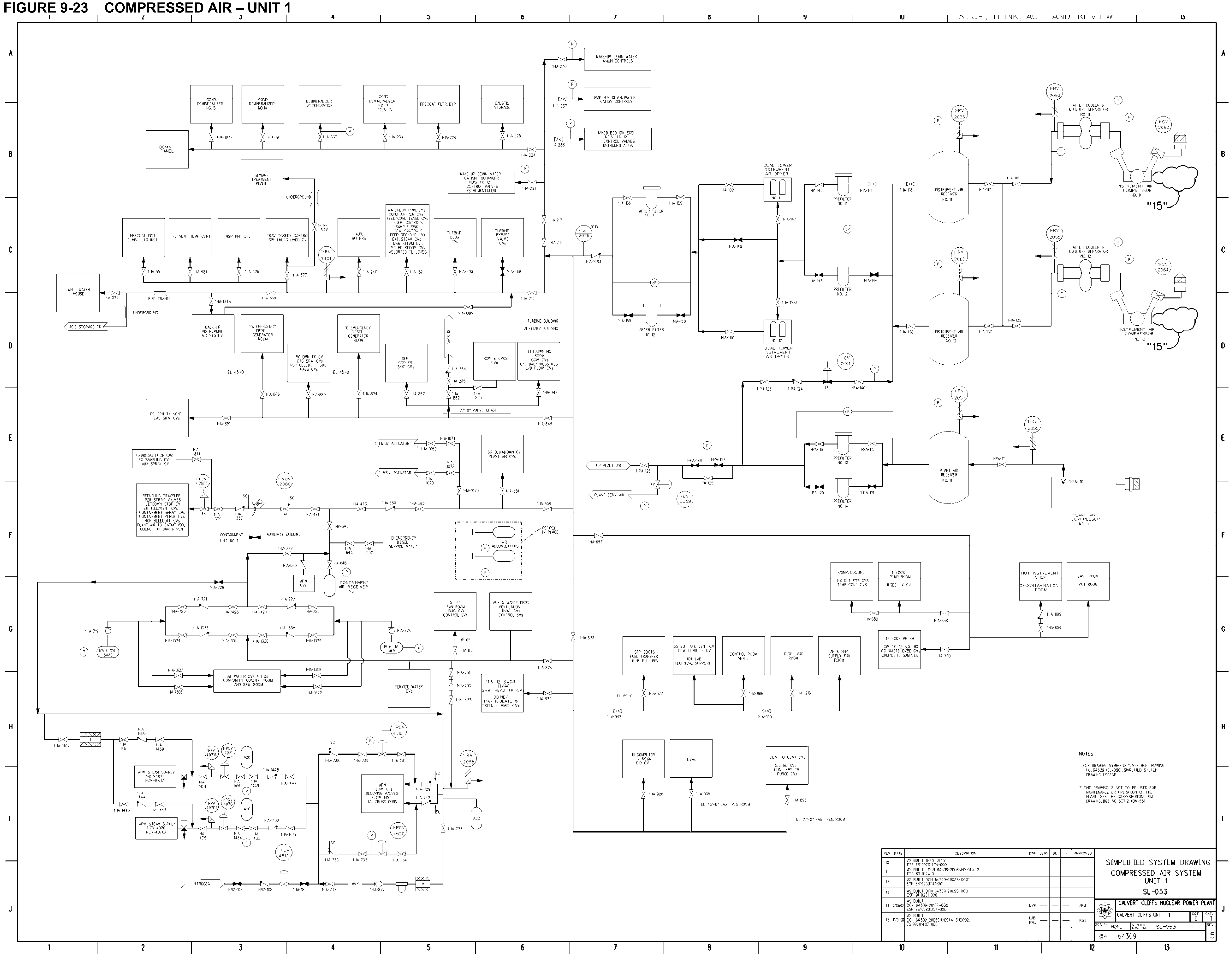
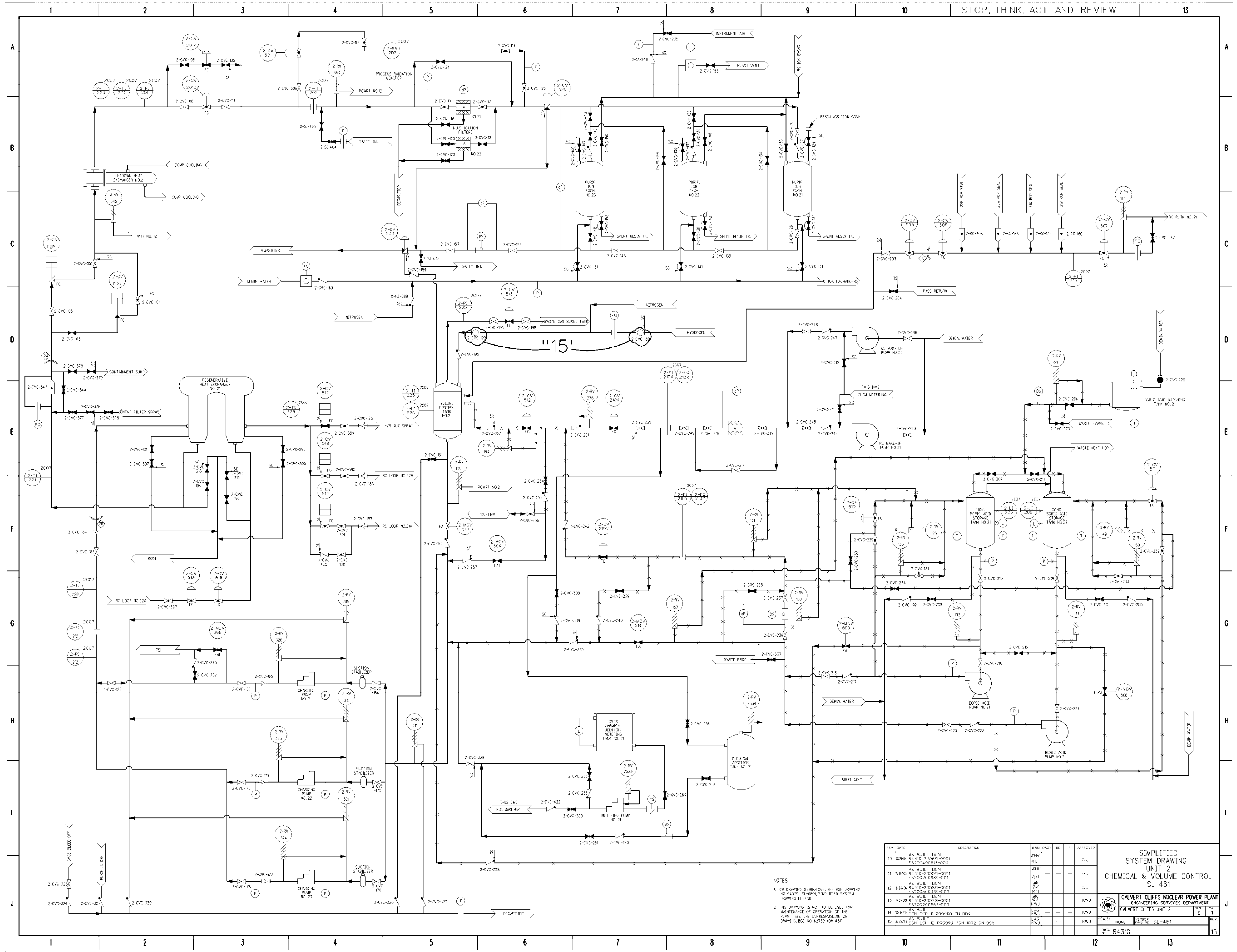


FIGURE 9-24 CHEMICAL AND VOLUME CONTROL SYSTEM – UNIT 2



Revision 46

FIGURE 9-25 COMPONENT COOLING WATER – UNIT 2

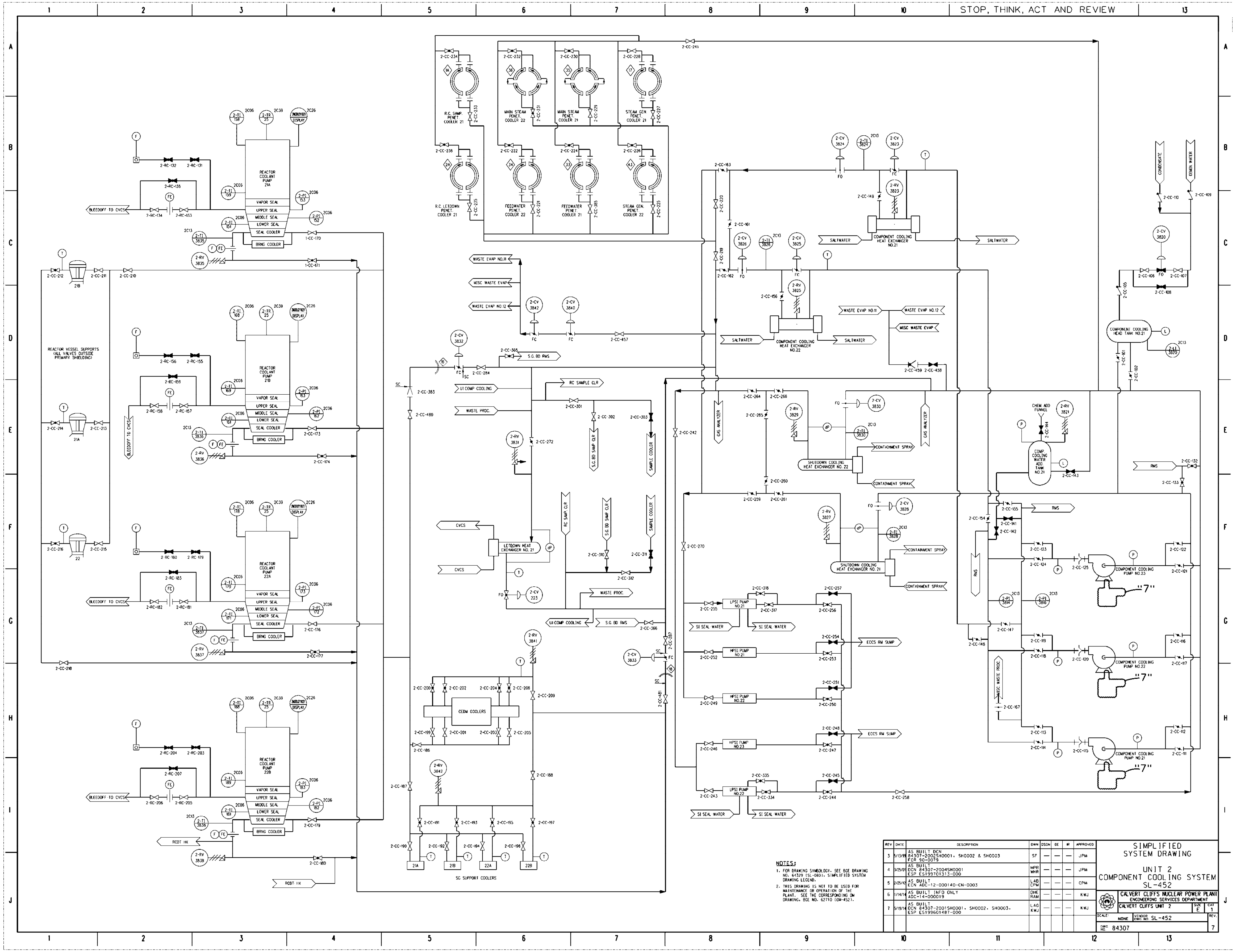


FIGURE 9-26 CIRCULATING AND SALTWATER – UNIT 2

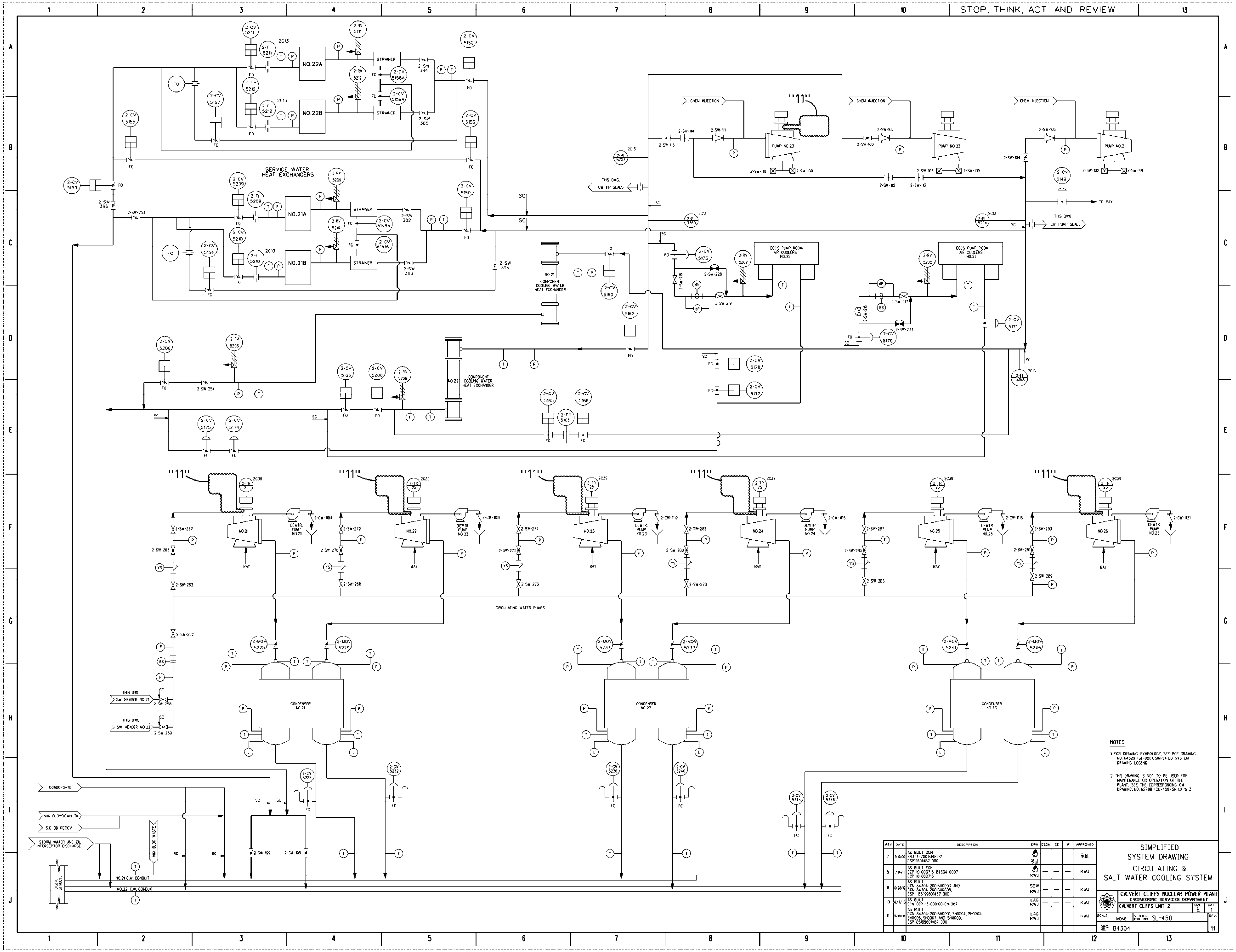


FIGURE 9-27 SERVICE WATER – UNIT 2

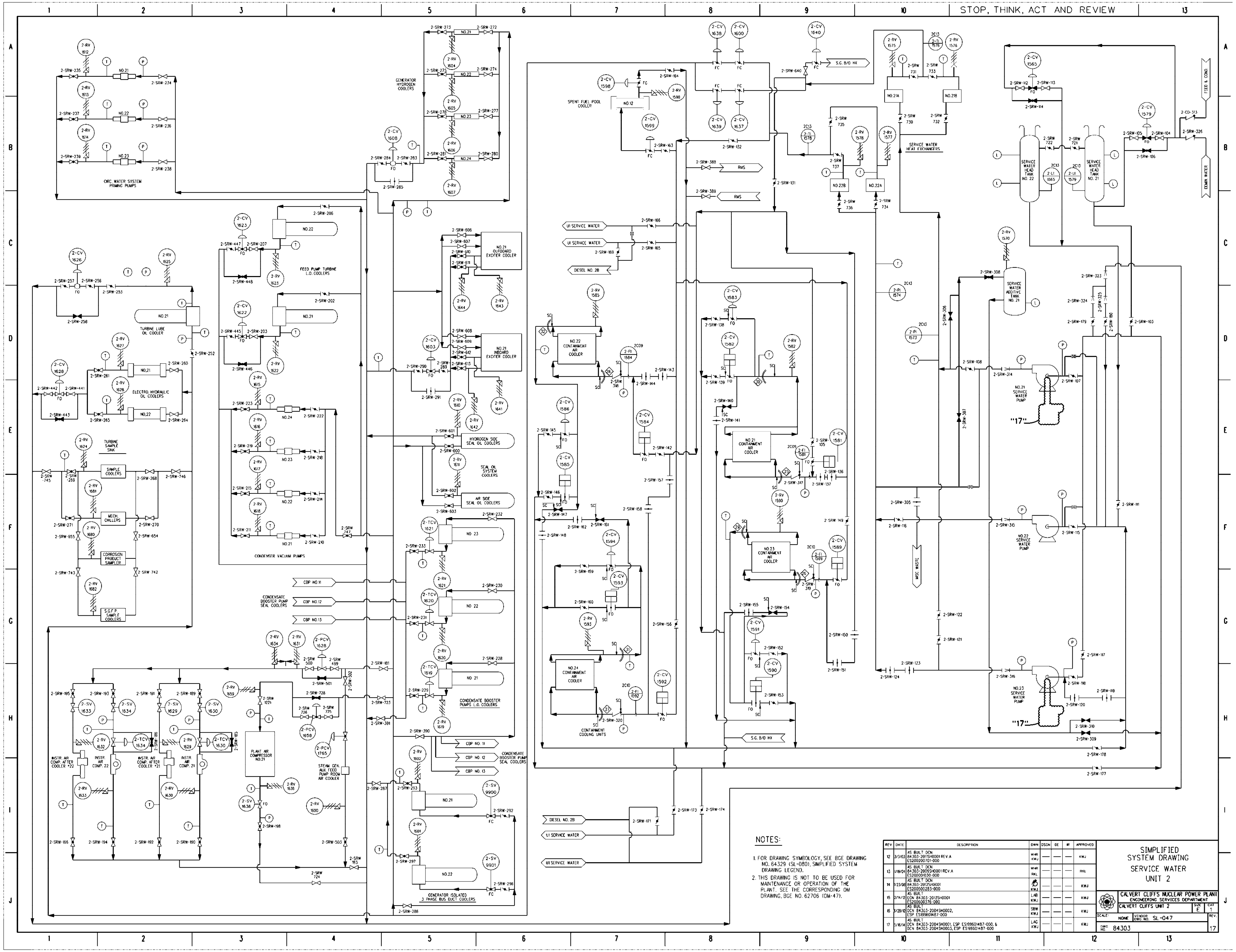


FIGURE 9-28 COMPRESSED AIR - UNIT 2

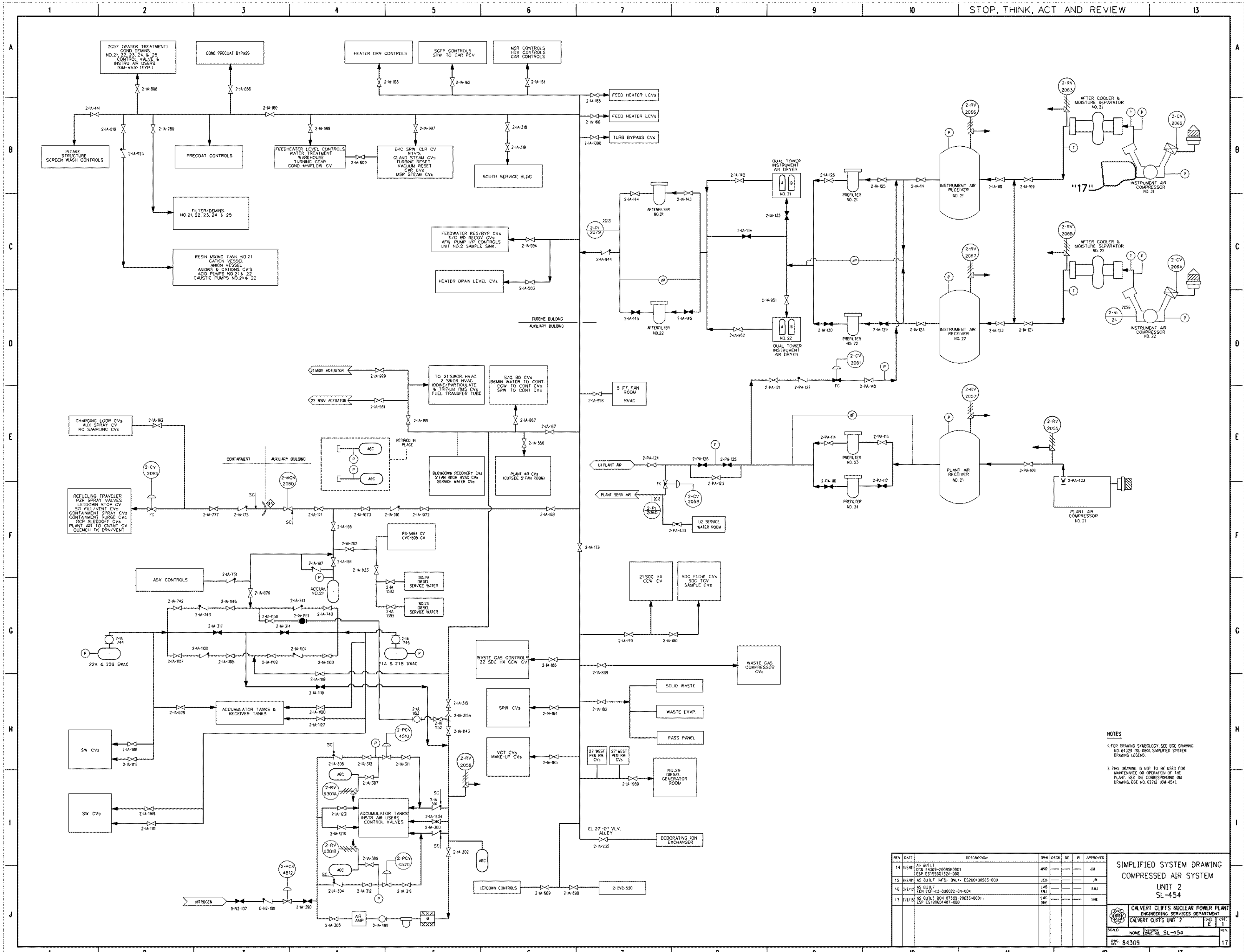


FIGURE 9-29 PLANT WATER AND AIR SERVICE SYSTEM

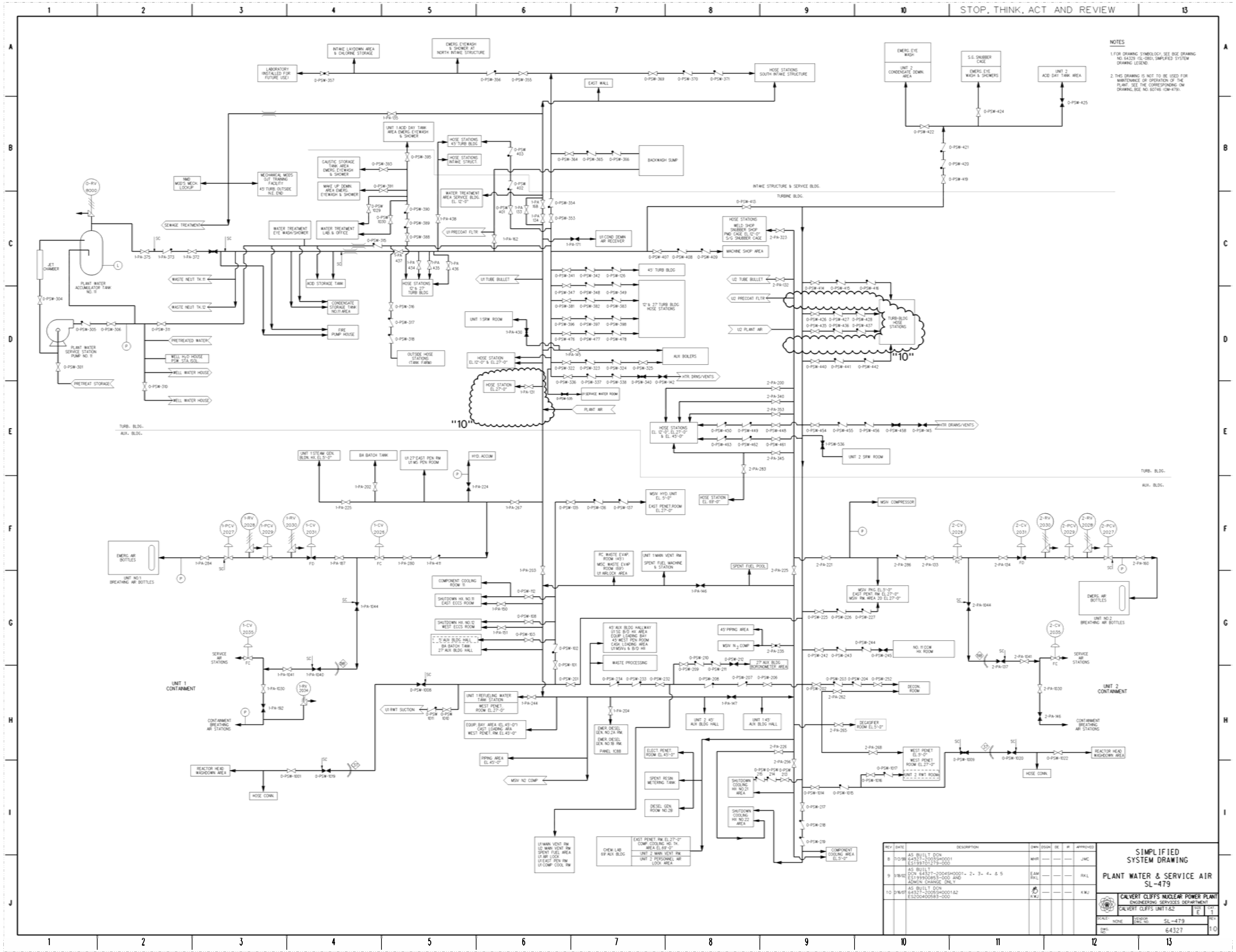
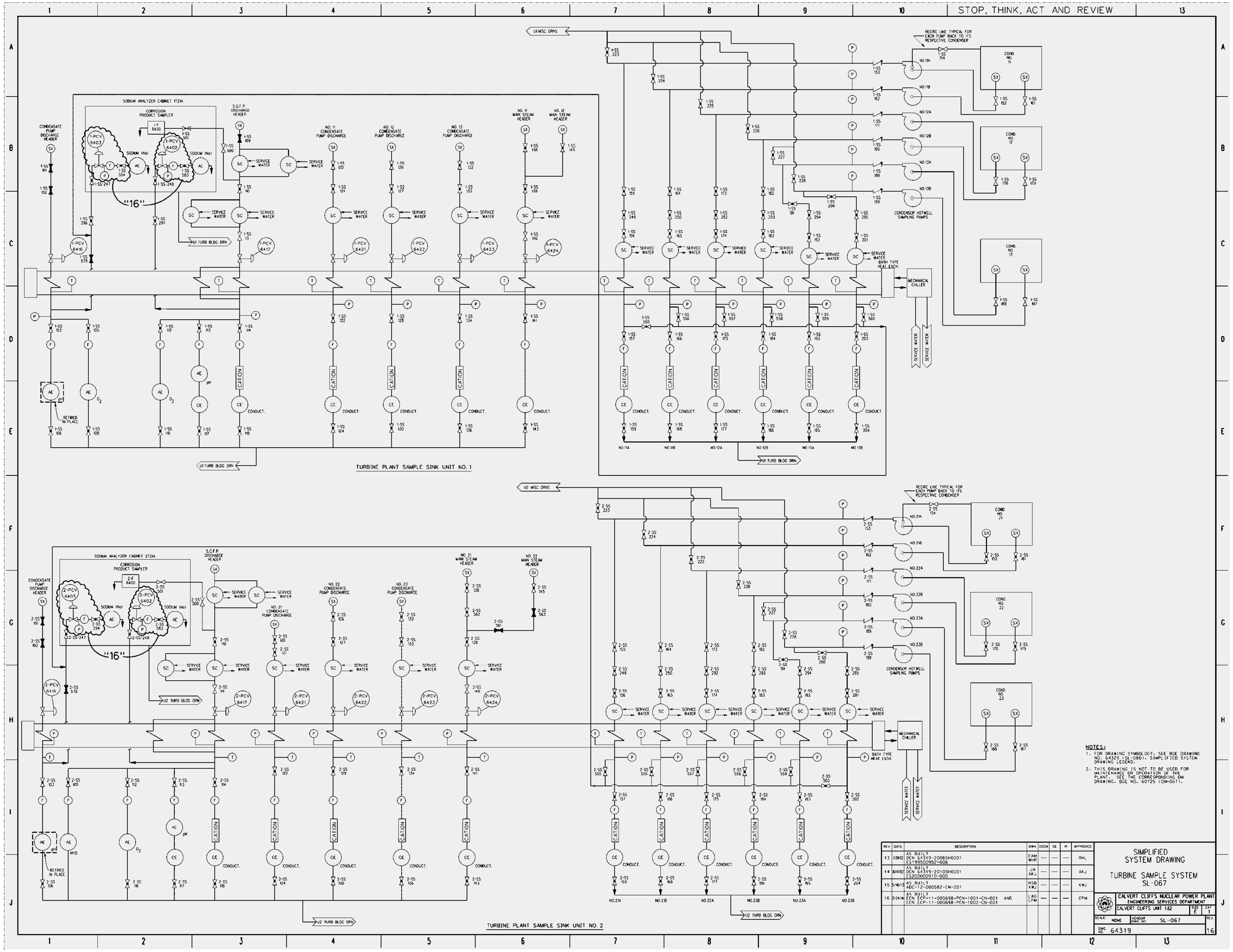


FIGURE 9-30 TURBINE SAMPLING SYSTEM



CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

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STEAM AND POWER CONVERSION SYSTEM

LIST OF ACRONYMS

AFW	Auxiliary Feedwater
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
CST	Condensate Storage Tank
ESFAS	Engineered Safety Feature Actuation System
FCR	Facility Change Request
MSIV	Main Steam Isolation Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valve
NEMA	National Electrical Manufacturers Association
NPSH	Net Positive Suction Head
RCS	Reactor Coolant System
SGFP	Steam Generator Feed Pump
TEMA	Tubular Exchanger Manufacturers Association

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 MAIN STEAM SYSTEM

10.1.1 DESIGN BASIS

The Main Steam System (MSS) is designed to transfer steam from the steam generators to the turbine throttle stop valves, the reheaters, and the turbine-driven pumps. The MSS also controls steam generator pressure by means of steam bypass, dump, or safety valves (high pressure) and main steam isolation valves (MSIVs) (low pressure).

The system is designed to accommodate electrical load changes from 15 to 100% power at a rate of 5% per minute and at greater rates over smaller load change increments, up to a step change of 10%. This is normally accomplished by manual control element assembly movement and adjustment of Reactor Coolant System (RCS) soluble boron concentration. Component design data is contained in Table 10-1.

10.1.2 SYSTEM DESCRIPTION

The MSS is shown in Figures 10-1 (Unit 1) and 10-9 (Unit 2).

Steam generated in each of the two steam generators is piped through the containment wall in separate 34" OD lines. The lines have the flexibility to take the relative movement due to thermal expansion. Each steam line has eight spring-loaded safety valves which discharge to the atmosphere, and an atmospheric dump valve system which relieves at a pressure lower than the setting of the safety valves (Section 10.1.2). This system also discharges to the atmosphere.

The main steam lines are designed to withstand the maximum possible discharge from any one relief valve locally and the full steam generator capacity with all eight valves discharging simultaneously. A stress analysis was performed to determine the effects on the main steam lines, assuming that all steam generator relief valves discharge concurrently. The analysis includes the effects of system weight, internal pressure, seismic and blowdown thrust loads. Bending, torsional loads, and deflections were generated in the analysis.

Mechanical restraints and supports are designed to sustain all the concurrently acting thermal, seismic and seismic anchor movement loads on the piping systems. The supports are designed to carry dead weight, insulation and the contents in the piping system. Supports permit free movement in longitudinal and lateral direction caused by thermal expansion or contraction and earthquake. Restraints and supports at the isolation valves and stop valves are designed to carry, in addition to the above mentioned loads, the dynamic loads due to sudden closure of stop valves.

10.1.2.1 Main Steam Safety Valves

Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve is provided by 16 spring-loaded ASME Code main steam safety valves (MSSVs) which discharge to the atmosphere. Eight of these safety valves are mounted on each of the main steam lines upstream of the steam line isolation valves, but outside the containment. The MSSVs are designed for full flow relief pressure of 1085 psig, thereby ensuring that the secondary system pressure will be limited to within 110% of its design pressure of 1015 psia during the most severe anticipated system operational transient. The opening pressure of the valves is set in accordance with ASME Code allowances, with the minimum set pressure at 935 psig, and the maximum set pressure at 1050 psig. At the nominal set pressure, these valves can pass a steam flow equivalent to an Nuclear Steam Supply System power level of

2737 MWt, coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser). The total relieving capacity for all valves on both of the steam lines in either unit is 12.18×10^6 lbs/hr of saturated steam at 100% rated thermal power (6.088×10^6 lbs/hr per steam generator).

Thrust loadings due to the sudden opening of the MSSVs as well as the unbalanced internal pressure in the discharge stack, which contribute to the force and moments acting on the main steam headers and MSSV inlet piping, were considered in the stress analysis of the MSS. Support and restraint locations, as well as material thicknesses, were selected to accommodate expected loads and reduce material stress. The main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code.

In hot standby, two MSSVs per steam generator are required to provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the RCS via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and operability testing of the MSSVs, which can only be conducted when the RCS is at or above 500°F. It allows operation within these operating conditions with a minimum number of functional MSSVs so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

Startup and/or power operation is allowable with MSSVs inoperable within the limitations of the Technical Specifications. The number of inoperable MSSVs will determine the necessary level of reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{x}\ x\ 106.5$$

where:

SP	=	reduced reactor trip setpoint [percent of rated thermal power]
V	=	maximum number of inoperable safety valves per steam line
106.5	=	Power Level - High Trip Setpoint
X	=	Total relieving capacity of all safety valves per steam line [lbs/hour]
Y	=	Maximum relieving capacity of any one safety valve [lbs/hour]

10.1.2.2 Atmospheric Steam Dump and Bypass System

The steam dump and bypass system is used to rapidly remove RCS stored energy and to limit secondary steam pressure following a turbine-reactor trip. The atmospheric steam dump system consists of two automatically-actuated atmospheric dump valves which exhaust to the atmosphere. The turbine bypass system consists of four turbine bypass valves which exhaust to the main condenser. The power-operated steam dump valves and steam bypass valves reduce, but do not eliminate, the probability of the MSSVs opening following turbine and reactor trips from full power.

The system also provides a means of heat removal during hot standby and during a plant cooldown. The atmospheric steam dump valves are capable of removing reactor decay heat when the condenser is not available.

The removal of reactor decay heat via the atmospheric steam dump system is not a normal mode of operation. Main condenser vacuum will be maintained to enable removal of reactor decay heat through the steam bypass valves until the shutdown cooling system can be initiated. In the event of a loss-of-condenser vacuum, the turbine bypass valves close automatically and the atmospheric steam dump valves open to exhaust the steam generated by decay heat.

Incidents may occur which release steam through the steam dump valves for a short period of time. If an incident occurs which results in atmospheric steam dump system operation, the radioactivity released can be estimated using the steam generator activity obtained from a sample, the duration of dumping, the estimated flow rate during dumping and the measured meteorological conditions. Steam generator blowdown system, main steam line system, and condenser air removal system radiation monitors are also available.

The atmospheric dump valves are positioned by the reactor coolant average temperature error signal when the turbine is tripped. In the event of a turbine trip above a preset power level, a quick-opening signal is provided to fully open both the atmospheric dump valves and the turbine bypass valves. When T_{avg} is reduced to less than T_{γ} (Figure 7-15, Sheet 3), the steam dump valves are modulated as a function of T_{avg}, and the turbine bypass valves are modulated as a function of main steam header pressure.

The total respective capacities of the atmospheric steam dump and turbine bypass valves are 5% and 40% of steam flow with the reactor at full power.

The system controls are arranged for either automatic operation or remote manual control.

10.1.2.3 Main Steam Line Isolation

One main steam line isolation valve assembly is provided on each main line header. The MSIV is a "y"-type, bi-directional, balanced disk globe valve with an American National Standards Institute (ANSI) 600 primary pressure rating; the bonnet closure is of a pressure seal design body and disk seating surfaces, disk guides and backseat are integrally hardfaced with Stellite 21; limit switches are mounted on the yoke to provide full open, full closed and intermediate position indication. The valve is capable of shutting against pressure from either side. A motor operated bypass valve is provided to equalize pressure across the valve before opening. The MSIVs and their actuator systems are designed to Seismic Category I requirements. Descriptive drawings of the valve are shown in Figures 10-2, 10-2A, and 10-2B.

Closure of the MSIV within a maximum of six seconds after a trip signal is initiated, prevents rapid flashing and blowdown of water stored in the shell side of the steam generator, thus avoiding a rapid uncontrolled cooldown of the RCS. Also, the isolation valves prevent release of the contents of the secondary side of both steam generators to the containment in the event of the rupture of one main steam line inside the Containment Structure. During normal operation, these valves remain open; upon low steam generator pressure or high containment pressure, a steam generator isolation signal energizes the closing mechanism of the valves to stop the steam flow.

The manufacturer, Rockwell Manufacturing Company, has supplied fast acting, hydraulic-actuated stop and stop - check valves for use in steam and feedwater systems of the Fort St. Vrain HTGR. These valves are in 6 to 20" size, pressure class 600 to 2500 and of y-type or vertical stem construction. The manufacturer

has supplied air actuated y-type balance disc valves in 16 to 26" size for use as steam isolation valves in U.S. boiling water reactors.

The actuator consists of a hydraulic cylinder directly coupled to a nitrogen accumulator and a control system designed to close the MSIV in case of either an upstream or downstream rupture using stored energy. The gas/hydraulic power package for each valve is separate. Physical separation, missile shielding and power supplies are arranged to prevent a single failure from disabling two hydraulic systems. Figure 10-3 shows a schematic diagram of the hydraulic actuation system for one valve.

The actuator piston is capable of counteracting the stem force resulting from a differential steam pressure of 1000 psi in either direction, in addition to the maximum calculated impingement forces. Speed control is accomplished by throttling oil flow at valves 2-26 and 3-26 (Figure 10-3) which act as an automatic adjusting orifice and a fixed orifice during closure. The orifices provide a constant oil flow rate independent of upstream pressure. Hydraulic pressure under the piston adjusts itself as necessary to counteract the stem force.

Valve closure forces were calculated using the peak magnitude stem and impingement forces mentioned above. The valve actuator was tested on a load cell under no flow and peak force conditions to ensure the 6.0 second closure time assumed in the accident analysis will be met under accident conditions. The maximum measured closure time difference between peak and no flow conditions is 0.8 seconds. The MSIVs' closure time is surveilled to 5.2 seconds, thus ensuring the valve will close within 6.0 seconds under accident conditions. These closure times do not include signal processing delay time.

The MSIV actuation system is a Rockwell A-180 valve-mounted gas/hydraulic stored-energy actuator. Energy for closure of the valve is provided by high pressure nitrogen contained in a spherical chamber above the hydraulic piston. The MSIV is held open by hydraulic fluid exerting pressure underneath the hydraulic piston. Closure of the valve occurs when either solenoid valve 2-89 or 3-89 opens releasing pilot pressure from either hydraulic dump valve 2-28 or 3-28 allowing the valve to open, thus releasing the hydraulic fluid from beneath the MSIV hydraulic piston. The fluid flows through independent manifolds to a common reservoir. A manual override device is available on each of the dump solenoid valves (2-89 and 3-89) which adds local closure capability to the valve. The MSIV will close within 6.0 seconds if either dump valve opens. The stored nitrogen energy provides the force to close the valve. Nitrogen pressure is continuously monitored with local annunciation of pressure, and Control Room annunciation of MSIV trouble, alerting plant personnel of an unusual condition. This ensures adequate nitrogen pressure is available at all times to close the MSIV.

The valve is a "Y"-pattern globe where the bonnet of the valve is at 45° to the flow centerline. This allows the body flow passage to conform to a high efficiency which results in a low pressure drop. The advantage of the balanced disk design is a reduced actuator size due to a reduction in disk loading. The design consists of a small check disk assembly located in the center of the main disk. The assembly opens during reverse flow conditions to equalize pressure above and below the disk. This balances not only disk pressure but also balances disk forces. With the disk pressures balanced, the nitrogen in the accumulator provides the force required to keep the valve closed for at least one hour against full

reverse steam pressure. This provides each MSIV with the capability of holding reverse pressure until a cooldown to below 300°F can be accomplished.

Four pressure transmitters are monitored by four independent bistables for each steam generator. Each steam generator provides two independent pressure channel inputs to redundant engineered safety features actuation system (ESFAS) steam generator isolation two-out-of-four logic matrices. Upon actuation of a single two-out-of-four logic matrix on low steam generator pressure, both gas/hydraulic-operated main steam line isolation valves will close (Section 7.3). The valves are also closed by a containment spray actuation signal.

Automatic closing of the gas/hydraulic-operated MSIVs can be blocked by operating the isolation block switches as the steam pressure is decreased toward the isolation setpoint. The isolation block is automatically removed by a three-out-of-four logic when the steam generator pressure rises to 100 psi above the isolation setpoint pressure.

The operation of the Calvert Cliffs steam generator isolation valves will reliably prevent excessive energy release from the steam generators in the event of a major steam line rupture. This conclusion is based on design analysis, design verification tests, and inservice tests.

The analyses include:

a. A design analysis of the valve body, disk, check element, piston, bonnet, and pressure seal gasket to ensure the requirements of American Society of Mechanical Engineers (ASME) B&PV Code, 1971 Edition through Summer 1971 Addenda, was met.

Design verification tests include:

a. Hydrostatic tests of the system at the factory to ensure adequate system strength;
b. Operational tests of the actuation system at the factory to ensure system flow rates and pressures are in accordance with design;
c. Operational tests of the complete system, including the valve, after installation, to ensure adequate system force and time response; and,
d. Extensive prototype, qualification, and industry experience has demonstrated the ability of the actuator to perform its design function under all anticipated conditions.

The inservice tests will be performed both on-line and during shutdown periods over the life of the plant to ensure continued system reliability. These tests include:

a. A full stroke test performed per the Inservice Test Program; and,
b. Special tests, performed as necessary, to establish nitrogen and hydraulic fluid replacement requirements, and to confirm the strength adequacy of the pressure boundary.

10.1.2.4 <u>Flow Restrictors</u>

Each main steam line is equipped with an integral venturi flow restrictor in the main steam outlet nozzle and an in-line verturi flow restrictor inside the Containment Building. The worst case steam line break would occur between the integral verturi flow restrictor and the in-line venturi flow restrictor. Any break downstream of the in-line venturi flow restrictor would further restrict the steam flow.

10.1.3 DESIGN EVALUATION

The components of the MSS are conventional and of the type that have been extensively used in fossil fuel plants and in other nuclear power plants. Adequate instruments, controls, and protective devices are provided to assure reliable and safe operation. The equipment is designed to the applicable codes and standards and to the best commercial standards and practices.

Seismic Category I requirements are placed on the system up to and including the first isolation valve outside the containment. This includes main steam and feedwater piping up to the isolation valves, the atmospheric dump, the main steam safety valves, the MSIVs, and the steam generators. All other components are designed to Seismic Category II requirements.

10.1.4 TESTS AND INSPECTIONS

Equipment, instruments, and controls are regularly inspected in order to ensure proper functioning of the system. The turbine stop valves, reheater intercept valves, and extraction line non-return valves may be tested while the turbine is in operation.

10.1.5 SECONDARY SYSTEM-SPECIFIC ACTIVITY

To ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR 50.67 limits in the event of a steam line rupture, secondary system-specific activity is controlled by the Technical Specifications. The maximum dose allowed by the Technical Specifications also includes the effects of a coincident 100 gpd primary-to-secondary tube leak in the steam generator of the affected steam line, and concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

TABLE 10-1

DESIGN DATA FOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

Turbine-Generator

		Unit 1	Unit 2
a.	Turbine		
	Throttle pres.[e] (rat./VWO)	815/811 psia	815/815 psia
	Steam moisture (max.)[a]	0.374%	0.374%
	kW @ rating	883,053	950,285
	kW @ VWO	913,719	968,116
	Makeup	0%	0%
	Turbine back-pressure	2" Hg abs.	2" Hg abs.
	No. of extraction	6	6
b.	Generator		
	Rating (kVA)	1,020,000	1,011,900
	Power factor	0.9	0.9
	Voltage	25,000	22,000
	Hydrogen pressure	60 psig	75 psig

Steam Generator Feed Pumps

Type	Double suction, double volute, single stage, vertically split, horizontal centrifugal	
Quantity	2	2
Capacity each (gpm)	15,000	15,000
Head (ft)	2392	2316
Material		
Case	ASTM A296, Gr CA15 or ASTM A487, Gr CA6NM Class A	
Impeller	ASTM A296, Gr CA15 or ASTM A743, Gr CA6NM	
Shaft	ASTM A276, Type 410	
Driver	Condensing, nonextracting, dual admission, horizontal steam turbine	
Codes	ASME B&PV Code, Section VIII, Standards of the Hydraulic Institute, National Electrical Manufacturers Association (NEMA), ANSI	

Steam Generator Steam-Driven Auxiliary Feed Pumps

Type	Horizontal, split case, multistage, centrifugal
Quantity	2 (per unit)
Capacity each (gpm)	700
Head (ft)	2490
Material	
Case	ASTM A217, Gr C5
Impeller	ASTM B148, Gr 9A, ASTM A743 Gr CA6NM or ASTM A296, Gr CA6NM
Shaft	ASTM A276, Type 410; or ASTM A479, Type 410
Driver	Single stage, noncondensing steam turbine
Seismic requirements	Category I
Codes	ASME B&PV Code, Sections VIII and IX, Standards of the Hydraulic Institute, ANSI

TABLE 10-1

DESIGN DATA FOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

<u>Steam Generator Motor-Driven Auxiliary Feed Pumps</u>	
Type	Horizontal, split case, multi-stage, single suction, opposed impeller centrifugal pump
Quantity	1 per unit
Capacity each (gpm)	450
Head (ft)	2800
Material	
Case	ASME SA-216-WCB
Impeller	ASTM A217, Gr CA-15
Shaft	ASTM A276, Type 410, Cond. T
Driver	Electric motor 500 hp, 4160 Volts, 60 Hz, 3 phase, 3500 rpm
Seismic requirements	Category I
Codes	ASME B&PV Code, Sections III, V, IX, Standards of the Hydraulic Institute, NEMA, ANSI
<u>Condensate Pumps</u>	
Type	Vertical centrifugal
Quantity	3 (per unit)
Capacity each (gpm)	8250
Head (ft)	490
Material	
Case	ASTM A48, C1. 40
Impeller	ASTM A296, Gr CA15
Shaft	ASTM A276, Type 410
Driver	Electric motor 1250 hp, 4160 Volts, 60 Hz, 3 phase 1160 rpm
Codes	ASME B&PV Code, Sections VIII and IX, Standards of the Hydraulic Institute, NEMA, ANSI
<u>Condensate Booster Pumps</u>	
Head (ft)	750
Capacity each (gpm)	8540
Material	
Case	ASTM A217, Gr C5
Impeller	ASTM A296, Gr CA15
Shaft	ASTM A276, Type 410
Driver	Electric motor 2000 hp, 4160 Volts, 60 Hz, 3 phase, 1780 rpm
Codes	ASME B&PV Code, Sections VIII and IX, Standards of the Hydraulic Institute, NEMA, ANSI

TABLE 10-1

DESIGN DATA FOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

Heater Drain Pumps	
Type	Vertical centrifugal
Quantity	2 (per unit)
Capacity each (gpm)	4290 (Unit 1); 4870 (Unit 2)
Head (ft)	900 (Unit 1); 740 (Unit 2)
Material	
Case	13-4 SS
Impeller	13-4 SS
Shaft	ASTM A276, Type 410 HT
Driver	Electric motor 1250 hp, 4160 Volts, 60 Hz, 3 phase 1760 rpm
Codes	ASME B&PV Code, Sections VIII and IX, Standards of the Hydraulic Institute, NEMA, ANSI
Condenser	
Type	Three shell, single pass with divided water boxes, surface condenser
Quantity	1 (per unit)
Design duty (Btu/hr)	5.910×10^9 (Unit 1); 5.886×10^9 (Unit 2)
Heat transfer area (ft^2)	453,000 (each)
Design pressure	Shell: 29.5" Hg vacuum Water box: 25 psig
Material	
Shell	ASTM A284, Gr C
Tubes	ASTM B676, Austenitic Stainless Steel and ASTM B338, Gr 2, Titanium (Unit 1) ASTM B338, Gr 2, Titanium (Unit 2)
Tube sheets	ASTM B169, Alloy 614 (Unit 1) ASTM SB265, Gr 1 and Gr 2 bonded to ASTM SA516, Gr 55 (Titanium on Carbon Steel) (Unit 2)
Codes	Standards of the Heat Exchange Institute
Feedwater Heaters	
Type	Closed, U-tube
Material	
Shell	ASTM A212, Gr B
Tubes	ASTM A249, Type 304L
Tube sheets	ASTM A212, Gr B
Codes	ASME B&PV Code, Section VIII, Heat Exchange Institute[(b)]

TABLE 10-1

DESIGN DATA FOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

HEATER NO.	DESIGN DUTY EACH (Btu/hr)	DESIGN PRESSURE (psig) SHELL	TUBE	DESIGN TEMP. (°F) SHELL	TUBE	HEAT TRANSFER AREA EACH FT^2
11 A,B,C	168.40×10^6	50 +Vac.	300	300	300	10,160
12 A,B,C	182.00×10^6	50 +Vac.	300	300	300	13,585
13 A,B	311.00×10^6	75 +Vac.	700	350	325	16,905
14 A,B	162.40×10^6	125 +Vac.	700	375	375	11,830
15 A,B	297.00×10^6	225	700	425	400	22,210
16 A,B	370.50×10^6	475	1500	500	475	23,390
11 A,B,C (Drain Cooler)	20.30×10^6	50 +30" Hg Vac	300	300	300	2,780
21 A,B,C	134.20×10^6	50 +Vac.	300	300	300	11,320
22 A,B,C	140.16×10^6	50 +Vac.	300	300	300	10,555
23 A,B	156.63×10^6	50 +Vac.	700	300	300	12,645
24 A,B	151.30×10^6	75 +Vac.	700	350	350	11,360
25 A,B	51.71×10^6	150	700	400	375	21,120
26 A,B	595.10×10^6	450	1500	500	460	29,625
21 A,B,C (Drain Cooler)	23.30×10^6	50 +Vac.	300	300	300	2,490

TABLE 10-1

DESIGN DATA FOR STEAM AND POWER CONVERSION SYSTEM COMPONENTS

SYSTEM	**APPLICABLE CODE**	**DESIGN PRESSURE (psig)**	**DESIGN TEMP. (°F)**
Main Steam Piping	ANSI B31.1	1000	580
Main Steam Penetration Piping (including atmospheric dump)	ANSI B31.7, CL2	1000	580
Main Steam Line Flow Restrictors	ASME Sec III, CL II	1000	580
Main Steam Isolation Valves	ASME Sec III, CL II	1085	580
Main Steam Safety Valves	---	985	550
Feedwater	ANSI B31.1	1500[(c),(d)]	460
Feedwater Penetration	ANSI B31.7, CL2	1500[(c)]	460
Condensate	ANSI B31.1	612	400
Auxiliary Steam	ANSI B31.1	225	425
Auxiliary Feedwater Discharge (Steam Driven Train)	ANSI B31.1	1440	100
Auxiliary Feedwater Discharge (Motor Driven Train)	ASME Sec III, CL III	1480	100
Auxiliary Feedwater Suction (Steam-Driven Train)	ANSI B31.1	285	100
Auxiliary Feedwater Suction (Motor-Driven Train)	ASME Sec III, CL III	285	100

(a) This is the turbine design value for optimum operation. However, higher steam moisture values are within the total design envelope for turbine operation. Normal inspection, maintenance, repair, and replacement of the steam generator moisture separating equipment will limit the moisture carryover of the outlet steam, which feeds the turbine. This will prevent excessive turbine erosion and eliminate turbine damage, within the normal turbine inspection program.

(b) Also see Sections 10.2.2.1 and 10.2.3.

(c) Design pressure was revised from 1650 psig for feedwater penetration piping and 1600 psig for feedwater piping.

(d) 1400 psig may be used as the design pressure of the feedwater lines inside Containment when overpressure protection, as provided by the MSSVs, is demonstrated to be acceptable. The feedwater penetration pipe is excluded from this change.

(e) Actual turbine throttle pressure may be less than the design value due to steam generator tube plugging. However, for reduced throttle pressure the turbine-generator may not be capable of producing rated power output.

10.2 <u>CONDENSATE AND FEEDWATER SYSTEM</u>

10.2.1 DESIGN BASIS

The Condensate and Feedwater System is designed to provide a means for transferring the condensate from the condenser hotwell to the steam generators (while at the same time raising the temperature and pressure) and providing a means for controlling the quantity of feedwater into the steam generators. Component design data is contained in Table 10-1.

In Bulletin 79-13, the NRC required an examination of feedwater piping welds. Baltimore Gas and Electric Company completed limited radiographic and visual examinations and found no evidence of cracking. After inspection of the radiographic procedure and film, the NRC found this examination acceptable.

10.2.2 SYSTEM DESCRIPTION

The Condensate and Feedwater System is shown in Figure 10-4 (Unit 1) and Figure 10-11 (Unit 2).

Condensate from the hotwells is pumped by two electric motor-driven condensate pumps (with another pump held on standby) through the gland steam condenser, the condensate demineralizer and precoat filtering system, the lowest feedwater heating stage drains coolers, and the two lowest pressure feedwater heating stages (three heaters per stage) to the suction of the three condensate booster pumps (two operating, one on standby). These pumps deliver the condensate to the two turbine-driven steam generator feed pumps (SGFPs) through two parallel sets of three feedwater heaters. The SGFPs pump the feedwater through two parallel high pressure heaters to the steam generators.

The heating steam for the feedwater heaters, as shown on Figure 10-5 (Unit 1) and Figure 10-10 (Unit 2), is extracted from the turbine as follows: extraction for High Pressure Heaters No. 16A and B and for the first stage reheater (two in parallel for Unit 1, four in parallel for Unit 2) is from the high pressure turbine; for Heaters No. 15A and B (25A and B) it is from the cold reheat line; for the remaining four heating stages it is from different stages of the low pressure turbine.

The drains from the second stage reheater are routed to the high pressure heaters, whereas those from the first stage are routed to Heaters No. 15A and B (25A and B). The steam which condenses in the three higher pressure feedwater heaters is collected (by cascading) in two heater drain tanks, which also collect drains from the moisture separator vessels. From this point, the drains are pumped via two heater drain pumps into the condensate system between Heaters No. 14 (24) and 15 (25). The three lower pressure feedwater heating stages cascade their drains back to the condenser hotwell. Chemicals are added to the condensate flow for oxygen scavenging and pH control. The chemicals approved for use are listed and controlled by Chemistry Section procedures.

10.2.2.1 <u>Feedwater Pumps</u>

Two turbine-driven pumps supply the required feedwater flow rate to the steam generators to match the steam flow demand by the plant turbine generator and auxiliaries. The driving steam for high load operation is hot reheat, whereas for low load, an automatic changeover to main steam takes place. Auxiliary steam from the auxiliary boiler may also be used. Several provisions, i.e., low vacuum, loss-of-turbine lube oil pressure, thrust bearing wear, turbine overspeed, and manual turbine trip, are made available to protect the turbine-driver/pump units

during operation. Specific critical values of the operating parameters for the turbine drivers and the feed pumps are established; a trip of a driver-pump unit set will occur when any of the preselected values is exceeded. The feed pump high discharge pressure trip and pump speed controls are credited as the means to limit the feedwater system pressure to within the system design pressure rating.

10.2.2.2 Automatic Control in Conjunction With Feedwater Regulating Valves

Each steam generator is equipped with three-element control in order to produce a demand signal for feedwater flow which is a function of the difference between the feedwater and steam flows, trimmed by the steam generator downcomer level error. The feedwater flow demand signal for each steam generator is sent to the corresponding feedwater regulating valve and in combination with the turbine driver speed control system, controls the level in each steam generator by modulating the feedwater flow.

Between approximately 2 and 100% power, the feedwater control system automatically controls the steam generator water level. Manual control capability can be utilized at any power level.

Upon reactor trip, the main feedwater valves are automatically closed and the feedwater bypass valves are automatically opened to approximately 3.8% of main feedwater valve flow. When an abnormally high steam generator level is sensed, the turbine is tripped.

10.2.2.3 Main Feedwater Pump Turbine Speed Control

The speed of each main feedwater pump turbine is controlled as a function of the main feedwater control valve demand signal which regulates the flow of condensate to the SGs.

The main feedwater control system demand signal is converted to a pump speed demand signal by a function that produces a relatively constant differential pressure across the control valve with the varied flow demands. Each set of feedwater pump manual/auto control stations have manual bias adjustment to adjust the fraction of feedwater flow through each pump to accommodate for any process abnormalities.

The steam supply source for normal operation up to approximately 40% of the SGFP output is main steam. Above this point, hot reheated steam is used. A connection to the auxiliary steam supply is provided on the hot reheat inlet and can be used if required.

10.2.2.4 Radioactivity Monitoring

A radioactivity monitor is located in the steam generator blowdown line. When a high radioactivity level is detected, an alarm is annunciated in the Control Room and the liquid effluent from the steam generators is diverted to the miscellaneous waste processing system. This blowdown liquid monitor has an alarm setpoint selected and adjusted in accordance with the Offsite Dose Calculation Manual. The steam generator blowdown tank vents alternatively to the heater drain tank or to the Feedwater Heaters No. 13A and 13B shell side for Unit 1 (Figure 10-1), and to the heater drain tank only for Unit 2 (Figure 10-9). Chapter 11 contains additional information on the radiation monitoring system.

10.2.3 DESIGN EVALUATION

The plant can carry more than half load with one-half of the feedwater heaters, i.e., No. 13, 14, and 15 (23, 24, and 25), shut down and isolated entirely from the system, with only one condensate pump in operation, and/or with only one SGFP in service.

The feedwater heaters are designed to meet Tubular Exchanger Manufacturers Association (TEMA) Standards and the ASME B&PV Code, Section VIII, Pressure Vessels, with the exception that the SGFP speed controls and high discharge pressure trips (in addition to the thermal relief valves) are credited for providing overpressure protection.

Because overpressure protection of the high pressure feedwater heaters (16 A/B and 26 A/B) is not strictly in accordance with ASME B&PV Code, Section VIII, a variance was obtained from the Maryland boiler and pressure vessel safety requirements defined in the Code of Maryland Regulations. The basis for this variance was that the feedwater system is protected by the SGFP steam turbine governors and modified high discharge pressure trips. The modification to each pump's high discharge pressure trip included the addition of two pressure switches and corresponding contacts in the alarm and trip circuits to ensure that the failure of one pressure switch will not disable the SGFP high discharge pressure trip (Facility Change Request [FCR] 88-0128 and FCR 90-26).

10.2.4 TESTS AND INSPECTIONS

Equipment, instruments, and controls are regularly inspected in order to ensure proper functioning of the system. The motor-driven pumps and controls can be given preoperational tests after erection and before plant warm-up.

A long-term program, consisting of systematic measures to ensure that erosion/corrosion does not affect high energy carbon steel systems, is in place. This program, which meets the intent of the Nuclear Management and Resources Council guidelines per NUREG-1344, Appendix A, applies to both single-phase and two-phase systems.

10.2.4.1 Tank Reliability

The Quality Control requirements for Condensate Storage Tank Nos. 11 and 21 and Fuel Oil Storage Tank Nos. 11 and 21 were included entirely within the applicable equipment specification and not supplemented by the requirements of the Bechtel generic specification BQC-200. The specification for these tanks invoked specific codes and standards to set forth material, fabrication and erection requirements, as well as to stipulate performance of certain tests and the documentation of the results of those tests.

The vendor for these tanks was selected from the Bechtel Approved Bidders' List which includes only those vendors who have undergone a quality assurance audit by Bechtel in accordance with Bechtel's requirements. Vendors on this list were audited periodically to ensure that Bechtel's quality assurance standards were satisfied.

After release of the order to the vendor for these tanks, a Bechtel shop inspector was assigned to follow this order by the Bechtel Procurement Department to assure that the tank materials were produced in accordance with the specification, drawings and purchase order requirements. Since these tanks are field erected, shop work was limited to the cutting and forming of plate and the Bechtel shop inspector's duties included the checking of incoming material for heat numbers and mill test certifications and the inspection of materials after shop operations.

At the site, the vendor erected the tanks and was responsible for ensuring that code and specification requirements were met. For Condensate Storage Tank Nos. 11 and 21, these requirements included welder and welding procedure qualifications according to ASME Code, Section IX, spot radiography, vacuum box testing and hydrostatic testing according to American Water Works Association Code D-100. For Fuel Oil Storage Tank Nos. 11 and 21, these requirements included welder and welding procedure qualifications according to ASME Code, Section IX, spot radiography, vacuum box testing and hydrostatic testing according to American Petroleum Institute Code 650. During the erection, Bechtel welding personnel maintained surveillance over welding and non-destructive examination. Documentation of satisfactory acceptance of the required tests has been completed and is on file at the job site. These surveillance activities included reviewing of welding procedures and welder qualifications, accepting non-destructive examination results, including radiographs, and maintaining a surveillance of storage and general construction practices followed at the site.

Additional design and quality control requirements for the condensate storage tank are given in Section 6.3.5.1.

10.3 AUXILIARY FEEDWATER SYSTEM

10.3.1 DESIGN BASIS

The Auxiliary Feedwater (AFW) System (Figures 10-13 and 10-14) is designed to provide feedwater to the steam generators for the removal of sensible and decay heat, and to cool the primary system to 300°F in case the condensate or the main feedwater systems are inoperative. The AFW trains may also be used for normal system cooldown to 300°F. Component design data is contained in Table 10-1.

Reactor decay heat and sensible heat are transferred to the steam generators by natural circulation of the reactor coolant if power is not available for the reactor coolant pumps. Generic Letter 81-21 required an assessment of the facility's ability to properly manage a natural circulation cooldown event. In a natural circulation event, AFW must be available during the time the RCS is cooled to 300°F plus an additional period to allow cooling of the reactor vessel head. The assessment revealed that the facility has three condensate storage tanks, a demineralized water tank and a pretreated water storage tank, all of which can be used to mitigate this event. The emergency diesel generators can supply power to the well water pumps which transport water from the ground to the pretreated water storage tank and provide an unlimited supply of water. The NRC concluded, based on this information, there is sufficient water for a natural circulation cooldown (Sections 1.2.9.8, 9.4.4, 10.3.2).

Generic Letter 81-14 requested a walkdown of the non-seismically-qualified portions of the AFW systems to identify apparent and practically-correctable deficiencies of seismic qualification.

Baltimore Gas and Electric Company conducted an evaluation which concluded that the AFW systems exhibit a high degree of inherent seismic resistance. The majority of the system's original design was accomplished using the seismic design criteria typical for other Calvert Cliffs Category I systems. For those few portions of the system that were identified as non-seismic, the as-built configuration of the systems was examined and it was determined that the safe shutdown earthquake would not have a significant effect on system function.

The NRC concurred with the results except for the system manual valves. Subsequently, these valves were seismically-qualified by analysis and the results were accepted by the NRC.

10.3.2 SYSTEM DESCRIPTION

Three AFW pumps are installed, consisting of one motor-driven and two non-condensing steam turbine-driven pumps. For a shutdown, only one pump is required to be operating, the others are in standby. Upon automatic initiation of AFW, one motor-driven and one turbine-driven pump automatically start.

The turbine driver is supplied with steam from the steam generator as long as the pressure is above 50 psig. Each turbine has a manually-set governor for controlling turbine speed. Once set for a certain speed, the governor is designed to maintain approximately constant speed with a minimum of 50 psig steam pressure. The motor-driven pump is supplied from an electrical bus which can be powered by a site emergency diesel generator.

There is a bypass valve to each turbine steam supply valve. To avoid turbine overspeed on startup, initial steam flow is controlled by first opening the bypass valve and then opening the main supply valve after an appropriate time delay.

The steam-driven pumps have a capacity of 700 gpm at 2490' total discharge head at 3990 rpm. Shutoff head for the turbine-driven pump is 2950'. The motor-driven pump has a capacity of 450 gpm at 2800' total discharge head at 3560 rpm. Shutoff head for the motor-driven pump is 3300'. A feed rate of 300 gpm to the steam generator(s) is necessary to remove decay heat and reduce the RCS temperature to 300°F.

These pumps take suction from a 350,000 gallon condensate storage tank which is protected against tornadoes and horizontal tornado-generated missiles. The tank provides a net positive suction head (NPSH) in excess of that required. Tornado protection for the tank consists of a Seismic Category I concrete structure of sufficient thickness to stop horizontal tornado-generated missiles and to resist tornado wind pressures. Bursting pressures are relieved by baffled, missile-proof vents.

In addition to the enclosure for No. 12 Condensate Storage Tank (CST), there is an enclosure for the piping header from the CSTs to the AFW pump suctions. This Category I reinforced concrete enclosure is in the Tank Farm located adjacent to No. 11 CST and No. 21 CST. This enclosure protects the Category I AFW header, connecting piping, and associated valves from the various natural phenomenon discussed above.

At the low level alarm point, the CST provides 300,000 gallons of water for decay heat removal and cooldown of both units. By adjusting the feedwater flow to the permissible cooldown rate, decay heat removal and cooldown of both units can be accomplished in six hours. The 300,000 gallons are also adequate to maintain the RCS at hot standby conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power, or to remove decay heat from both units for more than 10 hours after initiation of cooldown and still maintain normal no-load water level in the steam generators. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The motor-driven AFW pump has been provided with a fire hose connection on its suction as an alternate water source. Likewise, on loss of electric power (Appendix R scenario), a siamese fire hose connection may be installed at the automatic recirculation valve on the discharge side of the pump.

The steam generated during decay heat removal and cooldown during an electric power failure will be discharged by the atmospheric dump valves, except for what is used for the turbine-driven auxiliary feed pump. If electric power and the condenser are available, it is planned to use the condenser as a heat sink and condense the generated steam except for the amount used in the non-condensing turbine.

10.3.3 DESIGN EVALUATION

These turbine-driven pumps operate reliably as long as there is steam pressure in excess of 50 psig in one of the steam generators. If necessary at this point, with site power available, the steam supply can be switched to the auxiliary boiler steam system. In addition, in an emergency the steam-driven train can operate independent of offsite power and the diesels for up to two hours. The AFW air accumulators provide a sufficient control air source until operators can manually regulate the system. The steam generator's AFW system is initiated by remote manual control or on low level in either steam generator.

Flow control valves are installed in each flow leg in order to allow automatic flow initiation to a value selected by the operator. The valves can be set for automatic operation or placed in remote manual control from the Control Room or auxiliary shutdown panel. Maximum flow to the steam generators from the motor-driven AFW pump powered from the diesel is 300 gpm when feeding both generators. The flow control valves installed in each leg supplied from the motor-driven AFW pump are set at a flow setpoint not to

exceed 150 gpm per leg. If the flow is only being directed to one steam generator, it is acceptable to deliver a maximum of 330 gpm because the flow error associated with the non-used loop is eliminated. These motor-driven AFW pump capacity limits are imposed to prevent exceeding the emergency diesel generator load limit. If diesel generator loading is not a limiting concern, the delivered flow from the motor-driven AFW pump may be increased to the maximum capacity of the motor, which will deliver 575 gpm of feedwater. These upper flow limits do not apply to the steam-driven pumps.

Each AFW discharge flow leg is supplied with two block valves in series. During a Main Steam Line Break, the block valve in the flow legs to the ruptured steam generator will automatically shut. A motor-driven pump discharge header cross connect is available to provide flow, if necessary, to a unit if either No. 13 or 23 pump is inoperable. On/off flow control is provided by a remote manual valve, with the affected unit's flow control valve being used for modulation. High or low level in No. 12 CST is annunciated in the Control Room and level indication for each tank is provided in the Control Room. Indication of flow, discharge pressure, and annunciation of low suction pressure, excess discharge flow, and pipe rupture are also provided in the Control Room.

Auxiliary feedwater flow and response times are conservatively accounted for in the analyses of design basis events. In main steam line break and excess load analyses where AFW flow would increase the consequences of the accidents, the delay time for AFW actuation is minimized and AFW flow is maximized. In feedline break, loss of feedwater, and loss of non-emergency AC power analyses, in which AFW flow would decrease the consequences of the accidents, the delay time for AFW actuation is maximized and AFW flow is minimized. At 10 minutes after an Auxiliary Feedwater Actuation Signal, the operator is assumed to be available to increase or decrease AFW flow to that required by the existing plant conditions.

10.3.4 TESTS AND INSPECTIONS

The AFW System can be tested during normal operation by recirculating the AFW pump flow to the CST.

10.4 AUXILIARY BOILER STEAM SYSTEM

10.4.1 DESIGN BASIS

The Auxiliary Boiler Steam System is shown on Figure 10-6. Two auxiliary boilers are provided, each having the following principal characteristics:

Steam generating capacity:	125,000 lbs/hr
Steam pressure:	180 psig (operating) 250 psig (design)
Steam temperature:	380°F
Feedwater temperature:	180°F (normal) 40°F (at startup)

Each auxiliary boiler was sized to satisfy the plant heating plus the condensate startup deaeration steam requirements. If required, the auxiliary boilers can be used to supply the necessary steam supply for the steam generator auxiliary feed pumps and the SGFPs. Component design data is contained in Table 10-1.

10.4.2 SYSTEM DESCRIPTION

Three fuel oil pumps are provided for the two boilers. Each pump has a capacity of 10,000 lbs/hr at a discharge pressure of 150 psig.

Three feedwater pumps are provided for the auxiliary boilers and take suction from a common deaerator. The design flow of each pump is 265 gpm at 580' total discharge head.

The plant fuel oil system consists of two outdoor bulk storage tanks for No. 2 fuel oil serving the auxiliary boilers, the emergency diesel generators, the Station Blackout Emergency Diesel Generator, and the diesel-driven fire pump. There is an additional fuel oil system for the Societe Alsacienne De Constructions Mecaniques De Mulhouse, safety-related diesel generator, No. 1A. This fuel oil has a viscosity range as specified in American Society for Testing and Materials (ASTM) D975-81, Table 1. Two of the storage tanks are tornado-protected. Truck unloading pumps and tank transfer pumps are also provided.

10.5 TURBINE-GENERATOR AND CONDENSER SYSTEM

10.5.1 DESIGN BASIS

The turbine-generator is designed to receive steam from the steam generators and convert it into electric energy. The condenser transfers unusable heat to the condenser cooling water and deaerates the condensate. The closed regenerative turbine cycle heats the condensate and returns it to the steam generators. Component design data is contained in Table 10-1.

10.5.2 SYSTEM DESCRIPTION

10.5.2.1 Turbine-Generator

The turbines are 1800 rpm tandem compound, six-flow exhaust, indoor units. Saturated steam is supplied to the turbine throttle from the steam generators through four stop valves and four governing control valves. The steam flows through a two-flow, high-pressure turbine and then through combination moisture separator-reheaters (two in parallel for Unit 1, four in parallel for Unit 2) to three double-flow, low-pressure turbines which exhaust to the main condenser system.

Unit 1 is a General Electric turbine and Unit 2 is a Westinghouse turbine. The two units are similar in construction and type, and have similar performance characteristics and generating capacity.

Each turbine is equipped with an automatic stop and emergency trip system which trips the stop and control valves to a closed position for various conditions, including turbine overspeed, low bearing oil pressure, low vacuum, or thrust bearing failure. Upon occurrence of a turbine trip from any of the above causes, and when above a fixed reactor power level, a signal is supplied to the Reactor Protective System to automatically trip the reactor.

Each turbine lubricating oil system supplies oil for lubricating the bearings. A bypass stream of turbine lubricating oil flows continuously through an oil conditioner to remove impurities.

Each generator has the capability to accept the gross rated output of the turbine at rated steam conditions. The generator shafts are oil-sealed to prevent hydrogen leakage. Each generator has its own shaft-driven excitation equipment.

10.5.2.2 Condenser

Each unit has one three-shell, single-pass, deaerating-type condenser with divided water boxes. The condenser is capable of condensing the exhaust steam from the main turbine and the SGFP turbines under full plant load. Each of the three shells is internally equipped to provide for dumping main steam equal to 20% of the Nuclear Steam Supply System thermal capacity. However, only two shells are presently connected to the dump system, thus yielding a heat equivalent absorbing capacity of 40% or 1080 MW(t).

The condenser shells are connected to the turbine exhaust by a belt type, rubber expansion joint. Two low pressure feedwater heaters are installed in the neck of each condenser. The condenser vacuum is maintained by means of mechanical vacuum pumps.

The Condenser Air Removal and Priming System, shown on Figures 10-7 (Unit 1) and 10-12 (Unit 2), removes noncondensable gases from the condenser to the

plant vent. A radiation detector located in each vacuum pump suction line continuously monitors these gases for the presence of radioactivity which would indicate a reactor-coolant-to-secondary-system leak in the steam generators. All noble gases due to a leak will pass by these detectors, which are part of the condenser off-gas radiation monitoring system. The condenser off-gas radiation monitoring system has an alarm setpoint that results in alarm actuation at a primary-to-secondary system leak rate of less than 100 gpd from either Unit.

The monitor alarm setpoint is low enough to ensure detection and monitoring of a sudden and/or rapid increase in reactor-coolant-to-secondary-system leakage. The alarm setpoint provides warning to the plant operators of such an occurrence but has sufficient margin to minimize spurious alarms.

In addition, N-16 activity of the main steam lines upstream of the isolation valves is also monitored and recorded. These radiation monitors will provide annunciation when activity has increased to levels approaching 3.00E-05 µCi/cc. The alarm setpoints provide warning to the plant operators, and the recorded data is available for trending of the activity changes.

Circulating saltwater for condenser cooling for each unit is supplied by six pumps with a capacity of 200,000 gpm each.

The deep intake of cooling water, together with a mechanical condenser tube cleaning system (Section 9.3.2.2), precludes the need for use of chlorine to minimize fouling in the condenser tubes.

FIGURE 10-1 MAIN AND REHEAT STEAM - UNIT 1

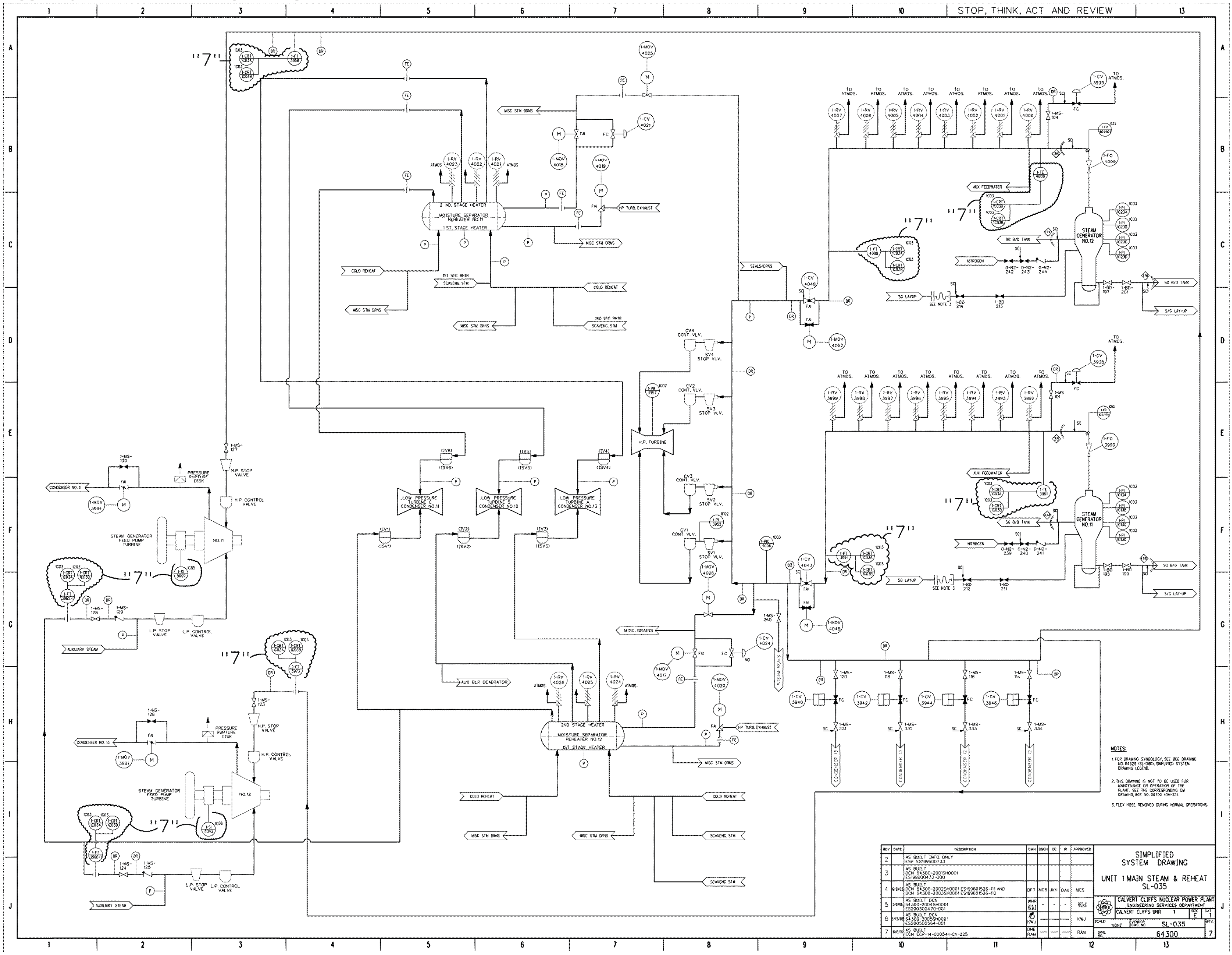
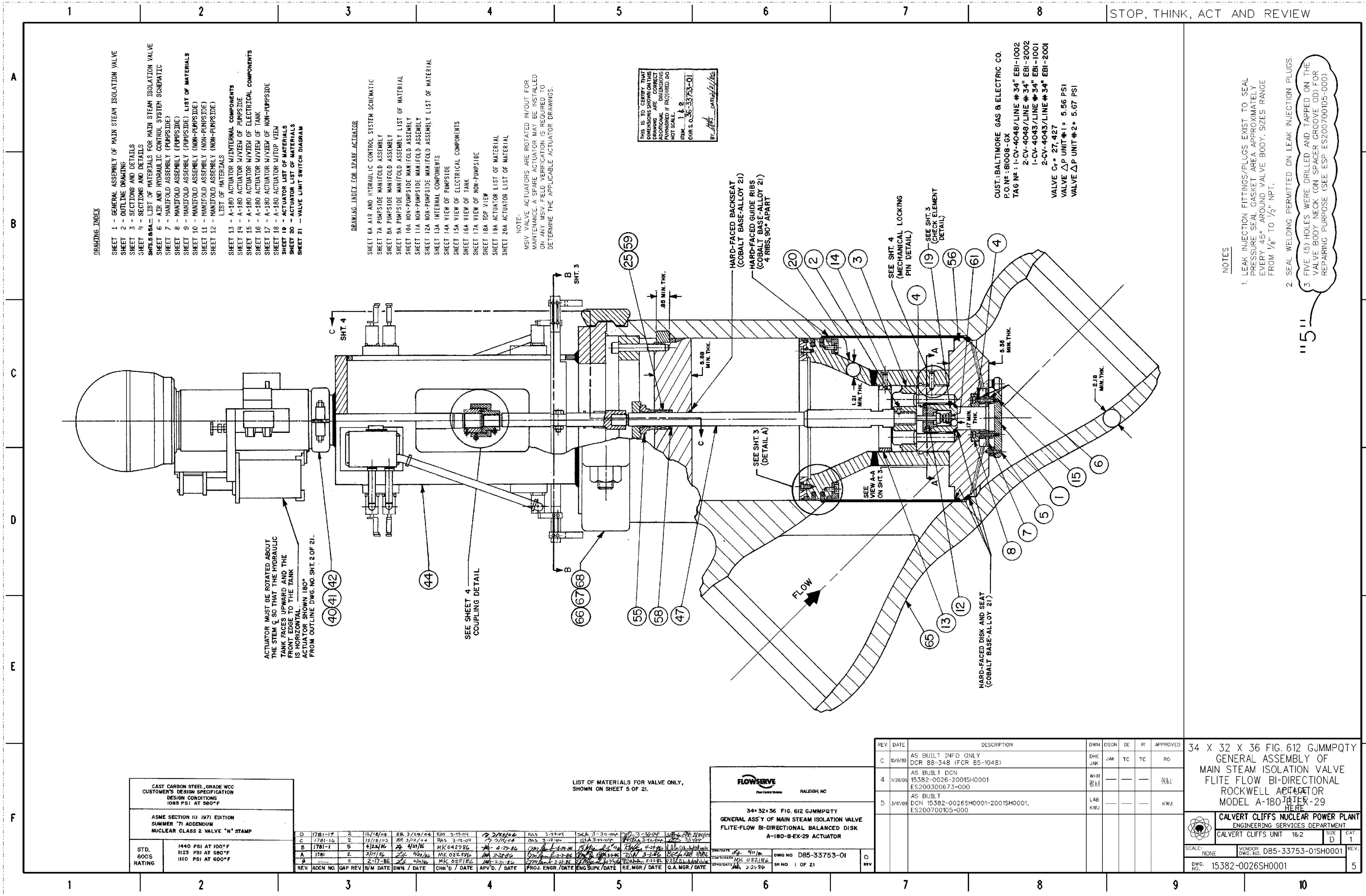
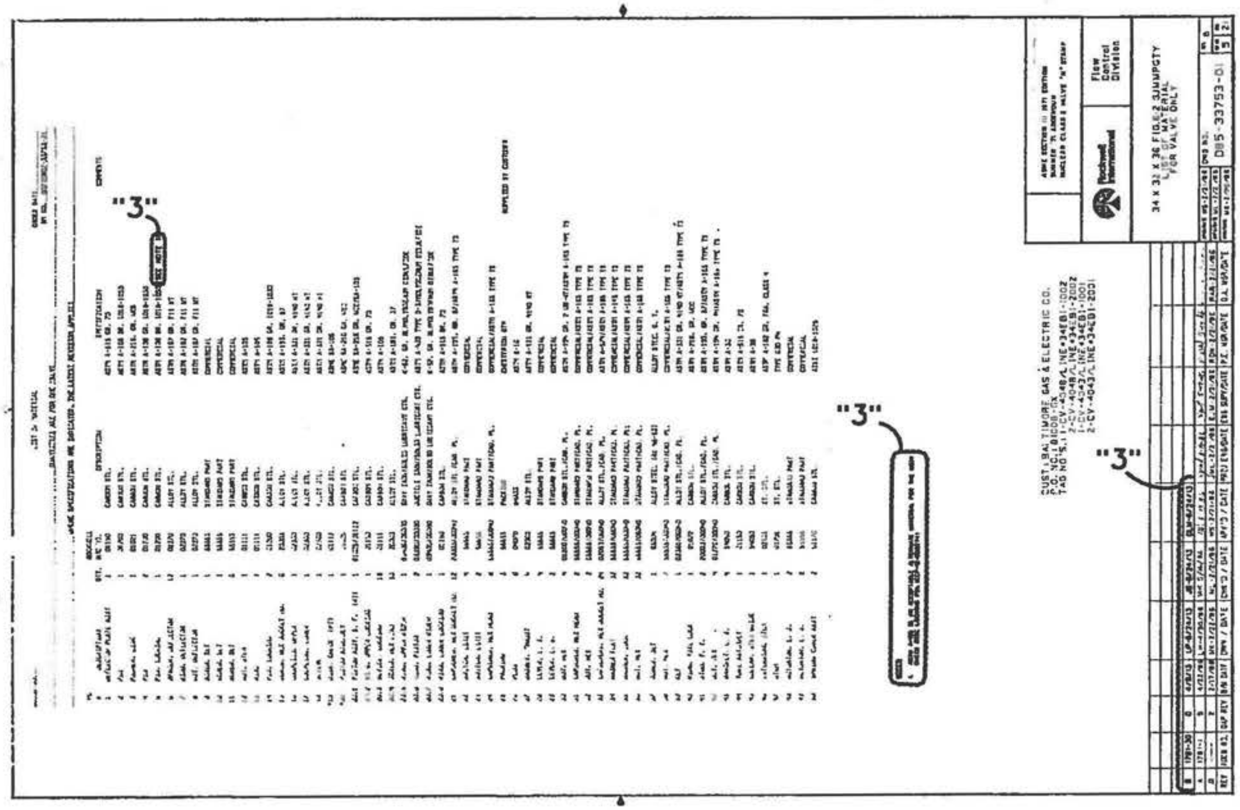


FIGURE 10-2 MAIN STEAM ISOLATION VALVE – FLITE-FLOW BI-DIRECTIONAL BALANCED DISK



STOP, THINK, ACT, AND REVIEW



REV	DATE	DESCRIPTION	DWN	DSGN	DC	IR	APPROVED
3	8/5/14	AS BUILT ECN ECP-13-000274-CN-001	NSB KWJ	—	—	—	KWJ

34 X 32 X 36
FIG.612 GJMMPOTY
LIST OF MATERIAL
FOR VALVE ONLY

CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 1&2 | SIZE B | CAT. 1

SCALE: NONE | VENDOR DWG. NO. D85-33753-01SH0005 | REV.

DWG. NO. 15382-0030 | 3

FIGURE 10-2B MAIN STEAM ISOLATION VALVE LIST OF MATERIAL FOR VALVE ONLY

STOP, THINK, ACT AND REVIEW

LIST OF MATERIALS
QUANTITIES ARE FOR ONE VALVE
WHERE SPECIFICATIONS ARE INDICATED, THE LATEST REVISION APPLIES

ORDER DATE ______
BM NO. 00720997-33753-01

PC NO.	DESCRIPTION	QTY.	ROCKWELL RMC NO.	DESCRIPTION	SPECIFICATION	COMMENTS
51	FLANGE, GLAND	1	24074/06080	ALLOY STL./SOLID LUBRICANT CTG.	4140/MOLYBDENUM DISULFIDE	
52	RING, SPACER	1	01111	CARBON STEEL	ASTM A-105 (SEE NOTE 2)	USE EXISTING PART
*53	BONNET (PT)	1	01112	CARBON STL.	ASME SA-105	
54	STUD, SPECIAL	2	20003/06040	ALLOY STL./CAD. PL.	ASTM A-193, GR B7/ASTM A-165 TYPE TS	"8"
55	GLAND ASSEMBLY	1	02360/06080	ALLOY STL./SOLID LUBRICANT CTG.	ASTM A-331 GR.4140 HT/MOLYBDENUM DISULFIDE	
55.1	GLAND	1	02360	ALLOY STL.	ASTM A-331 GR.4140 HT (SEE NOTE 3)	
55.2	BUSHING, GLAND	2	35091	ALUMINUM BRONZE	ASTM B-148 GR. C95200, OR GR. C95400 (SEE NOTE 4)	
*56	DISK (PT)	1	01112	CARBON STL.	ASME SA-105	
57	GASKET, P.S.	1	05101/06150	CARBON STL./SILVER	AISI 1005-1010/BRIGHT SILVER	
58	RING, JUNK	1	04200	NICKEL BASE ALLOY	WAUKESHA-88	
59	PACKING	2	55555	PACKING	CHESTERTON STYLE 1	SUPPLIED BY CUSTOMER
60	COVER, SPRING GUIDE	2	01200	CARBON STL.	ASTM A-108 GR. 1018-1030	
61	SPRING, CONICAL	1	04980	INCONEL X750	AMS 5699	
62	WASHER, BELLEVILLE	72	01430	HIGH CARBON STEEL	C1045740&T	
63	RETAINER, BONNET	1	01150	MILD CARBON STEEL	ASTM A-515, GR. 70	USE EXISTING PART
*64	RETAINER, GASKET	1	02841	STAINLESS STEEL	ASTM A-461, GR. 660 (SEE NOTE 1)	USE EXISTING PART
*65	BODY ASSEMBLY	1	01023	CAST CARBON STEEL	ASTM A-216, GR. WCC	USE EXISTING PART
66	RING, YOKE LOCK	1	01021	CAST CARBON STEEL	ASTM A-216, GR. WCB	USE EXISTING PART
67	NUT, HEX.	4	01290	ALLOY STEEL	ASTM A-194, GR. 7	USE EXISTING PART
68	STUD F.T.	2	02862	ALLOY STEEL	ASTM A-540, GR. B23, CL.4	USE EXISTING PART
69	CONNECTOR, 45°	2	55555	STANDARD PART	COMMERCIAL	
70	CONNECTOR 90°	6	55555	STANDARD PART	COMMERCIAL	
71	CONN./CABLE ASSY. 10" LG.	6	55555	STANDARD PART	COMMERCIAL	
72	ADAPATOR & LOCK RING ASSY.	6	55555	STANDARD PART	COMMERCIAL	
73	BOX, JUNCTION	2	55555	STANDARD PART	COMMERCIAL	
74	HUB, MEYERS	6	55555	STANDARD PART	COMMERCIAL	
75	HUB, MEYERS	2	55555	STANDARD PART	COMMERCIAL	
76	CONNECTOR , WIRE	54	55555	STANDARD PART	COMMERCIAL	
77	TERMINAL BLOCK	6	55555	STANDARD PART	COMMERCIAL	
78	TERMINAL MARKER, VINAL	1	55555	STANDARD PART	COMMERCIAL	
79	PANEL, JUNCTION BOX	2	55555	STANDARD PART	COMMERCIAL	
80	TUBING, METAL FLEXIBLE	8' LG.	55555	STANDARD PART	COMMERCIAL	
81	CONNECTOR, STRAIGHT	4	55555	STANDARD PART	COMMERCIAL	
82	SCREW, RD. HD. MACHINE	8	55555/06040	STANDARD PART/CAD. PL.	COMMERCIAL/ASTM A-165 TYPE TS	
83	SCREW, RD. HD. MACHINE	20	55555/06040	STANDARD PART/CAD. PL.	COMMERCIAL/ASTM A-165 TYPE TS	

ASME SECTON III 1971 EDITION
SUMMER 71 ADDENDUM
NUCLEAR CLASS 2 VALVE "N" STAMP

NONDESTRUCTIVE EXAMINATION CODES: (PT) - LIQUID PENETRANT TEST
(RT) - RADIOGRAPH TEST

*PRESSURE RETAINING PARTS

NOTES:

1. REPLACEMENT MATERIAL TO BE SA638 GR.660 TYPE 2. (REF. ES199701757-000 REV.0)
2. REPLACEMENT MATERIAL TO BE ASTM A-668, GR.4140 CL.L, ALLOY STEEL/CAD.PL., RMC NO.02353/06040
3. ASTM A-322 GR. 4140 HT IS AN ACCEPTABLE ALTERNATE MATERIAL (ECP-13-000568).
4. ASTM B505 GR. C95400 IS AN ACCEPTABLE ALTERNATE MATERIAL (ECP-13-000568).

"8"

CUST.: BALTIMORE GAS & ELECTRIC CO.
P.O. NO.: 81008-GX
TAG NO'S. 1-CV-4048/LINE NO. 34EBI-1002
2-CV-4048/LINE NO. 34EBI-2002
1-CV-4043/LINE NO. 34EBI-1001
2-CV-4043/LINE NO. 34EBI-2001

THIS DRAWING WAS REPRODUCED FROM ROCKWELL INTERNATIONAL MEASUREMENT & FLOW CONTROL DIVISION DWG. NO. D85-33753-01, REV. C, DATED 10/06/88, BGE F.P. DOC. NO. 15382-0031SH0005A.

REV	DATE	DESCRIPTION	DWN	DSGN	DC	IR	APPROVED
6		AS BUILT, INFO ONLY ESP ES199800048-000					
7	[illegible]	AS BUILT, INFO ONLY ESP ES199800897-000	JA	—	—	—	JMC
8	8/5/14	AS BUILT ECN ECP-13-000568-CN-001	NSB KWJ	—	—	—	KWJ

34 X 32 X 36
FIG. 612GJMMPQTY
LIST OF MATERIAL
FOR VALVE ONLY

CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 2 SIZE C CAT. 1
SCALE: NONE VENDOR DWG. NO. D85-33753-01 REV.
DWG. NO. 15382-0031SH0005A 8

FIGURE 10-3 MAIN STEAM ISOLATION VALVE HYDRAULIC ACTUATION SYSTEM

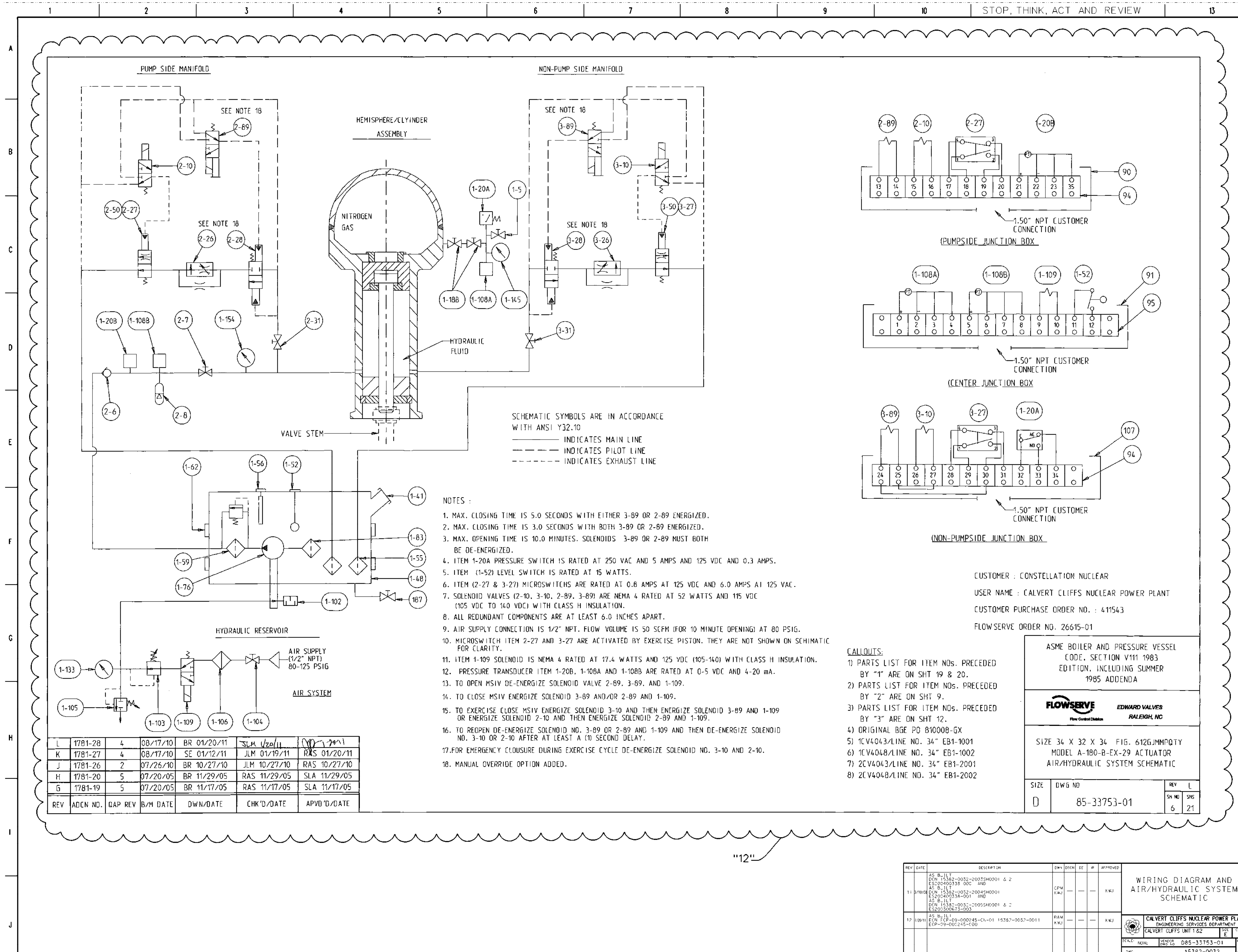


FIGURE 10-4 CONDENSATE AND FEEDWATER – UNIT 1

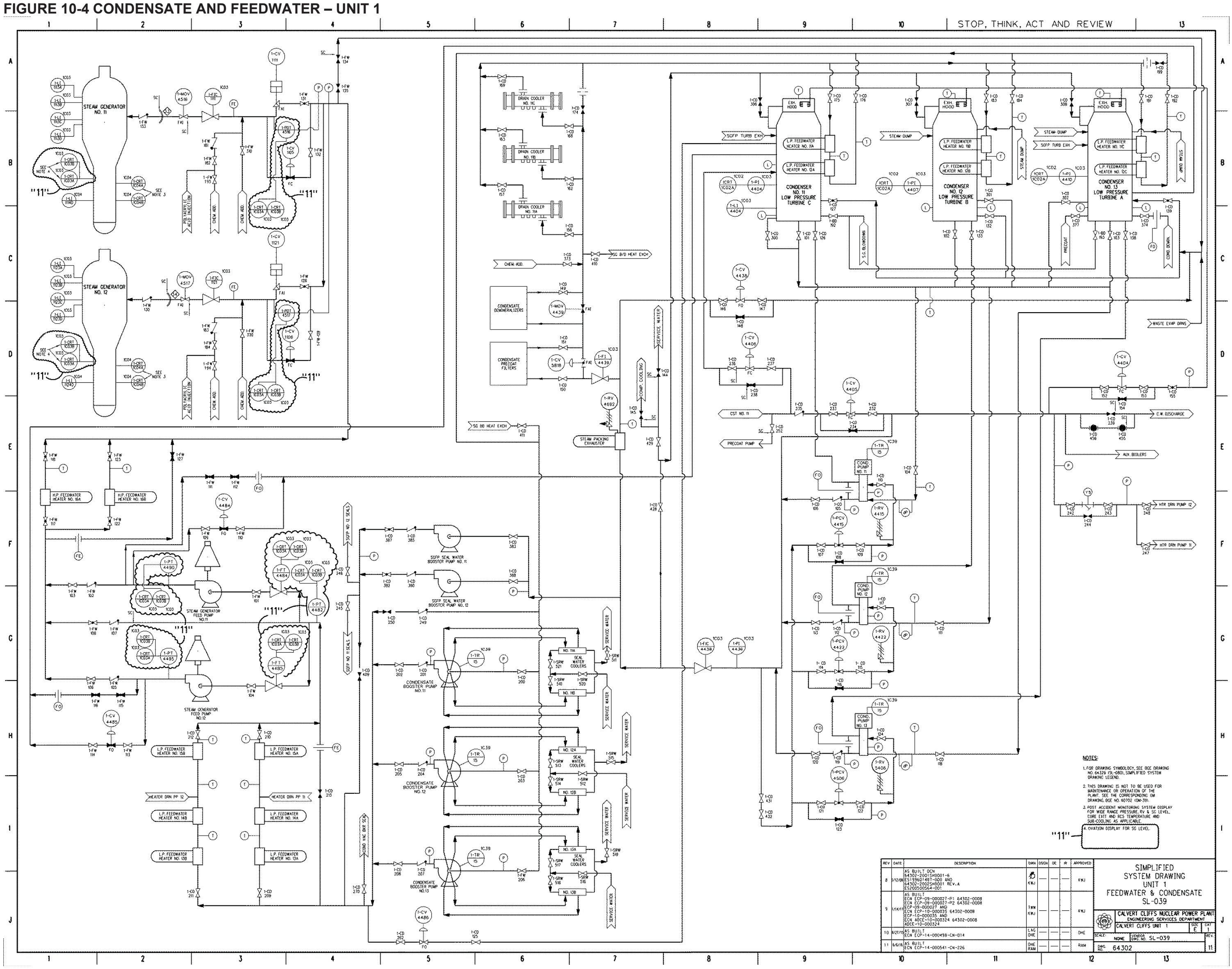
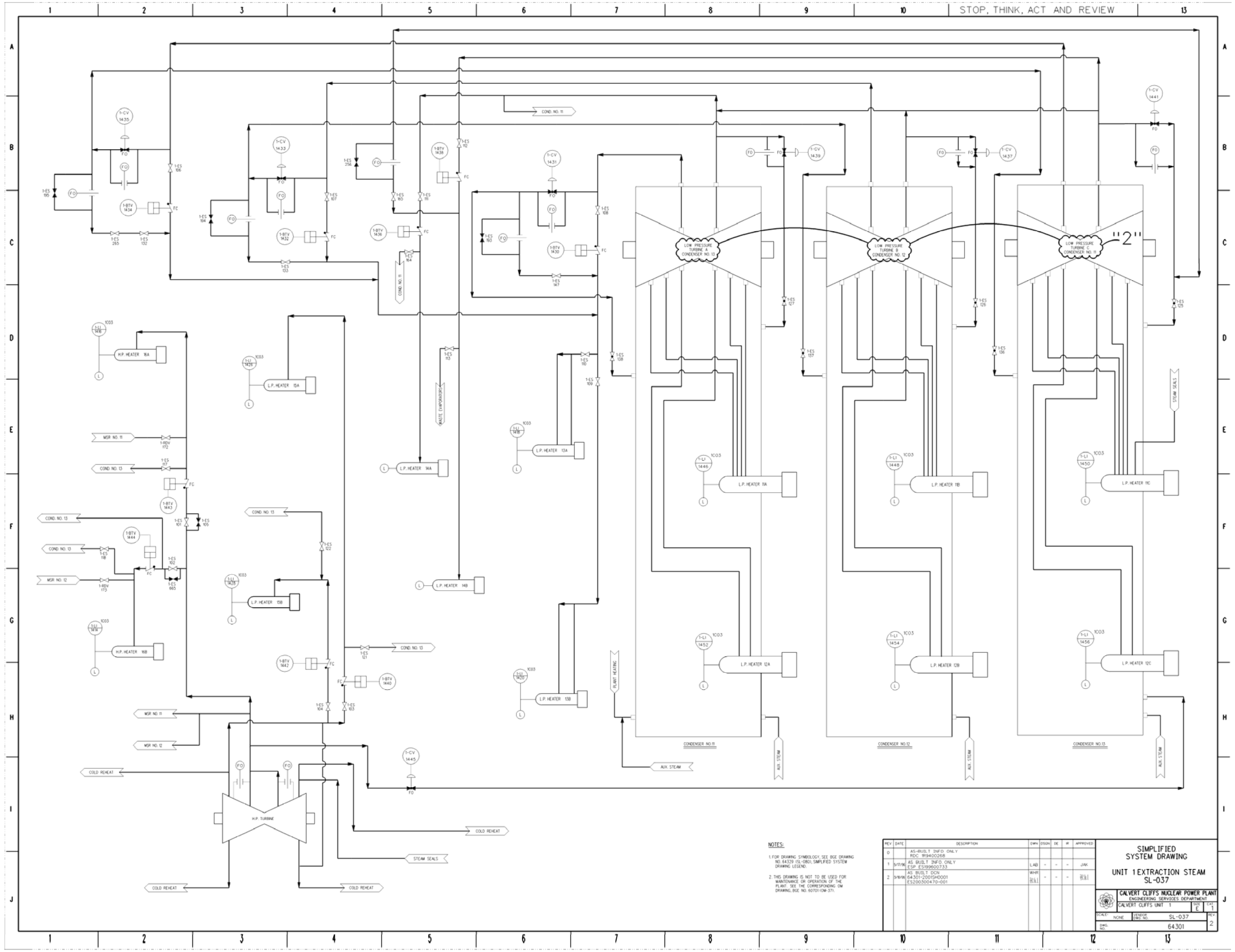
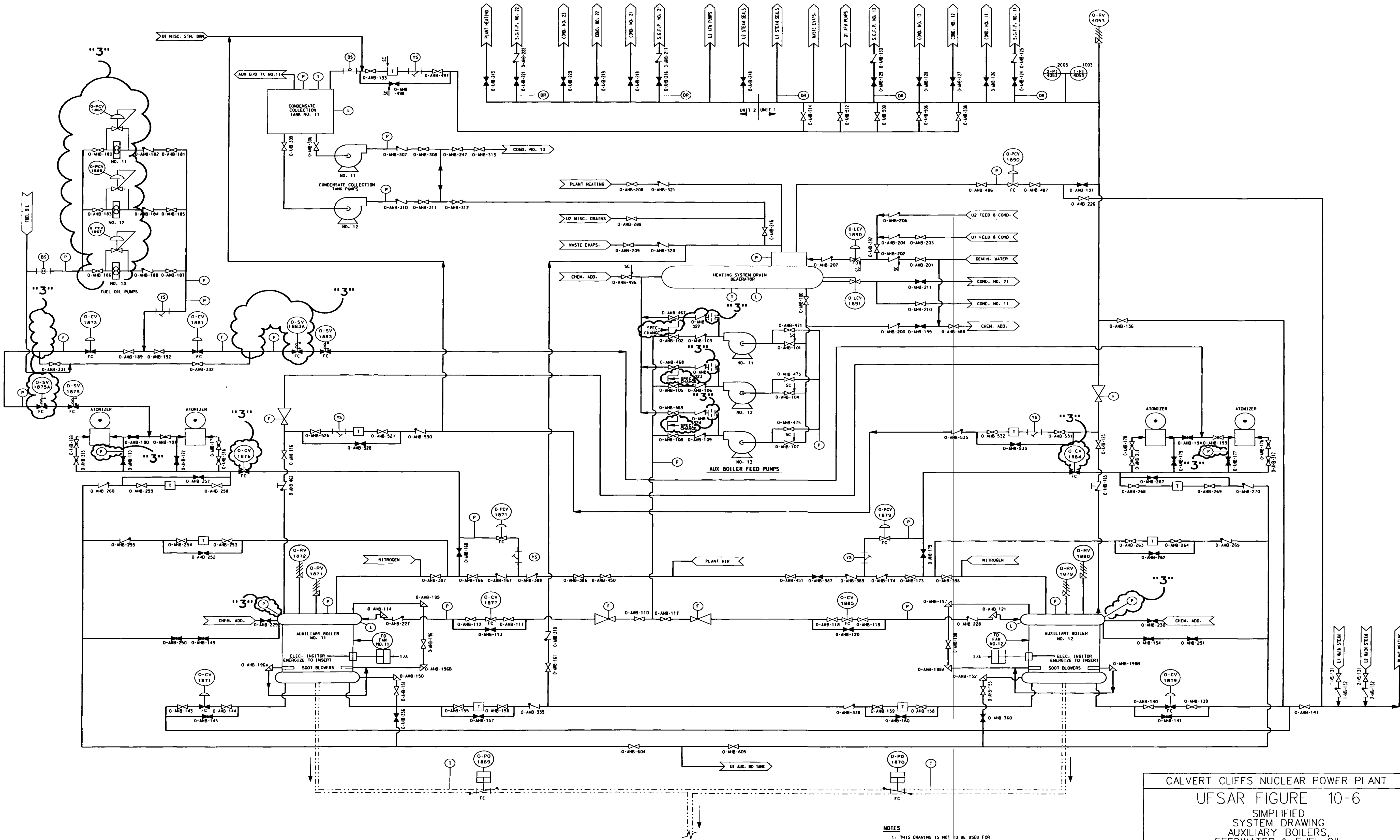


FIGURE 10-5 EXTRACTION STEAM - UNIT 1





NOTES

1. THIS DRAWING IS NOT TO BE USED FOR MAINTENANCE OR OPERATION OF THE PLANT. SEE THE CORRESPONDING OM DRAWING, BGE NO. 60707 (OM-48).
2. FOR DRAWING SYMBOLOGY, SEE BGE DRAWING NO. 64329 (SL-080), SIMPLIFIED SYSTEM DRAWING LEGEND.

CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 10-6

SIMPLIFIED SYSTEM DRAWING AUXILIARY BOILERS, FEEDWATER & FUEL OIL SL-048

BGE DRAWING 64328 REV. 3

REVISION 26

FIGURE 10-7 CONDENSER AIR REMOVAL – UNIT 1

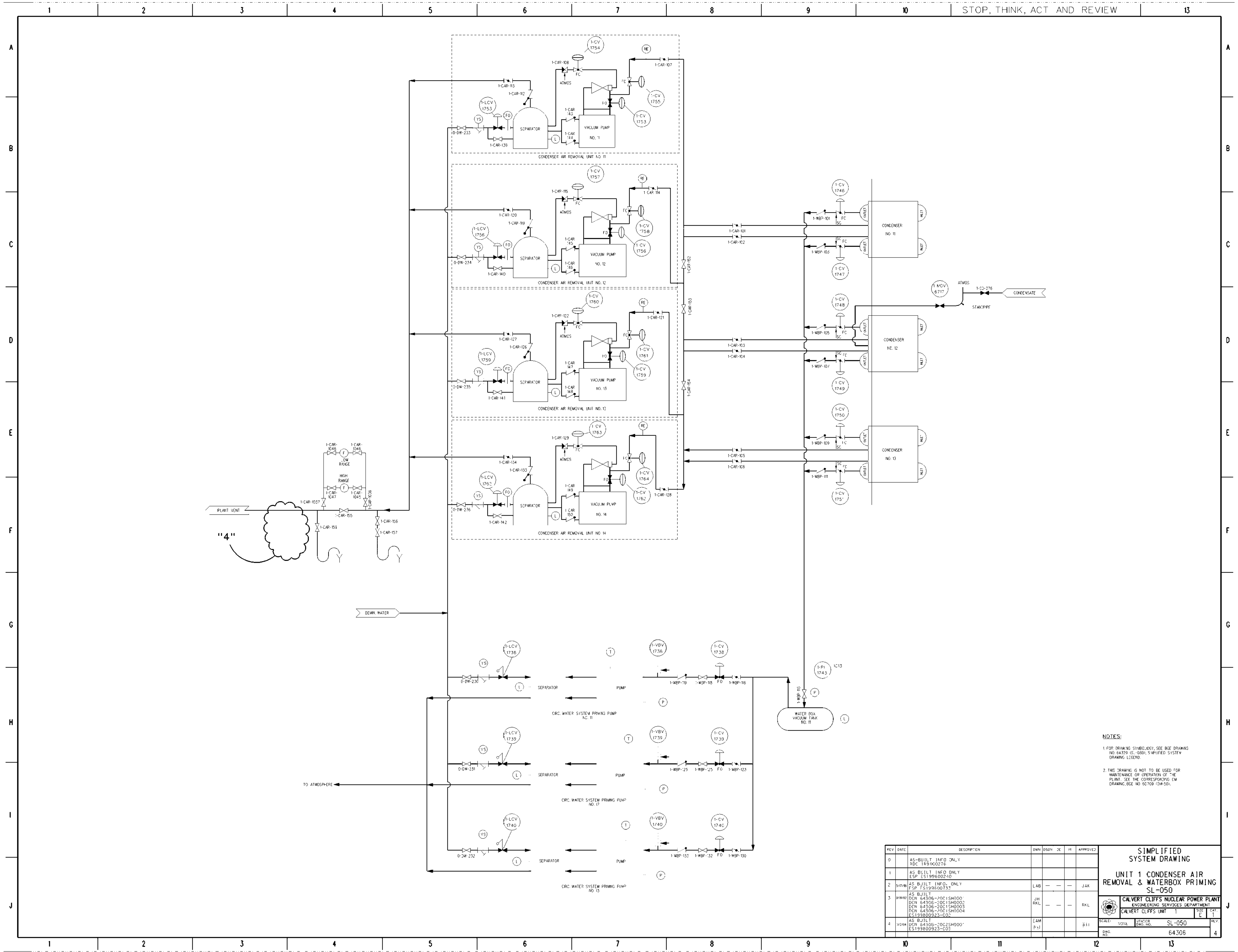
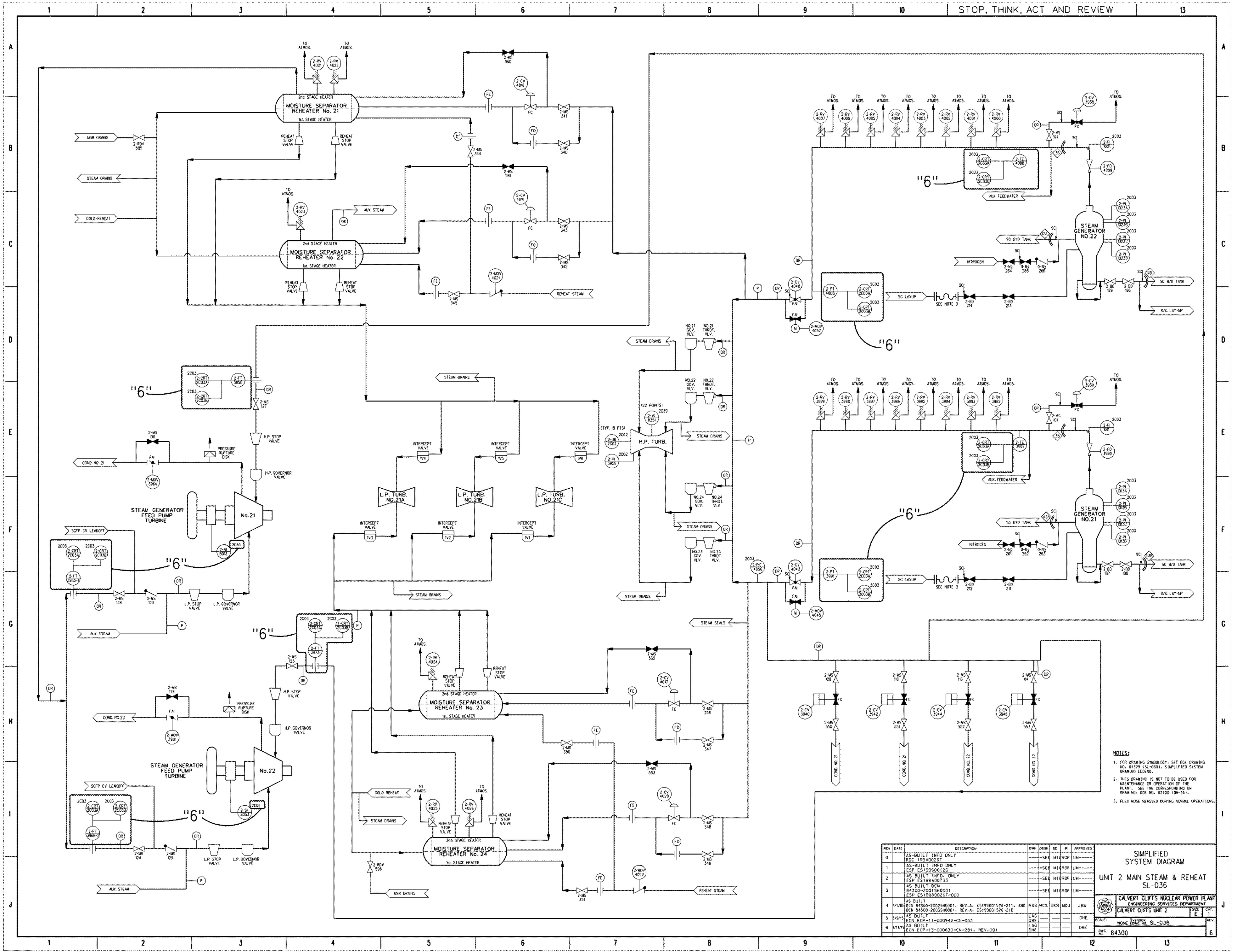
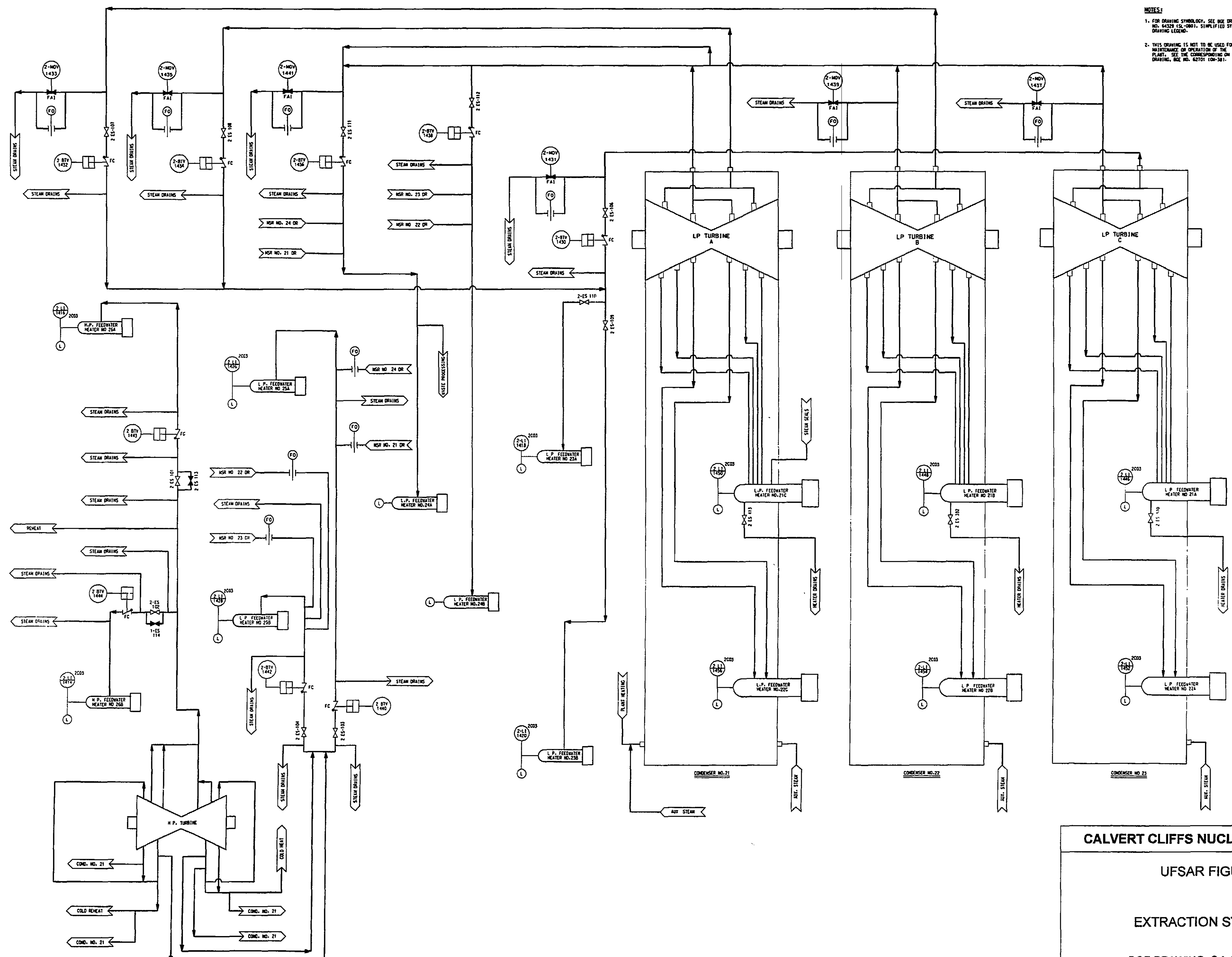


FIGURE 10-9 MAIN AND REHEAT STEAM - UNIT 2





CALVERT CLIFFS NUCLEAR POWER PLANT

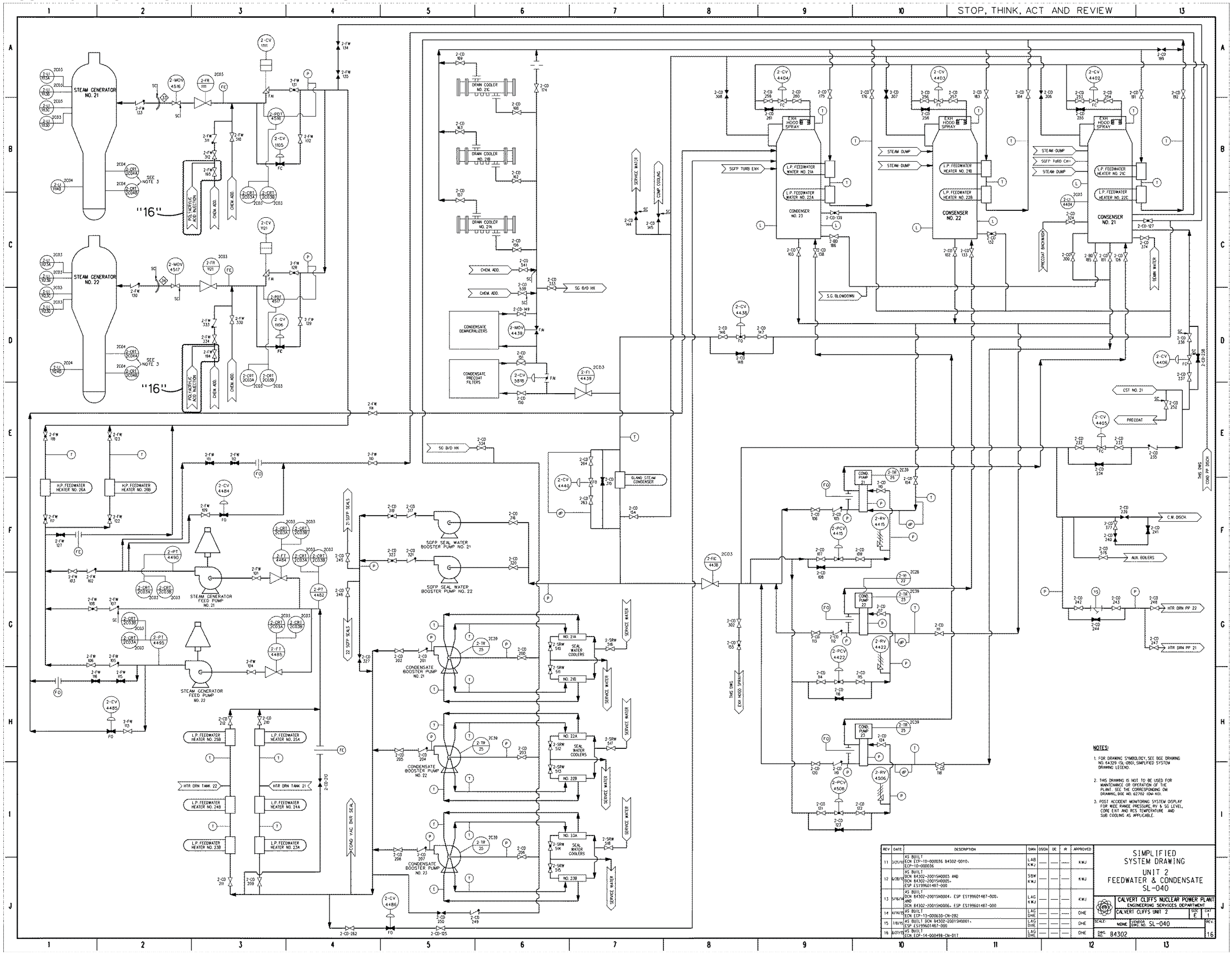
UFSAR FIGURE 10-10

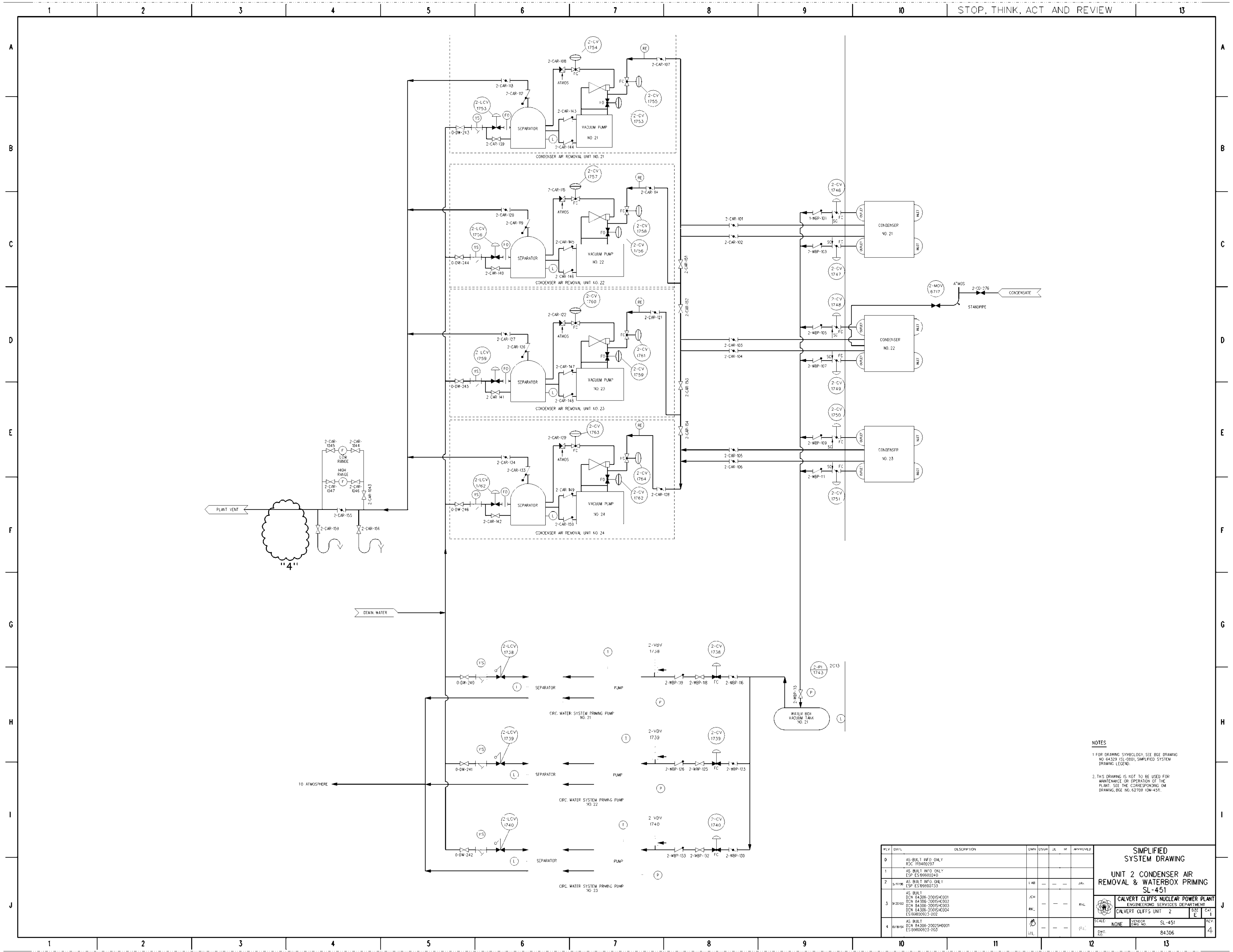
EXTRACTION STEAM - UNIT 2

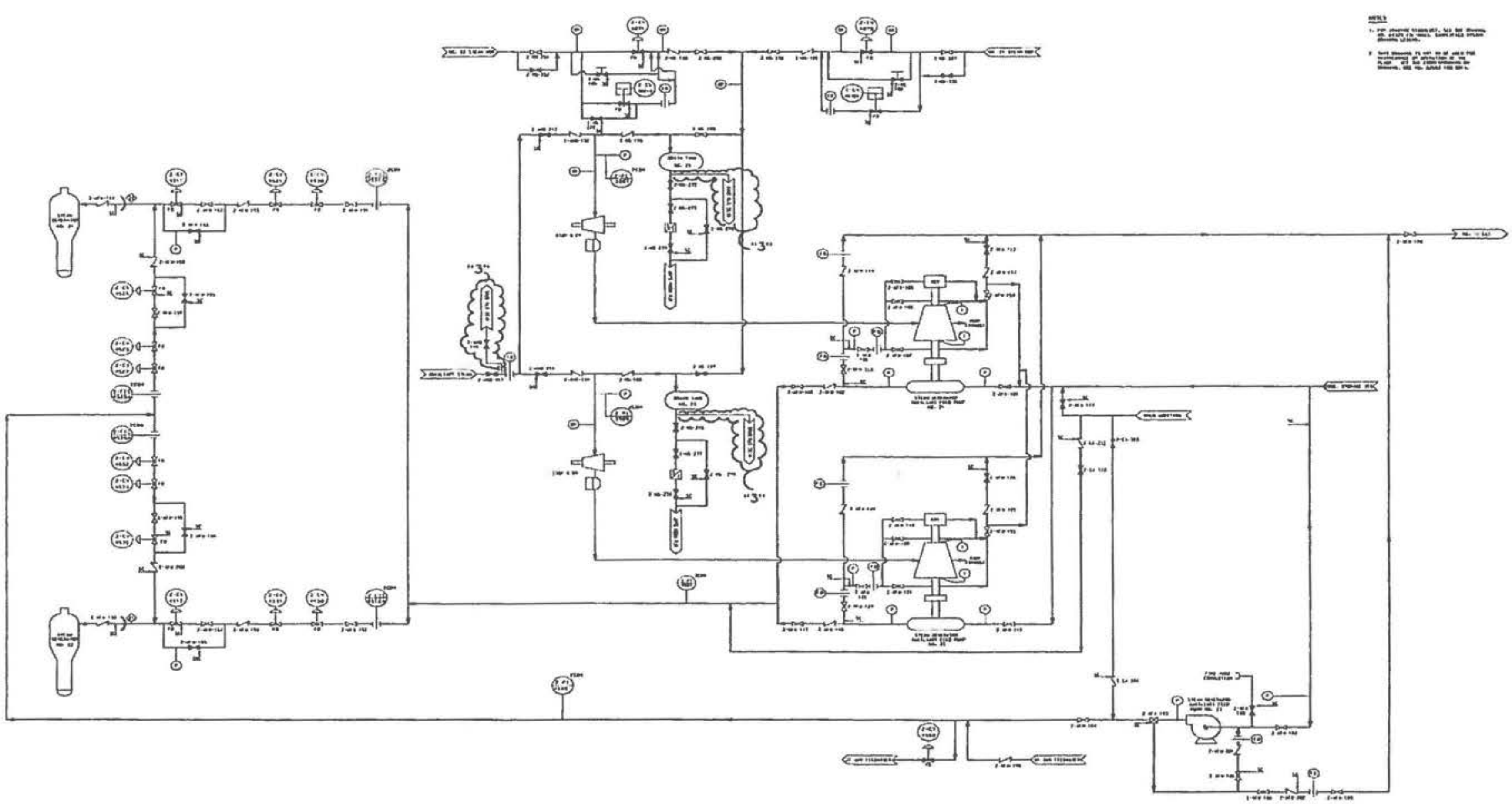
BGE DRAWING 84-301, REV 1

Revision 21

FIGURE 10-11 CONDENSATE AND FEEDWATER - UNIT 2




STOP, THINK, ACT AND REVIEW
SIMPLIFIED SYSTEM DRAWING
UNIT 2 CONDENSER AIR REMOVAL & WATERBOX PRIMING
SL-451
CALVERT CLIFFS NUCLEAR POWER PLANT
CONDENSER AIR REMOVAL UNIT NO. 21
CONDENSER AIR REMOVAL UNIT NO. 22
CONDENSER AIR REMOVAL UNIT NO. 23
CONDENSER AIR REMOVAL UNIT NO. 24
CIRC. WATER SYSTEM PRIMING PUMP NO. 21
CIRC. WATER SYSTEM PRIMING PUMP NO. 22
CIRC. WATER SYSTEM PRIMING PUMP NO. 23
PLANT VENT
DEMIN. WATER
TO ATMOSPHERE
CONDENSATE
NOTES



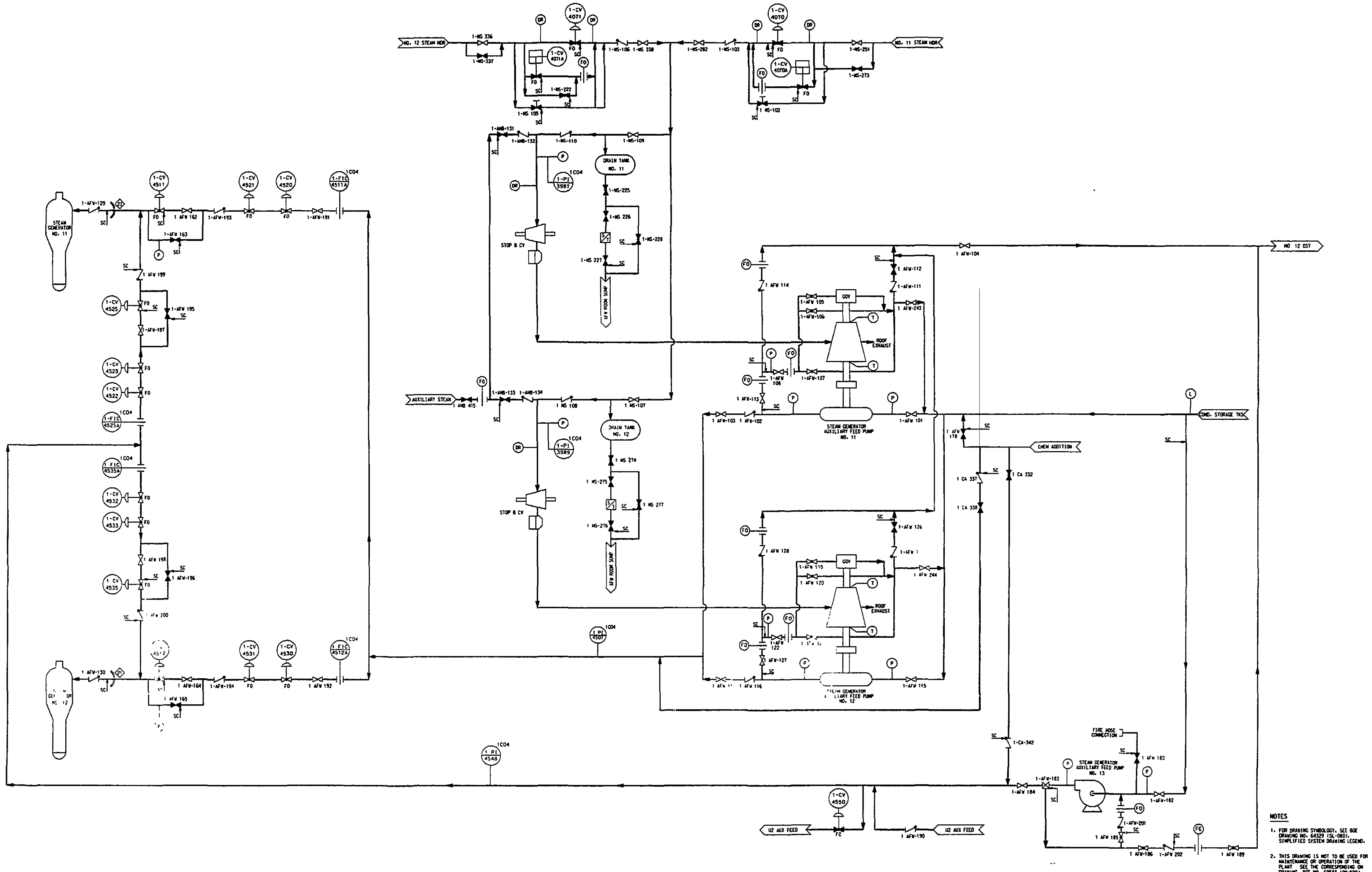
CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 10-13

SIMPLIFIED SYSTEM DRAWING UNIT 2 AUX FEEDWATER SL-801

BGE DRAWING 84312 REV. 3

REVISION 26



CALVERT CLIFFS NUCLEAR POWER PLANT

UFSAR FIGURE 10-14

AUXILIARY FEEDWATER - UNIT 1

BGE DRAWING 64-312, REV 1

Revision 21

APPENDIX 10A
HIGH ENERGY PIPE RUPTURES OUTSIDE CONTAINMENT

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LIST OF ACRONYMS

ACI	American Concrete Institute
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AFWP	Auxiliary Feedwater Pump
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
CCW	Component Cooling Water
CV	Control Valve
CVCS	Chemical and Volume Control System
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
fL/D	Flow Resistance Coefficient
FCR	Facility Change Request
HPSI	High Pressure Safety Injection
HX	Heat Exchanger (used in Tables only)
ISI	Inservice Inspection
LPSI	Low Pressure Safety Injection
MFIV	Main Feedwater Isolation Valve
MFW	Main Feedwater
MOV	Motor-Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
SDC	Shutdown Cooling
SG	Steam Generator (used in Tables only)
SGFP	Steam Generator Feed Pump
SRW	Service Water
SSC	System, Structure and/or Components
SSE	Safe Shutdown Earthquake
UFSAR	Updated Final Safety Analysis Report

APPENDIX 10A

10A HIGH ENERGY PIPE RUPTURES OUTSIDE CONTAINMENT

10A.0 INTRODUCTION

In 1972, the Nuclear Regulatory Commission (NRC) determined that additional information was needed to evaluate our compliance with General Design Criteria 4 regarding pipe ruptures outside containment. They issued a letter to Calvert Cliffs (Reference 1) describing the types of analyses and information that were needed to ensure compliance with the General Design Criteria. This Appendix responds to that request for information.

The NRC listed 21 items that needed to be evaluated by the licensee. Each section of Appendix 10A is, therefore, broken down into 21 corresponding sections. Below are listed the basic requirements for each item. A complete copy of the NRC request can be found on the Electronic Docket or in the Nuclear Regulatory Matters Unit. The text of the letter is also contained in Appendix B of Branch Technical Position SPLB 3-1 (Standard Review Plan 3.6.1).

1. Identify the systems which must be restrained for pipe whip. Systems do not need to be restrained if any of the following conditions exist: the service temperature is less than 200°F and the design pressure is less than or equal to 275 psig; the piping is physically separated from systems, structures and components (SSCs) important to safety; following a break, the whipping pipe cannot impact any SSC important to safety; or the internal energy of the whipping pipe is low enough that it does not impair the safety function of any SSC.
2. Design basis break locations must be determined for ASME Code Class 1, 2 and 3 piping. In general, break locations are at the terminal ends and some intermediate locations. In addition, critical cracks are postulated to determine the effects of jet impingement on SSCs important to safety. The critical crack size is one-half the pipe diameter in length, and one-half the wall thickness in width.
3. The orientation of the pipe break is as follows: longitudinal breaks in pipe 4" and larger, and/or circumferential breaks exceeding 1" in size.
4. A summary should be provided of the dynamic analyses which determines the loadings on Category I piping as the result of a postulated pipe break.
5. A description should be provided of the measures to protect against pipe whip, blowdown jet and reactive forces. This description could include pipe restraints, physical separation and other measures.
6. Describe the procedures that are used to evaluate the structural adequacy of Category I structures. Those procedures should include the method of evaluating stresses, the allowable design stresses, the load factors and load combinations.
7. The structural design loads, i.e., pressure, temperature, dead, live, equipment, static, dynamic and dynamic loads should be described.
8. Seismic Category I structural elements, such as floors, interior walls, exterior walls, building penetrations and the whole building should be analyzed for load reversal.
9. If new openings are made in existing structures, the load bearing capability of the modified structure should be demonstrated.
10. Verify that structural failure will not cause failure of any other structure in a manner that adversely affects the ability to mitigate the consequence of the accident, and the capability to bring the unit to cold shutdown.
11. Verify that the pipe rupture will not result in the loss of required redundancy in any portion of the protection system, Class 1E electrical system, engineered safety feature (ESF) equipment, cable penetrations or their interconnecting cables that are required to mitigate the consequences of that accident and place the unit in cold shutdown. Verify that the pipe rupture will not result in environmentally-induced failures which do not

result in a protective action, but does disable a protective function. In this case, a loss of redundancy is permitted, a loss of function is not.

12. Assurance that the Control Room or the alternate shutdown panel will be habitable and its equipment functional after a feedwater line or steam line break.
13. Environmental qualification should be demonstrated by test for electrical equipment required to function following a high energy line break. The information should include the following: identification of all electrical equipment necessary to meet the requirements of Item 11, above (the time after the high energy line break in which they are required to operate should be given); the test conditions and the results of the test data for environmental qualification; the results of a jet impingement study; an evaluation of the safety-related equipment in the Control Room; and an evaluation that the onsite power distribution system will remain operable.
14. Provide design drawings of the steam and feedwater lines which show the elevations of the pipe, the safety-related equipment, including ventilation equipment, intakes and ducts.
15. Discuss the potential for flooding due to a high energy line break.
16. Describe the quality control programs and inspection programs that are used for piping systems outside containment.
17. If leak detection equipment is used, a discussion of its capabilities should be provided.
18. Describe the emergency procedures that would be followed after a pipe rupture, including automatic and manual operations required to place the reactor unit in cold shutdown. The estimated times following the accident for all equipment and personnel actions should be included in the procedure summary.
19. Describe the seismic and quality classification of the high energy piping systems (including steam and feedwater piping) which run near SSCs important to safety.
20. Describe the assumptions, method and results of analyses used to calculate pressure and temperature transients in compartments, pipe tunnels and buildings following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified. The capability of systems required to function to meet a single active failure should be described.
21. Describe the methods or analyses performed to demonstrate that there will be no adverse effects on the primary Containment Structure due to a pipe rupture outside the containment.

In accordance with Atomic Energy Commission (AEC) criteria, the following systems with changes as summarized below, are considered high energy[1] systems and each is discussed in one of the following sections.

10A.1 **<u>MAIN STEAM (MS)</u>:** Encapsulated and restrained at terminal end and high stress points within the Auxiliary Building. Encapsulation along with an added vent stack, limits blowdown from a postulated pipe break to below compartment allowable pressurization. Added walls with watertight doors limit pressurization and flooding to the MS Valve Room and the connecting pipe tunnel to the turbine area. This item was reanalyzed in November 1981, to determine qualification requirements to meet IE Bulletin 79-01B, "Environmental Qualification of Electrical Equipment." This analysis supplements and expands the original analysis. It is discussed where applicable in the above sections.

10A.2 **<u>MAIN STEAM TO AUXILIARY STEAM GENERATOR FEED PUMP TURBINE</u>:** Sleeved and restrained at terminal ends and high stress points. Changed CV-4070 and CV-4071 from open to closed positions and added two gate valves and one globe

[1] Pressure greater than 275 psig and/or temperature greater than 200°F.

bypass valve. This item was modified under Facility Change Request (FCR) 89-26 during the addition of 1/2-CV-4070A and 1/2-CV-4071A.

Changed 300 psi gate valves to 600 psi gate valves to protect Auxiliary Steam System from MS pressure. Added 300 psi gate valves outside the Auxiliary Feedwater Pump (AFWP) Room at the auxiliary steam supply header to exempt the auxiliary steam supply line inside the AFWP room from consideration of a pipe rupture.

10A.3 **STEAM GENERATOR BLOWDOWN:** The small pipe used in this system together with the choking behavior of the saturated fluid indicate that a blowdown line rupture is not of sufficient magnitude to adversely affect secondary side inventory. Further, there is no safety-related piping or equipment close enough to the steam generator blowdown piping that can be damaged by pipe whip or fluid impingement if these lines were to rupture. This item was reanalyzed in November 1981 to determine qualification requirements to meet IE Bulletin 79-01B, "Environmental Qualification of Electrical Equipment." This analysis supplements and expands the original analysis. It is discussed where applicable in the above sections. (See Section 10A.3 details for a description of blowdown effects on Unit 2 main steam drains 5 and 6 in the Turbine Building.)

10A.4.1 **MAIN FEEDWATER (MFW) - INSIDE AUXILIARY BUILDING:** Complete sleeving of lines in the Auxiliary Building outside the MS valve room to direct all blowdown to the valve room or Turbine Building. In the valve room, lines are sleeved and restrained at points of high stress. The feedwater check valve is relocated to inside Containment to prevent blowdown from the steam generator.

10A.4.2 **MAIN FEEDWATER AND HEATER DRAIN SYSTEM - INSIDE TURBINE BUILDING:** Pipe whipping restraints and jet impingement barriers are added, as required, to protect the AFWP Room and the common wall between the Turbine and Auxiliary Buildings. (See Section 10A.4.2 details for a description of feedwater effects on main steam drains 5 and 6 in the Turbine Building.)

10A.5 **AUXILIARY FEEDWATER (AFW):** A check valve has been added inside Containment to eliminate the line outside of Containment as a high energy system.

10A.6 **SHUTDOWN COOLING (SDC):** Based on level of quality control, periodic inservice inspection (ISI), low usage factor, the short time the system has high energy and the strict administrative control when system is in use, a break in this system is not considered credible.

10A.7 **CHEMICAL AND VOLUME CONTROL (CVCS):** Encapsulated and restrained at high stress points with an added excess flow check valve inside Containment to terminate flow after a postulated pipe break inside the Auxiliary Building.

10A.8 **SAMPLING:** Line sizes are less than 1", therefore only a pipe crack is postulated. Shielding is added as required to protect ESFs from postulated jet impingement forces.

10A.9 **AUXILIARY STEAM:** Lines operate at low pressure, therefore only a pipe crack is postulated. Shielding is added as required to protect ESFs from postulated jet impingement forces.

10A.0.1 REFERENCES

1. Letter from A. Giambusso (NRC) to J. W. Gore (BGE), dated December 15, 1972, Request for Additional Information Concerning the Consequences of Postulated Pipe Failures Outside the Containment Structure

10A.1 MAIN STEAM

Each steam generator is connected with a single 34" pipe line to the steam header near the turbine. Each steam line will carry approximately 5.6 million pounds of steam per hour during rated power operation. These lines penetrate the Containment at Elevation 38'0" and pass through the Auxiliary Building to the Turbine Building. The MS System, shown in Figure 10A.1-1, will vary normally between 900 and 850 psia for no-load and full-load operation, respectively. A flow-limiting nozzle, located in the Containment, will protect the primary system against an excessive cooldown rate in the event of a main steam line break (MSLB). Pressure in the MS system is maintained primarily by the reactor coolant temperature. A turbine by-pass system, with a capacity of 40% of the rated steam flow, and an atmospheric dump system, with a capacity of 5% of the rated steam flow, provide additional control of the MS pressure during load changes. In addition, 16 relief valves protect the MS system from abnormal pressure above 1050 psia.

10A.1.1 PIPE WHIP

The MS system normally operates at a pressure above 275 psig and 200°F and, therefore, protection is provided for pipe whip following a longitudinal or circumferential break.

10A.1.2 CRITERIA FOR PIPE BREAK LOCATION

Pipe breaks are postulated to occur at the following locations:

a. Terminal ends;

b. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically-calculated basis under the loadings associated with seismic events and operational plant conditions exceed $0.8\ (S_h + S_A)^*$ or the expansion stresses exceed $0.8\ S_A$; and,

c. Two additional intermediate locations are selected on the following reasonable bases:

   1. The points of highest stress, Figure 10A.1-2 identifies the location of the high stress points. Table 10A-1 lists the stress values for these points; and,
   2. No break in short-run pipes up to five pipe diameters.

d. A critical crack defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width is postulated to occur at any location.

10A.1.3 CRITERIA FOR PIPE BREAK ORIENTATION

A longitudinal pipe break is considered for lines 4" and larger. The break is assumed to be parallel to the pipe axis and oriented at any point around the pipe circumference. A circumferential break is considered for lines exceeding a nominal pipe size of 1". The break is assumed to be oriented perpendicular to the pipe axis. A critical crack is assumed to be oriented at any point around the pipe circumference.

10A.1.4 SUMMARY OF PIPE WHIP DYNAMIC ANALYSIS

10A.1.4.1 Location of Number of Breaks

The locations and number of design basis breaks are chosen in accordance with the criteria in Section 10A.1.2. Two types of breaks, longitudinal break and circumferential break, are considered in accordance with the criteria in Section 10A.1.3.

* Pressure greater than 275 psig and/or temperature greater than 200°F.

The critical crack is considered to occur anywhere on the line. Figure 10A.1-2 shows the postulated pipe break locations for the MS line.

10A.1.4.2 The Postulated Rupture Orientation

The longitudinal break is parallel to the pipe axis and oriented at any point around the pipe circumference. The longitudinal break area is equal to the effective cross-sectional flow area upstream of the break location. The circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from a circumferential break are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

The critical crack is oriented at any point around the pipe circumference.

10A.1.4.3 Description of Forcing Function

Design parameters to estimate steam-water blowdown thrust and jet impingement forces expressed in term of F_T/P_OA_B as a function of the friction parameter. Flow resistance coefficient (fL/D), and upstream area restriction parameter, A_B/A_R, are respectively presented in Figures 10A.1-9 and 10A.1-10. In addition, graphical solutions to predict the impingement force experienced by the target object as a function of fL/D are also plotted in Figures 10A.1-11 and 10A.1-12.

DISCUSSION AND APPLICATION

1. Blowdown Thrust Loads

Thrust and jet impingement forces are produced during a rapid blowdown of a high pressure vessel. Thrust reaction force is a summation of the momentum expulsion rate and the exit plant pressure force. Momentum flow rate is the product of velocity and mass flow rate. Furthermore, the blowdown mass flow rate, velocity, and exit plane pressure are determined by vessel pressure, fluid properties, and the escape geometry. It follows that, for a given pressure vessel blowdown, thrust reaction force is totally determined. The total steady thrust reaction force may be written as follows:

$$\frac{F_T}{A_B} = (P_2 - P_\infty) + \frac{(V_2)^2}{\psi_2 g_c} \tag{1}$$

or

$$\frac{F_T}{A_B} = (P_2 - P_\infty) + \frac{(G_2)^2\ \psi_2}{g_c} \tag{2}$$

For definition of terms, refer to the list of notations at the end of this section.

Figure 10A.1-8 shows a blowdown of steam or saturated water from a vessel through an arbitrary pipe. The impingement target is located sufficiently far away that full jet expansion to environ pressure, P_∞ has occurred. It follows from conservation of momentum equations that the steady thrust reaction force and total jet impingement force per unit break area A_B can be expressed by:

$$\frac{F_T}{A_B} = (P_2 - P_\infty) + \frac{(G_2)^2\ \psi_2}{g_c} \tag{2}$$

$$\frac{F_j}{A_B} = \frac{A_B}{A}\frac{(G_2)^2\ \psi_2}{g_c} = \frac{A_\infty}{A_B}\frac{(G_2)^2\ \psi_\omega}{g_c} \tag{3}$$

Furthermore, a simple force balance on the steady jet which impinges normally on the flat wall shows that the total jet force and the total thrust are equal but opposite in direction giving

$$\frac{F_T}{A_B} = \frac{F_j}{A_B} \quad (4)$$

If ideal gas is assumed as the fluid flowing and blowdown through an isentropic nozzle (zero friction), it follows that for $k=C_p/C_v = 1.3$ the thrust may reach the maximum value

$$\frac{F_T}{A_B} = \frac{F_j}{A_B} = 1.26P_o - P \quad (5)$$

However, pipe friction and upstream area restrictions significantly affect the steady thrust loads.

Pipe friction effects on steam or saturated water blowdown steady thrust can be incorporated in the steady thrust loads from Figure 10A.1-9. Figure 10A.1-9 can also be used to estimate the thrust load for pipe break of any water line that is directly connected to the pressure vessel provided that subcooling of the blowdown zone in the pressure vessel is not greater than 22 Btu/lb.

If the postulated rupture pipe has an upstream area restriction such as flow-limiting venturi or feedwater orifice, the steady thrust loads can be seriously affected. Figure 10A.1-10 should be used to determine steady thrust loads for various break-to-restriction-area ratios in circumferentially ruptured pipes that initially contained steam or water.

2. Jet Impingement Loads

Blowdown flow will form a jet which can produce impact forces on pipes or other mechanical target objects in its path.

Total steady-state jet impingement force per unit break area is given in Equation (3). It follows from Equations (4) and (5) that the maximum value of total steady-state jet load for saturated steam or steam/water mixture blowdown through an isentropic nozzle where entire jet intercepted by target is

$$\frac{F_j}{A_B} = 1.26P_o - P_\infty \quad (6)$$

If the blowdown pipe friction is significant, Figure 10A.1-9 would be used to determine F_j/A. If there is an area restriction in the line, use Figure 10A.1-10 with $F_j = F_T$.

Total force on target objects, which are submerged in a jet (i.e., target area A_T is less the fully expanded free jet area A_∞) and do not fully intercept the jet, can be estimated from the product of "jet pressure" F_j/A_∞ (Figure 10A.1-11) and projected target area, A_T. If A_T is greater than A_∞ (target intercepts the jet), the full jet load F_j, which is equal to total thrust in Figure 10A.1-9 should be used. If the target is very close to a break where jet originates, full expansion will not occur so that Figure 10A.1-11 is invalid. Data of Faletti (Reference 1) indicates that full jet expansion probably occurs about five pipe diameters of axial travel after leaving the break. Therefore, whenever $L_\infty/D \geq 5$, jet pressure of Figure 10A.1-11 is valid. However, if $L_\infty/D < 5$, a jet pressure equal to F_T/A, and jet area A would be more appropriate.

Whether or not the target is fully submerged in a jet can be determined from the jet expanded area as follows:

A) $L_\infty/D \geq 5$.

Figure 10A.1-12 gives the expanded area, A_∞.

If $A_T < A_\infty$ target is fully submerged and impingement load = jet pressure x A_T.

If $A_T \geq A_\infty$ target intercepts entire jet and impingement load = F_j.

B) $L_\infty/D < 5$.

If $A_T < A$ target is fully submerged and impingement load = $F_j A_T / A$.

If $A_T > A$ target intercepts entire jet and impingement load = F_j.

Notations

A	=	Jet area, ft^2
A_B	=	Pipe flow area or break flow area, ft^2
A_R	=	Restriction flow area, ft^2
A_∞	=	Fully expanded free jet area, ft^2
A_T	=	Projected target area, ft^2
D	=	Pipe hydraulic diameter, ft
F_T	=	Total thrust, lbf
F_j	=	Jet impingement force, lbf
f	=	Moody friction factor
L	=	Equivalent pipe length for pressure loss from vessel
L_∞	=	Distance from pipe break to target, ft
P_o	=	Vessel pressure, lbf/ft^2
P_{sat}	=	Saturation pressure, lbf/ft^2
P_2	=	Exit plane pressure, lbf/ft^2
P	=	Atmospheric pressure, lbf/ft^2
ρ	=	Fluid density, lbm/ft^3
ψ	=	Fluid specific volume, ft^3/lb
G	=	Mass flow rate per unit area
V	=	Velocity
C_P	=	Constant pressure specific heat
C_V	=	Constant volume specific heat
g_c	=	Newton's constant, 32.2 $lbm\text{-}ft/lbf\text{-}sec^2$

10A.1.4.4 Mathematical Model and Dynamic Analysis

A large pipe break is assumed to be a one-time event, requiring a plant shutdown and necessary repairs. Permanent deformation of the pipe and restraint are allowed.

An energy balance method was used for the pipe whip restraint design. This method is similar to the maximum deflection of a system subjected to a long duration loading relative to the natural period as presented on Page 222 of Reference 6. The mathematical model is shown in Figure 10A.1-5. When required to accommodate the thermal movement of the pipe, gaps were provided

between the pipe and the restraint, and the effects of these gaps were considered in the dynamic analysis. Thus, the formula shown on Page 222 of "Introduction to Structural Dynamics" is modified as follows:

$$F\left(\frac{Y_g}{Y_{el}} + \mu\right) = R_m(\mu - 1/2)$$

where:

F	=	Jet force acting upon the pipe
Y_g	=	Gap between the pipe and the restraint
m	=	Ductility ratio, i.e., Y_{max}/Y_{el}
R_m	=	Restraint resistance force
Y_{el}	=	Deflection of the pipe and the restraint at the yield stress

As suggested by the AEC, a comparison was made between a time-history analysis method and the energy balance method of the restraint design. The results of this comparison are shown in Table 10A-7. A comparison was made on four different restraints. As shown in this table, a stepped forcing function was used in the time-history analysis, compared to a straight line forcing function (Figure 10A.1-15) used in the energy balance method of analysis. Two different sets of analyses were performed in the comparison. In the first set of calculations, the elastic deflections given by the time-history method, were used in the energy-balance method. In the second set of calculations, properties of a given restraint were used in the energy-balance method.

The results of this comparison indicated that the resistance force on the restraint (yield capacity of the restraint) will be similar in both analyses.

10A.1.4.5 Unrestrained Motion of the Ruptured Line

The MS line is restrained at the postulated break locations and additional restraints are provided to preclude axial movement within the encapsulation sleeve. No damage, therefore, can occur to structures, systems and components important to the plant safety due to a MSLB.

10A.1.5 PROTECTION AGAINST PIPE WHIP, JET IMPINGEMENT, AND REACTIVE FORCE

10A.1.5.1 Pipe Whip Restraints and Encapsulation
10A.1.5.2

The MS line is encapsulated and restrained to prevent pipe whip, jet impingement, or reactive forces from damaging other plant components and structures required for safety following a longitudinal or circumferential break.

The MS line encapsulation sleeve is designed in accordance with the following criteria:

a. The encapsulation sleeve is designed and supported in a manner which will not introduce significant strain concentrations on the encapsulated section of piping.

b. The piping beyond the encapsulation sleeve is provided with pipe whip restraints (or anchors) which restrict its axial displacement and motion within the sleeve following a postulated circumferential pipe break.

c. The encapsulation sleeve is designed (a) to withstand the dynamic forces of internal pressurization resulting from the escape of high energy fluid at the postulated pipe break location assuming complete pipe severance and axial separation to the extent permitted by the pipe restraints, and (b) to

restrict the flow at the open ends of the sleeve to a level required to preclude compartment pressurization beyond the allowable structure design limits.

d. The stresses imposed on the encapsulation sleeve during dynamic pressurization are limited to the design limits associated with "emergency condition" as permitted by the rules of American Society of Mechanical Engineers (ASME) Section III - Nuclear Power Plant Components Code, for Class 2 components.

e. All material for use in the encapsulation sleeves was procured to the requirements of Article NC-2000 of ASME Code, Section III, 1971.

f. Fabrication of the encapsulation sleeves is in accordance with the requirements of Article NC-4000 of ASME Code, Section III, 1971.

g. Full-penetration shop welds were radiographed in accordance with ASME Code, Section III, Class 2, and Code Case 1554.

h. Full-penetration field welds of the encapsulation sleeve were magnetic particle or liquid penetrant examined in accordance with the procedures described in Appendix IX-3500 or IX-3600 of ASME Code, Section III, 1971, with the acceptance standards of paragraph NB-5320 of the Code. Examinations were performed at the one-third level, two-thirds level and of the final welded surface.

i. The design of the encapsulation sleeve permits either its removal by machinery or flame-cutting techniques, or the replacement of encapsulated pipe section in the event leaks develop which require repair or replacement of the pipe.

j. Pipe weld joints located within the encapsulation sleeve and not accessible for subsequent ISI were non-destructively examined prior to the assembly of the encapsulation sleeve. The results satisfy the acceptance standards of ASME Section XI, Inservice Inspection Code.

k. The encapsulation sleeve is provided with open vent and drain pipe nipples as a means of monitoring the encapsulated pipe section for any leaks which might develop in service. These nipples extend beyond the pipe insulation.

l. The piping welds not encapsulated within the piping runs traversing safety-related areas, or within compartments adjoining safety-related areas were subjected to periodic inservice examinations in accordance with ASME Section XI Code Class 2 component requirements except that 100% of such welds were examined during each inspection interval. Alternatively, a risk-informed process for piping outlined in Reference 8 may be used for the weld selections and the determination of additional examinations when defects are discovered. This applies to the MFW and MS systems within the Auxiliary Building.

Figure 10A.1-3 shows an encapsulation detail for the MS System. The fluid head at the containment penetration is designed for pressure build-up or movements due to a pipe break in the Auxiliary Building.

The jet forces from a critical crack of less than 10 kips are not significant enough to create a pipe whip affecting Category I structures, systems, and components. The jet forces will produce low bending stresses well within the elastic range of the pipe with an expected pipe movement of less than 1/4" in the worst case. The jet impingement force resulting from a single critical crack will be shielded as required to prevent damage to the safety-related components, systems, and equipment. For further discussion see Section 10A.1.13C.

For those locations where the postulated break area would exceed 28.9 in^2, (the area of the largest branch line) the pipe is encapsulated to limit the blowdown to less than 291 lbm/sec (analysis in Section 10A.1.20). This is accomplished by limiting the release area between the ends of the encapsulation and the pipe to a net area less than 28.9 in^2. The encapsulation will also dissipate the jet impingement forces.

Following a steam line break, the pressure will instantaneously build up inside the encapsulation because of the restriction of blowdown through the gap. Supports are located between the encapsulation and the pipe to prevent displacement of the pipe normal to its axis. Whipping restraints are located so that the encapsulations are rigidly held in place and the MS lines are prevented from pulling out of the encapsulations.

A vent stack has been provided to vent the MS line Penetration Room to a compartment pressure below the acceptable level, which would affect the integrity of the Category I structures, system or components important to plant safety (Section 10A.1.20).

10A.1.5.3 Separation Provisions
10A.1.5.4

The MS lines are run parallel to each other approximately 5'10" apart. Separation of redundant features of the MS lines is accomplished by a combination of encapsulation and properly placed restraints.

The safety relief valves are arranged such that jet forces from the safety relief valves on one line will not affect the valves on the adjacent line.

The existing exhaust stack support steel (12" structural members) between each relief valve inlet will provide protection and separation of adjacent MS relief valves from the jet impingement force resulting from a circumferential or slot break at the 6" MS nozzle to the relief valves.

Additional steel was provided in the area of the relief valves where required to ensure complete protection against jet impingement from a 6" MSLB.

A jet impingement barrier, which consists of a steel plate, is provided between the MS line and L_b wall to protect the wall from the jet force resulting from a 6" MSLB.

10A.1.5.5 Description of a Typical Pipe Whip Restraint

The pipe whipping restraints are provided at the postulated break locations. Additional restraints are provided near elbows and other critical locations to control the pipe whip impact and axial movement due to a full break at the postulated break locations. Figure 10A.1-2 shows the location of restraints for the MS line.

The design and detail of a pipe whip restraint depends upon many variables, such as physical location, amount of force to be sustained and thermal movement of the pipe. A typical pipe whip restraint is a rigid structure of heavy structural steel members and/or steel plates. It is supported from the existing structural components, such as floors, walls and columns. When the restraint loads cannot be sustained by the existing structure of structural components, these loads are transferred to the foundation level using additional supports. Figure 10A.1-4 shows details of a pipe whip restraint for the MS line.

10A.1.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

10A.1.6.1 Method of Evaluating Stresses

Category I existing and added structures were evaluated for structural adequacy following a postulated rupture using the design bases shown in Appendix 5A. Ultimate strength design method for concrete was used as given in the above reference.

10A.1.6.2 Allowable Design Stress

Design stresses are proportioned such that the combined stresses are within the limits established in Appendix 5A.

10A.1.6.3 Load Factors and Load Combinations

Load factors and load combinations are discussed in Section 10A.1.7. A further discussion of load factors and load combination is provided in Section 5.

10A.1.6.4 Stresses in Category I Structure

Main steam line is encapsulated at the postulated break locations. The jet impingement forces resulting from a postulated pipe break are retained in the encapsulation pipe and are not taken by the structure or structural components. Any jet forces escaping from the encapsulation pipe are distributed such that they will not affect the structure.

The magnitude of a jet impingement force, due to a critical crack in the MS line, will be less than 10 kips. A simplified approach to impingement forces assumes the jet to disperse uniformly at the half angle of incidence between jet axis and the target surface. The half angle, ϕ, is taken as 10°. Thus, the pressure at distance X is:

$$P_j = F_j/A_j$$

where,

P_j = Effective jet pressure on the target

F_j = Jet impingement forces in kips

A_j = The cross-sectional area, in square inches, normal to the jet

Table 10A-2 shows the concrete and steel stresses due to jet impingement forces resulting from a critical crack plus 1 psi compartment pressurization on various structural components in the vicinity of the MS line.

Table 10A-3 shows the concrete and reinforcing steel stresses due to the pressurization of 2.6 psi resulting from a postulated pipe rupture.

The calculated stresses shown in the above tables are combined stresses, including the effects of pipe rupture, plus the effects of live load, dead load, equipment load, and Safe Shutdown Earthquake (SSE) loads. Allowable stress for the concrete is taken at 85% of the ultimate strength. Concrete, having an ultimate strength of 4,000 psi, is used. The allowable stress for the reinforcing steel is taken at 90% of the yield strength. Reinforcing steel, having a minimum yield of 40,000 psi, is used. The structures are also evaluated for the effects of pipe breaks which are transmitted through the restraints.

10A.1.6.5 Erosion of Concrete from Jet Impingement Forces

Since encapsulation pipes are used to prevent full area pipe rupture jet forces from effecting the structure, the only jet impingement force that must be considered is

from the critical crack. The most severe jet force condition occurs where the steam line is 1' away from the concrete. The exit velocity is expected to be approximately 1500 fps with a total force of 10,000 lbs distributed over an area of 84 in^2.

Most of the work done relating to blast erosion of concrete has been with reference to blast from jet engines of aircraft. Some of the effects of jet blast have been discussed in Reference 7. The work has been done at 1250°F and velocities at 3500 fps. The results of these tests and actual service showed that concrete pavement suffered light damage. Since our velocities and temperatures are considerable less than those obtained from the jet engine, excessive erosion of our structure concrete will not be a problem. The jet impingement forces which are expected will be on a local area for a relatively short duration and should not damage the structure adequacy.

However, if the effects of these jet forces are determined to significantly erode the concrete, steel shielding plates will be provided as required.

10A.1.7 STRUCTURAL DESIGN LOADS

The following design loads are used to evaluate the adequacy of Category I structures following a postulated rupture:

Dead Load - Actual weight of structural elements supported.

Live Load - Maximum expected live load in the area under consideration.

Equipment Load - Actual static load of equipment.

Pipe Load - Maximum calculated forces expected under normal operating and upset conditions. The forces include dead load, seismic forces, and thermal forces.

Pressurization - The maximum expected compartment pressure build up that would result from a postulated rupture.

Jet Impingement - Jet impingement forces resulting from full pipe area breaks are retained in the encapsulation pipe and are not taken by the structure. The forces resulting from critical cracks were considered.

Temperature - The effects of temperature increase from a pipe rupture are considered to be short term increases and will not affect the structure adequacy.

Seismic Forces - Seismic forces as shown in Appendix 5A.

These loads are combined using the following load combination equations to evaluate the structural integrity of a Category I structure following a postulated high energy pipe line rupture.

$Y = 1/\phi\ (1.25D + 1.00R + 1.25E)$
$Y = 1/\phi\ (1.25D + 1.25H + 1.25E)$
$Y = 1/\phi\ (1.00D + 1.00R + 1.00E')$
$Y = 1/\phi\ (1.00D + 1.00H + 1.00E')$

Y = required yield strength of the structure

D = dead load of structure, actual static weight of equipment, expected live load in the area under consideration. In addition, any other permanent loads contributing stress, such as soil or hydrostatic loads.

R = reactions from the pipe whip restraints, the maximum expected compartment pressure build-up that would result from a postulated rupture and jet impingement forces resulting from the critical crack (jet impingement forces

resulting from a postulated pipe break are retained in the encapsulation pipe and are not taken by the structure).

H = maximum calculated forces expected under normal operating and upset conditions. The forces include dead loads, seismic loads and thermal expansion of restrained pipes under normal operating conditions.

E = Operating Basis Earthquake (OBE) load.

E' = SSE load.

ϕ = yield capacity reduction factor as defined in Appendix 5A.

10A.1.8 REVERSAL OF LOADS ON THE STRUCTURE

The forces which could cause reversal of loadings due to the postulated accident, on the Seismic Category I structures or structural components are:

a. Jet Impingement Force
b. Compartment Pressurization
c. Reaction from Pipe Whip Restraint

Since the MS line is encapsulated at the postulated full break locations, the existing Category I structures or structural components will not be affected by the jet impingement forces.

A vent stack is provided to vent the MS line compartment at Elevation 27'0" (Figure 10A.1-7). The pressure in the MS line compartment, due to a postulated full break, will be limited to an acceptable level by providing the vent stack and the encapsulation pipe (Section 10A.1.20). The maximum pressure, in the MS line compartment, will not affect the integrity of the Category I structures or structural components.

Pipe whip restraints are supported by the existing structural components. When the restraint loads cannot be sustained by the existing structure of structural components, these loads are transferred to the foundation level using additional supports.

The effects of jet impingement forces and the pressurization due to the postulated single critical crack were insignificant except in the existing pipe tunnel. The roof of this pipe tunnel was adequately strengthened in order to make the tunnel safe against the reversal of loads due to the postulated single critical crack.

10A.1.9 STRUCTURAL EFFECTS OF OPENINGS

The openings are designed and located such that no adverse structural effects are incurred. Venting from the MS compartment was accomplished by the use of the existing pipe tunnel and the addition of a vent to the roof. The vent to the roof was made through existing tendon access openings, which required no additional reinforcing.

10A.1.10 EFFECT OF STRUCTURAL FAILURE

There will not be a failure of any structure, including Category II (non-seismic Category I) structures, due to the accident, that could cause failure of any other structure in a manner to adversely affect:

a. Mitigation of the consequences of the accident; and
b. Capability to bring the unit(s) to a cold shutdown condition.

10A.1.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

In the event of a MSLB in the Auxiliary Building the only region affected is the MS Penetration Room (Figure 10A.1-2).

The structures are designed to contain the escaping high energy fluid and to vent and/or drain this fluid safely.

The only passages through which steam can pass are the vents to the outside atmosphere and the tunnel to the Turbine Building (Section 10A.1.20.4). All doors, piping penetrations, and electrical penetrations are leak tight and capable of withstanding the pressure in the Penetration Room. The plant ventilation system does not communicate with this Penetration Room. A separate duct to the atmosphere is provided for normal ventilation in this room. Both the vent stack (Figure 10A.1-7) and the normal ventilation duct are designed to withstand a Penetration Room pressure of 5.0 psig so that steam cannot break through to any other region in the Auxiliary Building. As this paragraph illustrates, steam is prevented from propagating into the other areas of the Auxiliary Building.

Any steam escaping to the Turbine Building will not reach any vital equipment, instruments, electrical supplies, or cables, and will not flow back into the Auxiliary Building.

The only safety-related equipment located in the MS Penetration Room are:

a. Main steam isolation valves (MSIVs)
b. Main feedwater isolation valves (MFIVs)
c. Control valves (CVs) on the MS line to the AFWP turbines
d. Cable trays

Section 10A.1.13 describes the qualification of items a, b, and c above to properly function in the steam environment. Section 10A.1.20 describes the methods used to determine the standards to be met for environmental qualification and lists the resulting values. Section 10A.1.5 describes the protection provided against pipe whip, jet impingement, and reactive forces. The cable trays are enclosed in metal shielding to prevent damage from jet forces or a steam environment.

The steel conduits will withstand a pressure of 5 psig and 300°F of steam environment. The junction boxes are designed to withstand the above conditions (5 psig and 300°F).

In the event of a MS line rupture in the Turbine Building, the only regions affected are in the Turbine Building. The pressure levels that would result are insufficient to cause damage to the adjoining Auxiliary Building structures or security doors (Section 10A.1.20). The steam will not propagate into the Auxiliary Building.

Safety-related portions of main steam drains 5 and 6 penetrate the K-line wall and are located adjacent to it on the 12’ and 27’ Elevations of the Turbine Building. While some of this piping downstream of the included level switches exceeds 1”, it is supplied by 1” piping. Therefore, no pipe breaks are required to postulated. Even if a break were postulated, there are no other safety-related components around this piping on the Unit 1 side and only the 36” saltwater ram’s head on the Unit 2 side. Clearly, this small piping poses no threat to the ram’s head’s integrity. However, the main steam headers pass over these drains on the 27’ Elevations of both Units 1 and 2. A postulated break in these main steam headers could potentially rupture either or both of these drains. This condition was evaluated, and the results showed that this event would not impair the ability to achieve shut down and would not increase the consequences beyond that of the

ruptured team line alone. Therefore, no barriers or restrains are required to protect these drains from a break in the main steam headers.

The location of the instrumentation associated with the Reactor Protection System and the ESF Systems in relation to the high energy piping is such that the instruments will not be affected by pipe whip or jet impingement. The instruments have been qualified for a high temperature and pressure environment.

The safety relief valve operation will not be affected by the steam environment in the MS valve room after a pipe rupture. The turbine bypass line and the turbine bypass valve are located in the Turbine Building and their operation will not be affected by the steam environment.

The atmospheric steam dump valves are discussed in Section 10.1.2.1. These valves are located in the Auxiliary Building immediately above the MS Penetration Room and are in an enclosure that communicates with the MS Penetration Room (Section 10A.1.20.5). The atmospheric dump valves are not safety-related and are not required to operate during this accident. Should one of these valves inadvertently open, the operator has sufficient time to feed the unaffected steam generator with the AFW System.

10A.1.12 EFFECT ON CONTROL ROOM

The results of the analysis presented in Section 10A.1.20 show that the Control Room will not be affected by a break in the MS line.

10A.1.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

A. Identification

The following electrical equipment and valves must be qualified to meet the requirements of Section 10A.1.20:

1. Both MSIVs and both gas/hydraulic actuators
2. Main Steam to AFWPs CVs, 1/2-CV-4070/4071 and 1/2-CV-4070A/4071A
3. Both MFIVs, Motor-Operated Valves (MOVs)-4516 and 4517

B. Testing

The Limitorque valve operators on the MFIVs are similar to Limitorque valve operators which have been tested in simulated reactor containment post-accident steam environment conditions. These tests were performed by Franklin Institute Research Laboratories and are summarized in their report Number F-C3441. The valve operators were exposed to a steam environment for 30 days, including two temperature cycles going to 340°F during the first day. The resulting pressure/temperature profile closely followed that recommended by a cognizant IEEE committee. (Reference 3)

The AFWP steam isolation valves are Fisher Controls Valves [1/2-CV-4070/4071] and Rockwell bypass valves adapted to accept Valtek actuators [1/2-CV-4070A/4071A]. These valves and associated appurtenances have been analyzed to be qualified for the anticipated environments.

The worst-case environment in the MS piping Penetration Room following a MS line critical crack will be a wet steam and air mixture at 2.23 psig and 331°F. The MSIV actuator has been tested and demonstrated its ability to perform its design function under the above environmental conditions.

C. Criteria for Protecting Category I Systems, Components, or Equipment

The MS line is encapsulated at the postulated break locations. The jet impingement forces due to a postulated break are retained in the encapsulation pipe and the Category I systems, components, or equipment will not be affected by the jet impingement forces.

The magnitude of a jet impingement force resulting from a critical crack in the MS line will be less than 10 kips. A simplified approach to impingement forces assumes the jet to disperse uniformly at the half angle of incidence between the jet axis and the target surface. The half angle, ϕ, is taken as 10°. Thus, the jet pressure at distance X is:

$$P_j = F_j/A_j$$

where,

F_j = Jet impingement force

A_j = Cross-sectional area normal to the jet (expanded jet area)

When the target (Category I equipment or systems such as cable, cable trays, instruments) is fully submerged in a jet, jet impingement force = $P_j \times A_T$

where,

P_j = Jet pressure

A_T = Target area

When the target intercepts the jet, that is, target area is larger than the expanded jet area.

Jet impingement force = F_j

When necessary, barriers are provided to protect Category I systems, components, and equipment against the jet impingement forces. A detail of a typical barrier for protecting cable trays is shown in Figure 10A.1-13. The design criteria for barrier design are similar to the Category I structure design criteria and are discussed in Section 10A.1.7.

To prevent steam from escaping into areas affecting vital equipment and instrumentation, pressure seals designed to withstand the necessary pressure and temperature have been provided where required. The doors in the compartment walls are also designed as pressure retaining doors and are sealed accordingly.

All conduits at the junction boxes in the MS valve room are sealed against the steam environment such that no steam can pass through conduits to any other areas.

D. Control Room

A break at any of the postulated locations has no effect on the Control Room environment.

E. Onsite Power

The steam is prevented by walls from propagating into the Switchgear and the Emergency Diesel Generator (EDG) Rooms and, therefore, the onsite power sources and distribution systems will remain operable.

10A.1.14 DESIGN DIAGRAMS AND DRAWINGS

Figure 10A.1-1 is the MS System diagram. The routing of the MS line through the Auxiliary Building is shown on Figure 10A.1-2. This drawing shows the location of safety-related equipment located near the MS lines. It also shows the pressure retaining walls that will prevent the propagation of steam, and the vents that limit the pressure rise in the MS Penetration Room.

Figure 10A.1-3 shows a pipe encapsulation detail and Figure 10A.1-4 shows a whipping restraint detail. Figure 10A.1-5 shows a mathematical model for pipe whip restraint.

Figure 10A.1-14 shows the AFWP Room and Service Water (SRW) Pump Room ventilation system.

10A.1.15 FLOODING

The postulated break of the MS line in the Auxiliary and Turbine Buildings will release high quality steam, most of which is vented so as not to damage vital equipment or structures. Any moisture separated from the escaping steam or formed by condensation on cold surfaces can be adequately handled by two 6"-diameter drain lines penetrating the tunnel wall at floor level and gravity draining to the turbine room floor drain system at floor Elevation 12'0". Watertight doors are provided in the Auxiliary Building to prevent flooding the Penetration Room to other parts of the building. There are administrative controls (including Technical Specifications) on the open/closed status of the doors.

10A.1.16 QUALITY CONTROL

The quality control and quality assurance for the safety-related piping is in accordance with Appendix 1B. The level of quality control coverage for the remainder of the piping runs was selected on the basis of importance to plant operating reliability and it is intended that the same degree of quality controls will be maintained.

10A.1.17 LEAK DETECTION

Temperature switches are located within the MS line Penetration Room, which will alarm and will alert the operator to the abnormally high temperatures that could result from a small crack. No credit is taken for these switches. In the event there is a large rupture, the instrumentation associated with the steam generators will alert the operator to an MSLB (Section 10A.1.18).

10A.1.18 EMERGENCY PROCEDURES

Following a steam line rupture in the Auxiliary Building or the Turbine Building, the applicable emergency operating procedure would be implemented.

Depending on the size of the leak, indications may or may not be received. If indications are not received after a small leak has occurred, the operator would note the leak on his rounds in the Auxiliary Building and Turbine Building (four hours maximum). The operator would then evaluate the need to shut down the plant.

10A.1.19 SEISMIC AND QUALITY CLASSIFICATION

The MS lines are designed and constructed to meet American National Standards Institute (ANSI) B31.1 requirements with 100% radiograph of butt-welds in piping greater than 2" NPS, except for the portion that penetrates the Containment out through the MSIV. This portion is designed and constructed to meet the requirements of ASME Section III, Class 2. The non-destructive examination requirements of butt-welds in those

portions of the MS piping built to ANSI B31.1 and outside of the ISI boundary have been revised to the following requirements:

NPS > 8" 100% Radiographic Examination

NPS ≤ 8" The weld root pass is to be fabricated by GTAW method. A surface examination will be performed on the weld root and the final weld. Examination method is to be magnetic particle when practical, otherwise the liquid penetrant method shall be used. Also, a radiographic examination may be performed as an alternative to the above requirements. For 2" NPS and under piping, weld inspection shall be per code requirements.

The MS line from the steam generator outlet to the Turbine Building wall is designed as a Category I (seismic) system. The design data for the MS line are given in Table 10-1. The seismic requirements for Category I (seismic) systems are described in Appendix 5A.

10A.1.20 DESCRIPTION OF ASSUMPTIONS, METHODS AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

The opening at the end of the tunnel between the MS Penetration Room and the Turbine Building serves as a vent to relieve the pressure buildup. A wall obstructs the end of the tunnel; however, this wall is of lighter construction than any other structural component of the tunnel. With the wall in place, the pressure will build up until the wall collapses (at less than 1.0 psig). The pressure will immediately decay in the tunnel. The results of this analysis indicate, therefore, that the maximum pressure in that area will not exceed 1.0 psig.

The wall will be designed to fail at 0.5 psi or a hydrostatic pressure of 3' of water. The construction will consist of gypsum wallboard with 24 gauge metal frame work. The retainer clips used in the construction of the wall will be designed to fail on the application on either of the above design loads.

No credit is taken in the following analyses for the tunnel or the blow-out wall.

10A.1.20.1 Circumferential Break of 34" Line - 1608 in^2

In accordance with the break location criteria, presented in Section 10A.1.2, circumferential breaks are postulated in the MS line Penetration Room in the Auxiliary Building and in the Turbine Building (Figure 10A.1-2).

The effect of a full, double-ended break and its associated blowdown was not considered in the Auxiliary Building due to the encapsulation design which limits the steam release rates (Section 10A.1.20.3).

The postulated break in the Turbine Building, however, is not encapsulated and the effects of a full, double-ended break were studied. The break is located at the turbine nozzle (terminal end). The mass and energy release data was computed by the Nuclear Steam Supply System supplier for the accident postulated in Section 14.14. The assumptions used to generate system blowdown data for this accident are applicable to a break located in the Turbine Building. This blowdown data, which is given in Table 10A-4, was used to evaluate the pressure transient in the Turbine Building.

The Bechtel computer code Compartment Pressure Analysis (COPRA) was used to analyze the pressure transient for this problem. Compartment Pressure Analysis is a computer code for predicting the pressure differential across the walls of two adjoining compartments following rapid steam and water blowdown within

one of the compartments. The code is intended as a design tool. It is written in FORTRAN IV for the Honeywell 635 computer.

The equations and corresponding solutions are divided into two phases: an initialization or steady-state phase, and a calculational or transient phase. The initialization phase sets up quantities such as pressure, masses and temperatures in the compartment atmospheres for the steady state just prior to the blowdown accident. The transient phase described the transient behavior of these quantities during and after blowdown.

The following assumptions are incorporated in the analysis:

A. The steam, water, and air throughout each compartment are in thermal equilibrium at all times.

B. Water, steam, and air entering a compartment are mixed homogeneously and instantaneously; no accumulation of water occurs on the walls or in the sump.

C. There is no heat transfer to the compartment walls or floor.

D. The blowdown expands into Compartment 1 by the following thermodynamic process: first the mass expands isenthalpically to the total compartment pressure. The water present at that time could form more steam only by relatively slow evaporation. The water is assumed to undergo no further change of phase, maintaining thermal equilibrium with steam and air. The steam then completes its isenthalpic expansion to the partial pressure of the steam already in the compartment.

E. If equilibrium calculations result in a superheated atmosphere, then a sufficient quantity of the water suspended in the atmosphere is flashed into steam such that the atmosphere is just saturated. The energy to flash to water is taken from the atmosphere. If all the suspended water were ever used up, the containment atmosphere would remain superheated.

F. For masses passing between compartments, the thermodynamics differ slightly. The steam-air-water mixture entering from the other compartment will be brought to thermodynamic equilibrium without the intermediate step of flashing at the total pressure.

G. The mass flowing between the compartments is a homogeneous, two-phase, water-steam-air mixture. The flow equations, in addition, assume a frictionless, compressible, adiabatic, no-slip model.

H. Two completely separate flow equations are used: one for sharp-edged orifices, and one for all other apertures.

I. The first law of thermodynamics for an open system with no heat transfer, as applied to each compartment, is:

$$\frac{\partial E}{\partial t} = \Sigma_1 \frac{dM_i}{dt} h_i$$

where:

E	=	Total system energy
dM_i	=	Mass transfer into compartments
hi	=	Enthalpy of mass being transferred
t	=	Time

The break was postulated at the turbine nozzles which are below the operating deck. The pressure buildup in this region is relieved by venting through the floor grating areas to the upper and lower elevations. These openings were treated as sharp-edged orifice type passages.

The following assumptions are made for the Turbine Building:

Room Volume below operating deck	=	869,000 ft^3
Room Volume above operating deck	=	$9.0x10^6$ ft^3
Vent Area	=	1492 ft^2
Vent Flow coefficient	=	Orifice type coefficient (supplied by COPRA)

Additional vent areas will occur when the Turbine Building siding breaks off. The siding is released when the differential pressure across the siding exceeds 0.45 psi. The value of 0.45 psi is based on the failure of the retaining clips which hold the siding in place. The failure load of the retaining clips was determined by the siding manufacturer's laboratory test of the ultimate strength of the clips. On this basis there is no reason to expect any pressure greater than 0.45 psig in the region above the operating deck.

The COPRA analysis indicated that the differential pressure across the operating deck reached a maximum value of 0.39 psi and was decreasing before the pressure above the operating deck reached 0.45 psig. This decrease in differential pressure is caused by more mass flowing through the grating area than is coming out of the break; hence, the entire Turbine Building is tending toward an equilibrium pressure. The differential pressure is below 0.30 psi across the operating deck when 0.45 psig is reached above the operating deck. Since the pressure will not exceed 0.45 psig above the operating deck after the siding is released, the pressure below the operating deck will not exceed 0.75 psig after this occurrence. An examination of the COPRA analysis also shows that the pressure below the operating deck never exceeds 0.71 psig prior to the release of the siding.

The conclusion of this analysis is that the pressure will not exceed 0.45 psig above the operating deck, and will not exceed 0.75 psig below the operating deck, following a double-ended MSLB in the Turbine Building.

The wall separating the Auxiliary Building and the Turbine Building is a 3'-thick reinforced concrete wall capable of withstanding an external pressure of 15 psi.

All ventilation openings between the Turbine Building and the Auxiliary Building are protected with quick closing dampers which are actuated by a gauge pressure of 0.125" water column in the Turbine Building. These dampers are designed and tested to withstand a 1.0 psi differential pressure.

The roll-up doors which are located below the operating deck are reinforced by adding removable vertical columns at the third points. This strengthens the doors to withstand pressures in excess of 1 psi.

The remaining doors to the Auxiliary Building are personnel doors which swing open into the Turbine Building. Personnel doors located above the operating deck have been tested to withstand 90 psf (0.65 psi) differential pressure. Personnel doors located below the operating deck will be designed to withstand 1.0 psi differential pressure (Table 10A-8).

10A.1.20.2 Circumferential Break (Single Ended) of 6" Line - 28.9 in^2

The MS branch lines to the relief valves (6" lines), to the AFWP turbines (6" lines) and to the atmospheric dump valves (4" lines) join the 34" MS line in the MS Penetration Room (Figures 10A.1-2 and 10A.2-2). These connections are

considered terminal ends which are circumferential break locations as defined in Section 10A.1.2.

The maximum mass and energy release rates for a 6" circumferential break were calculated using the two-phase, single component, annular flow model developed by Moody (References 4 and 5). To account for the stored energy in the MS lines, an "infinite" reservoir was conservatively assumed just up-stream of the break. To account for the break entrance and exit effects, one velocity head loss was assumed. Normal maximum system pressure of 900 psia and temperature of 532°F were assumed. With this approach, the maximum blowdown is 1450 lb/sec-ft^2. The maximum exit conditions at the break were found to be as follows:

Mass discharge rate, lb/sec	-	291
Enthalpy (average), Btu/lbm	-	1150 (November 1981 analysis used 1200)

The Bechtel computer code COPATTA was used in November 1981, to analyze the pressure transient in this compartment following the postulated break. The COPATTA program is discussed in Section 14.20. The discharge mass rate and enthalpy were assumed constant at the maximum values given above.

The following assumptions are made for the Penetration Room:

Room Volume	=	24,000 ft^3
Vent Area	=	44.1 ft^2
Vent flow coefficient	=	0.71
Initial Room Temperature	=	160°F
Relative Humidity	=	70%
Total Heat Sink Area	=	6765 ft^2
(Concrete Walls, Floor, Ceiling)		

The following additional assumptions were used in the analysis.

a. No credit is taken for equipment such as heat sinks.

b. Room volume does not include the adjacent pipe tunnel nor is credit taken for steam flow into that area.

c. No credit is taken for heating, ventilation and air conditioning operation and the capability to remove heat.

d. Process heat loads are 130,285 Btu/hr until isolation and 65,142 Btu/hr for 3-1/2 hours thereafter. Heat load becomes zero 3-1/2 hours after isolation.

Results of the analysis are presented in Table 10A-1A. This table indicates that the maximum sustained temperature continues until the blowdown is isolated (also Figures 10A.1-15 and 10A.1-16).

10A.1.20.3 <u>Circumferential Break Inside Encapsulation - 53.8 in^2 Clearance</u>

The MS line encapsulations are located inside the MS valve compartment of the Auxiliary Building as shown in Figure 10A.1-2. The construction tolerances imposed on the design limit the gap between the MS line and the closure plates on the ends of the encapsulation and between the MS branch lines and the encapsulation. The total escape area of all openings in any encapsulation section will not exceed 53.8 in^2 for a guillotine break inside the encapsulation. This area does not include the 1" drain and 3/4" vent or take credit for the restraint bolt

obstructions; however, these are insignificant in the pressure calculations. This will correspond to a break flow of approximately 545 lbm/sec using the same mass flux employed in the preceding section.

The November 1981, COPATTA analysis of this break assumed the same room characteristics as listed above. The flow rate of 545 lbm/sec and the enthalpy of 1200 Btu/lbm are assumed constant.

Results of this analysis are presented in Table 10A-1A. This table indicates that the maximum sustained temperature continues until the blowdown is isolated (also Figures 10A.1-17 and 10A.1-18.)

10A.1.20.4 <u>Critical Crack in 34" Line - 8.5 in^2</u>

A major MS line rupture in the tunnel between the MS valve compartment in the Auxiliary Building and the Turbine Building is not a credible event because it does not meet the criteria presented in Section 10A.1.2. Specifically, the steam lines in the tunnel are long, straight sections that have no high stress points or terminal ends. A critical crack, however, is postulated anywhere in the MS line.

The maximum mass and energy released from a critical crack were calculated in the same manner as discussed in Section 10A.1.20.2. The maximum exit conditions were found to be as follows:

Mass discharge rate, lbm/sec	-	86
Enthalpy (average), Btu/lbm	-	1150 (November 1981 analysis used 1200)

The Bechtel computer code COPATTA was used in the November 1981 analysis of this problem. The discharge mass rate and enthalpy were assumed constant at the maximum values given above. Assumptions are listed in 10A.1.20.2.

Results of this analysis are presented in Table 10A-1A. This table indicates that the maximum sustained temperature continues until the blowdown is isolated (also Figures 10A.1-19 and 10A.1-20.)

10A.1.20.5 <u>Critical Crack in 4" Atmospheric Dump Line-0.32 in^2</u>

The 4" lines between the MS header in the MS Penetration Room and the atmospheric dump valves are high energy lines. The atmospheric dump valves are normally closed; hence, the lines downstream of these valves are not in the high energy class.

All of the postulated break locations in the high energy portion of this line are in the MS Penetration Room. These breaks in the 4" lines are smaller than the encapsulated guillotine break discussed in Section 10A.1.20.3, and result in smaller peak compartment pressure.

To protect against critical cracks in those portions of the high energy atmospheric dump valve lines in the compartment above the MS Penetration Room, the lines are completely enclosed from the floor penetrations to a point beyond the valves.

These enclosures are sealed around the lines downstream of the valves and attached to the floor to prevent propagation of steam into the compartment. Steam leakage will be vented from the enclosure into the MS Penetration Room through the floor penetrations.

Therefore, the postulated break discussed in Section 10A.1.20.4 is controlling for the design of the enclosure. That break resulted in a maximum pressure in the MS Penetration Room of 2.2 psig. The enclosure is designed to contain an internal pressure of 5 psig. This affords protection against a critical crack in the atmospheric dump line and also against the postulated line breaks in the MS Penetration Room.

10A.1.21 INTEGRITY OF THE CONTAINMENT STRUCTURE AND A PIPE RUPTURE OUTSIDE THE CONTAINMENT

The Containment Structure is designed using load combinations as discussed in Appendix 5A. The method used in the analysis of the Containment Structure is discussed in Chapter 5.

Since the MS line is encapsulated from the containment boundary to the first high stress point, there will be no jet impingement forces due to a full postulated break to impair the structural integrity of the prestressed concrete Containment Structure.

10A.1.22 EFFECT OF THE BABCOCK & WILCOX, CANADA REPLACEMENT STEAM GENERATORS

The replacement of the original steam generators with Babcock & Wilcox, Canada replacement steam generators does not involve changes to the design of the main steam line piping or its supports and restraints located outside Containment. Babcock & Wilcox, Canada has evaluated the effect of the replacement steam generators on those hydraulic parameters that are significant in determining the magnitude of the forces caused by a pipe break. It is concluded that the loads from secondary side pipe breaks are either unchanged or are reduced with the replacement steam generators in place.

10A.1.23 REFERENCES

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3. Proposed Guide for Type Test of Class I Electrical Valve Operators for Nuclear Power Generating Stations, Draft 13, IEEE Project Number 382, JCNPS/SC2.3, June 1972
4. Maximum Flow Rate of a Single Component, Two Phase Mixture," by F.J. Moody; ASME Transactions, Series C, Volume 87, 1965
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6. Introduction to Structural Dynamics by Professor John M. Biggs, 1964 Edition, published by McGraw-Hill Book Company
7. Perry H. Peterson "Resistance to Fire and Radiation," American Society for Testing and Materials (ASTM) Special Technical Publication No. 169, page 201, dated 1956
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TABLE 10A-1A

MAIN STEAM PENETRATION ROOM ANALYSIS RESULTS

INITIAL ROOM TEMP. = 160°F	6"	34" (Encapsulation)	CRITICAL CRACK
Maximum Sustained Temperature[d]	316°F for 13.5 min[a]	316°F for 7.7 min[b]	308-320°F for 4 hrs[c]
Peak Temperature	327°F < 20 sec	318°F < 20 sec	331°F < 20 sec
Peak Pressure	0.52 psig at 0.35 sec	1.49 psig at 0.50 sec	2.23 psig at 101.5 sec
Time to Return to 160°F	16.9 hrs	9.7 hrs	40.4 hrs

(a) Based on time to isolate feedwater plus time to empty one steam generator.

(b) Based on time to empty one steam generator

(c) Based on leak being located and isolated during required four-hour tours.

(d) Times listed are those taken to isolate blowdown; i.e., maximum temperature persists until blowdown isolation in each analyzed case.

TABLE 10A-1

MAIN STEAM STRESS VALUES

Following is a stress summary of the intermediate points considered between the terminal ends. The postulated full break locations are shown on Figure 10A.1-2. All values are in psi.

		PRIMARY STRESS					
POINT NUMBER	SECONDARY STRESS[a] ($0.8S_A$)	LONGITUDINAL PRESSURE	LONGITUDINAL WEIGHT	SEISMIC OBE	OTHER[c]	TOTAL S_{PR}[b,d] ($<0.8\ S_h$)	TOTAL STRESS[e] [$<0.8\ (S_A + S_h)$]
2	14,513	4,910	415	50	569	5,944	20,457
3	18,121	4,910	1,968	210	323	7,411	25,532
4	22,131	4,910	4,154	343	304	9,711	31,842
6	10,943	4,910	315	22	539	5,786	16,729
7	18,845	4,910	1,726	88	375	7,099	25,944
8	20,069	4,910	2,908	326	107	8,148	28,217

(a) S_A = The larger of f [$1.25\ S_c + 0.25\ S_h + (S_h - S_{PR})$] or

f($1.25\ S_c = 0.25\ S_h$) as per paragraph 102.3.2(c) and (d) of the USAS Code for Pressure Piping, USAS B31.1.0-1967, and as per NC-3600 of Section III (Nuclear Power Plant Components), ASME Boiler and Pressure Vessel (B&PV) Code. 0.85 S_A taken as 21,000 psi.

(b) S_c and S_h are the allowable stresses at cold and hot conditions, respectively, for Class 2 and Class 3 components as per ASME B&PV Code, Section III (Nuclear Power Plant Components).

(c) Other stresses are:
1) Due to steam or water hammer
2) Due to relief valve discharge.

(d) S_{PR} is the total of columns 3 through 6.

(e) 0.8 ($S_A + S_h$) taken as 35,000 psi.

TABLE 10A-2

STRESSES ON STRUCTURAL COMPONENTS DUE TO JET IMPINGEMENT FORCES RESULTING FROM A CRITICAL CRACK OF THE MAIN STEAM LINE

					CALCULATED STRESSES[(a)] DUE TO JET FORCE PLUS 1 psi		CALCULATED STRESSES VS. ALLOWABLE STRESSES	
STRUCTURAL COMPONENT	THICKNESS	DIST FROM RUPTURE	JET FORCE 1.26 PA	CROSS SECTION TARGET AREA ft^2	COMPRESSIVE CONC (psi)	TENSILE REINF (psi)	CONCRETE	REINFORCING
North Tunnel Wall	1'0"	2'0"	9,640	1.40	710	26,000	0.209	0.722
South Tunnel Wall	2'0"	3'0"	9,640	2.40	539	18,400	0.158	0.511
Tunnel Floor	1'3"	1'0"	9,640	0.60	1,012	20,950	0.297	0.582
Tunnel Ceiling EL 33'6"	1'4"	2'6"	9,640	2.00	650	14,039	0.191	0.390
New Wall @ Col. Line 6.1	1'3"	4'0"	9,640	3.90	1,065	25,885	0.313	0.719
Wall L_b	1'9"	3'6"	9,640	3.15	518	16,791	0.152	0.466
Wall 17	2'0"	3'0"	9,640	2.50	539	18,400	0.158	0.511
New Wall @ Col. Mc	1'3"	10'0"	9,640	17.30	376	9,129	0.110	0.253
Floor EL 27'0"	2'6"	1'0"	9,640	0.60	655	20,600	0.193	0.572
Ceiling EL 45'0"	1'3"	11'5"	9,640	21.80	480	11,640	0.141	0.323
Ceiling EL 45'0"	2'6"	3'0"	9,640	2.50	50	1,580	0.01	0.044

(a) Stresses shown are maximum stresses and include live load, dead load, equipment load & SSE seismic stresses.

TABLE 10A-3

STRESSES ON STRUCTURAL COMPONENTS IN THE MAIN STEAM COMPARTMENT DUE TO POSTULATED MAIN STEAM PIPE RUPTURE

		CALCULATED STRESSES[a] DUE TO PRESSURIZATION OF 2.6 psi[b]		CALCULATED STRESSES VS. ALLOWABLE STRESSES	
STRUCTURAL COMPONENT	**THICKNESS**	**COMPRESSIVE IN CONCRETE (psi)**	**TENSILE IN REINFORCING (psi)**	**CONCRETE**	**REINFORCING**
North Tunnel Wall	1'0"	230	8,420	0.068	0.234
South Tunnel Wall	2'0"	492	16,312	0.145	0.453
Tunnel Floor	1'3"	555	11,480	0.163	0.319
Tunnel Ceiling EL 33'6"	1'4"	245	5,292	0.072	0.147
New Wall @ Col. Line 6.1	1'3"	592	14,388	0.174	0.40
Wall L_b	1'9"	438	13,454	0.129	0.374
Wall 17	2'0"	492	16,312	0.145	0.453
New Wall @ Col. Mc	1'3"	592	14,388	0.174	0.40
Floor EL 27'0"	2'6"	615	19,330	0.181	0.537
Ceiling EL 45'0"	1'3"	525	12,710	0.154	0.353
Ceiling EL 45'0"	2'6"	38	1,184	0.01	0.033

(a) Stresses shown are maximum stresses which include live load, dead load, equipment load, and SSE seismic forces.

(b) This pressure is based on original design analyses, later reviews have shown this value to be conservative - Section 10A.1.

TABLE 10A-4

STEAM LINE RUPTURE INCIDENT NO LOAD, 1-LOOP OPERATION BREAK INSIDE TURBINE BUILDING

TIME (sec)	TOTAL BLOWDOWN (lb/sec)	TOTAL MASS (lbs)
0.00	6655.01	0.00
1.00	6215.92	6.409×10^3
2.00	5836.57	1.241×10^4
3.00	5509.71	1.806×10^4
4.00	5219.13	2.338×10^4
5.00	4959.82	2.838×10^4
6.00	4731.27	3.315×10^4
7.00	4523.20	3.767×10^4
8.00	4339.18	4.201×10^4
9.00	4177.27	4.619×10^4
10.00	4033.25	5.022×10^4
20.00	1860.79	8.048×10^4
30.00	1725.64	9.850×10^4
40.00	1485.61	1.146×10^5
50.00	1300.55	1.283×10^5
60.00	1175.84	1.407×10^5
70.00	1067.60	1.518×10^5
80.00	980.12	1.620×10^5
90.00	908.10	1.714×10^5
100.00	848.02	1.802×10^5
150.00	658.80	2.171×10^5
180.00	585.99	2.358×10^5
200.00	546.93	2.470×10^5

TABLE 10A-7

COMPARISON OF TIME-HISTORY PIPE WHIP DESIGN METHOD WITH ENERGY BALANCE DESIGN METHOD

TIME-HISTORY DESIGN METHOD					CALVERT CLIFFS ENERGY BALANCE					
					USING GIVEN RESTRAINT DEFLECTION			USING PROPERTIES OF GIVEN RESTRAINT		
LOAD. COND.	FORCING FUNCTION	TIME SECONDS	MAX. LOAD ON RESTRAINT (kips)	RESTRAINT CAPACITY AT YIELD (kips)	FORCING FUNCTION (kips)	RESTRAINT CAPACITY AT YIELD (kips)	DUCTILITY RATIO	FORCING FUNCTION (kips)	RESTRAINT CAPACITY AT YIELD	DUCTILITY RATIO
1	258	0.0027	615	930	324	1080	5	324	994	5
	180	0.11								
	103	0.30								
2	258	0.0018	646	1050	324	1080	5	324	994	5
	180	0.0625								
	281	0.30								
3	310	0.0063	793	930	324	1080	5	324	994	5
	216	0.031								
	303	0.09								
	257	0.30								
4	310	0.0063	850	1050	324	1080	5	324	994	5
	216	0.031								
	303	0.09								
	257	0.30								

TABLE 10A-8

LOCATION OF DOORS BETWEEN TURBINE BUILDING AND AUXILIARY BUILDING

UFSAR FIGURE NO.	ELEVATION	LOCATION[a]	CAPABILITY
1-6	5'0"	Between columns 105 & 106	10 psig
1-6	5'0"	Between columns 107 & 108	10 psig
1-10	5'0"	Between columns 206 & 207	10 psig
1-10	5'0"	Between columns 208 & 209	10 psig
1-7	27'0"	Between columns 105 & 106	1 psig
1-7	27'0"	Between columns 107 & 108 (roll-up door)	1 psig
1-7	27'0"	Between columns 110 & 111	1 psig
1-7	27'0"	Between columns 112 & 113	1 psig
1-11	27'0"	Between columns 203 & 204	1 psig
1-11	27'0"	Between columns 206 & 207	1 psig
1-11	27'0"	Between columns 208 & 209	1 psig
1-8	45'0"	Between columns 105 & 106	0.65 psig
1-8	45'0"	Between columns 107 & 108 (roll-up door)	1 psig
1-8	45'0"	Between columns 110 & 111	0.65 psig
1-8	45'0"	Between columns 112 & 113 (double door)	0.65 psig
1-12	45'0"	Between columns 201 & 202 (double door)	0.65 psig
1-12	45'0"	Between columns 203 & 204	0.65 psig
1-12	45'0"	Between columns 206 & 207 (roll-up door)	1 psig
1-12	45'0"	Between columns 208 & 209	0.65 psig

[a] All doors are through the Turbine Building/Auxiliary Building wall.

10A.2 MAIN STEAM TO AUXILIARY STEAM GENERATOR FEED PUMP TURBINES

Main steam (normally at 900 psig and 550°F) is supplied to the two auxiliary steam generator feed pump (SGFP) turbines automatically upon low level in the steam generator or by remote manual control from the Control Room.

Each turbine has a manually set governor for controlling turbine speed. Once set for a certain speed, the governor holds the speed constant even if the steam pressure decreases to a minimum of 50 psig.

Main steam drives the AFWP turbine as long as the pressure is equal to or greater than 50 psig. Should the steam pressure in both of the steam generators drop below 50 psig, the steam supply can be switched to the auxiliary boiler steam system.

The branch line from the MS lines to the CVs (1/2-CV-4070, 4071, 4070A and 4071A) before the auxiliary SGFP turbines, will normally contain high energy fluid (MS). A bypass line is provided around each CV station in order to allow for a single failure of the valves. The bypass valves can be actuated by an extended remote manual operator.

Based on the level of quality control, periodic ISI, the low usage factor, the short time this system exceeds 200°F and 275 psig, and the strict administrative control of the system, a break in this system between the CVs and the AFWP turbines is not considered credible.

10A.2.1 PIPE WHIP

The MS line to the auxiliary SGFP turbine contains high energy fluid (above 275 psig and 200°F) only from the MS lines to the CVs and manual bypass valves. Therefore, pipe whip protection will be provided in this area.

10A.2.2 PIPE BREAK LOCATIONS

The criteria used for determining the location of pipe breaks has been presented in Section 10A.1.2. The pipe break locations are shown on Figures 10A.2-2A and 10A.2-2B.

10A.2.3 PIPE BREAK ORIENTATION

The criteria used for determining the orientation of pipe breaks has been presented in Section 10A.1.3.

10A.2.4 SUMMARY OF DYNAMIC ANALYSIS

10A.2.4.1 Location and Number of Design Basis Breaks

The location and number of design basis breaks are chosen in accordance with criteria discussed in Section 10A.1.2, except that one break in a line with a run less than 5 pipe diameters was postulated on Unit 1.

10A.2.4.2 The Postulated Rupture Orientation

The longitudinal break is assumed to be parallel to the pipe axis and oriented at any point around the pipe circumference. The circumferential break is assumed to be perpendicular to the pipe axis. A further discussion of the break area and jet impingement forces is provided in Section 10A.1.4.2.

10A.2.4.3 Description of the Forcing Function and the Mathematical Model

A forcing function is assumed to be a straight line with changes so slow that the variation, up to the time of maximum response, is negligible.

The dynamic analysis method, used in pipe restraint design for this line, is similar to the one used for the MS line which is described in Section 10A.1.4.4. The mathematical model is shown in Figure 10A.1-5.

10A.2.4.4 Unrestrained Motion of the Ruptured Lines

Except for the horizontal runs at the break location upstream of 2-CV-4070, there will not be any unrestrained motion of the MS to auxiliary SGFP turbine line to damage structures, systems, and components important to the plant safety. Movement of this line is restricted by adjacent structures (evaluated as adequate restraint).

10A.2.5 PROTECTION AGAINST PIPE WHIP, JET IMPINGEMENT, AND REACTIVE FORCES

10A.2.5.1 Pipe Whip Restraints and Sleeves

All break locations in the MS to auxiliary SGFP turbine lines up to the bypass and CVs, are restrained and/or sleeved to prevent any pipe whip, jet impingement, or reactive forces from damaging structures, systems, and components important to plant safety following a longitudinal or circumferential break. Sleeves meet the criteria for the encapsulations referenced in Section 10A.1.5 and are securely anchored to surrounding permanent structures to prevent movement. An exception to this statement is the piping at valve station 2-CV-4071. A longitudinal break at this location could result in a jet which could impact the motor bonnet yoke of feedwater isolation valve 2-MOV-4517. An evaluation using the guidelines in Section 10A.1.4 was conducted. This evaluation determined that, due to the construction and configuration of 2-MOV-4517, a protective barrier is not required.

Adequate venting is provided in the MS piping Penetration Room to prevent compartment pressurization from a break or crack in the 6" MS line to the auxiliary SGFP turbines from causing damage to safety-related structures and components.

The jet impingement force due to a critical crack in the 6" MS to auxiliary SGFP turbine lines is 526 lbs. This force is not large enough to cause damage to any safety-related structures, systems, or components. The structures and equipment in the area of these lines have been evaluated for the much greater jet impingement forces resulting from a crack in the 34" MS line.

10A.2.5.2 Separation Provisions

Redundant features of the MS line to the auxiliary SGFP turbines are separated by a combination of sleeves and properly placed restraints. A break or crack in one MS line to the auxiliary SGFP turbines will not affect the integrity of the other steam generator if the CVs 1/2-CV-4070/4071 and 1/2-CV-4070A/4071A and bypass valves are closed. Further redundancy is included with a check valve downstream of each CV.

10A.2.5.3 Description of a Typical Pipe Whip Restraint

The pipe whip restraints are provided in the vicinity of the postulated full break locations to control the pipe whip impact and axial movement due to a full postulated break.

Description of a typical pipe whip restraint is given in Section 10A.1.5.5.

10A.2.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

10A.2.6.1 Method of Evaluating Stresses

Category I structures were evaluated for structural adequacy following a postulated rupture using the design bases shown in Appendix 5A. Ultimate strength design method was used in the structural evaluations. All Category I structures and structural components were found to be adequate against the loadings due to the postulated break.

10A.2.6.2 Allowable Design Stresses

Design stresses are proportioned such that the combined stresses are within the limits established in Appendix 5A.

10A.2.6.3 Load Factors and Load Combinations

Load factors and load combinations used in the design are discussed in Appendix 5A.

10A.2.7 STRUCTURAL DESIGN LOADS

The design loads used to evaluate the adequacy of Category I structures or structural components are discussed in Section 10A.1.7 of the Updated Final Safety Analysis Report (UFSAR).

10A.2.8 REVERSAL OF LOADS ON THE STRUCTURES

The forces causing reversal of loadings, due to the postulated accident, on the Category I structures or structural components are:

a. Jet Impingement Forces
b. Compartment Pressurization
c. Reactions from Pipe Whip Restraints

Since the MS to Auxiliary SGFP Turbine line is sleeved at the postulated pipe break locations, the sleeved pipe will resist the jet impingement forces, and protect the structures, systems and components important to the plant safety. Where encapsulation was not used, jet impingement effects on structures were reviewed at the break locations shown on Figures 10A.2-2A and 10A.2-2B.

The pipe whip restraints are supported from the existing structural components. When the restraint loads cannot be sustained by the existing structure or structural components, these loads are transferred to the foundation level using additional supports.

10A.2.9 STRUCTURAL EFFECTS OF NEW OPENINGS

The vent is discussed in Section 10A.1.9.

10A.2.10 EFFECTS OF STRUCTURAL FAILURE

There will not be a failure of any structure, including Category II (non-seismic Category I) structures, due to the accident, that could cause failure of any other structure in a manner to adversely affect:

a. Mitigation of the consequences of the accident; and
b. Capability to bring the unit(s) to a cold shutdown condition.

10A.2.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

Verification that a rupture in the MS line to auxiliary SGFP turbines will not affect safety is presented in Section 10A.1.11.

10A.2.12 EFFECT ON CONTROL ROOM

The results of the analysis presented in Section 10A.2.20 show that the Control Room will not be affected by a break in the MS lines to the auxiliary SGFP turbines.

10A.2.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Section 10A.1.13 discusses environmental qualification of affected required equipment in the area of the MS lines including the lines to the auxiliary SGFP turbines.

10A.2.14 DESIGN DRAWINGS

Figure 10A.2-1 shows a piping schematic diagram for the MS line to the auxiliary SGFP turbines.

Figures 10A.2-2 and 10A.2-3 show the routing of the MS line to the auxiliary SGFP turbines.

Figures 10A.2-2A and 10A.2-2B show line break locations and whip restraint locations.

10A.2.15 FLOODING

The postulated break in the MS line to the auxiliary SGFPs will release high quality steam, which will be vented to prevent damage to vital equipment or structures. Any moisture separated from the escaping steam or formed by condensation on cold surfaces can be adequately handled by the floor drainage system in the Auxiliary Building and Turbine Building, so that no flooding will occur.

10A.2.16 QUALITY CONTROL AND INSPECTION PROGRAMS

The quality control and inspection programs are presented in Section 10A.1.16.

10A.2.17 LEAK DETECTION

Temperature sensors are located within the MS line Penetration Room, which will alarm and will alert the operator to abnormally high temperatures that could result from the release of steam from a break or crack in the MS line to the auxiliary SGFP turbines.

10A.2.18 EMERGENCY PROCEDURES

Emergency Procedures are discussed in Section 10A.1.18.

10A.2.19 SEISMIC AND QUALITY CLASSIFICATION

The MS lines to the auxiliary SGFP turbines are designed and constructed to meet ANSI B31.7, Class II requirements from the MS line through 1/2-CV-4070/4071 and 1/2-CV-4070A/4071A, and their manual bypass valves. The rest of the lines from these valves through to the auxiliary SGFP turbines are designed and constructed to meet ANSI B31.1 requirements with additional non-destructive examination requirements as follows:

a. Either MT, PT, or radiography for all circumferential butt-welds; and
b. 100% MT or PT for 4" and under welded branch connections and all socket weld pipe and fittings (FCR 89-26 and future modifications).

These lines are designed to withstand a SSE in combination with normal design loads.

10A.2.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

All the postulated breaks in the MS to AFWP turbine lines are located in the MS Penetration Room (Figure 10A.2-2). The largest postulated break is a 6" circumferential break. The pressure and temperature effects of a break of this size in the MS Penetration Room are presented in Section 10A.1.20.3.

10A.2.21 INTEGRITY OF THE CONTAINMENT STRUCTURE AND A PIPE RUPTURE OUTSIDE THE CONTAINMENT

Since the MS to Auxiliary SGFP Turbine line is only 6" in diameter, any forces acting upon the Containment Structure will not be significant enough to impair the integrity of the prestressed concrete Containment Structure.

10A.3 STEAM GENERATOR BLOWDOWN

Each steam generator has an upper and lower blowdown line which can be used to control the build-up of soluble and particulate concentrations within the steam generator. The blowdown system, as shown in Figure 10A.3-1, will normally be operated continuously at a rate of 100-200 gpm per steam generator.

The steam generator pressure will vary between 900 and 850 psia for no-load and full-load operation, respectively. The corresponding saturation temperatures are 532°F and 525°F.

10A.3.1 PIPE WHIPS

The steam generator blowdown system normally operates above 275 psig and 200°F; however, as discussed in Section 10A.3.5, no pipe whip restraints are necessary.

10A.3.2 CRITERIA FOR PIPE BREAK LOCATIONS

The criteria used for determining the location of pipe breaks has been presented in Section 10A.1.2.

10A.3.3 CRITERIA FOR PIPE BREAK ORIENTATION

The criteria used for determining the orientation of pipe breaks has been presented in Section 10A.1.3.

10A.3.4 SUMMARY OF DYNAMIC ANALYSIS

Following a postulated break anywhere in the steam generator blowdown line between the containment penetration and the blowdown tank, the stored energy will disperse and the pressure in the pipe will rapidly decay to about half of its initial value due to choking in the small pipe.

Because of the small size of the line and the reduced pressure in the line, the force of the pipe whipping against adjacent structures is very small; therefore, no dynamic analysis is required.

10A.3.5 PROTECTION AGAINST PIPE WHIP, JET IMPINGEMENT, AND REACTIVE FORCES

In the vicinity of the steam generator blowdown lines in the Auxiliary Building there is no safety-related piping which can be broken by the whipping of a ruptured steam generator blowdown line. Also, there are no vital systems or equipment close enough to the steam generator blowdown lines to be damaged by such pipe whip. Adjacent walls and structures will not fail if impacted by these small blowdown lines. Therefore, no pipe whip restraints are necessary.

In the Auxiliary Building, there are no vital systems or equipment close enough to be damaged by jet impingement. Because of the small size of the blowdown lines, the force from a jet will not fail any adjacent walls or structures. Hence, no impingement protection is necessary.

A portion of the steam generator blowdown piping passes through the K-line in the Unit 2 Turbine Building on the 12’ Elevation in the area of the safety-related main steam drains 5 and 6. One of the two 6” branches goes to the 21 and 23 condensers, while the other goes to the circulating water discharge. The normal pressure and temperature in these lines is 135 psig and 360°, respectively. Therefore, while the temperature exceeds the 200° high energy line criteria, the pressure is less than the 275 psig criteria. An evaluation was done that shows that a postulated break in an area main steam header

that resulted in both main steam drains being ruptured would not impair the ability to achieve shutdown and would not increase the consequences beyond that of the ruptured steam line alone. The results of the main steam header break are considered more severe than those of a break in a blowdown line. Therefore, no barriers are required to protect this safety-related piping from a rupture or jet impingement from the nearby blowdown piping.

10A.3.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

The method of evaluating the adequacy of Category I Auxiliary Building has been summarized in Section 10A.1.6.

10A.3.7 STRUCTURE DESIGN LOADS

There will be no significant additional loads on the structure due to a postulated break in the steam generator blowdown line.

10A.3.8 REVERSAL OF LOADS ON STRUCTURE

There will be no reversal of loads on the structure due to a steam generator blowdown line break.

10A.3.9 STRUCTURAL EFFECT OF OPENINGS ADDED TO THE STRUCTURE

No openings are required to vent the structure following a break in the blowdown line.

10A.3.10 VERIFICATION THAT ANY STRUCTURE FAILURE WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

There will be no failure of a structure due to a postulated break in the steam generator blowdown line.

10A.3.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

Safety-related equipment which could be affected by the environment resulting from a pipe rupture is qualified to function under those conditions (Tables 10A-5 and 10A-6). There is insufficient energy release to cause a hazard to any structure by pipe whip or pressurization so the steam environment cannot spread to other areas (Sections 10A.3.1, 10A.3.4, and 10A.1.20).

Therefore, pipe ruptures in the steam generator blowdown system will not affect safety.

10A.3.12 EFFECT ON CONTROL ROOM

The results of the analysis presented in Section 10A.1.20 shows that the Control Room will not be affected by a blowdown line break.

10A.3.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Environmental qualification is discussed in Section 10A.3.20.

10A.3.14 DESIGN DIAGRAMS AND DRAWINGS

The steam generator blowdown system is shown on Figure 10A.3-1.

The routing of the steam generator blowdown lines through the Auxiliary Building is shown on Figures 10A.3-2 and 10A.3-3.

10A.3.15 FLOODING

Approximately two-thirds of the mass released from a blowdown line break will be in the form of water. This will amount to about 210 gpm which can be adequately drained by the floor drain system.

10A.3.16 QUALITY CONTROL AND INSPECTION PROGRAM

The quality control and inspection programs are presented in Section 10A.1.16.

10A.3.17 LEAK DETECTION

Temperature switches are located in the areas where the steam generator blowdown lines pass through, and will alarm to alert the operator of abnormally high temperatures that could result from a line rupture. However, no credit is taken for these switches (Section 10A.1.17).

10A.3.18 EMERGENCY PROCEDURES

Upon leak or rupture of the steam generator blowdown piping in the Auxiliary Building, the applicable emergency operating procedure would be implemented.

10A.3.19 SEISMIC AND QUALITY CLASSIFICATION

The blowdown lines are designed and constructed to meet ANSI B31.1 requirements, except for the portions of the line that penetrate the Containment up to and including the containment isolation valves, which are designed and constructed to meet B31.7, Class II requirements.

10A.3.20 DESCRIPTION OF ASSUMPTIONS, METHODS AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENT IN COMPARTMENT

The locations of the postulated circumferential or longitudinal breaks are shown in Figures 10A.3-2 and 10A.3-3. The flow from these breaks will be choked to a maximum rate of 43 lbs/sec because the fluid in the steam generator is saturated liquid.

The temperature and pressure build-up in the rooms where the blowdown lines are located was determined with the Bechtel computer code COPATTA. This computer code is described in Section 14.20.

The parameters used with this analysis are as follows:

Initial Room Conditions:	
Temperature	100°F (EL 45')/140°F (EL 27')
Pressure	0 psig
Relative humidity	70%
Blowdown Tank Room (above Elevation 45'0"):	
Volume	29,000 ft^3
Concrete surfaces	8250 ft^2
Vent opening	173.6 ft^2
Vent coefficient	1.0

Penetration Room (below Elevation 45'0"):

Volume	45,565 ft^3
Concrete surfaces	11,210 ft^2
Metal surfaces (grating)	800 ft^2
Vent opening (total)	7.8165 ft^2
Vent coefficient	0.85

For the blowdown tank room the peak pressure and temperature were found to be 0.27 psig at 8.4 minutes, and 234°F at 150 seconds, respectively. Some steam will dissipate into the corridors of the Auxiliary Building through an open passageway, but this small quantity is diluted and exhausted through the ventilation system without adverse effects (Figures 10A.3-4 and 10A.3-5).

For the Penetration Room, the pressure and temperature build-up is slowed by venting through a check damper into the MS valve room. From here the escaping steam has a path directly to the outside of the building (Section 10A.1.20). However, the check damper will not pass the full steam flow from the ruptured blowdown line so the pressure and temperature will continue to rise. Doors to the adjacent ventilation equipment room are not designed as pressure retaining and will fail at a pressure less than 1 psig. With these doors open, there is insufficient mass and energy flow from the broken pipes to sustain a room pressure above 0.5 psig. Hence, the pressure in the room will always be less than 1 psig.

The escaping steam can propagate to other areas of the Auxiliary Building but is diluted and exhausted through the ventilation systems without any adverse effects. The maximum temperature reached in the room where the break occurred is 212°F and the maximum pressure is less than 0.1 psig.

10A.3.21 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURE DUE TO PIPE RUPTURE OUTSIDE

The primary Containment Structure will not be affected due to a postulated break in the steam generator blowdown line.

