ARKANSAS NUCLEAR ONE - UNIT 1

SAR AMENDMENT 30

RENEWED FACILITY OPERATING LICENSE NUMBER DPR-51

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SAFETY ANALYSIS REPORT

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1 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Final Safety Analysis Report is submitted as required by 10 CFR 50.71. Arkansas Nuclear One - Unit 1 is located in Pope County, Arkansas approximately six miles west-northwest of Russellville, Arkansas and shares the site with ANO, Unit 2.

Arkansas Nuclear One - Unit 1 is designed to operate at core power levels up to 2,568 MWt which, when the 21 MWt contribution from the reactor coolant pumps is included, corresponds to a gross electrical output of 911.5 MWe. Site parameters, principal structures, engineered safeguards, physics, and certain hypothetical accidents are all evaluated at the design power level of 2,568 MWt.

The Nuclear Steam Supply System is a pressurized water reactor type which is similar to many other operating PWRs. It uses chemical shim and control rods for reactivity control and generates steam with a small amount of superheat in once-through steam generators. The Nuclear Steam Supply System and fuel cores were supplied by the Babcock & Wilcox Company. The replacement once-through steam generators are supplied by AREVA/Framatome-ANP.

Entergy Operations, Inc. is fully responsible for the safe operation of the station. The design, construction and aid in the initial testing and startup of the unit have been and will be supplied principally by Bechtel Corporation and the Babcock & Wilcox Company. Assistance has been and will be rendered by other consultants as required.

On October 23, 1996 (0CNA109617), the NRC issued Amendment No. 187 to the facility operating license to revise the name of the plant’s owner from Arkansas Power & Light Company to Entergy Arkansas, Inc. The name “Arkansas Power & Light” or “AP&L” remains in the SAR where the context is historical.

1.2 DESIGN SUMMARY

1.2.1 SITE CHARACTERISTICS

The site consists of approximately 1,100 acres providing for a 0.65 mile exclusion radius. The site is characterized by remoteness from population centers; freedom from flooding; sound, hard rock for structure foundations; a reliable network for emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

1.2.2 POWER LEVEL

The design and license power level for the reactor core will be 2,568 MWt, and all physics and core thermal hydraulics information in this report is based on that power level. An additional 21 MWt is available to the cycle from the contribution of the reactor coolant pumps resulting in a design gross electrical output of 911.5 MWe.

1.2.3 PEAK SPECIFIC POWER LEVEL

For cycle one, the peak specific power level in the fuel for operation at 2,568 MWt results in a maximum thermal output of 17.63 kW/ft of fuel rod. This value is comparable with other reactors of this size presently under construction, and with reactors in the 400-500 MWe class such as San Onofre, Ginna, and Connecticut Yankee, and therefore did not represent an extrapolation of technology.

1.2.4 CONTAINMENT SYSTEM

The reactor building is a fully continuous reinforced concrete structure in the shape of a cylinder on a flat foundation slab with a shallow domed roof. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a 3-way post-tensioning system. Hoop tendons are placed in 3-240 degree systems using three buttresses that run the full height of the cylinder as anchorages. The foundation slab is conventionally reinforced with high-strength reinforcing steel. A continuous access gallery is provided beneath the base slab for installation and inspection of vertical tendons. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The base liner has been installed on top of the structural slab and was covered with concrete. The structure provides shielding for both normal and accident conditions.

The reactor building will completely enclose the entire reactor and the Reactor Coolant System and ensure that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded even if gross failure of the Reactor Coolant System were to occur. The building encloses the Pressurized Water Reactor, steam generators, reactor coolant loops and portions of the auxiliary systems and engineered safeguard systems.

1.2.5 ENGINEERED SAFEGUARDS

The Engineered Safeguards (ES) have sufficient redundancy of component and power sources such that under the conditions of the worst postulated Loss of Coolant Accident (LOCA) the system can maintain the integrity of the containment and keep the exposure of the public below the limits of 10 CFR 50.67.

The engineered safeguards provided for this plant are the following:

A. High Pressure Injection System prevents uncovering of the core for small coolant piping leaks at high pressure and delays uncovering of the core for intermediate sized leaks. This system is normally operated as part of the Makeup and Purification system.

B. The Core Flooding system automatically floods the core when the Reactor Coolant System pressure reaches a level of approximately 600 psig.

C. The Low Pressure Injection System provides core cooling after the reactor coolant pressure has reached approximately 200 psia following a LOCA. This system normally operates as a part of the Decay Heat Removal System during shutdowns.

D. The Reactor Building Spray System sprays borated water into the reactor building atmosphere following a Design Basis Accident to provide cooling and iodine removal. This system is redundant to the Reactor Building Cooling System. The Containment Sump Buffering Agent System uses baskets of sodium tetraborate to provide pH buffering which aids in the iodine removal following initiation of recirculation.

E. The Emergency Feedwater System provides a minimum flow to remove decay heat in the event of a loss of Main Feedwater.

F. The Reactor Building Cooling System provides a heat sink to cool the building atmosphere under the conditions of a LOCA. This system also provides normal building cooling and ventilating requirements.

G. The Reactor Building Isolation System provides automatic isolation of all reactor building penetrations not required for limiting the consequences of an accident.

1.2.6 ELECTRICAL SYSTEMS AND EMERGENCY POWER

Arkansas Nuclear One - Unit 1 is part of the Entergy Arkansas, Inc. system. Unit 1 has the following redundant sources of electrical power:

- Unit 1 generates electrical power at 22 kV and is connected to the 500 kV station switchyard through its unit step-up transformer and a single dead-end span of 500 kV overhead transmission line. A bus-tie autotransformer bank interconnects the 500 kV and 161 kV systems in the station switchyard.

- Startup Transformer 1 is energized from the 22 kV tertiary winding of the 500 to 161 kV autotransformer while Startup Transformer 2, which is common to both Units 1 and 2, is supplied from the 161 kV ring bus. Thus, either of these two startup transformers, which can be fed from different sources, can provide the necessary startup and emergency load requirement of Unit 1. The normal power supply for the unit is the Unit 1 Auxiliary Transformer.

- In the event of loss of normal and preferred auxiliary power sources, the ES loads can be supplied from the onsite emergency power sources. The emergency onsite power sources consist of two independent and completely segregated emergency diesel generators, each of which has an adequate capacity to meet the loads required for

safe shutdown of the reactor. Details of the electrical systems are provided in Chapter 8.

The ES redundant systems have been designed and segregated so that failure of one train will not jeopardize proper functioning of the other train in meeting the shutdown and post-shutdown requirements of the reactor.

Stringent quality assurance standards and programs have been followed in the design, selection, and installation of all Class 1E type equipment and systems. These are discussed in detail under appropriate sections of Chapter 8.

1.3 DESIGN CHARACTERISTICS

1.3.1 DESIGN CHARACTERISTICS

The important design and operating characteristics of the Nuclear Steam Supply System are summarized in Table 1-1. None of the design features parameters of the Nuclear Steam Supply System differ significantly from those of Oconee Nuclear Units.

1.3.2 SIGNIFICANT DESIGN REVISIONS

The more significant design revisions made to the unit between the time the Preliminary Safety Analysis Report was issued and the time the license was issued are listed in the SAR Section 1.3.2.

1.4 GENERAL DESIGN CRITERIA

1.4.1 CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Discussion

Quality standards for those structures, systems and components important to safety are presented in the appropriate sections of this Safety Analysis Report (SAR). The Quality Assurance Program for these structures, systems, and components is presented in the Quality Assurance Program Manual (QAPM). Specific codes and standards for the reactor are presented in Chapter 3, for the Reactor Coolant System (RCS) in Chapter 4, for the reactor building in Chapter 5, for the engineered safeguards in Chapter 6, for the instrumentation and control systems in Chapter 7, and for the electrical systems in Chapter 8.

The following codes and standards are also utilized within the QA program as applicable to those activities to which they are referenced within various sections of this document and the QA Program Manual:

ASME B&PVC Section III, Division 1, 1992 Edition, No Addenda – Nuclear Power Plant Components [In lieu of the original Construction Code, all or portions of later editions/addenda of ASME Section III may be specified for repair or replacement (including system changes) of components or systems, within the rules of ASME B&PVC Section XI and 10 CFR 50.55a. If later editions/addenda are selected, design, fabrication, and examination requirements shall be reconciled with the Owner's specification.] In addition, the replacement OTSGs are manufactured per the requirements of Sections II, III, and V, 1998 Edition, 2000 Addenda. When performing repair/replacement activities in accordance with ASME Section XI, 2001 Edition / 2003 Addenda, to comply with IWA-4540(a)(2), nondestructive examination method and acceptance criteria of the 1992 Edition (or later) of ASME Section III shall be met for welds when a system leakage test is performed in lieu of a hydrostatic pressure test.

ASME B&PVC Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components. The testing of ASME Class 1, 2, 3, MC and CC components shall be performed in accordance with periodically updated versions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55(a). Specific code editions and addenda required by 10 CFR 50.55(a) are referenced in the ISI Program. These requirements apply except when relief has been provided in writing by the Commission.

ASNT SNT-TC-1A, 1984 Edition – Recommended Practice for Nondestructive Testing Personnel Qualification and Certification [This edition shall be used in lieu of earlier editions that might be referenced in other codes or standards to which ANO is committed.]

AWS D1.1, 1992 or later approved Edition – Structural Steel Welding Code. As an alternative to AWS requirements, Welding Procedure Specifications (WPS's), and/or Welder Performance Qualification (WPQ), meeting the ASME Section IX Code requirements may be utilized for structural steel applications provided all other applicable provisions of AWS D1.1 are met unrelated to WPS's and WPQ's.

1.4.2 CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.

Discussion

Systems and components identified in Criterion I have been designed to performance standards that will enable the facility to withstand, without loss of capability to protect the public, the forces or effects that might be imposed by natural phenomena. Additionally, combinations of severe loadings have been considered. The designs are based upon the most severe natural phenomena recorded for the site, with an appropriate safety margin to account for uncertainties in the historical data, or upon the most severe conditions that are susceptible to synthetic analyses.

The referenced sections of the FSAR are as follows:

A. Earthquakes - Sections 2.7, 5.1, 5.2;

B. Floods - Sections 2.4, 5.1;

C. Wind and Tornadoes - Sections 2.3, 5.1, 5.2;

D. Live Loads - Sections 5.1, 5.2;

1.4.3 CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Discussion

The ANO-1 fire protection program has been revised and updated in accordance with NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, as allowed by 10 CFR 50.48. This improved program includes procedures, personnel training, and risk-informed, performance-based analysis of the plant in terms of fire protection. Design details are found in Chapter 9.

1.4.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Discussion

The basic design criterion for structures, systems, and components important to safety prohibits loss of function associated with environmental conditions during normal operation, maintenance, testing, and postulated accidents. Protective walls and slabs, local missile shielding, or restraining devices protect the reactor building liner plate and safeguards systems within the reactor building against damage from missiles generated by failures resulting from loss of coolant. The concrete surrounding the RCS serves as radiation shielding and is an effective barrier against missiles. A local missile barrier is provided for Control Rod Drive Mechanisms.

For those parts of the engineered safeguards systems susceptible to missile damage, physically separated redundant equipment is provided to assure required operation.

High Pressure and Low Pressure Injection within the reactor building consists of four and two injection lines, respectively. The High Pressure Injection lines are connected to the reactor coolant inlet piping on opposite sides of the reactor vessel. For most of the routing, these lines are outside the reactor and steam generator shielding, protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area. A discussion on the prevention of damage from pipe rupture effects is presented in Section 4.2.6.6.

The Reactor Building Spray System headers are outside and above the reactor and steam generator concrete shield. During operation, a movable shield also provides missile protection for the area immediately above the reactor wall. The spray headers are thus protected from missiles originating beneath the shield.

All equipment, piping, valves, and instrumentation in the reactor building associated with the reactor building coolers are located to minimize the possibility of missile damage. The cooling units and associated piping are outside the secondary concrete shielding.

1.4.5 CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety function will not be significantly impaired by the sharing.

Discussion

As discussed in Section 1.7, "Interrelationship With Unit 2," the safety of either unit would not be impaired by the failure of any shared facilities and systems.

1.4.6 CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Discussion

The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady state operation, including the transients listed in the criterion. The integrity of fuel cladding is assured under all normal and abnormal modes of anticipated operation by avoiding overstressing and overheating of cladding. The core design, together with reliable process and Decay Heat Removal Systems, provides for this capability under all expected operating conditions and transients.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine mismatch. Above approximately 22 percent rated power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to dampen the effects of power transients. The RCS will maintain the reactor operating parameters within preset limits and the Reactor Protection System will shut down the reactor if normal operating limits are exceeded by preset amounts.

The Reactor Coolant Pumps have sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated.

1.4.7 CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Discussion

The overall power coefficient, which is the fractional change in neutron multiplication per unit change in core power level, is negative in the power operating range.

1.4.8 CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed.

Discussion

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature when the reactor is operated above approximately 22 percent rated power. Power oscillations from spatial xenon effects are minimized by the negative power coefficient and Axial Power Shaping Rod Assemblies.

The ability of the RCS to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations is provided. Increases in average coolant temperature provide negative feedback and enhance reactor stability during the major portion of core life in which the moderator temperature coefficient is negative. When the moderator temperature coefficient is positive, rod motion will compensate for the positive feedback. The maximum rate of power change resulting from temperature oscillations within the control system dead band is less than one percent per minute. The unit is designed to follow ramp load changes of at least five percent per minute, which is well within the capability of the control system as described in Section 7.2.3.1.

Reactor Protection System action to prevent excessive power peaking is provided by the power/imbalance/flow trip system in conjunction with the pressure/temperature and overpower trips. Reactor power peaking is not a directly observable plant variable; therefore, hot channel reactor power peaking limits are provided by placing limits on the reactor power imbalance. Power imbalance is defined as the power in the top half of the core minus the power in the bottom half of the core. The power imbalance limits provide both Departure from Nucleate Boiling (DNB) protection and fuel melt limit protection. The power/imbalance/flow trip system produces a power level trip which provides limits for the reactor power imbalance. The power level trip and imbalance limits are reduced in accordance with the RCS flow rate to account for less than 4-pump operation.

Control flexibility, with respect to xenon transients, is provided by a combination of control rods and nuclear instrumentation. Axial, radial, or azimuthal neutron flux oscillations will be detected by the nuclear instrumentation. One or more control rod groups can be positioned to suppress and/or correct flux changes.

1.4.9 CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the Reactor Coolant Pressure Boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Discussion

Adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety.

Instrumentation systems include the nuclear instrumentation system, which monitors the neutron flux level from source to 125 percent of rated power; the non-nuclear process instrumentation which measures temperatures, pressures, flows, and levels in the RCS, steam system, and reactor auxiliary systems; and the incore instrumentation system which measures neutron flux at specific locations within the reactor core.

Control is provided by two basic systems:

A. Protection Systems

   The protection systems, which consist of the Reactor Protection System and the safeguards actuation system, perform the most important control and safety functions. The protection systems extend from the sensing instruments to the final actuating devices, such as circuit breakers and pump or valve motor contactors. See discussion in Criterion 20.

B. Regulating Systems

   The reactivity control system monitors power output and regulates output by means of movable control rods and soluble boron. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis. Reactivity is also controlled with Burnable Poison Rod Assemblies. Reactivity control systems are designed to maintain the selected system variables and appropriate nuclear systems within prescribed operating ranges.

Instrumentation is provided for monitoring reactor building pressure and temperature during normal operation and accident conditions. Reactor building sump water level is also monitored.

The Integrated Control System maintains constant average reactor coolant temperature and constant steam pressure at the turbine during steady state and transient operation between approximately 22 and 100 percent full power.

Detailed information is provided in Section 7.

1.4.10 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The Reactor Coolant Pressure Boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

Discussion

The Reactor Coolant Pressure Boundary meets this criterion by means of the following procedures and programs:

A. Material selection, design, fabrication, testing, and certification in accordance with recognized codes, such as ASME.

B. Manufacture and erection in accordance with procedures which reflect code requirements and which are approved by the manufacturer, vendor, customer, and other parties having jurisdiction.

C. Selection of material properties with due consideration to the effects of neutron flux and general radiation.

D. System analysis to account for cyclic effects of thermal transients, mechanical shock, and seismic loadings.

E. Quality assurance program.

1.4.11 CRITERION 15 - RCS DESIGN

The RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the Reactor Coolant Pressure Boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Discussion

An analysis and evaluation of all normal and abnormal operating conditions and transients is integrally related to all RCS and associated systems design. For all anticipated transients, plots of critical variables, e.g. temperature, pressure, and the rate of temperature change, are generated for critical components. Also, the number of lifetime cycles was determined for each transient. All of these analysis results were invoked as functional requirements on the detailed design of the affected systems. Margins for uncertainties were included in (1) the basic analysis assumptions, (2) the assessment of lifetime cycles, and (3) in the code dictated procedures for stress analysis.

1.4.12 CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Discussion

The reactor building system, which consists of the reactor building and the reactor building isolation system, is designed to protect the public from the consequences of an unlikely event such as a LOCA.

The reactor building is designed to safely sustain all credible internal and external loading conditions that may occur during the life of the station. It is designed for an internal pressure of 59 psig with a coincident temperature of 286 °F combined with the Design Basis Earthquake. Due consideration has been given to all site factors and local environment as they relate to public health and safety.

See Chapters 5 and 14 for details.

1.4.13 CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the Reactor Coolant Pressure Boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. For specific details regarding the electrical distribution system, see Chapter 8.

The onsite electric power supplies, including the batteries and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the Reactor Coolant Pressure Boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

<u>Discussion</u>

The electrical power systems conform to this criterion through provision of an offsite 500 kV/161 kV transmission system and onsite diesel engine generators and batteries. Each of these systems provides sufficient capacity and capability to assure operation of the necessary safety functions during anticipated operational occurrences and postulated accidents, assuming the other system is not functioning.

The onsite power supplies, including the diesel generators, batteries and distribution system have sufficient independence, redundancy, and testability to perform their safety functions including a single failure.

Electric power is supplied to the station switchyard by five separate transmission lines. Three lines, one from the Mabelvale substation, one from the Ft. Smith substation, and one from the Pleasant Hill substation, feed the 500 kV ring bus. The remaining two lines, one line from the Russellville East substation and the other from the Pleasant Hill substation, feed the 161 kV ring bus. Two physically independent circuits with startup transformers are provided from the station switchyard to the onsite electrical distribution system. Startup Transformer 1 is supplied by the autotransformer bank through underground cables, and Startup Transformer 2 is supplied by the 161 kV ring bus.

The onsite electrical distribution system arrangement minimizes the vulnerability of vital circuits to physical damage. Two independent circuits can supply power from the different offsite transmission lines through the corresponding transformer to safety oriented components during operating and postulated accident and environmental conditions. Each of these circuits (Startup Transformer 1 and 2) are designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the Reactor Coolant Pressure Boundary are not exceeded. One of these circuits is designed to be available within a few seconds following a loss-of coolant accident to assure that core cooling, containment integrity, and other vital safety function are maintained.

Both ANO-1 and ANO-2 are prevented from automatic transfer to Startup Transformer 2 during normal power operations for all buses except Buses A1/A3 (Unit 1) and 2A1/2A3 (Unit 2). Procedures administratively control Unit 1 and Unit 2 access to Startup Transformer 2. Procedures may allow other fast transfer capabilities to Startup Transformer 2 for specifically analyzed conditions and restrictions defined in approved Engineering Calculations and Evaluations. Both offsite power sources are designed to be capable of non-interrupted availability during a LOCA and the two diesel generators are designed for a maximum starting of 15 seconds from admission of starting air to being ready for loading. Normally, the two 4160-volt engineered safeguards buses are fed on a one-for-one basis from the main buses. Upon loss of all available offsite power sources, the two 4160-volt engineered safeguards buses are each energized from their respective diesels. The DC system is designed to provide continuous power for control, instrumentation, reactor protection and engineered safeguard systems, safeguard actuation control systems, and other loads for normal operation and orderly shutdown. Two independent and physically separated Class 1E 125-volt batteries and DC control centers are provided for the vital instrumentation, distribution panels, and motors. Each Class 1E battery is sized to carry the continuous emergency and vital AC load for a minimum period of two hours in addition to supplying power for the operation of momentary loads during the 2-hour period. A Non-Class 1E 125 VDC battery and DC control center provide power for non-1E load such as emergency lighting, instrumentation, and motors.

Provisions are included to minimize the probability of losing electrical power from the remaining sources as a result of, or coincident with, a loss of the nuclear power unit, the transmission network, or the onsite power sources. With a loss of the electrical power generated by the nuclear power unit, auxiliary plant loads will be shifted automatically by fast-acting bus transfer devices to the offsite power source fed through the startup transformers. The 4160-volt and 6900-volt loads continue to receive offsite power. With the loss of offsite power to the engineered safeguards buses, the associated bus is cleared of all auxiliaries and ties prior to application of the associated diesel generator.

This prevents loss of the diesel generator as a result of an offsite power fault. Because of the design of the fault protection system there is a low probability that loss of the onsite diesel generator power sources could cause loss of either the offsite or nuclear unit electrical power sources.

SINGLE FAILURE ANALYSIS FOR ELECTRICAL POWER SYSTEMS

This analysis establishes that:

A. In the event of main unit trip, the auxiliary plant load will be shifted automatically to the offsite power source.

B. With the loss of offsite power to the safeguard buses, the associated bus is cleared of all auxiliary loads and ties prior to the application of the associated diesel generator.

C. It further establishes that a single failure will not jeopardize the redundancy and the safety of the engineered safeguard system.

EVENT	FAILURE	COMMENTS
Main Unit Trip	Relay 286-G1-1 or Relay 286-G1-2	Failure of one of these two relays will not prevent tripping of the Unit Auxiliary Transformer circuit breaker (C.B. 152-112). One normally open contact of each of the two relays is connected in parallel in the trip circuit of the circuit breaker (152-112) so that the closure of either contact will trip this breaker and permit transferring to off-site power. However, failure of the relay 286-G1-2 will prevent the fast automatic closing of the Startup Transformer 1 circuit breaker (152-113) or the Startup Transformer 2 feeder circuit breaker (152-111), consequently blocking the fast transfer scheme. But through the undervoltage relay on bus A1 and the selector switch (143), one of the two circuit breakers (152-113 or 152-111) will be closed and transfer to the selected offsite power source will thereby be accomplished.
	DC Power Feed to the lockout relays	Both lockout relays (286-G1-1,2), as well as other unit protective relays, have a common DC power supply. With the fault in the unit and in the event of the DC power failure to the lockout relays (which will be annunciated in the control room through alarms), these relays will not trip the Unit Auxiliary Transformer feeder breaker 152-112. When the bus A1 voltage reaches a preset low value, the undervoltage relay on this bus will trip all the connected loads, trip circuit breaker 152-112, and will close circuit breaker 152-113 or 152-111. Therefore, transfer to the preferred off-site power source will be accomplished.
LOCA		In the event of an ES signal, the ES signal logic will both trip breaker 152-112 immediately and start the Emergency Diesel Generator. The bus A1 undervoltage relay will transfer power to the selected offsite source after generator lock-out. Unavailability of both offsite power sources will be sensed by the safeguards bus A3 under-voltage relay which in turn will open the tie breaker between buses A1 and A3 and set the conditions for emergency diesel breaker 152-308 to close after attaining preset voltage level. The indiscriminate tripping of breaker 152-309 and closing of breaker 152-308 is prevented by coordination of undervoltage relays on buses A1 and A3.

1.4.14 CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connectors, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power buses, the offsite power system, and the onsite power system.

Discussion

All important passive components of the emergency power system such as wiring, insulation, connections, and switchboards are designed to permit appropriate periodic inspection and testing to assess their continuity and condition.

System design provides for the following periodic Emergency Diesel Generator electrical tests:

A. Each diesel generator is manually started each month and demonstrated to be ready for loading within 15 seconds. On this manual start, the signal initiating the start of the diesel is varied from one test to another to verify all starting circuits are operable. The generator is synchronized from the control room and loaded.

B. A test is conducted as directed by the Surveillance Frequency Control Program to demonstrate the overall automatic operation of the emergency power system. The test is initiated by a simulated simultaneous loss of normal and standby power sources and a simulated ES signal. Proper operations are verified by bus load shedding and automatic starting of selected motors and equipment to establish that restoration with emergency power has been accomplished within a limited time interval, approximately 70 seconds.

1.4.15 CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 rem TEDE, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Discussion

The control room is designed to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. As discussed in Sections 11.2.4 and 14.2.2.6, adequate radiation protection has been provided to ensure that radiation exposures to personnel occupying the control room following an accident will not exceed 5 rem TEDE, for the duration of the accident.

The control room is designed so that one man can operate the unit during normal steady state conditions. During other operating conditions, other operators will be available to assist the control operator. In the event that the control room must be evacuated, equipment at appropriate locations outside the control room is provided with a design capability for prompt hot standby, > 525 °F of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot standby, > 525 °F. In addition, the potential capability for subsequent cold shutdown will be provided for use in the event that the control room has to be evacuated and is not accessible for a long period of time under the conditions where no accident has taken place and the control room is still intact.

The following design features are provided to insure continuous control room access:

A. Adequate shielding to maintain tolerable radiation levels in the control room during a Design Basis Accident (DBA).

B. Three points of entry from outside the control room.

C. Nonflammable construction.

D. Cables and switchboard wiring pass flame test per IPCEA publication S-61-402 and NEMA WC5-1961.

E. Combustibles such as furniture have been evaluated.

F. Smoke protection and detection equipment is provided.

The Reactor Protection System is designed to be essentially fail-safe without operator control. Thus, safe shutdown can be achieved without operator action.

If the reactor is tripped, and the control room evacuated, reactor decay heat is removed by the steam generators, with steam exhausting through the main turbine bypass valve and/or atmospheric dump valve. Either the main or an emergency feedwater pump will continue to supply feedwater to the steam generators. Additionally, a motor driven auxiliary feedwater pump, normally used for startup and connected in parallel with the main feedwater pumps, could be used to supply feedwater to the steam generators. Under these conditions, a balance will be maintained between heat removal and decay heat generation, the RCS will be maintained at normal hot standby, > 525 degrees F temperature, and no significant makeup will be required for several hours. Any makeup can be supplied by operating the makeup pump, taking suction from the Borated Water Storage Tank and discharging through the normal makeup system lineup. These makeup operations can be conducted locally and the controls and instrumentation are adequate to maintain the plant in a safe hot standby, > 525 °F condition during the period of control room inaccessibility.

1.4.16 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Discussion

Safety analyses have been conducted to assure that acceptable fuel design limits are not exceeded during operational occurrences.

The Reactor Protection System design meets this criterion by monitoring variables that sense the accident condition and provide a reactor trip and safety features action. The Reactor Protection System limits reactor power that might result from unexpected reactivity changes and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits.

The safeguards actuation system senses RCS pressure and reactor building pressure and initiates, automatically, emergency core coolant injection (High and Low Pressure Injection), reactor building isolation and emergency cooling, and the Reactor Building Spray System.

The Emergency Feedwater Initiation and Control (EFIC) System functions and initiate logic are described in Section 7.1.4.

The Core Flooding Tanks are self-actuating.

1.4.17 CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Discussion

The design of protection systems meets this criterion by specific instrument location, component redundancy, and in-service testing capability. The major design criteria as stated have been applied to the design of the instrumentation. In addition, the protection systems meet the single failure criterion of IEEE 279-1968.

Removal of any protection system module from service produces a trip signal and leaves the remaining system with the required minimum redundancy to provide required system reliability.

Test connections and capabilities are built into the protection systems to provide for:

A. Preoperational testing to ensure that the protection systems can fulfill their functions.

B. On-line testing to insure availability and operability.

Test circuits are supplied to utilize the redundant, independent, and coincidence features of the protection systems. This makes it possible to manually initiate on-line trip signals in any single protection channel in order to test the trip capability in each channel without affecting the other channels.

1.4.18 CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions, on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Discussion

The Reactor Protection System provides four independent channels with 2-out-of-4 coincidence logic for trip initiation. The Engineered Safeguards Actuation System provides three analog channels with 2-out-of-3 coincidence logic for initiation of a trip requirement. All protection system functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and associated elements are electrically independent and packaged to provide independence and physical separation.

1.4.19 CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Discussion

The Reactor Protection System consists of four completely independent and separate protection channels.

Loss of power to any individual reactor protection channel will trip that individual channel. Loss of power to any two or more will cause the Reactor Protection System to trip the reactor.

The Engineered Safeguards Actuation System consists of three independent and separate analog channels and two independent and separate digital channels. Each analog channel is supplied from a different power source. The two digital channels are connected to different power sources. Loss of power to an analog channel will cause the channel output to go to a trip state. Loss of power to a digital channel will prevent the digital logic from assuming a tripped condition.

Manual trips are provided for the Reactor Protection System and the Engineered Safeguards Actuation System. Manual reactor trip in the Reactor Protection System is completely independent of the automatic trip circuitry and unaffected by the operating state of the automatic protection system. Safeguards equipment can be manually initiated by the operator at any time, even if power is lost to the actuation system.

The protection systems are designed for continuous operation under adverse environmental conditions.

1.4.20 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited to assure that safety is not significantly impaired.

Discussion

Control system input supplied from the protection system is maintained absolutely separate from the protection system functions by isolation amplifiers which guarantee that a failure or failures in the control system component or channel cannot prevent the protection channel from performing its intended function.

The remaining protection system channels meet all single failure criteria of IEEE-279, dated 1971, when the system experiences a failure of a protection system channel common to protection and control.

1.4.21 CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Discussion

Safety analyses demonstrate that the acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems. The safety analyses also demonstrate that protection system input has been selected to provide the protection required to assure that acceptable fuel design limits are not exceeded.

The reactor design meets this criterion by reactor trip provisions and safety features action. The Reactor Protection System limits reactor power that might result from unexpected reactivity changes and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits.

The reactor design meets the criterion under both normal operating conditions and accident conditions for any single malfunction of the reactivity control systems. The reactor is capable of providing a shutdown margin of at least one percent $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot zero power condition.

Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the reactor cooldown rate of 100 °F/hr. Thus, subcriticality is assured during cooldown with the most reactive control rod totally unavailable.

A reactor trip will protect against continuous withdrawal of the control rods.

1.4.22 CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Discussion

Two independent reactivity control systems of different design principles are provided. The first of these utilizes control rods. The Control Rod Drive System provides a positive means for inserting the rods. The total Control Rod Drive System is capable of reliably controlling the rate of reactivity changes and assures that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified fuel design limits are not exceeded.

The second reactivity control system is the Makeup and Purification System. By means of soluble boron, this system is capable of controlling the rate of reactivity changes resulting from planned normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded.

The Makeup and Purification System also has the ability to initiate and maintain a cold shutdown condition in the reactor.

1.4.23 CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Discussion

The reactivity control systems consist of control rods and soluble boron addition. The reactor is designed so that the control rods provide a hot standby, > 525 °F, shutdown margin of at least one percent $\Delta k/k$ with the single most reactive control rod fully withdrawn. The ECCS provides for injection of water with sufficient boron concentration to make the reactor subcritical. Post-accident injection of borated water by the ECCS is sufficiently fast to limit the core thermal transient and maintain a coolable geometry so that long-term post-accident core cooling capability is maintained.

1.4.24 CRITERION 28 - REACTIVITY LIMITS

The reactivity control system shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the Reactor Coolant Pressure Boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents will include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Discussion

The reactor design meets this criterion via safeguards which limit the maximum reactivity insertion rate. These include rod-group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution-time cutoff. In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions. Ejection of the maximum-worth control rod will not lead to further coolant boundary rupture or to internal damage which would interfere with emergency core cooling. Provisions have been included such that in the event of a steam line rupture, the affected steam generator will be isolated in a time and manner that the core will remain intact for effective core cooling. The reactor system has been designed such that there are no postulated reactivity accidents that will result in a reactor coolant temperature or pressure change of sufficient magnitude to damage the Reactor Coolant Pressure Boundary. The RCS has been designed such that no credible addition of cold water to the core will result in RCS damage.

1.4.25 CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control system shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Discussion

The protection systems and reactivity control systems meet this criterion by specific instrument location, component redundancy, and in-service testing capability. The protection systems meet the standards of IEEE Criteria for Nuclear Power Generating Station Protection Systems, IEEE-279, dated 1968.

The major design criteria stated below have been applied to instrumentation design.

A. No single failure will prevent the protection systems from fulfilling their protective function when action is required.

B. No single failure will initiate unnecessary protection system action provided implementation does not conflict with the criterion above.

Test connections and capabilities are built into the protection system to provide for:

A. Preoperational testing to assure that the protection system can fulfill its required functions.

B. On-line testing to assure immediate operability.

For details concerning the reliability of the Control Rod Drive System see Section 7.2.2. For details concerning the reliability of the boric acid addition system see Section 9.2.2.2.

1.4.26 CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the Reactor Coolant Pressure Boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting, and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Discussion

The RCS pressure boundary meets the criterion through the following:

A. Material selection, design, fabrication, inspection, testing, and certification are in accordance with ASME and ANSI codes.

B. Manufacture and erection are in accordance with approved procedures.

C. Inspection is in accordance with ASME and ANSI code requirements plus additional requirements imposed by the manufacturer.

D. System analysis accounts for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibratory loadings.

E. Selection of reactor vessel material properties gives due consideration to neutron flux effects and the resultant increase of the Nil Ductility Transition Temperature (NDTT).

F. Quality Assurance (QA) program described in the original Appendix 1B.

Materials, codes, cyclic loadings, and nondestructive testing are discussed further in Chapter 4.

The means provided for RCS leak detection and identification are outlined in Section 4.2.3.8.

1.4.27 CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The Reactor Coolant Pressure Boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

Discussion

The Reactor Coolant Pressure Boundary design meets this criterion by the following:

A. Development of reactor vessel plate material properties opposite the core to a specified Charpy V-notch test result of 30 ft-lb or greater at a nominal low NDTT.

B. Determination of the fatigue usage factor resulting from expected static and transient loading during detailed design and stress analysis.

C. Quality Control (QC) procedures including permanent identification of materials and nondestructive testing.

D. Operating restrictions to prevent failure towards the end of design life resulting from increase in the NDTT due to neutron irradiation as predicted by a material irradiation surveillance program.

The reactor vessel is the only RCS component exposed to a significant level of neutron irradiation and is, therefore, the only component subject to material irradiation damage. However, sufficient testing and analysis of ferritic materials in RCS pressure boundary components assure that the required NDT limits specified in the criterion are met. Unit operating procedures limit the operating pressure to 20 percent of the design pressure when the RCS temperature is below NDTT +60 °F throughout unit life. Analysis has shown no potential reactivity-induced conditions which result in energy release to the RCS in the range expected to be absorbed by plastic deformation.

1.4.28 CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the Reactor Coolant Pressure Boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

Discussion

The layout of the RCS and appurtenances is such that appropriate space and means of access is provided for direct and/or approved indirect methods of inspection and nondestructive testing which may be conducted during periods of plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM E-185-66 has been established.

1.4.29 CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the Reactor Coolant Pressure Boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the Reactor Coolant Pressure Boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Discussion

The Makeup and Purification System is capable of supplying borated water makeup for small leaks in the Reactor Coolant Pressure Boundary. A source of water for coolant injection is available from the Borated Water Storage tank. The Makeup and Purification System meets the requirements for offsite and onsite power supplies.

1.4.30 CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the Reactor Coolant Pressure Boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities, shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

Discussion

Reactor decay heat is removed through either one of the two steam generators until the RCS is cooled to where the Decay Heat Removal System becomes operational. Steam generated by decay heat will supply the steam-driven emergency feedwater pump turbine and can also be vented to atmosphere and/or bypassed to the condenser. The steam generators can also be supplied feedwater from one condensate pump and the auxiliary feedwater pump or by the motor-driven emergency feedwater pump. The steam generators provide a long-term capability for decay heat removal.

The main feedwater pumps supply water from the condensate pumps and the condenser hotwell to the steam generators. The emergency feedwater pumps take suction from either the Condensate Storage Tank or the Service Water System. During normal operations, the Condensate Storage Tank is sufficient to provide for decay heat removal after reactor shutdown with the condenser isolated for several hours until additional condensate grade water can be made available. The condenser is normally available so that water inventory is not depleted. During emergency operations, the safety grade Condensate Storage Tank has a 30 minute (minimum) supply of emergency feedwater protected by a tornado missile shield wall, giving operators ample time to align the emergency feedwater pump suctions to the Service Water System. The Reactor Coolant Pumps are provided with sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated. The Decay Heat Removal System will remove the decay heat until the RCS temperature is at a level at which refueling or maintenance may be safely performed. If leakage occurs during system operation, provisions are made for isolation. The Decay Heat Removal System serves as an engineered safeguards system for emergency core cooling; consequently, it is capable of operation from either onsite or offsite power supplies.

1.4.31 CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

Discussion

Low Pressure Injection, High Pressure Injection, and the Core Flooding System comprise an ECCS which maintains the core in a coolable geometry in the event of a LOCA. This system prevents clad melting and minimizes metal-water reaction for the entire spectrum of RCS failures ranging from the smallest break to the complete severance of the largest reactor coolant pipe. Redundancy of components, power supplies, initiation logic, and separation of function is provided so that a single failure will not prevent the ECCS from fulfilling its function. Provisions are made so that component leakage may be isolated. The ECCS may be operated from either onsite or offsite power supplies.

The dynamic post-accident performance analysis demonstrates the capability of the ECCS to terminate the core thermal transient, limit the clad metal-water reaction, and insure maintenance of a coolable geometry.

1.4.32 CRITERION 36 - INSPECTION OF ECCS

The ECCS shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Discussion

The ECCS is designed to permit appropriate periodic inspection of important components of the system.

1.4.33 CRITERION 37 - TESTING OF ECCS

The ECCS shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Discussion

The ECCS is comprised of three independent subsystems: High Pressure Injection, Low Pressure Injection, and Core Flooding. Portions of the High Pressure Injection and the Low Pressure Injection subsystems serve both normal and emergency functions; consequently, the active components regularly demonstrate functional capability. In addition, the systems are designed so that those portions which do not normally operate may be functionally tested during normal operation or at shutdown.

The delivery capability of the ECCS pumps may be checked periodically as required. The capability for delivering water to the core from the Core Flooding System may be demonstrated at shutdown.

The ECCS is designed to test, as far as practical, the full operational sequence including protection system initiation and operability of associated cooling water systems. The discussion of Criterion 18 describes the testing of the power supplies for the ECCS.

1.4.34 CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

Discussion

Either the Reactor Building Spray System or the Reactor Building Cooling System, operating independently, will reduce the post-accident building pressure and temperature to a low level following an accident and will maintain this low level thereafter. This is demonstrated by the post-accident reactor building pressure analysis.

Both of the two diverse systems are designed with redundant components so that a single failure of a component of either system, or a complete failure of either system, will not prevent the function from being fulfilled. Portions of the system may be isolated for leakage control. The Reactor Building Heat Removal Systems may be operated from either onsite or offsite power supplies.

1.4.35 CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Discussion:

The Reactor Building Heat Removal Systems are the Reactor Building Spray System and the reactor building cooling units.

Performance testing of all active components of the Reactor Building Spray System is accomplished as described in Criterion 40. During these tests, the equipment is visually inspected for leaks. Valves and pumps are operated and inspected after any maintenance to ensure proper operation.

The equipment, piping, valves, and instrumentation of the reactor building cooling units are arranged so they can be visually inspected. The cooling units and associated piping are outside the secondary concrete shield around the Reactor Coolant System loops. Personnel can enter the reactor building periodically to inspect and maintain this equipment. Service water piping and valves outside the reactor building, except buried pipe, are inspectable at all times. Operational tests and inspections are performed periodically.

1.4.36 CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Discussion

The components of the Reactor Building Heat Removal System are tested on a regular schedule as discussed in Sections 6.2.4 and 6.3.4. The capability is provided to test under conditions as close to design as practical the full operational sequence that would bring the reactor building pressure reducing system into action just prior to the spray system isolation valves. Test of the transfer to alternate power sources upon loss of normal station power is accomplished by using breaker handswitches in the control room.

Arrangements for testing the Reactor Building Spray System are shown on Figure 6-3. A test connection is provided for the air test of the building spray nozzles and a line is provided for an operational test of the spray pumps by recirculating to the Borated Water Storage Tank.

1.4.37 CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished assuming a single failure.

Discussion

Post-accident reactor building fission product control is accomplished by the Reactor Building Spray System.

Post-accident combustible gas control is accomplished by a Combustible Gas Control System using redundant internal hydrogen recombiners and is discussed in Section 6.6.

1.4.38 CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

Discussion

All critical parts can be physically inspected.

1.4.39 CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components , (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Discussion

Testing of the Reactor Building Spray System and cooling units are discussed with Criterion 40.

The operation of the Reactor Building Purge System can be tested. The operation of the atmospheric cleanup systems can be tested completely except that spray flow is recirculated through a test line to the Borated Water Storage tank.

1.4.40 CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities, shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

Discussion

Structures, systems, and components important to safety are cooled by the Service Water System. The Service Water System is redundant with two 100 percent capacity trains and three 100 percent capacity pumps which can be operated either from offsite power or from onsite emergency power. Two sources of cooling water are available for reactor equipment to use as an ultimate heat sink, the Emergency Cooling Pond and the Dardanelle Reservoir.

1.4.41 CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure system integrity and capability of the system.

Discussion

The Service Water System is designed to permit the required periodic inspections.

1.4.42 CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence bringing the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Discussion

The Service Water System is designed to permit the required testing. See Paragraph 9.3.2.1.

1.4.43 CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with ambient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin will reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Discussion

The reactor building, including access openings, penetrations, and the Reactor Building Heat Removal Systems, has a design pressure of 59 psig at 286 °F. The greatest transient peak pressure (54.0 psig) is associated with the hypothetical 5.0 ft² rupture in the RCS. The reactor building and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA. (See Section 14.2.2.5)

Effects of additional energy sources have been considered in the design. Section 14.2.2.5 discusses the conservatism of the calculational model and input parameters.

1.4.44 CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design will reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Discussion

The Reactor Building Ventilation System is sized to control the interior air temperature to approximately 110 °F bulk average during operation and a minimum of 50 °F during shutdown. In August of 1987, AP&L performed a detailed evaluation of all the components in the reactor building to verify their qualification to a measured 160 °F reference temperature at El 486'. The results of this evaluation were submitted to the NRC in the form of a Justification for Continued Operation (JCO). In the JCO, AP&L committed to lower the ambient reactor building temperature to as low as practicable. Report 92-R-1002-01 documents the improvement to the 110 °F level as a result of HVAC and insulation upgrades. Since the reactor building steel liner plate is not a principal load carrying component and is completely enclosed by the thick concrete walls, slab, and roof of the reactor building, it is not subject to sudden variations due to changes in external temperatures. In addition, the bottom liner plate is protected by a nominal 1-foot, 6-inch thick concrete cover. Nil ductility of the reactor building boundary, including the liner plate, penetrations, valves, and piping, is likely neither at the higher temperatures associated with accident conditions nor at the lower temperatures associated with operation, maintenance, and testing.

Safety of the structure under extraordinary circumstances and performance of the reactor building structure at various loading stages are the main considerations in establishing the structural design criteria. In addition to providing for the leak tight integrity of the liner plate under all loading conditions, the structural criteria requiring reactor building low-strain elastic response to predict its behavior under all design loadings has been applied. See Paragraphs 5.1.4.1 and/or 5.2.1.4 for details of the design methods used to meet these criteria.

The reactor building is designed for all credible conditions of loading, including normal loads, loads during a DBA, test load, and loads due to adverse environmental conditions. See Section 5.2.1.2 for a detailed discussion of loading considerations.

The load capacity of all load carrying structural elements is reduced by a yield capacity reduction factor providing for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision, while individually within required tolerance and the limits of good practice, may combine to result in under-capacity.

1.4.45 CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic Integrated Leakage Rate Testing can be conducted at containment design pressure.

Discussion

The initial Integrated Leakage Rate Test measures the percentage of air that can leak from the reactor building per day at the peak pressure. The preoperational test program will be in accordance with the Reactor Building Leakage Rate Testing Program specified in Technical Specifications 5.5.16.

The reactor building equipment and other equipment which could be subjected to reactor building test conditions are designed so that periodic Integrated Leakage Rate Testing can be conducted. The Reactor Building Ventilation System is used as needed throughout the test to achieve complete air mixing and control of air temperature.

1.4.46 CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

Discussion

The reactor building is designed to permit inspection of all penetrations and for the implementation of a surveillance program. Penetrations with resilient seals are limited to those seals on the personnel lock, escape lock, equipment hatch, and fuel transfer tube closure plate which are designed with double O-ring seals. The annular space between the seals can be pressurized to check for leak tightness. Details regarding the provisions for reactor building leak testing are given in the Technical Specifications and Chapter 5.

1.4.47 CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Discussion

The general design basis governing isolation valve requirements is: leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building, and various types of isolation valves. See Section 5.2.5 for a detailed discussion of the reactor building isolation system.

Testing requirements for containment isolation valve operability and leakage are contained in the Technical Specifications. Piping systems penetrating reactor containment are designed with the capability to perform the functions required in these two paragraphs.

1.4.48 CRITERION 55 – REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the Reactor Coolant Pressure Boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) one locked closed isolation valve inside and one locked closed isolation valve outside containment; (2) one automatic isolation valve inside and one locked closed isolation valve outside containment; (3) one locked isolation valve inside and one automatic isolation valve outside containment; or (4) one automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, will include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Discussion

Penetration and valving design meet the intent of this criterion. Table 5-1 lists reactor building isolation valve information for each penetration. Valves covered by this criterion are defined in Section 5.2.5.2.

Although the pressurizer sample line does not explicitly meet this criterion, the design is acceptable on the basis that the system is used infrequently and the remotely operated valves inside and outside the reactor building are normally closed. In addition to the physical design of the sampling system, administrative control is exercised by requiring the sample station operator to request the control room operator to open the inner and outer reactor building isolation valves.

1.4.49 CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) one locked closed isolation valve inside and one locked closed isolation valve outside containment; (2) one automatic isolation valve inside and one locked closed isolation valve outside containment; (3) one locked closed isolation valve inside and one automatic isolation valve outside containment; or (4) one automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Discussion

Valves which are covered by the above criterion are defined in Section 5.2.5.2. The applicable penetrations are listed in Table 5-1. The penetration and valving design meet the intent of this criterion.

1.4.50 CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the Reactor Coolant Pressure Boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Discussion

Penetration and valving design meet the intent of this criterion. Valves covered by the above criterion are defined in Section 5.2.5.2. The applicable penetrations are listed in Table 5-1.

Although the Core Flooding Tank and steam generator sample lines do not explicitly meet this criterion when the system is in use, the design is acceptable on the basis that the system is used infrequently and both the remotely operated valve inside the reactor building and the manual valve outside the reactor building are normally closed. In addition to the physical design of the sampling systems, administrative control is exercised to ensure that the manual isolation valve will be closed in a timely manner.

1.4.51 CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operations, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Discussion

The Radioactive Waste System collects, segregates, processes, and holds up radioactive solids, liquids, and gases to suitably control releases in order to meet the criterion for as low as practicable.

Liquid and solid wastes are normally processed in batches for offsite disposal. Gaseous waste released to the environment is monitored and discharged to assure tolerable activity levels on the site and at the exclusion distance.

The Gaseous Waste System can store accumulated gas generated during operation. The contents of the decay tanks are periodically sampled, and a release rate is established consistent with the prevailing environmental conditions. In-line monitoring provides a continuous check on the release of activity.

Permanently installed area detectors and the plant vent detectors monitor the discharge levels. In addition, portable monitors are available on site for supplemental surveys if necessary.

Radioactive liquid leakage into the cooling water systems is detected by monitors. These monitors are used for normal operational protection as well as for accident conditions. Detectors monitor the gaseous activity prior to discharge.

1.4.52 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The Fuel Storage and Handling, Radioactive Waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability of a reliability and testability that reflects the safety importance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Discussion

The Fuel Storage and Handling, Radioactive Waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. The systems have the capability for periodic inspection and testing of components important to safety. The shielding design considerations are discussed in Section 11.2.3.

Damage to a fuel assembly in the spent fuel pool releasing radioactive gases to the auxiliary building was evaluated. Exhaust of these gases through the plant vent without filtration results in offsite doses below the 10 CFR 50.67 acceptance criteria.

Accidents assuming rupture of a waste gas tank have been evaluated and the consequences of the release were shown to be well below the guideline values of 10 CFR 100.

Radioactive liquid effluent which might accidentally leak into the Intermediate Cooling Water System will be detected by a radiation monitor. Any accidental leakage from liquid waste storage tanks will be collected in the auxiliary building sump and transferred to other tanks to prevent release to the environment.

A small purification loop removes fission products and other contaminants in the spent fuel storage pool water. The basis for the design of the Spent Fuel Cooling System reflects the importance of this system to safety. The capability for appropriate testing has been provided.

The spent fuel pool is designed to withstand all credible conditions of loading, including normal loads and loads from a Design Basis Earthquake. The fuel pool cannot be drained by gravity since water must be pumped out and siphoning below safe levels is precluded by design.

1.4.53 CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage racks and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Discussion

Criticality in the new fuel storage is prevented by designing storage facilities to maintain a safe geometric spacing of 21 inches, center to center, between assemblies. Fuel assemblies cannot be placed in other than the prescribed location. (See Section 9.6.2.1)

The high density spent fuel assembly storage racks were analyzed using various computer codes and critical experiments were performed. Conservative assumptions were used in these analyses and all uncertainties were taken into account. From these analyses, a maximum neutron multiplication factor of 0.95 or less was obtained.

1.4.54 CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and Radioactive Waste Systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Discussion

Monitoring and alarm instrumentation is 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 exposure. For waste storage and handling, see Chapter 11. Chapter 9 describes the Service Water System and the fuel pool system.

Heat generated in the waste storage facilities, considered to be of a low magnitude, will not require a heat removal system. Monitoring and alarm instrumentation for high radiation levels will be provided to indicate and annunciate high radiation levels of the Radioactive Waste System discharges.

The continuity of decay heat removal from the fuel pool system is assured by two fuel pool pumps and two fuel pool coolers. The system reliability is described in Chapter 9.

Automatic initiation of the Spent Fuel Cooling System is unnecessary on pool high temperature due to the long time available for corrective action to be taken by the operators.

No automatic actions are provided on radiation alarms since there is sufficient time for corrective operator action.

1.4.55 CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Discussion

Monitoring all station solid, liquid, and gaseous releases is accomplished with the appropriate instrumentation (Chapters 7 and 11). Releases from the Reactor Building Ventilation System prior to release are monitored systematically (Chapter 5). The plant ventilation (Chapter 9) is similarly monitored and air flow quantity is controlled to achieve acceptable concentrations. All liquid effluents are monitored both before and after treatment with appropriate holdup available for decay purposes (Chapter 11). The solid effluents are labeled and checked for radioactivity before shipment offsite. Hence, monitoring of the releases within the facility environs keeps them as low as practicable. Beta-gamma detectors located in selected areas of the station, operated according to procedures, will assure that personnel dose does not exceed 10 CFR 20 limits (Chapter 11). The environmental program is designed to establish environmental radiation levels and detect any changes which may occur. Sampling points are chosen both onsite and offsite.

Process radiation monitors are listed in Table 11-7 and area radiation monitors are listed in Table 11-15.

1.5 RESEARCH AND DEVELOPMENT

A number of areas in which research and development would be carried out to finalize design details were identified during the course of the construction permit review. A summary of the status of each of those programs follows.

1.5.1 ORIGINAL ONCE-THROUGH STEAM GENERATOR TEST

Testing necessary to prove the adequacy of the original once-through steam generator design for service at the initial power level and to confirm that the size and configuration of the units has been completed.

The results of the tests have been evaluated to the extent necessary to establish that the design criteria have been met and to establish final design characteristics for manufacture of the steam generators. Topical Report BAW-10027, "Once-Through Steam Generator Research and Development Report," presents the results of the once-through steam generator research and development.

1.5.2 CONTROL ROD DRIVE LINE TEST

The test assembly for this program was a full-sized fuel assembly with associated control rod and control rod guide, adjacent internals, and control rod drive. The purpose of this program was to seek out potential material and/or design problems prior to production unit testing.

The test program of the roller nut mechanism was performed by the B&W Research Center in Alliance, Ohio in sufficient scope and depth to establish that the performance of the mechanism is satisfactory. Topical Report BAW-10029, Revision 1, "Control Rod Drive System Test Program," provides the results of the testing.

1.5.3 SELF-POWERED DETECTOR TESTS

The self-powered detector tests consisted of qualification testing with sufficient longevity to ensure that both neutron flux and power information can be reliably measured. The self-powered detectors have received an integrated dose which is equivalent to over four full power years and have shown no fault that would prohibit their use in PWR service.

The results of the self-powered detector test program are provided in Topical Report BAW-10001, "Incore Instrumentation Test Program."

1.5.4 THERMAL AND HYDRAULIC PROGRAMS

Section 3.3.2 discusses the thermal and hydraulic tests. The core power level for Cycle 1 was justified on the basis of the well-known W-3 DNB correlation. Using the W-3 correlation, the only information required for licensing at the design core power level is that obtained from the reactor vessel model flow tests in which a 1/6-scale model of the vessel and internals was used to measure the flow distribution to the core, fluid mixing in the vessel and core, and the distribution of the pressure drop within the reactor vessel. All of the tests relating to the safety analysis, maximum power rating, and fabrication of reactor internals have been satisfactorily completed. Test data analysis and documentation were conducted and a final report was submitted as proprietary Topical Report BAW-10012, "Reactor Vessel Model Flow Tests."

Data for Cycle 2 through Cycle 8 are based on the B&W-2 correlation and are given in the respective cycle reload report and its supporting documentation. The current cycle reload report is Chapter 3A.

The B&W-2 correlation has been developed and used in place of the W-3 correlation for Cycle 2 through Cycle 8. The B&W-2 correlation is a realistic predication of the burnout phenomenon, and predicts DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. In applying the B&W-2 correlation, the following modifications to Cycle 1 DNB data are used:

A. The limiting design DNBR of 1.30 is used, corresponding to a 95 percent confidence level of a 95 percent probability that DNB will not occur;

B. The pressure range applicable to the correlation has been extended downward from 2,000 to 1,750 psia; and,

C. The design overpower is changed from 114 percent to 112 percent of rated power.

The B&W-2 correlation is fully described in the B&W Topical Report BAW-10000(A), "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water."

During Cycle 9, the Mark-BZ Zircaloy grid fuel design, was introduced. The design change required implementation of the BWC correlation described in the B&W Topical Report BAW-10143P-A, "BWC Correlation of Critical Heat Flux."

During Cycle 20, Mark-B-HTP fuel assemblies were introduced. This new fuel design required implementation of the BHTP correlation described in the Framatome-ANP Topical Report BAW-10241P-A, “BHTP DNB Correlation Applied with LYNXT.”

1.5.5 EMERGENCY CORE COOLING AND INTERNALS VENT VALVES

Analytical evaluations of the effects of blowdown forces on the reactor internals and tests of the performance of the internals vent valves have been completed. The results of the analysis of the pressure-time history in the primary system following a LOCA and the resultant stresses and deflections in the reactor internals are reported in Topical Report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to a Loss-of-Coolant Accident and Maximum Hypothetical Earthquake." A similar investigation was performed to determine the stresses and deflections in the core and was submitted as Topical Report BAW-10035, Revision 1, "Fuel Assembly Stress and Deflection Analysis Due to Loss-of-Coolant Accident and Seismic Excitation."

A full-sized prototype internals vent valve has been analyzed and experimentally tested to assure that the valve is structurally adequate to withstand hydraulic loadings and subsequently perform its steam venting function during a LOCA resulting from a postulated pipe rupture. The results of this investigation are discussed in Section 3.2.3.2.4.

1.5.6 FUEL ROD CLAD FAILURE

A study of clad failure mechanisms associated with a LOCA has been completed. This study included identification of the potential failure mechanisms, a search of the literature to obtain applicable data, evaluation and application of existing data, and scoping tests to obtain data on potential failure mechanisms. The initial results of this study included the identification of the

failure mechanisms, an evaluation of the information available in the literature concerning these mechanisms, and an evaluation of the effects of these mechanisms on the reactor system design.

The objective of the study was to assure that there are no potential failure mechanisms that might interfere with the ability of the ECCS to terminate the core temperature transient and remove decay heat in the event of a LOCA. These potential failure mechanisms include clad melting, zirconium-water reaction, eutectic formation between Zircaloy clad and the Inconel 718 spacer grids, the possibility of clad embrittlement as a result of the quenching during core flooding, and clad perforation or deformation accompanying its failure. In the case of clad melting and zirconium-water reaction, our present design limit for peak clad temperature precludes these as possible failure modes. Information available in the literature, along with experimental evidence from tests conducted by B&W, shows that brittle fracture of the cladding will not occur as a result of quenching following a LOCA and that eutectic formation between dissimilar core materials will not interfere with the flow of emergency core coolant after the accident.

Preliminary tests showed that clad expansion is localized and that any significant pressure is relieved by perforation at temperatures of the order of 1,000 to 1,400 °F. The force for continued expansion therefore is dissipated. Extensions of these preliminary tests evaluated the effects of other variables in order to verify the conclusion that coolant channels will remain sufficiently open to permit core cooling.

Data available indicated that the cladding deformation would be of a random nature and of small magnitude. The interpretation of this data leads to the conclusion that this phenomenon will not affect ECCS performance significantly. Thus, the testing was of a confirmatory nature to more specifically evaluate the effect of clad swelling on the fuel and clad temperature during a LOCA.

Completion of B&W's program provided confirmation that coolant channel restrictions due to clad swelling will not limit ECCS effectiveness. The results of this work were filed as Topical Report BAW-10009, "Effect of Fuel Rod Failure in Emergency Core Cooling Effectiveness."

1.5.7 XENON OSCILLATIONS

This program was concerned with establishing the stability of the core and with evaluating the effects of part-length control rods and burnable poison clusters on core stability. If required, mechanisms for control of diverging xenon oscillations were to be developed.

The xenon program consisted of the following:

A. Modal analysis

B. One-dimensional digital analysis.

C. Two- and three-dimensional digital analysis.

The results of the modal analysis were submitted as Topical Report BAW-10010, Part I, "Stability Margin for Xenon Oscillations - Modal Analysis."

The results of the modal analysis were submitted as Topical Report BAW-10010, Part I, "Stability Margin Per Xenon Oscillations Two- and Three-Dimensional Digital Analyses." As a result of the analysis, the following conclusions were reached:

A. Diverging azimuthal or radial oscillations will not occur.

B. Diverging axial oscillations could occur but can be controlled with the axial power shaping rod assemblies.

1.5.8 IODINE REMOVAL SPRAY

Research and development programs have been carried out by B&W, ORNL, and others to provide data on the use of chemical spray systems for iodine removal.

Results of test programs conducted by B&W are included in proprietary Topical Report BAW-10017, Revision 1, "Research and Development Report on the Stability and Compatibility of Sodium Thiosulfate Spray Solution."

The analysis of test programs to establish the effectiveness of iodine removal sprays are presented in Topical Report BAW-10024, "Effectiveness of Sodium Thiosulfate Sprays for Iodine Removal."

Based on a 1975 B&W analysis indicating deficiencies in the Reactor Building Spray System (RBS) that might result in uneven drawdown of the RBS tanks and unacceptable spray water chemistry during a LOCA, sodium thiosulfate was eliminated from the RBS for iodine removal.

Based on an ANO engineering evaluation (Report No. 89-R-1006-02), the maximum reactor building spray is limited to less than the original design due to pump NPSH considerations.

The Spray Additive System which injected sodium hydroxide was eliminated from the RBS System and replaced with a passive Containment Sump Buffering Agent System in order to assure effective iodine retention post Design Basis Accident (DBA). The Reactor Building sump water pH must be maintained in a range from 7.0 to 10.5. This pH range also minimizes material corrosion for systems and components exposed to the fluid.

The Pressurized Water Reactor Owners Group (PWROG) performed testing on buffering agents for post DBA Reactor Building sump water and selected Sodium Tetraborate (NaTB) as the preferred buffering agent for the application. The PWROG activities were reported in WCAP-16596-NP, "Evaluation of Alternative Emergency Core Cooling System Buffering Agents" dated July 2006.

1.6 QUALITY ASSURANCE

1.6.1 QUALITY ASSURANCE DURING OPERATIONS

NRC accepted Arkansas Power and Light Company's (AP&L's) Topical Report, AP&L-TOP-1A, "Quality Assurance Manual Operations" by letter dated May 5, 1980. This report describes the Quality Assurance (QA) Program which AP&L applies to those operational phase activities involving safety-related structures, systems and components for ANO.

With the acceptance of AP&L-TOP-1A, the chapter section 1.6.1, "Introduction," was deleted since this information is discussed in the Quality Assurance Manual Operations.

As of June 10, 1986, the Quality Assurance Manual Operations was reclassified as a quality program description in lieu of a Topical Report. This change did not reduce AP&L's quality commitments previously accepted by the NRC on May 5, 1980. Revisions to the Quality Assurance Manual Operations, which do not reduce commitments, shall be submitted in conjunction with the ANO-1 Safety Analysis Report.

As of April 30, 1999, the ANO QA Manual Operations was deleted and replaced with the EOI QA Program Manual (QAPM).

1.7 INTERRELATIONSHIP WITH UNIT 2

Separate systems and equipment are provided for Unit 1 with few exceptions. Any item or system not specifically identified in the text or on the system diagrams as being shared is used exclusively for Unit 1. The safety of either unit would not be impaired by the failure of the shared facilities and systems. A summary of the major shared facilities and equipment follows.

1.7.1 ELECTRICAL SYSTEM

Startup Transformer 2 is common to Units 1 and 2 and is available as a standby transformer for its respective unit auxiliary or startup transformers. Its availability as an additional, independent source of power (Section 8.2) to the two sources normally provided for each unit (Unit Auxiliary and Startup Transformer 1 for Unit 1 and Unit Auxiliary and Startup Transformer 3 for Unit 2) allows operational flexibility without loss of the redundancy required for the engineered safety features power supply.

1.7.2 CONTROL ROOM

The plant is provided with a control room located adjacent to the Unit 2 control room. The control panels and equipment are physically separated by a partition wall, eliminating interaction between the Unit 1 and Unit 2 systems. The partition does not extend to the ceiling so the air conditioning and ventilation systems are shared by both units.

There are two normal ventilation systems, one for the Unit 1 half and one for the Unit 2 half, of the common control room. The air intake into each system is monitored for radiation and chlorine. Upon receiving a high radiation or high chlorine concentration signal from any of the normal air intakes, the entire control room, both Unit 1 and Unit 2 sections, is sealed except for filtered outside air used for pressurization to minimize unfiltered air inleakage. This arrangement assures redundancy in the monitoring system. Air conditioning for both control rooms under isolated control room conditions is maintained by emergency air handling and condensing units located in Unit 2. The emergency air conditioning trains are normally powered from vital buses in Unit 2 but one train can be supplied power from a vital bus in Unit 1. Only one emergency air conditioning train is required to operate during and following a shutdown of both units.

1.7.3 EMERGENCY COOLING POND

The cooling pond serves as a heat sink for normal plant shutdown of either Unit 1 or 2, as well as the source of emergency cooling water for simultaneous shutdown of both Units 1 and 2 in the unlikely event of a loss of the Dardanelle Reservoir water inventory. Under controlled conditions, with the Dardanelle Reservoir available, the ECP may provide SW/ACW to Unit 1 with Unit 2 and/or Unit 1 providing normal makeup as necessary to preserve ECP inventory. It is sized to contain sufficient water for dissipating the total combined heat transferred to the Unit 1 and 2 Service Water Systems as a result of the DBA in one unit and a normal plant shutdown of the other unit while limiting the cooling pond temperature to a maximum of 116 °F.

Natural surface drainage was used for initial fill and Dardanelle reservoir water is used to maintain pond level. A suitable spillway is provided to dispose of excess water.

Separate suction and discharge water lines are used for supplying pond water to the Unit 1 and Unit 2 Service Water Systems. The end of the lines terminating at the cooling pond are housed in Seismic Category 1 structures to prevent blockage of the pipe entrance and outlet.

The cooling pond is excavated in an impervious clay strata with the bottom of the pond about 4 to 16 feet above rock. Due to erosion concerns, the original spillway has been replaced by a reinforced concrete ogee shaped spillway located about 60 feet downstream of the original spillway. The original spillway is left in place.

The stability analysis of the ECP embankments is in accordance with the modified Swedish Slip Circle method of analysis. The shear strength used in the analysis was the residual shear strengths obtained from drained direct shear tests on undisturbed samples of the clay from the pond site. Stability analysis for the added embankments for the reinforced concrete ogee spillway used the Simplified Bishops Method, the Simplified Janbu Method, and Spencer's Method. Soil shear strength properties for the clay layers for the added embankments were obtained from laboratory tests of the clay materials used.

For the slopes added with construction of the reinforced concrete spillway for the ECP, the stability analyses were made in accordance with the Simplified Bishop Method, the Simplified Janbu Method, and the Spencer Method. These methods are also part of the family of "The Method of Slices," and provide equivalent or conservative results compared to the Modified Swedish Method (see Army Corps of Engineers, "Slope Stability," EM 1110-2-1902, 2003). For the seismic analysis of these added embankment slopes, the equivalent seismic DBE acceleration of 0.355g was used, which was obtained from an updated soil-structure interaction analysis. The same load conditions as for the original ECP slopes were used for the analysis of these added embankments.

The pond has been excavated using conventional earth moving equipment. The concrete Seismic Category 1 intake and discharge structures are keyed into the existing clay and backfilled using impervious material to form a seal at these points.

Soundings of the pond will be taken annually to determine the amount of silting.

1.7.4 WASTE DISPOSAL SYSTEM

Certain portions of the Waste Disposal System located in the auxiliary building are common for Units 1 and 2. These include:

A. Solid waste handling facilities

B. Laundry Waste System

C. Waste and boric acid concentrators (located in Unit 2)

D. Regenerative Waste System (located in Unit 2)

1.7.5 MISCELLANEOUS

The list below contains other common facilities or systems of some significance; it is not intended to list all shared facilities, systems, or components. A functional evaluation of these components is shown in Table 1-3. Failure of any of these shared systems will be of no serious consequence since such failure will not prevent the safe shutdown of either unit.

A. Raw Water Storage Tank
B. Diesel Fuel Oil Bulk Storage Tank
C. Communications Systems
D. Clean Chemistry Laboratory, Count Room
E. Fire Water Main and Fire Pumps
F. Fuel Handling Crane (Auxiliary Building)
G. On-Site and Off-Site Environmental Monitoring
H. Sodium Bromide/Sodium Hypochlorite System
I. Turbine Building Crane, Elevator
J. Intake Structure Gantry Crane
K. Startup Boiler
L. Office Buildings
M. Railroad Spur, Access Roads, Parking Facilities
N. Switchyard, Transmission Lines
O. Telemetering and Load Dispatching Equipment
P. Shops (Clean Instrument, Hot Instrument, Hot Machine, Machine, and Welding)
Q. Storeroom, Maintenance Facility, Warehouse, and Gas Bottle Storage
R. Reservoir Water Canals
S. Tendon Maintenance Scaffolding
T. Instrument Air Systems
U. Service Air Compressor
V. Breathing Air System
W. Control Room Kitchen and Sanitary Facilities
X. Oily Water Separator
Y. Sewage System
Z. Station Security System
AA. High Purity Hydrogen System
BB. Common Feedwater System
CC. Liquid Nitrogen Storage Tank
DD. Bulk Hydrazine Transfer
EE. Post Accident Sampling System (PASS) Building
FF. Independent Spent Fuel Storage Installation (ISFSI)
GG. Low Level Radwaste Storage Building (LLRWSB)

1.7.6 CONDENSATE STORAGE TANK

A safety grade Condensate Storage Tank, T41B , is shared by ANO-1 and ANO-2. The tank is a seismic qualified source of condensate supply for the Emergency Feedwater System. Both ANO-1 and ANO-2 are connected to the tank.

1.7.7 SAFETY PARAMETER DISPLAY SYSTEM

The SPDS is a computer-based system designed to monitor and display to the operator a concise set of parameters from which the safety status of the plant can be readily and reliably ascertained. The system functions as the SPDS for both the ANO-1 and ANO-2 Control Rooms and provides plant status information for the Technical Support Center (TSC) and Emergency Operations Facility (EOF).

1.7.8 ALTERNATE AC (AAC) POWER SOURCE AND BUILDING

The AAC power source, which consists of a 4400 kW diesel generator (continuous rating) and supporting auxiliaries, is designed to manually pick up the loads on one 4160V ESF bus in either Unit 1 or Unit 2 in the event of a station blackout in either unit. Additionally, it can be connected to a non-1E bus in either unit for performance testing or peaking.

1.7.9 COMMON FEEDWATER SYSTEM

The Common Feedwater System provides a non-safety related alternate source of feedwater to the Unit 1 and Unit 2 steam generators in the event that the Emergency Feedwater system is not available in certain fire scenarios.

1.8 IDENTIFICATION OF AGENTS AND CONTRACTORS

Arkansas Power & Light Company, as owners, arranged for the purchase of equipment and consulting, engineering, and construction services for the installation of Unit 1. As sole owners, Arkansas Power & Light Company was responsible for design and construction. Entergy Operations, Inc. is responsible for the operation and maintenance of the unit.

The Bechtel Corporation was retained for engineering, procurement, and construction services. They also provided assistance in obtaining licenses and permits, in employee training, in acceptance testing, in quality control, and in initial startup of the project.

The Babcock & Wilcox (B&W) Company was contracted with to design, manufacture, and deliver to the site a complete Nuclear Steam Supply System and fuel. In addition, the B&W Company supplied competent technical and professional consultation for erection, initial fuel loading, testing, and initial startup of the complete Nuclear Steam Supply System. B&W participated in initial plant personnel training.

The Westinghouse Electric Company supplied the turbine generator and its auxiliaries. AREVA/ Framatome-ANP supplied the replacement steam generators.

1.9 GLOSSARY OF ITEMS

This section presents a glossary of symbols, indices, legends and other aids to facilitate review of this Safety Analysis Report. Information was compiled from applicable specifications, drawings, SAR sections, and related publications developed throughout the design, construction, and documentation of Unit 1.

1.9.1 DRAWING INDEX AND SYMBOLS

Engineering drawings are cross-referenced with figure numbers in Table 1-5.

1.9.2 PIPING IDENTIFICATION

Piping identification is given in Table 1-6 and corresponds to the codes shown on Piping and Instrumentation Diagrams (P&IDs).

1.10 BABCOCK & WILCOX TOPICAL REPORT REFERENCES

1. BAW-10008, Part 1, Revision 1, Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake

2. BAW-10009, Effect of Fuel Rod Failure on Emergency Core Cooling

3. BAW-10010, Part 1, Stability Margin for Xenon Oscillations - Modal Analysis

4. BAW-10017, Revision 1, Stability & Compatibility of Sodium Thiosulfate Spray Solution - R&D Report (Proprietary)

5. BAW-10024, Effectiveness of Sodium Thiosulfate Sprays for Iodine Removal (Non-proprietary Version of BAW-1002)

6. BAW-10027, Once-Through Steam Generator Research & Development Report (Nonproprietary Version of BAW-10002)

7. BAW-10029, Revision 1, Control Rod Drive System Test (Nonproprietary version of BAW-10007, Revision 1)

8. BAW-10037, Revision 2, Reactor Vessel Model Flow Tests (Non-proprietary version of BAW-10012)

9. BAW-10143P-A, BWC Correlation of Critical Heat Flux (Proprietary)

10. BAW-10241P-A, BHTP DNB Correlation Applied with LYNXT

Table 1-1

**DESIGN PARAMETERS
ARKANSAS NUCLEAR ONE - UNIT ONE**

Data in Table 1-1 are applicable to Cycle 1 operation except for the steam generator, which has been updated for the replacement OTSGs. For data applicable to the current cycle, see Chapter 3A, Reload Report, and its supporting documentation. Note this table has bee updated for the replacement of the NaOH chemical buffer with NaTB per EC-84449.

1. Hydraulic and Thermal Design Parameters

Design Heat Output (core), MWt	2,568
Design Heat Output (core), Btu/h	8,765 x 10^6
Design Overpower, %	114
System Pressure (nominal), psia	2,170
System Pressure (minimum steady state), psia	2135
Power Distribution Factors	
Heat Generated in Fuel and Cladding, %	97.3
F∆h (nuclear)	1.78
Fq (nuclear)	3.03
Hot Channel Factors	
Fq (nc. and mech.)	3.12
DNB Ratio at Rated Conditions	2.0 (W-3)
Minimum DNB Ratio at Design Overpower	1.55 (W-3)
Coolant Flow	
Total Flow Rate, lb/hr	131.3 x 10^6*
Effective Flow Rate for Heat Transfer, lb/hr	124.2 x 10^6
Effective Flow Area for Heat Transfer, ft^2	49.19
Average Velocity Along Fuel Rods, ft/s	15.7
Coolant Temperature, °F	
Nominal Inlet	554*
Average Rise in Vessel	47.8
Average Rise in Core	49.3
Average in Core	579.7
Average in Vessel	578.9
Nominal Outlet of Hot Channel	647.1
Average Film Coefficient, Btu/hr-ft^2-°F	5,000
Average Film Temperature Difference, °F	31
Heat Transfer at 100% Power	
Average Heat Flux, Btu/hr-ft^2	171,470
Maximum Heat Flux, Btu/hr-ft^2	534,440
Average Thermal Output, kW/ft	5.656
Maximum Thermal Output, kW/ft	17.63
Maximum Clad Surface Temperature at Nominal Pressure, °F	654

Table 1-1 (continued)

Fuel Central Temperature, °F	
Maximum at 100% Power at Hot Spot	4220
Maximum at Overpower	4540
Thermal Output, kW/ft at Design Overpower Maximum	20.1

2. <u>Core Mechanical Design Parameters</u>

Fuel Assemblies	
Number	177
Design	CRA canless
Rod Pitch, in.	0.568
Overall Dimensions, in.	8.536 x 8.536
Total Weight, lb/assembly	1,550
Number of Spacer Grids per Assembly	8
Fuel Rods	
Number	36,816
Outside Diameter, in.	0.430
Diametral Gap, in.	0.007
Clad Thickness, in.	0.0265
Clad Material	Zircaloy-4
Fuel Pellets	
Material	UO_2 sintered
Density, % of theoretical	92.5
Diameter, in.	0.370
Length, in.	0.7
Control Rod Assemblies (CRA)	
Neutron Absorber	5% Cd-15% In-80% Ag
Length of Poison Section, in.	134
Cladding Material	304 SS, cold worked
Clad Thickness, in.	0.021
Number of Assemblies	61
Number of Control Rods per Assembly	16
Axial Power Shaping Rod Assemblies (APSRA)	
Neutron Absorber	5% Cd-15% In-80% Ag
Length of Poison Section, in.	36
Cladding Material (Poison Section)	304 SS, cold worked
Clad Thickness, in.	0.021
Number of Assemblies	8
Number of Control Rods per Assembly	16

Table 1-1 (continued)

Burnable Poison Rod Assemblies (BPRA)	
Neutron Absorber	$A1_2O_3$-B_4C
Length of Poison Section, in.	126
Cladding Material	Zircaloy-4, cold worked
Cladding Thickness, in.	0.035
Number of Assemblies	68
Number of Rods per Assembly	16
Orifice Rod Assemblies (ORA)	
Rod Material	304 SS, annealed
Number of Orifice Rods per Assembly	16
Core Structure	
Core Barrel ID/OD, in.	141/145
Thermal Shield ID/OD, in.	147/151

3. Preliminary Nuclear Design Data

Structural Characteristics	
Fuel Weight (as UO_2), metric tons	93.1
Clad weight (active zone), lb	42,200
Core Diameter, in. (equivalent)	128.9
Core Height, in. (active fuel)	144
Reflector Thickness and Composition	
Top (water plus steel), inches	12
Bottom (water plus steel), inches	12
Side (water plus steel), inches	18
Number of Fuel Assemblies	177
Fuel Rods/Fuel Assembly	208
Performance Characteristics	
Loading Technique	3 region
Fuel Discharge Burnup, MWd/mtU (Average-First Cycle)	14,400
Feed Enrichments, wt % ^{235}U (First Cycle)	2.62 avg.
Control Characteristics	
Effective Multiplication (Beginning of Life)	
Cold, No Power, Clean	1.237
Hot, No Power, Clean	1.170
Hot, Rated Power, Xe Equilibrium	1.109
Control Rod Worth ($\Delta k/k$), %	11.1
Boron Concentrations, To Shut Reactor Down With All Rods	
Inserted (clean), cold/hot ppm	929/463
Boron Worth (hot), % ($\Delta k/k$), ppm	1/100
Boron Worth (cold), % ($\Delta k/k$), ppm	1/75

Table 1-1 (continued)

Kinetic Characteristics (Range During Life Cycle)	
Moderator Temperature Coefficient, $(\Delta k/k)/°F$	$+0.28x10^{-4}$ to $-3.0x10^{-4}$
Moderator Pressure Coefficient, $(\Delta k/k/psi)$	$-3.0x10^{-7}$ to $+3.0x10^{-6}$
Doppler Coefficient, $(\Delta k/k)/°F$	$-1.1x10^{-5}$ to $-1.7x10^{-5}$

4. Principal Design Parameters of the Reactor Coolant System

Design Heat Output, MWt	2,584
Operating Pressure, psig	2,155
Reactor Inlet Temperature, °F	554
Reactor Outlet Temperature, °F	604
Number of Loops	2
Design Pressure, psig	2,500
Design Temperature, °F	650
Hydrostatic Test Pressure (cold), psig	3,125
Coolant Volume, including pressurizer, ft^3	11,478
Total Reactor Flow, gpm	352,000

5. Reactor Coolant System Code Requirements

Reactor Vessel and Closure Head	ASME III, Class A
Steam Generator	
Tube Side	ASME III, Class 1
Shell Side	ASME III, Class 2
Pressurizer	ASME III, Class A
Pressurizer Safety Valves	ASME III, Art. 9
Reactor Coolant Piping	ANSI B31.7 (See Note 2)
Reactor Coolant Pump Casing	(Not Code Stamped)

6. Principal Design Parameters of the Reactor Vessel

Material	
Reactor Vessel:	SA-533, Grade B, clad with 18-8 Stainless Steel
Reactor Vessel Closure Head:	SA-508 Class 3, clad with 308L & 309L Stainless Steel
Design Pressure, psig	2,500
Design Temperature, °F	650
Operating Pressure, psig	2,155
Inside Diameter of Shell, in.	171
Outside Diameter Across Nozzles, in.	249
Overall Height of Vessel and Closure Head (over CRD and instrument nozzles), ft/in.	40/8-3/4
Reactor Vessel Minimum Clad thickness, in.	1/8
Reactor Vessel Closure Head Minimum clad thickness, in.	3/16

Table 1-1 (continued)

7. Principal Design Parameters of the Steam Generators

Number of Units	2
Type	Vertical, once-through with integral superheater
Tube Material	Alloy 690
Shell Material	Low Alloy Steel
Tube Side Design Pressure, psig	2,500
Tube Side Design Temperature, °F	650
Tube Side Design Flow, lb/hr	65.66×10^6
Shell Side Design Pressure, psig	1,150
Shell Side Design Temperature, °F	605
Operating Pressure, Tube Side, Nominal, psig	2,155
Operating Pressure Shell Side, Nominal, psig	910
Superheat at Outlet at Rated Load, °F	> 35
Hydrostatic Test Pressure, tube side, cold, psig	3,125
Hydrostatic Test Pressure, shell side cold, psig	1,438

8. Principal Design Parameters of the Reactor Coolant Pumps

Number of Units	4
Type	Vertical, single stage
Design Pressure, psig	2,500
Design Temperature, °F	650
Operating Pressure, Nominal, psig	2,155
Suction Temperature, °F	554
Design Capacity, gpm	88,000
Design Total Developed Head, ft	396
Motor Type	a-c induction single speed
Motor Rating (nameplate), hp	9,000

9. Principal Design Parameters of the Reactor Coolant Piping

Material	Carbon steel clad with SS
Hot Leg (ID), in.	36
Cold Leg (ID), in.	28

10. Engineered Safety Features

Safety Injection System	
Number of High Head Pumps	3
Number of Low Head Pumps	2
Reactor Building Coolers	
Type	Recirculation Fan-Coil Unit
Number of Units	4
Capacity, Each,	
at Containment Design Temperature, Btu/hr	56.5×10^6

Table 1-1 (continued)

Core Flooding System	
Number of Tanks	2
Total Tank Volume, Each, ft^3	1,410**
Reactor Building Spray	
Number of Pumps	2
Capacity, Each, gpm	1500
Emergency Power	
Type	Diesel Engine Generators
Quantity/Capacity	2/2750 kW, 4160 volts
Exclusion Radius, mi	0.65
Low Population Radius, mi	4.0
Reactor Building	
Type	Pre-stressed concrete
Leak Rate, %	0.2
Gross Internal Volume, ft^3	2.048×10^6
Minimum Net Free Volume, ft^3	1.81×10^6
Design Pressure, psig	59
Emergency Feedwater Pump	
Type	Horizontal center.(One steam turbine drive one motor drive)
Head, ft	2600
Flow, gpm	780
Containment Sump Buffering Agent System	Sodium Tetraborate
Type	Passive
Quantity of Baskets	3
Size	6' x 6' x 4.5'
Sump pH	7.0 - 9.0

* Design values do not represent nominal operating parameters.

** 1410 ft^3 is original design; Total volume has been calculated to be 1421 ft^3.

Note 1: Iodine Removal is supported by the Containment Sump Buffering Agent System, which utilizes sodium tetraborate contained in baskets near the reactor building sump to buffer the pH of reactor coolant for effective iodine removal.

Note 2 Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

Table 1-2

**ARKANSAS NUCLEAR ONE-UNIT 1
NUCLEAR QUALITY ASSURANCE
Q LIST**

The purpose of this "Summary Level" Q-List (SLQL) is to generally identify those items within the scope of the Nuclear Quality Assurance Program. Items listed require special consideration during design, manufacture, and construction as described by the Quality Assurance Program Manual.

This SLQL is provided as general reference information only. It preserves the summary level information provided in the SAR "Q-List" with the exception of "Q-numbers" which are no longer used and "Line or Equipment Numbers." The specifications listed are preserved as reference information only and may not be the most current design requirements for specific components or structures. Entergy Operations, Inc. has implemented a detailed "Component Level Q-List" (CLQL) which provides classification of "Q" devices at the component level. Some non-safety related components are included in the SLQL. See CLQL for current classification of these components. The current CLQL is controlled in a computer database in accordance with approved procedures and current design basis requirements for both the SLQL and the CLQL are specified in a combination of documents such as calculations, specifications, procurement documents, and other associated engineering and licensing documents. A classification of "Q" whether for the CLQL or SLQL is intended to be synonymous with the term "Safety Related" defined by Entergy Operations, Inc. in accordance with criteria of 10CFR100, Appendix A. Consequently, the CLQL or SLQL are intended to encompass those structures, systems, and components required to assure:

- The integrity of the Reactor Coolant Pressure Boundary,
- The capability to shutdown the reactor and maintain it in a safe shutdown condition, or
- The capability to prevent or mitigate the consequences of accidents which could result in potential offsite doses comparable to the guideline doses of 10 CFR 100.

When structures, systems, or equipment as a whole are on this list, portions not associated with a loss of safety function are not meant to be included. This includes Engineered Safeguards as defined in Chapter 6.

The Q List is divided into five major sections:

1. CIVIL STRUCTURAL
2. NUCLEAR MECHANICAL
3. CONVENTIONAL MECHANICAL
4. AUXILIARY ELECTRICAL
5. ARCHITECTURAL

Table 1-2 (continued)

Description	Spec No.	Remarks
1.0 CIVIL STRUCTURAL SECTION		
1.1 Prestressed and Reinforced Concrete		
Reinforcing Steel	C-29	
Cadwelds	C-32	
Concrete Including Protective Mat	C-26 C-302 C-28 C-19	C-302 was superseded by ANO-C-2406
Pre-stressing System (Excluding Tendon Sheathing)	C-34	
Sheathing Filler	C-36	
1.2 Liner Plate		
Reactor Building Structure Liner Plate (Excluding leak chase piping system not mating with liner plate and tie nuts)	C-30/C-58	
Locks and Hatch	C-31	
1.3 Fuel Racks		
New Fuel Racks	C-42	
Spent Fuel Racks	C-42	
1.4 Structural Steel		
Containment Structural Steel	C-38/C-37	
Auxiliary Building Structural Steel	C-40	
Miscellaneous Metals	C-48/C-49/C-75	
1.5 Emergency Reservoir	C-64	
1.6 R.C. Pumps Hydraulic Shock Absorbers	C-68	
1.7 Intake Structure Sluice Gate	C-22/APL-C-2414	

Table 1-2 (continued)

Civil-structural items within the scope of the Nuclear Quality Assurance Program are limited to the following:

Reactor building

Auxiliary building housing the engineered safeguard systems, Emergency Diesel Generators, control room, and radioactive materials.

Diesel fuel storage facilities

Enclosures for service water pumps

Supports for Class 1 system components

Emergency Reservoir and pipelines

Spent fuel pool (excluding liner plate)

Description	Spec No.	Remarks
2.0 NUCLEAR MECHANICAL SECTION		
2.1 Reactor Equipment		
Reactor Vessel Equipment		
Vessel	CS-(F)-3-22-T	B&W
Vessel Internals	CS-3-76	B&W
Incore Instrument Tanks	M-85A/M-85B	
Piping (Incore Instrumentation)	M-101A/M-101B/M-101	
Supports	M-119	
Internals Indexing Fixture		B&W
Fuel Handling Equipment		
Fuel Transfer Tube Penetration	CS-272	B&W
Fuel Transfer Canal Recirc Penetration Piping	M-101A/ M-101B/M-101	B&W
Valves	M-123	
Main Fuel Handling Bridge (Trolleys Only)	CS-2-71	B&W

Table 1-2 (continued)

Description	Spec No.	Remarks
Fuel Handling Equipment (continued)		
Auxiliary Fuel Handling Bridge (Trolleys Only)	CS-2-71	B&W
Fuel Storage Handling Bridge (Trolleys Only)	CS-2-71	B&W
Fuel Handling Crane*	M-62	
Shroud Tube Holder		B&W
Control Rod Drive Equipment		
CRD Cooling Penetration Piping	M-103/M-103A	
Valves	M-114/M-114A	
Instrumentation & Controls		B&W
Spent Fuel Pool Cooling and Purification System Valves		
Drive Mechanism Instrumentation & Controls	1089/0969	B&W
Control Rods	CS-3-87	
Drive Mechanism	CS-3-87	B&W
Fuel Assemblies and Miscellaneous Items		
Fuel Assemblies		B&W
Neutron Sources		B&W
Incore Detectors		B&W
Incore Detector Seal Assemblies		B&W

* Reclassified to Non-Q by the Component Level Q-List Project (CLQL)

2.2 Reactor Primary Systems

Description	Spec No.	Remarks
Reactor Coolant System		
Reactor Coolant Pumps	CS-3-36	B&W
Steam Generators	08-5015531 M-559	AREVA Framatome-ANP
Piping	M-101/M-103/M-101A M-101B/M-103A	B&W

Table 1-2 (continued)

Description	Spec No.	Remarks
Reactor Coolant System (continued)		
Valves	M-123	Bechtel
Supports	M-119	B&W / Bechtel
I&C		B&W / Bechtel
Reactor Coolant Pump Motor Flywheels	CS-3-16 ANO-E-108(JI)	B&W ANO
Pressurizer		
Pressurizer Vessel	C-3-32	B&W
Pressurizer/RC Sample Penetration Pipe	M-103/M-103A	
Pressurizer Surge Line		B&W
Pressurizer Spray Line		B&W
Pressurizer Relief Piping	M-101/M-101A/M-101B	
Pressurizer Auxiliary Spray Line	M-103/M-103A	
Valves	M-123	B&W / Bechtel
Pressurizer Relief Valves	CS-3-79/CS-3-28	B&W
Supports	M-119	
I&C		B&W / Bechtel
Makeup and Purification System		
Makeup Pumps	1130/0369	B&W
Letdown Coolers	SPEC-16-00001-A	ANO
Seal Injection Filter		B&W
High Pressure Injection Piping	M-101/M-102/M-101A M-101B/M-103A	B&W
RCP Seal Water Piping	M-103/M-103A	
Makeup System Piping (Partial)	M-101/M-102/M-101A M-101B/M-103A	

Table 1-2 (continued)

Description	Spec No.	Remarks
Makeup and Purification System (continued)		
Valves	M-233/M-235	B&W / Bechtel
Supports	M-119	
I&C		B&W / Bechtel
Core Flooding Systems		
Core Flooding Tanks (CFT)		B&W
CFT Fill, N_2 Supply, Sample and Bleed Penetration Piping	M-101/M-103/M-103A M-101A/M-101B	
Core Flooding Tank Piping	M-101/M-101A/M-101B	
Valves	M-123	B&W / Bechtel
Supports	M-119	
I&C		B&W / Bechtel
Reactor Building Spray System		
Hydroxide Tank		B&W
Spray Pumps	1130/0369	B&W
Piping	M-101/M-103/M-103A M-101A/M-101B	
Valves	M-123	B&W / Bechtel
Supports	M-119	
I&C		B&W / Bechtel
Spray Pump Motors	1125/1268	B&W
Valves	M-233/M-235	B&W / Bechtel
Supports	M-119	

Table 1-2 (continued)

Description	Spec No.	Remarks
Decay Heat Removal System		
Borated Water Storage Tank	M-291	
Decay Heat Removal Pumps	1130/0369	B&W
Decay Heat Removal Coolers	CS-3-24	B&W
Piping	M-101/M-103/M-103A M-101A/M-101B	
Valves	M-123/M-245	B&W / Bechtel
Supports	M-119	
I&C		B&W / Bechtel
Decay Heat Removal Pump Motors	1125/1268	B&W
2.3 Radioactive Waste Treatment		
Gaseous Radwaste System		
Reactor Bldg Vent Header Penetration Piping	M-103/M-103A	
Valves	M-114/M-123/M-124 M-235/M-235A	
Supports	M-119	
I&C		
Liquid Radwaste System		
Quench Tank Gas Space Sample Penetration Piping	M-103/M-103A	
Quench Tank Drain Penetration Piping	M-110	
Quench Tank Condensate Supply Penetration Piping		
Reactor Bldg Sump Drain Penetration Piping	M-101/M-101A/M-101B	
Reactor Bldg Drain Header to Aux Bldg Drain Tank Penetration Piping	M-110	

Table 1-2 (continued)

Description	Spec No.	Remarks
Liquid Radwaste System (continued)		
Valves	M-123	
Supports	M-119	
I&C		
2.3 Reactor Plant Service Systems		
Intermediate Cooling Water System		
Intermediate Cooling Penetration Piping	M-101/M-101A/M-101B	
Valves	M-123	
I&C		
Treated Water Systems		
Filtered Water Supply Penetration Piping	M-103/M-103A	
Valves	M-114/M-115/M-114A	
Air/Nitrogen System		
Nitrogen Supply Penetrations Piping	M-103/M-103A	
Service Air Supply Penetration Piping	M-103/M-103A	
Instrument Air Supply Penetration Piping	M-103/M-103A	
Containment Test Connection Penetration Piping	M-101/M-101A/M-101B	
Valves	M-114/M-114A	
I&C		
Process Radiation Monitoring System	M-217	
Erection Cleaning on Nuclear SS Pipe	M-230	

Table 1-2 (continued)

Description	Spec No.	Remarks
3.0 CONVENTIONAL MECHANICAL SECTION		
3.1 Reactor Secondary Systems		
Main Steam System		
Piping from S/G to Main Steam Isolation Valves	M-101/M-101A/M-101B	
S/G Sample Penetration Piping	M-103/M-103A	
Emergency FW Pump Steam Line		
S/G Secondary Drain Penetration Piping	M-101/M-101A/M-101B	
S/G Recirc to Main Condenser Penetration Piping	M-101/M-101A/M-101B	
Main Steam Safety Valves	CS-2-103	B&W
Other Valves Including Atmospheric Dump Block Valves	M-123/M-233	
Main Steam Isolation Valves	M-120	
Supports	M-119	
I&C		
Feedwater System		
Emergency Feedwater Pumps	M-18/APL-M-2416	
FW Piping from S/G to Isolation Valve	M-101/M-101A/M-101B	
Emergency Feedwater Piping	M-101/M-101A/M-101B	
Valves	M-123/M-245	
Supports	M-119	
I&C		B&W
Emergency FW Pump Turbine Driver	M-18	
Condensate Storage Tank T41B	APL-M-2511	

Table 1-2 (continued)

Description	Spec No.	Remarks
3.2 Conventional Plant Service Systems		
Service Water System	M-11	
Service Water Strainers	M-293	
Piping	M-101/M-103/M-107 M-101A/M-101B/M-103B	
Valves	M-123/m-242/M-244	
Supports	M-119	
I&C		Bechtel
Service Water Pump Motors	M-11	
Heating, Ventilation, and Air Conditioning		
Reactor Building		
Cooler Housing, Service Clg Coil, Fans	M-61	
Penetration Piping (Chilled Water, Plant Heating, H_2 Purge)	M-101/M-101A/M-101B	
Isolation Valves (Chilled Water, Heating System)	M-123	
Purge Isolation Valves	M-239	
H2 Purge Isolation Valves		
Ducts, Duct Supports, Dampers	M-52A	
Duct Relief Valves	M-15	
H2 Recombiners		
Auxiliary Building		
Unit Coolers (Swgr Rm, Decay Heat Rm Makeup Pump Rm)	M-61	
EDG Room Exhaust Fans	M-57A	
Eng. Safeguards Equip Rms Purge Valves	M-123	

Table 1-2 (continued)

Description	Spec No.	Remarks
Ducts, Duct Supports, Dampers	M-52A	
Emergency Chilled Water Chillers	M-265	
Control Room		
Emergency Supply Air Filters & Fan	M-61	
Penetration Room Ventilation System		
Filters	M-267	
Lines	M-133/M-133A	
Fans	M-266	
Duct Isolation Valves	M-268	
Filter and Bleed Line Shutoff Valves	M-269	
Fuel Handling Area Ventilation System**		
Exhaust Fans**		
Filters**		
Ducts and Duct Supports**		
Supports	M-119	Bechtel
I&C	M-12	
Emergency Diesel Generator including:		
Starting Air	M-12	
Jacket Water Heat Exchanger	M-12	
Fuel Oil (FO) Day Tanks	M-12	
Fuel Tanks	M-85/M-85A/M-85B	
Diesel Oil Transfer Pumps	M12	

** Power, control, and electrical components are Non-Class 1E. This system will be maintained with the quality requirements associated with a safety related system on all future work as per NRC requirements.

Table 1-2 (continued)

Description	Spec No.	Remarks
Emergency Diesel Generator (continued)		
Piping (Air and FO)	M-101/M-103/M-107 M101A/M-101B/M103A	
Combustion Air Intake Duct	M-52A	
Wall Louvers	M-215	
Valves (Aire and FO)	M-123	
Supports	M-119	
I&C	M-12	
Electrical Control Cabinets	M-12	
Diesel Oil Transfer Pump Motors	M-12	
Reactor Building Crane	M-73	
Fire Water Penetration Piping	M-101/M-101A/M-101B	
Miscellaneous Items		
Spare Penetration Pipe Caps	M-101/M-101A/M-101B	
Flued Heads	M-289	
Miscellaneous Instruments or Sample Adapter Piping	M-103/M-103A	
Valves	M-123	
Control Panels	M-201/M-217	Bechtel / B&W
4.0 AUXILIARY ELECTRICAL SECTION		
4.1 Switchgear For Engineered Safeguards		
4.16 kV Switchgear	E-8	
4.16 kV Switchgear A3 & A4	ANO E-2451 & E-8	
4.16 kV Switchgear A6 & A8	APL-E-2417	

Table 1-2 (continued)

Description	Spec No.	Remarks
Load Center Transformer Engineered Safeguards		
4160 – 480 V Trans. 1000 kVA X5 & X6	E-10	
AC Instr. Transf. 120/208 V 30 kVA S/B: X6A & X6B, X-51 & X-61	E-19	
AC Instr. Transf. 120/208 V 45 kVA S/B: X-52 & X-62	APL-E-19	
Load Centers Engineered Safeguards		
480 V Load Center Bus B5 & B6	E-10	
Reactor Building Crane	M-73	
Fire Water Penetration Piping	M-101/M-101A/M-101B	
Motor Control Center (MCC) Engineered Safeguard		
MCC B51, B52, B53, B55, B56, B61, B62, & B63	E-11	
4.2 Switchboards and Panels		
AC Instrument Distribution Panels		
120/208 V AC Instrument Distr. Panel Y1 and Y2	E-19	
120/208 V AC Instrument Distr. Panel Y3 and Y4	APL-E-19	
Power and Distribution Panels Engineered Safeguard		
120/208 V AC Reactor Protection and Engineered Safeguard Dist. Panel RS1, RS2, RS3 and RS4	E-17	
4.3 Raceways Associated With Engineered Safeguards		
Conduit-Rigid Steel Seal Tight Flexible Including Fittings	E-59 (Drawing)	Partial
Cable Tray-Associated with Electrical Engineered Safeguard Equipment Power & Control Cables	E-59 (Drawing)	Partial
Cable Tray Fire Barriers	E-59 (Drawing)	Partial
Pull Boxes and Terminal Boxes	E-58 (Drawing)	Partial
Underground Duct, Fittings, and Encasement	E-59 (Drawing)	Partial
Manholes, Handholes, and Fittings	E-59 (Drawing)	Partial

Table 1-2 (continued)

Description	Spec No.	Remarks
4.4 Cable Associated With Engineered Safeguard		
Instrument Cable		Partial
Triaxial		Partial
Coaxial		Partial
Shielded		Partial
Thermocouple Cable	E-26	Partial
600 V Control Cable Switchboard Wire	E-25	Partial
600 V Power Cable	E-25	Partial
5 kV and 8 kV Power Cable	E-24	Partial
4.5 DC Equipment		
Battery Bank		
Battery Bank D06 & D07	E-16	
Battery Chargers		
Battery Charger D03A, D03B, D04A, & D04B	E-18	
AC Inverters		
Inverter Channel Y11, Y13, Y15, Y22, Y24, & Y25	E-18	
DC Motor Control Centers		
DC Motor Control Center D01, D02, D15 & D25	E-17	
Battery Control Panel		
Battery Bank Fuse and Relay Cabinet D16 & D17	E-16	
125 V DC Power Panels		
125 V DC Distribution Panel D11 & D12	E-17	

Table 1-2 (continued)

Description	Spec No.	Remarks
4.5 DC Equipment (continued)		
125 V DC Distribution Panels		
125 V DC Safeguard Actuation System Control Panel RA1 & RA2	E-17	
125 V DC Safety Disconnect Switches		
Battery Bank D13 & D14		
4.6 Miscellaneous Electrical Equipment		
Containment Electrical Penetrations		
12” Penetrations for Power	E-23	Partial
12” Penetrations for Control	E-23	Partial
12” Penetrations for Instrumentation	E-23	Partial
DC Transfer Switches	E-35	
Isolation Relays	E-36	
RPS Relays	E-38	
5.0 ARCHITECTURAL SECTION		
5.1 Impingement Door	A-38	
5.2 Watertight Door	A-41/A-027	A-41 was superseded by APL-A-2407

Table 1-3

EVALUATION OF MISCELLANEOUS SHARED SYSTEMS
(See Section 1.7.5 and Unit 2 SAR Section 1.2.2.10.E, Table 1.2-1)

	System	Serves Shutdown Function	Serves Emergency Function	Condition of Maximum Demand	PSAR Section Describing Sufficient Redundancy
A.	Raw Water Storage Tank	No	No	NA	NA
B.	Diesel Fuel Oil Bulk Storage	No	No	Loss of Offsite Pwr	NA
C.	Communications Systems	Yes	Yes	Variable	7.4.4
D.	Clean Chemistry Laboratory, Count Room	No	No	NA	NA
E.	Fire Water Main and Fire Pumps	No	Yes	Fire	9.8
F.	Fuel Handling Crane (Auxiliary Building)	No	No	NA	NA
G.	On-site and Off-Site Environmental Monitoring	Yes	Yes	DBA	11.2.6
H.	Sodium Bromide / Sodium Hypochlorite System	No	No	NA	NA
I.	Turbine Building Crane, Elevator	No	No	NA	NA
J.	Intake Structure Gantry Crane	No	No	NA	NA
K.	Startup Boiler	No	No	NA	NA
L	Office Buildings	No	No	NA	NA
M.	Railroad Spur, Access Roads, Parking Facilities	No	No	NA	NA
N.	Switchyard, Transmission Lines	Yes	Yes	Simultaneous Trip of Both Units	8.2

Table 1-3 (continued)

	System	Serves Shutdown Function	Serves Emergency Function	Condition of Maximum Demand	PSAR Section Describing Sufficient Redundancy
O.	Telemetering and Load Dispatching Equipment	No	No	NA	NA
P.	Shops (See 1.7.5.P)	No	No	NA	NA
Q.	Storeroom, Maintenance Facility, Warehouse, and Gas Bottle Storage	No	No	NA	NA
R.	Reservoir Water Canals	No	No	NA	NA
S.	Tendon Maintenance Scaffolding (portable)	No	No	NA	NA
T.	Instrument Air Systems (only when cross-connected)	No	No	NA	NA
U.	Service Air Compressor	No	No	NA	NA
V.	Breathing Air System	No	No	NA	NA
W.	Control Room Kitchen and Sanitary Facilities	No	No	NA	NA
X.	Oily Water Separator	No	No	NA	NA
Y.	Sewage System	No	No	NA	NA
Z.	Station Security System	No	No	NA	NA
AA.	High Purity Hydrogen System	No	No	NA	NA
BB.	Common Feedwater System	No	Yes	Fire	NA
CC.	Liquid Nitrogen Storage Tank	No	No	Outage	NA

Table 1-3 (continued)

	System	Serves Shutdown Function	Serves Emergency Function	Condition of Maximum Demand	PSAR Section Describing Sufficient Redundancy
DD.	Bulk Hydrazine Transfer	No	No	NA	NA
EE.	Post Accident Sampling System (PASS) Building	No	No	Post-DBA	NA
FF.	Independent Spent Fuel Storage Installation (ISFSI)	No	No	NA	NA
GG.	Low Level Radwaste Storage Building (LLRWSB)	No	No	NA	NA

Table 1-4

DELETED

Table 1-5

FIGURE - DRAWING CORRELATIONS

Drawing Number	SAR Title	Figure Number
A-411	Radiation Zones Plant Elevation 317'0"	Deleted
A-412	Radiation Zones Plant Elevation 335'0"	Deleted
A-413	Radiation Zones Plant Elevation 354'0"	Deleted
A-414	Radiation Zones Plant Elevation 372'0"	Deleted
A-415	Radiation Zones Plant Elevation 386'0"	Deleted
A-416	Radiation Zones Plant Elevation 404'0" and 422'0"	Deleted
C-110	Reactor Building Structure Penetration Details (P-1)	Deleted
C-111	Reactor Building Structure Penetration Details (P-2)	Deleted
C-134	Reactor Building Reinforcing Steel Equipment Hatch Details	Deleted
M-002	Equipment Location Fuel Handling Floor Plan	Deleted
M-003	Equipment Location Operating Floor Plan	Deleted
M-004	Equipment Location Intermediate Floor Plan	Deleted
M-005	Equipment Location Ground Floor Plan	Deleted
M-006	Equipment Location Plan Below Grade	Deleted
M-007	Equipment Location Sections A-A and F-F	Deleted
M-008	Equipment Location Section B-B	Deleted
M-009	Equipment Location Section C-C	Deleted
M-010	Equipment Location Section D-D	Deleted
M-011	Equipment Location Misc. Plans and Sections	Deleted
M-200	Instrumentation and Component Symbols	Deleted
M-201	Instrumentation Symbols	Deleted
M-206	MSIV Operator Controls	Deleted

The following SAR Figures are based on the referenced controlled drawings:

Drawing Number	SAR Title	Figure Number
1-MS-2 1-MS-102 1-MS-103 1-MS-104	Large Pipe Isometric From North Steam Generator	Deleted
1-MS-5	Isometric - ADV Line From North Steam Generator	Deleted
1-MS-5	Isometric - ADV Line From North Steam Generator	Deleted
1-MS-4 1-MS-101 1-MS-102 1-MS-105	Large Pipe Isometric From South Steam Generator	Deleted
1-MS-5	Isometric - ADV Line From South Steam Generator	Deleted
1-MS-5 1-MS-118	Isometric - Main Steam to Emergency Feedwater Pump	Deleted

Table 1-5 (continued)

Drawing Number	SAR Title	Figure Number
C-67	Emergency Cooling Reservoir Pipe Intake and Discharge	9-34
C-663, Sht 3,5,6	Emergency Pond Spillway	9-33A,B,C
C-106	Reactor Building Instrumentation for Pressure Proof Test	5-18
C-294	Protective Barrier - Main Steam to Emergency Feedwater Pump	A-10
E-001	Station Single Line Diagram	8-1
E-001	Single Line Diagram 500 KV Switchyard Auxiliary Power	8-9
M-204	Condensate and Feedwater, Emergency Feedwater	10-1
M-204	Emergency Feedwater Initiation and Control System	10-2
M-206	Steam Generator Secondary System	7-22
M-209	Circ Water, Service Water, Fire Water, Condenser Vacuum, Screen Wash, Intake Structure, and Discharge Structure	9-10
M-210	Service Water System	9-6
M-211	Auxiliary Cooling Water System	9-9
M-213	Dirty Liquid and Laundry Radwaste, Containment / Aux Bldg Sumps	11-2
M-214	Clean Liquid Radioactive Waste	11-1
M-215	Gaseous Radioactive Waste	11-3
M-217	Emergency Diesel Generators, Fuel Oil Storage and Starting Air Systems	8-3
M-218	Instrument, Service, and Breathing Air Systems	9-14
M-219	Fire Protection System	9-16
M-230	Reactor Coolant System	4-1
M-231	Makeup and Purification System	9-3
M-232	Decay Heat Removal System	6-1
M-233	Chemical Addition System	9-4
M-234	Intermediate Cooling Water System, Sheets 1 & 2	9-7
M-235	Spent Fuel Cooling System	9-47
M-236	Reactor Building Spray System	6-1
M-237	Sampling Systems	9-5
M-238	CRDM and Misc. Reactor Coolant Pump Connections	7-21
M-261	Reactor Building Ventilation System	5-7
M-264	Reactor Building Penetration Room Ventilation System	6-10
M-502	Equipment Arrangement Plan Main Control Room	7-25
V-6600-M1Q-229	Reactor Protection System	7-1

NOTE: For the latest revision of a drawing see ANO Drawing Control.

Table 1-6

PIPING IDENTIFICATION

Piping classes are designated by a three-letter code. The first letter indicates the primary valve and flange pressure rating, the second letter the type of material and the third letter the code to which the piping is designated.

The designations are as follows:

First Letter	Pressure Rating, psi	First Letter	Pressure Rating, psi
C	1500	H	150
D	900	J	Note 1
E	600	K	Note 1
F	400	L	Note 1
G	300	M	Note 1
		T	Tubing

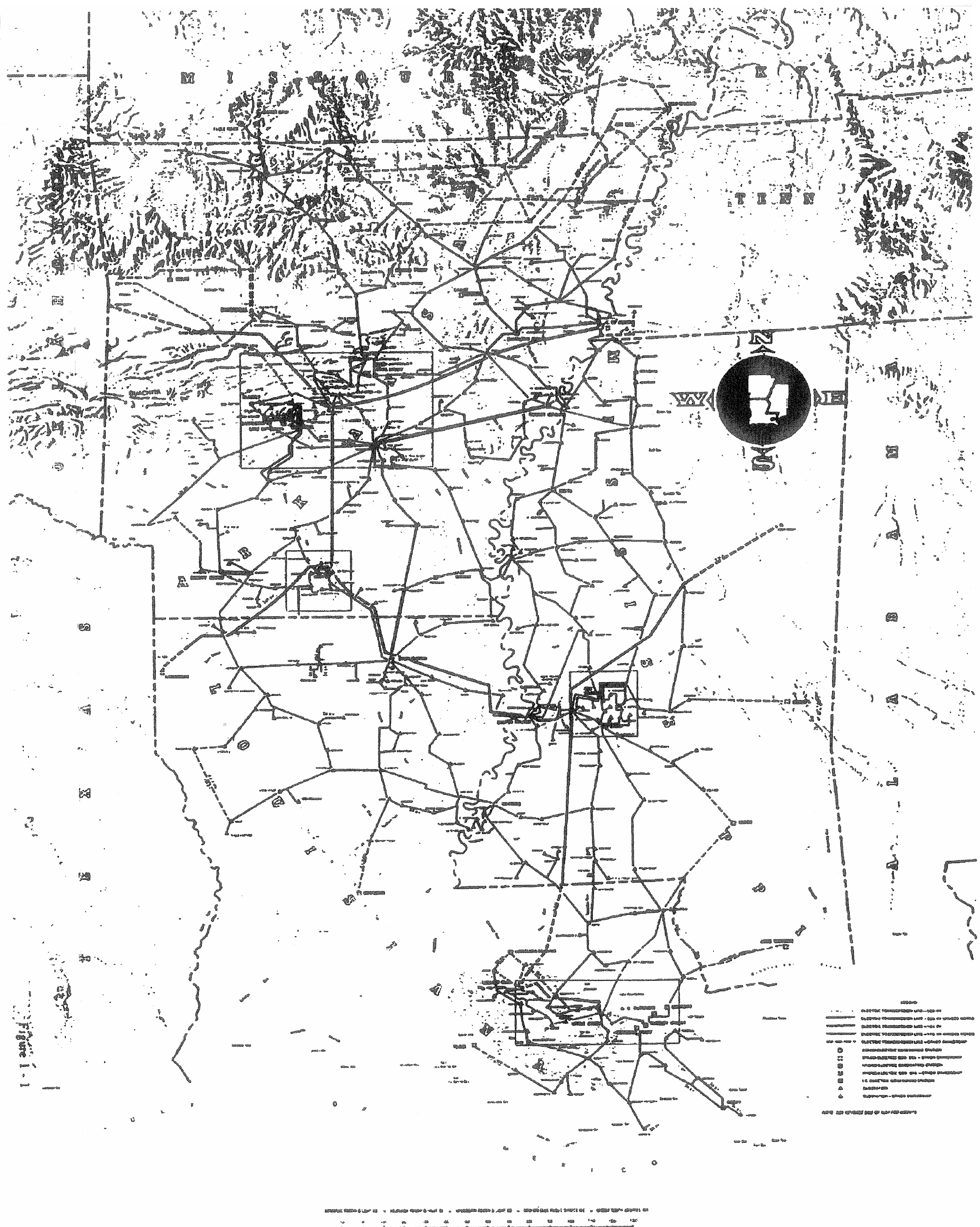
Second Letter	Material	Second Letter	Material
B	Carbon Steel	N	Galvanized Carbon Steel
C	Stainless Steel	R	Carbon Steel Radwaste
D	Copper	S	Stainless Steel Radwaste
E	Note 1		
F	Note 1		
G	Note 1		

Third Letter	Design Code
(See Note 2)	
A	Nuclear Power Piping, ANSI B31.7, Class I
B	Nuclear Power Piping, ANSI B31.7, Class II
C	Nuclear Power Piping, ANSI B31.7, Class III
D	Power Piping Code, ANSI B31.1.0
F	National Fire Protection Association Codes

NOTES:

1. For general use as designated on piping Class sheets.

2. Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.



SAR FIGURE NO. 1-1

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
2.7.6	Design Change Package 82-1047, "Seismic Instrumentation."
Amendment 5	
2.3.2.1.2	Design Change Package 85-2036A, "ANO-1 Meterological Tower Data."
Amendment 9	
2.3.6.2.3	ANO Unit 1 Technical Specifications, Amendment 140.
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2.3.2.1.2	Design Change Package 87-2096, "Dose Assessment System Upgrade."
Amendment 11	
2.3.2.1.2	Condition Report C-92-0023, "Removal of MET Tower Site Printer."
Amendment 19	
Figures 2-1 thru 2-27, 2-29 thru 2-31, 2-36 thru 2-51	License Document Change Request 1-1.1-0008, "Reformatting of Various Drawing Labels"
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2.3.2.1.2	License Document Change Request 1-2.3-0006, "Clarification of Meteorological Tower Instrumentation Location"
2.2.6	License Document Change Request 1-2.2-0003, "Removal of Specified Natural Gas Company Title"
All Figures	License Document Change Request 1-1.1-0009, "Deletion/replacement of Excessive Detailed Drawings from the SAR"
Amendment 21	
2.7.3	License Document Change Request 07-045, "Clarification of OBE and DBE Terms with Respect to License Basis"
Amendment 24	
2.3.2.1.2	License Document Change Request 10-055, "Correction to Met Tower Data Collection System"

UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
2.8.4	Condition Report CR-ANO-1-2012-0353, "Correct SAR Land Use Census Surveillance Frequency"
Amendment 25	
2.7.6	Engineering Change EC-33710, “Seismic Monitoring System Upgrade”
2.7.6	Licensing Basis Document Change LBDC 13-001, “Clarification of Seismic Monitoring Requirements”
2.3.6.2.5 2.3.6.2.6 2.3.6.2.7	Condition Report CR-ANO-C-2011-3015, "Provide Consistent Use of X/Q Term with respect to Routine Gas Releases"
Amendment 26	
Figure 2-32	Licensing Basis Document Change LBDC 14-031, "Add Detail to OSFSFs and LLRW Building"
Figure 2-32	“Condition Report CR-ANO-C-2014-1356, "Add Detail of Outside Radioactive Storage Facilities”
2.3.2.1.2	Engineering Change EC-46343, "Fire Protection for Expanded Protected Area"
Amendment 27	
2.4.4.3	Engineering Change EC-57218, "External Flooding Protection Update"

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2 SITE AND ENVIRONMENT

Sections of this chapter that do not describe the facilities at ANO are not encompassed by the FSAR update rule (10 CFR 50.71(e)) and therefore are not required to be updated. This includes such information as projected population densities, land use surveys, and groundwater users. This information was developed in support of initial licensing and, even if updated, would be unlikely to affect plant operation, the plant design bases, or the conclusions of safety analyses relative to public health and safety.

2.1 GENERAL DESCRIPTION

Data are presented in this section as a basis for the selection of design criteria for Arkansas Nuclear One - Unit 1 and to verify the adequacy of concepts for controlling routing and accidental release of radioactive effluents to the environment. A series of studies (meteorology, ground and surface water hydrology, geology, seismology, population, and land use) have been conducted.

The site is located in Pope County, Arkansas, about six miles west-northwest of Russellville, Arkansas.

Cooling water for Unit 1 will be drawn from and returned to the Dardanelle Reservoir.

The exclusion area around the reactor has a minimum radius of 0.65 miles. The boundary of the Low Population Zone (LPZ) lies at a 4-mile radius from the reactor building. There will be seven population centers with a projected population of 25,000 or more in the year 2012 within a 100-mile radius of the plant site.

The principal structures are founded on rock and the foundation materials have presented no special problems in either design or construction.

2.2 SITE AND ADJACENT AREAS

2.2.1 LOCATION

The Arkansas Nuclear One - Unit 1 plant is located in southwestern Pope County, Arkansas, as shown in Figure 2-1. The site is located at latitude 35°-18'-36"N and longitude 93°-13'-53"W.

2.2.2 LAND OWNERSHIP

The property within the 0.65 mile radius of the nuclear station will be controlled to the extent necessary by Entergy Operations, Inc. This area includes certain portions of the bed and banks of the Dardanelle Reservoir, which are owned by the United States Government. An easement has been obtained which will allow Entergy Operations, Inc. to exclude all persons from these areas during periods of emergency. The property boundary (shown in Figure 2-2) will be posted and a perimeter fence will be erected around the immediate station area.

2.2.3 VICINITY

The plant site is located about two miles southeast of the village of London, Arkansas on a peninsula formed by the Dardanelle Reservoir. The reservoir is part of the "Multiple Purpose Improvement Plan for the Arkansas River" and includes the Arkansas River and the former Illinois Bayou. The area of the reservoir is 36,600 acres; its normal pool elevation (338 feet) is controlled downstream by the Dardanelle Lock and Dam No. 10 on the Arkansas River.

2.2.4 RESIDENT POPULATION

Population centers within a 100-mile radius of the site are shown on Figure 2-4. Figures shown are from the 1960 Census. The largest city, Little Rock, Arkansas, is located 57 miles southeast of the site, and had a 1960 population of 128,929. The nearest population center, Hot Springs, Arkansas, is 55 miles south of the site and had a 1960 population of 37,286.

Figures 2-5, 2-6, and 2-7 show the estimated residential 1967 and 2012 population by sectors (16) at varying radii from the plant site. The estimated permanent population within the 4-mile radius Low Population Zone (LPZ) is 2,264 for 1970 and 3,709 for 2012.

The 1960 census showed mostly decreasing population in this area from 1950 to 1960. According to the "Arkansas River Region Comprehensive Development Plan 1980" (prepared by Associated Planners, Inc. for the Arkansas Planning Commission), this trend reversed and the population increased from 1960 to 1965. The 1970 population projection is a linear extrapolation of the 1960-1965 trend based on the data of the "Arkansas River Region Comprehensive Development Plan 1980." In some peripheral areas, which are not covered by the Development Plan, linear projections were used from the 1950 and 1960 census if the population increased. If the population decreased between 1950 and 1960, the 1960 census data were used without adjustment. These 1970 estimates are the latest available since the 1970 census will not be published until 1971.

The 2012 projection is taken from the Arkansas River Region Report (Volume I - Section B, prepared by Industrial Research and Extension Center, College of Business Administration, University of Arkansas). The increase used in the 1967-2012 projection is an average of the high and low estimates of this report.

2.2.5 PART-TIME POPULATION

It is expected that the Dardanelle Reservoir will be a major contributing factor to part-time population within a 5-mile radius. The recreational activities offered by the reservoir, such as boating, fishing, camping, etc., will increase the population of the area during the summer months. It is anticipated that the 75 miles (approximately) of shoreline of the Dardanelle Reservoir and Arkansas River will be developed, and weekend or holiday population will increase. Figure 2-8 shows the estimated population within the 4-mile radius LPZ is 1,049 for 1970 and 1,991 for 2012. The 1970 projection is a linear interpolation taken from the 4-mile radius on Figure 2-8.

2.2.6 LAND USE

Figure 2-9 shows the area of land in square miles that is pastured and cultivated within a 50-mile radius of the site. The land use is shown in 16 directional sectors centered on the site and within 5-, 10-, 20-, 30-, 40-, and 50-mile radii.

The land data were based on the 1964 United States Census of Agriculture Preliminary Report for Arkansas.

Figure 2-10 shows the number of dairy animals, by counties, that will be grazing within a 50-mile radius of the site.

There is no major airport with a control tower within 50 miles of the plant site. The closest airports are the Russellville Municipal Airport (8 miles) and the Clarksville Municipal Airport (15 miles). There are two low altitude airways nearby, as shown in Figure 2-11. This airway, V71, whose centerline is about five miles east of the plant site, has one or two scheduled commercial low altitude turbo-prop flights daily.

The centerline of airway V74N is two miles south of the plant site. This is an alternate route of V74, which is the main route between Fort Smith and Little Rock, 20 miles south of the plant. V74N carries only very light, unscheduled traffic.

The closest high altitude (jet) airway is Route J6-14/279 about 30 miles south of the plant site.

A natural gas transmission line crosses the plant site property (see Figure 2-2). The existing pipe is 10-3/4 inch O.D. operating at 500 psig. It is coated with coal tar enamel and is under cathodic protection. The pipe has been rerouted to pass under the discharge canal with four feet of earth cover. There is a minimum of 600 feet between the pipe and the reactor building. Consideration has been given to the possible safety hazards that the pipe represents to the plant: explosive rupture, ignition of discharged gas and effects of gas itself were examined. It has been concluded that the proximity of the gas line represents no safety hazard to the safe operation of the plant.

2.3 METEOROLOGY

2.3.1 GENERAL

This section presents the available data and the analysis thereof in regard to diffusion climatology for the Arkansas Nuclear One site. A detailed study has been made of all the available onsite meteorological data from September 1967 through May 1970. This study includes atmospheric data from September 1967 through May 1970; and atmospheric stability category frequency distributions, relative concentration values, and frequency distributions. These data are used to develop conservative diffusion models for this site applicable for use in estimating radiation dosage in normal or abnormal plant operations.

The climate of the Arkansas River valley in the region of the site is primarily continental in character. The Boston Mountains, with elevations up to 2,700 feet and oriented generally east-west on the north side of the valley, have an influence on the annual precipitation. The annual precipitation on the south slope is of the order of two to four inches greater than in the valley. Within the valley, in an east-west direction, the climatology is homogeneous. Directly to the south, hills range up to 300 feet. Snowfall in the region is of small consequence in winter. The mean temperature at Russellville, Arkansas, in July is 82 °F and in January 42 °F. The range in mean daily maximum and minimum temperature is 94 °F and 70 °F in July and 53 °F and 30 °F in January. The highest and lowest temperature recorded is 113 °F and 50 °F in July and 82 °F and -15 °F in January.

From 33 months of onsite wind data values of relative concentrations, X/Q sec/m^3, were calculated for two hours at the site boundary and low population zone and two to 24 hours and one to 30 days at the low population zone.

2.3.2 SITE LOCATION

The site of Arkansas Nuclear One is adjacent to Dardanelle Reservoir on the Arkansas River.

Physically, the site is centrally situated on a "peninsula," about two miles wide and two miles long, that extends into the Dardanelle Reservoir. On three sides, the site is surrounded by reservoir water; the shortest stretch of which is approximately one mile in a southeast direction. Generally, the site peninsula is at an elevation of about 400 feet, but with rises above 500 feet. Ground surface in the immediate vicinity of the plant site is predominantly meadow.

To the north of the site, the land mass gradually ascends to 1,000 feet altitude at a distance of about 15 miles in the Boston Mountains. The maximum height of 2,700 feet is 41 miles north-northwest of the site. Generally, the Arkansas River follows along the base of the Boston Mountains. The higher portions of the mountains are located west northwest to east northeast of the site.

To the south and west of the site location, across the Arkansas River and Dardanelle Reservoir, another range of hills extends from a direction of due south to due west. Directly south is Mount Nebo, elevation 1,880 feet at a distance of about eight miles. Further to the west-southwest at about 25 miles is Magazine Mountain at 3,042 feet altitude, the highest point in the state.

To the east, and extending to the south southeast, the land area is moderately level, interspersed with rolling hills frequently covered with woods.

2.3.2.1 Site Meteorological Program Description

The onsite meteorological program is the principal source of data used in this report. This program consists of two separate studies. One is called the Early Program and one is called the Tower Program. The wind data portions of these two programs have been combined to yield one record of two years and nine months of continuous data. The analyses of the data yield documentation of onsite meteorological diffusion conditions. Figure 2-35 shows the locations of the wind instrumentation on the site.

2.3.2.1.1 Early Program

The location of the Early Program wind system is approximately 0.4 miles east-northeast of the power plant. This system is a single B&W Model WS-101 wind speed and direction sensor, set atop a 30-foot wooden pole at a ground elevation of approximately 380 feet. Continuous data are recorded on strip charts.

The data are read directly from the recorder charts. Spot wind speed and direction are taken at the hour. Thirty-minute direction variation, or range, is taken from 15 minutes before to 15 minutes after the hour. Chart readings are made on a 3-hour basis.

The Early Program was operated from September, 1967 through May, 1969. It also was operated during October and November, 1969 to permit the correlation with the Tower Program data.

2.3.2.1.2 Tower Program

A 190-foot high instrumented meteorology tower was also placed in operation at the site. The tower is located at latitude 35°18'N and longitude 93°13'20"W at a position 0.51 miles due east of the reactor building. The tower is about 800 feet southeast of the Early Program pole and is placed at a ground elevation of approximately 360 feet.

The tower instrumentation includes sensors to measure wind direction, wind speed, air temperature and differential temperatures, and the occurrence of precipitation. The wind sensors are located at the 20-foot and the 190-foot levels. The temperature sensors are located at the 5-, 85-, and 190-foot levels. The precipitation sensor is located at the 20-foot level.

The wind direction instruments are Litton Model 510D-2 sensors utilizing Model 510V-4 vanes. The wind speed instruments are Litton Model 511S-4 sensors utilizing Model 510C-2 stainless steel cup assemblies. The temperature sensors utilize precision resistance bulbs mounted in ventilated solar radiation shields. The precipitation sensor consists of two printed circuit grids and the occurrence of precipitation reduces the electrical impedance of the circuit.

The sensor outputs are recorded on analog strip charts housed in a meteorological tower site building. This building is kept at constant temperature by air conditioner/heat pump equipment.

The Tower Program operated from June, 1969 through May, 1970. Data from the charts were analyzed by Litton similar to the method described for the Early Program, but with added analyses for precipitation and temperature/temperature difference. The two programs provide a two year, nine month record of site meteorological data.

In March 1986, the original 190-foot high instrumented meteorological tower, primary sensors, signal conditioners, and control room recorders were replaced by an upgraded meteorological system. This system consists of a 197-foot high meteorological tower with sensors at 10 meters and 57 meters, lightning protection (for the tower, sensor input signal lines, signal conditioning AC power, and the data lines between the tower site and the control room), signal conditioning electronics, a 2.5 KVA liquid propane fueled backup generator with auto start/auto transfer capability, a dedicated line FSK transmitter/receiver pair, new control room recorders, and data acquisition system.

The 197-foot high tower is located at approximately the same location as the original 190-foot high tower and is serviced by the same tower site building. Mounted on the tower is an electric winch driven sensor elevator which provides for automatically raising and/or lowering of the sensor carriage assemblies to ground level for servicing of the sensors. Parameters measured at each tower level are: (a) wind direction (0° to 540°), (b) wind speed (0 to 100 mph), (c) ambient temperature (-50 °C to +50 °C), and (d) dew point (-50 °C to +50 °C).

The signal conditioning electronics are comprised of signal conditioners for: (a) each level wind direction, (b) each level wind speed, (c) each level dew point, (d) lower level reference ambient temperature, (e) delta temperature between levels, and (f) standard deviation of upper level wind direction (sigma theta). Liquid crystal displays of each of these parameters in engineering units were provided and are located along with the signal conditioners and FSK transmitter at the tower site building. The signals from the signal conditioners are digitized, multiplexed, and transmitted back to the Unit 1 control room where they are de-multiplexed and converted back to analog signals to be sent to the recorders, as well as Radiological Dose Assessment Computer System (RDACS) via the SPDS.

The sensors, signal converter electronics, FSK transmitter, and data acquisition system, are all on the backup power system. The fuel supply sizing has been selected to provide at least eight (8) hours of continuous use in the event of loss of off-site power. The existing uninterruptible power supply is used in conjunction with the generator to provide for battery backup power during the transition from off-site to generator power. Indication of transfer to backup power is provided on the front panel of the meteorological cabinet located in the Unit 1 control room.

The data acquisition system provides for transfer of the meteorological data to a remote location for data reduction. Data validation is also provided.

In 2008, meteorological data storage was moved to the Plant Data Server (PDS). Data validation is performed using data from the PDS system.

2.3.2.1.3 <u>Observation Record</u>

A record of the monthly percent of possible observations of the weather instrument systems at the site is shown in Table 2-11. Comments relating to this table follow.

The 30-foot wind system is the Early Program instrumentation. Two major periods of data loss occurred, one from April through July, 1968, and one from March through May, 1969. During these periods, the electrical contact on the direction sensor operated only intermittently and eventually was replaced. During January, 1968, extended periods of ice storms resulted in a frozen vane for periods up to 36 hours. This also occurred for periods in November 1968 and in January-February, 1969 for periods of two to five days. Other outages were random intervals of

operations problems. These consisted of various difficulties such as recorder drive not wound (one week in April, 1969), jammed paper, faulty pens, lack of ink, loss of power to the site, etc. There was no distinctive diurnal pattern to these outages.

The 20-foot wind system is on the meteorological tower. The first 4½ months of operation of this system were plagued by integrated circuit board malfunctions, inferior electrical contacts, and photo-cell failure. To some extent, these same problems persisted throughout the year of system operation. The record shows that when data are missing due to these causes, large blocks of consecutive hours or days of data are lost. Also, for this system, some random loss of data was experienced due to operations problems as described for the 30-foot system.

The temperature sensors positioned at the top and the midpoint of the meteorological tower, with differential reference to the temperature sensor at the 5-foot tower level, suffered loss of data principally due to paper jamming or tearing, pen sticking, ink flooding or clogging and power outage. Some trouble was experienced on the temperature difference to 190-foot level due to integrated circuit board signal problems. The loss of data for all these events occurred principally as consecutive periods of days, until normal operation was restored.

2.3.3 GENERAL SITE CLIMATOLOGY

2.3.3.1 Surface Temperatures

Monthly mean and annual temperatures for the towns along the Arkansas River valley between Little Rock and Fort Smith are given in Table 2-12. The towns are Conway, Morrilton, Russellville, and Ozark. There is a marked similarity in mean temperature among the towns along this 130 mile-long reach of the Arkansas River (See Reference 1).

At Little Rock, the highest temperature recorded during the past 77 years was 110 °F on August 10, 1936, and the lowest was 13 °F below zero on February 12, 1899. July has the highest monthly mean temperature of 81.99 °F, and January the lowest monthly mean of 41.8 °F. Temperatures of 90 °F or higher occur on a average of 54 days during the year. Temperatures of "zero" or below have been recorded only 14 times during the past 77 years. The normal number of heating degree days per year is 2,982. The greatest monthly normal is 719 degree days in January.

The Little Rock area enjoys a relatively long growing season. The average number of days between the last killing frost or freeze in the spring, and the first in the fall is 242. The latest recorded killing frost in the spring was April 26, 1920, and the earliest in the fall on October 22, 1898.

At Russellville, annual mean daily maximum temperature is 73 °F; the minimum is 49.4 °F. The highest temperature on record is 113 °F, and the lowest is -15 °F. Russellville has 92 annual mean number of days with a temperature equal to or greater than 90 °F and 74 days equal to or less than 32 °F.

At Fort Smith, the highest temperature ever recorded was 113 °F; the lowest was 15 °F below zero.

The climatological survey for the town of Ozark (see Reference 2) describes temperature in the region. The Boston Mountains in the northern sections of the county provide a slight barrier for cold air penetration from the north in the winter. In the summer, the opposite effect is noticed. Year around, monthly temperatures in Franklin County averaged two to three degrees higher than Arkansas counties immediately to the north in the Ozark plateau.

2.3.3.2 Precipitation

2.3.3.2.1 Rain

The mean monthly and annual precipitation (mostly rain) data from the towns between Little Rock and Fort Smith are also shown in Table 2-12. The results show roughly similar rainfall characteristics from Little Rock to Russellville, with rainfall gradually diminishing thereafter to Fort Smith. In all towns, September rains are markedly similar. Later in the fall, rain is again less in the upper river region near Fort Smith.

The mean annual precipitation for Little Rock is 48.66 inches. It occurs principally in the form of rain and is well distributed throughout the year. The wettest month is May with a normal rainfall of 5.28 inches, and the driest is August with a normal of 2.82 inches. The greatest monthly total ever recorded was 18.04 inches in January, 1937. No measurable precipitation was recorded in June, 1952. The greatest amount in any 24 hours was 9.58 inches on April 8-9, 1913. There is an average of 106 days with measurable precipitation during the year and an average of 59 thunderstorms occur during the year.

During the past 63 years, Little Rock has received an average of 62 percent of the possible sunshine. During the last 77 years, there has been an average of 136 clear days, 108 partly cloudy days, and 121 cloudy days during the year. The least cloudy month is October and the cloudiest is January. Heavy fog occurs on an average of 10 days per year.

At Fort Smith, the greatest yearly rainfall was 61.81 inches; the least was 19.80 inches. The average annual rainfall was 42.2 inches.

The climatological survey for Ozark, Franklin County, (see Reference 2) is considered appropriate to the Russellville site. The information which follows is extracted from this survey. Rainfall is ample for farming. Late spring and early summer is the wettest period. Monthly totals drop off by almost 50 percent from May to August. Annual precipitation extremes range from around 23 inches to as much as nearly 80 inches.

Springtime rain is dependable. There is a 90 percent probability that May will produce almost three inches of rain or more. General rains of a heavy and widespread nature are the result of frontal passages. This time of the year, strongly contrasting air masses are to be found over Arkansas, particularly in the northwest counties. Single storm totals of five inches or more are not uncommon.

There is a 10 percent chance of any fall or early winter month producing under one inch of rainfall. The summertime convective or afternoon heating variety of thunderstorms are a very unreliable source of good general rains. Even during a dry spell the state can be dominated by humid air from the Gulf of Mexico that results in widely scattered afternoon showers.

Russellville weather records (see Reference 1) indicate, for seven years of observations, 70 mean number of days with precipitation equal or greater than 0.1 inch and 32 mean number of days with precipitation equal or greater than 0.50 inch.

2.3.3.2.2 Snow

It is believed that the description of occurrence of snow shown in the climatological survey for Ozark, Franklin County (see Reference 2) will apply also to the power plant site. This survey states that by more northern standards, snow is a poor moisture source. The average annual snowfall represents less than one inch of water content. The snowfall at the City of Ozark averaged 3.7 inches per year. The record year for snow was 1921, when some sections of Franklin County reported 20 inches of snow. The annual snowfall at Russellville for 10 years of record, 1951 through 1960, is 4.1 inches (see Reference 1).

Snowfall is not of major importance in the Little Rock area. During the past 72 years, there has been an annual average of 4.5 inches. The greatest monthly total on record is 19.4 inches in January, 1918, and the heaviest 24-hour fall was 13 inches on January 2, 1893. Snow does not generally remain on the ground for an extended period of time and seldom causes serious inconvenience.

2.3.3.3 Winds

2.3.3.3.1 Correlation of Early Program and Tower Program Data

As previously described, the meteorological study utilized two sets of wind instruments at different locations and heights. These are called the Early Program and the Tower Program. The two sets of data from these programs are offset in time except for one 2-month period of overlap. To permit full use of the entire period of record of two years and two months, a correlation of the data from the two programs for the overlap period is required.

The correlation study results in a set of rules to be applied to the Tower Program data so that the complete record of site data is homogeneous with respect to height and location. The adjustments permit development of site wind statistics and diffusion models with uniform data. Figure 2-35 shows the location of the Early Program and Tower Program instrumentation. The Early Program pole is located about 800 feet northwest of the tower. Both instrument sites have similar exposure characteristics. Surface cover is grass with an occasional grove of trees. There appears to be no naturally occurring obstructions which might cause the wind flow to be unrepresentative of the general area.

The wind speed characteristics of the two systems are quite similar, but the Early Program wind direction vane is expected to exhibit a smaller amount of wind direction fluctuation than the Tower Program system. This difference is not critical in the analysis.

Table 2-13 shows the sector frequency and mean speed for the two wind systems. To correlate wind speed, an analysis was made by the least squares method using coincident occurrences of data from each system. This was used to establish a linear regression equation connecting the 20-foot wind to the 30-foot wind during the period of overlap. The equation of this line is:

$$U_{30} = 1.7 + 0.96\, U_{20} \qquad (1)$$

where:

U_{30} is the wind speed at 30 feet, kt

U_{20} is the wind speed at 20 feet, kt

This equation is used to convert all the 20-foot wind data from the tower program to appear as if it came from 30 feet so that these data can be combined with the Early Program data.

To correlate wind direction, a comparison of the wind rose statistics for the period of overlap was made. No particular differences in direction for the two sets of data were noted except for direction differences in the northwest quadrant between the two systems. This may be influenced by the instrument shack adjacent to the tower.

The frequencies from the west-northwest and west are higher for the 20-foot system than for the 30-foot system. However, if the frequencies for north-northwest, northwest, west-northwest, and west are added, about the same total is found. It is also noted that the frequencies are rather uniformly distributed over these four directions in the Early Program data.

The procedure adopted to correlate the direction data was to leave the 20-foot wind direction frequencies the same except for the four directions identified as west, west-northwest, northwest and north-northwest. For these four directions, the frequencies were summed and then equally distributed over these four directions.

To correlate the stability category data, Table 2-14 was prepared. This table shows the coincident Percent Occurrence of Stability Categories using identical methods of analyses on the wind direction traces from 20-foot and 30-foot data. The significant differences, from the standpoint of affecting probability levels associated with diffusion models developed for the site, occur in the total columns. At 20 feet there is more "D" and less "E" than at 30 feet; and more nighttime nonsteady categories and less "F" at 20 feet.

This difference is consistent with the more sensitive (to fluctuations) wind vane at 20 feet and to the presence of the instrument building near the tower.

Although a rather complex probability surface could be devised to describe the relationship between the 20-foot and the 30-foot stability categories in Table 2-14, it is sufficiently accurate to assign every fifth 20-foot "D" as an "E" and every other nighttime nonsteady condition at 20 feet as an "F" when analyzing the data from the tower. This will result in statistics quite compatible with the Early Program data.

The cumulative probabilities of X/Q, relative concentration, are the important objective in the diffusion model development. The above adjustments will permit this calculation with uniform data and with reasonable accuracy for the entire sampling period.

2.3.3.3.2 <u>Surface Winds</u>

The following presentations of surface wind data contain the combined correlated 3-hourly record of the Early Program and the Tower Program.

Seasonal wind rose data are shown in Table 2-15. Complete wind roses are shown in Figures 2-36 to 2-39. Summer and fall have lower wind speeds than the winter and spring season although the differences are not large. In computing the seasonal mean speeds, calm is included at zero knots. The dominant directional frequency of all seasons except summer is associated with an east wind and ranges seasonably from 14.1 to 19.0 percent. In summer, the wind is predominately from the south with a frequency of 12.0 percent and from the east and southeast with a combined frequency of 19.9 percent. The dominant easterly frequency along with a smaller secondary maximum of frequency from the west has caused channeling of low level winds by the Arkansas River Valley. In all seasons, the data indicate that the speed of the westerly winds is systematically higher than the speed of the easterly winds.

These seasonal characteristics of the wind roses carry over to the annual wind rose summary, Table 2-16. See Figure 2-40 for annual wind rose. These data indicate that the annual mean speed including calm at zero knots, is 4.3 knots, that the frequency of calms is 7.6 percent, and that the distribution with azimuth is affected by channeling. As will become evident later, this distribution due to channeling leads to a relative concentration, X/Q, rose that is favorable in regard to the low population zone boundary.

Table 2-17 shows the annual percent frequency of occurrence of wind speed and direction when precipitation occurs. The wind speed classes are taken directly from the 20-foot level of the tower and have not been correlated to appear as if they came from the 30-foot level of the Early Program. This table shows that better than one-half of the time precipitation is associated with winds of one to three knots. Precipitation is associated most frequently with winds from the east or east-southeast.

Table 2-18 shows the annual percent frequency of wind direction and mean speed in knots for the 190-foot level of the Tower Program. Figure 2-41 shows the 190-foot wind rose. The table shows that for winds aloft, the amount of calm (0.4 percent) is greatly reduced and the mean wind speed (6.9 kt) is increased as compared to the surface winds. The directional frequency shows the effect of dominant winter winds from the west and spring and summer winds from the south. Local topography effects are not as pronounced at 190 feet as at the surface. Wind speeds generally are higher with the westerly winds and are the highest with winds from the northwest sector. Most winds (66.0 percent) are found between 4-10 knots in speed.

2.3.3.3.3 Strong Winds and Thunderstorms

Strong winds and storms can and do occur in the Arkansas River Valley. The mean annual number of days with thunderstorms (see Reference 3) at Russellville is about 55. At Fort Smith, the highest wind speed recorded in 21 years (see Reference 4) was 58 mph from the north in June, 1951. In Little Rock, the highest wind speed recorded in 19 years (see Reference 5) was 61 mph from the northwest in May, 1952.

2.3.3.3.4 Tornadoes

Tornadoes have occurred in the Arkansas River Valley. For the 35 years of record (see Reference 6), 1916 to 1950, 11 tornadoes were observed in Pope County; three were observed in the vicinity of the site.

About 40 tornadoes were observed in the 46-year period of 1916 through 1961 in Franklin County and the adjoining six counties of west central Arkansas (see Reference 2).

In the whole state of Arkansas, in the years 1953 to 1963, 189 tornadoes (see Reference 7) were observed. The site lies within an area of frequency of 50-100 tornadoes (see Reference 8) during the 40-year period of 1916 to 1955. In addition, the site is within an area reporting 42 tornadoes per 1-degree square between 1916-1961 (see Reference 9).

2.3.4 STABILITY WIND CATEGORIES

2.3.4.1 General

Over the last 30 to 40 years, many investigators have worked, experimentally and theoretically, on the problem of atmospheric diffusion. See Sutton (see Reference 51), Pasquill (see Reference 44), and Meteorology and Atomic Energy (see Reference 50) for review.

Gifford (see Reference 22) has suggested modifying the usual plume axis prediction equation of previous investigators to:

$$\frac{X}{Q} = \frac{1}{(\pi\ \sigma_y\ \sigma_z + cA)\ \overline{u}} \qquad (2)$$

assuming a reflection boundary condition

where:

X = concentration, units/m^3

Q = release rate, units/sec

σ_y = crosswind standard deviation, m

σ_z = vertical standard deviation, m

$\overline{u}$ = mean wind speed, m/sec

c = an empirical constant

A = cross-sectional area of building complexes from which source is being emitted, m^2

Equation 2 accounts for the increased dilution in the wake of a building. The constant c has to be determined experimentally. Values of 1/2 to 2 are widely quoted but extrapolation of the best available data (see References 18 & 31) to about 1000 meters yield values between 1 and 3. In this study the conservative value of c = 1/2 was used in Equation 2. This is the equation used for the 2-hour models developed in this study.

For long period releases, such as two to 24 hours and one to 30 days, the use of Equation 2 would be overly conservative due to plume meander. Therefore, to develop models for these periods, the relative concentration for a surface source is averaged over a 22.5° wind sector width. This is given by Equation 3 modified from Meteorology and Atomic Energy (see Reference 50).

$$X = \frac{(2/\pi)^{1/2}\ Qf}{\sigma_z\ \overline{u}(2\pi x/n)} \qquad (3)$$

where:

X = the sector averaged concentration units/m^3

Q = the average release rate, units/sec

σ_z = vertical standard deviation, m

$\overline{u}$ = mean wind speed, m/sec

x = distance downwind, m

n = number of wind direction sectors in a circle

f = fraction of the time that a particular wind direction occurs

Of all the meteorological parameters measured in various diffusion tests, i.e. σ_e, σ_u, ΔT, Ri and $\overline{u}$, Slade (see Reference 50) found that a measure of the horizontal wind direction fluctuation (σ_θ) was the best for the construction of a set of diagrams summarizing diffusion data under different stability conditions. The diagrams resemble those proposed by Pasquill (see Reference 45) and the Pasquill categories can be related to σ_θ. The relationships between the Pasquill categories and sigma theta are given in Table 2-19.

In order to obtain σ_θ, Slade (see Reference 49) suggests taking a 30-minute period of analog chart data, determining the range in wind direction, and dividing this range by six to obtain σ_θ. Six standard deviations account for 99.6 percent of the directional variability. Slade's procedures were followed to calculate σ_θ for 30-minute periods centered on the hour with data from the Early Program onsite wind data at 30 feet above the ground.

Once σ_θ has been determined, the stability category was named according to Pasquill's categories using the relationships in Table 2-19.

In processing of the data for the entire period of record, a separate account was taken of the nonsteady winds. These are not directly amenable to stability categorization by the method of Slade. These nonsteady data amounted to 7.5 percent of data sample. An adjustment was therefore made as suggested by Slade. When the nonsteady wind occurs during the night, stability Category E was assigned. When the nonsteady wind occurs during the day, stability Category B was assigned. This adjustment is believed to be conservative.

2.3.4.2 Summary of Pasquill Stability Categories

Table 2-20 shows a summary of Pasquill Stability Categories and mean wind speed for the seasons. These are also shown in Figures 2-42 through 2-45 for each season. Generally, summer is the season of low frequencies of occurrence of Categories E and F and winter the season of most Categories of E and F. The mean wind speed in winter for Categories E and F is stronger than in the other seasons. This is true also for the seasonal total mean wind speeds.

Table 2-21 shows the Annual Percent Frequency of Pasquill stability category and mean wind speed versus direction. This is also shown in Figure 2-46. Category E occurs 28.4 percent of the time with a mean speed of 2.3 m/sec and Category F occurs 7.1 percent of the time with a mean speed of 1.2 m/sec. The unstable categories (A, B and C) occur 36.5 percent of the time and their mean speed is 1.6 m/sec. When Category F occurs, the data suggest that there is a 0.34 probability the wind is directed up the river away from the Russellville quadrant, and only a 0.18 probability that the wind would be toward Russellville. Since Russellville is the key population group in the low population zone, this site characteristic is favorable.

