NT APPENDIX G

RESPONSES TO 70 CRITERIA

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CRITERION 1 - QUALITY STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could effect 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.

This criterion is met. Components of the engineered safeguards systems are designed and fabricated in accordance with established codes and/or standards as required to assure that their quality is in keeping with the safety function of the component.

The following codes, standards, and procedures judged to be required to assure such quality will be applied. It is not intended, however, to limit quality standards requirements to this list.

High Pressure Injection, Low Pressure Injection, and Containment Spray Pumps

a) Pressure containing material have been tested and examined per ASME Code, Section VIII. Castings have been liquid penetrant inspected in accordance with Appendix VIII of Section VIII of ASME B&PV Code. Acceptance standards are in accordance with USAS B31.1 Case N-10.

b) Butt welds have been fully radiographed in accordance with ASME Code, Section VIII, paragraph UW-51.

c) Fillet welds have been liquid penetrant inspected in accordance with ASME B&PV Code Section VIII, Appendix VIII.

d) The pump supplier submitted certified mill test reports of pressure containing materials.

e) Pressure containing parts were hydrostatically tested to 1½ times design pressure by the supplier in accordance with ASME Code, Section VIII.

f) The pumps will undergo periodic leak testing after installation.

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g) At least one pump of each type has been hydraulic-performance tested for capacity and head, in accordance with the requirements of the Hydraulics Institute and the Power Test Code PTC-8.2. One pump of each type has been tested for operation at stated NPSH and stated thermal transient conditions.

h) Special consideration was given the design of the pump seals to provide a high degree of assurance of their proper operation, including compatibility of seal materials with water chemistry conditions and minimum dependence on externally supplied cooling water.

i) Pump drive motors conform to NEMA standards. The pump supplier provided motor test data for at least one pump of each type.

<u>Stored Energy Tanks</u>

ASME Code, Section III, Class C

<u>Safety Injection and Containment Spray System Motor Operated Valves and Control Valves</u>

a) The design criteria for pressure containing parts is in accordance with USASI B16.5. Castings were radiographed in accordance with ASTM-E-71-64 or ASTM-E-186-65T as applicable. Radiographs were made in accordance with ASTM-E-94-62.

b) Pressure containing materials were tested and examined per ASME Code, Section VIII.

c) Radiographic inspection of pressure containing welds was performed in accordance with the requirements of ASME Code, Section VIII, ASTM-E-99-63 and ASTM-E-94-62.

d) Certified mill test reports of pressure containing materials were provided by the supplier.

e) Valve motors are in accordance with NEMA standards. Particular attention was given to the design of the valve motors to ensure their proper operation in a post-accident environment.

f) Pressure containing parts were hydrostatically tested in accordance with USASI B16.5.

g) Periodic leak testing will be performed on the valves after installation.

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Shutdown Heat Exchangers

a) Pressure containing materials were tested and examined per ASME Code, Section III, Class A (on tube side) and Class C (on shell side).

b) Heat transfer design and physical design are in accordance with TEMA standards.

c) Certified mill test reports of pressure containing materials were provided by the supplier.

d) Radiographic inspection of pressure containing welds was performed in accordance with the requirements of ASME Code, Section VIII, ASTM-E-99-63 and ASTM-E-94-62.

e) Pressure containing parts were hydrostatically tested in accordance with ASME Code, Section III.

Omaha Public Power District reviewed tests and inspections during material procurement and fabrication of the components to assure conformance with the quality control techniques of the applicable codes and standards. Records of all test and inspection results will be maintained.

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CRITERION 2 - PERFORMANCE STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect 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 for 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.

This criterion is met. The systems and components of the Fort Calhoun reactor facility essential to the prevention or mitigation of accidents which could affect the public health and safety are designed and will be fabricated and erected 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, floods, winds, ice and other local site effects.

The containment will be designed for simultaneous stresses produced by the dead load, by 60 psig internal pressure at the associated design temperature, and by the application of forces resulting from an earthquake whose ground motion is 0.08g horizontally and 0.053g vertically. Further, the containment structure will be designed to withstand a sustained wind velocity of 90 mph in combination with the dead load and design internal pressure and temperature conditions. The wind load is based on the highest velocity wind at the site location for 100 year period or recurrence: 90mph base wind at 30 feet above ground level. Other Class I structures will be designed similarly except that no internal pressure loading is applicable. Class I systems will be designed for their normal operating loads acting concurrently with the earthquake described above.

The containment structure is predicted to withstand without loss of function the simultaneous stresses produced by the dead load, by 75 psig internal pressure and temperature associated with this pressure and by an earthquake whose ground motion is 0.10 horizontally and 0.07 vertically.

The containment structure is predicted to withstand without loss of function 125% of the force corresponding to a 90mph wind impinging on the building concurrently with the stresses associated with the dead load and 75 psig internal pressure.

With no earthquake or wind acting, the structure is predicted to withstand 90 psig internal pressure without loss of function.

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Under each of these conditions, stresses in the structural members will not exceed 0.95 yield.

The facility is designed so that the plant can be safely shutdown and maintained in a safe shutdown condition during a tornado. Design considerations associated with tornadoes are further explained in Section 5.4.7 of the FSAR.

Flooding of the Fort Calhoun Station is considered highly unlikely. All plant openings into functional areas are at 1007 MSL or higher whereas the 0.1% flood peak stage is 1004.2 MSL. Further information is available in Section 2.7.1.2 of Fort Calhoun Station Unit No. 1 Facility Description and Safety Analysis Report.

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CRITERION 3 - FIRE PROTECTION

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.

This criterion is met. The reactor facility is designed to minimize the probability of such events as fires and explosions and to minimize potential effects of such events to safety. Noncombustible fire resistant materials are used whenever practical throughout the facility. The facility is provided with a fire protection system which includes detectors, alarms, water supply, secondary water supply, deluge systems, sprinklers, hose lines and portable extinguishers. For further information the fire protection system is described in Section 9-11 of the Fort Calhoun Station - Unit No. 1 Final Safety Analysis Report.

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CRITERION 4 - SHARING OF SYSTEMS

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

This criterion is met. The design of Ft. Calhoun Station Unit No. 1 is not based on sharing of systems and components with a future reactor facility.

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CRITERION 5 - RECORDS REQUIREMENT

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.

This criterion is met. The Omaha Public Power District is the owner and operator of the Fort Calhoun nuclear facility. The Omaha Public Power District will maintain records of the design, fabrication, and construction of essential components of Fort Calhoun Station throughout the life of the reactor.

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CRITERION 6 - REACTOR CORE DESIGN

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.

This criterion is met. The following bases are applied in the design of the fuel for all expected conditions of normal operation and for transient situations which can be anticipated:

a) The value of the DNBR will not be less than 1.30 as determined by the W-3 correlation.

b) The peak temperature in the fuel will be less than the melting point of irradiated UO_2.

c) The maximum tensile stress in the zircaloy cladding at normal operating conditions will not exceed two-thirds of the minimum yield strength at the expected local temperature, including the effects of fission gas and of other gas and vapor releases from the UO_2.

d) The predicted net average cladding strain will be less than three-fourths of 1% at the end of life, including the effects of fuel-clad interaction, of fission gas, and of other gas and vapor releases from the UO_2.

The minimum DNB ratio has sufficient margin to ensure a low probability of cladding failure under all anticipated conditions. The occurrence of DNB does not necessarily signify cladding failure, but, rather a local increase in cladding temperature which may or may not cause thermal damage, depending upon its severity and duration.

The average burnup of the initial core loading is 18,549 Mwd/MTU, and the design batch average discharge value is about 27,360 Mwd/MTU when equilibrium conditions are reached. As evidence currently available (1) indicates that UO_2 fuel performance is satisfactory at burnups of 50,000 Mwd/MTU, there is sufficient margin to allow for the ratio of peak-to-average burnup.

The design is adequate to satisfy the design bases in the event of a reactor coolant system depressurization transient at the end of a fuel cycle.

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The thermal margins during normal operation are sufficient to assure that the minimum thermal margins during anticipated transient conditions do not exceed the design bases.

The reactor coolant and control systems provide for this core capability.

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The Nuclear Steam Supply System design provides adequate protection against fuel failure in the event of loss of power to the reactor coolant pumps by providing sufficient rotational inertia to maintain an adequate coastdown period. This assures that the minimum DNBR will not be less than 1.30 even in the event of a total loss of power to all four pumps. (Cf. FSAR Section 14.6 for the results of an analysis of the loss-of-flow accident.)

If power to the reactor coolant pumps is lost, natural circulation of the reactor coolant following pump coastdown will be sufficient to transfer core decay heat to the main steam system without fuel damage. If the main condenser and the feedwater system are still functioning after the loss of the pumps, the energy transferred to the main steam system can be dissipated by bypassing steam to the main condenser. If there is a complete loss of off-site power and the main condenser is not available as a heat sink, steam from the main steam system can be dumped to the atmosphere through the safety valves or the atmospheric dump valve. In this case, the auxiliary feedwater pumps supply water to the shell side of the steam generators from an on-site storage tank. Power for this pump is available from a diesel generator and sufficient stored water is available to bring the core to and maintain it in a safe condition.

In an instance of turbine trip, the reactor is scrammed and energy transfer to the secondary system continues by means of automatic bypassing of steam directly to the main condenser. In addition, relief valves are provided to directly limit reactor coolant system pressure and temperature in the event of isolation of the reactor from its primary heat sink.

1. Preliminary Safety Analysis Report, Appendix 3A, Calvert Cliffs Nuclear Power Plant, Baltimore Gas and Electric Company, AEC Docket No. 50-317.

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CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS

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.

This criterion is met. The reactor core is designed not to have sustained power oscillations. If any power oscillations occur, the control system is sufficient to suppress such oscillations and prevent fuel damage in excess of acceptable limits.

The basic stability of a low enrichment pressurized water reactor with UO_2 fuel is due to the fast acting negative contribution to the power coefficient provided by the Doppler effect. To produce short term power oscillations, a forcing function would have to be imposed. Space-time analyses have been performed to assure that the plant is stable and exhibits a well damped transient behavior.

A modified Randall-St. John method was used to assess the xenon stability of the core. This analysis showed that radial and azimuthal oscillations induced in the core will be damped, but unstable axial oscillations could exist during later stages of the burnup cycle; however, these oscillations can be controlled and suppressed by manipulation of the control rods so that the thermal design bases are not exceeded. Xenon transients are characterized by long periods and slow changes in power distributions. The nuclear instrumentation will provide the information necessary to detect these changes.

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CRITERION 8 - OVERALL POWER COEFFICIENT

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

This criterion is met. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity; i.e., the overall power coefficient is not positive.

The reactivity coefficients for the Ft. Calhoun Station reactor are listed in Table 3.4-1 of the FSAR and are discussed in detail in Section 3.4.2 thereof.

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CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY

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.

This criterion is met. Reactor coolant system components are designed for a pressure of 2500 psia and a temperature of 650°F. The nominal operating conditions of 2100 psia and an average reactor coolant system temperature of 572.5°F permit an adequate margin for normal load changes and operating transients. The components are designed and constructed in accordance with the ASME Boiler & Pressure Vessel Code, Section III, and as delineated in Criterion 1. Reactor coolant loop piping is designed in accordance with ANSI B 31.1 plus nuclear code cases. Other reactor coolant boundary piping is in accordance with the intent of ANSI Draft Code for Nuclear Piping B 31.7 of February 1968. Quality control, inspection, and testing as required by these standards ensure the integrity of the reactor coolant system and is described in Appendix A & Section 4.5 of the FSAR.

In addition to the code requirements listed, the reactor coolant loop piping is designed to meet the cyclic loading requirements and transient conditions stated for the reactor pressure vessel in Section 4.2.2 of the FSAR. This piping is designed to withstand the dynamic seismic loadings for Class I structures under the rules listed in Section 2.0, Appendix F, of the FSAR.

Also, cf. Criteria 33 to 36.

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CRITERION 10 - CONTAINMENT

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.

This criterion is met. The containment structure is designed to sustain the initial effects of gross equipment failures, such as a large reactor coolant boundary break, without loss of required integrity and, together with other engineered safety features as necessary, to retain for as long as necessary, the functional capability to protect the public.

The containment building is designed to withstand an internal pressure of 60 psig at 305°F including all thermal loads resulting from the temperature associated with this pressure with a leakage rate of 0.2% or less of the contained volume per 24 hours and will be subject to a leak rate and pressure test to demonstrate compliance with the design.

Engineered safeguards in the form of the safety injection system; the containment spray system; and the containment air recirculation, particulate filtering, iodine filtering and cooling system are provided and are designed in accordance with the latest applicable codes. Sufficient redundancy exists in these separate systems to assure the availability of containment cooling in the event of a reactor coolant system rupture.

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CRITERION 11 - CONTROL ROOM

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 shutdown and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it is a safe condition if access to the control room is lost due to fire or other cause.

This criterion is met. The Fort Calhoun facility is equipped with a control room located in the reactor auxiliary building which is a Class I structure. The control room contains all controls, instrumentation displays and alarms essential for continuous operation under normal and accident conditions to safely start up, operate and shutdown the plant.

The control room is provided with an air conditioning system which maintains the control room under a slight positive pressure at all times and which can be isolated and operated on a recirculation basis. The control room air system is constructed such that following an accident which causes a release of radioactive material outside the building, the fresh air makeup to the control room can be manually re-routed through a high efficiency particulate filter and an activated charcoal carbon filter which will remove nearly all particulate matter and halogen vapors. In the event of a loss of coolant accident, the air conditioning system goes into the recirculation mode and the filters are placed in service automatically.

The control room air conditioning system and the control room shielding are designed to permit the necessary access and occupancy of the control room to allow termination of accidents and resulting consequences.

The whole body dose due to fission products in the containment atmosphere following the postulated MHA would not exceed 2.5 Rems; this dose is based upon a continuous occupancy for 30 days following the accident. The dose due to gaseous fission products which could enter the control room after a MHA is discussed in FSAR Section 14.21.3-3.

If for any reason the control room is evacuated without first initiating a reactor shutdown, a safe shutdown can be initiated and the shutdown condition monitored from various equipment within the plant.

Reactor shutdown can be initiated from the following locations among others:

1. The turbine can be tripped from the trip device on the turbine front standard, this will cause a reactor trip.

2. Reactor clutch power supplies can be de-energized from the electrical equipment load center.

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Continuation of a safe shutdown condition can be achieved with removal of heat from the steam generators by the turbine dump and bypass valves or safety valves and operation of a feedwater pump(s) to maintain level in the steam generators. These operations can be initiated, controlled and monitored external to the control room.

Reactor coolant system boron concentration can be monitored at the sample station to verify adequate shutdown margin is maintained. Boron can be added as required, using the charging pumps, from the concentrated boric acid storage tanks. These operations can be initiated, controlled and monitored external to the control room.

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CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS

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

This criterion is met. Instrumentation is provided for continuous measurement of all significant process variables. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation. The instrumentation conforms to applicable IEEE standards.

The principal process variables monitored include neutron level (reactor power); reactor coolant temperature, flow, and pressure; pressurizer liquid level; and steam generator level. In addition, instrumentation is provided for continuous automatic monitoring of radiation level.

The instrumentation and control systems are described in detail in Section 7 of the PSAR.

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CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS

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.

This criterion is met. The plant is provided with means to monitor and maintain control over the fission process throughout core life.

The following are provided to monitor and maintain control over the fission process during both transient and steady state periods over the lifetime of the core:

a) Ten independent channels of nuclear instrumentation, which constitute the primary monitor of the fission process. Of these channels, four are used to monitor the reactor during startup and power range operation; four monitor the reactor in the power range and are used to initiate a reactor shutdown in the event of overpower; and two are used in the automatic control system to regulate the reactor in response to turbine demand.

b) Two independent rod position indicating systems.

c) In-core instrumentation for determination of core power distribution.

d) A boronometer, which determines the neutron absorption by the coolant, is provided as a backup to determination of soluble poison concentration by sampling and analysis of reactor coolant water.

e) Manual and automatic control of reactor power by means of control rods.

f) Manual regulation of coolant boron concentrations.

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CRITERION 14 - CORE PROTECTION SYSTEMS

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.

This criterion is met. The reactor is protected by the reactor protection system from reaching a condition at which fuel damage might occur. The protection system is designed to monitor the reactor operating conditions and initiate a fast shutdown if any of measured variables exceed the operating limits.

The signals which will provide automatic reactor trip are identified in Table 7.2.-1 of the Fort Calhoun FSAR. The parameters and conditions which will initiate a trip are the following:

a) High Neutron Level (reactor power) (ΔT power is a backup)

b) High Startup Rate (low power level only)

c) High Pressurizer Pressure

d) Thermal Margin/Low Pressure

e) Loss of Load

f) Low Steam Generator Pressure

g) Low Reactor Coolant Flow

h) Low steam Generator Liquid Level

i) Containment Building High Pressure

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CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEM

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

This criterion is met. Protection systems are provided for sensing accident conditions which result in a loss of reactor coolant system pressure, pressurization of the reactor containment building, and abnormally high radiation levels in the reactor containment building.

Operation of the safety injection system is initiated by either low reactor coolant system pressure or high containment pressure.

Operation of the containment spray system is initiated by coincident low reactor coolant system pressure and high containment pressure. Containment air recirculation and cooling units not already in operation are started by either loss of reactor coolant system pressure or pressurization of the reactor containment building. Complete isolation of the containment building is initiated by high containment pressure, or low reactor coolant system pressure. Isolation of containment building vents only is initiated by high containment radiation.

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CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.
This criterion is met. The reactor coolant pressure boundary is monitored by the following means for detecting leakage of reactor coolant:

a) Containment Building Radiation Level - A gas monitor and a filter paper airborne particle monitor are arranged with a vacuum pump for continuous sampling of the containment building atmosphere. The particulate and gas monitor is sufficiently sensitive to detect small quantities of leaking coolant at a fraction of the design value for fuel assembly clad failures.

b) Condenser Offgas - A gas monitor is provided to detect any radioactive noble gases in the air ejector discharge. Presence of such gases at that point indicates the possibility of steam generator reactor coolant to secondary system leakage.

c) Steam Generator Blowdown Water - The blowdown sampling stream is monitored continuously in each S.G. blowdown sample line. A sudden increase in blowdown gamma activity indicates the possibility of a steam generator reactor coolant to secondary system leak.

d) Containment Humidity and Temperature - The humidity and temperature of the air in the containment are continuously monitored. An increase in the readings of these monitors could be an indication of leakage from the reactor coolant pressure boundary.

e) Containment Sump Level - Reactor coolant leakage reaching the containment building sump would be annunciated in the control room by activation of the sump high level alarm.

f) Volume Control Tank (VCT) Level - Loss of inventory from the Reactor Coolant System would be detected by level changes in the VCT.

The control room operator would be alarmed to the existence of larger leaks by low pressurizer level, closing of letdown orifice isolation valves, and continued operation of standby charging pumps.

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CRITERION 17 - MONITORING RADIOACTIVITY RELEASES

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.

This criterion is met. The containment atmosphere is monitored for radioactivity by a gas monitor and a filter paper type airborne particle monitor. Both monitors are arranged in a loop with a vacuum pump to sample continuously the containment building atmosphere. The particulate monitor is upstream of the gas monitor serving as a filter to minimize contamination of the gas monitor.

Plant gaseous effluents are vented to the atmosphere through the ventilation discharge duct. A gas monitor is installed in the duct in order to view the largest available gas volume and thus attain required sensitivity. Particulate monitoring of gaseous effluent is accomplished by 2 fixed filter air samplers located at ground level. A gas monitor similar to that provided in the containment building is provided to detect any radioactive noble gases in the air ejector discharge.

Plant liquid effluents are monitored just upstream of the point where plant effluent is discharged to the condenser cooling water discharge for dilution to ensure adequate sensitivity.

The above monitors are capable of detecting radioactivity released from normal operations, from anticipated transients and from accident conditions.

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CRITERION 18 - MONITORING FUEL AND WASTE STORAGE

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.

This criterion is met. Continuous decay heat removal is assured by monitoring of the water temperature in the spent fuel pool. Any deviation will be alarmed in the control room. Area monitoring of dose rates is supplied in the fuel and waste storage areas. A panel in the control room contains an indicator and alarm for each channel, plus power supplies and a multipoint recorder. Local alarms and indicators are provided at each monitor.

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CRITERION 19 - PROTECTION SYSTEMS RELIABILITY

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

This criterion is met. Design of protection systems includes specification of high quality components, ample design capacity, component redundancy, and in-service testability. The following principal design criteria have been applied:

a) No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.

b) No single component failure shall initiate unnecessary protection system action provided implementation does not conflict with the criterion above.

Testing facilities are built into the protection systems to provide for:

a) Preoperational testing to give assurance that the protection systems can fulfill their required functions.

b) In-service checking of protective channels from the process sensor to the channel trip unit (bistable).

c) In-service testing of the channel trip units (bistables) and associated coincidence logic and the outputs of that logic through to the final actuator.

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CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE

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.

This criterion is met. Two channels are available to monitor each critical parameter. Protection system logic is designed such that failure or removal from service of any one of the channels does not result in loss of the protection function.

Channel independence is maintained by independent power supplies, primary sensors, and bistable trip units. Physical separation of individual channel components and wiring is maintained wherever practicable.

Where necessary, different principles are used to achieve true independence of redundant instrumentation components. In many instances, an abnormal condition will cause tripping of more than one protective system due to inherent overlapping of protective functions. For example, while overpower or inadvertent reactivity addition will normally be terminated by the high power trip initiated by neutron level reading, such a transient may also be terminated by the low thermal margin/pressure trip, since both overpower and decreasing thermal margin are indicative of a limiting situation.

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CRITERION 21 - SINGLE FAILURE DEFINITION

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

This criterion is met. The design of the Fort Calhoun Station is based on the concept that no single failure of active components will inhibit necessary safety action when required. The concept is interpreted that multiple failures resulting from a single event are considered as a single failure. Redundancy of components is provided in the instrumentation and controls associated with vital process systems, the containment isolation system and the engineered safeguards.

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CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS

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

This criterion is met. The protection systems are separated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protection system.

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CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS

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.

This criterion is met. The protection systems are designed to provide the maximum practical degree of redundancy of channels and power sources. Physical separation of the multiple detectors and signal leads for these systems minimizes the vulnerability to a single accident of an individual protective system. Any accident or adverse condition which causes a loss of power to a protective system will result in automatic reactor shutdown.

The protective systems are designed to operate in the environment of normal plant operation and to terminate any transient condition before a measured variable reaches a level where the reactor or the respective instrumentation is damaged. Therefore, the equipment is not normally subject to adverse conditions which would result in failure. Because of the redundancy provided, loss of a single channel will not result in a loss of protection from accident(s) which is afforded by that protective system.

Components located inside the containment building which are required to operate following a DBA will withstand the post DBA environment for the required time period.

A major fire in the control room could result in extensive damage to instrumentation, controls, and protective circuitry and would almost certainly cause an automatic reactor shutdown. In the event that an automatic shutdown does not occur and the operator does not manually trip the reactor, the reactor can be safely shutdown from outside the control room.

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CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS

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.

This criterion is met. Emergency power is available from two completely independent Diesel generator sets and from the two completely independent 125v dc systems for essential dc loads.

The independent diesel generator supply systems are located in the plant and are connected to separate buses. Both generator sets are independently automatic starting upon loss of auxiliary power and will be ready to accept load within 10 seconds of loss of normal supply power. Starting power is self-contained within each unit. Each unit has sufficient capacity to start sequentially the loads that must be supplied for the engineered safeguards equipment for the hypothetical accident concurrent with loss of outside power. This capacity is adequate to provide a safe and orderly plant shutdown and maintain the plant in a safe condition.

Each of the two 125v dc batteries is capable of supplying essential station dc load for 8 hours and may be charged by the generator power supply.

Facilities are included to permit periodic starting and running the Diesel generator sets without interrupting plant operation. Diesel units are synchronized to the bus and loaded periodically to ensure readiness for emergency services.

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CRITERION 25 - DEMONSTRATION OF FUNCTION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEM

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.

This criterion is met. Protection systems, from the sensors up to the final protection element, will be capable of being checked during reactor operation, as follows:

a) Measurement channels used in protection systems will be checked by observing outputs of similar channels and cross checking with related measurements which are presented on indicators and recorders on the control board.

b) Trip units and logic will be tested by inserting a signal into the measurement channel ahead of the readout and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays.

c) The logic output relays will be tested individually for initiation of trip action.

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CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN

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.

This criterion is met. Protection systems are designed to fail into a safe state in the event of loss of power supply or disconnection of the system. Sufficient redundancy, channel independence, and physical separation are incorporated in the protective system design to preclude the possibility of the loss of a protection function under adverse environmental conditions.

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CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL

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

This criterion is met. The reactivity control system employs two separate methods of adjusting reactivity, viz., (1) mechanically driven control element assemblies and (2) adjustment of the concentration of boric acid in the reactor coolant.

The CEA system controls short term reactivity changes such as power changes and power distribution shaping, and is also used for rapid shutdown for reactor protection. The boric acid shim control compensates for long term reactivity changes such as those associated with fuel burnup, variation in the xenon and samarium concentrations, and plant cooldown and heatup.

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CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY

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.

This criterion is met. Two independent systems are provided for controlling reactivity by the addition or removal of poison from the core, viz.,the mechanically driven control element assemblies and the variation, by feed and bleed, of the concentration of dissolved boric acid in the reactor coolant.

Either system acting independently is capable of making the core subcritical for a hot operating condition and holding it subcritical in the hot standby condition; in this context, hot standby implies a reactor coolant temperature not less than 515°F.

With the shutdown margin available in the control element assemblies, a temperature reduction of at least 100°F from the hot standby condition can be sustained by the inserted control element assemblies (with the most reactive rod stuck) before boron injection is necessary to prevent any return to criticality.

It is also a requirement that either system be able to insert negative reactivity at a sufficiently fast rate to prevent exceeding acceptable fuel damage limits as the result of a power change. The two transients in this category which will require the maximum rate of negative reactivity insertion are normal plant cooldown and xenon burnup at power. For normal cooldown following a change from full power to hot standby condition (75°F/hr) and the maximum rate of xenon decay, the maximum rate of reactivity addition is 1.7 percent/day. This rate is achieved only when the temperature coefficient of reactivity is at its most negative value (end of core life). The second transient is the positive reactivity added by burnup of xenon following the increase from hot standby to full power, with maximum xenon concentration in the core at the time the reactor is being brought to full power. Calculations show that the maximum reactivity rate produced by this transient is 1.1 percent/hr. Any one of the three charging pumps can inject boric acid solution at a sufficient rate to decrease reactivity by 9.5 percent/hr (cf. Section 9.2.2.4 and Table 3.4-1 of the FSAR) and is hence able to control either of these transients.

The control element assemblies are inherently capable of inserting negative reactivity at a rate greater than that achieved by the charging pumps and the concentrated boric acid solution.

Cf. Section 9.2.2.4 of the FSAR for further discussion of reactivity control after shutdown.

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CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY

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.

This criterion is met. The mechanical control system is capable of making the core subcritical under any condition, including anticipated operational transients, sufficiently fast to prevent fuel damage in excess of acceptable limits. This control system is designed to provide a minimum shutdown margin of 2.4% $\Delta\rho$, assuming all control element assemblies, except the one of highest worth, are inserted in the core.

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CRITERION 30 - REACTIVITY HOLD DOWN CAPABILITY

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

This criterion is met. The chemical control system changes reactivity slowly, but its range of reactivity worth is very large. It can handle the total excess reactivity plus a large shutdown margin (greater than 4% with all control rods withdrawn). It can also shut the reactor down with appropriate margins for contingencies from any normal operating condition without the use of control rods.

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CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION

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.

This criterion is met. Limits have been placed on the maximum rate at which reactivity can be increased by unplanned continuous withdrawal of control element assemblies (CEA's). The number of CEA's in the core, the assignment of CEA's into operating groups and the design rate of withdrawal were established to assure fuel integrity in the event of uncontrolled CEA withdrawal.

While an inadvertent withdrawal of CEA's is considered unlikely, the reactor protective system is designed to terminate any such transient with an adequate margin to DNB. The analysis which supports this is described in Section 14.2 of the FSAR. This analysis shows that sufficient protection is provided by the high power level trip, the high pressurizer pressure trip, the thermal margin trip and the steam generator water level trip to prevent the minimum DNB ratio from falling below 1.3 in the event of continuous withdrawal of CEA's.

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CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods of 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.

This criterion is met. Limits have been placed on the maximum reactivity worth of individual control element assemblies (CEA's) so that the rate at which reactivity can be increased by CEA ejection, for example, will not cause an unacceptable rupture of the coolant pressure boundary or disrupt the core or reactor internals sufficiently to impair the effectiveness of emergency core cooling. The number and extent of insertion of CEA's in the core was selected to assure that the maximum reactivity worth of a single CEA is within a preselected safe limit.

To confirm this, an analysis was made for the assumption that a CEA is ejected instantaneously from the core. The analysis, which is described in Section 14.13 of the FSAR, shows that the energy increase at the core hot spot is limited to the extent that no fuel rods suffer significant damage following CEA ejection from full or zero power at the beginning or end of cycle. Also, it has been calculated that the pressure surge associated with this excursion will not rupture the reactor coolant boundary.

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CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY

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.

This criterion is met. The reactor coolant pressure boundary is designed to be within the minimum yield stress for all static and dynamic loads associated with any inadvertent and sudden release of energy to the coolant.

The safety analyses in FSAR Sections 14.2 through 14.12 describe the core and coolant boundary protection that is provided for energy release due to sudden or uncontrolled reactivity increases. The CEA ejection accident is analyzed in Section 14.13, and the increase in reactivity due to the addition of cold water in Section 14.7 & 14.12 of the FSAR. The results of both analyses indicate that the sudden energy release associated with these accidents will neither rupture the reactor coolant pressure boundary nor result in more than limited plastic deformations.

Rod dropout is not possible in view of the Ft. Calhoun Station reactor core design.

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CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating 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) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

This criterion is met. Carbon and low alloy steel materials which form part of the pressure boundary meet the requirements of the ASME Code, Section III, paragraph N-330 at a temperature of +40°F. The actual nil-ductility transition (NDT) temperature of the materials were determined by drop weight tests in accordance with ASTM-E-208. For the reactor vessel, Charpy tests were also performed and the results were used to plot a Charpy transition curve. The NDT temperatures as determined by drop weight test were used to correlate the Charpy transition curve and establish base points for the surveillance program. See Criterion No. 36 CF. FSAR Section 4.42 and 4.5.3.

The combined static and transient stresses will be limited, whenever the temperature is below NDT +60°F, to sufficiently low values to make the probability of a rapidly propagating failure extremely remote. The required stress limits will be maintained by operating restrictions as noted in the following.

All the reactor coolant pressure boundary components are constructed in accordance with the applicable codes noted in Criterion 1 and comply with the test and inspection requirements of these codes. These test inspection requirements will assure that flaw sizes will be limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding code requirements are performed. The tests and inspections performed on the reactor vessel are summarized in FSAR Section 4.5.

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Excessive embrittlement of the reactor vessel material due to neutron radiation is prevented by providing an annulus of coolant water between the reactor core and the vessel. The thermal shield, which is located in this annulus, contributes toward reducing vessel embrittlement. The expected maximum integrated fast neutron flux exposure of the reactor pressure vessel wall opposite the midplane of the core is less than 2.0×10^{19} nvt. This value applies to a 40-year vessel design life, with the plant at full power 80% of the time, and corresponds to a maximum increase in transition temperature of 217°F. A surveillance program will be conducted (see Criterion 36) to allow monitoring of the NDT temperature shift of the vessel material during its lifetime. Based on the determined NDT temperature, operating restrictions to limit vessel stresses would be applied as necessary. The reactor coolant system pressure is not increased to more than 500 psia below a temperature corresponding to NDTT +60°F. Vessel stresses resulting from pressurizing to only 20% of design pressure are sufficiently low to preclude brittle fracture. Cf. PSAR Section IV-2.1.2.

The Technical Specifications establishes limits for coolant system pressure and allowable stresses based on the NDTT and the system temperature which are sufficiently low to preclude brittle fracture.

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CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION

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 (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

This criterion is met. During normal startup for power operation, the reactor will not be made critical until the reactor coolant system temperature is 120°F higher than the predicted nil ductility transition temperature based on plant records of fast neutron dose to the vessel. The operational restrictions that will be invoked will maintain the temperature above NDT + 120°F for reactor operation. This will assure that a reactivity-induced loading which would contribute to elastic or plastic deformation cannot occur below NDT + 120°F in this plant. (Cf. FSAR Section 4.4.2)

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CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE

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 ASTM-E-185-66 shall be provided.

This criterion is met. The location of the more highly stressed portions of the Nuclear Steam Supply System components have been identified, and these areas are made accessible and equipped with removable insulation to allow for visual and appropriate nondestructive examination. The design of the reactor vessel cavity will permit examination of the outside of the vessel. The outside surfaces of the other major components, such as the steam generators, pressurizer, reactor coolant pumps, and piping, are readily accessible and can be examined by various means after stripping off insulation.

Sample pieces taken from the shell plate material of which the reactor vessel is fabricated, installed between the core and the vessel inside wall, will be removed and tested at intervals over the course of vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests will be performed on the samples to develop a Charpy transition curve. Comparison of this curve with the Charpy curve and drop weight tests for specimens taken at the beginning of the vessel life will permit determination of the change in NDT temperature, and operating instructions will be adjusted as required. This material surveillance program conforms with ASTM-E-185-66.

The entire reactor coolant pressure boundary can be hydrostatically tested.

Cf. FSAR Section 4.5.3 for a detailed discussion of the surveillance program.

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CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN

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.

This criterion is met. The engineered safety features of the system comprise the safety injection systems, the containment spray system and the containment air recirculation and cooling system, which are designed to cope with any size reactor coolant pressure boundary breakup to and including the circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.

The engineered safeguards consists of high and low pressure safety injection pumps, stored energy tanks (water accumulators), containment sprays, and containment air recirculation (fan) cooling. The systems are discussed in detail in Section 6 of the FSAR; some specific aspects of their design, operation and testability are described in the answers to Criteria 38 to 48 and 58 to 61 and their effectiveness is discussed in the analysis of the loss-of-coolant accident in Section 14.15 of the FSAR.

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CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES

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 systems, 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.

This criterion is met. The design of the systems and their components as well as their basic mode of operation is based on proven engineering principles, which ensures performance capability and reliability. Basic components of the type used in the design have demonstrated their performance capability and reliability in existing power plants both nuclear and conventional. All systems were tested for performance capability prior to plant operation and provisions are made for testing components and systems during the life of the plant.

The redundancy principle adhered to in the design of these systems ensures that failure of any single active component in any engineered safeguards system will not prevent that system from fulfilling its requirement.

The engineered safeguards systems are discussed in detail in Section 6 of the FSAR.

The inspection and testing of the engineered safeguards systems are covered in Criteria 45 through 48 and Criteria 58 through 65.

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CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES

Alternate power systems 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.

This criterion is met. Offsite power to the plant is available via the 161 kV line from Blair, and after the unit is tripped, via backfeed from the 345 kV system through the main and unit auxiliary transformers.

When the unit is tripped and the 161 kV supply is not available, the motor-operated disconnect switch in the generator main leads is opened and the supply to the unit auxiliary transformers is re-established. Switch operation is accomplished by a motor operator supplied from the station battery.

Onsite power is provided by two diesel generator sets. Each independent diesel generator set is adequate for supplying the minimum engineered safeguards equipment for the hypothetical accident concurrent with loss of outside power.

Station batteries provide onsite power for instrument and control systems. These batteries will be subject to rigorous inspection and maintenance. The charger voltage will periodically be manually lowered to test batteries capability to assume load at the appropriate bus voltage.

The diesel generator facilities permit periodic starting and running during normal plant operations.

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CRITERION 40 - MISSILE PROTECTION

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

This criterion is met. The high-pressure equipment in the reactor coolant system is surrounded by reinforced concrete and steel structures designed to stop all credible missiles and withstand the forces generated in a loss-of-coolant accident for break sizes up to and including the double-ended rupture of the reactor coolant pipe. The containment liner, the reactor coolant loops, the steam and feedwater piping, the auxiliary cooling piping and the containment cooling system are protected from missiles generated within the containment building. Barriers are provided where the use of radiation shielding and/or support structures for missile shielding is not feasible.

Two of the containment air recirculation and cooling units are located on the operating floor, and two on a concrete platform above the first pair. They are protected from missiles by the walls of the reactor coolant equipment compartments and by the missile shield placed over the reactor. Auxiliary coolant enters the air handling unit from below the operating floor, so that it is remote from any missiles.

The most critical plant missile external to the containment building has been determined to be a turbine last stage wheel fragment. Analyses show this missile will not perforate or impair the structural integrity of the containment building.

The emergency core cooling system will be designed to prevent loss of design capability during the emergency of a pipe rupture or earthquake. Piping connecting vessels will be engineered to restrict movement to certain maximum values during these emergencies. The piping system will be designed to accept these emergency imposed movements and still remain within code allowable limits for stress. Flexibility calculations will be according to the Code for Nuclear Piping USASI B31.7.

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CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY

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.

This criterion is met. The emergency core cooling systems and the containment heat removal systems are designed to ensure that failure of a single active component in each subsystem does not prevent meeting the required safety function of the individual subsystems.

a) Containment Heat Removal Systems - There are two independent systems, i.e., the containment spray system, which utilizes three pumps, and the containment air recirculation system, which utilizes four coolers. Failure of a single active component in either system will be compensated for by the other system. The two systems operating in parallel have sufficient installed capacity to compensate for the simultaneous loss of one active component in each system.

b) Safety Injection Systems (Emergency Core Cooling) - The high pressure safety injection system consists of three full-capacity pumps; failure of one pump will not limit the performance of the system.

The low pressure safety injection pumps and the stored energy tanks are designed for rapid flooding of the reactor core after blowdown. Failure of one pump to function will not impair the effectiveness of low pressure injection, since two full capacity pumps are provided. Also, two of the four stored energy tanks will cover the core hot spot and three will cover the core in the absence of pump action. Cf. Section 6 of the FSAR for the operation and design of these engineered safety systems.

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CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY

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.

This criterion is met. The engineered safety features consist of three individual and separate systems, as follows:

The Containment Spray System
The Safety Injection System
The Containment Air Recirculation and Cooling System

The major components of the containment spray system and the safety injection system, including the pumps and the shutdown heat exchangers, are located outside the containment. The spray nozzles, the spray and injection piping and the safety injection valves located inside the containment are designed for operation in the environment produced by a major loss-of-coolant accident. Safety injection piping is designed to accept any reactor vessel motion resulting from forces generated by a loss-of-coolant accident.

The containment air recirculation and cooling system components are located entirely within the containment. All components necessary for cooling and iodine filtration after the design basis accident are designed to withstand the accident and the environmental conditions following the accident. The heat transfer coil configuration and heat transfer capability have been tested by the manufacturer on a reduced version of the coil under post accident conditions. Motor control devices will be outside of the containment. A high temperature lubrication system and a high temperature, encapsulated insulation system for the motors has been specified. The means for providing reliable supplemental cooling, for control of moisture content of the cooling air, and for avoiding or neutralizing large air pressure differences across bearings has been specified. Frequent and careful preventative maintenance and periodic tests, particularly for insulation resistance, will be made.

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CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION

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.

This criterion is met. The engineered safeguards consists of three individual systems.

1. The containment spray system

2. The safety injection system

3. The containment air recirculation and cooling system

The actuation of any of these systems would in no way accentuate the adverse after-effect of the loss of normal cooling.

The after-effects of the containment spray system will be avoided by insulating equipment subject to thermal shock, electrical equipment and instrumentation will be selected to withstand direct spray and cables will be moisture resistant.

The safety injection system will introduce highly borated water into the reactor coolant at a pressure appreciably below operating pressure and will therefore cause no adverse level, pressure or reactivity effects. The thermal shock effects have been analyzed.

(Cf. Section 14.15.2.6 and Section 1.5.4)

A portion of the containment air recirculating and cooling system is operating at all times and will in no way cause adverse after-effects.

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CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY

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 loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

This criterion is met. The emergency core cooling systems are covered in detail in Section 6 of the FSAR.

There are three systems; the high pressure injection system, which uses pumps; the low pressure injection system, which also uses pumps, and the stored energy tanks, or accumulators. These systems are designed to meet the criteria stated above in reference to the prevention of fuel and clad damage (cf. Section 6.2.0 of the FSAR). The systems do not share active components other than the valves controlling the suction headers of the high and low pressure safety injection pumps. These valves are in no way associated with the function of the stored energy tanks. The partial loss of capacity due to failure of one of these valves to function automatically will not compromise system effectiveness.

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CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS

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

This criterion is met. Section 6.2.3 of the FSAR describes the arrangement and location of the main components in the emergency core cooling system. All pumps, the shutdown heat exchangers, and valves and piping external to the containment building are accessible for physical inspection at any time. All safety injection valves and piping inside the containment building, and the stored energy tanks, can be inspected whenever access to the containment building is possible.

The reactor vessel internals, reactor coolant piping and items such as the water injection nozzles are accessible for physical inspection.

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CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS

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.

This criterion is met. All active components in the emergency core cooling systems can be tested periodically for operability and functional performance. The high pressure and low pressure safety injection pumps and the motor operated valves can be tested, with the reactor at operating pressure, by flow recirculation. Two recirculation paths are available. One path, through the pump recirculation lines, verifies the functional performance of the pumps. The second path, through the leakage coolers, verifies flow path continuity and provides a check on the operability of the motor operated valves and all check valves except the final check valve in each safety injection header.

During normal plant cooldown, the low pressure pumps function as shutdown cooling pumps and feed the reactor coolant system via the safety injection headers. This operation verifies that the final check valve in each header is operable.

Operation of the check valve in the stored energy tank discharge line can be verified by bleeding through the leakage cooler and the recirculation line. Cf. Section 6.2.7 of the FSAR.

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CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS

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 practical.

This criterion is met. The emergency core cooling systems are designed to permit periodic testing of their delivery capability up to the reactor coolant pipe.

The low pressure safety injection pumps will be used as shutdown cooling pumps during normal plant cooldown. The pumps will discharge into the safety injection header via the shutdown heat exchangers and the low pressure injection lines.

With the plant at operating pressure, the high pressure safety injection pumps and valves may be operated so as to discharge through the leakage coolers back to the SIRW tank. This will verify flow path continuity in all high pressure injection lines.

Borated water from the stored energy tanks may be bled through the drain heat exchanger and the recirculation line to verify flow path continuity from each tank to its associated main safety injection header.

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CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS

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.

This criterion is met. Testing of the safety injection system active components, including pumps and valves, will be initiated by simulating the initiating signals for the safeguards and will utilize the control circuit components to the maximum practicable extent. System operability is demonstrated without injection of borated water into the reactor coolant system. (During testing of the injection valves, the pump motor control circuit is bypassed). The test method will preclude the possibility of disabling the safety injection initiating circuit during the test. (Cf. Section 7.3 of the FSAR).

Automatic starting of the emergency diesel generators will be tested periodically, as will the switchgear utilized in load shedding and sequential pickup of emergency loads. (Cf. Section 7.3.4.2 of the FSAR for further details).

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CRITERION 49 - CONTAINMENT DESIGN BASIS

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 loss-of-coolant accident, 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.

This criterion is met. The design of the containment structure, including access openings and penetrations, and the necessary heat removal systems is stated in Criterion No. 2 and No. 10 and in greater detail in the FSAR. The effects from metal-water or other chemical reactions including hydrogen burning have been investigated and determined to cause temperatures and pressures less than the design temperatures and pressure even with the failure of emergency core cooling.

The design pressure is 60 psig and temperature is 305°F. The maximum pressure and temperature from a loss-of-coolant accident is 59.3 psig and 285°F. These figures include the effects of the zirconium-water and other chemical reactions and take no credit for the safety injection system. A margin of 0.7 psig and 20°F is provided above the maximum accident conditions and design (cf Section 14.16).

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CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperature under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

This criterion is met. All ferritic components projecting outside of the containment are located within the enclosure provided by the auxiliary building, and the lowest service metal temperature is 50°F. The lowest service metal temperature within the containment is also 50°F. The metal used for these components conforms to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, which requires that the Charpy V-specimens be tested at least 30°F lower than the lowest service metal temperature in accordance with the requirements of Paragraphs N-331, N-333, N-511.2 and N-515.

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CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT

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.

This criterion is met. The only instance of a reactor coolant pressure boundary outside the containment is a single 3/8" diameter sampling line. This line is protected by a pair of remotely operated isolation valves, one inside and one outside of the containment. Each of these valves can be operated from the control room. In all other instances where reactor coolant leaves the containment, pressure and temperature are reduced substantially below operating pressure and temperature prior to entering the reactor auxiliary systems outside the containment.

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CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS

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.

This criterion is met. Two fully independent cooling systems are supplied to provide cooling of the containment building atmosphere following the MHA. Each system utilizes a different operating principle. The containment spray system reduces the temperature of the containment atmosphere by direct contact of the cool spray with the hotter containment atmosphere. The containment air recirculation and cooling system cools the containment atmosphere by recirculation of the hot gases through water cooled surface coolers. Each of these systems operating by itself at full effectiveness is sufficient to prevent containment building pressure following the MHA from exceeding design.

At the beginning of operation, the containment spray pumps take suction from the safety injection and refueling water tank. When this supply is depleted, pump suction is shifted to draw water from the containment building sump.

The containment air recirculation and cooling system consists of four recirculation and cooling units. One, two or three units operate continuously during reactor operation. The other unit or units are automatically started on initiation of the safety injection system. The coolers have two sources of cooling water.

In the event of loss of outside power, equipment in the containment spray and the containment air recirculation and cooling systems which is required for containment cooling will receive power from the emergency diesel generators.

Both systems are considered highly reliable due to the inclusion of multiple components. In addition to system redundancy, the containment spray pumps and shutdown heat exchangers are located in separate rooms outside the containment building; consequently, it is possible to perform maintenance on this equipment during long-term operation.

ARCHIVED TEXT

CRITERION 53 - CONTAINMENT ISOLATION VALVES

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

This criterion is met. Piping penetrating the containment is provided with valves as necessary to assure no unrestricted release of radioactive materials can occur. There are six basic classes of penetrations:

1. Direct Communicating System - two automatic control isolation valves are provided for penetrations by this system.

2. Reactor Coolant - Exposed System - two automatic control isolation valves are provided for penetrations by this system, one of which is located inside the containment.

3. Containment Atmosphere - Exposed System - (System inlet is maintained above 60 psig at all times) - one automatic control isolation valve is provided for penetrations by this system.

4. Containment Atmosphere - Exposed System - (System inlet not maintained above 60 psig) - two automatic control isolation valves are provided for penetrations by these systems. If system outside containment is not closed, one isolation valve will be inside the containment.

5. Closed System Inside Containment - one automatic closing valve is provided outside of the containment for penetrations by these systems.

6. Closed System Outside Containment - two automatic control isolation valves are provided for penetrations by these systems. If operating pressure of the system is below 60 psig one valve will be located inside containment.

7. Containment Pressure Switch Lines - these penetrations must remain open to initiate containment spray. They are protected by a remote manual isolation valve to permit isolation should an accident happen to this piping following the DBA.

All isolation valves, excluding those which are part of engineered safeguard systems and must remain open, are automatically closed. The position of each valve is displayed in the control room. An operator manually closes all valves external to the containment in the unlikely event a valve did not close.

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CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING

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.

This criterion is met. The containment will be tested at the design pressure of 60 psig during construction. All penetrations will be in place during this test. The test will be conducted after all systems are installed and individually tested so that performance of the containment valves can be included in the containment leakage test. The test will be conducted in accordance with Standard ANS-7.60 and will extend at least forty-eight hours. If leakage rate is greater than the design rate of 0.2% of the containment volume per day, corrective action will be taken and the test repeated. This procedure will be followed until a successful design pressure leak rate test has been demonstrated.

The containment design leak rate is 0.2 percent of the contained volume in 24 hours at 60 psig. With a containment leak rate of 0.2 percent per day and one of two iodine filter equipped fan units operating the off-site potential iodine inhalation thyroid exposure to the public will be a factor of 3.3 below 10 CFR 100 limits for the hypothetical loss-of-coolant accident with no credit for the Safety Injection System in limiting core meltdown. (Cf. FSAR, Section 3.4.8)

The personnel access lock is provided with interlocked double doors so containment integrity is maintained at all times. Access hatches are sealed in place, using flexible double seals or gaskets to assure tightness. The fuel transfer tube penetration is sealed by an isolation valve in the spent fuel storage pit and a blind flange on the containment end.

ARCHIVED TEXT

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING

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

This criterion is met. Post-operational leakage testing of the entire containment structure may be performed at the containment design pressure of 60 psig, during a plant shutdown, such as refueling periods. Leakage rate tests at design pressure will require venting of the quench tank and primary system drain tank to the containment atmosphere to avoid possible damage to the tanks. Instruments that might be damaged by full containment design pressure will be removed during tests.

Normally post-operational containment leakage testing will be conducted at a reduced pressure and results extrapolated to a leakage rate at design pressure. After the successful completion of the initial containment leakage rate test at design pressure (60 psig) the test will be repeated at 30 psig without further modification. If scatter of the test data makes it necessary, in order to establish the relationship between the leakage rates at the two test pressures, the test will be repeated at intermediate pressures.

Subsequent leak rate tests will be conducted at 30 psig in accordance with a schedule based on measured leak rates. The allowable leakage rate during these tests will be established from the relationship between the leak rates of the initial 30 psig and 60 psig tests.

ARCHIVED TEXT

CRITERION 56 - PROVISION FOR TESTING PENETRATIONS

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

This criterion is met. Provisions are made to individually leak test all penetrations in the containment building at any time throughout the operating life of the plant.

All electrical and piping penetrations are designed with double seals, the personnel access lock is provided with interlocked double doors and access hatches are provided with double seals. In all cases the interior of the penetration can be pressurized to the design pressure of 60 psig to check for leaks whenever a test is required.

ARCHIVED TEXT

CRITERION 57 - PROVISION FOR TESTING OF ISOLATION VALVES

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.

This criterion is met. The operability of containment isolation valves can be tested during shutdowns by tripping the control mechanism on all containment isolation valves. The control valves in the decay heat removal circuit must be manually operated during this test.

Measurement of the leak rate for each system which has two containment isolation valves can be made by pneumatically pressurizing the branch of the system lying between them during a shutdown. The systems which are singly valved are cooling water lines, steam and feedwater lines, air lines.

The leakage rate of each pair of containment valves (inlet and outlet) on cooling water lines can be tested by pressurizing the branch of the cooling system lying between them during a shutdown.

The leakage rate of each set of steam and feedwater valves as a group can be determined by pressurizing each steam generator during a shutdown.

The service air penetration leakage rate can be tested by pressurizing the service air system by means of the instrument air system and vice versa. The control valves in the decay heat removal system must be manually operated during this test.

The transfer tube isolation valve leakage rate can be tested by pneumatically pressurizing the tube between the isolation valve and the blank flange.

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CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

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.

This criterion is met. The containment air recirculation and cooling system consists of four fan cooler units; two particulate iodine filters are associated with two of the units. The fan coolers are equipped with inspection doors and all ductwork is flanged and exposed, so that all components of the system can be inspected. In normal operation, at least one fan cooler unit will be idle and therefore available for inspection.

The containment spray system consists of three pumps, two half capacity heat exchangers, associated valving and piping, and two spray headers. The entire system, with the exception of the spray headers, is external to the containment. Periodic physical inspection of all components is provided for in the design. The pump suction is taken either from the SIRW tank or the containment sump; both can be physically inspected.

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CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS

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.

This criterion is met. All components in the containment pressure reducing systems will be tested periodically for operability and performance. In the spray system, pumps will be operated with the spray line valves shut and with borated water recirculated to the SIRW tank; valves will be operated with the pumps not running. Shutdown heat exchangers will be periodically pressure and leak tested; operability will be verified during normal use for plant cooldown.

The containment air recirculation and cooling system consists of four fan cooler units. Normally, one to three of the units will be in operation.

The standby unit or units can be started for test purposes. The installed instrumentation and operation of the system provides frequent checks of the functional performance of the entire system. (Cf. Section 6.4 of the FSAR for details).

ARCHIVED TEXT

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS

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.

This criterion is met. The containment spray system is designed to permit periodic testing of system delivery capability.

A recirculation line is located in each spray line immediately upstream of the spray line valves. Operation of the containment spray pumps with the spray line valves shut will recirculate water to the SIRW tank to demonstrate flow path continuity to the valve; continuity through the remainder of the spray line and nozzles may be checked by air flow testing, using smoke or other means of verifying flow through each spray nozzle.

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CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

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.

This criterion is met. Active components of the containment spray system, such as spray pumps and valves, will be tested by simulating the initiating signals for the safeguards, utilizing the control circuit components to the maximum practicable extent. System operability is demonstrated without injection of borated water into the reactor coolant system or into the containment building. (During testing of a spray valve, the associated spray pumps are removed from the appropriate auto start sequencer circuits. However, any spray pump will always respond to a manually initiated start signal from its own control switch.) The test method will preclude the possibility of disabling the containment spray initiating circuit during the test.

The containment air recirculation and cooling system consists of four fan cooler units. Normally one to three of the units will be in operation. The nonoperating unit or units can be started by simulating the initiating signals for safeguards; the control circuit components will be utilized to the maximum practicable extent. Installed instrumentation will indicate operating conditions. Cf. Section 6.4 of the FSAR.

The automatic start of the emergency diesel generators will be tested periodically, as will the switchgear utilized in load shedding and sequential pickup of emergency loads. Cf. Section 7.3.4.2 of the FSAR.

ARCHIVED TEXT

CRITERION 62 - INSPECTION OF AIR CLEAN-UP SYSTEMS

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.

This criterion is met. The containment air cooling cleanup system consists of two 70,000 cfm fan cooler units, two 100,000 cfm particulate iodine fan cooler units, and a clean air distribution system is constructed with flanged ducts and dampers and is exposed.

During normal operations limited personnel access within the containment is possible and at least one fan cooler will be idle, therefore, a fan cooler could be inspected while operating.