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
F-2	1-CV-5160	Saltwater Inlet Component Cooling (CC) Heat Exchanger (HX) 11	A	A	A	A
F-2	1-SV-5160	Saltwater Inlet CC HX 11	A	A	A	A
F-2	1-CV-5206	Saltwater Outlet CC HX 11	A	A	A	A
F-2	1-SV-5206	Saltwater Outlet CC HX 11	A	A	A	A
F-2	1-SV-5206A	Saltwater Outlet CC HX 11	A	A	A	A
F-2	1-I/P-5206	Saltwater Outlet CC HX 11	A	A	A	A
J-1	1-HIC-5206	Saltwater Outlet CC HX 11	A	A	A	A
F-2	1-CV-5162	Saltwater Inlet CC HX 12	A	A	A	A
F-2	1-SV-5162	Saltwater Inlet CC HX 12	A	A	A	A
F-2	1-CV-5208	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-SV-5208	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-SV-5208A	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-I/P-5208	Saltwater Outlet CC HX 11	A	A	A	A
J-1	1-HIC-5208	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-CV-5163	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-SV-5163	Saltwater Outlet CC HX 12	A	A	A	A
F-2	1-CV-5150	Saltwater Inlet SRW HX 11	A	A	A	A
F-2	1-SV-5150	Saltwater Inlet SRW HX 11	A	A	A	A
F-2	1-CV-5209	Saltwater Outlet SRW HX 11A	A	A	A	A
F-2	1-FIC-5209	Saltwater Outlet SRW HX 11A	A	A	A	A
F-2	1-CV-5210	Saltwater Outlet SRW HX 11B	A	A	A	A
F-2	1-FIC-5210	Saltwater Outlet SRW HX 11B	A	A	A	A
J-2	1-PIC-5154	Saltwater Bypass SRW HX 11	A	A	A	A
F-2	1-CV-5154	Saltwater Bypass SRW HX 11	A	A	A	A
J-2	1-I/P-5154	Saltwater Bypass SRW HX 11	A	A	A	A
F-2	1-CV-5152	Saltwater Inlet SRW HX 12	A	A	A	A
F-2	1-SV-5152	Saltwater Inlet SRW HX 12	A	A	A	A
F-2	1-CV-5211	Saltwater Outlet SRW HX 12A	A	A	A	A
F-2	1-FIC-5211	Saltwater Outlet SRW HX 12A	A	A	A	A
F-2	1-CV-5212	Saltwater Outlet SRW HX 12B	A	A	A	A
F-2	1-FIC-5212	Saltwater Outlet SRW HX 12B	A	A	A	A
J-2	1-PIC-5157	Saltwater Bypass SRW HX 12	A	A	A	A
J-2	1-CV-5157	Saltwater Bypass SRW HX 12	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
J-2	1-I/P-5157	Saltwater Bypass SRW HX 12	A	A	A	A
F-2	1-CV-5153	Saltwater Outlet SRW HX 12	A	A	A	A
F-2	1-SV-5153	Saltwater Outlet SRW HX 12	A	A	A	A
F-2	1-CV-1645	SRW to EDG 1B	A	A	A	A
F-2	1-SV-1645	SRW to EDG 1B	A	A	A	A
F-2	1-CV-1646	SRW Outlet EDG 1B	A	A	A	A
F-2	1-SV-1646	SRW Outlet EDG 1B	A	A	A	A
F-5	1-CV-1588	SRW Inlet EDG 1B	A	A	A	A
F-5	1-SV-1588	SRW Inlet EDG 1B	A	A	A	A
K-0	1-SV-10241	EDG 1A1 Starting Air	A	A	A	A
K-0	1-SV-10242	EDG 1A1 Starting Air	A	A	A	A
K-0	1-SV-10271	EDG 1A2 Starting Air	A	A	A	A
K-0	1-SV-10272	EDG 1A2 Starting Air	A	A	A	A
F-5	0-SV-4834	EDG 1B Starting Air	A	A	A	A
F-5	1-SV-4835	EDG 1B Starting Air	A	A	A	A
F-2	1-CV-3824	CC HX 11 Outlet[(a)]	A	A	A	A
F-2	1-SV-3824	CC HX 11 Outlet[(a)]	A	A	A	A
F-2	1-CV-3826	CC HX 12 Outlet[(a)]	A	A	A	A
F-2	1-SV-3826	CC HX 12 Outlet[(a)]	A	A	A	A
F-1	1-CV-3828	CC Outlet Shutdown HX 11[(a)]	A	A	A	A
F-1	1-SV-3828	CC Outlet Shutdown HX 11[(a)]	A	A	A	A
F-1	1-CV-3830	CC Outlet Shutdown HX 12[(a)]	A	A	A	A
F-1	1-CV-3830	CC Outlet Shutdown HX 12[(a)]	A	A	A	A
F-2	1-LIT-206	Boric Acid Storage Tank 11 Level[(a)]	A	A	A	A
J-1	1-LIA-206	Boric Acid Storage Tank 11 Level[(a)]	A	A	A	A
F-2	1-LIT-208	Boric Acid Storage Tank 12 Level[(a)]	A	A	A	A
J-1	1-LIA-208	Boric Acid Storage Tank 12 Level[(a)]	A	A	A	A
H-1	1-PT-212	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
J-1	1-PIA-212	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
J-1	1ZL224XR,G	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
J-1	1ZL224YR,G	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
J-1	1ZL224ZR,G	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
J-1	1ZL224ZAR,G	Charging Pumps Discharge Header Flow[(a)]	A	A	A	A
F-2	1-MOV-514	Boric Acid Pumps to Chg Pump Suction	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
F-1	1-MOV-508	Boric Acid Tank 12 to Chg Pump Suction	A	A	A	A
F-1	1-MOV-509	Boric Acid Tank 11 to Chg Pump Suction	A	A	A	A
L-1	1-CV-515	Letdown Line Isolation	A	A	A	A
L-1	1-SV-515	Letdown Line Isolation	A	A	A	A
L-1	1-CV-516	Letdown Line Isolation	A	A	A	A
L-1	1-SV-516	Letdown Line Isolation	A	A	A	A
F-1	1-PT-302X	Low Pressure Safety Injection (LPSI) Pump 11 Discharge	A	A	A	A
J-1	1-PI-302X	LPSI Pump 11 Discharge	A	A	A	A
F-1	1-PT-302Y	LPSI Pump 12 Discharge	A	A	A	A
J-1	1-PI-302Y	LPSI Pump 12 Discharge	A	A	A	A
F-1	1-MOV-658	LPSI Pumps Discharge to SDC HX[(a)]	A	A	A	A
F-1	1-PT-303X	SDC HX 11 Inlet[(a)]	A	A	A	A
J-1	1-PT-303X	SDC HX 11 Inlet[(a)]	A	A	A	A
F-1	1-PT-303Y	SDC HX 12 Inlet[(a)]	A	A	A	A
J-1	1-PI-303Y	SDC HX 12 Inlet[(a)]	A	A	A	A
F-1	1-TE-303X	SDC HX 11 Outlet[(a)]	A	A	A	A
J-1	1-TI-303X	SDC HX 11 Outlet[(a)]	A	A	A	A
F-1	1-TE-303Y	SDC HX 12 Outlet[(a)]	A	A	A	A
J-1	1-TI-303Y	SDC HX 12 Outlet[(a)]	A	A	A	A
F-2	1-CV-657	SDC HX Flow[(a)]	A	A	A	A
F-2	1-I/P-657	SDC HX Flow[(a)]	A	A	A	A
J-1	1-HIC-3657	SDC HX Flow[(a)]	A	A	A	A
F-2	1-CV-306	LPSI Flow[(a)]	A	A	A	A
F-2	1-I/P-306	LPSI Flow[(a)]	A	A	A	A
J-1	1-FIC-306	LPSI Flow[(a)]	A	A	A	A
F-4	1-FY-306	LPSI Flow[(a)]	A	A	A	A
F-2	1-FT-306	LPSI Flow[(a)]	A	A	A	A
F-1	1-PT-307	LPSI Header Pressure[(a)]	A	A	A	A
J-1	1-PI-307	LPSI Header Pressure[(a)]	A	A	A	A
F-2	1-TE-351X	LPSI Header Temperature[(a)]	A	A	A	A
F-2	1-TT-351X	LPSI Header Temperature[(a)]	A	A	A	A
F-2	1-FT-312	LPSI Flow to Loop 11A[(a)]	A	A	A	A
J-1	1-FI-312	LPSI Flow to Loop 11A[(a)]	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
F-2	1-FT-322	LPSI Flow to Loop 11B[(a)]	A	A	A	A
J-1	1-FI-322	LPSI Flow to Loop 11B[(a)]	A	A	A	A
H-1	1-FT-332	LPSI Flow to Loop 12A[(a)]	A	A	C	A
J-1	1-FI-332	LPSI Flow to Loop 12A[(a)]	A	A	A	A
H-1	1-FT-342	LPSI Flow to Loop 12B[(a)]	A	A	C	A
J-1	1-FI-342	LPSI Flow to Loop 12B[(a)]	A	A	A	A
H-1	1-MOV-615	LPSI Flow to Loop 11A	A	A	C	A
H-1	1-MOV-625	LPSI Flow to Loop 11B	A	A	C	A
H-2	1-MOV-635	LPSI Flow to Loop 12A	A	A	A	A
H-2	1-MOV-645	LPSI Flow to Loop 12B	A	A	A	A
H-1	1-MOV-651	SDC Return Header[(a)]	A	A	C	A
H-1	1-MOV-652	SDC Return Header[(a)]	A	A	C	A
F-1	1-PT-301X	High Pressure Safety Injection (HPSI) Pump 11 Discharge	A	A	A	A
J-1	1-PI-301X	HPSI Pump 11 Discharge	A	A	A	A
F-1	1-PT-301Y	HPSI Pump 12 Discharge	A	A	A	A
J-1	1-PI-301Y	HPSI Pump 12 Discharge	A	A	A	A
F-1	1-PT-301Z	HPSI Pump 13 Discharge	A	A	A	A
J-1	1-PI-301Z	HPSI Pump 13 Discharge	A	A	A	A
F-1	1-MOV-654	HPSI Header	A	A	A	A
H-1	1-MOV-616	HPSI Flow to Loop 11A	A	A	C	A
H-1	1-MOV-626	HPSI Flow to Loop 11B	A	A	C	A
H-2	1-MOV-636	HPSI Flow to Loop 12A	A	A	A	A
H-2	1-MOV-646	HPSI Flow to Loop 12B	A	A	A	A
F-2	1-FT-311	HPSI Flow to Loop 11A	A	A	A	A
J-1	1-FI-311	HPSI Flow to Loop 11A	A	A	A	A
F-2	1-FT-321	HPSI Flow to Loop 11B	A	A	A	A
J-1	1-FI-321	HPSI Flow to Loop 11B	A	A	A	A
H-1	1-FT-331	HPSI Flow to Loop 12A	A	A	B	A
J-1	1-FI-331	HPSI Flow to Loop 12A	A	A	A	A
H-1	1-FT-341	HPSI Flow to Loop 12B	A	A	B	A
J-1	1-FI-341	HPSI Flow to Loop 12B	A	A	A	A
L-1	1-LT-110X	Pressurizer Level	A	A	A	A
J-1	1-LIC-110X	Pressurizer Level	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
L-1	1-LT-110Y	Pressurizer Level	A	A	A	A
J-1	1-LIC-110Y	Pressurizer Level	A	A	A	A
L-1	1-PT-102A-D	Pressurizer Pressure	A	A	A	A
J-1	1-PI-102A-D	Pressurizer Pressure	A	A	A	A
L-1	1-TT-112HA-HD	Reactor Loop 11A-B Temperature	A	A	A	A
L-1	1-TT-122HA-HD	Reactor Loop 12A-B Temperature	A	A	A	A
L-1	1-LT-1111	Steam Generator (SG) 11 Downcomer Level	A	A	A	A
J-1	1-LR-1111	SG 11 Downcomer Level	A	A	A	A
L-1	1-LT-1121	SG 12 Downcomer Level	A	A	A	A
J-1	1-LR-1121	SG 12 Downcomer Level	A	A	A	A
L-1	1-PT-1013A-D	SG 11 Pressure	A	A	A	A
J-1	1-PI-1013A-D	SG 11 Pressure	A	A	A	A
L-1	1-PT-1023A-D	SG 12 Pressure	A	A	A	A
J-1	1-PI-1023A-D	SG 12 Pressure	A	A	A	A
H-1	1-MOV-617	Auxiliary HPSI Flow to Loop 11A	A	A	C	A
H-1	1-MOV-627	Auxiliary HPSI Flow to Loop 11B	A	A	C	A
H-1	1-MOV-637	Auxiliary HPSI Flow to Loop 12A	A	A	A	A
H-1	1-MOV-647	Auxiliary HPSI Flow to Loop 12B	A	A	A	A
F-4	1-CV-4043	MS Isolation	B	B	A	B
F-4	1-CV-4048	MS Isolation	B	B	A	B
E-1	1-PT-4507	AFW Discharge Header - Steam Train	A	A	A	A
F-2	1-PT-4548	AFW Discharge Header - Motor Train	A	A	A	A
J-1	1-PI-4507	AFW Discharge Header - Steam Train	A	A	A	A
J-1	1-PI-4548	AFW Discharge Header - Steam Train	A	A	A	A
H-0	1-FT-4509B	AFW Flow to SG 11 - Steam Train	A	A	A	A
H-0	1-FT-4510B	AFW Flow to SG 12 - Steam Train	A	A	A	A
F-2	1-FT-4524A	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-2	1-FT-4534A	AFW Flow to SG 12 - Motor Train	A	A	A	A
J-1	1-FIC-4511A	AFW Flow to SG 11 - Steam Train	A	A	A	A
J-1	1-FIC-4512A	AFW Flow to SG 12 - Steam Train	A	A	A	A
J-1	1-FIC-4525A	AFW Flow to SG 11 - Motor Train	A	A	A	A
J-1	1-FIC-4535A	AFW Flow to SG 12 - Motor Train	A	A	A	A
H-1	1-I/P-4511A	AFW Flow to SG 11 - Steam Train	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
H-1	1-I/P-4512A	AFW Flow to SG 12 - Steam Train	A	A	A	A
F-3	1-I/P-4525A	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-3	1-I/P-4535A	AFW Flow to SG 12 - Motor Train	A	A	A	A
H-1	1-CV-4511	AFW Flow to SG 11 - Steam Train	A	A	B	A
H-1	1-CV-4512	AFW Flow to SG 12 - Steam Train	A	A	B	A
F-3	1-CV-4525	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-3	1-CV-4535	AFW Flow to SG 12 - Motor Train	A	A	A	A
F-5	1-I/E-4509B5	AFW Flow to SG 11 - Steam Train	A	A	A	A
F-5	1-FY-4509B	AFW Flow to SG 11 - Steam Train	A	A	A	A
F-5	1-E/I-4509B3	AFW Flow to SG 11 - Steam Train	A	A	A	A
F-4	1-I/E-4524A5	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-4	1-FY-4524A	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-4	1-E/I-4524A3	AFW Flow to SG 11 - Motor Train	A	A	A	A
F-4	1-I/E-4534A5	AFW Flow to SG 12 - Steam Train	A	A	A	A
F-4	1-FY-4534A	AFW Flow to SG 12 - Steam Train	A	A	A	A
F-4	1-E/I-4534A3	AFW Flow to SG 12 - Steam Train	A	A	A	A
F-5	1-I/E-4510B5	AFW Flow to SG 12 - Motor Train	A	A	A	A
F-5	1-FY-4510B	AFW Flow to SG 12 - Motor Train	A	A	A	A
F-5	1-E/I-4510B3	AFW Flow to SG 12 - Motor Train	A	A	A	A
H-1	1-CV-4521	Isolation AFW Flow to SG 11-Steam Train	A	A	B	A
H-1	1-CV-4520	Isolation AFW Flow to SG 11-Steam Train	A	A	B	A
F-2	1-CV-4522	Isolation AFW Flow to SG 11-Motor Train	A	A	A	A
F-3	1-CV-4523	Isolation AFW Flow to SG 11-Motor Train	A	A	A	A
H-1	1-CV-4530	Isolation AFW Flow to SG 12-Steam Train	A	A	B	A
H-1	1-CV-4531	Isolation AFW Flow to SG 12-Steam Train	A	A	B	A
F-2	1-CV-4532	Isolation AFW Flow to SG 12-Motor Train	A	A	A	A
F-3	1-CV-4533	Isolation AFW Flow to SG 12-Motor Train	A	A	A	A
H-1	1-SV-4520	Isolation AFW Flow to SG 11-Steam Train	A	A	B	A
H-1	1-SV-4521	Isolation AFW Flow to SG 11-Steam Train	A	A	B	A
F-2	1-SV-4522	Isolation AFW Flow to SG 11-Motor Train	A	A	A	A
F-3	1-SV-4523	Isolation AFW Flow to SG 11-Motor Train	A	A	A	A
H-1	1-SV-4530	Isolation AFW Flow to SG 12-Steam Train	A	A	B	A
H-1	1-SV-4531	Isolation AFW Flow to SG 12-Steam Train	A	A	B	A
F-2	1-SV-4532	Isolation AFW Flow to SG 12-Motor Train	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	MS	MS TO AFWP TURBINES	STEAM GENERATOR	MAIN FEEDWATER
F-3	1-SV-4533	Isolation AFW Flow to SG 12-Motor Train	A	A	A	A
F-4	1-CV-4070	MS Header 11 to AFW Turbines	B	B	A	B
F-4	1-CV-4071	MS Header 12 to AFW Turbines	B	B	A	B
H-1	1-SV-4070	MS Header 11 to AFW Turbines	A	A	B	A
H-1	1-SV-4070A	MS Header 11 to AFW Turbines	A	A	B	A
H-1	1-SV-4071	MS Header 12 to AFW Turbines	A	A	B	A
H-1	1-SV-4071A	MS Header 12 to AFW Turbines	A	A	B	A
F-4	1-CV-4070A	MS Header 11 to AFW Turbines	B	B	A	B
F-4	1-CV-4071A	MS Header 12 to AFW Turbines	B	B	A	B
F-2	1-PCV-4510	AFW Air Accumulator 11A	A	A	A	A
F-2	1-PCV-4520	AFW Air Accumulator 11B	A	A	A	A
L-1	1-LT-1114A-D	SG Wide Range Level	A	A	A	A
L-1	1-LT-1124A-D	SG Wide Range Level	A	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
F-2	1-CV-5160	Saltwater Inlet CC HX 11	-	-	A	A	A
F-2	1-SV-5160	Saltwater Inlet CC HX 11	-	-	A	A	A
F-2	1-CV-5206	Saltwater Outlet CC HX 11	-	-	A	A	A
F-2	1-SV-5206	Saltwater Outlet CC HX 11	-	-	A	A	A
F-2	1-SV-5206A	Saltwater Outlet CC HX 11	-	-	A	A	A
F-2	1-I/P-5206	Saltwater Outlet CC HX 11	-	-	A	A	A
J-1	1-HIC-5206	Saltwater Outlet CC HX 11	-	-	A	A	A
F-2	1-CV-5162	Saltwater Inlet CC HX 12	-	-	A	A	A
F-2	1-SV-5162	Saltwater Inlet CC HX 12	-	-	A	A	A
F-2	1-CV-5208	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-SV-5208	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-SV-5208A	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-I/P-5208	Saltwater Outlet CC HX	-	-	A	A	A
J-1	1-HIC-5208	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-CV-5163	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-SV-5163	Saltwater Outlet CC HX 12	-	-	A	A	A
F-2	1-CV-5150	Saltwater Inlet SRW HX 11	-	-	A	A	A
F-2	1-SV-5150	Saltwater Inlet SRW HX 11	-	-	A	A	A
F-2	1-CV-5209	Saltwater Outlet SRW HX 11A	-	-	A	A	A
F-2	1-FIC-5209	Saltwater Outlet SRW HX 11A	-	-	A	A	A
F-2	1-CV-5210	Saltwater Outlet SRW HX 11B	-	-	A	A	A
F-2	1-FIC-5210	Saltwater Outlet SRW HX 11B	-	-	A	A	A
J-2	1-PIC-5154	Saltwater Bypass SRW HX 11	-	-	A	A	A
F-2	1-CV-5154	Saltwater Bypass SRW HX 11	-	-	A	A	A
F-2	1-I/P-5154	Saltwater Bypass SRW HX 11	-	-	A	A	A
F-2	1-CV-5152	Saltwater Inlet SRW HX 12	-	-	A	A	A
F-2	1-SV-5152	Saltwater Inlet SRW HX 12	-	-	A	A	A
F-2	1-CV-5211	Saltwater Outlet SRW HX 12A	-	-	A	A	A
F-2	1-SV-5211	Saltwater Outlet SRW HX 12A	-	-	A	A	A
F-2	1-FIC-5211	Saltwater Outlet SRW HX 12A	-	-	A	A	A
F-2	1-CV-5212	Saltwater Outlet SRW HX 12	-	-	A	A	A
F-2	1-FIC-5212	Saltwater Outlet SRW HX 12B	-	-	A	A	A
J-2	1-PIC-5157	Saltwater Bypass SRW HX 12	-	-	A	A	A
F-2	1-CV-5157	Saltwater Bypass SRW HX 12	-	-	A	A	A
F-2	1-I/P-5157	Saltwater Bypass SRW HX 12	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
F-2	1-CV-5153	Saltwater Outlet SRW HX 12	-	-	A	A	A
F-2	1-SV-5153	Saltwater Outlet SRW HX 12	-	-	A	A	A
F-2	1-CV-1645	SRW to EDG 1B	-	-	A	A	A
F-2	1-SV-1645	SRW to EDG 1B	-	-	A	A	A
F-2	1-CV-1646	SRW Outlet EDG 1B	-	-	A	A	A
F-2	1-SV-1646	SRW Outlet EDG 1B	-	-	A	A	A
F-5	1-CV-1588	SRW Inlet EDG 1B	-	-	A	A	A
F-5	1-SV-1588	SRW Inlet EDG 1B	-	-	A	A	A
K-0	1-SV-10241	EDG 1A1 Starting Air	-	-	A	A	A
K-0	1-SV-10242	EDG 1A1 Starting Air	-	-	A	A	A
K-0	1-SV-10271	EDG 1A2 Starting Air	-	-	A	A	A
K-0	1-SV-10272	EDG 1A2 Starting Air	-	-	A	A	A
F-5	0-SV-4834	EDG 1B Starting Air	-	-	A	A	A
F-5	1-SV-4835	EDG 1B Starting Air	-	-	A	A	A
F-2	1-CV-3824	CC HX 11 Outlet[a]	-	-	A	A	A
F-2	1-SV-3824	CC HX 11 Outlet[a]	-	-	A	A	A
F-2	1-CV-3826	CC HX 12 Outlet[a]	-	-	A	A	A
F-2	1-SV-3826	CC HX 12 Outlet[a]	-	-	A	A	A
F-1	1-CV-3828	CC Outlet Shutdown HX 11[a]	-	-	A	A	A
F-1	1-SV-3828	CC Outlet Shutdown HX 11[a]	-	-	A	A	A
F-1	1-CV-3830	CC Outlet Shutdown HX 12[a]	-	-	A	A	A
F-1	1-CV-3830	CC Outlet Shutdown HX 12[a]	-	-	A	A	A
F-2	1-LIT-206	Boric Acid Storage Tank 11 Level[a]	-	-	A	A	A
J-1	1-LIA-206	Boric Acid Storage Tank 11 Level[a]	-	-	A	A	A
F-2	1-LIT-208	Boric Acid Storage Tank 12 Level[a]	-	-	A	A	A
J-1	1-LIA-208	Boric Acid Storage Tank 12 Level[a]	-	-	A	A	A
H-1	1-PT-212	Charging Pumps Discharge Header Flow[a]	-	-	C	C	A
J-1	1-PIA-212	Charging Pumps Discharge Header Flow[a]	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
J-1	1ZL224XR,G	Charging Pumps Discharge Header Flow[a]	-	-	A	A	A
J-1	1ZL224YR,G	Charging Pumps Discharge Header Flow[a]	-	-	A	A	A
J-1	1ZL224ZR,G	Charging Pumps Discharge Header Flow[a]	-	-	A	A	A
J-1	1ZL224ZAR,G	Charging Pumps Discharge Header Flow[a]	-	-	A	A	A
F-2	1-MOV-514	Boric Acid Pumps to Chg Pump Suction	-	-	A	A	A
F-1	1-MOV-508	Boric Acid Tank 12 to Chg Pump Suction	-	-	A	A	A
F-1	1-MOV-509	Boric Acid Tank 11 to Chg Pump Suction	-	-	A	A	A
L-1	1-CV-515	Letdown Line Isolation	-	-	A	A	A
L-1	1-SV-515	Letdown Line Isolation	-	-	A	A	A
L-1	1-CV-516	Letdown Line Isolation	-	-	A	A	A
L-1	1-SV-516	Letdown Line Isolation	-	-	A	A	A
F-1	1-PT-302X	LPSI Pump 11 Discharge	-	-	A	A	A
J-1	1-PI-302X	LPSI Pump 11 Discharge	-	-	A	A	A
F-1	1-PT-302Y	LPSI Pump 12 Discharge	-	-	A	A	A
J-1	1-PI-302Y	LPSI Pump 12 Discharge	-	-	A	A	A
F-1	1-MOV-658	LPSI Pumps Discharge to Shutdown HX[a]	-	-	A	A	A
F-1	1-PT-303X	SDC HX 11 Inlet[a]	-	-	A	A	A
J-1	1-PT-303X	SDC HX 11 Inlet[a]	-	-	A	A	A
F-1	1-PT-303Y	SDC HX 12 Inlet[a]	-	-	A	A	A
J-1	1-PI-303Y	SDC HX 12 Inlet[a]	-	-	A	A	A
F-1	1-TE-303X	SDC HX 11 Outlet[a]	-	-	A	A	A
J-1	1-TI-303X	SDC HX 11 Outlet[a]	-	-	A	A	A
F-1	1-TE-303Y	SDC HX 12 Outlet[a]	-	-	A	A	A
J-1	1-TI-303Y	SDC HX 12 Outlet[a]	-	-	A	A	A
F-2	1-CV-657	SDC HX Flow[a]	-	-	A	A	A
F-2	1-I/P-657	SDC HX Flow[a]	-	-	A	A	A
J-1	1-HIC-3657	SDC HX Flow[a]	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
F-2	1-CV-306	LPSI Flow[a]	-	-	A	A	A
F-2	1-I/P-306	LPSI Flow[a]	-	-	A	A	A
J-1	1-FIC-306	LPSI Flow[a]	-	-	A	A	A
F-4	1-FY-306	LPSI Flow[a]	-	-	A	A	A
F-2	1-FT-306	LPSI Flow[a]	-	-	A	A	A
F-1	1-PT-307	LPSI Header Pressure[a]	-	-	A	A	A
J-1	1-PI-307	LPSI Header Pressure[a]	-	-	A	A	A
F-2	1-TE-351X	LPSI Header Temperature[a]	-	-	A	A	A
F-2	1-TT-351X	LPSI Header Temperature[a]	-	-	A	A	A
F-2	1-FT-312	LPSI Flow to Loop 11A[a]	-	-	A	A	A
J-1	1-FI-312	LPSI Flow to Loop 11A[a]	-	-	A	A	A
F-2	1-FT-322	LPSI Flow to Loop 11B[a]	-	-	A	A	A
J-1	1-FI-322	LPSI Flow to Loop 11B[a]	-	-	A	A	A
H-1	1-FT-332	LPSI Flow to Loop 12A[a]	-	-	C	C	A
J-1	1-FI-332	LPSI Flow to Loop 12A[a]	-	-	A	A	A
H-1	1-FT-342	LPSI Flow to Loop 12B[a]	-	-	C	C	A
J-1	1-FI-342	LPSI Flow to Loop 12B[a]	-	-	A	A	A
H-1	1-MOV-615	LPSI Flow to Loop 11A	-	-	A	B	A
H-1	1-MOV-625	LPSI Flow to Loop 11B	-	-	A	B	A
H-2	1-MOV-635	LPSI Flow to Loop 12A	-	-	B	B	A
H-2	1-MOV-645	LPSI Flow to Loop 12B	-	-	B	B	A
H-1	1-MOV-651	SDC Return Header[a]	-	-	A	B	A
H-1	1-MOV-652	SDC Return Header[a]	-	-	A	A	A
F-1	1-PT-301X	HPSI Pump 11 Discharge	-	-	A	A	A
J-1	1-PI-301X	HPSI Pump 11 Discharge	-	-	A	A	A
F-1	1-PT-301Y	HPSI Pump 12 Discharge	-	-	A	A	A
J-1	1-PI-301Y	HPSI Pump 12 Discharge	-	-	A	A	A
F-1	1-PT-301Z	HPSI Pump 13 Discharge	-	-	A	A	A
J-1	1-PI-301Z	HPSI Pump 13 Discharge	-	-	A	A	A
F-1	1-MOV-654	HPSI Header	-	-	A	A	A
H-1	1-MOV-616	HPSI Flow to Loop 11A	-	-	A	C	A
H-1	1-MOV-626	HPSI Flow to Loop 11B	-	-	A	C	A
H-2	1-MOV-636	HPSI Flow to Loop 12A	-	-	C	C	A
H-2	1-MOV-646	HPSI Flow to Loop 12B	-	-	C	C	A
F-2	1-FT-311	HPSI Flow to Loop 11A	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
J-1	1-FI-311	HPSI Flow to Loop 11A	-	-	A	A	A
F-2	1-FT-321	HPSI Flow to Loop 11B	-	-	A	A	A
J-1	1-FI-321	HPSI Flow to Loop 11B	-	-	A	A	A
H-1	1-FT-331	HPSI Flow to Loop 12A	-	-	C	C	A
J-1	1-FI-331	HPSI Flow to Loop 12A	-	-	A	A	A
H-1	1-FT-341	HPSI Flow to Loop 12B	-	-	C	C	A
J-1	1-FI-341	HPSI Flow to Loop 12B	-	-	A	A	A
L-1	1-LT-110X	Pressurizer Level	-	-	A	A	A
J-1	1-LIC-110X	Pressurizer Level	-	-	A	A	A
L-1	1-LT-110Y	Pressurizer Level	-	-	A	A	A
J-1	1-LIC-110Y	Pressurizer Level	-	-	A	A	A
L-1	1-PT-102A-D	Pressurizer Pressure	-	-	A	A	A
J-1	1-PI-102A-D	Pressurizer Pressure	-	-	A	A	A
L-1	1-TT-112HA-HD	Reactor Loop 11A-B Temperature	-	-	A	A	A
L-1	1-TT-122HA-HD	Reactor Loop 12A-B Temperature	-	-	A	A	A
L-1	1-LT-1111	SG 11 Downcomer Level	-	-	A	A	A
J-1	1-LR-1111	SG 11 Downcomer Level	-	-	A	A	A
L-1	1-LT-1121	SG 12 Downcomer Level	-	-	A	A	A
J-1	1-LR-1121	SG 12 Downcomer Level	-	-	A	A	A
L-1	1-PT-1013A-D	SG 11 Pressure	-	-	A	A	A
J-1	1-PI-1013A-D	SG 11 Pressure	-	-	A	A	A
L-1	1-PT-1023A-D	SG 12 Pressure	-	-	A	A	A
J-1	1-PI-1023A-D	SG 12 Pressure	-	-	A	A	A
H-1	1-MOV-617	Auxiliary HPSI Flow to Loop 11A	-	-	A	C	A
H-1	1-MOV-627	Auxiliary HPSI Flow to Loop 11B	-	-	A	C	A
H-1	1-MOV-637	Auxiliary HPSI Flow to Loop 12A	-	-	C	C	A
H-1	1-MOV-647	Auxiliary HPSI Flow to Loop 12B	-	-	C	C	A
F-4	1-CV-4043	MS Isolation	-	-	A	A	A
F-4	1-CV-4048	MS Isolation	-	-	A	A	A
E-1	1-PT-4507	AFW Discharge Header - Steam Train	-	-	A	A	A
F-2	1-PT-4548	AFW Discharge Header - Motor Train	-	-	A	A	A
J-1	1-PI-4507	AFW Discharge Header - Steam Train	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
J-1	1-PI-4548	AFW Discharge Header - Steam Train	-	-	A	A	A
H-0	1-FT-4509B	AFW Flow to SG 11 - Steam Train	-	-	A	B	A
H-0	1-FT-4510B	AFW Flow to SG 12 - Steam Train	-	-	A	B	A
F-2	1-FT-4524A	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-2	1-FT-4534A	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
J-1	1-FIC-4511A	AFW Flow to SG 11 - Steam Train	-	-	A	A	A
J-1	1-FIC-4512A	AFW Flow to SG 12 - Steam Train	-	-	A	A	A
J-1	1-FIC-4525A	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
J-1	1-FIC-4535A	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
H-1	1-I/P-4511A	AFW Flow to SG 11 - Steam Train	-	-	A	B	A
H-1	1-I/P-4512A	AFW Flow to SG 12 - Steam Train	-	-	A	B	A
F-3	1-I/P-4525A	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-3	1-I/P-4535A	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
H-1	1-CV-4511	AFW Flow to SG 11 - Steam Train	-	-	A	B	A
H-1	1-CV-4512	AFW Flow to SG 12 - Steam Train	-	-	A	B	A
F-3	1-CV-4525	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-3	1-CV-4535	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
F-5	1-I/E-4509B5	AFW Flow to SG 11 - Steam Train	-	-	A	A	A
F-5	1-FY-4509B	AFW Flow to SG 11 - Steam Train	-	-	A	A	A
F-5	1-E/I-4509B3	AFW Flow to SG 11 - Steam Train	-	-	A	A	A
F-4	1-I/E-4524A5	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-4	1-FY-4524A	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-4	1-E/I-4524A3	AFW Flow to SG 11 - Motor Train	-	-	A	A	A
F-4	1-I/E-4534A5	AFW Flow to SG 12 - Steam Train	-	-	A	A	A
F-4	1-FY-4534A	AFW Flow to SG 12 - Steam Train	-	-	A	A	A
F-4	1-E/I-4534A3	AFW Flow to SG 12 - Steam Train	-	-	A	A	A
F-5	1-I/E-4510B5	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
F-5	1-FY-4510B	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
F-5	1-E/I-4510B3	AFW Flow to SG 12 - Motor Train	-	-	A	A	A
H-1	1-CV-4521	Isolation AFW Flow to SG 11- Steam Train	-	-	A	B	A
H-1	1-CV-4520	Isolation AFW Flow to SG 11- Steam Train	-	-	A	B	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
F-2	1-CV-4522	Isolation AFW Flow to SG 11-Motor Train	-	-	A	A	A
F-3	1-CV-4523	Isolation AFW Flow to SG 11-Motor Train	-	-	A	A	A
H-1	1-CV-4530	Isolation AFW Flow to SG 12-Steam Train	-	-	A	B	A
H-1	1-CV-4531	Isolation AFW Flow to SG 12-Steam Train	-	-	A	B	A
F-2	1-CV-4532	Isolation AFW Flow to SG 12-Motor Train	-	-	A	A	A
F-3	1-CV-4533	Isolation AFW Flow to SG 12-Motor Train	-	-	A	A	A
H-1	1-SV-4520	Isolation AFW Flow to SG 11-Steam Train	-	-	A	B	A
H-1	1-SV-4521	Isolation AFW Flow to SG 11-Steam Train	-	-	A	B	A
F-2	1-SV-4522	Isolation AFW Flow to SG 11-Motor Train	-	-	A	A	A
F-3	1-SV-4523	Isolation AFW Flow to SG 11-Motor Train	-	-	A	A	A
H-1	1-SV-4530	Isolation AFW Flow to SG 12-Steam Train	-	-	A	B	A
H-1	1-SV-4531	Isolation AFW Flow to SG 12-Steam Train	-	-	A	B	A
F-2	1-SV-4532	Isolation AFW Flow to SG 12-Motor Train	-	-	A	A	A
F-3	1-SV-4533	Isolation AFW Flow to SG 12-Motor Train	-	-	A	A	A
F-4	1-CV-4070	MS Header 11 to AFW Turbines	-	-	A	A	A
F-4	1-CV-4071	MS Header 12 to AFW Turbines	-	-	A	A	A
H-1	1-SV-4070	MS Header 11 to AFW Turbines	-	-	A	B	A
H-1	1-SV-4070A	MS Header 11 to AFW Turbines	-	-	A	B	A
H-1	1-SV-4071	MS Header 12 to AFW Turbines		-	A	B	A
H-1	1-SV-4071A	MS Header 12 to AFW Turbines	-	-	A	B	A
F-4	1-CV-4070A	MS Header 11 to AFW Turbines	-	-	A	A	A

TABLE 10A-5

INSTRUMENTATION REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

LOCATION	INSTRUMENT/ VALVE NO.	SERVICE	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	AUXILIARY STEAM
F-4	1-CV-4071A	MS Header 12 to AFW Turbines	-	-	A	A	A
F-2	1-PCV-4510	AFW Air Accumulator 11A	-	-	A	A	A
F-2	1-PCV-4520	AFW Air Accumulator #11B	-	-	A	A	A
L-1	1-LT-1114A-D	SG Wide Range Level	-	-	A	A	A
L-1	1-LT-1124A-D	SG Wide Range Level	-	-	A	A	A

LEGEND:

A. Located outside those areas which experience a steam environment or jet impingement.
B. Qualified to be operated in a steam environment or not adversely affected by jet impingement.
C. Not required for a break in this system.
E-1 Turbine Building EL 12'0
E-2 Turbine Building EL 27'0
E-3 Turbine Building EL 45'0
F-1 Auxiliary Building EL (-)15'0 & (-)10'0
F-2 Auxiliary Building EL 3'0
F-3 Auxiliary Building EL 14'9
F-4 Auxiliary Building EL 27'0
F-5 Auxiliary Building EL 45'0
F-6 Auxiliary Building EL 69'0
G-1 Intake Structure EL 3'0
H-1 Penetration Room EL 27'0
H-2 Penetration Room EL 45'0
J-1 Control Room
L-1 Containment
H-0 Penetration Room EL 5'0"
K-0 Diesel Generator Building No. 1A

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[a] These items are not required to function immediately, but will be used during subsequent operation to achieve a cold shutdown condition.

[b] Break not credible in this system.

[c] Handswitches for instrumentation and associated equipment are not located in the Control Room or in areas that are not subject to a steam environment.

[d] There are no environmental considerations which requires analysis: refer to respective high energy analysis discussion.

TABLE 10A-6

MECHANICAL AND ELECTRICAL EQUIPMENT REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

HIGH ENERGY SYSTEMS

LOCATION	MECHANICAL	MS	MS TO AFWP TURBINES	STEAM GENERATOR BLOWDOWN	MAIN FEEDWATER
F-1	HPSI Pumps	A	A	C	C
E-1	AFWPs (steam-driven)	A	A	A	A
F-5	EDG & Auxiliaries	A	A	A	A
F-2	SRW Pumps	A	A	A	A
F-3	SRW HX	A	A	A	A
F-3	Saltwater Strainers	A	A	A	A
G-1	Saltwater Pumps	A	A	A	A
F-6	Control Room Heating, Ventilation & Air Conditioning	A	A	A	A
F-1	LPSI Pumps[a]	A	A	A	A
F-1	SDC HX[a]	A	A	A	A
F-2	CC Pumps[a]	A	A	A	A
F-2	CC HX[a]	A	A	A	A
F-2	Boric Acid Storage Tanks[a]	A	A	A	A
F-2	Boric Acid Pumps[a]	A	A	A	A
F-1	Charging Pumps	A	A	A	A
F-4	Spent Fuel Pool Cooling Pumps	A	A	A	A
F-4	Spent Fuel Pool Cooling HX[a]	A	A	A	A
F-2	Penetration Room Ventilation[a]	C	C	A	C
F-4	Reactor Trip Breakers	A	A	A	A
F-4	Batteries	A	A	A	A
F-4	480 Volt Bus &	A	A	A	A
	4160 Volt Bus	A	A	A	A
F-5	Switchgear Room	A	A	A	A
F-4	Cable Spreading Room	A	A	A	A
F-5	Control Room	A	A	A	A
F-5 & F-6	Motor Control Center for above equipment	A	A	A	A
F-4	Actuators for MSIVs	B	B	A	B
F-2	AFWP (motor-driven)	A	A	A	A

TABLE 10A-6

MECHANICAL AND ELECTRICAL EQUIPMENT REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

HIGH ENERGY SYSTEMS

LOCATION	MECHANICAL	AFW[b]	SHUTDOWN COOLING[b]	CVCS	SAMPLING	MAIN FEEDWATER
F-1	HPSI Pumps	-	-	A	C	C
E-1	AFWPs (steam-driven)	-	-	A	A	A
F-5	EDG & Auxiliaries	-	-	A	A	A
F-2	SRW Pumps	-	-	A	A	A
F-3	SRW HX	-	-	A	A	A
F-3	Saltwater Strainers	-	-	A	A	A
G-1	Saltwater Pumps	-	-	A	A	A
F-6	Control Room Heating, Ventilation & Air Conditioning	-	-	A	A	A
F-1	LPSI Pumps[a]	-	-	A	A	A
F-1	SDC HX[a]	-	-	A	A	A
F-2	CC Pumps[a]	-	-	A	A	A
F-2	CC HX[a]	-	-	A	A	A
F-2	Boric Acid Storage Tanks[a]	-	-	A	A	A
F-2	Boric Acid Pumps[a]	-	-	A	A	A
F-1	Charging Pumps	-	-	A	A	A
F-4	Spent Fuel Pool Cooling Pumps	-	-	A	A	A
F-4	Spent Fuel Pool Cooling HX[a]	-	-	A	A	A
F-2	Penetration Room Ventilation[a]	-	-	C	A	C
F-4	Reactor Trip Breakers	-	-	A	A	A
F-4	Batteries	-	-	A	A	A
F-4	480 Volt Bus &	-	-	A	A	A
	4160 Volt Bus	-	-	A	A	A
F-5	Switchgear Room	-	-	A	A	A
F-4	Cable Spreading Room	-	-	A	A	A
F-5	Control Room	-	-	A	A	A
F-5 & F-6	Motor Control Center for above equipment	-	-	A	A	A
F-4	Actuators for MSIVs	-	-	A	A	A
F-2	AFWP (motor-driven)	-	-	A	A	A

TABLE 10A-6

MECHANICAL AND ELECTRICAL EQUIPMENT REQUIRED TO PLACE THE PLANT IN A SAFE SHUTDOWN CONDITION AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION

HIGH ENERGY SYSTEMS

LEGEND:

A. Located outside those areas which experience a steam environment.
B. Qualified to be operated in a steam environment.
C. Not required for a break in this system.
E-1 Turbine Building, EL 12'0"
E-2 Turbine Building, EL 27'0"
E-3 Turbine Building, EL 45'0"
F-1 Auxiliary Building, EL (-)15'0" & (-)10'0"
F-2 Auxiliary Building, EL 3'0"
F-3 Auxiliary Building, EL 14'9"
F-4 Auxiliary Building, EL 27'0"
F-5 Auxiliary Building, EL 45'0"
F-6 Auxiliary Building, EL 69'0"
G-1 Intake Structure, EL 3'0"

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[a] These items are not required to function immediately, but will be used during subsequent operation to achieve a cold shutdown condition.

[b] Break not credible in this system.

10A.4.1 MAIN FEEDWATER SYSTEM (INSIDE AUXILIARY BUILDING)

The objective of the MFW System is to provide a dependable supply of feedwater to the steam generators. Two turbine-driven pumps supply the required feedwater flow rate to the steam generators to match the steam flow demand by the plant turbine generator and auxiliaries. The driving steam for high load operation is hot reheat, whereas for low load, an automatic changeover to MS takes place. Auxiliary steam from the auxiliary boiler is also readily available.

In the event of a postulated break in the Auxiliary Building, the blowdown from either steam generator into the Auxiliary Building is prevented due to a check valve provided on each MFW line inside the Containment Structure. The blowdown from the pump side into the Auxiliary and Turbine Buildings is handled by the drain systems (Section 10A.4.1.15).

The MFW line is completely sleeved with a 22" OD SCH 80 pipe below the floor Elevation 27'0" to retain jet impingement forces, reactive forces and pressurization due to a full break or a critical crack.

The feedwater pipe sleeve (Figure 10A.4-7) is also designed to accept thermal growth of the feedwater line and to withstand seismic loadings.

The pressure and temperature in the MFW lines at the discharge of the steam generator feedwater pumps will be 1150 psia and 345°F, respectively, at valves wide open.

10A.4.1.1 PIPE WHIP

The MFW System normally operates at a pressure above 275 psig and 200°F, and therefore, requires restraints to limit damage from a pipe whip following a longitudinal or circumferential break.

10A.4.1.2 CRITERIA FOR LOCATING PIPE BREAKS

The postulated break locations are chosen in accordance with the criteria stated in Section 10A.1.2. The circumferential or longitudinal stresses derived on an elastically-calculated basis, under the loadings associated with a seismic event and operational plant conditions did not exceed 0.8 (S_h + S_A); nor did the expansion stresses exceed 0.8 S_A. Therefore, pipe breaks were postulated to occur only at the terminal ends and the two highest intermediate stress locations (Table 10A-9).

10A.4.1.3 CRITERIA FOR PIPE BREAK ORIENTATION

Criteria for pipe break orientation is presented in Section 10A.1.3.

10A.4.1.4 SUMMARY OF DYNAMIC ANALYSIS

10A.4.1.4.1 Location and Number of Design Basis Breaks

The locations and number of design basis breaks are chosen in accordance with the criteria discussed in Section 10A.1.3. A further discussion is given in Section 10A.1.4.1.

10A.4.1.4.2 The Postulated Rupture Orientation

The longitudinal break is assumed to be parallel to the pipe axis and oriented at any point around the pipe circumference. The circumferential break is assumed to be perpendicular to the pipe axis. A further discussion of the break area and the dynamic forces is provided in Section 10A.1.4.2.

10A.4.1.4.3 Description of the Forcing Function

The jet impingement force, caused by the momentum change of fluid flowing through the break, is a function of the upstream fluid conditions, fluid enthalpy, source pressure, pipe flow restrictional friction and dimensions.

The method used to compute the jet forces acting upon the pipe are computed using a method outlined in Section 10A.1.4.3.

10A.4.1.4.4 Dynamic Analysis and Mathematical Model

The dynamic analysis method used for the MFW pipe whip restraint design is similar to the one used for the MS line pipe whip restraint design. This method is explained in Section 10A.1.4.4 and the mathematical model is shown in Figure 10A.1-5.

10A.4.1.4.5 Unrestrained Motion of the Ruptured Line

Since the MFW line is restrained at the postulated break location and additional restraints are provided to preclude axial movement within the sleeve, no damage can occur to structures, systems and components important to plant safety due to a MFW line break.

10A.4.1.5 PROTECTIVE MEASURES

10A.4.1.5.1 Pipe Whip Restraints

Pipe whip restraints are provided to prevent the pipe whip impact, due to a full instantaneous break, on structures, systems and components important to the plant safety. A further discussion of the pipe whip restraint is provided in Section 10A.1.5.1.

10A.4.1.5.2 Protective Measures - MFW Pipe Sleeve

The MFW line below floor Elevation 27'0" is sleeved completely in order to retain jet impingement forces and prevent pressurization of the Auxiliary Building below Elevation 27'0".

A detailed description of design criteria for sleeves is given in Section 10A.1.5.

The feedwater pipe sleeve is also designed to accept thermal growth of the feedwater line and to withstand seismic loadings. The sleeve is supported by hangers and anchors, as required, through the Auxiliary Building.

10A.4.1.5.3 Separation Provisions
10A.4.1.5.4

The MFW lines run parallel to each other approximately 8'0" apart. Separation of redundant features in the MFW lines is accomplished by a combination of sleeving and properly placed restraints.

10A.4.1.5.5 Description of the Pipe Whip Restraint

The pipe whipping restraints are provided at the postulated break locations and at other critical locations, such as elbows, to control the pipe whip impact and axial movement due to a full break at the postulated break locations.

Figure 10A.4-2 shows the location of restraints for the MFW line.

A description of a pipe whip restraint is given in Section 10A.1.5.5.

10A.4.1.6 EVALUATION OF CATEGORY I STRUCTURES

10A.4.1.6.1 Method of Evaluating Stresses

The Auxiliary Building, a Category I reinforced concrete structure, was evaluated for a structural adequacy following a postulated rupture using ultimate strength design method as outlined in "Building Code Requirements for Reinforced Concrete" - American Concrete Institute (ACI) 318-63.

10A.4.1.6.2 Allowable Design Stresses

American Concrete Institute standard "Building Code Requirements for Reinforced Concrete" – ACI 318-63 was used for computing allowable stresses for concrete and reinforcing steel. Allowable stresses for concrete is taken at 85% of the ultimate strength and allowable stresses for the reinforcing steel is taken 85% of the yield strength.

10A.4.1.6.3 Load Factors and Load Combinations

Load factors and load combinations used in the evaluation of the Auxiliary Building are given in Section 10A.1.7.

10A.4.1.6.4 Stresses in the Category I Structure

The MFW line is completely sleeved with a 22" OD pipe below the floor Elevation 27'0" to retain jet impingement forces and pressurization resulting from postulated breaks in the MFW line below Elevation 27'0". The MFW line above floor Evaluation 27'0" is also sleeved at the postulated break locations.

The pressure inside the sleeve, above or below Elevation 27'0", will be released from the open ends of the sleeve and into the Turbine Building. A new vent stack has been provided to vent the MS Penetration Room. Thus, the pressure due to a postulated break in the MFW line inside the room will not be greater than 2.6 psi.

Table 10A-3 shows the concrete and steel stresses due to the pressurization of 2.6 psi resulting from a postulated pipe rupture.

Table 10A-2 shows the concrete and steel stresses due to a jet impingement force of 10 kips plus 1 psi compartment pressurization resulting from a critical crack in the MS line. These stresses are in various structural components in the MS Penetration Room.

The magnitude of jet impingement force and compartment pressurization, in the MS Penetration Room at Elevation 27'0", resulting from a critical crack in the MFW line will be less than 6 kips and 1 psi, respectively. Thus, the stresses in the MS Penetration Room due to a critical crack in the MFW line will be less than those listed in Table 10A-2.

It can be seen from Tables 10A-2 and 10A-4 that the stresses in the Category I structure resulting from a postulated break or critical crack in the MFW line will be well within the allowable stresses.

10A.4.1.6.5 <u>Erosion of Concrete from Jet Impingement Forces</u>

The MFW line is sleeved completely below Elevation 27'0". It is also sleeved above floor Elevation 27'0" at the postulated break location. Thus, the jet impingement forces resulting from a postulated break will be retained inside the sleeve. The jet impingement force resulting from a critical crack will be less than 6 kips. This force is considerably less than the jet impingement force resulting from a critical crack in the MS line to cause any concrete erosion.

A further discussion of concrete erosion due to a jet impingement force is given in Section 10A.1.6.5.

10A.4.1.7 STRUCTURAL DESIGN LOADS

The design loads used to evaluate the adequacy of Category I structures or structural components are discussed in the Section 10A.1.7.

10A.4.1.8 REVERSAL OF LOADS ON THE STRUCTURE

The forces causing reversal of loading due to the postulated accident on the Seismic Category I structures or structural components are:

A. Jet Impingement Force
B. Compartment Pressurization
C. Reaction from Pipe Whip Restraint

10A.4.1.8.1 <u>Reversal of Loading Due to Jet Impingement Forces</u>

The MFW line inside the Auxiliary Building and below Elevation 27'0" is sleeved completely. The postulated full break locations above Elevation 27'0" are also sleeved. The sleeve will resist jet impingement forces due to a pipe break or a critical size crack. The sleeve, therefore, will prevent Seismic Category I structures or structural components below floor Elevation 27'0" from being affected by the jet impingement forces.

The MFW line above Elevation 27'0" is also sleeved where full-break locations are postulated. The jet impingement forces due to a critical size crack break will not be significant enough to cause damage to Category I structure or structural components due to a reversal of loading.

10A.4.1.8.2 <u>Reversal of Loading Due to Compartment Pressurization</u>

The MFW line sleeve at Elevation 5'0" (below Elevation 27'0") will release the pressure due to a postulated break or critical crack at Elevation 27'0", so there will not be any reversal of loading on Category I structures or structural components at Elevation 5'0".

The pressure inside the sleeve, due to a postulated full break above or below Elevation 27'0", will be released above Elevation 27'0" in the area of the MS line compartment. A new vent stack has been provided to vent the MS line compartment. Thus, the pressure in the MS line compartment will be limited by the new vent to an acceptable level and

will not affect the integrity of the Category I structures or structural components.

10A.4.1.8.3 Pipe Whip Restraints

Pipe whip restraints are provided in accordance with the criteria stated in Section 10A.1.5.5. They are supported by the existing structural components. When the restraint loads cannot be sustained by the existing structure or structural components, these loads are transferred to the foundation level using additional supports.

The restraints for the MFW line above floor Elevation 27'0" required additional supports to transfer restraint loads from Elevation 27'0" to the floor Elevation 5'0", a foundation level.

10A.4.1.9 STRUCTURAL EFFECTS OF OPENINGS

A vent stack has been provided to vent the MS line Penetration Room to a compartment pressure below the acceptable level, and is described in Section 10A.1.9.

10A.4.1.10 EFFECTS OF STRUCTURAL FAILURE

There will not be a failure of any structure, including Category II (non-seismic Category I) structures, due to the accident, that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident; and

B. Capability to bring the unit(s) to a cold shutdown condition.

10A.4.1.11 VERIFICATION THAT HIGH ENERGY PIPE RUPTURES WILL NOT AFFECT SAFETY

Since the MFW is completely sleeved below floor Elevation 27'0" in the Auxiliary Building, a rupture of the pipe below this elevation will not affect any safety equipment. In the event a crack break occurs in the feedwater line portion which is not sleeved above Elevation 27'0" in the Auxiliary Building, the only region affected is the MS Penetration Room which is designed so there will be no effect on plant safety. Section 10A.1.11 describes the high energy pipe rupture in this room.

10A.4.1.12 EFFECT ON CONTROL ROOM

A feedwater line rupture will not affect the Control Room since there is no direct or indirect access from the affected area to the Control Room.

10A.4.1.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

A detailed description of environmental qualification of affected equipment above Elevation 27'0" in the Auxiliary Building is given in Section 10A.1.13.

The MFW check valves on the MFW lines inside the Containment are supplied by Rockwell Manufacturing Company. They are 16" nominal size with a rating of 900 lbs. Each valve is cast carbon steel A-216 Gr. WCB or WCC, butt-welded, pressure seal cap or bolted-body joint, welded or integral stellited seat ring, and stellited tilting disc with standard trim for steam or water service at 400°F.

The check valve closure can be verified when the plant is shut down by upstream pressure indicators.

10A.4.1.14 DRAWINGS

Figures 10A.4-2, 10A.4-3, 10A.4-4 and 10A.4-5 show the routing of the MFW piping from the containment penetration through the Auxiliary Building. The complete condensate and feedwater system is shown in Figure 10-4. (Figure 10A.4-1 shows that portion of the system diagram affected by the high energy system criteria.)

10A.4.1.15 FLOODING

In the event a feedwater line ruptures at Elevation 27'0" or within the sleeved area below, approximately three-fourths of the mass released would be a source of flooding.

Flooding of the MS Valve Room will be controlled by means of a lightly constructed (gypsum) wall located at the turbine room end of the steam line tunnel. The gypsum wall design and construction will be based on the collapse of the wall when subjected to a hydraulic pressure of 3' of water.

Minor flooding of the MS valve room at Elevation 27'0" resulting from a crack in the feedwater line will be handled by two 6" diameter drain lines penetrating the tunnel gypsum wall at floor level and gravity draining to the turbine room floor drain system at Elevation 12'0".

Pressure retaining walls and doors enclosing the MS valve room prevent communication of steam or water resulting from a pipe break to other areas of the Auxiliary Building.

10A.4.1.16 QUALITY CONTROL AND INSPECTION

A full description of the quality control and inspection requirements is given in Section 10A.1.16.

10A.4.1.17 LEAK DETECTION

A. Crack break above Elevation 27'0":

   A temperature switch is located in the room to give an indication in the Control Room if a crack break should occur in the MFW line.

B. Crack break within Sleeve:

   Vents and drains are provided for leak detection.

10A.4.1.18 EMERGENCY PROCEDURE

Following rupture of the MFW system in the Auxiliary Building or Turbine Building, the applicable emergency operating procedure(s) would be implemented.

10A.4.1.19 SEISMIC AND QUALITY CLASSIFICATION

The MFW lines in the Auxiliary Building to the isolation valve outside the Containment are designed and constructed in accordance with ANSI B31.1 and between the stop valve and the Containment in accordance with ANSI B31.7, Class II requirements. The feedwater line in the Auxiliary Building is designed to withstand a SSE in combination with normal design loads.

10A.4.1.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

In the Auxiliary Building, the MFW line is protected by a sleeve at all locations below the floor at Elevation 27'0" (Figure 10A.4-2). This ensures that no high energy fluid can be released in these areas.

The MFW lines above floor Elevation 27'0" in the Auxiliary Building pass through the MS Penetration Room. This portion of the MFW line is partially sleeved to provide jet impingement protection. Any circumferential break in the sleeved portion would vent into the MS Penetration Room or into the Turbine Building. In the MS Penetration Room, the open end of the sleeve is partially restricted to limit the rate of feedwater release into this room.

The flow restriction is a closure plate similar to the ones used on the MS encapsulation (Figure 10A.1-3). The construction tolerances imposed on the design will limit the escape area to a maximum of 56.2 in^2.

To determine the pressure transient in the Penetration Room, the Bechtel computer code COPDA (Reference 1) was utilized. The parameters were as follows:

Initial conditions:		
Temperature	-	160°F
Pressure	-	14.7 psia
Humidity	-	70%
Penetration Room:		
Volume	=	24,000 ft^3
Vent to atmosphere	=	44.1 ft^2
Vent coefficient	=	0.71

The mass and energy release rates are determined by the methods presented in Section 10A.1.20.2. The feedwater pressure is reduced to near saturation as the fluid expands into the sleeve. The feedwater temperature is assumed to be 436°F. The discharge rate from the sleeve is conservatively determined to be a maximum of 4500 lbm/sec-ft^2. This is based on a frictionless Moody two-phase flow model. For an escape area of 56.2 in^2 this is equivalent to 1755 lbm/sec. The enthalpy is assumed to be 416 Btu/lbm. The results of the COPDA analysis show that the maximum compartment pressure is 2.4 psig.

The pressure effects of a full feedwater line circumferential break in the Turbine Building will be less than that of a MS line circumferential break, which is discussed in Section 10A.1.20.1.

10A.4.1.21 INTEGRITY OF THE CONTAINMENT STRUCTURE WITH A PIPE RUPTURE OUTSIDE THE CONTAINMENT

The MFW line is sleeved completely below Elevation 27'0", and is also sleeved at the postulated break locations above Elevation 27'0", to retain jet impingement forces and compartment pressurization resulting from a postulated break.

The pressure inside the sleeve due to a postulated break will be released from the open ends of the sleeve into the MS Penetration Room and the Turbine Building. A new vent stack inside the MS Penetration Room will reduce the compartment pressure below 2.6 psi. The partial pressurization of the Containment walls will not affect the integrity of the prestressed concrete Containment Structure.

The magnitude of jet impingement forces resulting from a critical crack will not be greater than 6 kips, in addition to a compartment pressurization of 1 psi. These forces are not large enough to affect the 3'9"-thick prestressed concrete walls of the Containment Structure.

Thus, the integrity of the prestressed concrete Containment Structure will not be impaired due to a postulated full break of the MFW line.

10A.4.1.22 REFERENCE

1. BN-TOP-4, Revision 1, "Subcompartment Pressure and Temperature Transient Analysis," October, 1977

TABLE 10A-9

MAIN FEEDWATER STRESS VALUES

Following is a stress summary of the two intermediate points considered between the terminal ends. The postulated full break locations are shown on Figures 10A.4-2 and 10A.4-3.

PRIMARY STRESS psi

POINT NUMBER	SECONDARY STRESS (<0.8 S_A) psi[(a)]	LONGITUDINAL PRESSURE STRESS	LONGITUDINAL WEIGHT STRESS	SEISMIC OBE	PRIMARY STRESS TOTAL S_{PR}[(d)] TOTAL 0.8 S_h[(a,b)]	TOTAL STRESS [<0.8 (S_A + S_h)] psi[(c)]
2	12,354	5,204	432	572	6,208	18,562
3	10,862	5,204	663	352	6,219	17,081
6	15,320	5,204	458	378	6,040	21,360
7	15,998	5,204	379	405	5,988	21,986

(a) S_A = The larger of f [1.25 S_c + 0.25 S_h + (S_h - S_{PR})] or f (1.25 S_c + 0.25 S_h) as per paragraph 102.3.2(c) and (d) of the USAS Code for pressure piping, USAS B 31.1.0-1967 and as per NC-3600 of Section III (Nuclear Power Plant Components), ASME B&PV Code. 0.8 S_A taken as 21,000 psi.

(b) S_c and S_h are the allowable stresses at cold and hot conditions, respectively, for Class 2 and Class 3 components as per ASME B&PV Code, Section III (Nuclear Power Plant Components).

(c) 0.8 (S_A + S_h) taken as 35,000 psi.

(d) S_{PR} is the total of columns 3 through 5.

10A.4.2 MAIN FEEDWATER AND HEATER DRAIN SYSTEM (INSIDE TURBINE BUILDING)

A general description of the feedwater system is given in Section 10A.4.1. In the Turbine Building, the feedwater line from the pumps to Feedwater Heaters 16A & B, and the drain line from Heater 16B to the condenser, are investigated for a full break.

The pressure and temperature in the feedwater line at the discharge of feedwater pumps are 1150 psig and 345°F at valves wide open, and in the drain line from Heater 16B to the condenser is 384 psia and 440°F, respectively.

Both of these high energy lines pass near the steam generator AFWP Room which is a Category I structure. Figure 10A.4-6 shows the routing of these high energy systems inside the Turbine Building.

10A.4.2.1 PIPE WHIP

The feedwater line and the heater drain line from Feedwater Heater 16B to the condenser normally operates at a pressure above 275 psig and 200°F, and therefore, requires protection from pipe whip following a longitudinal or circumferential break.

10A.4.2.2 CRITERIA FOR LOCATING PIPE BREAKS

Since the piping in the Turbine Building is not designed to seismic conditions, a break can occur anywhere in the Turbine Building.

10A.4.2.3 CRITERIA FOR PIPE BREAK ORIENTATION

Criteria for pipe break orientation is presented in Section 10A.1.3.

10A.4.2.4 SUMMARY OF PIPE WHIP DYNAMIC ANALYSIS

10A.4.2.4.1 Location and Number of Breaks

Since the feedwater line inside the Turbine Building is not designed as a Category I (Seismic) system, a pipe break is postulated to occur anywhere along the line, either in the feedwater line from the feed pumps to Feedwater Heaters 16A & B, or in the drain line from Heater 16B to the condenser. Since a pipe break is postulated to occur anywhere along the line, the critical crack is not governing these lines. A large pipe break is assumed to be a one-time event, requiring a plant shutdown and necessary repairs.

10A.4.2.4.2 The Postulated Rupture Orientation

The longitudinal break is assumed to be parallel to the pipe axis and oriented at any point around the pipe circumference. The longitudinal break area is equal to the effective cross-sectional area upstream of the break location. The circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from a circumferential break are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

10A.4.2.4.3 Description of the Forcing Function and the Mathematical Model

10A.4.2.4.4

The jet thrust forces, caused by the momentum change of fluid flowing through the break, is a function of the upstream fluid conditions, fluid enthalpy, source pressure, pipe flow restrictional friction, and dimensions.

The jet impingement forces acting upon the pipe are computed using the formula described in Section 10A.2.4.3. The jet forces are assumed to be instantaneous. The forcing function is assumed to be a straight line with changes so slow that the variation, up to the time of maximum response, is neglected.

The dynamic analysis method, used for the pipe whip restraint design, is similar to the maximum deflection of a structure subjected to a long duration loading relative to the natural period as presented on page 222 of Reference 1. The mathematical model is shown in Figure 10A.1-5.

10A.4.2.4.5 Unrestrained Motion of the Ruptured Line

The feedwater line from the feed pumps to Feedwater Heaters 16A & B, and the drain line from Heater 16B to the condenser, are sleeved along the length of the AFWP Room. The sleeve will retain jet impingement forces and a pressure due to a postulated break in these feedwater lines.

Restraints are provided along the length of these pipes and additional restraints are provided to preclude any axial movement within the sleeve; no damage can occur to the AFWP room walls and Category I equipment inside this room.

10A.4.2.5 PROTECTIVE MEASURES

10A.4.2.5.1 The Pipe Whip Restraint
10A.4.2.5.2

The feedwater line from the feed pumps to Feedwater Heaters 16A & B, and the drain line from Heater 16B to the condenser, are located inside the Turbine Building, a Seismic Category II structure. The steam-driven AFWP are located in the vicinity of these feedwater lines. These AFWPs are located inside the AFWP Room, a Category I (seismic) enclosure. The motor-driven AFW pumps are located in the SRW Heat Exchanger Room.

The Category I (seismic) enclosure is protected against the effects of jet impingement forces and pipe whip impact.

The pipe whip restraints are provided to protect the AFWP Room against a pipe whip impact due to a postulated beak in these lines. The pipe movement is restricted by limiting the distance between pipe restraints to some dimensions less than the critical plastic hinge length of the pipe.

A. Rupture at an Elbow (Circumferential Break)

When the postulated rupture is at a pipe elbow or fitting for a circumferential break, the critical plastic hinge length (L_1) is determined by the moment resisting capabilities of the pipe (M_p) and the magnitude of the jet thrust (F_j), assuming the pipe acts as a simple cantilever member. The critical plastic hinge length of this condition is determined as:

$$L_1 < \frac{M_p}{F_j}$$

B. Longitudinal rupture

The longitudinal break is considered in the straight pipe run. The most severe loading condition for this type of rupture is when it occurs midway between the restraints. The critical plastic hinge length (L_2) is analyzed if the pipe is considered as a continuous beam along a run of restraints.

$$L_2 < \frac{8M_p}{F_j}$$

Where, M_p is the plastic moment resisting capabilities of the pipe and the F_j is the jet thrust force.

10A.4.2.5.3 Separation Criteria
10A.4.2.5.4

Since the feedwater lines inside the Turbine Building are designed as a Category II system, no separation of redundant features are required.

10A.4.2.5.5 Description of the Pipe Whip Restraint

The pipe whip restraints are provided to prevent the pipe from the feedwater line from feed pump to the Feedwater Heaters 16A & B or the drain line from Heater 16B to the condenser to the AFWP Room. The pipe whip restraints are located using the criteria outlined in Section 10A.4.2.5.1&2.

Description of a typical pipe whip restraint is given in Section 10A.1.5.5.

10A.4.2.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

Category I structures were evaluated for structural adequacy following a postulated pipe break, using the design bases shown in Appendix 5A. Ultimate strength design method for concrete was used as given in the above reference.

The design stresses were proportioned such that the combined stresses are within the limits established in Appendix 5A.

The load factors and load combination used in the evaluation of the existing structures or in the design of new structures or structural components are discussed in Appendix 5A.

10A.4.2.7 STRUCTURAL DESIGN LOADS

The design loads used to evaluate the adequacy of Category I structures or structural components are discussed in Section 10A.1.7.

10A.4.2.8 REVERSAL OF LOADS ON THE STRUCTURE

The forces causing reversal of loadings due to the postulated break on the Category I structures or structural components are:

1. Jet Impingement Force
2. Compartment Pressurization
3. Reaction from Pipe Restraint

The feedwater line from the feed pump to Feedwater Heaters 16A & B, and the drain line from Heater 16B to the condenser are sleeved along the length of the AFWP Room wall. The sleeve will retain jet impingement forces and pressure resulting from

a postulated break in these lines. Any jet forces escaping from the sleeve ends are distributed away from the AFWP Room walls. Thus, there will not be any jet impingement forces or pressurization of the AFWP Room walls due to a postulated rupture in these lines in the vicinity of the AFWP Room.

10A.4.2.9 STRUCTURAL EFFECTS OF OPENINGS

No new openings are provided that would affect the integrity of Category I structures or structural components.

10A.4.2.10 EFFECTS OF STRUCTURAL FAILURE

There will not be a failure of any structure, including Category II (non-Seismic Category I) structures, due to the accident that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident; and

B. Capability to bring the unit(s) to a cold shutdown condition.

The feedwater line from the feed pump to Heaters 16A & B, and the drain line from Heater 16B to the condenser, in the vicinity of the AFWP Room, are restrained due to a postulated break in these lines.

Any break in these lines beyond the boundary of the Auxiliary Building will have no adverse effects on the structural integrity of the Turbine Building.

10A.4.2.11 VERIFICATION THAT HIGH ENERGY PIPE RUPTURES WILL NOT AFFECT SAFETY

The Steam Generator AFWP room is considered with respect to damage from feedwater line breaks inside the Turbine Building. The AFWP Room is a Category I structure and is protected against jet impingement forces, reactive forces and pipe whipping.

The "Ram's Heads" of the Saltwater System are also located in the Turbine Building. Due to the size and location of the "Ram's Heads" with respect to the feedwater line in the Turbine Building, damage caused by rupture and whipping of these pipes and by jet impingement is not considered credible.

A portion of the feedwater piping is located near the K-line in the Turbine Building on the 12’ Elevation in the area of the safety-related main steam drains 5 and 6. If either of the two 16” feedwater lines in this area were to rupture, it could potentially break either or both of these small drain lines and require a unit shutdown to perform repairs. An evaluation of concurrent breaks in a feedwater line and both main steam drain lines were performed. This evaluation showed that this event would not impair the ability to achieve shutdown and would not increase the consequences beyond that of the ruptured feedwater line alone. Therefore, no barriers are required to protect these safety-related main steam drains from a rupture or jet impingement from the feedwater piping.

The other Auxiliary Building wall will not be impacted due to a postulated break in the feedwater line from the feed pump to Heaters 16A & B or a drain line from the Heater 16B to the condenser.

10A.4.2.12 EFFECT ON THE CONTROL ROOM

A high energy line rupture inside the Turbine Building will not affect the Control Room since there is no direct or indirect access from the affected area to the Control Room.

10A.4.2.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Not applicable (Section 10A.4.2.20).

10A.4.2.14 DRAWINGS

Figure 10A.4-6 shows the routing of the feedwater line from the feed pumps to Feedwater Heaters 16A & B, and the drain line from Heater 16B to the condenser.

10A.4.2.15 FLOODING

In the event a feedwater line ruptures inside the Turbine Building, floor drains are provided to handle the resulting blowdown without causing flooding.

10A.4.2.16 QUALITY CONTROL AND INSPECTION

The quality control and inspection programs are presented in Section 10A.1.16.

10A.4.2.17 LEAK DETECTION

Leak detection is discussed in Section 10A.1.18.

10A.4.2.18 EMERGENCY PROCEDURES

Upon leak or rupture of piping in the MFW and heater drain system inside the Turbine Building, the applicable emergency operating procedure would be implemented.

10A.4.2.19 SEISMIC AND QUALITY CLASSIFICATION

The feedwater lines and the heater drain lines to the condenser are designed and constructed in accordance with ANSI B31.1.

10A.4.2.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENT IN COMPARTMENTS

In the Turbine Building, a guillotine break at the MFW line is not as severe as a guillotine break of the MS line. Therefore, the MSLB as discussed in Section 10A.1.20, determines the maximum pressure in this region.

10A.4.2.21 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURE DUE TO PIPE RUPTURE OUTSIDE

Since the system is in the Turbine Building, a rupture in the feedwater system will not affect the Containment Structure.

10A.4.2.22 REFERENCES

1. "Introduction to Structural Dynamics" by professor John M. Biggs, 1964 edition, published by McGraw-Hill Book Company

10A.5 AUXILIARY FEEDWATER SYSTEM

The AFW System, shown in Figure 10A.5-1, is designed to provide feedwater for the removal of sensible and decay heat, and to cool the primary system to 300°F in case the MFW and Condensate Systems are not available. The AFW System may also be used for normal system cooldown to 300°F. During normal operation, the only portion of the AFW System that would have contained a high energy fluid (900 psia and 532°F) is the section of pipe downstream of the isolation valve before the steam generator. This system has been modified by installing a check valve inside the Containment, thus eliminating the line outside of Containment as a high energy system.

The AFW System is used for any one of the following conditions:

1. Loss of offsite electric power.
2. Complete loss of feedwater flow to the steam generators if any of the following conditions occur:
   a. Equipment malfunction (condensate pumps or SGFPs).
   b. Malfunction in the feedwater regulating systems for both steam generators cause all feedwater regulator CVs to close.
   c. In manual feedwater control, the operator either closes each of the feedwater regulator CVs or closes each feedwater stop valve.
   d. A MFW header ruptures.

The turbine-driven pump may be used for normal startup and shutdown. The MS line to the auxiliary SGFP turbine has been analyzed for 21,999 rapid full temperature cycles, which is equivalent to starting the plant from the cold condition to hot condition three times in every two days of plant life. The stress range reduction factor is chosen in accordance with Table 102.3.2(c) of the ANSI Code for Pressure Piping B31.1.

The check valves in the AFW lines located inside the containment are 4" nominal size with either a 600 or 900 lb ANSI rating. The original valves have a 600 lb rating but are being replaced on an as-needed basis with 900 lb tilting disc check valves, which have removable body sub-assemblies to facilitate maintenance. Each valve is cast carbon steel, A216 Gr. WCB, butt-weld ends, pressure seal cap or bolted body joint, stellited, welded or integral seat ring, tilting disc, with standard trim for steam or water service to 550°F. A pressure indicator is located upstream of the check valves to alert the operator of possible back-leakage through the check valves.

10A.5.1 PIPE WHIP

There are no postulated breaks or cracks in the piping of the pump train AFW System because a check valve inside Containment prevents high energy fluid from entering the lines outside Containment. Therefore, no pipe whip, jet impingement, or other reactive forces will occur.

The motor-driven pump trains were added in 1982-83. No pipe whips or breaks are postulated. Jet impingement from pipe cracks has been analyzed and is discussed in Section 10A.5.22.

10A.5.2 PIPE BREAK LOCATIONS

Not applicable (Section 10A.5.1).

10A.5.3 PIPE BREAK ORIENTATION

Not applicable (Section 10A.5.1).

10A.5.4 SUMMARY OF DYNAMIC ANALYSIS

Not applicable (Section 10A.5.1).

10A.5.5 PROTECTIVE MEASURES

Since the AFW System is not a high energy system outside the Containment, no protective measures are considered.

10A.5.6 SEISMIC CATEGORY I STRUCTURE EVALUATION

Since this system is only used for a short duration, a pipe break is not considered to be credible. Therefore, there will be no additional loadings to effect the adequacy of Category I structures which are designed in accordance with the design bases in Appendix 5A.

10A.5.7 STRUCTURAL DESIGN LOADS

There will be no additional loads in the Category I structures or structural components since this system is no longer a high energy system. All Category I structures are designed using the loads listed in Appendix 5A.

10A.5.8 LOAD REVERSAL ANALYSIS

There will be no reversal of loadings in Category I structures or structural components since this system is no longer a high energy system.

10A.5.9 EFFECTS OF NEW OPENINGS ON STRUCTURE

No new openings are required.

10A.5.10 VERIFICATION THAT ANY STRUCTURAL FAILURES WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

No structures will fail (Section 10A.1.10).

10A.5.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

Not applicable (Section 10A.5.1).

10A.5.12 EFFECT ON CONTROL ROOM

Since there are no postulated ruptures in the system piping, there will be no effect on the Control Room.

10A.5.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Not applicable (Section 10A.5.1).

10A.5.14 DESIGN DRAWINGS

Figures 10A.5-1 and 10A.5-2 show the AFW System.

10A.5.15 FLOODING

Not applicable (Section 10A.5.1).

10A.5.16 QUALITY CONTROL AND INSPECTION PROGRAMS

The quality control and inspection programs are presented in Section 10A.1.16.

10A.5.17 LEAK DETECTION

Not applicable (Section 10A.5.1).

10A.5.18 EMERGENCY PROCEDURES

Not applicable (Section 10A.5.1).

10A.5.19 SEISMIC AND QUALITY CLASSIFICATION

The turbine-driven train is designed and constructed in accordance with the ANSI B31.1 requirements, except for the isolation valve and the section penetrating the Containment which is designed and constructed to ANSI B31.7, Class II, requirements. The motor-driven train meets the requirements of ASME Section III, Class 3 and penetration systems are Class 2. The entire line is designed to withstand seismic loadings.

10A.5.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

Not applicable (Section 10A.5.1).

10A.5.21 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURE DUE TO PIPE RUPTURE OUTSIDE

Not applicable (Section 10A.5.1).

10A.5.22 MODERATE ENERGY PIPE CRACK ANALYSIS

This section discusses an evaluation of the modification of Calvert Cliffs Units 1 and 2 AFW System against the NRC Standard Review Plan (SRP 3.6), NUREG 0800, dated July 1981. All postulated leakage crack locations were determined on the basis of that SRP.

10A.5.22.1 Analysis

The methods used in the analysis were:

a. Assume damage and evaluate the results
b. Verify by calculations whether any damage actually would occur.

The following assumptions were made for this analysis:

1. For subcooled fluid or cold water, the discharging fluids at the exit plane of the pipe are expanded at a uniform 10° half angle.
2. Jet deflections off solid objects (such as concrete walls, mechanical components) are assumed to result in dissipated flow energy. Impingement from deflection jets, therefore, does not require any analysis.
3. The effect of gravity on jet trajectory is assumed to be negligible and does not require any analysis.
4. Jet impingement against rigid steel electrical conduits or instrument sensing lines is assumed to cause no damage if the impacted portion is mounted flush against a wall or other structural member.
5. When a jet is obstructed by a floor grating which can be shown to remain in place, the jet effect downstream of the grating is assumed to be diminished and redirected, and does not require any further analysis.
6. Maximum jet distance is calculated based on a final pressure of 1 psig. This distance was calculated to be 12'.

7. Fluid flow from a crack is based on a circular opening area equal to that of a rectangle one half pipe diameter in length and one-half pipe wall thickness in width (SRP 3.6.2-18). The crack calculated for the 6" lines will also be used for the 4" lines to assure a conservative approach.
8. The boundary limits for the moderate energy pipe crack analysis are defined as being from the new AFW pump suction connection to the existing line, through to the check valve upstream of the tie into the existing AFW System, including recirculation piping (no portion of the turbine-driven system is included).
9. All the impingement forces on the targets are calculated taking into consideration a shape factor. That factor is a measure of the target's potential for changing the momentum of the jet as described in ANSI/ANS-58.2-1980, Appendix D.
10. A crack in a motor-driven AFW pump's discharge line is taken as resulting in loss of the use of that pumping train. Therefore, no analysis is required for impingement of a jet on equipment associated with that pump discharge line, except for equipment which could affect the steam-driven train.
11. The AFW pipe crack analysis does not involve flooding nor wetting of the components.
12. All safe shutdown evaluations are based on the Baltimore Gas and Electric Company Interactive Cable Analysis/Safe Shutdown Study generated in response to 10 CFR Part 50, Appendix R.
13. Instrument air copper tubing will be protected as necessary to assure operability of AFW System.
14. The jet impingement forces on the targets are not combined with seismic loads to see the effect on the targets.

10A.5.22.2 Evaluation

Following the performance of calculations to determine the physical dimensions of a leakage crack and its associated jet, all safety-related instruments, their associated conduits, cable trays, and instrument air line within that area, were located. Crack locations are shown in Figures 10A.5-3 Sheets 1 & 2 and 10A.5-4 Sheets 1 & 2. Both units are tabulated in Figure 10A.5-5.

Three preliminary evaluations were made at the outset to limit the scope of the review:

1. On the basis of SRP 3.6.2 method for determining postulated leakage cracks, no crack has been calculated to occur on the suction piping of Unit 1 or Unit 2 AFW motor-driven train. Therefore, it was concluded than no analysis was required.
2. The new AFW motor-driven train instrument tubing and conduits are routed in such a way that Unit 1 instrument tubing and conduits will not be effected by Unit 2 pipe cracks, and vice versa. Moreover, it has been assumed that any crack in the motor-driven pump AFW train will cause the loss of that train. Therefore, it was concluded that no analysis was required for the jet impingement on AFW instrument tubing and conduits associated with the new pump and its discharge. Tubing associated with the steam-driven pump train will be analyzed (Section 10A.5.22.1-10).
3. Only those items located within 12' of the crack location are considered as targets (Section 10A.5.22.1-6).

In all cases, the safety-related targets were shown to fail in the safe position if damage was assumed to occur due to pipe crack jet forces. Thus, no effect on safe shutdown of the plant would result.

To further confirm that no effects from jet impingement could occur, the jet forces were calculated in accordance with SRP 3.6.2 and the forces were applied to the targets - refer to Figure 10A.5-5 for a tabulation of these forces. A generalized stress analysis was then performed for tubing, conduit, or piping which assumes a maximum support span and an impact due to the jet force.

10A.5.22.3 Conclusions

All safety-related targets were shown to sustain the jet load and stay within their allowable stress limits. Tubing 3/4" and 1/2" diameter - both have a small thickness of Burmingham Wire Gauge 16 - were determined by the calculations to be able to withstand the associated load beyond 8.5' from the crack.

10A.6 SHUTDOWN COOLING

The SDC System is used to cool the Reactor Coolant System from 300°F to ≤ 140°F. The maximum temperature and pressure at which SDC can be initiated is 300 psig and 300°F.

The SDC mode is a manually-initiated operation and is under strict administrative control during the entire cooldown period of approximately 36 hours. It is further noted that the SDC System conditions are above 200°F and 275 psig for less than 2% of the total operating time.

Based on the level of quality control, periodic ISI, the low usage factor, the short time this system exceeds 200°F and 275 psig, and the strict administrative control of the SDC System, a break in this system is not considered credible.

10A.6.1 PIPE WHIP

Since no pipe breaks or critical cracks are considered credible due to short duration, no pipe whip is considered.

10A.6.2 PIPE BREAK LOCATION

Not applicable (Section 10A.6.1).

10A.6.3 PIPE BREAK ORIENTATION

Not applicable (Section 10A.6.1).

10A.6.4 DYNAMIC ANALYSIS

Not applicable (Section 10A.6.1).

10A.6.5 PIPE WHIP, JET IMPINGEMENT, AND REACTIVE FORCES PROTECTION

No protective measures are required.

10A.6.6 SEISMIC CATEGORY I STRUCTURE EVALUATION

Since a break in this system is not considered credible, there will be no additional loading to affect the adequacy of Category I structures. Category I structure are designed in accordance with the design bases shown in Appendix 5A of the UFSAR.

10A.6.7 STRUCTURAL DESIGN LOADS

There will be no additional loads on the Category I structures or structural components because a break in this line is not considered to be credible. All Category I structures are designed using loads combinations listed in Appendix 5A of the UFSAR.

10A.6.8 ANALYSIS OF REVERSAL OF LOADS

Not applicable (Section 10A.6.1).

10A.6.9 STRUCTURAL EFFECT OF OPENINGS ADDED TO THE STRUCTURES

No new openings are required.

10A.6.10 VERIFICATION THAT ANY STRUCTURAL FAILURE WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

No structures will fail (Section 10A.1.10).

10A.6.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

Not applicable (Section 10A.6.1).

10A.6.12 EFFECT ON CONTROL ROOM

Since there are no postulated ruptures in the system piping, therefore, there will be no effect on the Control Room.

10A.6.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Not applicable (Section 10A.6.1).

10A.6.14 DESIGN DIAGRAMS AND DRAWINGS

Since no changes are required in this system, no design diagrams or drawings are included.

10A.6.15 FLOODING

Not applicable (Section 10A.6.1).

10A.6.16 QUALITY CONTROL AND INSPECTION PROGRAM

The quality control and inspection programs are presented in Section 10A.1.16.

10A.6.17 LEAK DETECTION

Not applicable (Section 10A.6.1).

10A.6.18 EMERGENCY PROCEDURES

Not applicable (Section 10A.6.1).

10A.6.19 SEISMIC AND QUALITY CLASSIFICATION

The SDC System is designed and constructed to meet ANSI B31.7, Class I requirements. The system is also designed to withstand an SSE in combination with normal design loads.

10A.6.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

Not applicable (Section 10A.6.1).

10A.6.21 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURE DUE TO PIPE RUPTURE

Not applicable (Section 10A.6.1).

10A.7 CHEMICAL AND VOLUME CONTROL SYSTEM

The CVCS is designed to perform the following functions:

(a) Maintain reactor coolant activity at the desired level by removing corrosion and fission products.

(b) Inject chemicals into the Reactor Coolant System to control coolant chemistry and minimize corrosion.

(c) Control the reactor coolant volume by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature and other coolant losses or additions.

The above functions are accomplished by a continuous letdown and charging of the coolant loop at a normal letdown and purification flowrate of 40 gpm and charging flowrate of 44 gpm. FCR 87-0074 reduced the normal letdown and purification flowrate from 40 gpm to 38 gpm. The changes in this analysis as a result of FCR 87-0074 are insignificant.

The maximum letdown flowrate is 128 gpm. (As for the normal letdown and purification flowrate noted above, FCR 87-0074 reduced the maximum letdown flowrate to 126 gpm.) The maximum normal pressure and temperature in the letdown line at the coolant loop nozzle are 2250 psig and 550°F, respectively, occurring when the plant is in normal operation. The letdown CVs, which regulate the reactor coolant flow from the regenerative heat exchanger as required by the pressurizer level regulating system, will reduce the pressure to 460 psig. The letdown fluid temperature is reduced to 232°F after passing through the regenerative heat exchanger, and is further lowered to 120°F through the letdown heat exchanger.

An excess flow check valve has been added to the letdown line inside the Containment downstream of the regenerative heat exchanger. This valve is designed to shut in the event that the flow through the letdown line reaches 200 ± 20 gpm as would occur in the event of a double-ended guillotine or slot rupture, thus limiting the letdown flow in the Auxiliary Building.

The maximum charging flowrate is 132 gpm, supplied when all three positive displacement charging pumps are operating. Normally, only one pump is operating. The maximum normal pressure and temperature at the charging pump discharge are 2310 psig and 120°F. A short term discharge pressure of 3010 psig can be maintained. The charging fluid passes through the regenerative heat exchanger, which increases the temperature to 415°F. The fluid then enters the reactor coolant loop.

The high energy CVCS lines are shown in Figures 10A.7-1 and 10A.7-4.

10A.7.1 PIPE WHIP

Motor-operated valves and instrumentation on the Safety Injection System piping in the west piping Penetration Room at Elevation 27'0" will be protected against possible damage caused by pipe whip occurring at the high stress points of the letdown line because this is an ESFs System. Protection will also be provided for certain safety-related cable trays in the same piping Penetration Room in the vicinity of the letdown line.

Pipe whip protection measures for the letdown line are discussed in Section 10A.7.5.

10A.7.2 CRITERIA FOR PIPE BREAK LOCATIONS

The criteria used for determining the location of pipe breaks has been presented in Section 10A.1.2. The pipe break locations for the letdown line are shown in Figure 10A.7-5.

10A.7.3 CRITERIA FOR PIPE BREAK ORIENTATION

No restriction has been placed on pipe break orientation.

10A.7.4 SUMMARY OF DYNAMIC ANALYSIS

10A.7.4.1 Location and Number of Design Basis Breaks

The locations and number of design basis breaks are chosen in accordance with the criteria discussed in Section 10A.1.2.

10A.7.4.2 The Postulated Rupture Orientation

The circumferential break is assumed to be perpendicular to the pipe axis. A further discussion of circumferential break area and the dynamic forces is provided in Section 10A.1.4.2.

10A.7.4.3 Description of the Forcing Function

The method used to compute the jet impingement forces and description of the forcing function used in the pipe whip analysis are discussed in Section 10A.2.4.3.

10A.7.4.4 Dynamic Analysis and Mathematical Model

The dynamic analysis method used for the CVCS lines is similar to the one used for the MS line and is described in Section 10A.1.4.4. The mathematical model is shown in Figure 10A.1-5.

10A.7.4.5 Unrestrained Motion of the Ruptured Line

There will not be any unrestrained motion of the CVCS line to damage structures, systems and components important to the plant safety.

10A.7.5 PROTECTION AGAINST PIPE WHIP, JET IMPINGEMENT, AND REACTIVE FORCES

10A.7.5.1 Pipe Whip Restraints and Sleeves

All high stress points of the CVCS letdown line (2" schedule 160) in the west piping Penetration Room are sleeved by a 6", schedule 160 pipe. The sleeve pipe is securely anchored to surrounding permanent structures and restrained to prevent its movement. This eliminates any danger from impact on structures, systems and components important to plant safety due to uncontrolled pipe whip resulting from a slot or double-ended rupture. Blowdown jet impingement or other reactive forces resulting from the break will also be effectively contained within the sleeve. Sleeves meet the criteria for the encapsulations referenced in Section 10A.1.5.

The jet impingement forces due to a critical size crack (0.204 in^2 crack area, 551 lbs forces) are not significant enough to damage any safety-related structure, systems or components. There will not be enough steam released from a critical crack to cause compartment pressurization.

10A.7.5.2 Excess Flow Check Valve

The excess flow check valve in the letdown line shuts immediately when the letdown flowrate reaches 210 gpm. This increased flow will occur in the event of a line rupture because of the reduced system pressure drop. Protection against the instantaneous pipe whip and jet impingement from a break will be provided for the time before the valve shuts, limiting the blowdown.

The spring-loaded excess flow check valve on the CVCS letdown line inside the Containment is supplied by Marotta Scientific Controls, Incorporated. It is 2" nominal size with a rating of 1500 lbs. It is designed for 2485 psig and 650°F. The valve body and internals will be constructed of Type 316 stainless steel. It closes automatically at a flowrate of 200 gpm $\pm$ 20 gpm. The operation of the check valve is tested periodically. A bypass orifice, designed to pass 10 gpm, permits resetting of the excess flow check valve from outside the Containment.

10A.7.5.3 Description of a Typical Pipe Whip Restraint

The pipe whipping restraints are provided at the postulated break locations and at other critical locations, such as elbows, to control the pipe whip impact and axial movement due to a full break at the postulated break locations.

Description of a typical pipe whip restraint is given in Section 10A.1.5.5.

10A.7.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

10A.7.6.1 Method of Evaluating Stresses

Category I structures were evaluated for structural adequacy following a postulated rupture using the design bases shown in Appendix 5A. Ultimate strength design method was used for structural evaluations. All Category I structures and structural components were found to be adequate against the loading due to the postulated break.

10A.7.6.2 Allowable Design Stresses

Design stresses are proportioned such that the combined stresses are within the limits established in Appendix 5A.

10A.7.6.3 Load Factors and Load Combinations

Load factors and load combinations used in the design are discussed in Appendix 5A.

10A.7.7 STRUCTURAL DESIGN LOADS

The design loads used to evaluate the adequacy of Category I structures or structural components are discussed in the Section 10A.1.7.

10A.7.8 REVERSAL OF LOADS ON THE STRUCTURES

The forces causing reversal of loadings due to the postulated accident on the Category I structures or structural components are:

1. Jet Impingement Forces
2. Compartment Pressurization
3. Reactions from Pipe Whip Restraints

Since the CVCS letdown line is only 2" in diameter, the magnitude of the jet impingement forces resulting from a critical crack will not be greater than 6 kips. This force is smaller compared to a jet impingement force of 10 kips resulting from a critical crack in the MS line.

The excess flow check valve, located inside the Containment Structure, will close and limit the release of the high energy fluid following a postulated rupture. With the excess flow check valve closed, only 10 gpm can be released via the bypass orifice. This quantity is insignificant and will not overpressure the room in which the break occurs.

Table 10A-10 gives stresses in various structural components of the pipe Penetration Room in the vicinity of the letdown line. These stresses are due to a compartment pressurization of 4.85 psig. It can be seen from this table that the stresses in the structural components of the Penetration Room are well within the allowable stresses.

10A.7.9 EFFECTS OF NEW OPENINGS ON STRUCTURE

No openings are required to vent the compartment structure following a break in the letdown or charging lines.

10A.7.10 VERIFICATION THAT ANY STRUCTURAL FAILURE WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

No structures will fail (Section 10A.1.10).

10A.7.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

A break or crack in the letdown line will result in flashing a maximum of approximately 36% of the blowdown released into the West Piping Penetration Room or Letdown Heat Exchanger Room in the Auxiliary Building. The crack will cause compartment pressurization which will automatically close the letdown line isolation valves inside the Containment, upstream of the regenerative heat exchanger (Section 10A.7.17). This will terminate the letdown line blowdown before causing any adverse effects on plant safety.

Pressure relief for the letdown heat exchanger room is provided by an open block-out connecting to the west piping Penetration Room. Pressure in the Penetration Room will be allowed to gradually decay.

The interconnecting Penetration Room ventilation system will maintain negative pressure in these rooms. Backdraft dampers will prevent propagation of the steam environment into the electrical Penetration Rooms.

There will be no steam released from a charging line break because the system temperature is too low.

Shutting off the charging pumps by use of the remotely-located hand switches (one per pump) will serve to terminate blowdown from a rupture in the charging system lines. Accessibility to these hand switches will not be affected by a charging line rupture.

Discussion of specific emergency procedures to be followed in the event of a letdown or charging line rupture is in Section 10A.7.18.

10A.7.12 EFFECT ON CONTROL ROOM

A letdown or charging line rupture will not affect the Control Room, since there is no direct access from the affected area to the Control Room.

10A.7.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

The release of high energy fluid as a result of circumferential or longitudinal break will be immediately limited to 10 gpm by the closure of the excess flow check valve. The 10 gpm released via the bypass orifice is not significant and can be easily handled by the ventilation system; therefore, no significant environmental change will occur.

Environmental conditions, leak detection and blowdown limitation from a crack in the letdown line are discussed in Sections 10A.7.17 and 10A.7.18. Equipment that must

function and equipment locations are shown in Tables 10A-5 and 10A-6. Equipment which must function in the steam environment has been qualified for that environment.

10A.7.14 DESIGN DRAWINGS

Figures 10A.7-2, 10A.7-3, and 10A.7-5 show the routing of the letdown and charging lines and the proposed break locations for the letdown line.

10A.7.15 FLOODING

No excessive amounts of water will be released from a slot or double ended rupture in the letdown line because the excess flow check valve will seat and terminate blowdown. However, a crack occurring at another point in the line in the piping Penetration Room may not cause enough blowdown to seat the excess flow check valve and terminate the flow. Water released from the crack will fall through the Penetration Room grating at Elevation 27'0" to the floor at Elevation 5'0". The room at Elevation 5'0" is equipped with floor drains which will carry the water off to the Waste Processing System. Similarly, should a break occur in the line in the Letdown Heat Exchanger Room, the floor drains in the room will be adequate to take the water to the Waste Processing System.

No appreciable amounts of water will be released from a break or crack in the charging line because of the rapid system pressure decay. Floor drains in the piping Penetration Room and Charging Pump Rooms are adequately sized to handle the resulting blowdown from the charging line without causing flooding.

10A.7.16 QUALITY CONTROL AND INSPECTION PROGRAMS

The quality control and inspection programs are presented in Section 10A.1.16.

10A.7.17 LEAK DETECTION EQUIPMENT

Four pressure sensors are installed in the West Piping Penetration Room and Letdown Heat Exchanger Room to detect the rise in ambient pressure resulting from blowdown release in the event of a crack in the letdown line. Two-out-of-four trip logic (signal from two of the four pressure sensors) is used to close each of the two letdown line isolation valves (CV-515 and CV-516) located upstream of the regenerative heat exchanger. Each pressure sensor is set to send the actuation signal automatically when room pressure reaches 0.5 psig, which will occur in the Penetration Room at 0.75 seconds after the crack. The room vapor temperature will be 111°F at this time. The valve will close within 9 seconds after the signal. The pressure in the Penetration Room after this time period will be 4.85 psig with a room vapor temperature of 168°F.

These instruments are type tested to ensure they operate under the maximum environmental conditions experienced in the area they are located.

10A.7.18 EMERGENCY PROCEDURES

Following rupture of the CVCS in the Auxiliary Building, the applicable emergency operating procedure(s) would be implemented. The excess flow check valve would shut if the leak exceeded 200 $\pm$ 20 gpm to limit the severity of the casualty.

10A.7.19 SEISMIC AND QUALITY CLASSIFICATION

The letdown and charging lines are designed to withstand a SSE in combination with normal design loads. The lines are constructed to ANSI B31.7, Class II standards in the Auxiliary Building to the letdown heat exchanger, and from the charging pump discharge.

10A.7.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

The locations of the postulated circumferential or longitudinal breaks are shown on Figure 10A.7-5. The flow from these breaks will cause the excess flow check valve, which is located inside the Containment, to close and immediately limit the release of the high energy fluid to the bypass flow of 10 gpm, which is not significant. Thus, a pressurization problem from this type of break cannot occur.

Information on temperature and pressure in the piping Penetration Room and in the letdown cooling heat exchanger compartment due to a critical crack in the CVCS letdown line was developed with the aid of the Bechtel computer code COPATTA. This computer program is described in Section 14.20.

The piping Penetration Room and the letdown heat exchanger compartment are adjoining rooms connected by a large opening above a shield wall. Therefore, for the purpose of evaluating the environmental consequences of pipe cracks, the two rooms can be considered as one.

The parameters used with COPATTA for this analysis are as follows:

Initial Room Conditions		
Temperature	-	120°F
Pressure	-	14.7 psia
Humidity	-	50%
Volume	-	29000 ft^3
Heat Sinks (west piping Penetration Room)		
Walls	-	5600 ft^2 of painted concrete
Floors	-	100 ft^2 of concrete
Grating	-	100 ft^2 of galvanized steel

The mass flow rate was chosen to be just under the value that would close the excess flow check valve. The assumed blowdown conditions are:

Mass blowdown	=	24.6 lbm/sec
Enthalpy	=	545.3 Btu/lbm

The pressure exceeds 0.5 psig in 0.75 seconds at which time the temperature is 111°F. When the pressure-actuated isolation valve closes 9 seconds later, the pressure reaches a maximum of 4.85 psig and the temperature reaches a maximum of 168°F. Section 10A.7.17 for discussion on the operation and control logic for the isolation valve.

10A.7.21 INTEGRITY OF THE CONTAINMENT STRUCTURE AND A PIPE RUPTURE OUTSIDE THE CONTAINMENT

Since the CVCS letdown line is only 2" in diameter, any forces acting upon the Containment Structure (Section 10A.7.8 for magnitude of jet impingement forces) will not be significant enough to impair the integrity of the prestressed concrete Containment Structures.

TABLE 10A-10

STRESSES IN STRUCTURAL COMPONENTS IN THE PIPE PENETRATION ROOM DUE TO PRESSURIZATION FROM POSTULATED RUPTURE OF LETDOWN LINE

		CALCULATED STRESSES DUE TO PRESSURIZATION OF 4.85 psig COMPREHENSIVE TENSILE		RATIO OF CALCULATED STRESSES VS. ALLOWABLE STRESSES[(a)]	
STRUCTURAL COMPONENT	**THICKNESS**	**STRESS IN CONCRETE (psi)**	**STRESS IN REBAR (psi)**	**CONCRETE**	**REBAR**
North-South Walls	3'0"	970	22,000	0.285	0.45
East-West Walls	2'0"	880	33,000	0.258	0.312
Ceiling EL 45'0"	2'3"	30	3,025	0.0265	0.0577

(a) Allowable stresses for the concrete are taken at 85% of the ultimate strength. Allowable stresses for the rebar are taken at 90% of the ultimate strength.

10A.8 SAMPLING SYSTEM

Reactor Coolant Sampling: The reactor coolant sampling system for each unit includes piping, valves, sample bomb connections, instrumentation, and a cooler enclosed in a fume hood. The reactor coolant sample sink is located on the 45' Elevation of the Auxiliary Building. The fume hood is ventilated by a blower through a high efficiency filter. Interlocking high-density concrete block shielding is provided to protect personnel in the rest of the sample room. Additional equipment in the sample room includes the steam generator blowdown sample panel, the steam generator sampling system, and, in Unit 1 only, the radioactive miscellaneous sampling system.

The reactor coolant samples can be taken from three points: Reactor coolant system hot leg, pressurizer liquid, and pressurizer vapor. Remotely operated valves, controlled by handswitches located on the steam generator blowdown sample panel, may be used to select one of the three sample points. Another remotely operated valve, controlled by a handswitch in the Control Room, must be open before samples can be taken. All four of these remotely operated valves are closed by a Safety Injection Actuation Signal from the Engineered Safety Features Actuation System. Manual isolation (entry valves) is provided for each sample line.

A high temperature sample is cooled in a sample cooler supplied by component cooling water. Once cooled, the sample pressure is reduced and the sample is passed through a metering valve for flow control. Both routine and post-accident samples may be obtained by the reactor coolant sampling system. Samples are normally transported to a laboratory in the Auxiliary Building for analysis.

A radiation monitor is located near the reactor coolant sample sink. It provides continuous monitoring of radiation levels, a local alarm, and a remote alarm in the Control Room.

Steam Generator Blowdown Sampling: The steam generator blowdown sampling system for each unit consists of a sample conditioning rack, fume hood, and instrument panel. This system is located in the sample room on the 45' Elevation of the Auxiliary Building. The sample conditioning rack contains sample entry valves, piping, primary coolers, pressure reducers, and a chiller-cooled isothermal bath. The primary sample coolers and chiller for the isothermal bath are both cooled by component cooling water.

Each high pressure sample flows through its respective entry valve to a primary cooler, pressure reducer, isothermal bath, and is then routed to a fume hood. The fume hood contains grab sample valves, piping, flow indicators, pH and conductivity elements, temperature indicators, and cation columns. The fume hood is ventilated through a high efficiency filter by an individual blower. The Unit 1 steam generator blowdown sample fume hood also contains the Radioactive Miscellaneous Sample System.

The instrument panel houses the following equipment: pH and conductivity displays, alarm annunciators, chiller controls, and primary sample valve position indication and handswitches. The pH and conductivity samples are continuously monitored. Alarms are provided for high sample outlet temperature, high conductivity, and high or low pH. Any of these alarms will cause a common trouble alarm in the Control Room.

10A.8.1 PIPE WHIP

In accordance with the proposed AEC criteria, pipe breaks in lines under 1" nominal pipe size need not be considered. The primary sampling system line sizes are 1/2" and 3/4", and the secondary system lines are 1/4". Therefore, no pipe whip will occur.

10A.8.2 CRITERIA FOR PIPE BREAK LOCATIONS

A critical crack defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width is postulated to occur at any location.

10A.8.3 CRITERIA FOR PIPE BREAK ORIENTATION

A critical crack is assumed to be oriented at any point around the pipe circumference.

10A.8.4 SUMMARY OF DYNAMIC ANALYSIS

Not applicable (Section 10A.8.1).

10A.8.5 PROTECTIVE MEASURES

Engineered Safety Feature Systems in the vicinity of high energy sampling system lines are protected by shielding, if necessary, to protect from damage as a result of jet impingement from a crack in these lines.

10A.8.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

There will be no effect on Category I structures as a result of a crack in the sampling line.

10A.8.7 STRUCTURAL DESIGN LOADS

Structural design loads are discussed in Section 10A.1.7.

10A.8.8 LOAD REVERSAL ANALYSIS

There will be no reversal of loads on the structure due to a crack in the sampling line.

10A.8.9 EFFECTS OF NEW OPENINGS ON STRUCTURE

No new openings are required.

10A.8.10 VERIFICATION THAT ANY STRUCTURAL FAILURES WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

No structures will fail.

10A.8.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

There will be no environmental consequences as a result of a crack in the sampling system piping.

The primary sampling system lines can be isolated by activating any or all of the three remotely-located hand switches, which will close the isolation valves inside the Containment on the three sample drawoff lines (two from the pressurizer and one from the hot leg). Accessibility to the hand switches will not be affected by a primary sampling system line crack.

The secondary sampling system lines can be isolated by manually closing any one of the three valves on the top and bottom steam generator blowdown sampling lines. A crack in these sampling lines will not hinder access to these valves.

10A.8.12 EFFECT ON CONTROL ROOM

A sample line crack will not affect the Control Room, since there is no direct access from the affected area to the Control Room.

10A.8.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

There will be no significant environmental effect on the ESF equipment as a result of a crack in the sampling system lines because of the small blowdown rate.

10A.8.14 DESIGN DRAWINGS

Figure 10A.8-1 is a drawing of the high energy portions of the primary and secondary sampling system lines. Figure 10A.8-2 shows the routing of the sampling system.

10A.8.15 FLOODING

No flooding of ESF equipment will occur as a result of a crack in the sampling system lines because of the small blowdown rate.

10A.8.16 QUALITY CONTROL AND INSPECTION PROGRAMS

The quality control and inspection programs for the lines outside the Containment are presented in Section 10A.1.16.

10A.8.17 LEAK DETECTION

A crack in the primary or secondary system lines would result in a loss of sample in the hot and cold labs.

10A.8.18 EMERGENCY PROCEDURES

Emergency procedures for the sampling system are outlined in Section 10A.1.18.

10A.8.19 SEISMIC AND QUALITY CLASSIFICATION

The sampling system lines are constructed to ANSI B31.7, Class I standards in the Auxiliary Building to the sampling rooms. They are designed to withstand a SSE in combination with normal design loads.

10A.8.20 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

No applicable (Section 10A.8.13).

10A.8.21 DESCRIPTION OF ASSUMPTIONS, METHODS, AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURES DUE TO PIPE RUPTURE OUTSIDE

There will be no effect on the Containment Structure as a result of forces from a crack in the sampling system.

10A.9 <u>AUXILIARY STEAM SYSTEM</u>

The Auxiliary Steam System is routed in various areas in the Auxiliary Building and is the steam supply to the waste evaporators which are located on Elevations 45'0" and 69'0". Normal auxiliary steam pressure is reduced from 195 psia and 320°F to a pressure of 70 psia and 302°F by means of a pressure-reducing station located in the Turbine Building. A full-sized safety relief valve is provided downstream of the pressure reducing station to prevent overpressurization in the event of a failure of the station.

10A.9.1 PIPE WHIP

Since the pressure in the steam lines to the waste evaporators is below 275 psig, there is not sufficient energy to cause a pipe whip following a crack break.

10A.9.2 PIPE BREAK LOCATIONS

A critical crack, defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width, is postulated to occur at any location.

10A.9.3 PIPE BREAK ORIENTATION

A critical crack is assumed to be oriented at any point around the pipe circumference.

10A.9.4 SUMMARY OF DYNAMIC ANALYSIS

Not applicable (Section 10A.9.1).

10A.9.5 PROTECTION AGAINST PIPE WHIP, JET BLOWDOWN, AND REACTIVE FORCES

10A.9.5.1 <u>Pipe Whip Restraints</u>

Not applicable (Section 10A.9.1).

10A.9.5.2 <u>Jet Forces</u>

The jet force from a critical crack is 100 lbs. A force of this magnitude will have a negligible effect on the massive structural components.

10A.9.6 EVALUATION OF SEISMIC CATEGORY I STRUCTURES

Category I structures were evaluated for structure adequacy following a postulated rupture using the design bases shown in Appendix 5A. Design stresses are such that the combined stresses are within the limits established in Appendix 5A. Load factors and load combinations are also discussed in Appendix 5A.

10A.9.7 STRUCTURAL DESIGN LOAD

The design loads, such as dead load, live load, pipe load, jet impingement and pressurization, are used to evaluate the adequacy of Category I structures following a postulated rupture. A detailed discussion of these loads is given in Section 10A.1.7.

10A.9.8 REVERSAL OF LOADS ON STRUCTURE

There will be no reversal of loads on the structure due to an auxiliary steam line break.

10A.9.9 STRUCTURAL EFFECT OF OPENINGS ADDED TO THE STRUCTURE

No openings are required to vent the structure due to auxiliary steam line break.

10A.9.10 VERIFICATION THAT ANY STRUCTURAL FAILURE WILL NOT AFFECT OTHER STRUCTURES REQUIRED FOR SAFETY

There will be no failure of a structure due to a postulated break in the auxiliary steam line.

10A.9.11 VERIFICATION THAT PIPE RUPTURE WILL NOT AFFECT SAFETY

The "Ram's Heads" of the Saltwater System are located in the Turbine Building. Due to the size and location of the "Ram's Heads" with respect to the Auxiliary Steam Line, damage caused by rupture and whipping of this line and by impingement is not considered credible.

A portion of the auxiliary steam piping is located near the K-line in the Unit 1 Turbine Building on the 12' Elevation in the area of the safety-related main steam drains 5 and 6. Although only cracks are postulated (see Section 10A.9.20), it has been shown that even if this caused a full rupture of both of these small drain lines, safe shutdown of the plant could be achieved with no increase in consequences over those previously obtained for other breaks in the Turbine Building. Therefore, no barriers are required to protect this safety-related main steam piping from a jet impingement by this auxiliary steam piping.

The effect on safety from a pipe rupture is discussed in Section 10A.9.20.

10A.9.12 EFFECT ON CONTROL ROOM

An auxiliary steam line rupture will not affect the Control Room.

10A.9.13 ENVIRONMENTAL QUALIFICATION OF AFFECTED REQUIRED EQUIPMENT

Tables 10A-5 and 10A-6 shows the equipment and instrumentation required to place the plant in a safe shutdown condition. As indicated by these tables, there is no vital equipment or instruments exposed to steam environment due to a pipe rupture in the Auxiliary Steam System (Section 10A.9.20).

10A.9.14 DESIGN DIAGRAMS AND DRAWINGS

Figures 10A.9-1, 10A.9-2, and 10A.9-3 show auxiliary steam to the waste evaporators.

10A.9.15 FLOODING

No flooding of the ESF equipment will occur as a result of a crack in the auxiliary steam line due to the appreciably small amount of mass released.

10A.9.16 QUALITY CONTROL AND INSPECTION

The quality control and inspection programs are presented in Section 10A.1.16.

10A.9.17 LEAK DETECTION

Temperature switches are located in various compartments along the steam line to give an indication in the Control Room if a crack should occur in the auxiliary steam line. No credit is taken for these switches, however.

10A.9.18 EMERGENCY PROCEDURE

Emergency procedures for this system are similar to those outlined in Section 10A.1.18.

10A.9.19 SEISMIC AND QUALITY CLASSIFICATION

The auxiliary steam line is designed and constructed in accordance with ANSI B31.1.

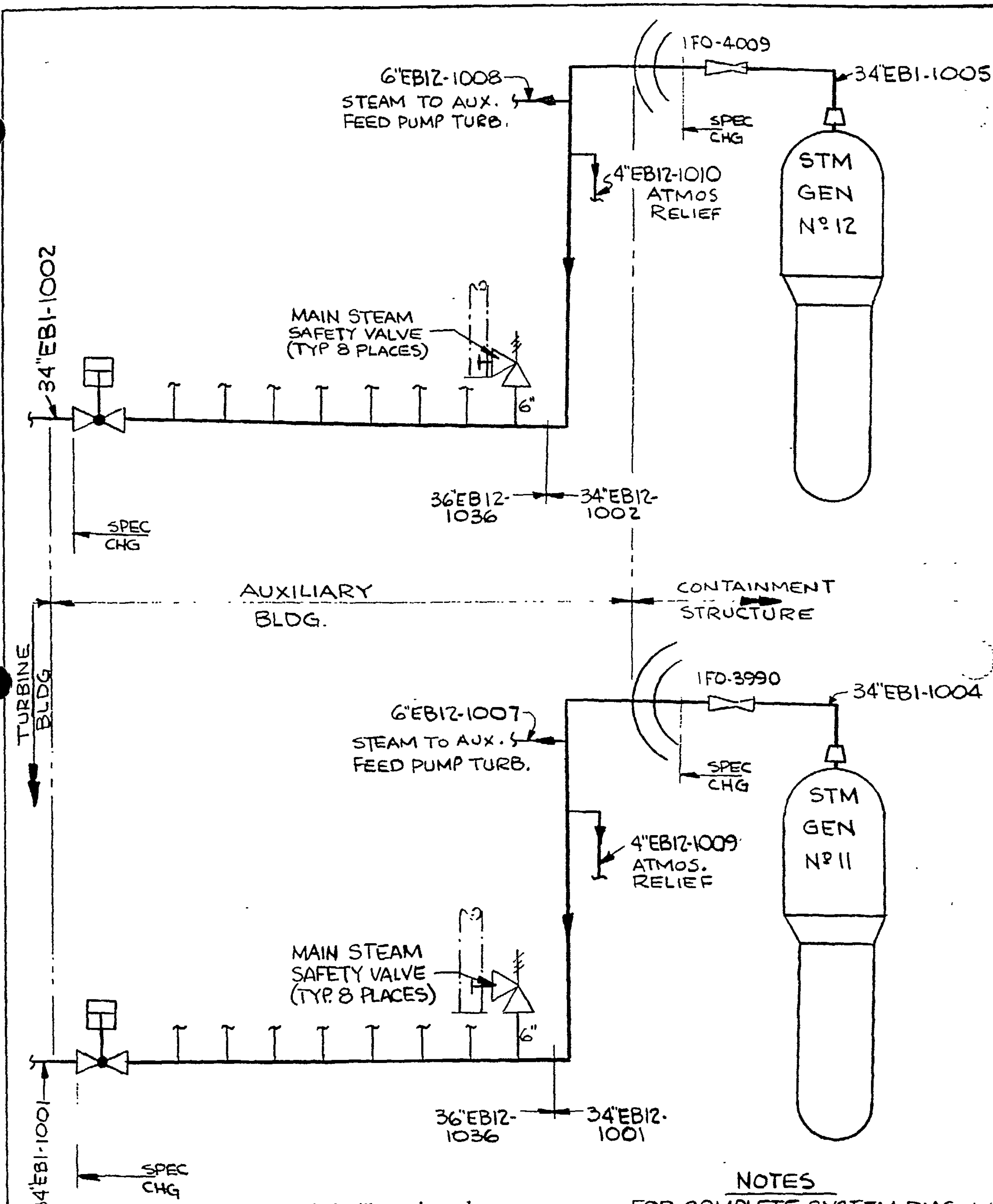
10A.9.20 DESCRIPTION OF ASSUMPTIONS, METHOD AND RESULTS OF ANALYSIS FOR PRESSURE AND TEMPERATURE TRANSIENTS IN COMPARTMENTS

In accordance with the pipe break location criteria presented in Section 10A.9.2, circumferential or longitudinal breaks are not credible accidents for this system and, therefore, will not cause a pressure or temperature problem. Critical cracks, however, are postulated to occur anywhere and the pressure and temperature consequences have been studied. Because of the low system pressure and the small size of a critical crack, the mass release rate will not cause a pressurization problem.

Pressure reducing and safety relief valves limit the system pressure to 70 psig inside the Auxiliary Building. A critical crack of the largest auxiliary steam line (12" line) in the Auxiliary Building has an area of 1.17 in^2. The maximum steam released from this rupture will be only 3.25 lbm/sec. Local pressure increases will be undetectable and local temperatures will not exceed 160°F.

10A.9.21 DESCRIPTION OF ASSUMPTIONS, METHODS AND RESULTS OF ANALYSIS FOR EFFECT ON PRIMARY OR SECONDARY CONTAINMENT STRUCTURE DUE TO PIPE RUPTURE OUTSIDE

A crack in the auxiliary steam line will not affect the Containment Structure.



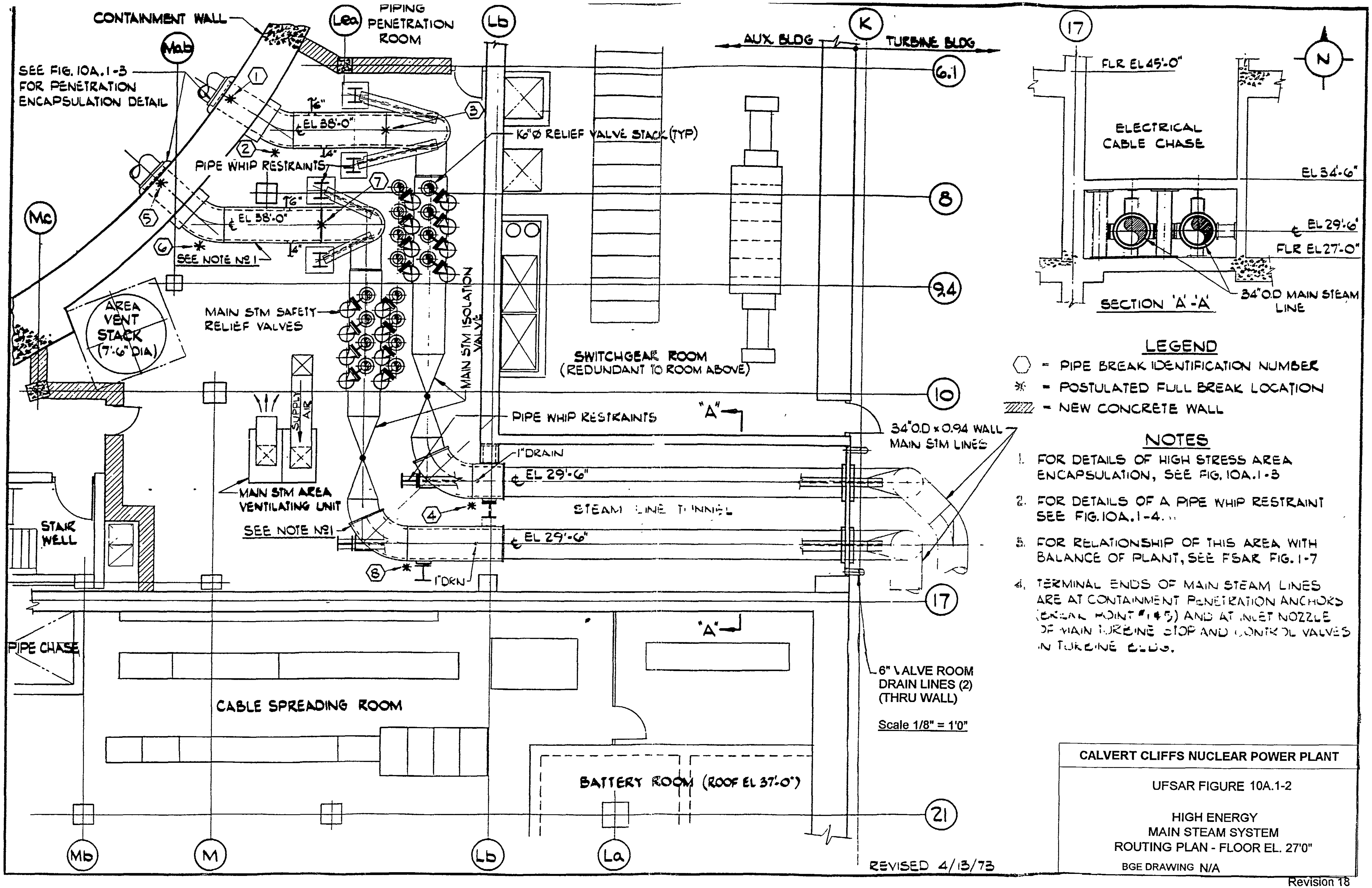
NOTES

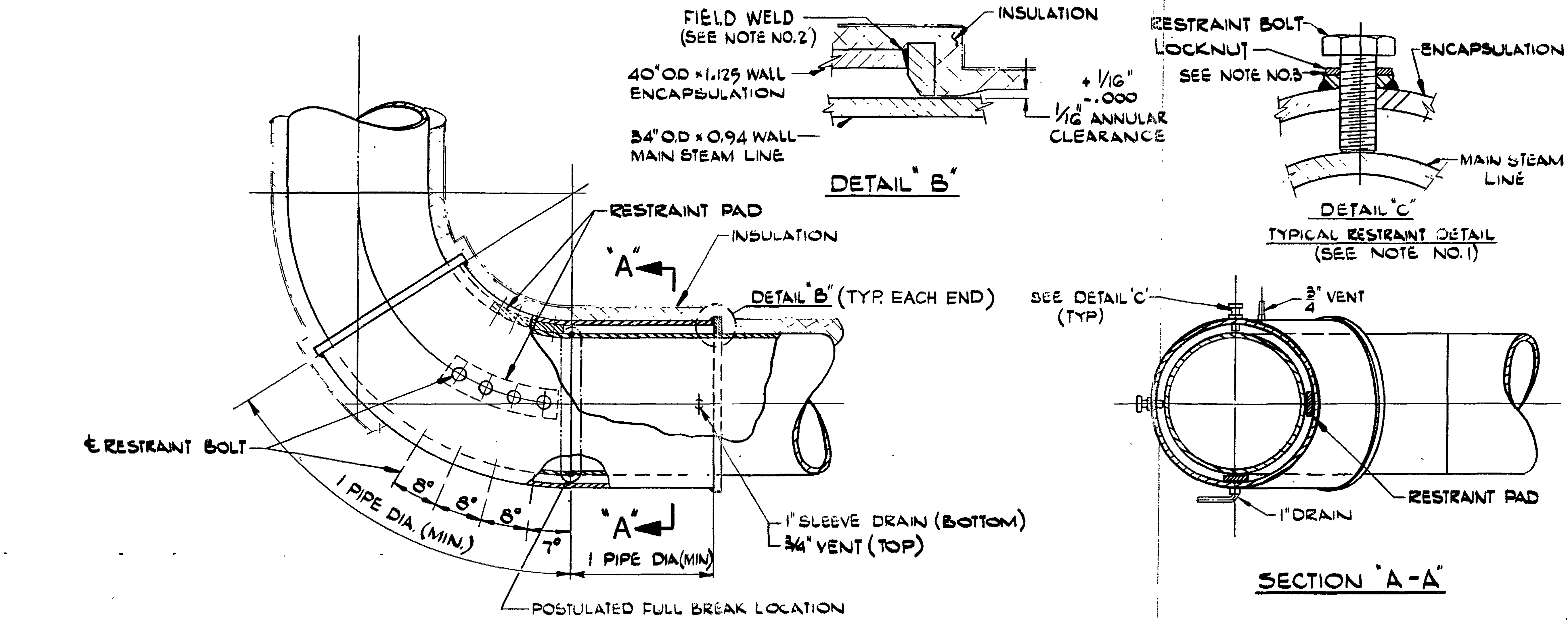
This sketch is for illustration only.
It is not intended to show all piping system details.

FOR COMPLETE SYSTEM DIAGRAM SEE FSAR FIG. 10-1

Revision 20

BALTIMORE GAS AND ELECTRIC CO. CALVERT CLIFFS NUC. PWR. PLNT.	MAIN STEAM SYSTEM	FIG. 10A.1-1


CONTAINMENT WALL
PIPING PENETRATION ROOM
AUX. BLDG
TURBINE BLDG
SEE FIG. 10A.1-3 FOR PENETRATION ENCAPSULATION DETAIL
EL 38'-0"
16"Ø RELIEF VALVE STACK (TYP)
PIPE WHIP RESTRAINTS
SEE NOTE Nº 1
AREA VENT STACK (7'-6" DIA)
MAIN STM SAFETY RELIEF VALVES
MAIN STM ISOLATION VALVE
SWITCHGEAR ROOM (REDUNDANT TO ROOM ABOVE)
SUPPLY AIR
PIPE WHIP RESTRAINTS
1" DRAIN
EL 29'-6"
MAIN STM AREA VENTILATING UNIT
STEAM LINE TUNNEL
SEE NOTE Nº1
1" DRN
STAIR WELL
PIPE CHASE
CABLE SPREADING ROOM
BATTERY ROOM (ROOF EL 37'-0")
34"O.D x 0.94 WALL MAIN STM LINES
6" VALVE ROOM DRAIN LINES (2) (THRU WALL)
Scale 1/8" = 1'0"
REVISED 4/13/73
FLR EL 45'-0"
ELECTRICAL CABLE CHASE
EL 34'-6"
EL 29'-6"
FLR EL 27'-0"
SECTION 'A'-'A'
34"O.D MAIN STEAM LINE
LEGEND
= PIPE BREAK IDENTIFICATION NUMBER
= POSTULATED FULL BREAK LOCATION
= NEW CONCRETE WALL
NOTES
1. FOR DETAILS OF HIGH STRESS AREA ENCAPSULATION, SEE FIG. 10A.1-3
2. FOR DETAILS OF A PIPE WHIP RESTRAINT SEE FIG. 10A.1-4.
3. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT, SEE FSAR FIG. 1-7
4. TERMINAL ENDS OF MAIN STEAM LINES ARE AT CONTAINMENT PENETRATION ANCHORS (BREAK POINT #1 & 5) AND AT INLET NOZZLE OF MAIN TURBINE STOP AND CONTROL VALVES IN TURBINE BLDG.
CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 10A.1-2
HIGH ENERGY MAIN STEAM SYSTEM ROUTING PLAN - FLOOR EL. 27'0"
BGE DRAWING N/A
Revision 18



NOTES

1. PRIOR TO INSTALLATION OF ENCAPSULATION SECTIONS, FIELD MEASUREMENTS ARE TO BE TAKEN OF THE STEAM LINE AT THE POINTS OF CONTACT BY THE END CLOSURE RING.

2. RESTRAINT BOLTS TO BE ADJUSTED TO CENTER THE STEAM LINE IN THE WELDED UP ENCAPSULATION SECTIONS WHILE IN THE COLD CONDITION TO PERMIT FIT-UP OF THE END CLOSURE RINGS. CLOSURE RINGS ARE TO BE SHIMMED BETWEEN RING AND PIPE TO INSURE CLEARANCE AS SHOWN IN DETAIL 'B' PRIOR TO WELDING TO 40" DIA. SLEEVE.

3. RESTRAINT BOLTS TO BE MARKED AFTER COLD POSITIONING OF ENCAPSULATION AND BACKED OUT AS REQUIRED (TO PERMIT THERMAL GROWTH OF STEAM LINE) PRIOR TO BLOCKING WITH LOCKNUT

NO SCALE

BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2

BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND

ENCAPSULATION DETAILS
MAIN STEAM SYSTEM

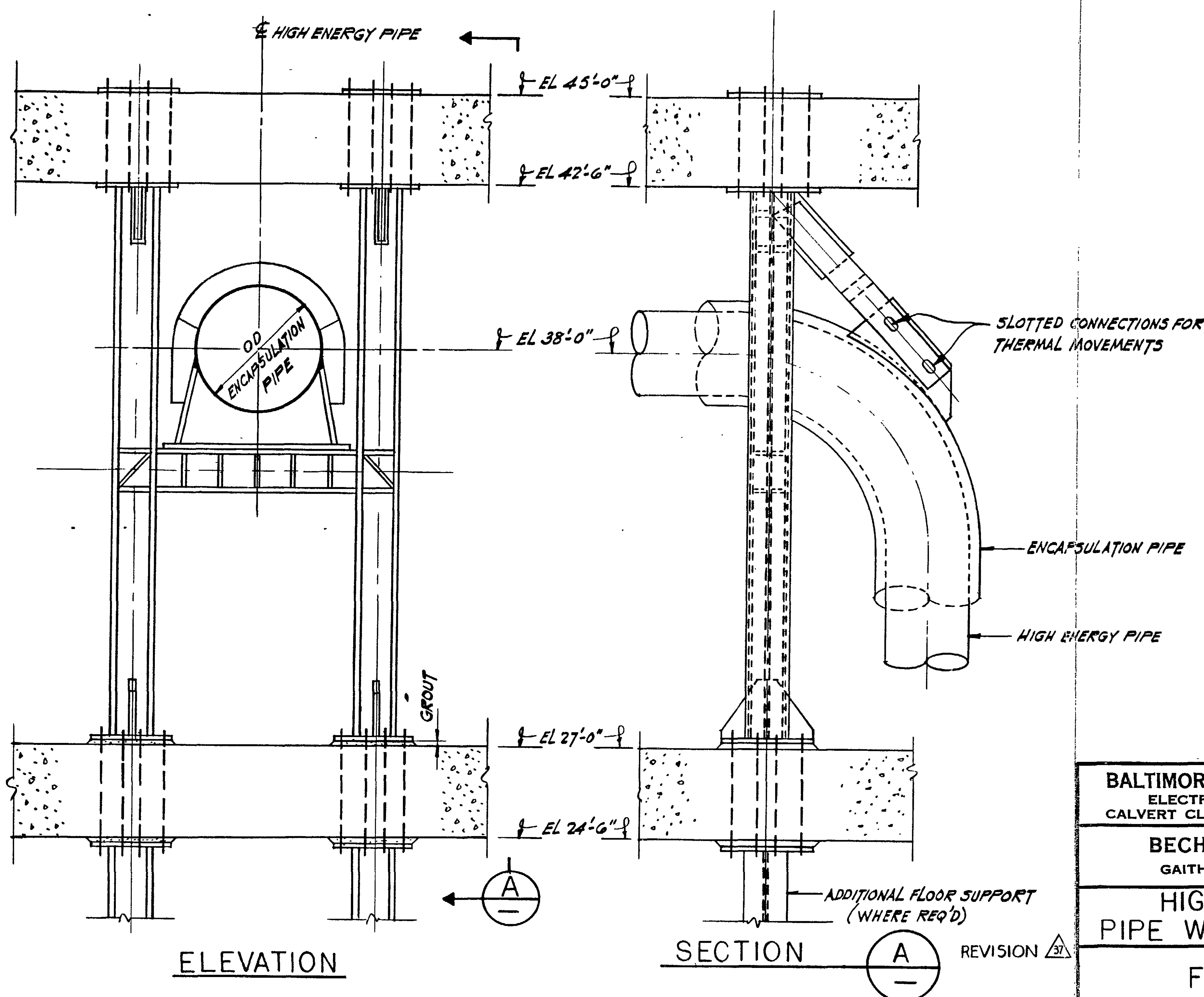
JOB NO.		REV.
6750	FIG. 10A.1-3	D

REVISION 37

REVISED 12/9/74 PB

10A-2

10A3



REVISED 12-6-74 REV. C

BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2

BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND

HIGH ENERGY PIPE WHIP RESTRAINT

FIG. 10A.1-4

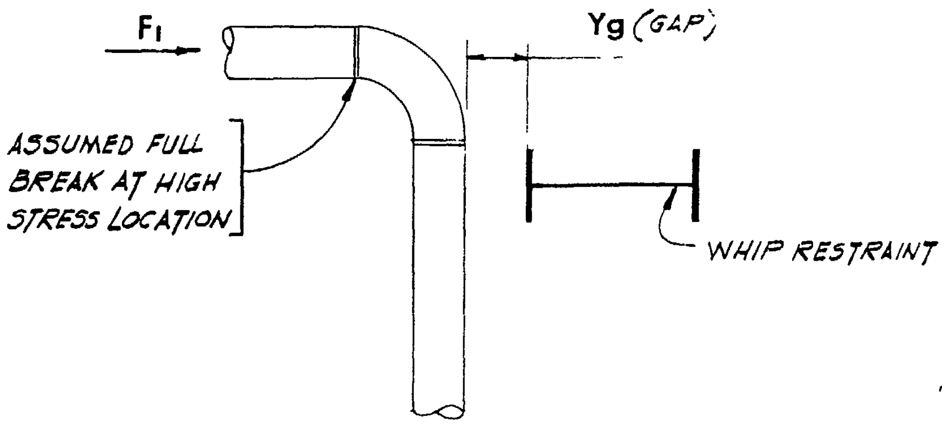
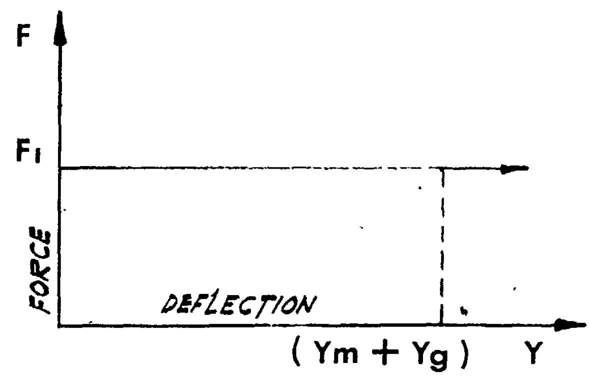
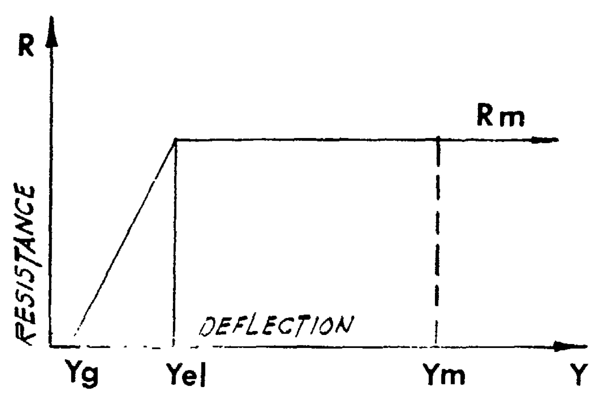


FIGURE 10A.1-5A



FORCING FUNCTION

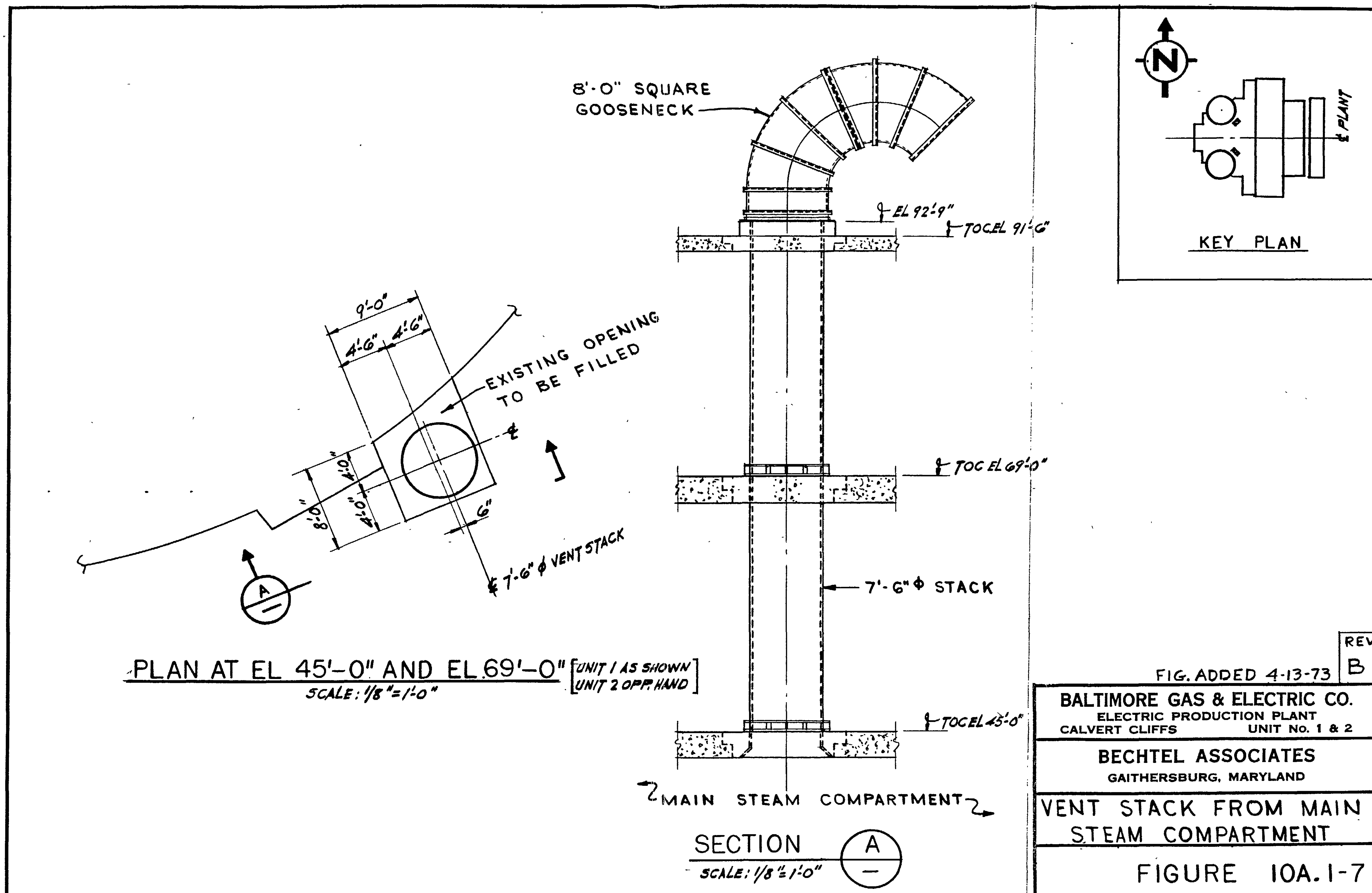
FIGURE 10A.1-5B

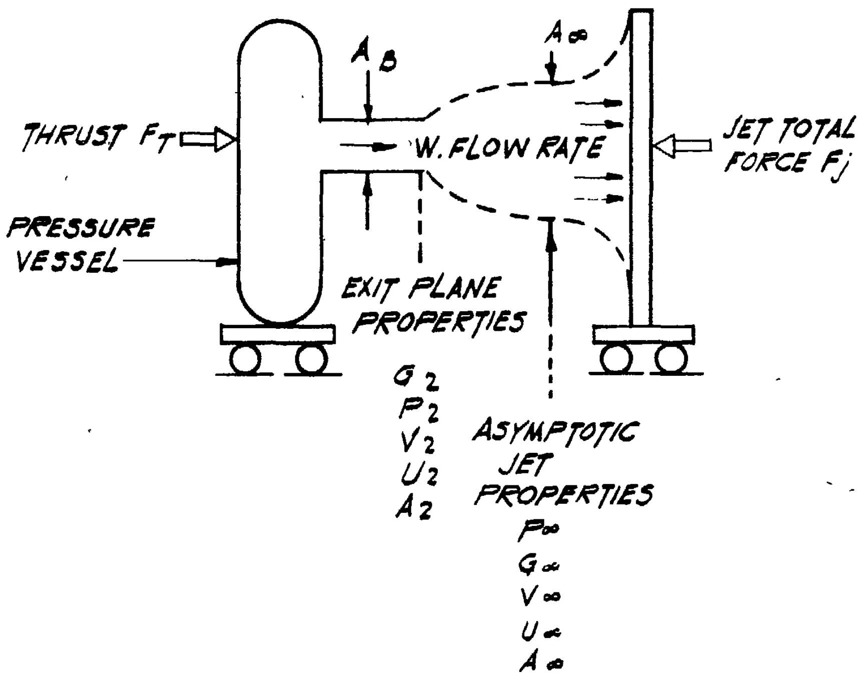


RESISTING FUNCTION

FIGURE 10A.1-5C

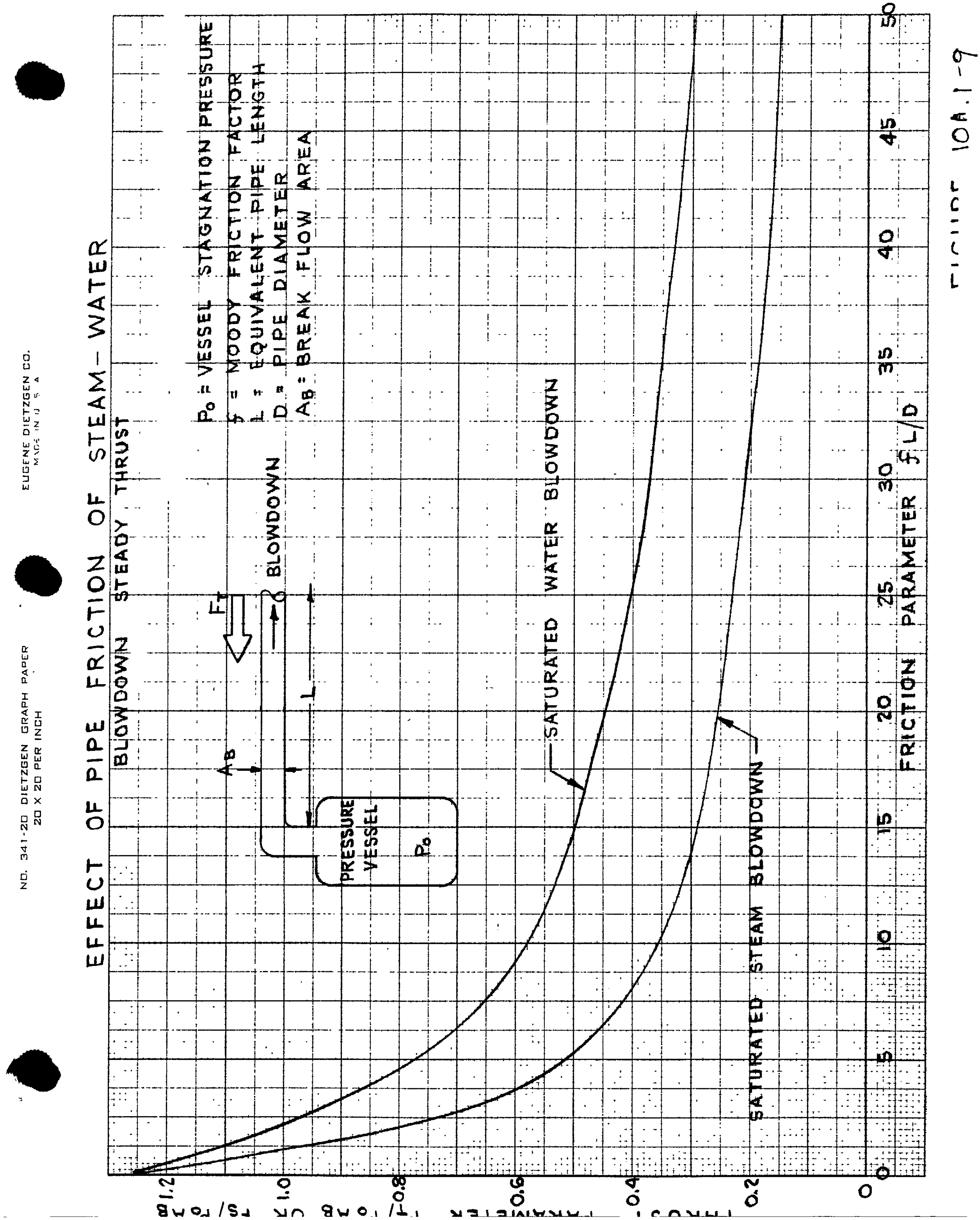
BALTIMORE GAS & ELECTRIC CO. ELECTRIC PRODUCTION PLANT CALVERT CLIFFS UNIT No. 1 & 2	MATHEMATICAL MODEL FOR PIPE WHIP RESTRAINT
BECHTEL ASSOCIATES GAITHERSBURG, MARYLAND	**FIGURE 10A.1-5**


8'-0" SQUARE GOOSENECK
EL 92'-9"
TOC EL 91'-6"
TOC EL 69'-0"
7'-6" Φ STACK
TOC EL 45'-0"
MAIN STEAM COMPARTMENT
SECTION A
SCALE: 1/8" = 1'-0"
9'-0"
4'-6"
4'-6"
EXISTING OPENING TO BE FILLED
4'-0"
4'-0"
8'-0"
6"
℄ 7'-6" Φ VENT STACK
A
PLAN AT EL 45'-0" AND EL 69'-0"
UNIT 1 AS SHOWN
UNIT 2 OPP. HAND
SCALE: 1/8" = 1'-0"
N
℄ PLANT
KEY PLAN
REV
FIG. ADDED 4-13-73
B
BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
VENT STACK FROM MAIN STEAM COMPARTMENT
FIGURE 10A.1-7



BLOWDOWN MODEL

FIGURE 10A. 1-8


EFFECT OF PIPE FRICTION OF STEAM-WATER
BLOWDOWN STEADY THRUST
Po = VESSEL STAGNATION PRESSURE
f = MOODY FRICTION FACTOR
L = EQUIVALENT PIPE LENGTH
D = PIPE DIAMETER
AB = BREAK FLOW AREA
Ft
BLOWDOWN
AB
L
PRESSURE VESSEL
Po
SATURATED WATER BLOWDOWN
SATURATED STEAM BLOWDOWN
FRICTION PARAMETER fL/D
THRUST PARAMETER Ft/PoAB OR Fs/PoAB
0
5
10
15
20
25
30
35
40
45
50
1.2
1.0
0.8
0.6
0.4
0.2
0
FIGURE 10A.1-9
NO. 341-20 DIETZGEN GRAPH PAPER
20 X 20 PER INCH
EUGENE DIETZGEN CO.

EFFECT OF AREA RESTRICTION ON STEAM WATER

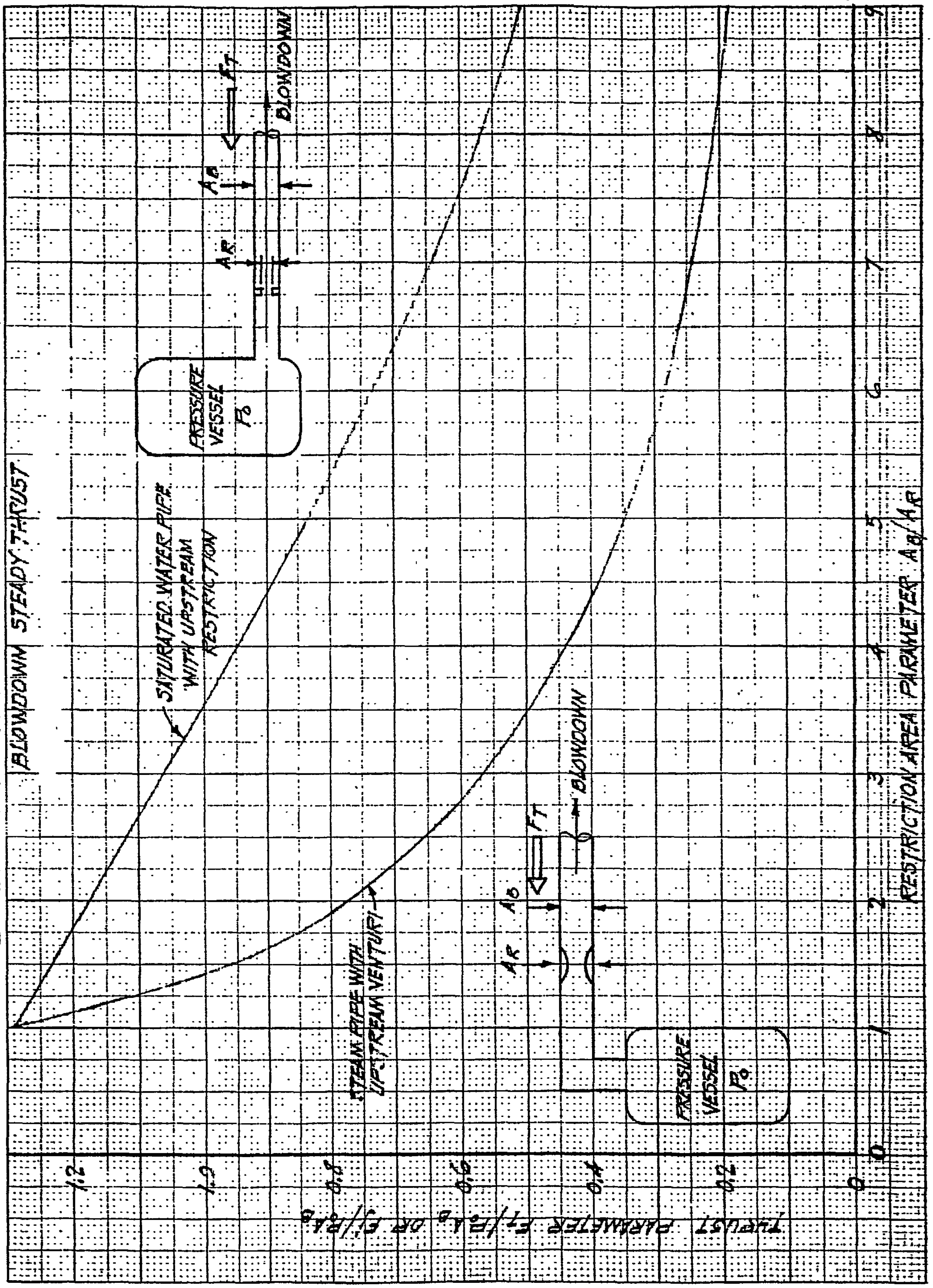


FIGURE 10A.1-10

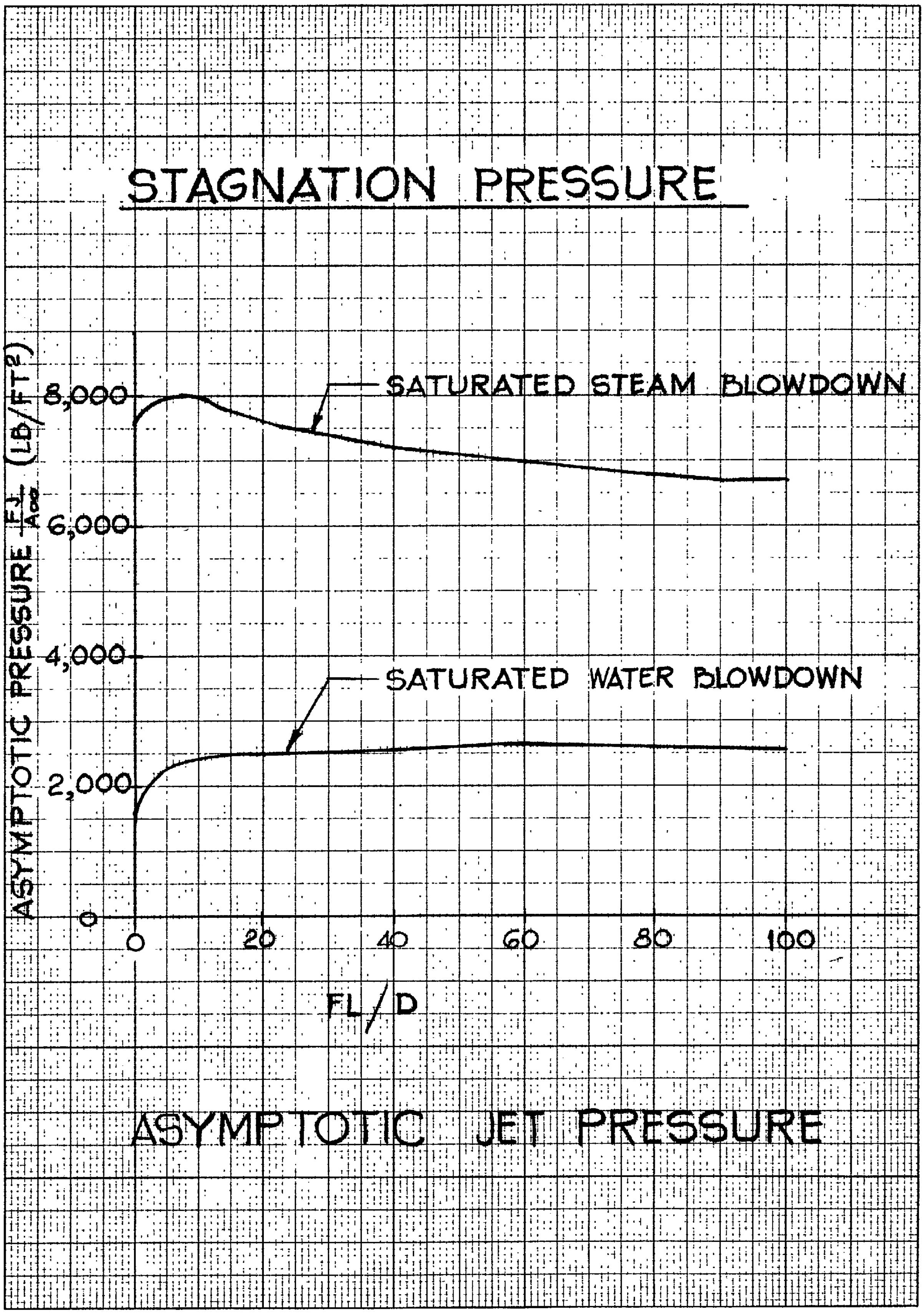

STAGNATION PRESSURE
SATURATED STEAM BLOWDOWN
SATURATED WATER BLOWDOWN
ASYMPTOTIC PRESSURE $\frac{FJ}{A_{\infty}}$ (LB/FT²)
8,000
6,000
4,000
2,000
0
0
20
40
60
80
100
FL/D
ASYMPTOTIC JET PRESSURE

FIG. 10A.1-11

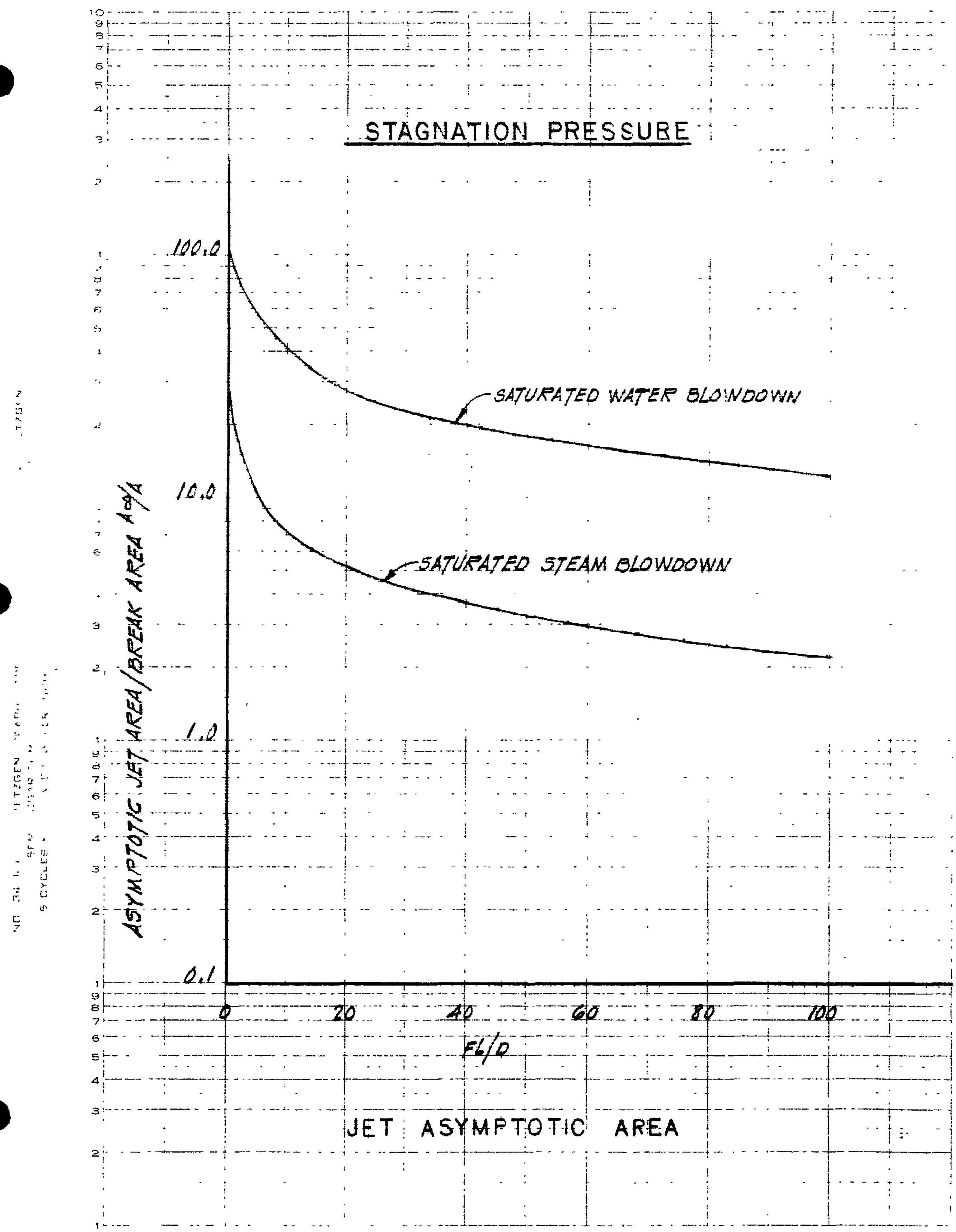


FIGURE 10A.1 – 12

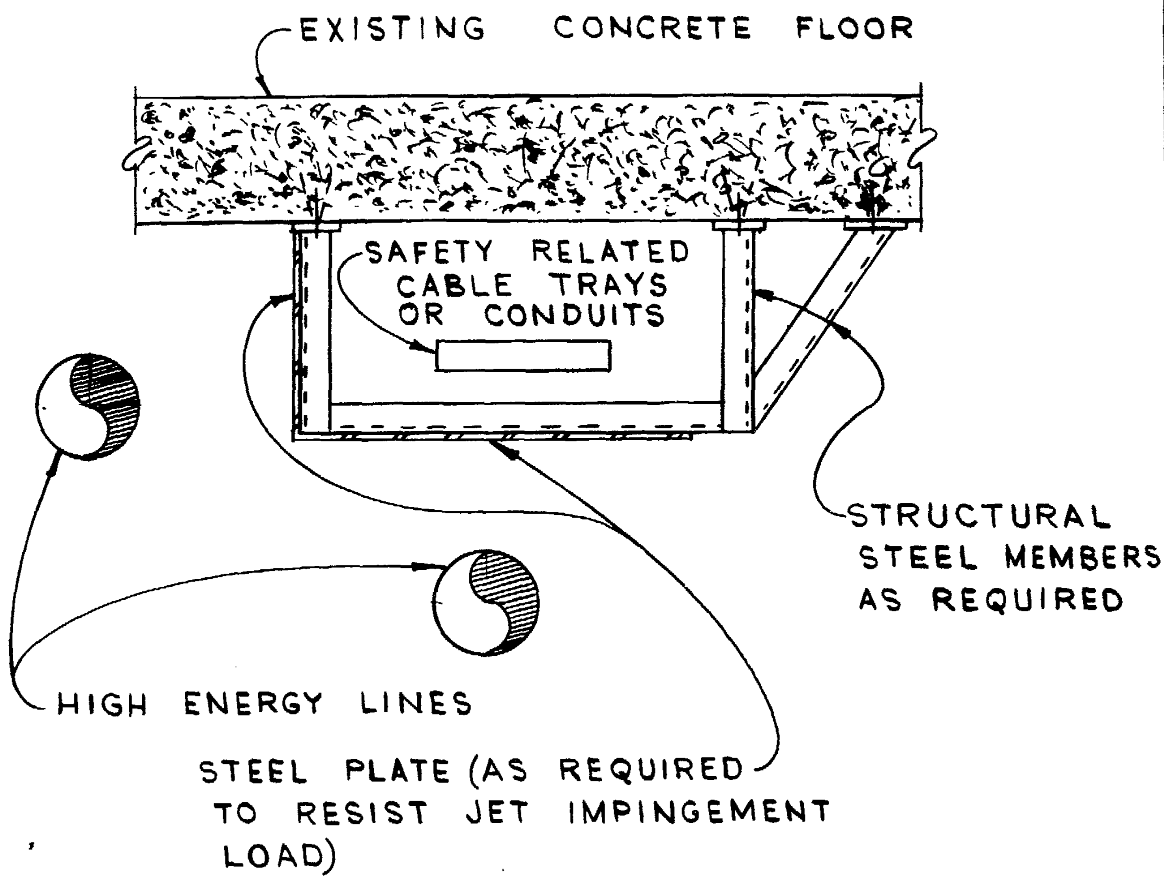
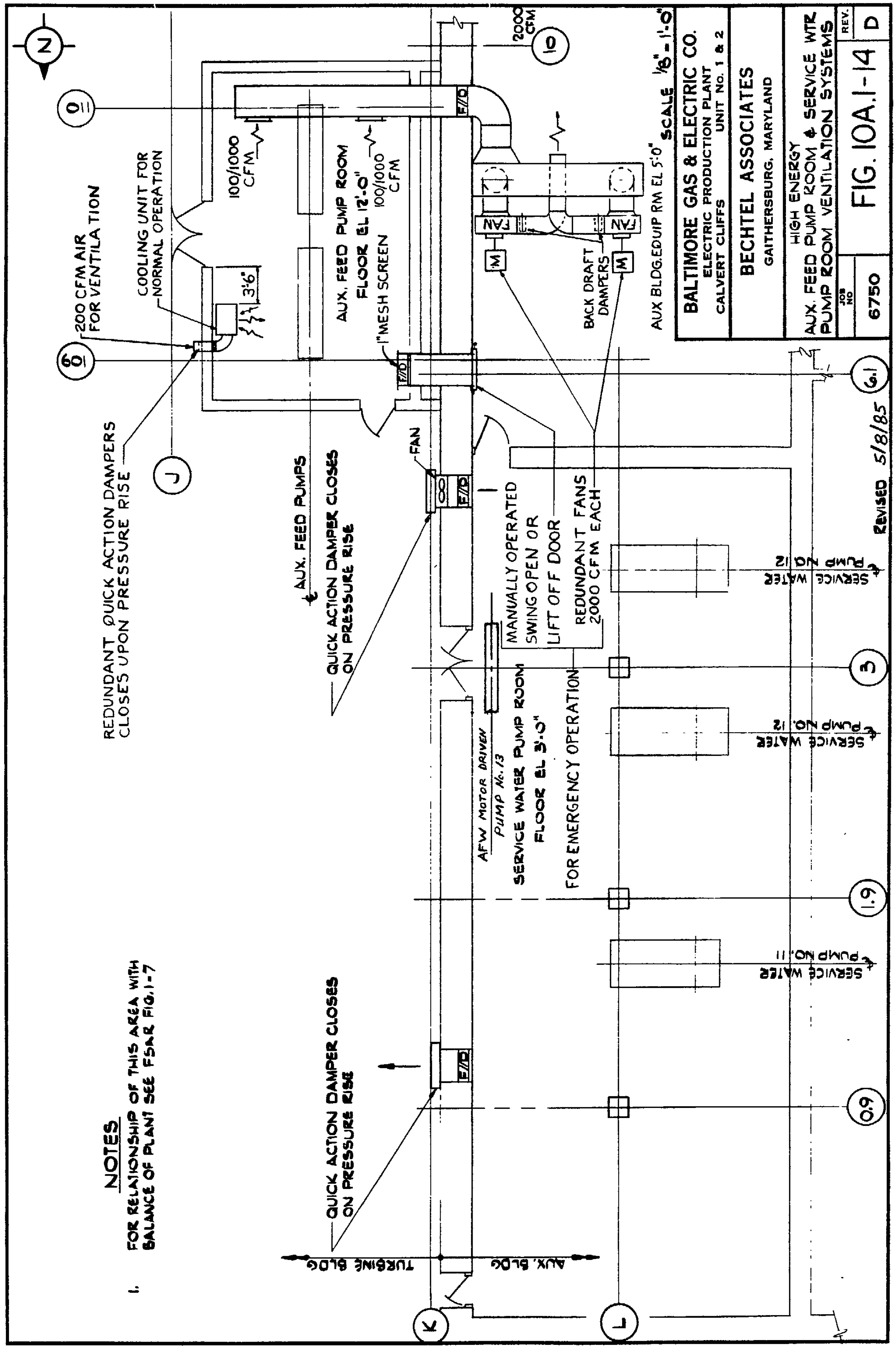
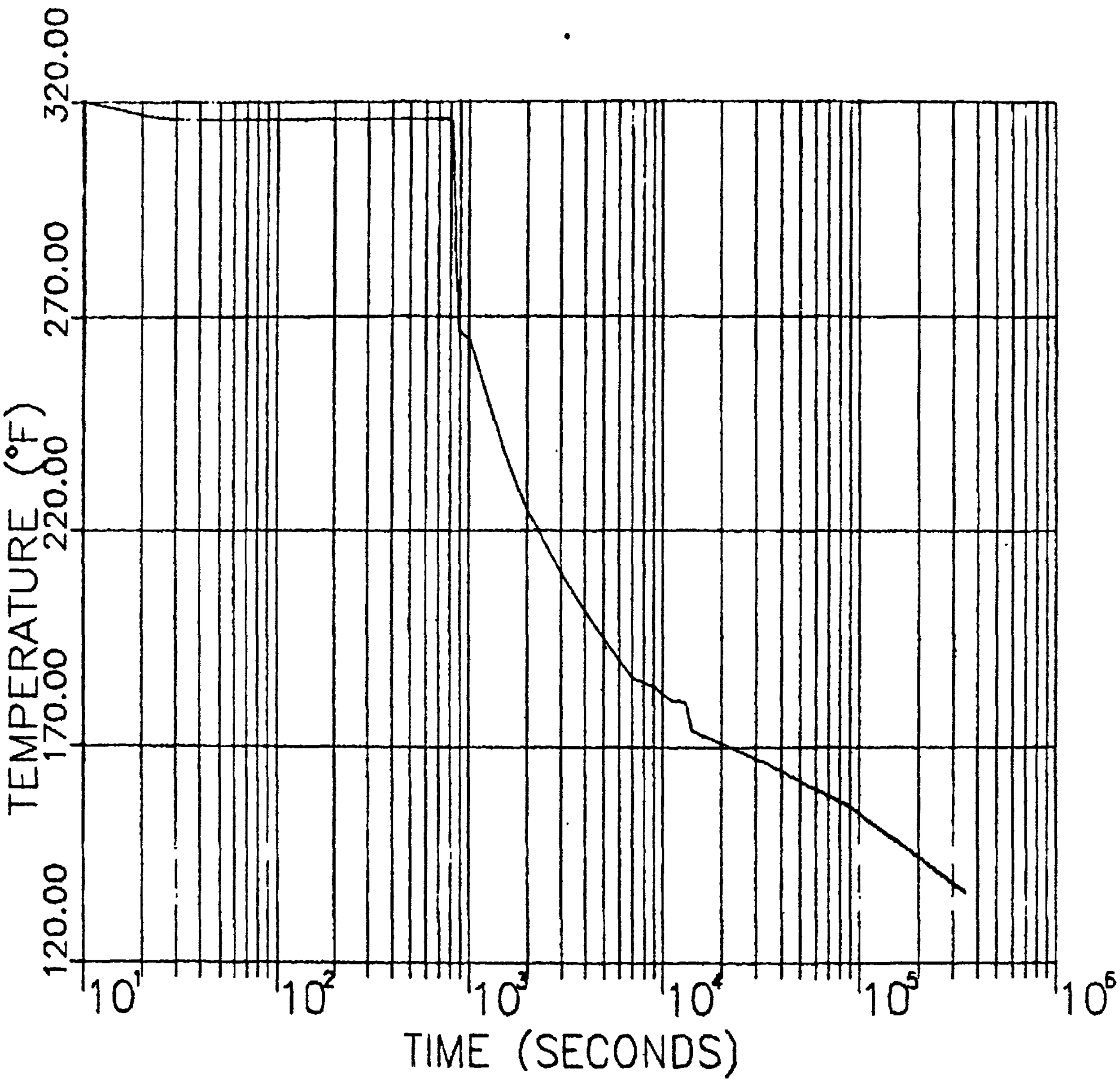


FIGURE ADDED 4-13-73 | REV. B

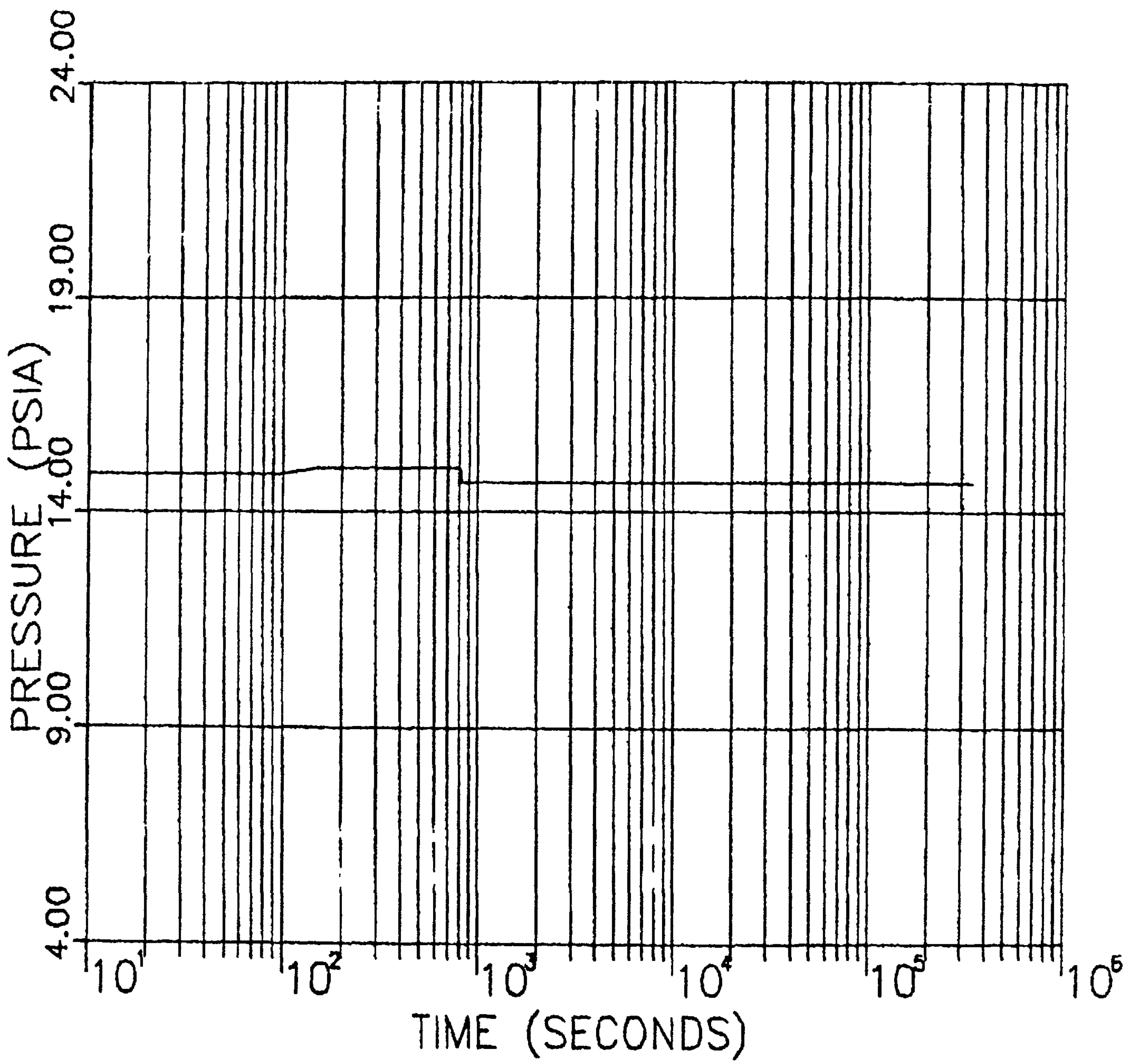
BALTIMORE GAS & ELECTRIC CO. ELECTRIC PRODUCTION PLANT CALVERT CLIFFS UNIT No. 1 & 2	TYP. JET IMPINGEMENT BARRIER FOR CABLE TRAYS
BECHTEL ASSOCIATES GAITHERSBURG, MARYLAND	FIGURE 10A.1-13



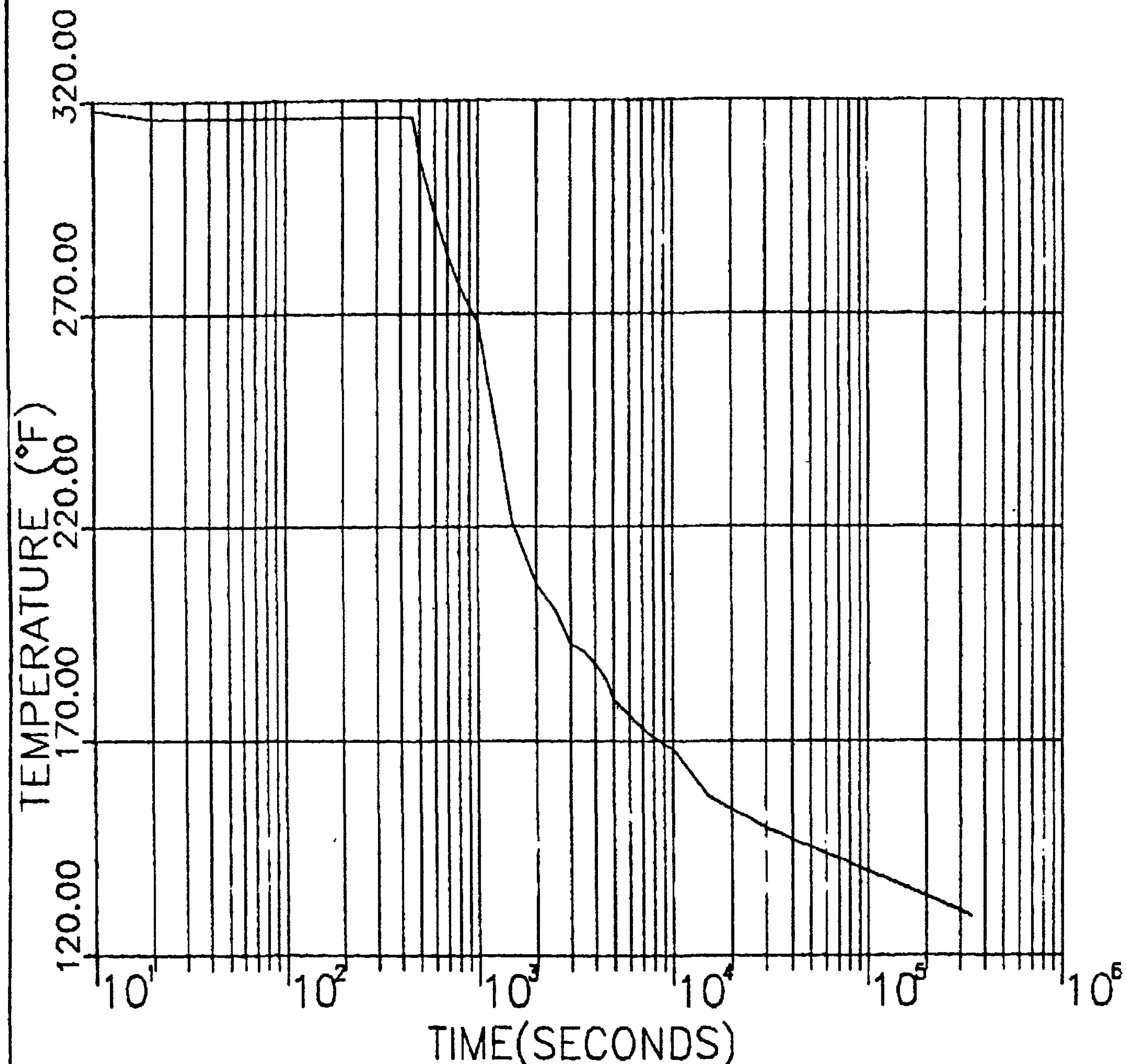
N
2000 CFM
10
100/1000 CFM
COOLING UNIT FOR NORMAL OPERATION
200 CFM AIR FOR VENTILATION
3'6"
AUX. FEED PUMP ROOM FLOOR EL 12'-0"
100/1000 CFM
1" MESH SCREEN
F/D
FAN
M
BACK DRAFT DAMPERS
AUX BLDG. EQUIP RM EL 5'-0"
SCALE 1/8" = 1'-0"
BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
HIGH ENERGY
AUX. FEED PUMP ROOM & SERVICE WTR PUMP ROOM VENTILATION SYSTEMS
JOB NO 6750
FIG. 10A.1-14
REV. D
REDUNDANT QUICK ACTION DAMPERS CLOSES UPON PRESSURE RISE
J
AUX. FEED PUMPS
QUICK ACTION DAMPER CLOSES ON PRESSURE RISE
MANUALLY OPERATED SWING OPEN OR LIFT OFF DOOR
REDUNDANT FANS 2000 CFM EACH
REVISED 5/8/85
6.1
SERVICE WATER PUMP NO. 12
3
AFW MOTOR DRIVEN PUMP No. 13
SERVICE WATER PUMP ROOM FLOOR EL 3'-0"
FOR EMERGENCY OPERATION
SERVICE WATER PUMP NO. 12
1.9
SERVICE WATER PUMP NO. 11
0.9
NOTES
1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT SEE FSAR FIG. 1-7
QUICK ACTION DAMPER CLOSES ON PRESSURE RISE
TURBINE BLDG
AUX. BLDG
K
L



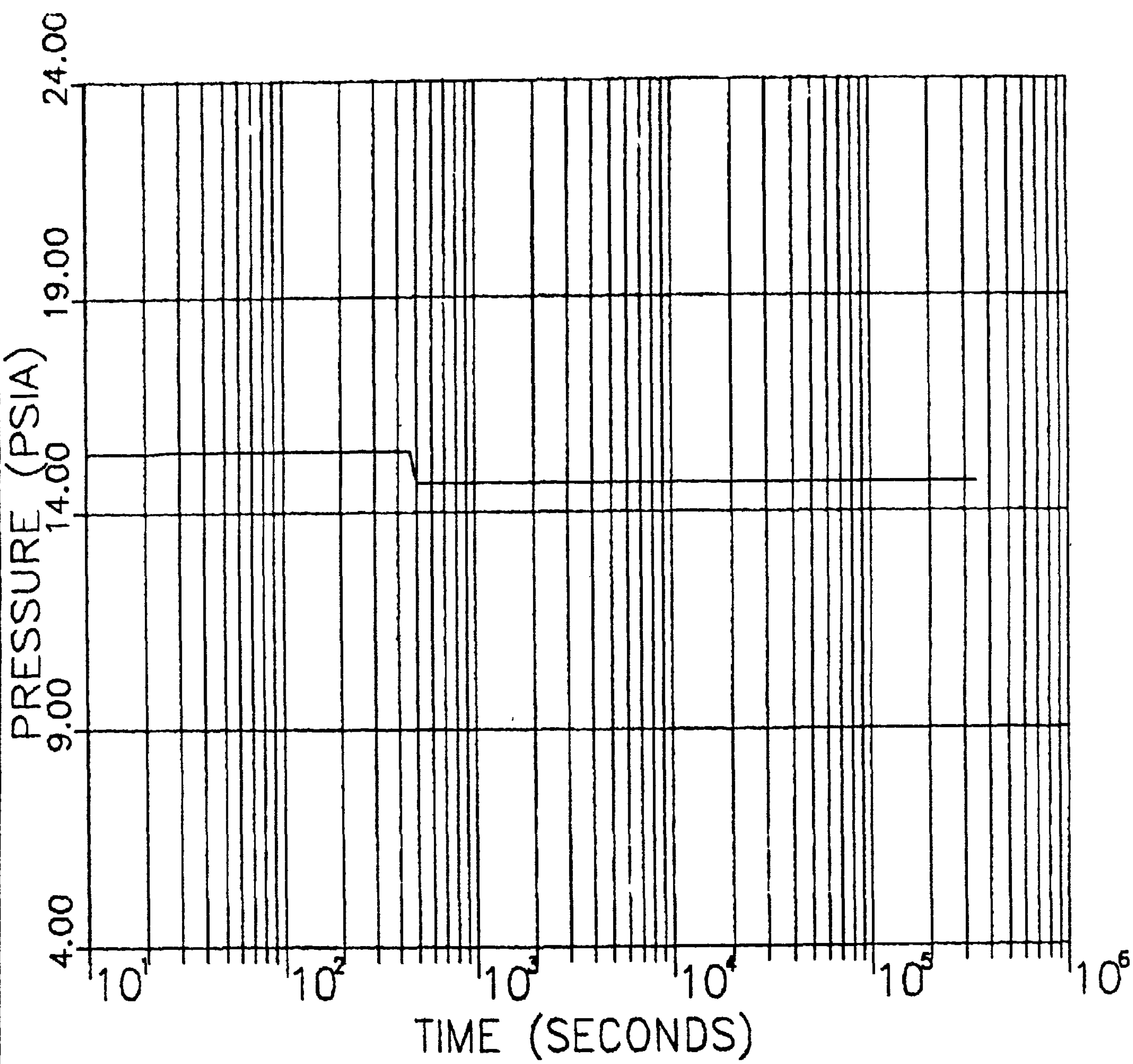
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	6" STEAM LINE BREAK MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-15



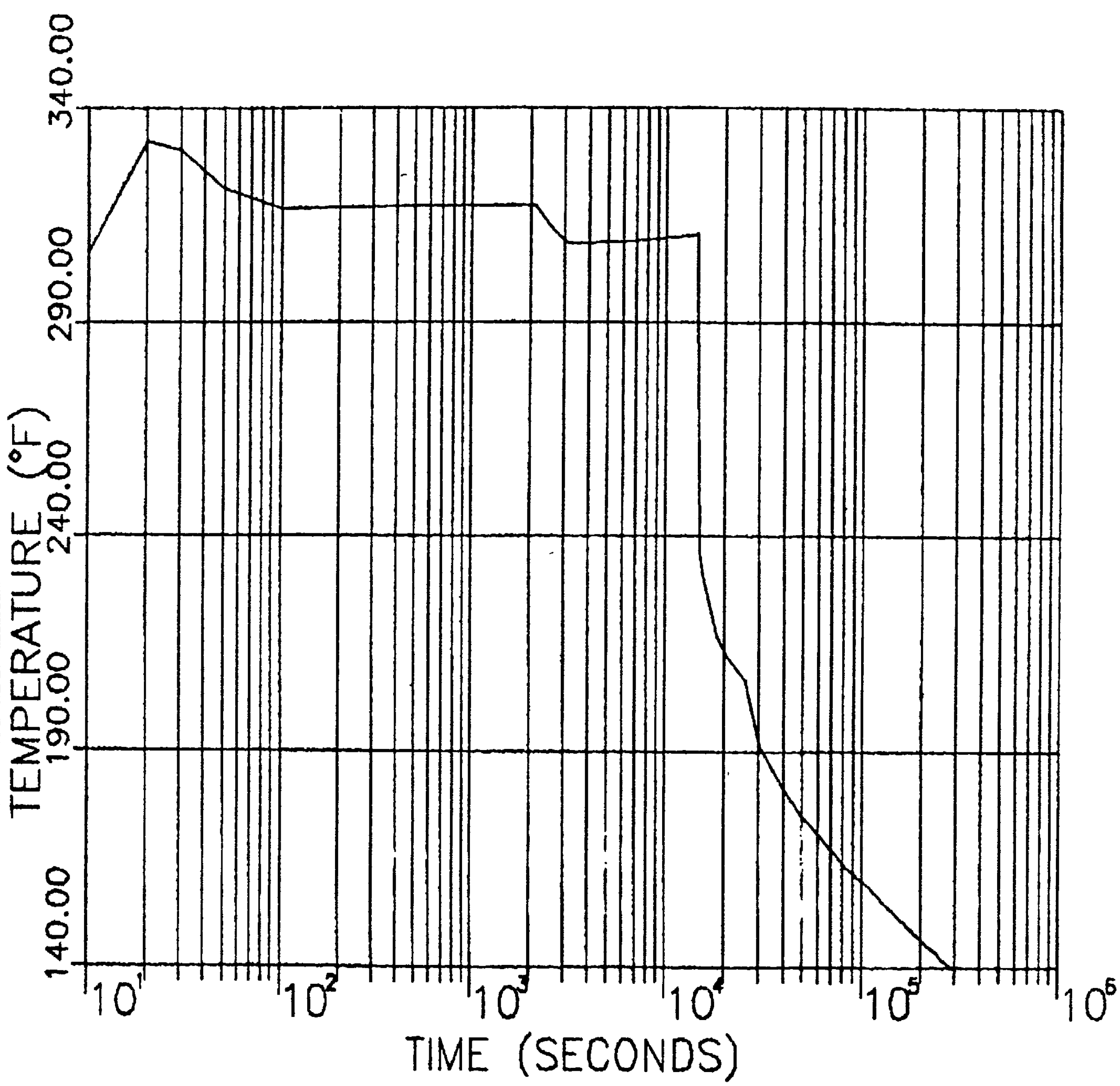
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	6" STEAM LINE BREAK MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-16



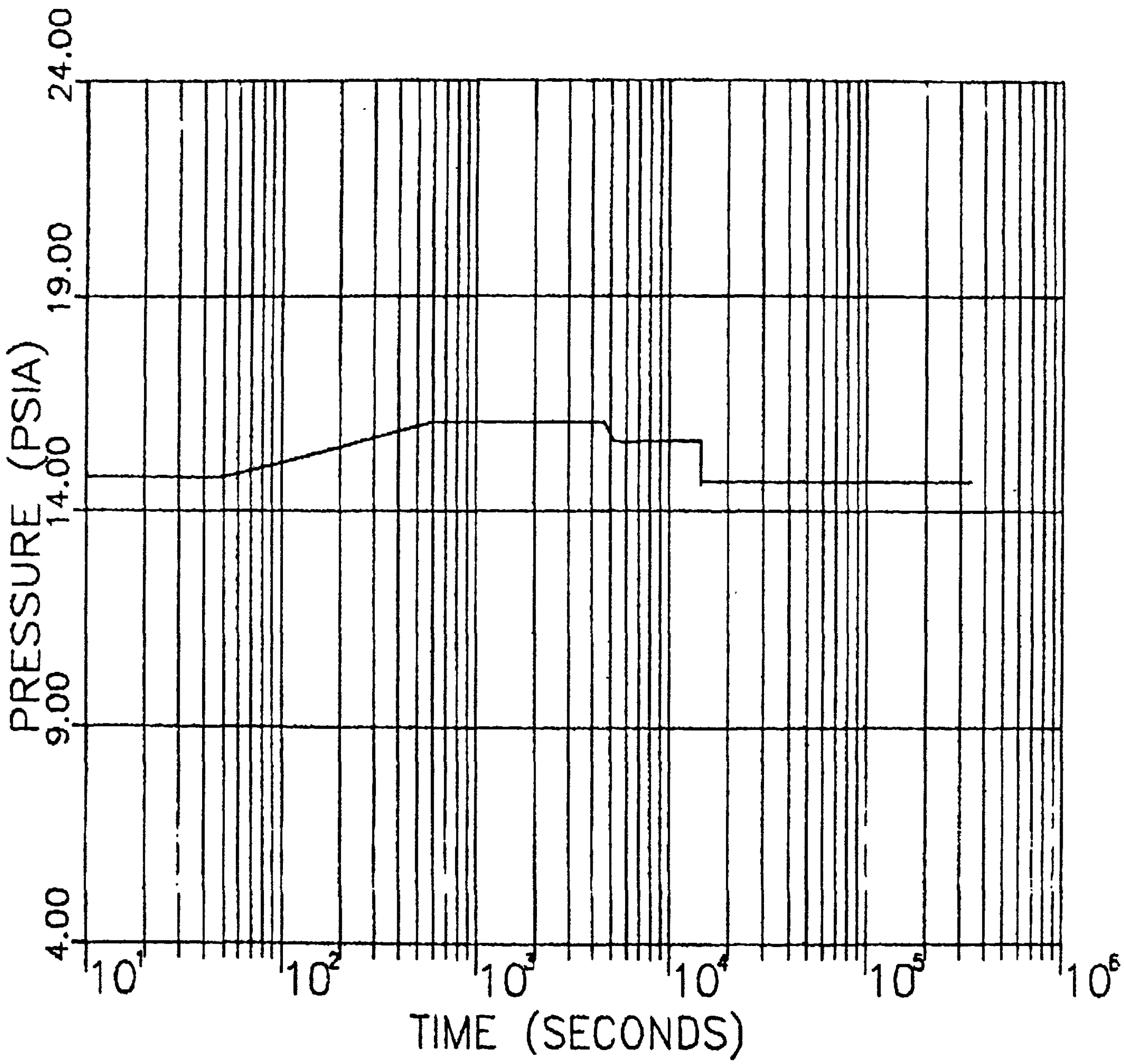
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BREAK OF ENCAPSULATED STEAM LINE MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-17



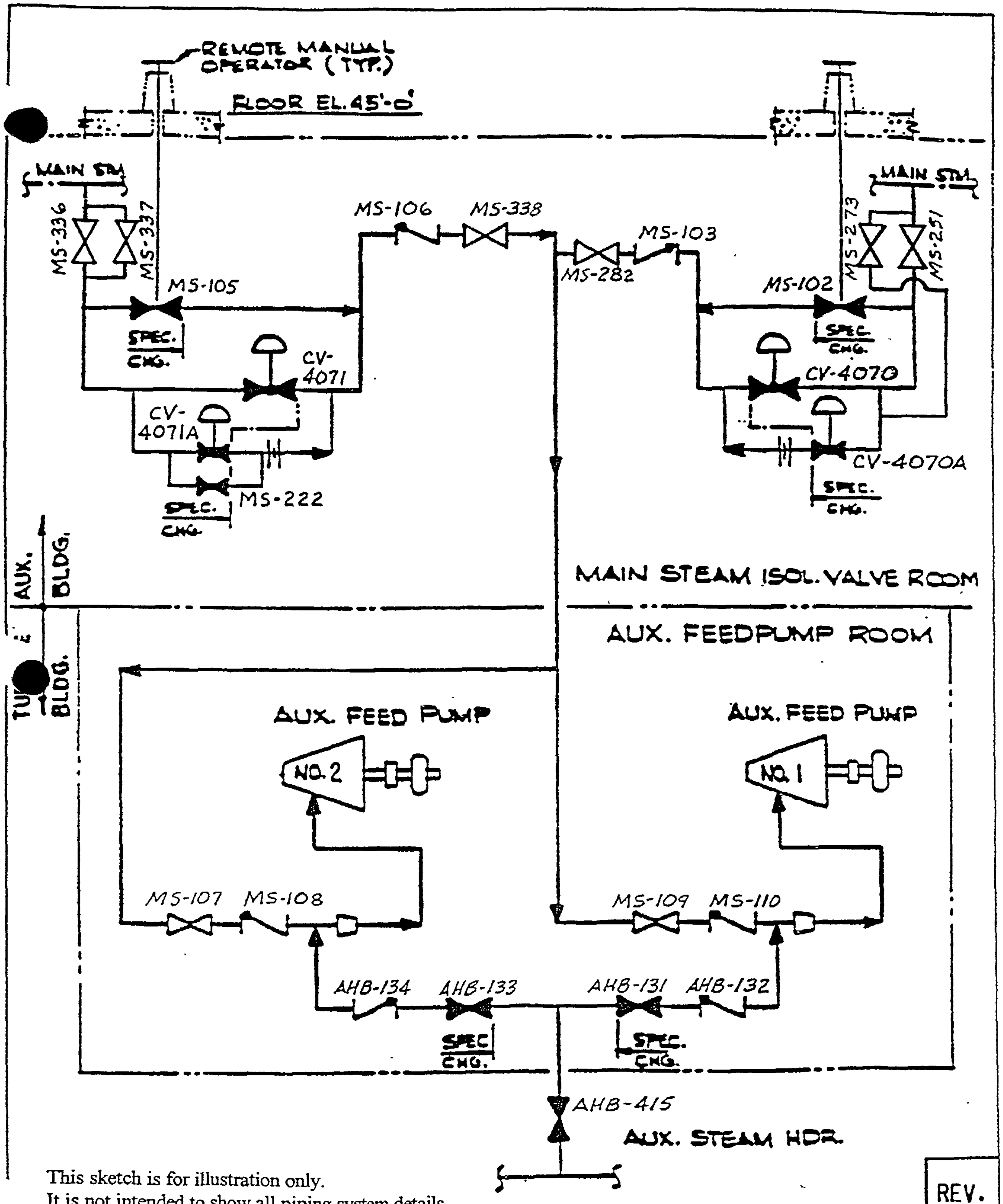
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	BREAK OF ENCAPSULATED STEAM LINE MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-18



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CRITICAL CRACK IN MAIN STEAM LINE MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-19



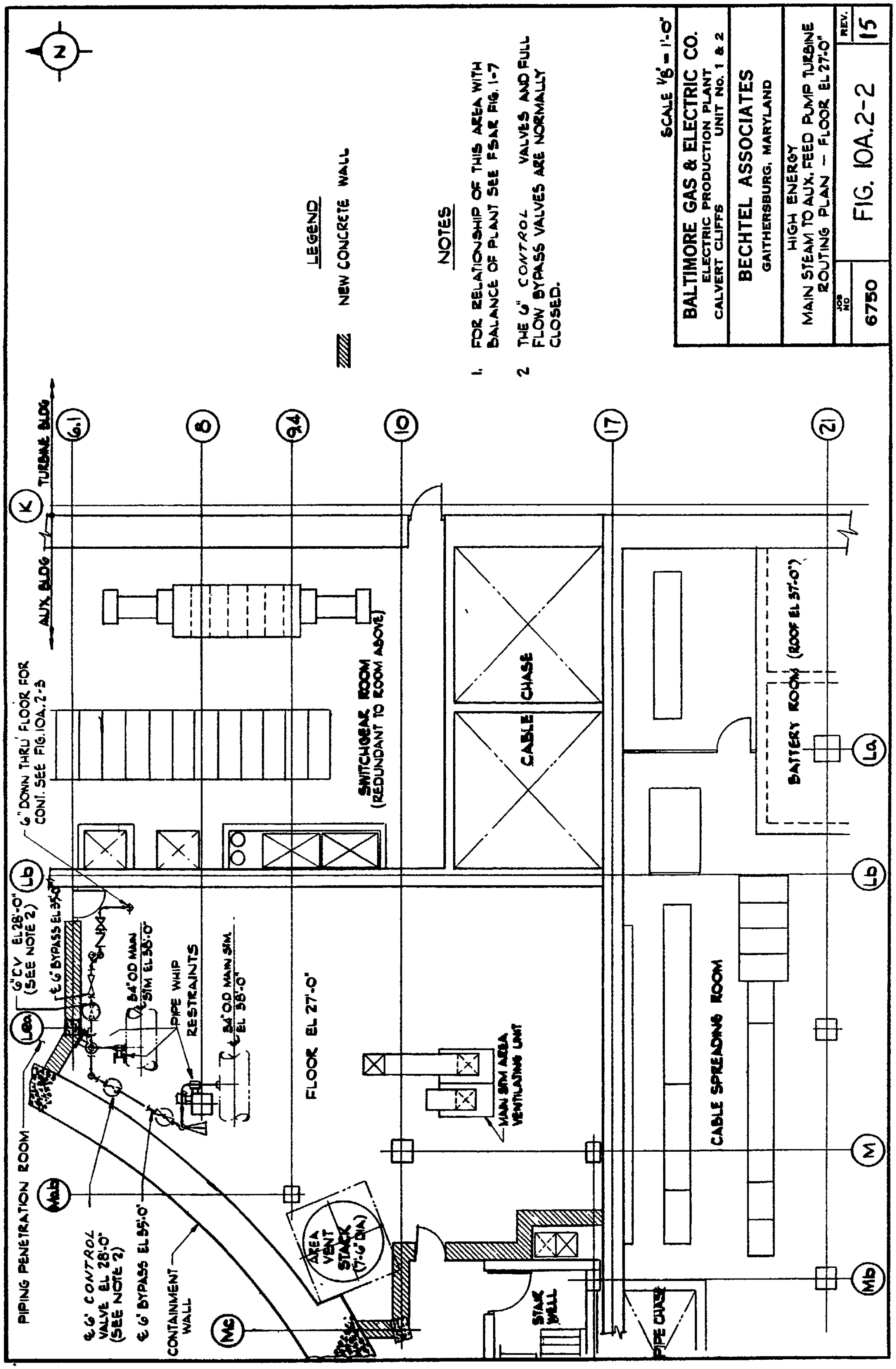
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CRITICAL CRACK IN MAIN STEAM LINE MAIN STEAM PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.1-20

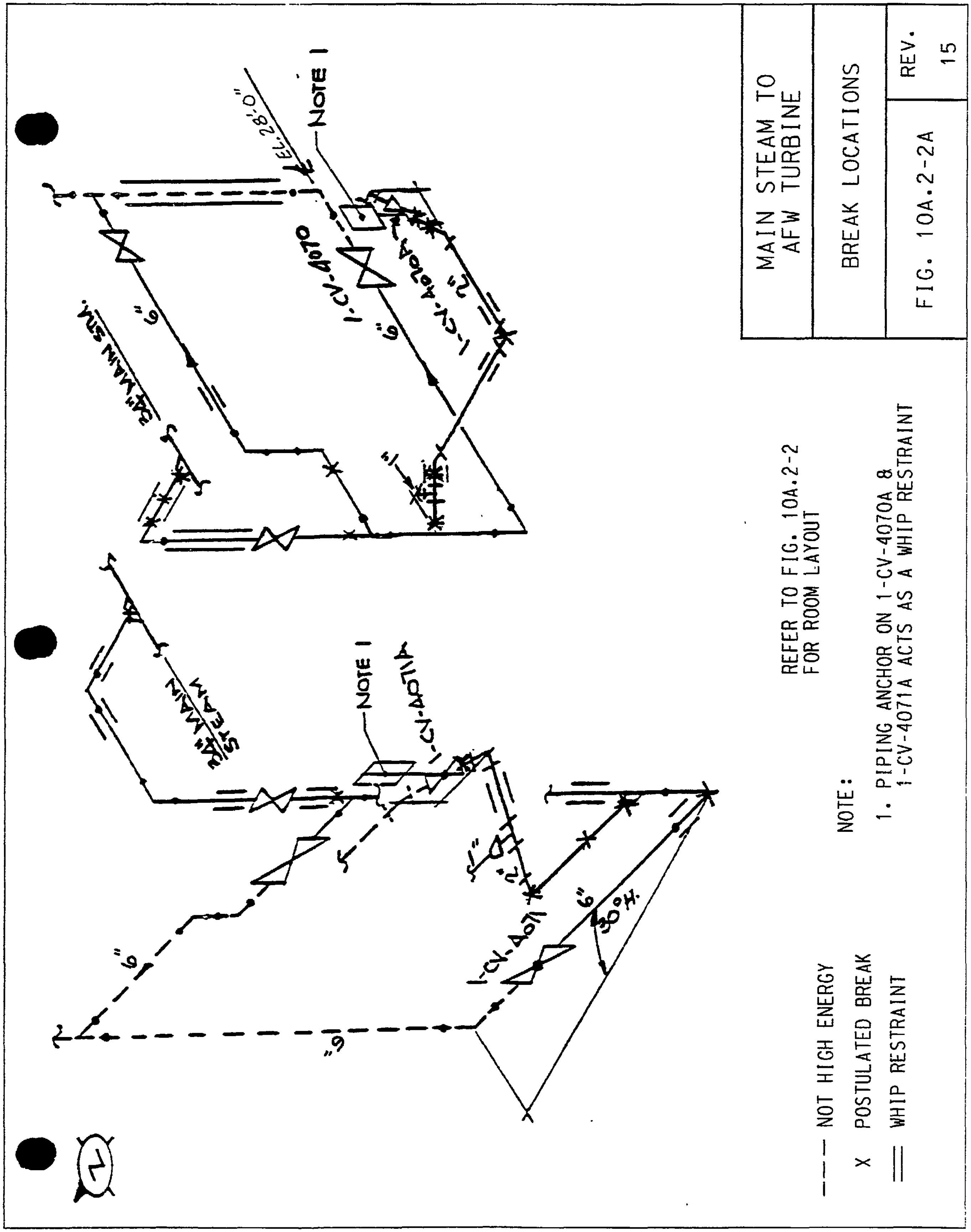


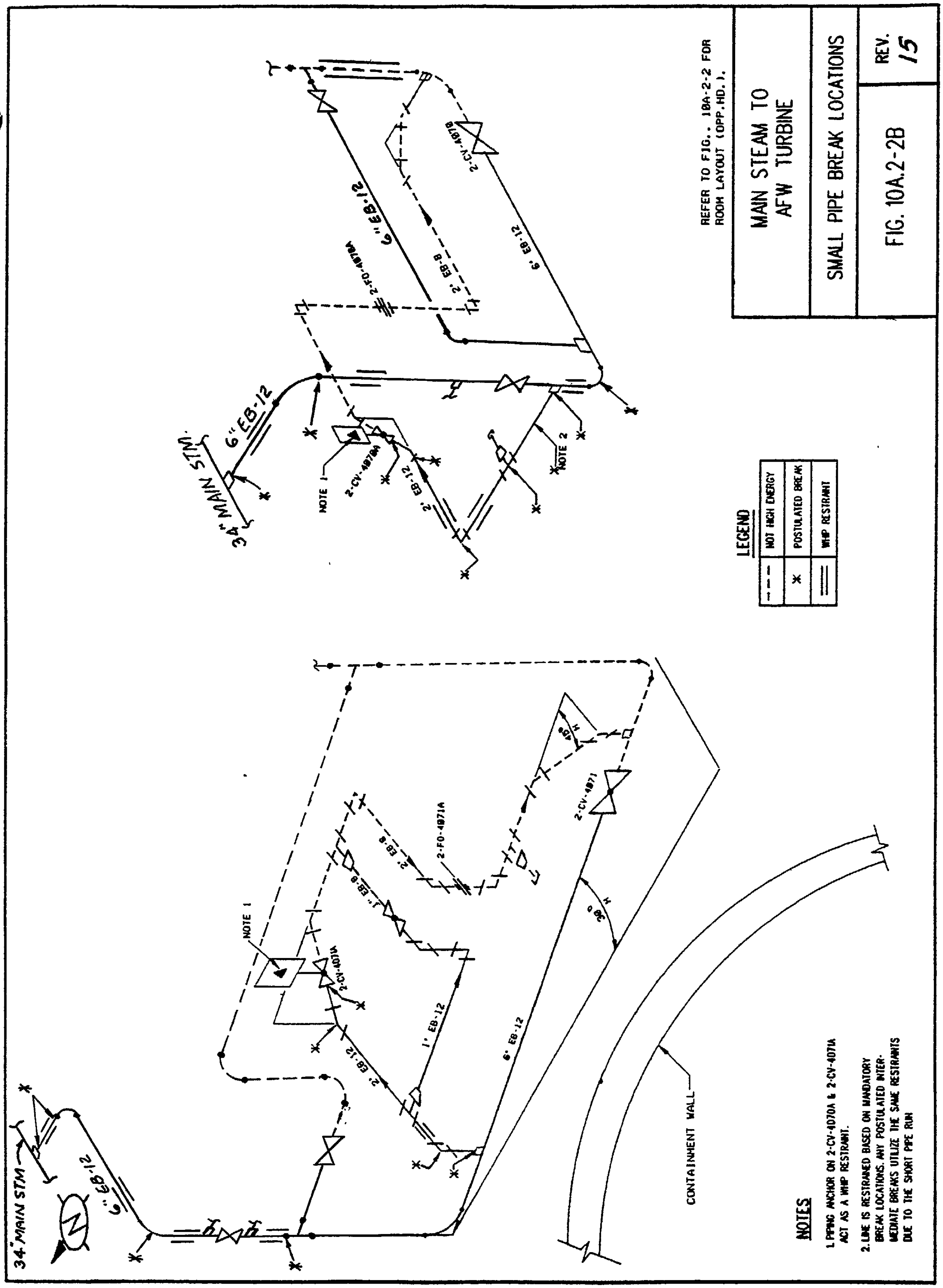
This sketch is for illustration only.
It is not intended to show all piping system details.

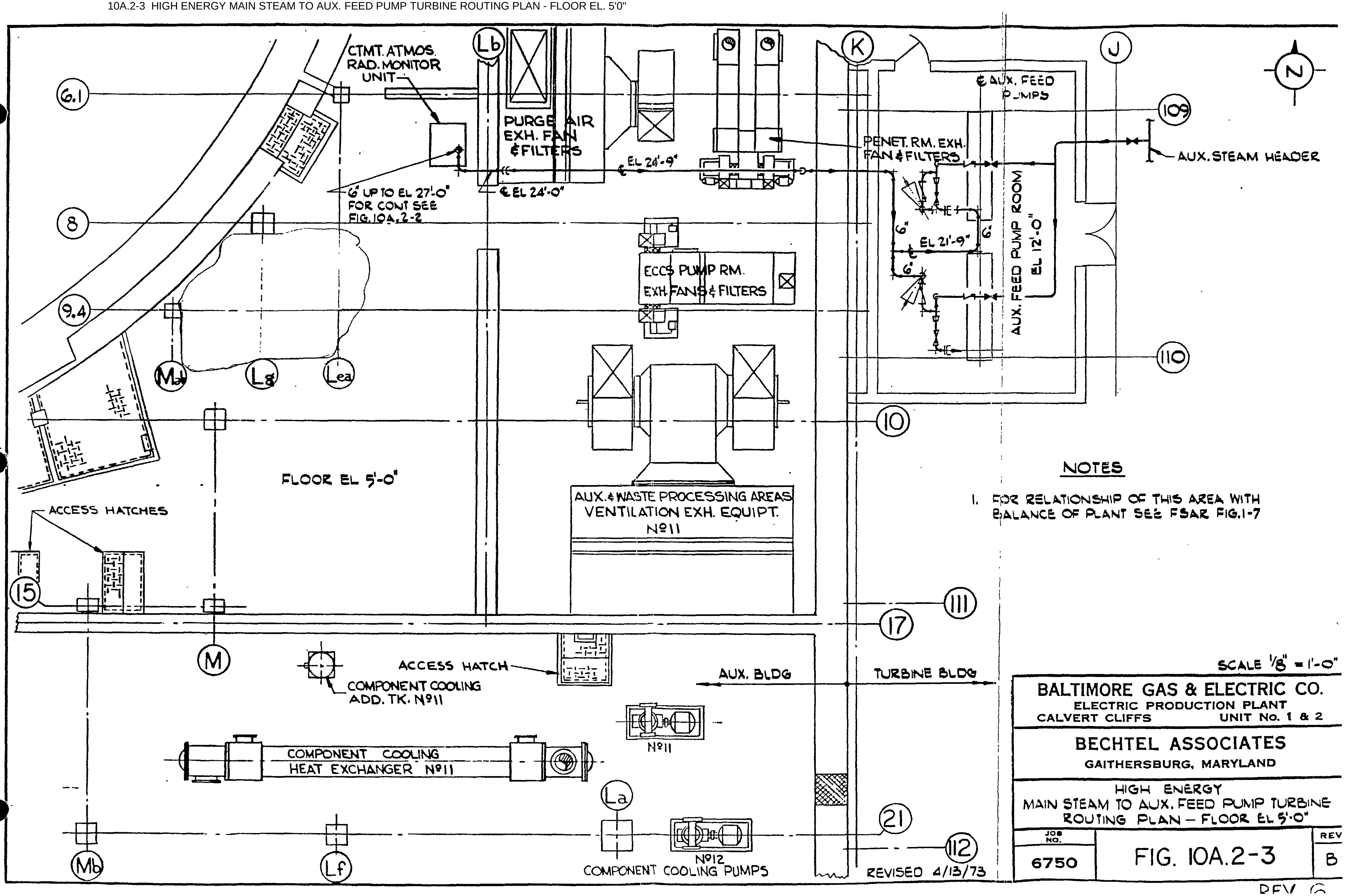
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	MAIN STEAM TO AUX. STEAM GEN. FEED PUMP TURBINE	FIG. 10A.2-1

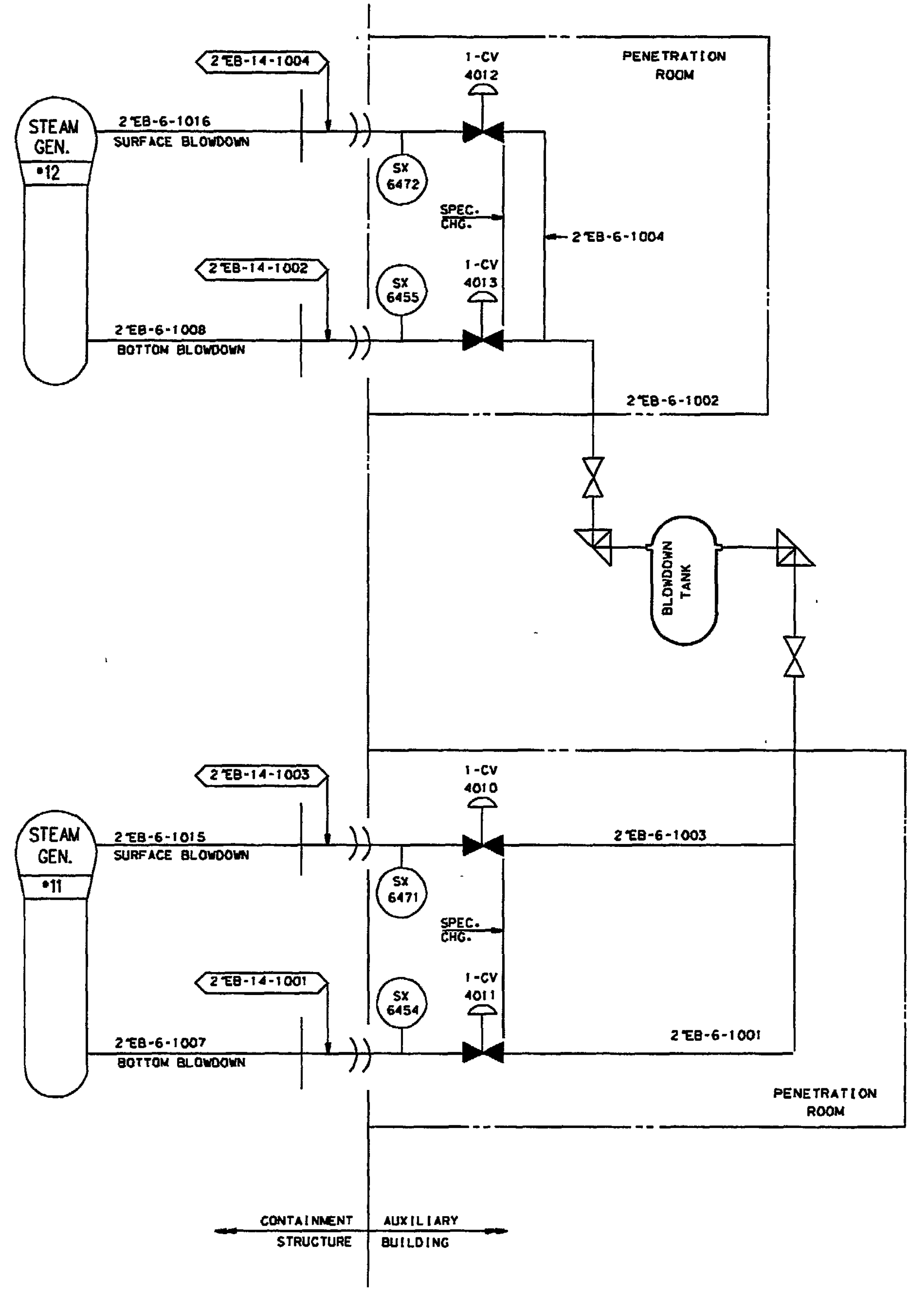
REV. 20


LEGEND
NEW CONCRETE WALL
NOTES
1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT SEE FSAR FIG. 1-7
2. THE 6" CONTROL VALVES AND FULL FLOW BYPASS VALVES ARE NORMALLY CLOSED.
SCALE 1/8" = 1'-0"
BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
HIGH ENERGY
MAIN STEAM TO AUX. FEED PUMP TURBINE
ROUTING PLAN — FLOOR EL 27'-0"
JOB NO
6750
FIG. 10A.2-2
REV.
15
TURBINE BLDG
AUX. BLDG
SWITCHGEAR ROOM
(REDUNDANT TO ROOM ABOVE)
CABLE CHASE
BATTERY ROOM (ROOF EL 37'-0")
CABLE SPREADING ROOM
FLOOR EL 27'-0"
PIPING PENETRATION ROOM
CONTAINMENT WALL
AREA VENT STACK (7'-6" DIA)
STAIR WELL
PIPE CHASE
MAIN STM AREA VENTILATING UNIT
PIPE WHIP RESTRAINTS
6" DOWN THRU FLOOR FOR CONT. SEE FIG. 10A.2-3
6" CV EL 28'-0" (SEE NOTE 2)
℄ 6" BYPASS EL 35'-0"
34" OD MAIN STM EL 38'-0"
℄ 6" CONTROL VALVE EL 28'-0" (SEE NOTE 2)
℄ 6" BYPASS EL 35'-0"


24" MAIN STM.
6"
1-CV-4070
1-CV-4070A
2"
NOTE 1
EL. 28'-0"
1"
24" MAIN STEAM
NOTE 1
1-CV-4071A
1-CV-4071
30°H.
MAIN STEAM TO AFW TURBINE
BREAK LOCATIONS
FIG. 10A.2-2A
REV. 15
REFER TO FIG. 10A.2-2 FOR ROOM LAYOUT
NOTE:
1. PIPING ANCHOR ON 1-CV-4070A & 1-CV-4071A ACTS AS A WHIP RESTRAINT
NOT HIGH ENERGY
X POSTULATED BREAK
WHIP RESTRAINT

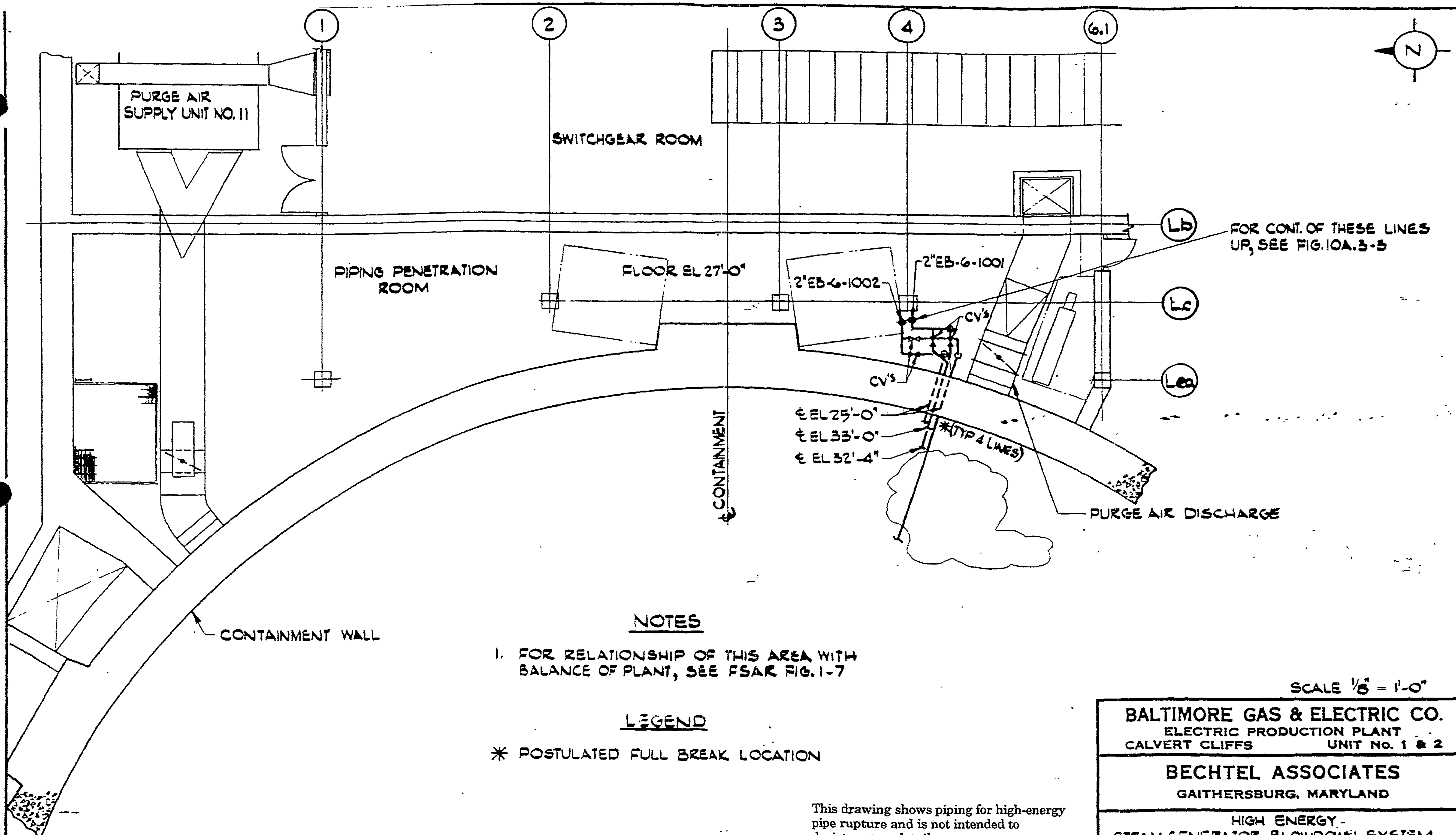

MAIN STEAM TO AFW TURBINE
SMALL PIPE BREAK LOCATIONS
FIG. 10A.2-2B
REV. 15
REFER TO FIG. 10A-2-2 FOR ROOM LAYOUT (OPP.HD.).
LEGEND
NOT HIGH ENERGY
POSTULATED BREAK
WHP RESTRAINT
NOTES
1. PIPING ANCHOR ON 2-CV-4070A & 2-CV-4071A ACT AS A WHP RESTRAINT.
2. LINE IS RESTRAINED BASED ON MANDATORY BREAK LOCATIONS. ANY POSTULATED INTER-MEDIATE BREAKS UTILIZE THE SAME RESTRAINTS DUE TO THE SHORT PIPE RUN
34" MAIN STM
6" EB-12
2" EB-12
1" EB-12
2" EB-8
1" EB-8
2-CV-4070A
2-CV-4071
2-CV-4071A
2-CV-4071B
2-FO-4071A
NOTE 1
NOTE 2
CONTAINMENT WALL


CTMT. ATMOS. RAD. MONITOR UNIT
PURGE AIR EXH. FAN & FILTERS
6" UP TO EL 27'-0" FOR CONT SEE FIG. 10A.2-2
℄ EL 24'-0"
EL 24'-9"
PENET. RM. EXH. FAN & FILTERS
℄ AUX. FEED PUMPS
AUX. STEAM HEADER
AUX. FEED PUMP ROOM EL 12'-0"
EL 21'-9"
ECCS PUMP RM. EXH. FANS & FILTERS
FLOOR EL 5'-0"
ACCESS HATCHES
AUX. & WASTE PROCESSING AREAS VENTILATION EXH. EQUIPT. Nº11
ACCESS HATCH
COMPONENT COOLING ADD. TK. Nº11
AUX. BLDG
TURBINE BLDG
COMPONENT COOLING HEAT EXCHANGER Nº11
Nº11
Nº12
COMPONENT COOLING PUMPS
REVISED 4/13/73
NOTES
1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT SEE FSAR FIG. 1-7
SCALE 1/8" = 1'-0"
BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
HIGH ENERGY MAIN STEAM TO AUX. FEED PUMP TURBINE ROUTING PLAN – FLOOR EL 5'-0"
JOB NO. 6750
FIG. 10A.2-3
REV B
REV 6



THIS DRAWING SHOWS PIPING FOR HIGH-ENERGY PIPE RUPTURE AND IS NOT INTENDED TO DEPICT SYSTEM DETAILS.

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR BLOWDOWN	REV. 24	Figure 10A.3-1



NOTES

1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT, SEE FSAR FIG. 1-7

LEGEND

* POSTULATED FULL BREAK LOCATION

SCALE 1/8" = 1'-0"

BALTIMORE GAS & ELECTRIC CO. ELECTRIC PRODUCTION PLANT CALVERT CLIFFS UNIT No. 1 & 2		
BECHTEL ASSOCIATES GAITHERSBURG, MARYLAND		
HIGH ENERGY- STEAM GENERATOR BLOWDOWN SYSTEM ROUTING PLAN – FLOOR EL 27'-0"		
JOB NO. 6750	FIG. 10A.3-2	REV. B

This drawing shows piping for high-energy pipe rupture and is not intended to depict system details.

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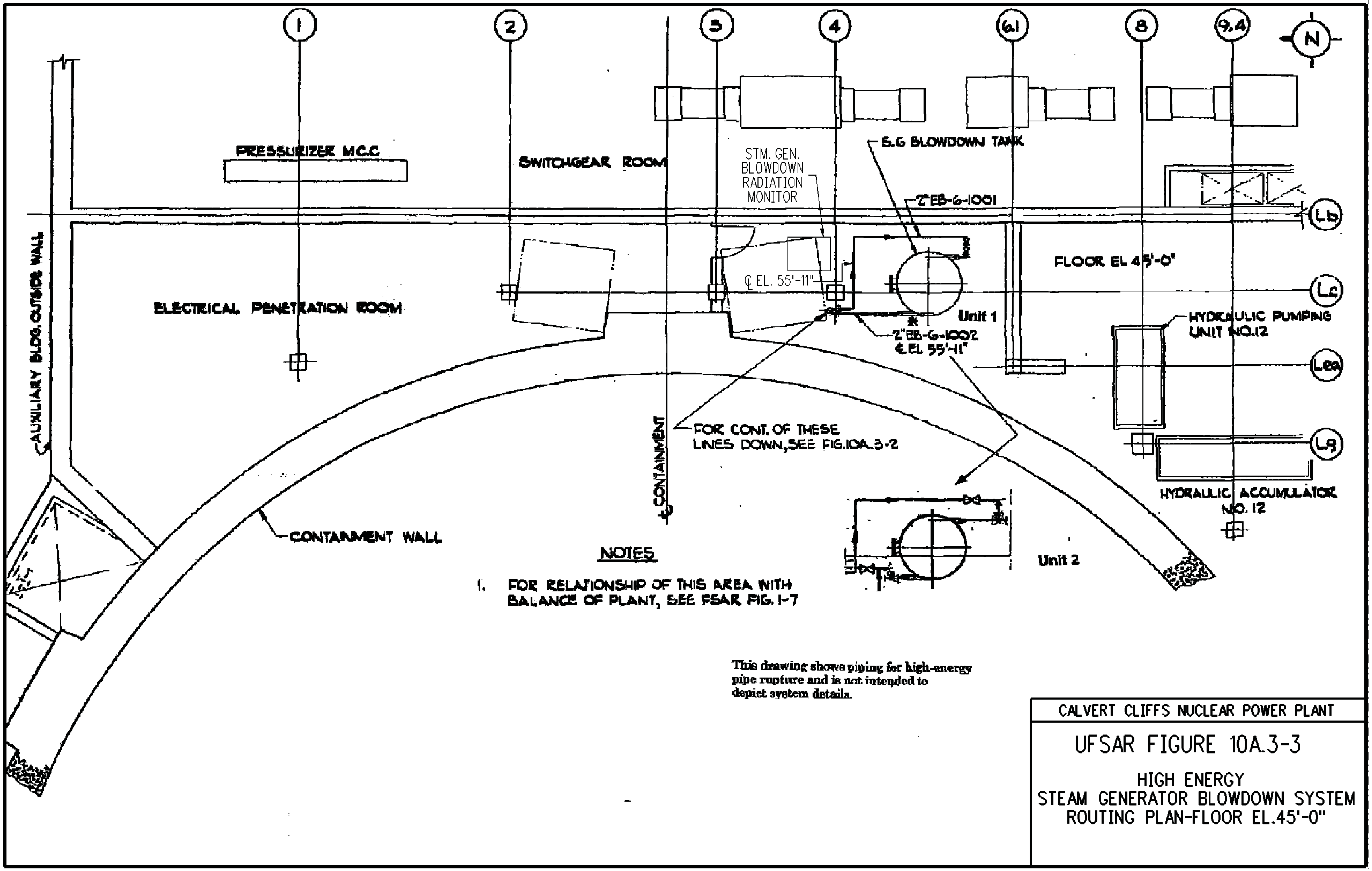
FIGURE 10A.3-3 HIGH ENERGY STEAM GENERATOR BLOWDOWN SYSTEM ROUTING PLAN – FLOOR EL. 45'0"

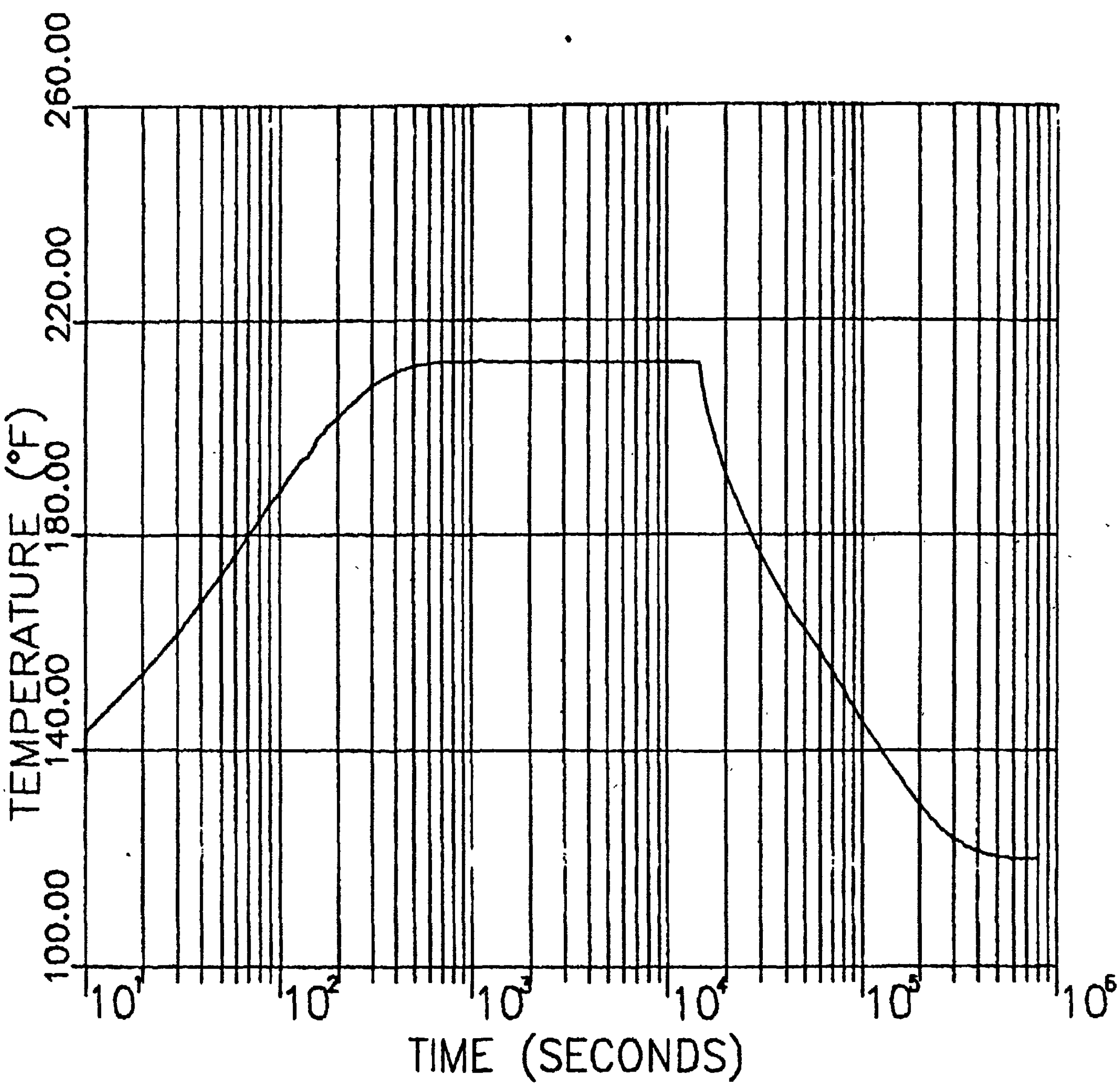


CALVERT CLIFFS NUCLEAR POWER PLANT

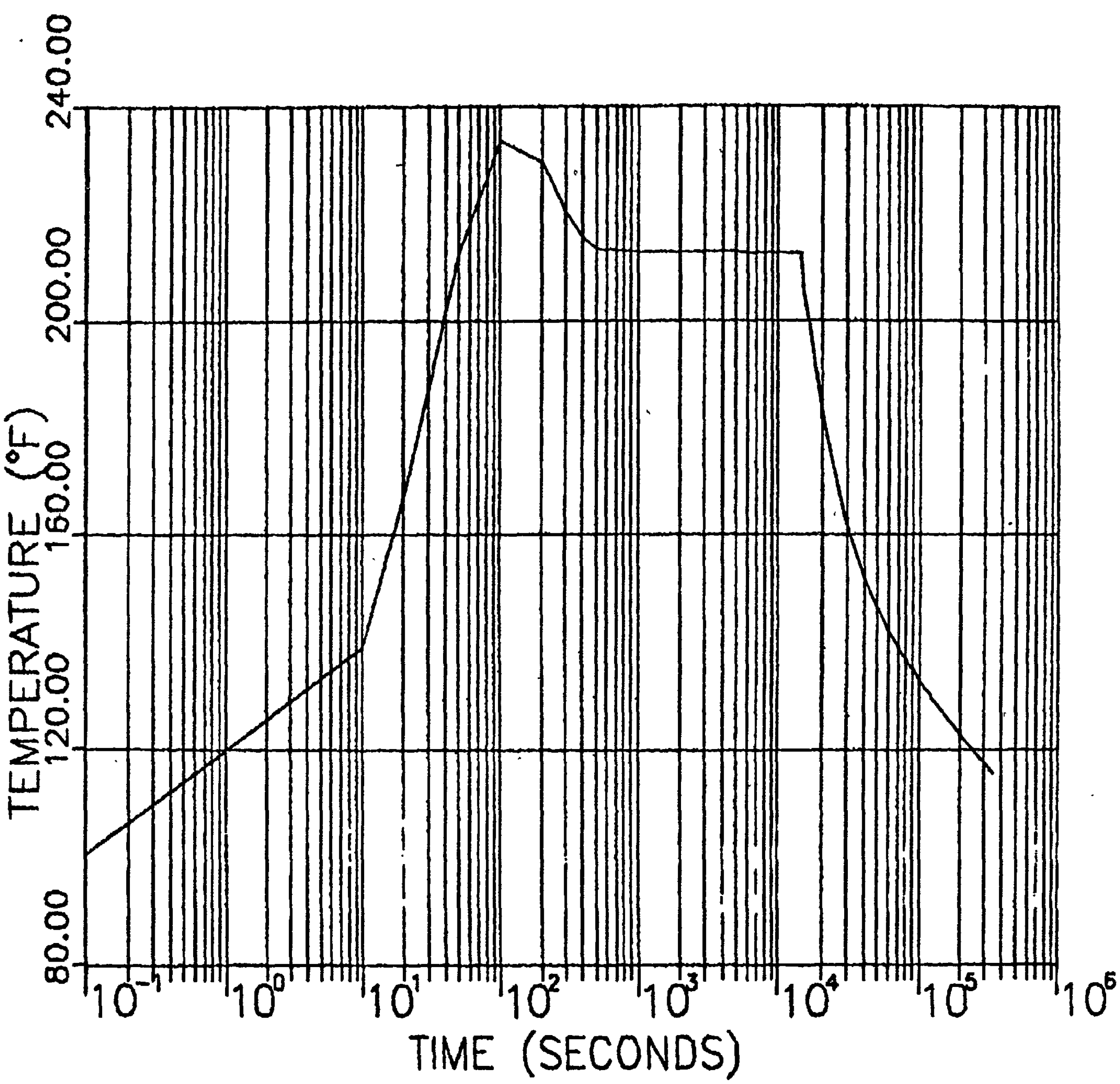
UFSAR FIGURE 10A.3-3

HIGH ENERGY
STEAM GENERATOR BLOWDOWN SYSTEM
ROUTING PLAN-FLOOR EL.45'-0"

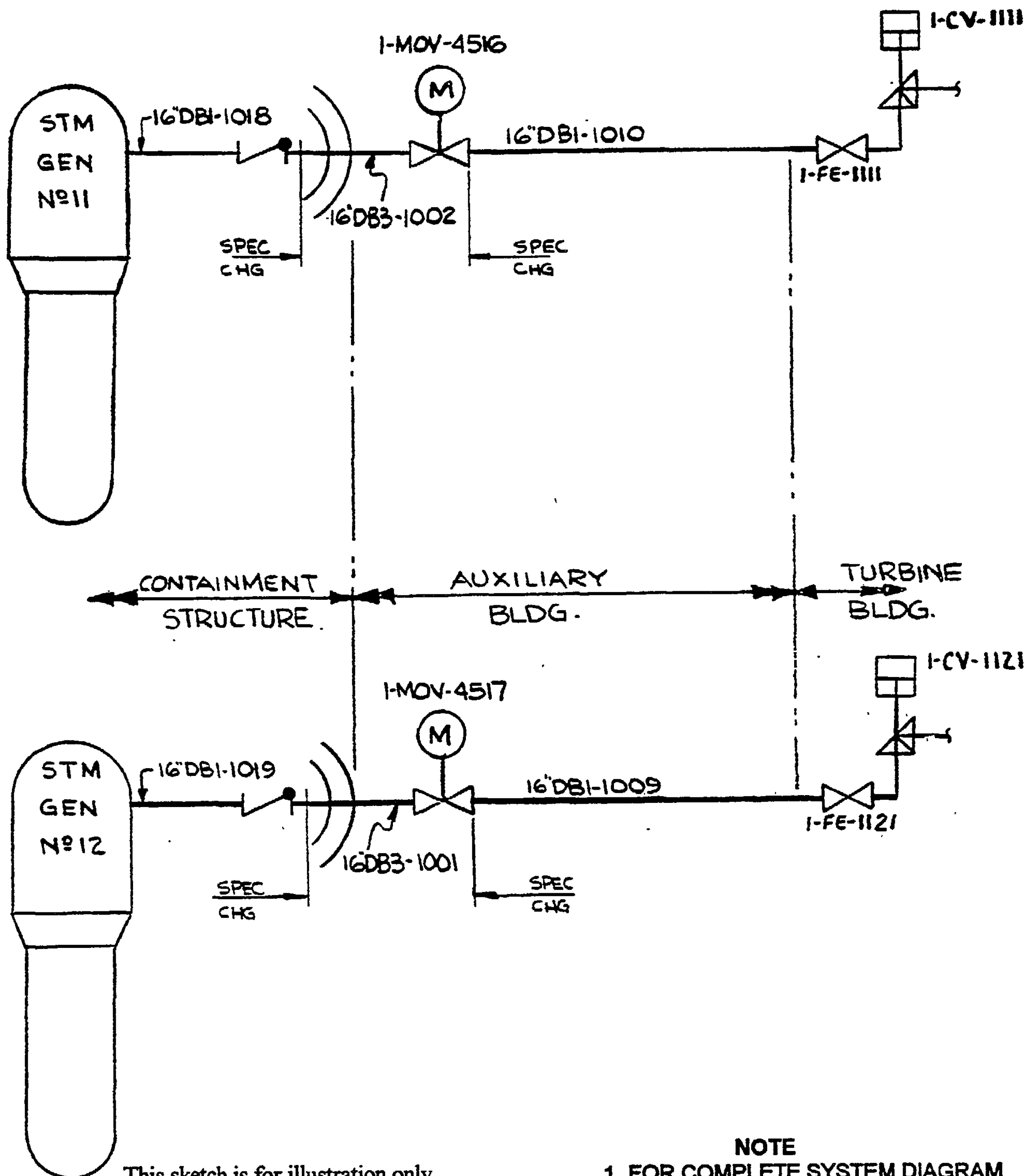
REVISION 21



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR BLOWDOWN LINE BREAK EAST PIPING PENETRATION ROOM INITIAL TEMPERATURE = 160F	Figure 10A.3-4



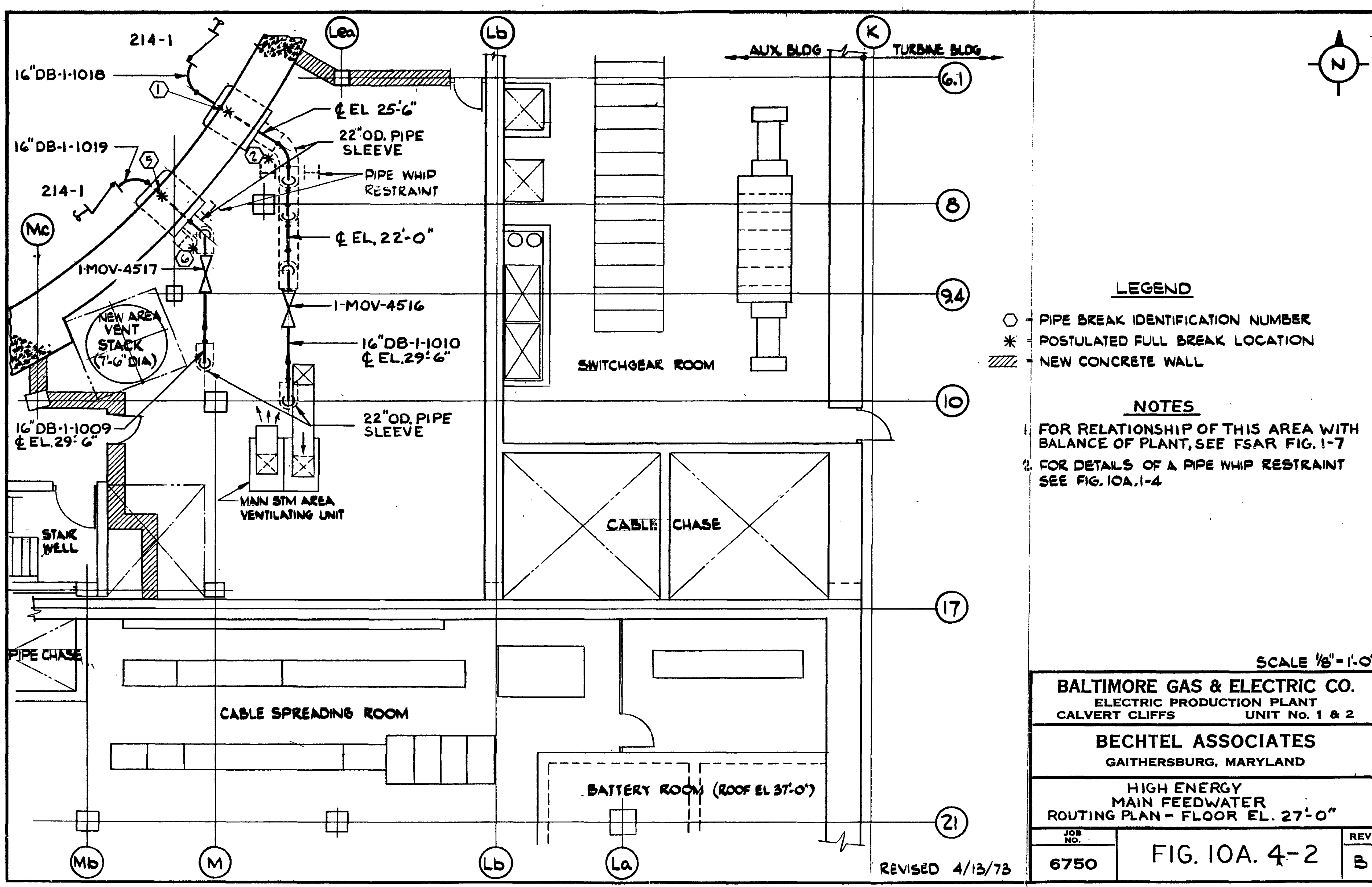
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR BLOWDOWN LINE BREAK ELEVATION 45' INITIAL TEMPERATURE = 160F	Figure 10A.3-5

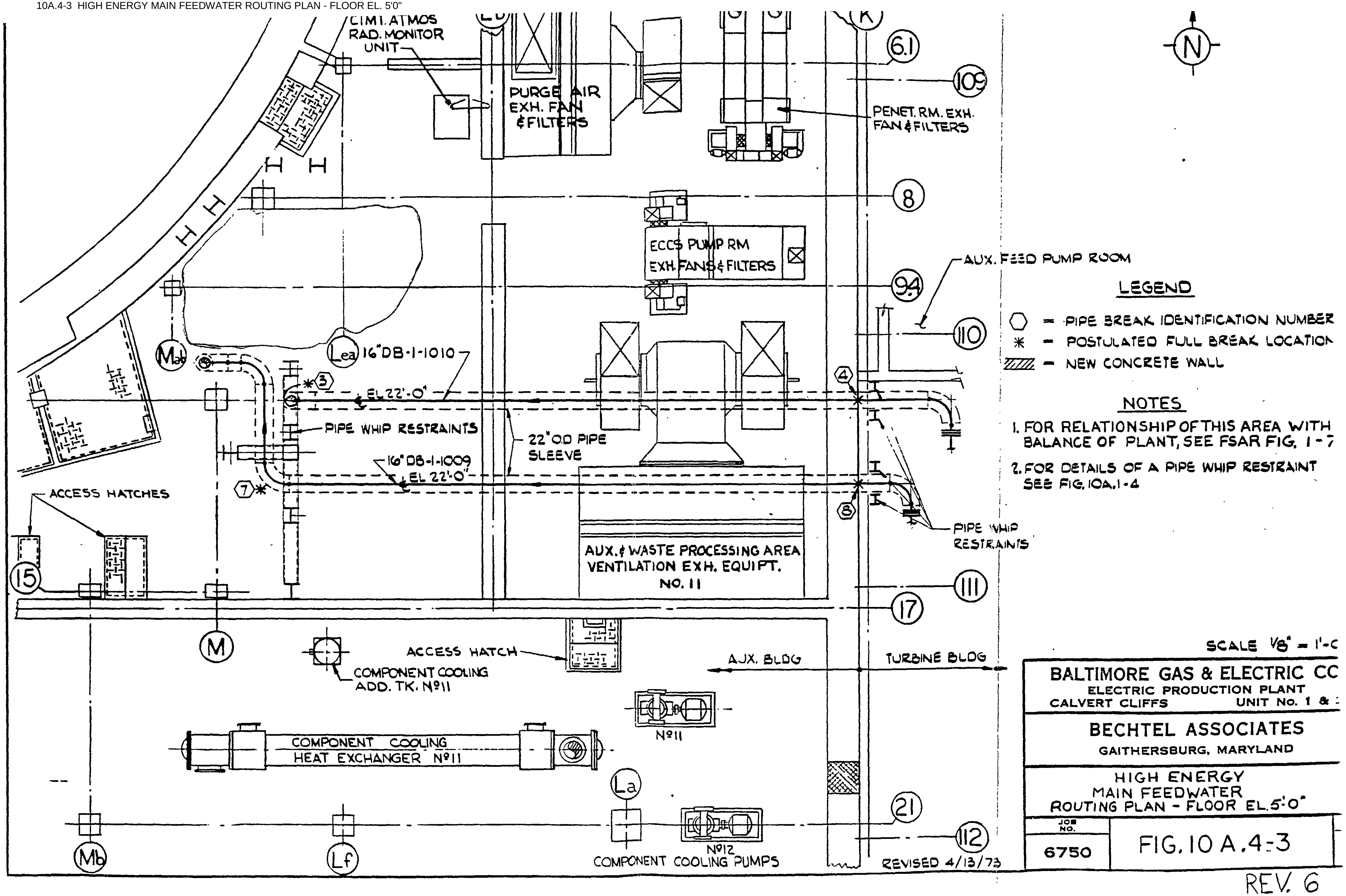


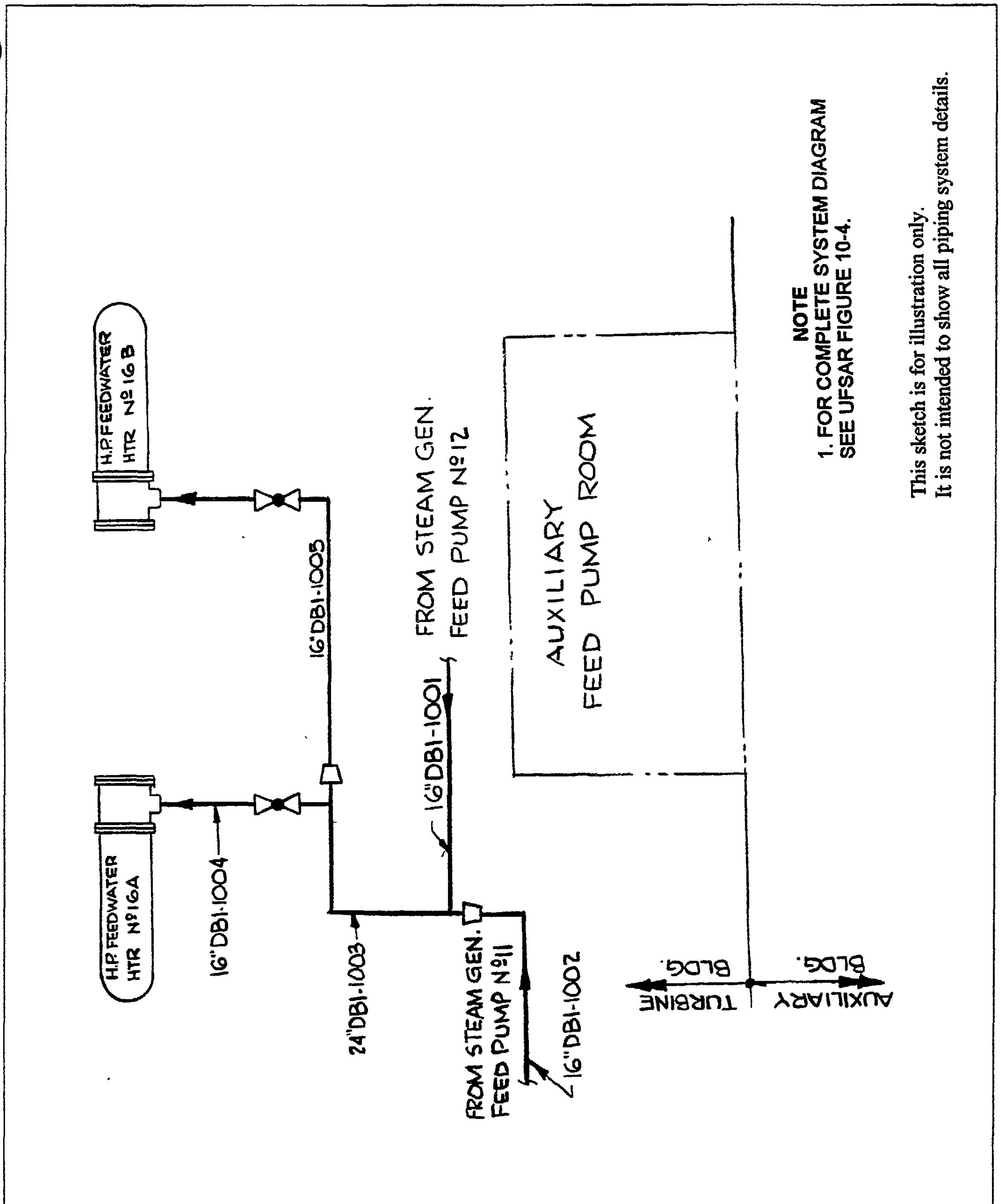
This sketch is for illustration only.
It is not intended to show all piping system details.