2.3.4.3 Temperature Difference Correlation With Stability Category

As part of the Tower Program, temperature difference measurements are obtained between five and 190 feet above grade. Wind speed and direction measurements are available at the 20-foot level. It is of interest to investigate, for this site, the relationships between full-tower temperature gradients and wind direction fluctuations at the 20-foot level indices at atmospheric stability.

The change of temperature with height, the lapse rate, is a measure of atmospheric stability. A lapse rate of 1 °C/100 m for dry adiabatic conditions occurs with neutral conditions. Non-neutral lapse rates are a function of height above ground. The relationship between the lapse rate and the diffusive properties of the atmosphere are site dependent. Therefore, it is not possible at the Arkansas Nuclear One site to use relationships between temperature difference and atmospheric diffusive properties developed for other locations. It is possible to examine the gross stability properties of the site using both methods. This is shown in Table 2-22.

The temperature difference categories used in Table 2-22 are arbitrarily chosen to define a small range near neutral with stable and unstable on either side. The significant points from this table are as follows:

A. All Slade categories occur under all temperature gradient categories.

B. All categories, A through F, occur in the stable temperature gradient category. However, if a broader neutral temperature gradient range had been used, many of the Cs and Ds would have fallen into the neutral range.

C. Excluding the nonsteady data, the unstable category plus one-half of the neutral category occurred 15.3 percent of the time, and Categories A through C occurred 18.9 percent of the time. Excluding the nonsteady data, the stable plus one-half of the neutral category occurred 55.5 percent of the time whereas Categories D through F occurred 61.3 percent of the time.

If the diffusion parameters σ_y and σ_z were used for Categories A and B under stable conditions, the resulting σ_z would be too large. The data in Table 2-22 indicate that this could happen 2.6 percent of the time. Conversely, the use of the diffusion parameters for Categories E and F under unstable conditions could result in too small a σ_z. This could happen 2.4% of the time.

It is concluded that the Slade stability category type of analysis is consistent with a stability analysis based on temperature data and that any errors in the calculated relative concentration probabilities caused by Categories A and B occurring under stable conditions will be compensated by the occurrence of Categories E and F under unstable conditions. Since the site diffusion models are based on relative concentration probability distributions, specifically those distributions under stable conditions, it can be concluded from the table that the probability distributions based on Slade's method are conservative.

2.3.4.4 Wind Direction and Mean Speed Correlation With Temperature Difference Stability Category

Summaries have been prepared of the annual percent frequency of wind direction and mean speed for various temperature difference (full tower, five to 190 ft) categories, "a" through "g". These are shown in Tables 2-23 through 2-29. The temperature difference stability categories, as requested and defined by the DRL staff, are shown below.

Stability Category	Temperature Difference, °C/100 m
a	≤-1.9
b	> -1.9 to -1.7
c	> -1.7 to -1.5
d	> -1.5 to -0.5
e	> -0.5 to +1.5
f	> +1.5 to +4.0
g	> +4.0

Lower case letters are used above as no data have been found which uniquely relates these temperature gradients to the Pasquill-Gifford diffusion categories, A through F.

In preparing Tables 2-23 through 2-29, the hourly data from the 20-foot level of the Tower Program, adjusted as described in Section 2.3.3.3.1, were used. In acquiring the temperature difference data from the charts, an average time of 10 minutes was used for each hourly value.

Salient features of Tables 2-23 through 2-29 are discussed below.

A. The majority of the total calms (7.2 percent out of a weighted mean of 10.4 percent) occur under stability category "g". This contributes significantly to the low value of the "g" category mean wind speed because calms are included as zero speed in calculating the mean speed.

B. The total occurrence of stable atmospheric conditions appears to be physically correct because of the correct definition of category "d", neutral.

C. The wind directional preferences are for winds from the west-northwest and east under all stability categories. Under category "a", there is a directional preference for west-northwest winds which is interpreted to be associated with post-frontal convective activity. Under categories "e", "f", and "g", there is a pronounced preference for occurrence of east winds relative to west-northwest winds. The east winds are away from Russellville.

It should be noted that the diffusion conditions associated with Pasquill-Gifford Category G have been obtained by pure extrapolation, as pointed out by Yanskey, Markee, and Richter (see Reference 57). In addition, the relationship between the diffusion characteristics of the atmosphere and any temperature gradient data are site dependent. These site dependent characteristics are included in the following analysis of wind direction fluctuations.

This is illustrated by a comparison of coincident Pasquill-Gifford stability categories and temperature difference stability categories for the full tower shown in Table 2-30. The Pasquill-Gifford stability categories were determined by the method proposed by Slade (Reference 49). The temperature difference stability categories are as previously defined.

In preparing Table 2-30, the hourly data from the 20-foot level of the Meteorology Tower was used. These data were not adjusted to be consistent with the Early Program data.

The salient features of Table 2-30 follow.

A. The percentage frequency of categories "e" through "g" are consistent with the stable categories shown in Table 2-22.

B. The stability category "g" occurs predominately under diffusion conditions which are neutral or non-steady at night. This is generally true of category "f" also.

C. It should be noted that Pasquill-Gifford Categories A and B are associated with temperature gradient categories "e", "f", and "g" 3.3 percent of the time, whereas Pasquill-Gifford Categories E and F are associated with temperature gradient categories "a" and "b" 2.4 percent of the time. These features are roughly compensating in their affect on X/Q probability distributions.

D. Under Pasquill-Gifford Category F, there is about an equal probability of better or worse categories based on the temperature gradient categorizations. Because of this feature the X/Q values used in the 2-hour model are not biased by the influence of temperature gradient considerations on the stability categories.

Because of the site location on a small peninsula, surrounded on three sides by water, in a valley, one would expect a tendency for colder temperatures in the lower layers of the atmosphere than would otherwise exist. This would bias temperature gradient stability categories toward stable categories. On the other hand, the use of wind direction fluctuation statistics for stability category characterization reflects the effect of mechanical turbulence and the variability of the winds at this site.

Additional temperature stability category analyses were prepared using temperature differences between the upper half of the tower, 85 to 190 feet, and the 20-foot adjusted wind data. These analyses are shown in Tables 2-31 through 2-37. Because temperature gradients decrease with height above the ground, one would expect less "a" and less "g" from these data than from the full tower data. This is shown in comparing Table 2-23 with 10.5 percent "a" to Table 2-31 with 8.8 percent "a". The comparison is even more significant in comparing Table 2-29 with 29.7 percent "g" with Table 2-37 with 9.4 percent "g". These data illustrate the difficulty in using a fixed temperature gradient characterization of atmospheric stability without mention of applicable heights.

The other salient features of Tables 2-31 through 2-37 have been pointed out previously, i.e., under stable conditions, "e" through "f", there is a high preference for winds from the east and east-southeast which would carry any releases away from Russellville. Calms occur preferentially under "f" and "g". This latter situation makes the mean speed low under "f" and "g", since calm is counted as zero speed and included in computing the mean speed.

As pointed out in Table 2-30, many of the "f" and "g" categories are associated with neutral or nonsteady conditions as determined by an analysis of the 20-foot wind direction fluctuation data.

2.3.5 SITE DIFFUSION MODELS

2.3.5.1 General

For the Arkansas Nuclear One site, the distance to the exclusion distance is 1,046 meters, and to the low population zone is 6,436 meters. These are the distances used to obtain the horizontal and vertical standard deviation of the plume (see Reference 49). Equation 2 is used for the 0-2 hour model and Equation 3 is used to calculate the sector-averaged concentration for the two to 24 hour model, the one to 30 day model, and the annual model. In the cavity dilution term of Equation 2, a conservative value of c = 1/2 is used. The cross-sectional area of only the reactor building part of the power plant complex is used in calculating the cavity dilution term. Thus, A = 2205 m^2.

The following paragraphs discuss the method of calculating the models and the results.

2.3.5.2 Two-Hour Diffusion Model, Exclusion Distance

From 33 months of onsite surface wind data, 3-hourly values of relative concentration, X/Q, were calculated at the exclusion distance. The data was taken from the wind speed and wind direction trace. Stability was determined using Slade's method. Equation 2 was used with cA = 0. The values of σ_y and σ_z are from Reference 49. The values of X/Q are given as a function of wind direction in Table 2-38. The total frequencies of occurrence of each X/Q class are plotted in a cumulative fashion on probability paper. This plot is given as Figure 2-47.

The value of X/Q which is not exceeded more than five percent of the time, with calms excluded, was selected from Figure 2-47 as 2.5×10^{-4} sec/m³. Values of σ_v and σ_z were selected (midway between those for stability Categories E and F at this distance) along with the associated wind speed which is consistent with the above X/Q. These values for σ_y and σ_z were inserted into Equation 2 along with c = 1/2 and A = 2205 m^2. The value of c = 1/2 is conservative because observations indicate values between 1 and 3. For A, the cross-sectional area of the reactor building only is used. This is also conservative because the total building cross-sectional area is 3,955 m^2 including the contiguous auxiliary building.

Because any leaking material from the reactor building would be vertically distributed over the 194-foot height of the reactor building in the downwind cavity, it is appropriate to modify the wind speed, consistent with the surface wind X/Q analysis in Figure 2-47, to account for the wind speed increase with height. The value of X/Q was adjusted to take into account the ratio of the mean layer wind speed to the surface wind speed.

The use of the cavity dilution and wind speed adjustments to the values of X/Q given at the exclusion distance which will not be exceeded more than five percent of the time in Figure 2-47 gives a 2-hour model X/Q = 1.2×10^{-4} sec/m^3. The effective cavity dilution factor is a conservative 1.4.

2.3.5.3 Two-Hour Diffusion Model, LPZ

The development of the 2-hour model for the Low Population Zone (LPZ) distance of 6,436 meters used the same data and procedures as above. The only exception is that at this distance the effect of the cavity dilution term is negligible. Table 2-39 gives X/Q probabilities as a function of wind direction for the 6,436 meter distance. The cumulative totals excluding calm are plotted on Figure 2-48. From this a value of X/Q = 1.5×10^{-5} sec/m³ which is not exceeded more than five percent of the time with calms excluded is given.

The modification of this X/Q value by the ratio of mean wind speed surface (190 feet to surface wind speed), gives a 2-hour diffusion model at the LPZ, X/Q = 1.2×10^{-5} sec/m^3.

2.3.5.4 Two to 24-Hour Diffusion Model, LPZ

For the two to 24 hour diffusion model at the LPZ, the X/Q for the 0-2 hour LPZ model is simply converted from Equation 2 (maximum centerline concentration but without cavity dilution) to Equation 3 (plume meander in a 22.5° sector). The maximum wind direction frequency factor is 0.146, from the east, taken from Table 2-16. The resultant conversion factor is 0.0348. The relative concentration, X/Q, for the two to 24 hour model is 4.2×10^{-7} sec/m^3. The probability of this relative concentration occurring is 0.73 percent.

Regarding the basis for sector averaging, it is pointed out that the two to 24 hour LPZ model is based on the 0-2 hour exclusion distance value of X/Q that has only a five percent chance of being exceeded. This value of X/Q is modified for range and sector averaging. It should be noted that values of X/Q exceeding the 0-2 hour value of X/Q are associated with light wind conditions at the site. Light winds are not normally those associated with great persistence of wind direction. Such plume axes would be perfectly coincident for three hours during the two to 24 hour period. To require this additional conservatism on top of a five percent X/Q value is regarded as unreasonable.

An analysis was made showing the persistence of the wind at 20 feet as a function of direction. This shows the following hours of persistence versus percentage frequency.

Persistence of Hours	Percent Frequency
2	58.8
3	22.3
4	8.7
5	3.7
6 & more	6.5

This shows that the chances of a wind in a given direction during the 2-hour model being followed by a wind in the same direction for the three hours immediately following is 6.5 percent. If the case of three scattered hours in the two to 24 hour period were considered, it is not likely that these three "hourly" plumes would be superimposed perfectly. For these reasons, sector averaging in the two to 24 hour model period has been adopted.

2.3.5.5 One to 30-Day Diffusion Model, LPZ

From the same data described for the 0-2 hour model for the LPZ, a value of X/Q associated with the 50 percent probability level on Figure 2-48 was selected. This was 0.25×10^{-5} sec/m³. This value is converted from Equation 2 (maximum centerline concentration but without cavity dilution effects) to Equation 3 (sector averaged). The maximum wind direction frequency factor is 0.146. The resultant conversion factor is 0.0593. Also, the value is modified by the ratio of mean layer speed to surface wind speed. The resultant one to 30 day model relative concentration, X/Q, is 1.1×10^{-7} sec/m³ taken at the LPZ. There is a 7.3 percent probability that this relative concentration will be exceeded on an annual basis.

Regarding the basis for use of the 50 percent occurrence value of X/Q, the dose contribution is about 25 percent of the total calculated long-term dose. If a value of X/Q that is exceeded only 20 percent of the time were used, then this contribution to the dose (from the one to 30-day period) would double. However, the total dose estimate would change by only 25 percent. Hence, it would appear that the LPZ long-term dose is not very sensitive to the value of X/Q used in this model, and it is reasonable to use a value of X/Q with average expectation of occurrence.

2.3.5.6 Annual Diffusion Model, Site Boundary

From Figure 2-47, basis of the 0-2 hour model at the exclusion distance, a value of X/Q at 50 percent probability of occurrence was selected. This was 0.43×10^{-4} sec/m^3. This value was converted from Equation 2 (with cavity dilution effect of 1.1) to Equation 3, using the frequency factor of 0.146. The resultant conversion factor is 0.0697. The value is also modified by the ratio of mean layer speed to surface wind speed. The final annual model relative concentration, X/Q, is 2.1×10^{-6} sec/m³ at the exclusion distance.

2.3.6 METEOROLOGICAL ANALYSES FOR THE YEAR JULY 29, 1971 TO JULY 29, 1972

The meteorological instruments were relocated to conform to current meteorological practices on July 29, 1971. Analyses of the new data and estimates of the atmospheric dilution qualities for both accident and routine operations are provided in the following sections.

Instruments were located on the meteorological tower as follows:

Wind speed and direction at 40 feet and at 190 feet above grade.

Temperature difference measurements between 30 feet and 85 feet and between 30 feet and 190 feet. The former difference span was used herein as a backup in the event the 30 to 190 foot difference was not observed due to instrument malfunction.

Hourly average values of wind speed, direction, temperature difference and temperature were read off the strip charts for this year of record.

2.3.6.1 Data Recovery

2.3.6.1.1 Wind Data

When analysis of the data was initiated, it became apparent that the wind speed measurement instrument at the 40-foot level was malfunctioning. To determine the nature of the malfunction, an MRI 1071 wind instrument was mounted on the tower below the suspect 40-foot Litton Industries instrument. A comparison of concurrent values of wind speed showed that the suspect instrument was defective and that the wind speeds recorded by it were below those actually occurring as recorded by the MRI 1071 instrument. The latter instrument, which counts revolutions of the anemometer cups in an hour through a mechanical linkage, can only malfunction by slowing down, thereby recording lower than actual winds. Further, by means of the Litton Industries wind direction recordings, it was possible to define those hours when actual winds were above the direction vane threshold, and then compare these times with those intervals when the suspect wind speed recordings were below their speed threshold. This comparison confirmed that the 40-foot level Litton wind speed instrument was malfunctioning on July 29, 1971 and thereafter. It was, therefore, decided to use the 190-foot wind speed data for the analysis, reducing these winds to those appropriate to the 40-foot level by means of the familiar Power Law formula;

where:

$$U_{40} = \frac{(40)P}{190} U_{190} \qquad (4)$$

and where U_{40} and U_{190} are wind speeds at 40 and 190 feet above grade and where P is a function of the stability class and is defined with locally observed wind speeds at the two levels by regression analyses.

Using a 60-day sample of concurrent 40-foot wind speeds observed with the MRI 1071 instrument and the 190-foot speeds observed with the valid 190-foot Litton Industries instrument, the following exponents were defined for each stability class, using the tower 30- to 190-foot temperature differences to define Pasquill classes.

Pasquill Classes	Best Fit Power Law Exponent	Number of Wind Speed Pairs in Sample
A,B,C,D	0.18	650
E	0.225	296
F,G	0.45	325

For the period July 29, 1971 to February 7, 1972, the 190-foot wind speeds and directions are used in all analyses, with the speeds reduced to those appropriate to the 40-foot level by the Power Law. For the period February 7 to July 29, 1972, the wind data is from the MRI 1071 instrument mounted just below the 40-foot level on the tower. The observed 40- and 190-foot wind data for these periods are shown in Tables 2-40 and 2-41, respectively. Because of difficulties in data reduction, the width of the wind sectors is 20° for all sectors except N, S, E, and W, which are 30°. Calculations using sector spread equations recognized these differences.

At the end of the period of record running from February 7, 1972 through July 1972, the MRI 1071 wind instrument, which was placed on the weather tower just below the defective 40-foot level Litton Industries instrument, was returned to the MRI factory for calibration and maintenance. Their report shows that this instrument was within calibration standards. It is concluded that the readings from this instrument were accurate and provide the proper basis for assessment of the meteorological conditions at the site during this period.

The report from Meteorology Research, Inc. is quoted as follows:

> "On Thursday, September 14, 1972, the above referenced Mechanical Weather Station was functionally tested in the MRI Wind Tunnel to verify its operation in accordance with the standard specifications. The following report indicates the various parameters tested.
>
> "Wind Direction - Starting threshold, delay distance and damping ratio were all found to be within the original manufactured tolerances, and met the specifications.
>
> "Wind Run - With the chart drive operating, the tunnel was set as 15 miles per hour and the station chart record checked for a 1-1/2 hour run. The overall accuracy proved to be well within one percent of a ±2 percent standard. Starting threshold was less than 0.75 miles per hour again meeting MRI's specifications.
>
> "Temperature - Temperature records were checked at several times during the period of testing and the chart markings found to be within our relative accuracy of +1 °F.
>
> "In all tests, the Mechanical Weather Station was found to be within the published MRI Specifications for accuracy. All tests were completed on the station as it arrived at our facility."

This report is signed by Keith H. Storey, dated September 26, 1972.

2.3.6.1.2 Data Recovery Percentages for the Year

Using the 30- to 85-foot delta temperature data as backup for the 30- to 190-foot data, and the 190-foot wind speeds reduced to 40-foot level as backup for the 40-foot wind speeds, data recovery is better than 90 percent as shown in Table 2-42. Sustained periods of missing data are quite limited as shown in Table 2-43.

2.3.6.1.3 Results of the New Data Program

Relocation of the temperature sensors and wind sensors and readjustment of data reduction to hourly average from spot readings, both in accordance with current meteorological practice for diffusion calculations, caused significant changes in the frequencies of the stable Pasquill classes as shown in Table 2-44.

The frequency of Pasquill G observations is lower by a factor of over 3 and Pasquill F by a factor of about 1.5 due to the relocation of the lower thermal sensor.

The frequencies of calm winds were also reduced by a factor of approximately 3 because of the change to reading hourly average speed values instead of instantaneous values.

The representativeness of the onsite data record running from August 1971 through July 1972 has been investigated by a comparison of Little Rock, Arkansas data for this year with that for the 5-year period 1966 through 1970.

The Pasquill-Turner method was used to determine the Pasquill Class conditions in both data sets, and the concurrent wind speed Pasquill Classes were examined. The year August 1971 through July 1972 experienced higher frequencies of low wind speeds during stable conditions than is normal as illustrated by Table 2-46. This data has been extracted from the basic data reproduced in Table 2-47. The long-term dilution qualities at the Arkansas Nuclear site should, therefore, be more favorable than the conditions observed during August 1971 through July 1972.

2.3.6.2 Diffusion Calculations

2.3.6.2.1 Two-Hour Model, Exclusion Distance

Relative concentrations were calculated at the exclusion distance of 1,046 meters using the standard centerline invariant wind formula with a wake factor cA = 1,103 square meters, the temperature difference Pasquill class boundaries given in Safety Guide 23, and the concurrent wind Pasquill class data given in Tables 2-40 and 2-41. Values were ranked and placed in cumulative frequency distributions as shown in Figures 2-49 and 2-50. The relative concentration value, which is exceeded only five percent of the full year, is at 6.5×10^{-4} sec/m^3. The difference between the five percentile values of Figure 2-49 and 2-50 is not due to the fact that wind data for Figure 2-49 were reduced by the Power Law. Rather, this difference is due to the higher frequency of very light winds in the spring, as confirmed by Table 2-15, with the independent data sample from the period September 1967 through May 1969.

2.3.6.2.2 Eight-Hour Model, LPZ

Relative concentrations were calculated at the low population distance of 6,436 meters and ranked in a cumulative frequency distribution using the invariant centerline wind equations, but ignoring the wake factor, as shown in Figure 2-51. The relative concentration value, which is exceeded only five percent of the time, is 9.7×10^{-5} sec/m^3.

2.3.6.2.3 8-24 Hour, 1-4 Day, 4-30 Day Models at the LPZ

The relative concentrations at the LPZ for evaluation of accidental conditions beyond eight hours are the highest values which occurred in a sector during the year of data averaged over the period using the standard sector spread equations given in Safety Guide 4. Thus, for evaluation of the relative concentrations appropriate to the 8th to the 24th hour of an accidental release, the relative concentration for each hour was calculated during the year. All possible consecutive 16-hour averages for the full year in each sector were determined and the highest 16-hour average was selected from among all 16-hour averages in all sectors. Values are, therefore, the highest which occurred in the year, and apply only to the sector of occurrence. The maximum 16-hour average concentration values occurring in the other 15 sectors are less than those cited herein.

The maximum average values appropriate for each of the longer duration accident conditions are as follows:

Sector Affected	Maximum Average Relative Concentration sec/m^3	Time Interval of Average	Interval After the Initiation of Accidental Release
W	9.78×10^{-6}	16 hours	8 - 24 hours
NW	3.868×10^{-6}	3 days	1 - 4 days
W	1.107×10^{-6}	26 days	4 - 30 days

NRC Safety Evaluation Report (SER) dated June 6, 1973 provided the values listed in Table 14-49, which have been used since that time.

2.3.6.2.4 Average Annual Relative Concentrations

Average annual concentrations were calculated using the sector spread equations of the Safety Guide 4 for each sector and pertinent distances from the containment building, as shown in Table 2-45.

The average value for all sectors is 4.87×10^{-6} sec/m^3 at the exclusion distance of 1,046 meters, and 2.82×10^{-7} sec/m^3 at the low population distance of 6,436 meters. Calm winds were included in these averages by the conservative assumption of a wind speed of 0.2 mph equally distributed in all directions. Because certain routine releases are subject to operational control, it is appropriate to consider average annual concentrations with calm winds omitted. These average values are 4.14×10^{-6} sec/m^3 at the exclusion distance and 2.38×10^{-7} sec/m^3 at the edge of the LPZ.

2.3.6.2.5 Average Annual Release Rate Limit

The design objective in the ODCM relating to the average annual release rate of noble gases and other radioactive isotopes, except I-131 and particulate radioisotopes with half-lives greater than eight days, discharged from the plant provides that the resulting annual dose at the site boundary should be less than 10 mrem to the whole body or any organ of an individual.

Using the methodology of Regulatory Guide 1.111, the highest annual average X/Q value for a ground level release is 2.0×10^{-5} sec/m^3 in the west-southwest sector at a distance of 0.65 miles. This dispersion factor (X/Q) will be used to calculate the annual dose at the site boundary. Dose calculations are based on the dose models in the Standard Radiological Technical Specifications (NUREG-0472).

2.3.6.2.6 Quarterly Release Rate Limit

The maximum allowable quarterly release rate of gross gaseous activity may not exceed eight times the maximum allowable release rate averaged over any 1-hour period. This method of calculating the maximum allowable quarterly release rate is based on the highest annual average X/Q value of 2.0×10^{-5} sec/m^3, and the dose models in NUREG-0472.

2.3.6.2.7 Hourly Release Rate Limit

The ODCM limits the release rate of radioactive materials and gaseous wastes (except I-131 and particulate radioisotopes with half-lives greater than eight days) when averaged over 1-hour period to that rate which would result in an off-site concentration equal to the maximum permissible concentration as defined in 10 CFR 20, Appendix B, Table II, Column 1, . These calculations will be based on the highest annual average X/Q value of 2.0×10^{-5} sec/m^3 consistent with the X/Q value used in calculating annual and quarterly limits.

2.4 SURFACE WATER HYDROLOGY

2.4.1 PURPOSE AND SCOPE

In connection with the safety aspects of the nuclear power plant presently under construction, surface water investigations were made to determine the availability and dependability of a cooling water supply, the magnitude of possible floods, and the possible failure of upstream dams.

2.4.2 WATER SUPPLY

The plant site is on a peninsula formed by the former Illinois Bayou to the east and the Dardanelle Reservoir and Arkansas River to the south and west. The surrounding water (normal pool elevation 338 feet) is controlled downstream by the Dardanelle Lock and Dam No. 10 and upstream by Ozark Lock and Dam No. 11.

Unit 1 has a once-through cooling system that requires 1,700 cfs of cooling water to condense the steam during normal operation.

2.4.2.1 Operation of Dardanelle Reservoir

Dardanelle Reservoir is part of the Arkansas River navigation project. The project will provide a minimum 9-foot navigation depth from the mouth of the Arkansas River at the Mississippi River to Catoosa, Oklahoma, near Tulsa, on the Verdigris River, a distance of more than 500 miles. There are 17 locks and dams in the system. Thirteen of these are simple locks and dams, providing navigation lifts of 30 feet or less. Dardanelle, Ozark, Robert S. Kerr, and Webbers Falls Dams (in order from downstream to upstream) are higher, with lifts up to 54 feet, and include some storage for hydropower generation.

Upstream from the head of navigation, there are seven large multi-purpose reservoirs. These reservoirs control the flow from about 140,000 square miles of drainage area with about 12,000,000 acre-feet of storage, of which, about 6,000,000 acre-feet are reserved for flood control.

Dardanelle Reservoir is 259 miles upstream from the mouth. A navigation lift of 54 feet raises shipping to the top of the pool at an elevation of 338 feet. The minimum navigation pool elevation is 336 feet, providing a normal two feet of storage in the reservoir for power generation. Power generation is on the basis of mean daily inflow equaling mean daily outflow, within the 336-338 feet limits.

2.4.2.2 Historical Records of Flows

Daily stream flow records for the period of January 1923 to September 1957, collected at the Dardanelle gauging station just below the dam, have been adjusted by the Corps of Engineers to reproduce flows as they would have been regulated by the complete system of dams upstream. The minimum daily average flow as computed by the Corps of Engineers was 4,000 cfs during the driest critical month of the year.

2.4.2.3 Low Flow

It is possible for the inflow to the reservoir to be zero under very exceptional circumstances, but these conditions would exist for only a few hours, during which time there would be more than enough water in storage in the reservoir to supply the consumptive use of the plant.

2.4.3 LAKES AND RESERVOIRS IN VICINITY

Lakes and reservoirs with a surface area of 100 acres or more within a 50-mile radius of the site, and details on ownership, location, use, and size of these bodies of water, are shown in Figure 2-12. The Arkansas River is used by the City of Russellville as a supplementary source of drinking water during periods of extremely low flow of the Illinois Bayou.

2.4.4 FLOODS

The highest flood experienced at the Dardanelle Dam site occurred in 1943, with a peak flow of 683,000 cfs. Dardanelle Dam is designed to hold a water level no higher than 338 feet and to discharge 900,000 cfs. The levees along the river channel in this area are designed for flows of 830,000 cfs.

2.4.4.1 Maximum Probable Flood

(see SAR Section 2.11)

2.4.4.2 Failure of Upstream Dams

(see SAR Section 2.11)

2.4.4.3 Design Flood Elevation

(see SAR Section 2.11)

2.5 GROUNDWATER HYDROLOGY

2.5.1 GENERAL GEOLOGY

The site is located in an area where clay and silty clay deposits overlie bedrock which consists of Pennsylvanian McAlester formation shale. The thickness of this clayey overburden varies from about 13 to 24 feet in the vicinity. The bedrock sequence forms the trough of the east-west trending Scranton syncline. Hard, fine grained sandstone of the Harthshorne formation was encountered during drilling at a depth of about 150 feet.

2.5.2 GROUNDWATER MOVEMENT

The piezometric surface slopes about 24 feet per mile southwest toward the lake as shown on Figure 2-13. Domestic wells located down groundwater slope from the plant site extend into bedrock; therefore, any contaminated water accidentally spilled at the plant will migrate slowly through relatively impermeable clay toward the lake and have no affect on water supplies from the artesian bedrock aquifer. The estimated rate of migration through the overburden is 0.1 to 1 foot per year.

2.5.3 GROUNDWATER SUPPLY

The only use of groundwater in the vicinity of the site is for local domestic purposes. Good groundwater bearing zones are not present in the overburden material at or near the site. Limited supplies are pumped from joint systems in the shale and sandstone bedrock aquifers. Most wells drilled into these bedrock artesian water producing zones are less than 150 feet in depth except for the Bunker Hill area. These wells are capable of only relatively low yields and a "good" well may produce up to 50 gpm.

Shallow domestic wells in the general vicinity are located up groundwater slope from the plant site; therefore, contamination from the plant is not possible. Deeper wells are not subject to contamination from the plant due to the presence of an impervious cap over the artesian aquifer. This aquiclude, or impervious cap, is the upper portion of the shale zone which acts as a confining media preventing the upward flow of the confined water. This same impervious material would prevent the downward percolation of any surface water from the plant area.

Piezometers bottomed in the overburden at the plant site were dry five days after installation, but a nearby piezometer drilled into bedrock encountered confined water which rose to within 9.5 feet of the ground surface.

As shown on Figure 2-13, a profile of the piezometric surface indicates that the recently filled Dardanelle Reservoir and the pre-reservoir water table have not reached equilibrium. This is apparently due to the low permeability of the soil cover over the bedrock.

2.5.4 WATER QUALITY

Ten samples of groundwater, two samples from Dardanelle Reservoir water, and one of Arkansas River water from downstream of the dam (located about nine miles southeast of the site) were sent to a laboratory for chemical analysis. Refer to Figure 2-14 for groundwater sample location and to Table 2-1 which summarizes the results of the chemical analyses.

Portions of the above mentioned surface water samples were sent to Trapelo-West Laboratories in Richmond, California (formerly Tracerlab) for gross beta analyses. These tests yielded values varying from 18.0 to 27.9 picoCuries per liter indicating normal surface water background radioactivity.

The quality of the river and reservoir water is generally chemically superior to about 50 percent of the groundwater samples. These surface water samples have pH values of 7.4 and average total dissolved solids of 432 parts per million. The groundwater from the sampled wells, all from the bedrock artesian system, varies from acidic to alkaline; from a high bicarbonate content of 444 ppm to a low of 5 ppm, and from a high of 1,559 ppm total dissolved solids to a low of 34 ppm. Total hardness ranges between 4 and 830 ppm.

Due to the wide range of quality of groundwater from the artesian aquifer, if groundwater development for potable use is considered, careful well site selection is required. Treatment of water to reduce mineral content from some wells sampled would be required to obtain water which is within U.S. Department of Health water quality standards. A preliminary evaluation of the quality of the sampled wells indicated that a zone (Zone A) extending between Water Wells No. 6 and 3, north and east of the plant site, produces the best quality water. The average total dissolved solids of the samples from this zone is 245 ppm (excluding the low 34 ppm analysis). However, two wells within this zone have some undesirable constituents. Well No. 2 has an iron (Fe) content of 1.3 ppm, which is slightly above the recommended maximum of 0.3 ppm. Wells No. 4 and 5 have manganese (Mn) contents of 0.3 and 4.0 ppm which are above the recommended drinking water maximum of 0.05 ppm. Also, it should be noted that Wells No. 8 and 9, located in Zone B about 2,500 feet west of the site area, are undesirable either due to nitrate (NO_3) content or due to amount of total dissolved solids.

The two sample wells in the vicinity of the site (No. 1 and 7) produce alkaline bicarbonate water with total dissolved solids of 773 and 635, respectively, as compared with the recommended maximum of 500 ppm. In all other respects, this water is chemically satisfactory for human consumption.

Thus, the samples indicate that potable water of suitable quality can be obtained from the vicinity of the site or from Zone A located about 3,000 feet north and east from the site area. The cost of piping the low quantity required for potable water from Zone A would be low, thus the primary consideration for a groundwater supply is the comparison of the total dissolved solids of 245 ppm (Zone A) vs. 700 ppm (near site).

The exact depth of production of the better quality wells is not known; however, the field study indicated that the wells should be about 150 feet in depth.

The previous discussion has considered only the quality of the water. It should be noted that the field study indicated that the domestic wells in the area are capable of only low productivity.

2.5.4.1 Well Survey

A survey of the wells within a 3-mile radius of Arkansas Nuclear One was conducted and the results are shown in Table 2-9 and Figure 2-31.

2.5.5 MIGRATION OF RADIOACTIVE IONS

As has been previously stated, the thick clayey cover over the bedrock precludes contamination of the bedrock groundwater system. The possibility of contamination of the ground surface and migration of contaminants to the lake is very remote because of the extremely low permeability of the clayey overburden at the site and the affinity of the radionuclides in solution for this clay. These factors should negate any significant or long distance travel of contaminated water.

2.6 GEOLOGY AND FOUNDATION CONDITIONS

2.6.1 INTRODUCTION

This portion of the report describes the geologic conditions affecting the design, construction, and operation of Unit 1. The object of the geologic investigations was to determine the physical characteristics of the foundation materials and verify their suitability for supporting the structures.

All soil sampling and core drilling were supervised by a Bechtel geologist. The soil samples were taken and tested by Grubbs Engineers, Inc. of Little Rock, Arkansas, who also did the core drilling in rock. The rock samples were tested in the Bechtel Geologic Laboratory in San Francisco.

2.6.2 SITE DESCRIPTION

The plant is at an elevation of about 353 feet (MSL) on a broad flat saddle from which surface water drains in a generally southerly direction.

Prior to construction, most of the area was covered with brush and scattered scrub trees. All of the area supports vegetation, which is quite dense in the summer and is occasionally practically impenetrable.

Interstate 40 and U.S. 64 run east-west one and one-half miles from the site; all other roads are improved dirt roads with the exception of Arkansas 333 leading from Arkansas 64 to the site which is a hard surfaced road.

2.6.3 REGIONAL GEOLOGY AND STRUCTURE

Physiographically, the site is situated in the center of the Arkansas Valley section of the Ouachita province (see Figure 2-15). The Arkansas Valley is a gently undulating, east-west-trending plain of lowland, 25 to 35 miles wide, that extends generally from Searcy westward to Fort Smith. Many long, sharp ridges and several broad-topped hills rise above the general level of the valley. In most parts of the valley the topography is an expression of the east-west-trending structure. Broad, open synclines are expressed by high, flat-topped mountains; steeply tilted limbs of anticlines and synclines are generally expressed by sharp ridges, some of which are miles long.

The Arkansas Valley lies within the area of outcrop of Paleozoic rocks which occupy essentially the northwestern half of the state. The rocks of the Arkansas Valley are nearly all of sedimentary origin. They include only a few bodies of igneous rock (see Figure 2-16). The beds in the valley consist chiefly of non-fossiliferous shale and sandstone, all of Carboniferous age, and most of them belonging to the lower part of the Pennsylvanian series. The rocks are generally highly carbonaceous and in some places are coal bearing. They contain little or no calcareous material.

Bedrock in the area of the site is sandstone and shales of Pennsylvanian age overlain in places by stiff clay and silty clay which has been eroded from higher elevation bedrock exposures (see Figure 2-17).

The Arkansas Valley section, which is a trough both structurally and topographically, lies between the essentially flat-lying rocks of the Boston Mountains on the north and the complexly folded strata of the Ouachita Mountains on the south. Its structure, therefore, has some of the characteristics of both its bounding regions. Near the northern border of the Arkansas Valley the faulting is normal, the folds are gentle, and most of the dips south of the anticlinal crests are steeper than those north of them. Near the southern border of the province, thrust faulting is common, the folds are pronounced, and the dips north of the axes of the anticlines are steeper than those south of them. The true intermediate structural character of the central part of the Arkansas Valley is shown by the occurrence of both normal and thrust faults in close proximity, by the increase in the number of symmetrical folds, and by the character of the folding itself which is intermediate between the close folding on the south and the gentle open folding on the north. The Arkansas Valley section, unlike the Ouachita section, has been folded down to form a synclinorium.

The nature of the structural features of the Arkansas Valley and their relation to the structure of the adjacent Ouachita and Boston Mountains show that the dominant force in the production of those features was horizontal pressure exerted from the south. It is believed that the folding and thrust faulting developed in post-Paleozoic pre-Cretaceous time due to the subsidence of a land mass to the south which supplied sediments in the place where this land mass once stood. The normal faults were also connected with this episode of deformation (see Reference 17). These are believed to have formed either contemporaneously with folding, before folding, or as a result of tension developed after the period of folding (see References 17 & 29). Some normal faults are also believed to have formed during deposition and subsidence of the Pennsylvanian sediments. In the Arkansas Valley, the southward-dipping faults terminate against northward-dipping normal faults (See Reference 27). The precise dating of the folds and faults of the Arkansas Valley cannot be done with assurance (See Reference 13), but it is known that they are very old geologic structures and probably were formed before Cretaceous time (see Reference 17).

East-west trending folds are mapped in the area of the site. From about three miles north to three miles south of the site are two anticlines and one syncline. From north to south these structures are as follows: London anticline, Scranton syncline, and the Prairie View anticline. The plant site is located on the Scranton syncline in which the maximum dip rarely exceeds 10 degrees except locally where contorted beds may dip as steeply as 20 degrees (see Figure 2-17).

The London anticline extends from about four and one-half miles northwest of the town of London, east to about three miles northwest of the site. This anticline may continue east to connect with the anticline which is mapped northeast of the site to a point three and one-half miles north of Russellville. The south limb of the London anticline is steeper than the north limb (see Reference 40).

The Scranton syncline, also known as the Ouita syncline, extends from northeast of Russellville, west to a point where it probably dies out about five miles northwest of Scranton. Here it is presumably terminated by the Dublin fault (see Reference 27), which is an east-west-trending fault over nine miles west of the site. On the western part of the Scranton syncline the south limb is steeper than the north limb. However, on the eastern portion of the syncline the north limb is steeper than the south limb (see Reference 40).

The Prairie View anticline is mapped from a point three miles southwest of the town of Scranton, east to at least as far as Russellville. This nearly symmetrical anticline is broken by northward-dipping normal faults (see Reference 27).

There are a number of old faults mapped in the vicinity of the Arkansas River, particularly to the northwest and west of the site. These are in the main, east-west-trending faults. Of these, the London and Prairie View faults are the closest to the site (see Figures 2-16 and 2-17). The London fault and an accompanying unnamed small branch fault trend east-west about four miles north of the site. A small fault which branches from the unnamed fault lies about two and one-half miles northeast of the site. The London fault is a high-angle south-dipping normal fault. It is best exposed about eight miles northwest of the site at Big Piney Creek where the western part of the fault ends. At this locality, the fault plane dips 58° south and has an apparent displacement of about 20 feet. The fault is also exposed to the east along Flat Rock Creek and in a drainage ditch along a country road about eight miles from the site. The unnamed east-west-trending fault lies about one mile south of the London fault. The trend of this unnamed fault changes toward the east to generally northwest-southeast and appears to intersect the London fault in the vicinity of the north fork of Mill Creek (see Reference 25). This fault is at least pre-Cretaceous in age.

The Prairie View fault is concealed under alluvium about six miles west of the site. Another six miles to the west, the fault is exposed. It is an east-west-trending high-angle north-dipping normal fault. (see Reference 40). This fault extends to the west as far as three miles northwest of Subiaco. The maximum displacement along the fault is 350 feet two miles west of the town of Prairie View (see Reference 27). Near the western end of the fault there is about 170 feet of maximum displacement and the fault plane dips about 65° northward. About three miles from the western end of the fault is a small branch fault which is downthrown on the southside and is assumed to be dipping south. The Prairie View anticline is faulted by the Prairie View fault and is similar to the faulted anticlines discussed by Hendricks and Parks (see Reference 29). It is their opinion that the faulted anticlines in the Fort Smith district were either faulted then folded, or contemporaneously faulted and folded (see Reference 25). Therefore, the Prairie View fault is at least as old as the folding in this area, which is pre-Cretaceous.

In summary, the only faults within five miles of the site are the Prairie View fault, and the London fault and its branch faults. The closest fault is only a small branch fault and approaches within about two and one-half miles of the site. The last movement of these faults is believed to have occurred prior to the Cretaceous.

2.6.4 SITE INVESTIGATIONS

2.6.4.1 Previous Investigations

In 1967 and 1968 a 2-stage exploration program was carried out. Thirteen diamond core holes in rock and 44 auger holes to rock were drilled at the plant site and surrounding area at that time. The graphic drill hold logs from these first two field investigations are presented in Figures 2-18 and 2-19.

Approximately 20 feet of core was sent to the Bechtel Geologic Laboratory in San Francisco for testing. The results of these tests are summarized in Table 2-2.

In 1968, approximately 7,500 feet of seismic refraction surveying was performed on and near the site. Refer to Figures 2-20 and 2-21 for location and profiles, respectively, of the seismic lines. In general, the seismic determination of bedrock elevations coincides well with the depths to bedrock determined by drilling. The profiles extended the bedrock configuration beyond that determined by drilling and confirmed bedrock relief. Figure 2-20 shows the location and extent of all investigations at the site.

2.6.4.2 Present Investigations

Sub-grade construction and excavation of the plant since the Spring of 1969 have borne out the 1967 and 1968 geologic investigation data.

During June and July of 1970 a detailed subsurface investigation was conducted. Eleven diamond core holes and 24 soil borings were drilled to obtain 692 feet of core samples and 51 undisturbed soil samples. The graphic logs of the core holes are presented on Figures 2-22 and 2-23. All soil samples were retained by Grubbs Consultant Engineers of Little Rock for testing and storage; approximately 20 feet of rock core was sent to the Bechtel Geologic Laboratory in San Francisco for testing. The results of the rock tests are given in Table 2-3.

Figure 2-20 shows the location of all core holes and soil borings from the three stages of field investigations. The elevations and depths to bedrock for all holes in the area of the site are presented in Table 2-4.

2.6.5 SITE GEOLOGY

The site is underlain by 8 to 30 feet of moderate to stiff, plastic, red and tan clay with occasional zones of silty clay, which overlies black, dense, horizontally bedded shale and sandstone of the McAlester formation. The rock is only moderately jointed with the joints varying in dip from 40° to vertical, but dipping predominantly between 50° and 70°. The joints are tight and essentially impervious. Drill water was lost only at the weathered shale contact, and in all holes the return was nearly 100 percent while coring in shale. Examination of fresh shale cores showed a tendency for them to part along bedding planes and to expand slightly upon removal and exposure to air. The data from 93 drill holes indicates that the bedrock surface has low relief and slopes gently to the southwest.

The deepest hole, DH-203, penetrated interbedded shale and sandstone at a depth of 97.0 feet, and dense fine-grained sandstone, apparently of the Hartshorne formation, at 159.5 feet. This sandstone was also found in DH-1 at 149.0 feet and crops out near lake level 3,200 feet east of the site. No evidence of faulting or offset of any of the beds was found during these investigations.

Overlying the unweathered shale bedrock is four to eight feet of highly weathered shale which grades from hard stratified tan clay at the top to very soft, light grey, weathered shale in the lower portion. Standard penetration test in the top of this zone typically required 25 plus blows to penetrate six inches.

2.6.6 FOUNDATION MATERIALS TESTING

2.6.6.1 Bedrock Samples

Twenty-eight samples from core holes DH-201 through DH-211 were tested in the Bechtel Geologic Laboratory in San Francisco. Standard test procedures were used in these physical property determinations. Data from these tests are presented in Table 2-3. Twenty-seven additional samples had been tested during the 1967-1968 exploration program. Table 2-2 summarizes the data obtained in that testing program.

2.6.6.2 Soil Samples

Thirteen 3-inch diameter Shelby tube samples were taken in the areas surrounding the major, heavier units of the plants. Although the major units will be founded on bedrock, lighter structures will be supported on soil in the general site area. The soils found in these three holes (DH-1A, -3A and -5) are typical of the surrounding area.

The testing program utilized standard ASTM procedures except for the consolidation tests which were specified to be performed for values relative to anticipated loading with approximate safety factors. A detailed soil test report has been prepared by Grubbs Consulting Engineers, Inc., Little Rock, Arkansas.

2.6.7 SITE FOUNDATION EVALUATION

The foundations for all Seismic Category 1 structures, except for the structures at the emergency pond as discussed in Section 9.3.2.4, will be shale which is part of the Pennsylvanian McAlester formation. The properties of this material are summarized in Table 2-5.

The seismic velocities indicate a dynamic modulus of elasticity ranging from a minimum of 2.8×10^6 psi to over 5×10^6 psi.

These values are indicative of good foundation rock. From the general investigation it is apparent that no foundation problems should be anticipated. The bearing capacity of the rock is well above the maximum loads of about 12TSF under operating conditions and about 19TSF under earthquake conditions.

The investigation borings indicated that two to seven feet of light grey, weathered shale would be encountered over sound bedrock. Unweathered shale, upon drying, showed a tendency to air slake. For this reason, excavation was delayed until exposed bedrock could be immediately covered with a protective layer of concrete. An inspection by a qualified geologist during excavation work confirmed that geological conditions exposed during final excavation were essentially as anticipated in previous reports. The geological conditions were found to be entirely suitable for Class 1 structures and no unusual problems were encountered during excavation.

2.6.7.1 Stability of Intake and Discharge Canal Slopes

An evaluation of the stability of the intake and discharge canal slopes was made in accordance with the modified Swedish Slip Circle method of analysis, known as the "Method of Slices". The earthquake load used in the analysis was the Maximum Earthquake; therefore, the intake and discharge canal slopes are qualified as Seismic Category 1.

For normal operating conditions a factor of safety of 1.5 was required, and for unusual conditions which may occur rarely, if at all, a factor of safety of 1.0 was considered acceptable.

The results of the analyses demonstrating the stability of the slopes in terms of factors of safety are presented below.

LONG-TERM STABILITY INTAKE AND DISCHARGE CANAL SLOPES

Structure	Excavation Slope	Loading Condition	Minimum Computed Factors of Safety
Intake Canal	2-1/2 to 1	Normal	2.4
Discharge Canal	2-1/2 to 1	Normal	1.8
Intake Canal	2-1/2 to 1	Seismic	1.1
Discharge Canal	2-1/2 to 1	Seismic	1.0
Intake Canal	2-1/2 to 1	Rapid Drawdown	1.5
Discharge Canal	2-1/2 to 1	Rapid Drawdown	1.6

2.7 SEISMOLOGY

2.7.1 GENERAL GEOLOGY

The site is located in an area where clay and silty clay overlie bedrock consisting of Pennsylvanian McAlester formation shale. The thickness of this clayey overburden varies from about 13 to 25 feet in the vicinity of the site. The bedrock sequence forms the trough of the east-west trending Scranton syncline. Hard, fine-grained sandstone of the Harthshorne formation occurs at a depth of about 150 feet beneath the site.

2.7.2 SITE SEISMIC EVALUATION

No active or recent faulting has been mapped in the area of the proposed site. The London and Prairie View faults located five and six miles, respectively, from the site are the closest known faults.

All Seismic Category 1 structures utilize the shale bedrock as a foundation. This rock has good strength properties and will result in no amplification of ground motion from an earthquake. Because of the excellent formation conditions and the recent general quiescence of the area, a low earthquake intensity could be selected. However, considering the New Madrid earthquake, which is the only severe earthquake experienced in the central United States in historic time, an intensity of VII (Maximum Earthquake) has been assigned for the site. This is considered to be conservative and corresponds to a design spectrum of 0.10g for Operating Basis Earthquake (OBE) and 0.20g for Design Basis Earthquake (DBE) (safe shutdown). Note that during the original design phase for ANO-1, the OBE was called the “Design Earthquake.” The DBE (safe shutdown) was called the “Maximum Earthquake.” These alternate terms remain in use in many current site documents. Care must be exercised because the ANO-1 “Design Earthquake” is not equivalent to the DBE or to the “Safe Shutdown Earthquake.”

2.7.3 EARTHQUAKE HISTORY

Figure 2-24 shows the locations of significant earthquakes - those of intensity IV-V (Modified Mercalli Scale - see Table 2-6) or greater - which have occurred within the area bounded by latitudes 33 and 38 degrees north, and longitudes 90 and 96 degrees west. This includes an area within about 150 miles of the site. Earthquakes which occurred outside this area, but were probably felt at the site, are also listed in this study. This includes the famous New Madrid Earthquakes.

2.7.3.1 Significant Earthquakes Between 33°-38°N and 90°-96°W

The significant earthquakes within this area are those of an intensity of IV-V and greater. Each earthquake is shown on Figure 2-24 by a Roman numeral at the epicenter. The Roman numeral indicates the greatest reported intensity for that quake. A list of the epicenters for this map is given in Table 2-7. Most of the epicenters are located by latitude and longitude. These locations are given in the references to denote either field epicenters or instrumentally located epicenters. Many of the older locations are field epicenters, that is, an epicenter was placed at the place or center of the area receiving the greatest effects from the quake. The references used for this study are shown in Figure 2-24 and include References 12, 15, 19, 20, 28, 35, 36, 37, 38, 42, 43, 46, 48, 52, 53, 54, 55 & 56. The first historical quake in the central United States is listed in the references as occurring in l699, so all quakes on this map have occurred since then. The references for this map have been checked for earthquakes occurring through December 1969.

The closest epicenter to the site is 48 miles. This was the October 22, I882 earthquake of maximum intensity VI-VII. The U. S. Coast and Geodetic Survey has approximated the location of this epicenter, but state that it was difficult to locate it accurately because of insufficient reports from the region most affected. Some investigators place this epicenter near El Reno, Oklahoma. In I969 an intensity V quake occurred 50 miles from the site. No other epicenters have occurred within 50 miles of the site.

2.7.3.2 The New Madrid Earthquakes

The earthquakes which have most significantly affected Arkansas and all adjoining states in the Mississippi Valley region occurred near New Madrid, Missouri in 1811 and 1812. These earthquakes consisted of a succession of shocks of varying intensities, beginning December 15, 1811 and lasting throughout the years 1812 and 1813. The maximum intensity of these shocks was estimated to be XII, with a probable magnitude of 8 (Richter).

The nearest points at which systematic attempts were made to record the shocks from the New Madrid earthquake were at Louisville, Kentucky and Cincinnati, Ohio. At Louisville, 250 miles from the epicenter, 1,764 shocks were recorded between December 16, 1811 and May 6, 1812, at which time the recording of shocks ceased. At Cincinnati, 340 miles from the epicenter, 41 periods of shocks were recorded, eight of which occurred between May 5, 1812 and December 12, 1813. The available data indicate that there were at least 1,882 earthquake shocks.

A good description of the area affected is given by Myron L. Fuller in his I912 paper, "The New Madrid Earthquake":

> "The area affected by the New Madrid Earthquake may be subdivided into an area of marked earth disturbances, an area of slight earth disturbances, and an area of tremors only. In the first is included the territory characterized by pronounced earthquake phenomena, such as domes and sunk lands, fissures, sinks, sand blows, large landslides, etc. This district includes the New Madrid region, originally considered a relatively small area, including the villages of New Madrid and Little Prairie (Caruthersville). It is now known, however, to be somewhat larger, extending from a point west of Cairo on the north to the latitude of Memphis on the south, a distance of more than 100 miles, and from Crowley Ridge on the West of Chickasaw Bluffs on the east, a distance of over 50 miles. The total area characterized by disturbance of the type mentioned is from 30,000 to 50,000 square miles.
>
> "In the area of slight earth disturbances will be included districts in which such minor features as the caving of banks, etc. took place. We have records in the narratives of Latrobe[1] and others of the occurrence of such phenomena along the Mississippi and Ohio, while Bradbury[2] records similar disturbances as far down the Mississippi as the mouth of the St. Francis, near Helena. The disappearance of Island 94 near Vicksburg has been described by August Warner[3]. In fact, there is little doubt that such phenomena as caving were prominent northward nearly to Herculaneum, north-eastward to a point beyond the Wabash, and southward at least to the mouth of the Arkansas. Although no records from the White River region have been seen, it was probably included in the area of slight disturbance. It is also possible that the lower Arkansas was affected to some extent.

> "The area of tremors was naturally far more extensive. On the north they are reported to have been felt in "Upper Canada," on the northwest they are reported to have been felt by the Indians in the region of the upper portions of the Missouri country, and in region between the headwaters of the Arkansas and the Missouri, a distance of more than 500 miles from New Madrid. Southwestward the shocks were felt in the Red River settlements and on the Washita River, an equal distance from the center of disturbance. To the south the shock was felt at New Orleans, also 500 miles distant; to the northeast at Detroit, 600 miles away; and to the east at Washington, over 700 miles and at Boston, 1,100 miles distant. A total area of over 1,000,000 square miles, or half that of the entire United States, was so disturbed that the vibrations could be felt without the aid of instruments."

Figure 2-25 lists some of the effects of the New Madrid earthquake, as described by Fuller and other investigators. The references are listed on the figure.

2.7.3.3 Other Earthquakes Probably Felt at the Site

A review of the literature was made to locate other earthquakes which have probably been felt in the area of the site. Table 2-8 presents a chronological list of those earthquakes, their locations by latitude and longitude and a brief description of the extent of their felt areas. Figure 2-26 shows the location of these earthquakes. A Roman numeral indicates the epicenter and the maximum intensity for each shock.

2.7.4 SEISMIC RISK MAP

Figure 2-27 is a seismic risk map for the conterminous United States. The map was made by a group of research geophysicists headed by Dr. S. T. Algermissen of the U. S. Coast and Geodetic Survey and was issued in January, 1969.

The map divides the conterminous United States into four zones: areas where there is thought to be no reasonable expectancy of earthquake damage; areas of expected minor damage; areas where moderate damage could be expected; and, areas where major destructive earthquakes may occur. The site lies well within the zone of expected minor damage.

The zones are based principally on the known distribution of damaging earthquakes, their intensities and geological considerations.

2.7.5 FOUNDATION VS. SITE INTENSITY

Figure 2-28 shows the earthquake frequency versus intensity at Little Rock, Arkansas. Figure 2-29 shows the active teleseismic stations in the area. The historical data indicates that Arkansas is not an area of earthquake centers. However, the effects of distant shocks should be considered, and in particular, those along the Mississippi River north and south of New Madrid. The New Madrid shock, which occurred over 150 years ago, has been considered in evaluating the seismic design of sensitive structures at the site. In this evaluation, the site has several recognized and distinct advantages.

A. The sensitive structures will be founded on unweathered firm shale bedrock.

B. The site is about 220 miles from the New Madrid epicentral zone. According to a mathematical formula relating distance to earthquake shock attenuation, developed by H. Kawasumi of the Earthquake Research Institute of Tokyo, a site intensity of about VI could be anticipated from a New Madrid area earthquake of intensity XII.

C. From Myron Fuller's report it may be estimated that St. Louis, Missouri experienced an intensity of VII during the New Madrid shock. St. Louis is about 60 miles closer to New Madrid than the site and is founded, in part, on thick alluvial deposits overlying shale and limestone bedrock. Consequently, the intensity at the site due to the New Madrid shock was apparently less than VII.

D. The above discussion and the previous paragraphs of this section have demonstrated that assignment of a maximum intensity of VII (MM) for this site is conservative.

2.7.6 TIME-HISTORY ACCELEROGRAPH

Three strong motion triaxial accelerographs, ACS-8001, ACS-2002, and ACS-8003, are mounted on the outside surface of the reactor building wall. These three instruments are placed in the same vertical axis; ACS-8002 at elevation 531 feet, six inches and ACS-8001 and ACS-8003 at elevation 335 feet (top of base slab). The vertical axis of ACS-8002 splits the difference between ACS-8001 and ACS-8003, which are mounted as close together as physically possible such that ACS-8003 provides a redundant backup to ACS-8001. The locations of these instruments represent two key elements in the dynamic model for seismic analysis. Two strong motion triaxial accelerometers, ACS-8001 and ACS-8003, located at the base slab will provide alarms to the Unit 1 control room via the seismic network control center, C529-NCC. One alarm from C529-NCC will be triggered when a setpoint of 0.01g has been exceeded. This alarm will indicate that an earthquake has occurred and the seismic monitoring system is recording seismic data. Another alarm from C529-NCC will be triggered when the pre-determined value of 0.1g, indicating the OBE has been exceeded. Detailed procedures will be included in the plant operating procedures describing the actions to be taken in the event of actuation of these alarms.

In addition to the above instrumentation, strong motion triaxial accelerographs are installed at the following locations:

Instrument Number	
XR - 8007	U1 Spent Fuel Pool
XR - 8009	Intake structure

All Seismic Category 1 structures are founded on rock. Therefore, soil-structure interaction is not significant at this site. The specific instruments to be selected will have a simple design which would allow plant personnel to calibrate and maintain the equipment. This seismic instrumentation system conforms to Safety Guide 12, "Instrumentation for Earthquake," published March 19, 1971.

Following an earthquake, the resulting measurements will be evaluated by qualified seismology and engineering personnel. If the analysis indicates the shock may have caused stresses exceeding design limits specified for OBE to structures or components, specific action will be taken as recommended by the evaluating personnel.

2.7.7 CONSULTANT'S REPORT

The consultant's report on site seismology is Figure 2-30.

2.8 ENVIRONMENTAL MONITORING

2.8.1 GENERAL

The environmental monitoring program has been designed to accomplish three objectives.

The first objective was to determine the existing level of background radioactivity resulting from natural occurrence and global fallout in the Arkansas Nuclear One Plant environs before radioactive materials were delivered to the site. This preoperational phase began approximately one year before nuclear fuel was received at the site and continued until the first nuclear reactor achieved criticality.

During the preoperational phase, procedures were established, methods and techniques were developed, and a continuing review of the program evaluated the suitability and adequacy of the monitoring program.

The second objective has been to identify pathways of potential significance by which radioactive materials attributable to station operation may enter human food, to determine what the concentration factors for isotopes of concern may be in these pathways, and to establish practical ways of monitoring the significant pathways. River fish, milk, and garden vegetables are probably the most significant food material of interest.

The third objective of the environmental monitoring program is to determine the effect of the operation of the nuclear units on the environment.

Significant quantities of radioactive materials should not be released to the environment during the operation of the nuclear units. The monitoring program is designed to demonstrate this and to document compliance with the NRC Branch Technical Position, Revision 1, November 1979, for Regulatory Guide 4.8 "Environmental Technical Specifications for Nuclear Power Plants." Sampling locations are specified in Table 4-1 of the ODCM.

All gaseous radioactive discharges from the Unit 1 plant site will be well within the limits established by 10 CFR 20. To assure meeting these requirements, a radioactive waste processing system is provided. The system will be designed specifically to monitor and control the release of radioactive materials to the atmosphere.

Radioactive liquid releases from Unit 1 will be well within the limits established by 10 CFR 20. Liquids are processed in filters and demineralizers to eliminate nearly all the removable radioactive material prior to release of such liquids from the plant. After such release and dilution in the river, concentrations of specific radioisotopes will be essentially unchanged from present levels. All radioactive discharges, both liquid and gas, are as low as practical.

The accuracy of detection of the various nuclides will be verified via the Interlaboratory Comparison Program.

Entergy Operations, Inc. through Plant Management assumes the responsibility for seeing that this program is carried out.

2.8.2 AIR ENVIRONMENT

The air environmental monitoring program was designed to determine existing natural background radioactivity and to detect changes in radiation levels in the air environment which may be attributed to the operation of the nuclear unit.

Airborne radioactivity is detected by collection of radioiodine, air particulates, and measurements of external radiation as outlined in the ODCM.

2.8.3 WATER ENVIRONMENT

In the preoperational phase of the monitoring program, emphasis was placed on the determination of existing natural background radioactivity in the Dardanelle Reservoir, groundwater, potable water, sediment in Dardanelle Reservoir, and biota present in the water environment. Biota samples included collections of fish and underwater plants.

It is the intent of the water environmental program to determine long-term accumulation effects of radionuclides in the water environment. Accordingly, the program description for analyses of waterborne radionuclides is outlined in the ODCM.

2.8.4 LAND ENVIRONMENT

In the land environmental monitoring program, as in the water monitoring program, it is the intent to detect any change in the concentration of radionuclides in the land environment. Accordingly, samples of vegetation are collected and analyzed for gamma radionuclides and for specific radionuclides in accordance with Table 4-1 of the ODCM.

As an additional tool to be used to gauge the potential for the operational releases of radionuclides to be incorporated into the food chain via this route, a Land Use Census is performed once per 24 months that reflects all ODCM requirements (see ODCM Appendix 1, Section 2.6.2) has been incorporated into the radiological Environmental Monitoring Program. This census includes the location of milk animals, residences, and gardens exceeding 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of 5 miles from the ANO-1 reactor building. Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

2.8.5 MONITORING PROGRAM MODIFICATIONS

The monitoring program will remain flexible to accommodate events like those described below:

A. In the event of a radioactive discharge above normal operating conditions, a complete area monitoring will be undertaken to determine any change in environmental radioactivity due to the discharge.

B. If a sample location should no longer provide data required by the ODCM, the sample site location will be changed to provide the necessary sample.

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Table 2-1

CHEMICAL ANALYSIS OF GROUND AND RIVER WATER
(parts per million)

Sample Number*	pH	NO_3	Cl	SO_4	HCO_3	CO_3	Na	K	Ca	Mg	Alk	Total Hardness	TDS	SO_2	Fe	Mn	F
1	8.8	3	206	12	444	10	308	0.5	5	3	388	24	773	9	< 0.1	< 0.1	1.1
2	6.2	13	52	22	34	0	30	2.5	8	10	28	60	197	26	1.3	< 0.1	0
3	8.4	1.5	90	41	134	2	48	7.3	28	24	116	168	332	13	< 0.1	< 0.1	0.22
4	5.7	2.6	10	4	5	0	5	0.75	1.6	1.5	4	10	34	19	< 0.1	0.3	0
5	8.5	1.5	36	11	163	2	27	0.45	33	14	140	140	218	27	< 0.1	4.0	0.33
6	7.0	< 0.1	41	6	183	0	40	0.75	24	17	150	128	234	29	< 0.1	< 0.1	0.22
7	8.7	2.3	125	5	417	24	261	0.25	0.8	0.8	402	4	635	9	< 0.1	< 0.1	1.2
8	8.4	3.8	116	687	293	17	150	2	180	92	282	830	1559	22	< 0.1	< 0.1	0.22
9	6.8	210	196	178	107	0	216	3.1	36	41	88	258	1042	24	< 0.1	< 0.1	0.22
10	8.4	2.5	28	182	280	6	100	1.1	46	30	245	240	535	24	< 0.1	< 0.1	0.27
11 (Reservoir)	7.4	2.5	160	62	107	0	103	3.9	42	10	88	146	460	4	< 0.1	< 0.1	0.44
12 (Reservoir)	7.4	2.4	144	56	100	0	94	3.6	40	9	82	136	417	3	< 0.1	< 0.1	0.33
13 (D/S Dam)	7.4	4.5	141	58	100	0	91	3.9	43	8	82	142	419	4	0.25	< 0.1	0.44

Table 2-2

SUMMARY OF LABORATORY ROCK TESTS

1967 and 1968 Investigations

1967 Investigation - Data from DH-2, 38.5-40.5 feet, and DH-5, 28.9-38.0 feet

TESTS	NO. OF TESTS	RESULTS High	Low	Average
A. Specific Gravity	10	2.59	2.53	2.57
B. Porosity	10	2.0%	5.5%	3.8%
C. Absorption	10	3.1%	2.1%	2.4%
D. Unconfined Compression	9	3740 psi	3140 psi	3479 psi
E. Modulus of Elasticity	9	1.0×10^6 psi	0.7×10^6 psi	0.8×10^6 psi
F. Poisson's Ratio	4	.26	0.10	0.18
G. Triaxial Compression	10	Ø = 37° & C = 890 psi (average)		

1968 Investigations-Data from D-120, 30.0-62.0 feet, D-121, 25.0-51.0 feet

TESTS	NO. OF TESTS	RESULTS High	Low	Average
A. Unconfined Compression	17	4680 psi	2940 psi	3800 psi
B. Modulus of Elasticity	9	1.08×10^6 psi	$.25 \times 10^6$ psi	$.64 \times 10^6$ psi

Table 2-3

LABORATORY ROCK TEST DATA - 1970 INVESTIGATION

DH Hole No.	Depth	Sp.Gr.	% Poro.	% Aps.	Compressive Strength	E x 10^6	µ
201	32.0	2.53	6.68	2.64	2730	0.44	
	51.0	2.57	5.26	2.04	3420	0.74	0.33
	66.5	2.60	5.00	1.92	2670	0.76	
	79.6				3450		
	109.0	2.59	4.54	1.75	2760		
	143.0	2.63	2.79	1.13	4690		
202	28.5	2.51	7.14	2.84	2940	0.45	0.34
	37.0	2.60	4.93	1.89	3120	0.87	0.25
	37.5				3150		
	54.0				2790	0.64	0.32
	54.4				3390		
203	33.3				2790		
	39.5	2.55	5.97	2.33	3150	0.50	
204	30.7	2.56	6.17	2.40	2540	0.46	0.28
	43.5				4000		
205	40.5	2.50	6.45	2.51	3390	0.83	
206	31.5	2.52	7.81	3.09	2420	0.36	
					2730		
207	41.5	2.59	5.86	2.25	3760	0.76	
	42.0				4130		
	42.5				4060	0.76	0.34
208	26.0	2.57	6.55	2.54	2670		
	31.0				3540		
	37.5				3060	0.80	
	49.5	2.60	5.76	2.22	3670		
209	26.2	2.67	6.48	2.42	2420	0.49	
210	22.0	2.59	6.48	2.50	2970	0.95	0.24
211	30.1	2.54	7.80	3.09	2770		

	7 Samples			28 Samples	15 Samples	7 Samples
HIGH VALUE	2.67	7.81	3.09	4690	0.95	0.34
LOW VALUE	2.51	2.97	1.13	2420	0.36	0.24
AVERAGE	2.57	5.99	2.32	3255	0.65	0.30

Table 2-4

SUMMARY OF DRILL HOLE DATA

Auger Holes

HOLE NO.	SURFACE ELEVATIONS	DEPTH TO ROCK	ELEVATIONS TO TOP OF ROCK
AH-1	333	18	315
AH-2	340	22	318
AH-3	345	23	322
AH-4	350	24	326
AH-5	352	23	329
AH-6	352	23	329
AH-7	354	12	342
AH-8	353	26	337
AH-9	353	17	336
AH-10	353	15	338
AH-11	354	16	338
AH-12	353	21	332
AH-13	352	21	331
AH-14	352	26	326
AH-15	383	16	367
AH-16	349	17	332
AH-17	348	9	339
AH-18	345	7	338
AH-19	357	13	344
AH-20	354	10	344
AH-21	353	17	336
AH-22	353	21	332
AH-23	349	23	326
AH-24	343	16	327
AH-25	343	13	330
AH-26	339	10	329

Soil Borings

A-101	341.9	17.0	324.9
A-102	352.2	22.5	329.7
A-103	351.5	18.0	333.5
A-104	352.9	17.5	335.4
A-105	352.4	21.0	331.4
A-106	352.4	17.7	334.8
A-107	352.6	12.7	335.9
A-108	352.8	18.5	334.3
A-109	353.5	16.0	337.5
A-110	353.4	13.0	340.4
A-111	347.7	13.5	334.2
A-112	353.9	15.0	338.9
A-113	354.3	24.0	330.3
A-114	353.7	24.0	329.7
A-125	353.0	27.0	326.0

Table 2-4 (continued)

Soil Borings (continued)

HOLE NO.	SURFACE ELEVATIONS	DEPTH TO ROCK	ELEVATIONS TO TOP OF ROCK
A-212	353.3	26.5	326.8
A-213	354.7	27.0	327.7
A-214	354.6	27.0	327.6
A-215	355.4	28.0	327.4
A-216	NOT DRILLED	----	----
A-217	353.5	28.5	326.8
A-218	356.7	16.0	340.9
A-219	355.0	8.0	347.0
A-220	361.5	18.0	343.5
A-221	355.5	19.0	336.5
A-222	351.1	18.0	333.1
A-223	349.8	17.0	332.8
A-224	348.5	18.0	330.5
A-225	347.7	22.5	325.2
A-226	346.7	18.5	328.2
A-227	351.2	10.0	332.0
A-228	352.9	18.5	334.4
A-229	352.5	18.5	334.0
A-230	352.5	17.0	335.5
A-231	348.0	15.0	333.0
A-232	352.0	29.0	321.0
A-233	350.1	26.0	324.1
A-234	348.4	32.0	316.4
A-235	348.3	26.5	321.8
A-236	348.3	22.0	326.3

Core Holes

HOLE NO.	SURFACE ELEVATIONS	DEPTH TO ROCK	ELEVATIONS TO TOP OF ROCK
AD-115	352.2	20.0	332.2
AD-116	352.4	21.0	331.4
AD-117	353.1	18.0	335.1
AD-118	353.3	14.5	338.8
AD-119	350.0	17.0	333.0
D-120	352.5	23.5	329.0
D-121	352.6	18.0	334.6
D-122	352.7	18.0	334.7
DH-201	353.5	27.0	326.5
DH-202	354.0	27.0	327.0
DH-203	353.7	26.5	327.2
DH-204	354.5	29.5	325.0
DH-205	353.3	27.0	326.3
DH-206	352.9	27.0	325.9
DH-207	353.4	29.5	323.9
DH-208	348.3	24.0	324.3
DH-209	353.5	16.5	337.0
DH-210	357.4	16.0	341.4
DH-211	353.3	28.0	325.3

Table 2-5

PROPERTIES OF SHALE

		HIGH	LOW	AVE. VALUE	NO. OF SAMPLES
1.	Specific Gravity	2.67	2.51	2.57	17
2.	Porosity	8.0%	2.97%	5.88%	17
3.	Absorption	3.09%	1.13%	2.35%	17
4.	Unconfirmed Compressive Strength	4690 psi	2420 psi	3444 psi	55
5.	Modulus of Elasticity	1×10^6 psi	$.36 \times 10^6$ psi	0.67×10^6 psi	33
6.	Poisson's Ratio	.34	.10	0.25	11
7.	Triaxial Compression			Ø=42° & C=775 psi	10
8.	Seismic Velocity(compressional)			10,000 to 14,500 psi	

Table 2-6

MODIFIED MERCALLI EARTHQUAKE INTENSITY SCALE OF 1931 (ABRIDGED)

I Not felt except by a very few under especially favorable circumstances (I Rossi-Forel Scale).

II Felt only by a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing (I to II Rossi-Forel Scale).

III Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing truck. Duration estimated (III Rossi-Forel Scale).

IV During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably (IV to V Rossi-Forel Scale).

V Felt by nearly everyone; many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop (V to VI Rossi-Forel Scale).

VI Felt by all; many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight (VI to VII Rossi-Forel Scale).

VII Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motor cars (VIII Rossi-Forel Scale).

VIII Damage slight in specially designed structures; considerable in ordinary substantial buildings, with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Change in well water. Disturbed persons driving motor cars (VIII + to IX Rossi-Forel Scale).

IX Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken (IX+ Rossi-Forel Scale).

X Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks (X Rossi-Forel Scale).

XI Few, if any, (masonry) structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

XII Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

Table 2-7

CHRONOLOGICAL LIST OF EPICENTER LOCATIONS FOR MAP SHOWING SIGNIFICANT EARTHQUAKE EPICENTERS

DATE MO/DAY/YR	LOCATION	INTENSITY AND MAGNITUDE IF KNOWN	DISTANCE FROM SITE IN MILES
1/4/1843	35.2°N 90.0°W	VIII	180
11/18/1878	36.7°N 90.4°W	VI	180
7/20/1882	38.0°N 90.0°W	V	250
10/22/1882	35.0°N 94.0°W	VI-VIII	48
12/5/1883	36.3°N 91.8°W	V	100
7/19/1889	35.2°N 90.0°W	VI	180
8/21/1905	Southeastern Missouri	VI	195
7/4/1907	37.7°N 90.4°W	IV-V	218
3/31/1911	33.8°N 92.2°W	V	115
10/4/1918	34.7°N 92.3°W	V	67
10/13/1918	Black Rock, Arkansas 120 miles NE of Little Rock	V	126
11/3/1919	36.2°N 90.0°W	IV-V	140
10/28/1923	35.5°N 90.3°W	VII	162
12/31/1923	35.4°N 90.3°W	V	162
12/9/1933	35.8°N 90.2°W	VI	169
4/11/1934	33.9°N 95.5°W	V	160
3/14/1936	34.9°N 95.2W	V	145
5/16/1937	35.9°N 90.4°W	IV-V	158
9/17/1936	35.5°N 90.3°W	IV-V	162
6/19/1939	24.1°N 93.1°W	V	85
2/8/1950	37.4°N 92.4°W	V	150
2/2/1954	36.7°N 90.3°W	VI	185
4/26/1954	35.2°N 90.1°W	V	175
1/25/1955	35.6°N 90.3°W	VI	162
4/2/1956	34.2°N 95.4°W	V	147
10/30/1956	Catoosa, Oklahoma	VII	154
11/25/1956	37.1°N 90.6°W	VI	188
1/26/1958	35.1°N 90.6°W	V	180
1/10/1961	½ way Hartshorne and Wilburton, Oklahoma	V	127
4/27/1961	35.0°N 95.0°W	V	102
3/3/1963	36.7°N 90.1°W	VI	195
6/4/1967	33.6°N 90.0°W	VI/3.8	173
6/29/1967	33.6°N 90.0°W	V/3.4	173
7/21/1967	37.5°N 90.4°W	VI/3.9	203
1/1/1969	34.8°N 92.6°W	V/4.2	50

Table 2-8

OTHER EARTHQUAKE EPICENTERS PROBABLY FELT AT SITE

1811-1813- Has been described previously. See Figure 2-25.

1867, April 24: 39.5°N 96.7°W, Maximum intensity VII (MM). Lawrence Kansas, where objects were thrown from shelves and plaster was cracked. Walls were cracked in Manhattan, Kansas. Felt in Arkansas, Illinois, Indiana, Missouri, Nebraska and Kentucky.

1895, October 31: 37.0°N 89.4°W, Maximum intensity VIII (MM). Heaviest near Charleston, Missouri, where land sank. At Cairo, Illinois chimneys were demolished. Felt from Canada to Mississippi and from Georgia to Kansas.

1903, November 4: 38.5°N 90.3°W, Maximum intensity VI-VII (MM). St. Louis. Felt in southern Illinois, Kentucky, Mississippi, Arkansas, Missouri, and Tennessee.

1909, September 27: 39.0°N 87.8°W, Maximum intensity VII (MM). Indiana. Felt area included southwest half of Indiana, all of Illinois, eastern Iowa and Missouri, west half of Kentucky and northern Arkansas.

1909, October 23: 37.0°N 89.5°W, Maximum intensity V (MM). Center in southeastern Missouri. Felt in Missouri, Arkansas, Mississippi, Tennessee, Kentucky, Illinois and Indiana.

1915, December 7: 36.7°N 89.1°W, Maximum intensity V-VI (MM). A sharp earthquake near mouth of the Ohio River. It was felt in Illinois, Kentucky, Tennessee, Arkansas, Mississippi and Missouri.

1917, April 9: 38.1°N 90.6°W, Maximum intensity VI (MM). Epicentral region between St. Louis and New Madrid, Missouri, where windows were broken and plaster cracked. Felt in Kansas to Ohio and from Wisconsin to Mississippi.

1927, May 7: 36.5°N 89.0°W, Maximum intensity VII (MM). Arkansas and adjacent states. Strongest at North Jonesboro, Arkansas, where brick chimneys were tumbled down. Although it was felt strongly in Arkansas, the area over which it was felt indicates that the epicenter was further to the east near the position listed.

1940, November 23: 38.2N 90.1°W, Maximum intensity VI (MM). Felt throughout Illinois, Kentucky, Tennessee, Missouri, and Arkansas.

1952, April 9: 35.4°N 97.8°W, Maximum intensity VII (MM). Center near El Reno, Oklahoma, where chimneys fell and walls were cracked. Felt in western Arkansas (I-III) at Clarksville and Dardanelle which are within 20 miles of the site.

1962, July 23: 36.1°N 89.8°W, Maximum intensity VI (MM). Plaster cracked in Dyersburg, Tennessee. Site area just on edge of felt area according to map but not reported felt by towns near the site area which lies at the limit of perceptibility.

1965, October 20: 37°51'N 91°05'W, Maximum intensity VI (MM). Centered in east central Missouri, VI at St. Louis. Not reported felt in towns near site area which lies at the limit of perceptibility.

1968, November 9: 38.0°N 88.5°W, Maximum intensity VII (MM). Southern Illinois. It was reported not felt at Russellville even though it was felt at Dardanelle. It was reported felt with intensity V in the extreme northeast part of Arkansas and intensities I-III as far south as Murfreesboro 90 miles southwest of the site.

Table 2-9

WELL SURVEY
(3-Mile Radius of ANO)

AREA I*

Total Number of Wells	35
Well Usage	
Residential	34
Business	1
Depth Range	60-315 ft
Location Within Area	
East Side	14
West Side	14
West Side	3
North Side	4

(All wells within this area are located either above plant grade level or out of the path of any drainage from plant)

AREA II*

Total Number of Wells	79
Well Usage	
Residential	71
Business	3
Churches	2
Poultry Farm	1
London Water Works	2
Depth Range (Excluding London Water Works)	40-105 ft
Depth - London Water Works Wells	285 ft, 450 ft

AREA III*

Total Number of Wells	60
Well Usage	
Residential	58
Business	1
Church	1
Depth Range	60-150 ft

AREA IV*

Total Number of Wells	88
Well Usage	
Residential	80
Business	2
Church	1
Poultry Farm	1
Livestock Farm	1
Arkansas State Park	2
Trailer Court (7 trailers)	1
Depth Range	35-300 ft

Table 2-9 (continued)

AREA V*

Total Number of Wells	22
Well Usage	
Residential	22
Depth Range	28-140 ft

AREA VI*

Total Number of Wells	0

*See Figure 2-31 for area location

Table 2-10

DELETED

SEE ODCM

Table 2-11

MONTHLY PERCENT OF POSSIBLE OBSERVATIONS, ARKANSAS NUCLEAR ONE SITE, WEATHER SYSTEMS

	J	F	M	A	M	J	J	A	S	O	N	D
1967												
30' Wind									52	72	62	100
1968												
30' Wind	68	96	86	2	10	0	47	94	91	70	52	75
1969												
30' Wind	67	72	33	52	21	0	0	0	0	90	96	
20' Wind						60	1	9	0	66	100	94
∆T, Full						100	68	99	99	97	100	95
∆T, Half						99	68	78	98	97	98	100
1970												
20' Wind	71	83	73	74	56							
∆T, Full	76	88	78	77	99							
∆T, Half	96	69	99	86	96							

Table 2-12

COMPARISONS OF TEMPERATURES AND PRECIPITATIONS IN THE ARKANSAS RIVER VALLEY

	Little Rock	Conway	Morrilton	Russellville	Ozark	Fort Smith
	MEAN TEMPERATURE 30-YEAR NORMAL (°F)					
January	41	43	41	42	41	40
February	44	46	44	46	44	44
March	52	53	51	53	51	51
April	62	63	62	63	62	62
May	71	70	70	70	69	70
June	79	79	78	78	78	79
July	82	82	82	82	82	83
August	81	82	81	81	82	82
September	74	75	74	74	75	75
October	63	64	64	64	64	64
November	50	51	51	51	50	50
December	42	44	43	44	42	42
ANNUAL	62	63	62	62	62	62
	MEAN PRECIPITATION, 30-YEAR NORMAL (Inches)					
January	5.22	4.78	4.27	4.04	3.41	2.66
February	4.33	4.62	3.94	4.24	3.94	3.43
March	4.81	5.03	4.73	4.84	3.84	3.47
April	4.93	5.40	4.94	5.06	4.76	4.24
May	5.28	5.82	5.33	5.61	5.68	5.26
June	3.61	3.96	3.90	4.14	4.14	4.35
July	3.34	3.76	2.91	4.17	3.34	2.80
August	2.82	2.98	2.95	3.79	3.10	2.92
September	3.23	3.26	3.01	3.53	3.29	3.64
October	2.88	3.22	2.90	3.28	3.31	3.45
November	4.12	4.39	3.90	4.13	3.29	3.18
December	4.09	4.46	4.01	3.47	3.04	2.82
ANNUAL	48.66	51.68	46.79	49.74	45.14	42.22

Table 2-13

PERCENTAGE FREQUENCY OF WIND DIRECTION AND MEAN SPEED (KT)

Early Program (30-ft) Versus Tower Program (20-ft)

	October, 1969				November, 1969			
	30-Foot		20-Foot		30-Foot		20-Foot	
Sector	%	Mean Speed	%	Mean Speed	%	Mean Speed	%	Mean Speed
N	2.9	5.0	4.2	3.5	3.9	3.6	1.3	1.8
NNE	2.7	3.6	5.3	3.0	1.9	4.6	4.6	2.5
NE	4.9	5.0	7.7	4.3	4.4	2.5	6.4	2.4
ENE	8.1	8.6	10.9	5.9	6.5	3.7	7.3	2.5
E	19.7	5.7	10.5	4.0	15.1	3.7	16.2	2.3
ESE	13.9	6.4	4.6	3.2	6.0	3.5	6.1	2.4
SE	8.8	6.1	7.2	2.2	4.5	4.8	4.0	3.8
SSE	6.5	6.2	11.2	3.3	5.7	5.6	5.0	4.8
S	7.6	8.4	3.5	2.4	6.0	6.6	6.6	4.3
SSW	1.4	4.3	1.5	3.4	1.9	6.5	2.1	4.4
SW	1.3	4.8	1.5	3.0	1.5	4.0	1.6	2.5
WSW	1.1	3.0	1.1	3.0	2.3	4.0	2.7	2.1
W	3.2	8.0	9.2	5.3	7.8	4.6	14.7	5.5
WNW	6.7	6.9	11.6	7.2	10.0	7.4	13.9	6.9
NW	6.7	7.7	5.0	3.8	12.3	6.7	5.3	5.6
NNW	4.5	6.3	5.0	2.7	10.2	7.5	2.2	2.2
% Mean	100	6.3	100	4.2	100	5.2	100	4.0
% Calm		6.3		6.5		10.2		13.0

Note: Percent calm shown separately and not included in table.

Table 2-14

COINCIDENT PERCENT OCCURRENCE OF STABILITY CATEGORY, 30-FOOT EARLY PROGRAM VS 20-FOOT TOWER PROGRAM

Arkansas Nuclear One Site October and November, 1969

		Slade Stability Category 30-foot						Non-Steady			CUM
		A	B	C	D	E	F	Night	Day	Total	%
Slade Stability Category 20-foot	A	1.3	0.4	0.5	0.4	0.0	0.0	0.8	1.6	5.0	5.0
	B	0.5	0.6	1.1	0.4	0.0	0.0	1.1	1.1	4.8	9.8
	C	0.2	1.1	2.9	2.1	0.1	0.0	2.4	0.4	9.2	19.0
	D	0.6	1.3	4.2	12.0	3.7	0.1	1.8	0.8	24.5	43.5
	E	0.0	0.1	1.5	12.0	14.4	0.7	2.0	0.5	31.2	74.7
	F	0.1	0.0	0.4	2.3	5.8	1.0	0.9	0.3	10.8	85.5
NS - Night		0.2	0.2	0.6	0.3	0.4	0.0	3.7	1.6	7.0	92.5
NS - Day		0.2	0.2	0.0	0.5	0.2	0.0	4.4	2.0	7.5	100.0
Total		3.1	3.9	11.2	30.0	24.6	1.8	17.1	8.3	100.0	
Cumulated Percent-Age		3.1	7.0	18.2	48.2	72.8	74.6	91.7	100.0		

Table 2-15

PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED (KT), SEASONAL SUMMARY, SURFACE WINDS

Arkansas Nuclear One Site Data, Sept. 1967 through May 1969

	Winter		Spring		Summer		Fall	
Sector	%	ū	%	ū	%	ū	%	ū
N	2.8	3.9	1.2	4.1	2.3	1.8	4.4	3.2
NNE	2.8	3.5	2.3	5.6	3.6	2.9	4.8	2.8
NE	5.2	3.5	3.0	4.8	3.3	2.3	6.8	2.6
ENE	7.2	4.0	7.0	4.5	5.7	3.5	8.0	3.3
E	19.0	4.9	15.7	6.8	9.7	3.1	14.1	3.9
ESE	10.8	5.4	8.9	5.3	10.2	3.6	6.7	4.1
SE	5.0	4.5	6.3	4.9	7.8	3.4	6.4	4.2
SSE	3.3	4.9	6.6	5.0	10.0	4.0	7.0	4.7
S	3.5	5.6	7.9	5.8	12.0	4.6	4.4	4.6
SSW	1.2	4.0	2.0	6.4	5.3	4.9	2.7	3.6
SW	0.9	5.2	1.7	6.4	4.0	4.3	2.3	3.3
WSW	2.3	4.8	2.2	6.0	4.6	4.7	4.1	4.0
W	8.9	6.3	4.4	8.1	4.8	3.4	7.1	5.2
WNW	9.1	7.4	3.9	6.6	5.8	4.8	5.9	6.0
NW	7.3	6.9	4.2	7.1	4.9	3.5	5.3	5.4
NNW	5.4	7.1	3.7	7.6	3.8	4.3	6.0	4.7
CALM	5.3	-	18.9	-	2.2	-	4.2	-
Total	100		100		100		100	
Mean		5.1		4.5		3.7		4.0

Table 2-16

PERCENTAGE FREQUENCY OF WIND DIRECTION AND MEAN SPEED (KT), ANNUAL SUMMARY, SURFACE WINDS

Arkansas Nuclear One Site Data, September 1967 through May 1970

Sector	1-3	4-6	7-10	11-15	16-20	21-25	26+	Total Freq.	Mean Speed
N	1.7	0.7	0.2	0.1				2.7	3.2
NNE	1.9	1.0	0.4					3.4	3.5
NE	2.9	1.2	0.5	+				4.6	3.2
ENE	3.5	2.5	0.8	0.1				7.0	3.8
E	5.9	5.9	2.6	0.2	0.1			14.6	4.3
ESE	3.5	3.7	1.5	0.4				9.2	4.6
SE	2.6	2.7	1.1	+				6.3	4.2
SSE	2.4	3.0	1.2	0.1	+			6.7	4.5
S	2.0	3.0	1.8	0.1				6.9	5.1
SSW	1.0	1.1	0.6	0.1				2.8	4.8
SW	1.1	0.5	0.5	0.1				2.2	4.5
WSW	1.3	1.2	0.6	0.1	+	+		3.3	4.7
W	2.1	1.9	1.6	0.6	0.1			6.3	5.8
WNW	1.8	1.8	1.6	0.8	0.2		+	6.2	6.3
NW	1.7	1.8	1.2	0.6	0.1			5.4	5.8
NNW	1.5	1.6	0.9	0.6	0.1			4.7	5.9
CALM	___	___	___	___	___	___	___	7.6	___
Total	36.9	33.7	17.2	3.9	0.6	0.0	0.0	100.0	4.3

Notes:

1. Includes 4657 of 7964 possible observations
2. Data based on 3-hourly observations
3. Includes correlated data

Table 2-17

ANNUAL SURFACE WINDS WHEN PRECIPITATION OCCURS, PERCENT FREQUENCY OF WIND SPEED (KT) AND DIRECTION

Arkansas Nuclear One Site June 1969 through May, 1970

			Wind Speed			Total	Mean
Sector	1-3	4-6	7-10	11-15	16+	Frequency	Speed
N	3.3	1.0	1.0	0.2	5.5	4.1	
NNE	3.9	1.4	0.8	0.5		6.6	4.2
NE	4.4	1.7	1.0	0.2	7.3	3.9	
ENE	3.6	1.0	1.0			5.6	3.7
E	6.9	2.2	2.1	0.3		11.5	4.1
ESE	6.2	3.4	4.2	0.4	0.3	14.4	5.2
SE	2.7	1.0	0.6			4.3	3.6
SSE	3.0	1.7	0.6			5.3	3.7
S	2.7	0.8	2.1			5.6	5.0
SSW	0.4	0.2	0.3			0.9	4.9
SW	0.2	0.2		0.2		0.6	6.7
WSW	1.8	1.4	0.5	0.2		3.9	4.5
W	5.6	1.0	0.6	0.2		7.4	3.2
WNW	4.3	1.8	2.7			8.8	4.6
NW	2.5	0.3	0.2	0.2		3.2	3.4
NNW	3.5	0.5				4.0	2.4
CALM						5.1	
TOTAL	55.0	19.6	17.7	2.4	0.2	100.0	3.9

Table 2-18

PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED (KT), ANNUAL SUMMARY, 190-FOOT LEVEL

Arkansas Nuclear One Site Data, June 1969 through May 1970

Sector	1-3	4-6	Speed 7-10	11-16	17+	Total Frequency	Mean Speed
N	0.9	0.6	0.7	0.3		2.5	6.1
NNE	0.5	0.4	0.3	0.3		1.5	6.9
NE	0.8	1.1	1.1	0.4	0.3	3.7	4.5
ENE	1.0	2.1	1.6	0.7	0.2	5.6	5.6
E	1.8	4.1	2.8	1.0	0.1	9.8	6.3
ESE	0.9	3.4	2.3	1.0	0.1	7.7	6.9
SE	1.0	1.5	1.8	1.2	0.2	5.7	7.8
SSE	0.9	1.2	1.5	0.5	0.1	4.2	6.9
S	1.1	3.0	4.0	2.1	0.1	10.3	7.9
SSW	0.9	2.5	2.6	1.3	0.1	7.4	7.3
SW	0.8	1.6	1.5	1.2	0.1	5.2	7.5
WSW	1.4	2.6	2.5	0.7	0.2	7.4	6.5
W	2.0	6.4	3.1	0.8	0.2	12.5	5.9
WNW	1.1	3.2	2.3	1.0	1.1	8.7	8.8
NW	1.0	1.8	1.1	1.1	0.3	5.3	9.0
NNW	0.4	0.8	0.5	0.3	0.1	2.1	6.6
CALM	___	___	___	___	___	0.4	___
TOTAL	16.5	36.3	29.7	13.9	3.2	100.0	6.9

Notes:

1. Includes 4263 of 8760 possible observations
2. Data based on hourly observations

Table 2-19

PASQUILL CATEGORY/SIGMA THETA RELATIONSHIPS

Atmospheric Condition	Pasquill Category	Sigma Theta Degrees
Extremely Unstable	A	22.5
Unstable	B	17.6 - 22.5
Slightly Unstable	C	12.6 - 17.5
Neutral	D	7.6 - 12.5
Slightly Stable	E	2.6 - 7.5
Stable	F	2.6

Table 2-20

SEASONAL PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY

(M/Slade Method) and Mean Speed (m/sec)

Surface Winds, Arkansas Nuclear One Site (September 1967-May 1970)

	WINTER	SPRING	SUMMER	FALL
Cat. A, %	6.2	9.8	16.6	12.2
$\bar{u}$	1.3	1.3	1.5	1.4
Cat. B, %	6.1	8.6	10.1	14.4
$\bar{u}$	1.4	1.1	1.3	1.7
Cat. C, %	11.9	14.5	20.0	15.6
$\bar{u}$	2.1	2.0	1.8	2.1
Cat. D, %	30.3	24.7	32.4	24.6
$\bar{u}$	2.8	3.0	2.3	2.6
Cat. E, %	34.1	33.8	17.7	28.2
$\bar{u}$	3.0	2.6	1.5	2.2
Cat. F, %	11.5	8.6	3.2	5.0
$\bar{u}$	2.9	0.9	0.6	1.0
Total, %	100	100	100	100
Mean, $\bar{u}$	2.6	2.2	1.8	2.1

Table 2-21

ANNUAL PERCENT FREQUENCY OF PASQUILL STABILITY CLASS (SLADE METHOD) AND MEAN SPEED (M/SEC) VS DIRECTION FOR SURFACE WINDS, ARKANSAS NUCLEAR ONE SITE

Wind Direction From	PASQUILL A	AVE SPEED	PASQUILL B	AVE SPEED	PASQUILL C	AVE SPEED	PASQUILL D	AVE SPEED	PASQUILL E	AVE SPEED	PASQUILL F	AVE SPEED	TOTAL ALL CLASSES FREQUENCY	TOTAL ALL CLASSES MEAN SPD
N	0.5	0.7	0.2	0.8	0.3	1.3	0.8	1.5	0.7	1.5	0.2	0.8	2.7	1.2
NNW	0.6	1.4	0.3	1.1	0.5	1.6	1.0	1.9	0.7	1.7	0.1	0.5	3.4	1.6
NE	0.5	0.8	0.7	1.5	0.5	1.5	1.3	1.9	1.3	2.1	0.2	1.1	4.6	1.7
ENE	0.6	1.1	0.7	1.4	1.3	2.0	2.2	2.1	1.6	2.1	0.5	1.3	7.0	1.9
E	1.4	1.4	1.0	1.5	2.2	1.7	5.0	2.5	4.1	2.6	0.9	1.4	14.6	2.2
ESE	0.9	1.5	1.0	1.8	1.6	1.9	3.0	2.7	2.1	2.4	0.6	1.2	9.2	2.2
SE	0.8	1.0	0.6	1.6	1.5	2.2	1.8	2.7	1.2	2.6	0.4	1.2	6.3	2.2
SSE	1.1	1.6	0.7	1.4	1.1	2.3	1.9	2.9	1.6	3.0	0.3	1.4	6.7	2.4
S	1.3	1.9	0.7	1.7	1.0	2.6	2.2	3.0	1.4	2.7	0.4	2.1	6.9	2.5
SSW	0.7	0.9	0.3	1.5	0.3	2.0	0.9	3.4	0.6	2.8	0.1	0.7	2.8	2.3
SW	0.5	1.2	0.2	1.5	0.4	1.9	0.5	2.9	0.5	1.7	0.0	0.6	2.2	1.8
WSW	0.5	1.0	0.3	1.2	0.9	2.4	1.0	3.5	0.7	2.4	0.0	0.3	3.3	2.4
W	0.5	1.0	0.5	1.2	1.3	2.0	2.0	2.7	1.8	3.0	0.2	1.6	6.3	2.4
WNW	0.4	1.2	0.4	1.6	0.9	2.7	1.8	2.9	2.3	3.1	0.4	1.8	6.2	2.7
NW	0.3	0.5	0.5	1.3	0.5	1.9	1.3	2.7	2.1	3.3	0.7	2.2	5.4	2.5
NNW	0.4	1.0	0.5	1.5	0.6	2.0	1.2	3.1	1.7	3.4	0.5	1.4	4.7	2.6
CALM	0.2	0	1.3	0	0.6	0	0.1	0	3.9	0	1.6	0	7.6	0
TOTAL	11.2		9.8		15.5		28.0		28.4		7.1		100.0	
MEAN		1.2		1.3		2.0		2.6		2.3		1.2		2.0

Table 2-22

ANNUAL PERCENT FREQUENCY OF COINCIDENT 20-FOOT SLADE CATEGORY AND FULL TOWER ΔT

Arkansas Nuclear One Site June 1969 through May 1970

		Full Tower ΔT			
		UNSTABLE	NEUTRAL	STABLE	TOTAL
20-Foot Slade Stability Category	A	0.7	0.6	1.0	2.3
	B	0.8	0.7	2.5	4.0
	C	2.2	1.7	8.7	12.6
	D	5.1	3.7	25.2	34.0
	E	2.3	1.9	20.7	24.9
	F	0.1	0.1	2.2	2.4
Non-Steady		0.8	0.5	18.5	19.8
Total		12.0	9.2	78.8	100.0

NOTES:

1. Includes 5291 observations of 8760 possible observations

2. Full Tower ΔT defined below for each of three categories

   Unstable $\Delta T/\Delta Z \leq -1.7$ °C/100m
   Neutral $\Delta T/\Delta Z = -0.5$ to -1.6 °C/100m
   Stable $\Delta T/\Delta Z \geq -0.4$ °C/100m

3. 20-Foot slade stability category data is not adjusted by correlation procedures to correspond with the 30-foot early program data.

Table 2-23

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY A

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N			0.2				0.2	8.5
NNE	0.4	0.8		0.2			1.4	5.3
NE		2.2	1.0	0.2			3.4	6.5
ENE	0.2	1.4	1.6	0.4			3.6	7.3
E	0.2	4.6	5.5	0.6			10.9	7.2
ESE		4.8	3.2	0.2			8.2	6.6
SE		3.4	2.2	0.2			5.8	6.6
SSE	0.2	4.0	3.6	0.2	0.2		8.2	7.0
S	0.4	2.0	4.2	0.2			6.8	7.2
SSW		0.4	2.0				2.4	7.9
SW		1.0	1.2	0.2			2.4	7.4
WSW		1.0	1.8	0.4			3.2	8.0
W		2.5	5.0	2.6	0.2		10.3	9.0
WNW		1.4	5.6	3.1			10.1	9.4
NW	0.4	1.0	4.1	3.6	0.4		9.5	10.0
NNW	0.2	1.3	4.9	3.0	0.2		9.6	9.5
CALM								
TOTAL	2.0	31.8	46.1	15.1	1.0		100.0	7.7

NOTE:
1. Data based on hourly observations
2. This category occurred 503 times or 10.5 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is ≤ -1.9 °C/100 m

Table 2-24

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY B

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N								
NNE			1.1				1.1	8.5
NE			1.1				1.1	8.5
ENE		2.2					2.2	5.0
E	2.2	6.9	6.7	1.1			16.9	6.5
ESE	1.2	10.1	6.7	1.1			19.1	6.5
SE		4.5	2.2				6.7	6.1
SSE		1.1	4.5				5.6	7.8
S		3.4					3.4	5.0
SSW		1.1					1.1	5.0
SW		2.2					2.2	5.0
WSW		6.7	1.2		1.1		9.0	7.0
W		1.1	5.7	2.3			9.1	9.2
WNW		4.5	4.5				9.0	6.8
NW		3.4	3.3				6.7	6.7
NNW	1.1	1.1	1.2		1.1		4.5	8.3
CALM								
TOTAL	4.5	48.3	38.2	4.5	2.2		100.0	6.7

NOTES:

1. Data based on hourly observations
2. This category occurred 89 times or 1.9 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is > -1.9 to -1.7 °C/100 m

Table 2-25

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY C

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N								
NNE		1.4					1.4	5.0
NE		2.8	1.4				4.2	6.2
ENE		2.8					2.8	5.0
E		9.8	2.8				12.6	5.7
ESE		10.0		1.4			11.4	6.0
SE		4.3	2.8				7.1	6.4
SSE	1.4		7.1				8.5	7.4
S		2.8	2.8				5.6	6.8
SSW		2.8					2.8	5.0
SW		4.2					4.2	5.0
WSW		5.6					5.6	5.0
W		1.4	4.3	1.4			7.1	8.7
WNW		4.2	1.4				5.6	5.9
NW		2.9	4.2	1.4	1.4		9.9	9.5
NNW		2.8	1.4				4.2	6.2
CALM								
TOTAL	1.4	57.8	28.2	4.2	1.4		100.0	6.1

NOTES:

1. Data based on hourly observations
2. This category occurred 71 times or 1.5 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is > -1.7 to -1.5 °C/100 m

Table 2-26

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY D

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N			0.3	0.2			0.5	10.4
NNE	0.3	0.5	1.0				1.8	6.4
NE		1.8	0.5	0.3			2.6	6.6
ENE	0.5	1.8	1.0	0.3			3.6	6.2
E	1.0	8.6	3.9				13.5	5.7
ESE	0.3	6.2	3.9	0.5			10.9	6.5
SE	0.8	4.4	2.3				7.5	5.7
SSE	0.5	6.0	3.1	0.5			10.1	6.3
S	0.7	4.2	4.4	0.3			9.6	6.6
SSW	0.3	1.0	1.6	0.5			3.4	7.5
SW	0.5	0.8	0.8	0.5		0.3	2.9	8.4
WSW	0.8	0.8	0.7				2.3	5.0
W	0.8	2.3	3.7	1.0	0.8		8.6	8.3
WNW		2.1	2.6	1.3			6.0	9.2
NW		1.3	1.6	2.3			5.2	9.6
NNW	0.5	1.3	1.8	1.6	0.8		6.0	9.6
CALM							5.5	
TOTAL	7.0	43.1	33.2	9.3	1.6	0.3	100.0	6.6

NOTE:

1. Data based on hourly observations
2. This category occurred 385 times or 8.0 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is > -1.5 to -0.5 °C/100 m

Table 2-27

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY E

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

Sector	1-3	4-6	7-10	11-15	16-20	21+	Total Freq.	Mean Speed
			Speed, Knots					
N	0.5	0.4	0.1	0.1			1.1	4.7
NNE	0.7	2.1	0.8				3.6	5.1
NE	0.1	2.4	2.2				4.7	6.5
ENE	0.7	2.6	2.9	0.4			6.6	6.7
E	1.1	9.6	6.9	1.8	0.3		19.7	6.9
ESE	0.7	3.3	6.0	0.9	0.1		11.0	7.4
SE	0.9	2.4	1.9				5.2	5.7
SSE	0.8	3.9	2.2	0.2			7.1	5.9
S	0.9	3.7	2.5	0.3			7.4	6.1
SSW	0.2	0.9	0.4	0.1			1.6	6.0
SW	0.4	0.5	0.1	0.2			1.2	5.6
WSW	0.2	0.9	0.5				1.6	5.7
W	0.4	2.3	1.7	1.5	0.6		6.5	8.7
WNW	0.4	2.1	2.5	1.1	0.4		6.5	8.3
NW	0.6	2.3	2.1	1.4	0.1		6.5	7.7
NNW	0.6	2.3	2.1	1.3	0.1	0.1	6.5	7.9
CALM							3.2	
TOTAL	9.2	41.7	34.9	9.3	1.6	0.1	100.0	6.7

NOTE:

1. Data based on hourly observations
2. This category occurred 1345 times or 28.1 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category > -0.5 to 1.5 °C/100 m

Table 2-28

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY F

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

Sector	1-3	4-6	7-10	11-15	16-20	21+	Total Freq.	Mean Speed
			Speed, Knots					
N	0.4	0.8	0.1				1.3	4.3
NNE	0.8	1.8	0.3				2.9	4.5
NE	1.5	2.3	0.2				4.0	4.1
ENE	0.8	4.4	0.2				5.4	4.7
E	2.6	8.1	3.6				14.3	5.3
ESE	1.9	5.4	1.6				8.9	5.0
SE	1.0	4.1	1.2	0.1			6.4	5.3
SSE	0.5	4.5	1.1				6.1	5.4
S	0.7	3.3	2.9	0.7			7.6	6.8
SSW	0.1	1.2	0.4	0.6			2.3	7.6
SW	0.3	0.6	0.9	0.1			1.9	6.6
WSW	0.6	1.3	0.3	0.1			2.3	5.0
W	0.5	2.5	2.2	1.4			6.6	7.6
WNW	1.2	2.3	2.8	1.4	0.3		8.0	7.7
NW	1.7	2.5	3.0	1.1	0.1		8.4	6.8
NNW	0.6	3.2	2.2	1.1	0.3		7.4	7.5
CALM							6.2	
TOTAL	15.2	48.3	23.0	6.6	0.7		100.0	5.7

NOTE:

1. Data based on hourly observations
2. This category occurred 975 times or 20.3 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is > 1.5 to 4.0 °C/100 m

Table 2-29

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, FULL TOWER, CATEGORY G

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 Through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N	0.8	0.4					1.2	3.0
NNE	1.9	0.8					2.7	2.9
NE	3.4	1.9	0.2				5.5	3.3
ENE	4.1	3.8	0.2				8.1	3.6
E	7.4	6.3	0.5				14.2	3.6
ESE	3.8	2.8	0.1				6.7	3.4
SE	1.7	1.1	0.1				2.9	3.4
SSE	1.0	1.5	0.3				2.8	4.3
S	0.8	1.0	0.7	0.1			2.6	5.3
SSW	1.2	0.4	0.1				1.7	3.1
SW	1.0	0.6	0.1				1.7	3.1
WSW	1.6	1.1	0.2				2.9	3.6
W	2.5	2.7	0.8	0.1			6.1	4.4
WNW	2.3	2.6	0.6	0.1			5.6	4.3
NW	2.8	1.9	1.0	0.3			6.0	4.6
NNW	2.2	2.0	0.7	0.2			5.1	4.5
CALM							24.2	
TOTAL	38.5	30.9	5.6	0.8			100.0	2.9

NOTE:

1. Data based on hourly observations
2. This category occurred 1426 times or 29.7 percent of the total observations
3. Total of all categories includes 4794 of 8760 possible observations
4. Temperature difference for this category is > 4.0 °C/100 m

Table 2-30

**ANNUAL PERCENT FREQUENCY OF COINCIDENT
20-FOOT PASQUILL-GIFFORD AND FULL TOWER TEMPERATURE DIFFERENCE
STABILITY CATEGORY**

Arkansas Nuclear One Site, June, 1969 through May 1970

		Full Tower T							
		A	B	C	D	E	F	G	Total
Pasquill – Gifford 20-Foot	A	0.5	0.1	0.1	0.6	0.7	0.1	0.3	2.4
	B	0.5	0.2	0.2	0.4	1.0	0.5	0.7	3.5
	C	1.9	0.4	0.3	1.3	2.8	2.3	3.3	12.3
	D	4.6	0.7	0.6	3.5	12.5	6.9	6.8	35.6
	E	2.0	0.3	0.3	1.5	9.0	6.1	3.9	23.1
	F	0.1			0.1	0.6	0.6	0.6	2.0
NS - Day		0.8	0.1	0.2	0.5	1.5	1.4	2.4	6.9
NS - Night						0.8	1.8	11.6	14.2

NOTES:

1. Includes 5365 of 8760 possible observations
2. 20-foot slade categories not adjusted
3. Temperature difference categories defined in text
4. Pasquill-Gifford categories defined by Slade
5. NS is non-steady wind

Table 2-31

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY A

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N	0.4	0.5					0.9	3.7
NNE	0.9	1.2	0.2				2.3	4.1
NE	0.7	3.0	0.7				4.4	5.1
ENE	0.9	2.7	0.9	0.2			4.7	5.4
E	0.5	6.2	4.2	0.7	0.2		11.8	4.4
ESE	0.2	4.9	3.3	0.9			9.3	7.0
SE	1.0	2.8	0.9				4.7	5.0
SSE	0.3	3.9	2.6	0.2			7.0	6.4
S	1.2	1.9	3.0				6.1	6.1
SSW	0.9	1.0	0.9				2.8	5.2
SW		1.1	0.5				1.6	6.1
WSW	0.5	0.9	0.7				2.1	5.5
W	0.9	2.3	4.0	1.2	0.5		8.9	8.1
WNW	0.9	1.6	2.9	1.9	0.2		7.5	8.4
NW	1.2	1.6	3.5	1.9			8.2	7.9
NNW	0.7	2.6	3.2	1.2	0.5		8.2	8.1
CALM							9.5	
TOTAL	11.2	38.2	31.5	8.2	1.4		100.0	6.1

Notes:

1. Data based on hourly observations
2. This category occurred 428 times or 8.8 percent of the total observations
3. Total of all categories included 4923 of 8760 possible observations
4. Temperature difference for this category is ≤ -1.9 °C/100 m

Table 2-32

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY B

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N		1.1	0.6	0.6			2.3	6.6
NNE	0.6	4.4	2.2				7.2	5.8
NE	1.1	2.2	1.6				4.9	5.5
ENE		1.1	3.3	0.6			5.0	8.4
E	2.2	5.5	4.4	1.1			13.2	6.3
ESE		6.0	4.4				10.4	6.5
SE	5.0	2.8					7.8	6.3
SSE		1.6	2.8				4.4	7.2
S	0.6		1.1				1.7	6.2
SSW		1.1	0.6				1.7	6.2
SW		1.6	1.1				2.7	6.4
WSW	1.1	0.6					1.7	3.1
W		3.9	5.0	1.6	0.6		11.1	8.4
WNW			5.5	3.3	0.6		9.4	10.7
NW		3.3	2.2	1.6	0.6		7.7	8.7
NNW	0.6	1.6	1.6	2.2			6.0	8.6
CALM							2.8	
TOTAL	6.2	39.0	39.2	11.0	1.8		100.0	7.2

Note:

1. Data based on hourly observations
2. This category occurred 181 times or 3.7 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > -1.9 to -1.7 °C/100 m

Table 2-33

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY C

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N	0.4			0.4			0.8	7.5
NNE		2.4	0.4				2.8	5.5
NE	0.5	2.8	1.2				4.5	5.6
ENE		0.8	4.1	0.8			5.7	8.6
E	0.9	7.7	7.3	2.8	0.4		19.1	7.3
ESE	0.8	2.4	6.1	0.9			10.2	7.6
SE		3.7	2.0				5.7	6.2
SSE	0.8	2.5	1.2				4.5	5.4
S		2.8	3.7				6.5	7.0
SSW	0.4	0.8	0.4				1.6	5.1
SW				0.4			0.4	13.0
WSW			0.4				0.4	8.5
W		0.9	2.4	2.0			5.3	9.6
WNW	0.8	3.7	4.9	2.0	0.4		11.8	8.0
NW	0.7		3.7	3.3	0.8		8.5	10.6
NNW		2.8	3.3	3.7	1.2		11.0	10.2
CALM							1.2	
Total	5.3	33.3	41.1	16.3	2.8		100.0	7.8

Note:

1. Data based on hourly observations
2. This category occurred 246 times or 4.9 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > -1.7 to -1.5 °C/100 m

Table 2-34

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY D

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots							
Sector	1-3	4-6	7-10	11-15	16-20	21+	Total Freq.	Mean Speed
N	0.4	0.4	0.3				1.1	4.9
NNE	0.7	1.6	0.7				3.0	5.1
NE	0.7	2.4	2.0	0.2			5.3	6.2
ENE	1.0	2.9	1.7	0.3			5.9	5.9
E	1.3	8.2	6.0	1.0	0.1		16.6	6.6
ESE	0.5	4.0	3.9	0.4	0.1		8.9	6.9
SE	0.6	2.5	1.5	0.1			4.7	5.9
SSE	0.4	3.3	1.6	0.2			5.5	6.1
S	0.7	2.7	1.6	0.1			5.1	5.8
SSW	0.2	0.8	0.5	0.3			1.8	7.0
SW	0.4	0.8	0.5	0.2		0.1	2.0	6.9
WSW	0.3	1.5	0.9	0.1	0.1		2.9	6.5
W	0.7	2.7	2.8	2.3	0.5		9.0	8.5
WNW	0.4	2.4	3.3	1.6	0.2		7.9	8.3
NW	0.7	2.7	3.0	2.1	0.1		8.6	8.1
NNW	0.7	1.8	2.6	1.7	0.2	0.1	7.1	8.5
CALM							4.6	
TOTAL	9.7	40.7	32.9	10.6	1.3	0.2	100.0	6.7

Note:

1. Data based on hourly observations
2. This category occurred 1528 times or 31.1 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > -1.5 to -0.5 °C/100 m

Table 2-35

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY E

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N	0.5	0.2					0.7	2.9
NNE	0.8	0.8	0.3				1.9	4.3
NE	1.3	1.8	0.4				3.5	4.3
ENE	1.3	3.9	0.6	0.1			5.9	4.8
E	3.1	8.9	3.2	0.1	0.1		15.4	7.6
ESE	2.2	5.1	2.2	0.1			9.6	5.2
SE	1.1	3.2	1.7	0.1			6.0	5.7
SSE	0.9	4.0	2.3	0.1	0.1		7.4	6.0
S	1.0	3.4	3.8	0.4			2.7	6.7
SSW	0.5	1.2	0.7	0.4			2.7	6.7
SW	0.6	0.6	0.5	0.2			1.9	5.8
WSW	1.1	1.3	0.2			0.1	2.6	4.0
W	1.1	3.4	1.5	0.3	0.1		6.4	5.9
WNW	1.3	2.8	2.1	0.4	0.1		6.7	6.2
NW	1.5	2.5	1.5	0.5	0.1		6.2	4.4
NNW	1.1	2.8	1.5	0.3	0.1		5.8	6.0
CALM							8.7	
TOTAL	19.4	45.9	22.5	2.9	0.5	0.1	100.0	5.1

Note:

1. Data based on hourly observations
2. This category occurred 1420 times or 28.8 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > -0.5 to 1.5 °C/100 m

Table 2-36

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY F

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

	Speed, Knots						Total	Mean
Sector	1-3	4-6	7-10	11-15	16-20	21+	Freq.	Speed
N	0.4	0.8	0.2				1.4	4.6
NNE	2.0	1.1	0.1				3.2	3.2
NE	2.4	2.0	0.2				4.6	3.6
ENE	4.6	4.3	0.9				9.8	3.9
E	7.3	7.3	1.4				16.0	3.9
ESE	2.9	3.5	0.9	0.2			7.5	4.5
SE	2.0	1.7	0.2				3.9	3.6
SSE	0.8	2.7	0.9				4.4	5.2
S	1.5	2.1	0.5	0.6			4.7	4.8
SSW	0.8	1.8	0.3	0.2			3.1	5.1
SW	1.1	0.1					1.2	1.9
WSW	1.2	0.9	0.3				2.4	3.9
W	2.0	1.4	0.6	0.1			4.1	4.2
WNW	1.8	2.0	0.7				4.4	4.4
NW	2.5	2.1	0.6				5.2	4.0
NNW	2.0	1.5	0.5	0.1			4.1	4.2
CALM							20.0	
TOTAL	35.2	35.3	8.3	1.2			100.0	3.3

Note:

1. Data based on hourly observations.
2. This category occurred 655 times or 13.3 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > 1.5 to 4.0 °C/100 m

Table 2-37

ANNUAL PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPEED, UPPER HALF TOWER, CATEGORY G

Adjusted Surface Winds, Arkansas Nuclear One Site (6-69 through 5-70)

Sector	1-3	4-6	Speed, Knots 7-10	11-15	16-20	21+	Total Freq.	Mean Speed
N	0.2						0.2	2.0
NNE	1.5	1.1					2.6	3.3
NE	3.4	1.1	0.4				4.9	3.2
ENE	2.4	2.4	0.2				5.0	3.7
E	7.3	7.1	0.4				14.8	3.6
ESE	4.3	2.8	0.2	0.2			7.5	3.6
SE	1.3	1.1					2.4	3.4
SSE	1.5	1.9	0.2				3.6	3.9
S	1.9	1.7	1.2	0.7			5.4	5.9
SSW	0.4	0.2	0.2				0.8	4.4
SW	0.9	0.9	0.2				2.0	4.0
WSW	1.5	1.7	0.4				3.6	4.1
W	1.3	1.7	1.6	0.2			4.8	5.7
WNW	2.4	1.4	0.7	0.4			4.9	4.7
NW	2.6	1.1	0.4	0.9			4.9	5.3
NNW	1.7	1.3	0.4	0.6			4.0	5.3
CALM								
TOTAL	34.6	27.5	6.5	3.0			100.0	3.0

NOTE:

1. Data based on hourly observations
2. This category occurred 465 times or 9.4 percent of the total observations
3. Total of all categories includes 4923 of 8760 possible observations
4. Temperature difference for this category is > 4.0 °C/100 m

Table 2-38

ANNUAL PERCENT FREQUENCY OF X/Q VERSUS DIRECTION SECTOR FOR 1046 METERS SURFACE WINDS, ARKANSAS NUCLEAR ONE SITE

AVERAGE HOURLY X/Q TIMES 10^{-4}, SEC PER CUBIC METER

WIND DIRECTION FROM	0.001 to 0.030	0.031 to 0.060	0.061 to 0.100	0.101 to 0.250	0.251 to 0.400	0.401 to 0.700	0.701 to 1.000	1.001 to 2.000	2.001 to 4.000	4.001 to 8.000	8000 Plus	TOTAL
N	0.23	0.24	0.13	0.25	0.12	0.41	0.30	0.56	0.38	0.04	0.02	2.67
NNE	0.38	0.24	0.20	0.33	0.46	0.45	0.53	0.40	0.27	0.11	0.02	3.38
NE	0.25	0.40	0.34	0.52	0.40	0.48	0.67	0.92	0.54	0.06	0	4.58
ENE	0.42	0.32	0.30	0.80	0.78	1.48	1.10	1.01	0.66	0.09	0	6.95
E	1.07	0.50	0.46	1.52	2.46	3.13	2.42	1.82	0.95	0.25	0.02	14.62
ESE	0.61	0.75	0.54	1.10	1.31	1.83	1.34	1.23	0.38	0.10	0	9.18
SE	0.67	0.28	0.37	1.03	1.32	0.96	0.51	0.75	0.39	0.07	0	6.35
SSE	0.88	0.46	0.47	0.89	0.98	1.27	0.79	0.69	0.24	0.07	0	6.73
S	1.09	0.31	0.45	0.99	1.34	1.32	0.61	0.55	0.22	0.06	0	6.93
SSW	0.47	0.32	0.13	0.31	0.52	0.46	0.23	0.22	0.09	0.05	0	2.81
SW	0.40	0.18	0.06	0.48	0.41	0.14	0.04	0.34	0.13	0.05	0	2.23
WSW	0.42	0.18	0.22	0.61	0.56	0.48	0.25	0.40	0.13	0.05	0	3.29
W	0.27	0.32	0.42	1.21	1.29	1.25	0.65	0.59	0.20	0.12	0	6.32
WNW	0.28	0.29	0.17	1.07	0.99	1.24	0.87	0.80	0.37	0.05	0.03	6.18
NW	0.23	0.17	0.30	0.59	0.81	1.12	0.84	0.86	0.44	0.07	0	5.43
NNW	0.25	0.24	0.26	0.60	0.80	1.14	0.43	0.58	0.36	0.06	0	4.73
CALM									7.62			
TOTAL	7.93	5.18	4.81	12.30	14.55	17.17	11.58	11.72	5.75	1.30	0.08	100.00

NOTES:
1. Based on 3.75 years of record, 670909-700531
2. Includes 4657 of 7964 possible observations
3. Observations based on 3-hourly recordings
4. Does not include dilution due to cavity diffusion
5. Sigma-Z is limited by monthly mean mixing depth

Table 2-39

ANNUAL PERCENT FREQUENCY OF X/Q VERSUS DIRECTION SECTOR FOR 6436 METERS SURFACE WINDS, ARKANSAS NUCLEAR ONE SITE

AVERAGE HOURLY X/Q TIMES 10^{-4}, SEC PER CUBIC METER

WIND DIRECTION FROM	0.001 to 0.030	0.031 to 0.060	0.061 to 0.100	0.101 to 0.250	0.251 to 0.400	0.401 to 0.700	0.701 to 1.000	TOTAL
N	1.14	0.54	0.56	0.29	0.13	0	0.02	2.67
NNE	1.77	0.82	0.36	0.27	0.14	0	0.02	3.38
NE	2.11	0.95	0.92	0.44	0.17	0	0	4.58
ENE	3.32	1.82	0.98	0.48	0.29	0	0	6.95
E	7.34	4.24	1.68	0.81	0.53	0	0.02	14.62
ESE	5.10	2.34	1.14	0.44	0.16	0	0	9.18
SE	4.01	1.13	0.70	0.28	0.22	0	0	6.35
SSE	4.28	1.46	0.59	0.23	0.17	0	0	6.73
S	4.68	1.42	0.54	0.17	0.12	0	0	6.93
SSW	1.89	0.55	0.20	0.09	0.08	0	0	2.81
SW	1.60	0.12	0.33	0.12	0.08	0	0	2.23
WSW	2.22	0.49	0.40	0.13	0.05	0	0	3.29
W	4.19	1.22	0.55	0.19	0.17	0	0	6.32
WNW	3.32	1.58	0.79	0.30	0.16	0	0.03	6.18
NW	2.62	1.42	0.72	0.43	0.24	0	0	5.43
NNW	2.58	1.15	0.54	0.26	0.20	0	0	4.73
CALM								7.62
TOTAL	52.21	21.25	10.98	4.94	2.92	0.	0.08	100.00

NOTES:

1. Based on 3.75 years of record, 670909-700531
2. Includes 4657 of 7964 possible observations
3. Observations based on 3-hourly recordings
4. Does not include dilution due to cavity diffusion
5. Sigma-Z is limited by monthly mean mixing depth

Table 2-40

WIND FREQUENCY DISTRIBUTION
(Frequency in Number of Occurrences)

ARKANSAS POWER & LIGHT
DATA PERIOD: JULY 29, 1971 THROUGH FEBRUARY 7, 1972

PASQUILL A (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	0	0	0	0	0	0	0	0	0	0	0	0	0	0.
NE	0	1	0	0	0	0	0	0	0	0	0	0	1	2.00
ENE	0	0	0	1	1	0	0	0	0	0	0	0	2	4.50
E	0	1	1	1	1	1	1	0	1	0	1	0	8	5.88
ESE	0	1	1	3	3	0	3	0	0	0	0	0	11	4.82
SE	0	0	0	0	1	0	0	0	1	0	0	0	2	7.00
SSE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
S	0	0	0	1	2	2	0	1	0	0	1	0	7	7.14
SSW	0	0	0	0	0	0	0	0	0	1	0	0	1	10.00
SW	0	0	2	0	1	0	0	1	1	0	0	0	5	5.60
WSW	0	1	1	0	1	0	0	0	0	0	0	0	3	3.33
W	1	2	2	2	2	1	1	0	0	0	1	4	16	6.94
WNW	0	0	1	0	0	1	1	2	1	0	0	3	9	9.44
NW	0	0	0	0	1	0	1	0	0	0	0	2	4	10.25
NNW	0	0	0	0	0	0	0	1	0	0	1	0	2	9.50
N	0	0	2	0	0	0	0	0	1	0	0	0	3	5.00
CALM													0	
TOTAL	1	6	10	9	13	5	7	5	5	1	3	10	75	6.64

Table 2-40 (continued)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
NE	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ENE	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
E	0	0	2	3	3	0	1	2	0	1	0	0	12	5.50
ESE	0	1	1	3	0	1	0	1	1	2	0	1	11	6.55
SE	0	0	3	2	1	0	1	0	0	0	0	0	7	4.14
SSE	0	0	0	1	3	1	1	2	0	0	0	0	8	6.00
S	0	0	1	1	1	0	2	0	0	2	1	1	9	7.75
SSW	0	0	0	0	0	2	0	0	0	0	0	0	2	6.00
SW	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
WSW	0	0	3	0	0	0	0	0	1	0	0	0	4	4.50
W	0	0	3	2	2	0	1	1	0	1	2	3	15	7.33
WNW	0	0	0	0	0	1	0	0	3	1	1	0	6	9.00
NW	0	0	0	0	0	0	1	0	0	0	0	1	2	9.50
NNW	0	0	0	1	0	0	0	0	0	0	0	0	1	4.00
N	0	0	0	0	0	0	0	0	0	0	0	0	0	0
CALM													0	
TOTAL	0	1	15	14	10	5	7	6	5	7	4	6	80	6.40

Table 2-40 (continued)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	0	0	0	0	0	0	0	2	0	0	0	0	2	8.00
NE	0	0	1	0	1	0	1	1	0	0	1	0	5	6.80
ENE	0	0	2	0	0	2	0	0	0	0	0	0	4	4.50
E	0	1	2	1	4	2	4	2	0	0	1	0	17	5.82
ESE	0	3	3	5	4	0	1	0	0	0	0	0	16	3.88
SE	0	0	2	5	0	2	1	1	2	0	0	0	13	5.46
SSE	0	0	0	2	2	2	1	1	0	1	0	0	9	6.11
S	0	1	1	1	0	0	1	2	1	1	0	1	9	7.22
SSW	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SW	0	0	1	0	0	0	0	0	0	0	0	0	1	3.00
WSW	0	1	2	2	0	0	1	0	0	0	0	0	6	3.83
W	0	5	3	3	4	1	0	2	0	0	0	2	20	5.55
WNW	0	1	1	0	1	2	0	0	2	0	0	2	9	7.89
NW	0	0	0	0	0	0	0	0	1	1	0	1	3	10.33
NNW	0	0	0	0	0	0	0	1	0	1	1	0	3	9.67
N	0	0	0	0	0	0	0	1	1	0	0	0	2	8.50
CALM													0	
TOTAL	0	12	18	19	16	11	10	13	7	4	3	6	119	5.92

Table 2-40 (continued)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	0	3	2	6	2	2	2	0	0	0	1	3	21	5.90
NE	0	0	6	6	2	6	4	2	1	4	1	0	32	6.00
ENE	1	5	13	9	11	10	5	4	1	1	1	1	62	5.27
E	8	13	30	31	47	27	26	12	13	18	9	21	255	6.28
ESE	0	7	16	23	15	13	18	11	8	9	5	9	134	6.38
SE	1	3	6	19	9	14	5	8	6	2	2	0	75	5.67
SSE	1	1	5	15	9	16	10	3	2	5	1	2	70	5.99
S	1	2	9	4	10	3	3	8	7	10	2	7	66	7.24
SSW	0	2	0	3	2	2	3	1	1	2	1	2	19	7.00
SW	1	1	4	2	3	3	1	1	1	5	1	5	28	7.36
WSW	2	5	3	3	1	5	0	2	0	0	1	1	23	4.70
W	2	9	12	9	9	13	12	12	5	7	4	35	129	9.03
WNW	2	5	3	1	7	5	6	7	10	6	6	32	90	10.23
NW	2	1	1	1	3	3	2	2	1	2	2	9	29	8.48
NNW	1	1	3	3	0	1	0	1	2	3	5	0	20	7.05
N	1	2	4	2	5	1	4	4	3	5	2	3	36	7.00
CALM													0	
TOTAL	23	60	117	137	135	124	101	78	61	79	44	130	1089	6.97

Table 2-40 (continued)

PASQUILL E (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	3	3	3	9	10	7	5	13	2	5	3	4	67	6.61
NE	6	6	11	12	9	8	10	7	5	7	2	0	83	5.48
ENE	2	5	18	22	25	30	14	13	6	5	1	2	143	5.60
E	11	14	20	52	65	83	63	47	48	24	21	57	505	7.18
ESE	1	7	9	23	28	34	34	33	31	31	12	20	263	7.53
SE	1	5	4	16	9	26	17	10	12	14	3	3	120	6.77
SSE	2	1	5	9	12	8	11	17	7	8	5	13	98	7.81
S	2	7	6	8	8	8	18	25	5	6	4	8	105	7.06
SSW	2	1	2	2	1	6	5	0	3	5	3	8	38	8.21
SW	1	0	1	1	1	0	2	2	2	3	2	12	27	10.81
WSW	9	3	3	5	4	0	0	0	0	0	3	3	30	4.90
W	8	12	14	24	18	8	13	13	11	7	9	38	175	7.78
WNW	3	6	9	12	9	15	12	13	13	15	16	38	161	8.75
NW	1	2	4	4	2	5	4	4	4	5	5	7	47	7.74
NNW	2	4	3	7	3	5	7	6	6	1	2	3	49	6.43
N	1	3	4	1	6	6	3	5	3	0	3	1	36	6.19
CALM													3	
TOTAL	55	79	116	207	210	249	218	208	158	136	94	217	1950	7.20

Table 2-40 (continued)

PASQUILL F (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	1	2	4	3	0	0	0	0	0	0	0	1	11	3.82
NE	2	6	8	10	6	2	0	1	2	1	0	0	38	4.11
ENE	4	6	13	17	21	17	12	2	1	0	0	1	94	4.83
E	5	5	11	26	38	46	21	17	10	7	2	3	191	5.87
ESE	3	2	7	10	11	7	5	7	3	6	4	2	67	6.21
SE	0	0	4	3	6	4	3	2	5	0	0	0	27	5.93
SSE	1	0	0	3	6	6	0	5	4	0	0	0	25	6.20
S	2	3	2	1	2	1	4	1	1	1	1	2	21	6.05
SSW	2	0	1	0	1	1	1	0	1	1	2	2	12	7.67
SW	5	2	2	1	1	0	0	0	0	0	0	3	14	4.71
WSW	5	4	1	0	0	1	0	0	1	0	1	0	13	3.23
W	8	8	3	8	10	6	1	2	1	1	0	3	51	4.53
WNW	2	5	3	4	6	8	2	1	3	0	0	1	35	5.03
NW	2	4	9	1	1	1	3	0	0	0	1	2	24	4.54
NNW	1	5	3	0	1	1	0	0	1	1	0	0	13	3.85
N	0	2	1	1	0	1	1	0	0	1	0	0	7	4.86
CALM													3	
TOTAL	43	54	72	88	110	102	53	38	33	19	11	20	646	5.31

Table 2-40 (continued)

PASQUILL G (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

	UPPER CLASS INTERVALS OF WIND SPEED (KTS)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	2	1	0	0	0	0	0	0	0	0	0	0	3	1.33
NE	3	3	3	1	2	0	0	0	1	1	0	1	15	4.20
ENE	3	1	7	19	6	3	1	0	0	1	0	2	43	5.23
E	7	6	10	15	19	17	9	9	4	0	2	1	99	5.16
ESE	4	4	4	9	5	2	2	1	0	2	0	2	35	4.83
SE	1	1	1	5	6	4	1	2	0	0	0	0	21	4.90
SSE	0	0	1	0	0	1	2	1	0	0	0	1	6	7.33
S	2	0	0	0	0	0	1	2	2	0	1	0	8	6.75
SSW	1	1	0	1	2	1	1	0	1	0	0	0	8	4.88
SW	0	6	2	0	1	0	0	1	0	0	0	1	11	4.18
WSW	1	5	2	3	1	0	0	0	0	0	0	0	12	2.83
W	6	6	9	9	4	7	4	1	1	0	0	2	49	4.41
WNW	2	5	3	2	3	3	5	0	0	1	0	2	26	5.12
NW	0	0	1	1	0	0	0	0	1	0	0	1	4	7.00
NNW	1	1	0	1	0	1	0	0	1	0	0	0	5	4.40
N	0	2	0	0	0	0	0	0	1	1	0	0	4	5.75
CALM													0	
TOTAL	33	42	43	66	49	39	26	17	12	6	3	13	349	4.91

Table 2-40 (continued)

PASQUILL A-G (FROM AEC/DELTA T CRITERIA, 195-30 FEET) 190-FOOT WINDS (UNREDUCED)

SECTOR	UPPER CLASS INTERVALS OF WIND SPEED (KTS) 1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	MEAN SPEED
NNE	6	9	9	19	12	9	7	15	2	5	4	8	105	6.03
NE	11	16	29	29	20	16	15	11	9	13	4	1	174	5.18
ENE	10	17	54	68	64	62	32	19	8	7	2	6	349	5.26
E	31	40	76	129	177	176	125	89	76	50	36	82	1087	6.51
ESE	8	25	41	76	66	57	63	53	43	50	21	34	537	6.72
SE	3	9	20	50	32	50	28	23	26	16	5	3	265	6.09
SSE	4	2	11	31	32	34	25	29	13	14	6	16	217	6.87
S	7	13	19	16	23	14	29	39	16	20	9	20	225	7.04
SSW	5	4	3	6	6	12	10	1	6	9	6	12	80	7.47
SW	7	9	13	4	7	3	3	5	4	8	3	21	87	7.40
WSW	17	19	15	13	7	6	1	2	2	0	5	4	91	4.20
W	25	42	46	57	49	36	32	31	18	16	16	87	455	7.27
WNW	9	22	20	19	26	35	26	23	32	23	23	78	336	8.48
NW	5	7	15	7	7	9	11	6	7	8	8	23	113	7.42
NNW	5	11	9	12	4	8	7	9	10	6	9	3	93	6.24
N	2	9	11	4	11	8	8	10	9	7	5	4	88	6.41
CALM													6	
TOTAL	155	254	391	540	543	535	422	365	281	252	162	402	4308	6.62

Table 2-41

WIND FREQUENCY DISTRIBUTION
(Frequency in Number of Occurrences)

ARKANSAS POWER & LIGHT
DATA PERIOD: FEBRUARY 7, 1972 - JULY 29, 1972

PASQUILL A (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
NE	0	0	1	1	0	0	1	1	0	2	0	0	6	7.00
ENE	0	0	0	2	2	0	0	0	1	1	0	0	6	6.17
E	0	0	1	9	6	1	4	1	3	2	3	7	37	7.46
ESE	0	0	1	1	3	4	1	0	0	0	1	2	13	7.08
SE	0	0	1	3	3	0	0	1	1	0	0	0	9	5.22
SSE	0	2	0	2	1	1	4	2	2	1	0	0	15	6.33
S	0	0	0	2	0	3	1	2	0	1	0	0	9	6.56
SSW	0	0	3	1	0	0	0	0	0	0	0	0	4	3.25
SW	0	0	0	0	0	1	1	0	0	0	0	1	3	8.33
WSW	0	0	0	2	0	0	2	1	1	0	1	1	8	7.88
W	0	1	1	2	2	2	2	2	6	4	3	11	36	9.47
WNW	0	1	0	0	0	0	0	1	1	1	2	12	18	12.56
NW	0	0	0	1	1	0	0	0	0	0	0	3	5	13.20
NNW	0	0	0	0	0	0	0	0	0	0	0	0	0	0
N	0	0	0	2	1	3	0	0	0	1	0	0	7	5.86
CALM													0	
TOTAL	0	4	8	28	20	15	16	11	15	13	10	37	177	8.07

Table 2-41 (continued)

PASQUILL B (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	1	1	0	0	0	1	0	0	2	0	0	0	5	5.40
NE	0	0	1	0	0	0	0	0	0	1	0	1	3	8.33
ENE	0	0	0	1	0	0	1	0	0	0	0	0	2	5.50
E	0	1	6	8	4	3	4	1	2	4	2	2	37	6.32
ESE	0	0	0	4	4	3	1	1	0	1	2	2	18	7.00
SE	0	0	3	3	5	2	1	1	2	0	0	0	17	5.35
SSE	0	0	0	0	3	2	3	4	3	0	0	0	15	7.13
S	0	0	1	1	3	2	2	1	2	0	0	2	14	7.14
SSW	0	0	0	0	1	1	2	0	0	0	1	0	5	7.20
SW	0	0	0	3	0	0	0	1	0	0	0	0	4	5.00
WSW	0	0	1	1	3	1	1	1	2	0	1	1	12	7.25
W	0	0	1	0	1	1	3	1	5	2	4	11	29	11.00
WNW	1	1	0	0	0	1	0	1	2	2	3	10	21	11.81
NW	0	0	0	0	0	0	0	0	0	1	0	2	3	17.33
NNW	0	0	0	0	1	0	0	1	0	0	0	0	2	6.50
N	0	0	1	0	2	0	1	0	0	0	0	0	4	5.00
CALM													0	
TOTAL	2	3	14	21	27	17	19	13	20	11	13	31	191	7.94

Table 2-41 (continued)

PASQUILL C (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	1	0	2	0	0	2	0	0	1	0	0	0	6	4.67
NE	0	0	3	0	3	0	1	1	0	0	0	0	8	4.88
ENE	0	0	1	3	2	1	3	0	0	0	0	0	10	5.20
E	0	3	7	9	11	5	5	3	5	1	4	5	58	6.47
ESE	0	0	1	5	6	0	4	1	2	3	1	3	26	7.50
SE	0	0	2	4	9	5	5	5	0	2	0	0	32	6.00
SSE	0	0	1	4	4	3	1	3	3	0	0	0	19	6.05
S	0	0	1	4	3	2	3	0	0	0	1	2	16	6.38
SSW	0	0	2	0	0	1	2	1	0	0	1	0	7	6.43
SW	0	0	1	1	1	0	0	0	0	0	0	4	7	9.86
WSW	0	0	0	4	1	2	1	0	0	0	0	2	10	6.80
W	0	0	0	2	1	3	2	1	3	1	1	6	20	9.50
WNW	1	1	0	0	1	1	0	2	1	0	0	16	23	11.83
NW	0	0	0	0	0	0	1	1	0	0	0	7	9	15.22
NNW	0	0	0	0	1	0	0	0	0	0	0	0	1	5.00
N	1	0	0	0	0	0	0	0	0	0	0	0	1	1.00
CALM													1	
TOTAL	3	4	21	36	43	25	28	18	15	7	8	45	254	7.42

Table 2-41 (continued)

PASQUILL D (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	2	3	5	2	1	3	4	3	1	2	2	0	28	5.61
NE	6	12	9	3	7	3	1	1	1	0	1	1	45	3.76
ENE	1	2	17	21	12	8	8	5	3	4	0	0	81	5.07
E	3	9	28	36	47	45	28	28	13	18	16	39	310	7.15
ESE	1	3	6	22	19	19	15	14	4	5	6	30	144	7.63
SE	1	1	9	18	23	16	9	7	14	4	2	2	106	6.08
SSE	0	1	3	14	13	15	9	5	6	3	0	0	69	5.93
S	2	1	2	6	6	7	10	7	6	7	8	3	65	7.51
SSW	0	2	2	1	5	2	6	4	5	8	3	8	46	8.41
SW	1	0	2	2	4	4	4	4	1	2	2	11	37	9.14
WSW	0	2	2	7	5	7	2	2	1	1	0	4	33	6.39
W	3	5	4	8	14	12	10	15	15	11	7	37	141	8.95
WNW	2	2	3	4	2	8	6	6	7	5	11	43	99	10.43
NW	0	2	2	1	2	2	2	5	4	4	2	15	41	10.20
NNW	0	2	0	0	3	0	2	1	2	2	2	6	20	9.40
N	3	2	3	0	2	3	6	2	4	1	0	0	26	5.69
CALM													2	
TOTAL	25	49	97	145	165	154	122	109	87	77	62	199	1293	7.41

Table 2-41 (continued)

PASQUILL E (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	6	5	5	9	6	17	4	4	1	2	0	0	59	4.88
NE	6	17	19	19	17	4	6	2	1	4	0	0	95	4.09
ENE	3	13	23	29	30	20	6	3	1	1	0	0	129	4.41
E	5	19	30	43	48	34	20	20	14	14	11	18	276	6.02
ESE	2	8	10	18	15	11	5	14	3	8	4	6	104	6.09
SE	3	4	14	16	13	12	8	13	4	4	3	0	94	5.56
SSE	2	3	8	9	15	13	10	10	1	2	1	0	74	5.55
S	4	4	6	6	13	6	11	4	2	1	1	1	59	5.34
SSW	2	2	1	7	6	6	7	9	7	4	1	3	55	6.85
SW	0	2	2	3	4	2	8	7	3	6	1	3	41	7.34
WSW	2	3	3	3	4	3	3	6	1	0	0	2	30	5.80
W	6	3	7	2	3	6	5	6	5	3	5	8	59	6.92
WNW	1	3	4	13	8	7	4	3	3	4	3	7	60	6.63
NW	2	4	1	3	0	6	4	1	3	4	0	7	35	7.69
NNW	1	2	5	6	3	4	6	2	4	1	3	8	45	7.36
N	4	8	12	11	8	10	1	5	2	1	0	1	63	4.56
CALM													3	
TOTAL	49	100	150	197	193	161	108	109	55	59	33	64	1281	5.73

Table 2-41 (continued)

PASQUILL F (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	10	9	3	3	2	0	0	0	0	0	0	0	27	2.19
NE	16	19	15	8	5	1	1	0	0	0	0	0	65	2.60
ENE	21	20	29	29	12	4	0	0	0	0	0	0	115	3.03
E	12	26	20	18	13	7	5	1	0	1	0	0	103	3.46
ESE	6	4	4	3	4	1	2	0	0	0	0	0	24	3.25
SE	2	4	2	4	3	0	0	1	0	0	0	0	16	3.44
SSE	0	1	2	2	3	1	0	0	0	1	0	0	10	4.70
S	2	2	0	5	2	4	1	0	0	0	0	0	16	4.19
SSW	6	0	1	4	1	0	0	0	0	1	0	1	14	3.79
SW	2	0	3	3	1	1	0	0	0	0	0	1	11	4.45
WSW	2	4	1	0	0	2	0	1	0	0	0	0	10	3.30
W	11	6	3	3	2	3	0	1	0	0	0	0	29	2.76
WNW	3	5	10	7	2	2	2	0	0	0	0	0	31	3.45
NW	3	5	4	2	1	0	3	2	4	0	0	0	24	4.63
NNW	3	4	2	1	1	0	0	0	0	0	0	0	11	2.36
N	3	4	1	4	0	1	0	0	0	0	0	0	13	2.77
CALM													15	
TOTAL	102	113	100	96	52	27	14	6	4	3	0	2	534	3.13

Table 2-41 (continued)

PASQUILL G (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	2	2	0	0	0	0	0	0	0	0	0	0	4	1.50
NE	5	9	2	2	0	0	0	0	0	0	0	0	18	2.06
ENE	12	11	9	7	0	1	0	1	0	0	0	0	41	2.51
E	23	26	12	6	8	5	0	1	1	0	0	0	82	2.71
ESE	6	0	1	1	0	0	0	0	0	0	0	0	8	1.63
SE	4	1	0	1	0	0	0	0	0	0	0	0	6	1.67
SSE	1	0	0	0	2	0	0	0	0	0	0	0	3	3.67
S	7	1	1	0	0	2	0	0	0	0	0	0	11	2.18
SSW	3	1	0	0	0	0	0	0	0	0	1	0	5	3.20
SW	1	0	1	0	0	0	0	0	0	0	0	0	2	2.00
WSW	7	2	0	0	0	0	0	0	0	0	0	0	9	1.22
W	11	10	2	1	1	0	0	0	0	0	0	0	25	1.84
WNW	9	3	3	1	3	0	0	0	0	0	0	0	19	2.26
NW	5	3	2	1	0	0	0	0	0	0	0	0	11	1.91
NNW	5	2	0	0	0	1	0	0	0	0	0	0	8	1.88
N	7	3	1	0	0	0	0	0	0	0	0	0	11	1.45
CALM													25	
TOTAL	108	74	34	20	14	9	0	2	1	0	1	0	288	2.08

Table 2-41 (continued)

PASQUILL A-G (FROM AEC/DELTA T CRITERIA, 195-30 FEET) WINDS AT THE 40-FOOT LEVEL

	MID-POINT OF WIND SPEED INTERVAL (MPH)													MEAN
SECTOR	1	2	3	4	5	6	7	8	9	10	11	> 11	TOTAL	SPEED
NNE	22	20	15	14	10	23	8	7	5	4	2	0	130	4.38
NE	33	57	50	33	32	8	10	5	2	7	1	2	240	3.63
ENE	37	46	79	92	58	34	18	9	5	6	0	0	384	3.99
E	43	84	104	129	137	100	66	55	38	40	36	71	903	5.91
ESE	15	15	23	54	51	38	28	30	9	17	14	43	337	6.63
SE	10	10	31	49	56	35	23	28	21	10	5	2	280	5.58
SSE	3	7	14	31	41	35	27	24	15	7	1	0	205	5.83
S	15	8	11	24	27	26	28	14	10	9	10	8	190	6.08
SSW	11	5	9	13	13	10	17	14	12	13	7	12	136	6.82
SW	4	2	9	12	10	8	13	12	4	8	3	20	105	7.68
WSW	11	11	7	17	13	15	9	11	5	1	2	10	112	5.78
W	31	25	18	18	24	27	22	26	34	21	20	73	339	7.81
WNW	17	16	20	25	16	19	12	13	14	12	19	88	271	8.59
NW	10	14	9	8	4	8	10	9	11	9	2	34	128	8.39
NNW	9	10	7	7	9	5	8	4	6	3	5	14	87	6.64
N	18	17	18	17	13	17	8	7	6	3	0	1	125	4.09
CALM													46	
TOTAL	289	347	424	543	514	408	307	268	197	170	127	378	4018	5.98

Table 2-42

ARKANSAS POWER & LIGHT DATA RECOVERY
(hours)

Month	Total Available	190' ΔT Missing	Both ΔT Missing	Temp. Missing	40' WD Missing	40' WS Missing	190' WD Missing	190' WS Missing
July 71	62	0	0	0	0	62	0	0
Aug 71	744	260	21	23	0	744	0	97
Sept. 71	720	190	27	30	4	720	4	39
Oct. 71	744	29	0	42	0	744	5	0
Nov. 71	720	247	5	70	195	720	5	9
Dec. 71	744	7	1	17	30	744	7	77
Jan. 72	744	135	0	35	2	744	4	53
Feb. 72	696	15	12	38	12	156	1	84
March 72	744	2	2	13	0	0	0	139
Apr. 72	720	27	0	16	0	0	28	101
May 72	744	102	98	101	0	0	28	15
June 72	720	8	8	5	0	0	46	0
July 72	744	10	5	6	0	0	0	0
TOTAL	8846	1032	179	396	243	4634	128	614
% Data Recovery	___	88.3	98.0	95.5	97.3	47.6	98.6	93.1

Table 2-43

**SUSTAINED PERIODS OF MISSING DATA
(Times Inclusive)**

190' - 30' ΔT

0700	8/1/71	-	1200	8/2/71
0000	8/11/71	-	1100	8/13/71
1600	8/13/71	-	0900	8/16/71
1700	8/27/71	-	1200	8/30/71
1500	9/13/71	-	1400	9/14/71
1900	9/22/71	-	1300	9/28/71
0800	11/10/71	-	0900	11/20/71
0600	1/4/71	-	0900	1/6/72
0000	1/15/72	-	1500	1/17/72
0200	4/22/72	-	0100	4/23/72
1000	5/18/72	-	1100	5/22/72

190'-30' ΔT and 85'-30' ΔT Simultaneously

1000	5/18/72	-	1100	5/22/72

40' Wind Direction

0400	11/21/71	-	0900	11/26/71
1200	12/2/71	-	1100	12/3/71

40' Wind Speed

(1000	7/29/71	-	1100	2/7/72 ?)
0800	11/23/71	-	1000	11/24/71
1000	11/25/71	-	0900	11/26/71
0800	1/4/72	-	1300	1/5/72

190' Wind Direction

1800	6/17/72	-	1500	6/19/72

190' Wind Speed

2100	8/15/71	-	1300	8/17/71
1200	8/30/71	-	1200	8/31/71
1600	12/3/71	-	1200	12/6/71
0700	1/4/72	-	1100	1/6/72

Table 2-44

PERCENT FREQUENCY OF PASQUILL CLASSES

	Pasquill Classes						
	A	B	C	D	E	F	G
June 1969 - May 1970[(1)]	10.5	1.9	1.5	8.0	28.1	20.3	29.7
July 29, 1971 - July 29, 1972[(2)]	3.0	3.3	4.5	28.6	38.8	14.2	7.6

(1) Delta temperatures observed through layer 5 - 190 feet above grade.