The containment purge air supply fan units and the exhaust fan units are located in the reactor auxiliary building. The containment purge air supply system consists of an outside fresh air duct, two steam heating, automatic filter fan units and a valved purge air duct to the containment. The containment purge air exhaust system consists of a valved air exhaust duct from the containment, two water eliminators, particulate filter, fan units, a fresh air dilution duct system, and an exhaust duct system to the plant ventilation discharge duct. All ducts are of flanged construction and are exposed. All filters, coils and water eliminators on the fan units can be visibly inspected. The containment purge air system can be inspected when not in service.

ARCHIVED TEXT

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS

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.

This criterion is met. The containment integrated air cooling and cleanup system consists of two fan cooler units, two particulate iodine fan cooler units, and a clean cooled air distribution system. Under normal operating conditions one, two, or three fan cooler units will be in service dependent on the cooling water temperature which is subject to seasonal variations. Installed instrumentation and operation of the system provide frequent checks of the functional performance of the entire system.

The containment purge air supply fans and exhaust particulate filter fan units are located in the reactor auxiliary building. During normal operations of the reactor, the purge system is not operated, however, it will be periodically checked for functional performance.

ARCHIVED TEXT

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS

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.

This criterion is met. Two of the four containment fan coolers are equipped with the following components in order of flow sequence: louvers (spring loaded to fail open), mist eliminator, absolute filter, iodine filter, bypass louvers (designed to fail closed), cooling coil and fan.

The filters will normally be bypassed. As stated in "Criterion 62" the filters can be inspected during normal operation of the reactor. Installed instrumentation will indicate abnormal changes in filter pressure drops during routine operations. In place testing and out-of-containment testing of removed filter modules is also possible. Containment filters and the particulate air purge filters will be periodically evaluated to ensure they are operating within acceptable limits.

ARCHIVED TEXT

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEM

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.

This criterion is met. The containment air cleanup and cooling system, as stated in Criterion 63 will have one or more fan units in service at all times during normal operations. The functional capability of idle fan cooler units, and the transfer to alternate power sources can be independently tested.

ARCHIVED TEXT

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY

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.

This criterion is met. The criterion for the design of the spent fuel storage racks is that the k_{eff} of the fuel array shall remain less than 0.96 during normal use and in the event of postulated accidents or mishandling.

The safe geometry criteria are established as follows:

a) The receptacles or cavities containing the new or spent fuel assemblies will be arranged vertically in a square lattice, and the dimensions of the storage rack will be such that the clear space between adjacent receptacles will be sufficient to yield a k_{eff} less than 0.96 with unborated water.

b) The insertion of new or spent fuel assembly into any part of water slab between receptacles is prevented by the top frame. The openings in the top frame at each side of a receptacle are not large enough to receive a fuel element.

c) The vertical dimensions of the storage rack will be such that the top of the active fuel portion of the fuel assembly will be a sufficient distance below the top frame to assure that the k_{eff} is less than 0.96 even with a fresh fuel assembly located at the level of the top frame.

d) Distortion of the structure due to seismic loading shall be prevented by the use of lateral bracing so that there will be no reduction of the water slab between the cavities.

In addition, criticality is precluded by maintaining the pit boron concentration at 1700 ppm which is sufficient to maintain the critical core configuration subcritical at all times of life.

The spent fuel storage racks are further described in Section 9.5.3.2 of the FSAR.

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CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT

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.

This criterion is met. The fuel pool cooling system removes decay heat from the fuel stored in the spent fuel pool. The system is capable of removing decay heat from one-third of the core freshly discharged from the reactor while maintaining the pool water temperature below 110°F. The pool can accommodate 1-1/3 cores. With 1-1/3 cores in the pool, the pool water temperature is maintained below 140°F which will not damage the fuel.

The cooling system consists of two full capacity pumps, a full capacity heat exchanger, a filter and a demineralizer. The filter and demineralizer maintain the clarity and purity of the water.

Blanked off connections are provided for temporary tie-in to the shutdown cooling system to provide a backup for the fuel pool heat exchanger.

Piping is arranged so that the pool cannot be accidentally drained.

The system will be tested with regard to flow paths, flow capacity, heat transfer capability and mechanical operability prior to initial fuel loading.

ARCHIVED TEXT

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING

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 20.

This criterion is met. The fuel storage pool is constructed of reinforced concrete and lined with stainless steel. The fuel storage pool is filled with borated water, which provides a transparent radiation shield and a cooling medium for removal of decay heat.

In the refueling canal and the spent pool, a common water level is maintained to provide at least 10 feet of water shielding over the top of the fuel. This water will provide sufficient shielding to permit normal occupancy of the area by operating personnel.

The liquid waste disposal containers are built of carbon steel in accordance with ASME Code Section III. The tank lining is hole-free, continuous, and of uniform thickness. The tanks are located in the reactor auxiliary building (a Class I structure) in shielded compartments.

The gas hold-up tanks are built of carbon steel in accordance with ASME Code Section III. The tanks are located in the reactor auxiliary building (a Class I structure) in separate shielded compartments.

The solid wastes will be collected and compressed in drums stored in a shielded area. Spent resins are dewatered and sealed in Shielded Containers.

The shielding for radiation protection meets the requirements of 10 CFR 20 for all fuel and waste storage areas.

ARCHIVED TEXT

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE

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

This criterion is met. The spent fuel pool and waste handling facilities are located in the auxiliary building (a Class 1 structure). These areas provide confinement capability in the event of accidental release of radioactive materials and are ventilated with dilution air for discharge to the ventilation discharge duct. Each gaseous waste decay tank is located in its own shielded structural compartment and is so designed that an accidental release of radioactive material resulting from a rupture will not exceed a whole body dose of 0.5 rems at the site boundary. The liquid waste hold-up tanks are each located in separate compartments. Each compartment is provided with a floor drain which drains into an auxiliary building sump and is then pumped to the spent regenerant tanks in the auxiliary building.

ARCHIVED TEXT

CRITERION 70 - CONTROL OF RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT

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 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 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.

This criterion is met. The radioactive waste system is designed to limit to acceptable levels that potential release to the environs from radionuclides generated or assembled as a consequence of normal plant operations or emergency conditions, even under unfavorable environmental situations.

The liquid wastes are collected, treated (filtration, demineralization, or evaporation) as appropriate, and analyzed prior to release. A radiation monitor, recorder-controller monitors all liquid discharges from the Radioactive Waste Disposal and Control System. An automatic shut-off valve in the discharge line closes if control values are exceeded. The effluent discharge, when diluted with the circulating water discharge, will not exceed requirements of 10 CFR 20.

The gaseous wastes resulting from reactor operations are monitored for release to the ventilation stack or stored in the Gas Decay Tanks. Waste gas found suitable for discharge, in accordance with the requirements set forth in 10 CFR 20, are released under controlled conditions to the Auxiliary Building Ventilation System for dilution prior to discharge. A radiation monitor, recorder controller in the ventilation system automatically interrupts the flow of waste gas through a control valve in the gas discharge header, should the activity approach predetermined limits.

Space for storage of the solid wastes is provided so that packaging, handling and shipping can be carried out under favorable environmental conditions. All solid waste will be monitored, labeled, packaged and handled according to applicable regulations.

APPENDIX M

POSTULATED HIGH ENERGY LINE RUPTURE OUTSIDE THE CONTAINMENT

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The following are controlled drawings and can be viewed and printed from the applicable listed aperture card.

1. SUMMARY

An investigation was undertaken to determine the effects of a postulated rupture outside the containment of a pipe containing high energy fluid. During the course of the investigation, pipe stress analyses were performed in order to determine postulated break locations. Dynamic analyses of pipe response to rupture at the locations thus obtained were performed to determine the effects of these breaks on safety related equipment.

During the course of this study, certain modifications were identified which would be necessary to assure the availability of certain essential equipment and structures. Design and installation of these modifications has been completed.

The results of the study, which included consideration of the modifications to the facility, indicate that there is a fully adequate safe shutdown capability of the plant in the unlikely event of a rupture of a pipe containing high energy fluid outside the containment.

Subsequent to the study, the NRC issued criteria which provides relaxation in arbitrary intermediate pipe rupture requirements. This criteria is stated in Section M.2.4 and can be applied to all piping systems at Fort Calhoun Station. Using these relaxed rules an analysis was performed on the B31.1 portions of the Main Steam and Feedwater piping outside containment in Room 81 with overlap into the Turbine Building. All break locations on the B31.7 portion of the Main Steam and Feedwater piping remain as identified in Attachments A and B.

2. DISCUSSION

2.1 General

In response to an AEC letter to the Omaha Public Power District dated December 14, 1972 transmitting the document entitled "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" and the two (2) pages of Errata subsequently issued by the AEC on January 11, 1973, herein referred to as "AEC Criteria", a study was performed to determine the consequences of a postulated rupture outside the containment of a pipe containing high energy fluid. With regard to this matter, on December 22, 1972, OPPD transmitted to the AEC a schedule for completion of such a study by May 15, 1973. On January 18, 1973, representatives of OPPD met with members of the AEC-DRL staff to present the results of the analyses up to that time. Subsequently, additional meetings were held on February 16, 1973, March 7, 1973 and April 19, 1973. On March 14, 1973, OPPD transmitted a letter to the AEC reviewing the entire analysis and modification program and indicating the status of the program, including a list of specific modifications to be installed prior to plant operation at power levels above 5 percent Also submitted at the time were the preliminary floor strengthening provisions in Room 81 of the auxiliary building and the pipe restraint configuration.

This submittal deals with the analyses performed for the study and their results. It also defines and describes the modifications which were completed prior to the completion of the first refueling.

Subsequent to the study, NRC Generic Letter 87-11 was issued, which eliminated the requirement to postulate arbitrary intermediate pipe ruptures. A reanalysis of the non-safety-related portions of the Main Steam and Feedwater piping was performed applying the Generic Letter 87-11 criteria and resulted in fewer postulated pipe rupture locations. Hardware associated with certain previously-identified pipe ruptures is, therefore, not necessary, but provides an additional margin of safety and need not be removed.

2.2 Identification of High Energy Systems

All systems outside the containment whose design temperatures exceed 200°F or whose design pressures exceed 275 psig are considered to be high energy systems. For the purposes of this study, those systems which are not normally pressurized were excluded from consideration. Table M.2-1 lists those systems which were investigated.

Ruptures in pipes containing high energy fluid, up to and including the circumferential break of the pipe, were considered for those systems where the service temperature of the fluid exceeds 200°F and the design pressure exceeds 275 psig. Only the "critical" crack was assumed to occur in the piping of those systems where the service temperature of the fluid exceeds 200°F or the design pressure exceeds 275 psig. The size of the "critical" crack was assumed to be one-half (½) the pipe diameter in length and one-half (½) the wall thickness in width.

Table M-2-1 - "Systems Outside Containment Exceeding 200°F Service Temperature and/or 275 psig Design Pressure"

System	Service Temperature, °F	Design Pressure, psig	Maximum Line Size, in.
Main Steam	550	985	36
Feedwater	438	1335	20
	438	985	16
Charging	130	2735	2½
Letdown	550	2485	2
	550	650	2½
Auxiliary Steam	300	50	10
Condensate Return	212	10	6
Steam Generator Blowdown	550	985	5
Sampling	600	2485	3/8
Nitrogen	100	2400	1/2
	100	275	1½
Hydrogen	100	2400	1/2

2.3 Identification of Essential Structures and Equipment

A review was completed identifying the essential structures and equipment outside of the containment which would be required to place and maintain the plant in a cold shutdown condition following a postulated rupture outside the containment of a pipe containing high energy fluid with the simultaneous loss of off-site power. These structures and equipment consist of the following:

1. Control room
2. Room 81 of the auxiliary building
3. Auxiliary feedwater system
    a. Auxiliary feedwater panel, AI-179
    b. Emergency feedwater tank
    c. Auxiliary feedwater pumps
    d. Piping, valves, etc.
4. Cable spreading room
5. Switchgear area
6. Electrical penetration area
7. Diesel generators
8. Regulating and shutdown control element assemblies
9. Main steam isolation valves
10. Main steam safety valves
11. Safety injection system
12. Raw water system
13. Pressurizer pressure and level control

2.4 Revised NRC Line Break Criteria

For high energy fluid piping systems the criteria for determining the location of pipe ruptures will be as provided in NRC Generic Letter 87-11.

3. MAJOR HIGH ENERGY SYSTEMS

3.1 Identification of Major High Energy Systems

The main steam and feedwater systems have been identified as the major high energy systems since they are the two systems, because of line sizes, fluid energy levels and plant arrangement, which would have the greatest potential to inhibit a safe shutdown of the facility in the event of the postulated pipe rupture incident. Each steam generator is designed to produce 3.112 x 10^6 lb/hr of saturated steam at 770 psia and 544°F when provided with feedwater at 437.8°F. The arrangement of the main steam and feedwater lines in the area of concern is shown in Figures M-1 and M-2. Isometric drawings of these piping runs are shown in Figures M-3 and M-4. As can be seen from the figures, the main steam and feedwater lines pass from the containment through Room 81 of the auxiliary building and then into the turbine building.

The other high energy systems, because of smaller line sizes, lower fluid energy levels and plant arrangement, offer a lower potential for hindering a safe shutdown of the facility in the event of the postulated pipe break incident. These systems are discussed in greater detail in Section 4.

3.2 Stress Analyses, Blowdown Analyses and Dynamic Analyses

The locations of the postulated main steam and feedwater lines circumferential and longitudinal break points were determined in accordance with the AEC Criteria for that part of each main steam and feedwater line between the containment and the first isolation valve, and was designed in accordance with ANSI Code B31.7 (see Attachments A&B). The remaining part of each of these lines outside the containment was designed in accordance with ASME Code B31.1 1980. The corresponding stress analyses were performed with break locations determined using the revised NRC criteria discussed in Section M.2.4 (see Attachments D&E).

Attachment A provides detailed information with regard to stress analyses, system blowdown analyses and piping response to a postulated rupture of the B31.7 main steam lines outside the containment. Attachment B provides similar information for a postulated rupture of a feedwater line outside the containment. A non-linear, elastic-plastic finite element, dynamic analysis using a mathematical model of the restraint system and a lumped mass mathematical model of the piping system was performed for each of the main steam and feedwater lines in the areas of concern to confirm the adequacy of the initial design analyses. The dynamic analyses took into account the load-time variation of jet forces including the impact effects resulting from the motion within the gap between the pipe and the restraint. Figures M-5 and M-6 provide the results of the stress analysis of the main steam lines in the auxiliary building and turbine building respectively. Figure M-7 summarizes the results of the stress analysis of the feedwater B31.7 lines.

Applying the guidelines set forth in the AEC Criteria concerning the selection of assumed break locations to the B31.7 main steam lines (see Attachment A) in the auxiliary building, it was postulated that breaks could occur at the terminal points of each line, i.e., at the point of connection to the containment penetration. Intermediate breaks were postulated to occur at all points where the AEC Criteria of 2.0 Sm was exceeded. It can be seen that in each case, the intermediate breaks were postulated to be in the area between the containment penetration and the main steam isolation valve. These locations are shown on Figure M-5.

Applying the AEC Criteria to each of the B31.7 feedwater lines (see Attachment E) resulted in postulating that breaks could occur at the terminal points on each line, i.e., at the point of connection to the containment penetration. Applying the AEC Criteria to each of the feedwater lines resulted in postulating that breaks could occur at the terminal points on each line, i.e., at the point of connection to the containment penetration and, for the north line, the point of connection with the header in the turbine building and, for the south line, the point of connection with feedwater heater 6B. In addition, breaks were postulated to occur at points 15 and 50 on the north feedwater line and points 510 and 530 on the south feedwater line. As in the case of the main steam lines, the intermediate break locations for each feedwater line were postulated to be in the area between the containment penetration and the feedwater isolation valve. These locations are shown in Figure M-1. In addition, the AEC Criteria further require that a small "critical crack" be postulated to occur at any point on the B31.7 lines.

Applying the NRC Criteria discussed in Section M.2.4 to the B31.1 Main Steam piping (see Attachment D) and Feedwater piping (see Attachment E) resulted in postulating that breaks would occur at the terminal points on each line. For the Feedwater lines these occur at the two connections to High Pressure Feedwater Heaters 6A & B in the Turbine Building. For the Main Steam lines, these occur at the connections to the Turbine Stop Valves and at anchor mark number MSH-5 on the 36" header in the Turbine Building.

3.3 Pressurization of Room 81

In the event of a break in a main steam line in Room 81 of the auxiliary building, the room would be pressurized by the steam blown out of the break. Pressurization of the room would also occur if a feedwater line broke in Room 81 and the feedwater was at an elevated temperature. In this case, a portion of the feedwater flashing into steam would cause the pressurization. It was determined that the energy and mass blowdown from the circumferential break of a main steam line in Room 81 was much greater than the blowdown from a longitudinal break of a main steam line or the circumferential or longitudinal break of a feedwater line. Therefore, the pressurization analysis for Room 81 was based on a postulated circumferential rupture of a main steam line since this incident would have the greatest potential for damaging the room due to over-pressurization. Pipe restraints were provided to preclude the possibility of a main steam or feedwater line contacting another main steam or feedwater line following its postulated failure.

The mass blowdown rate was calculated using the following assumptions:

1. When the postulated rupture occurs, the main steam line is filled with steam at the design pressure of 1000 psia.

2. Pressure in the steam generators remain constant at 1000 psia.

3. Initially, the flow out of the break exceeds the flow out of the steam generators. This occurs because the flow restrictors do not affect the large volume of steam already downstream of them in the main steam lines. Thus, the steam lines offer very little resistance to the inventory already in the piping.

4. Initial steam flow rate is based on a differential pressure between the main steam line and Room 81 of 985 psi.

5. As the initial inventory of steam in the lines is exhausted, the steam flow rate decreases due to the effects of the flow restrictors.

The parameters used in the mass blowdown calculation are shown in Table M.3-1. Figure M-8 presents the curve of calculated steam flow vs. time.

Table M.3-1 - "Parameters Used in Steam Blowdown Calculation"

Initial Steam Pressure	1000 psia
Volume of Steam Lines	2,363 ft^3
Main Steam Line Inside Diameter at Break	36 inches
Rated Choke Flow of Flow Restrictor	10,867,000 lb/hr. at 770 psia

In the Gibbs & Hill computer program (TPD) used for these pressurization calculations, the transient conditions (pressure, temperature and steam-air inventory) inside Room 81 were calculated. The transient conditions inside the room depend on the blowdown rates of mass and energy, the volume of the room and its initial conditions, and the size and type of port opening. Mass and energy balances were used in each time interval to determine the transient conditions. Thermodynamic properties of steam and the perfect gas law for air were used in the iterative process.

The following conditions were used in the calculation:

1. Mass blowdown rate and energy blowdown rate was known.

2. The chamber was vented through a port opening to ambient atmosphere. The ambient atmosphere serves as a infinite sink.

3. The port opening was treated as an orifice.

4. The steam-air mixture inside the chamber was assumed homogeneous.

5. Air was assumed to behave as a perfect gas.

6. No credit was taken for heat transfer from the steam-air mixture to the chamber structure.

At the instant of rupture, steam starts to flow from the ruptured pipe into the room. The energy and mass added to the room cause the pressure and temperature inside the room to rise. When the pressure differential between the room and the outside atmosphere reaches 30 lb/sq. ft. (0.21 psi), the plastic pressure relief domes blow out. Due to the unbalance of the pressure between the room and the outside atmosphere, some of the mass and energy is forced to flow through the "vent area" into the atmosphere. The amount of flow is dependent on the size of the "vent area," the type of edge the opening has, the pressure differential resulting from the steam blowdown and finally, the specific weight of the fluid that is being expelled. The volume of Room 81 is 193,200 ft^3. The vent area is composed of eight (8) openings in the roof which are covered by plastic pressure relief domes. The vent area is approximately 1200 ft^2; which has been reduced to approximately 1060.5 ft^2 by the addition of intrusion barriers. Figure M-1 shows the locations of these openings.

Figure M-9 shows the curve of pressure build-up in Room 81 as a function of time. As can be seen from the curve, the peak pressure differential between Room 81 and the outside atmosphere was calculated to be 1.2 psi. The vent area has been reduced from that on which Figure M-9 is based. Supplemental calculation has shown that the differential pressure will not exceed the design differential pressure of 1.5 psi for the steam flow given in Figure M-8.

Refer to Section M.3.6.5, "Switchgear Area" for a discussion on design features required to prevent HELB steam, migrating thru ductwork, from adversely affecting safety related equipment in areas communicating with Room 81 via ductwork.

3.4 Flooding in Room 81

In the event of a feedwater line rupture in Room 81, feedwater would flow from the break onto the floor of the room. Some water would flow into the turbine building through the door in the north-east part of the room. This is a conservative assumption as two doors into the room exist. The floor of the turbine building is made of grating at this point, open to the floor below.

It was determined that a circumferential rupture of a feedwater line with the plant at the hot standby condition would present the most serious flooding incident. Under these conditions, since the water in the lines would be at approximately room temperature, there would be a negligible amount of flashing. Thus, most or all of the water would flow onto the floor. Under other load conditions, feedwater temperature could be as high as 437.8°F, which would result in up to 24 percent of the water flowing from the break flashing into steam, and hence, less water flowing onto the floor.

For the postulated feedwater line break, it was assumed that the check valve inside the containment on the affected feedwater line would fail to operate. Thus, feedwater would flow into the room from both the steam generator side and the steam generator feed pump side of the break. It was assumed that the feedwater pump flow rate would increase above normal flow to the run-out condition. It was further assumed that the other pumps in the system would supply water to the steam generator feed pumps at sufficient flow and pressure to permit the continued operation of the feed pumps in the run-out condition.

Calculations indicated that flow from the break would be maximum during the first 0.66 seconds, with water flowing from both sides of the break. The water from the steam generator side of the break would be the volume of water in the feedwater line between rupture point and the steam generator. The water would then flow onto the floor at a slightly reduced rate, due to the flashing of a portion of the hotter water from the steam generator. After approximately 13.4 seconds, flow from the steam generator side of the break would cease. This is due to the fact that the water level in the steam generator would have then fallen below the level of the feedwater distribution ring. Approximately 9.7 seconds later, the water flow rate onto the floor would again decrease. At this point, blowdown of the feedwater piping water inventory up to the steam generator feed pumps would be terminated and flow would be from the feed pumps only. Approximately 223.4 seconds after the time of rupture, flow from the break would cease, since the inventory of the secondary system would be exhausted. Figure M-10 shows the calculated time history curve of the feedwater flow rate from the postulated rupture.

During the postulated incident, water would flow from Room 81 into the turbine building through the door at the north-east end of the east wall of the room. Investigation indicated that the door would act like a weir. A computer code was written to accept the flow rate into the room described above and, using a Francis weir formula, to calculate the depth of flooding in the room. The results of the calculation are shown in Figure M-11. It can be seen that the depth of water reaches a maximum of approximately 1.36 ft. The floor of Room 81 is capable of withstanding a water depth of 3.2 ft. at working stress design allowable. The effect of room pressurization for this case is negligible, since the feedwater would be approximately room temperature.

Subsequent to the original flooding analysis, an additional source of potential flooding was added in Room 81. Under certain conditions, both the main feedwater pumps and the engine driven auxiliary feedwater pump could be in operation. Under this scenario, the inventory available for flooding would be increased by the runout flow of the engine driven auxiliary feedwater pump. Calculations show the increase in flood level due to the increased flooding inventory would not exceed the maximum allowable flood level in the room (1.5 feet).

3.5 Modifications for the Protection of Essential Equipment

All of the essential structures and equipment which would be required to place and maintain the plant in a cold shutdown condition following a postulated rupture outside the containment of a pipe containing high energy fluid with the simultaneous loss of off-site power have been identified in Section M.3.3. Special attention has been given to the protection of these vital items. Analyses indicated that certain modifications were required to provide additional protection and thereby ensure the availability of equipment needed for a safe shutdown in the event of the postulated incident. The modifications necessary for the protection of vital equipment in the event of a rupture of a major high energy line are discussed below. The design, procurement and installation of these modifications was completed prior to operation of the facility at power levels above five percent of full power.

3.5.1 Relocation of auxiliary feedwater panel AI-179 --

Investigation indicated that the original location of the auxiliary feedwater panel AI-179 in Room 81 of the auxiliary building could give rise to a potentially undesirable situation in the event of the postulated high energy line rupture. To eliminate any possible hazards, the panel AI-179 was relocated from Room 81 to the electrical penetration room below Room 81. Separation of cables from this panel to essential equipment located in Room 81 was maintained.

The panel provides indication of steam generator level and pressure as well as local control of the steam driven auxiliary feedwater pump, auxiliary feedwater containment isolation valves, safety pump recirc valves and steam supply valves to the turbine-driven auxiliary feedwater pump.

By means of this modification, access to and operation of the auxiliary feedwater panel AI-179 in the event of the postulated pipe rupture incident is assured.

3.5.2 Main steam and feedwater piping restraints --

As discussed in Section M.3.2, it is postulated that a circumferential rupture or an equivalent longitudinal large break could occur within Room 81 between the containment penetration and the first isolation valve outside of the containment on any of the main steam or feedwater lines. A review of the potential consequences of a rupture in one of these lines indicated that pipe whip protection would be required for adjacent pipes and valves, for the emergency feedwater tank, for the wall between Room 81 and the control room, and for building columns and floor systems. To provide this protection additional restraints were required. New restraints were added for the main steam and feedwater lines in Room 81 of the auxiliary building to supplement existing restraints. Also, restraints were added for the main steam line in the turbine room immediately adjacent to the wall of the auxiliary building. The function of these restraints is to prevent the pipe whipping that could otherwise occur following a rupture of a main steam or a feedwater line. To prevent a whipping pipe from impacting against the Room 81 wall facing the control room, a yoke is provided at the north end of Room 81 to restrain the pipes in the unlikely event of a circumferential rupture of one of the main steam or feedwater lines.

The yoke was fabricated of heavy steel members made up of plates and welded together to form a single structure. To minimize the energy of impact, the hot clearance was minimized by stops with adjustable shimming which was installed between the vertical members of the restraint and the outside walls of the pipes. The yoke was also provided with U-bolts, mounted to restrain downward movement of the main steam line. The downward movement can occur as a result of a circumferential break in the upward sloping section of the lines at the first elbow outside of the containment penetrations. Plastic deformation of the U-bolts is permitted to absorb the energy of impact.

The yoke was anchored with "thru" bolts to the floor of Room 81 over a sufficient length so that the slab, acting as a large diaphragm, can carry the loads to the walls and foundation. Short six inch diameter pipe sections, welded to the underside of the yoke and set in grout into recesses drilled into the slab, transmit shear loads from the yoke to the floor.

In the vicinity of the pipes between the containment penetrations and the first isolation valves, additional restraints were provided to prevent a whipping main steam or feedwater pipe from hitting an adjacent pipe or valve, from striking a nearby tank or building column, from overloading the floor system and from forming a plastic hinge with consequent pipe whip. The restraints were designed to absorb the impact energy of a ruptured pipe and to transfer the loads into the existing building system. The locations and directions of resistance of the restraints in Room 81 are shown on Figure M-12. The existing restraints for the north feedwater pipe were constructed of reinforced concrete and are anchored with reinforcing steel to the original Room 81 floor. To minimize the impact of these restraints, provisions were made to reduce the hot clearance between them and the outside of the pipe walls.

The remaining restraints were constructed of welded heavy steel plates and sections that are anchored to the floor. Shear lugs and rebars comprise the anchors for some of the steel restraints and other restraints were cast in the supplementary floor and anchored to the original one with "thru" bolts. As with the other restraints, the hot clearance between the outside of the pipe and face of the restraints was minimized by providing for adjustable shimming. This minimizes the impact energy.

In order to protect the wall between the switchgear area and the turbine building from the postulated rupture of a main steam line or feedwater line at the point where these lines join their respective headers in the turbine building, a grillage of steel plates supported by steel beams was installed. However, due to revised NRC HELB criteria and a revised piping analysis, these are no longer required to be postulated circumferential break locations (see Section M.3.2). This grillage was designed to transfer the loads from the postulated breaks into the reinforced concrete floors of the auxiliary building at Elev. 1036' 0" and Elev. 1011'0". These loads would then be resolved by the deep beam diaphragm system of the noted floors. In addition, a vertical restraint was installed to restrain the upward movement of a main steam or feedwater pipe in the event of the postulated rupture.

The restraints in the turbine room were fabricated of heavy steel plates and shapes anchored to the turbine building steel structure. Reinforcement was provided for the steel structure so that the restraint forces are transmitted to the turbine building foundations. Adjustments were provided to minimize the hot clearance between the outside of the pipes and the restraints.

Design of all the restraints was based on forces and responses that have been verified by non-linear, elastic-plate dynamic analyses of the piping systems and restraints. Stresses in the restraints are below the yield strengths of the materials utilized except for the U-bolts utilized around the pipe in several of the restraints. Plastic deformation under initial impact is permitted for these bolts.

The pipe whip restraints installed near points 170 and 175 on the Main Steam line and Feedwater point 340 were eliminated as postulated line break locations by the code qualification analysis performed subsequent to using the Generic Letter 87-11 requirements. Therefore, the hardware installed for these restraints in the turbine room is not necessary and only provides additional protection against pipe break.

3.5.3 Reinforcement of floor in room 81 --

A supplementary floor, heavily reinforced in two directions, was placed under the main steam and feedwater pipes in those areas of Room 81 where a major rupture of a main steam or a feedwater pipe was postulated. This supplementary floor is designed to absorb the energy of impact of a downward moving pipe as well as to provide support for restraints and jet impingement barriers. In this same area steel beams and columns were added at the underside of the existing floor system as additional reinforcement to carry rupture loads to the foundation mat.

3.5.4 Protection from critical crack jet --

An analysis was made to determine the effects of jet impingement from a critical crack. A critical crack is defined as a longitudinal crack in a high energy line with a length equal to one-half (½) of the pipe wall thickness. It was postulated that a critical crack could occur at any point along the length of a high energy line.

In Room 81, the protective enclosure, described in Section M.3.5.8, around the main steam and feedwater lines between the containment and the first isolation valve provides protection from critical crack jet impingement on adjacent structural elements, safety valves and valve operators.

The floor of Room 81 is adequate to resist the jet impingement forces from a critical crack. However, in areas not covered by the supplementary concrete slab, a one-eighth (1/8) inch thick steel plate has been placed on the floor under the main steam and feedwater pipes to provide protection against possible erosion of the concrete by the steam jet. A similar plate has been placed along the wall between Room 81 and the control room alongside the pipes for protection against erosion of the concrete.

Protective steel barriers were provided to protect safety valves and isolation valve operators from jet impingement from a critical crack in those areas where the main steam and feedwater pipes are not covered with a protective enclosure.

In the turbine room it was determined that the wall between the turbine building and the switchgear room had to be reinforced or protected from jet impingement. Both solutions were utilized. To reinforce the existing wall, plates were attached to both surfaces and appropriate shear lugs were introduced to transform the wall into a composite section capable of resisting the postulated jet impingement force. Where this technique did not provide adequate strength a barrier of rolled steel sections and plate was fabricated and attached to the auxiliary building at the floor level so that the horizontal forces would be transmitted directly into the reinforced concrete floor system.

3.5.5 Additional steam supply line to turbine driven auxiliary feedwater pump --

The steam supply to the turbine driven auxiliary feedwater pump was redesigned to include a line from each main steam line. The connection to the main steam lines is upstream of the main steam isolation valves. The routing and support of the two lines is such that the rupture of a main steam line cannot damage the steam supply line associated with the other main steam line. Each steam supply line has a remotely operated isolation valve and a check valve. Each check valve was installed as close to the junction of the two supply lines as practical. The piping between each main steam line and the associated isolation valve was designed in accordance with USAS Code B31.7 1968. The remainder of the steam supply lines was designed in accordance with USAS Code B31.1 1967. The supply piping to the turbine driven auxiliary feedwater pump, from the main steam line to the pump, has been seismically restrained. The routing of these lines in room 81 is shown in Figure M-1. The steam supply lines are completely enclosed in the switchgear area to prevent adverse environmental conditions from affecting essential equipment. This modification ensures the availability of an uninterrupted source of steam to the turbine driven auxiliary feedwater pump in the event of the postulated pipe rupture incident.

3.5.6 Roof vent area in Room 81 --

Analyses indicated that approximately 1200 ft^2 of roof vent area would be required to limit the consequences of a steam line break in Room 81 with regard to pressurization of the room. Accordingly, vent area was provided to bring the total opening up to approximately 1200 ft^2. The addition of security intrusion barriers has reduced this area to approximately 1060.5 ft^2. Supplemental calculation has shown that the differential pressure resulting from this reduced area will not exceed the design differential pressure of 1.5 psi for the steam flow given in Figure M-8. The vent area is covered by plastic pressure relief domes, rated at 30 lb/sq. ft. The locations of these openings are shown in Figure M-1.

This vent area ensures that the design pressure of Room 81 will not be exceeded in the event of the postulated main steam line rupture. (See Section M.3.3 for additional information).

3.5.7 Protection against flooding in Room 81 --

In the event of the postulated feedwater line rupture in Room 81,water would accumulate on the floor of the room. Certain modifications have been made to ensure that water will not pass through the floor of Room 81 around piping, cable trays, conduit and ventilation ductwork into the switchgear area and electrical penetration area on the floor below. These modifications also serve to protect the electrical areas below from being subjected to a steam environment following a steam line break in Room 81.

Pipes carrying cold fluid have been imbedded in concrete where they pass through the floor of Room 81. Pipes carrying fluid at elevated temperatures have been surrounded by sleeves at the floor level extending at least two (2) feet above the floor. Seals, flexible where required by movement of the pipe, have been installed between the pipes and the sleeves. These seals will withstand the increased pressure in Room 81 following the postulated steam line break. The seals are adequately protected from the jet force caused by a postulated high energy line rupture.

Conduit has been imbedded in concrete where it passes through the floor of Room 81. Cable trays are surrounded by water-tight four-sided enclosures of steel plate embedded in the floor opening extending at least two (2) feet above the floor and the openings sealed. These seals will withstand the increased pressure in Room 81 following the postulated steam line break. The seals are adequately protected from the jet force caused by a postulated high energy line rupture. All ventilation ductwork which formerly passed through the floor of Room 81 was rerouted so that it no longer penetrates the floor openings which have been sealed.

Refer to Section M.3.6.5, "Switchgear Area" for a discussion on design features required to prevent HELB steam, migrating thru ductwork, from adversely affecting safety related equipment in areas communicating with Room 81 via ductwork.

3.5.8 Protective enclosures around main steam and feedwater lines in Room 81--

The investigation of the consequences of the postulated rupture of a high energy line outside of the containment led to the determination that the jet impingement from a longitudinal rupture of a main steam or feedwater line could have serious effects. It was determined that a jet in Room 81, unless some means of protection was provided, would have had the potential to damage or destroy nearby tanks, safety valves, isolation valve operators, building columns, walls and the roof over the room. Studies showed that it was not practical to provide individual protective barriers for every item requiring protection. It was determined that the most efficient and practical way to provide protection against jet impingement was to install a protective enclosure around those sections of the piping runs in which a rupture was postulated. These enclosures would contain the jets to the degree that they would not impair the function of structures or equipment necessary for a safe shutdown.

A protective enclosure was provided around the main steam and feedwater lines between the penetration sleeves and the first isolation valves, where a large rupture is postulated. This enclosure, although designed primarily to limit the effects of jet impingement, also serves to minimize the reaction effects of a longitudinal rupture by containing the jet and preventing the formation of an unbalanced external force. A description of this enclosure is provided below. The main steam and feedwater pipe protective enclosures consist of a series of longitudinal flat steel bars placed around the outside of the insulation and held in place by wire rope wrapping. The wire rope wrapping consists of a series of independent sections of three loops each spaced to limit the deformation of the flat bars. The longitudinal flat bars were fabricated from steel having minimum yield strength 36 ksi and 50 ksi. The wire rope was of Monitor AA grade steel. Batten plates were placed under the flat bars to fill the space between adjacent bars where necessary to prevent small jets from impinging on nearby valves, valve operators or building columns. Intermittent clips hold the flat bars in place against the wire rope wrapping following a pipe rupture so that a large space cannot form between adjacent bars. The arrangements of the protective enclosures are shown in Figure M-13.

The protective enclosures were designed for two loading conditions:

1. Jet impingement from a longitudinal rupture acting on the inner surface of the enclosure at the rupture.

2. Uniform internal pressure on the enclosure equal to the pressure in the ruptured pipe.

The design considers the dynamic effects of the postulated pipe rupture. Under the dynamic loading plastic deformation of the flat bars is permitted. Tension in the wire rope is less than 80 percent of the minimum guaranteed tensile strength. For static loading conditions stresses in the flat bars are below the yield strength. A design summary of the protective enclosures is provided in Attachment C.

3.6 Effect of Postulated Rupture on Essential Structures and Equipment After Modification

During the course of the investigation, an evaluation was made, giving consideration to the modifications, of the effects of the postulated rupture of a major high energy line on essential structures and equipment. The results of this evaluation are given below. The results show that, in the event of the postulated rupture of a major high energy line, essential structures and equipment will not be damaged and the capability to place and maintain the plant in a safe shutdown condition will not be impaired.

3.6.1 Control Room --

Careful investigation has revealed the control room cannot be damaged by the postulated rupture of a major high energy line. This is true whether the postulated rupture considered is the circumferential or longitudinal rupture or the small "critical crack." The location of the control room with relation to the major high energy lines outside the containment is shown in Figure M-1.

Section M.3.5 describes the various modifications made to protect essential structures and equipment from all postulated major high energy line breaks. Attachments A and B provide the blowdown analyses and the dynamic analyses of pipe response to the postulated ruptures. As can be seen from these sections and attachments, pipe movement from a postulated main steam or feedwater line circumferential rupture will be effectively arrested by the restraint system, thereby precluding the possibility of any of these lines damaging the control room. Jet forces from the postulated longitudinal rupture of a major high energy line would be controlled by the protective enclosures around these lines. In addition, the steel plates on the wall and floor of Room 81 will prevent erosion of concrete by jets from critical cracks. All of these items are discussed in greater detail in the section dealing with Room 81 of the auxiliary building (Section M.3.6.2).

Analysis for the critical crack in the main steam lines shows that the jet impingement loads will not impact the structural integrity of the control room outside air ductwork on filter units in Room 81.

Analysis for the critical crack in the feedwater lines shows that the jet impingement loads on the control room filter unit adjacent to the line are high enough that a failure of this unit must be postulated. The radiological releases in this case are insignificant and operation of the filter units is not required.

Thus, it can be seen that the required operation of the control room would continue satisfactorily in the event of the postulated high energy line rupture.

3.6.2 Room 81 of the auxiliary building --

A thorough investigation was made to determine the effects of a postulated rupture of a high energy line in Room 81. Consideration was given to both a longitudinal and a circumferential break as well as a critical crack.

A longitudinal failure in any of the four major high energy lines at the containment penetration will be absorbed by the containment sleeve. The anchor point is to the inside containment end of the penetration, so there is sufficient length of sleeve to absorb the force of such a break. Longitudinal ruptures in the main steam and feedwater lines in the other locations postulated for Room 81 would be contained by the pipe enclosures described in Section 3.5.

A circumferential rupture of the north main steam line at any locations between containment and the first elbow would be effectively controlled by the restraints located on either side of the elbow. The yoke spanning the steam and feedwater lines at the north end of the room would limit the movement of the north main steam line in the event of a circumferential rupture of this line anywhere from the first elbow to the main steam isolation valve, preventing this line from impacting the wall at the north end of this room which separates Room 81 from the corridor outside the control room.

Circumferential breaks in the south main steam line are controlled in a similar fashion by the restraints located on the line. Attachment A provides, in more detail, the results of the dynamic analyses of pipe response to the postulated circumferential rupture of the main steam lines in Room 81. These analyses were not performed at every discrete location between the containment and the isolation valves. Rather, specific locations which would demonstrate the complete adequacy of protection for all postulated rupture locations were selected for analysis.

Pipe movement due to a circumferential rupture of either north or south feedwater line at any postulated break location in Room 81 would be effectively controlled by the restraints located near the first elbow of each line and by the yoke which spans all four major high energy lines at the north end of the room. The restraints also prevent the formation of plastic hinges. Attachment B provides a more detailed description of the results of the dynamic analyses of pipe response to the postulated circumferential rupture of the feedwater lines in Room 81. As was the case with the main steam line analyses, the rupture locations selected were chosen so as to demonstrate the complete adequacy of protection against a feedwater line circumferential rupture at any postulated rupture locations.

The jet force from a critical crack in the main steam and feedwater lines in Room 81 will not pose a danger to the integrity of the room. Steel plates on the wall and floor will prevent erosion of the concrete by the critical crack jet.

The room is protected from overpressurization by means of pressure relief domes in the roof. These domes provide a vent area of 1060.5 square feet. Maximum pressurization of the room would be caused by the postulated circumferential rupture of a main steam line. The vent area serves to limit the differential pressure in the room to below the room design differential pressure of 1.5 psi. The required operation of essential equipment would not be affected adversely by the steam environment.

Analysis of the postulated feedwater line break and resultant flooding indicated that the maximum depth of flooding would be approximately 1.36 feet. The floor is capable of withstanding a depth of water of 3.2 feet. The required operation of essential equipment would not be affected adversely by the water level on the floor or the environmental conditions caused by the flashing of feedwater.

All openings in the floor of Room 81, through which pipes, cable trays and conduits pass, are adequately sealed to prevent water or steam from passing into the switchgear room below. Ventilation ductwork has been re-routed so that it no longer penetrates the floor of Room 81. It has been determined, therefore, that the structural integrity of Room 81 would not be jeopardized in the event of the postulated rupture.

Refer to Section M.3.6.5, "Switchgear Area" for a discussion on design features required to prevent HELB steam, migrating thru ductwork, from adversely affecting safety related equipment in areas communicating with Room 81 via ductwork.

3.6.3 Auxiliary feedwater system --

A thorough examination of the effects of the postulated rupture of a major high energy line on the auxiliary feedwater system has shown that the required functioning of system components would not be compromised.

The auxiliary feedwater panel, AI-179, has been moved from Room 81 and relocated in the upper electrical penetration room, 57W, below. It would not be affected by the postulated rupture. The emergency feedwater tank would not be affected by a jet from the postulated rupture of a main steam or feedwater line due to the protective enclosures around these lines. The restraint system would prevent the occurrence of pipe whips following the postulated circumferential break and thereby protect the emergency feedwater tank from these consequences of a postulated break. The safety related auxiliary feedwater pumps are located in the basement of the auxiliary building and cannot be damaged by the postulated ruptures of pipes in Room 81. Steam for the turbine driven auxiliary feedwater pump is supplied from each main steam line. A failure of one main steam supply will not prevent the required steam from reaching the turbine driven feedwater pump due to the redundancy of the steam supply. The two steam supply lines in Room 81 to the turbine driven auxiliary feedwater pump and the auxiliary feedwater pumps' discharge line are adequately restrained in Room 81 so that they can resist the jet forces imposed on them in the event of the postulated rupture. The main steam and feedwater lines protective enclosures eliminate the jet force of a postulated longitudinal rupture from consideration. Essential valves are protected by jet deflectors where required, but in any case these valves are fail-safe valves. The essential electrical cables are of the same design as the cables used inside the containment and were designed and tested to withstand the LOCA conditions in the containment (288°F, 60 psig, 100 percent humidity.)

The environment imposed on the cables by the postulated break would not be as severe as those conditions for which the cables were designed. Pressurization of flooding in Room 81 will not affect the required operation of essential system equipment.

It can be seen, therefore, that the postulated rupture and the subsequent effects would not inhibit the proper functioning of the auxiliary feedwater system.

3.6.4 Cable spreading room --

The cable spreading room is located in the auxiliary building, immediately below the control room. The location of the room is such that the only potential sources of damage to it are the 36 inch diameter main steam line and the 20 inch diameter feedwater line, both located in the turbine building. The cable spreading room and its relation to these lines is shown in Figure M-2.

The only postulated circumferential or longitudinal break location for the feedwater line in this area is at the connection of the north feedwater line to the 20 inch feedwater line in the turbine building. However, due to revised NRC HELB criteria and a revised piping analysis, this is no longer required to be a postulated circumferential break location (see Section M.3.2). The restraint system would prevent the ruptured line from whipping into and penetrating the wall separating the turbine building from the cable spreading room in the event of the postulated double-ended break incident. The jet impingement force from the postulated longitudinal break would be resisted by the plates attached to the existing wall. Critical crack jet impingement is not considered to be a problem. The only postulated large breaks of a main steam line in this area are at either side of the elbow of the 36 inch steam header in the turbine building. However, due to revised NRC HELB criteria and a revised piping analysis, these are no longer required to be postulated circumferential break locations (see Section M.3.2). These locations, points 170 and 175, are shown in Figure M-2. As in the case of the feedwater line, the restraint system would prevent the 36 inch main steam line from whipping into and penetrating the wall separating the turbine building from the cable spreading room in the event of the postulated circumferential break.

The plates attached to the existing wall would provide a sufficient protection from the jet impingement force of the postulated longitudinal break. Critical crack jet impingement is not considered to be a problem.

It can be seen, therefore, that the postulated high energy line ruptures would not damage the cable spreading room.

3.6.5 Switchgear area --

The switchgear area is located in the auxiliary building immediately below Room 81. The location of the room is such that the only potential sources of damage are the main steam and feedwater lines in Room 81 and the main steam line in the turbine building. The switchgear area and its relation to these lines is shown in Figures M-1 and M-2.

Fusible link actuated trap doors with 160°F links are installed in each duct (supply and return) connecting VA-19 and VA-41, located in Room 81 to the switchgear room. In the event of a high energy line break (HELB) in Room 81, migrating steam passing through these ducts will quickly heat the fusible link elements to their melting point allowing the trap doors to open. This will vent most of the migrating steam into the turbine building. Some steam will continue to flow through the switchgear wall openings due to the turning vanes located in the elbows containing the trap doors.

Venting the ducts reduces steam flow through the wall mounted Fire Dampers FD-35, 36, 37 and 38 thereby preventing excess differential pressure which might otherwise interfere with damper closure. These dampers also contain 160°F fusible links and close by gravity when the melting point is reached. Once closed, the fire dampers isolate the communicating areas from further steam migration and the trap doors prevent pressure buildup and potential damper failure. These design features mitigate the consequences of a HELB in Room 81 from creating adverse conditions in areas which communicate via the HVAC ductwork associated with VA-19 and VA-41.

The postulated longitudinal break of a major high energy line in Room 81 would not affect the switchgear area largely because the break locations are not in the area above the switchgear room. The postulated circumferential break would be effectively resisted by the restraint system, eliminating the possibility of damage to the switchgear area. Steel plates on the floor of Room 81 under the high energy lines would prevent erosion of concrete by the critical crack jet. In the event of the postulated double-ended break of a main steam line either at its point of connection to the 36 inch header in the turbine building, or at the terminal point of the header the restraint system would prevent the ruptured pipe from whipping into and penetrating the wall between the turbine building and the switchgear area. However, due to revised NRC HELB criteria and a revised piping analysis, these are no longer required to be postulated circumferential break locations (see Section M.3.2). The plates attached to the wall and the rolled steel sections would effectively resist the jet impingement force from a postulated longitudinal break. Critical crack jet impingement is not considered to pose a problem.

It can be seen, therefore, that the postulated rupture of a main steam or feedwater line, either in Room 81 of the auxiliary building or in the turbine building, would not damage the switchgear area.

3.6.6 Electrical penetration area --

The electrical penetration area is located in the auxiliary building, immediately below Room 81 and adjacent to the switchgear room. The location of the room is such that the only potential sources of damage to the area are the main steam and feedwater lines in Room 81. Figures M-1 and M-2 show the electrical penetration area in relation to these lines.

Jet impingement from the postulated longitudinal rupture of a main steam or feedwater line was removed from consideration as a potential source of damage due to the fact that the protective enclosures around these high energy lines would effectively contain the jet from the postulated rupture. Pipe movement due to the postulated major high energy line circumferential break would be resisted by the restraint system and the supplementary reinforced concrete slab. It is not considered possible that either a main steam or feedwater line could penetrate the floor of Room 81 and enter the electrical penetration area. Protection from critical crack jet impingement of that part of the floor of Room 81 which is directly above the electrical penetration area is provided by two methods. The supplementary floor is considered sufficient to withstand the jet from the critical crack. In those areas not covered by the supplementary concrete slab, a one-eighth (1/8) inch thick steel plate located on the floor under the main steam and feedwater lines provides protection from possible erosion of concrete by the jet. Neither pressurization nor flooding of Room 81 would damage the electrical penetration area since neither of these consequences of the postulated break will cause structural failure in Room 81. All openings through which pipes, cable trays and conduits pass from Room 81 into the electrical penetration area are sealed so as to prevent steam or water from entering the area through them.

Therefore, it has been determined that no damage would occur to the electrical penetration area or to the equipment within it in the event of the postulated major high energy line break.

3.6.7 Diesel generators --

The location of the diesel generators is such that neither they nor their associated control cables are located near major high energy lines.

Thus it can be seen that a postulated high energy line break would not impair the proper operation of the diesel generators.

3.6.8 Regulating and shutdown control element assemblies --

The proper functioning of the regulating and shutdown control element assemblies would not be hindered by the postulated rupture of a major high energy line outside the containment. The control element assemblies themselves are, of course, located in the containment, making them immune to the break. The only vulnerable parts of the system would be the control room, the cables and the room through which these cables pass. The adequate protection of these items is discussed elsewhere in Section M.3.

It has therefore been determined that the regulating and shutdown control element assemblies would continue to perform properly in the event of the postulated rupture.

3.6.9 Main steam isolation valves --

The main steam isolation valves are located in Room 81 of the auxiliary building as shown in Figure M-1.

The protective enclosures around the main steam and feedwater lines would provide adequate protection for these valves in the event of the postulated longitudinal rupture. These enclosures, together with the restraint system, would prevent damage to the main steam isolation valves in the event of a postulated double-ended break. Protection from critical crack jets would be provided by means of deflector plates. In addition, these valves are fail-safe.

Careful investigation has revealed that the required operation of the main steam isolation valves would not be impaired in the event of the postulated rupture.

3.6.10 Main steam safety valves --

The main steam safety valves are located in Room 81 of the auxiliary building as shown in Figure M-1.

As in the case of the main steam isolation valves, the protective enclosures around the main steam and feedwater lines would provide adequate protection for these valves in the event of the postulated longitudinal break. These enclosures, working in conjunction with the restraint system, would prevent damage to the main steam safety valves in the event of the postulated double-ended break. Protection from critical crack jets would be provided by deflector plates where necessary.

It can be seen that the main steam safety valves would still function properly in the event of the postulated rupture of a major high energy line outside the containment.

3.6.11 Safety injection system --

The pumps, valves and other components of the safety injection system are so located that they cannot be damaged in the event of the postulated rupture. The only vulnerable parts of the system are the electrical cables and the rooms through which they pass. The adequate protection of these rooms is discussed elsewhere in Section M.3.

It has been determined, therefore, that the availability and proper operation of the safety injection system would be ensured in the event of the postulated rupture.

3.6.12 Raw water system --

The only parts of the raw water system that are located near a major high energy line are two 16 inch diameter raw water supply and return headers in the north end of Room 81. These lines provide back-up cooling water for the containment air cooling and filtering units.

The protective enclosures around the main steam and feedwater lines would prevent damage to the raw water pipes in the event of the postulated rupture. In the event of the postulated double-ended break, the restraint system would act to control the movement of the major high energy lines and would thereby stop damage to the two raw water pipes from this source. The effect of the jet force that would be imposed on these two pipes in the event of a critical crack has been investigated. It is not considered possible that the critical crack jet would affect the proper operation of the raw water system. The raw water pipes would not be adversely affected by the pressurization or flooding of Room 81.

It can be seen, therefore, that the required functioning of the raw water system would not be impaired by the postulated rupture of a major high energy line outside the containment.

3.6.13 Pressurizer pressure and level control --

The pressurizer pressure and level control transmitters, namely A/, B/, C/ and D/PT-102, PT-103X, PT-103Y, LT-101X and LT-101Y and their sensing lines are located within the reactor containment. The related controls are located in the control room. The room through which the cables, for the instruments, are routed would not be damaged by the postulated high energy line rupture. The proper operation of the pressurizer pressure and level control system in the event of the postulated high energy line ruptures is thus assured.

4. HIGH ENERGY SYSTEMS OTHER THAN MAJOR HIGH ENERGY SYSTEMS

4.1 Identification of High Energy Systems Other Than Major High Energy Systems

All systems outside the containment whose service temperatures exceed 200°F or whose design pressures exceed 275 psig are considered to be high energy systems. For the purpose of this investigation, those systems which are not normally pressurized were excluded from consideration. The main steam and feedwater systems have already been identified as the major high energy systems since they are the two systems, because of line sizes, fluid energy levels and plant arrangement, which would have the greatest potential to inhibit a safe shutdown of the plant in the event of the postulated pipe rupture incident (see Section M.3). The other high energy systems, because of smaller line sizes, lower fluid energy and plant arrangement, offer a lower potential for hindering a safe shutdown of the facility in the event of the postulated rupture. These other high energy systems are the following:

1. Charging
2. Letdown
3. Auxiliary steam
4. Condensate return
5. Steam generator blowdown
6. Sampling
7. Nitrogen
8. Hydrogen
9. Auxiliary Feedwater System (Non Safety Class Portion)

A study was performed to determine what modifications were necessary to protect essential structures and equipment from a postulated rupture in one of these systems. The results of the study, including the effects of the postulated break, are discussed below.

4.2 Effects of Postulated Rupture on Essential Structures and Equipment

4.2.1 Charging --

It was determined that light steel deflector plates would be required to protect nearby safeguard cables from the effects of jet impingement. Installation of these deflector plates has been completed. With these modifications in place, the postulated rupture of a charging line would not adversely affect the safe shutdown capability of the plant.

4.2.2 Letdown --

An investigation indicated that the postulated break of the high energy portion of the letdown line would not adversely affect the safe shutdown capability of the plant.