NOTE
1. FOR COMPLETE SYSTEM DIAGRAM SEE UFSAR FIGURE 10-4.

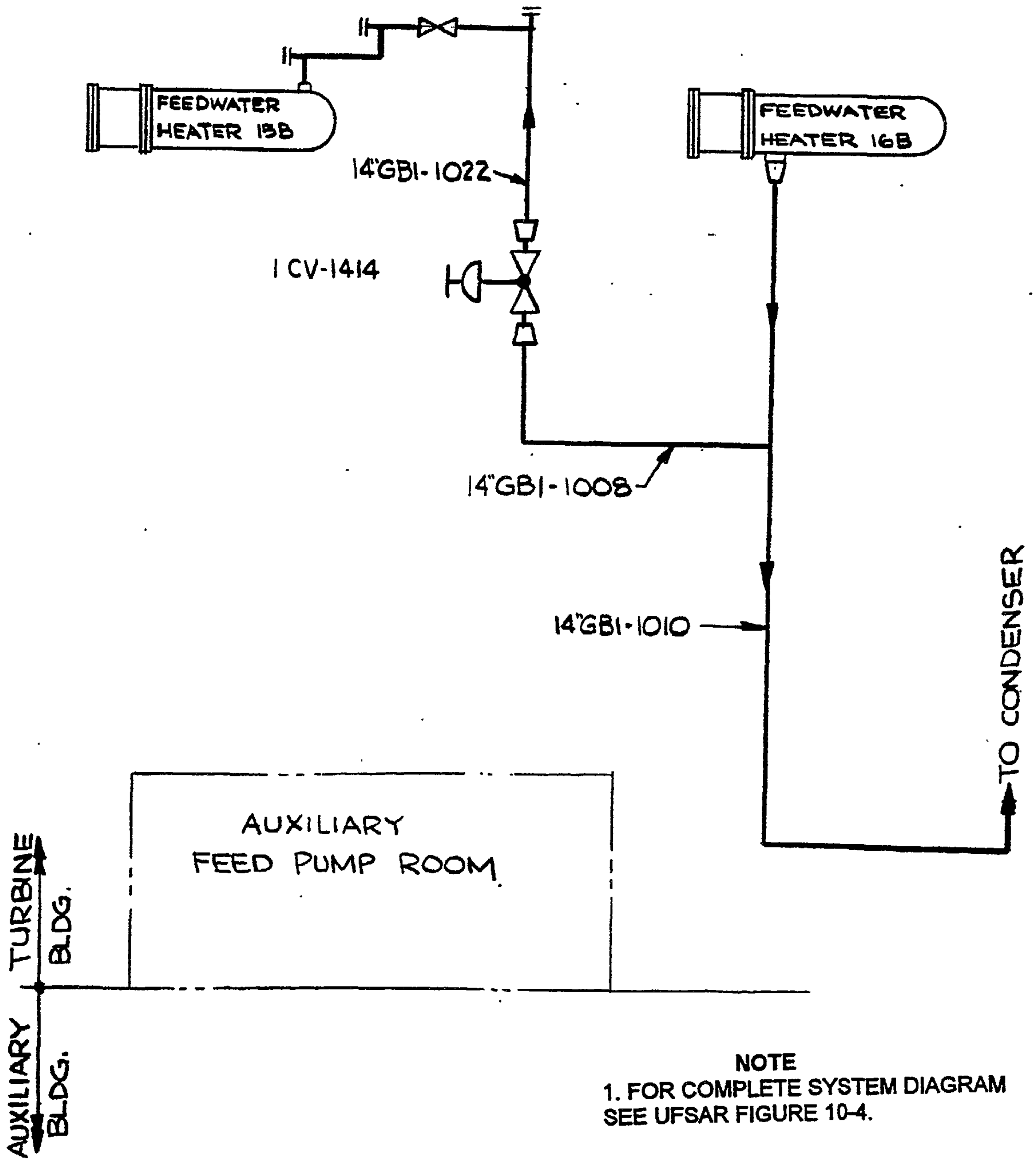
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	MAIN FEEDWATER SYSTEM	Figure 10A.4-1 Rev. 18


214-1
16"DB-1-1018
¢ EL 25'-6"
22" OD. PIPE SLEEVE
16"DB-1-1019
PIPE WHIP RESTRAINT
214-1
¢ EL. 22'-0"
1-MOV-4517
1-MOV-4516
NEW AREA VENT STACK (7'-6" DIA)
16"DB-1-1010 ¢ EL. 29'-6"
16"DB-1-1009 ¢ EL. 29'-6"
22" OD. PIPE SLEEVE
MAIN STM AREA VENTILATING UNIT
STAIR WELL
PIPE CHASE
CABLE SPREADING ROOM
SWITCHGEAR ROOM
CABLE CHASE
BATTERY ROOM (ROOF EL 37'-0")
AUX. BLDG
TURBINE BLDG
N
LEGEND
PIPE BREAK IDENTIFICATION NUMBER
POSTULATED FULL BREAK LOCATION
NEW CONCRETE WALL
NOTES
1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT, SEE FSAR FIG. 1-7
2. FOR DETAILS OF A PIPE WHIP RESTRAINT SEE FIG. 10A.1-4
SCALE 1/8"=1'-0"
BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
HIGH ENERGY MAIN FEEDWATER ROUTING PLAN - FLOOR EL. 27'-0"
JOB NO. 6750
FIG. 10A. 4-2
REV. B
REVISED 4/13/73


CIMT. ATMOS RAD. MONITOR UNIT
PURGE AIR EXH. FAN & FILTERS
PENET. RM. EXH. FAN & FILTERS
ECCS PUMP RM EXH. FANS & FILTERS
AUX. FEED PUMP ROOM
16" DB-1-1010
EL 22'-0"
PIPE WHIP RESTRAINTS
22" OD PIPE SLEEVE
16" DB-1-1009
EL 22'-0"
ACCESS HATCHES
AUX. & WASTE PROCESSING AREA VENTILATION EXH. EQUIPT. NO. 11
PIPE WHIP RESTRAINTS
ACCESS HATCH
COMPONENT COOLING ADD. TK. Nº11
AUX. BLDG
TURBINE BLDG
COMPONENT COOLING HEAT EXCHANGER Nº11
Nº11
Nº12
COMPONENT COOLING PUMPS
REVISED 4/13/73
LEGEND
PIPE BREAK IDENTIFICATION NUMBER
POSTULATED FULL BREAK LOCATION
NEW CONCRETE WALL
NOTES
1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT, SEE FSAR FIG. 1-7
2. FOR DETAILS OF A PIPE WHIP RESTRAINT SEE FIG. 10A.1-4
SCALE 1/8" = 1'-0
BALTIMORE GAS & ELECTRIC CO
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2
BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND
HIGH ENERGY MAIN FEEDWATER ROUTING PLAN - FLOOR EL. 5'-0"
JOB NO. 6750
FIG. 10 A.4-3
REV. 6



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER TO HP HEATERS	Figure 10A.4-4 Rev. 18

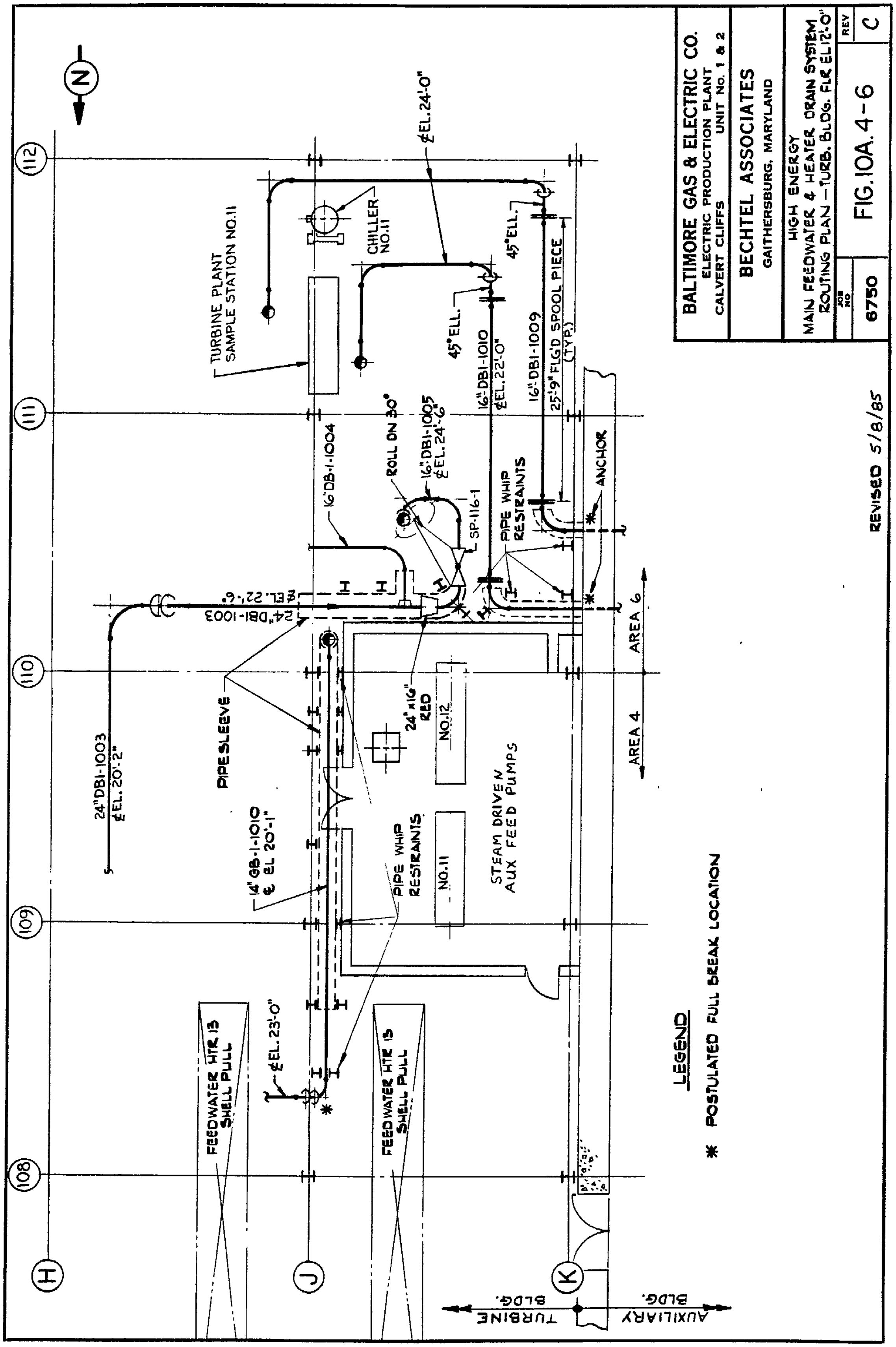


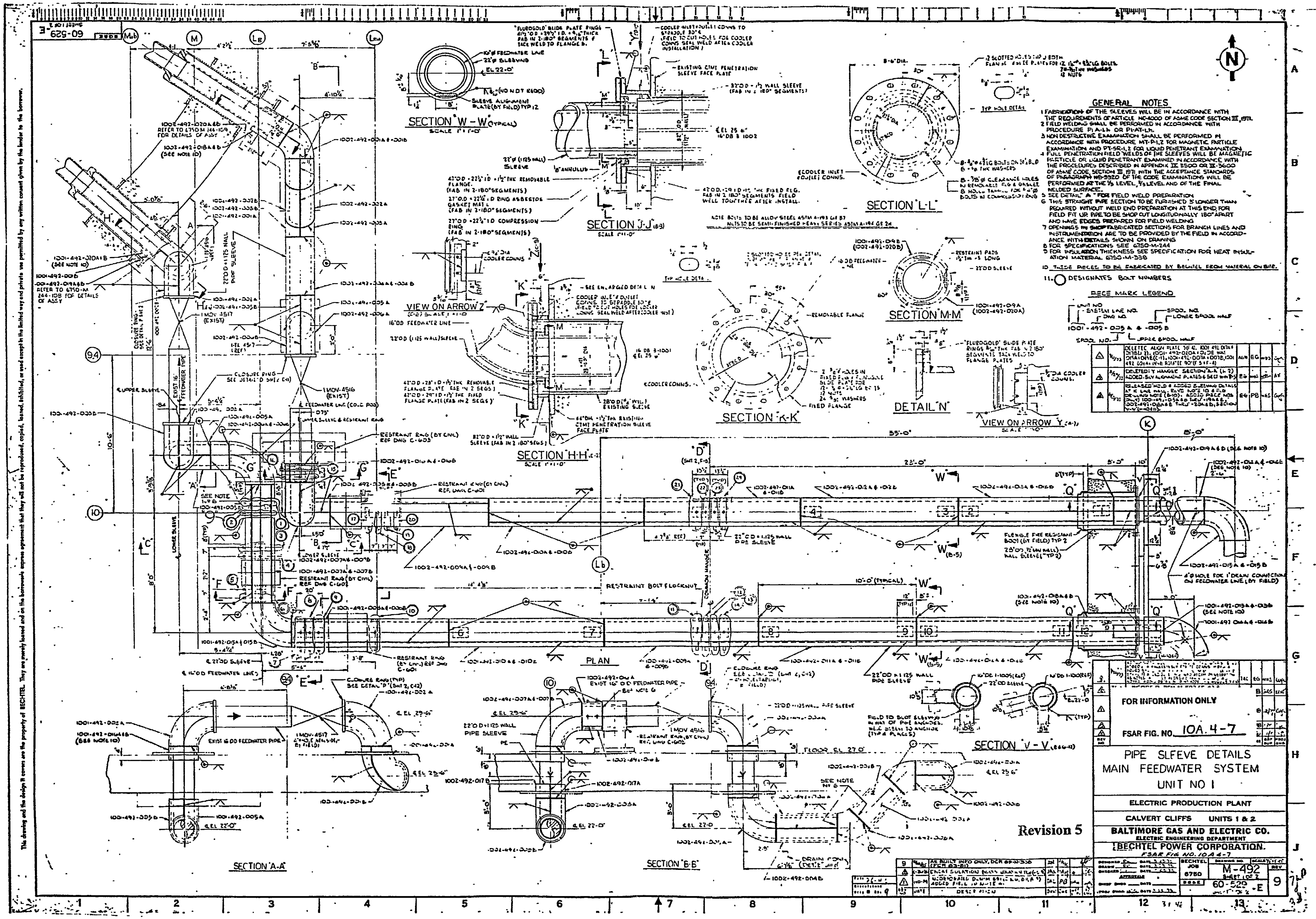
NOTE
1. FOR COMPLETE SYSTEM DIAGRAM SEE UFSAR FIGURE 10-4.

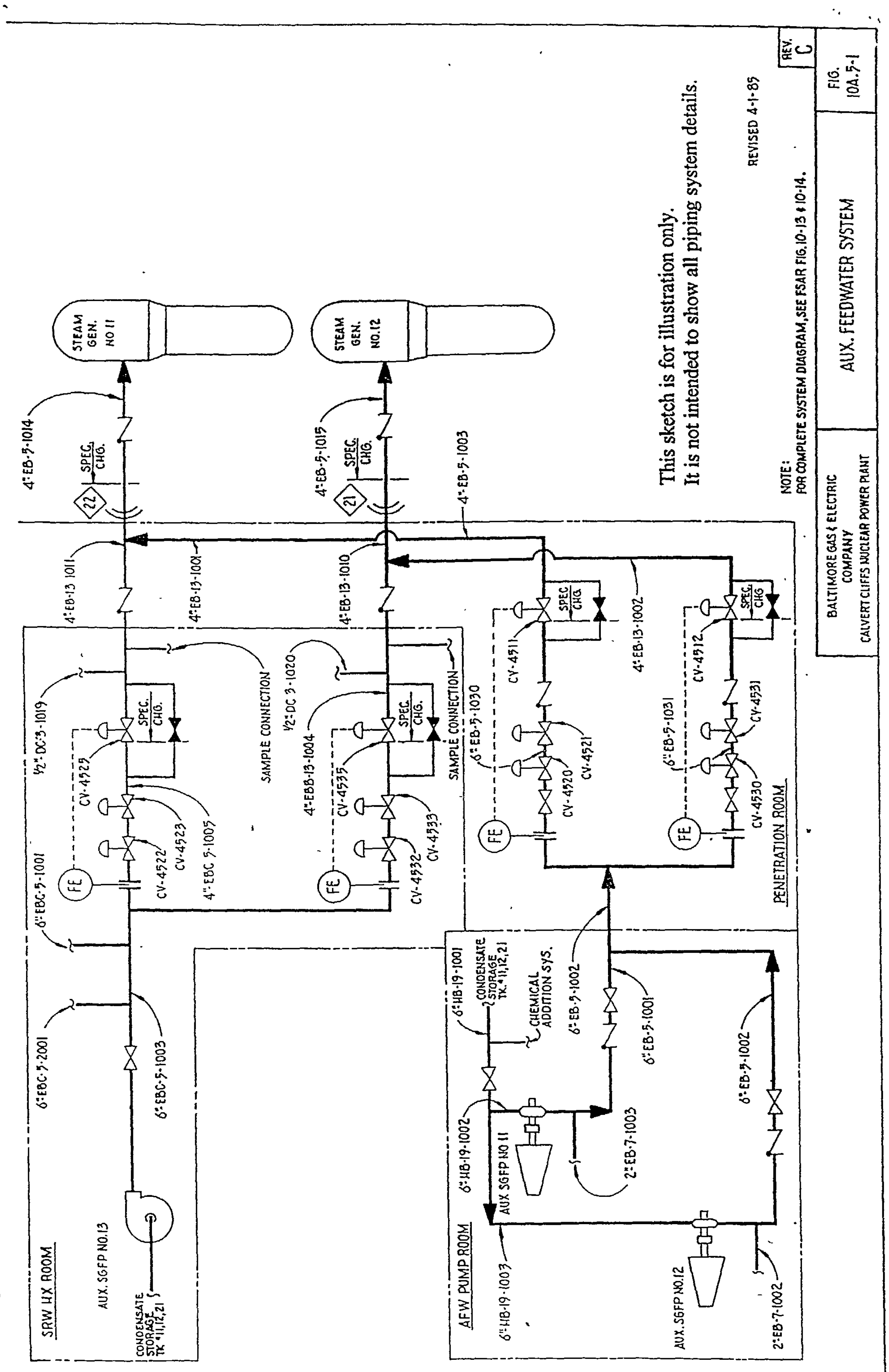
This sketch is for illustration only.
It is not intended to show all piping system details.

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER HEATER DRAIN TO CONDENSER	Figure 10A.4-5 Rev. 18

10A.4-6 HIGH ENERGY MAIN FEEDWATER AND HEATER DRAIN SYSTEM ROUTING PLAN - TURB BLDG. FLR EL. 12'0"




GENERAL NOTES
PIECE MARK LEGEND
SECTION 'W - W' (TYPICAL)
SECTION 'J-J'
SECTION 'L-L'
SECTION 'M-M'
SECTION 'K-K'
DETAIL 'N'
VIEW ON ARROW 'Y'
VIEW ON ARROW 'Z'
SECTION 'H-H'
PLAN
SECTION 'A-A'
SECTION 'B-B'
SECTION 'V - V'
FOR INFORMATION ONLY
FSAR FIG. NO. 10A.4-7
PIPE SLEEVE DETAILS
MAIN FEEDWATER SYSTEM
UNIT NO 1
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNITS 1 & 2
BALTIMORE GAS AND ELECTRIC CO.
ELECTRIC ENGINEERING DEPARTMENT
BECHTEL POWER CORPORATION
Revision 5
M-492
60-529


This sketch is for illustration only.
It is not intended to show all piping system details.
NOTE:
FOR COMPLETE SYSTEM DIAGRAM, SEE FSAR FIG. 10-13 & 10-14.
REVISED 4-1-85
REV. C
FIG. 10A.5-1
AUX. FEEDWATER SYSTEM
BALTIMORE GAS & ELECTRIC COMPANY
CALVERT CLIFFS NUCLEAR POWER PLANT
STEAM GEN. NO 11
STEAM GEN. NO.12
SRW HX ROOM
AFW PUMP ROOM
PENETRATION ROOM
AUX. SGFP NO.13
AUX SGFP NO 11
AUX. SGFP NO.12
CONDENSATE STORAGE TK. #11,12,21
CHEMICAL ADDITION SYS.
SAMPLE CONNECTION
SPEC. CHG.
FE
CV-4511
CV-4512
CV-4520
CV-4521
CV-4522
CV-4523
CV-4529
CV-4530
CV-4531
CV-4532
CV-4533
CV-4535
4"EB-5-1014
4"EB-5-1015
4"EB-5-1003
4"EB-13-1011
4"EB-13-1001
4"EB-13-1010
4"EB-13-1002
1/2"DC-3-1019
1/2"DC 3-1020
4"EBB-13-1004
6"EB-5-1030
6"EB-5-1031
4"EBC 5-1005
6"EBC-5-1001
6"EBC-5-2001
6"EBC-5-1003
6"HB-19-1001
6"HB-19-1002
6"HB-19-1003
6"EB-5-1002
6"EB-5-1001
2"EB-7-1003
2"EB-7-1002

Revision 20

UNIT 1

POSTULATED PIPE CRACK "A"
PIPE NO 4"EBC-5-1005, DWG NO SK-ME-105 SH 1

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1			BIA2253-¾"C (FROM 1-LS-1651)	PB	1-LS-1651	BIK52EL	1J236-1-LS-1651	
2		1-LS-1651						
3	¼" COPPER INST AIR TO 1-CV-5150							23* SEE NOTE 3 (TYP)
4			BIA2256-¾"C (FROM 1-PS-2256)	1-PS-1600	PB	BIK11BV	1K01-1-PS-1600	
5			A1A2238-1"C (FROM 1-CV-5152)	1-SV-5152	PB	A1V5152A	2C24A-1-SV-5152	42*
6			ZA1A2297-1"C (FROM 1-CV-5150)	PB	1-SV-5150	ZA1V5210C	2C24A-1-SV-5150	29*
7			ZA1A4252 (NEW AFW)			SEE NOTE 2 (TYPICAL)		
8			ZB1A2299-1"C (FROM 1-CV-5153)	1J459	1-SV-5153	ZB1V5212G	1J459-1-SV-5153	42*
9			ZA1A4251 (NEW AFW)					
10			ZA1A4237 (NEW AFW)					

POSTULATED PIPE CRACK "B", PIPE NO 4"EBB-13-1004
"C", 4"EBB-13-1002
DWG NO. SK-ME-105 SH 1

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1		JUNCTION BOX 1J575						
2			A1A4232- (NEW AFW)					
3			B1A9323-¾"C (FIRE PROT. CONDUIT)	OAE 226E	OAE 226F	B1FP226J	OAE226E - OAE226F	
4	¼" COPPER INSTR. AIR TO CV-4070 & CV 4071							10*

POSTULATED PIPE CRACK "D" PIPE NO 6"-EBC 5-1001
"E" 6"-EBC 5-2001
"E'" 6"-EBC 5-2001
DWG NO. SK-ME-105 SH 1

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESIGNATION	FORCE ON TARGET
1			A1A3849-2"C (1AA51)	TEE 1A3916/3848	PB	A1B0333P A1B0336P	1B003-1M0333 1B003-1M0336	
2			A1A3828-1½"C (1C107)	1C107	PB	A1B0336B B1V4016B	1C107-1PS4080 1C107-1SV4016	
3			B1A3844-2"C (1C107)	1C107	PB	B1V4087A B1V4087B B1V4087C	1C107-1TIS4087 1C107-1SV4087 1C107-1SV4088	
4			B1A3833-¾"C (1FT-4089)	PB	1FT 4089	B1F4089A	1C107-1FT4089	
5			B1A3840-1"C (2AC26)	1AG56	1RE 4095	B1RAD23A	1C22D-1RE4095	
6			A1A0398-1½"C	PB	TEE 1A0410/1215	A1F1581A	1FT1581A-1R01A 1FT4509-1R01A	
7			B1A0388-1½"C	PB	TEE 1A0390/0391	B1F1589A B1F4510A	1FT1589-1R01A 1FT4510-1R01A	
8			ZA1A1547-1"C(RS) (FROM 1-MOV-6579)	1-MOV-6579	1AA61	ZA1B1415P	1B014-1MOV6579	75*
9			ZA1A1546-1"C (FROM 1-MOV-6579)	1-MOV-6579	1AA61	ZA1B1415B	1B014-1MOV6579	75*
10			ZA1A0380-2½"C	1AA61	TEE 1A0381/82	ZA1B1407P ZA1B1408P	1B014-1MOV615 1B014-1MOV625	86*
11			ZA1A0383-2½"C	1AA61	TEE 1A0384/85	ZA1B1407B ZA1B1408B	1B014-1MOV615 1B014-1MOV625	86*
12			ZA1A1631-1"C(RS)	1AA53	1-MOV-617	ZA1B1416P	1B014-1MOV617	35*
13			ZA1A1632-1½"C	1-MOV-617	1AA53	ZA1B1416B	1B014-1MOV617	53*
14			ZB1A1636-1½"C	1-MOV-616	1AC07	ZB1B0416C	1B004-1MOV616	32*
15			ZB1A1635-1"C(RS)	1-MOV-616	1AC07	ZB1B0416P	1B004-1MOV616	49*
16	(8) ½" SAMPLE TUBING FROM 2"-EB-44 LINE STM. GEN BLOWDN SYSTEM							17*
17	TUBING FOR HPSI FLOW TO LOOP 11A GOING TO 1FT-311							27*
18	¼" COPPER INST AIR FROM A.F.W. ACCUM A&B							48*

POSTULATED PIPE CRACK "F", PIPE NO. 4"EBB-13-1004 *
"G", 4"EBB-13-1002 } ** SEE NOTE 1
"H", 4"EBB-13-1004 }
DWG NO SK-ME-105 SH 1

	ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESIGNATION	FORCE ON TARGET
**	1			B1A3840-1"C	1AG56	1RE 4095	B1RAD23A	1C22D-1RE4095	
**	2			B1A3833-¾"C (1FT-4089)	PB	1FT-4089	B1F4089A	1C107-1FT4089	
**	3			B1A3834-¾"C	PB	1TE 4086	B1T4086A	1C07-1TE4086	
**	4			B1A3844-2"C (1C107)	1C107	PB	B1V4087A B1V4087B B1V4087C	1C107-1TIS4087 1C107-1SV4087 1C107-1SV4088	
**	5			B1A3831-1½"C (FROM PB)	1C107	PB	B1F4089A B1K107B B1K107C B1P4092A B1P4093A B1T4086A	1C107-1FT4089 1C107-1PDIS4090 1C107-1PDIS4092 1C107-1PDT4092 1C107-1PDT4093 1C09-1TE4143	
*	6			ZA1A1547-1"C(RS) (FROM 1-MOV-6579)	1-MOV-6579	1AA61	ZA1B1415P	1B014-1MOV6579	75*
*	7			ZA1A1546-1"C (FROM 1-MOV-6579)	1-MOV-6579	1AA61	ZA1B1415B	1B014-1MOV6579	75*
**	8			ZB1A4198 (NEW AFW)					
**	9	¼" COPPER INST AIR TO CV 4070 & CY-4071							10*

POSTULATED PIPE CRACK "J", PIPE NO. 6"EBC-5-1001
"K", 6"EBC-5-2001
DWG NO. SK-ME-105 SH 2

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESIGNATION	FORCE ON TARGET
1	¾" HB 35 PIPE TO 1-SV-5464							76*
2			A1A3863-1"C (BREATHING AIR)	1AA66	PB	A1P0331P	1B003-1HS2026	
3			A1A3916-3"C	1B003	TEE 1A3848/49	A1B0333A A1B0333P A1B0336A A1B0336P	1B003-1C107 1B003-1M0333 1B003-1C107 1B003-1M0336	

UNIT 2

POSTULATED PIPE CRACK "L", PIPE NO. 6"EBC-5-1001
"M", 6"EBC-5-2001
DWG NO. SK-ME-106 SH 2

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1			B2A0326-6"C	2AA35	MCC-205W/P	B2B321P B2B321Q	2B003 2B03B	
2			A2A3404-3"C	2AB13	PB	A2B0330A A2B0330P A2B0332A A2B0332P A2P0331P	2B003-2C107 2B003-2M0330 2B003-2C107 2B003-2M0332 2B003-2M0331	
3			A2A0774-1½"C	2B003	2M0345	A2B0345P	2B003-2M0345	
4			A2A0745-1½"C	2B003	2M0333	A2B0333P	2B003-2M0333	
5			B2A3429-1½"C	2J497	2AM16	B2V4016C	1C107-2J497	
6			A2A3098-¾"C	2AA68	PB	A0SCCTVE	1C30C-0SCCTV3	
7			ZD2A3054-¾"C	2AH85	2-PT-5316A	ZD2C91P	2C91-2-PT-5316A	
8			ZE2A3055-¾"C	2AG62	2-PT-5316B	ZE2C92P	2C92 2-PT-5316B	
9	SAFETY INJECTION TUBING POST-ACCIDENT SAMPLING U4402							
10	¾" HB 71 -2003 ¾" HB 71 -2004 [illegible]							

POSTULATED PIPE CRACK "N"
PIPE NO. 4"EBC-5-2002, DWG NO. SK-ME-106 SH 1

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1	¼" COPPER FOR 2I/P5210							14*
2		2I/P 5210						
3			ZA2A1312-1"C	PB	2-SV-5210	A2V5212D ZA2V5210D	DELETED 2C24A-2SV5210	
4			ZA2A2469-¾"C (CONDUIT FROM SV-5210)	PB	2-SV-5210	A2V5212B ZA2V5210B	DELETED 2C24A-2SV5210A	
5	¼" COPPER TO PI-5210							
6			A2A1459-¾"C	2I/P-5210	PB	A2V5210A	C2I3-2I/P5210	
7	¼" COPPER TO CV-4599 & CV-1657							23*

POSTULATED PIPE CRACK "P"
PIPE NO. 6"EBC-5-1001 DWG NO SK-ME-106 SH 2

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1			A2A2824-2"C	2J290	PB	A2B1459A A2B1461A A2K21CS	2B014-2J290 2B014-2J290 2J290-2K01	
2			A2A2821-¾"C	2J409	2J290	A2B1459C	2J290-2J409	
3		ELECT JCT BOX 2J-290						
4		ELECT JCT BOX 2J-291						
5		HYDRAULIC PUMPING UNIT PANEL						2120* (APPROX)
6			A2A2817-1½"C	2AA59	2M1459	A2B1459P	2B014-2M1459	
7			B2A2818-1½"C	2AC19	2M0456	B2B0456P	2B004-2M0456	
8			A2A2816-1½"C	2AA59	2M1461	A2B1461P	2B014-2M1461	
9	¾" HB AIR SUPPLY TO SV505 & 5464							28.5*

POSTULATED PIPE CRACK "R", PIPE NO. 6"EBC-5-1001
"S", 6"EBC-5-2001
DWG NO SK-ME-106 SH 1

ITEM	TUBING	INST	CONDUITS	LOCATION FROM	LOCATION TO	INCLUDES CABLES	CABLE DESTINATION	FORCE ON TARGET
1			ZA2A2675-¾"C	2AA68	2FT 6901	ZA2P1435A	1C100-2FT6901	29*
2	¾" HB AIR SUPPLY TO SV505 & 5464							44*

POSTULATED PIPE CRACK "T"
PIPE NO. 4"EBB-13-2004, DWG NO SK-ME-106 SH 1

ONLY AFW (MOTOR DRIVEN TRAIN) INSTRUMENT AIR TUBING LOCATED IN THIS AREA IS AFFECTED

NOTES:

1. ASTERISKS RELATE TARGETS TO ASSOCIATED CRACK POINTS
2. NO DETAILED CABLE LISTING INDICATED AS CONDUITS FOR NEW AFW (MOTOR DRIVEN TRAIN) INCLUDE CABLES FOR ONLY NEW AFW
3. WHERE FORCES ARE LISTED, TARGETS ARE SAFETY-RELATED & ARE TO BE INCLUDED IN THE "AUXILIARY FEEDWATER SYSTEM POSTULATED PIPE CRACK ANALYSIS REPORT"

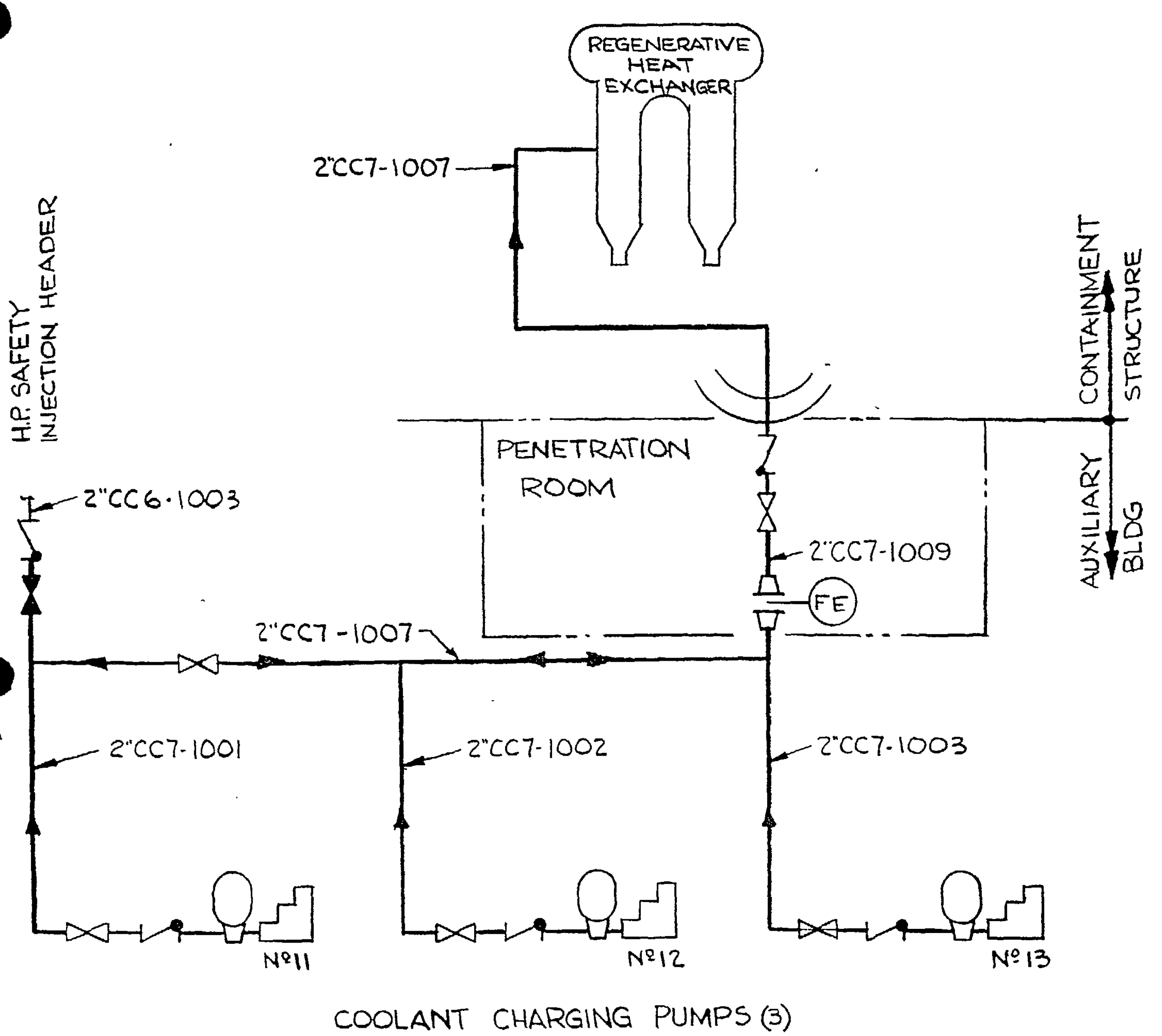
BALTIMORE GAS & ELECTRIC COMPANY DRAWING NO. 60-575-E (SK-ME-107) REVISION 2

CALVERT CLIFFS NUCLEAR POWER PLANT
UPDATED FINAL SAFETY ANALYSIS REPORT

UFSAR FIGURE 10A.5-5

POSTULATED PIPE CRACK TARGETS
AUX FEEDWATER SYSTEM
UNITS 1 & 2

Revision 15



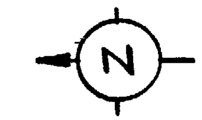
NOTES

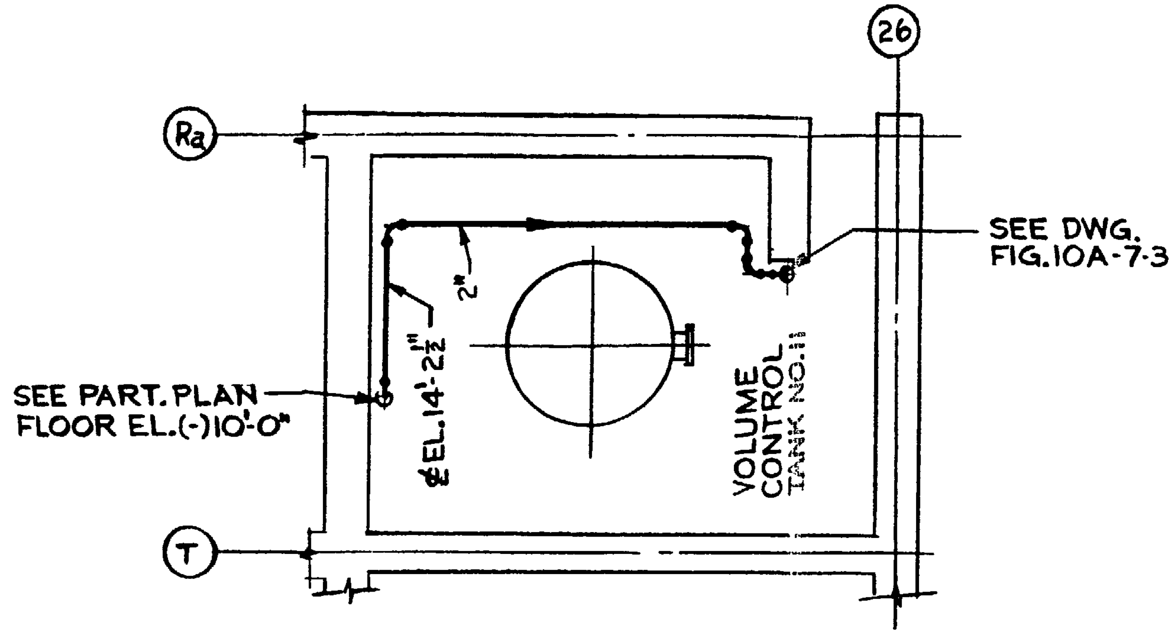
1. FOR COMPLETE SYSTEM DIAGRAM SEE FSAR FIG. 9-3.

This sketch is for illustration only.
It is not intended to show all piping system details.

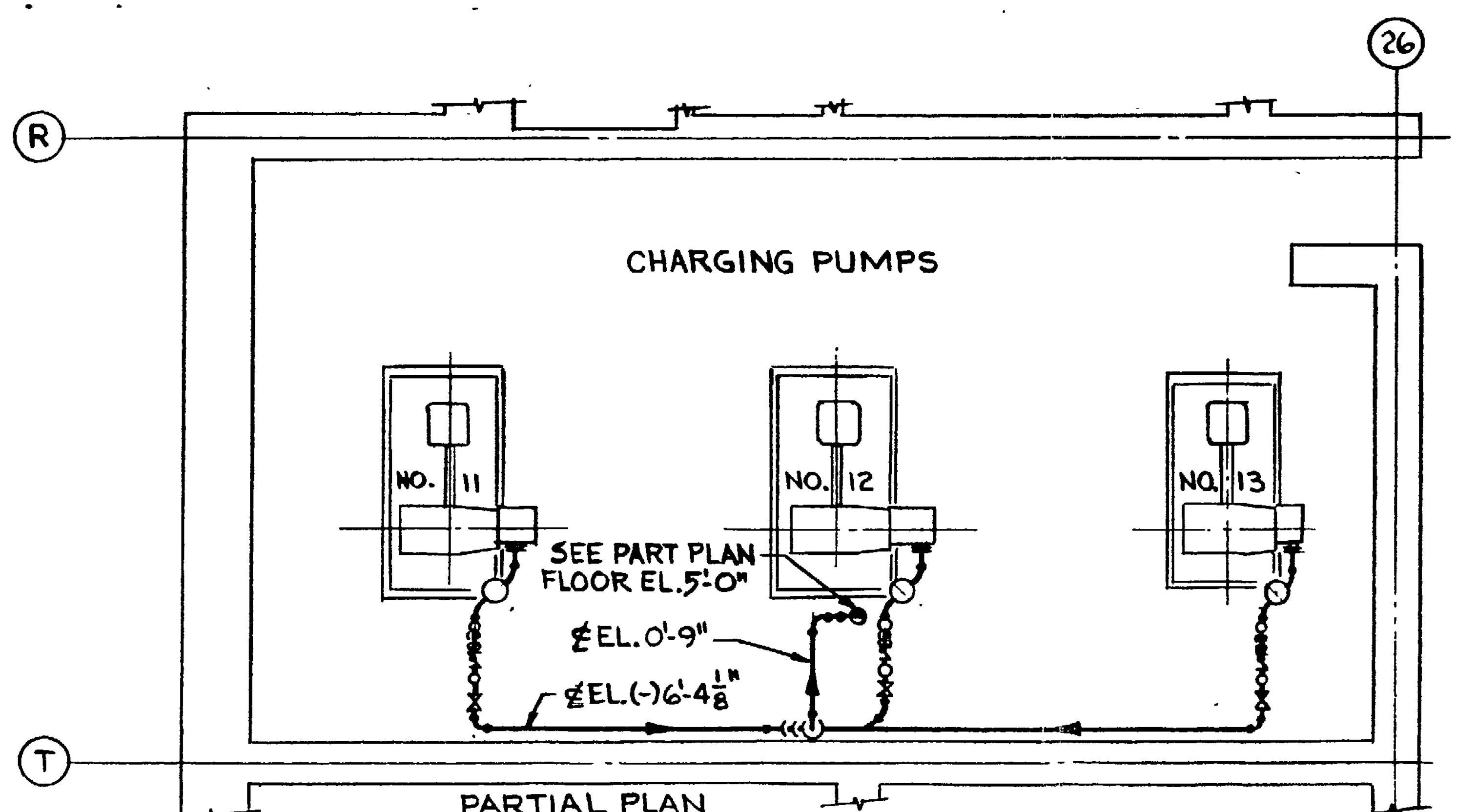
Revision 20

BALTIMORE GAS AND ELECTRIC CO. CALVERT CLIFFS NUC. PWR. PLNT.	CVC SYSTEM REACTOR COOL. CHARGING LINE	FIG. 10A.7-1





PARTIAL PLAN
FLOOR EL.5'-0"



PARTIAL PLAN
FLOOR EL.(-)10'-0"

NOTES

1. FOR RELATIONSHIP OF THIS AREA WITH BALANCE OF PLANT, SEE FSAR FIG.1-5 & 1-6

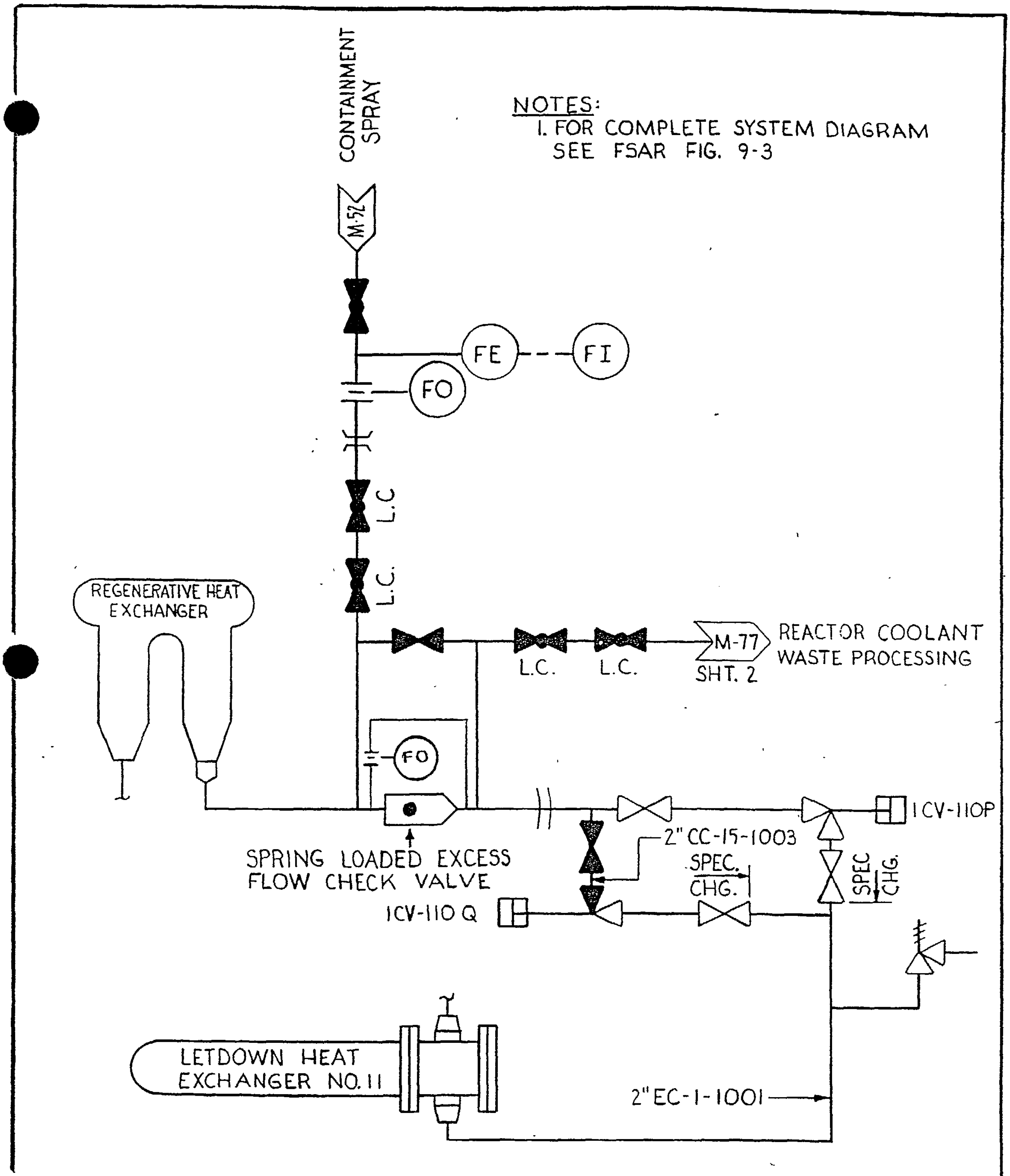
REVISED 4-13-73

BALTIMORE GAS & ELECTRIC CO.
ELECTRIC PRODUCTION PLANT
CALVERT CLIFFS UNIT No. 1 & 2

BECHTEL ASSOCIATES
GAITHERSBURG, MARYLAND

HIGH ENERGY
R.C. CHARGING LINE - ROUTING
PARTIAL PLANS @ FLOOR EL. 5'-0" & (-)10'-0"

JOB NO.	B.G. & E. DRAWING NO.	BECHTEL DRAWING NO.	REV.
6750		FIG.10A-7-2	B



This sketch is for illustration only.
It is not intended to show all piping system details.

Revision 20

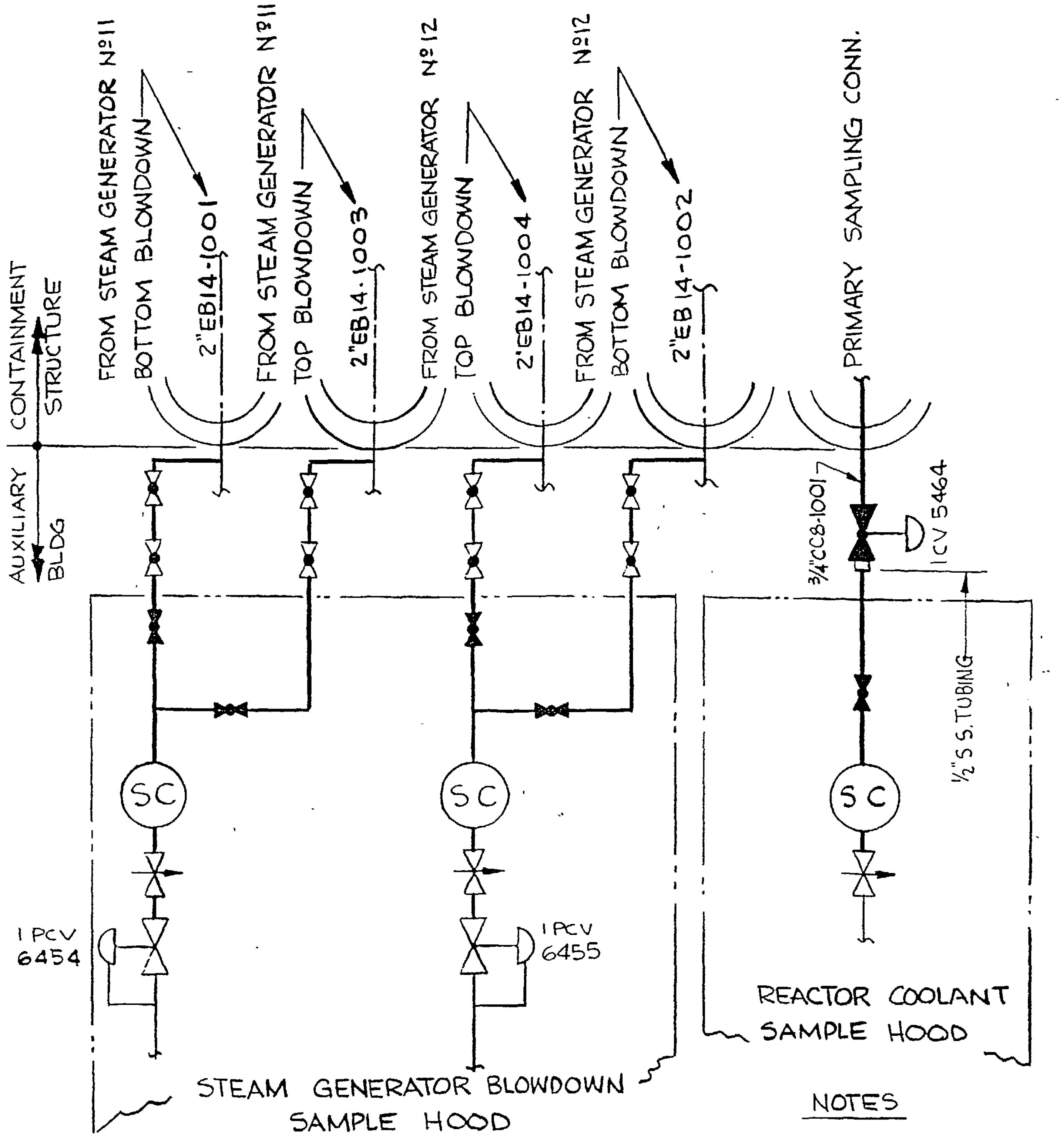
REVISED 10-1-76

REV. D

BALTIMORE GAS AND ELECTRIC CO.
CALVERT CLIFFS NUC. PWR. PLNT.

CVC SYSTEM
REACTOR COOLANT LETDOWN LINE

FIG. 10A.7-4



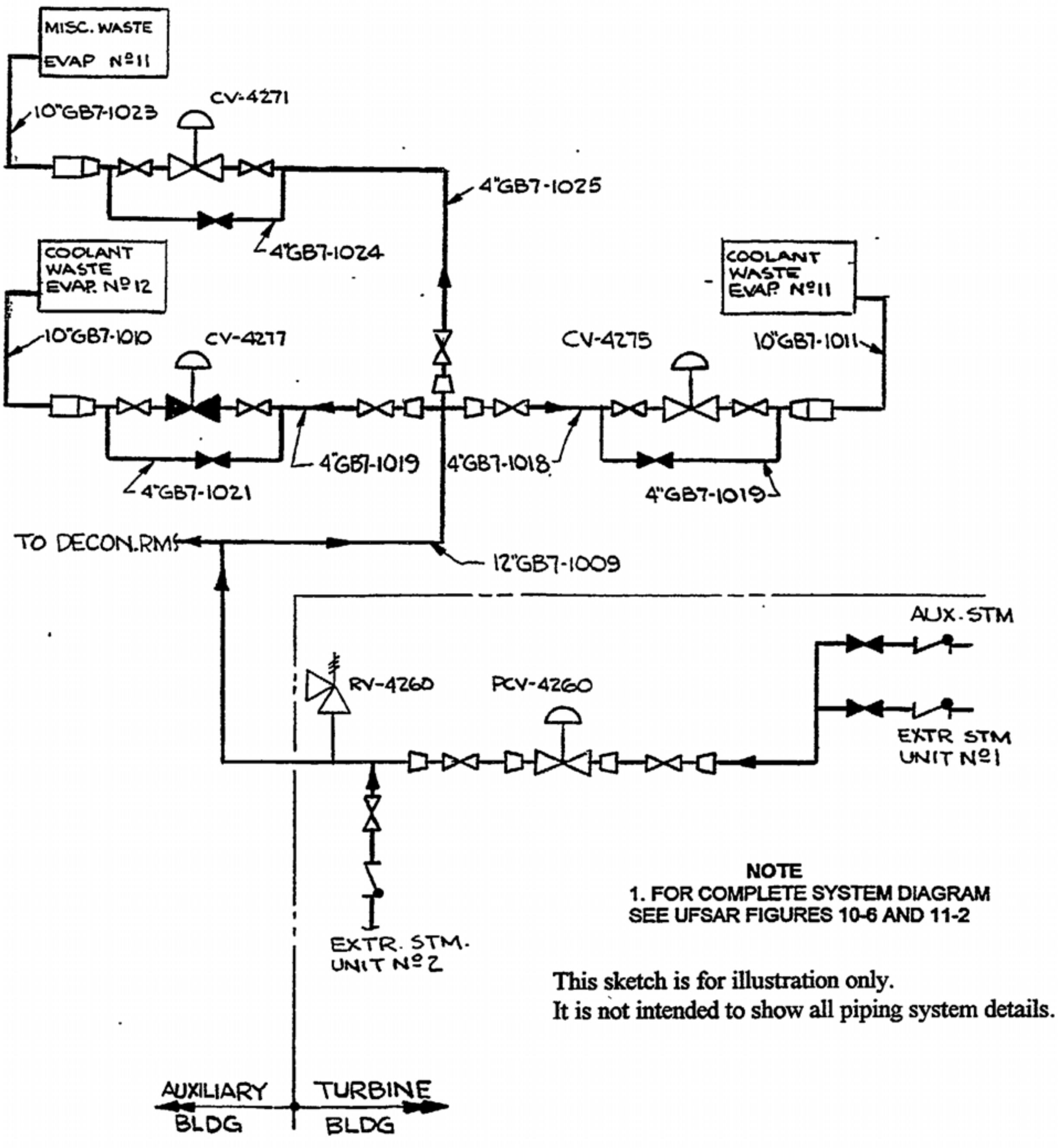
NOTES

1 FOR COMPLETE SYSTEM DIAGRAM SEE FSAR FIG. 9-10.

This sketch is for illustration only.
It is not intended to show all piping system details.

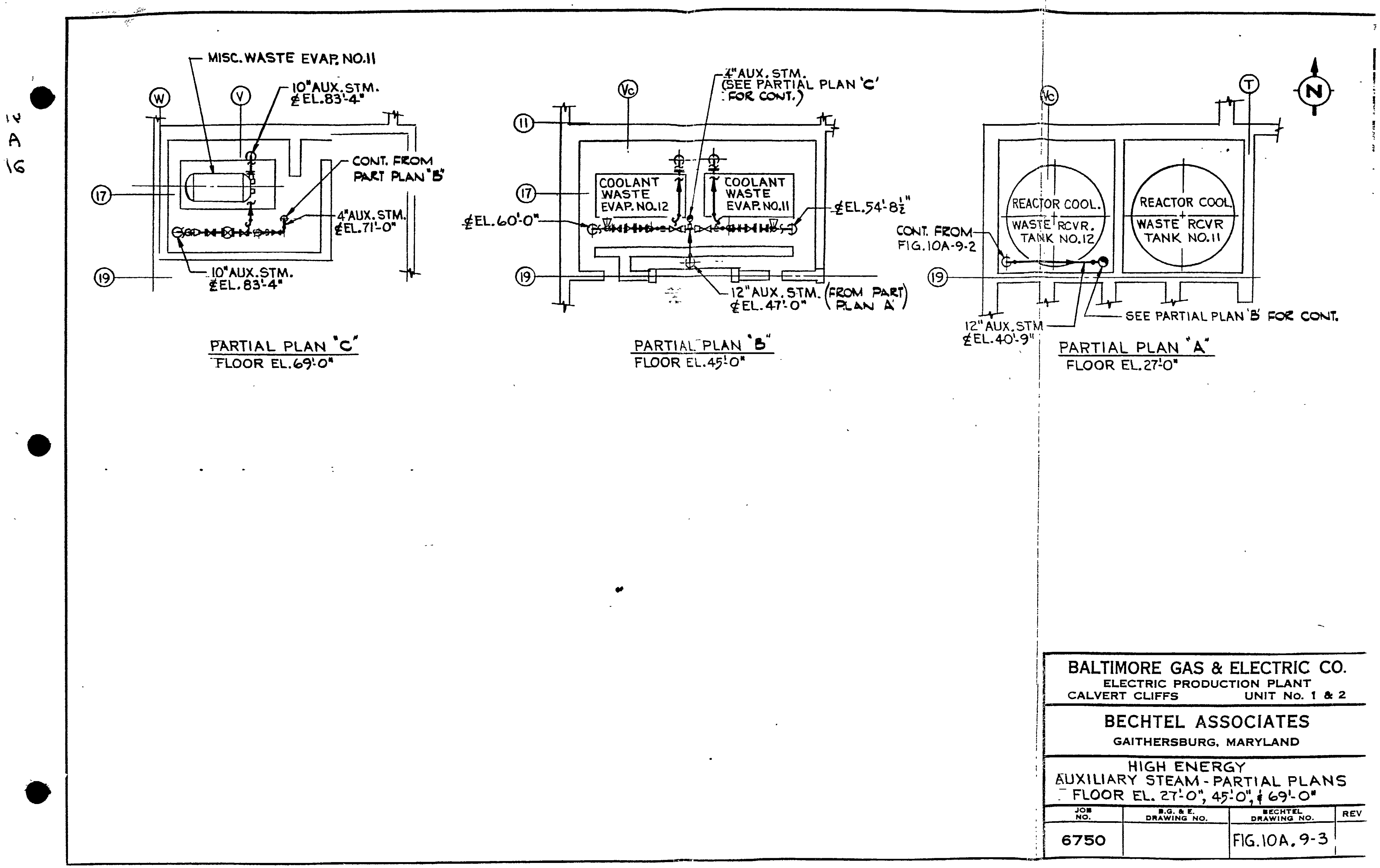
Revision 20

BALTIMORE GAS AND ELECTRIC CO. CALVERT CLIFFS NUC. PWR. PLNT.	STEAM GEN. & REACTOR COOL. SAMPLING SYSTEMS	FIG. 10A.8-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	AUX. STEAM TO WASTE EVAPORATORS	Figure 10A.9-1 Rev. 18

10A.9-3 HIGH ENERGY AUXILIARY STEAM ROUTING PLAN - FLOOR EL. 27'0"



CHAPTER 11
WASTE PROCESSING AND RADIATION PROTECTION

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WASTE PROCESSING AND RADIATION PROTECTION

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WASTE PROCESSING AND RADIATION PROTECTION

LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CCNPP	Calvert Cliffs Nuclear Power Plant, Inc.
CVCS	Chemical and Volume Control System
DAW	Dry Active Waste
DLR	Dosimeter of Legal Record
DRD	Direct-Reading Dosimeter
ECCS	Emergency Core Cooling System
ED	Electronic Dosimeter
GL	Generic Letter
GM	Geiger Mueller
HEPA	High Efficiency Particulate Air
LOCA	Loss-of-Coolant Accident
MHA	Maximum Hypothetical Accident
MPF	Materials Processing Facility
MWIE	Miscellaneous Waste Ion Exchanger
MWMT	Miscellaneous Waste Monitor Tank
MWPS	Miscellaneous Waste Processing System
MWRT	Miscellaneous Waste Receiver Tank
NEMA	National Electrical Manufacturers Association
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OSG	Original Steam Generator
RCA	Radiologically Controlled Area
RCDT	Reactor Coolant Drain Tank
RCS	Reactor Coolant System
RCW	Reactor Coolant Waste
RCWPS	Reactor Coolant Waste Processing System
SFP	Spent Fuel Pool
SRW	Service Water
SWPS	Solid Waste Processing System
WGPS	Waste Gas Processing System

11.0 WASTE PROCESSING AND RADIATION PROTECTION

11.1 WASTE PROCESSING

11.1.1 DESIGN BASIS

The waste processing systems are designed to provide controlled handling and disposal of radioactive liquid, gaseous and solid wastes from both units. Design criteria were established to maintain the release of radioactive material from the plant to the environment at levels which are as low as reasonably achievable (ALARA).

The design of the waste processing systems was based upon processing reactor coolant and miscellaneous waste during operation with 1% failed fuel. The annual radioactive waste releases for this design were shown to meet the dose guidelines of 10 CFR Part 50, Appendix I.

All releases meet the Offsite Dose Calculation Manual (ODCM) limits. By meeting the ODCM limits, the guidelines of 10 CFR Part 50, Appendix I will be met. This is confirmed by the effluent data and doses reported to the Nuclear Regulatory Commission (NRC) in the Radioactive Effluent Release Reports required by the Technical Specifications and 10 CFR 50.36a.

The ODCM is a program governed by requirements described in Technical Specifications. It provides limits for offsite radioactive waste releases, calculational methods to determine those releases, and alternative methods of accounting for and controlling release of radioactive materials. In the event that the radiation monitors described in this chapter for use in the release of liquid and gaseous radioactive material are not available for use, the alternative methods described in the ODCM may be used.

11.1.2 WASTE PROCESSING SYSTEMS

The waste processing systems include the Reactor Coolant Waste Processing System (RCWPS), the Miscellaneous Waste Processing System (MWPS), the Waste Gas Processing System (WGPS) and the Solid Waste Processing System (SWPS). Information on the design and fabrication of the components of these systems is listed in Table 11-1. Radwaste systems seismic requirements are provided in Section 5A.2.1.2.

11.1.2.1 Liquid Waste Processing Systems

11.1.2.1.1 Design Bases

Liquid waste is processed by two systems: the RCWPS and the MWPS. The RCWPS is designed to process reactor coolant concurrent with the letdown flow from the Chemical and Volume Control System (CVCS). The MWPS processes waste from miscellaneous sources. These liquid waste systems are designed to provide adequate dilution through a variable release rate of waste from the designated tanks to the circulating water discharge conduits of either or both units.

The design performance requirements for the liquid waste processing systems were established by examining seafood consumption routes to determine limiting discharge concentrations. Liquid effluent is controlled such that the increases in boron concentration of the circulating water are minimized. Sampling and release of liquid wastes were designed to be accomplished on a batch basis rather than on a continuous basis in order to provide for increased control over effluent discharge.

On order to determine the sizing of waste processing components, the preoperational design for the RCWPS assumed that 14 system volumes

of reactor coolant per unit would be processed annually. Operating experience has shown, however, that the amount of reactor coolant typically processed during a year is higher. At any one time, the waste receiver tanks and waste monitor tanks can accommodate six system volumes of reactor coolant.

For the preoperational design estimate, annual average limiting concentrations in the discharge conduits were calculated for each isotope expected to be discharged in the liquid effluents from Calvert Cliffs. These limiting concentrations were calculated based on 1% of the concentrations in 10 CFR Part 20, Appendix B, Table II, Column 2. The intake factors were taken from Oak Ridge National Laboratory ORNL 3721, Supplement 3.

The concentration factors used were taken from the "recommended values" in Table 1 of NUS Report TM-S-121, entitled Concentration Factors of Chemical Elements in Aquatic Organisms in the Chesapeake Bay, May, 1971. For elements not listed in this report, concentration factors were obtained from Lawrence Radiation Laboratory Report UCRL-50564, entitled Concentration Factors of Chemical Elements in Edible Aquatic Organisms, December 30, 1968. The intake and concentration factors presently used to evaluate dose commitment to members of the public from radioactive materials in liquid effluents are given in the ODCM. Preoperational design estimates of expected annual average concentrations from all liquid releases from Calvert Cliffs are listed for each isotope in Table 11-2.

The thyroid dose due to the ingestion of seafood assumed to be grown in the discharge conduits is shown in Table 11-3. This is based on an estimated total discharge of 3.73 curies in the liquid effluents from the plant. It is concluded, therefore, that expected liquid releases will be below the dose objectives listed in the ODCM, considering the seafood ingestion pathway of exposure. The Radioactive Effluent Release Reports confirm that the doses to the maximum exposed individual from ingestion of seafood assumed to be grown in the discharge conduits are indeed within 10 CFR Part 50, Appendix I guidelines.

11.1.2.1.2 Reactor Coolant Waste Processing System

The RCWPS is designed to provide controlled handling and disposal of radioactive liquid wastes from both reactor plants. The system is designed to provide temporary storage for reactor coolant wastes (RCWs) and to process the liquid wastes prior to disposal. This allows the release of radioactive material to the environment to be maintained ALARA, and the concentration of the effluent maintained below the limits set forth in the ODCM.

The RCWPS is shown on Figure 11-1.

The RCWPS consists of two reactor coolant drain tanks (RCDTs), three cartridge filters, two degasifiers, four RCW ion exchangers, two RCW receiver tanks, two evaporators (the evaporators are no longer used), two RCW monitoring tanks and various system pumps. A provision has been made to process RCW using a radioactive waste processing skid. All system tie-in piping for the skid is designed and built to the original construction code and meets the intent of Regulatory Guide 1.143,

Revision 1. The system is designed to simultaneously process reactor coolant and CVCS letdown flow from both Unit 1 and Unit 2.

The RCW liquid, which contains recoverable boron, is initially stored in one of two RCW receiver tanks. The RCW receiver tanks receive waste liquid from the volume control tank letdown inlet diversion, the RCDTs, and the waste gas surge and decay tanks.

Reactor coolant is diverted to the RCWPS via the CVCS when changes in the Reactor Coolant System (RCS) inventory or boron concentration are necessitated by startups, shutdowns, fuel depletion, or draining of the RCS for maintenance. Any condition which causes a high level signal in the volume control tank of the Unit 1 or Unit 2 CVCS will direct flow to the RCWPS. The flow is directed to the in-service RCW receiver tank via a cartridge filter, degasifiers, and two reactor coolant ion exchangers. The ion exchangers may be bypassed when conditions permit.

Radioactive liquid waste is also diverted to the RCWPS via the RCDTs. The RCDTs are located in the Unit 1 (RCDT 11) and Unit 2 (RCDT 21) Containment Buildings, and receive drains from several sources in the Containment. A high level signal from the RCDT alerts the operator that the collected liquid waste must be pumped to the RCW receiver tank. To preclude any unnecessary alarm conditions, the operator, at his discretion, may pump down the drain tanks at any time.

Condensate from the waste gas system surge and decay tanks is drained to the reactor coolant liquid waste degasifiers. The condensate is then pumped to the RCW receiver tanks.

Waste liquid that is sent to the RCWPS enters through a cartridge filter so that insoluble corrosion products are removed. If the filter becomes plugged, a high differential pressure alarm is actuated on a local control panel. From the filter, the liquid waste enters a degasifier which removes hydrogen, nitrogen, and fission gases and diverts them to the waste gas system's surge tank. When the liquid level in the degasifier reaches the high setpoint, the degasifier pump starts automatically to pump the liquid waste to the ion exchangers. The pump stops when the waste level is sufficiently reduced, and the process of filling the degasifier starts again. The degasifier pump continues to operate in a cyclic manner until the liquid level in the degasifier is stabilized or until the operator stops the evolution.

The degasifier liquid is pumped through the reactor coolant ion exchangers, which remove soluble ions, to the RCW receiver tanks for storage. There are four ion exchangers that can be aligned in various ways.

When requested, the in-service RCW receiver tank is isolated and the other waste receiver tank is placed in service. The liquid in the isolated tank may then be placed in recirculation for sampling prior to discharge or evaporation, or the liquid may be sent to the other RCW receiver tank or to one of the reactor coolant waste monitor tanks.

The purpose of the RCW evaporators is to distill the liquid to remove any remaining radioactive isotopes and to concentrate boric acid. The influent comes from the primary coolant loops in the plant and thus contains boric

acid which serves as a "chemical shim" for reactor control operation. The boric acid concentrate is sampled at the evaporator for purity. Depending upon the results, the concentrate is pumped either to the boric acid storage tanks or the Spent Fuel Pool (SFP) for reuse or disposed of in accordance with the Process Control Program. The evaporator distillate is pumped to one of the two RCW monitor tanks.

If the activity level is unacceptable, the liquid is reprocessed through the ion exchangers, the miscellaneous waste system ion exchanger prefilters, temporary portable processing equipment, or any combination thereof.

If the activity level in the monitor tank is within station and regulatory limits, a waste discharge permit is prepared. The liquid from the monitor tank is normally pumped by the monitor tank pump through a flow element to the liquid waste proportional composite sampler. The flow is monitored on the associated flow indicator while throttling the discharge valve to achieve the flow rate specified by the discharge permit. The liquid then passes through a radiation monitor that continually measures the liquid's activity. If excessive radioactivity is detected, an alarm is actuated and the two air-operated liquid waste discharge isolation valves shut to stop the discharge flow. If the liquid is acceptable for discharge, it is mixed with water from the circulating water system and discharged into the Chesapeake Bay. This discharge liquid can be directed to the discharge conduits of either unit using approved plant procedures with appropriate administrative approvals.

The discharge permit will specify required controls such that the concentration of the radioactive material will be within the liquid effluent concentration limits as described in the ODCM.

A condition that results in an alarm from the liquid waste discharge radiation monitor will automatically shut the two discharge isolation valves to stop the flow of liquid to the discharge conduit(s). Should this condition occur, the radiation monitor must be flushed to the MWPS until the alarm clears. The contents of the monitor tank can then be sampled and the results evaluated in order to determine if the liquid requires further processing, and to determine the source of the radiation monitor alarm.

The liquid waste discharge radiation monitor setpoint is established so that the concentration of radioactive material released to unrestricted areas in liquid effluents does not exceed the limits of the ODCM. The setpoint is established as described in the ODCM.

The liquid waste batch in process is required to be isolated from the time the tank is placed in a recirculation mode in preparation for sampling until the discharge is terminated. If any additions have been made to the tank, the contents will be resampled prior to any discharge to the circulating water system.

11.1.2.1.3 Miscellaneous Waste Processing System

The MWPS is designed to provide controlled handling and disposal of various liquid wastes from both reactor plants, as shown in Table 11-4. The system is designed to provide temporary storage for these liquid wastes, and to process the wastes prior to their disposal. The release of radioactive material to the environment is then maintained ALARA, and the concentration of the effluent is below the limits set forth in the ODCM.

The MWPS is shown on Figure 11-2.

The MWPS consists of two tanks, one duplex filter housing, one ion exchanger, and various pumps, strainers and instruments. Connections are installed to provide portable filtration between the Solid Waste Processing System and the MWPS. The MWPS evaporator and the MWPS heat exchanger are no longer used. The system receives liquid waste from these major sources: Auxiliary Building gravity drains, soapy drains, Materials Processing Facility (MPF) laundry, and containment normal sump and pumped sumps. The MWPS also collects the process liquid wastes from the Solid Waste Processing System, Service Water (SRW) System, Component Cooling System, Blowdown Recovery System, Refueling Water Tanks, Refueling Water Tank Room sump pump, and the SFP.

The primary system interface for the MWPS is with the RCWPS. The portions of the RCWPS utilized by MWPS include the RCW receiver and monitor tanks and the reactor coolant ion exchangers. Flow from the discharge of the RCWPS joins the discharge from the MWPS and goes to the circulating water system. Another system interface is the SWPS. The SWPS receives the spent resin from the miscellaneous waste ion exchanger (MWIE) and the cleansing flush from the MWIE wye strainer.

The MWPS heat exchanger is retired in place, but should it be called into service, its cooling water is supplied by the Component Cooling System. The ion exchanger is serviced by the demineralized water system and the nitrogen system during the process of resin discharge and resin fill.

Miscellaneous liquid waste is collected in the miscellaneous waste receiver tank (MWRT) and the miscellaneous waste monitor tank (MWMT). All miscellaneous liquid waste is directed to the MWRT, except for hot laboratory and soapy drains which are normally directed to the MWMT. The system is designed to process over a million gallons of miscellaneous waste annually.

The liquid from the MWRT may be pumped through the miscellaneous waste filter, which removes suspended solids from the fluid. The MWIE may be used to lower the activity in the liquid. A wye-type strainer retains any resin beads that might become dislodged from the resin bed. The liquid then flows to the RCWPS.

If activity levels are acceptable, the liquid from the MWMT is pumped to join the discharge flow path of the RCWPS monitor tank pump upstream of the discharge flow element and radioactivity monitor, as shown on Figure 11-2. If activity levels are not acceptable, the liquid is processed via filtration, ion exchanger, or both, or the water is sent to one of the RCW monitor tanks prior to release.

11.1.2.1.4 Liquid Waste Releases

Liquid waste releases consist of RCW processing, miscellaneous waste processing, and a few other system effluents. Radioactivity released via the liquid pathway is measured and maintained within the ODCM limits and is reported to the NRC in the Radioactive Effluent Release Reports. The dose consequences of these releases are determined using the methodology in the ODCM.

Reactor Coolant Waste Processing System Effluents

The RCWPS was originally sized to process 14 RCS volumes from each unit per year based on conservative processing assumptions for evaporator operations, reactor coolant activity, etc. The system was designed to provide maximum flexibility in processing, consequently, larger volumes of liquid radioactive waste with lower radioactivity levels can be processed. In addition, the concentrations of radioisotopes in the reactor coolant are normally lower than the preoperational design estimates due to less failed fuel than originally assumed.

Table 11-4 shows the systems which contribute to the generation of RCWPS waste. Table 11-5 shows the preoperational design estimates for the equilibrium concentration of radioisotopes in the reactor coolant prior to any processing and the associated annual activity discharged after processing through the RCWPS. These values were based upon 1% failed fuel in each reactor, the recycle of boric acid and processed reactor coolant, and conservative decontamination factors for filters, demineralizers, evaporators, and degasifiers.

Miscellaneous Waste Processing System Effluents

The sources of miscellaneous wastes processed by the MWPS are shown in Table 11-4. Table 11-6 shows the preoperational design estimates for the annual isotopic discharges after processing waste through the MWPS. The average specific activity value for the RCDT was assumed to be equal to the reactor coolant activity. The remaining miscellaneous liquid was assumed to have a specific activity equal to 1% of the reactor coolant values. Steam generator blowdown specific activity was calculated based on 1% failed fuel, 50 gallons/day per steam generator tube leakage, and 0.5 gallons/minute per steam generator blowdown. The calculation of specific activity of the liquid entering the MWPS from the blowdown tank drain includes radioactive decay in the steam generators. The noble gases contained in this liquid are assumed to be released via the gaseous pathways and are included in the calculated total gaseous releases. Conservative decontamination factors were used for estimating purposes.

Other Liquid Effluents

Steam generator blowdown can be recycled by use of the steam generator blowdown recovery system. However, if the analysis of the samples taken from the steam generators indicates that gross activity is within limits, blowdown can be sent to the blowdown tank which can be discharged directly to the circulating water discharge conduits. A radiation monitor is installed in the line from the blowdown tank to the discharge conduit which would alarm on high activity and automatically shut the discharge valve which causes the flow to be routed to the MWPS. This feature provides a backup to the analysis and also terminates the discharge if the gross activity of the steam generator increases above the level allowable for direct discharge during the blowdown.

Additionally, it is expected that there will be leaks in the secondary systems. If there is any iodine or particulate activity in the steam generator, some slight quantities of activity will be carried over and be released by the leaks to the Turbine Building. Most of this activity will

remain in the liquid phase and be released via the Turbine Building drains. The preoperational design estimates for the total annual release from the Turbine Building drains are shown in Table 11-7. The drain sumps are sampled in accordance with the ODCM.

The yard oil interceptor is not normally a contaminated system; however, it could potentially become contaminated through cross-over contamination from the Turbine Building drains. The existing set of acceptance criteria found in the ODCM is used to establish the allowed contamination levels in the yard oil interceptor.

For Unit 1 only, the tendon access gallery sumps are operated as a potentially radiologically contaminated system. The existing set of acceptance criteria found in the ODCM is used to establish the allowed contamination levels in the tendon access gallery sumps.

<u>Total Liquid Effluents</u>

Liquid effluents containing, or potentially containing, radioactivity can be released from the plant to unrestricted areas through the following pathways:

a. RCWPS
b. MWPS
c. Low activity steam generator blowdown
d. Turbine Building drains

Table 11-8 lists the preoperational design estimates of annual discharges from the RCWPS, MWPS, steam generator blowdown, and Turbine Building drains. This table lists estimated releases from normal release pathways. Other potential release pathways are sampled according to approved plant procedures. Occasional releases from abnormal pathways are quantified and recorded. All releases, including these occasional releases, are maintained within the ODCM limits.

The Liquid Radwaste Processing System will be used to reduce the radioactive materials in liquid wastes prior to their discharge when the calculated doses due to the liquid effluent released to unrestricted areas exceeds limits specified in the ODCM. The dose limits apply to the combined effluent of Units 1 and 2. This requirement provides assurance that the releases of radioactive materials in liquids will be kept ALARA by implementing the requirements of 10 CFR 50.36a, 10 CFR Part 50, Appendix A, General Design Criteria 60 and the design objectives given in 10 CFR Part 50, Appendix I, Section II.D.

<u>Liquid Effluent Concentration Limits</u>

The ODCM limits on liquid effluent concentration are provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in (1) exposures to a member of the public within the design objectives of 10 CFR Part 50, Appendix I, Section II.A, and (2) exposures to the population within the limits of 10 CFR 20.1301.

Liquid Effluent Dose Limits

The ODCM liquid effluent dose limits are provided to implement the requirements of 10 CFR Part 50, Appendix I, Sections II.A, III.A and IV.A. These limits assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept ALARA, and at the same time providing the required operational flexibility. The actual effluents are provided in the Radioactive Effluent Release Report. The dose calculation methodology and parameters in the ODCM implement the requirements in Appendix I, Section III.A that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated.

The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977; Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977; and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants."

11.1.2.2 Gaseous Waste Processing System

11.1.2.2.1 Design Bases

The WGPS is designed to store the gases removed from liquid waste to allow radioactive decay of the short-lived isotopes before the gases are released from the plant. During normal operation, there is sufficient waste gas decay tank capacity to allow for adequate decay of the noble gases contained in the gases vented from the reactor coolant.

11.1.2.2.2 Waste Gas Processing System

The WGPS is designed to provide controlled handling and disposal of radioactive gaseous wastes from both reactor plants. The system is designed to store the gases removed from liquid waste and other sources to allow radioactive decay of the short-lived isotopes before the gases are released from the plant. The WGPS is shown on Figure 11-2.

The WGPS consists of a surge tank, two compressors, three waste gas decay tanks, and a high efficiency particulate air (HEPA) filter. The WGPS collects, stores, and disposes of gaseous waste from the degasifiers, pressurizer quench tanks, RCDTs, the volume control tanks, and other miscellaneous hydrogenated sources. The gaseous waste is a mixture of hydrogen, nitrogen, water vapor, ammonia, and isotopes of xenon and krypton. The principal gas released from the coolant is hydrogen; radioactive fission products, activated dissolved gases, etc., contribute a small fraction to the total volume of liberated gases.

Waste gases are collected in three headers. One header collects radioactive gases from the RCDTs and the pressurizer quench tanks and discharges to the waste gas surge tank. A second header collects radioactive gases from Auxiliary Building sources, including the RCWPS

degasifiers and evaporators and directs them to the waste gas surge tank. The third header collects other vents and relief valve discharges and connects to the gas release header upstream of the gas discharge radiation monitor. The waste gas surge tank stores the waste gas and provides a suction reservoir for the waste gas compressors. A relief valve that relieves to the plant vent upstream of the radiation monitor protects the surge tank from overpressurization. When the gas pressure in the surge tank increases to above a certain setpoint, a waste gas compressor will start and pump the gas to one of the three waste gas decay tanks where the gas is stored at a pressure not to exceed the design pressure of 150 psig. Relief valves on each decay tank discharge to the waste gas surge tank prevent the uncontrolled release of gases to the atmosphere in the event of overpressurization of the decay tanks. Rupture disks are installed upstream of each decay tank relief valve to facilitate relief valve maintenance. The decay tank is sampled, and when the activity level has decayed to an acceptable level, the contents are discharged by permit at a controlled rate, through the release header, to the plant vent. A relief valve that relieves to the surge tank protects the release header from overpressurization. The rate of release is controlled by throttling a valve. The release header contains an absolute filter, a radiation monitor, and redundant, automatic discharge isolation valves. A high radiation monitor alarm, which is annunciated in the Control Room, automatically shuts the isolation valves.

Refer to Figures 9-20A, 9-20B, and 9-21 for the plant ventilation system and to Figure 11-4 for locations of the radiation monitoring device in the main plant vent. Ventilation flow rates are shown in Table 9-18. All major subsystems exhaust into the main plant vent through a common header. An automatic damper installed in the fresh air duct leading into the main exhaust plenum will maintain a constant flow into the main exhaust plenum, even though some of the subsystems are not in operation; and thus, the flow rate from the plant vent remains approximately constant. Other inflows to the plant vent, such as the waste gas system, the condenser vacuum pump discharge and the vent from the containment gaseous monitor contribute negligible flow when compared to the ventilation flow rate.

11.1.2.2.3 Gaseous Waste Releases

Waste Gas Processing System Effluents

The WGPS is sized to allow for adequate decay prior to release of the gaseous isotopes contained in the RCS from each unit and other miscellaneous sources of gaseous wastes. Table 11-4 shows the sources of these gaseous wastes. Table 11-9 shows the preoperational design estimates for annual isotopic releases from the WGPS.

Other Gaseous Effluents

There are other potential sources of gaseous releases from the plant which are not collected in the WGPS for holdup. If leaks occur in systems containing reactor coolant, radioactive gases could be released as a

result of purging the Containment Structures, from the Condenser Air Removal Systems, Turbine Building ventilation systems and aerated tank vents. All of these potential sources except the Turbine Building ventilation are released through the plant vent.

Table 11-9 shows the preoperational design estimates for isotopic discharge from these sources. In order to estimate these releases, assumptions were made for failed fuel, volume of liquid waste processed, steam generator tube leakage, reactor coolant leakage to the containment, containment filtration and containment purging.

Total Gaseous Effluents

Gaseous effluents containing, or potentially containing, radioactivity can be released from the plant to unrestricted areas through the following pathways:

a. WGPS
b. Containment Structure purge
c. Auxiliary Building ventilation
d. Condenser air removal system and gland seal exhauster
e. Aerated tank vents
f. Turbine Building ventilation

Table 11-8 lists the preoperational design estimates of annual discharges from these pathways. This table lists normal release pathways. Other potential release pathways are sampled according to approved plant procedures. Occasional releases from abnormal pathways are quantified and recorded. All releases, including these occasional releases, are maintained within ODCM limits.

The Gaseous Radwaste Processing System will be used to reduce the radioactive materials in gaseous wastes prior to their discharge whenever a suitable fraction of the dose design objectives set forth in 10 CFR Part 50, Appendix I, Sections II.B and II.C are reached. The dose limits apply to the combined effluent of Units 1 and 2. This requirement provides assurance that the releases of radioactive materials in gaseous effluents will be kept ALARA by implementing the requirements of 10 CFR 50.36a, 10 CFR Part 50, Appendix A, General Design Criteria 60 and the design objectives given in 10 CFR Part 50, Appendix I, Section II.D.

Gaseous Effluent Dose Rate

The ODCM limits on gaseous effluent dose rate is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area, exceeding the limits specified in 10 CFR Part 20, Appendix B, Table II (10 CFR 20.1301). For members of the public who may at times be within the site boundary, the amount of time that they are expected to remain at the site will be

sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary.

Gaseous Effluent Dose Limitations

The ODCM limits on gaseous effluent dose are provided to implement the requirements of 10 CFR Part 50, Appendix I, Sections II.B (for noble

gases), II.C (for Iodine-131 and radionuclides in particulate form), III.A and IV.A. These limits will ensure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept ALARA, and at the same time provide the required operational flexibility. The ODCM surveillance requirements implement the requirements in Appendix I, Section III.A that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive matter in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977; Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Reactors," Revision 1, July 1977; and NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." The ODCM equations for determining the air doses at and beyond the site boundary are based upon the historical annual average atmospheric conditions.

The release rate limits for Iodine-131 and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

11.1.2.3 Solid Waste Processing System

11.1.2.3.1 Design Bases

Solid waste is to be packaged so as to meet the applicable requirements of 10 CFR Parts 61 and 71 and 49 CFR for burial and transportation. The SWPS provides the capability for preparing solid waste for shipment to an offsite disposal facility or processor. The system is designed to minimize radiation exposure to personnel during the handling of solid wastes. The process parameters are included in the Process Control Program.

11.1.2.3.2 Solid Waste Processing System

The SWPS is designed to provide for the controlled handling and offsite disposal of solid waste originating from both units.

The SWPS equipment is located at Elevation 45'0" and 30'0" in the Auxiliary Building.

Spent radioactive ion exchanger resin is sluiced to a shielded spent resin metering tank at Elevation 45'0", where it is stored and partially dewatered via portable filtration to the MWRT. The portable filtration system may be bypassed as permitted. It is then sluiced to a suitable container where it is prepared for shipment. This operation is controlled

from Elevation 45'0", so that the operators are shielded from primary radioactivity sources.

Reactor Coolant Waste Processing System evaporator bottoms are normally recycled. If they are not recycled, they may be disposed of in accordance with an approved process control program.

Radioactive filters are transported from each filter housing to the waste disposal area. A shielded filter transfer cask is available to transport the filters if required due to high radiation levels. The filters are lowered from the transfer cask to a shipping container or placed in a suitable storage area.

11.1.2.3.3 Solid Waste Releases

All solid wastes are packaged in containers suitable for burial or transfer to an offsite processor prior to leaving the site.

11.1.2.4 Materials Processing Facility

The MPF was designed in accordance with Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," to provide interim storage of dry active waste (DAW) until they can be shipped to a permanent disposal facility. The storage capacity of the MPF does not exceed the expected waste generated at Calvert Cliffs for five years, based upon normal rates of operation and generation when the facility was designed. Provisions have been provided for expansion for an additional five years, if required.

The MPF is located on the plant site, outside of the protected area, and has an established physical security program consisting of locked doors and periodic patrols. Radiation protection has been incorporated into the structural design to prevent dose rates outside the facility from exceeding established limits in uncontrolled, unlimited access areas. A restricted area for radiation protection has also been established. Area radiation monitors or portable survey meters are used to monitor the general area radiation levels at various locations in the MPF.

The MPF is included in the site's Structural Monitoring Program. Design criteria were established according to the requirements of GL 81-38 to maintain the onsite radiation exposure ALARA and the calculated offsite contribution to less than 1 mrem per year. Design evaluations have demonstrated that the offsite dose resulting from accident scenarios involving the MPF would be negligible when compared to the Waste Evaporator Incident in Section 14.23.

The overall functions of the MPF are:

a. Interim storage of DAW and low-level processed wastes.
b. Decontamination of clothing, respirators, tools, hardware, and radioactive material.
c. Provide the capability for temporary hold-up of liquid wastes generated during laundry and decontamination activities.
d. Provide capability for receiving, sorting, compacting, packaging and offsite return shipment of DAW. Receipt of offsite shipments of site's DAW is only permitted if the DAW was originally generated at Calvert Cliffs, then shipped offsite for volume reduction and returned to the site for interim storage.
e. Provide office space for radwaste management activities.
f. Additional capability is provided for storage of spare plant equipment and components.

g. The processing of liquid waste in the Decontamination Area of the MPF in preparation of offsite shipment. The volume of liquid waste in the MPF is limited to two 55-gallon drums (one being processed and one staged for processing), with secondary containment devices designed to hold the contents of both drums. The quantity of liquid waste to be processed and staged in the MPF is limited so that curie content will be within the limits of the following.

Isotope	Activity (μCi)
Sn-113	20.10
CE-144	16.10
Zr-95	38.20
Co-57	16.30
Co-58	2871.00
Co-60	779.00
Cs-134	50.00
Cs-137	736.00
Mn-54	151.00
Sb-125	121.00
Total	4798.70

To accomplish these functions, the MPF is physically divided into several areas: interim storage, DAW processing, decontamination, office space, miscellaneous storage, and shipping and receiving.

Packages containing gaseous wastes, wastes containing free liquids, solidified wastes, or dewatered wastes are prohibited from storage in the MPF since the facility is a non-seismic and non-safety-related structure. The liquid waste generated in the MPF laundry facility is collected, sampled, when necessary, and then transferred to the Auxiliary Building, via a transfer truck, for processing through the existing plant radwaste system.

Dry active waste is collected from central sites at both units and delivered to the MPF's DAW processing area where it is sorted and segregated according to activity levels. Activities performed in the decontamination facility include the removal of residual surface and fixed thin layer contamination from plant equipment. For respirators, a wet cleaning method is utilized. All waste is then packaged, shipped offsite for processing and volume reduction (if appropriate), then returned to the site for storage in the interim storage area, or shipped to an offsite burial facility. As a precaution, the stored waste is isolated from the MPF processing areas.

11.1.2.5 Interim Resin Storage Facility

The interim resin storage facility is located in the Lake Davies area of the site (Figure 1-1) and was designed in accordance with Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites." Waste to be stored at the facility is limited to spent resins and filters and is processed by the Solid Waste Processing System (Section 11.1.2.3). The following waste forms are specifically excluded from storage at the interim resin storage facility: mixed wastes, spent fuel or fuel-related wastes, liquid or gaseous wastes, and wastes not generated by Calvert Cliffs Nuclear Power Plant.

Design criteria were established according to the requirements of Generic Letter 81-38 to maintain the onsite radiation exposure ALARA, and the calculated offsite contribution to the nearest permanent resident to less than 1 mrem per year.

11.1.2.6 West Road Cage

The West Road Cage is located due west of the Auxiliary Building on the 45' Elevation and evaluated in accordance with Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," to provide interim storage for up to five years (unless NRC approval for an extended storage time is obtained). Waste to be stored at the facility is limited to spent resins and filters and is processed by the Solid Waste Processing System (Section 11.1.2.3). Other radioactive materials contaminated by byproduct material may be stored in the area provided the materials are controlled in accordance with the Materials Safety Program (Section 11.4.1).

11.1.2.7 Original Steam Generator Storage Facility

The Original Steam Generator Storage Facility is a reinforced concrete structure with a footprint measuring approximately 75' x 85'. The building size allows for the storage of four original steam generator (OSG) lower assemblies in two bays. The building is located on the West side of the plant, just North of the Independent Spent Fuel Storage Installation. The structure is constructed of cast-in-place reinforced concrete with the exception of the East wall, which is constructed of pre-cast concrete sections. These sections allow for placement of the OSG lower assemblies inside the building and simplify the eventual removal of the OSG lower assemblies from storage. The facility is designed to house the OSG lower assemblies until the plant is decommissioned and the OSG lower assemblies are permanently disposed of.

Nuclear Regulatory Commission Generic Letter 81-38, was considered in the design of the Original Steam Generator Storage Facility, since this document applies to onsite storage of radwaste. However, in accordance with the intent of NRC Inspection Procedure 50001, "Steam Generator Replacement Inspection," for purposes of onsite storage of the OSGs, NRC Generic Letter 81-38 was reviewed only for applicability of proper controls for facility access and dose rates at the perimeter to ensure compliance with the limits of 10 CFR Part 20.

11.1.3 MONITORING INSTRUMENTATION

The monitoring instrumentation used to measure, record, and control the release of radioactivity to unrestricted areas includes the liquid waste discharge, the steam generator blowdown, and the main vent radiation monitoring systems. A list of the effluent monitors is provided in Table 11-10. The methodology for establishing the monitor trip setpoints is described in the ODCM. Calibration frequency is discussed in Section 11.2.3.

The liquid waste discharge, steam generator blowdown and main vent gaseous radiation monitoring systems are able to measure and control effluent releases within the requirements of the ODCM.

Liquid waste and steam generator blowdown monitoring must be in compliance with the ODCM during liquid releases from these pathways. These monitors are used to automatically terminate the release should the setpoint be reached, and to estimate the quantity of radioactive material discharged as a backup to laboratory analyses. Release of expected volumes of liquids below the monitor setpoints will ensure that ODCM levels will not be exceeded.

The main vent monitoring systems consist of a gas monitor and a fixed filter for particulate and iodine sampling. A laboratory isotopic analysis of the samples collected on the fixed filters is used to demonstrate compliance with the release limits for particulates and iodine.

Under normal operation, the blowdown tank vent is routed to Feedwater Heaters No. 13A and 13B, shell side for Unit 1, and to the heater drain tanks for Unit 2. Line-up of the blowdown tank vent to the heater drain tank is preferred during Unit 1 startup up to 30% power to prevent water hammering in the Feedwater heaters No. 13A and 13B. Therefore, there will be no direct pathway for radioactive material from blowdown vents to unrestricted areas. Any radioactivity in the blowdown tank vents will be monitored by the condenser offgas and the main vent monitoring systems.

11.1.4 TESTS AND INSPECTIONS

Functional tests and inspections to the waste processing system are made as required to ensure performance consistent with the requirements of 10 CFR Part 50, Appendix I and the ODCM. Routine surveillance is conducted during operator tours for detection of system leaks and for monitoring of system performance. Radiation detectors and monitors are routinely checked for operability. Alarm circuits and automatic features of flow diversion for waste liquid and gaseous effluents are routinely tested. The monitors are calibrated in accordance with the ODCM and approved plant procedures.

11.1.5 CONCLUSIONS

The waste processing systems are designed to allow flexible and reliable operations. All waste processing components in the RCWPS are redundant. In the event the components in the MWPS are inoperative, liquid waste normally processed by this system can be transferred to the RCWPS. If both redundant RCWPS components and the MWPS components are inoperative simultaneously, sufficient tank capacity exists in the RCWPS to hold 360,000 gallons. Both the RCWPS and MWPS are designed to allow reprocessing of liquid in the event activity levels are too high after the initial processing. Three gas decay tanks are provided, each of which is sized to store the gaseous waste produced by a simultaneous cooldown and degasification of both units. The waste processing systems are designed to prevent uncontrolled releases to the environment in the event of the failure of a single active component or an operator error.

For the preoperational design estimates, the total annual radiological impact as a result of operating Calvert Cliffs Unit 1 and Unit 2 with 1% failed fuel is shown in Table 11-3. The actual releases are provided in the Radioactive Effluent Release Reports.

Actual effluent data provided in the Radioactive Effluent Release Reports has been used as input to the calculational methods provided in the ODCM in order to calculate the offsite doses. These calculated doses are within the guidelines of 10 CFR Part 50, Appendix I and the limits of the ODCM.

To meet the dose limitations of 40 CFR Part 190, the ODCM requires the preparation and submittal of a special report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. The special report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR Part 190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible.

TABLE 11-1

WASTE PROCESSING SYSTEMS COMPONENT DESCRIPTION

MISCELLANEOUS WASTE RECEIVER TANK PUMP

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	One
Capacity (gpm)	120
Head (ft of H_2O)	110
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	7.5 hp, 3 phase, 60 Hz, 480 Volt
Code	National Electrical Manufacturers Association (NEMA), American Society of Mechanical Engineers (ASME) - Pumps and Valves for Nuclear Power

MISCELLANEOUS WASTE MONITOR TANK PUMP

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	One
Capacity (gpm)	120
Head (ft of H_2O)	110
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	7.5 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

MISCELLANEOUS WASTE MONITOR TANK METERING PUMP

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	One
Capacity (gpm)	10
Head (ft of H_2O)	60
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	1 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

MISCELLANEOUS WASTE RECEIVER TANK[a]

Type	Horizontal
Quantity	One
Design Pressure	Atmospheric
Volume (gal)	4000
Code	ASME Section III, Class C
Material	304 Stainless Steel

TABLE 11-1

WASTE PROCESSING SYSTEMS COMPONENT DESCRIPTION

MISCELLANEOUS WASTE MONITOR TANK[(a)]

Type	Horizontal
Quantity	One
Design Pressure	Atmospheric
Volume (gal)	4000
Code	ASME Section III, Class C (Code “N” Stamp Removed)
Material	304 Stainless Steel

ION EXCHANGERS

Type	Mixed bed, non-regenerable
Quantity	Five
Design Pressure (psig)	200
Flow (gpm)	128
Material:	
Vessel Shell	ASME Section III, Class C; Section VIII, (Paragraph UW-2a, applies)
Vessel Head	ASME SA240, Type 304
Internals	ASME SA240, Type 304 Austenitic Stainless Steel

FILTERS

Type	Cartridge
Quantity	Three
Design Pressure (psig)	125
Flow (gpm)	120
Code	ASME Section III, Class C

WASTE GAS COMPRESSORS

Type	Single stage, 1 head, diaphragm
Quantity	Two
Capacity (scfm)	4.0 to 7.0
Design Discharge Pressure (psig)	150
Motor	3.0 hp, 3 phase, 60 Hz, 460 Volt
Code	NEMA

WASTE GAS SURGE TANK[(a)]

Type	Vertical
Design Pressure (psig)	50
Volume (ft^3)	610
Code	ASME Section III, Class C
Material	304 Stainless Steel

WASTE GAS DECAY TANKS[(a)]

Type	Vertical
Quantity	Three
Design Pressure (psig)	150
Volume (ft^3)	610
Code	ASME Section III, Class C
Material	American Society for Testing and Materials (ASTM) A264, Type 304 Stainless Steel clad

TABLE 11-1

WASTE PROCESSING SYSTEMS COMPONENT DESCRIPTION

REACTOR COOLANT DRAIN TANK PUMPS

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	Two
Capacity (gpm)	100
Head (ft of H_2O)	113
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	7.5 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

DEGASIFIER PUMPS

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	Two
Capacity (gpm)	150
Head (ft of H_2O)	221
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	20 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

REACTOR COOLANT WASTE RECEIVER TANK PUMPS

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	Two
Capacity (gpm)	120
Head (ft of H_2O)	204
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	15 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

REACTOR COOLANT WASTE MONITOR TANK PUMPS

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	Two
Capacity (gpm)	120
Head (ft of H_2O)	204
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	15 hp
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

TABLE 11-1

WASTE PROCESSING SYSTEMS COMPONENT DESCRIPTION

REACTOR COOLANT WASTE MONITOR TANK METERING PUMP

Type	Horizontal centrifugal, end suction, mechanical seal
Quantity	One
Capacity (gpm)	10
Head (ft of H_2O)	60
Material:	
Case	316 Stainless Steel
Impeller	316 Stainless Steel
Shaft	316 Stainless Steel
Motor	15 hp, 3 phase, 60 Hz, 480 Volt
Code	NEMA, ASME - Pumps and Valves for Nuclear Power

REACTOR COOLANT DRAIN TANK[(a)]

Type	Horizontal
Quantity	Two
Design Pressure (psig)	50
Volume (gal.)	900
Code	ASME Section III, Class C
Material	304 Stainless Steel

REACTOR COOLANT WASTE RECEIVER TANKS[(a)]

Type	Vertical
Quantity	Two
Design Pressure (psia)	15
Volume (gal.)	90,000
Code	ASME Section VIII
Material	304 Stainless Steel

REACTOR COOLANT WASTE MONITOR TANKS[(a)]

Type	Vertical
Quantity	Two
Design Pressure (psia)	15
Volume (gal.)	90,000
Code	ASME Section VIII
Material	304 Stainless Steel

REACTOR COOLANT DEGASIFIERS

Type	Packed tower
Quantity	Two
Design Pressure (psia)	75
Reactor coolant bleed (gpm)	0-120
Code	ASME Section III, Class C; ANSI B31.1

TABLE 11-1

WASTE PROCESSING SYSTEMS COMPONENT DESCRIPTION

LIQUID WASTE EVAPORATORS

Quantity	Two
Type	Horizontal, vacuum
Design Pressure (psia)	30
Design Distillate Flow (gpm)	20
Code	ASME Section III, Class C
Mode of Operation	Batch (RCWPS), Continuous (MWPS)
Feed Tank Capacity (gal)	1000

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[a] Additional design and quality control requirements for these tanks are given in Section 6.3.5.1.

TABLE 11-2

PREOPERATIONAL DESIGN ESTIMATES OF LIQUID EFFLUENT CONCENTRATIONS

(Operation with 0.1% Failed Fuel)

ISOTOPE	PREOPERATIONAL ESTIMATED DISCHARGE CONCENTRATIONS (μCi/cc)
Sr-89	4.56×10^{-14}
Sr-90	9.22×10^{-15}
Sr-91	3.30×10^{-14}
Y-90	8.94×10^{-15}
Y-91	3.04×10^{-13}
Mo-99	2.33×10^{-11}
Ru-103	1.04×10^{-13}
Ru-106	8.41×10^{-15}
Te-129	2.25×10^{-13}
Te-132	3.90×10^{-12}
I-129	1.68×10^{-17}
I-131	3.56×10^{-10}
I-132	4.90×10^{-11}
I-133	2.86×10^{-10}
I-134	2.72×10^{-11}
I-135	1.24×10^{-10}
Cs-134	3.46×10^{-12}
Cs-136	4.53×10^{-13}
Cs-137	1.14×10^{-11}
Ba-140	1.08×10^{-13}
La-140	6.13×10^{-14}
Ce-144	1.01×10^{-11}
Pr-143	5.19×10^{-14}
Cr-51	4.61×10^{-14}
Mn-54	3.54×10^{-14}
Mn-56	2.75×10^{-11}
Fe-59	2.52×10^{-14}
Co-58	4.84×10^{-12}
Co-60	6.34×10^{-13}
Zr-95	2.96×10^{-15}
H-3	1.79×10^{-7}

TABLE 11-3

PREOPERATIONAL DESIGN ESTIMATES OF TOTAL ANNUAL RADIOLOGICAL IMPACT

(Operation With 0.1% Failed Fuel in Both Reactors)

ANNUAL LIQUID RELEASES

		ANNUAL AVERAGE LIQUID EFFLUENT CONCENTRATION IN CIRCULATING WATER (μCi/cc)	TOTAL CURIES RELEASED (Ci/yr)	MAXIMUM INDIVIDUAL DOSE (mrem/yr)	REMARKS
1.	Tritium	1.79×10^{-7}	682	4.13×10^{-4}	Whole body dose from seafood ingestion.
2.	All isotopes except tritium and noble gases	9.40×10^{-10}	3.73	5.84×10^{-2}	Whole body dose from seafood ingestion.
				3.37×10^{-4}	Thyroid dose from seafood ingestion.

ANNUAL GASEOUS RELEASES

		ANNUAL AVERAGE CONCENTRATION (μCi/cc)	TOTAL CURIES RELEASED (Ci/yr)	MAXIMUM INDIVIDUAL DOSE (mrem/yr)	REMARKS
1.	Noble gases in liquid	1.22×10^{-13}	1.61	2.29×10^{-4}	Whole body dose at site boundary assuming gases come out of the solution at the end of discharge conduit.
2.	Noble gases (excluding noble gases in liquid)	6.41×10^{-10}	9.19×10^{3}	1.23	Whole body dose at site boundary.
3.	Iodine-131	6.79×10^{-17}	0.0305	0.192	Child thyroid dose due to ingestion of milk.
4.	All iodine isotopes	6.22×10^{-15}	0.0890	0.0308	Thyroid inhalation dose at site boundary.
5.	Particulates	1.77×10^{-18}	7.99×10^{-4}	7.52×10^{-5}	Dose calculation assumes all particulates are Cs-137 (actually Cs-137 is less than 40%). Dose to liver of standard man due to the ingestion of milk and beef.

TABLE 11-4

RADIOACTIVE WASTE PROCESSING SYSTEM SOURCES

REACTOR COOLANT WASTE PROCESSING SYSTEM	MISCELLANEOUS WASTE PROCESSING SYSTEM	WASTE GAS PROCESSING SYSTEM
Chemical and Volume Control System	Laboratory Sink Drains	RCDT No. 11
Regenerative Heat Exchanger No. 21	Spent Fuel Cask	RCDT Washdown Area Drains
Safety Injection Tanks Leakage	Spent Fuel Storage Area Drains	Pressure Quench Tank No. 11
Loop 11A Drains	Low Point Piping Drains	Pressure Quench Tank No. 21
Loop 11B Drains	Auxiliary Building Gravity Drains	Volume Control Tank No. 11
Loop 12A Drains	Auxiliary Building Pumped Sumps	Volume Control Tank No. 21
Loop 12B Drains	Laundry	Degasifier No. 11 and No. 12
Quench Tank Drains	Showers	Evaporator No. 11 and No. 12
Flange Leakage Detector Drain	Component Cooling System Relief Valves	Miscellaneous Sources
Loop No. 11 Hot Leg Drain	Boric Acid Preparation Area Drains	
Leakoff from Valves in Containment	Equipment Drains	
Reactor Coolant Pump Seal Leakage	High & Low Pressure Safety Injection Pump Seal and Bearing Water	
	Blowdown Tank Drain	
	Charging Pumps	

TABLE 11-5

PREOPERATIONAL DESIGN ESTIMATES OF RADIOACTIVITY CONCENTRATIONS IN REACTOR COOLANT AND REACTOR COOLANT WASTE PROCESSING SYSTEM EFFLUENTS

(Operation With 1% Failed Fuel)

ISOTOPE	CONCENTRATION IN REACTOR COOLANT (μCi/cc)	ANNUAL DISCHARGE (Ci/yr)	ISOTOPE	CONCENTRATION IN REACTOR COOLANT (μCi/cc)	ANNUAL DISCHARGE (Ci/yr)
Br-84	4.66×10^{-2}	1.53×10^{-2}	Xe-133	181	1.49x10
Kr-85m	1.49	1.2×10^{-1}	Te-134	2.62×10^{-2}	8.60×10^{-3}
Kr-85	8.85×10^{-1}	7.28×10^{-2}	I-134	6.20×10^{-1}[a]	1.0
Kr-87	8.1×10^{-1}	6.66×10^{-2}	Cs-134	1.0×10^{-1}	3.29×10^{-2}
Kr-88	2.6	2.14×10^{-1}	I-135	2.7[a]	4.4
Rb-88	2.55	8.0×10^{-1}	Xe-135	7.53	6.20×10^{-1}
Rb-89	6.4×10^{-2}	2.11×10^{-2}	Cs-136	2.55×10^{-2}	8.40×10^{-3}
Sr-89	5.07×10^{-3}	1.66×10^{-3}	Cs-137	3.20×10^{-1}	1.05×10^{-1}
Sr-90	2.61×10^{-4}	0.85×10^{-1}	Xe-138	3.60×10^{-1}	2.96×10^{-2}
Y-90	1.02×10^{-3}	0.33×10^{-3}	Cs-138	6.90×10^{-1}	2.27×10^{-1}
Sr-91	3.56×10^{-3}	1.17×10^{-3}	Ba-140	6.11×10^{-3}	2.01×10^{-3}
Y-91	1.11×10^{-1}	3.65×10^{-2}	La-140	5.85×10^{-3}	1.92×10^{-3}
Mo-99	2.03	6.68×10^{-1}	Ce-144	4.0×10^{-3}	1.36×10^{-3}
Ru-103	4.13×10^{-3}	1.36×10^{-3}	Pr-143	5.80×10^{-3}	1.92×10^{-3}
Ru-106	2.48×10^{-4}	0.82×10^{-4}	Cr-51	3.8×10^{-5}	1.25×10^{-4}
Te-129	2.51×10^{-2}	0.83×10^{-2}	Mn-54	2.75×10^{-5}	9.06×10^{-5}
I-129	7.21×10^{-8}	1.19×10^{-7}	Mn-56	2.30×10^{-2}	7.57×10^{-2}
I-131	3.97[a]	6.74	Fe-59	2.13×10^{-5}	7.01×10^{-5}
Xe-131m	1.48	1.22×10^{-1}	Co-58	4.66×10^{-3}	1.53×10^{-2}
Te-132	3.30×10^{-1}	1.09×10^{-1}	Co-60	5.19×10^{-4}	1.71×10^{-3}
I-132	1.09[a]	1.8	Zr-95	9.35×10^{-7}	3.08×10^{-6}
I-133	5.66[a]	9.3	H-3	1.309×10^{-1}	1.077×10^{3}

[a] Conservative preoperational estimate for design purposes.

Total non-H^3 = 41.5 Ci/yr

Total non-H^3 Less Gases = 25.4 Ci/yr

TABLE 11-6

PREOPERATIONAL DESIGN ESTIMATES OF MISCELLANEOUS WASTE PROCESSING SYSTEM EFFLUENTS

(Operation With 1% Failed Fuel)

ISOTOPE	BLOWDOWN (Ci/yr)	OTHER (Ci/yr)	TOTAL (Ci/yr)
Br-84	2.44×10^{-5}	4.47×10^{-4}	4.71×10^{-4}
Rb-88	7.56×10^{-6}	2.44×10^{-2}	2.44×10^{-2}
Rb-89	7.08×10^{-11}	6.13×10^{-4}	6.13×10^{-4}
Sr-89	1.57×10^{-7}	4.86×10^{-5}	4.88×10^{-5}
Sr-90	1.59×10^{-6}	2.50×10^{-6}	4.09×10^{-6}
Y-90	5.84×10^{-7}	9.75×10^{-6}	1.03×10^{-5}
Sr-91	3.34×10^{-7}	3.41×10^{-5}	3.44×10^{-5}
Y-91	4.72×10^{-4}	1.07×10^{-3}	1.54×10^{-3}
Mo-99	1.21×10^{-3}	1.94×10^{-2}	2.06×10^{-2}
Ru-103	1.54×10^{-5}	3.96×10^{-5}	5.50×10^{-5}
Ru-106	1.43×10^{-6}	2.37×10^{-6}	3.80×10^{-6}
Te-129	2.76×10^{-7}	2.41×10^{-4}	2.41×10^{-4}
I-129	2.20×10^{-10}	3.47×10^{-9}	3.69×10^{-9}
I-131	2.88×10^{-2}	1.88×10^{-1}	2.17×10^{-1}
Te-132	2.28×10^{-4}	3.17×10^{-3}	3.40×10^{-3}
I-132	1.21×10^{-4}	5.24×10^{-2}	5.25×10^{-2}
I-133	5.64×10^{-3}	2.72×10^{-1}	2.78×10^{-1}
Te-134	1.60×10^{-7}	2.51×10^{-4}	2.51×10^{-4}
I-134	2.68×10^{-5}	2.98×10^{-2}	2.98×10^{-2}
Cs-134	5.94×10^{-4}	9.58×10^{-4}	1.55×10^{-3}
I-135	8.78×10^{-4}	1.30×10^{-1}	1.31×10^{-1}
Cs-136	5.20×10^{-5}	2.44×10^{-4}	2.96×10^{-4}
Cs-137	1.97×10^{-3}	3.07×10^{-3}	5.04×10^{-3}
Cs-138	3.64×10^{-6}	6.62×10^{-3}	6.62×10^{-3}
Ba-140	1.24×10^{-5}	5.86×10^{-5}	7.10×10^{-5}
La-140	2.16×10^{-6}	5.61×10^{-5}	5.83×10^{-5}
Pr-143	5.56×10^{-9}	5.56×10^{-5}	5.56×10^{-5}
Ce-144	2.32×10^{-5}	3.83×10^{-5}	6.15×10^{-5}
Cr-51	1.20×10^{-5}	3.66×10^{-5}	4.86×10^{-5}
Mn-54	1.56×10^{-5}	2.64×10^{-5}	4.20×10^{-5}
Mn-56	5.80×10^{-3}	2.21×10^{-2}	2.79×10^{-2}
Co-58	2.10×10^{-3}	4.48×10^{-3}	6.58×10^{-3}
Fe-59	8.24×10^{-6}	2.05×10^{-5}	2.87×10^{-5}
Co-60	3.14×10^{-4}	4.99×10^{-4}	8.13×10^{-4}
Zr-95	4.02×10^{-7}	9.00×10^{-7}	1.30×10^{-6}
H-3	4.02×10^{1}	6.29×10^{1}	1.03×10^{2}
Total Non-H-3			8.09×10^{-1}
Total H-3			1.03×10^{2}

TABLE 11-7

PREOPERATIONAL DESIGN ESTIMATES OF TOTAL ANNUAL LIQUID RELEASE VIA TURBINE BUILDING DRAINS

(Operation with 1% Failed Fuel)

ISOTOPE	ANNUAL LIQUID RELEASE (Ci/yr plant)
I-129	4.44×10^{-7}
I-131	5.72
I-132	2.42×10^{-2}
I-133	1.14
I-134	5.38×10^{-3}
I-135	1.76×10^{-1}
Total Iodine	7.07
Total Particulates	5.0×10^{-1}
Total H-3	4.04×10^{-2}

TABLE 11-8

PREOPERATIONAL DESIGN ESTIMATES OF ANNUAL DISCHARGES

(Operation With 0.1% Failed Fuel)

LIQUID RELEASES (Ci/yr)

ISOTOPE	RCWPS	MWPS	LOW ACTIVITY STEAM GENERATOR BLOWDOWN	TURBINE BUILDING DRAINS	TOTAL
Br-84	$1.53x10^{-3}$	$4.71x10^{-5}$	$4.01x10^{-6}$	--	$1.58x10^{-3}$
Kr-85m	$1.20x10^{-2}$	--	--	--	$1.20x10^{-2}$
Kr-85	$7.28x10^{-3}$	--	--	--	$7.28x10^{-3}$
Kr-87	$6.66x10^{-3}$	--	--	--	$6.66x10^{-3}$
Kr-88	$2.14x10^{-2}$	--	--	--	$2.14x10^{-2}$
Rb-88	$8.00x10^{-2}$	$2.44x10^{-3}$	$1.26x10^{-4}$	--	$8.25x10^{-2}$
Rb-89	$2.11x10^{-3}$	$6.13x10^{-5}$	$1.17x10^{-9}$	--	$2.17x10^{-3}$
Sr-89	$1.66x10^{-4}$	$4.88x10^{-6}$	$2.58x10^{-6}$	--	$1.74x10^{-4}$
Sr-90	$8.50x10^{-6}$	$4.09x10^{-7}$	$2.63x10^{-5}$	--	$3.52x10^{-5}$
Y-90	$3.30x10^{-5}$	$1.03x10^{-6}$	$9.65x10^{-6}$	--	$4.37x10^{-5}$
Sr-91	$1.17x10^{-4}$	$3.44x10^{-6}$	$5.52x10^{-6}$	--	$1.26x10^{-4}$
Y-91	$3.65x10^{-3}$	$1.54x10^{-4}$	$7.79x10^{-3}$	--	$1.16x10^{-2}$
Mo-99	$6.68x10^{-2}$	$2.06x10^{-3}$	$2.00x10^{-2}$	--	$8.89x10^{-2}$
Ru-103	$1.36x10^{-4}$	$5.50x10^{-6}$	$2.54x10^{-4}$	--	$3.96x10^{-4}$
Ru-106	$8.20x10^{-6}$	$3.80x10^{-7}$	$2.35x10^{-5}$	--	$3.21x10^{-5}$
Te-129	$8.30x10^{-4}$	$2.41x10^{-5}$	$4.55x10^{-6}$	--	$8.59x10^{-4}$
I-129	$1.19x10^{-8}$	$3.69x10^{-10}$	$7.28x10^{-9}$	$4.44x10^{-8}$	$6.40x10^{-8}$
I-131	$6.74x10^{-1}$	$2.17x10^{-2}$	$9.54x10^{-2}$	$5.72x10^{-1}$	1.36
Xe-131m	$1.22x10^{-2}$	--	--	--	$1.22x10^{-2}$
Te-132	$1.09x10^{-2}$	$3.40x10^{-4}$	$3.71x10^{-3}$	--	$1.49x10^{-2}$
I-132	$1.80x10^{-1}$	$5.25x10^{-3}$	$3.97x10^{-4}$	$2.42x10^{-3}$	$1.87x10^{-1}$
I-133	$9.30x10^{-1}$	$2.78x10^{-2}$	$1.87x10^{-2}$	$1.14x10^{-1}$	1.09
Xe-133	1.49	--	--	--	1.49
Te-134	$8.60x10^{-4}$	$2.51x10^{-5}$	$2.98x10^{-6}$	--	$8.88x10^{-4}$
I-134	$1.00x10^{-1}$	$2.98x10^{-3}$	$8.85x10^{-5}$	$5.38x10^{-4}$	$1.04x10^{-1}$
Cs-134	$3.29x10^{-3}$	$1.55x10^{-4}$	$9.77x10^{-3}$	--	$1.32x10^{-2}$
I-135	$4.40x10^{-1}$	$1.31x10^{-2}$	$2.89x10^{-3}$	$1.76x10^{-2}$	$4.74x10^{-1}$
Xe-135	$6.20x10^{-2}$	--	--	--	$6.20x10^{-2}$
Cs-136	$8.40x10^{-4}$	$2.96x10^{-5}$	$8.60x10^{-4}$	--	$1.73x10^{-3}$
Cs-137	$1.05x10^{-2}$	$5.04x10^{-4}$	$3.24x10^{-2}$	--	$4.34x10^{-2}$
Xe-138	$2.96x10^{-3}$	--	--	--	$2.96x10^{-3}$
Cs-138	$2.27x10^{-2}$	$6.62x10^{-4}$	$6.00x10^{-5}$	--	$2.33x10^{-2}$
Ba-140	$2.01x10^{-4}$	$7.10x10^{-6}$	$2.04x10^{-4}$	--	$4.12x10^{-4}$

TABLE 11-8

PREOPERATIONAL DESIGN ESTIMATES OF ANNUAL DISCHARGES

(Operation With 0.1% Failed Fuel)

LIQUID RELEASES (Ci/yr)

ISOTOPE	RCWPS	MWPS	LOW ACTIVITY STEAM GENERATOR BLOWDOWN	TURBINE BUILDING DRAINS	TOTAL
La-140	1.92×10^{-4}	5.83×10^{-6}	3.59×10^{-5}	--	2.34×10^{-4}
Ce-144	1.36×10^{-4}	6.15×10^{-6}	3.84×10^{-2}	--	3.85×10^{-2}
Pr-143	1.92×10^{-4}	5.56×10^{-6}	9.20×10^{-8}	--	1.98×10^{-4}
Cr-51	1.25×10^{-4}	4.86×10^{-5}	1.99×10^{-6}	--	1.76×10^{-4}
Mn-54	9.06×10^{-5}	4.20×10^{-5}	2.57×10^{-6}	--	1.35×10^{-4}
Mn-56	7.57×10^{-2}	2.79×10^{-2}	9.57×10^{-4}	--	1.05×10^{-1}
Fe-59	7.01×10^{-5}	2.87×10^{-5}	1.36×10^{-6}	--	1.00×10^{-4}
Co-58	1.53×10^{-2}	6.58×10^{-3}	3.48×10^{-5}	--	2.22×10^{-2}
Co-60	1.71×10^{-3}	8.13×10^{-4}	5.34×10^{-5}	--	2.58×10^{-3}
Zr-95	3.08×10^{-6}	1.30×10^{-6}	6.80×10^{-8}	--	4.45×10^{-6}
H-3	5.43×10^{2}	5.19×10^{1}	1.33×10^{-2}	8.74×10^{1}	6.82×10^{2}
Particulates	--	--	--	5.27×10^{-2}	5.27×10^{-2}
	Total activity in liquid excluding H-3 and noble gases				3.73
	Total H-3 in liquid				6.82×10^{2}
	Total noble gases in liquid				1.61

TABLE 11-8

PREOPERATIONAL DESIGN ESTIMATES OF ANNUAL DISCHARGES

(Operation With 0.1% Failed Fuel)

GASEOUS RELEASES (Ci/yr)

ISOTOPE	CONTAINMENT PURGE	GLAND SEAL AND CONDENSER AIR REMOVAL	AUXILIARY BUILDING VENT	WGPS AND AERATED TANK VENTS	TURBINE BUILDING VENTILATION	TOTAL
Br-84	--	--	--	--	--	--
Kr-85m	1.03	$4.60x10^{1}$	$5.73x10^{-1}$	--	--	$4.76x10^{1}$
Kr-85	$4.16x10^{1}$	$2.72x10^{1}$	$3.40x10^{-1}$	$7.25x10^{2}$	--	$7.94x10^{2}$
Kr-87	$2.82x10^{-1}$	$2.50x10^{1}$	$3.12x10^{-1}$	--	--	$2.56x10^{1}$
Kr-88	1.37	$8.02x10^{1}$	1.00	--	--	$8.26x10^{1}$
Rb-88	--	--	--	--	--	--
Rb-89	--	--	--	--	--	--
Sr-89	--	--	--	--	--	--
Sr-90	--	--	--	--	--	--
Y-90	--	--	--	--	--	--
Sr-91	--	--	--	--	--	--
Y-91	--	--	--	--	--	--
Mo-99	--	--	--	--	--	--
Ru-103	--	--	--	--	--	--
Ru-106	--	--	--	--	--	--
Te-129	--	--	--	--	--	--
I-129	$1.42x10^{-10}$	$1.31x10^{-10}$	$2.78x10^{-10}$	$7.28x10^{-12}$	$4.44x10^{-10}$	$1.00x10^{-9}$
I-131	$7.42x10^{-3}$	$1.73x10^{-3}$	$1.53x10^{-2}$	$4.01x10^{-4}$	$5.72x10^{-3}$	$3.05x10^{-2}$
Xe-131m	$3.30x10^{1}$	$4.56x10^{1}$	$5.69x10^{-1}$	$3.82x10^{1}$	--	$1.17x10^{2}$
Te-132	--	--	--	--	--	--
I-132	$1.87x10^{-3}$	$7.30x10^{-6}$	$4.20x10^{-3}$	$1.10x10^{-4}$	$2.42x10^{-5}$	$6.21x10^{-3}$
I-133	$9.50x10^{-3}$	$3.37x10^{-4}$	$2.16x10^{-2}$	$5.72x10^{-4}$	$1.14x10^{-3}$	$3.21x10^{-2}$
Xe-133	$2.16x10^{3}$	$5.58x10^{3}$	$6.96x10^{1}$	$5.83x10^{1}$	--	$7.87x10^{3}$
Te-134	--	--	--	--	--	--
I-134	$9.22x10^{-4}$	$1.66x10^{-6}$	$2.38x10^{-3}$	$6.26x10^{-5}$	$5.38x10^{-6}$	$3.38x10^{-3}$
Cs-134	--	--	--	--	--	--
I-135	$5.92x10^{-3}$	$5.24x10^{-5}$	$1.04x10^{-2}$	$2.73x10^{-4}$	$1.76x10^{-4}$	$1.68x10^{-2}$
Xe-135	9.44	$2.32x10^{2}$	2.09	--	--	$2.44x10^{2}$
Cs-136	--	--	--	--	--	--
Cs-137	--	--	--	--	--	--
Xe-138	--	$1.11x10^{1}$	$1.34x10^{-1}$	--	--	$1.12x10^{1}$
Cs-138	--	--	--	--	--	--

TABLE 11-8

PREOPERATIONAL DESIGN ESTIMATES OF ANNUAL DISCHARGES

(Operation With 0.1% Failed Fuel)

GASEOUS RELEASES (Ci/yr)

ISOTOPE	CONTAINMENT PURGE	GLAND SEAL AND CONDENSER AIR REMOVAL	AUXILIARY BUILDING VENT	WGPS AND AERATED TANK VENTS	TURBINE BUILDING VENTILATION	TOTAL
Ba-140	--	--	--	--	--	--
La-140	--	--	--	--	--	--
Ce-144	--	--	--	--	--	--
Pr-143	--	--	--	--	--	--
Cr-51	--	--	--	--	--	--
Mn-54	--	--	--	--	--	--
Mn-56	--	--	--	--	--	--
Fe-59	--	--	--	--	--	--
Co-58	--	--	--	--	--	--
Co-60	--	--	--	--	--	--
Zr-95	--	--	--	--	--	--
H-3	--	--	--	--	--	--
Particulates	1.21×10^{-6}	1.57×10^{-4}	2.44×10^{-6}	1.11×10^{-6}	5.27×10^{-4}	7.99×10^{-4}
		Total noble gases airborne				9.19×10^{3}
		Total iodine-131 airborne				3.05×10^{-2}
		Total short-lived iodine airborne				5.85×10^{-2}

NOTE: Although all tritium released from the plant has been assumed to be released via the liquid discharges for the purpose of these calculations, a small percentage of the total may be released via the gaseous pathways (primarily the containment purge).

All noble gases that leak to the steam generator are assumed to be released via the condenser air removal system.

TABLE 11-9

PREOPERATIONAL DESIGN ESTIMATES OF TOTAL PLANT GASEOUS EFFLUENTS

(Operation With 1% Failed Fuel)

ANNUAL DISCHARGE (Ci/yr)

ISOTOPE	CONTAINMENT PURGE	TURBINE BUILDING VENTILATION	GLAND SEAL AND CONDENSER AIR REMOVAL	AUXILIARY BUILDING VENTILATION	WGPS AND AERATED TANK VENTS	TOTAL
Kr-85m	$1.03x10^{1}$	--	$4.60x10^{2}$	5.73	--	$4.76x10^{2}$
Kr-85	$4.16x10^{2}$	--	$2.72x10^{2}$	3.40	$7.25x10^{3}$	$7.94x10^{3}$
Kr-87	2.82	--	$2.50x10^{2}$	3.12	--	$2.56x10^{2}$
Kr-88	$1.37x10^{1}$	--	$8.02x10^{2}$	$1.00x10^{1}$	--	$8.26x10^{2}$
Xe-131m	$3.30x10^{2}$	--	$4.56x10^{2}$	5.69	$3.82x10^{2}$	$1.17x10^{3}$
Xe-133	$2.16x10^{4}$	--	$5.58x10^{4}$	$6.96x10^{2}$	$5.83x10^{2}$	$7.87x10^{4}$
Xe-135	$9.44x10^{1}$	--	$2.32x10^{3}$	$2.09x10^{1}$	--	$2.44x10^{3}$
Xe-138	--	--	$1.11x10^{2}$	1.34	--	$1.12x10^{2}$
Particulates	$1.12x10^{-5}$	$5.00x10^{-3}$	$1.49x10^{-3}$	$2.26x10^{-5}$	$6.35x10^{-4}$	$7.16x10^{-3}$
I-129	$1.42x10^{-9}$	$4.44x10^{-9}$	$1.31x10^{-9}$	$2.78x10^{-9}$	$7.28x10^{-11}$	$1.00x10^{-8}$
I-131	$7.42x10^{-2}$	$5.72x10^{-2}$	$1.73x10^{-2}$	$1.53x10^{-1}$	$4.01x10^{-3}$	$3.05x10^{-1}$
I-132	$1.87x10^{-2}$	$2.42x10^{-4}$	$7.30x10^{-5}$	$4.20x10^{-2}$	$1.10x10^{-3}$	$6.21x10^{-2}$
I-133	$9.50x10^{-2}$	$1.14x10^{-2}$	$3.37x10^{-3}$	$2.16x10^{-1}$	$5.72x10^{-3}$	$3.21x10^{-1}$
I-134	$9.22x10^{-3}$	$5.38x10^{-5}$	$1.66x10^{-5}$	$2.38x10^{-2}$	$6.26x10^{4}$	$3.38x10^{-2}$
I-135	$5.92x10^{-2}$	$1.76x10^{-3}$	$5.24x10^{-4}$	$1.04x10^{-1}$	$2.73x10^{-3}$	$1.68x10^{-1}$

NOTE: All noble gases that leak to the steam generator are released via the condenser air removal system.

TABLE 11-10

PROCESS AND EFFLUENT RADIATION MONITORS

RADIATION MONITORING SYSTEM	DETECTION EQUIPMENT	SENSITIVITY μCi/cc	RANGE
Steam Generator Blowdown Monitor (R-4014)	Off-Line Scintillation	4.52×10^{-8} Cs-137	10 to 10^{6} CPM
Liquid Waste Discharge Monitor (R-2201)	Off-Line Scintillation	1.5×10^{-5} Cs-137	10 to 10^{6} CPM
Main Vent Gaseous Monitor (R-5415)	Off-Line Geiger Mueller (GM) Tube	5×10^{-6} Xe-133	5×10^{-6} to 1 μCi/cc 30 to 10^{6} CPM
Condenser Vacuum Pump Suction (R-1752)	In-Line Scintillation	9.3×10^{-7} Xe-133	1.4×10^{-6} to 0.3 μCi/cc 90 to 10^{7} CPM
Control Room Vent Supply (R-5350)	Off-Line GM Tube	3×10^{-6} Xe-133	3×10^{-6} to 10^{0} μCi/cc 10 to 10^{6} CPM
Fuel Handling Vent Exhaust (R-5420)	Off-Line GM Tube	3×10^{-6} Xe-133	3×10^{-6} to 1 μCi/cc 10 to 10^{6} CPM
Access Control Vent Exhaust (R-5424)	Off-Line GM Tube	3×10^{-6} Xe-133	3×10^{-6} to 1 μCi/cc 10 to 10^{6} CPM
Waste Processing Vent Exhaust (R-5410)	Off-Line GM Tube	3×10^{-6} Xe-133	3×10^{-6} to 1 μCi/cc 10 to 10^{6} CPM
Emergency Core Cooling System (ECCS) Pump Room Vent Exhaust (R-5406)	Off-Line GM Tube	3×10^{-6} Xe-133	3×10^{-6} to 1 μCi/cc 10 to 10^{6} CPM
Gaseous Waste Discharge (R-2191)	In-Line GM Tube	2.5×10^{-7} Kr-85	10^{-2} to 10^{2} μCi/cc 10^{1} to 10^{6} CPM
Containment[(a)] Atmosphere Particulate (R-5280)	Off-Line Scintillation	1.26×10^{-11} Cs-137	2.43×10^{-11} to 2.43×10^{-5} μCi/cc
Containment[(a)] Atmosphere Gaseous (R-5281)	Off-Line Scintillation	2.55×10^{-7} Xe-133	3.34×10^{-7} to 3.34×10^{-1} μCi/cc
Component Cooling (R-3819)	Off-Line Scintillation	4.5×10^{-8} Cs-137	10 to 10^{6} CPM
SRW (R-1595)	Off-Line Scintillation	4.52×10^{-8} Cs-137	10 to 10^{6} CPM
Steam Generator Blowdown Recovery System Discharge (R-4095)	Off-Line Scintillation	1.5×10^{-8} Cs-137	10 to 10^{6} CPM
Main Steam Header (R-5421, 5422)	Adjacent-to-Line Gieger Mueller (GM) Tube	10^{-2}	10^{-2} to 10^{+5} μCi/cc
Main Steam N-16 Header (R-5421A, 5422A)	Adjacent-to-Line Scintillation	10^{-7}	10^{-7} to 10^{-1} μCi/cc
Wide Range Effluent (R-5416, 5417, 5418)	Off-Line Scintillation (R-5416) Solid State (R-5417, 5418)	10^{-7}	10^{-7} to 10^{5} C/ml

TABLE 11-10

PROCESS AND EFFLUENT RADIATION MONITORS

[a] Operability of this detector ensures that RCS boundary leakage detection can be monitored during normal operation. The setpoint is reduced to a point as close as practical to the background level without incurring spurious alarms in order to enhance leak detection capabilities. The detectors are described in Sections 11.2.3.2.7 and 4.3.

11.2 RADIATION PROTECTION AND MONITORING

11.2.1 DESIGN BASIS

Radiation protection design features, including radiation shielding, are provided to facilitate compliance with the personnel dose limits specified in 10 CFR Part 20 for normal operation. Additional shielding and other design features are provided to facilitate compliance with General Design Criterion 19, 10 CFR Part 50, Appendix A, as amplified in NUREG-0737, Item II.B.2. Specifically, the design dose for personnel in a vital area should not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of design basis accidents. In this case, vital areas are those areas of the plant requiring access or occupancy by an operator to aid in the mitigation or the recovery from an accident. These areas are not necessarily the same as the vital areas defined in 10 CFR 73.2 for security purposes, and would normally include the Control Room, the Technical Support Center, the post-accident sampling system, the sample analysis room, etc.

11.2.2 SHIELDING DESIGN AND EVALUATION

11.2.2.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel; the radiation consists of neutrons diffusing from the core, prompt fission gammas, fission product gammas, and gammas resulting from the slowing down and capture of neutrons.

The primary shielding is designed to:

a. Attenuate the neutron flux to prevent excessive activation of unit components and structures.
b. Reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between the primary and secondary shields.
c. Reduce residual radiation from the core to a level which does not limit access to the region between the primary and secondary shields at a reasonable time after shutdown.

The primary shield consists of 7'0" of reinforced concrete that surrounds the reactor vessel. The cavity between the primary shield and the reactor vessel is air cooled to prevent overheating and dehydration of the concrete primary shield wall. The primary shield arrangement and shield thickness are shown on Figures 11-3A through C.

In addition to the above concrete shield, an additional permanent neutron shield has been designed and installed over the Reactor Vessel cavity annulus to attenuate neutrons. The neutron shield consists of a 13-1/2"-thick ring of borated concrete poured into a carbon steel support assembly. The reactor vessel cavity neutron shield and the permanent cavity seal ring are shown in Figures 11-5 and 11-6.

The neutron shield is mounted on existing support beams and has been analyzed to meet Seismic Class II/I Category, although the neutron shield, itself, is classified as non-safety-related. The neutron shield is underneath the permanent cavity seal ring which forms a water-tight barrier to retain refueling water above the reactor vessel during refueling operations. Both the seal ring and the neutron shield have removable hatches located at various points that allow access to the annulus area below the ring and shield. These hatches allow maintenance on the nuclear instrumentation detectors and inspection of the cavity, as needed.

11.2.2.2 Secondary Shielding

Secondary shielding is provided to reduce the radiation from the RCS to levels which allow limited access to the containment during operation and to supplement primary shielding. Nitrogen-16 is the major source of radioactivity in the reactor coolant during operation, and establishes the thickness of the secondary shield. The secondary shielding consists of a minimum of 2'6" of reinforced concrete surrounding the reactor coolant piping, pumps, steam generators, and pressurizer. Secondary shielding is shown on Figures 11-3A through C. The original design basis N-16 activity concentration at the reactor vessel outlet nozzle was 3.63×10^6 dis/sec-cc.

11.2.2.3 Containment Shielding

The containment provides the shielding necessary to minimize the radiation level at the outside surface during full power operation (2737 MWt), and to ensure that radiation levels at the site boundary are below the recommended guideline values of 10 CFR 50.67 in the event of a maximum hypothetical accident (MHA). It consists of a reinforced, prestressed concrete and steel structure with 3'9"-thick cylindrical walls and a 3'3"-thick dome (Figure 5-1).

11.2.2.4 Fuel Handling Shielding

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the SFP. It is designed to protect personnel against the radiation emitted from the spent fuel and control rod assemblies.

The refueling cavity above the reactor vessel is flooded to Elevation 67'0" to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height is thus approximately 23' above the reactor vessel flange. This height assures adequate water for shielding a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate from the spent fuel assembly is less than 7.0 mrem/hr at the water surface.

The permanent cavity seal ring installed around the reactor vessel cavity consists of a stainless steel ring that will hold refueling water above the reactor vessel. The seal ring has two L-shaped, stainless steel flexure seals that include an inner seal that absorbs radial expansion of the reactor vessel, and an outer seal that absorbs vertical movements of the vessel. Both seals, working together, accommodate the seismic lateral motions of the reactor vessel.

The seal ring is designed such that the flexible seal membranes which provide the sealing function are structurally supported by other components of the ring. That is, the top plate, the inner support ring, and the outer leveling screws of the seal ring provide the structural support for the weight and pressure loads.

The seal ring includes hatches with captured bolts; each hatch is fitted with a lifting ring. The hatches facilitate access to excore nuclear instrumentation, provide openings for air and steam flow during plant operation, and provide a path to the containment sump for containment spray. The small circular hatches which provide access to the neutron detector well latches are not required to be opened during plant operation. Prior to refueling, the hatches are closed to provide the sealing function of the refueling water. Each hatch incorporates two captured O-rings to provide seals for the refueling water. The inner O-ring provides a

backup seal for each hatch and creates a sealed chamber to accommodate leak testing.

The seal ring provides a convenient working surface during refueling, allows visual inspection of all seal welds, and does not interfere with refueling operations.

All exposed metal parts of the seal ring are fabricated from Type 304 stainless steel. The hatch bolts are stainless steel. The O-ring seals used on the hatches are ethylene propylene elastomer.

Fuel is removed from the reactor vessel and moved to the SFP by the fuel transfer mechanism, via the fuel transfer tube. Concrete shielding is provided around the reactor internals storage pool and the steam generator for personnel protection during refueling. The SFP in the Auxiliary Building is permanently flooded to provide adequate water for shielding above a fuel assembly when being withdrawn from the fuel transfer tube and raised by the spent fuel handling machine prior to insertion in the spent fuel storage rack. The sides of the SFP are 6'0"-thick concrete to minimize the dose rate on the outer surface.

The areas adjacent to the SFP where personnel will be working are low radiation areas. In order to ensure that low dose rates are maintained, a filter and demineralizer processes part of the flow passing through the SFP cooling system to remove radioactive corrosion and fission products from the SFP water. The SFP filter and demineralizer are located within shielded rooms which protect operating personnel from excessive radiation.

11.2.2.5 Auxiliary Building Shielding

The function of the Auxiliary Building shielding is to protect personnel working near various system components, such as those in the CVCS, the Waste Processing Systems, and the Sampling System. Controlled access to the Auxiliary Building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down or decontaminate the entire system.

All ion exchangers and contaminated filters are located at the 27' level of the Auxiliary Building. Each ion exchanger or filter is enclosed in a separate, shielded compartment. The concrete thicknesses provided around the shielded compartments are sufficient to reduce the dose rate in normally accessible areas to less than 2.5 mrem/hr.

For the unlikely event of a massive fuel failure, additional concrete block and lead brick shield walls were constructed in the Auxiliary Building. These shield walls were installed in response to NUREG-0737, Item II.B.2 and are designed to allow for continuous occupancy of the Control Room during the entire course of a LOCA using Regulatory Guide 1.4 source term assumptions. This shielding also allows for access to the non-affected unit during a LOCA. Placement of these shield walls, as well as expected dose rates for this situation, are depicted in Figures 11-7 through 11-11.

11.2.3 RADIATION MONITORING

11.2.3.1 General

The radiation monitoring system shown on Figure 11-4 consists of monitors, instrumentation and alarms that serve to warn plant personnel of increasing

radiation/ radioactivity levels in various plant areas and effluents and provide early warning of a plant malfunction which may result in a health hazard.

The electronic circuitry is solid state except for photomultiplier-tubes and GM tubes. Circuits and their components are designed to operate with 0 to 100% relative humidity, where required, or with a humidity range appropriate for the environment where they are installed and will withstand temperatures from -20°F to 120°F outside and 50°F to 104°F inside. In addition, those components required to operate in a LOCA environment are designed for 100% relative humidity at 276°F. These monitors have also been evaluated for operation at the revised maximum vapor temperature in Section 14.20.

Detector ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR Part 20 and the access control zoning. The area radiation monitoring system instruments and detectors have been chosen based on their proven reliability in other plants, and spare items and portable units are provided to permit operation during periods of prolonged maintenance.

Process/effluent and area monitoring systems have remote visual and audible alarms and visual meter indication at the detector location, plus control room alarm and meter indication. All process/effluent monitoring systems display trend recordings, and several provide signals to the plant computers.

Most radiation monitors are provided with solenoid-operated check sources which, when actuated from the Control Room, will cause a response on each instrument channel. Instrument channels are also provided with electronic test features which, together with the solenoid-operated check sources, provide the operator with a convenient means to routinely check instrument function and response. For those channels not equipped with a check source mechanism, the detector's crystal is doped with an isotope to act as a "keep alive" source. If the indicated count rate decreases below a pre-determined value, the channel's operate alarm actuates.

Calibration of all area and process radiation monitors will be accomplished at least once each refueling cycle. The calibration procedure will entail exposing the detector of each monitoring system to a known quantity of radiation in a constant and reproducible geometry. By using portable calibration equipment, it will be possible to calibrate the entire instrument channel from the detector to the Control Room recorder. Radioactive sources used in the portable calibration equipment are selected so that calibration points will be determined for at least two decades of instrument response.

The laboratory instrumentation at Calvert Cliffs is capable of detecting radioactivity levels normally encountered in environmental sampling programs. The equipment is, therefore, readily capable of detecting and measuring in-plant levels of radioactivity prior to dilution in the environment. Equipment is capable of monitoring specific radionuclides within the concentration ranges of the design objectives and expected levels. The ODCM includes requirements for radioactive effluent sampling and analysis, including detection sensitivities.

The radioactive effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous and liquid effluents during actual or potential effluent releases. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM. The alarm/trip for radioactive gaseous effluent monitoring instrumentation will occur prior to exceeding the ODCM limits, based on

average annual χ/Q, and the alarm/trip for radioactive liquid effluent monitoring instrumentation will occur prior to exceeding 10 CFR Part 20 limits. The functionality and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

11.2.3.2 Digital Processing Radiation Monitors

The Unit 1 and 2 main steam line radiation monitoring system consists of a Geiger-Mueller tube detector, a local digital microprocessor, and Control Room indicator/controller. The equipment is qualified for the normal environment of the equipment locations. The configuration of the equipment provides sensitivity improvements and is capable of monitoring 10^{-2} μCi/cc of activity in the main steam lines.

The local microprocessors have local alarm indication on their interfaces. Any alarm condition at the local microprocessor is communicated to the Control Room for appropriate operator notification. The Control Room Microprocessors have alarm indication on their interfaces and are connected to the Control Room annunciator windows to indicate alarm conditions. The setpoints have been established to comply with Technical Requirements Manual leakage limits. In the event of experiencing an alarm, the initiating equipment must be manually reset when alarm conditions are corrected.

The detectors are equipped with check sources, which when actuated, will cause activity indication at the local Microprocessor and Control Room indicator/controller. Additional electronic operation checks can be performed by implementing internal test programs. These capabilities provide the operator with a convenient means to verify equipment operation.

All equipment will be checked for calibration at least once each refueling cycle. The calibration procedure will entail exposing the detector of each monitoring system to known quantity of radiation and simulated reactor power level to assure correct and accurate operation is maintained.

11.2.3.3 Process and Effluent Radiation Monitoring

Process radiation monitors are provided in the following areas to monitor radioactivity in systems and to ensure that plant effluents are released in accordance with ODCM limitations. Table 11-10 lists the locations, ranges, and sensitivities of these monitors.

11.2.3.3.1 Plant Vent Monitors and Samplers

The plant vent effluent monitoring and sampling system consists of a gas monitor (1/2-RE-5415) and a pair of particulate, iodine, and tritium samplers on skids (1/2-RE-5320A, and 1/2-RE-5320B).

The plant vent effluent is continuously sampled through an isokinetic nozzle. The motive force for this sampling is the sample pump associated with the gas monitor and the sample pump for the fixed operating sampler skid. The sample flow is divided into two flow paths. A portion of the sample flows through the noble gas sample chamber, where its activity is monitored by a GM tube. The second sample flow path is the fixed particulate, iodine, and tritium filters located on the redundant sampler skids. This sample is pulled through the operating sampler skid by a sample pump with a flow rate of approximately

1 SCFM. Each of these samples is exhausted back to the main vent exhaust plenum.

The filters used for the iodine and particulate sampling are removable cartridges that are replaced weekly. Tritium samples are collected at least once per month. To collect the samples, it is necessary to have a person enter the Main Vent Exhaust Equipment Room and remove the cartridges for laboratory analysis. The noble gas monitor is audibly and visually alarmed in the Control Room.

11.2.3.3.2 Waste Gas Discharge Monitor

This detector monitors gases discharged from the waste gas decay tanks after they have passed through an absolute filter prior to their release from the plant vent. A high activity alarm on the monitor automatically shuts redundant isolation valves to prevent release of radioactive gas in excess of 10 CFR Part 20 concentration limits. Although the monitor has an automatic trip function, it consists of only a single detector. Requirements to ensure the functional capability of the waste gas discharge monitor are provided in the ODCM. The intermittent mode of waste release provides ample opportunity between waste releases for necessary maintenance and operational checks to assure proper operation of the monitor. In addition, grab samples are analyzed before each release to ensure that the monitor is functioning properly.

11.2.3.3.3 Liquid Waste Processing Discharge Monitor

A monitor is installed in the common discharge header from the RCWPS and MWPS. The monitor provides an alarm signal on high activity to shut redundant, downstream isolation valves. Closure of these isolation valves prevents release of radioactive effluents which exceed 10 CFR Part 20 concentration limits for unrestricted areas. The monitor utilizes only one detector as does the waste gas discharge monitor.

11.2.3.3.4 Condenser Air Removal Discharge Monitor

Noncondensable gases are continuously removed from the condensers by the Condenser Air Removal and Priming System. Vacuum pumps within this system take suction on the condenser and discharge to the plant vent. In the event of a reactor-coolant-to-secondary leak through the steam generator tubes, the gas removed from the condenser could be radioactive. Therefore, a monitor is installed in each vacuum pump suction line upstream of the plant vent, to warn plant operations personnel of such an occurrence.

11.2.3.3.5 Steam Generator Blowdown Tank Discharge Monitor

A single detector monitor is installed in the discharge from the blowdown tank to detect possible leakage of reactor coolant through steam generator tubes. A high activity alarm on this monitor automatically shuts the blowdown valves from the steam generator. In addition, depending on the valve line-up in the steam generator blowdown recovery system, either this monitor or the recovery system discharge radiation monitor will shut the recovery system discharge lines to the circulating water channel and the condenser, and open the valve to the MWPS.

11.2.3.3.6 Steam Generator Blowdown Recovery Monitor

This monitor measures radioactivity in the combined discharge from the steam generator blowdown and recovery system ion exchangers. The alarm sounds locally and in the Control Room to warn operators of high radioactivity levels in the discharge. At the same time, the trip signal closes discharge valves to the condenser and circulating water system and diverts the flow to the MWPS. This trip prevents the release of radioactivity to the plant secondary system and subsequently to the environment.

11.2.3.3.7 Containment Atmosphere Monitor

The containment atmosphere monitor and associated equipment are mounted on a common base and their function is to detect RCS leakage in the containment. The monitor consists of a moving filter particulate detector, a gaseous radioactivity detector, piping, valves, and pumping system. The pump draws a representative sample from the containment cooling fans exhaust plenums or the containment purge exhaust fan inlet header. The sample is tested and exhausted to the main vent.

11.2.3.3.8 Atmosphere Monitors

These monitors detect particulate and gaseous activity within the plant atmosphere and alarm individually in the Control Room upon high activity.

a. Control Room Ventilation Supply
b. Fuel Handling Area Ventilation Exhaust
c. Access Control Ventilation Exhaust
d. Waste Processing Ventilation Exhaust
e. Emergency Core Cooling System (ECCS) Pump Room Ventilation Exhaust

11.2.3.3.9 Component Cooling System Monitor

The Component Cooling System monitor is installed in piping which extends from the discharge side to the suction side of the component cooling pumps to detect possible in-leakage of radioactive liquid through tube leaks in certain heat exchangers. Flow through the monitor is produced by the CCW monitor pump, which continuously draws a sample flow from the discharge side of the CCW pumps and discharges back to the CCW pump inlet header. The alarm is annunciated locally and in the Control Room to alert operators to the presence of radioactivity.

11.2.3.3.10 Service Water System Monitor

The SRW system monitor is installed in the SRW return headers of the SFP coolers to detect possible tube leakage in the coolers. Flow through the monitor is produced by the SRW monitor pump which continuously draws a sample flow from the SRW return header and discharges back to the same header. The alarm is annunciated both locally and in the Control Room.

11.2.3.3.11 Main Steam Header Monitor

The Unit 1 and 2 monitors measure potential fission product releases to the environment in the event of a steam generator tube failure which would allow leakage of primary coolant to the secondary steam. The monitors have digital processing circuitry for self-checking and activity alarm communication. The detectors of the monitoring system are mounted adjacent to the main steam lines upstream of the isolation valves and safety relief valves. Setpoints are established in accordance with plant procedures.

11.2.3.3.12 Wide Range Effluent Monitor

These monitors are used to measure the release of radioactivity from the unit vents in the event of an accident. The ranges of these monitors are considerably greater than those of the original plant vent monitors. However, the original plant vent monitors are still maintained for normal operation and to provide redundant monitoring at the low end of the concentration range. The wide-range effluent monitors have three detectors with overlapping ranges for measuring noble gases. A high flow rate sample is measured by the low-range detector. A separate low flow rate sample is filtered for particulates and iodine and is then measured by the intermediate- and high-range noble gas radioactivity detectors. The filtration serves two purposes: to keep particulate and iodine plateout from contaminating the noble gas detection chamber and to collect grab samples for laboratory analyses, especially for I-131. Lead shielding is provided for the grab sample cartridges. Readouts and alarms are provided in the Control Room.

11.2.3.4 Area Radiation Monitoring

Monitoring Program

The area radiation monitoring system reads out and records gamma radiation levels in selected areas throughout the station and alarms (audibly and visually) if these levels exceed a preset value or if the detector malfunctions. Each detector reads out and alarms in the Control Room and at its station location. They can be checked with a test source to verify proper operation.

The alarm setpoint of each area monitor is variable, and is set at a level sufficiently above the normal background radiation level in the respective area to minimize the occurrence of spurious alarms.

Table 11-11 lists the locations of the installed in-plant area radiation monitors, including the range and sensitivities of these instruments.

The containment area monitors provide a signal for securing the containment purge system, should a fuel handling incident occur during refueling. The setpoints for these monitors are far below the dose rates resulting from a postulated fuel handling incident so these monitors can also fulfill the normal functions of area monitors. Four of the area monitors located in the Containment Structure provide control signals in addition to indication and alarms. Upon alarm on high activity, their output is used in a two-out-of-four logic matrix to stop the containment purge fans and close the containment isolation valves (Section 7.3.2.2).

11.2.3.5 Radiological Environmental Monitoring

The radiological environmental monitoring program required by the ODCM provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides resulting from the plant operation that lead to the highest potential radiation exposures to members of the public. This monitoring program implements 10 CFR Part 50, Appendix I, Section IV.B.2 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. Program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection. The lower limits of detection required by the ODCM are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the lower limits of detection is defined as a limit representing the capability of a measurement system (a priori, or before the fact) and not as a limit for a particular measurement (a posteriori, or after the fact).

Land Use Census

To satisfy the requirements of 10 CFR Part 50, Appendix I, Section IV.B.3, the ODCM requires an annual land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required. The best information from the door-to-door survey, aerial survey or consulting with local agricultural authorities shall be used. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

Interlaboratory Comparison Program

The ODCM requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as a part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of 10 CFR Part 50, Appendix I, Section IV.B.2.

TABLE 11-11

AREA RADIATION MONITORS

LOCATION	RANGE	SENSITIVITY
Containment – Area Monitors (RI-5316 A,B,C, & D)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Auxiliary Building – SFP Platform (RI-7025)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Sample Rooms 413, 424 (RI-7006)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Penetration Rooms 211, 221 (RI-7011)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
SFP Heat Exchanger Room 320 (RI-7020)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Drum Storage Area Room 418 (RI-7021)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Chemistry Lab 14 (RI-7023)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Liquid Waste Evaporator Room 420 (RI-7022)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
ECCS Pump Rooms (East, West) (RI-7004, RI-7005)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Miscellaneous Waste Pump Area Room 110 (RI-7017)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Miscellaneous Waste Rec. Tank Room 113 (RI-7016)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Conc. Boric Acid Storage Tank Rooms 215, 217 (RI-7010)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Waste Gas Equipment Room 208 (RI-7018)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Decontamination Room 210 (RI-7019)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
SFP Area Room 414 (RI-7024A)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV

TABLE 11-11

AREA RADIATION MONITORS

LOCATION	RANGE	SENSITIVITY
SFP Area Room 414 High Range (RI-7024B)	1 to 10^4 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Blowdown Tank Areas (RI-7012)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
New Fuel Storage Area (RI-7026)	.1 mr/hr to 10 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Containment High Range (RI-5317A, RI-5317B)	1 to 10^8 R/hr	Energy dependence of ± 20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV
Gas Analyzer Equipment Room (RI-7027)	.1 mr/hr to 10 R/hr	Energy dependence of ±20% of the actual radiation intensity over an energy response from 80 keV to 3 MeV

11.3 RADIATION SAFETY

11.3.1 GENERAL

The Plant General Manager-Calvert Cliffs Nuclear Power Plant Department is responsible for the Radiation Safety Program at Calvert Cliffs. This responsibility is shared by all supervisors and plant personnel. Personnel assigned to the plant and all visitors are required to follow all rules and procedures established for protection against radiation, contamination and airborne activity.

Administration of the Radiation Safety Program at Calvert Cliffs is the responsibility of the General Supervisor-Radiation Protection. The implementation of this program is the responsibility of the Health Physics Operations Unit and Health Physics Support Unit, with their primary purpose being to administrate, control, and eliminate, if possible, any and all radiological hazards within the plant. Additional responsibilities are in the areas of assisting the various plant training, operations, and maintenance sections in assuring the plant is operated and maintained in a safe condition, and that compliance with Company, State, and Federal regulations in regard to radiation safety are adhered to.

11.3.2 ACCESS CONTROL

A radiologically controlled area (RCA) is defined as an area within the plant site in which radioactive materials and/or radiation are present, or where there is a potential for their release in sufficient quantities (as designated by 10 CFR Part 20) to require protective measures. Entry into radiologically controlled areas is limited to those persons authorized to accomplish a specific task.

Radiologically controlled areas are designated by appropriate signs and barriers. Prior to entering these areas, personnel will meet the requirements for dosimetry, protective clothing and procedures as detailed in Calvert Cliffs Radiation Safety Procedures.

11.3.3 PERSONNEL PROTECTIVE AND MONITORING EQUIPMENT

11.3.3.1 Protective Equipment

All personnel entering a RCA may be required to wear anti-contamination clothing. This is directly dependent on the locations, plant condition and the task to be performed.

Generally the standard dress will be coveralls (over personal undergarments or medical “scrubs”), shoe covers, cotton glove liners, rubber gloves, rubber boots, and a hood. The requirements may be increased or decreased depending upon contamination and airborne radioactivity concentrations and the task to be performed.

Respiratory protection devices may be required when high or the potential for high airborne radioactive material exists. In such situations, the air will be monitored by the Health Physics Technicians and the required protective devices will be issued as appropriate for the type and concentrations of airborne radioactive material present. Monitoring and evaluating airborne radioactive material and the use of respiratory protection is performed according to the Calvert Cliffs Radiation Safety Procedures and 10 CFR Part 20.

11.3.3.2 Personnel Monitoring Program

The personnel monitoring program at Calvert Cliffs is based on the use of Dosimeter of Legal Record (DLR), Electronic Dosimeter (ED), and direct-reading dosimeters (DRD) for determining personnel exposures due to external Beta, Gamma, and neutron sources. When neutron exposures may be expected, a

combination of DLR and/or a direct reading portable neutron survey instrument will be used for exposure determination.

All personnel working in an RCA will be issued DLRs. Dosimeters of legal record will be processed routinely in accordance with Radiation Safety Procedures. Additional processing will be in accordance with applicable Radiation Safety Procedures. Additional DLRs will be issued for critical organs and/or extremity monitoring where prescribed by applicable Radiation Safety Procedures.

Under conditions specified in 10 CFR 20.1502 and the Calvert Cliffs Radiation Safety Manual or upon entering radiologically controlled areas of the plant, all personnel are required to wear a DLR and/or ED or DRD. In the case of visitors, they shall wear an DRD and be accompanied by an escort wearing a DLR and/or ED for the group. Direct-reading dosimeters shall be read daily or upon exit from the RCA to provide estimates of personnel exposure between DLR processing periods. The DRD also provides data as needed for evaluation of a lost or damaged DLR or evaluations of equipment, jobs, techniques, shielding, etc. Electronic Dosimeters may be used in place of DRDs.

The bioassay program requires all employees designated for assignment to Calvert Cliffs to pass a physical examination prescribed by the Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) Environmental, Safety & Health Department. For individuals required to work in RCAs of the plant, this examination will include a whole body count or passive screening. Periodic physical reexaminations, including in vivo counting or passive screening, will be given to plant employees who do significant work in an RCA. Employees terminating employment with CCNPP who have worked in an RCA at Calvert Cliffs will receive an in vivo analysis or passive screening.

Special health examinations may be required for any individual whose records show they are exceeding of yearly whole body or tissue/organ dose limits, or for any individual suspected of assimilating radioactive material. This examination may include in vivo analysis and/or in vitro analysis whenever uptake of significant radioactive materials is suspected.

Records showing the occupational dose accumulated at Calvert Cliffs of all individuals provided with personnel monitoring shall be maintained in accordance with 10 CFR 20.2106. Lists of the current status of personnel dose are available to plant supervision to aid in job planning. In addition, an alert list system will be used to emphasize those individuals who are approaching the administrative annual individual ALARA dose goal.

An individual radiation history record folder and/or electronic media will be maintained for each occupational worker. The folder contains records of: external and internal occupational dose received at Calvert Cliffs; prior occupational exposure history, to the extent required by revised 10 CFR Part 20; radiation orientation and/or training received; and special measurement results (in vivo, in vitro, respirator tests).

Additional records and reports to employees, other individuals, and the US NRC, shall be in accordance with 10 CFR 20.2202 through 20.2206.

11.3.4 RADIATION SAFETY FACILITIES

11.3.4.1 Change Room and Decontamination Facilities

Change room facilities are provided where personnel change into the protective clothing required for RCA work. Showers, sinks and appropriate monitoring equipment are provided to aid in personnel decontamination.

Equipment decontamination facilities are also provided at the plant for large and small equipment and components.

11.3.4.2 Health Physics Laboratory Facilities

The radiation safety laboratory contains facilities and equipment for detecting, analyzing, and measuring all types of ionizing radiation. In addition, a small source "calibrator" is available to perform operational checks of portable gamma survey instruments. The chemistry laboratory includes a counting room for analyzing environmental survey samples, including identification of specific radionuclides.

11.3.4.3 Health Physics Instrumentation (Excluding Process and Area Monitoring Systems)

Portable radiation survey instruments are provided for use to Health Physics Technicians as well as for operating and maintenance personnel. A variety of instruments are selected to cover the spectrum of radiation measurement requirements anticipated, i.e., instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. Sufficient quantities are provided to allow for routine and emergency use, and allowing for unavailability of instruments due to maintenance and calibration.

In addition to the portable instruments, appropriate monitoring instruments are located at the exits from an RCA and at various locations within an RCA. These instruments are intended to detect contamination on personnel, materials, or equipment, so as to prevent contamination from being spread within or beyond controlled areas. Portal monitors will also be utilized, as appropriate, to monitor for radioactive material at plant ingress and egress.

Details on the Health Physics instrumentation used are contained in the Calvert Cliffs Radiation Safety Procedures and Instrument Test Equipment and Calibration procedures.

11.4 RADIOACTIVE MATERIALS SAFETY

11.4.1 MATERIALS SAFETY PROGRAM

The overall radiation safety program includes provisions to assure the safe storage, handling, and use of sealed and unsealed special nuclear, source, and byproduct materials (Section 11.3).

In addition to indoctrination and training, personnel dosimetry, radiation and contamination surveys, and contamination control methods, the Radiation Safety Program includes specific provisions for radioactive material control. These provisions include procedures for the proper receipt of radioactive materials, the storage and movement of material within the plant, radioactive source control and inventory, and the packaging and labeling of radioactive materials for shipment.

11.4.2 SEALED SOURCE CONTAMINATION

Sealed sources containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material are periodically verified to be free of ≥ 0.005 microcuries of removable contamination. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation ensures that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. The test method has a detection sensitivity of at least 0.005 microcuries per test sample.

Each sealed source is tested for leakage and/or contamination by CCNPP personnel or other persons specifically authorized by the NRC or an Agreement State. Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) will be tested at the frequencies described below:

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:
    1. With a half-life greater than 30 days (excluding Hydrogen 3), and
    2. In any form other than gas.
b. Stored sources not in use - Each sealed source and fission detector is tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date are tested prior to being placed into use.
c. Startup sources and fission detectors - Each sealed startup source and fission detector is tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

Sealed sources with removable contamination in excess of 0.005 microcuries are immediately withdrawn from use and either decontaminated and repaired, or disposed of in accordance with NRC requirements.

11.4.3 FACILITIES AND EQUIPMENT

Plant laboratory facilities and equipment, survey and measuring instruments, and monitoring devices are described in Sections 11.2.3, 11.3.3, and 11.3.4.

11.4.4 PERSONNEL AND PROCEDURES

The experience and qualifications of key Radiation Safety personnel responsible for handling and monitoring radioactive materials are described in Sections 12.1 and 12.2.

11.4.5 REQUIRED MATERIALS

	MATERIAL	FORM AND USE	POSSESSION LIMIT
1.	Any byproduct, source, and special nuclear material	As reactor fuel; as sealed neutron sources for reactor start-up; as sealed sources for reactor instrument and radiation monitoring equipment calibration; and as fission detectors.	As required by Unit 1 or Unit 2 License.
2.	Any byproduct, material	Any form for sample analysis or counting equipment calibration.	As required by Unit 1 or Unit 2 License.
3.	Any source or special nuclear material	Any form for sample analysis or instrument calibration.	As required by Unit 1 or Unit 2 License.
4.	Sodium-24	Liquid form for tracer measurements for steam.	As required by Unit 1 or Unit 2 License.
5.	Byproduct and special nuclear materials	Possess but not separate such materials in such form as may be produced by operation of the facility.	As required by Unit 1 or Unit 2 License.

FIGURE 11-1 REACTOR COOLANT WASTE PROCESSING

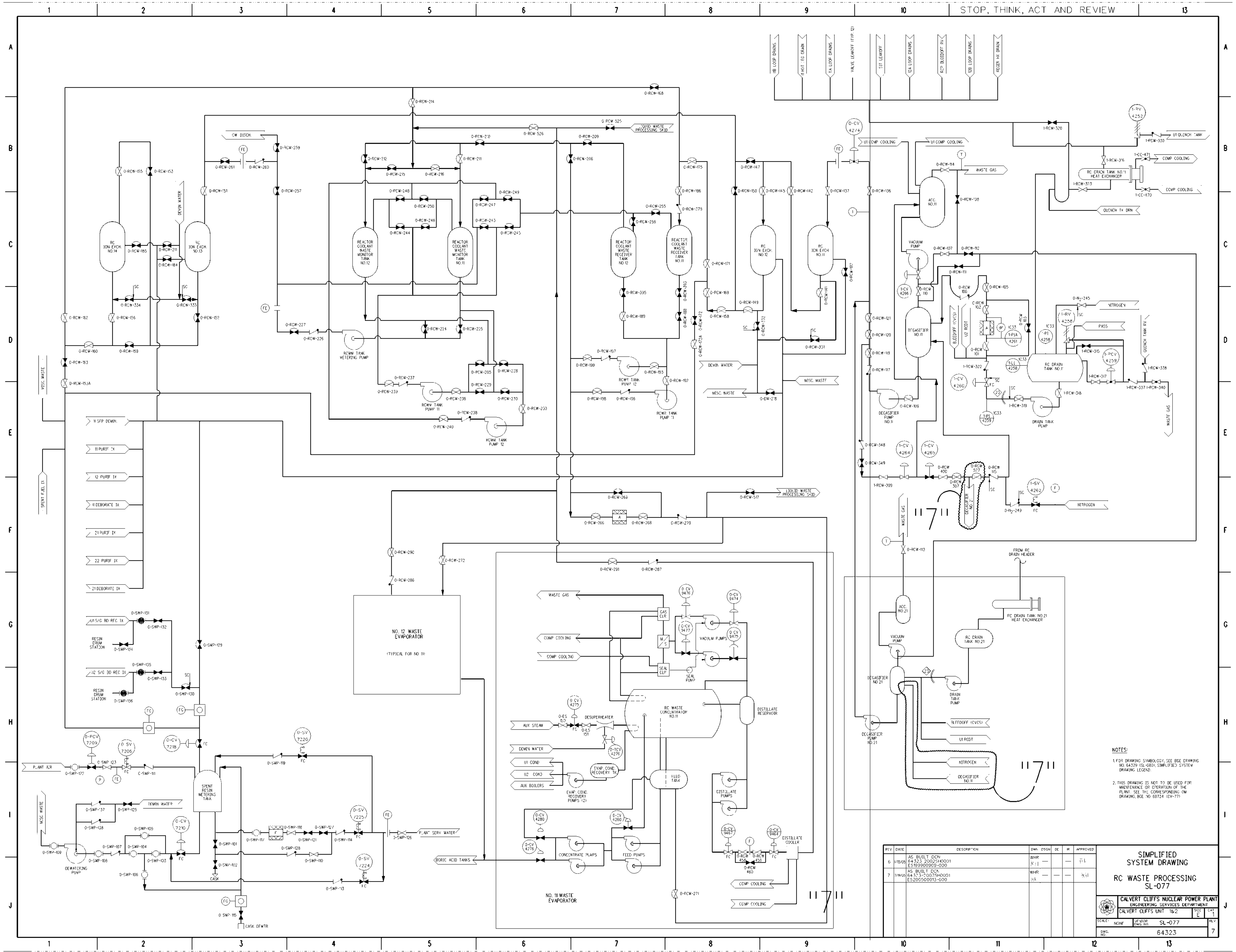


FIGURE 11-2 WASTE GAS AND MISCELLANEOUS WASTE PROCESSING

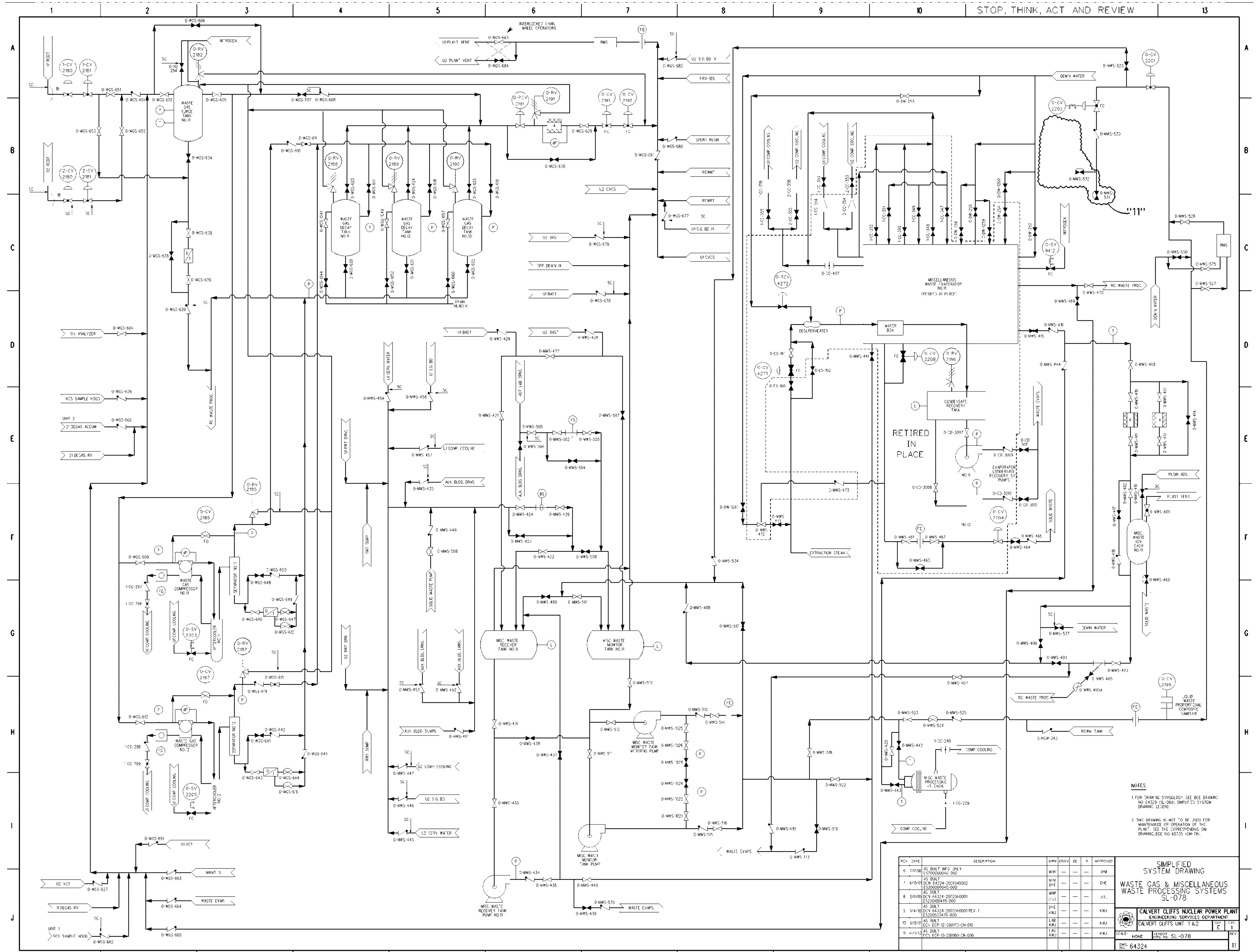


FIGURE 11-4 RADIATION MONITORING SYSTEM (Sheet 1)

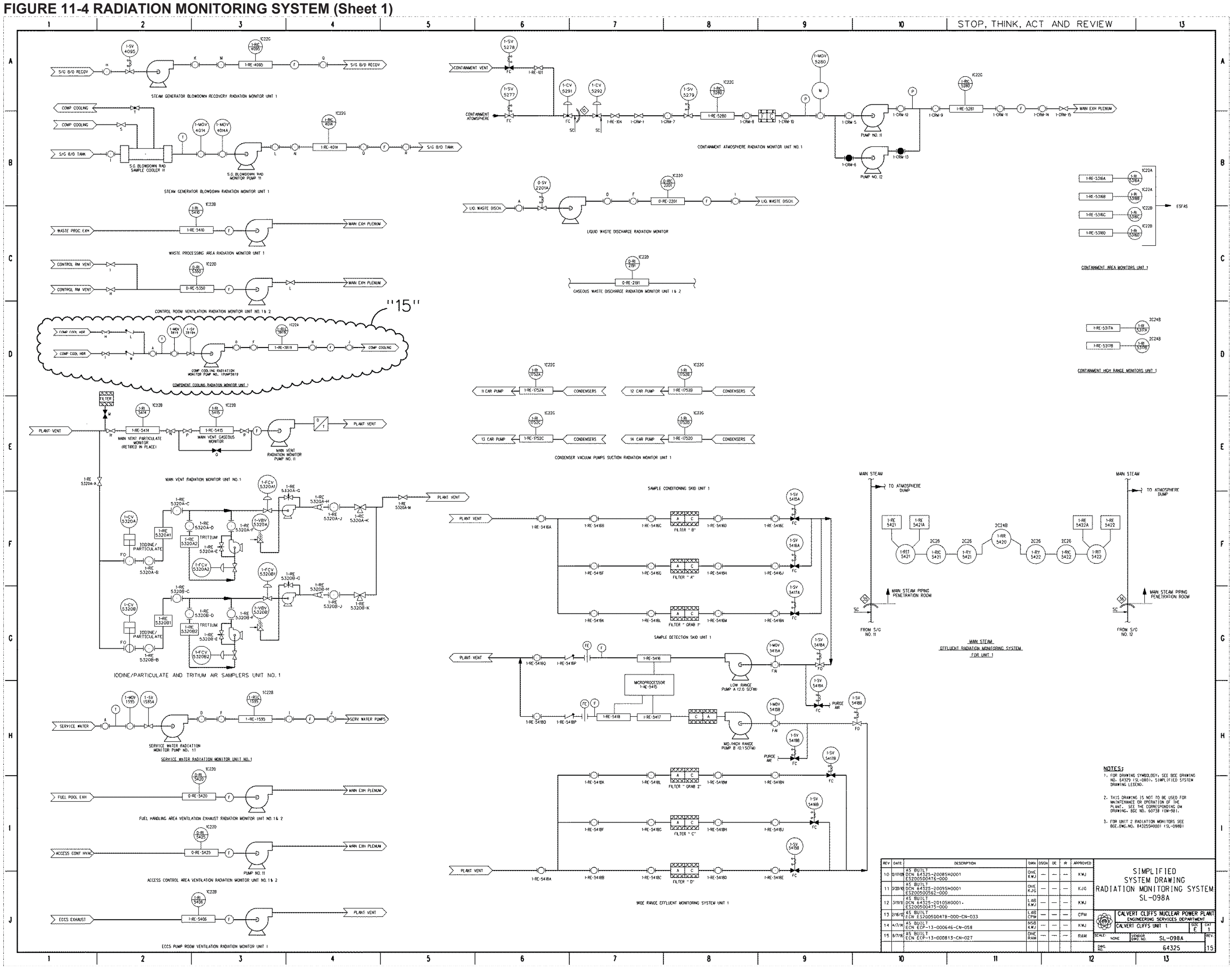
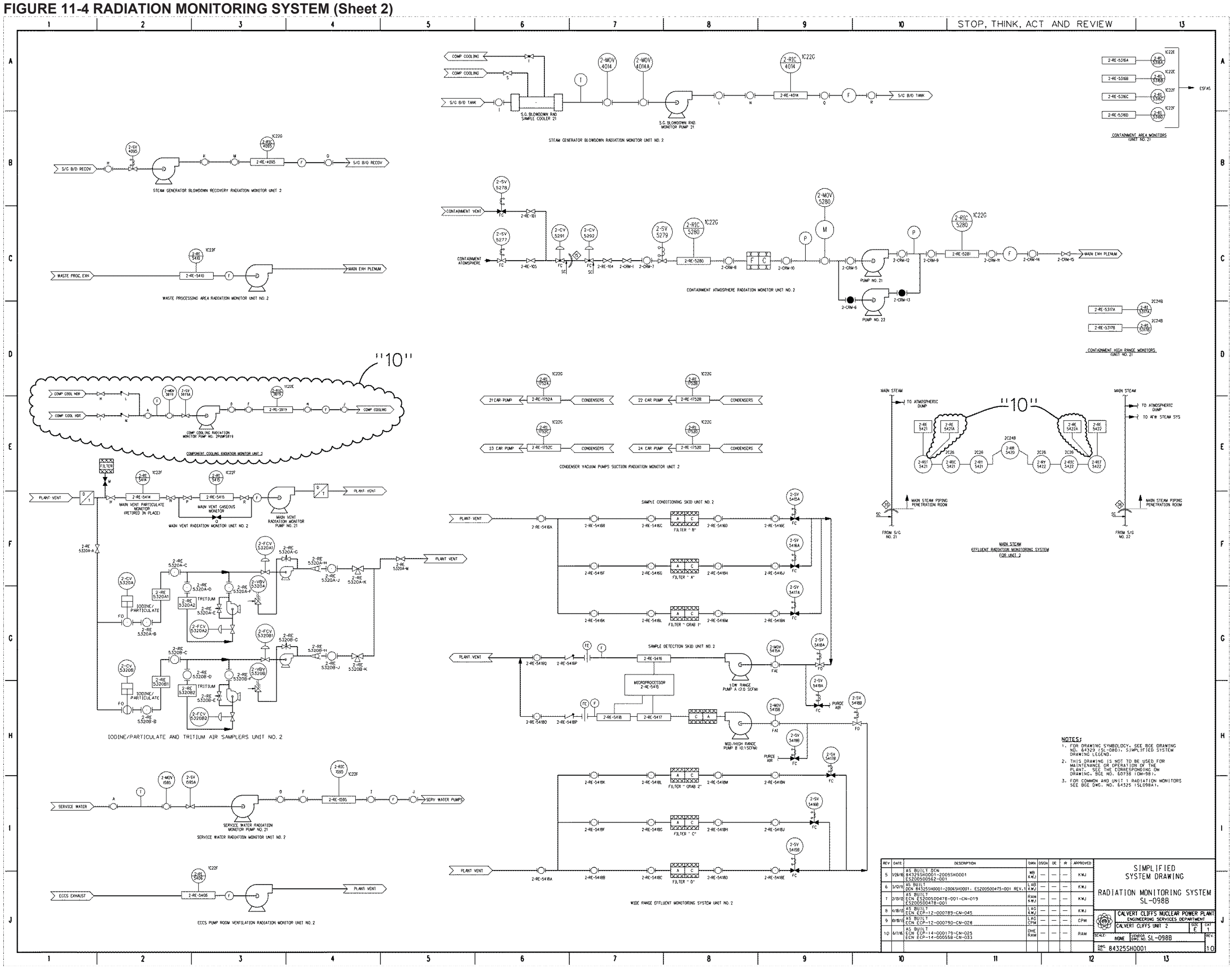
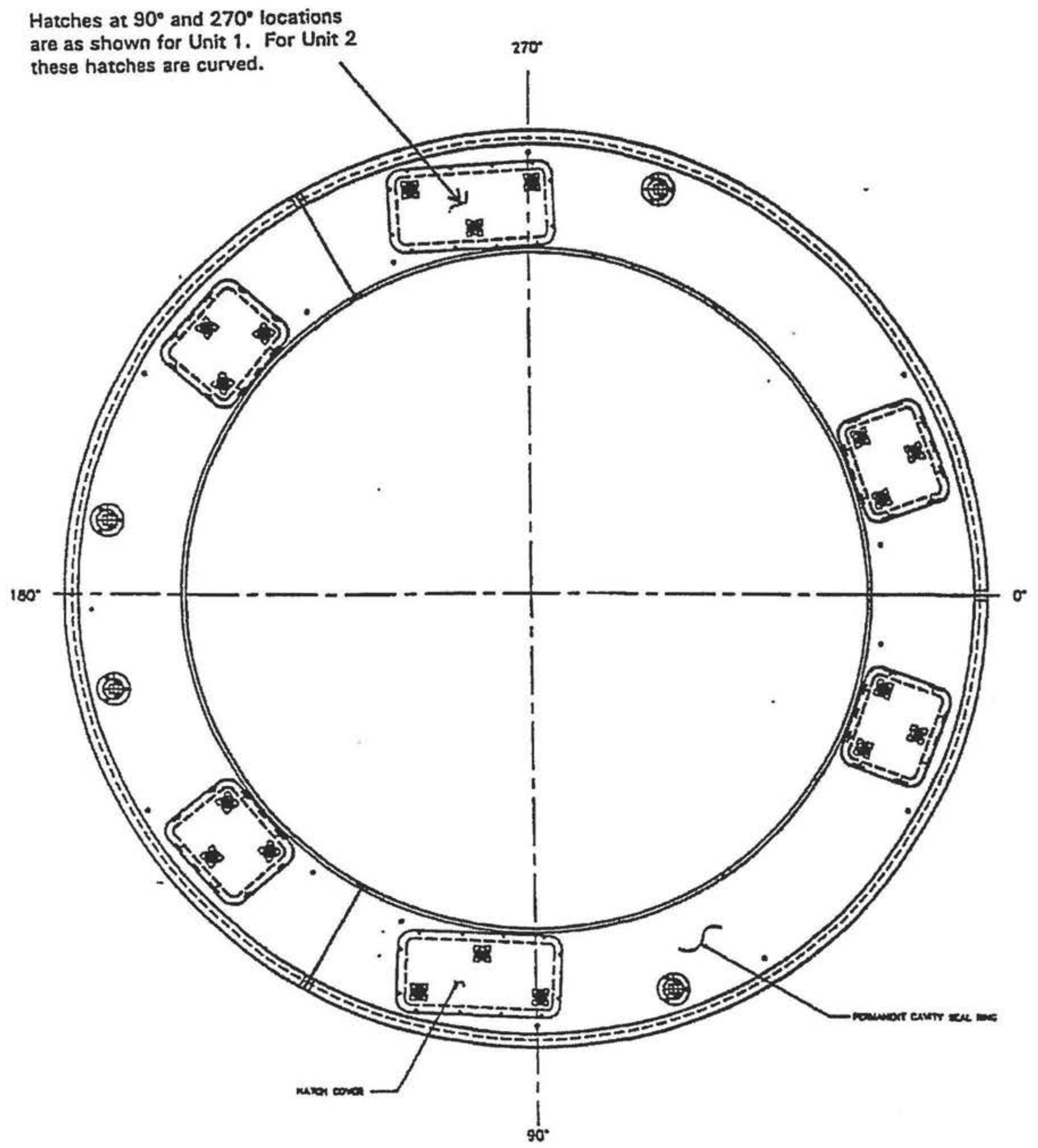
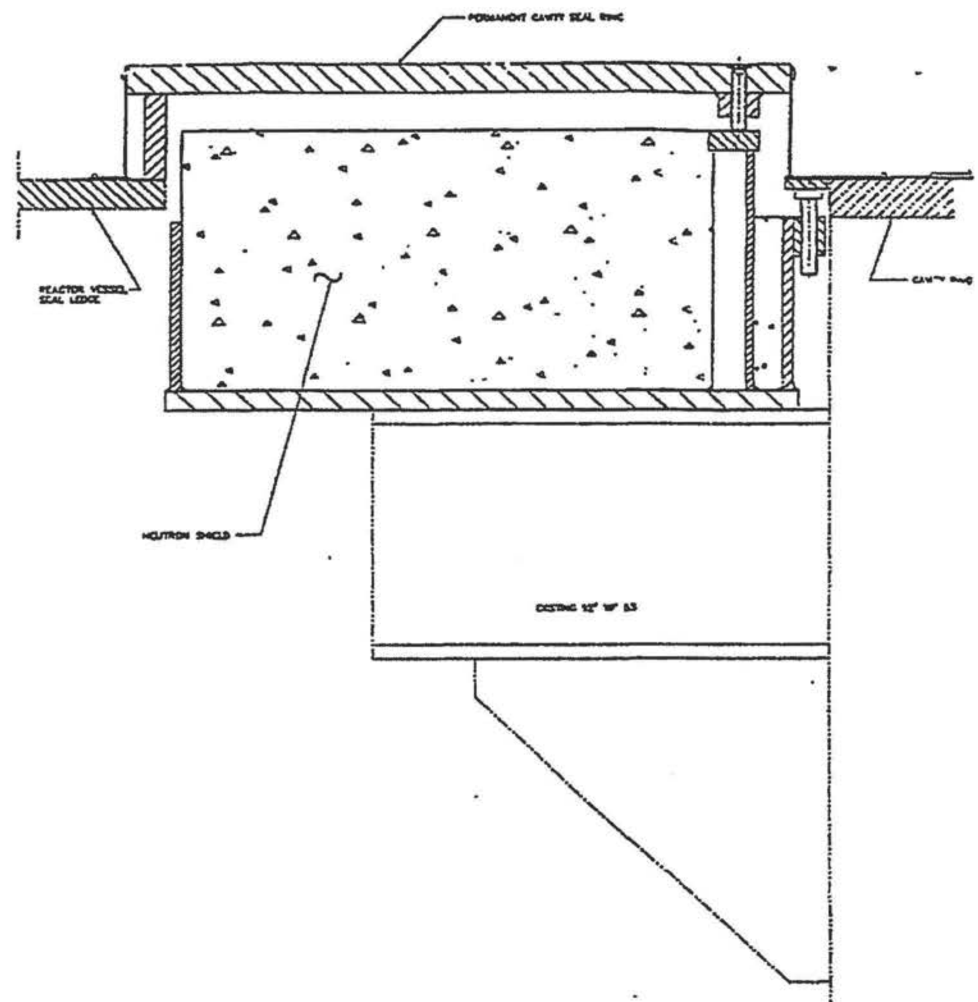


FIGURE 11-4 RADIATION MONITORING SYSTEM (Sheet 2)





BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	REACTOR CAVITY SEAL	Figure 11-5 Rev. 19



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	NEUTRON SHIELDING SECTION UNITS 1 AND 2	Figure 11-6 Rev. 18

CHAPTER 12
CONDUCT OF OPERATIONS

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CHAPTER 12
CONDUCT OF OPERATIONS

LIST OF ACRONYMS

CCNPP	Calvert Cliffs Nuclear Power Plant
ERP	Emergency Response Plan
ERPIP	Emergency Response Plan Implementation Procedures
QATR	Quality Assurance Topical Report

12.0 CONDUCT OF OPERATIONS

12.1 ORGANIZATION AND RESPONSIBILITY

The responsibilities of the Exelon Generating Company, LLC (Exelon) corporate organization and Calvert Cliffs Nuclear Power Plant (CCNPP) are described in the Quality Assurance Topical Report (QATR).

The qualifications for key positions within the plant organization have been established to ensure the high degree of knowledge and skill required to operate the plant in a safe and efficient manner. The minimum qualifications for all positions comply with Section 5.3 “Unit Staff Qualifications” of the Calvert Cliffs Unit 1 and Unit 2 Technical Specifications.
The Corporate and site organization is summarized in the QATR.

The positions described in the Technical Specifications are identified by the plant specific titles listed below:

Technical Specifications Positions	Plant Specific Titles
Plant Manager	Plant Manager
Specified Corporate Officer	Site Vice President
Operations Manager	Operations Director
Assistant Operations Manager	Shift Operations Superintendent
Shift Supervisor	Shift Manager

12.1.1 ORGANIZATIONAL STRUCTURE-HISTORICAL INFORMATION

The initial start-up of Unit 1 and Unit 2, including preoperational system tests, initial core loading, initial criticality, approach to full power commercial operation, and acceptance tests, were performed by CCNPP personnel with assistance from the Electric Production Department staff, other Baltimore Gas and Electric Company personnel, Combustion Engineering, Inc., and Bechtel Power Corporation, as needed.

Prior to fuel loading for Unit 1, "cold" (precritical) Atomic Energy Commission senior operator licenses were obtained by Nuclear Plant Engineer Operation’s six Shift Supervisors, five Senior Control Room Operators, one Control Room Operator, and one Performance Engineer. During the test period following initial criticality, approximately 15 Control Room Operators received the necessary operating experience to qualify for "hot" Atomic Energy Commission licenses.

The personnel assigned to the plant and site organization were essentially the same for both start-up and normal operation, though additional personnel were temporarily assigned to the plant for assistance during start-up and for training purposes.

12.2 PLANT STAFF TRAINING

Training programs exist for the following work groups:

a. Chemistry Personnel
b. Electrical Maintenance Personnel
c. Engineers and Technical Staff
d. Instrument Maintenance Personnel
e. Mechanical Maintenance Personnel
f. Nuclear Operations Personnel
g. Radiation Safety Personnel

These training programs are Institute of Nuclear Power Operations accredited and satisfy the standards set forth in Technical Specification 5.3. Details of each of these training programs are contained in their respective training program manual(s).

12.3 OPERATIONAL PROCEDURES AND CONTROLS

The plant is operated and maintained in accordance with approved procedures. (Technical Specifications) These procedures are reviewed and approved in accordance with administrative procedures.

Fuel Handling Procedures

These procedures prescribe the general preplanning for the fuel handling program and its associated safety measures and identify those aspects of the program for which procedures are to be prepared for each refueling outage.

Locking Devices

Locking Devices have been installed on selected valves or their controls to prevent their inadvertent operation. Administrative controls have been established to ensure that these devices remain in place as intended, and that they are re-installed after periods of approved removal.

12.4 RECORDS

Records required by NRC regulations and the Quality Assurance Topical Report are maintained.

12.5 REVIEW AND AUDIT OF OPERATIONS

The Quality Assurance Topical Report describes the administrative controls which have been established to ensure that all significant operations, maintenance, tests, emergencies, or unusual occurrences will be handled in accordance with written procedures which have been duly reviewed and approved.

12.6 EMERGENCY RESPONSE PLAN (FORMERLY SITE EMERGENCY PLAN)

12.6.1 OVERALL CONCEPT OF OPERATION

The CCNPP Emergency Response Plan (ERP) has been developed to protect the general public and site personnel from possible consequences of an emergency condition. This plan, combined with its implementation procedures and the Radiological Emergency Plans of the State and local agencies, allows for (a) early recognition and classification of a possible emergency condition; (b) prompt notification, via reliable communication channels, of agencies and personnel to augment the normal operating personnel; (c) prompt preplanned actions to be taken to protect the population-at-risk.

The CCNPP staff is trained to cope with emergencies. Written agreements with Federal agencies, private contractors, and coordinated State and local agency emergency plans (required by law) provide assistance to ensure resources can be readily available in as short a time as possible to cope with emergencies and protect the population-at-risk. The agencies and the resources they will provide are described in the ERP and the "Maryland Emergency Operations Plan, Annex Q, Radiological Emergency Plan." Both plans describe the roles of the various State and local agencies and their interfaces for carrying out protective and parallel actions in both a 10-mile-radius plume zone and a 50-mile-radius ingestion zone.

The ERP describes (1) the emergency classification system used at the plant; (2) the organizational control of emergencies including onsite, offsite, and augmentation organizations; (3) the emergency measures to be taken; and (4) available emergency facilities and equipment.

Procedures for implementation of the CCNPP ERP are contained in the Emergency Response Plan Implementation Procedures (ERPIPs). The procedures are distributed to those individuals, facilities, and organizations where immediate availability of such procedures would be required during an emergency. The ERPIPs provide the following information:

1. Means of classifying emergencies, lists of available equipment, and emergency information;
2. Directions for meeting notification requirements;
3. Directions for seeking emergency assistance; and,
4. Detailed instructions to individuals responsible for (a) assessing emergency conditions and (b) providing steps to be taken to mitigate the consequences of the accident.

The ERPIPs are used in conjunction with applicable plant operating, radiological control, and security procedures to correct the emergency condition and to mitigate the consequences of the accident.

12.6.2 ESSENTIAL ELEMENTS FOR ADVANCE PLANNING

The CCNPP Emergency Response Program, is defined by two separate but totally coordinated documents. The first document, the ERP, provides the basis for performing advance planning and for defining specific requirements and commitments to be implemented by other documents and procedures. The second document, the ERPIPs, provides the detailed information and procedures that will be required to implement the ERP in the event of an emergency at CCNPP. These two documents are briefly described below.

12.6.2.1 Emergency Response Plan

The CCNPP ERP ensures that all emergency situations, including those that involve radiation or radioactive material, are handled properly and efficiently. It covers the entire spectrum of emergencies from minor, localized emergencies to major emergencies involving action by offsite emergency response agencies and organizations. The CCNPP ERP includes a system for classifying emergencies. Thus, the CCNPP ERP provides the overall advance planning required for the development of methods of implementation that are included in the ERPIPs.

In summary, the CCNPP ERP provides the following:

1. A means for classifying emergency conditions in a manner compatible with a system utilized by Federal, State and County emergency response agencies and organizations;
2. A means of reclassifying such emergency conditions should the severity increase or decrease;
3. Identification of normal and emergency operating organizations;
4. General guidelines as well as specific details as to which County, State, and Federal authorities and agencies and other outside organizations are available for assistance;
5. Information pertaining to the Emergency Operations Facility, Technical Support Center, Operational Support Center, Joint Information Center, and equipment available both onsite and offsite;
6. Requirements for training, drills, reviews, and audits to ensure a high degree of emergency preparedness and operational readiness;
7. Figures and tables that display such information and data as organization charts, maps and population distributions;
8. Specific plans and agreements pertaining to participating offsite organization and agencies.

12.6.2.2 The Emergency Response Plan Implementation Procedures

The purpose of the ERPIPs is to provide (1) a single source of pertinent information and data and (2) the procedures that would be required by or useful to various emergency response agencies and organizations in the event of an emergency at CCNPP. The ERPIPs consolidate and integrate specific material required for personnel to implement or support the CCNPP ERP and the State Radiological ERP.

The ERPIP document is organized to provide the following:

1. Means of classifying emergencies, lists of available equipment, and emergency information.
2. Procedures that:
   a. Provide the CCNPP staff and supporting agencies with specific instructions for the implementation of the CCNPP ERP;
   b. Assign specific responsibility and authority to emergency response personnel;
   c. Provide a single source of pertinent information, forms, data and step-by-step instructions to ensure prompt actions and proper notifications and communications are carried out;
   d. Provide a record of the completed actions; and,
   e. Provide the mechanism by which emergency preparedness will be maintained at all times.

12.6.3 ORGANIZATIONAL CONTROL OF EMERGENCIES

12.6.3.1 Emergency Response Organization

The first line of control of any emergency at CCNPP lies with the normal shift personnel on duty at such time as an emergency situation should occur. Assistance is available within about one hour from other plant staff and operating personnel. Additional assistance is available from Corporate, local, State, Federal Agencies, and contractor personnel.

The Emergency Director/Recovery Manager is in charge of onsite emergency activities, described in this plan and further delineated in the ERPIPs, as the main contact at the site. He/she also directs communications to and from the site. In addition to directing the staff and operating personnel, he/she can call on additional Company and outside agency assistance as needed.

12.6.3.2 Recovery Organization

The Recovery Organization is activated at the direction of the Emergency Director/Recovery Manager.

The Recovery Organization is responsible for providing additional personnel and technical assistance from offsite sources.

12.6.3.3 Offsite Emergency Organization

Calvert Cliffs Nuclear Power Plant is equipped and staffed to cope with many types of emergency situations. However, if a fire or other type of incident occurs that requires outside assistance, such assistance is available from state agencies, local services, and contractors including:

A. State of Maryland
   1. Governor
   2. Maryland Emergency Management Agency
   3. Department of Health and Mental Hygiene
   4. Department of Agriculture
   5. Department of Natural Resources
   6. Maryland State Police
   7. Department of Human Resources
   8. Department of Transportation
   9. Department of Education
   10. Department of Housing and Community Development
   11. Maryland Military Department/National Guard
   12. Maryland Institute for Emergency Medical Service Systems
   13. Office of the Comptroller of the Treasury
   14. State Fire Marshall
   15. Maryland Department of Environment

B. Calvert Memorial Hospital

C. Local Fire and Rescue Service

D. Naval Oceanographic Command Detachment

E. Emergency Medical Assistance Program Contractor

The primary functions of the above-listed agencies and services can be found in the ERP.

12.6.4 EMERGENCY PLANNING ZONES

The ERP identifies the interface between CCNPP and the governmental agencies having action responsibilities to ensure the protection of the population-at-risk with Emergency Planning Zones of CCNPP. The Plume Exposure Pathway extends to 10 miles from CCNPP and the Ingestion Exposure Pathway extends 50 miles from the site.

CHAPTER 13
INITIAL TESTS AND OPERATION

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INITIAL TESTS AND OPERATION

LIST OF ACRONYMS

CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
RCS	Reactor Coolant System

13.0 INITIAL TESTS AND OPERATION

Sections 13.1 through 13.3 are historical information about tests that were conducted when the units were first started up. They can also be accessed through the NORMS Records System. The Document IDs (Doc ID) are “Section-13.1, Section-13.2, and Section-13.3.”

13.4 POST-REFUELING STARTUP TESTING

The following discussions represent the major startup tests conducted for Calvert Cliffs subsequent to each refueling. Sufficient data is obtained to verify that the plant operates in a safe condition within the bounds of the applicable acceptance criteria and, therefore, the Safety Analysis.

13.4.1 HOT FUNCTIONAL TESTING

13.4.1.1 CEDM Performance Testing

During this testing, the proper functioning of the control element assemblies (CEAs), control element drive mechanisms (CEDMs), and CEA position indication are verified through the insertion and withdrawal of the CEAs. Rod drop times are measured and evaluated. Proper latching of the single CEA to the respective CEA extension shaft is confirmed. Any irregularities are analyzed.

13.4.1.2 RCS Flow Verification

Reactor Coolant System (RCS) flow rates are verified based on differential pressure measurements obtained across the reactor coolant pumps and the reactor vessel. These values are compared for consistency to those obtained during previous testing.

13.4.2 INITIAL CRITICALITY

Approach to criticality commences with the withdrawal of the Shutdown CEA Groups, followed by the withdrawal, in sequence, of the Regulating CEA Groups, concluding with Group 5 at mid-core. Criticality is established through boron dilution. The plant is allowed to stabilize, then proceed to the low power physics tests to verify physics design parameters.

13.4.3 LOW POWER PHYSICS TESTING

13.4.3.1 Deleted

13.4.3.2 Critical Boron Concentration

Critical Boron Concentrations are determined for all CEAs withdrawn and with CEA Groups 1 through 5 inserted.

13.4.3.3 Isothermal Temperature Coefficient

The Isothermal Temperature Coefficient is determined by varying the RCS temperature. Control element assembly Regulating Group 5 is used to maintain flux and reactivity within a defined operating band.

13.4.3.4 CEA Group Worth Measurements

The RCS is diluted/borated while the CEAs are inserted/withdrawn in order to compensate for a change in reactivity. These changes are monitored via the reactivity computer.

13.4.4 POWER ASCENSION TEST

Power ascension testing consists of frequent monitoring of core power distribution with specific acceptance and review criteria established for 30, 60 and 85% power levels. Equilibrium conditions are established near full power. The Isothermal Temperature Coefficient is measured and compared with predictions.

13.4.5 ACCEPTANCE CRITERIA

Acceptance and review criteria for the above startup tests are listed below:

Parameters		Acceptance Criteria	Review Criteria
CEA Groups Worth		Greater of ± 15% Or ± 0.1% $\Delta\rho$	Greater of ± 15% Or ± 0.1% $\Delta\rho$
Total Regulating CEA Group Worth		± 10% of predicted	± 10% of predicted
Critical Boron Concentration		± 0.75% $\Delta\rho$ of predicted	± 0.5% $\Delta\rho$ of predicted
Isothermal Temperature Coefficient		Technical Specification limits for Moderator Temperature Coefficient	± $0.2\text{x}10^{-4}$ $\Delta\rho/^{\circ}$F
Power Distribution		Technical Specification limits on F_r^T and T_q	Measured radial assembly power distributions (the greater of)
	30%		± 15% or ± 0.15 RPD, or as limited by the applicable core misload detection analysis
	60%		± 10% or ± 0.10 RPD
	85%		± 10% or ± 0.10 RPD
	Full Power		± 10% or ± 0.10 RPD
			Where RPD is the relative power density
Core Symmetry Evaluation			
a.	Tilt		
	30% Power	None	± 3%
	60,85,97% Power	± 3%	± 2%
b.	Symmetric ICI Box Powers	None	± 10%

13.4.6 ACTION AND REVIEW PLAN

The Engineering Supervisor-Fleet Nuclear Fuels, will review the comparison of measurements with Review/Acceptance Criteria.

If any Review Criteria are exceeded, an evaluation will be made to determine first, the applicability of the prediction to the precise plant conditions under which the measurement was performed and, second, the accuracy of the measurement. As a result of this review, the measurement may be repeated and/or the prediction may be updated, if required, to reflect actual plant conditions at the time of measurement.

If any measurement from the lower power physics tests exceeds its Review Criteria, the Plant Operations and Safety Review Committee will review results of the low power physics tests and ensure that Acceptance Criteria are met prior to recommending operation above 5% of Rated Thermal Power. If, as a result of this review, it is determined that a Technical Specification limit has been exceeded, then appropriate action as required by Technical Specifications will be taken. A similar action plan for power ascension testing will be followed prior to increasing power beyond the 60 and 85% power plateaus.

If any Acceptance Criteria, except for bank worth, are exceeded, the validity of the physics data input to the Safety Analysis for the entire cycle will be determined. If it can be demonstrated that the measured value of the particular parameter in question, when

combined with the values of the other safety-related parameters, does not increase the severity or consequences of accidents or anticipated operational occurrences, the test results will be deemed acceptable. Additional measurements of safety-related parameters may be performed in order to support this demonstration.

If any regulating bank worth measurement falls outside of its acceptance criterion or if the total worth of the regulating banks falls outside of its acceptance criterion, shutdown Bank C shall be measured and compared with its acceptance criterion. If shutdown Bank C worth falls outside of its acceptance criterion or if the accumulated total worth of all the banks measured falls below their total worth acceptance criterion (after appropriate corrections and adjustments), then an evaluation shall be made of the validity of the safety analyses for the entire cycle, similar to the procedure discussed above for other measurement data.

If the combination of safety parameters determined above falls outside of the range of safety parameters used to support the proposed operation of the plant, the plant operating limits will be adjusted to prevent conditions that could result in exceeding the Specified Acceptable Fuel Design Limits.

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SAFETY ANALYSIS

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LIST OF ACRONYMS

ABB	Asea Brown Boveri
ADV	Atmospheric Dump Valve
AEC	Atomic Energy Commission
AFAS	Auxiliary Feedwater Actuation System
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
ASGPT	Asymmetric Steam Generator Protection Trip
ASI	Axial Shape Index
AST	Alternative Source Term
BOC	Beginning of Cycle
CAC	Containment Air Cooler
CBP	Condensate Booster Pump
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEAW	Control Element Assembly Withdrawal
CEDM	Control Element Drive Mechanism
CHF	Critical Heat Flux
CIS	Concurrent Iodine Spike
CSAS	Containment Spray Actuation Signal
CTM	Centerline Temperature Melt
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DEG	Double-Ended Guillotine
DEG/PD	Double-Ended Guillotine at Pump Discharge
DEQ	Dose Equivalent Curies
DES/PD	Double-Ended Slot at Pump Discharge
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EOC	End of Cycle
EOP	Emergency Operating Procedures
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FCM	Fuel Centerline Melt
FHI	Fuel Handling Incident
FLB	Feedline Break
FSAR	Final Safety Analysis Report
FTC	Fuel Temperature Coefficient
FTI	Framatome Technologies, Inc.
GIS	Generated Iodine Spike
HDP	Heater Drain Pump
HEPA	High Efficiency Particulate Air
HFP	Hot Full Power
HPSI	High Pressure Safety Injection

LIST OF ACRONYMS

HPT	High Power Trip
HTP	High Thermal Performance
HZP	Hot Zero Power
ICI	Incore Instrumentation
IFBA	Integral Fuel Burnable Absorber
LCO	Limiting Conditions for Operation
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOAC	Loss-of-Non-Emergency AC Power
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LPD	Local Power Density
LPSI	Low Pressure Safety Injection
LPZ	Low Population Zone
LSSS	Limiting Safety System Setting
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFIV	Main Feedwater Isolation Valve
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSS	Main Steam System
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OSG	Original Steam Generator
PCT	Peak Clad Temperature
PD	Pump Discharge
PDIL	Power Dependent Insertion Limit
PIS	Preaccident Iodine Spike
PLCEA	Part-Length Control Element Assembly
PLCS	Pressurizer Level Control System
PLHGR	Peak Linear Heat Generation Rate
PORV	Power-Operated Relief Valve
PPCS	Pressurizer Pressure Control System
PSV	Pressurizer Safety Valves
PWR	Pressurized Water Reactor
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protective System
RRS	Reactor Regulating System
RSG	Replacement Steam Generator
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RWT	Refueling Water Tank
SAFDL	Specified Acceptable Fuel Design Limit

LIST OF ACRONYMS

SDBS	Steam Dump and Bypass System
SDC	Shutdown Cooling
SDCHX	Shutdown Cooling Heat Exchanger
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPEVS	Spent Fuel Pool Exhaust Ventilation System
SG	Steam Generator
SGFP	Steam Generator Feedwater Pump
SGIS	Steam Generator Isolation Signal
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SLB	Steam Line Break
SRP	Standard Review Plan
TBV	Turbine Bypass Valve
TEDE	Total Effective Dose Equivalent
TID	Technical Information Document
TM/LP	Thermal Margin/Low Pressure
UFSAR	Updated Final Safety Analysis Report
VAP	Value Added Pellet
VHPT	Variable High Power Trip
WBD	Whole Body Dose
ZrB2	Zirc Diboride

14.0 SAFETY ANALYSIS

14.1 ORGANIZATION AND METHODOLOGY

This chapter presents analytical evaluations of the Nuclear Steam Supply System (NSSS) response to postulated disturbances in process variables and to postulated malfunctions or failure of equipment. These initiating Design Basis Events (DBEs) are postulated and their consequences analyzed despite the precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. Such occurrences are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such situations (or to identify the limitations of expected performance), and to assure the health and safety of the public in the event of even the most severe of the hypothetical occurrences analyzed.

All initiating events that required a revision to the methodology or that have been added subsequent to the original Final Safety Analysis Report (FSAR) will contain a brief discussion of the link to the current revision. During the latter stages of the licensing of Unit 1 for initial operation, the phenomenon of fuel densification and its adverse effects on plant operation was discovered (Reference 11). The alternative source term (AST) methodology per 10 CFR 50.67 and Reference 31, was approved by the Nuclear Regulatory Commission (NRC) in Reference 30.

Startup of an Inactive Reactor Coolant Pump (RCP) (Idle Loop Startup) has been eliminated from the Updated Final Safety Analysis Report (UFSAR) as Calvert Cliffs is not presently licensed for part loop operation. Part-length control element assembly (PLCEA) drop and malpositioning has also been eliminated as the PLCEAs have been removed from Calvert Cliffs cores. Excess Feedwater Heat Removal, Reactor Coolant System (RCS) Depressurization, Loss-of-Non-Emergency AC Power (LOAC), Transients Resulting from the Malfunction of one Steam Generator (SG) (Asymmetric SG events), Seized Rotor, Excessive Charging, and Feedline Break events have been subsequently added to the DBEs required by the operating license.

14.1.1 CLASSIFICATION OF TRANSIENTS AND ACCIDENTS

14.1.1.1 Categorization

The assignment of the initiating DBEs to categories is made by collecting events with the same major effect and similar occurrence rates into one of three major groups. The three categories discussed below are structured around the specified acceptance criteria described in Section 14.1.1.2. Table 14.1-1 lists all event groups and initiating events considered in the Calvert Cliffs Unit 1 and Unit 2 Safety Analysis.

a. Anticipated Operational Occurrences

Anticipated operational occurrences (AOOs) are initiating events that may occur during the lifetime of the plant. These events are subdivided into initiating events that require the action of the Reactor Protective System (RPS) and those requiring sufficient initial steady-state thermal margin and/or action of the RPS to prevent exceeding the fuel performance criteria, the RCS pressure upset criteria, and the offsite dose criterion.

b. Postulated Accidents

These initiating events are postulated even though they are not expected to occur during the lifetime of the plant based on the very low probability of occurrence. Nevertheless, such events are analyzed to evaluate the protection afforded by the continual upgrading of the plant design and characteristics, the extensive preventive precautions, and by mechanisms

incorporated into the plant design. For all postulated accidents that release radioactivity, the projected offsite dose is shown to be within the acceptance criteria.

c. Postulated Occurrences

Postulated occurrences are events which are not discussed in the above categories, but which could involve a release of radioactivity. For all postulated occurrences that release radioactivity, the projected offsite dose is shown to be within the acceptance criteria.

14.1.1.2 Acceptance Criteria

In order to permit the consistent application of design guidelines to each event, acceptance criteria discussed below have been established for each category.

Anticipated Operational Occurrences are analyzed to demonstrate that the fuel performance, RCS pressure, and offsite dose criteria are not exceeded. Postulated Accidents are analyzed against the RCS Pressure and offsite dose criteria. Postulated Occurrences are analyzed against the offsite dose criteria.

a. Fuel Performance Criterion

The acceptance criteria on fuel performance are the Departure from Nucleate Boiling Ratio (DNBR) and fuel centerline to melt temperature design limit. The steady-state and the transient criterion on clad failure is that the DNBR is greater than or equal to a value consistent with the methods discussed in Reference 47. The steady-state criterion on centerline melt temperature is that the peak fuel centerline temperature is in accordance with Technical Specification 2.1.1.2. For some cases, the reactor power rises rapidly for a very short period of time before the power transient is terminated. For those cases the total energy generated and the corresponding temperature rise at the hot spot are calculated for the duration of the transient to demonstrate that fuel centerline temperatures do not exceed UO_2 melt temperatures.

Section 14.1.4.4 discusses the fuel performance models and associated acceptance criteria.

b. Reactor Coolant System Pressure Criterion

The acceptance criterion on the reactor coolant pressure boundary is the 2750 psia pressure upset limit. This limit is based on 110% of the design pressure of the RCS.

c. Offsite Dose

The acceptance criteria on offsite radiation dose are based on 10 CFR 50.67 criteria. The two-hour site boundary dose limit is 25 REM total effective dose equivalent (TEDE).

d. Loss of Shutdown Margin

Acceptance criteria on loss of shutdown margin have been established to measure the time available to prevent loss of shutdown margin with the RCS cold or cooling. For all reactivity anomalies, the shutdown margin should not go to zero in less than 15 minutes for events initiated in operating Modes 3 through 5 and in less than 30 minutes for events initiated in operating Mode 6 (refueling) following an initiation of the dilution.

14.1.1.3 Section Numbering

The events analyzed in this chapter are titled and numbered as described in Table 14.1-1. For ease of cross-reference with the original FSAR, the table also lists corresponding section numbers for each event.

14.1.2 PLANT CHARACTERISTICS CONSIDERED IN SAFETY ANALYSIS

The following two Sections, 14.1.2.1 and 14.1.2.2, describe the principal parameters and their values used to analyze all the initiating DBEs in this chapter. Specific exceptions for a particular event will be described in the section for each event.

The values of initial conditions and calculated input parameters for each reload core design are compared to the values used to evaluate the UFSAR DBEs. The impact of any variable change on the Safety Analysis is evaluated. If all current cycle values for a particular event are bounded by (conservative with respect to) the UFSAR, no reanalysis is performed.

14.1.2.1 Initial Conditions

The events discussed in this chapter have been analyzed over the normal operating range for the initial conditions that significantly affect the boundary dose, RCS pressure, and fuel performance. These variables include core power level, core power distribution, core coolant inlet temperature, primary system pressure, RCS flow, and secondary system pressure.

The range of initial conditions used in the Safety Analysis is consistent with the Technical Specifications and bounds the space for which the Safety Analysis is valid. For conservatism in the Safety Analysis, all applicable uncertainties are assumed to occur simultaneously in the most adverse direction except as discussed in Section 14.1.4.

14.1.2.2 Input Parameters

The parameters used in the analysis of UFSAR DBEs are consistent with those listed in UFSAR Chapter 3. The following principal input parameters are discussed based on their significance in the Safety Analysis.

a. Doppler Coefficient

The effective fuel temperature coefficient (FTC) of reactivity (Doppler coefficient) is shown in UFSAR Chapter 3 for first cycle, beginning of cycle (BOC) and end of cycle (EOC) conditions. These curves were adjusted by 15% to conservatively account for uncertainties in determining the actual fuel temperature reactivity effects.

The effective fuel temperature is discussed in UFSAR Chapter 3. Equivalent fuel temperatures calculated in the S-RELAP5 computer code are used to interpolate Doppler reactivity changes during the transients.

b. Moderator Temperature Coefficient

A range of Moderator Temperature Coefficient (MTC) of reactivity is considered. These values include all uncertainties and bound the expected equilibrium cycle MTC for all the cycle exposures, power levels, control element assembly (CEA) configurations, and boron concentrations. The most conservative value of the MTC is assumed for each analysis.

The Steam Line Break (SLB) analysis uses an explicit table of reactivity versus moderator density to model moderator feedbacks instead of a single value of MTC. That table is consistent with the assumed stuck CEA. With the large moderator density variations that occur during an SLB event, an explicit moderator reactivity feedback is necessary.

c. Shutdown CEA Reactivity

The transient shutdown CEA reactivity (SCRAM) is dependent on CEA worth available on reactor trip, axial power distribution, the position of the regulating CEAs, and time in core life. The minimum total negative reactivity worth of CEAs available for the present cycle during a reactor trip from full power and zero power for both beginning and EOC is given in UFSAR Chapter 3. The net CEA worth available at trip is the minimum total worth reduced by the worth of the most reactive CEA. To increase operating flexibility, each initiating event uses a shutdown (SCRAM) reactivity that is consistent with conditions at the time of reactor trip. The Power Dependent Insertion Limit (PDIL) discussed in the Technical Specifications assures that the necessary CEA worth is available upon reactor trip.

Transient shutdown reactivity worth versus CEA position curves are dependent on Axial Shape Index (ASI) and Regulating CEA Group positions. The most conservative one for the particular set of initial conditions is used for each event. The shutdown worth versus CEA position curves predict a lower rate of reactivity insertion than is expected under any allowed initial condition. The shutdown worth versus CEA position curve is combined with a CEA position versus time curve to yield a shutdown worth versus time curve for the analyses. Consequently, a conservative representation of shutdown reactivity insertion rate is used for reactor trips that occur as a result of the events considered in this chapter.

d. Effective Neutron Lifetime Delayed Neutron Fraction

The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup.

For the analysis of each event, either BOC or EOC cycle specific values of the neutron lifetime and of the delayed neutron fraction are selected.

e. Decay Heat Generation Rate

AREVA analyses based on full power initial conditions conservatively assume a decay heat generation rate based on the 1971-1973 Proposed American Nuclear Society (ANS) Decay Heat Standard, which includes contributions from heavy element decay and is based on an infinite reactor operation period at full power.

f. Core Inlet Temperature (T_{cold})

A core inlet temperature range of 525°F to 535°F was originally used in the safety analysis for Hot Zero Power (HZP) events. These events have been evaluated to the minimum temperature of criticality allowed by the Technical Specifications, 515°F, and determined that sufficient analytical conservatisms exist to preserve the conclusions of the existing analysis. For at power events, core inlet temperature is assumed consistent with program temperature as shown in UFSAR Figure 4-9. A temperature indication uncertainty is normally applied in the safety analysis.

14.1.3 ASSUMED PROTECTION SYSTEM ACTIONS

During the course of any event, various systems are called upon to function. Such systems are described in UFSAR Chapters 6, 7, and 9. The manner in which these systems function during events is discussed in the Sequence of Events and Systems Operation subsections of each event description. Section 14.1.3.1 describes the Sequence of Events and Systems Operation. Section 14.1.3.2 describes the plant protection system analysis setpoints and delay times assumed in the analyses presented in this chapter. Section 14.1.3.3 reviews the status of control systems assumed in the transient analysis.

14.1.3.1 Sequence of Events and Systems Operation

The purpose of the Sequence of Events and Systems Operations subsections is to identify:

a. The sequence of events from event initiation to the final stabilized condition;

b. The extent to which normally operating plant instrumentation and controls are assumed to function;

c. The extent to which plant protection systems are required to function;

d. The credit taken for the functioning of normally operating plant systems; and

e. The operation of engineered safety systems.

14.1.3.2 Protection System Setpoints

The Calvert Cliffs Units have two protection systems: The RPS and the Engineered Safety Feature Actuation System (ESFAS).

The RPS is described in UFSAR Chapter 7. Analytical setpoints include instrument uncertainty and are conservative relative to the allowable limits listed in Table 7-1. Credited trip delay times are equal to or conservatively larger than values listed in Table 7-2. Delay times are defined as the elapsed time between the parameter reaching its analysis setpoint and the opening of the reactor trip breakers.

The time interval between opening of the trip breaker and the point at which the magnetic flux of the CEA holding coil has decayed enough to allow CEA motion is conservatively assumed to be 0.50 seconds. Finally, a conservative value of 2.6 seconds is assumed as the elapsed time from the beginning of CEA motion to the time the CEAs are 90% inserted into the reactor core. Thus, a time interval of 3.1 seconds is assumed between the interruption of power to the CEA holding coil and the point of 90% insertion.

The ESFAS is described in UFSAR Chapter 7. The analysis setpoints and systems actuated by the ESFAS during each event are described in the Sequence of Events subsection for each event. Credited response times are equal to or conservative to the values listed in Table 7-4.