(2) Delta temperatures observed through layer 30 - 190 feet above grade.

Table 2-45

**AVERAGE ANNUAL RELATIVE CONCENTRATION (SEC./CUBIC METER)
PER SECTOR PER DISTANCE IN METERS FROM PLANT SITE
(MODIFIED POWER LAW USED ON 190-FOOT WINDS)**

WINDS REDUCED FROM 190 FEET TO 40 FEET USING POWER LAW
EXPONENTS FOR PASQUILL CLASS A, B, C, & D = 0.18, FOR E = 0.225, FOR F & G = 0.45

PERIOD OF RECORD: JULY 29, 1971 - JULY 29, 1972

DISTANCE IN METERS

DIRECTION AFFECTED	100	500	1046	2000	3000	5000	6436	8000
NNE	1.7E-04	9.0E-06	2.4E-06	8.3E-07	4.3E-07	2.0E-07	1.4E-07	1.0E-07
NE	1.6E-04	8.5E-06	2.3E-06	7.9E-07	4.1E-07	1.9E-07	1.4E-07	9.8E-08
ENE	2.6E-04	1.4E-05	3.8E-06	1.3E-06	6.7E-07	3.1E-07	2.2E-07	1.6E-07
E	4.0E-04	2.2E-05	5.9E-06	2.0E-06	1.0E-06	4.8E-07	3.4E-07	2.4E-07
ESE	4.1E-04	2.2E-05	5.9E-06	2.0E-06	1.1E-06	4.9E-07	3.4E-07	2.5E-07
SE	1.8E-04	9.8E-06	2.7E-06	9.1E-07	4.8E-07	2.2E-07	1.6E-07	1.1E-07
SSE	1.8E-04	9.4E-06	2.6E-06	8.8E-07	4.6E-07	2.1E-07	1.5E-07	1.1E-07
S	1.6E-04	8.6E-06	2.3E-06	8.0E-07	4.2E-07	1.9E-07	1.3E-07	9.8E-08
SSW	2.2E-04	1.2E-05	3.2E-06	1.1E-06	5.7E-07	2.7E-07	1.9E-07	1.4E-07
SW	3.9E-04	2.1E-05	5.7E-06	1.9E-06	1.0E-06	4.7E-07	3.3E-07	2.4E-07
WSW	7.1E-04	3.8E-05	1.0E-05	3.6E-06	1.9E-06	8.7E-07	6.1E-07	4.4E-07
W	8.9E-04	4.8E-05	1.3E-05	4.4E-06	2.3E-06	1.1E-06	7.4E-07	5.4E-07
WNW	5.4E-04	2.9E-05	7.9E-06	2.7E-06	1.4E-06	6.4E-07	4.5E-07	3.2E-07
NW	2.7E-04	1.4E-05	3.9E-06	1.3E-06	6.9E-07	3.2E-07	2.2E-07	1.6E-07
NNW	2.0E-04	1.0E-05	2.8E-06	9.6E-07	4.9E-07	2.3E-07	1.6E-07	1.1E-07
N	2.2E-04	1.2E-05	3.2E-06	1.1E-06	5.6E-07	2.6E-07	1.8E-07	1.3E-07
AVERAGE				4.87E-6			2.82E-7	

Table 2-45 (continued)

DISTANCE IN KILOMETERS

DIRECTION AFFECTED	10	20	30	40	50	60	70	80
NNE	7.5E-08	2.9E-08	1.7E-08	1.2E-08	8.8E-09	6.8E-09	5.6E-09	4.7E-09
NE	7.2E-08	2.8E-08	1.6E-08	1.1E-08	8.4E-09	6.5E-09	5.3E-09	4.5E-09
ENE	1.2E-07	4.5E-08	2.6E-08	1.8E-08	1.4E-08	1.1E-08	8.6E-09	7.3E-09
E	1.8E-07	7.0E-08	4.0E-08	2.8E-08	2.1E-08	1.6E-08	1.3E-08	1.1E-08
ESE	1.8E-07	7.1E-08	4.1E-08	2.8E-08	2.1E-08	1.6E-08	1.3E-08	1.1E-08
SE	8.4E-08	3.3E-08	1.9E-08	1.3E-08	9.8E-09	7.6E-09	6.2E-09	5.2E-09
SSE	8.0E-08	3.1E-08	1.8E-08	1.3E-08	9.4E-09	7.3E-09	6.0E-09	5.1E-09
S	7.2E-08	2.8E-08	1.6E-08	1.1E-08	8.3E-09	6.5E-09	5.3E-09	4.5E-09
SSW	9.9E-08	3.9E-08	2.2E-08	1.5E-08	1.2E-08	9.0E-09	7.4E-09	6.2E-09
SW	1.8E-07	6.9E-08	4.0E-08	2.8E-08	2.1E-08	1.6E-08	1.3E-08	1.1E-08
WSW	3.3E-07	1.3E-07	7.4E-08	5.1E-08	3.8E-08	3.0E-08	2.4E-08	2.0E-08
W	4.0E-07	1.5E-07	8.9E-08	6.1E-08	4.6E-08	3.5E-08	2.9E-08	2.4E-08
WNW	2.4E-07	9.2E-08	5.3E-08	3.6E-08	2.7E-08	2.1E-08	1.7E-08	1.5E-08
NW	1.2E-07	4.5E-08	2.6E-08	1.8E-08	1.3E-08	1.0E-08	8.4E-09	7.1E-09
NNW	8.3E-08	3.2E-08	1.8E-08	1.3E-08	9.4E-09	7.3E-09	6.0E-09	5.0E-09
N	9.6E-08	3.7E-08	2.2E-08	1.5E-08	1.1E-08	8.6E-09	7.1E-09	5.9E-09

TOTAL NUMBER OF VALID OBSERVATION = 8309

NUMBER OF INVALID OBSERVATIONS = 475

THERE WERE 52 CALMS THAT WERE EQUALLY DISTRIBUTED OVER EACH SECTOR

Table 2-46

WIND SPEED FREQUENCIES DURING STABLE CONDITIONS FOR THE YEAR AUGUST 1971 THROUGH JULY 1972 AND FOR THE 5-YEAR PERIOD 1966 THROUGH 1970, AT LITTLE ROCK, ARKANSAS, AIRPORT

		WIND SPEED, kts.			
PERIOD OF RECORD	CALM	1-3	4-6	7-10	TOTAL
Aug. 71 - Jul. 72	5.7	5.2	20.9	6.8	38.6
1966 – 1970	4.0	3.8	21.6	7.3	36.7

Percentage frequencies during Pasquill E & F conditions, as defined by the Pasquill-Turner method.

Table 2-47

WIND SPEED, PASQUILL CLASS FREQUENCY DISTRIBUTIONS AT LITTLE ROCK, ARKANSAS, AIRPORT FOR 1-YEAR AND 5-YEAR PERIODS

(Pasquill classed by the Pasquill-Turner method)

August 1971 - July 1972

				WIND SPEEDS, kts.				
PASQUILL Class	CALM	1-3	4-6	7-10	11-16	17-21	21	TOTAL
A	2	1	20	0	0	0	0	23
B	1	27	117	68	0	0	0	213
C	7	10	85	211	24	0	0	213
D	15	30	280	501	361	28	6	1221
E,F	165	153	609	199	0	0	0	1126
TOTAL	190	211	1111	979	385	28	6	2920

1966 - 1970

PASQUILL Class	CALM	1-3	4-6	7-10	11-16	17-21	21	TOTAL
A	24	15	88	0	0	0	0	127
B	24	81	499	339	0	0	0	943
C	32	45	454	1194	159	6	0	1890
D	58	95	1201	2804	1951	163	8	6280
E,F	587	557	3151	1066	0	0	0	5359
TOTAL	723	793	5393	5403	2110	169	8	14,599

2.11 FLOOD EVALUATION

2.11.1 MAXIMUM PROBABLE FLOOD

Security-Related Information
Text Withheld Under 10 CFR 2.390

2.11.2 FAILURE OF UPSTREAM DAMS

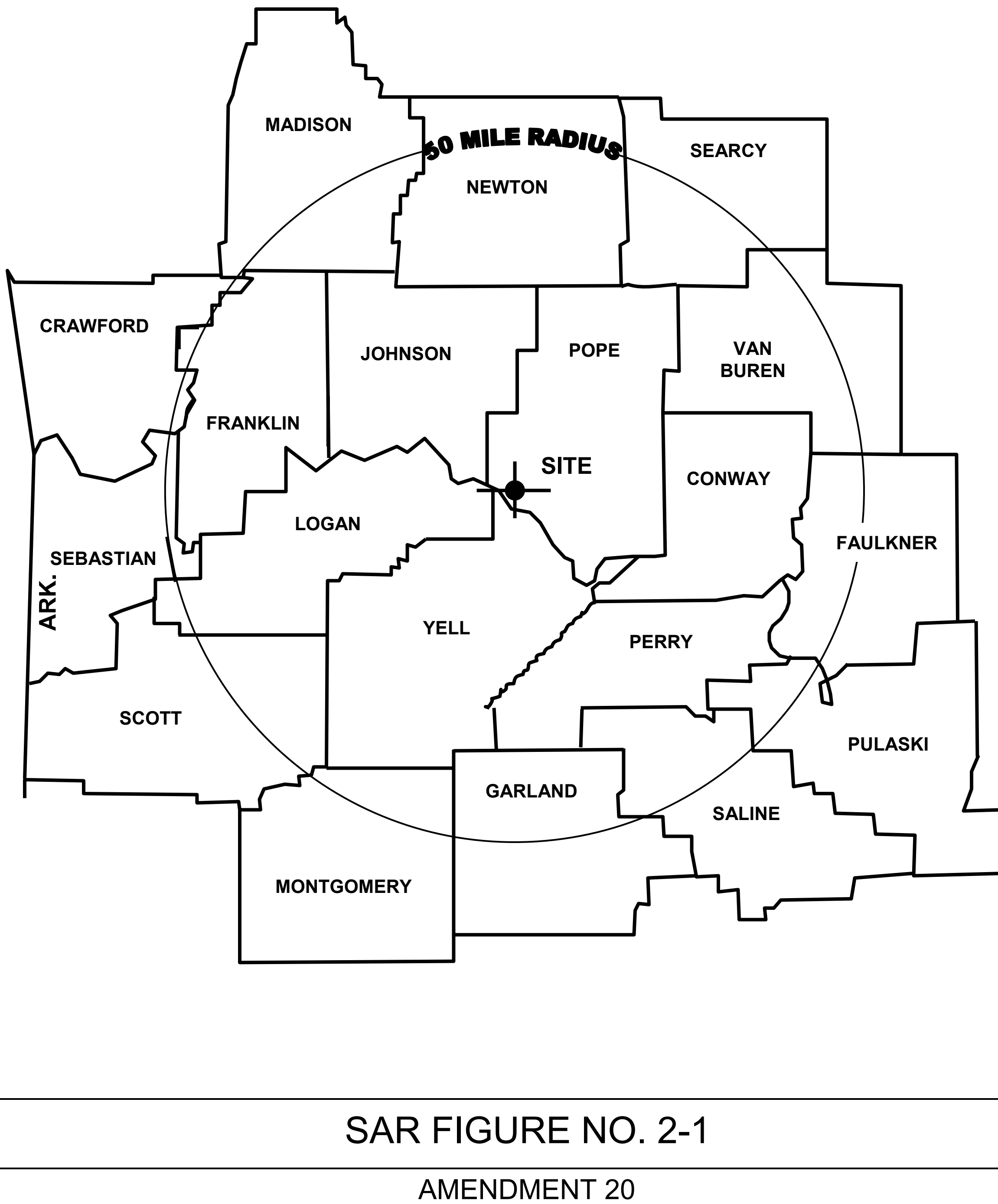
Security-Related Information
Text Withheld Under 10 CFR 2.390

Security-Related Information
Text Withheld Under 10 CFR 2.390

2.11.3 DESIGN FLOOD EVALUATION

Security-Related Information
Text Withheld Under 10 CFR 2.390

COUNTIES
WITHIN A 50-MILE RADIUS

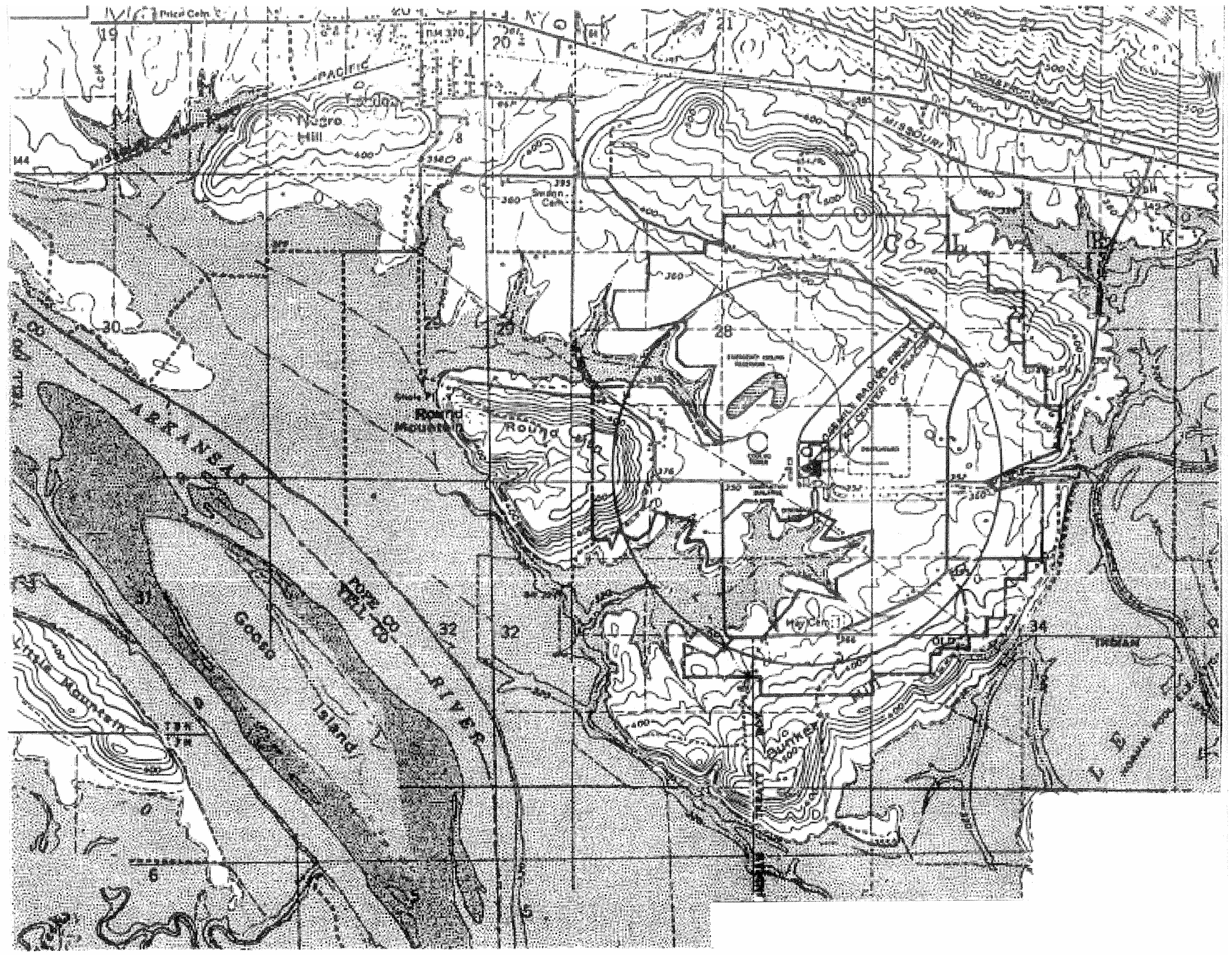


SAR FIGURE NO. 2-1

AMENDMENT 20

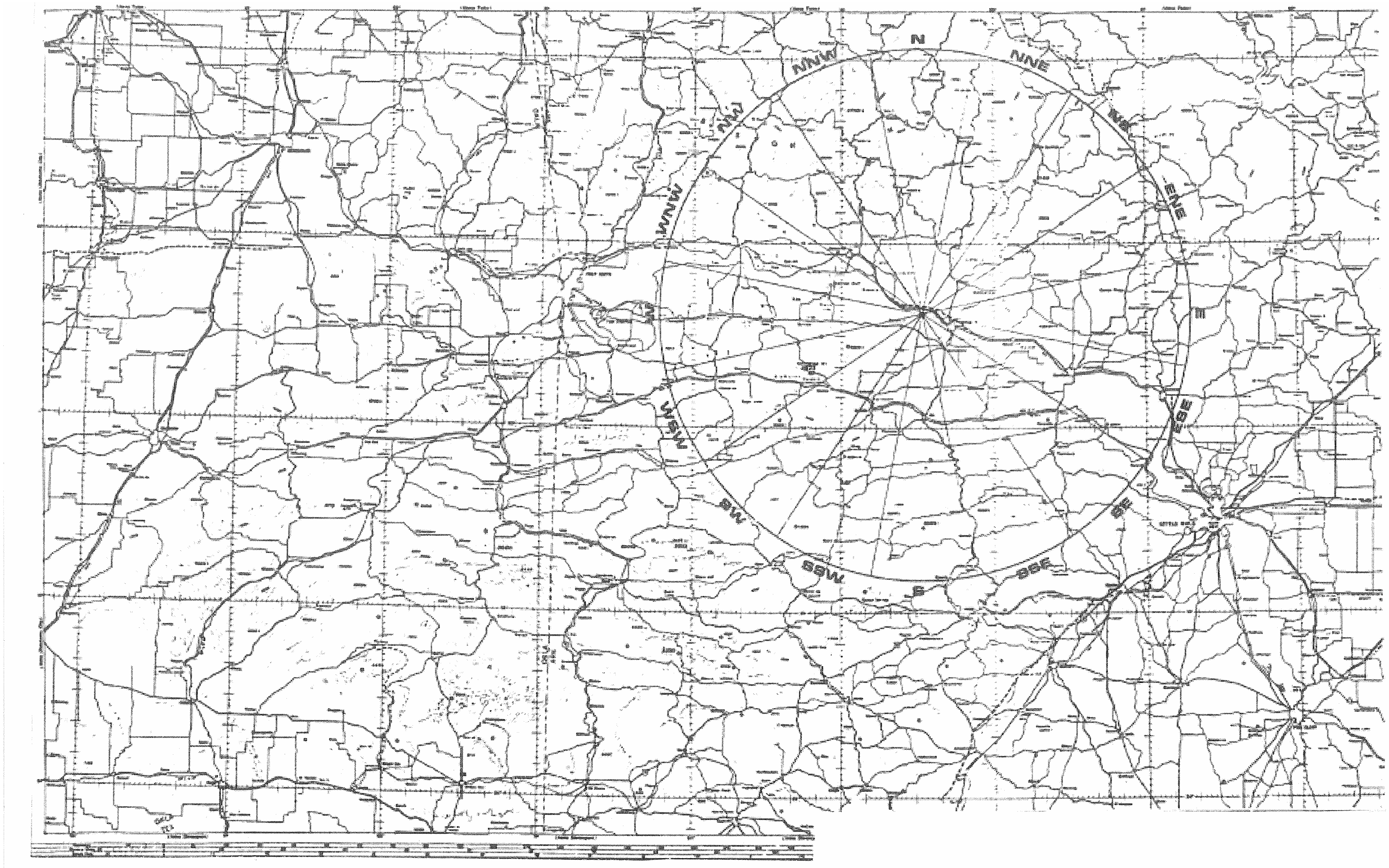
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

COUNTIES WITHIN A 50 MILE RADIUS	BASED ON DRAWING NO	SHEET	REV.



PLAN PLOT AND SITE BOUNDARY			SAR FIGURE NO. 2-2
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



GENERAL AREA MAP

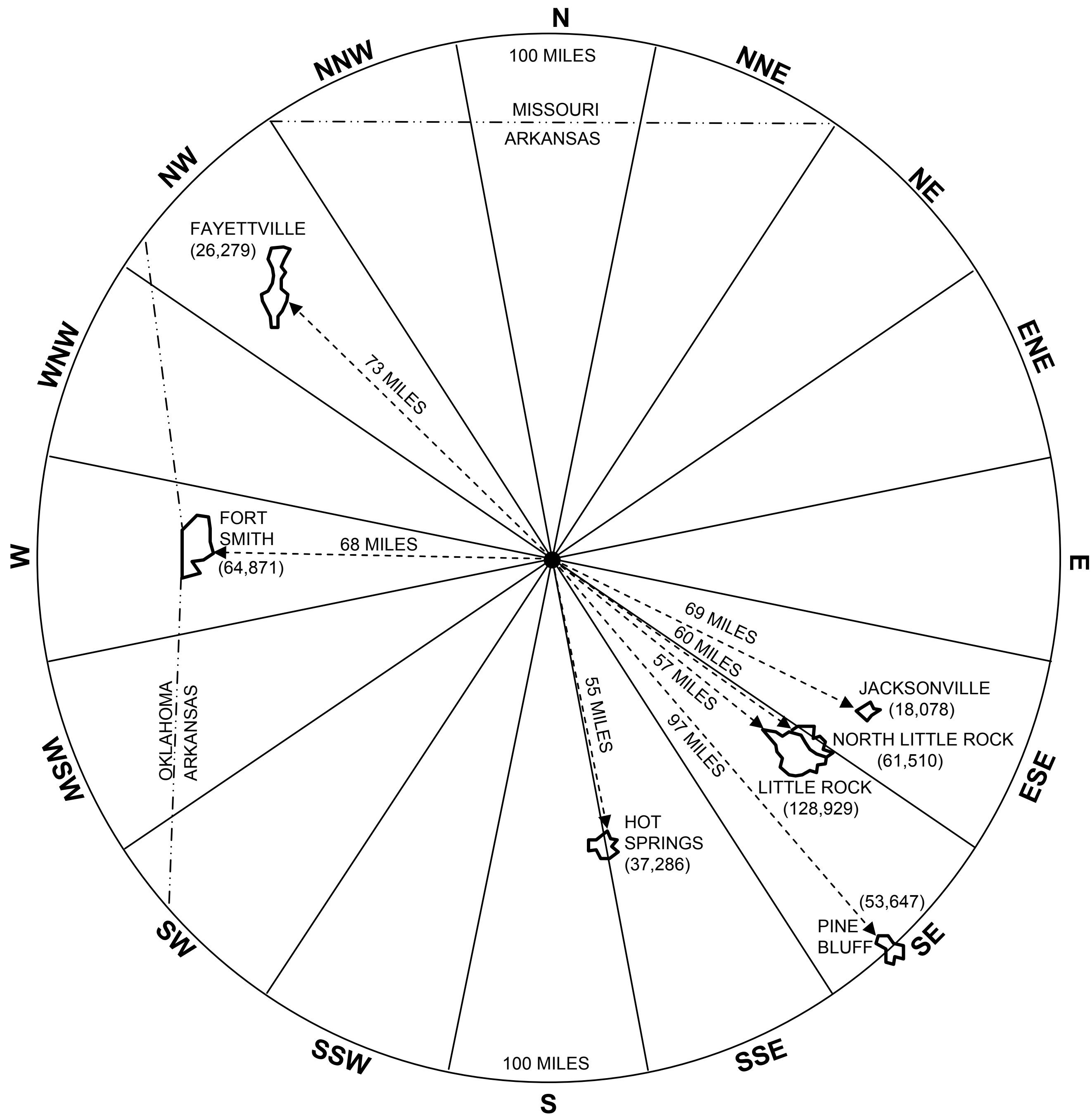
SAR FIGURE NO. 2-3

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	ENTERGY
		DESIGN:	ENTERGY
		CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

POPULATION CENTERS WITHIN A 100-MILE RADIUS

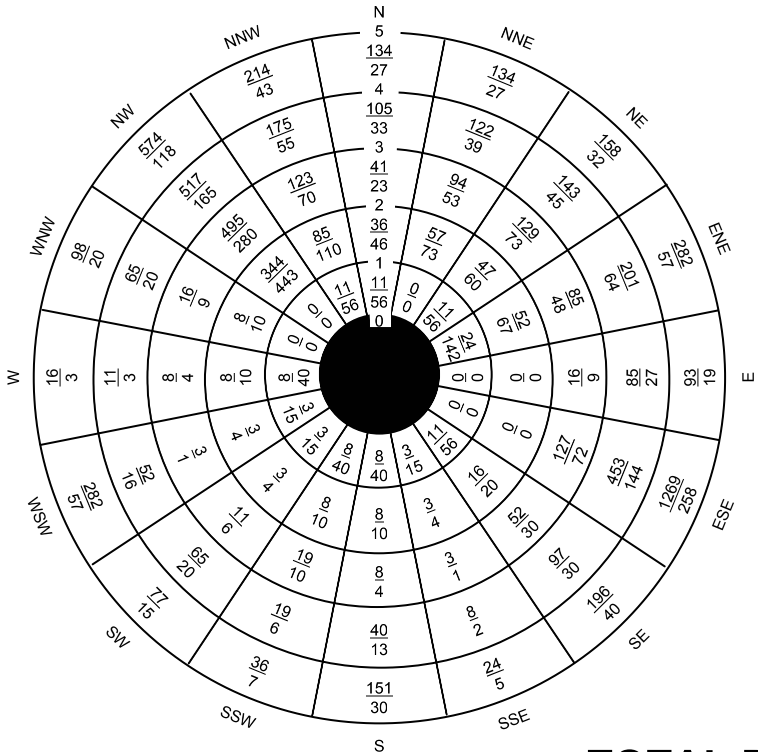


SAR FIGURE NO. 2-4

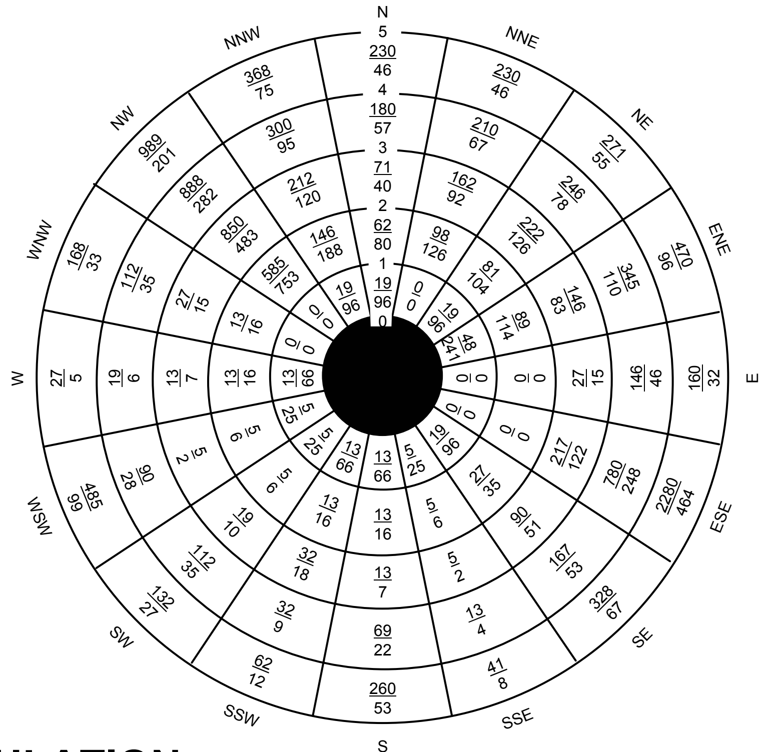
AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
POPULATION CENTERS WITHIN A 100 MILE RADIUS	BASED ON DRAWING NO	SHEET / REV.

1967
0 – 5 MILES



2012
0 – 5 MILES

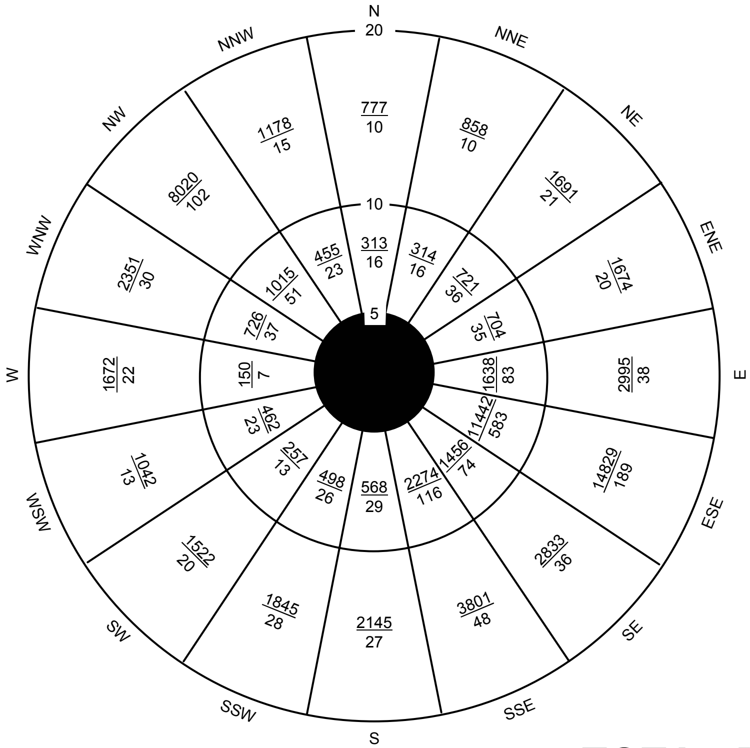


<u>TOTAL POPULATION</u>
PERSONS/SQ. MILE

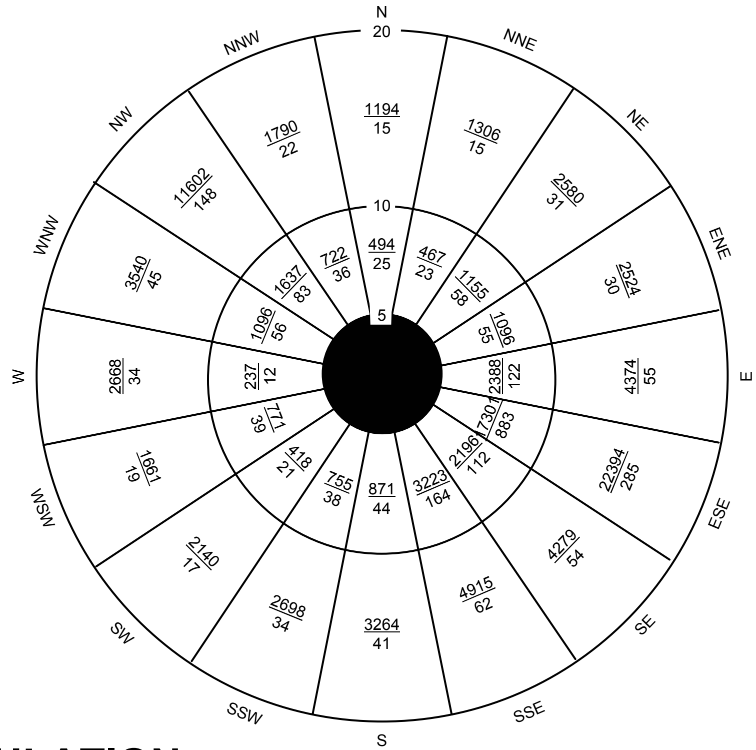
TOTAL POPULATION IS CUMULATIVE FROM THE CENTER

ESTIMATED POPULATION DISTRIBUTION (1967-2012) FROM 0 – 5 MILE RADII			SAR FIGURE NO. 2-5
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.

1967
5 – 20 MILES



2012
5 – 20 MILES

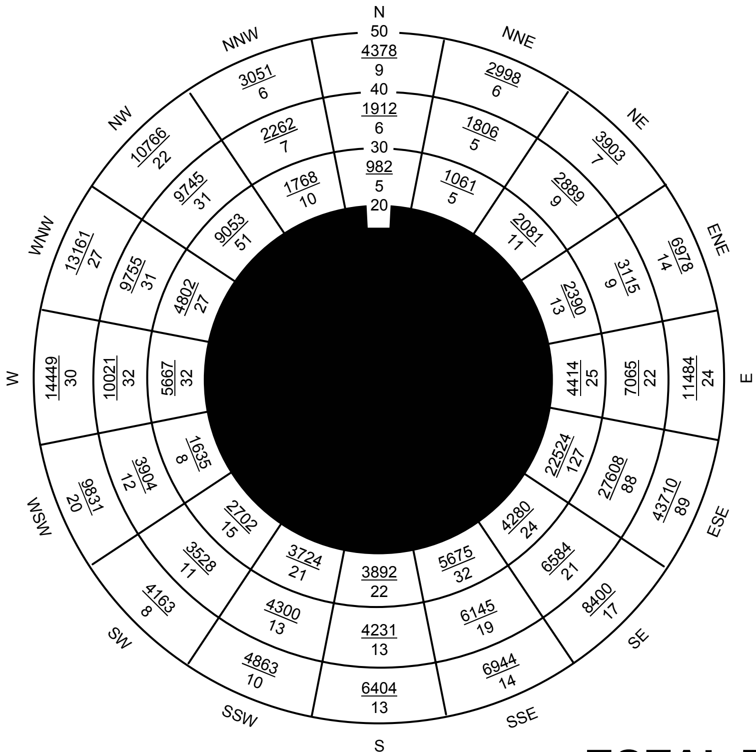


TOTAL POPULATION
PERSONS/SQ. MILE

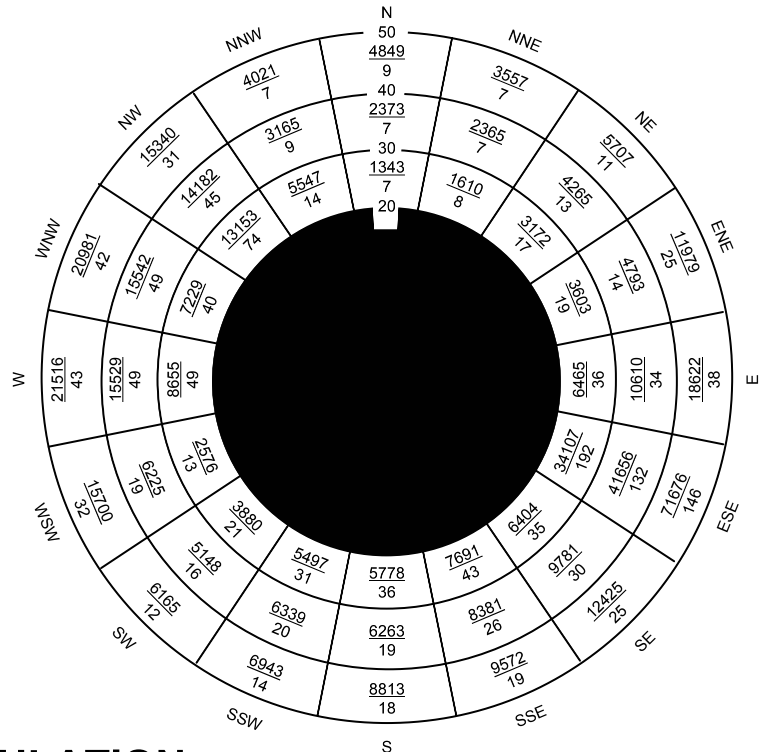
TOTAL POPULATION IS CUMULATIVE FROM THE CENTER

ESTIMATED POPULATION DISTRIBUTION (1967-2012) FROM 5 – 20 MILE RADII			SAR FIGURE NO. 2-6		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
			BASED ON DRAWING NO	SHEET	REV.

1967
20 – 50 MILES



2012
20 – 50 MILES



TOTAL POPULATION
PERSONS/SQ. MILE

TOTAL POPULATION IS CUMULATIVE FROM THE CENTER

ESTIMATED POPULATION DISTRIBUTION (1967-2012) FROM 20 – 50 MILE RADII

SAR FIGURE NO. 2-7

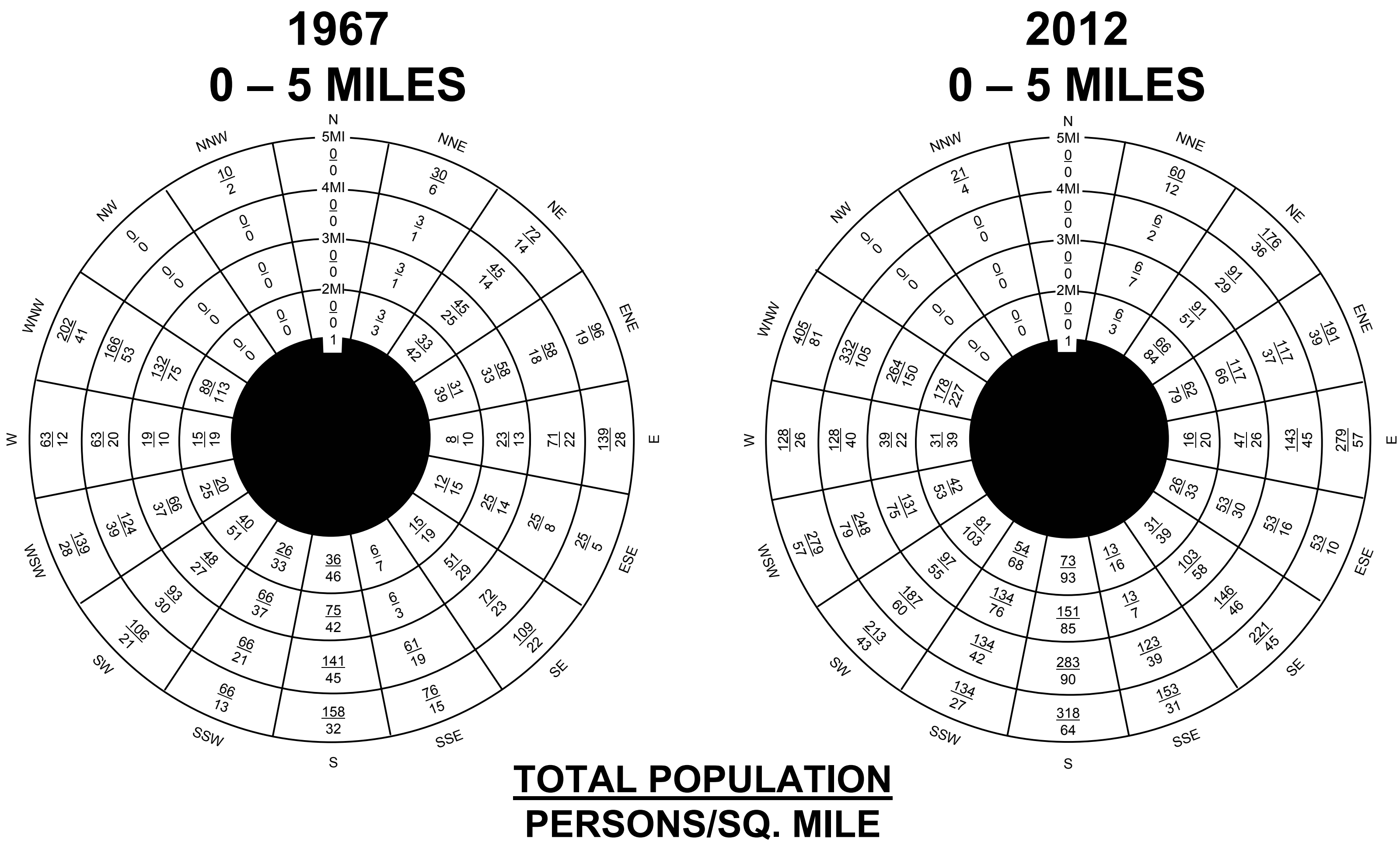
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



TOTAL POPULATION IS CUMULATIVE FROM THE CENTER

ESTIMATED POPULATION DISTRIBUTION (1967-2012) FROM 0 – 5 MILE RADII	SAR FIGURE NO. 2-8
	AMENDMENT 20

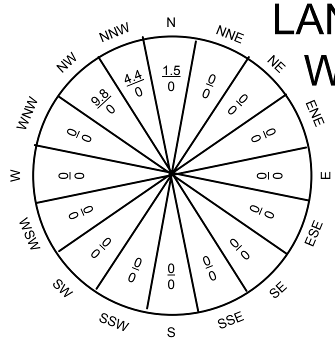
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

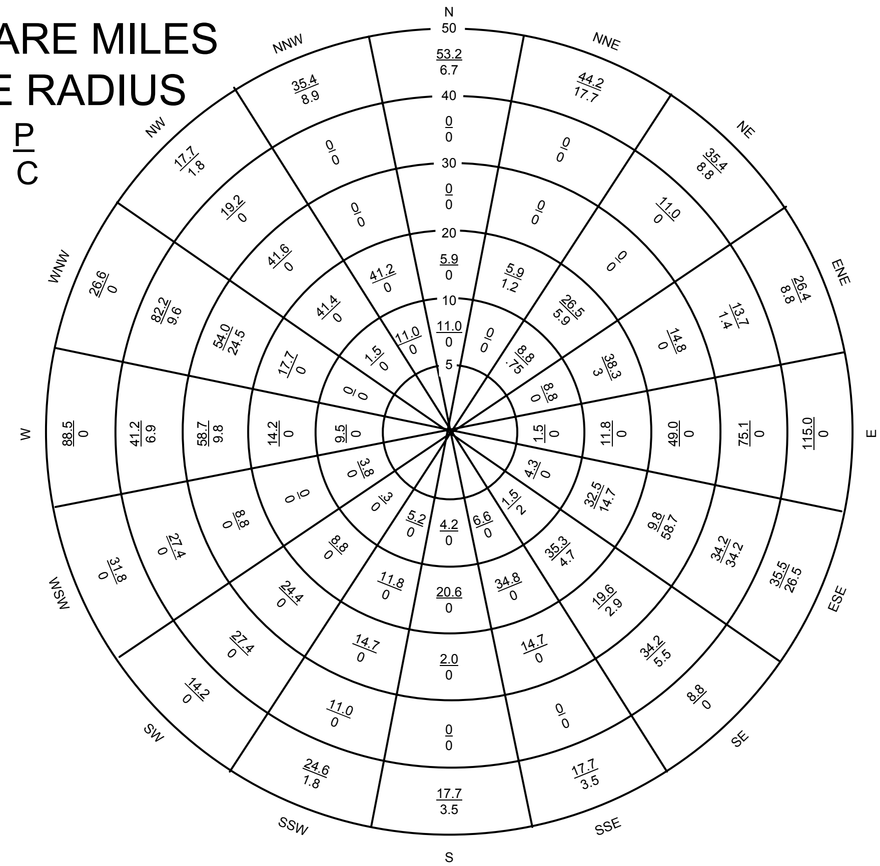
LAND USE IN SQUARE MILES WITHIN A 50 MILE RADIUS

PASTURED P
CULTIVATED C

0 – 5 MILES



5 – 50 MILES



CUMULATIVE TOTALS BY SECTORS

SECTOR	RADIUS IN MILES									
	10		20		30		40		50	
	P	C	P	C	P	C	P	C	P	C
	SQ.MI.		SQ.MI.		SQ.MI.		SQ.MI.		SQ.MI.	
N	12.5	0	18.4	0	18.4	0	18.4	0	71.6	6.7
NNW	15.4	0	56.6	0	55.6	0	56.6	0	92.0	8.9
NW	2.5	0	43.9	0	85.5	0	104.7	0	122.4	1.8
WNW	0	0	17.7	0	71.7	24.5	153.9	34.1	180.5	34.1
W	9.5	0	23.7	0	82.4	9.8	123.6	16.7	212.1	16.7
WSW	3.8	0	3.8	0	12.6	0	40.0	0	71.8	0
SW	0.3	0	9.1	0	33.5	0	60.9	0	75.1	0
SSW	5.2	0	17.0	0	31.7	0	42.7	0	67.3	1.8
S	4.3	0	24.9	0	26.9	0	26.9	0	44.6	3.5
SSE	6.6	0	41.4	0	56.1	0	56.1	0	73.8	3.5
SE	1.5	.2	36.8	4.9	56.4	7.8	90.6	13.3	99.4	13.3
ESE	4.3	0	36.8	14.7	46.6	73.4	80.8	107.6	116.3	142.8
E	1.5	0	13.3	0	62.3	0	137.4	0	252.4	0
ENE	8.8	0	47.1	3	61.9	3	75.6	4.4	102	13.2
NE	8.8	.75	35.3	6.7	35.3	6.7	46.3	6.7	81.7	15.5
NNE	0	0	5.9	1.2	5.9	1.2	5.9	1.2	50.1	18.9

LAND USE (PASTURED/CULTIVATED) WITHIN A 50 MILE RADIUS

SAR FIGURE NO. 2-9

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

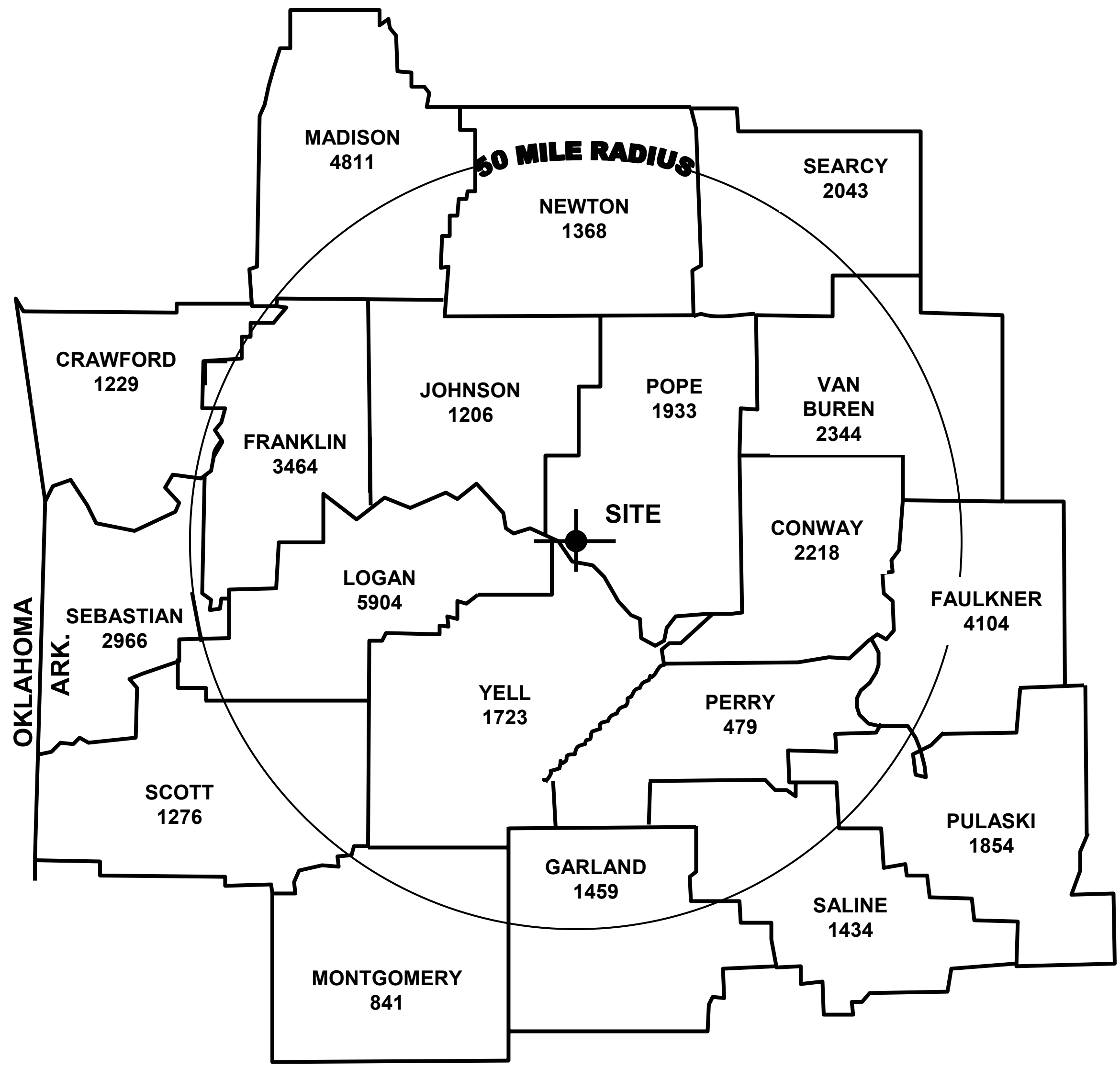
Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

DAIRY ANIMALS
WITHIN A 50-MILE RADIUS



SAR FIGURE NO. 2-10

AMENDMENT 20

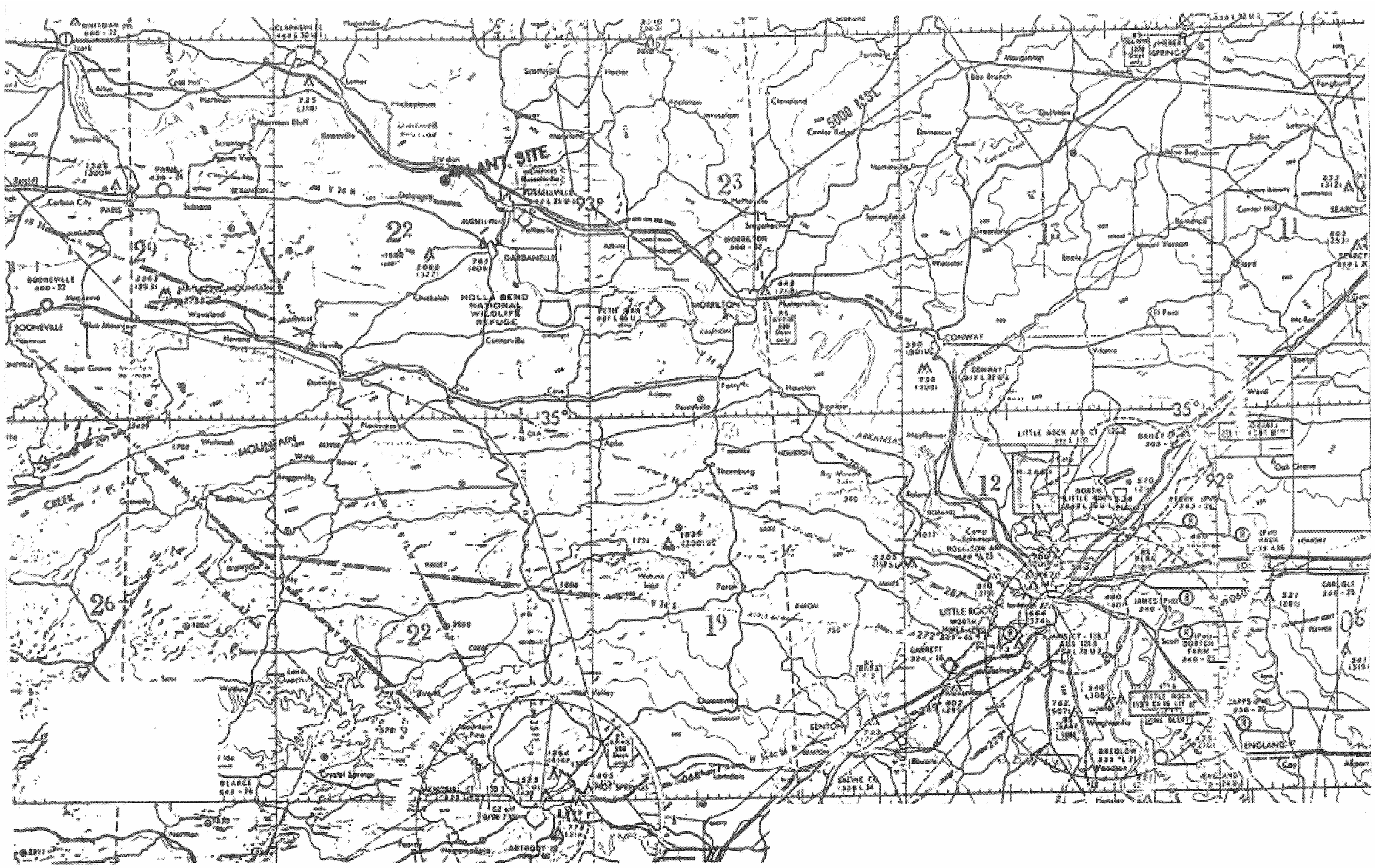
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

DAIRY ANIMALS WITHIN A 50 MILE RADIUS

BASED ON DRAWING NO	SHEET	REV.



AIRLINE ROUTES IN THE VICINITY OF RUSSELLVILLE

SAR FIGURE NO. 2-11

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

RUSSELLVILLE NUCLEAR STATION
DATA ON RESERVOIRS AND LAKES WITHIN A 50-MILE RADIUS
(MINIMUM SRFACE AREA – 100 ACRES)

No.	Reservoir-Lake (River)	Owner/Agency	Distance (miles)	Direction	Purpose[1]	Surface Area (acres)	Total Storage (10^3 ac-ft)	Dead Storage Surface (acres)	Dead Storage Volume (10^3 ac-ft)	Discharge Yearly Ave. (10^3 ac-ft/yr)
1.	Dardanelle Res (Arkansas R)	Corps of Engineers	0	S-WNW	N,P	34,300[2]	486.2	31,000[3]	420.8[4]	28,000
2.	Lake Atkins (Horsehead Branch)	Game and Fish[5]	17	ESE	FW	752	15	n.a.[6]	n.a.	n.a.
3.	Fish Lake (Point Removal Canal)	n.a.	25	ESE	n.a.	120	n.a.	n.a.	n.a.	n.a.
4.	Lake Overcup (Overcup Creek)	Game and Fish	29	ESE	FW	1,025	18.5	n.a.	n.a.	n.a.
5.	Beaver Fork L (Beaver Fork R)	City of Conway	47	ESE	W	710	n.a.	n.a.	n.a.	n.a.
6.	Harris Brake L (Fourch La Fave R)	Game and Fish	34	SE	FW	1,260	20	n.a.	n.a.	n.a.
7.	Big Maumelle L (Maumelle R)	Little Rock Water District	48	SE	W	8,850	196.5	n.a.	n.a.	93[7]
8.	Gibson Lake (Mill Creek)	Fish and Wildlife[8]	13	SSE	FW	480	n.a.	n.a.	n.a.	n.a.
9.	Winona Lake (Alum Fork Cr)	Little Rock Water District	40	SSE	W,P	1,170	41.8	n.a.	n.a.	28.1[7]
10.	Nimrod Res (Fourch La Fave R)	Corps of Engineers	24	S	F	18,300	336	3,550	29[9]	632
11.	Lake Ouachita (Ouachita R)	Corps of Engineers	50	S	F,P,W	48,300	2,768	20,900	865[9]	1,670
12.	Cove Lake (Cove Creek)	U.S. Forrest Service	23	WSW	R	166	2	n.a.	n.a.	n.a.
13.	Blue Mountain R (Petit Jean Cr)	Corps of Engineers	30	WSW	F	11,00	258	2,910	24.6[9]	413
14.	Hartman L (Old Arkansas R)	n.a.	24	WNW	n.a.	227	n.a.	n.a.	n.a.	n.a.
15.	Ozark Res[10] (Arkansas R)	Corps of Engineers	33	WNW	N,P,R	10,600	148.4	8,800	129	25,000
16.	Horsehead L (Horsehead Cr)	Game and Fish	29	NW	FW	100	3	n.a.	n.a.	n.a.
17.	Lake Ludwig (Spadra Creek)	City of Clarksville	20	NW	W	240	6	130	1	0.95
18.	Illinois Bayou (Same)	Russellville Water Co.	5*	NE	W	n.a.	n.a.	n.a.	n.a.	n.a.

Notes:
1. F-Flood Control, P-Power, N-Navigation, W-Water Supply, FW-Fish and Wildlife, R-Recreation
2. At Elevation 338 Feet
3. At Elevation 336 Feet
4. At Bottom of Power Pool Level
5. State of Arkansas Game and Fish Commission
6. n.a. – information not available
7. Safe Yield
8. U.S. Fish and Wildlife Service
9. At the Conservation Pool Level
10. Under Construction

* Upstream distance to water company station

RESERVOIR AND LAKE DATA WITHIN A 50 MILE RADIUS

SAR FIGURE NO. 2-12

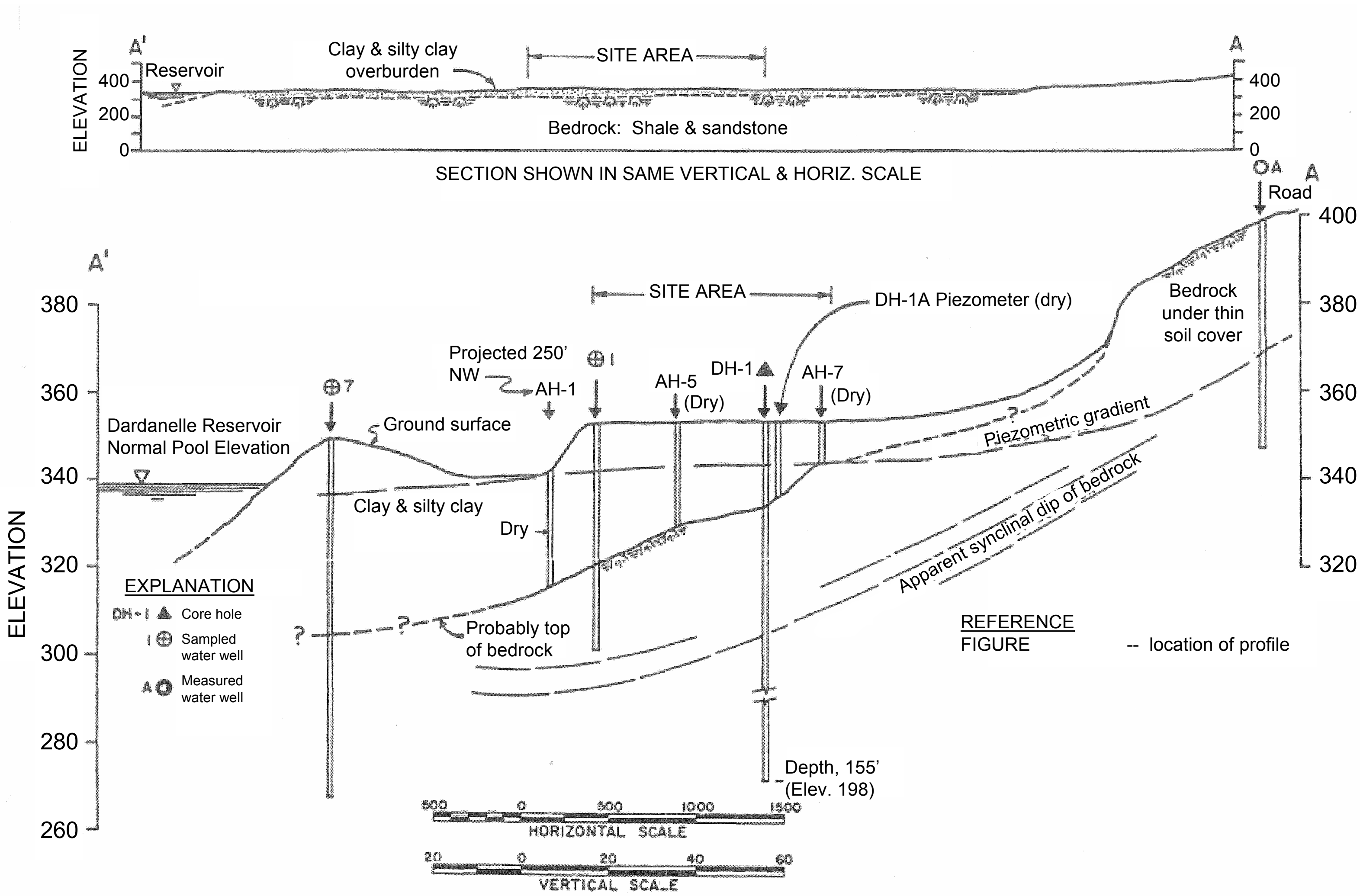
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



GEOLOGIC SECTION A' – A

SAR FIGURE NO. 2-13

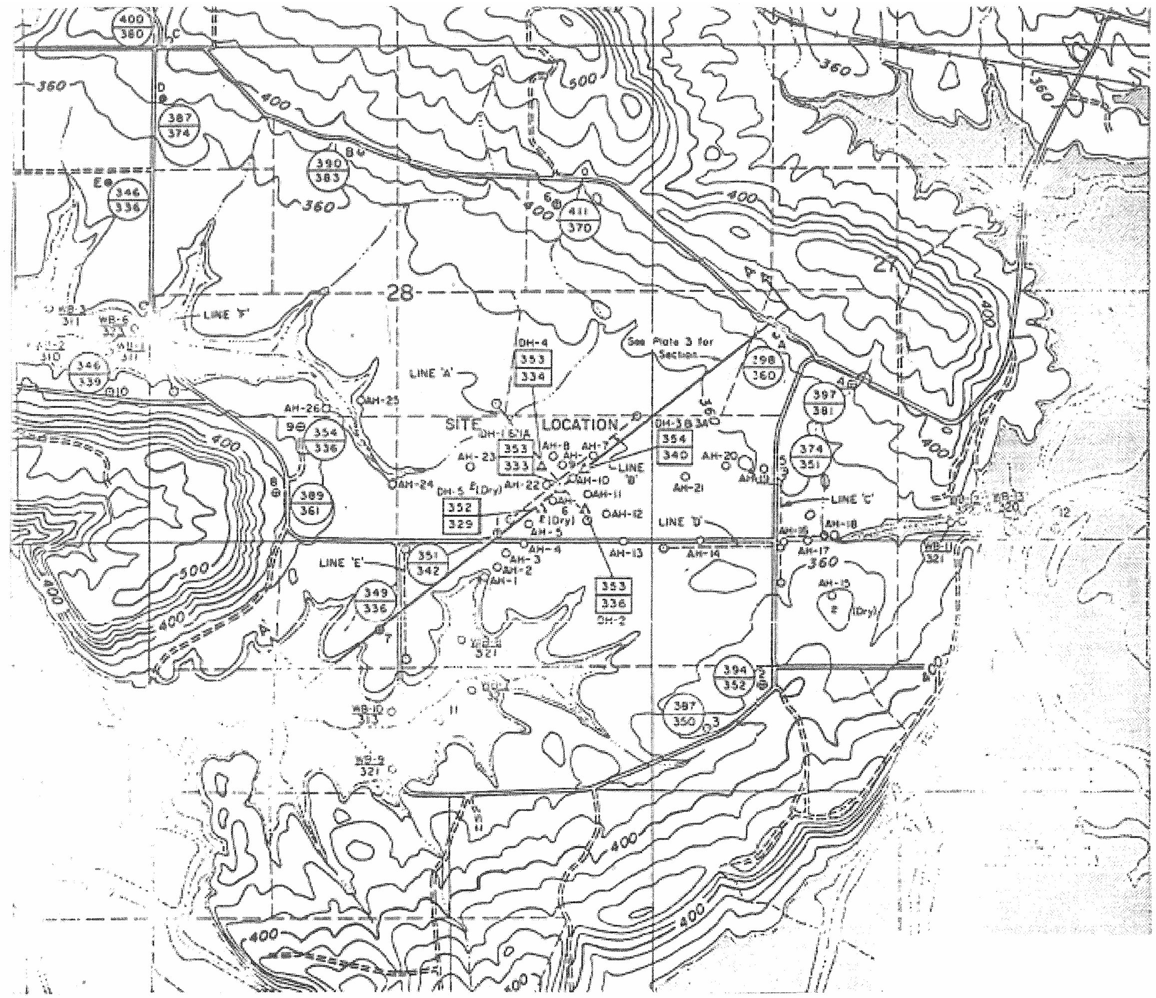
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



EXPLANATION

- Seismic Line
- Core Hole
- Sampled water well and identification (See Table 1)
- Measured water well
- Piezometer
- Auger Hole
- Wash bore hole / Bedrock elevation
- Surface elevation / Piezometric surface elevation
- Surface Elevation / Bedrock elevation

NOTES
1. Water sample 13 taken 300’ downstream Dardanelle Dam.
2. Wash bore 4 and 5 not shown.

AUGER HOLE DATA					
Hole No.	Surface El.	Bedrock El.	Hole No.	Surface El.	Bedrock El.
1	338’	315’	14	352’	327’
2	340’	318’	15	383’	367’
3	345’	322’	16	349’	332’
4	350’	326’	17	343’	339’
5	352’	329’	18	346’	338’
6	352’	329’	19	357’	344’
7	354’	342’	20	354’	344’
8	353’	337’	21	353’	336’
9	353’	336’	22	353’	332’
10	353’	338’	23	349’	326’
11	354’	338’	24	343’	327’
12	353’	332’	25	343’	330’
13	352’	331’	26	329’	329’

(Topography reporduced from U.S.G.S. 7.5 minute Russellville West, Ark. Map sheet)

500 0 500 1000 1500 Feet
SCALE

Contour interval = 20 feet

LOCATION OF FIELD INVESTIGATION

SAR FIGURE NO. 2-14

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

PHYSIOGRAPHIC DIVISIONS MAP OF ARKANSAS

EXPLANATION

Boundary of physiographic province

Boundary of physiographic section

REF.

Haley, B.R., Geology of Paris Quadrangle, Logan County, Arkansas, Arkansas Geological and Conservation Commission Information Circular No. 20-B, 1961.

SAR FIGURE NO. 2-15

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PHYSIOGRAPHIC DIVISIONS MAP OF ARKANSAS	BASED ON DRAWING NO	SHEET / REV.

EXPLANATION

RECENT
- Alluvium (Silt and clay)

PENNSYLVANIAN
- McAlester Group
  - Savanna sandstone (Sandstone & sandy shale)
  - Paris shale (Chiefly sandy shale)
  - Fort Smith formation (Sandstone-sandy shale)
- Winslow formation
  - Spadra shale (Contains coal beds)
  - Hartshorne sandstone (Chiefly massive sandstone)
  - Atoke formation (Shale and sandstone)
- Savanna, Paris, Fort Smith, Spadra, and Hartshorne formations
- Morrow Group
  - Bloyd shale and Hole formation (Shale, sandstone, Limestone, and coal)

MISSISSIPPIAN
- Pitkin limestone, Fayetteville shale, and Batesville sandstone
- Bloyd shale, Hole formation, Pitkin limestone Fayetteville shale, Batesville sandstone, and Moorefield shale
- Jackfork sandstone
- Stanley shade & Hot Springs sandstone
- Boone formation with St. Joe marble member Csj at base. (Limestone and chert)

DEVONIAN, SILURIAN, ORDOVICIAN
- Arkansas novoculite, Missouri Mtn. slate, Polk Creek shale, and Bigfork chert

SILURIAN
- Polk Creek shale and Bigfork chert

ORDOVICIAN
- Womble shale, Blakely sandstone, and Mazarn shale
- St. Peter sandstone and Everton limestone

Gas field

Faults

Axis of Anticline

T, overthrust side of thrust fault
D, downthrown side of normal fault
(type of fault not everywhere indicated;
Many minor faults are not shown)

REF
Geologic Map of Arkansas, 1929, prepared by the Arkansas Geological Survey.
Geologic Map of Delaware Quadrangle, 1961, Arkansas Geological Commission
Geologic Map of Russellville West Quadrangle, 1967, U.S.G.S. open file report

REGIONAL GEOLOGIC MAP

SAR FIGURE NO. 2-16

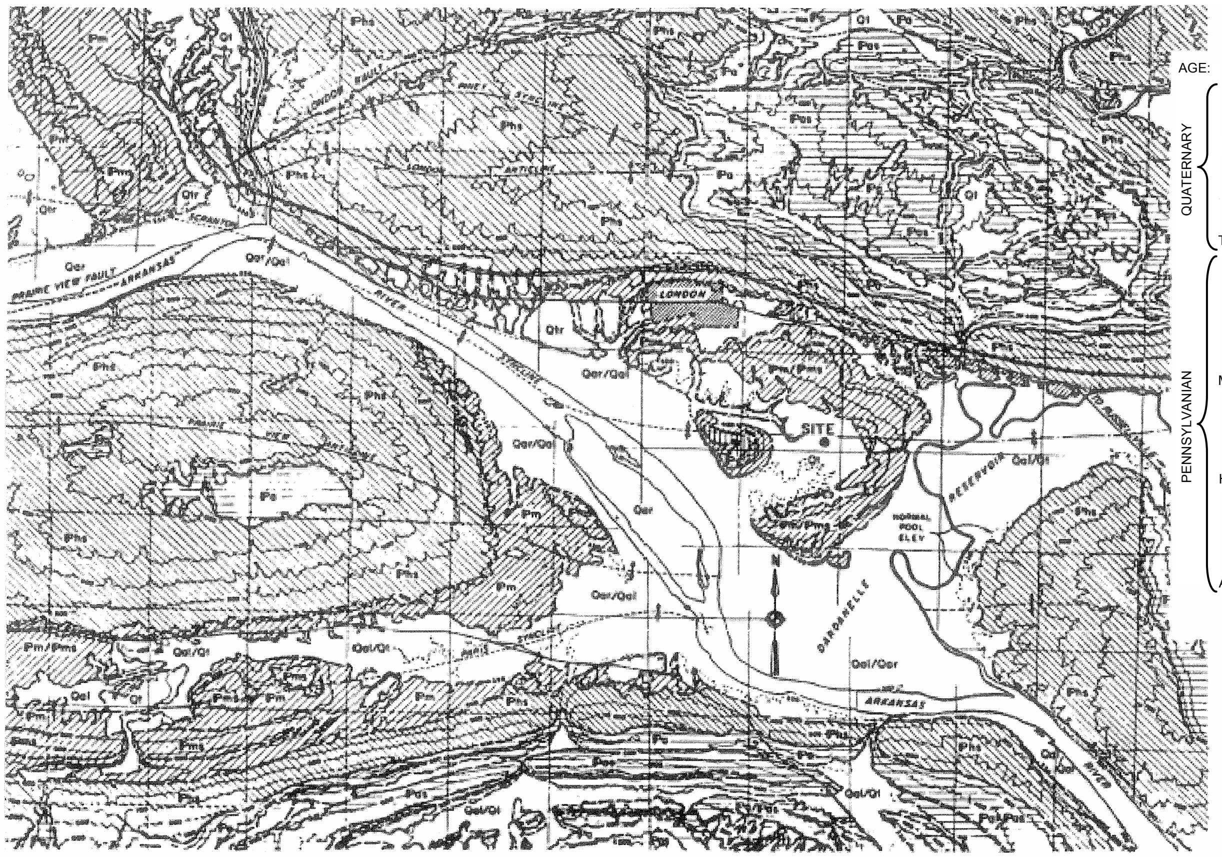
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



EXPLANATION

AGE:	SYMBOL:	DESCRIPTION:
QUATERNARY	Qal / Qar — Alluvium	Qal, alluvial deposits along stream channels and, in same places, parts of the lowermost terrace. Qar, alluvial deposits along Arkansas River
	Qt / Qtr — Terrace Deposits	Includes alluvial deposits in two undivided terrace levels: Qt, stream terrace: Qtr, river terrace.
PENNSYLVANIAN	Ps	Savanna formation
	Pm / Pms — McAlester formation	Pm, shale, siltstone, and thin beds of sandstone or silty sandstone. Pms, sandstone, silty sandstone, or interbedded sandstone, siltstone and shale.
	Ph / Phs — Hartshorne formation	Ph, shale, siltstone, and thin beds of sandstone or silty sandstone. Phs, sandstone, silty sandstone, or interbedded sandstone, siltstone and shale.
	Pa / Pas — Atoka formation	Pa, shale, siltstone, and thin beds of sandstone or silty sandstone. Pas, sandstone, silty sandstone, or interbedded sandstone, siltstone and shale.

EXPLANATION OF SYMBOS

- Geologic contact
- Fault
- Syncline
- Anticline

SCALE

NOTE: Geology is from the following tow sources -

1 Geology of Delaware Quadrangle and Vicinity Info Circular 20-A of Arkansas Geological Commission, 1961, Merewether and Haley

2. Geologic Map of Russellville West Quadrangle, by Boyd R. Haley, U.S. Geological Survey, prepared in cooperation with Arkansas Geological Commission

GENERALIZED GEOLOGIC MAP

SAR FIGURE NO. 2-17

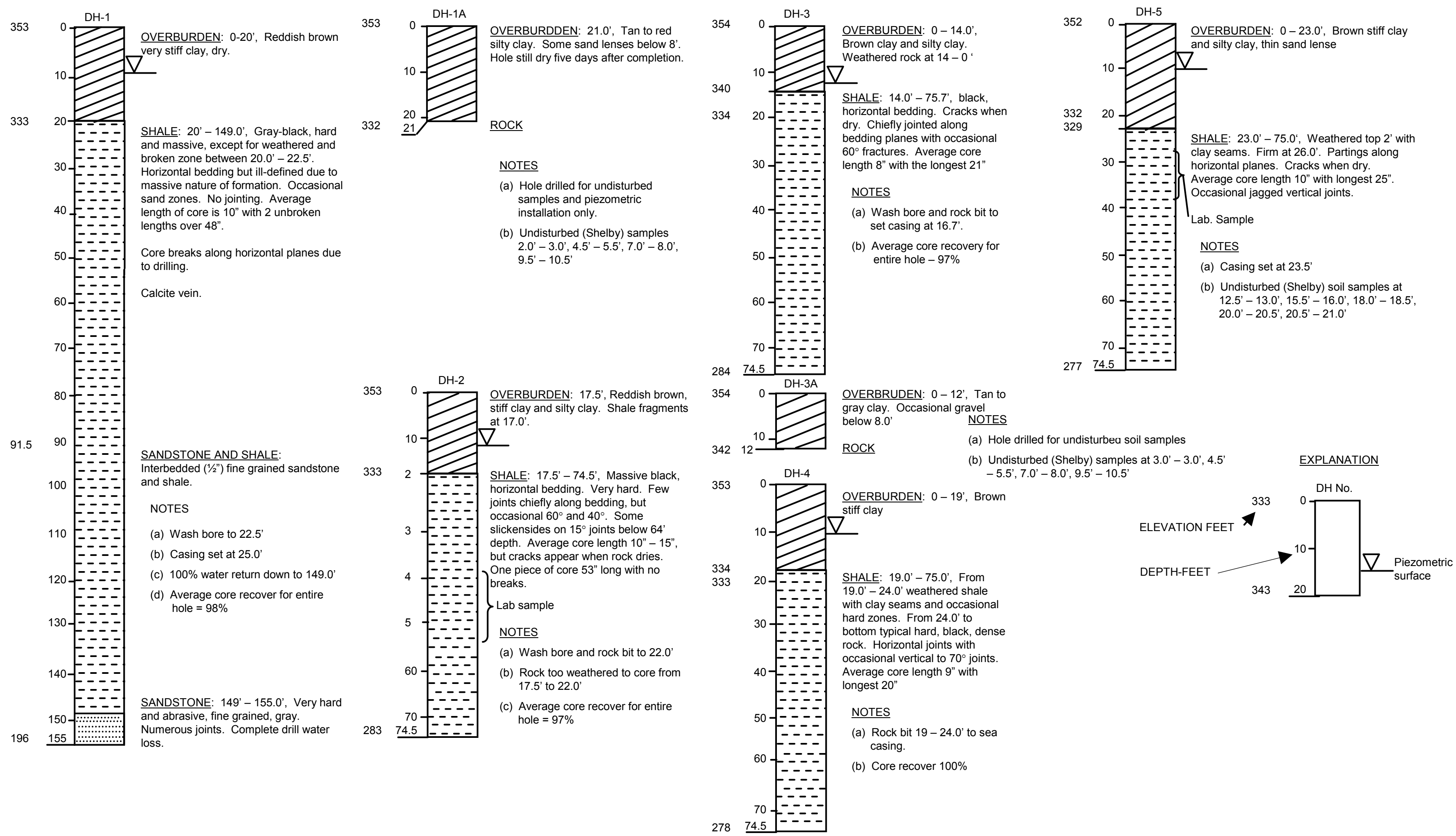
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



GRAPHIC LOG OF DRILL HOLES DH-1 THROUGH DH-5

SAR FIGURE NO. 2-18

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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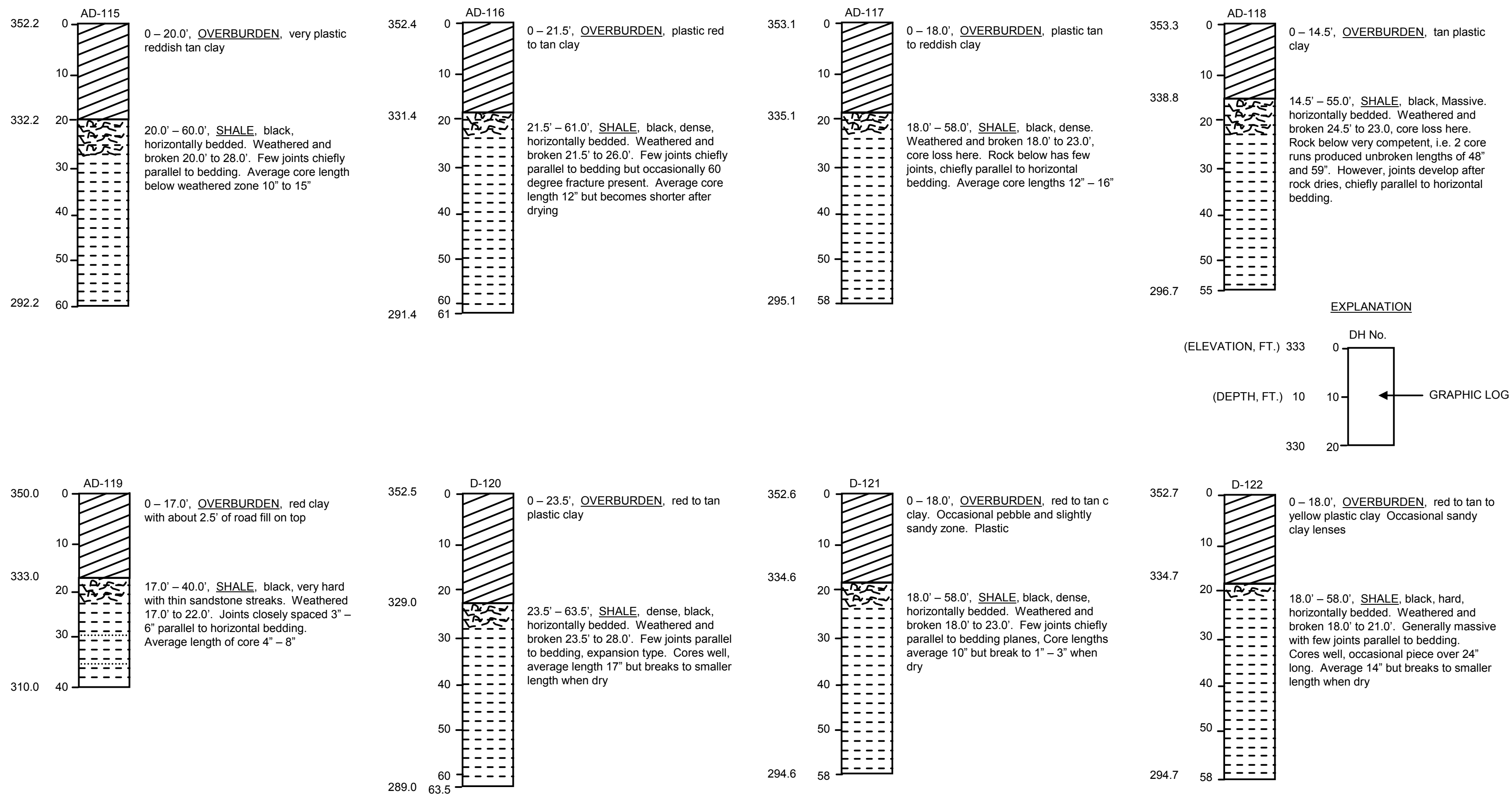
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



GRAPHIC LOG OF DRILL HOLES AD-115 THROUGH D-122

SAR FIGURE NO. 2-19

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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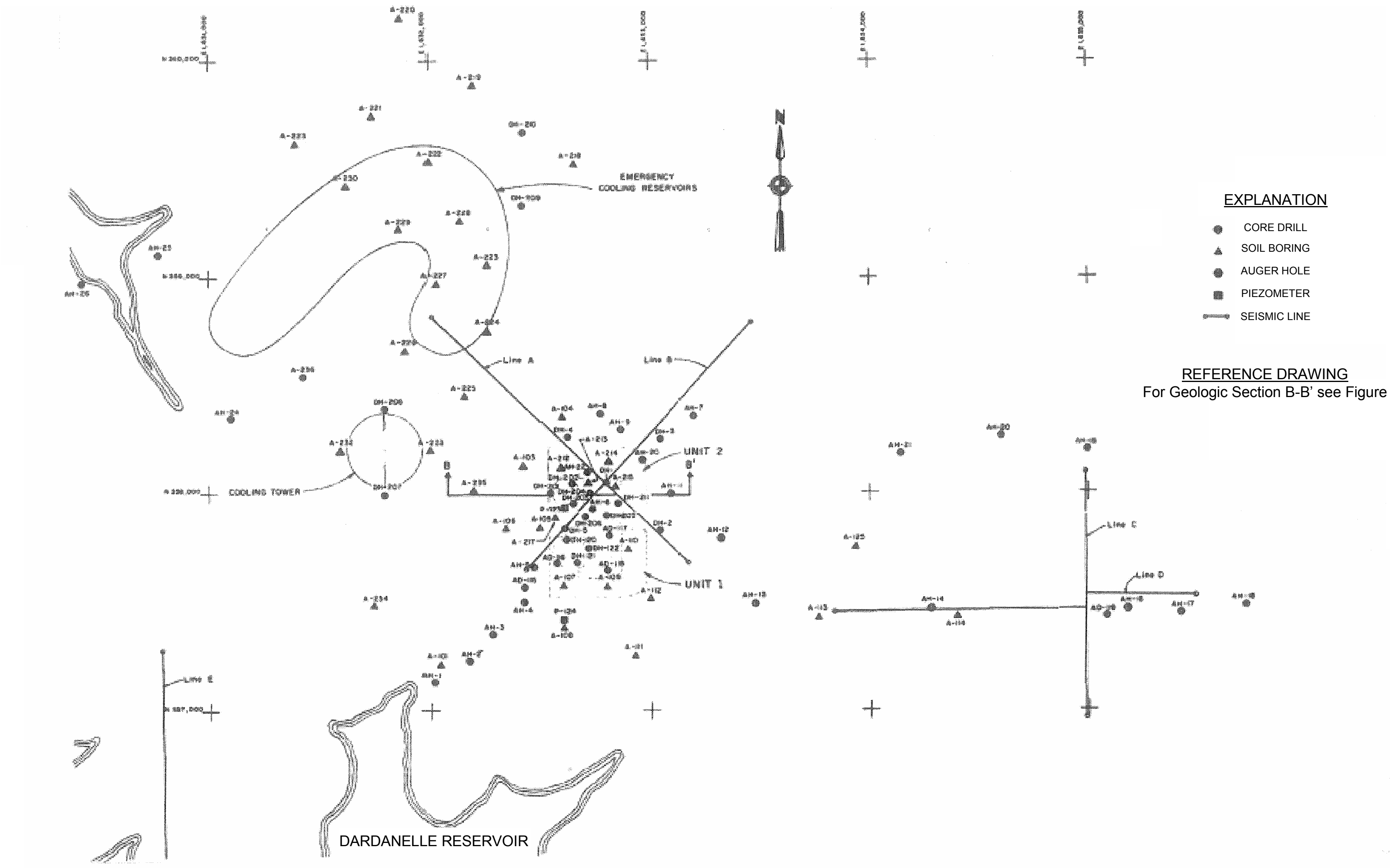
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



SITE INVESTIGATIONS

SAR FIGURE NO. 2-20

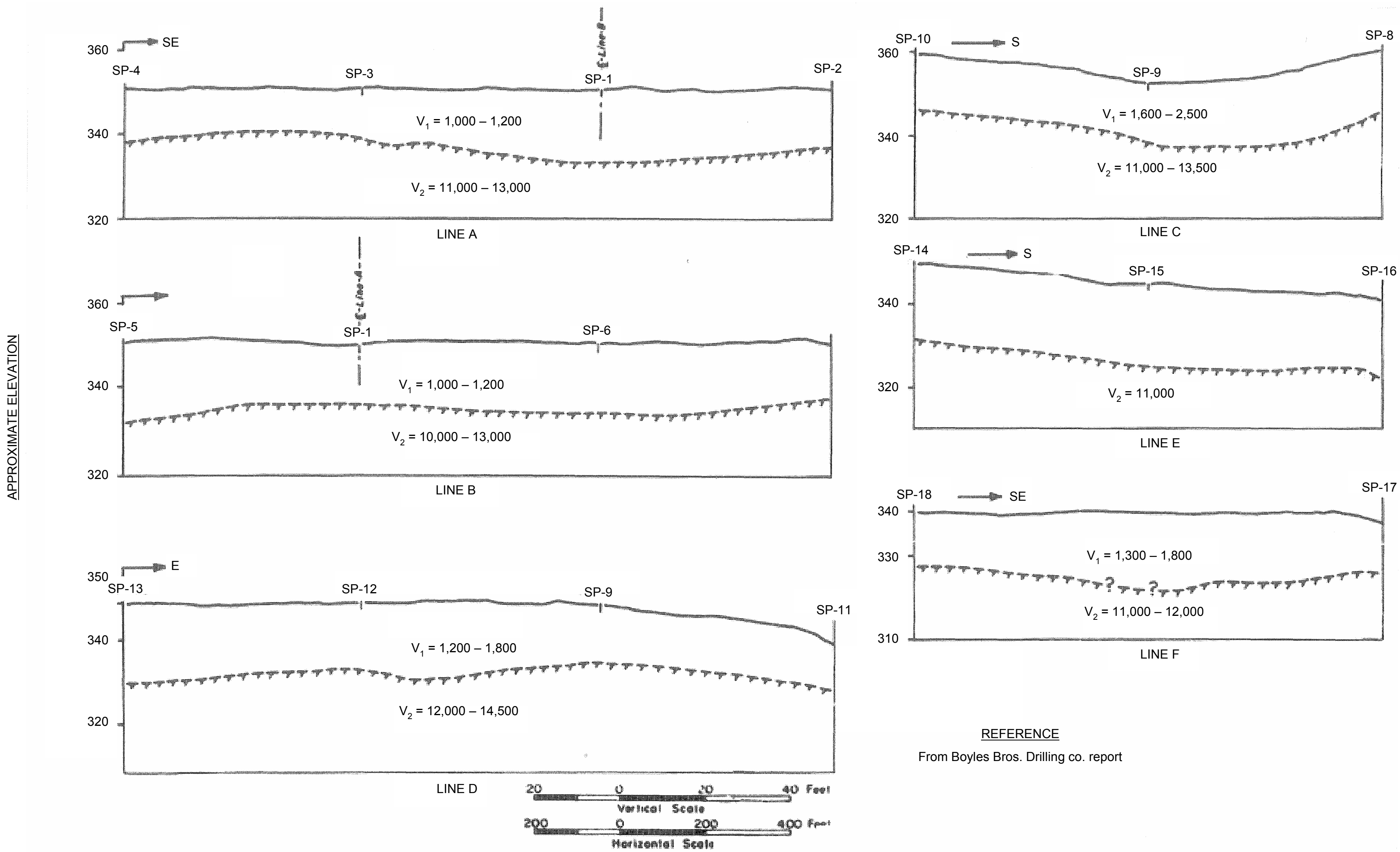
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

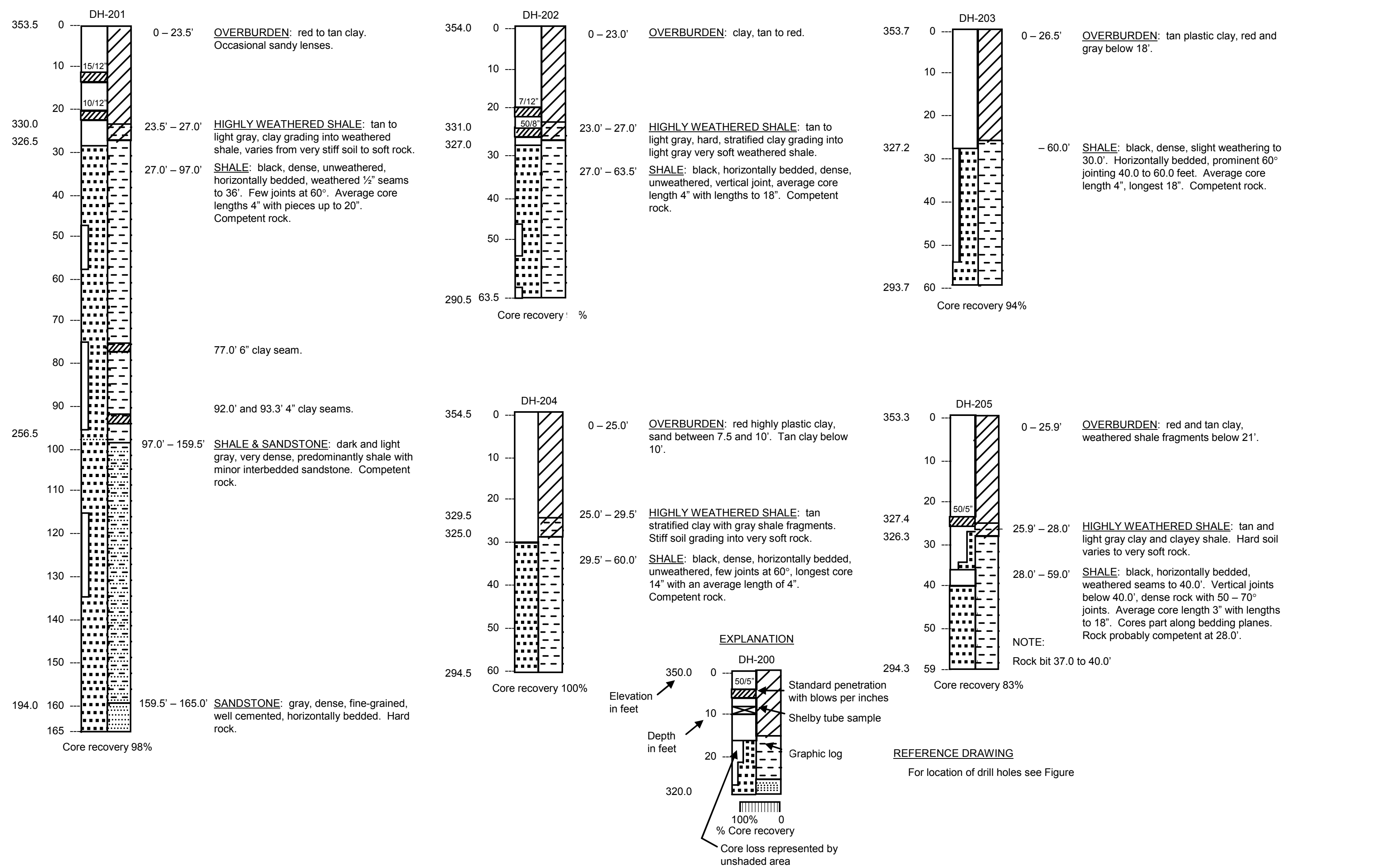
AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



SEISMIC PROFILES			SAR FIGURE NO. 2-21
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



GRAPHIC LOGS OF DRILL HOLES DH-201 THROUGH DH-205

SAR FIGURE NO. 2-22

AMENDMENT 20

ARKANSAS NUCLEAR ONE

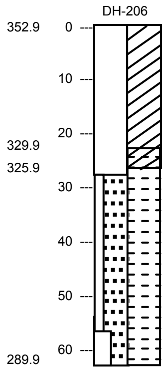
UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

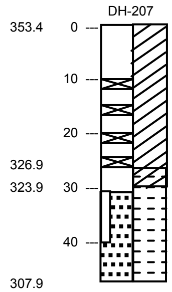


Core recovery 93%

0 – 23.0' OVERBURDEN: fill, clay and weathered shale to 10'. 10' and below tan, moderately plastic clay.

23.0' – 27.0' HIGHLY WEATHERED SHALE: brown to dark gray clay and weathered clay, stiff soil consistency increasing to very soft rock at 29.5'.

27.0' – 63.0' SHALE: black, horizontally bedded, some weathered seams and layers to 37', parts easily along bedding planes, clayey shale in zones, vertical joint. Average core length 3" with lengths to 18". Competent rock.

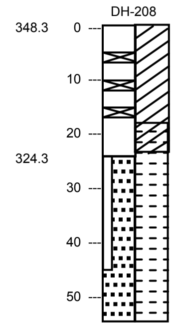


Core recovery 97%

0 – 26.5' OVERBURDEN: fill, clay and weathered shale to 6'. Tan, red and gray clay below.

26.5' – 29.5' HIGHLY WEATHERED SHALE: tan to light gray stratified clay with gray and brown weathered shale fragments. Stiff soil at 26.5' grades in to very soft rock at shale contact.

29.5' – 45.5' SHALE: unweathered, dark gray to black, horizontally bedded. One joint at 38" with 20° dip. Average core length 5", with pieces to 2.5'. Competent rock.

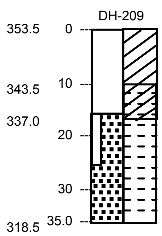


Core recovery 95%

0 – 18.0' OVERBURDEN: clay, slightly silty-sandy, tan, light gray and red.

18.0' – 24.0' HIGHLY WEATHERED SHALE: tan and light gray clay with weathered shale fragments grading down to dark gray weathered shale. Stiff soil varies to very soft rock.

24.0' – 55.0' SHALE: dark gray, unweathered, dense, horizontally bedded, unjointed, core parts along bedding planes in clean breaks, longest core 3' with an average length of 1.5'. Competent rock.

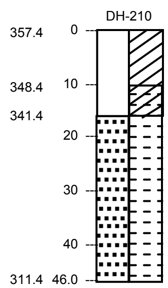


Core recovery 95%

0 – 10.0' OVERBURDEN: tan and red sandy clay.

10.0' – 16.5' HIGHLY WEATHERED SHALE: stratified tan stiff clay with weathered shale fragments to dark gray very soft weathered shale.

16.5' – 35.0' SHALE: dark, gray, horizontally bedded, unweathered. No jointing. Average core length 5" with pieces to 15". Competent rock.

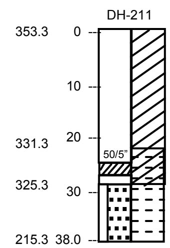


Core recovery 100%

0 – 9.0' OVERBURDEN: tan plastic clay.

9.0' – 16.0' HIGHLY WEATHERED SHALE: tan clay with shale fragments to gray weathered shale. Soil grades from stiff clay to a very soft rock.

16.0' – 46.0' SHALE: dense, unweathered, dark gray to black unjointed shale with interbedded 1/8 to 1/16" light gray well cemented sandstone lenses. 85% shale, 15% s.s. Average core length 4" with pieces to 18". Competent rock.



Core recovery 85%

0 – 22.0' OVERBURDEN: red and weathered shale fill to 9'. Red and tan clay 9 to 22'.

22.0' – 28.0' HIGHLY WEATHERED SHALE: tan stratified clay with weathered shale particles. 26' to 28' dark gray weathered shale. Hard soil increasing to very soft rock down thru zone.

28.0' – 38.0' SHALE: black, dense, unweathered horizontally bedded, few 60° joints. Longest core piece 12"; average core length 4". Competent rock.

REFERENCE DRAWINGS

1. See Figure for explanation
2. For location of drill holes see Figure

GRAPHIC LOGS OF DRILL HOLES DH-206 THROUGH DH-211

SAR FIGURE NO. 2-23

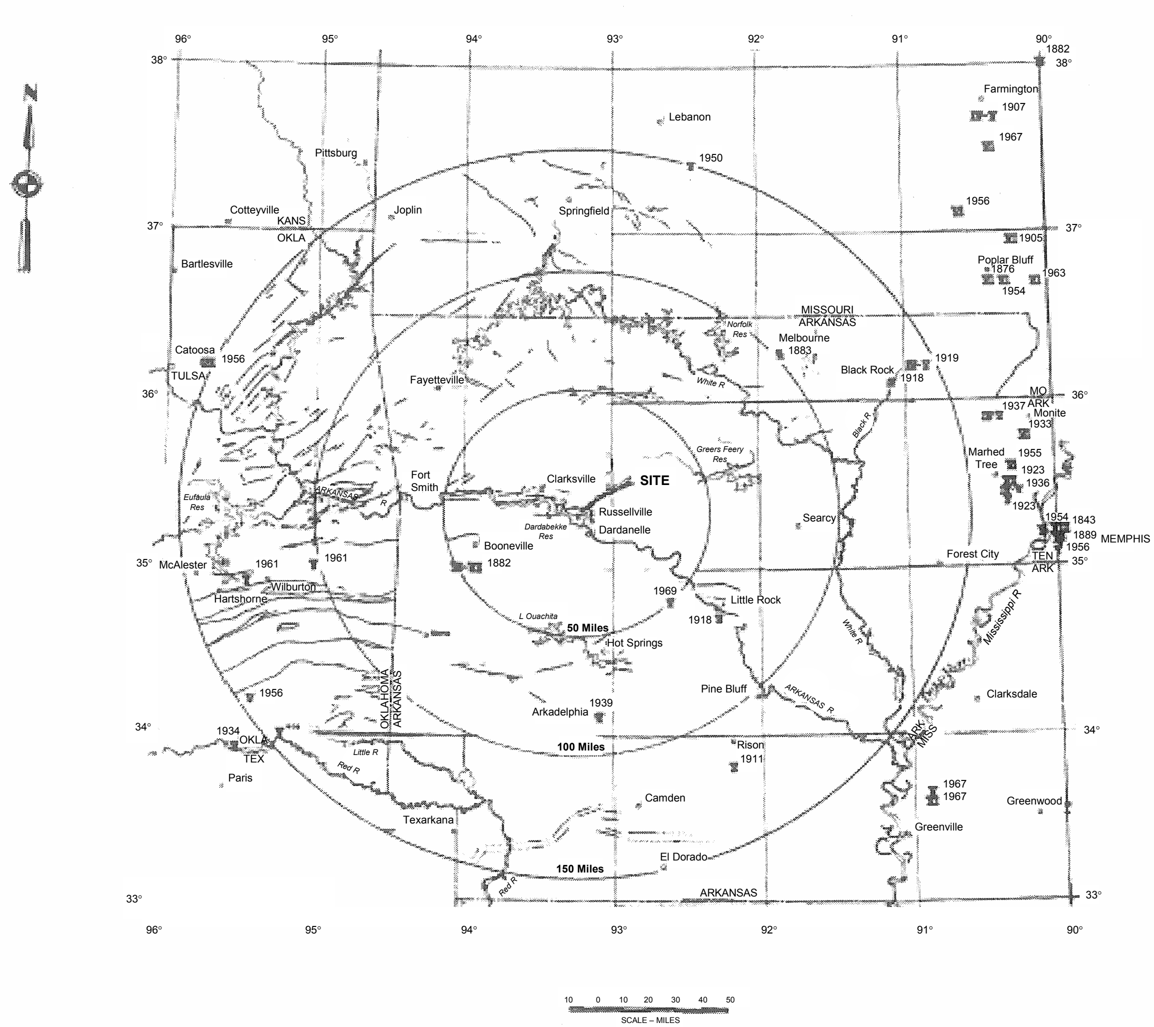
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



EXPLANATION

Roman Numeral indicates the epicenter of an earthquake, the epicentral intensity of which was given on the Modified Mercalli Scale.

Intensities IV – V and greater are shown.

Dates of earthquakes indicated.

Fault – dashed where inferred, dotted where concealed

REFERENCES:

U.S.C. & G.S., Earthquake History of the U.S. Part 1.

U.S.C. & G.S., United States Earthquakes 1928 – 1966.

U.S.C. & G.S., Abstracts of Earthquake Reports for the United States Jan. 1967 – Dec. 1968

E.S.S.A. Hypocenter Data Cards Jan. 1986 – Dec. 1968

B.S.S.A. Aug. 1969, Seismological Notes Jan. – Feb. 1969

B.S.S.A. Oct. 1969, Seismological Notes Mar. – Apr. 1969

B.S.S.A. Dec. 1969, Seismological Notes May – June 1969

B.S.S.A. Apr. 1969, Seismological Notes Sept. – Oct. 1969

B.S.S.A. Vol. 46, No. 2, History of Earthquakes in Kansas by D.F. Merriam

B.S.S.A. Vol. 31, No. 3, Contribution to the Seismic History of Missouri by R.R. Heinrich

Base map is taken from U.S.C. & G.S. World Aeronautical Charts 359, 360, 407, and 408

(1) Geologic Map of Arkansas, Geologic Survey of Arkansas

(2) Geologic Map of Russellville Wet Quadrangle, Arkansas, U.S.G.S. & Arkansas Geological Comm., 1967

(3) Map of Hartshorne Coal Bed, Keplinger and Wanenmacher, Petroleum Engineers, Tulsa, Oklahoma

(4) Geologic Map of Oklahoma, U.S.G.S. & Oklahoma Geological Survey, 1954

(5) Geologic Map of Missouri, Missouri Geological Survey, 1961

SIGNIFICANT EARTHQUAKE EPICENTERS & INTENSITIES AT EPICENTER 1699-1969

SAR FIGURE NO. 2-24

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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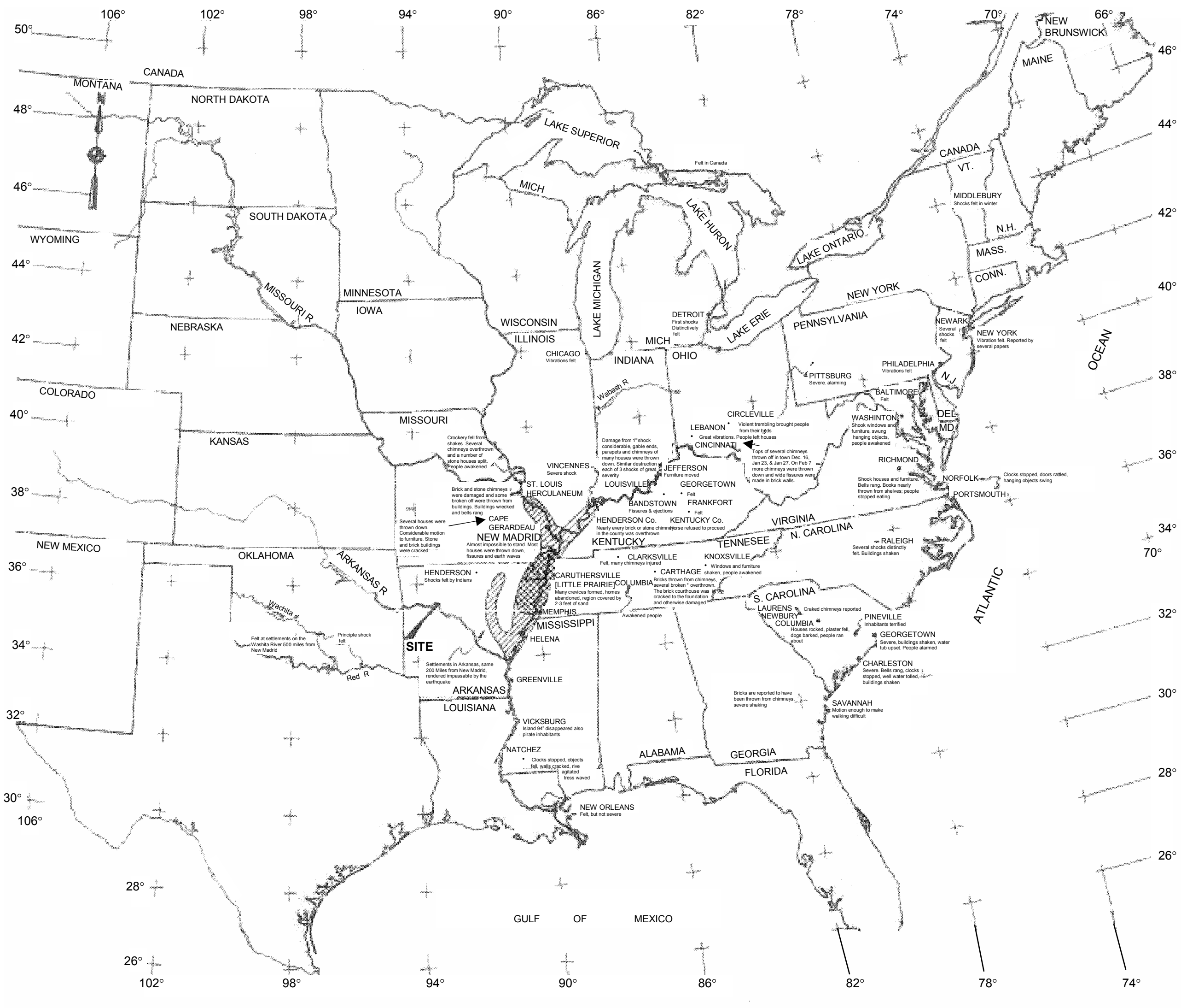
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



EXPLANATION

Map shows the extent of earthquake disturbances from the New Madrid area quakes from 1811 – 1812.

Area of principal disturbance, characterized by general warping, ejections, fissuring and severe landslides.

Area of minor disturbance, characterized by local fissuring, caving of stream banks, etc.

REFERENCES:

(1) Broadhead, G.C., "The New Madrid Earthquake" American Geologist vol., 30 August 1902.

(2) Clark, Blake, "America's Greatest Earthquake," Shreveport Magazine, March 1969.

(3) Fuller, Myron L., "The New Madrid Earthquake," U.S.G.S. Bull. 494, 1912.

(4) Heinrich, R.R., "A Contribution to the Seismic History of Missouri," B.S.S.A., vol., 31, No 3, 1941.

(5) Osterberg. J.O., " The Earthquake that Rearranged Mid-America," The Testing World No. 21, Winter 1967 – 1968.

(6) Sampson, F.A., "The New Madrid and Other Earthquakes of Missouri," B.S.S.A. vol. 3, No. 2, 1913.

(7) Shepard, E.M., "The New Madrid Earthquake," Journal Geology vol. 13, Jan. – Feb. 1905.

(8) U.S.C.&G.S. Earthquake History of the United States Part 1, 1965.

Base map is taken from imperial Map of the United States of America by Rand McNally and Company

THE NEW MADRID, MISSOURI EARTHQUAKES, 1811-1812

SAR FIGURE NO. 2-25

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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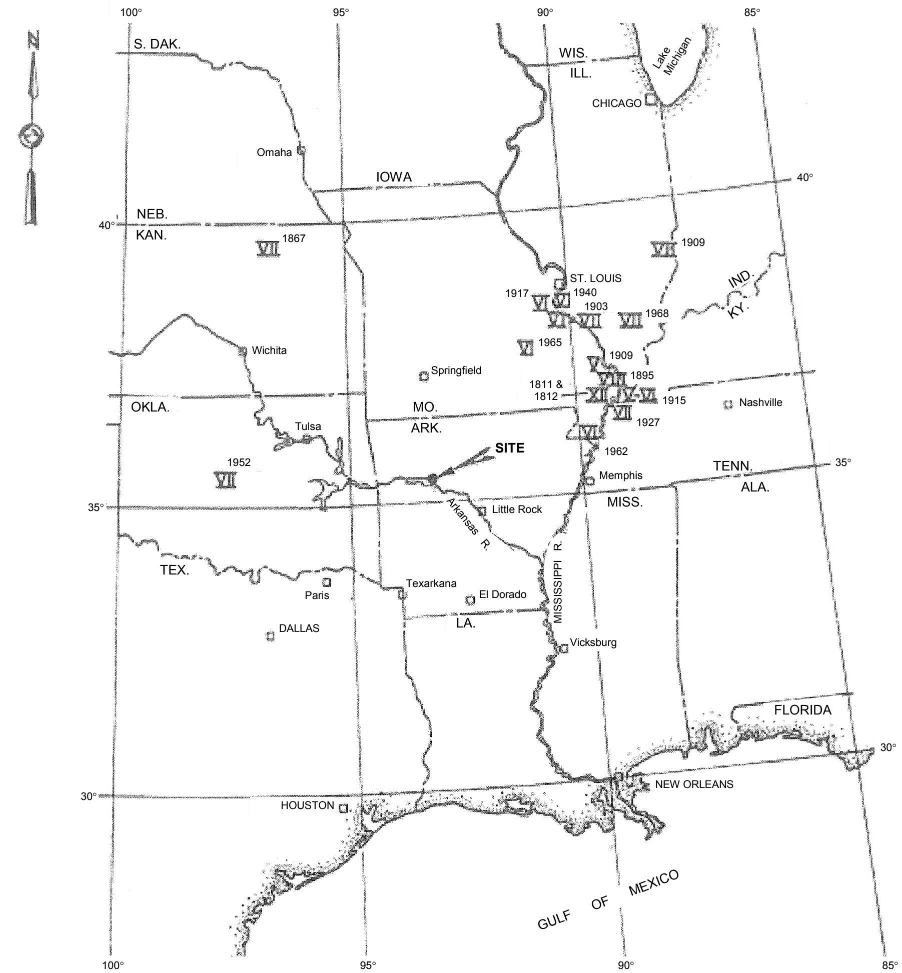
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



EXPLANATION

Roman Numeral indicates the epicenter of an earthquake, the epicentral intensity of which was given on the Modified Mercalli Scale.

Dates of earthquakes indicated.

REFERENCES:

U.S.C. & G.S., Earthquake History of the U.S. Part 1.

U.S.C. & G.S., United States Earthquakes 1928 – 1966.

U.S.C. & G.S., Abstracts of Earthquake Reports for the United States Jan. 1967 – Dec. 1968.

E.S.S.A. Hypocenter Data Cards Jan. 1968 – Dec. 1969.

B.S.S.A. Aug. 1969, Seismological Notes Jan. – Feb. 1969.

B.S.S.A. Oct. 1969, Seismological Notes Mar. – Apr. 1969.

B.S.S.A. Dec. 1969, Seismological Notes May – June 1969.

B.S.S.A. Apr. 1969, Seismological Notes Sept. – Oct. 1969.

B.S.S.A. Vol. 46, No. 2. History of Earthquakes in Kansas by D.F. Merriam.

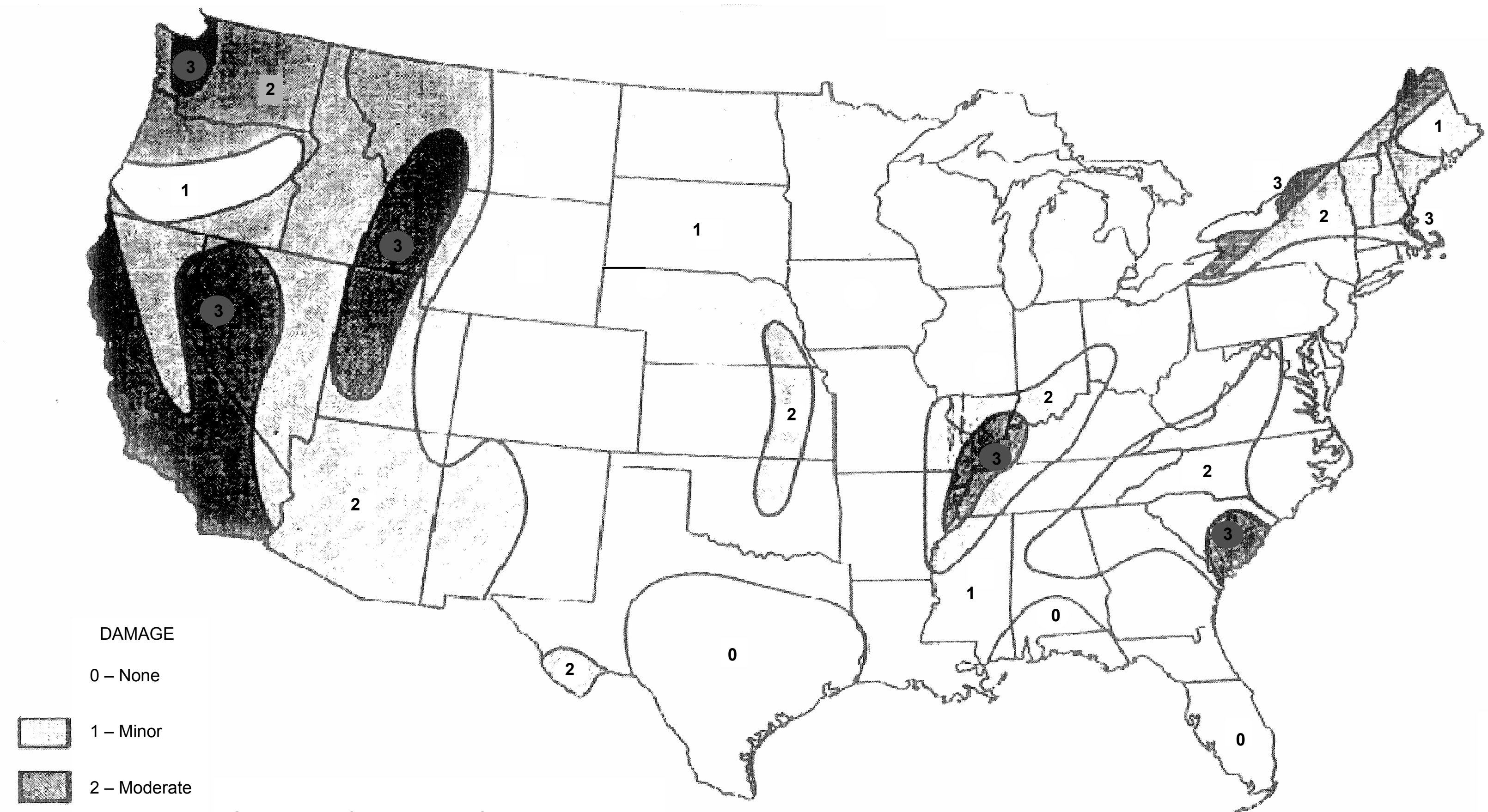
B.S.S.A. Vol. 31, No. 3. Contribution to the Seismic History of Missouri by R.R. Heinrich

Base map is taken from North America by Kümmerly and Frey, Berne, Switzerland.

0 50 100 200 300 400 500

SCALE - MILES

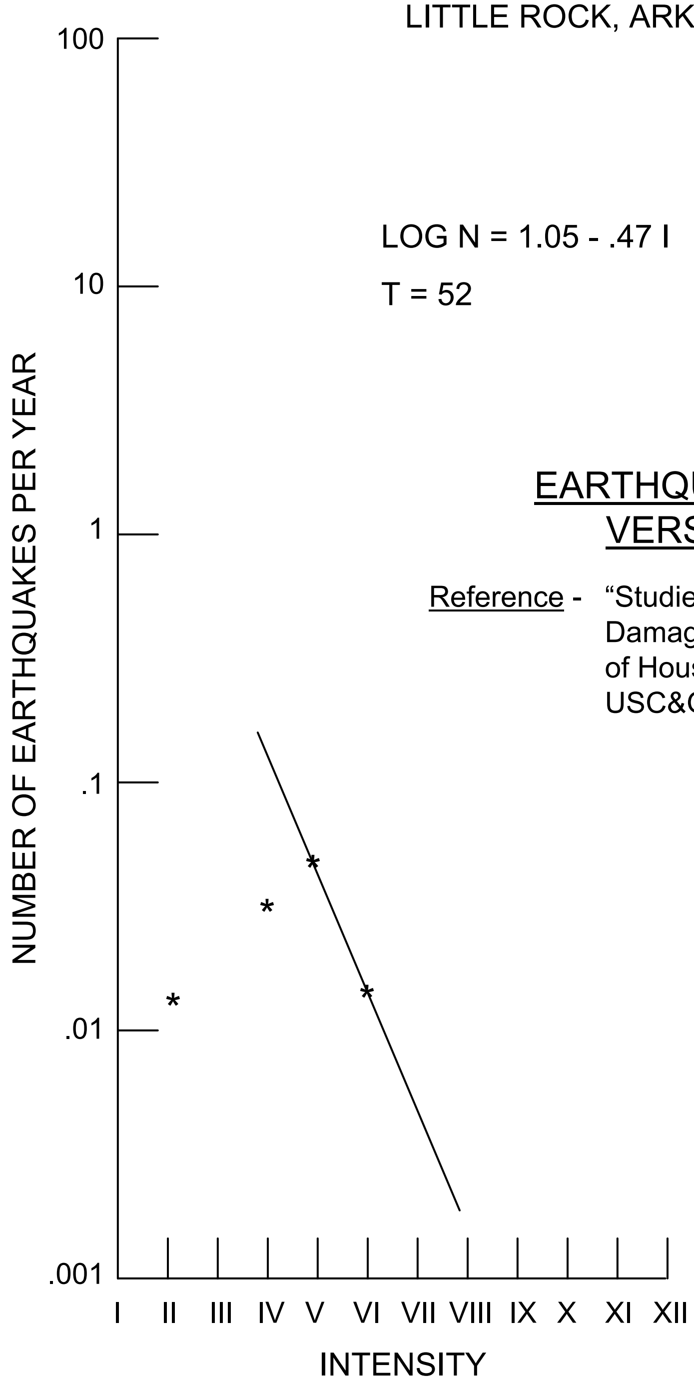
OTHER EARTHQUAKES PROBABLY FELT AT SITE		SAR FIGURE NO. 2-26		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
		BASED ON DRAWING NO	SHEET	REV.



Seismic risk map for conterminous U.S., developed by ESSA/Coast and Geodetic Survey and issued in January 1969. Subject to to revision as continuing research warrants, it is an updated edition of the map first published in 1948 and revised in 1951. The map divides the U.S. into four zones: Zone 0, areas with no reasonable expectancy of earthquake damage; Zone 1, expected minor damage; Zone 2, expected moderate damage; and Zone 3, where major destructive earthquakes may occur.

SEISMIC RISK MAP, CONTERMINOUS UNITED STATES			SAR FIGURE NO. 2-27		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
			BASED ON DRAWING NO	SHEET	REV.

LITTLE ROCK, ARKANSAS



EARTHQUAKE FREQUENCY VERSUS INTENSITY

Reference - "Studies in Seismicity and Earthquake Damage Statistics – Part prepared for Dept. of Housing and Urban Development by the USC&GS."

SAR FIGURE NO. 2-28

AMENDMENT 20

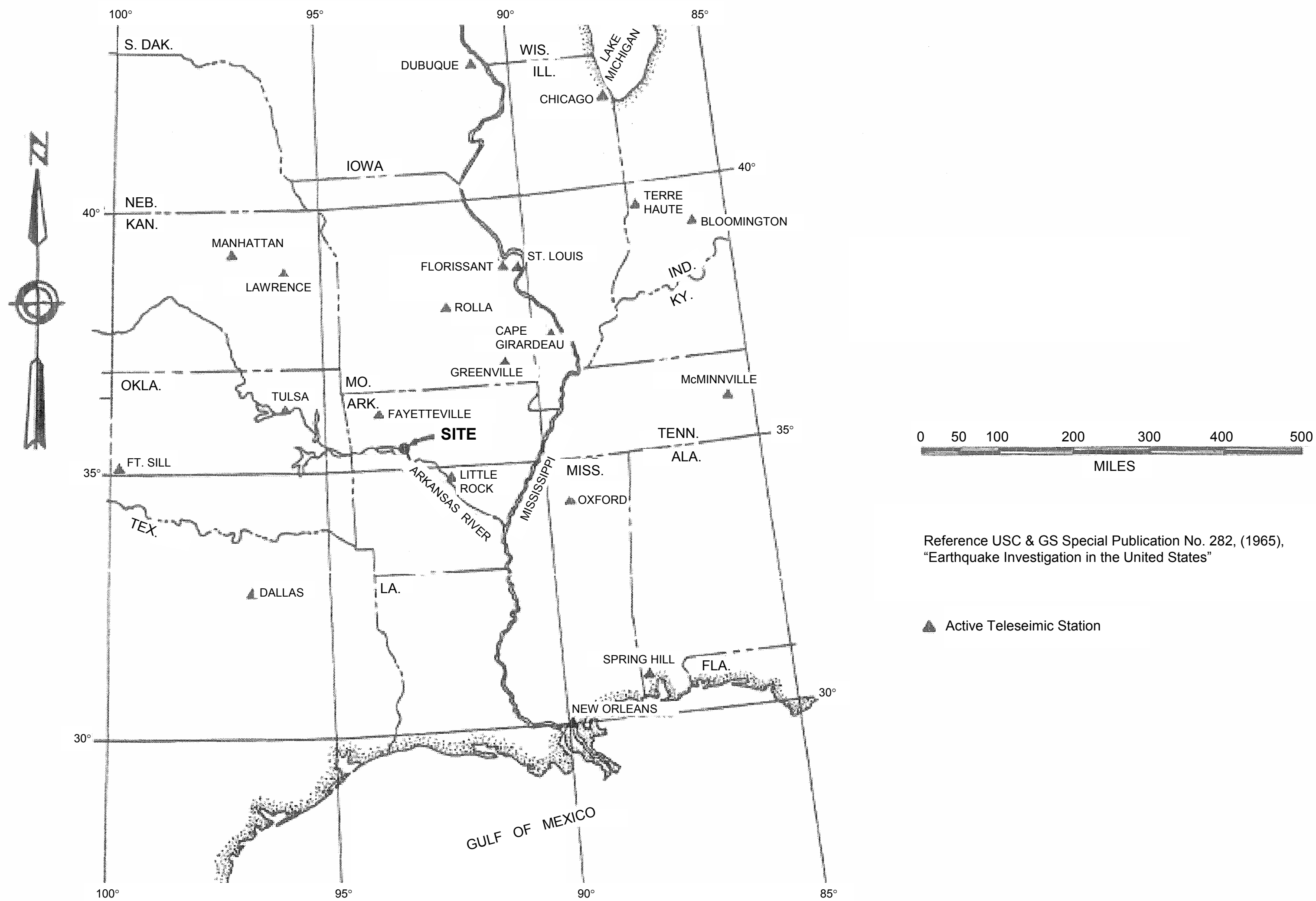
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	CHARLEY RANKIN
DESIGN:	ENTERGY
CAD NO:	fig2-28.sar

EARTHQUAKE FREQUENCY VS. INTENSITY AT LITTLE ROCK, ARKANSAS

BASED ON DRAWING NO	SHEET	REV.



ACTIVE TELESEISMIC STATIONS		SAR FIGURE NO. 2-29
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

Perry Byerly
6037 Contra Costa Road
Oakland, California 94618
Phone (415) 654-6893

August 5, 1970

Mr. Cole R. McClure, Jr.
Bechtel Incorporated
Fifty Beale Street
San Francisco, California 94119

Dear Mr. McClure:

I have read the section on seismology on the Russellville II site, submitted to me by Bechtel.

A study of the earthquake history of the area indicates that the greatest shaking in the area of the site was from the New Madrid earthquakes of 1811 and 1812.

I am impressed by the good foundation at the site, seismic speeds 11,000 to 14,500 ft/sec. I would estimate that the greatest intensity that could be expected at the site would be a high VI or low VIII, corresponding to 0.1 g. (I doubt if that great an intensity has been attained in historic time.)

So I endorse the value of 0.1 g for operating basis and 0.2 g for design basis.

Yours sincerely,

Perry Byerly

Perry Byerly

SAR FIGURE NO. 2-30

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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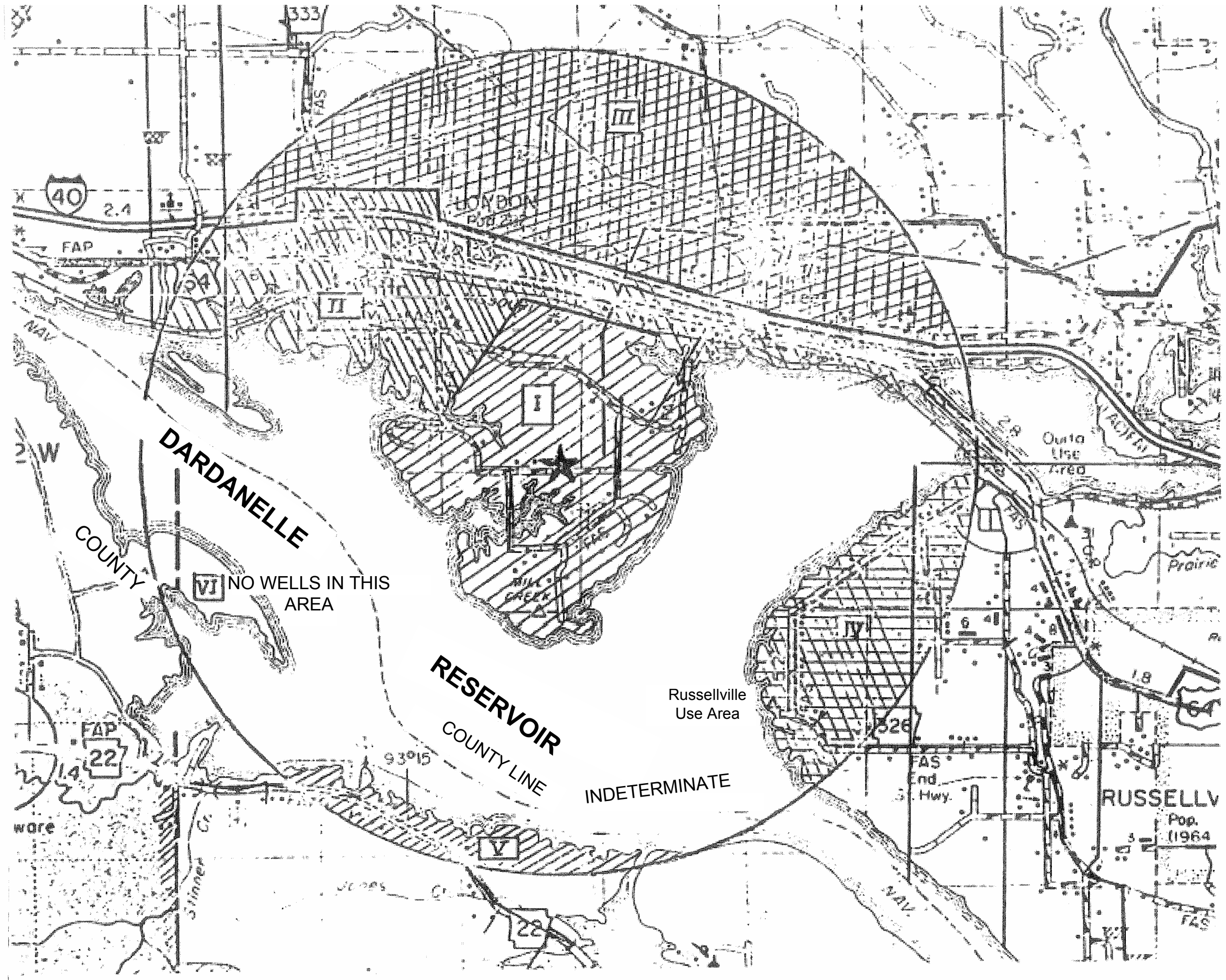
		DESIGN:	ENTERGY
		CAD NO:	
CONSULTANT'S REPORT	BASED ON DRAWING NO	SHEET	REV.



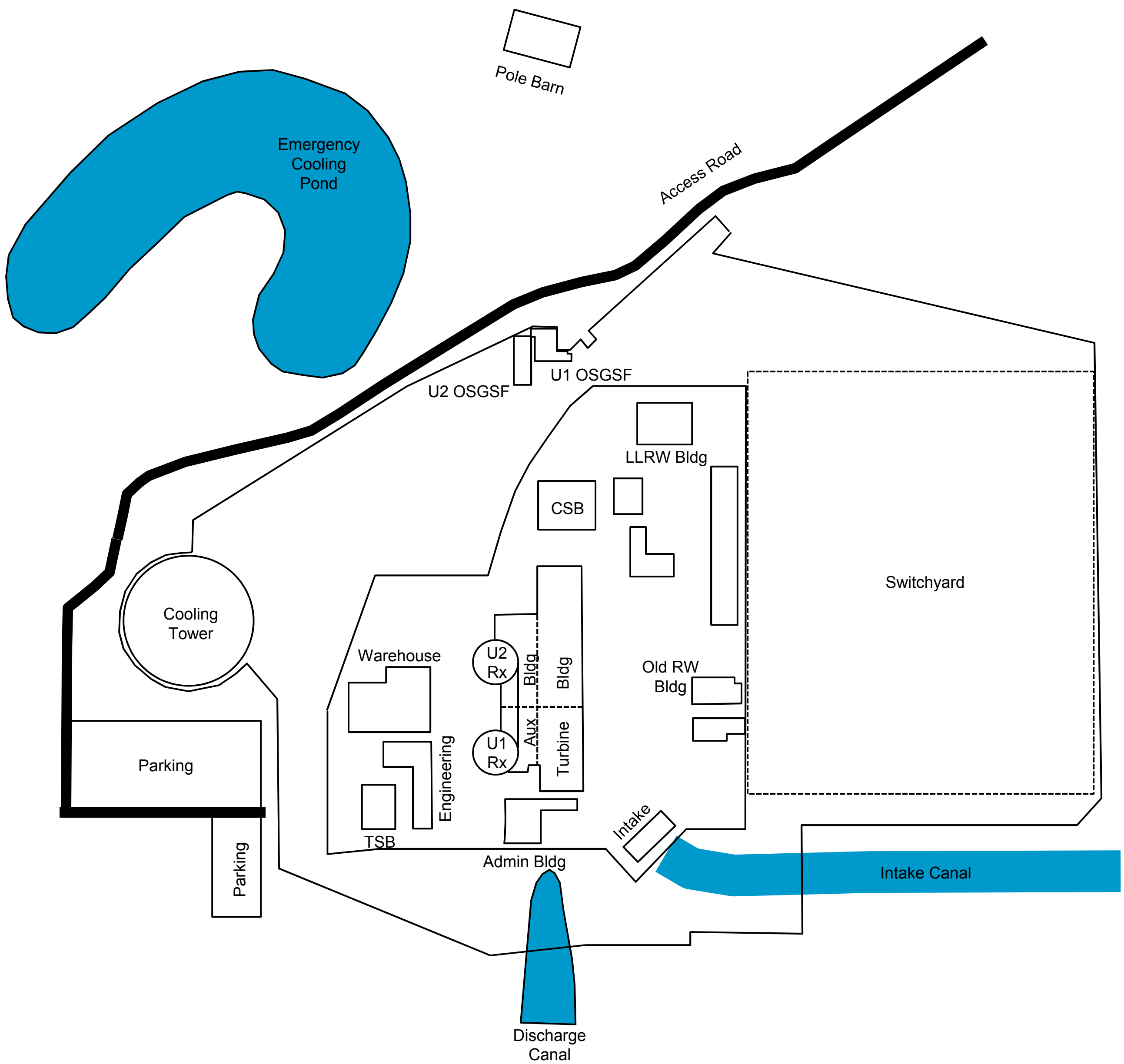
LOCATION OF WELL SURVEY

SAR FIGURE NO. 2-31

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 2-32

AMENDMENT 26

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	N/A

GENERAL SITE STRUCTURES

DRAWING NO	SHEET	REV.

DELETED

SEE ODCM

SAR FIGURE NO. 2-33

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
RESERVOIR SAMPLING POINTS	BASED ON DRAWING NO	SHEET / REV.

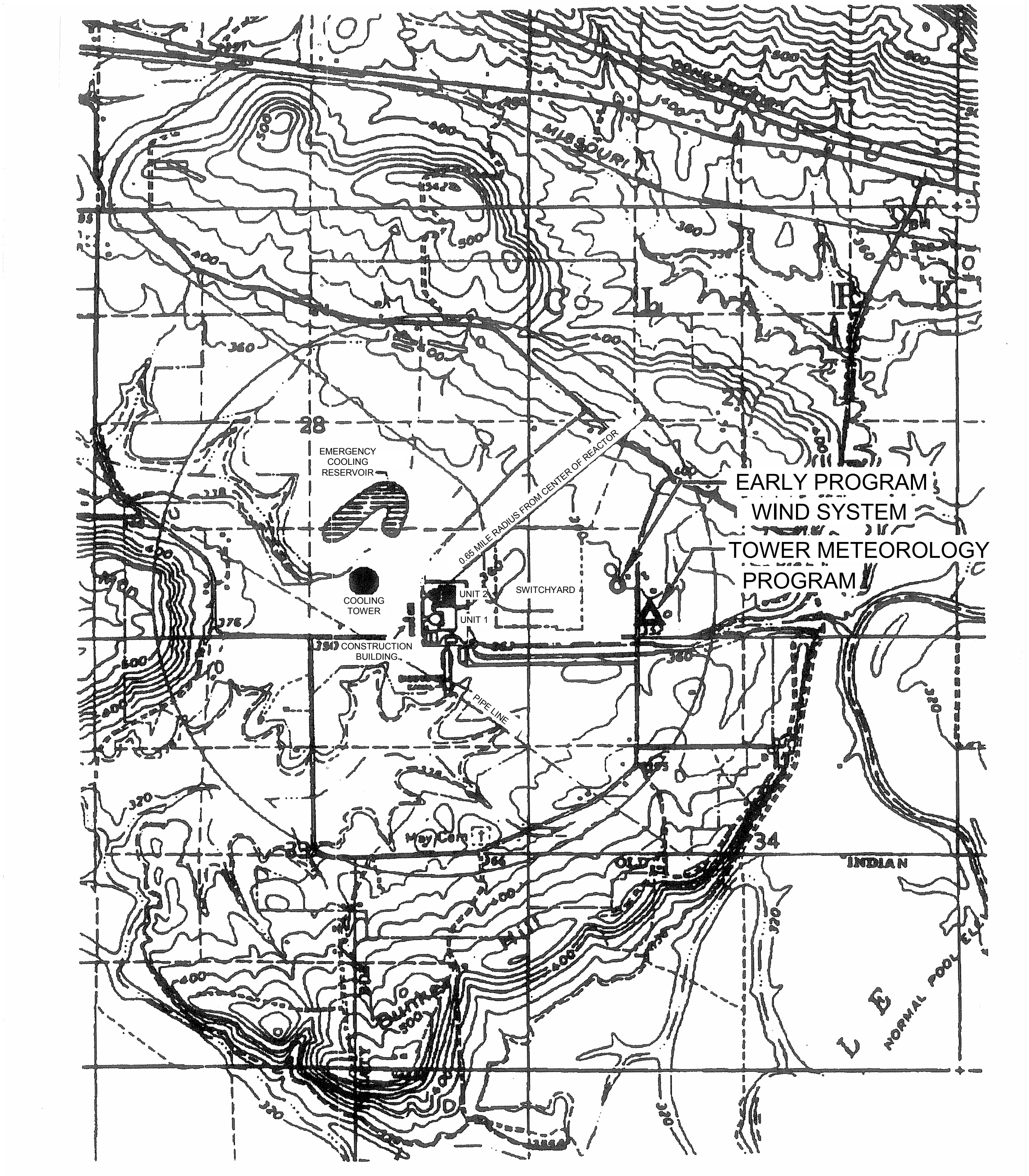
DELETED

SEE ODCM

SAR FIGURE NO. 2-34

AMENDMENT 20

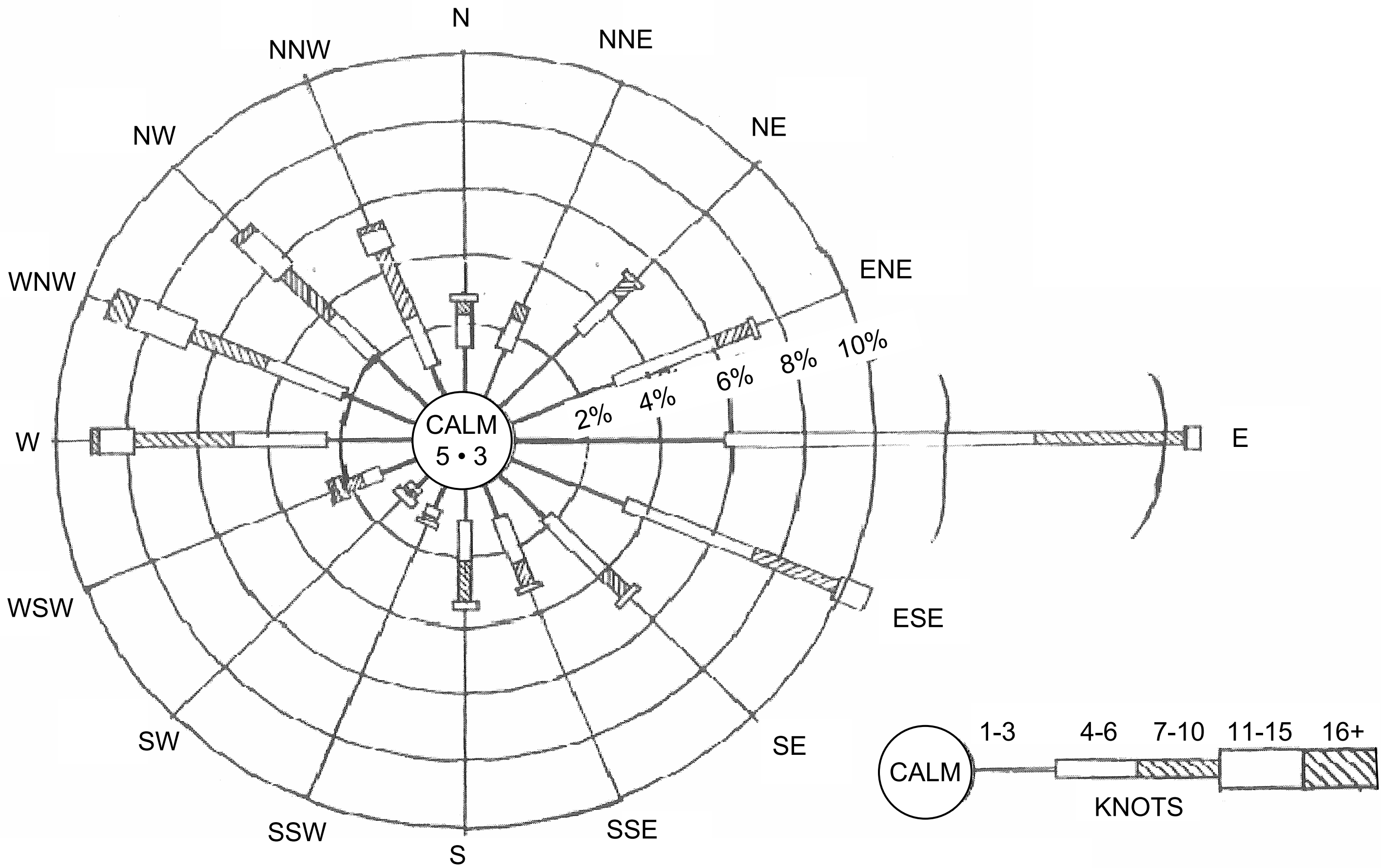
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OFF-SITE SAMPLING POINTS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 2-35

AMENDMENT 20

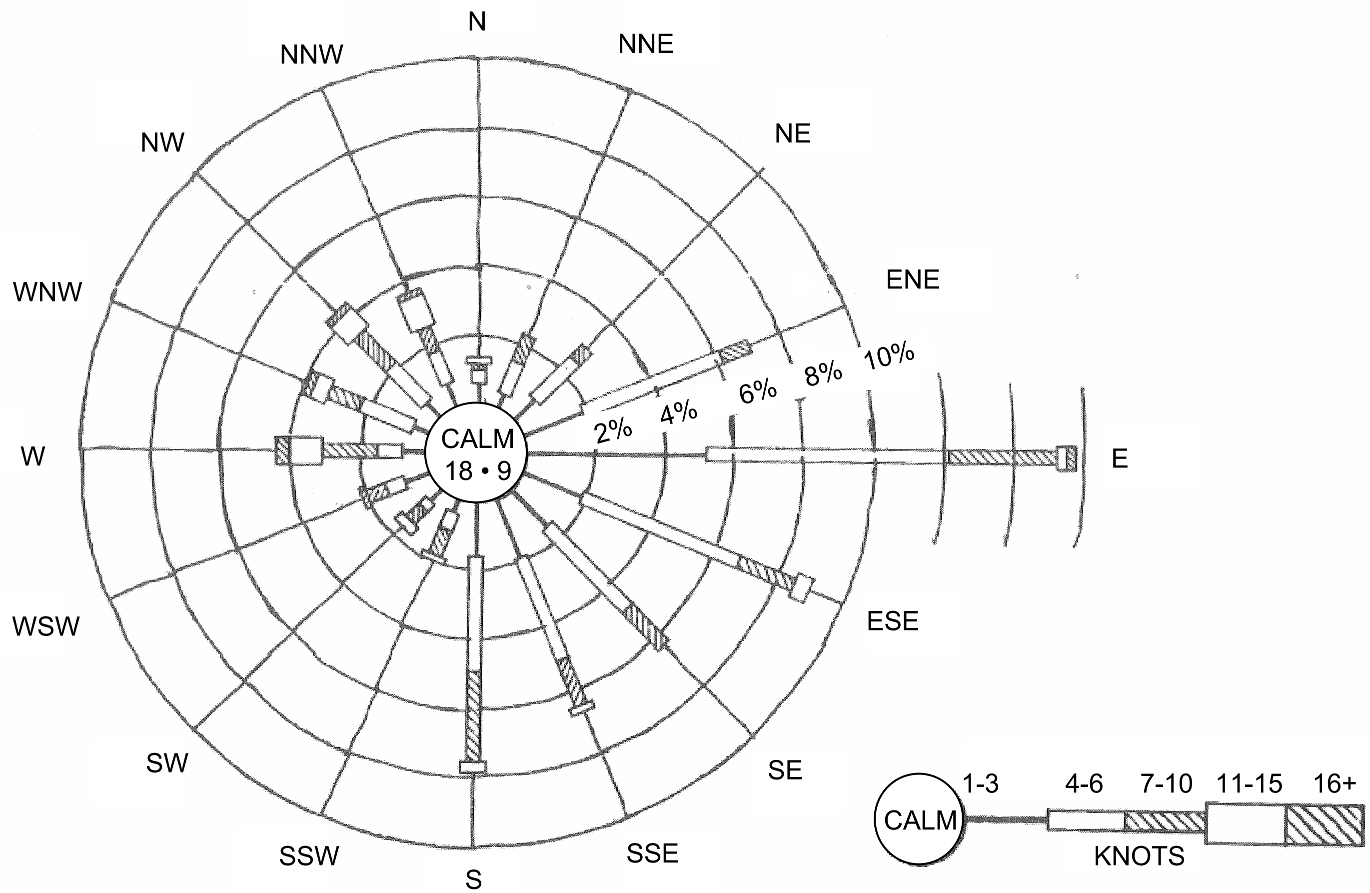
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LOCATIONS OF METEOROLOGY SYSTEMS	BASED ON DRAWING NO	SHEET / REV.



PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE WIND ROSE WINTER SEASON		SAR FIGURE NO. 2-36
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	AMENDMENT 20
SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:		BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

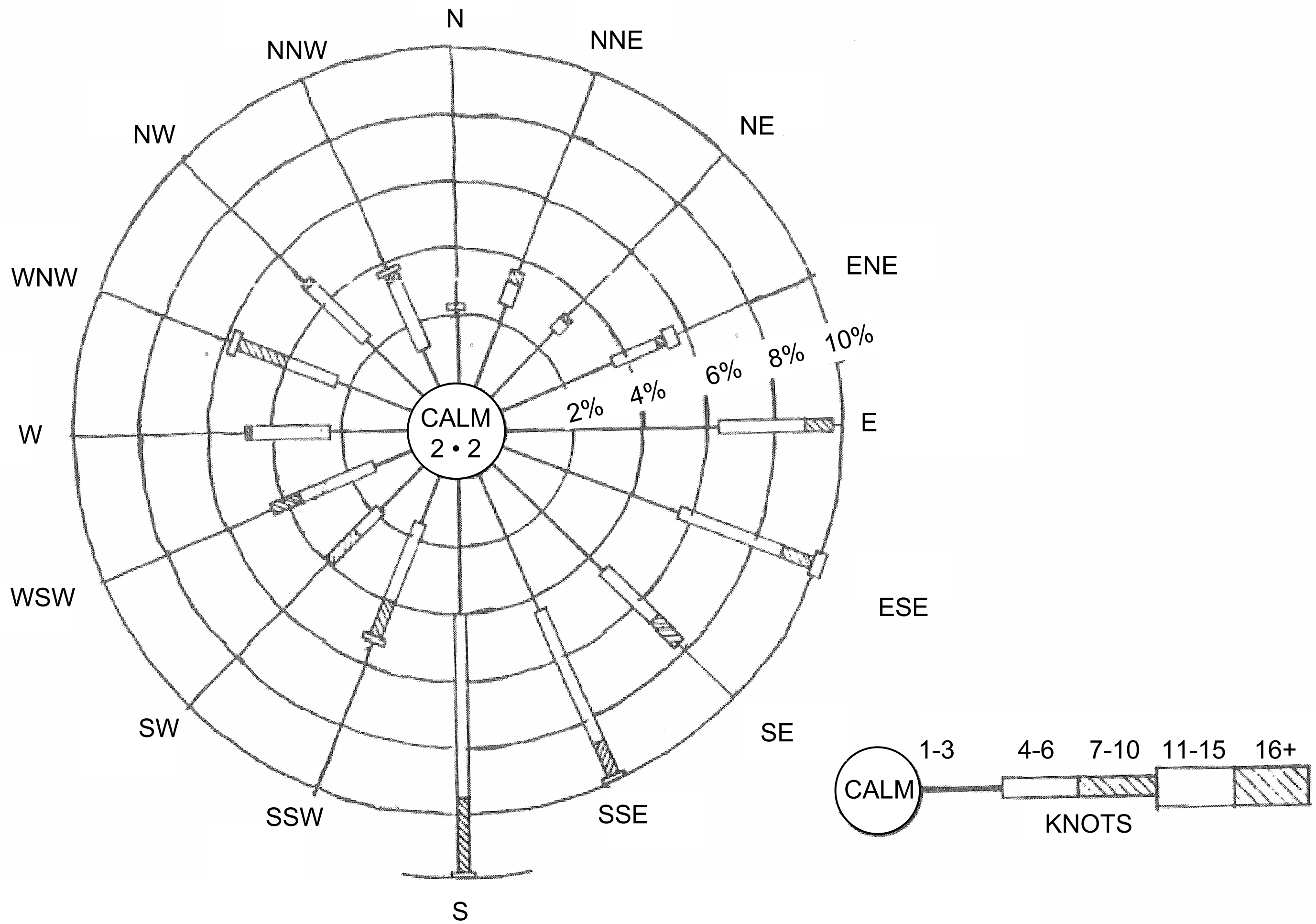
SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE WIND ROSE SPRING SEASON	SAR FIGURE NO. 2-37
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

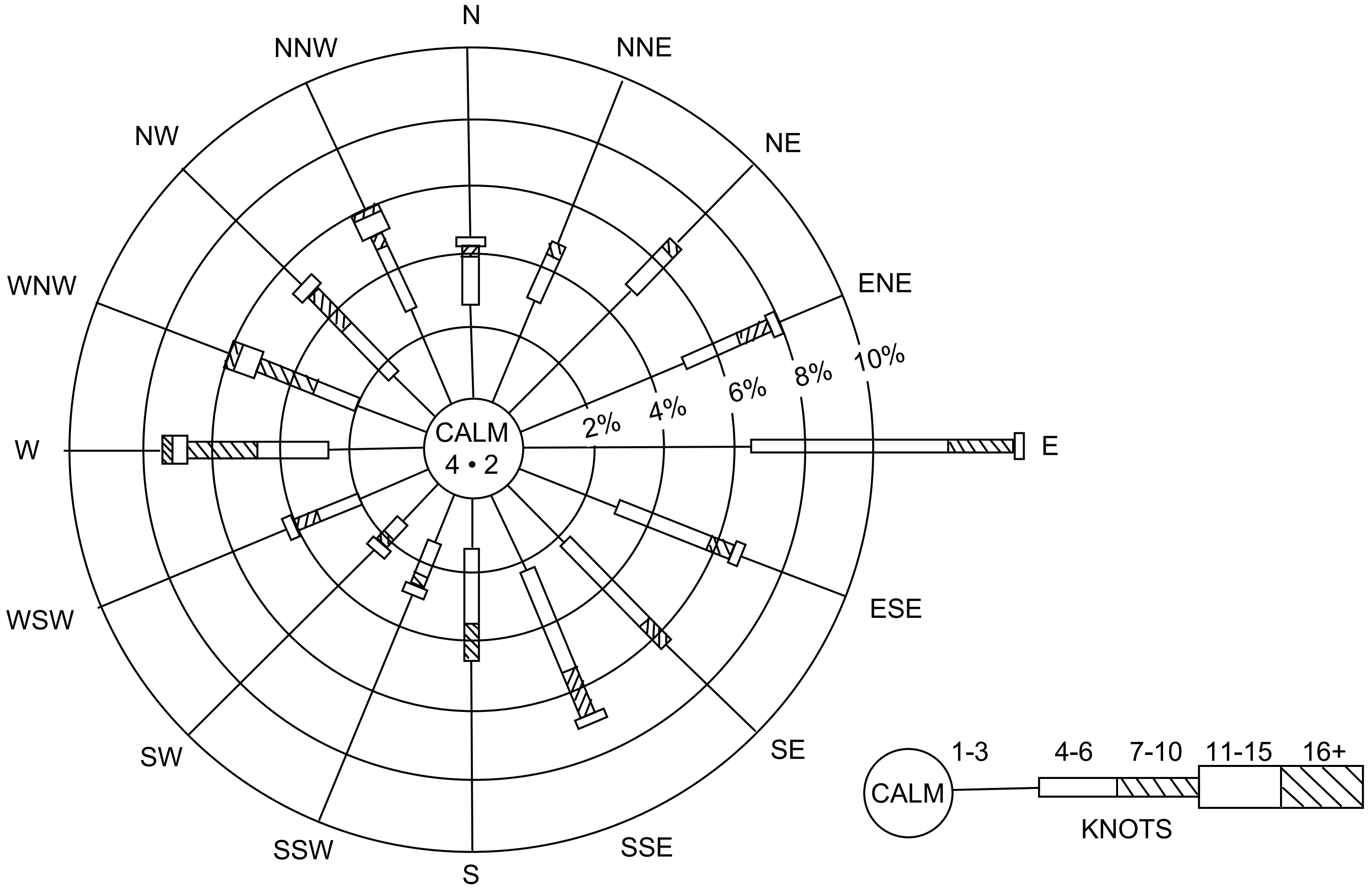
BASED ON DRAWING NO	SHEET	REV.



PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE WIND ROSE SUMMER SEASON			SAR FIGURE NO. 2-38
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20
			BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE WIND ROSE FALL SEASON	SAR FIGURE NO. 2-39
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

Entergy

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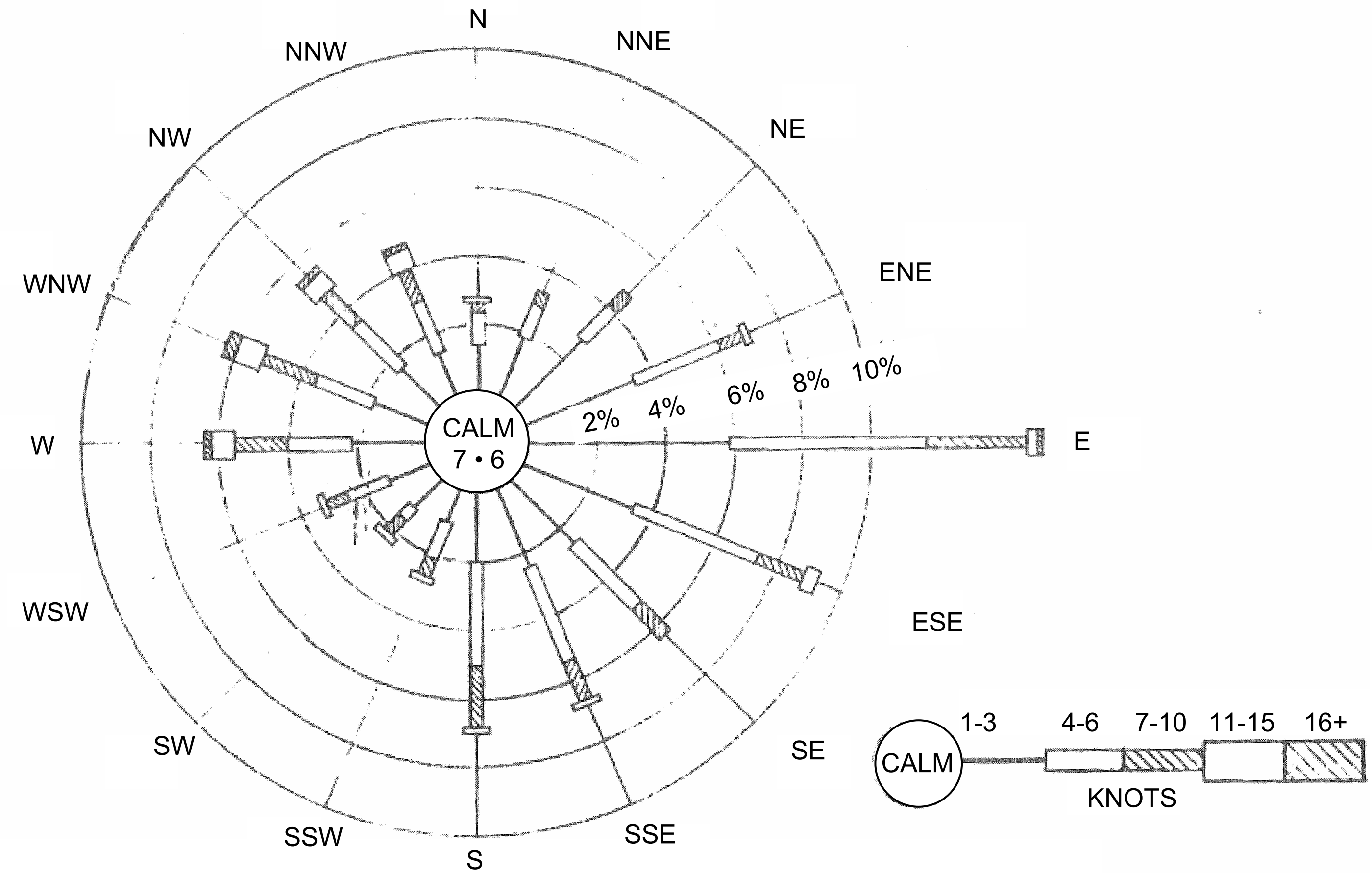
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE WIND ROSE ANNUAL			SAR FIGURE NO. 2-40		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
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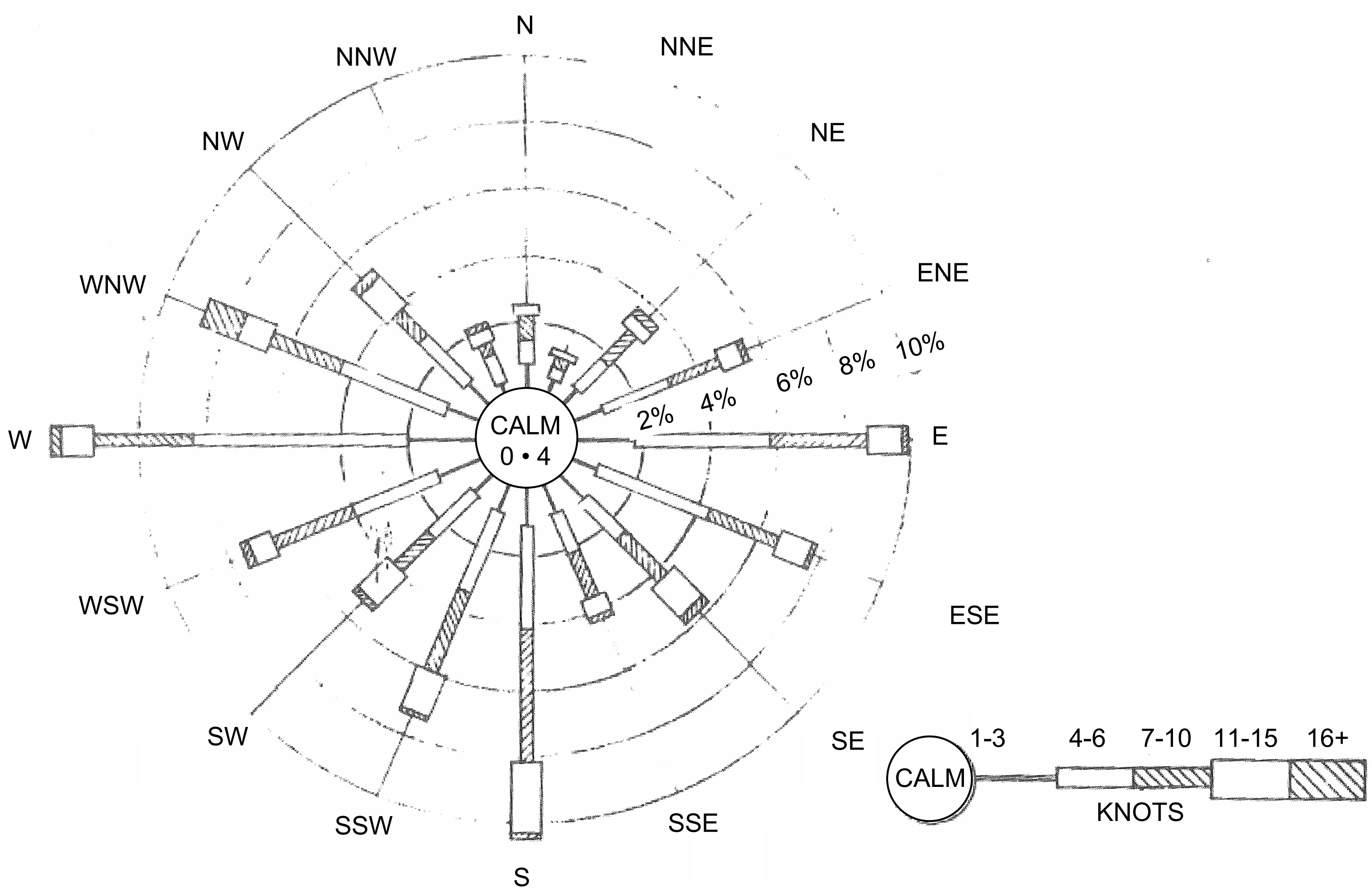
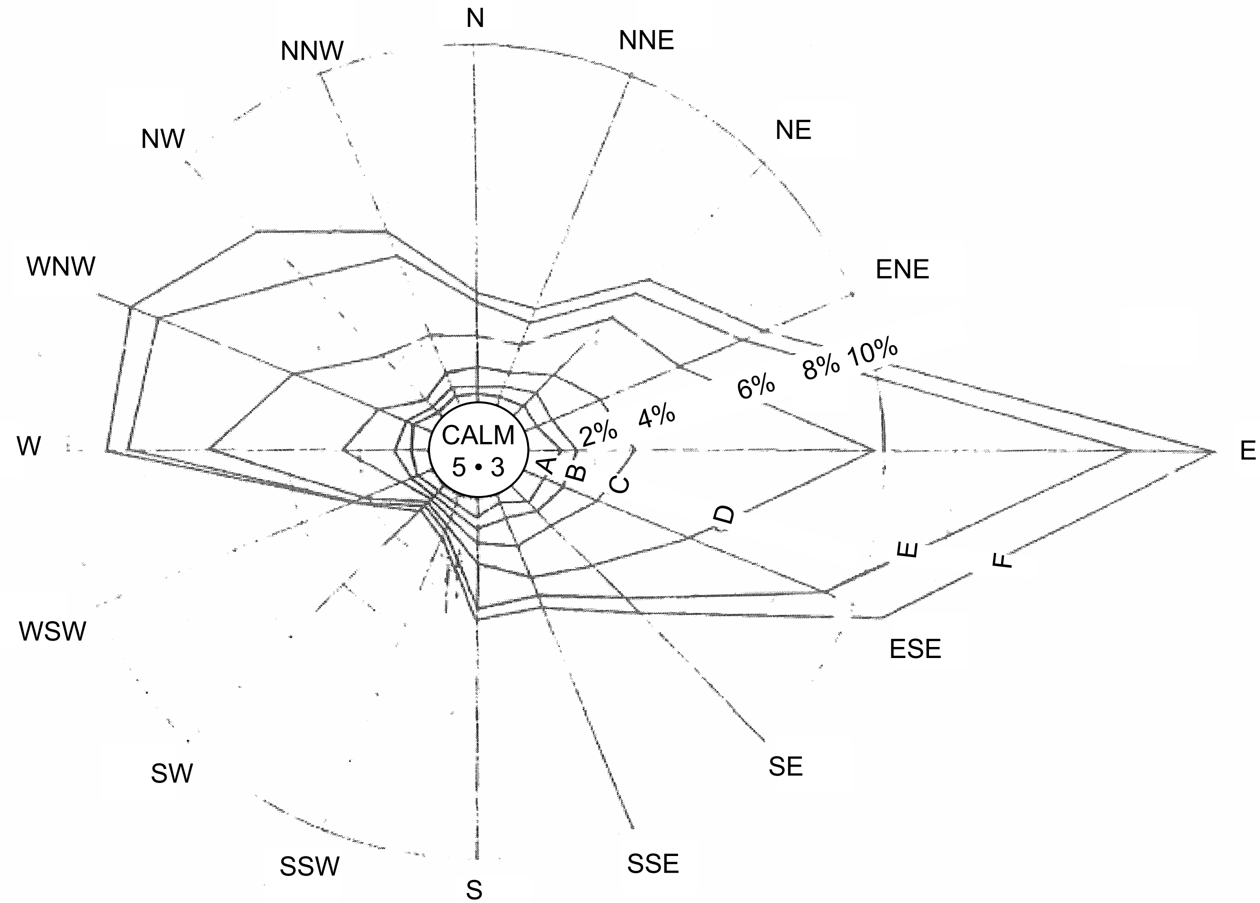


PERCENT FREQUENCY OF WIND DIRECTION AND MEAN SPPED (KT)

SEPTEMBER 9, 1967 – MAY 31, 1970

190 FOOT WIND ROSE ANNUAL			SAR FIGURE NO. 2-41
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY (SLADE METHOD)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE STABILITY ROSE WINTER SEASON			SAR FIGURE NO. 2-42		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
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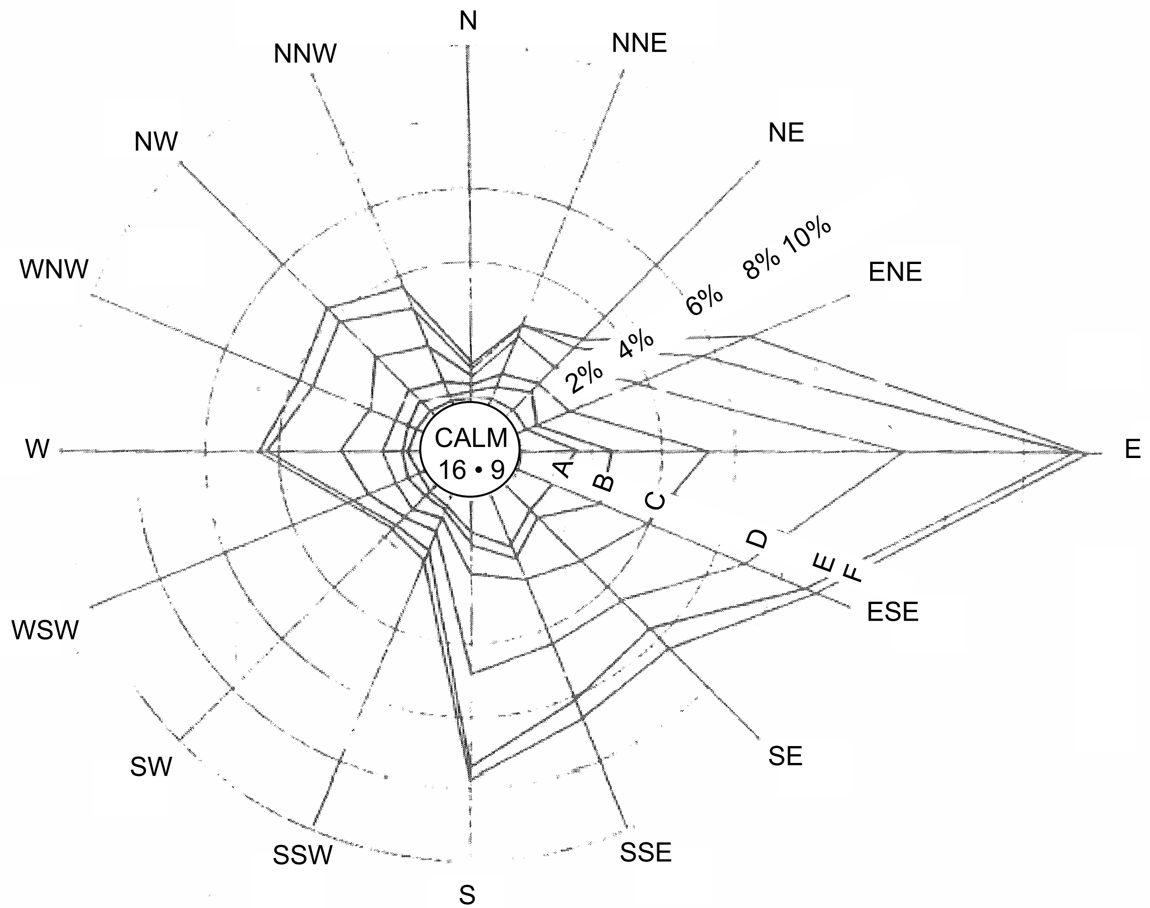
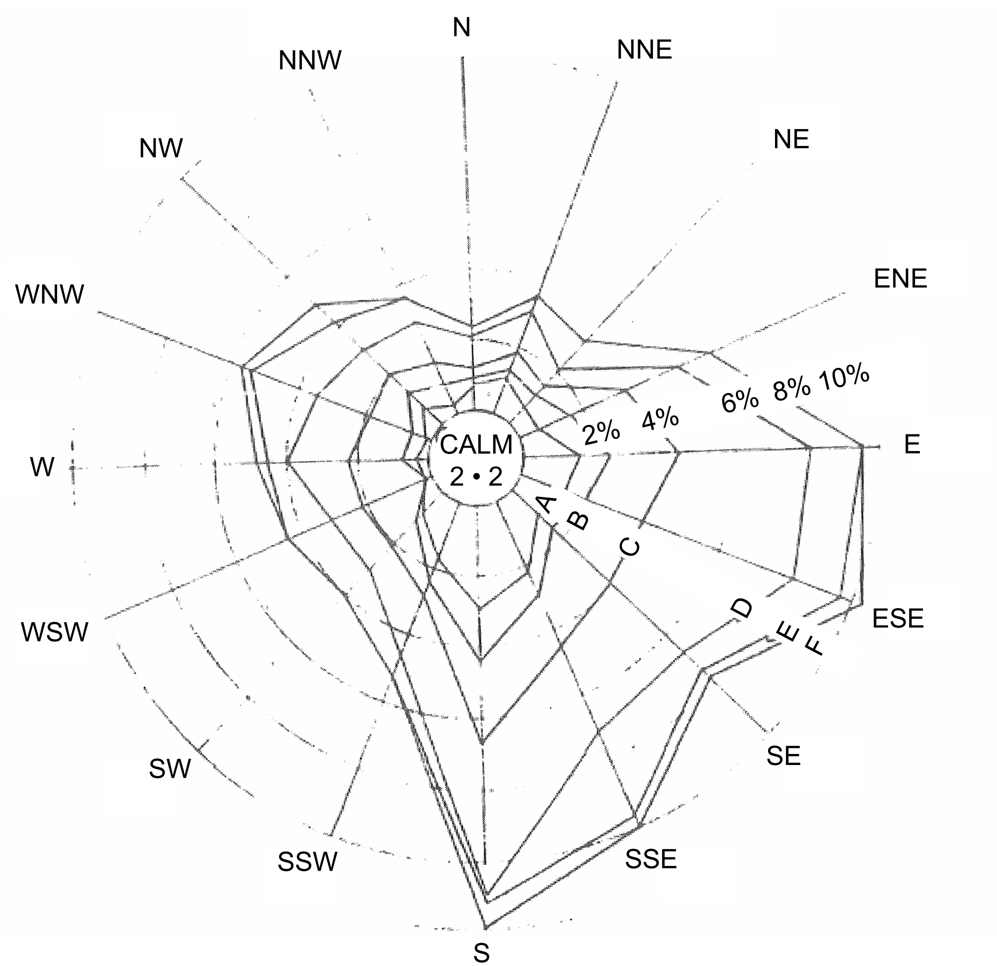


PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY (SLADE METHOD)

SEPTEMBER 9, 1967 – MAY 31, 1970

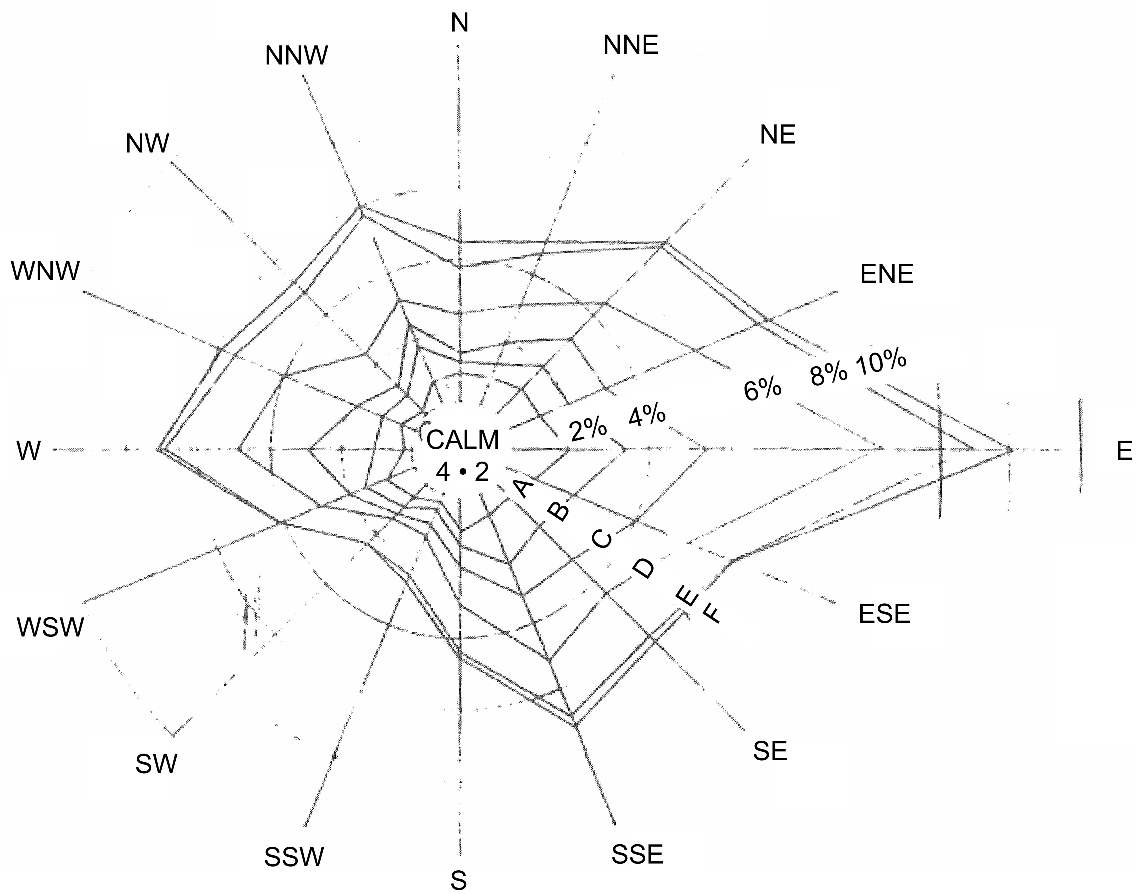
SURFACE STABILITY ROSE SPRING SEASON			SAR FIGURE NO. 2-43
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY (SLADE METHOD)

SEPTEMBER 9, 1967 – MAY 31, 1970

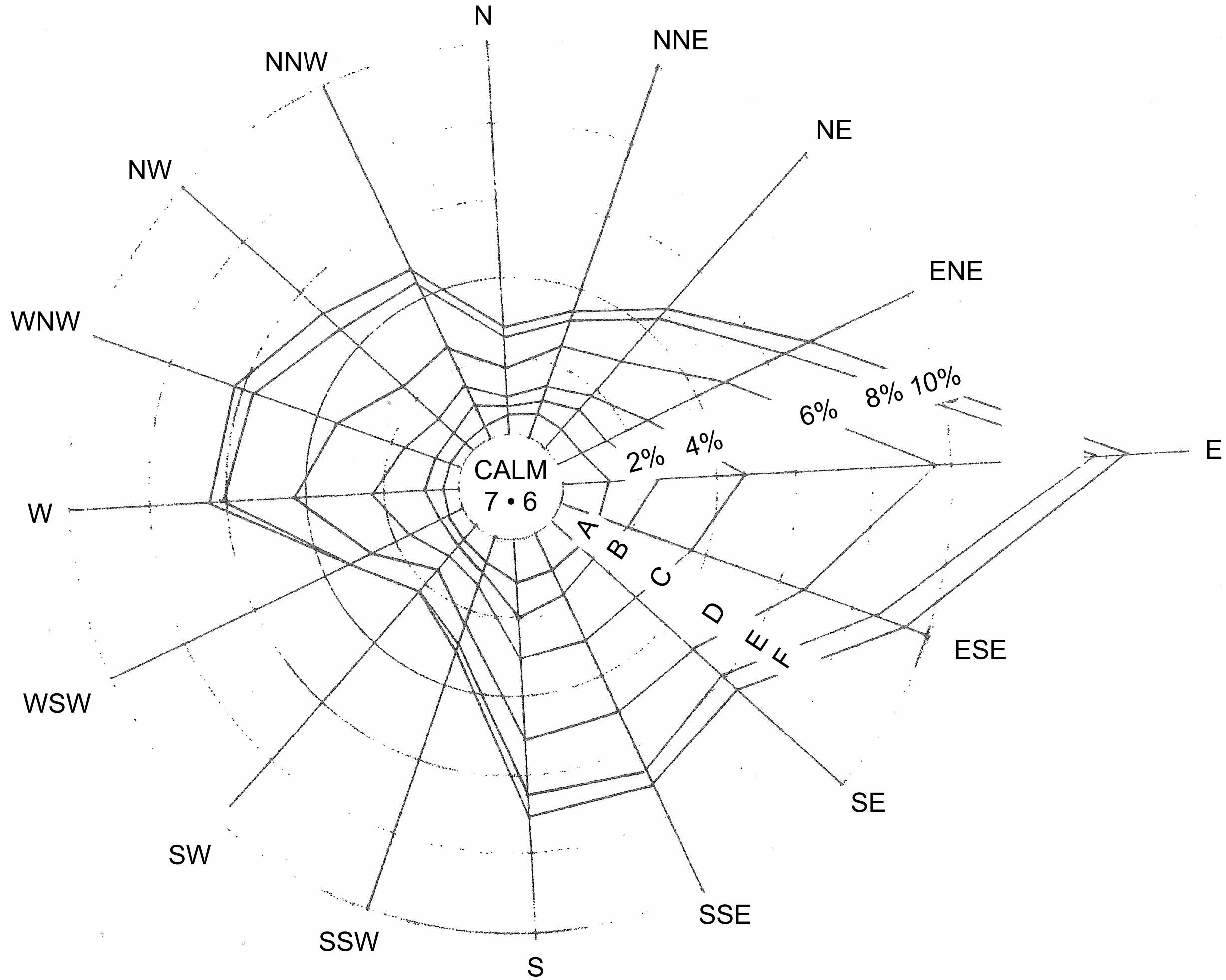
SURFACE STABILITY ROSE SUMMER SEASON			SAR FIGURE NO. 2-44
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY (SLADE METHOD)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE STABILITY ROSE FALL SEASON			SAR FIGURE NO. 2-45
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



PERCENT FREQUENCY OF PASQUILL STABILITY CATEGORY (SLADE METHOD)

SEPTEMBER 9, 1967 – MAY 31, 1970

SURFACE STABILITY ROSE ANNUAL			SAR FIGURE NO. 2-46		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
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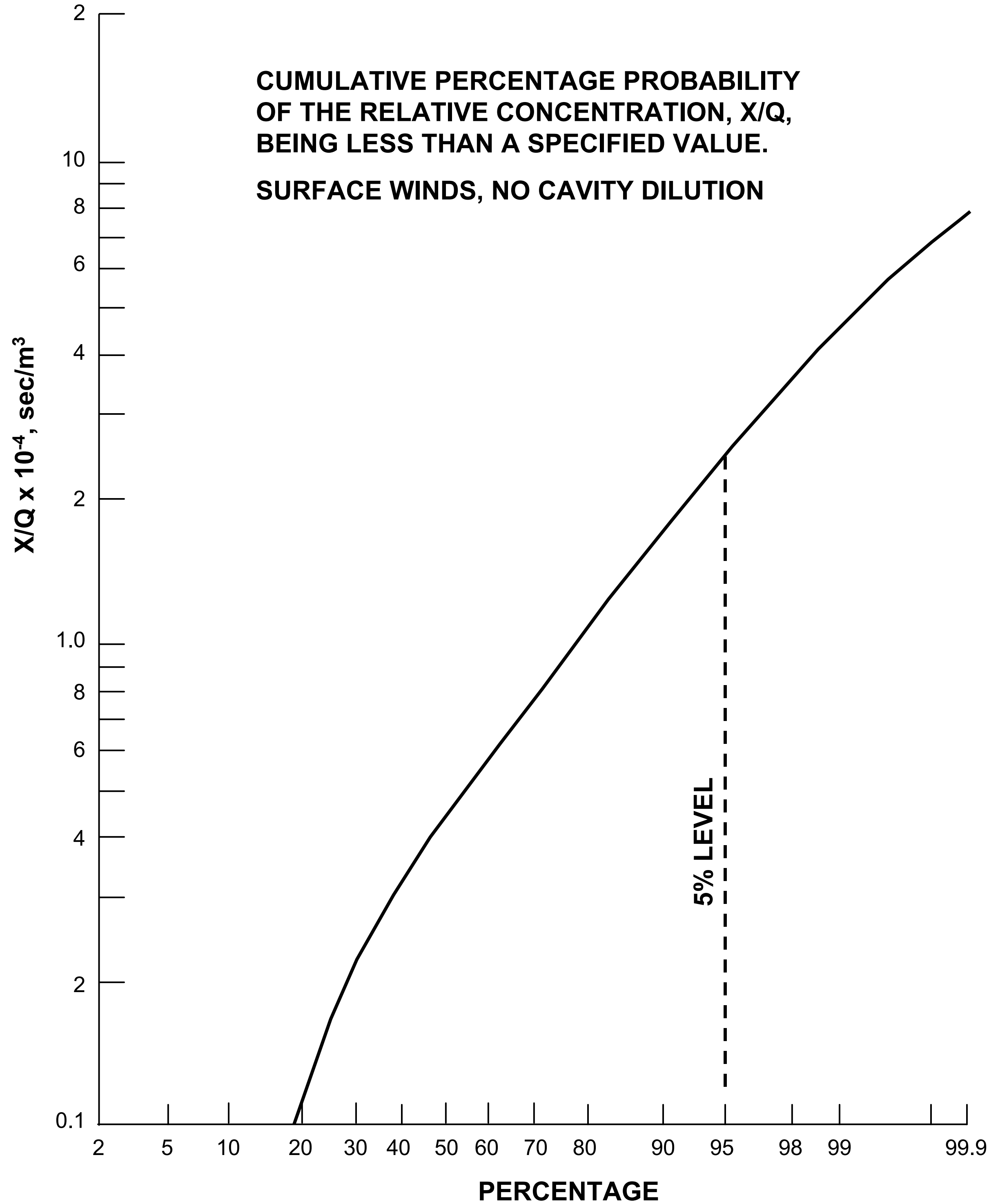
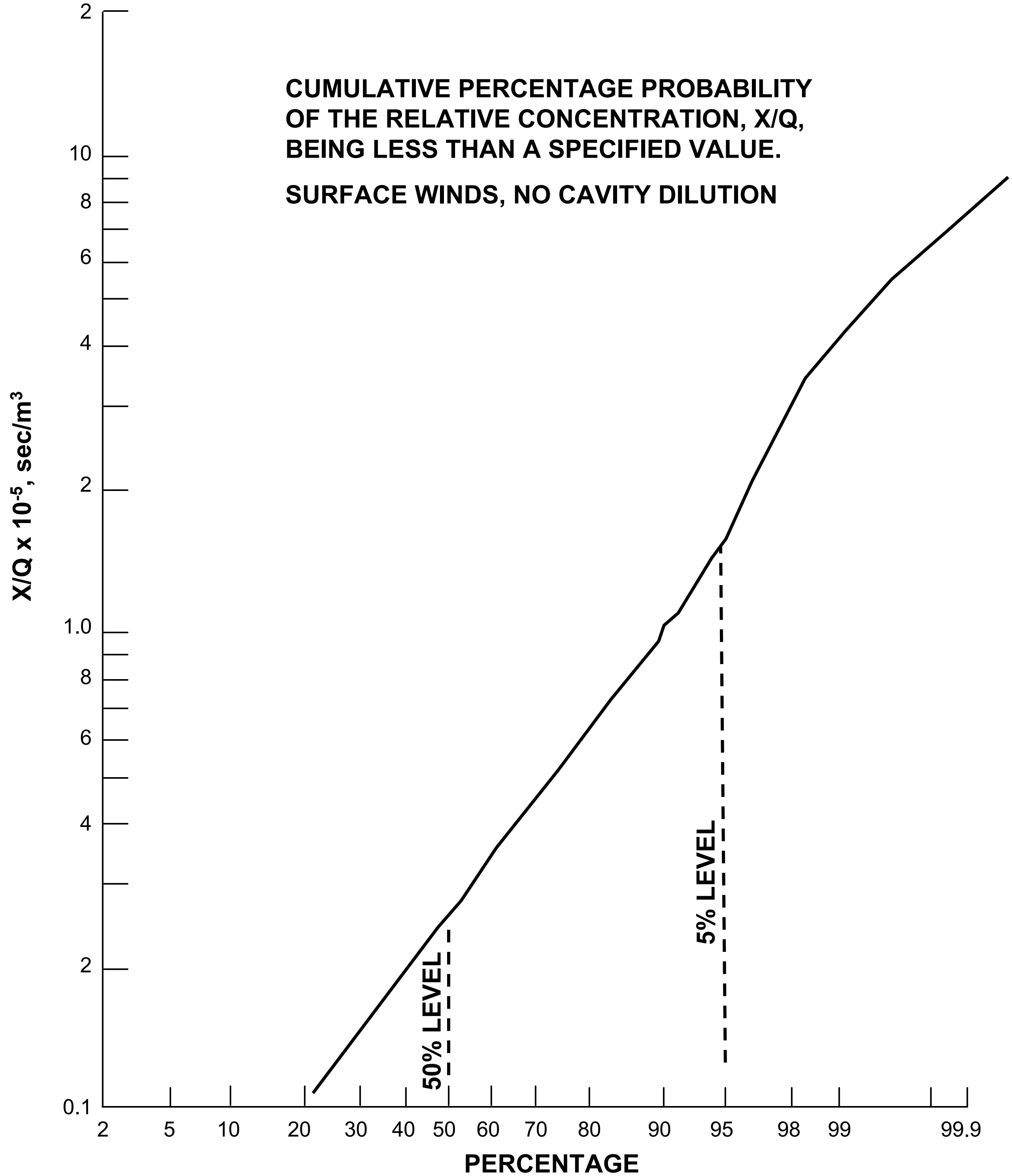


SAR FIGURE NO. 2-47

AMENDMENT 20

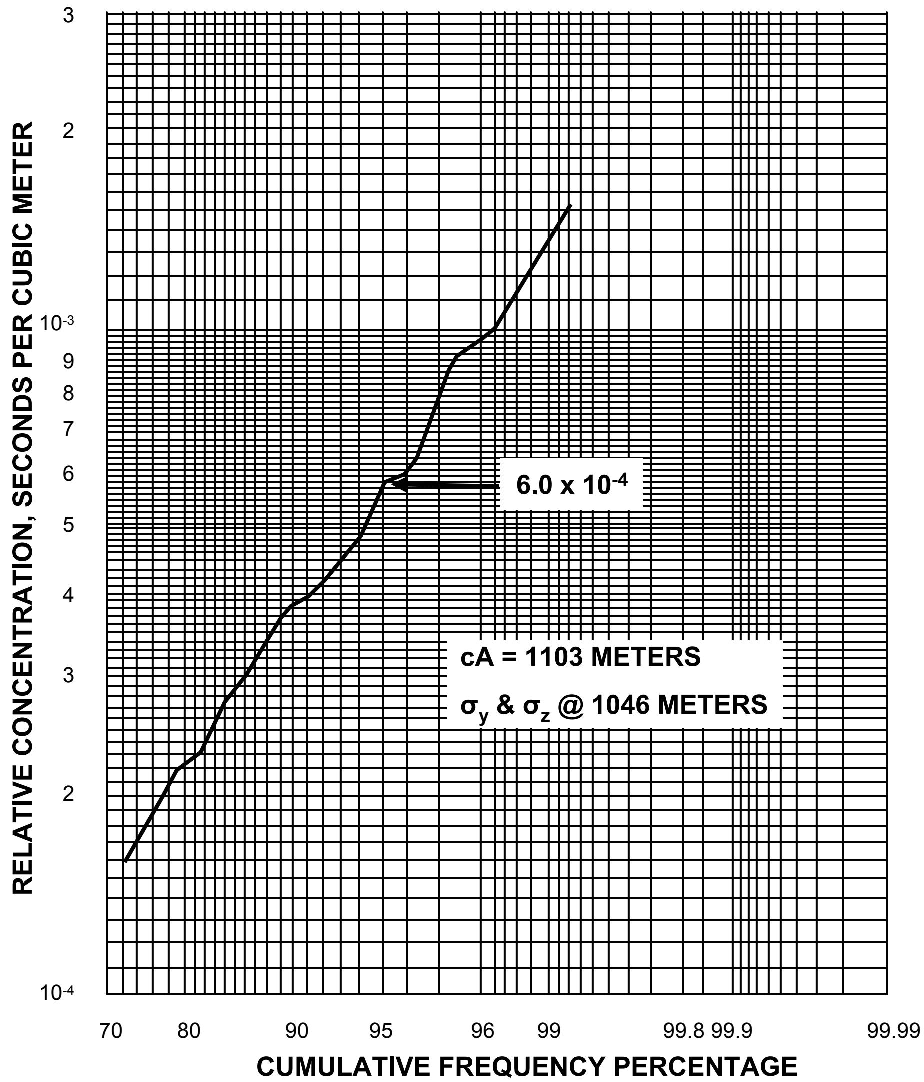
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
0 – 2 HOUR MODEL EXCLUSION DISTANCE	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 2-48

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
0 – 2 HOUR MODEL LPZ	BASED ON DRAWING NO	SHEET / REV.



σ_y & σ_z DETERMINED BY 30 – 190 FOOT ΔT PASQUILL CLASSES

WINDS OBSERVED AT 190 FEET AND REDUCED BY THE POWER LAW

$$U_{40} = U_{190}\ (40/190)^{P}$$

WHERE P = 0.18 FOR PASQUILL CLASSES A, B, C, & D

= 0.225 FOR PASQUILL CLASS E

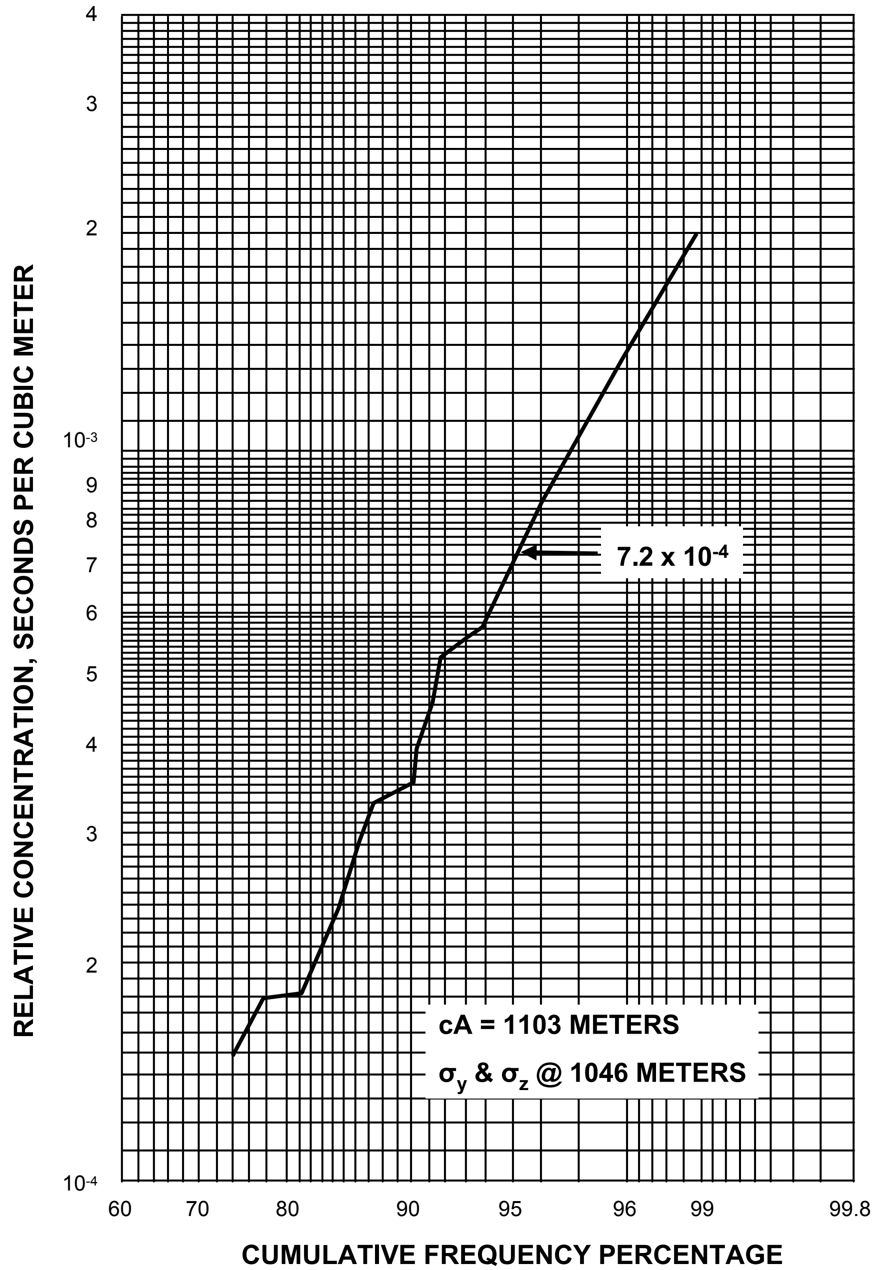
= 0.45 FOR PASQUILL CLASSES F & G

DATA PERIOD JULY 29, 1971 – FEB. 7, 1972

SAR FIGURE NO. 2-49

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
FREQUENCIES OF RELATIVE CONCENTRATIONS ASSUMING INVARIANT CENTERLINE WIND DIRECTION	BASED ON DRAWING NO	SHEET	REV.



σ_y & σ_z DETERMINED BY ΔT PASQUILL CLASSES AND OBSERVED BETWEEN 30 - 190 FT. ABOVE GRADE

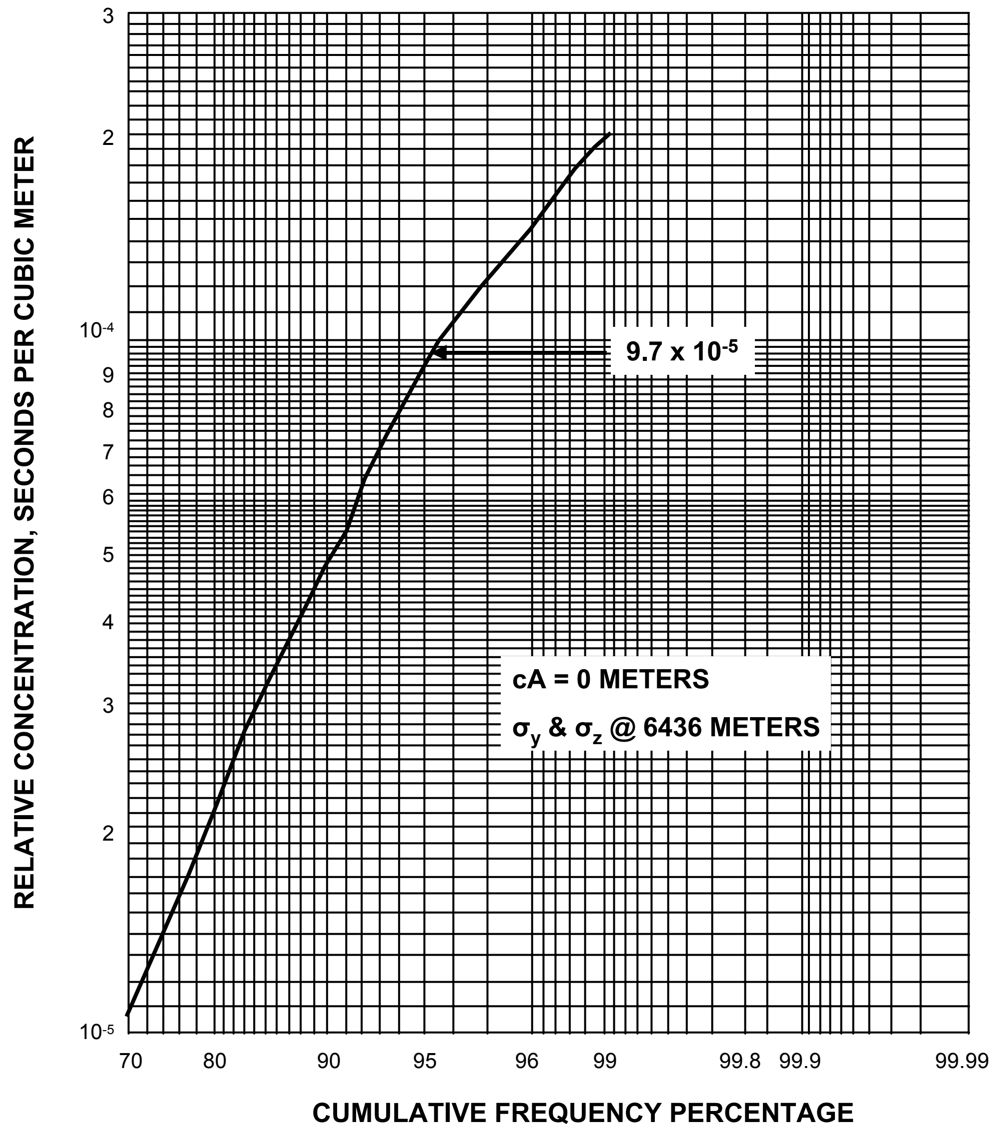
U OBSERVED AT 40 FT. ABOVE GRADE

DATA PERIOD FEB. 7, 1972 – JULY 29, 1972

SAR FIGURE NO. 2-50

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FREQUENCIES OF RELATIVE CONCENTRATIONS ASSUMING INVARIANT CENTERLINE WIND DIRECTIONS	BASED ON DRAWING NO	SHEET / REV.



σ_y & σ_z DETERMINED BY 30 – 190 FOOT ΔT PASQUILL CLASSES

U OBSERVED AT 40 FEET AND 190 FEET

WIND SPEEDS AT 190 FEET WERE REDUCED BY THE POWER LAW TO 40 FOOT LEVEL

DATA PERIOD JULY 29, 1971 – FEB. 7, 1972

SAR FIGURE NO. 2-51

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
FREQUENCIES OF RELATIVE CONCENTRATIONS ASSUMING INVARIANT CENTERLINE WIND DIRECTIONS	BASED ON DRAWING NO	SHEET	REV.

ARKANSAS NUCLEAR ONE
Unit 1

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UPDATE REFERENCE LIST

Section and references listed below denote documents that contain additional cross reference information used to update the SAR

Section	Cross Reference
3.1.2.3	Correspondence from Phillips, AP&L, to Ziemann, NRC, dated December 1, 1976. (1CAN127602)
3.3.4	Correspondence from Rueter, AP&L, to Ziemann, NRC, dated February 22, 1977. (1CAN027711)
3.2.4.1 3.2.4.1.2	Correspondence from Rueter, AP&L, to Howard, NRC, dated June 1, 1977. (1CAN067701)
3.1.2.3	Correspondence from Rueter, AP&L, to Davis, NRC, dated December 28, 1977. (1CAN127719)
3.1.2.3	Correspondence from Cavanaugh, AP&L, to Reid, NRC, dated November 9, 1978. (1CAN117804)
3.2.4.2.1	Correspondence from Trimble, AP&L, to Reid, NRC, dated October 21, 1980. (1CAN108018)
3.1.2.3 3.2.2 3.2.4	Correspondence from Cavanaugh, AP&L, to Reid, NRC, dated January 30, 1981. (1CAN018121)
3.2.4.1 3.2.4.1.2	Design Change Package 432, "Reactor Surveillance Specimens, Removal of Specimen Holder Tubes," 1976. (1DCP760432)
3.2.4.1.3 Table 3-39	Design Change Package 1019, "Core Components (BPRA & RNS Retainers)," 1979. (1DCP791019)
3.2.3.2.4 Table 3-21	Design Change Package 1049, "Internal Vent Valve Modification per B&W Field Change 164," 1980. (1DCP801049)
3.2.2.2.1.G	Correspondence from Marshall, AP&L, to Gagliardo, NRC, dated October 26, 1983. (1CAN108308)
3.2.4.1.2	Design Change Package 83-1063, "Upper Core Barrel Bolt Replacement."
3.2.4.3	Design Change Package 86-1030, "Replacement of Position Indicator Tubes."

Amendment 6

3.2.4.2 Table 3-27A Table 3-31 Figure 3-59 Figure 3-60	Design Change Package 84-1014, "Reactor Vessel Level Monitoring System."

UPDATE REFERENCE LIST (continued)

Section	Cross References
Amendment 7	
3.1.2 3.2.2 3.2.3 3.2.4 Reference List Associated Tables Associated Figures	Cycle 2 through Cycle 9 Reload Reports and References included in the Reload Report. The complete Reference List is included in the Chapter 3 Reference List.
Amendment 10	
3.2.4.2.1.1.C.1 3.2.4.2.1.1.C.5	Procedure 1409.296, Rev. 0, Permanent Change 1, "ANO-1 Fuel Assembly Component Inspection and Modification (Reconstitution)."
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3 REACTOR

3.1 DESIGN BASES

The reactor is designed to meet the performance objectives specified in Section 3.1.1 without exceeding the limits of design and operation specified in Section 3.1.2.

The objectives (Section 3.1.1) and limits of design and operation (Section 3.1.2) represent data and information pertinent to the first fuel cycle. For succeeding cycles of reactor operation, a reload report is prepared to document changes in nuclear, thermal hydraulic, and mechanical design of the reactor system. The most recent design data governing the operation of the Unit 1 core may be found in the Reload Report and its supporting documentation. The most current reload report is provided in Chapter 3A.

3.1.1 PERFORMANCE OBJECTIVES

The reactor is designed to operate at 2,568 MWt rated power with sufficient design margins to accommodate transient operation and instrument error without damage to the core and without exceeding the code pressure limits for the Reactor Coolant System (RCS).

The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients throughout core life. The effects of gas release, fuel dimensional changes, and corrosion- or irradiation-induced changes in the mechanical properties of cladding are considered in the design of fuel assemblies.

Core reactivity is controlled by Control Rod Assemblies (CRA), Burnable Poison Rod Assemblies (BPRA), and soluble boron in the coolant. Sufficient CRA worth is available to shut the reactor down with at least a one percent $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. Redundant equipment is provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA, and the rate at which reactivity can be added, is limited to ensure that credible reactivity accidents cannot cause a transient capable of damaging the RCS or significant fuel failure.

3.1.2 LIMITS

3.1.2.1 Nuclear Limits

The core has been designed to the following nuclear limits:

A. Fuel has been designed for a maximum local burnup of 55,000 MWd/mtU.

   In Cycle 5, a high burnup test assembly, designated Mark-BEB, was inserted in the core to undergo up to 4 cycles of irradiation. At the end of Cycle 8, this assembly was removed after having achieved a burnup of 57,152 MWd/mtU.

B. The power Doppler coefficient is negative. However, the control system is capable of compensating for reactivity changes resulting from either positive or negative nuclear coefficients.

C. The core will have sufficient excess reactivity to produce the design power level and lifetime without exceeding the control capacity or shutdown margin.

D. Controlled reactivity insertion rates have been limited to a maximum value of 1.1×10^{-4} (Δk/k)/s for a single regulating CRA group withdrawal, and 4.4×10^{-6} (Δk/k)/s for soluble boron removal.

E. Reactor control and maneuvering procedures will not produce peak-to-average power distributions greater than those listed in Table 3-1. The low reactivity worth of CRA groups inserted during power operation limits power peaks to acceptable values.

3.1.2.2 Reactivity Control Limits

The control system and operational procedures will provide adequate control of the core reactivity and power distribution. The following control limits will be met:

A. Sufficient control will be available to produce an adequate shutdown margin.

B. The hot standby, > 525 degrees F, shutdown margin of one percent Δk/k will be maintained throughout core life with the CRA of highest worth stuck out of the core.

C. CRA withdrawal rate limits the reactivity insertion rate to a maximum of 1.1×10^{-4} (Δk/k)/s for a single regulating group. Boron dilution is limited to a reactivity insertion rate of 4.4×10^{-6} (Δk/k)/s.

3.1.2.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic conditions based on the applicable CHF correlation.

The W-3 CHF correlation was used for DNBR analysis of Cycle 1. The following limits are applied to the W-3 correlation:

A. No central melting in the fuel at the design overpower (114 percent of rated power).

B. A 99 percent confidence that at least 99.5 percent of the fuel rods in the core are in no jeopardy of experiencing a Departure from Nucleate Boiling (DNB) during continuous operation at the design overpower.

C. The minimum allowable DNBR during normal operation and anticipated transients is 1.30.

D. The generation of net steam in the hottest core channels is permissible, but steam voids will be below the threshold for flow instabilities and, in any case, local steam quality is less than 22 percent at the point of minimum DNBR.

The B&W-2 (BAW-10000A) correlation has been developed and used in place of the W-3 correlation for Cycle 2 and all subsequent fuel cycles containing Mark-B fuel assemblies. The B&W-2 correlation is a realistic prediction of the burnout phenomenon and predicts DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. In applying the B&W-2 correlation, the following modifications to Cycle 1 DNB data are used:

A. The limiting design DNBR of 1.30 is used, corresponding to a 95 percent confidence level of a 95 percent probability that DNB will not occur;

B. The pressure range applicable to the correlation is extended downward from 2,000 to 1,750 psia; and,

C. The design overpower is changed from 114 percent to 112 percent of rated power.

D. The mass flow velocity range is between 0.94×10^6 to 4.0×10^6 lb/hr-ft^2.

E. The quality range is from -3% to 22%.

F. The equivalent diameter range is between 0.2 and 0.5 inches.

Beginning in Cycle 9 and subsequent cycles containing Mark-BZ fuel assemblies, the BWC correlation (BAW-10143-A) has been used. In applying the BWC correlation, the following limit changes, relative to the B&W-2 correlation, are used:

A. The limiting design DNBR is reduced from 1.30 to 1.18;

B. The pressure range applicable to the correlation is extended downward from 1,750 psia to 1,600 psia;

C. The quality range applicable to the correlation is extended outward from -3% to 22% to -20% to 26%.

D. The mass velocity range is between 0.43×10^6 to 3.8×10^6 lb/hr-ft^2.

Beginning in Cycle 20 and subsequent cycles containing Mark-B-HTP assemblies, the BHTP correlation (BAW-10241P-A) has been used. Refer to BAW-10241 for limitations on use of the BHTP correlation.

3.1.2.4 Mechanical Limits

3.1.2.4.1 Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with one or more reactor coolant pumps running, and shutdown conditions. No damage to the reactor internals will occur as a result of loss of pumping power.

Reactor internals are fabricated primarily from SA-240 (Type 304) material and designed within the allowable stress levels permitted by the ASME Code, Section III, for normal reactor operation and transients. Structural integrity of all core support assembly circumferential welds is assured by compliance with ASME Code, Sections III and IX, radiographic inspection acceptance standards and welding qualification.

The core support structure is designed as a Class 1 structure to resist the effects of seismic disturbance. Class 1 structures are those whose failure could cause uncontrolled release of radioactivity or those essential for safe reactor shutdown and the immediate and long-term operation following a Loss of Coolant Accident (LOCA). The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of CRAs. Core drop in the event of failure of the normal supports is limited by guide lugs so that the CRAs do not disengage from the fuel assembly guide tubes (Section 3.2.4.1).

The structural internals are designed to maintain their functional integrity in the event of any major LOCA. The dynamic loading resulting from the pressure oscillations because of a LOCA will not prevent CRA insertion.

Internal vent valves are provided to relieve pressure resulting from steam generation in the core following a postulated reactor coolant inlet pipe rupture so that the core will be rapidly recovered by coolant.

3.1.2.4.1.1 Allowable Stresses

The loading combinations and corresponding stress criteria, including the analytically predicted values of internals deflection for the combined maximum seismic and LOCA loadings, are given in Topical Report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss of Coolant Accident and Maximum Hypothetical Earthquake." Additional criteria for stresses due to flow-induced vibratory loads are given in B&W Topical Report BAW-10051, Revision 1, "Design of Reactor Internals and Incore Instrumentation Nozzles for Flow Induced Vibrations."

The effects of asymmetric loading due to a LOCA have been analyzed in detail in the B&W Topical Report BAW-1621, "B&W 177 FA Owners Group: Effects of Asymmetric LOCA Loadings - Phase II Analysis," July, 1980 and in Supplement 1 to this report dated June 1981.

In this report, linear elastic loadings are developed for the components and structures affected by the postulated primary piping guillotine ruptures. These loadings are developed for components and structures within the reactor vessel subcompartment and for the steam generator supports and unbroken primary coolant piping within the steam generator subcompartment. Each component or structure is then evaluated for the applied loadings and the resultant stresses are compared to the acceptance limits. Additional analyses are presented to verify the assumptions used in the loadings analysis and to evaluate vessel support stability and function. The results of the asymmetric LOCA loadings analysis for ANO-1 are summarized in the following paragraphs.

For postulated pipe ruptures within the reactor vessel subcompartment, the resulting stresses are within the acceptance limits except for the reactor cavity walls, which exceed limits by 10 percent. The analysis shows that the walls would have experienced localized yielding only, which is considered acceptable for structural integrity for reinforced concrete. Core flood line supports near the nozzle are overloaded, but these supports are not necessary for LOCA and the effects on piping stresses were included in the core flood piping analysis. The report results further demonstrate that core coolable geometry, core flood line integrity, and reactor vessel stability are maintained for the postulated pipe ruptures within the reactor vessel subcompartment.

For postulated pipe ruptures within the steam generator subcompartment, the steam generator support stresses and the unbroken piping stresses are within the acceptance limits.

3.1.2.4.1.2 Methods of Load Analysis Employed for Reactor Internals and Core

Static or dynamic analysis is used as appropriate. In general, dynamic analysis is used for earthquakes and the subcooled portion of the LOCA. For the relatively steady state portion of the LOCA, a static analysis is used.

Where it is indicated that substantial coupling, i.e. interrelationship, exists between major components of the Nuclear Steam System (NSS), such as the steam generator, the piping, and the vessel, the dynamic analysis includes the response of the entire coupled system. However, where coupling is found to be small, the component or groups of components are treated independently of the overall system.

The dynamic analysis for LOCA uses predicted pressure-time histories as input to a lumped-mass model. For an earthquake, actual earthquake records, normalized to appropriate ground motion, are used as input to the model. The output from the analysis is in the form of internal motions (displacements, velocities, and accelerations), motions of individual fuel assemblies, impact loads between adjacent fuel assemblies, and impact loads between peripheral fuel assemblies and the core shroud. Motions of the reactor vessel, internals, and core have been confirmed using a time history excited lumped-mass solution.

In addition, seismic analysis is also performed using a modal superposition and response spectra approach.

For the simultaneous occurrence of LOCA and the maximum earthquake, both time-history excitations are input to the system simultaneously such that maximum structural motions, indicating maximum stresses, are obtained. Outputs are those mentioned above.

The output from the lumped-mass model and additional information, such as pressure-time histories on separate internals and core components (including control rods), are used to calculate stresses and deflections. These stresses and deflections are compared to the allowable limits for the various loading combinations to ensure that they are less than the allowables. The allowable stress limits are shown in Section 4.1.2.

3.1.2.4.2 Core Components

3.1.2.4.2.1 Fuel Assembly

Fuel assemblies are designed for structural adequacy and reliable performance during core operation, handling, and shipping. Design criteria for core operation include steady state and transient conditions under combined effects of flow induced vibration, temperature gradients, and seismic disturbances.

Topical Report BAW-10133PA, Rev. 1 -- “Mark-C Fuel Assembly LOCA - Seismic Analyses”, gives an analysis of a Mark-C fuel assembly in which loads caused by hot leg and cold leg breaks and seismic activity are analyzed. The procedure outlined in this analysis is also applicable to the Mark-B fuel assemblies of ANO-1.

Spacer grids, located along the length of the fuel assembly, position fuel rods in a square array and are designed to maintain fuel rod spacing during core operation, handling, and shipping. Spacer grid to fuel rod contact loads are established to minimize fretting but to allow axial relative motion resulting from fuel rod irradiation growth and differential thermal expansion.

The fuel assembly upper end fitting is indexed to the plenum assembly by the grid immediately above the fuel assemblies to assure proper alignment of the fuel assembly guide tubes to the control rod guide tube. The guidance of the control rod assembly and axial power shaping rod assembly is designed such that these assemblies will never be disengaged from the fuel assembly guide tubes during operation.

The fuel assembly structural design criteria for the loads and permanent deflection of the Operating Basis Earthquake (OBE) are as follows:

A. Loads on the fuel assembly spacer grid are not to exceed the elastic limit of the spacer grid determined from tests performed on production grids.

B. There can be no permanent deformation of the fuel assembly spacer grids.

For the Design Basis Earthquake (DBE), LOCA (including effects of asymmetric loads), and simultaneous LOCA and DBE, the loads and permanent deflection limits are as follows:

A. Loads on the fuel assembly spacer grid are allowed to exceed the elastic limit, but the permanent deformation of the spacer grid cannot exceed that which could distort the guide tubes and prevent the insertion of the control rods. This value of permanent deformation is determined by tests on production grids.

B. To provide stability, loads on the control rod guide tubes and end spacer grid assembly are limited to 85 percent of the critical Euler buckling load. The value of 85 percent is chosen as a value, based on engineering judgment, so as not to design to failure.

C. Loads on the spacer grid welds are limited to 85 percent of the load that could cause failure. The value of this load is determined by tests on production grids.

D. Loads on the mechanical attachment of the end spacer grid skirt to the end fitting are limited to 85 percent of the load that could cause the attachment to fail. The value of this load is determined by tests on production end grid assemblies.

Actual numerical limits are shown in Table 3-1a and the test conducted to determine these limits are described in Section 3.3.

The Zircaloy-4 cladding is designed to withstand strain resulting from combined effects of reactor pressure, fission gas pressure, fuel expansion, and thermal and irradiation growth. Clad strain resulting from normal and abnormal operating conditions is limited as follows:

A. Cladding stress levels must be less than certain limits. The design criteria is:

   The stress intensity value of the primary membrane stresses in the fuel rod cladding, which are not relieved by small material deformation of the cladding, shall not exceed ⅔ of the minimum unirradiated yield strength at temperature (650 °F).

   Pellet-cladding interaction (PCI) and creep collapse induced stresses are not addressed here as small deformations of the cladding will relieve those stresses. Limits are based on ASME criteria. Stress level intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities are compared to Sm. Sm is equal to ⅔ of the minimum specified unirradiated yield strength of the material at the operating temperature (650 °F). The limits are as follows:

Primary general membrane stress intensities (Pm) must not exceed Sm.

Local primary membrane stress intensities (P1) must not exceed 1.5 * Sm. These include the contact stresses from spacer grid stop-fuel rod contact. Primary membrane + bending stress intensities (P1 +Pb) must not exceed 1.5 * Sm.

Primary membrane + Bending + Secondary stress intensities (P1+Pb+Q) must not exceed 3.0 * Sm. (1.5 * Sm is commonly used where possible).

Stress intensity calculations combine the stresses so that the stress intensity is maximized.

Classifications of Stresses:

Loading Condition	Stress Category
Pressure Stresses	Pm
Ovality Stresses	Pb
Spacer Grid Interaction	Pl
Flow Induced Vibration	Pb
Radial Thermal Expansion	Q
Differential Rod Growth	Q

Pressure and temperature inputs to the stress intensity analyses are chosen so that the operating conditions for all Condition II transients are enveloped.

B. Stresses relieved by small material deformation are permitted to exceed the yield strength. The strain limits for these conditions are:

Transient: The uniform transient strain (elastic and inelastic) should not exceed 1%. This is defined as the transient induced deformation with gage lengths corresponding to the cladding dimensions. The transient strain is calculated from the change in diameter of the fuel pellet during the maximum power transient the fuel rod is expected to undergo. This is done as the maximum strain would occur at the cladding inside diameter and the cladding deformation is driven by the fuel pellet expansion.

Total: The total strain (elastic and inelastic) of the cladding at End of Life (EOL) should not exceed 1.42%. This is defined as the change in cladding diameter from the time of pellet-cladding contact to the maximum diameter during a maximum power transient at the EOL.

C. For strain conditions which may result in low cycle fatigue, the following limits apply.

The total fatigue usage factor for all condition I, II events and one condition III event shall not exceed 90% of the material fatigue life.

D. Minimum clad collapse pressure margins are as follows:

1. Ten percent margin over system design pressure, on short-time collapse, at fuel rod end voids.

2. End voids must not collapse (must be either freestanding or have adequate support) on a long-time basis.

3. Ten percent margin over system operating pressure, on short-time collapse, at hot spot average temperature of the clad wall.

4. Clad must be freestanding at design pressure on a short-time basis at approximately 733 °F hot spot temperature averaged through the clad wall.

E. Creep collapse of the cladding due to creep ovalization will not be predicted during the incore life of the fuel rod. Creep collapse is predicted to occur when:

Ovality induced stress levels in the cladding are more than the yield stress.

Creep ovalization rate is greater than 0.1 mil ovality per hour.

The creep collapse analysis methodology is described in BAW-10084.

3.1.2.4.2.2 Control Rod Assembly (CRA)

Absorber material cladding used on the CRA, Axial Power Shaping Rod Assembly (APSRA), and the BPRA is designed to the same criteria as the fuel rod clad as applicable to absorber material characteristics. Clearance is provided between the rods of each of these assemblies and fuel assembly guide tubes to permit coolant flow to limit operating temperature of the absorber materials. In addition, this clearance is designed to permit rod motion as required during reactor operation under any condition, including seismic disturbances.

Excessive stress in the CRA components during trip of the rod drive mechanism is prevented by use of conservative design stress limits and by hydraulic snubbing in the drive mechanism to minimize shock.

3.1.2.4.2.3 Orifice Rod Assembly

The Orifice Rod Assembly (ORA) is designed to have adequate clearance when inserted into the fuel assembly guide tubes to permit coolant flow without unacceptable mechanical interference between the rod assembly and guide tubes under any operating condition.

3.1.2.4.3 Control Rod Drive Mechanisms

3.1.2.4.3.1 Shim Safety Drive

The shim safety control rod drives provide CRA insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which have been established to limit power peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. The normal insertion and withdrawal velocity has been established as 30 inches per minute.

The drive provides a trip of the CRA which results in a rapid shutdown of the reactor for conditions that cannot be handled otherwise by the RPS. The trip setpoint is based on the results of various reactor emergency analyses, including instrument and control delay times and the amount of reactivity that must be inserted before deceleration of the CRA occurs. The maximum travel time for a ⅔ insertion on a trip command of a CRA has been established as 1.40 seconds.

The control rod drives can be coupled and uncoupled to their respective CRAs without any withdrawal movement of the CRA.

Materials selected for the control rod drive are capable of operating within the specified reactor environment for the life of the mechanism without any deleterious effect. Adequate clearances are provided between the stationary and moving parts of the control rod drives so that the CRA trip time to full insertion will not be adversely affected by mechanical interference under all operating conditions and seismic disturbances.

3.1.2.4.3.2 Axial Power Shaping Drive

The axial power shaping drives operate similarly to the shim safety drives except that the trip function has been eliminated. They have the same insertion and withdrawal, velocity of 30 inches per minute and can be coupled and uncoupled to their respective APSRA without any withdrawal movement of the assembly.

3.2 REACTOR DESIGN

3.2.1 GENERAL SUMMARY

The important core design, thermal, and hydraulic characteristics are tabulated in Table 3-1.

For succeeding cycles of reactor operation, a Reload Report is prepared to document changes in nuclear, thermal hydraulic, and mechanical design of the reactor system. The most recent design data governing the operation of the Unit 1 core may be found in the Reload Report and its supporting documentation. The most current reload report is provided in Chapter 3A.

Changes made in the general core design in succeeding cycles include the following:

APSR position limits were initiated in Cycle 3 to provide additional control of power peaking and assurance that LOCA kW/ft limits will not be exceeded.

Cycle 5 initiated push-pull operation throughout the entire cycle. Cycle 6 initiated a complete APSR withdrawal near the end of cycle. This mode of operation has continued throughout subsequent cycles. Operation in the push-pull mode reduced power peaking in comparison to the rodded mode of Cycles 1 through 4 and withdrawal of the APSRs near the end of cycle allowed the cycle to operate longer at full power.

Starting in Cycle 7, the LBP length was shortened by removing burnable poison material from the top of the poison stack. In Cycle 7 the poison length was 117 inches (as compared to the standard length of 126 inches) and in Cycle 8 a poison length of 121.5 inches was used. Cycle 9 used the same LBP length as Cycle 8. This shortening of the LBP stack length was to provide operation at a more positive imbalance, which would give increased effective maneuvering room at the beginning of cycle.

Starting with Cycle 8, the center control rod was removed to allow the inclusion of a vent valve in the reactor head. Removing this control rod reduced the number of full length control rods from 61 to 60. Due to potential interference with the support structure, no CRA or BPRA inserts are allowed in the center core location.

The black APSRs were replaced with gray APSRs in Cycle 9. The black APSRs used in Cycles 1 through 8 contained 36 inches of Ag-In-Cd poison material while the gray APSR poison material consists of 63 inches of Inconel. This change was affected to reduce power peaking.

Starting in Cycle 22, an alternate extended life control rod will replace the existing extended life control rods that have reached the end of their design lifetime. These alternate control rods can take the place of any control rod in the reactor.

3.2.2 NUCLEAR DESIGN AND EVALUATION

This section presents the evaluation of significant core parameters and shows that the basic design of the core satisfies the performance limits and objectives of Sections 3.1.2.1 and 3.1.2.2.

Demonstration of satisfying core performance limits and objectives for the current fuel cycle are provided in Section 5 of Chapter 3A, Reload Report, and its supporting documentation.

3.2.2.1 Nuclear Characteristics of the Design

A summary of the nuclear characteristics of the initial core and nominal reload core is given in Table 3-2.

3.2.2.1.1 Excess Reactivity

The excess reactivity associated with various core conditions is tabulated in Table 3-3.

The minimum critical mass, with and without xenon and samarium poisoning, may be specified as a single assembly or as multiple assemblies in various geometric arrays. The unit fuel assembly has been investigated for comparative purposes. A single cold, clean assembly containing a maximum probable enrichment of 3.5 weight percent is subcritical in unborated water. Two assemblies side-by-side would be supercritical under these conditions.

3.2.2.1.2 First Cycle Reactivity Control Distribution

The control of the excess reactivity shown in Table 3-3 is provided by soluble boron, burnable poison rod assemblies, and control rod assemblies. The distribution of this control is shown in Table 3-4 for the beginning of the first fuel cycle.

A typical variation in critical boron concentration over the initial fuel cycle is shown in Figure 3-1. The curve reflects the burnout of the burnable poison of the fuel assemblies depicted in Figure 3-2.

The general use of movable Control Rod Assemblies in controlling excess reactivity is illustrated in Table 3-4, and a detailed account of movable control rod use is provided in Section 3.2.2.1.3.

Cycle 1 operated in a rodded mode of operation throughout much of its life. The control rods inserted in the core (usually Bank 7 almost totally inserted and Bank 6 slightly inserted) were called the transient banks. Midway through Cycle 1, the control rod groupings were changed so that different control rods were assigned to Banks 1 through 7 (the designations Transient Group 1 and Transient Group 2 denote this rod interchange.) Near the end of Cycle 1, Bank 7 was withdrawn so that the core operated in a push-pull mode for the last several weeks. The availability of control rod movement, soluble boron concentration changes, and burnable poison depletion allowed reactivity to be controlled by three different means. This is reflected in Table 3-4.

Cycles 2 through 4 were also operated in a rodded mode, but no rod interchange occurred in mid-cycle since these were much shorter cycles than Cycle 1. Cycles 2 and 3 did not contain burnable poison clusters, thus, reactivity changes due to fuel burnup was controlled by soluble boron means. Burnable poison, clusters were re-introduced in Cycle 4. The use of control rods, burnable poison, and soluble boron to control reactivity in Cycle 4 was similar to that of Cycle 1.

Cycle 5 saw the initiation of operating the entire cycle in a push-pull mode. This mode of operation has continued through all succeeding cycles. Burnable poison clusters are present in all these cycles. In this mode, the burnable poison clusters and long-term changes in soluble boron control reactivity changes due to fuel burnup. The control rods, coupled with temporary changes in soluble boron concentration, are used primarily for transient xenon effects and power Doppler effects during power level changes.

3.2.2.1.3 Reactivity Shutdown Analysis

The ability to shut down the core under hot conditions is illustrated in Table 3-5. In this tabulation, the first cycle is evaluated at the Beginning of Life (BOL) and the End of Life (EOL) for shutdown capability.

Examination of Table 3-5 for excess control rod worth (Item 3) shows that, with the highest worth Control Rod Assembly (CRA) stuck out, the core can be maintained in a subcritical condition with operating boron concentrations. The following conservatisms are included in the table:

Item 1 (a) - The Doppler deficit shown reflects the maximum expected fuel temperatures for the particular time in core life.

Item 2 (b) - The calculated control rod worth has been reduced 10 percent.

Item 2 (d) - The calculated worth of the stuck rod was not reduced.

Under conditions where a cooldown to reactor building ambient temperature is required, concentrated soluble boron will be added to the reactor coolant to produce a shutdown margin of at least one percent $\Delta k/k$. BOL boron levels for several core conditions are listed in Table 3-7 along with boron worth values. The conditions shown with no CRA illustrate the highest requirements.

Table 3-5a illustrates shutdown margin for Cycle 8, which is a typical calculation for those reload cycles operating in the push-pull mode and with an APSR withdrawal near the end of cycle. Shutdown margin is shown at beginning of Cycle 8, just before the planned APSR withdrawal at 380 EFPD, and at the planned end of cycle of 420 EFPD. The following conservatisms are used in calculating shutdown margin for these cycles:

1. The bounding effect due to poison burnup in the control rods throughout the cycle is included.
2. The worth of the inserted Banks 1-7 (less the stuck rod) has been decreased by 10%.
3. Flux redistribution that occurs when going from full power to zero power, including potential effects due to xenon maldistribution, are included.

The required shutdown margin for reload cycles is 1% $\Delta k/k$, the same as in Cycle 1.

3.2.2.1.4 Control Rod Groups for Operation

Figures 3-5 and 3-6 show the position and function of all control rod groups over the first fuel cycle. Figure 3-5 shows the rod configuration for the first 250 full power days (2,568 MWt) and Figure 3-6 shows rod groupings over the remainder of the first cycle.

The worths of the transient rod groups are listed in Table 3-6.

Starting in Cycle 8, the control rod in H-8 (the center of the core) was removed to allow the installation of a vent valve in the reactor head. This reduced the number of full length control rods from 61 to 60. The control rod configuration for Cycle 8 is shown in Figure 3-6a. The worths of Banks 6, 7, and 8 for Cycle 8 are illustrated in Table 3-6a. Due to potential interference with the support structure, no CRA or BPRA inserts are allowed in the H-8 location.

The rod worth calculations were performed with the PDQ07 (see Reference 17) - HARMONY (see Reference 14) computer program in two dimensions. Rod Group 8 (APSRs) was assumed to be withdrawn. Rod worths for sequential withdrawal of the full length control rods is shown in Figures 3-7 and 3-8 for the BOL and EOL conditions for the first fuel cycle. The location and worth of the single most reactive control rod is also shown on these two figures.

Figures 3-9 and 3-10 give the potential ejected control rod worth at BOL for Transient Bank 1 and at EOL for Transient Bank 2. Two values are shown on each figure since the calculations were performed in two dimensions and the APSRs had to be considered either fully withdrawn or as full length rods. For safety analysis calculations in Chapter 14, the ejected rod worth is conservatively assumed to be 0.65 % $\Delta k/k$.

Table 3-6a also gives the maximum ejected rod worths for Cycle 8 at Hot Zero Power conditions with groups 5 through 7 fully inserted. With the push-pull mode of operation, the core operates with all full length rods out except for a small amount of rod bite from Bank 7. Therefore, there is little rod material to be ejected and the potential ejected rod worth is much less than that assumed in the Chapter 14 safety analysis.

3.2.2.1.5 Reactivity Coefficients

Reactivity coefficients form the basis for studies involving normal and abnormal reactor operating conditions. These coefficients have been investigated as part of the analysis of this core and are described below as to function and overall range of values.

A. Doppler Coefficient

The Doppler coefficient of reactivity reflects the change in reactivity as a function of fuel temperature. A rise in fuel temperature results in an increase in the effective absorption cross section of the fuel (the Doppler broadening of the resonance peaks) and a corresponding reduction in neutron production. The range for the Doppler coefficient under operating conditions is expected to be -1.1×10^{-5} to -1.7×10^{5} $(\Delta k/k)/°F$.

The Doppler coefficient is due primarily to Doppler broadening of the ^{238}U resonances with increasing fuel temperature. Temperature-dependent resonance integrals, which include Doppler broadening, are calculated by the B&W proprietary RIP program (Section 3.2.2.2.1.A) and are based on the experimental work of Hellstrand, Blomberg, and Horner (see Reference 37). A comparison of calculated to experimental resonance integrals for UO_2 rods of different radii is presented in Figure 3-11. The curves for r = 0.5 are representative of this core.

Uncertainties in the calculated values of the Doppler coefficient may be attributed in part to the slight mismatch between the RIP and Hellstrand calculations but primarily to uncertainties involved in predicting fuel temperatures. Since the Doppler coefficient is a function of fuel temperature, a value calculated at a certain power level is dependent on the accuracy of predicting the fuel temperature at that power level.

In the nuclear calculations, this uncertainty due to fuel temperature is applied conservatively. That is, the highest expected fuel temperature is used to determine the Doppler deficit for shutdown reactivity analysis and a lower fuel temperature is used for the conservatism in the xenon stability analysis.

B. Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. The expected range for the void coefficient is shown in Figure 3-12.

C. Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is opposite in sign and considerably smaller when compared to the moderator temperature coefficient. A typical range of pressure coefficients over a life cycle is -3×10^{-7} to $+3 \times 10^{-6}$ $(\Delta k/k)$/psi.

D. Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors using soluble boron as a reactivity control have a less negative moderator temperature coefficient than do cores controlled solely by movable or fixed CRA. The major temperature effect on the coolant is a change in density. An increasing coolant temperature produces a decrease in water density and an equal percentage reduction in boron concentration. The concentration change results in a positive reactivity component by reducing the absorption in the coolant. The magnitude of this component is proportional to the total reactivity held by soluble boron. Distributed poisons (burnable poison rods or inserted control rods) have a negative effect on the moderator coefficient for a specified boron concentration. That is, the moderator coefficient for a system with 1,200 ppm boron in the coolant and one percent rod worth inserted will be more negative than for a system with 1,200 ppm boron and no rods inserted. Depending on the core size, core loading, and power density, a plant may or may not require additional distributed poisons to yield the appropriate moderator temperature coefficient as determined by the safety analysis and the stability analysis of the core. This is illustrated in Table 3-8.

An examination of the data in Table 3-8 shows that the limiting factor on the moderator coefficient is the value used in the safety analysis in Chapter 14, i.e. $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F_m$. The physics startup program includes measurements of the moderator temperature coefficient. This data will allow an extrapolation to the full power moderator coefficient to assure that the $+0.5 \times 10^{-4}$ α_m will not be exceeded. Previously calculated data will be available during the test period for comparison with the measured values.

Procedures for calculating and physically measuring moderator coefficient at zero power and at power are similar. The former is obtained by raising or lowering the average temperature of the entire core and observing the corresponding change in reactivity. The change in reactivity divided by the change in T_{avg} (core) gives the temperature coefficient.

Measuring the moderator coefficients at power must be done in a less direct manner due to the inability to measure the fuel temperature. The moderator coefficient is measured by raising or lowering T_{avg} (moderator) without changing the power. The reactivity change divided by the change in T_{avg} (moderator) produces the measured moderator coefficient plus two other appreciable effects: (1) The Doppler effect due to increased fuel temperature, and (2) axial expansion of the fuel also due to an increase in fuel temperature.

Correction for the first effect is incorporated in the calculational model. A correction for the second effect was obtained by separate analysis. This produced an axial-expansion reactivity coefficient of -0.02 x 10^{-4} (Δk/k)/°F which can be applied directly to the change in average fuel temperature when calculating moderator coefficients at rated power.

Comparison of moderator temperature coefficients at zero power and low power plateaus and extrapolating to full power ensures that the limiting value of +0.5 x 10^{-4} (Δk/k)/°F_m is not exceeded at rated power. Applying the corrections mentioned above to the calculated moderator coefficients at power and comparing to measured moderator coefficients gives added assurance.

Inaccuracies during zero power tests will result primarily from uncertainties in system conditions and uncertainties in reactivity measurements. Uncertainties in temperature measurement and distribution (at power) will be relatively small. The zero power uncertainty will be approximately +0.1 x 10^{-4} (Δk/k)/°F_m in the measured values.

E. Power Coefficient

The power coefficient, α_p, is the fractional change in neutron multiplication per unit change in core power level. A number of factors contribute to α_p, but only the moderator temperature coefficient and the Doppler coefficient contributions are significant. The power coefficient can be written as:

$$\propto_p = \alpha_m \frac{\Delta Tm}{\Delta P} + \alpha_f \frac{\Delta Tf}{\Delta P}$$

where:

α_m = moderator temperature coefficient

α_f = fuel temperature coefficient

$\frac{\Delta Tm}{\Delta P}, \frac{\Delta Tf}{\Delta P}$ = change in moderator and fuel temperature per unit change in core power

The power coefficient was calculated for Cycle 1 at BOL (time zero) at full power. A boron concentration of ~1,200 ppm was assumed with criticality achieved by control rods. The three-dimensional PDQ07 Code with thermal feedback was used to include the effects of spatially distributed fuel and moderator temperatures. The calculated value is shown in Table 3-8.

F. pH Coefficient

Currently, there is no definite correlation which will permit prediction of pH reactivity effects. Some of the parameters needing correlation are the effects relating pH reactivity change for various operating reactors, pH effects versus reactor operating time at power, and changes in effects with varying clad, temperature, and water chemistry. Yankee, Saxton, and Indian Point Station, Unit 1 have experienced reactivity changes at the time

of pH changes but there is no clear-cut evidence that pH is the direct reactivity influencing variable without considering other items such as clad materials, fuel assembly crud deposition, system average temperature, and prior system water chemistry.

The pH characteristic of this design is shown in Table 3-9 where the cold values are measured and the hot values are calculated. Saxton experiments (see Reference 57) have indicated a pH reactivity effect of 0.0016 (Δk/k)/ΔpH with and without local boiling in the core. Considering the system makeup rate of 35,000 lb/hr and the core in the hot condition with 1,200 ppm boron in the coolant, the corresponding changes in pH are 0.02 pH units per hour for boron dilution and 0.05 pH units per hour for ^{7}Li dilution (starting with 0.5 ppm ^{7}Li). Applying the pH worth value quoted above from Saxton, the total reactivity insertion rate for the hot condition is 3.1 x 10^{-8} (Δk/k)/s. This insertion rate of reactivity can be easily compensated by the operator or the automatic control system.

3.2.2.1.6 Reactivity Insertion Rates

Figure 7-8 displays the integrated rod worth of three overlapping rod banks as a function of distance withdrawn. The indicated groups are those used in the core during power operation. Using approximately 1.2 percent Δk/k CRA group worth and a 30 inche per minute drive speed in conjunction with the reactivity response given in Figure 7-8 a maximum reactivity insertion rate of 1.1 x 10^{-4} (Δk/k)/s is yielded. The maximum reactivity insertion rate for soluble boron removal is 4.4 x 10^{-6} (Δk/k)/s.

3.2.2.1.7 Power Decay Curves

Figure 3-13 displays the BOL power decay curves for the CRA worths corresponding to the one percent hot standby, > 525 °F, shutdown margin with and without a stuck rod. The power decay is initiated by the trip of the CRA with a 300 msec delay from initiation to start of CRA motion. The time required for insertion of a CRA two-thirds of the distance into the core is 1.4 seconds.

3.2.2.2 Nuclear Evaluation

Analytical models and their application are discussed in this section. Core instabilities associated with xenon oscillation are also described.

3.2.2.2.1 Analytical Models

Reactor design calculations are made using a large number of computer codes. The choice of which code set(s) to use is set forth below by design area.

A. Nuclear Calculation Model

Calculation of the reactivity of a pressurized water reactor core is performed in one, two, or three dimensions. The geometry chosen depends on the type of calculations to be made. In a "clean" type of calculation where there are no strong, localized absorbers of a type differing from the rest of the lattice, a one-dimensional analysis is satisfactory. This type of problem is solved by the one-dimensional B&W proprietary depletion package code LIFET which utilizes the MUFT (see Reference 12), WANDA (see Reference 45), and the proprietary B&W RIP, TAME, and AMOP computer programs. A brief description of LIFET follows.

1. Calculation of Epithermal Cross Sections

   The epithermal neutron energy spectrum for each composition region is calculated using the B_1 (or P_1) transport approximation with the Fermi age slowing-down treatment for heavy nuclides and the Grueling-Goertzel (see Reference 35) treatment for the light nuclides. For nuclides exhibiting resonances in the epithermal energy range, the effective resonance integral, corrected for shielding effects and Doppler broadening, is calculated by either the Narrow Resonance or Narrow Resonance-Infinite Absorber Model for each epithermal multigroup. For tightly packed heterogeneous lattices, the Dancoff factor is calculated by the method of Sauer (see References 55 and 56). In computing the epithermal spectrum, both inelastic scattering and (n, 2n) reactions are taken into account. After calculating the neutron spectrum, macrogroup (3 or less groups) diffusion constants are obtained by averaging over the spectrum.

   The resonance integral calculation in the LIFET program is adapted from the RIP program. This program is based upon the work of Dresner (see Reference 27) and Adler, Hinmann, and Nordheim (see Reference 5). Effective resonance integrals are calculated at each resolved and unresolved peak for each resonance nuclide. The effective resonance integral for each energy interval in the multigroup structure is then the sum of the integrals over all peaks in the multigroup. These multigroup resonance integrals are subsequently used in the epithermal spectrum calculation.

2. Calculation of Thermal Cross Sections

   A thermal spectrum calculation is performed for each composition region of a reactor. Since these regions are not strictly homogeneous, a space-energy calculation should be performed to account for the heterogeneous configuration. For the case of uniformly distributed fuel rods in a moderating medium, the space-energy flux may be described in terms of separate space and energy-dependent terms. The AMOP routine in the proprietary LIFET program describes the heterogeneous effects by calculating effective microscopic cross sections for each nuclide. These cross sections reflect the geometrical configuration of the fuel rods, clad, and moderator. The TAME routine then calculates the energy dependence of the flux for an equivalent homogeneous region. Finally, the cross sections are averaged over the energy spectrum to obtain a single thermal cross section set for use in the multigroup diffusion calculation.

B. Reactivity Analysis of Critical Experiments

Thirty-one UO_2 and PuO_2-UO_2 critical assemblies were analyzed in order to verify the methods previously described. This verification program was undertaken for two purposes. The first objective was to determine the validity of these methods for predicting k_{eff} of critical assemblies by means of a one-dimensional model. The second purpose was to investigate the applicability of the present thermal and epithermal spectrum calculations for determining effective cross sections to be used in the analysis of large power reactors. The criticality portion of the LIFET program was used to determine k_{eff} for the critical assemblies. This one-dimensional (cylindrical) analysis was performed with 4-group cross sections; one thermal group and three epithermal groups. The group structure which was used for the group collapse is given below.

Group	Energy Range
1	9.12 keV - 10 MeV
2	61.4 eV - 9.12 keV
3	1.85 eV - 61.4 eV
4 (thermal)	0.0 eV - 1.85 eV

The 17 UO_2 criticals selected for analysis are described in References 15, 26, 31, 34, and 64. K_{eff} was calculated for each assembly assuming the "resonance" treatment for resonance absorption and using the 4-group structure. These values are referred to as the reference values. Statistically, k^r_{eff} = 0.9983 ± 0.0047, where 0.9983 is the arithmetic mean and the error corresponds to one standard deviation of a sample about the mean. All values of k^r_{eff} are between 0.9912 and 1.0068.

A comparison was also made between k^r_{eff} and k^r_{eff} calculated using a 2-group structure. Statistically, $k^{(2\text{-group})}$ = 1.0032 ± 0.0080 for the UO_2 criticals.

For all UO_2 criticals, the difference in k_{eff} between the 2-group and 4-group analysis was +0.0049 ± 0.0040 with the 2-group being higher. The important feature is that the difference decreases rapidly as the assembly radius increases. Hence, for a PWR with a radius of about 150 cm, the difference between the 2- and 4-group treatments would diminish to less than 0.05 percent. This fact justifies the use of a 2-group structure for large power reactors. The increased accuracy of the 4-group treatment appears in the calculation of the leakage, which is important only in small assemblies.

Eleven PuO_2-UO_2 and three UO_2 critical assemblies were studied. These UO_2 assemblies were considered because their lattice parameters were similar to the PuO_2-UO_2 assemblies and contained a larger ^{235}U weight percent than the other UO_2 criticals. These criticals are described in detail in References 41 and 59. A criticality calculation was performed on each PuO_2-UO_2 assembly using the "resonance" treatment for resonance absorption in the epithermal energy range and a 4-group structure for the calculation of k_{eff}. These values are referred to as the reference values. Statistically, k^r_{eff} = 1.001 + 0.0048. All values of k^r_{eff} are between 0.9889 and 1.0073. A second comparison was made between k^r_{eff} and $k_{eff}^{2-\text{group}}$ corresponding to a 2-group structure. $k_{eff}^{2-\text{group}}$ = 1.0135 ± 0.0070 for the PuO_2-UO_2 critical assemblies. For the 14 criticals, the mean difference between the 2-group k_{eff} and 4-group k_{eff} was 0.0134 ± 0.0061.

It is apparent that the difference between the 2- and 4-group structures is essentially the same for the PuO_2-UO_2 and UO_2 criticals. The very small assemblies, e.g. 12.94 cm radius, show errors between the 2-group structures in excess of two percent.

For all UO_2 and PuO_2-UO_2 critical assemblies, k^r_{eff} = 0.991 ± 0.0047. This indicates that the present methods, employing the "resonance" treatment for resonance absorption and the 4-group structure, predict k_{eff} for uniform lattices quite accurately. The one-dimensional model is adequate for assemblies which are nearly cylindrical.

However, for square assemblies, k^{r}_{eff} may differ from the mean value by as much as one percent. A two-dimensional criticality calculation is used for these assemblies. The 2-group treatment yields $k^{(2\text{-group})}$ = 1.0079 ± 00.0074 and $\Delta k_{eff}^{(2-\text{group})}$ = 0.0088 ± 0.0050 for all UO_2 and PuO_2-UO_2 critical assemblies.

For small assemblies, $\Delta k_{eff}^{(2-\text{group})}$ is greater than two percent, indicating the importance of the group structure on the leakage calculation. For large assemblies such as a PWR, the difference is less than 0.05 percent. For uniform lattices of fuel pins in a large reactor, the 2-group cross sections calculated according to the methods described in this section are adequate for design calculations.

C. Power Distribution Analysis of Critical Experiments

A series of detailed three-dimensional power distribution measurements have been performed at the B&W Company Critical Experiments Laboratory. These experiments were designed to provide detailed power distribution measurements with both part and full length control rods, against which analytical methods and results can be checked. A detailed description of the tests and results can be found in Reference 9.

The PDQ07 program was used to calculate the power distributions corresponding to the measured values. Since PDQ07 is a three-dimensional program, the core geometry can be described more exactly where necessary. The composition overlay accurately describes both fuel and poison pins in the control rod regions. Solution points are placed both along the boundary and in the center of the cells in the fuel rod regions. The thermal feedback option in PDQ07 is not exercised because the temperature variations are small. As in the actual measurements, the power densities are normalized to the value at mid-height of the twentieth pin (standard rod) on the west radius (shown in Figure 3-14).

The 2-group coefficients input to PDQ07 for fuel pins, borated water, and aluminum-water reflector are generated by the LIFET program. Homogenized poison pins are treated by using the method described in Section 3.2.2.2.1.H to obtain absorption coefficients. Figure 3-15 shows graphs of the normalized relative power distributions at 35 cm, mid-plane and 105 cm from core bottom for the east traverse from core center. These radial distributions correspond to the case where the control rod clusters are withdrawn 72.5 cm, 92.1 cm, and 72.5 cm for the north, central, and south positions respectively. The axial normalized power distributions at Positions A, B, and C (Figure 3-14) are shown in Figure 3-16.

The radial and axial normalized power distributions calculated with the PDQ07 program agree quite well with the measured distributions. There are, however, deviations near control rod locations.

D. Application of Computer Codes to Nuclear Design

PDQ05 (see Reference 18) and PDQ07, in conjunction with HARMONY, have been used in quarter core X-Y geometry to deplete the core through several cycles. This model was instrumental in choosing core enrichments, core loading schemes and shuffle patterns, and xenon-transient rod banks. EOL mass balances and megawatt days per tonne produced by each assembly were obtained from these cases. The same model was used to calculate rod bank worths, critical boron concentration, and total available rod worth as

well as isothermal moderator coefficients and other undistributed reactivity coefficients. Hot and cold ejected and stuck rod worths were calculated with two-dimensional PDQ, showing the full core.

In a depletion calculation, the time steps used are primarily 50 days in length except for the beginning and end of the fuel cycle. Boron reactivity worth is calculated by observing the k_{eff} associated with two boron concentrations. Since everything else is held constant, the change in reactivity divided by the change in boron concentration gives the boron worth. These worths are computed as a function of burnup and boron concentration.

The three-dimensional PDQ07 code is also an integral part of core design work. B&W has modified this program to include thermal feedback effects. This option is used in almost all three-dimensional calculations. Extensive analyses have been made of:

1. Control rod maneuvering

2. Xenon stability analysis and control

3. Reactivity coefficients

A three-dimensional analysis with feedback must be utilized for the above problems primarily to obtain an accurate description of the radial and axial power distributions. These distributions are necessary in the evaluation of thermal margin.

For reload designs, B&W has added two other nuclear design codes to the licensing analysis procedure:

1. FLAME: This is a nodal code used to calculate core reactivity and two- and three-dimensional power distributions with thermal hydraulic feedback effects. Control rod data are input by node. The code also includes a transient xenon capability. This code is widely used in performing maneuvering analyses. (Reference: BAW-10124 "FLAM3 – Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions")

2. NOODLE: This is a two-energy group neutronics program employing the analytical nodal method to calculate fluxes, power densities, and the core eigenvalue. Both two- and three-dimensional problems may be analyzed with thermal hydraulic feedback and isotopic depletion. This code is widely used in calculating control rod worths, soluble boron worths, reactivity coefficients and power distributions. (Reference: BAW-10152A "NOODLE - Multidimensional Two Group Reactor Simulator")

E. Fuel Cycle Analysis

The X-Y PDQ model with HARMONY was used to calculate fuel cycles for this reactor. The fuel assembly arrangement in the core is shown in Figure 3-60. First cycle and succeeding cycle average burnups are given in Table 3-2.

F. Control of Power Distributions

The core has been designed to allow power maneuvering on control rods. During steady state operation, the axial power shape will be very nearly symmetrical about the mid-plane, since the operator will be periodically adjusting the part-length control rod bank to minimize imbalance. However, during power maneuvering there exists the possibility of tilting the power towards the outlet or the inlet.

The resulting effects on Departure from Nucleate Boiling (DNB) and/or fuel melt design limits have been analyzed for a similar Babcock and Wilcox PWR design to establish the allowable limits on axial imbalance consistent with the thermal design case described in Section 3.2.3.1.1. Imbalance trip setpoints have been established (Figure 7-4) for a reactor protection system function (overpower trip based on flow and imbalance) which will ensure that DNB and fuel temperature limits will not be exceeded.

These trip setpoints will be verified by calculation and startup testing and will provide core protection against excessive axial power imbalance up to and including the design overpower level.

During startup testing, incore instrumentation will be available for confirmation that the out-of-core axial imbalance is as expected. That is, incore instrumentation will be used to establish that the out-of-core trip setpoints are consistent with the actual incore imbalance data.

G. Power Maldistributions

1. Misaligned Control Rods

The reactor has a control function to protect against a rod out of step with its group. The position of each rod is compared to the average of the group. If a fault is detected at power levels above 40 percent of rated power, a rod withdrawal inhibit is activated. If a rod is dropped, the Intergrated Control System (ICS) cannot maintain core power to match demand by withdrawal of other rods and the plant is run back to 40 percent of rated power.

Radial power tilts may be detected with the incore or out-of-core instrumentation. The operator can monitor the upper or the lower set of out-of-core detectors or the incore detectors using the computer to determine the radial power symmetry condition.

The case of one CRA being left out of the core while the remainder of the group is fully inserted could occur only during special configurations of CRAs established during low power physics testing. Since these tests are performed at reduced power, and since the operator can monitor both incore and out-of-core detectors, a radial tilt resulting from such a condition could be detected and appropriate action taken before the approach to thermal limits could be realized.

The APSR drives are also equipped with the position monitors and the alarm function for a rod out of step with the group average. These drives, however, do not permit rod drops. With the power removed from the rod drive windings of the APSR, the roller nut will not disengage and the rod remains in its position (Section 3.2.4.3.2). Since the maneuvering range for these rods will be only three to four feet, it is not likely that thermal limits will be exceeded if one of the rods was stuck and the rest of the group were moved.

Operating procedures provide adequate guidance for determining rod positions and recovery of misaligned control rods (1CAN108308).

2. Azimuthal Xenon Oscillations

This reactor is predicted to have a substantial margin to threshold for azimuthal xenon oscillations. Therefore, this mode is not considered a power peaking problem.

3. Fuel Misloading

Misloading the fuel pins in an assembly is prevented by loading controls and procedures. Each fuel rod is identified by an enrichment code and the design of the reactor is such that only one enrichment is used per assembly. The manufacturing process relies on administrative procedures and quality control checks to assure that fuel rods are placed in the proper assembly. One such administrative procedure which will be practiced to the extent practical is the "campaigning" of enrichments so that only a single enrichment is handled at a given time in fuel fabrication.

Gross fuel assembly misplacement in the core is prevented by administrative core loading procedures and the prominent display of identification markings on the upper end fitting of each assembly. This type of misplacement would be detected with the incore detectors during startup tests. During this phase, the response of the incore detectors will be compared to calculations. Even if an assembly were out of position the operator can monitor the incore detectors to determine if a trend is developing towards a radial power tilt. Upon return to power operation after refueling and, to an even greater extent, initial increase to power, the operator will carefully monitor both incore and out-of-core detectors to assure that core symmetry exists within Technical Specification limits.

H. Control Rod Analysis

B&W has developed a procedure for the analysis of the reactivity worth of small cylindrical Ag-In-Cd control rods. The procedure has been verified against a set of 14 critical experiments in which the variables included: number of rods per cluster, arrangement of rods within the cluster, number of clusters in the core, and soluble boron concentration. Approximately half of the experiments included water holes to simulate withdrawn rods. The comparison of calculated and experimental reactivity worths is shown in Table 3-10.

The experiments were performed at the Babcock & Wilcox Critical Experiment Laboratory with lattices of aluminum-clad uranium oxide fuel rods. Enrichment of the fuel was 2.46 weight percent ^{235}U. The Ag-In-Cd control rods used in the experiments had an absorber diameter of 0.400 inches. Geometrical arrangements of the control rods were chosen to encompass the reference design for the power cores. Experimental rod worths were determined by calibration against soluble boron concentration. The critical soluble poison concentration was determined for each configuration with rods in and again with rods out. (Soluble poison concentrations quoted in the table are for the rods in situation.) Soluble poison concentrations were measured with an absolute accuracy of ± 5 ppm and a precision of ± 3 ppm. References 23 and 24 describe the experiments in detail.

The analytical method used in this analysis is based upon the PDQ code with coefficients generated by the B&W LIFET program. Key features include the allowance for interference and overlap effects between resonances and isotopes in the Ag-In-Cd rod and calculation of the relative fluxes in the control rod and surrounding fuel in an 80-group thermal model.

3.2.2.2.2 Xenon Stability Analysis and Control

Modal and digital analysis of the core indicated that the tendency towards xenon instability in the axial mode could exist for a given combination of events. Therefore, eight part-length Axial Power Shaping Rod Assemblies (APSRAs) have been included in the design. They will be positioned during operation to maintain an acceptable distribution of power for any particular operating condition in the core, thereby reducing the tendency for axial oscillations.

Starting in Cycle 6, and continuing through all succeeding cycles, an operational change was implemented of completely withdrawing the APSRs near the end of cycle. This change was instituted to allow the cycle to operate longer at full power by depleting the reactivity gained by removing the APSRs. For each cycle, the stability and control of the core in the feed and bleed mode with APSRs removed was analyzed. The calculations demonstrated the axial stability of the core in each cycle. The APSR operating limits in the Technical Specifications were changed to show that no APSR insertion was allowed during these times in the applicable reload cycle.

Analysis also indicated that the core is stable azimuthally over the entire fuel cycle.

Modal analysis indicated that the core is stable with regard to radial oscillations.

A detailed description of the xenon analysis performed on the core will be found in Topical Report BAW-10010, "Stability Margin for Xenon Oscillations."

3.2.3 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

This section presents the evaluation of significant core thermal and hydraulic parameters and shows that the basic design of the core satisfies the performance limits and objectives of Section 3.1.2.3. Demonstration of satisfying core thermal and hydraulic limits and objectives for the current fuel cycle is provided in Section 6 of Chapter 3A, Reload Report, and its supporting documentation.

3.2.3.1 Thermal and Hydraulic Characteristics

Sections 3.2.3.1.1 through 3.2.3.2.3 predominantly describe Cycle 1 conditions predicted with the W-3 correlation with 114 percent design overpower except where denoted. Data for Cycle 2 and subsequent cycles containing Mark-B fuel assemblies are based on the B&W-2 correlation and are given in the respective cycle reload reports. Beginning in Cycle 9 and subsequent cycles containing Mark-BZ fuel assemblies, the BWC correlation was used. Beginning in Cycle 20 and subsequent cycles containing Mark-B-HTP fuel assemblies, the BHTP correlation was used. Associated data is provided in each cycle's reload report and its supporting documentation. The current cycle reload report is Chapter 3A.

In subsequent cycles after Cycle 1, new methods and computational tools were introduced that yielded more accurate predictions of thermal-hydraulic conditions in the core. The justifications of the new methodologies and computer codes are contained in the respective topical reports

identified in Section 3.2.3. Results of the application of these new methods and codes in thermal-hydraulic analyses are not explicitly shown in the SAR, however, the respective cycle reload reports acknowledge the impact of the changes. The general trends observed in the Cycle 1 results of Section 3.2.3 are similar to the trends that would be observed using the new methods and computational tools. The Cycle 1 results are comprehensive and are provided for information. With the implementation of new methods and computational tools beyond Cycle 1, appropriate analyses have been performed to justify acceptable thermal-hydraulic performance of the fuel.

The B&W-2, BWC, and BHTP correlations provide a realistic prediction of the burnout phenomenon and predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. In applying the B&W-2, BWC, and BHTP correlation, the following modifications to Cycle 1 DNB data are used:

A. The limiting design DNBR of 1.30 (B&W-2), 1.18 (BWC), or 1.132 (BHTP) is used, corresponding to a 95 percent confidence level of a 95 percent probability that DNB will not occur;

B. The pressure range applicable to the correlation has been extended downward from 2,000 to 1,750 psia for the B&W-2 correlation (BAW-10000A), downward to 1,600 psia for the BWC correlation (BAW-10143A), and downward to 1385 psia for the BHTP correlation (BAW-10241A); and,

C. The design overpower is changed from 114 percent to 112 percent of rated power.

The B&W-2 correlation is fully described in the B&W Topical Report BAW-10000(A), "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water."

The BWC correlation is fully described in the B&W Topical Report BAW-10143P-A, "BWC Correlation of Critical Heat Flux."

The BHTP correlation is fully described in the Framatome-ANP Topical Report BAW-10241P-A, “BHTP DNB Correlation Applied with LYNXT.”

3.2.3.1.1 Fuel Assembly Heat Transfer Design

A. Design Criteria

The criterion for the heat transfer design is to be safely below DNB at the design overpower. The analysis is described in detail in Section 3.2.3.2.2, Statistical Core Design Technique.

The input information for the statistical core design technique and for the evaluation of individual hot channels is as follows:

1. Heat transfer critical heat flux equations and data correlations.

2. Nuclear power factors.

3. Engineering hot channel factors.

4. Core flow distribution hot channel factors.

5. Maximum reactor overpower.

These inputs have been derived from test data, physical measurements, and calculations as outlined below.

B. Heat Transfer Equation and Data Correlation

The heat transfer relationship used to predict limiting heat transfer conditions for Cycle 1 operation is presented in References 60 and 61. The equations are as follows:

1. W-3 uniform flux DNB correlation for single channel with all walls heated:

$$\frac{Q''_{DNB,\,eu}}{10^6} = \{(2.022 - 0.000432P)\}$$

$$+ (0.1722 - 0.0000984\ P)\ \exp\ [(18.177 - 0.004129\ P)\chi]\}$$

$$\bullet\ [(0.1484 - 1.596\ \chi + 0.1729\ \chi\ |\chi|\ \frac{G}{10^6}) + 1.037]$$

$$\bullet\ [1.157 - 0.869\ \chi] \bullet [0.2664 + 0.8357\ \exp\ (-3.151\ D_e)]$$

$$\bullet\ [0.8258 + 0.000794\ (H_{sat} - H_{in})]$$

where:

Q'' = flux, btu/h-ft^2,

P = pressure, psia,

G = mass velocity, lb/h-ft^2,

χ = quality, expressed as fraction,

D_e = equivalent diameter, in.,

H = enthalpy, Btu/lb.

2. W-3 non-uniform flux DNB correlation for single channel with all walls heated:

$$Q''_{DNB,N} = Q''_{DNB,eu}/F$$

where:

$Q''_{DNB,N}$ = DNB heat flux for the non-uniformly heated channel,

$Q''_{DNB,eu}$ = equivalent uniform DNB flux,

$$F = \{\frac{C}{Q''_{\ell DNB}[1 - \exp(-C\ell_{DNB})]}\}$$

$$x\left\{ \int_{0}^{1_{DNB}} Q''(Z) \exp\left[-C(1_{DNB} - Z)\right] dZ \right\}$$

$$C = 0.44 \frac{\left(1-\chi_{DNB}\right)^{7.9}}{\left(\frac{G}{10^6}\right)^{1.72}} in^{-1}$$

1_{DNB} = distance from the inception of local boiling to the point of DNB, in.,

Z = distance from the inception of local boiling measured in the direction of flow, in.

The following new C-factor has recently been reported:(25)

$$C = 0.15 \frac{(1-x)^{4.31}}{\left(\frac{G}{10^6}\right)^{0.478}}$$

A comparison was made between the two C-factors for their effects on the DNB ratio. the results are shown in Figure 3-17 for the design overpower condition. Since the difference is small and since all the DNBR analyses were completed when the new C-factor was published, the DNB ratios reported herein will be based on the original C-factor.

3. W-3 uniform flux DNB correlation for single channel with unheated walls:

$$\frac{Q''_{\text{DNB, with unheated wall}}}{Q''_{\text{DNB, using } D_h \text{ to replace } D_e}} = (1.36 + 0.12e^{9\chi})$$

$$\times (1.2 - 1.6\, e^{-1.92D_h})$$

$$\times (1.33 - 0.237\, e^{5.66\chi})$$

where:

D_e = equivalent diameter based on only the wetted perimeter, in.

D_h = equivalent diameter based on only the heated perimeter, in.

Individual channels are analyzed to determine a DNB ratio, i.e. the ratio of the heat flux at which a DNB is predicted to occur to the heat flux in the channel being investigated. This DNB ratio is related to the data correlation as shown in Figure 3-18. A confidence and population value is associated with every DNB ratio as described in the Statistical Core Design Technique (Section 3.2.3.2.2). The plot of DNB versus population is for a confidence of 99 percent. The criterion for

evaluating the thermal design margin for individual channels or the total core is the confidence-population relationship. The DNB ratios required to meet the basic criteria or limits are a function of the experimental data and heat transfer correlation used and vary with the quantity and quality of data. The recommended minimum design DNB ratio for the W-3 correlation is 1.30.

The DNB and population relationship for a limit of 1.30 in the hot channel corresponds to a 99 percent confidence that at least 94.3 percent of the population of all such hot channels is in no jeopardy of experiencing a DNB. The DNB ratios and the fraction of the core in no jeopardy of experiencing a DNB at design conditions are considerably higher than those given in the design limits outlined in Section 3.1.2.3.

4. The heat transfer relationship used to predict limiting heat transfer conditions for the fuel containing zircaloy intermediate spacer grids, introduced into the ANO-1 core beginning in Cycle 9, is presented in BAW-10143P-A, “BWC Correlation of Critical Heat Flux”. The equation is as follows:

BWC Critical Heat Flux correlation:

$$q''_{BWC} = \frac{A_5\left[\left(A_1 G\right)^{A_3 + A_4 (P-2000)} - A_8 H_{fg} G x\right]}{\left[\left(A_2 G\right)^{A_6 + A_7 (P-2000)}\right] F}$$

where:

$$F = \frac{C q''_{av}}{q''_L \left(1 - e^{-CL}\right)} \int_0^L \left[\phi(Z)\right] e^{-C(L-Z)} dz$$

$$C = \frac{B_1 (1-x)^{B_2}}{\left(G/10^6\right)^{B_3}}$$

q''_{BWC} :	critical heat flux, Btu/hr-ft^2
G:	local mass velocity, lbm/hr-ft^2
P:	pressure, psia
x :	local quality
H_{fg} :	latent heat of vaporization, Btu/lbm
D_e :	equivalent diameter, in
L :	axial location of CHF, in
Z :	axial location, in

q''_{av}: rod average heat flux, Btu/hr-ft^2

$\phi(z)$: local average heat flux ratio

A_1 through A_8 and B_1 through B_3 are optimized coefficients as described in BAW-10143P-A.

5. The heat transfer relationship used to predict limiting heat transfer conditions for the Mark-B-HTP fuel, introduced into the ANO-1 core beginning in Cycle 20, is presented in BAW-10241P-A, "BHTP DNB Correlation Applied with LYNXT."

C. Nuclear Power Factors

The heated surfaces in every flow channel in the core are examined for heat flux limits. The heat input to the fuel rods in a coolant channel is determined from a nuclear analysis of the core and fuel assemblies. The results of this analysis are presented in Table 3-11.

The axial nuclear factor, Fz, is illustrated in Figure 3-19. The distribution of power expressed as P / P is shown for two conditions of reactor operation. The first condition is an inlet peak resulting from partial insertion of a CRA group for transient control following a power level change. This condition results in the maximum local heat flux and maximum linear heat rate. The second power shape is a symmetrical cosine which is indicative of the power distribution with xenon override rods withdrawn. Both of these flux shapes have been evaluated for thermal DNB limitations. The limiting condition is the cosine power distribution. The inlet peak shape has a larger maximum value. However, the position of the cosine peak farther up the channel results in a less favorable flux to enthalpy relationship.

This effect has been demonstrated in DNB tests of nonuniform flux shapes. The cosine axial shape has been used to determine individual channel DNB limits and to make the associated statistical analysis. The fuel rod peaking factor $F\Delta h$ is calculated for each rod in the core. A distribution curve of the fraction of the core fuel rods operating above various peaking factors is shown in Figure 3-20 for a typical fuel cycle condition with the fuel rod maximum design peaking factor.

Additional axial power distributions have been analyzed to determine the allowable power in the upper half of the channel. Figure 3-21 shows the position versus allowable peak for DNBR conditions equivalent to symmetrical cosine distribution. The maximum design radial-local of 1.78 has been used in Figure 3-21. Figure 3-22 presents the allowable conditions when the radial-local power factor is 1.65 instead of 1.78. A comparison of Figures 3-21 and 3-22 shows the additional allowable outlet peak for less than maximum design radial-local.

D. Engineering Hot Channel Factors

Power peaking factors obtained from the nuclear analysis are based on mechanically perfect fuel assemblies. Engineering hot channel factors are used to describe variations in fuel loading, fuel and clad dimensions, and flow channel geometry from perfect physical quantities and dimensions.

The application of hot channel factors is described in detail in Section 3.2.3.2.2, Statistical Core Design Technique. The factors are determined statistically from fuel assembly as-built or specified data where F_Q is a heat input factor, F_Q" is a local heat flux factor at a hot spot, and F_A is a flow area reduction factor describing the variation in coolant channel flow area. Several subfactors are combined statistically to obtain the final values for F_Q, F_Q", and F_A. These subfactors are shown in Table 3-12. The factor, the coefficient of variation, the standard deviation, and the mean value are tabulated.

Rod bowing and the application of rod bow penalty factors are described in Section 3.2.3.2.3, Part I.

E. Core Flow Distribution Hot Channel Factors

The physical arrangement of the reactor vessel internals and nozzles results in a non-uniform distribution of coolant flow to the various fuel assemblies. Reactor internal structures above and below the active core are designed to minimize unfavorable flow distribution. A 1/6-scale model test of the reactor and internals was performed to demonstrate the adequacy of the internal arrangements. The results of the test have confirmed the adequacy of the design values used. Model test results are given in Topical Report BAW-10037, Revision 2, "Reactor Vessel Model Flow Tests."

A flow distribution factor is determined for each fuel assembly location in the core. The factor is expressed as the ratio of fuel assembly flow to average fuel assembly flow. The finite values of the ratio are greater or less than 1.0 depending on the position of the assembly being evaluated. The flow in the central fuel assemblies is in general larger than the flow in the outermost assemblies due to the inherent flow characteristics of the reactor vessel.

The flow distribution factor is related to a particular fuel assembly location and the quantity of heat being produced in the assembly. A flow-to-power comparison is made for all of the fuel assemblies. The worst condition in the hottest fuel assembly is determined by applying model test isothermal flow distribution data and heat input effects at power as outlined in Section 3.2.3.2.3. Two assumptions for flow distribution have been made in the thermal analysis of the core as follows:

1. For the maximum design condition and for the analysis of the hottest channel, all fuel assemblies receive minimum flow for the worst power condition. The hottest assembly receives 95 percent of the average channel flow at isothermal conditions. This is supported by B&W Topical Report BAW-10037, Rev. 2.

2. For the most probable design conditions predicted, average flows have been assigned for each fuel assembly consistent with location and power.

F. Maximum Reactor Design Overpower

Core performance is assessed at the maximum design overpower. The selection of the design overpower is based on an analysis of the reactor protection system as described in Chapter 7. The maximum overpower will not be exceeded under any conditions.

G. Maximum Design Conditions Analysis Summary

The Statistical Core Design Technique described in Section 3.2.3.2 was used to analyze the reactor at the maximum design conditions described previously. The total number of fuel rods in the core that have a possibility of reaching DNB is shown in Figure 3-23 for a range of overpowers for the maximum design conditions. Point B on Line 1 is the maximum design point with the design $F\Delta h$ nuclear and minimum flow to every channel in the core. This Point B forms the basis for this statistical statement:

There is a 99 percent confidence that at least 99.96 percent of the fuel rods in the core are in no jeopardy of experiencing a Departure from Nucleate Boiling (DNB) during continuous operation at the design overpower.

At 100 percent power as shown by Point A, the statistical number of fuel rods in jeopardy is less than one, resulting in a fuel rod population protected in excess of 99.997 percent. The limit imposed by a W-3 DNB ratio of 1.3 would result in 70 fuel rods in jeopardy or a fuel rod population protected of 99.810 percent.

An additional analysis of fuel rods in jeopardy for all the maximum design conditions except fuel assembly flow distribution is shown as Line 2 in Figure 3-23. Each assembly was assumed to receive average flow for the assembly power conditions. The number of rods that have a possibility of a DNB are within the bounds of the two lines. Statistical results for the maximum design condition calculation, shown by Figure 3-23, are summarized in Table 3-13.

H. Most Probable Design Condition Analysis Summary

The previous maximum design calculation, as shown by Line 1 in Figure 3-23, indicates the total number of rods that may be in jeopardy when it is conservatively assumed that every rod in the core has the mechanical and heat transfer characteristics of a hot channel as described in Section 3.2.3.2.2. For example, all channels are analyzed with F_A (flow area factor) less than 1.0, F_Q (heat input factor) greater than 1.0, and with minimum fuel assembly flow. It is physically impossible for all channels to have hot channel characteristics. A more realistic indication of the number of fuel rods in jeopardy may be obtained by the application of the statistical heat transfer data to average rod power and mechanical conditions.

An analysis for the most probable conditions has been made based on the average conditions described in Section 3.2.3.2.2. The results of this analysis are shown in Figure 3-24. The statistical results are also tabulated in Table 3-13.

The analysis was made from Point D at 100 percent power to Point F at 122.5 percent power to show the sensitivity of the analysis with power. The worst condition expected is indicated by Point E at 114 percent power where it is shown statistically that there is a small possibility that 2.5 fuel rods may be subject to a DNB. This result forms the basis for the following statistical statement for the most probable design conditions:

There is at least a 99 percent confidence that at least 99.993 percent of the rods in the core are in no jeopardy of experiencing a DNB even with continuous operation at the design overpower.

I. Distribution of the Fraction of Fuel Rods Protected

The distribution of the fraction (P) of fuel rods that have been shown statistically to be in no jeopardy of a DNB has been calculated for the maximum design and most probable design conditions. The computer programs used provide an output of (N) number of rods and (P) fraction of rods that will not experience a DNB grouped for ranges of (P). The results for the most probable design condition are shown in Figure 3-25.

The population protected, (P), and the population in jeopardy, (1-P), are both plotted. The integral of (1-P) and the number of fuel rods gives the number of rods that are in jeopardy for given conditions as shown in Figures 3-23 and 3-24. The number of rods is obtained from the product of the percentage times the total number of rods being considered (36,816). Two typical distributions shown in Figure 3-25 are for the most probable condition analysis of Points E and F on Figure 3-24. The lower line of Figure 3-25 shows (P) and (1-P) at the 114 percent power condition represented by Point E of Figure 3-24.

The upper curve shows (P) and (1-P) at the 122.5 percent power condition represented by Point F of Figure 3-24. The integral of N and (1-P) of the lower curve forms the basis for the statistical statement at the most probable design condition described in Item "H" above.

J. Hot Channel Performance for 4-Pump Operation

The hottest unit cell with all surfaces heated has been examined for hot channel factors, DNB ratios, and quality for a range of reactor powers. The cell has been examined for the maximum value of $F\Delta h$ nuclear. The hot channel was assumed to be located in a fuel assembly with minimum flow, i.e. 95 percent of average fuel assembly flow. The heat generated in the fuel is 97.3 percent of the total nuclear heat. The remaining 2.7 percent is assumed to be generated in the coolant as it proceeds up the channel within the core and is reflected as an increase in ΔT of the coolant.

Operating pressure and inlet temperature error bands (± 65 psi, ± 2 °F) are reflected in the total core and hot channel thermal margin calculations in the direction producing the lowest DNB ratios or highest qualities. The engineering hot channel factors used in the design analysis are described in Section 3.2.3.2.2. The DNB ratio versus power is shown in Figure 3-26. The hot channel exit quality for various powers is shown in Figure 3-27. The engineering hot channel factors and summary results are listed in Table 3-14.

K. Hot Channel Performance Summary for Partial Pump Flow

The power limitations imposed on the reactor due to the loss of one or more pumps has been examined by studying DNB ratios and quality in the hot channel for a range of reactor powers. The system parameters used in the analysis are the same as the ones discussed in Section 3.2.3.1.1.J. A constant reactor vessel average temperature of 578.9 °F was used to determine inlet temperature. The DNB ratio versus power for the flow conditions caused by loss of pumps is illustrated in Figure 3-28. The hot channel quality at the minimum DNB ratio point (X_{DNB}) at the same conditions is shown in Figure 3-29.

Two limits have been placed on the analysis. One is the recommended minimum DNB ratio of 1.3 and the other is a quality limit of ± 15 percent at the point of the minimum DNB ratio. This limit has been used since the limits of the W-3 correlation are ± 15 percent quality at the DNB location. Table 3-15 summarizes the power limitations on reactor operation at 2,120 psig as defined by the hot channel conditions.

3.2.3.1.2 Fuel and Cladding Thermal Conditions

Fuel and cladding thermal conditions are determined for fuel rods pre-pressurized with helium.

A. Fuel

A digital computer code is used to calculate the fuel temperature. The program uses non-uniform volumetric heat generation across the fuel diameter and external coolant conditions and heat transfer coefficients determined for thermal-hydraulic solutions. The fuel conductivity is varied with fuel density and as a function of temperature. Conductivity values for 95 percent dense UO_2 are shown in Figure 3-30, a plot of fuel conductivity versus temperature.

The heat transfer from the fuel to the clad is calculated with a fuel and clad expansion model proportional to temperature and burnup. The temperature at BOL is calculated using a gas conductivity of 0.14 Btu-ft/h-°F-ft^2. The temperature drop during fuel life will change with variations in gas composition and gas temperature. The gas conduction model is used in the calculation until the fuel expansion relative to the clad closes the gap to a dimension equivalent to a contact coefficient. The contact coefficient is dependent upon interface pressure and gas conductivity.

A plot of fuel center temperature versus linear heat rate in kW/ft is shown in Figure 3-31 for BOL conditions for the maximum design cold diametral clearance. The supporting parameters and design conditions are shown in Table 3-16.

B. Clad

The assumptions in the preceding paragraph were applied in the calculation of the clad surface temperature at the maximum overpower.

Boiling conditions prevail at the hot spot. The Jens and Lottes relationship (see Reference 38) for the coolant-to-clad ΔT for boiling was used to determine the clad temperature. The resulting maximum calculated clad surface temperature is 654 °F at a core operating pressure of 2,185 psig.

3.2.3.1.3 End of Life Clad Transients

An investigation was carried out analyzing the ability of the cladding to withstand various EOL transients which, though not considered normal, could occur during the life of the plant. The specific transients examined were loss of flow at 108 percent power and power excursions up to 114 percent. The latter value is the maximum power attainable with an upper trip point setting of 107.5 percent. The effects of internal cladding pressure and system pressure on the integrity of the cladding during the normal shutdown for refueling and due to depressurization transients were also examined.

For the flow coastdown analysis, temperatures for the fuel and cladding during a coastdown from 108 percent power were obtained. Even starting from this overpower condition, there was no rise in temperature in either the fuel or cladding following the loss of pumping power. The fact that the pumps have been designed to include a rotational inertia equivalent to at least 73,500 lb-ft^2 (Allis Chalmers, A-C) and 68,605 lb-ft^2 (Jeumont Industrie, J-I) allows them to provide sufficient flow after the loss of power to avoid temperature increases and to maintain the DNB ratio for the hot channel at a value higher than the DNB ratio for continuous operation at the maximum overpower condition.

A power excursion transient to 114 percent was also considered. Since the DNB analysis has been done for steady state operation at this power level, DNB was not a consideration. For this transient it was expected that a greater release rate of fission gases, and consequently a greater internal pressure and stress of the cladding during the excursion would occur. The analysis of the internal pressure buildup and stress and strain in the cladding at 114 percent overpower was carried out with conservative assumptions. At the end of life, the calculated internal gas pressure is considerably below the design internal pressure. The tensile stress due to the maximum calculated fission gas pressure is less than 10 percent of yield strength.

During a normal reduction in power, or a cooldown for refueling, the internal gas temperature and pressure decreases. An investigation of depressurization transients indicates that, when coolant pressure is conservatively assumed to drop instantaneously to zero, the cladding stress due to internal pressure is less than yield strength. The design internal pressure in a hot fuel rod is 3,300 psi, (maximum calculated fission gas pressure is considerably less than this even at design power in a high burnup rod as described in Section 3.2.3.2.3.H). The internal pressure required to cause clad stress equal to the yield strength at the hot spot in the core at maximum overpower conditions (maximum average clad temperature 711 °F) is 4,500 psi. It is concluded that any conceivable depressurization transient cannot damage the clad.

Following the first core of ANO-1, reload fuel was evaluated to different stress and strain criteria where the fuel pin cladding is separately evaluated for pressure induced stresses and for pellet induced cladding strains. No specific overpower transient is evaluated for the effect on fuel pin cladding. The criteria used for cladding stresses is that the worst case pressure induced stresses must not exceed 2/3 of the minimum unirradiated yield stress of the cladding. The method of calculating cladding transient strain is specific to the fuel performance code used to model the fuel. Currently, the code TACO3 is used to set fuel rod local power (LHGR) limits such that neither the fuel center line temperature reaches the melting point nor the pellet induced transient strains (plastic plus elastic) exceed 1%. These limits insure that the cladding strain remains well within the ductile range of irradiated cladding and that cladding will not fail. These stress and strain limits were shown adequate to meet the guidelines of NUREG 0800 for reload fuel evaluation in BAW 10179PA.

3.2.3.2 Thermal and Hydraulic Evaluation

3.2.3.2.1 Introduction

Summary results for the characteristics of the reactor design are presented in Section 3.2.3.1. The Statistical Core Design Technique employed in the design represents a refinement in the methods for evaluating pressurized water reactors. Corresponding single hot channel DNB data were presented to relate the new method with previous criteria. A comprehensive description of the new technique is included in this section to permit a rapid evaluation of the methods used.

A detailed evaluation and sensitivity analysis of the design has been made by examining the hottest channel in the reactor for DNB ratio, quality, and fuel temperature. The W-3 correlation for Cycle 1 operation has been used in this analysis.

3.2.3.2.2 Statistical Core Design Technique

The core thermal design is based on a Statistical Core Design Technique developed by B&W. The technique offers many substantial improvements over older methods, particularly in design approach, reliability of the result, and mathematical treatment of the calculation. The method

reflects the performance of the entire core in the resultant power rating and provides insight into the reliability of the calculation. This section discusses the technique in order to provide an understanding of its engineering merit.

The statistical core design technique considers all parameters that affect the safe and reliable operation of the reactor core. By considering each fuel rod, the method rates the reactor on the basis of the performance of the entire core. The result then will provide a good measure of the core safety and reliability since the method provides a statistical statement for the total core. This statement also reflects the conservatism or design margin in the calculation.

A reactor safe operating power has been determined by the ability of the coolant to remove heat from the fuel material. The criterion that best measures this ability is the DNB, which involves the individual parameters of heat flux, coolant temperature rise, and flow area, and their intereffects. The DNB criterion is commonly applied through the use of the Departure from Nucleate Boiling Ratio (DNBR). This is the minimum ratio of the DNB heat flux (as computed by the DNB correlation) to the surface heat flux. The ratio is a measure of the margin between the operating power and the power at which a DNB might be expected to occur in that channel. The DNBR varies over the channel length, and it is the minimum value of the ratio in the channel of interest that is used.

The calculation of DNB heat flux involves the coolant enthalpy rise and coolant flow rate. The coolant enthalpy rise is a function of both the heat input and the flow rate. It is possible to separate these two effects; the statistical hot channel factors required are a heat input factor, F_Q, and a flow area factor, F_A. In addition, a statistical heat flux factor, $F"_Q$ is required; the heat flux factor statistically describes the variation in surface heat flux. The DNBR is more limiting when the burnout heat flux is based on minimum flow area (small F_A) and maximum heat input (large F_Q), and when the surface heat flux is large (large $F"_Q$). The DNB correlation is provided in a best-fit form, i.e. a form that best fits all of the data on which the correlation is based. To afford protection against DNB, the DNB heat flux computed by the best-fit correlation is divided by a DNB factor (BF) greater than 1.0 to yield the design DNB surface heat flux. The basic relationship involves as parameters

$$DNBR = \frac{Q"_{DNB}}{BF} \text{ x } f(F_A, F_Q) \text{ x } \frac{1}{Q"_{surface} \text{ x} F_{"Q"}}$$

statistical hot channel and DNB factors. The DNB factor (BF) above is usually assigned a value of unity when reporting DNB ratios so that the margin at a given condition is shown directly by a DNBR greater than 1.0, i.e. 1.30, in the hot channel.

Selected heat transfer data are analyzed to obtain a correlation. Since thermal and hydraulic data generally are well represented with a Gaussian (normal) distribution (Figure 3-32), mathematical parameters that quantitatively rate the correlation can be easily obtained for the histogram. These same mathematical parameters are the basis for the statistical factors.

In analyzing a reactor core, the statistical information required to describe the hot channel subfactors may be obtained from data on the as-built core, from data on similar cores that have been constructed, or from the specified tolerances for the proposed core. The design factors can be shown graphically (Figures 3-33 and 3-34).

All the plots have the same characteristic shape whether they are for subfactors, hot channel factors, or a burnout factor. The factor increases with either increasing population or confidence. The value used for the statistical hot channel and burnout factor is a function of the percentage of confidence desired in the result and the portion of all possibilities desired as well as the amount of data used in determining the statistical factor. A frequently used assumption in statistical analyses is that the data available represent an infinite sample of that data. The implications of this assumption should be noted. For instance, if limited data are available, such an assumption leads to a somewhat optimistic result. The assumption also implies that more information exists for a given sample than is indicated by the data; it implies 100 percent confidence in the end result. The B&W calculational procedure does not make this assumption, but rather uses the specified sample size to yield a result that is much more meaningful and statistically rigorous. The influence of the amount of data for instance can be illustrated easily as follows. Consider the heat flux factor which has the form

$$F_{Q''} = 1 + K\sigma_{FQ''}$$

where:

$F_{Q''}$ = the statistical hot channel factor for heat flux,

K = a statistical multiplying factor, and,

$\sigma_{FQ''}$ = the standard deviation of the heat flux factor including the effects of all the subfactors.

If $\sigma_{FQ''}$ = 0.05 for 300 data points, then a K factor of 2.608 is required to protect 99 percent of the population. The value of the hot channel factor then is and will provide 99 percent confidence for the calculation. If,

$$F_{Q''} = 1 + (2.608 \times 0.050) = 1.1304$$

instead of using the 300 data points, it is assumed that the data represent an infinite sample, then the K factor for 99 percent of the population is 2.326. The value of the hot channel factor in this case is

$$F_{Q''} = 1 + (2.326 \times 0.050) = 1.1163$$

which implies 100 percent confidence in the calculation. The values of the K factor used above are taken from SCR-607 (See Reference 50). The same basic techniques can be used to handle any situation involving variable confidence, population, and number of points.

Having established statistical hot channel factors and statistical DNB factors, we can proceed with the calculation in the classical manner. The statistical factors are used to determine the minimum fraction of rods protected or that are in no jeopardy of experiencing a DNB at each nuclear power peaking factor. Since this fraction is known, the maximum fraction in jeopardy is also known. It should be recognized that every rod in the core has an associated DNB ratio that is substantially greater than 1.0, even at the design overpower, and that theoretically no rod can have a statistical population factor of 100 percent, no matter how large its DNB ratio.

Since both the fraction of rods in jeopardy at any particular nuclear power peaking factor and the number of rods operating at that peaking factor are known, the total number of rods in jeopardy in the whole core can be obtained by simple summation. The calculation is made as a function of power and the plot of rods in jeopardy versus reactor overpower is obtained (Figure 3-23). The

summation of the fraction of rods in jeopardy at each peaking factor summed over all peaking factors can be made in a statistically rigorous manner only if the confidence for all populations is identical. If an infinite sample is not assumed the confidence varies with population. To form this summation then, a conservative assumption is required. B&W total core model assumes that the confidence for all rods is equal to that for the least-protected rod, i.e. the minimum possible confidence factor is associated with the entire calculation.

The result of the foregoing technique, based on the <u>maximum design</u> conditions (114 percent power for Cycle 1), is this statistical statement:

> There is at least a 99 percent confidence that at least 99.96 percent of the rods in the core are in no jeopardy of experiencing a DNB, even with continuous operation at the design overpower.

The maximum design conditions are represented by these assumptions:

A. The maximum design values of $F\Delta h$ (nuclear max/avg total fuel rod heat input) are obtained by examining the maximum, nominal, and minimum fuel assembly spacing and determining the worst value for rod peaking.

B. The maximum value of F_Z (nuclear max/avg axial fuel rod heat input) is determined for the limiting transient or steady state condition.

C. Every coolant channel in the core is assigned to have less than the nominal flow area represented by engineering hot channel factors, F_A, less than 1.0.

D. Every channel is assumed to have the minimum pressure drop associated with the core flow condition.

E. Every fuel rod in the core is assumed to have a heat input greater than the maximum calculated value. This value is represented by engineering hot channel heat input factors, F_Q and $F_{Q''}$, which are greater than 1.0.

The statistical core design technique may also be used in a similar manner to evaluate the entire core at the <u>most probable</u> mechanical and nuclear conditions to give an indication of the most probable degree of fuel element jeopardy. The result of the technique based on the <u>most probable design</u> conditions leads to a statistical statement which is a corollary to the maximum design statement:

> There is at least a 99 percent confidence that at least 99.993 percent of the rods in the core are in no jeopardy of experiencing a DNB, even with continuous operation at the design overpower.

The most probable design conditions are assumed to be the same as the maximum design conditions with these exceptions:

A. Every coolant channel is assumed to have the nominal flow area (F_A = 1.0).

B. Every fuel rod is assumed to have (1) the maximum calculated value of heat input, and (2) F_Q and $F_{Q''}$ are assigned values of 1.0.

C. The flow in each coolant channel is based on a power analysis without flow maldistribution factors.

D. Every fuel rod is assumed to have a nominal value for FΔh nuclear.

The full meaning of the maximum and most probable design statements requires additional comment. As to the 0.04 percent or 0.007 percent of the rods not included in the statements, statistically it can be said that no more than 0.04 percent or 0.007 percent of the rods will be in jeopardy and that in general the number in jeopardy will be fewer than 0.04 percent or 0.007 percent. The statements do not mean to specify a given number of DNBs but only acknowledge the possibility that a given number could occur for the 114 percent overpower conditions assumed. Analyses for 100 percent rated power conditions reported in Table 3-13 show that essentially none of the fuel rods are subject to a DNB.

In summary, the calculational procedure outlined here represents a substantially improved design technique in two ways:

A. It reflects the performance and safety of the entire core in the resultant power rating by considering the effect of each rod on the power rating.

B. It provides information on the reliability of the calculation and, therefore, the core through the statistical statement.

3.2.3.2.3 Evaluation of the Thermal and Hydraulic Design

A. Hot Channel Coolant Quality and Void Fraction

An evaluation of the hot channel coolant conditions provides additional confidence in the thermal design. Sufficient coolant flow has been provided to ensure low quality and void fractions. The quality in the hot channel versus reactor power is shown in Figure 3-27. The sensitivity of channel outlet quality with pressure and power level is shown by the 2,185 and 2,120 psig system pressure conditions examined. These calculations were made for the maximum design value of FΔh. Additional calculations for a 10 percent increase in FΔh were made at maximum overpower. The significant results of both calculations are summarized in Table 3-17.

The conditions of Table 3-17 were determined with all of the hot channel factors applied. Additional calculations were made for unit cell channels without engineering hot channel factors to show the coolant conditions more likely to occur in the reactor core. A nominal value for FΔh was examined with and without fuel assembly flow distribution hot channel factors at 2,185 psig as shown on Figure 3-35. These results show that the exit qualities from the hottest cells are lower than the maximum design conditions.

B. Core Void Fraction

The core void fractions were calculated at 100 percent rated power for the normal operating pressure of 2,185 psig and for the minimum operating pressure of 2,120 psig. The influence of core fuel assembly flow distribution was checked by determining the total voids for both 100 and 95 percent total core flow for the two pressure conditions. The results are presented in Table 3-17.

The most conservative condition of 95 percent flow at 2,120 psig results in an acceptable void volume in the core. Conservative maximum design values were used to make the calculation.

The void program uses a combination of Bowring's (see Reference 13) model with Zuber's (see Reference 63) correlation between void fraction and quality. The Bowring model considers three different regions of forced convection boiling. They are:

1. Highly Subcooled Boiling

   In this region, the bubbles adhere to the wall while moving upward through the channel. This region is terminated when the subcooling decreases to a point where the bubbles break through the laminar sublayer and depart from the surface. The highly subcooled region starts when the surface temperature of the clad reaches the surface temperature predicted by the Jens and Lottes equation. The highly subcooled region ends when

$$T_{sat} - T_{bulk} = \frac{\eta\phi}{V} \qquad \text{(A)}$$

where:

ϕ = local heat flux, Btu/h-ft^2

η = 1.863×10^{-5} (14 + 0.0068p),

V = velocity of coolant, ft/s,

P = pressure, psia.

The void fraction in this region is computed in the same manner as Maurer, (see Reference 46) except that the end of the region is determined by Equation (A) rather than by a vapor layer thickness. The non-equilibrium quality at the end of the region is computed from the void fraction as follows:

$$x^*_d = \frac{1}{1+\frac{\rho_f}{\rho_g}\left(\frac{1}{a_d}-1\right)} \qquad \text{(B)}$$

where:

x^*_d = non-equilibrium quality at end of Region 1,

a_d = void fraction at $T_{sat} - T_{bulk} = \frac{\eta\phi}{V}$,

ρ_f = liquid component density, lb/ft^3,

ρ_g = vapor component density, lb/ft^3.

2. Slightly Subcooled Boiling

In this region, the bubbles depart from the wall and are transported along the channel (condensation of the bubbles is neglected). This region extends to a point where the thermodynamic quality is equal to the apparent quality. In general, this is the region of major concern in the design of pressurized water reactors.

The non-equilibrium quality in this region is computed from the following formula:

$$x^* = x_d^* + \frac{P_h}{\dot{m}\, h_{fg}(1+\varepsilon)\int_{zd}^{z}(\theta - \theta_{SP})dz} \tag{C}$$

where:

x^* = nonequilibrium quality in Region 2,

h_{fg} = latent heat of vaporization, Btu/lb,

$\frac{1}{1+\varepsilon}$ = fraction of the heat flux above the single phase heat flux that actually goes to producing voids,

θ_{SP} = single phase heat flux, Btu/h-ft^2

$\dot{m}$ = mass flow rate, lb/hr,

P_h = heated perimeter, ft,

Z = channel distance, ft.