4.2.3 Auxiliary Steam --

A study indicated that there were many areas in which the auxiliary steam system had the potential to impinge on safeguard cables. The modifications required to protect cable trays from the effects of the postulated crack were light steel deflector plates. Installation of these items has been completed. With these modifications installed, the postulated crack of an auxiliary steam line would not adversely affect the safe shutdown capability of the facility.

4.2.4 Condensate return --

A study indicated that there were certain areas in which the condensate return system, although the fluid energy levels are relatively low, could have a potential for impinging on safeguards cables. The modifications required to protect safeguards cables from the effects of the postulated crack were light steel deflector plates. Installation of these modifications has been completed. With these modifications installed, the postulated crack of a condensate return line would not adversely affect the safe shutdown capability of the plant.

4.2.5 Steam generator blowdown --

An investigation indicated that the postulated pipe rupture in the steam generator blowdown system would not adversely affect the safe shutdown capability of the plant.

4.2.6 Sampling --

An investigation indicated that the postulated pipe rupture in the sampling system would not adversely affect the safe shutdown capability of the plant.

4.2.7 Nitrogen --

An investigation indicated that the postulated rupture of a pipe in the nitrogen system would not adversely affect the safe shutdown capability of the plant.

4.2.8 Hydrogen --

An investigation indicated that the postulated pipe rupture in the hydrogen system would not adversely affect the safe shutdown capability of the facility.

4.2.9 Auxiliary Feedwater --

An investigation indicated a postulated pipe rupture (critical crack) in the non safety grade portion of the auxiliary feedwater system would not adversely affect the safe shutdown capability of the plant.

It should be noted that the light steel deflector plates discussed above are different from Cable Tray Covers[11]. The cable tray covers are not taken credit for in HELB and are not a design basis requirement. Refer to Drawings D-4111 and D-4137 for locations of light steel deflector plates.

5. SINGLE FAILURE CRITERIA

5.1 General

An investigation was performed to determine the consequences of the imposition of the single failure criteria on the postulated rupture of a line containing high energy fluid outside the containment with the concurrent loss of off-site power. The study indicated that the only single failure which would have affected the safe shutdown capability of the plant was a failure of the motor driven auxiliary feedwater pump concurrent with the postulated steam line break. Since the determination of this problem, however, the steam supply system to the turbine driven auxiliary feedwater pump was modified.

As discussed in Section M.3.5.5, the steam supply system to the turbine driven auxiliary feedwater pump was modified such that it now consists of a separate line from each main steam line. The connection to the main steam line is upstream of the main steam isolation valves. The routing and support of the two steam supply lines is such that the postulated rupture of one main steam line cannot damage the steam supply line associated with the other main steam line. Each steam supply line has a remotely operated isolation valve and a check valve. Each check valve is located as close as practicable to the junction of the two supply lines at the turbine driven auxiliary feedwater pump. This ensures the availability of an uninterrupted source of steam for the turbine driven feedwater pump in the event of the postulated pipe rupture incident.

The study indicated that with this modification the facility would retain a fully adequate safe shutdown capability in the event of a single failure of an active component imposed on the postulated pipe rupture incident.

6. SUMMARY OF EMERGENCY OPERATING PROCEDURES

The postulated rupture of a main steam or feedwater line which results in a reactor trip, requires the implementation of the emergency operating procedures. The postulated rupture of the other high energy lines which do not result in a reactor trip, because of smaller line size, lower fluid energy level, and plant arrangement would be classified as off-normal operation. The abnormal operating procedure require isolation of the ruptured line (if possible) and should a reactor trip occur or be required, implementation of the emergency operating procedures.

The Emergency Operating Procedure for an Uncontrolled Heat Extraction provides operator action for an event which leads to an unexpected, rapid increase in Steam Generator steam flow or loss of Steam Generator inventory and subsequently requires and/or results in a reactor trip. The Uncontrolled Heat Extraction Procedure provides the operator actions which must be accomplished in the event of a steam line or feedline break resulting in the blowdown of the affected Steam Generator. These actions are implemented after performing the immediate actions in the Reactor Trip Emergency Operating Procedure and after an Uncontrolled Heat Extraction has been diagnosed. The actions in the Uncontrolled Heat Extraction Procedure are necessary to ensure the plant is placed in a safe stable condition. This procedure also ensures that the Emergency Plan is implemented. The Uncontrolled Heat Extraction Procedure directs the operator to cooldown the plant and to enter shutdown cooling when $T_H \leq 300°F$ and Pressurizer Pressure ≤ 250 psia.

The modifications that were made to the high energy lines and the subsequent analysis show that no special emergency procedures are necessary to cope with the postulated ruptures. The required equipment and operator action areas have been protected from the consequences of such an event.

Attachment A - "Nuclear Services Corporation"

Summary

This report, prepared for Gibbs, Hill, Durham & Richardson, Inc., presents the results of pipe rupture analyses of the main steam piping outside the containment for Fort Calhoun Station, Unit 1. The analyses were performed in accordance with, and in response to, the Atomic Energy Commission letter to Mr. J. L. Wilkins, Omaha Public Power District, dated December 14, 1972 with its attachments, and the amendment dated January 11, 1973. Included were stress analyses of the piping for the purpose of determining the design basis break locations, the selection of the locations in accordance with the criteria provided in the attachment to the aforementioned AEC letter. An evaluation was made for potential pipe whip and resulting potential damage to adjacent safety systems. At those break locations deemed by Gibbs, Hill, Durham & Richardson, Inc., to have potential for such damage, computerized nonlinear, dynamic pipe whip analyses were performed on the piping/restraint system to confirm design adequacy of the restraint system to prevent such whip.

Subsequent to the preparation of Attachment A by Nuclear Services Corporation, EAS Energy Services prepared Attachment D, which requalified only the B31.1 region of the Main Steam piping outside containment. Consequently, Attachment A now only qualifies the B31.7 region of the Main Steam piping outside containment.

Attachment B - "Nuclear Services Corporation"

Summary

This report, prepared by Gibbs, Hill, Durham, & Richardson, Inc., presents the results of pipe rupture analyses of the feedwater piping outside the containment for Fort Calhoun Station, Unit 1. The analyses were performed in accordance with, and in response to, the Atomic Energy Commission letter to Mr. J. L. Wilkins, Omaha Public Power District, dated December 14, 1972 with its attachments, and the amendment dated January 11, 1973. Included were stress analyses of the piping for the purpose of determining the design basis break locations, the selection of the locations in accordance with the criteria provided in the attachment to the aforementioned AEC letter. An evaluation was made for potential pipe whip and resulting potential damage to adjacent safety systems. At those break locations deemed by Gibbs, Hill, Durham & Richardson, Inc., to have potential for such damage, computerized nonlinear, dynamic pipe whip analyses were performed on the piping/restraint system to confirm design adequacy of the restraint system to present such whip.

Subsequent to the preparation of Attachment B by Nuclear Services Corporation, EAS Energy Services prepared Attachment E, which requalified only the B31.1 region of the Feedwater piping outside containment. Consequently, Attachment B now only qualifies the B31.7 region of the Feedwater piping outside containment.

Attachment C - "Main Steam and Feedwater Piping Protective Enclosures"

Design Summary

Those portions of the main steam and feedwater lines within Room 81 which are considered susceptible to a large break were provided with a protective enclosure designed to contain the steam or fluid to the extent that severe impingement of jets on safety-related equipment or structures cannot occur.

The protective enclosures consist of a series of longitudinal flat steel bars placed around the outside of the pipe insulation and held in position by a series of independent loops of wire rope.

Design Assumptions - Main Steam Line Enclosures

The critical break for the design of the enclosure is the longitudinal large break. In accordance with AEC criteria this is taken to be a slot break with a length equal to twice the pipe diameter and an area equal to the inside area of the pipe. In the case of the main steam line the area of the break is 531 square inches. The maximum reactive force of the steam issuing through this break was obtained by means of a time-dependent dynamic analysis. The maximum calculated force is 559 kips.

Initially the steam jet is considered to impinge either on one or on two contiguous longitudinal bars. Under an assumed instantaneous application of load the bars would deform plastically at a load of 1.2 times the maximum jet force based on the minimum specified yield strength of the bars. Plastic deformation is limited to approximately twice the elastic deformation.

For this initial phase the wire rope was designed to resist a force equal to twice the jet force on the assumption that full reversal of the jet occurs. Based on the yield design of the bars this force is increased by a factor of 1.2 and by an additional factor of 1.25 to account for variation in the yield strength of the bars. Any effect of the insulation in reducing the dynamic effect is conservatively disregarded.

Following this initial phase the entire annulus between the pipe and the enclosure is filled with steam. In this case there will be leakage through the bars due to initial small gaps and widening of these gaps as the wire rope stretches under load. As a result, the pressure within the annulus will be less than the initial pressure within the pipe. Two cases are considered for design of the wire rope. The first case is based on uniform pressure within the annulus. The second case is based on a combination of jet load on the bars directly in front of the assumed break and a uniform pressure within the remainder of the annulus.

Within several hundredths of a second the jet load reduces to a relatively steady state value. The load cases described above are considered for the appropriate reduced load and pressures. For these cases the wire rope is assumed to be at a temperature of 500°F and the reduced strength of the wire rope corresponding to this temperature is used.

At the main steam safety valves the spacing of the wire rope is increased to clear the valve nozzles and their reinforcing plates. At these locations the longitudinal enclosure bars are thickened and reinforced with stiffener plates to conform to the criteria given above. The wire rope size is increased and the spacing immediately adjacent to the nozzles is decreased to compensate for the increased spacing between wire rope loops in this area.

In order to maintain the configuration of the bars following pipe rupture, the individual bars are tightly fastened to the wire rope by means of wire rope clips. Each bar is fastened by clips at a spacing of approximately one foot.

The maximum gap that may form between individual bars during pressurization is conservatively calculated by assuming that only one or two bars are loaded by the jet and that the wire rope slides freely through the clips fastening the wire rope to the individual bars so that the entire stretch of the wire rope is concentrated at two gaps. For this calculation, the yield strength of the bars is taken as 45 ksi and the modulus of elasticity of the wire rope as 11×10^6 psi. Assuming a 10% reduction for the effects of the clamping force of the clips and friction between the wire rope and the bars over its entire perimeter, a maximum gap of 0.62 inches results. Wherever such a gap could result in a possible jet impingement on a safety-related item a 1/8" thick batten plate is provided under the bars to cover this gap and prevent release of a jet.

Feedwater Line Enclosure

The design assumptions for the protective enclosure around the feedwater lines are similar to those for the main steam line. The loads involved however, are far smaller.

Attachment D - "EAS Energy Services"

Summary

This report, prepared for Omaha Public Power District, presents the results of piping stress analysis of the Main Steam B31.1 piping outside the containment for Fort Calhoun Station, Unit 1 contained in EAS Energy Services Calculation No. 185-90-P-MS-001, Revision 0, dated 2-28-91. The analysis was performed in accordance with the ASME Boiler and Pressure - Section III (Division 1) - 1980 Subsection NC Piping Code. The criteria used for determining postulated pipe rupture locations was NRC Generic Letter 87-11.

Based on the results of this piping analysis, no intermediate postulated pipe rupture locations are required on any of the Main Steam B31.1 piping outside the containment.

Attachment E - "EAS Energy Services"

Summary

This report, prepared for Omaha Public Power District, presents the results of piping stress analysis of the Feedwater B31.1 piping outside the containment for Fort Calhoun Station, Unit 1 contained in EAS Energy Services Calculation No. 185-90-P-FW-002, Revision 1, dated 3-11-91. The analysis was performed in accordance with the ASME Boiler and Pressure - Section III (Division 1) - 1980 Subsection NC Piping Code. The criteria used for determining postulated pipe rupture locations was Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid Systems Piping Inside and Outside Containment," Revision 2, dated June 1987.

Based on the results of this piping analysis, no intermediate postulated pipe rupture locations are required on any of the Feedwater B31.1 piping outside the containment.

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APPENDIX N

RECLASSIFICATION OF SYSTEMS

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List of Tables

1. PURPOSE

The purpose of this appendix is to classify the FCS mechanical systems based upon regulatory and industry guidance (RG 1.26[1] and ANSI/ANS-51.1[2]) and to establish the corresponding piping and component codes of construction. The original licensing basis for the Fort Calhoun Station (FCS) did not categorize systems based on safety functions which current industry standards and NRC guidance documents have promulgated. The FCS safety-related systems are designated as Critical Quality Elements (CQE) and the Non-Nuclear Safety Related Systems are designated as either Limited-Critical Quality Elements (Limited CQE) or Non-CQE depending on the system's importance on plant safety.

Current industry standards (RG 1.26 and ANSI/ANS-51.1), correlate system safety classes to ASME Section III code classes. Because of the similarity between the ASME Section III and B31.7 code classes Appendix N will correlate the Safety Classes with B31.7 code classes.

The following sections provide the safety classification criteria, safety class barrier interface criteria, correlation between safety class and equipment design, and quality assurance requirements.

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1 Regulatory Guide 1.26, revision 3, 1976, "Quality Group Classifications and Standards for water, steam, and radioactive waste containing components of nuclear power plants"

2 ANSI/ANS-51.1, 1983, "Nuclear Safety Criteria for the design of stationary pressurized water reactors"

2. SAFETY CLASSIFICATION CRITERIA

American National Standard ANSI/ANS-51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactors" is the guidance document utilized in the establishment of the system safety classification criteria. Based upon this standard, original and replacement mechanical equipment shall be assigned to one of five Safety Classes (SC) identified as, SC-1, -2, -3, Non-Nuclear Safety (NNS) CL-1 or -2 in accordance with definitions provided herein. Assignments shall be made for all mechanical equipment of the nuclear power plant. Classification shall be done on the basis of definitions contained in section N.2.1 and the interface requirements of section N.3.0.

When the interface requirements described herein are not met, the equipment shall be assigned to the more stringent Safety Class corresponding to functions of the connected equipment or justification for other defined bases shall be provided (see N.3.4). Table N-1 provides the primary safety class designations and safety class interfaces for the FCS mechanical systems.

2.1 Safety Classes

Equipment shall be assigned to one of the five classes in accordance with N.2.1.1 through N.2.1.5. The equipment assigned to SC-1, -2, or -3 is that relied upon in the plant design to accomplish nuclear safety functions.

When more than one system is capable of accomplishing a nuclear safety function and one system, on its own, satisfies all nuclear safety-related systems requirements (e.g., redundancy, diversity, capacity), the latter shall be classified to the corresponding Safety Class and the additional system as NNS.

Equipment assigned to one of the five safety classes shall meet the minimum requirements as prescribed in N.4.0. Equipment or components with more stringent design requirements than those corresponding to the applicable safety class may be utilized, however, the design code/safety class designation as set forth in Table N-2 shall be retained.

Where a single item of equipment, or a portion thereof, provides two or more functions or classes, it shall be classified to the more stringent class. Different portions of the same equipment (e.g., the tube side versus the shell side of a heat exchanger) may perform different functions and be assigned to different classes provided the equipment contains a suitable interface boundary meeting the requirements of N.3.0.

A support shall be classified to the more stringent class corresponding to the function provided either (a) by the supported equipment, unless failure of the support could not jeopardize the function of the supported equipment; or (b) directly by the support itself (e.g., a Seismic Category II/I condition). Exceptions may be made for intermediate elements (e.g., load-bearing housings of electric motors, valve operators, heat exchangers, or diesel engines) and access structures, whether of concrete or structural steel that carry the weight of, or provide structural stability to SC-1, -2, or -3 equipment, either directly or through piping supports. Such intermediate elements shall be classified as SC-3 or NNS in accordance with either N.2.1.3h or N.2.1.4f, as applicable.

2.1.1 Safety Class 1

Safety Class 1 (SC-1) shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the Reactor Coolant Pressure Boundary (RCPB) whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability. The RCPB is defined in 10CFR50.2 of the Code of Federal Regulations as being all pressure-containing components, such as pressure vessels, piping, pumps, and valves, which are:

- Part of the reactor coolant system. In this context the reactor coolant system includes the reactor coolant loops between and including the primary side of the steam generators, reactor coolant pumps, reactor vessel and pressurizer or

- Piping connected to the reactor coolant system up to and including any and all of the following:

    a. The outermost containment isolation valve in system piping which penetrates primary reactor containment.

    b. The second of two valves which are normally closed during normal reactor operation in system piping which does not penetrate the primary reactor containment.

    c. The reactor coolant system safety and relief valves.

Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary may be of a lesser safety class provided the component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and one open). Each open valve must be capable of automatic or remote actuation and, assuming the other valve is open, its closure time must be such that in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

2.1.2 Safety Class 2

Safety Class 2 (SC-2) shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment that is not included in SC-1 and is designed and relied upon to accomplish the following nuclear safety functions:

a. Provide fission product barrier or primary containment radioactive material holdup or isolation.

b. Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g., containment spray).

c. Introduce emergency negative reactivity to make the reactor subcritical (e.g., boron injection system), or restrict the addition of positive reactivity via pressure boundary equipment.

d. Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., residual heat removal and emergency core cooling)

e. Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., SIRWT).

2.1.3 Safety Class 3

Safety Class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:

a. Provide for functions defined in SC-2 where equipment, or portions thereof, are not within the scope of the USAS B31.7 piping code.

b. Provide secondary containment radioactive material holdup, isolation, or heat removal.

c. Except for primary containment boundary extension function, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits.

d. Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room) containing SC-1, -2, or -3 equipment.

e. Introduce negative reactivity to achieve or maintain subcritical reactor conditions (e.g., boron makeup).

f. Provide or maintain sufficient reactor coolant inventory for core cooling (e.g., reactor coolant normal makeup system).

g. Maintain geometry within the reactor to ensure core reactivity control or core cooling capability (e.g., core support structures).

h. Structurally load-bear or protect SC-1, -2, -3 equipment.[3]

i. Provide radiation shielding for the control room or offsite personnel.

j. Ensure required cooling for liquid-cooled stored fuel (e.g., spent fuel storage pool and cooling system).

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3 This applies to concrete or steel structures that are not within the scope of the piping code or component supporting structure.

k. Ensure nuclear safety functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal of SC-1, -2, or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, or provide fuel oil to the emergency diesel engine).

l. Provide an acceptable environment for SC-1, -2, or -3 equipment and operating personnel.

2.1.4 Non-Nuclear Safety Class 1

Non-Nuclear Safety Class 1 (NNS CL-1) shall apply to equipment that is not included in SC-1, -2, or -3 that is designed and relied upon to accomplish one or more selected, but limited, requirements specified to ensure acceptable performance of specific NNS functions. The selected requirements are established on a case-by-case basis commensurate with the specific NNS function performed. The functions performed by the NNS Class 1 equipment are:

a. Process, extract, encase, or store radioactive waste.

b. Provide cleanup of radioactive material from the reactor coolant system or the fuel storage cooling system for normal operations,

c. Extract radioactive waste from, store, or transport for reuse irradiated neutron absorbing materials (e.g., boron compounds).

d. Resist failure that could prevent any SC-1, -2, or -3 equipment from performing its nuclear safety function.

e. Structurally load-bear or protect NNS equipment providing any of the functions listed in N.2.1.4.

f. Provide permanent shielding for protection of SC-1, -2, or -3 equipment or of onsite personnel.

g. Provide operational, maintenance or post-accident recovery functions involving radioactive materials without undue risk to the health and safety of the public.

h. Following a control room evacuation, provide an acceptable environment for SC-1, -2, or -3 equipment required to achieve or maintain a safe shutdown condition.

i. Handle spent fuel, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel.

j. Ensure reactivity control of stored fuel.

k. Protect SC-2 or -3 equipment necessary to attain or maintain safe shutdown following fire.

2.1.5 Non-Nuclear Safety Class 2

Non-Nuclear Safety Class 2 (NNS CL-2) shall apply to equipment that is not included in SC-1, -2, -3 or NNS CL-1. This equipment is not relied upon to perform a nuclear safety function.

3. SAFETY CLASS INTERFACES

If failure of Safety Class or NNS equipment connected to other Safety Class equipment could prevent the latter equipment from accomplishing its nuclear safety function, an interface barrier or isolation device shall be provided to protect the latter equipment.

3.1 Safety Class Interfaces for Pressure Integrity of Fluid Systems

A membrane (e.g., heat exchanger tube, bellows, piston, o-ring, blind flange) that provides pressure boundary separation of equipment classes serves as the required barrier between any two classes, and shall be categorized to the more stringent class. Where equipment of differing classes is interconnected, the more stringent class extends to and includes the cited barriers. Otherwise, the interface shall be in accordance with the following criteria.

Interface barriers or isolation devices connecting Safety Class or NNS Fluid system equipment to other Safety Class equipment shall be capable of limiting the loss of fluid from the latter equipment if the former were to fail. The fluid loss shall be limited such that (1) if the fluid contains radioactivity, the applicable dose criteria[4] are met and (2) if the fluid performs a nuclear safety function (e.g., heat transfer) sufficient inventory and pressure are maintained to perform that function. The closure time of open valves and the limitation of losses due to flow restrictions shall be evaluated when determining the loss of fluid. The loss of fluid shall be assessed for the full duration of the event. The loss of fluid limitation shall be met applying the single failure criterion to the actuation of the interface barriers or isolation devices.

4 The applicable dose criteria is provided in USAR, Section 14.

3.2 Reactor Coolant Pressure Boundary Interface

The reactor coolant pressure boundary (RCPB) interface from SC-1 to SC-2, 3 or NNS (as applicable) shall be as follows per the rules delineated in paragraph N.2.1.1 and N.3.1:

a. One Code Safety or Relief Valve

b. The second of the two valves normally closed during power operation.

c. The second of two valves which are capable of automatic or remote actuation with closure times sufficient to allow the reactor to be shutdown and cooled down in an orderly manner assuming make-up is provided by the reactor coolant make-up system only.

d. A flow restrictor sized to limit the flow though the postulated failure of the lower safety class component such that the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

3.3 Primary Containment Penetration Interface

Generally, any fluid system penetration through the primary containment and its corresponding primary containment isolation devices shall be no less than SC-2. Where system piping that penetrates the primary containment boundary is assigned a less stringent safety class than SC-2 (i.e., SC-3) on either side of the penetration, then the interface to SC-2 shall be at its connection to a primary containment penetration assembly. Detailed containment isolation provisions are provided in USAR Section 5.9.

3.4 Exceptions To The Defined Safety Class Interfaces

Exceptions to the safety class interface criteria defined in N.3.1, N.3.2 or N.3.3 with the explanation for the acceptability for the other defined basis are as follows:

Exception to Section N.3.1 second paragraph is taken for the isolation of Component Cooling Water (CCW) to the Vacuum Deaerator, and Primary Sample Cooler which is provided solely by valves HCV-2895A/B. These single remote manual valves provide the SC-3 to NNS CL-1 interface. Credit is taken for seismically supported NNS CL-1 USAS B31.1 piping and non-CQE isolation valves at the equipment served by the CCW. This is considered an acceptable substitute for a second remote manual valve for this limited section of NNS CL-1 piping.

3.5 Safety Class Interface for Other Equipment

The classification requirements for supports and mechanical movers or links shall be as required in N.2.1. No interface barrier is required when interconnected mechanical equipment consists of differing Safety Classes, mechanical equipment connected to structures or electrical equipment, or electrical equipment connected to structures.

Safety class interfaces between NNS CL-1 and NNS CL-2 shall be a single closed manual valve, check valve or flow restricting device. An open single remote manual valve is also an acceptable interface between these safety classes. An exception to this criteria is a single open manual valve, which can be used as a class boundary interface if operator action has been shown to be adequate to allow the NNS CL-1 system to perform its design function. An example of this exception is the Fire Protection System.

4. CORRELATION BETWEEN SAFETY CLASS AND EQUIPMENT DESIGN CODE

The USAS B31.7 "Nuclear Power Piping" 1968 Draft, defined Class I, II, and III prior to the development of ASME Section III Class 1, 2, and 3 for piping in the early 1970's. As the ASME piping code was an outgrowth of USAS B31.7, the Class I, II, and III designations of B31.7 correlate directly with Class 1, 2, and 3 of ASME Section III. ANSI/ANS-51.1 makes a direct correlation between Safety Classes and ASME Section III classes, hence the following correlation is made:

Safety Class	B31.7 Class	ASME Section III Class
1	I	1
2	II	2
3	III	3

The USAS B31.7 code constitutes the FCS design code for piping, pipe fittings, and supports but does not include piping system components (valves, heat exchangers, tanks, pumps, etc). The design codes for the piping system components, as defined above, will remain as specified in the original contract specifications, purchase orders, and USAR.

The original piping code of construction (USAS B31.1 with Code cases and ASME Section III NDE for the RCS loop and USAS B31.7 Class I for the remaining Safety Class 1 piping as defined in this Appendix N) shall be retained for all future plant modifications. For several systems the original piping "as-built" is B31.7 Class I for Safety Class 2 or 3 systems as defined in Table N-1, or B31.7 Class II now defined as NNS CL-1 or NNS CL-2. The original piping code of construction for these systems is reclassified to the piping code corresponding to safety class shown in Table N-1.

Component replacements shall meet the requirements of either the original code of construction or later editions of the construction code, e.g., ASME Section III for piping and fittings. Replacement components may meet all or portions of the requirements of later editions of the construction code provided the provisions of ASME Section XI Article IWA-7200 1980 Edition Winter 1980 Addenda are met.

Table N-2 contains a correlation between safety class, equipment design (code of construction), and quality assurance.

The Seismic Category I requirements specified for specific plant systems is shown in USAR Appendix F. Safety Class 1, 2, and 3 equipment is considered to be Seismic Category I.

5. QUALITY ASSURANCE

All mechanical equipment designated as SC-1, -2, -3 or NNS CL-1, 2 that provide functions listed in N.2.1.1 through N.2.1.5 have quality levels assigned to ensure the probability that the included structures, systems, and components will perform their functions. USAR Appendix A describes the OPPD Quality Assurance (QA) program for Fort Calhoun Station (FCS). The QA program is applied to CQE systems, (components designated Safety Class 1, 2, 3) and some of the Non-Nuclear Safety Class 1 systems such as Fire Protection, Radioactive Waste, and Limited CQE. The remaining Non-Nuclear Safety Class 1 and Non-Nuclear Safety Class 2 are termed Non-CQE with Special Requirements and Non-CQE respectively and have no formalized quality assurance program. Table N-2 correlates the level of quality assurance with respect to safety class.

Table N-1 - "System Safety Classification"

P&ID No.	System	Safety Class
11405-M-1	Containment HVAC	
	• Containment Penetrations	SC-2
	• Post-Accident H_2 Control	SC-3
	• Containment Air Cooling & Filtering	SC-3
	• Balance of System	NNS CL-1
11405-M-2	Auxiliary Bldg. HVAC	
	• Flow path from fuel storage area through filter VA-66 to stack	NNS CL-1
	• HVAC equip. and ductwork to cool safety related equipment	NNS CL-1
11405-M-5	Demineralized Water	
	• Containment Penetrations	SC-2
	• Connections to Safety Injection and Containment Spray Pumps	SC-2
	• Balance of System	NNS CL-2
11405-M-6	Waste Disposal System	
	• Containment Penetrations	SC-2
	• Interface with SIRWT	SC-2
	• Balance of System	NNS CL-1&2
11405-M-7, 8,9	Waste Disposal System	
	• Containment Penetrations	SC-2
	• Balance of System	NNS CL-1&2

Table N-1 (Continued)

P&ID No.	System	Safety Class
11405-M-10	Component Cooling System	
	• Control Room A/C piping	SC-3
	• Balance of System	SC-3
11405-M-11	Spent Fuel Pool Cooling	
	• Pool Cooling Water System	SC-3
	• Fuel Transfer Drain Pump Piping	SC-2
	• Spent Fuel Pool Cooling Emergency Cross-Tie Piping	NNS CL-1
11405-M-12	Primary Plant Sampling	
	• Containment Penetrations	SC-2
	• Sampling System Outside Containment	NNS CL-1&2
	• Volume Control Tank Piping	SC-3
	• Sample System Inside Containment	SC-2
	• Reactor Coolant Pressure Boundary	SC-1
11405-M-13	Plant Air	
	• Containment Penetration	SC-2
	• Balance of System	NNS CL-2

Table N-1 (Continued)

P&ID No.	System	Safety Class
11405-M-40	Component Cooling System	
	• Containment Penetrations	SC-2
	• Containment Air Cooling Piping inside Containment	SC-2
	• RCP Seal and Lube Oil Cooler Piping inside Containment	SC-2
	• Piping adjacent to the RCP Seal Coolers	SC-2
	• SI Leakage Cooler and Detector Well Cooler Piping	NNS CL-2
	• Balance of System	SC-3
11405-M-42	Nitrogen, Hydrogen, Methane, Propane, and Oxygen Gas	
	• Containment Penetrations	SC-2
	• Nitrogen Piping connected to Safety Injection Tanks	SC-2
	• Nitrogen Piping connected to Component Cooling Surge Tank	SC-3
	• Nitrogen Piping connected to Emergency FW Storage Tank	SC-3
	• Nitrogen and Hydrogen Piping adjacent to the Volume Control Tank	SC-2
	• Piping Connected To Waste Disposal System	NNS CL-1
	• Balance of System	NNS CL-2

Table N-1 (Continued)

P&ID No.	System	Safety Class
11405-M-97	Misc. HVAC System	
	• Control Room HVAC	SC-3
	• Emerg. Diesel Gen. Air Inlet, Radiator Exhaust Dampers, and Duct	SC-3
	• Balance of System	NNS CL-2
11405-M-98	Waste Disposal System	
	• Containment Penetrations	SC-2
	• Piping connection from Volume Control Tank	SC-3
	• Balance of System	NNS CL-1&2
11405-M-99	Waste Disposal System Aux. Bldg. Floor Drains	NNS CL-2
11505-M-100	Raw Water	
	• Containment Penetration Piping to and from Containment Cooling Units	SC-2
	• Raw Water Supply Piping	SC-3
	• Discharge Piping in the Aux. Bldg.	SC-3
	• Discharge piping in the Turbine Bldg.	NNS CL-1
	• Balance of System	NNS CL-2
11405-M-119	Component Cooling CEDM	
	• Component Cooling Supply and Return Piping	SC-2
	• Seal Leakage Piping	NNS CL-2
11405-M-252	Steam System	
	• Steam Piping from Steam Generators to Main Steam Isolation Valves	SC-2
	• Steam piping to Auxiliary Feedwater Pump Turbine Driver	SC-3
	• Steam Exhaust from Auxiliary Feedwater Pump Turbine Driver	SC-3
	• Balance of System	NNS CL-2

Table N-1 (Continued)

P&ID No.	System	Safety Class
11405-M-253	Steam Generator Feedwater and Blowdown	
	• Blowdown from Steam Generators through Containment Penetrations	SC-2
	• Balance of Blowdown System	NNS CL-2
	• Main Feedwater from Feedwater Isolation Valves to Steam Generator	SC-2
	• Balance of Main Feedwater System including Diesel Driven Feedwater Pump	NNS CL-1
	• Auxiliary Feedwater System from Containment Penetrations to Steam Generators	SC-2
	• Steam and Electric Drive Auxiliary Feedwater System from Storage Tank to Containment Isolation Valves	SC-3
	• Balance of System	NNS CL-1&2
11405-M-254	Condensate System	
	• Emergency Feedwater Storage Tank and Supply Piping to Auxiliary Feedwater Pumps	SC-3/NNS CL-1
	• Balance of Condensate System	NNS CL-1&2
11405-M-261	Condenser Evacuation	NNS CL-2
11405-M-262	Fuel Oil System/Lube Oil System	
	• Emergency Diesel Generator Fuel Oil System	SC-3
	• Diesel Driven Auxiliary Feedwater Pump Fuel Oil System	NNS CL-1
	• Balance of Systems	NNS CL-2
11405-M-264	Instrument Air System	
	• Containment Penetration	SC-2
	• Air Supply to SIRWT Bubblers	SC-3
	• Air Supply to AOV's Requiring Actuation Post DBA	SC-3
	• Balance of System	NNS CL-2

Table N-1 (Continued)

P&ID No.	System	Safety Class
11405-M-266	Fire Protection	
	• Control Room Charcoal adsorbers	SC-3
	• Safe Shutdown Plant Areas	NNS CL-1
	• Non-Safe Shutdown Plant Areas	NNS CL-2
627-D-8053	Waste Evaporator Package (abandoned in place)	NNS CL-1
C-4175	Typical Valve Air Source Valve Configurations	
	• Air Source for Safety Related Valves Required to Hold or Change Position	SC-3
	• All Other Air Sources for Valve Actuation	NNS CL-2
D-4078	Reactor Coolant Gas Vent System	
	• Reactor Coolant Pressure Boundary up to Orifice	SC-1
	• Piping from Orifice to HCV-180 and 181	SC-2
	• Piping Down Stream of HCV-180 and 181	NNS CL-1
E-4144	FW-10 Lube Oil Schematic	
	• Critical Lube Oil components	SC-3
	• All Other Components	NNS CL-2
E-23866-210-110	Reactor Coolant System	
	• Main R. C. Loop Pressure Boundary	SC-1
	• Pressurizer and Main Piping	SC-1
	• Pressurizer Safety and Relief Valve Discharge, Quench Tank and Tank Relief	NNS CL-1
	• Balance of System	NNS CL-2

Table N-1 (Continued)

P&ID No.	System	Safety Class
E-23866-210-120	Chemical and Volume Control System	
	• Reactor Coolant Pressure Boundary Piping	SC-1
	• Charging Flowpath through Regenerative Heat Exchanger (Shell Side)	SC-2
	• Containment Penetrations	SC-2
	• Charging Pump Flowpath from Boric Acid Supply to Containment Penetration	SC-2
	• Volume Control Tank and Associated Piping	SC-3
	• Chemical Addition Tank and Metering Pump Flow Path	NNS CL-2
	• Alternate Flowpath from S.I. and Refueling Water Tank through Valve LCV-218-3	SC-2
	• Relief Valve Discharge and Drain Piping to Waste Disposal and Pressurizer Quench Tank	NNS CL-1
	• Flowpath through Letdown Heat Exchanger Outside Containment	SC-3
	• CVCS Ion Exchangers and Associated Piping	SC-3
	• Resin Addition Tank and Drain Piping to Waste Disposal	NNS CL-1
	• Demineralized Water Supply Piping	NNS CL-2
	• Balance of System	NNS CL-2
E-23866-210-121	Chemical and Volume Control System	
	• Boric Acid Flowpath from Storage Tanks to Charging Pumps	SC-2
	• Boric Acid Batching Tank and Associated Piping	NNS CL-2
	• Demineralized Water Connection and Balance of System	NNS CL-2

Table N-1 (Continued)

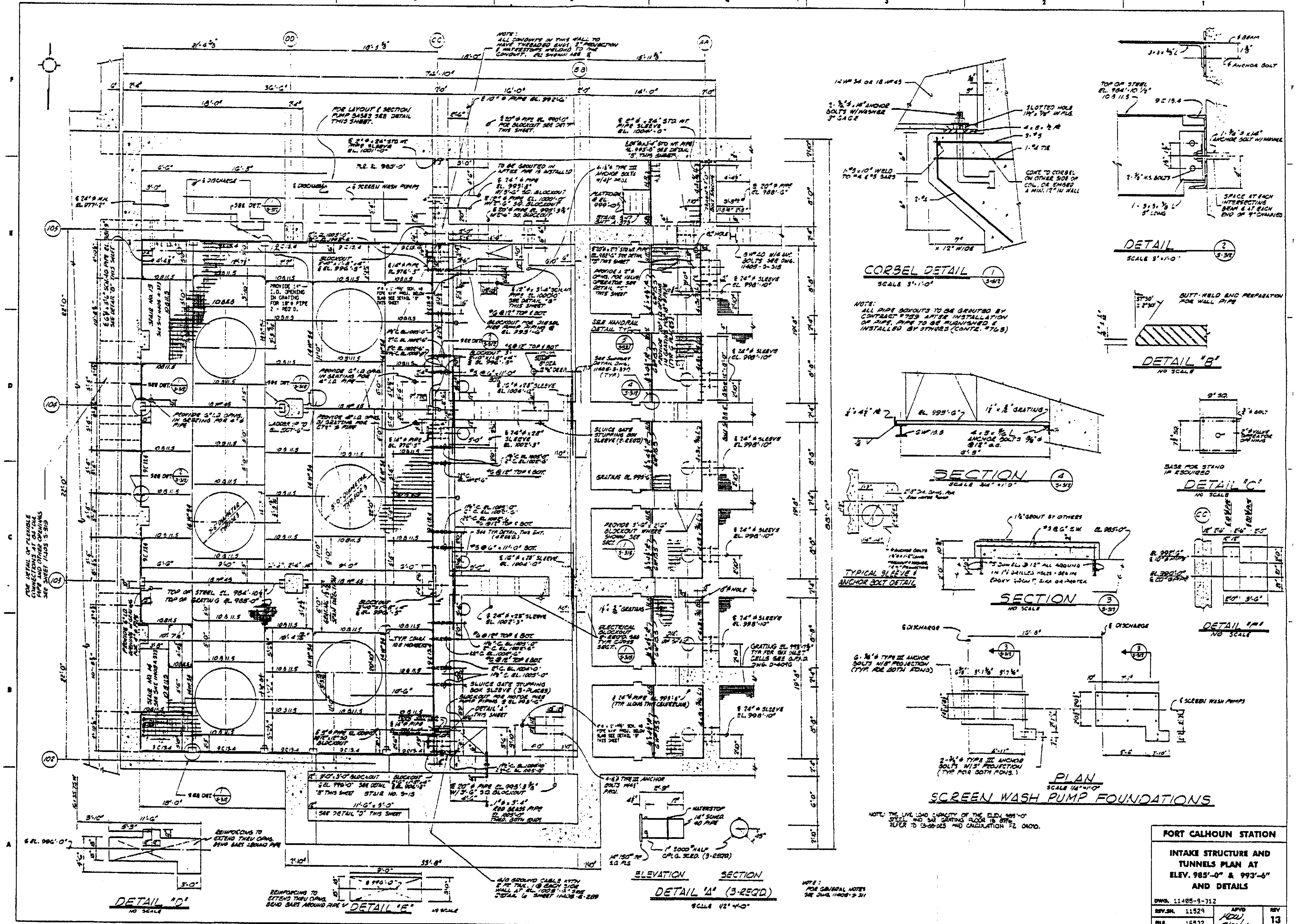
P&ID No.	System	Safety Class
E-23866-210-130	Safety Injection and Containment Spray System	
	• Reactor Coolant Pressure Boundary Piping	SC-1
	• Safety Injection Tank and Associated Piping	SC-2
	• Containment Spray Headers Inside Containment	SC-2
	• Safety Injection Headers Inside Containment	SC-2
	• Header from Safety Injection Leakage Coolers to CVCS inside containment	SC-2
	• Shutdown Cooling Suction Piping Containment Penetration (M-16)	SC-1
	• Balance of Containment Penetrations	SC-2
	• S.I. and Refueling and Water Tank and Containment Sump through Pumps and Heat Exchangers Including Recirculation Piping	SC-2
	• Alternate Flowpath from S.I. and Refueling Water Tank through Valve LCV-218-3 to Charging Pumps	SC-2
	• S.I. and Refueling Water Tank Vents	NNS CL-1
	• Relief Valve Discharge	NNS CL-1
	• Balance of System	NNS CL-2
B-120-F03001	Emergency Diesel Generator Lube Oil System	SC-3
B-120-F04002	Emergency Diesel Generator Jacket Water System	SC-3
B-120-F07001	Emergency Diesel Generator Starting Air	
	• Pressure Boundary from Air Receivers to Air Start Motors	SC-3
	• Air Compressors and Associated Piping	NNS CL-2
13007.54-EM-1A	Post Accident Sampling System	NNS CL-1&2

Table N-2 - "Correlation Between Safety Class, Equipment Design and Quality Assurance"

SAFETY CLASS	EQUIPMENT	CODE OF CONSTRUCTION [1]	QUALITY ASSURANCE [2]
SC-1	Piping [5]	B31.7, Class I - 1968 draft	CQE
	Piping System Components[4]	Original Contract Specifications	CQE
SC-2	Piping	B31.7, Class II - 1968 draft	CQE
	Piping System Components[4]	Original Contract Specifications	CQE
SC-3	Piping [5]	B31.7, Class III - 1968 draft	CQE
	Piping System Components[4]	Original Contract Specifications	CQE
NNS CL-1	Piping	B31.1[7]	(6)
	Piping System Components[4]	B31.1 and Note 3 & 7	(6)
NNS CL-2	Piping [5]	B31.1	Non-CQE
	Piping System Components[4]	B31.1 and Note 3	Non-CQE

NOTES:

1. "Code of Construction" connotes design, fabrication, installation, and testing.
2. 10CFR50 Appendix B applies to all CQE equipment.
3. NNS piping system components shall be commercial grade (e.g.,. pressure vessels to ASME section VIII, pumps to manufacturers standards, etc.).
4. Piping system components are valves, heat exchangers, pumps, etc.
5. Piping includes pipe fittings and pipe supports.
6. Limited CQE, Radioactive Waste Disposal, Fire Protection, or Non-CQE with special requirements.
7. Not applicable to Fire Protection.



FORT CALHOUN STATION

INTAKE STRUCTURE AND TUNNELS PLAN AT ELEV. 985'-0" & 993'-6" AND DETAILS

DWG. 11405-S-312			
REV.SH.	11529	APVD	REV
FILE	16532		13

FIG 1.2-11

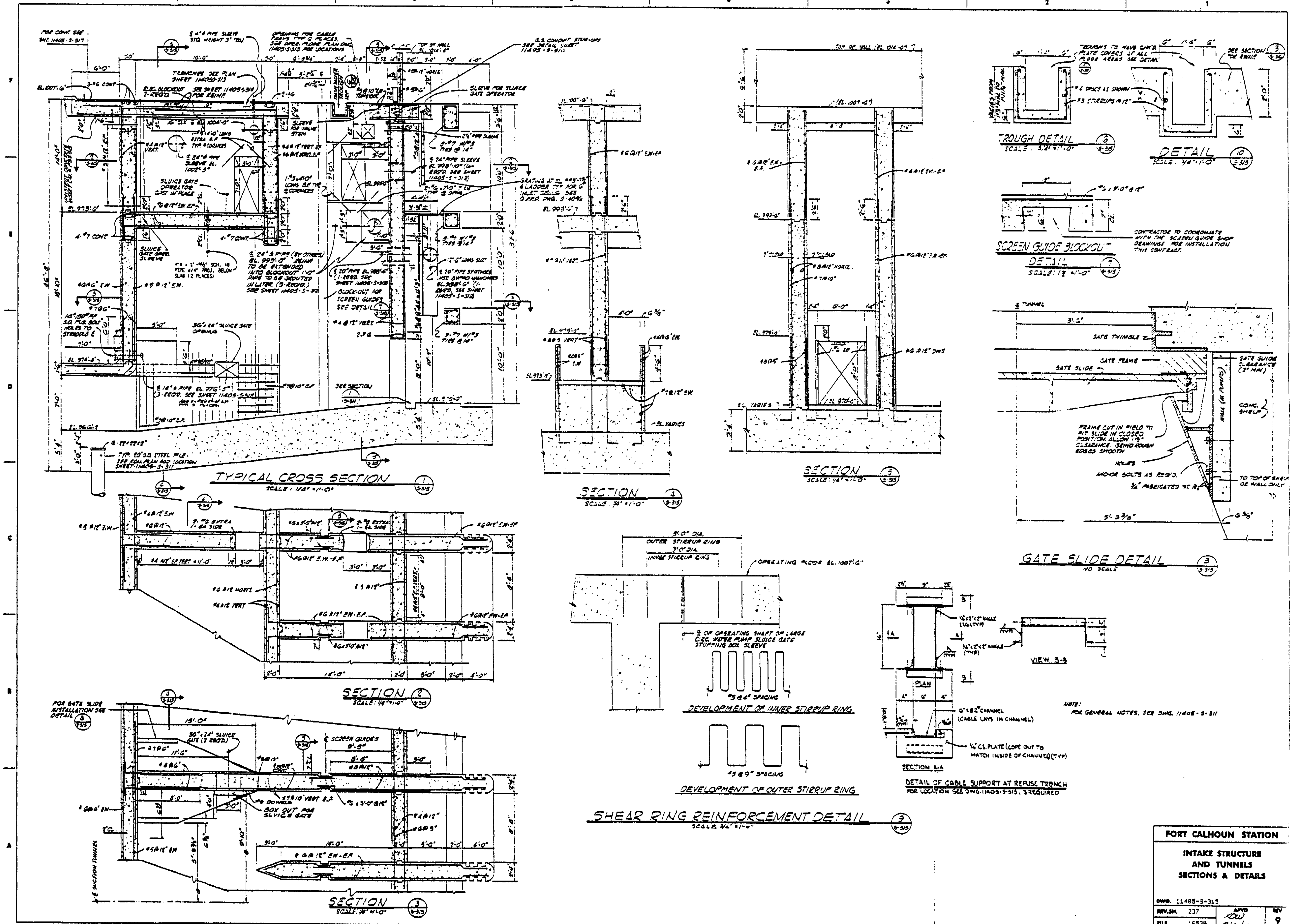
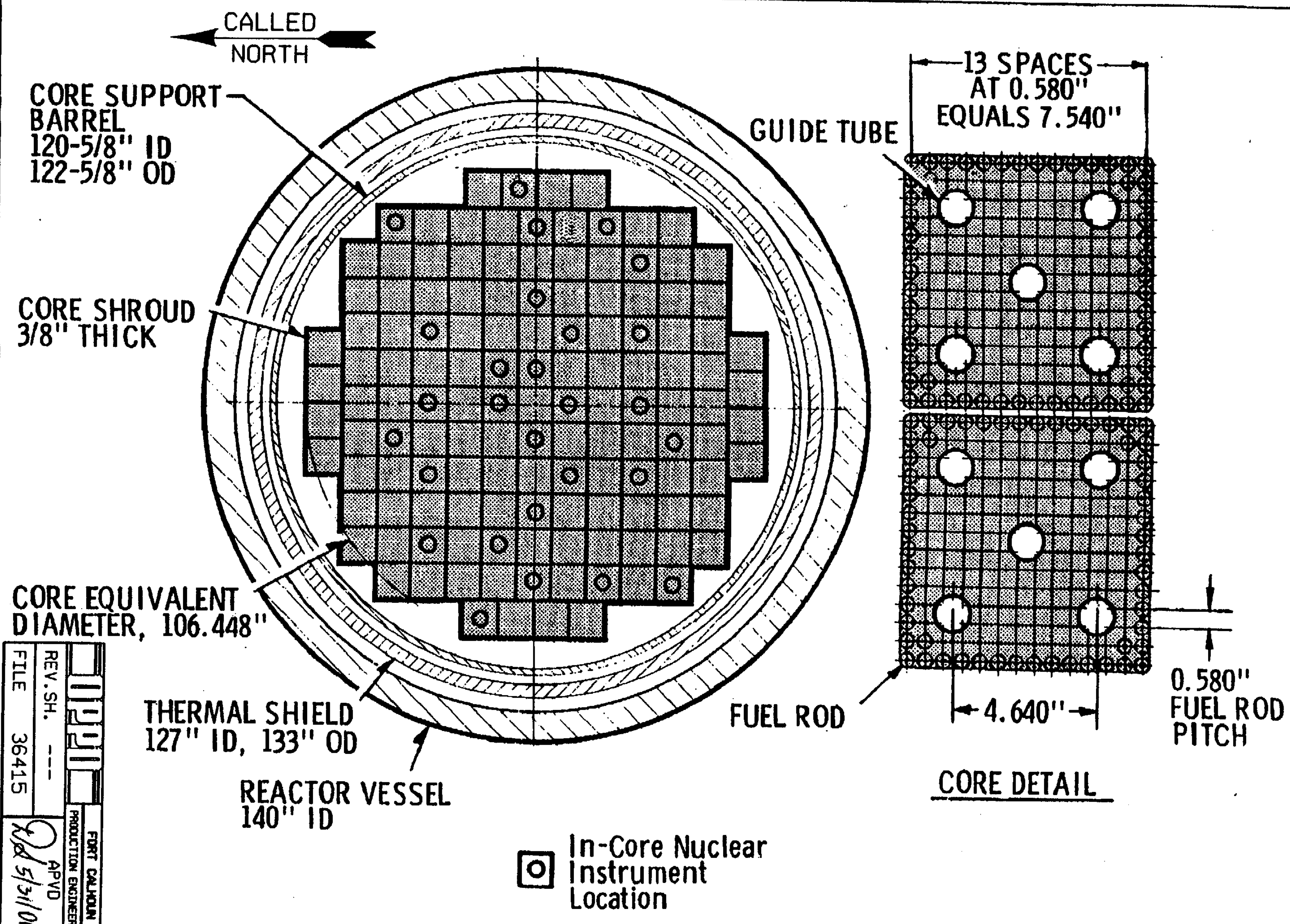

TYPICAL CROSS SECTION
SCALE: 1/4" = 1'-0"
SECTION 4
SECTION 5
SECTION 2
SECTION 3
TROUGH DETAIL
DETAIL
SCREEN GUIDE BLOCKOUT DETAIL
GATE SLIDE DETAIL
NO SCALE
DEVELOPMENT OF INNER STIRRUP RING
DEVELOPMENT OF OUTER STIRRUP RING
SHEAR RING REINFORCEMENT DETAIL
DETAIL OF CABLE SUPPORT AT REFUSE TRENCH
VIEW B-B
SECTION A-A
PLAN
NOTE:
FOR GENERAL NOTES, SEE DWG. 11405-S-311
FORT CALHOUN STATION
INTAKE STRUCTURE
AND TUNNELS
SECTIONS & DETAILS
DWG. 11405-S-315
REV.SH. 237
FILE 16535
APVD
REV
9

FIG 1.2-13



Reactor Core Cross Section

Omaha Public Power District
Fort Calhoun Station-Unit No.1

Figure 3.1-2

FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. ---
FILE 36415
APVD 5/31/01
REV 1

AA	FUEL ASSEMBLY LOCATION NUMBER
B.BBB	ASSEMBLY AVERAGE BURNUP (GWD/MTU)

				F1 27.832	H1 30.388	
		C2 37.668	D2 37.676	E2 21.263	G2 0.000	J2 0.000
	B3 38.231	C3 0.000	D3 0.000	E3 0.000	G3 19.474	J3 21.395
	B4 0.000	C4 0.000	D4 17.420	E4 28.546	G4 0.000	J4 31.281
A6 34.963	B5 22.051	C5 0.000	D5 27.960	E5 0.000	G5 22.621	J5 16.139
A8 29.129	B7 0.000	C7 19.821	D7 0.000	E7 22.452	G7 0.000	J7 21.902
	B9 0.000	C9 21.393	D9 30.563	E9 16.080	G9 21.821	J9 0.000

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION
REV.SH. 23643
FILE 36416
APVD DA 5/31/01
REV 6

Assy Avg Burnup Distribution Beginning of Cycle	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-1

AA	FUEL ASSEMBLY LOCATION NUMBER
B.BBB	ASSEMBLY AVERAGE BURNUP (GWD/MTU)

				F1 31.722	H1 35.470	
		C2 42.673	D2 45.995	E2 34.601	G2 19.026	J2 19.879
	B3 41.748	C3 16.975	D3 21.873	E3 23.178	G3 40.251	J3 41.436
	B4 3.997	C4 21.703	D4 39.072	E4 47.984	G4 23.857	J4 49.663
A6 40.856	B5 35.679	C5 23.266	D5 47.731	E5 23.819	G5 43.899	J5 38.017
A8 37.380	B7 20.510	C7 40.959	D7 24.074	E7 43.829	G7 23.761	J7 41.455
	B9 21.340	C9 41.957	D9 49.315	E9 38.081	G9 41.410	J9 23.436

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23644	APVD	REV
FILE 36417	5/31/01	6

Assy Avg Burnup Distribution End of Cycle (17,000 MWD/MTU)	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-2

AA	FUEL ASSEMBLY LOCATION NUMBER	
B	CEA GROUP IDENTIFICATION	
C	FUEL ASSEMBLY TYPE	

				F1 T7	H1 T3	
		C2 T7	D2 A T3	E2 X4	G2 2 Y6	J2 Y7
	B3 T8	C3 4 Y5	D3 Y9	E3 A Y3	G3 X3	J3 N X5
	B4 A Y1	C4 Y9	D4 B X2	E4 T1	G4 Y4	J4 T1
A6 T3	B5 X6	C5 A Y3	D5 T1	E5 B Y2	G5 X6	J5 3 X2
A8 T7	B7 2 Y8	C7 X3	D7 1 Y4	E7 X6	G7 Y2	J7 X1
	B9 Y3	C9 N X5	D9 T1	E9 3 X2	G9 X1	J9 4 Y10

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. 23645	APVD	REV
FILE 36419	5/31/01	6

Position of Fuel Assemblies And CEA Groups	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-4

AA B.BBB C.CCC	FUEL ASSEMBLY LOCATION NUMBER RELATIVE ASSEMBLY POWER DENSITY MAXIMUM F_r

					F1 0.222	H1 0.297	
		C2 0.263	D2 0.453	E2 0.792	G2 1.197	J2 1.250	
	B3 0.178	C3 1.023	D3 1.301	E3 1.347 1.605	G3 1.329	J3 1.301	
	B4 0.106	C4 1.251	D4 1.357	E4 1.166	G4 1.348	J4 1.136	
A6 0.305	B5 0.753	C5 1.303	D5 1.178	E5 1.295	G5 1.287	J5 1.389	
A8 0.439	B7 1.221	C7 1.308	D7 1.340	E7 1.287	G7 1.303	J7 1.159	
	B9 1.223	C9 1.285	D9 1.142	E9 1.389	G9 1.160	J9 1.286	

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. 23646 FILE 36420	APVD 5/31/01	REV 6

Core Radial Power Distribution Beginning of Cycle ARO	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-5