Critical assumptions made in the Safety Analysis about valve and pump responses are included in Table 14.1-2.

14.1.3.3 Control System Operational Status

Some normally operating control systems are assumed to function during the course of the events described in this chapter. The operability status of a control system is dependent on the criteria being addressed for a particular initiating event. In general, automatic operation of these normally available control systems is assumed unless operation in the manual mode would make the approach to acceptance criteria more limiting. In the manual mode, the control system is assumed to continue to function as it was prior to the event. For example, for a peak pressurizer pressure event, the Steam Bypass Control System is assumed to be in manual and will remain closed throughout the event, although it would normally be called upon to open at reactor trip. The assumed status of each control system is presented in the discussion of each event.

During normal operating conditions, the plant is operated with the Feedwater Regulating System in automatic and with the SG water level at the Normal Water Level. This mode is therefore assumed for events analyzed in this chapter.

The Pressurizer Level Control System (PLCS) regulates the water level in the pressurizer within ± 5% of the programmed level during normal plant operating conditions. For events analyzed in this chapter, pressurizer level is assumed to be between 116 and 242", except when three charging pumps are operating and letdown flow is less than 29 gpm. Under those conditions pressurizer level is assumed to be between 116 and 227".

Although the plant regulates the pressurizer pressure at 2250 psia for normal plant operating conditions, events analyzed in this chapter assume a value between 2200 and 2300 psia. A pressure indication uncertainty is applied as is appropriate. As of Unit 1 Cycle 19 and Unit 2 Cycle 17, analyses are assessed for initial pressurizer pressure values up to and including 2311 psia.

The boration system is a subset of the Chemical Volume and Control System (CVCS) and ensures that negative reactivity control is available during each mode of facility operation. Typically, the CVCS is assumed to be initially operating with one charging pump, with a total of six gpm to the RCPs and a resultant letdown flow rate, with minimum pressurizer spray, and with minimum heat input from the proportional heaters.

For Units 1 and 2, loss-of-coolant accident (LOCA) flow from the charging pumps is not required and is not credited.

No boration is assumed to occur via the CVCS upon a safety injection actuation signal (SIAS) since the system may be lined-up to dilute directly to the charging pump suction. In this line-up, the boric acid pump head may not overcome the reactor coolant makeup pump head when SIAS actuates. As a result, concentrated boric acid may not be injected upon a SIAS.

For all events, the CEA motion inhibit is assumed to be operable. Control element assembly motion inhibit insures that programmed CEA group overlap will be maintained and that a single CEA withdrawal will not occur.

14.1.4 CORE AND SYSTEM PERFORMANCE

14.1.4.1 Mathematical Models

This section briefly describes the computer codes used in analyzing the DBEs.

A. Non-Loss-of-Coolant Accident Events

1. CESEC

The CESEC digital computer code (Reference 1) simulates a Combustion Engineering, Inc. (CE) NSSS. The code calculates the plant response for non-loss-of-coolant accident initiating events for a wide range of operating conditions.

The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the SGs, and the RCPs. The secondary system components include the secondary side of the SGs, the steam system, the feedwater system, and the various steam control valves. In addition, the program models those plant control and protection systems needed to perform the safety analysis.

The secondary systems in CESEC are modeled as a single node. In events where the peak pressure of the secondary system is considered, the elevation head of the liquid in the steam generator downcomer is added to the CESEC peak steam generator pressure.

2. S-RELAP5

The S-RELAP5 plant transient thermal-hydraulic system code is used to simulate the overall transient response of the RCS and steam systems during the non-LOCA event. The S-RELAP5 plant model includes a thermal model of the fuel, a hydraulic model of the RCS, a point-kinetics model of the reactor, a hydraulic model of the steam system, and control logic which represent various RPS trips. The RCS hydraulic model simulates the hot legs, pressurizer, SGs (primary sides), cold legs, RCPs, reactor vessel, and core. The steam system hydraulic model simulates the SGs (secondary sides), main steam lines, and turbine.

3. FIESTA

FIESTA is no longer used due to AREVA performing the feed line break analysis.

4. AREVA Scram Insertion

The critical eigenvalue as a function of control rod position is calculated. The scram reactivity is constructed as the combination of a table of scram reactivity versus control rod position and control rod position versus time. The control rods are inserted into a bounding bottom-peaked shape to delay the reactivity insertion. The scram worth is reduced to account for the worst stuck control rod, as appropriate. Delays between the time the trip setpoint is reached and the start of control rod insertion are included in the transient analysis.

5. Statistical Setpoint/Transient Methodology

The analog protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying:

a) Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS;
b) Limiting Conditions for Operation (LCO) for reactor system parameters; and,

c) LCO for equipment performance.

The LSSS, combined with the LCO, established the thresholds for protection system action to prevent exceeding acceptable limits during DBEs where changes in DNBR and linear heat rate (LHR) are important. The limits addressed by the RPS are:

a) The reactor fuel shall not experience centerline melt; and
b) The DNBR shall have a minimum allowable limit corresponding to a 95% probability at a 95% confidence level that Departure from Nucleate Boiling (DNB) will not occur.

The RPS trips jointly provide protection for all AOOs. The RPS providing primary protection from centerline melt is the Local Power Density (LPD) LSSS. The RPS providing primary DNB protection is Thermal Margin/Low Pressure (TM/LP) LSSS.

The design of the RPS requires that correlations including uncertainties be applied to express the LSSS in terms of functions of monitored parameters. These functions are the trip limits that are then set into the RPS.

Two methods can be used to compensate for uncertainties. It can be assumed that all applicable uncertainties occur simultaneously in the most adverse direction even though not all of the uncertainties are systematic; some are random and some contain both systematic and random characteristics. This assumption is extremely conservative. Reference 45 documents the methodology used to statistically combine uncertainties explicitly in lieu of the credit previously used.

The scope of Reference 45 encompasses the following objectives:

a) To define the method used to statistically combine uncertainties to determine the TM/LP and LPD LSSSs and LCOs;
b) To describe the methods for statistically confirming the TM/LP and LPD LSSSs and LCOs.

Operation within the DNB and LPD LCOs provides the necessary initial DNB and LHR margin to prevent exceeding acceptable limits during DBEs where changes in DNBR and LHR are important.

With the introduction of AREVA fuel assemblies, a new method for the statistical combination of uncertainties was used to verify the TM/LP LSSS, the LPD LSSS, the DNB LCO, and LPD LCO.

The LCOs and LSSSs protect against fuel failure in LOCAs, prevent DNB, and meet the specified acceptable fuel design limits (SAFDLs) for fuel centerline melt (FCM). Loss-of-coolant accident limits are based on the LHGR used in the LOCA analyses, DNB limits are based on correlations which have been approved by the NRC, and FCM limits are calculated for each reload cycle and fuel design.

Statistically combining the uncertainties involved in calculating LCOs and LSSSs establishes conservative and meaningful values for those settings. The statistical approach provides an accurate method for

accounting for uncertainties and can require a large number of calculations.

A bounding approach is used to reduce the number of calculations for some cases. This approach is used for cases that have many nominal cases. The uncertainties are combined at each nominal point and a margin is defined. In general, the number of calculations used in the analysis can be reduced by statistically combining the bulk of the uncertainties at a single nominal point and applying this calculational uncertainty to every nominal point. The nominal point used is conservatively chosen to provide the greatest uncertainty in the calculated results and therefore, a conservative estimate at all other points.

The nominal point is chosen by finding the location where the difference between the nominal point and the deterministic calculation is maximum. This method is used in deriving and confirming the setpoints. Deriving an LCO or LSSS is somewhat more difficult than confirming an existing setpoint, because the power uncertainty coming from the ASI uncertainty depends on the functional form and the process becomes iterative.

<u>LPD LSSS and LCO</u>

Protection against FCM is provided by the LPD LSSS and the LPD LCO. The LPD LSSS protects against FCM by monitoring the power level of the reactor and tripping the reactor when the power level exceeds the trip setpoint corresponding to the ASI. The LPD LCO limits power operation based on the ASI. The function of the LPD LCO is to protect against the LPD exceeding the LHGR limit established in the LOCA analysis.

These functions are based on the worst axial (F_z) and radial (F_r) power distributions for a given ASI. Radial power distribution does not affect the ASI directly. Therefore, the radial peaking factor assumed for all values of ASI is the Technical Specification value. The F_r is augmented to account for increased peaking when CEAs are inserted.

Each axial power distribution has a value for ASI. However, an unlimited number of axial power distributions can correspond to the same ASI. In determining the LPD LSSS and LCO, the most limiting axial distribution for a given ASI sets the limit.

The LPD LSSS setting, without uncertainties, is generated by determining the power at which FCM is predicted to occur. The allowed power for each axial power shape and corresponding F_q is calculated. The resulting power and ASI points are plotted and a curve is drawn below all of the power versus ASI points. This curve represents the LPD LSSS, without accounting for uncertainties. The curve provides the power and ASI combination where any axial shape at this power level and ASI value will protect against FCM.

To confirm the LPD LSSS, a series of calculations is performed using each axial power shape. Those shapes for which the melt power exceeds the power and ASI combination by the offset of the Variable High Power Trip (VHPT), adjusted for uncertainties, are not

considered. For the remaining shapes, the nominal margin between the trip power and the FCM power is calculated, then adjusted for uncertainties. This adjustment is made by calculating the probability distribution in margin between the trip power and the power at which FCM would occur. Using a one-sided, lower 95% of the margin from the distribution, a table of margin versus ASI is created.

The uncertainties that must be accounted for are from two sources: measurement and calculation uncertainties. Calculation uncertainties include model structural deficiencies and parameter uncertainties. The uncertainties included in the LPD LSSS include;

- Engineering uncertainty
- Power peaking uncertainty
- Azimuthal tilt allowance
- Power uncertainty, which may be power dependent
- Trip power uncertainty (for LSSS only)
- Transient decalibration and trip overshoot bias (for LSSS only)
- ASI uncertainty

The ASI measurement drift and power measurement are pure measurement uncertainties and can be applied directly to the results of an uncertainty analysis in which they are not varied.

The LPD LCO is similar in form to the LPD LSSS and is based on preventing the plant from exceeding a reduced LHGR during operation. The value of the reduced LHGR is no greater than the value used in the LOCA analysis. The LPD LCO comes into effect only when the in-core detectors are not in service. Since the LPD LCO has no trip associated with it, uncertainties associated with the trip are not applied to the limits.

<u>TM/LP LSSS and LCO</u>

Protection against DNB is provided by the TM/LP LSSS and the DNB LCO. The TM/LP LSSS trips the reactor when conditions approach a 95% probability that the limiting fuel pin undergoes DNB. The DNB LCO ensures that AOOs will not result in DNB with at least a 95% probability at a 95% confidence level.

The TM/LP trip is actuated when the measured pressure falls below a calculated limit. The calculated limit is based on analysis of the DNBR as a function of pressure, ASI, power, and inlet temperature. The TM/LP has a minimum pressure that will result in a trip. The TM/LP trip pressure is the maximum of the calculated trip pressure and the floor.

The power term is adjusted, depending upon the ASI. Peaking factors are adjusted for part-power rodded configurations. The power input to the TM/LP is the maximum of that calculated by two different methods: the excore (neutron) detectors and a thermal calculation.

An iterative scheme is used to obtain the DNB portion of the variable pressure setting for a selected value of ASI, power, and T_{inlet}. The axial power shape that produces the worst DNBR for a range of ASI

values centered around a selected ASI is used to calculate the pressure corresponding to DNB. Values of power which are not permitted by the LPD LSSS or the MSSV settings are not considered in the analysis. The radial peaking factor is set at its nominal value. Flow is fixed at the thermal design limit and the DNBR target is set to the DNBR mean, adjusted for rod bow and mixed core penalties. Power and T_{inlet} are chosen and the pressure corresponding to DNB is calculated. The iteration provides a single point for DNB pressure.

The process is repeated for a set of ASI values ranging from -0.6 to 0.6 and over a large range of powers and T_{inlet}. The results of the nominal calculation set provides a series of pressures as a function of power, ASI, and T_{inlet} at which the CHF calculated by the DNBR correlation and corrected for rod bow and mixed core penalties, is equal to the calculated heat flux.

The TM/LP LSSS protects against hot-leg saturation and DNB during slow transients. The uncertainties that are accounted for include;

- Radial power peaking factor
- DNB correlation uncertainty
- Loop flow
- Pressure measurement
- Inlet temperature measurement uncertainty (RTDs)
- Power measurement uncertainty
- ASI uncertainty

The DNB LCO is designed to protect the DNBR SAFDL during an AOO. It is set and confirmed by the limiting transient events. The limiting transients are those that produce the largest decrease in DNBR from an initial steady-state power and ASI. Operation of the reactor within the limits of power versus ASI, disregarding uncertainties, means no AOO would result in DNBR less than or equal to the adjusted mean of the applicable DNBR correlation.

Uncertainties in the parameters and model structure are accounted for by using the probability distribution in DNB power at the most sensitive point. This point corresponds to the axial power shape that produces the greatest difference between the nominal and the deterministic percent allowed power. Protection against DNB is provided by limiting the reactor power based on the peripheral (external) ASI. The shape of the DNB LCO, in conjunction with radial power peaking limits and other LCOs, protects against DNB.

The uncertainties applicable to the DNB LCO that must be accounted for are listed below. Applicability of the uncertainty is dependent upon the event analyzed.

- Radial power peaking factor
- Pressure measurement
- DNB correlation uncertainty
- Loop flow
- Inlet temperature measurement uncertainty (RTDs)
- ASI uncertainty

- Power measure uncertainty
- Pump coastdown (for loss-of-coolant flow)
- Low flow trip uncertainty (for loss-of-coolant flow)
- Scram delay (for loss-of-coolant flow)
- Scram worth (for loss-of-coolant flow)
- Scram rate (for loss-of-coolant flow)

Either the excore or the incore detectors can be used to monitor the LHR LCO. The DNB LCO and ASI is monitored on the excore detectors.

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the axial power distribution trip will limit the allowed ASI during operation.

6. Control Element Assembly Withdrawal

AREVA analysis is performed at HFP and HZP conditions. Appendix C of the Technical Specifications prohibits changing Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 until an NRC accepted generic, or Calvert Cliffs specific, basis is developed for analyzing the CEA Rod Bank Withdrawal event, the CEA Drop event, and the CEA Ejection event (power level sensitive transients) at full power conditions only.

B. Containment Response

RELAP5/MOD2-B&W is used to generate the mass and energy release rates used in the containment analysis for the main steam line break (MSLB) inside Containment. RELAP5/MOD2-B&W is an advanced system analysis computer code designed to analyze a variety of thermal-hydraulic transients in light water reactor systems. RELAP5/MOD2-B&W is advanced over its predecessors due to its six-equation, full non-equilibrium, two-fluid model for the vapor-liquid flow field and partially implicit numerical integration scheme for more rapid execution. As a system code, it provides simulation capabilities for the reactor primary coolant system, secondary system, feedwater trains, control systems, and core neutronics. Special component models include pumps, valves, heat structures, electric heaters, turbines, separators, and accumulators. Code applications include the full range of safety evaluation transients, LOCAs, and operating events. RELAP5/MOD2-B&W is approved by the NRC for performing analysis on plants with recirculating steam generators.

The mass and energy released into Containment during a LOCA has been calculated by Westinghouse.

C. Loss-of-Coolant Accident

1. S-RELAP5

The S-RELAP5 code (Reference 38) is a RELAP5 based thermal-hydraulic system code developed by AREVA NP for performing small break LOCA (Reference 39) and realistic analyses of a large break LOCA (Reference 40) in PWRs. RELAP5 is a light water reactor transient analysis code developed at the Idaho National Engineering Laboratory for the NRC. The main purpose of the RELAP-5 code is to calculate the behavior of a RCS during a transient by simulating a wide variety of hydraulic and thermal transients in both nuclear and non-

nuclear systems involving mixtures of steam, water, non-condensable gas, and solute. The code includes models of hydrodynamic systems, heat transfer and heat conduction, fuel, reactor kinetics, control system, and trip system models.

AREVA NP has incorporated improvements to the code as required to provide congruency with the unmodified literature correlations and those required to obtain adequate simulation of key LOCA experiments.

2. RODEX3A

RODEX3A is a best-estimate fuel code which has been approved for use in the performance of realistic large break LOCA analyses. The RODEX3A code simulates the thermal and mechanical response of a fuel rod in a coolant channel as a function of exposure for the normal and power ramp conditions encountered in a PWR. Phenomenological rate-dependent models are used to evaluate the temperature-, stress-, and exposure-dependent changes in the state of the fuel and cladding materials and in the release of the inert gaseous fission products. A quasi-steady-state computational procedure is used to evaluate the response of a fuel rod as a function of time.

3. ICECON

The ICECON code has been incorporated into the S-RELAP5 code to provide the required containment boundary conditions for the large break LOCA analysis. ICECON was developed to predict the long-term behavior of PWR nuclear plant containment systems.

4. RODEX2

The RODEX2 code (References 43 and 44) is a fuel rod code which has been approved for use in the performance of small break LOCA analyses (Reference 39). The RODEX2 code simulates the thermal and mechanical response of a fuel rod in a coolant channel as a function of exposure for the normal and power ramp conditions encountered in pressurized (PWR) and boiling water reactors. The code incorporates models to describe the thermal-hydraulic condition of the fuel rod in a flow channel; the gas release, swelling, densification, and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion. The calculations are performed on a time incremental basis with conditions being updated at each calculated increment.

D. Thermal Hydraulics

1. XCOBRA-IIIC

Based on the overall core conditions calculated by S-RELAP5 at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code (Reference 41) is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single “channel”) linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single “channel”). The limiting assembly DNBR

calculations are performed using an approved AREVA DNB correlation.

2. COAST

In the original FSAR, the COAST (Reference 6) computer code was used to calculate the reactor coolant flow coastdown transient characteristic for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation for conservation of mass is written for the appropriate nodal points. Pressure losses due to friction, bends, and shock losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

The current coolant flow coastdown transient is based on actual plant data taken in 1981, adjusted for the effect of steam generator tube plugging as predicted by COAST.

E. Fuel Performance

The RODEX2-2A and RODEX2-3A fuel codes are used to calculate fuel performance for input to non-LOCA and LOCA transients. RODEX2-2A (References 43 and 44), was developed to perform calculations for a fuel rod under normal operating conditions. The code incorporates models to describe the thermal-hydraulic condition of the fuel rod in a flow channel; the gas release, swelling, densification and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion.

For realistic large break LOCA, fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. RODEX3A computes the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance at the reference fuel temperature and zero power. RODEX3A data is transferred to S-RELAP5 and a steady-state S-RELAP5 calculation is required to initialize the S-RELAP5 calculation at the power of interest.

F. Reactor Physics Computer Codes

Numerous computer codes are used to produce the reactor physics parameter input values required by the NSSS simulation and other codes previously described. These reactor physics codes are described in Chapter 3.

14.1.4.2 Operator Action Requirement

Operator actions assumed to mitigate the consequences of the events presented in this chapter are delineated in each event description requiring such action. The operator should be alerted to the need for action by an unambiguous alarm. An unambiguous alarm is one which, within the time period allowed for diagnosis, would make the operator aware of the need to take the action assumed. There

may or may not be redundant or diverse plant alarms for a particular action; however, there are always continuously operating, non-alarming visual indications of relevant plant process parameters in the Control Room which serve as back-ups.

A time delay is assumed between the unambiguous alarm and the accomplishment of any manually-initiated action. This delay conservatively accounts for the time required by the operator to diagnose the event, decide what action to perform, and then initiate this action.

After initiation of the Boron Dilution event, operator action is initiated within 15 minutes when in Modes 1 through 5 and within 30 minutes when in Mode 6.

14.1.4.3 Activity Release Methodology

This subsection summarizes the assumptions, parameters, and calculational methods used to determine the doses that result from postulated events. The total activity released to the site boundary is dependent upon the initial activity in the primary and secondary systems as well as any changes in activity resulting from the event. Depending on the initiating event, radioactivity could be released to the site boundary through the atmosphere dump valves, main steam safety valves (MSSVs), steam turbine-driven auxiliary feed pumps, condenser air removal system, and/or leak through the Containment and engineered safety feature (ESF) system.

a. Primary System Activity

   The primary system activity is based upon the initial activity in the primary coolant and the activity released to the coolant due to failed fuel rods during the event. The initial primary system concentration is conservatively assumed to be at the Technical Specification limit of 0.5 μCi/gm Dose Equivalent curies (DEQ) I-131 and 100/E μCi/gm gross activity.

b. Fuel Cladding Failure

   The release of gas gap activity from pins calculated to experience clad failure is considered. For each pin calculated to experience clad failure, a conservative fraction of the total pin activity of iodines and noble gases is assumed to be present in the gas gap. This amount is assumed to be released instantaneously to the RCS. This release increases the primary system specific activity.

c. Secondary Activity

   A conservative set of secondary side specific activities is used to calculate releases from the SGs. The DEQ I-131 activity limit is the Technical Specification limit of 0.1 μCi/gm.

d. Steam Generator Tube Leakage

   All events considered, with the exception of the Steam Generator Tube Rupture (SGTR) event, were analyzed assuming the Technical Specification limit of 100 gal/day through each SG. For the SGTR event, 200 gal/day is assumed to leak through the unaffected SG.

e. Containment Leakage

   The assumed leak rate from containment is the maximum Technical Specification limit of 0.16% by weight of containment air per day.

f. <u>Calculational Factors</u>

A conservative set of decontamination factors (i.e., reciprocal of partition factor) for iodine releases from the primary and secondary side was used to calculate activity releases to the atmosphere. The Decontamination Factor for dose calculations is functionally defined as the ratio of the concentration of iodine in water to that in steam. These values are presented in Table 14.1-3.

A breathing rate of $3.5x10^{-4}$ m^3/sec is assumed in the analysis which is characteristic of the active portion of a normal adult workday.

An atmospheric dispersion coefficient Chi/Q of ≥ $1.3x15^4$ sec/m^3 is assumed in the analysis. The relative concentration is conservative for the 0-2 hour site boundary condition described in Chapter 2.

g. <u>Calculational Methods</u>

The isotopic activities released from the failed fuel per unit power were generated by the isotope generation and depletion computer code SAS2H/ORIGEN-S (References 32 and 33) and multiplied by the relevant power level and release fractions. The SAS2H control module performs the depletion/decay analysis using the well-established codes and data libraries provided in the SCALE system. Problem-dependent resonance processing of neutron cross-sections is performed using the Bondarenko resonance self-shielding module BONAMI-S and the Nordheim Integral Treatment resonance self-shielding module NITAWL-II. The XSDRNPM-S module is used to produce spectral weighted and collapsed cross-sections for the fuel depletion calculations. COUPLE updates the cross-section constants included on an ORIGEN-S nuclear data library with data from the cell-weighted cross-section library produced by XSDRNPM-S. The weighting spectrum computed by XSDRNPM is applied to update all nuclides in the ORIGEN-S library that were not specified in the XSDRNPM analysis. The point-depletion ORIGEN-S module is used to compute time-dependent concentrations and source terms for isotopes simultaneously generated and depleted through neutronic transmutation, fission, and radioactive decay. The cross-section library 44GROUPNDF5 was utilized in this analysis. 44GROUPNDF5 is a 44-energy group library derived from the latest ENDF/B-V files with the exception of O-16, Eu-154, and Eu-155, which were taken from the improved ENDF/B-VI files. Note that the SAS2H/ORIGEN-S libraries include 689 light elements, such as clad and structural materials, 129 actinides including fuel nuclides and their decay and activation products, and 879 fission product nuclides.

The RADTRAD (References 34 through 36) computer code can calculate TEDE and thyroid doses to personnel at the site boundary, low population zone, and Control Room per the AST methodology of 10 CFR 50.67 and Reference 31 resulting from any postulated accident which releases radioactivity within the Containment, SFP, or within the primary system. RADTRAD models the transport of radioactivity from up to 63 radioisotopes from the sprayed and unsprayed regions of a primary containment or a SFP area, through the secondary containment if any, and then to the environment and to the Control Room. The code includes the capability to model time-dependent activity release; containment spray, filtration, and leakage; control room filtration and inleakage; primary and secondary containment purge filters; Control Room intake filters; atmospheric dispersion; and natural decay. The activity released to the environment is

transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients.

MICROSHIELD (Reference 37) is a comprehensive photon/gamma ray shielding and dose assessment point kernel program that is used for calculating shine doses from the Containment volume, Containment and Control Room filters, and plume activities. The MICROSHIELD point kinetics code is capable of modeling various geometries including distance and orientation between source and dose point; dimension of the source region; and the dimensions, locations, and orientations of intervening shields. MICROSHIELD has solution algorithms for 16 source geometries including points, lines, disks, spheres, cylinders, rectangular volumes, truncated cones, infinite planes, and infinite slabs with various shields. The source and shield material compositions, densities, and buildup factors can be specified. A source strength as a function of isotopic content or photon energy spectrum can also be specified.

14.1.4.4 Fuel Performance Models and Acceptance Criteria

The acceptability of fuel rod performance is assessed by comparing the predicted thermal and mechanical behavior to appropriate guidelines. The computer codes used to determine the acceptability of fuel rod performance are described in Section 14.1.4.1. S-RELAP5 predicts the transient core average heat flux, the core inlet average coolant condition, and the RCS pressure, the values of which are all input to XCOBRA-IIIC. XCOBRA-IIIC predicts the DNBR in the reactor core based on a closed-channel, thermal-hydraulic model. RODEX predicts the hot rod fuel and clad temperatures and the pressure differential across the clad. The analysis of all events has been taken to the point of showing decreasing fuel temperatures, increasing DNBR, and a sufficient coolant inventory to meet the criteria.

The following sections describe the models and fuel behavior acceptance criteria used in analyzing each category of DBEs.

A. Anticipated Operational Occurrences

1. Departure from Nucleate Boiling Ratio

The steady-state and transient SAFDL on DNBR is that the minimum DNBR shall provide at least a 95% probability with a 95% confidence of not experiencing DNB on the fuel rod with that DNBR. Compliance with this limit ensures that there is a low probability of fuel rods being damaged due to cladding overheating.

Values of the following system parameters are used to determine the minimum DNBR during an event. Uncertainties are included.

1. Thermal power,
2. Integrated radial peaking factor,
3. Core inlet temperature,
4. Pressurizer pressure,
5. Parametric analyses in axial shape, and
6. RCS core flow.

The XCOBRA-IIIC computer code is used for the evaluations of minimum DNBR by AREVA. The XCOBRA-IIIC code is a steady-state thermal-hydraulics code that calculates the axial and radial flow and

enthalpy distribution within assemblies and sub-channels for non-LOCA events. When used in conjunction with core boundary conditions from the S-RELAP5 transient analysis and high thermal performance DNB correlation; XCOBRA-IIIC also calculates the corresponding Minimum Departure from Nucleate Boiling Ratio (MDNBR). Minimum DNBR calculations are performed in a two-step process. Calculations are first performed on a core-wide basis to calculate the axially varying flow and enthalpy distribution in the peak powered fuel assembly. Next, these flow and enthalpy boundary conditions are applied to a sub-channel model of the peak powered fuel assembly. Then, these flow and enthalpy boundary conditions are applied to a sub-channel model of the peak powered assembly to determine the local conditions for the calculation of MDNBR.

2. Fuel Temperature

The SAFDL on fuel temperature is that no significant fuel melting will occur during steady-state operation or during a transient. The fuel melting point is assumed to be that described in the Technical Specifications. Compliance ensures that the fuel rod is not damaged as a result of material property changes and increases in fuel pellet volume which could accompany fuel melting.

3. Site Boundary Dose

During AOOs, the site boundary dose shall not exceed a small fraction of 10 CFR 50.67 guidelines. The site boundary dose is controlled by meeting the DNBR and fuel temperature SAFDLs described in Items 1 and 2 above, which preclude any significant cladding damage.

4. RCS Pressure Upset Limit

The RCS Pressure Upset Limit is 110% of design pressure and is equal to 2750 psia.

B. Postulated Accidents

1. Site Boundary Dose

For Postulated Accidents, the site boundary dose shall not exceed the 10 CFR 50.67 guidelines. This is met by minimizing the number of fuel rods predicted to fail.

For the CEA Ejection event, radioactive gases are assumed to be released from the pellet to the coolant if the deposited energy is predicted to exceed the threshold value. For this event, Reference 46 is used to calculate the deposited energy (Section 14.13).

The pre-trip SLB event assumes that all fuel rods with a DNBR less than the NRC-approved safety limit experience DNB. The safety limit DNBR gives a 95% probability, at a 95% confidence level, that the hottest fuel rod will not experience DNB. For dose calculations, allrods experiencing DNB are assumed to fail. For the return to power SLB analysis, DNB is assumed when the rod experiences a DNBR less than 1.135 calculated using the limit associated with the modified Barnett CHF correlation (Reference 42).

2. RCS Pressure Upset Limit

The RCS Pressure Upset Limit is 110% of design pressure and is equal to 2750 psia.

3. Containment Pressure

The containment design pressure limit is 50 psig. It is not explicitly analyzed for events that bypass Containment (SGTR) or have little or no mass release, such as Seized Rotor.

4. Emergency Core Cooling System Criteria

The LOCA event shall meet the criteria in Reference 10.

5. Coolable Core Geometry

Fuel geometry is maintained such that a path for removal of decay heat is ensured. This criteria is met by limiting the extent of fuel melting.

C. Postulated Occurrences

1. Site Boundary Dose

For postulated occurrences, the site boundary dose shall not exceed the 10 CFR 50.67 guidelines. This is met by minimizing the number of fuel rods predicted to fail.

14.1.5 REFERENCES

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TABLE 14.1-1

DESIGN BASIS EVENTS

A. Anticipated Operational Occurrences for which action of RPS is required to prevent exceeding acceptable limits:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.3	14.3	Boron Dilution
14.5	14.9	Loss of Load
14.6	14.10	Loss of Feedwater (LOFW) Flow
14.8	--	RCS Depressurization

Anticipated Operational Occurrences for which the RPS trips and/or sufficient initial steady-state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.2	14.2	Control Element Assembly (CEA) Withdrawal
14.4	14.11	Excess Load
14.7	--	Excess Feedwater Heat Removal
14.9	14.6	Loss-of-Coolant Flow
14.10	--	LOAC
14.11	14.4	CEA Drop
14.12	--	Asymmetric SG Events
14.25	--	Excessive Charging

B. Postulated Accidents:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.13	14.13	CEA Ejection
14.14	14.12	SLB
14.15	14.14	SGTR
14.16	--	Seized Rotor
14.17	14.15	LOCA
14.26	--	Feedline Break

C. Postulated Occurrences:

UFSAR NUMBER	ORIGINAL FSAR NUMBER	EVENT
14.18	14.5	Fuel Handling Incident
14.19	14.8	Turbine - Generator Overspeed Incident
14.20	14.16	Containment Pressure Analysis
14.22	14.17	Waste Gas Incident
14.23	14.20	Waste Processing System Incident
14.24	14.18	Maximum Hypothetical Incident

TABLE 14.1-2

SAFETY ANALYSIS VALVE AND PUMP ASSUMPTIONS

COMPONENT		INSTALLED QUANTITY	ANALYSIS TOTAL CAPACITY	ANALYSIS SETPOINT
Turbine Bypass Valves		4	40%[(a)]	Start open = 895 psia (820)[(f)] Full open = 905 psia (830)[(f)]
Atmospheric Dump Valves		2	5%[(a)]	Start open = 8°F (T_{avg}-532°F) Full open = 30°F (T_{avg}-532°F) Close = 3°F (T_{avg}-532°F)
Turbine Control Valves		4	---	Closure time ≤ 2 sec[(b)]
Turbine Stop Valves		4	---	Closure time 0.15 sec
Main Steam Isolation Valves (MSIVs)		2	---	Closure time 6 sec
Pressurizer Safety Valves (PSVs)		2	---	Full Open = 2575 psia and 2600 psia[(h)]
Main Steam Safety Valves (MSSVs)			---	Full Open (psia)
		4		1029.25
		4		1049.7
		8		1064.7
High Pressure Safety Injection (HPSI) Pumps		3	Variable	1195 psia[(e)]
CVCS Charging Pumps		3	Variable	---
AFW Pumps				
Unit 1	Steam-driven	2	Variable	Total Developed Head = 2490'
	Motor-driven	1	Variable	Total Developed Head = 2490'
Unit 2	Steam-driven	2	Variable	Total Developed Head = 2490'
	Motor-driven	1	Variable	Total Developed Head = 2490'

(a) % of HFP steam flow.
(b) Each valve is assumed to have a signal delay time of 0.90 seconds in the safety analysis, not included in the closure time.
(c) No longer used.
(d) Deleted.
(e) The RCS pressure at which the initiation of HPSI flow delivery to the RCS is credited, when only one HPSI pump is operating.
(f) A calculation has been performed to evaluate TBV setpoint at 820-830 psia. The setpoint of 895-905 psia is bounding as long as the SG level trip setpoint is nominally at -50.0".
(g) Deleted.
(h) The pressurizer safety valves and the MSSVs are assumed to be fully open at the maximum value allowed. See each event for specific opening setpoint.

TABLE 14.1-3

DECONTAMINATION FACTORS USED IN OFFSITE DOSE CALCULATIONS

RELEASE PATH	DECONTAMINATION FACTORS[a] IODINES	DECONTAMINATION FACTORS[a] OTHER ISOTOPES
Primary leak outside-containment[b]	1	1
Releases through dump valves or safety valves	100[c]	1

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[a] Decontamination factor equal to 10 means 1/10 of the total initial activity in the mass is released to the air.

[b] Certain events show appreciable releases from containment (e.g., CEA ejection). Treatment of these releases is discussed in the appropriate individual sections.

[c] In the event of SG dryout due to blowdown to the environment, the entire SG radionuclide inventory is assumed to be released to the environment.

14.2 CONTROL ELEMENT ASSEMBLY WITHDRAWAL EVENT

14.2.1 IDENTIFICATION OF EVENT AND CAUSE

The CEAs, in a pre-programmed sequence (according to the PDIL), are used to control (dampen) xenon oscillations and rapidly control core power. The Regulating and Shutdown CEA Groups also provide the required negative reactivity for shutdown during DBEs.

The action of the control element assembly withdrawal (CEAW) Prohibit will stop the CEAs from withdrawing under the following conditions and, thus, prevent the CEAs from aggravating the situation.

a. High neutron flux power level pre-trip;
b. High rate of change pre-trip; or,
c. Thermal Margin/Low Pressure pre-trip.

The action of the CEA motion inhibit will prevent any CEA from being raised or lowered under the following conditions.

a. PDIL alarm;
b. CEA Regulating group out of sequence alarm; or,
c. CEA Regulating or Shutdown group deviation alarm.

Consequently, the CEA motion inhibit prevents the groups from being moved outside the pre-programmed sequence and prevents a single CEA from being misaligned. Therefore, only a sequential CEA group withdrawal needs to be addressed.

A CEAW event is defined as any event caused by a single malfunction in the Reactor Regulating System (RRS) or Control Element Drive Mechanism (CEDM) control system that results in a continuous sequential CEA group withdrawal. The CEA position indication systems are programmed to produce no more than a 40% overlap between CEA Regulating groups with Group 1 being withdrawn first and Group 5 last.

The CEAW event has been re-analyzed to support the use of AREVA fuel at Calvert Cliffs. The results of the analysis are presented in Section 14.2.4 and support the transition fuel cycle, as well as, full core implementation of AREVA fuel at Calvert Cliffs.

14.2.2 SEQUENCE OF EVENTS

A CEAW event can approach the DNBR and linear heat generation rate (LHGR) SAFDLs and the RCS Pressure Upset Limit. Initial margins maintained by the LCOs in conjunction with the RPS (VHPT, TM/LP trip, or LPD trip) ensure that these design limits will not be exceeded. Since no pin failures are postulated to occur, the site boundary dose criteria in 10 CFR 50.67 guidelines will not be approached.

AREVA analyzed HFP and HZP conditions, based upon the assertion that the range of initial reactor power levels is bounded by analyzing only full-power cases due to the VHPT setpoint automatically resetting to track the current operating power level, resulting in a proportionately lower setpoint for a part-power case than for a full power case. The NRC asserts that the possible transient variations in the core power distribution may lead to a more limiting DNBR at lower power levels. The response to this concern incorporated an explicit analysis of part-power transients using an operating envelope that bounded Calvert Cliffs. However, the analysis did not provide a Calvert Cliffs specific basis to conclude that changes in the core operating limits are acceptable with respect to the part-power transient. Therefore, a licensing condition is imposed in the Technical

Specifications to restrict certain Core Operating Limits Report limits from being changed without prior NRC review and approval until an NRC-accepted, or Calvert Cliffs specific, basis is developed for analyzing the CEAW event at full power conditions only (Reference 3).

14.2.2.1 Zero Power Case

The zero power case is assumed to initiate at a HZP, critical condition. For events with high reactivity insertion rates, the positive reactivity insertion caused by CEA withdrawal will cause power to increase at an exponential rate. Since the event initiates at zero power with no heat being produced in the fuel, the heat flux will also be zero. If the reactor power goes above 10^{-4}% of rated power and the rate of change of neutron flux is greater than 1.5 decades per minute, a CEAW Prohibit will be initiated. If the rate exceeds 2.6 decades per minute, with the power between 10^{-4} and 15% of rated power, a reactor trip will be initiated. For conservatism, no credit for the High Rate-of-Change of Power Trip is taken in the analysis presented, i.e., for an event that initiates from a critical condition. However, the high rate-of-change of power trip is credited as justification for not analyzing subcritical CEAW events.

By the time core power reaches 1% of rated power, the neutron flux will be increasing exponentially at an extremely high rate. Although the reactor trip occurs at the minimum setting on the VHPT (30%; analysis assumes 36.4%), the core power could peak above 100% power depending on the worth of the CEAs. The core power peaks after trip due to the RPS electronic and the CEA holding coil delays, and the time necessary to insert enough SCRAM reactivity to offset the positive reactivity insertion. As core power increases, the fuel temperature will increase and result in Doppler feedback, which will reduce the peak core power. Due to the fuel time constant, the core heat flux will lag the core power and consequently result in a lower peak. The core power will rapidly decrease as the CEAs are inserted, thus terminating the power excursion.

The core average temperature will slowly increase as it follows the increase in core heat flux. Due to the loop cycle time, the core inlet temperature will lag the average temperature. With the exit temperature increasing faster than the inlet temperature, more moderator feedback will occur in the top portion of the core. Since the MTC is generally negative, more negative reactivity will be inserted in the top portion of the core resulting in the power peak going to the bottom of the core. For conservatism in the analyses, a positive MTC is used thereby adding positive reactivity.

During the event, the RCS pressure will increase and follow the core average temperature rise. Depending on the CEA worth, the temperature rise could increase the pressurizer pressure above the power-operated relief valve (PORV) setting. The action of the pressurizer pressure and level control systems will moderate the pressure peak. For peak pressure consideration, no credit is allowed for these systems.

The PORVs act to decrease primary pressure, resulting in more adverse DNBR consequences. The peak primary system pressure is not explicitly calculated. It is expected to be benign due to the MSSVs being available in Mode 3 and higher, and in Modes 1 and 2, the secondary system is available for heat removal until after reactor trip.

The SG temperature and pressure will increase as the core temperature increases. Upon the reactor trip, the atmospheric dump and steam bypass valves

will normally modulate the core average temperature and SG pressure to 532°F and below 900 psia, respectively. With the quick opening of the valves, the core inlet temperature will initially decrease and then follow core average temperature.

14.2.2.2 Full Power Case

The full power case is initiated at 100% of rated power and at the LCOs. As the CEAs are withdrawn at the preprogrammed rate, the core power will steadily increase at a rate dependent on the worth of the CEAs. If the CEAs are being withdrawn from a high worth region (i.e., a region in which the CEAs are suppressing the power), the core power will increase at a fast rate. Conversely, if the CEAs were initially in a low worth region, the power will increase at a slow rate.

The withdrawal of CEAs will cause the axial power distribution to shift to the top of the core. The associated increase in the axial peak is compensated by a decrease in the integrated radial peaking factor. The magnitude of the 3-D peak change depends primarily on the initial CEA configuration and the initial axial power distribution.

The withdrawal of CEAs will also cause the neutron flux power measured by the excore detectors to be decalibrated due to rod shadowing. This decalibration of excore detectors, however, is partially compensated for by neutron attenuation due to moderator density changes (temperature shadowing).

As the core power increases, the fuel temperature will increase and result in negative Doppler reactivity feedback. The core average heat flux will slowly increase and lag the core power at an increment dependent on the clad-fuel gap conductance. With the heat flux increasing, the core average temperature will increase. With the core average temperature increasing, the moderator feedback will increase or decrease the rate of reactivity addition depending on whether the MTC is positive or negative. As a result of the increase in core average temperature, the RCS pressure will increase. If the CEAs are fully withdrawn before any trip is reached, a new steady-state at a higher core power and core average temperature will result. With the turbine still demanding 100% of rated power, the atmospheric dump and bypass systems will pick up the additional power (load).

Assuming a large enough withdrawn CEA worth, a trip will occur on either the Variable High Power, Axial Flux Offset, TM/LP, or High Pressurizer Pressure Trips. The amount of withdrawn CEA worth to cause a trip depends on the MTC, Doppler coefficient, and the position of the CEAs.

During a CEAW with the fuel and the RCS heating up, the MTC (usually negative during power operation) and the Doppler coefficient (always negative during power operation) will offset part of the withdrawn CEA worth. With the RCS temperature increasing, the pressurizer pressure and the level will increase. Although no credit is taken in the analysis, the pressurizer sprays will partially suppress the pressure increase and the level control system will maintain the programmed level. For cases where the pressurizer pressure exceeds 2400 psia, a reactor trip will be initiated and the PORVs will open, thereby reducing the number of times the PSVs are actuated. For peak pressure consideration, the PORVs are assumed to be inoperable as they are non-safety grade. In addition, no credit is allowed for the action of the atmospheric dump and turbine bypass valves which would normally maintain the SG below 900 psia and regulate the average RCS temperature at 532°F.

When addressing the fuel DNB SAFDLs, the pressurizer sprays and PORVs are assumed operable to minimize system pressure. These systems act to maximize the margin required to account for transient shifts, which are necessary due to lack of dynamic compensation in the TM/LP trip. Transient shifts account for changes in monitored parameters that occur between the time a TM/LP trip pressure is sensed and the time of MDNBR.

14.2.3 CORE AND SYSTEM PERFORMANCE

14.2.3.1 Mathematical Models

The transient response of the RCS and steam systems to the CEAW event was simulated using the S-RELAP5 thermal-hydraulic system code, described in Section 14.1.4.1, consistent with the methodology in Reference 2. The XCOBRA-IIIC fuel assembly thermal-hydraulic code, described in Section 14.1.4.1, was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using an NRC-approved AREVA DNB correlation. The overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation. The limiting design axial power profile (a top peaked axial power distribution) was used for this simulation.

14.2.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.2-1. Those parameters that are unique to the analysis are discussed below.

For DNB cases, reactivity parameters were chosen such that a new steady-state power level was reached at the VHPT setpoint to maximize DNBR degradation. For FCM cases, reactivity parameters were set to ensure the greatest power excursion and maximum peak FCM.

The key parameters for the CEAW event initiated from both HFP and HZP for determining minimum DNBR are the reactivity insertion rate due to rod motion, the MTC and the FTC. The maximum CEAW rate is calculated by combining the maximum CEA differential worth ($\Delta\rho$/inch) and the maximum CEAW speed of 30 in/min. The minimum transient DNBR is calculated using the most adverse DNBR initial conditions. The analysis conservatively assumed a positive MTC value since this, in combination with increasing coolant temperatures, inserts a positive reactivity and thus maximizes the power and heat flux transients. The analysis also assumed a BOC FTC with uncertainty. This, in combination with increasing fuel temperatures, inserts the least amount of negative reactivity due to Doppler feedback. This minimizes the transient minimum DNBR.

For HZP conditions, the scram worth of the CEAs is set to the Technical Specification minimum shutdown margin, which is more limiting than assuming that the highest-worth CEA is stuck in the fully withdrawn position.

The initial power level used for the analysis is the lowest following an extended shutdown (assumed to be 10^{-9} times the rated power). The analysis assumes that the event is preceded by an extended shutdown because the extremely low neutron population under such a condition delays the power increase as the CEAs are withdrawn until a significant amount of positive reactivity has been added, which maximizes the subsequent power excursion. The combination of the highest reactivity addition rate and the lowest initial power level produces the

highest peak values of the fuel rod surface heat flux and centerline temperature, which result in limiting DNB and FCM values.

The VHPT setpoint is further decalibrated by a factor that accounts for changes in the peripheral power that may occur as CEAs are withdrawn from the interior of the core.

For HFP conditions, a spectrum of positive insertion rates is analyzed from very slow to fast, limited only by bank worth and maximum drive speed. Two reactivity feedback matrices of cases are evaluated: for most-positive reactivity feedback (most-positive MTC and least-negative Doppler coefficient) and the other for most-negative feedback (most-negative MTC and most-negative Doppler coefficient).

For both matrices of reactivity feedback cases, reactivity insertion rate ranges bounding the respective lowest MDNBR point and the maximum value for CEA bank withdrawal are considered. The lower bound of the reactivity insertion rate range analyzed is also considered to be bounding of a reasonable minimum reactivity insertion rate for bounding Mode 1 Boron Dilution.

Protection against violation of the SAFDLs is provided by the VHPT or the TM/LP trip in the analysis.

14.2.3.3 Results

Table 14.2-2 contains the sequence of events for the zero power case for the maximum withdrawal rate. Figures 14.2-1 through 14.2-4 present the transient behavior of the core power, core average heat flux, RCS temperatures, and RCS pressure as a function of time. Also, the analysis revealed that the fuel centerline temperatures are well below those corresponding to the acceptable FCM limit provided in the Technical Specifications.

Table 14.2-3 contains the sequence of events for the full power case for the limiting withdrawal case with respect to DNBR SAFDL. Figures 14.2-5 to 14.2-8 present the transient behavior of the core power, core average heat flux, RCS temperatures, and RCS pressure as a function of time for this case. The limiting case is a CEAW from EOC HFP conditions with a withdrawal rate of 7.55 pcm/second. The analysis also concluded that the fuel centerline temperatures are well below those corresponding to the acceptable FCM limit.

The S-RELAP5 plant simulation results from the analysis of the CEAW event were used as input into the MDNBR calculations. The S-RELAP5 plant simulation was adjusted to account for power uncertainty. The temperature, pressure, and flow measurement uncertainties are accounted for in the MDNBR calculations. The MDNBR was above the high thermal performance DNB correlation upper 95/95 limit plus a 2% mixed core penalty for both the zero and full power cases.

The FCM calculation for the HZP case results in a fuel centerline temperature that is significantly less than the melt temperature provided in the Technical Specifications. The HFP case results in LHGR less than the LHGR FCM safety limit.

The CEAW event is not limiting with respect to peak RCS pressure. With no loss of secondary load or feedwater and no loss of offsite power, a reactor trip on VHPT or high pressurizer pressure, along with primary safety valve capacity, is sufficient to maintain peak RCS pressure well below the over pressurization limit.

Other events, such as Loss of Load, Loss of Normal Feedwater and Feedline Break, all exceed this event with respect to peak RCS pressure.

The radiological consequences of opening the atmospheric dump valve during the most adverse CEAW event is less adverse than the LOAC event.

14.2.4 CONCLUSIONS

The analysis of the CEAW event demonstrates that the initial margin maintained by the LCOs in conjunction with the action of the RPS prevents exceeding the fuel SAFDLs and the RCS Pressure Upset Limit during an uncontrolled CEAW transient. The radiological consequences of opening the atmospheric dump valve upon reactor trip during the most limiting CEAW event is a site boundary dose, which is negligible compared to the 10 CFR 50.67 guidelines.

Since the DNBR and centerline temperature melt (CTM) design limits are not exceeded for this event and no fuel pins are predicted to fail, it is concluded that extended burnup has no adverse impact during this event.

14.2.5 REFERENCES

1. Deleted
2. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004
3. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

TABLE 14.2-1

INITIAL CONDITIONS AND INPUT PARAMETERS - CEAW EVENT

HZP

PARAMETER	UNITS	HZP VALUE
Initial Core Power	MWt	2.737×10^{-6}
Initial Core Inlet Temperature	°F	532
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
Combined Bank Differential Worth	pcm/inch	32.0
Bank Withdrawal Rate	inches/minute	30
Scram Worth	pcm	3500
VHP Trip Setpoint	%RTP	36.4
VHP Trip Delay	sec	0.4
MTC	pcm/°F	+7.0
Maximum Predicted FQ for HZP CEA Withdrawal	---	3.688
Rod Shadowing Power Decalibration	---	0.677

HFP

PARAMETER	UNITS	HFP VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
Combined Bank Differential Worth	pcm/sec	0.0002 to 5.0 for positive feedback 3.00 to 8.00 for negative feedback
Scram Worth	pcm	5277.6
VHP Trip Setpoint	%RTP	110.33
VHP Trip Delay	sec	0.9
MTC	pcm/°F	+7.0 for positive feedback -33 for negative feedback
Doppler Temperature Coefficient	pcm/°F	-0.80 for positive feedback -1.85 for negative feedback

TABLE 14.2-2

SEQUENCE OF EVENTS FOR ZERO POWER CEAW EVENT

TIME (sec)	EVENT	VALUE
0.0	Bank Withdrawal Begins	16.0 pcm/sec
37.55	Core Power Reaches VHP Trip Setpoint	53.767 %RTP
37.95	Reactor Trip Signal Generated	80.3 %RTP
38.45	Control Rods Released	98.9 %RTP
38.55	Maximum Nuclear Power	99.5 %RTP
40.16	Maximum Heat Flux Power	1198.8 MWt 43.8 %RTP

TABLE 14.2-3

SEQUENCE OF EVENTS FOR FULL POWER CEAW EVENT

TIME (sec)	EVENT	VALUE
0.0	Bank Withdrawal Begins	---
109.06	Trip Setpoint Reached	---
109.73	Maximum Power	111.01 %RTP
109.96	Reactor Trip Signal Generated	TM/LP
110.32	Maximum Heat Flux Power	110.58 %RTP
110.46	Control Rods Released	TM/LP

14.3 BORON DILUTION EVENT

14.3.1 IDENTIFICATION OF EVENT AND CAUSE

The CVCS regulates both the chemistry and the quality of coolant in the RCS. Changing the boron concentration in the RCS is a part of normal plant operation, compensating for long-term reactivity effects such as fuel burnup, xenon buildup and decay, and plant startup. During refueling operations, borated water is supplied from the refueling water tank (RWT).

Boron concentration in the RCS can be decreased either by controlled addition of demineralized water or by using a purification ion exchanger with a deborating resin. During normal operation, concentrated boric acid solution and demineralized water is introduced into the volume control tank in concentrations corresponding to the required concentration for proper plant operation. When the specified amount has been injected, the makeup controller automatically shuts the demineralized water and boric acid control valves. A purification ion exchanger with a deborating resin is normally used for boron removal when the boron concentration in the RCS is low (less than 50 ppm) and the feed and bleed method becomes inefficient.

The CVCS is equipped with the following indications which could inform the operator when a change in boron concentration in the RCS is occurring:

a. Volume control tank level;

b. Makeup controller flow; and,

c. CVCS valve position lineup.

A Boron Dilution event is defined as any event caused by a malfunction or an inadvertent operation of the CVCS that results in a dilution of the active portion of the RCS. The active portion of the RCS is defined as that volume of water that circulates through the core. For example, when in shutdown cooling (SDC), no credit is allowed for the volume of water in the SG and other stagnant portions of the RCS. A dilution of the RCS can be the result of adding borated water, which has a boron concentration that is less than the system boron concentration, or by the removal of boron using a purification ion exchanger with a deborating resin.

14.3.2 SEQUENCE OF EVENTS

The analysis of the Boron Dilution event covers the six modes of operation (Table 14.3-1). The modes of operation are defined in Technical Specification Table 1.1-1. A Boron Dilution event can approach the DNBR and LHGR SAFDLs and the RCS Pressure Upset Limit. In all cases, operator action is required to prevent exceeding these limits by securing the dilution and borating, if necessary, to maintain the required shutdown boron concentration.

The calculated time-to-criticality is dependent upon the critical and shutdown boron concentrations, the RCS coolant mass and the flow rate of the dilution stream. In addition, for the dilution front model, a range of SDC flow rates is required.

Assume the boron concentration is exactly the amount needed to maintain the required Technical Specification shutdown margin. Also assume that under the worst conditions the operator has 30 minutes in the refueling mode and 15 minutes in the other modes of operation from the time of initiation of the event to secure the dilution to prevent losing the minimum shutdown margin. The DNBR and LHGR SAFDLs and the RCS Pressure Upset Limit criteria will be met if the entire shutdown margin is not lost.

14.3.2.1 Power Operation and Startup

An inadvertent Boron Dilution event at power and startup (Modes 1 and 2) can be postulated as a result of various malfunctions of, or inadvertent operation of, the CVCS. The sequence of events starts with the decrease of the boron concentration in the RCS. All three charging pumps are on and adding 150 gpm of unborated (demineralized) water into the RCS. The effect of decreasing the boron concentration is to add positive reactivity. With the reactor initially critical, the core power, heat flux, and RCS temperatures will increase the pressurizer pressure and level. Although no credit is taken for them in the analysis, the pressurizer pressure and level control systems will maintain programmed pressure and level. The combination of the pressurizer sprays and RCS letdown will accommodate these slow increases in pressure and level respectively. The SG temperature and pressure will slowly increase with the increasing average RCS temperature. This is a similar sequence to a slow reactivity addition due to a CEA withdrawal.

The increasing fuel and moderator temperatures will result in a negative reactivity feedback due to the Doppler and the MTC being generally negative. The negative reactivity feedbacks will partially offset the positive reactivity insertion due to the dilution, thus further slowing down the power rise.

If the dilution is not secured, the reactor will be shut down by either the TM/LP or the VHPT. The action of the pressurizer control system will prevent the pressure from exceeding the High Pressurizer Pressure Trip setpoint. Operator action is required to secure the dilution.

14.3.2.2 Hot Standby, Hot Shutdown

For the Hot Standby (Mode 3) and Hot Shutdown (Mode 4) cases, the analysis assumes all three charging pumps are on (as in Section 14.3.2.1 above) and assumes a boron concentration to meet the required shutdown margin, and shutdown to critical boron concentration ratios are as in Table 14.3-1.

Mode 3 assumes RCPs are operating and mix the entire RCS loops.

While in Hot Shutdown the limiting shutdown to critical boron ratio occurs while on SDC. The active volume of the RCS includes only that volume to the top of the Hot Leg plus the SDC system. Rapid mixing cannot be assumed in the reactor vessel head or the SG.

14.3.2.3 Cold Shutdown

Two cases are run for Cold Shutdown (Mode 5): a three-pump case and a two-pump case. The three-pump case uses the same volume as the hot shutdown case. For the two-pump case, Cold Shutdown (Mode 5), the NSSS could be partially drained due to repairs or inspections on the RCS (e.g., RCS pump seal replacement, SG inspection, etc.). Therefore, the analysis assumes an active volume of water which is sufficient to fill the RCS to the bottom of the hot leg plus fill the SDC System. Technical Specifications require at least one train of SDC to be in operation. Since the Technical Specifications only allow 88 gpm when the pressurizer level is below 90” in Mode 5, the analysis uses 100 gpm when the level is at the bottom of the hot leg.

14.3.2.4 Refueling

The analysis for Refueling (Mode 6) uses an active volume in the RCS that is the same as cold shutdown. The refueling boron concentration is defined in terms of a ratio of refueling to critical concentration. Changes in the boron concentration

which occur each cycle, can be easily evaluated by comparing the ratio to the minimum allowable ratio presented in Table 14.3-1.

14.3.3 CORE AND SYSTEM PERFORMANCE

14.3.3.1 Mathematical Models

Since the NSSS response to a Boron Dilution event is basically the same as a slow CEA Withdrawal event, only the time to criticality is calculated. The rate of change of boron concentration as a function of time can be described by the below equation. The boron in the active volume will be uniformly mixed when sufficient flow exists. Instantaneous mixing is assumed when an RCP is operating. Therefore, the time to lose the prescribed shutdown is:

$$\frac{dC}{dt} = \frac{w}{M} Cx \frac{\rho_{chgg}}{\rho_{RCS}}$$

or

$$t_c = \frac{M}{W} x \frac{\rho_{RCS}}{\rho_{chgg}} In \frac{C_o}{C_c}$$

where:

M	=	active volume
C_o	=	initial boron concentration
C_c	=	critical boron concentration
W	=	charging volume flow rate
t_c	=	time to lose a prescribed shutdown margin
ρ_{RCS}	=	density of active volume
ρ_{chgg}	=	density of charging water

The ratio of shutdown boron to critical boron is:

$$\frac{C_{sdm}}{C_{RCS}} = e\left[\frac{t_{sdm}W}{M} \bullet \frac{\rho_{chg}}{\rho_{RCS}}\right]$$

where:

C_{sdm}	=	shutdown boron concentration
C_{RCS}	=	critical boron concentration
t_{sdm}	=	criterion for minimum time to lose prescribed shutdown margin

The dilution front model is used when the RCS flow is much slower than would occur with at least one RCP running. Typical loop transient times of several minutes are associated with these low flow conditions. The assumption of instantaneous mixing is no longer valid in these low flow conditions; however, the dilution flow is assumed uniformly mixed with the RCS coolant in the vessel downcomer and the lower plenum regions prior to reaching the core inlet.

The time for the first dilution front to reach the core is calculated by dividing the RCS mass from the mixing location to the bottom of the core by the shutdown cooling + dilution flow. The time for subsequent front is calculated by dividing the mass of the RCS and shutdown cooling systems by the shutdown cooling + dilution flow. The time to criticality is determined by iteratively tracking the number of dilution fronts.

14.3.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.3-1.

The instantaneous mixing and dilution front models are dependent upon the mass of liquid in the active volume, the dilution stream mass flow rate, and the initial to critical boron ratio. The shutdown margin requirement, inverse boron worth and time in cycle are inherently included in the boron ratio. The dilution front model is also dependent upon the RCS flow rate, which is set equal to the shutdown cooling flow rate.

14.3.3.3 Results

Table 14.3-2 contains the minimum time to lose the prescribed shutdown margin for Modes 3-6 of operation. The results show that the operator has sufficient time to take appropriate action to mitigate the consequences of this event for each operating mode.

For Modes 1 and 2, an inadvertent charging of unborated water at the maximum rate would result in a maximum rate of reactivity addition that is within the range evaluated for a CEA Withdrawal event and is therefore not as limiting.

14.3.4 CONCLUSION

In Modes 1 and 2, the RPS initially mitigates the consequences of a Boron Dilution event; after which the operator has sufficient time to terminate the dilution. In all other modes, sufficient time is provided to allow operator action to mitigate the consequences before shutdown margin is lost.

The worst time in life for this event is at BOC when boron concentration is highest and MTC is least negative. Therefore, increased burnup has no adverse effect on this transient.

14.3.5 NRC ACCEPTANCE LIMIT

The acceptance criteria for this event are that the times between initiation of a boron dilution event and loss of shutdown margin are not less than 15 minutes for Modes 2, 3, 4 (Reference 1), and 5, and 30 minutes for Mode 6 (Reference 2). The SER (Reference 2) also states that the analysis of boron dilution for power operation is acceptable because the operator has adequate time to terminate the boron dilution event due to the TM/LP trip and VHPT.

Standard Review Plan Section 15.4.6 requires plants licensed to these requirements to demonstrate that Control Room operations will have a positive alarm indicating the onset of a boron dilution event. Generic Letter 85-05, dated January 31, 1985 and titled "Inadvertent Boron Dilution Events," documents the NRC determination that the consequences of this event do not warrant backfitting this requirement to plants (such as Calvert Cliffs) that are not currently licensed to the Standard Review Plan for this event.

14.3.6 REFERENCES

1. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr., dated December 12, 1980, Issuance of Amendment No. 48 to Facility Operating License No. DPR-53
2. Letter from S. A. McNeil (NRC) to J. A. Tiernan, dated May 4, 1987, Issuance of Amendment No. 108 to Facility Operating License No. DPR-69

TABLE 14.3-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE BORON DILUTION EVENT

PARAMETER		
Minimum Ratio of Shutdown to Critical Boron Concentration	**Ratio**	**SDM % $\Delta\rho$**
Power Operation (Mode 1 and 2)	---	---
Hot Standby (Mode 3)	1.05	-3.5
Hot Shutdown (Mode 4), RCP Running	1.04	-3.5
Hot Shutdown (Mode 4), SDC	1.17	-3.5
Hot Shutdown (Mode 5), 3 Charging Pumps	1.16	-3.0
Hot Shutdown (Mode 5), 2 Charging Pumps	1.11	-3.0
Refueling (Mode 6), 3 Charging Pumps	1.28	-6.263
Refueling (Mode 6), 2 Charging Pumps	1.18	-6.263
RCS Volume and Charging Flow	**Volume ft^3**	**Method**
Power Operation (Mode 1 and 2)	---	---
Hot Standby (Mode 3)	8861	Instant Mix
Hot Shutdown (Mode 4), RCP Running	8861	Instant Mix
Hot Shutdown (Mode 4), SDC	4513	Dilution Front
Hot Shutdown (Mode 5), 3 Charging Pumps	4513	Dilution Front
Hot Shutdown (Mode 5), 2 Charging Pumps	3657	Dilution Front
Refueling (Mode 6), 3 Charging Pumps	3657	Dilution Front
Refueling (Mode 6), 2 Charging Pumps	3657	Dilution Front

TABLE 14.3-2

RESULTS OF BORON DILUTION EVENT

MODE	TIME TO LOSE PRESCRIBED SHUTDOWN MARGIN (MIN)	CRITERION FOR MINIMUM TIME TO LOSE PRESCRIBED SHUTDOWN MARGIN (MIN)[a]
Hot Standby	16.5	15
Hot Shutdown	16.5	15
Cold Shutdown – three pumps	21.8	15
Cold shutdown – two pumps	20.0	15
Refueling	30.9	30

[a] Assumed time between initiation of event and termination of the dilution by the operator.

14.4 EXCESS LOAD EVENT

14.4.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the turbine control valves (governor valves on Unit 2) is to regulate the steam to the high pressure turbine. These four valves are located just downstream of the main stop valves (throttle valves on Unit 2). When operating at HFP (2737 MWt), these valves can only demand an additional 10% steam flow when full open. To regulate the secondary pressure and to reduce the number of times the MSSVs are actuated, there are two dump valves to the atmosphere and four turbine bypass valves to the condenser with a capacity of 5% and 40% of rated thermal power (RTP), respectively. The atmospheric dump valves are located outside-containment and upstream of the MSSVs. The turbine bypass valves are upstream of the main stop valves and downstream of the MSIVs.

An Excess Load event is defined as any rapid, uncontrolled increase in SG steam flow other than an SLB (Section 14.14). The full opening of the turbine control valves, atmospheric dump valves, or turbine bypass valves during steady-state operation would result in the most limiting Excess Load event.

The most limiting Excess Load event at HFP is an inadvertent opening of the atmospheric dump and turbine bypass valves. The full opening of the atmospheric dump and turbine bypass valves is more limiting than the full opening of the turbine control valve at HFP because the combination of the atmospheric dump and turbine bypass valves has more heat removal capacity than the turbine control valves (i.e., 45% compared to less than 10%, respectively, at HFP).

For the zero power Excess Load event, the most limiting case is a complete opening of the turbine control valves. The turbine control valves have more heat removal capacity than the atmospheric dump and turbine bypass valves at HZP. Thus, the full opening of the turbine control valves causes a larger power rise due to a more severe steam flow mismatch than the full opening of the atmospheric dump and turbine bypass valves. Consequently, the full opening of the turbine control valves will result in the highest power and highest peak LHR.

14.4.2 SEQUENCE OF EVENTS

An Excess Load event can approach the DNBR and LHGR SAFDLs and the RCS Pressure Upset limit. For the most limiting events, the initial margins maintained by the LCOs in conjunction with the action of the VHPT will prevent exceeding these limits. Additional protection is provided by the TM/LP, Rate of Change of Power or Low SG Pressure Trip. Since no fuel pin failures are postulated to occur, the site boundary dose criteria contained in the 10 CFR 50.67 guidelines will not be approached.

14.4.2.1 Zero Power Case

The limiting Excess Load event at zero power is postulated to be initiated by the full opening of the turbine control valves. The result of the increase in steam flow is a power mismatch between the primary and secondary systems.

The immediate response to the additional steam flow demand is a rapid decrease in SG pressure. As the pressure rapidly decreases, the level in the SG initially increases due to void formation. The SG temperature will also rapidly decrease as more heat (steam) is being extracted than is being added. Since the reactor is not producing any heat, the secondary side cools down the primary side. The RCS temperatures will decrease and the pressurizer pressure and level will consequently decrease. With the decreasing level and pressure, the charging pumps and pressurizer heaters will automatically turn on. Since the pressurizer

pressure and level control systems are not safety grade, no credit is allowed for this automatic feature to mitigate the decrease in level and pressure.

In the presence of a negative MTC (depending on the boron concentration, the MTC can be either positive or negative) and a negative FTC (always negative), positive reactivity feedbacks will occur in response to the decreasing coolant and fuel temperatures. These feedbacks cause an increase in core power, and then an increase in core heat flux and slow down the decrease in core temperatures. Without any negative feedbacks to reduce the positive reactivity insertion, the core power will increase at an exponential rate until a VHPT signal initiates a reactor trip.

The HZP Excess Load event can be postulated to be initiated at various conditions, including:

- maximum cooldown rate maximum excess load and maximum MFW, or
- lower cooldown rate with less than maximum excess load, less or no MFW, or a reactivity feedback representative of anytime during the cycle.

The HZP events that present a slower approach to the VHPT may not reach the sensed VHPT setpoint under some conditions. The events that do not reach the sensed VHPT setpoint have other trips available to protect the SAFDLs. These RPS trips include the Low SG Pressure Trip, Low SG Level Trip, or the Rate of Change of Power Trip. The postulated transients initiated at HZP with a less than maximum cooldown or reactivity insertion rate would incur at least one of these trips, which would preclude or mitigate any significant increase in power that would challenge the SAFDLs.

A conservative HZP Excess Load event is analyzed as the licensing basis event regarding the SAFDLs. This event conservatively assumes a maximum excess load of 120% and maximum MFW flow to maximize the RCS cooldown rate and peak power. In addition, a most negative Technical Specifications MTC limit and least negative FTC are used to maximize the positive reactivity insertion during the RCS cooldown. Only the VHPT, including uncertainties, is credited for this event. This trip is delayed by conservatively modeling decalibration of the sensed excore detector power during the cooldown. Also, a very conservative Fq is used at the hot spot along with a thermal conductivity and melt temperature for 8 wt% gadolinia fuel rods to maximize the challenge to the FCM SAFDL. This analysis produces a very conservative simulation of the HZP Excess Load event.

The Excess Load event presented herein at HZP was initiated at the conditions given in Table 14.4-1.

The sequence of events for the HZP case is presented in Table 14.4-2. Figures 14.4-7 to 14.4-12 show the NSSS transient response for core power, core heat flux, RCS temperature, RCS pressure, SG pressures and reactivities.

HZP cases were run to evaluate return-to-power following reactor scram. The reactor returned to power following scram when a MSIV is assumed to fail to close for a minimum shutdown margin of 3500 pcm. Boration from HPSI returns the core to a subcritical condition.

If MFW is assumed to not respond to the increased steam load, then the sensed VHPT setpoint may not be reached. The scenario with minimum MFW reaches the Low SG Pressure Trip setpoint. It is conservatively assumed that the AFAS occurs at event initiation, resulting in an early delivery of AFW flow to both SGs. Closure of the MSIV on one SG causes SG pressure to diverge, which results in AFW Block, diverting AFW flow to the "intact" SG. The water level in the intact SG increases. The analysis of this scenario shows that the operator has greater than ten minutes after event initiation to control AFW and stabilize the plant and to prevent further progression of the event. This minimum MFW case also produces the most limiting post scram return to power of those initiated from the HZP condition.

The HZP Excess Load event also provides input to the mechanical design analysis of cladding strain described in UFSAR Chapter 3.

14.4.2.2 Full Power Case

The HFP Excess Load event is analyzed via an MTC spectrum for purposes of identifying the DNBR and FCM limiting cases. The initial margin maintained by the LCOs in conjunction with the action of the VHPT prevents the DNBR from going below the SAFDL and protects against FCM.

The full power case is initiated at 100% of rated power and at the LCOs. The event is postulated to be initiated by the full opening of the atmospheric dump and bypass valves. The result of the increase in steam flow is a power mismatch between primary system output and secondary system demand.

The immediate NSSS response to the increase in steam flow demand is a rapid decrease in SG pressure and temperature. The rapid decrease in pressure will cause the SG level to initially increase due to void formation. The decrease in SG temperature will cause the RCS temperatures to decrease. The decreasing RCS temperatures will cause the RCS pressure and coolant volume to rapidly decrease. As the pressurizer pressure and level continue to decrease, the charging pumps and pressurizer heaters will automatically turn on and RCS letdown will stop. Since the pressurizer control system is not safety grade, the analysis does not take credit for these automatic features to mitigate the consequences of the event.

The analysis assumes a negative MTC and a negative FTC that will result in a positive reactivity feedback. The additional positive reactivity will cause an increase in core power and then an increase in core heat flux. The increasing core heat flux will slow down the decrease in RCS temperatures.

With a high fuel rod gap conductance, H_{gap}, the core heat flux will closely follow the core power and reach a maximum slightly below the peak core power.

The Excess Load event initiated from RTP is one of the events analyzed to establish input to the statistical setpoint calculation.

Main Feedwater is modeled to follow steam demand until SG isolation occurs. The thermal margin cases (DNBR and FCM) are completed prior to SG isolation. For the return to power (long term) cases, SG pressures drop to the SG isolation setpoint. The limiting cases assume failure of an MSIV to close. The SG connected to the stuck open MSIV depletes to AFAS setpoint, AFW Block due to SG differential pressure occurs. AFW Block results in no AFW flow to the

"ruptured" SG and full flow to the unaffected SG. Operator action after 10 minutes is required to control AFW and stabilize the plant.

The reactor remained subcritical following a scram for all HFP cases except for those assuming failure of an MSIV to close. The DNB and FCM analyses at the time of peak post-scram power did not result in fuel failure. At 600 seconds (10 minutes), the analysis assumes the operator shuts off steam flow to the atmospheric dump valves, which isolates steam flow from the SG with the closed MSIV. The SG with the stuck open MSIV would continue to blowdown until dry, or until operator action successfully isolates that SG.

14.4.3 CORE AND SYSTEM PERFORMANCE

14.4.3.1 Mathematical Models

The transient response of the RCS and steam systems to the Excess Load event was simulated using the S-RELAP5 thermal-hydraulic code consistent with the methodology in Reference 1. The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using an approved AREVA DNB correlation. For the thermal margin calculations, the overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation. The limiting design axial power profile (a top peaked axial power distribution) was used for this simulation for conservatism. These computer codes are described in Section 14.1.4.1.

The Excess Load Event can exhibit a prolonged cooldown after scram has occurred. The prolonged cooldown results in safety systems being activated, which introduces the potential for single failures. The worst single failure is a failure of a MSIV to shut. This failure results in conditions that are generally the same as those encountered during a Steam Line Break; therefore, the Steam Line Break methodology is used to analyze the Excess Load with failure of an MSIV to shut. The methodology is described in Section 5.4 of Reference 1. The Steam Line Break methodology requires an assumption that the worst CEA does not insert. The potential exists to overcome the negative reactivity inserted by the CEAs, resulting in very high power peaking factors in the vicinity of the stuck CEA. The Steam Line Break methodology includes iteration between a neutronics and thermal hydraulics code to converge on the thermal hydraulic and power conditions in the vicinity of the stuck CEA in order to determine post-scram DNB and FCM.

14.4.3.2 Input Parameters and Initial Conditions

The key input parameters and initial conditions assumed in the analysis of the fuel SAFDLs at or near the time of trip, are given in Table 14-4.1.

For the HZP case, it was assumed that the plant is in Mode 2 conditions. A Moderator Density Table, biased to support the HFP most negative MTC limit was used. The excore power signal was credited as input to the VHPT. This results in the closest approach to the DNBR and LHGR SAFDLs. For the HFP case, a spectrum of negative MTCs and a least negative Doppler fuel temperature feedback were analyzed. Turbine operation in automatic and manual modes was assumed.

An FTC corresponding to EOC conditions with an uncertainty causing a least negative FTC was used in the analysis since this FTC causes the least amount of negative reactivity change for mitigating the transient increase in core heat flux.

For short term HZP thermal margin analyses, all three charging pumps are assumed to be operating and injecting a maximum flow of 50 gpm. All letdown control valves are assumed closed. This configuration maximizes the primary side cooldown and increases the positive reactivity insertion.

The Pressurizer Pressure Control System (PPCS) was assumed to be inoperable because this minimizes the RCS pressure during the event and, therefore, reduces the calculated DNBR. All other control systems were assumed to be in manual mode of operation and have no impact on the results of this event.

An analysis was performed to evaluate the return-to-power. The excess load event initiated from HFP conditions resulted in an excessive cooldown of the RCS. An FTC and MTC corresponding to EOC conditions were used since EOC values add the most positive reactivity during the cooldown. The HFP case assuming failure of a MSIV to close with delayed operator response to control AFW, shut the atmospheric dump valves and isolate the SG by shutting the MSIV or turbine control valves (10 minute response, which is past the time of return-to-power) resulted in the limiting scenario. The return-to-power scenario assumed no charging flow and assumed feedwater isolation valve leakage. This scenario resulted in a turnaround in the power decrease (a return-to-power) that is shortly thereafter arrested by boron being injected via HPSI.

The Excess Load event is categorized as an AOO for which the RPS trips and/or sufficient initial steady-state thermal margin, maintained by the LCOs, prevent acceptable limits from being exceeded. The analysis that evaluates the approach to the fuel SAFDLs at or near the time of reactor trip is protected by RPS trips and initial thermal margin. The subcritical margin calculation is protected by RPS trips, initial thermal margin, and ESFAS.

14.4.3.3 Results

Table 14.4-2 contains the sequence of events for the zero power case to determine the approach to the DNBR and LHGR SAFDL. Figures 14.4-7 through 14.4-12 present the transient behavior of the core power, core average heat flux, RCS temperatures, RCS pressure, SG pressure, and various reactivities versus time.

Table 14.4-3 contains the sequence of events for the full power case to determine the approach to the DNBR and LHGR SAFDLs. Figures 14.4-1 through 14.4-6 present the transient behavior of the core power, core average heat flux, RCS temperatures, RCS pressure, SG pressure, and various reactivity versus time.

The S-RELAP5 plant simulation results from the analysis of the Excess Load event were used as input into the MDNBR calculations. The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the MDNBR calculations. The MDNBR was above the NRC-approved DNB correlation 95/95 limit plus a 2% mixed core penalty. The peak LHGR is less than the LHGR FCM safety limit. The fuel centerline temperature for the HZP event is well below that required to melt fuel.

The radiological consequence of stuck open atmospheric dump and turbine bypass valves during an Excess Load event is less adverse than the Loss of Non-

Emergency AC Power event. Since non-emergency AC power is still available in the Excess Load event, steam may be directed to the condenser after 10 minutes for controlled plant cooldown. When this happens, the steam (and any activity in it) is no longer being released directly to the atmosphere through the atmospheric dump valves and MSSVs.

EOC cases at HZP and HFP were also evaluated for long-term subcriticality assuming single active failure of either a HPSI or MSIV. A return-to-power is predicted for cases assuming failure of a MSIV to close. The return-to-power peak power does not result in a power excursion that challenges the fuel SAFDLs.

14.4.4 CONCLUSION

The analysis of the Excess Load event demonstrates that the action of the RPS in conjunction with the initial margins maintained by the LCOs prevents exceeding the fuel SAFDLs. The RCS pressure upset limit is not exceeded since the RCS pressure decreases during the event. The radiological consequence of the stuck open atmospheric dump and turbine bypass valves during the event is a site boundary dose that is negligible compared to the 10 CFR 50.67 guidelines. The MTC is the only key parameter which is adversely impacted by extended burnup. The negative reactivity inserted due to the CEAs and boron injected via the HPSI pumps is sufficient to arrest the return-to-power shortly after turnaround.

14.4.5 REFERENCES

1. EMF-2310(P)(A), Revision 1, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, May 2004

TABLE 14.4-1
INITIAL CONDITIONS AND INPUT PARAMETERS TO DETERMINE APPROACH TO SAFDLs FOR THE EXCESS LOAD EVENT

PARAMETER	UNITS	HFP VALUE	HZP VALUE
Initial Core Power	MWt	2754	2.737×10^{-6}
Initial Core Inlet Temperature	°F	548	532
Initial RCS Pressure	psia	2250	2250
Initial Vessel Flow Rate	gpm	370,000	370,000
Effective MTC	pcm/°F	-5 to -33	-33
Minimum CEA Worth Available at Trip	pcm	5277.6	3500
EOC Kinetics, β_{eff}	---	0.005237	0.005237
ASI for MDNBR (Limiting Design Axial Profile)	---	-0.3	-0.3
Doppler Temperature Coefficient	pcm/°F	-1.1	-1.1
VHP Trip Setpoint	% RTP	112.0	40.0
VHP Trip Delay	sec	0.4	0.4
Temperature Shadowing or Decalibration Factor	%/°F	0.70	0.70
Resistance Temperature Detector Response Time, Hot/Cold	sec	0/12	Not Credited
Atmospheric Dump Valve Capacity (2 Valves)	10^6 lbm/hr	0.2925/valve	0.2925/valve
Turbine Bypass Valve Capacity (4 Valves)	10^6 lbm/hr	1.173/valve	1.173/valve
Maximum Predicted FQ	---	---	9.7

TABLE 14.4-2

SEQUENCE OF EVENTS FOR THE ZERO POWER EXCESS LOAD CONDITIONS TO CALCULATE MAXIMUM LHR

TIME (sec)	EVENT	SETPOINT OR VALUE
0.00	Turbine Admission Valve Opens	120% of Full Power Steam Flow
18.19	VHPT Setpoint Reached	40% of full power
18.25	Maximum Neutron Power	82.4% RTP
18.58	Trip Breakers Open	---
19.07	CEAs Begin to Drop into Core	---
20.17	MDNBR	> DNB Limit
20.17	Maximum Clad Surface Heat Flux	575.8 MWt
21.3	Maximum Fuel Centerline Temperature	< CTM Limit
22.1	CEAs Fully Inserted	---

TABLE 14.4-3

SEQUENCE OF EVENTS FOR APPROACH TO SAFDLs FOR THE FULL POWER EXCESS LOAD EVENT

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Complete Opening of Dump and Bypass Valves at Full Power. Turbine Control Valves move to fully open position.	---
24.79	VHPT Setpoint Reached	112% of full power
25.19	Trip Breakers Open	---
25.6	MDNBR	> DNB Limit
25.67	CEAs Begin to Drop into Core	---
25.7	Peak Neutron Power	121.62% RTP
25.7	Maximum Clad Surface Heat Flux	3285.5 MWt

14.5 LOSS OF LOAD EVENT

14.5.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the turbine stop valves (throttle valves on Unit 2) are to quickly shut off steam flow to the high pressure turbine. There are four valves in parallel off a common header which are located upstream of the turbine control valves (governor valves on Unit 2) and downstream of the MSIVs. The quick closure of the stop valves prevents overspeeding the turbine when there is a turbine trip. A turbine trip can be the result of a reactor trip, loss of electrical load, loss of condenser vacuum, low oil pressure, etc.

A Loss of Load event is defined as any event that results in a reduction in the SGs heat removal capacity through the loss of secondary steam flow. Closure of all MSIVs, turbine stop valves, or turbine control valves will cause a Loss of Load event. Of the three types of valves in the steam lines between the SG and the high pressure turbine, the turbine stop valves have the quickest closure time.

The most limiting Loss of Load event for primary system overpressure is a turbine trip without a concurrent reactor trip or an inadvertent closure of the turbine stop valves at HFP. A turbine trip would result in the closure of the turbine stop valves.

14.5.2 SEQUENCE OF EVENTS

A Loss of Load event can result in an approach to the DNBR and LHGR SAFDLs and the RCS Pressure Upset Limit. The action of the TM/LP, the Variable High Power, or the High Pressurizer Pressure Trip will prevent exceeding these limits. Since no fuel pin failures are postulated to occur, the site boundary dose criteria in the 10 CFR 50.67 guidelines will not be approached.

The most limiting criteria for the Loss of Load event are the RCS and Secondary Pressure Upset Limit of 110% of design. Normally the non-safety grade turbine trip would initiate a reactor trip and lessen the peak pressure. In analyzing this event, no credit is allowed for this trip (Section 7.2.3.8).

A Loss of Load event is initiated at HFP by the termination of steam flow to the turbine. The immediate system response is a rapid increase in SG pressure and temperature with the RCS adding heat and without any steam being extracted from the SG. To maximize the pressure and temperature increase, no credit is allowed for the Steam Dump and Bypass System (SDBS) which would reduce the pressure transient. With the SDBS inoperable, the secondary pressure will rapidly reach the SG safety valve analysis setpoints.

With the inability of the SG to remove the heat from the RCS, the RCS temperature will rapidly begin to increase. The pressurizer pressure and level will increase with the increasing RCS temperature. To maximize RCS pressure no credit is taken for the pressurizer pressure and level control system. Consequently, the pressure will rapidly approach the PORV analysis setpoint and the High Pressurizer Pressure Analysis Trip setpoint. To maximize the peak RCS pressure, no credit is taken for the action of the PORVs.

To maximize secondary peak pressure, credit is taken for the pressurizer pressure control system. This will delay the high pressurizer pressure trip and thus add more energy to the secondary system.

The analysis assumes a positive MTC which will add positive reactivity with the increasing RCS temperature. The MTC is normally negative at HFP. The core power and core

average heat flux will increase and further increase the rate of the primary system pressurization.

A High Pressurizer Pressure Trip will be initiated when the pressure reaches the analysis setpoint. The reactor trip will terminate the core power increase after some delay reflecting RPS response, CEA holding coil, and CEA insertion time intervals. The core power will then rapidly decrease to the decay power level. The core average heat flux will follow the core power but will lag the core power due to the fuel time constant.

After the reactor trip, the pressurizer pressure will continue to increase and approach the PSVs' opening pressure analysis setpoint. This is caused by the above-mentioned delays and the core heat flux lagging behind the core power, which results in additional heat being added to the coolant.

The pressurizer pressure will peak and then start to rapidly decrease once the heat flux decays as the SGs begin removing heat through the MSSVs. Consequently, PSVs that opened will close.

The RCS temperatures will slowly decrease and approach the saturation temperature corresponding to the lowest MSSV pressure analysis setpoint.

14.5.3 CORE AND SYSTEM PERFORMANCE

14.5.3.1 Mathematical Models

The NSSS response to the Loss of Load event was simulated using the S-RELAP5 computer code described in Section 14.1.4.2.

14.5.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis to maximize peak RCS pressure are listed in Table 14.5-1 for the present cycles of Unit 1 and Unit 2. The input parameters and initial conditions used in the analysis to maximize peak secondary pressure are listed in Table 14.5-3. Those parameters, which are unique to the analysis, are discussed below.

The most positive MTC was assumed. This MTC, in conjunction with the increasing coolant temperatures, maximizes the rate of change of heat flux and the pressure at the time of reactor trip. A FTC corresponding to BOC conditions was used in the analysis. This FTC causes the least amount of negative reactivity feedback to mitigate the transient increases in both the core heat flux and the pressure. The uncertainty on the FTC used in the analyses is shown in Table 14.5-1. Sensitivity studies were performed to determine the most limiting set of initial conditions, provided in Tables 14.5-1 (RCS Pressure) and 14.5-3 (Secondary Peak Pressure). Pressurizer pressure and level, SG level, RCS inlet temperature and RCS flow rate were ranged, one parameter at a time. The most limiting set of initial conditions is provided in the Tables. The lower limit on initial RCS pressure is used to maximize the rate of change of pressure, and thus peak pressure, following trips. For the case to maximize RCS peak pressure, the lower limit on T_{in} is assumed, which results in a lower initial second pressure, delays the opening of MSSVs, and maximizes RCS pressure.

14.5.3.3 Results

The Loss of Load event has been analyzed to ensure that the significant pressure increase experienced during the event remains below 110% of design. Table 14.5-2 contains the sequence of events to calculate maximum RCS pressure. Figures 14.5-1 through 14.5-6 present the transient core power, core

average heat flux, RCS pressure, RCS temperature behavior, SG pressure, and pressurizer water volume.

Table 14.5-4 contains the sequence of events for the maximum peak secondary pressure analysis. No additional parameter plots for this case are provided as they are very similar to Figures 14.5-1 through 14.5-6.

The results show the peak RCS pressure and peak secondary pressure remain below 110% of design. Due to the prominent pressure spike combined with the limited power increase, the minimum DNBR and peak LHR will not challenge the SAFDLs. As such, no explicit calculations are included for this event. Cases were analyzed to verify the applicability of the MSSV out-of-service power levels as stated in the Technical Specifications and to determine the peak pressurizer level following a Loss of Load. The radiological consequences of opening the MSSVs during the most adverse Loss of Load event are less adverse than the LOAC event.

14.5.4 CONCLUSIONS

The analysis of the Loss of Load event demonstrates that the action of the High Pressurizer Pressure Trip, PSVs, and MSSVs is sufficient to ensure that the integrity of the RCS and Main Steam System are maintained without any credit for the SDBS and the pressurizer PORVs. The radiological consequence of opening the MSSVs during the event is a site boundary dose which is negligible compared to the 10 CFR 50.67 guidelines.

TABLE 14.5-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM RCS PRESSURE

PARAMETER	UNITS	VALUE
Initial Core Power Level	MWt	2754[b]
Initial Core Inlet Coolant Temperature	°F	546
Vessel Flow Rate	gpm	412,000
Initial PZR Pressure	psia	2164[a]
Initial Pressurizer Liquid Level	---	67.2% span
Initial SG Pressure	psia	N/A
Initial SG Level	%NR	69
MTC	X 10^{-4} $\Delta\rho$/°F	+0.15
Doppler Coefficient	X 10^{-4} $\Delta\rho$/°F	-0.08[e]
Doppler Coefficient Uncertainty	%	N/A
Number of Plugged SG Tubes	%	10
Axial Shape Index	---	N/A
CEA Worth at Trip	% $\Delta\rho$	-5.0
Time to 90% Insertion of SCRAM Rods	sec	3.1
RRS	Operating Mode	Manual
SDBS	Operating Mode	Inoperative
MSSV Opening Pressure	psia	1029.25
Pressurizer Pressure Control System	Operating Mode	Manual
Pressurizer Level Control System	Operating Mode	Manual
Turbine Stop Valve Stroke Time	Sec	0.0[d]

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty on indicated pressurizer pressure.

(b) Value does not include 17 MWt of pump heat.

(c) Deleted.

(d) A faster turbine stop valve stroke time results in higher peak primary pressure.

(e) The Doppler reactivity feedback includes an uncertainty of 10%.

TABLE 14.5-2

SEQUENCE OF EVENTS FOR LOSS OF LOAD EVENT TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Event Initiation	---
5.2, 5.5	MSSVs Open	---
6.17	High Pressurizer Pressure Trip Setpoint	2420 psia
7.55	Peak Reactor Power	102.5% RTP
7.58	CEAs Begin to Insert	---
8.10	PSV RC-200 Opens	---
8.87	PSV RC-201 Opens	---
8.75	Peak RCS Pressure	2706.6 psia
9.2	Peak Secondary Pressure	1094.8 psia
10.8	PSVs RC-201 Closes	---
11.25	Peak Pressurizer Level	75.32% span
11.6	PSVs RC-200 Closes	---
14.55	Peak Reactor Vessel Inlet Temperature	565.9

TABLE 14.5-3

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM SECONDARY PRESSURE

PARAMETER	UNITS	VALUE
Initial Core Power Level	MWt	2754[b]
Initial Core Inlet Coolant Temperature	°F	550
Vessel Flow Rate	gpm	370,000
Initial PZR Pressure	psia	2164[a]
Initial Pressurizer Liquid Level	---	32.2% span
Initial SG Pressure	psia	N/A
MTC	$X\ 10^{-4}\ \Delta\rho/°F$	+0.15
Doppler Coefficient	$X\ 10^{-4}\ \Delta\rho/°F$	-0.08[d]
CEA Worth at Trip	$\%\ \Delta\rho$	-5.0
Number of Plugged SG Tubes	%	0
Time to 90% Insertion of SCRAM Rods	sec	3.1
RRS	Operating Mode	Manual
SDBS	Operating Mode	Inoperative
MSSV Opening Pressure	psia	1029.25
Pressurizer Pressure Control System	Operating Mode	Auto
Pressurizer Level Control System	Operating Mode	Auto
Turbine Stop Valve Stroke Time	sec	0.15-2[c]

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty on indicated pressurizer pressure.

(b) Value does not include 17 MWt of pump heat.

(c) A range of turbine stop valve stroke times between 0.15 seconds and 2 seconds has been analyzed.

(d) The Doppler reactivity feedback includes an uncertainty of 10%.

TABLE 14.5-4

SEQUENCE OF EVENTS FOR LOSS OF LOAD EVENT TO MAXIMIZE CALCULATED SECONDARY PEAK PRESSURE

TIME(sec)	EVENT	SETPOINT OF VALUE
0.0	Event Initiation	---
8.2	SG Safety Valves Begin to Open	1029.35 psia
15.4	High Pressurizer Pressure Trip Setpoint	2420 psia
16.2	Peak Reactor Power	102.8% RTP
16.3	Trip Breakers Open	Setpoint + 0.9 sec
16.8	CEA Insertion Begins	Breakers open + 0.5 sec
18.6	Peak RCS Pressure[a]	2517.2 psia
19.8	Peak Pressurizer Level	44.1% span
22.6	Peak Secondary Pressure[a]	1101.8 psia
24.6	Peak Reactor Vessel Inlet Temperature	568.8

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[a] Peak Pressure includes elevation head.

14.6 LOSS OF FEEDWATER FLOW EVENT

14.6.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the feedwater regulating system is to maintain the liquid inventory in the SG at the normal operating level. The feedwater regulating valve controller, which is part of the three element control system (Chapter 10), automatically adjusts the position of the regulating valves to modulate the feedwater flow. During normal operation the controller matches the feedwater flow to the steam flow.

The feedwater flow trains to the SG consist of three electric motor-driven condensate pumps in parallel (usually one in standby), which take suction on the three condensers and discharge to the three parallel electric motor-driven condensate booster pumps (usually one in standby). These pumps deliver the condensate to the two parallel, steam turbine-driven SG feed pumps (Chapter 10).

Check valves in the condensate and feedwater system reduce the likelihood of a total LOFW and blowdown of both SGs due to a pipe break in the condensate or feedwater systems.

An LOFW Flow event is defined as a reduction in feedwater flow to the SG without a corresponding reduction in steam flow from the SG. The closure of the feedwater regulating valves, the loss of condensate or feedwater pumps, or a pipe break in the condensate or feedwater systems during steady-state operation would result in a LOFW Flow event.

The most limiting LOFW Flow event at HFP is an inadvertent closure of both feedwater regulating valves. An instantaneous closure of the regulating valves would cause the largest steam and feedwater flow mismatch and result in the most rapid reduction in the SG inventory.

The LOFW event has been analyzed with respect to RCS peak pressure, Main Steam System (MSS) peak pressure, and maximum SG inventory depletion. As the LOFW event results in increasing pressure and only a small reactor power increase, both the DNB and peak linear heat rate SAFDLs are bounded by more limiting events. The SG inventory depletion case is analyzed to ensure that the AFW System has sufficient capacity to provide its long-term heat removal function.

14.6.2 SEQUENCE OF EVENTS

An LOFW Flow event can result in the approach to the RCS and MSS Pressure Upset Limits (110% of design pressure). The action of either the high pressurizer pressure or low SG level reactor trip functions will provide event mitigation and prevent exceeding these limits. Since the SAFDLs are bounded by other events, no fuel pin failures are assumed to occur, and therefore, the site boundary dose limits specified in 10 CFR 50.67 are not approached.

The event is initiated at 102% of 2700 MW and at the RCS temperature and pressure Technical Specification limits. The event is postulated to occur due to a failure in the feedwater regulating system that instantaneously shuts both feedwater regulating valves. Closure of both valves results in a complete cessation of feedwater flow to both SGs.

The LOFW event relies on AFAS to provide AFW System actuation and maintain acceptable SG liquid inventories. The AFAS actuates based on wide range SG level indication. This signal is used to actuate AFW flow and isolate SG blowdown. The AFW System (Chapter 10) consists of two non-condensing, steam turbine-driven AFW pumps

and one motor-driven AFW pump. Steam for the turbine-driven pumps can be supplied from the MSS or from the auxiliary boiler steam system.

The immediate system response is a steady decrease in SG liquid inventory. The temperature in the SG will increase due to the loss of subcooled feedwater flow and cause the SG pressure to increase correspondingly until the MSSV opening setpoint is reached. The RCS temperature and pressure will increase due to mismatch in primary-to-secondary heat transfer.

With the SG liquid inventory depleting and RCS pressure rising, a reactor trip will occur on either high pressurizer pressure or low SG water level. Both analysis setpoints protect the Technical Specification values and include uncertainty.

For the analyses that maximize RCS and SG pressures, pressures will continue to increase until the point at which the PSV and MSSV setpoints are reached and the valves lift to relieve pressure. For these analyses, neither the turbine bypass valves (TBVs) nor ADVs are credited, as this would relieve pressure. The impact on MSS overpressure of a reactor trip near concurrent with the first MSSV opening was investigated. Staggered MSSV setpoints compensate for tripping at a potentially higher secondary pressure. For the analysis that maximizes SG inventory depletion, the TBVs and ADVs are evaluated as this would increase the SG steam flow and could result in a more rapid SG liquid inventory depletion. The pressurizer PORVs are not credited in either the overpressure or SG inventory depletion cases.

The reactor trip rapidly decreases the core heat flux to decay heat levels. A turbine trip is assumed to occur on reactor trip. The analyses show that neither the RCS nor the MSS upset pressure limits are exceeded.

For the maximum inventory depletion case, the SG water level continues to decrease following the trip due to decay heat. The turbine and RCPs continue to run. The RCPs add heat. An AFAS is generated once the analysis setpoint is reached. The initiation of AFAS causes the AFW pumps to start and the SG blowdown isolation valves to close. The worst single-failure is assumed to occur in the AFW System resulting in the failure of the motor-driven AFW pump to deliver flow.

Reactor Coolant System temperature and pressure will initially decrease after the trip. After the initial decrease, RCS temperature and pressure will begin to slowly rise due to the inability of the SG to adequately remove decay and RCP heat. Steam generator pressure will initially stabilize just above the setpoint of the lowest credited pressure relief device. At ten minutes post-trip, operators are credited with an action to increase AFW flow provided by the turbine-driven AFW pump. Once the SG level has decreased to the point that heat transfer is significantly degraded, SG pressure begins decreasing. The decrease in SG pressure will cause the RCS pressure and temperature to begin to decrease. If sufficient AFW flow was not supplied soon after, RCS pressure and temperature would again increase due to the lack of heat transfer. Once the AFW flow is sufficient to exceed the decay and RCP heat removal demand, SG pressure and level increases and RCS temperature and pressure stabilizes.

14.6.3 CORE AND SYSTEM PERFORMANCE

14.6.3.1 Mathematical Models

The NSSS response to the LOFW event is simulated using the S-RELAP5 computer code described in Section 14.1.4.2.

14.6.3.2 Input Parameters and Initial Conditions

Tables 14.6-1 and 14.6-2 present the limiting input parameters and initial conditions used to analyze the LOFW event to maximize peak RCS pressure, peak secondary pressure, and SG inventory depletion.

Sensitivity studies were performed to determine the most limiting set of initial conditions, provided in Tables 14.6-1 (RCS Pressure) and 14.6-2 (SG Inventory). Pressurizer pressure and level, SG level, RCS inlet temperature, and RCS flow rate were ranged, one parameter at a time.

14.6.3.3 Results

The SAFDLs are not explicitly determined for this event, since both SAFDLs criteria are bounded by other more limiting events. The LOFW event is an increasing pressure event and, therefore, the DNBR SAFDL is not limiting for this event. The LOFW event also results in small power increases, and therefore the linear heat rate limit is not challenged. Additional cases demonstrate that the pressurizer does not overfill.

a. To Maximize RCS Peak Pressure

   The sequence of events for the LOFW Flow event analyzed to determine the peak RCS pressure is presented in Table 14.6-3. Figures 14.6-1 through 14.6-4 present the transient behavior of core power, RCS temperatures, RCS pressure, and SG pressure. Note that the SG pressure profile, Figure 14.6-4, does not include the liquid downcomer head. The worst case peak RCS analysis included a LOAC assumed to occur at the time of reactor trip. The maximum RCS pressure, including elevation head, was determined to be within 110% of the design RCS pressure (2750 psia), and is therefore acceptable.

b. To Maximize Steam Generator Pressure

   The sequence of events for the LOFW Flow event analyzed to determine the peak SG pressure is presented in Table 14.6-4. Figures 14.6-5 through 14.6-8 present the transient behavior of core power, RCS temperatures, RCS pressure, and SG pressure. Note that the SG pressure profile, Figure 14.6-8, does not include the liquid downcomer head. The worst case peak SG pressure analysis was without a LOAC. The maximum SG pressure, including liquid downcomer head, was determined to be within 110% of the design MSS pressure (1116.5 psia), and is therefore acceptable.

c. To Maximize Steam Generator Inventory Depletion

   The sequence of events for the LOFW Flow event analyzed to determine the minimum SG inventory is presented in Table 14.6-5. Figures 14.6-9 through 14.6-13 present the transient behavior of core power, RCS inlet temperature, RCS pressure, SG pressure, and SG inventory. Note that the SG pressure profile, Figure 14.6-12, does not include the liquid downcomer head. The worst case peak SG inventory depletion analysis was without a LOAC.

   As shown on the figures, RCS temperature, RCS pressure, and SG pressure initially peak before the trip. Due to the reactor trip, the RCS parameters rapidly decrease. After the initial decrease, RCS parameters begin to rise again due to the primary-to-secondary heat transfer mismatch.

Steam generator pressure initially remains steady and then begins to drop due to the decreasing SG level and decreased SG heat transfer. Reactor Coolant System temperature and pressure peak at the values indicated in Table 14.6-5. As the SG inventory reaches its minimum level, AFW flow becomes sufficient to remove RCP and decay heat, following an assumed operator action to increase the turbine-driven AFW flow at ten minutes. Steam Generator inventory begins increasing and the RCS parameters begin to stabilize. Within one hour of event initiation, plant parameters have stabilized. Therefore, the actions of the RPS, AFAS, and AFW Systems are adequate to ensure long-term heat removal from the RCS, and ultimately recover the depleting steam generator inventory.

Operation of the ADVs and TBVs was found to be less limiting without the additional steam release. Additionally, MSSV setpoints at the low end of the allowable range were also found to be less limiting than those at the high end.

14.6.4 CONCLUSIONS

The analysis of the LOFW event demonstrates that the actions of the RPS, AFAS, and AFW Systems prevent exceeding the fuel SAFDLs, and the RCS and SG Pressure Upset Limits. The radiological consequence of opening the MSSVs during the event is a site boundary dose that is negligible compared to the 10 CFR 50.67 guidelines.

The LOFW Flow event is a heatup transient and is most limiting at BOC. Therefore, extended burnup has no adverse impact on this event.

TABLE 14.6-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED PEAK PRESSURE

PARAMETER	UNITS	PEAK RCS PRESSURE	PEAK SECONDARY PRESSURE
Initial Core Power Level	MWt	2754[a]	2754[a]
Initial Core Inlet Coolant Temperature	°F	546	550
RCS Flow Rate	gpm	412,000	370,000
Initial Pressurizer Pressure	psia	2164	2164
Initial Pressurizer Liquid Level	% span	67.2	32.2
Pressurizer Spray	---	no	yes
Plugged SG Tubes	%	10	0
MTC	x 10^{-4} $\Delta\rho$/°F	+0.15	+0.15
CEA Worth at Trip	%$\Delta\rho$	-5.0	-5.0
High Pressurizer Pressure Analysis Trip Setpoint	psia	2420.0	2420.0
Low SG Water Water Level below normal	inches	N/A	116.4
Trip Delay Time	sec	0.9	0.9
LOAC at time of trip	---	yes	no
PSV Setpoint – RC-200	psia	2575	2575
PSV Setpoint – RC-201	psia	2600	2600
MSSV Bank 1 Setpoint (2 per SG)	psia	1019.7[b]	1019.7
MSSV Bank 2 Setpoint (2 per SG)	psia	1049.7	1049.7
MSSV Bank 3 Setpoint (4 per SG)	psia	1064.7	1064.7

[a] Value does not include approximately 17 MWt of pump heat calculated by S-RELAP5.

TABLE 14.6-2

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE SG INVENTORY DEPLETION

PARAMETER	UNITS	SETPOINT OR VALUE
Initial Core Power Level	MWt	2754[a]
Initial RCS Inlet Coolant Temperature	°F	550
Vessel Flow Rate	gpm	412,000
Initial Pressurizer Pressure	psia	2311
Initial Pressurizer Liquid Level	% span	67.2
Pressurizer Spray	--	Yes
Plugged SG Tubes	%	0
MTC	$\times 10^{-4}\ \Delta\rho/°F$	+0.15
CEA Worth at Trip	$\%\ \Delta\rho$	-5.0
SG Water Level	% NR	69
High Pressurizer Pressure Analysis Trip Setpoint	psia	2420.00
Trip Delay Time	sec	0.9
AFW Actuation Analysis Setpoint	% WR	42.4
Steam-Driven AFW Response Time	sec	180
Steam-Driven AFW Flow Credited with Operator Action (per SG)	gpm	200
SG Blowdown Flow (total from both SG)	lbm/hr	~150,000
SG Blowdown Isolation Response Time	sec	35
ADVs (Begin to open/fully opened/fully closed)	°F	540/557/535
ADVs (Begin to open/fully opened/fully closed)	°F	540/557/535
TBVs (begin to open/fully opened)	psia	895/905

[a] Value does not include approximately 17 MWt of pump heat.

TABLE 14.6-3

SEQUENCE OF EVENTS FOR LOFW EVENT TO MAXIMIZE CALCULATED PEAK RCS PRESSURE

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Event Initiation	---
40.5	High Pressurizer Pressure Trip Setpoint Reached	2420 psia
41.4	Reactor and Turbine Trip	---
41.7	MSSVs Open	---
41.9	CEA Insertion Begins	---
41.9	Peak Reactor Power	101.4% RTP
44.3	PSV RC-200 Opens	---
45.0	Peak RCS Pressure	2658.9 psia
47.8	PSV RC-200 Closes	---

TABLE 14.6-4

SEQUENCE OF EVENTS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED PEAK SECONDARY PRESSURE

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Termination of Feedwater Flow	---
47.2/47.3	MSSVs Open	---
52.3	Peak Reactor Power	101.49% RTP
53.98	Low SG Level Trip Setpoint	0% NR
54.89	Reactor and Turbine Trip	---
55.39	CEAs Insertion Begins	---
56.0	Peak RCS Pressure	2451.3 psia
60.6	Peak MSS Pressure	1115.7 psia
64.75	Peak Reactor Vessel Inlet Temperature	571.19°F

TABLE 14.6-5

SEQUENCE OF EVENTS FOR THE LOFW EVENT TO MAXIMIZE STEAM GENERATOR INVENTORY DEPLETION

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Event Initiation	---
38.58	High Pressurizer Pressure Trip	2420 psia
39.49	Trip Breakers Open	---
40.0	CEAs Insertion Begins	---
55.88	AFAS Setpoint is Reached	42.4% WR
90.88	SG Blowdown Isolation Valves Close	---
239.40	AFW Flow Begins	100 gpm/SG
639.5	Operators take Action to Increase AFW Flow	200 gpm/SG
639.6	Minimum SG-2 Inventory is Reached	490.7 lbm
639.6	Minimum SG-1 Inventory is Reached	490.2 lbm
1171.13	RCS Peak Pressure Occurs (second peak)	2365.3 psia
3013.85	Maximum RCS Inlet Temperature Occurs	582.9°F

14.7 <u>EXCESS FEEDWATER HEAT REMOVAL EVENT</u>

14.7.1 INTRODUCTION

The condensate and feedwater system is designed to provide a means for transferring the condensate from the condenser hotwells to the SGs (while at the same time raising the temperature and pressure) and providing a means for controlling the quantity of feedwater into the SGs (Section 10.2).

Condensate from the three condenser hotwells is pumped first through the two lowest pressure feedwater heating stages (three heaters per stage), and then through two parallel sets of three low pressure feedwater heaters to the SG feed pumps. The feedwater is then pumped through two parallel feedwater heaters to the SGs. Steam generator level is modulated by the feedwater regulating valves, the feedwater regulating bypass valves, and the associated control systems.

An Excess Feedwater Heat Removal event is defined as a reduction in SG feedwater temperature without a corresponding reduction in steam flow from the SGs. This could be caused by the loss of one or more of the feedwater heaters, or due to a feedwater controller malfunction at steady-state power that causes an increase in feedwater flow.

Proposed General Design Criterion 6, Reactor Core Design, requires that the reactor core function without exceeding fuel damage limits under all normal operating conditions and plant transients. This transient, the Excess Feedwater Heat Removal event, was analyzed to ensure the DNB and LHR SAFDLs are not exceeded. The computer models and methods used in this analysis are those described in Section 14.1.4, specifically S-RELAP5 and XCOBRA-IIIC. As discussed in the following sections, during the core and system response to the Excess Feedwater Heat Removal event, the SAFDLs are within the required limits and the proposed General Design Criterion is met.

14.7.2 PHYSICAL DESCRIPTION OF EVENT

The most limiting Excess Feedwater Heat Removal event is postulated to occur at HFP and is caused by the assumed loss of both high pressure feedwater heaters. This is modeled by a reduction in SG feedwater enthalpy. The immediate system response to this malfunction is a decrease in feedwater temperature to the SGs. The cooler water entering the SGs causes the SG temperature and pressure to slowly decrease, and more heat is extracted from the RCS. In response, the RCS temperature and pressure will decrease and cause pressurizer level to decrease.

When there is a negative MTC, a positive reactivity feedback occurs in the core in response to the decreasing core average temperature. This increases core power. The core average heat flux will also increase and partially offset the RCS temperature decrease resulting from the feedwater temperature decrease, and the reactor reaches a new (higher) steady-state power. Although the VHPT is approached, no reactor trip on nuclear instrument power occurs due to the temperature shadowing of the excore detectors. The delta T portion of the VHPT and the TM/LP trip are not credited. The plant remains at the steady-state power until operators manually trip the plant. Table 14.7-2 depicts the sequence of events for the Excess Feedwater Heat Removal event.

An increase in feedwater flow rate to 155% of rated full power flow has also been analyzed. However, the results of the increased feedwater flow transient were bounded by the results of the loss of feedwater heater transient.

14.7.3 METHODOLOGY

The NSSS response to the Excess Feedwater Heat Removal event was simulated using S-RELAP5. The S-RELAP5 results were subsequently used as input to the XCOBRA-IIIC

code to evaluate the DNB response. Fuel centerline melt is bounded by that calculated for an Excess Load event initiated at HFP.

14.7.4 INPUTS AND ASSUMPTIONS

Initial Conditions

Steam and main feedwater flow are initially assumed equal. The remaining initial plant conditions for the Excess Feedwater Heat Removal event were selected to maximize the NSSS cooldown and the core power increase to ensure the SAFDLs are maintained. Key inputs such as power, T_{in}, RCS pressure, core mass flow rate, MTC, and the feedwater enthalpy were selected to achieve these conditions (Table 14.7-1).

Concurrent Events/Single Failures

There are no concurrent events or single failures assumed in the analysis.

Automatic RPS/ESFAS Functions

No RPS actuations occurred. No ESFAS equipment is actuated during this event.

Other Equipment Safety Functions

The pressurizer pressure and level control systems are not credited. Since this is an overcooling event and the RCS/SG pressure upset limits are not approached, the PSVs, PORVs, and MSSVs are not actuated. In addition, the AFW system is not actuated.

Operator Actions

The analysis assumed that operator actions mitigate the event (i.e., manually trip the plant) at 1800 seconds in accordance with applicable plant procedures.

Status of Non-safety Related Control Systems

The steam dump and bypass system is not actuated.

14.7.5 RESULTS

Figures 14.7-1 through 14.7-6 present, as a function of time, the transient core power, core average heat flux, RCS temperatures, RCS pressure, SG pressure, and SG temperature. These results support the determination that the DNB and FCM SAFDLs are not exceeded.

Results of all cases show that this event is bounded by the Excess Load event for all criteria.

14.7.6 CONCLUSIONS

The loss of both high pressure feedwater heaters is the most limiting HFP Excess Feedwater Heat Removal event (i.e., results in higher core power and lower RCS temperature and pressure). The analysis demonstrates that the SAFDLs (DNB and FCM) are not exceeded. Since this is an overcooling event, the RCS pressure upset limit is not approached. In addition, since there are no fuel failures, the radiological consequences of the Excess Feedwater Heat Removal event are negligible.

TABLE 14.7-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE EXCESS FEEDWATER HEAT REMOVAL EVENT

PARAMETER	UNITS	HFP VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
Effective MTC	pcm/°F	-33
EOC Kinetics, β_{eff}	---	0.005237
ASI for MDNBR (Limiting Design Axial Profile)	---	-0.3
Doppler Coefficient	pcm/°F	-1.11
Integrated Radial Peak Factor (F_r)	---	1.65
Maximum Feedwater Temperature Decrease	°F	100.0

TABLE 14.7-2

SEQUENCE OF EVENTS FOR THE EXCESS FEEDWATER HEAT REMOVAL EVENT

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of Both High Pressure Feedwater Heaters	---
160.2	Secondary Pressure Reaches a Minimum Value	815.0 psia
161.4	RCS Pressure Reaches a Minimum Value	2228.5 psia
162.6	Core Power Reaches a Peak Value	3208 MW 117.2% of 2737 MWt
162.6	Minimum DNBR is Reached	> MDNBR SAFDL
163.6-169.4	Core Inlet Temperature Reaches a Minimum Value	540.0°F
167.6	Core Average Heat Flux Reaches a Maximum Value	3205.53 MW 117.1% of 2737 MWt
1800	Operator Action Mitigates the Event	---

14.8 <u>REACTOR COOLANT SYSTEM DEPRESSURIZATION</u>

14.8.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the PSVs is to prevent over-pressurization of the RCS. There are two valves in the system which are located on two parallel pipes off the top of the pressurizer. These valves are spring-loaded and have an opening pressure of 2485 and 2550 psig, respectively. To reduce the number of challenges to the PSVs and to prevent over-pressurization at low system temperature, two PORVs are also installed. These valves tee-off the pipes to the PSVs. Both the PSVs and the PORVs discharge to the quench tank.

An RCS Depressurization event is defined as a rapid, uncontrolled decrease in RCS pressure other than a loss of coolant (Section 14.17). Inadvertent opening the PSVs or PORVs during steady-state operation would result in an RCS Depressurization event.

The most limiting RCS depressurization at HFP is an inadvertent opening of both PORVs. The two PORVs have a larger relieving capacity than one PSV.

If the RCS Depressurization event is not terminated, the event would turn into a small break loss-of-coolant accident (Section 14.17). Therefore, this analysis will only follow the RCS Depressurization event until just after a reactor trip.

14.8.2 SEQUENCE OF EVENTS

An RCS Depressurization event can approach the DNBR SAFDLs. The action of the TM/LP Trip will prevent exceeding the DNBR limit. The LHGR SAFDL and RCS Pressure Upset Limit will not be approached as there is essentially no power rise and no pressure increase for this event. Since no fuel pin failures are postulated to occur, the site boundary dose criteria in the 10 CFR 50.67 guidelines will not be approached.

An RCS Depressurization event is postulated to be initiated at HFP by the inadvertent opening of both PORVs. The immediate system response is a rapid depressurization of the RCS. The level in the pressurizer will initially increase as voids form in response to the decrease in pressure. As the pressure continues to decrease, the level in the pressurizer will decrease due to the steam mass leaving the pressurizer. The discharged steam goes to the quench tank where it is condensed and stored.

To compensate for the decreasing pressure, the water in the pressurizer flashes to steam and the PPCS actuates the proportional heaters in an attempt to maintain pressure. As the pressurizer level decreases, the PLCS will reduce RCS letdown flow to a minimum of 29 gpm and actuate the remaining charging pumps. With the pressure continuing to decrease, all backup heaters will be energized to assist in maintaining pressure. For conservatism in the analysis, no credit is allowed for the pressurizer pressure control system. The pressurizer level control system was not modeled. This has little impact on MDNBR and does not significantly impact the depressurization rate. For both Units at this time, RCS temperatures, core power, core average heat flux, and secondary system pressure will be essentially constant.

With a maximum depressurization rate and the pressurizer pressure and control system inoperable, the RCS pressure will rapidly approach the TM/LP Analysis Trip setpoint. Upon reactor trip, the core power and core average heat flux will rapidly decay. The RCS will approach the saturation temperature corresponding to the normal main steam bypass analysis setpoint pressure of 900 psia.

14.8.3 CORE AND SYSTEM PERFORMANCE

14.8.3.1 Mathematical Models

The transient response of the RCS and steam systems to the RCS Depressurization event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 1. The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly as part of the AREVA TM/LP setpoint verification analysis (Reference 2). The limiting assembly DNBR calculations were performed using a NRC-approved DNB correlation. Both of these computer codes are described in Section 14.1.4.1.

14.8.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.8-1. Those parameters which are unique to the analysis are discussed below.

For this event, the TM/LP Trip is the primary reactor trip. A single calculation is performed at BOC HFP conditions, maximum Technical Specification core inlet temperature, and minimum Technical Specifications RCS flow rate. This produced the minimum margin to the DNB limit. A conservative moderator density reactivity feedback is used, which is based on the HZP Technical Specification MTC.

The event is assumed to be caused by an inadvertent opening of both pressurizer PORVs while operating at RTP. This results in a rapid drop in the RCS pressure and, consequently, a rapid decrease in DNBR.

The initial axial power shape and the corresponding SCRAM worth versus insertion used in the analysis is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.

The pressurizer heaters were assumed inoperable. The charging and letdown system is not modeled. This is due to the event presenting a benign challenge to MDNBR and the event being used to determine the TM/LP pressure bias used in the setpoint verification analysis. The depressurization rate at the time of trip is not significantly impacted by letdown.

14.8.3.3 Results

Table 14.8-2 contains the sequence of events for the event at HFP. Figures 14.8-1 through 14.8-4 present the transient core power, core average heat flux, RCS temperatures, and RCS pressure behavior.

14.8.4 CONCLUSION

The analysis of the RCS Depressurization event demonstrates that the action of the RPS prevents exceeding the fuel SAFDLs. The radiological consequence of opening the atmospheric dump valves upon reactor trip during the most limiting RCS Depressurization event is a site boundary dose which is negligible compared to the 10 CFR 50.67 guidelines.

The key transient parameters for this event are independent of burnup, therefore, extended burnup has no impact on this event.

14.8.5 REFERENCES

1. EMF-2310(P)(A), Revision 1, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, May 2004
2. EMF-1961(P)(A), Statistical Setpoints for Combustion Engineering Type Reactors

TABLE 14.8-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR RCS DEPRESSURIZATION EVENT

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
TM/LP Trip Delay	sec	0.9
Scram Worth	pcm	5277.6
MTC	pcm/°F	+7
Doppler Reactivity Coefficient	pcm/°F	-0.8
Effective Cross-sectional Area of the PORVs	ft^2	0.02008
PLCS	operating condition	Not Modeled
PPCS	operating condition	Heaters Disabled Spray Available
SDBC	operating condition	Not actuated
SG Tube Plugging	% per SG	0

TABLE 14.8-2

SEQUENCE OF EVENTS FOR THE RCS DEPRESSURIZATION EVENT

TIME (sec)	EVENT	VALUE
0.0	Inadvertent Opening of PORVs	---
37.45	TM/LP Analysis Trip Setpoint is Reached	1862.8 psia
38.35	TM/LP Trip Breakers Open	---
38.80	Time of Maximum Heat Flux	2819.9 MW
38.85	CEAs Begin to drop into Core	---
53.85	Transient Evaluation Terminated[a]	---

[a] Since the event trips on TM/LP for all cases, the DNBR calculation is covered in the TM/LP setpoint verification analysis. No event specific DNBR calculation is required.

14.9 LOSS-OF-COOLANT FLOW EVENT

14.9.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the RCPs is to provide forced coolant flow through the core. There are four RCPs in the RCS which are located in the SG cold legs. The RCS is a two-loop, two SG system with four cold legs.

Electrical power for the RCPs is provided from separate busses which are connected to a service transformer. Both Unit 1 and Unit 2 service transformers receive their power from the main switchyard which is arranged in a ring bus configuration. The power from each unit's main generator goes directly to the main switchyard and then comes back into the service transformers.

In the event that the main turbine-generator should trip, electrical power for the RCPs is provided by offsite sources through the ring bus. If a turbine trip occurs on Unit 2 during a period when offsite power sources are unavailable, the four RCPs would be initially energized by the transient output of the turbine-generator. During this period, the high rotational energy of the turbine-generator during initial stages of coast down is used to provide power to the RCPs. This turbine-generator coast down assist feature provides a slower decrease in RCS flow rate than would occur due to the influence of the RCP flywheels alone. Unit 1 does not have this assisted coast down feature and coasts down on the RCP flywheels. Should a failure occur in the service transformer, the affected pumps will coast down with their own flywheel energy. The turbine trip would result in a reactor trip. However, no credit is allowed for this trip in the analysis.

A Loss-of-Coolant Flow event is defined as a loss of forced reactor coolant through the core with offsite power available, but without a seized RCP rotor (Section 14.16). A loss-of-coolant flow with offsite power unavailable is discussed in Section 14.10.

The most limiting Loss-of-Coolant Flow event is a concurrent loss of power to all four RCPs. A loss of four RCPs is more limiting than a loss of one, two, or three RCPs as the coolant through the core will decrease faster with a loss of all four RCPs.

14.9.2 SEQUENCE OF EVENTS

A Loss-of-Coolant Flow event can approach the DNBR and LHGR SAFDLs and the RCS Pressure Upset Limit. The action of the low flow trip in conjunction with the steady-state margin ensured by the LCOs will prevent exceeding these limits. Since no fuel pin failures are postulated to occur, the site boundary dose criteria in 10 CFR 50.67 guidelines will not be approached.

A Loss-of-Coolant Flow event is initiated at HFP and within the LCOs by the concurrent loss of power to all four RCPs. The immediate system response to the coast down of the pumps is a rapid decrease in coolant mass flow rate through the core. In one second, the flow decreases by approximately 11% and in ten seconds, the flow is down to approximately 50% of full flow.

With the coolant mass flow rate decreasing, the core temperatures and the enthalpy rise across the core will start increasing. In the presence of a positive MTC that is normally negative, the increasing core temperatures will result in positive reactivity feedback. The core power will subsequently increase. Due to the fuel time constant and increased enthalpy, the core average heat flux will lag the core power rise.

Depending on the initial core flow measured by the SG differential pressure transmitters, the low flow analysis trip setpoint is reached in approximately one second. After sufficient time for trip signal processing and decay of the CEA holding coils (i.e., 1 second), the

CEAs begin to drop into the core and insert negative reactivity. Consequently, the increase in core power is terminated and starts to decrease. Since the core flow is still decreasing and the heat flux has not decreased, the DNBR will continue to decrease at this time. After the CEAs have been sufficiently inserted (depending on the axial power distribution) and taking into account the lagging heat flux, the DNBR transient will terminate within 3 to 5 seconds of the event.

Shortly after the core heat flux starts to decrease, the RCS temperatures and pressurizer pressure will begin to decrease. Since the loop cycle time is longer than the DNBR transient time (i.e., approximately 10 seconds compared to less than 5 seconds), the SG temperature and pressure remain essentially constant during the initial sequence of events. After the SDBS actuates, the system will stabilize to a HZP condition.

14.9.3 CORE AND SYSTEM PERFORMANCE

14.9.3.1 Mathematical Models

The transient response of the RCS and steam system to the Loss of Coolant Flow event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 1. The S-RELAP5 code is described in Section 14.1.4.1. The time-dependent coolant flow data was developed based on measured data for a zero plugged tube condition. The data was conservatively adjusted for the effect of SG tube plugging using the COAST computer code. The transient results were subsequently used as input for the evaluation of MDNBR using the XCOBRA-IIIC computer code. The code is described in Section 14.1.4.1.

14.9.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.9-1. Those parameters which are unique to the analysis are discussed below.

The MTC and FTC are the only key parameters which are impacted by extended burnup. Since this transient is more limiting at BOC, corresponding MTC and FTC values were assumed in the analysis. Hence, extended burnup has no adverse impact on this event.

The coolant flow coast down calculated by S-RELAP5 was benchmarked to flow coast down data reflecting the effects of 10% SG tube plugging. Reactor trip for the Loss of Coolant Flow event was initiated by low coolant flow rate as determined by a reduction in the sum of the total loop flow. The plant protection system setpoint credited in the analysis was adjusted to account for uncertainties and time delays. The initiation of scram was delayed after the setpoint was reached to account for appropriate instrumentation and other time delays in the safety system including initiation of actual rod movement. The analysis is conducted at HFP BOC conditions, using a limiting axial power distribution. The BOC most-positive MTC Technical Specification value, independent of power level, and the BOC HFP nominal FTC (biased less negative) were used. The minimum transient DNBR is calculated using the most adverse DNBR initial conditions.

14.9.3.3 Results

Table 14.9-2 contains the sequence of events for the Loss-of-Coolant Flow event. Figures 14.9-2 through 14.9-5 present the transient core power, core average heat flux, RCS temperatures, and RCS pressure behavior.

Core boundary conditions from each case are used for the evaluation of the MDNBR, via the AREVA DNB LCO setpoint verification analysis (Reference 2). The MDNBR is above the NRC-approved DNB correlation upper 95/95 limit plus a 2% mixed core penalty.

The results show that the DNBR SAFDL is not exceeded. The peak RCS pressure scenario is bounded by the more limiting Feedline Break event.

14.9.4 CONCLUSION

The analysis of the Loss-of-Coolant Flow event demonstrates that the action of the RPS in conjunction with the LCOs will prevent exceeding the fuel SAFDLs and RCS Pressure Upset Limits. The radiological consequences of opening of the atmospheric dump valves upon reactor trip is a site boundary dose which is negligible compared to the 10 CFR 50.67 guidelines.

Since DNBR design limits are not exceeded and no fuel pins are predicted to fail, extended burnup has no adverse impact during this event.

14.9.5 REFERENCES

1. EMF-2310(P)(A), Revision 1, SRP Chapter Non-LOCA Methodology for Pressurized Water Reactors, May 2004
2. EMF-1961(P)(A), Statistical Setpoints for Combustion Engineering Type Reactors

TABLE 14.9-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR LOSS-OF-COOLANT FLOW EVENT

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
Minimum RCS Flow Trip Setpoint	% of 370,000 gpm	90
RCS Flow Trip Response Time	sec	0.5
Minimum CEA Worth Available at Trip	pcm	5277.6
Doppler Fuel Temperature Coefficient	pcm/°F	-0.8
Effective MTC	pcm/°F	+7
4-Pump RCS Flow Coast Down (Includes effects of 10% plugged tubes)	---	Figure 14.9-1

TABLE 14.9-2

SEQUENCE OF EVENTS FOR LOSS-OF-COOLANT FLOW EVENT

TIME (sec)	EVENT	VALUE
0.0	Loss of Power to all Four RCPs	---
0.91	Low Flow Trip Analysis Setpoint Reached	90% of 370,000 gpm
1.40	Trip Breakers Open	---
1.89	CEAs Begin to Drop Into Core	---
Various[a]	Minimum DNBR Occurs	> 1.164

[a] Less than 5 seconds. Time depends on axial shape.

14.10 LOSS-OF-NON-EMERGENCY AC POWER

14.10.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the AC power on the plant's ring bus is to provide power to the NSSS and the balance of plant electrical loads. Plant AC power goes to emergency and non-emergency AC power loads. Emergency power loads are classified as those loads that are essential to safely shut down the plant and maintain the plant in a safe shutdown condition.

Unit 1 and Unit 2 turbine-generators separately provide AC power to the plant ring bus which supplies AC power to the main grid and each Unit's service transformer. When the turbine-generators are off line, the main grid (i.e., offsite power) supplies power to both of the plant's service transformers. During normal operation, the service transformers supply AC power to all emergency and non-emergency AC loads. When a Unit's turbine-generator is off line during emergency conditions, and neither offsite nor station power is available, the emergency diesel generators supply AC emergency power to all of the plant's vital electrical loads. Non-vital electrical loads, such as RCPs and condensate pumps, are lost when there is a loss of all non-emergency AC power.

A Loss-of-Non-Emergency AC Power (LOAC) event is defined as a loss of the plant's 500 kV/13 kV service transformers. A loss of load to the plant's turbine-generator with offsite and plant power (i.e., other Unit's turbine-generator) unavailable would result in an LOAC event. In the following presentation, a LOOP will mean a loss of the main grid in conjunction with the loss of the other Unit's turbine-generator.

The most limiting LOAC event is a loss of turbine load at HFP with offsite AC power unavailable.

14.10.2 SEQUENCE OF EVENTS

An LOAC event can result in an approach to the DNBR and LHGR SAFDLs and the RCS Pressure Upset Limit. The action of the Low RCS Flow RPS Trip in conjunction with the steady-state margin ensured by the LCOs will prevent exceeding these limits.

The response of the RCS to an LOAC event during the first five seconds is identical to a Loss-of-Coolant Flow event. During this time interval, the secondary system (i.e., SG feedwater side) has not had enough time to affect the RCS due to the loop cycle time. Consequently, the analysis of the Loss of Flow event ensures the fuel SAFDLs will not be exceeded during an LOAC event.

Since the analysis of the Loss of Flow event ensures the fuel SAFDLs will not be exceeded, the analysis of LOAC event presented herein will address the approach to the RCS Pressure Upset Limit and the approach to the site boundary dose criteria in 10 CFR Part 100 guidelines.

An LOAC event is initiated at HFP by loss of turbine-generator load with offsite power unavailable. The immediate system response is similar to a simultaneous Loss of Load event, LOFW event, and Loss-of-Coolant Flow event. The loss of voltage to the 4 kV emergency busses generates an undervoltage signal initiating the starting of the diesel generator. The normal supply busses are automatically separated from the emergency and non-emergency busses. Once the diesel generator is up to speed, the load sequencer starts loading the emergency busses with the vital equipment in a sequential manner to avoid overloading the diesel generator.

With power lost to the RCP motors, the pumps start to coast down immediately. In the analysis, no credit is allowed for the loss of power to the CEA holding coils that would

release the CEAs to shut down the reactor. A low RCS flow signal will be reached in less than one second, which will trip the reactor. In less than five seconds, the DNBR transient will be terminated (Section 14.9).

The SG response to the LOFW and the termination of the main steam flow to the turbine is an increase in SG pressure and temperature. The analysis assumes that the feedwater flow is instantaneously reduced to zero with the loss of the condensate system pumps, and that the steam flow to the turbine is instantaneously reduced to zero with the closure of the turbine stop valves upon initiation of turbine trip.

With the atmospheric steam dump and turbine bypass systems inoperable, the SG pressure will rapidly approach the MSSVs' opening pressures. The MSSVs will open as this is the only path for removal of decay heat (i.e., steam). With reactor power decreasing to decay heat levels, the RCS will continue to transfer heat to the SGs, thereby keeping the main steam safeties open.

Due to the inability of the SGs to remove all of the generated heat, the RCS temperature, and then the pressure, will almost immediately start increasing. For a short period of time after the CEAs are inserted into the core, the RCS pressure will continue to increase and approach the PSVs' analysis setpoints. The pressurizer pressure and level control systems and the pressurizer PORVs would partially mitigate the pressure transient. For the analysis, no credit is allowed for these systems or for the PORVs. The opening of the MSSVs in conjunction with only decay heat being generated will result in the RCS pressure peaking and then decreasing. In one to two minutes, the RCPs will have completely coasted down and the RCS will be in natural circulation.

The SG liquid inventory will slowly deplete due to the steam blowdown through MSSVs and the LOFW to the SG. As the SG liquid inventory decreases and temperature increases, the SG heat transfer capability will be reduced. The reduction in heat transfer to the SG will prevent the RCS from cooling down. The combination of decay heat, natural circulation, and degraded SG heat transfer will then cause the core average temperature to increase.

Once the core decay heat rate equals the SG heat removal rate, the core average temperature will start slowly decreasing. After approximately 200 seconds, the decay heat rate will remain relatively constant. The RCS average temperature will approach the saturation temperature corresponding to the MSSV opening analysis setpoint.

At 600 seconds (10 minutes), the analysis assumes the operator initiates AFW via remote-manual operation from the Control Room. The present AFW System automatically initiates AFW flow in less than 180 seconds following a low SG level signal initiated by the ESFAS during a loss of AC power.

The subcooled AFW decreases the SG temperature and starts to cool down the RCS. At 900 seconds (15 minutes), the analysis assumes the operator, by remote-manual operation of the atmospheric dump valves, initiates plant cooldown.

14.10.3 CORE AND SYSTEM PERFORMANCE

14.10.3.1 Mathematical Models

The NSSS response to the LOAC event was simulated using the CESEC computer code described in Section 14.1.4.1. The site boundary dose was calculated as described in Section 14.1.4.3.

14.10.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.10-1 for the present cycles of Unit 1 and Unit 2. Those parameters which are unique to the analysis are discussed below.

To maximize the power increase with the increasing core temperatures, a positive MTC is assumed.

Of the key input parameters, only primary and secondary coolant activity, in principle, is burnup dependent. However, the analysis assumed conservative values for the primary and secondary coolant activities.

14.10.3.3 Results

Table 14.10-2 contains the sequence of events for the LOAC event at HFP. Figures 14.10-1 through 14.10-5 present the transient behavior of the core power, core average heat flux, RCS temperatures, RCS pressure, and SG pressure.

The sequence of events and NSSS response plots are based on the OSG plant configuration. These trends are representative of the RSG plant configuration.

Table 14.10-3 presents the radiological assumptions and the 0-2 hour site boundary dose for the event. The results show that the thyroid and WBD is 0.04 and 0.0006 REM compared to the 10 CFR Part 100 guidelines of 300 and 25 REM, respectively.

14.10.4 CONCLUSION

The analysis of the LOAC event demonstrates that the action of the RPS and LCOs prevent exceeding the fuel SAFDLs. The action of the low RCS flow trip in conjunction with the MSSVs ensures the integrity of the RCS. The radiological consequences of opening the MSSVs during the event and cooling down through the atmospheric dump valves is a site boundary dose which is negligible in comparison to the 10 CFR Part 100 guidelines.

The Technical Specification limits on primary and secondary activities will bound the effects of extended burnup.

This event is not affected by the transition to AREVA Advanced CE-14 HTP™ fuel because the key parameters for this event are plant related system responses which are unchanged from, or bounded by, the current analysis. These parameters are not adversely affected by either the transition cycle or full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

TABLE 14.10-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR LOSS-OF-NON-EMERGENCY AC POWER EVENT

PARAMETER	UNITS	UNIT 1[a,b]	UNIT 2[a,b]
Initial Core Power Level	MWt	102% of 2700	102% of 2700
Core Inlet Temperature	°F	552	552
Vessel Flow Rate	gpm	370,000	370,000
RCS Pressure	psia	2200	2200
SG Pressure	psia	875	875
Low RCS Flow Trip Response	sec	0.65[d]	0.65[d]
MTC	10^{-4} $\Delta\rho$/°F	+0.5	+0.5
Doppler Coefficient Uncertainty	%	15	15
CEA Worth on Trip	10^{-2} $\Delta\rho$	-5.14	-5.14
AFW System Response	---	Manual[e]	Manual[e]

(a) These values represent inputs to the limiting transient scenario analyzed for each unit. In general, a range of initial conditions and input parameters, including uncertainties, were evaluated to determine the limiting case.

(b) For the RSG configuration, an evaluation was performed to determine the effects of RSGs on the consequences of the event. The evaluation concluded that any impact on the reported dose consequences associated with the RSGs (i.e., changes in liquid volume and metal mass) was more than compensated by the use of a RCS activity of 2.5 μCi/gm versus the Technical Specification limit of 0.5 μCi/gm.

(d) Present Technical Specification limit is 0.50 second.

(e) While the Technical Specification response time for AFAS is ≤ 180 seconds, this analysis assumes manual actuation of AFW after 10 minutes.

TABLE 14.10-2

SEQUENCE OF EVENTS FOR LOSS-OF-NON-EMERGENCY AC POWER EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	Loss of All Non-Emergency AC Power	---
1.00	Low Flow Trip Analysis Setpoint is Reached	93%[a,b]
1.65	Trip Breakers Open	---
2.15	CEAs Begin to Drop Into Core	--
3.8	SG Safety Valves Start to Open	1000 psia
8.5	Maximum SG Pressure	1041 psia
12.0	Maximum RCS Pressure	2493 psia
602.4	AFW Flow Initiated	---
618.5	SG Safety Valves Close	---
900.0	Operator Activates the Remotely-Operated Atmospheric Steam Dump Valves and Initiates Plant Cooldown	---
10,236.0	SDC Initiated	300 psia

[a] Percent of initial 4-RCP Flow.

[b] An evaluation has been performed to determine the effects of the RSG configuration on the consequences of the event. The evaluation found that the current analysis is still bounding.

TABLE 14.10-3

RADIOLOGICAL ASSUMPTIONS AND RESULTS FOR LOSS-OF-NON-EMERGENCY AC POWER EVENT

PARAMETER		UNITS	UNIT 1	UNIT 2
Primary to Secondary Leak Rate		gpm	1[a]	1[a]
RCS Volume		ft^3	10,400	10,400
RCS Maximum allowable concentration (DEQ I-131)		μCi/gm	2.5[b]	2.5[b]
Secondary Maximum allowable concentration (DEQ I-131)		μCi/gm	0.1[a]	0.1[a]
RCS Maximum allowable concentration of Noble Gases (DEQ Xe-133)		μCi/gm	102.5/E[c]	102.5/E[c]
RCS Cooldown Rate		°F/hr	100	100
Steam Required to Drive AFW Pump Turbines		lbs/hr	26,555	26,555
SG Partition Factor		---	0.01	0.01
Atmospheric Dispersion Coefficient		sec/m^3	$1.8x10^{-4}$	$1.8x10^{-4}$[d]
Breathing Rate		m^3/sec	$3.47x10^{-4}$	$3.47x10^{-4}$
Dose Conversion Factor (I-131)		REM/Ci	$1.48x10^6$	$1.48x10^6$
Dose Conversion Factor (Xe-133)		$REM/MeV/m^3$	$1.33x10^{-11}$	$1.33x10^{-11}$
Site Boundary Dose:	Thyroid	REM	0.04	0.04
(0-2 hrs)	Whole Body	REM	0.0006	0.0006

---

[a] Technical Specification Limit.

[b] Present Technical Specification Limit is 0.5 μCi/gm.

[c] Present Technical Specification Limit is 100/E - μCi/gm.

[d] This X/Q is very conservative (actual X/Q = $1.3x10^{-4}$ sec/m^3) and is a compensatory measure for an error in the decay heat model of CESEC.

14.11 CONTROL ELEMENT ASSEMBLY DROP EVENT

14.11.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the CEA is to control the core axial power distribution and to provide instantaneous reactivity to shut down the reactor during controlled procedures and during abnormal and emergency conditions. Under normal operating conditions (i.e., controlled procedures), CEAs are inserted and withdrawn in a pre-programmed group sequence according to the PDIL curve in the Technical Specifications. There are 3 shutdown groups and 5 regulating groups for a total of 77 CEAs. Presently there are no PLCEAs (Chapter 3) in the core. The shutdown groups are dual CEAs (i.e., two CEAs with one extension shaft) and the regulating groups are single CEAs.

The CEAs are withdrawn, inserted, and held by the CEDMs which are located on top of the reactor vessel. The CEDM is a magnetic jack-type drive system which operates at a constant speed. A CEA is released from the CEDM holding coil grippers by removing the power to the holding coil. After approximately half a second, the CEDM holding coil magnetic flux will decay and the weight of the CEA and the CEA extension shaft will cause the CEA to drop into the core.

A CEA Drop event is defined as an uncontrolled insertion of a CEA. A loss of power to the CEDM holding coil or a mechanical fault in the CEDM magnetic jack drive will result in a CEA Drop event.

The most limiting CEA Drop event is an uncontrolled CEA insertion at HFP. Operation at HFP is most limiting as the reactor is then operating closest to the fuel SAFDLs. Appendix C of the Technical Specifications prohibits changing Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 until an NRC accepted generic, or Calvert Cliffs specific, basis is developed for analyzing this event at full power conditions only.

The CEA Drop event represents the limiting event with respect to challenging the LHGR FCM safety limit. The LHGR FCM safety limit that is imposed to compensate for the RODEX2 methodology, which does not explicitly model degraded fuel thermal conductivity, is 21 kW/ft (Reference 3).

14.11.2 SEQUENCE OF EVENTS

A CEA Drop event can approach the DNBR and LHGR SAFDLs. The steady-state margin ensured by the LCOs will prevent exceeding these limits. The RCS Pressure Upset Limit is not approached as the system cools down during the event. Since no fuel pin failures are postulated to occur, the site boundary dose criteria in 10 CFR 50.67 guidelines will not be approached.

A CEA Drop event is initiated at HFP from within the LCOs by a failure of the CEDM holding coil. The immediate system response caused by the inserted negative reactivity worth is a reduction in the local core power in the vicinity of the dropped CEA. The local heat flux will be suppressed and the local moderator temperature will decrease. The azimuthal core power tilt will increase due to shutting down of part of the core. The core power and core average heat flux will correspondingly decrease. The amount of decrease is dependent on the dropped CEA worth.

The reduction in core average temperature, in conjunction with a negative moderator temperature (normally negative), will result in positive moderator feedback. The resulting positive reactivity addition from the moderator feedback partially compensates for Doppler and the dropped CEA negative reactivity.

The RCS pressure will decrease in response to the reduction in the core average temperatures. The analysis assumes the pressurizer pressure and level control systems, which would mitigate the pressure decrease, are inoperable. A decreased RCS pressure results in a lower minimum DNBR.

The decrease in core outlet temperature will cause the SG temperature and pressure to decrease. In the analysis, the turbine load is assumed to remain at full power. To maintain the same load, the turbine control valve is assumed to open further to compensate for the reduction in SG pressure. The result is an additional decrease in core inlet temperature and positive moderator feedback. Normal operating procedures maintain the turbine control valve at a set valve position which would prevent any further decrease in core inlet temperature. The cooldown of the RCS continues until the power mismatch between the RCS and the power demand is eliminated.

The core average temperature continues to decrease until sufficient positive moderator reactivity feedback offsets the Doppler and the dropped CEA negative reactivity and returns the core power to its pre-drop level. Consequently the power mismatch will be eliminated and no further cooldown of the RCS will occur. The RCS coolant temperatures will reach a new equilibrium value that is slightly lower than the initial values.

During the return to the initial core average power level and with part of the core power suppressed, the local power peaks will increase to make up the power difference. The local peaks that will occur are dependent upon the worth and position of the dropped CEA.

After detection of CEA drop/misalignment, the operator will initiate a power reduction as required by Technical Specifications.

14.11.3 CORE AND SYSTEM PERFORMANCE

14.11.3.1 Mathematical Models

The transient response of the reactor coolant and steam systems to the CEA Drop event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 1. The S-RELAP5 results were subsequently used as input for the evaluation of MDNBR and FCM. S-RELAP5 is described in Section 14.1.4.1.

14.11.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.11-1. Those parameters that are unique to the analysis are discussed below.

The analysis assumes the most negative MTC and FTC of reactivity (including uncertainties), because these coefficients produce the minimum RCS coolant temperature decrease upon return to 100% power level and lead to a minimum DNBR.

Charging pumps and pressurizer heaters are assumed to be inoperable during the transient. This maximizes the pressure drop during the event. All other systems are assumed to be in manual mode of operation and have no impact on this event.

The analysis uses the maximum radial peaking distortion factors which, for conservatism, are the ratio of the post-drop to the pre-drop F_r. Calculations were performed to cover a range of dropped CEA worths from 10 pcm to 200 pcm. The

dropped CEA event is usually terminated by a TM/LP trip, a VHPT, or may potentially reach a new equilibrium state without trip.

As seen from the above discussion, the MTC and FTC are the only key parameters which are impacted by extended burnup. The analysis conservatively assumed an EOC MTC value of -3.3×10^{-4} $\Delta\rho/°F$. In addition, the analysis assumed an EOC FTC value with an uncertainty and bias. Hence, the effects of extended burnup have been explicitly and conservatively included in the analysis.

The event was initiated by dropping a full-length CEA over a period of 3.0 seconds. The maximum increases in radial peaking factors in either rodded or unrodded planes were used in all axial regions of the core once the power returns to the initial level. The axial power shape in the hot channel is assumed to remain unchanged, therefore, the increase in the three-dimensional peak is proportional to the maximum increase in radial peaking factor. Since there is no trip assumed, and the secondary side continues to demand 100% power, the peaks will stabilize at these asymptotic values after a few minutes.

14.11.3.3 Results

Table 14.11-2 contains the sequence of events for the CEA Drop event at HFP. Figures 14.11-1 through 14.11-4 present the transient behavior of the core power, core average heat flux, RCS temperatures, and RCS pressure as a function of time for the 200 pcm case.

Core boundary conditions from each case are used for the evaluation of the MDNBR, via the AREVA DNB LCO setpoint verification analysis (Reference 2) and FCM. The resultant peak LHGR is less than the more limiting FCM LHGR limit provided by either Reference 3, or by cycle specific analysis, and the MDNBR is above the NRC-approved DNB correlation upper 95/95 limit plus a 2% mixed core penalty.

14.11.4 CONCLUSION

The analysis of the CEA Drop event demonstrates that operating within the LCOs will prevent exceeding the fuel SAFDLs, maintain the integrity of the RCS, and ensure negligible radiological release to the site boundary compared to 10 CFR 50.67 guidelines.

14.11.5 REFERENCES

1. EMF-2310(P)(A), Revision 1, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, May 2004
2. EMF-1961(P)(A), Statistical Setpoints for Combustion Engineering Type Reactors
3. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 - Amendment RE: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

TABLE 14.11-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR CEA DROP EVENT

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
Effective MTC	pcm/°F	-33
Excore Detector Decalibration Factor	%/°F	0.70
Axial Power Distribution	ASI	-0.20 to +0.20
Maximum F_z	---	1.485
Distortion Factor (Full Power)	F_r post/ F_r pre	1.13

TABLE 14.11-2

SEQUENCE OF EVENTS FOR THE CEA DROP EVENT

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Rod Drop Initiated	---
0.0	Core Heat Flux Reaches Minimum	---
3.0	CEA Fully Dropped	-200 pcm
3.0	Core Power Reaches Minimum	78.02% of RTP
300.0	MDNBR Boundary Conditions: Core Heat Flux Reaches Final Value	100.76% of RTP
	Core Inlet Temperature	542.14
	RCS Pressure	2214.1
	MDNBR[(a)]	> 1.164

(a) Results shown for the maximum dropped CEA worth. Core boundary condition from each dropped CEA case are used for the evaluation of MDNBR, via the setpoint verification analysis.

14.12 ASYMMETRIC STEAM GENERATOR EVENT

14.12.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the SGs is to remove heat from the RCS. Any perturbation within the SGs will affect the RCS due to the close coupling of both systems.

An Asymmetric SG event is defined as any initiator that affects only one of the two SGs. A loss of load, an excess load, a LOFW, or an excess feedwater to only one SG would result in an Asymmetric SG event.

Asymmetric SG events, which are the result of a malfunction of one SG, cause a non-uniform core inlet temperature distribution. The non-uniform core inlet temperature distribution in conjunction with the moderator temperature reactivity feedback produces asymmetric local power peaking in the core.

The most limiting Asymmetric SG event is a loss of load to one SG. An asymmetric loss of load would produce the largest core inlet temperature differential across the core. Based on a negative MTC, the RCS temperature tilt across the core will cause an increase in local core power peaking and an approach to the fuel SAFDLs.

14.12.2 SEQUENCE OF EVENTS

An Asymmetric SG event can approach the DNBR and LHGR SAFDLs. The action of the Asymmetric Steam Generator Protection Trip (ASGPT), and the Low SG Pressure, Low SG Level, TM/LP, and HPTs will prevent exceeding these limits. The primary trip for the most adverse Asymmetric SG event is the ASGPT. The RCS Pressure Upset Limits will not be approached during the event. Since no fuel pin failures are postulated to occur, the site boundary dose criteria in 10 CFR 50.67 guidelines will not be approached.

14.12.2.1 Asymmetric Excess Feedwater

An Asymmetric Excess Feedwater event is initiated at HFP from within the LCOs by a malfunction in one of the feedwater controllers, which instantaneously fully opens the feedwater regulator valve to one SG. The full opening of the feedwater regulator valve causes additional subcooled feedwater to enter the SG which lowers the temperature and pressure. The result is a reduction in the steam flow from the affected SG. The excess feedwater also causes the affected SG cold leg temperature to decrease because additional heat is being extracted.

The analysis assumes the turbine demand remains constant, which causes the unaffected SG to pick up part of the load by further opening the turbine control valve. Present operating practices maintain the turbine control valve flow area constant, thus avoiding the increased demand. The increased steaming rate results in lowering the temperature of the SG and therefore the cold leg temperatures.

The result of the asymmetric decrease in the core inlet temperature is a temperature and power tilt across the core. Since the increased feedwater flow rate only decreases the temperature slightly, there will be a small increase in radial peaks and core power. The event will be terminated by the ASGPT. This event is less limiting than a loss of load to one SG (Section 14.12.2.4) because it produces a smaller temperature tilt across the core.

14.12.2.2 Asymmetric Loss of Feedwater

An Asymmetric LOFW event is initiated at HFP from within the LCOs by a malfunction in one of the feedwater controllers which instantaneously shuts the

feedwater regulator valve to one SG. The closure of the feedwater regulator valve causes a LOFW to the SG. The LOFW will cause the temperature and pressure to increase in response to the decreasing SG level. The temperature and pressure in the unaffected SG (i.e., with feedwater flow available) also increases in response to the increased turbine header pressure. The core inlet temperature from both SGs will increase with the decreased secondary heat transfer. A slight core inlet temperature asymmetry occurs with the higher inlet temperature resulting from the affected SG.

The small core inlet temperature tilt will not cause a significant radial power tilt. The slight increase in core temperatures in conjunction with a negative MTC will result in a decrease in core average power. The event will be terminated by the Asymmetric SG Pressure Trip or a Low SG Level Trip. This event is less limiting than a loss of load to one SG (Section 14.12.2.4) because it produces a smaller temperature tilt across the core.

14.12.2.3 Asymmetric Excess Load

An Asymmetric Excess Load event is initiated at HFP from within the LCOs by the inadvertent opening of a single secondary safety valve on one SG. The excess load on a single SG causes its pressure and temperature to decrease which results in a decrease in the core inlet temperature. Since the temperature from only one SG decreases, a core inlet temperature distribution tilt occurs across the core. In the presence of a negative MTC, positive moderator reactivity feedback occurs that increases the core power. A new steady-state condition is obtained once the core power increases to match the excess load demand. The event will be terminated by the Asymmetric SG Pressure Trip or Low SG Level Trip. This event is less limiting than a loss of load to one SG (Section 14.12.2.4) because it produces a smaller temperature tilt across the core.

14.12.2.4 Asymmetric Loss of Load

An Asymmetric Loss of Load event is initiated at HFP from within the LCOs by an inadvertent closure of a single MSIV on one SG. The loss of load to a single SG causes the pressure and temperature on the SG to increase. With the decrease in SG heat transfer, the core inlet temperature from the isolated SG will increase. The isolated SG water level drops rapidly as the increasing pressure collapses the steam bubble in the liquid inventory. The pressure will continue to increase until the MSSVs open.

The analysis assumes the turbine load demand remains constant, which causes the turbine control valves to open further. The increased load demand will decrease the other (i.e., unaffected) SG pressure and temperature. In response to the decreased temperature, the core inlet temperature from the SG will also decrease. Present operating practice maintains the turbine control valve flow area constant, which will lessen the severity of the event.

The result of the outlet temperature increase and decrease from their respective SGs is a severe core inlet temperature maldistribution. In the presence of negative MTC and FTC (normally negative at power), the coolant temperature tilt will cause a radial power shift toward the cold side of the core. The power in the outermost fuel bundles, where there is almost no mixing of the inlet flow, will experience the greatest local power increase. The power on the hot side of the core will decrease due to the negative moderator reactivity feedback.

The ASGPT will initiate a reactor trip to terminate the event when the absolute differential SG pressure (i.e., P_{sg1}-P_{sg2}) exceeds a preselected analysis setpoint value.

14.12.3 CORE AND SYSTEM PERFORMANCE

14.12.3.1 Mathematical Models

The transient response of the RCS and steam systems to the Asymmetric Steam Generator Loss of Load event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 1. The event is analyzed with an S-RELAP5 model which captures the asymmetric core inlet temperature distribution and applies local peaking augmentation factors (Reference 2).

The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using an approved AREVA DNB correlation. The overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation. The limiting design axial power profile (a top peaked axial power distribution) was used for this simulation for conservatism. Both of these computer codes are described in Section 14.1.4.1.

14.12.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis of the Asymmetric Loss of Load event are listed in Table 14.12-1 and Figure 14.12-1. Those parameters that are unique to the analyses are discussed below.

The analysis used the radial peaking distortion factor as a function of core inlet temperature for calculating the minimum DNBR and peak linear heat generation rate (PLHGR). To increase the power tilt, a negative MTC was assumed. During a severe asymmetric transient, the cooler core inlet temperature and the temperature tilt decalibrate the neutron power and the DT power, respectively.

Asymmetric tube plugging is bounded by assuming no tube plugging in both generators and minimum MSSV setpoints. No tube plugging increases the initial pressures in both generators and causes earlier opening of the affected SGs MSSVs, thereby delaying actuation of the ASGPT.

The MTC is the only key parameter which is adversely impacted by extended burnup. The analysis assumed an EOC MTC value. Hence, the effects of extended burnup have been explicitly and conservatively included in the analysis.

14.12.3.3 Results

Table 14.12-2 contains the sequence of events for the Asymmetric Loss of Load event at HFP. Figures 14.12-2 through 14.12-6 present the transient behavior of the core power, core average heat flux, RCS temperatures, pressurizer pressure, and SG pressure.

The Asymmetric Loss of Load event at HFP conditions peak power result combined with the Technical Specification peaking factor, inlet temperature augmentation and uncertainties, and accounting for control rod position, axial peak, and engineering uncertainty, results in a conservative PLHGR that is below the cycle specific limit and is below the LHGR limit imposed in Reference 2.

The S-RELAP5 plant simulation results from the analysis of the Asymmetric SG Loss of Load event were used as input into the minimum DNBR calculations. The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations. The MDNBR was above the NRC-approved DNB correlation upper 95/95 limit plus a 2% mixed core penalty. In addition, adequate FCM margin exists for this event.

14.12.4 CONCLUSION

The analysis of the Asymmetric SG event demonstrates that operating within the LCOs in conjunction with the LSSS will prevent exceeding the fuel SAFDLs, and maintain the integrity of the RCS. The radiological consequence of opening the atmospheric dump valve upon reactor trip is a site boundary dose that is negligible compared to the 10 CFR 50.67 guidelines.

14.12.5 REFERENCES

1. EMF-2310(P)(A), Revision 1, May 2004, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors

2. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)

TABLE 14.12-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD TO ONE STEAM GENERATOR

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Effective MTC	pcm/°F	-33
Fuel Doppler Temperature Feedback	---	EOC Fuel Temperature Dependent
ASGPT Setpoint	psid	186
Minimum CEA Worth Available at Trip	pcm	5740.8
SG Tube Plugging	%	0
Asymmetry in SG Tube Plugging Assumed	% difference between SGs	0
Initial Vessel Flow Rate	gpm	370,000

TABLE 14.12-2

SEQUENCE OF EVENTS FOR THE LOSS OF LOAD TO ONE STEAM GENERATOR

TIME (sec)	EVENT	VALUE
0.0	Spurious closure of MSIV on SG-1	---
0.0	Steam flow from unaffected SG increases to maintain turbine power	---
6.04	ASGPT Analysis Setpoint reached (differential pressure)	186.0 psid
6.94	Trip breakers open	---
7.44	CEAs begin to insert	---
Varies[a]	Minimum DNBR occurs	> 1.164

[a] Near time of CEA insertion. Time depends on core conditions.

14.13 CONTROL ELEMENT ASSEMBLY EJECTION

14.13.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the CEA is to control the core axial power distribution and provide instantaneous reactivity to shut down the core during controlled procedures and during abnormal and emergency conditions. The CEAs are connected to the CEA drive shafts which are enclosed in the CEDM pressure housing. The CEDM pressure housings are an extension of the reactor vessel closure head and provide a part of the reactor coolant boundary. There are a total of 61 CEDM housings (Chapter 3) on the replacement reactor vessel closure head, 57 CEAs and 4 spare CEDMs. Eight of the extension shafts that were connected to PLCEAs have been removed.

The CEAs are withdrawn, inserted, and held by the CEDMs. The CEDMs are a magnetic jack-type drive system which operate at a constant speed. A CEA is withdrawn in a programmed sequence by the CEDM lifting coil with a total drive stroke of 137". Due to the physical dimensions of the CEA extension shaft and the CEDM housing, the withdrawal of a CEA cannot breach the pressure housing.

A CEA Ejection event is defined as a rapid, uncontrolled, total withdrawal of a single or dual CEA. A dual CEA is two CEAs connected to a single CEA extension shaft. The event is postulated to occur as a result of a complete instantaneous circumferential rupture of either the CEDM pressure housing or the CEDM nozzle from the reactor vessel closure head. The pressure of the RCS causes the ejection of the extension shaft through the rupture and the movement of the CEA to a fully-withdrawn position.

The most limiting CEA Ejection event is a rapid total withdrawal of the highest worth CEA.

14.13.2 SEQUENCE OF EVENTS

A CEA Ejection event can result in an approach to the fuel SAFDLs. The action of the Variable High Power, or the TM/LP trip in conjunction with the LCOs will prevent exceeding these limits. For this event, due to the postulated breach of the CEA pressure housing, the RCS pressure boundary will also be breached and the site boundary dose guidelines will be approached.

14.13.2.1 Zero Power Case

A CEA Ejection event is initiated from HZP (10^{-9} RTP) and from within the LCOs by a rapid uncontrolled total withdrawal of a CEA within 0.10 seconds. At this point, the RCS is assumed to be breached for the purpose of determining the radiological consequences at the site boundary. The immediate reactor core response is an exponential increase in nuclear power. The delayed neutron fraction consistent with the time in cycle (BOC or EOC) is used.

At 40% (30% plus 10% uncertainty) of RTP, a VHPT is initiated. As the fuel temperature starts increasing, negative Doppler feedback partially negates the ejected CEA reactivity worth and terminates the power excursion. After the High Power Trip, RPS response and CEA holding coil delay times have elapsed, the CEAs will insert and terminate the event.

14.13.2.2 Full Power Case

A CEA Ejection event is initiated at HFP from within the LCOs by a rapid uncontrolled total withdrawal of a CEA within 0.10 seconds and the breaching of the RCS pressure boundary. The immediate reactor core response is an

exponential increase in nuclear power. The delayed neutron fraction consistent with the time in cycle (BOC or EOC) is used.

At 110.33% (i.e., maximum analysis setpoint including uncertainties) of RTP, a VHPT is initiated. The negative Doppler feedback due to the increasing fuel temperature partially offsets the ejected CEA worth and terminates the power excursion. The insertion of the CEAs will terminate the event after the RPS response time and CEA holding coil delay time has elapsed.

The peak deposited energy is a function of the initial stored energy, the amount of energy generated in the fuel rod, and the amount of energy released to the coolant during the transient. The initial stored energy is a function of initial LHGR and fuel-clad gap conductivity. The energy generated in the fuel rod during the transient is a function of the ejected CEA worth and the change in the radial and axial power distribution. The amount of heat transferred out of the fuel rod is a function of the fuel-clad gap conductivity and coolant-fuel rod film coefficient. To maximize the peak deposited energy during the transient, the analysis assumes the simultaneous occurrence of the most limiting combination of these parameters.

14.13.3 CORE AND SYSTEM PERFORMANCE

14.13.3.1 Mathematical Models

The Control Rod Ejection event is analyzed to the following acceptance criteria:

1. The radial average fuel pellet enthalpy at the hot spot must be < 200 cal/g
2. A calculation was performed to ensure maximum RCS pressure is below 110% of design pressure.

The average fuel pellet enthalpy limit is imposed by Reference 12. To preserve a coolable geometry and avoid molten fuel-to-coolant interaction, the modified criterion from peak radial average fuel enthalpy is 200 cal/g.

The peak RCS pressure for the analysis of record is that performed to support Reference 2.

The control rod ejection analysis is broken into three components: (1) fuel enthalpy calculation, (2) S-RELAP5 transient simulation, and (3) XCOBRA-IIIC DNB calculation and FCM calculation.

Fuel enthalpy is calculated using the methodology described in Reference 11. The fuel enthalpy calculation is based upon cycle-specific parameters, calculated at four state points; HFP-BOC, HFP-EOC, HZP-BOC, and HZP-EOC. The burnup dependence of cycle-specific parameters (ejected rod worth, post-ejected F_q, Doppler reactivity, and delayed neutron fraction) is bounded by evaluating at BOC and EOC only. The use of cycle-specific parameters instead of bounding data requires calculations to be performed each reload.

The PDIL contains a breakpoint which allows rod banks to be partially inserted at HZP conditions. A partially inserted rod bank may lead to an ejected worth and power peak that does not bound the ejected worth and power peak produced by rods that are fully inserted. Therefore, the HZP peak average radial fuel enthalpy is calculated based upon a modified PDIL (modified in analytical space) that assumes CEA Bank 3 to be fully inserted at HZP (Reference 12).

The S-RELAP5 hot-spot model is used to calculate the fuel centerline temperature, consistent with the methodology in Reference 10. In addition, S-RELAP5 is used to determine the conditions for subsequent DNBR calculations.

The XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distribution for the entire core and the DNB performance for the DNB limiting assembly. The limiting assembly DNBR calculations were performed using an approved DNB correlation at the statepoints HZP-BOC, HZP-EOC, HFP-BOC, and HFP-EOC. Both of these computer codes are described in Section 14.1.4.1.

The RCS pressure used in the DNBR calculations is held constant at the initial core exit pressure. For the HFP cases, a design axial power shape was assumed that bounds the LCO and LSSS ASI limits.

14.13.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.13-1. Those parameters which are unique to the analysis are discussed below.

Analyses were performed using either BOC or EOC physics parameters. Although there is a burnup dependence associated with the delayed neutron fraction (smaller with exposure, prompts higher ejected worth) and Doppler coefficient (more negative with exposure, prompts more negative reactivity feedback), the calculations at the extremes of the burnup window adequately covers the variability in key physics parameters. The maximum positive BOC MTC was used for the BOC case, consistent with the power level analyzed. The EOC MTC, calculated at EOC HZP conditions, is used for the EOC HFP and HZP cases.

Hot zero power and HFP BOC and EOC analyses were performed. Conservative (i.e., bounding) post-ejection peaking and ejected worth were used to determine deposited energy values.

The analyses assumed the CEA is ejected in 0.10 seconds. The HFP analysis assumes the CEA is ejected from the transient insertion limit. The ejected worth is increased in analysis to ensure that the power excursion is arrested by Doppler reactivity feedback, followed by a reactor trip signal, within 5 seconds of when the power excursion begins.

The full power case assumes the core is initially operating at 2754 MWt while the zero power ejection assumes an initial power level of 2.8 W. A VHPT is conservatively assumed to initiate at 110.33% of 2737 MWt for the full power case and 40% (30% + 10% uncertainty) of 2737 MWt for the zero power case to terminate the event. The plant protection system setpoints were adjusted to account for uncertainty and time delay. The initiation of scram was delayed to account for holding coil decay.

The RCPs were assumed to be running and no loss of offsite power is assumed, consistent with the current licensing basis. Pressurizer spray and PORVs were assumed to be available but pressurizer heaters were assumed unavailable.

14.13.3.3 Results

Table 14.13-2 contains the results for the zero power and HFP CEA Ejection events. Figures 14.13-1 and 14.13-2 present the core power response for the full power and zero power cases for the analysis. The results show the calculated hot

spot centerline fuel temperature remains below the fuel melting point (corrected for Gadolinia content and burnup).

Fuel damage assumptions used in the dose analysis are conservative and are described below.

14.13.4 DOSE ANALYSIS

The CEA Ejection event dose analysis conforms to the regulatory requirements of 10 CFR 50.67 and Reference 9 using AST methodology. The doses resulting from a postulated CEA Ejection event would be a composite of doses resulting from portions of the release going out via the Containment and portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequences of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios. The fuel melting temperature criterion used for release of large fractions of fission gases will correspond to the initiation of melting. If the temperature is insufficient to cause incipient fuel melt but is sufficient to cause clad damage, then the gas gap activity of the affected fuel pins is assumed to be released instantaneously and uniformly into the primary system. For this AST analysis, the worst-case historical fraction for incipient centerline melting of 8% including clad failure is assumed. However, based on a 1987 SER (Reference 6), Calvert Cliffs accepted a 10% fuel damage fraction. Since 8% of the fuel is assumed to melt with clad failure, an additional 2% of the fuel is assumed to experience clad failure with no fuel melt releasing the gas gap contents. Note that since only the highest power fuel pins are assumed to be damaged, the releases are increased by the pin power peaking factor of 1.7.

For the 30-day containment release path scenario, the failed/melted fuel activity resulting from a postulated CEA Ejection event (consisting of 100% of the noble gases, 25% of the iodines, 30% of the alkali metals, 5% of the tellurium metals, 2% of the bariums and strontiums, 0.25% of the noble metals, 0.05% of the cerium group, and 0.02% of the lanthanides contained in the fuel which is estimated to reach initiation of melting and 10% of the noble gases, 10% of the iodines, and 24% of the alkali metals which are contained in the gas gaps of the fuel which experience clad failure) is released into the primary system, which is then released in its entirety into the Containment via the ruptured control rod drive mechanism housing. The released activity is instantaneously and uniformly mixed in the free volume of the Containment and is then released at the containment Technical Specification leak rate into the environment. Cleanup via aerosol natural deposition using the 10^{th} percentile Powers aerosol decontamination model, and containment filtration using two 20,000 $\pm$ 2,000 cfm recirculation filtration units at 90% inorganic and 30% organic iodine efficiency with manual initiation at 20 minutes post-accident is credited. The analysis was performed using the RADTRAD computer code.

For the 8-hour secondary release path scenario, the failed/melted fuel activity, resulting from a postulated CEA Ejection event (consisting of 100% of the noble gases, 50% of the iodines, 30% of the alkali metals, 5% of the tellurium metals, 2% of the bariums and strontiums, 0.25% of the noble metals, 0.05% of the cerium group, and 0.02% of the lanthanides contained in the fuel which is estimated to reach initiation of melting, and 10% of the noble gases, 10% of the iodines, and 24% of the alkali metals which are contained in the gas gaps of the fuel which experience clad failure) is released into the primary system, which is then transmitted into the secondary system via the Technical Specification SG tube leakage. The condenser is assumed to be unavailable due to loss-of-offsite power. Environmental releases occur from both SGs via the ADVs and MSSVs. The SG tubes remain covered for the duration of the event; therefore the gap iodines have a partition coefficient of 100 in the SGs. A conservative flashing fraction is assumed (10% for the first 15 minutes and 1% thereafter); however, no credit for scrubbing in the SG is

assumed. The steam release from the SG for the first 1800 seconds was taken directly from a CESEC calculation. The steam release from 1800 seconds to 8 hours is based on a simple energy balance methodology; that is, the steam released from 1800 seconds to 8 hours is based on the amount of steam required to remove the residual heat from the primary and secondary systems, the decay heat generated in the core, and the reactor coolant pump heat. The steam release rates are divided by a partition coefficient of 100 per Reference 9 and entered into the RADTRAD computer code.

The activity released to the environment is transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients. A constant Control Room inleakage of 3500 cfm was assumed in this analysis. Control Room filtration is credited based on a nominal flow of 10,000 ± 1,000 cfm per train. A charcoal filter efficiency of 90% is credited for elemental and organic iodine, while a high efficiency particulate air (HEPA) efficiency of 99% is credited for particulate iodine. The Control Room and site boundary doses are calculated based on the appropriate breathing rates and occupancy factors and on References 7 and 8 dose conversion factors. Additional inputs are included in Table 14.13-3.

The EAB, LPZ, and Control Room doses for the design-basis CEA Ejection event are detailed in the following table. Note that all values are below the regulatory limits.

Results	EAB Rem	LPZ Rem	Control Room Rem
8 hour Secondary Pathway	0.32798	0.088148	4.76271
30 day Containment Pathway	0.8513	0.2190	2.0281
Regulatory Limits	6.3000	6.3000	5.0000

The most limiting case is a CEA ejection from HFP at BOC conditions. The sequence of events for this condition is presented in Table 14.13-2. Figures 14.13-1 and 14.13-2 present core power as a function of time for this condition.

The S-RELAP5 plant simulation results from the analysis of the CEA Ejection event were used a input to DNBR calculations. The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations. The MDNBR was above the NRC-approved DNB correlation upper 95/95 limit plus a 2% mixed core penalty. In addition, adequate FCM margin exists. The total average fuel pellet deposited enthalpy was determined to be less than 200 cal/g using the Reference 11 methodology.

14.13.5 CONCLUSION

The analysis of the CEA Ejection event demonstrates that operating within the LCOs and in conjunction with the LSSS will limit fuel clad failure to less than 10%, will prevent exceeding the RCS Pressure Upset Limit, and will therefore limit the radiological site boundary dose to well within the criteria in 10 CFR 50.67 guidelines.

The analysis of this event explicitly included the effects of extended burnup and since the resultant site boundary doses are within 10 CFR 50.67 limits, it is concluded that the consequences of the CEA Ejection event are acceptable.

The most limiting case is a CEA Ejection from HFP at BOC conditions. The sequence of events for this condition is presented in Table 14.13-2. Figures 14.13-1 and 14.13-2 present the core power as a function of time for this condition.

The S-RELAP5 plant simulation results from the analysis of the CEA Ejection event were used as input to DNBR calculations. The plant simulations were adjusted to account for power, temperature, pressure and flow measurement uncertainties in the minimum DNBR calculations. The MDNBR was above the NRC-approved DNB correlation upper 95/95 limit plus a 2% mixed core penalty. In addition, adequate FCM margin exists. The total average fuel pellet deposited enthalpy was determined to be less than 200 cal/g using the Reference 11 methodology.

14.13.6 REFERENCES

1. CENPD-190A, "CEA Ejection, CE Method of Control Element Ejection," January 1976

2. Letter from Mr. Alexander N. Chereskin (NRC) to Mr. Bryan C. Hanson (Exelon), dated December 30, 2015, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment RE: Revision to Pressurizer Safety Valve Technical Specifications

3. Deleted

4. Deleted

5. Deleted

6. Letter from NRC to BGE, “Revised Safety Evaluation Supporting Amendment No. 108 to Facility Operating License No. DPR-69,” June 30, 1987

7. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988

8. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993

9. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000

10. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004

11. XN-NF-87-44(A), “Generic Analysis of the Control Rod Ejection Transient for PWRs”

12. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)

TABLE 14.13-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE CEA EJECTION EVENT

PARAMETER	UNITS	VALUE
FULL POWER		
Core Power Level	MWt	2754
Core Inlet Temperature	°F	548
RCS Pressure	psia	2250
Vessel Flow Rate	gpm	370,000
Effective MTC (Density vs Reactivity)	pcm/°F	+1.5 (BOC) -10.36 (EOC)
Doppler Reactivity Coefficient	pcm/°F	-0.9 (BOC) -1.1 (EOC)
Ejected CEA Worth	pcm	350 (BOC) 150 (EOC)
Delayed Neutron Fraction	---	0.006469 (BOC) 0.005227 (EOC)
Post-Ejection F_q	---	2.875 (BOC) 2.7175 (EOC)
CEA Bank Worth at Trip	pcm	-5420
CEA Drop Time	sec	3.10
ZERO POWER		
Core Power Level	W	2.8
Core Inlet Temperature	°F	532
RCS Pressure	psia	2250
Vessel Flow Rate	gpm	370,000
Effective MTC (Density vs Reactivity)	pcm/°F	+7.0 (BOC) -10.36 (EOC)
Doppler Reactivity vs. Fuel Temperature	---	Nominal for time in life, reduced by uncertainty, and biased 10% less negative
Ejected CEA Worth	pcm	660 (BOC) 530 (EOC)
Delayed Neutron Fraction	---	0.006469 (BOC) 0.005227 (EOC)
Post-Ejection F_q	---	6.884 (BOC)\ 9.260 (EOC)
CEA Bank Worth at Trip	pcm	-3500
CEA Drop Time	sec	3.10

TABLE 14.13-2

CEA EJECTION EVENT RESULTS

FULL POWER	**BOC CONDITIONS**	
Time (sec)	**EVENT**	**VALUE**
0.0	Beginning of reactivity insertion	---
0.023	High Power scram setpoint reached	110.33%
0.10	Ejected CEA Fully Withdrawn	---
0.14	Maximum Nuclear Power	~ 210%
0.923	CEA Insertion Begins	---
1.865	Minimum DNBR	> MDNBR Limit
1.9	Maximum Core Heat Flux	3518 MWt
3.25	Maximum Fuel Centerline Temperature	4398.2 °F

ZERO POWER	**BOC CONDITIONS**	
Time (sec)	**EVENT**	**VALUE**
0.0	Beginning of reactivity insertion	---
0.10	Ejected CEA Fully Withdrawn	---
1.099	High Power scram setpoint reached	40.0%
1.275	Maximum Nuclear Power	~ 160%
2.199	CEA Insertion Begins	---
3.70	Minimum DNBR	> MDNBR Limit
3.725	Maximum Core Heat Flux	1495.1 MWt
4.9	Maximum Fuel Centerline Temperature	4309.6 °F

TABLE 14.13-3

ASSUMPTIONS FOR RADIOLOGICAL CONSEQUENCES OF THE CEA EJECTION EVENT

(1) Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. Iodine released from the failed fuel is assumed to be 95% particulate, 4.85% elemental, and 0.15% organic.

(2) The maximum allowable containment leakage rate L_a contained in the Containment Leakage Rate Testing Program of Technical Specification 5.5.16 is 0.16 percent per day at P_a. Per Reference 9, the Containment should be assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.

(3) The breathing rate and Control Room occupancy factors are per Reference 9.

- Breathing rate:
  - 3.5E-4 m^3/s for 0-8 hours
  - 1.8E-4 m^3/s for 8-24 hours
  - 2.3E-4 m^3/s for 24-720 hours
- Occupancy factors:
  - 1.0 for 0-24 hours
  - 0.6 for 24-96 hours
  - 0.4 for 96-720 hours

(4) The ADV to site boundary 2-hour, atmospheric dispersion coefficient is 1.44E-4 sec/m^3.

(5) The Containment to site boundary 2-hour, atmospheric dispersion coefficient is 1.30E-4 sec/m^3.

(6) The atmospheric dispersion coefficients from the ADV to the Control Room are:

- 3.83E-3 sec/m^3 for 0-2 hours
- 3.25E-3 sec/m^3 for 2-8 hours
- 1.32E-3 sec/m^3 for 8-24 hours
- 9.92E-4 sec/m^3 for 24-96 hours
- 7.92E-4 sec/m^3 for 96-720 hours

(7) The atmospheric dispersion coefficients from the Containment to the Control Room are:

- 1.11E-3 sec/m^3 for 0-2 hours
- 7.29E-4 sec/m^3 for 2-8 hours
- 3.19E-4 sec/m^3 for 8-24 hours
- 2.36E-4 sec/m^3 for 24-96 hours
- 1.98E-4 sec/m^3 for 96-720 hours

14.14 <u>STEAM LINE BREAK EVENT</u>

14.14.1 IDENTIFICATION OF EVENT AND CAUSE

The Main Steam System carries steam from the SGs to the turbine-generators and to other auxiliary equipment. An SLB may occur as a result of thermal stress or cracking in the main steam line. The guillotine-type break assumed in the Safety Analysis is the most adverse transient scenario and its probability of occurrence is extremely low.

A rupture in the Main Steam System increases the rate of heat extraction by the SGs and causes cooldown of the reactor coolant. With a negative moderator coefficient of reactivity, the cooldown will produce a positive reactivity addition. A severe decrease in main steam pressure will cause the MSIVs in the main steam lines to trip closed (Chapter 10). The MSIV is a gas hydraulic, bi-directional, balanced disk, Y-pattern, globe valve designed to hold pressure from either direction. If the steam line rupture occurs between the SG and the isolation valve, blowdown of the affected SG would continue. However, termination of flow from the intact SG occurs with closure of both isolation valves, either of which is capable of stopping flow. A detailed description of the MSIV verification program is provided in Section 14.14.2.

The SLB transient was analyzed in two distinct portions, referred to as pre-trip and post-trip. For the pre-trip portion of the event, the main concern is the power excursion seen due to the cooldown in combination with a negative MTC. A loss of power on reactor trip is also assumed. A parametric analysis of break size and MTC was performed to determine the limiting case with respect to the DNB SAFDL. Cases run for an inside Containment break credited high containment pressure. Cases run for an outside Containment break credited low SG pressure, high power-NI, and delta-T power trips.

For the post-trip portion of the event, the concern is a return to power in the vicinity of an assumed stuck rod. Full load and no load cases, with and without a loss of power, were analyzed to find the limiting cases with respect to the DNB and LHR SAFDL. A guillotine break of the main steam line was assumed, as this provides the largest cooldown of the RCS and the greatest potential for a post-trip return to power.

The one-loop full load and one-loop no load cases were not analyzed since Technical Specifications prohibit operation in these modes. Since the SLB event is classified as a postulated accident, site boundary doses must not exceed the 10 CFR 50.67 guidelines. Since the SGs are designed to withstand coolant system operating pressure on the tube side with atmospheric pressure on the shell side, the continued integrity of the RCS barrier is assured. The most limiting of the pre- and post-trip events are summarized in Section 14.14.4.3.

14.14.2 DISCUSSION OF MAIN STEAM ISOLATION VALVE TESTING

The MSIVs have been designed to close in less than six seconds under the pressure, temperature, and flow conditions applicable to the assumed accident. The design techniques used to ensure that the MSIVs will function reliably under accident conditions have been extensively verified by testing different valves that employ similar principles under conditions of pressure, temperature, and flow similar to those applicable to the assumed accident.

The testing under accident conditions has been performed on Y-type balanced disc valves. Construction of the MSIVs is described in UFSAR Chapter 10. The balanced disc valves which have been tested differ from the Calvert Cliffs MSIVs as follows:

1. The balanced disk valves are actuated by air and springs. They are only capable of closing when line pressure forces the valves shut because the actuators do not exert sufficient force to close the valves against full reverse pressure. The Calvert

Cliffs balanced disk MSIVs employ a hydraulic cylinder directly coupled to a nitrogen accumulator that is capable of closing the valves with pressure from either side. They are arranged in a similar manner to the tested balanced disk valves so that normal flow tends to close the valves. Testing has been performed on the Calvert Cliffs actuators showing they deliver the required force to close the valve under the maximum calculated (1085 psi) reverse flow force.

2. The balanced disk valves utilize hydraulic dash-pots to control the disk speed during the closure. Flow control valves throttle the flow of oil from one side of the piston to the other, effectively damping the motion and preventing steam flow from slamming the valves closed. The Calvert Cliffs MSIVs use automatic adjusting orifice and a fixed orifice to control valve speed during closure. The orifices provide a constant oil flow rate independent of upstream pressure. These orifices throttle the oil from below the piston to the oil reservoir.
3. The balanced disk valves have pilot assemblies in the disk which equalize pressure across the valves for opening. The Calvert Cliffs MSIVs have a check disk assembly in the center of the main valve disk. This equalizes steam pressures above and below the disk to balance the forces on the disk. This design reduces the forces necessary to close the valve during reverse flow conditions.

Testing of balanced disk valves under conditions similar to those existing in the assumed accident has verified that the Y-type valve configuration shuts reliably and does not bind.

The Calvert Cliffs MSIVs incorporate a bidirectional balanced disk design. This design balances the forces above and below the disk, which reduces the forces necessary to close the valve during reverse flow conditions. The maximum force required to close the valve under full reverse flow conditions has been calculated. The MSIV actuator was tested on a load cell showing that it can deliver the required force to close the valve in less than six seconds. Therefore, the actuator will function and close the valve quickly against reverse flow.

The manufacturer of the Calvert Cliffs MSIVs has tested the closure of his 16" American National Standards Institute Class 600 air-actuated balanced disk valve under rupture conditions using 1500 psig air as the flowing medium. Mass flow rates were typically 1800 pounds per second during closure. Closure occurred in as little as 2-1/2 seconds, and typically before the pressure had decayed below 1200 psig. Closure time under rupture conditions was within 10% of closure time under no-flow conditions using pressure compensated flow control valves for speed control. The Calvert Cliffs MSIVs are of 32" bore size and would experience mass flow rates of about 3330 pounds per second at the beginning of an accident. The variation in closing time between no-flow and rupture-flow conditions is less than 0.8 seconds.

This number has been verified by load testing of the actuators. This 0.8 seconds difference was applied to the accident analysis valve closure time (less than 6.0 seconds at that time), resulting in the Technical Specifications valve closure limit of less than 5.2 seconds. The accident analysis valve closure time has since been increased to 7.0 seconds to incorporate margin in the analysis. The manufacturer has also functionally tested balanced disk valves for 20,000 operations containing 575°F saturated steam under no flow conditions. In all cases the valves functioned reliably and the design techniques were verified.

In addition, a 20" balanced disk valve of similar construction, but made by a different manufacturer, has been tested for closure by General Electric Company and Commonwealth Edison Company (Reference 1). In this test, boiler steam of varying moisture contents was used.

After replacement of the MSIV actuator and modification of the valve internals from a stop valve to a bidirectional balanced disk, the valve was operated through at least four complete cycles with no flow. The correlation between test closure time and accident closure time is discussed in Chapter 10.

Periodic testing after installation is discussed in Chapter 10.

14.14.3 SEQUENCE OF EVENTS

The rupture of a main steam line will cause the affected SGs pressure and temperature to rapidly decrease. The steam released through the break in the affected SG extracts heat from the primary side which causes the primary coolant temperatures and pressure to rapidly decrease. The decrease in the primary coolant temperature, in combination with a negative MTC, results in positive reactivity addition which causes the core power level and core heat flux to increase.

The uncontrolled blowdown of the affected SG will initiate one of the following reactor trips: high containment pressure, high power, delta-T power, low SG pressure, or TM/LP. The power increase will be terminated after the CEAs are inserted. The drop in the SG pressure will also initiate a SGIS. Following appropriate delays, the MSIVs on both the affected and on the unaffected SGs will close and terminate the blowdown from the unaffected SG.

The blowdown from the affected SG will continue since the MSIV is downstream from the break area. The continued blowdown causes the RCS pressure to decrease until SIAS is initiated, which automatically starts the HPSI Pumps. Since the shutoff head of the HPSI Pumps is approximately 1200 psia, no safety injection (SI) flow is delivered immediately. Therefore, the pressure continues to decrease and the pressurizer empties.

The conservative assumption for the pre-trip portion of the event is that a LOOP after a turbine trip results in a coast down of the RCPs, which causes the core flow to decrease. For the post-trip portion of the event, both the LOOP and no LOOP cases are considered with respect to the return to power potential.

The cooldown of the RCS will insert positive reactivity from moderator and fuel temperature feedbacks. This positive reactivity addition will erode the negative reactivity added by the CEAs. The magnitude of core subcriticality depends on the SCRAM worth and the moderator and fuel temperature reactivity feedbacks. Since it is expected that all CEAs will be inserted after a reactor trip signal, the negative reactivity addition from the inserted CEAs will be sufficient to offset the positive reactivity inserted by the moderator and fuel temperature feedbacks. Consequently, the core will remain sufficiently subcritical to prevent any power increase over decay power levels if all CEAs are inserted.

The cooldown of the RCS is terminated when the affected SG blows dry and AFW flow is isolated to the ruptured SG. As the coolant temperature decreases, additional positive reactivity insertion occurs from the moderator reactivity feedback. However, the insertion of additional positive reactivity is not sufficient to significantly erode the negative reactivity inserted by the CEAs following the trip. In addition, the HPSI Pump flow will insert negative reactivity, which also ensures that the core remains subcritical during the event.

The HPSI pump flow will add sufficient coolant mass such that the pressurizer level will be reestablished and the RCS pressure will be maintained. The cooldown of the RCS is terminated once action is taken to terminate the AFW flow to the affected SG.

In summary, during this event various reactivity feedback mechanisms are observed. The various components of reactivity are summarized below.

1. Moderator Reactivity - The decrease in primary coolant temperatures, in combination with an effective negative MTC, inserts positive reactivity. The positive reactivity insertion continues until the coolant temperatures increase in response to the production of instantaneous fission power and the eventual dryout of the ruptured SG. The resulting increase in coolant temperature reduces the reactivity inserted during the moderator cooldown. This trend continues until the coolant temperatures once again begin to decrease due to the cooling action of AFW flow delivered to the ruptured SG. The decrease in coolant temperatures causes an increase in core reactivity.
2. Doppler Reactivity - The fuel temperatures rapidly decrease after reactor trip (i.e., from HFP to HZP temperatures) which leads to a significant amount of positive reactivity insertion. Thereafter, the fuel temperatures gradually decrease and the positive reactivity insertion continues although at a reduced rate.
3. Boron Reactivity - The decrease in pressure initiates SIAS, which causes the startup of the HPSI pumps. After the pumps reach full speed and the pressure drops below the shutoff head of the pumps, HPSI flow is delivered to the core. The boron injected via the HPSI pumps inserts negative reactivity.

   It is conservatively assumed that the RCS is initially assumed to be at zero ppm boron, for both the HFP and HZP post-trip SLB analyses. Assuming no boron in the RCS negates the negative reactivity insertion that would occur due to concentrating the boron during the cooldown. The charging pumps are not modeled.
4. SCRAM Worth - The reactor trip on low SG pressure causes the CEAs to drop into the core and insert negative reactivity. In the analysis, the temperature dependence of SCRAM worth has been included in the moderator temperature defect.
5. Total Reactivity - The total reactivity is the sum of each individual reactivity component. The total reactivity initially is very negative following reactor trip when the CEAs are fully inserted into the core. The insertion of positive reactivity from moderator and Doppler feedback causes the core to approach criticality. The total reactivity decreases as the moderator reactivity component decreases and boron enters the core.

The major difference between the description given above and the licensing analysis is the assumption on stuck CEA. For the licensing analysis, both the moderator cooldown curve and the SCRAM worth are calculated assuming a stuck CEA, which greatly reduces the magnitude of SCRAM worth and increases the magnitude of moderator reactivity feedback effects. The net effect causes the core reactivity to approach criticality during transient conditions. As the core approaches criticality, subcritical multiplication occurs and instantaneous fission power is produced. That is, a return to power causes temperatures to increase, which inserts negative reactivity from moderator and fuel temperature feedback mechanisms and limits the return to power.

The conservatisms and credits included in the licensing analysis are enumerated below.

1. The most reactive CEA is assumed to stick in the withdrawn position. A SCRAM worth calculated assuming no stuck CEAs would increase the available SCRAM worth.
2. The accident analysis credits the tripping of the main feedwater pumps upon SGIS. The inventory in the main feedwater system is allowed to blow down to the affected steam generator until closure of the MFIVs. The time delays for the

starting of the HPSI pumps and for the closure of the MFIVs are the most conservative values allowed by Technical Specifications. No credit is taken for closure, or rampdown, of the main feedwater regulation valves after reactor trip. Gross leakage to the affected steam generator through the MFIVs (400 gpm per valve) is assumed in the analysis.

3. The accident analysis models the delivery of AFW to the affected steam generator by crediting the installed minimum 20 second delay in the AFW response after AFAS. After identification of the intact steam generator by the Auxiliary Feedwater Actuation System based upon differential SG pressure, the AFW flow is diverted to the intact steam generator within the Technical Specification 20 second response time. Gross leakage through the AFW block valves (40 gpm per valve) is assumed in the analysis. The licensing case thus has delivery of more cold AFW to the affected steam generator than expected.
4. Cases with and without LOOP on turbine trip are examined. With LOOP, the RCPs begin their coastdown. This results in negligible inlet plenum mixing which maximizes the RCS cooldown used in the moderator reactivity calculation. Furthermore, the elevated core outlet temperature results in slightly higher RCS pressures and consequently reduces safety injection flow.
5. Zero tube-plugging is assumed because this will maximize the RCS cooldown rate, total cooldown of the RCS, and moderator reactivity feedback.

14.14.4 CORE AND SYSTEM PERFORMANCE

14.14.4.1 Mathematical Models

The transient response of the RCS and steam systems to the SLB event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 14. The overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation for the pre-scram SLB. The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using an approved AREVA DNB correlation. The limiting design axial power profiles were used for this simulation. Both these computer codes are described in Section 14.1.4.1. A spectrum of SLB sizes, MTCs, and locations were analyzed for the pre-trip power excursion. Post-trip return to power analysis included full and no load, with and without LOOP, and single failure cases.

14.14.4.2 Input Parameters and Initial Conditions

The analysis is consistent with requirements for the analysis of Steam System Piping Failures as described in the Standard Review Plan, Section 15.1.5 (Reference 7). Assumptions made are consistent with SRP requirements:

1. Both cases with LOOP and without LOOP are examined.
2. The maximum worth CEA is assumed held fully withdrawn. Analyses have been performed assuming the most reactive CEA at the nominal zero power conditions as the stuck rod and analyses have been performed assuming the most reactive CEA at the minimum affected core sector temperature during the event as the stuck rod. The latter assumption produced the most limiting combination of scram worth and reactivity addition from moderator feedback.
3. The worst single active component failure was determined and assumed to occur.

Single failures considered in the analysis include, but are not limited to, the following:

1. A single failure within the SI system that would reduce the addition of the high concentration boric acid to the core.
2. A single failure within the feedwater system that leaves one condensate booster pump operating after a CSAS, resulting in further feedwater delivery prior to closure of the MFIVs.
3. A failure of a MSIV to close resulting in additional heat removal from the RCS.
4. A single failure of an MFIV to close upon SGIS resulting in further feedwater delivery.

For postulated failures 1, 2, 3, and 4, no credit is taken for the closure of the main feedwater regulating valves after trip.

In order to determine the worst single active component failure, the following scenarios were investigated for both full load and no load cases:

1. Post-trip SLB scenario with a stuck CEA, with and without LOOP, and a single active component failure resulting in the failure of one HPSI to start.
2. Post-trip SLB scenario with a stuck CEA, with LOOP, and a single active component failure resulting in the failure of an MFIV to close on the affected SG.

A conservatively high value of the AFW flow was assumed; limited only by the maximum possible flow rate through the suction line to the AFW pumps. An AFW flow value of 1550 gpm was used in the analysis. This flow is directed only to the unaffected SG, except for leakage past the AFW block valves after AFAS block occurs.

The limiting analyses assumed that the MFW pumps are tripped upon SGIS. The blow down of the inventory in the MFW system to the affected SG was assumed to continue until closure of the MFIVs. Averaged over this period, the rate of uncontrolled feedwater delivery to the affected SG was more than 50% of the full power feedwater flow rate to that generator.

The unrestrained delivery of MFW until closure of the MFIV bounds the postulated failure within the feedwater system that leaves a condensate booster pump operating.

The scenario of a single failure of the MSIV to close for the unaffected SG was considered, thus extending the blowdown from the unaffected SG until the MSIV for the affected SG closes. In this scenario credit for normal operation of the main turbine steam inlet valves, and the steam dump and bypass valves was taken as a backup to the failed MSIV. Although these valves are not safety-related, credit for operation of these valves is acceptable as a backup to the MSIV per SRP 15.1.5. The results of this scenario are less limiting than those for the HPSI pump failure condition.

The post-trip SLB initiated from both full and no load conditions uses the modified Barnett CHF Correlation (Reference 16) to calculate the minimum DNBR. In addition, the peak LHGR initiated from both full and no load conditions is determined via hand calculation. The three-dimensional power distribution peak (F_q) associated with the worst stuck CEA are used in the calculation.

a. <u>SLB – Post-Trip Return to Power</u>

The post-trip SLB assumes that the event is initiated by the double-ended rupture (guillotine break) downstream of steam outlet nozzle flow restrictor which has an area equivalent to 1.9 ft^2. The SLB event was initiated from an array of initial conditions. The most limiting conditions are listed in Table 14.14-1. The MTC of reactivity assumed in the analysis corresponds to EOC, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the steam line rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density, rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed in the analysis is given in Figure 14.14-1. This moderator cooldown curve was conservatively calculated assuming that on reactor trip the CEA is stuck in the fully-withdrawn position that yields the most severe combination of SCRAM worth and reactivity insertion.

The charging system is not modeled. Since the initial boron concentration is very low, direct dilution from charging in the absence of letdown may help maintain pressurizer level, but has little impact on the results.

The reactivity defect associated with the fuel temperature decrease was also based on an EOC Doppler defect. The Doppler defect based on an EOC FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the SLB event. The Doppler coefficient multiplier (uncertainty) on the FTC assumed in the analysis is a power based Doppler feedback model. The beta fraction assumed is the nominal for EOC conditions.

Conservatively, the boron concentration present in the RCS prior to the event is assumed to be 0 ppm. This negates the negative reactivity insertion that would occur due to concentrating the boron during the cooldown. No charging is assumed.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip was calculated for the stuck rod that produced the moderator cooldown curve in Figure 14.14-1.

During a return to power, negative reactivity credit was assumed in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. The three-channel S-RELAP5 core model can only accommodate relatively simple radial and axial power distributions, associated reactivity feedback, and feedback weighting models. This tends to result in simple and conservative representation of highly complex neutronics and thermal-hydraulic phenomena. The inherent conservatisms are demonstrated by comparing the reactivity change calculated with S-RELAP5 against that calculated with the neutronics code.

A manual reactor trip at t=0 seconds is used to maximize the cooldown of the RCS.

The analysis assumptions regarding the AFW actuation analysis setpoint, the associated time delays, and the AFW flow and leakage through each leg are given below. They were conservatively chosen to deliver the

maximum AFW flow that maximizes the primary cooldown and enhances the potential return to power.

Auxiliary feedwater was conservatively assumed to initiate at the start of the transient with credit taken for minimum AFW actuation delay time, which in all cases resulted in AFW initiation at a level far above the Technical Specification actuation setpoint plus uncertainties. This was done to ensure the analysis results would remain bounding in the event of any future revision of the Technical Specification or uncertainty and is consistent with the enveloping nature of the analysis.

Due to the early initiation of AFW, some flow reaches the affected SG prior to automatic isolation of that SG. The AFW block valve leakage is assumed to enter the damaged SG. Auxiliary feedwater pump flow is assumed to be at a runout value of 1550 gpm.

The analysis included isolation of the ruptured SG when the SG differential pressure reached the analysis setpoint. In addition, a time delay was assumed in the analysis to close the AFW isolation (i.e., block) valves.

One of the scenarios conservatively assumed that on a SIAS, only one HPSI pump starts. In addition, a maximum time delay for HPSI pumps to accelerate to full speed was assumed in the analysis.

The analysis assumes that a limited amount of fluid must be removed or “swept out” of the SI piping before the highly borated SI water reaches the RCS. A conservatively large sweep out volume of 180 ft^3 was assumed. This sweep out volume corresponds to the volume from the SI nozzle, through the check valve and to the first MOV.

b. SLB – Pre-Trip Power Excursion

The results of the post-trip analysis presented above are limiting with respect to return to power. For break sizes less that the maximum double-ended guillotine break, the potential for a pre-trip power excursion becomes the concern. Therefore, the assumptions for the pre-trip analysis listed below are designed to maximize the power excursion prior to reactor trip, rather than maximizing the post-trip return to power.

The limiting pre-trip SLB event was initiated from the conditions listed in Table 14.14-2. The assumptions for the pre-trip event that differ from the post-trip event are listed below.

The reactivity defect associated with the fuel temperature change was based on a nominal Doppler reactivity feedback consistent with the time in cycle (EOC). The pre-trip SLB analysis is performed over a range of MTC versus break size cases. Biasing the Doppler coefficient may result in a different limiting MTC value, but the total reactivity will not be altered. The Beta fraction assumed is the EOC nominal value. An EOC nominal Beta fraction without biasing is sufficient due to the power rise during the pre-trip SLB event being slow.

A spectrum of MTCs was employed to determine the effect of MTC on power range detector response during the pre-trip SLB.

Dose calculations are also performed to calculate the amount of dose as a result of a SLB.

The assumptions made to maximize the site boundary dose are listed in Table 14.14-3. During the event, two sources of radioactivity contribute to the site boundary dose; (1) the initial activity in the SG and (2) the activity associated with the primary-to-secondary leakage. The primary activity includes the maximum initial activity allowed by the Technical Specifications and any activity released to the coolant due to fuel failure. In calculating the site boundary dose, the analysis conservatively assumed that all activity is released to the atmosphere with a decontamination factor of 1.0.

The scenario that would result in the worse dose is an outside containment break. The dose analysis assumes that the SLB would result in 0.80% failed fuel.

14.14.4.3 Results

a. SLB – Post-Trip Return to Power

The post-trip SLB scenario with a stuck CEA and a single active component failure resulting in the failure of one HPSI pump to start is the limiting event. The limiting case among full load and no load, with and without LOOP, depends upon cycle-specific core physics parameters. In many instances, the limiting scenario for approach to the DNBR SAFDL is different from the limiting scenario for approach to the LHGR SAFDL. The post-trip SLB with a stuck CEA, a single active component failure resulting in a failure of one HPSI pump to start, LOOP, and initiated from a full load condition, while not always limiting, is presented here for example.

A maximum break size is limiting for the post-trip SLB scenario. This break size is limited to an area of 1.9 ft^2 by the SG steam nozzle flow restrictors.

Table 14.14-4 lists the sequence of events for the post-trip SLB with a stuck CEA, a single active component failure resulting in a failure of one HPSI pump to start, no-LOOP, and initiated from a full load condition. Moderator reactivity versus moderator density is presented in Figure 14.14-1. The NSSS responses during the transient are given in Figures 14.14-2 through 14.14-7.

The post-trip SLB results show that the minimum DNBR is above the modified Barnett SAFDL of 1.158 including a 2% mixed core penalty and the peak LHGR remains below the SAFDL of the more limiting of 21 kW/ft (per Reference 15) or the cycle specific limit.

b. SLB – Pre-Trip Power Excursion

The results of the parametric in break size and MTC demonstrated that an outside containment break size of 2.0 ft^2 resulted in a maximum consequence. This break size is limiting since it results in the most adverse conditions for DNBR and PLHR. Therefore, the results of the 2.0 ft^2 pre-trip SLB are presented here.

The sequence of events for the 2.0 ft^2 pre-trip SLB is given in Table 14.14-5. The reactivity insertion as a function of time is presented in Figure 14.14-12. The NSSS responses during the transient are given in Figures 14.14-8 through 14.14-13.

The limiting pre-trip SLB results presented use the average core power, as decalibrated by downcower density changes, to determine the time of trip for the VHPT. The use of the average core power does not account for change in the signal to the limiting excore detector due to change in the radial power distribution during the transient.

Cases were rerun to include consideration of the effects of the radial power change on the limiting excore detector. These cases resulted in the trip signal being reached sooner. The limiting pre-trip SLB results presented are based on the use of average core power.

The results of the analysis show that the SLB causes the core power to increase until a reactor trip signal is generated at the high power setpoint (NI power). The trip breakers open and the CEAs drop into the core, terminating the power and heat flux increases.

A LOOP on turbine trip is assumed to occur at the time of reactor trip.

When the SG Isolation Analysis Setpoint is reached, the MSIVs begin to close and are completely closed seconds later. The blowdown from the intact SG is terminated at this time.

The minimum DNBR during the transient was calculated based on AREVAs thermal-hydraulic code (XCOBRA-IIIC).

14.14.5 CONCLUSIONS

The post-trip SLB results show that the minimum DNBR is above the modified Barnett SAFDL of 1.158 including a 2% mixed core penalty and the peak LHGR remains below the SAFDL of the more limiting of 21 kW/ft (per Reference 15) or the cycle specific limit. The pre-trip SLB results show that the DNBR and FCM limits are not exceeded.

The radiological consequences of a power excursion in the pre-trip portion of an SLB are small when compared to the guidelines of 10 CFR 50.67:

EAB TEDE	0.2180 rem
LPZ TEDE	0.0577 rem
Control Room TEDE	4.6301 rem

These limiting doses are applicable to Calvert Cliffs Unit 1 and Unit 2 assuming a SLB resulting of 0.80% failed fuel (Reference 11).

14.14.6 REFERENCES

1. General Electric Company, "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED - 5750, March 1969. This report has been submitted to the Atomic Energy Commission (AEC) as a topical report.
2. Deleted
3. Deleted
4. Deleted
5. Deleted
6. Deleted

7. NUREG-0800, "Standard Review Plan," Section 15.1.5, Revision 2, July 1981 [Steam System Piping Failures Inside and Outside of Containment (PWR)] (Formerly NUREG-75/087)

8. Deleted

9. Deleted

10. Deleted

11. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), “Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846),” dated August 29, 2007

12. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988

13. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993

14. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004

15. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), “Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832),” dated February 18, 2011

16. IN-1412, TID-4500, “A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,” Idaho Nuclear Corporation July 1970

TABLE 14.14-1

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED FOR THE POST-TRIP SLB EVENT INITIATED FROM FULL POWER

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2737
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
SG Differential Pressure Analysis Setpoint	psid	250
SIAS Setpoint	psia	1640
Minimum CEA Worth Available at Trip	pcm	6026
Moderator Cooldown Curve	pcm vs. Density	Figure 14.14-1
Effective MTC	pcm/°F	-33
Beta Fraction (EOC nominal)	---	0.005215
Percentage of SG Plugged Tubes	%	0

TABLE 14.14-2

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED FOR THE PRE-TRIP SLB EVENT INITIATED FROM FULL POWER

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
High Power Trip Setpoint	% RTP	112
High Power Trip Response	sec	0.4
TM/LP Trip Delay	sec	0.9
High Containment Pressure Trip Setpoint	psia	19.45
High Containment Pressure Trip Delay	sec	0.9
Low SG Pressure Trip Setpoint	psia	600 (harsh) 650 (non-harsh)
Low SG Pressure Trip Delay	sec	0.9
Minimum CEA Worth Available at Trip	pcm	5470.8
Doppler Reactivity Coefficient	pcm/°F	-1.48
Effective MTC	pcm/°F	-8 to -33
Kinetics, β_{eff}	---	0.005237
Radial Peak Temperature Dependence (Asymmetric case only)	---	Case Dependent
Temperature Shadowing or Decalibration Factor	%/°F	0.70
Resistance Temperature Detector Response, Hot/Cold	sec	0.0/25.0

TABLE 14.14-3

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE SLB EVENT

PARAMETER	UNITS	VALUE
RCS Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.5
Secondary Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.1
Partition Factor Assumed for All Doses	---	1.0
Atmospheric Dispersion Coefficient[b]	sec/m^3	$1.44x10^{-4}$
Breathing Rate	m^3/sec	$3.5x10^{-4}$
Dose Conversion Factor	REM/Ci	[c]

(a) Technical Specification limits.

(b) 0-2 hour accident condition.

(c) Dose conversion factors obtained from References 12 and 13.

TABLE 14.14-4

SEQUENCE OF EVENTS FOR THE POST-TRIP SLB EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	SLB Occurs, AFW startup process is initiated	---
0.0	Manual Trip Occurs	---
0.5	CEAs Begin to Drop into Core	---
10.8	SG Isolation Setpoint is Reached	600 psia
11.7	MSIV Begins to Close	---
15.9	SIAS Generated	1640 psia
33.6	SG Differential Pressure Analysis Value is Reached	250 psid
46.8	HPSI pump at full speed, but does not inject due to RCS cold leg pressure being higher than HPSI pump shutoff head	---
53.6	AFW Block Valve Closed on Affected SG	---
63.0	Lowest RCS cold leg pressure reaches HPSI pump shutoff head and HPSI flow begins filling SI lines with borated water	---
75.8	Affected SG MFW isolation valves fully closed (65.0 second delay)	---
209.0	Shutdown worth fully overcome by moderator and Doppler feedback	0.0 $
258.0	Maximum Post-Trip Reactivity	0.03160 $
426.4	HPSI pump delivers enough borated water to completely fill the SI lines and begins delivering boron	---
428.0	Peak Return to Power	258.76 MW

TABLE 14.14-5

SEQUENCE OF EVENTS FOR PRE-TRIP SLB EVENT WITH LOOP ON TURBINE TRIP INITIATED FROM FULL POWER

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	SLB Occurs	2.0 ft^2
22.44	VHPT Setpoint is Reached	112% RTP
22.84	Trip Breakers Open	---
22.90	Maximum Core Average Heat Flux Occurs	3519.4 MW
23.34	CEA Begin to Drop into Core	---
24.40	Minimum DNBR Occurs	> MDNBR Limit

14.15 STEAM GENERATOR TUBE RUPTURE EVENT

14.15.1 IDENTIFICATION OF EVENT AND CAUSES

The SG is the interface heat exchanger between the RCS (primary) and the main steam system (secondary). The reactor coolant flows through tubes in the SG and transfers its heat to the feedwater on the shell side, thereby generating saturated steam. There are two SGs per reactor unit.

The Steam Generator Tube Rupture (SGTR) event is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety, in that a leaking SG tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would then mix with water in the secondary side of the affected SG. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the turbine bypass valves, or directly to the atmosphere via the MSSVs or the ADVs. Any noncondensible radioactive gases entering the condenser are removed by the condenser priming and air removal system and discharged to the plant vent.

Experience with nuclear SGs indicates that the probability of complete severance (double-ended break) of a tube is remote. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. In the event of a SG tube leakage or rupture, the reactor coolant leaks into the secondary side of the SG. The reactor coolant transfer causes the level in the affected SG to increase and the pressurizer level to decrease, provided that the tube leak rate exceeds the capacity of the charging pumps. In the case of a double-ended tube rupture (design basis SGTR event), the leak rate far exceeds the charging pump capacities and, consequently, the pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The rate of RCS depressurization is determined by the leak rate, the charging flow rate and the pressurizer heater capacity. Furthermore, when the pressurizer level decreases to the point where the heaters would be uncovered, this mitigation for the pressure decrease is lost.

The drop in the RCS pressure will also initiate a reactor trip on the TM/LP pressure limit, ensuring that the SAFDLs are not exceeded. Following sufficient time for trip signal processing delays and decay of the CEA holding coil flux, the CEAs enter the core and add negative reactivity, which rapidly reduces the core fission power and heat generation rate, and causes the reactor coolant temperature to decrease.

At approximately the time of reactor trip, the pressurizer empties and the RCS pressure rapidly decreases to the hot leg saturation pressure. The decrease in RCS pressure will also initiate a SIAS. As the pressure drops below the HPSI pump shut-off head, SI flow is delivered to the core. The RCS pressure gradually increases following the initiation of the SIAS and the SI flow, and stabilizes at a pressure near that of the HPSI pump head. The SI flow offsets the coolant mass loss due to the ruptured tube, and results in slowing the depressurization of the RCS. Note that the larger the pressure difference between the primary and secondary, the larger the leak rate.

The SG pressure remains constant until the reactor trip on low pressurizer pressure occurs. The rapid closure of the turbine control and stop valves following turbine trip sharply reduces the secondary steam flow and causes a secondary pressure "spike" to occur. The quick opening of the steam dump and bypass control system (not credited in the safety analysis) following turbine trip, however, limits the magnitude of the secondary pressure spike, and gradually reduces the secondary pressure as the RCS residual heat reaches decay levels.

Based on available indications (i.e., reactor trip, pressurizer level indicators, SG level indicators, condenser off-gas radiation monitor, radiation monitors in the SG blowdown sample lines, SG level indicators, etc.), the operator can identify the nature of the event and manually isolate the SG with the ruptured tube. Once the isolation has occurred, the operator can initiate cooldown per the Emergency Operating Procedures (EOPs). During the cooldown period, the operator may steam the affected SG in order to prevent it from overfilling. The analysis credits backflow from the SG to the primary.

The objective of this analysis is to determine the maximum 0-2 hour EAB TEDE, the 30 day LPZ TEDE, and the 30 day Control Room TEDE which would result from a design basis SGTR event. Doses from this event must meet 10 CFR 50.67 and Reference 1 limits: a) below 10% of the 10 CFR 50.67 limits for the EAB and LPZ limits due to Concurrent Iodine Spike (CIS), b) below the 10 CFR 50.67 EAB and LPZ limits for the fuel damage and Preaccident Iodine Spike (PIS), and c) below the 10 CFR 50.67 Control Room limit for all SGTR events.

The SGTR event analysis accounts for SG tube plugging. Tube plugging reduces the heat transfer surface area and the flow area in the SG, which reduces RCS flow rate and lowers SG pressure. Tube plugging increases the activity release due to increased SG DP.

Isolation of an ADV may occur when an ADV begins to leak at an excessive rate. The ADV is isolated to prevent further leakage and damage to the valve. The SGTR event assumes that the ADV of the unaffected SG is isolated at the onset of the event. Thus, the initial plant cooldown is accomplished using the ADV of the affected SG only. The operator will be required to identify the blocked ADV, initiate actions to unblock the ADV of the intact SG, and isolate the affected SG to mitigate the release of radioactivity to the environment. After the operator isolates the affected SG, the operator will continue cooling down the RCS using the intact SG. The affected SG level will be maintained by using backflow to the RCS. The operator continues the cooldown until the shutdown entry conditions are reached.

The use of the affected ADV in this analysis is for the purpose of maximizing the radiological releases during the event.

14.15.2 SEQUENCE OF EVENTS AND SYSTEMS OPERATION

The sequence of events for a typical limiting case is presented in Table 14.15-2. Several cases were analyzed to examine the effect of time of reactor trip, initial SG pressure, AFW actuation and flow, subcooling, plugged tubes, and cooldown rate on radiological dose consequences. The results, in most cases, did not differ significantly and the sequence of events for the presented case utilizes several assumptions regarding system operation that are chosen to maximize the radiological doses. The operator actions assumed in the analysis are consistent with EOPs.

The analysis assumed a loss of forced circulation following the reactor trip, which results in higher hot leg temperature, higher fraction of the leak flow flashing into the affected SG, slower cooldown and RCS depressurization, and reduces the capability to cool down the plant via the unaffected SG. All of these effects result in higher doses.

No credit was taken in the analysis for operation of the turbine bypass valves to the condenser. All of the steam releases are assumed to be directly to the atmosphere via the MSSVs or the ADVs.

The SG blowdown is assumed to be unavailable for level control.

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere. Furthermore, minimum AFW flow was assumed based on the automatic action of the AFAS, which maximizes SG pressures and ADV releases to the atmosphere during the post-trip period prior to operator action.

The ADV of the unaffected or intact SG is isolated at the onset of the event. Therefore, initially, all of the heat removal is through the ADV of the affected SG. Also, the unblocking of the isolated ADV may comprise up to a 2 hour delay as personnel need to access the manual control station which is outside the Control Room or manually operate the ADV using the handweel.

The operator actions assumed in this analysis are consistent with the Calvert Cliffs EOPs. The first operator action is assumed at 15 minutes following the reactor trip. Subsequently, a time delay of two minutes between each discrete operator action is assumed. The major post-trip EOP analysis assumptions regarding operator actions are:

1. <u>Operate the ADV on the affected SG</u>: 15 minutes after reactor trip, the operator takes manual control of the ADV on the affected SG to prevent further cycling of the MSSVs.
2. <u>Take manual control of the AFW to the SGs</u>: Two minutes after opening the affected SG ADV, the operator takes manual control of the AFW flow to each SG, with flow initially delivered to both SGs.
3. <u>Stabilize the plant and maintain cold leg temperature</u>: The operator quickly diagnoses the event and stabilizies the RCS to a temperature which precludes a challenge to the MSSVs using the SG ADVs and AFW. The length of the stabilization period is assumed to be no more than 10 minutes from the time that the operator takes manual control of the ADVs. As a result of this diagnosis, the operator initiates action to unisolate the ADV of the intact SG, which is assumed to be isolated at this time. The actions may take up to 1 hour after taking control.
4. <u>Cool the RCS before isolating the Affected SG</u>: After the stabilization period, the operator begin to cool the RCS at a rate of up to 100°F/hr to maximum steam releases.
5. <u>Isolate the Affected SG</u>: The operator isolates the affected SG when T_{HOT} is less than 515°F (including uncertainties). The analysis assumes no opening of the ADV or MSSVs of the affected SG after 2 hours. However, the ADV of the affected SG may be opened 24 hours into the accident to hasten shutdown.
6. <u>Plant cooldown after isolation of the affected SG</u>: Following the isolation of the affected SG, the operator cools down the plant using the ADV on the intact SG at a maximum of 35°F/hr to maximize steam releases.
7. <u>Maintain SG pressure and level</u>: The pressure and level of the affected SG will initially be controlled by steaming to atmosphere for up to 2 hours. In addition, the RCS will be aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.
8. <u>Maintain subcooling margin during the event</u>: A target subcooling margin of 50°F is maintained by the operator. This value consists of 25°F required by the EOPs and 25°F of core exit thermocouple uncertainty.
9. <u>Maintain pressurizer level</u>: The pressurizer level is maintained by controlling safety injection flow. In addition, the RCS is aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.
10. <u>Pressurizer control actions and control systems</u>: The operator uses the HPSI system and the pressurizer vent (or auxiliary spray) to control RCS inventory and subcooling.

The combination of the assumed cooldown rate and the high subcooling margin including instrument uncertainties result in a conservatively slow depressurization of the RCS, which maximizes the tube leakage. The increased leak rate raises the final activity level released through the affected SG. It also leads to a high liquid level in the SG early in the event resulting in the opening of the affected SG ADV and more frequent releases to the environment. However, at 2 hours into the event, the affected SG is completely isolated. Thus, the affected SG level is maintained by using backflow to the RCS. The ADV steaming is increased by the assumption of a lower actual SG level to accommodate instrument uncertainties.

Together, these assumptions, in combination with the radiological assumptions presented in Section 14.15.3.2, assure that the radiological dose results from the analysis conservatively bound the expected doses for this event.

14.15.3 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.15.3.1 Core and System Performance

A. Mathematical Models

The thermal hydraulic response of the NSSS to the SGTR was simulated using the Reference 6 computer program up to the time the operator takes control of the plant (15 minutes after trip). Operator actions to mitigate the effects of the SGTR event and bring the plant to shutdown cooling entry conditions were simulated using a CESEC-based cooldown algorithm, referred to as the COOL code.

B. Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.15-1 for the present cycles of Unit 1 and Unit 2. The selected values of these inputs maximize the radiological releases to the atmosphere during the transient.

The maximum allowed Technical Specification core inlet temperature, including instrument uncertainties, results in a correspondingly high initial SG pressure. This increases the steam released through the MSSVs and the ADVs throughout the event.

The minimum core flow results in higher than average coolant temperature and higher enthalpy fluid entering the SG, a resultant increase in flashing fraction, and higher activity releases through the MSSVs and ADVs.

A maximum initial pressure and a maximum initial pressurizer liquid volume, delay the reactor trip. Delaying reactor trip is conservative because it increases the amount of heat to be removed and increases steam releases.

The SG level is maintained within a small range during operation, the limits of which would have no effect on the trip time and insignificant effect on the AFW actuation time.

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere.

The selection of fuel and moderator temperature coefficients are not significant, as there is no change in the core power or temperature prior to reactor trip. The TM/LP trip uncertainty is applied to lower the setpoint to

delay trip. Three HPSI pumps are assumed to be started on SIAS, thus maximizing the flow delivered to the RCS upon SIAS. These assumptions result in higher post-trip RCS pressures, and maximize the tube leakage.

The radiological consequences of the SGTR event are also dependent on the break size. As the break size is decreased from that of a double-ended rupture, the integral leak is reduced and the radiological consequences will be less severe. Therefore, the most adverse break size is the largest assumed break of a full double-ended rupture of a SG tube.

C. Results

Table 14.15-2 presents the sequence of events for the double-ended rupture of a SG tube event with the loss of forced circulation upon reactor trip. Figures 14.15-1 through 14.15-16 present the dynamic behavior of important NSSS parameters during this event. The only scenario presented is the one that assumes isolation of the affected SG 2 hours into the transient while maintaining the highest subcooling possible by accounting for core exit thermocouple uncertainty.

The sequence of events and NSSS response plots are based on the RSG configuration.

The double-ended break of a SG tube results in a primary-to-secondary leak rate which exceeds the capacity of the charging pumps. As a result, pressurizer level and pressure gradually decrease from their initial values. For the case discussed here, maximum charging flow and zero letdown was assumed to delay the time of reactor trip. As the pressure decreases, the proportional heaters and then backup heaters are turned on to prevent further depressurization. All heaters are turned off automatically as the pressurizer level is decreasing to levels which result in uncovery of the heaters. The depressurization of the RCS and pressurizer level decrease continue, resulting in an approach to DNB SAFDL. The TM/LP trip is designed to trip the reactor before the DNB SAFDL is reached. The analysis of the SGTR event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded, since the rate of depressurization for this event is less than the rate of depressurization for the RCS Depressurization event. The analysis credits a reactor trip only when the low pressurizer pressure floor of the TM/LP trip is reached. The loss of forced circulation (RCP pumps tripping) is assumed to occur 3 seconds after the trip breakers are opened, resulting in the initiation of the RCS flow coastdown.

The analysis also assumes the steam bypass system to the condenser will become unavailable and that the unaffected SG ADV is blocked for 60 minutes into cooldown. The affected SG ADV automatically opens at trip time and then modulates on a program based on RCS average temperature. The turbine valve closure due to the reactor trip causes the SG pressures to rise, and leads to the opening of the MSSVs. They reopen and close several times during the period until the operator takes action to cool the plant.

The loss of forced circulation and the RCS flow coastdown result in reduction of flow into the upper head region of the reactor vessel. This region becomes thermal-hydraulically decoupled from the rest of the RCS,

and due to flashing caused by the depressurization and boiloff from the metal structure to coolant heat transfer, voids begin to form in this region.

The pressurizer empties due to the continued primary-to-secondary leak and the post-trip RCS liquid shrinkage. The continued RCS and pressurizer depressurization results in SIAS generation and delivery of the HPSI flow to the RCS when the RCS pressure decreases below the HPSI pump head.

The AFW actuation setpoint is reached in the unaffected SG and the AFW is delivered to both SGs following system and piping delays.

Fifteen minutes following the trip, the operator takes manual control of the plant, which consists of manual control of ADVs, AFW and HPSI. The analysis of the limiting case assumes that at this point the operator has diagnosed the event.

Following the diagnosis, the operators begin to cooldown the RCS at approximately 100°F/hr, using the ADV on the affected SG and the AFW system until the hot leg temperature of the affected loop reaches an isolation temperature of 493.21°F (515°F per EOPs minus 21.79°F uncertainty).

14.15.3.2 Radiological Consequences

The limiting SGTR event as re-analyzed by Reference 4 is considered to be a complete double-ended tube break. The SGTR event allows primary coolant to leak into the secondary side via the SG. In the case of the double-ended tube rupture, the leak rate far exceeds the charging pump capacities and, consequently, the pressurizer level decreases. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The drop in the pressure will cause a rector trip on TM/LP, ensuring that the DNB SAFDL is not exceeded. Peak linear heat rate is of no concern because there is no appreciable power increase during the transient. Thus, no fuel damage is postulated to occur during this event. The reactor trip also generates a turbine trip causing the secondary pressure to rapidly increase due to closure of the turbine valve. In the assumed evolution, the turbine bypass valves are not available to mitigate the rise in secondary pressure. The action of the ADVs and MSSVs will limit the secondary pressure until the operator is able to assume control. After the operator identifies the event, the operator initiates a cooldown of the RCS. In this analysis, the ADV of the intact SG is assumed to be isolated at the beginning of the event for up to 2 hours. Thus, this initial cooldown is carried out using the ADV of the affected SG only. After 2 hours, the operator isolates the affected SG and continues cooling down the RCS using the intact SG. The affected SG level will be maintained by using backflow to the RCS. The operator continues the cooldown via the ADV of the unaffected SG until the SDC entry conditions are reached. A 30 day cooldown via the ADV of the unaffected SG is conservatively assumed. Note that the operators can reopen the ADV of the affected SG for up to 8 hours after an initial cooldown of 24 hours post-accident to attain SDC in 32 hours post-accident.

The AST methodology of 10 CFR 50.67 and Reference 1 is used to calculate offsite and Control Room doses for a SGTR event. If no or minimal fuel damage is postulated, the activity is the maximum coolant activity allowed by the Technical Specifications, assuming 2 cases of iodine spiking. The PIS case assumes that a reactor transient has occurred prior to the postulated SGTR event and has raised

the primary coolant iodine concentration to the maximum value permitted by the Technical Specifications, 30 μCi/gm. The CIS case assumes that the transient associated with the SGTR event causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value with an 8 hour duration.

A. Assumptions and Conditions

The assumptions and parameters employed for the evaluation of radiological releases are:

(1) CIS doses are calculated assuming that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (0.5 μCi/gm DEQ I-131 activity). The primary CIS activities are released homogeneously into the primary system over the 8 hour duration of the CIS spike.

(2) PIS doses are calculated assuming that a reactor transient has occurred prior to the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value permitted by the Technical Specifications: 30 μCi/gm. The primary PIS activities are assumed to be homogeneously distributed throughout the primary system at the beginning of the accident.

(3) The specific activity of the primary coolant is assumed to be 100/E μCi/gm noble gas per Technical Specifications.

(4) An initial DEQ I-131 secondary activity of 0.1 μCi/gm is assumed (Technical Specification limit). The secondary activities are assumed to be homogeneously distributed throughout the secondary system at the beginning of the accident.

(5) The dose conversion factors were extracted from References 2 and 3.

(6) The iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic.

(7) The main Control Room inleakage points include the West Road inlets, the Turbine Building, and Access Control Units 11 and 13 on the Auxiliary Building roof. Installation of automatic isolation dampers and radiation monitors at Access Control Units 11 and 13 on the Auxiliary Building roof were credited.

- A Control Room inleakage rate of 3500 cfm was based on measured inleakage measurements.
- Control Room recirculation filtration is credited assuming 10,000 ± 10% cfm flow at 90% filter efficiency for elemental and organic iodine and 99% for particulates with a 20 minute delay time.
- 0-8, 8-24, and 24-720 hour breathing rates of 3.5E-04, 1.8E-04, and 2.3E-04 m^3/sec are assumed.
- 0-24, 24-96, and 96-720 hour Control Room occupancy factors of 1.0, 0.6, and 0.4 are assumed.

(8) The primary to secondary ruptured tube leakage and Technical Specification leakage of 200 gpd are assumed to continue until SDC conditions defined as 300°F and 270 psia are attained and releases from the SGs have been terminated. Per Reference 1, the Technical Specification leakage should be apportioned between affected and unaffected SGs in such a manner that the calculated dose is

maximized. Thus, since the primary to secondary flow from the RCS to the affected SG was maximized in Reference 4 for the worst-case thermal-hydraulic conditions, all of the Technical Specification primary to secondary leakage is assumed to flow to the unaffected SG.

(9) The portion of the primary fluid leaking into the SG that flashes into steam is dependent on the enthalpy of the primary liquid and the saturation enthalpy of the SG. When there is a steam release to the atmosphere, the flashed portion is released before the steam in the SG. The flashing portion has a decontamination factor of 1.0. The non-flashing portion of the primary leak flow is assumed to mix uniformly with the liquid in the SG.

(10) The SG is assumed to have a decontamination factor of 100, so that the concentration of radioactivity in the steam phase is 1/100 of the concentration in the liquid phase.

Additional inputs and assumptions are detailed in Table 14.15-3.

B. Mathematical Model

The behavior of the primary and secondary systems during and after a double-ended tube break SGTR event was modeled by Reference 4. The CESEC-III NSSS simulation code was used to model the SGTR for primary and secondary response during the initial portion of the event. However, CESEC-III does not have the capability to model the multiple operator actions credited in the SGTR event. Thus, the remainder of the event was simulated using the COOL-II code, which can model explicit operator actions. The COOL-II Code is a thermal-hydraulic code that simulates the plant cooldown by operator actions based upon the Calvert Cliffs EOPs. Because the COOL-II code does not have a kinetics model, CESEC-III is run to approximately 15 minutes past reactor trip to ensure all power being generated is from decay heat and a conservative decay heat curve is input to COOL-II.

The SGTR occurs at a time t=0 with the PIS primary activity and the Technical Specification secondary activity uniformly distributed throughout their respective systems. The SGTR occurs at a time t=0 with the Technical Specification secondary activity uniformly distributed throughout the secondary system and with the CIS primary activity released homogeneously into the primary system over an 8 hour duration. The primary noble gases are released at a 200 gpd rate into the unaffected SG and at the time-dependent tube rupture leak rate into the affected SG and then directly through the ADVs and MSSVs into the environment, when the ADVs and MSSVs are in the open position. The primary iodines are released at a 200 gpd rate into the unaffected SG and at the time-dependent tube rupture leak rate into the affected SG, where a percentage is vented directly through the ADVs and MSSVs into the environment via flashing. The remaining iodines are added to the secondary system, which is released by steaming with a partition factor of 100 out of the ADVs, when the ADVs and MSSVs are in the open position. No cleanup mechanisms (spray, filtration, plateout) are assumed in the primary or secondary systems. The activity released to the environment is transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients. Control Room filtration is credited in this analysis. The Control Room and site boundary doses are calculated based on appropriate breathing rates and occupancy factors and on References 2 and 3 dose conversion factors.

The Control Room and offsite doses are calculated for the SGTR event based on the AST methodology of Reference 1. This was accomplished by utilizing the RADTRAD computer transport code. The RADTRAD computer code calculates TEDE and thyroid doses to personnel at the site boundary, low population zone, and Control Room per 10 CFR 50.67 resulting from any postulated accident which releases radioactivity within any primary or secondary system. RADTRAD models the transport of up to 63 radionuclides from the source region, through a secondary region, and then to the environment and to the Control Room. The code includes the capability to model time-dependent activity release; time-dependent spray/filtration/deposition removal processes, piping/filter/inleakage transfer mechanisms, atmospheric dispersion; and natural decay.

C. Results

The EAB, LPZ, and Control Room doses for the design-basis CIS and PIS SGTR event for the two cooldown modes described previously are detailed in the following table:

SGTR Event Results

	EAB Rem	LPZ Rem	Control Room Rem
CIS			
Unaffected ADV 0-30 days	0.1964	0.0484	1.7081
Affected ADV 0-2/24-32 hr	0.1964	0.0476	1.6929
Regulatory Limits	2.5	2.5	5
PIS			
Unaffected ADV 0-30 days	0.4910	0.1164	4.1590
Affected ADV 0-2/24-32 hr	0.4910	0.1162	4.1655
Regulatory Limits	25	25	5

Note that all values are below the regulatory limits.

14.15.4 CONCLUSION

The analysis of the SGTR event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded. All doses are within 10 CFR 50.67 and Reference 1 limits, as approved by Reference 5.

This event is not affected by the transition to AREVA Advanced CE-14 HTP™ fuel because the key parameters for this event are plant related system responses which are unchanged from, or bounded by, the current analysis. These parameters are not adversely affected by either the transition cycle or full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

14.15.5 REFERENCES

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

2. Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988

3. Federal Guidance Report (FGR) 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993

4. CA06595, Westinghouse Calculation CN-TAS-05-13, Revision 000, Calvert Cliffs Units 1 & 2 Steam Generator Tube Rupture Event

5. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846)," dated August 29, 2007

6. CESEC-III, Mod 5 computer program (ABB Topical Report "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System" Enclosure 1-P to LD-82-001, December, 1981

TABLE 14.15-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	UNITS	UNIT 1[(a)]	UNIT 2[(a)]
Core Power	MWt	2754	2754
T_{in}	°F	550	550
RCS Pressure	psia	2286	2286
SG Tubes Plugged		2500	2500
Core Mass Flow Rate	$x10^6$ lbm/hr	134.0	134.0
Secondary Pressure	psia	890.5	890.5
Tube ID	inches	0.654	0.654
Pressurizer Liquid Level at Full Power	ft^3	952	952
Low Pressurizer Pressure (TM/LP Floor) Setpoint	psia	1829	1829
Safety Injection Actuation (SIAS) Setpoint	psia	1765	1765

(a) These values represent inputs to the limiting transient scenario analyzed for each unit. In general, a range of initial conditions and input parameters, including uncertainties, were evaluated to determine the limiting case.

TABLE 14.15-2

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME	EVENT	SETPOINT OR VALUE
0.0	Tube Rupture Occurs	---
8.4	Proportional Pressurizer Heaters Setpoint Reached, psia	2275
66.4	Backup Pressurizer Heaters Setpoint Reached, psia	2200
348.4	Pressurizer Heaters De-energize due to Low Pressurizer Level, ft^3	270
417.8	Low Pressurizer Pressure Trip Analysis Setpoint is Reached, psia	1829
418.7	Trip Breakers Open	---
	ADVs Open, °F	535
420.8	MSSVs Open, psia	935
421.7	Loss of Forced Circulation, RCPs Begin to Cost Down	
426.2	Maximum SG Pressure is Reached, psia	986
430.9	SIAS Setpoint is Reached, psia	1765
438.3	Pressurizer Empties	---
456.7	MSSVs Close, psia	878
	The MSSVs subsequently cycle repeatedly	
478.4	Safety Injection Flow Begins to Enter the RCS, psia	1351
749.9	AFW Actuation Setpoint is Reached Unaffected SG	204” BNL
1018.5	AFW is Initiated to Unaffected SG	100 gpm
1318.7	Operator Takes Manual Control of the Plant and Begins Cooldown at Rate of 100°F/hr by Adjusting the ADVs on the affected SG	---
1438.7	AFW Increase to Both SGs (2 minutes past takeover time)	200 gpm/SG
1800	Operator Opens the Pressurizer Vent	
2270	Hot Leg Reaches Isolation Temperature, °F	493.21
2280	Adequate Pressurizer Level, Inches (Operator Begins to Throttle HPSIs)	101
5040	Operator Unblocks ADV of Intact SG	

TABLE 14.15-3

ASSUMPTIONS FOR RADIOLOGICAL CONSEQUENCES OF THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	DESIGN BASIS ASSUMPTION
Primary system activity:	
Pre-existing iodine spike (PIS), μCi/gm	30
Event GIS, μCi/gm	0.5
Spiking factor	335
Secondary system activity, μCi/gm	0.1
Primary-to-secondary leak rate in the unaffected SG, gpd	200
EAB Atmospheric Dispersion factor (X/Q) sec/m^3, 0 - 2 hr	1.44×10^{-4}
LPZ Atmospheric Dispersion Factor (X/Q), sec/m^3	
0 - 2 hr	3.39×10^{-5}
2 - 24 hr	2.2×10^{-6}
24 - 720 hr	5.4×10^{-7}
Decontamination factor between the water and steam phases in the SGs	100
Breathing rate, m^3/sec	
0 - 8 hr	3.5×10^{-4}
8 - 24 hr	1.8×10^{-4}
24 - 720 hr	2.3×10^{-4}
Control Room Atmospheric Dispersion Factor (χ/Q), sec/m^3	
0 - 2 hr	3.83×10^{-3}
2 - 8 hr	3.25×10^{-3}
8 - 24 hr	1.32×10^{-3}
1 - 4 days	9.92×10^{-4}
4 - 30 days	7.92×10^{-4}

14.16 SEIZED ROTOR EVENT

14.16.1 IDENTIFICATION OF EVENT AND CAUSE

The primary function of the RCPs is to provide forced coolant flow through the reactor core. There are four RCPs in the RCS which are located in the SG cold legs. The RCS is a two-loop two-SG system with four cold legs.