The void fraction in this region is computed from the following formula:

$$a = \frac{x^*}{C_o\left[x^* + \frac{\rho_g}{\rho_f}(1 - x^*)\right] + \frac{38.3 A_f \rho_g}{\dot{m}}\left[\frac{\sigma g g_c(\rho_f - \rho_g)}{\rho_f^{\,2}}\right]^{0.25}} \tag{D}$$

where:

g = acceleration due to gravity, ft/s^2,

g_c = constant in Newton's Second Law = $32.17\frac{lb_m\ ft}{lb_f\ s^2}$

C_o = Zuber's distribution parameter,

A_f = flow area, in^2,

σ = surface tension,

a = void fraction.

Equation (D) results from rearranging equations found in Reference 11 and assuming bubbly turbulent flow in determining the relative velocity between the vapor and the fluid. Zuber has shown that Equation (D) results in a better prediction of the void fraction than earlier models based on empirical slip ratios.

3. Bulk Boiling

In this region, the bulk temperature is equal to the saturation temperature and all the energy transferred to the fluid results in net vapor generation. Bulk boiling begins when the thermodynamic (heat balance) quality, x, is greater than the nonequilibrium quality, x*. The void fraction in this region is computed using Equation (D) with the thermodynamic quality, x, replacing x*.

C. Coolant Channel Hydraulic Stability

Flow regime maps of mass flow rate and quality were constructed in order to evaluate channel hydraulic stability. The confidence in the design is based on a review of both analytical evaluations (see References 11 & 36) and experimental results obtained in multiple rod bundle burnout tests. Bubble-to-annular and bubble-to-slug flow limits proposed by Baker (see Reference 8) are consistent with the B&W experimental data in the range of interest. The analytical limits and experimental data points have been plotted to obtain the maps for the four different types of cells in the reactor core. These are shown in Figures 3-36, 3-37, 3-38, and 3-39. The experimental data points represent the exit conditions in the various types of channels just previous to the burnout condition for a representative sample of the data points obtained at design operating conditions in the nine rod burnout test assemblies. In all of the bundle tests, the pressure drop, flow rate, and rod temperature traces were repeatable and steady and did not exhibit any of the characteristics associated with flow instability.

Values of hot channel mass velocity and quality at 114 and 130 percent power for both maximum design and most probable conditions are shown on the maps. The potential operating points are within the bounds suggested by Baker. Experimental data points for the reactor geometry with much higher qualities than the operating conditions have not exhibited unstable characteristics.

D. Hot Channel DNB Comparisons

DNB ratios for the hottest channel have been determined for the W-3 correlation, and the results are shown in Figure 3-26. DNB ratios are shown for the design 1.50 axial max/avg symmetrical cosine flux shape from 100 to 140 percent power. The W-3 DNB ratio at the maximum design power of 114 percent is 1.55. This compares with the suggested W-3 design value of 1.3. A ratio of 1.3 is reached at 122.5 percent power at an exit quality of 9.2 percent which is within the prescribed quality limits of the correlation.

The sensitivity of the DNB ratio with F_z nuclear was examined from 100 to 140 percent power. The detailed results are labeled in Figure 3-40. A cosine flux shape with an F_z of 1.80 and an $F\Delta h$ of 1.78 results in a W-3 DNB ratio of 1.30 at 114 percent power. Similar results are shown for a value of F_z of 1.65 and for the design value of 1.5.

The influence of a change in $F\Delta h$ was determined by analyzing the hot channel for an $F\Delta h$ of 1.96. This value is 10 percent above the maximum design value of 1.78. The resulting W-3 DNB ratio is 1.23 at 114 percent power. This value is well above the correlation best-fit values of 1.0 for the severe conditions assumed.

E. Reactor Flow Effects

Another significant variable to be considered in evaluating the design is the total system flow. Conservative values for system and reactor pressure drop have been determined to ensure that the required system flow is obtained in the as-built plant. The reactor vessel model test and the production pump tests have confirmed the design conditions.

The difference between the reactor system flow and the reactor core flow is the leakage flow. The leakage flow is defined as that part of the flow that does not contact the active heat transfer surface area. This part of the flow exists primarily through three different paths. These paths are (1) through the core shroud, (2) through the control rod guide tubes and instrument tubes, and (3) between all interfaces separating the inlet and outlet regions.

The leakage flow rates are determined through an iterative process. The total core flow is obtained by taking the difference between the total system flow and an assumed leakage value. Pressure drops are now calculated between the vessel inlet and outlet. These pressure drops are then used to calculate the flow rates through all the predefined leakage paths. The process is repeated until the calculated leakage value is equal to the assumed leakage value.

The leakage flow rates are determined in an appropriate manner to yield a minimum, nominal, and maximum value based on conservative assumptions for minimum and maximum values. For example, in determining the maximum leakage, the flow through the interfaces separating the inlet and outlet plenums is calculated by assuming the maximum tolerances at hot conditions to result in the minimum flow resistance. An example of the resulting maximum leakage through the various paths is shown in Table 3-18.

The reactor core flow and power capability were evaluated by determining the steady state power DNB ratios versus flow. Analyses were made for (a) variations of power capability with total reactor flow for a constant DNB ratio of 1.30, (b) DNB ratios for design flow with variations in hot channel mixing coefficients, and (c) DNB ratios for gross flow variations of ± 10 percent. The results are shown in Figures 3-41 and 3-42. For the analysis shown in Figure 3-41 for design hot channel condition, the flow was determined that would give a DNB ratio of 1.30 for a range of reactor powers. This analysis shows, for example, that a DNB ratio of 1.30 can be maintained in the hot channel at 114 percent power with a total reactor flow of 118 x 10^6 lb/h as compared with the available design flow of 131.3 x 10^6 lb/hr. The results shown by Line 2 in Figure 3-42 are the DNB ratios for rated flow of 131.3 x 10^6 lb/hr versus power. The limiting condition is 122.5 percent power for a DNB ratio of 1.30. Lines 1 and 3 show the DNB ratios versus power where the total system flow has been varied by ± 10 percent. Adequate DNB ratios can be maintained with a substantial reduction in reactor coolant system flow.

The foregoing sensitivity analyses were made using a fuel assembly design mixing coefficient of 0.02. A sensitivity analysis for a range of coefficients was made for the rated flow condition. The results are shown by Lines 4 and 5 of Figure 3-42 and discussed in more detail in Section 3.2.3.2.3.J.

F. Reactor Inlet Temperature Effects

The influence of reactor inlet temperature on power capability at design flow was evaluated. A variation of 1 °F in reactor inlet temperature will result in a power capability change of 0.6 percent at a given DNB ratio.

G. Fuel Temperature

1. Method of Calculation

A fuel temperature and gas pressure computer code was developed to calculate fuel temperatures, expansion, densification, equiaxed and columnar grain growth, center piping of fuel pellets, fission gas release, and fission gas pressure. Program and data comparisons were made on the basis of the fraction of the fuel diameter within these structural regions:

a. Outer limit of equiaxed grain growth - 2,700 °F.

b. Outer limit of columnar grain growth - 3,200 °F.

c. Outer limit of molten fuel (UO_2) - 5,080 °F.

Data from References 6, 37, 47, and 48 were used to compare calculated and experimental fractions of the rod in grain growth and central melting.

The radial expansion of the fuel pellet is computed from the mean fuel temperature and the average coefficient of linear expansion for the fuel over the temperature range considered. This model combined with the model for calculating the heat transfer coefficient was compared with the model developed by Notley et al (see Reference 49) of AECL. The difference in fuel growth for the two calculation models was less than the experimental scatter of data.

The fuel may be divided into as many as 30 radial and 70 axial increments for the analysis. An iterative solution for the temperature distribution is obtained. The thermal conductivity as shown in Figure 3-30 is input as a function of temperature and corrected for variations in fuel density. The relative thermal expansion of the fuel and cladding is taken into account when determining the temperature drop across the gap between the fuel and cladding surfaces.

The temperature drop across the gap is calculated using a gap conductance model based on the methods reported in Reference 40. The model, which has the capability of calculating gap conductance before fuel-to-clad contact as well as after contact, is an extension of the methods suggested by Ross and Stoute and is referred to as the TAFY-3 code (see Reference 54). This fuel-to-clad heat transfer is a function of gap width, gas conductivity, mean conductivity of the interface material, mean surface roughness, material hardness, and fuel-to-clad contact pressure. Before total

fuel-to-clad contact is made, a fraction of the fuel, based on fuel OD and gap size, is in contact with the clad. A constant contact pressure is applied to this fraction to simulate the effects of fuel cracking (see References 30 & 10).

The analytical model computes the amount of central void expected whenever the temperature approaches the threshold temperature for fuel migration and readjusts the density according to the new geometry.

The program uses a polynomial fit relationship for fuel thermal conductivity. The B&W reference design (see Reference 43) curve illustrated in Figure 3-30 for 95 percent dense UO_2 is a modification of the relationship presented in GEAP-4624 (see Reference 44). The curve yields a conservative integrated thermal conductivity of 93 W/cm with relatively little increase in conductivity beyond 3,000 °F.

2. Fuel Center Temperature Results at Beginning and End of Life

The results of the analysis for center temperatures with the methods described above are shown in Figures 3-43 and 3-44 for Beginning and End of Life (BOL and EOL) conditions. The BOL and EOL gas conductivity values are 0.14 and 0.05 Btu/hr-ft-°F. Christensen's fuel melting data (see Reference 19) is used to establish fuel melting temperatures.

A sensitivity analysis was made for a range of cold diametral clearances to show the effect of clearance on fuel center temperature. Temperatures for the nominal design clearance (0.007 inch), the maximum design clearance (0.0085 inch), and the maximum possible (0.0095 inch) are shown in Figures 3-43 and 3-44.

The calculated EOL center fuel temperatures are higher than the BOL values because of the reduction in the conductivity of the gas in the gap. The effect is apparent even though the fuel to clad diametral gap decreases. The calculation includes the effect of fuel swelling due to irradiation and accounts for the flux depression in the center of the rod because of the self-shielding effect of UO_2 (non-uniform power generation). The effect of clad irradiation and thermal expansion is also considered.

The most conservative assumptions using the B&W design curve with relatively little increase in thermal conductivity above 3,000 °F result in central fuel melting at about 23.4 kW/ft, which is 3.3 kW/ft higher than the maximum design value of 20.1 kW/ft at 114 percent power as shown in Figure 3-43.

The transient analyses at accident and normal conditions have been made using the design fuel thermal conductivity curve (Figure 3-30) to reflect a conservative value for the maximum average temperature and stored energy in the fuel. Use of this curve results in a higher temperature and, therefore, a lower Doppler coefficient since it decreases with temperature. Thus, the resultant Doppler effect is also conservative.

3. Fuel Center and Average Temperature Variations with Fuel Burnup

Maximum fuel temperature conditions are affected by the fuel-to-clad heat transfer coefficient. The coefficient is determined by fuel-to-clad clearance and gas conductivity. Fuel swelling due to burnup decreases the clearance and results in improved heat transfer; however, the conductivity of the gas decreases with the addition of xenon and krypton gas to the helium fill gas. A combination of these effects for BOL and EOL conditions was described in the previous section. It was conservatively assumed that the peak power could be obtained in a fuel rod with the maximum burnup. It is not likely that the peak power will be experienced in a rod with any significant fuel depletion; however, an additional sensitivity analysis of fission gas conductivity and fuel growth from zero to maximum burnup has been made for the maximum design cold diametral clearance of 0.0085 inch. This analysis shows that the worst combination of gas conductivity and fuel-to-clad clearance occurs at the maximum burnup or EOL. Center and average fuel temperature peak at EOL. The fuel center temperature for a fuel rod with the maximum design diametral clearance, maximum enrichment, and maximum linear heat rate will change with burnup as shown in the upper curve in Figure 3-45. The maximum design linear heat rate of 17.63 kW/ft for the 100 percent power condition is given in Table 3-1. The lower curve in Figure 3-45 is a comparison of the fuel center temperature at EOL and BOL as a function of heat rate.

4. Equilibrium Cycle Average Fuel Temperatures

An analysis has been made to show equilibrium average fuel conditions in the core. A typical fuel cycle, EOL condition was used to determine the fraction of fuel at a given average condition. The results are shown in Figure 3-46 for 100 percent power. The average fuel temperature in the core is 1,280 °F. A typical reactor power distribution at EOL as shown in Figure 3-47 was used to obtain fuel rod heat rates. A symmetrical cosine axial power distribution with a 1.50 max/avg value as shown in Figure 3-19 was used to predict the axial heat rate distribution. It was assumed that 97.3 percent of the power is generated in the fuel. The core radial power, assembly local power, and fuel rod axial power distributions were used to obtain the temperature distribution for this analysis.

The B&W design thermal conductivity was used to provide conservative values for fuel conductivity. The maximum powers occurred in fuel assemblies with one and two cycles of operation as shown in Figure 3-47 and the assemblies with the highest burnup did not exceed 1.087 times the average power for the typical case analyzed. Typical 6- and 10-kW/ft rod radial temperature profiles are shown in Figure 3-48.

H. Fission Gas Release

The fission gas release is based on results reported in GEAP-4596 (see Reference 39). Additional data from GEAP-4314 (see Reference 58), AECL-603 (see Reference 52), and CF-60-12-14 (see Reference 51) have been compared with the suggested release rate curve. The release rate curve (see Reference 39) is representative of the upper limit of release data in the temperature region of most importance. A maximum internal pressure of 3,300 psi is used to determine the clad stresses reported in Section 3.2.4.2.

The design values for fission gas release from the fuel and for the maximum clad internal pressure were determined by analyzing various operating conditions and assigning suitable margins for possible increases in local or average burnup in the fuel. A detailed analysis of the design assumptions for fission gas release and the relationship of burnup, fuel growth, and initial diametral clearance between the fuel and clad are summarized in the following paragraphs. An evaluation of the effect of having the fuel pellet internal voids available as gas holders is also included.

1. Design Assumptions

   a. Fission Gas Release Rates

      The fission gas release rate is calculated as a function of fuel temperature at 114 percent of rated power. The procedures for calculating fuel temperatures are discussed in Section 3.2.3.2.3.G. The fission gas release curve and the supporting data are shown in Figure 3-49. Most of the data are on or below the design release rate curve. A release rate of 51 percent is used for the portion of the fuel above 3,500 °F. The fuel temperatures were calculated using the B&W design fuel thermal conductivity curve which yields conservatively high values for fuel temperatures.

   b. Axial Power and Burnup Assumptions

      The temperature conditions in the fuel are determined for the most severe axial power peaking expected to occur. Two axial power shapes have been evaluated to determine the maximum release rates. These are 1.50 and 1.70 max/avg shapes as shown in Figure 3-19. The quantity of gas released is found by applying the temperature-related release rates to the quantities of fission gas produced along the length of the fuel rod.

      The quantity of fission gas produced in a given axial location is obtained from reactor core axial region equilibrium burnup studies. Three curves showing the axial distribution of burnup as a local-to-average ratio along the fuel rod are shown in Figure 3-50. Values at 100 and 300 days of operation and EOL are shown.

      The EOL axial burnup distribution is the condition with the maximum fission gas inventory. The average burnup at the EOL in the hot fuel rod is 40,900 MWd/mtU which has been determined as follows:

Calculated Hot Bundle Average Burnup, MWd/mtU	35,400
Hot Fuel Rod Burnup Factor	1.05
Margin for Calculation Accuracy	1.10
Hot Rod Maximum Average Burnup, MWd/mtU	40,900

      The local burnup along the length of the fuel rod is the product of the hot rod maximum average value above and the local-to-average ratio shown in Figure 3-50. The resulting hot rod local maximum burnup for the end of life condition is about 44,950 MWd/mtU.

c. Hot Rod Power Assumptions

Maximum fuel temperature was determined as a function of fuel burnup for the maximum design heat rate of 17.63 kW/ft by operating continuously at the rated power level throughout life. Conservative fuel and clad properties and gas conditions were used to determine the fuel-to-clad heat transfer coefficient. A study of the power distribution in the core through several cycles to equilibrium conditions shows the assembly average burnups as a function of power for all assemblies. The power burnup history for fuel rods is also determined by considering the local peaking factors. A conservative margin for calculation accuracy was included in the reference design power history. Fission gas release and internal rod pressure were determined for rated and maximum overpower conditions.

d. Fuel Growth Assumptions

The fuel growth was calculated as a function of burnup as indicated in Section 3.2.4.2.1. Fuel pellet dimensions in the thermal temperature and gas release models were increased to the EOL conditions as determined above.

e. Gas Conductivity and Contact Heat Transfer Assumptions

The quantity of fission gas released is a function of fuel temperature and fuel burnup. The temperatures are influenced by three factors: (a) the conductivity of the fission gas in the gap between the fuel and clad, (b) the diametral clearance between fuel and clad, and (c) the heat transfer conditions when the fuel expands enough to contact the clad. Burnup is influenced by two factors: (a) the power history of the fuel and (b) the initial fuel concentration.

Gas conductivity varied with burnup in the analysis. Diametral clearances of 0.0025- to 0.0065-inch reflecting minimum and maximum design clearances after fuel growth were analyzed. The contact heat transfer coefficients were calculated as suggested in Reference 40 and are illustrated in Figure 3-51. The gap conductance is plotted as a function of heat rate for two fuel-to-clad gap clearance conditions to show the dependence of fuel-to-clad heat transfer on this parameter. Heat transfer models presented in the literature (see References 43 & 62) suggest that gap conductance is higher than the design values used in this analysis.

2. Summary of Results

The reference design power history, Section 3.2.3.2.3.H.1.c, was used to determine the maximum internal fuel rod pressure and corresponding fission gas release rate. Pressures and rates were determined for various cold diametral clearances and axial power peaking shapes.

Fission gas release rate results are shown in Figure 3-52. The highest release rate is for a 1.70 max/avg power shape as shown in Figure 3-52 with an EOL axial burnup shape and closed fuel porosity. The increase in release rate with diametral clearance results from higher fuel temperatures. The release rate at the minimum

clearance, 0.0045 inch, is 5.0 percent. This condition is equivalent to the minimum gap after irradiation growth and produces the maximum clad stress (maximum sized pellets with minimum internal diameter cladding). The release rate is 16.0 percent for the maximum design diametral clearance (0.0085 inch).

An additional case was examined to check the sensitivity of the calculations to axial power shapes. The results are shown in Figure 3-52. The effect of open fuel porosity on fission gas release is also shown in Figure 3-52.

Maximum internal pressure due to the release of fission product gases are shown in Figure 3-53. Internal clad pressure is plotted as a function of cold diametral gap for the 1.50 max/avg and 1.70 max/avg axial power shapes with the expected EOL burnup distribution. The lower curve for the 1.70 max/avg power shape assumed that 7.5 percent of the fuel volume is available to hold the released gas (open porosity). The remaining curves correspond to closed porosity. The present design condition being used to determine the maximum internal pressure assumes a closed pore condition with all released gas contained outside the fuel pellets in the spaces between the expanded dished ends of the pellets, the radial gaps (if any), and the void spaces at the ends of the fuel rod. The effects of fuel densification and grain growth described in Section 3.2.3.2.3.G are included in the analysis. The expected maximum internal pressures are not strongly influenced by the axial power shape.

The allowable design internal pressure of 3,300 psi is well above the maximum values of internal pressures calculated for open or closed pores and the maximum internal pressure should only occur with the maximum design diametral clearance condition. An increase in maximum fuel burnup can be tolerated within the prescribed internal pressure design limits.

It has been indicated in Reference 49 and in AECL-1598 that the UO_2 fuel is plastic enough to flow under low stresses when the temperature is above 1,800 °F. That fraction of the fuel below this temperature may retain a large portion of the original porosity and act as a fission gas holder. The hottest axial locations producing the highest clad stresses will have little if any fuel below 1,800 °F. However, the end of the fuel rods will have some fuel below this temperature.

The approximate fraction of the fuel below 1,800 °F at maximum overpower (114 percent) for a 1.5 axial power shape is as follows for various cold diametral clearances. The bundle average powers shown in Figure 3-47 were used to determine the heat rates.

Clearance, in.	Percent of Fuel Below 1,800 °F, %
0.0045	79
0.0070	69
0.0085	62

The retention of fuel porosity in the low temperature and low burnup regions will result in modest reductions in internal gas pressure.

Gas pressure at rated and overpower conditions are shown in Figure 3-53. The overpower condition is not expected to occur except for brief periods during operating transients.

I. Hot Channel Factors Evaluation

1. Rod Pitch and Bowing

A flow area reduction factor is determined for the as-built fuel assembly by taking channel flow area measurements and statistically determining an equivalent hot channel flow area reduction factor. A fuel assembly has been measured, and the results are shown in Table 3-12. Interior channel measurements and measurements of the channels formed by the outermost fuel rods with adjacent assemblies have been analyzed. Coefficients of variation for each type of channel have been determined. In the analytical solution for a channel flow, each channel flow area is reduced over its entire length by the F_A factors shown in Figure 3-34 for the desired population protected at a 99 percent confidence. The hot channels have been analyzed during values for 95 percent population protected, or F_A in the interior cells of 0.98 and F_A in the wall cells of 0.97 as listed in Section 3.2.3.1.1.J.

Special attention is given to the influence of water gap variation between fuel assemblies when determining rod powers. Nuclear analyses have been made for the nominal, maximum, and minimum spacing between adjacent fuel assemblies. The nominal and maximum hot assembly fuel rod powers are shown in Figures 3-54 and 3-55. The hot channel nuclear power factor ($F\Delta h$ nuclear) of 1.78 shown in Section 3.2.3.1.1 is based on Figure 3-55 for the worst water gap between fuel assemblies. The factor of 1.783 is a product of the hot assembly factor of 1.68 times the 1.061 hot rod factor. This power factor is assigned to the hottest unit cell rod which is analyzed for burnout. Peaking factors for other channels are obtained in a similar manner. In all cases, the combined flow spacing and power peaking producing the lowest DNB ratio is used.

A penalty factor for rod bowing is calculated for each operating cycle using the highest assembly burnup in the cycle core. The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup-dependent DNBR rod bow penalty for the applicable cycle minus a credit of one percent for the flow area reduction factor used in the hot channel analysis. All plant operating limits prior to Cycle 7 were based on an original method of calculating rod bowing penalties that are more conservative than those that would be obtained with new approved procedures. Prior to Cycle 7, this subrogation resulted in a DNBR margin in excess of 3.8 percent, which is partially used to offset the reduction in DNBR due to fuel rod bowing. The rod bow penalty factor was applied to all plant operating limits which were based on DNBR criteria through the inclusion of a minimum of 10 percent DNBR margin in each operating limit to offset the impact of any rod bow penalty. Beginning with Cycle 7, the DNB impact of fuel rod bowing was evaluated using the method described in BAW-10147PA-R1.

2. Fuel Pellet Diameter, Density, and Enrichment Factors

Variations in the pellet size, density, and enrichment are reflected in coefficients of variation Numbers 2 through 7 of Table 3-12. These variations have been obtained from the measured or specified tolerances and combined statistically as described in Section 3.2.3.2.2 to give a power factor on the hot rod. For 99 percent confidence and 95 percent population conditions, this factor, F_Q, is 1.011 and is applied as a power increase over the full length of the hot channel fuel rod. The local heat flux factor, $F_{Q''}$, for similar conditions is 1.014. These hot channel values are shown in Table 3-1. The corresponding values of F_Q and $F_{Q''}$ with 99.99 percent population protected are 1.025 and 1.03, respectively. A conservative value of $F_{Q''}$ of 1.03 for 99 percent confidence and 99.99 percent population is used for finding the maximum fuel linear heat rates as shown in Section 3.2.3.1.2.

These factors are used in the direct solution for channel enthalpies and are not expressed as factors on enthalpy rise as is often done.

Additional information concerning the effects of fuel densification on fuel performance is provided in B&W Topical Reports BAW-10055, Revision 1, Fuel Densification Report, and BAW-1391, Arkansas Nuclear One - Unit One, Fuel Densification Report.

3. Flow Distribution Effects

Inlet Plenum Effects

The inlet plenum effects have been determined from the 1/6-scale model flow test. It has been conservatively assumed that the flow in the hot bundle position is five percent less than average bundle flow under isothermal conditions corresponding to the model flow test conditions. An additional reduction of flow due to hot assembly power is described below.

Redistribution in Adjacent Channels of Dissimilar Coolant Conditions

The hot fuel assembly flow is less than the flow through an average assembly at the same core pressure drop because of the increased pressure drop associated with a higher enthalpy and quality condition. This effect is allowed for by making a direct calculation for the hot assembly flow. The combined effects of upper and lower plenum flow conditions and heat input to the hot assemblies have been used to determine hot assembly flows. The worst flow maldistribution effect has been assumed in the design and the minimum hot assembly flow has been calculated to be 87 percent of the average assembly flow at 114 percent overpower. Actual hot assembly flows are calculated rather than applying an equivalent hot channel enthalpy rise factor.

Physical Mixing of Coolant Between Channels

The flow distribution within the hot assembly is calculated with a mixing code that allows an interchange of heat between channels. Mixing coefficients have been determined from multirod mixing tests. The fuel assembly, consisting of a 15 x 15 array of fuel rods, is divided into unit, wall, control rod, and corner cells as shown by the heavy lines in Figure 3-54. The mixed enthalpy for every cell is determined

simultaneously so that the ratio of cell to average assembly enthalpy rise (Enthalpy Rise Factor) and the corresponding local enthalpy are obtained for each cell. Typical enthalpy rise factors are shown in Figures 3-54 and 3-55 for the hot and surrounding cells. The assumptions used to describe the channels for the peaking and enthalpy rise factors shown are given in Section 3.2.3.2.3.J below.

J. Evaluation of the DNB Ratios in the Unit, Wall, Control Rod, and Corner Cells

1. DNB Results at Rated Flow

The DNB ratios in the hot unit cell at the maximum design condition described in Section 3.2.3.1 are shown in Figure 3-26. The relationship shown is based on the application of the W-3 correlation. An additional sensitivity analysis of the assembly corner, wall, i.e. peripheral, and control rod cells has been made for the worst combination of fuel assembly spacing and power peaking.

The sensitivity of the assembly design with respect to variations of mass velocity (G), channel spacing, mixing intensity, and local peaking on the DNB ratios in the fuel assembly channels has been evaluated by analyzing the most probable conditions and the postulated maximum design condition. The summary results are shown in Table 3-19. The unit cell DNB ratios are repeated for comparison. All of the DNB ratios are for 114 percent overpower.

The DNB ratios in all channels are high enough to ensure a confidence-population relationship equal to or better than that outlined in Section 3.2.3.1.1 for the hot unit cell channel. All of the wall, corner, and control rod cells have DNB ratios equal to or greater than that of the unit cell hot channel. This results from a more favorable flow to power ratio in these cells associated with relatively larger flow areas.

The DNB ratios were obtained by comparing the fuel rod local heat fluxes and channel coolant conditions with the limitations predicted by the correlation. Typical results are shown in Figures 3-56 and 3-57 for the most probable and maximum design conditions in the unit cell.

2. Fuel Rod Power Peaks and Cell Coolant Conditions

The most probable case local-to-average rod powers and the local-to-average exit enthalpy rise ratios are shown in Figure 3-54 for the hot corner, wall, control rod, and unit cells in the hot fuel assembly. Values shown are for nominal water gaps between the hot fuel assembly and adjacent fuel assemblies with nominal flow to the hot fuel assembly and with a minimum intensity of turbulence, α, (The intensity of turbulence, α, is defined as

$$\sqrt{V_t'^2} \Big/ V$$

where V_t' is the transverse component of the fluctuating turbulent velocity and V is the coolant velocity in the axial direction. This method of computing mixing is described by Sandberg, R.O. and Bishop, A.A., CVTR Thermal-Hydraulic Design for 65 MW Gross Fission Power, CVNA-227.) equal to 0.02.

The maximum design case local-to-average rod power (nuclear peaking factor) and exit enthalpy rise factors in the hot fuel assembly are shown in Figure 3-55. The factors were determined for this case with the minimum water gap between the hot fuel assembly and adjacent fuel assemblies, with minimum flow to the hot fuel assembly, and with a minimum assumed intensity of turbulence, α, equal to 0.02. An evaluation of minimum, nominal, and maximum spacing between assemblies showed the minimum to have the lowest DNB ratios.

An intensity of turbulence of 0.02 was used for both most probable and maximum design case analyses. The influence of turbulence is shown in Figure 3-42, which shows values ranging from 0.01 to 0.06. The value of 0.02 is sufficiently conservative for design evaluation. The conditions analyzed to obtain the DNB ratios for various values of the intensity of turbulence shown in Figure 3-42 were outlined previously in Section 3.2.3.2.3.I.

3. Fuel Assembly Power and Rated Flow Conditions

The most probable and maximum design cases were run at 114 percent reactor power with the nominal and worst $F\Delta h$ factors shown in Section 3.2.3.1.1.C. The 1.50 modified cosine axial power shape of Figure 3-19 was used to describe the worst axial condition.

The hot assembly flow under most probable conditions without a flow maldistribution effect is 96 percent of the average assembly flow, and the reduction in flow is due entirely to heat input effects. The hot, assembly flow under the maximum design conditions is 87 percent of the average assembly flow and considers the worst combined effects of heat input and flow maldistribution.

K. DNB Results for Postulated Loss of an Internals Vent Valve

The reactor arrangement includes vent valves above the core to equalize the pressure between inlet and outlet regions during a LOCA. The effective core flow will be reduced in the unlikely event that an internals vent valve is open. A DNB analysis was made to show the design margin for a postulated accidental failure of one valve disc.

1. Four-Pump Operation

In the event the disc from one of these valves is completely removed, a small reduction in effective core flow for heat removal will be experienced. Approximately 5.7 percent of the incoming flow will bypass the core through the valve opening. However, the reduction of resistance results in an increase in total system flow of about 1.1 percent. The net reduction of flow for core heat removal is 4.6 percent.

A complete sensitivity analysis has been made to determine the effects of design flow and unexpected core bypass flow from the inlet to the outlet chambers in the reactor vessel. The results are shown in Figure 3-58. Bypass flow was varied from 0 to 10 percent while holding a constant reactor average temperature of 578.9 °F. A design allowance of two percent (2.63×10^6 lb/hr) bypass flow for vent valve seat and fitup leakage is included in all calculations for most probable or maximum design DNB ratios. This design condition is indicated by Line 2 and is identical with rated conditions in Figure 3-26 as previously discussed.

Line 5 shows the DNB ratios versus power for a condition of the loss of one vent valve disc.

A comparison of DNB ratios at various power levels for full flow and reduced flow with one vent valve open is given in Table 3-20. A minimum DNB ratio of at least 1.30 is maintained at the maximum design overpower with one vent valve open. DNB ratios were determined for the worst corner, control rod, wall, or unit cell. The DNB ratios in the hot unit cell were the lowest.

2. Partial Pump Operation

The power limitations imposed on the reactor due to the loss of one or more pumps has been examined for the full flow condition in Section 3.2.3.1.1.K. The results of a similar analysis for the reduced flow conditions caused by the loss of an internals vent valve are also tabulated in Table 3-20.

3.2.3.2.4 Evaluation of Internals Vent Valve

A vapor lock problem could arise if water is trapped in the steam generator blocking the flow of steam from the top of the reactor vessel to a cold leg leak. Under this condition, the steam pressure at the top of the reactor would rise and force the steam bubbles through the water leg in the bottom of the steam generator. This same differential pressure that develops a water leg in the steam generator will develop a water leg in the reactor vessel which could lead to uncovering of the core.

The most direct solution to this problem is to equalize the pressure across the core support shield, thus eliminating the depression of the water level in the core. This was accomplished by installing vent valves in the core support shield to provide direct communication between the top of the core and the coolant inlet annulus. These vent valves open on a very low pressure differential to allow steam generated in the core to flow directly to the leak from the reactor vessel. Although the flow path in the steam generator is blocked, this is of no consequence since there is an adequate flow path to remove the steam being generated in the core.

During the vent valve conceptual design phase, criteria were established for valves for this service. The design criteria were: (1) functional integrity, (2) structural integrity, (3) remote handling capability, (4) individual part-capture capability, (5) functional reliability, (6) structural reliability, and (7) leak integrity throughout the design life. The design criteria resulted in the selection of the hinged-disc (swing-disc) check valve, which was considered suitable for further development.

Because of the unique purpose and application of this valve, B&W recognized the need for a complete detailed design and development program to determine valve performance under nuclear service conditions. This program included both analytical and experimental methods of developing data. It was performed primarily by B&W and the selected valve vendor or his subcontractors.

Vent valve preliminary design drawings were prepared and analyzed both by B&W and the vendor/subcontractor. Specifications and drawings were prepared, and orders were placed with the vendor for the design, development, fabrication, and test of a full-size prototype vent valve. The prototype valve was completed and subjected to the tests described in Section 3.3.4. All testing was successfully completed and minor problems encountered during valve assembly handling or use were corrected to arrive at the final design for the production valve.

The only significant problem encountered during test was seizing of one jackscrew. This was attributable to an excessive thickness of "Electrolyze" which spalled off the screw threads. This problem was corrected by reducing the specified "Electrolyze" thickness from 0.0015 inch to 0.0004 inch maximum and no further galling was encountered. To further enhance resistance to galling, the final design jackscrew has a 1-1/8 inch-8 Acme thread form instead of a 1 inch-12 UNF and the material is an age hardened corrosion resistant alloy instead of 410 SS. No further jackscrew problems have occurred or are anticipated on the basis that the surfaces are separated by the low friction "Electrolyze," different materials of different hardnesses are used, loose fits are employed, and thread contact stresses are low (3,775 psi).

The final design of this valve is shown in Figure 3-62. The valve disc hangs closed in its natural position to seal against a flat, stainless steel seat inclined five degrees from vertical to prevent flow from the inlet coolant annulus to the plenum assembly above the core. In the event of LOCA, the reverse pressure differential will open the valve. At all times during normal reactor operation, the pressure in the coolant annulus on the outside of the core support shield is greater than the pressure in the plenum assembly on the inside of the core support shield. Accordingly, the vent valve will be held closed during normal operation. With four reactor coolant pumps operating, the pressure differential is 42 psi resulting in a several thousand pound closing force on the vent valve.

Under accident conditions, the valve will begin to open when a pressure differential of less than 0.15 psi develops in a direction opposite to the normal pressure differential. At this point, the opening force on the valve counteracts the natural closing force of the valve. With an opening pressure differential of no greater than 0.3 psi, the valve would be fully open. With this pressure differential, the water level in the core would be above the top of the core. In order for the core to be half uncovered, assuming solid water in the bottom half of the core, a pressure differential of 3.7 psi would have to be developed. This would provide an opening force of about 10 times that required to open the valve completely. This is a conservative limit since it assumes equal density in the core and the annulus surrounding the core. The hot, steam-water mixture in the core will have a density much less than that of the cold water in the annulus and somewhat greater pressure differentials could be tolerated before the core is more than half uncovered.

An analog computer simulation was developed to evaluate the performance of the vent valves in the core support shield. This analysis demonstrated that adequate steam relief exists so that core cooling will be accomplished. This analysis is described in detail in Section 14.2.2.5.

The behavior of the valve disc during LOCA conditions was investigated and the rather complex dynamic behavior of the disc during LOCA was analyzed as a series of simpler models which provide conservative predictions of peak stresses and deflections.

The valve disc remains closed initially for the LOCA hot leg (36-inch pipe) case and the disc opening on subsequent differential pulses is less than one-half of the initial disc to vessel wall impact velocity for the LOCA cold leg (28-inch pipe) case. Therefore, the disc motion and initial impact with the vessel inside wall was chosen as the worst case and the only one requiring consideration. The cold leg LOCA pressure-time history acting on the disc was approximated by a piecewise linear time function. The moment due to pressure was equated to the rotary inertia of the disc to determine the velocity of impact with the vessel inside wall.

The model chosen for the initial impact consisted of three effective springs and two masses to represent the disc with its log, the compliance of the disc, and the vessel inside wall.

Loads generated on impact were based on the conservation of energy. The stresses obtained for these loads indicated that the elastic model assuming conservation of energy was not valid and that the impact must assume plastic deformation.

The plastic analysis provided the following information:

A. Crush deformation of lug after disc corner contacts the vessel wall is predicted to be 0.165 inch.

B. The total deformation of lug from contact with the vessel wall until disc assembly motion is arrested is predicted to be 0.483 inch.

C. The total angular deformation at the plastic hinge is predicted to be 0.016 radians.

D. An analysis was performed on the reactor vessel wall for disc assembly impact and the results indicate that while the stainless steel cladding is deformed locally, the reactor maintains its structural and pressure boundary integrity.

Because of conservative assumptions used in the plastic analysis, actual deformations will be considerably less than the above predicted values. Although plastic deformation may occur as predicted above on impact, the disc will retain its structural integrity. Plastic deformation of the disc dissipates the stored kinetic energy stored in the disc effectively; thus, the energy available for rebound is less than one percent of the initial impact energy and is too low to overcome the pressure differential and cause impact on the valve body. Disc and body hinge components were analyzed for worst case disc impact loadings and the resulting stresses were found to be less than the allowable limits; therefore, the valve disc free-motion (venting) function will be unaffected.

From the above, it is concluded that vent valve performance will not be impaired during the course of an accident because disc free-motion part stresses remain within allowable limits, disc structural integrity is maintained, vessel pressure boundary integrity is maintained, and plastic deformation of the disc seating surface improves the venting function.

With reference to Figure 3-62, each jackscrew assembly consists of a jackscrew sleeve, jackscrew locking cup, bolted block capture cover, cover attachment fasteners (socket head cap screws), and bolt locking cups. The potential for loss of jackscrew assembly parts during the plant lifetime is considered remote on the basis that the jackscrews and capture parts are accessible for visual inspection during scheduled refueling outages. A jackscrew loss is considered remote because a failure in service is highly improbable with the low compressive load (1,000 psi) involved and the jackscrew is retained in the valve body by a central shoulder and the ends are threaded into the retaining rings. Capture cover failures and loss is highly improbable on the same basis that it is not loaded in service. The capture cover is attached to the upper retaining ring by socket head cap screws which are lock welded to the cover at installation. By design, these screws are retention rather than structural devices and are not loaded in service. Locking cups (welded to cover) with two crimps 180° apart are used to lock each screw head to the capture cover and in the absence of loads on both the cover and screws, the likelihood of lock weld failure and loss of screw heads is considered remote. With the capability to inventory these cap screw heads visually at scheduled refuelings, any problem related to the loss of these screws would be apparent early in the plant life and the valve assemblies could be removed for corrective action.

The internals vent valves are described, including materials and hinge part loose clearances, in Section 3.2.4.1.2.H.

The internals vent valves have been tested for ability to withstand the effects of vibratory excitations and for other functional characteristics as described in Section 3.3.4.

3.2.3.3 Cycle by Cycle Fuel Thermal Analysis Methodology

The following section describes the changes which have taken place in the fuel thermal analysis methodology since the original Cycle 1 evaluation as well as the significant changes to the fuel design that affect the fuel thermal performance. Cycle-specific Reload Reports are listed in Section 3.4.

Cycle 1 through Cycle 6

As stated in Section 3.2.3.2.3.G, the FSAR analysis was performed using the TAFY-3 thermal analysis code. This original analysis did not consider the effects of fuel densification. BAW-10054 set forth guidelines for evaluating densification effects in light water reactor fuel. Therefore, the methods for evaluating the fuel thermal performance of the ANO-1 fuel were revised in the ANO-1 Fuel Densification Report (BAW-1391). The new method assumed instantaneous densification and incorporated densified fuel parameters. These methods continued to be used, along with the TAFY-3 code (BAW-10044), through Cycle 6. The four Mark-BEB lead test assemblies, introduced during Cycle 5, were evaluated using TACO2 (BAW-10141A).

Cycle 7 through Cycle 11

Beginning with the Cycle 7 fuel thermal analysis, the TACO2 fuel performance code was used. TACO2 provided more accurate temperature predictions than TAFY-3 and also included a time dependent densification model. Therefore, using densified fuel parameters was no longer necessary for the fuel thermal analysis. Although a direct one-to-one comparison of the fuel temperature predictions between the TAFY and TACO2 code is not shown in the SAR, an extensive justification of the TACO2 code is provided in BAW-10141A. The overall trends observed in Figures 3-43 through 3-46 for the TAFY-3 code are similar to those observed using the TACO2 code.

Cycles 12 and 13

Cycle 12 marked the first use of the B9 fuel rod design which contains a fuel pellet with a slightly larger diameter resulting in a slightly smaller pellet-clad gap than those in earlier fuel rod designs. The fuel performance of the B9 fuel rod design was determined using the TACO3 code (BAW-10162A). The fuel performance of the previously irradiated fuel was evaluated with either the TACO2 or TACO3 code. Beginning with the Cycle 12 fuel thermal analyses, a transition began from the TACO2 code to the TACO3 code (BAW-10162A) for the fresh fuel designs. The TACO3 code uses best estimate inputs and statistical parameters to yield best estimate fuel thermal predictions. Statistical evaluations are performed by the TACO3 code to determine uncertainties and to provide conservative results for reload licensing analyses. The code was used in conjunction with the fuel thermal analysis criteria presented in BAW-10179P-A. Although a direct one-to-one comparison of the fuel temperature predictions between the TACO3 and TACO2 code is not shown in the SAR, an extensive justification of the TACO3 code is provided in BAW-10162A. The overall trends observed in Figures 3-43 through 3-46 for the TAFY-3 code are similar to those observed using the TACO3 code.

Cycle 14

For the first time, the predicted fuel rod internal gas pressure for some of the irradiated fuel during a cycle exceeded the reactor coolant system pressure. This action was permissible based on the approved methodology defined in BAW-10183P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)." BAW-10183P-A describes the method with which the fuel rod internal gas pressure is restricted to a value below that which would cause a) the fuel-clad gap to increase due to outward cladding creep during steady-state operation, and b) extensive DNB propagation to occur.

The reload licensing analyses were performed in accordance to the methodology of BAW-10179P-A, Revision 1. This revision incorporated the methods and criteria used for the ANO-1 core as defined in BAW-2149A, “Evaluation of Replacement Rods in BWFC Fuel Assemblies”, BAW-10156A, Revision 1, “LYNXT Core Transient Thermal-Hydraulic Program”, and BAW- 10183P-A, “Fuel Rod Gas Pressure Criterion (FRGPC)”.

3.2.3.4 Cycle by Cycle Thermal-Hydraulic Methodology

The following section describes the various revisions that have been made in thermal-hydraulic methodology since the original Cycle 1 evaluation. This section begins with a description of the changes which were incorporated in the ANO-1 Fuel Densification Report and proceeds with a description of the major changes which were incorporated for each successive reload.

Cycle 1 - Fuel Densification Report

The ANO-1 Fuel Densification Report incorporated the effects of a reduced stack height (as a result of densification) and a densification power spike into the DNB analysis. These effects resulted in a higher surface heat flux and a lower minimum DNBR. In response to the densification analysis, the overpower limit was reduced from 114% to 112%.

Cycle 2

As outlined in the Cycle 2 Reload Report (BAW-1433), several changes were made in the methods and models used for thermal-hydraulic evaluations between Cycles 1 and 2. For DNB analyses, the B&W-2 critical heat flux correlation was implemented with a 1.30 design limit. Changes to the analysis parameters included increasing the DNBR design limit by 5.9% to account for fuel rod bow while eliminating the use of densification power spikes and an open vent valve flow penalty. Based on actual measured flow during Cycle 1, the thermal design flow was increased to 106.5% of design flow for the Cycle 2 thermal-hydraulic evaluations. This resulted in a 1 °F increase in core inlet temperature. With the implementation of the new CHF correlation, the general analysis methodology described in section 3.2.3.1.1 was maintained.

Cycle 3

As outlined in the Cycle 3 Reload Report (BAW-1471), the thermal-hydraulic analysis for Cycle 3 used the same methods as the revised Cycle 2 analysis described in the Cycle 2 Reload Report. This method, as described previously, removed the densification power spike from the DNBR calculation and increased the DNBR design limit to account for fuel rod bow. This increase was calculated to be 11.2% for Cycle 3.

Cycle 4

As stated in the Cycle 4 Reload Report (BAW-1504), the Cycle 4 thermal-hydraulic evaluation methodology is the same for Cycles 3 and 4 with two exceptions. Based on performance anomalies at other B&W plants, all orifice rod assemblies were removed for Cycle 4 operation. In addition, retainer assemblies were installed on all BPRAs and on two assemblies containing regenerative neutron sources. The combined effect of these design changes resulted in a reduction in the minimum DNBR of approximately 0.02 relative to Cycle 3 for steady-state and transient predictions. The Cycle 4 analysis maintained the 11.2% increase in the DNBR design limit to account for fuel rod bow.

Cycle 5

As stated in the Cycle 5 Reload Report (BAW-1658, Revision 2), the Cycle 5 nuclear design allowed a reduction in the design radial-local peak from 1.78 to 1.71. This resulted in the minimum steady-state DNBR increasing from 1.88 to 2.05 for the 112% maximum design overpower condition. For Cycle 5, the fuel rod bow penalty was set at 4%. Four Mark BEB extended burnup lead test assemblies were introduced into the Cycle 5 core.

Cycle 6

As stated in the Cycle 6 Reload Report (BAW-1747), the Cycle 6 thermal-hydraulic design is identical to Cycle 5. For Cycle 6, the fuel rod bow DNBR reduction was calculated to be 1.1%. The Cycle 6 core contained four once-burned extended burnup lead test assemblies.

Cycle 7

As outlined in the Cycle 7 Reload Report (BAW-1840), several changes were made to the methods and models used in the thermal-hydraulic analysis for ANO-1 Cycle 7. Cycle 7 marked the first implementation for ANO-1 of crossflow modeling with the codes LYNX1 (BAW-10129A, “LYNX1: Reactor Fuel Assembly Thermal Hydraulic Analysis Code”), LYNX2 (BAW-10130A, “LYNX2: Subchannel Thermal-Hydraulic Analysis Program”), and LYNXT (BAW-10156A, “LYNXT - Core Transient Thermal-Hydraulic Program”). The crossflow codes were used according to the approved methodology in BAW-1829A, “Thermal-Hydraulic Crossflow Applications”. In addition, Cycle 7 used a 1.65 symmetric chopped cosine design axial flux shape, which was an increase from the 1.5 axial flux shape used previously. The minimum steady-state DNBR based on the B&W-2 correlation increased from 2.05 to 2.08 for the 112% maximum design overpower condition for the transition to crossflow modeling with the associated change in design axial peaking. With the implementation of crossflow modeling, the general analysis methodology was maintained in demonstrating adequate DNB protection using an approved CHF correlation, however, the new modeling technique incorporated different ways to address flow redistribution effects as defined in BAW-1829. Therefore, some of the modeling assumptions considered for Cycle 1 were no longer applicable.

It was also determined at this time that using densified fuel parameters provided an unnecessary conservatism in the thermal-hydraulic analysis as described in BAW-1829. Therefore, beginning with the Cycle 7 analyses, nominal undensified fuel parameters were used. Based on the B&W rod bow topical report (BAW-10147), the fuel rod bow penalty was eliminated for Cycle 7. Cycle 7 contained four twice burned extended burnup lead test assemblies.

Cycle 8

As stated in the Cycle 8 Reload Report (BAW-1918), the methods and models used in the thermal-hydraulic analysis of Cycle 8 are identical to those used for Cycle 7. Cycle 8 contained one thrice burned extended burnup lead test assembly.

Cycle 9

Mark-BZ fuel assemblies, with Zircaloy intermediate spacer grids, were first used in the core in Cycle 9. As presented in the Cycle 9 Reload Report (BAW-2027), the BWC critical heat flux (CHF) correlation (BAW-10143A), as described in section 3.2.3.1.1.B.4, was used for the analysis of these assemblies. The design configuration chosen for the Cycle 9 thermal-hydraulic analysis was a full core of Zircaloy-grid fuel assemblies, for which the core bypass flow rate was 8.8 percent of the RCS flow rate. The revised reference minimum steady-state DNBR for the 112% maximum design overpower condition was found to be 1.77 based on the BWC correlation design limit of 1.18. The use of a full core of Zircaloy grid fuel assemblies was demonstrated to be conservative for the actual Cycle 9 configuration, therefore, a transition core DNBR penalty was not necessary.

As presented in the Cycle 9 Reload Report (BAW-2027), the BWC critical heat flux correlation was used for the analysis of these assemblies. Methods and models for other aspects of the thermal-hydraulic analysis were identical to the Cycle 8 analysis.

Cycle 10

Cycle 10 (BAW-2114) was the second cycle in the transition from the Mark-B Inconel-grid fuel design to the Mark-BZ, Zircaloy-grid fuel design. The design of the fresh Mark-B8 fuel assemblies was such that the fuel was hydraulically, thermally, and geometrically similar to the fuel design used in the remainder of the Cycle 10 core. The methods, codes, and models used in the thermal-hydraulic analysis of Cycle 10 were identical to those used in Cycle 9.

Cycle 11

The methods, codes, and models used in the thermal-hydraulic analysis of Cycle 11 (BAW-2153) were identical to those for Cycle 10.

Cycle 12

The methods, codes, and models used in the thermal-hydraulic analysis of Cycle 12 (BAW-2194) were identical to those for Cycle 11 except for the use of the TACO3 code, use of a different method to calculate the engineering hot channel factor F_q, and the elimination of the engineering hot channel factor $F_{q''}$ from DNBR predictions. All methods and criteria used for Cycle 12 were consistent with those described in BAW-10179P-A.

The introduction of the Mark B8ZL fuel assembly into the Cycle 12 core was demonstrated not to alter the limiting hot pin DNB performance. The conservatism in the Cycle 9 reference DNB analysis was shown to offset the effect of the higher heat flux for the slightly shorter initial stack height in the B9 fuel rods.

Cycle 13

The methods, codes, and models used in the thermal-hydraulic analysis of Cycle 13 (BAW-2232) were identical to those used for Cycle 12. The reference DNB analysis was shown to bound the effect of the skirtless lower end grid of the Mark-B9ZL fuel assembly.

Cycle 14

The methods, codes, and models used in the thermal-hydraulic analysis of Cycle 14 (BAW-2276) were identical to those used for Cycle 13. The primary change for Cycle 14 was the tightening of the DNB-based pressure-temperature safety limit lines for 4, 3, and 2 reactor coolant pump operation. This action was taken to avoid a restriction on the control rod insertion limits. As a result, the variable low pressure trip was also tightened accordingly.

3.2.4 MECHANICAL DESIGN

This section presents the evaluation of significant mechanical design parameters for the reactor internals, core components, and control rod drives and shows that the basic mechanical design satisfies the performance limits and objectives of Section 3.1.2.4.

3.2.4.1 Reactor Internals

Reactor internal components include the plenum assembly and the core support assembly. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a turnaround baffle for the outlet flow. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, and thermal shield. Figure 3-59 shows the reactor vessel, reactor vessel internals arrangement, and the reactor coolant flow path. Figure 3-60 shows a cross section through the reactor vessel and Figure 3-61 shows the core flooding arrangement.

Reactor internal components do not include fuel assemblies, CRAs, or incore instrumentation. Fuel assemblies and CRAs are described in Section 3.2.4.2, control rod drives in Section 3.2.4.3, in Section 4.4.5, and incore instrumentation in Section 7.3.3.

The reactor internals are designed to support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain CRA guide tube alignment between fuel assemblies and control rod drives. They also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for incore instrumentation between the reactor vessel lower head and the fuel assemblies, provide support for surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall, and contain the internals vent valves. The vent valves are designed to prevent steam generated within the core from preventing the rapid recovering of the core by coolant following a reactor coolant inlet pipe rupture. All reactor internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the reactor vessel internal surface.

A shop fitup and checkout of the internal components in an as-built reactor vessel mock-up ensure proper alignment of mating parts before shipment. Dummy fuel assemblies and CRAs are used to check fuel assembly clearances and CRA free movement.

To minimize lateral deflection of the lower end of the core support assembly as a result of horizontal seismic loading, integral weld-attached, deflection-limiting guide lugs have been welded on the reactor vessel inside wall. These lugs will also limit the rotation of the lower end of the core support assembly which could result from flow-induced torsional loadings. The lugs allow free vertical movement of the lower end of the internals for thermal expansion throughout all ranges of reactor operating conditions. In the unlikely event that a flange, circumferential weld, or bolted joint might fail, the lugs will limit the possible core drop to one-half inch or less. The elevation plane of these lugs was established near the elevation of the vessel support skirt attachment to minimize dynamic loading effects on the vessel shell or bottom head. A 1/2-inch core drop will not allow the lower end of the CRA rods to disengage from their respective fuel assembly guide tubes even if the CRAs are in the full-out position. In this rod position, approximately 6 $^{1}/_{2}$ inches of rod length remains in the fuel assembly guide tubes. A core drop of 1/2-inch will not result in a significant reactivity change. The core cannot rotate and bind the drive lines because rotation of the core support assembly is prevented by the guide lugs.

The core internals are designed to meet the stress requirements of the ASME Code, Section III during normal operation and transients. Additional criteria and analysis are given in B&W Topical Report BAW-10051, Revision 1, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations." A detailed stress analysis of the internals under accident conditions has been completed and reported in Topical Report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss of Coolant Accident and Maximum Hypothetical Earthquake." This report analyzes the internals in the event of a major LOCA and for the combination of LOCA and seismic loadings. It is shown that although there is some internals deflection, failure of the internals will not occur because the stresses are within established limits. These deflections would not prevent CRA insertion because the control rods are guided throughout their travel and the guide-to-fuel assembly alignment cannot change because positive alignment features are provided between them and the deflections do not exceed allowable values. All core support circumferential weld joints in the internals shells are inspected to the requirements of the ASME Code, Section III.

To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

The effects of asymmetric loading due to a LOCA are analyzed in detail in the B&W Topical Report BAW-1621, "B&W 177 FA Owners Group: Effects of Asymmetric LOCA Loadings - Phase II Analysis," dated July 1980 and in Supplement 1 to this report dated June 1981.

3.2.4.1.1 <u>Plenum Assembly</u>

The plenum assembly is located directly above the reactor core and is removed as a single component before refueling. It consists of a plenum cover, upper grid, CRA guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is constructed of a series of parallel flat plates intersecting to form square lattices and has a perforated top plate and an integral flange at its periphery. The cover assembly is attached to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRA guide tubes. Lifting lugs are provided for remote handling of

the plenum assembly. These lifting lugs are welded to the cover grid. The CRA guide tubes are welded to the plenum cover top plate and bolted to the upper grid. CRA guide assemblies provide CRA guidance, protect the CRA from the effects of coolant cross-flow, and provide structural attachment of the grid assembly to the plenum cover.

Each CRA guide assembly consists of an outer tube housing, a mounting flange, 12 perforated slotted tubes, and four sets of tube segments which are oriented and attached to a series of castings so as to provide continuous guidance for the CRA full stroke travel. The outer tube housing is welded to a mounting flange which is bolted to the upper grid. Design clearances in the guide tube accommodate misalignment between the CRA guide tubes and the fuel assemblies. Final design clearances were established by tolerance studies and Control Rod Drive Line (CRDL) facility prototype test results. The test results are described in Section 3.3.3.4.

The plenum cylinder consists of a large cylindrical section with flanges on both ends to connect the cylinder to the plenum cover and the upper grid. Holes in the plenum cylinder provide a flow path for the coolant water. The upper grid consists of a perforated plate which locates the lower end of the individual CRA guide tube assembly relative to the upper end of a corresponding fuel assembly.

The grid is bolted to the plenum cylinder lower flange. Locating keyways in the plenum assembly cover flange engage the reactor vessel flange locating keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly is guided by the inside surface of the lower flange of the core support shield.

3.2.4.1.2 Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid assembly, flow distributor, thermal shield, incore instrument guide tubes, and internals vent valves. Static loads from the assembled components and fuel assemblies and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads are all carried by the core support assembly.

The core support assembly components are described as follows:

A. Core Support Shield

The core support shield is a flanged cylinder which mates with the reactor vessel opening. The forged top flange rests on a circumferential ledge in the reactor vessel closure flange. The core support shield lower flange is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum assembly relative to the core support shield. The cylinder wall has two nozzle openings for coolant flow. These openings are formed by two forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or internals. Eight vent valve mounting rings are welded in the cylinder wall. Internals vent valves are installed in the core support shield cylinder wall to control steam flow from the core following a postulated cold leg (reactor coolant inlet) pipe rupture as described in Section 3.2.4.1.

B. Core Barrel

The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The core barrel consists of a flanged cylinder, a series of internal horizontal former plates bolted to the cylinder, and a series of vertical baffle plates bolted to the inner surfaces of the horizontal formers to produce an inner wall enclosing the fuel assemblies. The core barrel cylinder is flanged on both ends. The upper flange of the core barrel cylinder is bolted to the mating lower flange of the core support shield and the lower flange is bolted to the lower grid assembly. The upper bolts are locked with crimped locking clips; the lower bolts are locked with welded clips. Coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel. A small portion of the coolant flows upward through the space between the core barrel outer cylinder and the inner baffle plate wall. Coolant pressure in this space is lower than the coolant pressure in the core, thus avoiding tension loads on the bolts attaching the plates to the horizontal formers.

C. Lower Grid Assembly

The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two grid structures, separated by short tubular columns, and surrounded by a forged flanged cylinder. The upper structure is a perforated plate while the lower structure consists of a machined forging. The top flange of the forged cylinder is bolted to the lower flange of the core barrel.

A perforated flat plate located midway between the two grid structures aids in distributing coolant flow prior to entrance into the core. Alignment between fuel assemblies and incore instruments is provided by pads bolted to the upper perforated plate.

D. Flow Distributor

The flow distributor is a perforated dished head with an external flange which is bolted to the bottom flange of the lower grid. The flow distributor supports the incore instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

E. Thermal Shield

A cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and reactor vessel inner wall. The thermal shield reduces the incident gamma absorption internal heat generation in the reactor vessel wall and thereby reduces the resulting thermal stresses. The thermal shield upper end is restrained against inward and outward vibratory motion by restraints bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk-fit on the lower grid flange and secured by 96 high strength bolts.

F. Surveillance Specimen Holder Tubes

The design of the reactor vessel internals allows for the installation of Surveillance Specimen Holder Tubes (SSHT) which contain the surveillance specimen assemblies. The holder tubes were initially installed in the reactor vessel but were removed in June 1976 due to damage sustained in operation. The design change to remove the specimen holder tubes involved removal of the damaged tubes from the core barrel, boring of the SSHT access holes, and roll-expansion of the SSHT shroud tubes over their topmost 1½ inches.

The reactor internals are designed to accept the installation of the holder tubes on the core support assembly. When installed, the tubes extend from the top flange of the core support shield down toward the lower end of the thermal shield. The holder tube has a 3¾ inch offset to place the centerline of the specimens approximately 2¼ inches from the vessel inside wall. Babcock and Wilcox Topical Report BAW-10006, Revision 2, "Reactor Vessel Material Surveillance Program," describes the holder tubes and specimen capsules in detail.

The Toledo Edison Company, Davis Besse Unit 1 reactor serves as the host reactor for the irradiation of ANO-1 reactor vessel surveillance capsules as part of the Babcock and Wilcox company integrated reactor vessel surveillance program for B&W plants.

G. Incore Instrument Guide Tube Assembly

The incore instrument guide tube assemblies guide the incore instrument assemblies from the instrument penetrations in the reactor vessel bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel instrument penetrations and the instrument guide tubes in the flow distributor to accommodate misalignment. Fifty-two incore instrument guide tubes are provided and are designed so they will not be affected by the core drop described in Section 3.2.4.1.

H. Internals Vent Valves

Internals vent valves are installed in the core support shield to prevent a pressure imbalance which might interfere with core cooling following a postulated inlet pipe rupture. Under all normal operating conditions, the vent valve will be closed. In the unlikely event of a pipe rupture in the cold leg of the reactor loop, the valve will open to permit steam generated in the core to flow directly to the leak and will permit the core to be rapidly recovered and adequately cooled after emergency core coolant has been supplied to the reactor vessel. The design of the internals vent valve is shown in Figure 3-62.

Each valve assembly consists of a hinged disc, valve body with sealing surfaces, split-retaining ring, and fasteners. Each valve assembly is installed into a machined mounting ring integrally welded in the core support shield wall. The mounting ring contains the necessary features to retain and seal the perimeter of the valve assembly. Also, the mounting ring includes an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly will be remotely handled as a unit for removal or installation. Valve component parts, including the disc, are of captured-design to minimize the possibility of loss of parts to the coolant system and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote inspection of disc function. Vent valve materials are listed in Table 3-21.

The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, anti-galling characteristics, and compatibility with mating materials in the reactor coolant environment.

The arrangement consists of eight 14-inch inside diameter vent valve assemblies installed in the cylindrical wall of the internals core support shield (refer to Figure 3-59). The valve centers are co-planar and are 42 inches above the plane of the reactor vessel coolant nozzle centers. In cross section, the valves are spaced around the circumference of the core support shield wall.

The hinge assembly consists of a shaft, two valve body journal receptacles, two valve disc journal receptacles, and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters and between the journal outside diameters and their receptacles. The hinge assembly is shown and the clearance gaps are identified in Figure 3-63. The bushing clearances are listed in Table 3-22.

The valve disc hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield.

The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of disc free-motion in service. In the event that one rotational clearance should bind in service, seven loose rotational clearances would remain to allow unhampered disc free-motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc free-motion.

In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact loads to the hinge parts as a result of a LOCA.

3.2.4.2 <u>Core Components</u>

The complete core has 177 fuel assemblies arranged in a square lattice to approximate the shape of a cylinder. All fuel assemblies are identical in mechanical construction and mechanically interchangeable in any core location. Each fuel assembly will accept any control assembly. The reactivity of the core is controlled by 60 CRAs and eight Axial Power Shaping Rod Assemblies (APSRA). APSRAs are identical in physical configuration to the CRAs but have different absorber material or length. There are two types of APSRs. The first type is the 'black' APSR which has the same absorber material as the CRA, but only in the lower portion of the rod. The 'black' APSRs were replaced in Cycle 9 with 'gray' APSRs. The 'gray' APSRs have a longer absorber of Inconel. In addition, the absorber section is pressurized with high purity helium in order to reduce differential pressure stresses in the cladding. Burnable poison rod assemblies are installed in some fuel assemblies which do not contain an APSR or a CRA. The burnable poison rod assemblies (BPRA) assure a negative moderator temperature coefficient through core lifetime. In Cycle 7, the length of the BPRA stack was shortened by 9 inches. The part of the stack which was removed was replaced by a tubular spacer of Zircaloy-4. In Cycle 8 the length of the BPRA stack was increased by 4½ inches and the Zircaloy spacer was shortened by the

same amount. In Cycle 9 the BPRA design was switched from the Mk-B4 to the Mk-B5 design in order to be compatible with the Mk-B5 (Mk-B6) fuel assembly design. The mechanical and geometric configuration of the CRAs, APSRs and BPRAs permit full interchange in any fuel assembly.

The ANO-1 reactor originally had 61 control rod assemblies. During the ANO-1 seventh refueling outage (1R7), the center CRA was removed to accommodate Inadequate Core Cooling (ICC) instrumentation. The center CRA was part of safety group 2. After removal of the center CRA, safety group 2 contained 8 CRAs, making the total number of CRAs 60. Due to potential interference with the support structure of the instrumentation, no CRA or BPRA inserts are allowed in the center core location.

An alternate CRA was introduced in Cycle 22 to extend the life of the control rods when operating in high fluence locations. These alternate CRAs take the place of any control rod in the reactor. The alternate CRAs will have a 3-piece Ag-In-Cd absorber which will minimize the swelling due to irradiation. The operation and worth of the alternate CRAs will not impact the performance of the control rods.

3.2.4.2.1 <u>Fuel Assemblies</u>

During Cycle 20, Mark-B-HTP fuel assemblies were introduced. See Figure 3-66C for a depiction of the basic Mark-B-HTP assembly. The Mark-B-HTP fuel assembly is fully described in BAW-2449(P).

3.2.4.2.1.1 <u>Fuel Assembly Description</u>

A. General

The fuel is sintered, low-enriched, uranium dioxide, cylindrical pellets. The pellets are clad in Zircaloy-4 tubing and sealed by Zircaloy-4 end caps welded at each end. The clad, fuel pellets, end caps, and fuel support components form a "fuel rod." Two hundred and eight fuel rods, 16 control rod guide tubes, one instrumentation tube assembly, seven segmented spacer sleeves, eight spacer grids, and two end fittings make up the basic "Fuel Assembly" (Figures 3-66A and 3-66B). The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly is reserved for instrumentation. Control rod guide tubes are located in 16 locations of the array. Fuel assembly components, materials, dimensions and irradiation conditions are listed in Table 3-23.

B. Fuel Rod

All fuel rods are internally prepressurized with helium in accordance with the procedure discussed in Section 3.3.3.3.4 and in accordance with the B&W specification for helium concentration. The fuel is in the form of sintered and ground pellets of low enriched uranium dioxide. Pellet ends are dished to minimize differential thermal expansion between the fuel and cladding. Radial growth of the fuel during burnup is accommodated by pellet porosity, radial clearance between the pellets and the cladding, and by a small amount (less than one percent) of permanent strain in the cladding. Fuel growth is calculated by the method given in Reference 25.

Above the fuel column is a spring spacer that separates the fuel from the fuel rod upper end cap. This spacer maintains the fuel column in place during shipping and handling. In operation, the spacer permits axial differential growth and thermal expansion between the fuel and the clad. This spacer also provides radial fuel rod cladding support.

Below each fuel column is a spring spacer which axially locates the bottom of the fuel column and separates the fuel from the lower fuel rod end cap. This spring is designed to deflect under high column loads to reduce axial strain in the cladding.

Tubular metal spacers of Zircaloy-4 are located between the fuel pellets and the spring spacers to thermally insulate and separate fuel pellets from spring spacers.

Fission gas release from the fuel is vented to voids within the pellets, the radial gap between the pellets and the cladding, and to the void spaces at top and bottom ends of the fuel rods.

C. Fuel Assembly

1. General

The two fuel assembly types are shown in Figures 3-66(a) and 3-66(b). These are the Mk-B4 and the Mk-B5Z reconstitutable fuel assembly designs. The Mk-B5 reconstitutable design with Zirc grids (Mk-B6) will be used in Cycle 9 and after. In both designs the eight spacer grids, end fittings, and the guide tubes form the basic structure. Fuel rods are supported at each spacer grid by contact points integral with the walls of the cell boundary. In the Mk-B4 fuel assembly design, the guide tubes are permanently attached to the upper and the lower end fittings. The Mk-B5 reconstitutable design allows for removal of the upper end fitting, so the guide tubes are not permanently attached to the upper end fitting. Use of similar material (Zircaloy-4) in the guide tubes and fuel rods result in minimum differential thermal expansion and irradiation growth. In Cycle 8, the guide tube material was changed from Cold-Worked to Annealed Zircaloy-4.

2. Spacer Grids

There are two types of spacer grids. These are the Inconel spacer grids and the Zirc spacer grids. The Zirc spacer grids will be used with the Mk-B5Z fuel assembly starting with Cycle 9.

Both spacer grids are constructed from strips which are slotted and fitted together in "egg crate" fashion. Each grid has 32 strips, 16 perpendicular to 16, which form the 15 x 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the wall of each square opening are integrally punched in the strips. On each of the two end spacer grids, the peripheral strip is extended and rigidly attached to the respective end fitting. The end spacer grids are all made of Inconel while the six intermediate spacer grids are made of Zircaloy-4 for the most recent design or Inconel for the older design. For the Mk-B6 design, the top end spacer grid is identical in dimensions to an intermediate spacer grid but is constructed of Inconel. The Mk-B6 top end spacer grid is positioned from the upper end fitting by 16 stainless steel sleeves.

3. Lower End Fitting

The lower end fittings position the assembly in the lower core grid plate. The lower ends of the fuel rods rest on the grid of the lower end fitting. Penetrations are machined in the lower end fitting to provide a means for attaching the control rod guide tubes and to provide access to the instrument tube. In Cycle 8 the design of the lower end fitting was changed to provide an anti-straddle feature. The anti-straddle end fitting prevents mispositioning the fuel assembly in the lower grid plate during refueling operations.

4. Upper End Fitting

The upper end fittings are the major design change between the Mk-B4 and the Mk-B6 fuel assemblies. Both end fittings position the upper end of the fuel assembly in the upper core grid plate structure and provide a means for coupling the handling equipment. An identifying number on each upper end fitting provides positive identification. The Mk-B6 upper end fitting is removable to provide for reconstitution of the fuel assembly.

Integral with each upper end fitting is a holddown spring and spider to provide a positive holddown margin to oppose hydraulic forces.

Penetrations in the upper end fitting grid are provided for the guide tubes.

In early 1980, a number of holddown springs which had failed from stress corrosion cracking were discovered at the Davis-Besse 1 plant. The failed springs were all fabricated from the same two heats of metal, which had an uncommonly large surface grain structure. B&W performed an analysis of the failure mechanism and a safety evaluation of core operation with broken holddown springs. Recommended remedial measures included increased inspections, replacement of defective material, and an increase in frequency of control rod exercising from once each two weeks to once per week. Holddown spring failures were also found at the Crystal River 3 and Oconee plants.

Following issuance of the B&W recommendations, ANO-1 operating procedures were modified to include exercising control rods once per week and following any occurrence during which the average primary system temperature is lowered below 480 °F. During the Cycle 5 reload shutdown in the first quarter of 1981, inspection of ANO-1 holddown springs revealed one defective spring, which was replaced. Frequency of control rod exercising was subsequently restored to once every two weeks.

5. Guide Tubes

The Zircaloy guide tubes provide continuous guidance to the control rods when inserted in the fuel assembly during operation and provide the structural continuity for the fuel assembly. The Mk-B4 fuel assembly design has flanged and threaded sleeves welded to each end of a guide tube which secure the guide tubes to each end fitting by lock welded nuts. The Mk-B5 reconstitutable design also contains flanged and threaded sleeves, but these guide tube ends pass through holes in the end fitting allowing for guide tube nuts and locking devices to be placed on the

threaded sleeves to secure the end fitting. Transverse location of the guide tubes is provided by the spacer grids. Wear characteristics of the control rods on the guide tubes are analyzed and discussed in Babcock and Wilcox Topical Report BAW-1623, "Control Rod Guide Tube Wear Measurement Program."

6. Instrumentation Tube

   This Zircaloy tube serves as a channel to guide, position, and contain the incore instrumentation within the fuel assembly. The instrumentation probe is guided up through the lower end fitting to the desired core elevation. It is retained axially at the lower end fitting by a retainer sleeve.

7. Spacer Sleeves

   The spacer tube segments fit around the instrument tube between spacer grids and prevent axial movement of the spacer grids during primary coolant flow through the fuel assembly.

3.2.4.2.1.2 Fuel Assembly Evaluation

A. General

   The basis for the design of the fuel rod is discussed in Section 3.1.2.4. Materials testing and actual operation in reactor service with Zircaloy cladding have demonstrated that Zircaloy-4 material has sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

B. Fuel assembly stress and deflection due to LOCA and seismic excitation.

   An analysis of fuel assembly stress and deflection due to LOCA and seismic excitation is based on an investigation of four separate loading conditions. The loading conditions are:

   1. the Operating Basis Earthquake (OBE) - a ground acceleration of 0.10g acting horizontally and 0.067g acting vertically and occurring simultaneously;

   2. the Design Basis Earthquake (DBE) - a ground acceleration of 0.20g acting horizontally and 0.133g acting vertically and occurring simultaneously;

   3. a Loss-of-Coolant Accident (LOCA); and,

   4. the simultaneous occurrence of a DBE and a LOCA timed so that the combined deflections are the maximum. LOCA loads are most severe for an outlet pipe rupture. Loads for an inlet pipe rupture are less severe because of the higher flow resistance of the smaller pipe and better equalization of pressures permitted by the internals vent valves. Based on the test discussed in Section 3.3, it is concluded that the fuel assemblies can withstand a LOCA, the combined effects of a DBE and LOCA, a DBE, and an OBE without exceeding the respective allowable limits.

      The results of the analysis of the effects of asymmetric loadings due to a LOCA are summarized in Section 3.1.2.4.1.

Details of the current horizontal analyses for the combined seismic and LOCA loads are presented in BAW-2292P. These analyses bound ANO-1 with Mark BZ fuel as described in BAW-2027. B&W Owner's Group reports BAW-1847, Rev. 1, and BAW-1889P, eliminate dynamic effects of large primary loop piping ruptures from the structural design basis (Leak-Before-Break). As such, the reactor vessel internal component, more specifically the fuel assemblies, may be analyzed for ruptures in RCS branch connections utilizing the same methods as in BAW-1621.

C. Clad Stress and Strain

The cladding of fuel rods is subjected to external hydrostatic pressure, gradually increasing internal pressure, thermal stresses, vibration, and to the effects of differential expansion of the fuel and cladding caused by thermal expansions and by fuel growth due to irradiation effects. In addition, the properties of the cladding are influenced by thermal and irradiation effects. The analysis of these effects is discussed below.

Stress analysis for cladding is based on several conservative assumptions that make the actual margins of safety greater than those calculated. For example, it is assumed that the clad with the thinnest wall, the smallest fuel-clad gap, and the greatest ovality permitted by the specification is operating in the region of the core where performance requirements are most severe. Fission gas release rates, fuel growth, and changes in mechanical properties with irradiation are based on a conservative evaluation of currently available data.

D. Pressure Effects

1. Beginning of Life Power Conditions

Clad stresses due to external and internal pressure are considerably below the yield strength. Circumferential stresses due to external pressure, calculated using those combinations of clad dimensions, ovality, and eccentricity that produce the highest stress, are shown in Table 3-24. The maximum compressive stress in the expansion void at the system design pressure is the sum of compressive membrane stress plus compressive bending stress due to ovality at the clad OD. Stress conditions are listed for BOL.

In the heat producing zone, the stress and temperature is such that the clad material may creep enough to allow an increase in clad ovality until further creep is restrained by support from the fuel. If fuel-clad contact occurs, the clad is subject to cyclic stresses and strains which are a result of power and pressure transients. To minimize clad fatigue damage, all fuel rods will be internally pressurized with helium. Fatigue analyses, based on conservative assumptions, show that the design limits previously specified (Section 3.1.2.4.2) are met for pressurized fuel rods.

2. End of Life Power Conditions

At the EOL, fission gas pressure does not exceed operating pressure (Section 3.2.3.2.3), however an internal pressure of 3,300 psi has been selected as the design basis. At this pressure the differential would result in a circumferential tensile stress at normal operating pressure. This stress value shown in Table 3-24 is

about one-fourth of the yield strength and, therefore, is not a potential source of short-time burst. The possibility of stress-rupture burst has been investigated using finite-difference methods to estimate the long-time effects of the increasing design pressure on the clad. The predicted pressure-time relationship produces stresses that are less than one-third of the stress levels that would produce stress rupture at the EOL. Outpile stress-rupture data were used but the greater than 3:1 margin on stress is more than enough to account for decreased stress rupture strength due to irradiation.

3. End of Life Shutdown Conditions

The primary coolant system can be cooled down at a maximum rate of 100 °F/hr. During this cooldown period it is desirable to maintain a compressive load on the fuel clad until the clad has been cooled to at least 425 °F where all significant hydrides have had a chance to precipitate under a favorable stress field. The pressurizer and primary system will be cooled down at rates producing a net compressive load on the clad until the clad has reached 425 °F. Coolant conditions and stress levels are shown for two stages of the normal cooldown cycle to the desired temperature level in Table 3-24.

4. Fuel Burnup, Temperature, and Gas Release Conditions

The total production of fission gas and maximum internal clad pressure is based on the analysis of fuel rod power and burnup histories resulting from fuel depletion and fuel cycling. The fission gas release is based on temperature versus release fraction as shown in Figure 3-49. Fuel temperatures are calculated for small radial and axial increments. The total fission gas release is calculated by integrating the incremental releases.

Fuel burnup, temperature and gas release conditions are determined by evaluating the following factors for the most conservative conditions:

a. Gas conductivity at EOL with fission gas present.

b. Influence of the pellet-to-clad radial gap and contact heat transfer coefficient on fuel temperature and release rate.

c. Unrestrained radial and axial thermal growth of the fuel pellets relative to the clad.

d. Hot rod local peaking factors.

e. Radial distribution of fission gas production in the fuel pellets.

f. The fuel temperatures used to determine fission gas release and internal gas pressure have been calculated at the reactor rated power and maximum design overpower condition. Fuel temperature, total free gas volume, fission gas release, and internal gas pressure have been evaluated for a range of initial diametral clearances. This evaluation shows that the highest internal pressure results when the maximum design diametral gap is assumed because of the resulting high average fuel temperature (Figure 3-53). The release rate increases rapidly with an increase in fuel temperature and

unrestrained axial growth reduces the relatively cold gas end plenum volumes. A conservative thermal expansion model is used to calculate fuel temperatures as a function of initial cold diametral clearance as outlined in Section 3.2.3.2.3.G.1.

g. Nuclear calculations of power peaks and power burnup histories are considered in evaluating fuel and cladding performance.

E. Collapse Margins

Short-time collapse tests have demonstrated a clad collapsing pressure in excess of 4,000 psi at the expansion void maximum temperature. The collapse pressure margin is approximately 2.2. Extrapolation to hot spot average clad temperature (733 °F) indicates a collapse pressure of 3,500 psi and a margin of 1.9, which exceeds requirements. Outpile creep collapse tests have demonstrated that the clad meets the long-time (creep collapse) requirement. However, backup radial support has been provided in the upper end void to assure clad dimensional stability in the event that in-pile creep rates are sufficiently high to allow creep collapse of unsupported cladding. Test results summarized in Section 3.3.3.3.1 show the end void spacers are capable of providing backup support. The results of the test show that creep collapse of the bottom end void will not occur since the clad temperature is about 90 °F lower than that in the upper void region. The spacer in the bottom end void is therefore not required to provide radial support. Its geometry, however, is similar to the upper spacer and it therefore provides added assurance of clad dimensional stability at the bottom void region.

After Cycle 2, the computer code CROV was used to predict the creep collapse life of the fuel pin. The current methods are outlined in References 5 and 6, the TACO2 and CROV topicals. In all cases the projected creep collapse life of the fuel exceeds the in-core exposure.

F. Fuel Irradiation Growth and Fuel-Clad Differential Thermal Expansion

The results of tests and the operation of Zircaloy-clad UO_2 fuel rods indicate that the rods can be safely operated to the point where total permanent strain is 1½ percent or higher in the temperature range applicable to PWR cladding (see Reference 3). The design allowable strain is about one percent (Section 3.1.2.4.2.C). Fuel rod operating conditions pertinent to fuel swelling considerations are listed in Table 3-23.

The capability of Zircaloy-clad UO_2 fuel in solid rod form to perform satisfactorily in service has been demonstrated through operation of the SA-1 assembly in the Dresden and Shippingport cores, and through results of their supplementary development programs, up to approximately 45,000 MWd/mtU.

As outlined below, existing experimental information supports the various individual design parameters and operating conditions up to and perhaps beyond the maximum design burnup of 55,000 MWd/mtU, but not in a single experiment. However, the proprietary B&W High Burnup Irradiation Program does combine the primary items of concern in a single experiment.

1. Application of Experimental Data to Design Adequacy of the Clad-Fuel Initial Gap to Accommodate Clad-Fuel Differential Thermal Expansion

    a. Experimental Work

    Six rabbit capsules, each containing three Zr-2 clad rods of 5-inch fuel length, were irradiated in the Westinghouse Test Reactor (see Reference 28) at power levels up to 24 kW/ft. The 94 percent Theoretical Density (TD) UO_2 pellets (0.430 OD) had initial clad-fuel diametral gaps of 6, 12, and 25 mils. No dimensional changes were observed. Central melting occurred at 24 kW/ft only in the rods that had the 25-mil initial gap.

    Two additional capsules were tested (see Reference 29). The specimens were similar to those described above except for length and initial gap. Initial gaps of 2, 6, and 12 mils were used in each capsule. In the A-2 capsule, three 38-inch-long rods were irradiated to 3,450 MWd/mtU at 19 kW/ft maximum. In the A-4 capsule, four 6-inch-long rods were irradiated to 6,250 MWd/mtU at 22.2 kW/ft maximum. No central melting occurred in any rod but diameter increases up to 3 mils in the A-2 capsule and up to 1.5 mils in the A-4 capsule were found in the rods with the 2-mil initial gap.

    b. Application

    In addition to demonstrating the adequacy of Zircaloy-clad, UO_2 pellet rods to operate successfully at the power levels of interest (and without central melting), these experiments demonstrate that the design initial clad-fuel gap of 4.5 to 8.5 mils is adequate to prevent unacceptable clad diameter increase due to differential thermal expansion between the clad and the fuel at BOL. A maximum local diametral increase of less than 0.001 inch is indicated for fuel rods having the minimum initial gap, operating at the maximum overpower condition.

2. Adequacy of the Available Voids to Accommodate Differential Expansion of Clad and Fuel Including the Effects of Fuel Swelling.

    a. Experimental Work

    Zircaloy-clad, UO_2 pellet-type rods have performed successfully in the Shippingport reactor up to approximately 40,000 MWd/mtU. Bettis Atomic Power Laboratory (see Reference 25) has irradiated plate-type UO_2 fuel (96 to 98 percent TD) up to 127,000 MWd/mtU and at fuel center temperatures between 1,300 °F and 3,800 °F. This work indicates fuel swelling rates of 0.16 percent $\Delta V/10$ (see Reference 18) $f/^3cm$ until fuel internal voids are filled, then 0.7 percent $\Delta V/10$ (see Reference 18) $f/^3cm$ after internal voids are filled. This point of "breakaway" appears to be independent of temperature over the range studied and dependent on clad restraint and the void volume available for collection of fission products. The additional clad restraint and greater fuel plasticity (from higher fuel temperatures) of rod-type elements tend to reduce these swelling effects by providing greater resistance to radial swelling and lower resistance to longitudinal swelling than was present in the plate-type test specimens.

This is confirmed in part by the work of Frost, Bradbury, and Griffiths of Harwell (see Reference 32) in which ¼-inch diameter UO_2 pellets clad in 0.020-inch stainless steel with a 2-mil diametral gap were irradiated to 53,300 MWd/mtU at a fuel center temperature of 3,180 °F without significant dimensional change.

In other testing (see Reference 33), 0.150-inch OD, 82 to 96 percent TD oxide pellets (20 percent Pu, 80 percent U) clad with 0.016-inch stainless steel with 6 to 8 mil diametral gaps have been irradiated to 77,000 MWd/mtU at fuel temperatures high enough to approach central melting without apparent detrimental results. Comparable results were obtained on rods swaged to 75 percent TD and irradiated to 100,000 MWd/mtU.

b. Application

Clad strain due to reactor operating conditions is calculated as follows:

Fuel irradiation induced swelling is determined using an empirical model based on the BAPL data (see Reference 25). The fuel swelling model accounts for the portion of swelling which is accommodated by fuel porosity. Initial external fuel swelling will occur at 0.16 percent ΔV/10 f/^{3}cm (see Reference 18) until the fuel pores are filled. For fuel density of 94 percent of theoretical, this occurs at a burnup of 11.1 x 10 f/^{3}cm (see Reference 18) (45,300 MWD/mtU). After the fuel pores are closed, the fuel will swell at a rate of 0.7 percent ΔV/10 f/^{3}cm (see Reference 18) until the maximum local peak design burnup of 13.5 x 10 f/^{3}cm (see Reference 18) (55,000 MWD/mtU) is reached. The design burnup exceeds, of course, the actual calculated local peak burnup.

The fuel is assumed to swell uniformly in all directions, conservatively neglecting axial plastic flow into the pellet end dishes. For uniform fuel swelling in all directions, the percent increase in diameter is one-third the percent volumetric swelling rate. If the fuel cracks, the crack voids will also be available to accommodate fuel growth. Fuel-clad differential thermal expansion in going from cold conditions to power is calculated as described in 3.2.3.2.3.G.

Studies of clad strain for various fuel-clad gaps indicate that the rod with the minimum gap will experience the greatest clad strain in spite of its improved gap conductivity. Clad permanent strain reaches a maximum at EOL.

The clad plastic strain for maximum fuel density maximum local design burnup (55,000 MWd/mtU) and minimum fuel-clad gap is 0.7 percent. Fuel rods with nominal gaps, nominal density, and average burnup will not grow sufficiently to cause any tensile loop stress or strain in the cladding.

3. Fuel Swelling Studies at B&W

Experimental fuel swelling studies under in-pile conditions simulating large reactor environments were undertaken by B&W. Parameters contributing to swelling are burnup, heating rate, fuel density and grain size, and clad restraint. These parameters were studied systematically by irradiating a series of capsules containing fuel rods.

Test variables are in Table 3-25. Variables include heat rate, burnup, clad thickness, and fuel-to-clad gap. Post-irradiation examination includes investigation of dimensional changes, metallographic examination of fuel and cladding, fission gas release correlations with test conditions, and other related observations.

Preliminary results from this proprietary program were discussed with the AEC Staff in October, 1971. These results, which included rods irradiated to burnups up to 55,000 MWd/mtU, support the design criteria used for ANO-1 fuel rods. As indicated in Section 3.3.3.3.3, the results of this program are essential to the design of advanced cores but not necessary for the safe operation of this plant.

Results from this program will not be placed in the public record or literature and will not be adopted for this application due to its proprietary nature.

G. Effect of Zircaloy Creep

The effect of Zircaloy creep on the amount of fuel rod growth due to fuel swelling has been investigated. Clad creep has the effect of producing a nearly constant total pressure on the clad ID by permitting the clad diameter to increase as the fuel diameter increases. Based on out-of-pile date (see Reference 4), one percent creep will result in 10,000 hours (corresponding approximately to the EOL diametral swelling rate) from a stress of about 22,000 psi at the 733 °F average temperature through the clad at the hot spot. At the start of this higher swelling period (roughly the last one-third of the core life), the reactor coolant system pressure would more or less be balanced by the rod internal pressure so the total pressure to produce the clad stress of 22,000 psi would have to come from the fuel. Contact pressure would be 2,400 psi. At the EOL, the rod internal design pressure exceeds the system pressure by about 1,100 psi, so the clad fuel contact pressure would drop to 1,300 psi. Assuming that irradiation produces a 3:1 increase in creep rates, the clad stress for one percent strain in 10,000 hours would drop to about 15,000 psi. Contact pressures would be 1,800 psi at the beginning of the high swelling period and 700 psi at the EOL. Since the contact pressure was assumed to be 825 psi in calculating the contact coefficient used to determine the fuel pellet thermal expansion, there is only a short period at the very end of life (assuming the 3:1 increase in creep rates due to irradiation) when the pellet is slightly hotter than calculated. The effect of this would be a slight increase in pellet thermal expansion and, therefore, in clad strain.

H. Overall Assembly

1. Assurance of Control Rod Assembly Free Motion

The 0.058-inch diametral clearance between the control rod guide tube and the control rod is provided to cool the control rod and to ensure adequate freedom to insert the control rod. As indicated below, studies have shown that fuel rods will not bow sufficiently to touch the guide tube. Thus, the guide tube will not undergo deformation caused by fuel rod bowing effects. Initial lack of straightness of fuel rod and guide tube, plus other adverse tolerance conditions, conceivably could reduce the 0.088-inch nominal gap between fuel rod and guide tube to a minimum of about 0.038 inch, including amplification of bowing due to axial friction loads from the spacer grid. The extended life control rods will have a 0.057-inch diametral clearance between the control rod guide tube and the control rod. The change will not impact control component drop times.

The maximum expected flux gradient of 1.176 across a fuel rod will produce a temperature difference of 12 °F which will result in a thermal bow of less than 0.002 inch. Under these conditions, for the fuel rod to touch the guide tube, the thermal gradient across the fuel rod diameter would have to be on the order of 300 °F.

The effect of a DNB occurring on the side of a fuel rod adjacent to a guide tube would result in a large temperature difference. In this case, however, investigation has shown that the clad temperature would be so high that insufficient strength would be available to generate a force of sufficient magnitude to cause a significant deflection of the guide tube. In addition, the guide tube would experience an opposing gradient that would resist fuel rod bowing, and its internal cooling would maintain temperatures much lower than those in the fuel rod cladding, thus retaining the guide tube strength.

2. Vibration

The semiempirical expression developed by Burgreen (see Reference 16) was used to calculate the flow-induced vibratory amplitudes for the fuel assembly and fuel rod. The calculated amplitude is less than 0.010 inch for the fuel assembly and less than 0.005 inch for the fuel rod. The fuel rod vibratory amplitude correlates with the measured amplitude obtained from a test on a 3 x 3 fuel rod assembly. In order to substantiate this conservatively calculated amplitude for the fuel assembly, a direct measurement has been obtained for a full size prototype fuel assembly during testing of the assembly in the Control Rod Drive Line (CRDL) facility at the B&W Research Center, Alliance, Ohio. The maximum assembly amplitude determined by measurement was 0.005 inch.

3. Demonstration

In addition to the specific items discussed above, the overall mechanical performance of the fuel assembly and its individual components has been and is continuing to be demonstrated in an extensive experimental program in the CRDL (Section 3.3.3.2).

3.2.4.2.2 Control Rod Assembly

Each of the three control rod assembly designs (Figures 3-67(a), (b), and (c)) has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded. The control rod drive is coupled to the CRA by a bayonet type connection. Full length guidance for the CRA is provided by the control rod guide tube of the upper plenum assembly and by the fuel assembly guide tubes. The CRAs and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke.

Each control rod has a section of neutron absorber material. The absorber material is a round bar of silver-indium-cadmium alloy. The original CRA design was clad in type 304 stainless steel tubing. A water and pressure tight container for the absorber were formed by welding stainless steel end pieces to the cladding. The extended life control rod design is clad in an Inconel tubing with Inconel end pieces welded to it. In addition, the absorber section is pressurized with high purity helium to reduce pressure differential stresses in the cladding. The cladding used in all designs provides the structural strength of the control rods and prevents corrosion of the absorber material. In the original design a flexible tubular spacer was used. The extended life design uses a spring. In all designs the spacers prevent absorber movement during shipping and handling and also permit differential expansion in service. Each control rod has a section of neutron absorber material. The absorber material is silver-indium-cadmium alloy. The alternate extended life control rods have three sections of absorber material.

Principle data pertaining to the CRA designs are shown in Tables 3-27 (a), (b), and (c).

A CRA prototype of the B&W design has been extensively tested at reactor temperature, pressure, and flow conditions in the B&W test loop at their alliance Research Laboratory. For test program description and results, refer to Topical Report BAW-10029, Revision 1, "Control Rod Drive System Test Program."

These control rods are designed to withstand all operating loads including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. The ability of the control rod clad to resist collapse has been established in a test program on cold-worked stainless steel tubing. Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material will not cause excessive stressing or stretching of the clad.

Because of their length and the possible lack of straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads results. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies will not encounter significant frictional resistance to their motion in the guide tubes.

3.2.4.2.3 Axial Power Shaping Rod Assembly (APSRA)

There are two axial power shaping rod designs: a "black" design, and the recently designed "gray" APSR, implemented in Cycle 9. For both designs, each axial power shaping rod assembly (Figures 3-68(a) and (b)) has 16 axial power shaping rods, a stainless steel spider, and a female coupling. The 16 rods are attached to the spider by means of a nut threaded to the upper shank

of each rod. After assembly all nuts are lock welded. The axial power shaping rod drive is coupled to the APSRA by a bayonet connection. The female couplings of the APSRA and CRA have a slight dimensional differences to ensure that each type of assembly can only be coupled to the correct type of drive mechanism. When the APSRA is inserted into the fuel assembly, it is guided by the guide tubes of the fuel assembly. Full length guidance of the APSRA is provided by the control rod guide tube of the upper plenum assembly. At the full out position of the control rod drive stroke, the lower end of the APSRA remains within the fuel assembly guide tubes to maintain the continuity of guidance throughout the rod travel length. The APSRAs are designed to permit maximum conformity with the fuel assembly guide tubes throughout travel.

Each design has a length of neutron absorbing material clad in cold-worked type 304 stainless steel tubing. The 'black' APSR has an absorber of silver-indium-cadmium. The 'gray' APSR has a longer absorber made of Inconel. The tubing used for the cladding provides structural strength and prevents corrosion of the absorber material. The section of cladding which contains the absorber is sealed by welded internal and end plugs as shown in Figures 3-68(a) and (b). In addition, the absorber section of the 'gray' APSR design is pressurized with high purity helium in order to reduce differential pressure stresses in the cladding. On both designs, the section above the absorber is vented to the primary system and is always filled with primary coolant. This reduces differential pressure stresses to a negligible level. Pertinent data on the APSRA designs is shown in Table 3-28(a) and (b).

These axial power shaping rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the axial power shaping rod clad to resist collapse due to the system pressure has been demonstrated by an extensive collapse test program on stainless steel tubing. Neither Ag-In-Cd alloy or Inconel material used as absorbers yields gaseous products under irradiation. Because of this, the internal pressure increase with increasing irradiation is negligible. Swelling of both absorber materials is negligible and will not result in unacceptable cladding strain.

Because of their great length and unavoidable lack of straightness, some slight mechanical interference between axial power shaping rods and the fuel assembly guide tubes must be expected. However, the parts involved are flexible and result in very small friction drag loads. Similarly, thermal distortions of the rods are small because of the low heat generation and adequate cooling. Consequently, the APSRAs will not encounter significant frictional resistance to their motion in the guide tubes.

3.2.4.2.4 Burnable Poison Rod Assembly (BPRA)

Each BPRA design (Figures 3-65) has up to 16 burnable poison rods and a stainless steel spider. In the original design, a coupling mechanism was attached to the spider. The coupling mechanism provided a means for positive coupling between the BPRA and the fuel assembly holddown latch. In Cycle 4 the coupling mechanism was replaced by a retainer assembly. Beginning in Cycle 9 the Mk-B5 BPRA design with a built-in retainer mechanism was used. The BPRA is inserted into the fuel assembly through the upper end fitting.

Each burnable poison rod has a section of sintered Al_2O_2-B_4C pellets which serves as burnable poison. In Cycle 7 the stack length of the pellets was reduced by 9 inches and the missing stack length replaced by a tubular spacer of Zircaloy-4. In Cycle 8 the stack length was increased by 4½ inches and the Zircaloy-4 spacer reduced by the same amount. The burnable poison is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the burnable poison rods.

In addition to their nuclear function, the BPRAs also serve to minimize guide tube bypass coolant flow. Pertinent data on the BPRA are shown in Table 3-29.

The burnable poison rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. To provide positive holddown against lift forces acting on the burnable poison rods and regenerative neutron sources, retainers are positioned over the burnable poison rods. These retainers were installed during the refueling outage prior to Cycle 4 operation. The positive holddown is accomplished by capturing the feet of the load arm assembly of the retainer between the fuel assembly and the upper core plate of the plenum. More complete information is provided in the BPRA Retainer Design Report (BAW-1496).

The ability of the burnable poison rod clad to resist collapse due to the system pressure and internal pressure has been demonstrated by an extensive test program on cold-worked Zircaloy-4 tubing.

3.2.4.2.5 Orifice Rod Assembly (ORA)

Orifice rod assemblies were used in Cycle 1 through Cycle 3. Each orifice rod assembly (Figure 3-64) has 16 orifice rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism provides a means for positive coupling between the ORA and the fuel assembly holddown latch when the orifice rods are inserted into the fuel assembly. The necked down section of the rods permits lateral movement in order to facilitate the installation of the orifice assembly into the guide tubes in the fuel assembly. The ORA serves to limit bypass flow through empty guide tubes. Pertinent data on the ORA is shown in Table 3-30.

3.2.4.2.6 Quality Control Program for Core Components

B&W-NPGD equipment specifications require that core components be fabricated under an approved quality control program. This includes shop quality control provisions, which are approved by B&W-NPGD quality assurance personnel, and special process procedures, which are approved by B&W-NPGD design personnel.

The B&W Commercial Nuclear Fuel Plant manufactures core components under a controlled manufacturing system which includes complementary written process procedures and inspection provisions. These fabrication activities are supported by quality control provisions, e.g. document control, control special processes, in process and final inspection, gauge control, corrective action, etc.

A. The tests and inspections described below are those as specified for the various components and assemblies. Additional testing and inspection are performed routinely to assure that the test and inspection program is adequate and to further assure the quality of the final product.

   1. Fuel Rods

      A program of testing and inspections is performed throughout the manufacturing of fuel rods to assure the integrity of the cladding in the as-fabricated fuel rods. Each tube is 100 percent non-destructively tested for defects using ultrasonic inspection. The inside and outside diameters and the wall thickness of the tubing is inspected for actual dimensions on a 100 percent basis. Also, the finished fuel rods are 100 percent

visually inspected for surface defects. The fuel rod end cap seal welds are 100 percent inspected for surface irregularities using an optical magnification of 6X. The surfaces of all fuel rod components and all completed fuel rods are 100 percent inspected to ensure that the criteria for cleanliness is satisfied. As a final inspection, all rods are helium leak checked in a vacuum.

Destructive testing of production components on a statistical basis is performed to further assure the integrity of the cladding. Specimens from each lot of tubing are subjected to a 14-day corrosion test after which the hydrogen content for each sample is determined. Samples of end cap to clad welds are also subjected to a corrosion test. An additional program testing on end cap weld samples is performed to determine the presence of cracks, voids, inclusions, porosity, lack of penetration, or other weld imperfections when metallurgically examined.

2. Fuel Pellets

The characteristics of fuel pellets are determined on a statistical basis through a program of tests and inspections described below.

Random samples from each lot of pellets are chemically analyzed. Certified reports are recorded with the following information for each lot of pellets:

Analytical test results including weight percent uranium, weight percent ^{235}U, O/U ratio, moisture, absorbed gas, impurities, and total equivalent boron content;

Variables data from the sampling plans for the average diameter, pellet weight, pellet density, and surface finish;

Weight of each stack and stack length;

Pellet symbol and stack identification records;

Uranium accountability documentation as required by applicable government regulations;

Identification of lots of isotopically blended uranium along with the measured enrichments, the enrichment tolerances, and the quantity of material involved.

In addition, B&W overchecks incoming material to assure that all requirements of the specifications are met.

3. Spacer Grids

Fuel assembly spacer grids are tested and inspected under a statistical program to assure that the dimensions of the spacer grids and the fuel assembly are acceptable.

Welds are inspected at 6X magnification to assure a 95/95 confidence level that the specifications for dimensional size and location and the absence of cracks, lack of penetration, or other imperfections are met.

The spacer grids are also 100 percent inspected for outside dimensions and flatness.

4. Fuel Assembly

The completed fuel assemblies are inspected on a statistical basis to assure a 95/95 confidence level that the assemblies are dimensionally correct. In addition, the fuel assemblies are subjected to an envelope inspection on a 100 percent basis. The assemblies are also 100 percent inspected to ensure that the critical specified end fitting dimensions, i.e. surfaces mating with the reactor internals, are correct. The guide tube assemblies of each completed fuel assembly are inspected full length by an inspection fixture which simulates a control rod assembly.

Fuel assemblies are dimensionally inspected on a statistical basis for fuel rod, guide tube, and instrumentation tube pitch to assure acceptable water channel dimensions in the final assembly.

Welds in the final assembly are inspected for imperfections using a 6X magnification. In addition, metallographic examination of welds is performed on a statistical basis.

Completed fuel assemblies are inspected to assure a 95/95 confidence level that the cleanliness specifications are met.

5. Control and Axial Power Shaping Assemblies

Control and axial power shaping assemblies and the constituent components are tested and inspected throughout manufacture to assure the integrity of the assembly and the absorber rods.

The control rod and axial power shaping rod cladding is inspected for inside and outside diameters and the minimum wall thickness on a 100 percent basis. The tubing is also nondestructively tested for defects on a 100 percent basis using ultrasonic inspection. The tubing is also 100 percent inspected for compliance with the cleanliness specification.

The finished control rod and axial power shaping rod closure welds are 100 percent vacuum leak tested and visually inspected at 6X magnification. The control rods and axial power shaping rods are also 100 percent inspected for surface defects.

The completed control rod and axial power shaping rod assemblies are dimensionally inspected on a statistical basis to assure the dimensional specifications are met. The control rod and axial power shaping rod spiders are tested on a 100 percent basis to assure proper alignment and engagement with the respective drive couplings. Each completed control rod and axial power shaping rod assembly is inspected full length in a specially designed fixture, which simulates the fuel assembly guide tubes.

6. Burnable Poison Assembly

The rods in the burnable poison assembly are tested and inspected throughout manufacture to assure the integrity of the as-fabricated rods.

Each tube is 100 percent nondestructively tested for defects using ultrasonic inspection. The inside and outside diameters and the wall thickness of the tubing are inspected for actual dimensions on a 100 percent basis. Also, the finished burnable

poison rods are 100 percent visually inspected for surface defects. The end plug closure welds are 100 percent helium leak tested. The finished rods are 100 percent visually inspected for surface defects.

Destructive testing on production burnable poison rod components on a statistical basis is performed to further assure the integrity of the cladding. Specimens from each lot of tubing are subjected to a 14-day corrosion test after which the hydrogen content for each sample is determined.

7. Program of Inspection After Shipment

   A visual inspection of new control rods, fuel assemblies, and their shipping containers is performed at the reactor site.

B. B&W continually reviews the fuel design to incorporate the latest design information from outside sources and company sponsored research and development programs. This is accomplished by a coordinated effort between the fuel design and manufacturing departments.

C. The B&W Quality Assurance Audit/Review Program includes the following steps to assure the quality and integrity of the fuel:

   1. All standard specifications for the manufacture of fuel assemblies and control components at QA are reviewed.

   2. A system audit is periodically performed at B&W's Commercial Nuclear Fuel Plant to its adherence to procurement documents released by NPGD Fuel Contracts.

   3. A QA review is made on all variation notices and concurrence requests for deviations of fuel components to the specifications and drawings.

   4. The data sheets for each fuel assembly are QA reviewed.

   5. A QA internal audit is periodically performed on the Fuel Engineering Section relative to its adherence to Customer Requirements.

D. Hydriding is a process which simultaneously occurs with oxidation of Zircaloy in PWR coolant. Hydriding cannot be prevented; however, steps can be taken to lessen the possibility of hydriding from sources other than the reactor coolant.

   The following steps are taken to minimize Zircaloy hydriding and thus assure the integrity of the clad.

   1. Fuel Pellets

      a. The moisture content is limited.

      b. The total absorbed and adsorbed gases are limited.

      c. The surface cleanliness of the finished pellets is specified to eliminate contamination by hydrogenous material which could contribute to hydriding.

d. The fuel pellet vendor is required to sample for moisture in accordance with a variables sampling plan.

e. B&W has procedures and a sampling plan to overcheck the fuel vendors for conformance with the specifications.

2. Cladding

a. The texture, i.e. orientation of the poles of the crystals, is specified to provide desired mechanical characteristics and to lessen the effects of precipitated hydride platelets.

b. The surface cleanliness of the tubing is protected by packaging that eliminates contaminants which could cause hydriding during in-reactor service.

3. Fuel Rod Fabrication

The fuel pellets are tested for moisture. Should a test show a moisture level higher than that specified, the lot of pellets is dried until the specified moisture content is achieved.

The moisture level in ceramic spacers used in the fuel rod is also controlled.

Internal and external surfaces of the Zircaloy tubing are cleaned just prior to fuel pellet loading.

The fuel rod loading is performed in a clean room which has temperature and humidity control.

Once the rods are loaded, there is a minimal time interval until the rods are capped and seal welded. The rods are pressurized with helium which is limited on moisture and total hydrocarbon. The rods are 100 percent leak checked to assure the integrity of the cladding and the welds.

4. Operation

The pressure and temperature conditions during cool down are specified to align the hydride platelets circumferentially as hydrogen is precipitated at lower temperatures.

E. The important design features which influence the effects of fuel-clad interaction are discussed below:

1. Diametral Gap Between Clad and Pellet

This gap provides space to accommodate thermal expansion, irradiation growth, and cracking of the pellet which occur throughout the life of the fuel rod. Failures have seldom occurred in cladding that had diametral gaps of one percent of the pellet diameter. This corresponds to approximately 4 mils in the B&W design.

2. Clad Physical Properties

   The B&W cladding specification requires a minimum uniform elongation while maintaining a minimum yield strength at operating temperature. Zircaloy quickly loses ductility and saturates during the first year of fuel assembly exposure. The higher the pre-irradiation ductility, the higher will be the ductility after saturation.

   When the power in a fuel rod is rapidly increased to a higher level, as in the Halden experiments, the thermal expansion of the fuel pellet will, under some conditions, locally strain the cladding beyond its elastic limit. This is the time when the extra margin of ductility reduces the failure probability. The rods which failed in the Halden experiments exhibited considerably less elongation than the minimum specified for B&W cladding.

3. Fuel Pellet Geometry

   One of the features of the B&W fuel pellet geometry is the chamferring of the sharp corners at the end of the pellet. This, along with dished ends, tends to minimize hourglassing and pellet-clad interaction.

4. Clad Thickness

   Clad thickness is an important design parameter affecting fuel rod integrity. Thicker clad is beneficial in two ways. First, it retards creep collapse of the clad, maintaining an effective diametral gap for a longer period, and, thus, postponing and reducing clad-pellet interaction. Second, when pellet contact does occur, the thicker clad exerts greater force on the hot fuel pellet requiring the pellet to plastically accommodate a larger amount of the dimensional change in the axial rather than diametral direction.

5. Pre-pressurized Fuel Rod

   The B&W standard fuel rod is pressurized with helium during fabrication. During operation, the resultant pressure differential across the clad decreases with increased temperature. The reduced differential pressure significantly decreases the clad creep collapse rate, thus maintaining an effective diametral gap for a longer period of time. The helium pre-pressurization also improves heat transfer between the hot fuel pellet and the clad. This results in a lower fuel pellet temperature for comparable heat flux. The lower fuel temperature produces less thermal expansion of the pellet, reducing pellet-clad interaction.

6. Top and Bottom Fuel Rod Void Regions

   Axial expansion of the fuel stack is accommodated by the void region at the top of the fuel rod. Unlike other designs, the pellets do not rest directly on the lower fuel rod end cap. Instead they are supported on a spring spacer. When fuel becomes lodged with the clad in the hot, central portion of the rod, further axial expansion of the fuel normally results in stretching of the clad. However, in the B&W design, the compression of the lower spring spacer reduces the extent of clad strain.

F. The design limits for fuel rod collapse are defined in Section 3.1.2.4.2, Item D. An evaluation of the conformation of the clad design to meet the criteria is presented in Section 3.2.4.2.1, Item E. Experimental results which support the clad design in resisting collapse are presented in Section 3.3.3.3.1. As indicated above, all fuel rods are prepressurized to lessen the differential pressure across the clad wall and reduce the potential for collapse.

3.2.4.3 Control Rod Drives

3.2.4.3.1 Description

The Control Rod Drive Mechanism (CRDM) positions the control rod within the reactor core and indicates the location of the control rod with respect to the reactor core. The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the shim safety drive mechanism releases the CRA and supporting CRDM components permitting the CRA to move by gravity into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or accident condition. The control rod is decelerated at the end of the rod trip insertion by a snubber assembly in the CRDM upper housing. The snubber assembly supports the control rod in the fully inserted position. Criteria applicable to drive mechanisms for both control shim rod assemblies and axial power shaping rod assemblies are given below. Additional requirements for the mechanisms which actuate only control shim rod assemblies are also given below.

General Design Criteria

A. Single Failure

No single failure shall inhibit the protective action of the control rod drive system. The effect of a single failure will be limited to one CRDM.

B. Uncontrolled Withdrawal

No single failure or sequence of dependent failures will cause uncontrolled withdrawal of any CRA.

C. Equipment Removal

The disconnection of plug-in connectors, modules, and subassemblies from the protective circuits will be annunciated or shall cause a reactor trip.

D. Position Indication

Continuous position indication, as well as an upper and lower position limit indication, shall be provided for each CRDM. The accuracy of the position indicators shall be consistent with the tolerance set by reactor safety analysis.

E. Drive Speed

The control rod drive control system shall provide a single uniform mechanism speed. The drive controls, or mechanism and motor combination, shall have an inherent speed limiting feature. The speed of the mechanism for both insertion and withdrawal is shown in Table 3-31. The withdrawal speed is limited as described in Section 7.2.2.3.2.

F. Mechanical Stops

Each CRDM shall have positive mechanical stops at both ends of the stroke, or travel. The stops shall be capable of receiving the full operating force of the mechanisms without failure.

G. Control Rod Positioning

The control rod drives shall provide for controlled withdrawal or insertion of the control rods out of, or into, the reactor core to establish and hold the power level required.

Additional Design Criteria

The following criterion is applicable only to the mechanisms which actuate control rod assemblies:

CRA Trip

The shim safety drives are capable of rapid insertion or trip for emergency reactor conditions.

3.2.4.3.2 Control Rod Drive Mechanisms

The CRDMs provide for controlled withdrawal or insertion of the control rod assemblies out of or into the core and are capable of rapid insertion or trip. The drive mechanisms are sealed, reluctance motor-driven, screw units. The CRDM data is listed in Table 3-31.

Shim Safety Drive Mechanism

The shim safety drive mechanism consists of a motor tube which houses a torque tube, a leadscrew, its rotor assembly, and a snubber assembly. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

The control rod drive output element is a non-rotating, translating leadscrew coupled to the control rod. The screw is driven by separate anti-friction roller nut assemblies attached to segment arms which are rotated magnetically by a motor stator located outside the pressure boundary. Current impressed on the stator causes the separating roller nut assembly halves to close and engage the leadscrew. Mechanical springs disengage the roller nut halves from the screw in the absence of a current. For rapid insertion, the nut halves separate to release the screw and control rod, which move into the core by gravity. A hydraulic snubber assembly within the upper housing decelerates the moving CRA to a low speed a short distance above the CRA full-in position. The final CRA deceleration energy is absorbed by the down-stop buffer spring. The CRDM is a totally sealed unit with the roller nut assemblies and segment arms magnetically driven by the stator coil through the motor tube pressure housing wall. The leadscrew assembly is connected to the control rod by a bayonet type coupling. An anti-rotation device (torque taker) prevents rotation of

the leadscrew while the drive is in service. A closure and vent assembly is provided at the top of the motor tube housing to permit access to couple and release the leadscrew assembly from the control rod. The top end of the leadscrew assembly is guided by the torque taker assembly. Two of the six phase stator housing windings are energized to maintain the control rod position when the drive is in the holding mode.

The CRDM is shown in Figures 3-69 and 3-70. Subassemblies of the CRDM are described as follows:

A. Motor Tube

The motor tube is a 3-piece welded assembly designed and manufactured in accordance with the requirements of the ASME Code, Section III, for Class A nuclear pressure vessel. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding is performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. The motor tube wall between the rotor assembly and the stator is constructed of magnetic material to present a small air gap to the motor. This region of the motor tube is martensitic stainless steel. The upper end of the motor tube functions only as a pressurized enclosure for the withdrawn leadscrew and is made of stainless steel transition-welded to the upper end of the martensitic steel motor section. The lower end of the low alloy steel tube section is welded to a stainless steel machined forging which is flanged at the face which contacts the vessel control rod nozzle. Double gaskets, which are separated by a ported test annulus, seal the flanged connection between the motor tube and the reactor vessel.

B. Motor

The motor is a synchronous reluctance unit with a slip-on stator. The rotor assembly is described in Item F. The stator is a 48-slot, 4-pole arrangement with water cooling coils in the outside casing. The stator is encapsulated after winding to establish a sealed unit. It is 6-phase star-connected for operation in a pulse-stepping mode and advances 15 mechanical degrees per step. The stator assembly is mounted over the motor tube housing as shown in Figure 3-69.

C. Vent Valve

The upper end of the motor tube is closed by a closure insert assembly containing a vapor bleed port and vent valve. The vent valve and closure insert have double seals. The insert closure is retained by a closure nut which is threaded to the inside of the motor tube. The sealing load for the closure insert is applied by a hydraulically preloaded spring washer that is retained by the closure nut.

D. Actuator

The actuator consists of the translating leadscrew, leadscrew nut assembly, and the torque taker assembly on the screw. The actuator leadscrew travel is 139 inches.

E. Leadscrew

The leadscrew has a lead of 0.750 inch. The thread is double lead with a single pitch spacing of 0.375 inch. Thread lead error is held to 0.0005 inch maximum in any six inches for uniform loading with the roller nut assemblies. The thread form is a modified ASME with a flank angle that allows the roller nut to disengage without lifting the screw.

F. Rotor Assembly

The rotor assembly consists of a ball bearing supported rotor tube carrying and limiting the travel of a pair of segment arms. Each of the two arms carry a pair of ball bearing supported roller (nut) assemblies which are skewed at the leadscrew helix angle for engagement with the leadscrew. The current in the motor stator (two of a six winding stator) causes the arms that are pivoted in the rotor tube to move radially toward the motor tube wall to the limit provided, thereby engaging the four roller nuts with the centrally located leadscrew. Also, four separating springs mounted in the segment arms keep the rollers disengaged when the power is removed from the stator coils. A second radial bearing mounted to the upper end of the rotor tube has its outer race pinned to both segment arms, thereby synchronizing their motion during engagement and disengagement. When a six phase rotating magnetic field is applied to the motor stator, the resulting force produces rotor assembly rotation.

G. Torque Tube and Torque Taker

The torque tube is a separate tubular assembly containing a key that extends the full length of the leadscrew travel. The tube assembly is secured in elevation and against rotation at the lower end of the closure assembly by a retaining ring, keys, and the insert closure. The lower end of the torque tube houses the snubber assembly and is the down stop. The leadscrew contacts the insert closure assembly for the upper mechanical stop.

The torque taker assembly consists of the position indicator permanent magnet, the snubber piston and a positioning keyway. The torque taker assembly is attached to the top of the leadscrew and has a keyway that mates with the key in the torque tube to provide both radial and tangential positioning of the leadscrew.

H. Snubber Assembly

The total snubber assembly is composed of a piston that is the lower end of the torque taker assembly and a snubber cylinder and belleville spring assembly which is attached to the lower end of the torque tube. The snubber cylinder is closed at the bottom by the snubber bushing and leadscrew. The snubber cylinder has a 12-inch active length in which the free-fall tripped leadscrew and CRA is decelerated without applying greater than 10 times gravitational force on the control rod. The damping characteristics of the snubber is determined by the size and position of a number of holes in the snubber cylinder wall and the leakage at the snubber piston and bushing. Leakage reduction at the snubber piston and bushing can only be reduced to a minimum amount caused by practical operating clearances. Therefore, at the end of the snubbing stroke, there is kinetic energy from a five foot per second impact velocity that is absorbed by the belleville spring assembly by a slight instantaneous overtravel past the normal down stop.

I. Leadscrew Guide

The leadscrew guide bushing acts as a primary thermal barrier and as a guide for the screw shaft. As a primary thermal barrier, the bushing allows only a small path for free convection of water between the mechanism and the closure head nozzle. Fluid temperature in the mechanism is largely governed by the flow of water up and down through this bushing. The diametral clearance between screw shaft and bushing is large enough to preclude jamming the screw shaft and small enough to hold the free convection

to an acceptable value. In order to obtain trip travel times of acceptably small values, it is necessary to provide an auxiliary flow path around the guide bushing. The larger area path is necessary to reduce the pressure differential required to drive water into the mechanism to equal the screw displacement. The auxiliary flow paths are closed for small pressure differentials (several inches of water) by ball check valves which prevent the convection flow but open fully during trip.

J. Position Indications

Two methods of position indication are provided; 1) an absolute position indicator and 2) other, a relative position indicator. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the leadscrew extension comes in close proximity. As the leadscrew (and the CRA) moves, switches operate sequentially producing an analog voltage proportional to position. The position indicator assemblies have been replaced with a new R4C type position indicator tube. The resolution of these position indicators under normal conditions is ± 2.5 inches. The R4C will allow continued operation of the position indicator system with a faulty reed switch. Resolution with one channel of the R4C isolated is ± 3.5 inches. For Group 8 only, there is an additional $+^{3}/_{5}$" bias or offset error which affects signal accuracy of the R4C assembly. Also, for Group 8, a one rod misalignment of up to an additional $^{5}/_{8}$" applies. Additional reed switches are included in the same tube with the absolute position transducer to provide full withdrawal and insertion signals. The relative position transducer is a small pulse-stepping motor driven from the power supply for the rod drive motor. This small motor is coupled to a potentiometer with an output signal accuracy of ± 0.97 inch, producing a readout with an accuracy of ± 2.4 inches.

K. Motor Tube Design Criteria

The motor tube design complies with Section III of the ASME Boiler and Pressure Vessel Code for a Class A vessel. The operating transient cycles, which are considered for the stress analysis of the reactor pressure vessel, are also considered in the motor tube design.

Quality standards relative to material selection, fabrication, and inspection are specified to insure safety function of the housings essential to accident prevention. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. These design and fabrication procedures establish quality assurance of the assemblies to contain the reactor coolant safely at operating temperature and pressure.

In the highly unlikely event that a pressure barrier component or the control rod drive assembly does fail catastrophically, i.e. rupture completely, the following results would ensue:

1. Control Rod Drive Nozzle

   The assembly would be ejected upward as a missile until it was stopped by the missile shield over the reactor. This upward motion would have no adverse effect on adjacent assemblies.

2. Motor Tube

   The failure of this component anywhere above the lower flange would result in a missile like ejection into the missile shielding over the reactor. This upward motion would have no adverse effect on adjacent mechanisms.

Axial Power Shaping Rod Drive Mechanism

For actuating the partial-length control rods, which maintain their set position during a reactor trip of the shim safety drive, the CRDM is modified so that the roller nut assembly will not disengage from the leadscrew on a loss of power to the stator.

3.3 TESTS AND INSPECTIONS

3.3.1 NUCLEAR TESTS AND INSPECTION

3.3.1.1 Critical Experiments

An experimental program (see References 20 & 22) to verify the relative reactivity worth of the CRA has been completed. Detailed testing established the worth of the CRA under various conditions similar to those for the reference core. These parameters include control rod arrangement in a CRA, fuel enrichments, fuel element geometry, CRA materials, and soluble boron concentration in the moderator.

Gross and local power peaking were also studied and three-dimensional power peaking data were taken as a function of CRA insertion. Detailed peaking data were also taken between fuel assemblies and around the water holes left by withdrawn CRA. The experimental data have been analyzed and were used to benchmark the analytical models used in the design.

3.3.1.2 Zero Power, Approach to Power, and Power Testing

Boron worth and CRA worth (including stuck-CRA worth) will be determined by physics tests at the beginning of each core cycle. The boron worth and CRA worth at a given time in core life will be based on CRA position indication and calculated data as adjusted by operating data.

The reactor coolant will be analyzed in the laboratory periodically to determine the boron concentration and the reactivity held in boron will then be calculated from the concentration and the reactivity worth of boron.

The method of maintaining the hot standby, > 525 °F, shutdown margin (hence stuck-CRA margin) is related to operational characteristics (load patterns) and to the power peaking restrictions on CRA patterns at power. The CRA pattern restrictions will insure that sufficient reactivity is always fully withdrawn to provide adequate shutdown with the stuck-CRA margin. Power peaking as related to CRA patterns and shutdown margin will be predicted by calculations.

Operation under power conditions will normally be monitored by incore instrumentation and the resulting data will be analyzed.

3.3.2 THERMAL AND HYDRAULIC TESTS AND INSPECTION

3.3.2.1 Reactor Vessel Flow Distribution and Pressure Drop Test

A 1/6-scale model of the reactor vessel and internals has been tested to evaluate:

A. The flow distribution to each fuel assembly of the reactor core and to develop any necessary modifications to produce the desired flow distribution.

B. Fluid mixing between the vessel inlet nozzle and the core inlet and between the inlet and outlet of the core.

C. The overall pressure drop between the vessel inlet and outlet nozzles and the pressure drop between various points in the reactor vessel flow circuit.

D. The internals vent valves for closing behavior and for the effect on core flow with valves in the open position.

The reactor vessel, flow baffle, and core barrel were made of clear plastic to allow use of visual flow study techniques. All parts of the model except the core were geometrically similar to those in the production reactor. However, the simulated core was designed to maintain dynamic similarity between the model and production reactor.

Each of the 177 simulated fuel assemblies contained a calibrated flow nozzle. The test loop was capable of supplying cold water (80 °F) to three inlet nozzles and hot water (180 °F) to the fourth. Temperature was measured in the inlet and outlet nozzles of the reactor model and at the inlet and outlet of each of the fuel assemblies. Static pressure taps were located at suitable points along the flow path through the vessel. This instrumentation provided the data necessary to accomplish the objectives set forth for the tests. The results of the test were reported in B&W Topical Report BAW-10037, Revision 2, "Reactor Vessel Model Flow Test."

3.3.3 FUEL ASSEMBLY, CONTROL ROD ASSEMBLY, AND CONTROL ROD DRIVE MECHANICAL TESTS AND INSPECTION

To demonstrate the mechanical adequacy and safety of the fuel assembly, CRA, and control rod drive, a number of functional tests have been performed.

3.3.3.1 Prototype Testing

A full-scale prototype fuel assembly, CRA, and control rod drive have been tested in the Control Rod Drive Line (CRDL) facility located at the B&W Research Center, Alliance, Ohio. This full-sized loop is capable of simulating reactor environmental conditions of pressure, temperature, and coolant flow. To verify the mechanical design, operating compatibility, and characteristics of the entire control rod drive fuel assembly system, the drive was stroked and tripped approximately 200 percent of the expected operating life requirements.

A portion of the testing was performed with maximum misalignment conditions. Equipment was available to record and verify data such as fuel assembly pressure drop, vibration characteristics, and hydraulic forces and to demonstrate control rod drive operation and verify scram times. All prototype components were examined periodically for signs of material fretting, wear, and vibration/fatigue to insure that the mechanical design of the equipment met reactor operating requirements. Test results are given in B&W Topical Report BAW-10029, Revision 1, "Control Rod Drive System Test Program."

3.3.3.2 Model Testing

Many functional improvements have been incorporated in the design of the fuel assembly as a result of model tests. For example, the spacer grid to fuel rod contact area was fabricated to 10 times reactor size and tested in a loop simulating the coolant flow Reynolds number of interest. Thus, visually, the shape of the fuel rod support areas was optimized with respect to minimizing the severity of flow vortices and pressure drop. A 9-rod (3 x 3) assembly using stainless steel spacer grid material has been tested at reactor conditions (640 °F, 2,200 psi, 13 fps coolant flow) for 210 days. Two full-sized canned fuel assemblies with stainless steel spacer grids have been tested at reactor conditions, one for 40 days and the other for 22 days. A prototype canless fuel assembly using Inconel 718 spacer grids has been tested for approximately 90 days, approximately half of that time at reactor conditions. The principal objectives of these tests were to evaluate fuel assembly and fuel rod vibration and/or fretting wear resulting from flow-induced vibration. Vibratory amplitudes have been found to be very small and, with the exception of a few isolated instances which are attributed to pretest spacer grid damage, no unacceptable wear has been observed.

3.3.3.3 Component and/or Material Testing

3.3.3.3.1 Fuel Rod Cladding

Extensive short-time collapse testing was performed on Zircaloy-4 tube specimens as part of the B&W overall creep-collapse testing program. Initial test specimens were 0.436 inch OD with wall thicknesses of 0.020 inch, 0.024 inch, and 0.028 inch. Ten 8-inch-long specimens of each thickness were individually tested at 680 °F at slowly increasing pressure until collapse occurred. Collapse pressures for the 0.020-inch wall thickness specimens ranged from 1,800 to 2,200 psig, the 0.024-inch specimens ranged from 2,800 to 3,200 psig, and the 0.028-inch specimens ranged from 4,500 to 4,900 psig. The material yield strength of these specimens ranged from 65,000 to 72,000 psi at room temperature and was 35,900 psi at 680 °F.

Additional Zircaloy-4 short-time collapse specimens were prepared from material with a yield strength of 78,000 psi at room temperature and 48,500 psi at 615 °F. Fifteen specimens having an OD of 0.410 inch and an ID of 0.365 inch (0.0225-inch nominal wall thickness) were tested at 615 °F at increasing pressure until collapse occurred. Collapse pressures ranged from 4,470 to 4,960 psig.

Creep-collapse testing was performed on the 0.436-inch OD specimens. Twelve specimens of 0.024-inch wall thickness and 30 specimens of 0.028-inch wall thickness were tested in a single autoclave at 680 °F and 2,050 psig. During this test, two 0.024-inch wall specimens collapsed during the first 30 days and two collapsed between 30 and 60 days. None of the 0.028-inch wall specimens had collapsed after 60 days. Creep-collapse testing was then performed on 30 0.410-inch OD by 0.365-inch ID (0.0225-inch nominal wall) specimens for 60 days at 615 °F and 2,140 psig. None of these specimens collapsed and there were no significant increases in ovality after 60 days.

The results of the 60-day creep-collapse testing on the 0.410-inch OD specimens showed no indication of incipient collapse. The 60-day period for creep-collapse testing was used since it exceeds the point of primary creep of the material yet is sufficiently long to enter the stage when fuel rod pressure begins to build up during reactor operation, i.e. past the point of maximum differential pressure that the cladding would be subjected to in the reactor.

These tests were followed by additional creep-collapse tests in which 60 specimens of variable wall thickness were subjected to a pressure of 2,085 psi at 685 °F until collapse occurred. The cladding wall thickness was 0.0285, 0.0263, 0.0251, and 0.0240 inches. The cladding thickness included the range of tolerances for production cladding and the pressure represented the fuel rod maximum pressure differential at operating conditions. The temperature was selected to conservatively approximate in-pile creep rates. It was found that the 0.024-inch wall specimens collapsed in less than a month, and several 0.0263-inch wall specimens collapsed in less than three months. In view of the unknown increase of in-pile creep rates as compared with out-of pile creep rates, it was decided to provide backup support for the cladding in the upper end void region where cladding temperatures of 650 °F occur in hot channels.

A spring spacer has been designed as a backup spacer. The spacer provides radial support to the cladding without causing excessive axial restraint to the fuel expansion. Analyses have been performed indicating that the spacer can withstand the shipping acceleration of the fuel pellets without permanent deformation. Testing has been performed to demonstrate that the spring spacer will provide backup support to the cladding. The spacers were enclosed in

production Zircaloy cladding and subjected to 2,500 psi at 750 °F. This represents the design system external pressure for the cladding and the nominal operating temperature of the spacer. Post-autoclave examinations revealed that the cladding was adequately supported.

3.3.3.3.2 Fuel Assembly Structural Components

The structural characteristics of the fuel assemblies which are pertinent to loadings resulting from normal operation, handling, earthquake, and accident conditions are investigated experimentally in test facilities such as the CRDL facility. Structural characteristics such as natural frequency and damping are determined at the relatively high (up to approximately 0.300 inch) amplitude of interest in the seismic, LOCA, and asymmetric LOCA loadings analyses. Natural frequencies and amplitudes resulting from flow-induced vibration are measured at various temperatures and flow velocities up to reactor operating conditions.

3.3.3.3.2.1 Analysis of Fuel Assembly Stress and Deflection Due to LOCA and Seismic Excitation

A discussion of analyses of the fuel assembly for loads caused by the depressurization transient following an instantaneous reactor coolant pipe rupture and/or seismic excitation is provided in this section.

The following sections reflect historical analyses to the Mark-B (Inconel Intermediate Grids) fuel assemblies. Assumptions, methods, and models are described in Topical Report BAW-10035, Revision 1. The LOCA forcing functions are taken from Topical Report BAW-10008, Revision 1. The loading conditions investigated for the Mark-B fuel assemblies are an Operating Basis Earthquake (OBE), Design Basis Earthquake (DBE), LOCA, and a simultaneous occurrence of a DBE and a LOCA timed so that the combined deflections are at a maximum.

Similar assumptions, methods, models, and results applicable to the Mark-BZ (Zircaloy Intermediate Grids) and Mark-B-HTP (M5 spacer grids) fuel assemblies are provided in Topical Report BAW-10133PA, Rev. 1, Topical Report BAW-1781P, Rev. 0, Topical Report BAW-2292P, Rev. 0, Topical Report BAW-10227PA, Rev. 0, and/or Topical Report BAW-2449(P), Rev. 1. The effects of asymmetric LOCA loadings are considered for the vertical faulted analyses and taken from B&W Topical Report BAW-1621, "Effects of Asymmetric LOCA Loadings. - Phase II Analysis," July 1980. The effects of leak-before-break LOCA loadings are considered for the horizontal faulted analyses and are provided in Topical Report BAW-2292P, "Mark-B Fuel Assembly Spacer Grid Deformation in B&W Designed 177 Fuel Assembly Plants."

On investigation of the effects of the loadings, two areas can be identified for closer study:

A. Horizontal - contact between fuel assemblies due to motions in a horizontal plane, where the contact occurs primarily at mid-span grid spacers.

B. Vertical - contact of fuel assemblies with the internals due to upward pressure, where the contact occurs between the end fittings and the grid plates.

Seismic accelerations used in the analysis are those specified for the Arkansas Nuclear One site. Seismic effects are evaluated by using the time-history method. The calculation of LOCA forces is provided from Topical Report BAW-10008, Part 1, Revision 1.

All safety margins in this analysis are calculated as follows:

$$\text{Margin} = \frac{[\text{Allowable (load) - Applied (load)}] \text{ x } 100\%}{\text{Applied (load)}}$$

Since allowable loads are based on criteria described in Section 3.1.2.4.2, the margins quoted herein are in excess of those required. Any positive margin, including zero, is acceptable. A zero margin indicates that the criterion is just met.

3.3.3.3.2.1.1 <u>Loads</u>

3.3.3.3.2.1.1.1 <u>Vertical Loads on Core During LOCA</u>

The total force acting on a single fuel assembly for the outlet rupture is given in Figure 3-71. Figure 3-71 is a combination of Figure 6 (ΔP across core for a 36-inch-ID outlet break) and Figure 10 (shear force on core for a 36-inch-ID outlet break) from Topical Report BAW-10008, Part 1, Revision 1. It is found in the following manner:

$$\text{Fig. } 3-71 = \frac{(\text{Fig } 6)\ (\text{blocked area of core}) + (\text{Fig. } 10) - (\text{weight of core})}{177 \text{ fuel assemblies}}$$

This combined pressure and fluid friction force is sufficient to cause the fuel assemblies to lift off the lower grid and contact the upper grid. They deflect the upper grid, causing axial loads in the control rod guide assemblies and subsequent deflection of the plenum cover beams. The resisting force from the plenum cover stops the fuel assemblies and causes them to return to the lower grid.

3.3.3.3.2.1.1.2 <u>Horizontal Thrust Force During LOCA</u>

The LOCA thrust force acting at the vessel's outlet nozzle is analyzed using the FLASH computer code and the relationship

Thrust = pressure x area.

Testing associated with the LOFT program tends to confirm that the horizontal thrust can be calculated by this relationship. The FLASH program is used to correlate the vessel pressure and, therefore, the thrust for some of the semi-scale blowdown tests.

The results for a 36-inch outlet pipe rupture are shown in Figure 3-72.

3.3.3.3.2.1.1.3 <u>Seismic Excitation</u>

The specific time history used in this analysis is that supplied by the Bechtel Corporation for the elevation of the reactor vessel support pad. The method of analysis used to obtain the time history, based upon the 1940 N-S EL Centro Earthquake, is described in Chapter 5.

3.3.3.3.2.1.2 <u>Models Used in Analysis</u>

3.3.3.3.2.1.2.1 <u>Horizontal Contact Analysis</u>

Structurally, the fuel assemblies are long, slender beams which are responsive to horizontal excitations. Because of the proximity of the assemblies, these motions could result in midspan contact. The concern is that such contacts could produce unacceptable damage to the spacer

grids and thus reduce coolant flow or restrict control rod motion. Two possible forms of horizontal excitation are seismic and LOCA. Seismic excitation occurs at the vessel's foundation and the LOCA produces a thrust force at the nozzle. The vertical component of the earthquake is considered with the horizontal analysis. However, because of the vertical stiffness of the reactor internals, the seismic contribution to the displacement of the core is negligible (about 0.002 to 0.0003 inch) with respect to the horizontal motion.

Overall Model

Both of these excitations can produce horizontal motion to the fuel assemblies so that a dynamic model including all the components involved (the reactor vessel, the control rod drives, the internals, and the fuel assemblies) is needed.

First Segment

This overall model is divided into two segments. The first segment includes all components named above except individual fuel assemblies and involves recording the motions of the upper and lower grid plates, the core support shield, and earth velocity versus time. The motions are input excitations for the second segment of the overall model.

The first step in the solution is to determine a model that accurately represents the structure being investigated. The more masses used, the more accurate (but also the more complex) the solution. The investigation of different models shows that for these horizontal contacts a 9-mass model (Figure 3-73) was sufficient to describe the motions of the components.

The method used in this dynamic model was the far coupled, "lumped mass" approach, which may be considered as the vibration equivalent of the finite element technique in static stress analysis problems. The distributed mass of components of the structure is considered to be concentrated at discrete points. These mass points are connected by massless flexible elements. The behavior of the total structure is then determined from the response of these mass points. No damping is included.

Once the model is fixed, a set of simultaneous equations is written for the structure. For a system of N masses, the equations in matrix form are:

$$\underset{NxN}{[M]}\ \underset{Nx1}{\{X\}} + \underset{NxN}{[K]}\ \underset{Nx1}{\{X\}} = \underset{Nx1}{\{F\}}$$

where:

[M] = mass matrix,

{X} = acceleration matrix,

[K] = stiffness matrix,

{X} = displacement matrix,

{F} = force matrix.

This model is then programmed for a digital computer. A modification is made to invert the final flexibility matrix generated and thus obtain a stiffness matrix. Each row of the stiffness matrix is then divided by the mass corresponding to that row to obtain a "K/M" matrix. These values are then substituted into scaled equations and solved on the analog computer.

To validate the analog representation of the model, initial displacement tests are performed. The digital program generates frequencies and mode shapes. The nine-mass model on the analog is displaced into a particular mode shape and then allowed to vibrate. The frequency and mode shape of vibration from the analog compared well with the digital results.

Excitation of First Segment

As shown in Figure 3-73 the seismic excitation is applied at the base of the reactor vessel and the results are recorded. These time-history records are compared with the published spectrum.

The simultaneous occurrence of the DBE and LOCA is also recorded. The seismic excitation is applied at the vessel's skirt, as described above, and the LOCA thrust force is applied at the nozzle. Owing to the relative timing of these two events, maximum fuel assembly displacement is obtained. This investigation indicates that maximum displacement gives maximum contacts and hence maximum loads.

Second Segment

The second segment the model comprises five fuel assemblies, two core baffles, and associated circuitry. Each fuel assembly is modeled by far-coupling techniques with three lumped masses. Each mass has an individual damper and an elastic-plastic spring that represents the transverse structural properties of the grid. The core baffle is represented by an elastic spring. The clearances between assemblies and between the core baffle and outside assemblies are also established during the program. The elastic-plastic properties of the spacer grid are determined by test and used as program input. The frequency and damping properties of the fuel assembly are established by test and are used as program input.

Because of the input excitation, contacts occur between adjacent assemblies or between assemblies and the core baffle. The elastic-plastic grid spring allows a maximum value of force to be exerted and any remaining motion of the assembly creates permanent deformation of the grid. The program considers the energy loss involved as well as the change in cross-sectional size of the impacting grid. These effects influence the event as it occurs and also influences the succeeding contacts. The program sums the total grid deformation from all contacts during the seismic history and presents this value as program output. These results are compared with the general design criteria as well as the specific criterion.

3.3.3.3.2.1.2.2 Vertical Contact Analysis

This analysis is conducted to determine the loads acting on the various parts of the fuel assembly as a result of vertical contact with the upper grid assembly during a LOCA. The fuel assembly, when subjected to the upward pressure force caused by the instantaneous rupture of a primary outlet pipe, can suddenly accelerate vertically toward the upper grid assembly. When the fuel assembly does contact the upper grid assembly, the fuel rods tend to slip in the upper end spacer grid. Since the stiffness of the end spacer grid assembly is substantially greater than the axial stiffness of the guide tubes, once slippage occurs a compressive load is applied to the guide tubes by way of the lower end grid end fitting assembly. Since any dynamic buckling of the guide tubes during a LOCA could prevent control rod insertion, investigation of the loads applied to the guide tubes is the primary concern in this analysis.

Loads on the guide tube result from both the slip loads of the fuel rods on the end spacer grid and the intermediate spacer grids on the guide tubes. These loads, along with the end spacer grid buckling load, were determined by tests described below.

The mathematical model of the fuel assembly is shown in Figure 3-74. By appropriately combining springs in series and parallel, this model is simplified to a single-degree-of freedom system, i.e. a single mass and spring combination. However, the single spring reflects the nonlinear characteristics of the overall structural system. A typical load-deflection curve for this spring, based on data obtained from production fuel assemblies, is shown in Figure 3-75. It should be noted that this spring curve is for the beginning of life (BOL). Its shape changes considerably as a function of full-power operational time because of irradiation growth, other irradiation effects, and material yield strength variation. These effects are considered in the analysis. The forcing function acting on the model is based on the LOCA pressure curves given in Topical Report BAW-10008, Part 1, Revision 1 and is shown in Figure 3-71. Section 1a gives details of its determination.

To account for the nonlinear spring characteristics of the fuel assembly model and also for the rapidly changing forcing function, a digital program was developed. Utilizing a numerical integration routine based on the linear acceleration assumption, the program calculates the three dynamic parameters of interest (displacement, velocity, and acceleration) at every quarter-millisecond. A logic system monitors the displacement and adjusted the spring rate to reflect the nonlinearities of the spring. Calculation stops when a negative velocity, indicating that the fuel assembly is moving in a downward direction, is encountered.

From the digital output, the maximum displacement of the fuel assembly is used to enter the spring load-deflection curve and read directly the total fuel assembly load. This load is used to determine the maximum guide tube load.

3.3.3.3.2.1.3 <u>Tests Conducted</u>

3.3.3.3.2.1.3.1 <u>Frequency and Damping Tests</u>

The fuel assembly frequency and damping values are established from several test programs in which full-sized test specimens were used. Tests are performed in air, in still water at temperatures up to 200 °F, and in still and flowing water at reactor operating conditions (650 °F and 2,200 psi). Both displacement loading (pluck tests) and steady-state sinusoidal excitation are used.

This extensive testing confirms that the natural frequency of the assembly is in the low frequency range and provides the damping values for use in the analysis. Both frequency and damping are also dependent on the amplitude of vibration. This dependence is due to fuel rod slippage in the spacer grids, which is the prime source of the damping values. The tests also establish that damping increases with the coolant flow velocity owing to the effect of coolant flow on the spacer grids.

3.3.3.3.2.1.3.2 <u>Spacer Grid Compression Tests</u>

The analysis of fuel assemblies during conditions of horizontal acceleration and contact require knowledge of certain transverse characteristics of the spacer grids. These characteristics are (1) the elastic and plastic load abilities and (2) the amount of permanent deformation and energy that can be absorbed without interfering with control rod motion. This information is obtained by performing compression or crush tests on individual spacer grids.

Each grid is filled with simulated fuel rods and guide tubes and then mounted in a vertical plane in the tensile machine (Figure 3-76). A vertical compressive load is applied while recording the grid deflection and other significant data. During the loading, efforts are continually made to insert poison pin segments into the guide tubes. The load and the grid distortion at which this is no longer possible are recorded. These test results are corrected for temperature effects by applying a ratio of the grid material's (Inconel 718) yield strength at temperature (600 °F) to its yield strength at room temperature. The results of this procedure give the initial elastic load ability for the grids, the load cycling as the horizontal rows of the grid fail, and the permanent distortion of the grid at the time when the poison pins can no longer be inserted.

These results are obtained from essentially static tests. The first two results are used as input data for the horizontal contact analysis and the third is used as the acceptance criterion for grid deformation as if it is obtained dynamically. The results are checked qualitatively against the effects of dynamic loading with a drop test as described below.

3.3.3.3.2.1.3.3 Spacer Grid Drop Test

The results of the compression tests depend to some extent on the mode of failure, which is transverse displacement of individual fuel rod rows. To check this failure mode qualitatively under dynamic loading conditions, a single grid loaded with short lengths of fuel rod is dropped on a solid base so as to land on its side. Only those rows nearest the impact surface are crushed and this result supports the assumption that dynamic loading could decrease with distance from the impacting surface. Therefore, since the analysis of the static crush test assumes that the maximum impact load is applied uniformly over the three rows of spacer grids nearest the impact surface, the test indicates that the analysis is conservative.

3.3.3.3.2.1.3.4 Slip Tests

Tests are used to determine the slip load between the fuel rods and end grid, the slip load between guide tubes and intermediate spacer grids, and the end grid skirt buckling load. The fuel rod slip load is determined by placing Zircaloy dummy fuel rods in an end grid and slowly applying a uniform load to the fuel rods until the rods slip. When the rods became flush to the upper end of the spacer grid, the load is increased until the end spacer grid skirt buckles. The guide tube slip load is determined by taking several measurements of the force required to draw the production guide tube through the production intermediate spacer grid.

3.3.3.3.2.1.4 Results

3.3.3.3.2.1.4.1 Horizontal Contact Analysis

Design Criterion

The level of permanent distortion suffered during the design basis earthquake, LOCA, or DBE plus LOCA must not prevent control rod insertion.

Results and Margins of Safety

The results of the analysis show that the canless fuel assembly meets the general design criteria as well as the specific design criterion stated above. The margins for the three excitation levels are as follows:

Arkansas Nuclear One Unit 1 Time History (EL Centro N-S, 1940)

Excitation	Margin, %
OBE	25
DBE	80
DBE + LOCA	10

For the DBE case, for example, the allowable deformation is 150 mils. The margins are calculated as follows:

Arkansas Nuclear One level = 0.20g

Level to obtain 150 mils = 0.36g

$$\text{Margin} = \frac{0.36 - 0.20}{0.20} = 80\%$$

Conclusion

The reference fuel assembly design can withstand the horizontal contact loads.

3.3.3.3.2.1.4.2 Vertical Contact Analysis

Guide Tube Buckling

Specific Design Criteria

The compressive load in guide tubes should not exceed 85 percent of the static Euler buckling load. The holddown spider will be allowed to yield since it serves no safety function.

Results and Margins of Safety

The test results in Table 3-1a show that the load on the guide tubes from the combination of the fuel rod slip load and the slip load between the intermediate spacer grids and the guide tubes is 4420 lb. The margin of safety is 26 percent.

Conclusion

The fuel assembly design can withstand a vertical LOCA contact.

Upper End Spacer Grid Welds

As shown in Figure 3-78 the spacer grid is formed of strips assembled in egg crate fashion. The top and bottom of each such intersection are tungsten-inert gas welded. During a LOCA, the fuel assemblies contact the upper internals grid. The resultant fuel rod deceleration loads are transmitted through the end spacer grid. The load carried by the grid is limited to the slip load of the fuel rods in the grid.

The bending moment at the middle of a grid strip is calculated on the basis that the grid strip is a simply supported beam subjected to the slip load of the fuel rods. The moment divided by the grid depth is the force carried by the grid welds.

To verify this method of calculating the force on a spacer grid weld, a series of tests is conducted using portions of spacer grids loaded as shown in Figure 3-78. The grid strips are loaded to failure, and the corresponding force on the welds is compared with previous pull-test results for welds loaded only in tension. These tests indicate that the weld strength can be predicted using the analytical methods described.

Specific Design Criteria

The spacer grid welds are capable of supporting a tensile load of 225 pounds at room temperature. The comparable load at temperature (600 °F) is 200 pounds. The stresses for normal LOCA and earthquake loads are limited to 85 percent of ultimate stress, reducing the 200-pound load to an allowable load of 170 pounds.

The force carried by the end spacer grid welds is calculated using the method described above. For the total maximum possible fuel rod slip load, the maximum weld force during a LOCA is 58 pounds. The allowable load is 170 pounds and the margin is 93 percent. The loads due to a LOCA and/or earthquake are not additive to those due to normal operation because the maximum loads are limited by the available friction loads between the end grids and the fuel rods.

Conclusion

The end spacer grid welds can withstand the vertical contact loads.

<u>End Spacer Grid Assembly</u>

This assembly consists of an end spacer grid, a skirt that connects the end spacer grid to the end fitting, and the end fitting. The skirt is formed by extending the 20-mil outside strips of the spacer grid and reinforcing them with a 30-mil doubler. This composite plate is mechanically attached to the end fitting as well as butting against a shoulder on the end fitting.

The end spacer grid is tested to failure at room temperature by applying a uniform load across the grid. To obtain the allowable buckling load, the failure load is reduced to allow for loss of strength at operating temperature and then, for stability, multiplied by 0.85. This load is presented in Table 3-1a as the allowable end spacer grid assembly buckling load.