AA	FUEL ASSEMBLY LOCATION NUMBER
B.BBB	RELATIVE ASSEMBLY POWER DENSITY
C.CCC	MAXIMUM F_r

				F1 0.231	H1 0.306	
		C2 0.291	D2 0.489	E2 0.794	G2 1.160	J2 1.224
	B3 0.203	C3 1.012	D3 1.312	E3 1.392 1.580	G3 1.240	J3 1.194
	B4 0.221	C4 1.294	D4 1.295	E4 1.147	G4 1.378	J4 1.076
	B5 0.791	C5 1.381	D5 1.163	E5 1.365	G5 1.231	J5 1.282
A6 0.336						
	B7 1.205	C7 1.242	D7 1.382	E7 1.234	G7 1.363	J7 1.135
A8 0.472						
	B9 1.246	C9 1.201	D9 1.090	E9 1.286	G9 1.137	J9 1.398

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23647	APVD	REV
FILE 36421	5/31/01	6

Core Radial Power Distribution at 7000 MWD/MTU Burnup ARO	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-6

AA	FUEL ASSEMBLY LOCATION NUMBER
B.BBB	RELATIVE ASSEMBLY POWER DENSITY
C.CCC	MAXIMUM F_r

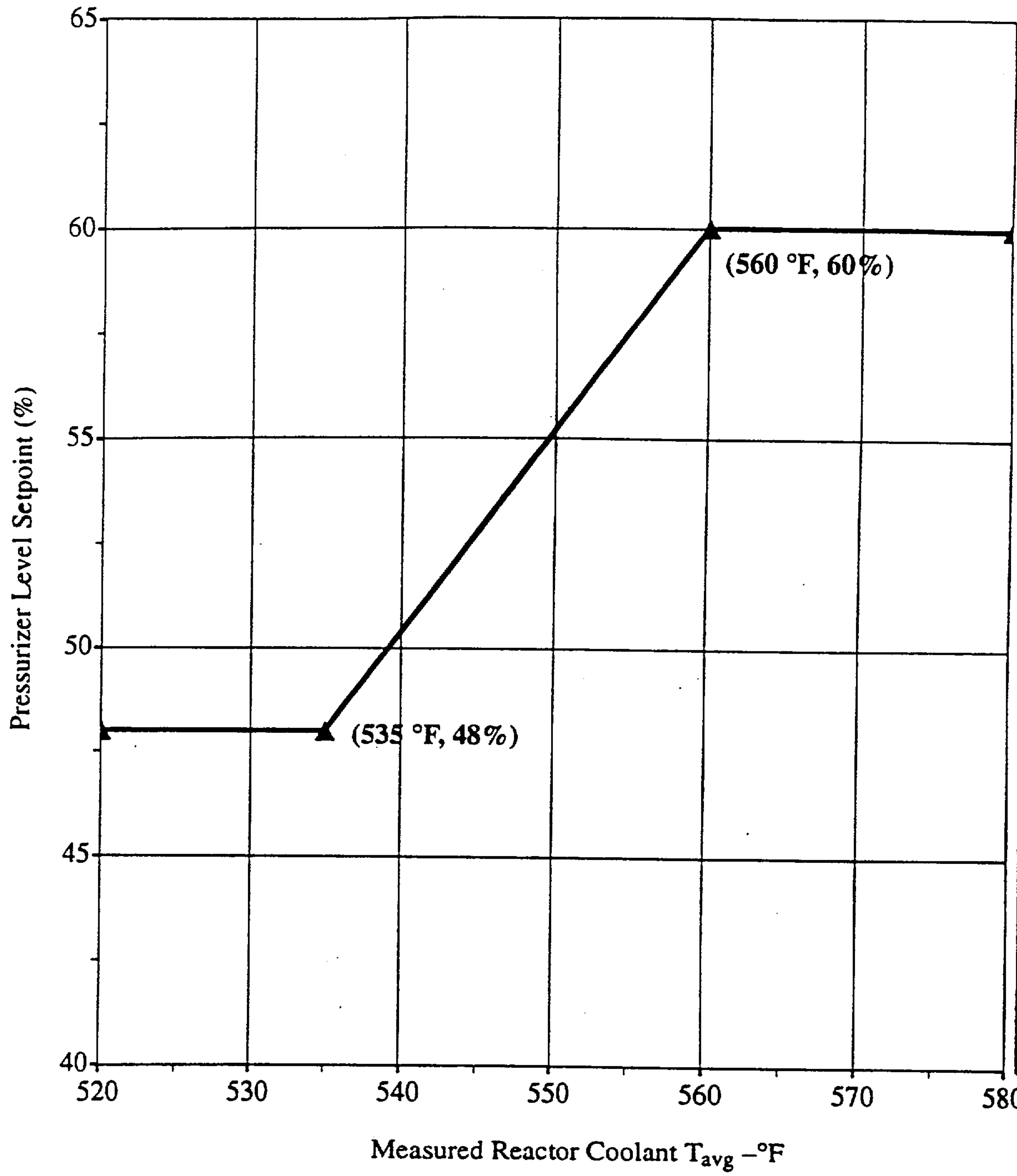
				F1 0.251	H1 0.317	
		C2 0.343	D2 0.542	E2 0.800	G2 1.084	J2 1.119
	B3 0.248	C3 1.021	D3 1.269	E3 1.337	G3 1.156	J3 1.106
	B4 0.362	C4 1.289	D4 1.217	E4 1.126	G4 1.408	J4 1.062
A6 0.412	B5 0.856	C5 1.373	D5 1.147	E5 1.417	G5 1.198	J5 1.218
A8 0.558	B7 1.226	C7 1.200	D7 1.427 1.523	E7 1.205	G7 1.397	J7 1.104
	B9 1.286	C9 1.160	D9 1.087	E9 1.225	G9 1.106	J9 1.340

OPPD
FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION
REV.SH. 23648
FILE 36422
APVD 5/31/01
REV 6

Core Radial Power Distribution EOC (17,000 MWD/MTU) ARO	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-7

USAR FIGURE 4.3–10

PRESSURIZER LEVEL SETPOINT VS. REACTOR COOLANT AVERAGE TEMPERATURE



FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION

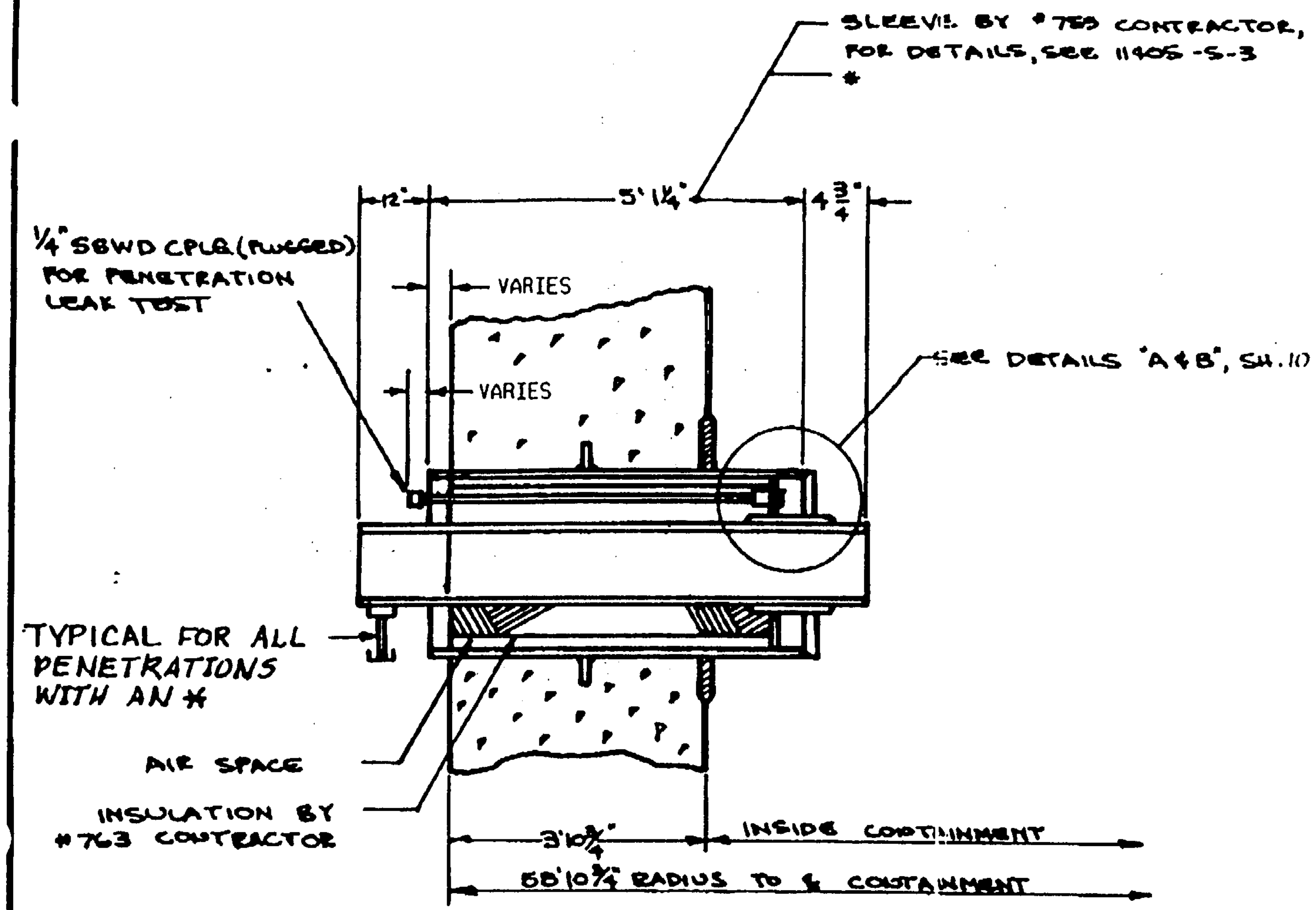
REV 1

APVD K. Perdue 12-11-95

REV. SH. 37931

FILE 36460

Pressurizer Water Level Program	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 4.3-10



TYPE III

NOTES:
1. FOR PIPE CONNECTIONS AT PENETRATION M-44, SEE DWG. 11405-M-54, SH. 8.05
2. SEE NOTE 1,2,3 & 4 ON SH.1
3. SEE DWG. B-4386
4. SEE S & W DWG. 13007.54-EM5A FOR ADDITIONAL DETAIL 6-6.
5. DUST COVER ONLY - NON-COE.

* 5'-9¼" L. FOR PENETRATION M-19, 24, & 25.
* 6'-1" L. FOR PENETRATION M-8, 14 & 15.

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. 1986	APVD	REV
FILE 16246	2/27/95	12

PIPING PENETRATION LIST

	PEN. NO.	SERVICE DESCRIPTION	PIPE SIZE	PIPE CLASS	PIPE TEMP. DEG. F	GUIDE TUBE SUPPORT TYPE 1 OR 2 (NOTE 3)	SLEEVE SIZE	SLEEVE SCHED.	AL SLEEVE COVER Y or N (NOTE 5)
	M-8	CONTAINMENT SUMP PUMP DISCH.	2"	151 R	50-300	TYPE 1	8"	80	Y
	M-14	VENT HEADER	3"	152 R	50-150		12"		
	M-15	NUCLEAR DETECTOR WELL COOLING UNIT SUP.	1½"	152	50-100		8"		
	M-24	PRESSURIZER QUENCH TANK GAS SAMPLE	¾"	301 R	50-300		8"		
	M-25	REACTOR COOLANT DRN. TANK GAS SAMPLE	¾"	301 R	50-140		8"		
*	M-30	HYDROGEN PURGE	4"	152			12"		
	M-38	CONTAINMENT H.P. CONTROL	1"	152		TYPE 1	12"		
*	M-39	C.W. SUPPLY TO S.I. TANK LEAKAGE COOLERS.	4"	152	50-140	TYPE 2	8"		
	M-57	HYDROGEN SAMPLING	1"	152		TYPE 1	8"		
*	M-42	NITROGEN TO SAFETY INJECTION TANKS	1"	152			8"		
*	M-43	NITROGEN TO QUENCH TANK & R.C. DRAIN TANK	1"	152			12"		
	M-44	SPARE	1"	1000	AMBIENT		8"		
	M-46	CONTAINMENT AIR SAMPLING	1"	151			8"		
	M-47	CONTAINMENT AIR SAMPLING	1"	151			8"		
	M-48	CONTAINMENT PRESS. RELIEF	2"	152			8"		
	M-50	CONTAINMENT H.P. CONTROL	1"	152			12"		
	M-51	" " " "	1"	152			8"		
	M-52	" " " "	1"	152			8"		
	M-58	HYDROGEN SAMPLING	1"	152			8"		
	M-73	INSTR. AIR	2"	301	50-130	TYPE 1	12"		
*	M-74	SERVICE AIR	4"	152	50-150	TYPE 2	8"		
	M-77	C.W. SUPPLY TO VA-8A	8"	152	50-120	TYPE 2	12"		
	M-78	C.W. SUPPLY TO VA-1A	10"	152	50-120	TYPE 3	14"		
*	M-79	DEAER WATER.	2"	151	32-85	TYPE 1	8"		
*	M-80	DEMIN WATER	2"	151	32-85	TYPE 1	12"		
	M-84	C.W. SUPPLY TO VA-8B	8"	152	50-120	TYPE 2	12"		
	M-85	C.W. SUPPLY TO VA-1B	10"	152	50-120	TYPE 3	14"		
	M-86	S.I. SPRAY	12"	301 R	120-160	TYPE 1	18"		
	M-89	S.I. SPRAY	12"	301 R	120-160		18"		
*	M-69	HYDROGEN PURGE	4"	152			12"		
	M-19	R.C. PUMP SEAL COOLER & L.O. COOLER C.W. SUP.	6"	152	50-100		12"		
*	M-31	Hydrogen Sampling	1"	SEE NOTE 4		TYPE 1	8"	80	Y
	M-40	Hydrogen Sampling							

USE THIS DWG. WITH SHEET 10

SHEET 4 CONT. ON 5

REV	DESCRIPTION	DATE	BY
6	ADDED PENET. M-41	10-1-73	LG
5	REV. FOR CONSTR. CONTR. # 763	3-14-72	LG
4	REV. FOR CONSTR. CONTR. # 763	3-18-70	BW
3	REV. FOR CONSTR. CONTR. # 763	8-6-69	
2	ISSUED FOR CONSTR. CONTR. # 763	4-29-69	
1	ISSUED FOR BRAND CONTR. # 763	11-19-68	

O.P.P.D. FORT CALHOUN STATION UNIT #1

CONTAINMENT PIPE PENETRATIONS

DESIGNED ____ CHECKED ____
DETAILED ____ DATE ____
APPROVED ____

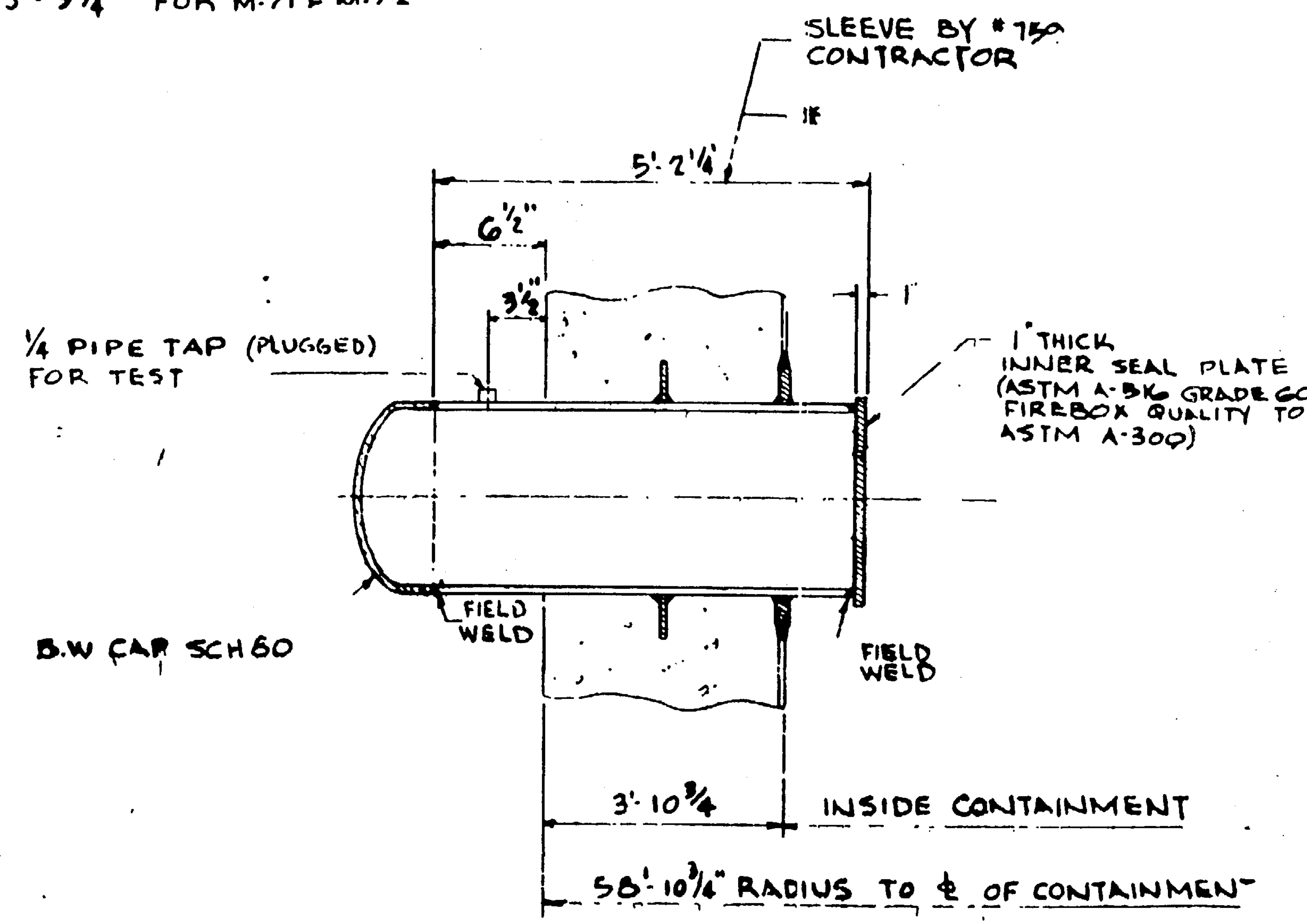
GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS OMAHA, NEBRASKA

USAR FIGURE 5.9-4

DRAWING NO. 11405-M-70

Ⓐ

* 5'-6 31/32" FOR M-26
6'-4 23/32" FOR M-1
6'-4 11/32" FOR M-4
5'-9 1/4" FOR M-71 & M-72



PIPING PENETRATION LIST

PEN NO.	SERVICE DESCRIPTION	SLEEVE SIZE	SLEEVE SCH.
M-1	SPARE	12	80
M-4		16	
M-21		12	
M-23		8	
M-26		16	
M-27		8	
M-28		12	
M-29		8	
M-32	SPARE	12	
M-33		8	
M-34		12	
M-35		8	
M-36		8	
M-37		8	
M-41	SPARE	8	
M-54		12	
M-55		8	
M-56		12	

PEN No.	SERVICE DESCRIPTION	SLEEVE SIZE	SLEEVE SCH.
M-59	SPARE	8	80
M-60		8	
M-61		8	
M-62		12	
M-64		8	
M-65		8	
M-66		12	
M-67		8	
M-68		8	
M-70		8	
M-71		8	
M-72		12	
M-81		8	
M-90		8	
M-92		8	
M-98		12	
M-99		8	

NOTES
1. SEE NOTES No 1, 2, 4 & ON SHT #1

OMAHA PUBLIC POWER DISTRICT
OMAHA, NEBRASKA

1987	6	2.5.1987	5	REV. NO.
5-84	TAW	11-13-79	JGK	DRAWN
5-21-84	WDK	11-21-79	VES	CHECK
5-21-84	JGK	1-9-80	MEE	ENG.
—	—	—	—	CIVIL DPT.
—	—	—	—	ELEC. DPT.
5-21-84	MEE	1-9-80	MEE	MECH. DPT.
—	—	—	—	Q CONTROL
—	—	—	—	P.E.

REGISTERED PROFESSIONAL ENGINEER
FRANK D. HUTCHINSON III
E-2817
STATE OF NEBRASKA

SHEET 5 CONT. ON 6

NO.	REVISIONS	DATE	BY
4	DELETED PENETRATION M-41	10-1-73	L.G.
3	REV. FOR CONSTR CONTR #763	3-19-72	L.G.
2	REV. FOR CONSTR. CONTR #763	8-6-69	[illegible]
1	CONSTR. CONTRACT #763	5-6-69	[illegible]

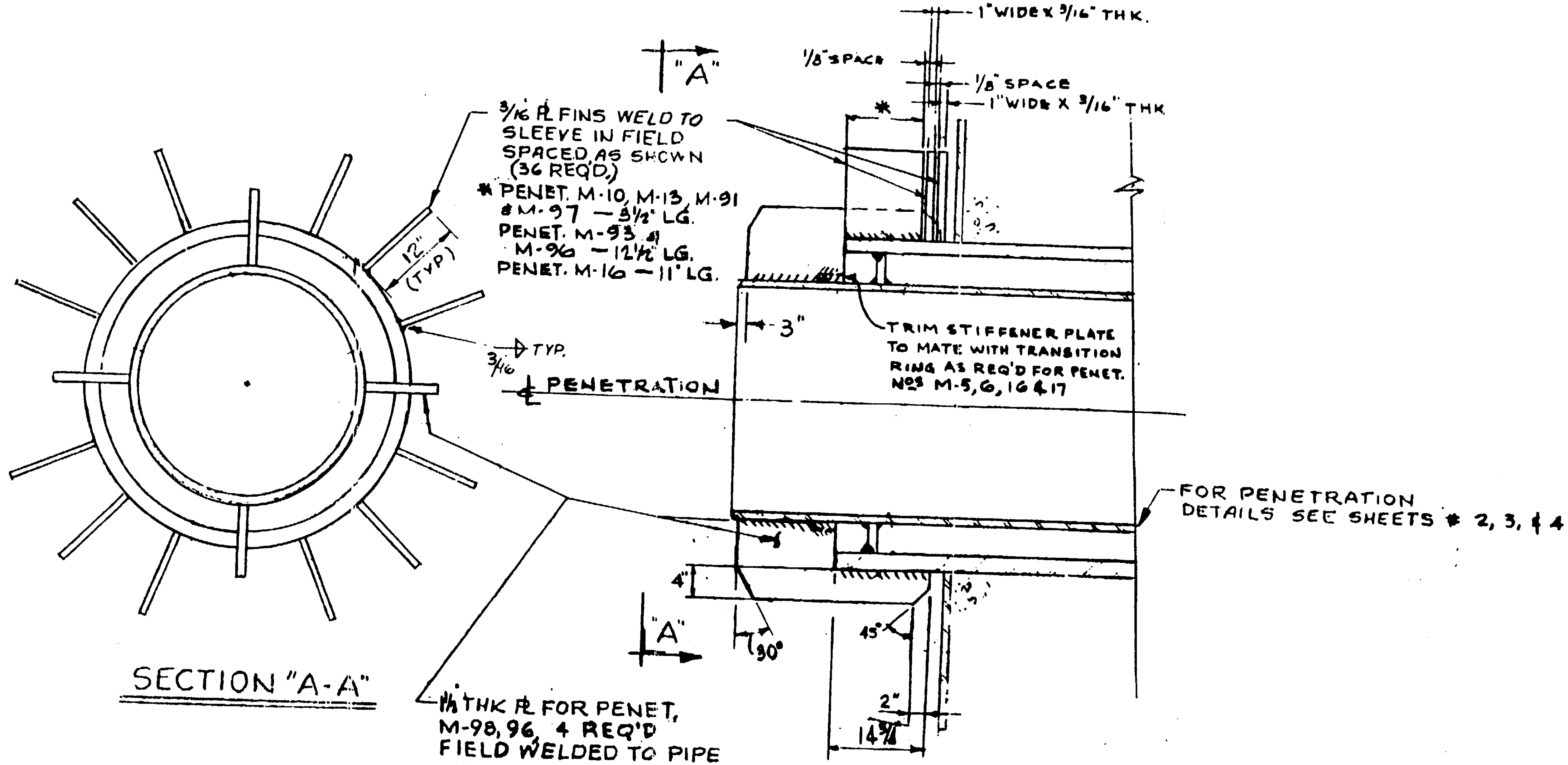
O.P.P.D. FORT CALHOUN STATION UNIT #1

CONTAINMENT PIPE PENETRATIONS

DESIGNED [illegible] CHECKED —
DETAILED — DATE —
APPROVED —

GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS
OMAHA, NEBRASKA

USAR FIGURE 5.9-5
DRAWING NO. 11405-M-70



PIPING CONTRACTOR SHALL BE RESPONSIBLE FOR FURNISHING, FABRICATING & INSTALLING SLEEVE COOLING FINS & STIFFENERS AS INDICATED ABOVE. FINS & STIFFENERS SHALL BE INSTALLED BEFORE CONTAINMENT PRESSURE TESTS.

OMAHA PUBLIC POWER DISTRICT
OMAHA, NEBRASKA

RB-		RB-1323	5	REV. NO.
		5-23-84		DRAWN
		5-25-84	DWD	CHECK
		5-24-84		ENG.
		—	—	ELECT. DPT.
		—	—	CIVIL DPT.
		5-25-84		MECH. DPT.
		—	—	NUC. DPT.
		—	—	Q. CONTROL
		—	—	P.E.

REGISTERED PROFESSIONAL ENGINEER
FRANK D. HUTCHINSON III
E-2817
STATE OF NEBRASKA

SHEET 6 CONT. ON 7

NO.	REVISIONS	DATE	BY
4	REV. FOR CONSTR. CONTR. #763	8-6-69	
3	REV. FOR CONSTR. CONTR #763	5-6-69	
2	ISSUED FOR CONSTR. CONTR. #763	4-21-69	
1	ISSUED FOR BIDDING CONTR. #763	11-19-68	

O.P.P.D. FORT CALHOUN STATION UNIT #1

PENETRATION COOLING FINS & STIFFENERS

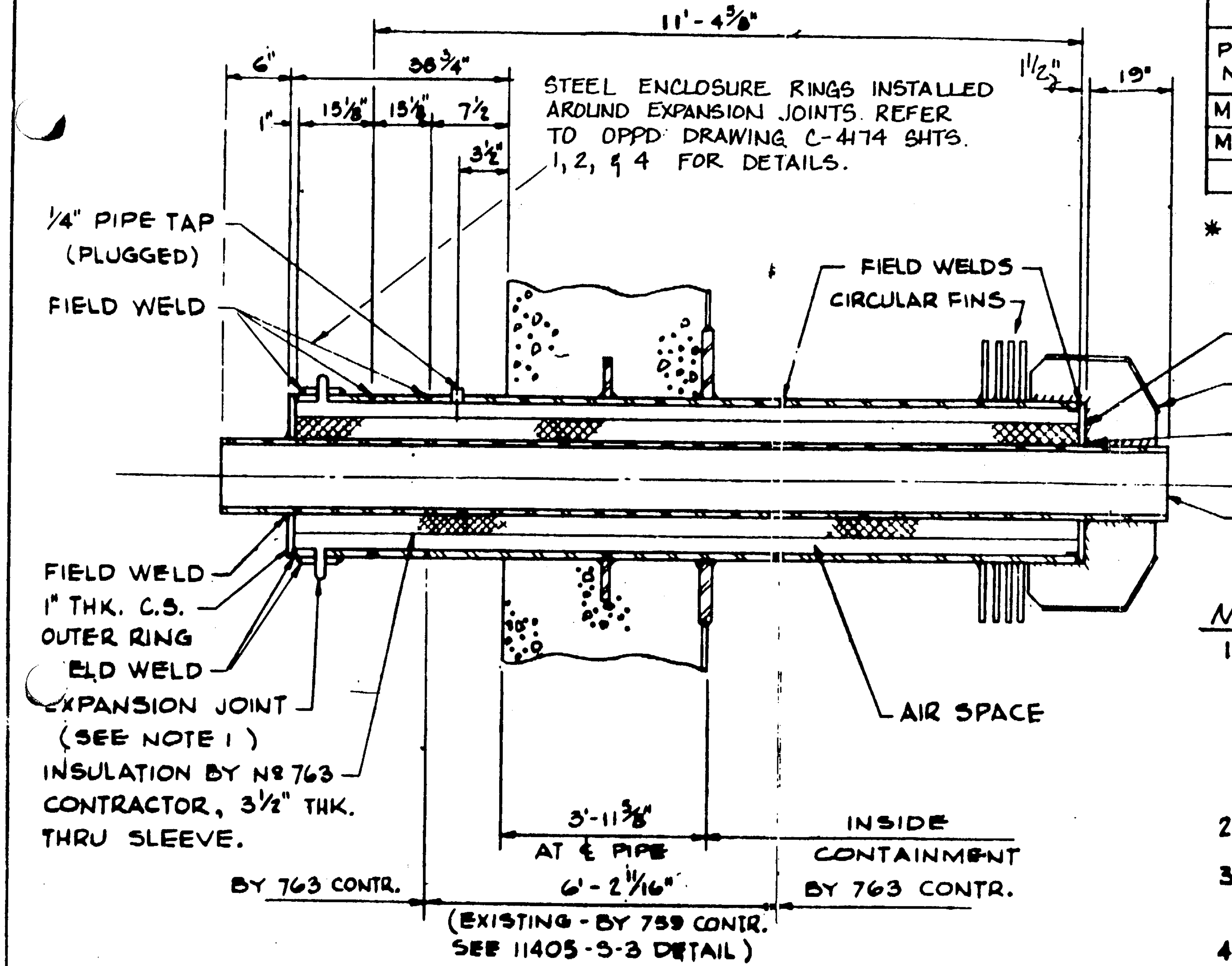
DESIGNED ______ CHECKED —
DETAILED — DATE —
APPROVED —

GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS
OMAHA, NEBRASKA

USAR FIGURE 5.9-6

DRAWING NO. 11405-M-70

16240



TYPE I

PIPING PENETRATION LIST

PEN Nº	SERVICE DESCRIPTION	PIPE SIZE	PIPE CLASS	PIPE TEMP. DEG. F		SLEEVE SIZE	SLEEVE SCH.
M-94	MAIN STEAM	28"	1000	545		44"	2" THK
M-95	MAIN STEAM	28"	1000	545		44"	2" THK

* EXCLUSIVE OF PARAGRAPHS N-810 THRU N-842, PAR N-1500 & PARAGRAPHS VW-60 AND VCS-115 OF SECTION VIII OF THE CODE.

NOTES:

1- *ORIGINAL S.S. ELEMENT REMOVED. EXPANSION JOINT ASSEMBLY SHOP FABRICATED IN TWO HALVES. EXPANSION JOINT ASSEMBLY TO BE WELDED TO EXISTING WELDING ENDS, WELDING ENDS* TO BE IDENTICAL IN MATERIAL O.D. & WALL THK. TO SLEEVE (SEE DWG. Nº 11405 S-3). REFER TO OPPD DWG D-4405 AND TEMPFLEX DWG D-51109 FOR DTLS.

2- SEE NOTES Nº 1, 2, 3, & 4 SH-1.

3- SEE SHEET 8 FOR PENETRATION COOLING FINS AND STIFFENERS.

4- Nº 763 CONTRACTOR TO EXAMINE SLEEVE ASSEMBLY BY Nº 759 CONTRACTOR AND ADAPT HIS WORK TO FIT. Nº 763 CONTRACTOR TO DETERMINE EXTENT OF FIELD FABRICATION NECESSARY TO MEET FIELD CONDITIONS. SHOP ASSEMBLY TO BE MAXIMIZED.

5- 28" STEAM LINE TO BE IN CONFORMANCE WITH CONTRACT Nº 763, SECTION HI, GROUP II PIPING.

6- ALL OTHER PRESSURE RETAINING PARTS INSIDE CONTAINMENT TO BE IN ACCORDANCE WITH ASME SECT. III CLASS B*.

7- PORTION OF 44" SLEEVE PROVIDED BY CONTRACT Nº 763 SHALL BE ASTM A-516 GRADE 60 FIREBOX QUALITY TO ASTM A-300. CHARPY VEE-NOTCH IMPACT TESTS IN ACCORDANCE WITH ASME SECT. III CLASS B SHALL BE PERFORMED AT MAX. TEMP. OF 0°F.

REGISTERED PROFESSIONAL ENGINEER
FRANK D. HUTCHINSON III
E-2817
STATE OF NEBRASKA

SHEET 7 CONT. ON 8

NO.	REVISIONS	DATE	BY
	REV. FOR CONSTR. CONTR #763	3-18-70	GW
	TRANSFERRED FROM SHT. 4 ISSUED	8-6-69	L.G.
	FOR CONSTR. CONTR. #763		

O.P.P.D. FT. CALHOUN STATION - UNIT # 1

CONTAINMENT PIPE PENETRATIONS

DESIGNED ________ CHECKED ________
DETAILED ________ DATE ________
APPROVED ________

GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS
OMAHA, NEBRASKA

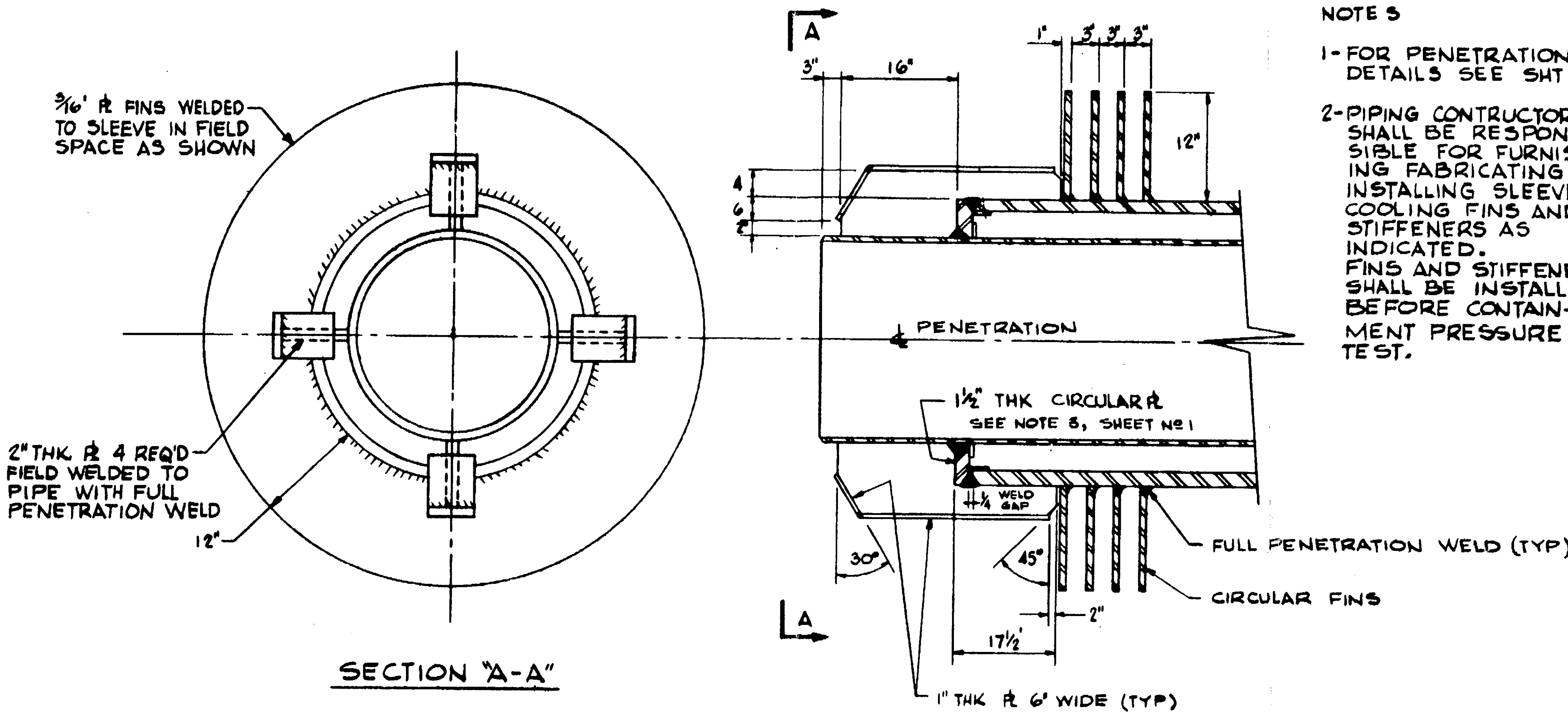
USAR FIGURE 5.9-7

DRAWING NO. 11405-M-70

FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION
OPPD
REV 5
APVD JLS 7-30-91
REV. SH. 13232
FILE 16243

r.s. 13232

16243 16243



NOTES

1- FOR PENETRATION DETAILS SEE SHT #7

2- PIPING CONTRUCTOR SHALL BE RESPONSIBLE FOR FURNISHING FABRICATING & INSTALLING SLEEVE COOLING FINS AND STIFFENERS AS INDICATED. FINS AND STIFFENERS SHALL BE INSTALLED BEFORE CONTAINMENT PRESSURE TEST.

OPPD — FORT CALHOUN STATION — PRODUCTION ENGINEERING DIVISION

REV.SH. 13233 | APVD [signature] 9/18/00 | REV 4

FILE 16244

NO	REVISIONS	DATE	BY
2	REV FOR CONSTR CONTR 763	3-18-70	GW
1	TRANSFERRED FROM SHT 6 &	8-6-69	LG
	ISSUED FOR CONSTR CONTR #763		

OPPD FT CALHOUN STATION UNIT #1

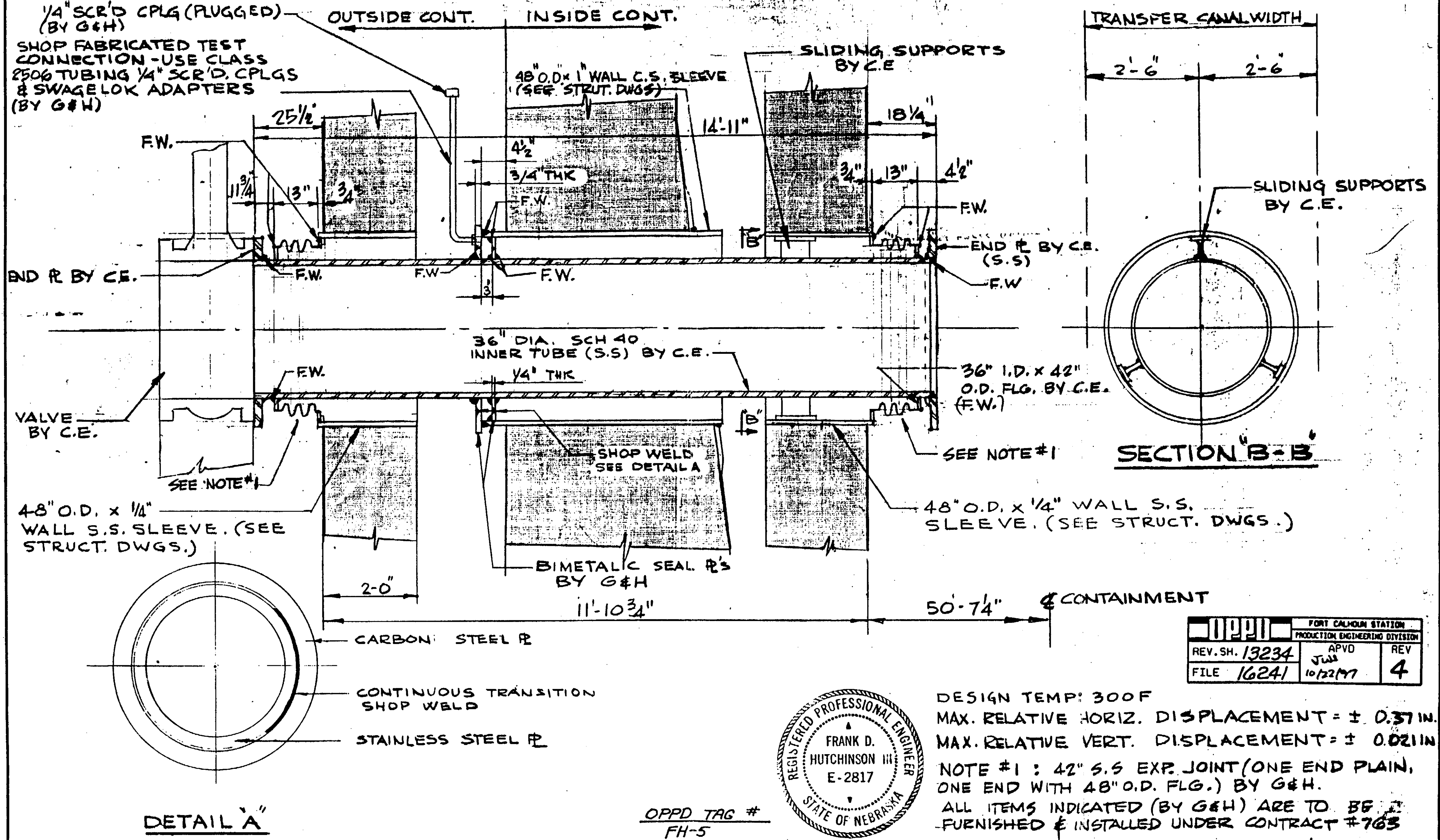
PENETRATION COOLING FINS & STIFFENERS

DESIGNED ____ CHECKED ____
DETAILED ____ DATE ____
APPROVED ____

GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS — OMAHA, NEBRASKA

USAR FIGURE 5.9-8

DRAWING NO 11405-M-70, SH.8



OPPD — FORT CALHOUN STATION — PRODUCTION ENGINEERING DIVISION

REV. SH. 13234	APVD JWJ 10/22/97	REV 4
FILE 16241		

DESIGN TEMP: 300F
MAX. RELATIVE HORIZ. DISPLACEMENT = ± 0.37 IN.
MAX. RELATIVE VERT. DISPLACEMENT = ± 0.021 IN

NOTE #1 : 42" S.S EXP. JOINT (ONE END PLAIN, ONE END WITH 48" O.D. FLG.) BY G&H.
ALL ITEMS INDICATED (BY G&H) ARE TO BE FURNISHED & INSTALLED UNDER CONTRACT #763

REGISTERED PROFESSIONAL ENGINEER — FRANK D. HUTCHINSON III — E-2817 — STATE OF NEBRASKA

OPPD TAG # FH-5

SHEET 9 CONT. ON 10

NO.	REVISIONS	DATE	BY
3	REV. FOR CONTR. CONTR. #763	8-6-69	
2	ISSUED FOR CONSTR. CONTR. #763	4-21-69	
1	ISSUED FOR BIDDING CONTR. #763	11-19-68	

O.P.P.D. FORT CALHOUN STATION - UNIT #1 FUEL TRANSFER PENETRATION DETAIL - M-100

DESIGNED ____ CHECKED ____
DETAILED ____ DATE ____
APPROVED ____

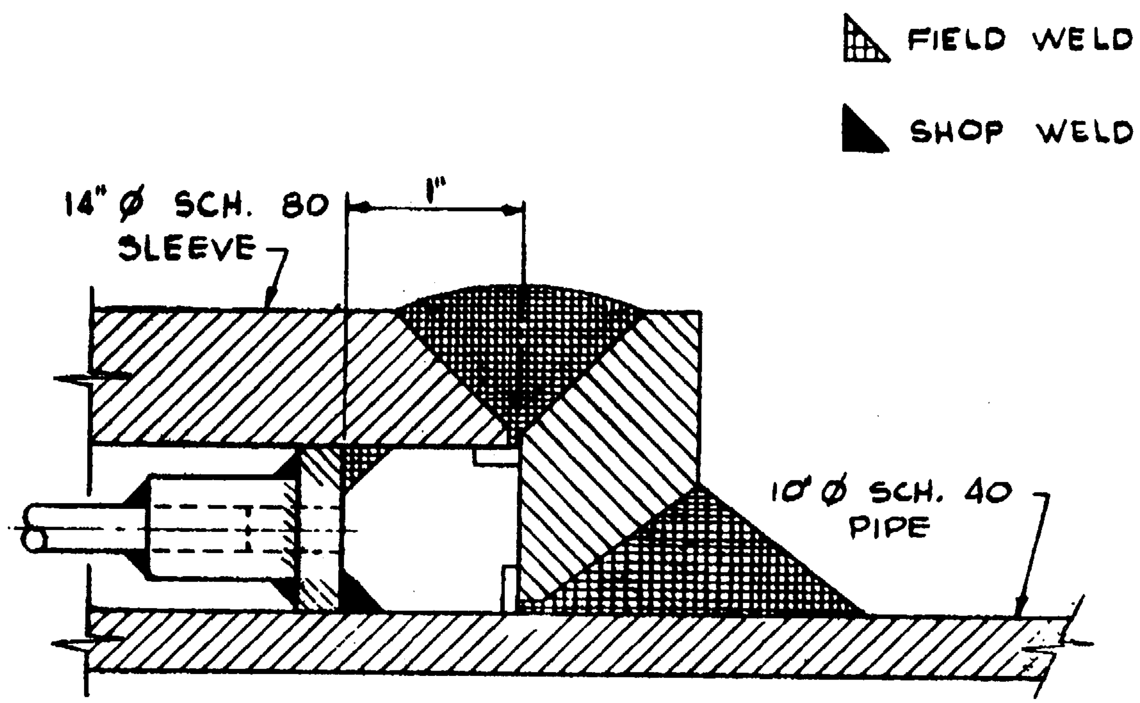
GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS — OMAHA, NEBRASKA

USAR FIGURE 5.9-9
DRAWING NO. 11405-M-70

16241

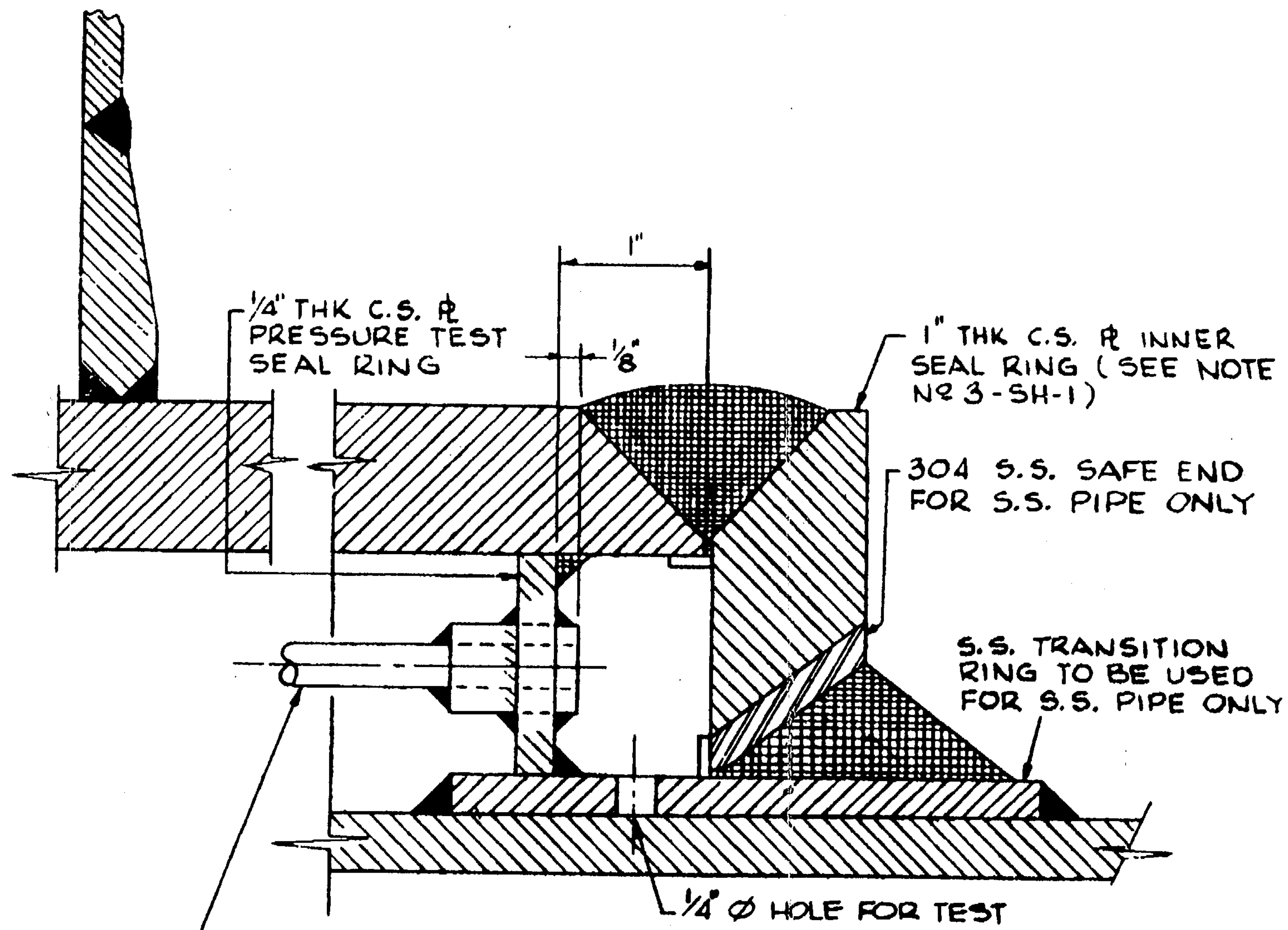
16241

USE THIS DWG. WITH SHEET 4



SLEEVE CLOSURE DETAIL
FOR PENETRATION M-78 & M-85 ONLY

DETAIL "B"



SLEEVE CLOSURE DETAIL FOR ALL
PENETRATIONS LISTED ON SHT#5 EXCEPT M-78, 85, 86 & 89

DETAIL "A"

SHOP FABRICATION TEST CONNECTION
USE CLASS 2506 TUBING, SWAGELOK
(OR EQUAL) ADAPTERS.