The shaft seal system for the RCPs consists of three sets of four similar, rubbing face seals mounted in a cartridge, plus a fourth rubbing face, low pressure, vapor seal mounted on top of the cartridge.

Each of the four seals consists of a rotating tungsten carbide ring riding over a carbon graphite stationary face. The reactor coolant leaking into the seal cavity is forced through the seal water cooler by the auxiliary impeller where it is cooled with component cooling water. The seals are kept at approximately 145°F by maintaining a controlled flow of this cooled reactor coolant through the seal cartridge.

Each pump is equipped with renewable casing wear rings, four tandem mechanical face seals, and a self-aligning, water-lubricated bearing mounted above the pump impeller. Additional shaft support is provided by the pump motor bearings. The pump motor-driver and pump shaft are connected with a rigid coupling. The pump impeller and pump shaft are bolted and pinned together to allow differential thermal expansion. The pump and motor are furnished as a unit complete with an oil lift pump, oil cooler, seal water cooler, and instrumentation. The instrumentation will alert the operators to any incipient failures in the pump motors or seals.

Non-reverse rotation devices are provided on the pump motors to prevent the pump from windmilling in the reverse direction, and to limit backflow through a stopped pump from thereby bypassing the core.

A Seized Rotor event is defined as a complete seizure (i.e., binding) of a single RCP shaft. The seizure is postulated to occur due to a mechanical failure or a loss of component cooling to the pump shaft seals.

The most limiting Seized Rotor event is an instantaneous RCP shaft seizure at HFP. The reactor coolant flow through the core would be asymmetrically reduced to three pump flow as the result of a shaft seizure on one pump.

14.16.2 SEQUENCE OF EVENTS

A Seized Rotor event can result in an approach to the fuel SAFDLs, to the RCS Pressure Upset Limit, and to the site boundary dose criteria in 10 CFR 50.67 guidelines. The action of the LSSS in conjunction with the LCOs will limit the number of fuel pins that experience DNB for a short period of time.

A Seized Rotor event is initiated at HFP from within the LCOs by an instantaneous complete seizure of a single RCP shaft. The immediate system response is a rapid reduction to three pump flow. With the loss of one RCP, the core inlet flow is non-uniform. The core inlet flow from the cold leg of the seized pump shaft is reduced.

Due to the reduction in core flow, the core coolant temperatures up the core will increase. Assuming a positive MTC, the core power will increase. Normally, the MTC is negative at power which will start shutting down the core. The core average heat flux will decrease slightly due to the increasing core temperatures. After the RPS response time and CEA holding coil delay time has elapsed, the insertion of the CEAs will terminate the power rise. The heat flux will then be reduced to the decay heat level.

Assuming the event initiated at the most adverse DNBR condition in conjunction with an instantaneous RCP seized shaft, a limited number of fuel pins may experience DNB for a short period of time. Since the time of minimum DNBR occurs in less than two seconds, the events occurring on the secondary side (i.e., SG) will not affect the results.

14.16.3 CORE AND SYSTEM PERFORMANCE

14.16.3.1 Mathematical Models

The transient response of the RCS and steam systems to the Seized Rotor event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 5. The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using the NRC-approved DNB correlation. The overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation. The limiting design axial power profile (a top peaked axial power distribution) was used for the simulation. Both of these computer codes are described in Section 14.1.4.1.

14.16.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.16-1. Those parameters which are unique to the analysis are discussed below.

A three pump, core inlet, maldistribution factor is used in the analysis to account for reduced flow into the assemblies near the core inlet cold leg with the seized pump shaft. The three pump core mass flow rate is the total core flow assuming three pump operation. The analysis accounted for core bypass.

The asymmetric core inlet flow distribution dictated in Reference 6 is used to perform the Seized Rotor event analysis in accordance with the methodology described in Reference 5. The inlet flow factors are depicted in Reference 7.

Reactor trip for the Seized Rotor event was initiated by a low coolant flow trip. The RPS trip setpoint was adjusted to account for uncertainties and time delays. The initiation of scram was delayed after the setpoint was reached to account for time delays including initiation of control rod movement.

The key transient parameters, which are impacted by extended burnup, are MTC, FTC and radioisotope gap concentrations. The BOC most-positive MTC Technical Specification value independent of power level was used for this analysis. The event occurs quickly and is not sensitive to other neutronic parameters. The BOC HFP nominal fuel temperature coefficient (biased less negative) was used. In addition, the radioisotope gap concentrations assumed in calculating site boundary doses correspond to a burnup of 62,000 MWD/MT. Hence, extended burnup effects have been conservatively included in the analysis of this event.

14.16.3.3 Results

Table 14.16-2 contains the sequence of events for the Seized Rotor event. Figures 14.16-1 through 14.16-4 present the transient behavior of the core power, core average heat flux, RCS temperatures, and RCS pressure.

The S-RELAP5 plant simulation results from the analysis of the Seized Rotor event were used as input into minimum DNBR calculations. The plant simulation

data were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations. The MDNBR was above the HTP DNB correlation upper 95/95 limit plus a 2% mixed core penalty. This event does not challenge the FCM SAFDL.

The potential for DNB propagation during the Seized Rotor event was not evaluated because no fuel rods are expected to experience DNB.

14.16.4 DOSE ANALYSIS

A Seized Rotor event is initiated at HFP by an instantaneous complete seizure of a single RCP shaft. With the reduction of core flow due to the loss of an RCP, the core coolant temperatures will increase. Assuming a positive MTC, the core power will increase. The core average heat flux will decrease slightly due to the increasing core temperatures. The insertion of the CEAs due to a low RCS flow trip will terminate the power rise; however, a maximum of 5% of fuel pins may experience DNB for a short period of time and are assumed to fail. The initial secondary activity, together with initial primary activity, and failed fuel activity released to the primary that then leaks into the secondary at the Technical Specification limit of 200 gpd, will escape out of the SGs via the ADVs and condenser. Note that per the requirements of Reference 4, the release of fission products from the secondary system should be evaluated with the assumption of a coincident LOOP. Thus, the use of condensers is not credited.

The Seized Rotor event dose analysis conforms to the regulatory requirements of 10 CFR 50.67 and Reference 4 using AST methodology.

For the 8-hour secondary release path scenario, the 16% I-131, 10% other iodines, 20% Kr-85, 10% other noble gases, and 24% alkali metals which are contained in the gas gaps of the fuel which experience clad failure, are released into the primary system, which is then transmitted into the secondary system via the Technical Specification SG tube leakage. Environmental releases occur from both SGs via the ADVs. The SG tubes remain covered for the duration of the event; therefore the gap iodines have a partition coefficient of 100 in the SGs. A conservative flashing fraction is assumed (10% for the first 15 minutes and 1% thereafter); however, no credit for scrubbing in the SG is assumed. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. A value of 0.001 was conservatively assumed. The steam release from the first 1800 seconds was taken directly from a CESEC analysis. The steam release from 1800 seconds to 8 hours is based on a simple energy balance methodology; that is, the steam released from 1800 seconds to 8 hours is based on the amount of steam required to remove the residual heat from the primary and secondary systems, the decay heat generated in the core, and the RCP heat. Radionuclide transport was modeled by the RADTRAD computer code.

The activity released to the environment is transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients. A constant Control Room inleakage of 3500 cfm was assumed in this work. Control Room filtration is credited based on a nominal flow of 10,000 $\pm$ 1,000 cfm per train. A charcoal filter efficiency of 90% is credited for elemental and organic iodine, while a HEPA efficiency of 99% is credited for particulate iodine. The Control Room and site boundary doses are calculated based on appropriate breathing rates and occupancy factors and on References 1 and 2 dose conversion factors. Additional inputs are included in Table 14.16-3.

The EAB, LPZ, and Control Room doses for the design-basis Seized Rotor event are detailed in the following table. Note that all values are below the regulatory limits.

Seized Rotor Event Results

Results	EAB Rem	LPZ Rem	Control Room Rem
8 hour Secondary Pathway	0.041	0.0095	0.7885
Regulatory Limits	2.5 (Reference 4)	2.5 (Reference 4)	5.0 (Reference 3)

14.16.5 CONCLUSION

For the case of the Loss-of-Coolant Flow resulting from a seizure of a RCP shaft, the Low Flow Trip in conjunction with the DNB LCOs results in no fuel rod expected to experience DNB. For an assumed maximum failure of 5% of the fuel pins, the resultant site boundary doses are well within the 10 CFR 50.67 limits. In addition, the maximum RCS pressure experienced during the event is expected to be well below the upset pressure limit of 2750 psia.

The analysis of this event explicitly included the effects of extended burnup. Since the site boundary doses are within 10 CFR 50.67 limits, it is concluded that the consequences of the Seized Rotor event are acceptable at extended burnup. Departure from nucleate boiling propagation was not evaluated since no fuel pins are expected to experience DNB.

14.16.6 REFERENCES

1. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988
2. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993
3. 10 CFR 50.67, “Accident Source Term”
4. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000
5. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004
6. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), “Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832),” February 18, 2011
7. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), “Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel,” December 30, 2010

TABLE 14.16-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR SEIZED ROTOR EVENT

PARAMETER	UNITS	VALUE
Initial Core Power Level	MWt	2754
Core Inlet Coolant Temperature	°F	548
4-Pump Vessel Mass Flow Rate	gpm	370,000
3-Pump Vessel Mass Flow Fraction	Frac. of 4-Pump	Approximately 0.75[a]
RCS Pressure	psia	2250
ASI (Range)	---	Top Peaked Design Axial
F_r	---	1.65

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[a] As established by the S-RELAP5 initialization.

TABLE 14.16-2

SEQUENCE OF EVENTS FOR SEIZED ROTOR EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	Seizure of One RCP	---
0.205	Low Coolant Flow Trip Analysis Setpoint is Reached	90% of Initial 4-Pump Flow
0.704	Trip Breakers Open	---
1.204	CEAs Begin to Drop Into Core	---
1.55	Minimum DNBR Occurs	> 1.164

TABLE 14.16-3

ASSUMPTIONS FOR SEIZED ROTOR DOSE CALCULATION

1. Initial thermal power is 2754 MWt.
2. The pin power peaking factor is 1.70.
3. The failed fuel fraction is 5%.
4. The damaged fuel rods are assumed to release their gas gap activities consisting of 16% I-131, 10% other iodines, 20% Kr-85, 10% other noble gases, and 24% alkali metals instantaneously and homogeneously throughout the primary system at the initiation of the accident. The primary Technical Specifications activities of 0.5 microCi/gram DEQ I-131 and 100/Ebar microCi/gram noble gas are assumed to be homogeneously distributed throughout the primary system at the beginning of the accident. The secondary Technical Specification activities of 0.1 microCi/gram DEQ I-131 are assumed to be homogeneously distributed throughout the secondary system at the beginning of the accident.
5. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic.
6. The primary to secondary leakage of 200 gpd per Technical Specification 3.4.13 is assumed to continue until the primary system pressure is less than the secondary system pressure or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity is assumed to continue until SDC is in operation and releases from the SGs have been terminated. For the secondary system release pathway, the duration of the cooldown from HFP, defined as 574.5°F and 2250 psia to SDC, defined as 300°F and 270°psia per the EOPs, is assumed to be 8 hours.
7. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. A value of 0.001 was conservatively assumed.
8. The breathing rates and Control Room occupancy factors are per Reference 4.
   - Breathing rate:
     - 3.5E-4 m^3/s for 0-8 hours
     - 1.8E-4 m^3/s for 8-24 hours
     - 2.3E-4 m^3/s for 24-720 hours
   - Occupancy factors:
     - 1.0 for 0-24 hours
     - 0.6 for 24-96 hours
     - 0.4 for 96-720 hours
9. The ADV to site boundary 2-hour, atmospheric dispersion coefficient is 1.44E-4 sec/m^3.
10. The atmospheric dispersion coefficients from the ADV to the Control Room are:
    - 3.83E-3 sec/m^3 for 0-2 hours
    - 3.25E-3 sec/m^3 for 2-8 hours
    - 1.32E-3 sec/m^3 for 8-24 hours
    - 9.92E-4 sec/m^3 for 24-96 hours
    - 7.92E-4 sec/m^3 for 96-720 hours

14.17 LOSS-OF-COOLANT ACCIDENT

14.17.1 INTRODUCTION AND SUMMARY

Title 10 CFR 50.46 (Reference 1) provides the acceptance criteria for Emergency Core Cooling Systems (ECCS) for light water nuclear power reactors. The ECCS performance analyses presented in this section demonstrate that the Calvert Cliffs Units 1 and 2 ECCS design satisfies these criteria.

Sections 14.17.2 and 14.17.3 describe the analyses for the large break LOCA and the small break LOCA, respectively. Sections 14.17.4.1 and 14.17.4.2 describe the ECCS performance of the current cycles for Units 1 and 2.

The ECCS performance analyses were performed for a spectrum of large and small break LOCA break sizes. The limiting break size, i.e., the break that results in the highest peak cladding temperature, was identified as the 0.09 ft^2 break in the cold leg pump discharge piping. The results of the analysis demonstrate that, for a PLHGR of 15.0 kW/ft the ECCS design meets the 10 CFR 50.46 Acceptance Criteria. Conformance is as follows:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The ECCS performance analysis yielded a peak cladding temperature of 1626°F for the 0.09 ft^2 break. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The ECCS performance analysis yielded a maximum cladding oxidation of 0.02460 times the total thickness before oxidation for the Double-Ended Guillotine break of 4.5832 ft^2/side.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The ECCS performance analysis assures the core remains amenable to cooling from the effects of fuel cladding rupture and swelling, and the effects of LOCA. The analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Since Criteria 1 and 2 are satisfied for the hot pin, it is clear that the hot pin remains amenable to cooling. It is therefore concluded that the remainder of the core also remains amenable to cooling. Therefore, the analysis demonstrates a coolable geometry.

Criterion (5) Long-Term Cooling. "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The ECCS performance analysis showed that the rapid insertion of borated water from the safety injection tanks (SITs) and the SI pumps suitably limited the peak cladding temperature and cooled the core within a short period of time. Subsequently, the SI pumps will continue to supply cooling water from the refueling water tank or the containment sump.

14.17.2 LARGE BREAK LOCA ANALYSIS

The purpose of the large break LOCA analysis is to verify typical Technical Specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10 CFR 50.46(b) criteria are met:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The ECCS performance analysis yielded a peak cladding temperature of 1620°F for the Double-Ended Guillotine break of 4.5832 ft^2/side. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The ECCS performance analysis yielded a maximum cladding oxidation of 0.02460 times the total thickness before oxidation.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The ECCS performance analysis assures the core remains amenable to cooling despite the effects of fuel cladding rupture and swelling, and the effects of LOCA. The realistic large break LOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Therefore, the analysis demonstrates compliance with Criterion 4.

Criterion (5) Long-Term Cooling. "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The ECCS performance analysis showed that the rapid insertion of borated water from the SITs and the SI pumps suitably limited the peak cladding temperature and cooled the core within a short period of time.

Subsequently, the SI pumps will continue to supply cooling water from the refueling water tank or the containment sump.

14.17.2.1 Event Description

A large break LOCA is initiated by a postulated large rupture of the RCS primary piping. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The break initiates a rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the large break LOCA analysis. The reactor is shutdown by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience DNB. Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core criticality ends. As heat transfer from the fuel rods is reduced, the cladding temperature increases.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate and leads to a period of positive core flow or reduced downflow as the RCPs in the intact loops continue to supply water to the reactor vessel (in no-LOOP conditions). Cladding temperatures may be reduced and some portions of the core may rewet during this period. The positive core flow or reduced downflow period ends as two phase conditions occur in the RCPs, reducing their effectiveness. Once again, the core flow reverses as most of the vessel coolant inventory flows out through the broken cold leg.

Mitigation of the large break LOCA begins when the SIAS is initiated. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single failure be considered. This single failure has been determined to be the loss of one ECCS pumped injection train. The AREVA realistic large break LOCA methodology conservatively assumes an on-time start and normal lineup of the containment spray to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes the loss of a diesel generator, LPSI injection into the broken loop and one intact loop, HPSI injection into all four loops, and all containment spray pumps are operating.

When the RCS pressure falls below the SIT pressure, fluid from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will drive some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery continues relying solely on pumped ECCS injection. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperature created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the HPSI and LPSI while the decay heat continues to fall. Steam generated from fuel rod rewet will entrain

liquid and pass through the core, vessel upper plenum, the hot legs, the SGs, and the RCPs before it is vented out the break. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core-wide cooling. Full core quench occurs within a few minutes after core-wide cooling. Long-term cooling is then sustained with coolant provided by LPSI.

14.17.2.2 Evaluation Model

The realistic large break LOCA methodology is documented in Reference 2. The methodology follows the Code Scaling, Applicability, and Uncertainty evaluation approach (Reference 3). This method outlines an approach for defining and qualifying a best estimate thermal hydraulic code and quantifies the uncertainties in a LOCA analysis.

The realistic large break LOCA methodology consists of the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel cladding gap conductance.
- S-RELAP5 for the system calculation (includes ICECON for containment response).

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions, and that the dominant phenomena expected during the large break LOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the RCPs or the SG separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant Technical Specifications or to match measured data. Additionally, the RODEX3A code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 14.17.2.3.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into the loop containing the pressurizer. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Containment pressure is also calculated by S-RELAP5 and provides direct feedback for the pressure calculation using containment models derived from ICECON (Reference 4). The methods used in the application of S-RELAP5 to the large break LOCA are described in Reference 2.

The final step of the best estimate methodology is to combine all the uncertainties related to the code and plant parameters, and estimate the PCT at a high probability level. The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, base RODEX3A and S-RELAP5 input files for the plant (including the containment input file) are developed. Code input development guidelines are applied to ensure that model nodalization is consistent with the model nodalization used in the code validation.

2. Sampled Case Development

   The non-parametric statistical approach requires that many "sampled" cases be created and processed. For every set of input created, each "key LOCA parameter" is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant Technical Specifications or data). Those parameters considered "key LOCA parameters" are listed in Table 14.17-1. This list includes both parameters related to LOCA phenomena (based on the Phenomena Identification and Ranking Table provided in Reference 2) and to plant operating parameters.

3. Determination of Adequacy of ECCS

   The realistic large break LOCA methodology uses a non-parametric statistical approach to determine values of PCT at the 95% probability level. Total oxidation and total hydrogen are based on the limiting PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the criteria set forth in Section 14.17.2.

   The following are deviations from the approved realistic large break LOCA evaluation model (Reference 2) that were necessary to either correct or improve the calculation and/or to respond to additional information requested by the NRC. Each of these items has been approved for use at Calvert Cliffs until a revision to EMF-2103 is approved and implemented (Reference 8).

   - **Reactor Power** - The assumed reactor core power for the Calvert Cliffs realistic large break LOCA accident is 2754 MWt. This value represents the plant RTP (i.e., total reactor core heat transfer rate to the RCS) of 2737 MWt with a maximum power measurement uncertainty of 0.62% added to the RTP. The power was not sampled in the analysis.
   - **Rod Quench** - The realistic large break LOCA analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900°F before the rod is allowed to quench.
   - **Film Boiling Heat Transfer Limit** - The realistic large break LOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15% of the total heat transfer at and above a void fraction of 0.9.
   - **Break Size** - The split versus double-ended break type is no longer related to break area. In concurrence with Regulatory Guide 1.157, both the split and the double-ended break will range in area between the minimum break area (A_{min}) and an area of twice the size of the broken pipe. The determination of break configuration, split versus double-ended, will be made after the break area is selected based on a uniform probability for each occurrence. A_{min} was calculated to be 28.7% of the DEG break area.
   - **10 CFR Part 50, Appendix A, General Design Criterion 35 (Emergency core cooling) - LOOP and No-LOOP Case Sets** - In concurrence with General Design Criterion 35, a set of 59 cases was run with a LOOP assumption and a second set with a no-LOOP assumption. The results from both case sets are shown in Figure 14.17-17.

- **Cold Leg Condensation Efficiency** - During recent realistic large break LOCA modeling studies, it was noted that cold leg condensation efficiency may be under-predicted. Water entering the downcomer post-SIT injection remained sufficiently subcooled to absorb the downcomer wall heat release without significant boiling. However, tests (Reference 5) indicate that the steam and water entering the downcomer from the cold leg, subsequent to the end of SIT injection, reach near saturation resulting from the condensation efficiency ranging between 80 to 100%. To assure that cold leg condensation would not be under-predicted, a realistic large break LOCA evaluation model update was made. Noting that saturated fluid entering the downcomer is the most conservative modeling scheme, steam and liquid multipliers were developed so as to approximately saturate the cold leg fluid at the cold leg pressure before it enters the downcomer. Providing saturated fluid conditions at the downcomer entrance conservatively reduces both the downcomer driving heat and the core flooding rate. The test results indicate that fluid conditions entering the downcomer range from saturated to slightly subcooled. Hence, it is conservative to force an approximation of saturated conditions for fluid entering the downcomer. The NRC stated in Reference 8 that it finds this departure from the previously approved realistic large break LOCA methodology acceptable because (1) the artificially saturated fluid conditions will conservatively reduce both the downcomer driving head and the core flooding rate, which becomes conducive to portions of the fuel remaining in a vapor-cooled environment, thus presenting a greater challenge to clad surface cooling, and (2) conditions in the downcomer following SIT discharge are expected to be slightly subcooled, meaning that assuming fully saturated conditions is conservative.
- **RODEX3A Temperature Compensation** - AREVA Inc. has acknowledged an issue concerning fuel thermal conductivity degradation as a function of burnup as raised by the NRC (Reference 6). In order to manage this issue, AREVA Inc. is modifying the way RODEX3A temperatures are compensated in the Reference 2 realistic large break LOCA methodology. In the current process, the realistic large break LOCA computes PCTs at many different times during an operating cycle. For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady-state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then a two-transformation adjustment to the base fuel centerline temperature is computed. The first transformation is a linear adjustment for an exposure of 10 MWd/MTU or higher. In the new process, a polynomial transformation is used in the first transformation instead of a linear transformation. The rest of the realistic large break LOCA process for initializing the S-RELAP5 fuel rod temperature should not be altered and the rest of LOCA transient should also continue in the original fashion. This approach was accepted by the NRC for first-cycle AREVA fuel only (Unit 1 Cycle 21 and Unit 2 Cycle 19) in Reference 8.

  The NRC has concluded that a license condition is necessary to restrict plant operation to a single-cycle under the current AREVA large break LOCA analysis of record, and to obtain NRC review and

approval of a generic disposition concerning the analysis of only first-cycle fuel in light of the fuel thermal conductivity degradation issue with the RODEX3A code (Reference 8).

In response to the license condition, the realistic large break LOCA analysis has been updated to specifically model both first and second cycle fuel rods. Third cycle fuel does not retain sufficient energy potential to achieve significant cladding temperatures nor cladding oxidation and is not included in the realistic large break LOCA individual pin calculations. The burnup for the individual first and second cycle rods analyzed is assigned according to the sampled time in cycle. The time in cycle is sampled once and is the same for both the fresh (first cycle) and once-burnt (second cycle) fuel. Burnup for the fresh and once-burnt rods is different in accordance with the cycle management. Likewise, pin pressure and thermal conductivity differ.

In addition to the thermal conductivity and fuel temperature adjustments for burnup, a burnup dependent reduction in allowed peaking is needed for the once-burnt fuel. For first cycle fuel, the realistic large break LOCA methodology increases the F_r to the Technical Specification maximum (including uncertainty) for the first cycle hot rods in the model. Shortly into the cycle, once-burnt fuel has insufficient energy potential to achieve this peaking. A burnup dependent reduction in allowed peaking is therefore applied through an adjustment in the second cycle F_r. This approach was accepted by the NRC in Reference 9 for both first-cycle and burned AREVA Advanced CE-14 HTP™ fuel.

14.17.2.3 <u>Plant Description and Summary of Analysis Parameters</u>

The analysis presented here is for a Combustion Engineering-designed PWR, which has 2x4 loop arrangement. There are two hot legs each with a U-tube SG and four cold legs each with a RCP. The RCS includes one pressurizer connected to a hot leg. The core contains 217 fuel assemblies. The AREVA fuel assemblies contain 2, 4, 6, and 8 w/o Gadolinia pins. The ECCS includes one HPSI, one LPSI, and one SIT injection path per RCS loop. The break is modeled in the same loop as the pressurizer as directed by the realistic large break LOCA methodology. The realistic large break LOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e., RAS) for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/HPSI path per RCS loop. The HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary side SG that is instantaneously isolated (closed MSIV and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 10% per SG was assumed.

As described in the realistic large break LOCA methodology, many parameters associated with large break LOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of both phenomenological and plant parameters are given in Table 14.17-1. The large break LOCA phenomenological uncertainties are provided in Reference 2. Values for process or operational

parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 14.17-2. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Table 14.17-3 presents a summary of the uncertainties used in the analysis. Where applicable, the sampled parameter ranges are based on Technical Specification limits or supporting plant calculations that provide more bounding values.

For the realistic large break LOCA evaluation model, dominant containment parameters, as well as NSSS parameters, were established via a Phenomena Identification and Ranking Table process. Other model inputs are generally taken as nominal or conservatively biased. The Phenomena Identification and Ranking Table outcome yielded two important (relative to PCT) containment parameters - containment pressure and temperature. As noted in Table 14.17-3, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the containment volume (Table 14.17-3). Containment heat sink data and material thermal properties are given in Table 14.17-7. The heat transfer coefficients are calculated internally by S-RELAP5 during the transient and are variable based on the air/steam ratio in Containment. The Containment initial conditions and boundary conditions are given in Table 14.17-8. The containment spray is modeled at maximum heat removal capacity. All spray flow is delivered to the Containment.

14.17.2.4 Analysis of Results

Two case sets of 59 transient calculations were performed by sampling the parameters listed in Table 14.17-1. For each case set, a PCT was calculated for a UO_2 rod and for Gadolinia-bearing rods with concentrations of 2, 4, 6, and 8 w/o Gd_2O_3. The limiting case set containing the highest PCT corresponds to that with no offsite power available. A limiting PCT of 1620°F occurred in Case 47 for a fresh 8 w/o Gd_2O_3 rod. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The major parameters for the limiting transient are presented in Table 14.17-4. Table 14.17-5 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit. The best-estimate PCT case is Case 1, which corresponded to the median case out of the 59-case set with no offsite power available. The nominal PCT was 1424°F for an 8 w/o Gd_2O_3 rod. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 196°F.

The case results, event times, and analysis plots for the limiting PCT case are shown in Tables 14.17-5 and 14.17-6 and in Figures 14.17-6 through 14.17-16. Figure 14.17-1 shows linear scatter plots of the key parameters sampled for the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter ranges used in the analysis. Figures 14.17-2 and 14.17-3 show the time of PCT and break size versus PCT scatter plots for the 59 calculations, respectively. Figures 14.17-4 and 14.17-5 show the maximum oxidation and total oxidation versus PCT scatter plots for the 59 calculations, respectively. Key parameters for the limiting PCT case are shown in Figures 14.17-6 through 14.17-16. Figure 14.17-6 is the plot of PCT independent of elevation; this figure clearly indicates that the transient exhibits a sustained and stable quench. A comparison of PCT results from both LOOP and no-LOOP case sets is shown in Figure 14.17-17. As seen in Figure 14.17-17 the peak PCT is from the LOOP case.

14.17.2.5 Conclusions

A realistic large break LOCA analysis was performed using NRC-approved realistic large break LOCA methods (Reference 2). Analysis results show that the limiting LOOP case has a PCT of 1620°F and a maximum oxidation thickness of 2.46% fall well within regulatory requirements. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

The analysis supports operation at a nominal power level of 2754 MWt (including 0.62% uncertainty), a SG tube plugging level of up to 10% in all SGs, a LHGR of 15.0 kW/ft, a total peaking factor (F_q) up to a value of 2.37, and a nuclear enthalpy rise factor (F_r) up to a value of 1.81 (including 6% uncertainty) with no axial or burnup dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For large break LOCA, the 10 CFR 50.46(b) criteria presented in Section 14.17.2 are met and operation of Calvert Cliffs Units 1 and 2 with AREVA Advanced CE-14 HTP™ fuel is justified.

14.17.3 SMALL BREAK LOCA ANALYSIS

The purpose of the small break LOCA analysis is to verify typical Technical Specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10 CFR 50.46(b) criteria are met:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The small break LOCA ECCS performance analysis yielded a peak cladding temperature of 1626°F. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The small break LOCA ECCS performance analysis yielded a maximum cladding oxidation of 0.0177 times the total thickness before oxidation.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The small break LOCA ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The small break LOCA ECCS performance analysis assures the core remains amenable to cooling despite the effects of fuel cladding rupture and swelling. The small break LOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Since Criteria 1 and 2 are satisfied for the hot pin, it is clear that the hot pin remains amenable to cooling. It is therefore

concluded that the remainder of the core also remains amenable to cooling. Therefore, the analysis demonstrates compliance with Criterion 4.

14.17.3.1 Event Description

A postulated small break LOCA is defined as a break in the RCS pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP (Reference 7). The break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected out through the break. This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b) criteria (Reference 1).

The small break LOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a low pressurizer pressure signal. The SIAS occurs when the system has further depressurized. The capacity and shutoff head of the HPSI pumps are important parameters in the small break LOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is large enough that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the SIT pressure or to recover core liquid level on HPSI flow. This tends to maximize the heat up time of the hot rod which produces the maximum PCT and local cladding oxidation. Core recovery for the limiting break begins when the SI flow that is retained in the RCS exceeds the mass flow rate out the break, followed by injection of SIT flow. For very small break sizes, the primary system pressure does not reach the SIT pressure.

14.17.3.2 Evaluation Model

The small break LOCA evaluation model for the event response of the primary and secondary systems and hot fuel rod used in this analysis (Reference 7) consists of two computer codes. The two computer codes used in this analysis are:

- The RODEX2-2A code was used to determine the burnup-dependent initial fuel rod conditions for the system calculations.
- The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot fuel rod response.

The fuel-to-clad gap conditions used to initialize S-RELAP5 are taken at EOC, consistent with an EOC top-peaked axial power distribution.

This methodology has been reviewed and approved by the NRC to perform small break LOCA analyses for Calvert Cliffs in Reference 8 with the following restrictions and deviations from Reference 7:

1. Since the generic break spectrum model was shown to predict a non-conservative peak cladding temperature, the NRC staff concludes that a license condition is necessary to capture the more restrictive design criteria for Calvert Cliffs reload designs (Reference 8).

   The small break LOCA performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to the Calvert Cliffs core reload designs starting with Unit 1 Cycle 21 and Unit 2 Cycle 19.

2. To support the acceptability of Calvert Cliffs operation at 2754 MWt and 15.0 kW/ft, the following NRC staff recommendations, modifications to the S-RELAP5 input modeling, and changes to the EOPs include the following:
   a. Modifications to the S-RELAP5 modeling to allow only one cold leg suction piping to clear of liquid for all small breaks with diameters of 4 inches or less.
   b. Removal of credit for the hot leg nozzle gaps and the upper core barrel flange.
   c. Leakage paths that represent communicate paths for fluid flow between the upper plenum and upper head directly into the upper downcomer region.
   d. Inclusion of a large reverse flow K-factor at the outlet of the core to prevent the downflow of liquid from above to cool the core hot bundle during periods of core uncovery.
   e. The HPSI head-flow curve input to the S-RELAP5 code will be verified against the surveillance testing to be conducted prior to power operation with the AREVA fuel loaded in the core. The heat flow curve should include adjustments for all measurement uncertainties associated with the surveillance test.
   f. The simulator operator training and qualification should be conducted periodically to ensure the operators can trip the RCPs following the limiting small break LOCA within 4 minutes following loss of 20°F subcooling.

3. The following restriction is imposed on the S-RELAP5 small break LOCA methodology for Calvert Cliffs:

   Should the PCT increase above the current limiting break PCT of 1626°F in any subsequent evaluation, the licensee will be expected to correct the ability of the S-RELAP5 code to more accurately compute the two-phase level and resultant heat-up of the fuel cladding in the core. Currently, the core nodalization produces core cells with a height of 6". Because of the homogeneous assumption regarding vapor and liquid mixing in a control volume (cell) in S-RELAP5, saturated conditions are imposed on the entire volume regardless of the amount of liquid contained in the volume. As such, no heat-up occurs in the cell containing the two-phase surface in the core until all of the liquid in the cell drains to the cell below. To correct this deficiency, more cells in the core region are required or modifications to compute the two-phase surface within the cell containing the level are necessary to properly account for the vapor superheat and cladding heat-up in this region.

14.17.3.3 <u>Plant Description and Summary of Analysis Parameters</u>

Calvert Cliffs Units 1 and 2 are Combustion Engineering-designed 2x4 PWRs with two hot legs, four cold legs, and two vertical U-tube SGs. The reactor has a rated core power of 2754 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 217 fuel assemblies. The hot legs connect the reactor vessel with the vertical U-tube SGs. Main feedwater is injected into the downcomer of each SG. There are three AFW pumps, one motor-driven and two turbine-driven. The ECCS contains three HPSI pumps (a minimum of 2 OPERABLE per Technical

Specifications), four SITs, and two LPSI pumps. Important system parameters and initial conditions used in the analysis are given in Table 14.17-9.

The RCS was nodalized in the S-RELAP5 model into control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SG tube plugging level of 10% in each SG was assumed. The HPSI system was modeled to deliver the minimum total SI flow asymmetrically to the broken loop and three intact loops in the S-RELAP5 model, with the highest individual loop HPSI flow and the flow from one SIT injected into the cold leg containing the break. The degraded HPSI flow used in the analysis is shown in Table 14.17-10. Low pressure safety injection flow was not modeled since the primary system pressure does not fall below the shutoff heat of the LPSI pumps until well after the time of PCT is reached.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed in 10 CFR Part 50, Appendix K.

The input model included details of both main steam lines from the SGs to the turbine control valve, including the MSSV inlet piping connected to the main steam lines. The MSSVs were set to open at their nominal setpoints plus 3% tolerance.

The analysis assumed loss of offsite power concurrent with reactor scram on low pressurizer pressure. The single-failure criterion required by 10 CFR Part 50, Appendix K was satisfied by assuming the loss of one emergency diesel generator, which resulted in the disabling of one HPSI pump and the motor-driven AFW pump. Thus, a single HPSI pump was assumed to be operable. Charging pump flow was not credited in the analysis. Initiation of the HPSI system was delayed by 30 seconds beyond the time of SIAS. The 30-second delay represents the time required for diesel generator startup and switching. The disabling of the motor-driven AFW pump leaves two turbine-driven pumps available, only one of which automatically starts. The initiation of the turbine-driven pump was delayed 180 seconds beyond the time of the AFAS indicating low SG level. Operator startup of the second turbine-driven AFW pump was not credited in the analysis.

A spectrum of cold leg break sizes (0.02 through 0.49 ft^2) was analyzed. The break spectrum calculations assumed RCP trip at reactor trip due to an assumed LOOP at reactor trip.

14.17.3.4 <u>Results of the Small Break Analysis</u>

The time sequence of events for the limiting break case is shown in Table 14.17-11. A 0.09 ft^2 break in the cold leg pump discharge piping with LOOP was determined to have the maximum PCT of 1626°F. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The maximum local cladding oxidation was calculated to be 1.77% of the total cladding thickness before oxidation. The limiting core-wide metal reaction was calculated to be less than 0.02% of the maximum hypothetical amount for the active core as required by 10 CFR 50.46. Trend plots for parameters of interest are shown in Figures 14.17-18 through 14.17-25. These results indicate that a coolable geometry would be maintained during a small break LOCA event.

14.17.3.5 Conclusions

The results of the small break LOCA analysis conform to the 10 CFR 50.46 ECCS acceptance criteria of 2200°F, 17%, and 1% for peak cladding temperature, maximum cladding oxidation, and maximum core-wide oxidation.

14.17.4 CURRENT CYCLE ANALYSES

14.17.4.1 Unit 1

The base large break LOCA and small break LOCA ECCS performance analyses presented in Sections 14.17.2 and 14.17.3, respectively, are applicable to the current cycle of Unit 1.

14.17.4.2 Unit 2

The base large break LOCA and small break LOCA ECCS performance analyses presented in Section 14.17.2 and 14.17.3, respectively, are applicable to the current cycle of Unit 2.

14.17.5 REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"
2. EMF-2103(P)(A), Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” Framatome ANP, Inc., April 2003
3. Technical Program Group, Quantifying Reactor Safety Margins, NUREG/CR-5249, EGG-2552, October 1989
4. XN-CC-39(A), Revision 1, “ICECON: A Computer Program to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants),” Exxon Nuclear Company, October 1978
5. G.P. Liley and L.E. Hochreiter, “Mixing of Emergency Core Cooling Water with Steam: 1/3 - Scale Test and Summary,” EPRI Report EPRI-2, June 1975
6. U.S. Nuclear Regulatory Commission, Information Notice 2009-23, Accession Number ML091550527, “Nuclear Fuel Thermal Conductivity Degradation,” October 8, 2009
7. EMF-2328(P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, March 2001
8. Letter from D. V. Pickett (NRC) to G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)
9. Letter from N. S. Morgan (NRC) to G. H. Gellrich (CCNPP), dated December 19, 2012, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Safety Evaluation of the Realistic Large-Break Loss-of-Coolant Accident Summary Report (TAC Nos. ME7672 and ME7673)

TABLE 14.17-1

SAMPLED LARGE BREAK LOCA PARAMETERS

Phenomenological

- Time in cycle (peaking factors, axial shape, rod properties, and burnup)
- Break type (guillotine versus split)
- Critical flow discharge coefficients (break)
- Decay heat[(a)]
- Critical flow discharge coefficients (surgeline)
- Initial upper head temperature
- Film boiling heat transfer
- Dispersed film boiling heat transfer
- Critical heat flux
- T_{min} (intersection of film and transition boiling)
- Initial stored energy
- Downcomer hot wall effects
- SG inlet plenum interfacial effects[(b)]
- Condensation interphase heat transfer coefficient[(b)]
- Metal-water reaction

Plant[(c)]

- Offsite power availability[(d)]
- Break size
- Pressurizer pressure
- Pressurizer liquid level
- SIT pressure
- SIT liquid level
- SIT temperature (based on containment temperature)
- Containment temperature
- Containment volume
- Initial RCS flow rate
- Initial operating RCS temperature
- Diesel start (for LOOP only)

(a) Not sampled in analysis, multiplier set to 1.0.

(b) Not sampled in analysis.

(c) Uncertainties for plant parameters are based on typical plant-specific data.

(d) This is no longer a sampled parameter. One set of 59 cases is run with LOOP and another set of 59 cases is run with no-LOOP.

TABLE 14.17-2

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

	EVENT	OPERATING RANGE
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.440”
	b) Cladding inside diameter	0.387”
	c) Cladding thickness	0.0265”
	d) Pellet outside diameter	0.3805”
	e) Pellet density	96% of theoretical
	f) Active fuel length	136.7”
	g) Gd_2O_3 concentrations	2, 4, 6, 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressure location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	14x14 AREVA NP HTP™ fuel
	e) SG tube plugging	≤ 10% [a]
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	2754 MWt [b]
	b) LHR	15.0 kW/ft
	c) F_Q	2.37
	d) F_r	1.810 [c]
	2.2 Fluid Conditions	
	a) Loop flow	370,000 gpm ≤ M ≤ 422,250 gpm
	b) RCS cold leg temperature	546.0°F ≤ T ≤ 554.0°F
	c) Pressurizer pressure	2164 psia ≤ P ≤ 2336 psia
	d) Pressurizer level	32.2% ≤ L ≤ 67.2%
	e) SIT pressure	194.7 psia ≤ P ≤ 264.7 psia
	f) SIT liquid volume	1090 ft^3 ≤ V ≤ 1179 ft^3
	g) SIT temperature	60°F ≤ T ≤ 125°F (Coupled with Containment temperature)
	h) SIT resistance fL/D	As-built piping configuration: Line 11A: 5.80 Line 11B: 5.72 Line 12A: 5.19 Line 12B: 5.35
	i) Minimum ECCS boron	≥ 2300 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge piping
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	0.2876 ≤ A ≤ 1.0 full pipe area (split) 0.2876 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one emergency diesel generator
	e) Offsite power	On or Off

TABLE 14.17-2

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

EVENT	OPERATING RANGE		
f) ECCS pumped injection temperature	100°F		
g) HPSI pump delay	30.0 sec (w offsite power) 30.0 sec (w/o offsite power)		
h) LPSI pump delay	45.0 sec (w offsite power) 45.0 sec (w/o offsite power)		
i) Containment pressure	13.7[d] psia, nominal value		
j) Containment temperature	60°F ≤ T ≤ 125°F		
k) Containment spray delay	20 sec		
l) HPSI flow	RCS Cold Leg Pressure (psia)	Broken loop flow (gpm)	Intact loop flow (gpm)
	14.7	164.96	155.5
	215	164.96	155.5
	615	125.25	116.8
	900	87.04	79.5
	1015	70.15	63.0
	1100	50.87	44.2
	1150	35.31	29.1
	1180	21.81	15.9
	1195	10.66	5.9
	1195.1	0	0.0
m) LPSI flow	RCS Cold Leg Pressure (psia)	Broken loop flow (gpm)	Intact loop flow (gpm)
	14.7	1713.52	1659.62
	64.7	1422.6	1377.41
	114.7	1029.27	995.9
	149.7	604.27	583.83
	159.7	393.04	379.15
	169.7	206.06	198.2
	169.8	0	0

(a) In the realistic large break LOCA analysis, only the maximum 10% tube plugging in each SG was analyzed. By independently sampling the break loss discharge coefficients, any flow differences attributed to asymmetry in the SG tube plugging is covered by use of the realistic large break LOCA methodology.

(b) Includes 17 MWt uncertainties.

(c) The radial power peaking for the hot rod includes 6% measurement uncertainty and 3.5% allowance for control rod insertion effect.

$F_{r\ limit} = F_r *(1+uncert\_F_r)*(1+uncert\_cr\_insertion)=1.65*(1.0+0.06)*(1+0.035)=1.810$

(d) Nominal containment pressure range is -1.0 psi to +1.8 psi. For realistic large break LOCA, a reasonable value in this range is acceptable.

TABLE 14.17-3

STATISTICAL DISTRIBUTIONS USED FOR PROCESS PARAMETERS

PARAMETER	OPERATIONAL UNCERTAINTY DISTRIBUTION	PARAMETER RANGE
Pressurizer pressure (psia)	Uniform	2164 - 2336
Pressurizer liquid level (%)	Uniform	32.2 - 67.2
SIT liquid volume (ft^3)	Uniform	1090.0 - 1179.0
SIT pressure (psia)	Uniform	194.7 - 264.7
Containment temperature (°F)	Uniform	60 - 125
Containment volume (ft^3)	Uniform	1.989E+6 - 2.148E+6
Initial RCS flow rate (gpm)	Uniform	370,000 - 422,250
Initial RCS operating temperature (T_{cold}) (°F)	Uniform	546.0 - 554.0
RWT temperature for ECCS (°F)	Point	100
Offsite power availability [(a)]	Binary	0, 1
Delay for containment spray (sec)	Point	20
LPSI pump delay (sec)	Point	30.0 (w offsite power) 30.0 (w/o offsite power)
HPSI pump delay (sec)	Point	45.0 (w offsite power) 45.0 (w/o offsite power)

(a) This is no longer a sampled parameter. One set of 59 cases is run with LOOP and one set of 59 cases is run with No-LOOP.

TABLE 14.17-4

SUMMARY OF MAJOR PARAMETERS FOR THE LIMITING PCT CASE

	FRESH 8% Gd_2O_3 FUEL	ONCE-BURNT UO_2 FUEL
Core average burnup (EFPH)	8897	8923
Core power (MWt)	2754	
Hot rod LHGR (kW/ft) / Total Peaking (F_Q)	14.3618 / 2.26884	
Radial Peaking (F_r)	1.62	1.73
ASI	-0.0878	-0.0949
Break type	Guillotine	
Break size (ft^2/side)	4.5832	
Offsite power availability	Not available	
Decay heat multiplier	1.0	

TABLE 14.17-5

SUMMARY OF HOT ROD LIMITING PCT RESULTS

CASE #47 (Offsite Power Unavailable)	**FRESH FUEL 8% Gd_2O_3 ROD**	**ONCE-BURNT UO_2 ROD**
PCT		
Temperature	1620°F*	1545°F
Time	8.52 sec	8.36 sec
Elevation	7.859 ft	7.859 ft
Metal-Water Reaction		
Pre-transient local oxidation (%)	1.214	1.997
Transient local oxidation maximum (%)	0.543	0.463
Total local oxidation maximum (%)	1.757	2.460
Total core-wide oxidation (%)	0.0111	

* The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

TABLE 14.17-6

CALCULATED EVENT TIMES FOR THE LIMITING PCT CASE

TIME (sec)	EVENT
N/A	RCP trip
0.0	Break opened
0.6	SIAS initiated
8.5	PCT occurred
13.8	Start of broken loop SIT injection
16.2, 16.1, and 16.1	Start of intact loop SIT injection (Loops 2, 3, and 4, respectively)
27.1	Beginning of core recovery (beginning of reflood)
30.6	Broken loop HPSI delivery began
30.6, 30.6, and 30.6	Intact loop HPSI delivery began (Loops 2, 3, and 4, respectively)
45.6	Broken loop LPSI delivery began
45.6, N/A, and N/A	Intact loop LPSI delivery began (Loops 2, 3, and 4, respectively)
72.0, 68.2, and 68.5	Intact loop SITs emptied (Loops 2, 3, and 4, respectively)
72.6	Broken loop SIT emptied
340.0	Transient calculation terminated

TABLE 14.17-7

CONTAINMENT HEAT SINK DATA

DESCRIPTION	SLAB MATERIAL	MATERIAL THICK. (ft)	AREA (ft^2)
Shell and Dome	Paint	2.50E-04	73230
	Carbon Steel	2.08E-02	
	Concrete	3.00E+00	
Unlined Concrete	Concrete	4.00E+00	53000
Galvanized Steel	Zinc	3.17E-04	100800
	Carbon Steel	8.33E-03	
Painted Thin Steel	Paint	2.50E-04	70250
	Carbon Steel	2.07E-02	
Painted Steel	Paint	2.50E-04	55000
	Carbon Steel	5.25E-02	
Painted Thick Steel	Paint	2.50E-04	2966
	Carbon Steel	2.01E-01	
Containment Penetration Area	Paint	2.50E-04	3000
	Carbon Steel	6.25E-02	
	Concrete	3.75E+00	
Stainless Steel Lined Concrete	Stainless Steel	1.56E-02	7925
	Concrete	4.00E+00	
Containment Liner Plate Stiffeners	Paint	2.50E-04	4000
	Carbon Steel	6.67E-01	
	Concrete	2.00E+00	
Base Slab	Concrete	8.00E+00	13300
Sump Strainer 1	Stainless Steel	1.31E-02	308.774
Sump Strainer 2	Stainless Steel	1.97E-02	161.338
Sump Strainer 3	Stainless Steel	9.83E-03	3
Sump Strainer 4	Stainless Steel	4.08E-03	3433.5
Additional H/S 1	Carbon Steel	1.00E-02	193.05
Additional H/S 2	Paint	2.50E-04	42.79
	Carbon Steel	2.08E-02	
Additional H/S 3	Paint	2.50E-04	56.54
	Carbon Steel	4.17E-02	
Improvised H/S	Stainless Steel	8.33E-02	10000

THERMAL PROPERTIES

Material	Thermal Conductivity (Btu/hr-ft-°F)	Heat Capacity Btu/ft^3-°F
Concrete	2.5	35
Carbon steel	35	55
Stainless steel	10	62
Paint	1.5	32
Zinc	70	45

TABLE 14.17-8

CONTAINMENT INITIAL AND BOUNDARY CONDITIONS

Containment Volume	
Net free volume, ft^3	1,989,000 - 2,148,090
Initial Conditions	
Compartment pressure (nominal), psia	13.7
Compartment temperature, °F	$60 \leq T \leq 125$
Outside temperature, °F	10
Humidity,%	90
Containment spray	
Number of pumps operating	2
Spray flow rate (total, both pumps), gpm	4,600
Minimum spray temperature, °F	40
Fastest post-LOCA initiation of spray, sec	20
Initial Time, sec:	
Spray flow (minimum)	20
Fans (minimum)	0

TABLE 14.17-9

SMALL BREAK LOCA ANALYSIS ECCS PERFORMANCE

Key System Parameters and Initial Conditions

Reactor power, MWt	2754[a]
Peak LHR, kW/ft	15.0
Radial peaking factor (1.65 plus uncertainties)	1.81[b]
RCS flow rate, gpm	370000
Pressurizer pressure, psia	2250
Core inlet coolant temperature, °F	548
SIT pressure, psia	194.7
SIT fluid temperature, °F	120
AFW temperature, °F	112
Low SG level AFAS setpoint for harsh conditions,% wide range	29.26[a]
HPSI fluid temperature, °F	100
Reactor scram low pressurizer pressure setpoint for harsh conditions, psia	1790[a]
Reactor scram delay time on low pressurizer pressure, sec	0.9
Scram CEA holding coil release delay time, sec	0.5
SIAS activation setpoint pressure for harsh conditions, psia	1640[a]
HPSI pump delay time on SIAS, sec	30
MSSV lift pressures	Nominal + 3% tolerance

(a) Includes uncertainty.

(b) Includes 1.06 F_r measurement uncertainty and 1.035 F_r rodded augmentation factor.

TABLE 14.17-10

SMALL BREAK LOCA ANALYSIS ECCS PERFORMANCE

HPSI FLOW RATE VERSUS COLD LEG PRESSURE

COLD LEG PRESSURE (psia)	TOTAL FLOW (to four loops) (gpm)	TOTAL FLOW TO 3 INTACT LOOPS (gpm)
15	631.48	466.52
215	631.48	466.52
615	475.57	350.32
900	325.56	238.52
1015	259.24	189.09
1100	183.56	132.69
1150	122.47	87.16
1180	69.47	47.66
1194	28.32	17.66
1195	0.0	0.0

TABLE 14.17-11

SMALL BREAK LOCA ECCS PERFORMANCE ANALYSIS

CALCULATED EVENT TIMES FOR LIMITING BREAK SPECTRUM CASE

Break Size (ft^2) - 0.09

	Time (sec)
Event initiation	0.0
Pressurizer pressure reaches low PZR pressure setpoint (1790 psia)	13.6
Reactor trip, offsite power lost, RCPs tripped, MFW terminated, and turbine tripped	15.0
Pressurizer pressure reaches SIAS setpoint (1640 psia)	19.8
HPSI flow begins	90
SG level reaches AFAS setpoint (29.26% wide range span)	2892
Steam-driven AFW delivery begins	3072
Loop seal 1A clears [a]	---
Loop seal 1B clears [a]	280
Loop seal 2A clears [a]	1580
Loop seal 2B clears (broken loop) [a]	---
Break uncovers	334
SIT flow begins	1564
Minimum reactor vessel mass occurs	1112
Hot rod rupture occurs	1285
PCT occurs	1568

[a] LOOP seal clearing times based on times when void fraction reached 0.50; break uncovery time based on time void fraction reached 0.90.

14.18 FUEL HANDLING INCIDENT

14.18.1 GENERAL

The likelihood of a fuel handling incident is minimized by administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a qualified supervisor. Before any refueling operations begin, verification of complete CEA insertion is obtained by tripping all CEAs to obtain indication that the CEAs are fully in. Boron concentration in the coolant is raised to the refueling boron concentration where the core would be at least 5% subcritical and is verified by chemical analysis.

After the vessel head is removed, the CEA drive shafts are removed from their respective assemblies. A load cell is used to indicate that the drive shaft is free of the CEA as the lifting force is applied.

The maximum elevation to which the fuel assemblies can be raised is limited by the design of the fuel handling hoists and manipulators to assure that the minimum depth of water above the top of a fuel assembly required for shielding is always present (Chapter 9). This constraint applies in fuel handling areas inside containment and in the spent fuel pool area. Supplementing the physical limits on fuel withdrawal, radiation monitors located at the fuel handling areas would provide both audible and visual warning of high radiation levels in the event of a low water level in the refueling cavity and fuel pool. Fuel pool structural integrity is assured by designing the pool and the spent fuel storage racks as Category I structures.

The design of the spent fuel storage racks and handling facilities in both the containment and fuel storage area is such that fuel will always be in a subcritical geometrical array, assuming zero boron concentration in the fuel pool water. The spent fuel pool and refueling pool water are normally at refueling boron concentration. Natural convection of the surrounding water provides adequate cooling of fuel during handling and storage. Adequate cooling of the water is provided by forced circulation in the spent fuel pool cooling system.

Fuel failure during refueling as a result of inadvertent criticality or overheating is not possible. The possibility of damage to a fuel assembly as a consequence of mishandling is minimized by thorough training, detailed procedures, and equipment design. The single-failure-proof design of the Spent Fuel Cask Handling Crane prevents the drop of heavy objects such as shipping/transfer casks on the spent fuel storage racks. Inadvertent disengagement of a fuel assembly from the fuel handling machine is prevented by mechanical interlocks; consequently, the possibility of dropping and damaging of a fuel assembly is remote.

Should a fuel assembly be dropped or otherwise damaged during handling, radioactive release could occur in either the containment or the Auxiliary Building. The air in both of these areas is monitored. The radiation monitors immediately indicate the increased activity level and alarm. The affected area would then be evacuated.

The effects of a Fuel Handling Accident on Control Room habitability are discussed in Section 9.8.2.3, Auxiliary Building Ventilating Systems.

Release of activity through the containment purge system would be prevented by automatic closure of the containment isolation dampers, as described in Chapter 9. With the exception of the personnel airlock doors, the equipment hatch opening, and penetration flow paths providing direct access from the containment atmosphere to the environment unisolated but under administrative controls, automatic containment closure capability is required during movement of irradiated fuel within the containment building.

Both doors of the containment personnel air lock (which leads to the interior of the Auxiliary Building) may be open during fuel movement if at least one door is operable and capable of being closed by a designated individual stationed immediately outside of the airlock. A temporary hatch cover plate may be used in place of an emergency personnel escape door. The equipment hatch opening may be open during fuel movement if the containment outage door is operable and capable of being closed within 30 minutes by a designated individual stationed near the door. Closing the containment outage door includes removal of any grating or truck ramps that are in the opening, with the use of a forklift, followed by door closure. Penetration flow paths providing direct access from the containment atmosphere to the environment may be unisolated under administrative controls. These controls minimize the potential for release of activity to the environment during the time it takes to evacuate the containment structure.

The spent fuel pool ventilation system draws air across the spent fuel pool area; this air is discharged to the atmosphere through the plant vent. If the cask loading hatch and all exterior hatches to the 69' level of the Auxiliary Building are closed, this is the only route for the release of activity from the spent fuel pool area to the environment.

Failed fuel rods that have released their active gas gap inventory can be stored in encapsulated fuel tubes. These encapsulated fuel tubes can be stored in the peripheral guide tubes of empty grid cages in the spent fuel pool. A single encapsulation tube containing a damaged fuel rod can be stored in an incore instrumentation (ICI) trash can, can be stored in an empty spent fuel pool (SFP) rack space that is inaccessible to both the spent fuel handling machine and the Auxiliary Building cask handling crane, can be laid temporarily atop the spent fuel pool storage racks with administrative restrictions on fuel movement in the laydown area, or can be placed at the bottom of an upender trench with the associated upender tagged out. Fuel failure resulting from inadvertent criticality is administratively precluded, as described in Section 9.7.2.10. Storage of the encapsulation tubes in the peripheral guide tubes of empty grid cages, in an ICI trash can or empty SFP rack space, will cause a decrease in maximum SFP reactivity due to a decrease in fissile inventory and will not create inadvertent criticality. An encapsulated fuel rod placed temporarily atop the spent fuel pool storage racks or at the bottom of the upender trench will be decoupled in reactivity space from the assemblies stored within the rack.

Undamaged fuel rods can only be stored in the encapsulation tubes in empty grid cages. This will guarantee that the consequences of a fuel handling incident will not be increased. Only damaged fuel rods with no gas gap activity can be stored in encapsulation tubes, in ICI trash cans or empty SFP rack space, temporarily atop the spent fuel pool storage racks, or at the bottom of an upender trench, thus precluding any fission gas release.

14.18.2 METHOD OF ANALYSIS

The analysis assumes that a fuel assembly is dropped during fuel handling in the Containment. Interlocks and procedural and administrative controls make such an event highly unlikely; however, if an assembly were damaged to the extent that one or more fuel rods were broken, the accumulated fission gases and iodines in the fuel element gap would be released to the surrounding water. Release of the solid fission products in the fuel would be negligible because of the low fuel temperature during refueling, which greatly limits their diffusion.

In the spent fuel pool the fuel assemblies are stored within the racks at the bottom of the spent fuel pool. The top of the rack extends above the tops of the stored fuel assemblies. A dropped fuel assembly could not strike more than one fuel assembly in the storage rack. Impact could occur only between the ends of the involved fuel assemblies, the bottom end fitting of the dropped fuel assembly impacting against the top end fitting of the stored fuel assembly. The results of an analysis of the end on energy absorption capability of a fuel

assembly indicate that a fuel assembly is capable of absorbing the kinetic energy of the drop with no fuel rod failures.

Reconstitution or inspection of a fuel assembly can take place in individual SFP storage racks with spent fuel assemblies placed on rack spacers and with their upper end fittings removed. In such a configuration, the structural integrity of the fuel assemblies is reduced, and the fuel rods may protrude above the SFP racks. Since fuel damage could occur if a heavy object is dropped on top of an assembly seated on a rack spacer with its upper end fitting or template removed, administrative controls will restrict movement of loads over the affected assemblies on rack spacers plus one storage rack cell on each side of the affected assemblies. Heavy loads may only be moved in this area via the single-failure-proof crane, if assemblies are seated on rack spacers with their upper end fittings or templates removed. Only the single-failure-proof crane or single-failure-proof rigging will be used over the reconstitution area in the SFP for loads other than tools. A knowledgeable and briefed person will be present for the entire time that the upper end fitting or template is removed from an assembly to restrict movement of loads other than tools in this area of the SFP. In addition, after the upper end fittings or templates have been removed, the spent fuel handling machine will be administratively prohibited from nearing the affected assemblies on rack spacers plus one storage rack cell on each side of the affected assemblies.

The analyses for a fuel handling incident in the refueling pool and the spent fuel pool both assume that gas gap activity from 176 fuel rods is released. Because of the high energy absorption required to rupture a fuel rod, this number represents the maximum number of damaged pins expected from any credible fuel handling incident scenario. Undamaged fuel rods can only be stored in the encapsulation tubes in empty grid cages. As with encapsulated failed rods, there can only be four in a grid cage. Only damaged fuel rods with no gas gap activity can be stored in encapsulation tubes, in ICI trash cans or empty SFP rack space, temporarily atop the spent fuel pool storage racks, or at the bottom of an upender trench, thus precluding any fission gas release.

The Fuel Handling Incident (FHI) analysis assumes a total iodine decontamination factor of 200 based on a minimum water depth of 23' per Reference 13. In the refueling pool this assumption is preserved by the Technical Specification requirement of 23' of water above fuel assemblies seated in the reactor core. In the SFP, the Technical Specifications only require 21.5' of water above fuel assemblies seated in the SFP storage racks. This Technical Specification was deemed sufficient to preserve the required 23' of water because a FHI was assumed to occur as a fuel assembly strikes the bottom of the SFP. When assemblies are placed on rack spacers and their upper end fittings are removed, a FHI from a dropped heavy object would require a lower decontamination factor based on reduced water coverage. A revised decontamination factor of 120 for a FHI during reconstitution/inspection with 20.4' of water between the top of the pin and the surface of the water was computed for a 20.5" rack spacer. Note that this is very conservative, since normal level control will result in at least 21.5' of water above exposed fuel pins. An FHI with 55 days of decay time post-shutdown and no Spent Fuel Pool Exhaust Ventilation System (SFPEVS) credit is less severe radiologically than an FHI with 72 hours of decay time post-shutdown, a reconstitution decontamination factor of 120, and SFPEVS credit. Thus after 55 days of decay, the SFPEVS operability requirements and the requirement of negative pressure in the SFP area are no longer necessary during irradiated fuel movement or reconstitution in the Auxiliary Building.

Fission product activity in the fuel rod gap has been determined for the highest power fuel assembly in the core for the design basis accident. The rod gap activity is computed as a function of total core inventory for the following isotopes: 20% of Kr-85, 10% of all other noble gases, 16% of I-131, and 10% of all other iodines. These gas gap activities have been validated for assembly average burnup of up to 62,000 MWD/MTU. Such a value

bounds Calvert Cliffs current peak pin burnup limit of 62,000 MWD/MTU. Total core inventory is calculated using the methodology described in References 13 and 14. Fuel assemblies will not be removed from the core within 72 hours after shutdown. Therefore, many of the short lived isotopes will have decayed significantly.

14.18.3 RESULTS

14.18.3.1 Fuel Handling Incident in Containment

Table 14.18-1 shows the activity that would be released from the failure of all 176 fuel rods. In the highest power fuel assembly, the activities tabulated are those released to the water in the fuel pool, and those released to the containment atmosphere.

Because iodine is readily absorbed by water and the fuel being handled is under water, much of the iodine released from the damaged rods would be retained in the refueling pool water. To account for this preferential retention of iodine by the pool water, a decontamination factor of 200 is assumed, which corresponds to the value suggested in Reference 13. No additional credit is taken for plate-out of iodines on surfaces within the containment.

The 0-2 hour dose at the exclusion boundary was calculated based on the following assumptions:

a. Gap activity releases from the damaged fuel rods to the refueling pool water and to the containment air are listed in Table 14.18-1.

b. An overall refueling pool iodine decontamination factor of 200 is assumed. This is based upon a decontamination factor of 285 for inorganic iodine and 1 for organic iodine. The iodine in the fuel rod gas gap is composed of 99.85% inorganic species and 0.15% organic species. After applying the appropriate decontamination factors for each species, the refueling pool is 70% inorganic species and 30% organic species. No credit is taken for noble gas retention by the pool water.

c. The site boundary atmospheric dispersion factor is 1.44×10^{-4} sec/m^3 (Chapter 2).

d. A breathing rate of 3.5×10^{-4} m^3/sec is assumed.

e. The gap activity from the damaged fuel rods is released to the environment over a two hour time period.

f. Because both doors of the containment personnel air lock, unisolated but administratively controlled penetration flow paths from the containment atmosphere to the environment, and the containment outage door may be open during the early stages of a fuel handling incident, all activity released from the containment is assumed to be unfiltered.

g. Doses were computed using the dose conversion factors and methodology described in References 15 and 16.

The results are tabulated in Table 14.18-2.

14.18.3.2 Fuel Handling Incident in the Spent Fuel Pool Area

If a fuel handling incident were to occur while handling fuel in the spent fuel pool area, the following assumptions would apply:

a. The activity release to the spent fuel pool water is identical to that assumed in the case of a fuel handling incident in containment. The overall spent fuel pool decontamination factor for iodine is 120. This is based upon a

water level of at least 20.4' covering a ruptured fuel assembly. This activity is tabulated in Table 14.18-1.

The overall SFP decontamination factor for iodine is 120 for a FHI during reconstitution with 20.4’ of water between the tops of the pins and the surface of the water.

b. The same breathing rates and atmospheric dispersion factor assumptions apply for the spent fuel pool as for the refueling pool case.

c. The iodine in the fuel rod gas gap is composed of 99.85% inorganic species and 0.15% organic species. After applying the appropriate decontamination factors for each species (146 for inorganic and 1 for organic), the iodine in the air above the spent fuel pool is 82% inorganic species and 18% organic.

d. For a FHI in the SFP, all of the activity released to the air above the spent fuel pool is assumed to be discharged to the outside atmosphere through the SFPEVS. An FHI with 55 days of decay time post-shutdown and no SFPEVS credit is less severe radiologically than an FHI with 72 hours of decay time post-shutdown, a reconstitution decontamination factor of 120, and SFPEVS credit. Thus after 55 days of decay, the SFPEVS operability requirements and the requirement of negative pressure in the SFP area are no longer necessary during irradiated fuel movement or reconstitution in the Auxiliary Building.

The results are tabulated in Table 14.18-2.

14.18.4 CONCLUSION

The 0-2 hour exclusion boundary doses resulting from a fuel handling incident are within the guidelines of 10 CFR 50.67. This is true even if all the rods of the highest power fuel assembly fail when the fuel assembly is dropped into the refueling pool or when a fuel assembly is damaged on a rack spacer during reconstitution or inspection with its upper end fitting removed in the SFP.

This event includes the transition to AREVA Advanced CE-14 HTP™ fuel (with Gd_2O_3 burnable poison irradiated to a maximum burnup of 62 GWd/MTU). A source term was developed that bounds the transition cycles and the full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

14.18.5 REFERENCES

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2. Deleted
3. Deleted
4. Deleted
5. Deleted
6. Deleted
7. Deleted
8. Deleted
9. Deleted
10. Deleted
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12. Deleted

13. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000

14. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), “Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846),” dated August 29, 2007

15. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988

16. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993

TABLE 14.18-1

SOURCE TERM FOR FUEL HANDLING ACCIDENT IN CONTAINMENT OR SPENT FUEL POOL BASED UPON ALTERNATIVE SOURCE TERM METHODOLOGY

Isotope	Decay Constant (1/sec)	Ci/MWt	Ci Released to Water	Ci Released to Cntmt Air	Ci Released to SFP Air
Kr-85	2.049E-09	3.718E+02	1.603E+03	1.603E+03	1.603E+03
Kr-85m	4.298E-05	7.968E+03	2.496E-01	2.496E-01	2.496E-01
Kr-87	1.514E-04	1.621E+04	3.161E-13	3.161E-13	3.161E-13
Kr-88	6.780E-05	2.266E+04	1.141E-03	1.141E-03	1.141E-03
I-131	9.978E-07	2.756E+04	7.346E+04	3.673E+02	6.122E+02
I-132	8.371E-05	3.946E+04	3.211E-05	1.605E-07	2.676E-07
I-133	9.257E-06	5.572E+04	1.091E+04	5.456E+01	9.093E+01
I-134	2.196E-04	6.286E+04	2.564E-20	1.282E-22	2.137E-22
I-135	2.913E-05	5.296E+04	6.011E+01	3.005E-01	5.009E-01
Xe-133	1.530E-06	5.571E+04	8.085E+04	8.085E+04	8.085E+04
Xe-135	2.118E-05	1.771E+04	1.577E+02	1.577E+02	1.577E+02
Xe-133m	3.663E-06	1.735E+03	1.449E+03	1.449E+03	1.449E+03
Xe-135m	7.551E-04	1.164E+04	2.530E-81	2.530E-81	2.530E-81
Xe138	8.193E-04	4.933E+04	6.261E-88	6.261E-88	6.261E-88

TABLE 14.18-2

OFFSITE AND CONTROL ROOM DOSES FOR A FUEL HANDLING ACCIDENT IN CONTAINMENT OR SPENT FUEL POOL BASED UPON ALTERNATIVE SOURCE TERM METHODOLOGY

	TEDE Dose (Rems)		
	FHI in Containment	**FHI in SFP**	**Regulatory Limit**
Exclusion Area Boundary	0.6958	1.1136	6.3
Low Population Zone	0.1638	0.2622	6.3
Control Room	2.3314	3.8538	5.0

14.19 TURBINE-GENERATOR OVERSPEED INCIDENT

This is an analyzed event.

For a discussion of this material, see Section 5.3.1.2.

14.20 CONTAINMENT RESPONSE

14.20.1 INTRODUCTION

The Containment Structure encloses the primary and secondary plant and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The Containment Structure must be capable of withstanding the pressure and temperature conditions resulting from a postulated LOCA or main steam line break (MSLB) accident. While other events, such as a feedwater line break also discharge mass and energy to Containment, the LOCA and MSLB have been confirmed to be the two most severe inside containment events with respect to maximizing the peak containment pressure and temperature.

The Calvert Cliffs Unit 1 and 2 Containments are each approximately 2.0 E6 ft^3 in net free volume, and have a design pressure and temperature of 50 psig and 276°F, respectively. These containment design values were selected as a result of the original analysis of the LOCA. While 50 psig reflects a maximum pressure in the vapor space, the design temperature of 276°F is a structural limit of the concrete wall inner surface and the steel liner plate. The acceptance criteria for the containment response analysis is that pressure and temperature remain below these limits. This ensures that the containment response analysis is performed in accordance with proposed General Design Criteria 10, 40, and 52 with respect to containment heat removal and design margin. In addition to demonstrating that the structural pressure and temperature limits discussed above are met, the DBEs presented in this section are used to develop an enveloping containment accident pressure and temperature profile for the purpose of demonstrating the Environmental Qualification of safety-related electrical equipment in Containment.

This section summarizes the LOCA and MSLB containment pressure and temperature as well as subcompartment analyses. The pressure and temperature analyses for LOCA and MSLB have been further divided into mass and energy release calculations and containment response calculations. These analyses have been updated to address plant changes and issues. The MSLB containment analysis was updated due to the replacement of Units 1 and 2 SGs.

Consistent with the original design analyses, the mass and energy released into Containment during a LOCA has been calculated by the NSSS vendor, Westinghouse Electric Corporation, previously Asea Brown Boveri/Combustion Engineering, Inc. (ABB/CE). This information was used to calculate the peak containment pressure and temperature and long-term containment response. The MSLB mass and energy release rates were calculated by Framatome Technologies, Inc. (FTI) as part of Unit 1 steam generator replacement. As presented in the following sections, the results of these analyses have been shown to be within the stated containment design criteria.

14.20.2 LOSS-OF-COOLANT ACCIDENT

14.20.2.1 Description of Event

The LOCA is characterized by the rapid discharge of the RCS inventory into the Containment. The initial enthalpy of this discharge is on the order of 600 Btu/lbm. The containment pressure and enthalpy determine the fraction of this discharge which flashes to steam and the fraction which falls as liquid to the containment sump. This discharge causes containment pressure and temperature to rapidly increase and the containment vapor space quickly becomes saturated. This increase in pressure will initiate a CSAS, a SIAS, and a CIS. The containment pressure increase also initiates a reactor trip, which is quickly followed by the turbine trip. The CSAS and SIAS initiate containment spray and SIAS initiates containment air coolers (CACs). High containment pressure will also initiate the

SGIS which, together with CSAS, initiates the closure of the main steam and feedwater isolation valves. Finally, SGIS and CSAS both initiate trips of the steam generator feedwater pumps (SGFPs), heater drain pumps (HDPs), and condensate booster pumps (CBPs) which rapidly terminate feedwater flow to the SGs.

The large-break LOCA causes a rapid depressurization of the RCS whose pressure quickly falls below the shut-off head of the HPSIs and LPSIs, respectively. The SI pumps, with inventory from the RWT, are the primary source of core cooling for the majority of the event and will start in response to a SIAS on containment high pressure or low pressurizer pressure. Once RCS pressure falls below the SIT cover gas pressure, the SITs will also empty their liquid inventory into the RCS to aid in cooling the core.

Three break locations have been analyzed for Calvert Cliffs: a hot leg break; a cold leg break on the RCP discharge leg; and a cold leg break on the RCP suction leg. A spectrum of break sizes has been analyzed for these locations. For the cold leg break locations, the limiting break size has been determined to be a double-ended slot break (9.8 ft^2) on pump discharge (DES/PD) with maximum SI. For the hot leg break location, the limiting break size has been determined to be a 2.0 ft^2 break. In the case of a discharge leg break, the SI flow to the ruptured cold leg will largely spill to the containment sump. Full SI flow to the core will occur for both the suction and hot leg breaks. The SIT nitrogen cover gas, which is also released to Containment, is only a small component of the overall containment pressure response.

Although the primary-to-secondary heat transfer continues for approximately five-seconds, the primary side temperature rapidly drops below the SG temperature causing reverse heat transfer to occur for the remainder of the transient. This is particularly evident in the cold leg breaks, which have a viable steam flow path through the SG tubes prior to discharging to the Containment. For the cold leg breaks, the addition of SG energy to the exiting break flow extends the duration of the initial peak pressure and temperature and for Calvert Cliffs slightly increases peak pressure and temperature.

For the limiting hot leg break, the peak pressure and temperature occurs in approximately one minute while for the limiting cold leg break, it occurs approximately three minutes after the break. During the long-term portion of the analysis, a second containment pressure and temperature peak occurs following the Recirculation Actuation Signal (RAS). This function initiates the SI pump delivery switch-over from the injection mode to the recirculation mode. The second peak following the RAS occurs after the source of the water for the spray and injection to the reactor vessel is switched from the RWT inventory to the sump. Despite the fact that the recirculation from the sump is passed through the shutdown cooling heat exchangers (SDCHX), the temperature of the spray is higher than that of spray from the RWT. This reduces the effectiveness of the spray and results in the secondary peak pressure and temperature.

Following the initial blowdown, the reactor is first refilled by the incoming SI flow, and then reflooded as the core becomes quenched. During the reflood and post-reflood phases of cold leg breaks, the exiting steam first passes through the SGs and acquires energy prior to exiting the RCS to the Containment. Therefore, a detailed analysis of the reflood and post-reflood phases is necessary for cold leg breaks. The RCS and SGs come into a quasi-equilibrium state during this process

and after the end of the post-reflood phase, the reactor vessel and the primary and secondary systems cool down together. This is the long-term phase of the event, which is typically simulated until the containment atmosphere temperature returns to its initial value. For the hot leg break, the greatest quantity of two-phase mass leaves the top of the core, bypasses the SGs, and discharges to the Containment. There is no mechanism for adding significant quantities of the secondary-side energy to the break flow stream. Hot leg break reflood and post-reflood energy release is, therefore, of minimal importance, and detailed reflood and post-reflood analyses are not performed for hot leg breaks. During this time frame, SI pump flow from the RWT continues to be fed to the reactor vessel. Following RAS, the LPSI pumps are shut off and the HPSI pumps recirculate sump water into the reactor vessel while the containment spray pumps recirculate sump water through the SDCHX to the containment atmosphere. Since the DES/PD with minimum SI results in the highest peak pressure and temperature, only the results of this LOCA are presented in this chapter.

14.20.2.2 Mass and Energy Release

The analytical simulation of the LOCA event is divided into four distinct phases. These are blowdown, reflood, post-reflood and long-term cooldown. For the Calvert Cliffs Units, the NSSS vendor calculates mass and energy release data for three of these four phases. As explained below, the long-term cooldown mass and energy release calculations were performed concurrent with the containment response simulation. The mass and energy release data for the first three phases was input to the Calvert Cliffs Containment model that was prepared with GOTHIC, Electric Power Research Institute's containment response code (References 37 through 42).

The blowdown phase of the LOCA lasts approximately 20 seconds, and is basically the complete discharge of the inventory in the RCS. Break mass and energy transfer rates for this portion of the LOCA are calculated with the NSSS vendor's CEFLASH-4A Code, Reference 2. While the original CEFLASH-4 Code referenced in the Standard Review Plan (SRP) was used in the original Calvert Cliffs analysis, this version of the CEFLASH computer code is no longer available for use and has been superseded by the NRC-approved 1985 Evaluation Model version of the CEFLASH-4A Code. In Appendix K analysis, the goal is to contain core heat to maximize the fuel/cladding temperature. Conversely, for the containment analysis, the goal is to maximize the heat removed from the core to maximize the severity of the mass and energy response. Thus, the biasing of key design inputs relative to the specific criteria for each analysis make the two sets of assumptions quite different.

While the SRP specifies that LOCA containment mass and energy release calculations should be done in general accordance with the Appendix K analysis, it states that additional conservatism should be included to maximize the release to Containment. This additional conservatism is addressed via the following inputs/assumptions:

a. The Appendix K prediction of fuel clad swell and rupture is not considered. This will maximize the energy available for release from the core.

b. Calculations of heat transfer from core to coolant assume nucleate boiling. This will maximize the energy transfer to the exiting RCS coolant. While nucleate boiling is assumed for a portion of the Appendix K transient, as core conditions change, different heat transfer correlations may be selected by the code. To maximize fuel/cladding temperature, once an alternate

means of heat transfer is selected, the correlation does not go back to this high heat transfer regime.

c. The initial mass of water in the RCS is based on the temperature and pressure conditions existing at 102% of full power. This differs from the Appendix K assumption of using nominal (cold) volumes without inclusion of expansion due to normal operating pressure and temperature.

d. Some typical Appendix K assumptions are to isolate the SGs at the initiation of the event and not include the addition of MFW during the blowdown. For the containment analysis, a time-dependent MFW addition and realistic isolation of the SGs is assumed.

e. The ECCS performance analysis assumes maximum loop pressure differences, while the containment analysis assumes nominal pressure differences.

Since the refill phase (the time period during which the reactor vessel fills with SI liquid from the bottom of the reactor vessel to the bottom of the active core) is conservatively omitted for containment calculations, the next phase of the transient simulation is reflood. This is only applicable to cold leg breaks. Full core quench occurs several minutes after reflood. The post-reflood phase is the next phase during which the energy in the RCS and SGs is transferred to Containment via the exiting break flow. The end of post-reflood occurs when the RCS and SGs are essentially in equilibrium.

The reflood and post-reflood phases of the LOCA are simulated in accordance with the NRC-approved FLOODMOD2 methodology, Reference 3. The reflood and post-reflood phases are only used for the cold leg break analysis. During the reflood and post-reflood phases, liquid entrainment in the exiting steam flow is calculated based on a carryout rate fraction specified in the SRP.

The long-term phase of the LOCA completes the transient simulation of this event. GOTHIC is used to calculate the containment pressure and temperature response (Section 14.20.2.3) throughout the transient. This is accomplished by application of the mass and energy release data as well as RCS and SG conditions provided by the NSSS vendor, from the beginning of the event up to the end of blowdown for hot leg breaks and end of post-reflood for cold leg breaks. During the long-term phase of the event, the GOTHIC Code also calculates the mass and energy release data concurrent with the transient containment pressure and temperature calculation. This follows the post-reflood mass and energy release calculation and defines the final phase of the event. Since the Calvert Cliffs Containment model for the GOTHIC Code does not have a mechanistic calculation of sensible heat addition from the primary and secondary metal (including SG inventory), the NSSS vendor's CONTRANS containment code, Reference 4, was also run for this long-term phase. The rate of energy addition due to this sensible heat was then added to the decay power and used as input to the GOTHIC mass and energy release calculation. In this manner, all sources of energy were explicitly accounted for. Table 14.20-1 provides a summary of the significant assumptions associated with the LOCA mass and energy release methodology.

14.20.2.3 <u>Containment Response Analysis</u>

Calvert Cliffs' containment response to a LOCA was previously analyzed by using the COPATTA computer program, Reference 5. COPATTA, a Bechtel code derived from the industry computer program CONTEMPT, Reference 6, analyzes the effects of a high energy line break on the Containment Structure. The present analysis is however performed with the GOTHIC Code (References 37 to 43).

Description of the GOTHIC Code

GOTHIC is a state-of-the-art program that solves the conservation equations for mass, momentum, and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy, and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single-phase flows. The interface models allow for the possibility of thermal nonequilibrium between phases and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms in multi-dimensional models, with optional models for turbulent shear and turbulent mass and energy diffusion. Conservation equations are solved for up to three primary and up to two secondary fields. The primary fields include steam/gas mixture, continuous liquid, and liquid droplet. The secondary field includes ice.

For the primary fields, GOTHIC calculates the relative velocities between the separate but interacting fluid fields, including the effects of two-phase slip on pressure drop. GOTHIC also calculates heat transfer between phases, and between surfaces and the fluid. Reduced equations sets are solved for the secondary fields by the application of appropriate assumptions. The three primary fluid fields may be in thermal nonequilibrium in the same computational cell.

The steam/gas mixture is referred to as the vapor phase and is comprised of steam and, optionally, up to eight different noncondensing gases. The noncondensing gases available in the model are defined by the user. A library of properties for nearly 50 different gases is available, although the user may include any noncondensing gas for which appropriate properties are known. Mass balances are solved for each component of the steam/gas mixture, thereby providing the volume fraction of each type of gas in the mixture.

The principal element of a model is a computational volume. GOTHIC features a flexible noding scheme that allows computational volumes to be treated as lumped parameter (single node) or one-, two- or three-dimensional, or any combination of these within a single model. Volumes partitioned with a one-, two- or three-dimensional computational grid are referred to as subdivided. As a minimum, a GOTHIC model consists of at least one lumped parameter volume. Subdivision of a volume into a one-, two-, or three-dimensional mesh is based on orthogonal coordinates. Adjacent cells in a subdivided volume communicate through parameters defined by discretization of the governing equations. Turbulence and hydrogen burn models augment the volume calculations. Separate volumes communicate through junctions or flow paths. A separate set of momentum equations are solved for junctions. Mass, momentum, and energy can be added or removed at boundary conditions that are connected to volumes by flow paths. Subdivided volumes also communicate through flow connectors. A single 3D flow connector can provide multiple connections between two subdivided volumes.

GOTHIC includes a general model for heat transfer between thermal conductors and the steam/gas mixture or the liquid. There is no direct heat transfer between thermal conductors and liquid droplets. Thermal conductors are modeled as one-dimensional slabs for which heat transfer occurs between the fluid and the conductor surfaces and, within a conductor, perpendicular to the surfaces. Thermal conductors can exchange heat by thermal radiation. Any number of conductors can be assigned to a volume. Nodalization of a conductor allows variation of material properties in the direction of heat transfer. Heat generation within thermal conductors may be specified on a node-by-node basis.

GOTHIC includes an extensive set of models for operating equipment. These items include pumps and fans, valves and doors, heat exchangers and fan coolers, vacuum breakers, spray nozzles, coolers and heaters, volumetric fans (annular fans, deck fans, etc.), hydrogen recombiners (forced and natural convection), ignitors (spark device used to ignite hydrogen burns), and pressure relief valves.

General Model Description

The GOTHIC model is concerned both with the pressure and temperature within the containment atmosphere and the temperatures in the Containment Structure. As discussed previously, the NSSS vendor provided the rate of mass and energy transfer from the RCS to the Containment up to the post-reflood phase. These are obtained from blowdown and core thermal behavior calculations, yielding mass and/or energy input rates from a variety of sources. These sources include the release of reactor coolant, the decay heat (causing heating or boiling of residing water in the reactor vessel), superheating of the steam as it passes through the RCS loops and SG, sensible heat from the RCS metal, and metal-water reactions. The long-term mass and energy release rates, from the vessel to the Containment during a LOCA, are calculated by GOTHIC. A control volume representing the reactor pressure vessel is used to simulate heat transfer from the core to the coolant during post-RAS injection to the RCS from the sump.

The basic Containment model describes the Containment and reactor vessel regions and the effects of heat sinks following a LOCA. Included in this basic model is the capability to accommodate operation of ESFs. Several options have been incorporated in the basic model to assist in using these features.

Superposition of heat input functions is assumed so that any combination of blowdown, metal-water reaction, decay heat generation, and sensible heat energy can be used with appropriate ESFs to determine the containment pressure and temperature-time history associated with a LOCA.

To model the containment response, the lumped volume option of GOTHIC is used, which is consistent with the methodology of Bechtel's COPATTA. Each lump volume consists of a pool and a vapor region. The pool region in the lumped volume for Containment represents the containment sump. The pool region for the reactor vessel represents the inventory of the downcomer, lower plenum, core and upper plenum. Evaporation from the surface of the liquid region is accounted for as GOTHIC calculates an interface temperature, which is then used to determine the rate of heat and mass transfer at the interface. Each region is assumed homogeneous, but a temperature difference can exist between regions. In GOTHIC, drops are treated separately. Hence, by accounting for heat and mass transfer between the vapor and the drop phase, GOTHIC allows for revaporization of the droplets while falling through the hot vapor region to the pool. The non-condensable gases are included in the vapor region.

Thermodynamic Assumptions

Based on the input model described in the previous section, GOTHIC calculates conditions in two separate regions of the Containment, the water region (sump) and the atmosphere region (vapor/drops). Following completion of the primary system post-reflood phase, GOTHIC also calculates conditions in the water and vapor/drop regions in the reactor vessel. GOTHIC allows transfer of mass and energy across these regions as well as between various phases in a region. GOTHIC solves the conservation equations for mass in each region for four phases of drops, liquid, ice, and vapor (steam/non-condensable gas) if any of

these phases are present in the control volume. GOTHIC solves the conservation of energy for drops, liquid, and vapor and for solid thermal conductors if any of these are present. GOTHIC also solves the conservation equation for momentum for drops, liquids, and vapor. The conservation equation for mass in GOTHIC expresses that the "Rate of change of mass in a control volume = mass transfer by convection + mass transfer by diffusion + boundary source + interface source + equipment source + combustion source." The conservation equation for energy in GOTHIC expresses that the "Rate of change of energy in a control volume = convection and flow work + energy carried by thermal diffusion + energy carried by mass diffusion + boundary source + interface source + equipment source + combustion source." Finally, the conservation equation for momentum in GOTHIC expresses that "Rate of change of momentum in a control volume + momentum transfer by convection = surface stress + body force + boundary source + interface source + equipment source."

Integration of the governing equations, with inclusion of the equations of state, for each region, from the start of the transient to any later time, provides the thermodynamic properties with which the state point conditions or pressure and temperature can be determined. Numerical integration of the governing equations and calculation of the properties within each region are based on the following assumptions for the Containment model:

a. At the RCS break point, the discharge flow separates into a steam phase that is added to the atmosphere (vapor) region, and, as specified by boundary condition parameters in the GOTHIC model, the water is added as drops. The steam and drops enter the containment atmosphere at the conditions specified by the boundary condition parameters. In general, the steam will approach saturation at the partial pressure of steam in the atmosphere, while the drops will approach saturation at the total pressure of the atmosphere.

b. The steam and air partial pressure comprise the total containment atmosphere pressure, which is also the sump pressure. Following blowdown, the primary-system (reactor vessel region) pressure remains slightly higher than the containment pressure, which provides sufficient driving force for discharge of steam to the Containment through the break.

c. Initially, the steam-air mixture, the drops, and the water phase are each assumed homogeneously mixed with uniform properties. Specifically, thermal equilibrium between the air and steam is assumed. However, a temperature difference may exist between each phase.

d. Steam that condenses from the atmosphere onto the structure is added to the sump. During any time interval, the condensate is allowed to revaporize if sufficient heat exists in the atmosphere to cause the phase change.

e. Mass and energy will be transferred from the liquid regions (sump and reactor vessel) to the containment atmosphere by flashing, if the calculation indicates that the containment pressure is less than the saturation pressure corresponding to the liquid temperature.

f. The sump region contains no water at the beginning of the design basis accident.

g. Condensation of steam due to a vapor pressure gradient between the steam in the containment atmosphere and the water in the sump is conservatively neglected.

h. Condensation of steam on structural heat sinks occurs at the saturation temperature corresponding to the partial pressure of the steam phase in the Containment.

Atmosphere and Sump Description

The free volume of the Containment is modeled by a single control volume. In the GOTHIC model, this control volume includes the atmosphere and sump regions of the Containment, as well as the equipment models and heat structures that apply within the Containment. The atmosphere generally contains air, steam, and drops. Liquid is added to the sump and the pool as spray and liquid spills out the broken loop. Condensate on the heat structures is also added to the pool. Initially, there are no drops in the atmosphere or liquid in the pool except for a very small, but non-zero, volume fraction. The equipment models include a control volume representing the reactor vessel, coolers representing the heat removal function of the fan coolers, and heat exchangers that remove heat from the fluid when water from the sump is recirculated to the spray system. The equipment models and heat structure models are discussed in the following paragraphs.

Reactor Vessel Region Description

A separate reactor vessel region is used in the GOTHIC model to include the effects of decay power, metal-water reaction, heat transfer from the primary system metal, and SI water cooling. This region is not used during the initial mass and energy release phases (blowdown, reflood, and post-reflood) when the primary coolant system pressure is not in equilibrium with the containment atmosphere. During the initial phases, reactor coolant stored energy, decay power, and other energy sources are normally added to the vapor in the containment atmosphere as blowdown energy releases furnished by the NSSS vendor. The reactor vessel is modeled in GOTHIC as a lumped control volume in which the decay and sensible heats, as provided by the NSSS vendor, are added via a GOTHIC heater. The post-RAS portion of the break event is initiated by opening a valve on a flow path connecting the reactor vessel to the Containment, simulating a cold leg or a hot leg. The valve opens at the termination of reflood/post-reflood phase. The pre-RAS SI is modeled by a GOTHIC flow boundary condition. The post-RAS SI is modeled by a GOTHIC "coupled boundary condition" that models injection of the sump water into the reactor vessel. The vessel pressure is the same as the containment pressure at the termination of the reflood/post-reflood phase. The vessel volume is just slightly larger than that used in COPATTA model to allow steam accumulation.

Calculation of long-term mass and energy transfer from the reactor vessel region takes place at the end of the blowdown phase for hot leg breaks or at the end of the post-reflood phase for cold leg breaks.

Heat Transfer Considerations

During a LOCA, heat transfer takes place between the containment atmosphere, sump or reactor vessel water, and the exposed surfaces inside the Containment Structure. These surfaces may be building structures, primary or secondary system components, equipment, or other possible heat sinks or sources. The rate of heat transfer between the containment regions and these conducting masses, is determined by the surface area, the surface temperature, the heat transfer coefficient, the physical arrangement of the conducting masses and the thermal properties of these masses. All of the above parameters are considered by the GOTHIC Code during the transient analyses as described in this section.

Additional heat transfer from the containment system can occur via a heat exchanger system connected to the sump water recirculation piping. Heat transfer calculations made in this manner permit the removal of residual or decay heat, that has been added to the containment sump during cooling of the RCS. The method of calculating this heat transfer is described below.

a. <u>Heat Conduction Calculations</u>

In the GOTHIC Code, there is no limitation on the number of thermal conductors in a model. These solid conductors can be described by a one-dimensional, multi-region, heat conduction equation given by:

$$\rho c \frac{dT}{dt} = \nabla \bullet (k \nabla T) + \dot{q}'''$$

where:

T	-	temperature (°F)
t	-	time (hr)
k	-	thermal conductivity (Btu/hr-ft-°F)
c	-	specific heat (Btu/lbm-F)
ρ	-	conductor density (lbm/ft^3)
$\dot{q}'''$	-	volumetric heat generation rate (Btu/hr-ft^3)

To perform heat transfer calculations for thermal conductors in GOTHIC, the del operator, ∇, is applied in the Cartesian and the cylindrical coordinates. To be consistent with the COPATTA methodology (Reference 22), a one-dimensional slab is used to model various conductors inside the Containment in the GOTHIC model. Thermal properties of the conductors are properly allowed to change as a function of temperature.

The input for the heat conduction calculations includes provisions for specifying geometry type, surface area, number of different material regions (with coordinates), mesh spacing, and material type for each heat conductor. The magnitude of heat generation, thermal properties, and boundary conditions for each portion of the heat conductors are specified in a similar manner.

Normally, each heat conductor will include all of the solid components comprising a structure in the Containment. For example, the containment structure walls are specified by four materials, consisting of a topcoat layer of paint, a layer of primer coating, a steel liner plate, and several feet of concrete. Similar specifications are made for the other structures within the Containment Structure.

Boundary conditions ranging from perfectly insulated (adiabatic) to zero resistance are applied to the heat conductor surfaces, as appropriate. These boundary conditions may indicate exposure to a constant temperature, a time-dependent temperature (such as the containment atmosphere temperature, the sump temperature, or the reactor vessel water temperature), or some combination of the above. Solution to the thermal conduction equation requires one initial and two boundary conditions to be specified. For the initial condition, the surfaces of all thermal conductors exposed to the containment atmosphere are assumed to be at thermal equilibrium with the containment atmosphere. For boundary conditions, the surfaces of thermal conductors are either exposed to a convection boundary (i.e., a specified bulk fluid temperature

and a heat transfer coefficient), or insulated. Condensing heat transfer coefficient values are based either on the steam/air ratio in the containment atmosphere (Uchida condensing heat transfer coefficient) or a turbulence parameter inside the Containment (Tagami condensing heat transfer coefficient).

b. <u>Heat Transfer Coefficients</u>

Heat transfer in the heat conductors within the Containment Structure is dependent upon the assumed heat transfer coefficient between the containment region and the heat conductor surface. Heat transfer to the heat sinks from the containment atmosphere region is determined by the Tagamiheat transfer coefficient correlation, References 8 and 9. This empirical correlation is applicable during the forced convection period following primary system blowdown. In the Tagami correlation, the maximum heat transfer coefficient is related to three parameters. These parameters are: the total energy released from the primary system during the beginning of blowdown to the first peak pressure in the Containment, time to the first peak pressure (t_p), and containment free volume. The shorter the time to the first peak pressure, the higher the heat transfer coefficient. Similarly, the larger the containment free volume, the higher the heat transfer coefficient. Since the time to the first peak pressure is not known initially, trial runs were made to determine the time to peak pressure. Having found the maximum value for the heat transfer coefficient, values at other points in time are found by linear interpolation. The Tagami correlation is only applicable up to time t_p. Afterward, another correlation is used as described next.

During the post-blowdown period of the transient, a quasi-steady-state condition develops due to decreasing turbulence in the Containment. Heat transfer under these conditions is dependent upon the steam-air mixture. Experimental work by Uchida, et al, Reference 13, has shown that during free convection cooling periods, the condensing heat transfer coefficient is dependent on the ratio of non-condensable gas to steam masses. Application of the Uchida data during the long-term cooling period, and adoption of a transition heat transfer coefficient between the "Modified Tagami" value and the Uchida value (based on the reduction of turbulence in the Containment), completes the specification of the condensing heat transfer coefficient during LOCA transients.

The heat transfer coefficient between the water regions of the sump and the basemat is conservatively neglected. A natural convection coefficient of 1 Btu/hr-ft^2-°F is assumed for the outer surfaces of the Containment Structure, which are exposed to the outside atmosphere. Zero heat transfer is specified at external surfaces exposed to the earth and at the liquid-vapor interface between the containment sump and atmosphere regions. These thermal boundary conditions contribute to prediction of conservative values of pressure and temperature within Containment.

c. <u>Heat Exchanger Calculations</u>

The GOTHIC Code provides models for both water-to-water heat exchanger and fan coolers. The heat exchanger model is used to determine the rate of heat removal from the sump during the long-term cooling period of the LOCA. Four types of heat exchangers may be specified with several optional combinations of containment spray or core SI water cooling (using either a single, two parallel, or two series units).

The available heat exchanger types are parallel flow, counter flow, cross-flow with mixed secondary flow, and parallel-counter flow (shell & tube) with mixed shell fluid. The physical and thermal characteristics of each heat exchanger are specified, as are the coolant inlet temperatures and flow rates.

The heat exchanger performance calculations are made using the method of heat transfer effectiveness as developed by Kays and London, Reference 14. This method employs the number of exchanger transfer units to evaluate an effectiveness for heat transfer. The effectiveness is defined as the ratio of the actual heat transfer to the maximum theoretical heat transfer. An effectiveness is calculated for any of the four heat exchanger types specified, and from the effectiveness, the heat exchanger duty, and system temperatures are determined.

<u>Safety Injection System</u>

The effect of the SI System on the containment pressure and temperature analysis is simulated in the GOTHIC model by a provision for pumped coolant injection into the reactor vessel region during the post-blowdown period. The accumulation of water in the reactor vessel region, the addition of heat to the region, and the effects of the pumped SI are all accounted for in the reactor vessel region calculations.

Safety injection water may be supplied from an external source or from recirculation from the containment sump. Variable injection flow rates and temperatures may be specified. The injection flow or a fractional part of the injection flow may be added directly to the containment sump, thereby permitting any combination of system ruptures to be simulated. Safety injection water supplied by sump recirculation may be passed through the heat exchangers.

The actions of core filling systems provided by high pressure injection or flooding are normally provided for in the NSSS blowdown data as input mass and energy entering the Containment from the primary system.

<u>Containment Cooling Systems, Air Coolers</u>

The GOTHIC Code is capable of simulating the effects of containment ESFs. These include the addition of either externally-supplied or internally-recirculated containment spray water and the cooling of the atmosphere region by fan cooling units.

Although CACs and the entire Service Water System can be explicitly modeled with GOTHIC, the air cooler heat removal rate as a function of steam saturation temperature is specified as boundary condition consistent with the COPATTA methodology. The number of units and the starting and terminating times for these units depend upon the assumed single-failure and operation of the containment cooling systems.

The effect of saturation temperature of the steam-air mixture on the heat removal capability of the CACs, assuming constant service water inlet temperature (105°F) and air flow rate (55,000 cfm), is shown in Figure 14.20-1, and is calculated in Reference 15.

<u>Containment Cooling Systems, Spray Trains</u>

The GOTHIC Code is capable of simulating the effects of containment sprays. Although Containment Spray Systems can be explicitly modeled with GOTHIC, a

spray flow rate and enthalpy is specified as a boundary condition to the GOTHIC model, which is consistent with the COPATTA methodology.

The Containment Spray System flow rate, inlet temperature, delay time after actuation, and termination time are specified as input data to the GOTHIC model. The water supply may be external or internal. Prior to RAS, the RWT is the source for the spray water. Following RAS, internally-recirculated water from the sump as a function of time is passed through the SDCHX before it is sprayed into the containment atmosphere as described in the portion of Section 14.20.2.3 labeled “Heat Exchanger Calculations.”

The heat removed from the atmosphere region due to spray water heating is added to the sump water. In a similar manner, any condensate due to this atmosphere cooling is also added to the sump.

To account for condensation degradation due to the presence of the non-condensable gases, containment codes such as COPATTA, CONTRANS, and CONTEMPT use a so-called spray efficiency or effectiveness. The GOTHIC Code does not require such a multiplier due to the explicit treatment of drops.

14.20.2.4 Inputs and Assumptions

Initial Conditions

Mass and Energy Analysis:

The initial plant conditions for the LOCA were selected to maximize the release of mass and energy to the Containment. The worst case 102% power level was evaluated for this effort. Presented in Table 14.20-2 is a summary of the key inputs and assumptions for the limiting DES/PD maximum SI case of this effort.

The inputs were selected to contribute to conservative predictions of the primary and secondary side mass and energy. Key inputs like initial SG water and steam mass, core flow rate, T_{cold}, SG tube heat transfer via SG tube plugging and tube removal, RCS and SG volumetric expansion, core power, and primary and secondary side metal mass were selected to achieve this condition. The trip logic associated with isolating main steam and MFW flow was also carefully biased, such that a conservative response would be obtained.

Containment Pressure and Temperature Analysis:

The containment pressure and temperature analysis input data have been developed based upon the design for the plant. A thorough compilation of geometric, thermodynamic, design, and initial operating conditions was prepared prior to the evaluation of the postulated LOCA. These physical and performance conditions were determined based upon estimates of design parameters that would contribute to conservative predictions of containment pressure and temperature. A summary of the data is given below.

The containment parameters used for the pressure and temperature transient analyses are given in Table 14.20-3. The net free internal volume was computed from a gross volume and occupied volume calculation, Reference 16. The initial thermodynamic conditions (pressure and temperature) in the Containment are the maximum expected values based on Technical Specifications.

The containment heat sink data specified for the heat transfer calculations during the LOCA are given in Tables 14.20-4 and 14.20-5, and are calculated in Reference 17. Table 14.20-5 lists the geometric configuration of the heat sinks,

including the materials and thermodynamic properties, thicknesses, and surface areas. A revised analysis was performed with the GOTHIC Code to determine the effect of the degraded properties of a new coating on the containment response to design basis accidents. The conservatively determined heat transfer coefficients used in the analysis are given in Table 14.20-4. These data, plus the geometric data, completely specify the necessary heat sink conditions for the calculations of heat transfer to and within the structures of the Containment during the LOCA.

Concurrent Events

A LOOP at the initiation of the event was assumed for the limiting cases evaluated in this analysis. Although cases with offsite power available were evaluated, the LOOP case produced a slightly more severe containment peak pressure and temperature response. No other concurrent events were assumed in this analysis.

Single Failures

A range of single failures was considered in the LOCA analyses for the three break locations analyzed (References 19 through 21). Failures involving the containment heat removal systems had the greatest negative impact. Failure of a diesel generator (coupled with a LOOP), which disables one train of spray and one train of containment coolers, was shown to be the worst case for the hot leg break. However, the failure of one train of containment coolers, with SI flow maximized (i.e., with two diesel generators functioning), provided the most limiting results for a cold leg break. This is true because under this scenario, SI flow to the RCS and, ultimately the Containment, is maximized resulting in a more severe mass and energy release to the Containment. The DES/PD cold leg break with maximum SI flow provided the most severe peak containment pressure and temperature, slightly worse than the DES/PD break with minimum SI. Therefore, the limiting case presented is the DES/PD break with a LOOP and the failure of one train of containment coolers. This case is referred to as the ‘DES/PD maximum SI’ case below.

A passive failure, which disables all of component cooling water was considered. However, this failure was determined not to be limiting with respect to peak containment pressure and temperature. The passive failure need only be assumed to occur after a RAS. In addition, when a passive failure is assumed to occur, no active failure need be assumed. Therefore, all ESF equipment would be available prior to a RAS and the peak pressure and temperature for this scenario is not limiting.

Automatic RPS/ESFAS Functions

Presented below is a list of the functions credited in the limiting DES/PD maximum SI LOCA mass and energy release and containment response analysis:

RPS/ESFAS FUNCTION	**EQUIPMENT FUNCTION**	**ANALYSIS SETPOINT**	**RESPONSE TIME SIGNAL + EQUIP (sec)**
CSAS	MFW, Condensate Booster, HDPs Trip	4.75 psig	0.9 sec trip delay with 10 sec pump coast down
SIAS/CSAS	Containment Spray Initiated	4.75 psig	70.9 sec
SIAS	CACs Start	4.75 psig	35.9 sec
SIAS	HPSI and LPSI Inject	4.75 psig	15 sec[a]
RAS	SI Suction from RWT to Sump	47.5”[b]	N/A[c]

[a] This response time represents a minimum (during a LOOP) since SI flow makes containment response more severe.

[b] The analytical setpoint shown is the level above the bottom of the RWT. This setpoint is conservatively high since it minimizes the relatively cool RWT inventory pumped to Containment.

[c] No explicit response time is credited. However, the response time must be rapid enough to ensure continuous SI flow as the pump suction is transferred to the sump.

Note that for breaks as large as that analyzed in the limiting case, the reactor will shut down on voids, therefore no RPS trip (insertion of shutdown rods) is credited for the limiting case.

Other Equipment Safety Functions

The analysis of the containment pressure and temperature response to a LOCA credits the proper functioning of the ESFs. The performance of the ESFs for the containment design basis is given in Table 14.20-6. Except when affected by single failure assumptions, ESFs operate at 100% capacity. For the minimum SI cases, only one high-pressure pump, one low-pressure pump, one spray pump, and two fan cooling units are assumed to be in operation.

Operator Actions

Consistent with plant emergency procedures, for the maximum SI case, the operator is assumed to terminate the operation of one train of sprays when the pressure in the Containment decreases to 4.75 psig following the peak.

14.20.2.5 Results

Mass and Energy Release Results

Containment responses to design basis accident (DBA) LOCAs were reanalyzed due to the vendor using a coarse mesh to edit data. Using a finer mesh (0.05 seconds versus 0.5 seconds) increased the peak mass flow rate in the blowdown phase of the LOCA, which was obscured in the coarse mesh edit. Using a finer mesh (0.01 seconds) did not change the data. Also standard test procedures on the SI pumps demonstrated that higher total dynamic head were produced by these pumps than previously measured. Therefore, new sets of

mass and energy release data were produced by Westinghouse for five LOCAs using CEFLASH-4A, CEFLOOD3, and CONTRANS2 (Reference 49). These five sets include a double-ended split (slot) break LOCA on the pump discharge with the maximum set of SI pumps, a double-ended split (slot) break LOCA on the pump discharge with a minimum set of SI pumps, a double-ended split (slot) break LOCA on the hot leg with the maximum set of SI pumps, a double-ended split (slot) break LOCA on the hot leg with a minimum set of SI pumps, and a 2 ft^2 break LOCA on the hot leg with a minimum set of SI pumps.

Using the above sets of mass and energy release data it was determined (References 46 through 48) that a double-ended split (slot) break LOCA on the pump discharge with a minimum set of SI pumps available results in highest peak pressure in the Containment. With an adjustment in the initial containment pressure in the Technical Specifications from 1.8 psig to 1 psig, the peak pressure of the most limiting LOCA is below the design limit of 50 psig (References 50 and 52). Figure 14.20-2 shows containment pressure as well as vapor and water temperatures. The sequence of events is shown in Table 14.20-7.

Containment response to a LOCA is also applicable to both RSGs. This is because the primary side of both OSG and RSG are nearly identical.

14.20.2.6 <u>Summary of LOCA Analysis and Effect of the RSGs</u>

The effect of the RSGs on the LOCA containment response was evaluated in the mid-1990s. The SG parameters that affect peak pressure and temperature during the large-break LOCA are the primary liquid mass and energy, and total system sensible heat. The peak pressure and temperature reached during the blowdown phase are solely a function of primary system mass. Because the RSG has a slightly smaller primary volume, it will have less primary mass and a slightly lower blowdown peak. Conservatively using the difference at cold conditions, the decrease in energy would be 662,000 Btu. The second, higher peak is reached during the reflood phase of the LOCA. The reflood mass release rates are not significantly affected by the RSGs because the RSG flow resistance at zero plugging is within one percent of that of the OSGs. The energy released to Containment at this time is dictated mainly by fuel-stored energy removal and secondary pressure. The RSGs will operate at a higher pressure than that assumed in the analysis. The increase in energy caused by the higher secondary pressure was compensated for by the decrease in energy due to the primary volume effect. Therefore, the peak pressure with the RSGs is bounded by the peak pressure with the OSGs.

The long-term pressure and temperature are controlled by total system sensible heat and the containment heat removal systems. The total sensible heat of the RCS with the RSGs in place was determined to be virtually identical to that of the RCS with the OSGs. Because the primary system volume is less, and the system sensible heat is virtually identical, the results of the current analysis remain applicable to the RSGs.

14.20.3 MAIN STEAM LINE BREAK

14.20.3.1 <u>Description of Event</u>

The MSLB containment event is characterized by the rapid blowdown of steam into Containment due to a rupture in the main steam line. The location of this break is at the SG outlet nozzle, upstream of the MSIVs. This location results in the largest possible break size. The initial portion of the transient is characterized by the blowdown of both SGs, including the main steam lines downstream of the

MSIVs. In this early phase of the event, steam also continues to flow to the turbine, until the reactor trips. Following the reactor trip, which occurs on containment high pressure, the turbine stop valves close. During this portion of the transient, MFW continues to feed the SGs.

Coincident with the reactor trip, CSAS and SIAS occur on containment high pressure. Containment spray actuation signal initiates the closure of the MSIVs, closure of the MFIVs, trips the SGFPs, CBPs and HDPs, and opens the containment spray valves. Safety injection actuation signal switches the CACs from the normal operation fast speed mode to the accident condition slow speed mode of operation.

Following the closure of the MSIVs, the contribution of steam to Containment from the intact SG ceases. This SG remains isolated for the reminder of the event. The remaining steam downstream of the MSIVs to the turbine stop valves is also isolated upon MSIV closure, and no longer contributes to the blowdown to Containment.

The isolation of MFW in response to the CSAS occurs via tripping of the SGFPs, CBPs, and HDPs. While some preferential flow is diverted to the ruptured SG due to the relatively high pressure difference between the SGs, the contribution of MFW flow is greatly reduced with the pump trips on high containment pressure. Following SGFP coastdown, the MFIVs shut within approximately 65 seconds to fully isolate MFW flow. The feedwater regulating valves and their associated bypass valves act to ramp feedwater flow to approximately 5% of rated feed flow within 20 seconds of the turbine trip; however, this function is not credited in this analysis. Upon closure of the MFIVs, feedline inventory upstream of the SG feedwater inlet nozzle to the MFIVs will flash into the ruptured SG.

In the event that the MFIV fails to be capable of closing completely under the analyzed conditions, action of the feedwater regulating valves and their associated bypass valves may be credited, along with a fully open MFIV, to isolate MFW flow to the affected SG.

During the initial phase of the event, the only source of heat removal is via condensation heat transfer to the heat sinks or containment walls. Following the generation of the CSAS, the containment spray line valves will open to allow spray flow to enter Containment at approximately one minute into the event. The containment spray pumps will start in response to a SIAS, also on containment high pressure.

To supplement the active containment heat removal provided by the containment sprays, the CACs also start in response to SIAS. This signal switches the fan speed from fast to slow, and also opens the appropriate service water valves to allow more cooling flow to the CAC units. There are two containment spray trains and four CACs (two trains of two) inside Containment. The operation of the containment sprays and CACs aids in terminating the increase in containment pressure and temperature.

Auxiliary feedwater is actuated on low SG level during the MSLB. Since the SG pressure differential between the ruptured and intact units quickly diverges due to the double-ended guillotine break, the high SG differential pressure setpoint for blockage of flow to the ruptured SG is quickly reached. Following approximately a 20-second delay for AFW block valve closure, the AFW flow is terminated. Thus, a relatively small integrated amount of AFW actually enters the ruptured SG for contribution to the overall mass and energy release.

With AFW isolated, the ruptured SG boils dry, thus terminating the mass and energy release to Containment. At this point, the active and passive containment heat removal devices continue to decrease containment pressure and temperature to their initial values.

14.20.3.2 Mass and Energy Release Methodology

The sensitivity study performed on the mass and energy release for the MSLB event, References 36 and 44, focused on a matrix of cases. This matrix included four different initial power levels and numerous different single failures. The goal of the analysis was to maximize the severity of the mass and energy release, which in turn maximizes the containment pressure and temperature response.

The methods followed for this analysis are consistent with FTI's NRC-approved RELAP5/MOD2-B&W MSLB methodology (Reference 45). A detailed feedwater system is included in the reference such that the modeling of feedwater flow is similar to that approved by the NRC as part of the NRC Bulletin 80-04 analyses for Calvert Cliffs.

A summary of the significant assumptions or methods which have been included in this analysis are provided in Table 14.20-8.

RELAP5/MOD2-B&W is the primary computer code used to model the mass and energy release during a MSLB. This model includes both primary and secondary systems, coupled through the SG tubes. As such, the model accounts for the reactor core, vessel, RCS piping, RCPs, and the pressurizer. The model for the secondary side of the SG includes the main feedwater and the main steam systems. Nuclear heat production in the core is modeled by one group point kinetics with six delayed neutron groups. The neutronics model also accounts for the effect of such thermal feedback as Doppler and moderator temperature on reactivity. Shutdown CEAs and decay heat generation are also modeled. As discussed in Section 14.20.3.5, the input data were conservatively biased to maximize the mass and energy release.

Framatome Technologies, Inc. used the NRC-approved containment code CONTEMPT to predict the timing of the containment high pressure set point.

The RELAP5/MOD2-B&W Code was used to calculate the contribution of main feedwater, including flashing, to the ruptured and intact SGs. This code was used to simulate the Calvert Cliffs MFW trains, which included SGFP, CBPs, and HDPs. These trains were represented by a total of 51 nodes, pumps, valves, and feedwater heaters. The resistance in the feedwater piping was conservatively minimized and no credit was taken for the closure of the MFW regulating valves. Although the feedwater heaters were conservatively represented as hot slabs, this analysis used a realistic representation of the cooldown of the feedwater heaters following the isolation of shell side steam flow following turbine trip.

A realistic coastdown of all feed train pumps was modeled after the time to reach the CSAS setpoint on containment high pressure.

14.20.3.3 Containment Response Analysis Methodology

The containment pressure and temperature response to a MSLB is calculated using the GOTHIC computer program, References 36 to 44. The program model description and thermodynamic assumptions are provided in Section 14.20.2.3. The primary differences between the LOCA analysis and the MSLB analysis are:

the reactor vessel region model is not employed in the MSLB analysis; and the Uchida correlation is used for the heat transfer coefficient to the structural heat sinks in the MSLB, rather than the Tagami, as described in the portion of Section 14.20.2.3 labeled “Heat Transfer Coefficients.”

14.20.3.4 <u>Inputs and Assumptions</u>

<u>Initial Conditions</u>

Mass and Energy Analysis:

The initial plant conditions for the MSLB analysis were selected to maximize the mass and energy release. The initial power levels assumed for this analysis were 102%, 75%, 50%, and 0% of 2700 MWt. An additional 17.1 MWt was included to account for the rate of heat addition from the RCPs to the reactor coolant. Since four power levels were evaluated in this analysis, a number of power dependent inputs were adjusted to conservatively reflect plant conditions for each power level. Presented in Table 14.20-9 is a summary of the key inputs and assumptions for the limiting case of this analysis. The design basis assumption of a full double-ended guillotine break with 0% moisture carryover is used in this analysis.

The RELAP5/MOD2-B&W inputs for each power level were selected to maximize the primary and secondary side inventories and energies. Key inputs, like initial SG water and steam mass, feedwater flow, T_{cold}, SG pressure, primary-to-secondary heat transfer via SG tube plugging and tube removal, RCS flow rate, RCS and SG volumetric expansion, core power, and primary and secondary side metal mass, were selected to achieve this condition. The trip logic associated with isolating the intact SG as well the MFW flow was also carefully biased, such that a conservative response would occur. Since the diversion of AFW to the intact SG reduces the energy of the RCS (albeit a small amount), AFW flow to the intact SG was conservatively omitted in this analysis.

Since a number of trade-offs are present in the MSLB event, the determination of the most limiting case is only possible by the computer code simulation. One key trade-off exists in the power level versus inventory and feedwater flow inputs. As initial power level increases, RCS temperature and core decay heat increases and more primary-to-secondary energy is present to boil off the SG inventory. Main feedwater flow rate and enthalpy increase accordingly. However, initial SG inventory decreases with increasing initial core power. Therefore, the MSLB analysis includes an evaluation of multiple power levels.

Containment Pressure and Temperature Analysis:

The containment pressure and temperature analysis input data have been developed based upon the design of the plant. A thorough compilation of geometric, thermodynamic, design, and initial operating conditions prior to the hypothetical occurrence of an MSLB has been prepared. These physical and performance conditions were determined based upon conservative estimates of the most adverse design parameters with respect to maximizing containment pressure and temperature. The initial plant conditions assumed in the analysis of the containment response to an MSLB are provided in Table 14.20-10. In addition to the Containment initial conditions, this table also lists several of the key assumptions concerning the actuation and performance of the CACs and the containment sprays.

<u>Concurrent Events</u>

A LOOP at the initiation of the event with the coincident failure of one diesel generator was evaluated as part of this analysis. It was determined that the LOOP

scenario produces a less severe containment response than with offsite power available. This is due to the continued operation of the RCPs with offsite power available, which maximizes the primary-to-secondary heat transfer. This offsets the loss of one train of CACs and one train of containment sprays for the LOOP scenario. Although the limiting case did include the worst single failure, no concurrent event (such as LOOP) was assumed.

Single Failures

The updated MSLB containment analysis included a detailed single failure analysis. Each of these failures was evaluated at four different power levels: 102%, 75%, 50%, and 0%. Note that for each case, no credit was taken for the closure of the main feedwater regulating valves, both the feedwater regulating valves and the associated bypass valves were assumed to fail as is for the duration of the analysis. This assumption is consistent with the NRC Bulletin 80-04 evaluation conducted to address the effects of extended feedwater addition. Presented below is a list of the single failures addressed in this analysis.

a. Loss-of-offsite power and one emergency diesel fails to start, resulting in the loss of one spray train and two CACs. This will leave one spray pump and two CAC units available. Reactor coolant pumps coast down on loss of power. This represents the LOOP case.
b. A service water pump fails which disables a train of CACs. This will leave two containment sprays and two CAC units (one train) available.
c. One containment spray pump fails to actuate. This will leave one containment spray and four CAC units available.
d. The CSAS fails to trip a SGFP.
e. The CSAS fails to trip a CBP.
f. The CSAS fails to trip a HDP.
g. The failure of a MSIV to close. This is the MSIV on the ruptured side since the steam in the steam lines downstream of the other MSIV will blowdown to Containment.
h. The failure of an MFIV to close.

The single failure of a service water pump (which disables one train of CACs) was shown to be more limiting than the loss of a containment spray pump. As expected, the LOOP with the coincident failure of a diesel generator was not a limiting case due to the tripping of the RCPs. This degraded the primary-to-secondary heat transfer, which drives the boil-off process.

The failure of an MSIV or MFIV was not shown to be a limiting single failure. In the case of the MSIV failure, the additional steam downstream of the MSIV on the intact SG was not significant relative to other more severe feedwater-related single failures. For the MFIV failure, the flow rate from the condensate pump was not significant and the water injected was relatively cool.

The most limiting case was determined to be the 75% power case with the failure of a SGFP to trip in response to an SGIS (CSAS) on containment high pressure. This event assumed the availability of offsite power and continued operation of RCPs since this condition results in a more severe containment response. Containment pressure and temperature for the RSG MSLB are shown in Figure 14.20-3.

Automatic RPS/ESFAS Functions

The following trips were relied upon for the mass and energy release analysis.

RPS/ESFAS FUNCTION	EQUIPMENT FUNCTION	ANALYSIS SETPOINT	RESPONSE TIME SIGNAL + EQUIP (sec)
RPS Trip on High Containment Pressure	Reactor Trip Breakers Open	4.75 psig	0.9
CSAS	MSIV Closure	4.75 psig	0.9 + 7.0 = 7.9
CSAS	MFIV Closure	4.75 psig	0.9 + 65 = 65.9 (No LOOP)
CSAS	SGFPs, CBPs, and HDPs trip on CSAS	4.75 psig	0.9 signal delay with 1 sec pump trip delay
AFAS	AFW Initiated to SG	66.5“ below normal water level	0.9 + 20.0 (timer delay) = 20.9 (minimum)
AFAS Block	AFW to Ruptured SG Isolated	250 psid	0.9 + 20.0 = 20.9
SIAS/CSAS	Containment Spray Initiated	4.75 psig	0.9 + 62.0 = 62.9 (No LOOP)
SIAS	CACs Start	4.75 psig	0.9 + 10.0 = 10.9 (No LOOP)

Other Equipment Safety Functions

For the MSLB containment analysis, the ESFs credited are the CACs and the containment sprays. Since the worst single failure has been identified to be failure of a SGFP to trip, all four CACs and both trains of containment sprays are assumed to be in operation. A summary of the performance parameters for these two ESFs is provided in Table 14.20-11.

The MSSVs were relied upon to remove energy from the intact SG. The opening setpoint of the first bank was 1010 psia, while the full open pressure of the last bank, including accumulation, was 1112.4 psia.

Operator Actions

No operator actions are credited in the MSLB analysis.

Status of Non-Safety-Related Control Systems

A normal initial SG water level is assumed. In addition, while the majority of the components relied on to trip the MFW-related pumps on CSAS are safety grade, a few of the components are non-safety-related. The operability of these non-safety-related components, however, was accepted by the NRC in Reference 18 during the extended MFW addition analysis conducted to respond to NRC Bulletin 80-04.

Closure of the turbine stop valves is assumed following reactor trip. These valves are closed instantaneously following reactor trip. Since delaying the closure of the turbine stop valves would reduce the severity of energy content of the SGs, this response is conservative. Closure of the turbine stop valves is not a safety function in this instance since their rapid closure makes containment response slightly worse.

14.20.3.5 Analysis Results

Mass and Energy Release Results

Mass and energy release data were calculated for a matrix of cases. As discussed in the portion of Section 14.20.3.5 labeled "Single Failures," this matrix included eight different single failures, each evaluated at four different power levels. Framatome Technologies, Inc. used the NRC-approved CONTEMPT containment code to determine the most limiting single failure at each power level, which resulted in the determination of the single most limiting event. The most limiting case was determined to be an MSLB initiated from 75% power with the failure of a SGFP to trip in response to a CSAS.

In general, when considering the range of single failures, the CONTEMPT analysis showed an increase in peak containment pressure with increasing power level. This is due to the increase in feedwater flow rate as power increases. For the limiting single failure (being the failure of SGFP to trip), the initial SG pressure and the rate of depressurization contributed to increased flashing and reduced rate of feedwater flow at full power. This resulted in the prediction of a lower peak pressure in the Containment. For the zero power case, the large initial SG inventory was the dominant effect. However, the peak pressures were less than those for the full power case.

Table 14.20-12 provides a sequence of events for the limiting case. For completeness, this table includes both mass- and energy-related, as well as containment response-related, information.

Containment Response Results

The pressure and temperature profiles for the containment response to the MSLB are calculated in Reference 43 and provided in Figure 14.20-3. Results are shown in Table 14.20-12.

The effect of increasing the full power SG mass inventory on the MSLB containment response was evaluated. The SG mass inventory may increase as a result of partially bypassing the pressure feedwater heaters, which lowers the feedwater inlet temperature to the SGs. The evaluation concluded that the effect of this increase in SG inventory would be to increase the peak containment pressure by less than 0.5 psi with no increase in peak containment temperature. Therefore, peak pressure in the Containment during a MSLB will remain below 50 psig.

14.20.3.6 Summary of MSLB Analysis

Framatome Technologies, Inc. produced the MSLB break mass and energy transfer rates from the RSGs into the Containment using RELAP5/MOD2-B&W Code. The single failure resulting in the most limiting containment response is due to the failure of SGFP to trip. To ensure a set of conservative mass and energy transfer rates, FTI performed a spectrum of analysis based on the reactor power level and single failure criterion. Various reactor power levels of 0%, 50%, 75%, and 102% were examined. The largest break size is limited to the venturi flow area of 1.9 ft^2 installed in the RSG steam outlet. In the calculation of the break mass and energy transfer rates, a 0% moisture carryover was assumed. The GOTHIC Code was used to predict the containment response to a MSLB using the FTI produced mass and energy transfer rates. Steam superheat was predicted in the Containment for a short duration of about 90 seconds.

14.20.4 SUBCOMPARTMENT ANALYSIS

14.20.4.1 Methodology

For the original compartment designs, the occurrence of a LOCA was postulated to result from the rupture of the RCS piping, including the main loop piping, either within the reactor cavity or the SG compartments of the Containment. For the current compartment designs, breaks are not postulated to occur in the main loop piping based on the Leak-Before-Break Evaluation documented in References 1 and 35. Since only main loop piping is present in the reactor cavity, no LOCA is currently postulated to occur in this compartment and no analysis is presented herein for the reactor cavity. However, smaller bore piping connected to the RCS is present in both SG compartments and the pressurizer compartment. Since Reference 1 only applies to main loop piping, a LOCA is postulated to occur in these compartments, and the original design analyses are presented herein for these compartments as they remain bounding.

Subcompartment analyses are made with the Bechtel computer program COPDA, Reference 30, and its predecessor COPRA, Reference 31, which calculate the transient pressure and temperature responses of interconnected volumes subject to high energy pipe break accidents. The differential pressures across structural components are calculated based on the time-dependent compartment pressures. COPDA (as well as COPRA) calculates a mass and energy balance of the two-phase, two-component steam-water-air mixture as reactor coolant enters the compartment during the LOCA and exits through vents and openings into adjacent compartments or into the main Containment Structure. There is no provision in the code for heat transfer to structures or for operation of ESFs, since these options generally have a negligible effect on compartment pressures for the short time following the rupture within which peak differential pressures occur. Nor is credit taken for operator action since peak differential pressures occur so quickly (well before the time typically assumed for operator action) after the start of the transient.

A pipe rupture of this type, within a compartment, results in the expulsion of high enthalpy water out of the ruptured pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-water-air mixture flows through the openings into the adjacent compartments or the main containment. The maximum differential pressure achieved between the compartments can determine the design strength of walls and supports between the compartments. The maximum pressure differential will depend on the number and shape of the openings leading between compartments, the volume of each compartment, and the blowdown rate from the broken pipe. The ensuing flow from the cavity or compartment follows orifice flow relations with the entrance and friction losses included in the flow coefficient for each case. The vent area from the cavity or compartment controls the differential pressure transient.

14.20.4.2 Inputs and Assumptions

Steam Generator Compartments

For the SG compartment, only the double-ended guillotine break in the hot leg is considered since it provides the largest rate of mass and energy release. The relief areas are divided into three classes: long sharp-edged nozzles; sharp-edged orifices; and well-rounded orifices. A coefficient of 0.61 is applied to the areas of the long nozzles, and a coefficient of 0.97 is applied to the areas of the well-rounded orifices. Coefficients for the sharp-edged orifices are supplied by the computer code. The initial conditions, main containment volume, SG compartment volumes, and total relief flow areas are listed in Table 14.20-13.

Pressurizer Compartment

The upper portion of the pressurizer is enclosed in a box-like structure that has been analyzed as a separate compartment. Because the nozzle for the pressurizer relief valve line is located within the boundaries of this compartment, a subcompartment analysis was performed using the mass and energy release rates from a postulated break at the nozzle. The initial conditions for the compartment volume and relief flow area are listed in Table 14.20-13. The relief openings consist of ventilation openings at the top of the compartment and the opening around the lower portion of the pressurizer. The latter is available only after a baffle plate, made to withstand normal ventilation pressure, blows open at a design pressure differential of 5 psi.

Jet Forces

Jet force pressures associated with postulated primary coolant pipe breaks are considered in the design of walls and slabs adjacent to those pipes. Both the slot and guillotine types of break are investigated. The opening area of a break is assumed to be, at most, the inside cross-sectional area of the pipe. After a break has been fully developed, the escaping reactor coolant produces a thrust, or a reaction on the piping system. The maximum magnitude of this thrust is found to be equal to system operating pressure times the rupture area, or the pipe inside cross-sectional area based on investigations of the Calvert Cliffs system blowdown data. The expanding reactor coolant which forms the jet plume is assumed to diverge at a half-angle of 10° from the break opening. The total jet force is assumed to be equal to the thrust reaction.

The jet impingement area is computed by using the distance from the postulated break to the target structure and the 10° half-angle divergence. The total jet force is assumed to act on this area and thus the jet pressures are found. Those breaks that result in highest pressures and/or stresses are chosen as the governing ones for each wall, slab, or their portions thereof.

14.20.4.3 Containment Internal Structure Evaluation Results

The results of the cavity pressurization analyses are listed below. The calculated compartment volumes and relief areas, combined with the computed mass and energy discharge rates, were used with the COPDA/COPRA model to ensure an accurate analysis of the differential pressures. The safety margins used in the design of the compartment walls are indicated in UFSAR Chapter 5.

For the SG compartments, the maximum pressure differentials across the compartment walls are 16.4 psi and 18.4 psi for the east and west compartments, respectively. These occur at about 0.1 seconds.

For the pressurizer compartment, the maximum differential pressure of 5.02 psi occurs 0.10 seconds after the break of the relief valve nozzle. This is the worst pressure gradient that can result from a break within this compartment; however, this compartment will be exposed to a greater pressure differential due to a LOCA hot leg break in the adjacent SG compartment. Hence, the pressurizer compartment walls are designed to the same differential pressure loads as the SG compartment.

COMPARTMENT	MAXIMUM ∇P, psid	CONCURRENT COMPARTMENT PRESSURE, psig
East SG Cavity	16.4	18.7
West SG Cavity	18.4	20.8
Pressurizer Cavity	16.4	18.7

The resulting jet pressures on the containment internal compartments vary from 30 ksf to 76 ksf. However, these loads act on, at most, one-third of the span for one-way walls or slabs.

14.20.5 CONCLUSIONS

The peak containment pressure for both the LOCA and MSLB DBEs remains below the containment design pressure of 50 psig. Although the peak vapor temperature exceeds the containment design temperature of 276°F, this occurs only for a brief period during a time when the containment atmosphere is superheated. Due to steam condensation on the colder surfaces, the surface temperature of revised structures and mechanical components in contact with the containment atmosphere remains at or below the saturation temperature for the steam in Containment. This temperature is below the design temperature of 276°F. The pressure differentials and jet impingement forces that could occur across the walls of enclosed compartments have been analyzed and incorporated in the current compartment design. Therefore, since the methods used to predict the peak pressure, temperature, and jet forces are conservative, the containment design is adequate and contains sufficient margin. This conclusion is consistent with the NRC Safety Evaluations documented in References 32, 33, and 34, that established the acceptability of the original containment design.

This event is not affected by the transition to AREVA Advanced CE-14 HTP™ fuel because the key parameters for this event are plant related system responses which are unchanged from, or bounded by, the current analysis. These parameters are not adversely affected by either the transition cycle or full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

14.20.6 REFERENCES

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33. Letter from Mr. K. R. Coller (NRC) to Mr. J. W. Gore, Jr. (BGE), dated May 4, 1973, Supplement No. 1 to the Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Baltimore Gas and Electric Company Calvert Cliffs Nuclear Station, Units 1 and 2 Docket Nos. 50-317 and 50-318, dated August 28, 1972

34. Letter from NRC to BGE, dated December 4, 1973, Supplement 2 to the Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Baltimore Gas and Electric Company Calvert Cliffs Nuclear Station, Units 1 and 2 Docket Nos. 50-317 and 50-318, dated August 28, 1972

35. Letter from Mr. D. G. McDonald (NRC) to Mr. R. E. Denton (BGE), dated February 3, 1994, Installation of a Neutron Shield/Pool Seal at the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M87176 and M87177)

36. Framatome Technologies, Inc. Calculation 32-5004396-04, “Calvert Cliffs SLB for Containment,” September 2000

37. GOTHIC Containment Analysis Package, Technical Manual, Version 8.0 (QA), NAI 8907-06, Revision 19, January 2012

38. GOTHIC Containment Analysis Package, User Manual, Version 8.0 (QA), NAI 8907-02, Revision 20, January 2012

39. GOTHIC Containment Analysis Package, Qualification Report, Version 8.0 (QA), NAI 8907-09 Revision 12, January 2012

40. Comparison of Results from GOTHIC to Results from Other Codes, NAI 9301-05, January 1996

41. GOTHIC Containment Analysis Package, Installation and Operations Manual, Version 8.0 (QA), NAI 8907-08 Revision 20, January 2012

42. Simulation of GOTHIC Large Scale Containment Experiments, Battelle Ingenieurtechnik GmbH, Eschborn, October 1995

43. Calculation CA06774, Revision 02, “Containment Response to LOCA & MSLB for Calvert Cliffs Units 1 & 2,” October 2008

44. FTI Calculation 5004396-03, Rev. 04, “Calvert Cliffs SLB for Containment,” Calculation CA05684, Revision 000, February 2001

45. Framatome Technologies, Inc. Topical Report BAW-10169P-A, “RSG Plant Safety Analysis,” October 1989

46. CA07460, Revision 0000, “Containment LOCA Blowdown Mass & Energy Release Data with Increased SI Pump Flow for Calvert Cliffs Units 1 & 2”

47. CA07461, Revision 0000, “Containment LOCA Reflood/Post-Reflood Mass & Energy Release Data with Increased SI Pump Flow for Calvert Cliffs 1 & 2”

48. CA07462, Revision 0000, “Containment LOCA Long-Term Decay and Sensible Heat Data with Increased SI Pump for Calvert Cliffs 1 & 2”

49. Westinghouse Electric Company letter and attachments, LTR-OA-10-84, Revision 0, “Transmittal of Verified Mass & Energy Release and Long Term Heat Rate Data for Calvert Cliffs LOCA Containment Analysis,” August 6, 2010

50. CA07725, Revision 0000, “LOCA Containment Peak Pressure Analysis”

51. Not used

52. CA07786, Revision 0000, “Containment Response Analysis in Support of CCNPP Units 1 & 2 License Amendment Request”

TABLE 14.20-1

SUMMARY OF SIGNIFICANT ASSUMPTIONS FOR LOSS-OF-COOLANT ACCIDENT MASS and ENERGY RELEASE METHODOLOGY

Sources of Energy

- Reactor power and decay heat, including metal water reaction
- Stored metal energy in the core and primary system, including the core and reactor vessel internals
- Steam generator steam and liquid inventory
- Steam generator metal energy, including tubes
- Main Feedwater prior to feedwater trip

Sources of Mass Release

- Primary system inventory
- Water and nitrogen from SITs
- Refueling water tank

Mass and Energy Release Calculations

CEFLASH-4A

- A double-ended slot break was assumed for the RCP suction and discharge cold leg breaks.
- A spectrum of break sizes was investigated for the hot leg break.
- A LOOP was assumed at the initiation of the LOCA. The maximum SI cases assumed the operation of two diesel generators, while the minimum SI cases assumed the failure of a diesel generator.
- The reactor is shut down because of voiding in the RCS for cold leg breaks and by shutdown control rod insertion for the hot leg break.
- The core is allowed to stay in pre-DNB condition except when in heat transfer to steam. In addition, it is allowed to return to nucleate boiling as conditions permit throughout the transient.
- A two-phase heat transfer coefficient, via the Jens-Lottes correlation, is used to calculate core to coolant heat transfer in the nucleate boiling regime.
- The critical flow of mass discharge was modeled via the Henry-Fauske/Moody choked flow model.
- Main Feedwater was conservatively modeled and then isolated in response to the MFW pump trip via SGIS on high containment pressure.
- Steam flow was conservatively isolated by rapid closure of the turbine stop valves following reactor trip on high containment pressure.
- Volumetric expansion due to pressure and temperature considerations was included in both the primary and secondary systems.
- A minimum Technical Specification RCS flow rate was assumed to maximize the energy release to Containment.
- Nominal RCS loop flow resistances were assumed.
- The initial cold leg temperature was maximized to increase the energy release to Containment.
- Consistent with the LOOP assumptions, the RCPs were tripped at the initiation of the LOCA.
- Decay heat was based on the original Combustion Engineering standard decay heat curve, with a multiplier of 20%. The curve is a conservative predecessor to the 1971 ANS Standard Curve.
- A nominal SIT liquid volume, initiated at the maximum Technical Specification pressure was conservatively assumed.

TABLE 14.20-1

SUMMARY OF SIGNIFICANT ASSUMPTIONS FOR LOSS-OF-COOLANT ACCIDENT MASS and ENERGY RELEASE METHODOLOGY

SIT Nitrogen Discharge

- The discharge of nitrogen from the SITs was conservatively modeled via a quality assured utility code, the main equations of which were extracted from the NRC approved model in the ABB/CE COMPERC Code topical report (CENPD-134P), Reference 29.

FLOODMOD2 Simulation

- In accordance with Section 6.2.1.3 of the SRP, the refill period was conservatively omitted. By this approach, the water level is assumed to be at the bottom of the active core at the end of blowdown. This results in a continuous high mass and energy release rate to Containment. Crediting the refill period would result in a time interval of decreased mass and energy release, which would allow the passive heat sinks time to reduce the severity of the transient containment response.
- Single failures resulting in both minimum and maximum SI pump flow cases were evaluated.
- Decay heat was based on the conservative 1971 ANS Standard Curve, with a multiplier of 20%.
- Nominal RCS loop flow resistances were assumed.
- Consistent with the design basis assumptions, credit was taken for 50% condensation of break flow in the limited interval from the annulus being full to the SITs completing the discharge of water. No credit for condensation due to SI pump flow was assumed. The condensation-related assumptions were consistent with the original FSAR design analyses.

CONTRANS Sensible Energy Addition Calculation

- Decay heat was based on the NRC Branch Technical Position ASB 9-2 Guidelines.
- Consistent with the reflood methodology for cold leg breaks, all primary and secondary system metal, and SG secondary inventory, was included in the calculation.
- Consistent with the blowdown assumptions, only the reactor vessel and internals energy below the elevation of the hot leg and the core stored energy contributed to the sensible heat addition.
- The pre- and post-RAS SI flow rates were consistent with Bechtel's long-term model assumptions.

TABLE 14.20-2

INITIAL CONDITIONS AND KEY ASSUMPTIONS FOR MASS AND ENERGY RELEASE ANALYSIS OF LOSS-OF-COOLANT ACCIDENT

	ITEM	VALUE/ASSUMPTION
1.	Methodology	
	Vintage	Current Day, Consistent with SRP
	Break Type	9.8 ft² Double-Ended Slot in RCP Discharge Leg (Suction leg and hot leg breaks also considered)
2.	Initial Nuclear Steam Supply System Parameters	
	Mode	1
	Power Level	102%
	Initial Primary Pressure	2250 psia
	Initial RCS Inlet Temperature	550°F
	Initial Secondary Pressure	815 psia
	Steam Generator Level	35.06' above tube sheet (Normal Water Level)
3.	Reactor Shutdown	On voids
4.	Reactor Coolant Pumps	Tripped at Initiation of Event
	Total RCS flow rate	370,000 gpm
5.	Safety Injection Pump Flow	Maximum SI (Two HPSIs/Two LPSIs) (Minimum SI 1 HPSI/1 LPSI also considered)
6.	Main Steam Isolation Valves	Not credited. SG isolation due to turbine trip
7.	Main Feedwater	
	Initial Flow Rate	102% of full flow
	MFW Enthalpy	425 Btu/lbm
8.	Main Feedwater Isolation Valves	Not credited, MFW contribution terminated following coastdown of SGFPs after SGIS pumps trip
9.	AFW Contribution	None

TABLE 14.20-3

CONTAINMENT PARAMETERS

Containment Design Pressure, psig	50
Containment Design Temperature, °F	276
Net Free Internal Volume, ft^3	1.989E6
Initial Pressure, psia	15.7
Initial Relative Humidity, %	20
Initial Inside Temperature, °F	125
Outside Temperature, °F	125
Service Water Temperature, °F	105
RWT Water Temperature, °F	100

TABLE 14.20-4

CONTAINMENT HEAT SINK THERMODYNAMIC DATA

HEAT SINK BOUNDARY	HEAT TRANSFER COEFFICIENT
Containment atmosphere to heat sink surfaces	"Modified Tagami" (Section 14.20.2.3)
Containment atmosphere to sump	0.00
Containment sump to heat sink surfaces	0.00
Containment walls and dome to outside atmosphere	1.0 Btu/hr-ft^2-°F

TABLE 14.20-5

CONTAINMENT HEAT SINKS

Heat Sink #1 - Containment Cylinder and Dome

Containment cylinder -	52080 ft^2
Containment dome -	21150 ft^2
	73230 ft^2

Material		Node Spacing (in.)	Thickness (in.)
Paint*		1.0×10^{-3}	18×10^{-3}
Primer		7.5×10^{-4}	3×10^{-3}
Carbon steel		6.25×10^{-2}	2.606×10^{-1}
Concrete region			
	2	5.4×10^{-2}	1.5
	3	2.1×10^{-1}	4.0
	4	1.3	6.5
	5	1.07	30.0

Heat Sink #2 - Miscellaneous Unlined Concrete

Steam generator compartment walls -	25508 ft^2
Equipment hatch loading platform -	800 ft^2
Reactor shield platform -	880 ft^2
Pressurizer wall and roof -	2040 ft^2
Refueling canal (outside) -	6750 ft^2
Fuel canal buttresses -	2432 ft^2
Steam generator buttresses -	3580 ft^2
	41900 ft^2

Material		Node Spacing (in.)	Thickness (in.)
Concrete region	1	6×10^{-2}	3.0
	2	1.2×10^{-1}	3.0
	3	1.2	12.0

(Exposed on one side to the containment atmosphere and insulated on the other.)

Heat Sink #3 - Outside Reactor Cavity (unlined concrete): Concrete – 6160 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Concrete	2×10^{-1}	12.0

(Exposed on one side to the containment atmosphere and insulated on the other.)

Heat Sink #4 - Galvanized Steel

Ventilation ductwork -	27788 ft^2
Grating -	51359 ft^2
Cable trays -	16436 ft^2
	95583 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Zinc	1.70×10^{-3}	3.4×10^{-3}
Carbon Steel	2.42×10^{-2}	9.7×10^{-2}

TABLE 14.20-5

CONTAINMENT HEAT SINKS

Heat Sink #5⁺ - Miscellaneous Steel 0.12 to 0.15 in. Thick

Polar crane -	20350 ft^2
Stairway framing -	1286 ft^2
Cable supports -	1872 ft^2
SG Platforms	1500 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	375 ft^2
	25383 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	1.0×10^{-3}	8×10^{-3}
Primer	7.5×10^{-4}	3×10^{-3}
Carbon Steel	9.79×10^{-3}	1.37×10^{-1}

Heat Sink #6⁺ - Miscellaneous Steel 0.18 to 0.24 in. Thick

Polar crane -	3759 ft^2
Stairway platforms -	1528 ft^2
Cable supports -	397 ft^2
Ventilation duct plenum supports -	623 ft^2
SG Platforms	3850 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	7383 ft^2
	17540 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	1.0×10^{-3}	8×10^{-3}
Primer	7.5×10^{-4}	3×10^{-3}
Carbon Steel	1.01×10^{-2}	2.03×10^{-1}

Heat Sink #7⁺ - Miscellaneous Steel 0.24 to 0.3 in. Thick

Polar crane -	2598 ft^2
SITs	138 ft^2
Stairway framing -	193 ft^2
Reactor head shroud -	1439 ft^2
Ventilation duct plenum supports -	3553 ft^2
Equipment hatch lifting rig -	563 ft^2
Cable supports -	323 ft^2
Shield barrier and SG Platforms	147 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	4207 ft^2
	13161 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	1.0×10^{-3}	8×10^{-3}
Primer	7.5×10^{-4}	3×10^{-3}
Carbon Steel	1.24×10^{-2}	2.48×10^{-1}

TABLE 14.20-5

CONTAINMENT HEAT SINKS

Heat Sink #8+ - Miscellaneous Steel 0.3 to 0.4 in. Thick

SITs	448 ft^2
Polar crane -	2643 ft^2
Shield barrier and lifting beam -	119 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	4231 ft^2
	7441 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	$1.0x10^{-3}$	$8x10^{-3}$
Primer	$7.5x10^{-4}$	$3x10^{-3}$
Carbon Steel	$1.78x10^{-2}$	$3.56x10^{-1}$

Heat Sink #9 - Miscellaneous Steel 0.4 to 0.5 in. Thick

Polar crane -	1692 ft^2
Main steam pipe supports -	371 ft^2
Shield barrier -	73 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	1578 ft^2
	3714 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	$1.0x10^{-3}$	$8x10^{-3}$
Primer	$7.5x10^{-4}$	$3x10^{-3}$
Carbon Steel	$1.21x10^{-2}$	$4.25x10^{-1}$

Heat Sink #10+ - Miscellaneous Steel 0.5 to 0.625 in. Thick

Polar crane -	5364 ft^2
Reactor head shroud -	1120 ft^2
Miscellaneous (I-beams, channel beams, angle iron, and plates)	2546 ft^2
	9030 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	$1.0x10^{-3}$	$8x10^{-3}$
Primer	$7.5x10^{-4}$	$3x10^{-3}$
Carbon Steel	$2.57x10^{-2}$	$5.15x10^{-1}$

Heat Sink #11+ - Miscellaneous Steel 0.625 to 0.75 in. Thick

SITs -	70 ft^2
Polar crane -	2160 ft^2
	2230 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	$1.0x10^{-3}$	$8x10^{-3}$
Primer	$7.5x10^{-4}$	$3x10^{-3}$
Carbon Steel	$3.38x10^{-2}$	$6.77x10^{-1}$

Heat Sink #12 - Miscellaneous Steel 0.75 to 1.0 in. Thick

SITs -	3682 ft^2
Polar crane -	1449 ft^2
	5131 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	$1.0x10^{-3}$	$8x10^{-3}$
Primer	$7.5x10^{-4}$	$3x10^{-3}$
Carbon Steel	$4.21x10^{-2}$	$8.42x10^{-1}$

TABLE 14.20-5

CONTAINMENT HEAT SINKS

Heat Sink #13+ - Miscellaneous Steel 1.0 to 1.5 in. Thick

Polar crane -	2649 ft^2
RCP supports -	962 ft^2
Reactor head shroud -	732 ft^2
	4358 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	1.0×10^{-3}	8×10^{-3}
Primer	7.5×10^{-4}	3×10^{-3}
Carbon Steel	5×10^{-2}	1

Heat Sink #14+ - Miscellaneous Steel 1.5 in. Thick or Greater

Polar crane -	1236 ft^2
Equipment hatch -	343 ft^2
Personnel hatch -	269 ft^2
Emergency airlock -	67 ft^2
	1915 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint*	1.0×10^{-3}	8×10^{-3}
Primer	7.5×10^{-4}	3×10^{-3}
Carbon Steel	4.82×10^{-2}	2.412

Heat Sink #15 - Containment Wall in Penetration Areas

Reinforcing plates for containment liner at penetration area -	2470 ft^2

Material		Node Spacing (in.)	Thickness (in.)
Paint*		1.0×10^{-3}	8×10^{-3}
Primer		7.5×10^{-4}	3×10^{-3}
Carbon Steel		7.5×10^{-2}	7.5×10^{-1}
Concrete region	1	5×10^{-2}	1.5
	2	2×10^{-1}	4
	3	1.3	6.5
	4	1.32	33

Heat Sink #16 - Stainless Steel Liner Plate

Refueling pool -	7750 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Stainless Steel	1.875×10^{-2}	1.875×10^{-1}

Heat Sink #17 - Containment Shield Barrier

Lead Shield -	203.3 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Lead	4.0×10^{-2}	1.0

Heat Sink #18 - Reactor Cavity Below El. 29’-4”

Reactor Cavity Wall -	1280 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	4.0×10^{-3}
Primer*	7.5×10^{-4}	6.0×10^{-3}
Carbon Steel	6.25×10^{-2}	2.5×10^{-1}
Air	1.429×10^{-1}	1.0
Carbon Steel	6.25×10^{-2}	2.5×10^{-1}
Concrete	1.6×10^{-1}	12.0

TABLE 14.20-5

CONTAINMENT HEAT SINKS

<u>Heat Sink #19</u> - Reactor Cavity Between El. 29'-4" and 44"
Reactor Cavity Wall - 968.8 ft^2

Material	Node Spacing (in.)	Thickness (in.)
Paint	$1.0x10^{-3}$	$4.0x10^{-3}$
Primer*	$7.5x10^{-4}$	$6.0x10^{-3}$
Carbon Steel	$6.25x10^{-2}$	$2.5x10^{-1}$
Concrete	$1.6x10^{-1}$	12.0

Material Properties

	Thermal Conductivity Btu/hr-ft-°F	Volumetric Heat Capacity Btu/ft^3-°F
Concrete	2.2	32.835
Carbon Steel	29.6	53.6
Stainless Steel	8.6	60.1
Paint	0.3	47.1
Primer	1.01	21.70
Zinc	62.2	42.0
Lead	19.6	22.3
Air	0.017	0.0156

<u>Heat Sink #20</u> - Containment Sump Strainer Between El. 10'-0" and 14'-0"

Thickness	Surface Area	El.
0.157 in.	307.9 ft^2	3'
0.157 in.	0.874 (H-3) ft^2	H in inches from 3" to 35"
0.236 in.	116.5 ft^2	3'
0.236 in.	14.961 (H-3) ft^2	H in inches from 3" to 30"
0.236 in.	29.877 (H-30) ft^2	H in inches from 30" to 34"
0.118 in.	3.00 H ft^2	H in inches from 3" to 35"
0.049 in.	381.5 ft^2	6.59 in.
0.049 in.	381.5 ft^2	10.18 in.
0.049 in.	381.5 ft^2	13.77 in.
0.049 in.	381.5 ft^2	17.36 in.
0.049 in.	381.5 ft^2	20.95 in.
0.049 in.	381.5 ft^2	24.54 in.
0.049 in.	381.5 ft^2	28.13 in.
0.049 in.	381.5 ft^2	31.72 in.
0.049 in.	381.5 ft^2	35.31 in.

<u>Material</u>
Uncoated stainless steel

* The analysis conservatively assumes that the entire surface of Heat Sink 1 has a paint thickness of 18 mils and Heat Sinks 5 – 15 have a paint thickness of 8 mils. The analysis conservatively assumes that Heat Sinks 18 and 19 have a primer thickness of 6 mils. The above coating assumptions conservatively envelop the existing coating in the Containment.

\+ The current paint and primer utilized within the Containments on carbon steel is Carboline 890 at a maximum thickness of 6 mils primer and 6 mils paint. The paint and primer thicknesses and material properties listed in the above table are more conservative.

TABLE 14.20-6

ENGINEERED SAFETY FEATURES PERFORMANCE FOR LOSS-OF-COOLANT ACCIDENT CONTAINMENT ANALYSES

SAFETY FEATURE			CONTAINMENT DESIGN BASIS MINIMUM SI	CONTAINMENT DESIGN BASIS MAXIMUM SI
1.	PASSIVE SI			
	Water Source: SITs			
	No. tanks and lines		4	4
	Quantity available (water at 120°F)		4,000 ft^3	4,000 ft^3
	Operating point (reactor pressure)		250 psig	250 psig
2.	ACTIVE SI			
	Water Sources: RWT (360,000 gal)[h] and recirculation from sump			
	No. of pumps (HPSI/LPSI)			
	Pre-RAS		1/1	2/2
	Post-RAS		1/0	2/0
	Flow rate			
	Pre-RAS		~5400 gpm[a]	~8000 gpm[c]
	Post-RAS		575 gpm[b]	1000 gpm[b]
3.	CONTAINMENT SPRAY			
	Water Sources: RWT (360,000 gal)[h] and recirculation from sump			
	No. of lines and headers		1	2(d)
	No. of pumps		1	2
	Flow rate		1250 gpm	2500 gpm
4.	CONTAINMENT ATMOSPHERE COOLING FAN			
	No. of units		2	2
	Flow rate:	(air side)	55,000 cfm/CAC	55,000 cfm/CAC
		(SRW side)	1400gpm/1900gpm[i]	1400gpm/1900gpm[i]
	Heat removal (Btu/hr)		Figure 14.20-1	Figure 14.20-1
5.	SHUTDOWN COOLING HEAT EXCHANGER			
	Cooling Water Supply: Component Cooling System			
	Type: Shell and tube; No. of Units		1[e]	2
	Heat transfer area		4990 ft^2	9980 ft^2
	Heat transfer coefficient (overall)		206 Btu/hr-ft^2-°F[f]	206 Btu/hr-ft^2-°F[f]
	Flow rate:	(sump water side)	1250 gpm	2500 gpm[d]
		(component cooling water side)	8.96×10^5 lbm/hr	1.79×10^6 lbm/hr
	Operating point (time)		4175 sec[g]	1920 sec

TABLE 14.20-6

ENGINEERED SAFETY FEATURES PERFORMANCE FOR LOSS-OF-COOLANT ACCIDENT CONTAINMENT ANALYSES

(a) For the hot leg break, the entire SI flow rate is injected into the core. For the cold leg break with minimum SI, the entire pre-RAS SI flow rate is assumed to flow to the reactor, with no spillage directly to the sump. This is conservative in that it maximizes the rate of energy transfer from the reactor to the Containment.

(b) For the cold leg break, 25% of this flow is assumed to spill from the broken loop directly to the containment sump. For the hot leg break, this entire flow rate is assumed to be injected into the core.

(c) Of the 8000 gpm, 2,000 gpm are spilled to the sump after being uniformly mixed in the vessel.

(d) Operation of two trains of spray is assumed for the initial 1.12 days ($9.65x10^4$ sec). By then, containment vapor pressure has decreased to 4.75 psig or less. After that time, it is assumed that one train of spray is manually turned off, and the flow rate decreases from 2500 gpm to 1250 gpm.

(e) Although only one SDCHX is credited, the system is operated such that both SDCHXs will receive component cooling water while only one SDCHX will have containment spray circulating through the tubes.

(f) This value is used for the initial 10,000 sec of the transient. Thereafter a value of 195 Btu/hr ft^{3}°F is used, taking into account the decreasing sump water temperature.

(g) For the Hot Leg Break with minimum SI, the operating point (time to RAS) is calculated to be approximately 4180 sec.

(h) This value represents the minimum volume available prior to RAS.

(i) 1400 gpm assumed pre-RAS, 1900 gpm assumed post-RAS.

TABLE 14.20-7

SEQUENCE OF EVENTS FOR DOUBLE-ENDED DISCHARGE LEG MINIMUM SI LOSS-OF-COOLANT ACCIDENT

TIME, sec	EVENT	COMMENT
0.0	Break occurs	Break area = 9.8 ft^2
	LOOP	RCPs begin coasting down
0.6	CSAS, Containment High Pressure analytical setpoint is reached, Reactor trip setpoint reached, Containment High Pressure.	Containment pressure = 4.75 psig
1.50	CSAS actuated, Main Feedwater, Condensate Booster, HDPs begin coasting down	0.9 sec signal delay time included
1.85	Turbine Stop Valves close	Includes a 0.25 sec valve stroke time plus a 0.1 sec signal delay time after reactor trip signal actuated
11.50	Main Feedwater rampdown completed	A 10-sec rampdown time was calculated
14.40	End of blowdown	
16.50	SI Pump Flow started	A 15 sec delay is assumed after SIAS on High Containment Pressure is actuated
36.50	CACs Full On	A 35.0-sec delay assumed after SIAS signal actuated
71.50	Containment Spray full on	A 70.0-sec delay assumed after CSAS signal actuated
200	Peak Containment Temperature	Vapor Temperature (F): 274.5
200	Peak Containment Pressure	Total Pressure (psig): 49.7
279.4	End of Post-Reflood	Long-term release model begins
1800.0	Time of RAS	SI pump flow switches from RWT to sump

TABLE 14.20-8

SUMMARY OF SIGNIFICANT ASSUMPTIONS FOR MAIN STEAM LINE BREAK MASS and ENERGY RELEASE CALCULATIONS

Sources of Energy

- Affected SG's metal, including the vessel tubing
- Affected SG's water (and steam) inventory
- Feedwater line inventory from MFIV to affected SG
- Unaffected SG inventory prior to closure of MSIVs
- Reactor power and decay heat
- Primary system metal

  Primary coolant to affected SG during blowdown; No credit taken for cold SI flow or AFW flow to intact SG

Sources of Mass Release

- Affected SG steam and liquid
- Feedwater line
- Feedwater transferred to affected SG prior to closure of MFIVs
- Steam from unaffected SG prior to closure of MSIV
- Steam line inventory from ruptured SG outlet nozzle to MSIV
- The steam line volume from the ruptured side SG to the turbine and intact SG

Mass and Energy Release Calculations

- Break flow calculations were based on the MOODY critical flow correlation
- In accordance with the original NRC approved design basis assumptions for the Calvert Cliffs plants, a double-ended guillotine break was assumed for all cases
- No moisture carryover was credited in the MSLB analysis
- SG heat transfer calculations were based on nucleate boiling heat transfer
- Volumetric expansion due to pressure and temperature considerations was included in both the primary and secondary systems

Main Feedwater Flow

- Diversion of flow from the intact SG due to high differential pressure between intact and ruptured SGs was considered in the analysis
- Feedwater flashing in affected feedline up to the MFIV caused by reduction in SG pressure was considered in the analysis
- The MFW regulating valve was not credited in this analysis

TABLE 14.20-9

INITIAL CONDITIONS AND KEY ASSUMPTIONS FOR ANALYSIS OF MASS AND ENERGY RELEASE FOR MAIN STEAM LINE BREAK

	ITEM	VALUE/ASSUMPTION
1.	Methodology	
	Mass and Energy Code	RELAP5/MOD2-B&W
	Containment Response Code	CONTEMPT, Version 24
	Moisture Carryover from Ruptured SG	0%
	Break Type	Guillotine at SG nozzle, 1.9 ft^2 effective area due to SG flow restrictor
	Water in Feed Pipe	Considered
	Steam in Header Pipe	Considered
2.	Initial Nuclear Steam Supply System Parameters	
	Mode	1
	Power Level	102% 75% (limiting case), 50%, 0% also considered
	Initial Primary Pressure	2250 psia
	Initial RCS Inlet Temperature	550.0°F[(a)]
	Initial Secondary Pressure	888 psia[(a)]
	Primary and Secondary Volumetric Expansion Due to Pressure and Temperature	Considered
	Steam Generator Level	35.95' above tube sheet (Normal Water Level)
	Number of tubes plugged	None
	Number of tubes removed	None
3.	Reactor Shutdown	
	Reactor Trip	At 4.75 psig containment pressure
	Rod Drop Time	3.1 sec including holding coil delay
4.	Reactor Coolant Pumps	On throughout Event
	Total RCS flow rate	422,250 gpm
5.	Main Steam Isolation Valves	
	Delay Time from SGIS	0.9 sec
	Closure Time	7.0 sec
	Lift vs Time	Linear
	Single Failure	Considered
	MSIV Logic	High Containment pressure 4.75 psig + 0.9 sec delay + 7.0 sec closure

TABLE 14.20-9

INITIAL CONDITIONS AND KEY ASSUMPTIONS FOR ANALYSIS OF MASS AND ENERGY RELEASE FOR MAIN STEAM LINE BREAK

	ITEM	VALUE/ASSUMPTION
6.	Main Feedwater	
	Initial Flow Rate	102% of full flow[a] (Consistent with initial power level)
	MFW Enthalpy	425 Btu/lbm[a]
	SGFP, CBP HDP Trip Logic	High Containment pressure 4.75 psig + 0.9 sec delay + 1.0 sec trip delay
	MFW Flow Diverted from Intact High Pressure SG to Ruptured Low Pressure SG	Considered
	Contribution of MFW Flashing Between Shut MFIV and Ruptured SG	Considered
	Limiting Single Failure	Failure of SGFP to Trip on SGIS (CSAS) on Containment High Pressure
7.	Main Feedwater Isolation Valves	
	Response Time from SGIS	65 sec (for no LOOP case)
	Single Failure	Considered
	MFIV Logic	High Containment Pressure Temperature Setpoint of 4.75 psig + 0.9 sec delay + 65 sec step closure
8.	MFIV Leakage	A total leakage of 200 gpm (based on full power conditions) is assumed to the ruptured SG
9.	AFW Contribution	After 20 sec initial timer delay, 1550 GPM until isolation at 20 sec after high differential pressure block signal (actuated 0.9 sec after 250 psid is reached). AFW leakage to the ruptured SG, after isolation, is assumed at 80 gpm total. AFW enthalpy corresponds to the maximum AFW temperature of 100°F.
10.	Main Steam Piping	
	Total Equiv. K factor for Main Steam Pipe Line Losses	Based on steam line geometry
	Contribution from Steam in Pipe to Ruptured SG's MSIV	Considered

[a] Power Dependent Values.

TABLE 14.20-10

INITIAL CONDITIONS AND KEY ASSUMPTIONS FOR ANALYSIS OF CONTAINMENT RESPONSE TO MAIN STEAM LINE BREAK

Free Volume	1.989×10^6 ft^3
Containment Spray Actuation Setpoint (SIAS/CSAS)	4.75 psig in Containment
Spray Delay Time (after CSAS)	62 sec, with off-site power available plus 0.9 sec signal delay
Spray (RWT) Temperature	100°F
Containment Spray Logic	2 spray trains (1250 gpm per train)
CAC Capacity	See Figure 14.20-1, Four CACs operational, with service water flow rate = 1400 gpm
CAC Actuation Setpoint (SIAS)	4.75 psig
CAC Delay	10 sec, with off-site power available plus 0.9 sec signal delay
Initial Containment Temperature	125°F
Initial Containment Pressure	1.8 psig
Initial Relative Humidity	20%
Heat Transfer Coefficient Used for Passive Heat Sinks	Uchida Correlation
Credit for Condensate Revaporization	No credit taken
Credit for Heat Transfer to Water in Containment Sump	No credit taken

TABLE 14.20-11

ENGINEERED SAFETY FEATURE PERFORMANCE PARAMETERS USED FOR CONTAINMENT ANALYSIS FOR MAIN STEAM LINE BREAK

SAFETY FEATURES	ACCIDENT OPERATION
Containment Spray [a]	
Water sources (RWT)	360,000 gal[d]
No. of lines and headers	2
No. of pumps	2
Flow rate	2500 gpm
Containment Atmosphere Cooling Fans[b]	
No. of units	4
Total flow rate (air side)	220,000 cfm
Heat removal (Btu/hr)[c]	Figure 14.20-1

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[a] The containment spray delay time after CSAS is 62.9 sec, with offsite power available.

[b] The containment fan cooler actuation analysis setpoint is 4.75 psig in Containment. The time delay is 10.9 sec.

[c] The heat removal rate is based on a service water flow rate of 1400 gpm per unit.

[d] This value represents the minimum volume available prior to RAS.

TABLE 14.20-12

SEQUENCE OF EVENTS FOR MAIN STEAM LINE BREAK INSIDE CONTAINMENT

TIME, sec	EVENT	COMMENT
0.0	Break occurs	Break area = 1.9 ft^2 (effective)
1.61	Reactor trip analytical setpoint reached, Containment High Pressure	Containment pressure = 4.75 psig
	SGIS/CSAS analytical setpoint reached, Containment High Pressure	
	Turbine Stop valves closed	TSVs conservatively closed at reactor trip setpoint
2.51	CEAs begin entering core	Reactor trip delay time = 0.9 sec
	SGIS/CSAS signal actuated	SGIS/CSAS signal delay time = 0.9 sec
3.51	SGFP, CBP, and HDP Coastdown Begins	Based on 1 sec response time for SGFP steam inlet valves closure
9.51	MSIV Closure	A 7.9 sec delay time after High Containment Pressure Reactor Trip analytical setpoint is reached
11.57	AFAS Block high SG ΔP analytical setpoint reached	SG ΔP = 250 psid
12.51	CACs full on	A 10 sec delay time after CSAS actuated
16.67	AFAS Low SG Water Level analytical setpoint reached	Ruptured SG level = 66.5” below normal water level
32.47	AFW to ruptured side SG isolated	A 20.9 sec delay time assumed after analytical setpoint is reached
64.50	Containment Sprays full on	A 62.0 sec delay assumed after CSAS signal actuated.
65.0	Containment peak temperature	Vapor temperature = 354.2°F
67.50	MFIVs closed	A MFIV stroke time of 65 sec assumed following actuation of SGIS on CSAS
250.0	Containment peak pressure	Total pressure = 49.1 psig

TABLE 14.20-13

INPUT PARAMETERS COMPARTMENT PRESSURIZATION ANALYSIS

Initial Conditions	
Containment Temperature, °F	120
Containment Pressure, psia	14.7
Containment Relative Humidity, %	50
Structure Geometries	
Main Containment:	
Net Free Volume, ft^3	2.00×10^6
East SG Compartment:	
Net Free Volume, ft^3	53,980
Long Sharp-Edged Nozzle Area, ft^2	356
Sharp-Edged Orifice Area, ft^2	770
Well-Rounded Orifice Area, ft^2	1,072
West SG Compartment:	
Net Free Volume, ft^3	51,500
Long Sharp-Edged Nozzle Area, ft^2	182
Sharp-Edged Orifice Area, ft^2	414
Well-Rounded Orifice Area, ft^2	1,072
Pressurizer Compartment:	
Net Free Volume, ft^3	3,394
Orifice-Type Relief Area (ventilation openings), ft^2	20.5
Orifice-Type Relief Area (available after baffle blows out), ft^2	85.2

14.21 DELETED

Hydrogen Accumulation in Containment, was deleted per License Amendment Nos. 262/239.

14.22 WASTE GAS INCIDENT

14.22.1 GENERAL

The most limiting waste gas incident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one waste gas decay tank.

As the components of the waste gas system are subjected to pressures no greater than 150 psig, a failure is certainly not likely. However, a rupture of a waste gas decay tank is analyzed to define the limit of the hazard that could result from any malfunction in the radioactive waste gas system.

14.22.2 METHOD OF ANALYSIS

It is assumed that the maximum activity in any one waste gas decay tank would occur shortly after a heatup and subsequent reactor startup from a cold shutdown condition of one unit near the end of a 24-month operating cycle. It is further assumed that this unit has been operating for an extended period with 1% defective VAP fuel and that all of the coolant is let down and the noble gases from one RCS volume are stored in one tank. On this basis the noble gas activity in the tank, neglecting decay after letdown, is as follows:

Isotope		Activity, curies (Reference 1)
Kr	85m	672
Kr	85	7,637
Kr	87	393
Kr	88	1,242
Xe	131m	1,749
Xe	133	106,943
Xe	135	4,325
Xe	138	188

The activity is assumed to be released into the Auxiliary Building with the ventilation system discharging into the turbulent wake of the plant buildings. Therefore, the 0-2 hour meteorological model described in Chapter 2 is applicable. Doses were computed using the dose conversion factors and methodology described in ICRP-30.

14.22.3 RESULTS

For this incident the whole body immersion dose at the nearest exclusion zone boundary is 0.182 REM, using Chi/Q = 1.3×10^{-4} sec/m^3.

14.22.4 CONCLUSIONS

In the unlikely event of rupture of a gas decay tank resulting in a release of the maximum stored gaseous activity from one RCS volume, the dose at the nearest exclusion zone boundary is a factor of approximately 100 less than the 10 CFR Part 100 dose guideline value. Therefore, a waste gas incident does not represent undue hazard to the public health and safety.

This event is not affected by the transition to AREVA Advanced CE-14 HTP™ fuel. The source term bounds the transition cycles and the full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

14.22.5 REFERENCES

1. J.R. Massari, "RC Waste Processing System Incident and Waste Gas Incident Dose Analysis," CA05994, October 18, 2002

14.23 WASTE PROCESSING SYSTEM INCIDENT

14.23.1 GENERAL

In the event of a seismically-induced failure of the reactor coolant Waste Processing System, it is hypothesized that the contents of some components in the system will be released. The release of the contents of the Waste Processing System is analyzed to the site boundary dose limits specified in 10 CFR 20.1301(a)(1).

14.23.2 METHOD OF ANALYSIS

The analysis was performed using the following methodology:

a. The fuel fission product source term is calculated for the most limiting assemblies using the SAS2H depletion sequence of the SCALE code system. The reactor coolant source term is determined based on 1% failed fuel damage, which is consistent with the source term used in UFSAR Sections 14.13 through 14.16.

b. The Alternative Source Term (AST) methodology of 10 CFR 50.67 and Regulatory Guide 1.183 is used to calculate doses due to a waste processing incident. Only noble gasses, halogens (iodine), and tritium are considered in the dose analysis. Site boundary Total Effective Dose Equivalent (TEDE) doses due to the waste processing incident are calculated using the RADTRAD computer code. The waste processing system incident model is constructed assuming that failure of all waste processing system components occurs at time, t = 0. The decay and daughter product options of RADTRAD are utilized. The rate of escape of activity to the environment is assumed to occur within a matter of minutes and is transported to the site boundary via appropriate atmospheric dispersion coefficients.

c. Dose consequences are calculated (Reference 1) for simultaneous failure of some Waste Processing System components downstream of the containment isolation valves (1/2-CV-4260).

14.23.3 ASSUMPTIONS

a. Feed to the Waste Processing System is assumed to be reactor coolant with 100/Ebar μCi/gm noble gas equilibrium activities (Reference 2) which are more limiting than the 1% failed fuel noble gas activities and are based on the Technical Specification limit. The reactor coolant iodine equilibrium activities are based on 1% failed fuel. A reactor coolant equilibrium tritium concentration of 3.5 μCi/cc is used.

b. Credit for processing prior to each Waste Processing System component is assumed. Decontamination factors are consistent with the original evaluation for the evaporators, with Reference 3, and are listed in Table 14.23-1. Contrary to the original evaluation for the evaporators, a decontamination factor of unity is assigned to the degasifier to bound actual plant system performance levels.

c. For the evaporator failure, it is assumed that the evaporator feed tanks and associated piping are filled with processed reactor coolant source term. Contrary to the original evaluation for the evaporators, this analysis does not postulate a concentrated liquid waste due to evaporator operation. This assumption applies to an evaporator system that is no longer operational, but is still physically connected to the Waste Processing System and therefore the feed tanks are postulated to contain activity due to leakage. Since the evaporator feed tanks are at 160°F, no flashing is assumed, and 10% of each evaporator feed tank volume is assigned to evaporate. The following release fractions are assumed:

   1. 10^{-3} of all of the iodine in solution is assumed to come out of solution;

2. All noble gases are released;
3. All of the iodine and tritium in the evaporated portion of the evaporator feed tanks is released.

The release is conservatively treated as an instantaneous puff release.

d. For other components of the Waste Processing System, no flashing of the continued fluid is assumed since piping design temperatures are < 180°F, and release fractions for tritium, iodine, and noble gases are assumed to be 10%, 10.1%, and 100%, respectively. Upon failure of an ion exchanger, 10% of the equilibrium halogen inventory on the resin is assumed to instantaneously and non-mechanistically transfer to the water. The rate of escape of all radionuclide activity to the environment is assumed to occur within a matter of minutes.
e. The ventilation stack-to-site boundary 0-2 hour atmospheric dispersion coefficient is 1.44×10^{-4} sec/m^3 (UFSAR Section 2.3.6).
f. A breathing rate of 3.5×10^{-4} m^3/sec is assumed.
g. Doses were computed using the dose conversion factors from Federal Guidance Report (FGR) Nos. 11 and 12.
h. The following components are postulated to fail due to a seismic event in the Waste Processing System incident:
    1. Spent Resin Metering Tank
    2. Miscellaneous Waste Monitor Tank
    3. Reactor Coolant Evaporator Feed Tanks (2)
    4. Reactor Coolant Evaporator System Piping

14.23.4 RESULTS

The calculated control room and offsite doses are as follows:

Dose, REM (TEDE)		
Control Room	EAB	LPZ
0.153	0.031	0.007

14.23.5 CONCLUSIONS

In the event of a failure of the reactor coolant Waste Processing System, the total radioactivity released to the atmosphere and carried to the site boundary will be less than the maximum allowable limits of 0.1 REM (TEDE) specified in 10 CFR Part 20.1301(a)(1).

The control room dose consequence due to this event is less than the allowable limit specified in 10 CFR 50.67.

14.23.6 REFERENCES

1. J.R. Massari, “Waste Processing Incident for Alternate Source Term,” CA06608, Revision 0000
2. CA06422, “Primary and Secondary Isotopic Calculations,” Revision 0000
3. T. Chandrasekaran, J.Y. Lee, C.A. Willis, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, NUREG-0017, Revision 1, April 1985
4. ECP-14-000889-004-CN-001 CA06608-0000, "Waste Processing System Incident for Alternate Source Term," Revision 0000

TABLE 14.23-1

SUMMARY OF COMPONENT DECONTAMINATION FACTORS AND AMOUNT OF PRIOR PROCESSING CREDITED FOR EACH WASTE PROCESSING SYSTEM COMPONENT

Pre-Component Processing DFs	CVCS IX	Degasifier	RCW IXs 11 & 12	Process Skid or RCW IXs 13 & 14	Total Pre-Component DF Used (NG/H/T)
Noble Gas (NG) DF	1	1	1	1	
Halogen (H) DF	10	1	10	10	
Tritium (T) DF	1	1	1	1	
Pre-Processing Credited (indicated by √) for Each Waste Processing System Component					
RCW Evaporators	√	√	√		1/100/1
RCW Processing Skid	√	√	√		1/100/1
RCW Receiver Tank & MW Tanks	√	√	√		1/100/1
RCW Monitor Tanks	√	√	√	√	1/1E3/1
RCW IXs (11 & 12)	√	√			1/10/1
RCW IXs (13 & 14)	√	√	√		1/100/1
RCW Degasifiers	√	√			1/10/1
RCW Filters	√				1/10/1
System Piping	√				1/10/1
MW Monitor Tank	N/A	N/A	N/A	N/A	4/4/4

14.24 MAXIMUM HYPOTHETICAL ACCIDENT

14.24.1 GENERAL

The MHA involves a gross release of fission products from the fuel to the Containment. During an accidental release, air containing radionuclides may enter the Control Room through inleakage into the Control Room ventilation system. The Control Room TEDE doses from inleakage and shine must meet 10 CFR Part 50, Appendix A, General Design Criteria 19 limits. Similarly, during an accidental release, air containing radionuclides may travel offsite. The offsite TEDE doses must meet 10 CFR 50.67 limits. The Control Room and offsite TEDE doses were calculated with the dose conversion factors extracted from References 5 and 6.

The design-basis MHA utilizes the AST methodology of 10 CFR 50.67 and Reference 1 to calculate offsite and Control Room doses for an MHA. Per Reference 1, the TEDE analysis should include all sources of radiation that will cause exposure to Control Room personnel; including the following pathways:

- Containment pathway
- Hydrogen Purge Line pathway
- Ventilation Stack pathway
- RWT pathway
- Containment Shine
- Plume Shine
- Control Room Filter Shine

The results of the design-bases MHA AST analysis (Reference 2) were submitted to the NRC in Reference 3. The NRC subsequently approved the license amendment request in Reference 4.

14.24.2 METHOD OF ANALYSIS

14.24.2.1 Control Room

The main Control Room inleakage points include the West Road inlets, the Turbine Building, and Access Control Units 11 and 13 on the Auxiliary Building roof. Installation of automatic isolation dampers and radiation monitors at the Access Control Units 11 and 13 on the Auxiliary Building roof were credited.

Assumptions used are:

- The Control Room volume is 289194 ft^3.
- A Control Room inleakage rate of 3500 cfm was based on measured inleakage measurements.
- Control Room recirculation filtration is credited assuming 10,000 $\pm$ 10% cfm flow at 90% filter efficiency for elemental and organic iodine and 99% for particulates for a 20 minute delay time.
- 0-8, 8-24, and 24-720 hour breathing rates of 3.5E-04, 1.8E-04, and 2.3E-04 m^3/sec are assumed.
- 0-24, 24-96, and 96-720 hour Control Room occupancy factors of 1.0, 0.6, and 0.4 are assumed.

14.24.2.2 Source Terms

The inventory of fission products in the reactor core and available for release to the containment atmosphere is based on the maximum full power operation of the

core with current licensed values for fuel enrichment (5.0 w/o), fuel burnup (62,000 MWd/MTU), and core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty (2754 MWt). The period of irradiation was of sufficient duration to allow the activity of dose-significant radionuclides to reach maximum values. The isotopic activities released from the failed fuel were generated utilizing the isotope generation and depletion computer code SAS2H/ORIGEN-S.

The core inventory release onset, duration, and fractions by radionuclide groups for the gap release phase (0.5-30 minutes) and early in-vessel damage phase (30-108 minutes) for a DBA LOCA were extracted from Reference 1. The activity released from each release phase is modeled as increasing linearly over the duration of the phase. The release fractions for the gap release and early in-vessel damage phases are as follows:

	Gap Release	Early Invessel Damage
• Noble Gases (Xe, Kr)	0.05	0.95
• Halogens (I, Br)	0.05	0.35
• Alkali Metals (Cs Rb)	0.05	0.25
• Tellurium Metals (Te, Sb, Se)	0.00	0.05
• Ba, Sr	0.00	0.02
• Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0.00	0.0025
• Cerium Group (Ce, Pu, Np)	0.00	0.0005
• Lanthanides, (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0.00	0.0002

Per Reference 1, of the radioiodine released from the RCS to the containment atmosphere in a LOCA, 95% of the iodine released should be assumed to be particulate iodine, 4.85% elemental iodine, and 0.15% organic iodine. This includes releases from the gap and fuel pellets. With the exception of elemental and organic iodine and noble gases, all other fission products should be assumed to be in particulate form.

14.24.2.3 Containment Pathway

Per Reference 8, the total release from Containment may be apportioned between the exposed and enclosed building surfaces. 72% of the released airborne activity post-MHA is assumed to leak out of the Containment through the containment walls.

Assumptions used are:

- The Containment volume is 1.989E+06 ft^3.
- Two 55,000 cfm cooling units are credited after a 60 second activation delay.
- Two containment iodine removal units are credited after a 63 second activation delay for the first unit and a 20 minute activation delay for the second unit. Each unit consists of activated charcoal filters preceded by HEPA filters. Each filter unit has a 20,000 ± 10% cfm flowrate with a filter efficiency of 90% for elemental and particulate species and 30% for organic species per Reference 9.
- Reduction in aerosol airborne radioactivity in the Containment by natural deposition within the Containment was credited per Reference 1. The 10th percentile Powers aerosol decontamination model (Reference 10) was

utilized. Aerosol particles grow by coagulating with other aerosol particles or because steam condenses on them thus, gravitational settling of aerosols is usually the dominant aerosol removal process.

- Reduction in airborne radioactivity by containment spray systems that have been designed and maintained in accordance with Reference 11 may be credited.
  - The spray removal constant for aerosols is 3.414/hr with no maximum decontamination factor but with a 90 second activation delay.
  - The spray removal constant for elemental iodine is 14.816/hr with a maximum decontamination factor of 14.04 and with a 90 second activation delay.
  - Organic iodides are not removed by spray.
- The maximum allowable containment leakage rate L_a contained in the Containment Leakage Rate Testing Program of Technical Specification 5.5.16 was assumed, 0.16 percent per day at P_a. Per Reference 1, the Containment is assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.
- The 0-2 hour Containment-to-EAB atmospheric dispersion coefficient (χ/Q) was calculated to be 1.30E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour Containment-to-LPZ atmospheric dispersion coefficients were calculated to be 3.330E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst-case 0-2, 2-8, 8-24, 24-96, and 96-720 hour Containment-to-Control Room atmospheric dispersion coefficients were calculated to be 1.11E-03, 7.29E-04, 3.19E-04, 2.36E-04, and 1.98E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

14.24.2.4 Ventilation Stack Pathway

Per Reference 8, the total release from Containment may be apportioned between the exposed and enclosed building surfaces. Containment leakage is more likely at penetrations rather than through liner plates or weld joints. Penetration rooms are built adjacent to the outside surface of each Containment and enclose the areas around the majority of the penetrations. Thus, a fraction of the containment leakage will leak into the Auxiliary Building penetration rooms, be processed by the penetration room emergency ventilation system, and be expelled to the atmosphere through the ventilation stacks. 28% of the released airborne activity post-MHA is assumed to leak out of the Containment through the containment penetrations into the Auxiliary Building penetration rooms.

Assumptions used are:

- The containment volume, circulation, and cleanup mechanisms are the same as described previously for the containment pathway.
- The 0-2 hour ventilation stack-to-EAB χ/Q was calculated to be 1.44E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour ventilation stack-to-LPZ χ/Qs were calculated to be 3.39E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst case 0-2, 2-8, 8-24, 24-96, and 96-720 hour ventilation stack-to-Control Room χ/Qs were calculated to be 1.68E-03, 1.34E-03, 5.14E-04, 3.84E-04, and 3.12E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

- The Penetration Room Emergency Ventilation System is designed to collect and process containment penetration leakage, so as to reduce to a minimum the environmental radioactivity levels from post-accident containment leaks. To minimize the release of radioactive material to the environment, penetration room ventilation is continuously routed through a prefilter, a HEPA filter, and an activated charcoal filter, positioned in series. Following a LOCA, a containment isolation signal will start both the two full-size blowers. The entire system is designed to operate under negative pressure up to the fan discharge. Per Technical Specification 5.5.11, each Penetration Room Emergency Ventilation System unit has a design flowrate of 2,000 ± 10% cfm with an efficiency of 90% for elemental particulate species and 30% for organic species.

14.24.2.5 Hydrogen Purge Line Pathway

Per Reference 1, if the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The containment 4" hydrogen purge line is used for an unlimited amount of pressure and containment radioactivity control purposes; however, the vent isolation valves may also be opened for surveillance testing. The 4" vent line isolates on a SIAS and containment radiation signal.

Assumptions used are:

- The purge release evaluation assumes that 100% of the radionuclide inventory in the RCS liquid is released to the containment atmosphere at the initiation of the LOCA. This inventory is based on the Technical Specification RCS equilibrium activity of 0.5 μCi/gm DEQ I-131. Iodine spikes need not be considered. The purge system is isolated before the onset of the gap release phase, thus release fractions associated with gap release and early in-vessel phases need not be considered.
- 100% of the Technical Specification primary iodine and noble gas activities are assumed to be released instantaneously and homogeneously throughout the containment atmosphere at the initiation of the accident.
- The radioiodine that is postulated to be available for releases from the RCS to the environment should be assumed to be 97% elemental and 3% organic.
- The hydrogen purge line flow rate value is conservatively assumed to be 1645 cfm.
- The χ/Q and Penetration Room Emergency Ventilation System values are identical to those for the ventilation stack pathway.

14.24.2.6 Refueling Water Tank Pathway

There is a potential for an unmonitored release pathway resulting from the post-LOCA leakage of isolation valves in the safety injection and containment spray system recirculation lines to the RWT, which is vented directly to the atmosphere (Reference 7). During the recirculation phase, sump water is recirculated through the ECCS pumps and could leak through various valves and reach the RWT. The two pathways include the two valves in series in the minimum flow recirculation line header (MOV659/660) and the valve from the containment spray pumps (SI459).

Assumptions used are:

- The liquid volume of the containment sump is 68,329 ft^3.
- The air volume of the RWT is 52,109.75 ft^3.
- The minimum time to RAS is 30 minutes.
- The leakage through SI459 and MOV659/660 is limited to 1,000 cc/hr. Per Reference 1, this leakage must be doubled to account for valve degradation between testing.
- A 10% flashing fraction is conservatively assumed.
- A 4.2 cfm leakrate from the RWT atmosphere to the environment is assumed.
- Radioiodine that is postulated to be available for releases from the RWT to the environment is assumed to be 97% elemental and 3% organic.
- The 0-2 hour RWT-to-EAB χ/Q was calculated to be 1.44E-04 sec/m^3.
- The 0-2, 2-24, and 24-720 hour RWT-to-LPZ χ/Qs were calculated to be 3.39E-05, 2.20E-06, and 5.40E-07 sec/m^3, respectively.
- The worst-case 0-2, 2-8, 8-24, 24-96, and 96-720 hour RWT-to-Control Room χ/Qs were calculated to be 2.57E-03, 2.13E-03, 8.50E-04, 5.71E-04, and 4.85E-04 sec/m^3, respectively, using the ARCON96 computational methodology.

14.24.2.7 Containment Shine

Per Reference 1, the MHA analysis should consider radiation shine from radioactive material in the Containment.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- These isotopic values together with the Containment and Control Room geometries are input into the Microshield point kernel computer code to calculate containment shine doses in the Control Room. Note that Microshield will allow parent decay and daughter buildup, so that subsequent isotopic quantities as a function of decay time are automatically calculated by Microshield.
- No removal of airborne radioactivity is assumed except for decay.

14.24.2.8 Plume Shine

Per Reference 1, the MHA analysis should consider shine from the external radioactive plume released from Containment.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- No removal of airborne radioactivity is assumed except for decay.
- The maximum allowable containment leakage rate L_a contained in the Containment Leakage Rate Testing Program of Technical Specification 5.5.16 was assumed, 0.16% per day at P_a. Per Reference 1, the Containment is assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for

the remaining duration of the accident. Thus the fraction of the containment activity that leaks from the Containment over 30 days can be calculated to be 0.0016+0.0008*29=0.0248.

- These isotopic values are input into the Microshield point kernel computer code to calculate plume shine doses in the Control Room. The Microshield results assume that all of the released activity is released from Containment at the beginning of the accident and sits over the Control Room for 30 days. To correct for release timing and dispersion via wind, a dilution factor is calculated which conservatively assumes that all containment release passes directly over the Control Room at a wind speed that is one-tenth the wind speed integrated over 8 years.

14.24.2.9 Control Room Filter Shine

Per Reference 1, the analysis should consider radiation shine from the radioactive material buildup of the Control Room recirculation filters.

Assumptions used are:

- All of the failed fuel isotopics emitted during the gas release and early in-vessel damage phases are assumed to be released into the containment atmosphere at the MHA initiation.
- The maximum allowable containment leakage rate is 0.16% per day at P_a. Per Reference 1, the Containment should be assumed to leak at the leak rate incorporated in the Technical Specifications for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.
- For the Control Room filter shine calculations, the containment atmosphere is cleansed by the iodine removal system, which filters particulates and elemental and organic iodine and by spray, which removes particulates.
- The activity released to the environment is transported to the Control Room via appropriate atmospheric dispersion coefficients.
- The Control Room inleakage is a constant 3,500 cfm in this analysis.
- Control Room filtration is credited based on a recirculation flow at a nominal 10,000 cfm. A charcoal filter efficiency of 100% is credited for elemental, organic, and particulate iodine.
- The resulting isotopic values are input into Microshield to calculate Control Room filter shine. Note that Microshield will allow parent decay and daughter buildup, so that subsequent isotopic quantities as a function of decay time are automatically calculated by Microshield.

14.24.3 RESULTS

The EAB, LPZ, and Control Room doses for the design-basis MHA are detailed in the following table.

MHA Results

Results	**EAB Rem**	**LPZ Rem**	**Control Room Rem**
Containment Pathway	1.70	0.423	3.780
Penetration Room Pathway	0.16	3.5 E-02	0.37
RWT Pathway	3.7 E-05	1.9 E-03	0.33
Hydrogen Purge Pathway	6.5 E-05	1.5 E-05	7.7 E-05
Containment Shine			5.5 E-02
Plume Shine			3.0 E-03

MHA Results

Results	EAB Rem	LPZ Rem	Control Room Rem
Control Room Filter Shine			1.4 E-02
Total	1.86	0.46	4.57
Regulatory Limits	25	25	5

Note that all values are below the regulatory limits.

This event includes the transition to AREVA Advanced CE-14 HTP™ fuel (with Gd_2O_3 burnable poison irradiated to a maximum burnup of 62 GWd/MTU). A source term was developed that bounds the transition cycles and the full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

14.24.4 REFERENCES

1. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000
2. CCNPP Calculation CA06449, Revision 1, “Maximum Hypothetical Accident Using Alternative Source Terms,” dated August 25, 2010
3. Letter from B. S. Montgomery (CCNPP) to Document Control Desk (NRC), License Amendment Request: Revision to Accident Source Term and Associated Technical Specifications, dated November 3, 2005
4. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), “Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846),” dated August 29, 2007
5. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988
6. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993
7. NRC Information Notice No. 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere,” dated September 19, 1991
8. Regulatory Guide 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessment at Nuclear Power Plants,” dated June 2003
9. Regulatory Guide 1.52, Revision 2, “Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,” dated March 1978
10. NUREG/CR-6189 SAND94-0407, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments”, dated July 1996
11. NUREG-0800, SRP 6.5.2, Revision 2, “Containment Spray as a Fission Product Cleanup System,” dated December 1988

14.25 EXCESSIVE CHARGING EVENT

14.25.1 IDENTIFICATION OF EVENT AND CAUSE

The Excessive Charging Event initiated from maximum pressurizer level was analyzed to assure that the operator has at least 15 minutes from initiation of a high pressurizer level alarm to take corrective action and terminate the event prior to filling the pressurizer solid. The Excessive Charging Event is assumed to occur by inadvertent initiation of charging flow.

14.25.2 CORE AND SYSTEM PERFORMANCE

14.25.2.1 Mathematical Models

The time required to fill the pressurizer solid was calculated using the following equation:

$$T = \frac{V_s - V_{sL} - V_T}{(F_{CH} - F_{LD})\frac{W_2}{W_1}}$$

where:

V_S = steam volume in the pressurizer;
V_{SL} = equivalent saturated liquid volume pressurizer steam volume;
V_T = volume above the spray nozzles;
F_{CH} = charging flow rate;
F_{LD} = letdown flow rate;
W_1 = specific volume of liquid at charging and letdown conditions; and,
W_2 = specific volume of liquid at pressurizer conditions.

14.25.2.2 Input Parameters and Initial Conditions

The analysis was performed for three combinations of charging and letdown flows. Table 14.25-1 presents the initial conditions assumed in the analysis and the results of the analysis. The initial conditions assumed in the analysis are consistent with the Technical Specification limits on initial pressurizer level including appropriate instrument loop uncertainties.

14.25.2.3 Results

As seen from Table 14.25-1, all three combinations of charging and letdown flow analyzed provide at least 15 minutes after initiation of high level alarm for the operator to take corrective actions and terminate the event prior to filling the pressurizer solid.

14.25.3 CONCLUSIONS

The operator has at least 15 minutes after initiation of a high level alarm to take corrective action and terminate the event prior to filling the pressurizer solid.

This event is not affected by the transition to AREVA Advanced CE-14 HTP™ fuel because the key parameters for this event are plant related system responses which are unchanged from, or bounded by, the current analysis. These parameters are not adversely affected by either the transition cycle or full core implementation of AREVA fuel. Therefore, this analysis remains applicable to plant operation with AREVA fuel.

TABLE 14.25-1

EXCESSIVE CHARGING EVENT - CORE AND SYSTEM PERFORMANCE FOR CHARGING FLOW AND LETDOWN FLOW

VOLUME CONTROL ASSUMPTIONS				MAXIMUM INITIAL PRESSURIZER LEVEL		HIGH LEVEL ALARM ANALYSIS SETPOINT		TIME TO FILL[a] PRESSURIZER
	CHARGING FLOW (gpm)	CBO	LETDOWN FLOW (gpm)	LIQUID VOLUME (ft^3)	LEVEL[c] (in)	LIQUID VOLUME (ft^3)	LEVEL[c] (in)	(minutes)
1.	144	6	0	853	218	853	218	16[d]
2.	144	6	29[b]	952	242	1014	257	15
3.	96	6	0	952	242	1094	276	15

(a) From time of initiation of high pressurizer level alarm.

(b) Conservatively lower than the 30 gpm minimum letdown flow listed in Figure 4-11.

(c) Referenced to the 1" level nozzle at the bottom of the pressurizer.

(d) One minute to recognize computer signal leaving 15 minutes for operator response.

14.26 FEEDLINE BREAK EVENT

14.26.1 IDENTIFICATION OF EVENT AND CAUSE

A Feedline Break (FLB) may occur as a result of thermal stress or cracking in the main feedline. The guillotine-type break assumed in the Safety Analysis is the most adverse transient scenario and its probability of occurrence is extremely low. Installation of a safety-grade AFAS forced the inclusion of this event into the scope of DBEs.

The safety-grade AFAS consists of two components.

a. Automatic initiation of AFW to both SGs is based on a low level signal from either one of the SGs. This signal is generated when the water level in either SG decreases below a nominal analysis setpoint value based on wide range level indication.

b. Isolation logic identifies and isolates a ruptured SG. A SG with lower pressure (in comparison to the other SG) is identified as being ruptured and is isolated when the SG differential pressure (i.e., P^{SG}[A] - P^{SG}[B]) exceeds the analysis setpoint value. The safety analysis was performed by including appropriate uncertainties to the nominal analysis setpoint.

The FLB event is analyzed to demonstrate that the RCS pressure limit of 2750 psia is not exceeded and that the site boundary doses do not exceed 10 CFR 50.67 guidelines. The event was analyzed with Loss of AC Power on turbine trip for primary system overpressure and without Loss of AC Power on turbine trip for secondary system overpressure and decay heat removal by AFW. A spectrum of break sizes were considered in the primary system overpressure analysis and the results of the limiting break size are presented here.

14.26.2 SEQUENCE OF EVENTS

The FLB event is initiated by a break in the MFW System piping. Depending on the break size and location and the response of the MFW System, the effects of a break can vary from a rapid heatup to a rapid cooldown of the RCS. In order to discuss the possible effects, breaks are categorized as small if the associated discharge flow is within the excess capacity of the MFW System, and as large if otherwise. Break locations are identified with respect to the feedwater line reverse flow check valve. The reverse flow check valve of concern is located between the SG feedwater nozzle and the containment penetration. Closure of the check valve, to prevent reverse flow from the SG, maintains the heat removal capability of that SG in the presence of a break upstream of the check valve.

Feedwater line breaks upstream of the reverse flow check valve can initiate one of the following transients. A break of any size, with MFW System unavailable, will result in a LOFW Flow event. A small pipe break with MFW System available will result in no reduction in feedwater flow. Depending on the break size, a large break with MFW System available will result in either a partial or a total LOFW Flow event. Since FLBs upstream of the reverse flow check valve result in transients no more severe than a LOFW Flow event, these FLBs were not analyzed.

In addition to the possibility of partial or total LOFW flow, FLBs downstream of the check valve have the potential to establish reverse flow from the affected SG back to the break. Reverse flow occurs whenever the MFW System is not operating subsequent to a pipe break, or when the MFW System is operating, but without sufficient capacity to maintain pressure at the break above the SG pressure. Feedline Breaks which develop reverse

flow through the break are limiting with respect to RCS overpressure. Thus, only these FLBs were considered in the analysis.

Feedline Breaks downstream of the check valve with reverse flow may result in either an RCS heatup or an RCS cooldown event, depending on the enthalpy of the reverse flow and the heat transfer characteristics of the affected SG. However, excessive heat removal through the feedwater line break is not considered in the analysis because the cooldown potential is less than that for the SLB event. This occurs because SLBs have a greater potential for discharging high enthalpy fluid because the steam piping is located above the feedwater piping within a SG.

Unlike SLBs, FLBs cause a decrease in feedwater flow, resulting in lower SG liquid inventory that reduces the heat removal capacity. The reduced heat transfer capability results in a rapid RCS overpressurization and, thus, it is the heatup potential of an FLB which was analyzed.

A general description of the FLB event downstream of the check valves, with the MFW System unavailable and with low enthalpy break discharge, is given below. The loss of subcooled feedwater flow to both SGs causes increasing SG temperatures, decreasing liquid inventories and decreasing water levels. The rising secondary temperature reduces the primary-to-secondary heat transfer, which results in a heatup and pressurization of the RCS. The heatup becomes more severe as the affected SG experiences a further reduction in its heat transfer capability due to decreasing liquid inventory. The heatup of the RCS and the depletion of liquid inventory in the SG will initiate a reactor trip on either high pressurizer pressure or SG low water level. The RCS heatup can continue even after a reactor trip, due to a total loss of heat transfer in the affected SG as the liquid inventory is completely depleted. The rise in RCS pressure causes the PSVs to open. The rise in secondary pressure is limited by the opening of the MSSVs. The opening of the PSVs and the MSSVs, in conjunction with the reactor trip (which reduces core power to decay level), mitigates the RCS overpressurization.

The reduction of liquid inventory in the unaffected SG in conjunction with low level SG signal initiates AFW flow to the unaffected SG. Automatic initiation of AFW in combination with operator action at 10 minutes to increase AFW flow is sufficient to provide a continued heat sink for the removal of decay heat.

14.26.3 CORE AND SYSTEM PERFORMANCE

14.26.3.1 Mathematical Models

The FLB event is simulated using the S-RELAP5 computer code described in Section 14.1.4.2. The simulation includes the effects of tripping the RCPs on Loss of AC Power. The automatic actuation of AFAS is credited, together with operator action after 10 minutes.

14.26.3.2 Input Parameters and Initial Conditions

The following is a discussion of the conservative assumptions and initial conditions chosen to maximize RCS pressure:

Blowdown of the SG nearest the feedwater line break is modeled using the Moody critical flow model with homogeneous flow and a sudden expansion loss coefficient. The break is conservatively modeled from below the feedwater ring in the SG downcomer. These modeling choices generate conservative break flow and primary-to-secondary heat transfer, and thus a conservative RCS overpressurization.

For smaller break sizes credit is only required for the High Pressurizer Pressure Trip. As the break size is increased, credit is required for the SG Low Pressure Trip. No credit was taken for the High Containment Pressure Trip or the Low SG Level Trip.

Table 14.26-1 presents the initial conditions chosen to maximize the RCS pressure. An MTC value corresponding to BOC conditions is assumed. The MTC, in conjunction with increasing coolant temperatures, adds positive reactivity, and, thus, maximizes the rate of change of heat flux and pressure at the time of trip. An FTC corresponding to BOC conditions is used in the analysis. This FTC causes the least amount of negative reactivity feedback, allowing higher increases in both the heat flux and RCS pressure. An uncertainty factor of 10% is applied to the FTC used in the analysis.

A minimum initial RCS pressure is used in the analysis to maximize the rate of change of pressure at time of trip and, thus, the peak pressure obtained following a reactor trip.

The SDBS, the PPCS, the PLCS and the PORVs are assumed to be in the manual mode of operation. This assumption enhances the RCS pressure increase, since the automatic operation of these systems mitigates the RCS pressure increase.

This analysis credited the automatic initiation of AFW on low SG level. At 10 minutes after reactor trip, operator action was credited to control AFW flow to maintain SG level.

The assumptions made to maximize the boundary site dose are given in Table 14.26-2. During the event, two sources of radioactivity contribute to the site boundary dose: (1) the initial activity in the SG, and (2) the activity associated with primary-to-secondary leakage. The leakage through the SG tubes is assumed to be 1.0 gpm. The initial primary and secondary activities are assumed to be at the Technical Specification limits of 0.5 μCi/gm (DEQ I-131) and 0.1 μCi/gm (DEQ I-131), respectively. The analysis assumes that all of the initial activity in the SGs and the primary activity due to the tube leakage are released to the atmosphere with a decontamination factor of 1.0, resulting in the maximum site boundary dose.

14.26.3.3 Results

The FLB event with Loss of AC (LOAC) power on reactor trip results in the maximum RCS pressure. This occurs because the LOAC power causes the RCPs to coast down. The reduced core flow decreases the rate of heat removal and, thus, maximizes the primary heatup and overpressurization. Thus, only the results of the FLB event with LOAC power on reactor trip are presented here.

Figure 14.26-1 shows the peak RCS pressure as a function of break size for the initial conditions and input parameters specified in Table 14.26-2. The peak RCS pressure occurred for a break size of 0.02 ft^2. The transient progression for key parameters can be observed on Figures 14.26-2 through 14.26-10 and the sequence of events for the limiting break size is included in Table 14.26-3. The limiting break size is so small that the damaged steam generator does not depressurize sufficiently to signal closure of the MSIVs and the differential pressures between the two steam generators does not reach the setpoint to automatically isolate the AFW to the damaged steam generator.

The resultant site boundary and Control Room doses calculated with the assumptions given in Table 14.26-2 are:

Exclusion Area Boundary	0.4 rem TEDE
Low Population Zone	0.01 rem TEDE
Control Room	1.10 rem TEDE

14.26.4 CONCLUSION

The results of the FLB event with LOAC power on turbine trip show that the peak pressure does not exceed the pressure upset limit of 2750 psia and that the site boundary doses are within 10 CFR 50.67 guidelines. The downward trend of the long-term RCS pressure and temperatures and increasing intact SG level show plant recovery.

14.26.5 REFERENCES

1. Not used.
2. Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988
3. Federal Guidance Report (FGR) 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993

TABLE 14.26-1

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED IN THE FEEDWATER LINE BREAK EVENT

PARAMETER	UNITS	VALUE[a]
Initial Core Power Level	MWt	2754[b]
Initial Core Inlet Coolant Temperature	°F	535[e]
Vessel Flow Rate	gpm	412,000
Initial Pressurizer Pressure	psia	2164
Initial Pressurizer Liquid Level	% span	32.2
Effective MTC	$x10^{-4}\ \Delta\rho/°F$	+0.15
Doppler Reactivity Feedback	---	Minimum
High Pressurizer Pressure Trip Setpoint[c]	psia	2470
AFAS setpoint inches below normal water level	inches BNWL	265.2
CEA Worth at Trip	$\% \Delta\rho$	-5.0
RRS	Operating Mode	Manual[d]
SDBS	Operating Mode	Manual[d]
Pressurizer Pressure Control System	Operating Mode	Manual[d]
Pressurizer Level Control System	Operating Mode	Manual[d]

(a) These values represent inputs to the limiting transient scenario analyzed for Unit 1. A range of selected parameters, including uncertainties, were evaluated to determine the most limiting set of initial conditions.

(b) Value does include approximately 17 MWt of pump heat calculated by S-RELAP5.

(c) The trip credited is dependent upon the break size. The limiting break size presented is associated with a small break size. Larger breaks are protected by the Low SG Pressure trip.

(d) These modes of control system operation maximize the peak RCS pressure.

(e) Minimum including EOC temperature coastdown conditions.

TABLE 14.26-2

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE FEEDLINE BREAK EVENT

PARAMETER	UNITS	VALUE
RCS Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.5
Secondary Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.1
Partition Factor Assumed for All Doses	---	1.0
Atmosphere Dispersion Coefficient[b]	sec/m^3	1.44E-04
Breathing Rate	m^3/sec	3.50E-4
Dose Conversion Factors per References 2 and 3		

[a] Technical Specification Limits.

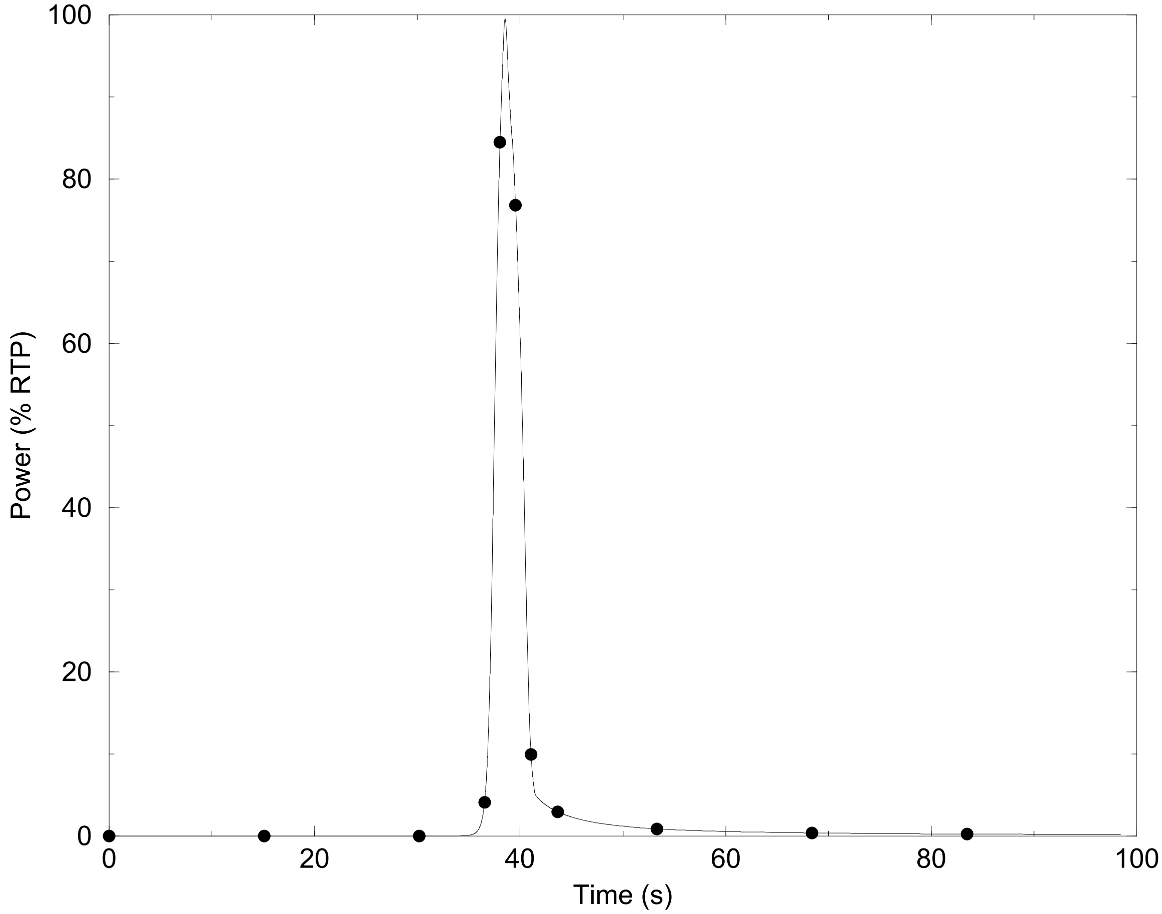
[b] 0-2 hour accident condition at EAB.

TABLE 14.26-3

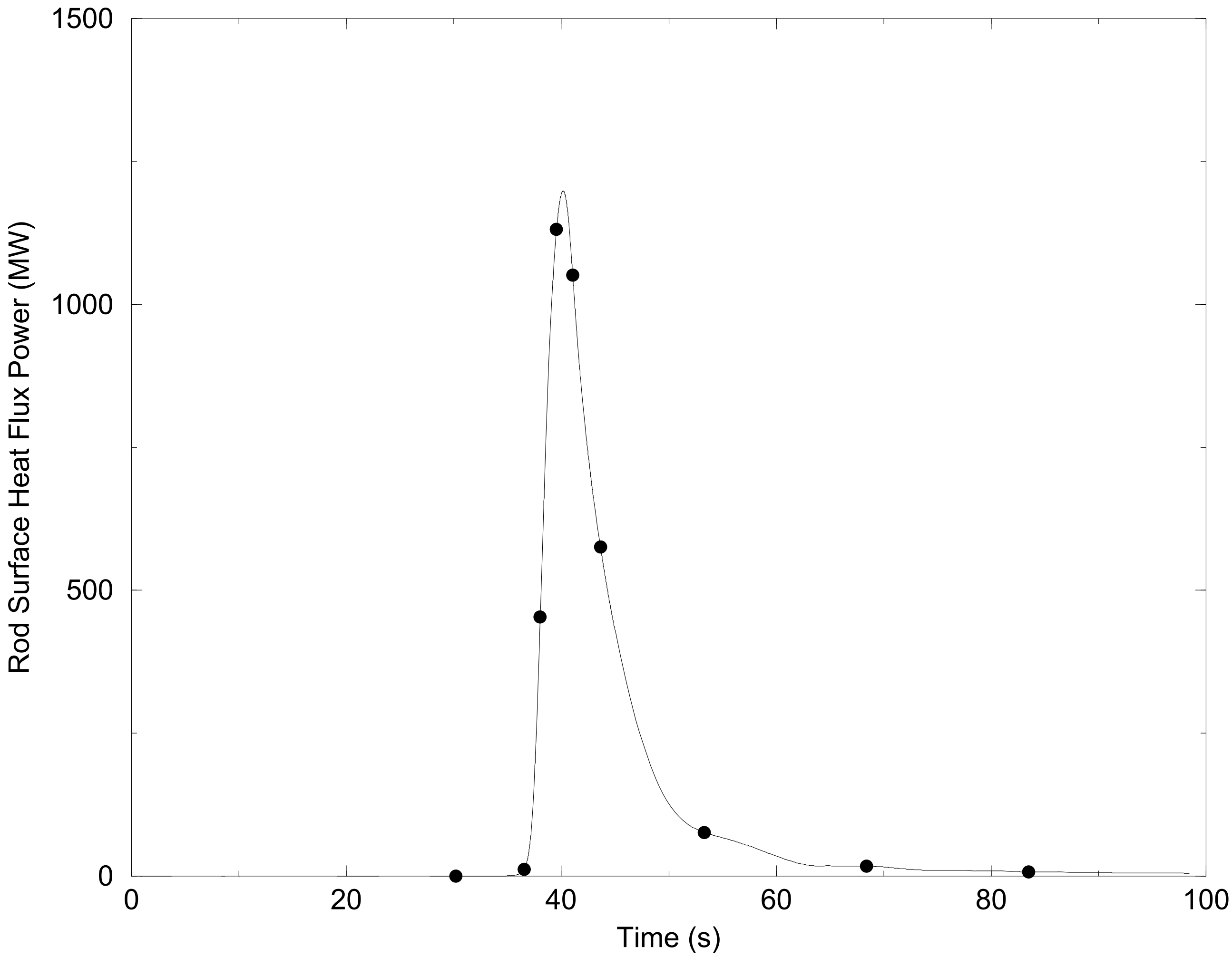
SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK WITH LOAC FOLLOWING REACTOR TRIP

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Event Initiation	0.02 ft^2
44.1	Low-Low SG level AFAS	----
57.8	High Pressurizer Pressure Trip Setpoint	2470.0 psia
58.7	Reactor and Turbine Trip	---
59.2	Scram Rods Fall	---
60.7	PSV RC-200 Opens	2575 psia
61.2	PSV RC-201 Opens	2601 psia
61.9	Peak RCS Pressure[a]	2730.7 psia
63.4	MSSVs Open	---
65.9	PSV RC-200 Closes	2497.0 psia
68.3	PSV RC-201 Closes	2472.2 psia
79.1	SG Blowdown Isolation	---

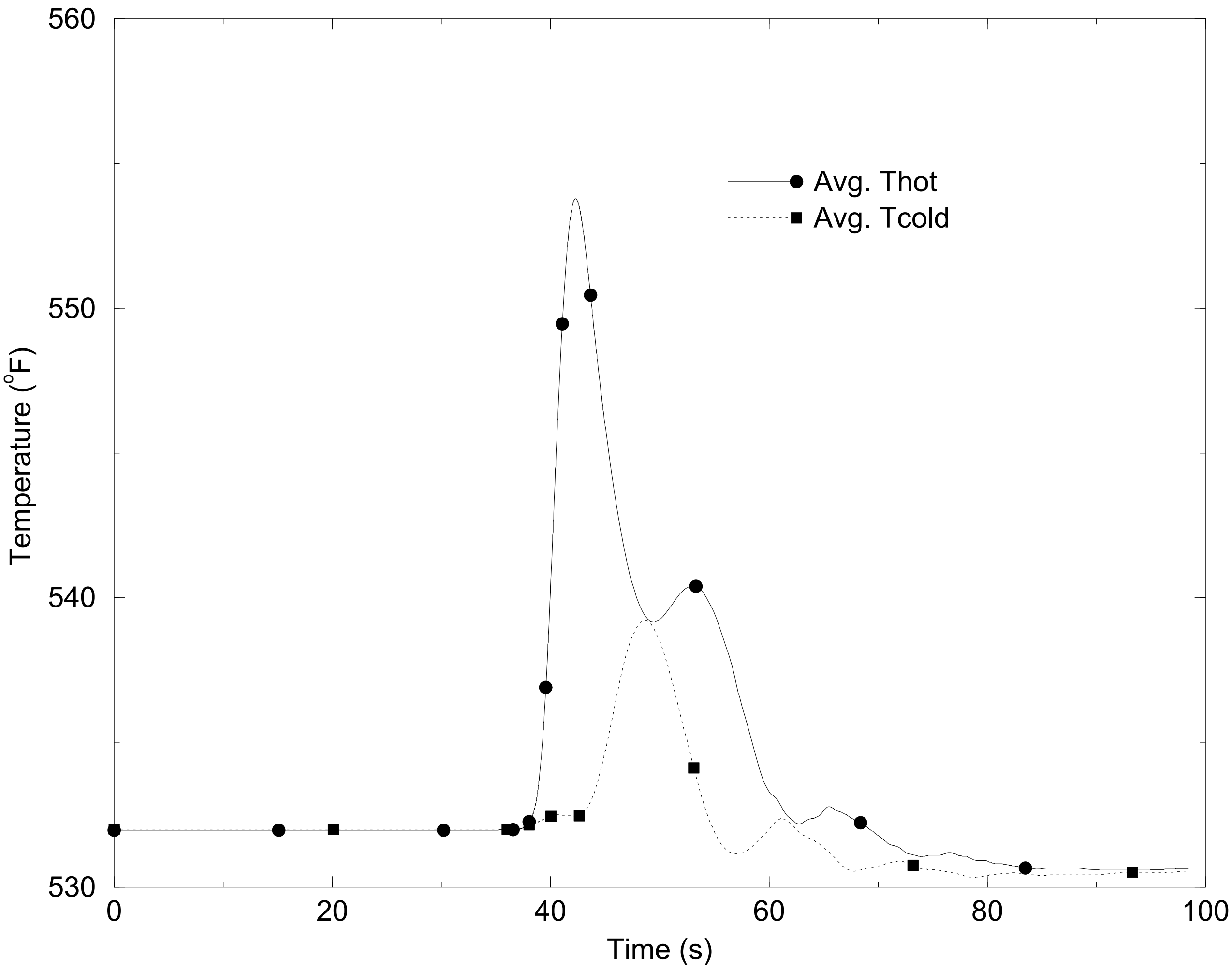
[a] Peak pressure includes elevation head.