The skirt and mechanical attachment are tested for a compressive load equal to or greater than the maximum possible slip load of the fuel rods in the end spacer grid due to vertical contact at the BOL.

The assembly is also evaluated for transverse loads. An arbitrary and conservative lateral deflection of one inch from the horizontal contact analysis was used and the moment at the mechanical attachment was determined.

Specific Design Criteria

The skirt must not buckle. The allowable load is 85 percent of the critical buckling load.

Conclusion

The end spacer grid assembly is adequate for the maximum anticipated loads as given in Table 3-1a. Minimum margins are as follows:

Component	Margin, %
Skirt buckling	299
Mechanical attachment	294

3.3.3.3.3 High Burnup Fuel Irradiations

The purpose of the B&W high burnup irradiation program is to determine the swelling rate of UO_2 as a function of burnup using fuel rods of a similar design to the core. In addition to determining the swelling rates, the effect of several other variables, including density, heat rate, and cladding restraint, are being investigated.

The program consists of capsules, some of which will operate at a heat rate of 18 kW/ft and others at heat rate of 21.5 kW/ft. The pellets, other than ^{235}U content, conform to the reactor fuel specifications. The target burnup ranges from 10,000 to 80,000 MWd/mtU with eight capsules exceeding 45,000 MWd/mtU. The specimens are not tested with an external pressure. However, two different cladding thicknesses, 0.015 and 0.025 inch, are used to vary the restraint offered by the cladding. The fuel rods are irradiated with a cladding surface temperature of 650 °F. The diametral gaps between the pellets and cladding vary from 4 - 5 and 7 - 8 mils to give smeared densities of about 92.3 and 90.8 percent respectively. These gaps and smeared densities are consistent with the fuel rod specifications. The insertion date for the first capsule was September 5, 1967. The schedule for the B&W high burnup irradiation program is shown in Table 3-26.

The tests are oriented toward the determination of the behavior of materials in an irradiation environment and to determine the optimum geometric and material properties for the specific application. The information is essential for design of advanced cores, but is not essential for the safe operation of ANO-1. Adequate information is available in the literature, as discussed in Section 3.2.4.2.1. Results of the tests in the program will not be placed in the public record or literature and will not be adopted for this application due to the proprietary nature of the B&W program.

3.3.3.3.4 Fuel Rod Pre-pressurization

Development and testing of the procedure by which B&W pre-pressurizes fuel rods assures that the actual fuel rod internal pressure is within tolerance of the specified pressure. The maximum time for the fuel rod to equalize pressure with the charging chamber is identified in the qualification of the pre-pressurization procedure. A regularly scheduled test is performed to check the time to equalization. Further, the procedure provides for a minimum of 1.5 times the maximum equalization time in the actual production pre-pressurization.

3.3.3.4 Control Rod Drive Tests and Inspection

3.3.3.4.1 Control Rod Drive Developmental Tests

The testing and development program for the roller nut drive has been completed. The prototype drive was tested at the B&W Research Center at Alliance, Ohio. Wear characteristics of critical components have indicated that material compatibility and structural design of these components would be adequate for the design life of the mechanism. The trip time for the mechanism as determined under test conditions of reactor temperature, pressure, and flow was well within the specification requirements. B&W Topical Report BAW-10029, Revision 1, summarizes the results of the test program.

3.3.3.4.2 Production Tests

Production tests discussed in this section have been performed either on the drives installed or on drives manufactured to the same specifications. The finished control rod drives have been proof-tested as a complete system, i.e. mechanisms, motor control, and system control working as a system. This proof-testing is above and beyond any developmental testing performed in the product development stages.

Mechanism production tests include the following:

A. Ambient Tests

   Coupling tests
   Operating speeds
   Position indication
   Trip tests

B. Operational Tests

   Operating speeds
   Position indication
   Partial- and full-stroke cycles
   Partial- and full-stroke trip cycles

3.3.4 INTERNALS TESTS AND INSPECTIONS

The internals upper and lower plenum hydraulic design is evaluated and guided by the results from the 1/6-scale model flow test which is described in Section 3.3.2.1. The test results have guided the design to obtain minimum flow maldistribution and test data allowed verification of vessel flow and pressure drops.

The effects of internal misalignment was evaluated on the basis of the test results from the CRDL tests described in Section 3.3.3.4. These test results, correlated with the internals guide tube design, ensure that the CRA can be inserted at specified rates under conditions of maximum misalignment.

System dynamic analysis methods and procedures used to determine dynamic responses of the reactor internals and Class 1 components of the reactor coolant pressure boundary which have an effect on the internals responses are described in B&W Topical Report BAW-10008, Part 1, Revision 1.

Analysis for flow-induced vibrations of the reactor internals has been performed and the results are presented in B&W Topical Report BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibrations."

To verify the integrity of the reactor internals for flow-induced vibration, a preoperational program in accordance with Safety Guide 20 is utilized. The details of the test program for the prototype internals similar to Arkansas Nuclear One - Unit 1 are discussed in B&W Topical Report BAW-10038, "Vibration Analysis and Preoperational Test Program for 177 Fuel Assembly Prototype Reactor Internals, plus its Supplement, Revision 4.

Internals shop fabrication quality control tests, inspection, procedures, and methods are similar to those for the pressure vessel described in Section 4.3.11.2. The internals surveillance specimen holder tubes and the material irradiation program are described in Section 4.4.5.

A listing is included herewith for all internals nondestructive examinations and inspections with applicable codes or standards applicable to all core structural support material of various forms. In addition, one or more of these examinations is performed on materials or processes which are used for functions other than structural support, i.e. alignment dowels, etc., so that virtually 100 percent of the completed internals materials and parts are included in the listing. Internals raw materials are purchased to ASME Code, Section II or ASTM material specifications. Certified material test reports are obtained and retained to substantiate the material chemical and physical properties. All internals materials are purchased and obtained to a low cobalt limitation. The ASME Code, Section III, as applicable for Class A vessels, is generally specified as the requirement for reference level non-destructive examination and acceptance. In isolated instances when ASME III cannot be applied, the appropriate ASTM Specifications for non-destructive testing are imposed. All welders performing weld operations on internals are qualified in accordance with ASME Code, Section IX, applicable Edition and Addenda. The primary purpose of the following list of non-destructive tests is to locate, define, and determine the size of material defects to allow an evaluation of defect, acceptance, rejection, or repair. Repaired defects are similarly inspected as required by applicable codes.

A. Ultrasonic Examination

   1. Wrought or forged raw material forms are 100 percent inspected throughout the entire material volume to ASME III, Class A.

   2. Personnel conducting these examinations are trained and qualified.

B. Radiographic Examination (includes X-ray or radioactive sources)

   1. Cast raw material forms are 100 percent inspected to ASME III, Class A or ASTM.

   2. All circumferential full penetration structural weld joints which support the core are 100 percent inspected to ASME III, Class A.

   3. All radiographs are reviewed by qualified personnel who are trained in their interpretation.

C. Liquid Penetrant Examination

1. Cast form raw material surfaces are 100 percent inspected to ASME III Class A or ASTM.

2. Full penetration, non-radiographic or partial penetration, structural welds are inspected by examination of root and cover passes to ASME III, Class A.

3. All circumferential full penetration, structural weld joints which support the core have cover passes inspected to ASME III, Class A.

4. Personnel conducting these examinations are trained and qualified.

D. Visual (5X Magnification) Examination

This examination is performed in accordance with and results accepted on the basis of a B&W Quality Control Specification which complies with NAV-SHIPS 250-1500-I. Each entire weld pass and adjacent base metal are inspected prior to the next pass from the root to and including the cover passes.

1. Partial penetration, non-radiographically or non-ultrasonically feasible structural weld joints are 100 percent inspected to the above specification.

2. Partial or full penetration, attachment weld joints for non-structural materials or parts are 100 percent inspected to the above specification.

3. Partial or full penetration weld joints for attachment of mechanical devices which lock and retain structural fasteners are 100 percent inspected to the above specification.

4. Personnel conducting these examinations are trained and qualified.

After completion of shop fabrication, the internals components are shopfitted and assembled to final design requirements. The assembled internals components undergo a final shopfitting and alignment of the internals with the "as-built" dimensions of the reactor vessel. Dummy fuel and CRAs are used to ensure that ample clearances exist between the fuel and internals structures guide tubes to allow free movement of the CRA throughout its full stroke length in various core locations. Fuel assembly mating fit is checked at all core locations. The dummy fuel and CRAs are identical to the production components except that they are manufactured to the most adverse tolerance space envelope and they contain no fissionable or absorber materials.

All internal components can be removed from the reactor vessel to allow inspection of all vessel interior surfaces. Internals components surfaces can be inspected when the internals are removed to the canal underwater storage location.

The internals vent valves were designed to relieve the pressure generated by steaming in the core following a LOCA so that the core will remain sufficiently cooled. The valves were designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe.

To verify the structural adequacy of the valves to withstand the pressure forces and perform the venting function, the following tests were performed:

A. A full-size prototype valve assembly (valve disc retaining mechanism and valve body) was hydrostatically tested to the maximum pressure expected to result during the blowdown.

B. Sufficient tests were conducted at zero pressure to determine the frictional loads in the hinge assembly, the inertia of the valve disc, and the disc rebound resulting from impact of the disc on the seat so that the valve response to cyclic blowdown forces may be determined analytically.

C. A prototype valve assembly was pressurized to determine the pressure differential required to cause the valve disc to begin to open. A determination of the pressure differential required to open the valve disc to its maximum open position was simulated by mechanical means.

D. A prototype valve assembly was successfully installed and removed remotely in a test stand to confirm the adequacy of the vent valve handling tool.

E. A 1/6-scale model valve disc closing force (excluding gravity) test as described in Section 3.3.2.1.

F. The full-size prototype valve's response to vibration was determined experimentally to verify prior analytical results which indicated that the valve disc would not move relative to the body seal face as a result of vibration caused by transmission of core support shield vibrations. The prototype valve was mounted in a test fixture which duplicated the method of valve mounting in the core support shield. The test fixture with valve installed was attached to a vibration test machine and excited sinusoidally through a range of frequencies which encompassed those which may reasonably be anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat was monitored and recorded during test. The test results indicated that there was a relative motion of the valve to its seat for conditions simulating operating conditions. After no relative motion was observed or recorded during test, the valve disc was manually forced open during test to observe its response. The disc closed with impact on its seat, rebounded open, and reseated without any adverse effects to valve seal surfaces, characteristics, or performance. From this oscillograph record, the natural frequency of the valve disc was conservatively calculated as approximately 1,500 Hz; whereas, the range of frequencies for the primary system (including internals components) has been established as 15 to 160 Hz. These frequencies are separated by an ample margin to conclude that no relative motion between the valve disc and its seat will occur during normal reactor operation.

Each production valve will be subjected to Tests B and C above except that no additional analysis will be performed in conjunction with Test B.

The valve disc, hinge shaft, shaft journals (bushings), disc journal receptacles, and valve body journal receptacles have been designed to withstand, without failure, the internal and external differential pressure loadings resulting from a LOCA. These valve materials will be non-destructively tested and accepted in accordance with the ASME Code III requirements for Class A vessels as a reference quality level.

During scheduled refueling outages, after the reactor vessel head and the internals plenum assembly have been removed, the vent valves will be accessible for visual and mechanical inspection. A hook tool will be provided to engage with the valve disc exercise lug described in Section 3.2.4.1.2.H.

With the aid of this tool, the valve disc will be manually exercised to evaluate the disc freedom and the amount of force necessary to hold the valve in its full open position. The hinge design will incorporate special features, as described in Section 3.2.4.1.2.H, to minimize the possibility of valve disc motion impairment during its service life. With the aid of the hook tool, the valve disc can be raised and a remote visual inspection of the valve body and disc sealing faces can be performed for evaluation of observed surface irregularities. In the unlikely event that a hinge part should fail during normal operation, the most significant indication of such a failure would be a change in the disc free-motion as a result of altered rotational clearances.

Remote installation and removal of the vent valve assemblies if, required, is performed with the aid of the vent valve handling tool which includes unlocking and operating features for the retaining ring jackscrews.

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ANP-3366	Arkansas Nuclear One – Unit 1, Cycle 26 Reload Report

ANP-3499	Arkansas Nuclear One – Unit 1, Cycle 27 Reload Report
ANP-3637	Arkansas Nuclear One – Unit 1, Cycle 28 Reload Report
ANP-3792	Arkansas Nuclear One – Unit 1, Cycle 29 Reload Report
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BAW-1391	Arkansas Nuclear One - Unit 1, Fuel Densification Report
BAW-1433, Part 1	Arkansas Nuclear One - Unit 1, Cycle 2 Reload Report
BAW-1471	Arkansas Nuclear One - Unit 1, Cycle 3 Reload Report
BAW-1496	Burnable Poison Rod Assembly Retainer Design Report
BAW-1504	Arkansas Nuclear One - Unit 1, Cycle 4 Reload Report
BAW-1621	Effects of Asymmetric LOCA Loadings - Phase II Analysis
BAW-1623	Control Rod Guide Tube Wear Measurement Program
BAW-1658 Revision 2	Arkansas Nuclear One - Unit 1, Cycle 5 Reload Report
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BAW-2114	Arkansas Nuclear One - Unit 1, Cycle 10 Reload Report
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BAW-2194	Arkansas Nuclear One - Unit 1, Cycle 12 Reload Report
BAW-2232	Arkansas Nuclear One - Unit 1, Cycle 13 Reload Report

BAW-2276	Arkansas Nuclear One - Unit 1, Cycle 14 Reload Report
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BAW-10124	FLAM3 - Three Dimensional Nodal Code for Calculating Core Reactivity and Power Distributors
BAW-10129	LYNX1: Reactor Fuel Assembly Thermal-Hydraulic Analysis Code
BAW-10130-A	LYNX2: Subchannel Thermal-Hydraulic Program
BAW-10141, Revision 1	TACO2 - Fuel Pin Performance Analysis
BAW-10143	BWC Correlation of Critical Heat Flux
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BAW-10152	NOODLE - Multi-dimensional Two Group Reactor Simulator
BAW-10156	LYNXT - Core Transient Thermal Hydraulic Program
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Unless indicated otherwise, data in Tables 3-1 through 3-31 are design data applicable to Cycle 1 operation. For data applicable to the current cycle, see Chapter 3A, Reload Report, and its supporting documentation.

Table 3-1

CORE DESIGN, THERMAL AND HYDRAULIC DATA

Reactor	
Design Heat Output, MWt	2,568
Vessel Coolant Inlet Temperature, °F	554
Vessel Coolant Outlet Temperature, °F	603.8
Core Outlet Coolant Temperature, °F	606.2
Core Operating Pressure, psig	2,185
Core and Fuel Assemblies	
Total No. of Fuel Assemblies in Core	177
No. of Fuel Rods per Fuel Assembly	208
No. of Control Rod Guide Tubes per Assembly	16
No. of In-Core Instr. Positions per Fuel Assembly	1
Fuel Rod Outside Diameter, in.	0.430
Cladding Thickness, in.	0.0265
Fuel Rod Pitch, in.	0.568
Fuel Assembly Pitch Spacing, in.	8.587
Unit Cell Metal/Water Ratio (Volume Basis)	0.82
Cladding Material	Zircaloy-4 (Cold Worked)
Fuel	
Material	UO_2
Form	Dished-End, Cylindrical Pellets*
Pellet Diameter, in.	0.370
Active Length, in.	144
Density, % of theoretical, nominal	92.5 - 96
Heat Transfer and Fluid Flow at Design Power	
Total Heat Transfer Surface in Core, ft^2	49,734
Average Heat Flux, Btu/hr-ft^2	171,470
Maximum Heat Flux, Btu/hr-ft^2	534,440
Average Power Density in Core, kW/l	83.39
Average Thermal Output, kW/ft of Fuel Rod	5.66
Maximum Thermal Output, kW/ft of Fuel Rod	17.63
Maximum Cladding Surface Temperature, °F	654
Average Core Fuel Temperature, °F	1,280
Maximum Fuel Central Temperature at Hot Spot, °F	4,220
Total Reactor Coolant Flow, lb/hr	131.32×10^6
Core Flow Area (Effective for Heat Transfer), ft^2	49.19
Core Coolant Average Velocity, fps	15.73
Coolant Outlet Temperature at Hot Channel, °F	647.1

Table 3-1 (continued)

Power Distribution

Maximum/Average Power Ratio, Radial x Local ($F\Delta h$ Nuclear)	1.78
Maximum/Average Power Ratio, Axial (F_Z Nuclear)	1.70
Overall Power Ratio (F_q Nuclear)	3.03
Power Generated in Fuel and Cladding, %	97.3

Hot Channel Factors

Power Peaking Factor (F_Q)		1.011
Flow Area Reduction Factor (F_A):	Interior Bundle Cells	0.98
	Peripheral Bundle Cells	0.97
Local Heat Flux Factor ($F_{Q''}$)		1.014
Hot Spot Maximum/Average Heat Flux Ratio (F_q Nuc and Mech)		3.12

DNB Data

Design Overpower (% Design Power)	114
DNB Ratio at Design Overpower (W-3)	1.55
DNB Ratio at Design Power (W-3)	2.0
Limiting DNB Ratio at Design Overpower (W-3)	1.3

*except Mark BEB test assembly, which included some annular pellets.

Table 3-1a

NUMERICAL LIMITS FOR FUEL ASSEMBLY STRUCTURAL DESIGN CRITERIA

	Deflection or Load		Margin,
	Calculated	Allowable	%
Horiz. Contact Analysis - Spacer Grid Permanent Deformation, in.			
OBE	0.000	0.000	25[a]
DBE	0.037	0.150	80[a]
DBE + LOCA	0.140	0.150	10[a]
Vertical Contact Analysis, lb			
Guide Tube Buckling	4,420	5,588	26
Upper End Spacer Grid Welds	58	170	193
End Spacer Grid Assembly-Buckling	2,500	9,970	299
End Spacer Grid Mechanical Attachment-Shear	155	610	294

[a] Calculated from the following relation:

$$\text{Margin } (\%) = \frac{\text{level to obtain limit - Arkansas Nuclear One Unit 1 level}}{\text{Arkansas Nuclear One Unit 1 level}} \text{ x } 100$$

The preceding criteria provide sufficient safety margins against failure.

Table 3-2

NUCLEAR DESIGN DATA

Fuel Assembly Volume Fractions		
Fuel		0.303
Moderator		0.580
Zircaloy		0.102
Stainless Steel		0.003
Void		0.012
		1.000
Total UO^2 (BOL, First Core)		
Metric Tons		93.1
Core Dimensions, in.		
Equivalent Diameter		128.9
Undensified Active Height - Nominal		142.0
Unit Cell H^2O/U Atomic Ratio (Fuel Assembly)		
Cold		2.88
Hot		2.06
Full-Power Lifetime, days		
First Cycle - Nominal		460
Each Succeeding Cycle		**
Fuel Irradiation, MWd/mtU		
First Cycle Average		14,400
Succeeding Cycle Average		**
Fuel Loading, wt % ^{235}U		
Core Average First Cycle		2.62
Control Data		
Control Rod Material		Ag-In-Cd
No. of Full-Length CRAs	Cycles 1-7	61
	Cycle 8 and following	60
Control Data		
No. of APSRs		8
APSR Material	Cycle 1-8	Ag-In-Cd
	Cycle 9 and following	Inconel
APSR Cladding Material		SS304
APSR Poison Length, inches	Cycle 1-8	36
	Cycle 9 and following	63
Control Rod Cladding Material	Cycles 1-6	SS304
	Cycle 7 and following	Inconel

** See Reload Report, Chapter 3A, and its supporting documentation.

Table 3-3

EXCESS REACTIVITY CONDITIONS

Effective Multiplication, k_{eff}[(a)]	
Cold, 70 °F, Clean	1.257
Hot, 532 °F, Clean, Zero Power	1.194
Hot, 580 °F, Clean, Full Power	1.168
Hot, 580 °F, Full Power, Equilibrium Xenon and Samarium	1.122
Single Fuel Assembly[(b)]	
Hot	0.77
Cold[(c)]	0.87

[(a)] First cycle at beginning of life (BOL), reflects burnable poison holddown.

[(b)] Based on highest probable enrichment of 3.5 wt %.

[(c)] A center-to-center assembly pitch of 21 in. is required for this k_{eff} in cold, unborated water with no xenon or samarium.

Table 3-4

FIRST CYCLE DESIGN BASIS REACTIVITY CONTROL DISTRIBUTION

% Δk/k	
1. Controlled by Soluble Boron	
a. Moderator Temperature Deficit (70 to 532 °F)	3.4
b. Equilibrium Xenon and Samarium	3.5
c. Fuel Burnup and Fission Product Buildup	10.9
2. Controlled by Burnable Poison Rod Assemblies (BPRA)	
a. Fuel Burnup and Fission Product Buildup	4.0
3. Controlled by Movable Control Rod Assemblies (CRA)	
a. Transient Xenon	1.0
b. Doppler Deficit (0 to 2,568 MWt)	1.5
c. Moderator Temperature Deficit (0 to 15% Power 532 to 579 °F)	0.8
d. Dilution Control	0.2
e. Shutdown Margin	1.0
Total Movable CRA Required	4.5

Table 3-5

SHUTDOWN REACTIVITY ANALYSIS

Reactivity, % Δk/k, First Cycle

		BOL	EOL[a]
1.	Required Movable Control Rod Worth		
a.	Doppler Deficit, 0-100% power	1.1	1.3
b.	Moderator Deficit (532°F to 580°F)	0.0	0.8
c.	Inserted Transient Control Rod Worth	1.0	1.4
d.	Possible Reactivity Feedback from Xenon Undershoot (below equilibrium)	0.4	0.4
e.	Possible Dilution Insertion	0.2	0.2
f.	Total Required Rod Worth	2.7	4.1
2.	Shutdown Analysis		
a.	Total Calculated Worth (61 rods)	11.1	10.2
b.	Rod Model Correction (10%)	10.0	9.2
c.	Rod nvt Effect	10.0	9.1
d.	Stuck Rod Worth (not reduced)	-3.4	-2.1
e.	Reduction in Rod Worth (580 °F to 532 °F)	-0.2	-0.2
f.	Net Rod Worth Available	6.4	6.8
3.	Excess Control Rod Worth		
a.	Net Rod Worth Available Minus Total Required Rod Worth	+ 3.7	+ 2.7

[a] EOL in this table refers to about 30 days before the end of the fuel cycle. (This is the point in time when the transient rod bank begins to move out of the core.)

TABLE 3-5a

SHUTDOWN MARGIN CALCULATIONS FOR ANO-1, CYCLE 8

	BOC % Δk/k	380 EFPD % Δk/k	420 EFPD % Δk/k
Available Rod Worth			
Total rod worth, HZP	8.85	9.38	9.15
Worth reduction due to poison material burnup	-0.10	-0.10	-0.10
Maximum stuck rod, HZP	-1.58	-1.86	-1.63
Net Worth	7.17	7.42	7.42
Less 10% uncertainty	-0.72	-0.74	-0.74
Total available worth	6.45	6.68	6.68
Required Rod Worth			
Power deficit, HFP to HZP	1.57	2.30	2.34
Allowable inserted rod Worth	.50	.60	.65
Flux redistribution	.84	1.20	1.20
Total required worth	2.91	4.10	4.19
Shutdown margin (total available minus total required worth)	.54	2.58	2.49

Note: The required shutdown margin is 1.00% Δk/k.

Table 3-6

TRANSIENT ROD BANK WORTH

Bank	Time in Life[a]	Worth, % Δk/k
Transient Bank #1	BOL	1.0
	250 days	1.3
Transient Bank #2	250 days	1.2
	435 days	1.5

[a] Effective full power days at 2,568 MWt.

Table 3-6a

CONTROL ROD WORTHS, CYCLE 8

Control Rod or Group	Worth, % $\Delta k/k$
Group 6, HFP, BOC 8	1.14
Group 7, HFP, BOC 8	1.49
Group 8, HFP, BOC 8	0.39
Group 7, FHP, EOC 8	1.52
Ejected Rod N12, BOC 8, HZP	0.55
Ejected Rod N12, 380 EFPD, HZP	0.60
Ejected Rod N12, EOC 8, HZP	0.59
Stuck Rod N12, BOC 8, HZP	1.58
Stuck Rod N12, 380 EFPD, HZP	1.86
Stuck Rod N12, EOC 8, HZP	1.63

Note:

HFP = Hot Full Power
HZP = Hot Zero Power
Ejected Rods calculated with Banks 5-7 inserted
Stuck Rods calculated with Bank 1-7 inserted

Table 3-7

SOLUBLE BORON LEVELS AND WORTH

Core Conditions	BOL Boron Levels, ppm
1. Cold, k_{eff} = 0.99	
a. No CRA in	1,609
b. All CRA in	1,094
c. One Stuck CRA	1,283
2. Hot, Zero Power k_{eff} = 0.99	
a. No CRA in	1,728
b. All CRA in	748
c. One Stuck CRA	1,088
3. Hot Rated Power, k_{eff} = 1.00	
a. No CRA in	1,441
b. Xenon Transient Rods in	1,341
4. Hot Rated Power With Equilibrium Xenon, k_{eff} = 1.00	
a. No CRA in	1,191
b. Xenon Transient Rods in	1,091
5. Hot Rated Power With Equilibrium Xenon and Samarium, k_{eff} = 1.00	
a. No CRA in	1,091
b. Xenon Transient Rods in	991
6. Boron Worth (% Δk/k)/ppm	
a. Hot	1/100
b. Cold	1/75

Table 3-8

MODERATOR TEMPERATURE COEFFICIENT

Conditions	
1. Core Size, number of fuel assemblies	177
2. Core Average Enrichment, wt % ^{235}U	2.62
3. Power Density, MWt/assembly	14.508
4. Initial Critical Conditions (hot, full power, clean)	
a. Boron Concentration, ppm	1,340
b. CRA Inserted Worth, % $\Delta k/k$	1.0
c. Burnable Poison Worth, % $\Delta k/k$	4.0
d. Moderator Temperature Coefficient, [10^{-4} ($\Delta k/k$)/°F] 3-D with feedback	+ 0.03
5. Threshold Value of MTC for Azimuthul Xenon Instability [10^{-4} ($\Delta k/k$)/°F]	>>1.0
6. Moderator Temperature Coefficient at End of First Cycle [10^{-4} ($\Delta k/k$)/°F]	- 2.6
7. MTC at End of Equilibrium Fuel Cycle [10^{-4} ($\Delta k/k$)/°F]	- 3.0
8. Maximum MTC Used in Safety Analysis [10^{-4} ($\Delta k/k$)/°F]	+ 0.5
9. Power Coefficient BOL, 2,568 MWt [10^{-4} ($\Delta k/k$)/% power]	- 1.1

Table 3-9

pH CHARACTERISTICS

^{7}Li, ppm	T_{mid}, °F	Boron Concentration, ppm	pH Units
0.5	70	1,800	5.0
2.0	70	1,800	5.6
0.5	580	1,200	7.0
2.0	580	1,200	7.5
0.5	580	17	7.2
2.0	580	17	7.8
0.5	70	17	7.9
2.0	70	17	8.5

Table 3-10

CALCULATED EXPERIMENTAL ROD AND ROD ASSEMBLY COMPARISON

Core No.	Clusters per Core	Ag-In-Cd Rods per Cluster	H_2O Holes, Fraction of Core	Soluble Boron, ppm	Control Rod Worth,[a] % $\Delta k/k$ Experimental	Calculated
5-B	4	4	0.051	1,232	2.02 ± 0.09	2.13 ± 0.02
4-F	4	9	0.0	1,219	3.36 ± 0.09	3.43 ± 0.02
5-C	2	12	0.056	1,167	2.36 ± 0.09	2.44 ± 0.02
4-D	1	16	0.0	1,390	1.45 ± 0.09	1.35 ± 0.02
5-D	2	16	0.058	1,118	2.86 ± 0.09	2.78 ± 0.02
4-E	1	20	0.0	1,365	1.52 ± 0.09	1.48 ± 0.02
5-E	2	20	0.059	1,082	3.06 ± 0.09	3.02 ± 0.02

[a] Variations in these data from those shown in the FSAR tabulation result from a change in the nuclear calculational model.

Table 3-11

NUCLEAR POWER FACTORS
CYCLE 1

1. The nominal nuclear peaking factors for the worst time in core life are

   $F\Delta h$ = 1.77
   Fz = 1.70
   Fq = 3.01

2. The design nuclear peaking factors for the worst time in core life are

   $F\Delta h$ = 1.78
   Fz = 1.70
   Fq = 3.03

   where:

   $F\Delta h$ = max/avg total radial x local nuclear power ratio
   Fz = max/avg axial power ratio (nuclear)
   Fq = $F\Delta h$ x Fz (nuclear total)

The nominal values are the maximum values calculated with nominal spacing of fuel assemblies. The design values are obtained by examining maximum, nominal, and minimum fuel assembly spacing and determining the worst values for the combined effect of flow and rod peaking.

Table 3-12

COEFFICIENTS OF VARIATION, Cycle 1

CV No.	Description	Standard Deviation of Variable (σ)	Mean Value of Variable ($\overline{X}$)	Coefficient of Variation ($\sigma/\overline{X}$)
1.	Flow Area			
	Interior Bundle Cells	0.00190	0.17740	0.01072
	Peripheral Bundle Cells	0.00346	0.21546	0.01608
2.	Local Rod Diameter	0.000647	0.430	0.00151
3.	Average Rod Diameter (Die-Drawn Local and Average Same)	0.000647	0.430	0.00151
4.	Local Fuel Loading			0.00698
	Subdensity	0.00647	0.935	0.00692
	Subfuel Area (Diameter Effect)	0.000094	0.1075	0.00088
5.	Average Fuel Loading			0.00557
	Subdensity	0.00485	0.935	0.00519
	Sublength	0.26294	144	0.00183
	Subfuel Area (Diameter Effect)	0.000094	0.1075	0.00088
6.	Local Enrichment	0.00421	2.30	0.00183
7.	Average Enrichment	0.00421	2.30	0.00183

Enrichment values are for worst case normal assay batch; maximum variation occurs for minimum enrichment.

Table 3-13

**STATISTICAL DNB RESULTS
(99% Confidence Level)**

Point	Power, % of 2,568 MWt	$F\Delta h$	No. of Channels Possible DNB	Population Protected, %	Hot Channel DNB Ratio (W-3)
DNB Results - Maximum Design Condition, Cycle 1					
A	100	1.78	< 1	> 99.997	2.00
B	114	1.78	14.0	99.962	1.55
C	122.5	1.78	70.0	99.810	1.30
DNB Results - Most Probable Condition					
D	100	1.77	< 1	> 99.997	2.25
E	114	1.77	2.5	99.993	1.82
F	122.5	1.77	9.0	99.976	1.54

Table 3-14

HOT CHANNEL DATA AND PERFORMANCE FOR FOUR PUMP OPERATION, CYCLE 1

Engineering Hot Channel Factors

$F_Q = 1.011$ $F_A = 0.98$ (interior cells)

$F_Q = 1.014$ $F_A = 0.97$ (wall cells)

Performance Summary

Reactor Power, %	DNB Ratio (W-3)	Exit Quality, %
100	2.00	0.9
107.5 (trip setting)	1.75	3.4
114 (maximum power)	1.55	5.8
122.5	1.30	9.2

Hot Channel Statistical Statement

The DNB ratio in the hot channel at the maximum overpower of 114 percent is 1.55 which corresponds to a 99 percent confidence that at least 99.34 percent of the fuel channels of this type are in no jeopardy of experiencing a DNB.

Table 3-15

HOT CHANNEL PERFORMANCE VS PUMPS IN SERVICE, CYCLE 1

Reactor Coolant Pumps Operating	3 Pumps	2 Pumps (2 Loops)
Hot Channel DNBR at Maximum Design Overpower	1.30	1.40
Hot Channel Quality at Minimum DNBR Point, %	7.0	15.0
Reactor Coolant Flow, % of Rated	74.7	49.0
Maximum Design Overpower, % of Rated Power (2,568 MWt)	101.5	77.0

Table 3-16

FUEL ROD DESIGN FACTORS AND CONDITIONS, CYCLE 1

Design Power Peaking Factors[(a)]

$F\Delta h$ = 1.78

Fz = 1.70

$F_{Q''}$ = 1.03

A conservative value of 1.03 was assumed for the local hot spot heat flux peaking factor, $F_{Q''}$. The assigned value corresponds to a 99 percent population-protected relationship as described in the statistical technique, and shown in Figure 3-33.

Maximum Local Heat Input Factor

F_q = 3.12 (nuclear and mechanical)

Nominal Clearance Fuel to Clad BOL

Δd = 0.007 in.

Maximum Design Clearance Fuel to Clad BOL

Δd = 0.0085 in.

Hot Spot Fuel Conditions at BOL

Power, %	Heat Rate, kW/ft	Fuel Center Temp, °F
100	17.63	4,220
114	20.1	4,540

Average Fuel Conditions at BOL

Power, %	Heat Rate, kW/ft	Fuel Center Temp, °F
100	5.66	1,280

(a) Reference 4, Section 3.4

Table 3-17

HOT CHANNEL AND CORE AVERAGE COOLANT CONDITIONS, CYCLE 1

Hot Channel Coolant Conditions

Power, %	FΔh	Quality, %	Exit Void Fraction, %	Operating Pressure, psig
100	1.78	(-)1.3[b]	0.9[a]	2,185
114	1.78	3.4	12.8	2,185
128	1.78	8.7	31.2	2,185
114	1.96	7.1	32.2	2,185
100	1.78	0.9	2.8	2,120
114	1.78	5.8	26.3	2,120
128	1.78	11.3	39.8	2,120
114	1.96	9.5	41.1	2,120

Core Average Void Fraction

Flow, %	Pressure, psig	Core Void Fraction, %
100	2,185	0.06
100	2,120	0.194
95	2,185	0.201
95	2,120	0.770

[a] Subcooled Voids.
[b] Negative indication of quality denotes subcooling.

Table 3-18

REACTOR CORE LEAKAGE FLOW, CYCLE 1

Path	% of System Flow[a]
1. Shroud	1.6
2. Control Rod Guide Tubes and Instrument Guide Tubes	1.7
3. Inlet to Outlet Interfaces	0.3
4. Assumed Allowance	1.8
Total Leakage	5.4

[a] Based on the design system flow of 131.3 x 10^6 lb/hr.

Table 3-19

DNB RATIOS IN THE FUEL ASSEMBLY CHANNELS (W-3), CYCLE 1

Most Probable Conditions

Cell Type	G, lb/hr-ft^2 x 10^{-6}	DNBR (W-3) (114% Power)
Unit	2.51	1.82
Corner	2.58	1.86
Wall (peripheral)	2.56	1.90
Control Rod	2.40	1.97
Maximum Design Conditions		
Unit	2.26	1.55
Corner	2.14	1.66
Wall (peripheral)	2.20	1.65
Control Rod	2.16	1.69

Table 3-20

DNB RATIOS AND POWER LIMITS WITH A VENT VALVE OPEN

Four Pump DNB Ratio Comparison

Percent Rated Power	DNBR (W-3) (Full Flow)	DNBR (W-3) Vent Valve Open)
100	2.00	1.84
107.5[a]	1.75	1.59
114	1.55	1.4
117	1.46	1.3

Partial Pump Power Limitations

Reactor Coolant Pumps Operating	3 Pumps	2 Pumps - 2 Loops
Hot Channel DNBR	1.3	1.45
Hot Channel Quality at Minimum DNBR Point, %	8.0	15.0
Reactor Coolant Flow, % of rated	70.1	44.4
Maximum Design Overpower, % of rated power (2,568 MWt)	97.5	73.0

[a] Trip set point.

Table 3-21

INTERNALS VENT VALVE MATERIALS

Valve Part Name	Material and Form	Material Specification No.
Valve Body	304 SS Casting[a]	ASTM A351-CF8
Valve Disc	304 SS Casting[a]	ASTM A351-CF8
Disc Shaft	431 SS Bar[b]	ASTM A276 Type 431 Cond. T
Shaft Bushings	Stellite No. 6	
Retaining Rings (Top and Bottom)	15-5 pH (H 1100) SS Forgings	AMS 5658
Ring Jackscrews	"A-286 Superalloy" SS[c]	AMS 5737 C
Jackscrew Bushings	431 SS Bar	ASTM A276 Type 431 Cond. A
Misc Fasteners, Covers, Locking Devices, etc.	304 SS Plate Bar, etc. Inconel 600 and Inconel 718	ASTM A240, ASTM A276 ASTM A479, ASTM A193, ASTM B166, ASTM B168, AMS 5662C

[a] Carbide solution annealed, C_{max} 0.08%, Co_{max} 0.2%.

[b] Heat treated and tempered to Brinell Hardness Number (BHN) range of 290-320.

[c] Heat treated to produce a BHN of 248 minimum.

Table 3-22

VENT VALVE SHAFT AND BUSHING CLEARANCES

Clearance Gaps are Illustrated in Figure 3-63

A. Cold Clearance Dimensions at 70 °F

Bushing ID	1.500	to	1.505	
Shaft OD	1.490	to	1.485	
	0.010	to	0.020	Clearance (Gaps 1, 2, 7, and 8)
Body ID	2.000	to	2.003	
Bushing OD	1.997	to	1.995	
	0.003	to	0.008	Clearance (Gaps 3, 4, 5, and 6)

Bushing End Clearance (Gaps 9 and 10)

Body Lugs	5.752	to	5.756	
Disc Hub	4.746	to	4.742	
	1.006	to	1.014	
	0.996	to	0.992	
	0.010	to	0.022	End Clearance (Gaps 9 and 10)

Bushing Flange 0.249 x 4 = 0.996
0.248 x 4 = 0.992

B. Hot Clearance Differential Change From 70 to 580 °F

Linear coefficient of thermal expansion of the materials for a temperature change of 70 to 600 °F.

Shaft:	A-286	9.8×10^{-6} in./in./°F
Bushing:	Stellite #6	8.1×10^{-6}
Bodies:	CF8 Stainless	9.82×10^{-6}

$\Delta T = 580 - 70 = 510$ °F

Shaft $\Delta D = D\alpha\Delta T = 1.5\,(9.8 \times 10^{-6})\,510 = 0.0075$
Bushing ID $= 1.5\,(8.1 \times 10^{-6})\,510 = \underline{0.0062}$
-0.0013 Decrease

Bushing OD $= 2\,(8.1 \times 10^{-6})\,510 = 0.0083$
Body ID $= 2\,(9.82 \times 10^{-6})\,510 = \underline{0.0100}$
+0.0017 Increase

Bushing Endplay Hot

CF8 Body $\Delta L = 1\,(9.82 \times 10^{-6})\,510 = 0.0050$
Stellite #6 Bushing Flange $= 1\,(8.1 \times 10^{-6})\,510 = \underline{0.0041}$
+0.0009 Increase

Table 3-23

FUEL ASSEMBLY DIMENSIONS AND IRRADIATION CONDITIONS

Fuel Assembly Component Dimensions

Item	Material	Dimensions, in.
Fuel Rod		
Fuel	UO_2 sintered pellet (92.5%theoretical density)	0.370 diameter
Fuel Clad	Zircaloy-4	0.430 OD x 0.377 ID x 153-1/8 long
Fuel Rod Pitch		0.568
Active Fuel Length		141.8
Metallic Spacer	Zircaloy-4	0.362 OD x 0.297 ID
Minimum Fuel to Clad Gap (BOL)		0.0059
Fuel Assembly		
Fuel Assembly Pitch		8.587
Overall Length		165-5/8 165-21/32 MK-B6
Control Rod Guide Tube	Zircaloy-4	0.530 OD x 0.016 wall
Instrumentation Tube	Zircaloy-4	0.493 OD x 0.441 ID
End Fittings	Stainless steel (castings)	
Spacer Grid	Inconel 718 strips	0.020 thick exteriors 0.016 thick interiors
Spacer Grid (Zirc)	Zircaloy-4	0.020 thick exteriors 0.018 thick interiors
Spacer Sleeve	Zircaloy-4	0.554 OD x 0.502 ID

Fuel Irradiation Conditions

Average Design Burnup, MWd/mtu	29,100
Hot Rod Maximum Average Burnup, MWd/mtU	40,900
Hot Rod Maximum Burnup, MWd/mtU	44,950
Maximum Design Local Burnup, MWd/mtU	55,000
Irradiation Fuel Growth, % by volume	9.5

Table 3-24

CLAD CIRCUMFERENCE STRESSES

Case	P External psig	P Internal psig	T Clad, °F	σBending + σMembrane	σTotal, psi	Yield[e] Strength, psi	Ultimate[e] Strength, psi
1. Beginning of Life - Pre-Operational	2,185[a]	690	532	-18,800	-18,800	48,000	57,000
Hot Standby % of Power	2,500[b]	690	532	-23,700	-23,700	48,000	57,000
2. Beginning of Life – Void Section	2,185	1,340	650	-10,100	-10,100	45,000	50,000
Section of Clad - 100% Power	2,500	1,340	650-14,200	-14,200	45,000	50,000	
3. Beginning of Life - Void Section	2,185	1,380	650	-9,600	-9,600	45,000	50,000
Section of Clad - 114% Power	2,500	1,380	650	-13,700	-13,700	45,000	50,000
4. Beginning of Life - Fueled	2,185	1,340	723	-10,100	-13,700	42,000	44,000
Section of Clad - 100% Power	2,500	1,340	723	-14,200	-17,800	42,000	44,000
5. Beginning of Life - Fueled	2,185	1,380	733	-9,600	-13,700	41,500	43,500
Section of Clad - 114% Power	2,500	1,380	733	-13,700	-17,800	41,500	43,500
6. End of Life - Hot Standby –	2,185	725	532	-18,300	-18,300	48,000	57,000
0% Power	2,500	725	532	-23,100	-23,100	48,000	57,000
7. EOL - Fueled Section of Clad -	2,185	3,300[c]	704	+10,100[d]	+16,200	43,000	46,000
100% Power - Design Internal Press	2,500	3,300	704	+7,200[d]	+12,100	43,000	46,000
8. EOL - Fueled Section of Clad -	2,185	3,300	711	+10,100[d]	+16,500	43,000	46,000
114% Power - Design Internal Press	2,500	3,300	711	+7,200[d]	+12,400	43,000	46,000
9. End of Life - Fueled Section of	2,185	2,160	704	-300	-3,000	43,000	46,000
Clad - 100% Power	2,500	2,160	704	-3,900	-6,600	43,000	46,000
10. End of Life - Fueled Section of	2,185	2,300	711	+1,100[d]	+4,400	43,000	46,000
Clad - 114% Power	2,500	2,300	711	-2,300	-5,300	43,000	46,000
11. End of Life - Immediately	2,185	1,450	535	-8,700	-8,800	48,000	57,000
After Shutdown	2,500	1,450	535	-12,700	-12,800	48,000	57,000
12. EOL at Clad Temperature of 425 °F	1,725	1,270	425	-5,300	-5,300	50,000	62,500

Table 3-24 (continued)

Notes:

(a) System operating pressure
(b) System design pressure
(c) Fuel rod clad internal design pressure
(d) Pressure stress only
(c) Fuel rod clad internal design pressure

(e) Cladding is specified with 45,000 psi minimum yield strength and 10 percent minimum elongation, both at 650 °F. Minimum room temperature strengths are approximately 75,000 psi yield strength (0.2 percent offset) and 85,000 psi ultimate tensile strength

Table 3-25

B&W HIGH BURNUP IRRADIATION PROGRAM - CAPSULE FUEL TEST

Identification			Burnup		Irradiation	Heat Rate			Clad
Capsule	Fuel Rod	Irradiation Facility	MWd/mtU $x\ 10^{-3}$	Fissions/cc $x\ 10^{-20}$(1)	Time Calendar Months(2)	Initial, kW/ft	Final, kW/ft	Diametral Gap, mils	Thickness, mils
B-1	B-1	RS-3(3)	10	2.5	4	18	17.5	4-5	25
	B-2			2.5			17.5	7-8	25
	B-3			2.5			17.5	7-8	15
B-2	B-20	RS-4	17	3.8	10	18	16.9	Powder	25
	B-19		26	6.5			16.9	4-5	25
B-3	B-7	RS-6	30	7.5	11	18	16.1	4-5	25
	B-8			7.5			16.1	7-8	25
	B-9			7.5			16.1	7-8	15
B-4	B-10	RS-1	45	10.05	17	18	14.9	Powder	25
	B-11		11.25				14.9	4-5	25
B-5	B-13	RS-5	55	13.75	21	18	14.1	4-5	25
	B-14		13.75				14.1	7-8	25
	B-15		13.75				14.1	7-8	15
B-6	B-31	RS-2	70	15.63	26	18	13.3	Powder	25
	B-17			17.5			13.3	7-8	25
	B-4			20.0			12.5	7-8	25
B-7	B-5	RS-5	80	17.87	30	18	12.5	Powder	25
	B-4			20.0			12.5	7-8	25
B-8	B-22	RL-1	30	7.5	11	21-1/2	20.5	4-5	25
	B-23			7.5			20.5	7-8	25
	B-33			7.5			20.5	7-8	15

Table 3-25 (continued)

Capsule	Identification Fuel Rod	Irradiation Facility	Burnup MWd/mtU x 10^{-3}	Burnup Fissions/cc x 10^{-20}(1)	Irradiation Time Calendar Months(2)	Heat Rate Initial, kW/ft	Heat Rate Final, kW/ft	Diametral Gap, mils	Clad Thickness, mils
B-9	B-25	RL-2	55	13.75	21	21-1/2	19.3	4-5	25
	B-26			13.75			19.3	7-8	25
	B-27			13.75			19.3	7-8	15
B-10	B-28	RL-3	80	20.0	30	21-1/2	17.8	7-8	25
	B-29			20.0			17.8	7-8	25
	B-30			20.0			17.8	7-8	15
B-11	B-24	RL-4	70	17.5	26	21-1/2	16.5	7-8	15
	B-34			17.5			16.5	7-8	25
	B-35			17.5			16.5	7-8	25
B-12	B-16	RS-3	65	14.50	24	18	13.3	Powder	25
	B-32			16.24			13.3	7-8	15

(1) Based on 200 MeV per fission.

(2) Based on 80 percent reactor efficiency.

(3) Irradiation Location in BAWTR.

Table 3-26

DELETED

Table 3-27a

CONTROL ROD ASSEMBLY DATA

Item	Data
Number of CRA	60
Number of Control Rods per Assembly	16
Outside Diameter of Control Rod, in.	0.440
Cladding Thickness, in.	0.021
Cladding Material	Type 304 SS, Cold-Worked
End Plug Material	Type 304 SS, Annealed
Spider Material	SS, Grade CF3M
Poison Material	80% Ag, 15% In, 5% Cd
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Section, in.	134
Stroke of Control Rod, in.	139

Table 3-27b

CONTROL ROD ASSEMBLY DATA - EXTENDED LIFE DESIGN

Item	Data
Number of CRA	60 (With Cycle 8)
Number of Control Rods per Assembly	16
Outside Diameter of Control Rod, in.	0.441
Cladding Thickness, in.	0.0225
Cladding Material	Inconel
End Plug Material	Inconel
Spider Material	SS, Grade CF3M
Poison Material	80% Ag, 15% In., 5% Cd
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Section, in.	139
Stroke of Control Rod, in.	139

Table 3-27c

CONTROL ROD ASSEMBLY DATA – ALTERNATE EXTENDED LIFE DESIGN

Item	Data
Number of CRA	60 (Starting Cycle 22)
Number of Control Rods per Assembly	16
Outside Diameter of Control Rod, in.	0.441
Cladding Thickness, in.	0.0225
Cladding Material	Inconel
End Plug Material	Inconel
Spider Material	SS, Grade CF3M
Poison Material	80% Ag, 15% In., 5% Cd
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Top Section, in.	127
Length of Poison Middle Section, in.	6
Length of Poison Bottom Section, in.	5.68
Stroke of Control Rod, in.	139

Table 3-28a

AXIAL POWER SHAPING ROD ASSEMBLY DATA - BLACK

Item	Data
Number of Axial Power Shaping Rod Assemblies	8
Number of Axial Power Shaping Rods per Assembly	16
Outside Diameter of Axial Power Shaping Rod, in.	0.440
Cladding Thickness, in.	0.021
Cladding Material	Type 304 SS, Cold-Worked
Plug Material	Type 304 SS, Annealed
Poison Material	80% Ag, 15% In, 5% Cd
Spider Material	SS, Grade CF3M
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Section, in.	36
Stroke of Rod, in.	139

Table 3-28b

AXIAL POWER SHAPING ROD ASSEMBLY DATA - GRAY

Item	Data
Number of Axial Power Shaping Rod Assemblies	8
Number of Axial Power Shaping Rods Per Assembly	16
Outside Diameter of Axial Power Shaping Rod, in.	0.440
Cladding Thickness, in.	0.027
Cladding Material	Type 304 SS, Cold-Worked
Plug Material	Type 304 SS, Annealed
Poison Material	Inconel
Spider Material	SS, Grade CF3M
Female Coupling Material	Type 304 SS, Annealed
Length of Poison Section, in.	63
Stroke of Control Rod, in.	139

Table 3-29a

BURNABLE POISON ROD ASSEMBLY DATA - MK - B4

Item	Data
Number of Burnable Poison Rods per Assembly	Up to 16
Outside Diameter of Burnable Poison Rod, in.	0.430
Cladding Thickness, in.	0.035
Cladding Material	Zircaloy-4, Cold-Worked
End Plug Material	Zircaloy-4, Annealed
Poison Material	B_4C in Al_2O_3
Length of Poison Section, in.	126
Cycle 7	117
After Cycle 7	121.5
Spider Material	SS, Grade CF3M
Coupling Mechanism Material	Type 304 SS, Annealed
Retainer	
Load arm assembly	Type 304 SS
Housing assembly	Type 304 SS
Spring	Inconel x-750

Table 3-29b

BURNABLE POISON ROD ASSEMBLY DATA - MK - B5

Item	Data
Number of Burnable Poison Rods per Assembly	Up to 16
Outside Diameter of Burnable Poison Rod, In.	0.430
Cladding Thickness, in.	0.035
Cladding Material	Zircaloy-4, Cold-Worked
Plug Material	Zircaloy-4, Annealed
Spider Material	SS, Grade CF3M
Poison Material	B_4C in Al_2O_3
Length of Poison Section, in.	121.5

Table 3-30

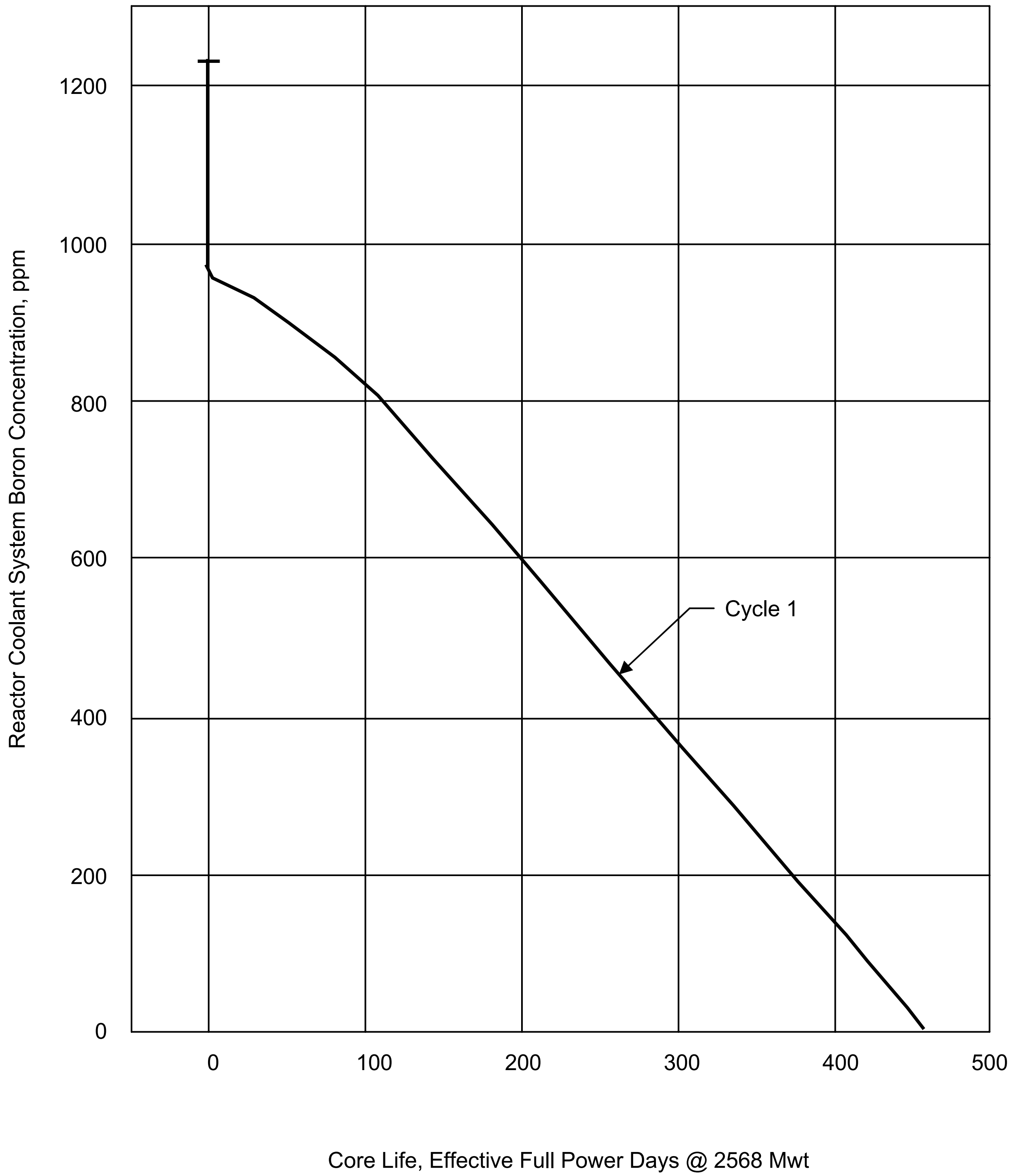
ORIFICE ROD ASSEMBLY DATA

Item	Data
Number of Orifice Rod Assemblies	40 First Cycle, 76 Equilibrium Cycle
Number of Orifice Rods per Assembly	16
Outside Diameter of Orifice Rod, in.	0.440
Orifice Rod Material	Type 304 SS, Annealed
Spider Material	SS, Grade CF3M
Coupling Mechanism Material	Type 304 SS, and 17-4PH, Condition H 1100

Table 3-31

CONTROL ROD DRIVE MECHANISM DESIGN DATA

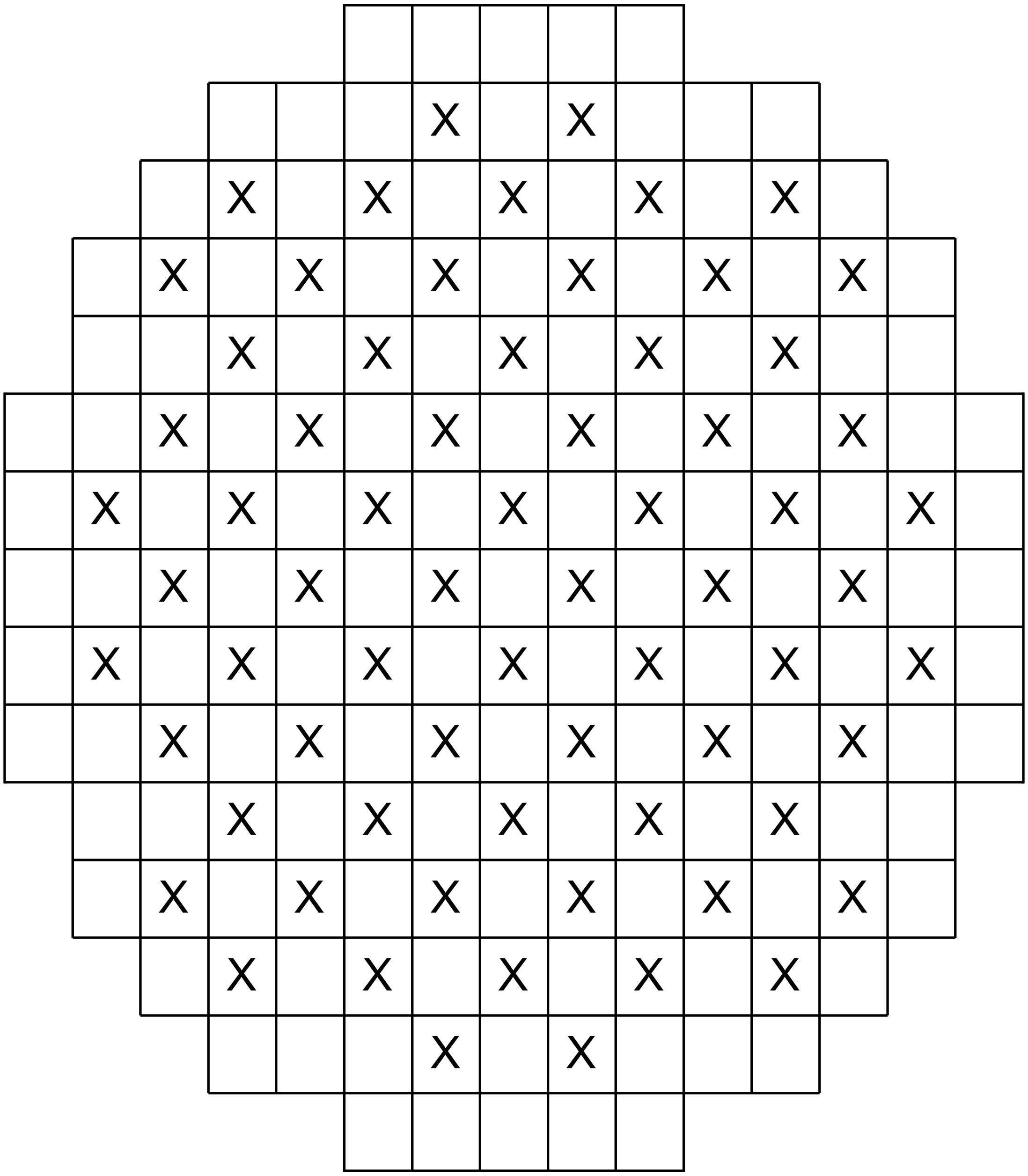
Mechanism Function	Shim Safety	Axial Power Shaping
Type	Roller Nut Drive	Roller Nut Drive
Quantity	60	8
Location	Top-mounted	Top-mounted
Direction of Trip	Down	Does Not Trip
Velocity of Normal Withdrawal and Insertion, in./min.	30	30
Maximum Travel Time for Trip		
2/3 Insertion, s	1.40	Drive Has No Trip Function
3/4 Insertion, s	1.50	Drive Has No Trip Function
Length of Stroke, in.	139	139
Design Pressure, psig	2,500	2,500
Design Temperature, °F	650	650
Weight of Mechanism (approx), lb	940	940



SAR FIGURE NO. 3-1

AMENDMENT 20

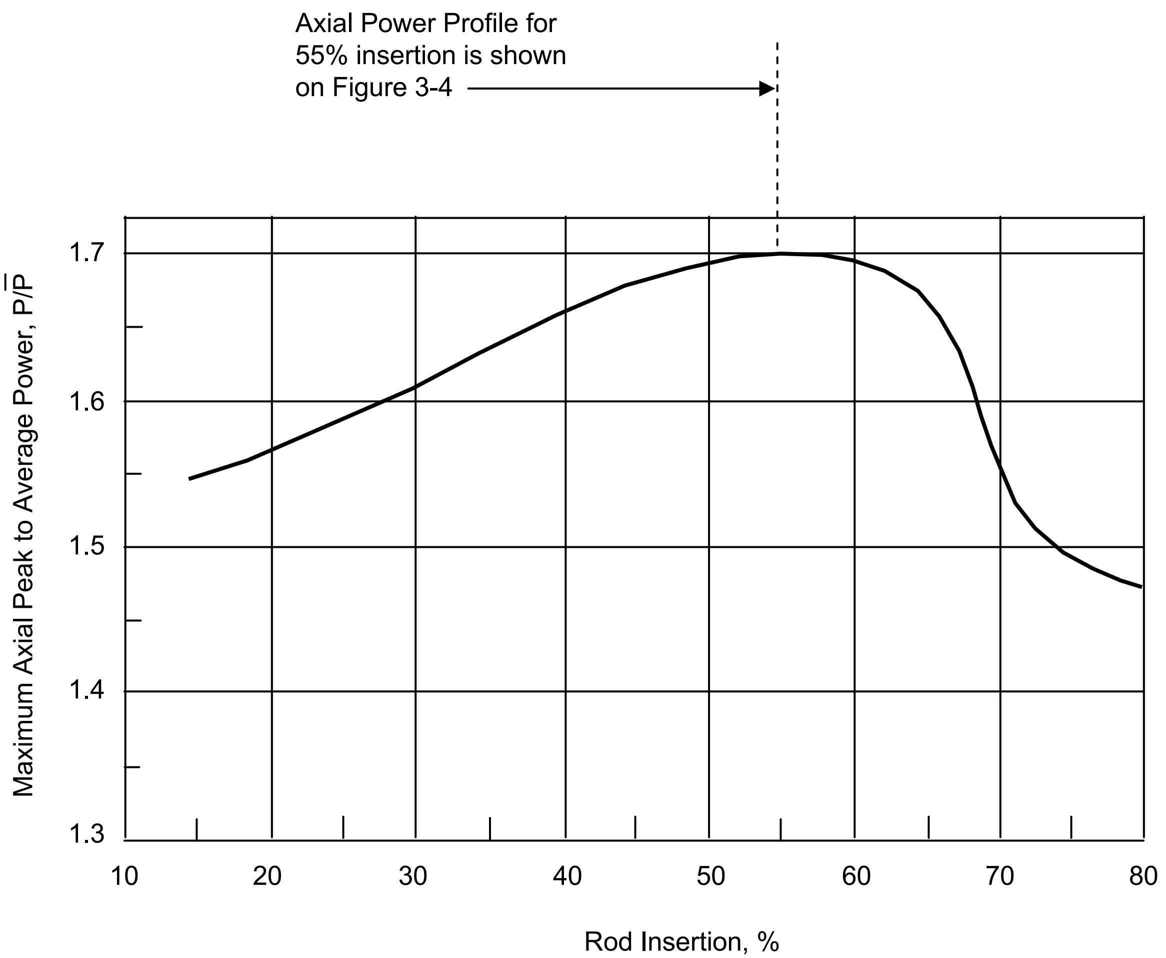
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
BORON CONCENTRATION VERSUS CORE LIFE	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-2

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LOCATION OF FUEL ASSEMBLIES CONTAINING BURNABLE POISON RODS	BASED ON DRAWING NO	SHEET 1 REV.



AXIAL PEAK TO AVERAGE POWER VERSUS XENON OVERRIDE ROD INSERTION

SAR FIGURE NO. 3-3

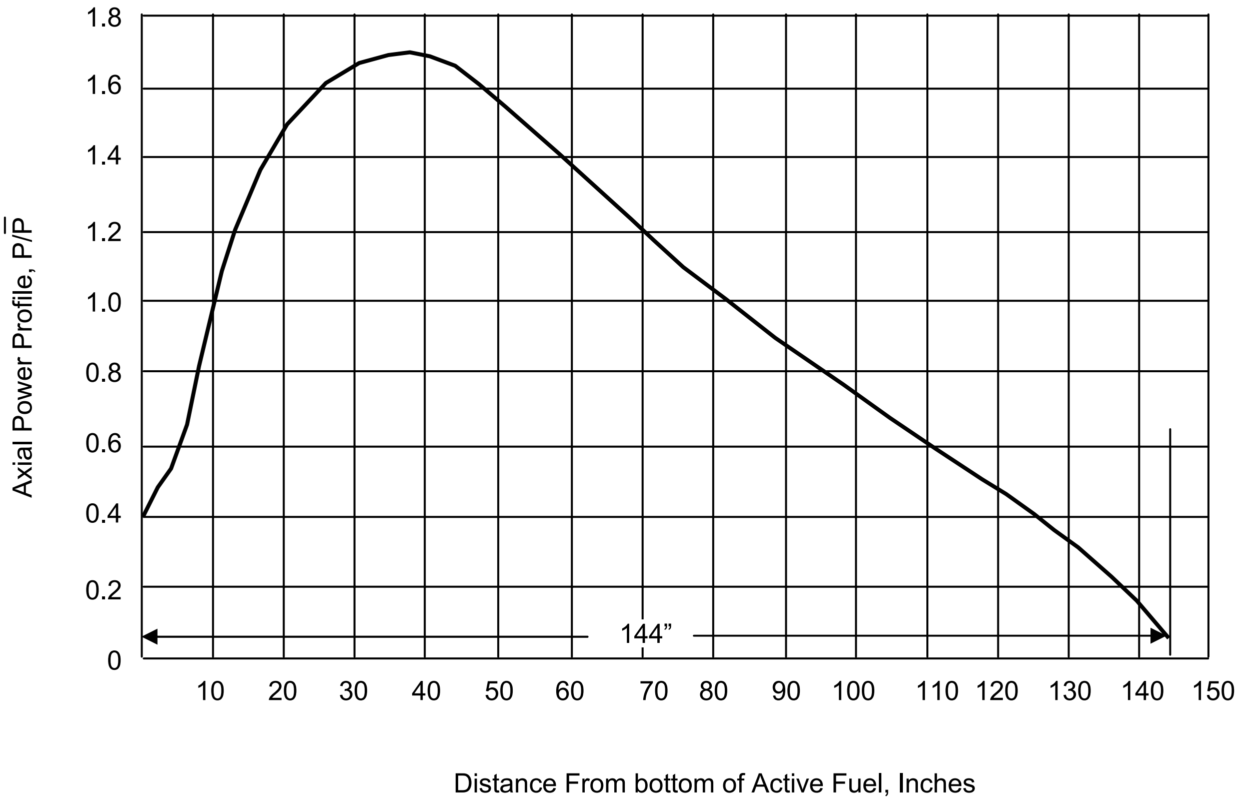
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

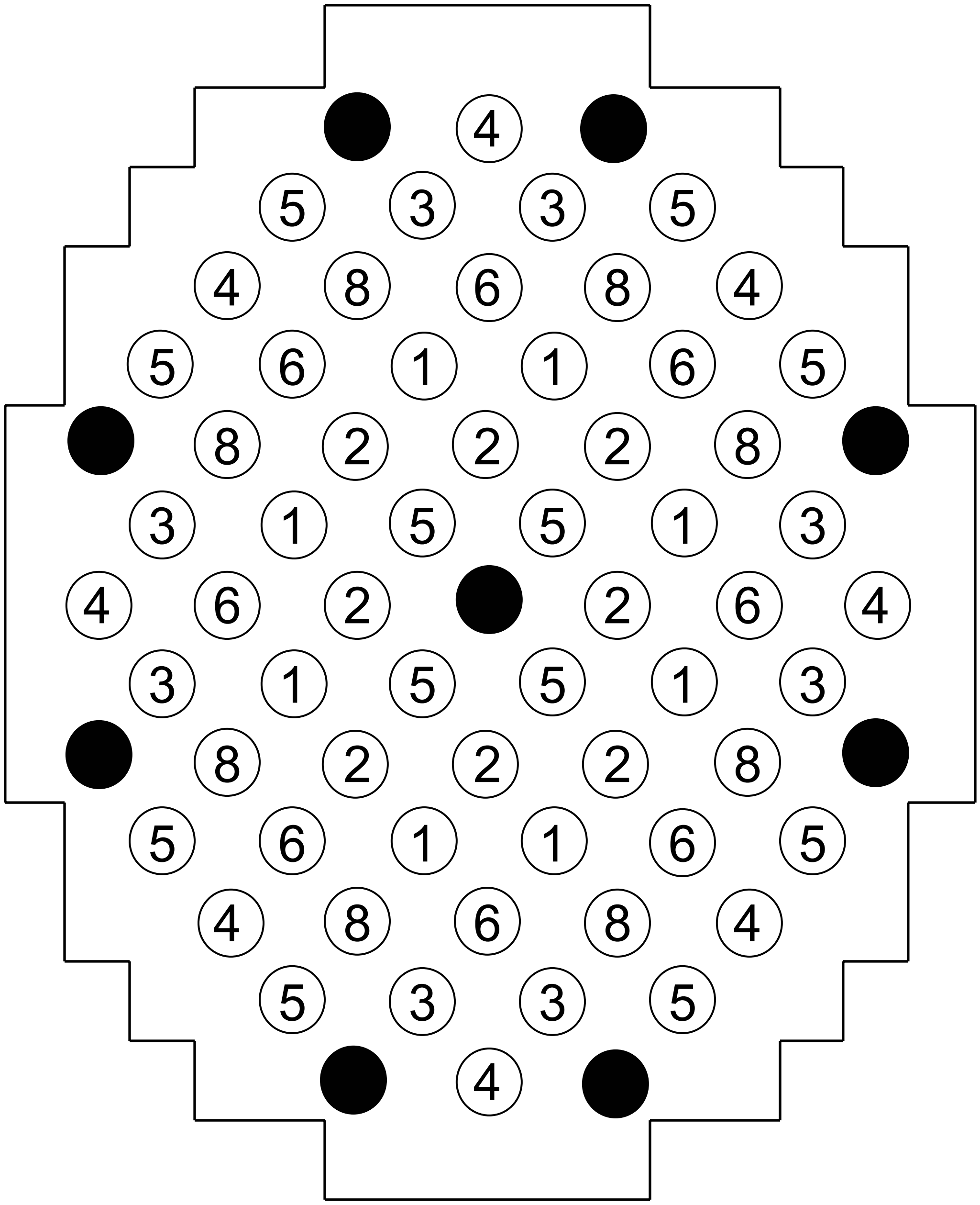
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



AXIAL POWER PROFILE, XENON OVERRIDE RODS 55 PERCENT INSERTED			SAR FIGURE NO. 3-4
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

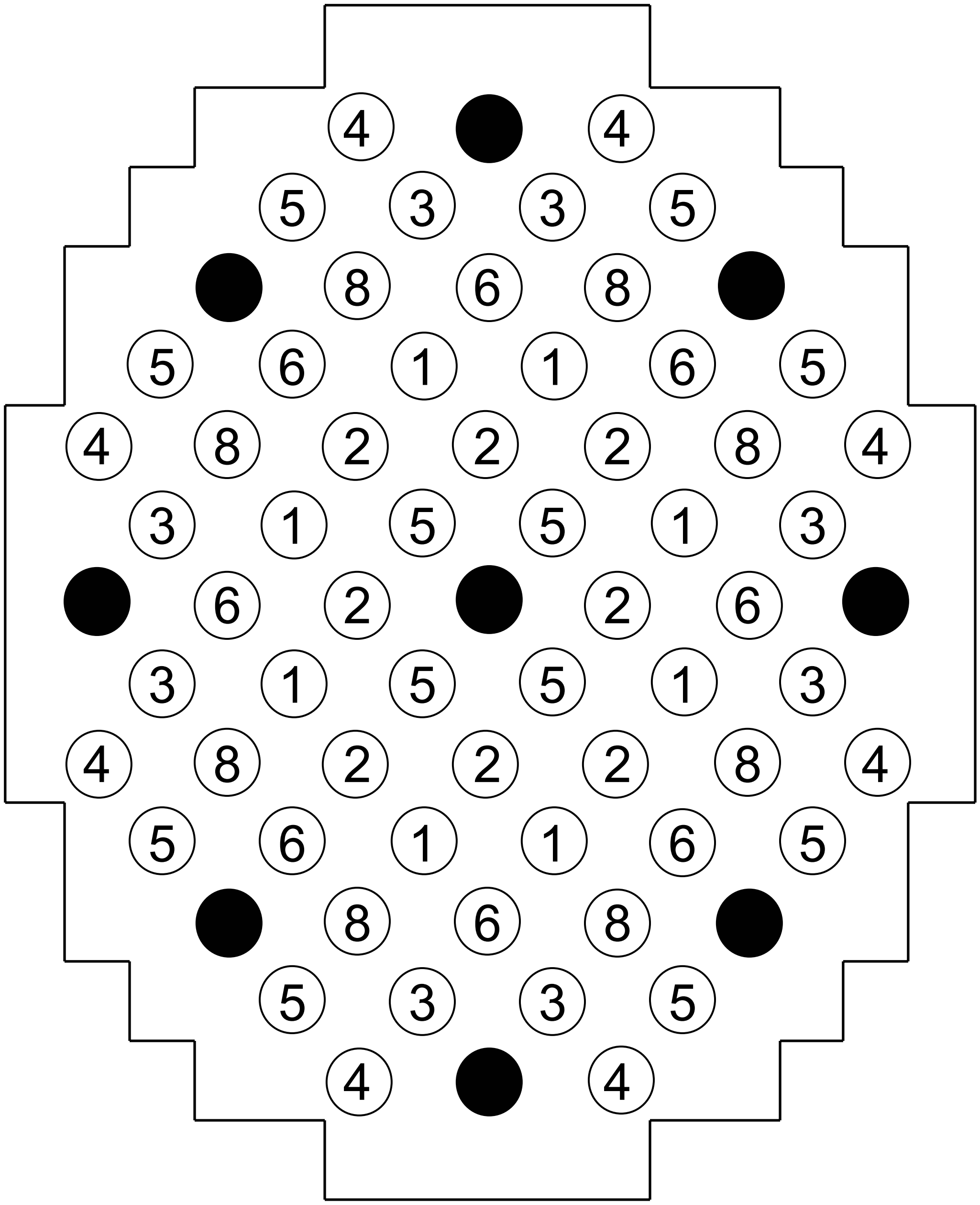


BANKS	PURPOSE
1, 2, 3, 4	Safety
5	Doppler
6	Doppler
● 7	Transient
8	APSR

SAR FIGURE NO. 3-5

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
0 – 250 DAY TRANSIENT BANK	BASED ON DRAWING NO	SHEET 1 / REV.



BANKS	PURPOSE
1, 2, 3, 4	Safety
5	Doppler
6	Doppler
● 7	Transient
8	APSR

SAR FIGURE NO. 3-6

AMENDMENT 20

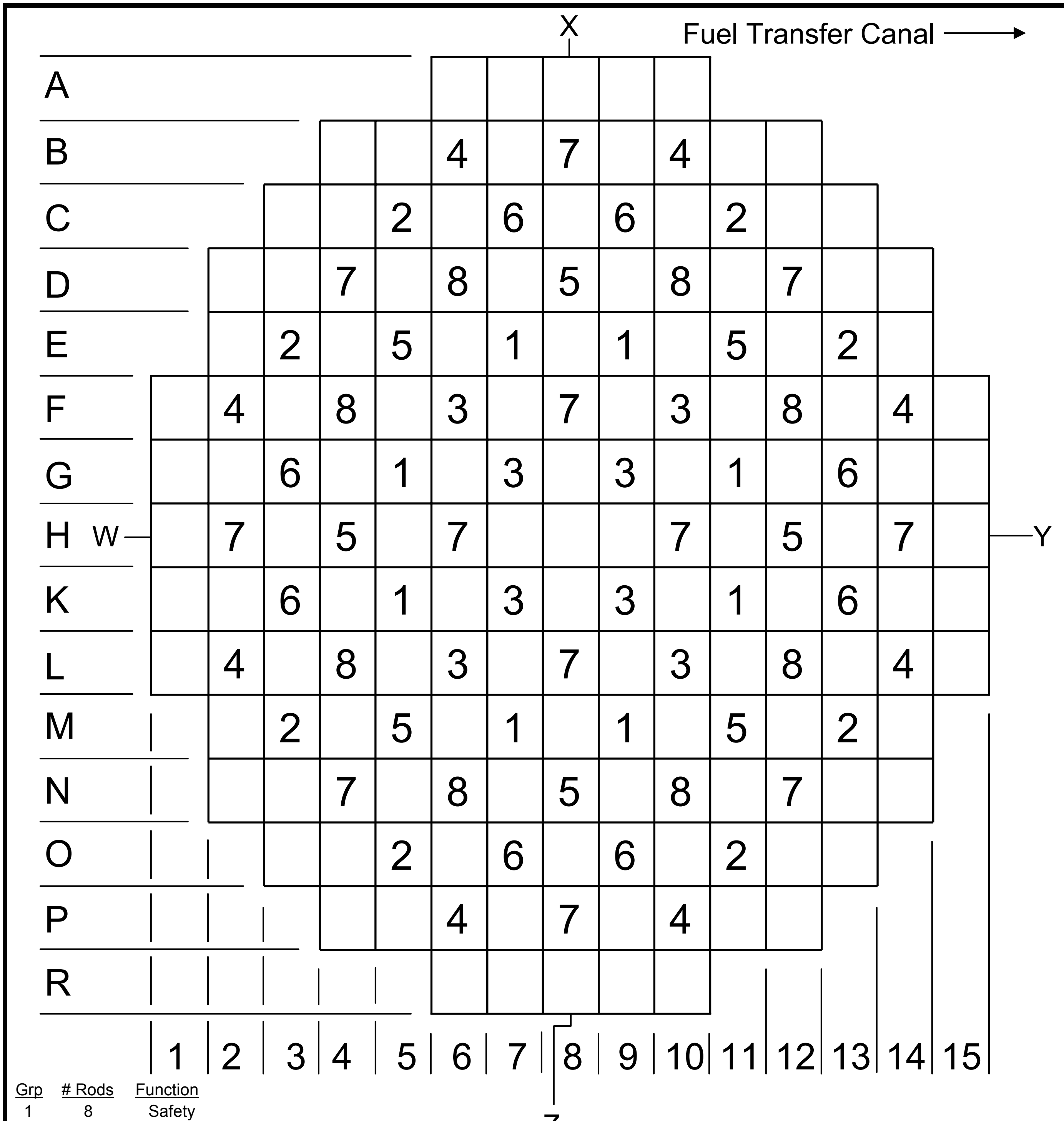
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

250 – 460 DAY TRANSIENT BANK

BASED ON DRAWING NO	SHEET	REV.
	1	



Grp	# Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs

X Group Number

SAR FIGURE NO. 3-6A

AMENDMENT 20

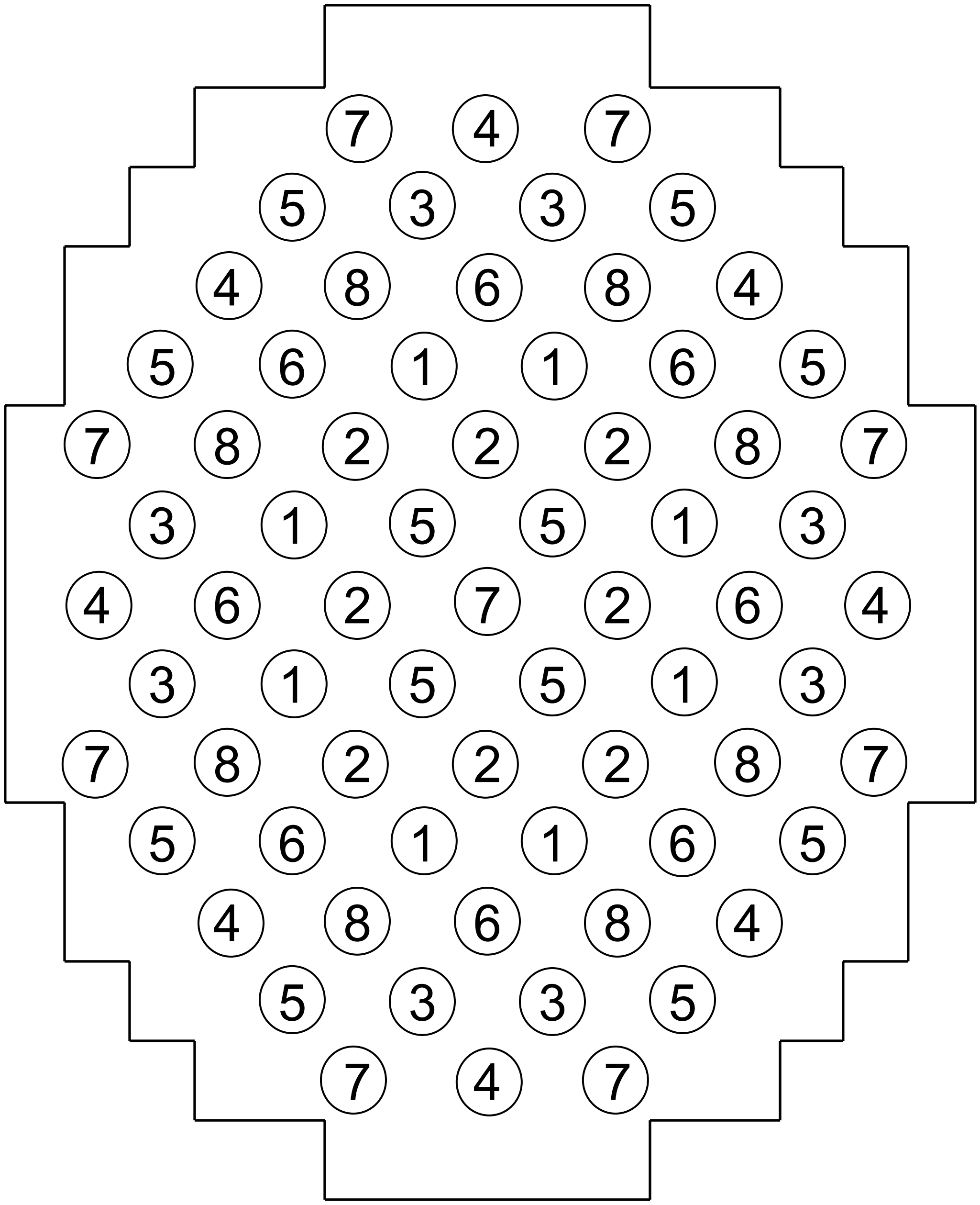
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

EJECTED ROD WORTH – EOL, HOT, FULL POWER

BASED ON DRAWING NO	SHEET	REV.
	1	



GROUP	NO. RODS	WORTH, %ΔK/K	PURPOSE
1	8	1.82	Safety
2	8	3.19	Safety
3	8	.86	Safety
4	8	1.28	Safety
5	12	1.42	Doppler
6	8	1.58	Doppler
7	9	.99	Transient
TOTAL	61	11.14	

SAR FIGURE NO. 3-7

AMENDMENT 20

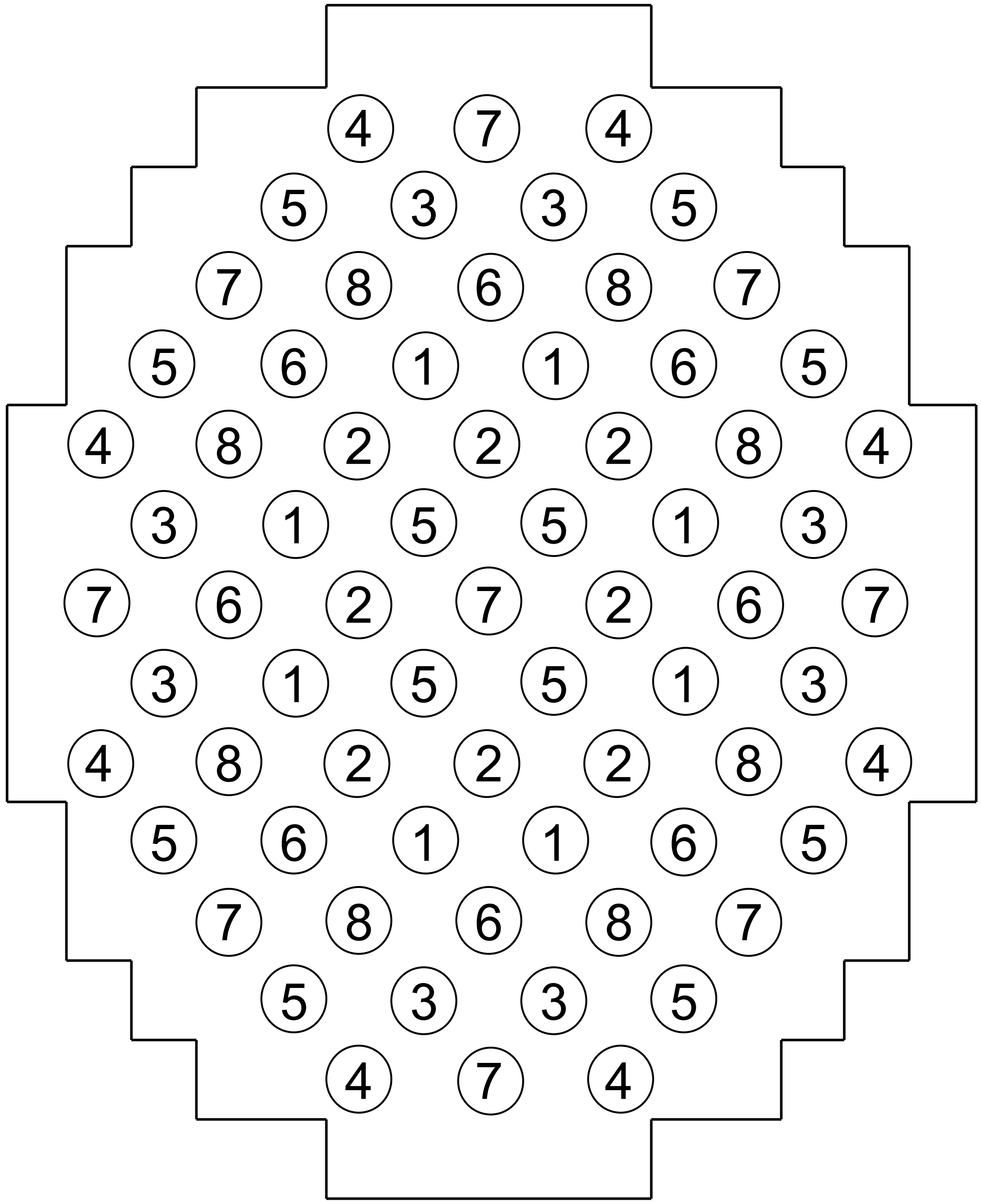
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ROD WORTH OF GROUPS WITHDRAWN SEQUENTIALLY (BOL)

BASED ON DRAWING NO	SHEET	REV.
	1	



GROUP	NO. RODS	WORTH, %ΔK/K	PURPOSE
1	8	1.04	Safety
2	8	2.62	Safety
3	8	0.96	Safety
4	8	1.32	Safety
5	12	1.71	Doppler
6	8	1.17	Doppler
7	9	1.37	Transient
TOTAL	61	10.19	

SAR FIGURE NO. 3-8

AMENDMENT 20

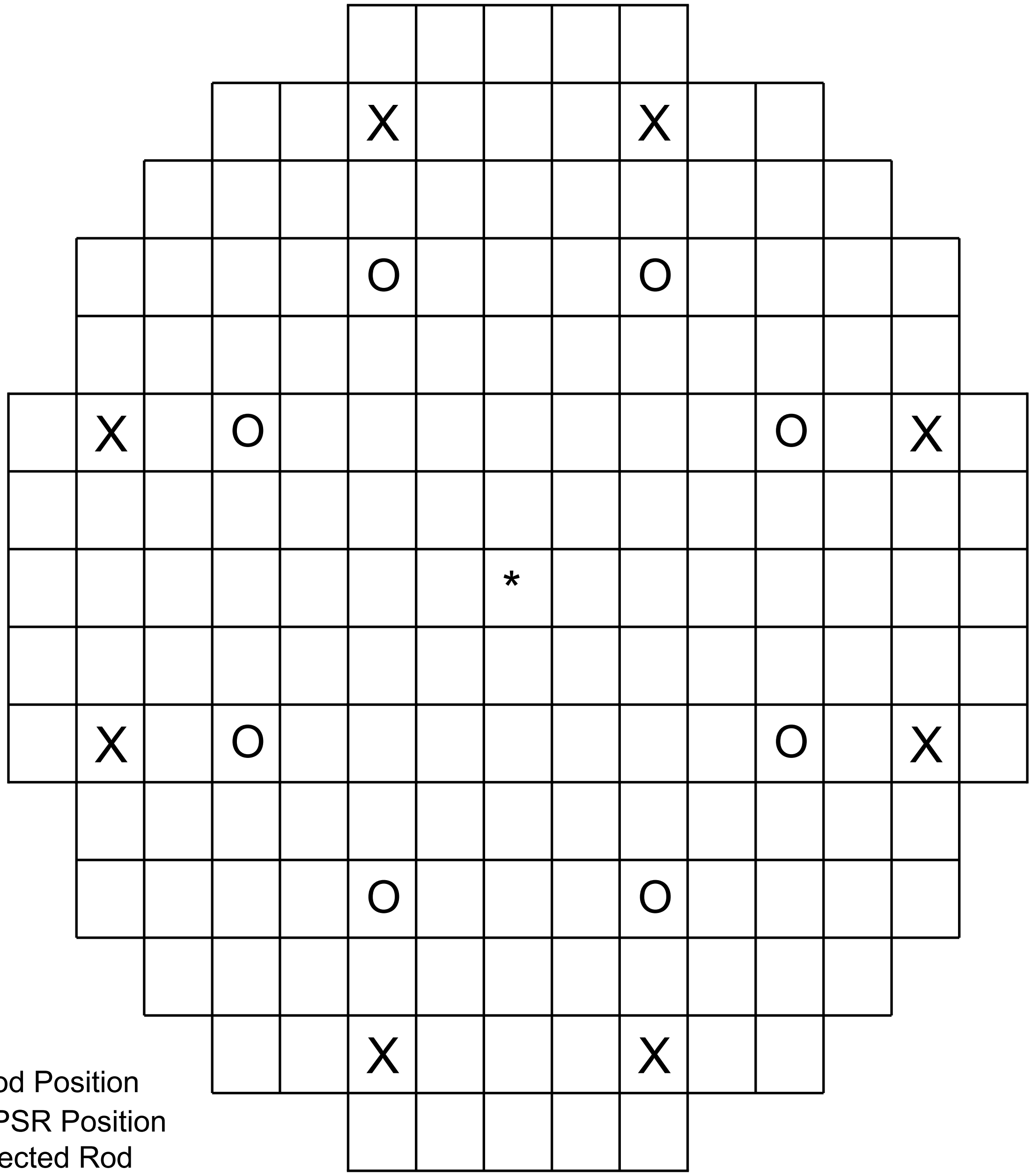
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ROD WORTH OF GROUPS WITHDRAWN SEQUENTIALLY (EOL)

BASED ON DRAWING NO	SHEET	REV.
	1	



X - Rod Position
O - APSR Position
* - Ejected Rod

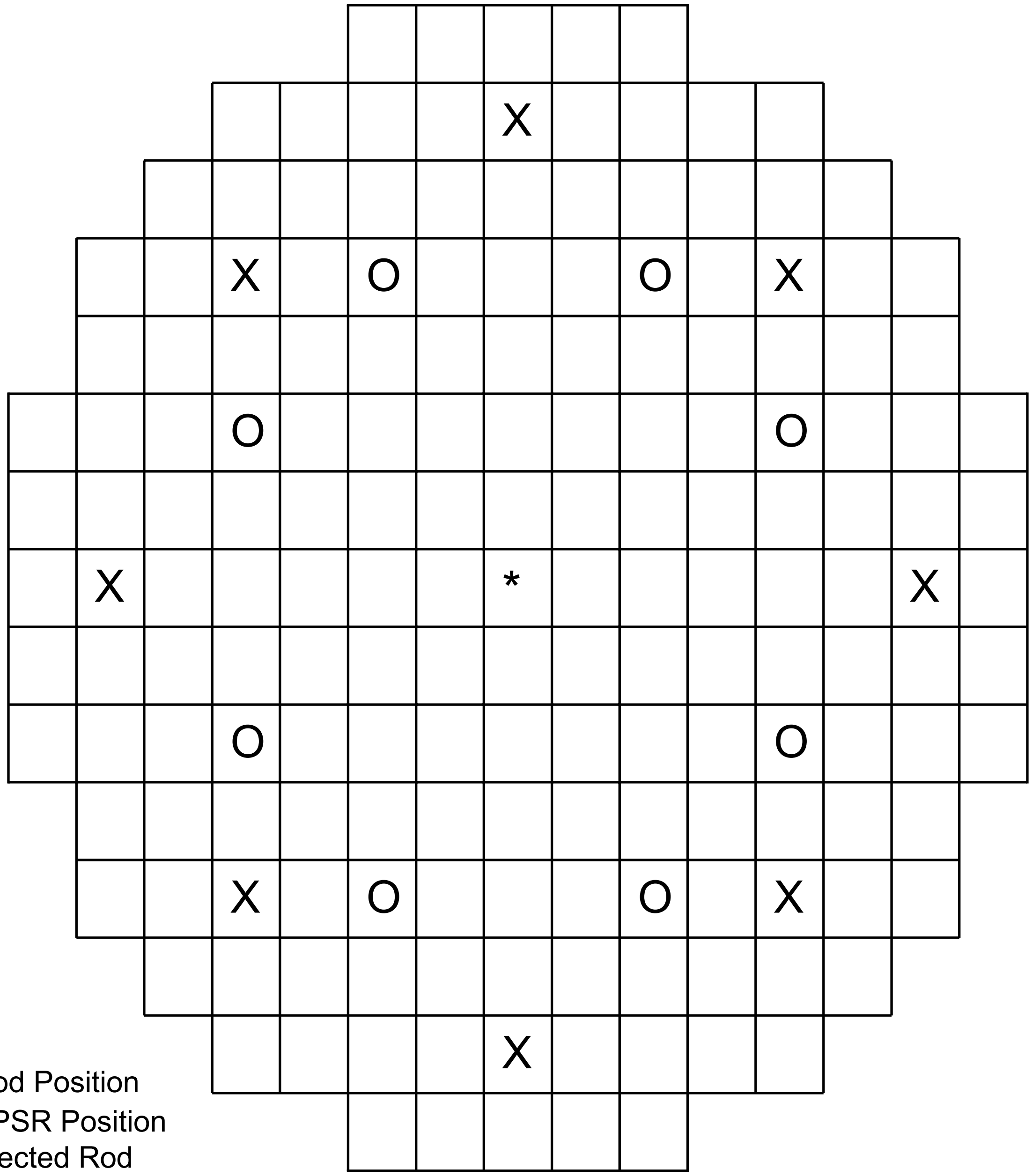
EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%ΔK/K)
Transient Bank	.27
Transient Bank & APSR's	.49

SAR FIGURE NO. 3-9

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
EJECTED ROD WORTH – BOL, HOT, FULL POWER	BASED ON DRAWING NO	SHEET 1	REV.



X - Rod Position
O - APSR Position
* - Ejected Rod

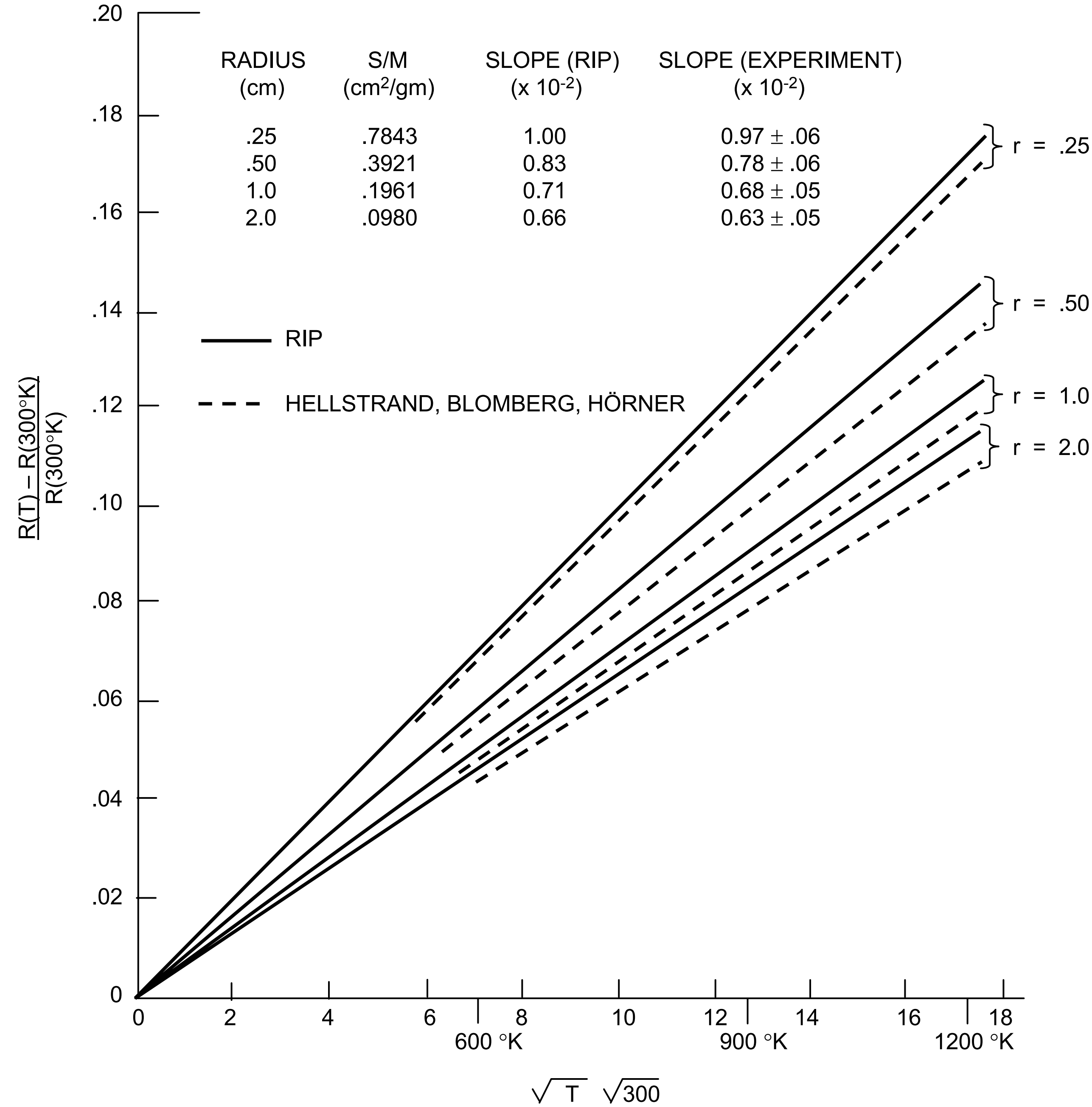
EJECTED ROD WORTH

RODS IN	WORTH OF EJECTED ROD (%ΔK/K)
Transient Bank	.37
Transient Bank & APSR's	.67

SAR FIGURE NO. 3-10

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
EJECTED ROD WORTH – EOL, HOT, FULL POWER	BASED ON DRAWING NO	SHEET 1	REV.

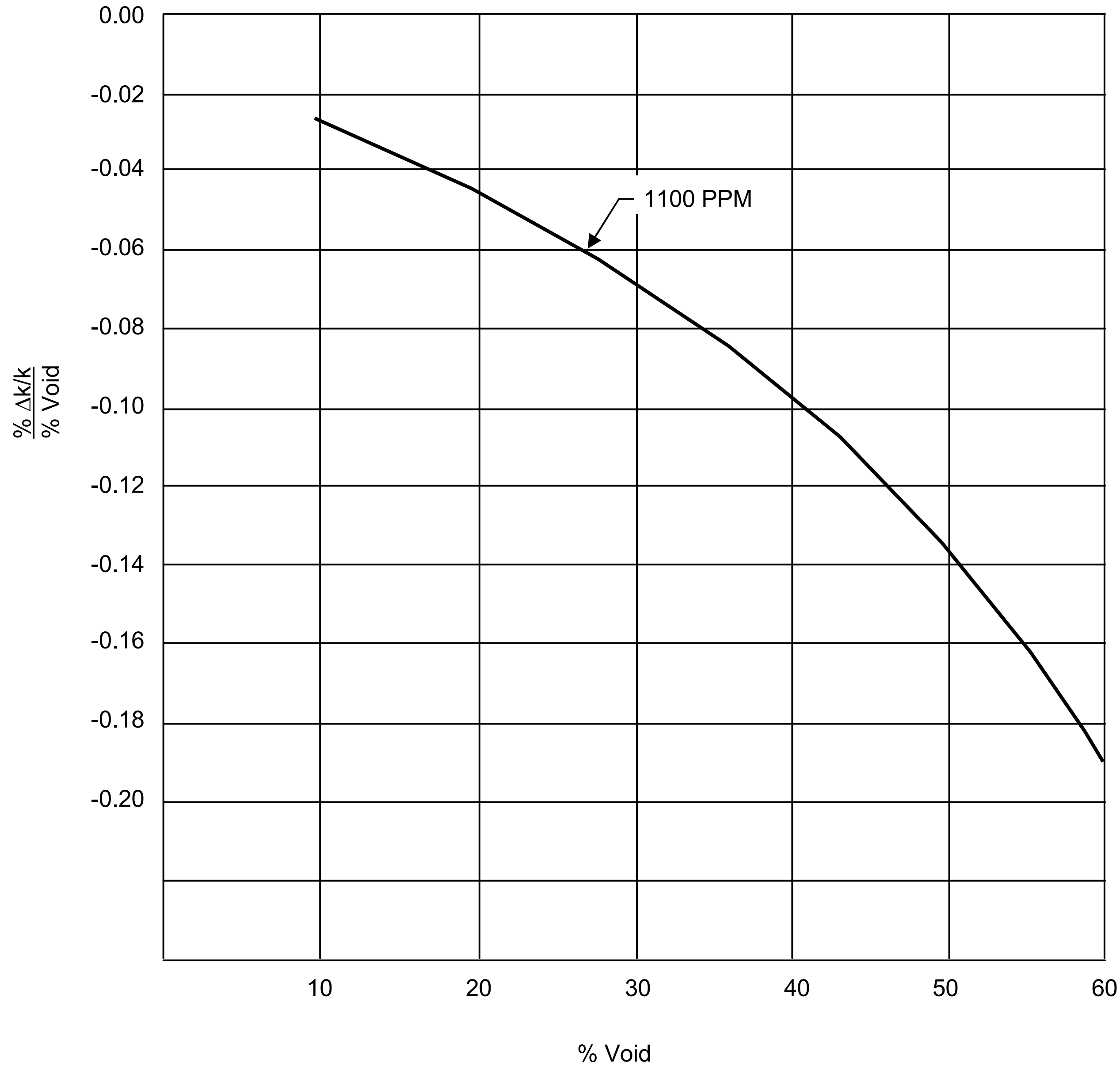


SAR FIGURE NO. 3-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

FRACTIONAL CHANGE IN THE RESONANCE INTEGRAL FUNCTION OF $\sqrt{T}$ - $\sqrt{300}$ FOR UO_2 RODS	BASED ON DRAWING NO	SHEET	REV.

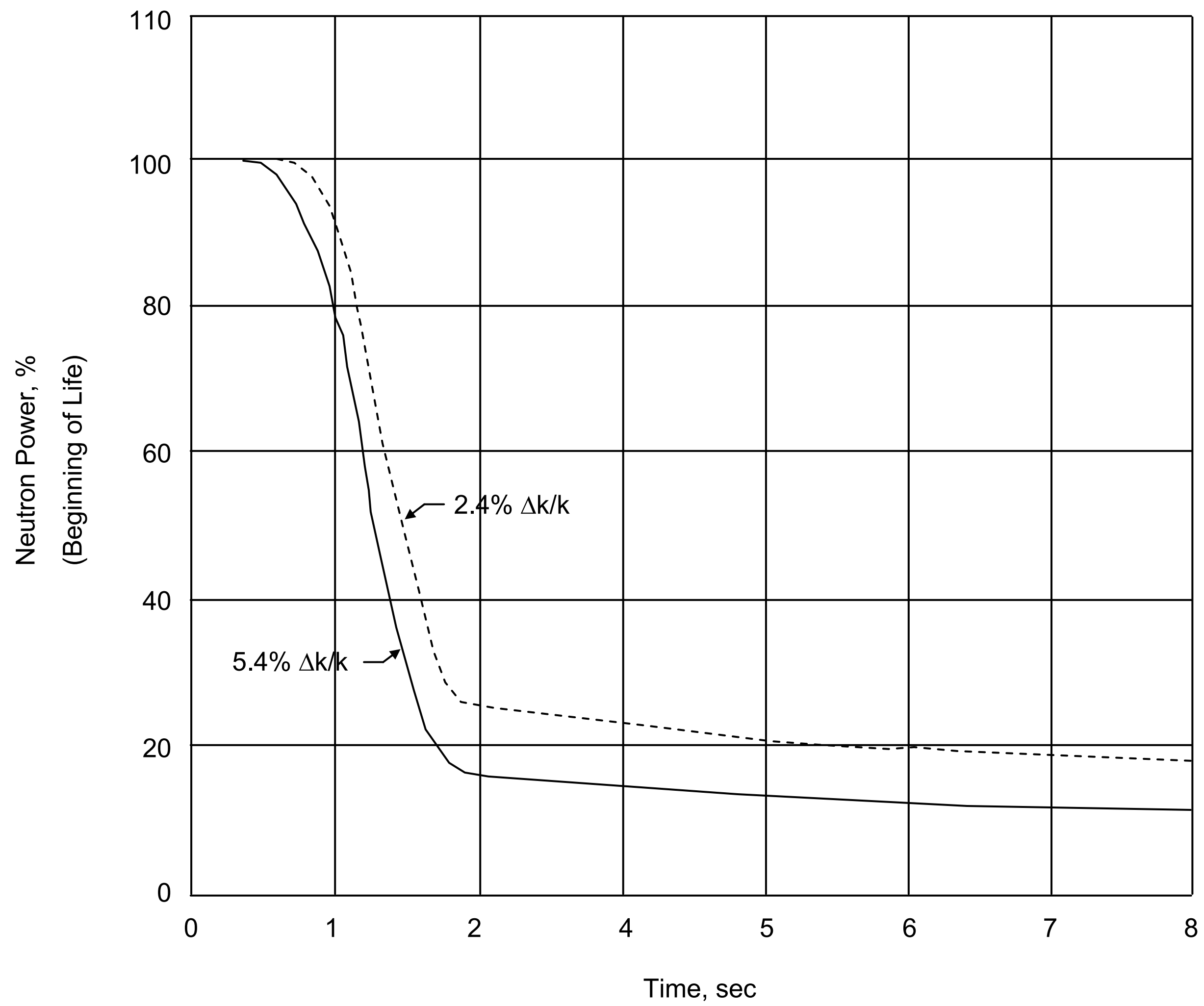


SAR FIGURE NO. 3-12

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

UNIFORM VOID COEFFICIENT FOR 177 ASSEMBLY CORE	BASED ON DRAWING NO	SHEET	REV.



PERCENT NEUTRON POWER VERSUS TIME FOLLOWING TRIP

SAR FIGURE NO. 3-13

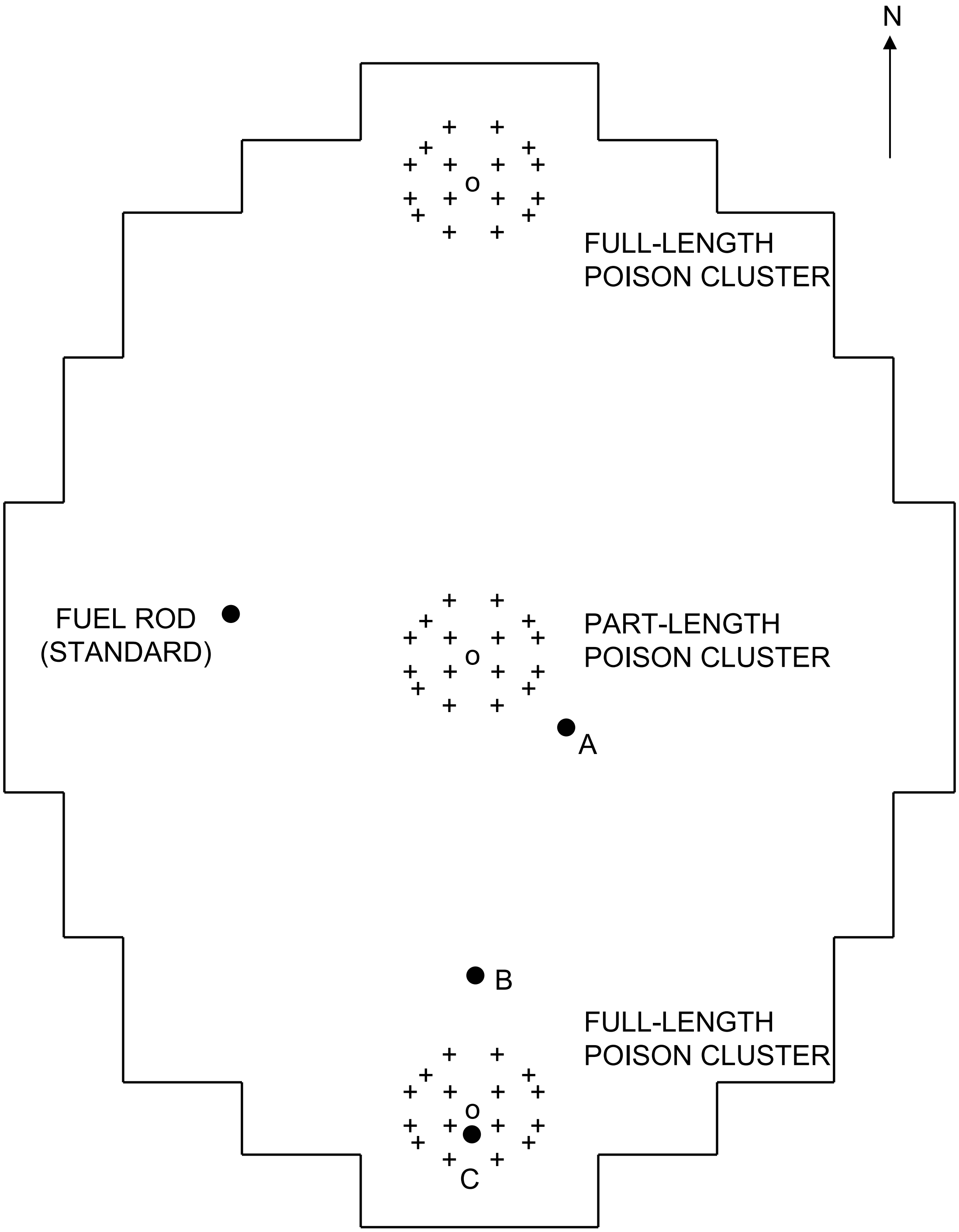
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

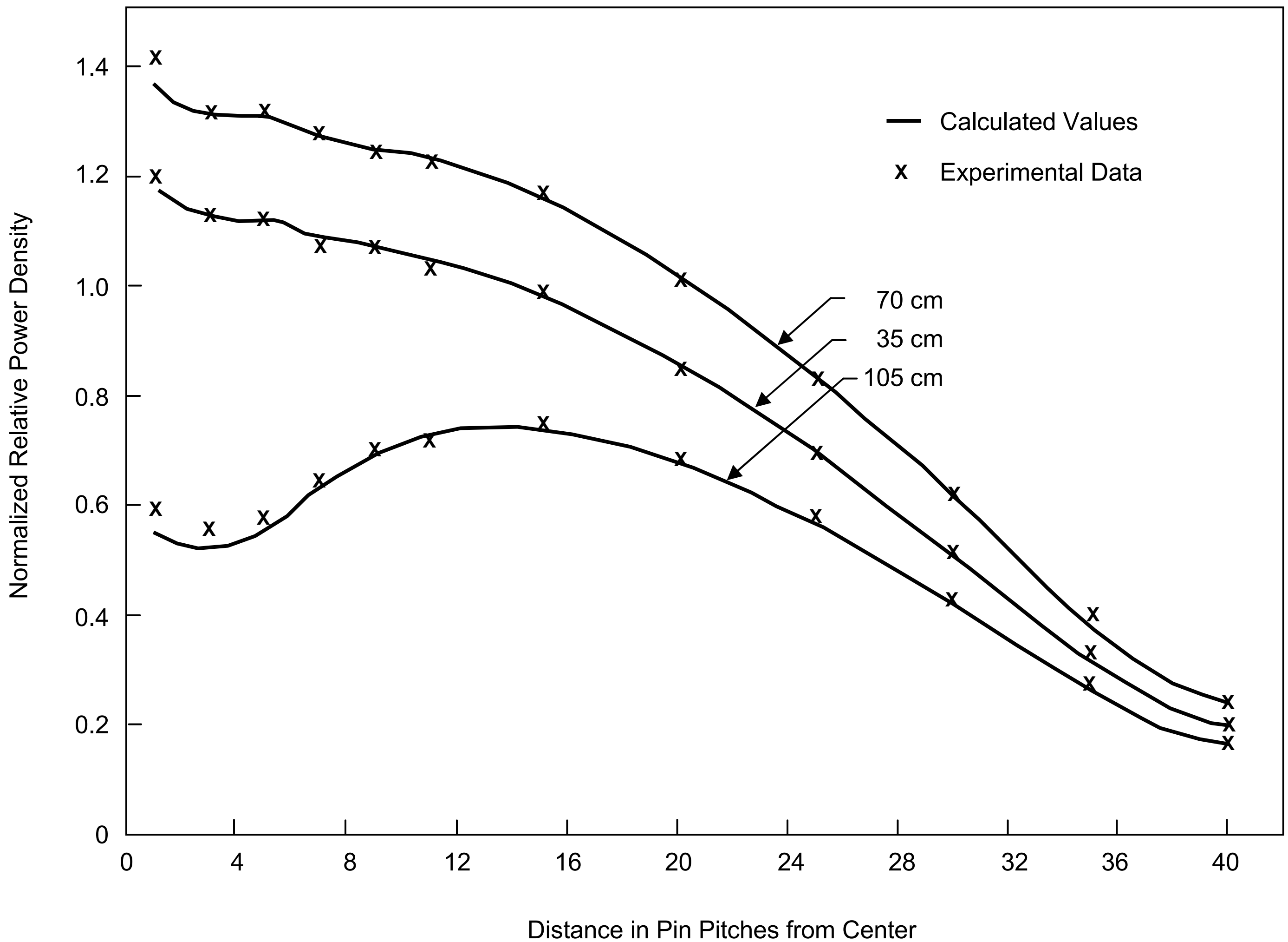
BASED ON DRAWING NO	SHEET	REV.



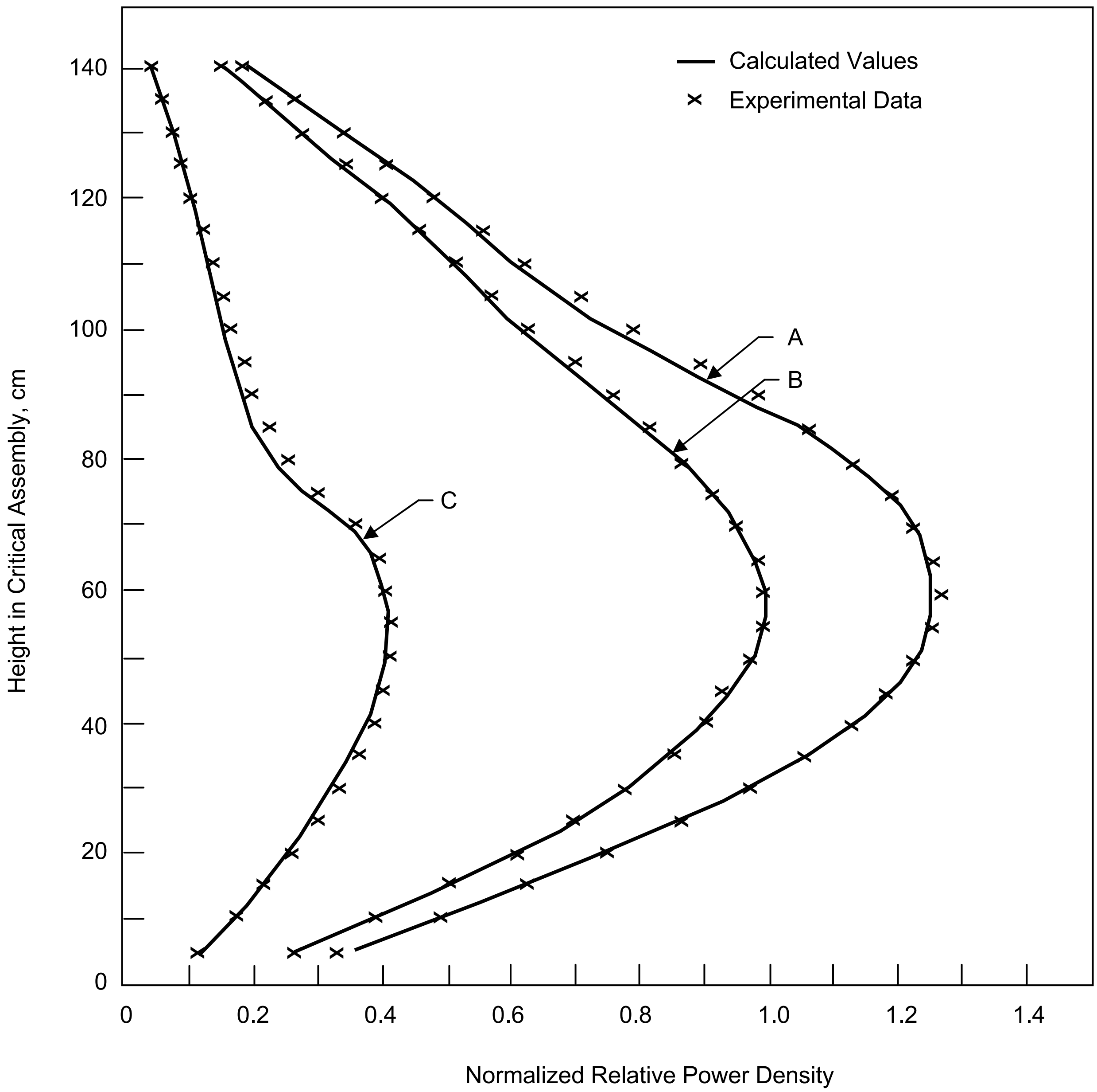
SAR FIGURE NO. 3-14

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LOADING DIAGRAM FOR CORE IV-H	BASED ON DRAWING NO	SHEET 1 / REV.



RADIAL POWER DISTRIBUTION AT THREE LEVELS FROM BOTTOM OF CORE			SAR FIGURE NO. 3-15		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
			BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-16

AMENDMENT 20

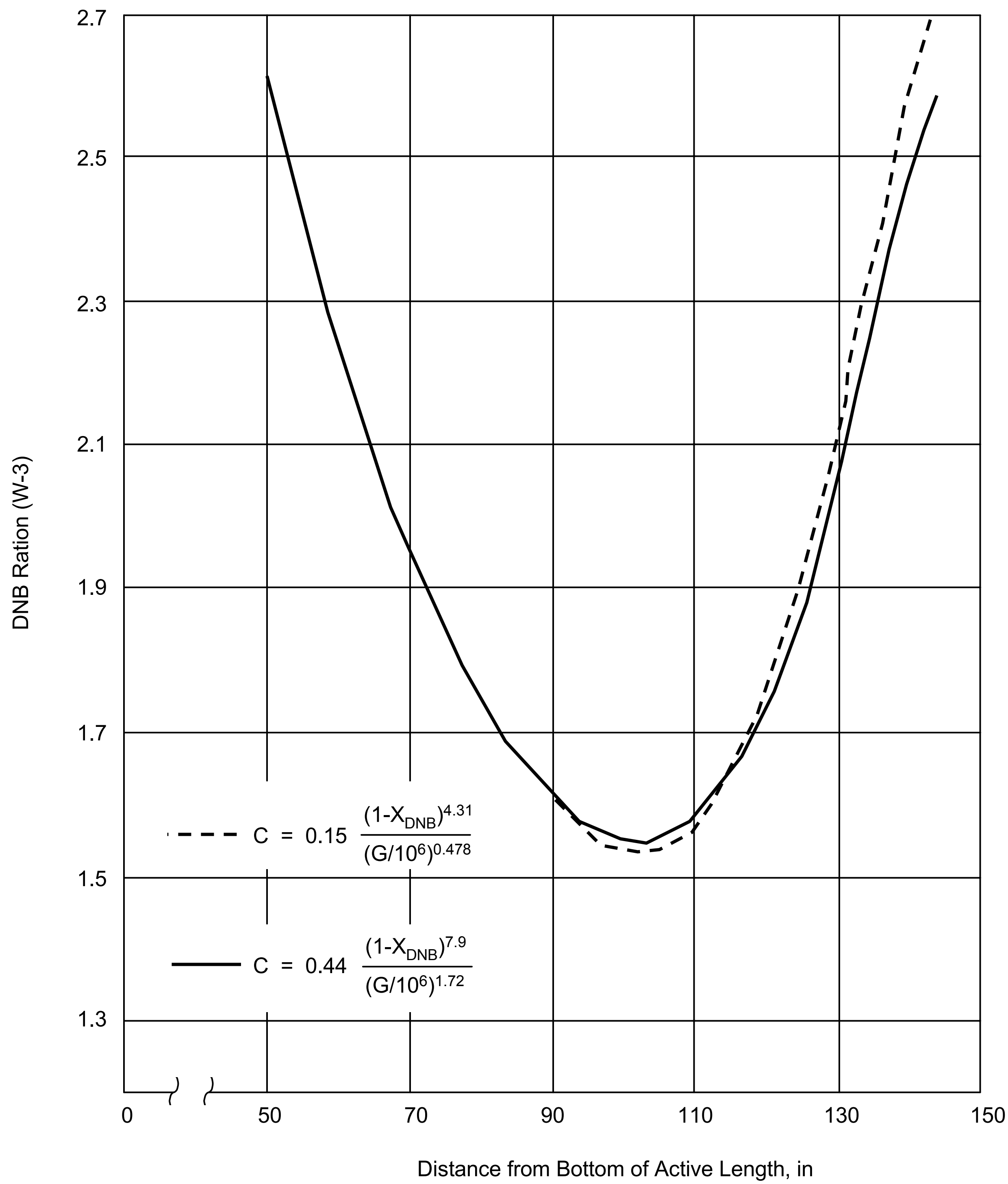
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

AXIAL POWER DISTRIBUTION AT THREE SELECT POSITION IN THE CORE

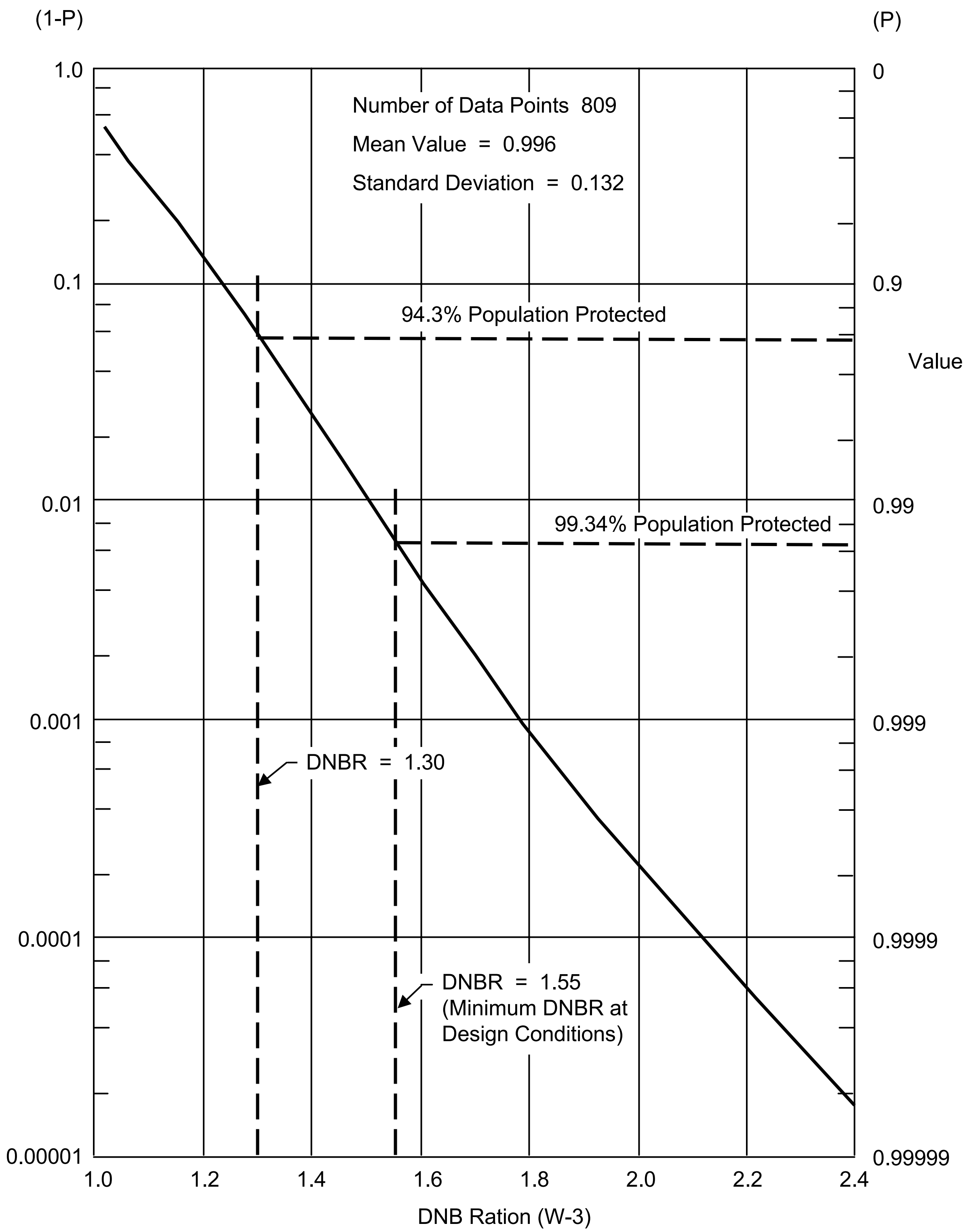
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-17

AMENDMENT 20

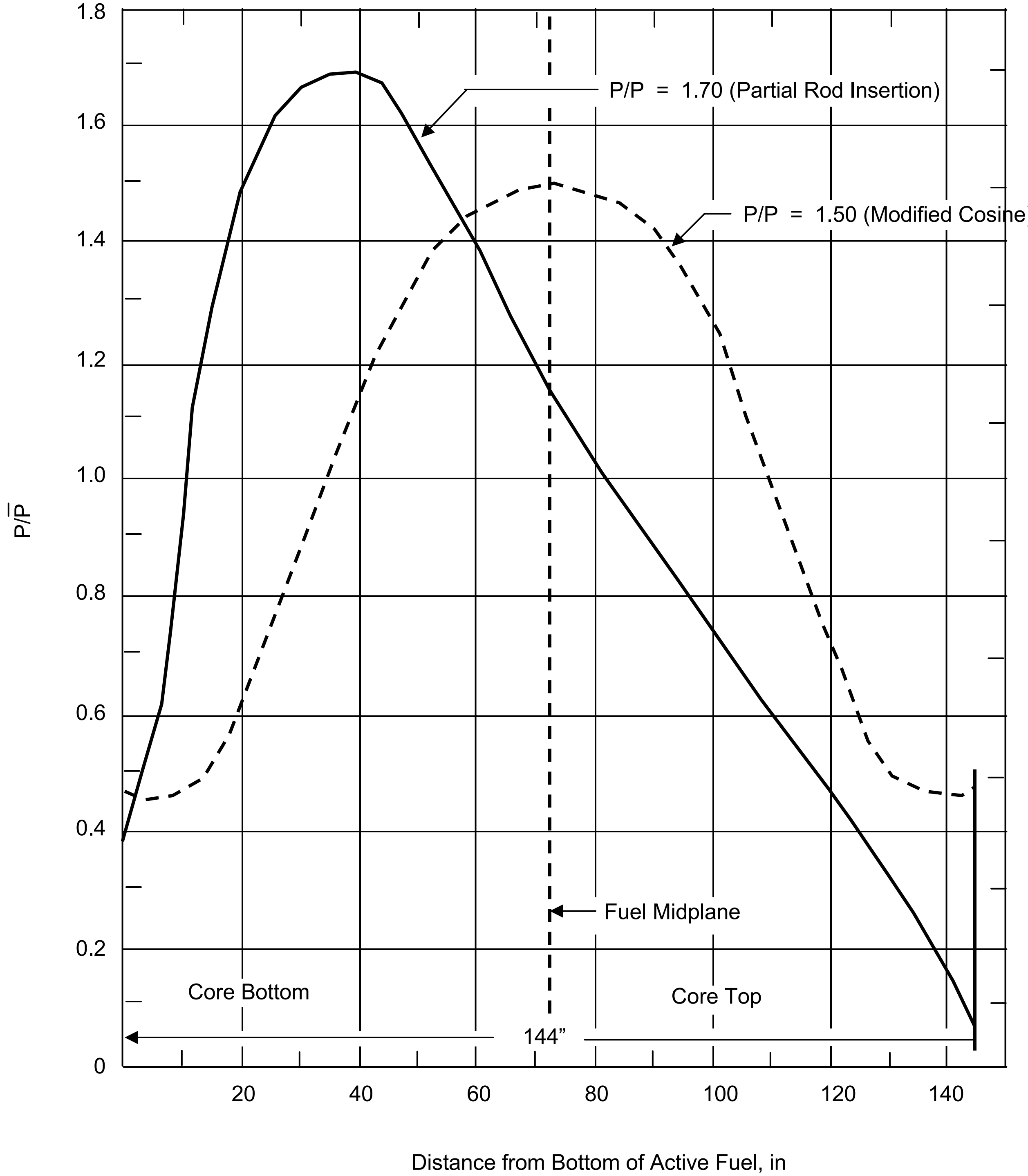
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
DNB RATIOS (W-3) IN THE HOT CELL FOR THE NEW AND OLD C FACTORS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-18

AMENDMENT 20

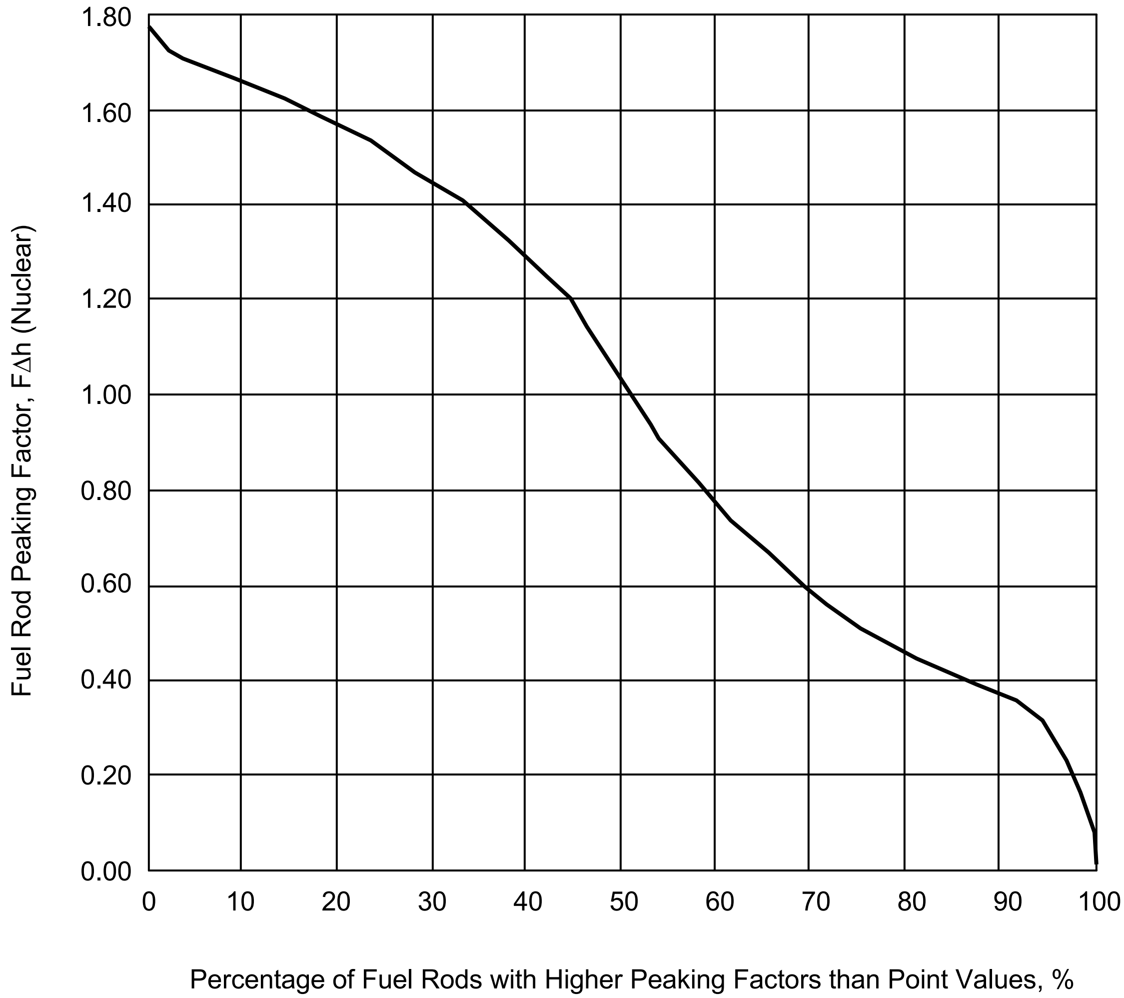
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
POPULATION PROTECTED, P, AND 1-P VERSUS DNB RATIO (W-3)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-19

AMENDMENT 20

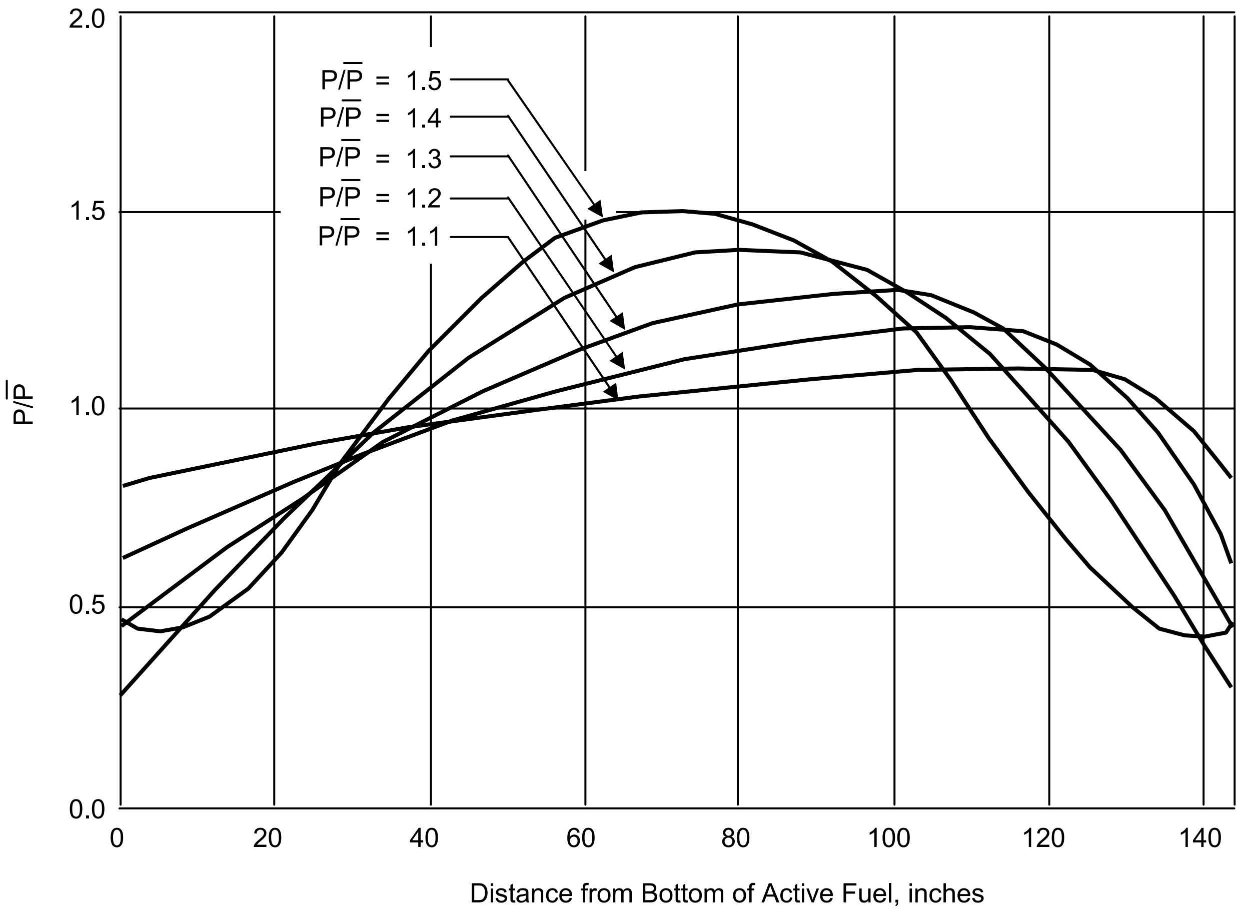
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
POWER SHAPE REFLECTING INCREASED AXIAL POWER PEAK FOR 144-INCH CORE	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-20

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
DISTRIBUTION OF FUEL ROD PEAKING FACTOR	BASED ON DRAWING NO	SHEET / REV.



ALLOWABLE AXIAL POWER DISTRIBUTIONS WITH MAXIMUM DESIGN RADIAL-LOCAL PEAKING FACTOR (1.78)

SAR FIGURE NO. 3-21

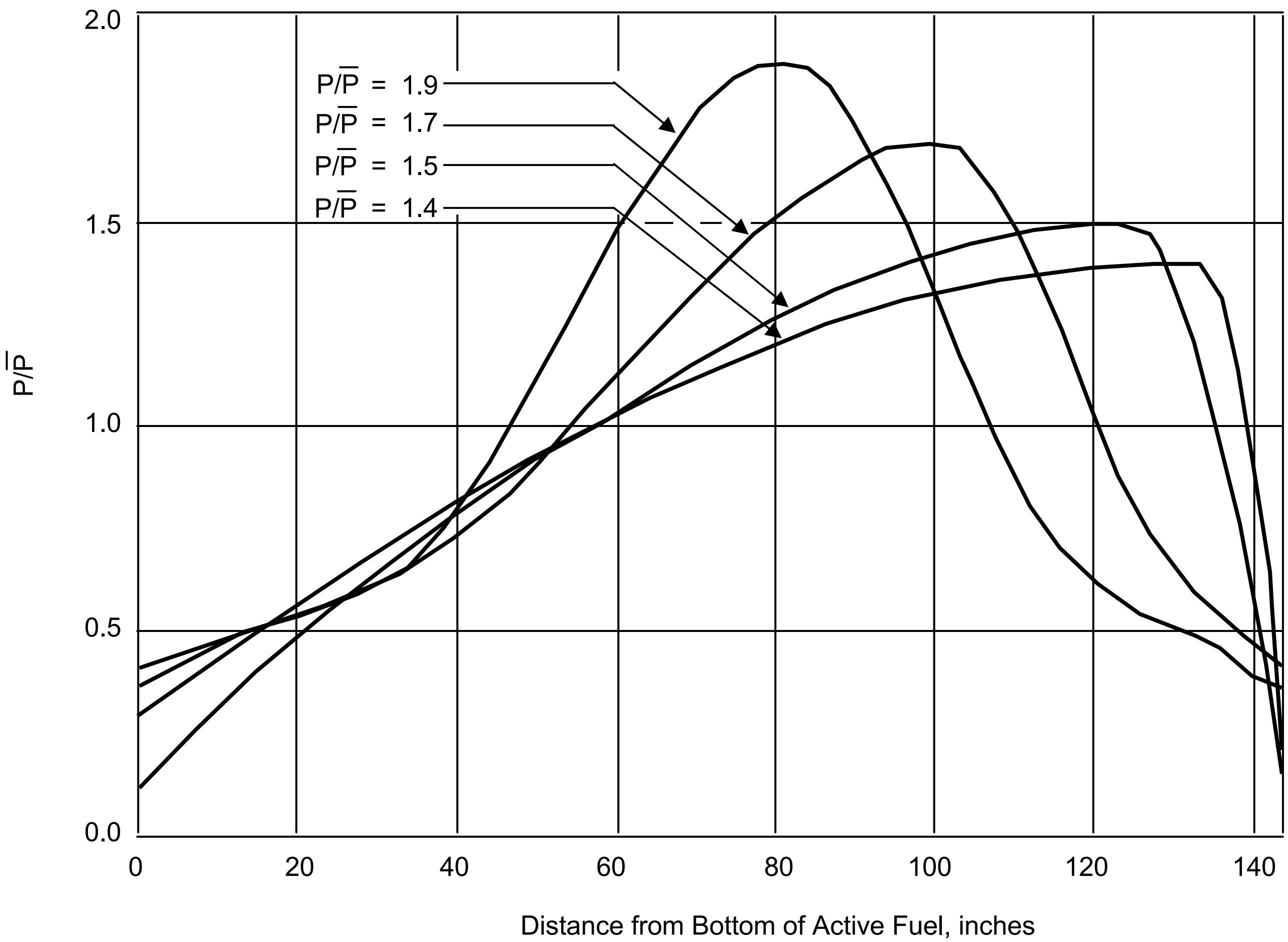
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

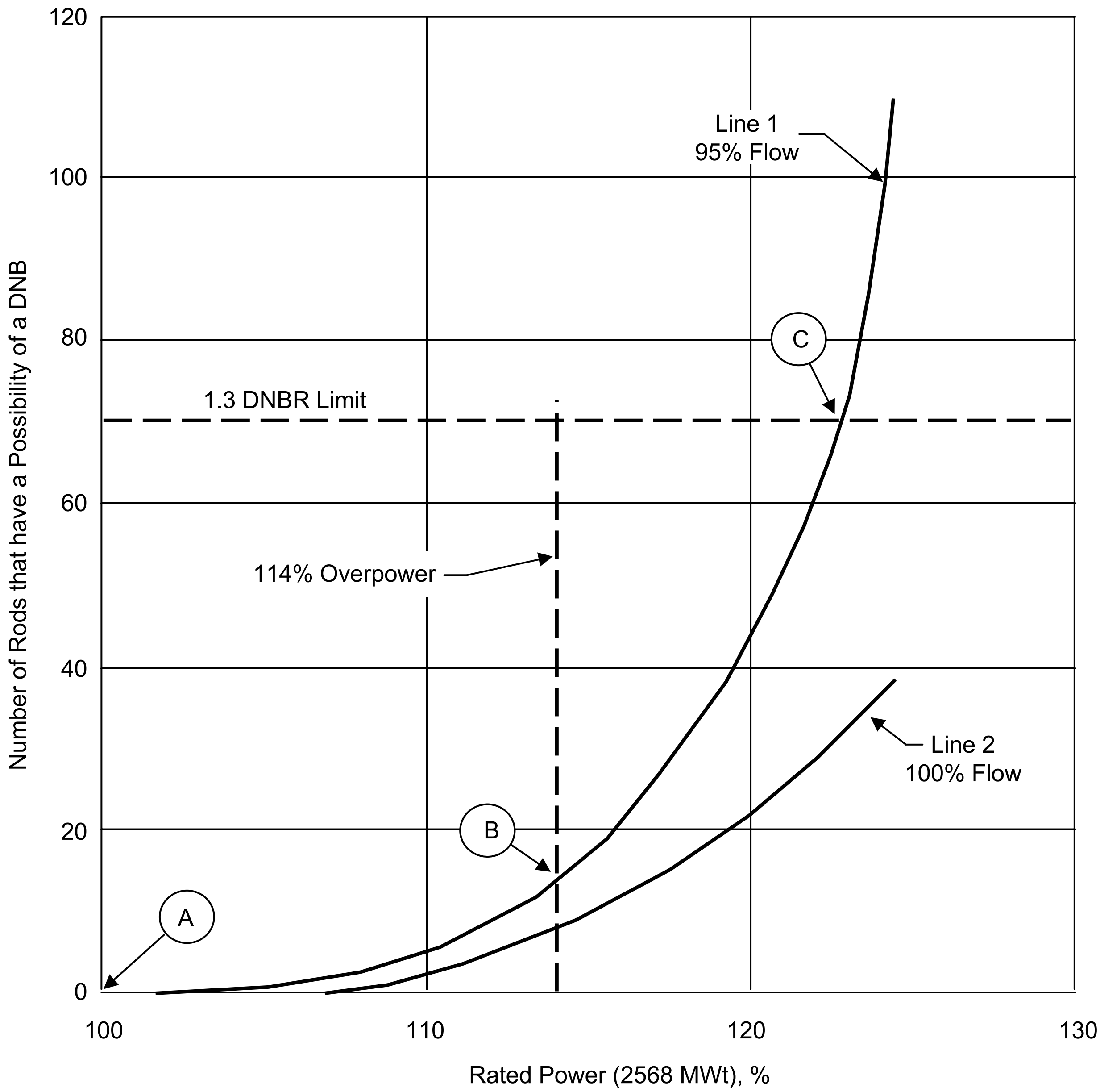
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



ALLOWABLE AXIAL POWER DISTRIBUTIONS WITH 1.65 RADIAL-LOCAL PEAKING FACTOR	SAR FIGURE NO. 3-22
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-23

AMENDMENT 20

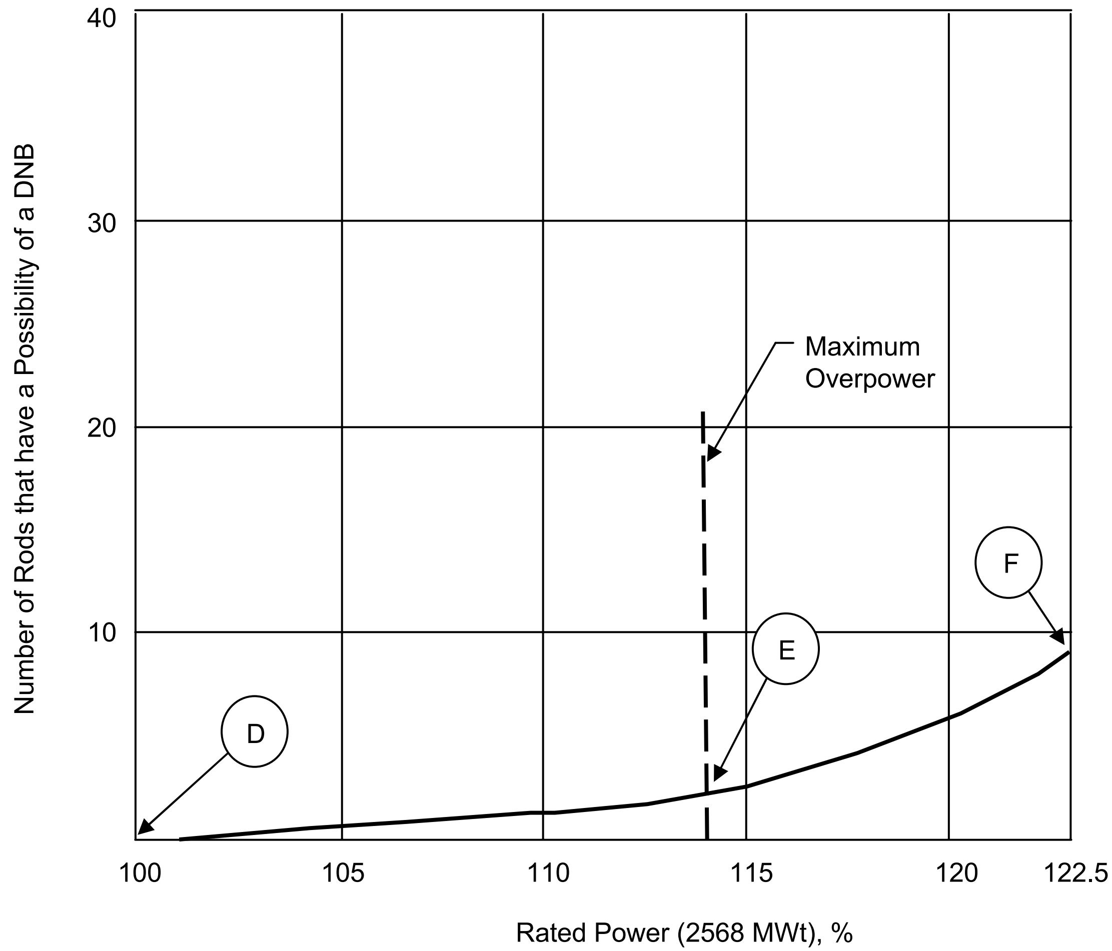
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

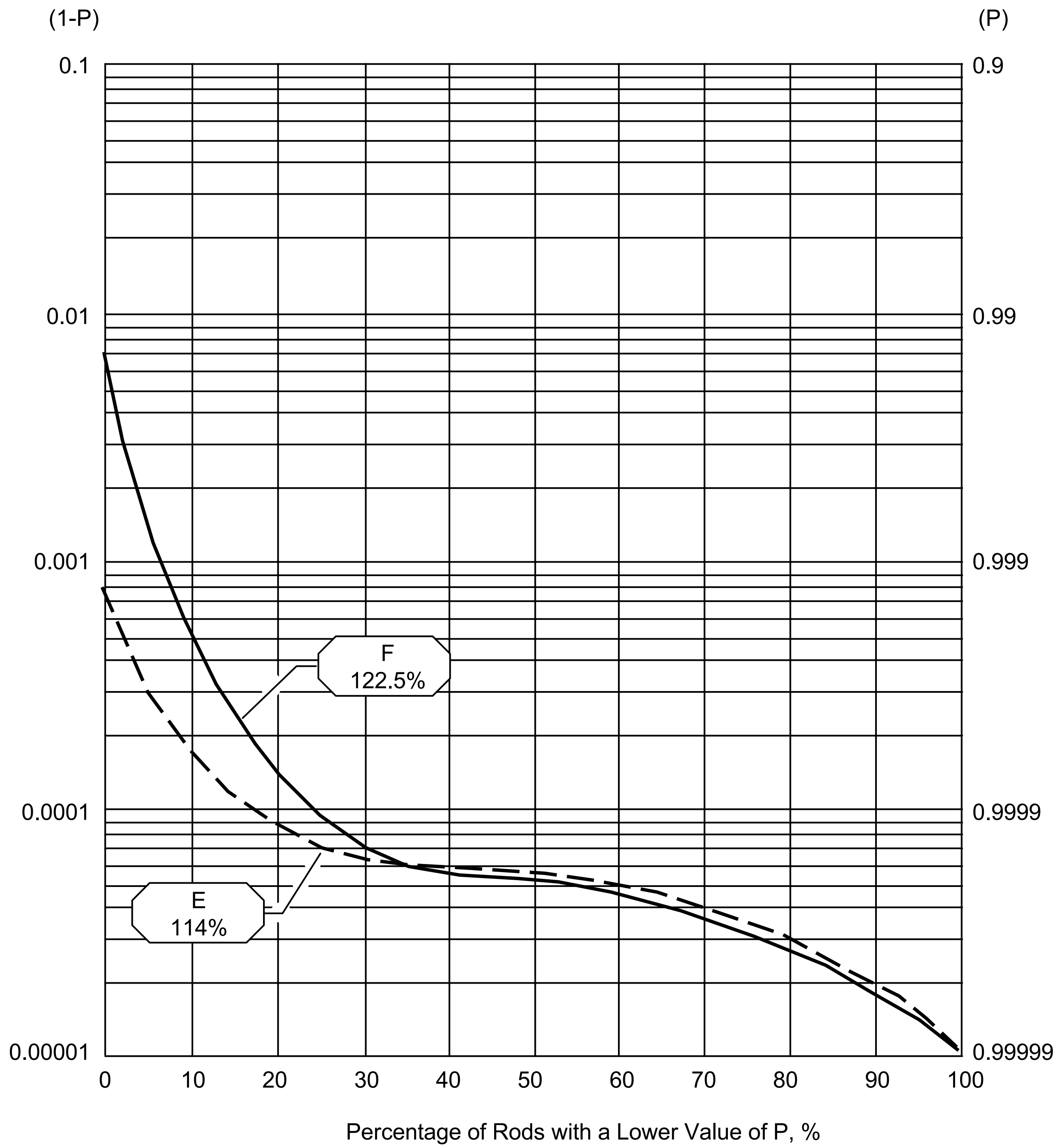
SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

POSSIBLE FUEL ROD DNB'S FOR MAXIMUM DESIGN CONDITIONS – 36, 816 – ROD CORE

BASED ON DRAWING NO	SHEET	REV.