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. 2293	APVD	REV
FILE 16245	9/16/00	5

NO.	REVISIONS	DATE	BY
2	CONSTRUCTION	10-1-70	L.G.
1	ISSUED FOR CONSTRUCTION CONTR #763	3-18-70	GW

O.P.P.D. FT. CALHOUN STATION - UNIT #1

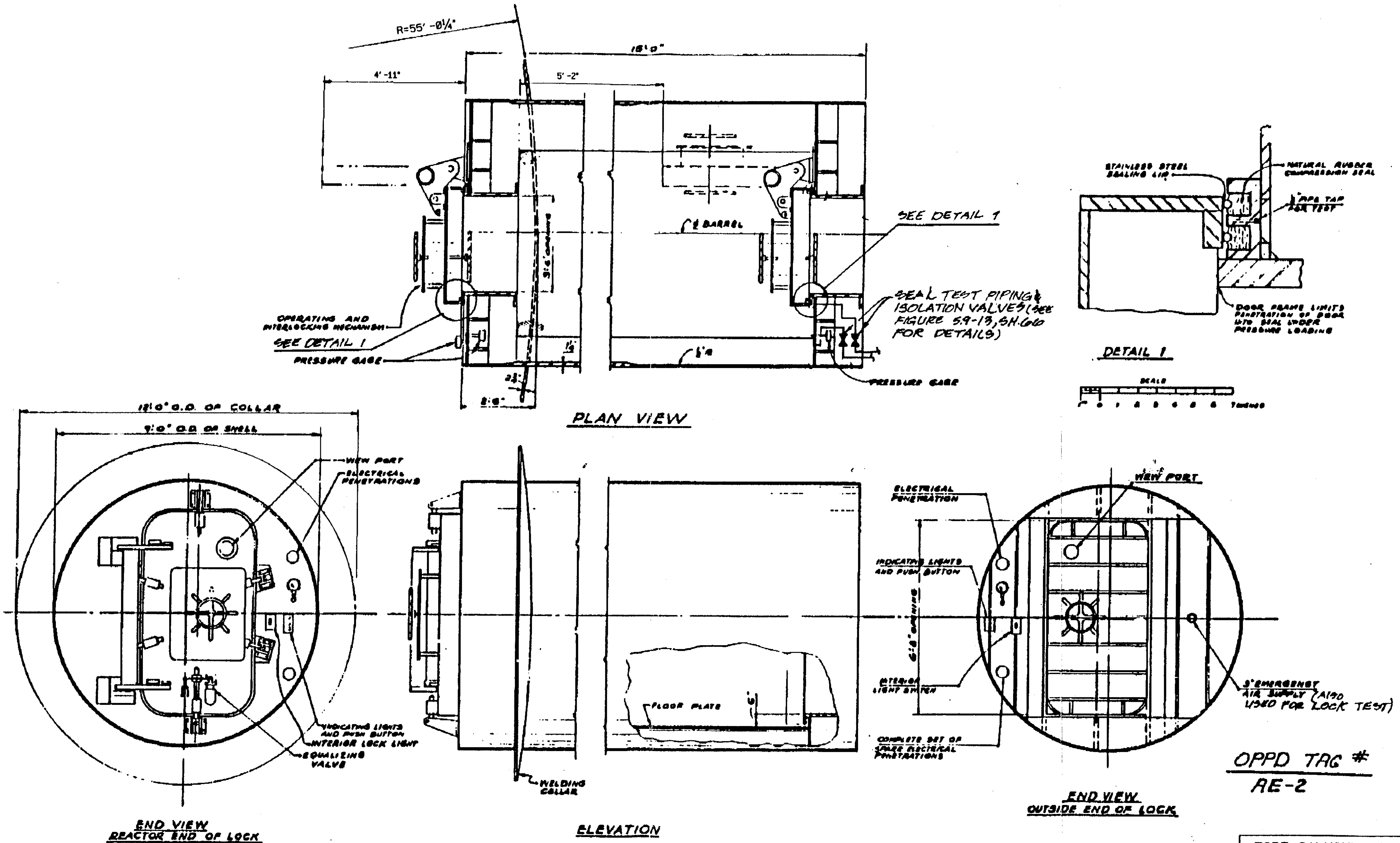
CONTAINMENT PIPE PENETRATIONS

DESIGNED GW CHECKED
DETAILED DATE MARCH 9, 1970
APPROVED

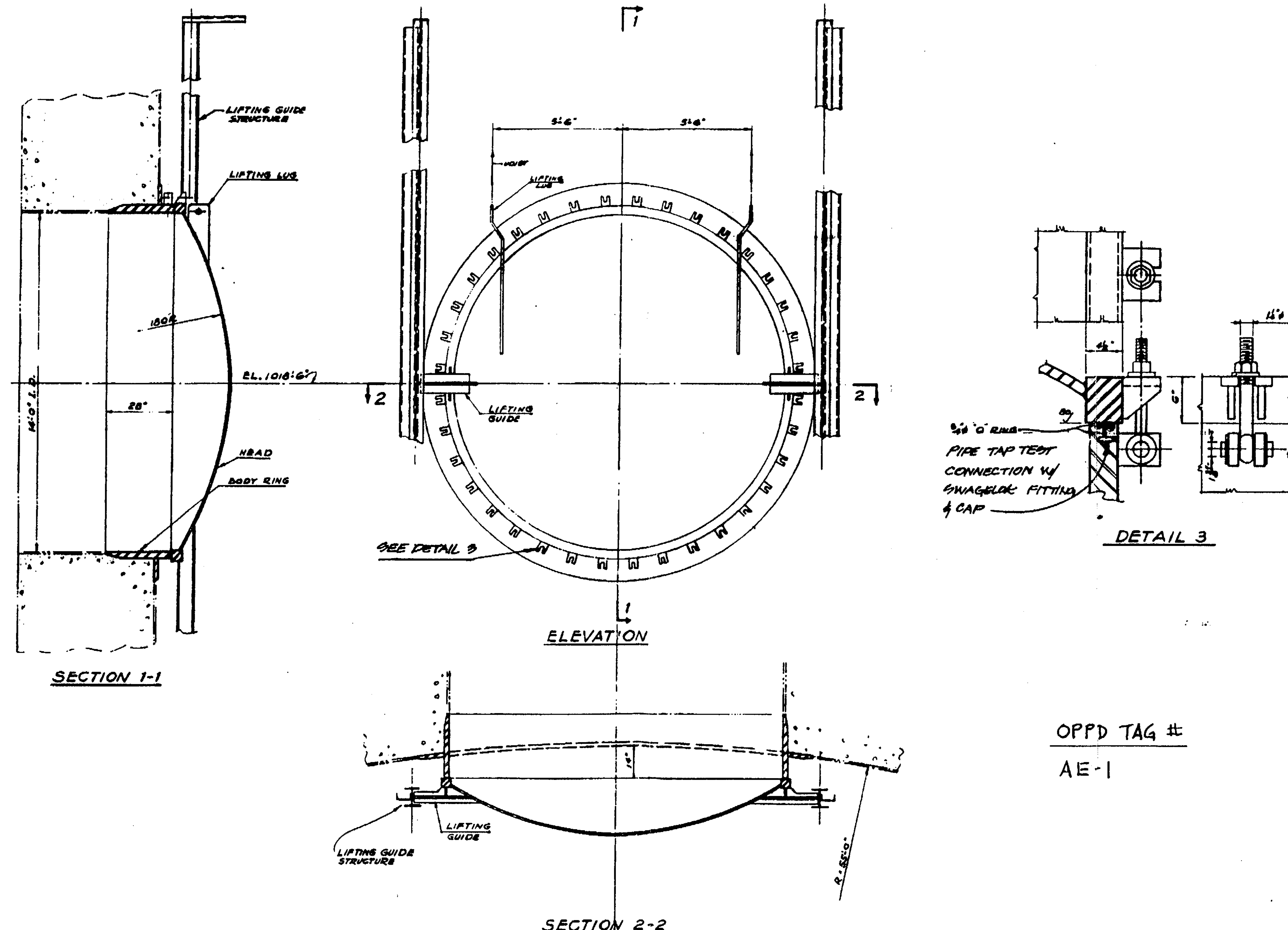
GIBBS, HILL, DURHAM & RICHARDSON, INC.
CONSULTING ENGINEERS
OMAHA, NEBRASKA

USAR FIGURE 5.9-10

DRAWING NO. 11405-M-70, SH.10



FORT CALHOUN STATION		
PERSONNEL AIR LOCK		
OPPD DWG. FIGURE 5.9-11		
REV.SH. 31459	APVD. TEB 11-9-2000	REV 3
FILE 36529		



OPPD TAG #

AE-1

FORT CALHOUN STATION

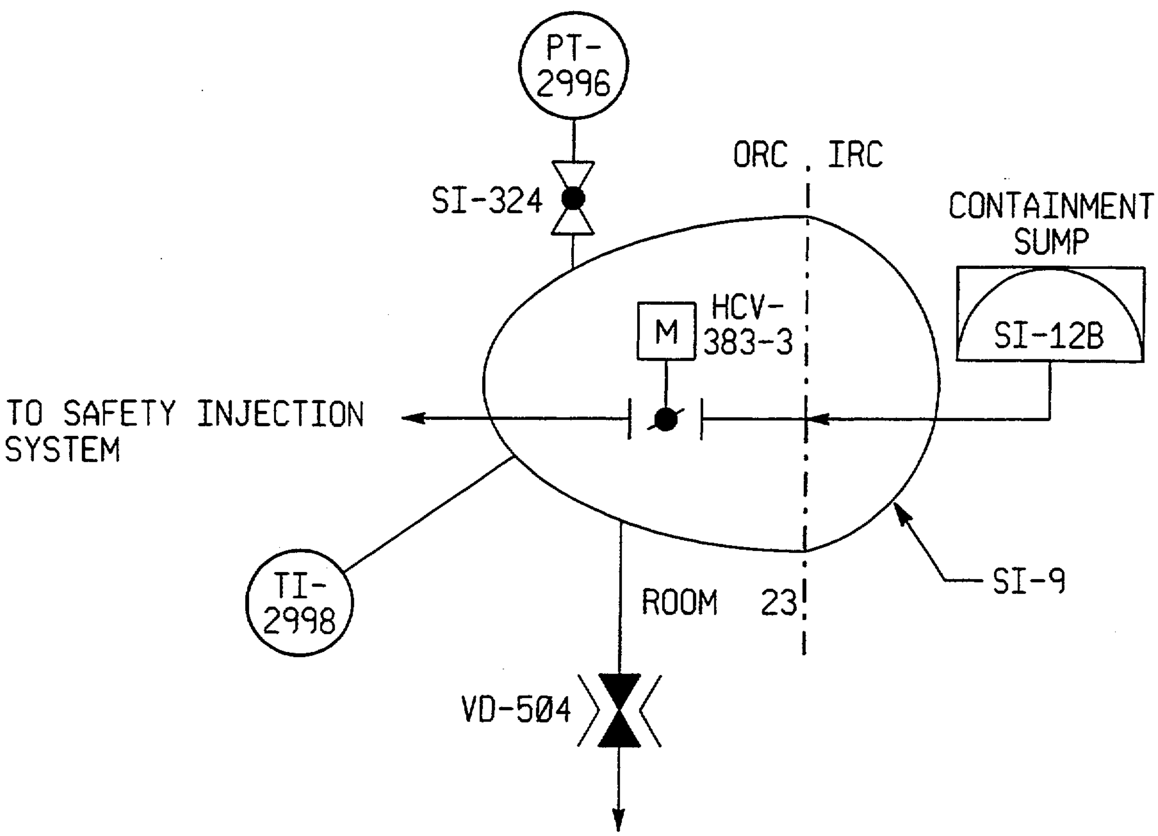
EQUIPMENT ACCESS HATCH

OPPD

DWG. FIGURE 5.9-12

REV.SH. 33758	APVD	REV
FILE 36531	TEP 11-9-2000	2

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-383-3	AS IS	OPEN	RAS
SI-324	——	——	——
VD-5Ø4	——	——	——

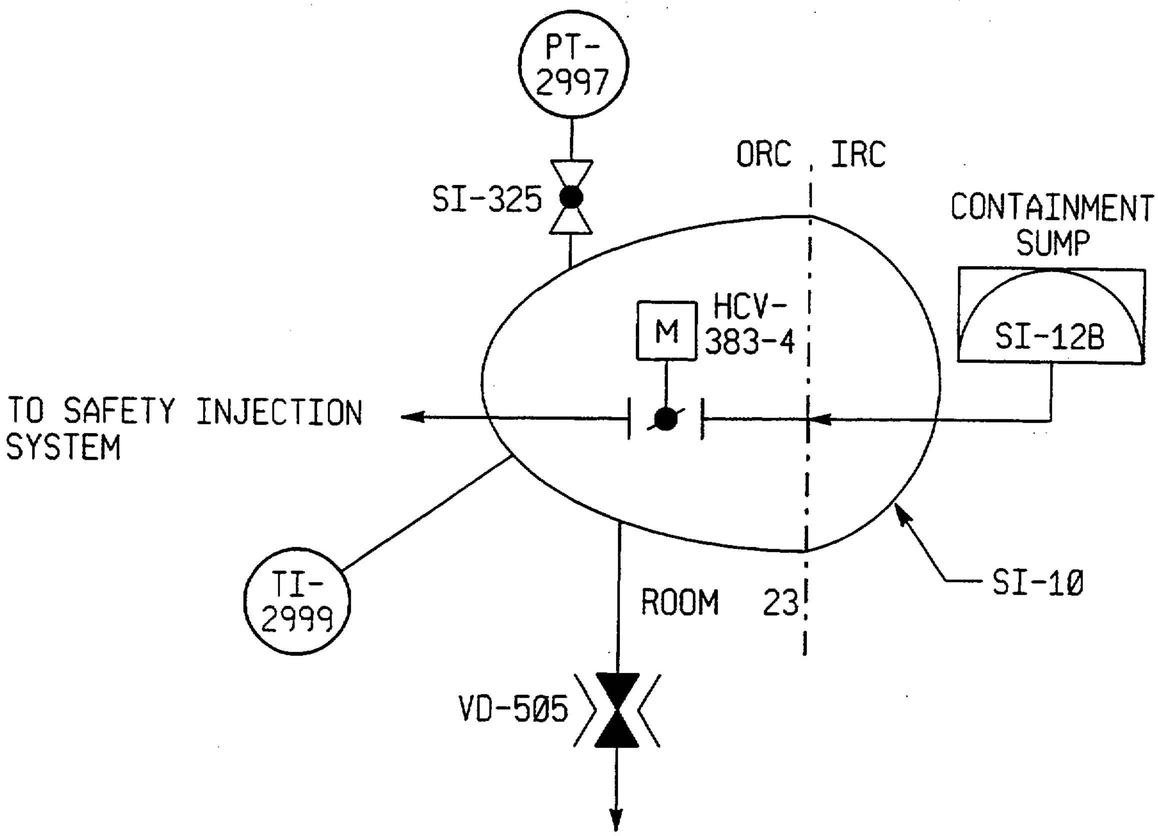


NOTE:
Vessel sealed to containment and pipe
(See Section 6.2.3.8)

Reference Drawings:
1. 114Ø5-MECH-1
2. E-23866-21Ø-13Ø, SH.3

OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	PENETRATION M-HCV-383-3 CONTAINMENT SUMP RECIRCULATION (Containment Atmosphere Exposed System)	**USAR** Figure 5.9-13 Sheet Ø1 REV.SH. 19986 · FILE 4Ø122 · APVD 3-21-01 · REV 5

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-383-4	AS IS	OPEN	RAS
SI-325	—	—	—
VD-505	—	—	—



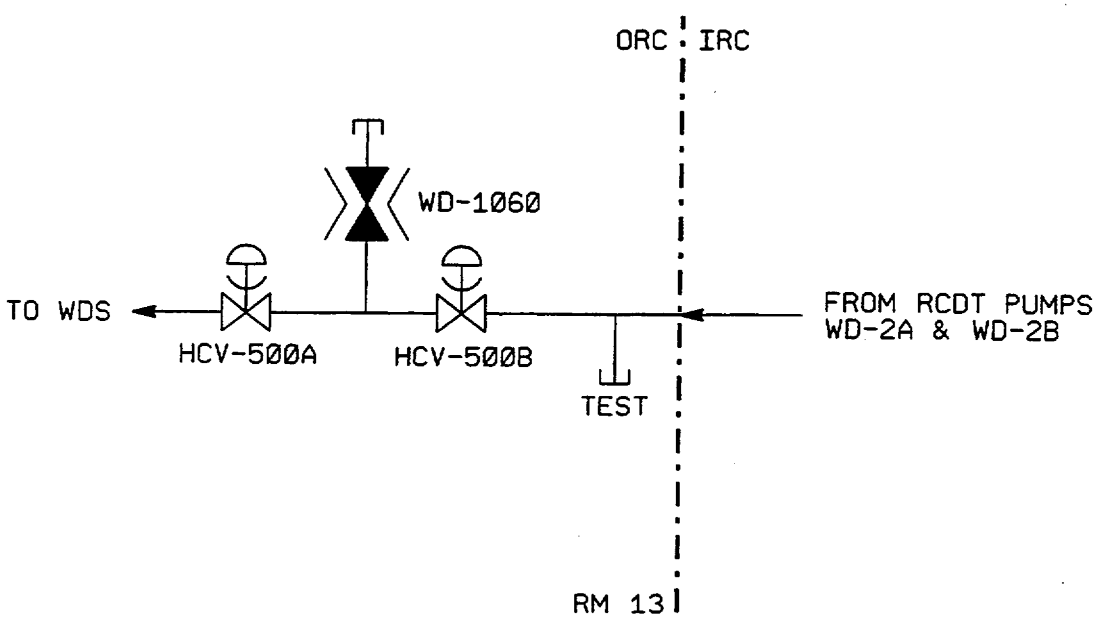
NOTE:
Vessel sealed to containment and pipe
(See Section 6.2.3.8)

Reference Drawings:

1. 11405-MECH-1
2. E-23866-210-130, SH.3

OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	PENETRATION M-HCV-383-4 CONTAINMENT SUMP RECIRCULATION (Containment Atmosphere Exposed System)	**USAR** Figure 5.9-13 Sheet 02 REV.SH. 19987 APVD REV 5 FILE 40123

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-500A	CLOSED	CLOSED	CIAS
HCV-500B	CLOSED	CLOSED	CIAS
WD-1060	—	—	—

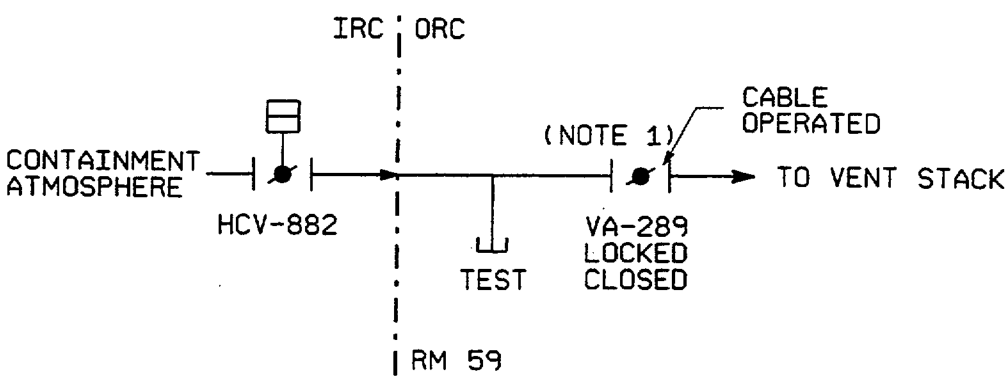


Reference Drawings:

1. 11405-MECH-1
2. 11405-M-6

OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION UNIT NO. 1	PENETRATION M-20 RC DRAIN TANK PUMP DISCHARGE TO WDS (Closed System)	USAR Figure 5.9-13 Sheet 19
		REV.SH. 20003 / FILE 40139 / APVD 3-27-01 JMc / REV 5

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-882	OPEN	CLOSED	CIAS
VA-289	—	—	—



NOTES:
1. OPEN DURING PURGING ONLY

Reference Drawings:
1. 11405-M-MECH-1
2. 11405-M-1

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO. 1

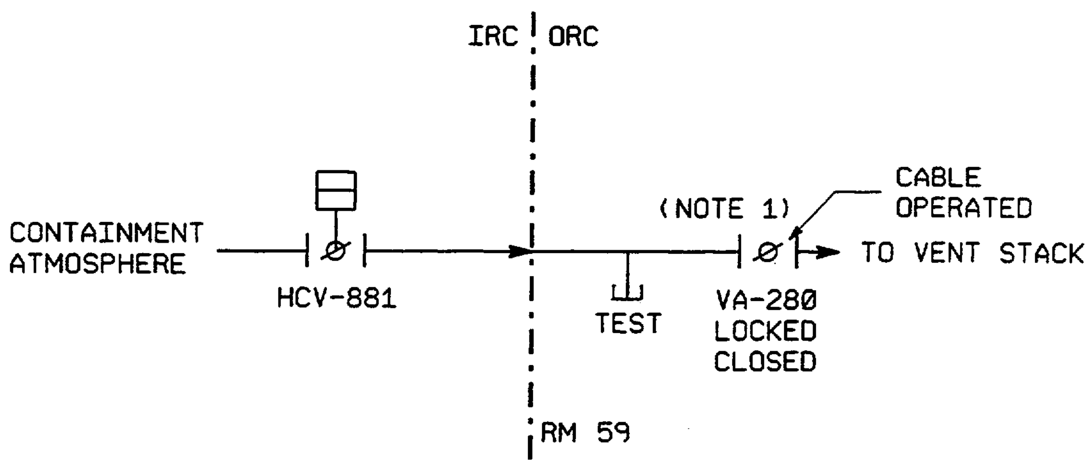
PENETRATION M-30
CONTAINMENT HYDROGEN PURGE
(Containment Atmosphere Exposed System)

USAR

Figure 5.9-13 Sheet 23

REV.SH. 20007	APVD 3-27-01	REV 3
FILE 40143		

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-881	OPEN	CLOSED	CIAS
VA-280	—	—	—

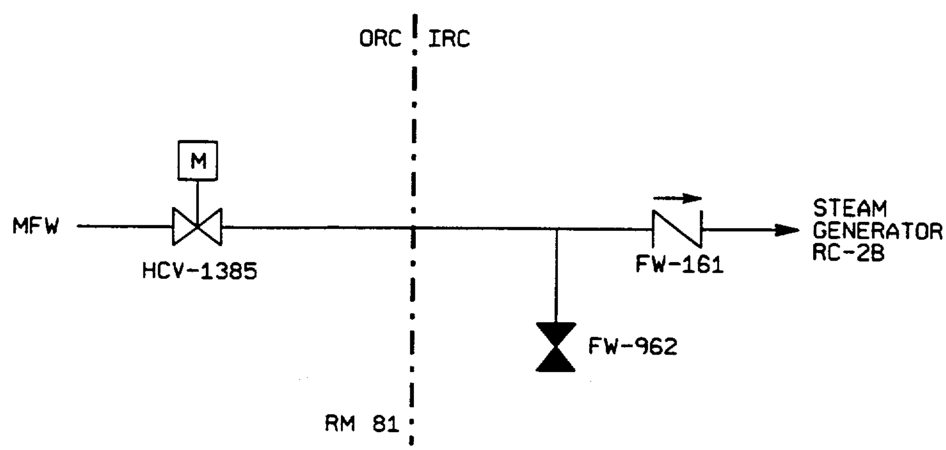


NOTES:
1. OPEN DURING PURGING ONLY.

Reference Drawings:
1. 11405-M-MECH-1
2. 11405-M-1

OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION UNIT NO. 1	PENETRATION M-69 HYDROGEN PURGE (Containment Atmosphere Exposed System)	**USAR** Figure 5.9-13 Sheet 43
		REV.SH. 20027 / FILE 40163 / APVD 3-27-01 JMc / REV 3

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-1385	AS IS	CLOSED	SGIS
FW-962	—	—	—



Reference Drawings:

1. 11405-M-MECH-1
2. 11405-M-253

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO. 1

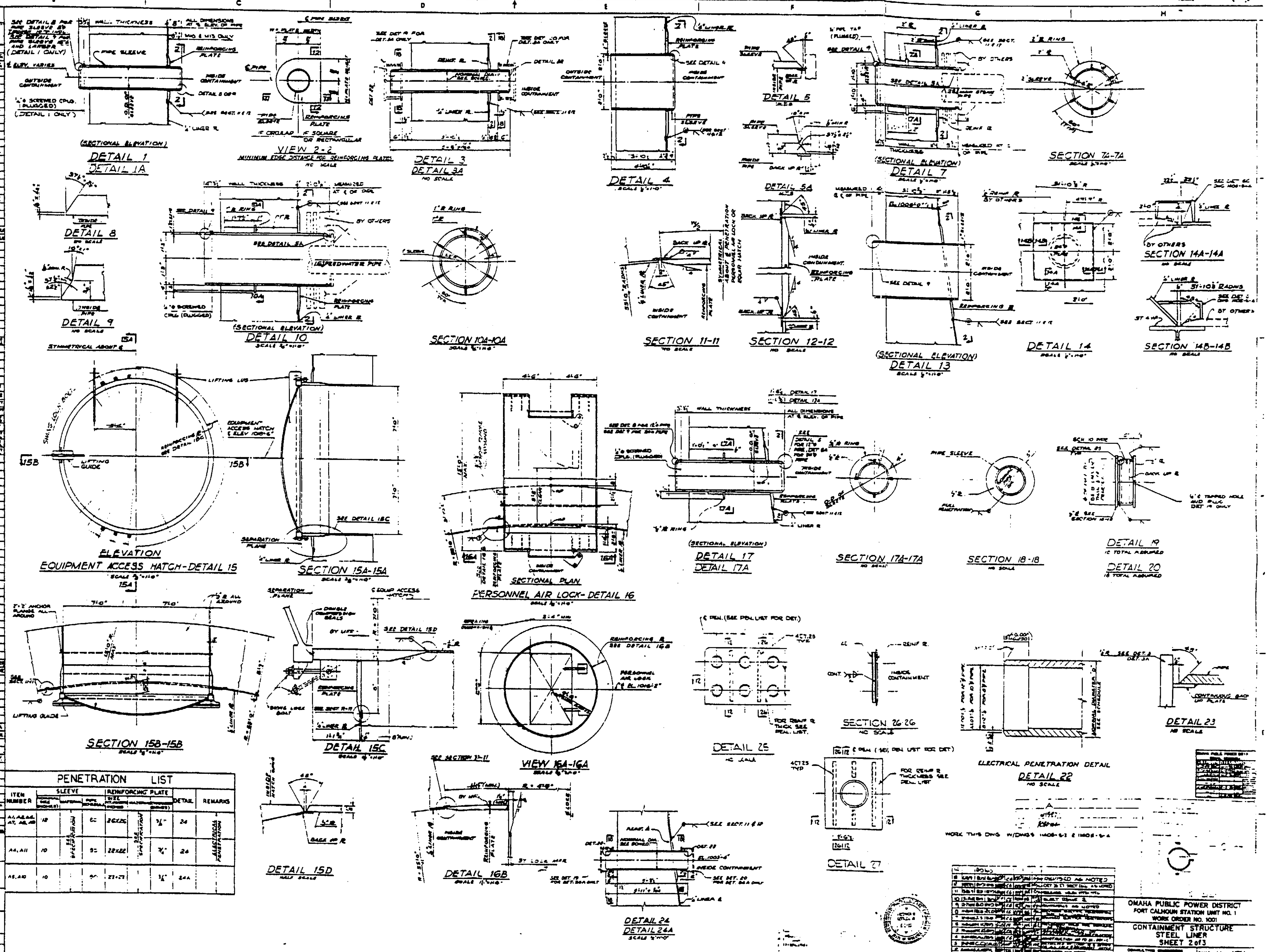
PENETRATION M-93
FEEDWATER TO RC-2B SUPPLY
(Closed System)

USAR

Figure 5.9-13 Sheet 61

REV.SH. 20046	APVD	REV
FILE 40182	9/18/00	2

PENETRATION LIST



PENETRATION LIST

ITEM NUMBER	SLEEVE			REINFORCING PLATE			DETAIL	REMARKS
A1, A2, A6, A7, A8, A9	12	SEE SPECIFICATION	60	26x26	SEE SPECIFICATION	3/4"	24	ELECTRICAL PENETRATION
A4, A11	10		90	22x22		3/4"	24	
A5, A10	10		90	22x22		3/4"	24A	

WORK THIS DWG. W/DWGS 11405-S-2 & 11405-S-4

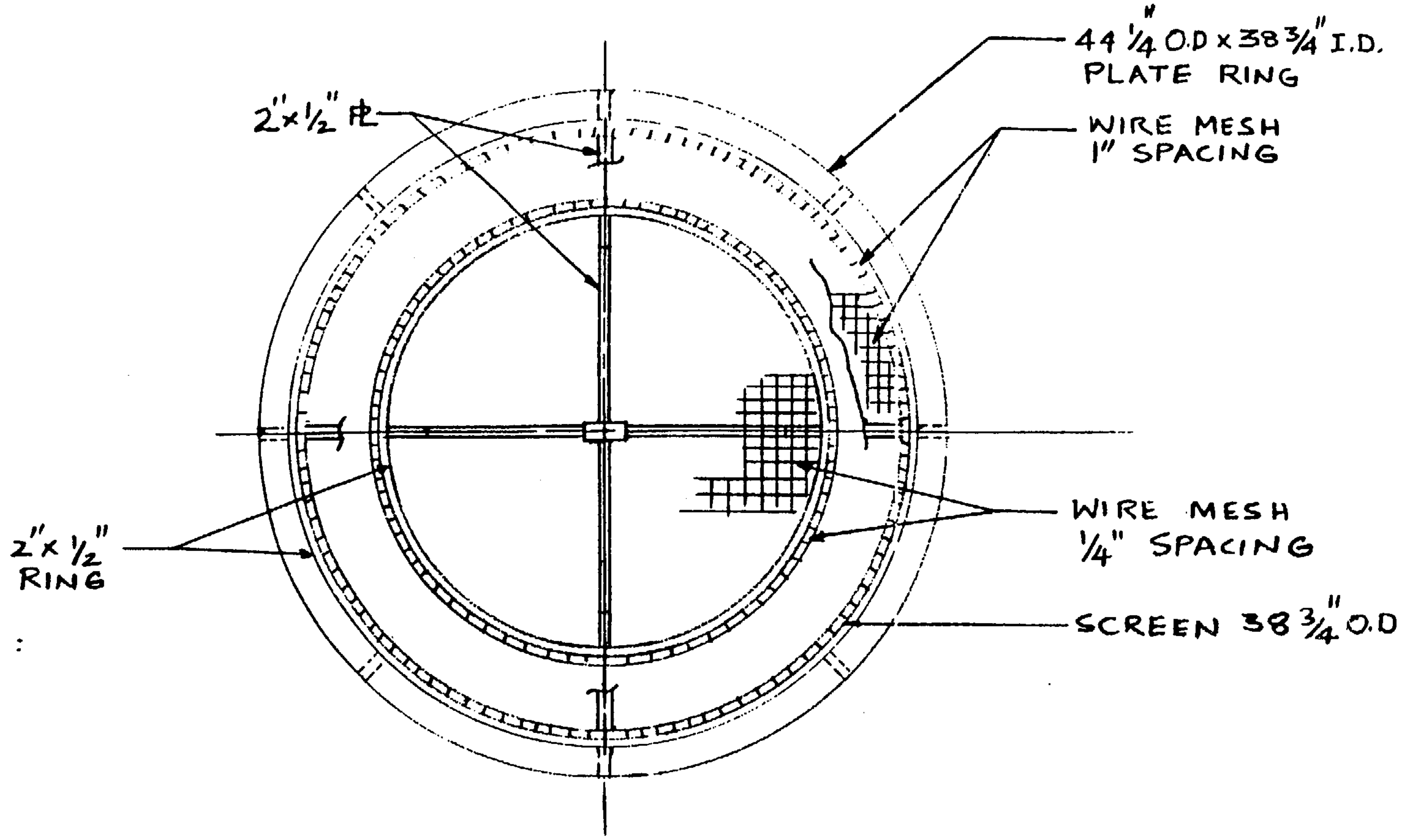
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO. 1
WORK ORDER NO. 1001

CONTAINMENT STRUCTURE
STEEL LINER
SHEET 2 of 3

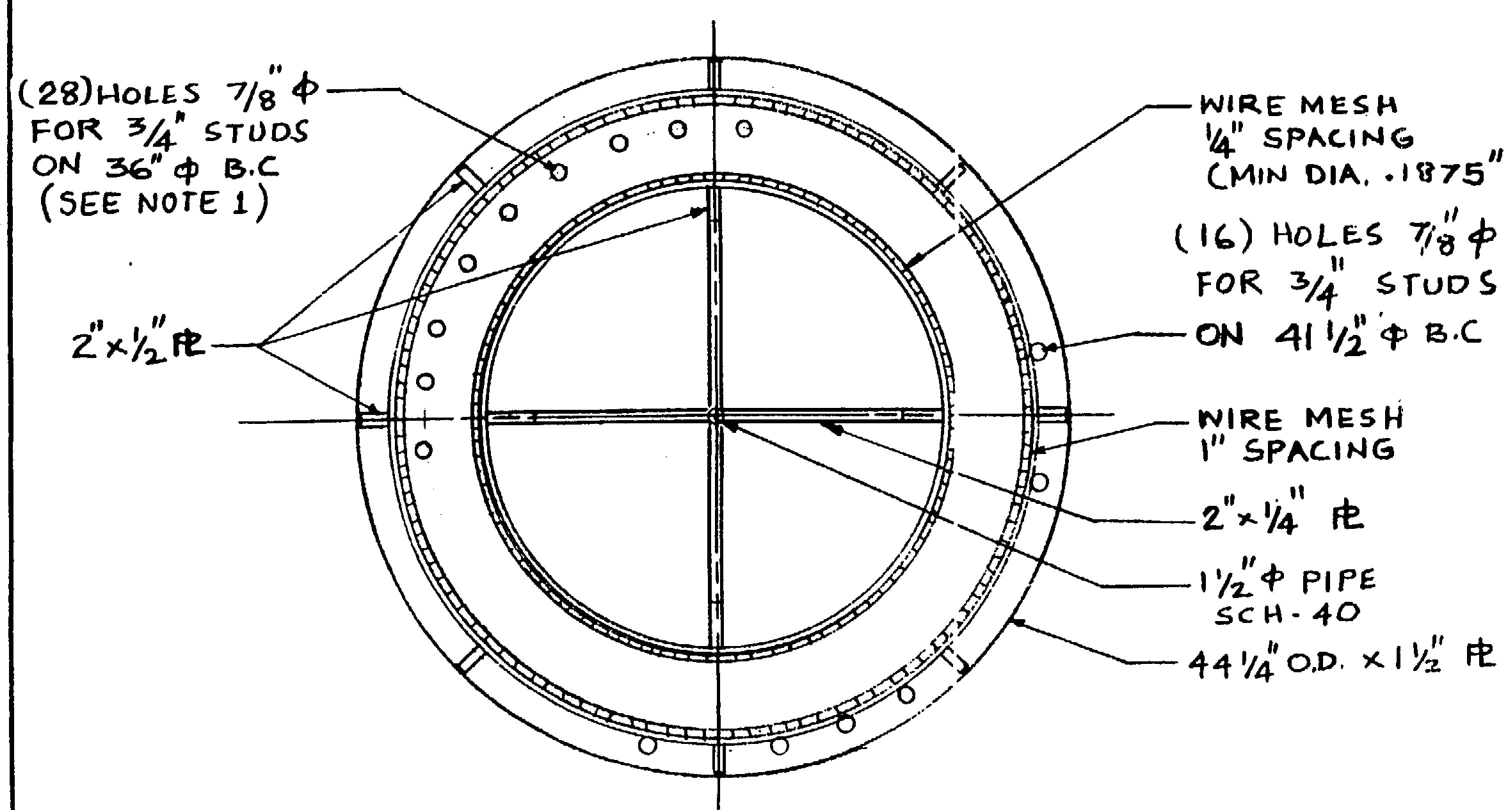
11405-S-3

FIGURE 5.9-15

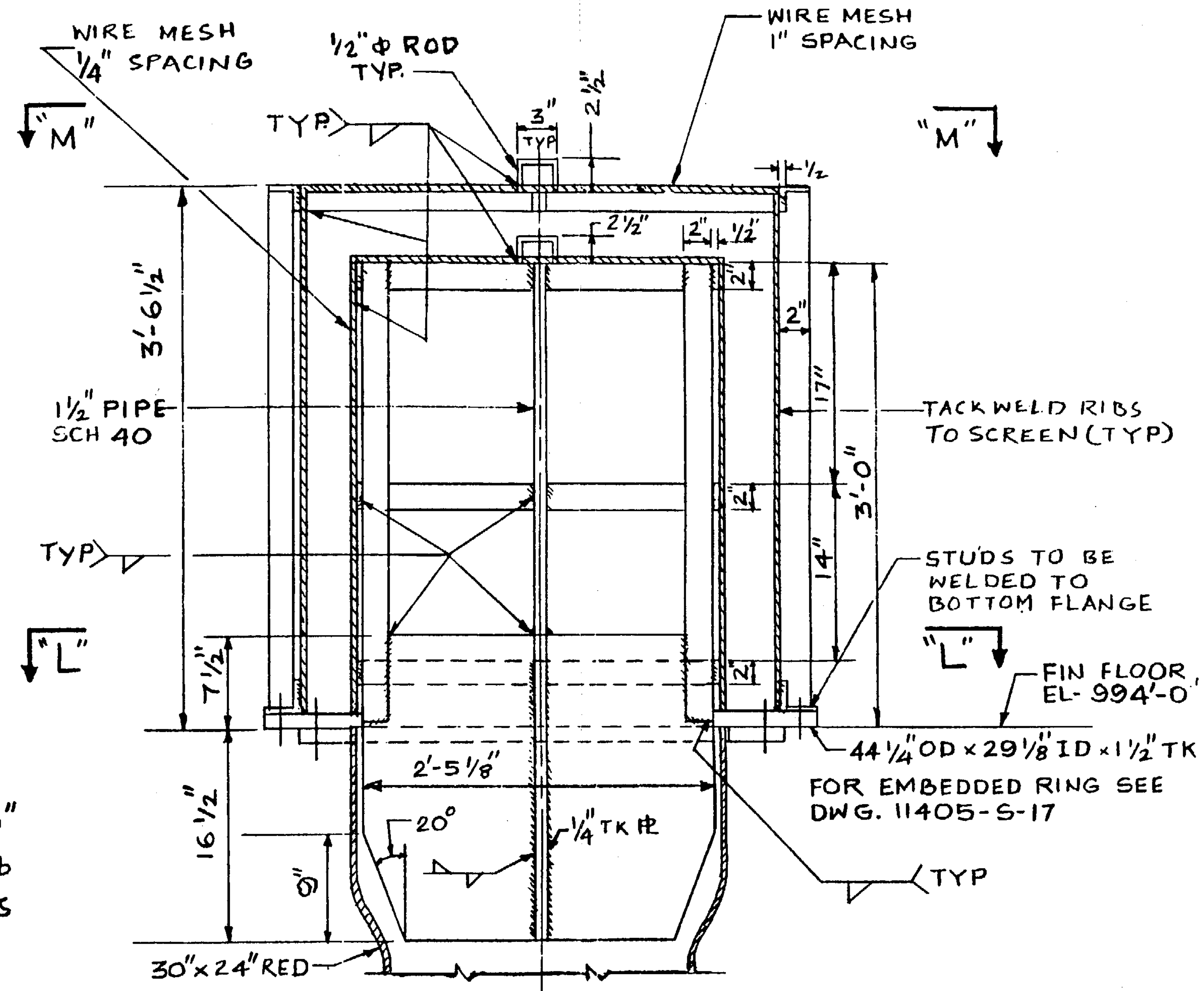
16382



SECTION "M-M"
SCALE: 1"=1'-0"



SECTION "L-L"
SCALE: 1"=1'-0"

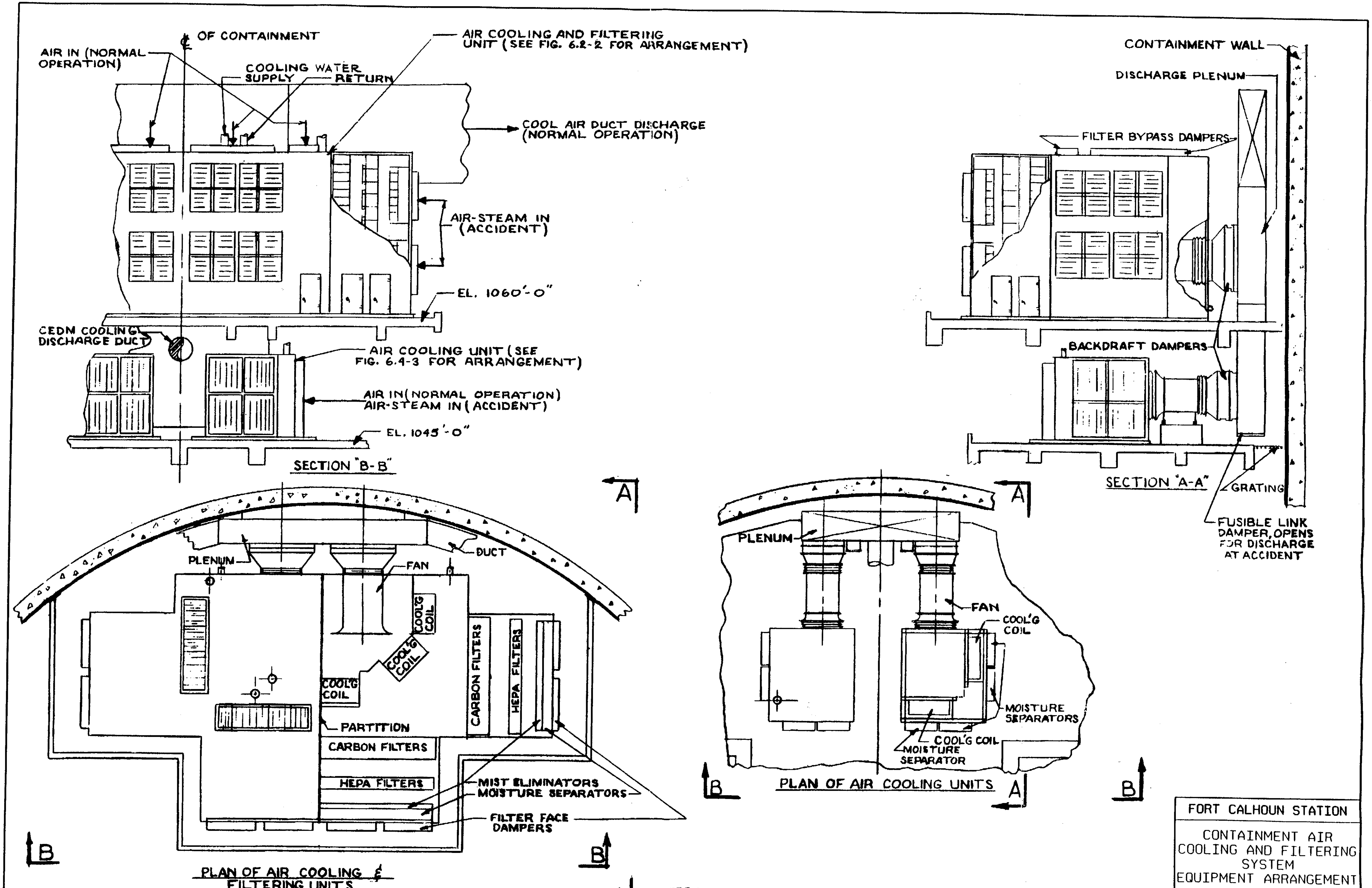


DETAIL "K"
SCALE: 1"=1'-0"

NOTE: 1. ONLY 13-3/4" STUDS ARE USED (EVERY OTHER ONE WITH ONE MISSING). THE MINIMUM NUMBER OF STUDS REQ'D IS 4 PER FC05981.

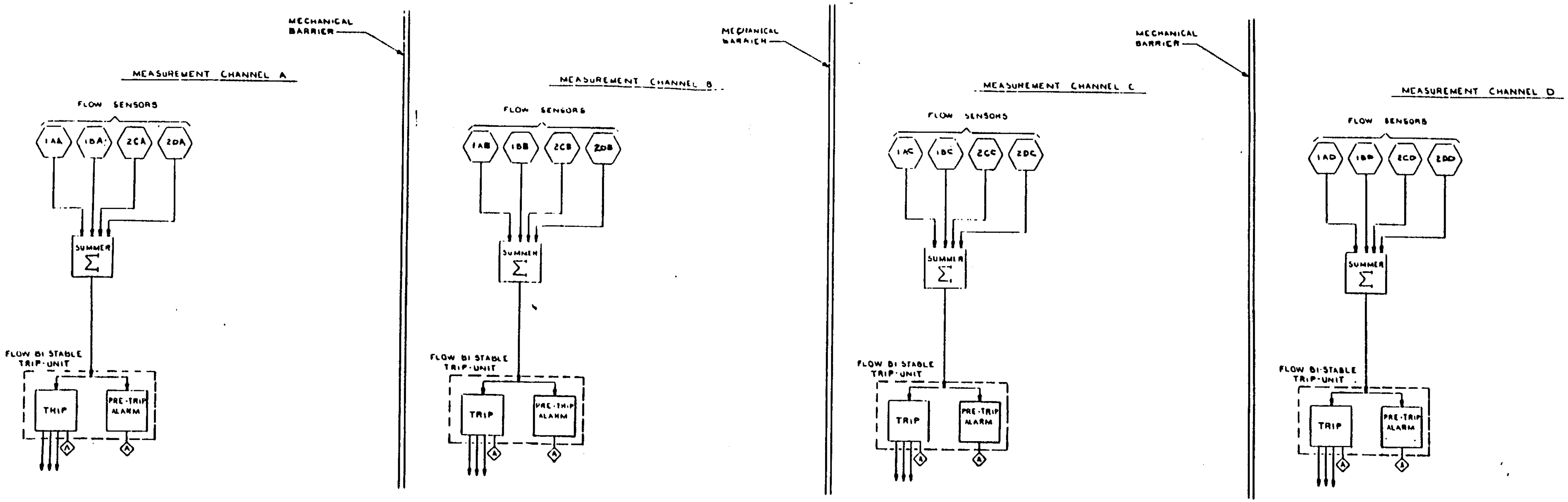
36537

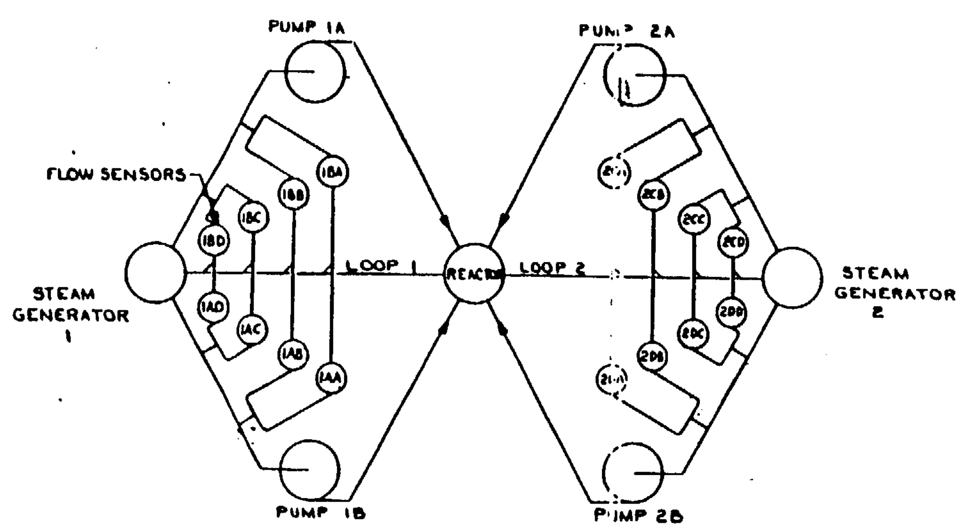
REV.SH. 34474	APVD	REV	Omaha Public Power District	S.I. Recirculation	FIGURE
FILE 36537	LLS 10-92	1	Fort Calhoun Station Unit No. 1	Inlet Strainer	6.2-3



FOR LOCATION OF UNITS IN CONTAINMENT SEE FIGS. 1.2-8 & 12-9

FORT CALHOUN STATION		
CONTAINMENT AIR COOLING AND FILTERING SYSTEM EQUIPMENT ARRANGEMENT		
DWG. USAR FIG. 6.4-1		
REV.SH. ---	APVD	REV
FILE 36541	5/31/01	1





OMAHA PUBLIC POWER DISTRICT OMAHA, NEBRASKA				
RS-		RS-19972	1	REV. NO.
		3-3[illegible]86	TW	DRAWN
		4-4-86	RDC	CHECK
		4-4-86	RA	ENG.
		5-8-86	[illegible]	ELECT.DPT.
		—	—	CIVIL DPT.
		—	—	MECH DPT
		—	—	NUC. DPT.
		—	—	Q.CONTROL
		—	—	P.E

Low Flow Protective System Functional Diagram

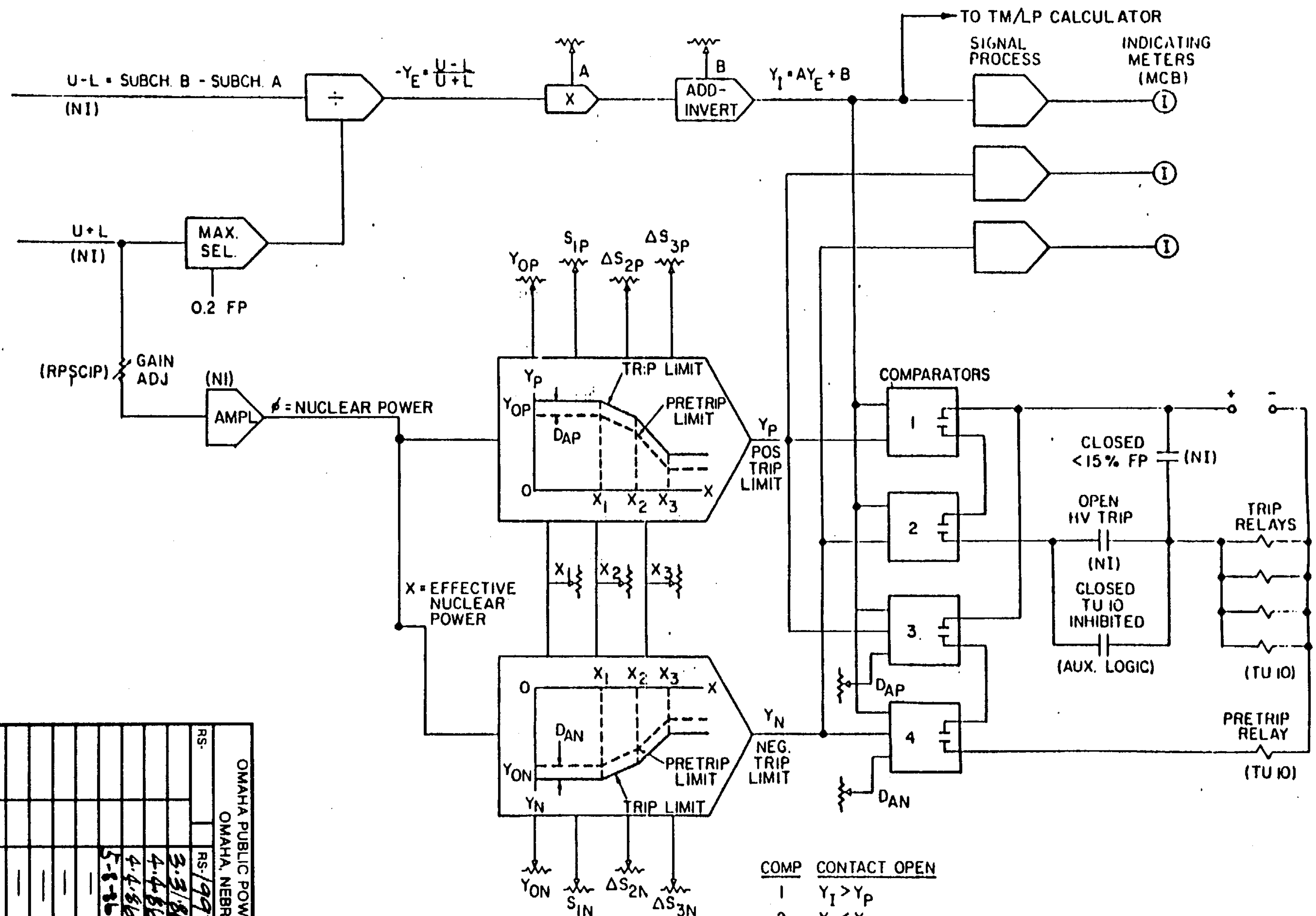
Omaha Public Power District Fort Calhoun Station-Unit No. 1

Figure 7.2-5

Omaha Public Power District
Fort Calhoun Station
Unit No. 1

Axial Power Distribution
Trip System
Functional Diagram

Fig. 7.2-9



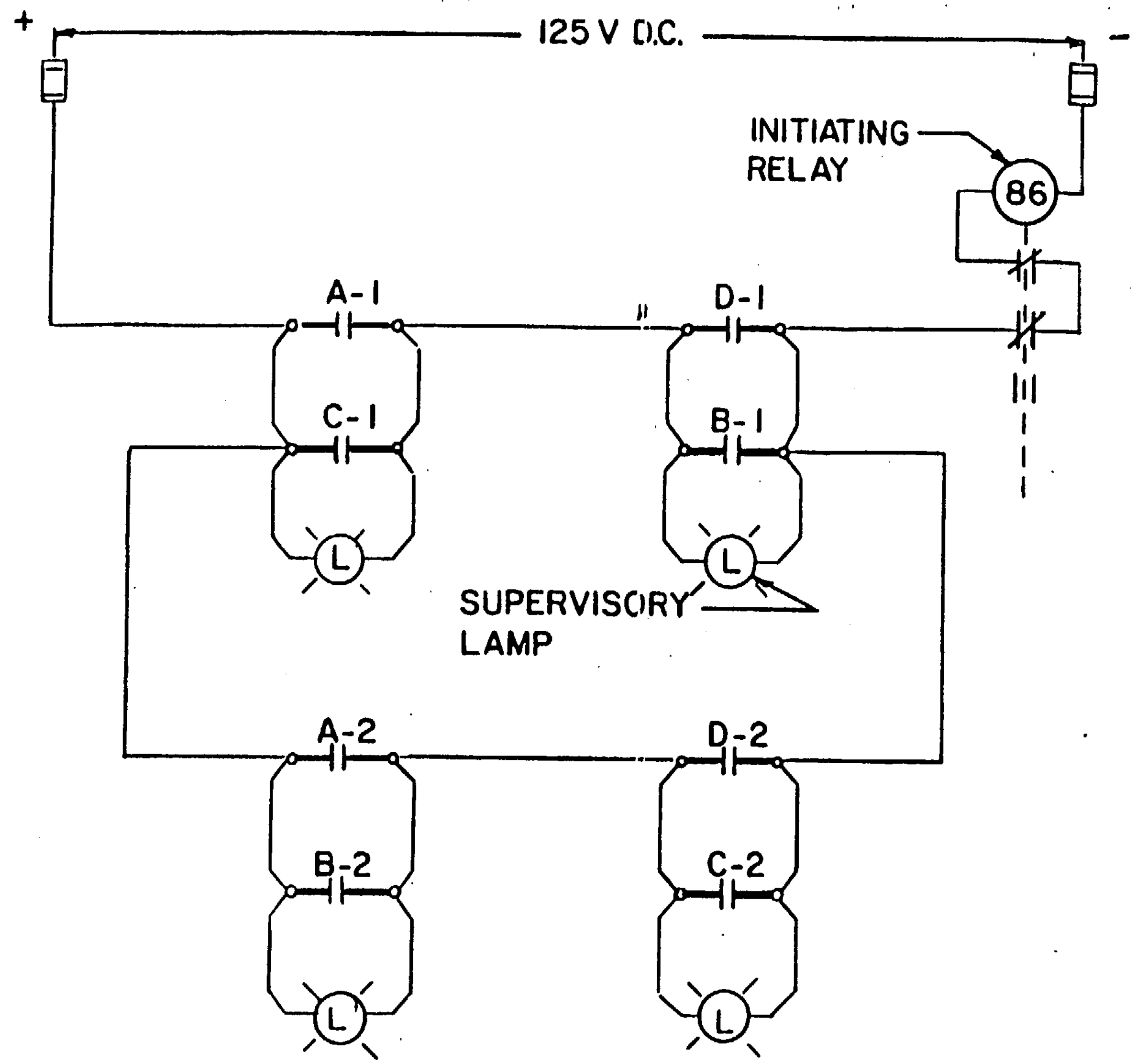
COMP	CONTACT OPEN
1	$Y_I > Y_P$
2	$Y_I < Y_N$
3	$Y_I > Y_P - D_{AP}$
4	$Y_I < Y_N + D_{AN}$

NOTE:
EACH SLOPE POTENTIOMETER SETS THE CHANGE IN SLOPE AT THE CORRESPONDING BREAK POINT. SEE TEXT FOR POTENTIOMETER SCALE FACTORS.