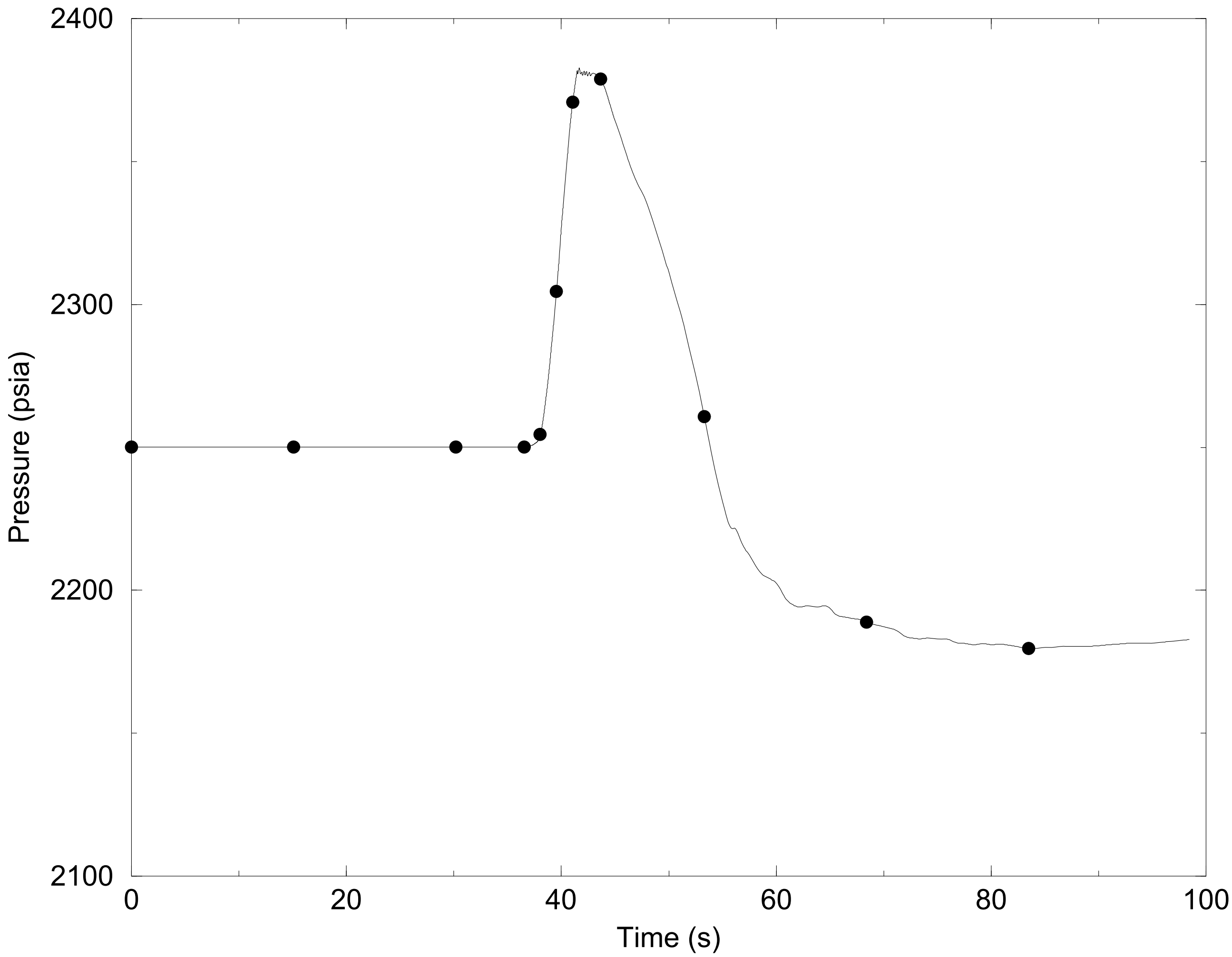
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HZP CORE POWER VS TIME	Figure 14.2-1 Revision 44



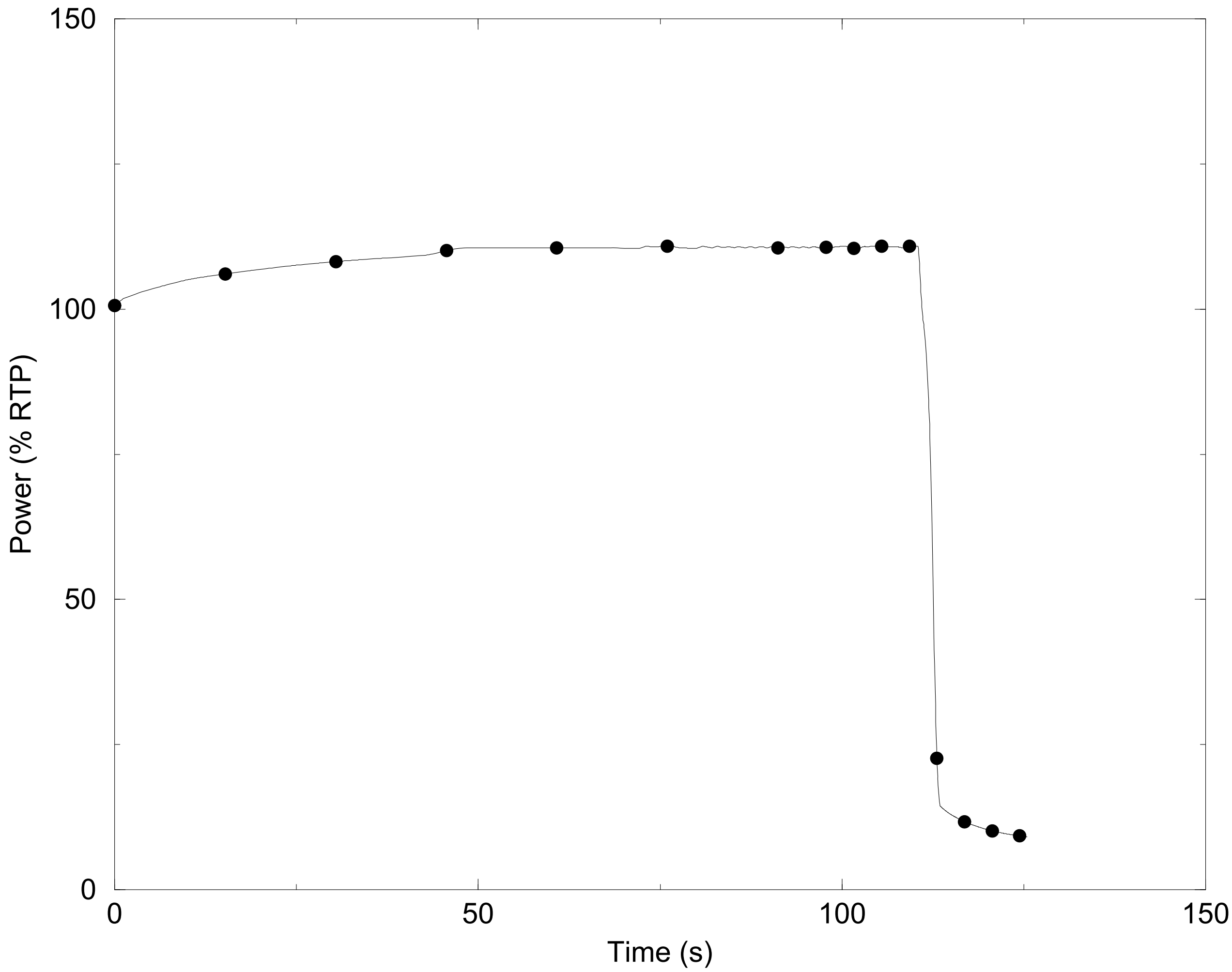
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HZP CORE HEAT FLUX VS TIME	Figure 14.2-2 Revision 44



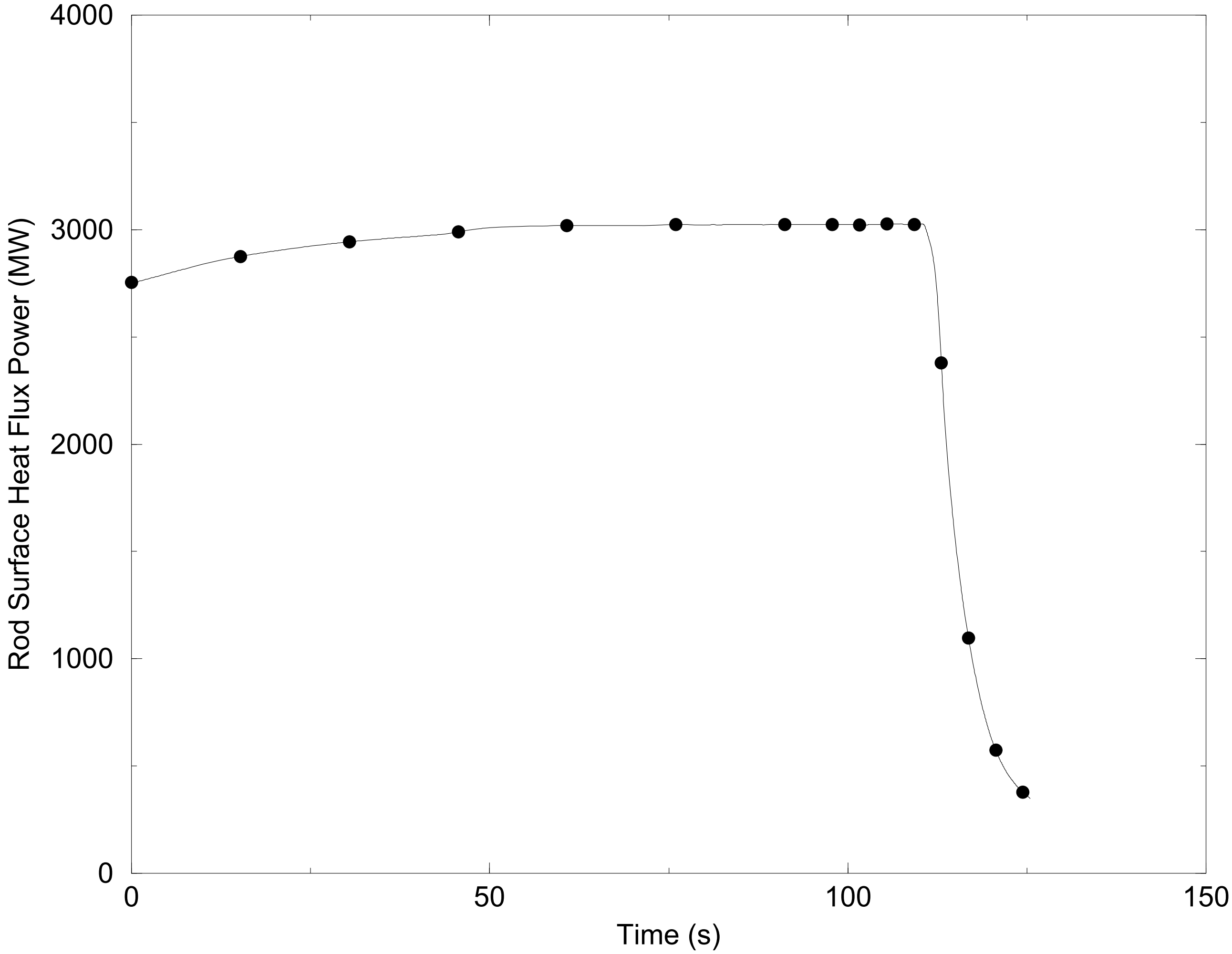
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HZP RCS TEMPERATURES VS TIME	Figure 14.2-3 Revision 44



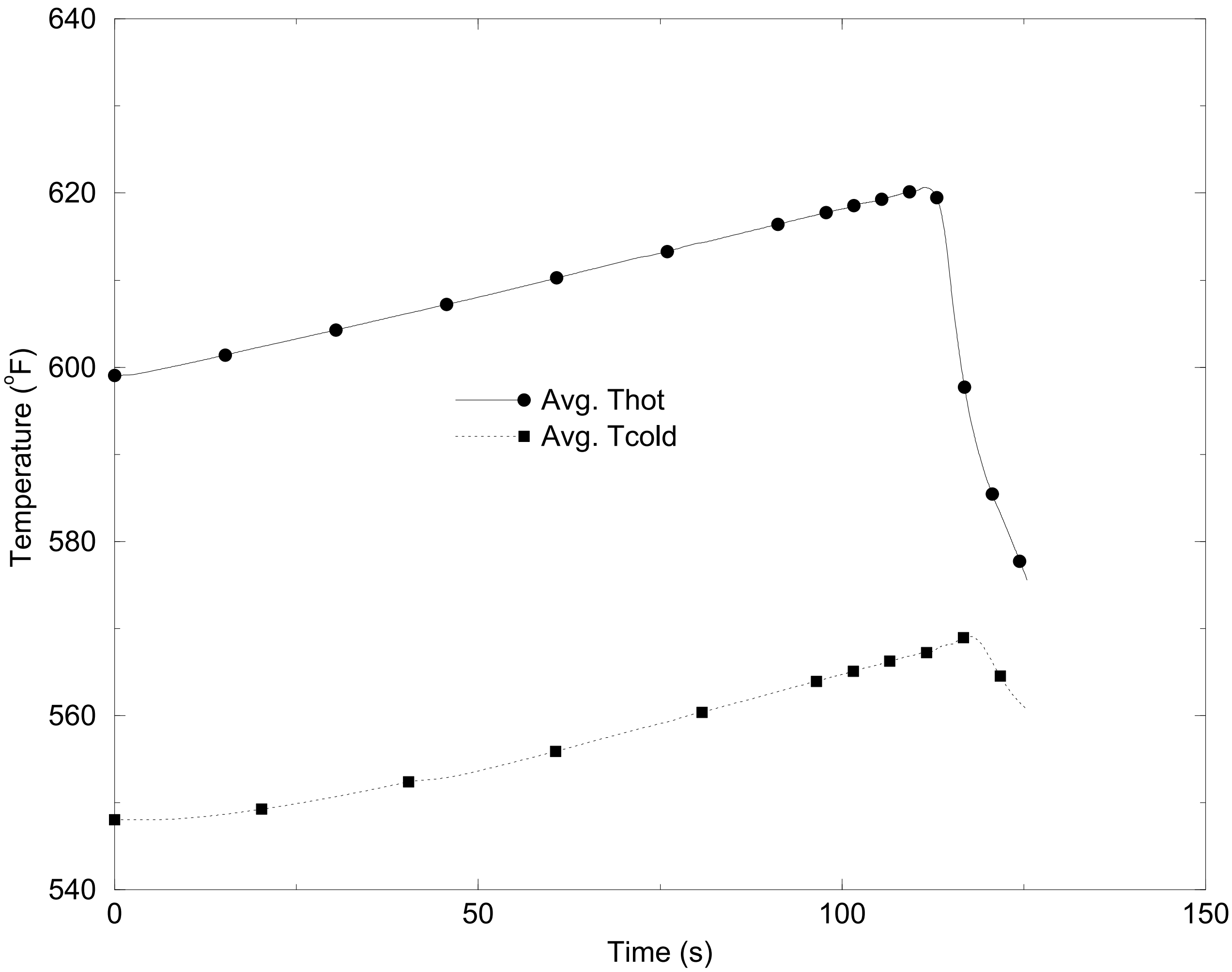
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HZP RCS PRESSURE VS TIME	Figure 14.2-4 Revision 44



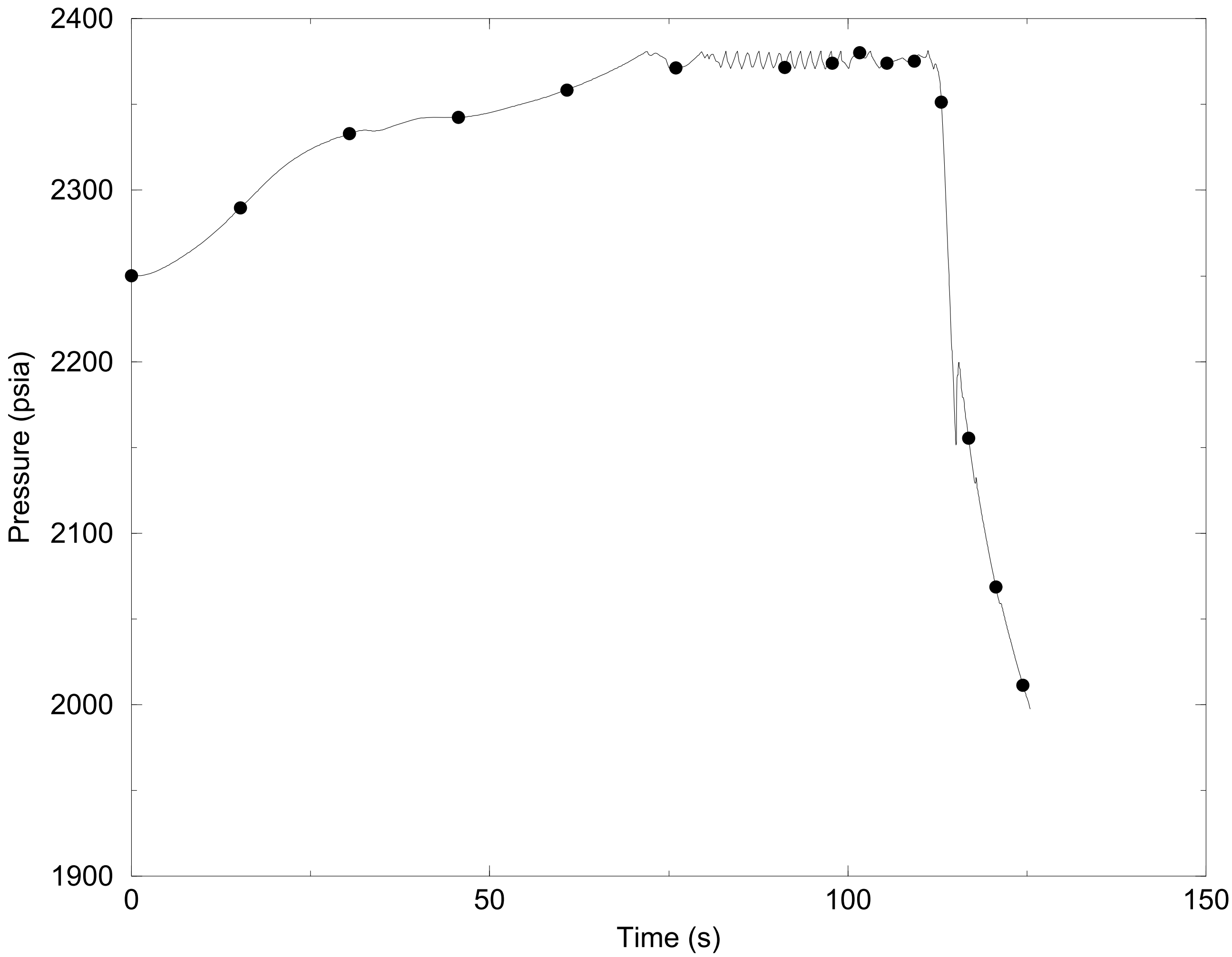
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HFP CORE POWER VS TIME	Figure 14.2-5 Revision 44



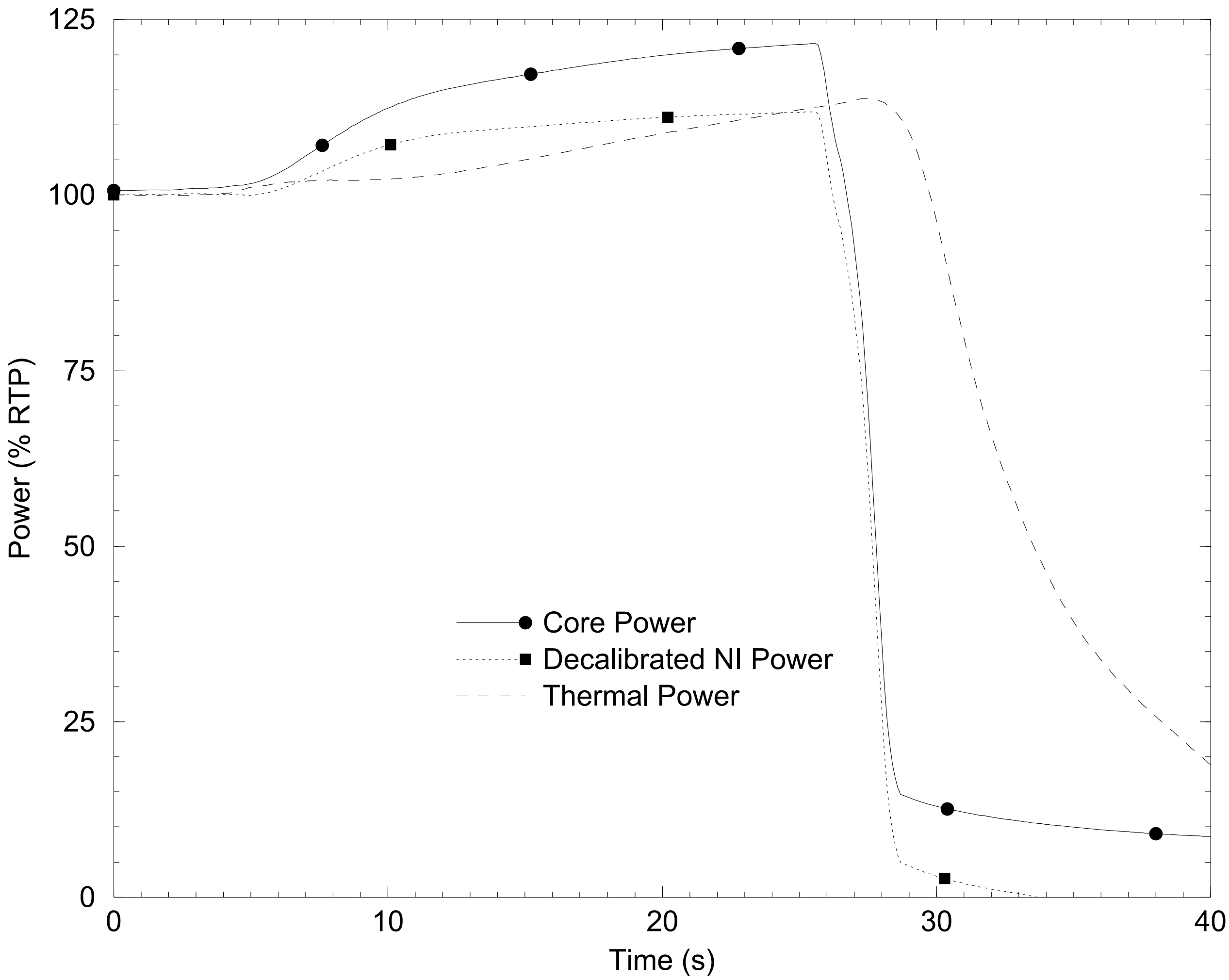
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HFP CORE HEAT FLUX VS TIME	Figure 14.2-6 Revision 44



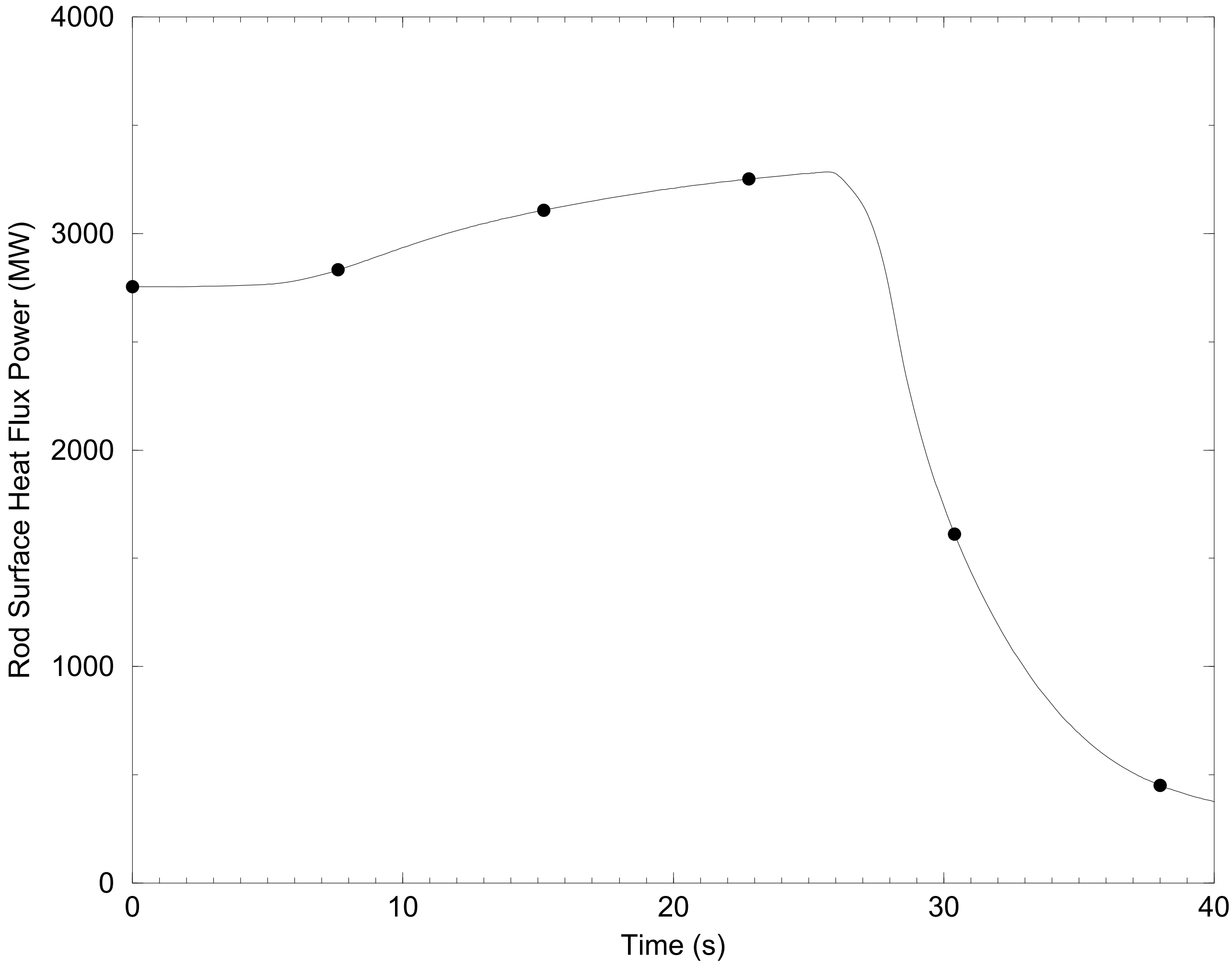
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HFP RCS TEMPERATURES VS TIME	Figure 14.2-7 Revision 44



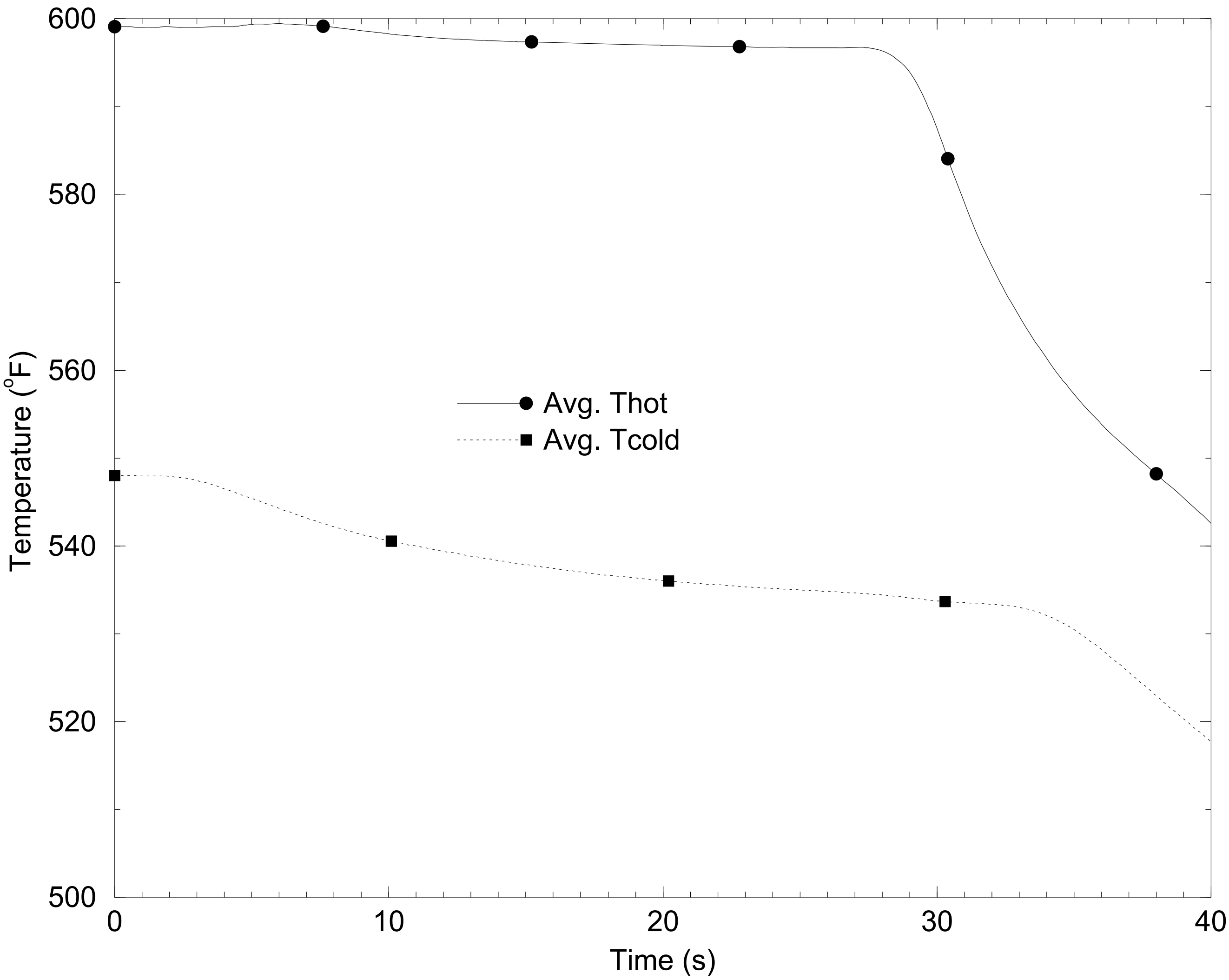
Calvert Cliffs Nuclear Power Plant	CEAW EVENT – HFP RCS PRESSURE VS TIME	Figure 14.2-8 Revision 44



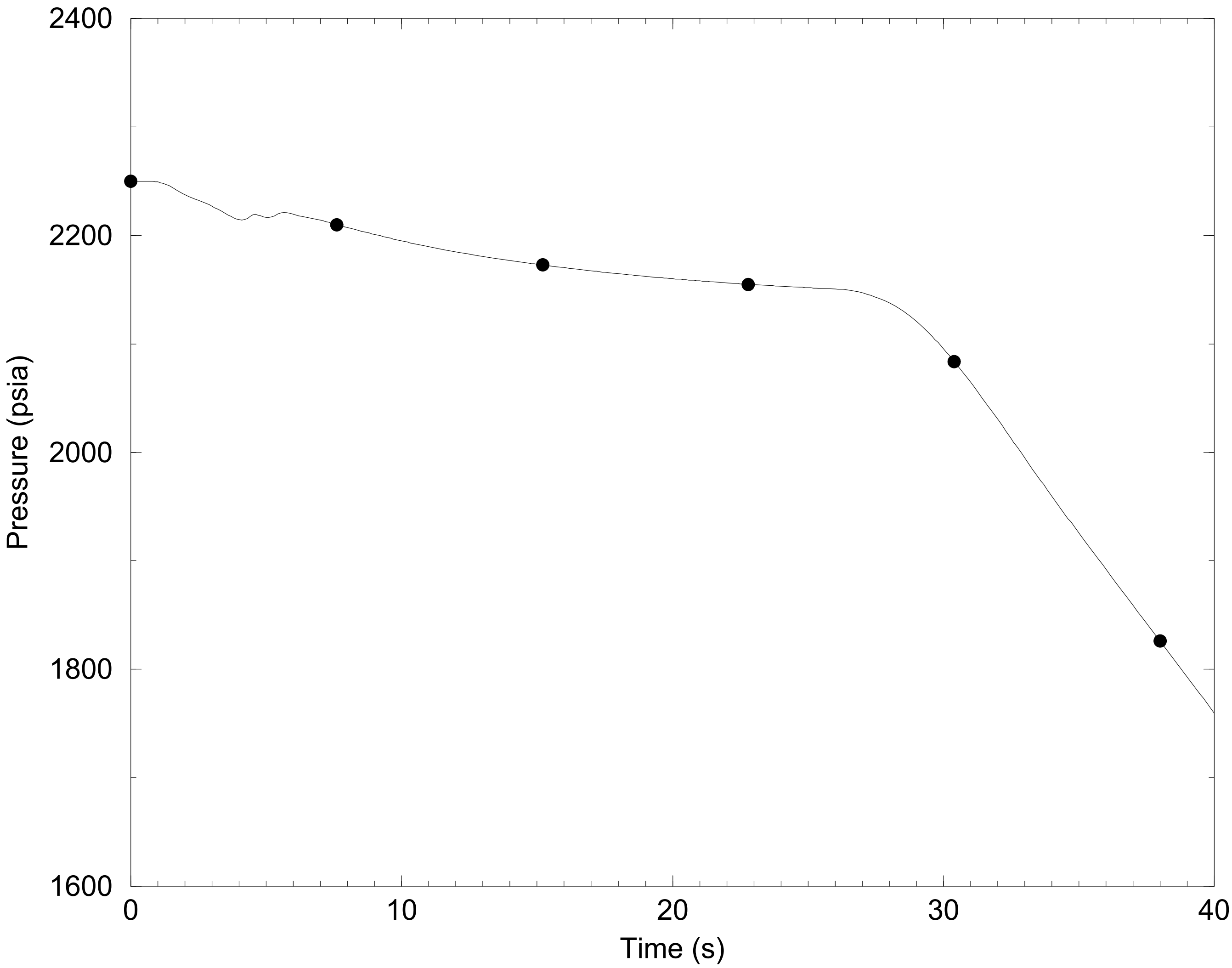
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE POWER VS TIME (HFP)	Figure 14.4-1 Revision 44



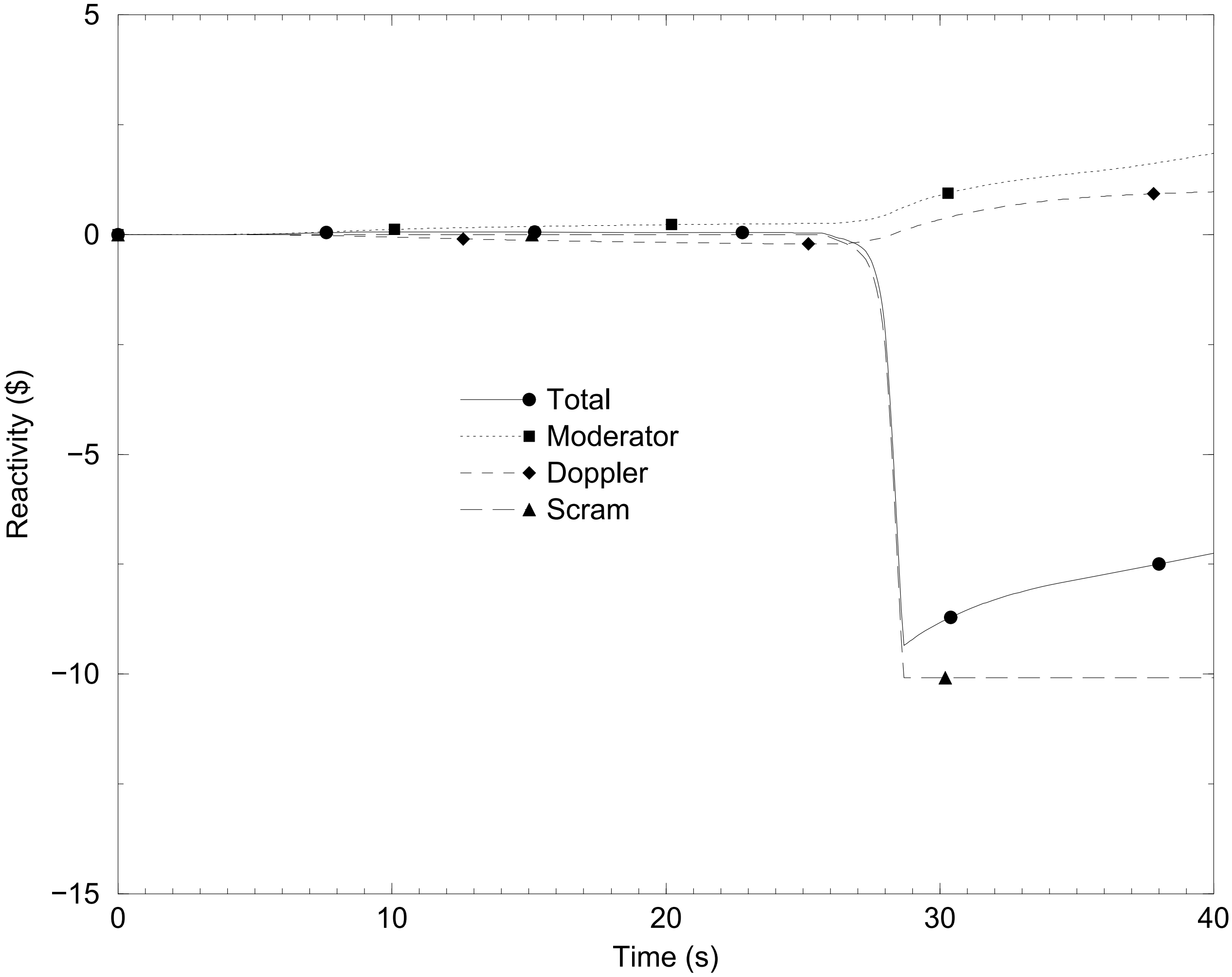
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE HEAT FLUX VS TIME (HFP)	Figure 14.4-2 Revision 44



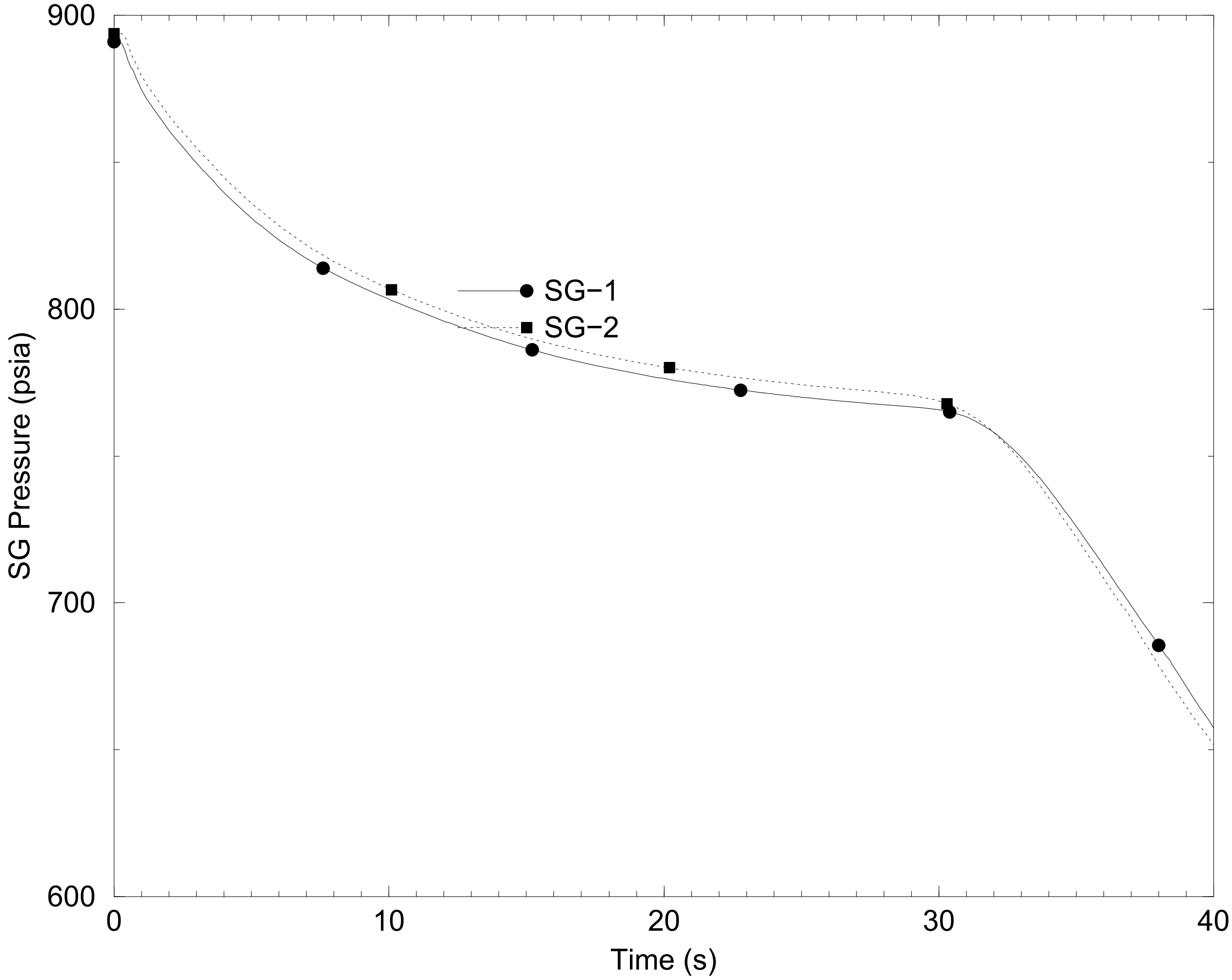
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT RCS TEMPERATURES VS TIME (HFP)	Figure 14.4-3 Revision 44



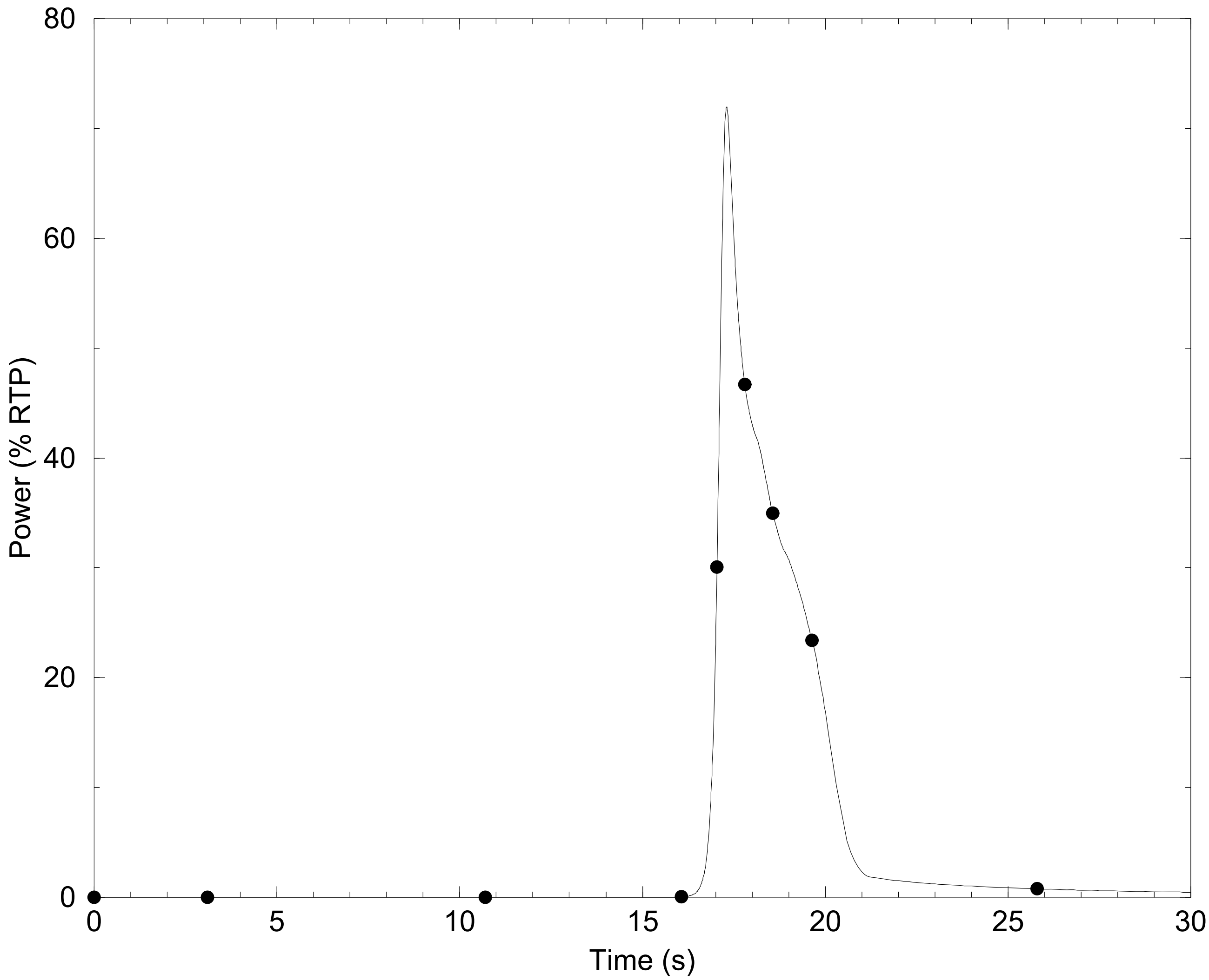
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT PRESSURIZER PRESSURE VS TIME (HFP)	Figure 14.4-4 Revision 44



Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTIVITIES VS TIME (HFP)	Figure 14.4-5 Revision 44



Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT SG PRESSURES VS TIME (HFP)	Figure 14.4-6 Revision 44

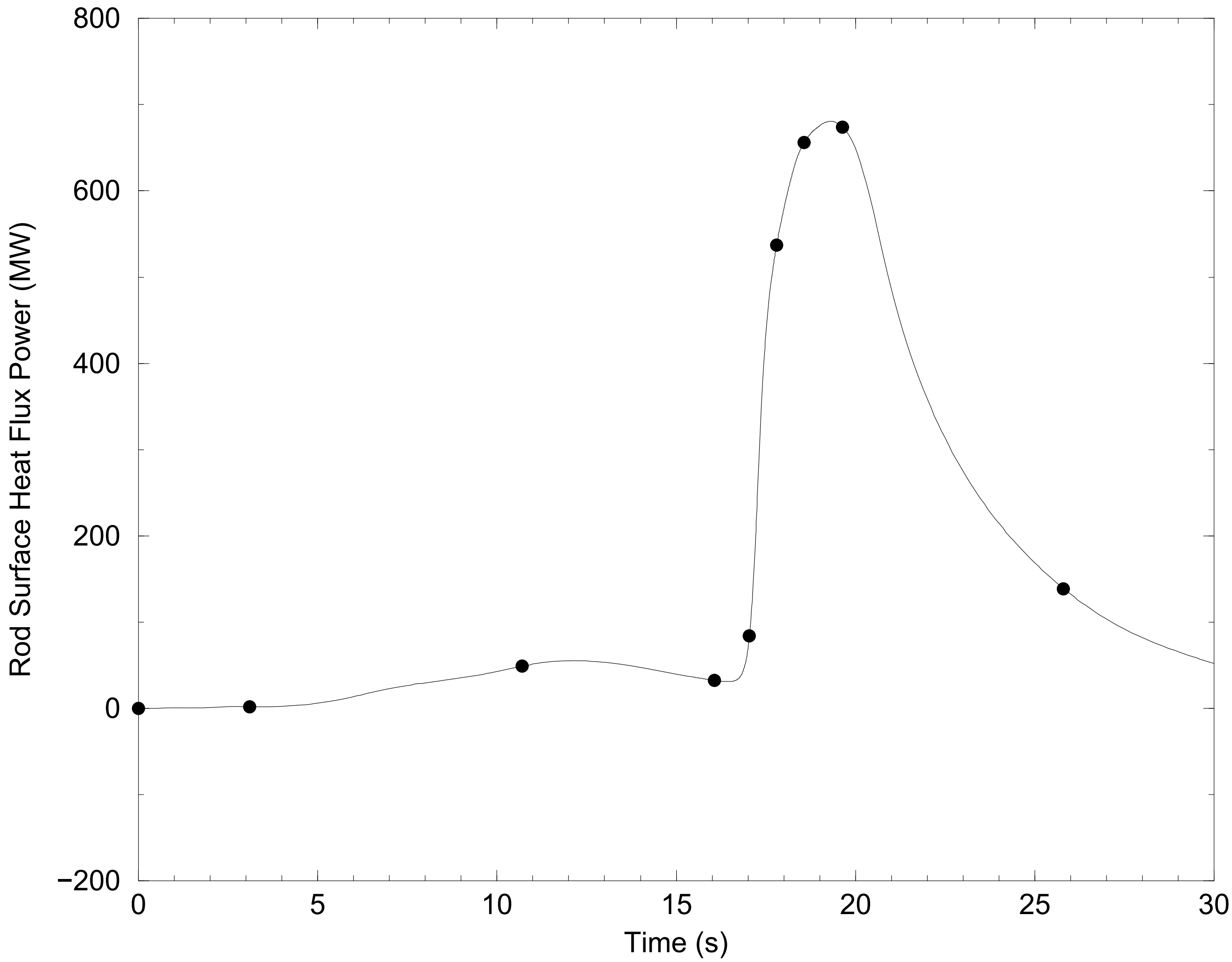


Calvert Cliffs Nuclear Power Plant

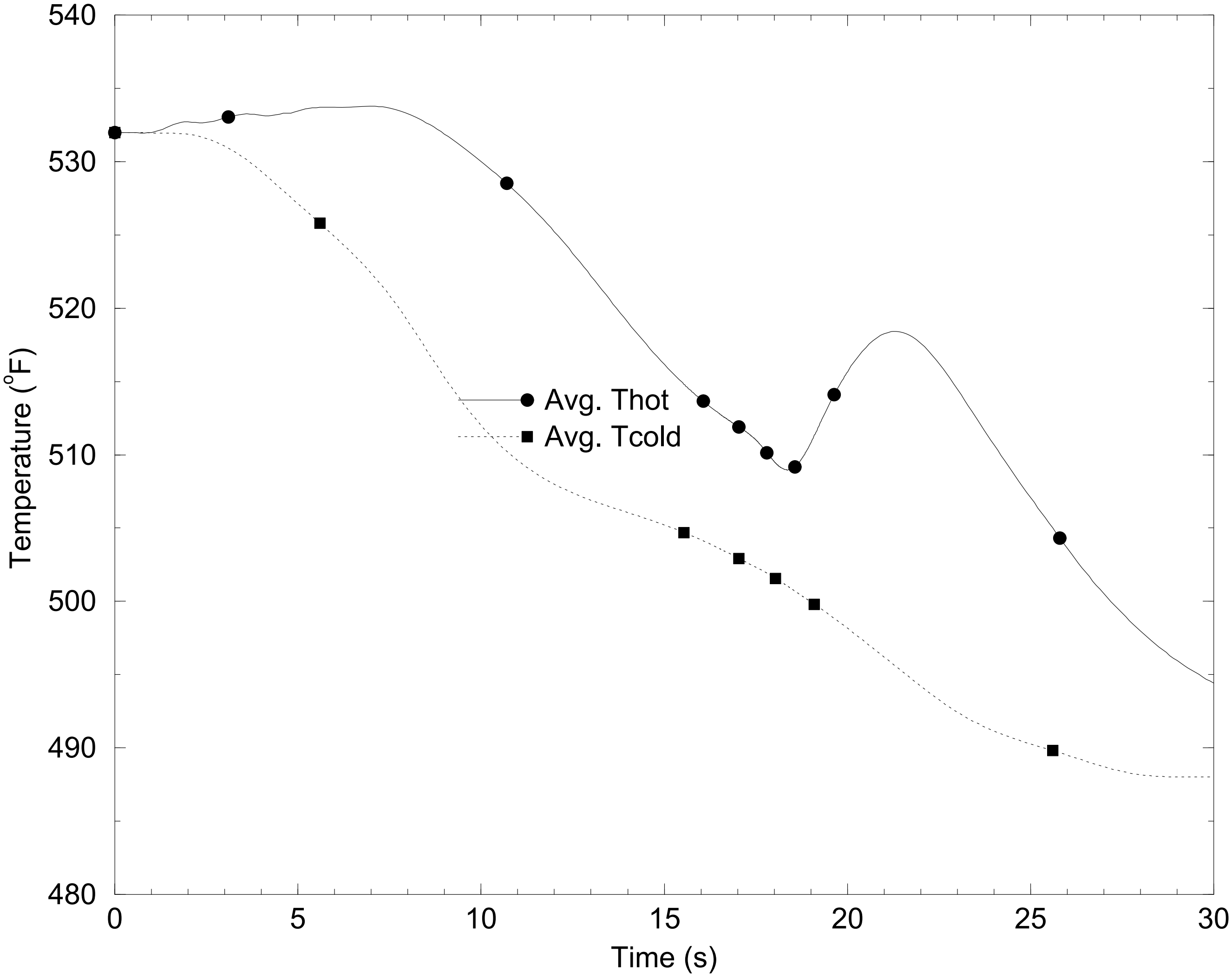
EXCESS LOAD EVENT
CORE POWER VS TIME (HZP)

Figure 14.4-7

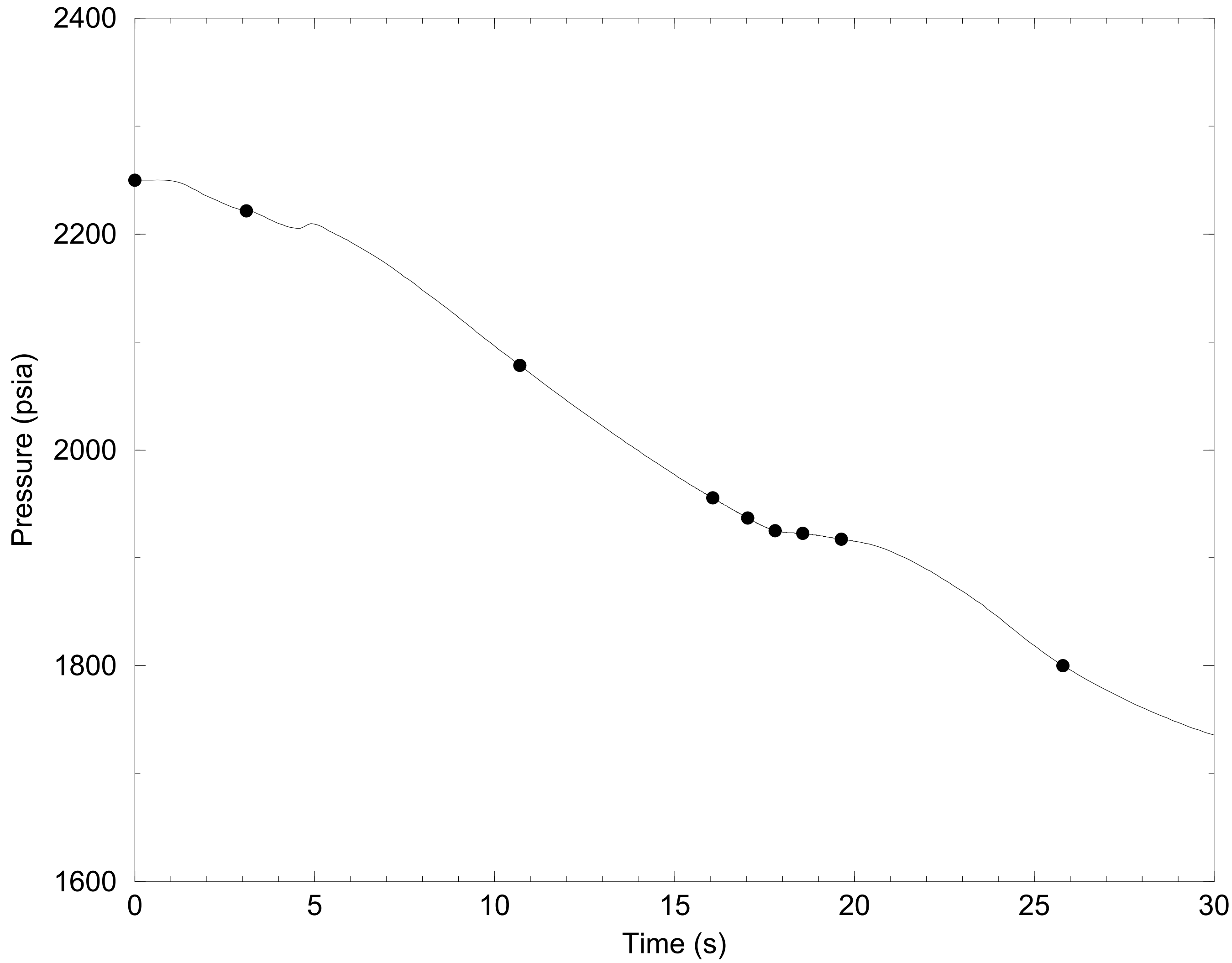
Revision 44



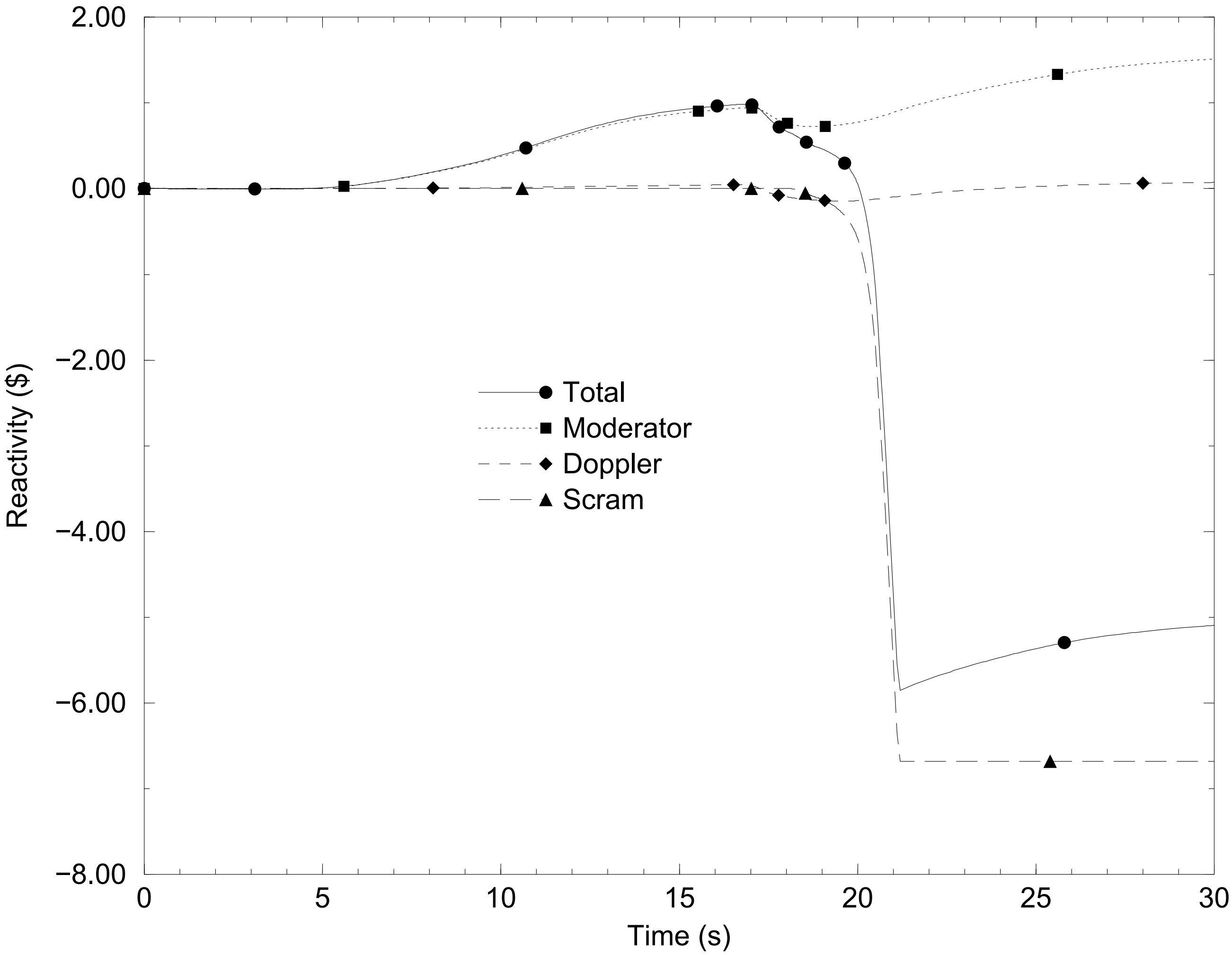
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE HEAT FLUX VS TIME (HZP)	Figure 14.4-8 Revision 44



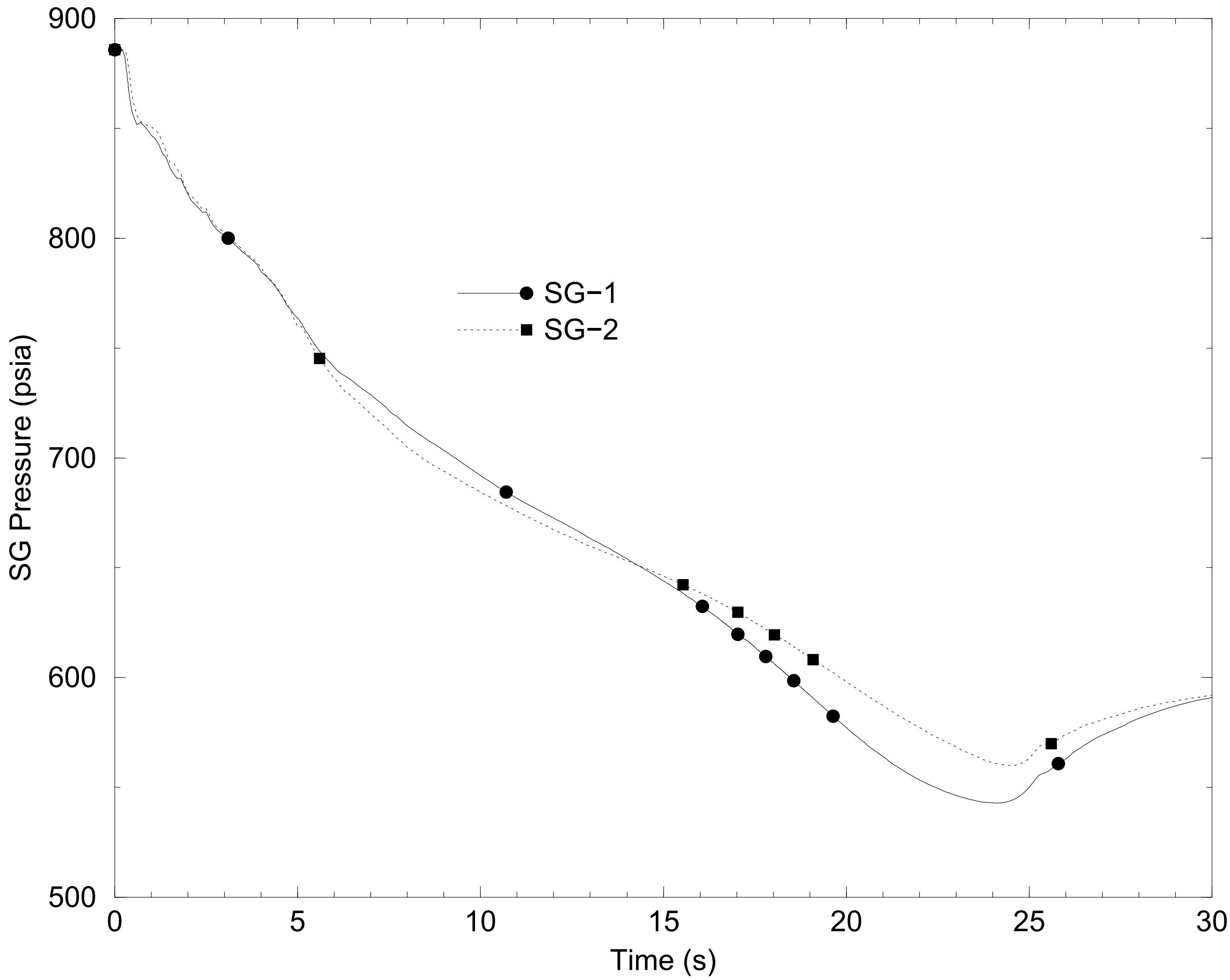
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT RCS TEMPERATURES VS TIME (HZP)	Figure 14.4-9 Revision 44



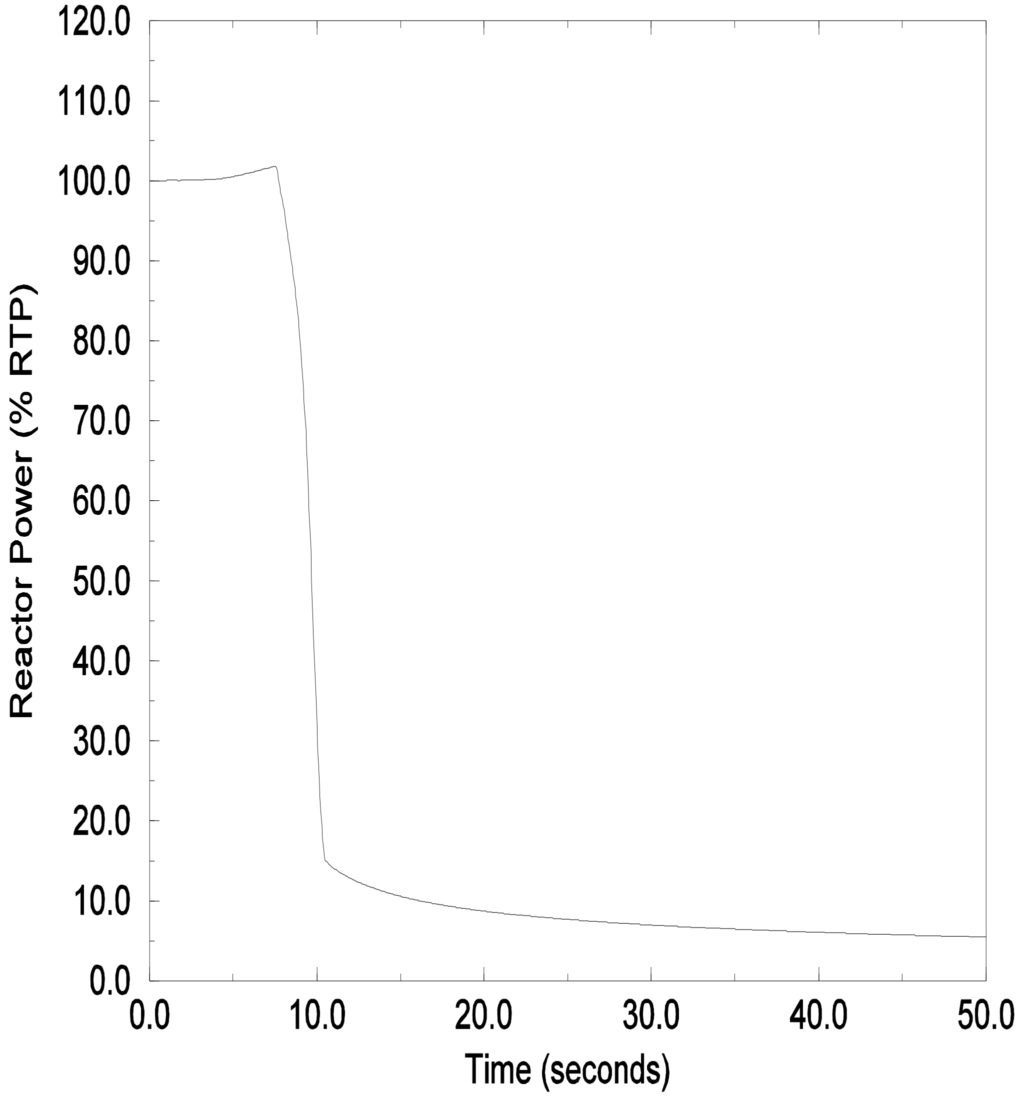
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT PRESSURIZER PRESSURE VS TIME (HZP)	Figure 14.4-10 Revision 44



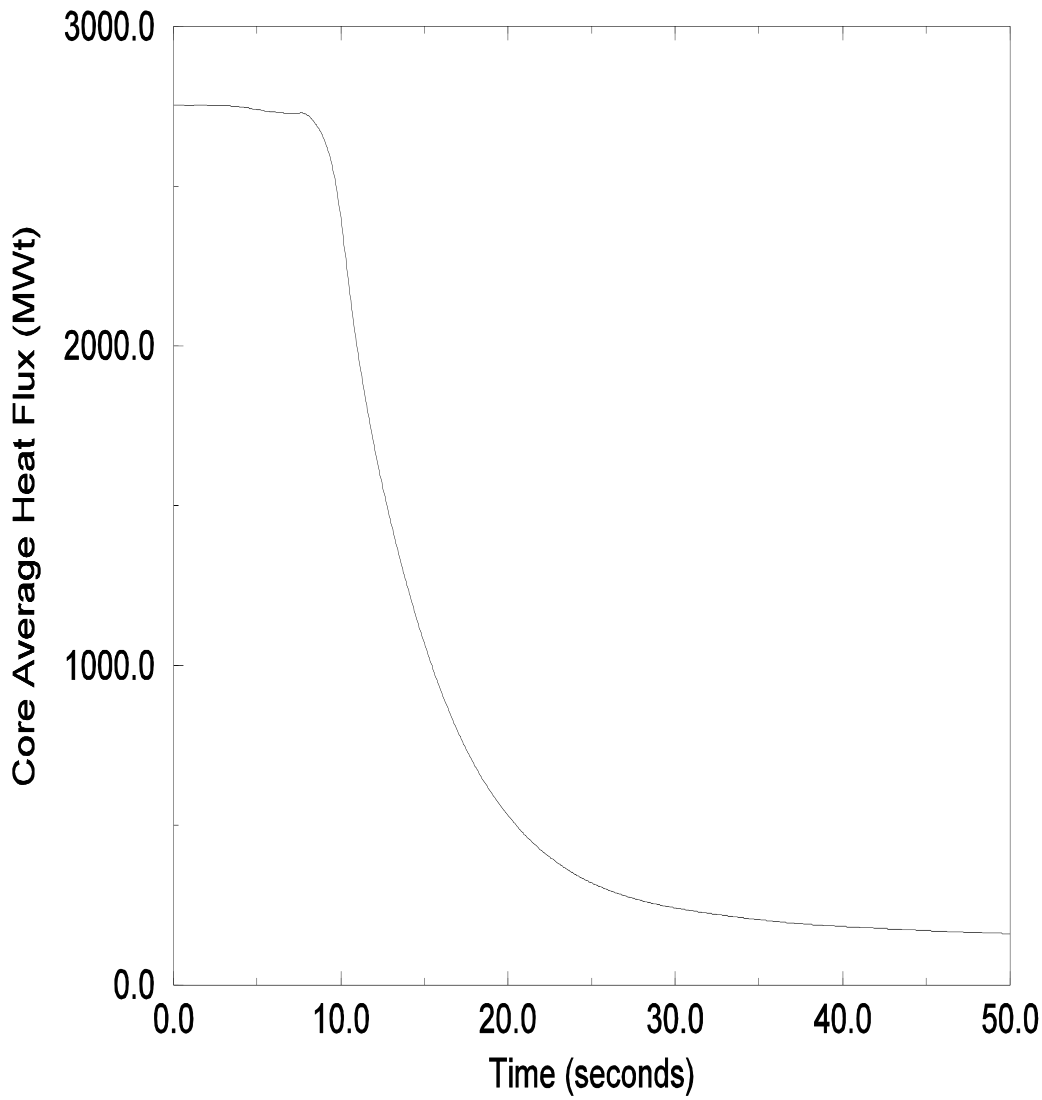
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTIVITIES VS TIME (HZP)	Figure 14.4-11 Revision 44



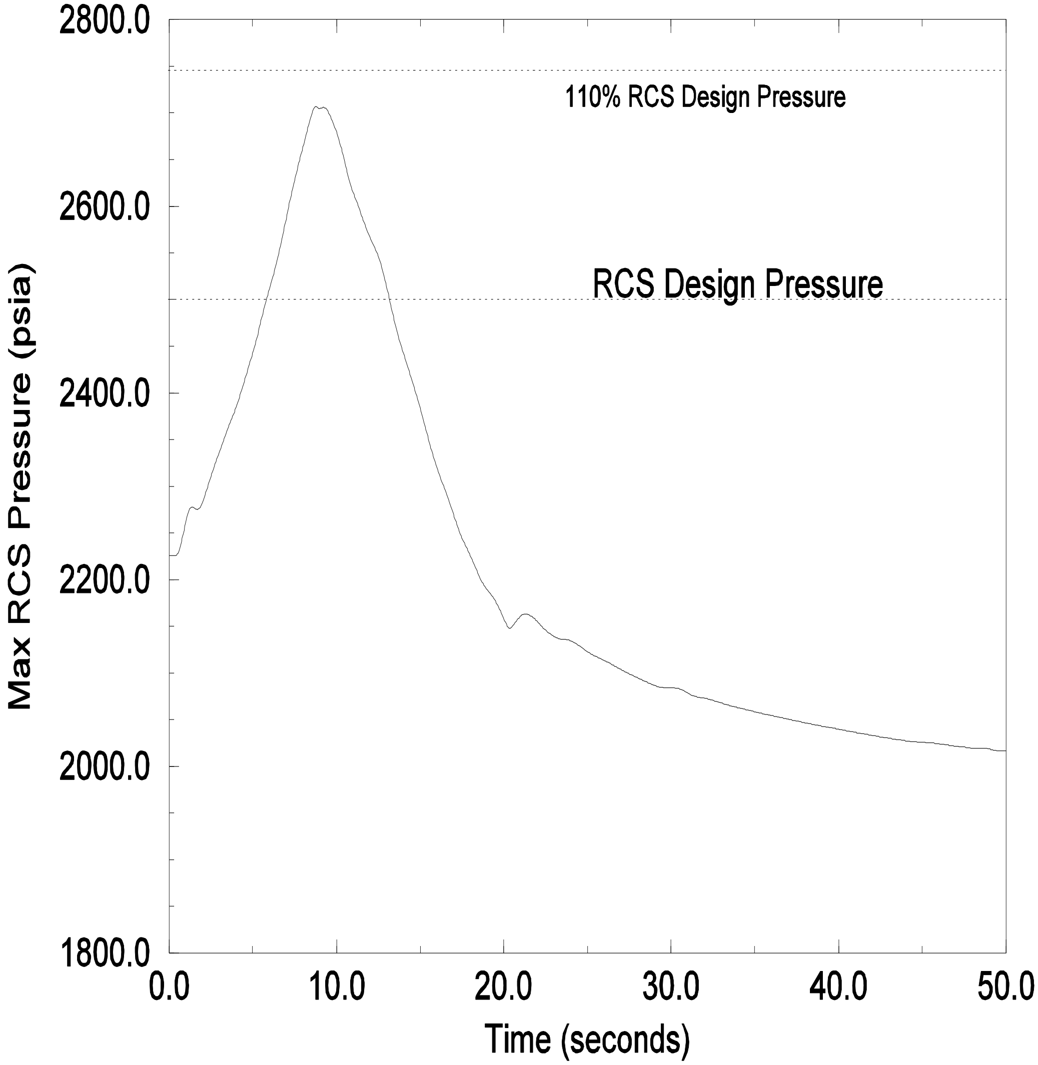
Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT SG PRESSURES VS TIME (HZP)	Figure 14.4-12 Revision 44



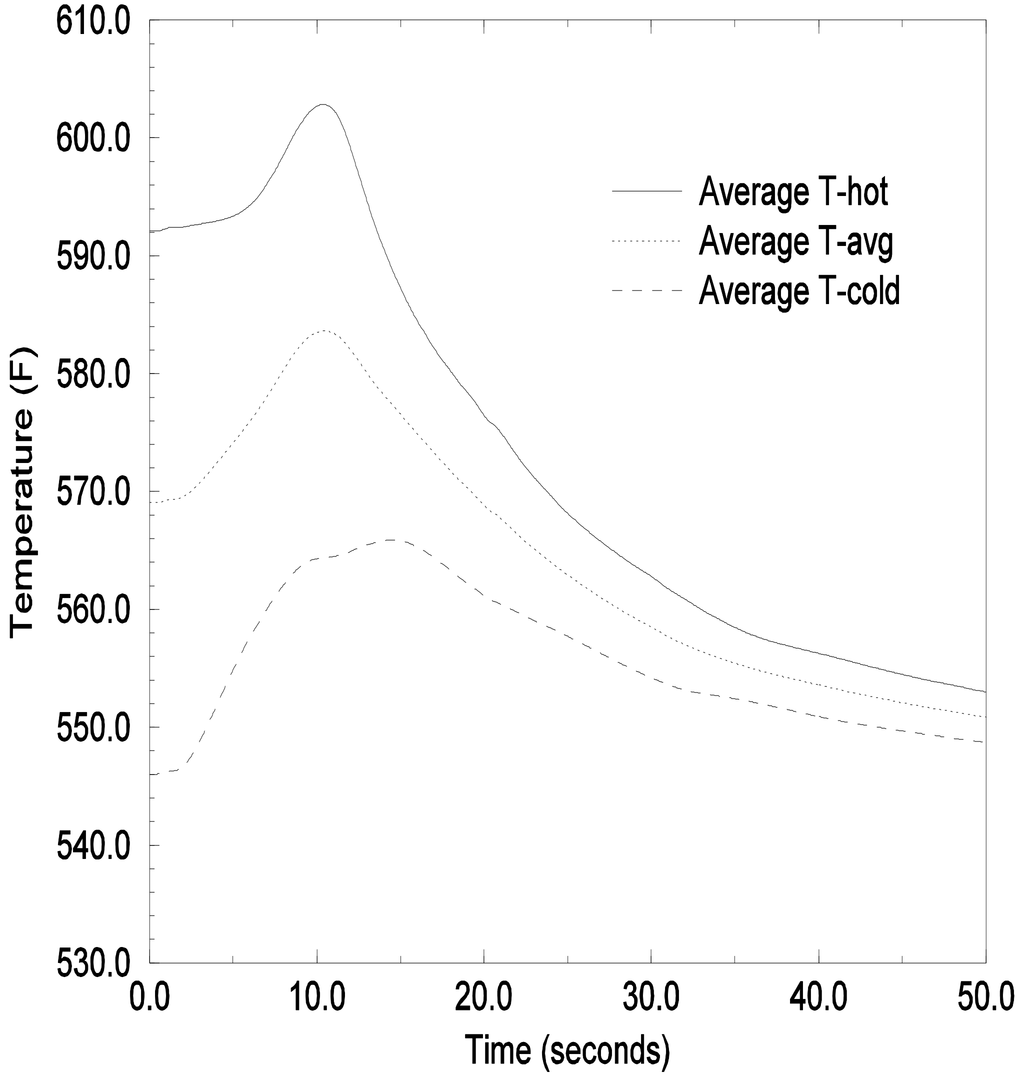
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE POWER VS TIME	Figure 14.5-1 Revision 49



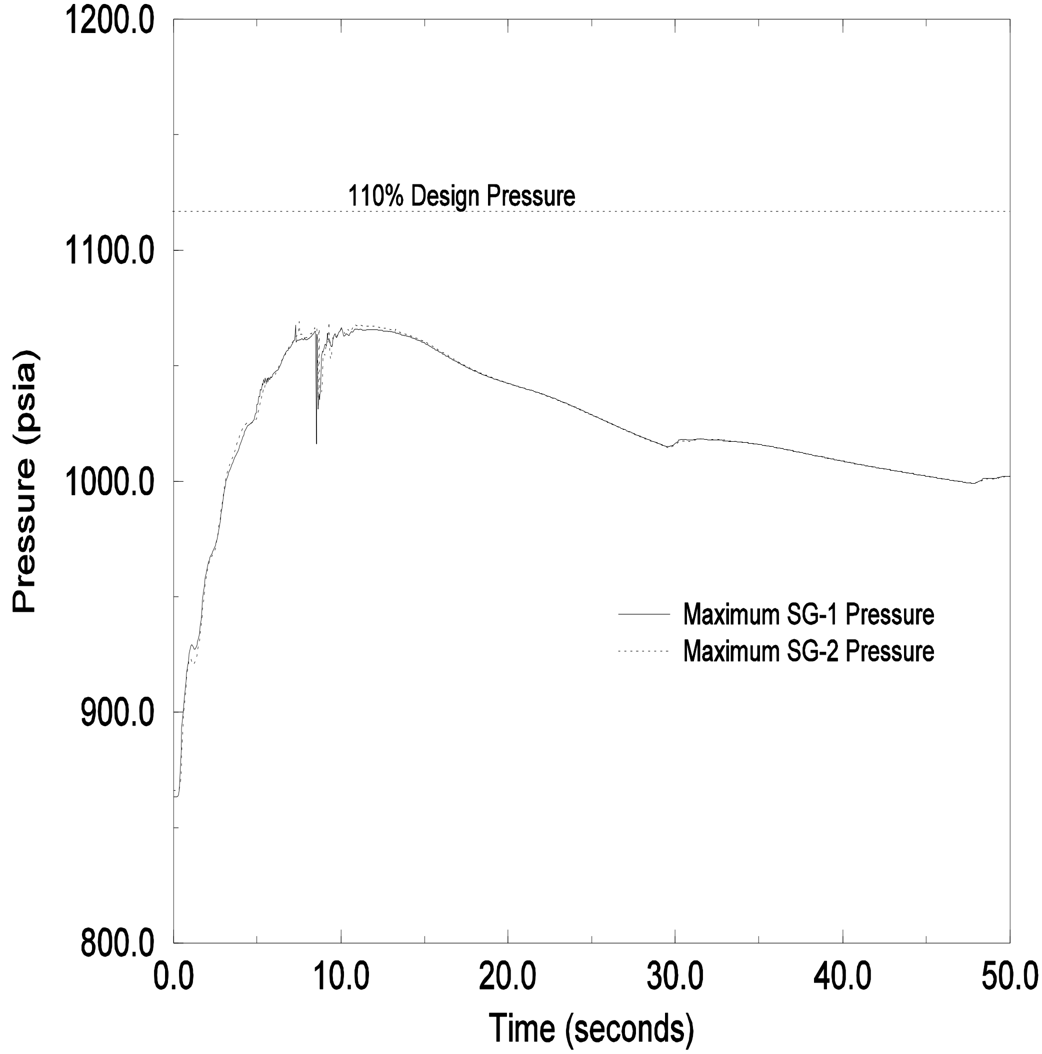
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE AVERAGE HEAT FLUX VS TIME	Figure 14.5-2 Revision 49



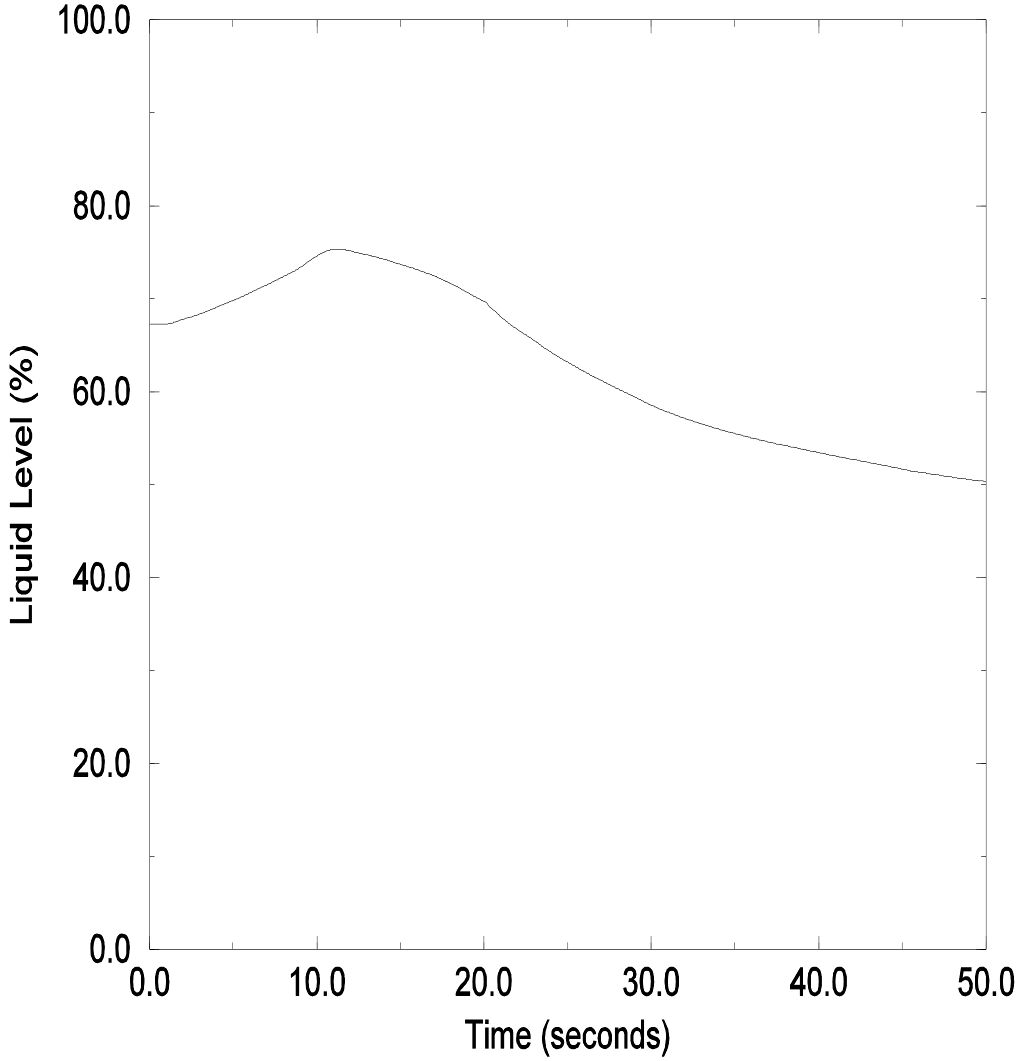
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT RCS PRESSURE VS TIME	Figure 14.5-3 Revision 49



Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT RCS TEMPERATURES VS TIME	Figure 14.5-4 Revision 49



Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT STEAM GENERATOR PRESSURE VS TIME	Figure 14.5-5 Revision 49



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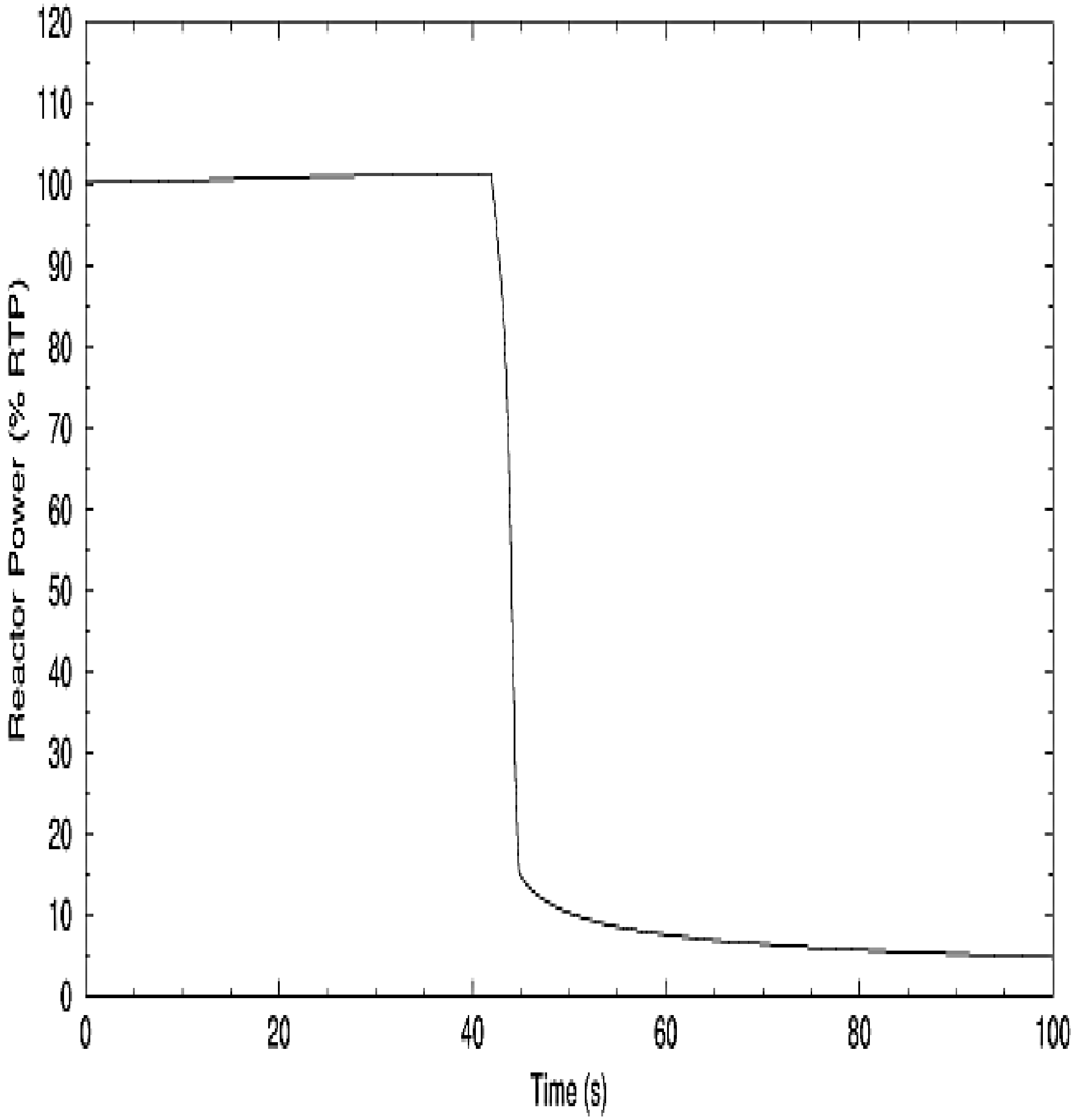
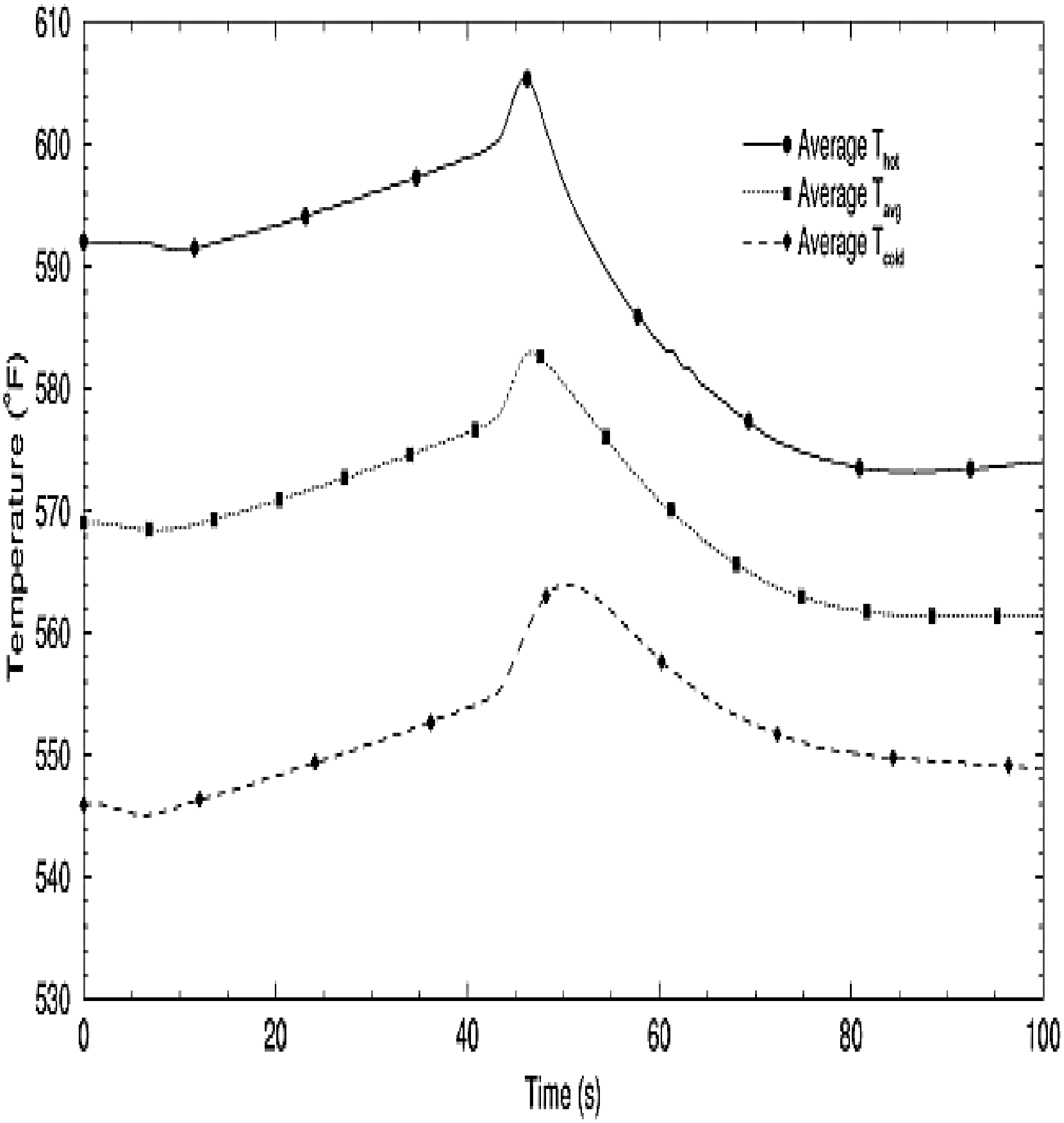
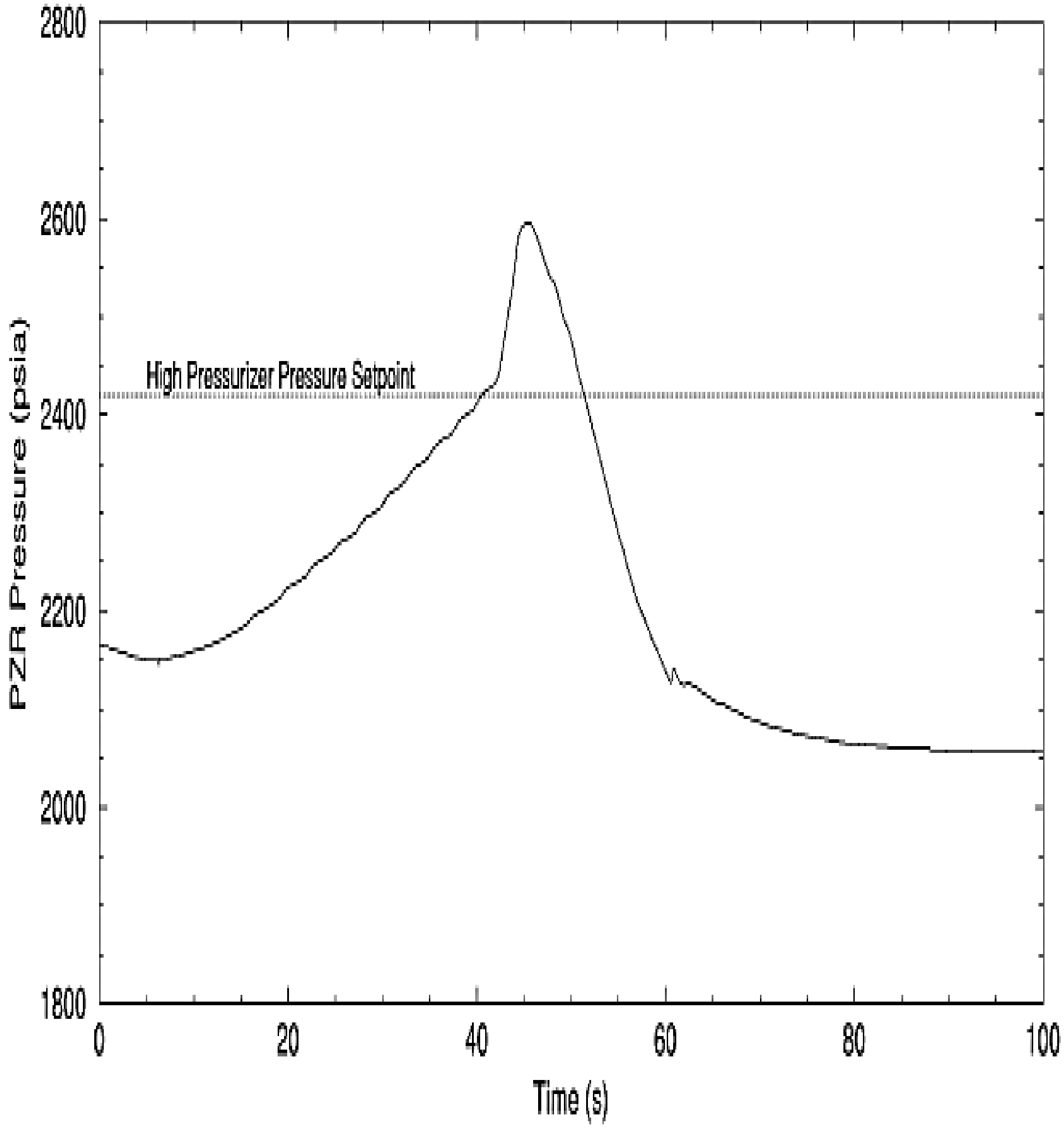


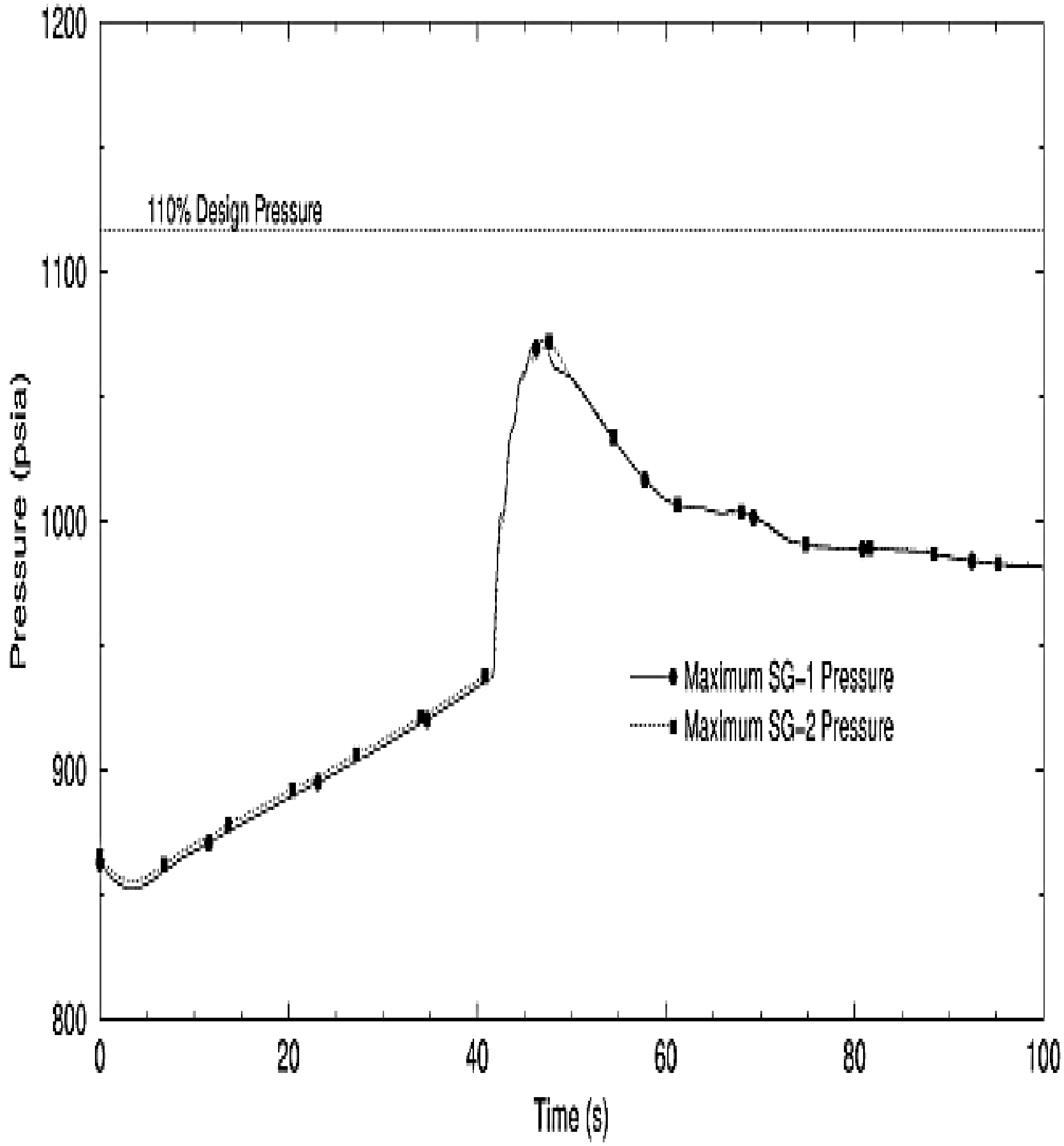
Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM RCS PEAK PRESSURE CORE POWER VS TIME	Figure 14.6-1 Revision 49



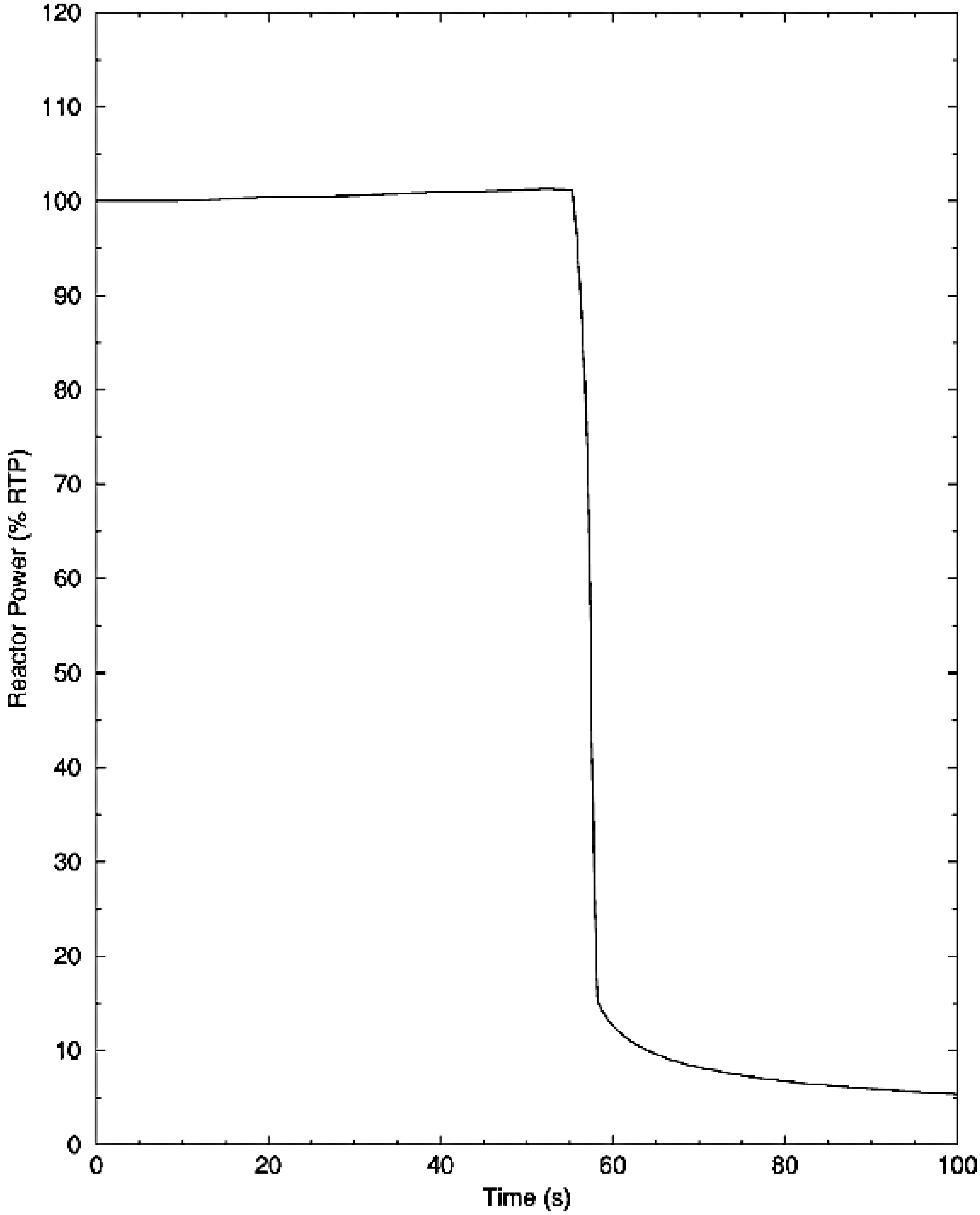
Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM RCS PEAK PRESSURE RCS TEMPERATURES VS TIME	Figure 14.6-2 Revision 49



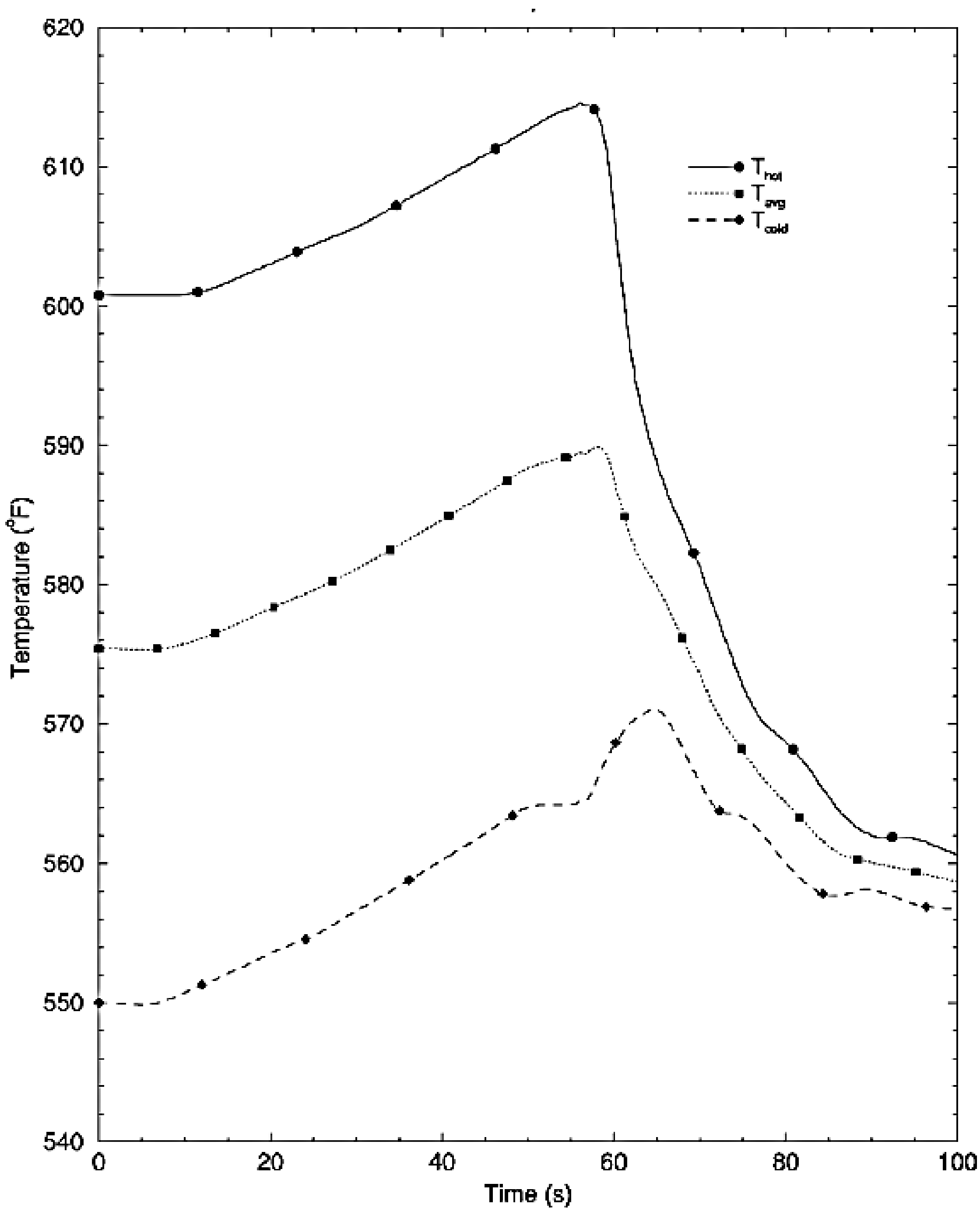
Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM RCS PEAK PRESSURE RCS PRESSURE VS TIME	Figure 14.6-3 Revision 49



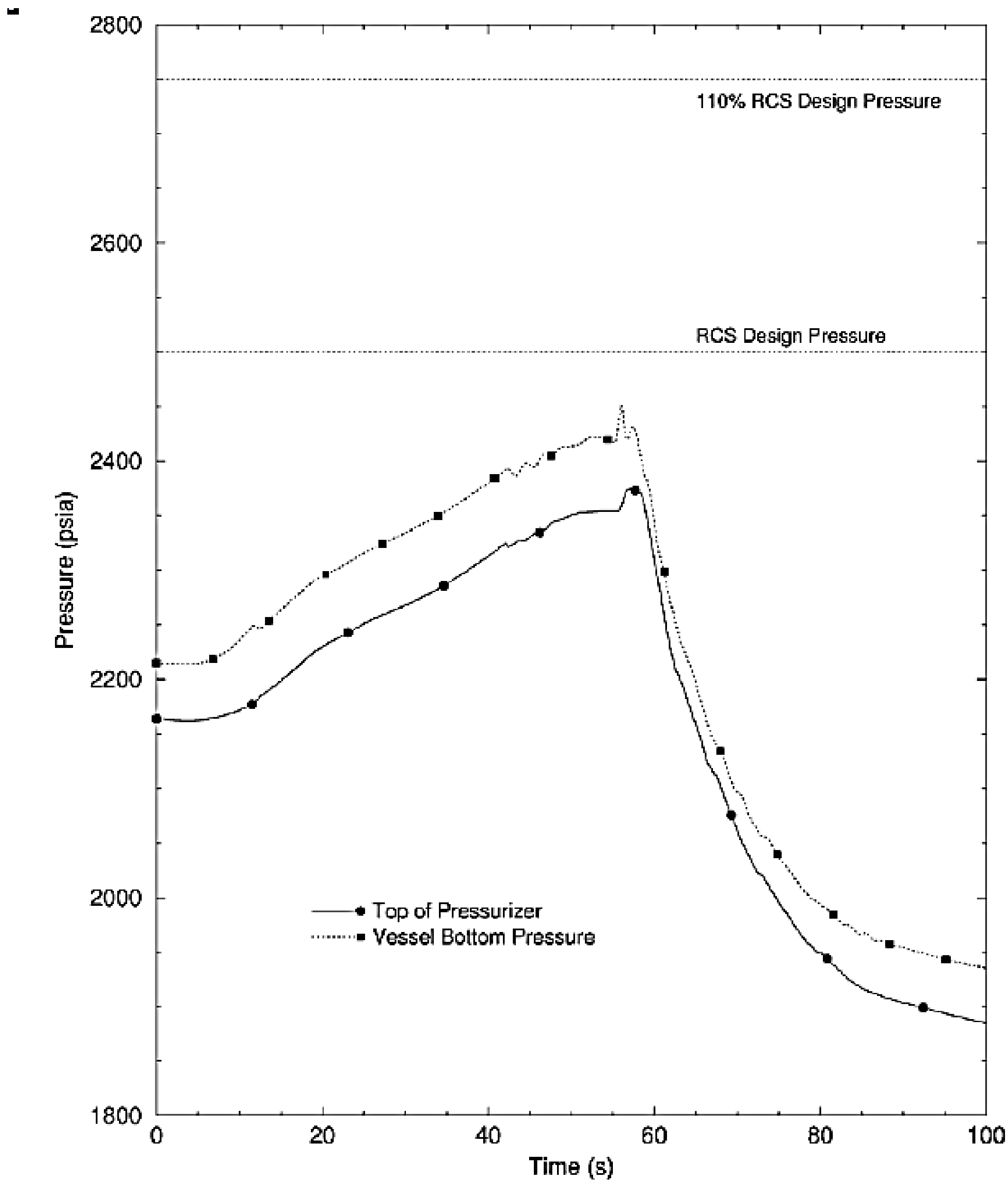
Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM RCS PEAK PRESSURE STEAM GENERATOR PRESSURE VS TIME	Figure 14.6-4 Revision 49



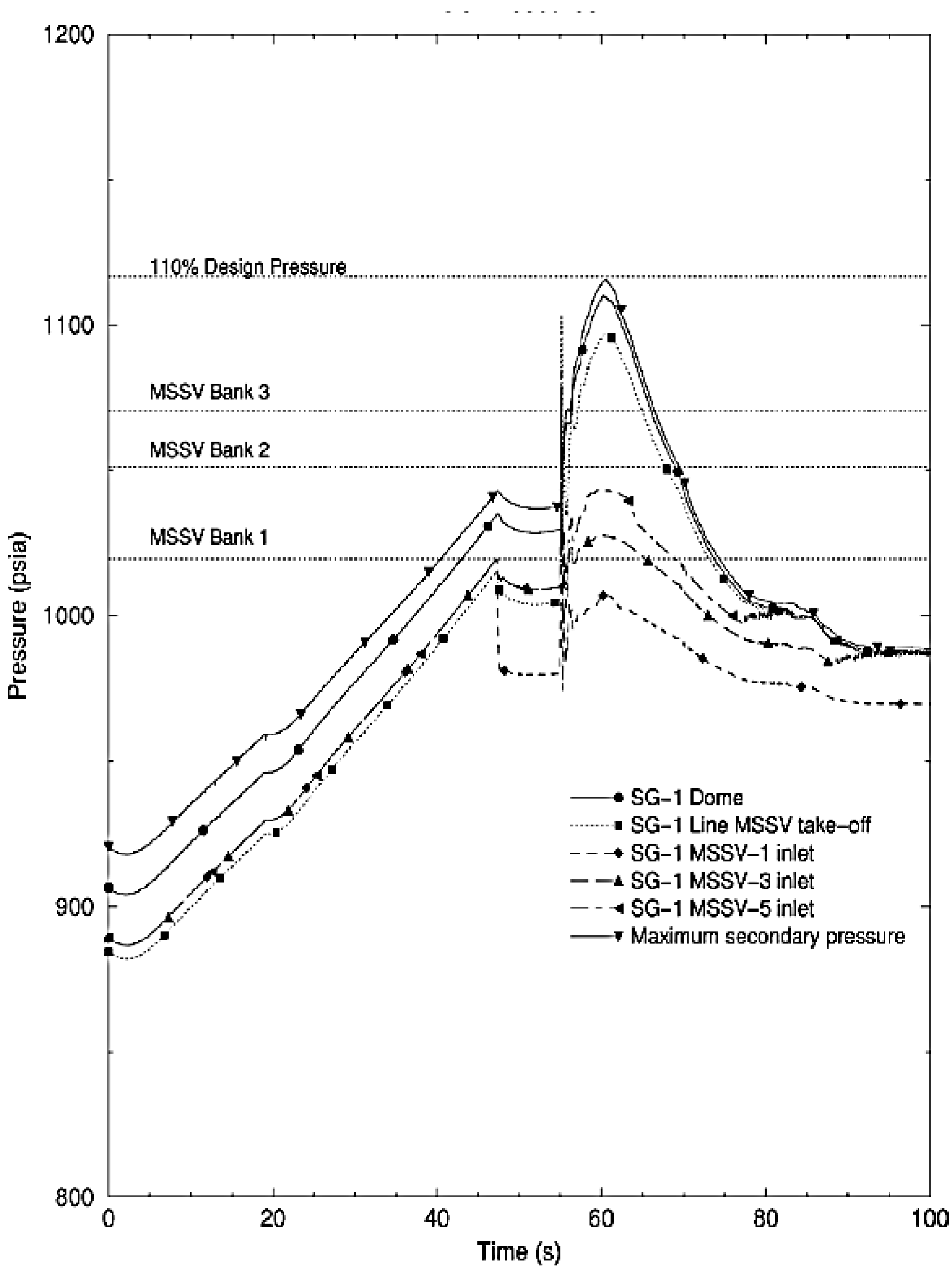
Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM SECONDARY PEAK PRESSURE CORE POWER VS TIME	Figure 14.6-5 Revision 49



Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM SECONDARY PEAK PRESSURE RCS TEMPERATURES VS TIME	Figure 14.6-6 Revision 49



Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM SECONDARY PEAK PRESSURE PRESSURIZER PRESSURE VS TIME	Figure 14.6-7 Revision 49



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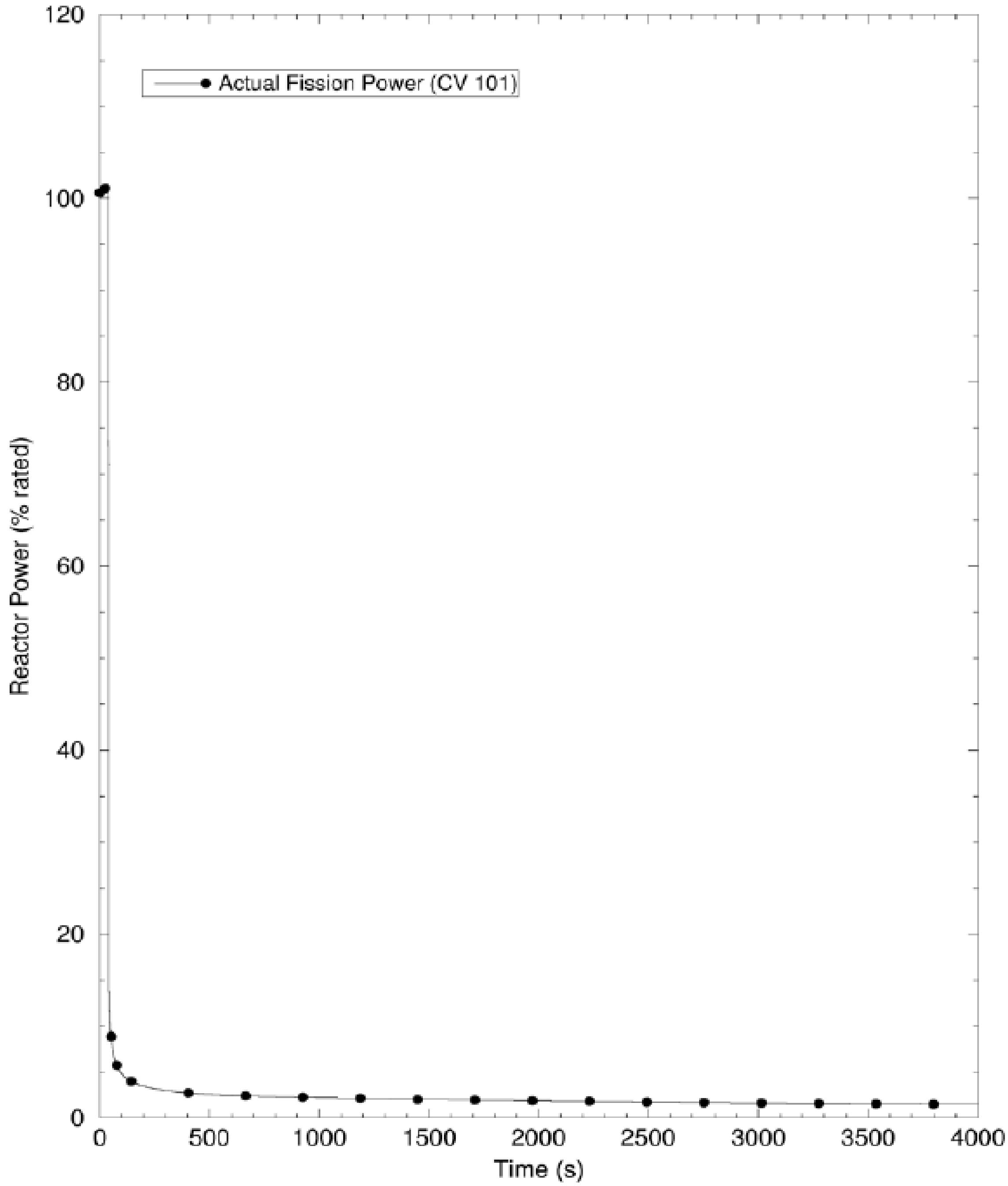
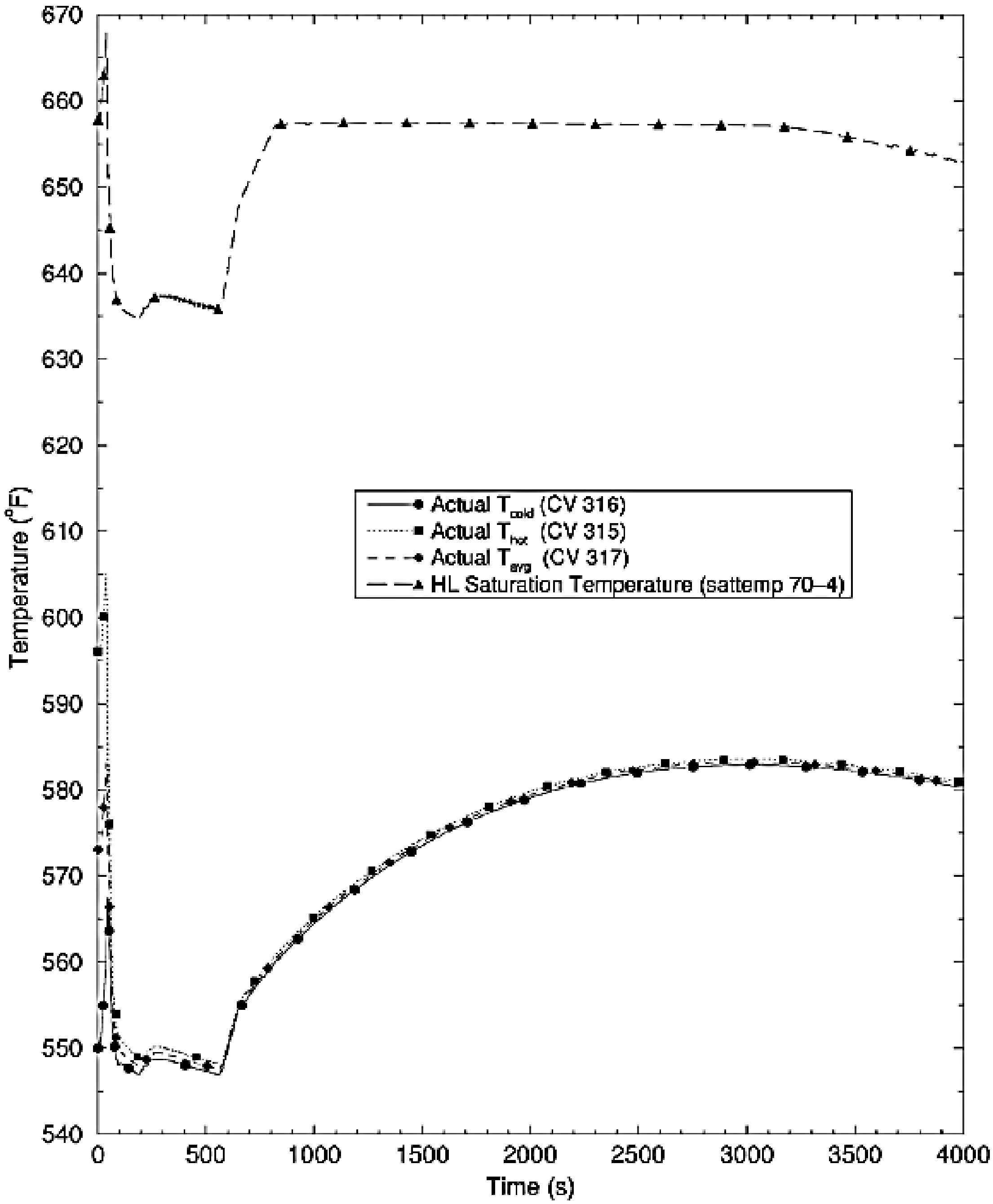


Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM STEAM GENERATOR INVENTORY DEPLETION CORE POWER VS TIME	Figure 14.6-9 Revision 49



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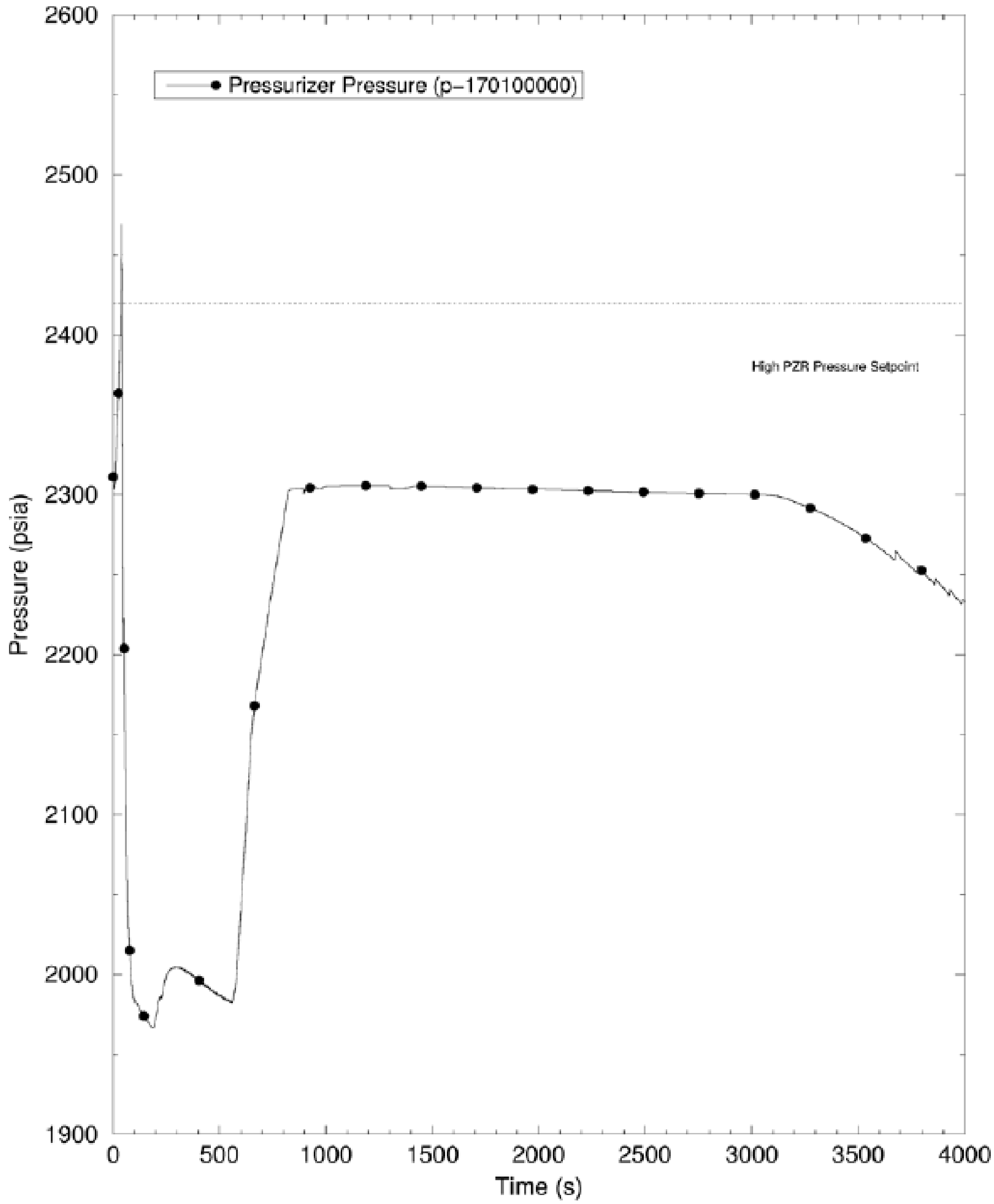
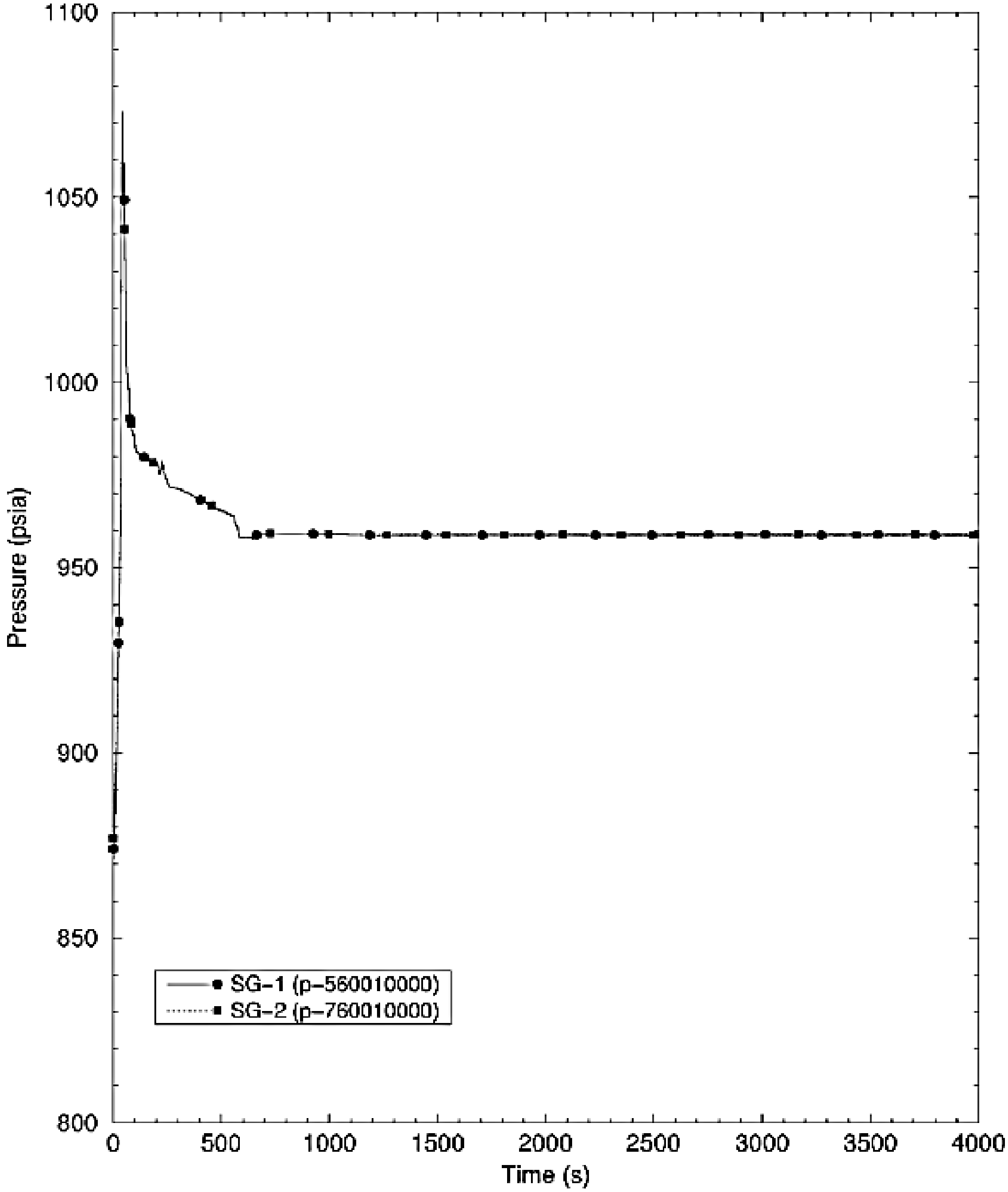
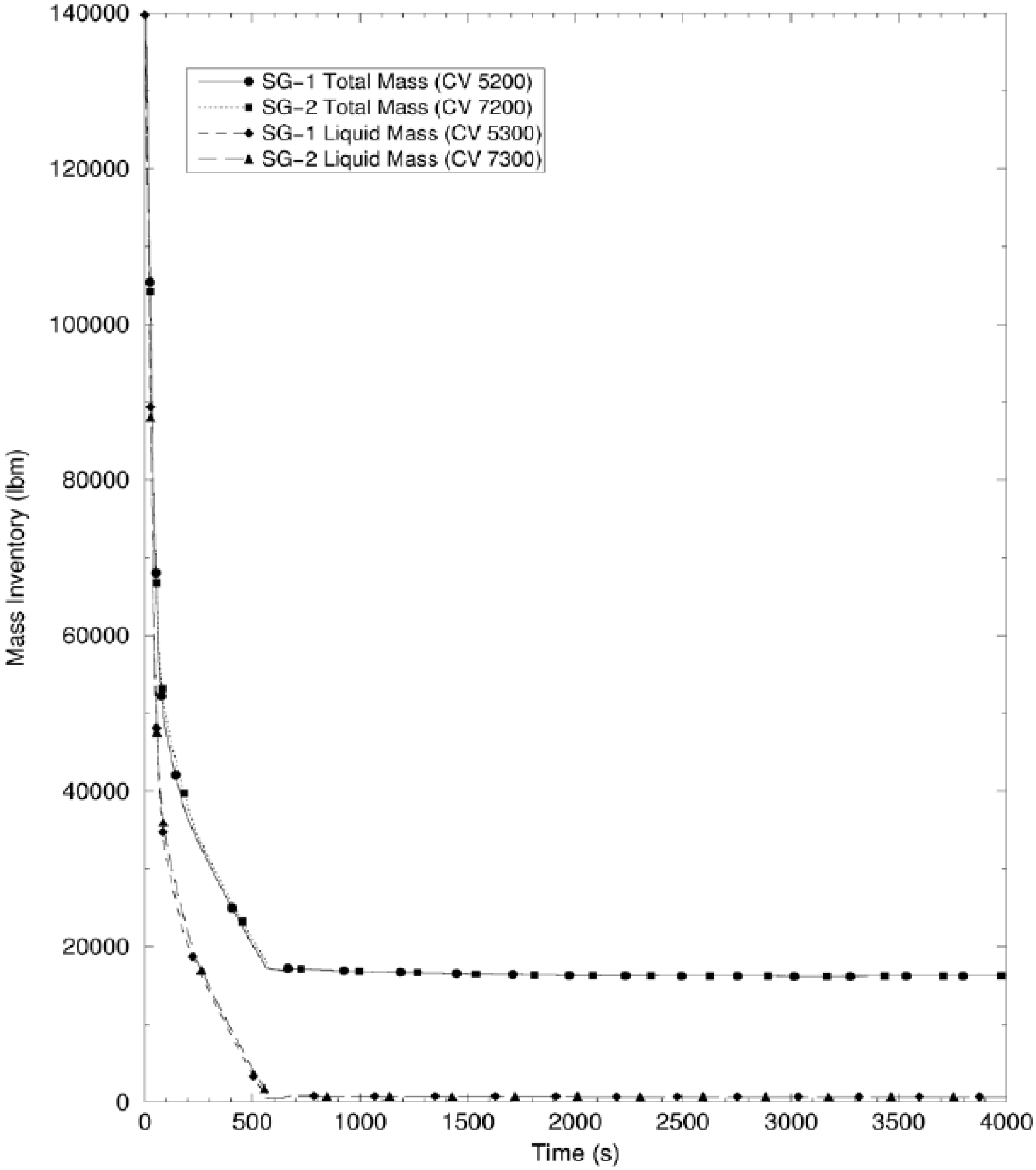


Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM STEAM GENERATOR INVENTORY DEPLETION PRESSURIZER PRESSURE VS TIME	Figure 14.6-11 Revision 49



Calvert Cliffs Nuclear Power Plant	LOSS OF FEEDWATER FLOW EVENT MAXIMUM STEAM GENERATOR INVENTORY DEPLETION STEAM GENERATOR PRESSURE VS TIME	Figure 14.6-12 Revision 49



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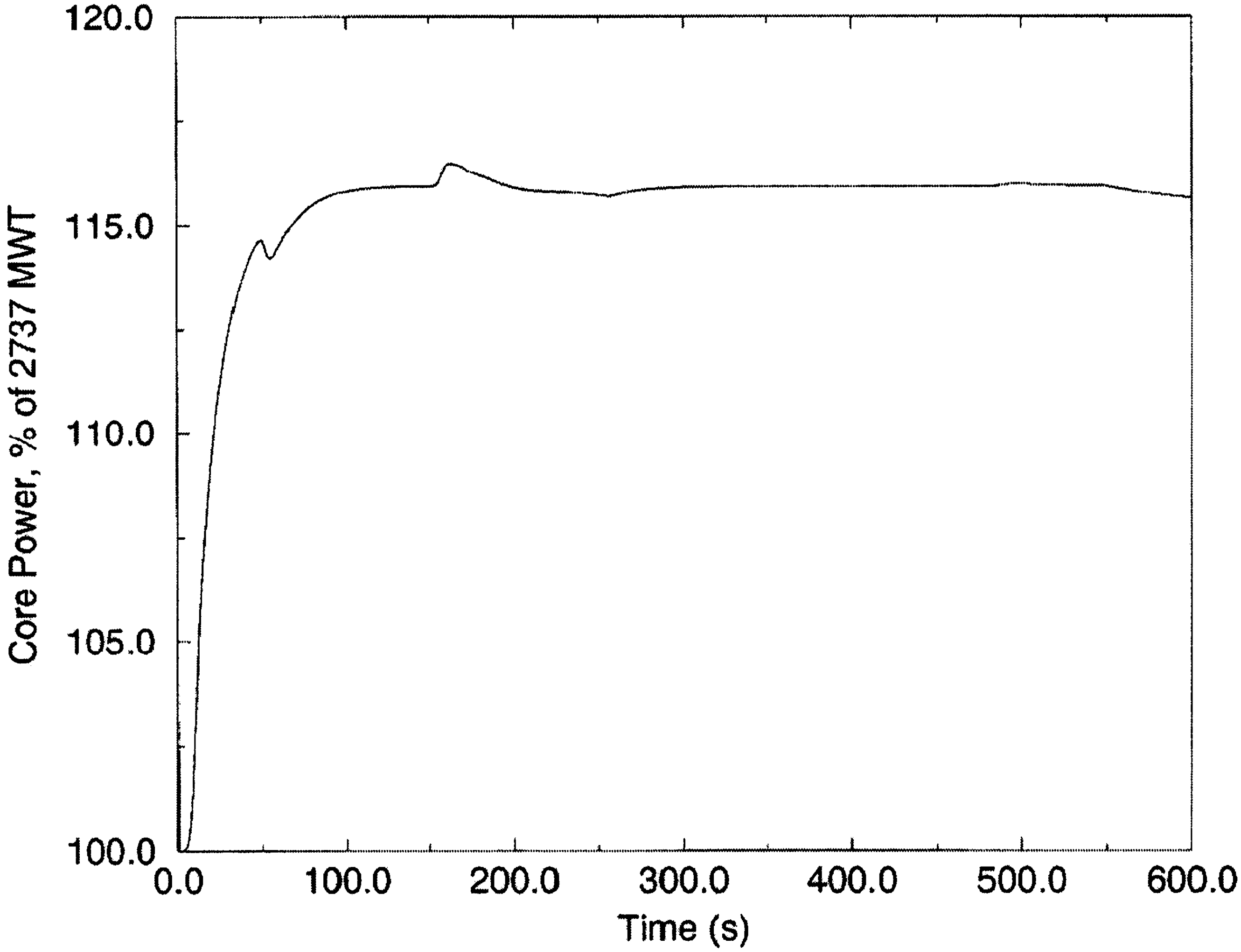
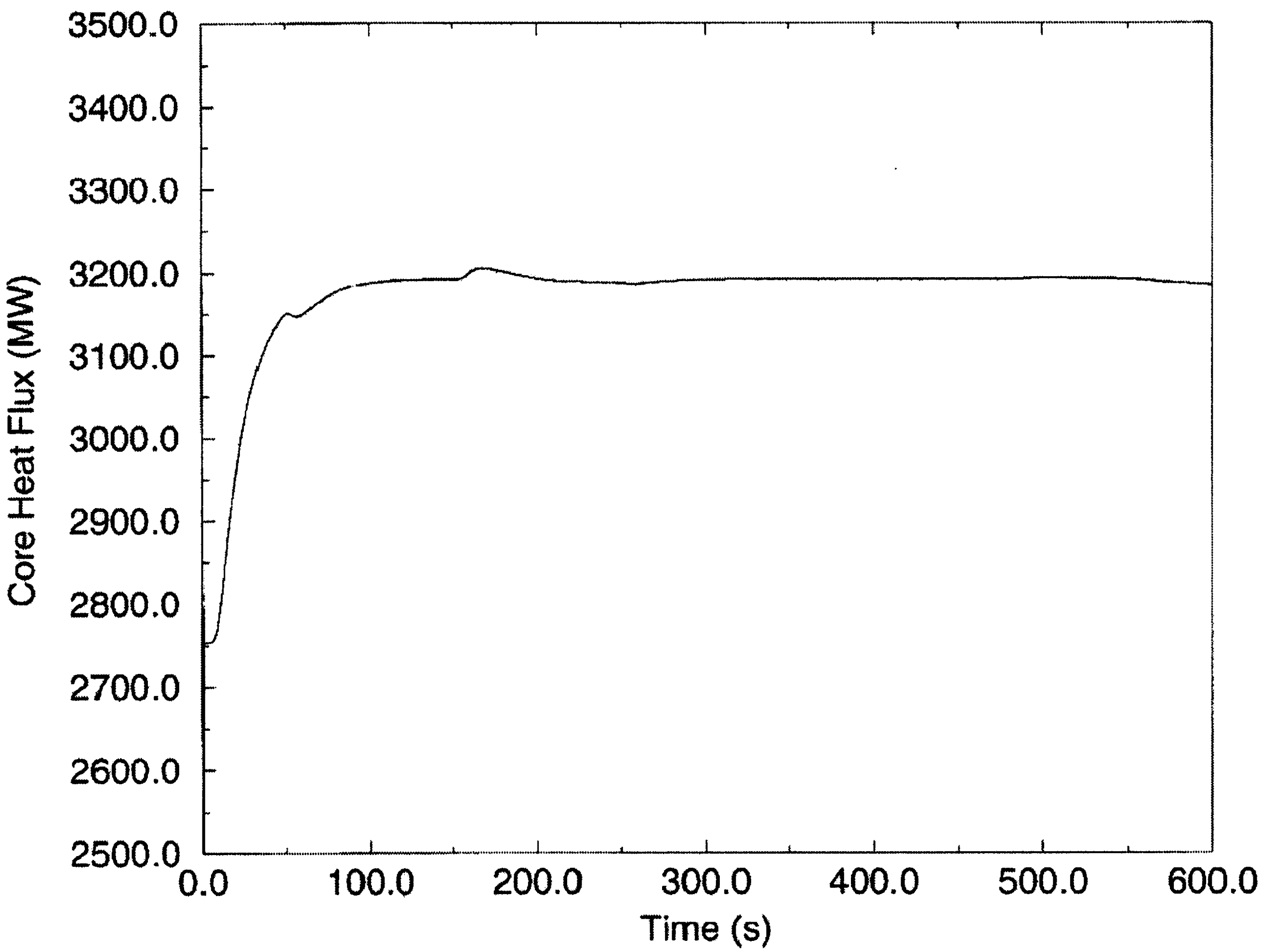


Calvert Cliffs Nuclear Power Plant	EXCESS FEEDWATER HEAT REMOVAL EVENT CORE POWER VERSUS TIME	Figure 14.7-1 Revision 44



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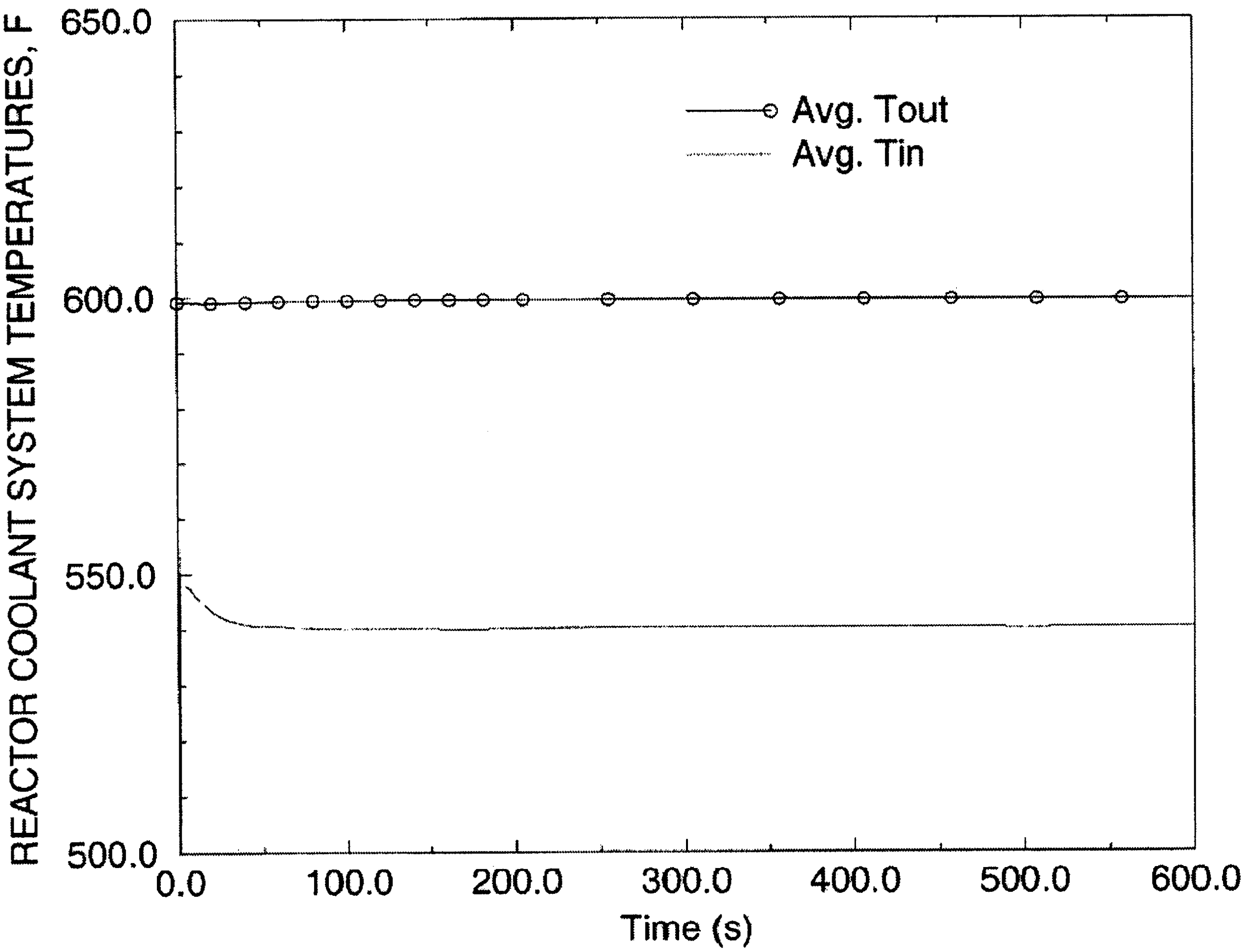
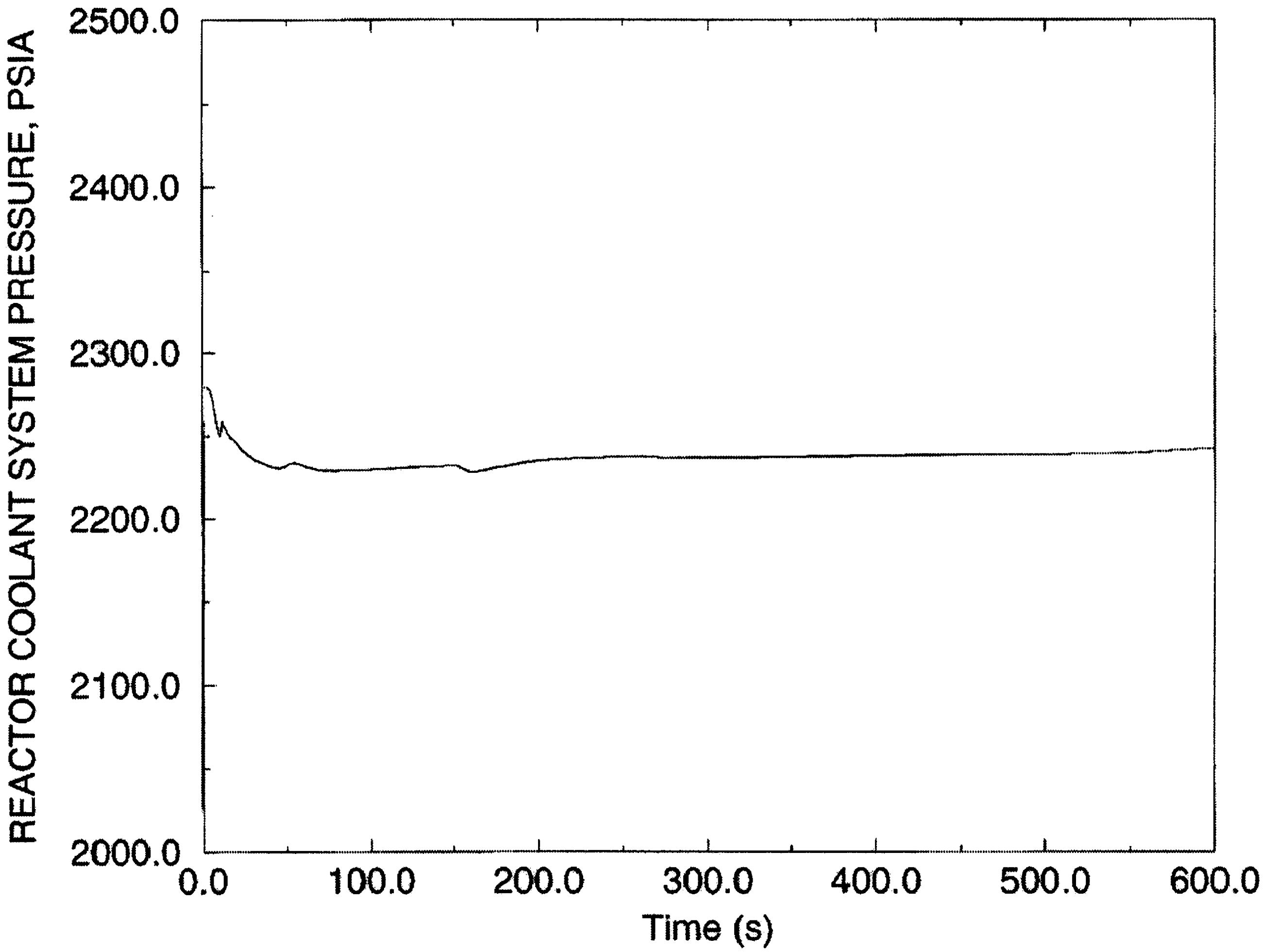


Calvert Cliffs Nuclear Power Plant	EXCESS FEEDWATER HEAT REMOVAL EVENT RCS TEMPERATURES VERSUS TIME	Figure 14.7-3 Revision 44



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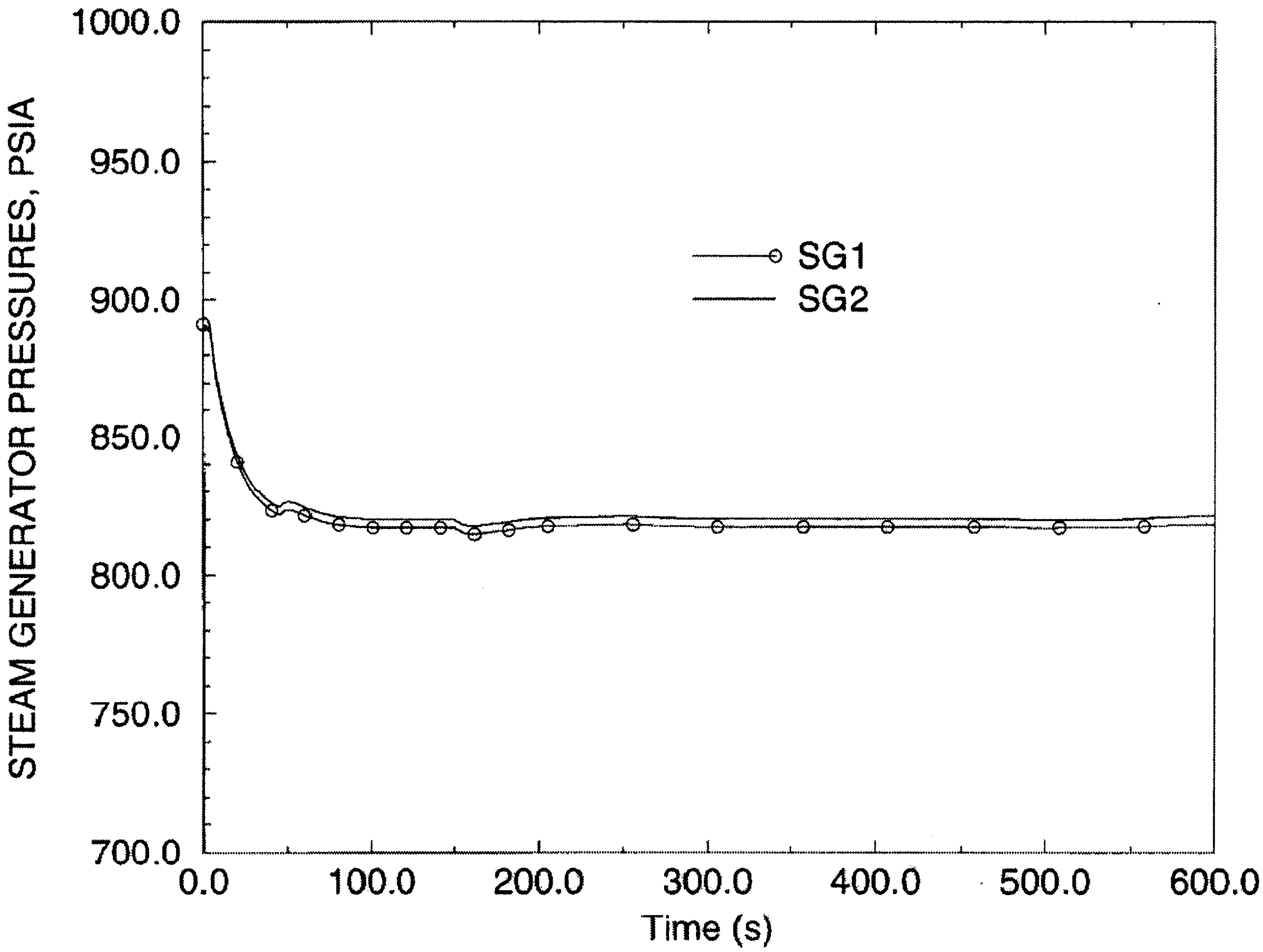
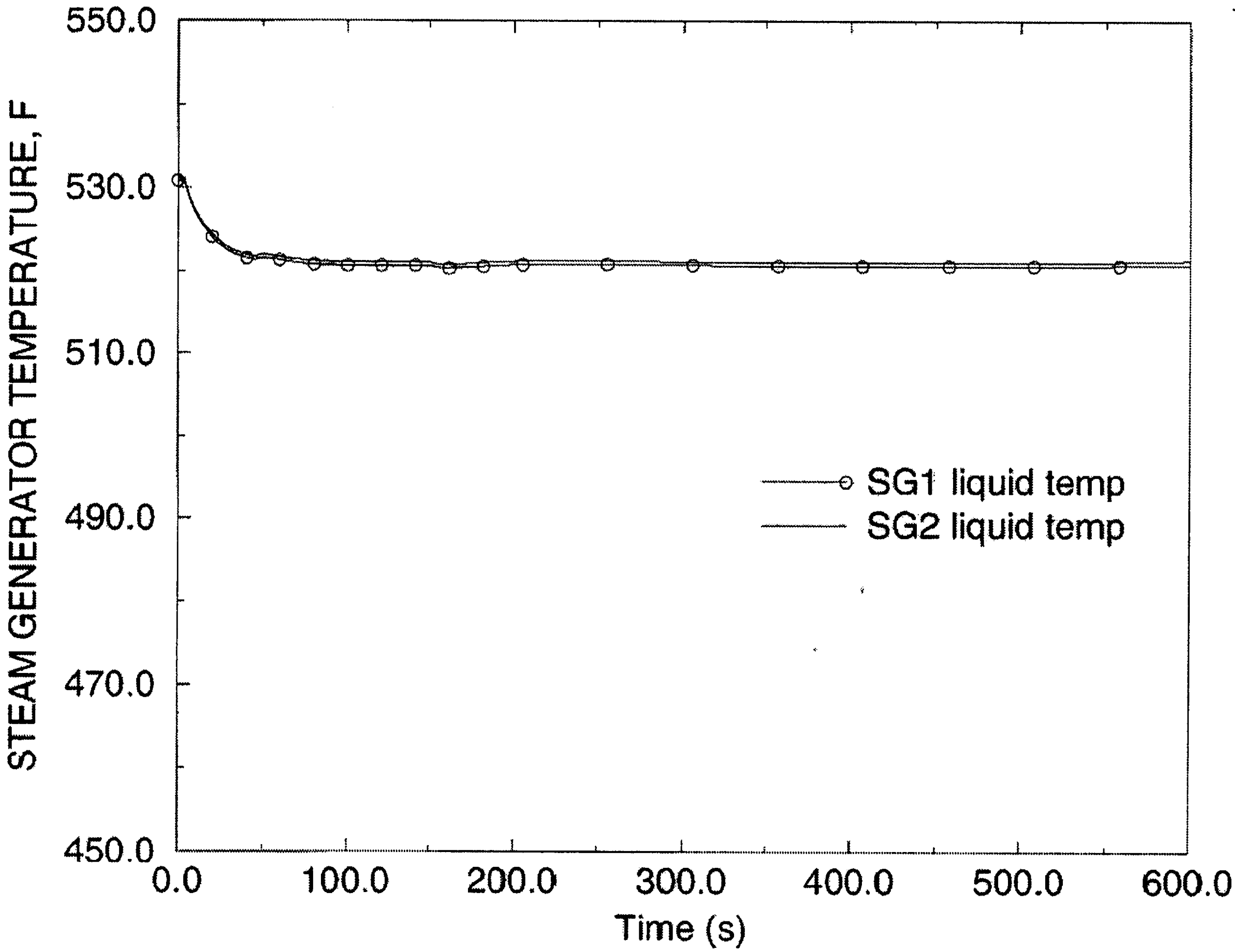
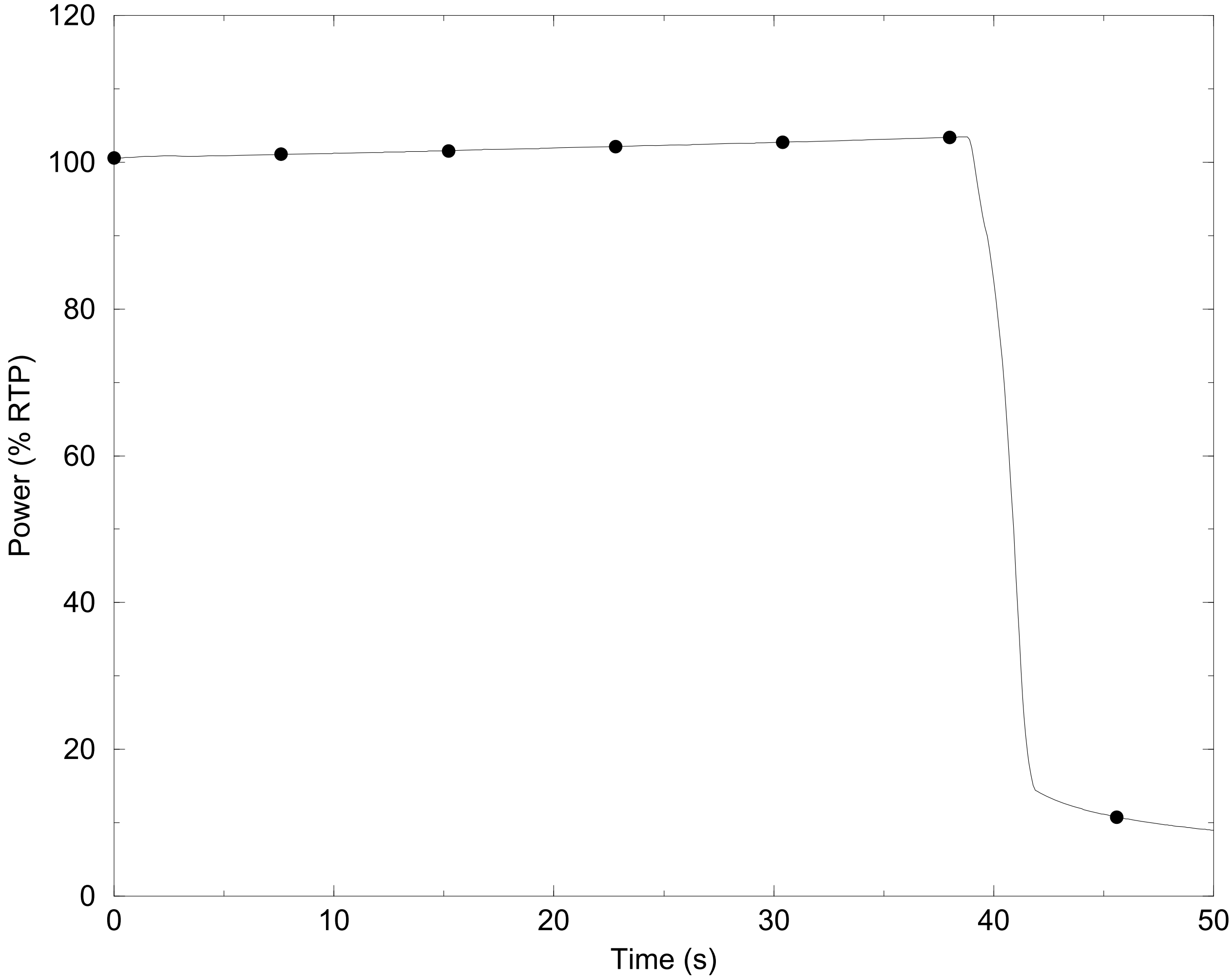


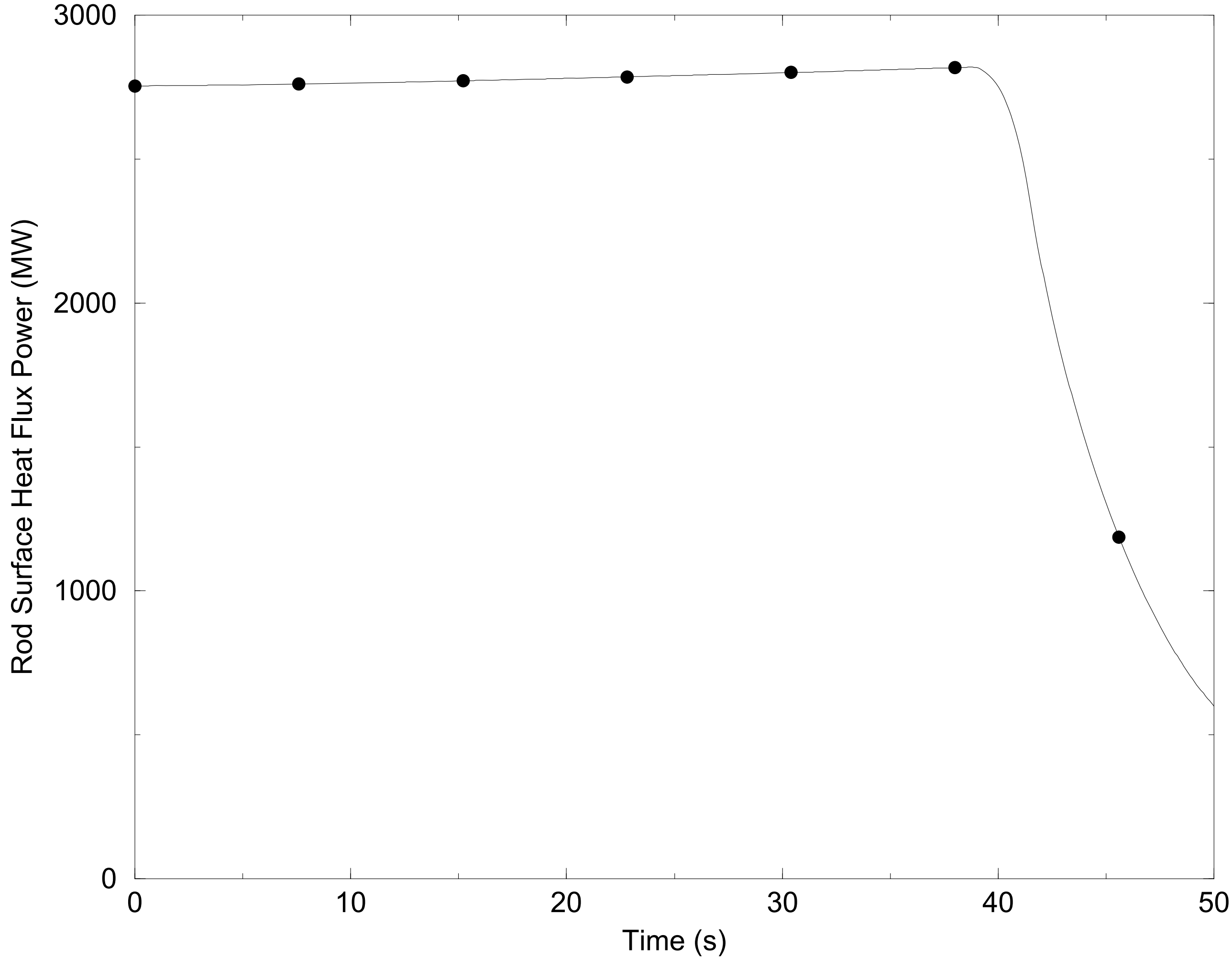
Calvert Cliffs Nuclear Power Plant	EXCESS FEEDWATER HEAT REMOVAL EVENT STEAM GENERATOR PRESSURES VERSUS TIME	Figure 14.7-5 Revision 44



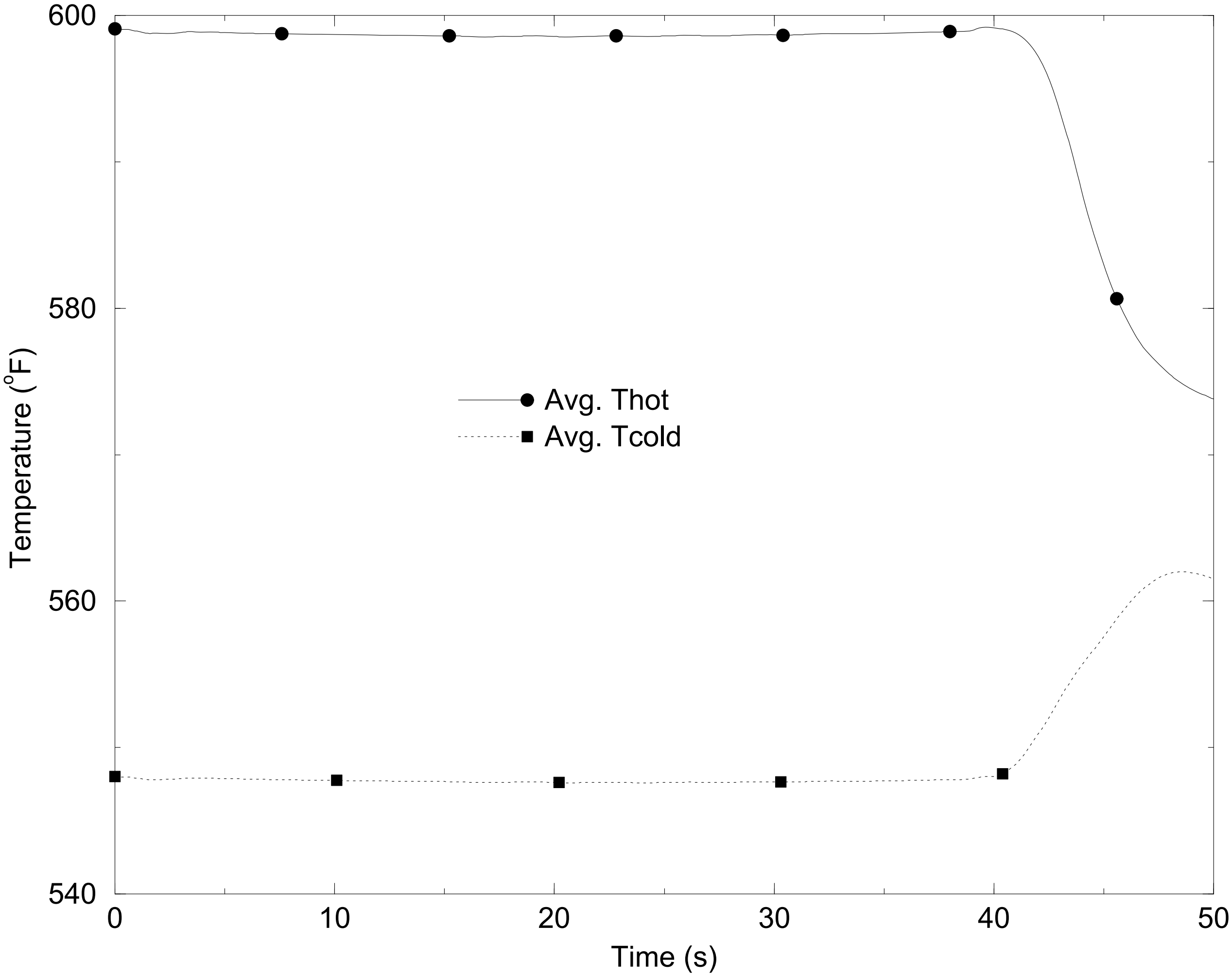
Calvert Cliffs Nuclear Power Plant	EXCESS FEEDWATER HEAT REMOVAL EVENT STEAM GENERATOR TEMPERATURE VERSUS TIME	Figure 14.7-6 Revision 44



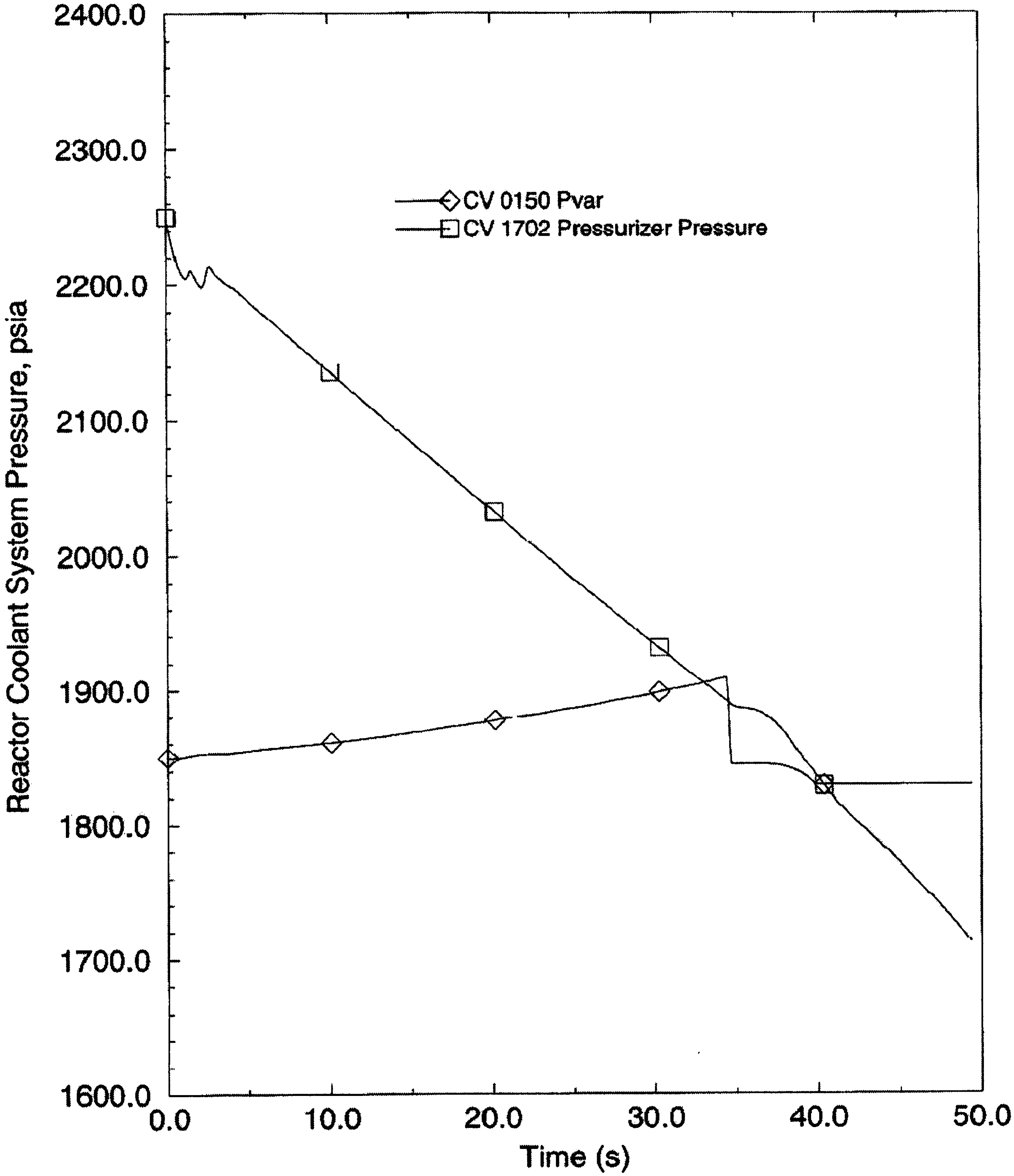
Calvert Cliffs Nuclear Power Plant	RCS DEPRESSURIZATION EVENT CORE POWER VS TIME	Figure 14.8-1 Revision 44



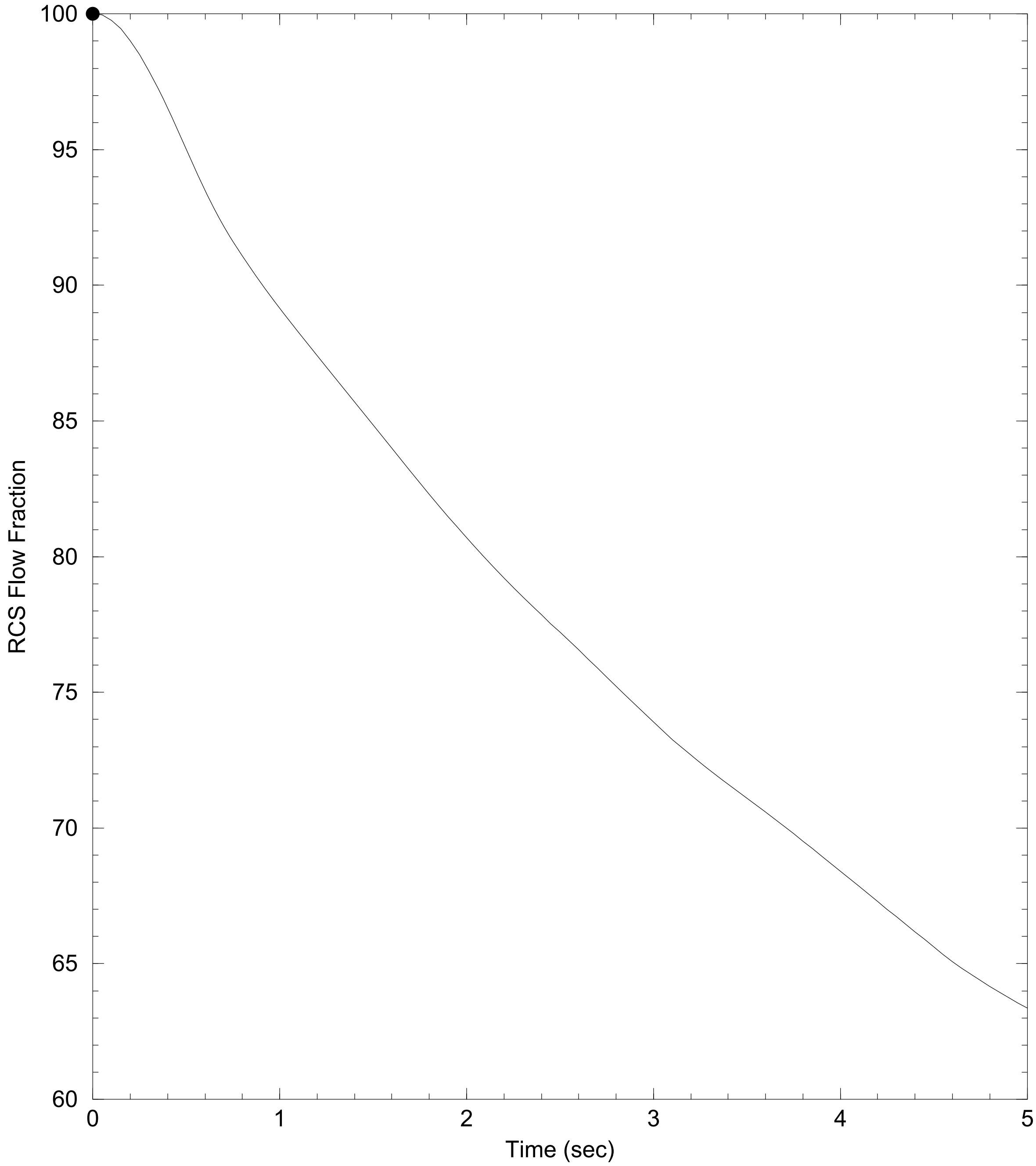
Calvert Cliffs Nuclear Power Plant	RCS DEPRESSURIZATION EVENT CORE AVERAGE HEAT FLUX VS TIME	Figure 14.8-2 Revision 44



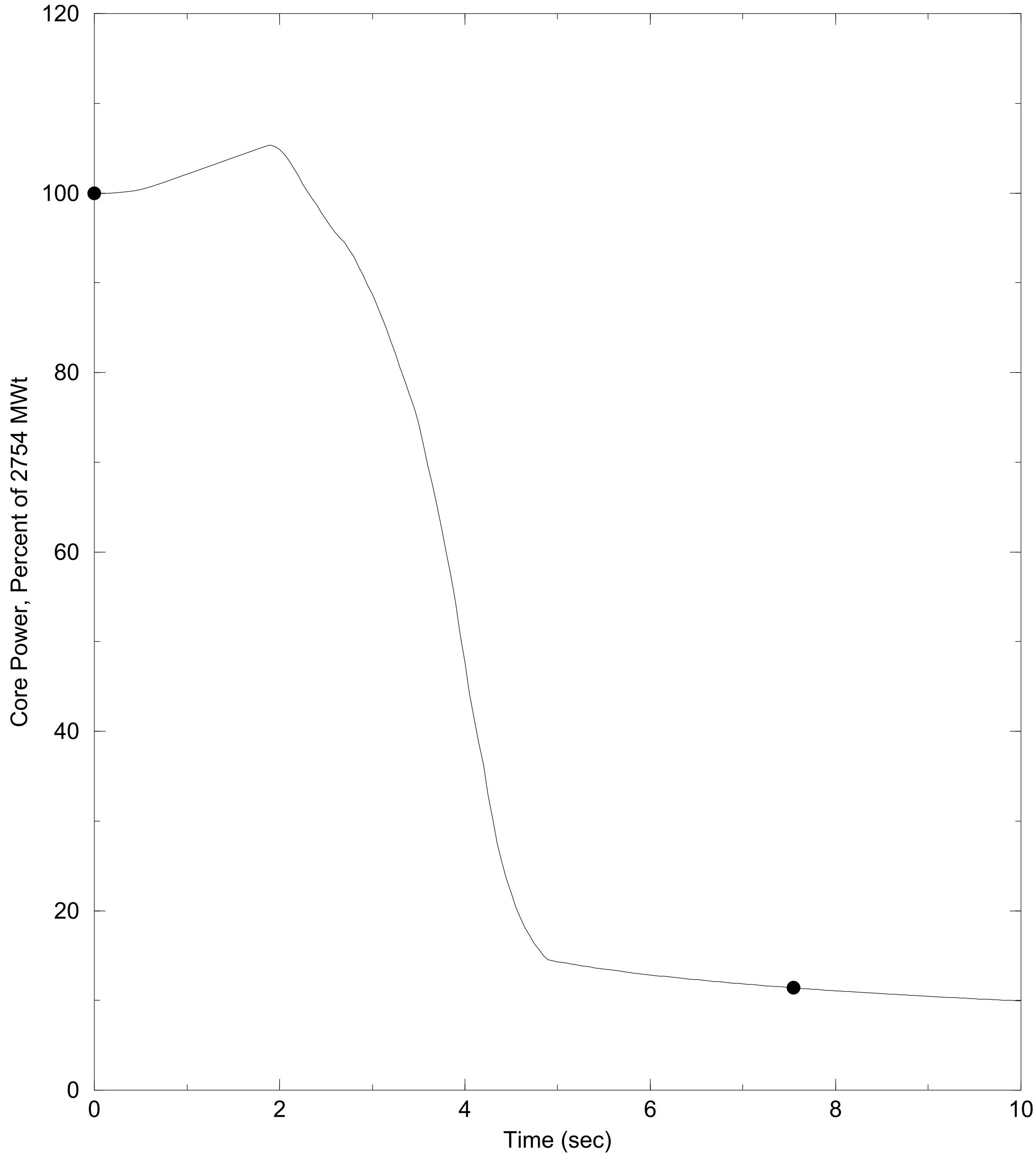
Calvert Cliffs Nuclear Power Plant	RCS DEPRESSURIZATION EVENT RCS TEMPERATURES VS TIME	Figure 14.8-3 Revision 44



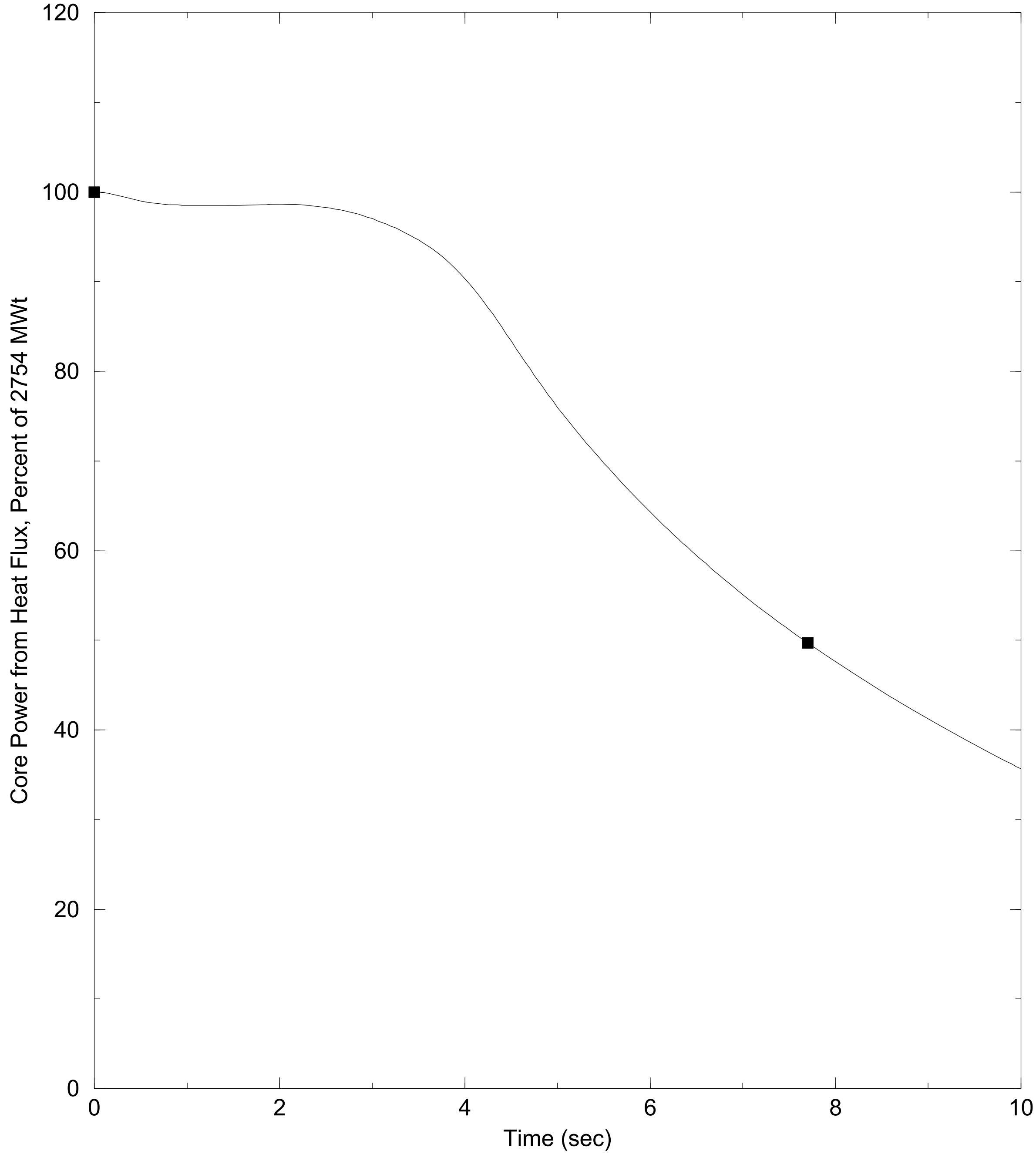
Calvert Cliffs Nuclear Power Plant	RCS DEPRESSURIZATION EVENT RCS PRESSURE VS TIME	Figure 14.8-4 Revision 44



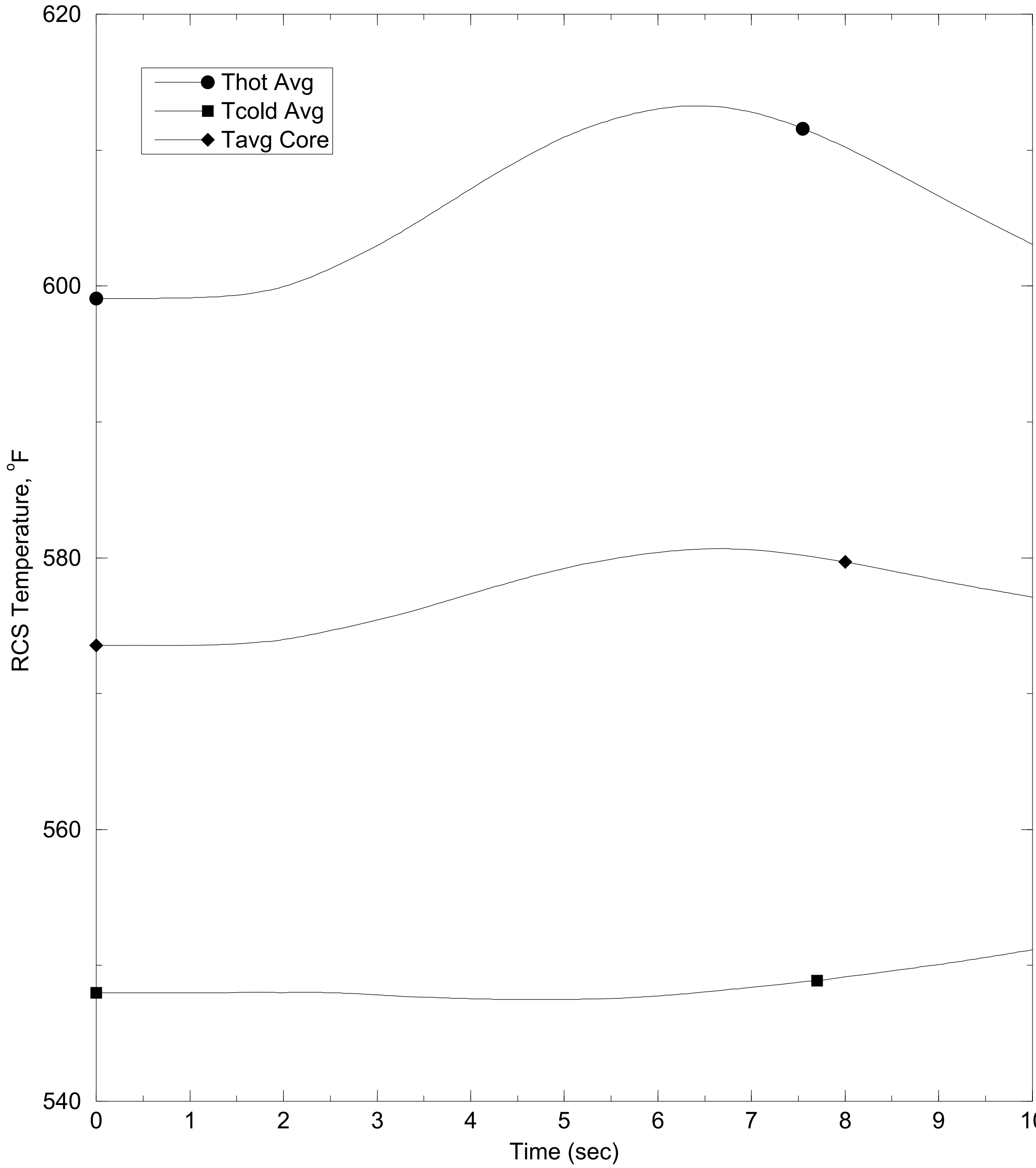
Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE FLOW FRACTION VS TIME	Figure 14.9-1 Revision 44



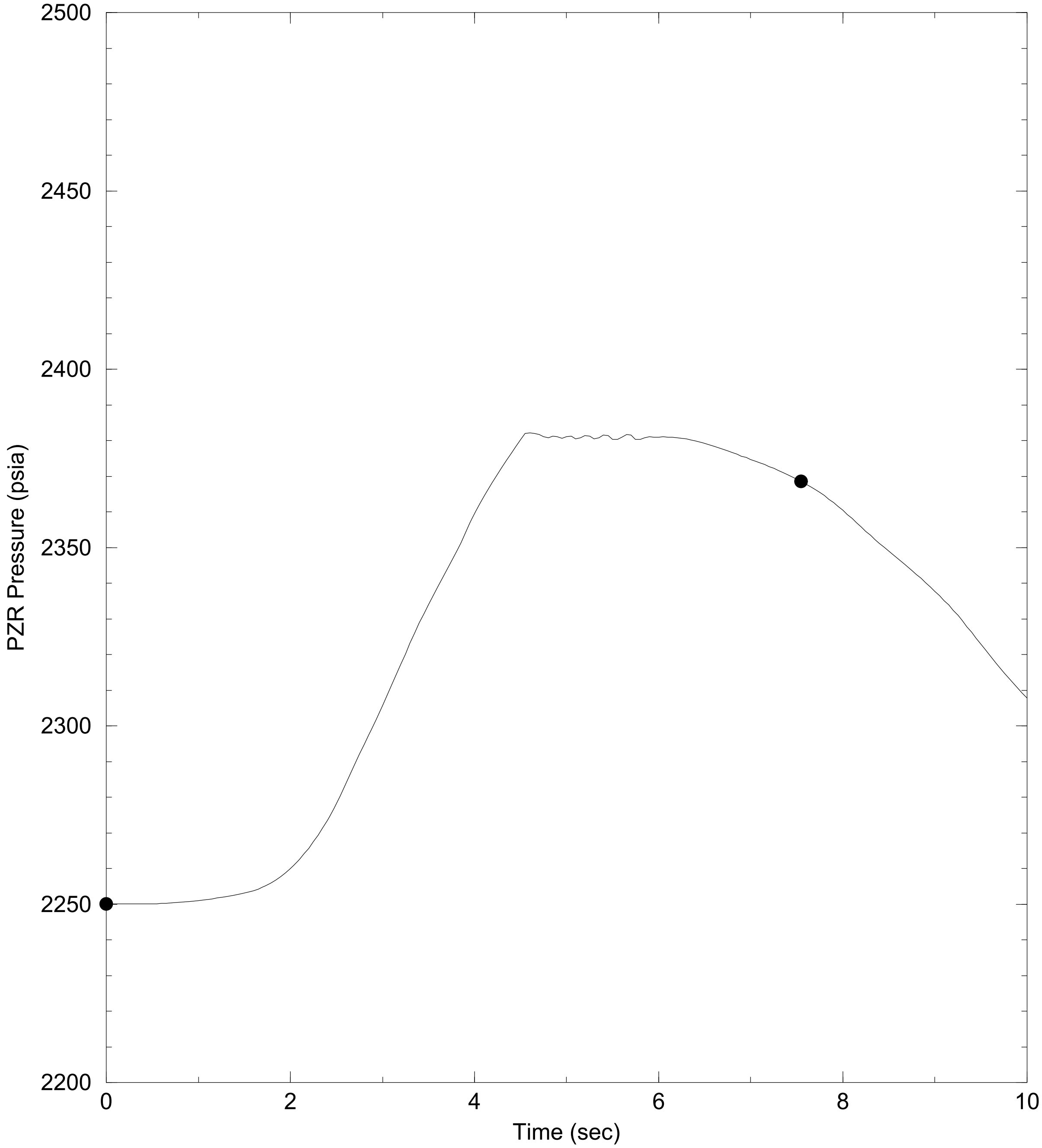
Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE POWER VS TIME	Figure 14.9-2 Revision 44



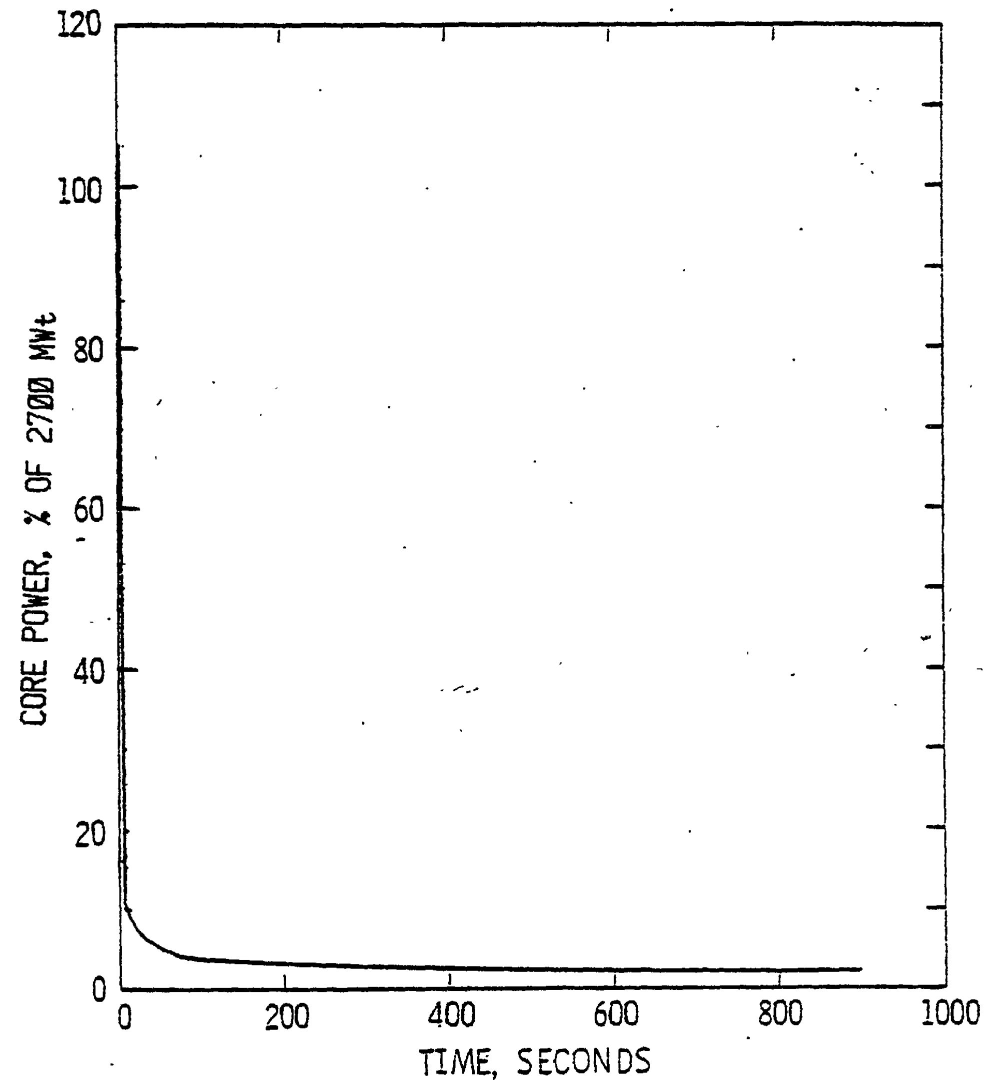
Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE HEAT FLUX VS TIME	Figure 14.9-3 Revision 44



Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT RCS TEMPERATURES VS TIME	Figure 14.9-4 Revision 44

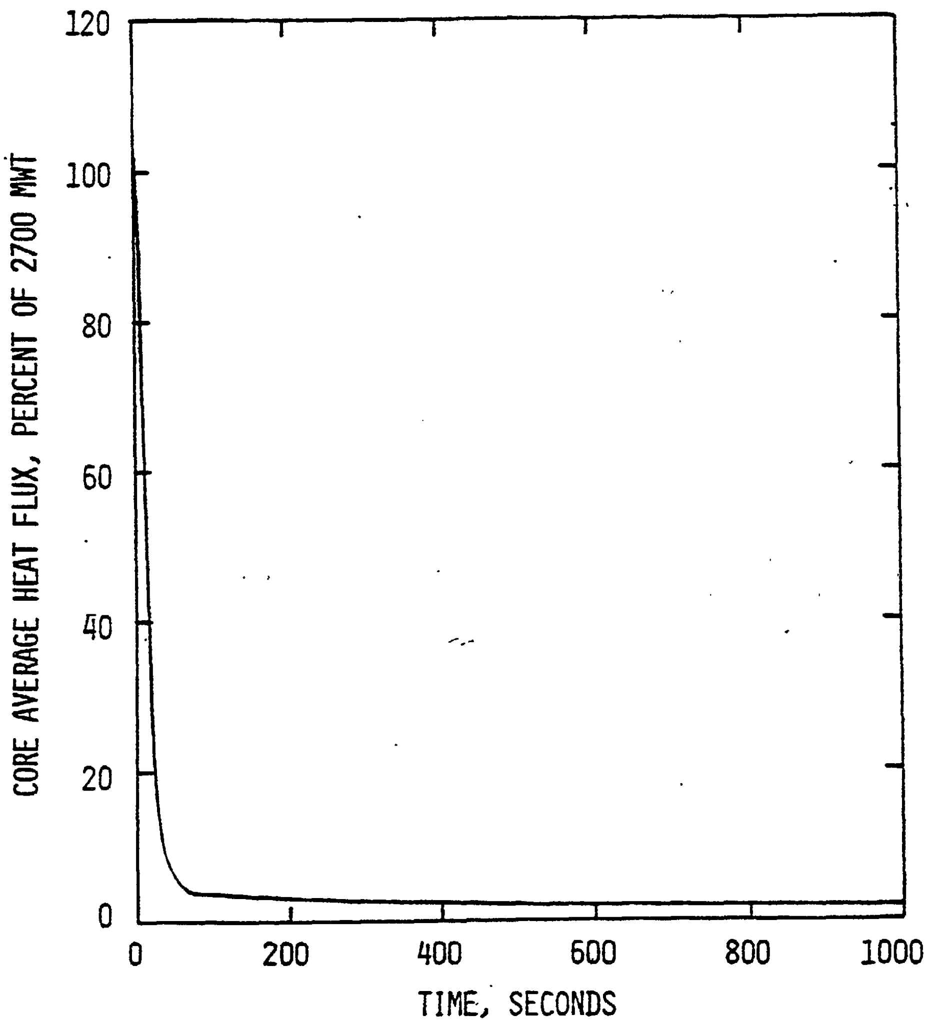


Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT RCS PRESSURE VS TIME	Figure 14.9-5 Revision 44



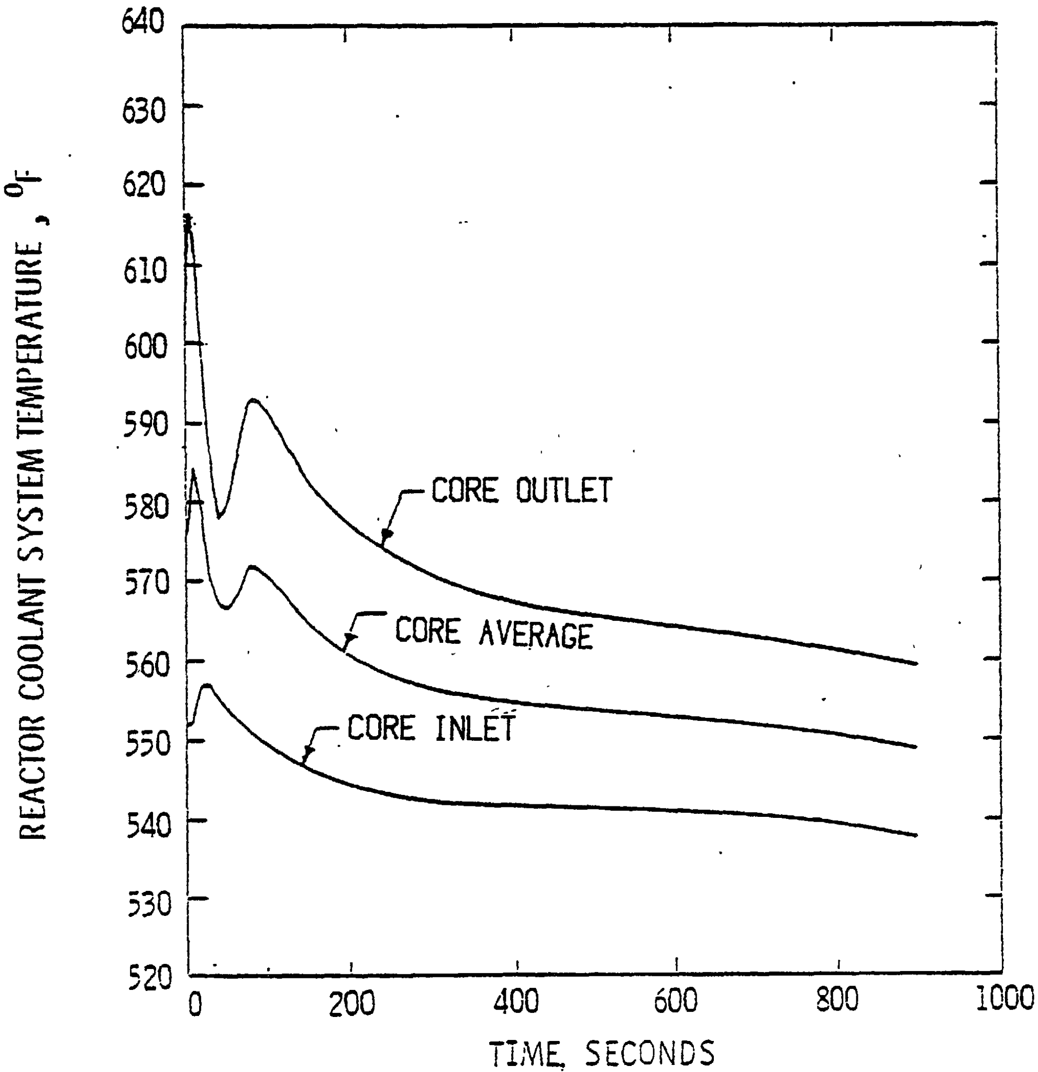
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF ALL NON-EMERGENCY AC POWER EVENT CORE POWER vs TIME	Figure 14.10-1

14.10-2 LOSS OF ALL NON-EMERGENCY AC POWER EVENT CORE AVERAGE HEAT FLUX VS TIME



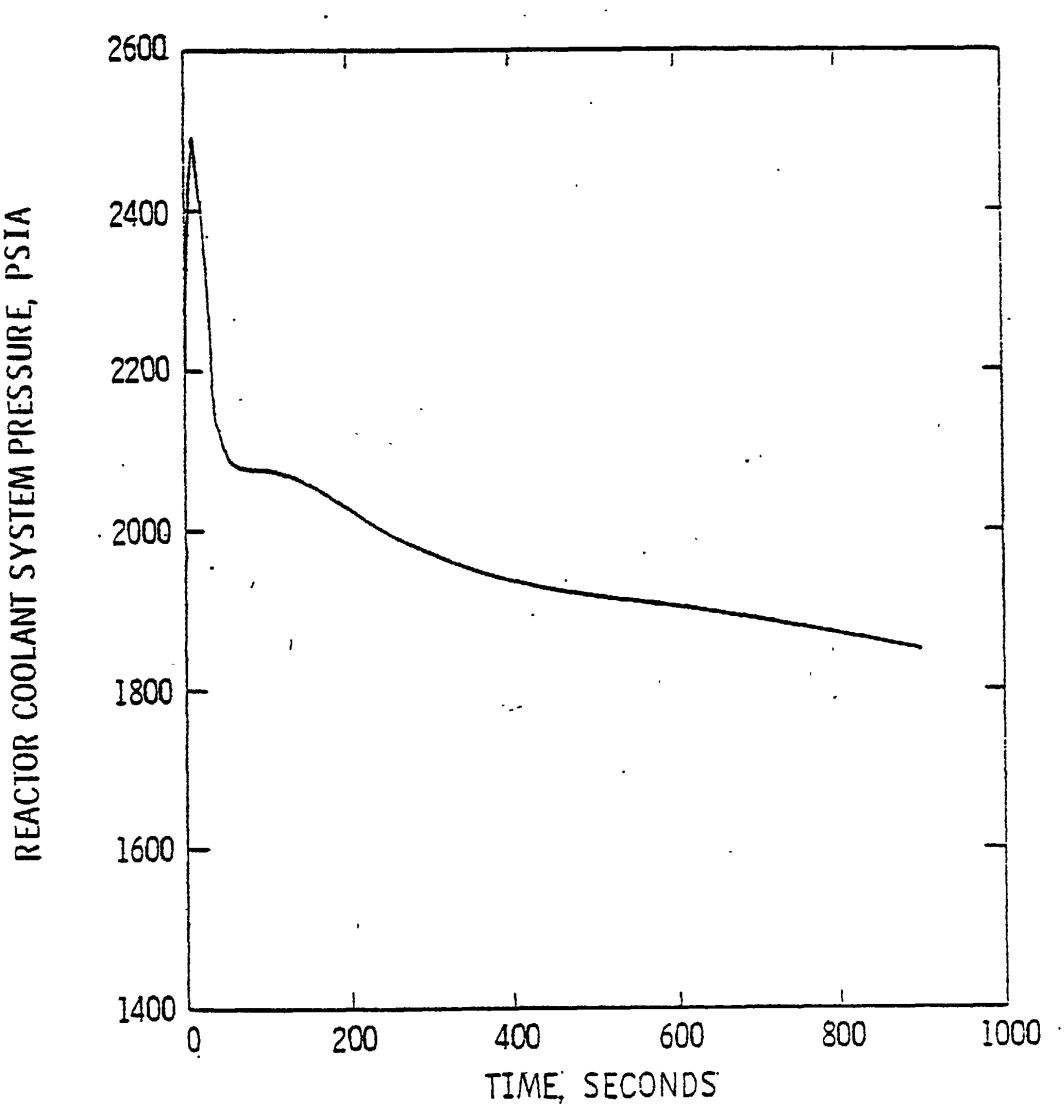
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF ALL NON-EMERGENCY AC POWER EVENT CORE AVERAGE HEAT FLUX vs TIME	Figure 14.10-2

14.10-3 LOSS OF ALL NON-EMERGENCY AC POWER EVENT REACTOR COOLANT SYSTEM TEMPERATURE VS TIME



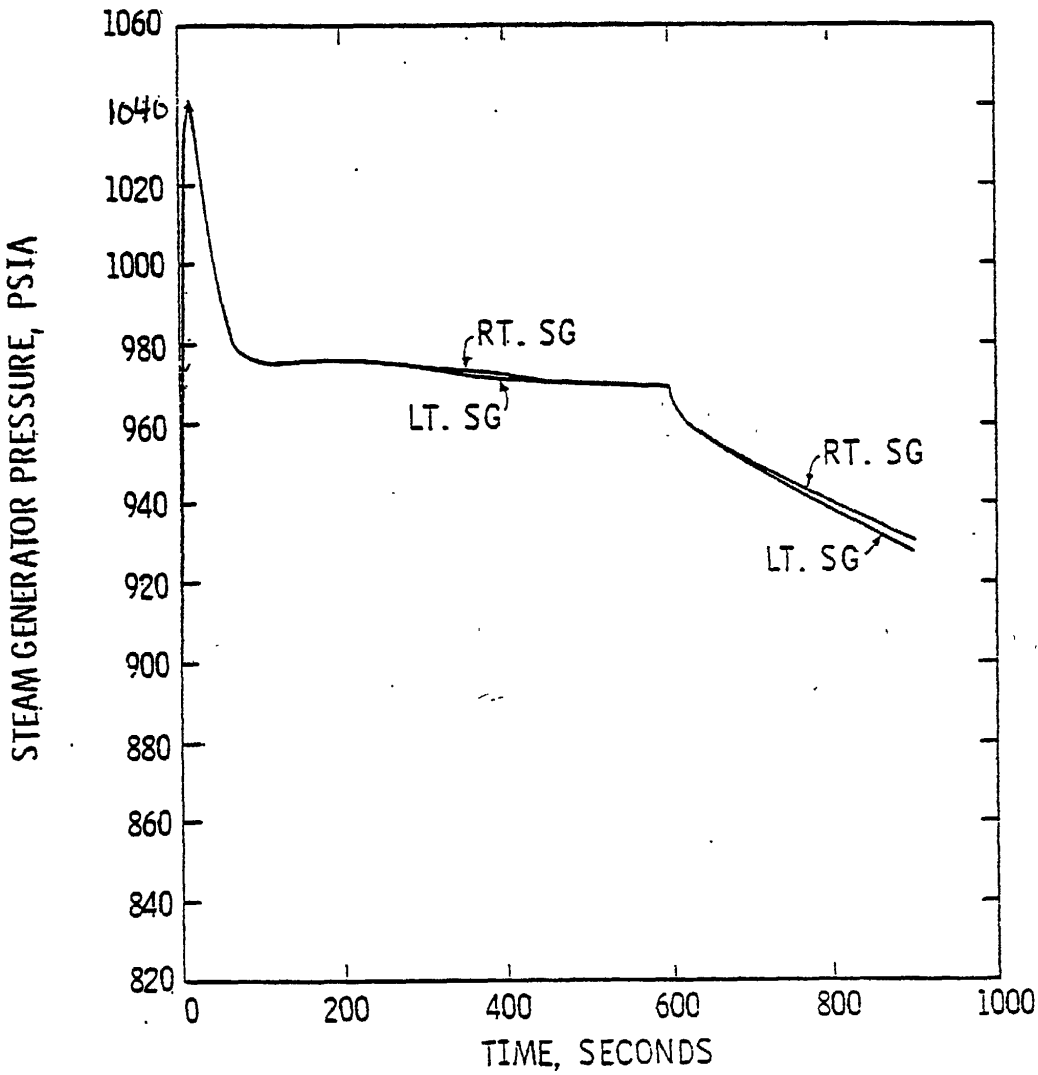
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF ALL NON-EMERGENCY AC POWER EVENT REACTOR COOLANT SYSTEM TEMPERATURE vs TIME	Figure 14.10-3

14.10-4 LOSS OF ALL NON-EMERGENCY AC POWER EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME

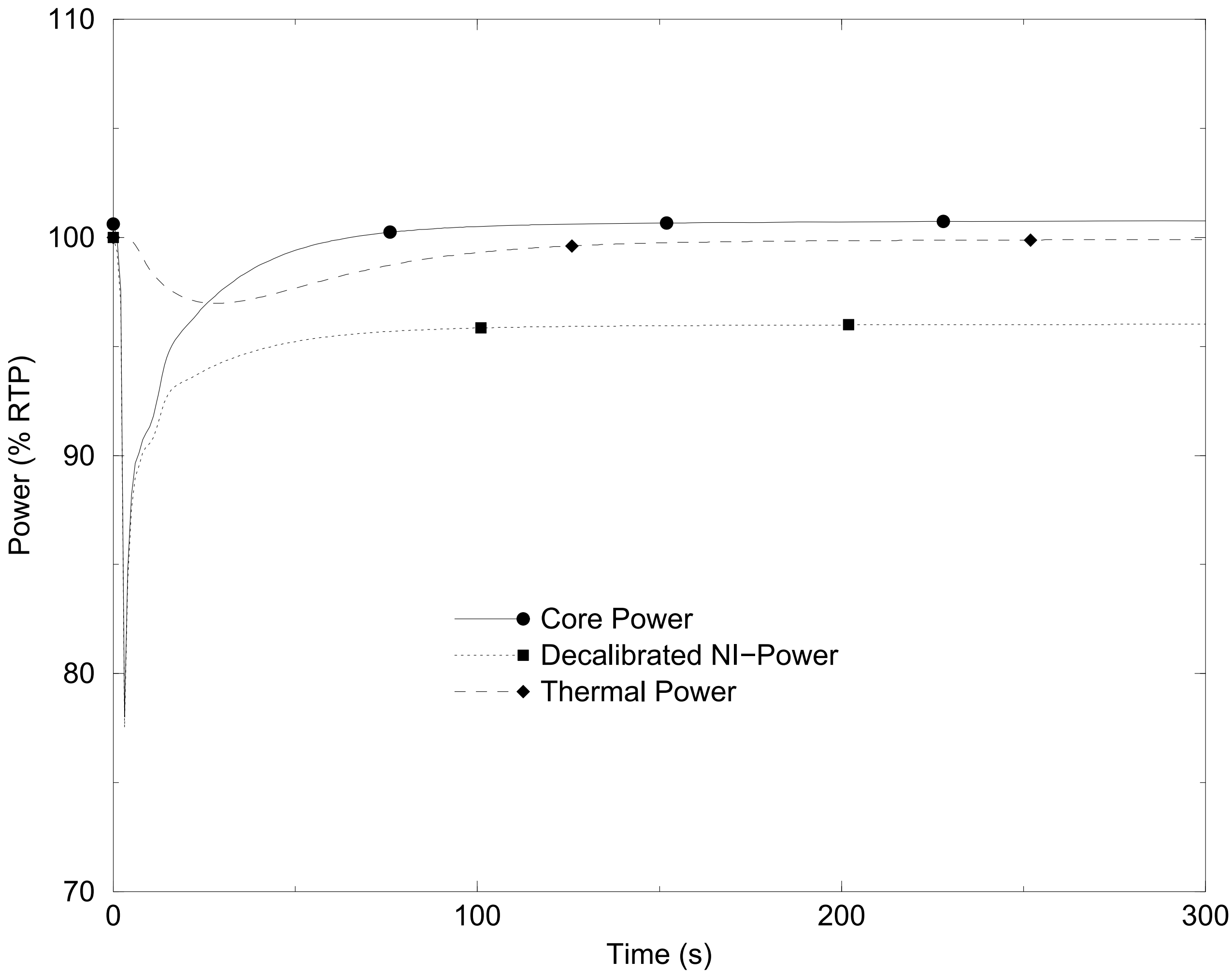


BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF ALL NON-EMERGENCY AC POWER EVENT REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 14.10-4

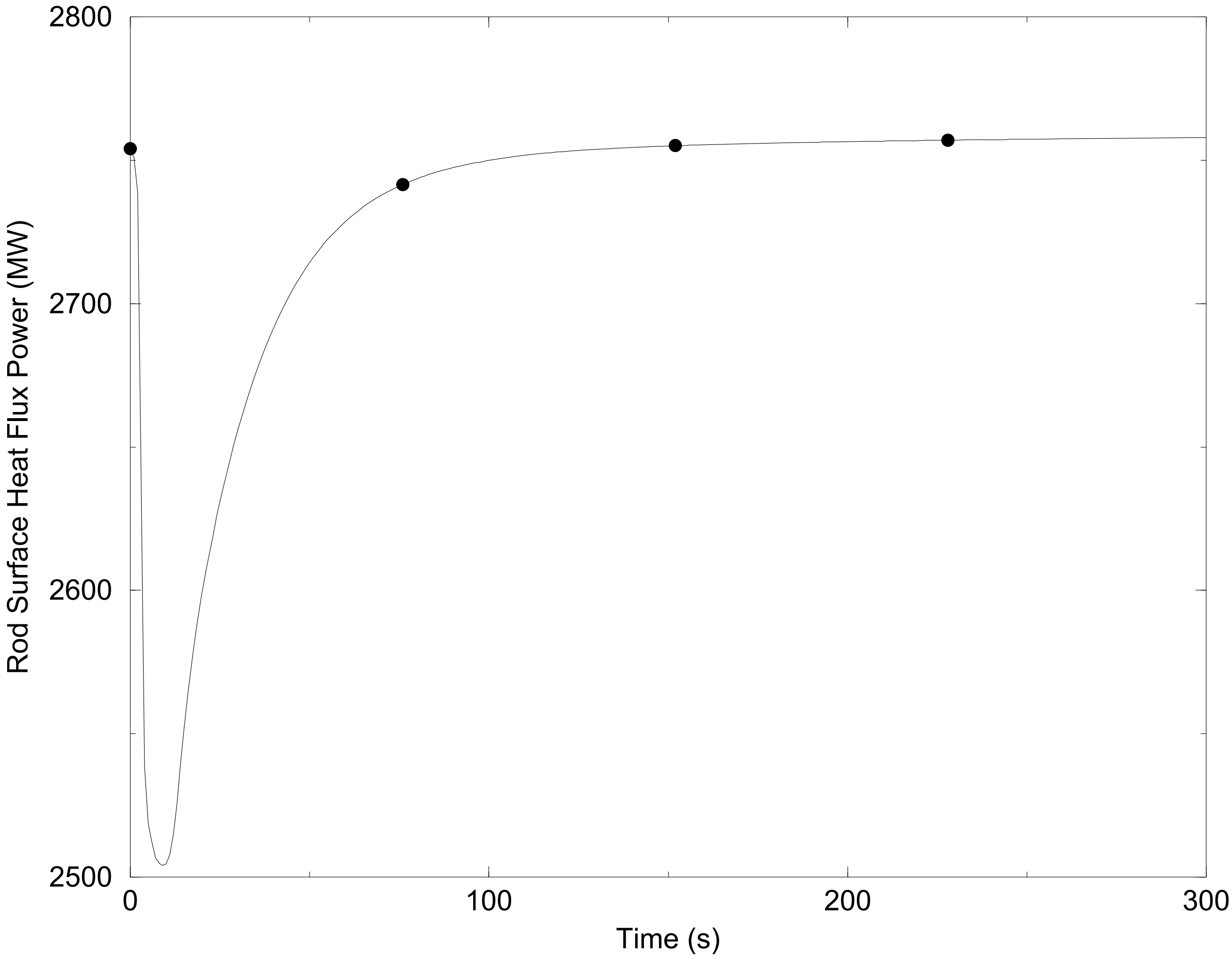
14.10-5 LOSS OF ALL NON-EMERGENCY AC POWER EVENT STEAM GENERATOR PRESSURE VS TIME



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF ALL NON-EMERGENCY AC POWER EVENT STEAM GENERATOR PRESSURE vs TIME	Figure 14.10-5



Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP CORE POWER VS TIME	Figure 14.11-1 Revision 44

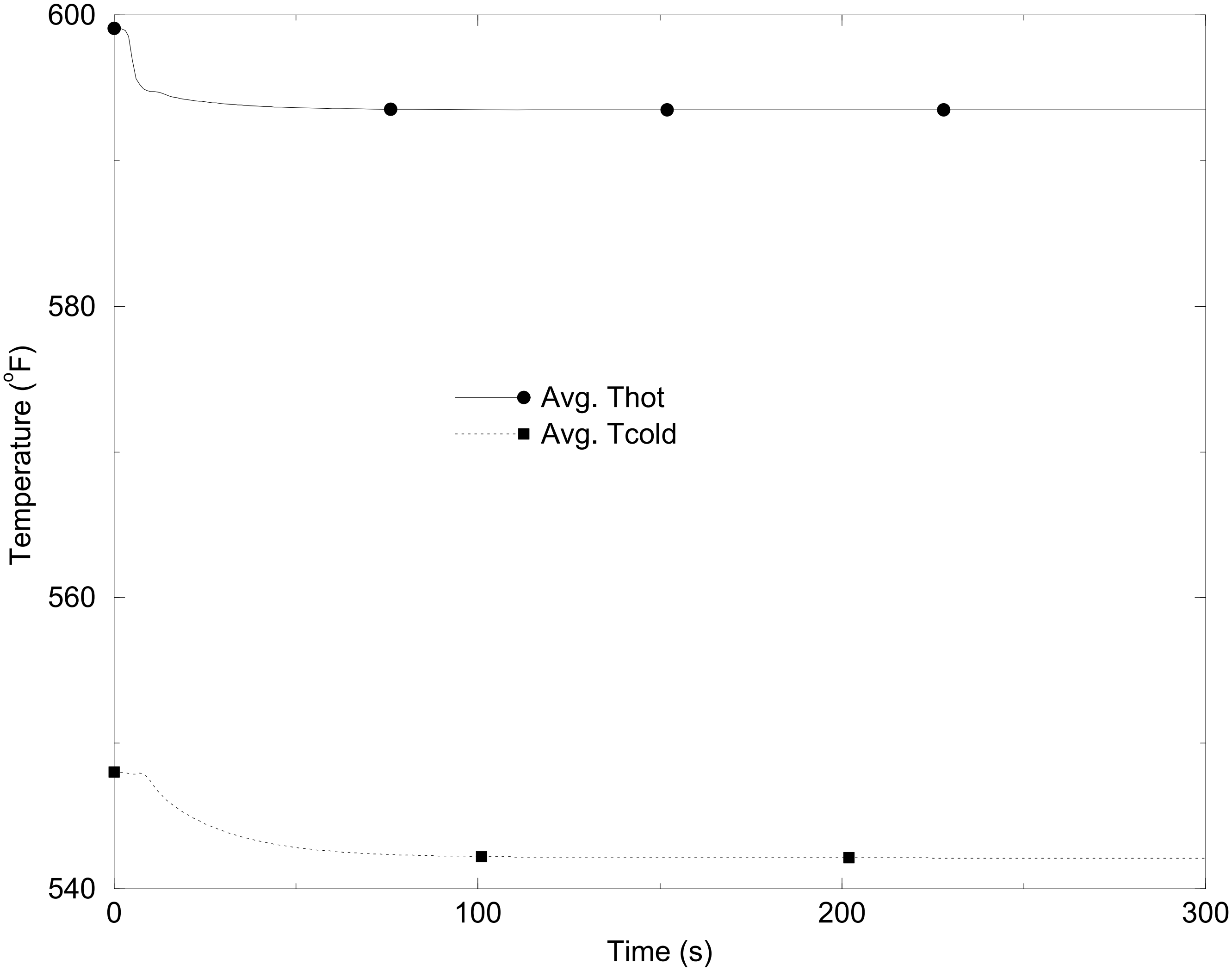


Calvert Cliffs Nuclear Power Plant

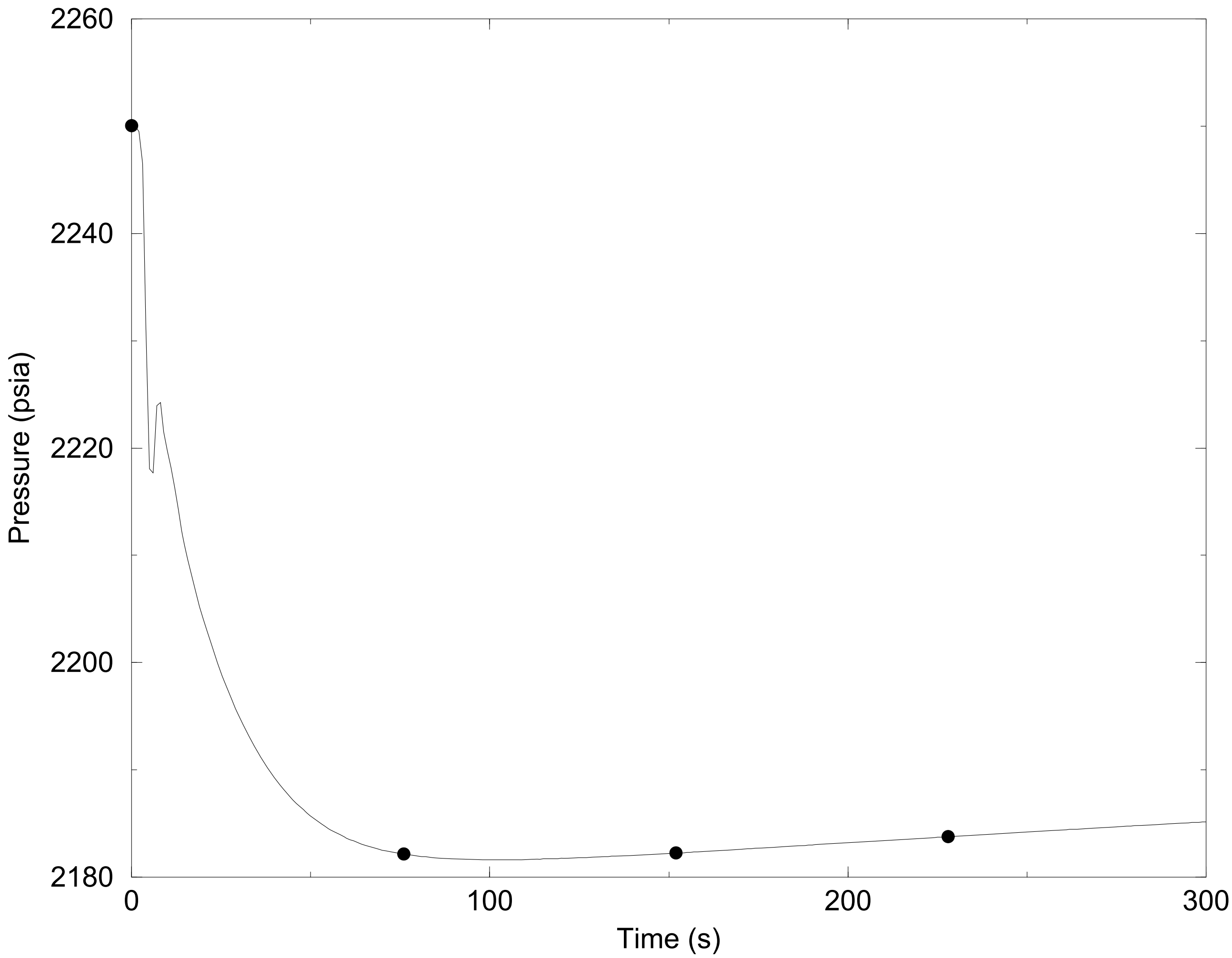
FULL LENGTH CEA DROP
CORE HEAT FLUX VS TIME

Figure 14.11-2

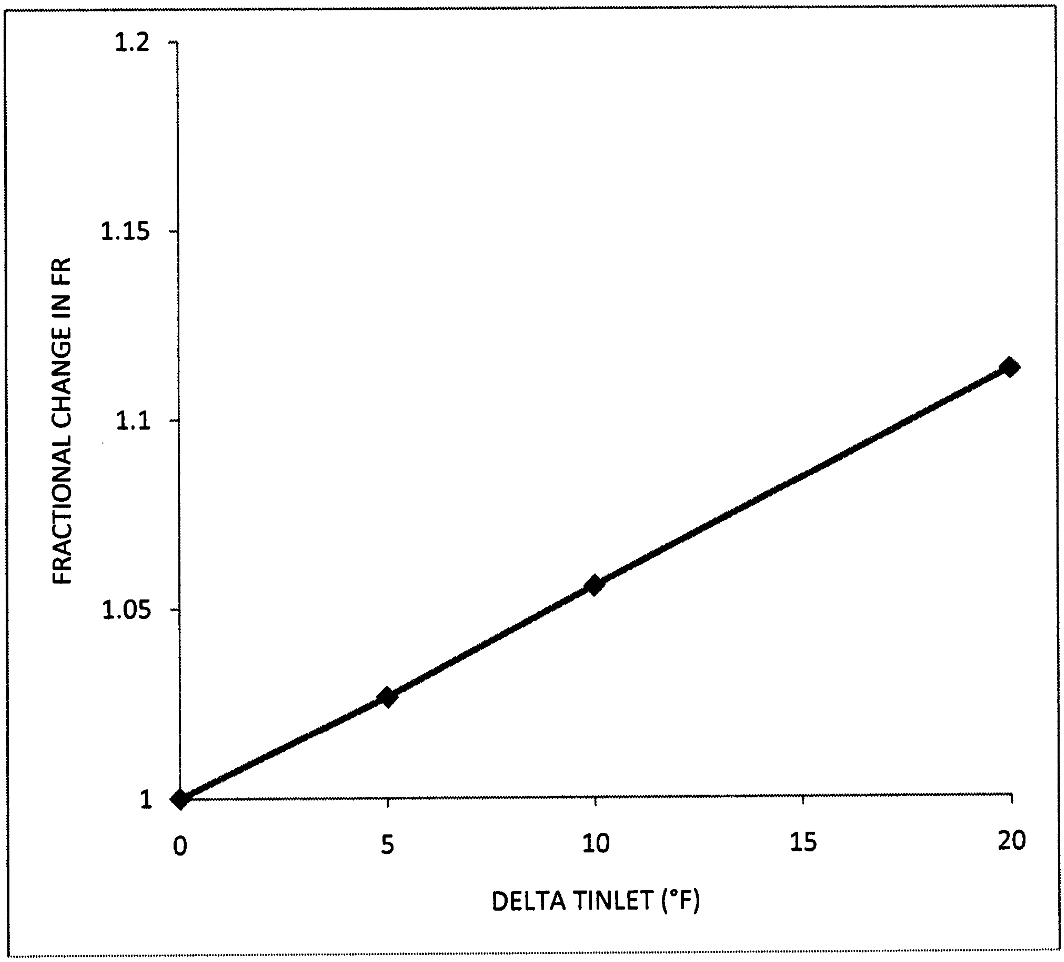
Revision 44



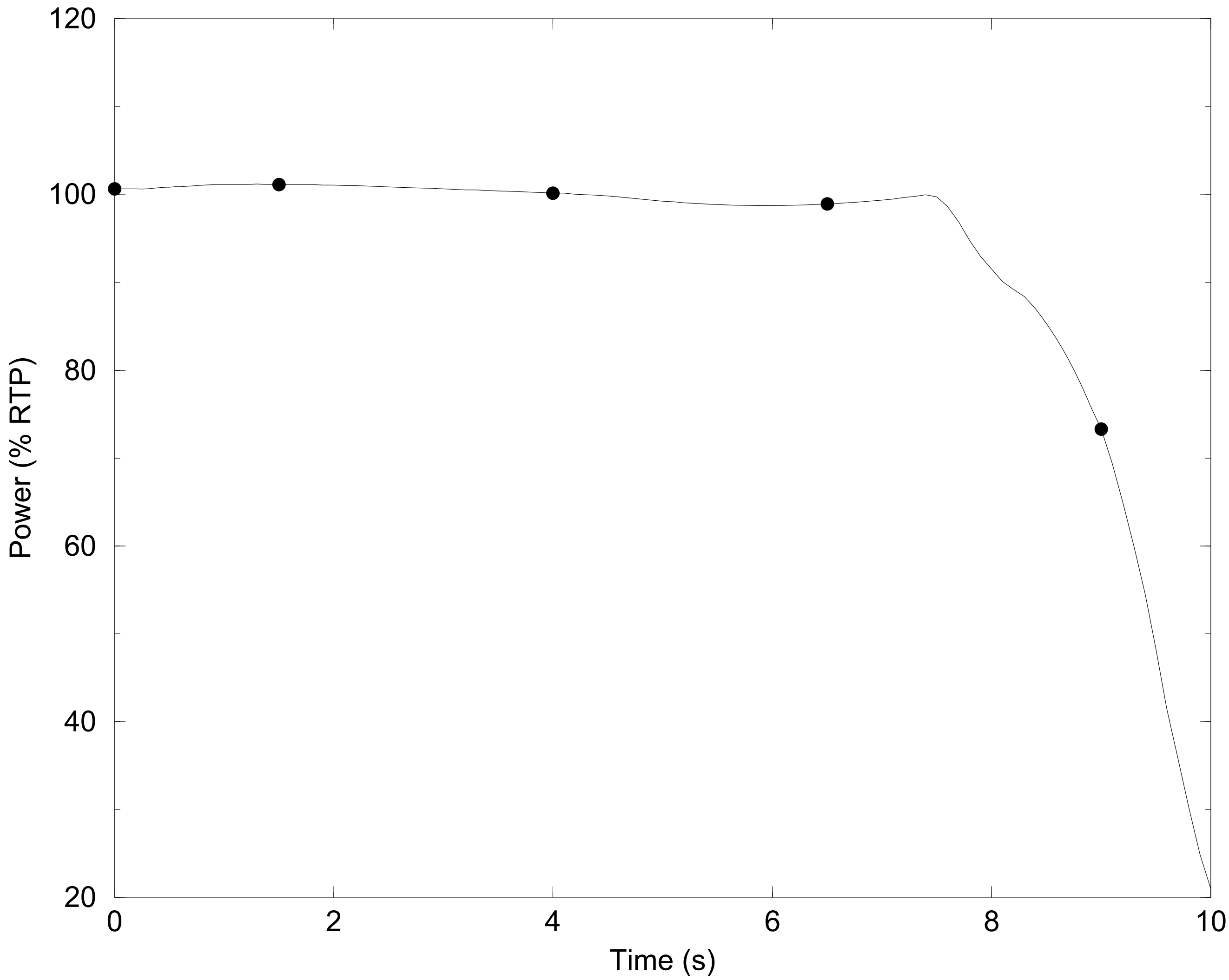
Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP RCS TEMPERATURES VS TIME	Figure 14.11-3 Revision 44



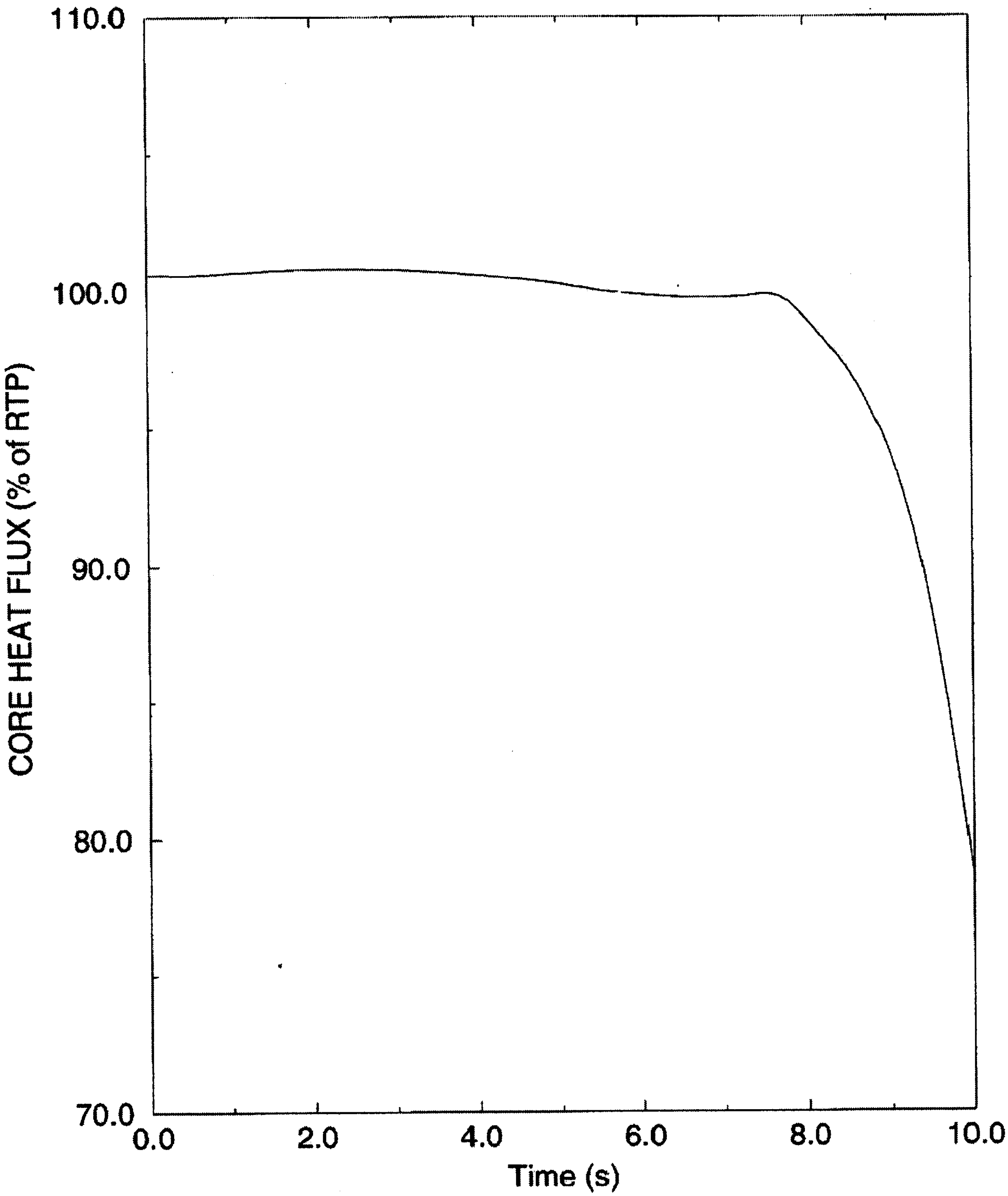
Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP RCS PRESSURE VS TIME	Figure 14.11-4 Revision 44



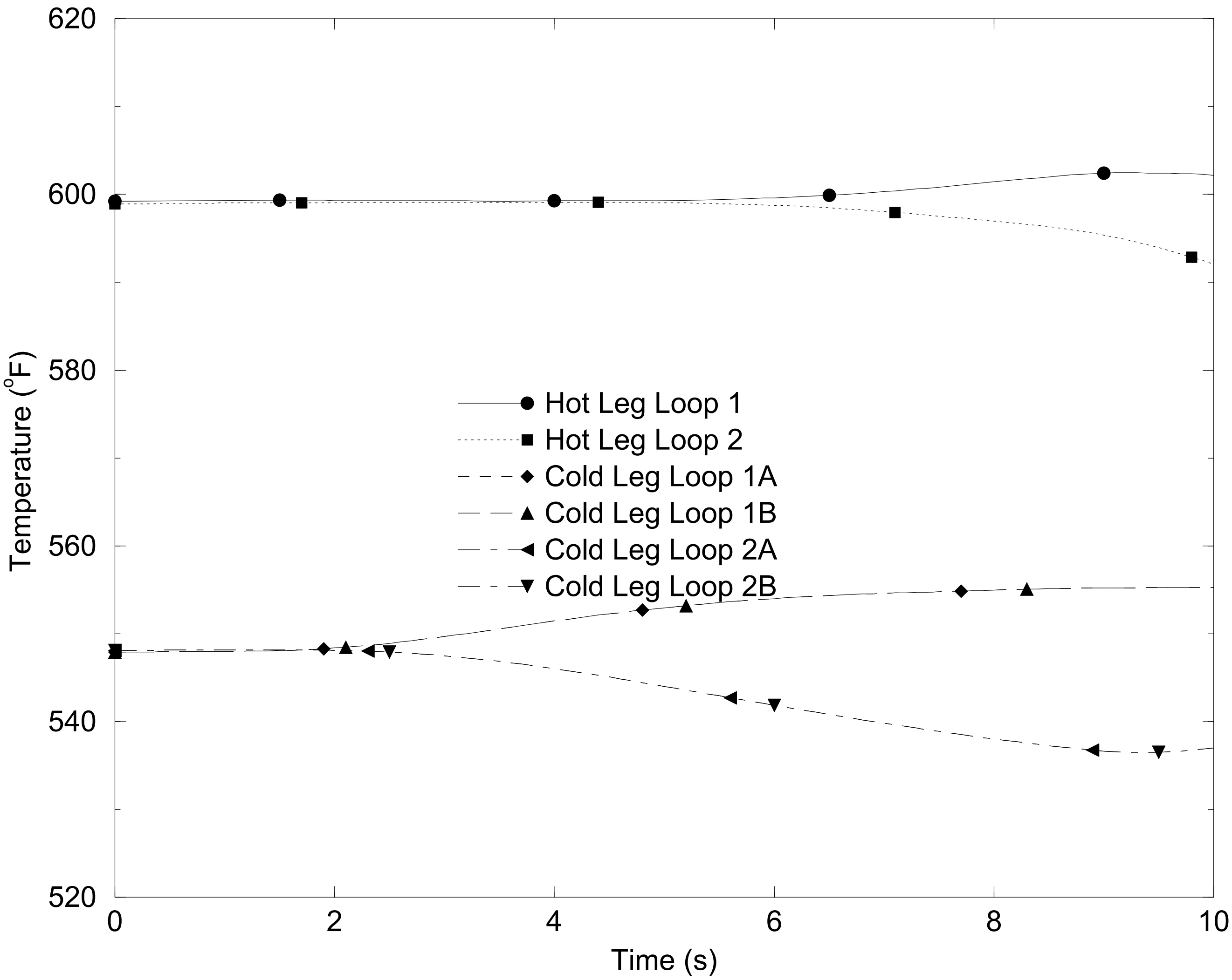
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT RADIAL DISTORTION FACTOR VS CORE INLET TEMPERATURE ASYMMETRY	Figure 14.12-1 Revision 44