POSSIBLE FUEL ROD DNB'S FORMOST PROBABLE CONDITIONS – 36,816 – ROD CORE			SAR FIGURE NO. 3-24
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



SAR FIGURE NO. 3-25

AMENDMENT 20

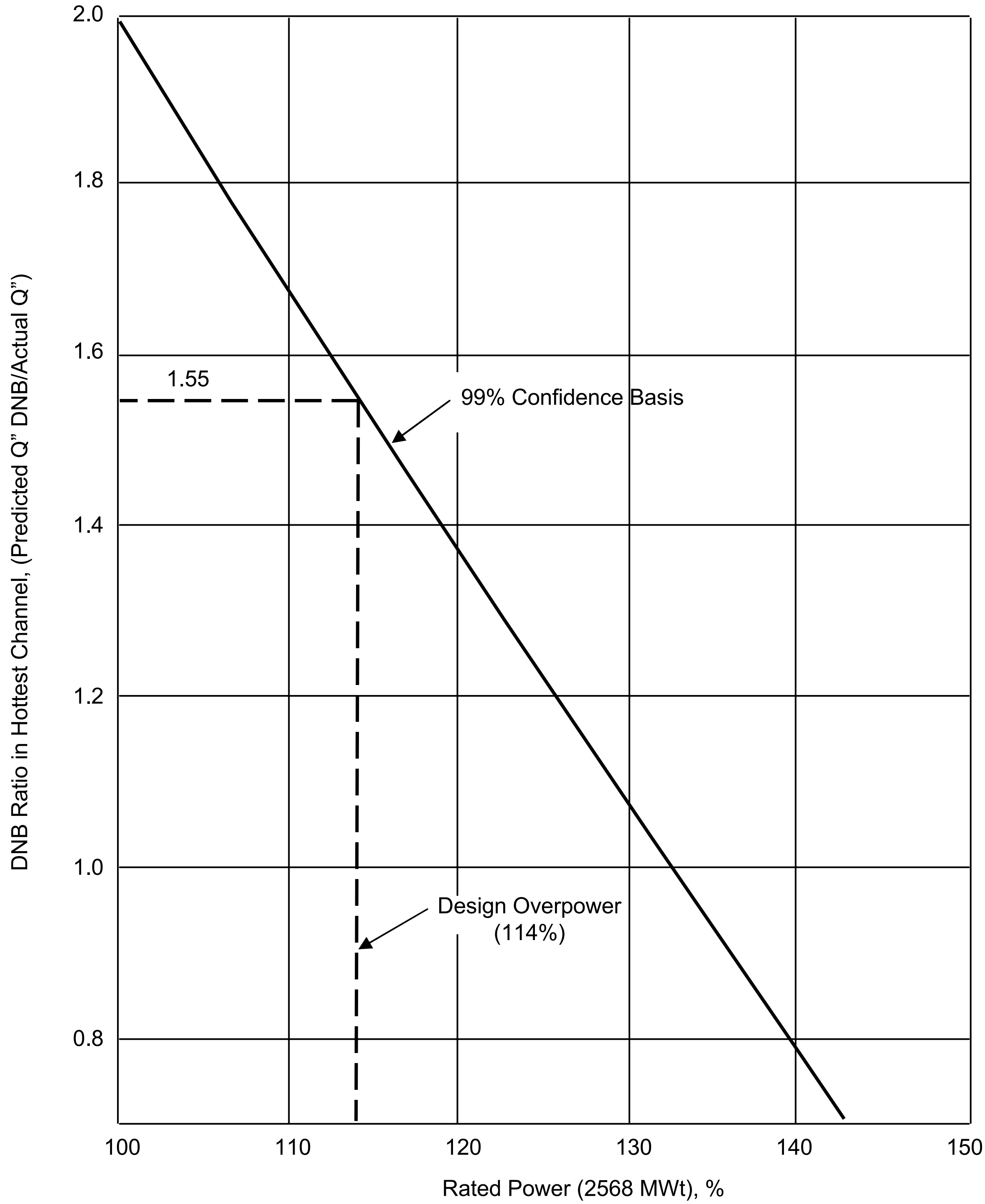
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

DISTRIBUTION OF POPULATION PROTECTED, P, AND 1-P VERSUS NUMBER OF RODS FOR MOST PROBABLE CONDTIONS

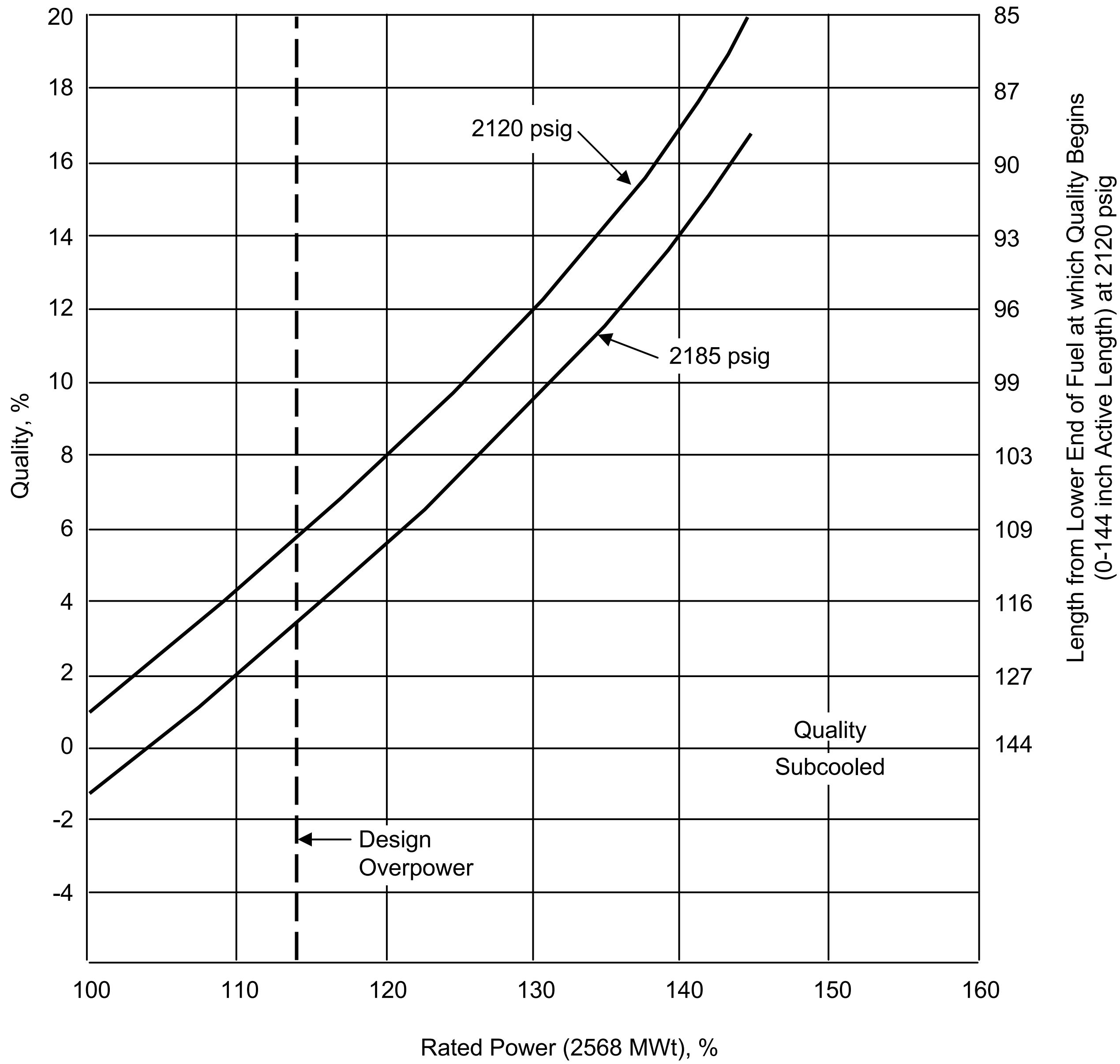
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-26

AMENDMENT 20

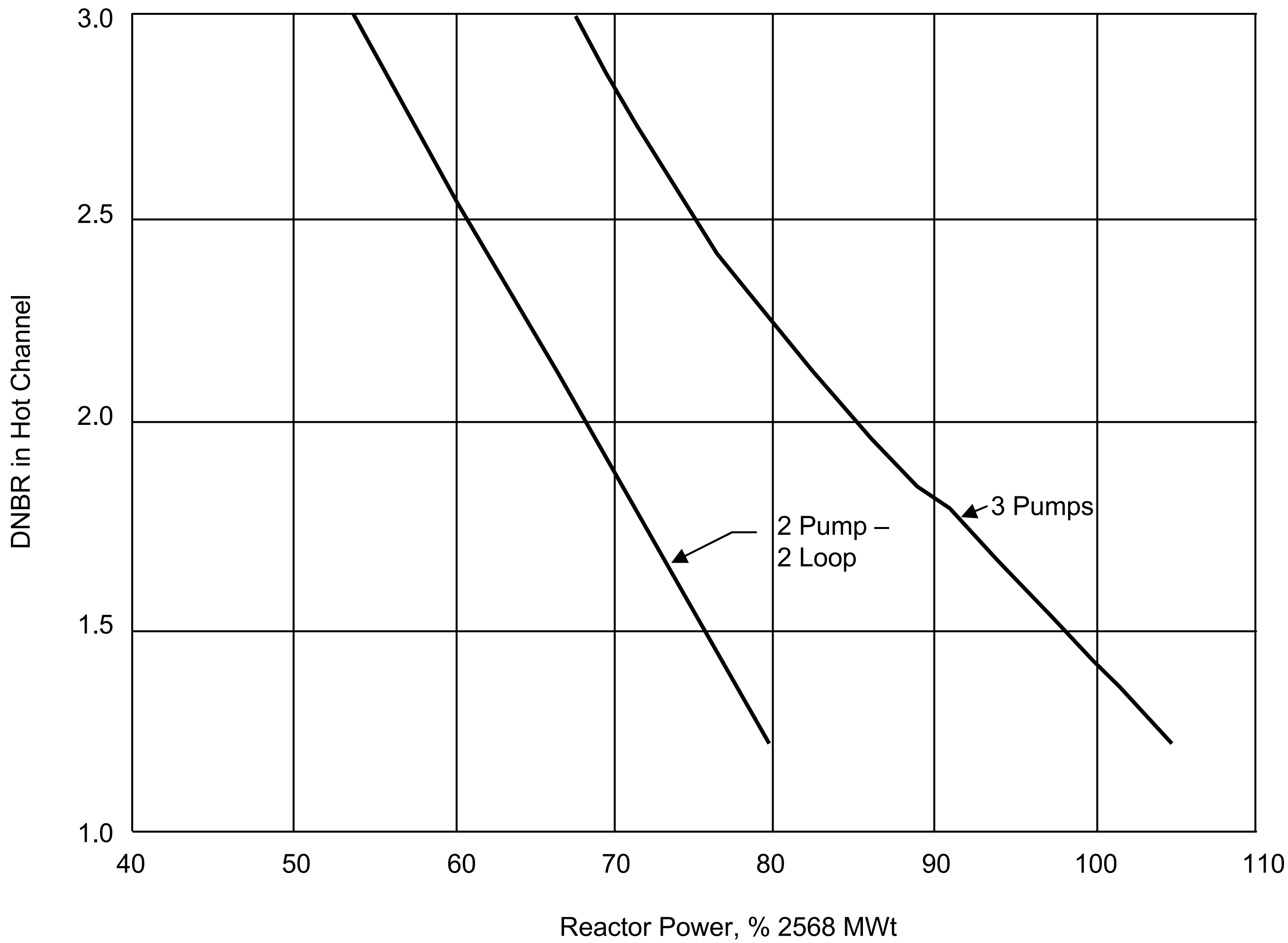
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
DNB RATIOS (W-3) IN HOT UNIT CELL VERSUS REACTOR POWER	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
MAXIMUM HOT CHANNEL EXIT QUALITY VERSUS REACTOR POWER	BASED ON DRAWING NO	SHEET	REV.



HOT CHANNEL DNB RATIO (W-3) VERSUS POWER FOR PARTIAL PUMP OPERATION

SAR FIGURE NO. 3-28

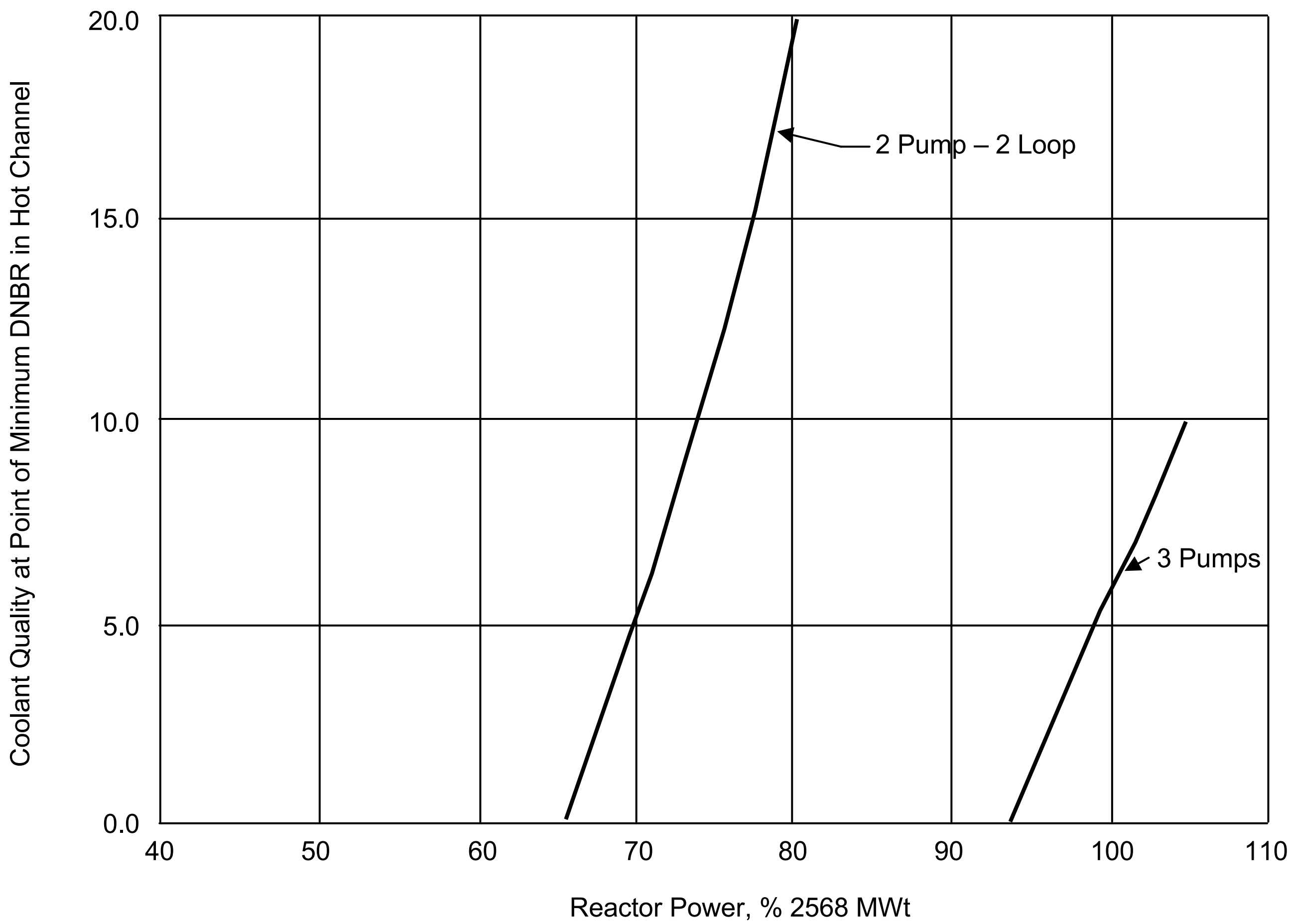
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



HOT CHANNEL QUALITY AT POINT OF MINIMUM DNBR VERSUS POWER FOR PARTIAL PUMP OPERATION

SAR FIGURE NO. 3-29

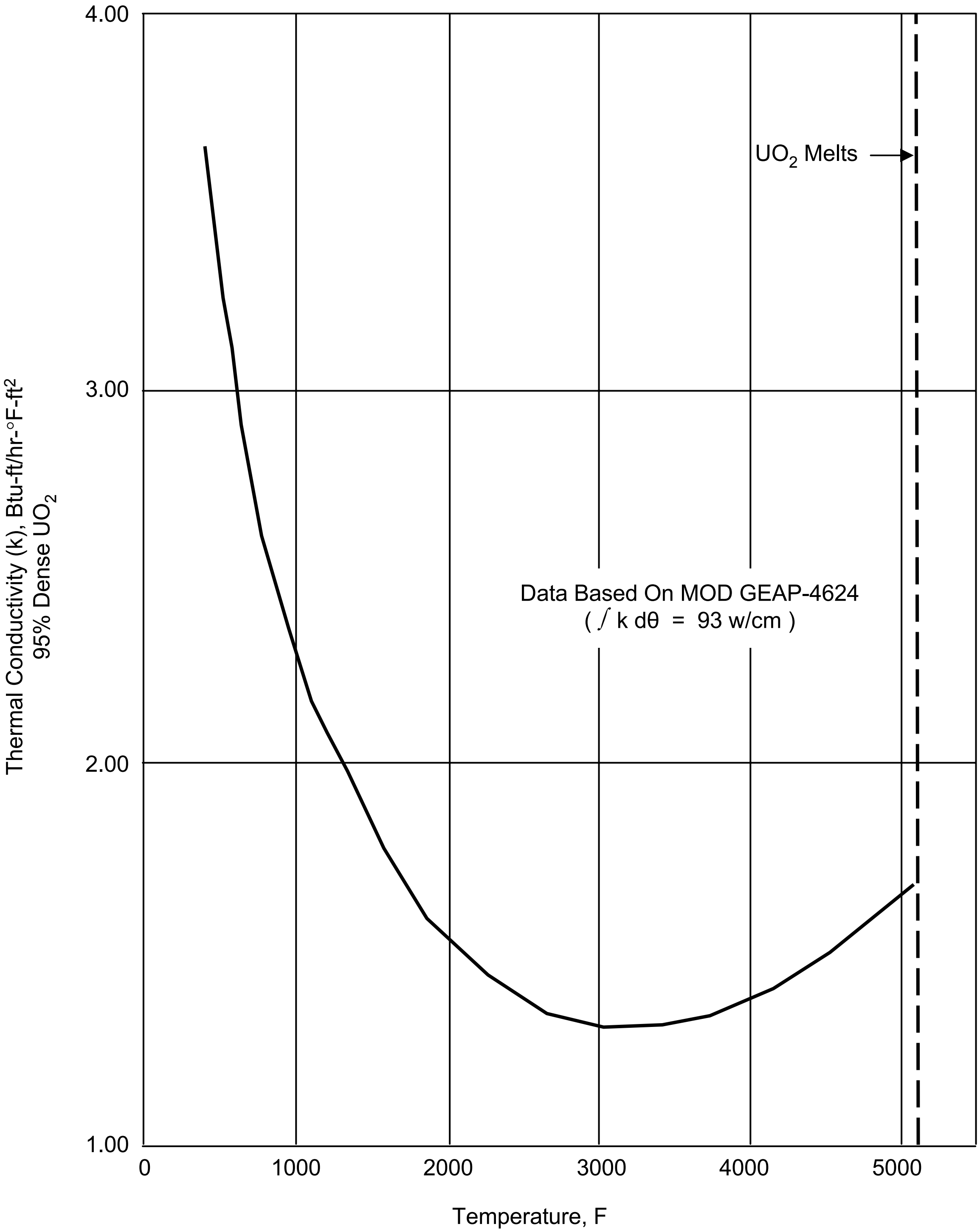
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

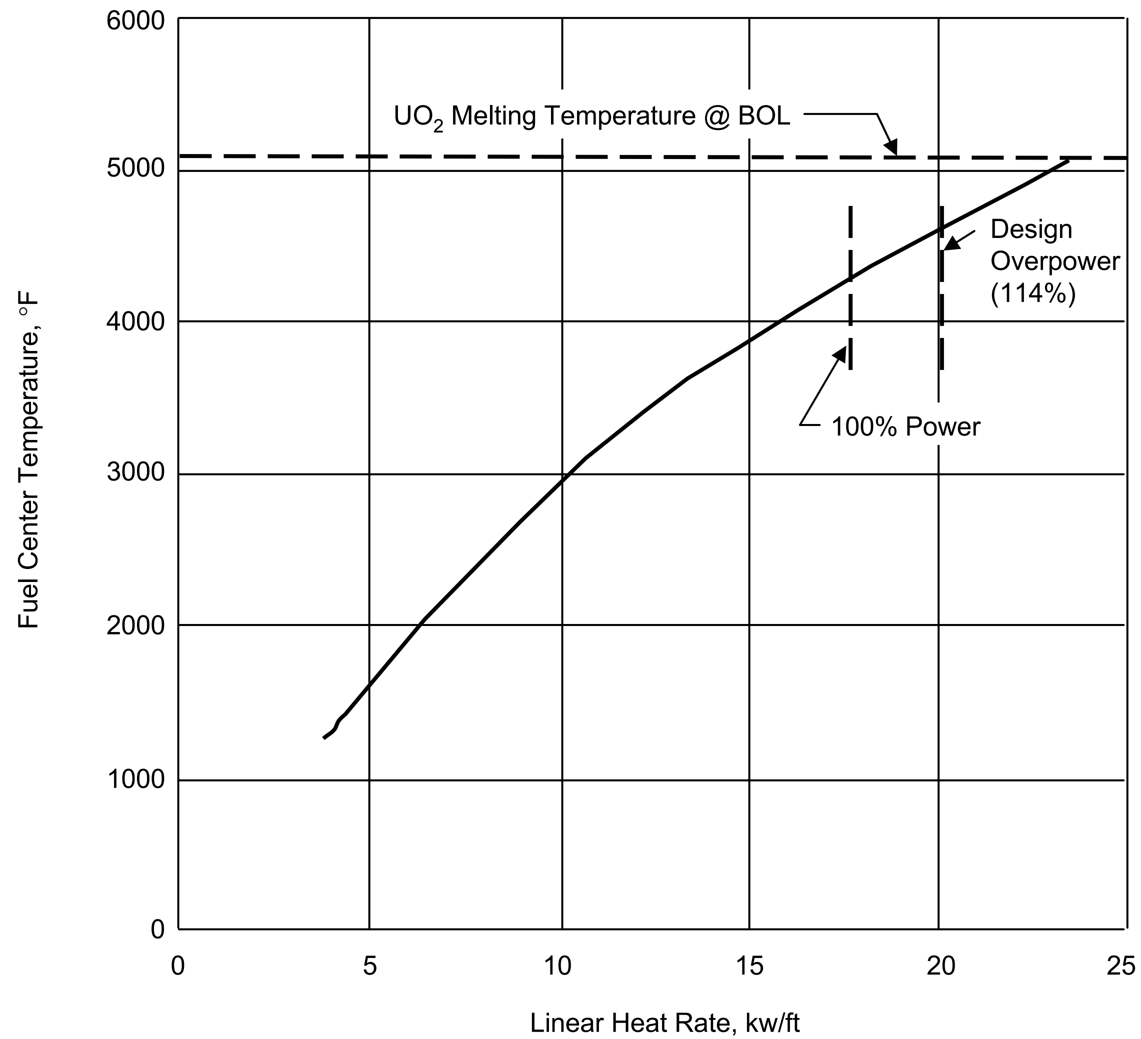
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-30

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
THERMAL CONDUCTIVITY OF UO_2	BASED ON DRAWING NO	SHEET	REV.

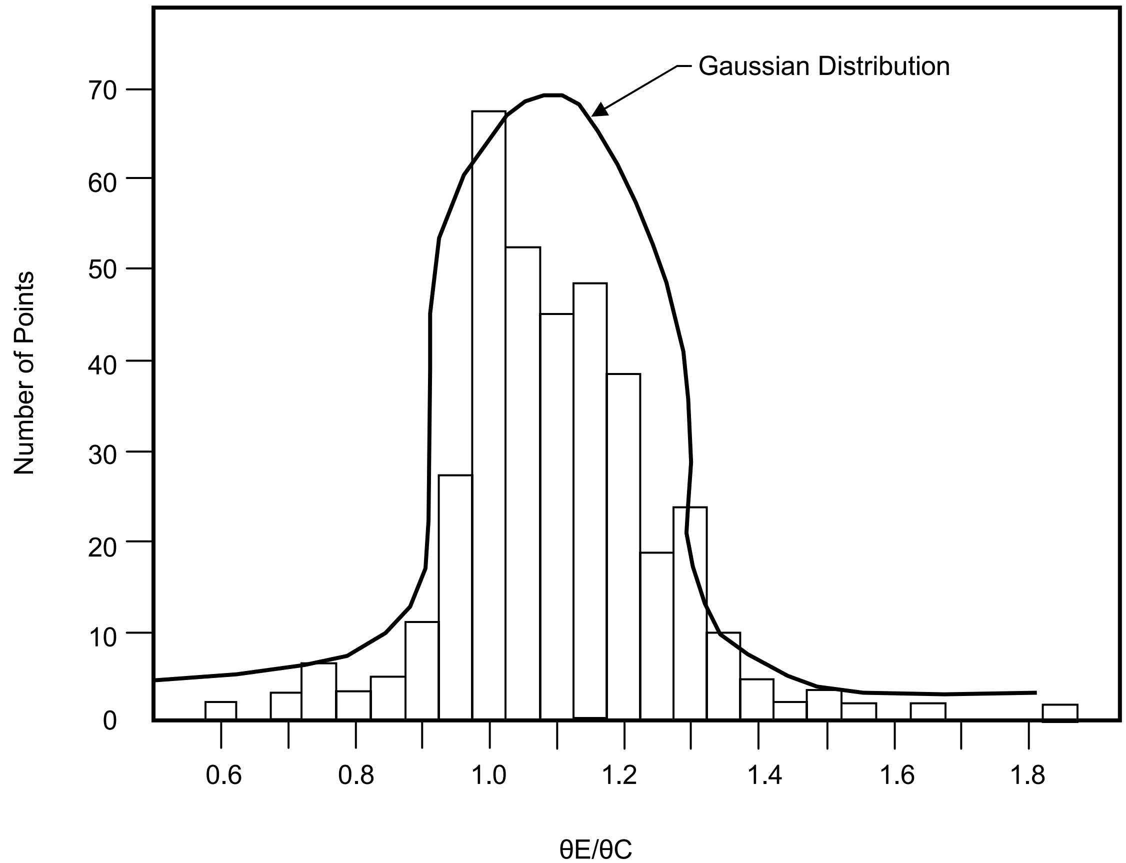


FUEL CENTER TEMPERATURE AT THE HOT SPOT VERSUS LINEAR HEAT RATE	SAR FIGURE NO. 3-31
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



NUMBER OF DATA POINTS VERSUS ΦE/ΦC

SAR FIGURE NO. 3-32

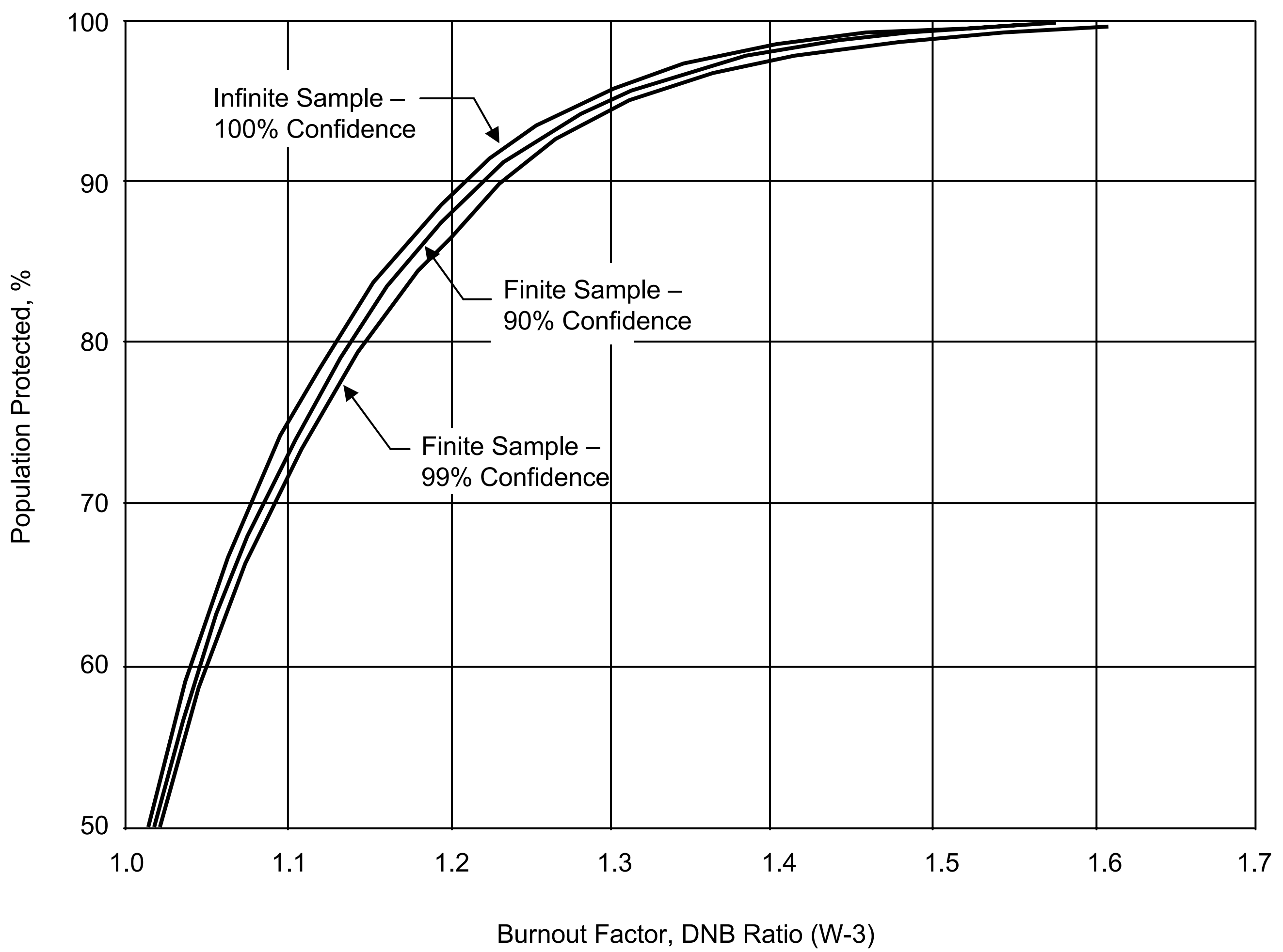
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



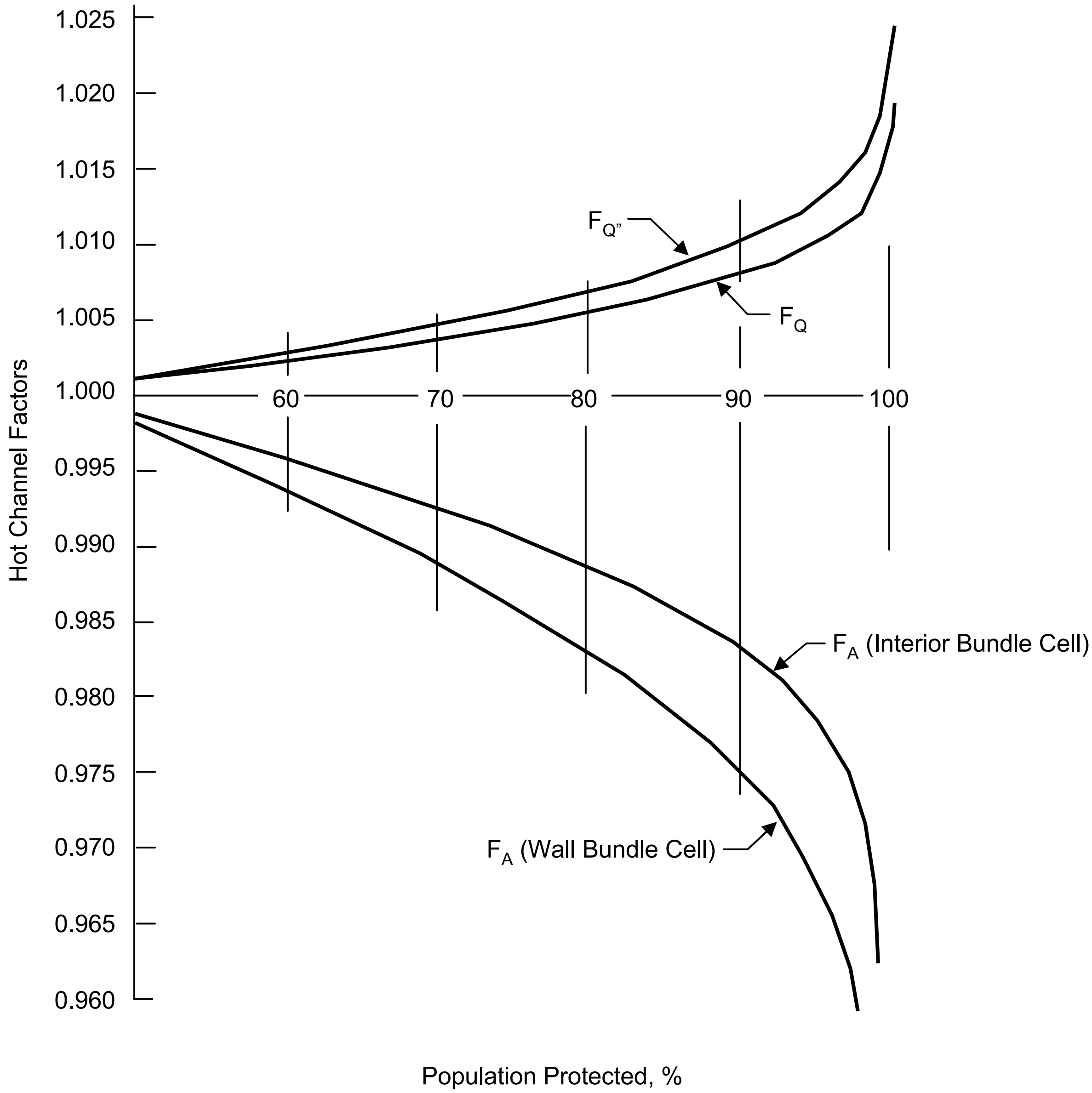
BURNOUT FACTOR (W-3) VERSUS POPULATION FOR VARIOUS CONFIDENCE LEVELS	SAR FIGURE NO. 3-33
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

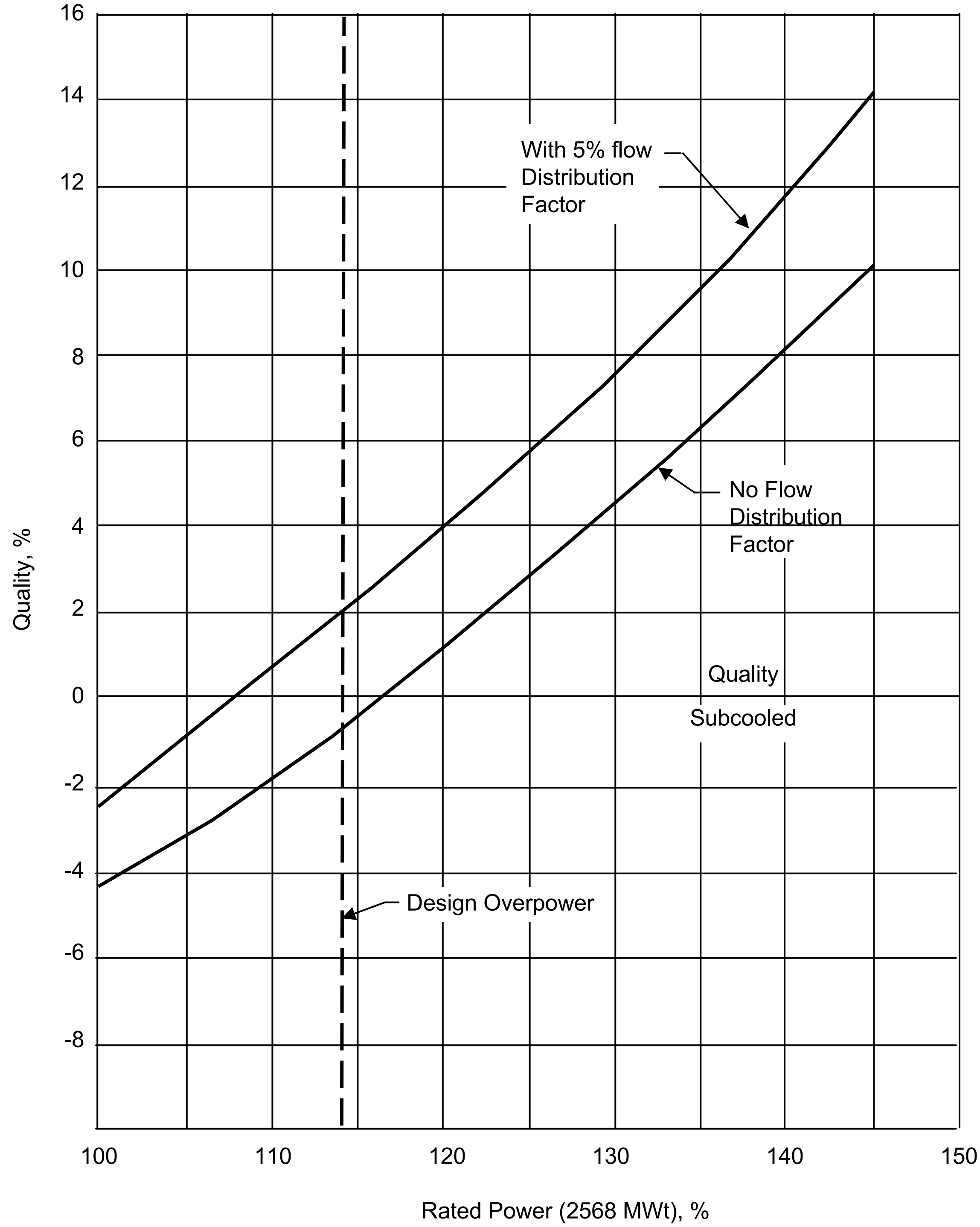
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-34

AMENDMENT 20

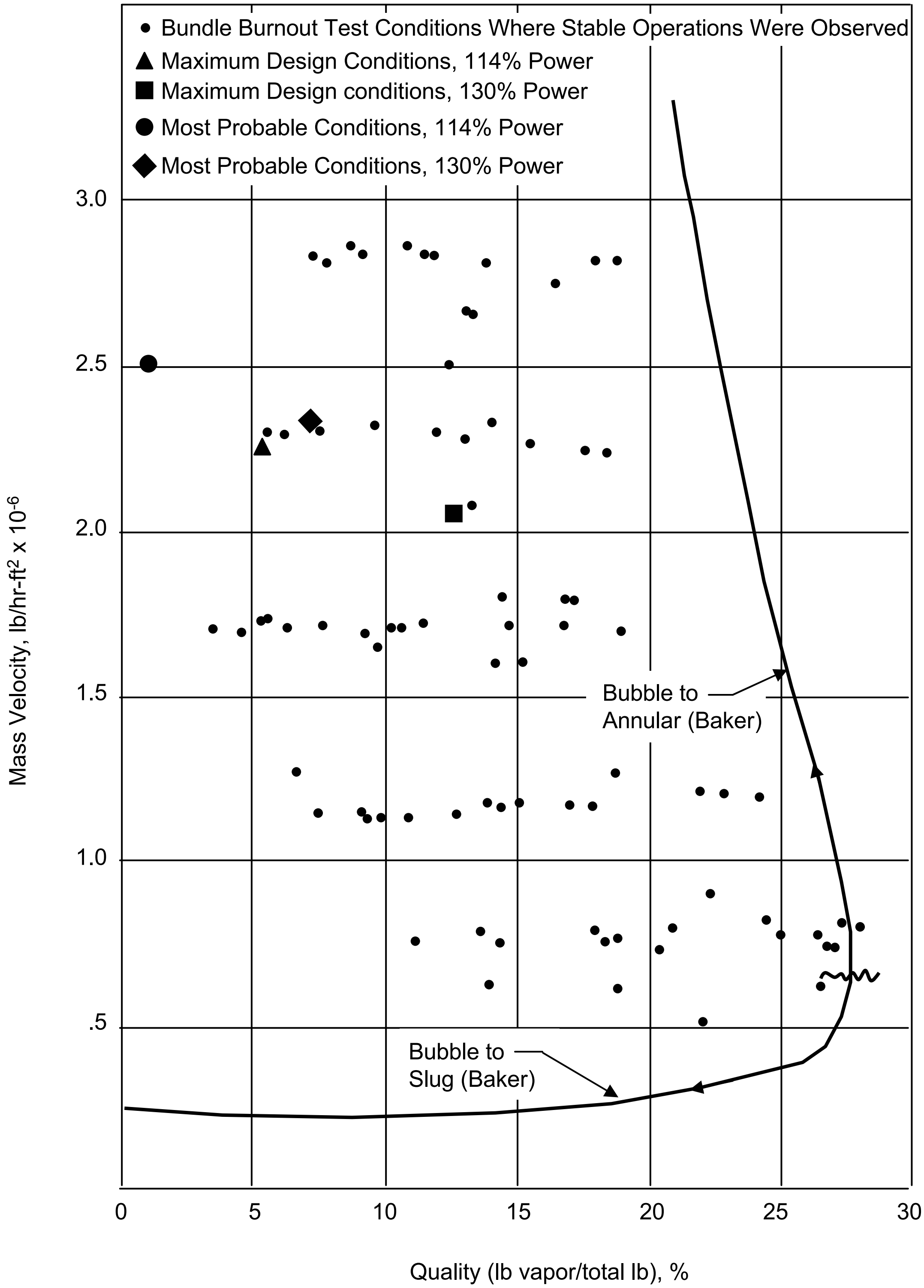
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
HOT CHANNEL FACTORS VERSUS PERCENT POPULATION PROTECTED	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-35

AMENDMENT 20

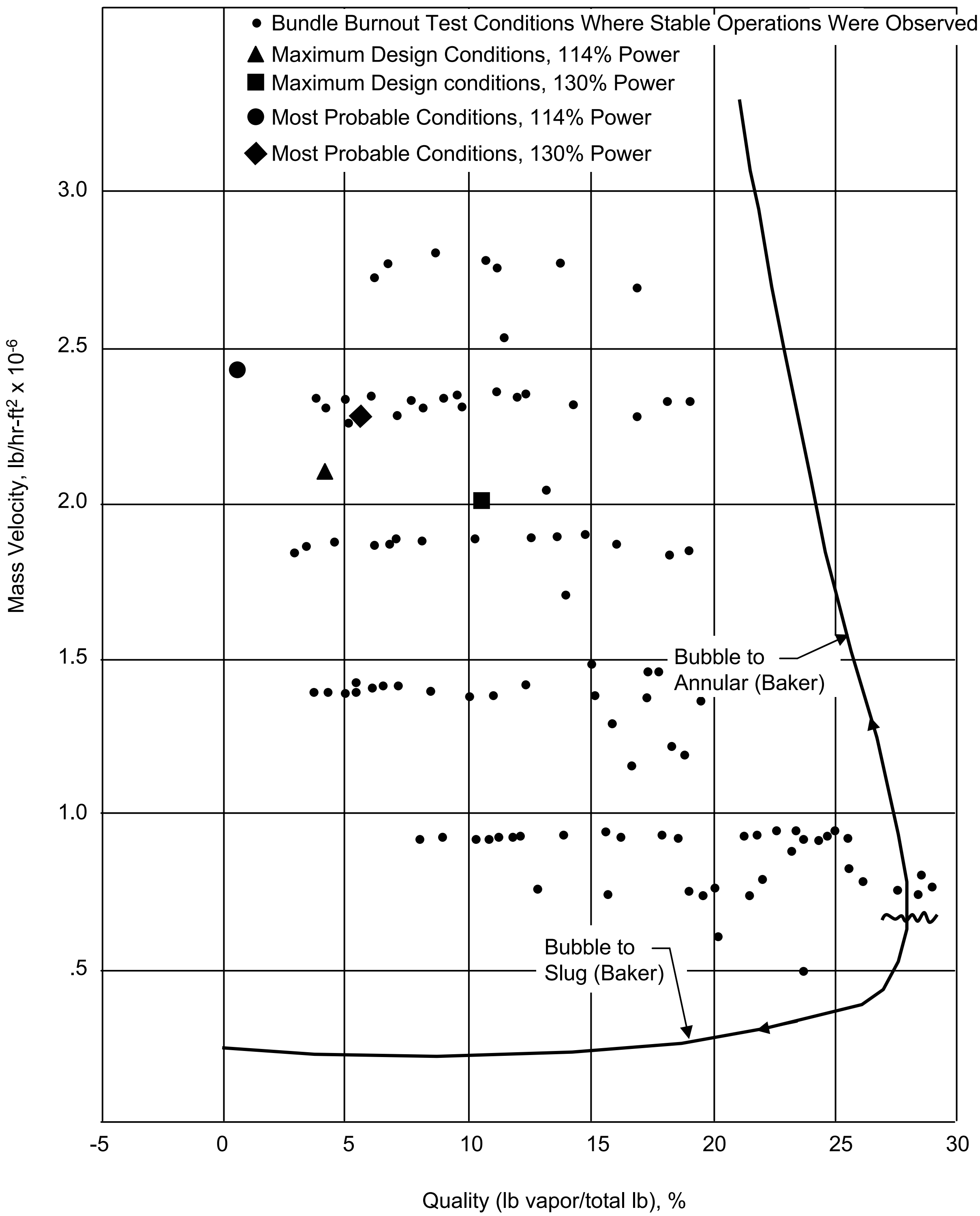
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
HOT CHANNEL AND NOMINAL CHANNEL EXIT QUALITIES VERSUS REACTOR POWER (WITHOUT ENGINEERING HOT CHANNEL FACTORS)	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-36

AMENDMENT 20

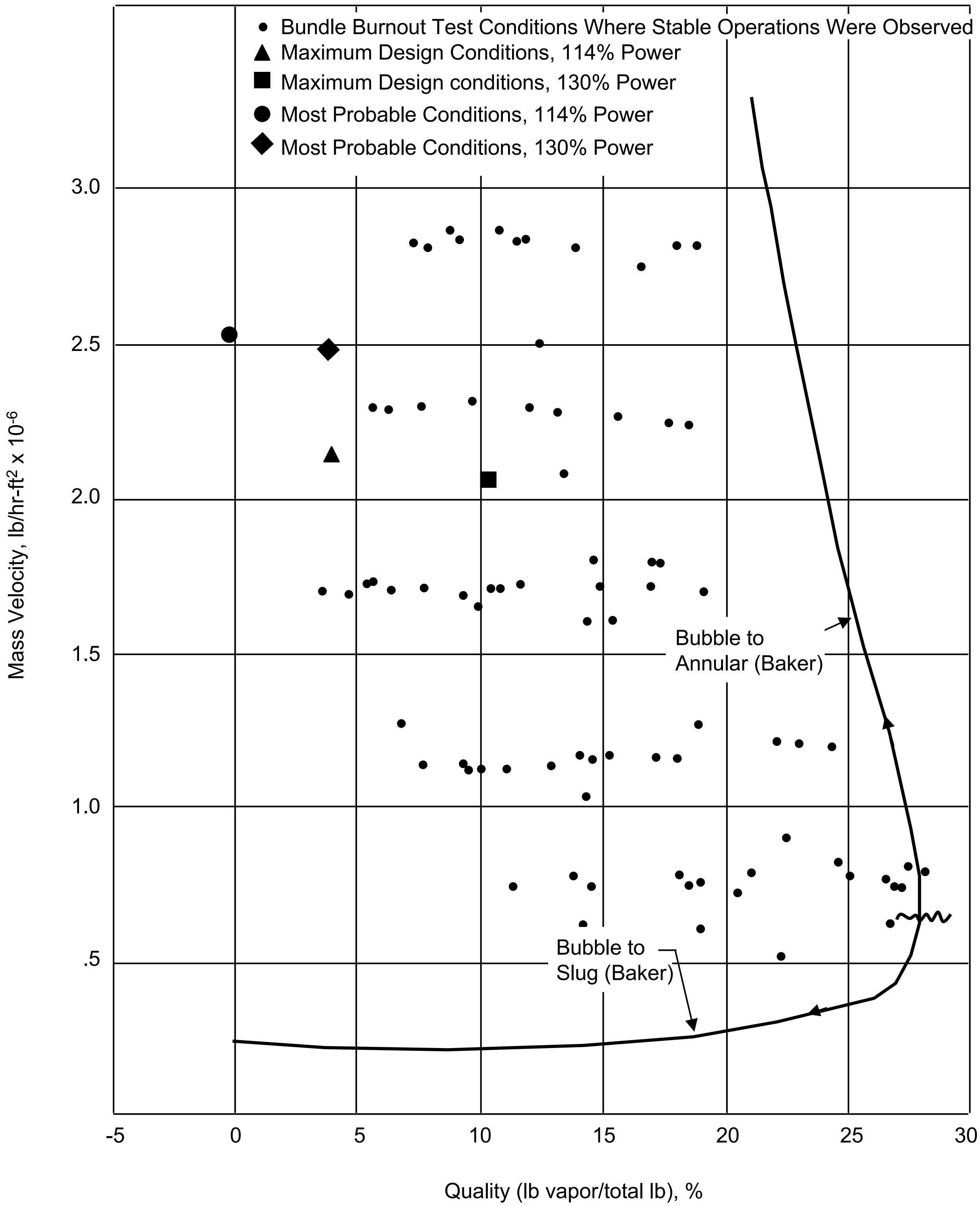
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
FLOW REGIME MAP FOR THE HOT UNIT CELL	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-37

AMENDMENT 20

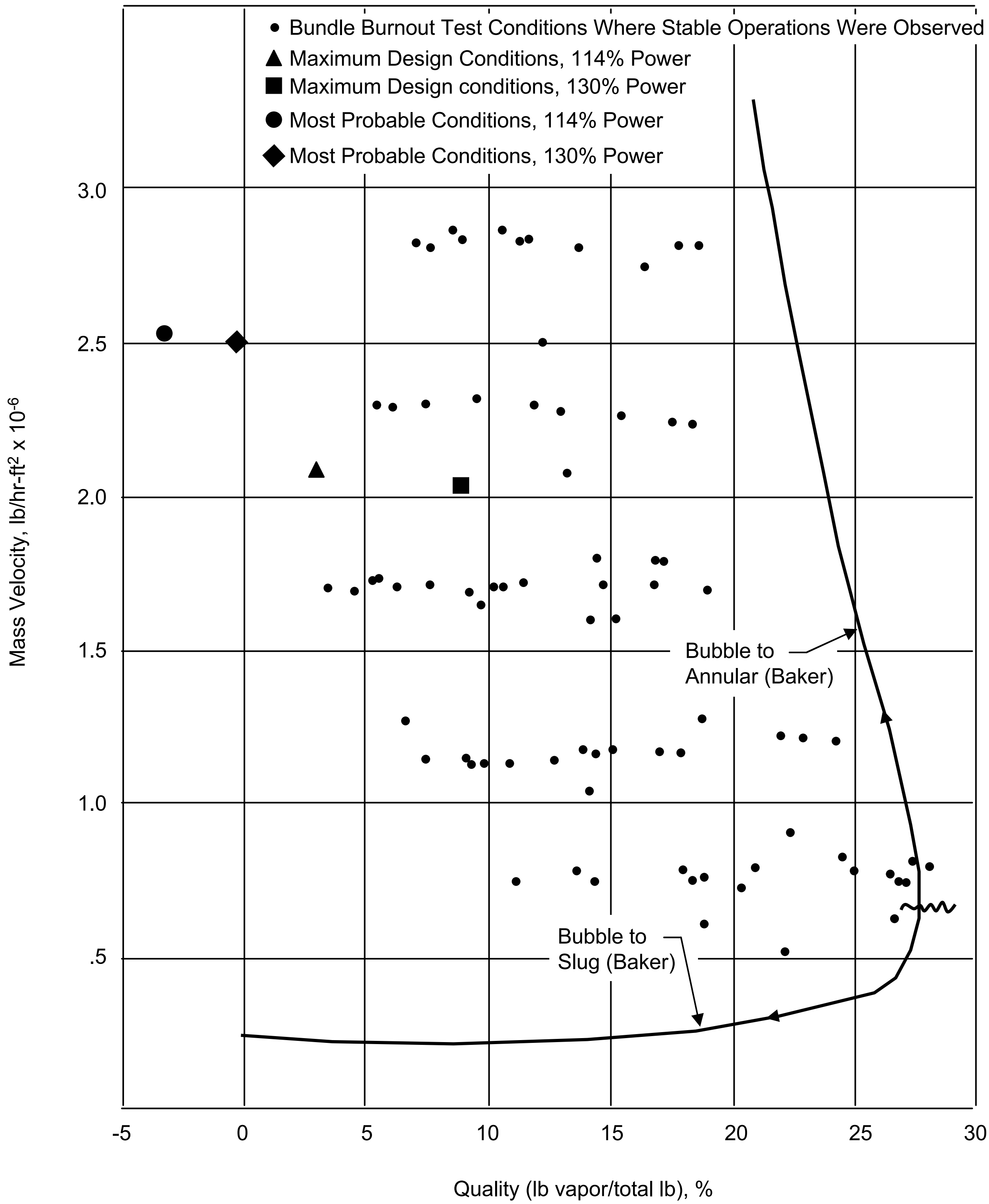
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FLOW REGIME MAP FOR THE HOT CONTROL ROD CELL	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-38

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FLOW REGIME MAP FOR THE HOT WALL CELL	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-39

AMENDMENT 20

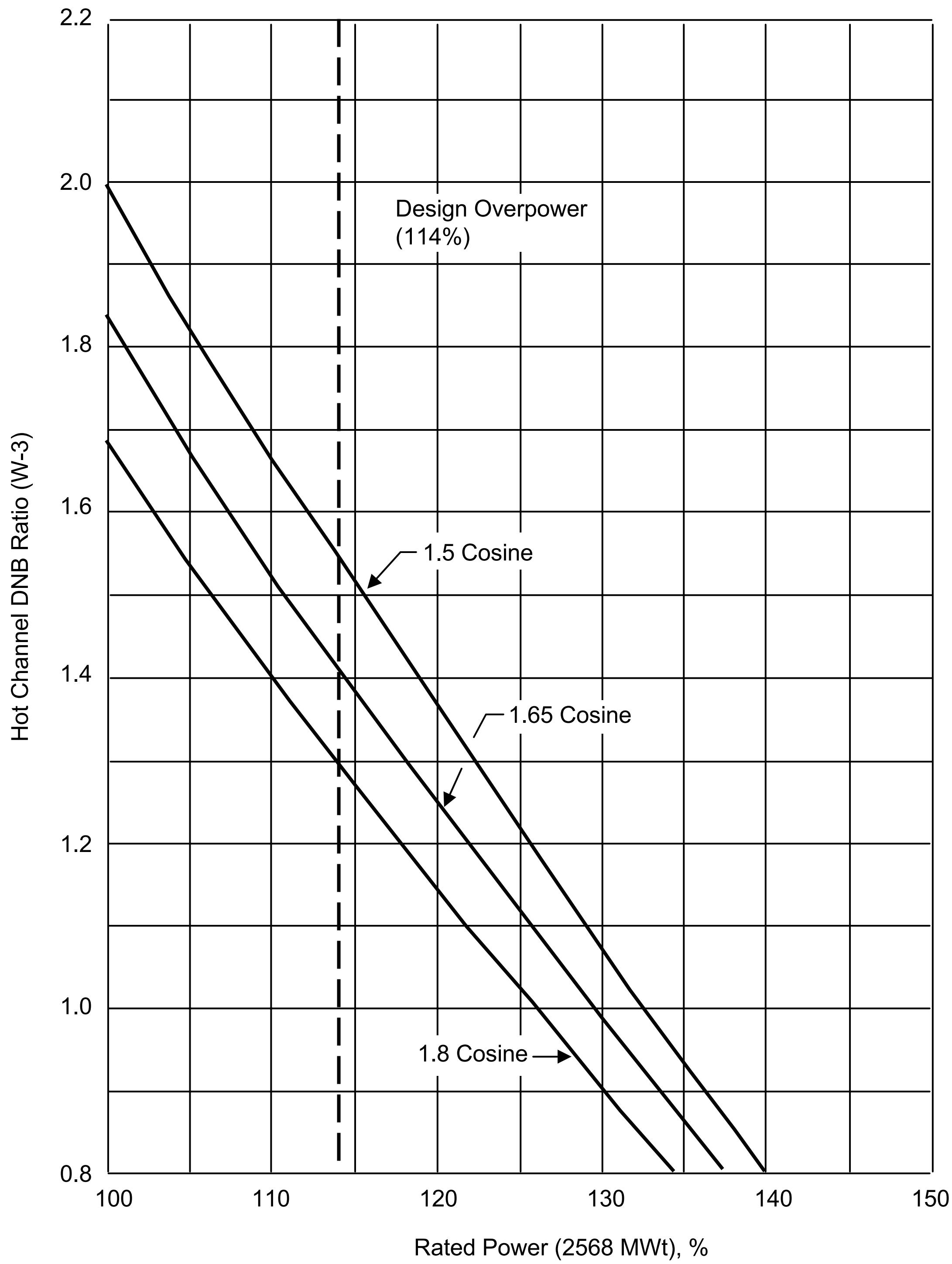
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

FLOW REGIME MAP FOR THE HOT CORNER CELL

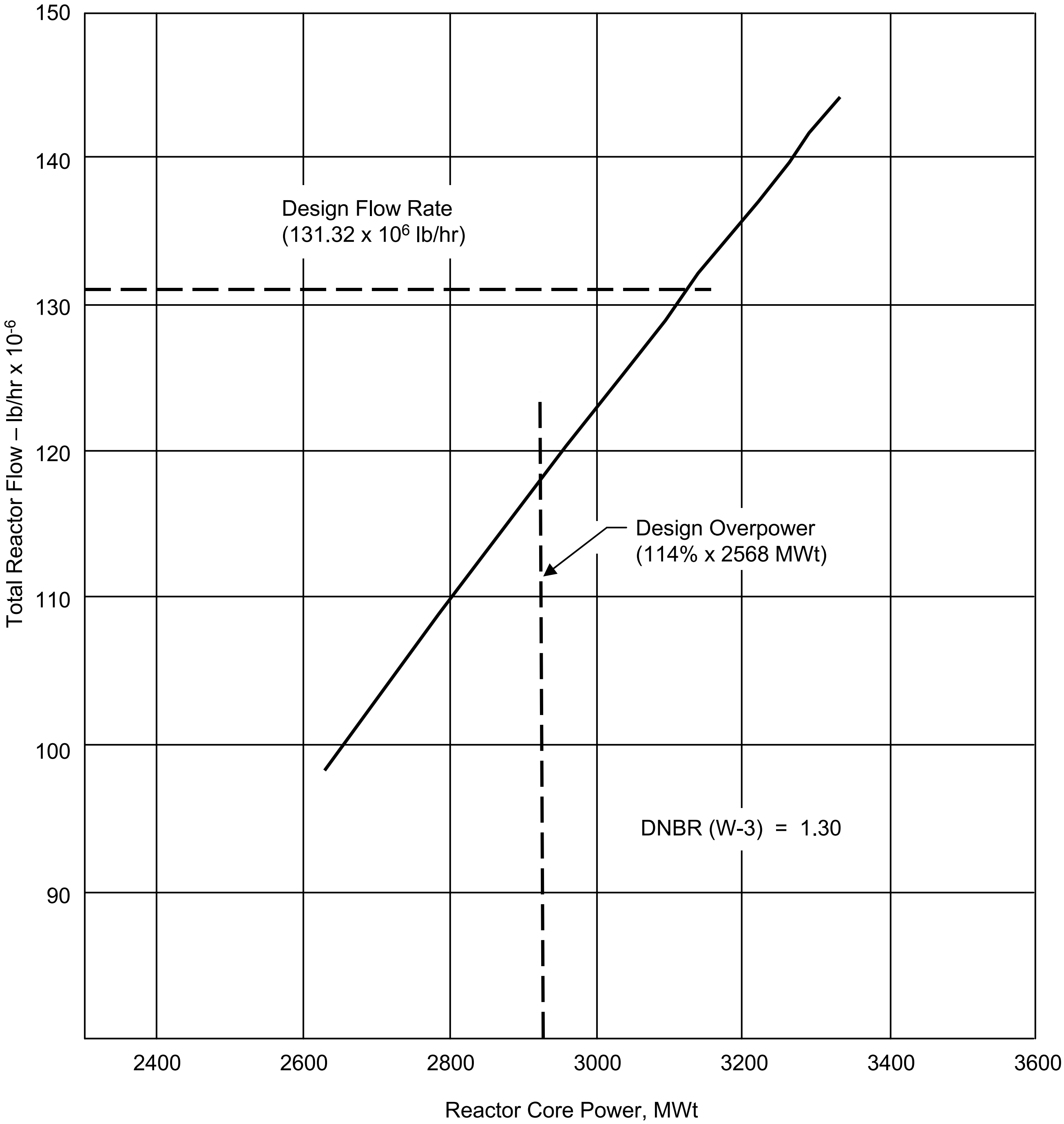
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-40

AMENDMENT 20

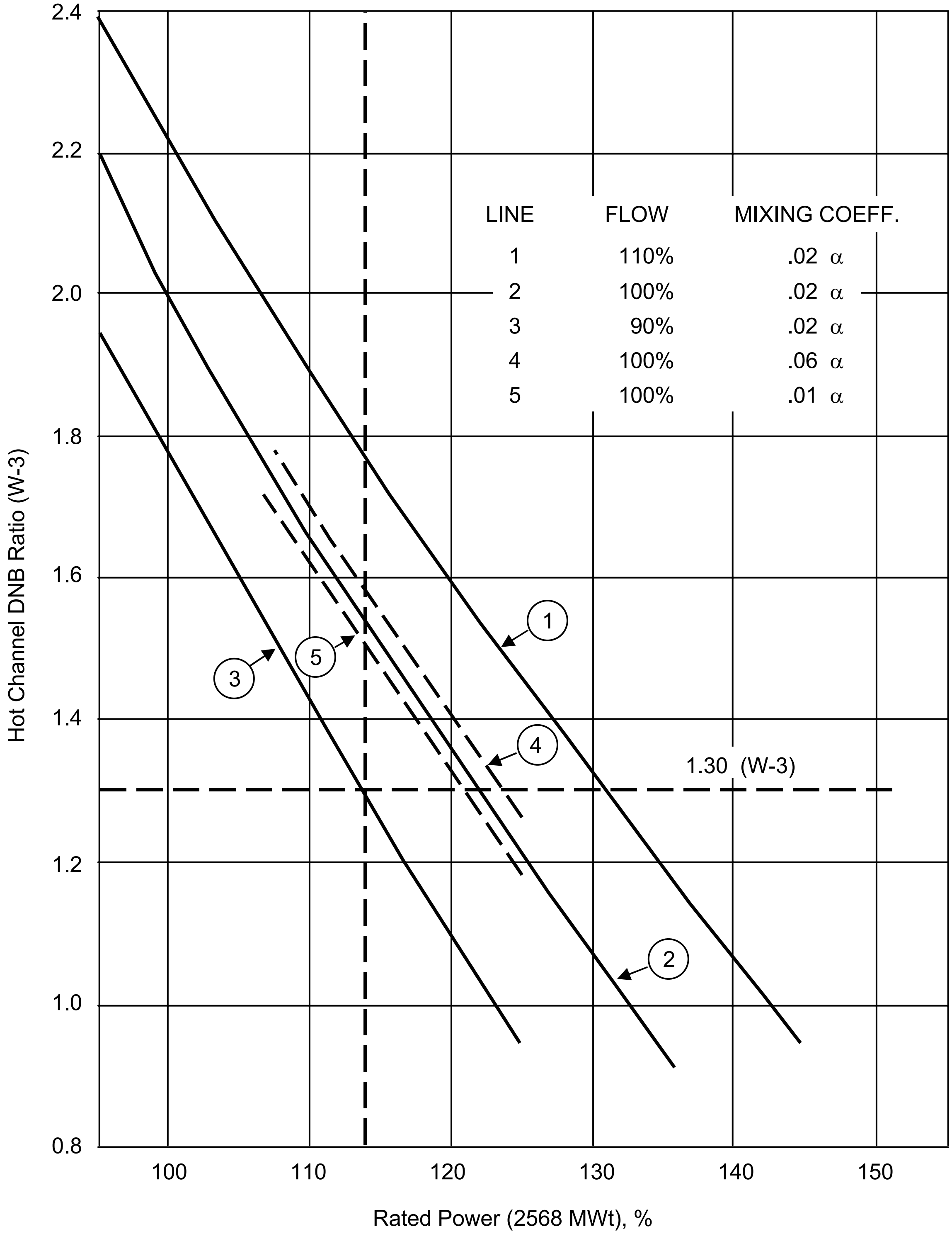
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
HOT CHANNEL DNB RATIO (W-3) VERSUS POWER FOR VARIOUS AXIAL FLUX SHAPES	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-41

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR COOLANT SYSTEM FLOW VERSUS POWER	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-42

AMENDMENT 20

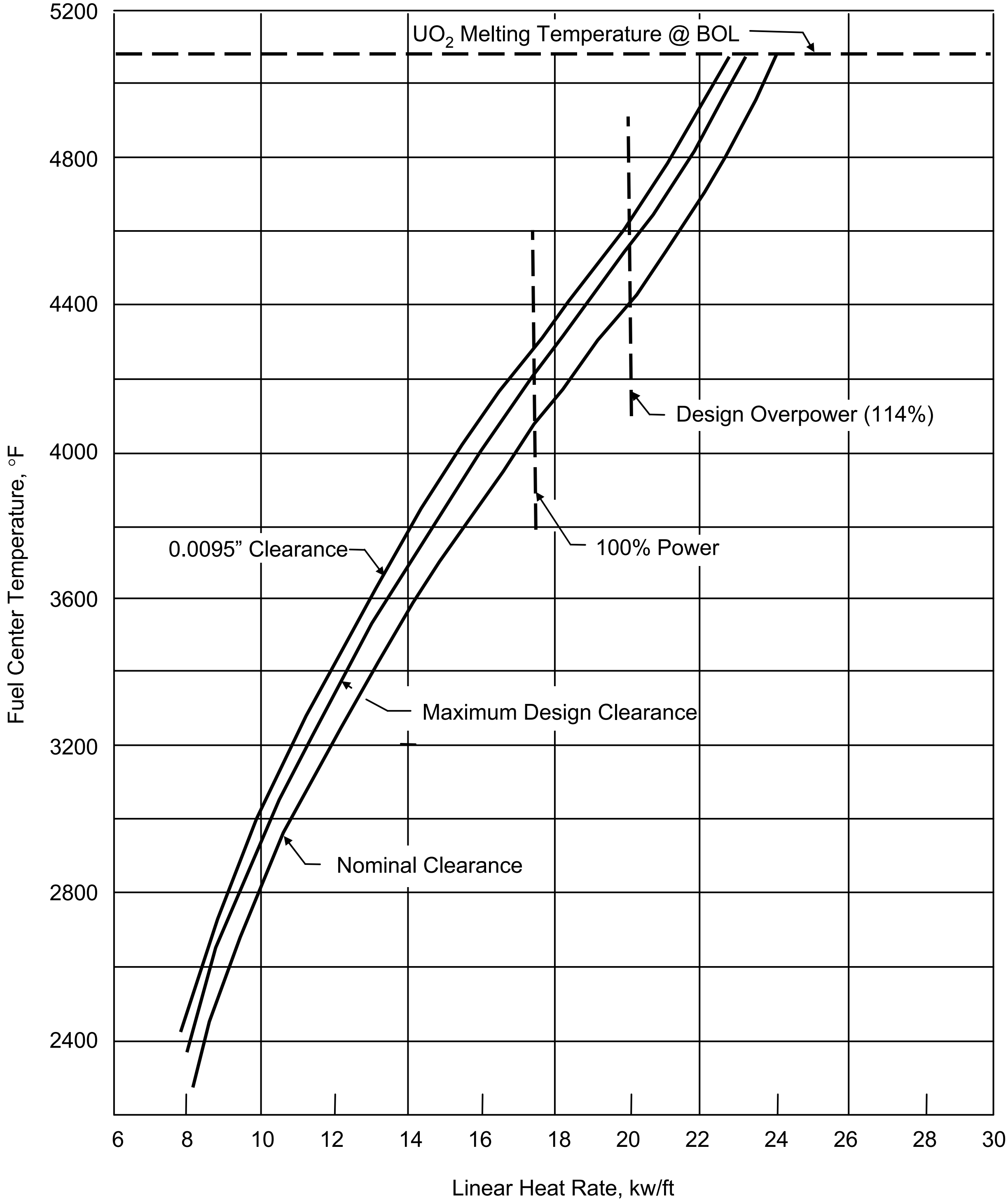
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

HOT CHANNEL DNB RATIO (W-3) VERSUS POWER WITH REACTOR SYSTEM FLOW AND ENERGY MIXING AS PARAMETERS

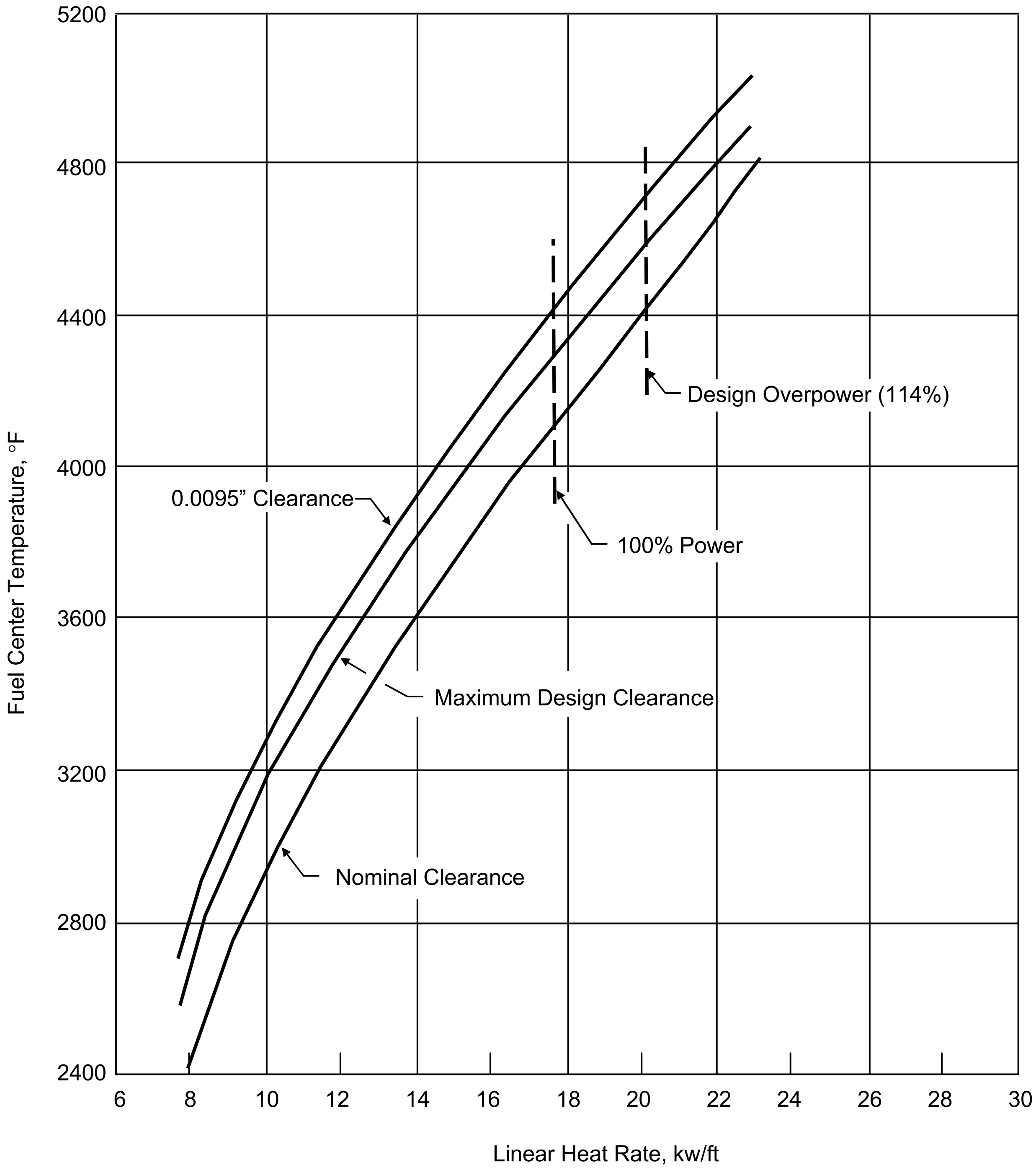
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-43

AMENDMENT 20

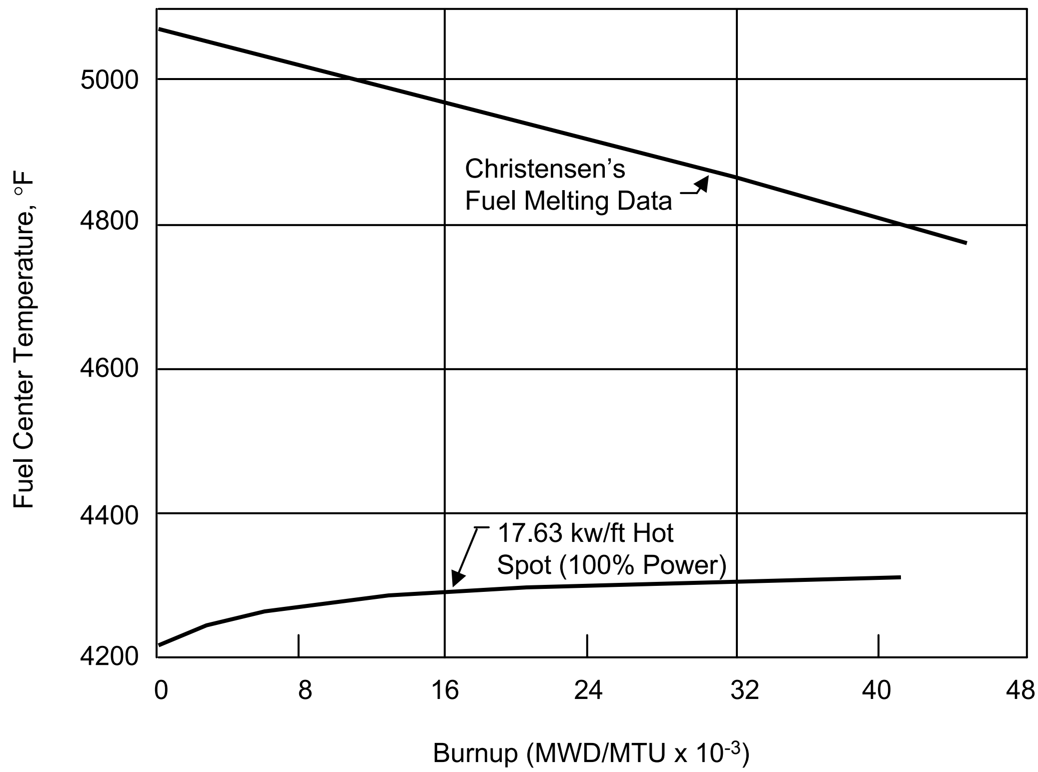
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FUEL CENTER TEMPERATURE FOR BEGINNING OF LIFE CONDITIONS	BASED ON DRAWING NO	SHEET / REV.

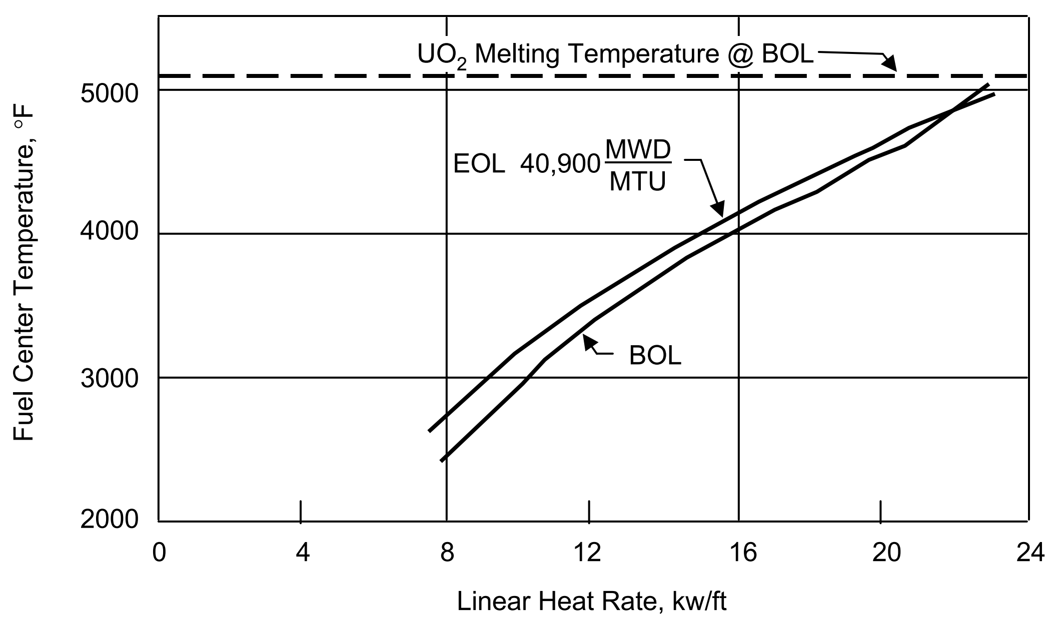


SAR FIGURE NO. 3-44

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FUEL CENTER TEMPERATURE FOR END OF LIFE CONDITIONS	BASED ON DRAWING NO	SHEET / REV.

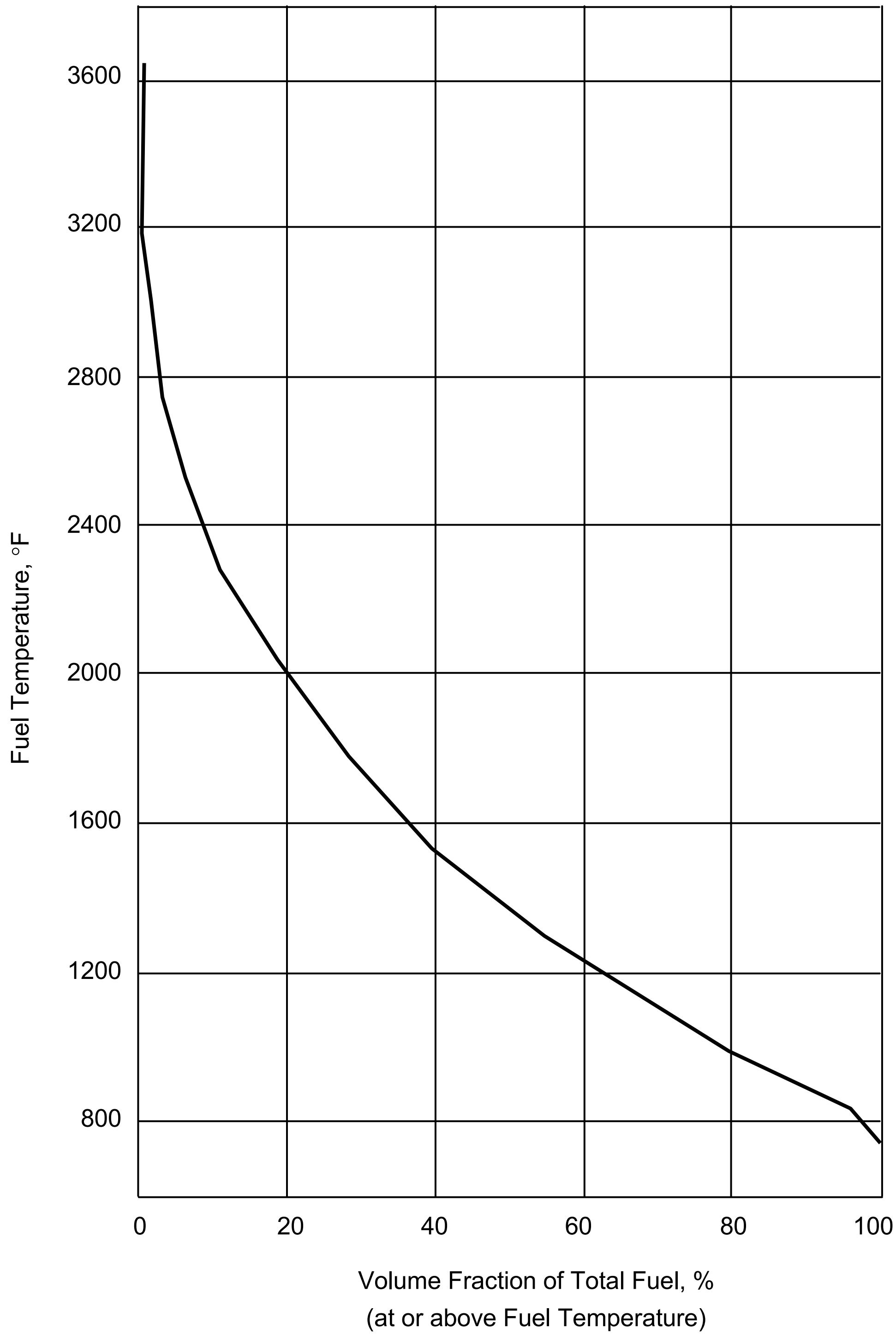




SAR FIGURE NO. 3-45

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
BURNUP EFFECT ON FUEL CENTER TEMPERATURE	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-46

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FUEL TEMPERATURE VERSUS TOTAL FUEL VOLUME FRACTION FOR EQUILIBRIUM CYCLE AT END OF LIFE	BASED ON DRAWING NO	SHEET / REV.

1 1.545	3 1.087	2 1.082	3 0.854	3 0.926	1 1.326	2 1.282	1 0.969
	2 1.173	3 0.958	2 1.007	3 0.916	2 1.202	3 1.019	1 0.887
		3 0.920	2 1.068	2 1.106	3 0.959	1 1.088	1 0.684
			2 1.054	3 0.972	2 1.078	1 0.917	
				3 1.014	1 1.110	1 0.690	
					1 0.793		

Number Cycles Burned

Assembly $P/\bar{P}$

SAR FIGURE NO. 3-47

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

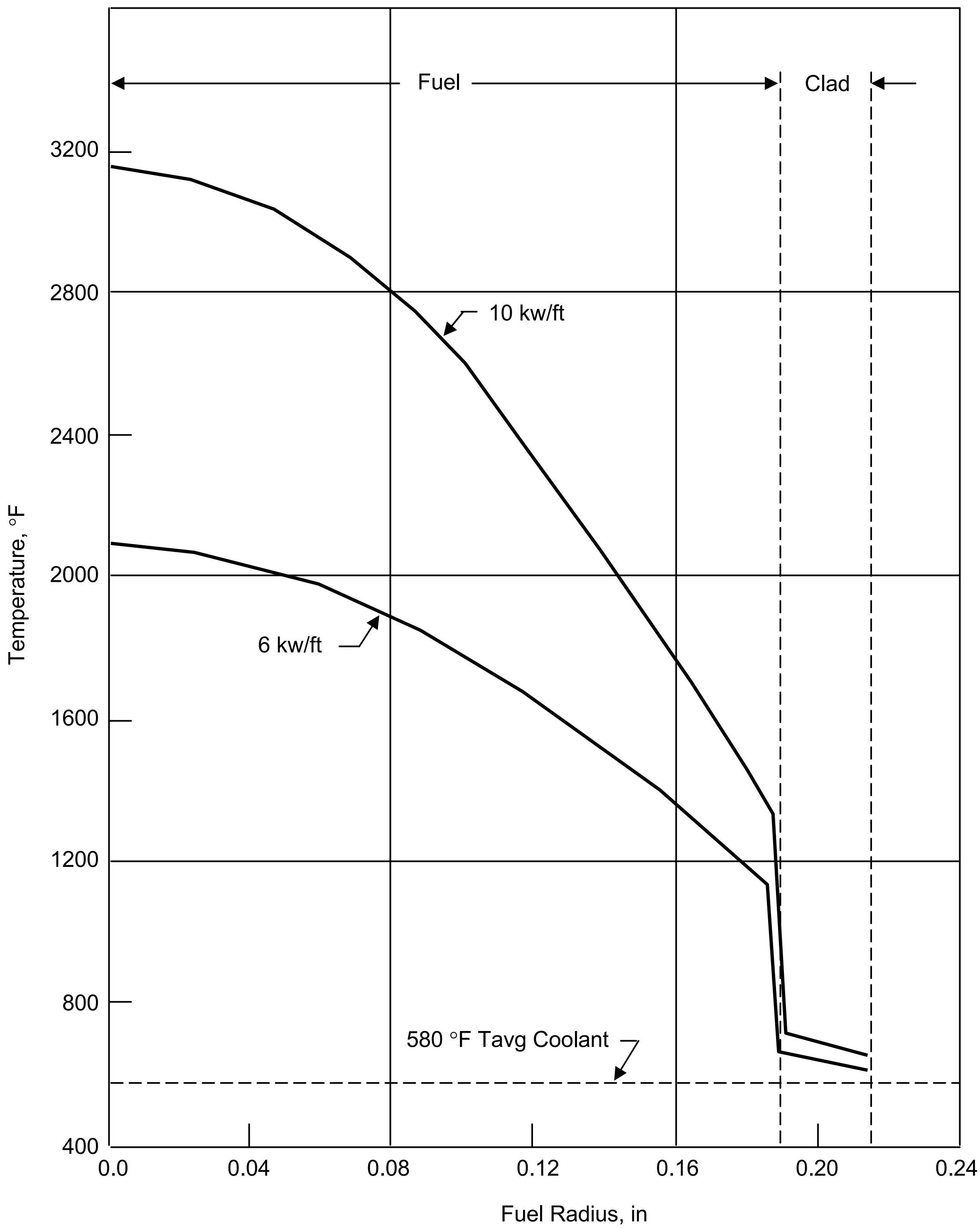
SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

TYPICAL REACTOR FUEL ASSEMBLY POWER DISTRIBUTION AT END-OF-LIFE EQUILIBRIUM CYCLE CONDITIONS FOR 1/8 CORE

BASED ON DRAWING NO

SHEET 1

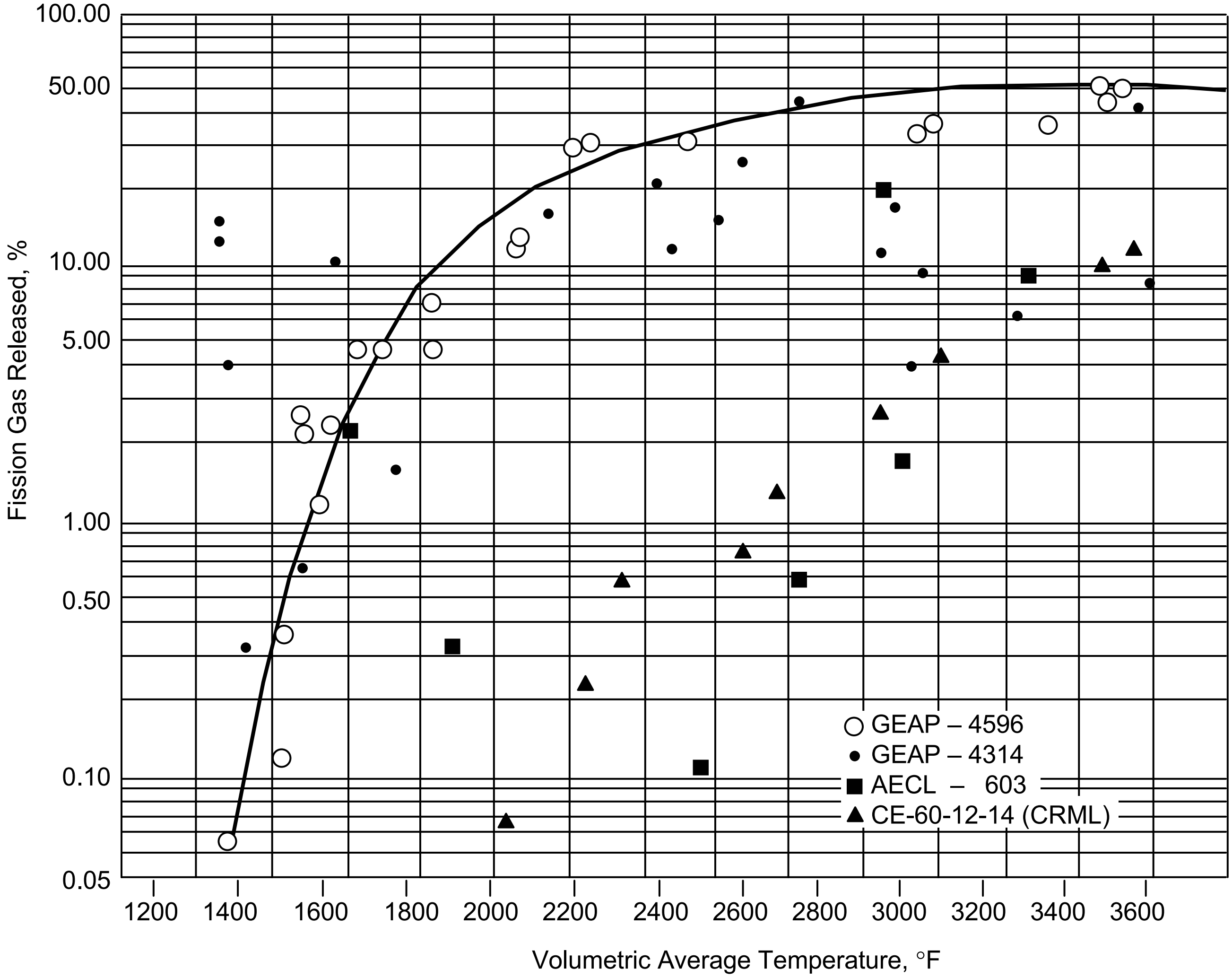
REV.



SAR FIGURE NO. 3-48

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FUEL ROD TEMPERATURE PROFILES AT 6 AND 10 KW/FT	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-49

AMENDMENT 20

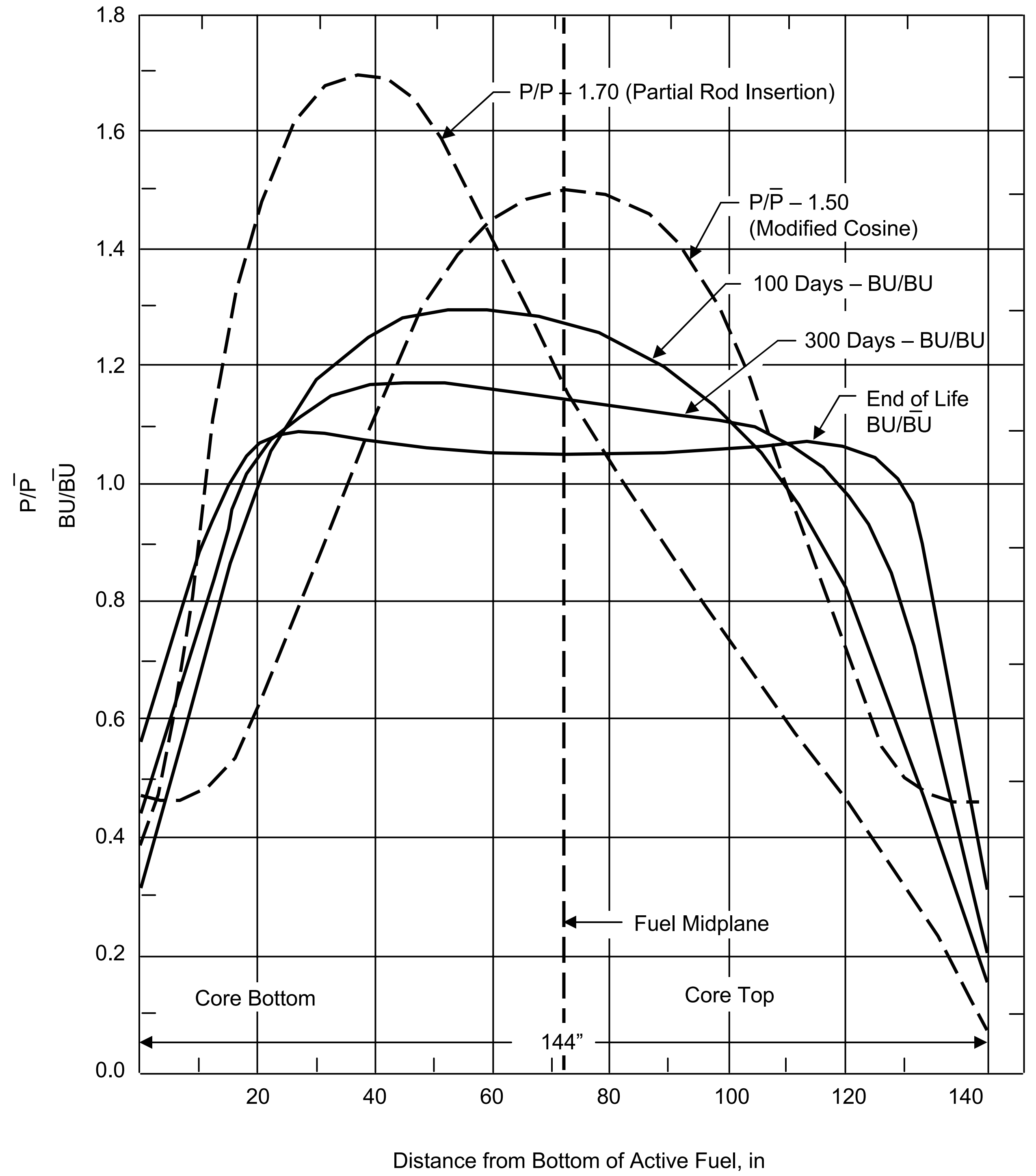
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

PERCENT FISSION GAS RELEASED AS A FUNCTION OF THE AVG TEMP OF THE UO_2 FUEL

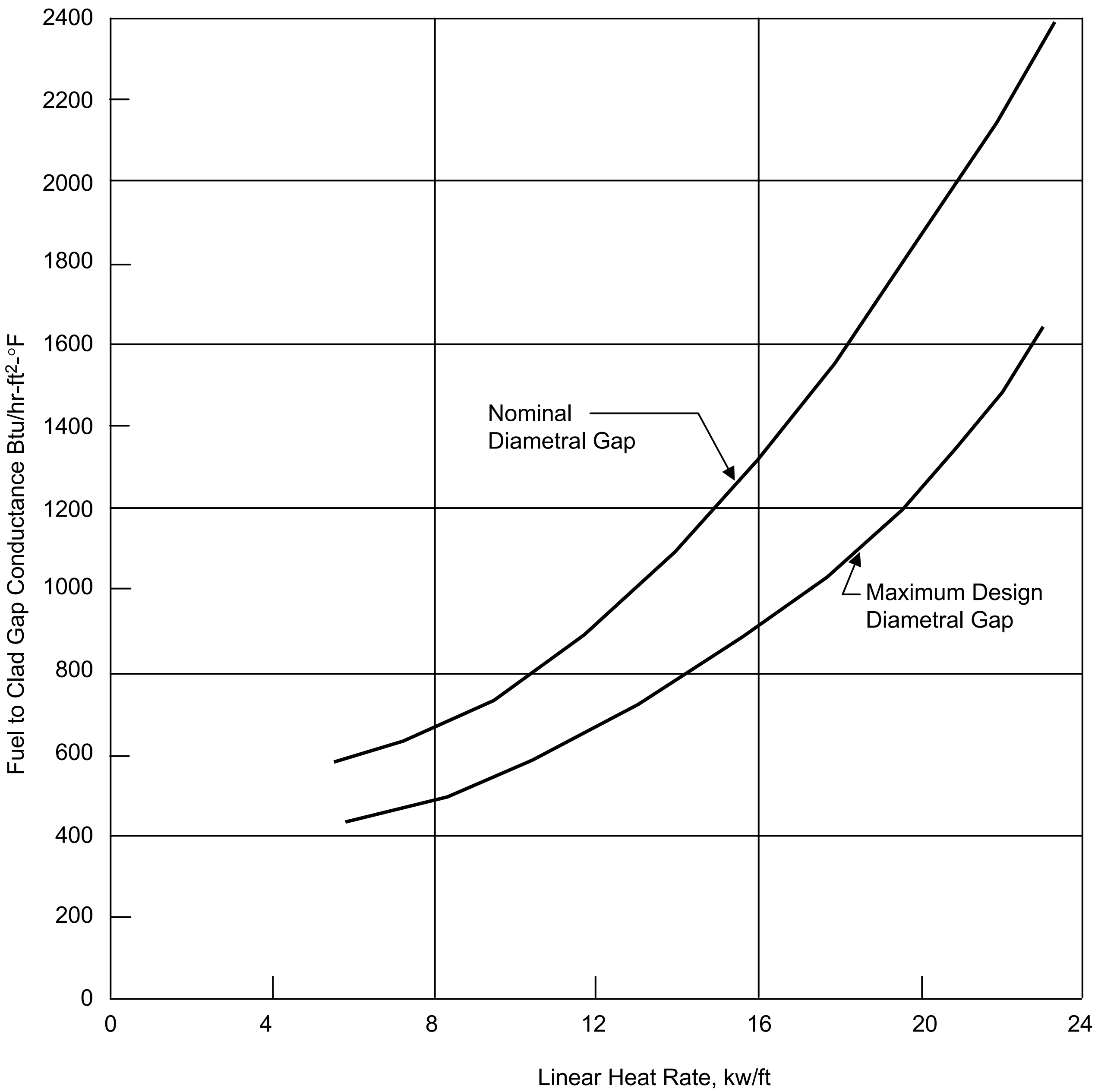
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-50

AMENDMENT 20

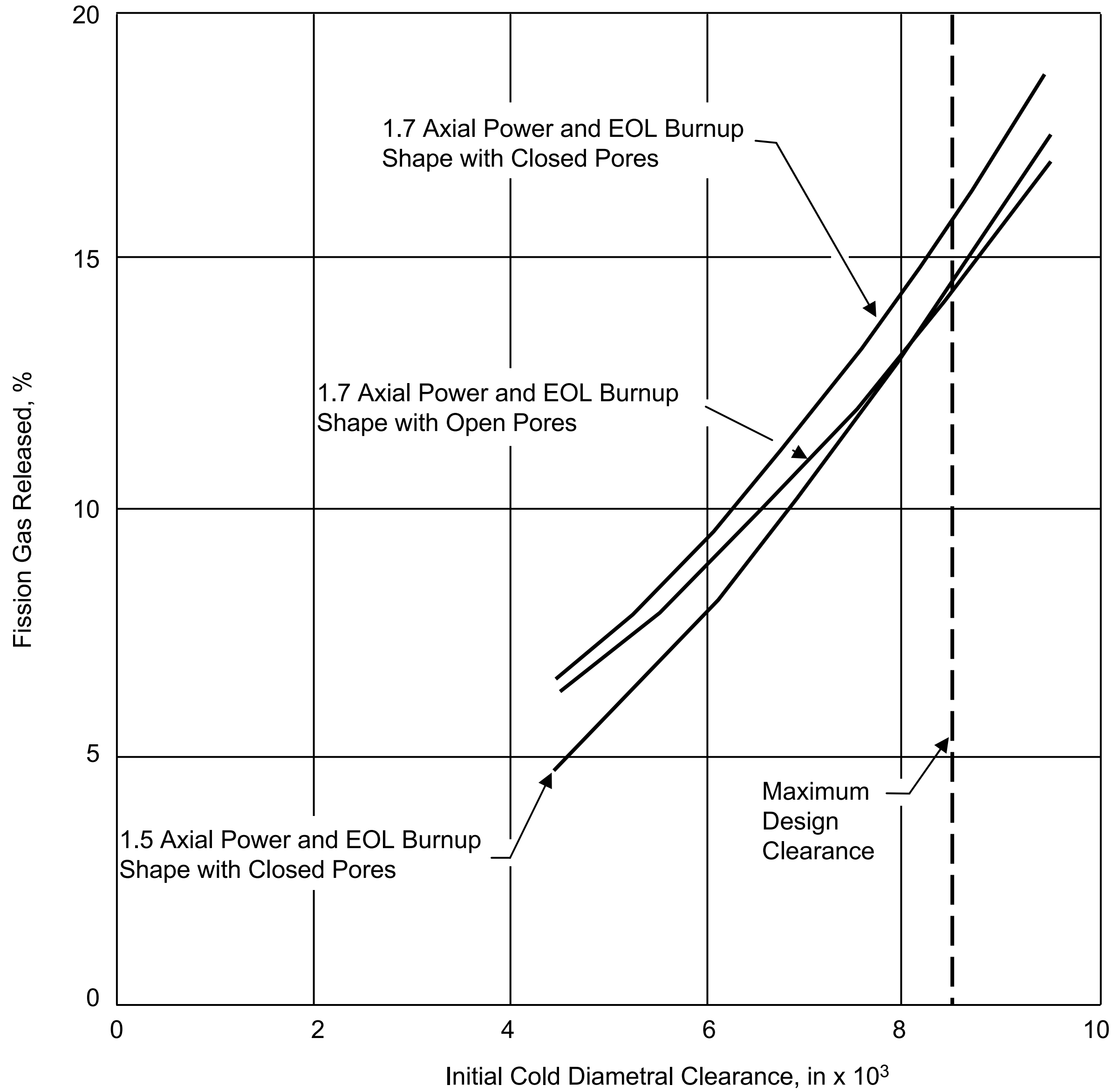
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
AXIAL LOCAL TO AVERAGE BURNUP AND INSTANTANEOUS POWER COMPARISONS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-51

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FUEL TO CLAD GAP CONDUCTANCE FOR END OF LIFE CONDITIONS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-52

AMENDMENT 20

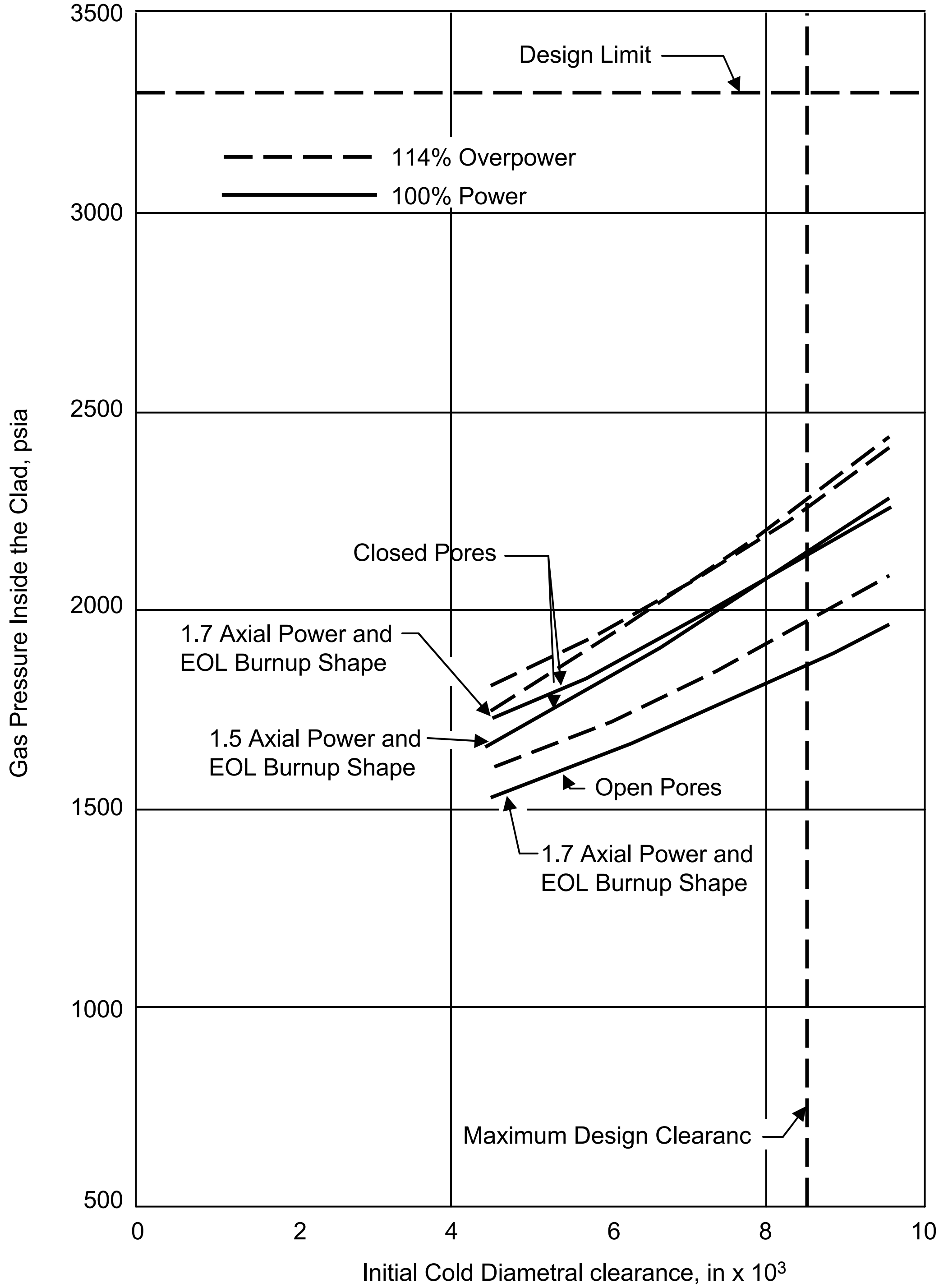
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

FISSION GAS RELEASE FOR 1.5 AND 1.7 MAX/AVG AXIAL POWER SHAPES

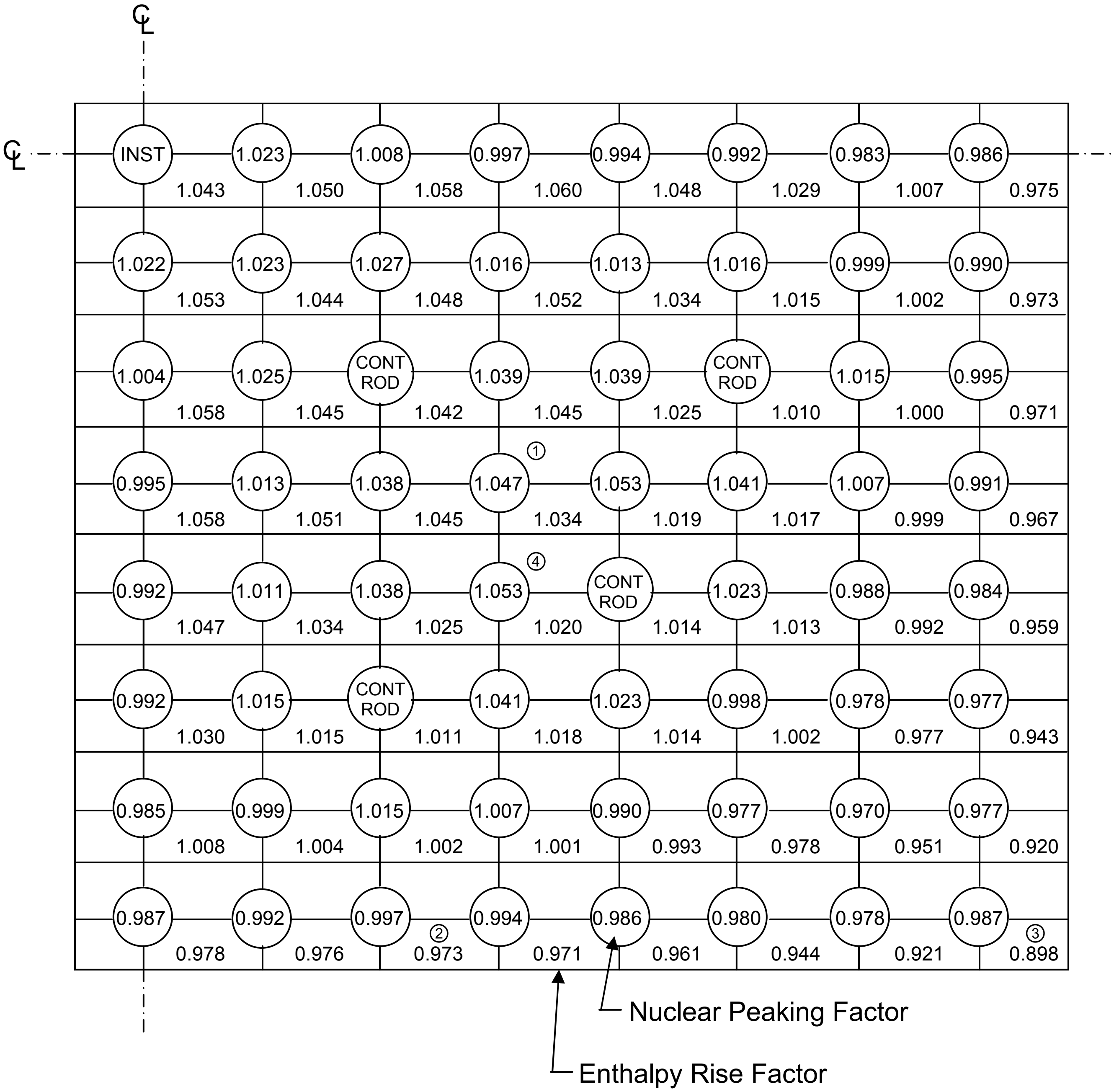
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-53

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MAXIMUM GAS PRESSURE INSIDE THE FUEL CLAD FOR VARIOUS AXIAL BURNUP AND POWER SHAPES	BASED ON DRAWING NO	SHEET / REV.



① HOT UNIT CELL

② HOT WALL CELL

③ HOT CORNER CELL

④ HOT CONTROL ROD CELL

SAR FIGURE NO. 3-54

AMENDMENT 20

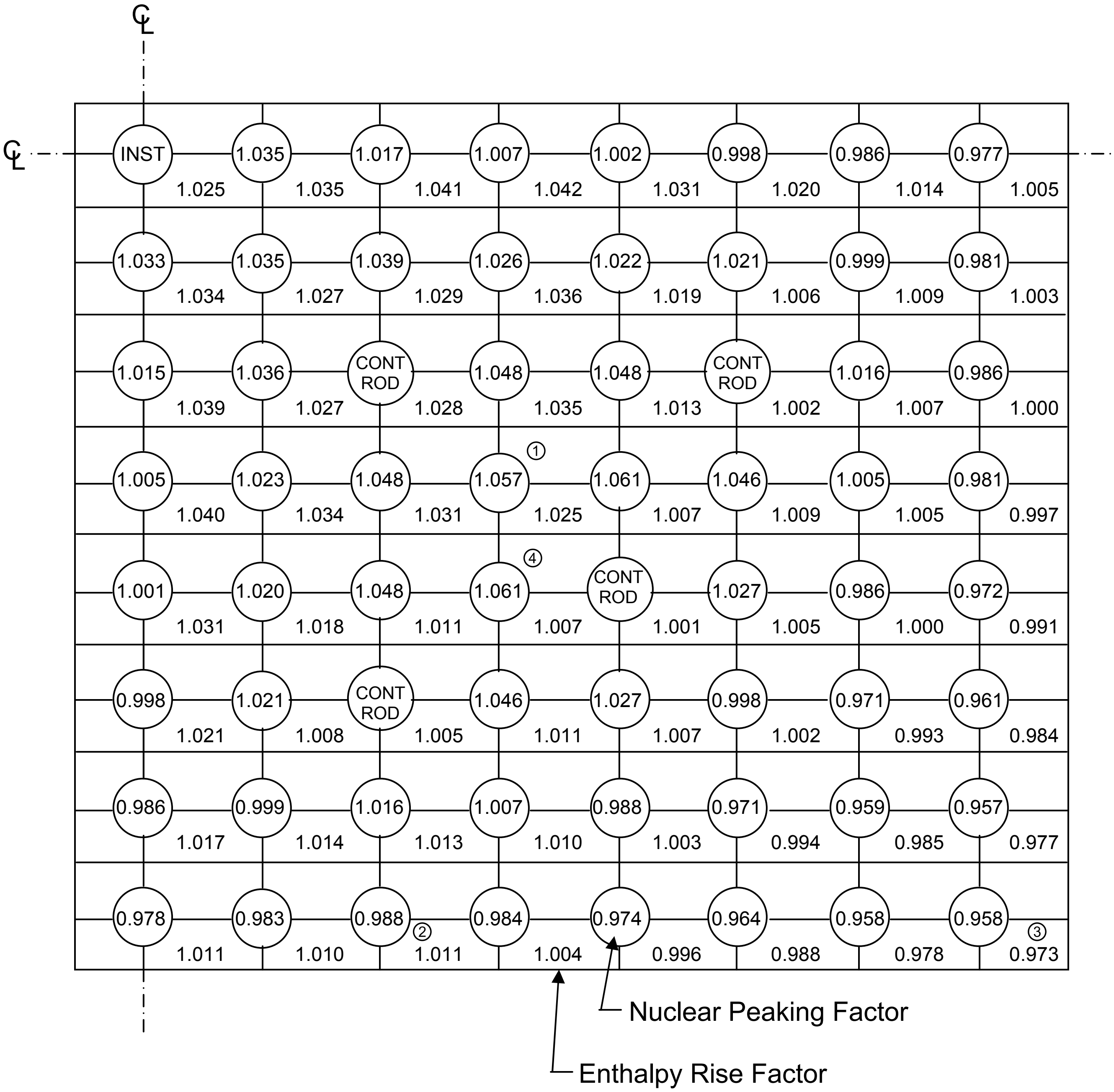
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

NOMINAL FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS

BASED ON DRAWING NO	SHEET	REV.
	1	



① HOT UNIT CELL

② HOT WALL CELL

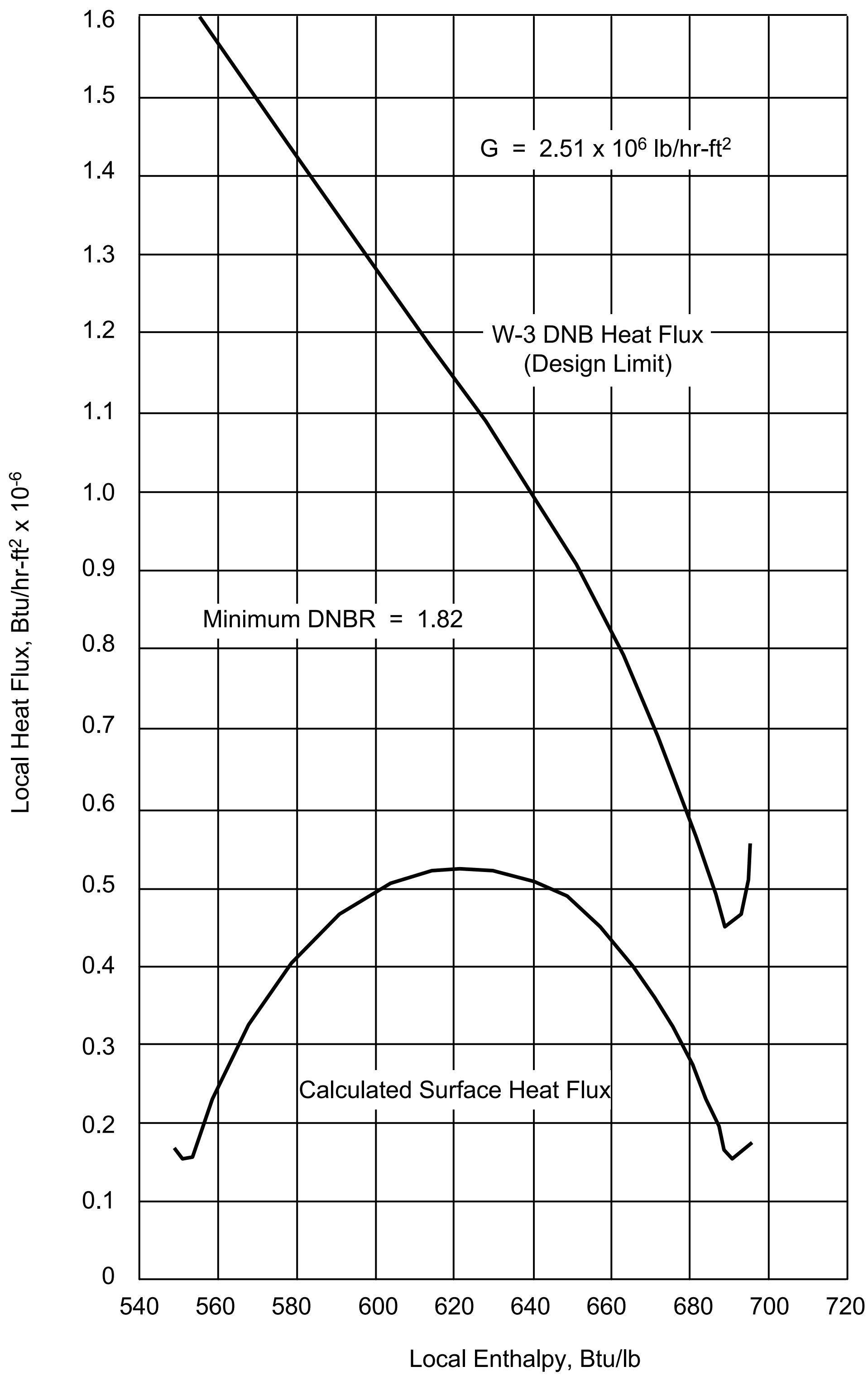
③ HOT CORNER CELL

④ HOT CONTROL ROD CELL

SAR FIGURE NO. 3-55

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MAXIMUM FUEL ROD POWER PEAKS AND CELL EXIT ENTHALPY RISE RATIOS	BASED ON DRAWING NO	SHEET 1 / REV.



SAR FIGURE NO. 3-56

AMENDMENT 20

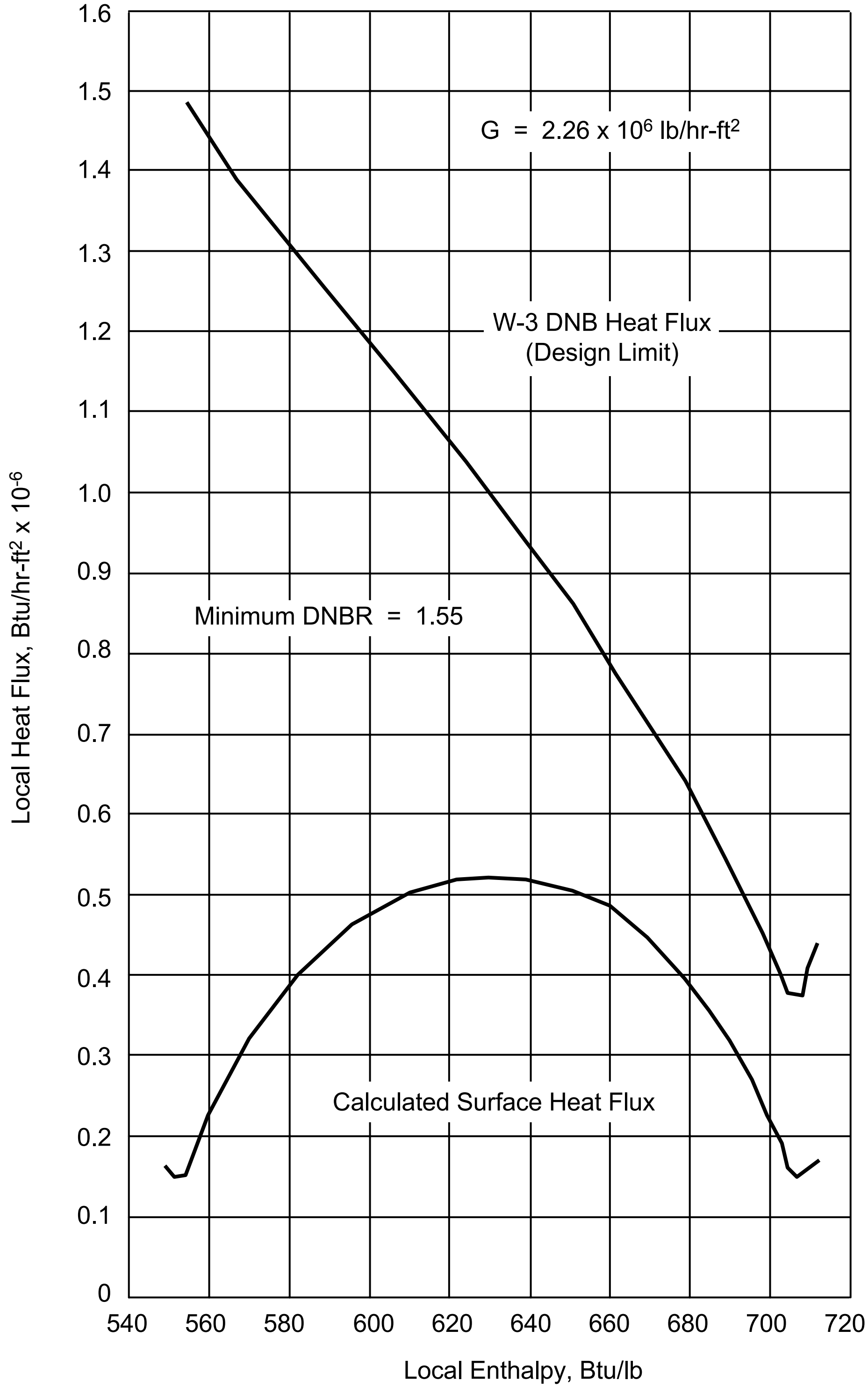
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

CALCULATED AND DESIGN LIMIT LOCAL HEAT FLUX VERSUS ENTHALPY IN THE HOT UNIT CELL AT THE MOST PROBABLE CONDITION

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-57

AMENDMENT 20

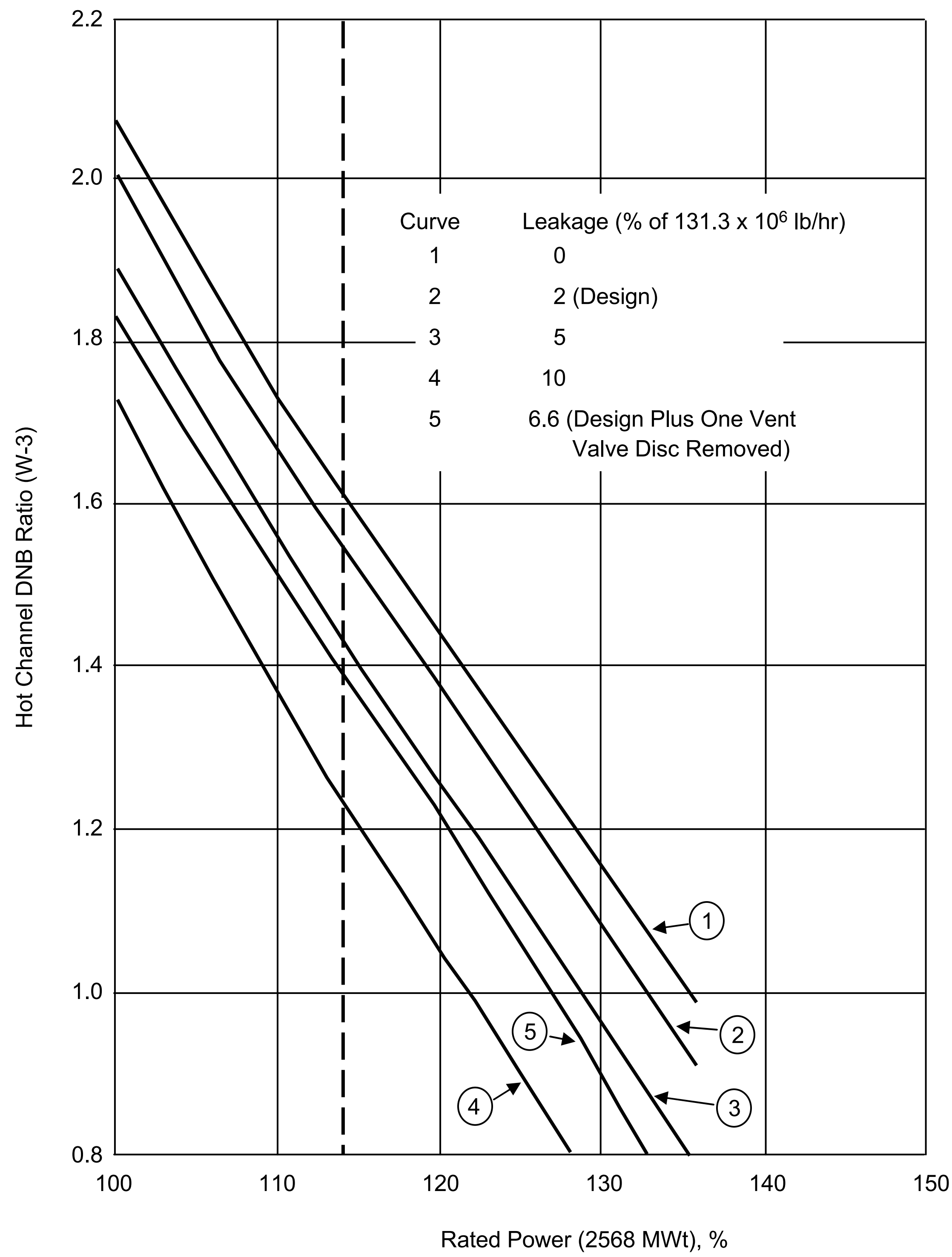
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

CALCULATED AND DESIGN LIMIT LOCAL HEAT FLUX VERSUS ENTHALPY IN THE HOT UNIT CELL AT THE MAXIMUM DESIGN CONDITION

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-58

AMENDMENT 20

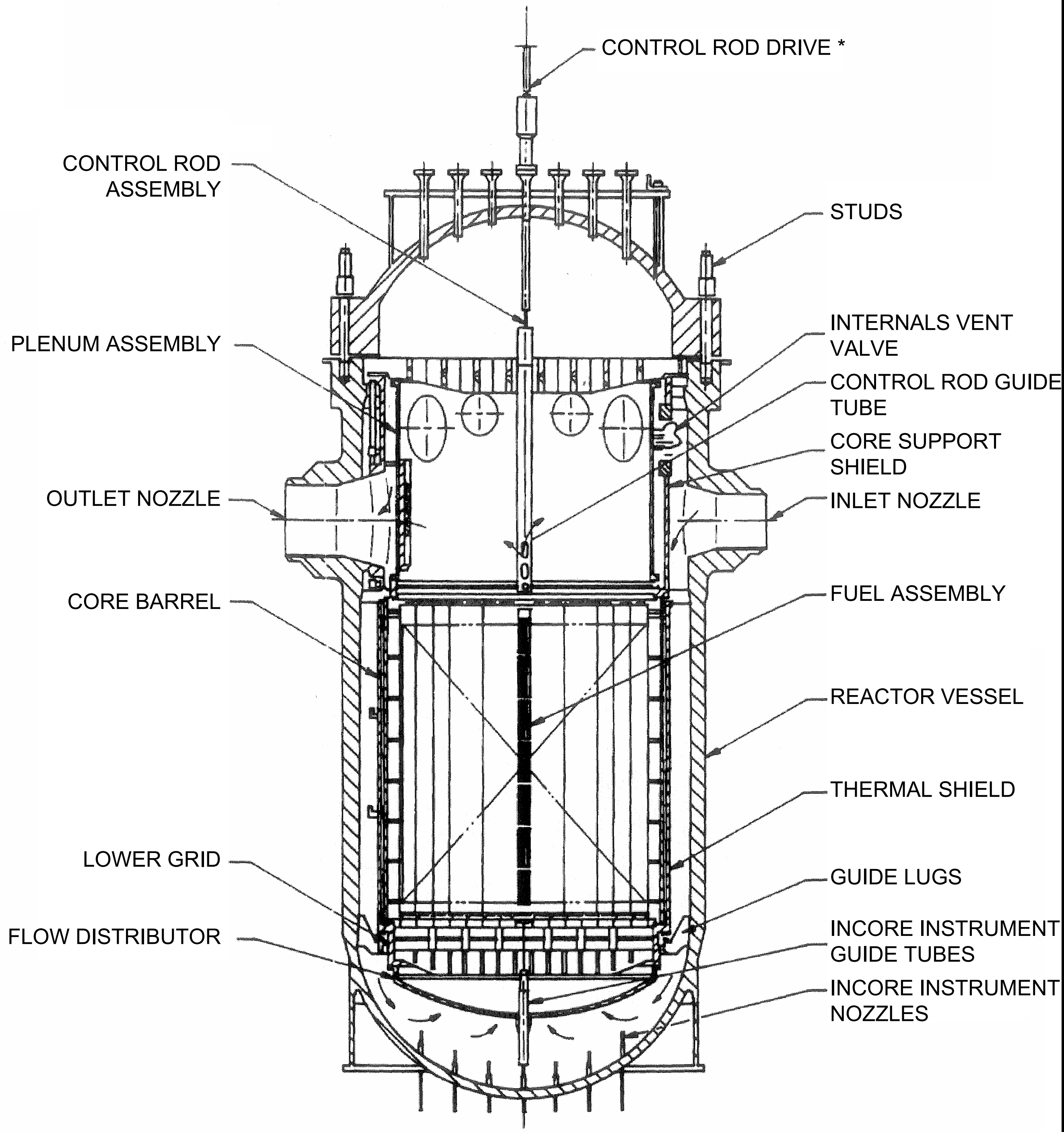
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

DNB RATIO (W-3) VERSUS POWER FOR VARIOUS INLET TO OUTLET CORE BYPASS LEAKAGE

BASED ON DRAWING NO | SHEET | REV.

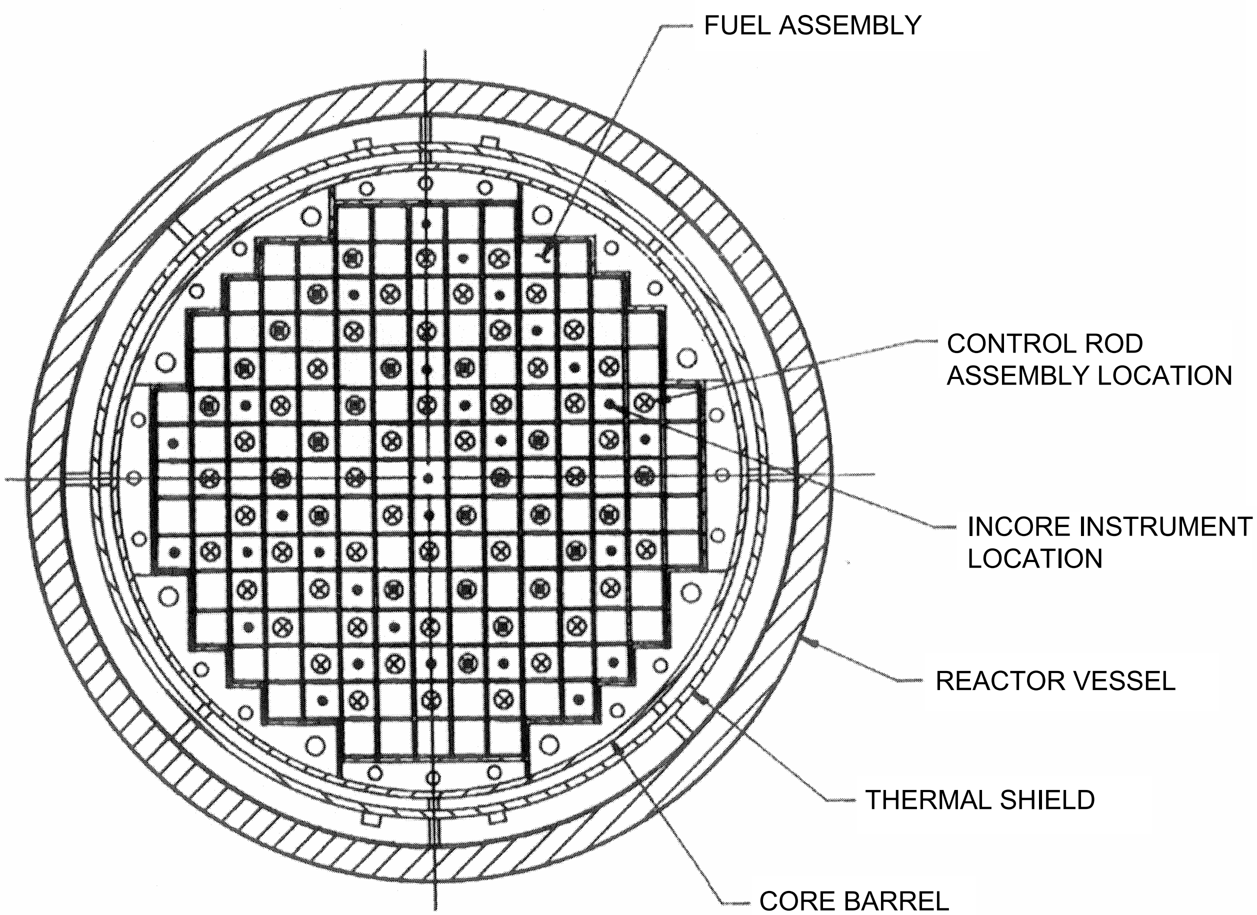


* CENTER CONTROL ROD DRIVE AND ASSEMBLY REMOVED DURING 1R7.

SAR FIGURE NO. 3-59

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR VESSEL AND INTERNALS – GENERAL ARRANGEMENT	BASED ON DRAWING NO	SHEET / REV.