OMAHA PUBLIC POWER DISTRICT
OMAHA, NEBRASKA

RS-	RS-19970	1	REV. NO.
	3-31-86 TW		DRAWN
	4-4-86 RPC		CHECK
	4-4-86 RPC		ENG.
	5-8-86		ELECT.DPT.
	—	—	CIVIL DPT.
	—	—	MECH.DPT.
	—	—	NUC. DPT.
	—	—	Q.CONTROL
	—	—	P.E

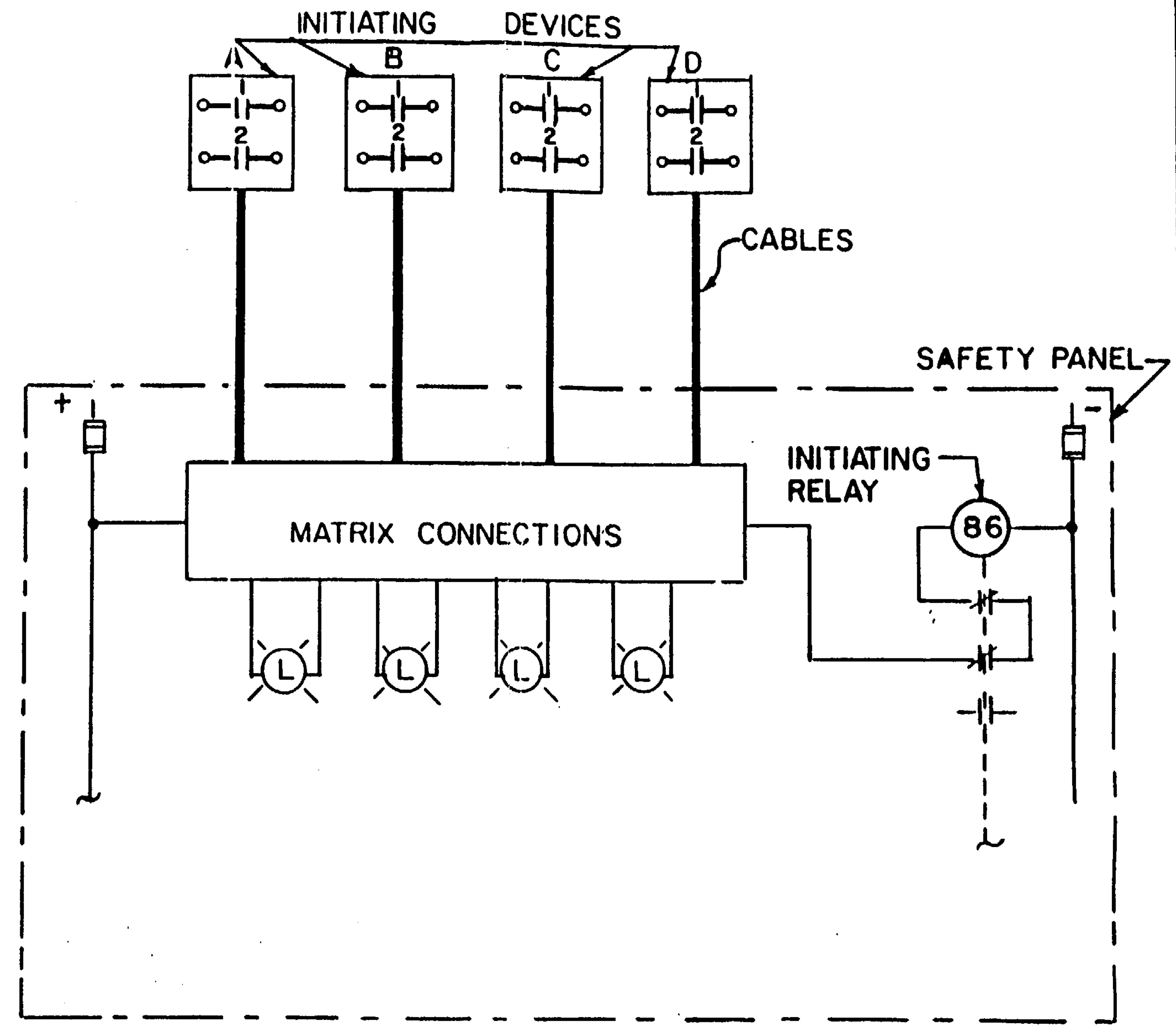
36552 26552



NOTE:

SUPERVISORY LAMPS WILL DETECT OPEN CIRCUIT, SHORT CIRCUIT, OR CONTACT CLOSURE.

POINT TO POINT WIRING DIAGRAM

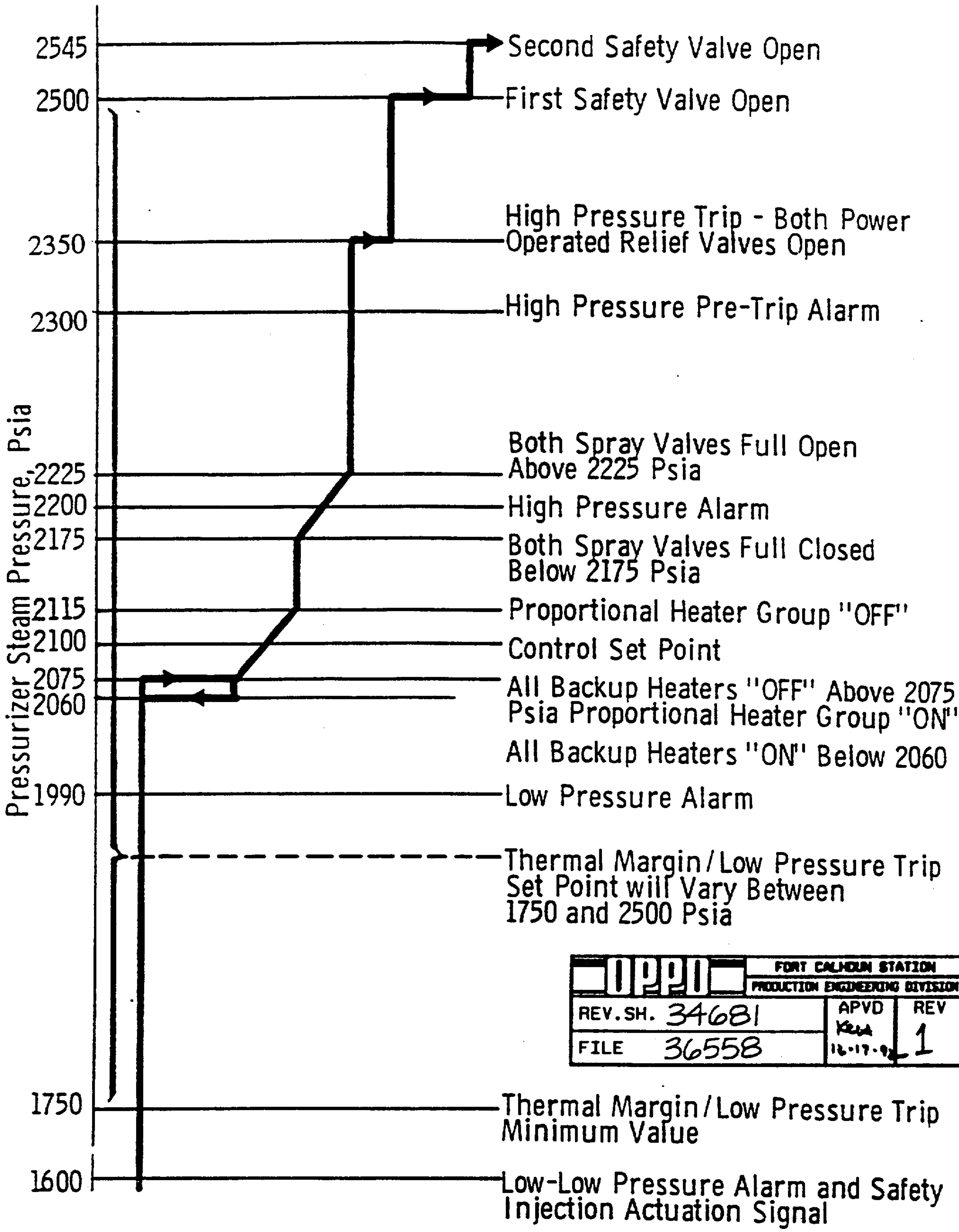


BLOCK WIRING DIAGRAM

36555

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	TYPICAL MATRIX SUPERVISION CHANNEL 'A' OR 'B'	
REV. SH. 37349 FILE 36555	APVD Kperdue 12-19-95	REV 1	OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION-UNIT No. 1

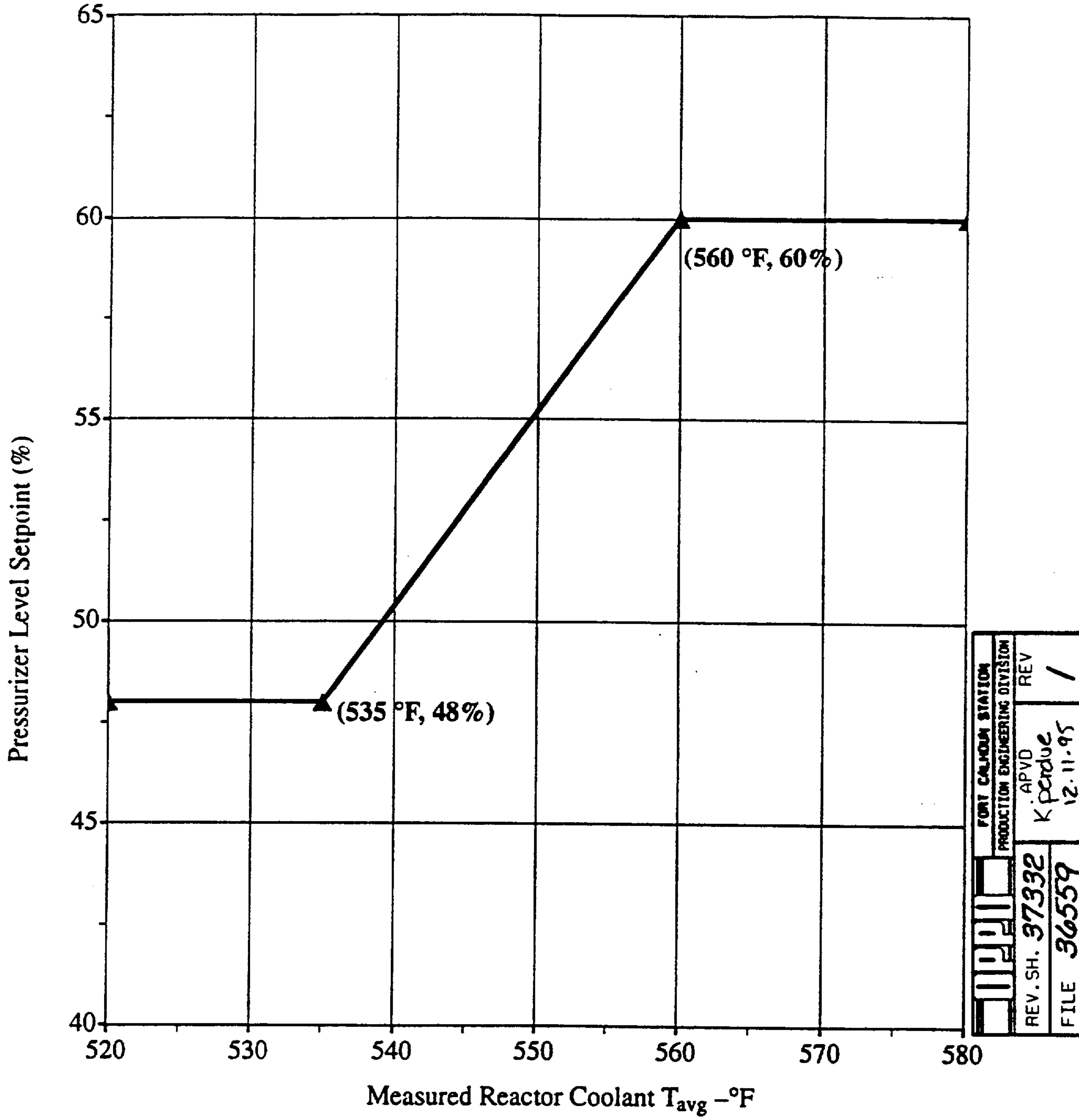
FIG. 7.3-3



Pressure Control Program	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.4-3

USAR FIGURE 7.4-4

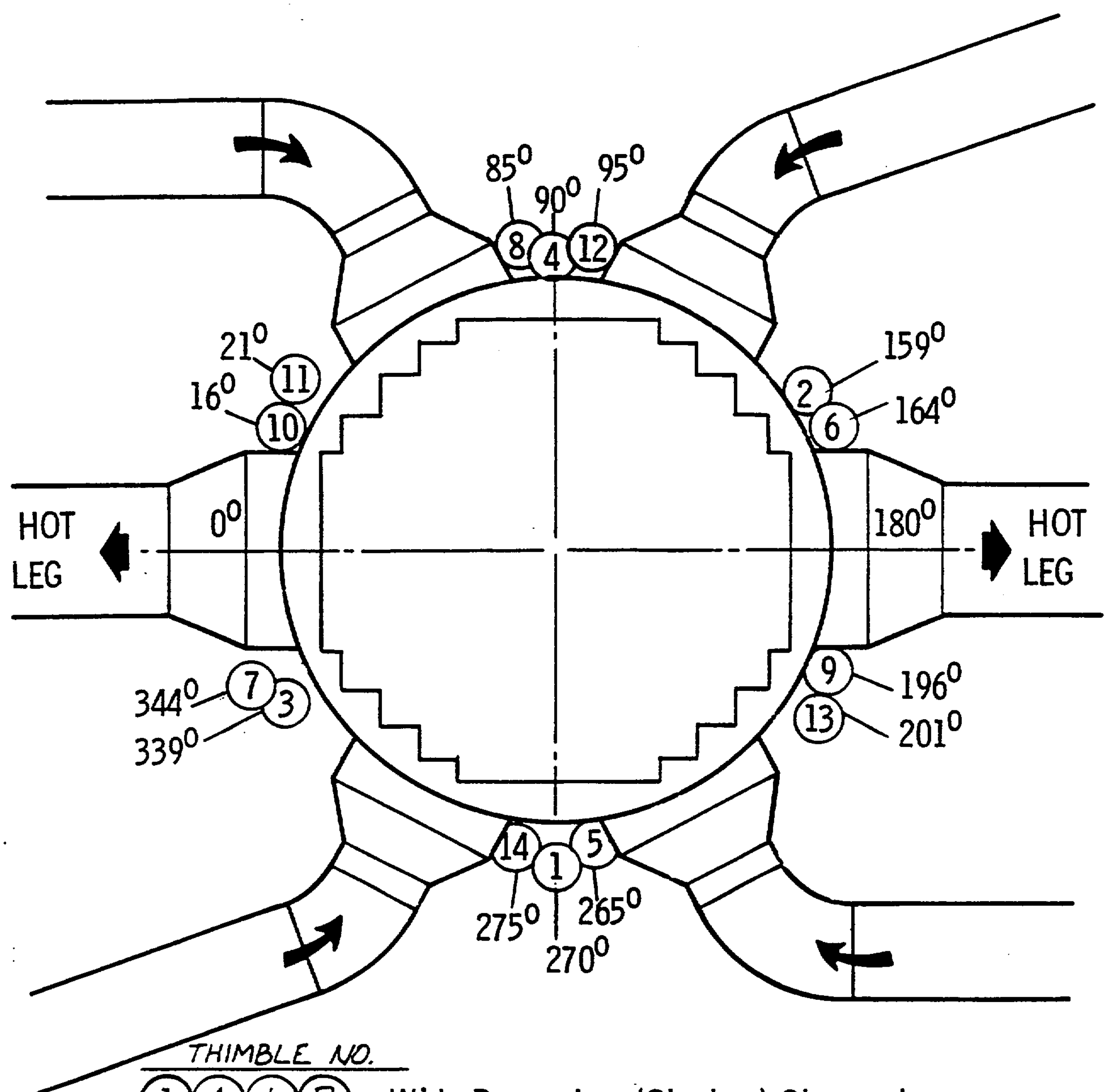
PRESSURIZER LEVEL SETPOINT
VS.
REACTOR COOLANT AVERAGE TEMPERATURE



Pressurizer Level Setpoint	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.4-4

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION		
REV. SH. 35490		APVD MMG 8/4/93	REV 1
FILE 36562			

NORTH



THIMBLE NO.

1 4 6 7 = Wide Range Log (Startup) Channels

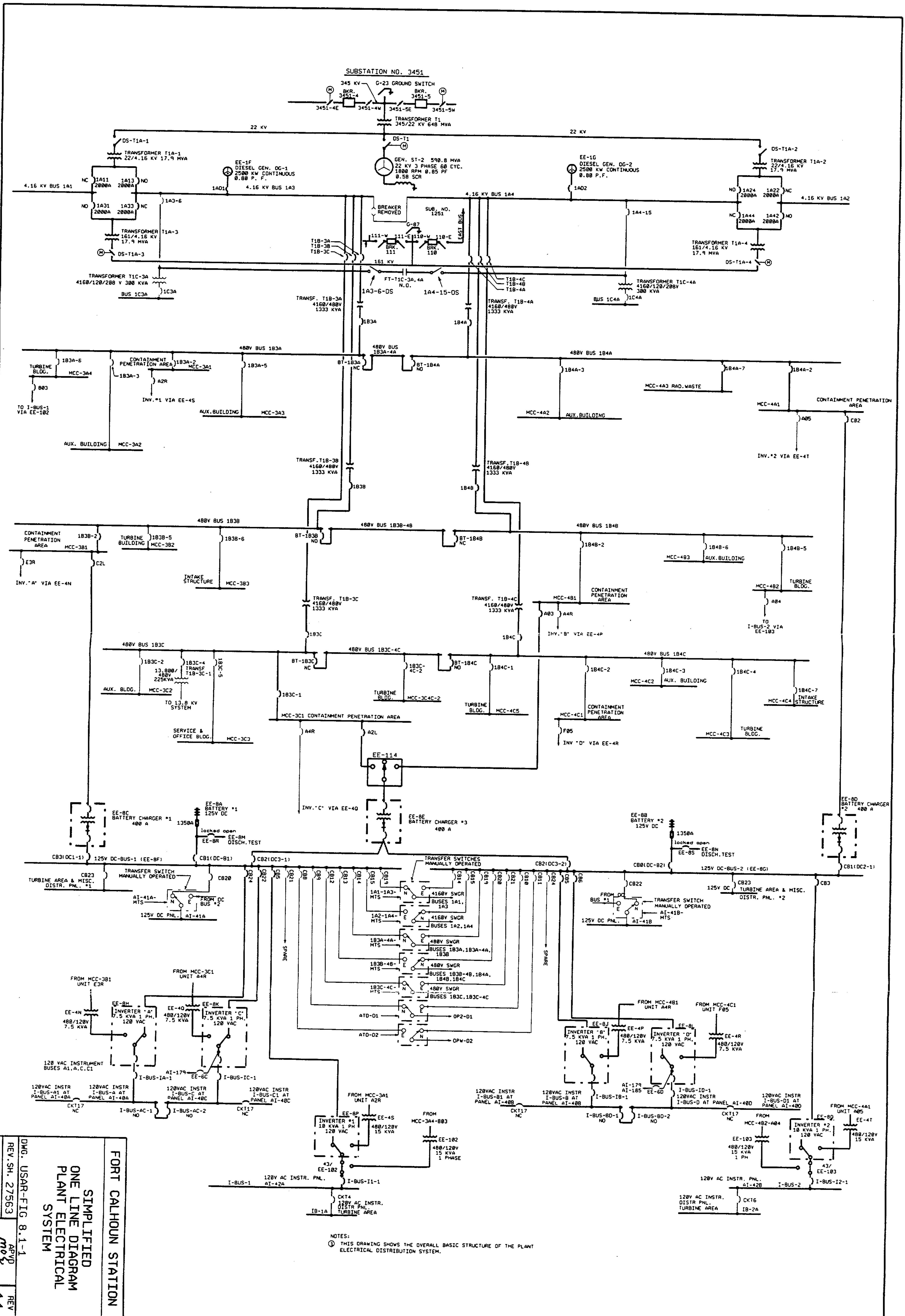
2 3 11 13 = Power Range Safety Channels

5 8 = Power Range Control Channels

9 10 12 14 = Spare Instrument Wells

36562

Nuclear Detector Location	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.5-1



SUBSTATION NO. 3451
345 KV
G-23 GROUND SWITCH
BKR. 3451-4
BKR. 3451-5
3451-4E
3451-4W
3451-5E
3451-5W
TRANSFORMER T1 345/22 KV 648 MVA
22 KV
DS-T1
GEN. ST-2 590.8 MVA 22 KV 3 PHASE 60 CYC. 1800 RPM 0.85 PF 0.58 SCR
DS-T1A-1
TRANSFORMER T1A-1 22/4.16 KV 17.9 MVA
DS-T1A-2
TRANSFORMER T1A-2 22/4.16 KV 17.9 MVA
EE-1F DIESEL GEN. DG-1 2500 KW CONTINUOUS 0.80 P. F.
EE-1G DIESEL GEN. DG-2 2500 KW CONTINUOUS 0.80 P.F.
4.16 KV BUS 1A1
4.16 KV BUS 1A3
4.16 KV BUS 1A4
4.16 KV BUS 1A2
BREAKER REMOVED
SUB. NO. 1251
TRANSFORMER T1A-3 161/4.16 KV 17.9 MVA
DS-T1A-3
TRANSFORMER T1A-4 161/4.16 KV 17.9 MVA
DS-T1A-4
161 KV
FT-T1C-3A,4A N.O.
1A3-6-DS
1A4-15-DS
TRANSFORMER T1C-3A 4160/120/208 V 300 KVA
BUS 1C3A
TRANSFORMER T1C-4A 4160/120/208V 300 KVA
BUS 1C4A
TRANSF. T1B-3A 4160/480V 1333 KVA
TRANSF. T1B-4A 4160/480V 1333 KVA
480V BUS 1B3A
480V BUS 1B3A-4A
480V BUS 1B4A
TRANSF. T1B-3B 4160/480V 1333 KVA
TRANSF. T1B-4B 4160/480V 1333 KVA
480V BUS 1B3B
480V BUS 1B3B-4B
480V BUS 1B4B
TRANSF. T1B-3C 4160/480V 1333 KVA
TRANSF. T1B-4C 4160/480V 1333 KVA
480V BUS 1B3C
480V BUS 1B3C-4C
480V BUS 1B4C
MCC-3C1 CONTAINMENT PENETRATION AREA
EE-114
EE-8C BATTERY CHARGER #1 400 A
EE-8A BATTERY #1 125V DC
EE-8E BATTERY CHARGER #3 400 A
EE-8B BATTERY #2 125V DC
EE-8D BATTERY CHARGER #2 400 A
125V DC-BUS-1 (EE-8F)
125V DC-BUS-2 (EE-8G)
TRANSFER SWITCHES MANUALLY OPERATED
INVERTER "A" 7.5 KVA 1 PH. 120 VAC
INVERTER "C" 7.5 KVA 1 PH. 120 VAC
INVERTER "B" 7.5 KVA 1 PH. 120 VAC
INVERTER "D" 7.5 KVA 1 PH. 120 VAC
INVERTER #1 10 KVA 1 PH 120 VAC
INVERTER #2 10 KVA 1 PH. 120 VAC
NOTES:
① THIS DRAWING SHOWS THE OVERALL BASIC STRUCTURE OF THE PLANT ELECTRICAL DISTRIBUTION SYSTEM.
FORT CALHOUN STATION
SIMPLIFIED
ONE LINE DIAGRAM
PLANT ELECTRICAL
SYSTEM
DWG. USAR-FIG 8.1-1
REV.SH. 27563
FILE 54187
APVD
REV
11

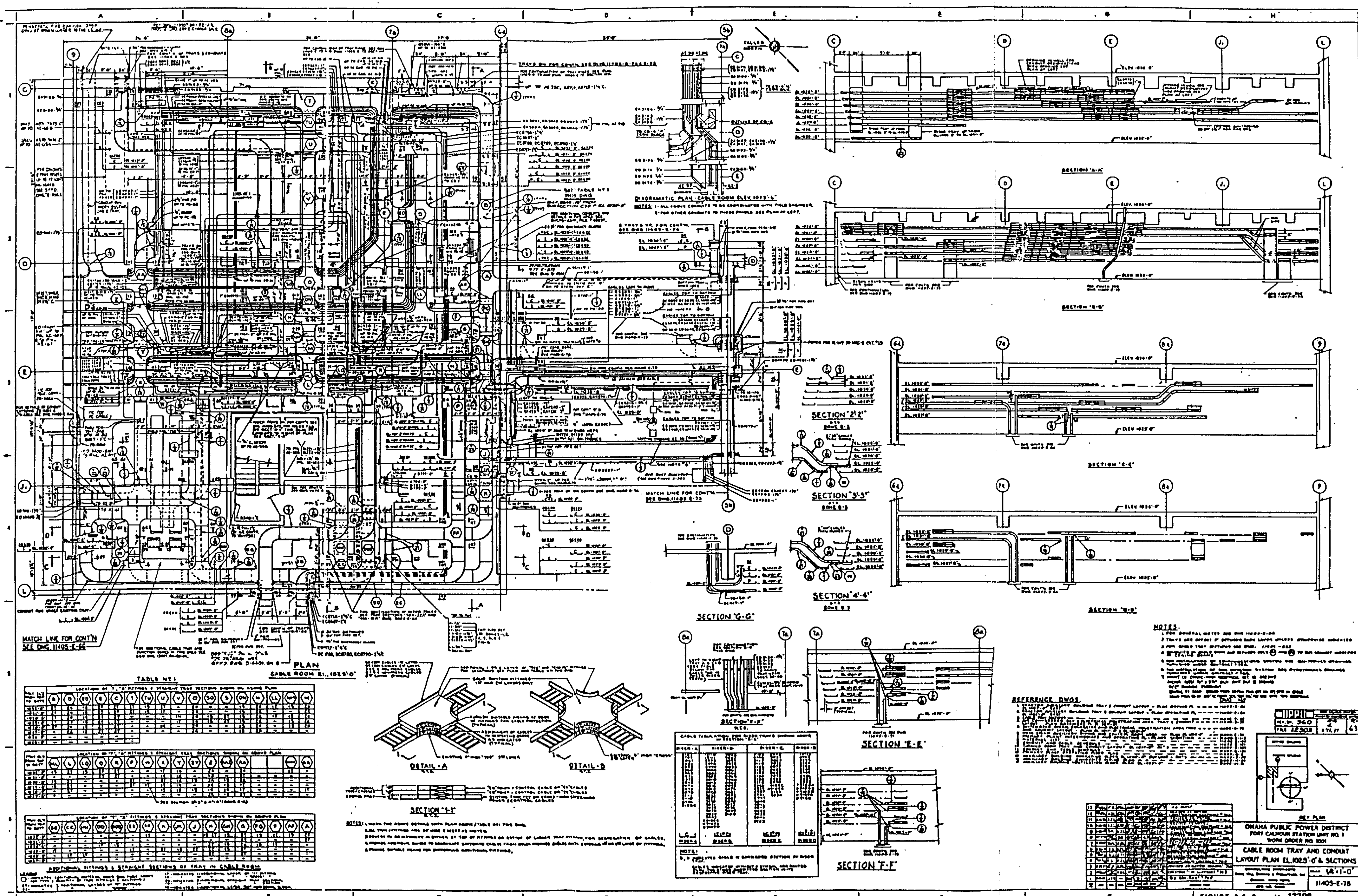
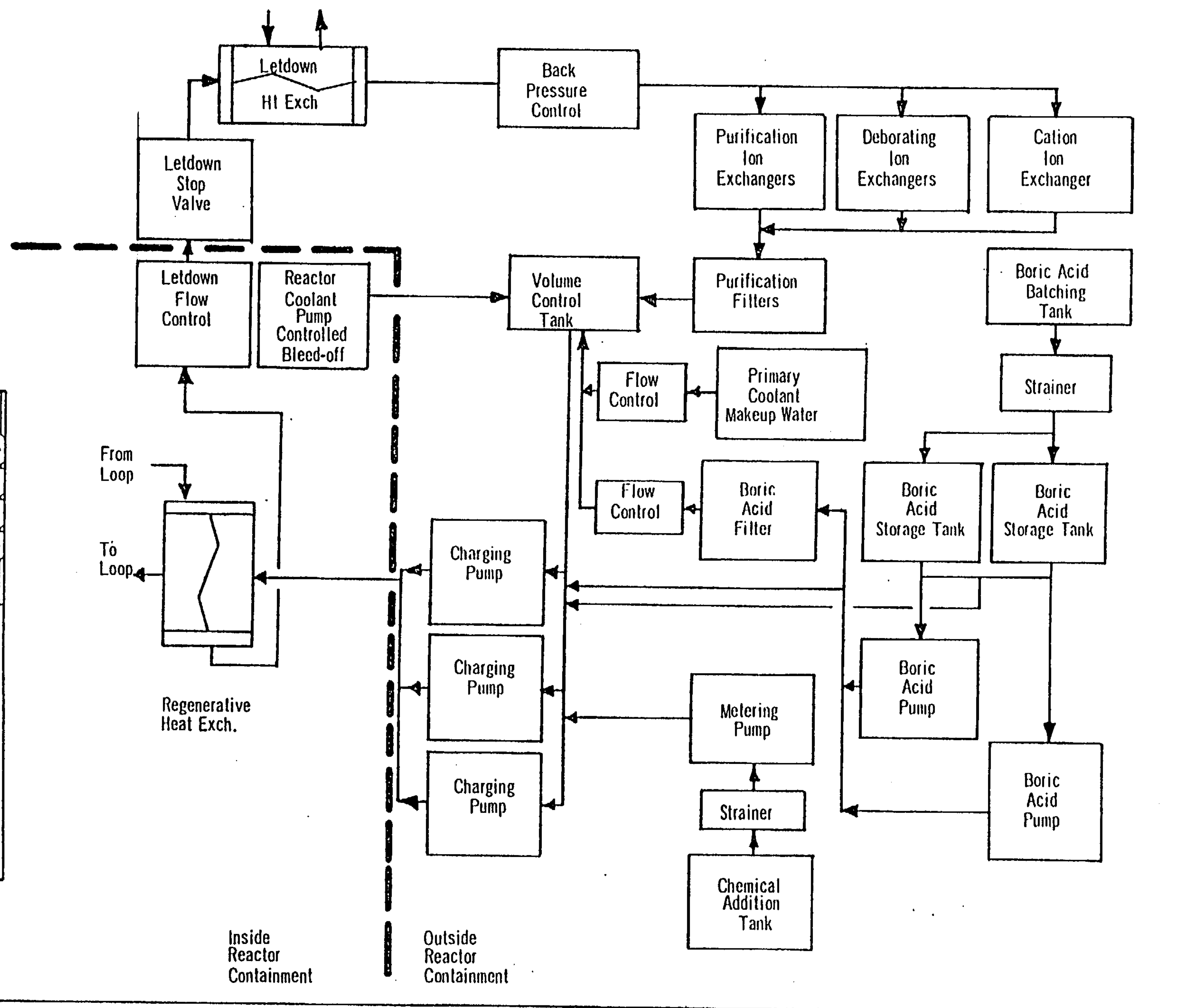


FIGURE 8.5-2

H 12309



Chemical And Volume Control Sys Flow Schematic

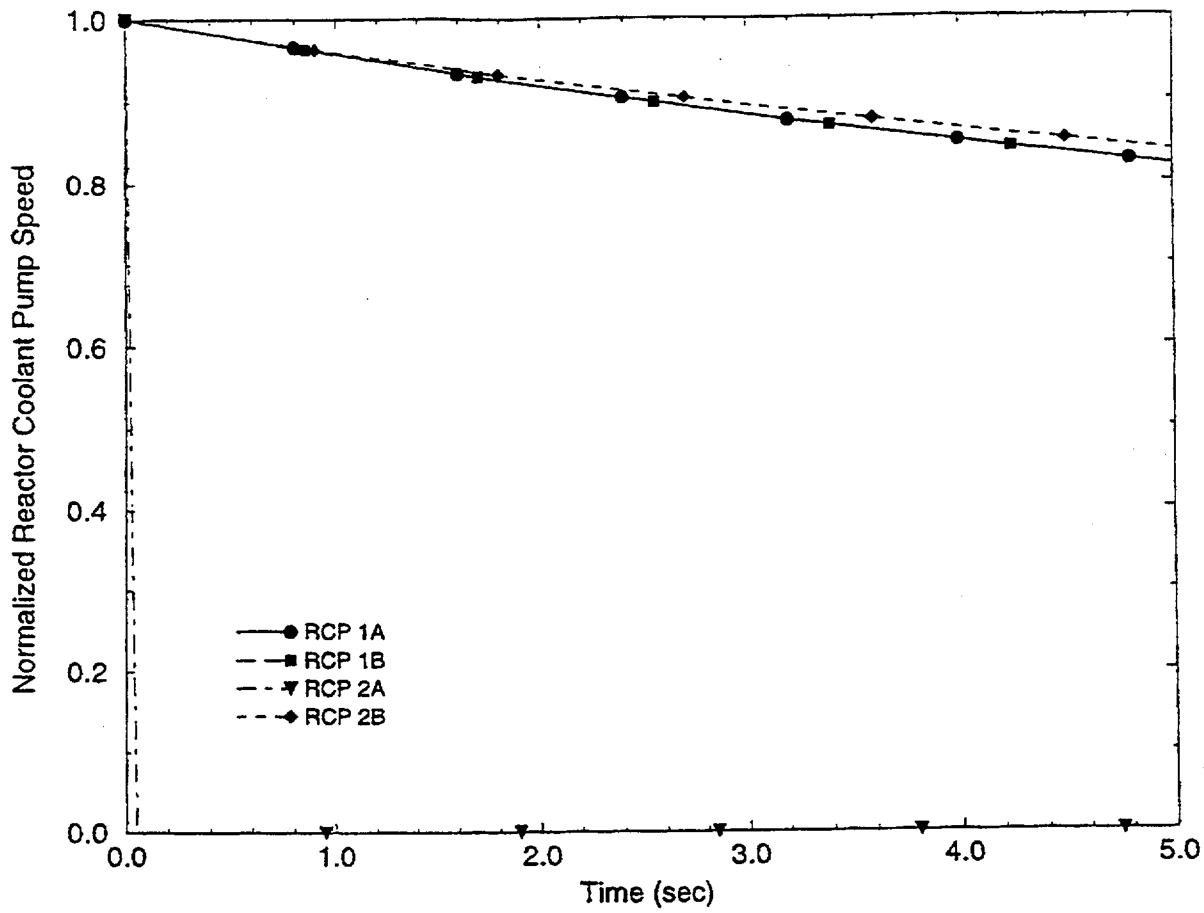
Omaha Public Power District
Fort Calhoun Station-Unit No.1

Figure 9.2-1

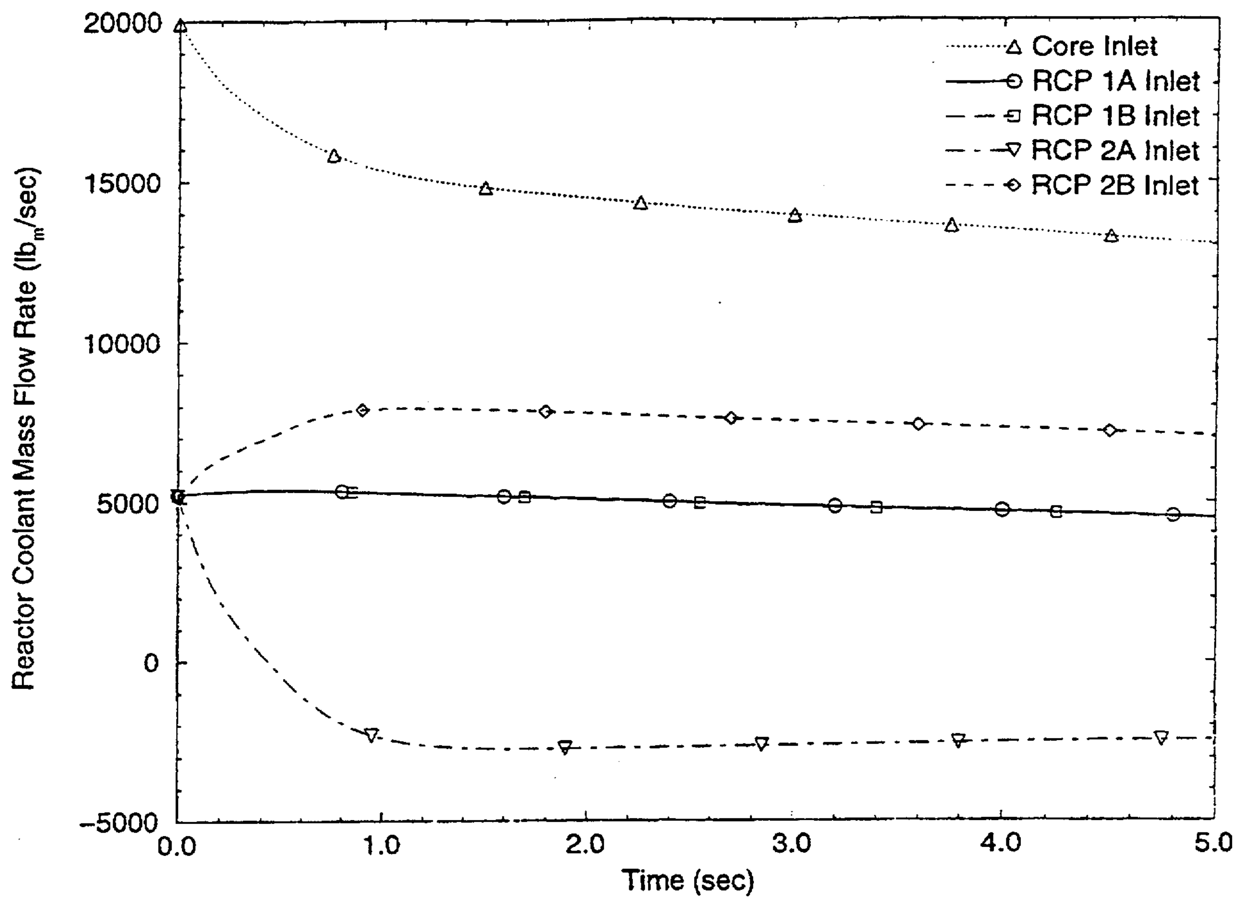
FILE 36570 | REV.SH. — | APVD 5/23/97 | REV 1

FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

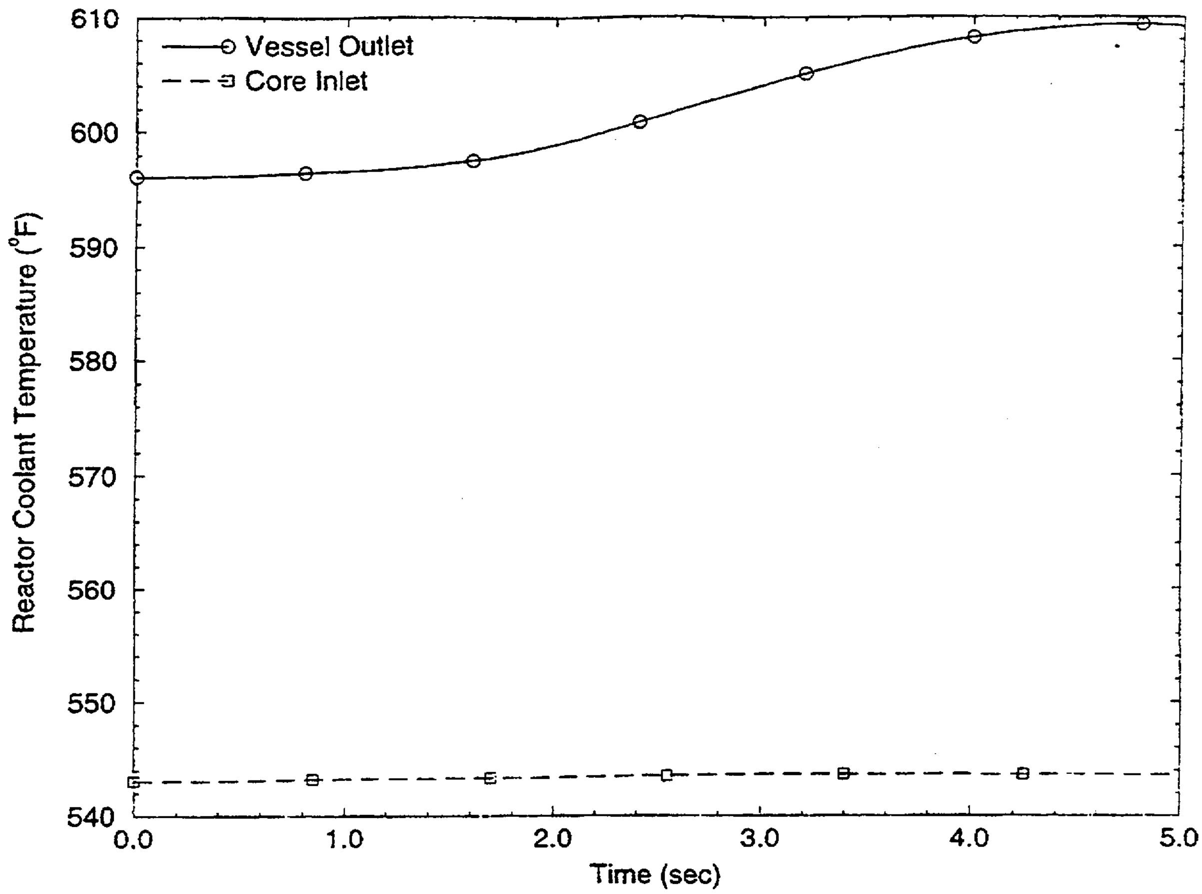
36570



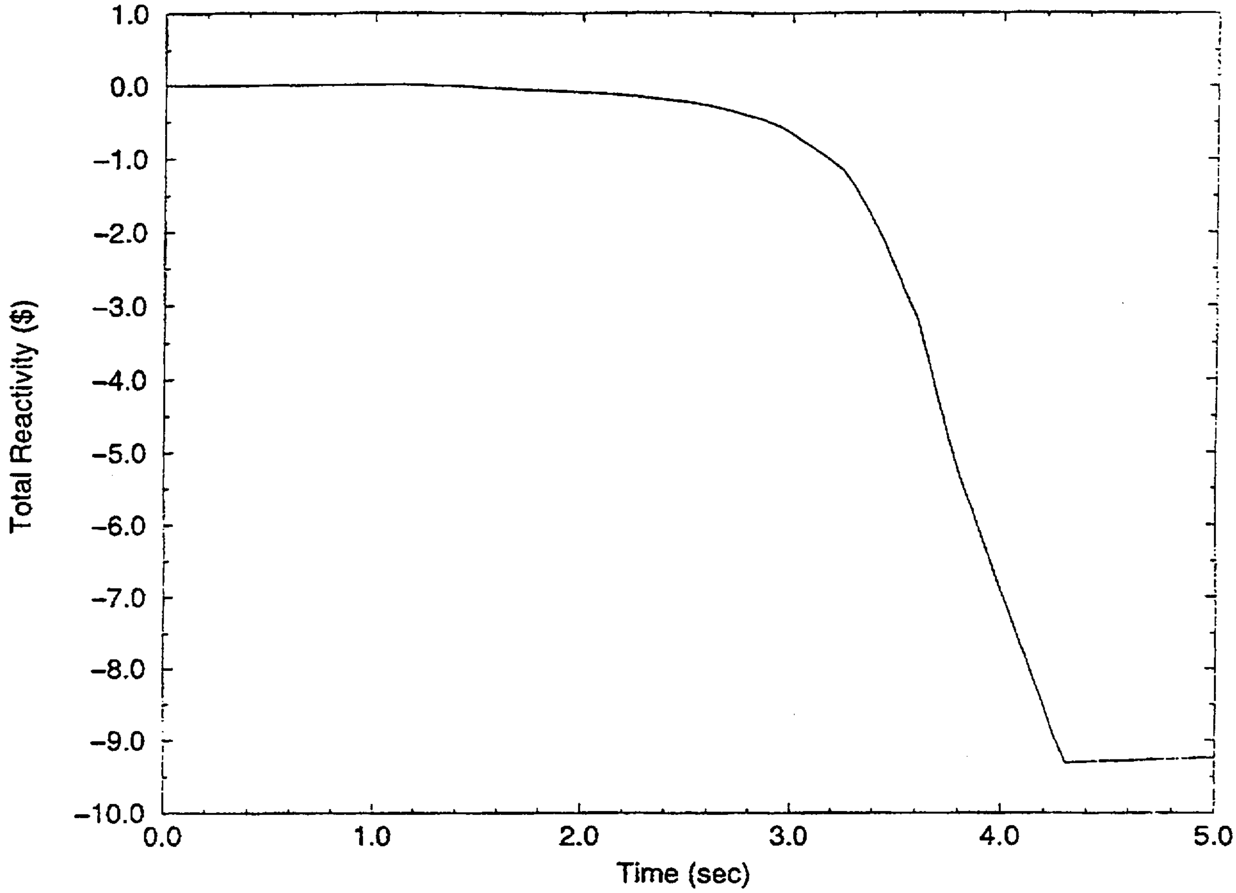
FORT CALHOUN STATION		
RCP SPEEDS FOR SEIZED ROTOR EVENT		
DWG. USAR FIGURE 14.6-6		
REV.SH. ---	APVD 6/1/01	REV Ø
FILE 48496		



FORT CALHOUN STATION		
CORE INLET AND RCS COLD LEG FLOW RATES FOR SEIZED ROTOR EVENT		
DWG. USAR FIGURE 14.6-7		
REV.SH. ---	APVD	REV
FILE 48497	DJ 6/1/01	Ø



FORT CALHOUN STATION		
VESSEL OUTLET AND CORE INLET TEMPERATURES FOR SEIZED ROTOR EVENT		
USAR DWG. FIGURE 14.6-8		
REV.SH. ---	APVD	REV
FILE 48498	Rd 6/1/01	Ø

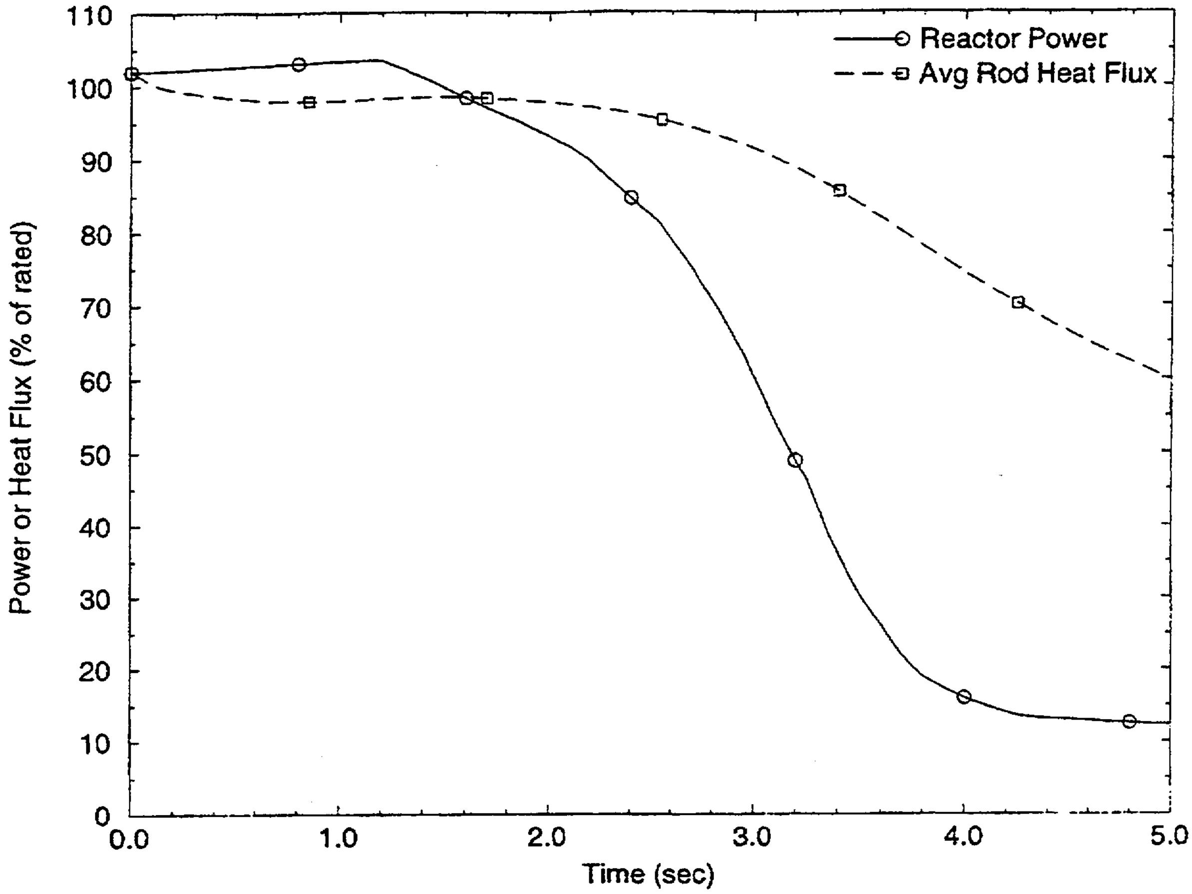


FORT CALHOUN STATION

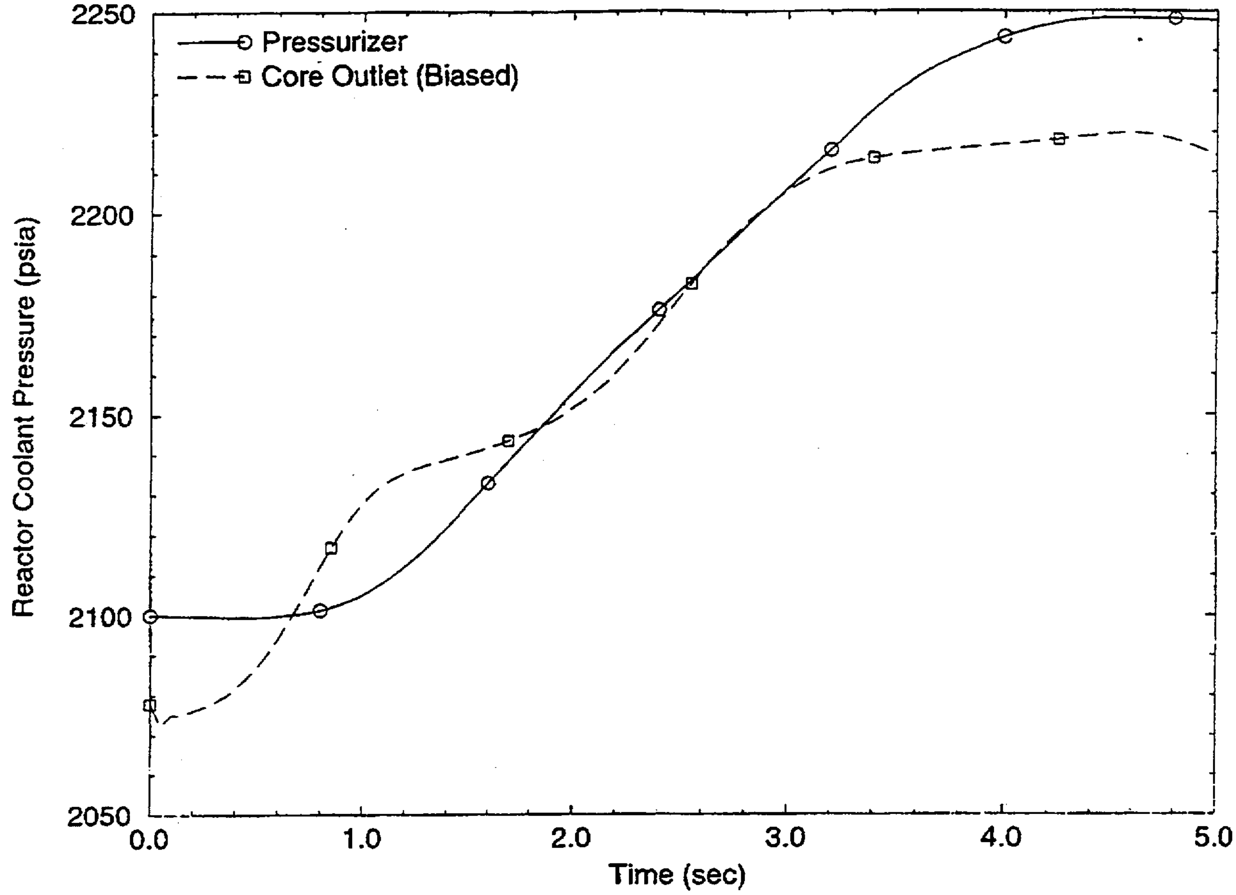
TOTAL REACTIVITY
FOR SEIZED
ROTOR EVENT

USAR
DWG. FIGURE 14.6-9

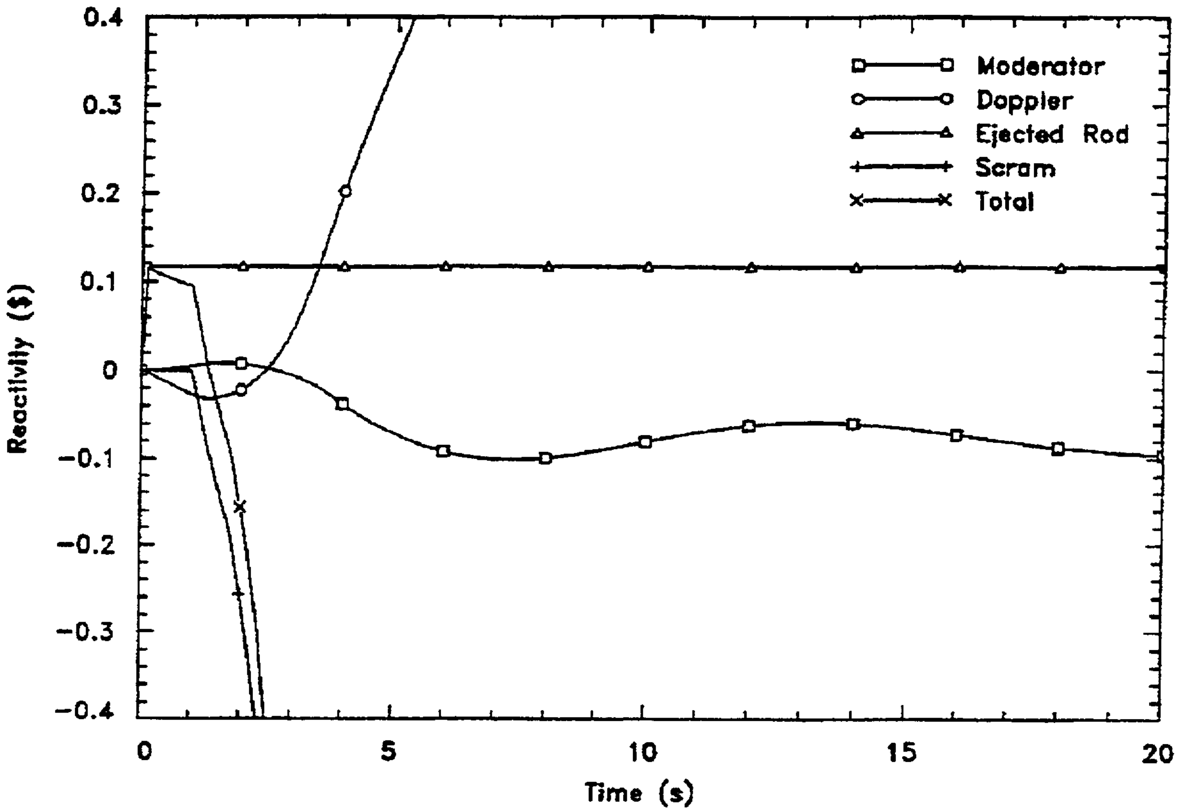
REV.SH.	---	APVD	REV
FILE	48499	6/1/01	Ø