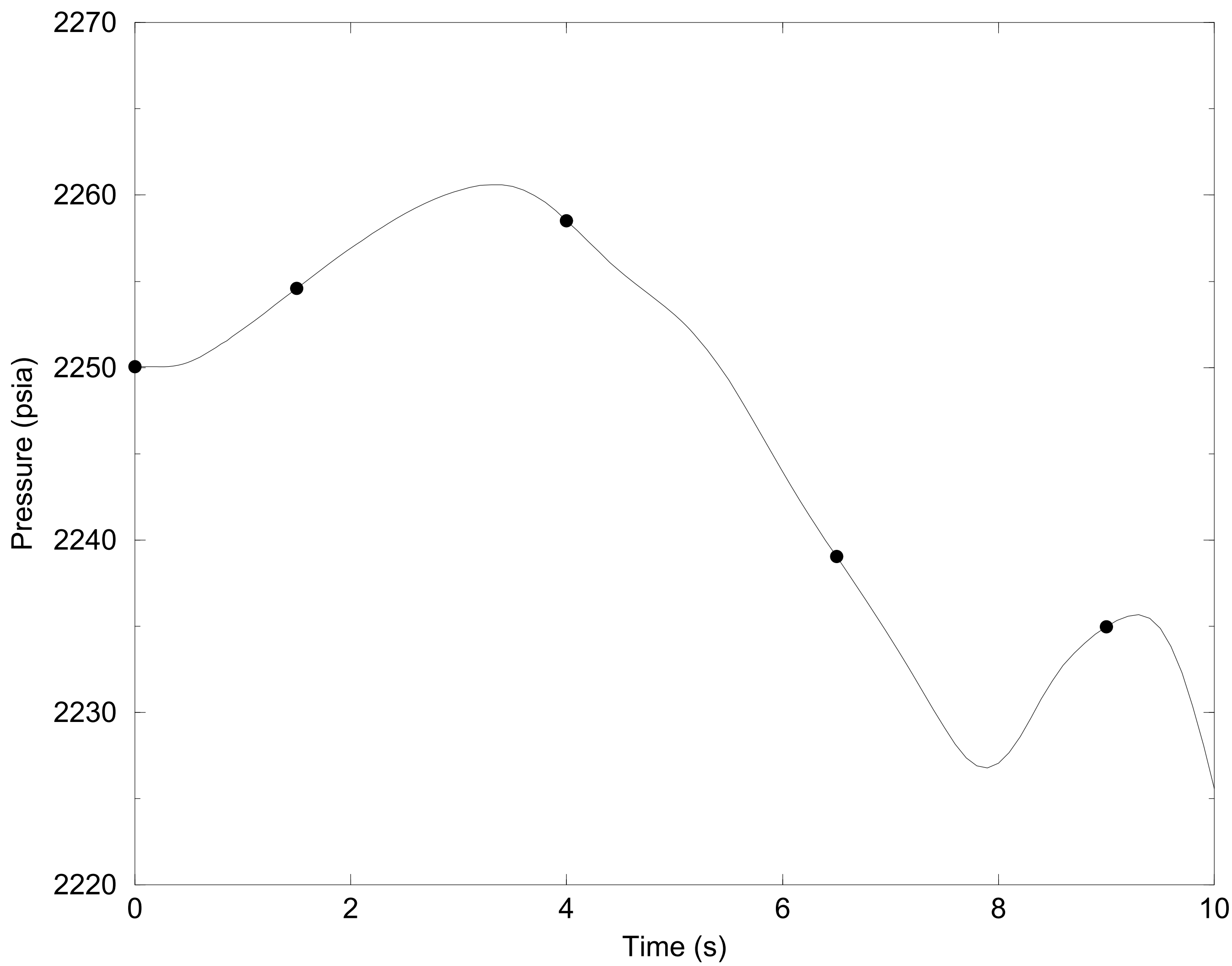
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 SG EVENT CORE POWER VS TIME	Figure 14.12-2 Revision 44



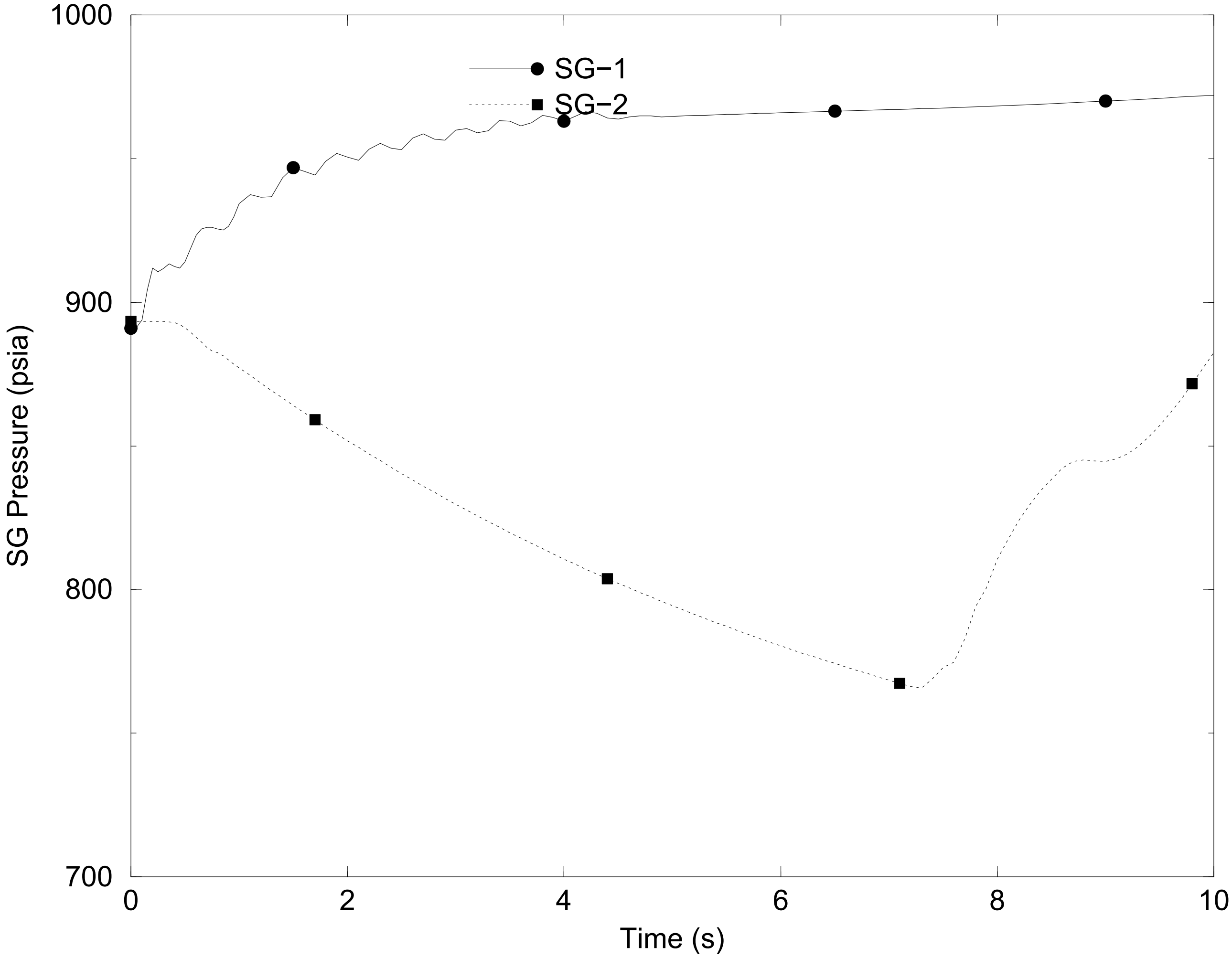
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 SG EVENT CORE HEAT FLUX VS TIME	Figure 14.12-3 Revision 44



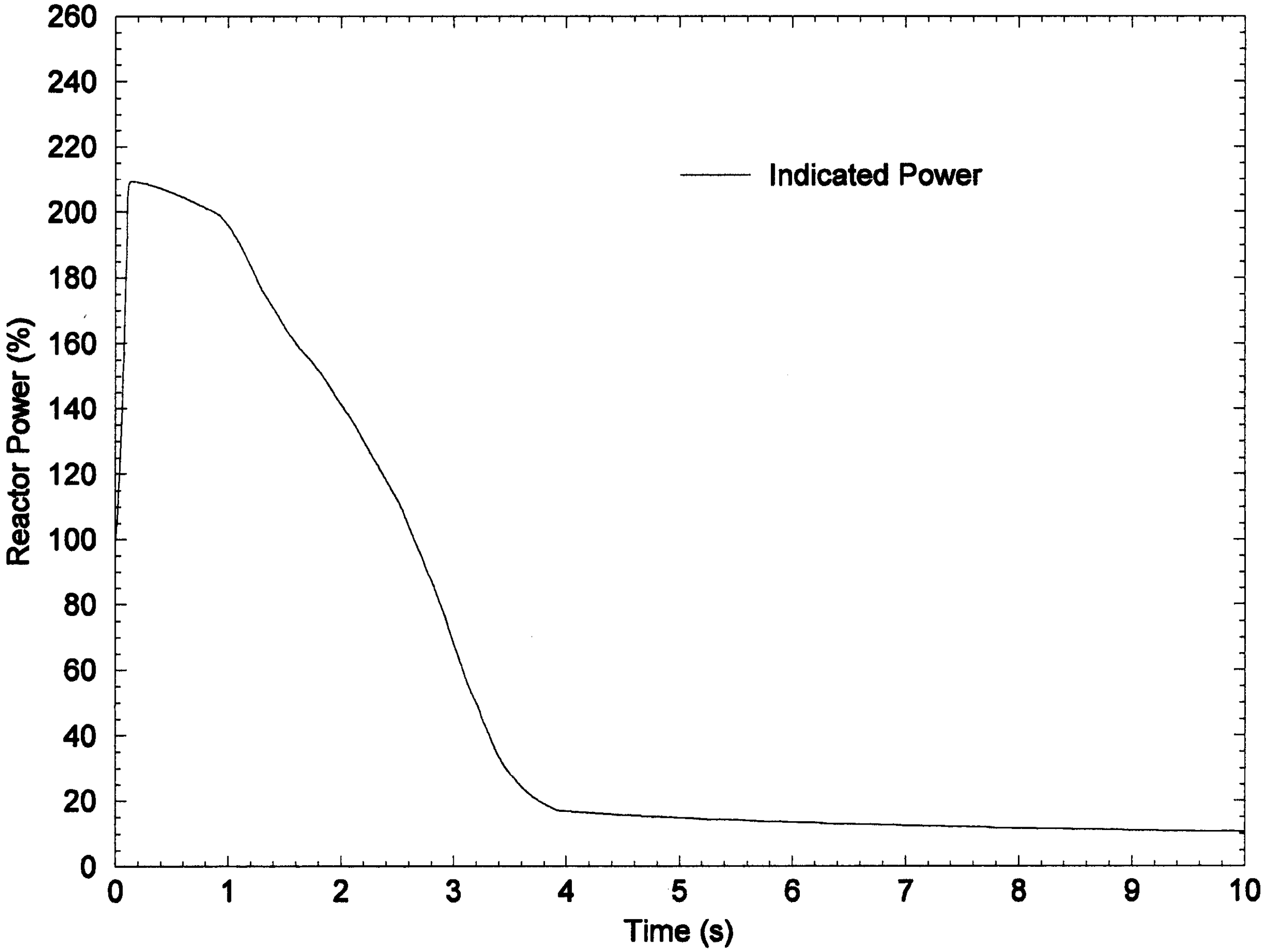
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 SG EVENT RCS TEMPERATURES VS TIME	Figure 14.12-4 Revision 44



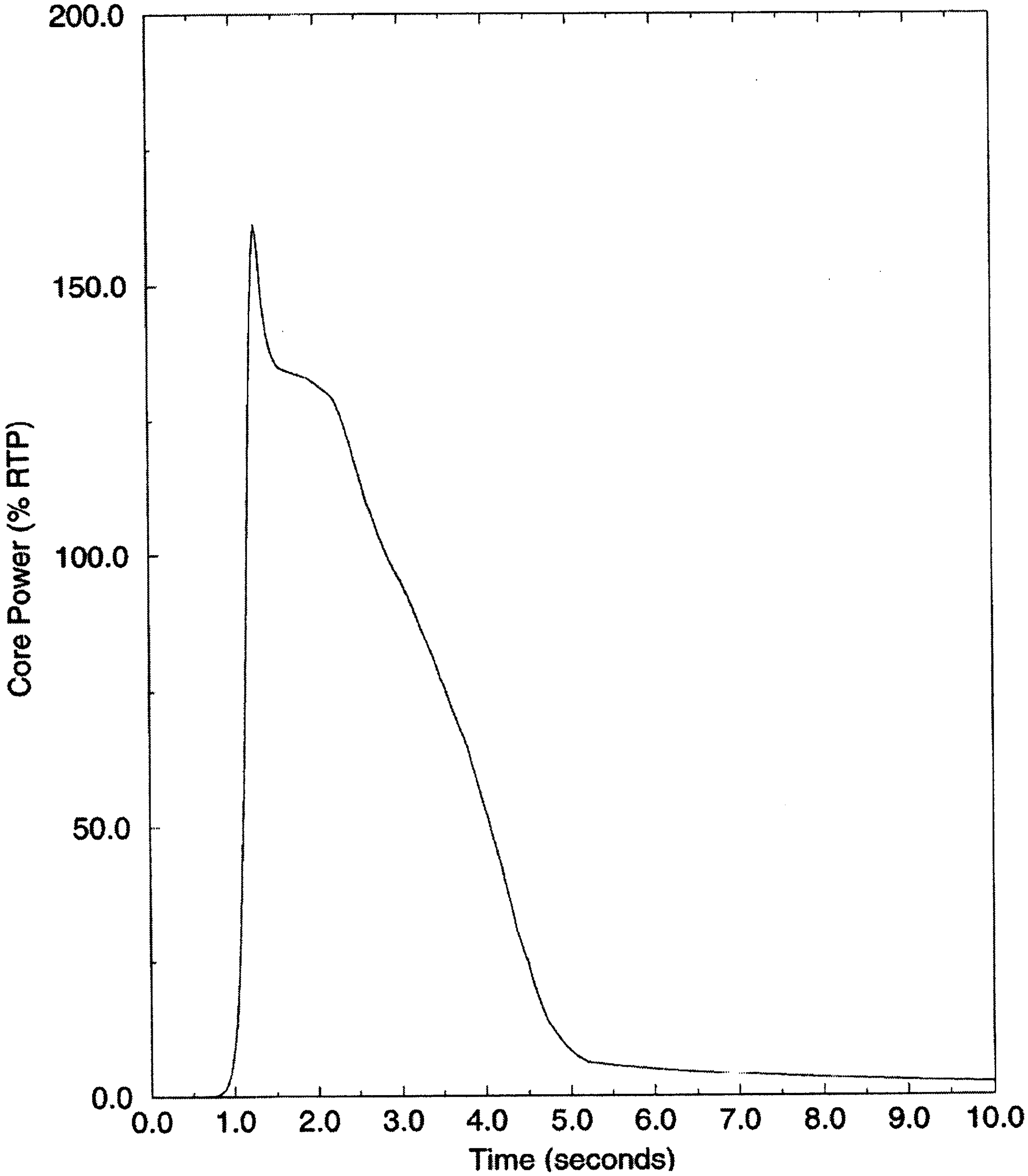
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 SG EVENT PRESSURIZER PRESSURE VS TIME	Figure 14.12-5 Revision 44



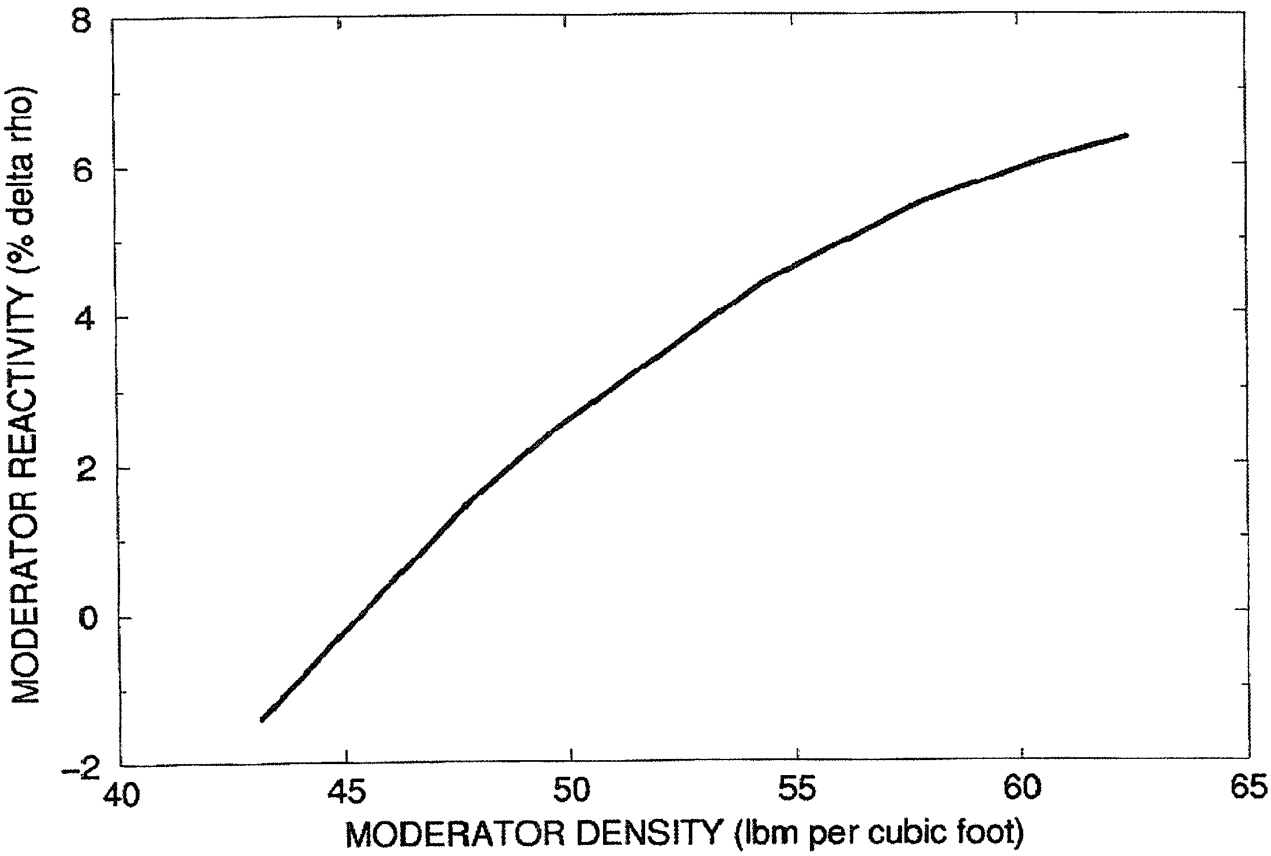
Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 SG EVENT SG PRESSURES VS TIME	Figure 14.12-6 Revision 44



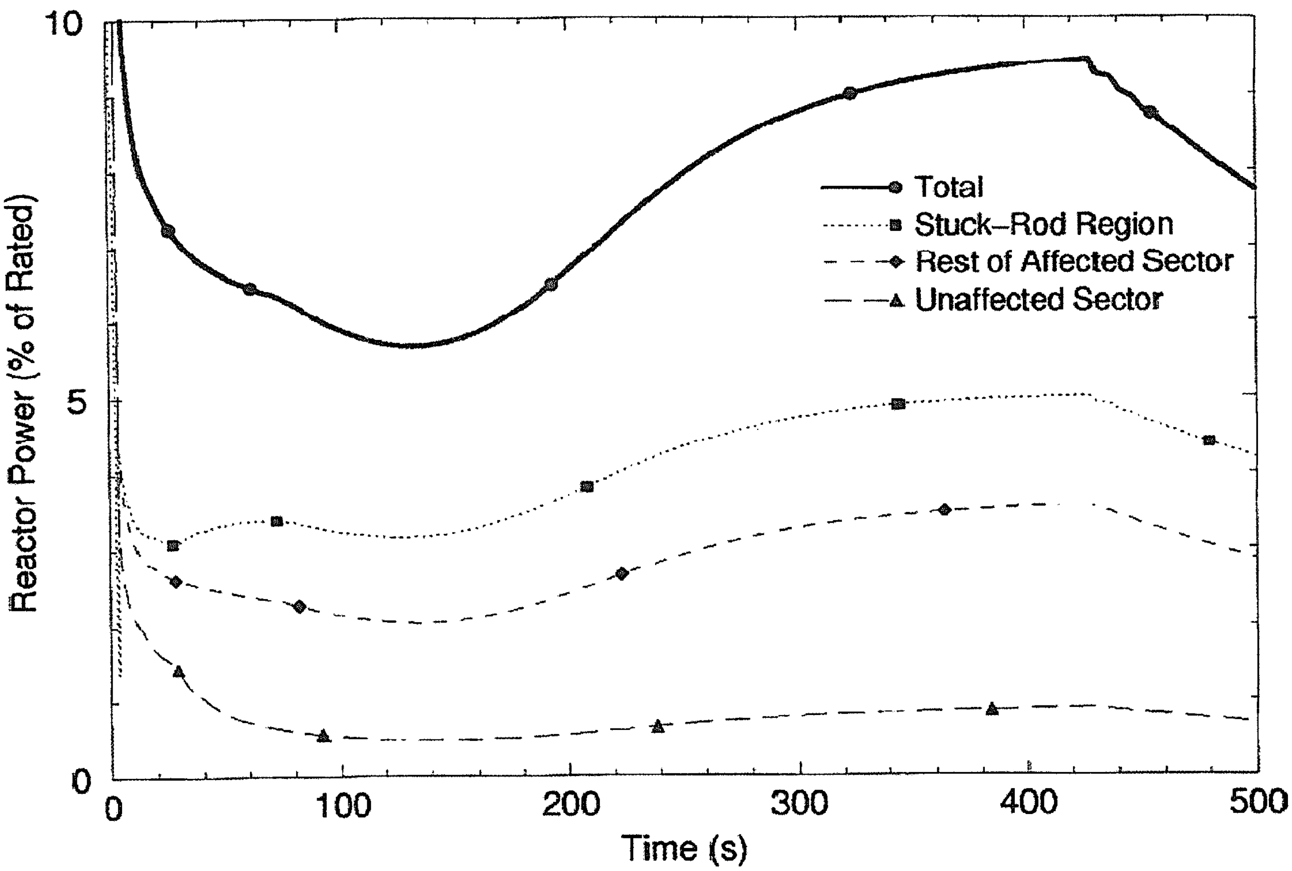
Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT HOT FULL POWER CORE POWER VS TIME	Figure 14.13-1 Revision 44



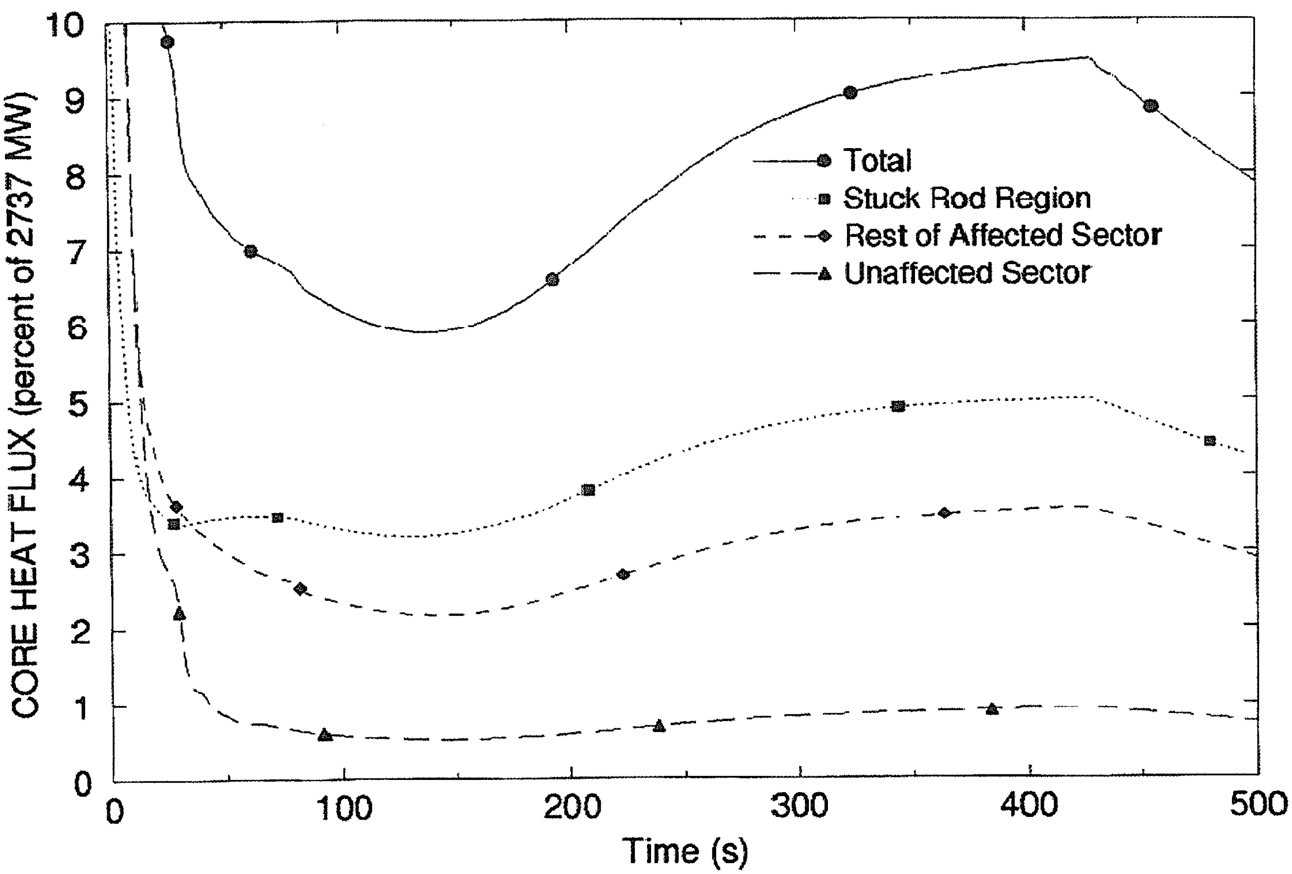
Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT HOT ZERO POWER CORE POWER VS TIME	Figure 14.13-2 Revision 44



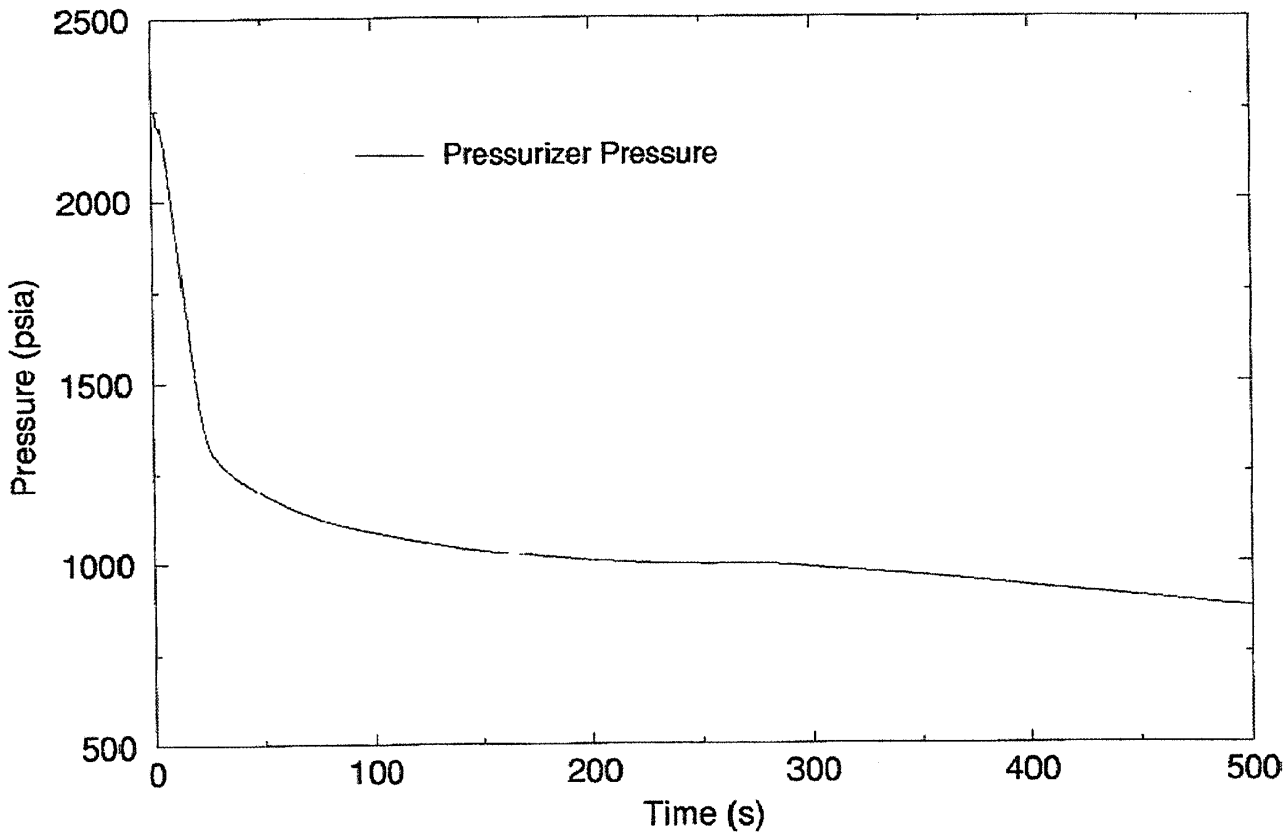
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP MODERATOR REACTIVITY VS MODERATOR DENSITY	Figure 14.14-1 Revision 46



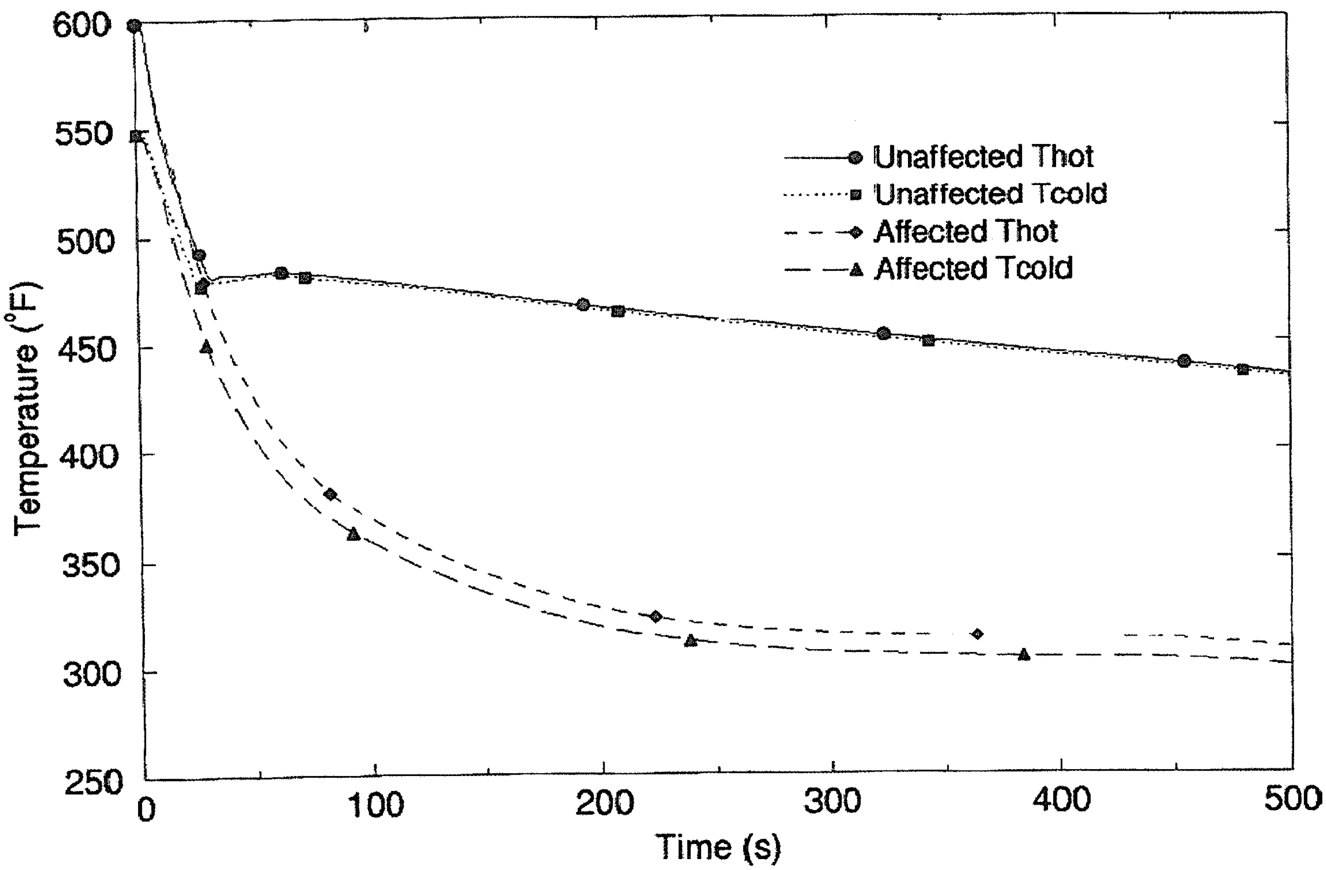
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP CORE POWER VS TIME	Figure 14.14-2 Revision 46



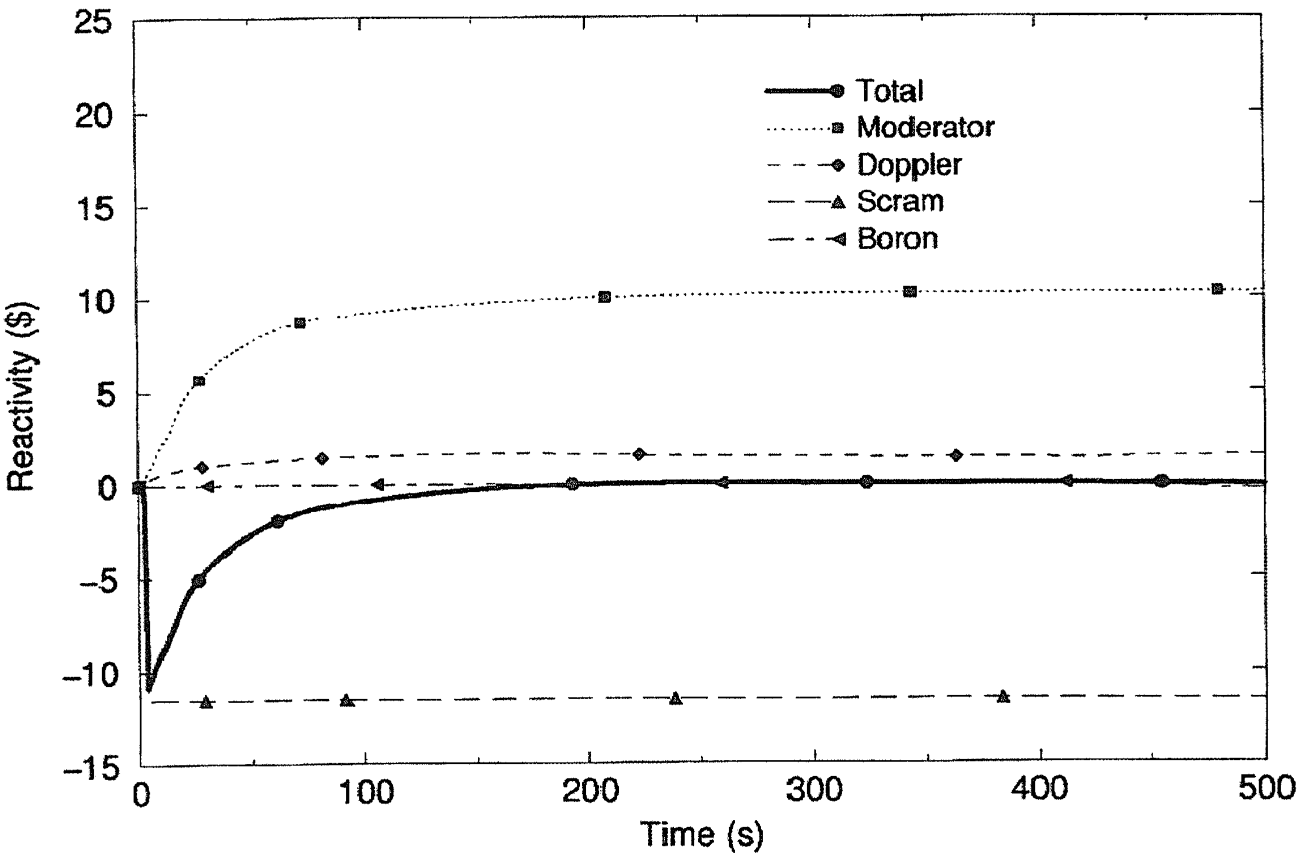
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP CORE HEAT FLUX VS TIME	Figure 14.14-3 Revision 46



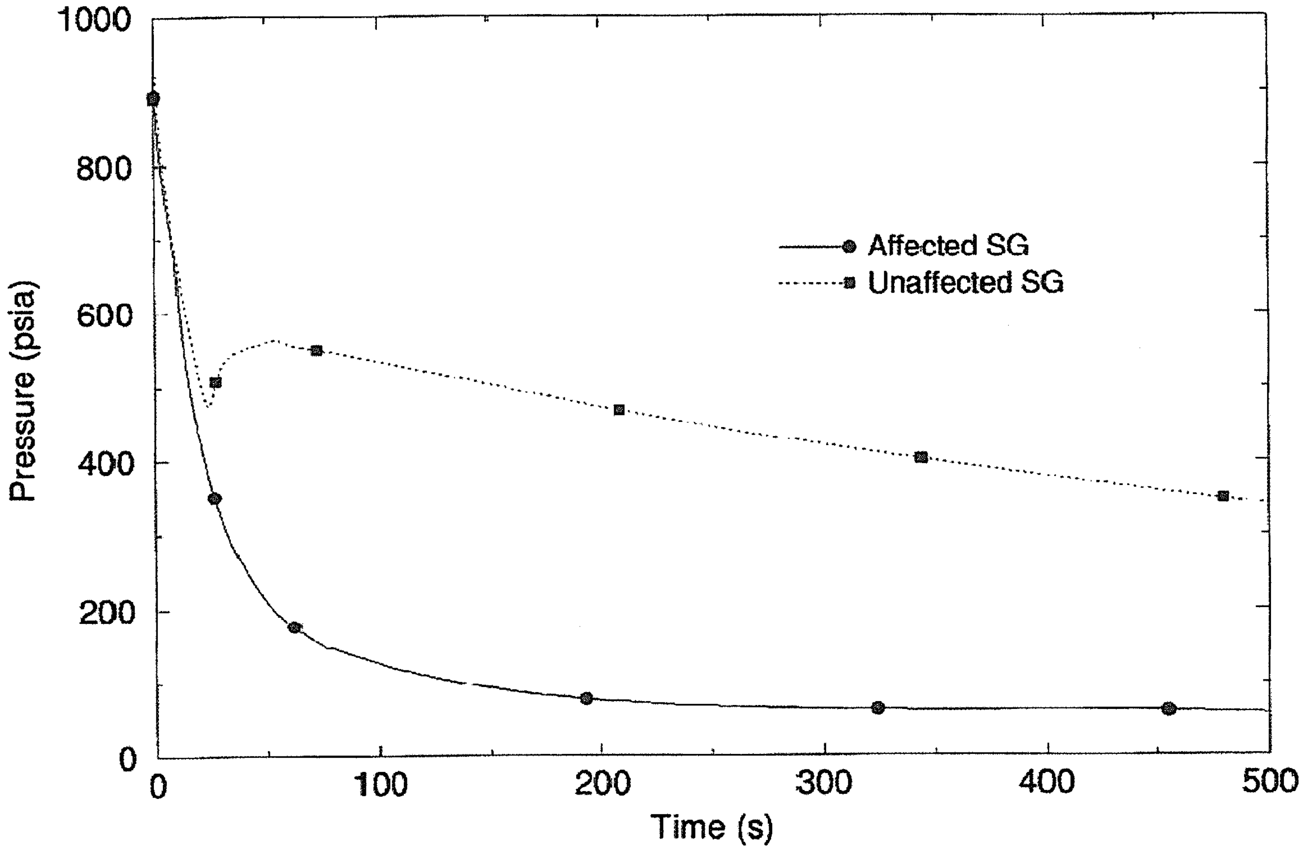
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP PRESSURIZER PRESSURE VS TIME	Figure 14.14-4 Revision 46



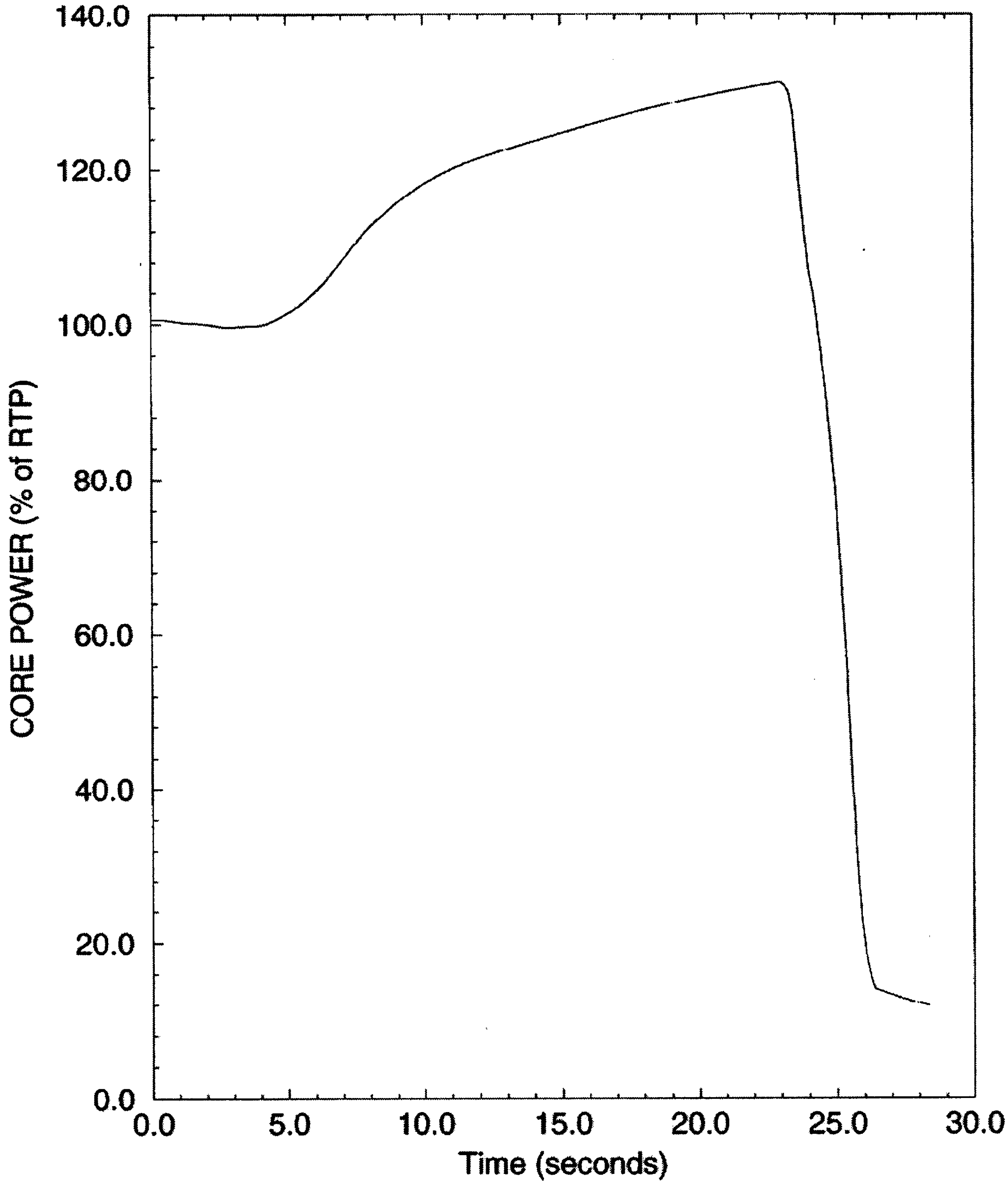
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP RCS TEMPERATURES VS TIME	Figure 14.14-5 Revision 46



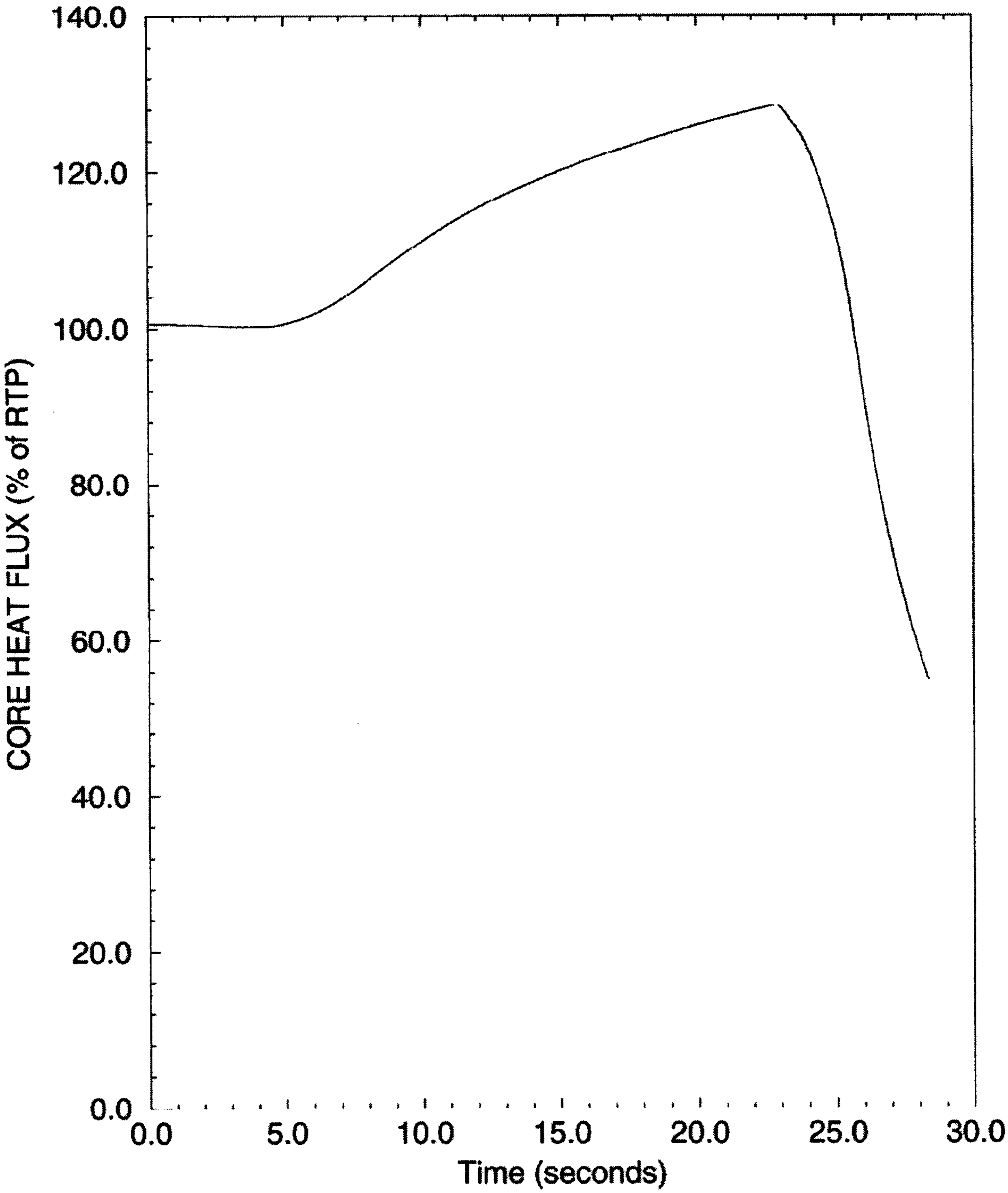
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP REACTIVITIES VS TIME	Figure 14.14-6 Revision 46



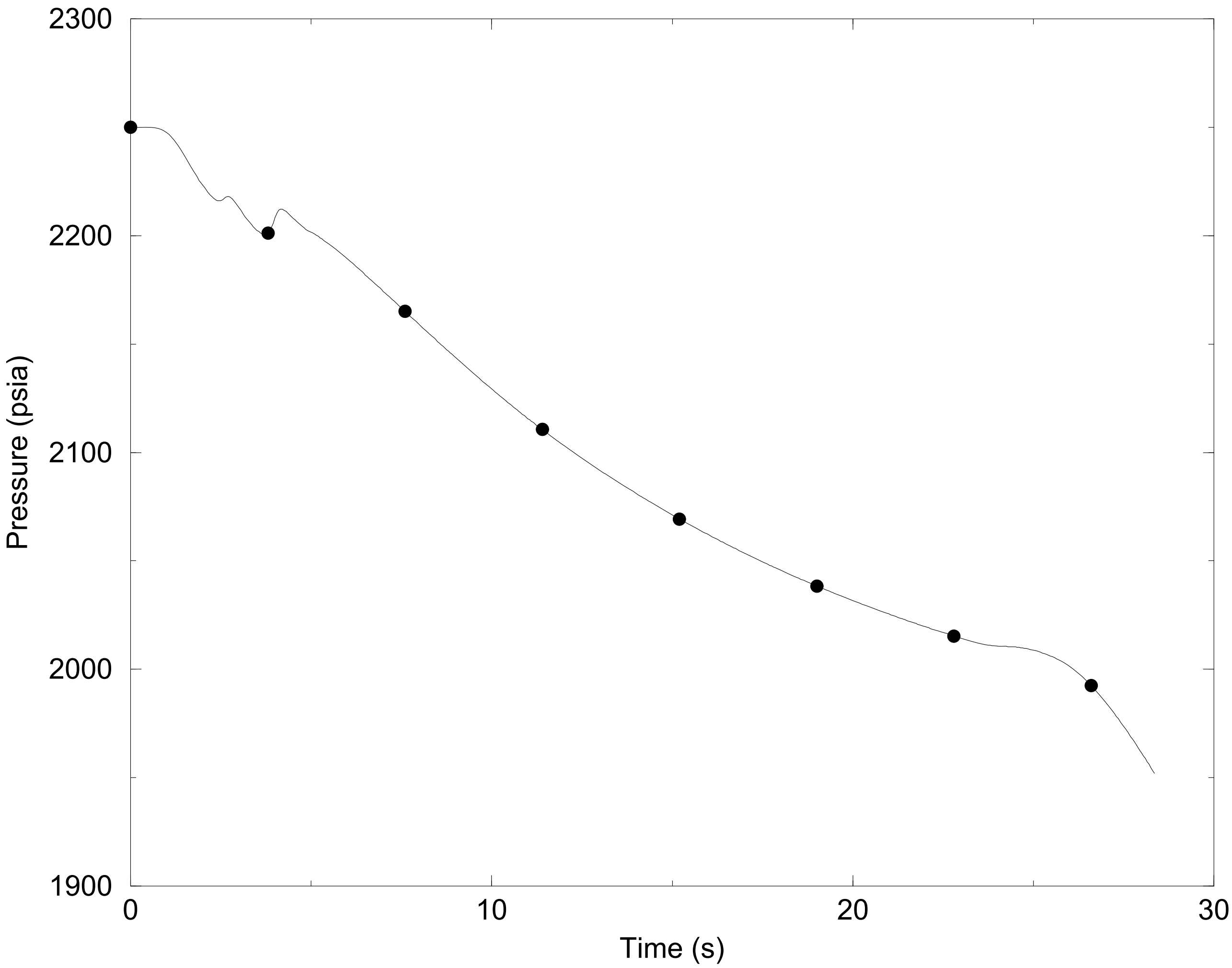
Calvert Cliffs Nuclear Power Plant	SLB EVENT POST-TRIP SG PRESSURE VS TIME	Figure 14.14-7 Revision 46



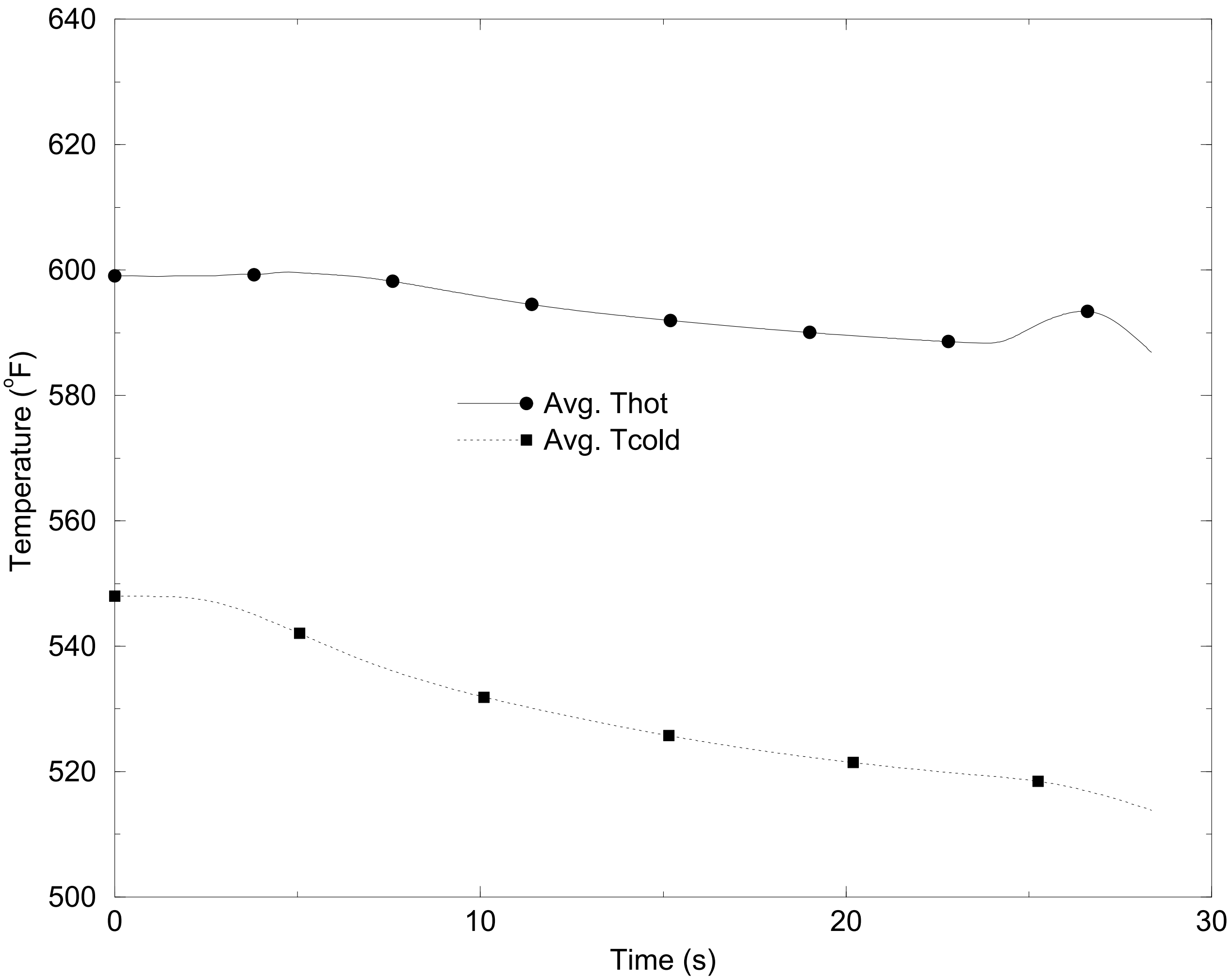
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP CORE POWER VS TIME	Figure 14.14-8 Revision 44



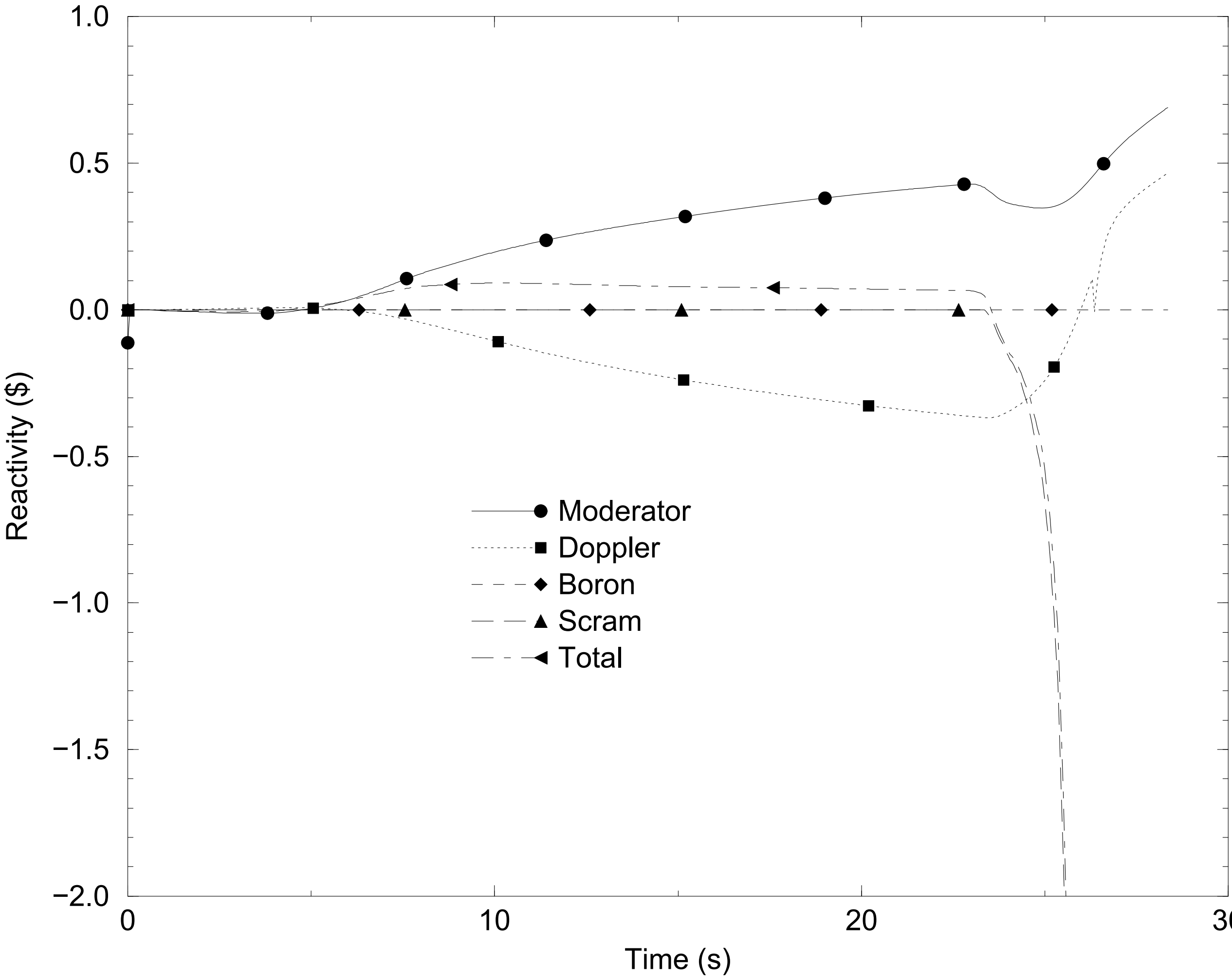
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP CORE HEAT FLUX VS TIME	Figure 14.14-9 Revision 44



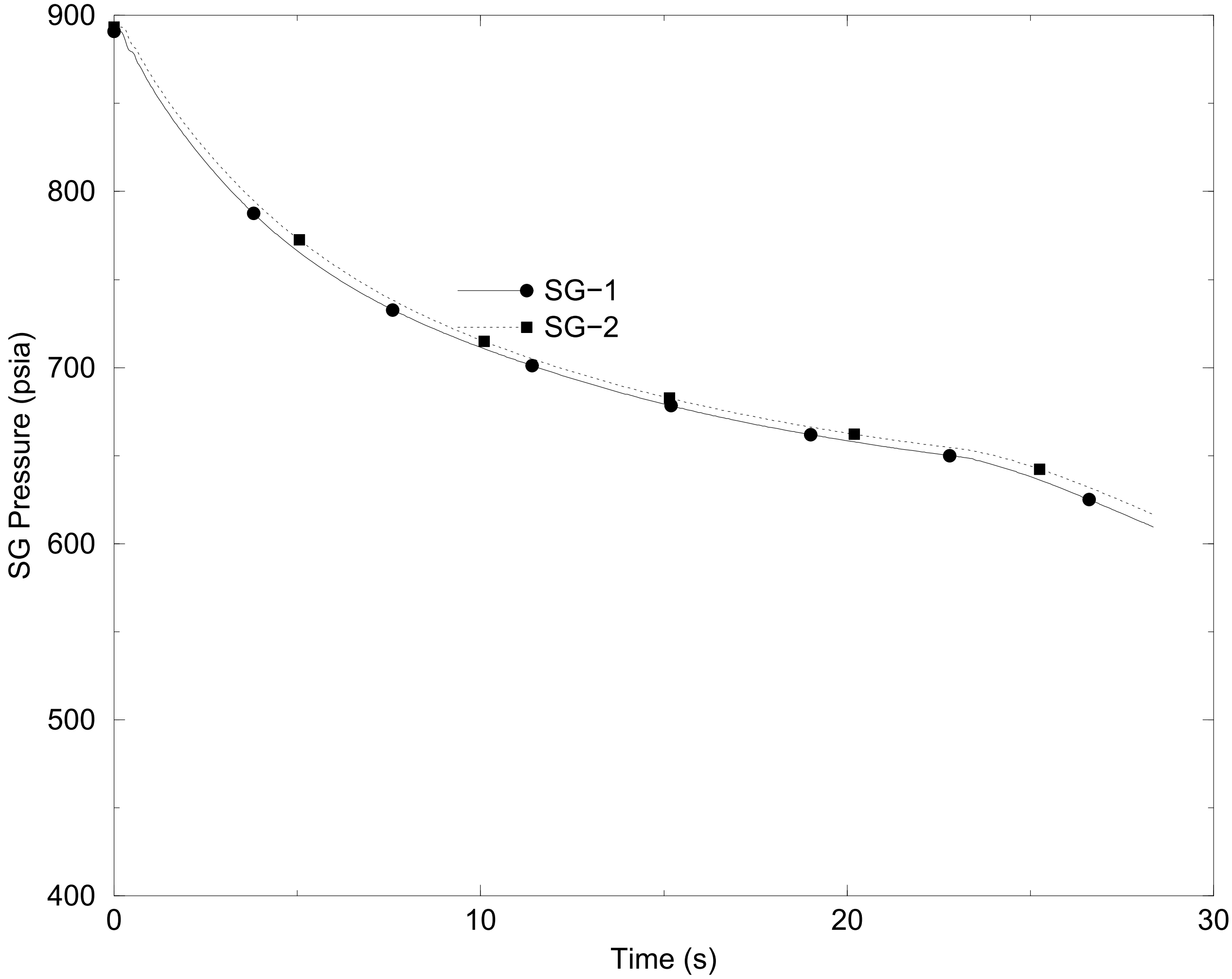
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP PRESSURIZER PRESSURE VS TIME	Figure 14.14-10 Revision 44



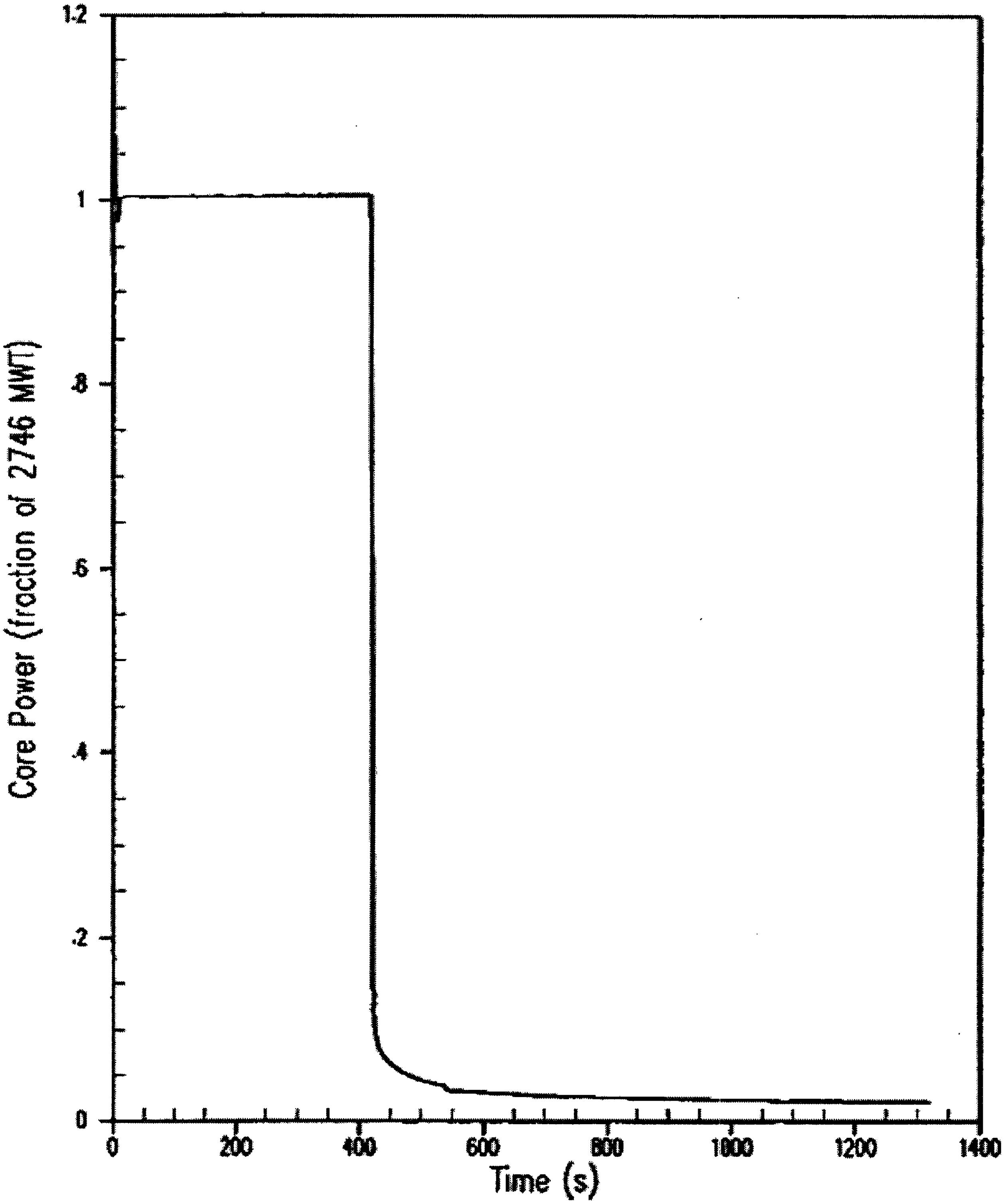
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP RCS TEMPERATURES VS TIME	Figure 14.14-11 Revision 44



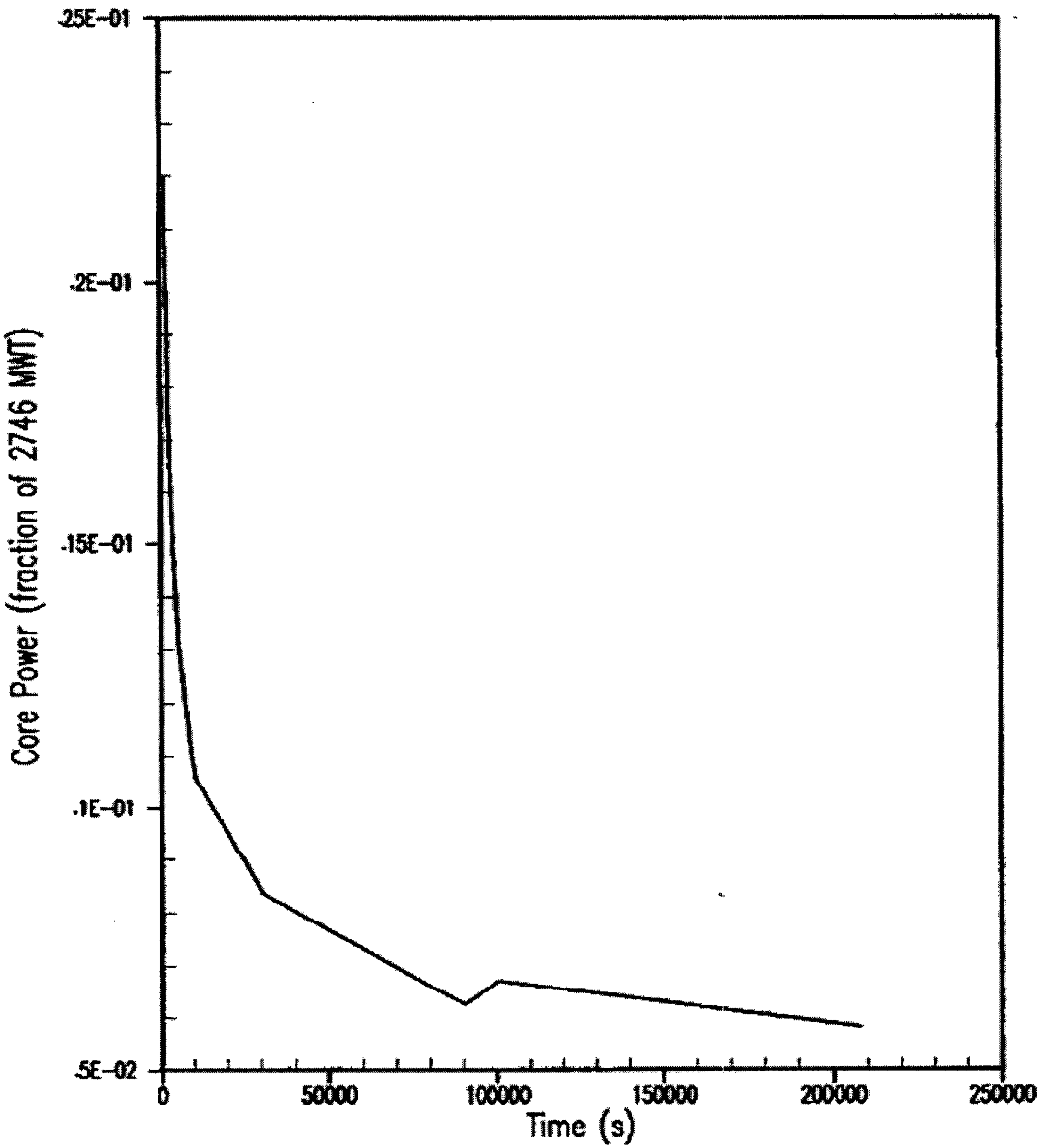
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP REACTIVITIES VS TIME	Figure 14.14-12 Revision 44



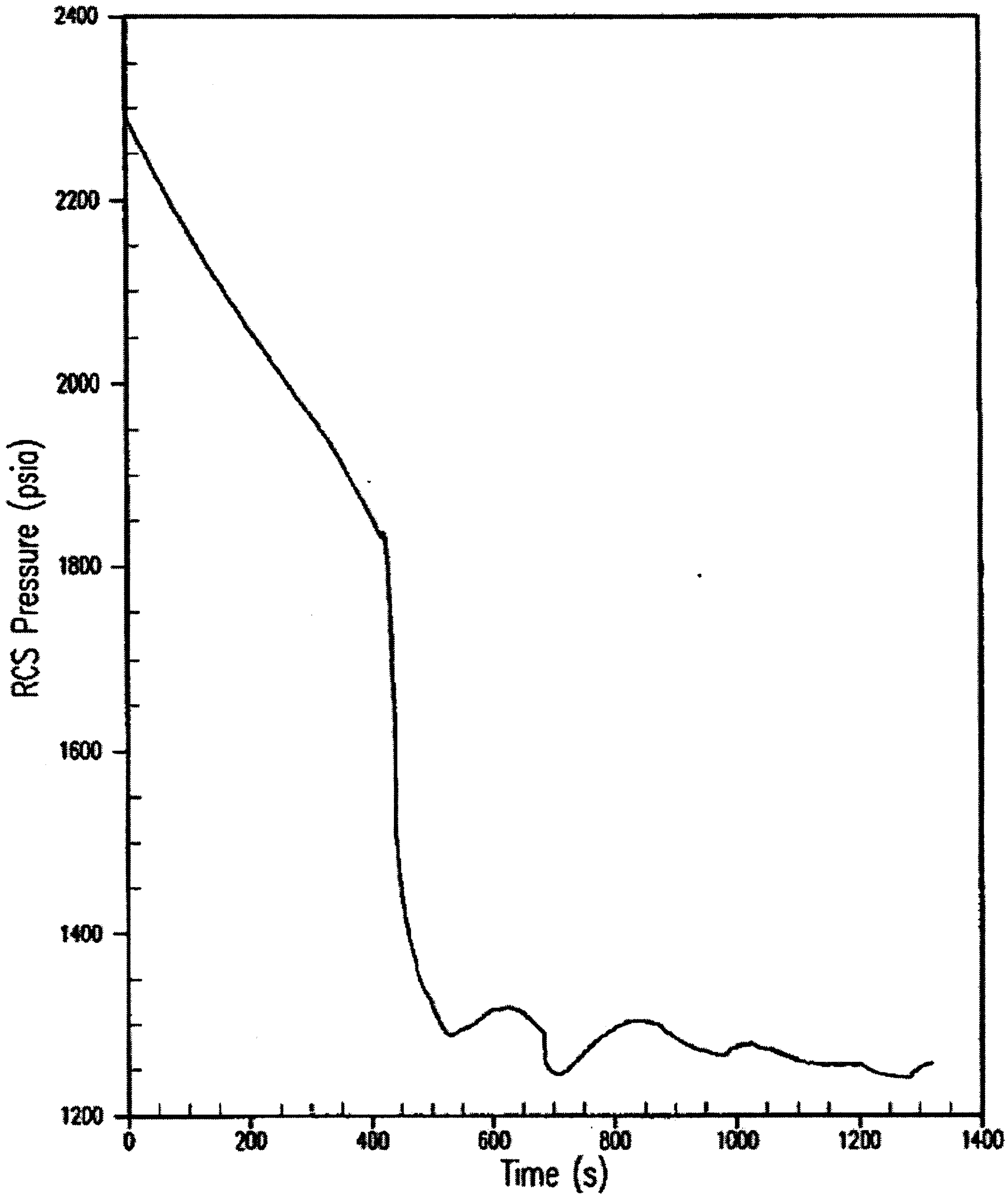
Calvert Cliffs Nuclear Power Plant	SLB EVENT PRE-TRIP SG PRESSURES VS TIME	Figure 14.14-13 Revision 44



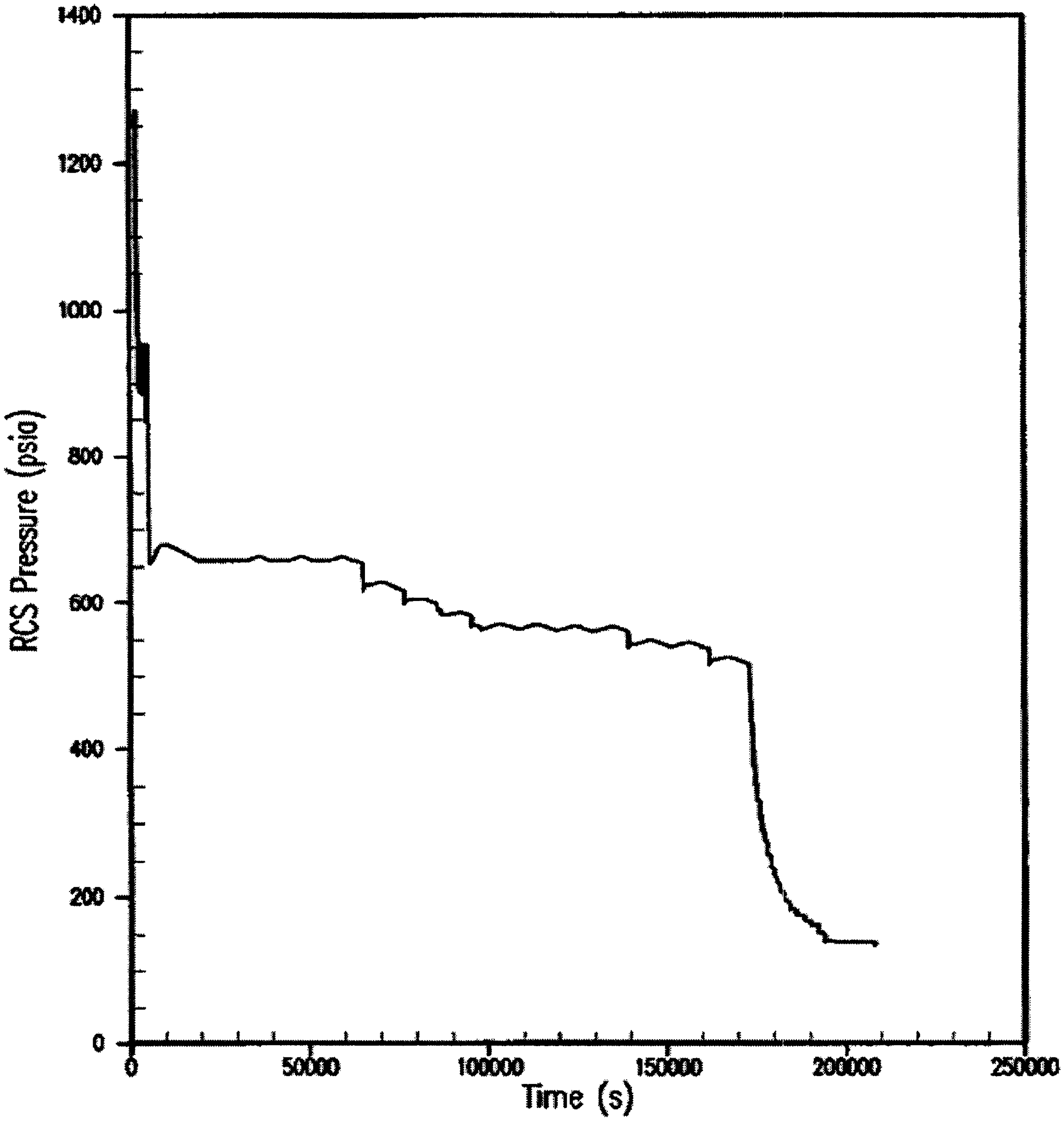
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS CORE POWER VS TIME (Sheet 1)	Figure 14.15-1 Rev. 43



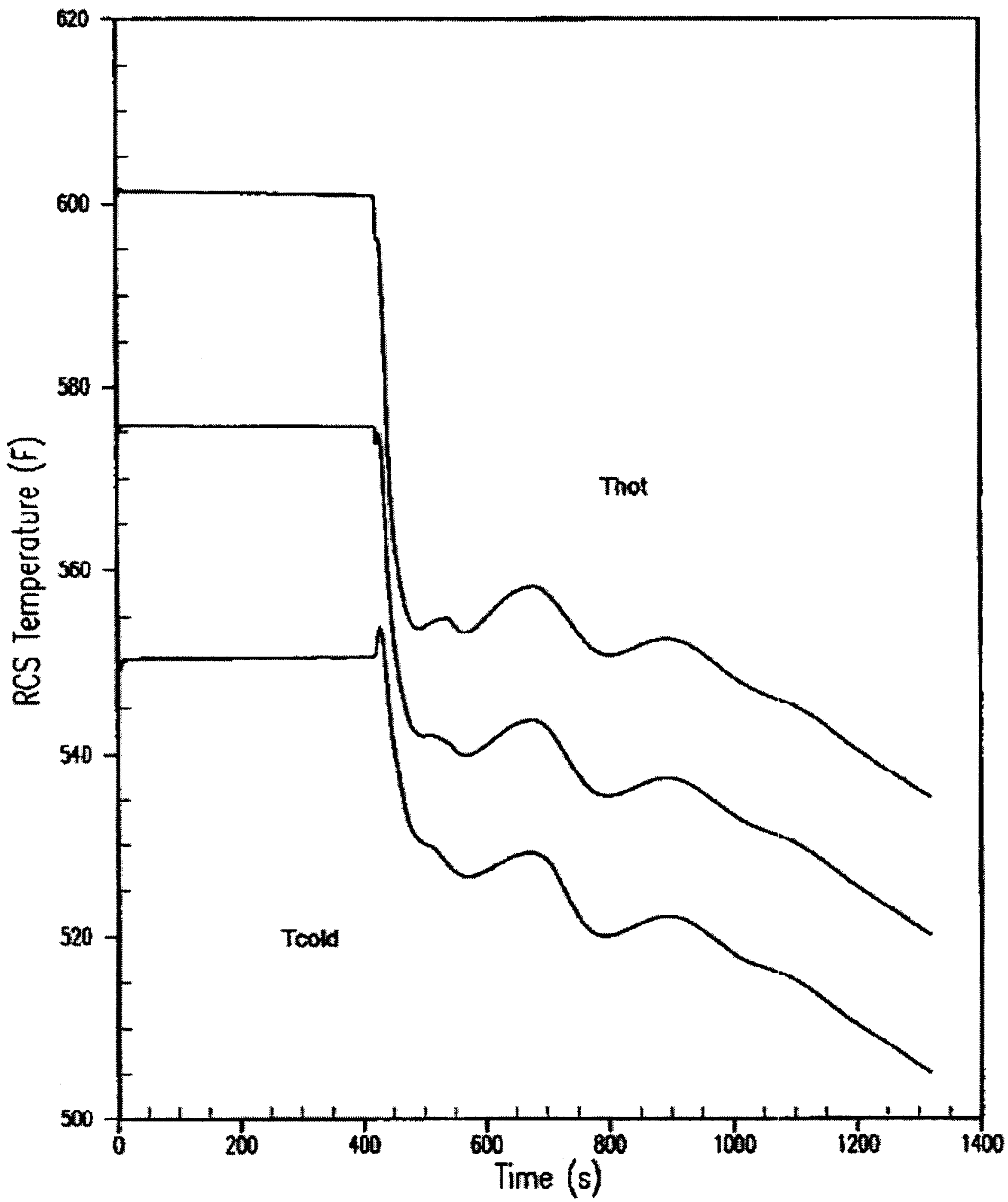
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS CORE POWER VS TIME (Sheet 2)	Figure 14.15-1 Rev. 43



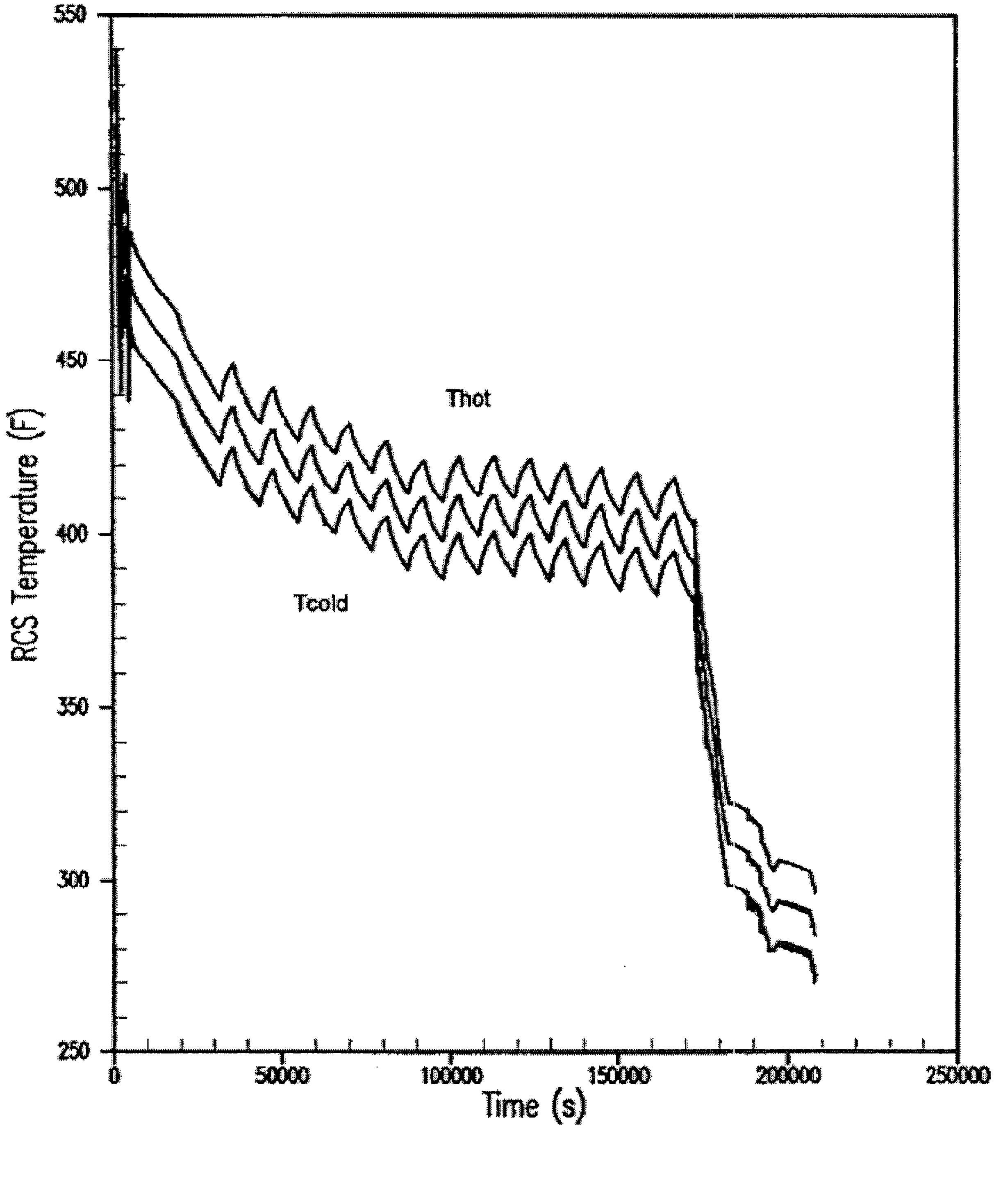
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS REACTOR COOLANT SYSTEM PRESSURE VS TIME (Sheet 1)	Figure 14.15-2 Rev. 43



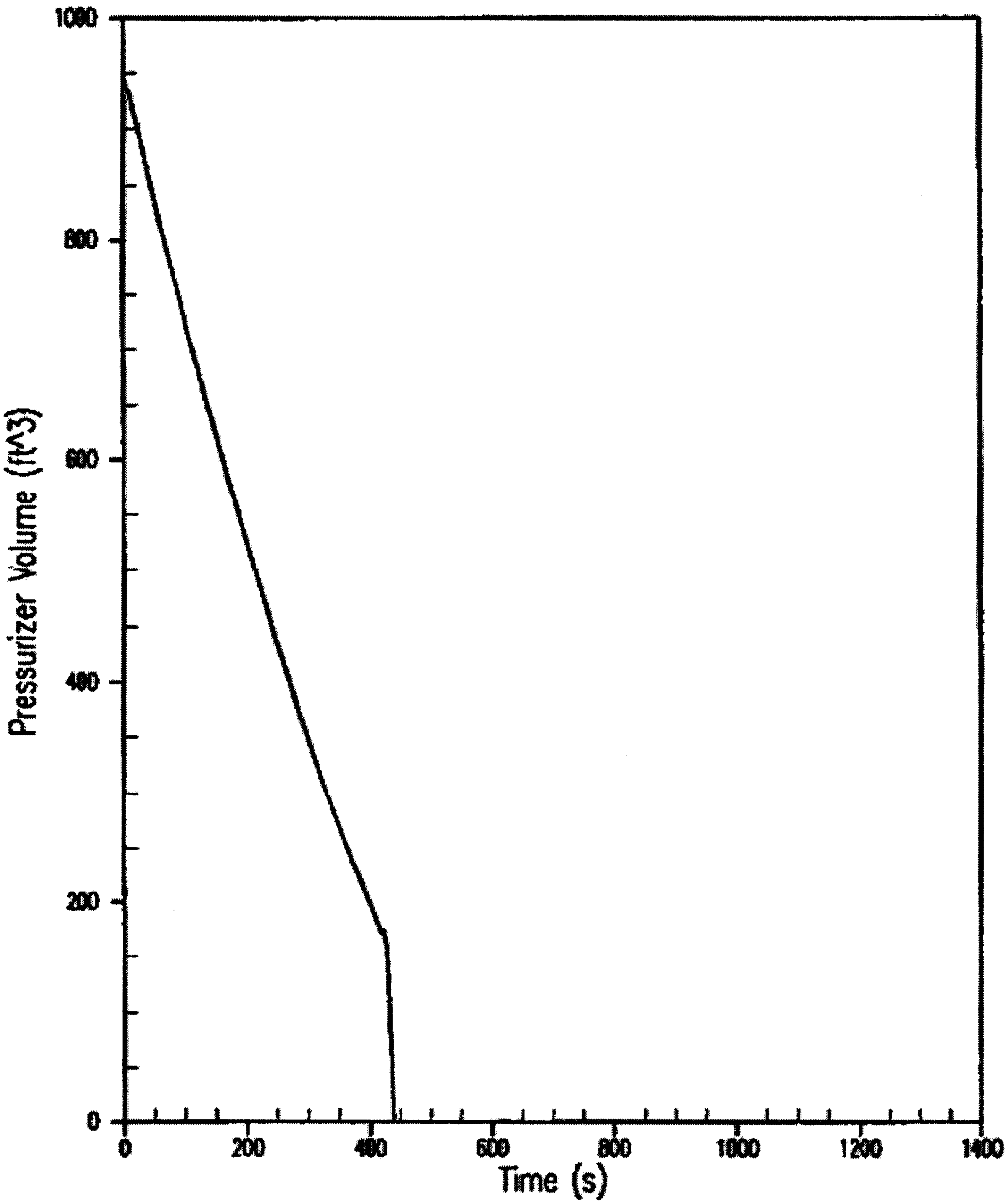
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS REACTOR COOLANT SYSTEM PRESSURE VS TIME (Sheet 2)	Figure 14.15-2 Rev. 43



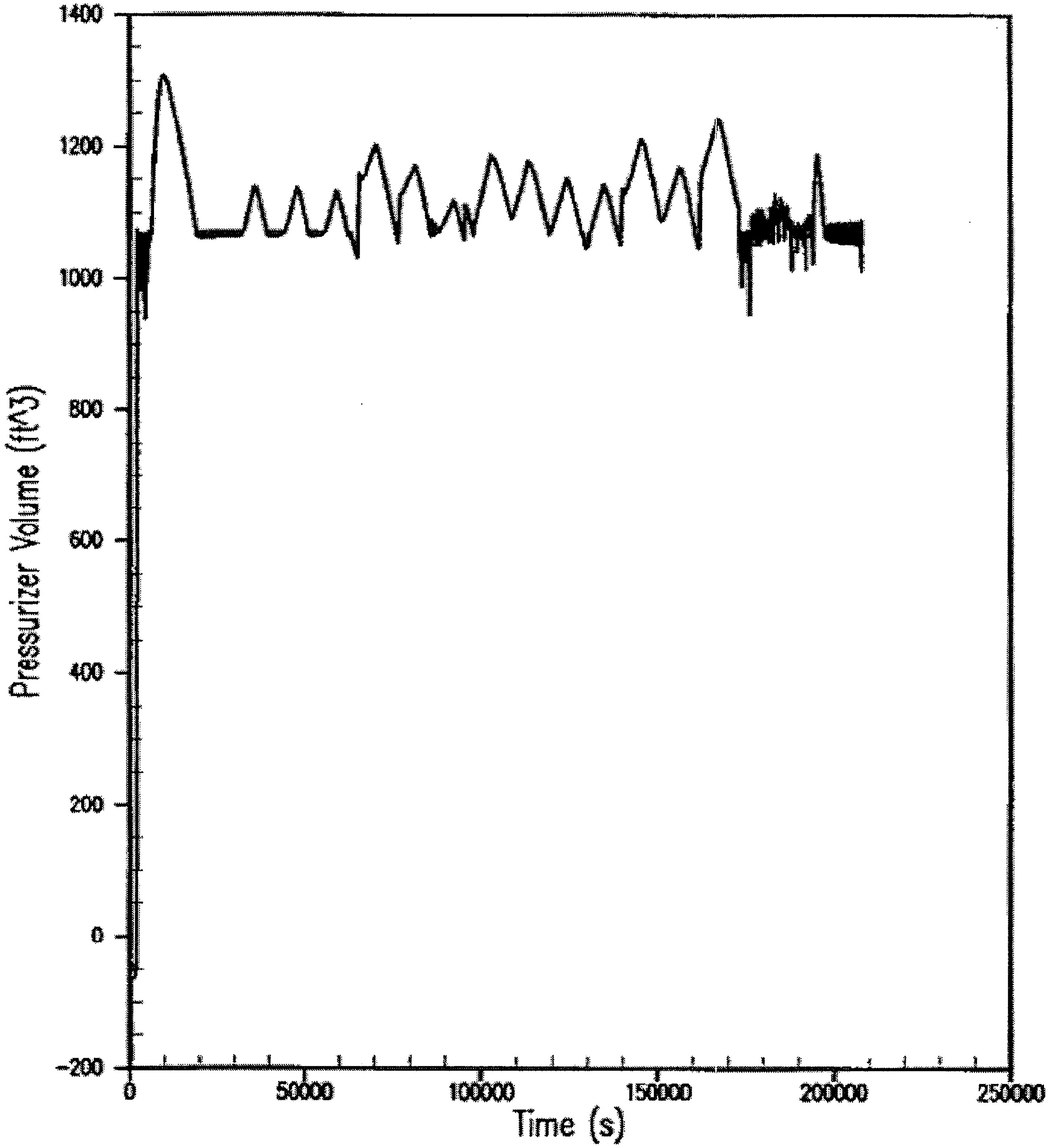
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS CORE COOLANT TEMPERATURE VS TIME (Sheet 1)	Figure 14.15-3 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS CORE COOLANT TEMPERATURE VS TIME (Sheet 2)	Figure 14.15-3 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS PRESSURIZER WATER VOLUME VS TIME (Sheet 1)	Figure 14.15-4 Rev. 43



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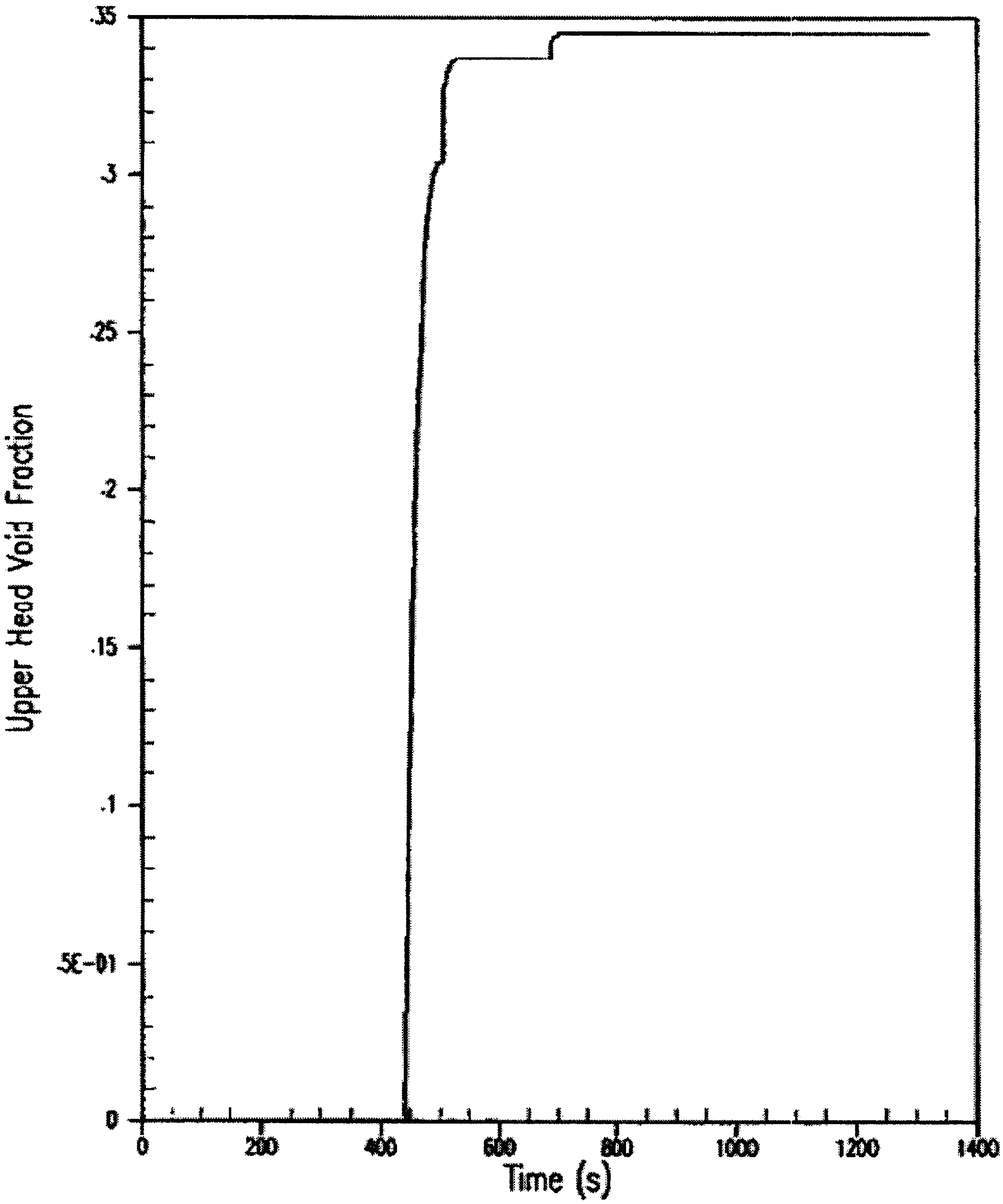
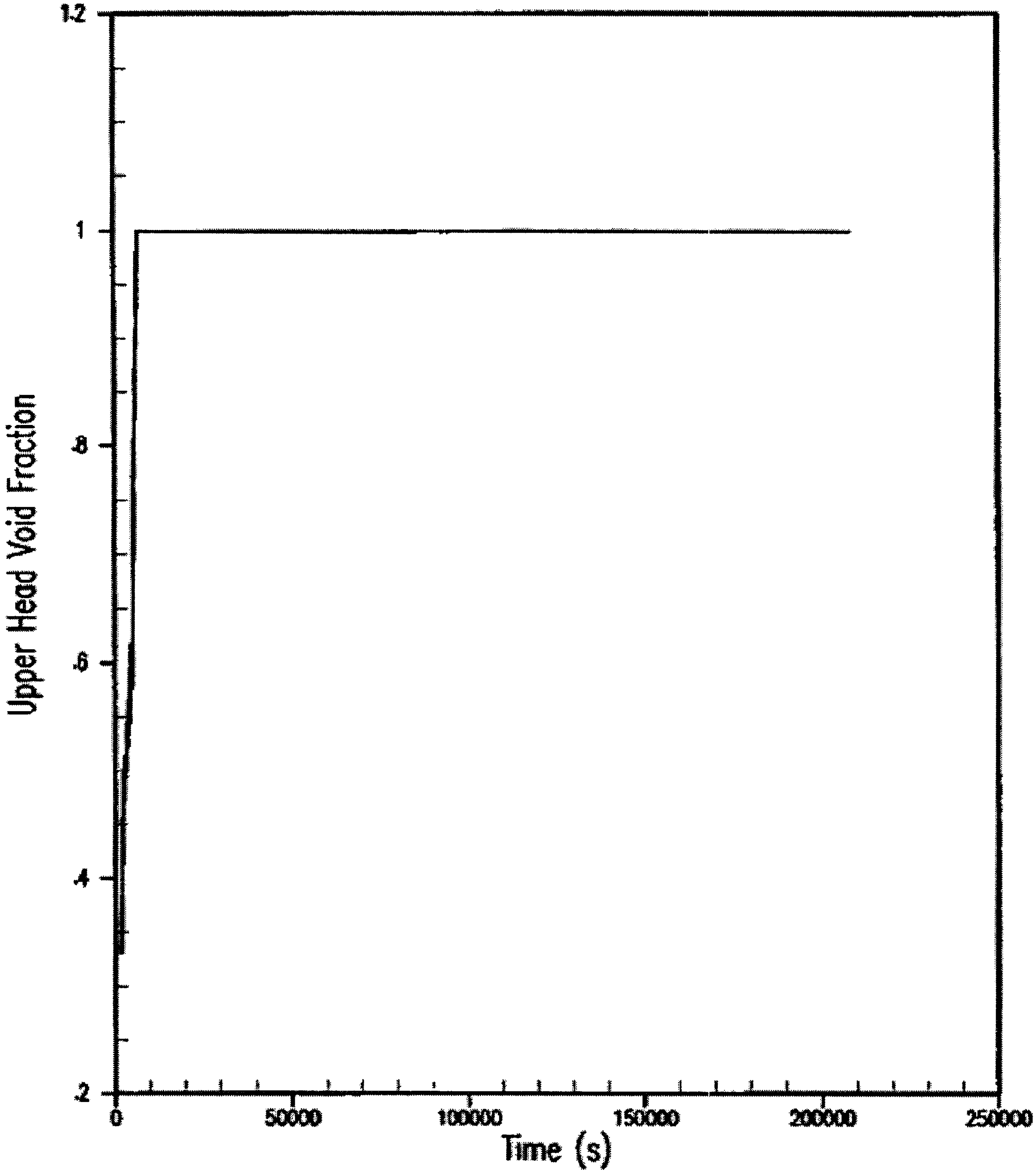


Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS UPPER HEAD VOID FRACTION VS TIME (Sheet 1)	Figure 14.15-5 Rev. 43



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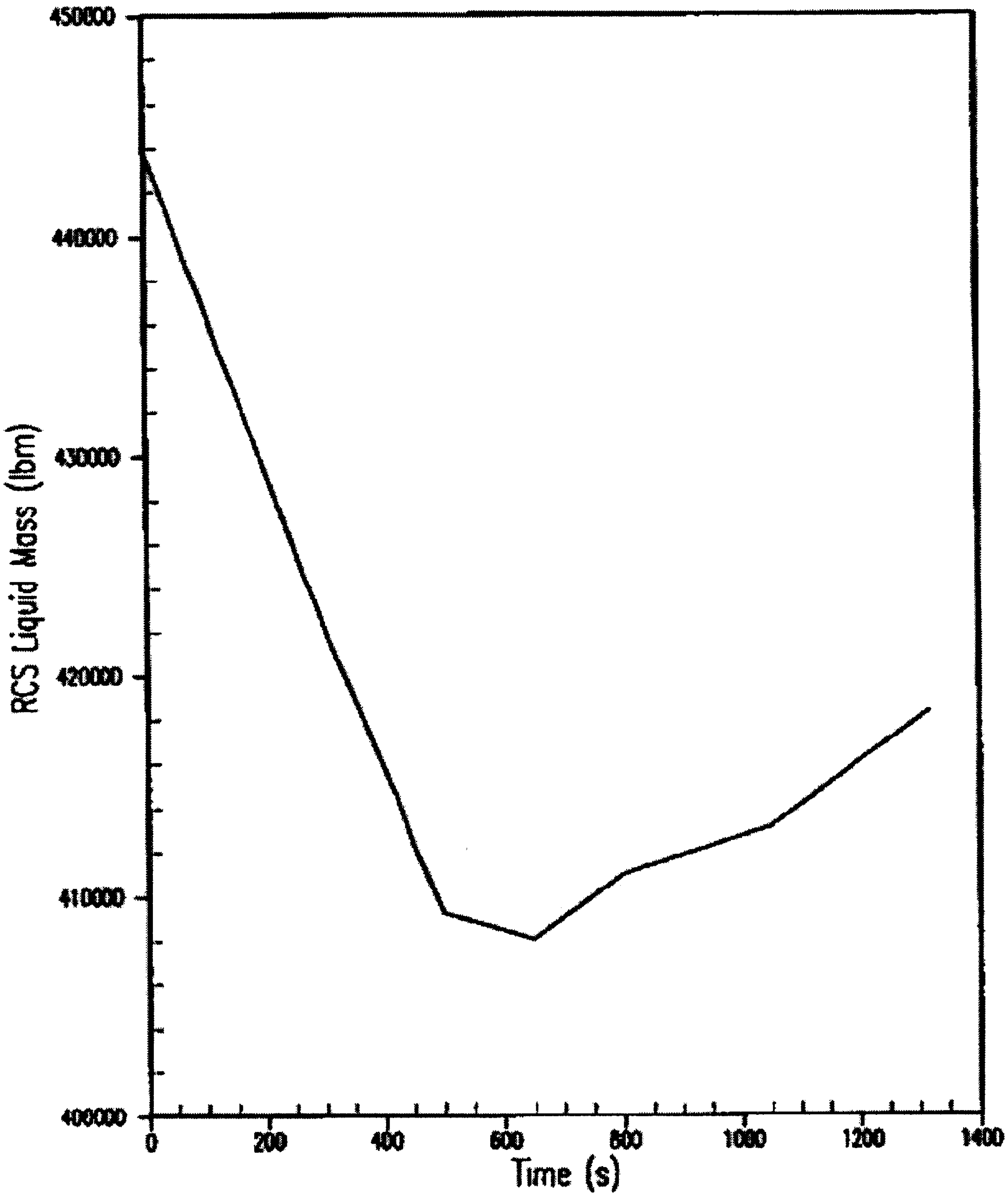
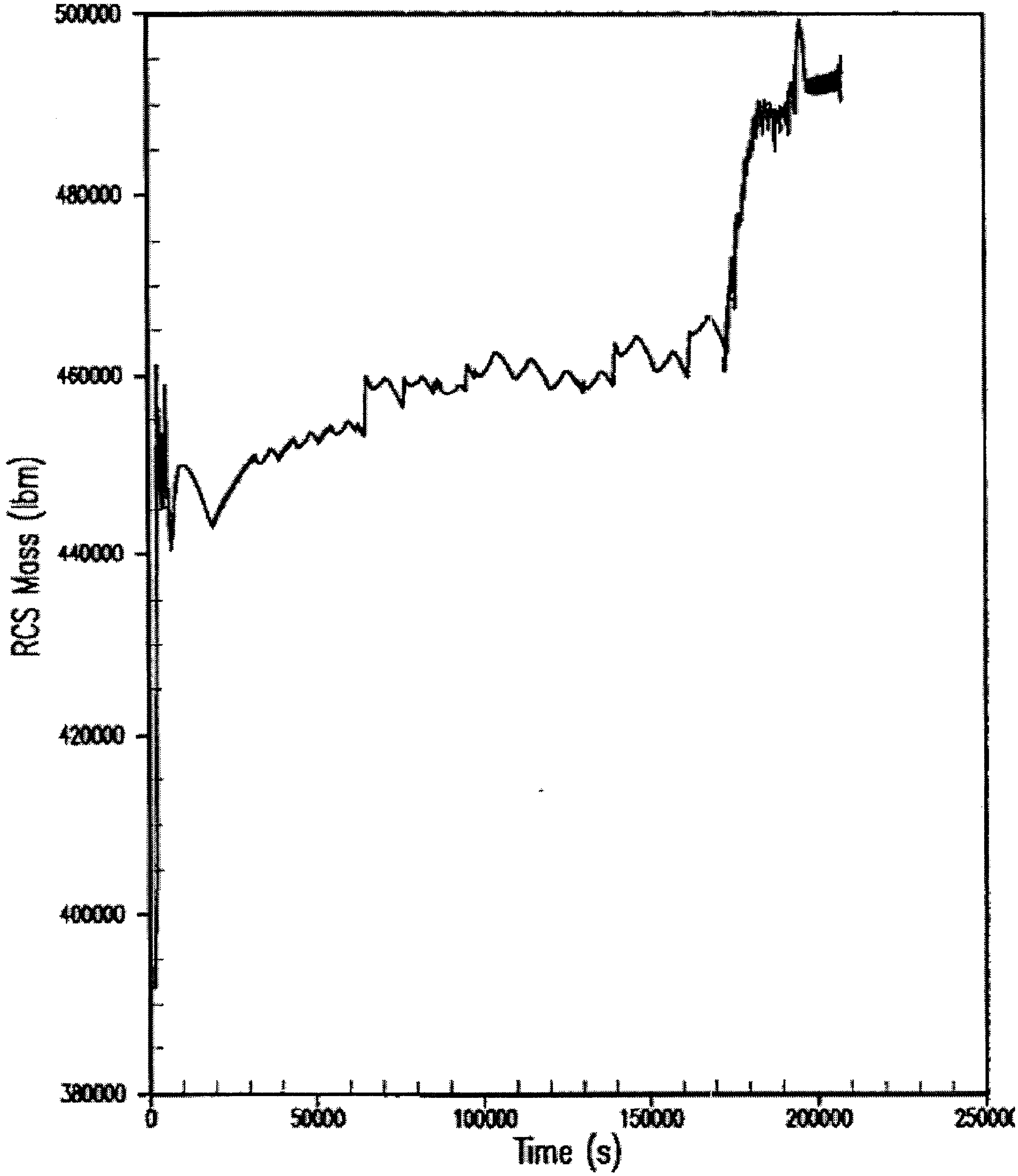
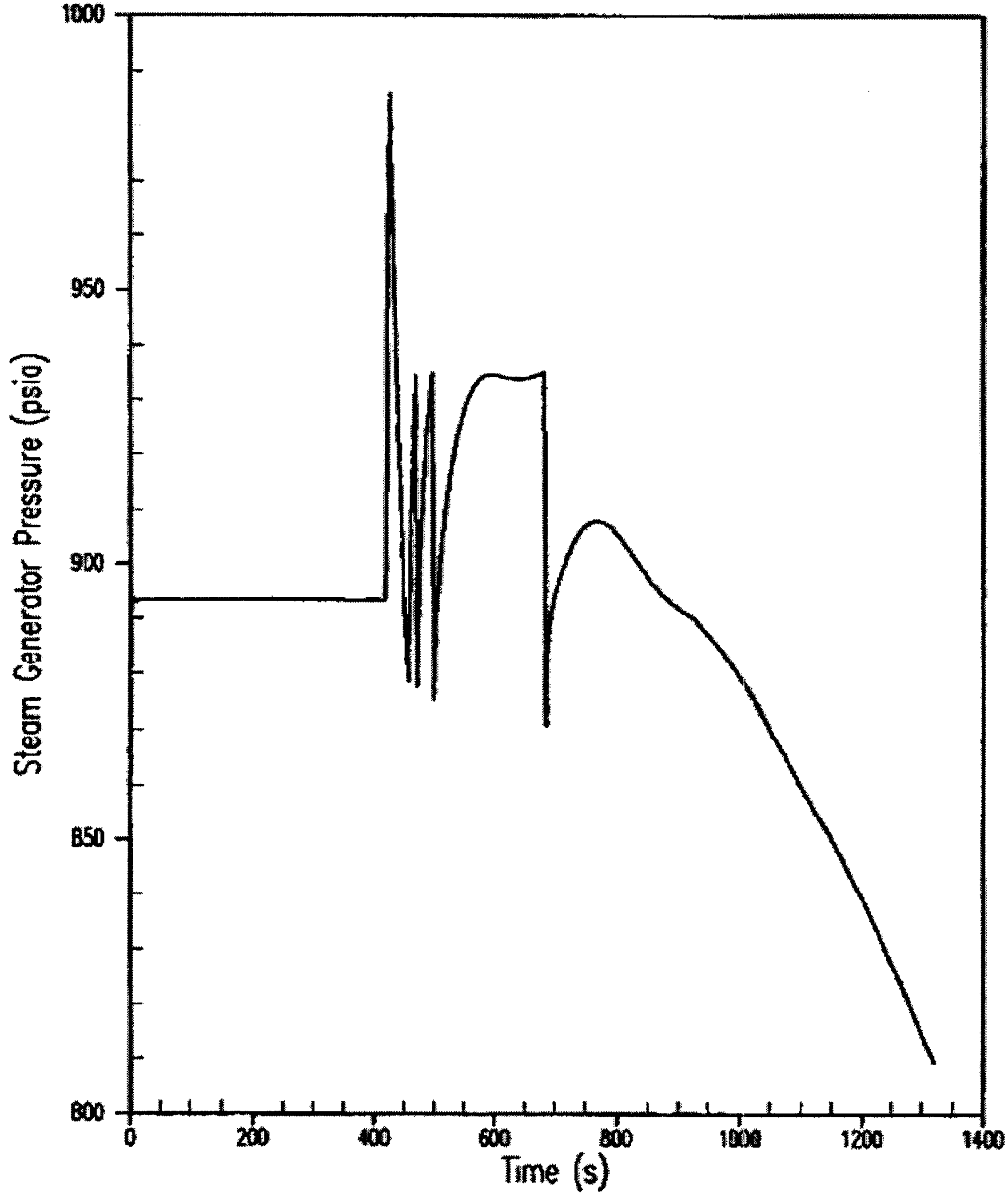


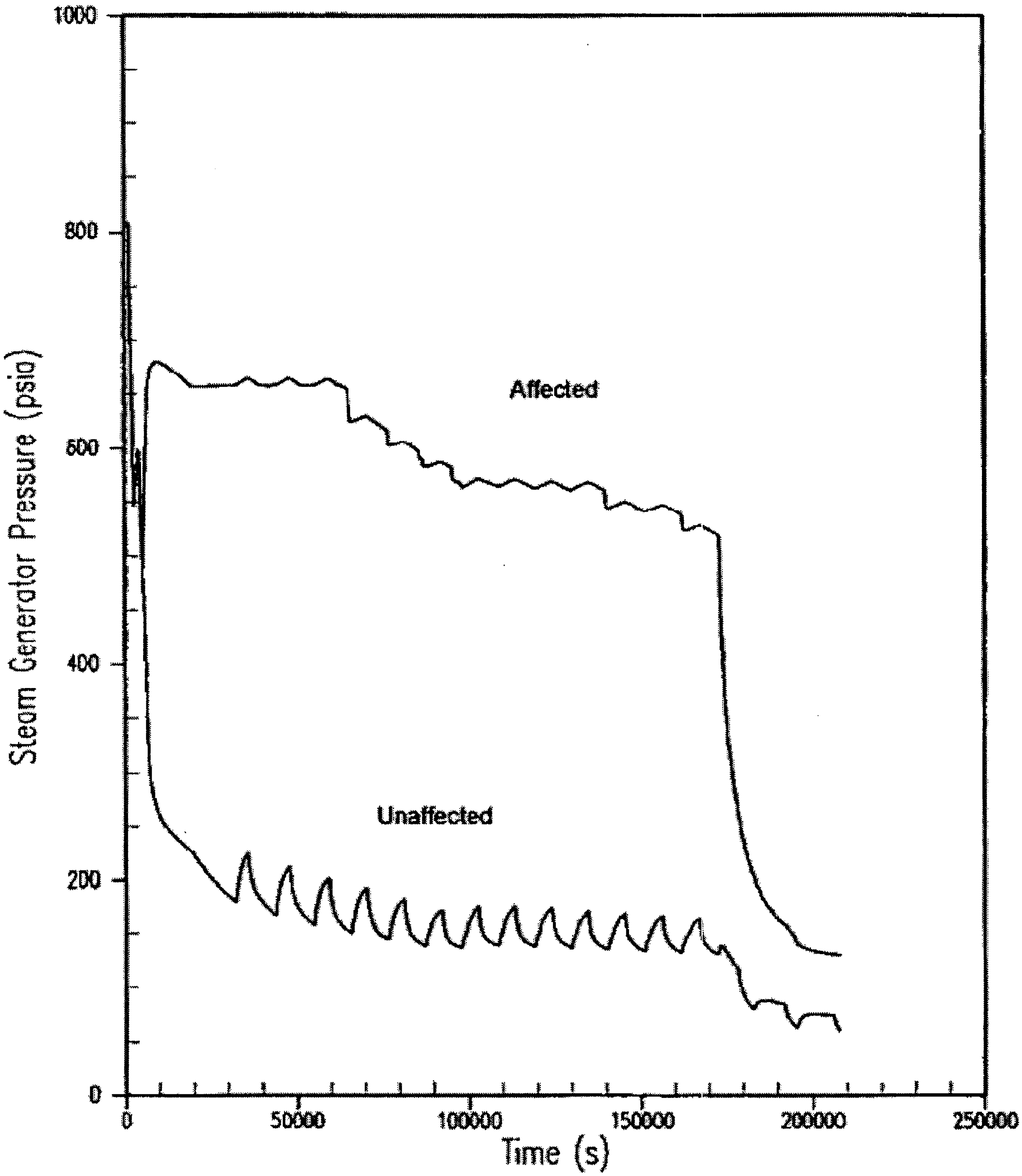
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS RCS LIQUID MASS VS TIME (Sheet 1)	Figure 14.15-6 Rev. 43



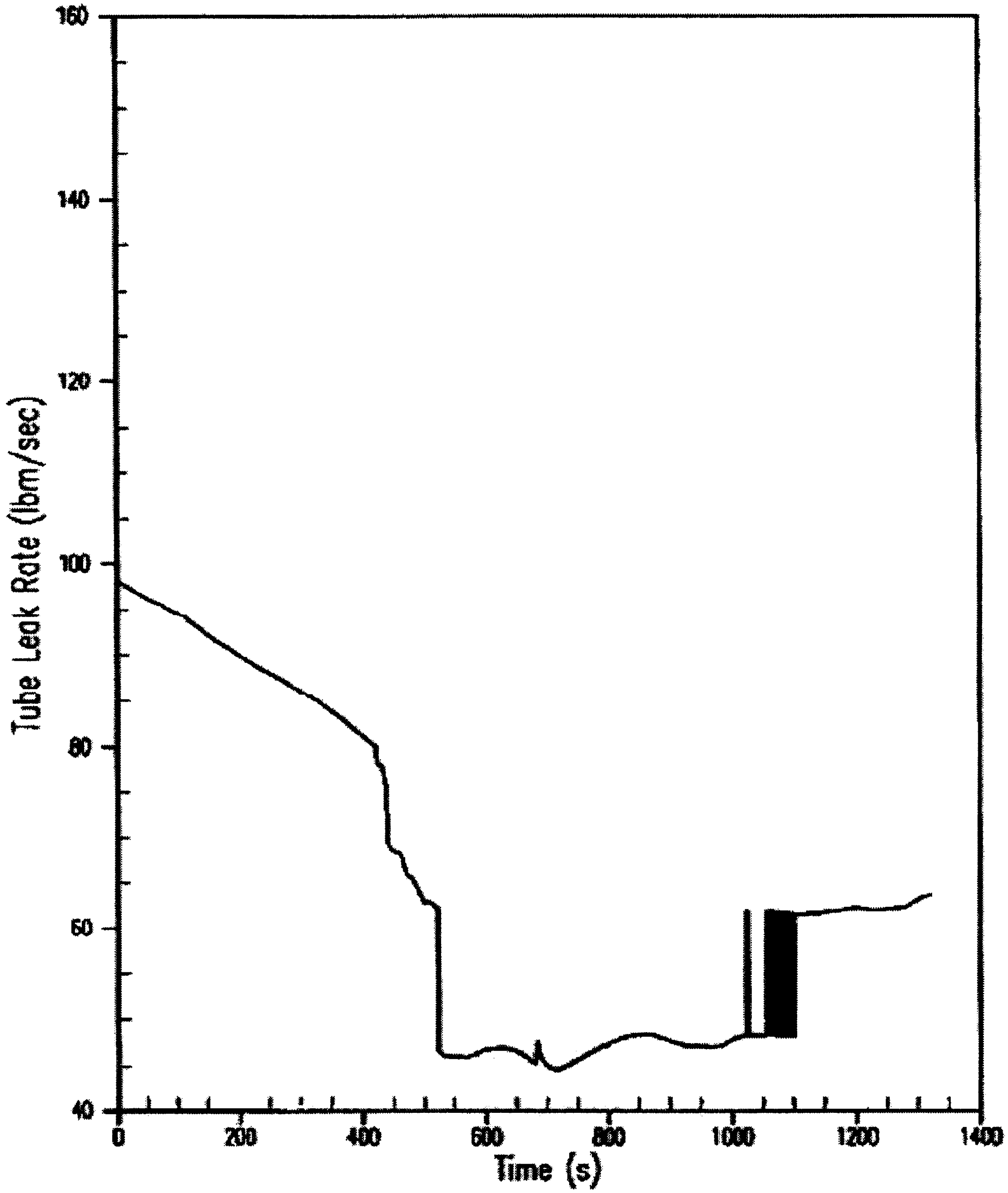
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS RCS LIQUID MASS VS TIME (Sheet 2)	Figure 14.15-6 Rev. 43



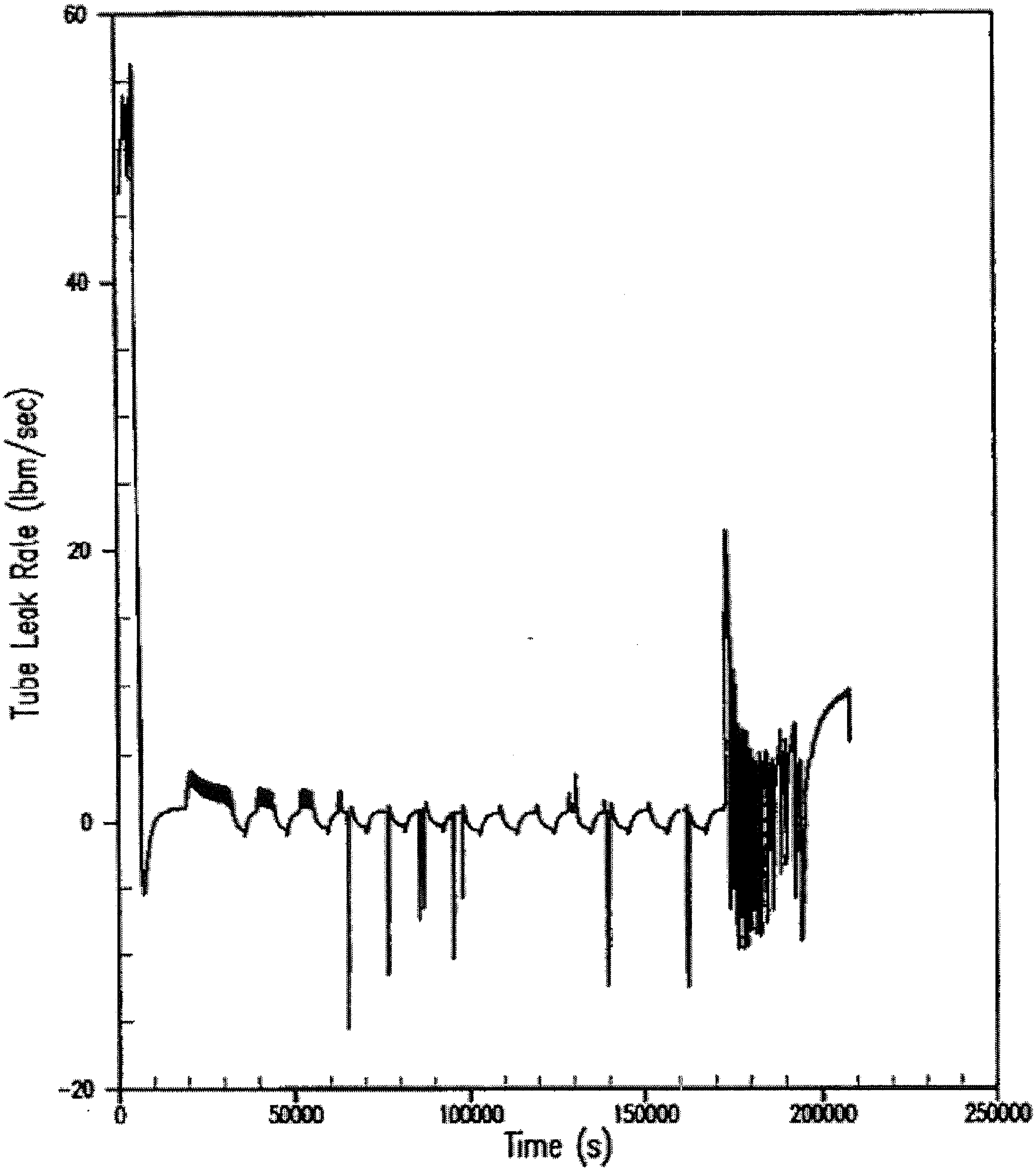
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS STEAM GENERATOR PRESSURE VS TIME (Sheet 1)	Figure 14.15-7 Rev. 43



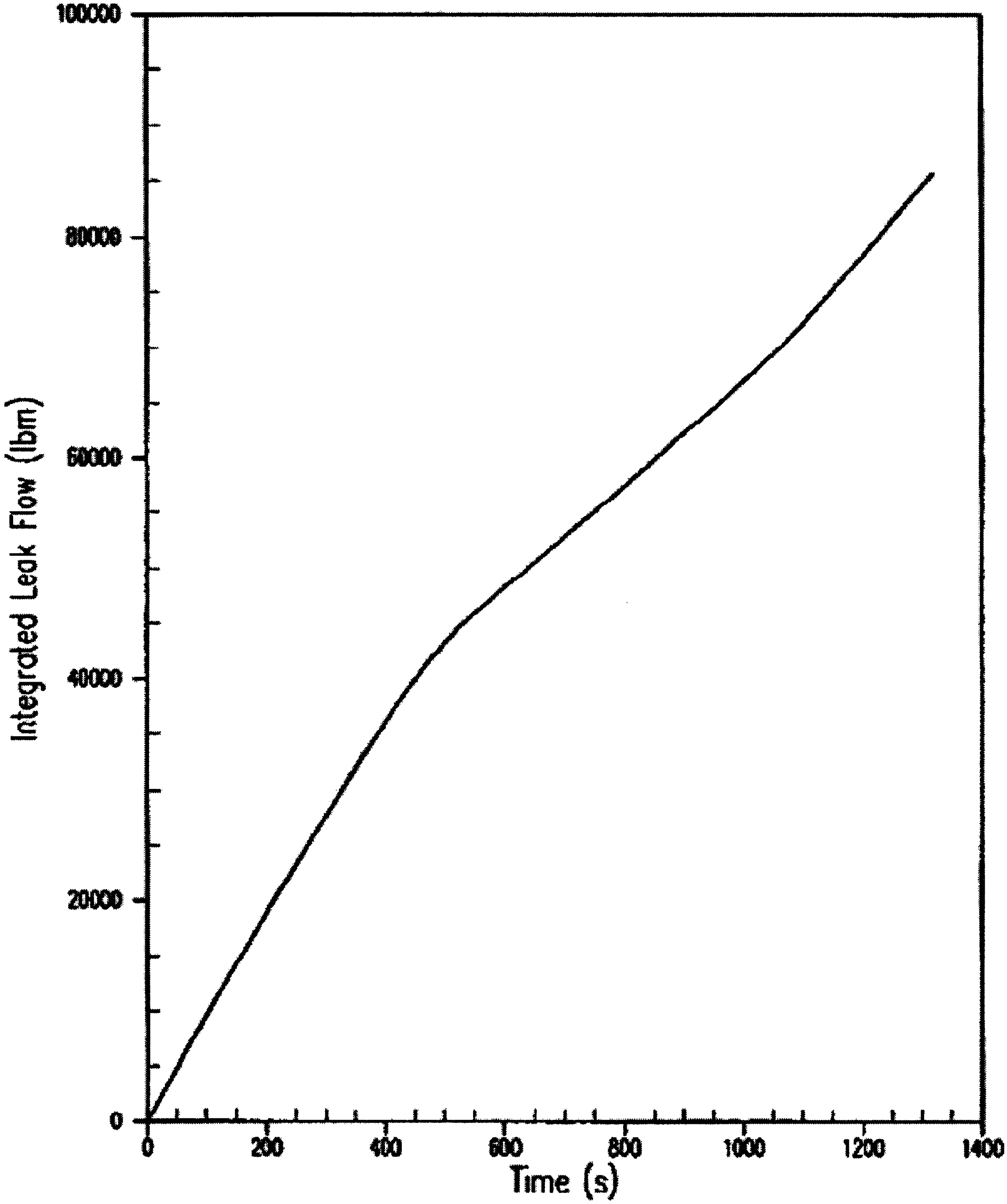
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS STEAM GENERATOR PRESSURE VS TIME (Sheet 2)	Figure 14.15-7 Rev. 43



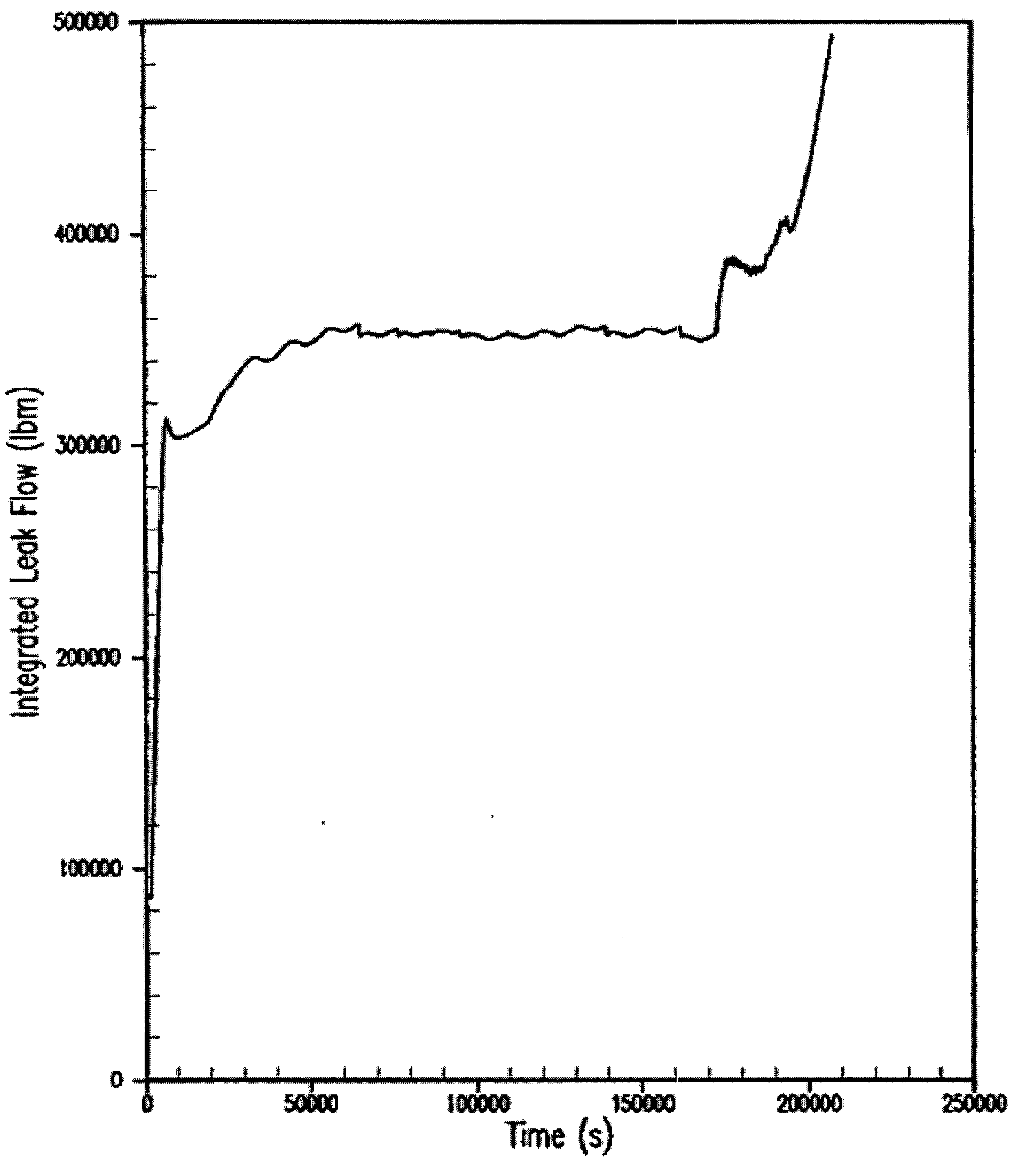
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS TUBE LEAK RATE VS TIME (Sheet 1)	Figure 14.15-8 Rev. 43



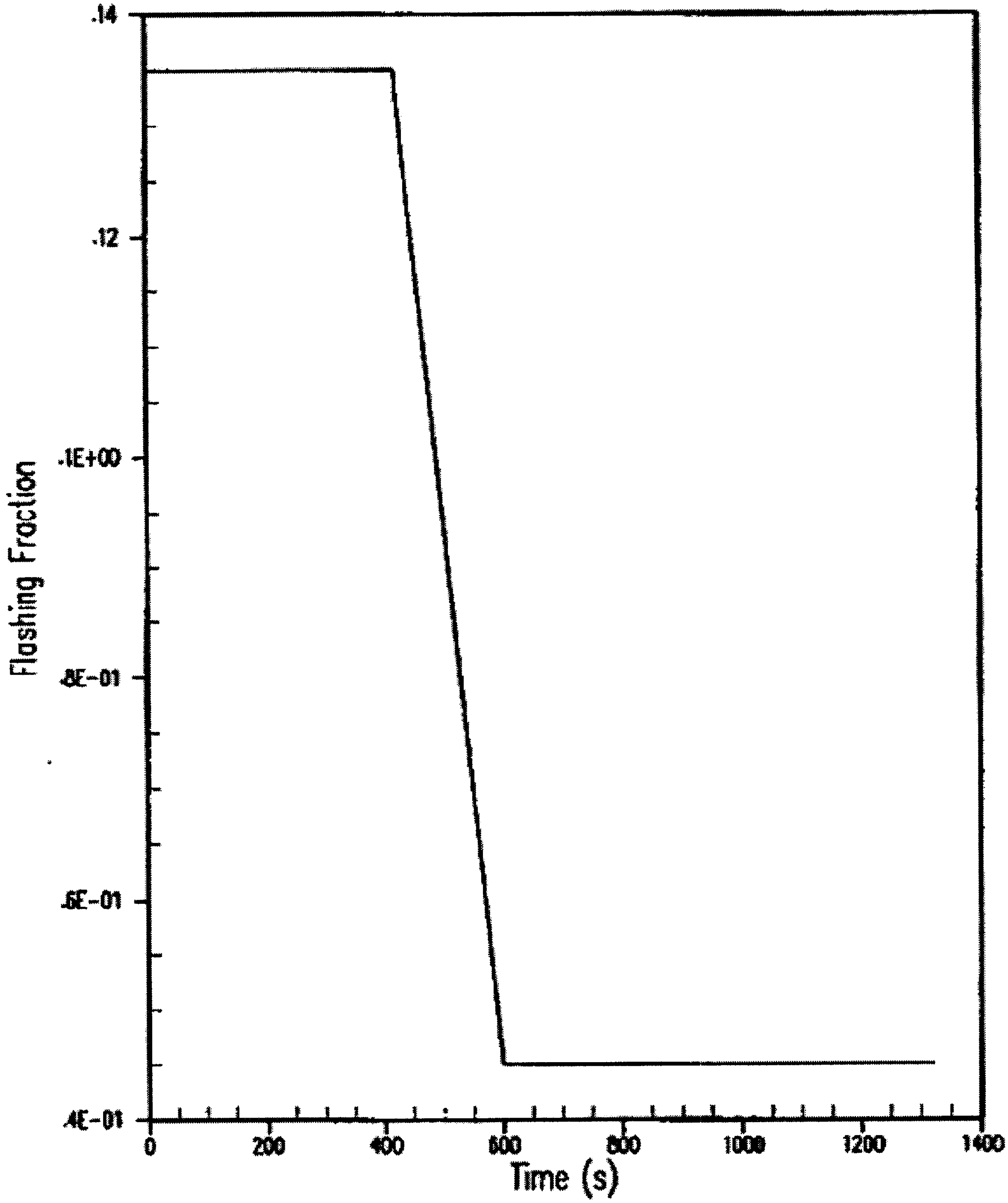
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS TUBE LEAK RATE VS TIME (Sheet 2)	Figure 14.15-8 Rev. 43



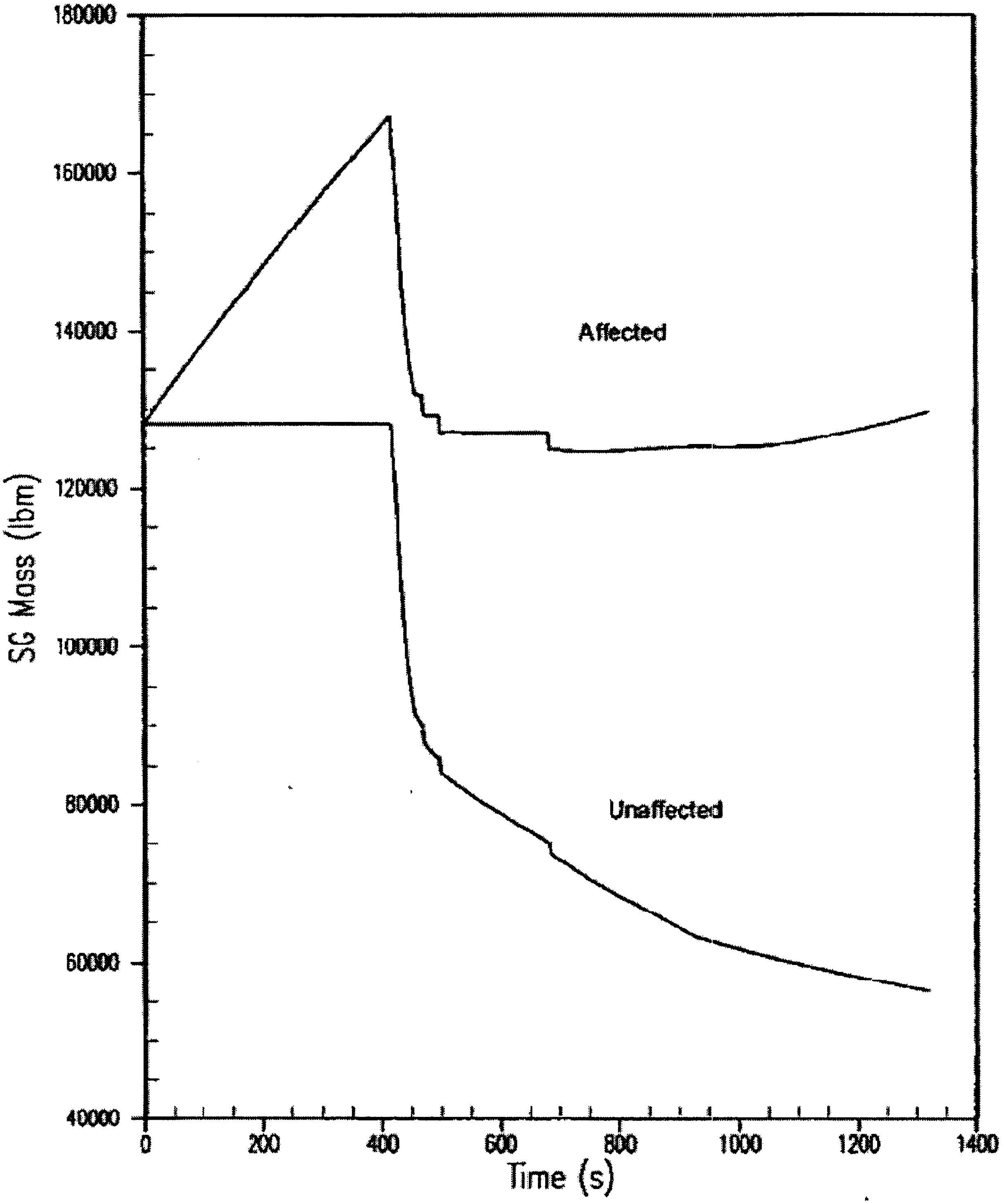
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED LEAK FLOW VS TIME (Sheet 1)	Figure 14.15-9 Rev. 43



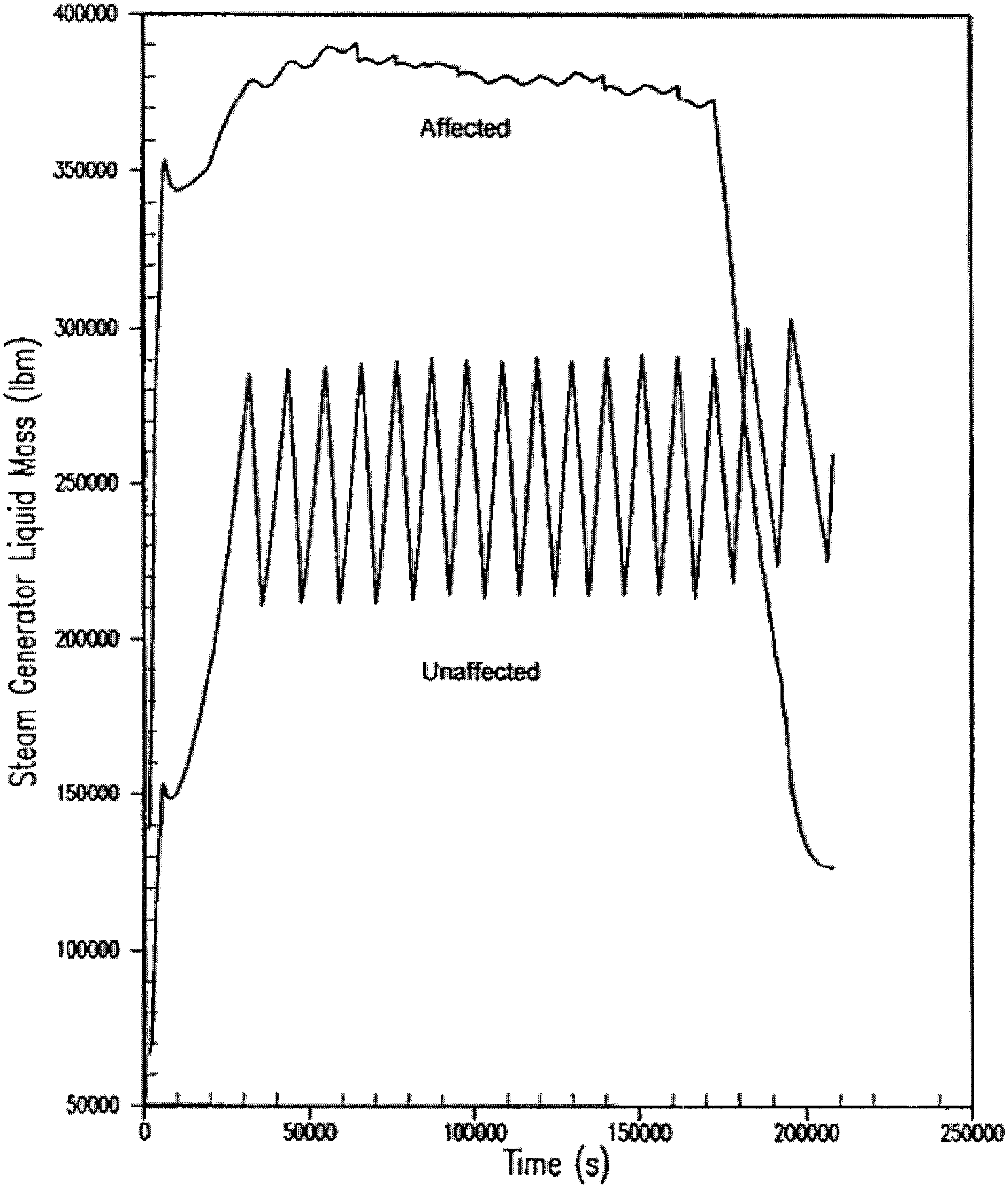
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED LEAK FLOW VS TIME (Sheet 2, Includes CESEC Portion of Event)	Figure 14.15-9 Rev. 43



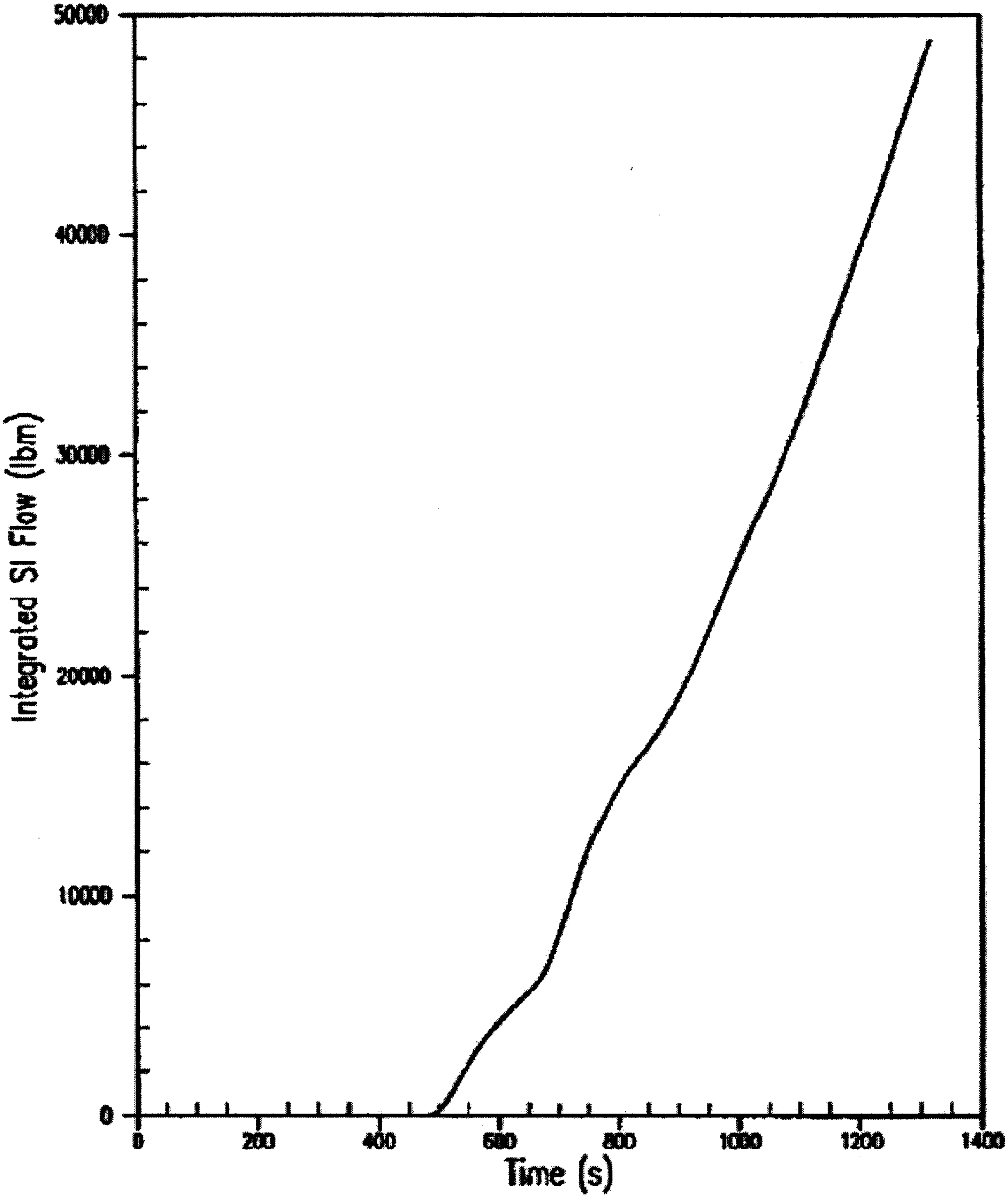
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS FLASHING FRACTION VS TIME	Figure 14.15-10 Rev. 43



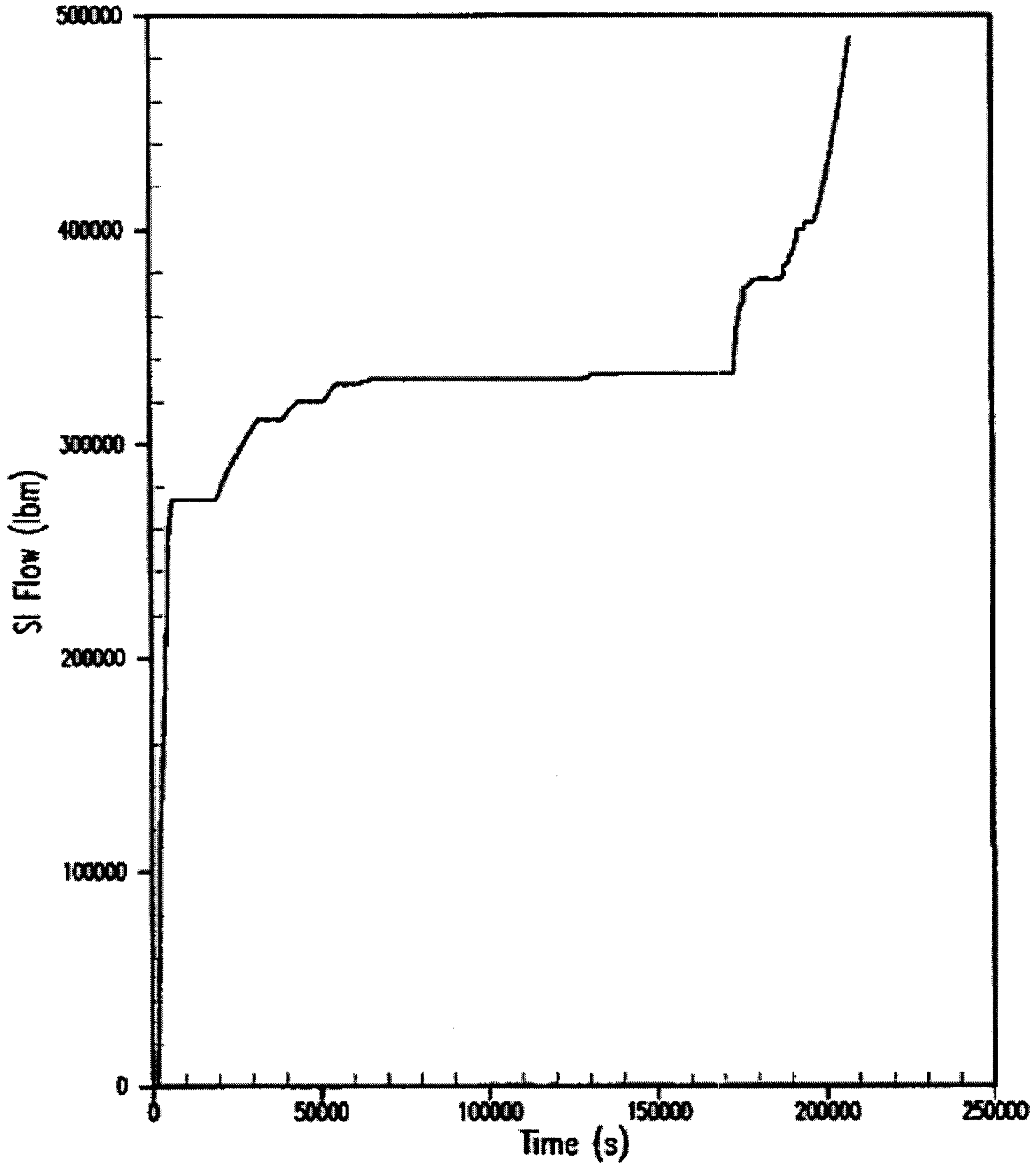
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS STEAM GENERATOR MASS VS TIME (Sheet 1)	Figure 14.15-11 Rev. 43



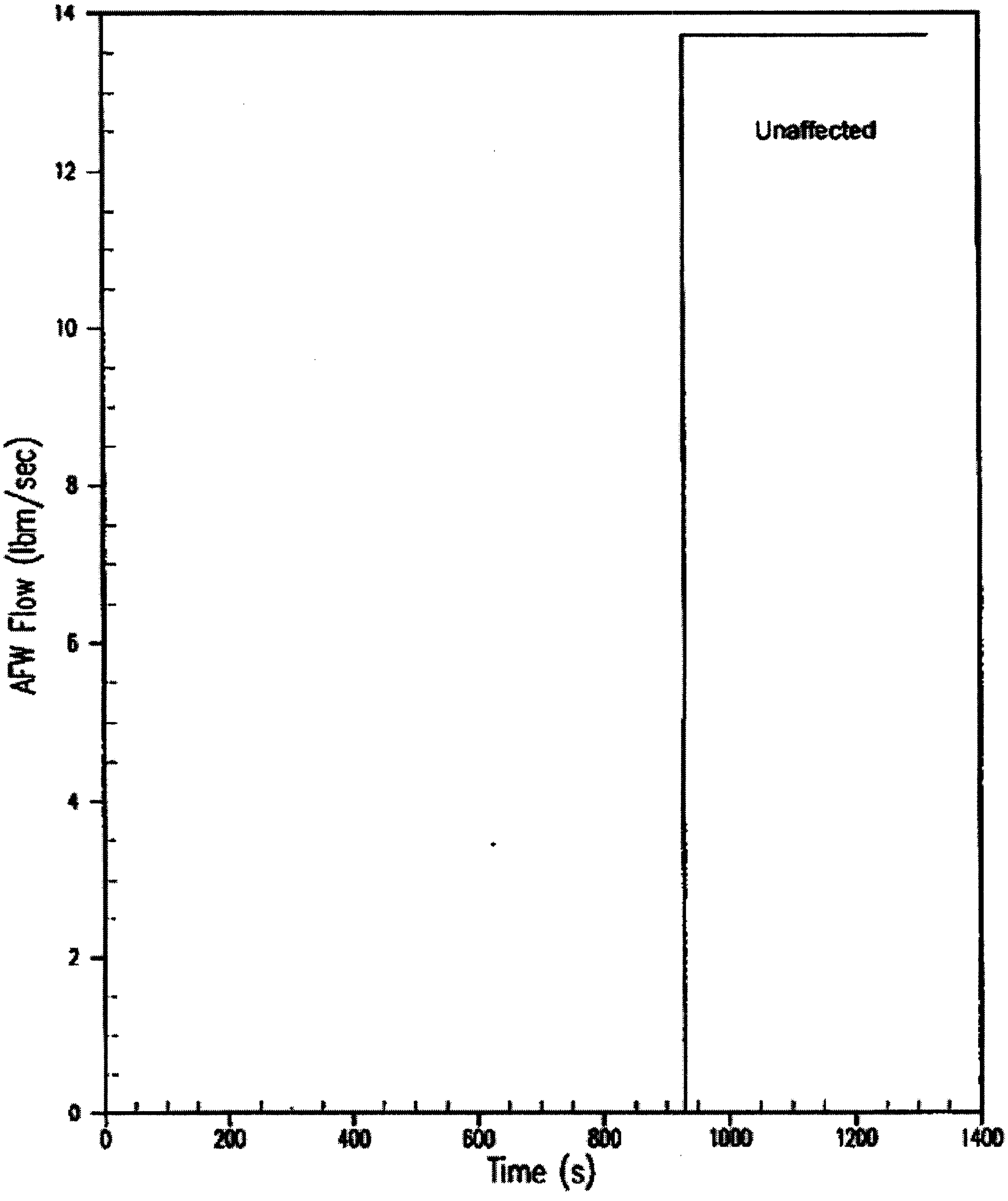
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS STEAM GENERATOR MASS VS TIME (Sheet 2)	Figure 14.15-11 Rev. 43



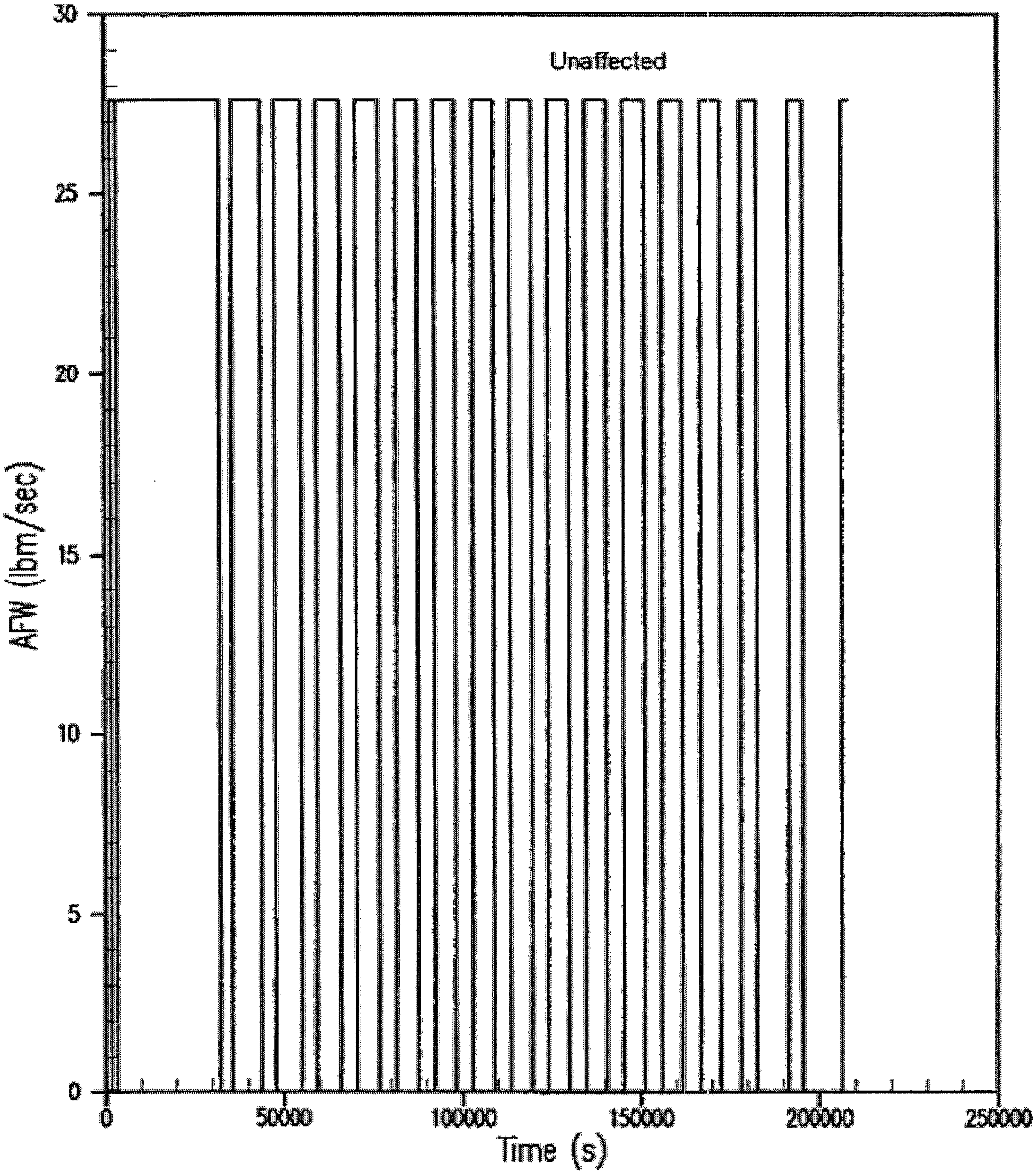
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED SAFETY INJECTION FLOW VS TIME (Sheet 1)	Figure 14.15-12 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED SAFETY INJECTION FLOW VS TIME (Sheet 2, Does Not Include CESEC Portion of Event)	Figure 14.15-12 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS AUXILIARY FEEDWATER FLOW VS TIME (Sheet 1)	Figure 14.15-13 Rev. 43



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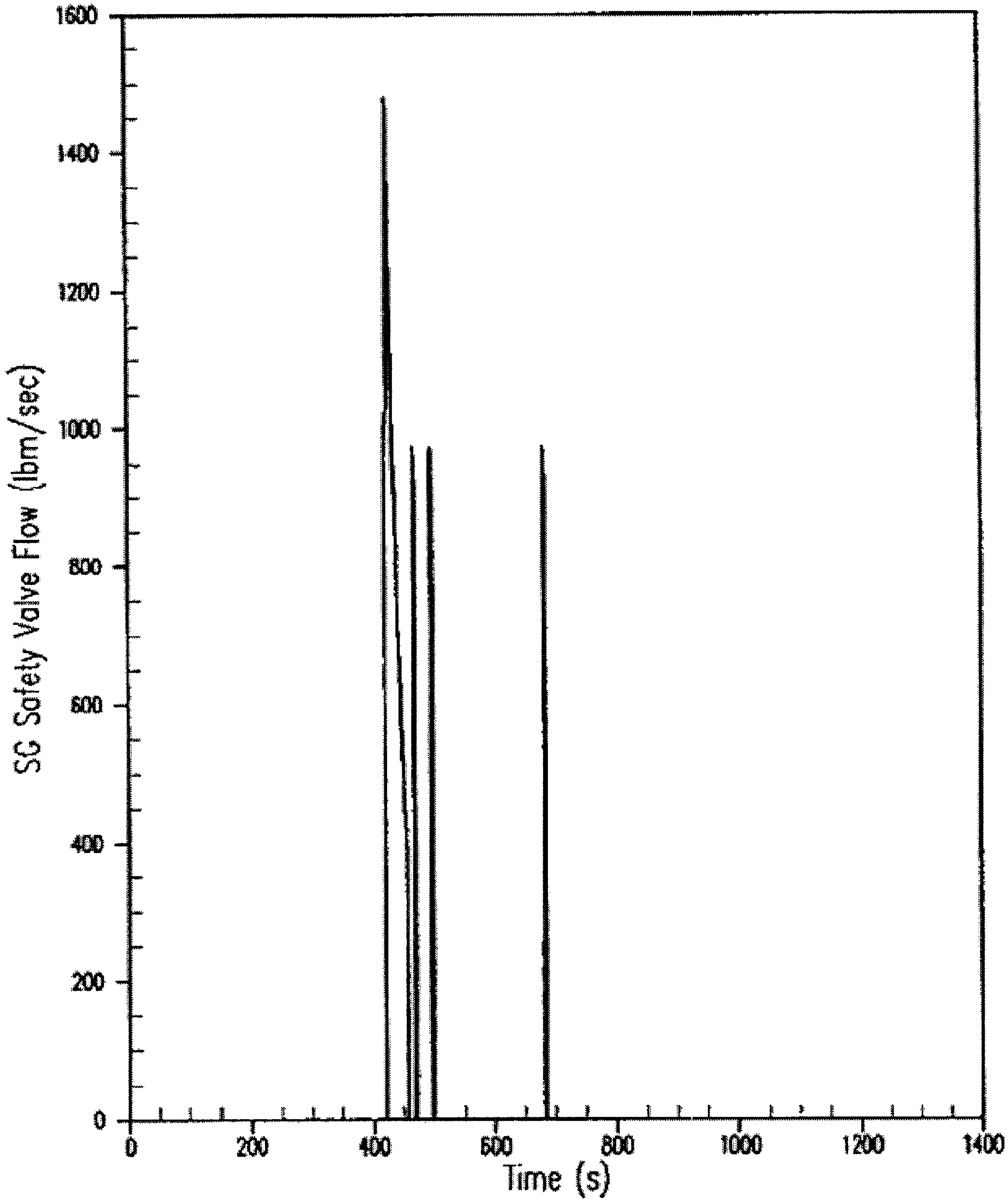

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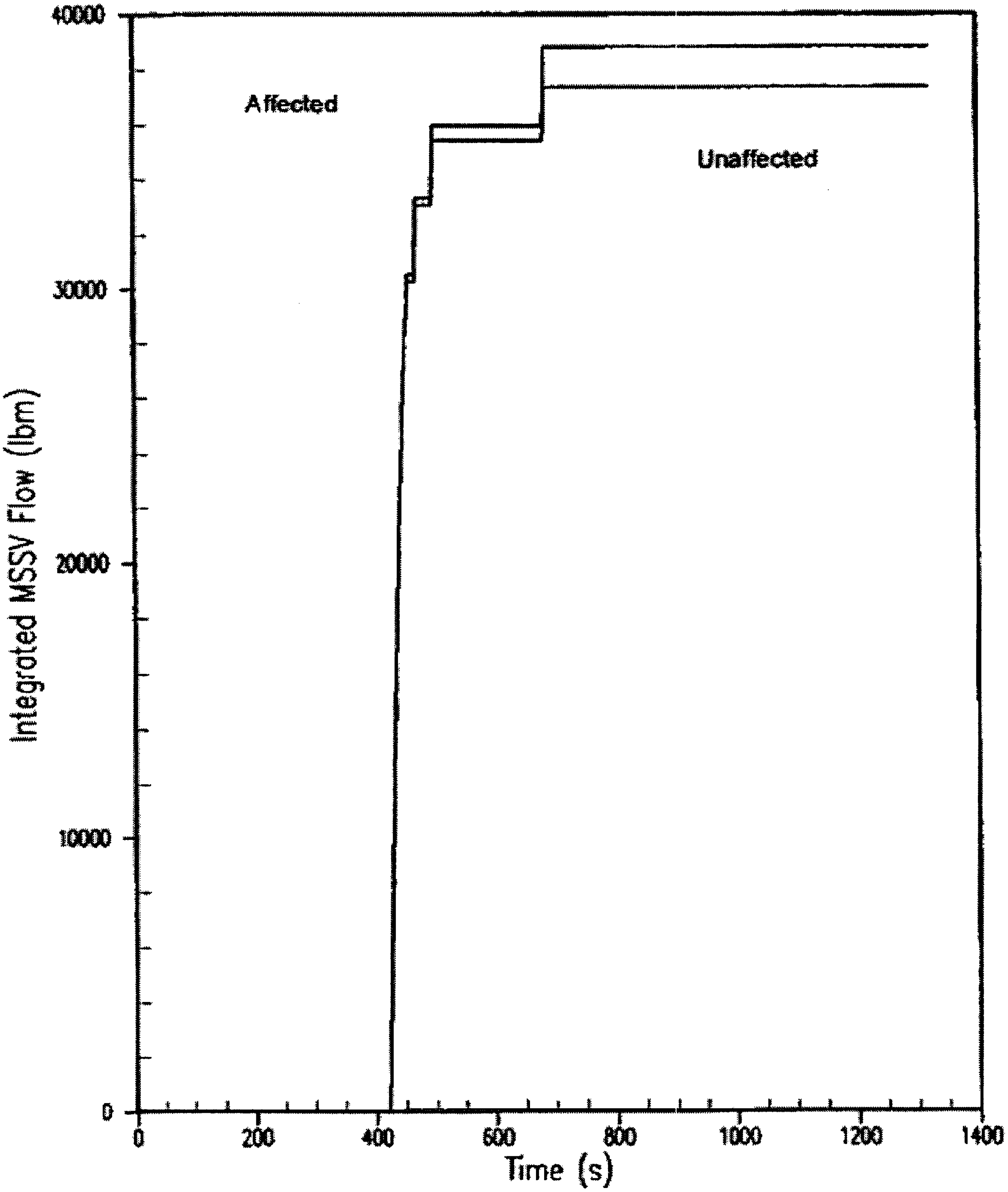
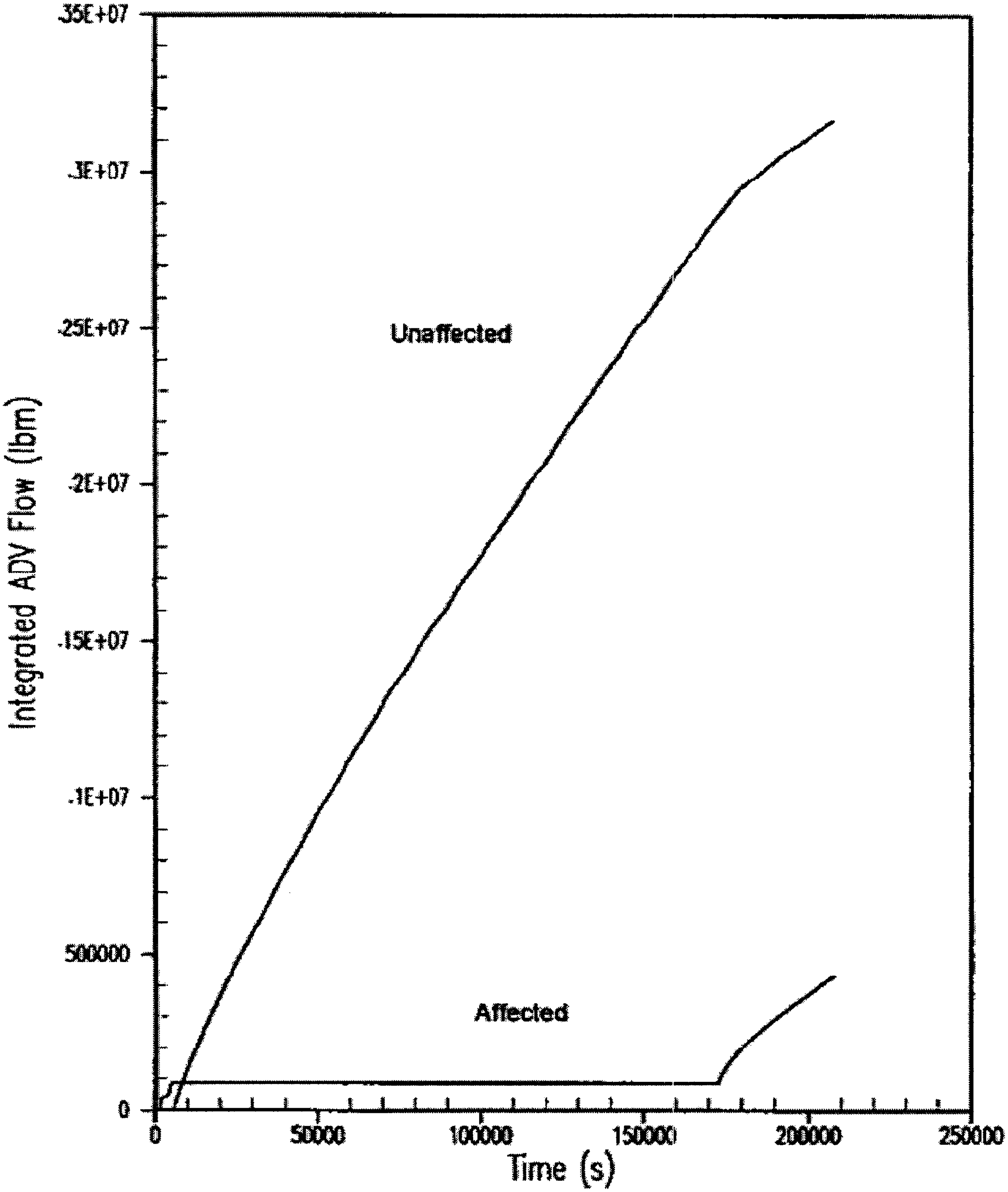
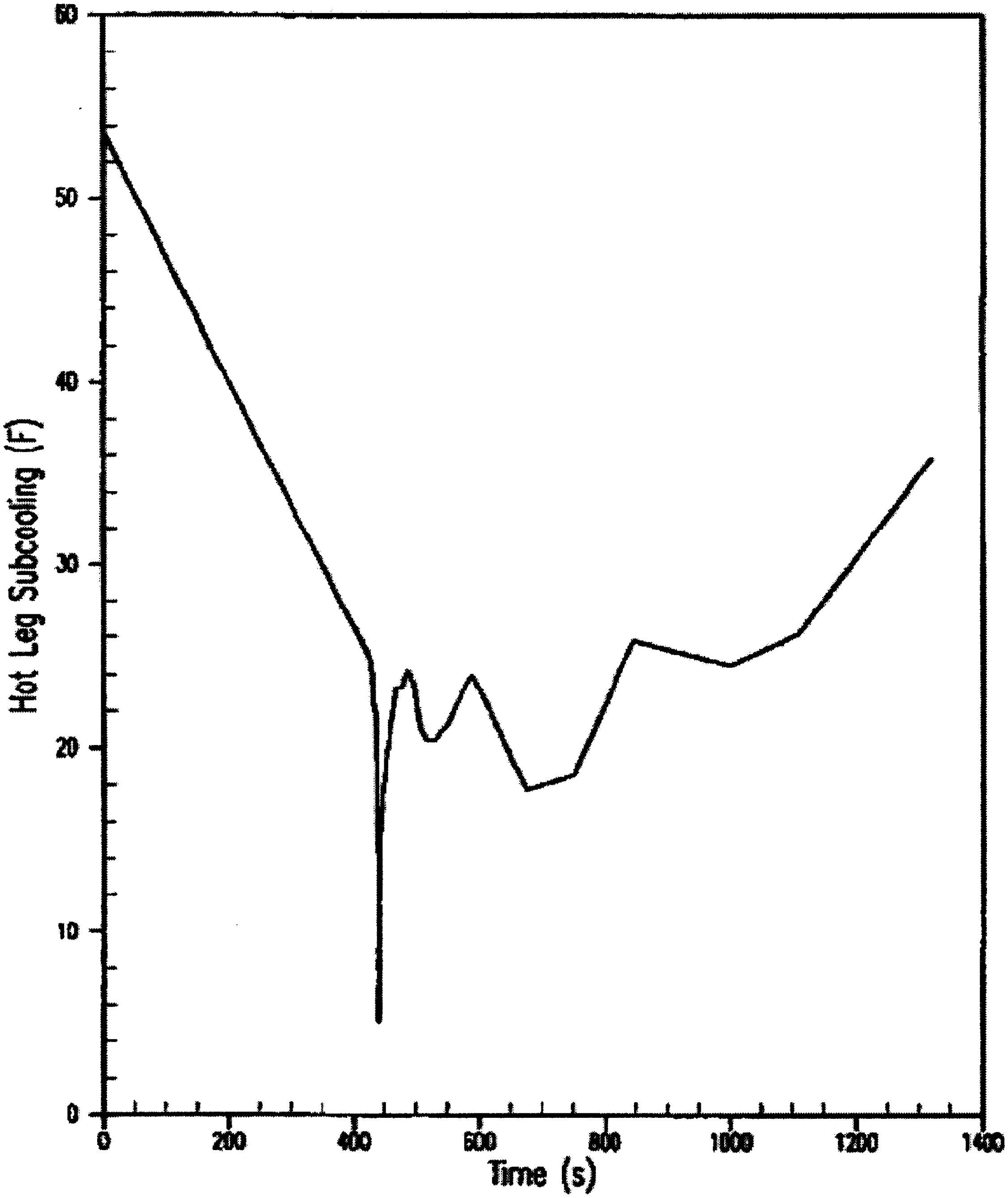


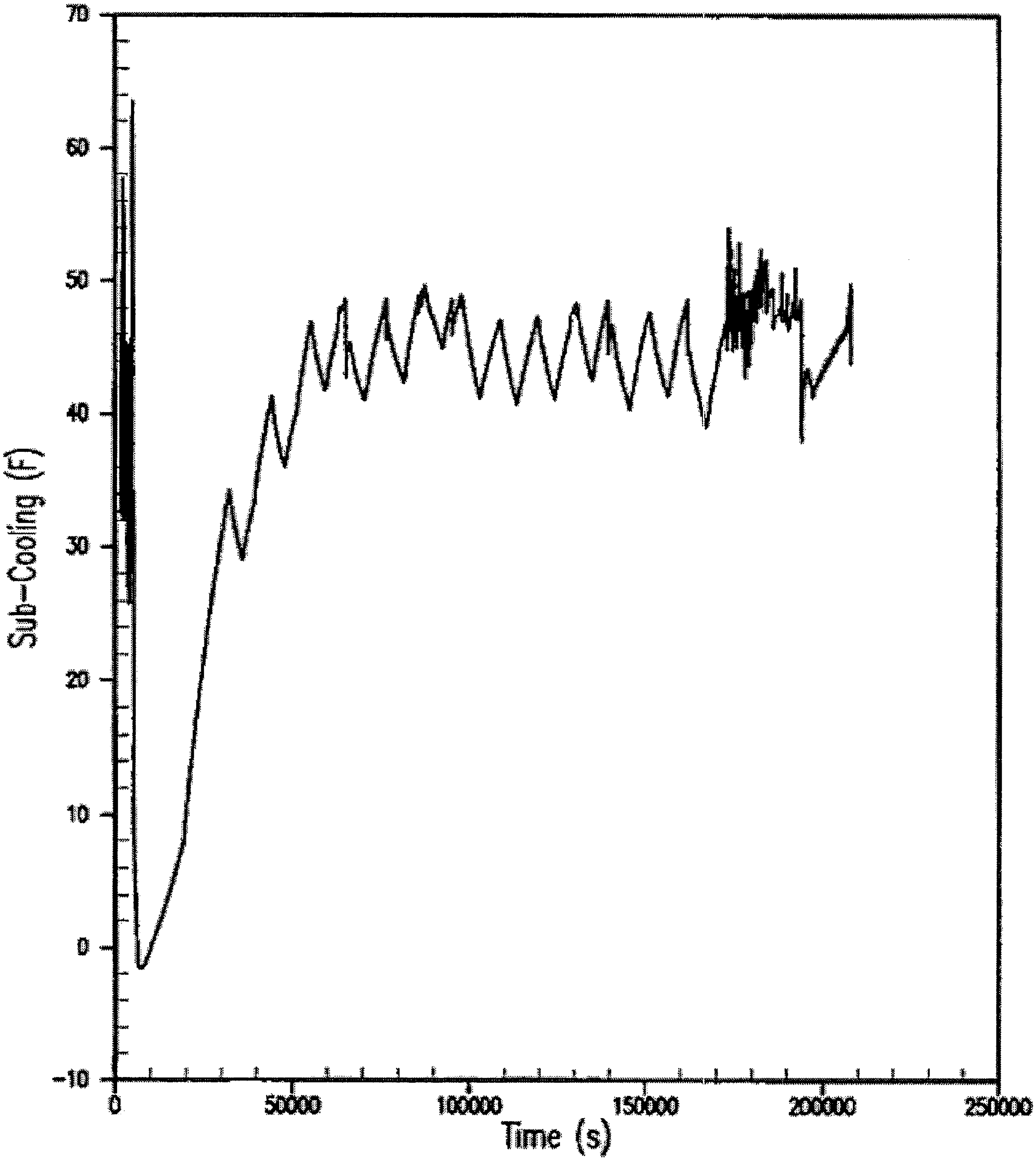
Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED MSSV FLOW VS TIME (Sheet 1)	Figure 14.15-15 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS INTEGRATED MSSV FLOW VS TIME (Sheet 2, Does Not Include CESEC Portion of Event)	Figure 14.15-15 Rev. 43



Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS HOT LEG SUBCOOLING VS TIME (Sheet 1)	Figure 14.15-16 Rev. 43



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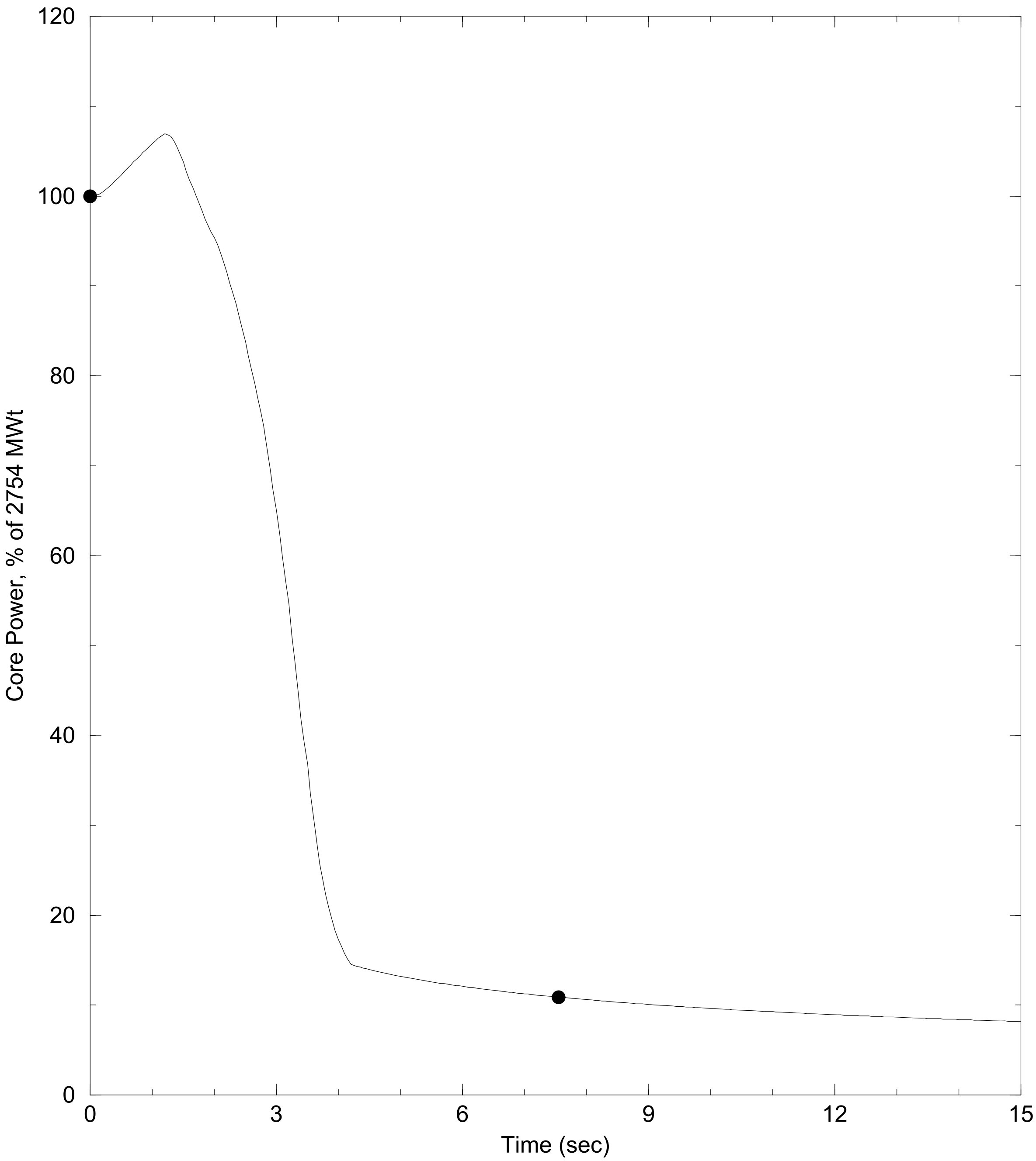
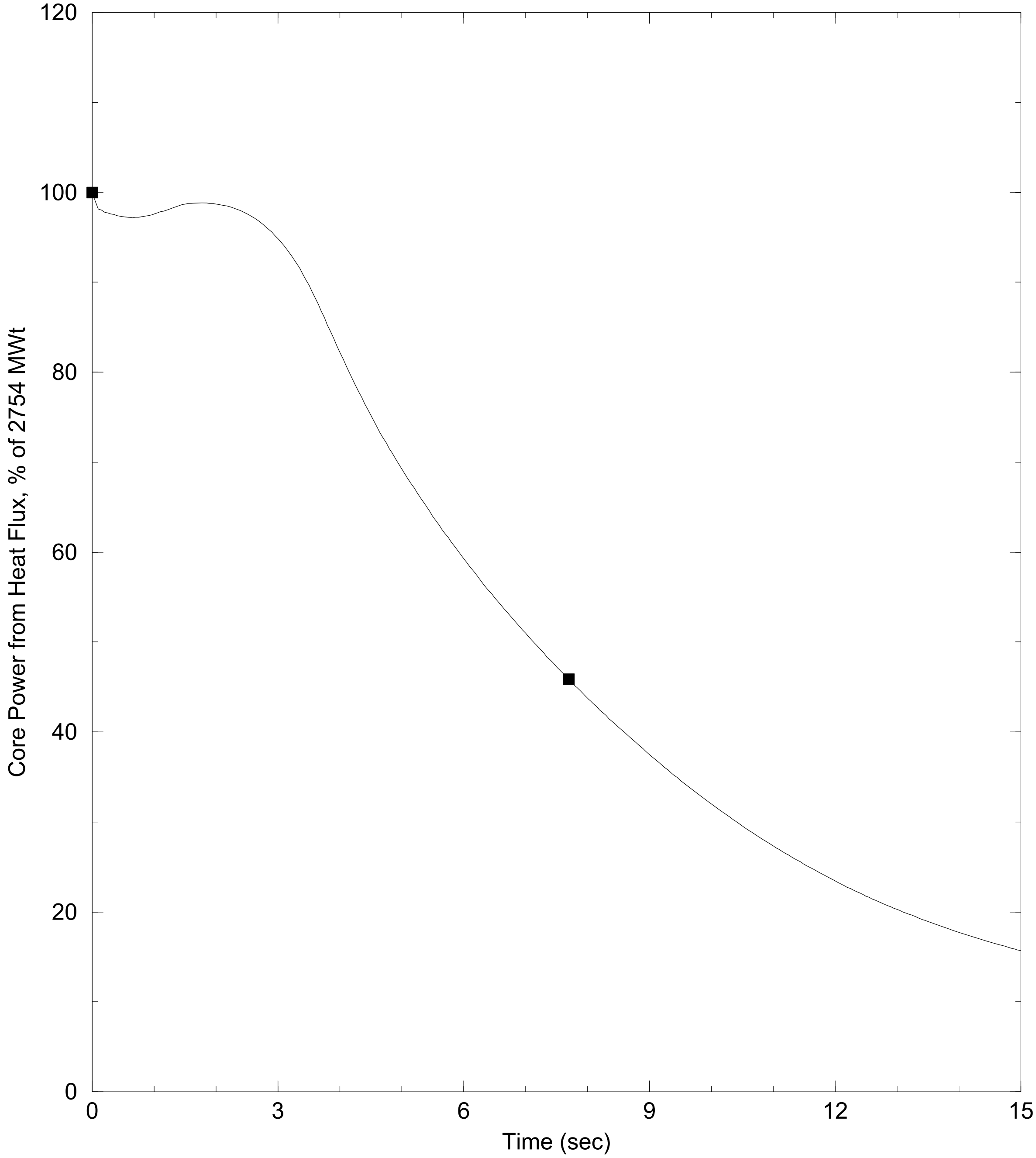
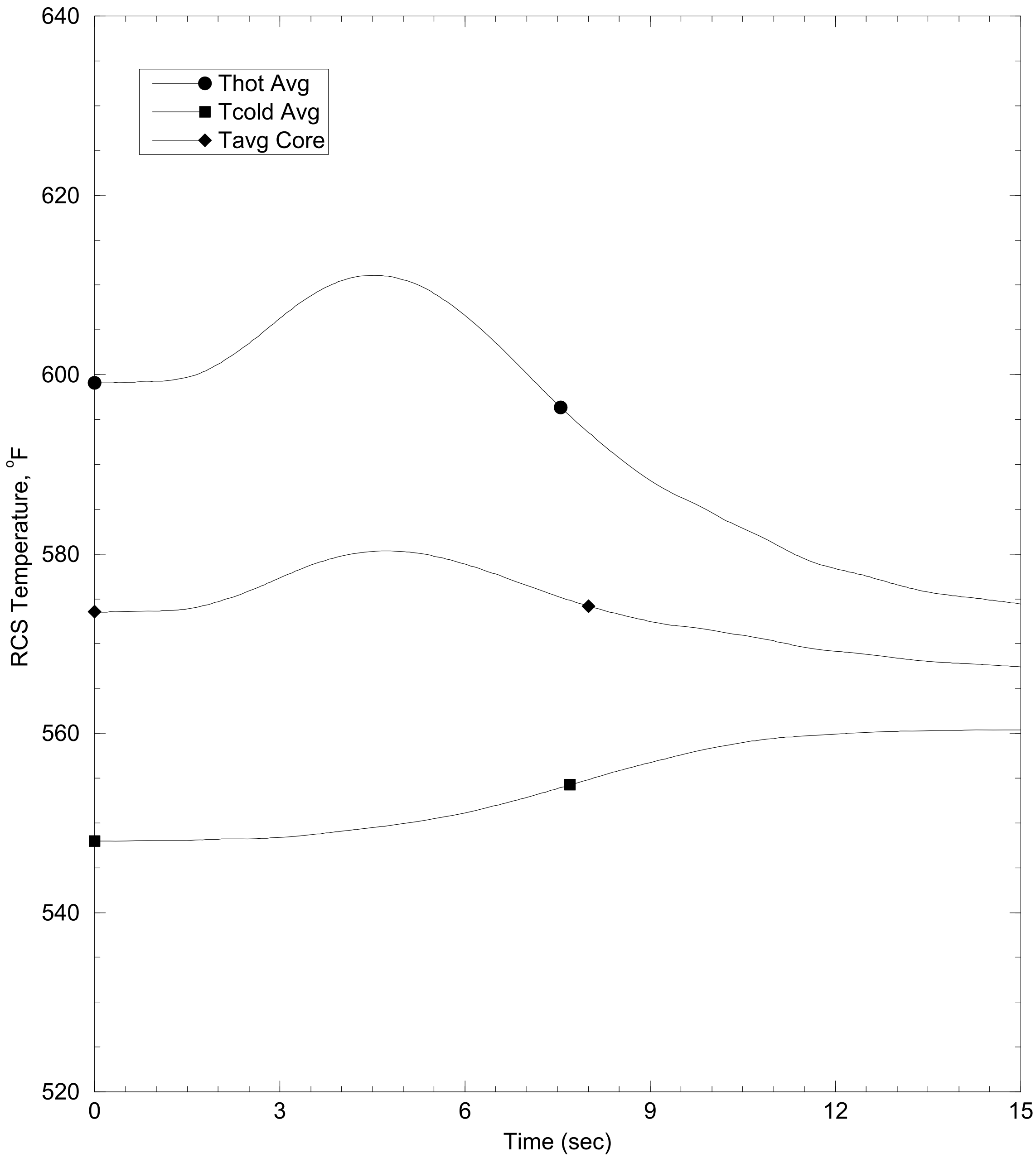


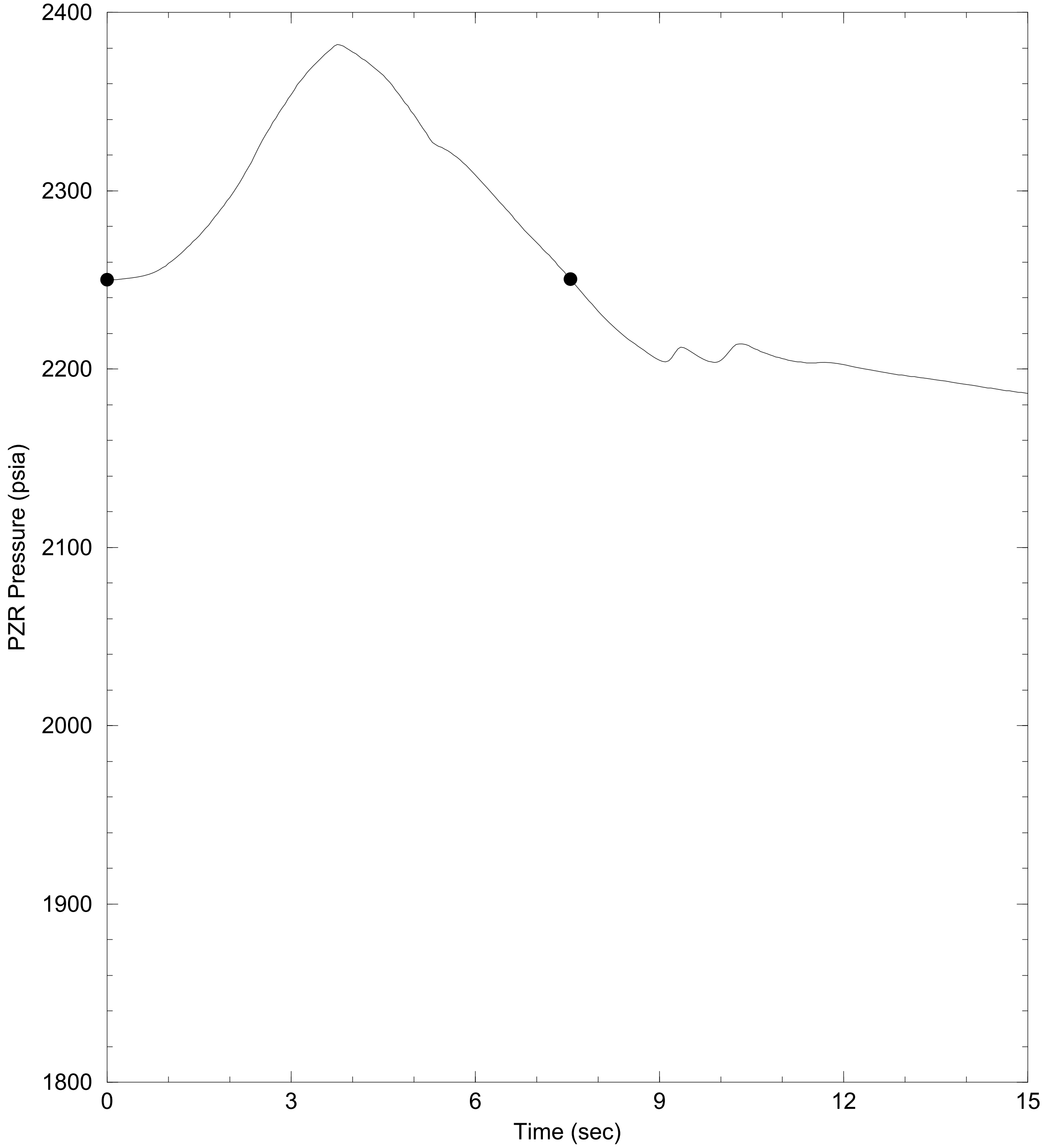
Calvert Cliffs Nuclear Power Plant	SEIZED ROTOR EVENT CORE POWER VS TIME	Figure 14.16-1 Rev. 44



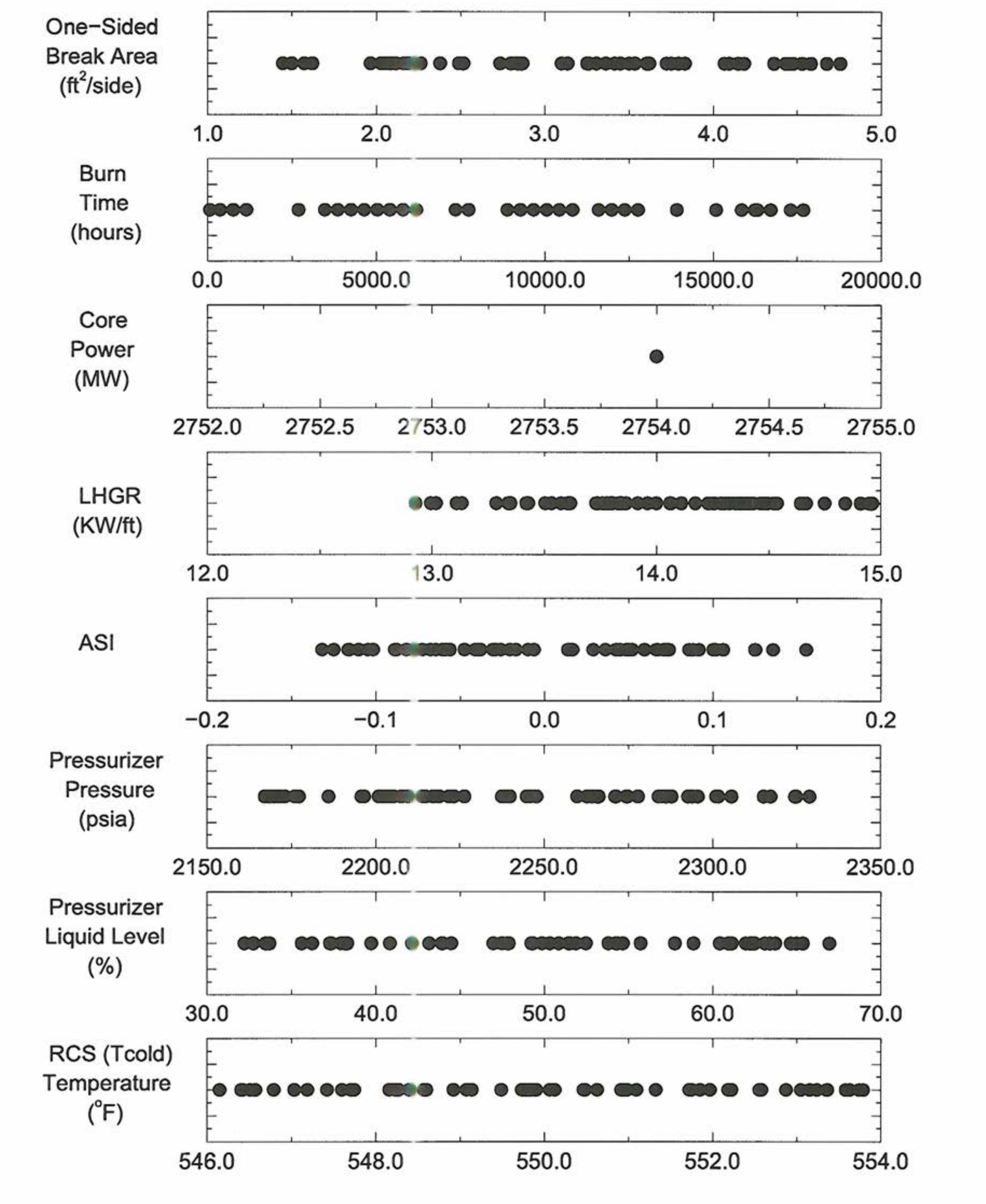
Calvert Cliffs Nuclear Power Plant	SEIZED ROTOR EVENT CORE AVERAGE HEAT FLUX VS TIME	Figure 14.16-2 Rev. 44



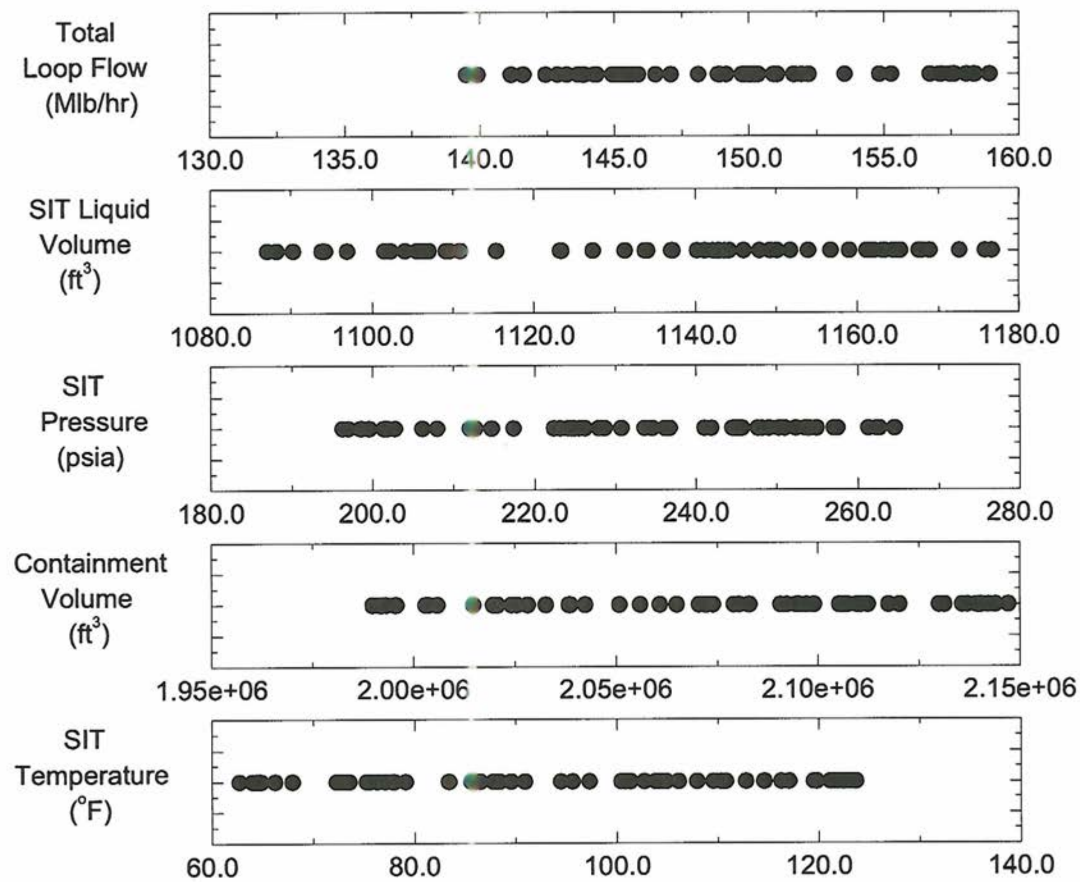
Calvert Cliffs Nuclear Power Plant	SEIZED ROTOR EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	Figure 14.16-3 Rev. 44



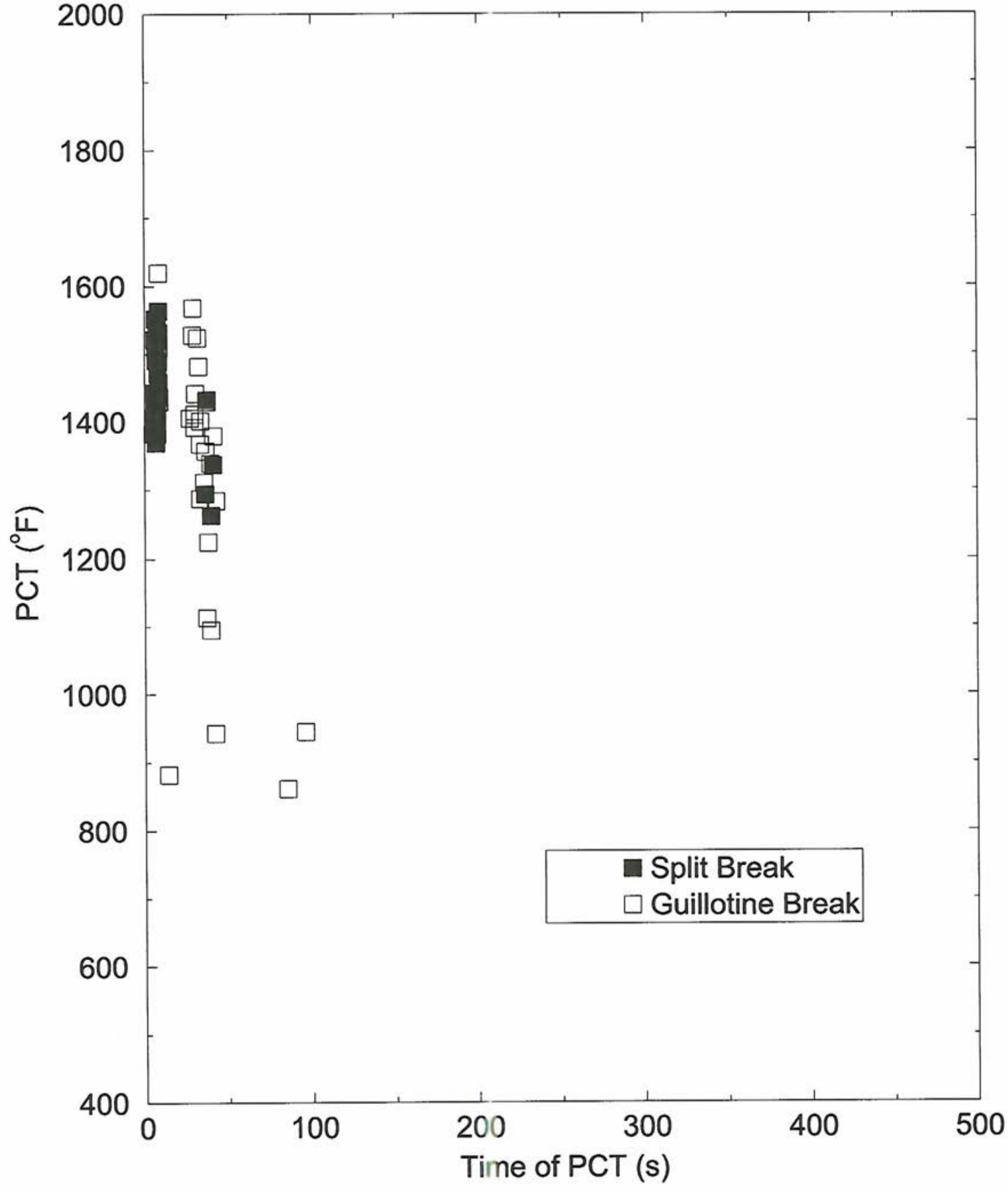
Calvert Cliffs Nuclear Power Plant	SEIZED ROTOR EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	Figure 14.16-4 Rev. 44



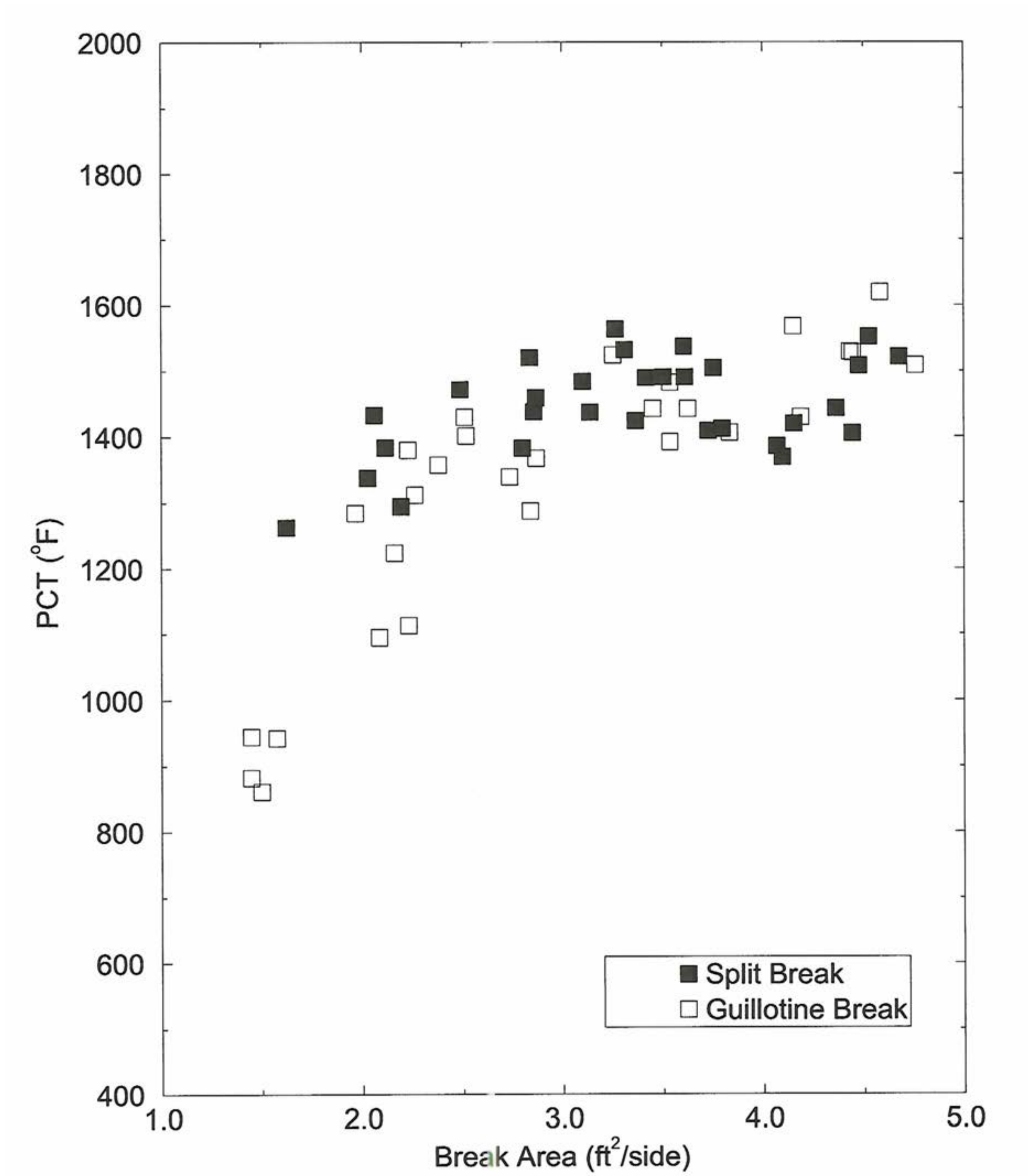
Calvert Cliffs Nuclear Power Plant	SCATTER PLOT OF OPERATIONAL PARAMETERS (Sheet 1)	Figure 14.17-1 Rev. 46



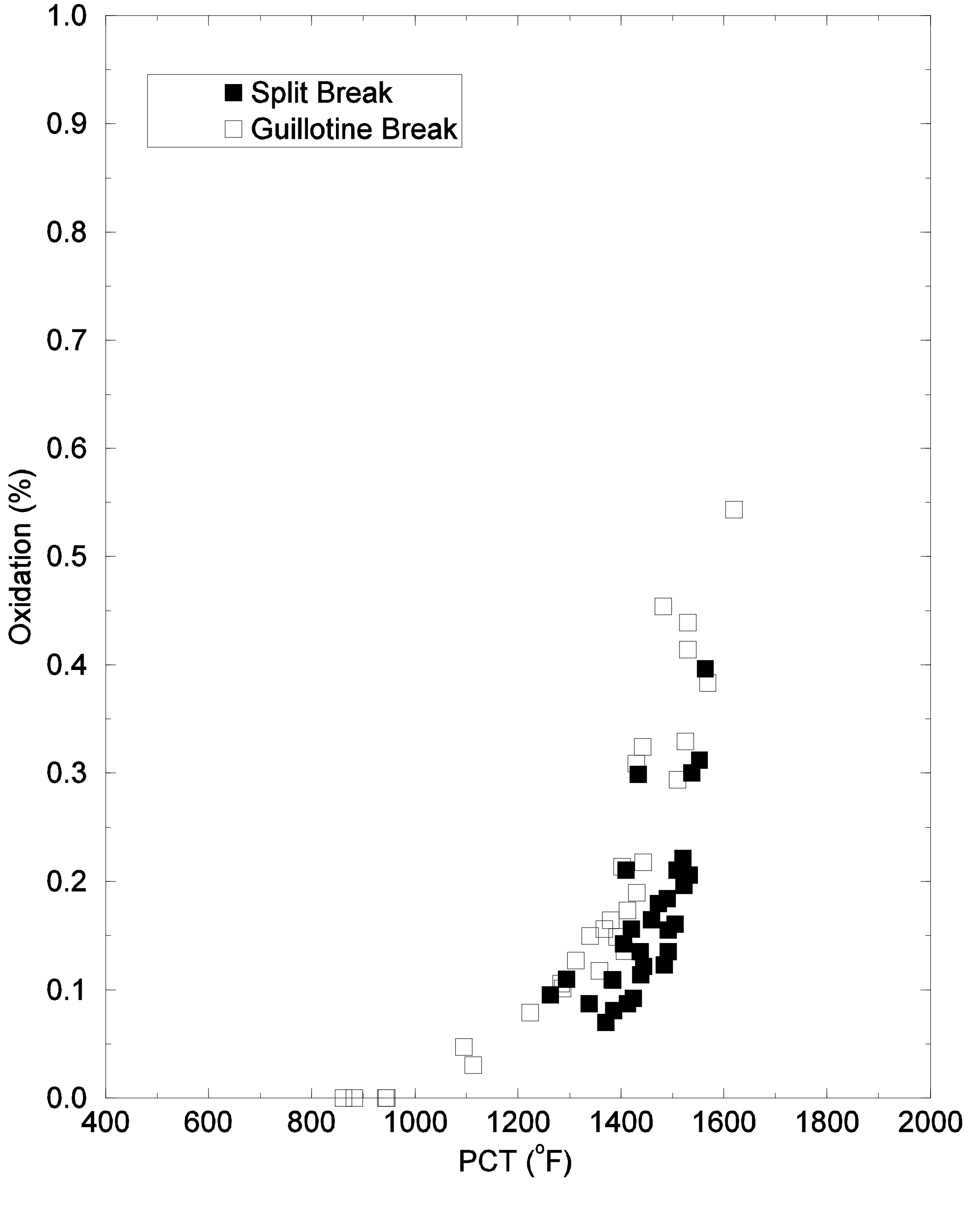
Calvert Cliffs Nuclear Power Plant	SCATTER PLOT OF OPERATIONAL PARAMETERS (Sheet 2)	Figure 14.17-1 Rev. 46



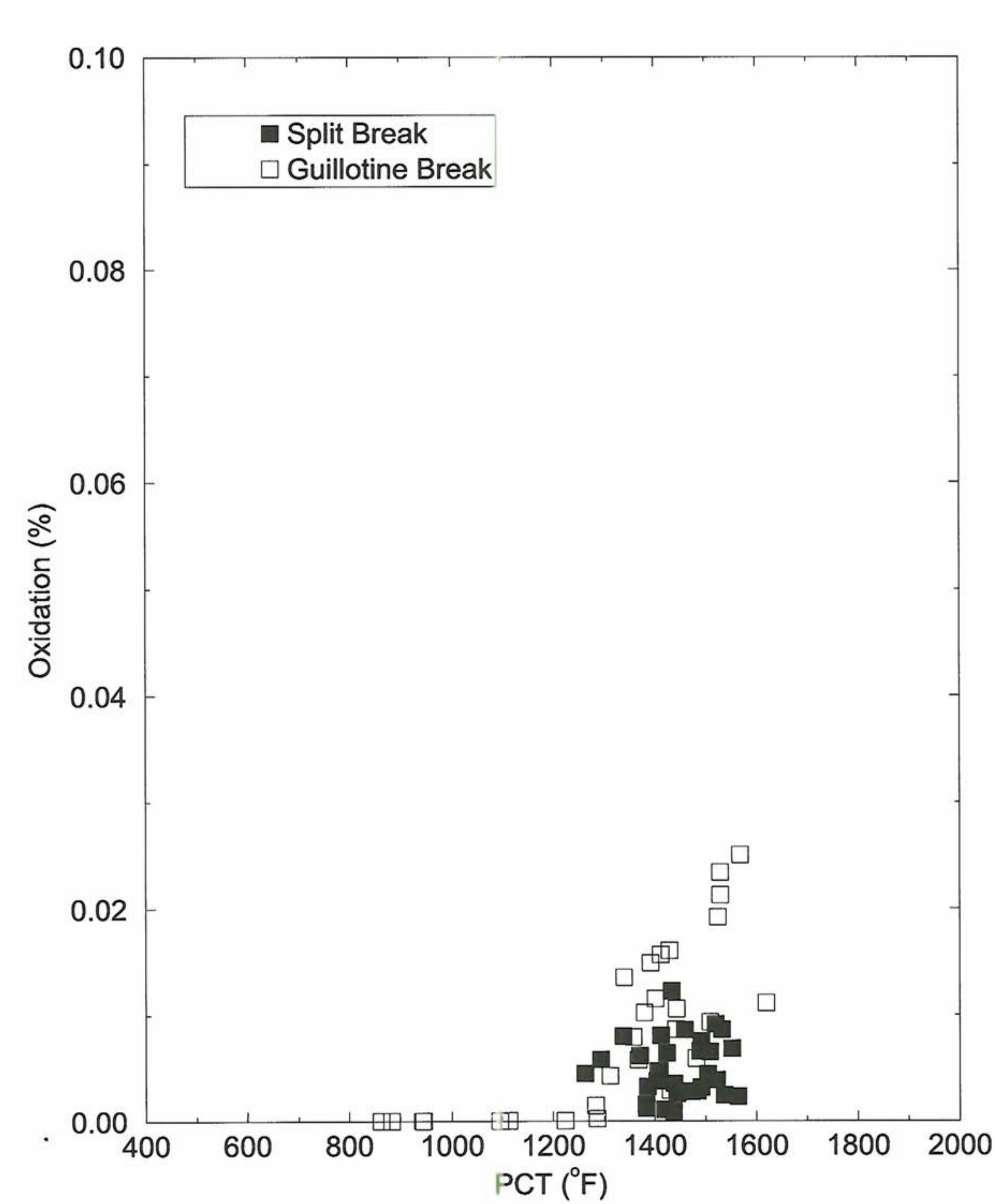
Calvert Cliffs Nuclear Power Plant	PCT VERSUS PCT TIME – SCATTER PLOT FROM 59 CALCULATIONS	Figure 14.17-2 Rev. 46



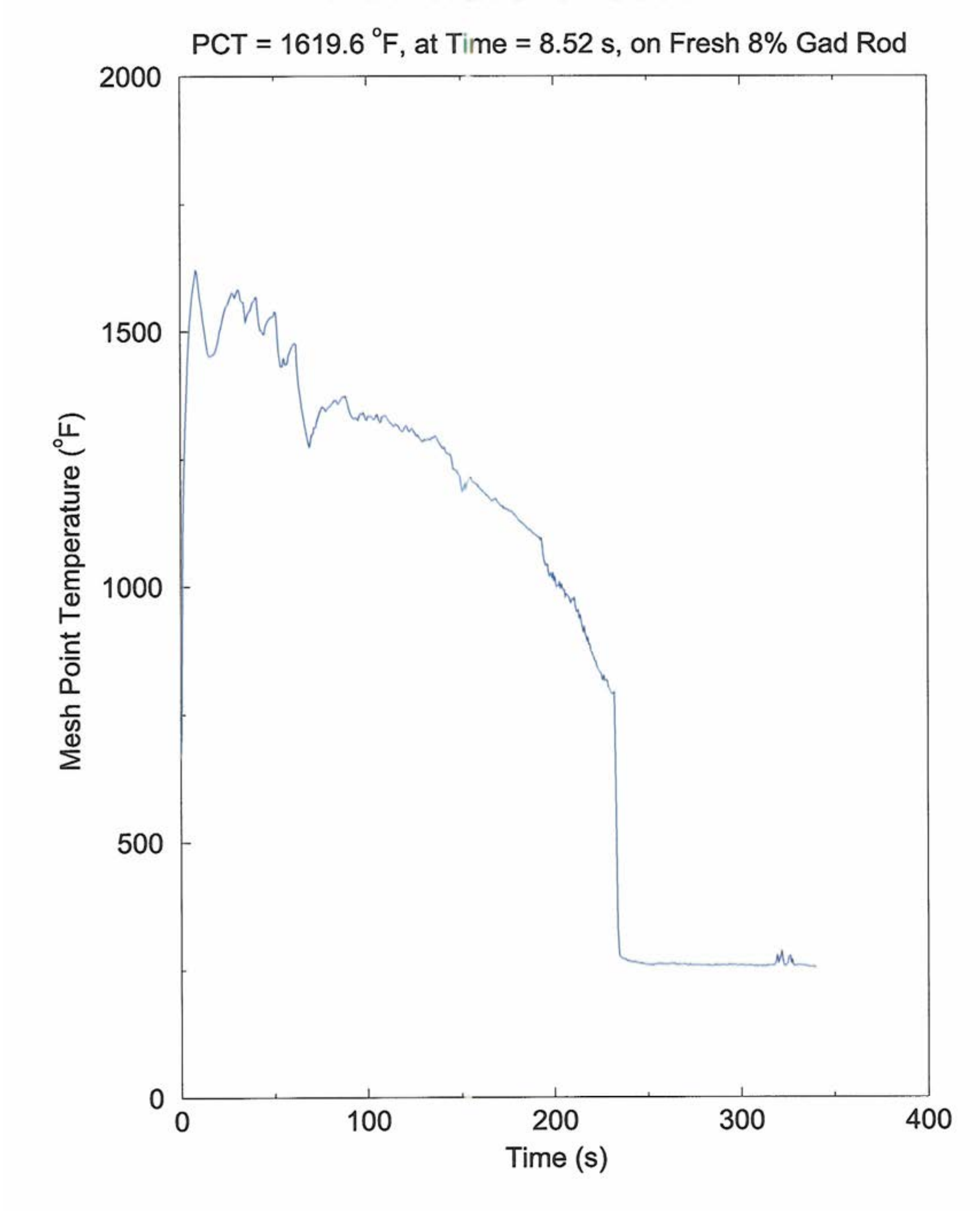
Calvert Cliffs Nuclear Power Plant	PCT VERSUS BREAK SIZE– SCATTER PLOT FROM 59 CALCULATIONS	Figure 14.17-3 Rev. 46



Calvert Cliffs Nuclear Power Plant	MAXIMUM OXIDATION VERSUS PCT - SCATTER PLOT FROM 59 CALCULATIONS	Figure 14.17-4 Rev. 46



Calvert Cliffs Nuclear Power Plant	TOTAL OXIDATION VERSUS PCT - SCATTER PLOT FROM 59 CALCULATIONS	Figure 14.17-5 Rev. 46



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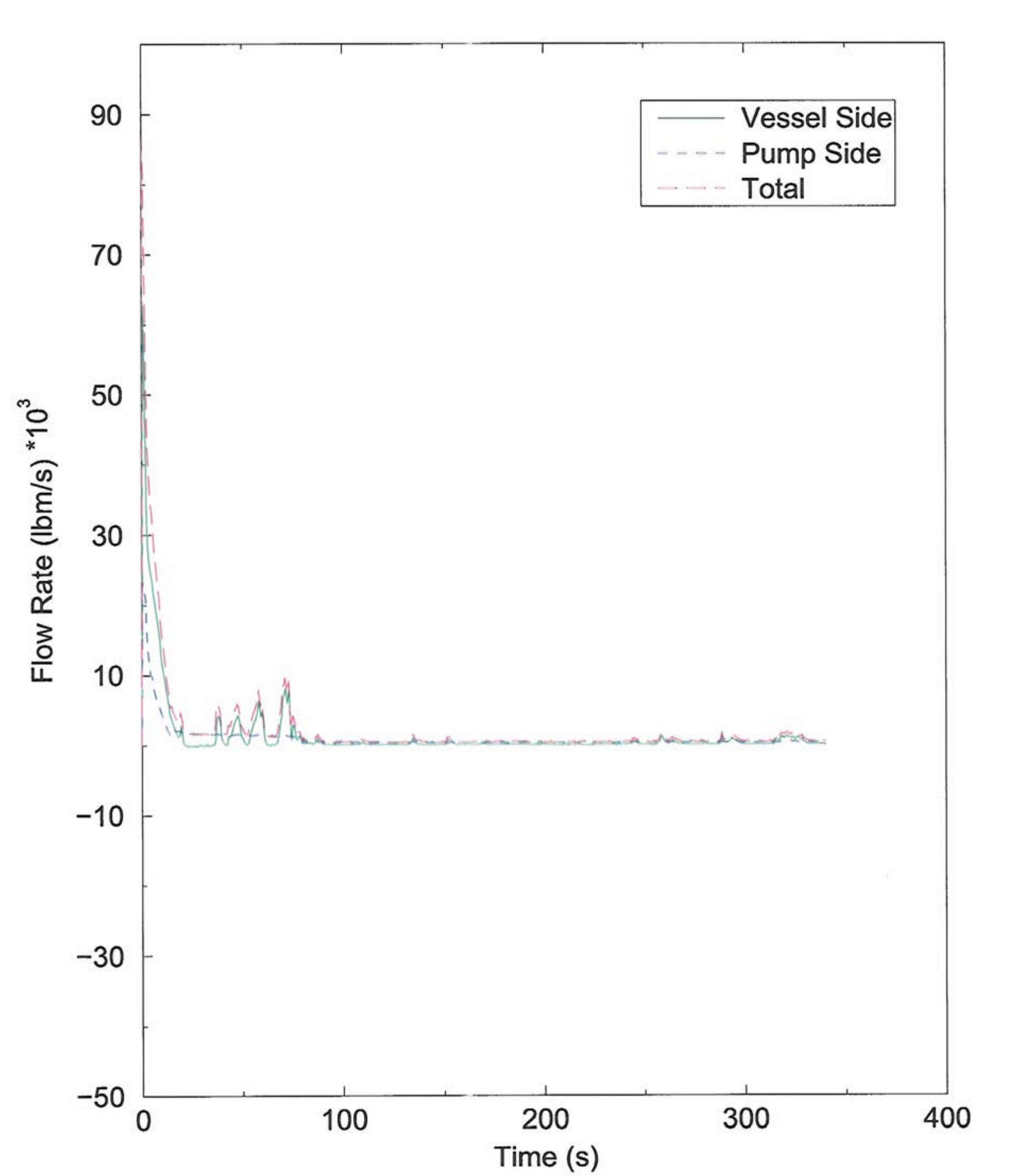