FORT CALHOUN STATION		
REACTOR POWER AND AVERAGE ROD HEAT FLUX FOR SEIZED ROTOR EVENT		
DWG. USAR FIGURE 14.6-10		
REV.SH. ---	APVD	REV
FILE 48502	6/1/01	0



FORT CALHOUN STATION		
PRESSURIZER AND BIASED CORE OUTLET PRESSURES FOR SEIZED ROTOR EVENT		
USAR DWG. FIGURE 14.6-11		
REV.SH. ---	APVD	REV
FILE 48511	6/1/01	Ø

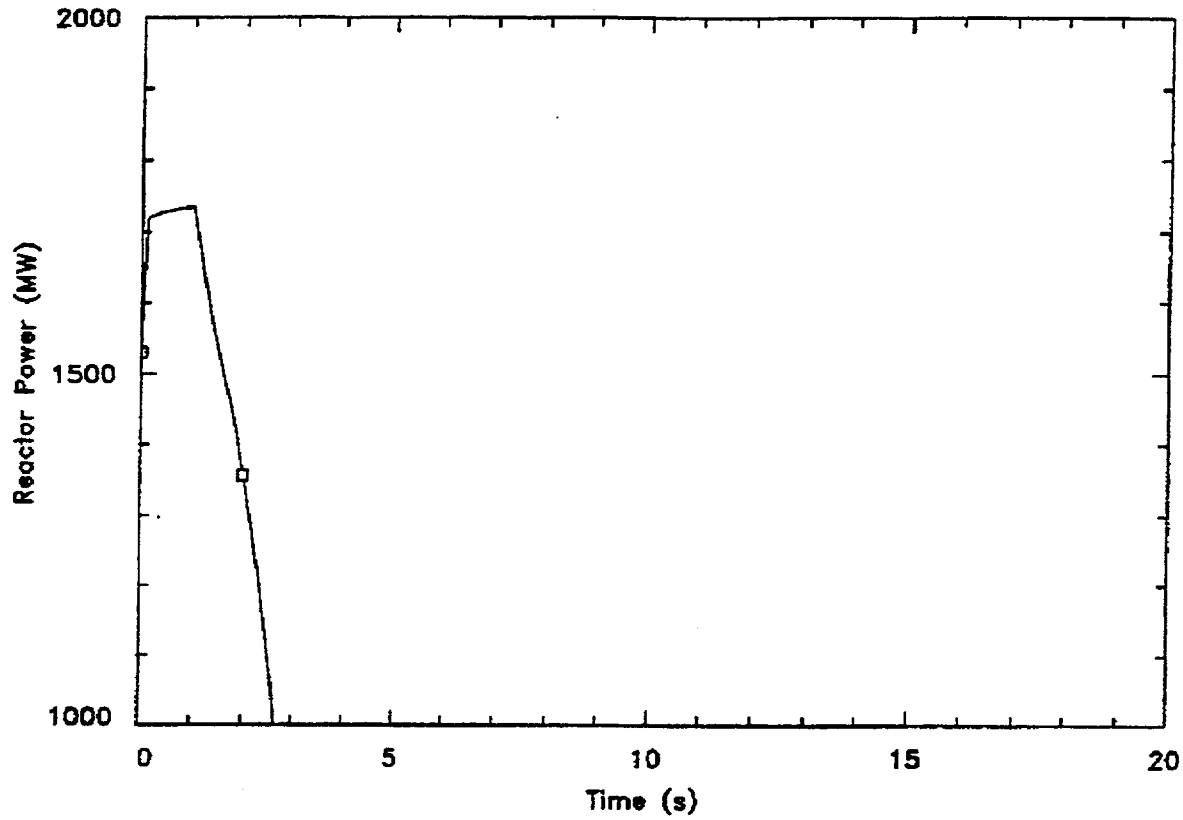


FORT CALHOUN STATION

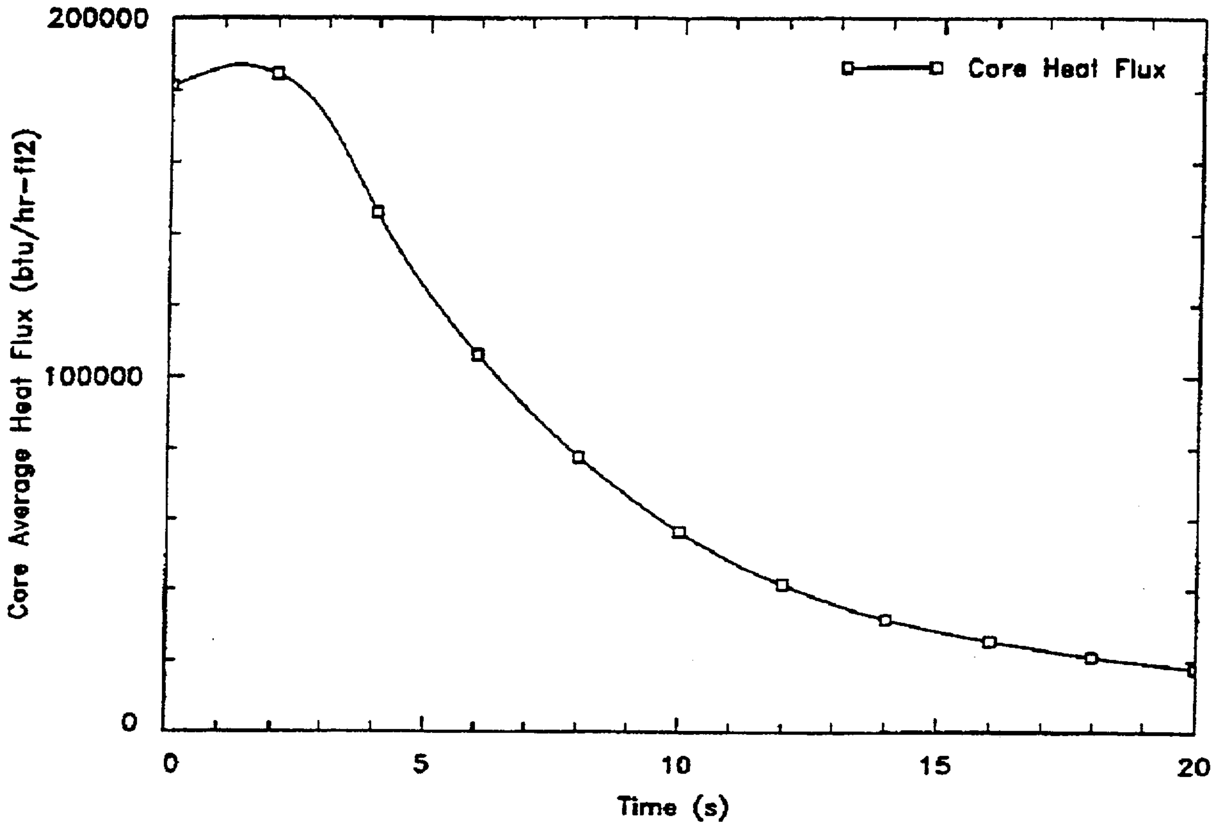
COMPONENT AND TOTAL REACTIVITY RESPONSES FOR THE HFP BOC CEA EJECTION CASE

DWG. USAR FIGURE 14.13-1

REV.SH. ---	APVD	REV
FILE 48513	DJ 6/1/01	Ø



FORT CALHOUN STATION			
CORE POWER RESPONSE FOR THE HFP BOC CEA EJECTION CASE			
DWG. USAR FIGURE 14.13-2			
REV.SH.	---	APVD	REV
FILE	48514	6/1/01	Ø

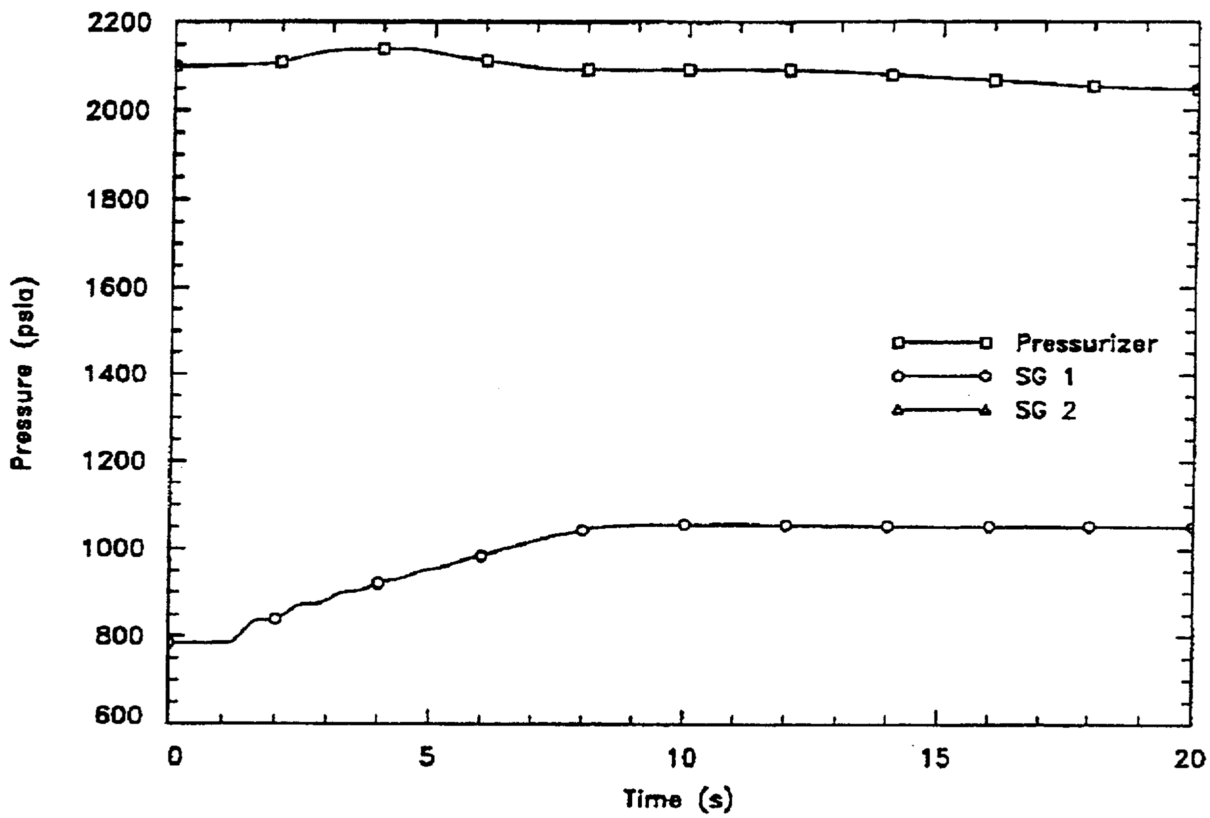


FORT CALHOUN STATION

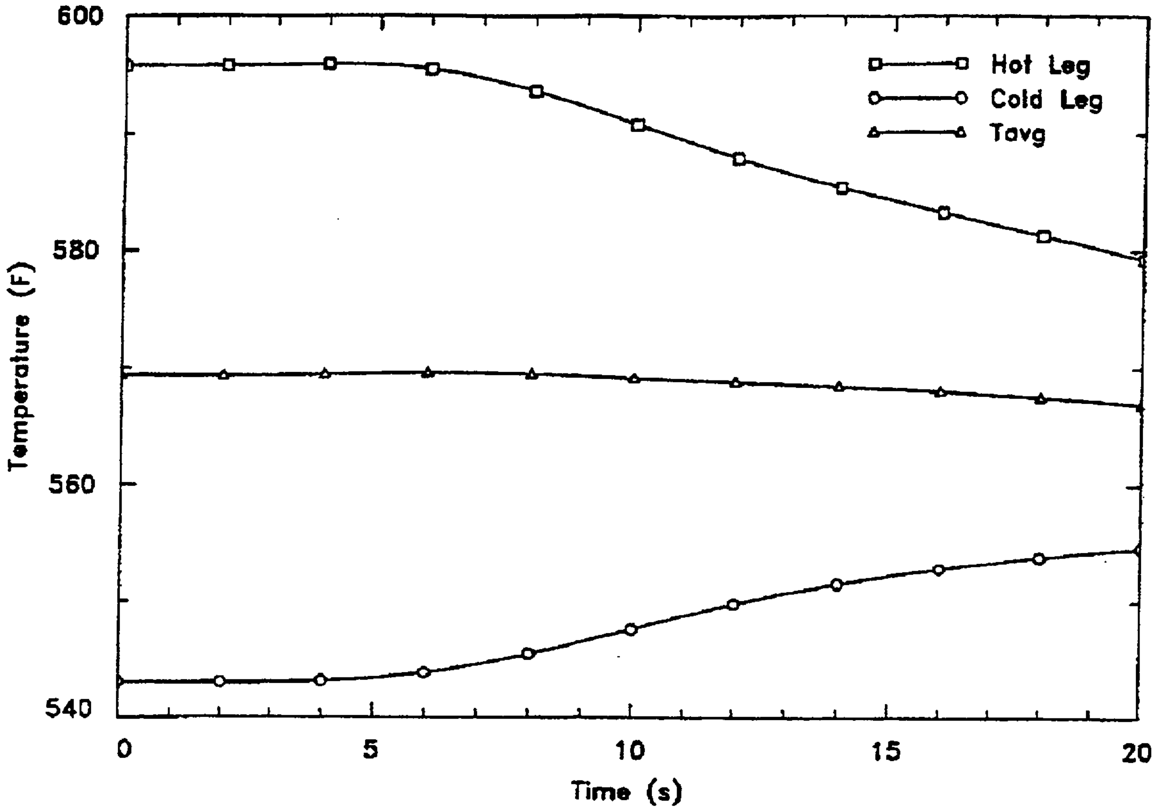
CORE AVERAGE HEAT FLUX RESPONSE FOR THE HFP BOC CEA EJECTION CASE

DWG. USAR FIGURE 14.13-3

REV.SH. ---	APVD	REV
FILE 48521	RJ 6/1/01	Ø



FORT CALHOUN STATION		
SYSTEM PRESSURE RESPONSES FOR THE HFP BOC CEA EJECTION CASE		
DWG. USAR FIGURE 14.13-4		
REV.SH. ---	APVD	REV
FILE 48522	6/1/01	Ø



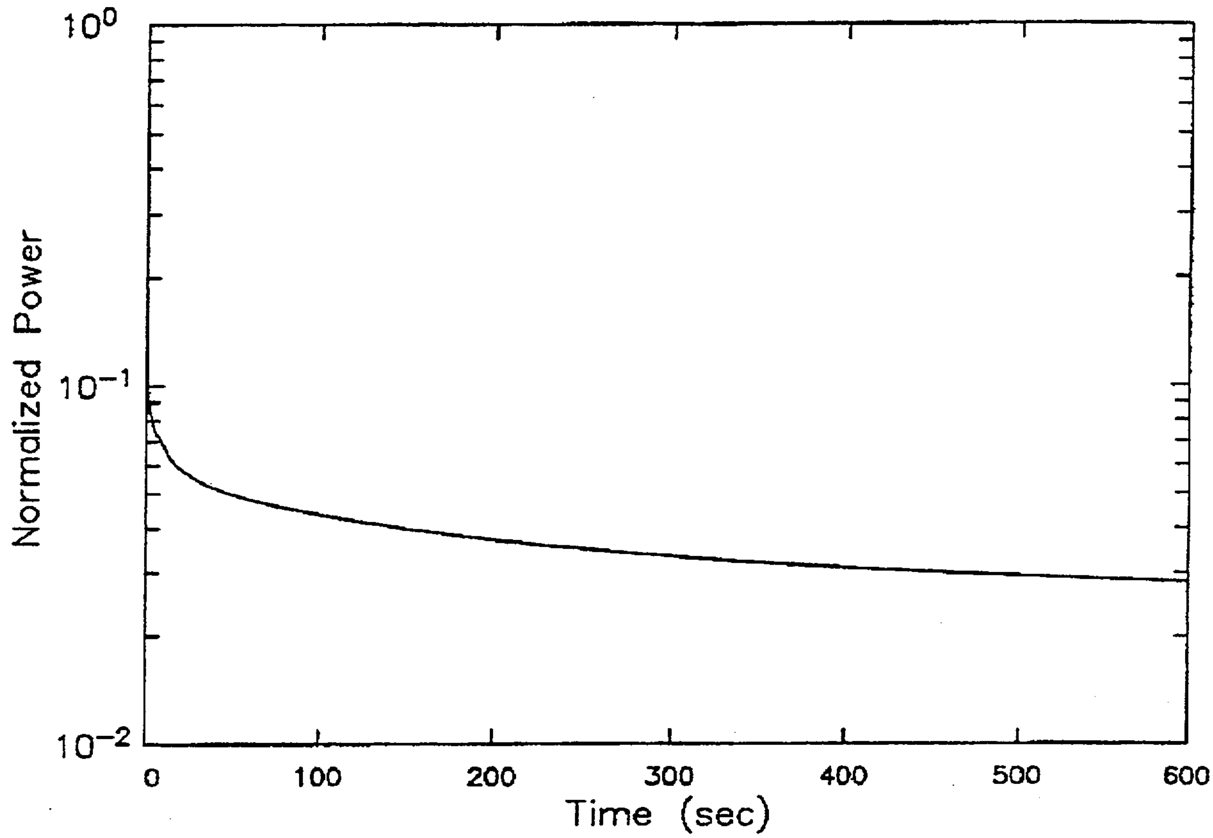
FORT CALHOUN STATION

SYSTEM FLUID TEMPERATURE RESPONSES FOR THE HFP BOC CEA EJECTION CASE

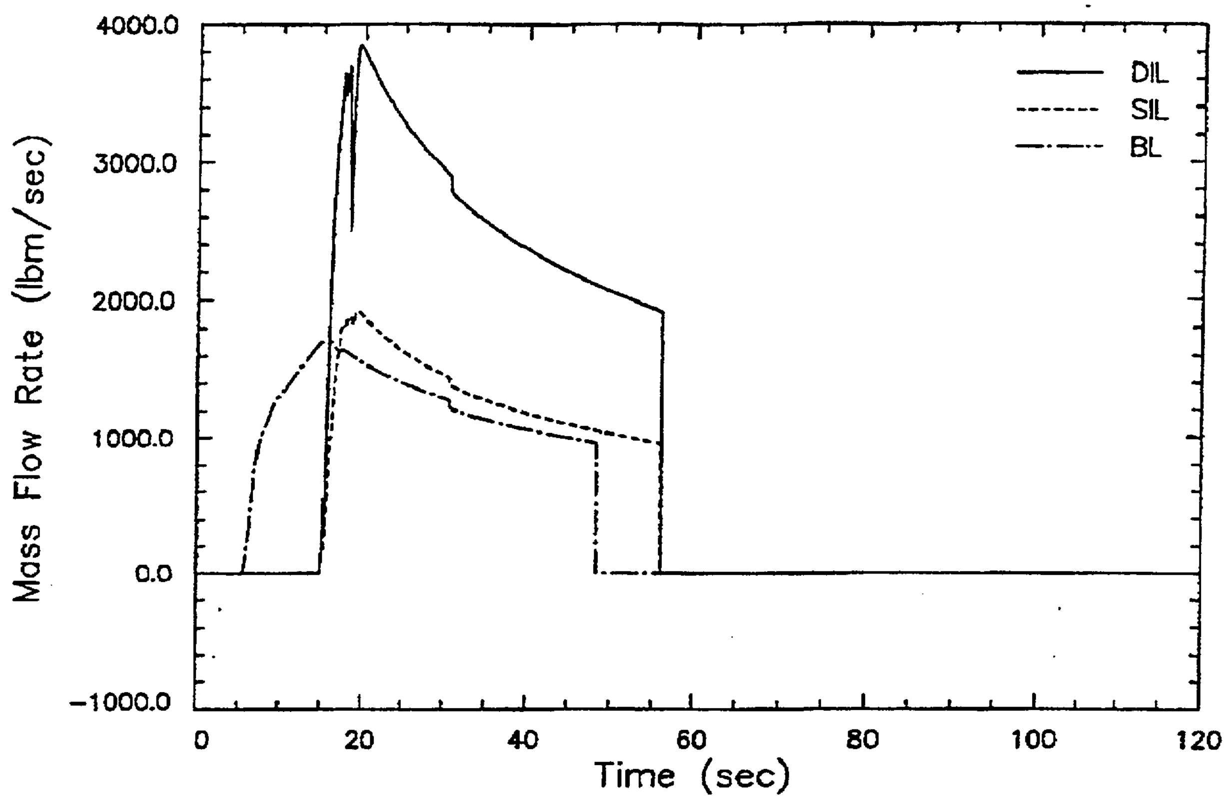
USAR

DWG. FIGURE 14.13-5

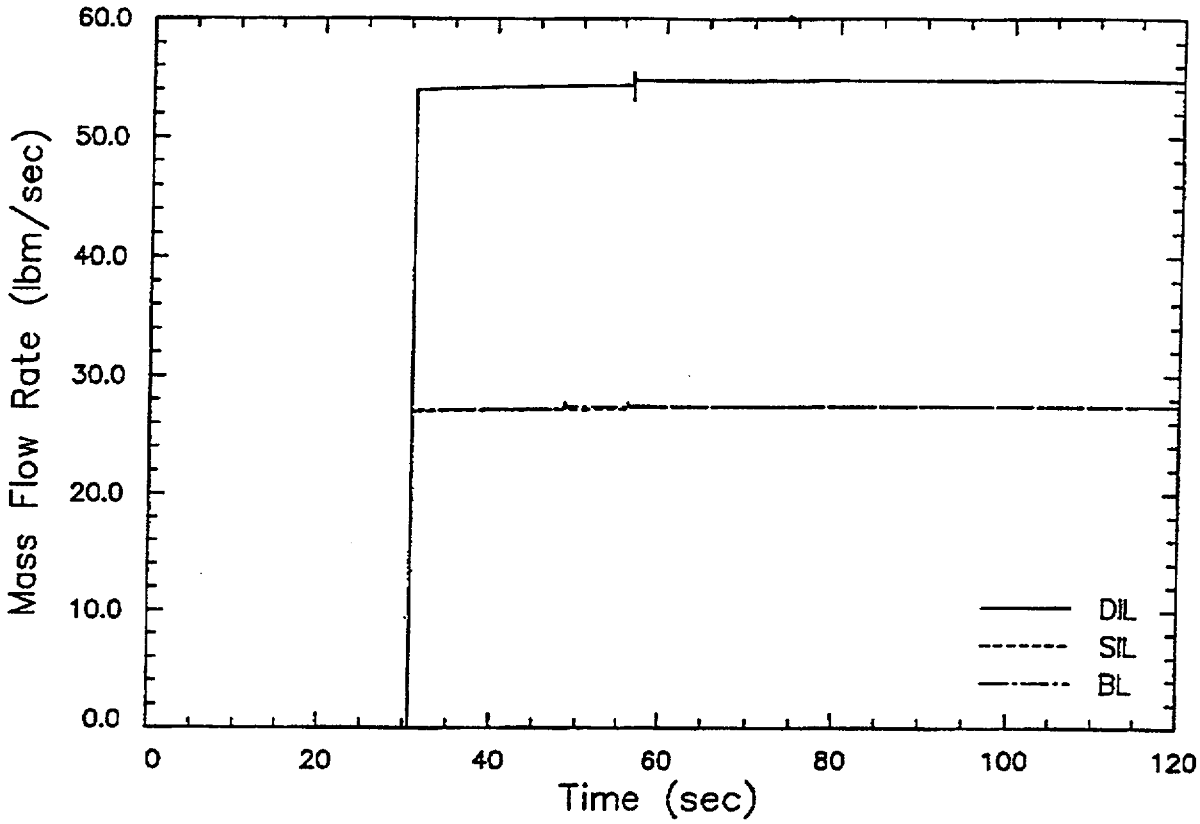
REV.SH. ---	APVD	REV
FILE 48523		Ø



FORT CALHOUN STATION		
NORMALIZED POWER: 1.Ø DECLG EOC NOLPSI		
USAR DWG. FIGURE 14.15-41		
REV.SH. ---	APVD	REV
FILE 48524	6/1/01	Ø



FORT CALHOUN STATION		
SIT DISCHARGE RATES: 1.Ø DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-42		
REV.SH. ---	APVD 6/1/01	REV Ø
FILE 48525		

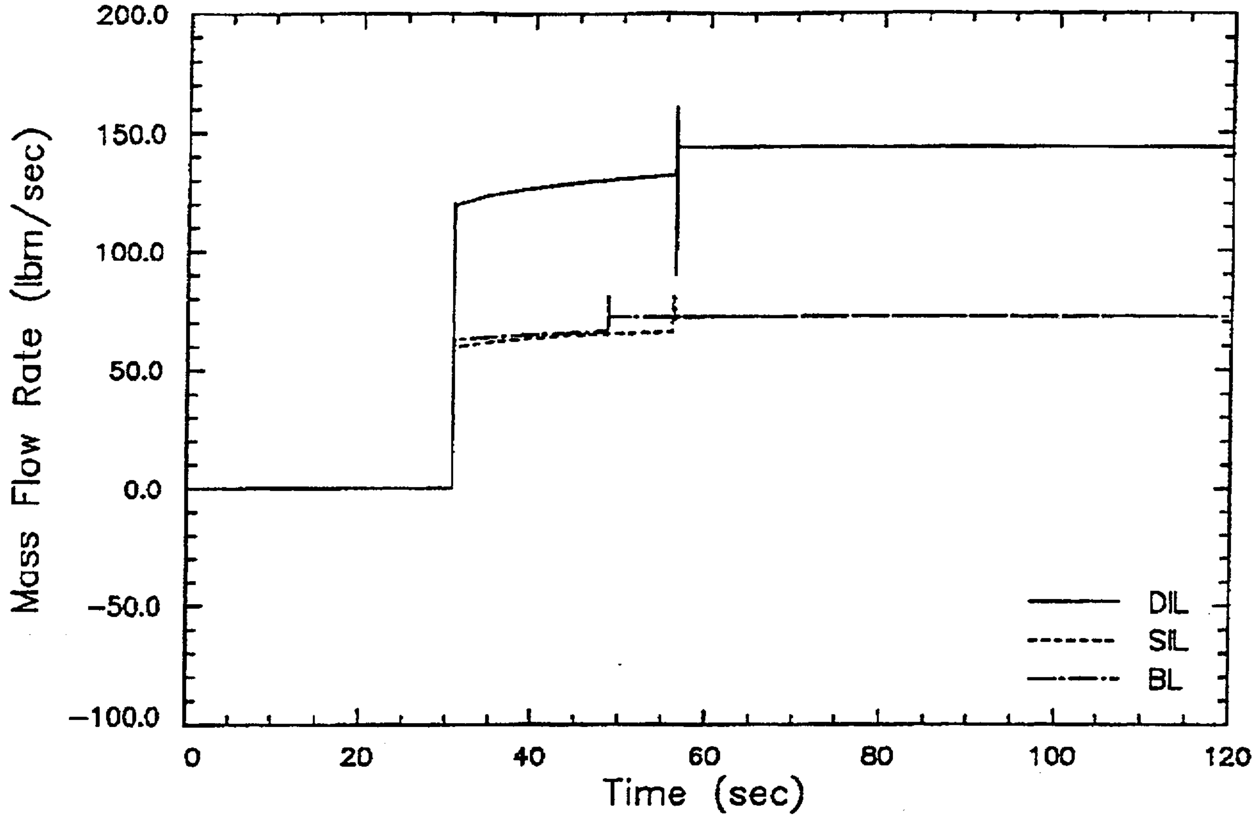


FORT CALHOUN STATION

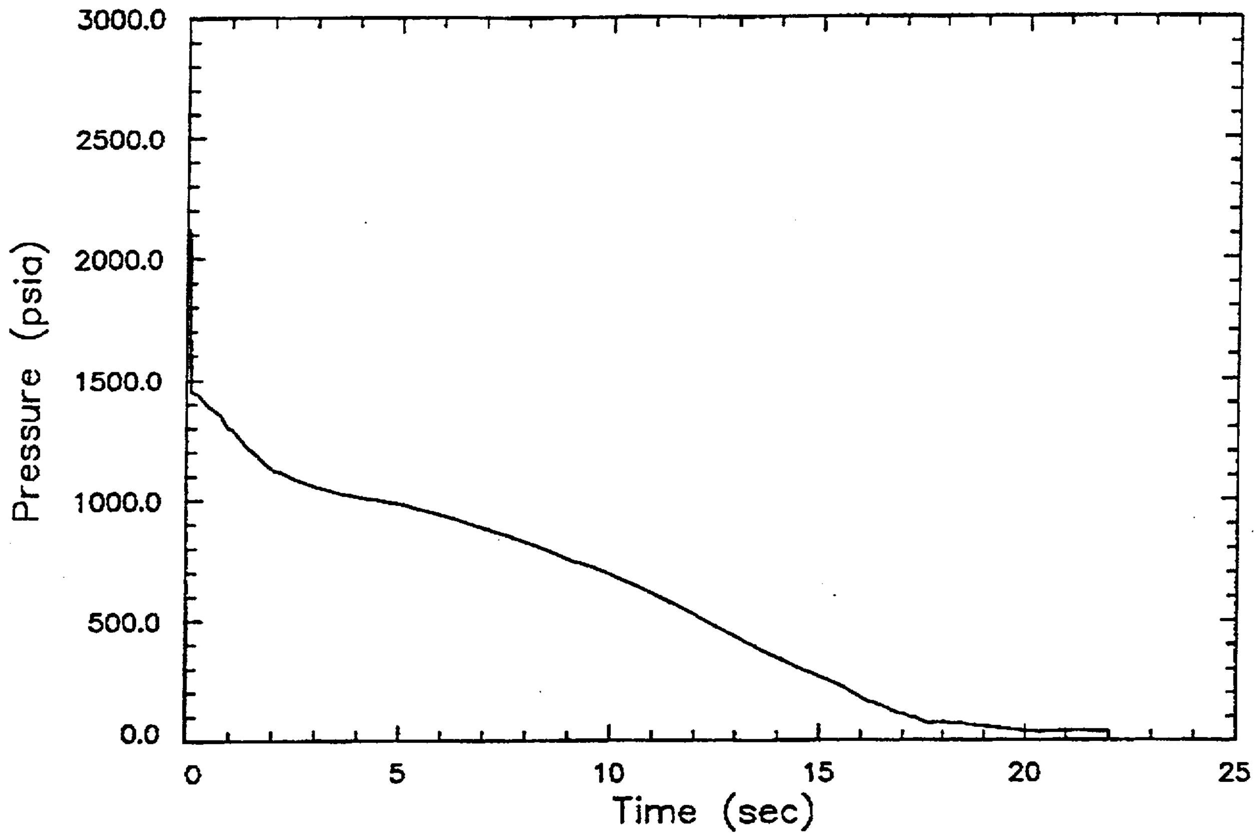
HPSI FLOW RATES:
1.Ø DECLG
EOC NOLPSI

USAR
DWG. FIGURE 14.13-43

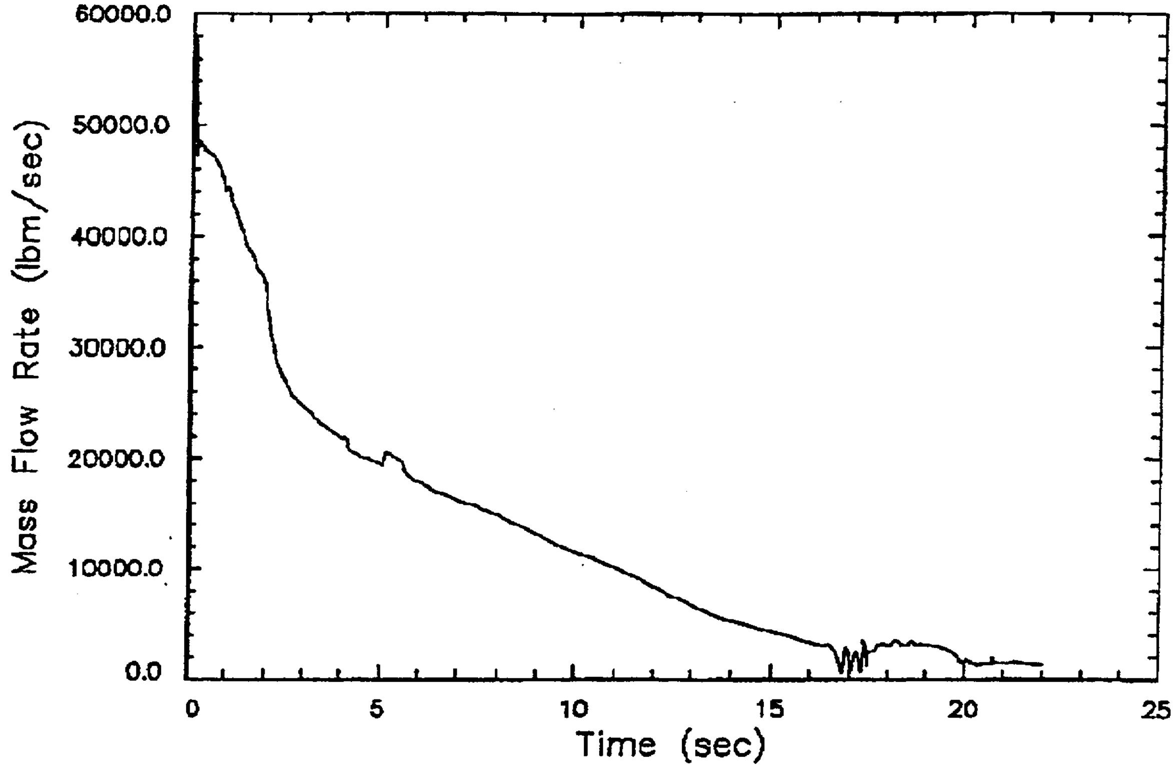
REV.SH.	---	APVD 6/1/01	REV Ø
FILE	48535		



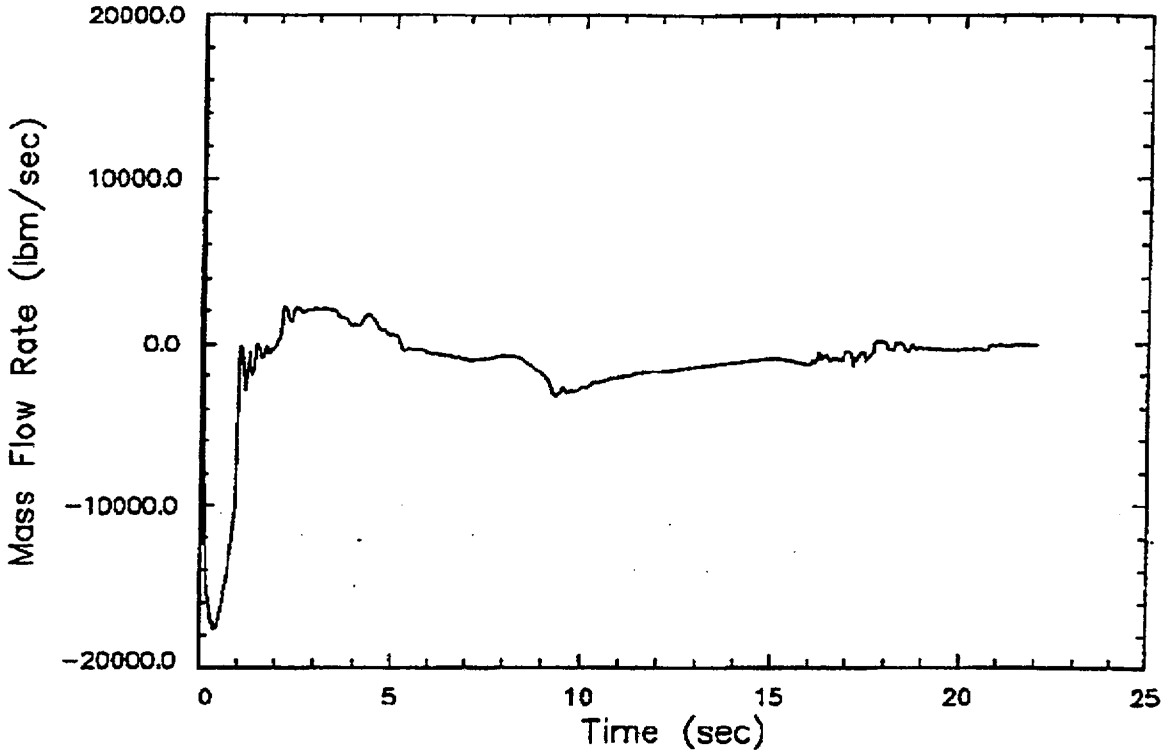
FORT CALHOUN STATION		
LPSI FLOW RATES: 1.Ø DECLG EOC NOLPSI		
USAR DWG. FIGURE 14.15-44		
REV.SH. ---	APVD	REV
FILE 48536	6/1/01	Ø



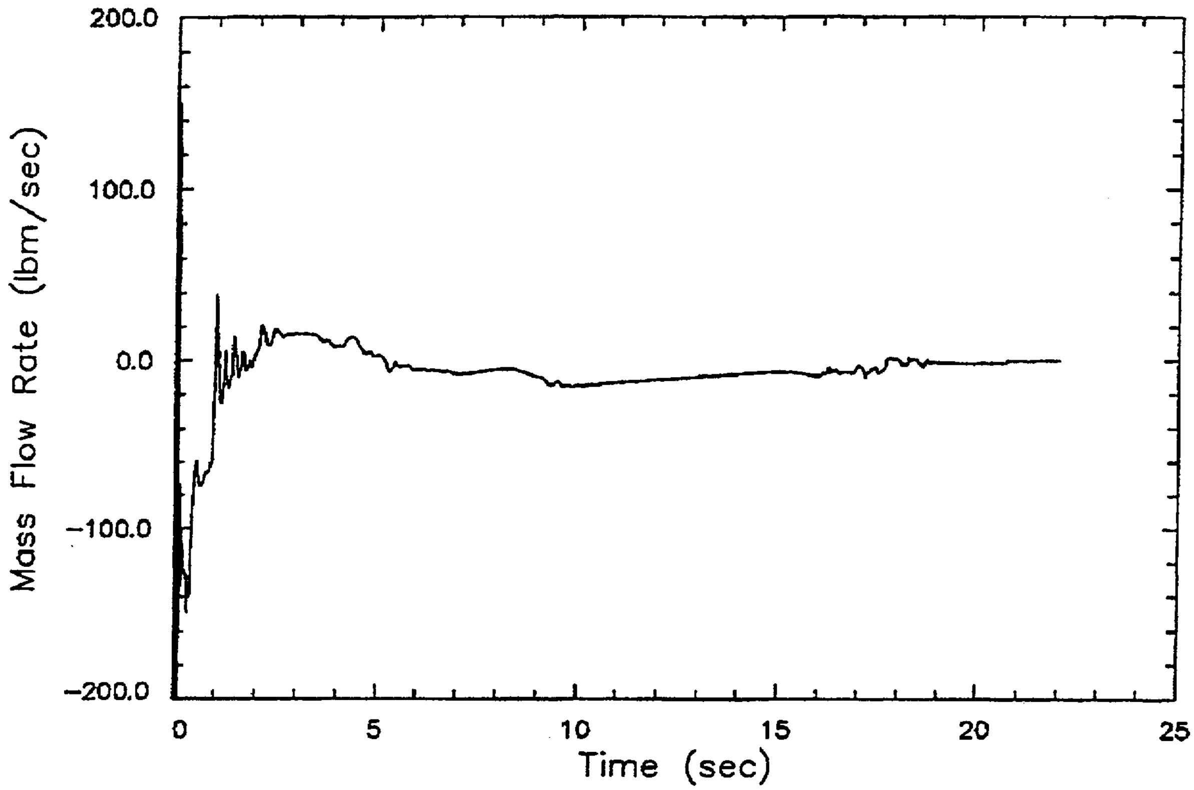
FORT CALHOUN STATION		
UPPER PLENUM PRESSURE: 1.Ø DECLG EOC NOLPSI		
USAR DWG. FIGURE 14.15-45		
REV.SH. ---	APVD	REV
FILE 48537	6/1/01	Ø



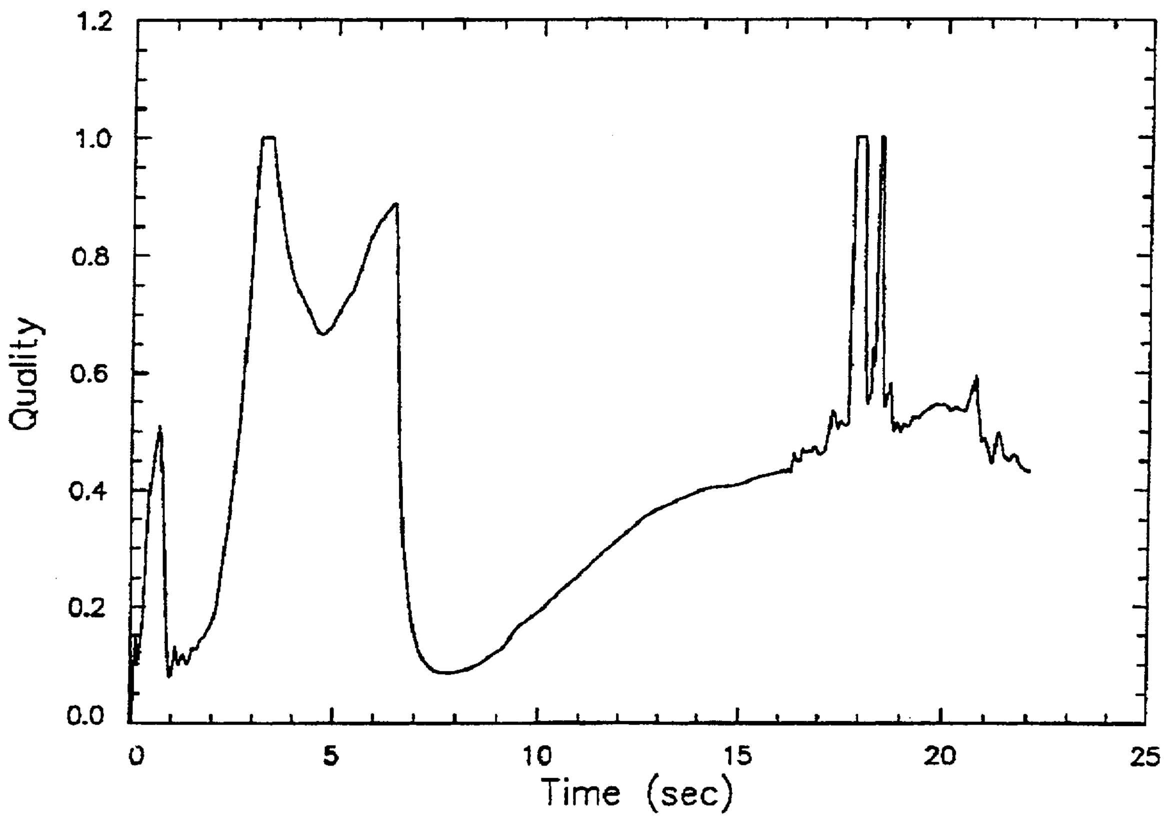
FORT CALHOUN STATION	
TOTAL BREAK FLOW RATE: 1.Ø DECLG EOC NOLPSI	
USAR DWG. FIGURE 14.15-46	
REV.SH. ---	APVD	REV Ø
FILE 48753		



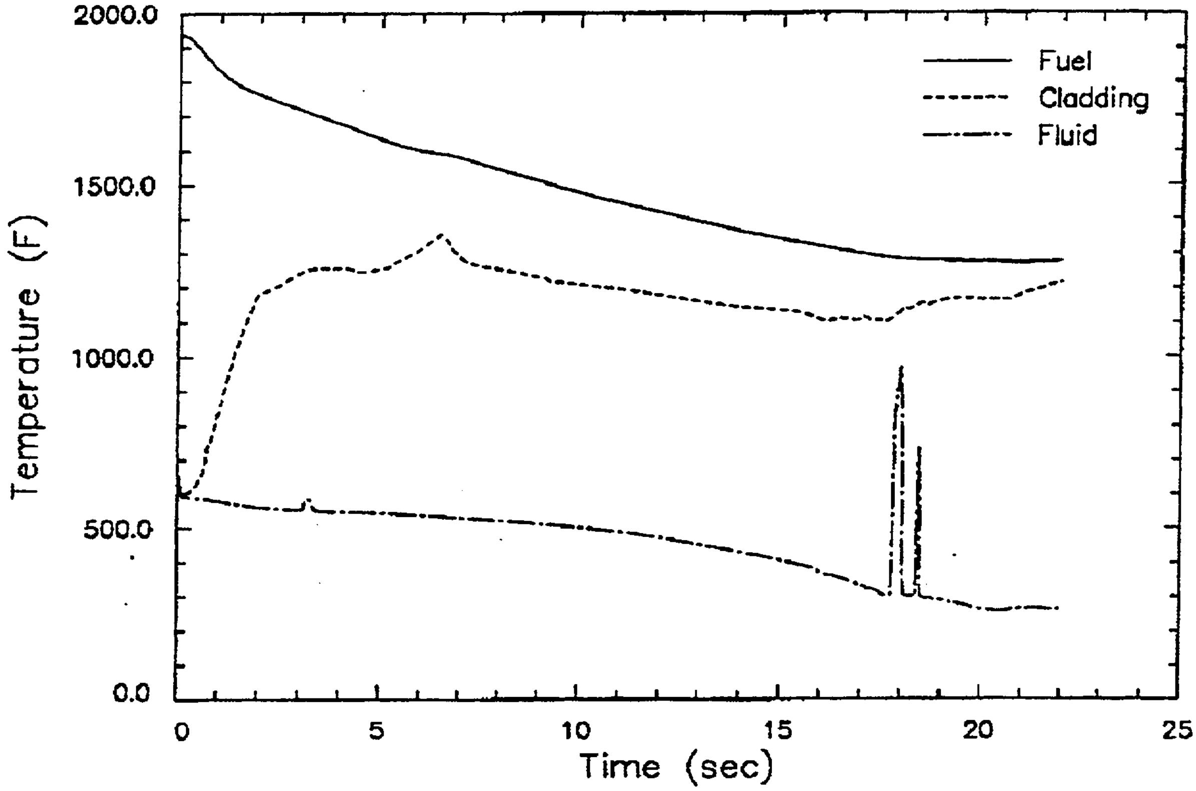
FORT CALHOUN STATION		
AVERAGE CORE INLET FLOW RATE: 1.Ø DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-47		
REV.SH. ---	APVD	REV
FILE 48754	6/1/01	Ø



FORT CALHOUN STATION		
HOT ASSEMBLY INLET FLOW RATE: 1.0 DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-48		
REV.SH. ---	APVD	REV
FILE 48755		Ø



FORT CALHOUN STATION		
PCT - NODE FLUID QUALITY: 1.Ø DECLG EOC NOLPSI		
USAR DWG. FIGURE 14.15-49		
REV.SH. ---	APVD	REV
FILE 48756	RA 6/1/01	Ø



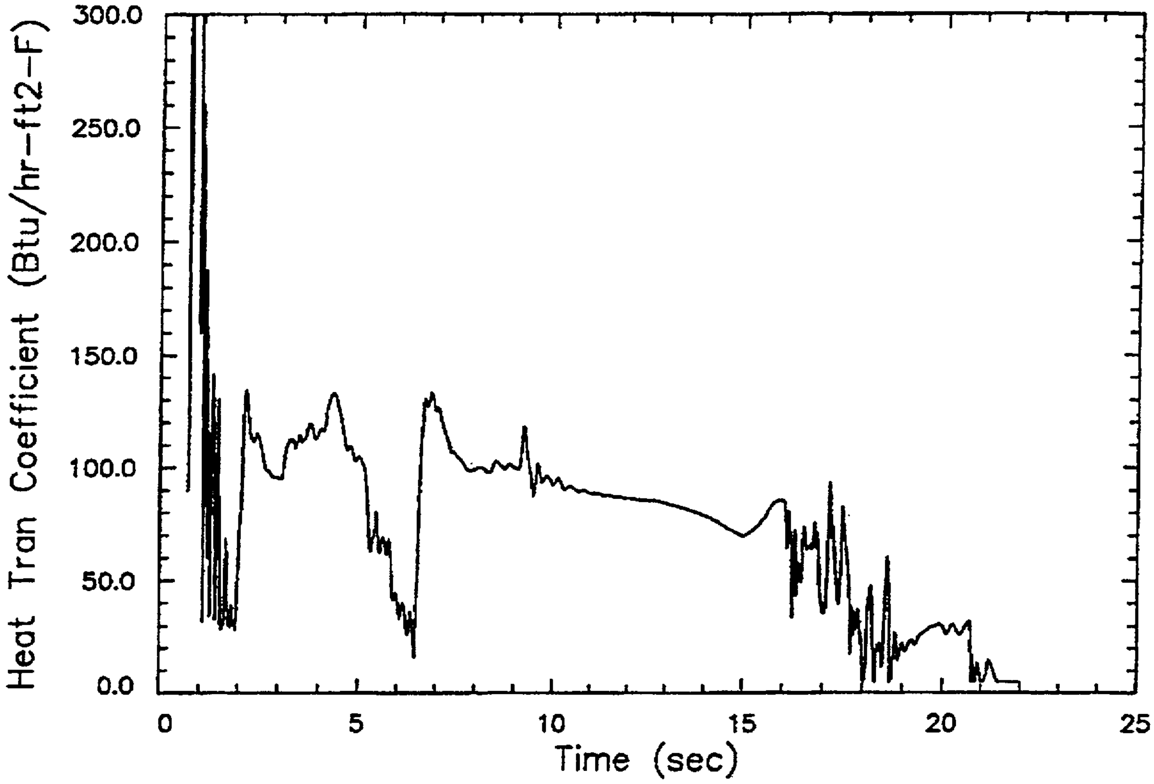
FORT CALHOUN STATION

PCT - NODE AVERAGE FUEL, CLADDING SURFACE, AND FLUID TEMPERATURES: 1.Ø DECLG EOC NOLPSI

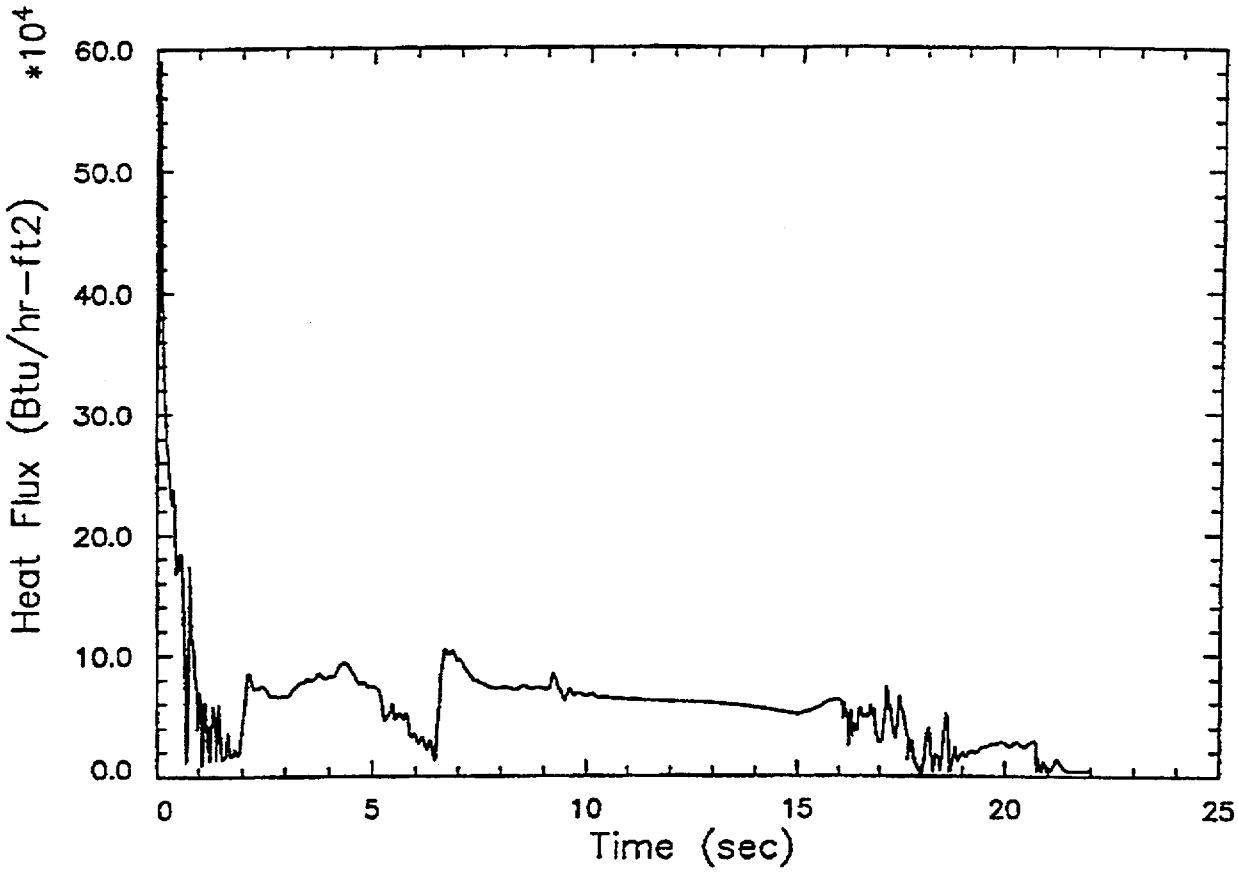
USAR

DWG. FIGURE 14.15-5Ø

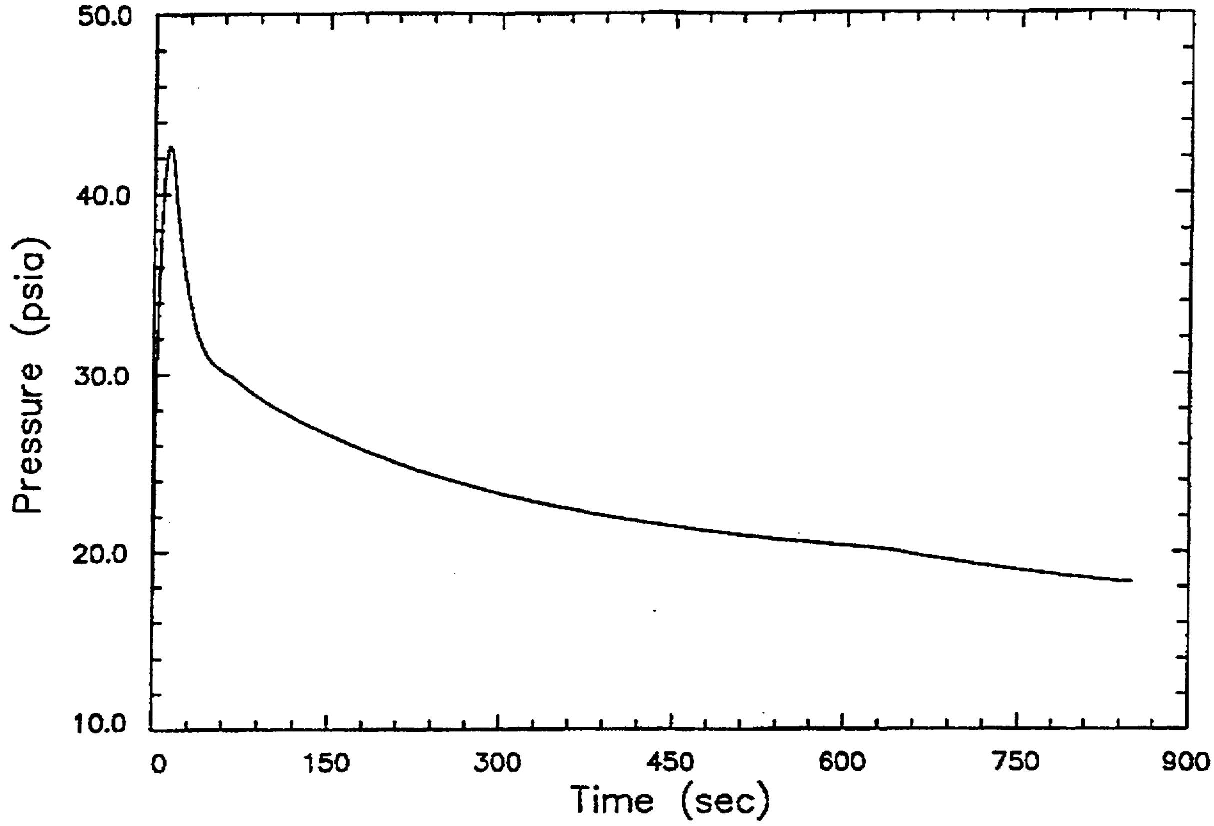
REV.SH. ---	APVD	REV
FILE 48757	6/1/01	Ø



FORT CALHOUN STATION		
PCT - NODE HEAT TRANSFER COEFFICIENT: 1.0 DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-51		
REV.SH. ---	APVD	REV
FILE 48758	DL 6/1/01	0



FORT CALHOUN STATION		
PCT - NODE HEAT FLUX: 1.0 DECLG EOC NOLPSI		
USAR DWG. FIGURE 14.15-52		
REV.SH. ---	APVD	REV
FILE 48759	6/1/01	0

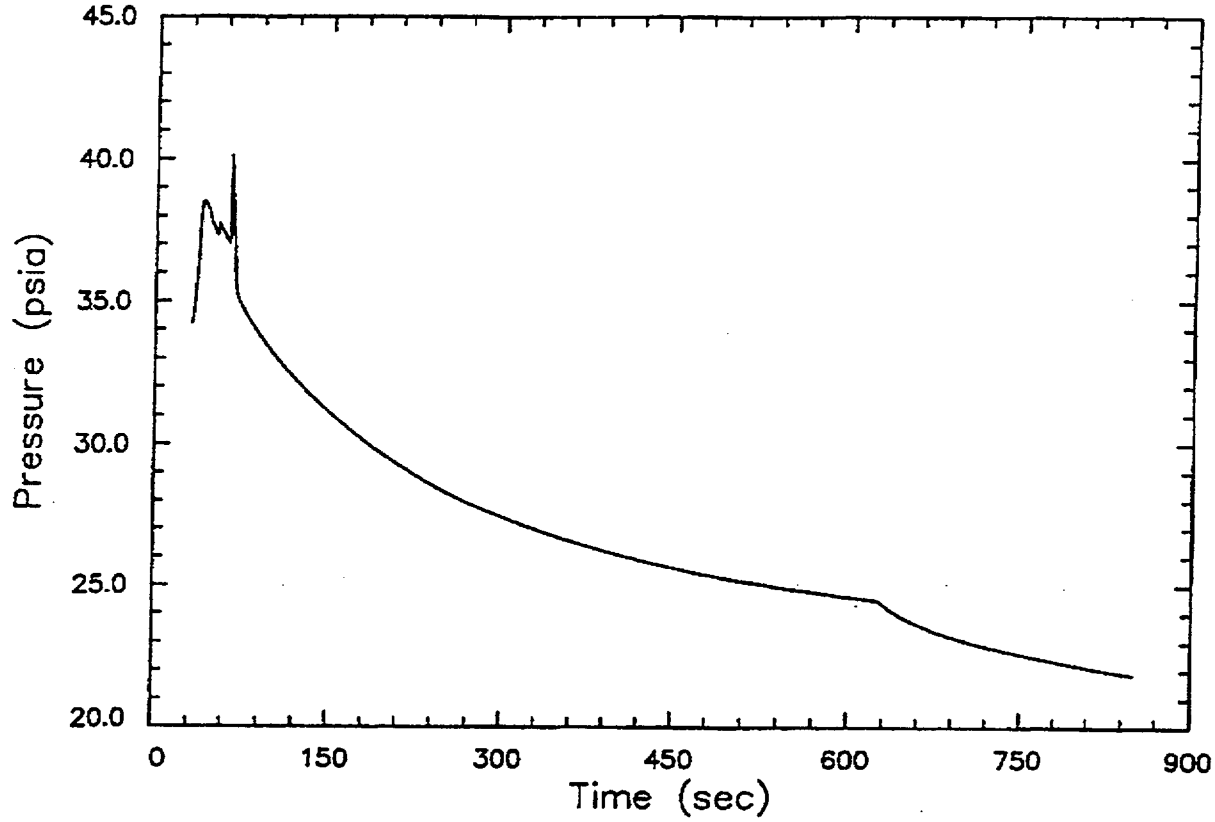


FORT CALHOUN STATION

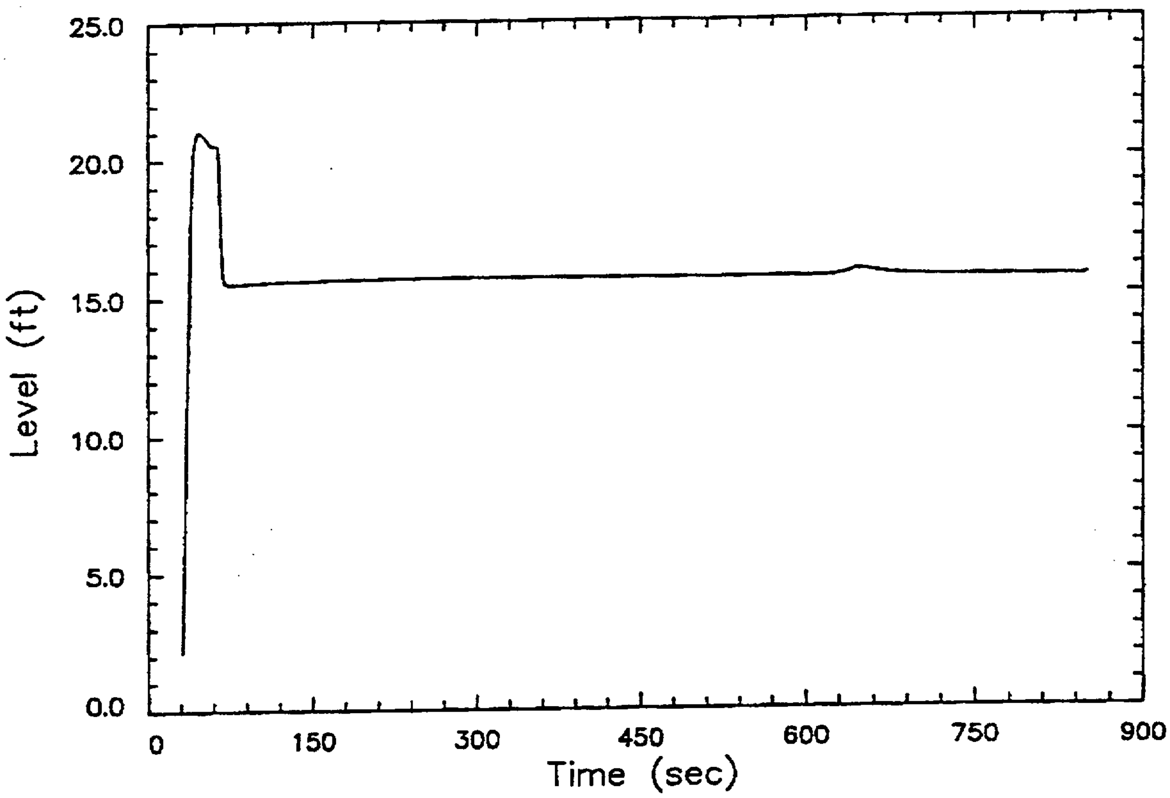
CONTAINMENT PRESSURE:
1.0 DECLG
EOC NOLPSI

USAR
DWG. FIGURE 14.15-53

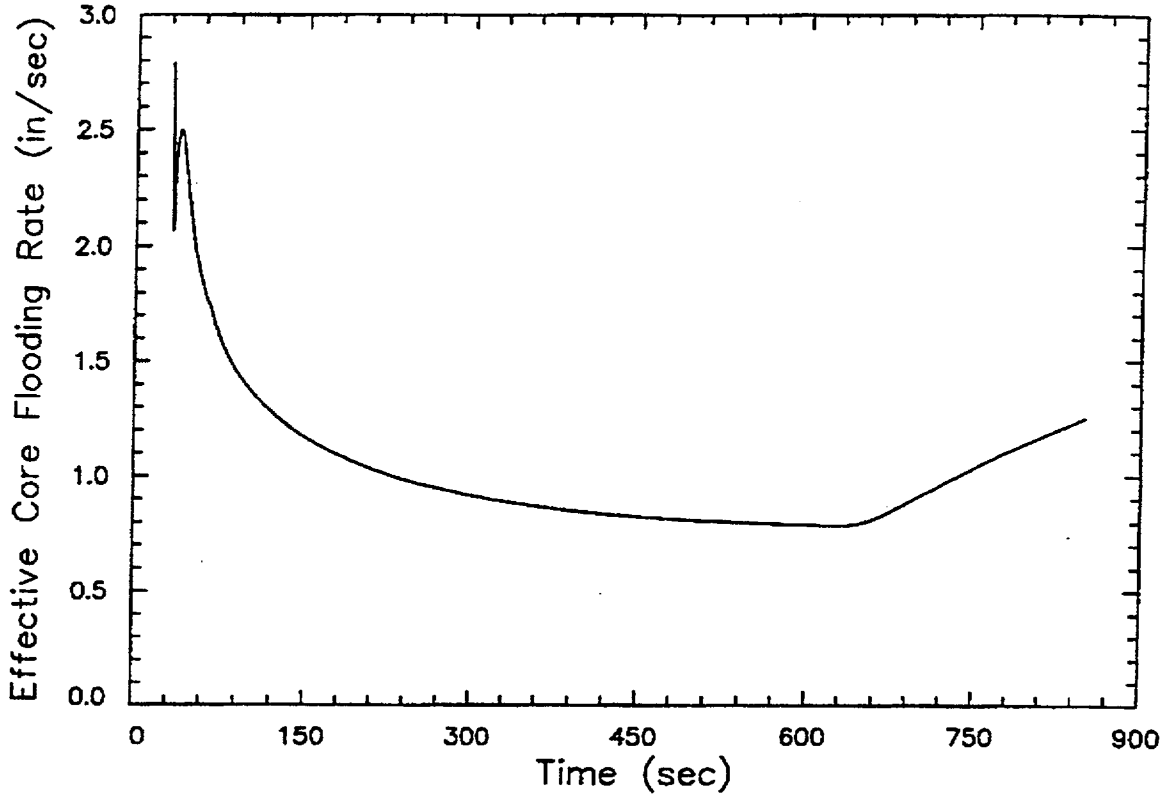
REV.SH.	---	APVD	REV
FILE	48760	6/1/01	0



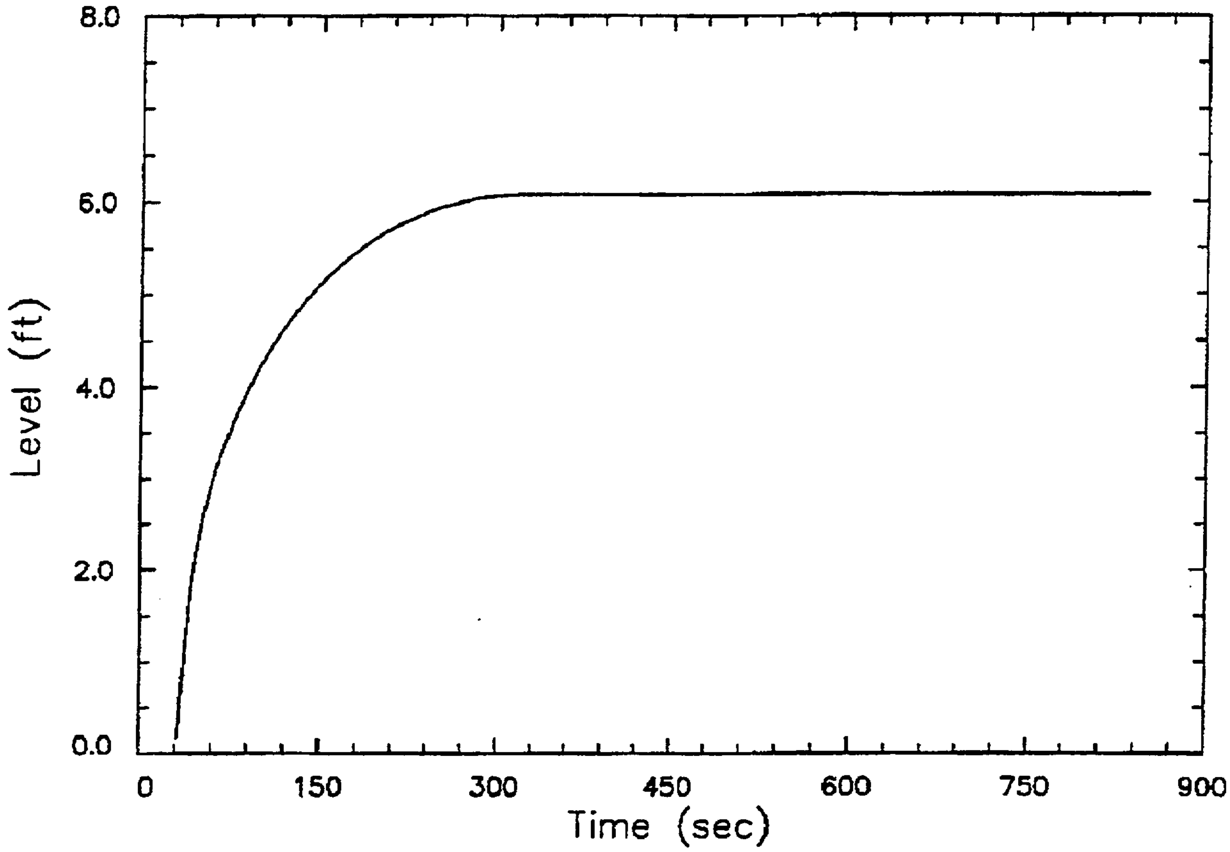
FORT CALHOUN STATION		
UPPER PLENUM PRESSURE AFTER BLOWDOWN: 1.Ø DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-54		
REV.SH. ---	APVD	REV
FILE 48761	6/1/01	Ø



FORT CALHOUN STATION			
DOWNCOMER MIXTURE LEVEL: 1.Ø DECLG EOC NOLPSI			
DWG. USAR FIGURE 14.15-55			
REV.SH. ---		APVD	REV
FILE 48763		PJ 6/1/01	Ø



FORT CALHOUN STATION		
CORE EFFECTIVE FLOODING RATE: 1.Ø DECLG EOC NOLPSI		
DWG. USAR FIGURE 14.15-56		
REV.SH. ---	APVD	REV
FILE 48765	6/1/04	Ø

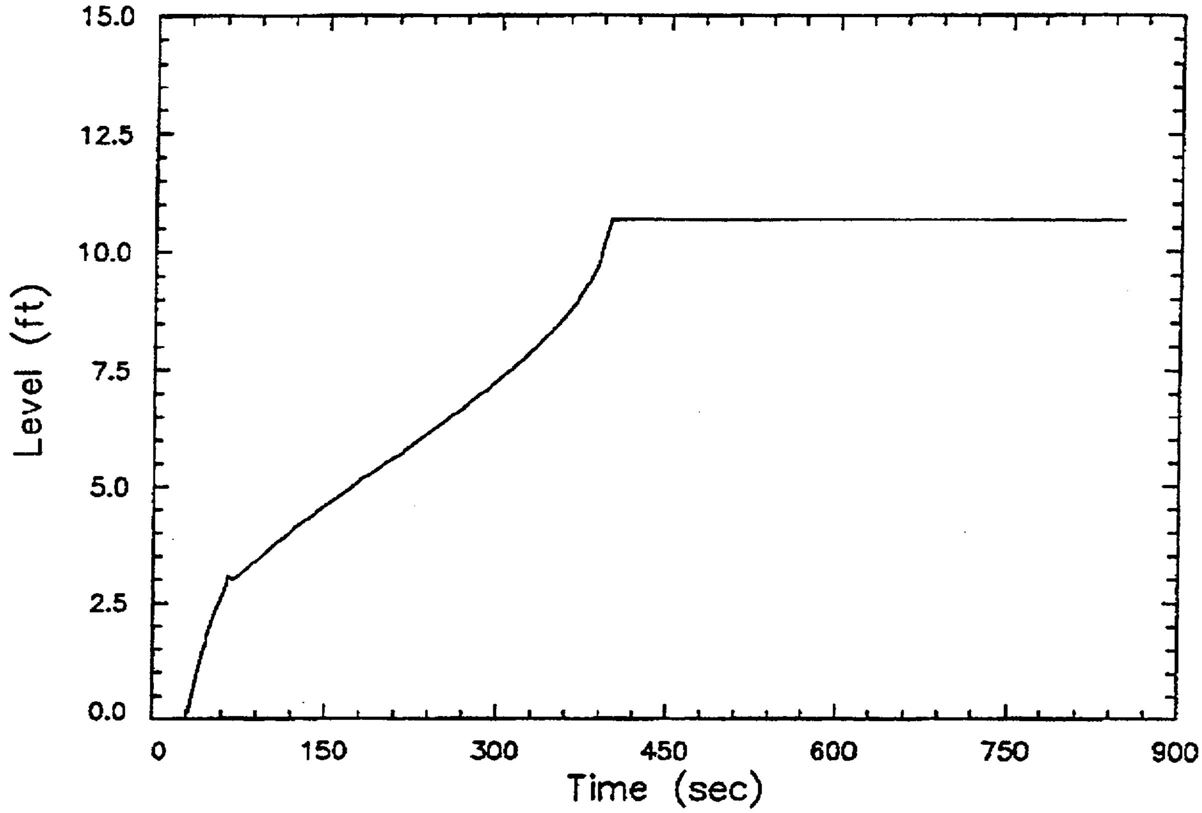


FORT CALHOUN STATION

CORE COLLAPSED
LIQIUD LEVEL:
1.Ø DECLG
EOC NOLPSI

USAR
DWG. FIGURE 14.15-57

REV.SH.	---	APVD	REV
FILE	48766	6/1/01	Ø

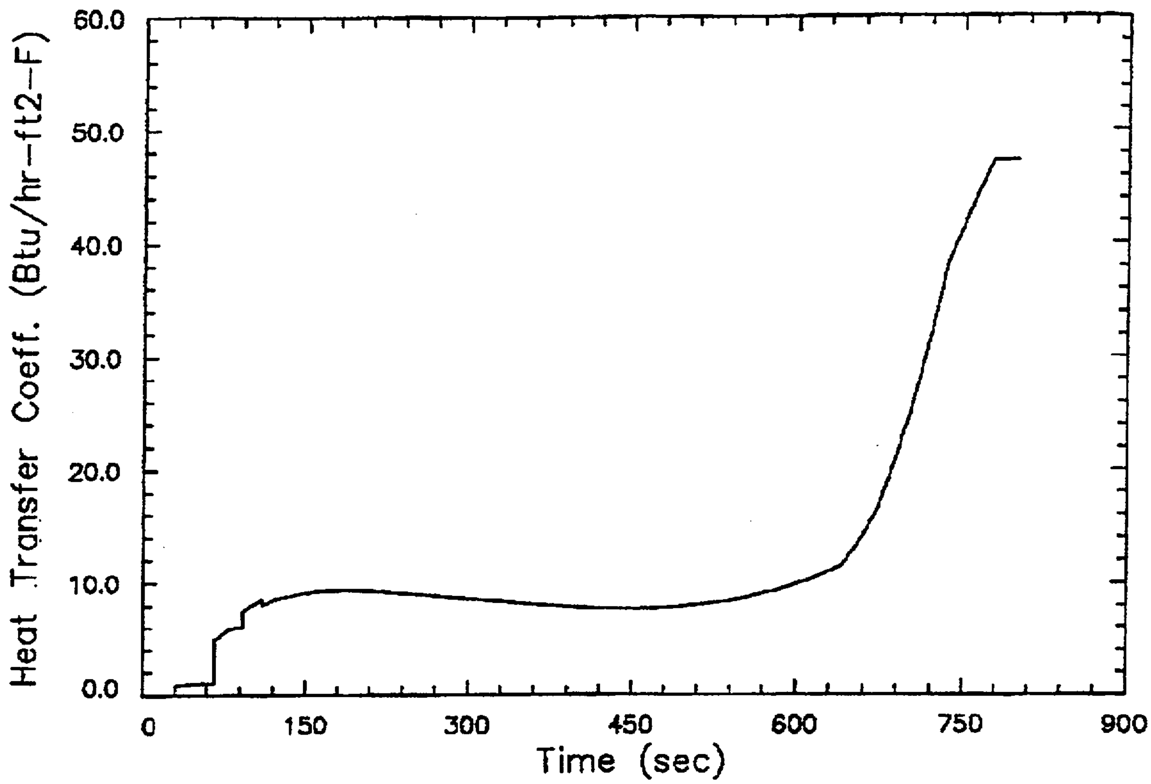


FORT CALHOUN STATION

CORE QUENCH LEVEL:
1.Ø DECLG
EOC NOLPSI

USAR
DWG. FIGURE 14.15-58

REV.SH.	---	APVD	REV
FILE	48768	RS 6/1/01	Ø



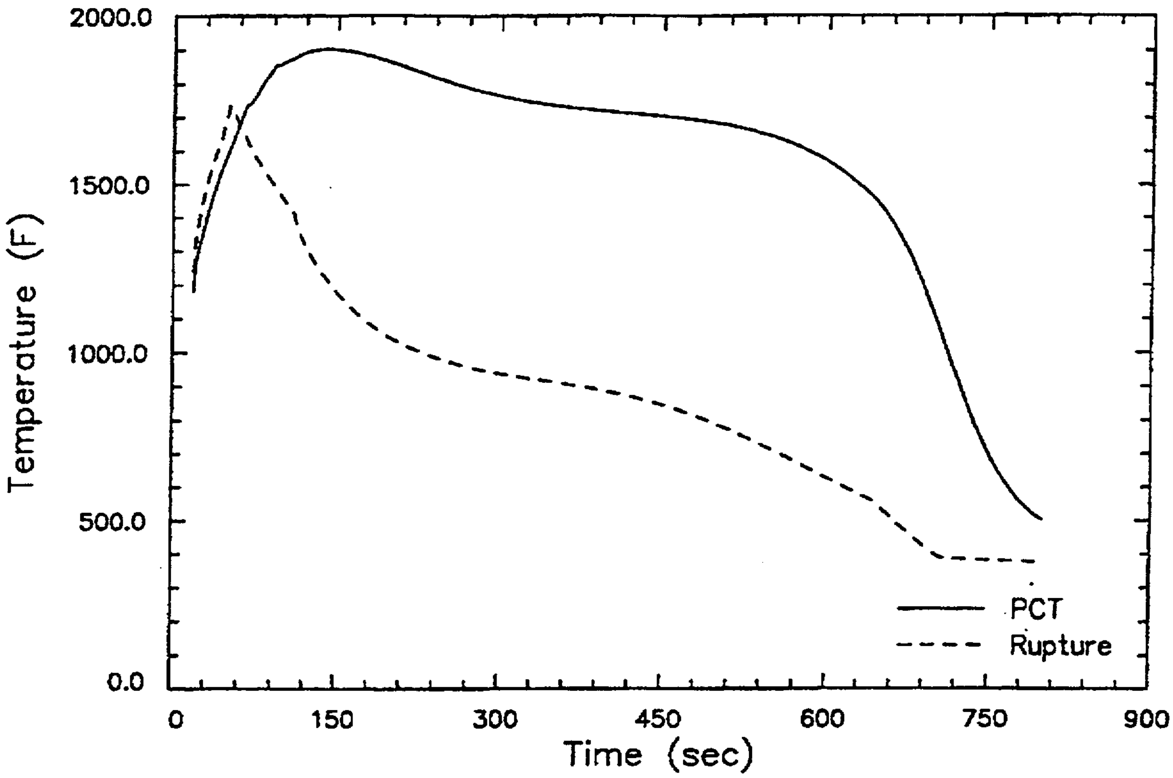
FORT CALHOUN STATION

PCT - NODE HEAT TRANSFER COEFFICIENT: 1.0 DECLG EOC NOLPSI

USAR

DWG. FIGURE 14.15-59

REV.SH.	---	APVD	REV
FILE	48771	6/1/01	Ø

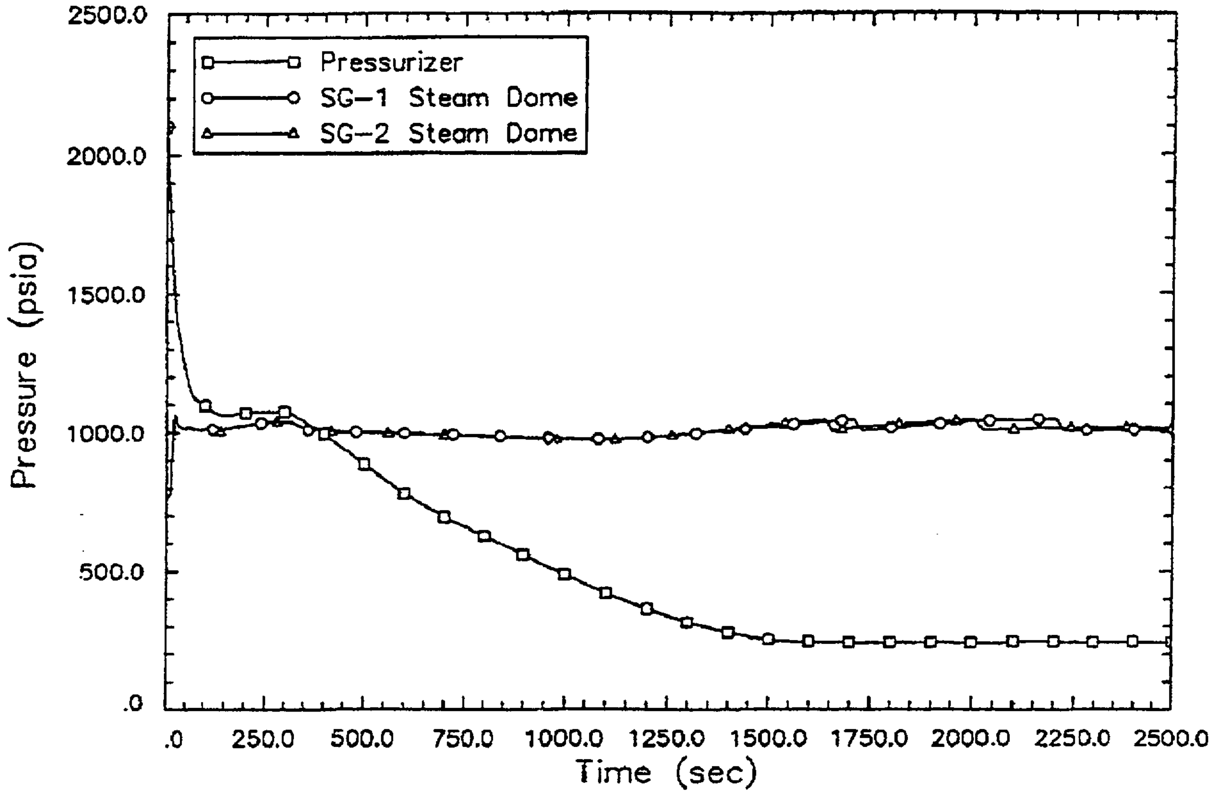


FORT CALHOUN STATION

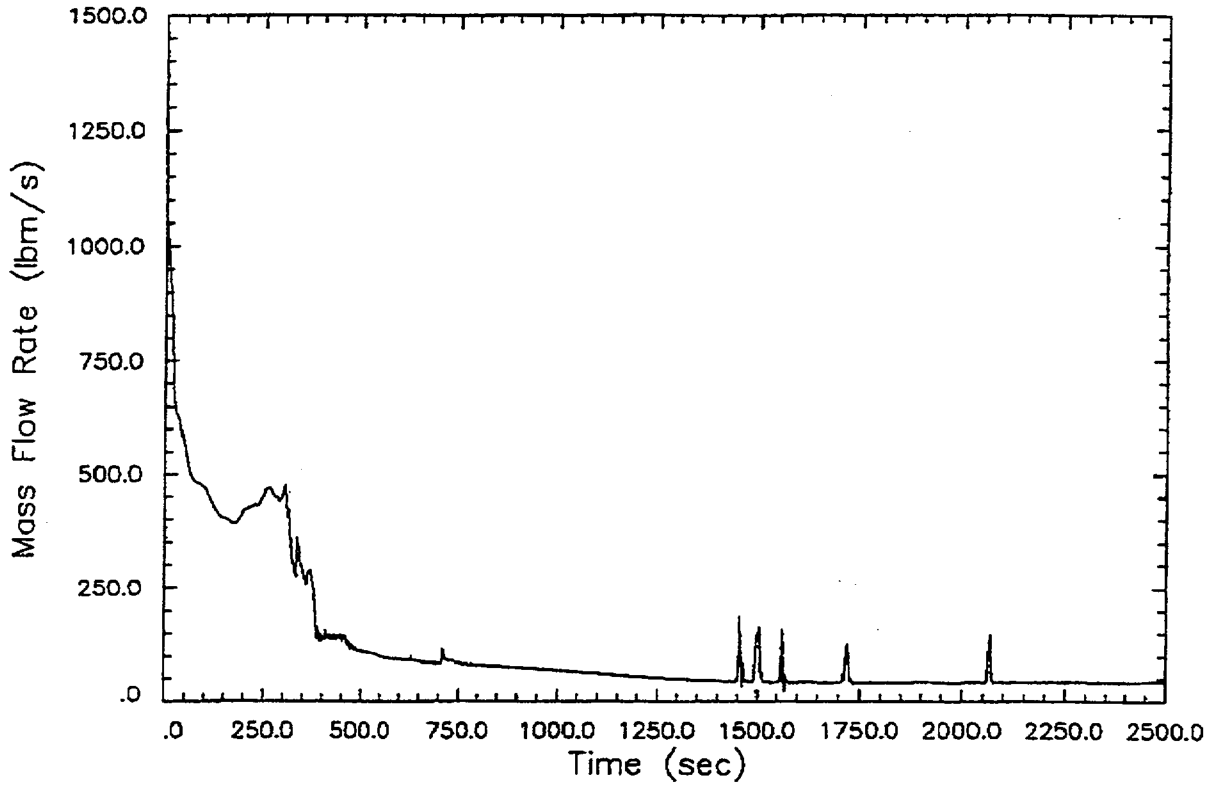
PEAK CLADDING TEMPERATURE:
1.Ø DECLG
EOC NOLPSI

DWG. USAR FIGURE 14.15-6Ø

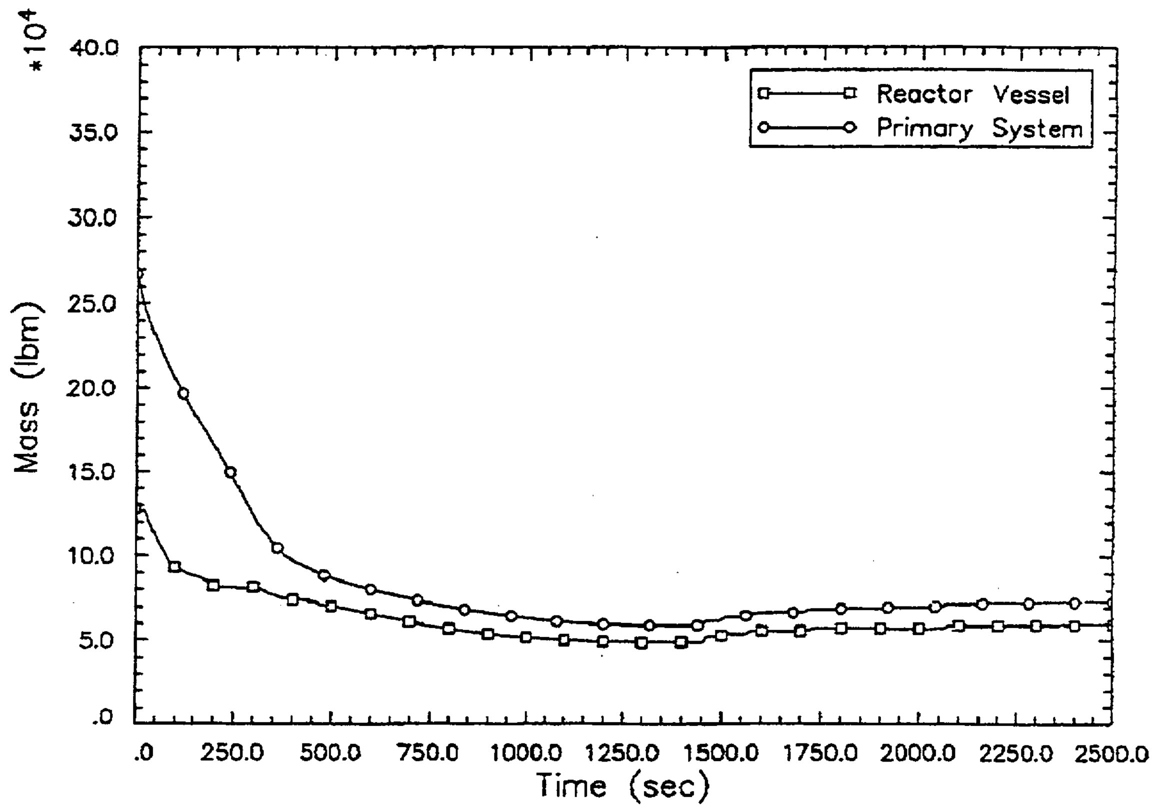
REV.SH. ---	APVD	REV
FILE 48772		Ø



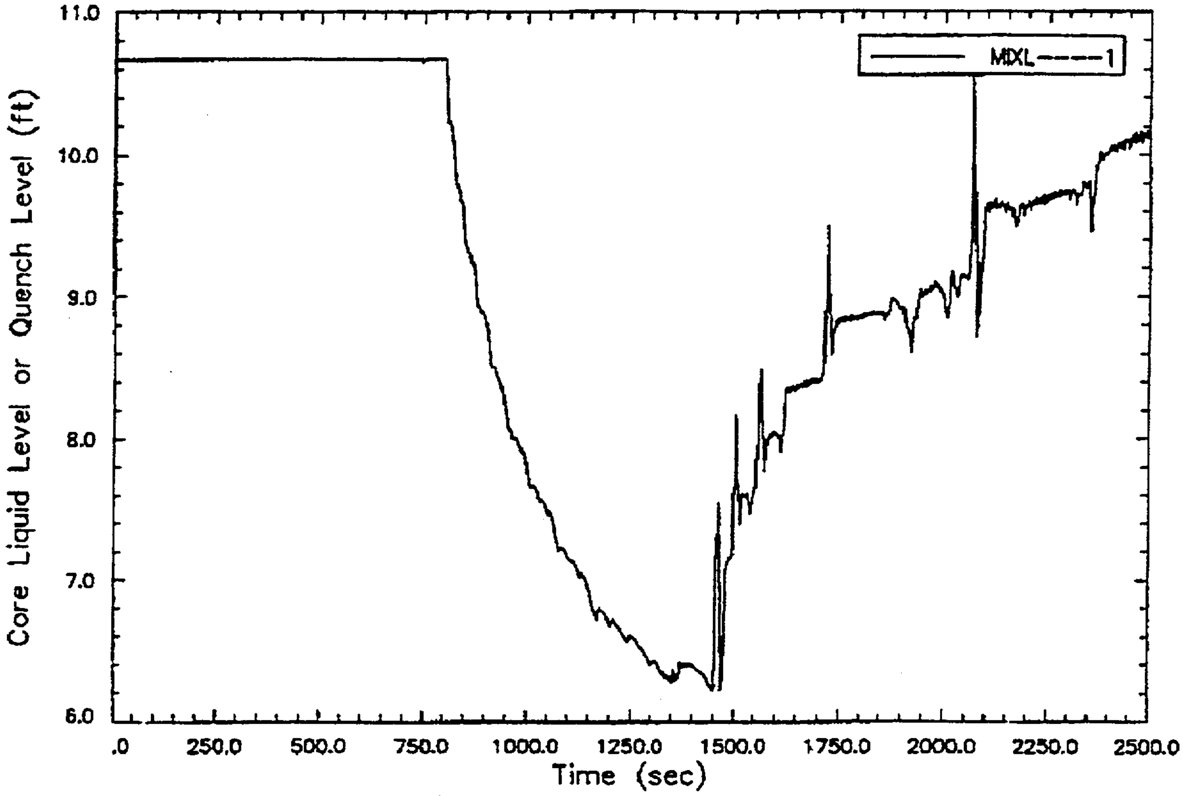
FORT CALHOUN STATION		
PRESSURIZER AND STEAM GENERATOR PRESSURES FOR Ø.Ø49 FT^2 BREAK CASE		
DWG. USAR FIGURE 14.15-61		
REV.SH. ---	APVD	REV
FILE 48773	6/1/01	Ø



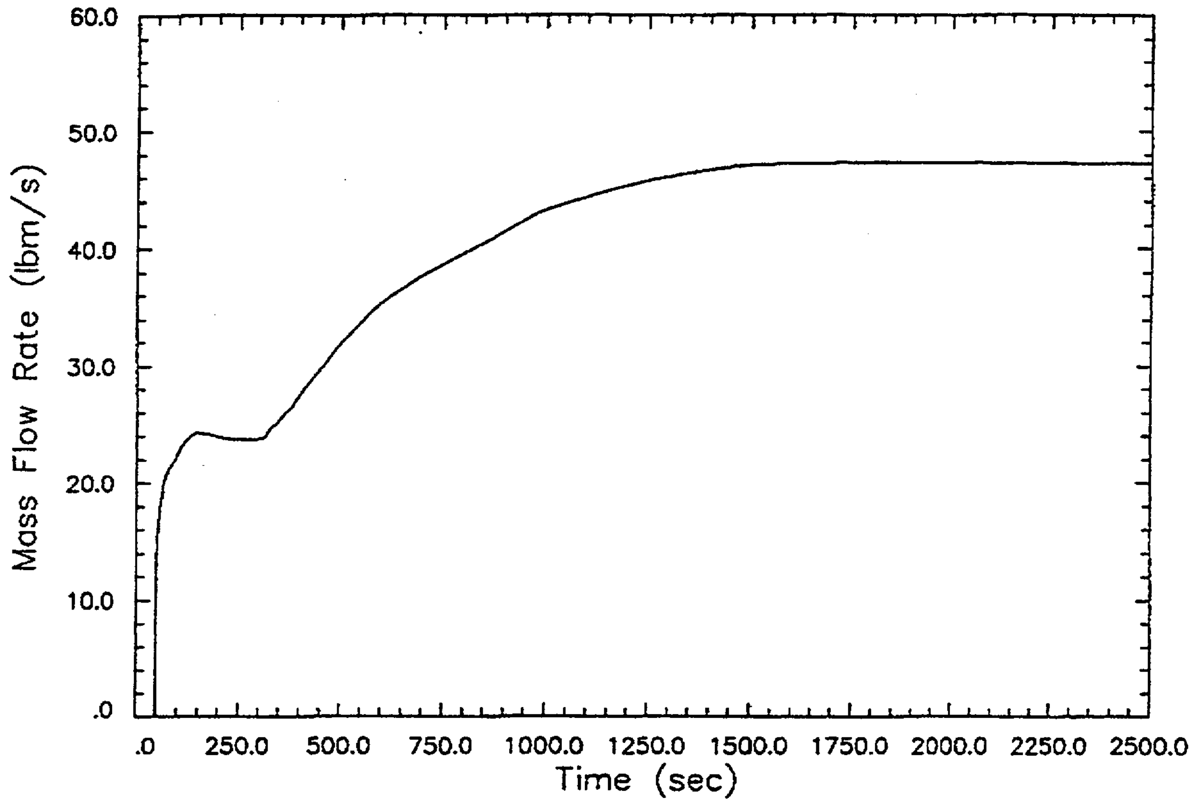
FORT CALHOUN STATION		
BREAK MASS FLOW RATE FOR Ø.Ø49 FT^2 BREAK CASE		
USAR DWG. FIGURE 14.15-62		
REV.SH. ---	APVD	REV
FILE 48776	RL 6/1/01	Ø



FORT CALHOUN STATION		
RCS MASS INVENTORY FOR Ø.Ø49 FT^2 BREAK CASE		
USAR DWG. FIGURE 14.15-63		
REV.SH. ---	APVD	REV
FILE 48777	6/1/01	Ø



FORT CALHOUN STATION		
TOODEE2 HOT CHANNEL MIXTURE LEVEL FOR Ø.Ø49 FT^2 BREAK CASE		
DWG. USAR FIGURE 14.15-64		
REV.SH. ---	APVD 6/1/01	REV Ø
FILE 48778		

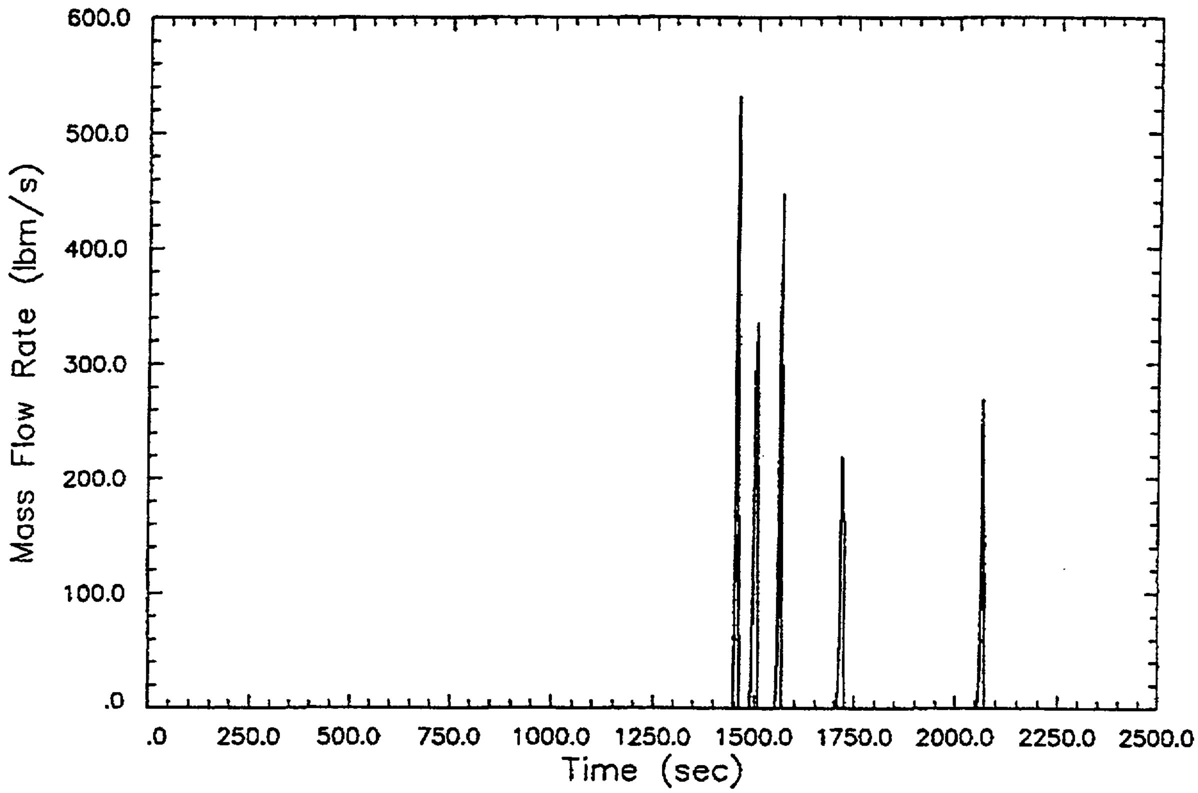


FORT CALHOUN STATION

TOTAL HPSI
MASS FLOW RATE
FOR Ø.Ø49 FT^2
BREAK CASE

USAR
DWG. FIGURE 14.15-65

REV.SH.	---	APVD	REV
FILE	48779	6/1/01	Ø

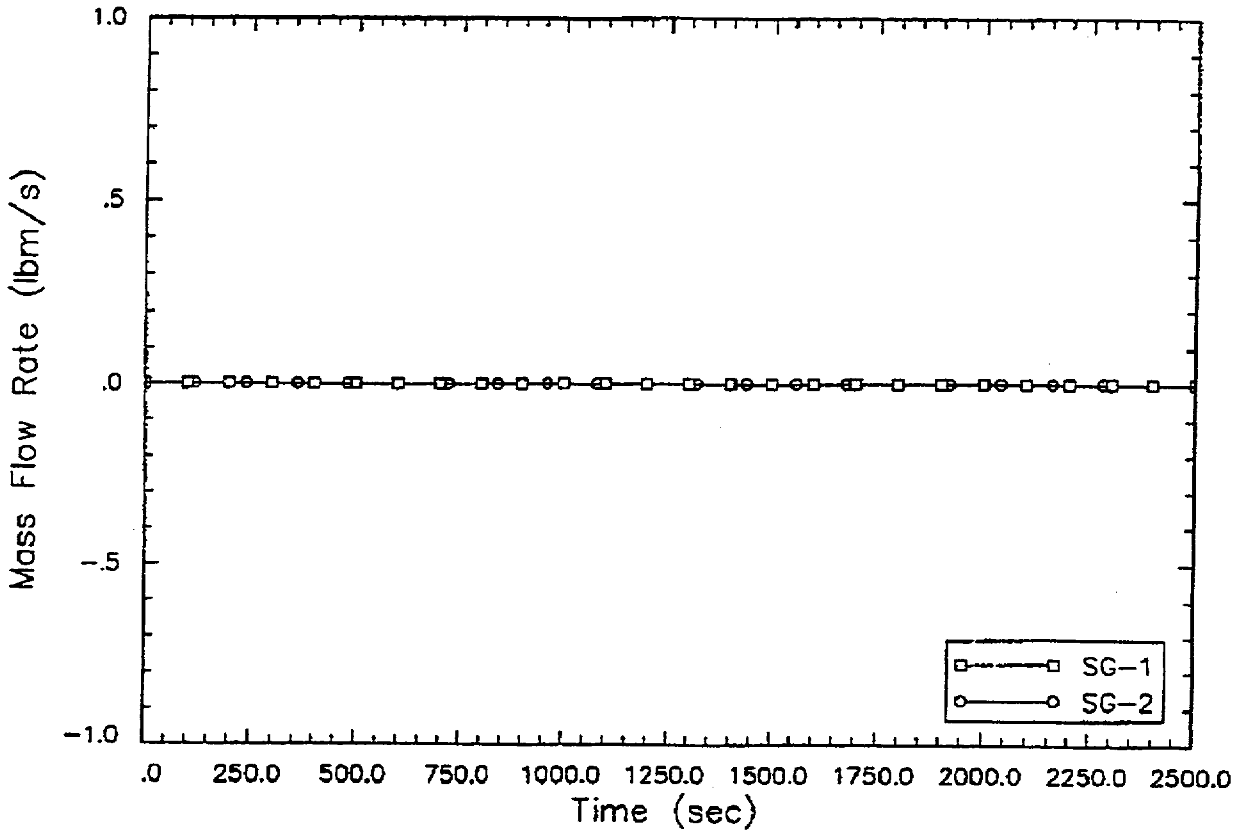


FORT CALHOUN STATION

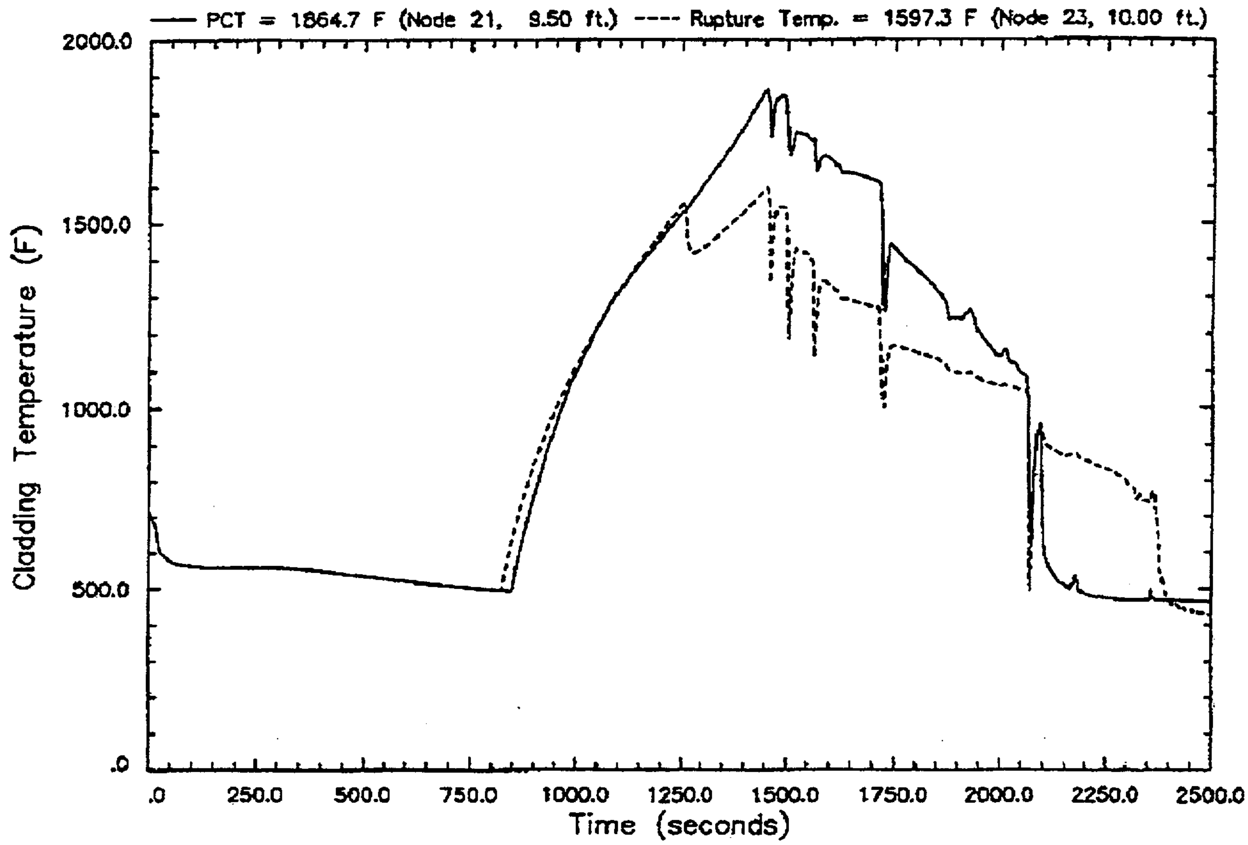
TOTAL SIT
MASS FLOW RATE
FOR Ø.Ø49 FT^2
BREAK CASE

DWG. USAR FIGURE 14.15-66

REV.SH. ---	APVD	REV
FILE 48781	PL 6/1/01	Ø



FORT CALHOUN STATION		
AFW MASS FLOW RATE FOR Ø.Ø49 FT^2 BREAK CASE		
USAR DWG. FIGURE 14.15-67		
REV.SH. ---	APVD 6/1/01	REV Ø
FILE 48782		



FORT CALHOUN STATION

TOODEE2 PCT NODE AND RUPTURE NODE CLADDING TEMPERATURES FOR 0.049 FT^2 BREAK CASE

USAR

DWG. FIGURE 14.15-68

REV.SH.	---	APVD	REV
FILE	48783		Ø