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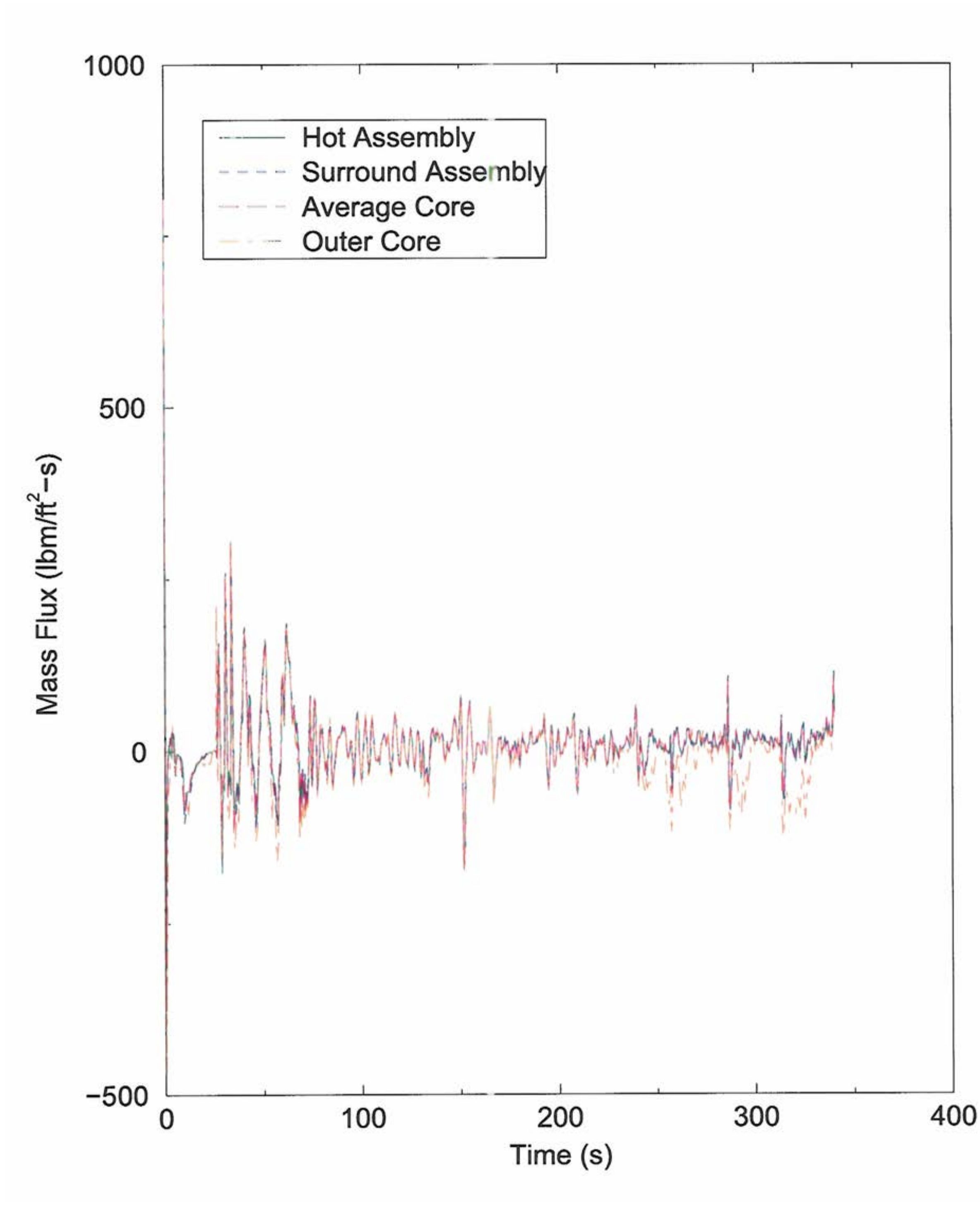
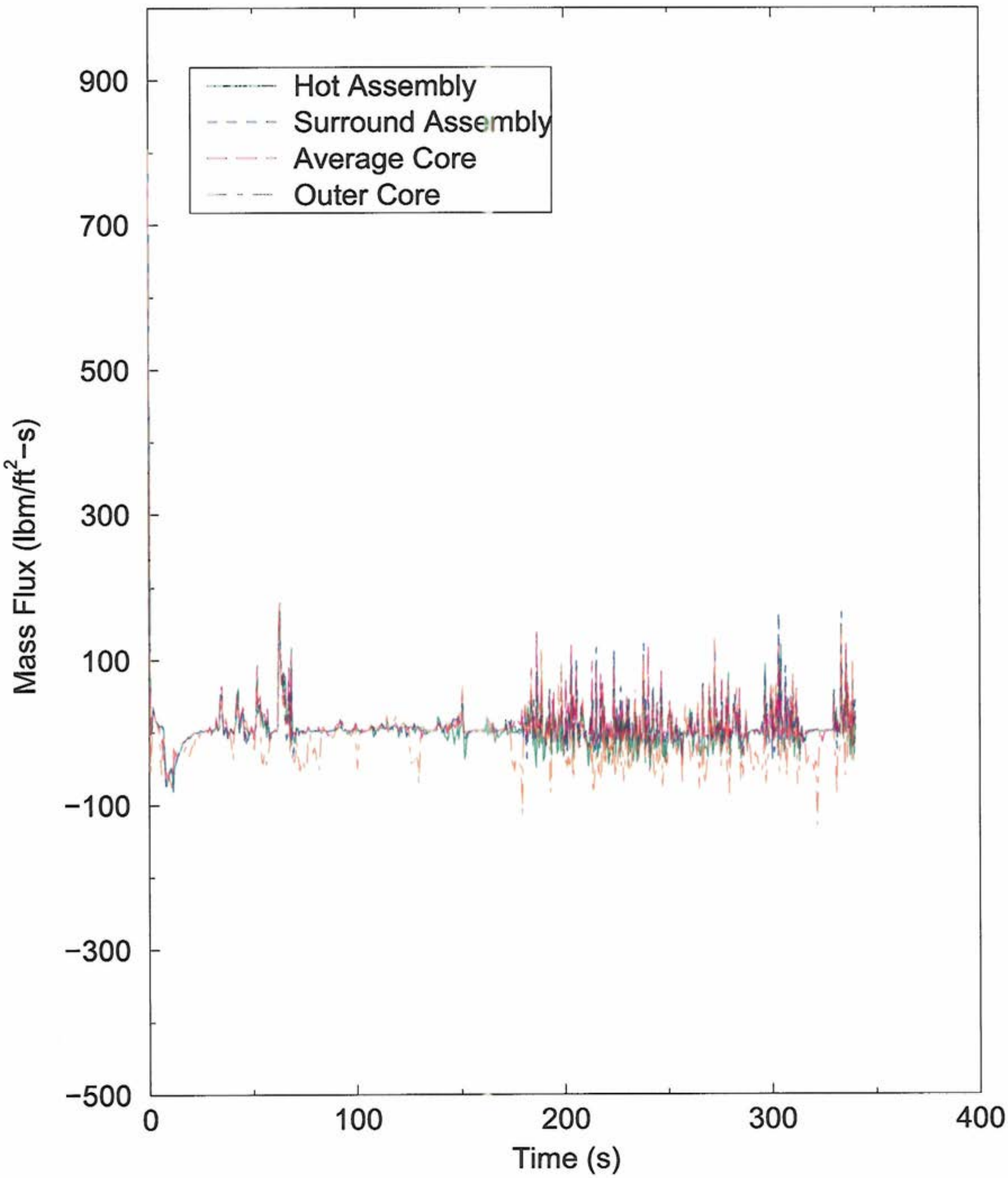
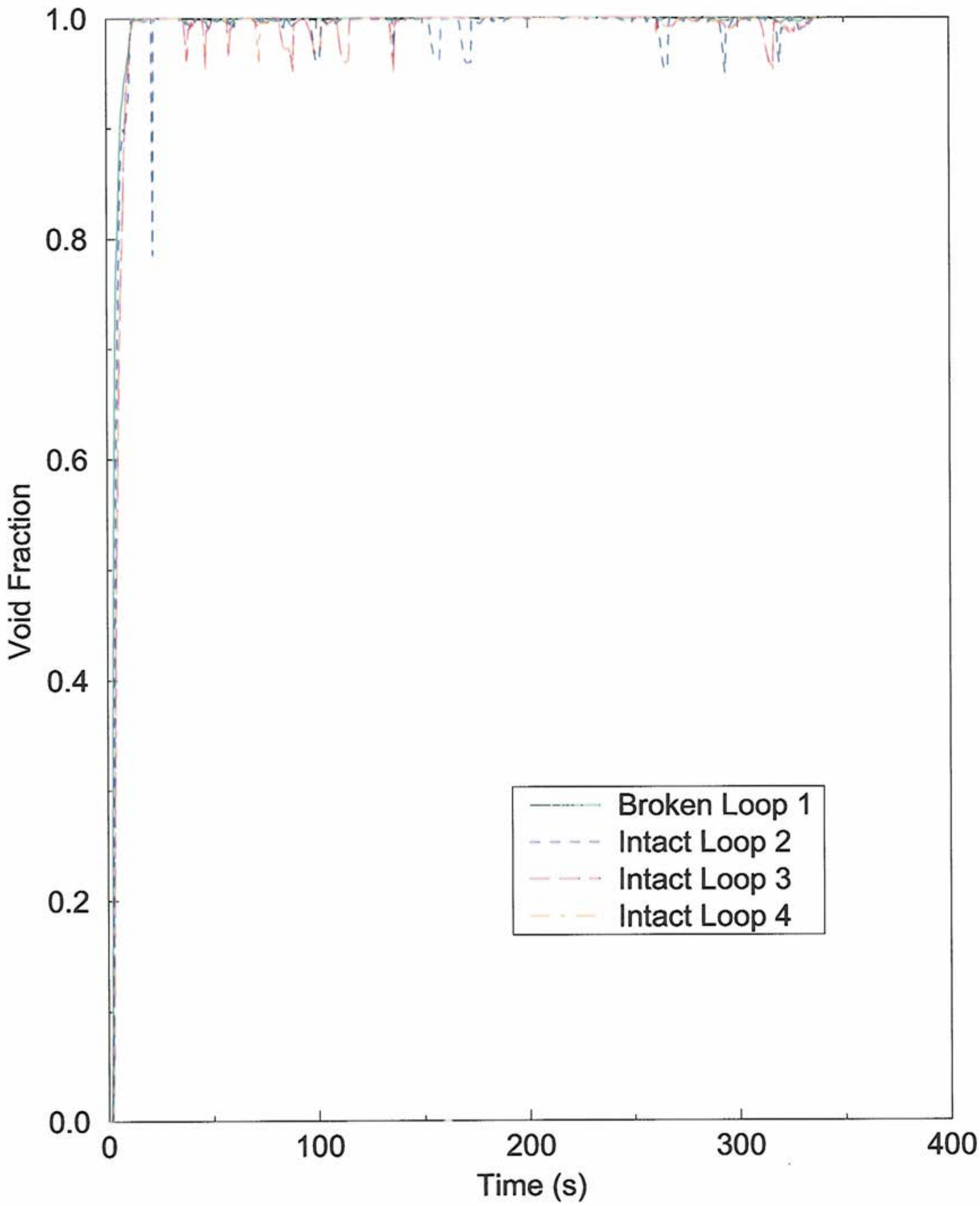


Calvert Cliffs Nuclear Power Plant	CORE INLET MASS FLUX FOR THE LIMITING CASE	Figure 14.17-8 Rev. 46



Calvert Cliffs Nuclear Power Plant	CORE OUTLET MASS FLUX FOR THE LIMITING CASE	Figure 14.17-9 Rev. 46



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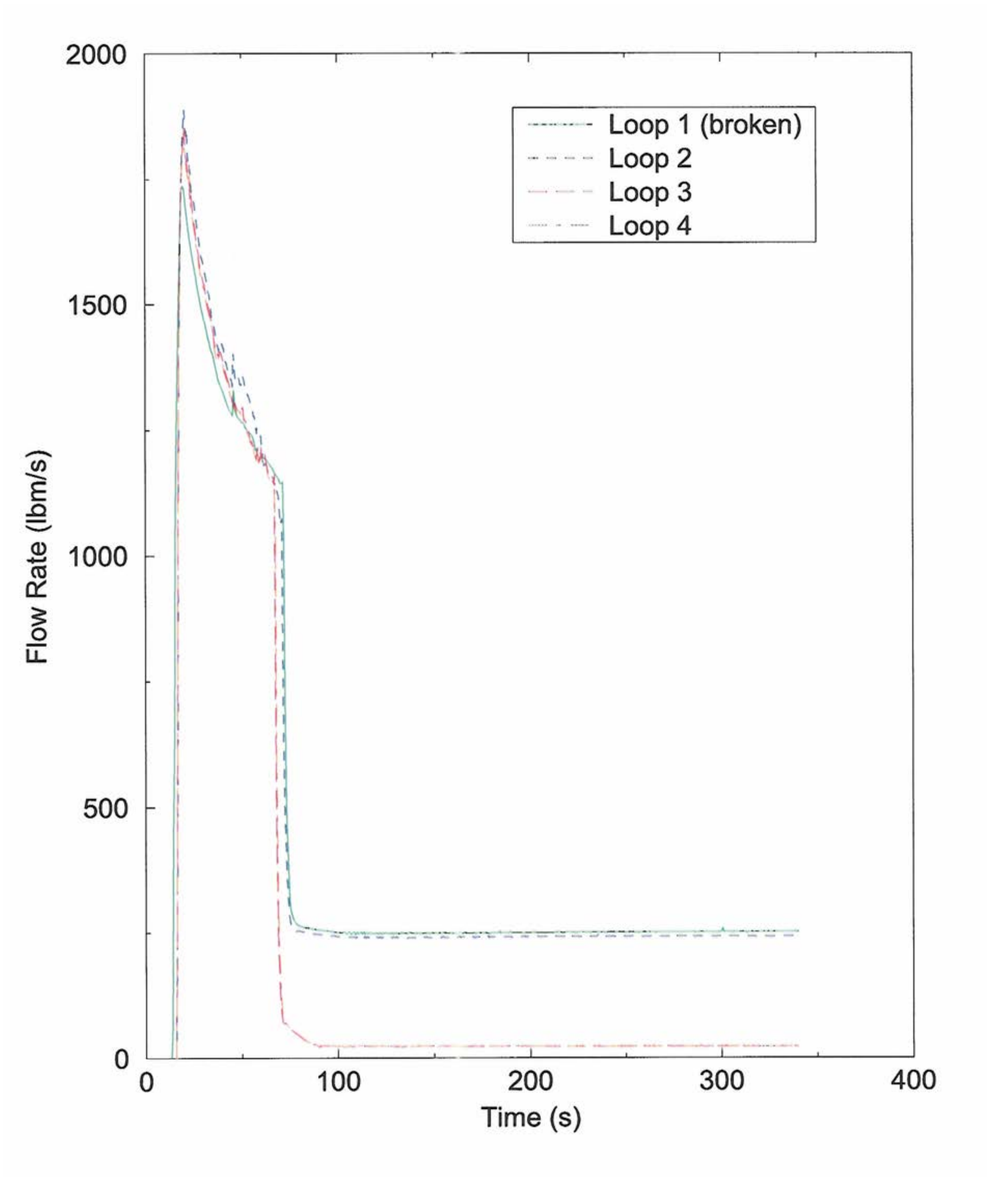

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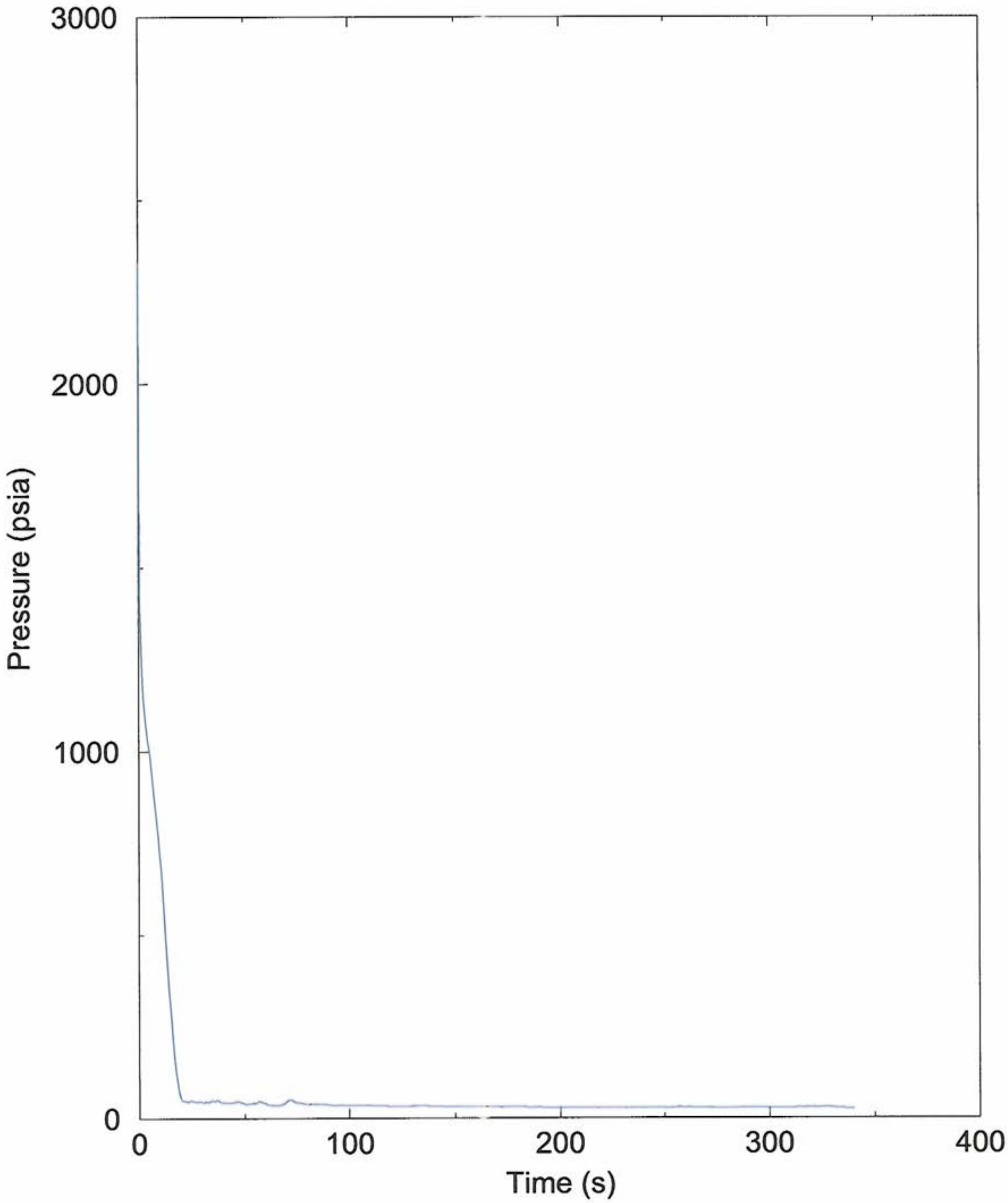

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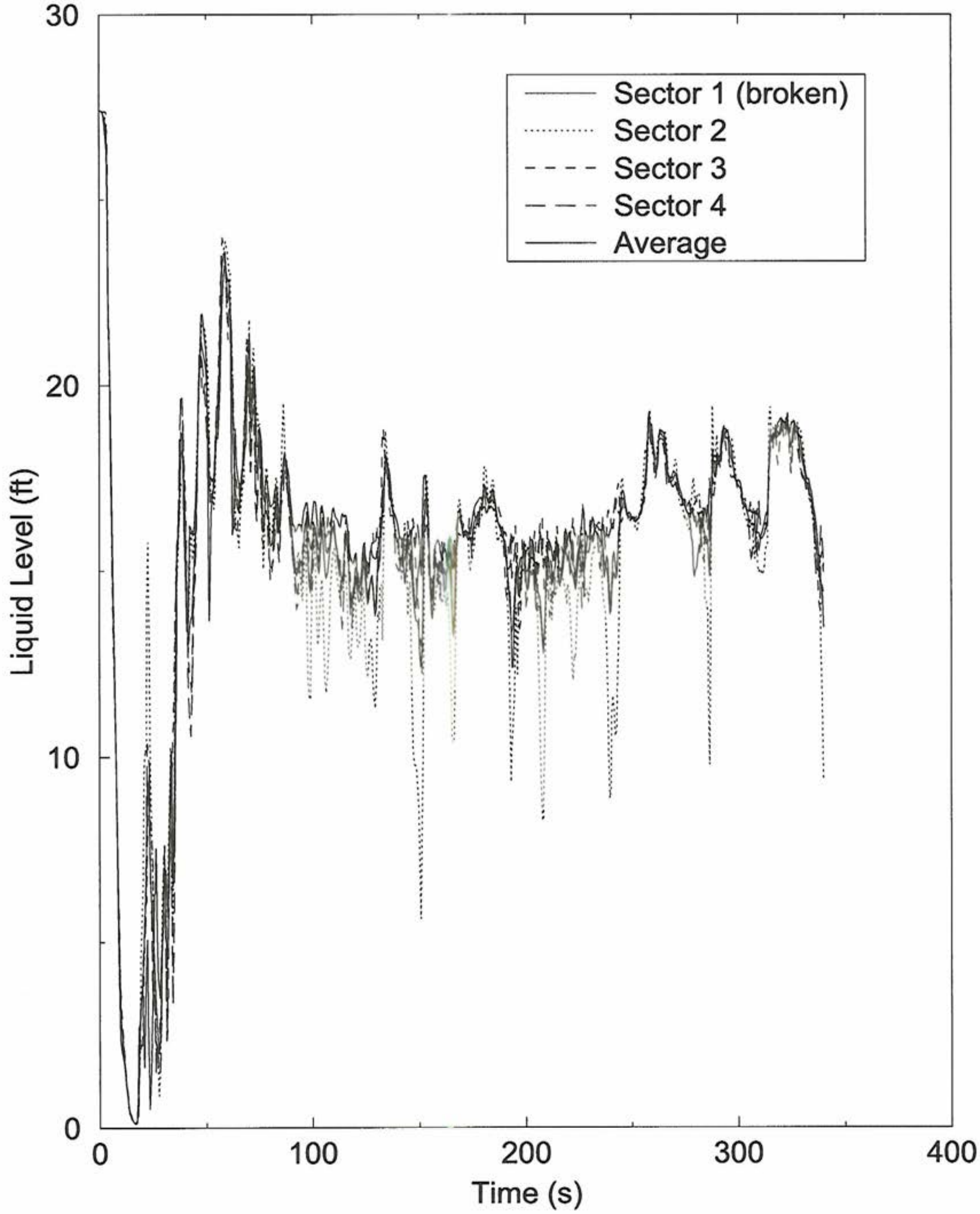

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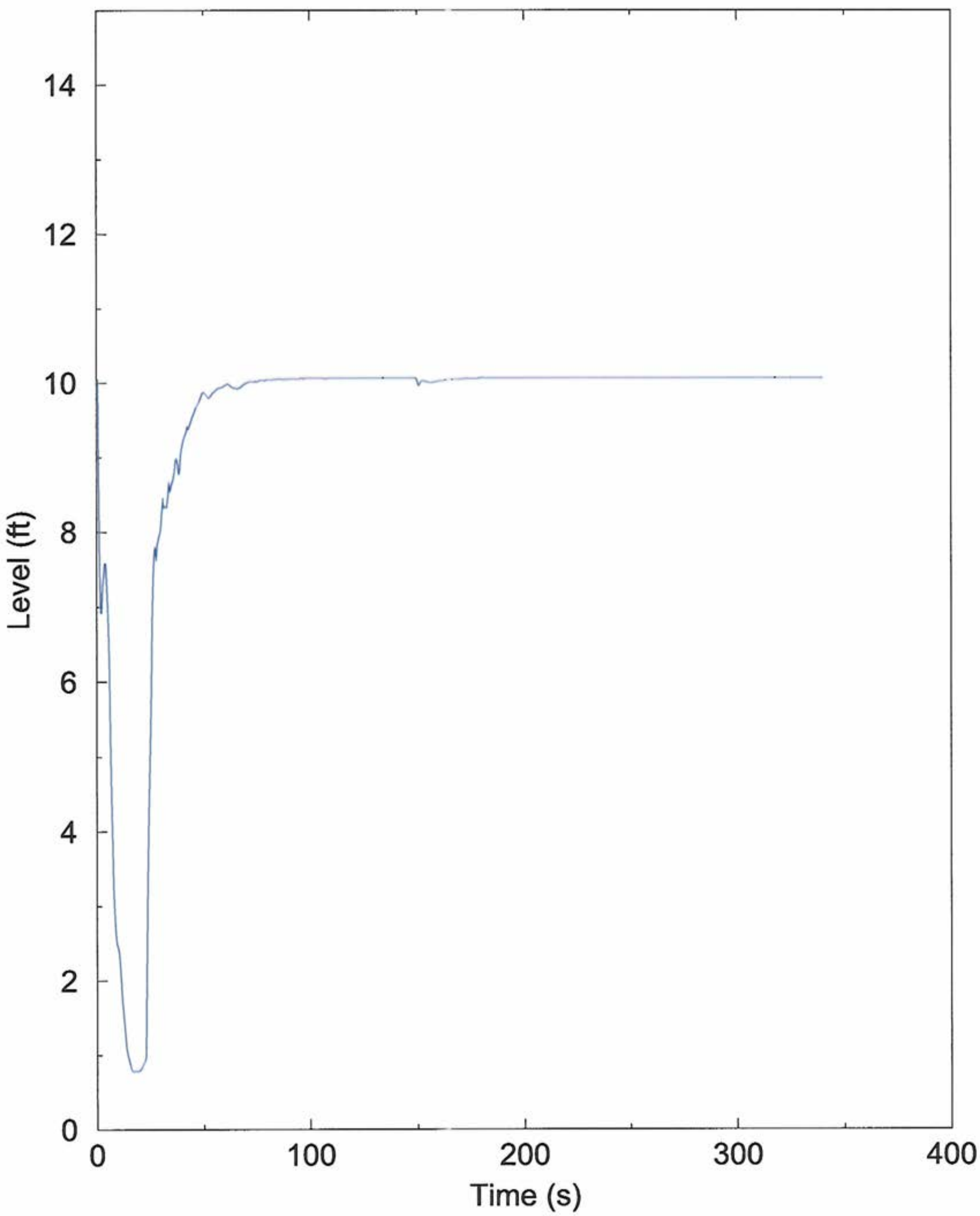

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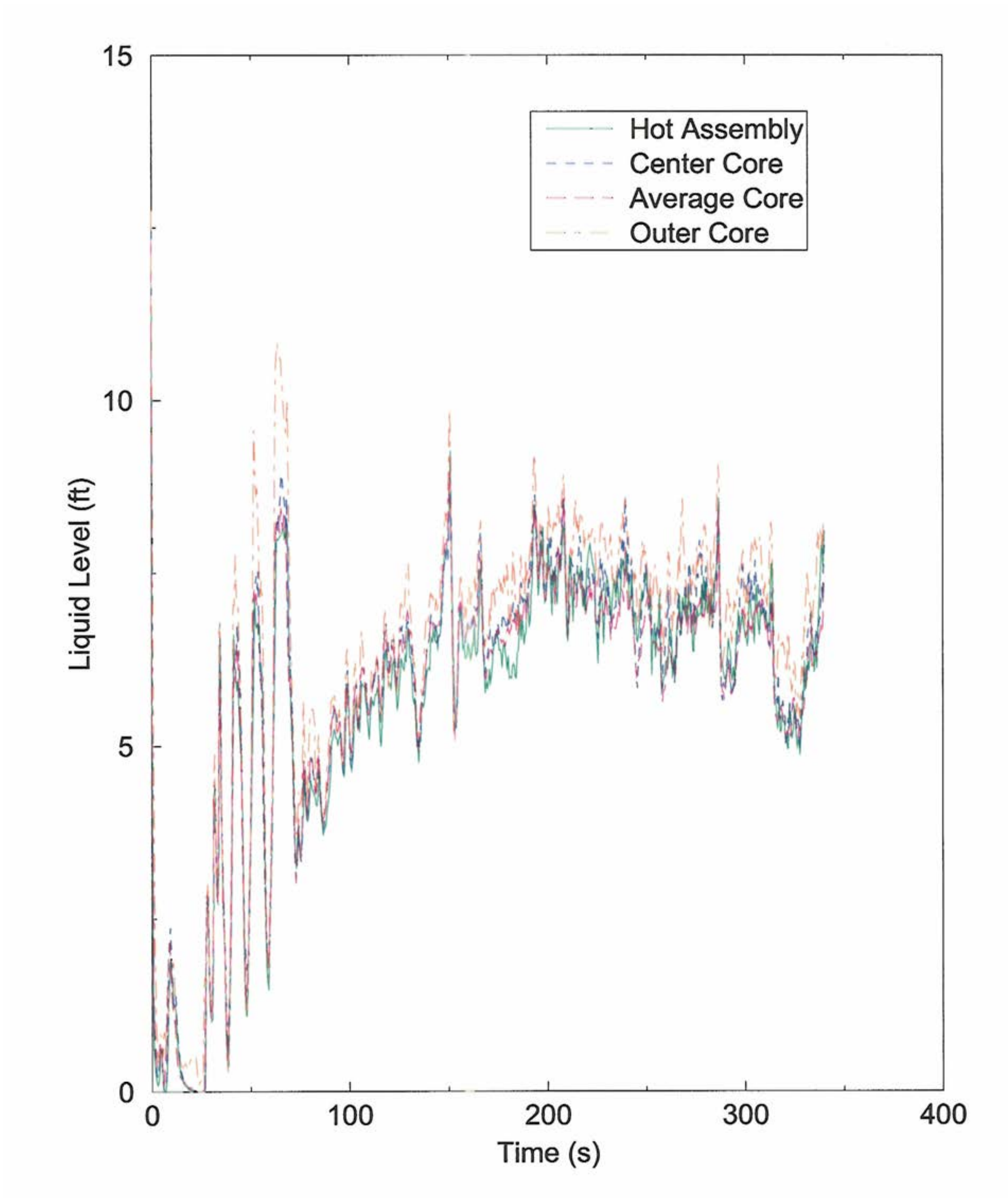
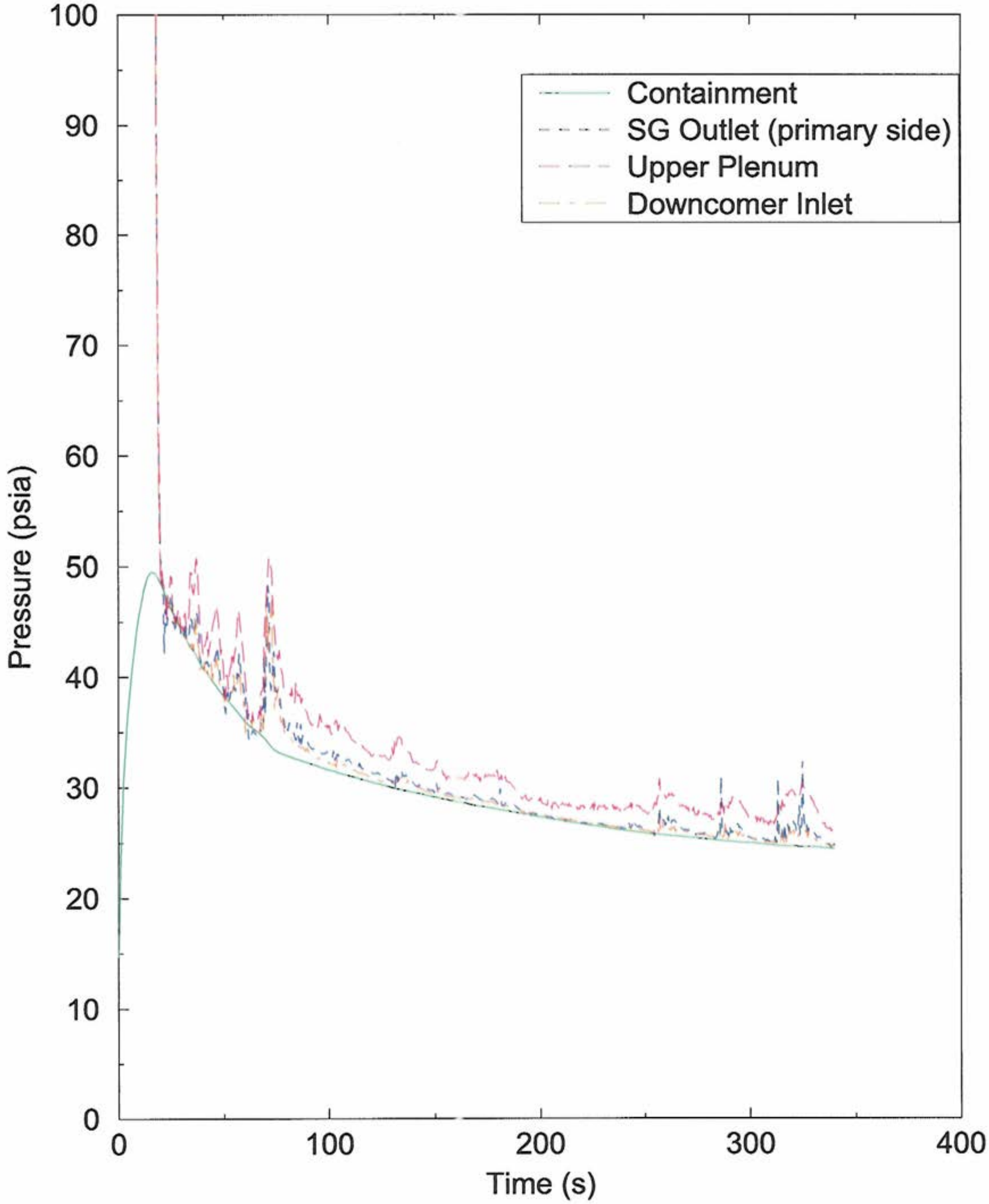
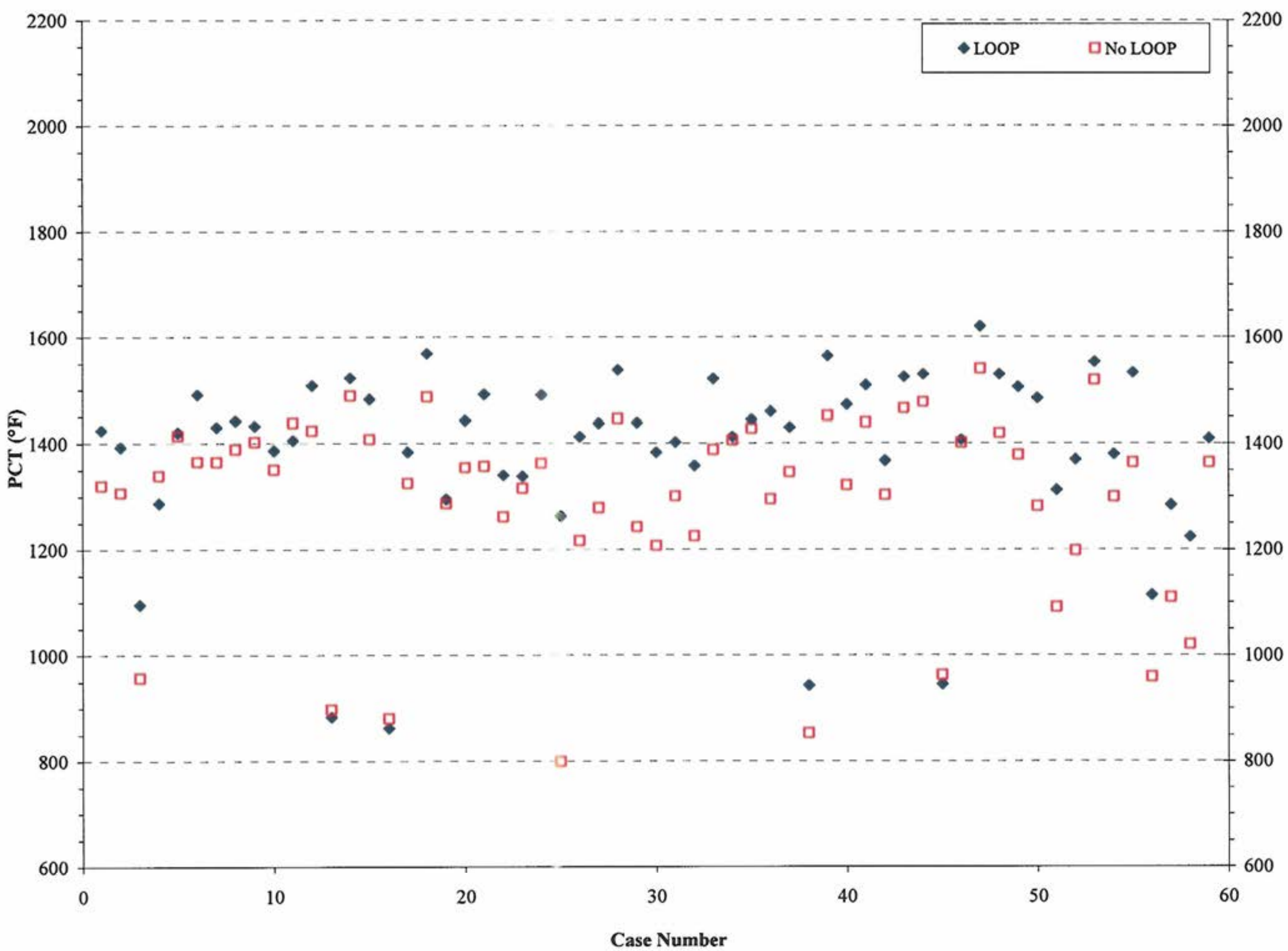


Calvert Cliffs Nuclear Power Plant	COLLAPSED LIQUID LEVEL IN THE CORE FOR THE LIMITING CASE	Figure 14.17-15 Rev. 46



Calvert Cliffs Nuclear Power Plant	CONTAINMENT AND LOOP PRESSURES FOR THE LIMITING CASE	Figure 14.17-16 Rev. 46



LOOP
No LOOP
PCT (°F)
Case Number
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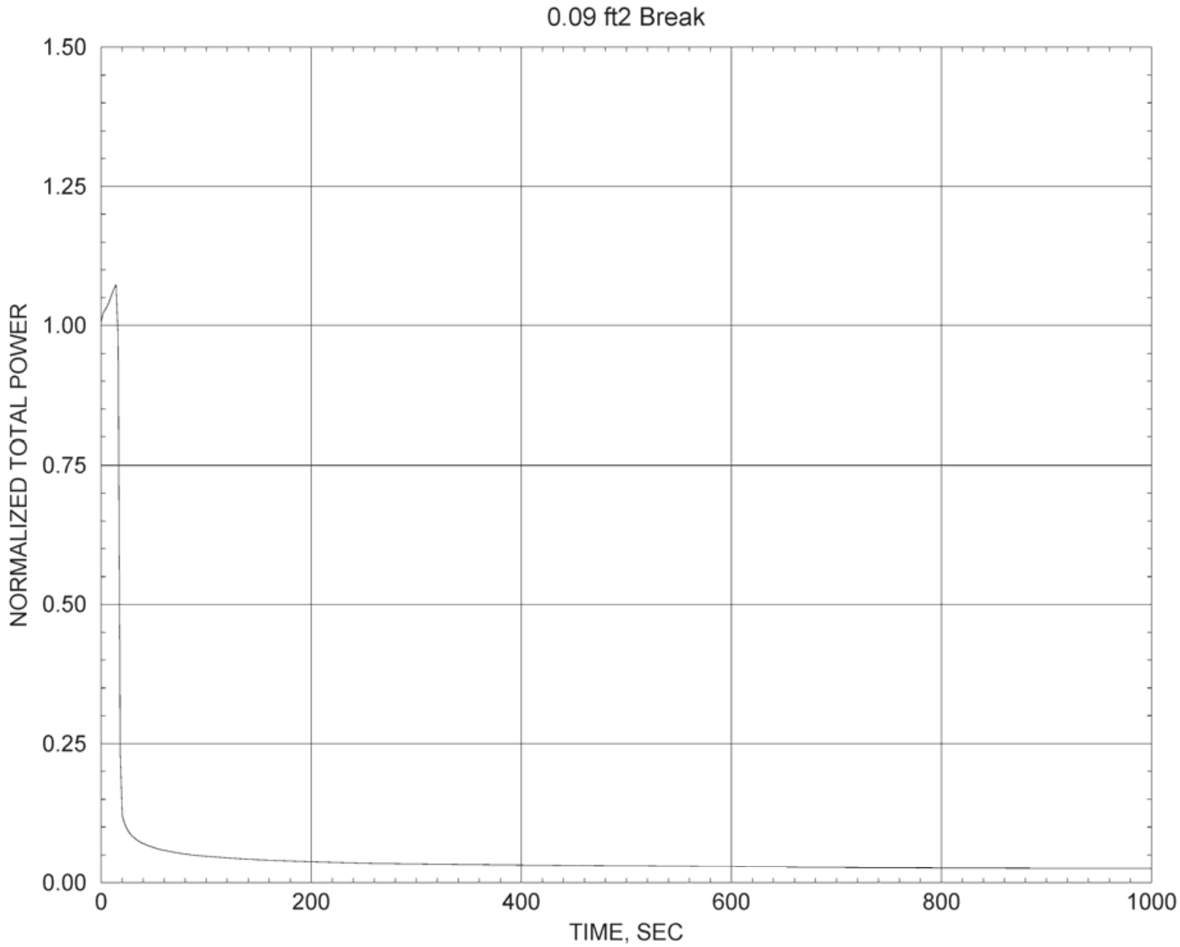
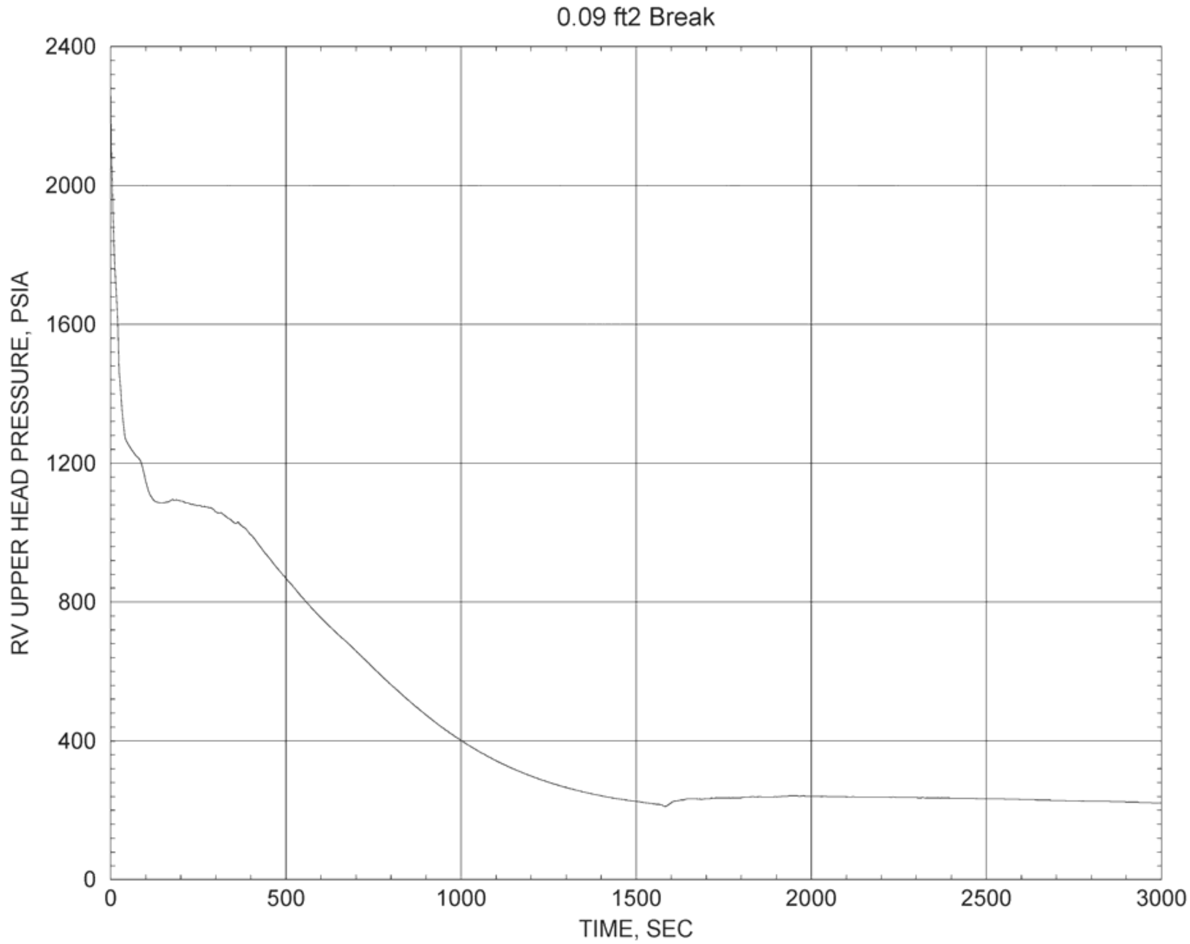
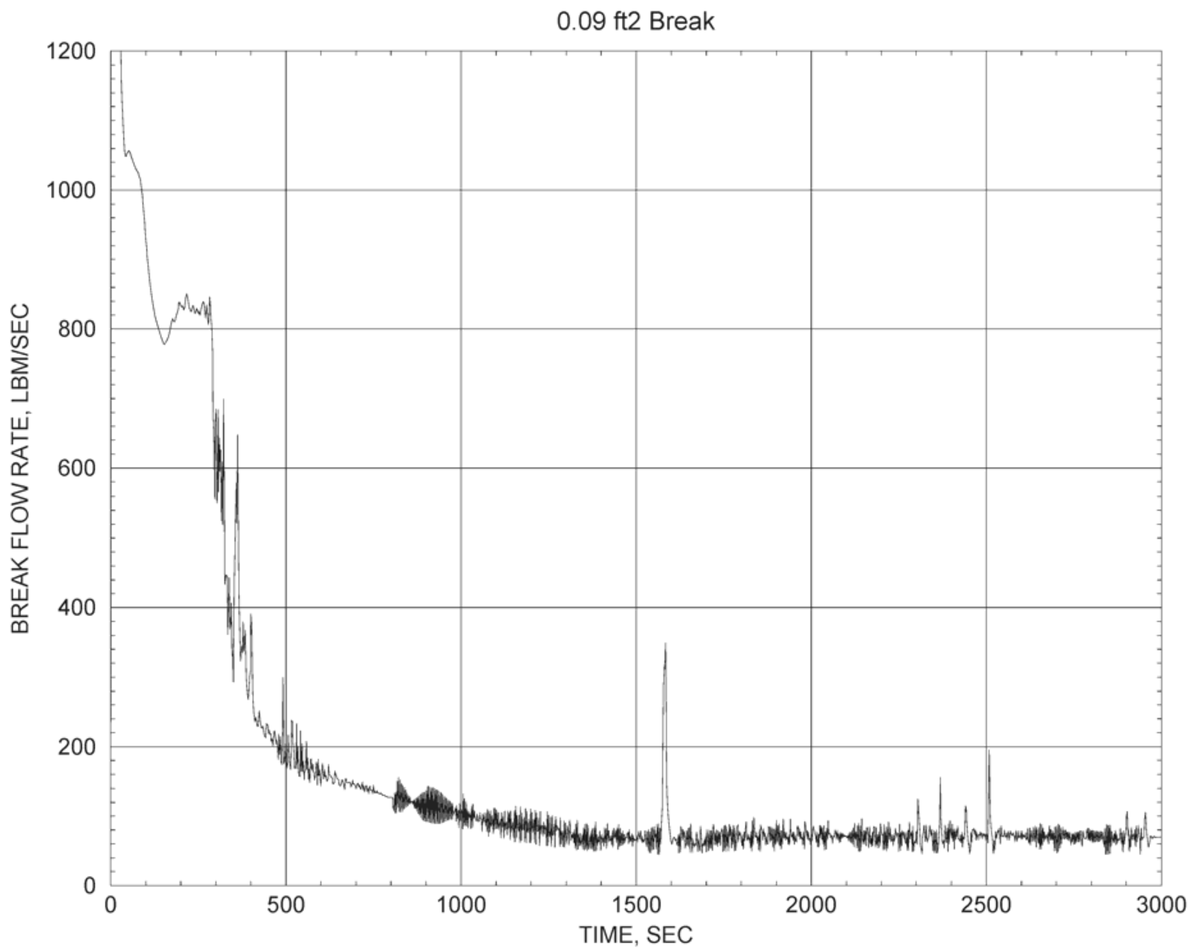
2000
1800
1600
1400
1200
1000
800
600
0
10
20
30
40
50
60



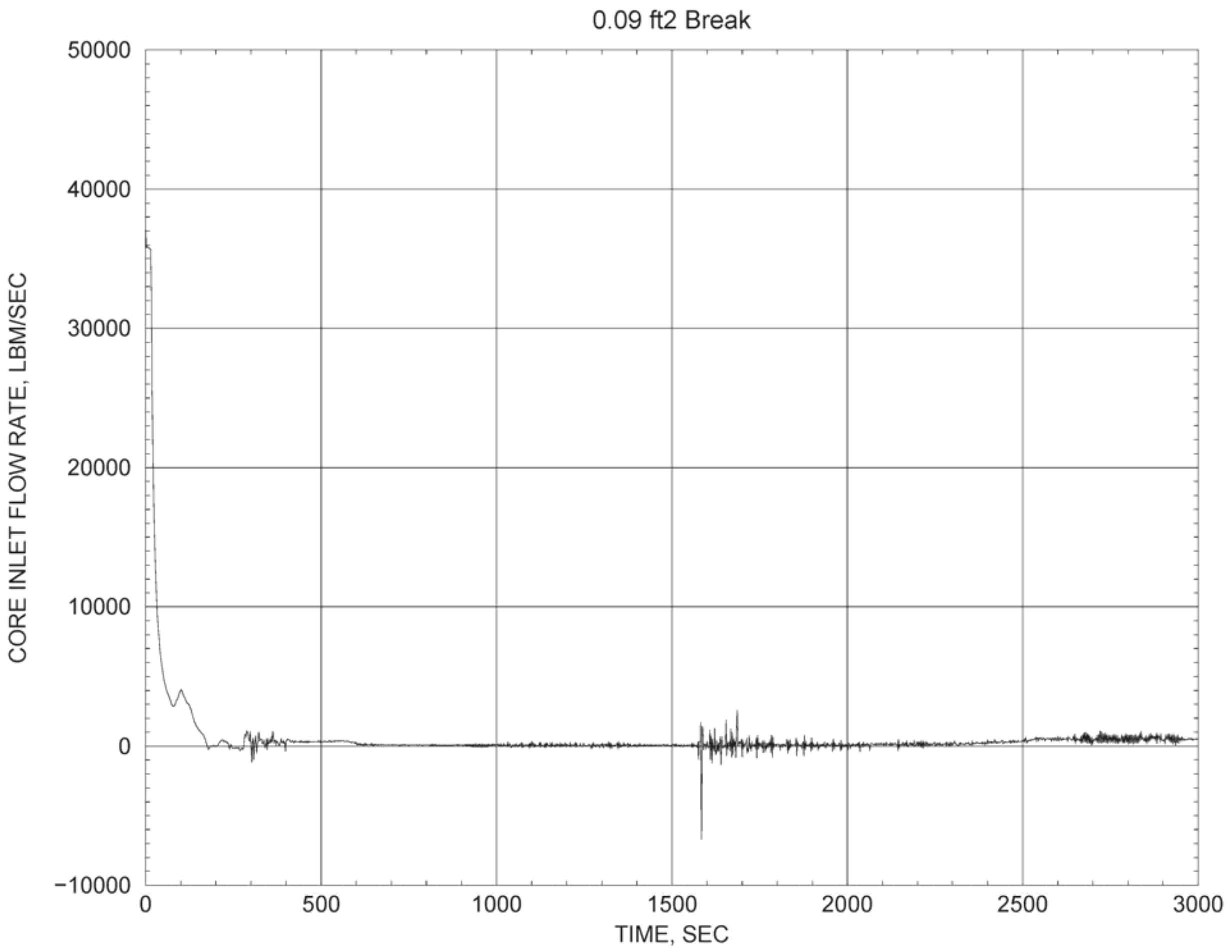
Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - REACTOR POWER	Figure 14.17-18 Rev. 44



Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - INNER VESSEL PRESSURE	Figure 14.17-19 Rev. 44



Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - BREAK FLOW RATE	Figure 14.17-20 Rev. 44



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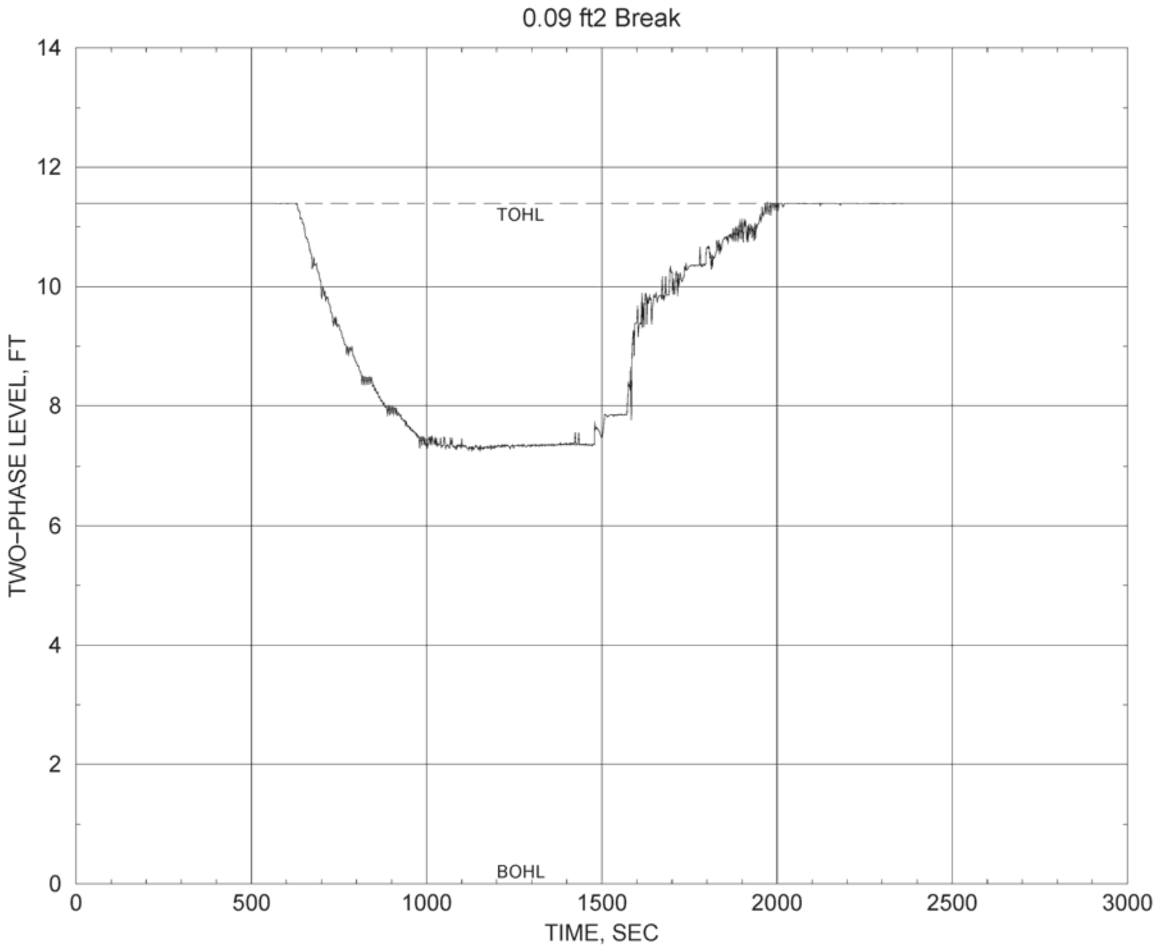

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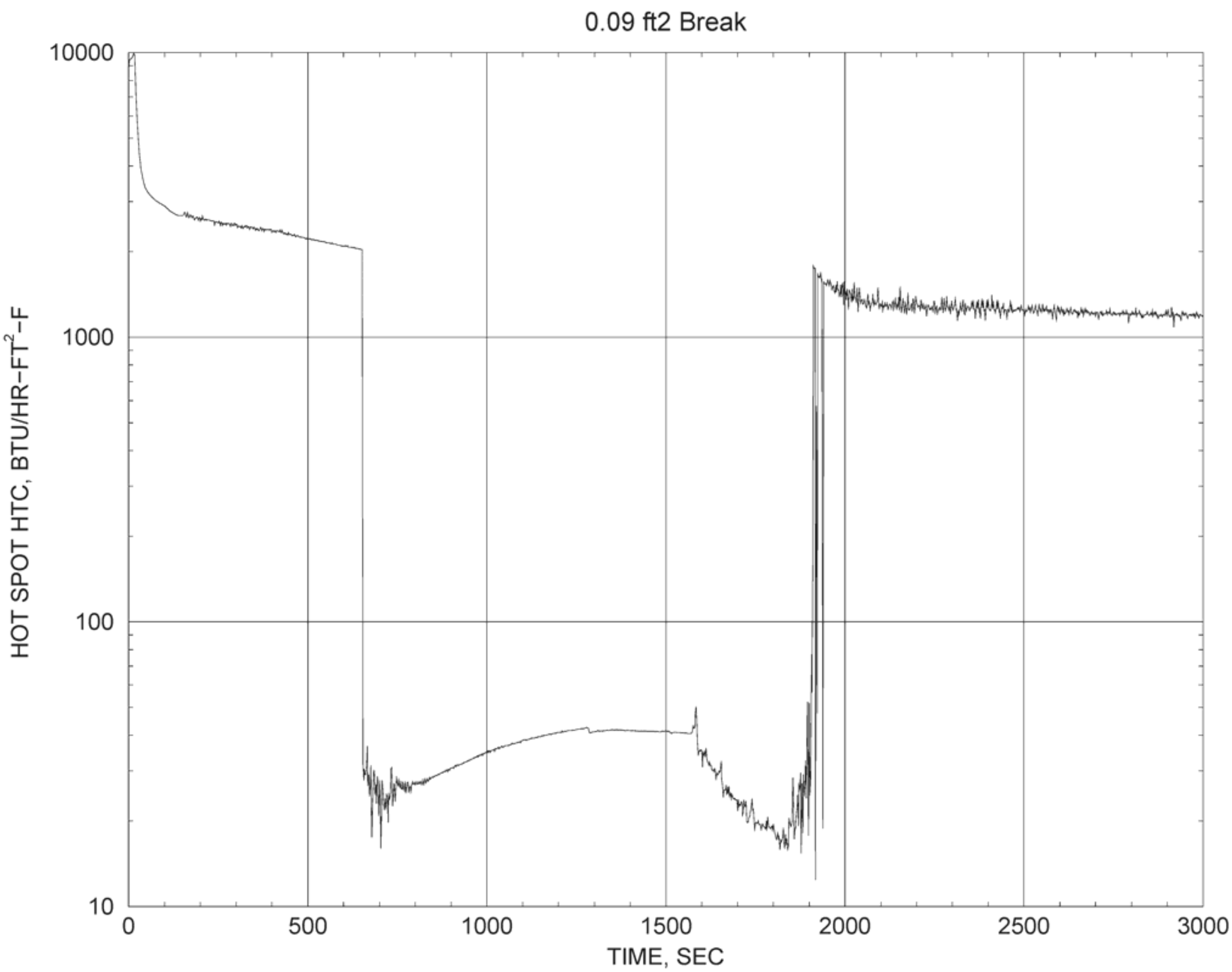
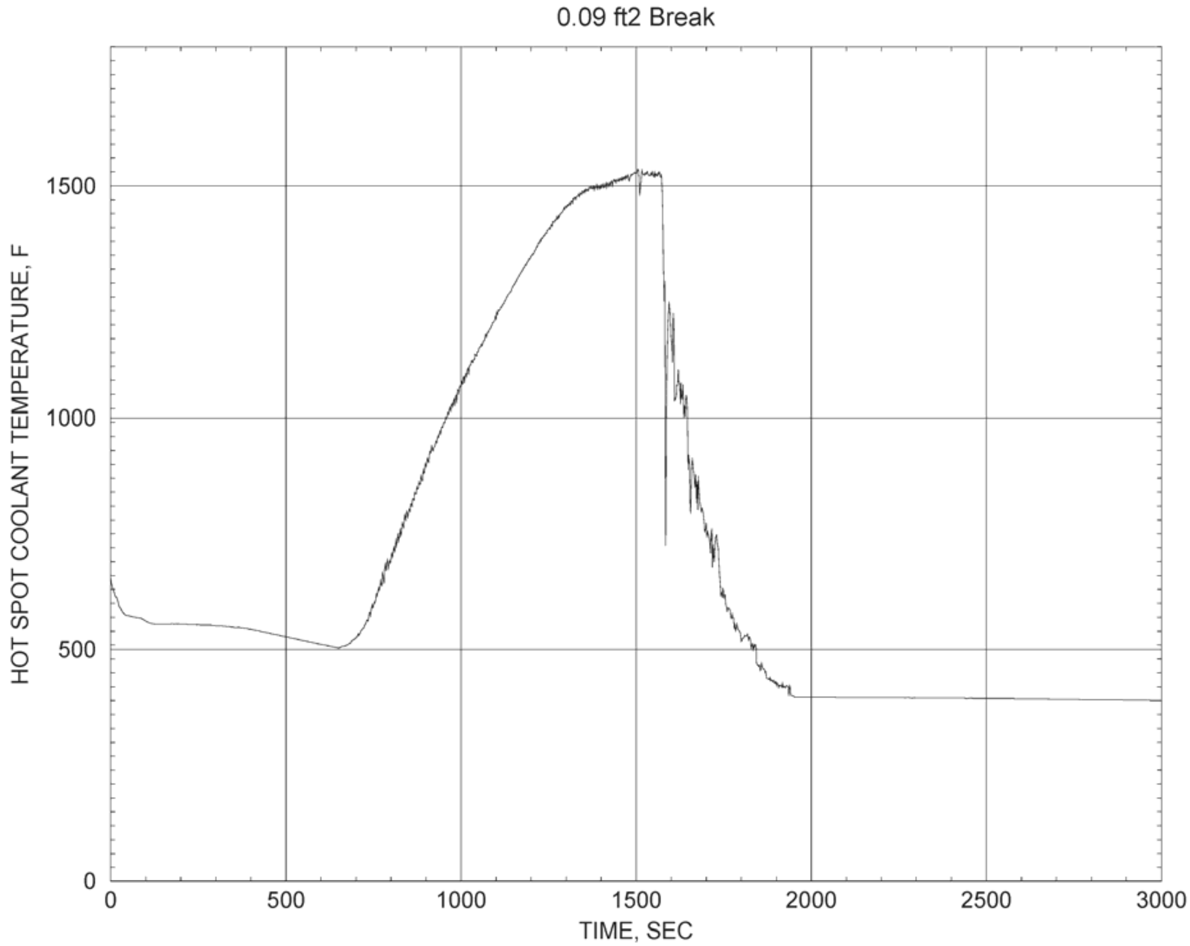
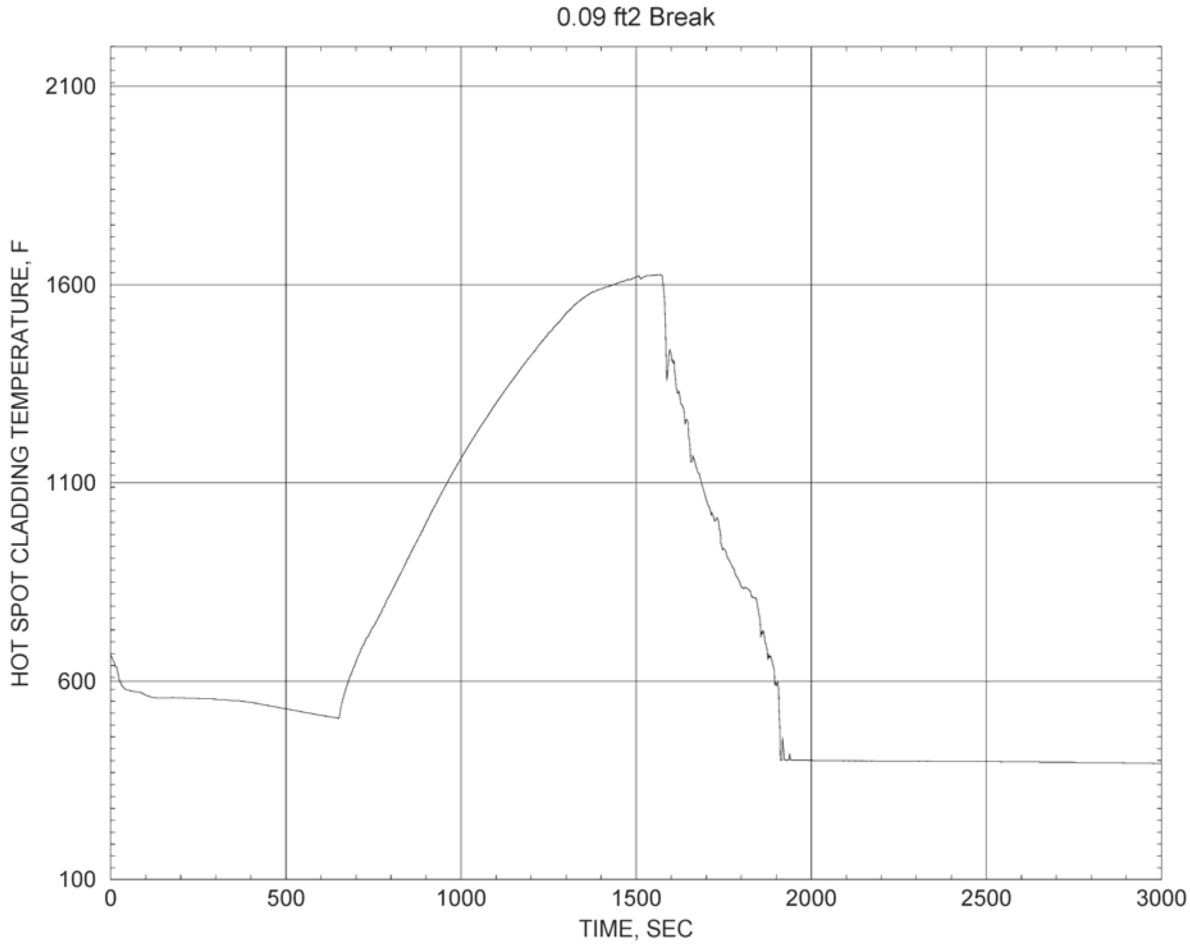


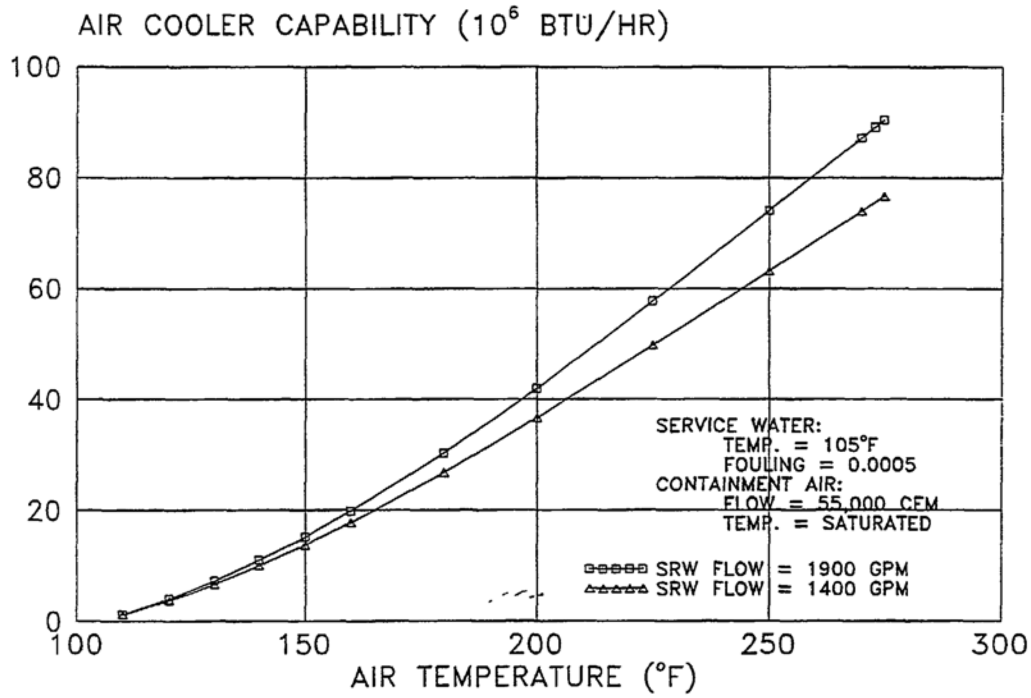
Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - HEAT TRANSFER COEFFICIENT AT HOT SPOT	Figure 14.17-23 Rev. 44



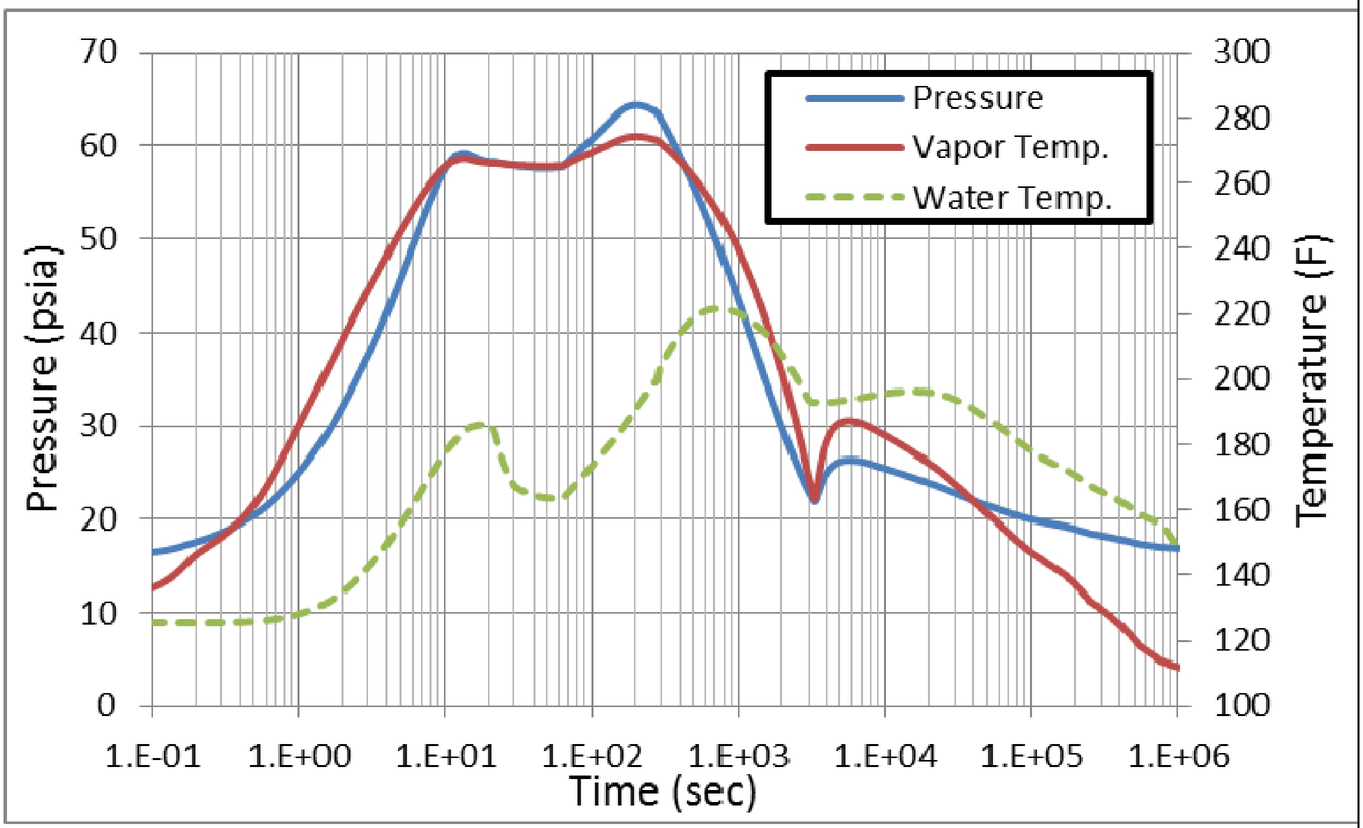
Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - COOLANT TEMPERATURE AT HOT SPOT	Figure 14.17-24 Rev. 44



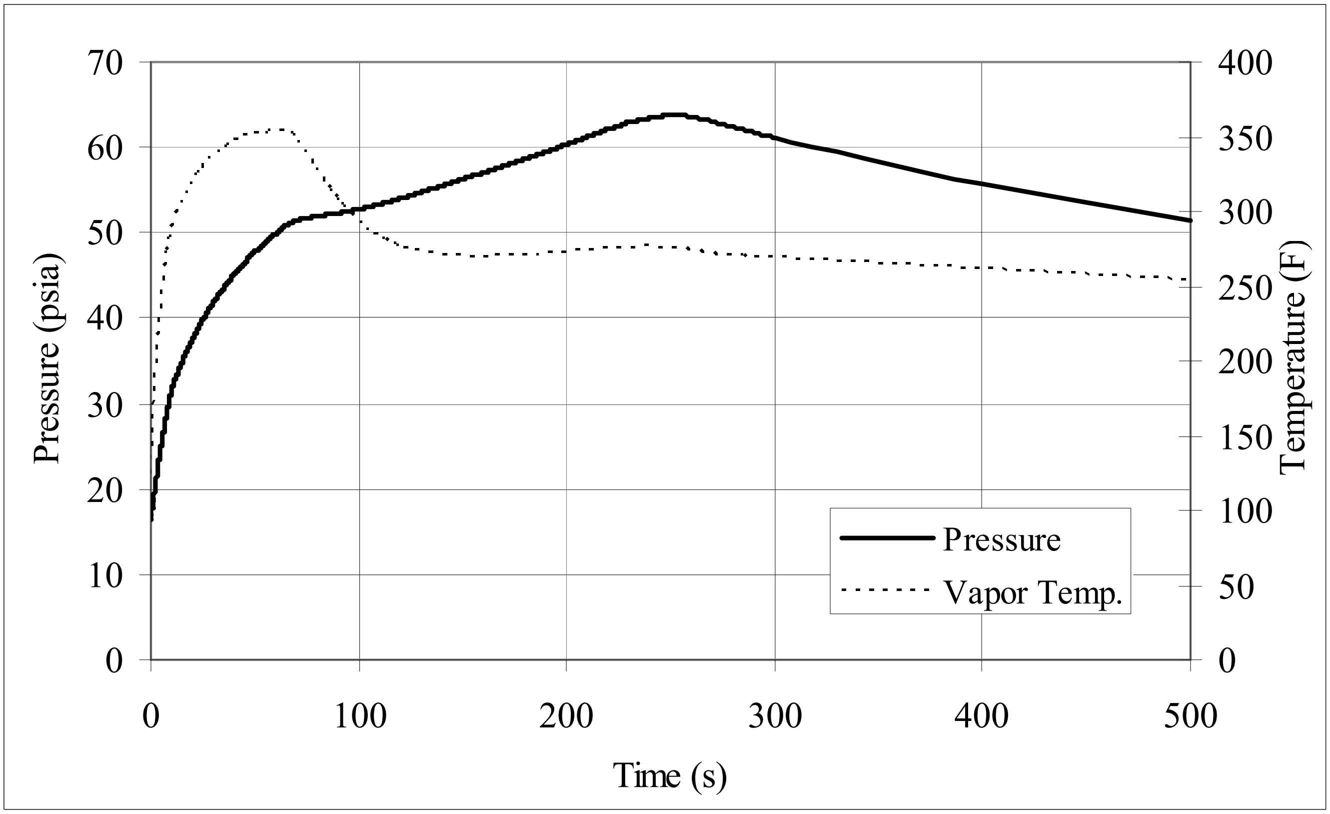
Calvert Cliffs Nuclear Power Plant	SMALL BREAK LOCA - ECCS PERFORMANCE ANALYSIS FOR THE LIMITING CASE - CLADDING TEMPERATURE AT HOT SPOT	Figure 14.17-25 Rev. 44



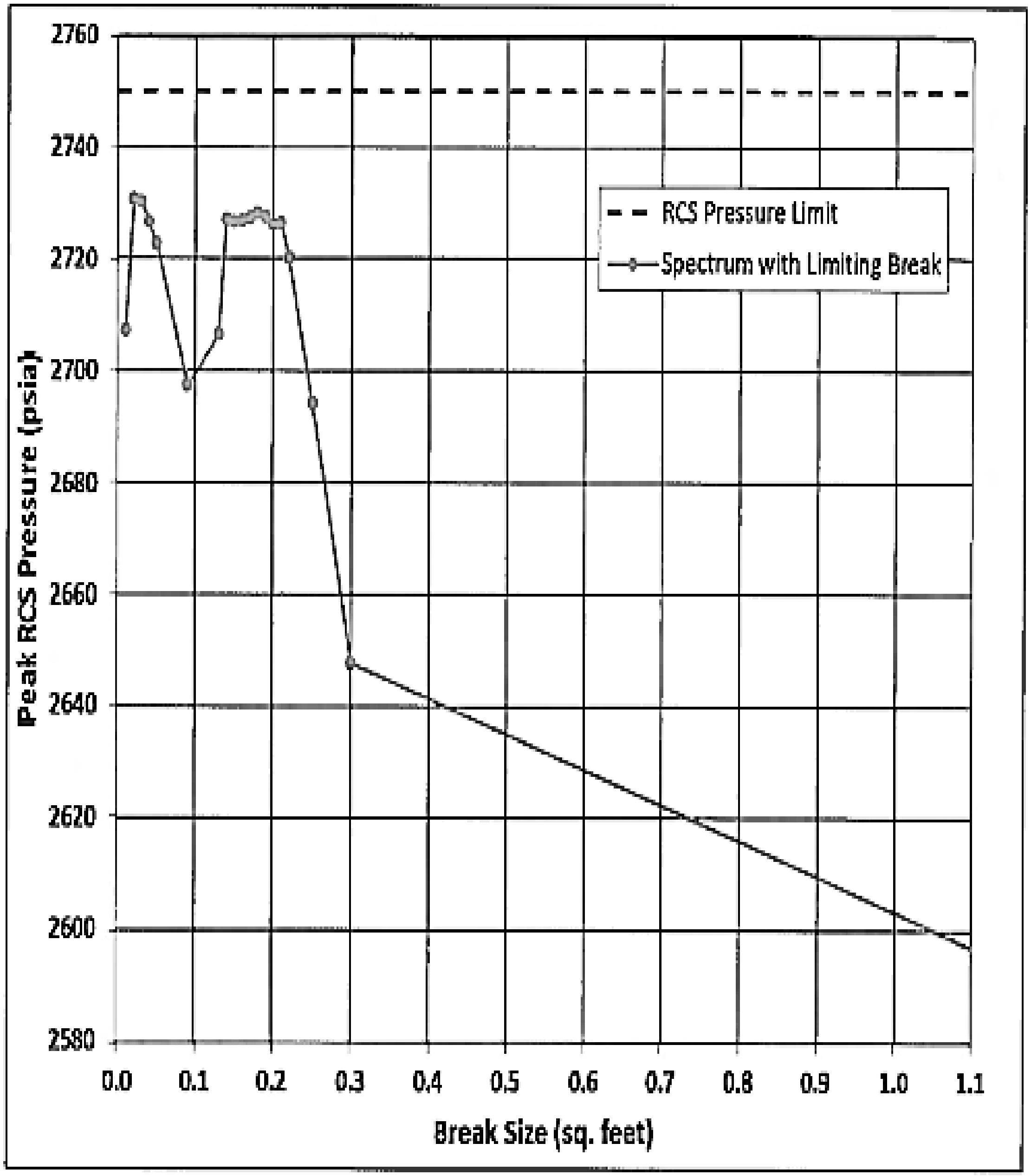
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CONTAINMENT AIR COOLER CAPABILITY	Figure 14.20-1 Rev. 17



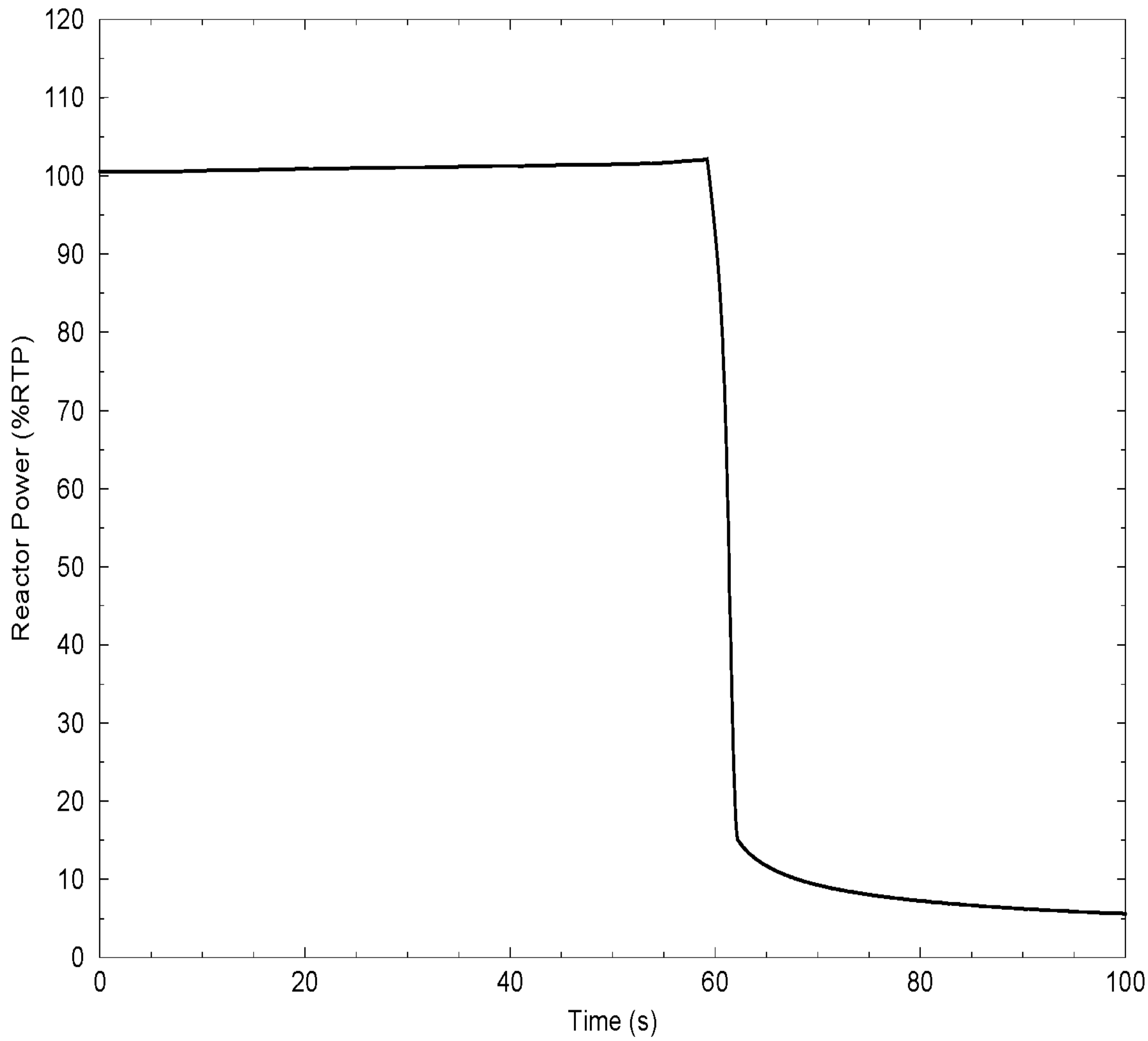
Calvert Cliffs Nuclear Power Plant	COLD LEG DISCHARGE LOCA – MAXIMUM SI, CONTAINMENT PRESSURE AND TEMPERATURE AND CONTAINMENT SUMP TEMPERATURE VERSUS TIME	Figure 14.20-2 Revision 47



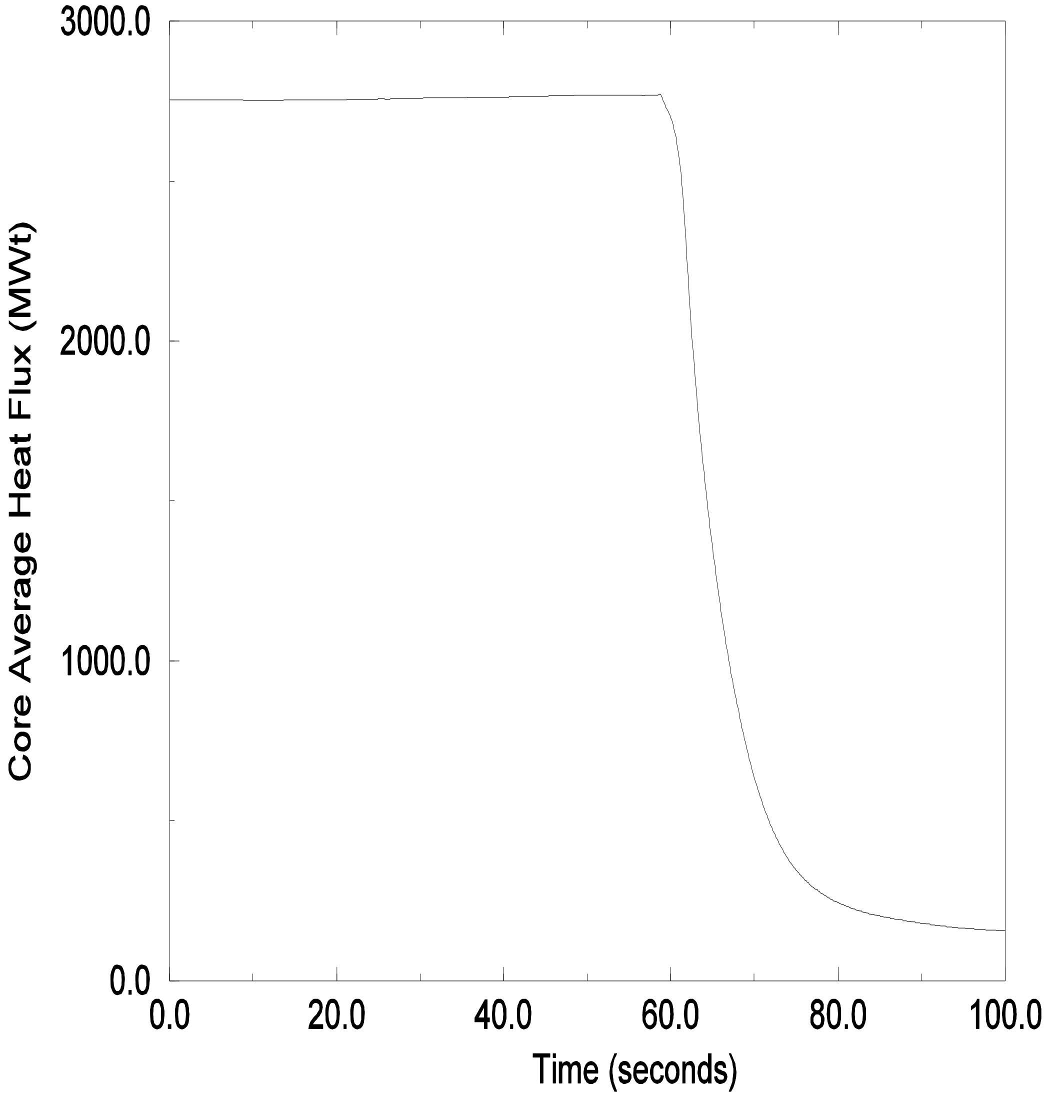
Calvert Cliffs Nuclear Power Plant	MAIN STEAM LINE BREAK, CONTAINMENT PRESSURE AND TEMPERATURE VERSUS TIME	Figure 14.20-3 Revision 47



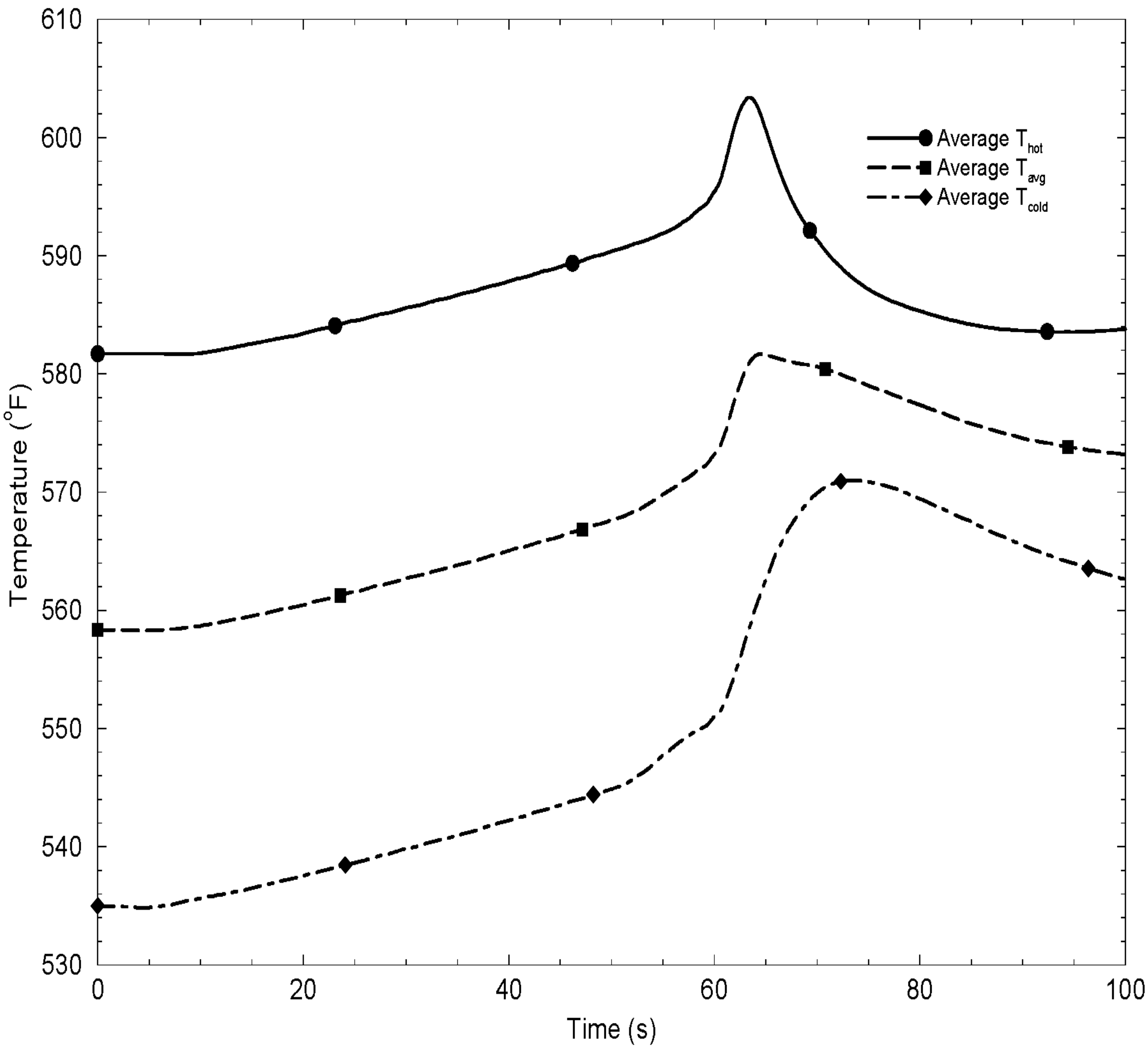
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP RCS PEAK PRESSURE VS BREAK SIZE	Figure 14.26-1 Revision 49



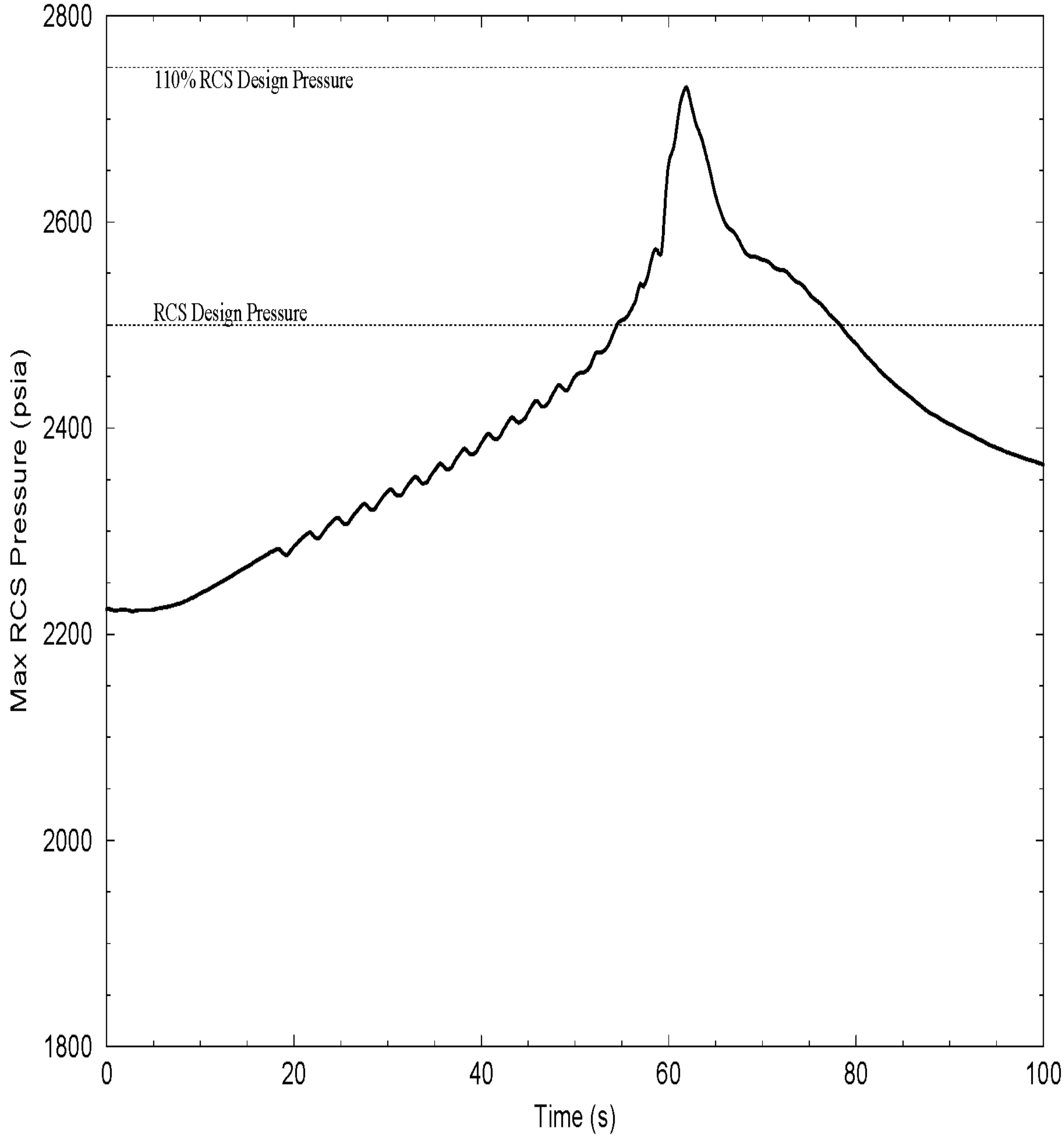
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP CORE POWER VS TIME	Figure 14.26-2 Revision 49



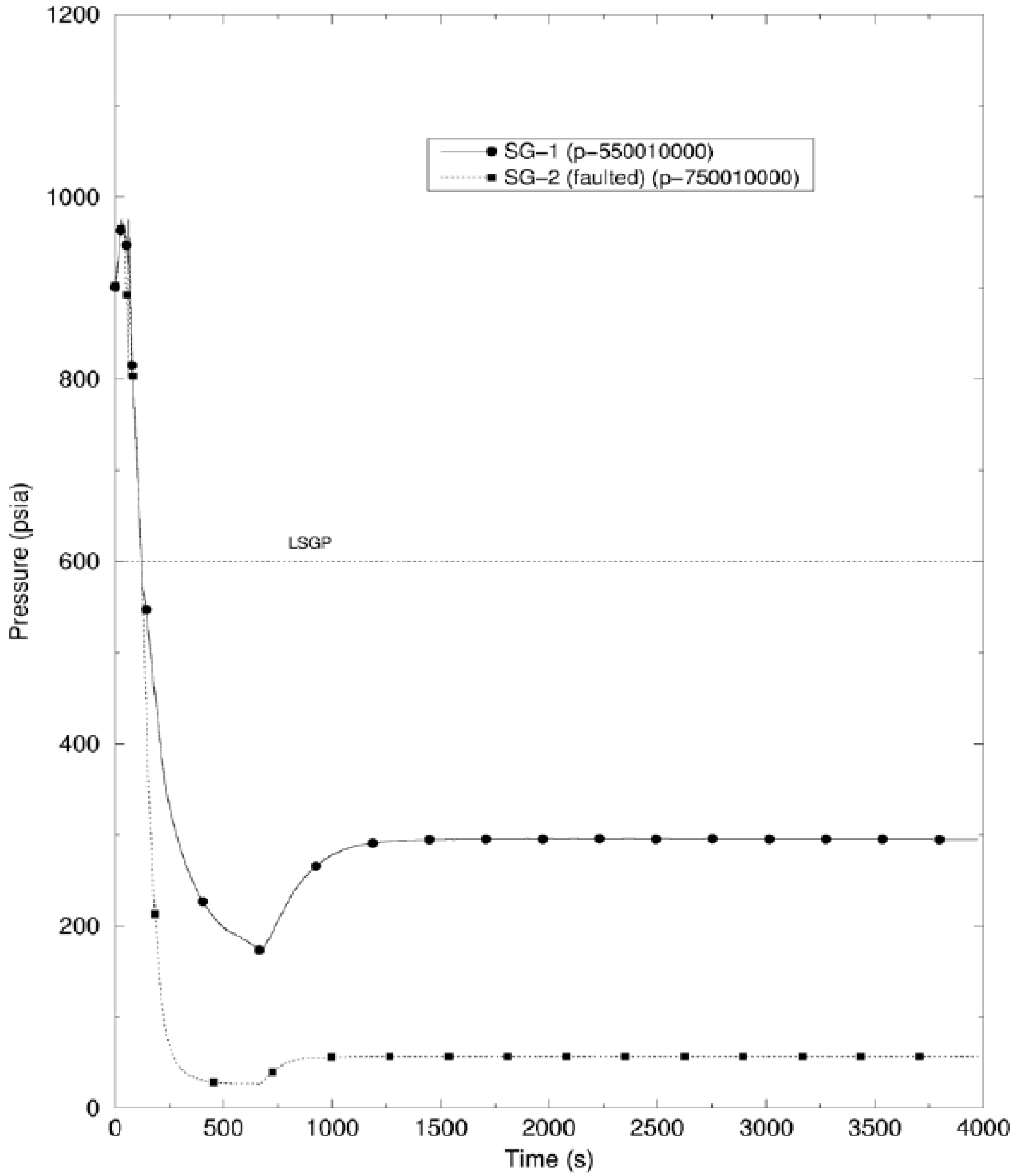
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP CORE AVERAGE HEAT FLUX VS TIME	Figure 14.26-3 Revision 49



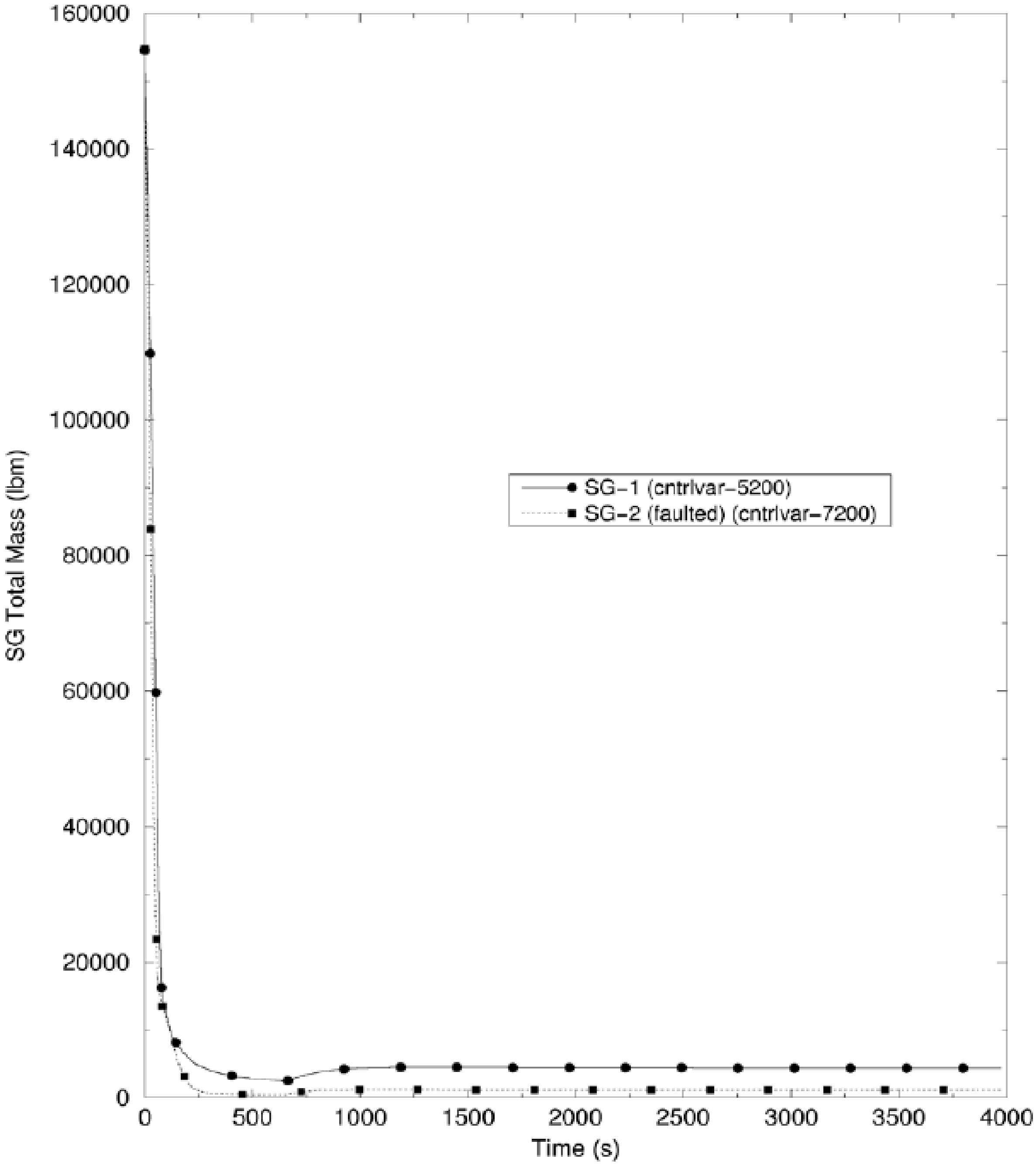
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP RCS TEMPERATURES VS TIME	Figure 14.26-4 Revision 49



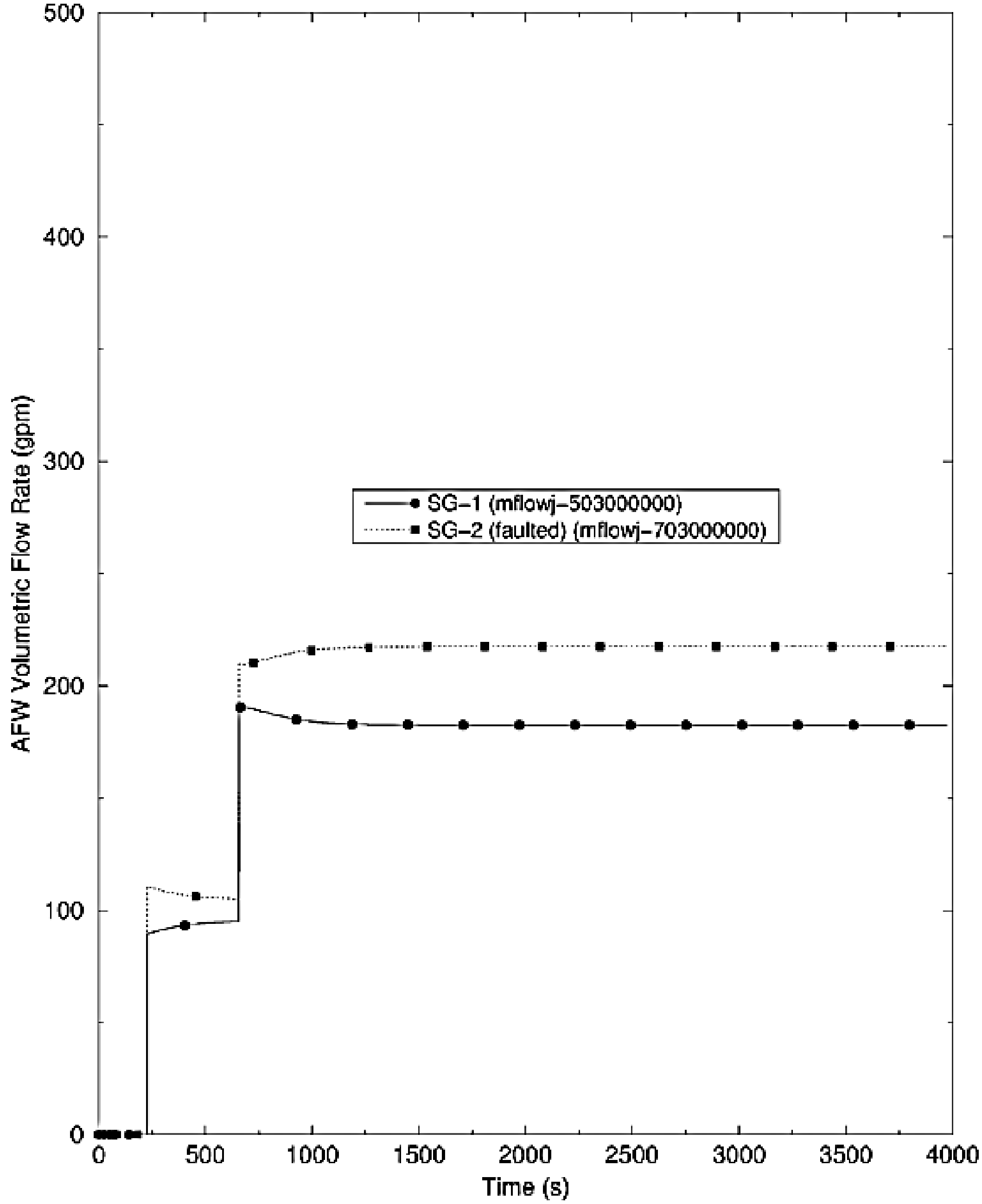
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP RCS PRESSURE VS TIME	Figure 14.26-5 Revision 49



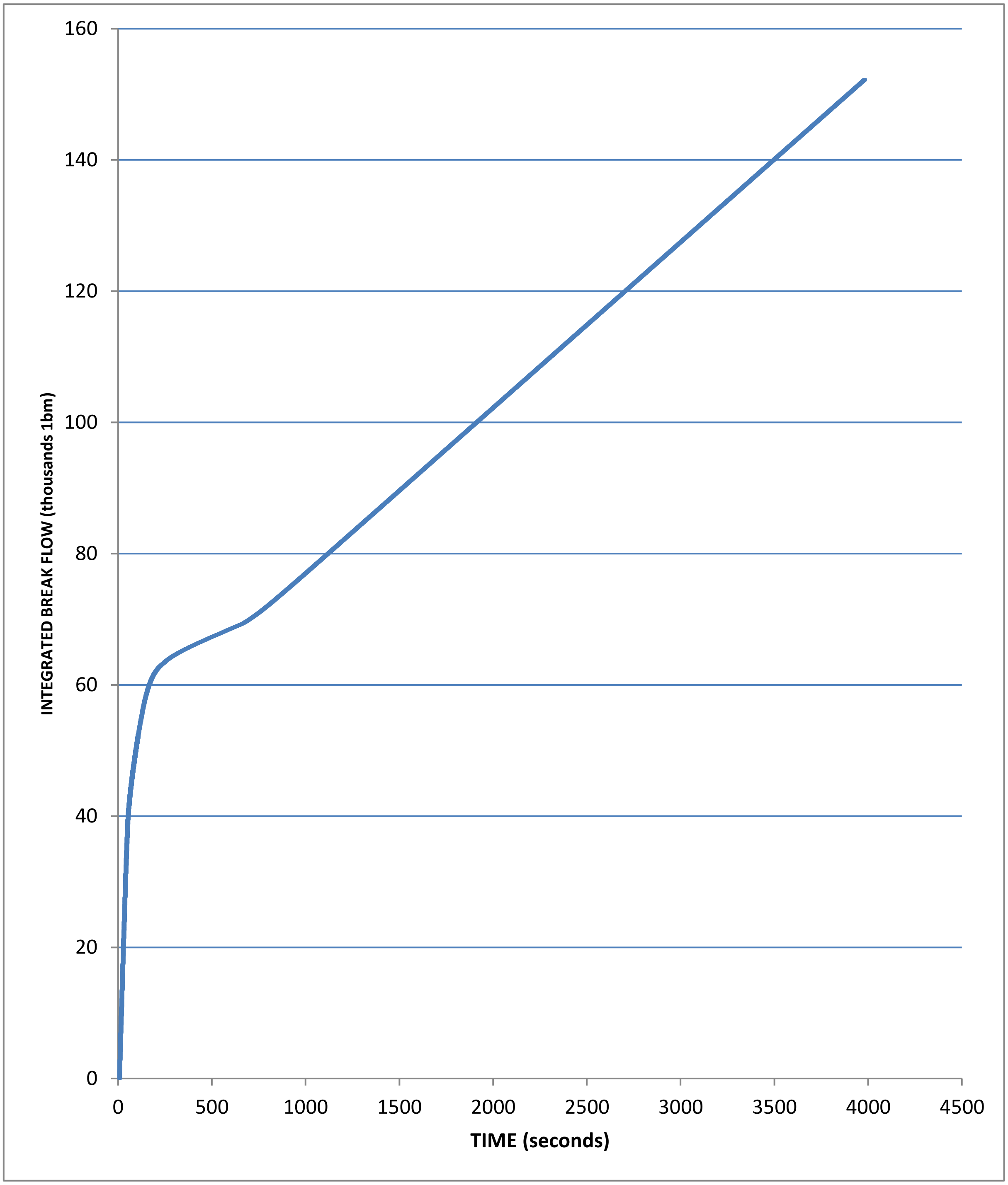
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH NO LOAC FOLLOWING REACTOR TRIP STEAM GENERATOR PRESSURE VS TIME	Figure 14.26-6 Revision 49



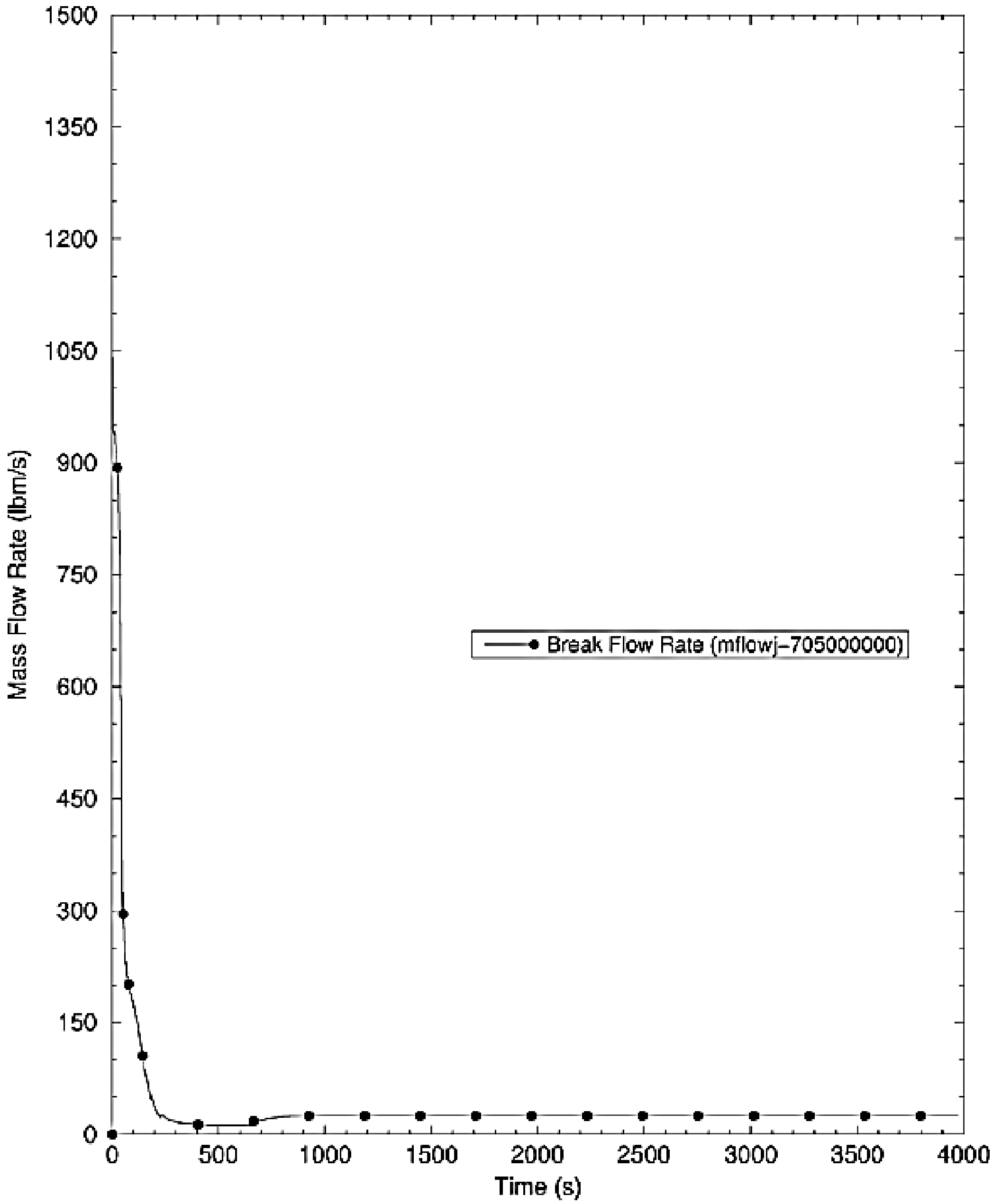
Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH NO LOAC FOLLOWING REACTOR TRIP STEAM GENERATOR INVENTORY VS TIME	Figure 14.26-7 Revision 49



Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH NO LOAC FOLLOWING REACTOR TRIP AUXILIARY FEEDWATER FLOW VS TIME	Figure 14.26-8 Revision 49



Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH NO LOAC FOLLOWING REACTOR TRIP INTEGRATED BREAK FLOW VS TIME	Figure 14.26-9 Revision 49



Calvert Cliffs Nuclear Power Plant	FEEDLINE BREAK EVENT WITH NO LOAC FOLLOWING REACTOR TRIP BREAK FLOW VS TIME	Figure 14.26-10 Revision 49

CHAPTER 15
TECHNICAL REQUIREMENTS MANUAL

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15.0 TECHNICAL REQUIREMENTS MANUAL

The Technical Requirements Manual consists of material that was removed from the Technical Specifications on conversion to Improved Standard Technical Specifications. The material removed was selected because it did not meet any of the four criteria the Nuclear Regulatory Commission has established for material that is in Technical Specifications. These four criteria, which set a level of safety significance for Technical Specification content, are detailed in 60 FR 36953. The four criteria are summarized in two general classes: (1) those related to the prevention of accidents, and (2) those related to mitigation of the consequences of accidents.

The Technical Requirements Manual is a stand-alone, licensee-controlled document, changes to which are controlled by the 10 CFR 50.59 process. It describes the normal operating condition for the systems and components listed below, and specifies actions that would be taken when these system and components are not in their normal conditions. These actions are consistent with the guidance provided in Nuclear Regulatory Commission Generic Letter 91-18 for degraded and nonconforming conditions and will the requirements of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance,” regarding removal of equipment from service for maintenance or testing.

The following systems, components, and process limits are addressed in the Technical Requirements Manual:

1. Boron dilution and flow paths
2. Control element assembly position indication
3. Radiation monitoring instrumentation
4. Meteorological instrumentation
5. Incore Detector System
6. Seismic monitoring instrumentation
7. Fire detection instrumentation
8. Reactor Coolant system chemistry
9. Pressurizer pressure/temperature limits
10. American Society of Mechanical Engineers Code components
11. Deleted
12. Letdown line excess flow
13. Reactor Coolant System vents
14. Containment structural integrity
15. Steam generator pressure/temperature limits
16. Snubbers
17. Sealed source contamination
18. Watertight doors
19. Fire suppression water system
20. Spray and sprinkler system
21. Halon system
22. Fire hose stations
23. Yard fire hydrants and hydrant hose houses
24. Fire barrier penetrations
25. Fuel decay time
26. Refueling communications
27. Refueling machine
28. Spent fuel pool crane travel
29. Explosive gas mixtures
30. Gas storage tanks
31. Fire detection instruments
32. Snubber visual inspection interval
33. Sprinkler locations
34. Fire hose stations

CHAPTER 16
LICENSE RENEWAL

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LIST OF ACRONYMS

AFW	Auxiliary Feedwater
ARDI	Age-Related Degradation Inspection
ARDM	Age-Related Degradation Mechanisms
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BACI	Boric Acid Corrosion Inspection
CASS	Cast Austenitic Stainless Steel
CC	Component Cooling
CCNPP	Calvert Cliffs Nuclear Power Plant
CE	Combusting Engineering, Inc.
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CVCS	Chemical and Volume Control System
DFO	Diesel Fuel Oil
EDG	Emergency Diesel Generator
EQ	Environmentally Qualified
FMP	Fatigue Monitoring Program
FOST	Fuel Oil Storage Tank
FP	Fire Protection
FWS	Feedwater System
HVAC	Heating, Ventilation, and Air Conditioning
IA	Instrument Air
ICI	Incore Instrumentation
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
ISI	Inservice Inspection
LLRT	Local Leak Rate Test
LRA	License Renewal Application
LOCA	Loss-of-Coolant Accident
MIC	Microbiologically-Induced Corrosion
MOV	Motor-Operated Valves
MRP	Materials Reliability Program
MSIV	Main Steam Isolation Valve
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSR	Non-safety Related
NSSS	Nuclear Steam Supply System
PEO	Period of Extended Operation
PM	Preventive Maintenance
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RRM	Reactor Refueling Machine

LIST OF ACRONYMS

RVI	Reactor Vessel Internals
RVLMS	Reactor Vessel Level Monitoring System
RWT	Refueling Water Tank
SA	Starting Air
SBO	Station Blackout
SCC	Stress Corrosion Cracking
SDC	Shutdown Cooling
SFHM	Spent Fuel Handling Machine
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SG	Steam Generator
SI	Safety Injection
SIT	Safety Injection Tank
SR	Safety-Related
SRW	Service Water
STP	Surveillance Test Procedure
SW	Saltwater

16.0 LICENSE RENEWAL

16.1 INTRODUCTION

This chapter contains summary descriptions of the Aging Management Programs (AMPs) and associated activities credited for managing the effects of plausible Age Related Degradation Mechanisms (ARDMs) of applicable structures, systems and components (SSCs) described in the Calvert Cliffs Nuclear Power Plant (CCNPP) License Renewal Application (LRA). This chapter also contains summary descriptions of the Time-Limited Aging Analyses (TLAAs).

Section 16.2 presents summary descriptions of the AMPs credited for managing plausible ARDMs, of applicable SSCs, as required by 10 CFR 54.21(d).

Section 16.3 presents summary descriptions of the TLAAs identified and evaluated during the Part 54 Integrated Plant Assessment (IPA) and included in the CCNPP LRA, as required by 10 CFR 54.21(d). It also contains a new TLAA that was identified during license renewal implementation.

Section 16.4 presents periodic update information as required by 10 CFR 54.37(b).

Section 16.5 presents a listing of 'General References.' These 'General References' provide background information/additional detail to support further research into the information presented in this UFSAR Chapter.

16.2 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

This section presents summary descriptions of the AMPs and associated activities credited for managing the effects of plausible ARDMs of applicable SSCs.

The majority of AMPs credit existing plant programs, procedures and associated activities. A number of AMPs credit existing plant programs, procedures and associated activities that have been modified prior to entering the Period of Extended Operation (PEO) for managing specific plausible ARDMs. A few AMPs credit new programs, procedures and associated activities established, prior to entering the PEO, to manage plausible ARDMs where existing plant programs, procedures and associated activities did not exist.

Common to all AMPs is the use of the Corrective Action Process to identify and resolve discovered degraded/non-conforming conditions and the use of the Operating Experience (OPEX) Process to identify and assess internal and external OPEX related to potential aging issues for applicability to CCNPP.

Table 16-2 presents the AMPs and the associated SSCs. Table 16-2 is arranged to align with the applicable LRA section number.

Table 16-2 contains the following 8 columns:

1. Line Item Number
2. Applicable LRA Section
3. Applicable SSC
4. Portion(s) of the SSC subject to a particular aging mechanism,
5. Aging mechanism(s) that is plausible for the SSC
6. Credited AMP

7. Reference to the applicable AMP Summary Description and, as applicable, attributes specific to the associated SSC line item
8. Implementation schedule.

NOTE: CCNPP Renewed Licenses for Units 1 and 2 (Condition 2.G) stipulated that the UFSAR be updated to include:

- Any future actions (i.e., regulatory commitments) listed in NUREG-1705; Appendix E (Reference 1).
- Section 16.2 and Table 16-2 document completion of these ‘future actions’ (’regulatory commitments’) prior to entering each Unit’s PEO.

16.2.1 ADDITIONAL BASELINE WALKDOWNS AGING MANAGEMENT PROGRAM

This is a new AMP consisting of a one-time inspection of components in the ‘Component Supports’ commodity group that were not subject to inspection under the In-Service Inspection (ISI) or the Seismic Verification Project (SVP). Fourteen systems within the scope of license renewal contain piping supports not subject to ASME Section XI Inservice Inspection (ISI) inspections or one-time inspection performed as part of the SVP walkdowns.

For these component supports (i.e., not inspected or partially inspected by the SVP walkdowns or subject to inspections under the ISI Program, and environmental or other differences prevented extrapolation of results to cover these component supports) additional one-time baseline walkdowns were performed. The additional baseline walkdown scope included inspection (visual examinations), on a sampling basis, for aging effects (i.e., corrosion and loose bolts).

The sampling approach was comparable to the approach required by ASME Class 3 systems. These walkdowns documented the condition of the piping support type, not including piping frames outside containment. If an aging effect was found during this one-time inspection, additional sampling one-time inspections for piping hangers outside containment. In addition, the inspection scope for that system was expanded to pipe frames outside containment.

The systems included in the additional baseline walkdown program were as follows:

Condensate	Liquid Waste
Condensate Storage	Nitrogen and Hydrogen
Compressed Air	Nuclear Steam Supply Sampling
Demineralized Water	Plant Drains
Diesel Fuel Oil	Plant Heating
Extraction Steam	Plant Water
Fire Protection	Well and Pretreated Water

This AMP has been completed and does not continue into either unit’s PEO.

NOTE: Based on the results of the walkdowns associated with this AMP, aging management of applicable component supports within the applicable systems have been implemented via the Structure and System Walkdown AMP (Section 16.2.29).

16.2.2 AGE-RELATED DEGRADATION INSPECTION (ARDI) AGING MANAGEMENT PROGRAM

This is a new AMP consisting of inspections, on a sampling basis, of mechanical systems, consistent with NRC GALL AMPs XI.M32 and XI.M35, based on groups of components comprised of the same material/environment combination, regardless of the system or unit where they are installed. All of the components in each of these material/environment combination groups are subject to the same plausible aging effects requiring management.

This is a discovery AMP that is typically credited in combination with a mitigation program. This AMP is typically paired with the Chemistry Program. The Water Chemistry AMP (Section 16.2.12) is utilized to mitigate corrosion in water filled piping systems. The Diesel Fuel Oil Chemistry AMP (Section 16.2.16) provides a similar mitigation function for Diesel fuel oil filled piping systems. For each identified controlled chemical environment, ARDI sampling inspections were used to validate that the Water and Fuel Oil Chemistry AMPs have mitigated plausible aging effects requiring mitigation (AERMs) from occurring.

There are a few systems included in the ARDI AMP that do not have a controlled chemical environment. For these systems, and associated material/environment combination groups, site operating experience had been that no AERMs have been occurring. The ARDI AMP is utilized to validate the plant's operating history of the non-presence of AERMs in these systems for the applicable material/environment combination groups. This is consistent with the One-Time Inspection program (Section XI.M32) guidance in Rev. 1 of NUREG-1801.

This AMP was initially based on Electric Power Research Institute (EPRI) TR-107514, *Age-Related Degradation Inspection Method and Demonstration: In Behalf of Calvert Cliffs Nuclear Power Plant License Renewal Application,* which based sampling methodology on statistical methods that provided a 90% probability that 90% of the components in a material/environment combination group would not exhibit the applicable potential AERMs. For the number of components that the methodology algorithms indicated were required to be inspected out of the total population of components within a material/environment combination group, as long as no AERMs were discovered for the components inspected, then the 90%/90% confidence criteria was satisfied and no further inspections of any kind were required during the PEO.

The systems that are included in this AMP are as follows:

Auxiliary Building HVAC
Auxiliary Feedwater
Chemical Addition
Chemical and Volume Control
Component Cooling
Compressed Air
Condensate
Condensate Storage
Containment Heating and Ventilation
Containment Spray
Control Room HVAC
Demineralized Water
Emergency Diesel Generator
EDG Building HVAC
Liquid Waste
Main Steam
Nitrogen and Hydrogen
Nuclear Steam Supply Sampling
Plant Drains
Plant Heating
Plant Water
Primary Containment HVAC
Radwaste HVAC
Radiation Monitoring
Reactor Coolant
Safety Injection
Saltwater
Service Water

Feedwater	Spent Fuel Pool Cooling
Fire Protection	Waste Gas

NUREG-1801; Revision 2 established a standardized sampling approach to the performance of One-Time Inspections (NUREG-1801, Section XI.M32). This sampling approach provides for the inspection of 20% of the total number of components in each material/environment combination group, not to exceed a maximum of 25 inspections for each group. CCNPP adopted this sampling approach.

Small bore piping inspections were also completed under this AMP consistent with the One-Time Inspection of ASME Code Class 1 Small Bore Piping guidance provided in NUREG-1801, Rev. 1 (Section XI.M35).

This AMP has been completed and does not continue into either unit's PEO.

NOTE: Based on the results of the ARDI inspections associated with this AMP, aging management of applicable components not meeting ARDI criteria, has been implemented via the Preventive Maintenance AMP (Section 16.2.23).

16.2.3 ALLOY 600 AGING MANAGEMENT PROGRAM

This AMP credits existing plant program (i.e., Alloy 600 Program). The Alloy 600 Program was developed after several instances of plant-specific and industry issues, relative to the primary water stress corrosion cracking (PWSCC) of Alloy 600 components, were identified.

This AMP is consistent with the Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program guidance in NUREG-1801, Rev. 2, (Section XI.M11B) with the enhancement that this AMP includes the RCS nozzle thermal sleeves and other non-pressure boundary components. In addition, welds and base metals are implicitly included.

Much of the Alloy 600 material (including all of the pressurizer heater sleeves) at the time the units' licenses were renewed, has been replaced with Alloy 690 and the RPV Head replacements eliminated Alloy 600 that was of concern in the previous design. This AMP includes all Nickel-Chromium (Ni-Cr) based alloys in the primary systems.

Specifically, this AMP inspects RCS nozzles during refueling outages for indications of leakage through the performance of the Boric Acid Corrosion Inspection (BACI) AMP (Section 16.2.7). The Alloy 600 Program also contains provisions for augmented inspections based on 10 CFR 50.55a.

16.2.4 ASME SECTION XI IN-SERVICE INSPECTION (ISI), SUBSECTIONS IWB, IWC, AND IWD AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., ASME Section XI). The ASME Section XI, Subsections IWB, IWC and IWD programs consist of periodic volumetric, surface, and/or visual examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components for evidence of degradation. Surface and visual examinations of integral attachments are also performed. The ASME Section XI program is implemented through the ISI Program per the requirements of 10 CFR 50.55(a). As such, the program is consistent with the ASME Section XI, Subsections IWB, IWC, and IWD program guidance in NUREG-1801, Rev. 2 (Section XI.M1).

16.2.5 ASME SECTION XI IN-SERVICE INSPECTION (ISI), SUBSECTIONS IWE AND IWI AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., ASME Section XI). The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of steel containment shells and their attachments, containment steel liners and their attachments, and containment air locks, hatches, and pressure retaining bolting for evidence of degradation. The ASME Section XI, Subsection IWL program consists of periodic visual inspection of concrete surfaces for reinforced and pre-stressed concrete containments, and periodic visual inspection and sample tendon testing of unbounded post-tensioning systems for pre-stressed concrete containments, for evidence of degradation, assessment of damage, and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with Regulatory Guide 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments*. The ASME Section XI program is implemented through the ISI Program per the provisions and requirements of 10 CFR 50.55(a). As such, the program is consistent with the ASME Section XI, Subsections IWE and IWL program guidance in NUREG-1801, Rev. 2 (Sections XI.S1 and XI.S2, respectively).

This AMP utilizes the Surveillance Testing AMP (Section 16.2.30) and Preventive Maintenance AMP (Section 16.2.23) to facilitate the periodic tests and inspections.

16.2.6 ASME SECTION XI IN-SERVICE INSPECTION, SUBSECTION IWF AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., ASME Section XI). The ASME Section XI, Subsection IWF program consists of periodic visual examinations of supports, associated with Class 1, 2, 3, and Metal Containment (MC) piping and components, for evidence of degradation. The ASME Section XI program is implemented through the ISI Program per the requirements of 10 CFR 50.55(a). As such, the program is consistent with the ASME Section XI, Subsection IWF program guidance in NUREG-1801, Rev. 2 (Section XI.S3).

16.2.7 BORIC ACID CORROSION INSPECTION (BACI) PROGRAM

This AMP credits an existing plant program (i.e., Boric Acid Corrosion Control). The Boric Acid Corrosion Control Program consists of discovery and management of general corrosion/oxidation and corrosion due to boric acid exposure, for those systems containing boric acid, by performing visual inspections. The scope of the Boric Acid Corrosion Control program is as follows: (a) identifies locations to be examined, (b) provides examination requirements and procedures for the detection of leaks, and (c) provides the responsibilities for initiating engineering evaluations and the necessary corrective actions.

During each refueling outage, designated personnel perform walkdown inspections to identify and quantify any leakage found at specific locations inside containment and in the Auxiliary Building. A second inspection of these components is performed prior to plant start-up (at normal operating pressure and temperature) if leakage was identified previously and corrective action taken. If either leakage or corrosion is discovered, the condition is entered into the corrective action process to document and resolve the deficiency. Corrective actions address the removal of boric acid residue and inspection of the affected components for general corrosion.

The Boris Acid Corrosion Control Program scope includes Reactor Coolant System (RCS) components in accordance with Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,* as well as non-RCS

mechanical, electrical, and structural components susceptible to boric acid corrosion which are potentially exposed to borated water leaks.

The Boric Acid Corrosion Control Program has been modified to include examinations during refueling outages of: (1) the reactor vessel cooling shroud anchorage to the reactor vessel head for evidence of boric acid leakage, (2) all reactor vessel cooling shroud structural support members for general corrosion/oxidation and (3) piping supports associated with the Spent Fuel Pool (SFP) demineralizer and filter.

An engineering review for stress corrosion cracking (SCC) at the Refueling Water Tank (RWT) penetrations determined that the RWT penetrations will leak before break. This engineering review confirmed that detection of minor leakage by preventive maintenance (PM) based periodic visual inspections performed will adequately manage the RWT penetration and associated welds prior to a challenge to the structural integrity under design basis conditions.

This AMP utilizes the Preventative Maintenance AMP (Section 16.2.23) to facilitate periodic BACIs and periodic RWT penetration/weld inspections.

16.2.8 BURIED PIPING INSPECTION AGING MANAGEMENT PROGRAM

This is a new AMP for buried pipe in the Auxiliary Feedwater and Diesel Fuel Oil Systems. This AMP provides assurance that the effects of plausible ARDMs are effectively managed in the PEO so that this buried piping remains capable of maintaining the system's pressure boundary function under all applicable current licensing basis loading conditions.

This AMP considers variations in environmental conditions (including cathodic protection) to select representative samples of buried piping (external surfaces) for inspection to ensure that the piping wrapping/coating and cathodic protection are adequately protecting the pipe from the external environment. Evidence of the effects of crevice corrosion, galvanic corrosion, general corrosion MIC or pitting will initiate corrective actions in accordance with the Corrective Action Program.

This AMP utilizes the Preventative Maintenance AMP (Section 16.2.23) to facilitate the piping inspections.

16.2.9 CABLE AGING MANAGEMENT PROGRAM

This is a new AMP initially developed to address a number of cables that were identified, during the IPA, as candidates for replacement before the end of the CCNPP Unit PEOs. CCNPP has since defined a cable condition monitoring AMP that provides the necessary inspection, testing, analysis and acceptance criteria to ensure that plausible ARDMs associated with in-scope cables are adequately managed in the PEO. Prior to entering the PEO, this AMP has been updated to be consistent with the Electrical Cables and Connections not Subject to 10 CFR 50.49 Environmental Qualification (EQ) Requirements, Non-EQ Electrical Cables and Connections Used in Instrumentation Circuits, and Non-EQ Inaccessible Power Cables program's guidance provided in NUREG-1801, Rev. 2 (Sections XI.E1, XI.E2, and XI.E3, respectively). To address NRC RIS 2003-09, this AMP includes EQ cables.

In-scope low and medium voltage cables and connections (non-EQ and EQ) in accessible areas exposed to adverse localized environments caused by heat, radiation, or moisture are inspected on a periodic basis. Visual inspections for cable and connector jacket surface anomalies such as embrittlement, discoloration, cracking, and surface containment are performed at least once every 10 years. (Table 16-2, Items 258 thru 261, and 266)

This AMP utilizes the Preventative Maintenance AMP (16.2.23) and Surveillance Testing AMP (Section 16.2.30) to facilitate cable tests and inspections.

16.2.10 CAST AUSTENITIC STAINLESS STEEL (CASS) AGING MANAGEMENT PROGRAM

This is a new AMP developed to manage the effects of thermal embrittlement of cast austenitic stainless steel (CASS) components in the RCS, Safety Injection System (SI), and Reactor Vessel Internals (RVI) with design temperatures >250°C (482°F) by identifying those components that may be susceptible to this ARDM. This AMP was developed to: (1) screen components; (2) review operating experience; (3) utilize either volumetric examination or EVT-1 visual examination; and (4) follow industry programs to evaluate thermal embrittlement and adjust the program accordingly.

NOTE: The CASS RVI components included within scope of this program are inspected in accordance with the RVI AMP (Section 16.2.26).

Susceptibility of individual components to thermal embrittlement was determined based on the delta ferrite content of the component, the casting method (static or centrifugal) and the molybdenum content. Delta ferrite content was determined using Hull's equation. For components that failed the screening and were deemed susceptible to thermal aging embrittlement, the preferred alternative is either a volumetric examination or an EVT-1 visual examination. A second alternative is to replace the component. The second alternative will be used if a component cannot be qualified for the license renewal term by either a volumetric examination or an EVT-1 examination or, if it is more cost effective to replace rather than perform either a volumetric examination or an EVT-1 examination. Replacement of the component will make the ARDM non-plausible for the respective component. The corrective actions taken as part of this program will ensure that the CASS components remain capable of performing their pressure boundary function under all applicable CLB conditions.

This AMP is implemented through the CASS Program Plan and is consistent with the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) program guidance in NUREG-1801, Rev. 2, (Section XI.M12).

16.2.11 CAULKING AND SEALANTS INSPECTION AGING MANAGEMENT PROGRAM

This is a new AMP initiated to provide requirements and guidance for the identification, inspection frequencies and acceptance criteria for non-fire barrier caulking and sealants used in the applicable in-scope plant structures (Turbine Building, Intake Structure, Auxiliary Building, and Safety Related/SBO Diesel Buildings) to ensure that their condition is maintained at a level that ensures that they will perform their intended functions. This new AMP consists of baseline inspections, completed prior to entering the PEO, to determine the material condition of the caulking and sealants along with periodic inspection activities during the PEO.

This AMP utilizes the PM AMP (Section 16.2.23) and the Surveillance Testing AMP (Section 16.2.30) to facilitate the periodic inspections.

16.2.12 CHEMISTRY (WATER) AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Plant Chemistry). The Plant Chemistry Program serves to minimize impurity ingress to plant systems; reduce corrosion product generation, transport, and deposition; reduce collective radiation exposure through chemistry; improve integrity and availability of plant systems; and extend component and

plant life. The Plant Chemistry Program is implemented through a series of procedures that provide for monitoring, maintaining, and/or controlling fluid chemistry in various plant systems.

Certain activities directed by these procedures are credited with mitigation of plausible aging effects by performing periodic measurement and evaluation of water chemistry parameters in process and supporting fluid systems. For certain fluid systems, activities include treatment with additives to aid in the prevention and control of corrosion mechanisms. Other portions of these procedures are credited with discovery of certain aging effects through measurement and evaluation of additional chemistry parameters in supporting systems. When the value of a measured parameter approaches or goes beyond predetermined warning limits, appropriate corrective actions are initiated as prescribed by the applicable procedure.

This AMP is credited for chemistry control, and thus mitigation aging management, in the following systems:

Auxiliary Feedwater	Main Steam
CVCS	Nitrogen and Hydrogen
Component Cooling	Nuclear Steam Supply Sampling
Containment Heating & Ventilation	RCS
Containment Spray	Safety Injection
Service Water	Emergency Diesel Generator
Spent Fuel Pool Cooling	Feedwater
Steam Generator Blowdown	Chemical Addition

16.2.13 COMPREHENSIVE REACTOR VESSEL SURVEILLANCE AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Comprehensive Reactor Vessel Surveillance Program). The Comprehensive Reactor Vessel Surveillance Program (CRVSP) implements the requirements of 10 CFR Part 50, Appendix H and provides the necessary data to monitor the embrittlement of the reactor vessels.

NOTE: A detailed discussion of the CRVSP is presented in Section 4.1.5.2.

Calvert Cliffs has five surveillance capsules for each unit to provide sufficient RPV material property changes and fluence information as suggested in American Society for Testing and Materials (ASTM) E185-82 to meet the requirements of 10 CFR Part 50, Appendix H, through the original license period. Each CCNPP unit also has one standby surveillance capsule to meet future needs (e.g., life extension, radical fuel management changes) as required. The regulations already require embrittlement and loss of upper shelf energy projections be updated to account for any significant changes in the projected values of RT_{PTS} or change in the expiration date for operation of the facility.

The CRVSP also provides for evaluation and incorporation of other research results, such as applicable coupon surveillance data obtained from other power plants.

CCNPP will continue to make periodic adjustments of neutron embrittlement and loss of upper shelf energy predictions, as needed, to account for any new information on the RPV beltline materials.

In addition, CCNPP addressed the following items in the CRVSP:

1. The capsule withdrawal schedule was revised to provide data at neutron fluences equal to or greater than the projected peak neutron fluence at the end of the PEO.
2. One capsule containing dosimeters will be removed during the final 5 years of the PEO.

16.2.14 CONTAINMENT LOCAL LEAKAGE RATE TESTING AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Containment Local Leak Rate Testing). Containment Local Leak Rate Testing (LLRT) is performed as part of an overall CCNPP Containment Leakage Rate Testing Program.

NOTE: A discussion of the Containment Leakage Rate Testing Program is presented in Section 5.5.2.1.

The CCNPP Containment Leakage Rate Testing Program implements the leakage testing of the containment, as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J 'Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors- Option B' and the CCNPP Technical Specifications. The Containment Leakage Rate Testing Program requires a quantitative assessment of the containment integrity on a periodic basis.

This AMP is credited for discovery and management of radiation and thermal related degradation for non-metallic portions of both EQ and Non-EQ electrical penetrations. Any significant degradation of the penetration elastomer seals would be detected and repaired as required to maintain integrity.

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) and Surveillance Testing AMP (Section 16.2.30) to facilitate the credited surveillance testing activities.

16.2.15 DESIGN CHANGE AND MODIFICATION IMPLEMENTATION AGING MANAGEMENT PROGRAM

This AMP credits an existing plant process (i.e., Design Change/ Modification Process). Implementation of the following modifications eliminated the plausible ARDMs and aging management activities that would have been required for the applicable plant components during the PEO:

1. Control Room air handling unit supports were replaced. The new supports eliminate the use of elastomers, thereby eliminating elastomer degradation for the Control Room HVAC air handler supports as a plausible ARDM in the Component Supports commodity group.
2. Certain Chemical and Volume Control System (CVCS) heat trace, installed on the borated water piping, was replaced with a different type of heat trace. This modification removed the corrosive adhesive associated with the original heat tracing, thus eliminating SCC of the external surfaces of the applicable stainless steel components as a plausible ARDM in the CVCS.

This AMP has been completed and does not continue into either Unit's PEO.

16.2.16 DIESEL FUEL OIL (TANKS AND CHEMISTRY) AGING MANAGEMENT PROGRAM

This AMP credits existing and new plant programs and activities. This AMP manages the effects of plausible ARDMs of the Diesel Fuel Oil (DFO) System components and various DFO storage tanks.

This AMP utilizes the PM AMP (see Section 16.2.23) and the Surveillance Testing AMP (Section 16.2.30) to facilitate the various inspections and tests.

16.2.17 ENVIRONMENTAL QUALIFICATION (EQ) AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., EQ Program). The EQ Program manages the environmental qualification of electrical equipment important to safety as required by 10 CFR 50.49. The EQ Program is credited for managing the effects of plausible ARDMs of organic subcomponents for EQ components within the scope of license renewal. The EQ Program provides for management of aging effects by performing periodic preventive maintenance and replacement activities.

NOTE: A discussion of the EQ Program is presented in Section 7.12.

The EQ Program will continue to be administered in accordance with the requirements of 10 CFR 50.49, the Division of Operating Reactors Guidelines as transmitted by NRC Bulletin 79-01B, and NUREG-0588. The EQ Program includes requirements for determining the components in-scope per 10 CFR 50.49 and options for management of the plausible thermal and radiative aging effects associated with these components. The EQ Program contains the provisions necessary to ensure that EQ components will remain qualified to perform their required 10 CFR 50.49 function(s) under applicable design bases conditions should a design basis event occur at the end of extended plant life. The EQ Program is consistent with the Environmental Qualification (EQ) of Electric Components AMP guidance provided in NUREG-1801, Rev. 2; Section X.E1.

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) to facilitate the various inspections, refurbishments and replacements.

16.2.18 FLOW ACCELERATED CORROSION (FAC) AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Flow-Accelerated Corrosion (FAC) Program). The FAC Program was established to provide early detection and prevention of pipe wall thinning caused by accelerated corrosion, cavitation or erosion that could lead to ruptures in high energy piping.

All piping within the scope of the FAC program is evaluated and categorized to determine inspection points where thickness measurements will be taken. Inspection points are determined through evaluations of site-specific data, failures at other plants and modeling of piping systems by CHECWORKS software developed by Electric Power Research Institute (EPRI). An ultrasonic non-destructive examination is used to determine the wall thickness at a number of grid locations for each inspection point. This data is used with a predictive model to determine additional inspection points, to adjust an inspection point’s priority and to extrapolate the time until the minimum wall thickness will be reached on the inspected component. This information is maintained in the FAC Program database and further inspections or replacements are scheduled to ensure no components reach their minimum thickness value while in-service.

The FAC Program is credited for aging management in the following systems:

Extraction Steam	Main Feedwater
Main Steam	Steam Generator Blowdown

16.2.19 FATIGUE MONITORING AGING MANAGEMENT PROGRAM

This AMP credits an existing program (i.e., Fatigue Monitoring). The Fatigue Monitoring Program (FMP) records and tracks the number of thermal and pressure test transients.

Cycle counting is performed as part of this program. The data for thermal transients is collected, recorded and analyzed using Safety Related software (FatiguePro). This software is used to analyze data that represents real transients and to predict the number of transients for 40 and 60 years of plant operation based on historical records.

The FMP tracks low-cycle fatigue usage using three methods: 1) cycle counting, 2) cycle-based fatigue analysis, and 3) stress-based fatigue analysis. In accordance with ASME Code Section III, the fatigue life of a component is based on a calculated cumulative usage factor of less than or equal to one.

Twenty-seven sentinel locations have been selected for monitoring for low-cycle fatigue usage. These locations represent the bounding locations for critical thermal and pressure transients and operating cycles.

The FMP assesses the effect of the environment using statistical correlations developed in NUREG/CR-5704 for stainless steels, NUREG/CR-6583 for carbon and low alloy steels and NUREG/CR-6909 for nickel-based alloys. The modified FMP uses these statistical correlations to calculate an effective environmental factor to account for the reduction in fatigue life due to the reactor water environment. This factor was applied to fatigue loads where the specified threshold criteria for strain rate and temperature have been exceeded.

The FMP provides fatigue monitoring for the following systems:

Chemical and Volume Control	Reactor Coolant
Control Element Drive Mechanism	Reactor Pressure Vessel
Feedwater	Reactor Vessel Internals
Nuclear Steam Supply Sampling	Safety Injection

16.2.20 FIRE BARRIER PENETRATION SEAL INSPECTION AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Surveillance Testing), in addition to Operations Department's Performance Evaluations (PEs) for managing the plausible ARDMs associated with penetration fire barriers located in the Turbine Building, Intake Structure, Auxiliary Building and Safety Related/SBO Diesel Buildings. These existing Surveillances and PEs provide instructions for visual inspection of fire barrier penetration seals in fire areas boundaries that protect safe shutdown and other areas. The scope, of these Surveillance Tests and PEs, is to visually inspect the following types of fire barrier and penetration seals:

- Electrical Conduit and cable tray penetration seals
- HVAC duct penetration seals (ducts without dampers)
- Mechanical pipe penetration seals
- Expansion joint penetration seals

Under these existing Surveillance Tests and PEs the penetration seals are inspected for damage, cracking, voids and proper installation, providing for separate failure criteria and repair criteria.

NOTE: A discussion of the Fire Protection Program is presented in Section 9.9.

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) and Surveillance Testing AMP (Section 16.2.30) to facilitate the associated Surveillance Test Procedure (STPs) and PE periodic inspections.

16.2.21 FIRE PROTECTION AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., Fire Protection) and procedures (i.e., Conduct of Operations).

NOTE: A discussion of the Fire Protection (FP) Program is presented in Section 9.9.

The FP Program is credited for managing the aging of specific components within the scope of license renewal. Various performance and/or condition monitoring activities implemented by this program provide for discovery of aging effects on pressure-retaining components (e.g., piping, spool-pieces, and valves), by performing periodic inspections and functional testing.

The FP Program is the integrated effort involving components, procedures and personnel used to carry out activities of the FP Program and fire prevention. The FP Program contains maintenance, periodic functional testing, and inspection criteria to provide reasonable assurance that various Non-Safety Related (NSR) systems are capable of performing their FP intended functions. Any abnormal condition would be detected and investigated to ensure that it does not have the ability to impact safety or adversely affect operation of the system. Any such condition would be repaired prior to impacting the passive FP intended function of the system. Fire protection equipment and systems are inspected and tested upon initial installation and periodically thereafter.

The inspection and testing is conducted following the guidance of applicable National FP Association Codes and Standards as well as recommendations and requirements of the insurance carrier and the NRC. Plant procedures mandate test frequencies and testing process. Applicability, compensatory actions, testing requirements and testing frequencies for those FP systems that protect safe shutdown and Safety Related (SR) equipment are contained in the CCNPP Technical Specifications. Plant procedures also identify compensatory actions to be taken when Appendix R, safe shutdown actions are identified as degraded/non-conforming.

The FP AMP credits ‘conduct of operations’ procedures for managing plausible aging mechanisms on NSR portions of various systems required for safe shutdown (Appendix R). The demands on most NSR systems and components during normal operation are the same as, or greater than, the demands placed on them during mitigation of fires. Therefore, satisfactory performance of periodic functional tests can be used to demonstrate that aging is adequately managed for the passive FP functions of NSR components. A system that is in continuous operation during normal operation can be characterized as undergoing a continuous FP function test if the system parameters (pressure, temperature, flow, etc.) encountered during performance of FP intended functions are bounded by the normal operating parameters of the system. The performance and conditioning activities conducted in accordance with Conduct of Operations ensure detection of abnormal conditions. The Conduct of Operations requires that operators be accountable for their immediate areas of responsibility.

This includes performing general inspections and checking conditions of areas and equipment. Operators assess degraded equipment conditions to ensure personnel and affected equipment safety while completing corrective actions. Where the above type of demonstration is successful, performance and condition monitoring activities during normal plant operation are credited for identifying the effects of system aging. Specific aging management programs are not necessary and no further evaluation is required.

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) and Surveillance Test AMP (Section 16.2.30) to facilitate the associated STPs.

16.2.22 LOAD HANDLING AND FUEL HANDLING EQUIPMENT AGING MANAGEMENT PROGRAM

This AMP credits existing CCNPP load handling procedures in addition to Operations Section Performance Evaluations (PEs) and Operating Instructions (OIs). These load handling procedures, PEs and OIs establish the requirements and assigns responsibilities for activities involving load handling. These procedures, etc. are credited for managing the aging of certain devices in the Fuel Handling Equipment (FHE) and Heavy Load Handling Cranes (HLHC) commodity group.

NOTE: A Discussion of 'Control of Heavy Loads' is presented in Section 5.7.

Certain steps of the load handling procedures, etc. provide for discovery of plausible ARDMs by performing periodic visual inspection and non-destructive examination. These activities provide reasonable assurance that aging of FHE and HLHC will be managed during the PEO.

The FHE and HLHC components addressed by this AMP are as follows:

Auxiliary Building Cask Handling Crane	Containment Purge Exhaust Monorail Hoists
Containment Building Jib Cranes	Fuel Upending Machines
Intake Structure Semi-Gantry Crane	New Fuel Elevator
Polar Cranes	Reactor Refueling Machines
Reactor Vessel Head Lift Rigs	Reactor Vessel Head Cooling Shroud/ Structural Support Members
Refueling Machine Auxiliary Hoists	Spent Fuel Inspection Elevator
Spent Fuel Handling Machine	Transfer Machine Jib Crane

This AMP utilizes the Preventive Maintenance AMP (16.2.23) to facilitate FHE and HLHC component inspections and examinations.

16.2.23 PREVENTIVE MAINTENANCE (PM) AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (i.e., PM Program). The CCNPP PM Program maintains plant structures, systems and components (SSCs) in a reliable condition for normal operation and emergency use, to minimize equipment failure and extend equipment and plant life. The PM Program covers PM activities for all nuclear power plant SSCs within the scope of license renewal. The PM tasks maintain plant SSCs through regular maintenance and inspection of SSCs for signs of damage and/or degradation.

Preventive maintenance encompasses a variety of maintenance actions taken to extend equipment life and maintain equipment within design operating conditions. These include periodic maintenance actions (accomplished on a routine basis) and certain other activities that may be initiated in response to predictive or periodic maintenance results, vendor recommendations or experience.

PM tasks are scheduled and implemented in accordance with PM Program procedures. Some PM tasks have been modified to inspect for effects of specific plausible ADRMs.

PM activities are automatically scheduled and implemented in accordance with PM Program procedures.

The following SSCs and commodity groups specifically credit the PM AMP for aging management:

Auxiliary Feedwater System	Reactor Pressure Vessel/Control Element Drive Mechanism
Component Cooling and Service Water Systems	Main Steam System
Electrical Commodities (Groups 1-7)	Spent Fuel Pool Cooling System
Heating, Ventilation, and Air Conditioning Systems	Piping Encapsulations
Feedwater System	Emergency Diesel Generators
Salt Water Cooling System	Instrument Air and Salt Water Air Systems
Intake Structure	ASME Section XI In-Service Inspection; Subsections IWE and IWL
Boric Acid Corrosion Inspection	Buried Piping Inspection
Cables	Caulking and Sealants Inspection
Containment Leakage Rate Testing	Diesel Fuel Oil (Tanks and Chemistry)
Environment Qualification	Fire Barrier Penetration Seal Inspection
Fire Protection	Load Handling and Fuel Handling Equipment
Protective Coating	Reactor Coolant
Spent Fuel Pool (Liner/ Neutron Absorbing Material)	Structure and System Walkdown

16.2.24 PROTECTIVE COATINGS AGING MANAGEMENT PROGRAM

This AMP credits an existing plant procedure. The CCNPP protective coatings procedure was established to control Service Level 1 protective coatings activities, performed inside Containment, to ensure they comply with NRC Regulatory Guide 1.54 and ANSI N101.4-1972. Service Level I coatings are those where failure of the coating could adversely affect the operation of mechanical fluid systems required for post-accident operation. The procedure provides for discovery of corrosion or of conditions that would allow corrosion to occur, such as degraded coatings by performing periodic visual inspections on all readily accessible containment surfaces.

The procedure requires that the responsible engineer perform a walkdown of the inside of containment to verify the condition of Service Level 1 coatings. The responsible engineer develops a list of all areas inside containment exhibiting deterioration. Repair areas are evaluated to ensure timely corrective action is taken.

This AMP provides reasonable assurance that aging of steel components inside containment will be managed during the PEO. This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) to facilitate Containment protective coating inspections and repairs.

16.2.25 REACTOR COOLANT AGING MANAGEMENT PROGRAM

This AMP is credits existing plant activities. This AMP specifically manages aging of the following components:

1. Reactor Vessel Head O-ring leak-off line utilizing the PM AMP (Section 16.2.23).
2. Reactor Coolant Pump (RCP) seal water heat exchanger tubes utilizing Operations use of Operating Instructions (OIs).

16.2.26 REACTOR VESSEL INTERNALS (RVI) AGING MANAGEMENT PROGRAM

This is a new AMP developed and implemented to manage the aging effects applicable to RVI components following the recommendations of GALL; Revision 2; Section XI.M16A and Materials and Reliability Program (MRP)-227-A. The RVI AMP provides for inspection, acceptance criteria and corrective actions. Under the guidance of NEI 03-08, Calvert Cliffs incorporated the recommendations for additional inspections and evaluations provide by industry guidelines. These additional industry evaluations and recommendations are documents in MRP-227-A.

Calvert Cliffs will continue to participate in industry programs through the MRP.

16.2.27 SPENT FUEL POOL (SFP) AGING MANAGEMENT PROGRAM

This AMP credits existing plant activities. This AMP specifically manages aging of the following components:

1. SFP Liner utilizing the PM AMP (Section 16.2.23) and Operations use of Operating Instructions (OI).
2. SFP (Unit 1) Storage Racks (Neutron Absorbing Material) utilizing the PM Program (Section 16.2.23) and an Engineering Test Procedure (ETP)

Regarding the SFP Liner, the applicable OI and related PM provide for the determination of SFP leakage. This OI provides detailed instructions for leakage monitoring of the SFP cooling System. During the performance of this OI, the ‘telltale’ valves are opened, drained and are monitored for 24 hours with catch devices installed at the outlet of each ‘telltale’ valve.

Regarding the SFP (Unit 1) Storage Racks Neutron Absorbing Material, the ETP and related PMs were initially developed on the basis of vendor recommendations for detecting degradation of neutron-absorbing materials. The ETP is designed to permit samples of the materials used in the Unit 1 Spent Fuel Pool (SFP) storage racks to be periodically removed from the SFP for examination. Through specific positioning of the designated sample packets, both accelerated and long-term exposure to gamma radiation and borated water is provided. Sufficient samples are available so that the principle properties (i.e., sample weight for the Carborundum material) can be determined as a function of exposure on a regularly scheduled basis. Visual condition is assessed on a graded scale and the results of physical property analysis are compared to historical results. The ETP was modified to refine the process for scheduling sample packet removal from the SFP.

16.2.28 STEAM GENERATOR (SG) AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (Steam Generator (SG) Program) that was applicable to the aging management of the Steam Generators that were installed in the plant at the time the operating licenses were renewed. Those SGs were replaced in 2002 and 2003 for CCNPP Units 1 and 2, respectively. The materials in the new SGs were upgraded to minimize aging degradation and, consequently, a new AMP was created for the new SGs.

The SG tubes are inspected in accordance with the SG Program. The SG Program prescribes the sample size for tube inspection, inspection process, evaluation and determination of tube status. The SG Program is credited to manage the aging effects of denting, intergranular attack (IGA) and SCC of the SG tubes with eddy current examinations. Tube plugging or sleeving are the prescribed methods for addressing tubes that are identified as susceptible to failure before the next inspection. The SG

Program provides for detection of leakage between Primary and Secondary sides of the SG which potentially could be caused by denting, wear, pitting SCC/IGSCC or IGA.

The SG Program (visual inspections) is also credited to manage the effects of wear, erosion, corrosion of the SG vessel, manway covers, handhole covers and tube supports.

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) and Surveillance Testing AMP (Section 16.2.30) to facilitate the periodic inspections.

16.2.29 STRUCTURE AND SYSTEM WALKDOWNS AGING MANAGEMENT PROGRAM

This AMP credits an existing plant procedure (Structure and System Walkdown). Structure and System Walkdown activities provide for discovery and mitigation of plausible aging effects through periodic walkdowns. These periodic walkdowns, performed by responsible personnel, assess the condition of applicable structures, systems and components to identify abnormal or degraded conditions. The walkdown procedure provides guidance for the identification of specific types of plausible degradation.

This AMP provides reasonable assurance that aging of the components in the following structures, systems and commodity groups will be managed during both units' PEOs:

Structures:

- Containment Structure
- Auxiliary Building
- Turbine Building Structure
- Intake Structure
- Emergency Diesel Generator Building Structures
- Miscellaneous Tank and Valve Enclosures

Systems:

- AFW
- Auxiliary Building Heating and Ventilation
- Component Cooling
- Compressed Air
- Condensate
- Condensate Storage
- Containment Heating and Ventilation Spent Fuel Pool
- Control Room HVAC
- Demineralized Water
- Diesel Fuel Oil
- Diesel Generator Building Heating and Ventilation
- Fire Protection
- Liquid Waste
- Main Steam
- Plant Drains
- Plant Healing
- Radiation Monitoring
- Saltwater
- Sampling System
- Service Water
- Spent Fuel Pool
- Refueling Water Tank
- Well and Pretreated Water
- Safety Injection System

Commodity Groups:

- Component Supports
- Instrument Lines

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) to facilitate the various System/Structure Walkdown /inspection activities.

16.2.30 SURVEILLANCE TESTING AGING MANAGEMENT PROGRAM

This AMP credits an existing plant program (Surveillance Testing). Surveillance Testing was established to implement certain surveillance requirements specified in the CCNPP Technical Specifications. A number of the Surveillance Test Procedures (STPs) are credited for managing the aging of certain SSCs within the scope of License Renewal.

The following SSCs and commodity groups specifically credit the Surveillance Testing AMP for aging management:

- Containment (Tendons, Liner Plates)
- Component Cooling and Service Water Systems
- Battery Terminals (Electrical Commodities-Group 1)
- Fuse Holders (Electrical Commodities-Group 7)

In addition, the following AMPs utilize the Surveillance Testing AMP to facilitate various inspections, refurbishment, replacements and tests for aging management:

- ASME Section XI In-Service Inspection (ISI); Subsections IWE and IWL
- Cables
- Caulking and Sealants Inspection
- Containment Leakage Rate Testing
- Diesel Fuel Oil (Tanks and Chemistry)
- Fire Barrier Penetration Seal Inspection
- Fire Protection
- Steam Generator

This AMP utilizes the Preventive Maintenance AMP (Section 16.2.23) to facilitate various surveillance test activities.

16.3 <u>EVALUATION OF TIME-LIMITED AGING ANALYSES</u>

As part of License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of the time-limited aging analyses (TLAAs) for the period of extended operation be provided. In addition, 10 CFR 54.21(d) requires that a summary description of the TLAAs be included in the License Renewal UFSAR Supplement. The following TLAAs have been identified and evaluated to meet these requirements.

16.3.1 CONTAINMENT TENDON PRESTRESS LOSS

There is a TLAA that provides acceptable Containment Tendon stress relaxation through the period of extended operation. Tendon prestress losses are determined by measuring tendon lift-off force. Technical Specification Surveillance Requirement 3.6.1.2 and Technical Specification 5.5.6 establish the surveillance schedule for measuring lift-off forces as discussed in Section 5.5.2.2 of this UFSAR. The measurement is performed in accordance with a Surveillance Test Procedure which provides the normalized lift-off forces required to be achieved during the surveillance test as a function of plant service after initial prestressing. These curves, which were originally projected to cover 40 years, were recalculated to project operation through the PEO.

This TLAA required reanalysis to project applicability through the PEO, and is therefore managed in accordance with 10 CFR 54.21(c)(1)(ii). This TLAA is also managed by an

existing AMP (Table 16-2, Item No. 19) and is therefore also managed in accordance with 10 CFR 54.21(c)(1)(iii).

16.3.2 POISON SHEETS IN SPENT FUEL POOL

The criticality analysis for the Unit 1 SFP credits the existence of Carborundum (i.e., neutron absorbing) sheets located between spent fuel assemblies. These analysis assume the Carborundum material has a minimum concentration of Boron 10. This assumption accounts for the potential loss of boron carbide and is based on experiments showing that the Carborundum sheets experience a loss of boron carbide due to aging.

The Spent Fuel pool criticality analyses are therefore a Time Limited Aging Analysis (TLAA) as defined by 10 CFR 54.3.

This TLAA required reanalysis to project applicability through the PEO, and is therefore managed in accordance with 10 CFR 54.21(c)(1)(ii).

16.3.3 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION PROGRAM

The TLAA aspect of EQ encompasses all long-lived equipment in the scope of the EQ Program, whether active or passive. At CCNPP, each EQ file for a group of long-lived components documents a TLAA. Environmentally-qualified equipment is replaced with qualified new or refurbished equipment prior to the end of its qualified life. Preventive maintenance is scheduled to initiate and execute these replacements. Qualified life re-evaluations are an ongoing activity and consider actual normal operating conditions as compared to design maximums (e.g., actual ambient temperatures are below the maximum design temperature that was used as the basis for the current qualified life). Qualified lives are adjusted up and down accordingly. Qualified life re-evaluations are performed under the current EQ Program and will continue to be performed during the PEO.

The EQ TLAAs were revised to project applicability through the PEO, and are therefore managed in accordance with 10 CFR 54.21(c)(1)(ii). The EQ TLAAs are also managed by an existing AMP (Table 16-2, Item No. 246) and are therefore also managed in accordance with 10 CFR 54.21(c)(1)(iii).

16.3.4 REACTOR PRESSURE VESSEL TOUGHNESS REQUIREMENTS

There are four TLAAs that address loss of toughness due to irradiation embrittlement.

- Plant Heatup/Cooldown (Pressure/Temperature (PT) Limit) Curves (Tech Spec Figures 3.4.3-1 and 3.4.3-2).
- Low Temperature Overpressure Protection (LTOP) PORV setpoints (Tech Spec Figure 3.4.12-1 is based on the PT limits).
- Pressurized Thermal Shock (PTS) requirements (10 CFR 50.61).
- The Reduction of Upper Shelf Energy (10 CFR 50, Appendix G).

These TLAAs use predictions of the cumulative damage to the reactor vessel from irradiation embrittlement and were originally based on the 40-year expected service life of the plant. The TLAAs have been updated to consider the projected neutron fluence for the 20-year period of extended operation.

These TLAAs are projected to the end of the PEO, and are therefore managed in accordance with 10 CFR 54.21(c)(1)(ii). These TLAAs are also managed by the Comprehensive Reactor Vessel Surveillance Program (CRVSP) (Table 16.2, Item No. 75) such that the effects of aging due to irradiation embrittlement on the reactor vessel are

adequately managed through the PEO and are therefore also managed in accordance with 10 CFR 54.21(c)(1)(iii).

16.3.5 REACTOR VESSEL INTERNALS

There is one TLAA that is affected by fatigue.

- Core Support Plate and Core Support Barrel Lower Flange Weld

This TLAA uses predictions of the cumulative fatigue damage to the Core Support Plate and Core Support Barrel Lower Flange Weld. This TLAA considered the projected 20-year period of extended operation. Results show that the fatigue cumulative usage factor is less than the ASME acceptance criterion of 1.00 for 60 years of operation when corrected for environmental effects.

Because this is a new analysis that remains valid through the period of extended operation instead of the reevaluation of an existing analysis, the TLAA is classified as being managed in accordance with 10 CFR 54.21(c)(1)(i).

16.3.6 CONTAINMENT LINER FATIGUE

The ASME codes require that the containment liner material be prevented from experiencing significant distortion due to the thermal load and that the stresses be considered from a fatigue standpoint. The following fatigue loads were considered in the design of the liner plate

- The annual outdoor temperature variation, assumed to be 40 cycles during the plants 40-year life;
- The interior temperature variations during the startup and shutdown of the RCS, assumed to be 500 cycles; and
- Thermal cycling due to a loss-of-coolant accident, assumed to occur once during plant life.

The design of the liner plate and penetration sleeves included consideration of thermal stress and fatigue for which there was an assumed number and severity of thermal cycles. Since this assumption was partly based on a 40-year operating life, the fatigue analyses had to be reviewed to assure they remain valid during the PEO. The review of this TLAA determined it was valid to the end of the PEO. The TLAA is, therefore, being managed in accordance with 10 CFR 54.21(c)(1)(i).

16.3.7 MAIN STEAM PIPING FATIGUE

The Main Steam supply lines to the AFW pump turbines provide the system pressure boundary function and are subject to thermal loadings. According to the UFSAR Chapter 10A discussion, 21,999 rapid full temperature cycles have been considered. However, even if the number of assumed cycles were limited to 7000 equivalent full temperature cycles, which is much more limiting, this piping would have to be cycled approximately once every 3 days over an extended plant life of 60 years. Under current plant operating practices, the system is operated only occasionally during plant heatups and cooldowns, during plant transients, and for periodic (monthly) testing. Plant heatups and cooldowns are limited to 500 each, and reactor trips are limited to 400 over plant life. Monthly testing over 60 years would contribute another 720 cycles. These actual and potential cycles combined equal slightly more than 2000 cycles for the AFW steam supply. It is, therefore, unlikely that the 7000 assumed cycles will be approached during the PEO. Thus, the existing analysis is considered to remain valid for the PEO, and there is reasonable assurance that the intended function will be maintained. This TLAA is, therefore, managed in accordance with 10 CFR 54.21(c)(1)(i).

16.3.8 NUCLEAR STEAM SUPPLY FATIGUE

Components in the NSSS are subject to a wide variety of varying mechanical and thermal loads that contribute to fatigue accumulation. The RCS components were designed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, and the American National Standards Institute (ANSI) Standard USAS B 31.7, Nuclear Power Piping Code. These codes require the design analysis for Class I components to address fatigue and establish limits such that initiation of fatigue cracks is precluded. Portions of UFSAR Section 4.1 and the certified design specification identify the different design cyclic transients used in the fatigue analysis required by code for various major components of the RCS including the reactor vessel, RCS piping, steam generators, pressurizer, pressurizer auxiliary spray piping, and pressurizer surge line.

This TLAA use predictions of the cumulative damage due to low cycle fatigue and were originally based on the 40 year expected service life of the plant. This TLAA has been updated to consider the projected 20 year period of extended operation and increase in fatigue damage due to environmental effects and is, therefore, managed in accordance with 10 CFR 54.21(c)(1)(ii).

This TLAA, being part of the Fatigue Monitoring Program is also being managed through the PEO by an existing AMP (Table 16-2; Items 56, 77, 109, ,203 and 205) and are therefore also managed in accordance with 10 CFR 54.21(c)(1)(iii).

16.3.9 CLASS 2 AND 3 PIPING COMPONENTS (OTHER THAN MAIN STEAM PIPING) FATIGUE

Systems that contain B31.7 Class II or III piping (other than main steam piping) have a stress limit based on 7000 cycles. These systems would have to be cycled approximately once every 3 days over an extended plant life of 60 years to reach 7000 cycles. Fatigue is not plausible for systems where the ΔT between system shutdown temperature and maximum operating temperature if the ΔT was small (equal to or less than 50°F). For systems where the ΔT is greater than 50°F, the number of thermal cycles for 60 years was conservatively estimated using plant operating history. Plant heatups and cooldowns are limited to 500 each, and reactor trips are limited to 400 over plant life. It is, therefore, unlikely that the 7000 cycles will be approached during the period of extended operation. This TLAA is, therefore, being managed in accordance with 10 CFR 54.21(c)(1)(i).

16.4 <u>10 CFR 54.37(B) UPDATE</u>

Newly identified Systems, Structures or Components that would have been subject to an aging management review are included in Table 16-2.

Newly identified Systems, Structures or Components that would have been subject to an evaluation of time-limited aging analyses are included in Table 16.3.

16.5 <u>REFERENCES</u>

1. NUREG-1705, Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2, dated December, 1999
2. AMBD-0001; Auxiliary Feedwater System
3. AMBD-0002; Reactor Coolant
4. AMBD-0003; Reactor Pressure Vessel & Control Element Drive Mechanisms
5. AMBD-0004; Component Cooling System/Service Water System
6. AMBD-0005; Main Steam

7. AMBD-0006; Diesel Fuel Tanks and Chemistry
8. AMBD-0007; Boric Acid Corrosion Inspection (BACI) Program
9. AMBD-0008; Fire Barrier Penetration Inspection
10. AMBD-0009; Age-Related Degradation Inspection (ARDI) Program
11. AMBD-0010; Water Chemistry Program
12. AMBD-0013; Flow Accelerated Corrosion (FAC) Program
13. AMBD-0014; Steam Generator
14. AMBD-0015; Electrical Commodities (Group 1); Battery Terminals/Chargers and Inverters
15. AMBD-0016; Electrical Commodities (Group 2)
16. AMBD-0017 Electrical Commodities (Group 3); Bus Cabinets
17. AMBD-0018; Electrical Commodities (Group 4); MCC Panels
18. AMBD-0019; Electrical Commodities (Group 5); Local Control Station Panels
19. AMBD-0020; Electrical Commodities (Group 6); Misc. Panels
20. AMBD-0021; Spent Fuel Pool Cooling
21. AMBD-0022; Containment Leakage Rate Testing
22. AMBD-0023; Comprehensive Reactor Vessel Surveillance Program
23. AMBD-0025; Fire Protection
24. AMBD-0026; Caulking and Sealants Inspection
25. AMBD-0027; Cast Austenitic Stainless Steel (CASS)
26. AMBD-0029; HVAC Systems Preventive Maintenance
27. AMBD-0030; Fuel Handling Equipment/Heavy Load Handling Cranes
28. AMBD-0033; Additional Baseline Walkdowns
29. AMBD-0034; Piping Encapsulations
30. AMBD-0035; Buried Pipe Inspection Program
31. AMBD-0036; Cable Aging Management Program
32. AMBD-0038; Check Valve Corrosion Inspection (Main Feedwater)
33. AMBD-0039; 10 CFR 50.49 Program
34. AMBD-0040; ASME Section XI (ISI) Components
35. AMBD-0041; Alloy 600 Program
36. AMBD-0042; Emergency Diesel Generator
37. AMBD-0043; Reactor Vessel Internals Program
38. AMBD-0045; Salt Water Cooling
39. AMBD-0049; Fatigue Monitoring Program
40. AMBD-0050; Air Quality Preventive Maintenance
41. AMBD-0051; Intake Structure

42. AMBD-0052; Structures and Systems Walkdowns
43. AMBD-0053; Containment Procedures
44. AMBD-0054; Auxiliary Building Procedures and Inspections
45. AMBD-0055; Electrical Commodities; Group 7- Fuse Holders

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
1.	3.1	Component Supports	Piping supports, cable raceway supports, HVAC ducting supports, equipment supports, frames and saddles (inside containment)/loss-of-coolant accident (LOCA) restraints, equipment anchorages (for Nuclear Steam Supply System (NSSS) hoods outside containment)	General Corrosion, Loading Due to Hydraulic Vibration or Water Hammer, Loading Due to Thermal Expansion, SCC of High Strength Bolts	Additional Baseline Walkdowns	See Section 16.2.1 for a general description.	Program was modified
2.	3.1	Component Supports	Piping supports, cable raceway supports, HVAC ducting supports, equipment supports, metal spring isolators and fixed bases (outside containment)/LOCA restraints, frames and saddles/LOCA restraints, frames and saddles/ring foundation for flat bottom vertical TKs, frames and saddles (inside containment)/LOCA restraints	General Corrosion, Wear, Erosion, Loss of Integrity at Bolted Connections, Welded Connections, Pinned Connections, Cracking (fatigue cracking and SCC of High Strength Bolts)	ASME Section XI, Subsection IWF	See Section 16.2.6 for a general description.	Existing Program
3.	Deleted						
4.	3.1	Component Supports	Control Room HVAC, Elastomer vibration isolators	Elastomer Hardening	Design Change and Modification Implementation	See Section 16.2.15 for a general description.	Program has been implemented
5.	3.1	Component Supports	CAC Fan Housing Spring Isolator Supports and Fixed Bases	General Corrosion	Preventive Maintenance (PM)	See Section 16.2.23 for a general description. For the CAC fan housing supports and fixed bases, the PM checklists, which open and inspect other components internal to the fan housing, have been modified to also inspect these spring isolator supports for signs of general corrosion.	Existing Program

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
6.	3.1	Component Supports	Piping (Snubber) supports	General Corrosion, Wear, Erosion, Loss of Integrity at Bolted, Pinned or Weld Connections, Cracking	ASME Section XI, Subsection IWF	See Section 16.2.6 for a general description.	Existing Program
7.	3.1	Component Supports	Piping supports, cable raceway supports, HVAC ducting supports, equipment supports, elastomer vibration isolators, metal spring isolators and fixed bases (outside containment)/ LOCA restraints, frames and saddles/ LOCA restraints, frames and saddles/ ring foundation for flat-bottom vertical TKs	General Corrosion, Loading Due to Hydraulic Vibration or Water Hammer, Loading Due to Thermal Expansion, Elastomer Hardening, Loading Due to Rotating or Reciprocating Equipment	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
8.	3.1A	For the following systems: (011) Service Water (SRW) Cooling (012) Saltwater (SW) Cooling (013) FP (015) Component Cooling (CC) (019) Compressed Air (023) Diesel Fuel Oil (DFO) (024) Emergency Diesel Generators (EDGs) (029) Plant Heating (036) Auxiliary Feedwater (AFW) (037)	Piping/supports beyond SR/non-safety-related (NSR) boundary providing structural support	Various	Structure and System Walkdowns	Aging management of the piping segments and supports beyond the SR/NSR boundary that provide structural support for the SR piping and boundary isolation valves is accomplished by the Structure and System Walkdowns Program described in Section 16.2.29. This program ensures that the intended structural support function of the NSR piping segments and supports beyond the SR/NSR boundary up to the first anchor point (or equivalent) is maintained.	Existing Program

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
		Demineralized Water/ Condensate Storage (038) Sampling System (NSSS) (041) Chemical and Volume Control (CVCS) (045) FWS (046) Extraction Steam (051) Plant Water (052) Safety Injection (SI) (053) Plant Drains (061) Containment Spray (064) Reactor Coolant (RCS) (067) Spent Fuel Pool Cooling (SFPC) (069) Waste Gas (071) Liquid Waste (074) Nitrogen and Hydrogen (077/79) Area and Process Radiation Monitoring (083) Main Steam					
9.	3.1A	For the following systems: (011) SRW Cooling (012) SW Cooling	Piping/supports beyond SR/NSR boundary providing structural support	Various	Various	Aging management of the piping segments and supports beyond the SR/NSR boundary that provide structural support for the SR piping and boundary isolation valves is accomplished by the same aging management programs credited to manage the SR portion of the piping systems for the identified systems, as described in various	Existing Program

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
		(013) FP (015) CC (019) Compressed Air (023) DFO (024) EDGs (029) Plant Heating (036) AFW (037) Demineralized Water/Condensate Storage (038) Sampling System (NSSS) (041) CVCS (045) FWS (046) Extraction Steam (051) Plant Water (052) SI (053) Plant Drains (061) Containment Spray (064) RCS (067) SFPC (069) Waste Gas (071) Liquid Waste (074) Nitrogen and Hydrogen (077/79) Area and Process Radiation Monitoring (083) Main Steam				items in this Table. These various Aging Management Programs ensure that the intended structural support function of the NSR piping segments and supports, beyond the SR/NSR boundary up to the first anchor point (or equivalent), is maintained.	
10.	3.2	Cranes, Reactor Vessel Cooling Shroud	Reactor vessel cooling shroud structural support members, cranes	Corrosion Due to Boric Acid	Boric Acid Corrosion Inspection (BACI)	See Section 16.2.7 for a general description.	Program was modified

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
11.	3.2	Fuel Handling, Cranes	Carbon steel fuel handling equipment and heavy load handling crane components, Carbon steel wire rope	General Corrosion, Fatigue, Wear, Mechanical Degradation	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description. Credited for discovery and management of fatigue, wear, and mechanical degradation of carbon steel wire rope and general corrosion of carbon steel components of fuel handling equipment, by performing visual inspections. PM repetitive tasks have been modified to explicitly present the inspection requirements.	Existing Program
12.	3.2	Fuel Handling	Carbon steel components of the Spent Fuel Handling Machine (SFHM) and the Reactor Refueling Machine (RRM)	General Corrosion	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description. Performance Evaluations provide for checks of the SFHM, RRM, and associated components prior to refueling campaigns (i.e., defuel/refuel or fuel shuffle). These procedures require performing a walkdown for foreign material and cleanliness, inspecting the SFHM and associated equipment for damaged, corroded, or deteriorated parts, and checking cleanliness of rail surfaces. The plant's nuclear operations procedures have numerous levels of controls and reviews, including assignment of responsibility for conducting performance evaluations as required, reviewing all the evaluations for accuracy and completeness, and analyzing data for trends, if applicable. Specific responsibilities are assigned to CCNPP personnel for monitoring these programs through periodic audits. These controls provide reasonable assurance that the associated activities will continue to be an effective means of monitoring the fuel handling equipment for the effects of general corrosion/oxidation.	Existing Program
13.	3.2	Fuel Handling	Stainless steel wire rope	Fatigue, Wear, Mechanical Degradation	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description. Performance Evaluations provide for visual inspection of hoisting ropes and drive cables for the fuel upending machines and transfer carriages which are visually inspected for damage. Performance Evaluations provide for wire rope inspection for the SFHM, RRM, the spent fuel inspection elevator, and the new fuel elevator prior to refueling campaigns. The checks for the SFHM and the elevators are also performed. The SFHM procedures require visual inspection of the hoisting rope while running the hoist through the full length of travel. The Refueling Machine Procedures require the same activities for the main hoist on each unit's RRM. The Fuel Elevators Procedures require visual inspection for damage to hoisting ropes for the spent fuel inspection elevator and the new fuel elevator.	Existing Programs

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
14.	Deleted						
15.	3.2	Fuel Handling, Cranes, RCS	Carbon steel parts of spent fuel cask handling crane, polar crane, intake structure semi-gantry crane, purge valve exhaust monorail hoist, reactor vessel head lift rig, RRM, containment roof jib crane, and transfer machine jib crane	General Corrosion	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description. Repetitive Tasks are credited for discovery and management of general corrosion effects in carbon steel parts by performing visual inspections.	Existing Program
16.	3.2	Fuel Handling	Stainless steel wire rope (SFHM and Fuel Elevators)	Fatigue, Wear, Mechanical Degradation, Distortion	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description. Repetitive tasks are credited for discovery and management of fatigue, wear, and mechanical degradation/ distortion of stainless steel wire rope, for the Auxiliary Building Handling Crane, Fuel Upending Machines, Transfer Jib Crane, and Fuel Elevators. These PM tasks have been modified to explicitly present the inspection requirements.	Existing Program
17.	3.2	Fuel Handling	Carbon steel chain (Containment purge monorail hoist)	General Corrosion, Wear	Load Handling and Fuel Handling Equipment	See Section 16.2.22 for a general description.	Existing Program
18.	Deleted						
19.	3.3A	Containment	Containment tendons	Prestress Losses	ASME Section XI, Subsections IWE and IWL	See Section 16.2.5 for a general description.	Existing Program
20.	3.3A	Containment	Containment tendons	General Corrosion	ASME Section XI, Subsections IWE and IWL	See Section 16.2.5 for a general description.	Existing Program
21.	3.3A	Containment	Grout under tendon bearing plates	Weathering	ASME Section XI, Subsections IWE and IWL	See Section 16.2.5 for a general description.	Existing Program
22.	3.3A	Containment	Containment liner	General Corrosion	ASME Section XI, Subsections IWE and IWL	See Section 16.2.5 for a general description.	Existing Program
23.	3.3A	Containment	Non-metallic portions of non-EQ electrical penetrations	Radiation and Thermal Damage	Containment Local Leakage Rate Testing	See Section 16.2.14 for a general description.	Existing Program
24.	3.3A	Containment	Steel components inside the Containment Structure	General Corrosion	Protective Coatings	See Section 16.2.24 for a general description.	Existing Program

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
25.	3.3A	Containment	Emergency air lock Personnel air lock and Equipment hatch gaskets	Elastomer Degradation	PM	See Section 16.2.23 for a general description. The Containment Personnel, Emergency Escape Air Lock and Equipment Hatch Adjustment, Lubrication, and Inspection Procedure, provides instructions for adjustment, lubrication, and inspection of the operating mechanism for the air lock. Degradation of the air lock gaskets is managed by routine replacement.	Existing Program
26.	3.3A	Containment	Embedded steel/rebar within the containment wall and dome	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
27.	3.3A	Containment	Steel components outside the Containment Structure	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
28.	3.3A	Containment	Refueling pool liner and permanent cavity seal ring	Intergranular stress corrosion cracking (IGSCC)	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
29.	3.3B	Turbine Building	Caulking and sealants that do not function as fire barriers	Weathering	Caulking and Sealant Inspection	See Section 16.2.11 for a general description.	Program has been implemented.
30.	3.3B	Turbine Building	Caulking and sealants that function as fire barriers	Weathering	Penetration Fire Barrier Inspection	See Section 16.2.20 for a general description.	Existing Program
31.	3.3B	Turbine Building	Carbon steel components	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
32.	3.3C	Intake Structure	Caulking and sealants that do not function as fire barriers	Weathering	Caulking and Sealant Inspection	See Section 16.2.11 for a general description.	Program has been implemented.
33.	3.3C	Intake Structure	Caulking, sealants, and expansion joints that function as fire barriers	Weathering	Penetration Fire Barrier Inspection	See Section 16.2.20 for a general description.	Existing Program
34a.	3.3C	Intake Structure	Concrete of fluid-retaining walls and slabs	Aggressive Chemical Attack	PM	See Section 16.2.23 for a general description. The repetitive tasks (Intake Structure Cavity Repairs and Cleaning) have been modified to include specific age-related degradation mechanisms (ARDMs) where they are not presently included and/or additional specified components/ subcomponents where they are not presently inspected.	Existing Program
34b.	3.3C	Intake Structure	Concrete of fluid-retaining walls and slabs	Aggressive Chemical Attack	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
35.	3.3C	Intake Structure	Embedded steel/rebar of fluid-retaining walls and slabs	General Corrosion	PM	See Section 16.2.23 for a general description. The repetitive tasks (Intake Structure Cavity Repairs and Cleaning) were modified to include specific ARDMs where they are not presently included and/or additional specified components/subcomponents where they are not presently inspected.	Existing Program
36.	3.3C	SW	Sluice gate wire rope and chain assemblies	Crevice Corrosion, MIC and Pitting	PM	See Section 16.2.23 for a general description.	Existing Program
37.	3.3C	Intake Structure	Carbon steel components	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
38.	3.3D	Miscellaneous Tank & Valve Enclosures	Carbon steel components	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
39.	3.3E / 5.12	Auxiliary Building (Main Steam/CVCS/ Feedwater Systems)	Pipe encapsulations	General Corrosion	PM	See Section 16.2.23 for a general description.	Existing Program
40.	3.3E	Auxiliary Building	Spent fuel pool (SFP) liner	IGSCC	Spent Fuel Pool	See Section 16.2.27 for a general description.	Existing Program
41.	NA	1A Diesel Building	Carbon steel members	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
42.	NA	1A Diesel Building	Caulking and sealants that function as fire barriers	Weathering	Fire Barrier Penetration Inspection	See Section 16.2.20 for a general description.	Existing Program
43.	NA	1A Diesel Building	Caulking and sealants that do not function as fire barriers	Weathering	Caulking and Sealants Inspection	See Section 16.2.11 for a general description.	Program has been implemented.
44.	3.3E	Auxiliary Building	SFP storage racks (Unit 1 side only)	Loss of Neutron-Absorbing Material	Spent Fuel Pool	See Section 16.2.27 for a general description.	Program has been modified.
45.	3.3E	Auxiliary Building and SBO Buildings	Caulking, sealants, and expansion joints that do not function as fire barriers	Weathering	Caulking and Sealant Inspection	See Section 16.2.11 for a general description.	Program has been implemented.
46.	3.3E	Auxiliary Building and SBO Buildings	Caulking, sealants, and expansion joints that function as fire barriers	Weathering, Galvanic Corrosion	Penetration Fire Barrier Inspection	See Section 16.2.20 for a general description.	Existing Program
47.	3.3E	Auxiliary Building and SBO Buildings	Carbon steel components	General Corrosion, SCC	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
48.	4.1	RCS	PUMP	Erosion, Galvanic Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
49.	4.1	RCS	HV	Galvanic Corrosion; SCC	BACI	See Section 16.2.7 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
50.	4.1	RCS	PP, CKV, CV, ERV, Motor-Operated Valve (MOV), PUMP, Steam Generator (SG) HX, PZV, RV, SV	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
51.	4.1	RCS	PP, CV, HV, PUMP and MOV	Wear	BACI	See Section 16.2.7 for a general description.	Existing Program
52.	4.1	RCS	PP, PUMP, RVI, and PZV (surge nozzle safe end)	Thermal Embrittlement	Cast Austenitic Stainless Steel (CASS)	See Section 16.2.10 for a general description.	Program has been implemented.
53.	4.1	RCS	PP and PZV	SCC/IGSCC and PWSCC	Alloy 600	See Section 16.2.3 for a general description.	Program has been modified.
54.	4.1	RCS	SG HX, SG Tubes and Tube Supports	Denting, Wear, Pitting, SCC/IGSCC, Intergranular Attack (IGA), Erosion, Erosion Corrosion	Steam Generator	See Section 16.2.28 for a general description.	Existing Program
55.	4.1	RCS	HV, MOV, SG Tubes	Wear, Pitting, SCC/IGSCC	Surveillance Testing	See Section 16.2.30 for a general description. Calvert Cliffs Surveillance Test Procedures (STPs), are credited for discovering these ARDMs for the SG HX tubes. These procedures will discover these ARDMs by determining if any of the SG HX tubes are leaking RCS coolant. These Calvert Cliffs procedures direct the user to perform calculations to determine the amount and potential source of RCS leakage. Any abnormal RCS leakage would be detected and actions taken to correct the leakage prior to a loss of the intended function. The CCNPP Technical Specifications provide the basis for the acceptance criteria of leakage rates.	Existing Program
55A.	4.1	RCS	PP (RPV Head Leak-off Line	SCC	Reactor Coolant	See Section 16.2.25 for a general description. Calvert Cliffs procedure implementation prevents SSC of the RPV head leak-off line by directing that the line be blown clear of fluids with compressed air. NOTE: Clearing the line of fluid eliminates the potential for this ADRM.	Existing Program
56.	4.1	RCS/RVI	PP, CKV, CV, ERV, SG HX, MOV, PUMP, PZV, and RV	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description. A one-time fatigue analysis was performed for the reactor coolant pumps (RCPs), MOVs, and pressurizer RVs. RCP Suction/Discharge and shutdown cooling (SDC) piping were added as locations to be monitored.	Program has been modified
57.	Deleted						
58.	Deleted						

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
59.	4.1	RCS	PP, PUMP, SG HX	General Corrosion	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
60.	4.1	RCS	PUMP, and MOV	Wear	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
61.	4.1	RCS	ERV, HV, RV, PUMP, MOV	Thermal Embrittlement	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
62.	4.1	RCS	SG HX	Denting, Pitting and SCC/IGSCC	Chemistry	See Section 16.2.12 for a general description. Procedures that control FWS chemistry, includes the SGs, condensate storage tank, feedwater, condensate, Main Steam System, heater drain tanks, condensate demineralizer effluent, SG blowdown ion exchanger effluent, and condensate precoat filters. These procedures control fluid chemistry in order to minimize the concentration of corrosive impurities (chlorides, sulfates, oxygen) and optimizes fluid pH. Control of fluid chemistry minimizes the corrosive environment for FWS components, and limits the rate and effects of corrosion/pitting, denting and SCC/IGSCC. Secondary chemistry parameters (e.g., pH, dissolved oxygen levels) are measured at procedurally-specified frequencies. The measured parameter values are compared against target values that represent a goal or predetermined warning limit. If a measured value is out of bounds, corrective actions are taken (e.g., power reduction, plant shutdown) in accordance with the plant secondary chemistry procedure. Remedial actions are specified to minimize corrosion degradation of components and to ensure that secondary system integrity is maintained.	Existing Program
63.	4.1	RCS	SG tube supports	Erosion-Corrosion	Chemistry	See Section 16.2.12 for a general description. See the description of Secondary Chemistry in Item 62 above.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
64.	4.1	RCS	RCP tube-in-tube seal water HX	IGA	Chemistry	See Section 16.2.12 for a general description. Primary Chemistry Procedures are credited with mitigating the effects of IGA on the RCP seal water HX (RCS side) by monitoring and maintaining the RCS chemistry. Calvert Cliffs Primary Chemistry lists the parameters to monitor (e.g., chloride, fluoride, sulfate, oxygen, pH), the frequency of monitoring these parameters, and the acceptable value or range of values for each parameter. The primary chemistry parameters are measured at procedurally-specified frequencies and are compared against target values, which represent a goal or predetermined warning limit. If a target value is approached or violated, corrective actions are taken as prescribed by the procedure, thereby ensuring timely response to chemical excursions.	Existing Program
65.	4.1	RCS	PP	PWSCC	Chemistry	See Section 16.2.12 for a general description. The Primary Chemistry Procedure is described in Item 64 above.	Existing Program
66.	4.1	RCS	SG HX	SCC/IGSCC and Pitting	Chemistry	See Section 16.2.12 for a general description. The Primary Chemistry Procedure is described in Item 64.	Existing Program
67.	4.1	RCS	RCP tube-in-tube seal water HX	IGA	Chemistry	See Section 16.2.12 for a general description. CC/SRW Chemistry Procedures are credited with mitigating IGA on the RCP seal water HX (CC System side) by monitoring and maintaining CC chemistry to control the concentrations of oxygen, chlorides, other chemicals, and contaminants. The water is treated with hydrazine to minimize the amount of oxygen in the water that aids in the prevention and control of most corrosive mechanisms. The procedures list the parameters to monitor, the frequency of monitoring these parameters, and the target and action levels for the CC System fluid parameters. The parameters currently monitored by the procedure are: pH, hydrazine, chloride, dissolved oxygen, dissolved copper, dissolved iron, suspended solids, gamma activity, and tritium activity (normally not a radioactive system). All of the parameters listed in this procedure currently have target values that give an acceptable range or limit for the associated parameter.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
68.	N/A	Various	Various	Various	Age-Related Degradation Inspection (ARDI)	See Section 16.2.2 for a general description.	ARDI inspections have been completed prior to entry into the Unit 1 and Unit 2 PEOs.
69.	Deleted						
70.	4.1	RCS	RCP tube-in-tube seal water HX	Wear	Reactor Coolant	See Section 16.2.25 for a general description. Calvert Cliffs utilizes Operating Instructions for management and discovery of wear in the RCP seal water HXs.	Existing Program
71.	4.2	Reactor Pressure Vessels (RPVs) and Control Element Drive Mechanisms (CEDMs)/ Electrical	RPV Alloy 600 components	SCC	Alloy 600	See Section 16.2.3 for a general description.	Program has been modified
72.	4.2	RPVs and CEDMs/ Electrical	RPV head and vessel, RPV studs, nuts and washers	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
73.	4.2	RPVs and CEDMs/ Electrical	RPV Head Nozzles and RPV anchor bolts	SCC	BACI	See Section 16.2.7 for a general description.	Existing Program
74.	4.2	RPVs and CEDMs/ Electrical	RPV/CEDM	Wear	BACI	See Section 16.2.7 for a general description.	Existing Program
75.	4.2	RPVs and CEDMs/ Electrical	RPV	Neutron Embrittlement	Comprehensive Reactor Vessel Surveillance	See Section 16.2.13 for a general description.	Program has been modified
76.	Deleted						
77.	4.2	RPVs and CEDMs/ Electrical	PZV (RPV), CEDM, Reactor Vessel Level Monitoring System (RVLMS)/Incore Instrumentation (ICI) Nozzles	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description. This AMP was modified to perform an engineering evaluation for CEDM and RVLMS/ICI nozzle components to ensure that the components are bounded.	Program has been modified
78.	4.2	RPVs and CEDMs/ Electrical	PZV (RPV), CEDM, RVLMS/ICI Nozzles	Fatigue	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
79.	4.2	RPVs and CEDMs/ Electrical	RPV components susceptible to general corrosion	General Corrosion	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
80.	4.2	RPVs and CEDMs/ Electrical	RPV anchor bolts	SCC	ASME Section XI, Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
81.	4.2	RPVs and CEDMs/ Electrical	RPV/CEDM	Wear	ASME Section XI; Subsections IWB, IWC, and IWD	See Section 16.2.4 for a general description.	Existing Program
82.	Deleted						
83.	4.2	RPVs and CEDMs/ Electrical	RPV head and vessel, O-ring flange sealing area	General Corrosion	PM	See Section 16.2.23 for a general description. A maintenance procedure is credited for the discovery of general corrosion on the RPV head and vessel O-ring flange sealing area. This procedure provides for inspection and acceptance criteria for minor pitting, nicks, and scratches near or on the O-ring sealing area. Any evidence of general corrosion would be found during the performance of this procedure.	Existing Program
84.	4.2	RPVs and CEDMs/ Electrical	RPV, studs, nuts, and washers	General Corrosion, Wear	PM	See Section 16.2.23 for a general description. A maintenance procedure is credited for the discovery of general corrosion and wear. This procedure specifies the procedural steps, materials, and acceptance criteria to be used in the cleaning and inspection of the RPV studs, nuts, and washers. The procedure describes what the inspection process should be looking for, and how to report any wear or damage that is found. The procedure also lists the acceptance criteria for contact between load bearing surfaces as a minimum of 70 percent.	Existing Program
85.	4.3	RVI	Combustion Engineering, Inc. (CE) Pressurized Water Reactor (PWR) RVI components as identified by Section IV.B3 of GALL, Rev 2	As identified for CE PWR RVI components in Section IV.B3 of GALL, Rev 2	RVI	See Section 16.2.26 for a general description.	Program has been implemented.
86.	Deleted						
87.	Deleted						
88.	Deleted						
89.	Deleted						
90.	Deleted						
91.	Deleted						
92.	Deleted						
93.	5.1	AFW	Class 'HB' PP	Crevice Corrosion General Corrosion MIC, Galvanic Corrosion and Pitting	Buried Pipe Inspection	See Section 16.2.8 for a general description.	Program has been implemented.
94.	Deleted						
95.	Deleted						
96.	Deleted						

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
97.	5.1	AFW	Pump Turbine and GOVs Valves and GOVs Cooler	Crevice Corrosion, General Corrosion, Erosion Corrosion, Pitting	PM	See Section 16.2.23 for a general description. Maintenance procedures for Auxiliary Feedwater Pump Turbine Overhaul, and AFW Pump Turbine Governor Valve Overhaul disassembles the turbine and turbine GOVs to inspect for damage. Measurements are taken to assure critical tolerances are within acceptance criteria. Specific subcomponents are inspected for wear, erosion, pitting, and/or surface cracking.	Program has been modified.
98.	5.1	AFW	CV	General Corrosion	PM	See Section 16.2.23. for a general description	Existing Program
99.	5.1	AFW	CKVs	Crevice Corrosion General Corrosion Pitting	Chemistry	See Section 16.2.12 for a general description. Demineralized Water Chemistry procedures have been established to: minimize impurity ingress to plant systems; reduce corrosion product generation, transport, and deposition; reduce collective radiation exposure through chemistry; improve integrity and availability of plant systems; and extend component and plant life. The demineralized water chemistry program controls fluid chemistry in order to minimize the concentration of corrosive impurities and dissolved oxygen. The demineralized water chemistry parameters (e.g., specific conductivity, dissolved oxygen, chloride, fluoride, sulfate) are measured at procedurally-specified frequencies. The measured parameter values are compared against target values, which represent a goal or predetermined warning limit. These procedures will mitigate the effects of crevice corrosion, general corrosion, and pitting of the internal surfaces of AFW System CKVs located at the interface with the Chemical Addition System.	Existing Program
100.	5.1	AFW	EB, HB, PP, CKVs, CVs, HVs, PUMP	Crevice Corrosion, General Corrosion, Pitting, Erosion Corrosion (for CVs)	Chemistry	See Section 16.2.12 for a general description. The Secondary Chemistry Procedure is described in Item 62.	Existing Program
101.	5.1	AFW	GOV valve pump turbine	Erosion Corrosion, Crevice Corrosion, General Corrosion, Pitting	Chemistry	See Section 16.2.12 for a general description. The Secondary Chemistry Procedure is described in Item 62.	Existing Program
102.	5.1	AFW	PUMP	Crevice Corrosion, Pitting	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
103.	5.1	AFW	PP	Crevice Corrosion, General Corrosion, Pitting	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
104.	5.1	AFW	TK (No. 12 Condensate Storage Tank)	Elastomer Degradation	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
105.	Deleted						
106.	Deleted						
107.	5.2	CVCS	All items exposed to borated water (due to leakage)	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
108.	5.2	CVCS	CVOPs with air internal environment	General Corrosion	PM	See Section 16.2.23 for a general description. The quality of Instrument Air (IA) System components that are within scope is verified approximately quarterly, based on operating experience and applicable industry standards, in accordance with PM Checklist. These checklists assure that the system is being maintained in accordance with industry standards for moisture (dew point) and particulate contamination. This procedure is used to mitigate the effects of general corrosion on CVOPs and PCVs.	Existing Program
109.	5.2	CVCS	Letdown Line piping, CKVs, CVs, HXs, HVs and TEs	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description.	Existing Program
110.	5.2	CVCS	Heat traced PP and components	SCC	Design Change and Modification Implementation	See Section 16.2.15 for a general description.	Commitment was completed
111.	5.2	CVCS	Letdown HX shell side	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. The CC/SRW Chemistry Procedure is described in Item 67.	Existing Program
112.	5.2	CVCS	Items with boric acid or borated water internal environments	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. The Primary Chemistry Procedure is described in Item 64.	Existing Program
113.	Deleted						
114.	Deleted						
115.	Deleted						
116.	Deleted						
117a.	5.3	CC	PUMP casings	General Corrosion, Crevice Corrosion, Pitting	PM	See Section 16.2.23 for a general description. The CC pumps are inspected for crevice corrosion/ pitting and general corrosion , using a maintenance procedure. This procedure instructs the user to inspect the pump impeller and shaft for erosion, corrosion/pitting, and inspect all pump parts for wear, corrosion, and mechanical damage.	Existing Program

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Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
117b.	5.3	CC	PUMP casings	General Corrosion, Crevice Corrosion, Pitting	Surveillance Testing	See Section 16.2.30 for a general description. Pump inspections rely on the CC pump periodic test of the pump active function (flow vs. head) to trigger the overhaul. The periodicity of this test is established by Technical Specifications.	Existing Program
118.	5.3	CC	PP, automatic vents, CKVs, CVs, HVs, PUMP casings, REs, RVs, SVs, TEs, TIs, TICs, HXs	Crevice Corrosion, Pitting	Chemistry	See Section 16.2.12 for a general description. The CC/SRW Chemistry Procedure is described in Item 67.	Existing Program
119.	5.3	CC	PP, CKVs, CVs, HVs, PUMP casings, RVs, TEs, TIs, HXs	General Corrosion	Chemistry	See Section 16.2.12 for a general description. The CC/SRW Chemistry Procedure is described in Item 67.	Existing Program
120.	5.3	CC	Automatic vents, CVs, HVs, RVs, SVs	Selective Leaching	Chemistry	See Section 16.2.12 for a general description. The CC/SRW Chemistry Procedure is described in Item 67.	Existing Program
121.	5.3	CC	HXs, PP	Erosion Corrosion	Chemistry	See Section 16.2.12 for a general description. The CC/SRW Chemistry Procedure is described in Item 67.	Existing Program
122.	Deleted						
123.	5.4	Compressed Air	All Compressed Air System carbon steel components	General Corrosion	PM	See Section 16.2.23 for a general description.	Existing Program
124.	Deleted						
125.	5.5	Containment Isolation Group	MOVs in borated water systems	Crevice Corrosion, General Corrosion and Pitting	BACI	See Section 16.2.7 for a general description.	Existing Program
126.	Deleted						
127.	5.6	Containment Spray	PP, CKVs, CVs, HVs, HXs, MOVs, and PUMPs that are exposed to borated water (due to leakage)	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
128.	5.6	Containment Spray	SDC HXs	General Corrosion, Crevice Corrosion, and/or Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
129.	5.6	Containment Spray	PP, CKVs, CVs, FEs, FOs, HVs, HXs, MOVs, PUMPs, RVs, TEs, and TIs) that are exposed to borated water (as process fluid)	General Corrosion, Crevice Corrosion, and/or Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 64 for a description of the Primary System Chemistry Procedure.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
130.	5.7	DFO	DFO TKs	Crevice Corrosion, General Corrosion, Pitting, Fouling, and MIC	DFO (Tanks and Chemistry)	See Section 16.2.16 for a general description. Under the DFO Storage Tank Water Check STP, water that may collect at the DFO tank bottom is periodically drained and fuel chemistry is analyzed. If the amount of drained water or fuel chemistry is found not to meet the established standards, corrective action is implemented as required. Draining the water will minimize the degradation of the internal surface of the carbon steel tank bottom, and will also minimize the possibility of MIC since microbes require water to survive and multiply. If more than one gallon of water is drained, the operator is required to notify the shift supervisor, and the situation will be investigated to determine and correct the source of the water.	Existing Program
131.	5.7	DFO	Buried PP	Crevice Corrosion, General Corrosion, Galvanic Corrosion, MIC, and Pitting	Buried Pipe Inspection	See Section 16.2.8 for a general description.	Program has been implemented.
132.	5.7	DFO	DFO Tanks	Crevice Corrosion, General Corrosion, Pitting, Fouling, and MIC	DFO (Tanks and Chemistry)	See Section 16.2.16 for a general description. Fuel oil chemistry is controlled, under this procedure, including testing for the presence of biologics. The procedure establishes surveillance frequencies, fuel oil specifications (e.g., viscosity, % water and sediment, particulate contamination, and biologics), and corrective actions. Sampling and analysis are performed on new fuel prior to unloading from fuel trucks. This procedure specifies limits for viscosity, water, and sediment for both receipt inspection and Technical Specification surveillance, in accordance with ASTM D975-81. This procedure requires the addition of a stabilizer/corrosion inhibitor prior to unloading fuel oil into the FOSTs. This approach provides assurance that the desired ratio of inhibitor to fuel oil exists. Corrosion inhibitor is added to the fuel to control corrosion of any exposed metal surfaces in the tank. A biocide is also added to the FOSTs for the initial addition, or if the presence of biological activity has been confirmed, to control MIC.	Existing Program
133.	5.7	DFO	All above-ground DFO items	Crevice Corrosion, General Corrosion, and Pitting	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
134.	5.7	DFO	DFO TKs	Weathering	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
135.	5.7	DFO	DFO TKs	Crevice Corrosion, General Corrosion, Pitting, Fouling, and MIC	DFO (Tanks and Chemistry)	See Section 16.2.16 for a general description. Draining water and chemistry testing/control of fuel oil provide a high degree of confidence that the effects of the plausible ARDMs will be minimized. However, the internal surfaces of the tank are not accessible during system walkdowns. CCNPP has developed PM Tasks to perform internal inspections of the FOSTs at periodic intervals, consisting of UT assessments of the condition of the tank interior conducted in accordance with American Petroleum Institute Standard 653 for FOST inspections. The results of these inspections are documented as part of the PM program. These results are used to assess the overall condition of the tank interior and to determine the appropriate intervals for future inspections.	Existing Program
136.	Deleted						
137.	Deleted						
138.	Deleted						
139.	Deleted						
140.	5.8	EDG	Jacket water expansion TKs and cooling water PP	General Corrosion, Crevice Corrosion, and Pitting	Chemistry	See Section 16.2.12 for a general description. Procedures are credited with mitigating the effects of general corrosion, crevice corrosion, and pitting of the cooling water piping and jacket water expansion tanks by monitoring and maintaining EDG jacket water chemistry (e.g., pH, dissolved oxygen). This procedure contains two different sets of chemistry parameters, one for the Fairbanks Morse EDGs, and one for the Societe Alsacienne De Constructions Mecaniques De Mulhouse EDG. The water is treated with hydrazine or corrosion inhibitors to minimize the amount of oxygen in the water, which aids in the prevention and control of most corrosive mechanisms. Continued maintenance of system water quality will mitigate EDG jacket water expansion tank and cooling water piping degradation. The procedure provides for a prompt review of EDG jacket water chemistry parameters so that steps can be taken to return chemistry parameters to normal levels, and, thus, minimize the effects of general corrosion, crevice corrosion, and pitting.	Existing Program

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
141.	5.8	EDG	EDG lube oil "Y" strainers	General Corrosion, Crevice Corrosion, and Pitting	PM	See Section 16.2.23 for a general description. PM checklists (Clean/Inspect 2B, 1B, and 2A EDG Lube Oil "Y" Strainers and Baskets) are credited with discovery of the effects of degradation on the "Y" strainer internal surfaces. The PMs tasks have been modified to check for signs of corrosion on the "Y" strainer internal surfaces during performance of the procedures.	Existing Program
142.	5.8	EDG	EDG Starting Air (SA) and CA intake piping and SA system CKVs	General Corrosion, Crevice Corrosion, and Pitting	PM	See Section 16.2.23 for a general description. A PM procedure (disassemble, Inspect and Overhaul EDG CKV) is credited with discovery of the effects of pitting, crevice corrosion, and general corrosion of the internal surfaces of EDG SA System CKVs. This procedure has been modified to inspect specifically for corrosion of piping and check for the presence of debris in valves that could indicate the piping in these systems is undergoing corrosion.	Existing Program
143.	5.8	EDG	EDG SA/CA intake/ exhaust piping, EDG intake filters and intake/ exhaust mufflers, EDG exhaust piping and flame arrestors	General Corrosion, Crevice Corrosion, Erosion Corrosion, Particulate Wear, Erosion, Fatigue, and Pitting	PM	See Section 16.2.23 for a general description. A PM procedure (Inspect EDG Air Intake Filters) is credited for the discovery of effects of corrosion on the internal surfaces of the EDG SA/CA intake piping, internal surfaces of the EDG intake filters and intake mufflers, and external surfaces of the EDG intake filters and exhaust mufflers. The MPM has been modified to inspect the attached piping for signs of corrosion.	Existing Program
144.	5.8	EDG	EDG SA and CA intake PP	General Corrosion	PM	See Section 16.2.23 for a general description. PM repetitive tasks are credited with the discovery of the effects of corrosion of internal piping surfaces for EDG SA, and CA intake systems. These PM repetitive tasks have been modified to inspect specifically for corrosion of piping and check for the presence of debris in valves that could indicate the piping in these systems is undergoing corrosion.	Existing Program
145.	5.8	EDG	EDG SA and CA intake PP	General Corrosion, Crevice Corrosion, and Pitting	PM	A PM procedure (Remove Relief Valve, Test and Reinstall) is credited with the discovery of degradation of the internal piping surfaces for EDG SA, and CA intake systems. This procedure has been modified to inspect specifically for corrosion of piping and check for the presence of debris in valves that could indicate the piping in these systems is undergoing corrosion.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
146.	5.8	EDG	DFO day TKs, drip TKs and associated level switches	General Corrosion, Crevice Corrosion, Pitting, and MIC	DFO (Tanks and Chemistry)	See Section 16.2.16 for a general description. A procedure is credited for mitigating the effects of crevice corrosion, general corrosion, MIC and pitting on the interior surfaces of the EDG DFO day tanks, and drip tanks with associated tank-mounted level switches. Under this procedure, fuel oil chemistry is controlled, including testing for the presence of biologics. The procedure establishes surveillance frequencies, fuel oil specifications (e.g., viscosity, % water and sediment, particulate contamination and biologics), and corrective actions. Sampling and analysis are performed on new fuel prior to unloading from fuel trucks. This procedure specifies limits for water, viscosity, and sediment for both receipt inspection and Technical Specification surveillance for fuel oil in the FOSTs in accordance with ASTM-D975-81. The procedure currently has target values and action levels that give an acceptable range or limit for a given parameter. This procedure now requires the addition of a stabilizer/corrosion inhibitor prior to unloading fuel oil into the FOSTs. This approach provides a better assurance that the desired ratio of inhibitor to fuel oil exists. This procedure was revised to incorporate criteria in accordance ASTM-D270-65 for taking quarterly samples from the diesel FOSTs. This revision involves taking multilevel samples from each diesel FOST rather than sampling only from the tank bottom.	Existing Program
147.	5.8	EDG	DFO day TKs, drip TKs	General Corrosion, Crevice Corrosion, Pitting, and MIC	Surveillance Testing	See Section 16.2.30 for a general description. STPs (Testing EDGs and the 4 kV LOCA Sequencers) are credited for mitigation of crevice corrosion, general corrosion, pitting, and MIC on the interior of the DFO day tanks. The procedures provide for periodic draining of DFO day tank of any water that may be present, which minimizes the corrosive effects of water on carbon steel. The tank sample is taken and visually examined for the presence of water in the fuel. This procedure is currently performed monthly after the EDGs are shut down from testing.	Existing Program
148.	Deleted						
149.	Deleted						

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
150.	5.9	FWS	PP	Erosion Corrosion	Flow Accelerated Corrosion	See Section 16.2.18 for a general description. All of the FWS piping subject to aging management review, as well as all the piping in the system not subject to aging management review, are included in this program.	Existing Program
151.	5.9	FWS	PP	Low Cycle Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description.	Existing Program
152.	5.9	FWS	CKVs	Erosion Corrosion	PM	See Section 16.2.23 for a general description.	Existing Program
153.	5.9	FWS	PP, CKVs, HVs, MOVs, and TEs	General Corrosion Crevice Corrosion, Erosion Corrosion, and/or Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 62 for a description of Secondary Systems Chemistry procedures.	Existing Program
154.	Deleted						
155.	5.10	FP	NSR portions of the below listed systems required for safe shutdown (Appendix R Fire Scenario) are included in the group 041 - CVCS 064 - RCS	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
156.	5.10	FP	NSR portions of the below listed systems required for safe shutdown (Appendix R scenario), are included in the group: 008 – Well Water 011 – Service Water 013 – FP 015 – Component Cooling 019 – Compressed Air 029 – Plant Heating 036 – AFW 037 – Demin Water and Cond Storage 044 - Condensate 053 – Plant Drains 071 – Liquid Waste 083 – Main Steam	For Plausible Aging Mechanisms Applicable to These Systems	Fire Protection	See Section 16.2.21 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
157.	5.10	FP	NSR portions of the below listed systems required for safe shutdown (Appendix R Fire Scenario) are included in the group 013 – FP 023 – DFO 036 – AFW 053 – Plant Drains	For Plausible Aging Mechanisms Applicable to These Systems	Fire Protection	See Section 16.2.21 for a general description.	Existing Program
158.	5.10	FP	NSR portions of the below listed systems required for safe shutdown (Appendix R Fire Scenario) are included in the group 008 – Well Water 011 – SRW 013 – FP 015 – CC 019 – Compressed Air 029 – Plant Heating 036 – AFW 037 – Demin Water & Cond Storage 044 – Condensate 053 – Plant Drains 071 – Liquid Waste 083 – Main Steam	For Plausible Aging Mechanisms Applicable to These Systems	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
159.	Deleted						
160.	Deleted						
161.	Deleted						
162.	5.11A	Auxiliary Building Heating and Ventilation	Fans	Dynamic Loading	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
163.	5.11A	Auxiliary Building Heating and Ventilation	Duct flexible collars, GD	Elastomer Degradation, Wear	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
164.	5.11A	Auxiliary Building Heating and Ventilation	Ducts, HXs	General and Crevice Corrosion, Pitting	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
165.	5.11A	Auxiliary Building Heating and Ventilation	HX	General and Crevice Corrosion, Pitting	PM	See Section 16.2.23 for a general description.	Existing Program
166.	Deleted						

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
167.	Deleted						
168.	5.11B	Primary Containment Heating and Ventilation	Cooling coil external surfaces	Crevice Corrosion, Pitting	PM	See Section 16.2.23 for a general description.	Existing Program
169.	5.11B	Primary Containment Heating and Ventilation	Fans	Dynamic Loading	PM	See Section 16.2.23 for a general description.	Existing Program
170.	5.11B	Primary Containment Heating and Ventilation	Fans	Dynamic Loading	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
171.	5.11B	Primary Containment Heating and Ventilation	Cooler housings, HXs	General Corrosion, Crevice Corrosion, Pitting, MIC	PM	See Section 16.2.23 for a general description.	Existing Program
172.	5.11B	Primary Containment Heating and Ventilation	Cooler rubber boots, GD	Radiation Damage, Elastomer Degradation, Wear	PM	See Section 16.2.23 for a general description.	Existing Program
173.	5.11B	Primary Containment Heating and Ventilation	Cooling coil internal surfaces	Crevice Corrosion, Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
174.	5.11B	Primary Containment Heating and Ventilation	Duct flexible collars, GD	Elastomer Degradation, Wear	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
175.	Deleted						
176.	5.11C	Control Room and Diesel Generator Buildings HVAC	Dampers, ducts, fans, filters, HXs	General Corrosion, Crevice Corrosion, MIC, and Pitting	PM	See Section 16.2.23 for a general description. Existing procedures have been modified to include specific items with respect to discovery of these ARDMs to help ensure each plausible ARDM is being adequately managed.	Existing Program
177.	Deleted						
178.	5.11C	Control Room and Diesel Generator Buildings HVAC	Fans	Dynamic Loading	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
179.	5.11C	Control Room and Diesel Generator Buildings HVAC	Duct flexible collars	Elastomer Degradation, Wear	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
180.	5.11C	Control Room and Diesel Generator Buildings HVAC	Dampers, ducts, fans, FD, filters, HXs	General Corrosion, Crevice Corrosion, MIC, and Pitting	Structure and System Walkdowns .	See Section 16.2.29 for a general description.	Existing Program
181.	Deleted						
182.	5.12	Main Steam, Extraction Steam	PP	Erosion Corrosion	Flow Accelerated Corrosion (FAC)	See Section 16.2.18 for a general description.	Existing Program
183.	5.12	Main Steam, Extraction Steam	PP, CKVs, CVs, FOs, HVs, HXs, and MOVs	General Corrosion, Crevice Corrosion, Pitting, and Erosion Corrosion	Flow Accelerated Corrosion (FAC)	See Section 16.2.18 for a general description.	Existing Program
184.	5.12	Main Steam, Compressed Air	PP, CVs, HVs, ACC	General Corrosion	PM	See Section 16.2.23 for a general description The quality of the air to IA System components that are within scope is periodically verified, in accordance with PM Checklists. These checklists assure that the system is being maintained in accordance with industry standards for moisture (dewpoint) and particulate contamination. Maintenance of dry air mitigates corrosion of compressed air components.	Existing Program
185.	5.12	Main Steam	CVs	General Corrosion, Erosion Corrosion	PM	See Section 16.2.23 for a general description. Main steam isolation valves (MSIVs) are periodically inspected. The PM activities require the periodic disassembly and inspection of these valves, per the requirements of the PM procedure. These regularly scheduled inspections would result in the detection of the effects of degradation such that corrective action would be taken. Existing procedures were modified to include specific items with respect to discovery of these ARDMs to help ensure each plausible ARDM is being adequately managed.	Program has been modified.
185A.	5.12	Main Steam	MSIV actuation system	Any Plausible Aging Effects	PM	See Section 16.2.23 for a general description.	Existing Program
186.	5.12	Main Steam	SG blowdown HXs	General Corrosion, Crevice Corrosion, Erosion Corrosion, and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
187.	5.12	Main Steam	SG blowdown radiation monitor cooler	General Corrosion, Crevice Corrosion, Pitting, and Selective Leaching	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of CC/SRW Chemistry Procedure.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
188.	5.12	Chemical Addition	HVs	General Corrosion, Crevice Corrosion, and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 99 for a description of the Demineralized Water Chemistry Procedure.	Existing Program
189.	5.12	Main Steam, Nitrogen & Hydrogen	PP, CKVs, CVs, HVs, FOs, HXs, MOVs, TEs, and TKs	General Corrosion, Crevice Corrosion, Pitting, Erosion Corrosion, Wear, Selective Leaching	Chemistry	See Section 16.2.12 for a general description. See Item 62 for a description of the Secondary Chemistry Procedure.	Existing Program
190.	Deleted						
191.	5.13	NSSS Sampling	Sample coolers, CVs, and HVs that are exposed to borated water (due to leakage)	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
192.	5.13	NSSS Sampling	CVOP	General Corrosion	PM	See Section 16.2.23 for a general description. The quality of IA System components that are within scope is periodically verified in accordance with PM Checklists. These checklists assure that the system is being maintained in accordance with industry standards for moisture (dew point) and particulate contamination.	Existing Program
193.	5.13	NSSS Sampling	PP and Valves in the RCS hot leg sampling line	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description. The FMP was modified to include an engineering evaluation of the piping and valves in the RCS hot leg sampling line.	Program has been modified
194.	5.13	NSSS Sampling	HXs that are exposed to chemically treated water from the CC System	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
195.	5.13	NSSS Sampling	Sample coolers, CVs, HVs, and SVs that are exposed to borated water (as process fluid)	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 64 for a description of Primary Chemistry Procedure.	Existing Program
196.	5.13	NSSS Sampling	Sample coolers and HVs that are exposed to steam and feedwater (as process fluid)	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 62 for a description of the Secondary Chemistry Procedure.	Existing Program
197.	5.13	NSSS Sampling	PUMP	Rubber Degradation	PM	See Section 16.2.23 for a general description.	Existing Program
198.	Deleted						
199.	Deleted						
200.	Deleted						

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
201.	5.15	SI	Refueling Water Tank (RWT) penetrations and associated welds	SCC	BACI	See Section 16.2.7 for a general description. Engineering review discussed in Item 204 concluded that periodic inspections of RWT penetrations and associated welds will be performed as part of BACI Program to manage the effects of aging on subject structures, systems, and components. PM tasks have been generated to perform the inspections.	Existing Program
202.	5.15	SI	PP (fasteners), CKVs, CVs, HVs, HXs, MOVs, RVs, PUMPs, and TKs that are exposed to borated water (due to leakage)	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
203.	5.15	SI	PP and valves in the safety injection tank (SIT) injection mode flowpath	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description. In response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," CCNPP identified the potential for thermal stratification in the piping between the SIT outlet CKVs and the loop inlet CKVs, and subsequently confirmed the natural convection phenomenon. Since the current piping analysis for the affected portions of the SI System does not include the additional stresses imposed by thermal stratification, CCNPP completed an engineering review of the industry's task reports and determined: (a) any necessary changes to the piping analyses of record for the SI System; and (b) the impact of such changes on fatigue usage parameters used by the FMP. SI System locations were added to the FMP.	Commitment was completed
204.	5.15	SI	RWT penetrations and associated welds	SCC	BACI	See Section 16.2.7 for a general description. An engineering evaluation determined that RWT penetrations will leak before break. This evaluation confirmed that detection of minor leakage by visual inspection will adequately manage the RWT penetrations prior to a challenge of the structural integrity under design basis conditions. See Item 201.	Existing program
205.	5.15	SI	PP and valves in the SIT injection and SDC mode flowpaths	Fatigue	Fatigue Monitoring	See Section 16.2.19 for a general description.	Program has been modified
206.	5.15	SI	Low pressure SI pump seal HXs and high pressure SI Pump seal coolers	General Corrosion, Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
207.	5.15	SI	All SI System device types that are exposed to borated water (as process fluid)	General Corrosion, Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 64 for a description of the Primary Chemistry Procedure.	Existing Program
208.	5.15	SI	RWT penetrations and associated that are exposed to borated water (as process fluid)	SCC	Chemistry	See Section 16.2.12 for a general description. See Item 64 for a description of the Primary Chemistry Procedure.	Existing Program
209.	5.15	SI	RWT HXs	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 99 for a description of the Demineralized Water Chemistry Procedure.	Existing Program
210.	Deleted						
211.	5.15	SI	RWT perimeter seal	Weathering	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
212.	Deleted						
213.	Deleted						
214.	5.16	SW	CC and SRW HXs	Crevice Corrosion, Erosion Corrosion, General Corrosion, MIC, Pitting, and Elastomer Degradation	PM	See Section 16.2.23 for a general description.	Existing Program
215.	5.16	SW	FOs	Crevice Corrosion, Erosion Corrosion, MIC, Particulate Wear Erosion, and Pitting	PM	See Section 16.2.23 for a general description.	Existing Program
216.	5.16	SW	Internally lined PP, BSs, CKVs, CVs, HVs, LC, LJ, and PUMPs	Crevice Corrosion, Galvanic Corrosion, General Corrosion, MIC, Particulate Wear Erosion, Pitting, Elastomer Degradation, and Selective Leaching	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected. Degradation of the saltwater strainer carbon steel drain lines is managed by routine replacement.	Existing Program
217.	5.16	SW	Emergency Core Cooling System Pump Room air coolers	Crevice Corrosion, General Corrosion, MIC, and Pitting	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
218.	5.16	SW	ACCs, CVs, HVs, and PCVs, with air internal environment	General Corrosion	PM	See Section 16.2.23 for a general description.	Existing Program

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Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
219.	5.16	SW	CC and SRW HXs	Crevice Corrosion, General Corrosion, and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
220.	5.16	SW	SW System bolting	General Corrosion, Crevice Corrosion, and Pitting	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
221.	Deleted						
222.	Deleted						
223.	Deleted						
224.	Deleted						
225.	5.17	SRW	Air-operated valves (CV, HV)	General Corrosion	PM	See Section 16.2.23 for a general description. The quality of IA System components that are within scope is periodically verified, in accordance with PM Checklists. These checklists assure that the system is being maintained in accordance with industry standards for moisture (dew point) and particulate contamination. Mitigation of general corrosion of the SRW air-operated valves. The exposure to moisture is minimal and short-term, and is not expected to result in significant levels of degradation of the carbon steel components.	Existing Program
226.	5.17	SRW	PP, AVV, CKVs, CVs, FEs, FOs, HVs, PUMPs, REs, RVs, TEs, TIs, TKs	Crevice Corrosion and Pitting	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
227.	5.17	SRW	PP, AVVs, CKVs, CVs, HVs, PUMPs, RVs, TIs	General Corrosion	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
228.	5.17	SRW	CVs, HVs, and PUMPs	Selective Leaching	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
229a.	5.17	SRW	PUMPs	General Corrosion, Crevice Corrosion and Pitting	PM	See Section 16.2.23 for a general description. The SRW pumps are inspected for general corrosion, crevice corrosion/pitting using the PM Pump Overhaul Procedure. This procedure instructs the user to inspect certain pump components for erosion, wear, and mechanical damage. The procedure has been modified to include inspections for general corrosion, crevice corrosion/pitting on the pump casing and bushings. The procedure directs the user to contact the System Engineer if any of these indications are found, and to replace parts as necessary.	Program has been modified.
229b.	5.17	SRW	PUMPs	General Corrosion, Crevice Corrosion and Pitting	Surveillance Testing	See Section 16.2.30 for a general description. Pump inspections rely on the SRW pump periodic test of the pump active function (flow vs. head) to trigger the overhaul. The periodicity of these tests is established by Technical Specifications.	Existing Program
230.	Deleted						
231.	5.18	SFPC	All items that are exposed to borated water (due to leakage)	General Corrosion	BACI	See Section 16.2.7 for a general description.	Existing Program
232.	5.18	SFPC	SFP demineralizer, filter and pipe supports	General Corrosion, Boric Acid Corrosion	PM	See Section 16.2.23 for a general description. PM tasks have been modified to explicitly call for inspection of the components for signs of boric acid corrosion.	Existing Program
233.	5.18	SFPC	SFPC PUMPs	Cavitation Erosion, Erosion Corrosion	PM	See Section 16.2.23 for a general description. PM tasks have been modified to explicitly present inspection requirements.	Existing Program
234.	5.18	SFPC	HXs	General Corrosion	Chemistry	See Section 16.2.12 for a general description. See Item 67 for a description of the CC/SRW Chemistry Procedure.	Existing Program
235.	6.1	Cables	For information concerning the Aging Management Programs for LRA Section 6.1, Cables, see items 258 through 266				

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
236a.	6.2	Electrical Commodities	Battery terminals/ charger and inverter cabinets	Electrical Stressors, General Corrosion, and Wear	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
236b.	6.2	Electrical Commodities	Battery terminals/ charger and inverter cabinets	Electrical Stressors, General Corrosion, and Wear	Surveillance Testing	See Section 16.2.30 for a general description.	Existing Program
237.	6.2	Electrical Commodities	Breaker cabinets	Electrical Stressors, Wear, and Fatigue	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
238.	6.2	Electrical Commodities	Bus cabinets	Electrical Stressors, Wear, and Fatigue	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
239.	6.2	Electrical Commodities	Bus cabinets	Electrical Stressors, Wear, and Fatigue	PM	See Section 16.2.23 for a general description. New PM Tasks have been created to include specific ARDMs for components/sub-components that were not initially inspected.	Existing Program
240.	6.2	Electrical Commodities	Motor-control cabinets panels	Electrical Stressors, Wear, Fatigue, and Dynamic Loading	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
241.	6.2	Electrical Commodities	Motor-control cabinets panels	Electrical Stressors, Wear, Fatigue, and Dynamic Loading	PM	See Section 16.2.23 for a general description. New PM Tasks have been created to include specific ARDMs for components/sub-components that were not initially inspected.	Existing Program
242.	6.2	Electrical Commodities	Miscellaneous panels	Electrical Stressors, Wear, Fatigue, and Dynamic Loading	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
243.	6.2	Electrical Commodities	Miscellaneous panels	Electrical Stressors, Wear, Fatigue, and Dynamic Loading	PM	See Section 16.2.23 for a general description. New PM Tasks have been created to include specific ARDMs for components/sub-components that were not initially inspected.	Existing Program

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AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
244.	6.2	Electrical Commodities	Local control station panels	Electrical Stressors, Wear, Fatigue, and General Corrosion	PM	See Section 16.2.23 for a general description. PM tasks have been modified to include specific ARDMs where they were not initially included and/or additional specified components/ subcomponents where they were not initially inspected.	Existing Program
245.	6.2	Electrical Commodities	Local control station panels	Electrical Stressors, Wear, Fatigue, and General Corrosion	PM	See Section 16.2.23 for a general description. New PM Tasks have been created to include specific ARDMs for components/sub-components that were not initially inspected.	Existing Program
245a.	NA	Electrical Commodities	Fuse Holders	Electrical Stressors, Corrosion, Fatigue	PM	See Section 16.2.23 for a general description.	Existing Program
245b.	NA	Electrical Commodities	Fuse Holders	Electrical Stressors, Corrosion, Fatigue	Surveillance Testing	See Section 16.2.30 for a general description.	Existing Program
246.	6.3	EQ Equipment	All long-lived components on the EQ Master List	Thermal, Radiative, and Kapton-Unique Aging Effects	Environmental Qualification	See Section 16.2.17 for a general description.	Existing Program
247.	6.3	EQ Equipment	EQ Penetrations	Thermal, Radiative	Containment Local Leakage Rate Testing	See Section 16.2.14 for a general description.	Existing Program
248.	6.3	EQ Equipment	EQ Penetrations	General Corrosion	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
249.	6.3	EQ Equipment	EQ Penetrations	General Corrosion	Protective Coatings	See Section 16.2.24 for a general description.	Existing Program
250.	Deleted						
251.	Deleted						
252.	Deleted						
253.	6.4	Instrument Lines	Instrument line supports	General Corrosion and Elastomer Hardening	Structure & System Walkdown	See Section 16.2.29 for a general description.	Existing Program
254.	Deleted						
255.	NA	Piping External Surfaces	External pipe surfaces	Various Corrosion Mechanisms	BACI	See Section 16.2.7 for a general description.	Existing Program
256.	NA	Piping External Surfaces	External pipe surfaces	Various Corrosion Mechanisms	Structure and System Walkdowns	See Section 16.2.29 for a general description.	Existing Program
257.	Deleted						
258.	6.1	Cables	In-Scope Random-Lay tray Electrical Cables and Connections (low and medium voltage)	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented
259.	6.1	Cables	In-Scope Maintained-Spacing Tray Electrical Cables and Connections (low and medium voltage)	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
260.	6.1	Cables	In-Scope Thermal and Radiative Synergy Electrical Cables and Connections (low and medium voltage)	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented
261.	6.1	Cables	In-Scope Large-Motor Terminations	Thermal Stress	Cable	See Section 16.2.9 for a general description.	Program has been implemented
262.	N/A	Cables	In-Scope Inaccessible Power Cables	Thermal Stress, Radiation Stress, Moisture/Submergence/Wetting	Cable	See Section 16.2.9 for a general description. This program requires that in-scope inaccessible or underground power cables (greater than or equal to 400 volts) exposed to adverse localized environments caused by significant moisture (e.g., cable wetting or submergence) are tested on a periodic (i.e., at least every 6 years) basis. The specific type of testing utilized is capable of detecting reduced insulation resistance of the cables' insulation system due to wetting or submergence. In addition this program requires that periodic (i.e., annually) visual inspections of applicable manholes an underground cable duct banks are performed to ensure that cables are not wetted or submerged, the cable support structures are intact and that the dewatering/drainage systems are operating properly.	Program has been implemented
263.	6.1	Cables	In-Scope Sensitive, Low Current Instrumentation Circuits	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description. This program requires that in-scope low current, high voltage instrumentation circuit cables and connections are assessed, via review/evaluation of periodic instrument calibration results/findings, to identify potential cable and connection insulation material degradation that could have an impact on circuit operation. When an instrument is found to be significantly out of calibration, additional evaluation is performed on the circuit, including the cable and connections, as required. The review of periodic instrument calibration results is performed at a maximum of 10 year intervals. This periodic review/ evaluation activity, combined with the performance of visual inspections of the cables in accessible areas exposed to adverse localized environments, is sufficient to manage potential ARDMs.	Program has been implemented
264.	6.1	Cables	In-Scope, 4 kV Bus Insulating Boots	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented
265.	6.1	Cables	In-Scope ITE (ITE Imperial Company) MCC Internal Wiring	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Item No.	LRA Section	System	Components	Aging Mechanism	Program	Program Description	Implementation Schedule
266.	N/A	Cables	In-Scope EQ Cables and Connections	Thermal Stress, Radiation Stress, Moisture	Cable	See Section 16.2.9 for a general description.	Program has been implemented

TABLE 16-2

AGING MANAGEMENT PROGRAMS (AMP), INDEXED BY LRA SECTION AND SYSTEM

Component Key

ACC	Accumulator	HX	Heat Exchanger
AVV	Auto Vent Valve	LC	Level Controller
BS	Basket Strainer	LJ	'LJ' Pipe Class
CA	Combustion Air	MC	Moisture/Humidity Controller
CKV	Check Valve	PCV	Pressure Control Valve
CV	Control Valve	PP	Piping
CVOP	Control Valve Operator	PUMP	Pump/Driver Assembly
EB	'EB' Pipe Class	PZV	Pressure Vessel
ERV	Electro-Relief Valve	RE	Radiation Element
FD	Fire Damper	RV	Relief Valve
FE	Flow Element	SV	Solenoid Valve
FO	Flow Orifice	TE	Temperature Element
GD	Gravity Damper	TI	Temperature Indicator
GOV	Governor	TIC	Temperature Indicating Controller
HB	'HB' Pipe Class	TK	Tank
HV	Hand Valve		