REACTOR VESSEL AND INTERNALS – CROSS SECTION

SAR FIGURE NO. 3-60

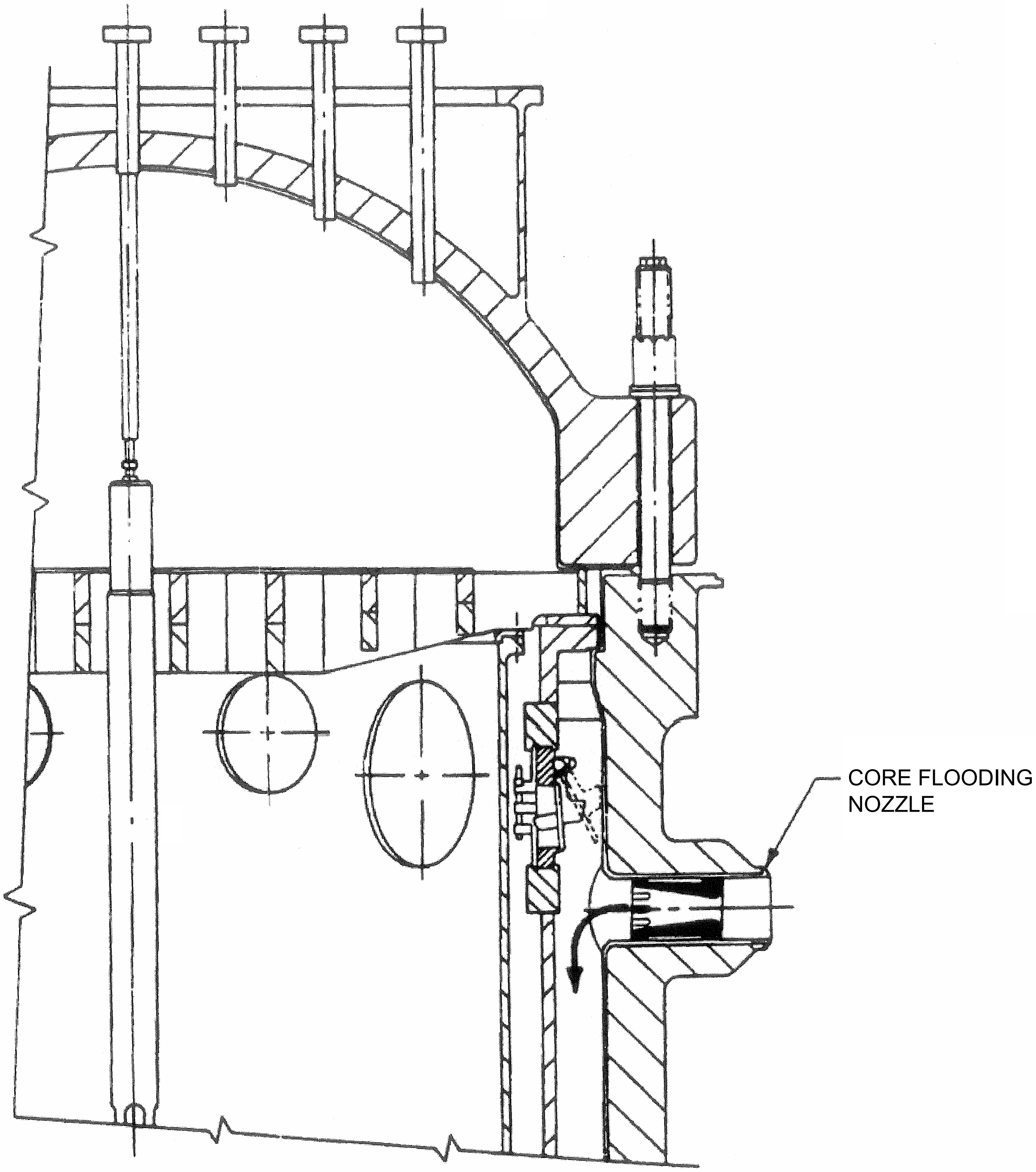
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

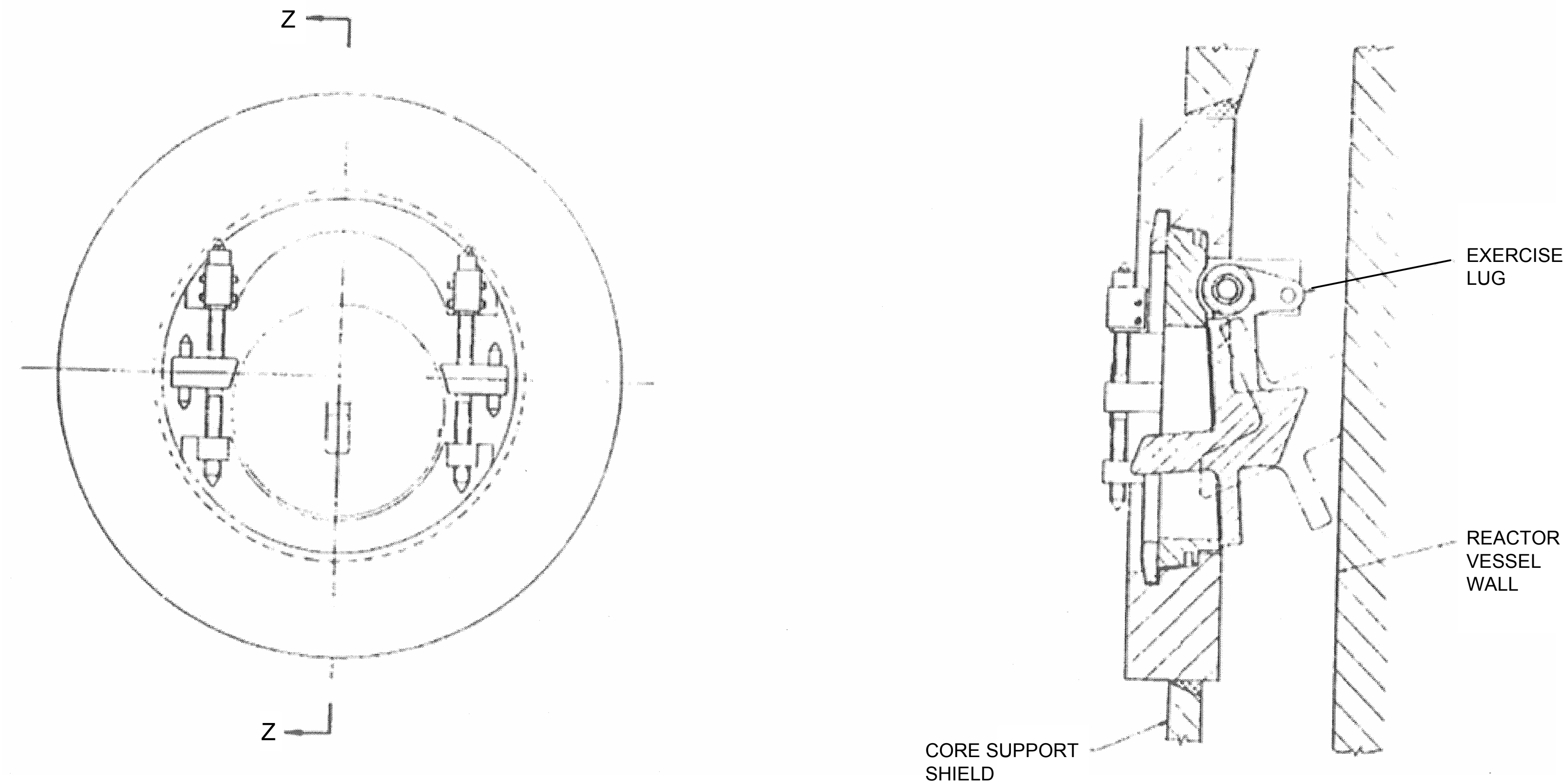
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-61

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
CORE FLOODING ARRANGEMENT	BASED ON DRAWING NO	SHEET / REV.



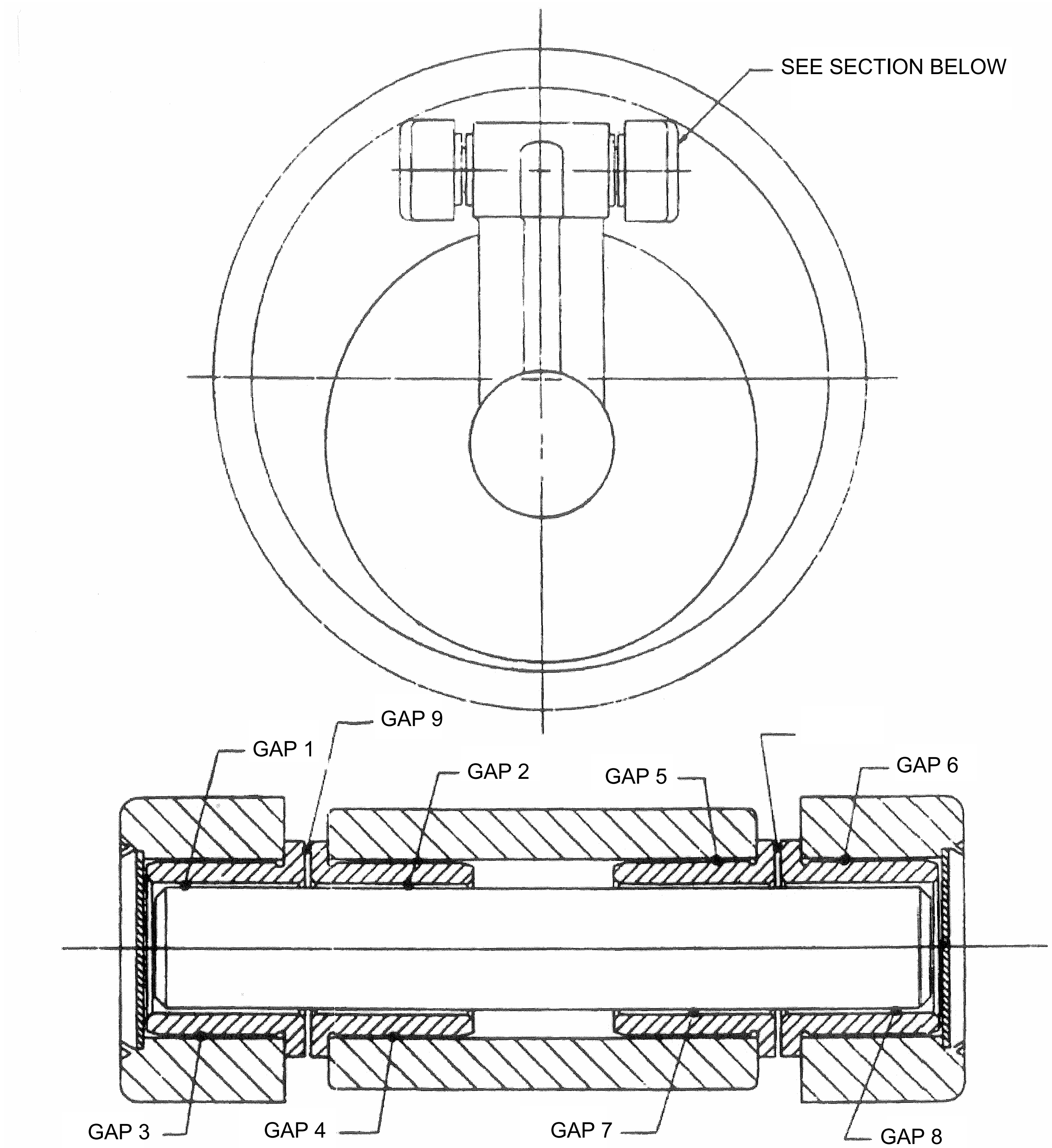
INTERNALS VENT VALVES	SAR FIGURE NO. 3-62
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

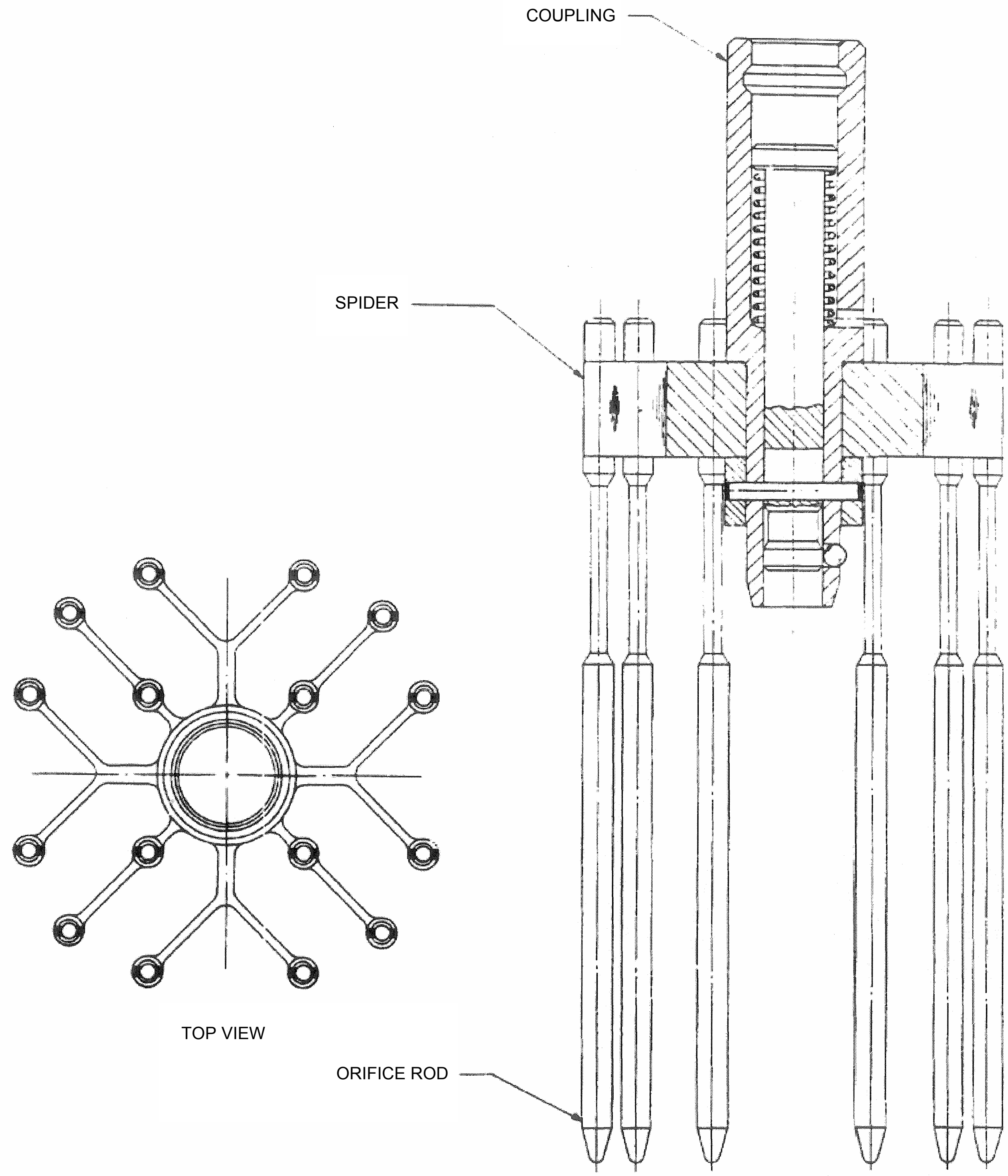
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-63

AMENDMENT 20

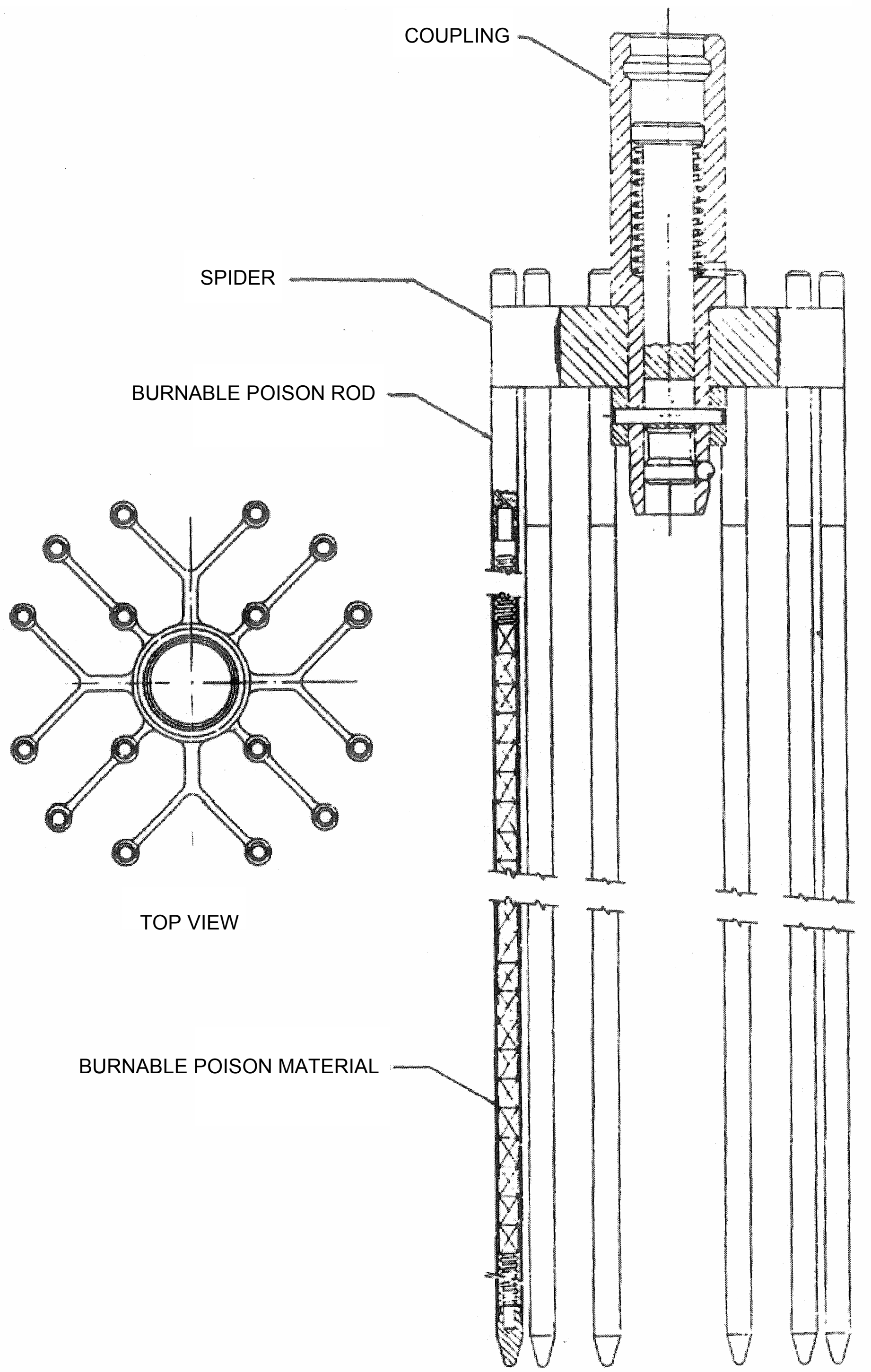
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
INTERNALS VENT VALVE CLEARANCE GAPS	BASED ON DRAWING NO		SHEET	REV.



SAR FIGURE NO. 3-64

AMENDMENT 20

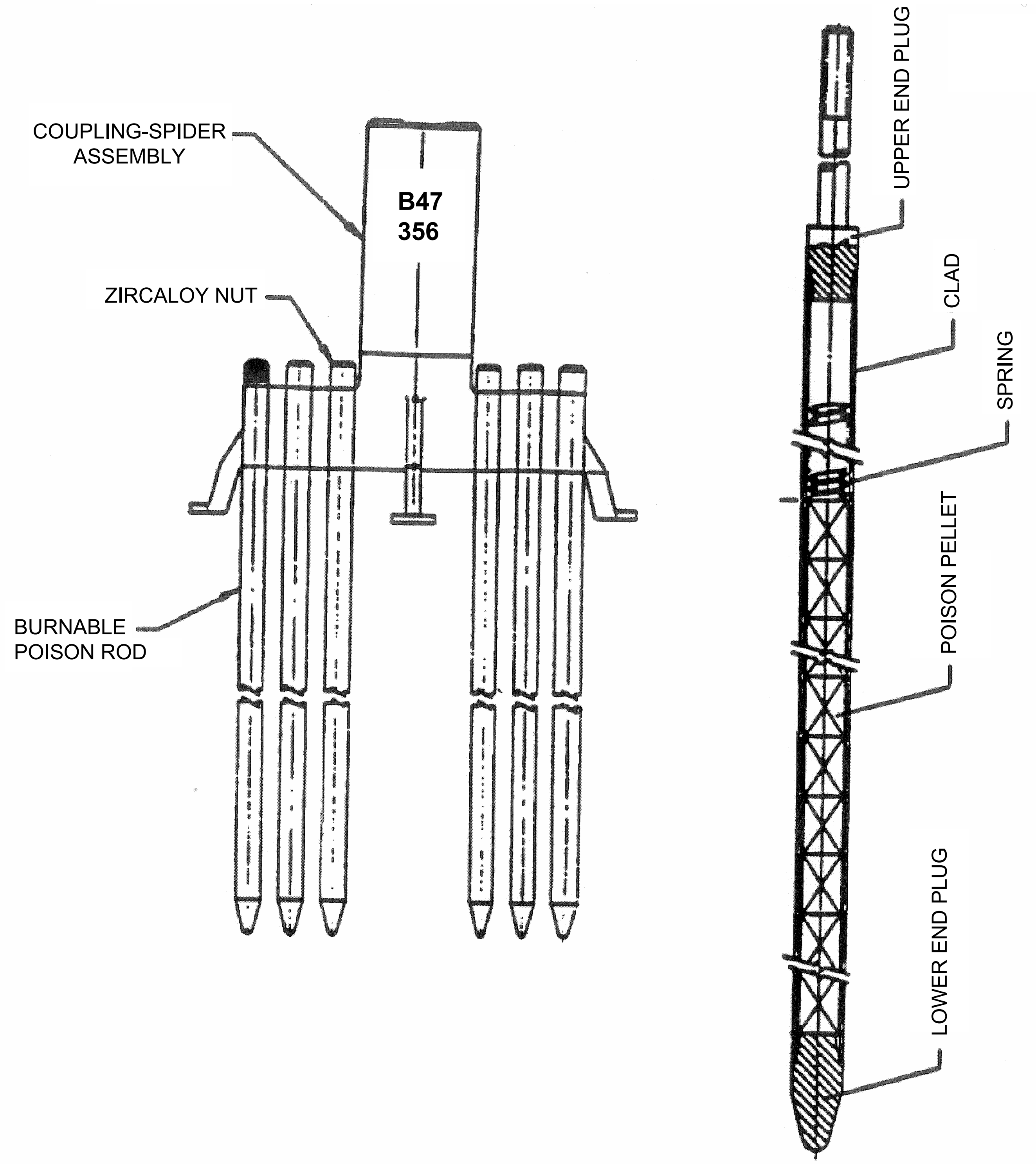
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ORIFICE ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-65A

AMENDMENT 20

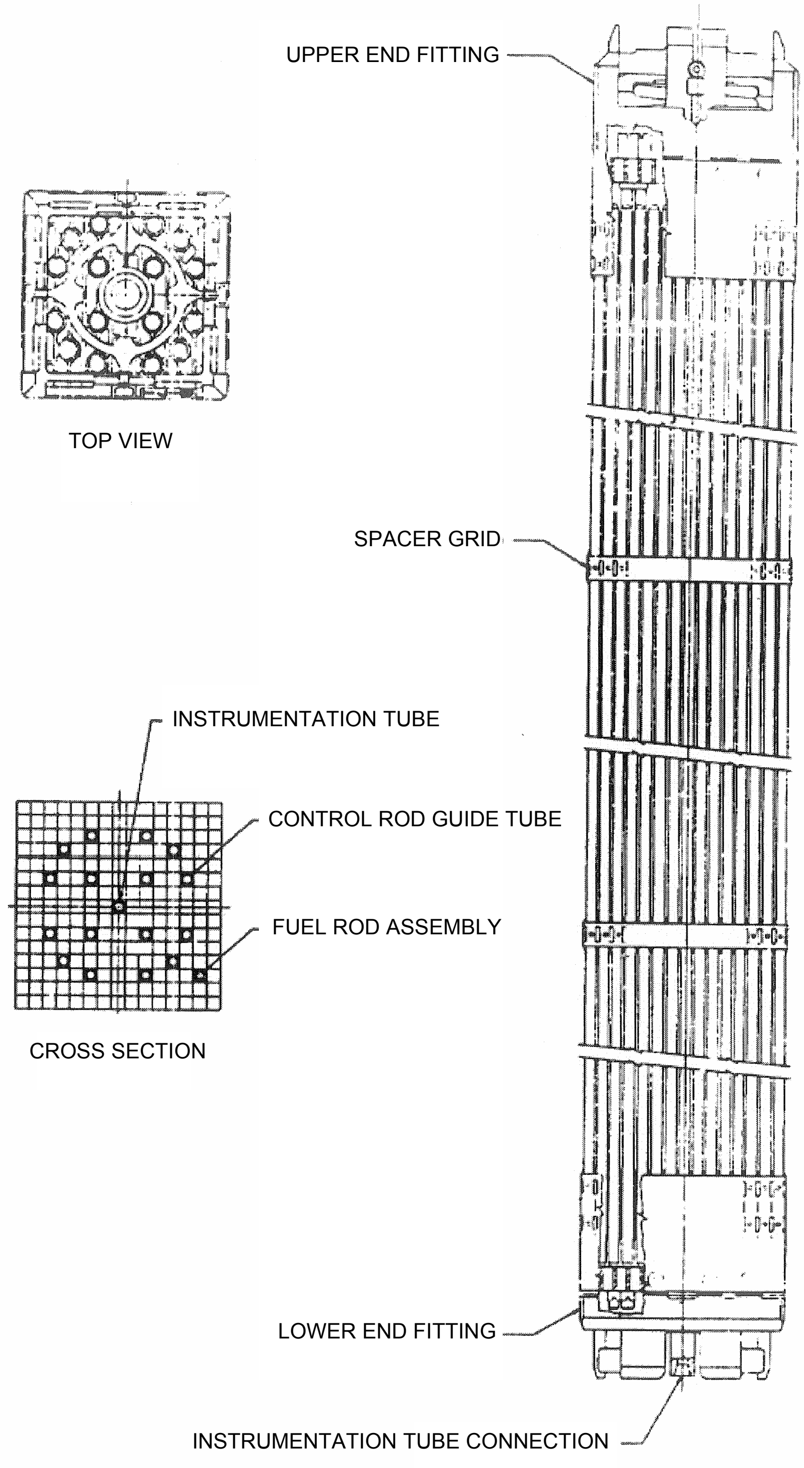
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
BURNABLE POISON ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-65B

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MK-B5 BURNABLE POISON ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.

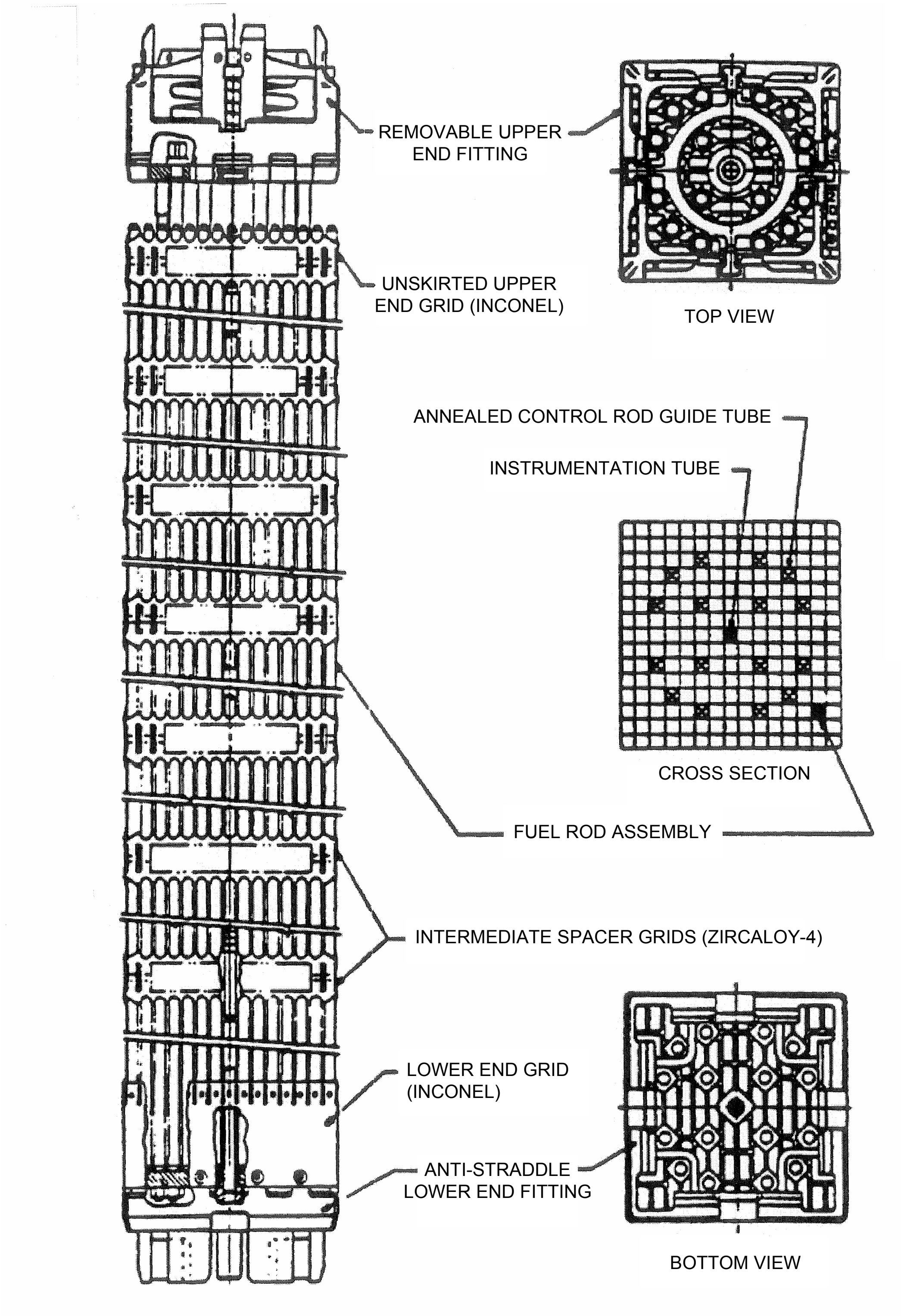


SAR FIGURE NO. 3-66A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

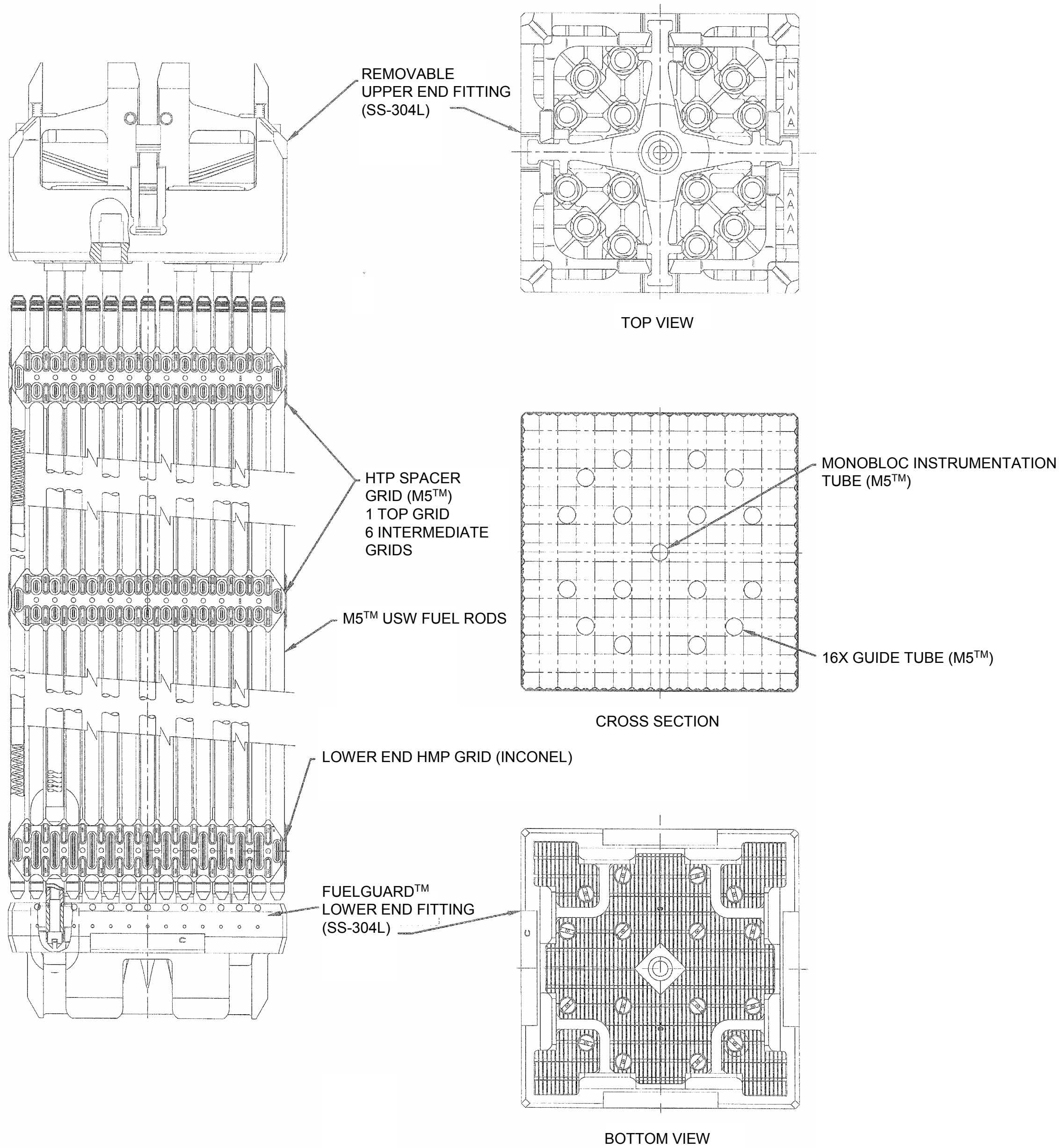
FUEL ASSEMBLY	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-66B

AMENDMENT 20

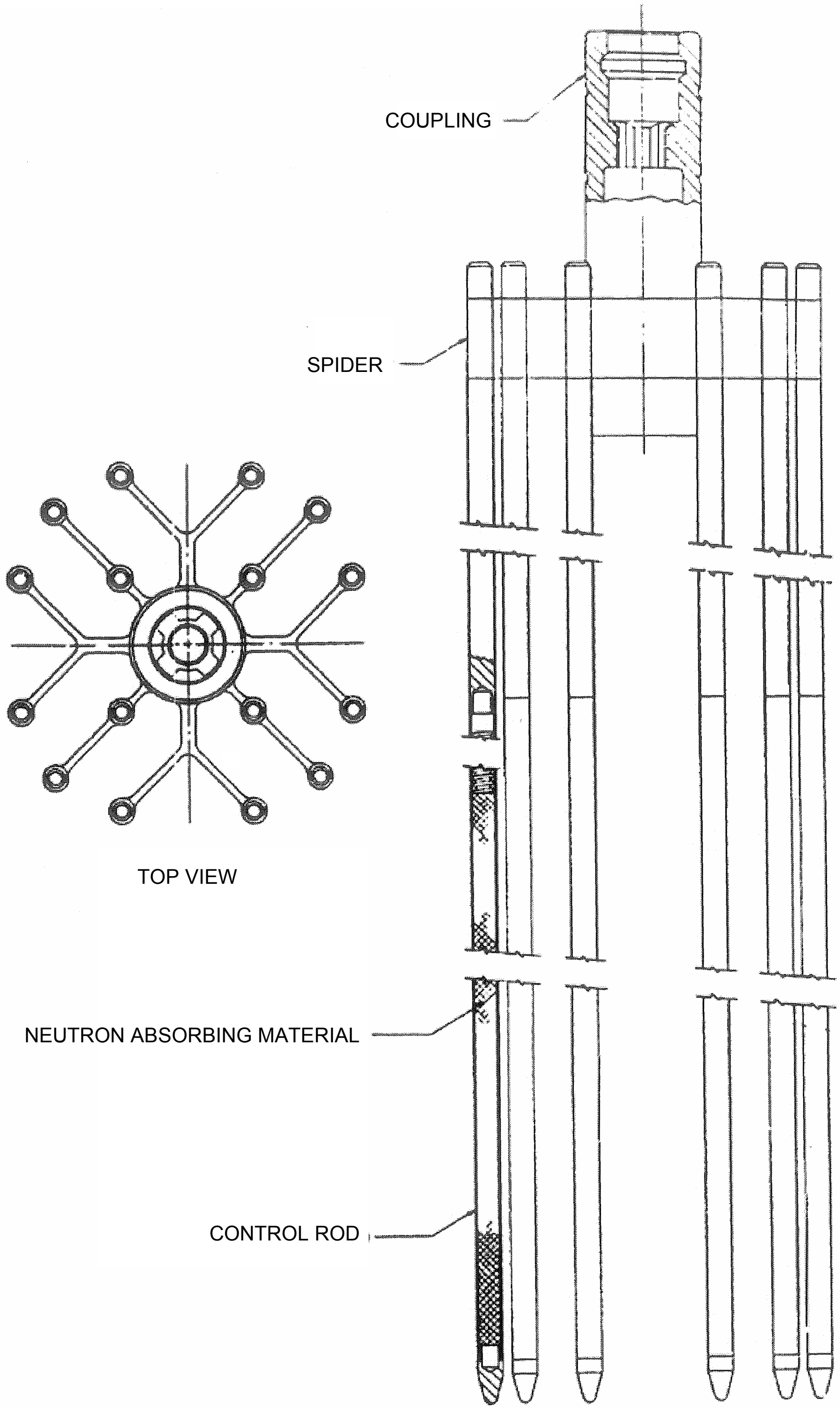
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MARK B2 FUEL ASSEMBLY (MARK – B6)	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-66C

AMENDMENT 20

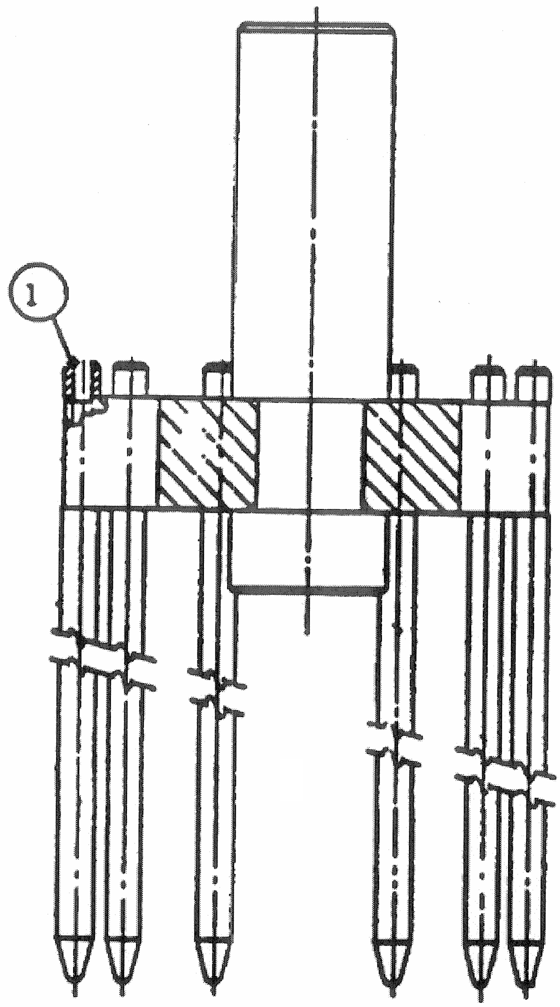
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MARK-B-HTP FUEL ASSEMBLY	BASED ON DRAWING NO	SHEET 1 REV.

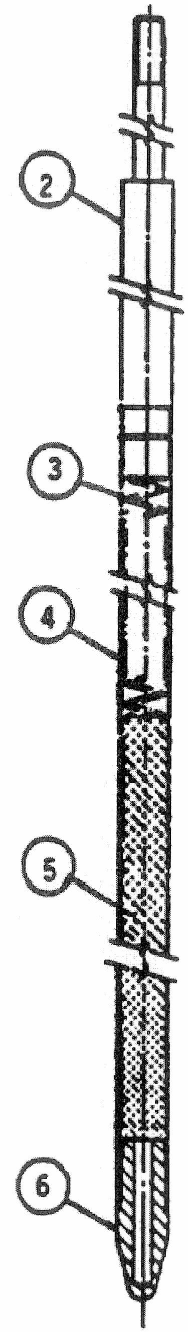


SAR FIGURE NO. 3-67A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
CONTROL ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.





Components
1. Nut
2. Upper End Plug
3. Spring
4. Cladding
5. Absorber
6. Lower End Plug

SAR FIGURE NO. 3-67B

AMENDMENT 20

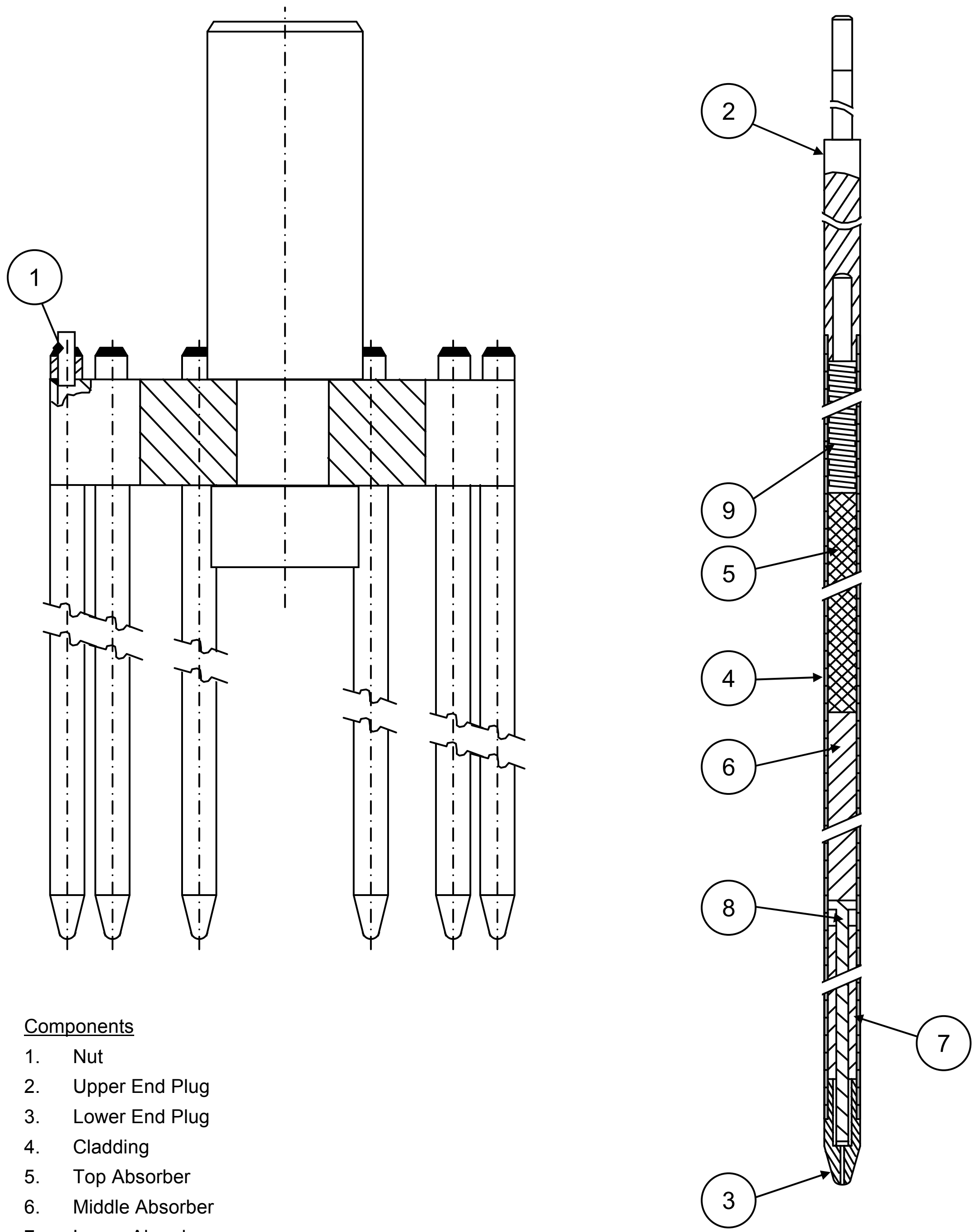
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

EXTENDED LIFE CONTROL ROD ASSEMBLY

BASED ON DRAWING NO	SHEET	REV.

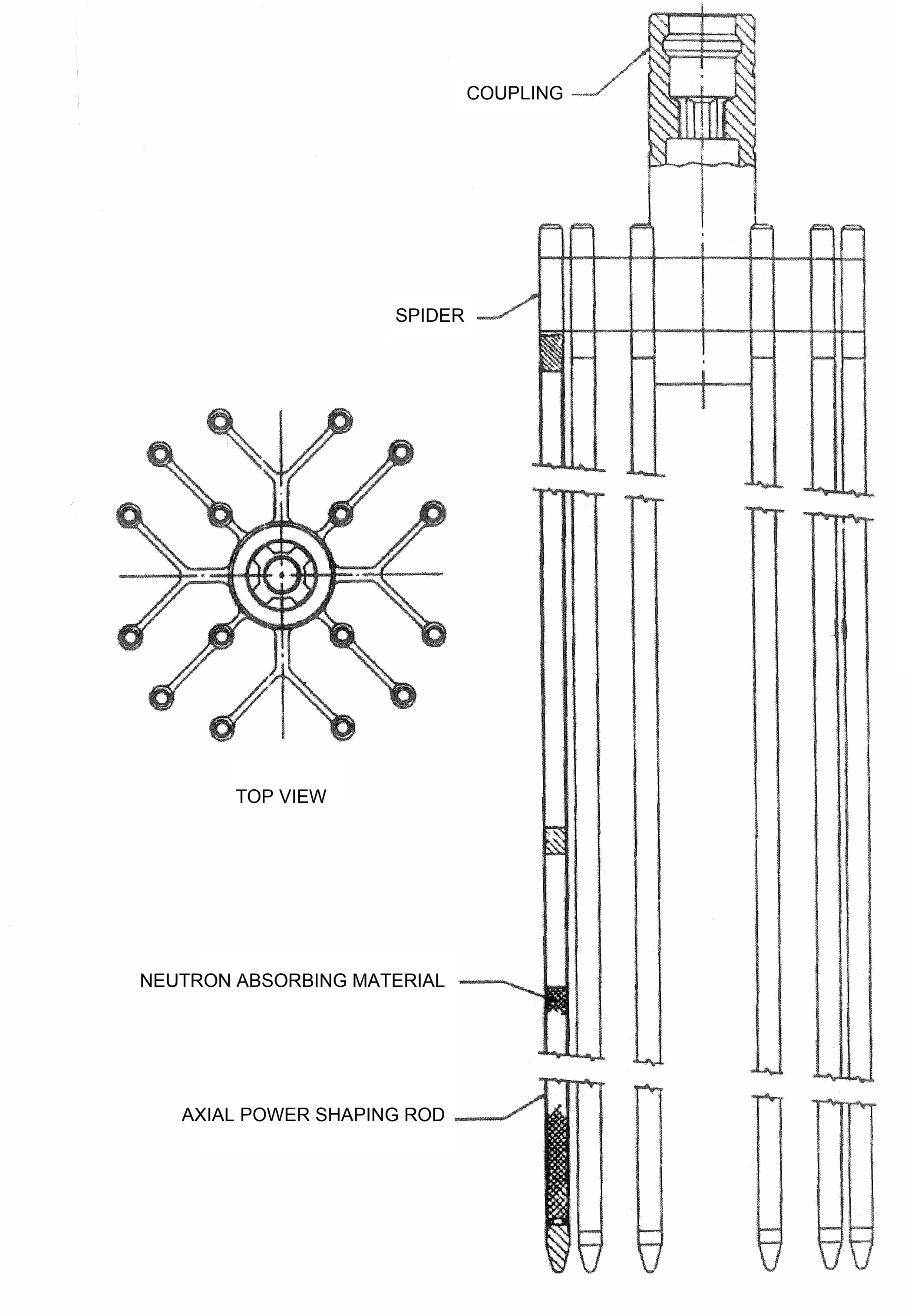


Components
1. Nut
2. Upper End Plug
3. Lower End Plug
4. Cladding
5. Top Absorber
6. Middle Absorber
7. Lower Absorber
8. Stack Support
9. Spring

SAR FIGURE NO. 3-67C

AMENDMENT 22

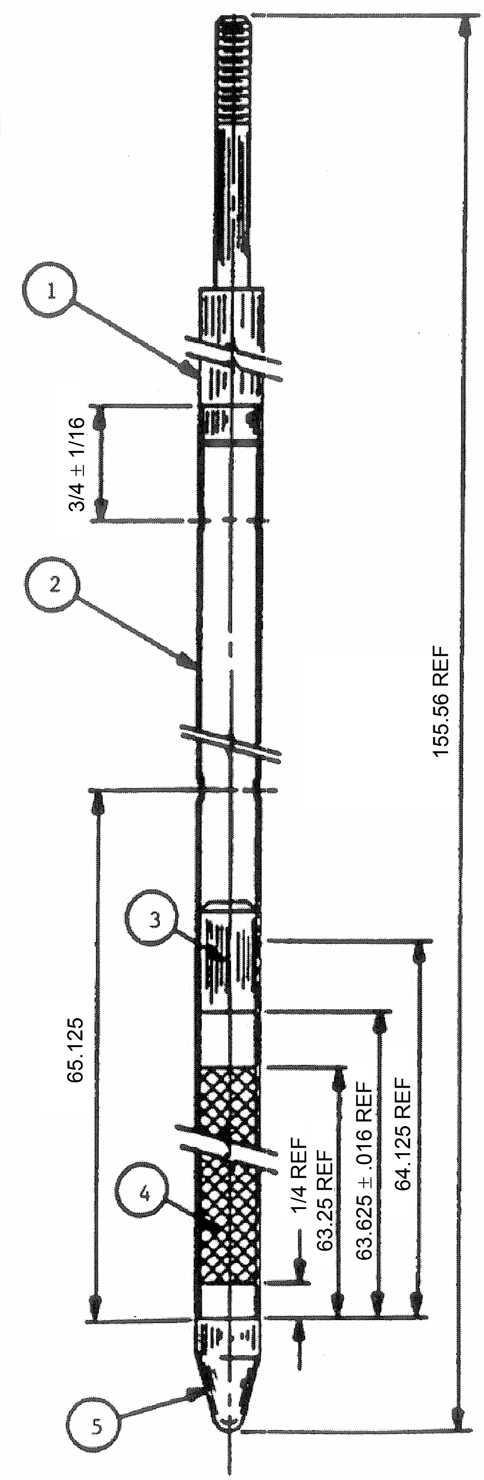
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
EXTENDED LIFE CONTROL ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-68A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
AXIAL POWER SHAPING ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.



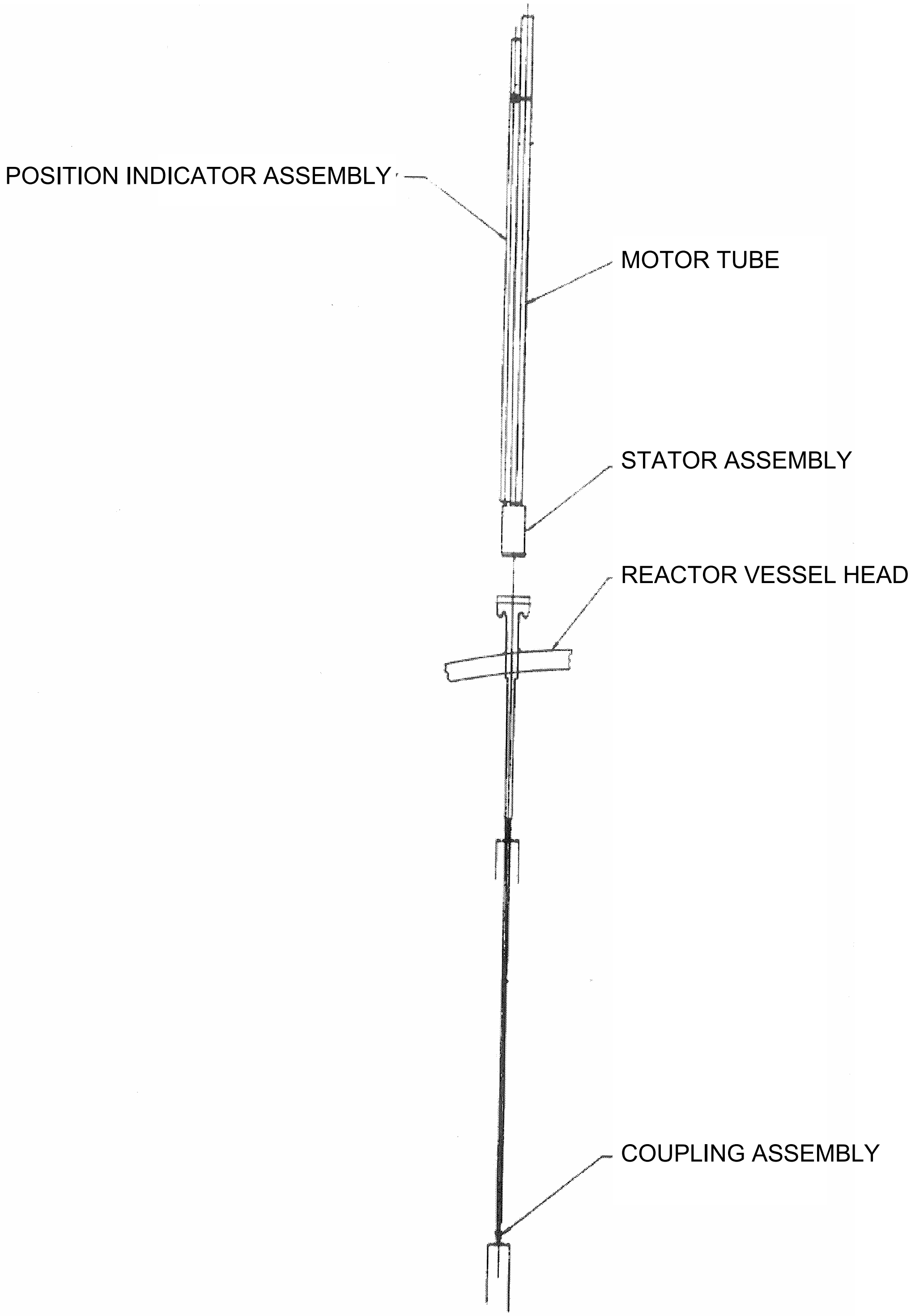
Components

1. Upper End Plug
2. APSR Cladding
3. Intermediate Plug
4. Inconel-600 Absorber
5. Lower End Plug

SAR FIGURE NO. 3-68B

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
(GRAY) AXIAL POWER SHAPING ROD ASSEMBLY	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 3-69

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
CONTROL ROD DRIVE – GENERAL ARRANGEMENT	BASED ON DRAWING NO	SHEET / REV.

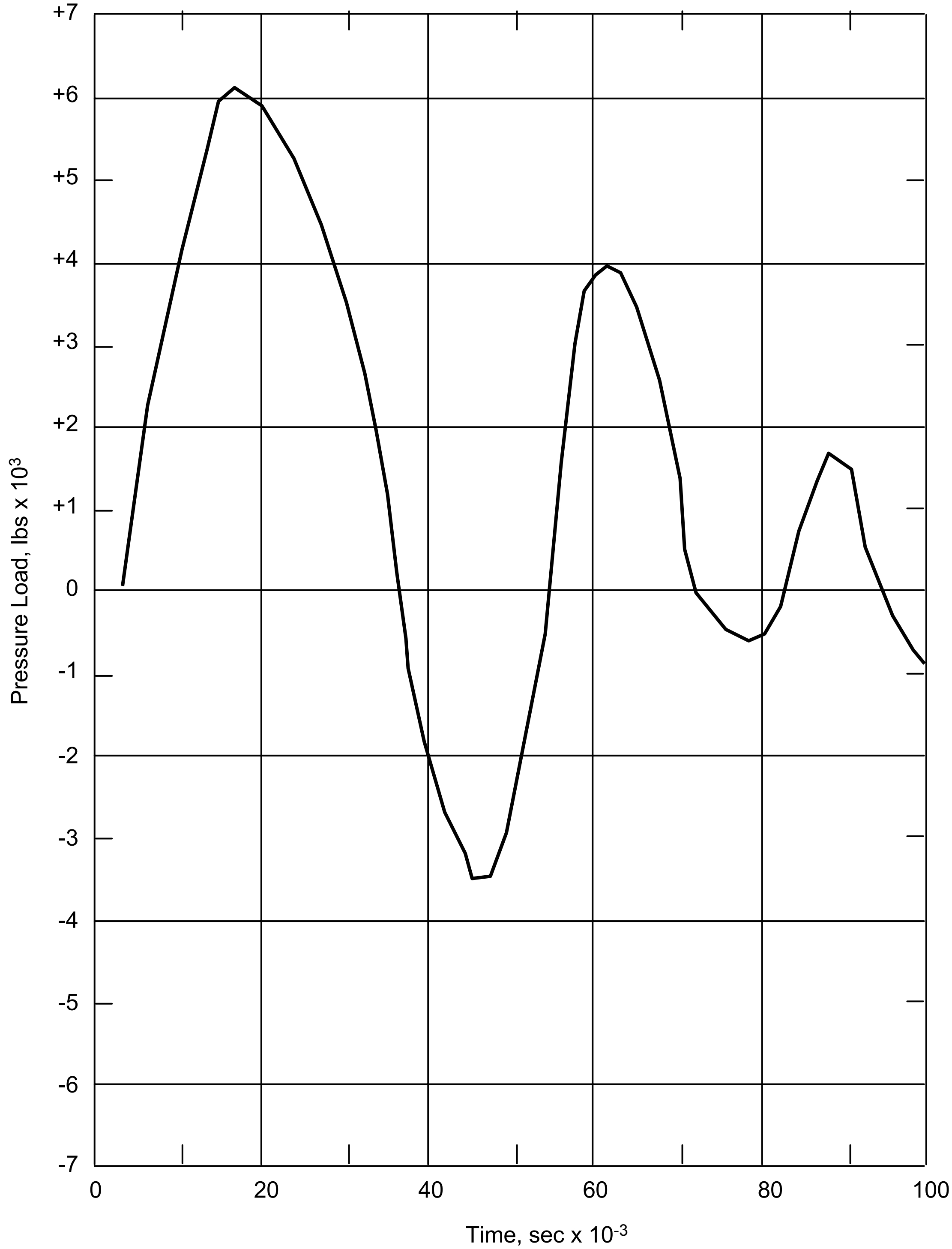
DELETED

SAR FIGURE NO. 3-70

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

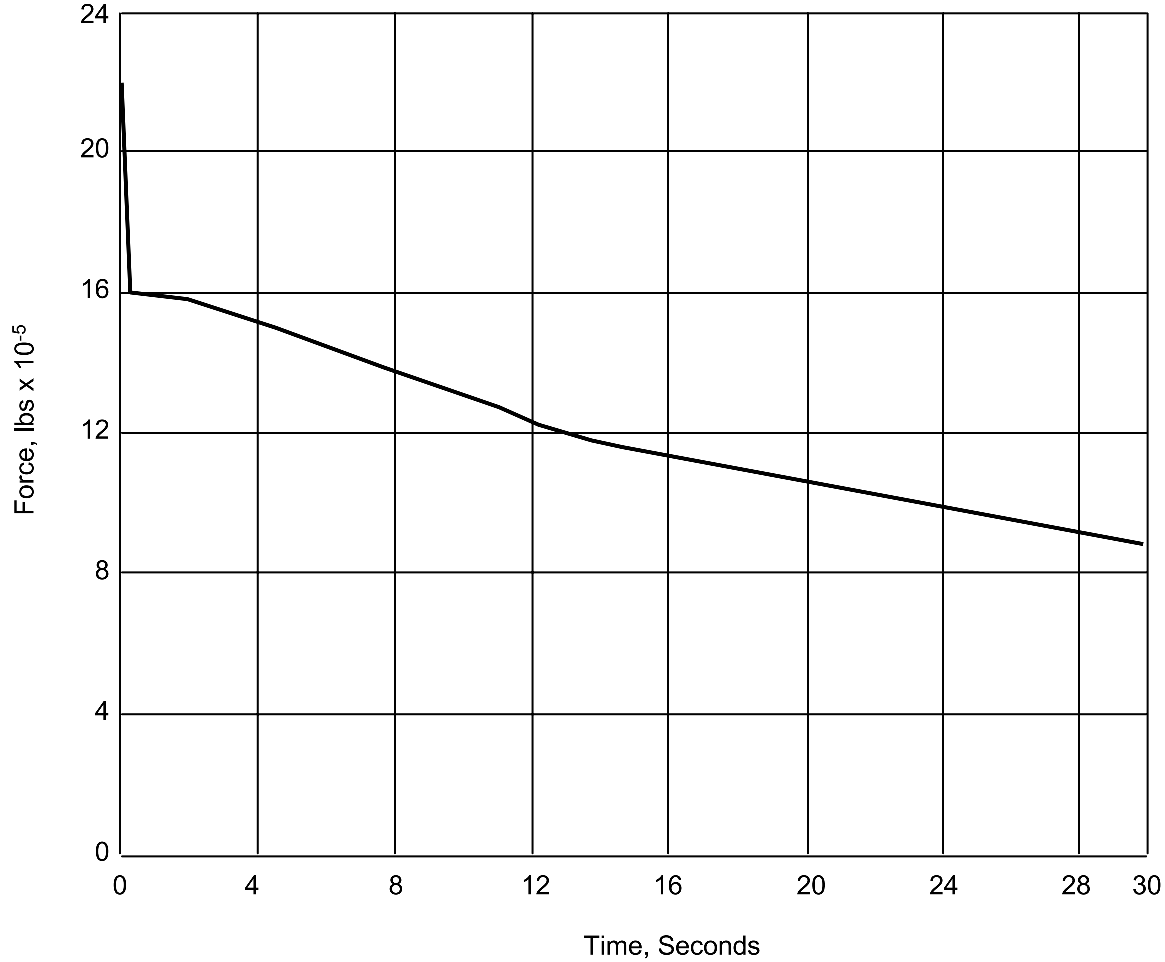
CONTROL ROD DRIVE VERTICAL SECTION	BASED ON DRAWING NO	SHEET 1	REV.



SAR FIGURE NO. 3-71

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
VERTICAL CONTACT LOADING CURVE	BASED ON DRAWING NO	SHEET	REV.



THRUST – TIME CURVE FOR CIRCUMFERENTIAL OR LONGITUDINAL BREAK OF 36-INCH-ID PIPE

SAR FIGURE NO. 3-72

ARKANSAS NUCLEAR ONE

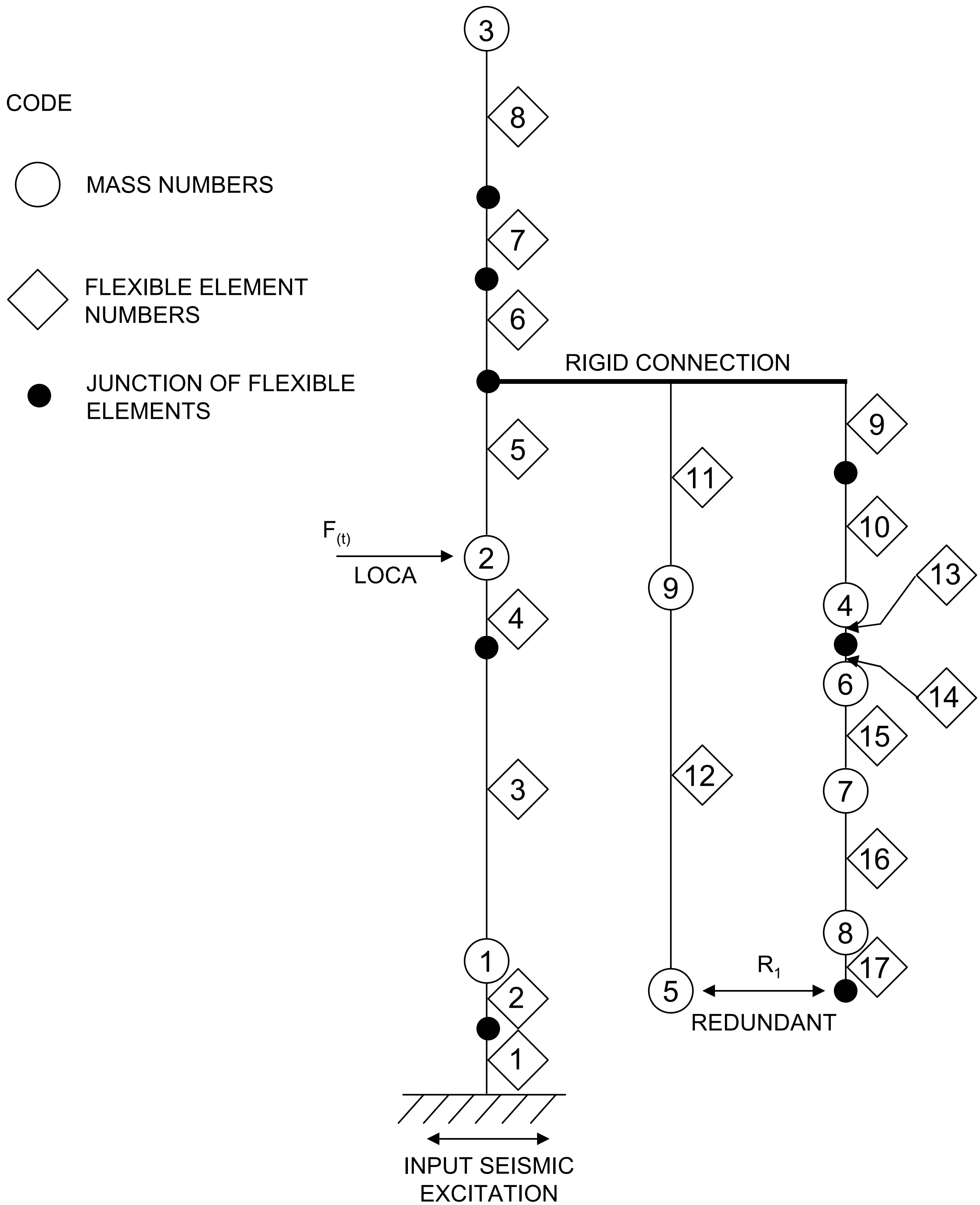
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

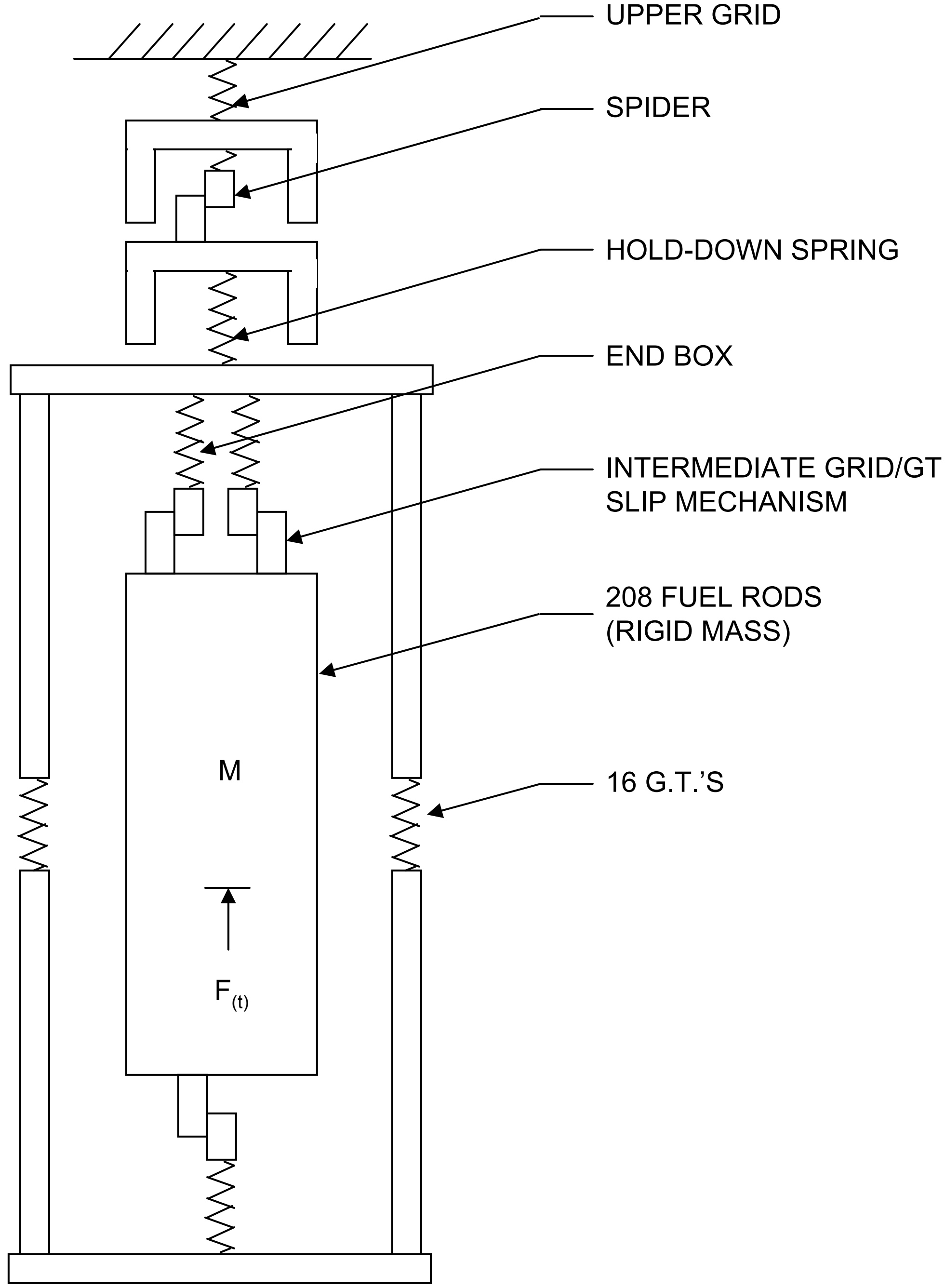
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3-73

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FIRST – SEGMENT MODEL	BASED ON DRAWING NO	SHEET 1 / REV.

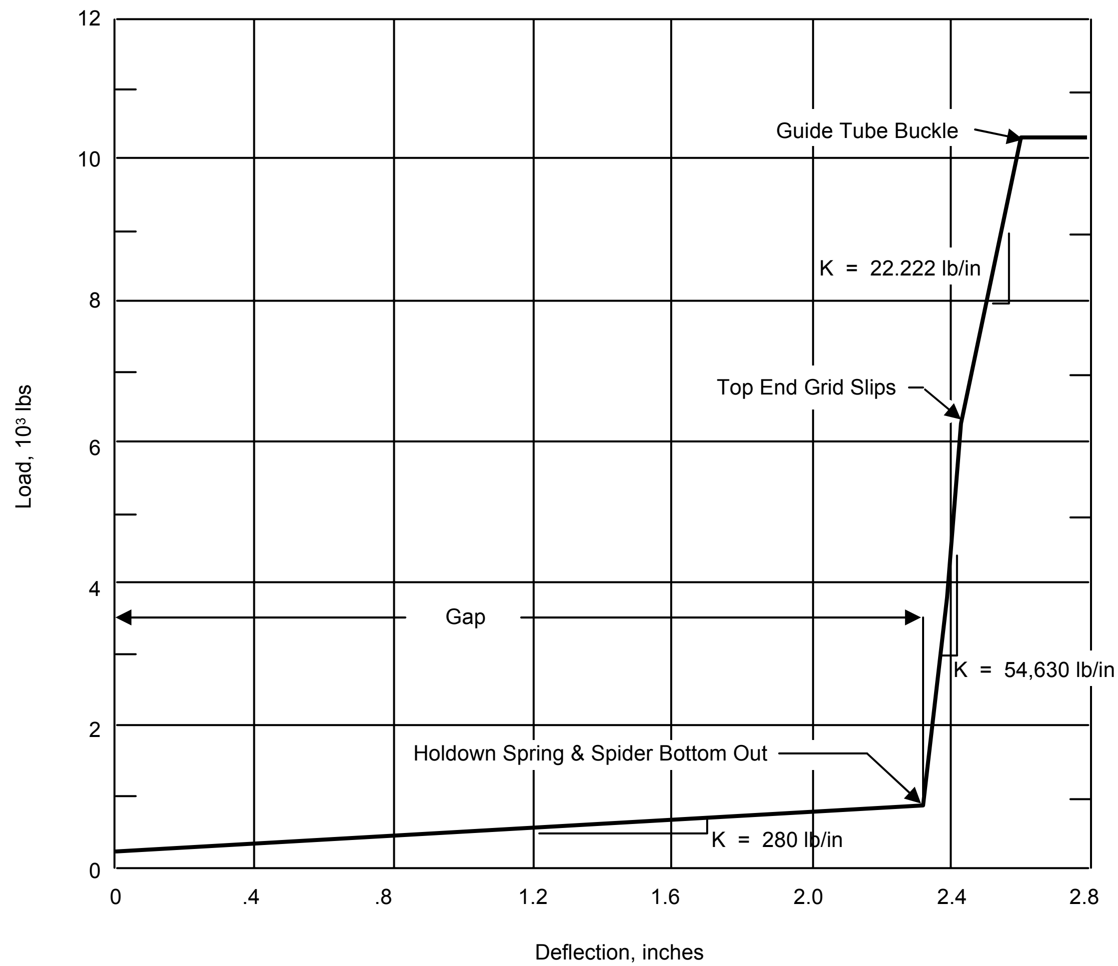


SAR FIGURE NO. 3-74

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

FUEL ASSEMBLY CONTACT MODEL	BASED ON DRAWING NO	SHEET	REV.
		1	



BEGINNING OF LIFE SPRING CURVE		SAR FIGURE NO. 3-75		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	AMENDMENT 20		
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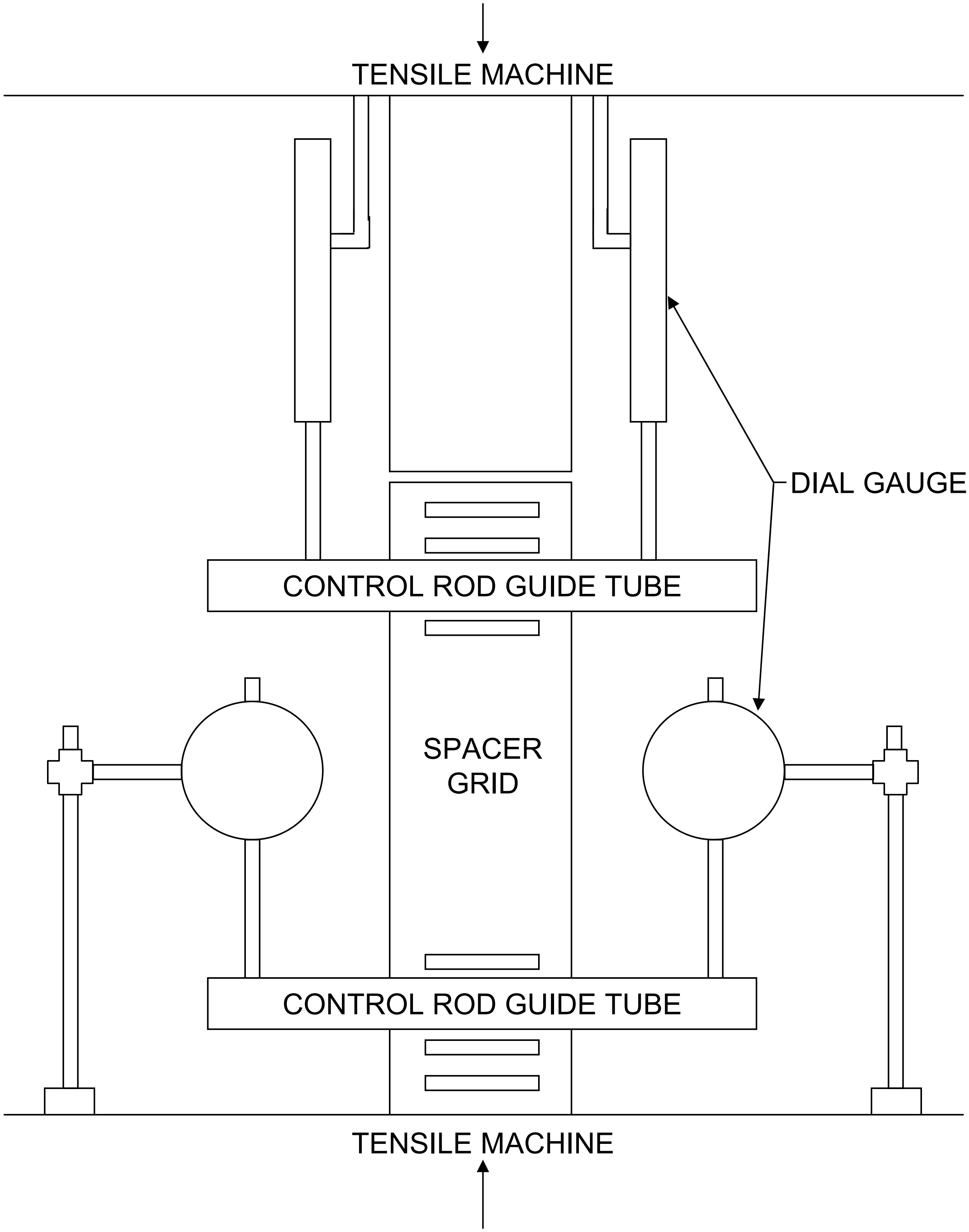

SAR FIGURE NO. 3-76

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ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
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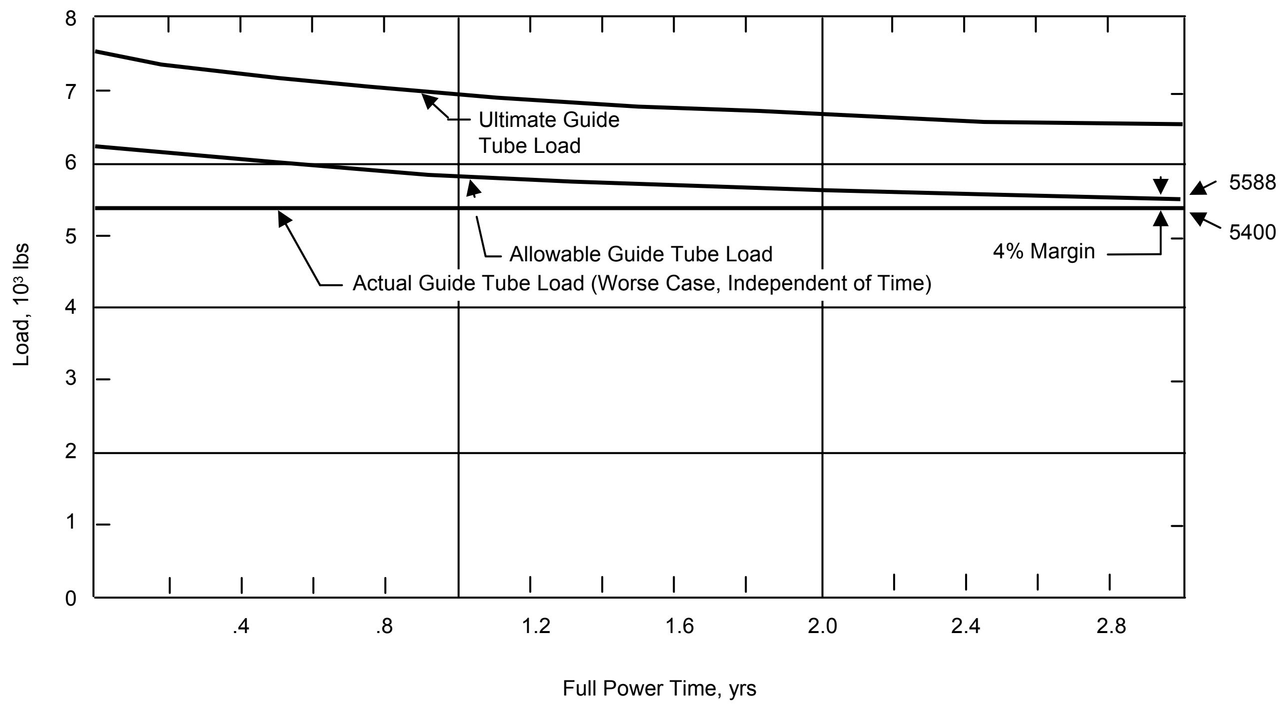

VERTICAL CONTACT ANALYSIS RESULTS	SAR FIGURE NO. 3-77
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

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DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

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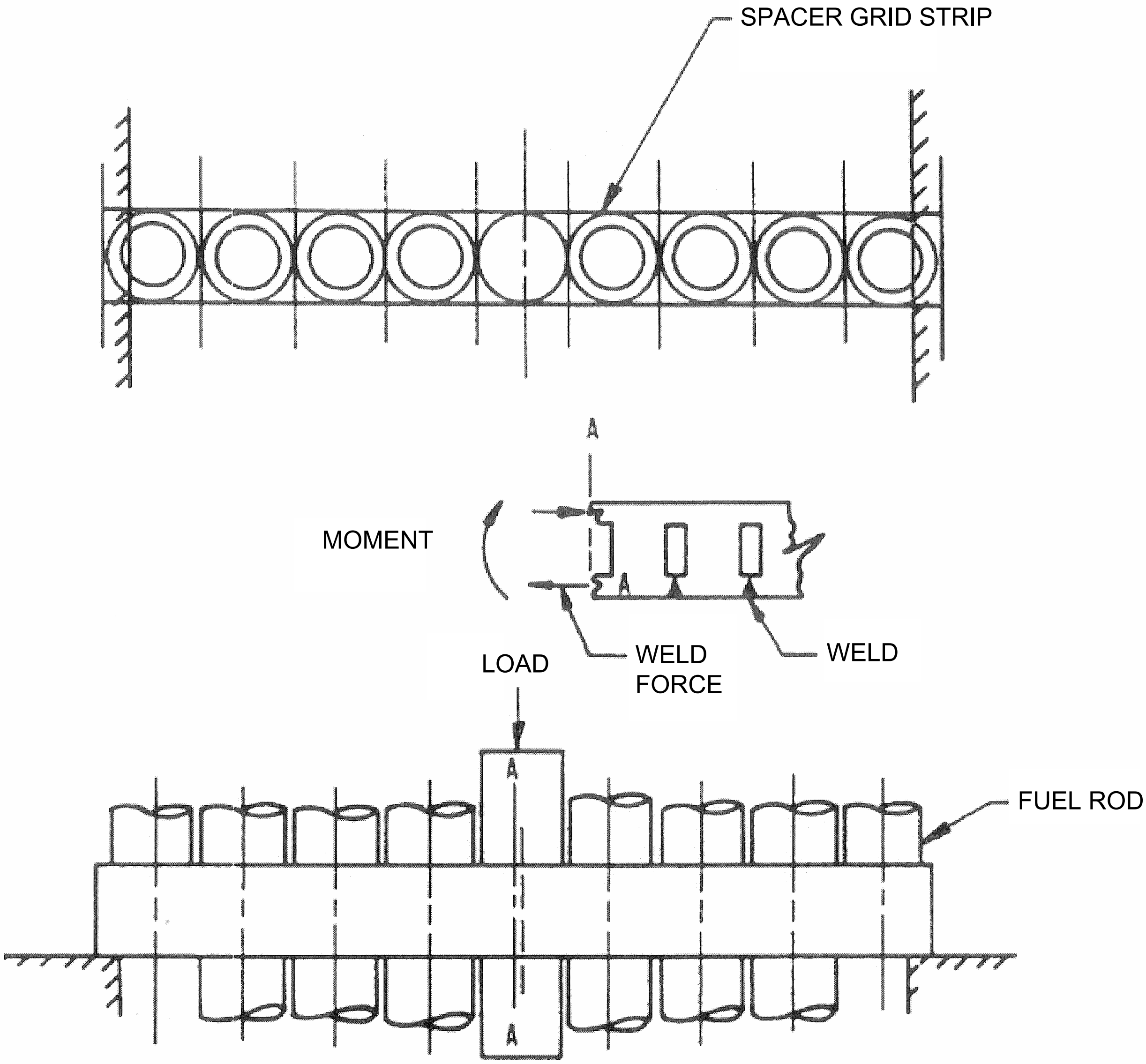

SAR FIGURE NO. 3-78

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UPDATE REFERENCE LIST

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3A ARKANSAS NUCLEAR ONE – UNIT 1, CYCLE 30 RELOAD REPORT

A reload report is prepared to document changes in nuclear, thermal-hydraulic, and mechanical design of the reactor system for each succeeding cycle of reactor operation. The most recent design data governing the operation of the Unit 1 core may be found in the reload report corresponding to the current fuel cycle of the Unit 1 core (Cycle 30). This reload report is provided herein as Chapter 3A. The creation of Chapter 3A in the Unit 1 Safety Analysis Report (SAR) to incorporate the most current reload report requires that the contents of Chapter 3A be replaced in their entirety with each newly issued reload report.

3A.1 INTRODUCTION AND SUMMARY

This report justifies the operation of Cycle 30 of Arkansas Nuclear One, Unit 1 (ANO-1) at the rated core power of 2568 MWt. Included are the required analyses as outlined in the United States Nuclear Regulatory Commission (USNRC) document, "Guidance for Proposed License Amendments Relating to Refueling" (Reference 1).

To support Cycle 30 operation of ANO-1, this report employs analytical techniques and design bases established in reports that have been submitted to and accepted by the USNRC and its predecessor, the United States Atomic Energy Commission (USAEC) (see references).

The Cycle 29 and 30 reactor parameters related to power capability are summarized briefly in Section 3A.5 of this report. The accidents analyzed in the SAR (Reference 2) have been reviewed for Cycle 30 operation. In those cases where Cycle 30 characteristics were conservative compared to those analyzed for previous cycles, new accident analyses were not performed.

The Cycle 30 design includes an end-of-cycle (EOC) hot full power (HFP) extension maneuver which includes the withdrawal of the APSRs, followed by a moderator temperature (T_{avg}) reduction of up to 7 °F (actual), the flexibility to perform a withdrawal of Control Rod Group 7 up to 94% WD, and a power coast down near the end of cycle. The impact of the Group 7 withdrawal flexibility on the Cycle 30 operation was evaluated and found to be acceptable. The effects of the EOC T_{avg} reduction on the reactor coolant system (RCS) structural, RCS operation, core mechanical (fuel), radiological dose consequences, nuclear (design-peaking), and thermal-hydraulic parameters as well as potential effects and/or consequences on loss of coolant accident (LOCA) and non-LOCA safety analyses were evaluated and found to be acceptable. The Cycle 30 design meets the criteria for the EOC HFP extension maneuver imposed by the emergency core cooling system (ECCS) analysis with regard to maximum analyzed temperature reduction and moderator temperature coefficient. The evaluations also verified that the operational maneuver at EOC will be accommodated by the core protective and operating limits. In addition, the EOC T_{avg} reduction maneuver will not significantly impact the results of the accident analyses. With respect to the T_{avg} reduction maneuver, it will be necessary to take into account any instrumentation error that may exist. Therefore, the indicated T_{avg} reduction should be less than the maximum allowable T_{avg} reduction of 7 °F by an appropriate instrumentation uncertainty allowance.

USNRC Generic Letter 88-16 (Reference 3) allows the placement of numeric values of certain cycle specific parameters in a Core Operating Limits Report (COLR). The Cycle 30 specific values of the core operating limits, protective limits, and trip setpoints are included in Section 3A.8 of this report. The COLR changes for Cycle 30 are included in Section 3A.8 of this report.

The Technical Specifications (TS) have been reviewed for Cycle 30 operation. Based on the analyses performed, which take into account the Final Acceptance Criteria for Emergency Core Cooling Systems (Reference 4), it has been concluded that ANO-1 can be operated safely for Cycle 30 at a rated power level of 2568 MWt.

For Cycle 30, 16 chrome coated (c-c) fuel rods are present in once-burned lead test assemblies (LTAs) designated as Batch 31C. Similarly, stainless steel rods (SSRs) are included in both burned fuel assemblies and twelve batch 32 feed fuel assemblies (as described in Section 3A.3) for potential baffle-related fuel rod wear during Cycle 30 and future fuel cycles.

The effects of the stainless steel rods, on the nuclear, thermal-hydraulic, mechanical, thermal design, LOCA, non-LOCA, radiation, and maneuvering analyses have been evaluated using NRC approved methodologies and the guidelines prescribed in Reference 25.

Revision 1 of this reload report is issued to restrict positive imbalance trip and alarm setpoints. The restricted imbalance limits provide additional operational margins for power peaking. The updated setpoints are found in Section 3A.8 and Figures 3A-7, 3A-8, 3A-15, 3A-16 and 3A-17 in Section 3A.12.

3A.2 OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analysis of ANO-1 is Cycle 29. Cycle 29 achieved criticality on November 18, 2019 and is expected to shut down in April 2021. Cycle 30 was analyzed to 506 effective full power days (EFPD) based on a Cycle 29 design cycle length of 495 +10/-30 EFPD. The analyses for Cycle 30 are based on a depletion of Cycle 29 that used core follow depletion up to 124.60 EFPD and nominal steady-state design depletion for the remainder of the cycle. Although this depletion did not include actual operational history of the entire Cycle 29, it is the best representation of actual Cycle 29 operation for the basis of the analysis described in this report.

3A.3 GENERAL DESCRIPTION

The ANO-1 reactor core is described in detail in Chapter 3 of the ANO-1 SAR (Reference 2).

The Cycle 30 core contains 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods (nominally), 16 control rod guide tubes, and one incore instrument guide tube. Four fresh fuel assemblies (Batches 32B1-32B4) contain four (4) stainless steel rods (SSRs) each and eight fresh fuel assemblies (Batches 32C1-32C8) contain contain three (3) SSRs each. Burned LTAs in Batch 31C contain one (1) SSR and two (2) c-c rods containing uranium dioxide fuel pellets each. Additionally, burned Batches 30B, 30C, and 31B1 contain SSRs as illustrated in Figure 3A-1. All fuel assemblies include dished-end, chamfered, cylindrical pellets of uranium dioxide and use fuel rods with $M5_{Framatome}$ cladding.

All fuel assemblies in the core have an average nominal fuel loading of 488.3 kg of uranium, excluding the impact of SSRs. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 3A-1 for all fuel assemblies.

Figure 3A-1 is the fuel shuffle diagram for ANO-1, Cycle 30. The initial enrichments of all fuel batches are presented in Figure 3A-2. Implementation of radially zone-loaded enrichments within the fuel assembly results in a reduction in the radial peak pin power relative to an assembly with a uniform enrichment loading. Selected fuel rod locations within the zone-loaded assembly, as shown in Figure 3A-2, contain reduced enrichment fuel to obtain this effect. Fifty-six Batch 29 assemblies, four batch 27A1 assemblies, and one Batch 26A4 assembly will be discharged at the end of Cycle 29.

The fifty-six twice-burned Batch 30 assemblies and the sixty once-burned Batch 31 fuel assemblies will be shuffled to new locations. The twice-burned Batch 26A5 fuel assembly will be reinserted to location H08 at the core center. The sixty fresh Batch 32 assemblies will be loaded in a symmetric checkerboard pattern throughout the core with predetermined core locations for each of the Batch 32B1-32B4 fuel assemblies and each of the Batch 32C1-32C8 fuel assemblies. Figure 3A-3 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 30. No fuel asymmetries exist in the Cycle 30 design as noted in Table 3A-1a.

Reactivity is controlled by 60 full-length silver-indium-cadmium (Ag-In-Cd) control rod assemblies (CRAs), 44 burnable poison rod assemblies (BPRAs), and soluble boron shim. In addition to the full-length CRAs, eight Inconel gray axial power shaping rod assemblies (APSRs) are provided for additional control of the axial power distribution. The 60 Ag-In-Cd CRAs are the extended life control rod assemblies (ELCRAs) with a three-segment design. Figure 3A-4 indicates the Cycle 30 locations of the 60 CRAs and 8 APSRs and the group designations, which are the same as those of the reference cycle. The core locations of the Burnable Poison Rod Assemblies (BPRAs) and concentrations of the lumped burnable poison (LBP) are shown in Figure 3A-5.

3A.4 FUEL SYSTEM DESIGN

3A.4.1 FUEL ASSEMBLY MECHANICAL DESIGN

The types of fuel assemblies and pertinent fuel design parameters for ANO-1, Cycle 30 are listed in Table 3A-1. Batches 26A5, 30, 31, and 32 are the Mark-B-HTP-1 fuel design. Throughout the document, the overall design will be referred to as the Mark-B-HTP design for simplicity unless otherwise noted. The Mark-B-HTP design is mechanically compatible with the previous fuel assemblies supplied by Framatome Inc. Detailed information on the Mark-B-HTP design is given in Reference 22.

The Mark-B-HTP implements the advanced low corrosion $M5_{Framatome}$ fuel rod cladding, guide tubes, instrument tube, and high thermal performance (HTP™) spacer grids. The Mark-B-HTP design includes the FUELGUARD debris resistant lower end fitting, an upper end fitting assembly with a six-leaf hold down spring, and the MONOBLOC instrument tube. The HMP™ lower end grid is manufactured from nickel-alloy 718.

All Cycle 30 fuel assemblies are zone-loaded, meaning that they have more than one fuel rod enrichment within each assembly. The Mark-B-HTP fuel grids are welded directly to the guide tubes. The lower end grid is secured by Zircaloy-4 capture rings welded to the guide tubes above and below the lower HMP grid.

Batches 26A5, 30, 31, and 32 fuel assembly grid corners have been designed (through the use of lead-in surfaces) to minimize the potential for grid hang-up.

SSRs are included in twelve Batch 32 feed fuel assemblies (as described in Section 3A.3) for the mitigation of potential baffle-related fuel rod wear during future fuel cycles. Cycle 30 will continue to operate with c-c fuel rods in Batch 31C LTAs. Aside from the c-c fuel rods, the Batch 31C LTAs are mechanically identical to the other Mark-B-HTP fuel assemblies.

The Mark-B-HTP fuel rod is designed to withstand a 4g axial loading during shipment and handling of the fuel assembly without gaps forming between pellets in the fuel stack. This design condition is achieved with stainless steel springs in the upper and lower plenums of the fuel rod.

Eight (8) gray axial power shaping rod assemblies (APSRAs) and sixty (60) silver-indium-cadmium (Ag-In-Cd) control rod assemblies (CRAs) will be used in Cycle 30. The CRAs are of the extended life design (ELCRA). Forty-four (44) Burnable Poison Rod Assemblies (BPRAs) will be used in Cycle 30.

A total of one hundred and four (104) stainless steel fuel rods (SSRs) will be used in Cycle 30. Forty (40) will be delivered in the fresh Batch 32 fuel assemblies, twenty four (24) will remain in the once-burned Batch 31 fuel assemblies, and forty (40) will remain in the twice-burned Batch 30 fuel assemblies.

3A.4.2 FUEL ROD ANALYSES

The fuel rod designs were analyzed for mechanical performance, and the results demonstrated that the applicable criteria, as specified in References 5 and 6 were met.

The fuel rod mechanical performance evaluations include consideration of SSRs and shutdown flexibility for the previous cycle. For Cycle 30, SSRs are present in both reinserted fuel

assemblies from Cycle 29, and fresh fuel assemblies. The effect of the inclusion of SSRs on the maximum predicted pin peaks for all batches was evaluated according to Reference 25 as approved by the NRC.

For Cycle 30, 16 c-c rods are present in once burned LTAs. The c-c UO_2 rods were determined to not be limiting for Cycle 30 and the ANO-1 $M5_{Framatome}$ UO_2 fuel rod analysis results were shown to be applicable to the c-c UO_2 rods.

3A.4.2.1 Cladding Collapse

The operating power history for the most limiting fuel assembly was determined for each of the Cycle 30 fuel batches. A single power history envelope was created to be conservative for all Cycle 30 batches. The analysis was based upon the method of Reference 7.

The COPERNIC (Reference 23) and CROV outputs confirm that collapse will not occur for Mark-B-HTP rods with chamfered pellets operating within the analyzed conditions. The analysis was performed for Mark-B-HTP fuel rods with chamfered pellets.

3A.4.2.2 Cladding Stress

The stress parameters for the Cycle 30 fuel rods are enveloped by a conservative fuel rod stress analysis. For design evaluation, certain stress intensity limits for all Condition I and II events must be met. Limits are based on American Society of Mechanical Engineers (ASME) criteria. Stress intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities are compared to S_m. S_m is equal to two-thirds of the minimum yield strength in the hoop direction as a function of operating temperature. The stress intensity limits for Conditions I and II are as follows:

- Primary general membrane stress intensities (P_m) shall not exceed 1.5 S_m for $M5_{Framatome}$ in compression, and S_m for $M5_{Framatome}$ in tension.
- Local primary membrane stress intensities (P_l) shall not exceed 1.5 S_m. These include the contact stresses from the spacer grid stop and the fuel rod.
- Primary membrane + bending stress intensities ($P_l + P_b$) shall not exceed 1.5 S_m.
- Primary membrane + bending + secondary stress intensities ($P_l + P_b + Q$) shall not exceed 3.0 S_m.

where:

P_m = General primary membrane stress intensity
P_l = Local primary membrane stress intensity
P_b = Primary bending stress intensity
Q = Secondary stress intensity

Condition	Enveloping Mark-B-HTP Rod Stress Intensity (psi)	Margin of Safety [(Allowed - Actual) / Actual] * 100%
P_m	20,709	38
$P_m + P_b$	26,248	9
$P_m + P_b + P_l$	28,187	2
$P_m + P_b + P_l + Q$	28,681	100

Stress intensity calculations combine stresses so that the resulting stress intensity is maximized. For the Mark-B-HTP rod, positive margins are retained to the design limit. The following are sources of conservatism used in the stress analysis to ensure that all Condition I and II operating parameters were enveloped:

1. Minimum pre-pressure.
2. High system pressure.
3. High thermal gradient across the cladding.
4. Minimum specified cladding thickness.

3A.4.2.3 Cladding Strain

The table below lists the results from the transient strain analysis, which are to be used as an operating envelope that will prevent the cladding from achieving 1.0% transient strain. The axial power imbalance protective limits and corresponding protection system trip setpoints in the COLR ensure that the linear heat rate limits specified in the table will not be exceeded.

Burnup (MWd/mtU)	Mark-B-HTP Fuel Rods LHR at 1.0% Strain (kW/ft)
20,000	23.72
25,000	22.85
30,000	22.26
35,000	21.92
40,000	21.58
45,000	20.71
50,000	20.50
55,000	19.59
60,000	17.73
62,000	17.59

3A.4.2.4 Cladding Oxide

Reference 6 contains the cladding oxide analysis methodology, model, and acceptance criterion, which limit the maximum oxide prediction for cladding waterside corrosion to 100 microns. The limit applies to the high burnup rod in each fuel batch.

Each Cycle 30 batch was modeled using final fuel cycle design data and maximum cycle lengths. Since the predicted results are below 100 microns, the Cycle 30 oxide performance is acceptable.

3A.4.2.5 <u>Cladding Fatigue</u>

Per Reference 5, the predicted cumulative fatigue usage factor (CUF) must be less than 0.90 for the fuel rod cladding over the life of each fuel rod. The fuel rod design is acceptable in terms of cladding fatigue up to a design life of 10 EFPY of operation. The fatigue analysis was performed for Mark-B-HTP fuel rods. The result in the table below shows that the fuel rods satisfy the cladding fatigue criterion for Cycle 30.

Design	CUF for UO_2 Fuel Rods
Mark-B-HTP with Chamfered Pellets	0.201

3A.4.3 THERMAL DESIGN

All fuel assemblies in the Cycle 30 core are thermally and hydraulically similar, with batches 26A5, 30, 31, and 32 having the same stack height. The design of the Batch 32 (A, B, C) Mark-B-HTP assemblies is such that the thermal performance of this fuel is similar to the fuel design used in the remainder of the core.

The COPERNIC (Reference 23) fuel thermal performance code was utilized for the centerline fuel melt (CFM), transient cladding strain (TCS), and pin pressure analyses for all of the fuel batches in accordance with the approved methods of Reference 5. The TACO3 fuel performance code (Reference 8) was utilized for the LOCA initialization analysis and performed in accordance with the approved methods of Reference 5.

Linear heat rate (LHR) to CFM limits are determined by the COPERNIC code. The Mark-B-HTP fuel rod internal pressure has been evaluated with the COPERNIC code. The internal fuel rod pressures for all batches at the EOC-30 are predicted to be within the fuel rod gas pressure criterion described in Reference 9.

The fuel rod thermal design evaluations include consideration of stainless steel rods (SSRs) and shutdown flexibility for the previous cycle. For Cycle 30, SSRs are present in both reinsert and fresh fuel assemblies. The effect of the inclusion of SSRs on the maximum predicted pin peaks for all batches was evaluated according to Reference 25 as approved by the NRC.

As stated in Section 3A.4.2, the c-c UO_2 rods were determined to not be limiting for Cycle 30 and the ANO-1 $M5_{Framatome}$ UO_2 fuel rod analysis results were shown to be applicable to the c-c UO_2 rods.

Table 3A-2 provides the thermal parameters for the fuel assembly batches to be utilized in the ANO-1 Cycle 30 core.

3A.4.4 MATERIAL COMPATIBILITY

The compatibility of all possible fuel-cladding-coolant-assembly interactions for Batch 32 fuel assemblies is identical to those of present fuel assemblies because no new materials were introduced in Cycle 30.

Chrome-coated fuel rods were introduced in cycle 29 within LTAs, which remain in the core for Cycle 30. Chrome-coated components have been used in the past in PWRs with no adverse

effects on the RCS. Interaction of the chrome coating with the spacer grids was shown to have no adverse impacts on fuel performance based upon testing and mechanical evaluations.

3A.4.5 OPERATING EXPERIENCE

Cycle 30 contains a full core of the Mark-B-HTP fuel assembly design. HTP™ spacers have been successfully implemented in over 50 nuclear power plant facilities world-wide. The implementation of the HTP™ design at ANO-1 has improved the resistance to grid to fuel rod fretting failures starting with Cycle 20.

3A.5 NUCLEAR DESIGN

3A.5.1 PHYSICS CHARACTERISTICS

Table 3A-3 lists the core physics parameters of design for Cycles 29 and 30. The Cycle 29 EOC parameters shown in Table 3A-3 are based on the original licensed length of 512 EFPD. The values for both cycles were calculated with the NEMO code (Reference 10). Figure 3A-6 illustrates a representative relative power distribution for the beginning of Cycle 30 at full power with equilibrium xenon and nominal rod positions. The differences in enrichment, shuffle pattern, number of BPRAs, and cycle length caused the changes in the physics parameters between Cycles 29 and 30. Calculated ejected rod worths and their adherence to criteria were considered at all times in life and at all power levels in the development of the rod position setpoints presented in Section 3A.8. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with Cycle 30 stuck rod worths is demonstrated in Table 3A-4. Rod position setpoints that ensure the minimum shutdown margin is preserved during power operation are specified in Section 3A.8. The following conservatisms were applied for the shutdown margin calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. Flux maldistribution allowance.
4. Allowance for shutdown flexibility of Cycle 29.

The flux maldistribution allowance is included in the shutdown margin calculations to accommodate operation with an off-nominal power distribution, such as that caused by transient xenon. Note that 0.02% Δk/k of the flux maldistribution allowance accounts for the effects of the sixty-three segment extended life control rod assemblies (ELCRAs) present in Cycle 30. The sixty three-segment ELCRAs have a less than minimal effect on rod worth compared to the previous single segment ELCRA design.

The reference fuel cycle shutdown margin is presented in the ANO-1 Cycle 29 Reload Report (Reference 11). The minimum boric acid concentration and minimum volume storage for the boric acid addition tank (BAAT), per Technical Requirements Manual (TRM) Section 3.5.1, and the borated water storage tank (BWST), per TS Section 3.5.4, are sufficient to meet the requirements of Section B 3.5.1 of the TRM (Reference 12).

3A.5.2 ANALYTICAL INPUT

The Cycle 30 incore measurement calculation constants to be used for computing core power distributions were prepared in the same manner as those for Cycle 29.

3A.5.3 CHANGES IN NUCLEAR DESIGN

The design changes for Cycle 30 consist of a decrease in feed batch enrichment, decrease in the number of BPRAs, and a shorter licensed cycle length. These changes were incorporated in the physics model. SSRs are used in twelve of the fresh fuel assemblies in Cycle 30 in addition to SSRs in several Batch 31 and Batch 30 fuel assemblies. There are no asymmetries in the full core loading by batch (enrichment) or burnup as noted in Table 3A-1a.

The calculational methods used to obtain the important nuclear design parameters for this cycle were the same as those used for Cycle 29, which is the reference cycle. The core design change did not affect the methods for defining the transient neutronic parameters and thus, changes to these calculational methods were not required.

The gray APSRs will be withdrawn from the core near the end of Cycle 30 (455 ± 10 EFPD) where the stability and control of the core in the feed-and-bleed mode with APSRs removed have been analyzed. The results of the analysis are reported in Section 3A.8. The recommended APSR withdrawal procedure for Cycle 30 will follow the power operation guidelines. The operating limits (changes to the COLR) for this reload cycle are given in Section 3A.8.

3A.6 THERMAL-HYDRAULIC DESIGN

The reference core analysis for Cycle 30 corresponds to a full core of Mark-B-HTP fuel assemblies. The departure from nucleate boiling (DNB) performance was determined using the LYNXT (Reference 13) core thermal-hydraulic computer code. The DNB design criterion for the M5$_{Framatome}$ grid fuel assemblies is the Statistical Design Limit, which indicates a 95% probability at a 95% confidence level that DNB at the hot fuel rod will not occur. However, design analyses were performed at the Thermal Design Limit based on the BHTP Critical Heat Flux (CHF) correlation (Reference 21, as corrected by Reference 26), which accommodates margin for other purposes as outlined in Reference 14. The approved methods in Reference 5 and the statistical core design (SCD) methodology in Reference 14 were used in the analysis.

The presence of SSRs in the ANO-1 Cycle 30 core design was considered in the thermal hydraulic analyses. The effects of the SSRs were evaluated in accordance with References 5 and 25. The chrome-coated fuel rods present in Cycle 30 fuel assemblies have also been considered, and the existing methods and analyses are applicable.

The calculation of the core bypass flow for the reference core configuration utilized a conservative number of open assemblies. The ANO-1 Cycle 30 core is comprised of 177 Mark-B-HTP fuel assemblies and has fewer open assemblies than the referenced core configuration which results in a core bypass flow less than the reference core configuration. The DNB evaluation performed for Cycle 20, the first implementation of the Mark-B-HTP fuel assembly, remains valid for Cycle 30 with a full core of Mark-B-HTP assemblies.

The limiting Condition II transient was used to establish the DNB-based operating limits as described in Reference 5.

Table 3A-5 summarizes DNB analysis parameters for Cycles 29 and 30.

3A.7 ACCIDENT AND TRANSIENT ANALYSIS

Each SAR accident analysis has been examined with respect to changes in the Cycle 30 parameters to verify that the SAR analyses are bounding for Cycle 30 operation, and to ensure that thermal performance during anticipated transients and accident events is not degraded.

3A.7.1 RADIATION ANALYSIS

The results of the radiological analysis provided in ANO-1 Alternate Source Term (AST) analysis remain valid for ANO-1 Cycle 30 with stainless steel rods (SSRs) in feed and burned assemblies, and chrome-coated rods. The ANO-1 Cycle 30 final fuel cycle design, Mark-B-HTP fuel assembly parameters, and rod pressure evaluation were reviewed. The addition of chrome-coated rods results in a minimal increase in chromium-51 activity and the resultant dose is minimal compared to dose significant isotopes such as iodine-131. In addition, the activation of chromium is expected to have a minimal impact on dose due to direct radiation during normal and accident conditions. It is concluded that the source term used for the dose analyses remain valid and is applicable for the Cycle 30 core. All doses meet the total effective dose equivalent (TEDE) limits in Part 50.67 (Accident Source Term) of 10 CFR 50 (Reference 24), and the General Design Criteria (GDC)-19 Control Room habitability dose limit of 10 CFR 50, Appendix A.

3A.7.2 NON-LOCA SAFETY ANALYSIS EVALUATION

Each cycle, the key parameters in the non-LOCA safety analyses are verified to remain bounding. A comparison of the key cycle-specific parameters and the corresponding analysis assumptions is given in Table 3A-6. Table 3A-7 lists the analysis status for each event in Chapter 14 of the SAR.

The key cycle-specific parameters for each of the events in Chapter 14 of the ANO-1 SAR were reviewed. It was concluded that the non-LOCA safety analyses remain bounding for Cycle 30 operation. This includes evaluation of c-c fuel, as well as SSRs in both fresh and burned fuel assemblies.

3A.7.3 LOCA EVALUATION

The ANO-1 LOCA analyses are based on the LOCA methodology described in Reference 5 and summarized below.

The BWNT LOCA ECCS Evaluation Model (EM) reported in BAW-10192P-A (Reference 16) as modified by Supplement 1P-A to the EM (Reference 15) has been approved for the analysis of large break and small break loss-of-coolant accidents (LBLOCA and SBLOCA) for the B&W designed plants. The EM is based on the RELAP5/MOD2-B&W code (Reference 17) for blowdown and SBLOCA analyses, the REFLOD3B code (Reference 18) for reflooding data, and the BEACH code (Reference 19) for refill pin thermal analysis. The NRC-approved RELAP5/MOD2–B&W-based topical report, BAW-10164P-A (Reference 17) includes the BHTP CHF correlation that is used for the Mark-B-HTP fuel assembly design. The fuel performance data input to the EM were provided by the TACO3 computer code (Reference 8).

The ANO-1 LOCA analyses demonstrate compliance with the criteria outlined in 10 CFR 50.46 (Reference 4):

1. Peak cladding temperature (PCT) shall not exceed 2200 °F.
2. The percentage of local cladding oxidation shall not exceed 17 percent.
3. The maximum hydrogen generated during the transient shall not exceed that which would be generated by the oxidation of one percent of the fuel cladding.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. The mode of long-term core cooling shall be established.

Consistent with the methods defined in Reference 5 for application of the LOCA analyses to the reload process, the ANO-1 LOCA analyses have been evaluated based on the Cycle 30-specific design, including fuel design, plant configuration, and planned operating conditions, to establish the Cycle 30 maximum-allowable LOCA LHR limits such that the LOCA analyses remain valid for Cycle 30 and ensuring continued compliance to the 10 CFR 50.46 criteria.

3A.7.3.1 Cycle 30 Core Design

Cycle 30 consists of a full core of Mark-B-HTP fuel assemblies. The ANO-1 LOCA analyses modeled the replacement steam generator with 5 percent symmetric steam generator tube plugging and considered all elevations and all times in life. The LOCA LHR limits are applicable for a full core of Mark-B-HTP fuel assemblies. All LOCA analyses used TACO3 fuel performance initializations and considered the same fuel data. The significant fuel parameters for Cycle 30 are compared with those assumed in the Mark-B-HTP LOCA analyses in Table 3A-8. The conclusion of the evaluations was that the application of the full core Mark-B-HTP LCOA LHR limits to Cycle 30 is acceptable.

In addition to the fuel parameters, all other issues not related to the fuel and cycle design, but that could potentially affect the applicability of the LOCA LHR limits, were evaluated. Those issues include limitations and restrictions on the approved LOCA EM and the resolution of LOCA-related Preliminary Safety Concerns (PSCs). All of these issues were reviewed and found to be acceptable for ANO-1 Cycle 30. The LHR limits provided in Table 3A-9A take into account any LHR limit reductions for all the condition reports associated with the Mark-B-HTP LOCA analyses.

The evaluation also incorporates the impacts of SSRs and c-c rods present in the core as described in Section 3A.1 of this report. This core design has been taken into account and the LOCA LHR limits provided in Table 3A-9A are valid for use in the maneuvering analysis to verify the core peaking.

3A.7.3.2 Allowable LHR Limits

All Mark-B-HTP fuel batches in the Cycle 30 core were reviewed and shown to be bounded by an acceptable LOCA analysis. The RELAP5-based LHR limits for UO_2 rods and c-c rods that were used in the core power distribution analysis to determine Cycle 30 core operating limits are shown in Table 3A-9A. At end-of-life (EOL) conditions, with a burnup of 62 GWd/mtU, the LHR limits are significantly reduced from the beginning of life (BOL) and middle of life (MOL) in order to maintain the internal fuel pin pressure less than or equal to the limits based on the NRC-approved fuel rod gas pressure criterion (Reference 9).

The latest LBLOCA analyses have been performed in accordance with the NRC-approved burnup-dependent fuel thermal conductivity degradation methodology as described in the Supplement 1P-A to the EM (Reference 15). In addition, an evaluation was performed to account for fuel pellet thermal conductivity degradation effects with burnup for SBLOCA analyses. The combination of these measures ensures that fuel thermal conductivity degradation has been adequately addressed in the LOCA LHR limits provided for Cycle 30.

The LOCA LHR limits are reported on a nuclear source basis consistent with the values used as input to the core power distribution analysis. The nuclear source basis accounts for all usable energy produced by the fuel rod, some of which is deposited in the moderator and surrounding fuel rods. The thermal source LHR is related to the nuclear source LHR by the steady-state energy deposition factor (EDF) (nuclear source LHR = EDF * thermal source LHR).

The EOC moderator temperature coefficient was determined to remain equal to or more negative than -10 pcm/°F. Therefore the LOCA LHR limits have been verified as applicable to an end-of-cycle moderator temperature reduction of up to 12 °F (actual) reduced for uncertainty. Additionally, the Cycle 30 moderator temperature coefficient at all power levels was verified to remain below the limits required to maintain the predicted consequences of a partial-power LOCA less severe than those predicted in the full-power LOCA analyses.

3A.7.4 CONCLUSIONS

It is concluded by the examination of Cycle 30 core thermal, thermal-hydraulic, and kinetics properties that this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 30. With application of the approved methods from Reference 5 and LOCA EM changes reported via 10 CFR 50.46, the transient evaluation of Cycle 30 is considered to be acceptable. The key safety analysis parameters for Cycle 30 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses. These conclusions have been verified to remain valid with the inclusion of SSRs and c-c rods in Cycle 30.

3A.8 PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS AND CORE OPERATING LIMITS REPORT

This section describes proposed revisions to the Core Operating Limits Report (COLR) for Cycle 30 operation based on reactivity, power peaking, and control rod worth in the Cycle 30 core design. Cycle 30 employs a very low leakage fuel cycle design comprising 177 Mark-B-HTP fuel assemblies (including 60 fresh fuel assemblies), and gray APSRs. The nominal design includes withdrawal of APSRs at 455 ± 10 EFPD and core average moderator temperature reduction of 7 °F minus measurement uncertainty. The analysis incorporates the impacts of the stainless steel rods and the UO_2 $M5_{Framatome}$ c-c rods present in the core, as described in Section 3A.3 of this report. The analysis also incorporates the impact of the Cycle 29 shutdown flexibility, with the Cycles 29 and 30 T_{avg} reduction flexibilities and the Cycle 30 Group 7 withdrawal (to 94%WD) flexibility. The maximum design lifetime of Cycle 30 is 506 EFPD. Shutdown flexibility for Cycle 29 is 495 +10/-30 EFPD. Long-term operation at reduced power has the potential to result in an unanalyzed core power distribution. Guidance for extended operation at reduced power is provided in the Power Operation Guidelines document.

In accordance with USNRC Generic Letter 88-16 (Reference 3) and TS 5.6.5, a cycle specific analysis was conducted to generate axial power imbalance protective limits, the corresponding maximum allowable trip setpoints, and core operating limits and alarm setpoints for rod index, axial power imbalance, and quadrant power tilt. This analysis is based on USNRC-approved methodology described in Reference 5. Transient xenon conditions were included in the analysis so that a power level cutoff hold is not required for Cycle 30 operation. Xenon stability was also examined after the withdrawal of APSRs at 455 EFPD. The xenon oscillation resulting from a design power transient was found to be bounded by the xenon stability index of -0.012288 $hours^{-1}$. A negative value indicates that the oscillation was naturally dampened without requiring control component movement.

The analysis incorporates DNB peaking limits based on the allowable increase in design (radial x local) peaking provided by statistical core design methodology described in Reference 14. The impact of local peaking increases caused by a dropped or misaligned control rod event initiated from within the allowable limits of normal operation has been evaluated in a generic analysis performed for B&W plants. The maximum hot full power dropped rod worth was confirmed to be less than the limit of 0.20% $\Delta k/k$. Additional checks are performed each cycle to confirm that the results based on the generic analysis are applicable. The analysis determined that the core operating limits and alarm setpoints provide protection for the overpower condition that could occur due to nuclear instrumentation (NI) error during an overcooling transient. Based on this analysis and the COLR revisions provided in this report, the ECCS Final Acceptance Criteria limits will not be exceeded, nor will the thermal design criteria be violated.

Axial power imbalance and quadrant power tilt are monitored by three independent measurement systems. The primary system comprises fixed incore detectors supplying signals to the on-line computer where they are processed by the fixed incore detector monitoring system (FIDMS) to calculate three-dimensional power distributions. The minimum incore detector system consists of a subset of the incore detectors that also supplies signals to an analog recorder without benefit of signal processing and signal-to-power conversion. The NI system employs excore detectors monitoring the fast neutron flux leakage from each quadrant of the core. Periodic heat balance calculations are performed to calibrate the NI system so that excore detector currents are scaled to provide an accurate indication of the percent of rated core thermal power. Measurement system-independent rod position and axial power imbalance limits were error adjusted to generate alarm setpoints for power operation. Normal operating limits for Cycle 30 are defined by the error-

adjusted alarm setpoints shown in Figures 3A-9 through 3A-17C. The shutdown margin requirements for operation in Modes 1 and 2 are provided in Table 3A-13. APSR insertion limits and alarm setpoints are specified in Table 3A-10. Quadrant power tilt limits and alarm setpoints are listed in Table 3A-11. Steady-state tilt values are dependent on the power level; larger tilts are permitted during low power operation up to 60% FP. The power imbalance alarm-setpoints in Figures 3A-15, 3A-16 and 3A-17 in Section 3A-12 have been restricted to provide additional operational margin for power peaking.

A description of the LOCA evaluation performed for the Cycle 30 core, including a tabular listing of the LOCA allowable LHR limits, is provided in Section 3A.7.3. The LHR limits specified in Table 3A-9A, as discussed in Section 3A.7.3, were determined based on the NRC-approved RELAP5/MOD2-B&W-based EM with the incorporation of the NRC-approved burnup-dependent fuel thermal conductivity degradation (TCD) method as described in Supplement 1P-A (Reference 15). Table 3A-9A illustrates the relationship between allowable LHR limits, burnup, and axial elevation for Mark-B-HTP fuel. Low EOL LHR limits are set in order to maintain the internal pin pressure less than or equal to the limit specified in the NRC-approved fuel rod gas pressure criterion methodology described in Reference 9.

Each cycle the maneuvering analysis determines if the LOCA LHR limits are in compliance with the axial peaking restrictions from the BWNT LOCA EM. For Cycle 30 it was found that no LOCA LHR limits required a reduction. The resulting LOCA LHR limits are provided in Table 3A-9B.

Positive imbalance trip setpoints are restricted to provide additional operational margins for power peaking. The updated protective limits and setpoints are found in Figures 3A-7 and 3A-8, respectively, in Section 3A.12. The trip setpoints account for additional uncertainties not assumed in the development of the DNBR design limit based on statistical core design methodology. Analysis methods used to establish transient cladding strain, centerline fuel melt, and DNB-based peaking limits are described in Sections 3A.4.2.3, 3A.4.3, and 3A.6, respectively. The DNB-based maximum allowable peaking (MAP) limits that protect the initial condition peaking factors assumed in the loss of forced reactor coolant flow analysis for the Mark-B-HTP fuel are specified in Table 3A-12. Allowable linear heat rates are also constrained to prevent reaching the centerline fuel melt temperature and exceeding 1% uniform transient cladding strain. The axial power imbalance protective limits and trip setpoints prevent these thermal limits from being exceeded. The Cycle 30 specific pressure-temperature protective limit lines are shown in Figure 3A-18. The Variable Low Pressure Trip (VLPT) setpoints for Cycle 30 are provided in Figure 3A-19. The nominal operating power levels for the various RCP combinations are specified in Table 3A-14.

A steady-state core power peaking evaluation of the c-c UO_2 fuel rods showed them to exhibit lower peaking than the peak power fuel rod in the core at all times-in-life. In addition, the Cycle 30 maneuvering analysis also examined the margins for CFM, TCS, DNB, and LOCA fuel limits and determined that the c-c rods were non-limiting for the operating and safety limits established for the ANO-1 Cycle 30 COLR.

3A.9 STARTUP PROGRAM –- PHYSICS TESTING

The planned startup testing program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide information for continued safe operation of the unit. Predicted physics parameters used during startup testing such as rod worths and differential boron worth are taken from the Physics Manual (or equivalent).

3A.9.1 PRECRITICAL TESTS

3A.9.1.1 Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Technical Specifications state that the acceptance criteria for rod drop time from fully withdrawn to 75% inserted shall be less than or equal to 1.66 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop from fully withdrawn to two-thirds inserted by worth. Since the most accurate position indication is obtained from the zone reference switch at the 75% inserted position, this position is used instead of the two-thirds inserted position for data gathering.

3A.9.1.2 RC Flow

Reactor coolant (RC) flow with four RC pumps running will be measured at hot standby conditions prior to criticality. The measured flow shall be within allowable limits.

3A.9.2 ZERO POWER PHYSICS TESTS

The scope of the Zero Power Physics Testing (ZPPT) has been implemented based upon testing described in NRC-approved Topicals BAW-10179P-A (Reference 5) and BAW-10242(NP)-A (Reference 20).

3A.9.2.1 Critical Boron Concentration

Once initial criticality is achieved, equilibrium boron is obtained and the critical boron concentration is determined. The measured critical boron concentration (CBC) is determined by correcting the measured boron for any insertion of control rod group (CRG) 7 by use of predicted rod worth and differential boron worth. This is compared to an appropriate CBC for the actual CRG 7 position. Per Reference 5, the acceptance criterion placed on CBC is that the actual boron concentration shall be within ± 50 ppm boron of the predicted value. Per Reference 20, if any predicted CRG worth is used to correct the measured CBC and the CBC deviation exceeds ± 45 ppm, additional testing is recommended.

3A.9.2.2 Temperature Reactivity Coefficient

The isothermal hot zero power (HZP) temperature coefficient is measured near the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. The change in reactivity is then calculated by the summation of reactivity (obtained from a reactimeter) associated with the temperature change. Per Reference 5, the acceptance criterion for the temperature coefficient is that the measured value shall not differ from the predicted value by more than $\pm 0.2 \times 10^{-4}$ $\Delta k/k/°F$.

The moderator temperature coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator temperature coefficient. Per Technical Specifications, this value shall be also be less than or equal to + 0.9 x 10^{-4} $\Delta k/k/°F$.

3A.9.2.3 Regulating Rod Group / Boron Reactivity Worth

Individual CRG reactivity worths (CRG 6 and 7 as a minimum) are measured at HZP conditions using the boron/rod swap method. This technique consists of deborating the RCS and compensating for the reactivity changes from this deboration by inserting individual CRG 7, 6, and if necessary, 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on reactimeter data, and differential control rod worths are obtained from the measured reactivity worth versus the change in control rod group position. The differential rod worths of each of the control groups are then summed to obtain integral rod group worths. Note that the integral rod group worth for CRG 7 may not include the entire group worth. The initial critical conditions obtained may result in a partial measurement of CRG 7 worth; however, the measured value will always be matched with an appropriate predicted value. Per Reference 5, the acceptance criteria for the CRG worths are as follows:

1. Individual group worth:

$$\frac{\text{predicted value - measured value}}{\text{predicted value}} \text{ x 100\% shall be} \leq 15\%$$

2. Sums of group worths measured:

$$\frac{\text{predicted value - measured value}}{\text{predicted value}} \text{ x 100\% shall be} \leq 10\%$$

The boron reactivity worth (differential boron worth) is measured by dividing the reactivity rate of change (from the reactivity computer) to the boron rate of change from measured boron samples at specific time intervals during the rod worth test. No specific acceptance criterion for measured differential boron worth applies.

3A.9.3 POWER ESCALATION TESTS

3A.9.3.1 Core Symmetry Test

The purpose of this test is to evaluate the symmetry of the core at low power during the initial power escalation following a refueling outage. Symmetry evaluation is based on incore quadrant power tilts during escalation to the intermediate power level. Per Reference 5, the following are review criteria for the core symmetry test:

- The absolute value of the quadrant power tilt should be less than the full power limit found in the COLR.
- Symmetric incore detector readings are within ± 10%.

3A.9.3.2 Core Power Distribution Verification at Intermediate Power Level (IPL) and 100% FP

Per Reference 5, core power distribution tests are performed at the IPL (40 - 80% FP) and at approximately 100% FP. Equilibrium xenon is established prior to the 100% FP test. The test at the IPL is essentially a check of the power distribution in the core to identify abnormalities before escalating to the 100% FP plateau. Peaking factor criteria are applied to the IPL core power distribution results to determine if additional tests or analyses are required prior to 100% FP operation.

The following acceptance criteria are placed on the IPL and 100% FP tests:

1. Per Technical Specifications, the $F^N_{\Delta H}$ and F_Q peaking margin at the core-limiting location shall be greater than zero.

2. Per Technical Specifications, the quadrant power tilt shall not exceed the limits specified in the COLR.

3. The measured radial (assembly) peaks for each fresh fuel location and any limiting burned fuel location shall be within the following limits:

$$\frac{\textbf{predicted value - measured value}}{\textbf{predicted value}} \times 100\% \text{ more positive than the limit in Reference 10}$$

4. The measured total (segment) peaks for each fresh fuel location and any limiting burned fuel location shall be within the following limits:

$$\frac{\textbf{predicted value - measured value}}{\textbf{predicted value}} \times 100\% \text{ more positive than the limit in Reference 10}$$

Items 1 and 2 of the acceptance criteria ensure that the initial condition limits are maintained. The verification at the IPL provides reasonable assurance that power escalation to 100% FP may be accomplished without exceeding limits specified by the safety analysis with regard to DNBR and linear heat rate criteria. Items 3 and 4 are established to determine if measured and predicted power distributions are within allowable tolerances assumed in the reload analysis. For initial startup testing following a refueling outage, the acceptance criteria are the approved reliability factors for the nuclear design code; these values are used for testing to provide assurance that the core and fuel are loaded properly, that the incore detector system hardware and software models are performing appropriately, and that measurements are consistent with design model predictions.

The following review criteria (per Reference 5) also apply to the core power distribution results at the IPL and at 100% FP:

1. The root mean square (RMS) of the differences between predicted and measured radial (assembly) peaking factors should be less than 0.05.

2. For all 1/8 core locations, the (absolute) difference between predicted and measured radial (assembly) peaking factors should be less than 0.10.

The review criteria are established to determine if measured and predicted power distributions are consistent.

3A.9.3.3 Incore vs. Excore Detector Imbalance Correlation Verification

Axial power imbalances, set up in the core by control rod positioning, are read simultaneously on the incore detectors and excore power range detectors. The excore detector offset slope versus incore detector offset slope shall be greater than or equal to 0.96 for the six point method and 0.98 for the three point method, and the y-intercept (excore offset) shall be between -2.5% and 2.5%, inclusive. If either of these criteria is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required slope and/or intercept.

3A.9.3.4 Hot Zero Power to Hot Full Power Reactivity Difference

The measured HFP all rods out critical boron concentration (AROCBC) is determined at ~100% FP by first recording the RCS boron concentration during equilibrium, steady state conditions. Corrections to the measured RCS boron concentration are made for control rod group insertion, xenon, samarium, and power deficit (if not at 100% FP) using predicted data. The review criterion placed on the HFP AROCBC is that the measured AROCBC should be within ± 50 ppm boron of the predicted value.

In addition, the predicted AROCBC at HFP is compared to the measured value at HFP AROCBC to obtain the HFP delta. This HFP delta is then compared to the difference between the predicted and measured HZP AROCBC values (HZP delta). Per Reference 5, the acceptance criterion for this test is that the HFP delta minus HZP delta is within ± 50 ppm.

3A.9.4 PROCEDURE FOR USE IF ACCEPTANCE/REVIEW CRITERIA NOT MET

If an acceptance criterion ("shall" as opposed to "should") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is allowed. Further specific actions depend on evaluation of results. These actions can include repeating the tests with more detailed test prerequisites and/or steps, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

If a review criterion ("should" as opposed to "shall") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is recommended. This evaluation is similar to that performed to address failure of an acceptance criterion.

3A.10 REFERENCES

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Table 3A-1

FUEL DESIGN PARAMETERS

Batch	26A5	30A; 30B; 30C	31A; 31B; 31C	32A, 32B, 32C
No. of Fuel Assemblies	1	44; 4; 8 (total 56)	48; 4; 8 (total 60)	48; 4; 8 (total 60)
Fuel Assembly Type	Mark-B-HTP FUELGUARD	Mark-B-HTP FUELGUARD	Mark-B-HTP FUELGUARD	Mark-B-HTP FUELGUARD
Fuel Rod Type	Mark-B-HTP	Mark-B-HTP	Mark-B-HTP	Mark-B-HTP
Fuel Rod OD, in.	0.430	0.430	0.430	0.430
Fuel Rod ID, in.	0.380	0.380	0.380	0.380
Undensified Active Fuel Length, in.	143.00	143.00	143.00	143.00
Pellet OD, in.	0.3735	0.3735	0.3735	0.3735
Fuel Pellet Initial Density, %TD	96.0	96.0	96.0	96.0
Fuel Batch Enrichment (wt% ^{235}U)	3.76[a]	3.49[a]; 3.30[a]; 3.30[a]	3.78[a]; 3.64[a]; 3.63[a]	3.70[a]; 3.48[a]; 3.48[a]
Average Assembly Burnup, BOC (MWd/mtU)[b]	38,755	35,223; 31,823; 26,003	19,684; 19,739; 13,878	0
Cladding Collapse Burnup (MWd/mtU)	65,000	65,000	65,000	65,000
Maximum Pin Burnup at EOC, 506 EFPD (MWd/mtU)[b]	53,010	54,251; 44,081; 39,421	41,223; 36,959; 29,653	22,214; 20,935; 19,024

(a) Average value.

(b) Burnups are based upon the nominal previous cycle length and are not adjusted for the impact of SSRs.

Table 3A-1A

SUMMARY OF ASYMMETRIC FUEL ASSEMBLY SHUFFLES

Quarter Core Location	Full Core Location	Fuel Batch	Average Initial Enrichment	BOC Burnup* MWd/mtU

There are no asymmetric fuel assembly shuffles for Cycle 30.

* From full core model

Table 3A-2

ANO-1, CYCLE 30 FUEL THERMAL PARAMETERS

Parameters	Batch 26A5	Batches 30A; 30B; 30C	Batches 31A; 31B; 31C	Batches 32A; 32B; 32C
No. of Assemblies	1	44; 4; 8	48; 4; 8	48; 4; 8
Initial Density, %TD	96.0	96.0	96.0	96.0
Initial Pellet OD, in.	0.3735	0.3735	0.3735	0.3735
Initial Stack Height, in.	143.00	143.00	143.00	143.00
Enrichment, wt% ^{235}U	3.757[a]	3.487[b]; 3.302[c]; 3.301[d]	3.777[e]; 3.641[f]; 3.632[g]	3.697[h]; 3.482[i]; 3.481[j]
Nominal LHR at 2568 MWt, kW/ft[j]	5.71	5.71	5.71	5.71
Average BOL Fuel Temperature at LHR_{NOM}, °F	1400	1400	1400	1400
Minimum LHR to Melt, kW/ft	20.01	20.01	20.01	20.01

(a) 192 rods at 3.78 wt% ^{235}U and 16 rods at 3.48 wt% ^{235}U.

(b) 192 rods at 3.51 wt% ^{235}U and 16 rods at 3.21 wt% ^{235}U.

(c) 192 rods at 3.32 wt% ^{235}U and 12 rods at 3.02 wt% ^{235}U.

(d) 192 rods at 3.32 wt% ^{235}U and 13 rods at 3.02 wt% ^{235}U.

(e) 192 rods at 3.80 wt% ^{235}U and 16 rods at 3.50 wt% ^{235}U.

(f) 192 rods at 3.67 wt% ^{235}U, 4 rods at 3.37 wt% ^{235}U, and 8 rods at 3.07 wt% ^{235}U.

(g) 192 rods at 3.67 wt% ^{235}U, 4 rods at 3.37 wt% ^{235}U, and 11 rods at 3.07 wt% ^{235}U.

(h) 192 rods at 3.72 wt% ^{235}U and 16 rods at 3.42 wt% ^{235}U.

(i) 192 rods at 3.50 wt% ^{235}U and 12 rods at 3.20 wt% ^{235}U.

(j) 192 rods at 3.50 wt% ^{235}U and 13 rods at 3.20 wt% ^{235}U.

(k) Based on nominal initial stack height, an Energy Deposition Factor of 0.973 and conservatively incorporating the impact of up to 104 SSRs in the core. Cycle 30 is planned to operate with 104 SSRs and 16 c-c fuel rods.

Table 3A-3

PHYSICS PARAMETERS FOR ANO-1, CYCLES 29 AND 30[(a)]

	Cycle 29[(b)]	Cycle 30 [(c)]
Cycle Length[(d)], EFPD	512	506
Design Cycle Burnup, MWd/mtU	15,211	15,033
Average Core Burnup – EOC, MWd/mtU	32,597	32,316
Initial Core Loading, mtU	86.4	86.4
Critical Boron – BOC, ppm (no Xe)		
HZP[(e)], Group 8 Inserted	1843	1904
HFP[(e)], Group 8 Inserted	1629	1698
Critical Boron – EOC, ppm (eq Xe)		
HZP, Group 8 Out	286	286
HFP[(f)], Group 8 Out	0	0
Control Rod Worths – HFP, BOC, %Δk/k		
Group 6	1.01	0.98
Group 7	0.84	0.91
Group 8 (maximum)	0.15	0.15
Control Rod Worth – HFP, EOC, %Δk/k		
Group 7	0.98	1.03
Maximum Ejected Rod Worth (L-10)[(k)] – HZP, %Δk/k		
BOC, Groups 5 – 8 Inserted	0.31 (L-10)	0.30 (L-10)
465 EFPD[(g)], Groups 5 – 8 Inserted	0.34 (L-10)	0.37 (N-12)
EOC, Groups 5 – 7 Inserted	0.33 (L-10)	0.40 (N-12)
Maximum Stuck Rod Worth (M-13)[(l)] – HZP, %Δk/k		
BOC, Groups 1 – 8 Inserted	0.84 (M-13)	1.07 (N-12)
465 EFPD[(g)], Groups 1 – 8 Inserted	1.06 (M-13)	1.34 (N-12)
EOC, Groups 1 – 7 Inserted	1.05 (M-13)	1.31 (N-12)
Power Deficit, HZP to HFP, %Δk/k		
BOC, Group 8 Inserted	1.65	1.59
EOC, ARO	2.82	2.86
Doppler Coefficient – HFP, 10^{-5} Δk/k/°F		
BOC (no Xe)	-1.60	-1.63
EOC (eq Xe)	-1.79	-1.82

Table 3A-3 (continued)[(a)]

	Cycle 29[(b)]	Cycle 30[(c)]
Moderator Coefficient – HFP, 10^{-4} Δk/k/°F		
BOC (no Xe, 1629 ppm, Group 8 Inserted)	-0.73	-0.60[(h]
EOC (eq Xe, 0 ppm, Group 8 Out)	-3.14	-3.15[(i)]
Steam Line Break Deficit[(j)], – %Δk/k		
EOC (eq Xe, 0 ppm, ARI/SRO)	0.71	0.73
Boron Worth – HFP, ppm/%Δk/k		
BOC	148	147
EOC	123	122
Xenon Worth – HFP, %Δk/k		
BOC (4 EFPD, equilibrium)	2.55	2.56
EOC (equilibrium)	2.69	2.71
Effective Delayed Neutron Fraction – HFP		
BOC	0.0063	0.0063
EOC	0.0053	0.0053

(a) Cycle 30 data are for the conditions stated in this report. The Cycle 29 core conditions are identified in Reference 11.

(b) Cycle 29 data are based on a Cycle 28 (cycle N-1) lifetime of 468 EFPD at 2568 MWt-.

(c) Cycle 30 data are based on a Cycle 29 (cycle N-1) lifetime of 495 EFPD at 2568 MWt.

(d) All end of cycle values calculated at the indicated end of cycle licensing window.

(e) HZP denotes hot zero power (532 °F T_{avg}); HFP denotes hot full power (579 °F T_{avg}).

(f) At HFP all rods out conditions, 0 ppm occurs at 496.8 EFPD for Cycle 29 and 490.8 EFPD for Cycle 30.

(g) Calculated at 475 EFPD for Cycle 29.

(h) The BOC, HZP MTC calculated at 2073 ppmB with no xenon is +0.27 x 10^{-4} Δk/k/°F.

(i) The EOC, HFP MTC calculated at 0 ppmB with equilibrium xenon and rod group 7 at 60% WD is -3.26 x 10^{-4} Δk/k/°F.

(j) Calculated at conservative steam line break conditions.

Table 3A-4

**SHUTDOWN MARGIN CALCULATION FOR ANO-1
CYCLE 30 FROM HFP CONDITIONS**

	BOC %Δk/k	465 EFPD %Δk/k	506 EFPD %Δk/k
Available Rod Worth			
Total rod worth, HZP	7.730	8.298	8.262
Worth reduction due to poison material burnup	-0.199	-0.237	-0.238
Maximum stuck rod, HZP	-1.070	-1.335	-1.314
Net Worth	6.461	6.726	6.710
Less 10% uncertainty	-0.646	-0.673	-0.671
Total available worth	5.815	6.053	6.039
Required Rod Worth			
Power deficit, HFP to HZP	1.595	2.789	2.856
Allowable inserted rod worth	0.332	0.516	0.527
Flux maldistribution allowance	0.370	0.370	0.370
Cycle 29 shutdown flexibility allowance	0.200	0.200	0.200
Total required worth	2.497	3.875	3.952
Shutdown margin (total available worth minus total required worth)	3.318	2.179	2.086

NOTE: The required shutdown margin is 1.00 %Δk/k.

Table 3A-5

DESIGN ANALYSIS CONDITIONS, CYCLES 29 AND 30

	Cycle 29	Cycle 30
Design power level, MWt	2568	2568
RCS design flowrate, gpm	352000	352000
DNBR modeling	Crossflow	Crossflow
With SCD Methodology		
Nominal system pressure, psia	2200	2200
Nominal flow fraction of design flowrate	1.09	1.09
Nominal core bypass flow, % (a)	6.24	6.24
Nominal design radial-local power peaking factor	1.734	1.734
Reference design axial flux shape	1.65 chopped cosine	1.65 chopped cosine
Nominal Hot channel factors		
Enthalpy rise	1.0(b)	1.0(b)
Heat flux	N/A(c)	N/A(c)
Flow area	1.0(d)	1.0(d)
Active fuel length (AFL), in.(e)	143.00(e)	143.00(e)
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	173	173
Max heat flux at 100% power, 10^3 Btu/h-ft^2	514	514
CHF correlation	BHTP	BHTP
CHF correlation DNB limit	1.132	1.132
Overpower Minimum DNBR		
4-P	1.915	1.915
3-P	2.097	2.097
2-P	2.495	2.495
Minimum 4-Pump DNBR		
at rated power	2.104	2.104
for limiting Condition II transient	1.7399	1.7399

(a) Used in the reference analysis. The actual bypass values are bounded for both Cycles 29 and 30.

(b) 1.015 enthalpy rise hot channel factor was used in the derivation of the SDL.

(c) The hot channel factor for heat flux is no longer applied in DNB calculations as discussed in Reference 5.

(d) 1.025 radial hot channel factor to account for flow area variability was used in the derivation of the SDL.

(e) Cold nominal stack height used in the reference analysis. The AFL for batches 26A5, 30, 31, and 32 is 143.00 inches. The steady-state analyses used an AFL of 142.75 inches, which is conservative compared to a stack height of 143.00 inches. The transient analyses were performed conservatively relative to a stack height of 143.00 inches.

Table 3A-6

COMPARISON OF KEY PARAMETERS FOR ACCIDENT ANALYSIS

Parameter	Safety Analysis Value	Cycle 30 Value
BOC [a] Doppler coefficient, x 10^{-5}, Δk/k/°F	-1.17[e]	-1.64
EOC [b] Doppler coefficient, x 10^{-5}, Δk/k/°F	-1.30	-1.80[f]
BOC moderator coefficient (HFP), x 10^{-4}, Δk/k/°F	0.0	-0.407
EOC moderator coefficient (HFP), x 10^{-4}, Δk/k/°F	-4.0	-3.26
BOC moderator coefficient (HZP), x 10^{-4}, Δk/k/°F	+0.9	+0.275
SLB reactivity deficit, %Δk/k	0.94505[c]	0.73[d]
Inverse boron worth (HFP), ppm/%Δk/k	158	150
Maximum ejected rod worth (HFP), %Δk/k	0.65	< 0.65[g]
Maximum dropped rod worth (HFP), %Δk/k	0.28	≤ 0.20[h]
Initial boron concentration (HFP), ppm	2270	1865

[a] BOC denotes beginning of cycle.

[b] EOC denotes end of cycle.

[c] Determined from the steam line break analysis.

[d] Calculated at conservative steam line break analysis conditions.

[e] A BOC Doppler coefficient of -1.30 x 10^{-5} Δk/k/°F was used in the most recent startup event analysis.

[f] The EOC Doppler coefficient of -1.80 X 10^{-5} Δk/k/°F was calculated for an all-rods-out (ARO) condition. An EOC Doppler coefficient of -1.73 X 10^{-5} Δk/k/°F was also provided for control rod bank 7 at 60% withdrawn, which defines the maximum EOC moderator temperature coefficient.

[g] The rod insertion limits ensure that the HFP maximum ejected rod worth with an applied 15% uncertainty remains less than 0.65 %Δk/k.

[h] The SAR maximum dropped rod worth is 0.65% Δk/k; however, the maximum dropped rod worth is limited to 0.28%Δk/k in the cycle-specific safety analysis evaluation. The cycle-specific maximum HFP dropped rod worth increased by 15% uncertainty is < 0.20% Δk/k.

Table 3A-7

ANALYSIS STATUS OF NON-LOCA SAFETY ANALYSIS

Event	Effective Cycle For Analysis Of Record	Cycle-Specific Parameters Bounded?
Startup Event	14[c]	Yes
Rod Withdrawal at Power Event	1	Yes
Moderator Dilution Event		
At Power	12	Yes
During Refueling	30	Yes
Cold Water Event	1	Yes
Loss of Coolant Flow System Response[a]		
Locked Rotor Event	1[e]	Yes
Four-Pump Coastdown Event	1[e]	Yes
Four-to-Two Pump Coastdown Event	1[e]	Yes
Dropped Rod Event	1[d]	Yes
Loss of Electric Power Events		
Loss of Load Event	1[c]	Yes
Complete Loss of AC Power Event	1	Yes
Turbine Overspeed Event	1	Yes
Fuel Handling Accident[b]	1	[b]
Steam Line Failure Event[b]	24	Yes
Steam Generator Tube Failure Event[b]	1	Yes
Rod Ejection Event[b]	1	Yes
Loss-of-Coolant Event[b]	Reload Report Section 3A.7.3	Reload Report Section 3A.7.3
Maximum Hypothetical Accident[b]	Reload Report Section 3A.7.1	Reload Report Section 3A.7.1
Waste Gas Decay Tank Rupture Event[b]	Reload Report Section 3A.7.1	Reload Report Section 3A.7.1

[a] The plant system response (including power, RCS flow, core inlet temperature, and system pressure) has been shown to be bounding for Cycle 30. The DNB analysis using the Statistical Core Design methodology is discussed in Reload Report Section 3A.6.

[b] For a discussion of the dose consequences of these events, refer to Reload Report Section 3A.7.1

[c] This event was reanalyzed for Cycle 17.

[d] This event was reevaluated for Cycle 19.

[e] This event was reanalyzed for Cycle 26.

Table 3A-8

COMPARISON OF KEY FUEL PARAMETERS FOR MARK-B-HTP LOCA ANALYSES

Parameter	Batch-Specific Value	Generic Mark-B-HTP Analysis Value	Bounded?
Fuel Rod/Assembly Type	All Cycle 30 Batches: Mark-B-HTP FA Zone loaded	Mark-B-HTP FA	Yes
Cold Fuel Dimensions			
Fuel OD, in.	0.3735	0.3735	Yes
Clad ID, in.	0.380	0.380	Yes
Clad OD, in.	0.430	0.430	Yes
Cold Fuel Rod Plenum Volume (in^3)	All Cycle 30 Batches: PROPRIETARY	PROPRIETARY	Yes
Fuel Rod Pre-Pressure (psig)	All Cycle 30 Batches: PROPRIETARY	PROPRIETARY	Yes
Fuel Enrichment, weight percent ^{235}U	All Cycle 30 Batches: > 3.0 – < 5.1 wt%	3.0 – 5.1 wt%	Yes
Generic Power History Envelope Bounded?	Yes	Enveloping Value	Yes

Table 3A-9A

CYCLE 30 ALLOWABLE LHR LIMITS (kW/ft) FOR MARK-B-HTP FUEL(1,2,3)
Nuclear Source

Allowable LOCA LHR Limits						
Core Elevation, ft		LHR Limit for 0 MWd/mtU, kW/ft		LHR Limit for 28,000 MWd/mtU, kW/ft		LHR Limit for 62,000 MWd/mtU, kW/ft
0.000		16.6		16.6		12.3
2.506		17.5		17.5		13.0
4.264		17.5		17.5		13.0
6.021		17.3		17.3		13.0
7.779		17.5		17.5		13.0
9.536		17.3		17.3		13.0
11.000		15.3		15.3		12.5
12.000		14.5		14.5		12.3

Allowable LOCA LHR Limits – Chrome-Coated Fuel Rods Only						
Core Elevation, ft		LHR Limit for 0 MWd/mtU, kW/ft		LHR Limit for 28,000 MWd/mtU, kW/ft		LHR Limit for 62,000 MWd/mtU, kW/ft
0.000		15.7		15.7		11.6
2.506		16.6		16.6		12.3
4.264		16.6		16.6		12.3
6.021		16.4		16.4		12.3
7.779		16.6		16.6		12.3
9.536		15.1		15.1		12.3
11.000		12.5		12.5		11.5
12.000		11.8		11.8		10.9

1. Limits may be linearly interpolated as a function of fuel rod burnup and as a function of core elevation.
2. The LOCA LHR limits above are reported on a nuclear source basis consistent with the values used as input to the core power distribution analysis. The nuclear source basis accounts for all useable energy produced by the fuel rod, which is related to the thermal energy deposited into the rod by the energy deposition factor (nuclear source = EDF * thermal source). These LHR values are adjusted as necessary for input to the plant online computer.
3. These LHR values are adjusted if necessary during the maneuvering analyses based on the cycle-specific axial peaking to ensure that the LHR limits are in compliance with the axial peaking restriction from the BWNT LOCA EM. Any reductions that may be applied are discussed in Section 3A.8, and are reflected in Table 3A-9B.

Table 3A-9B

CYCLE 30 ALLOWABLE LHR LIMITS (kW/ft) FOR MARK-B-HTP FUEL

(Referred to by Technical Specifications 3.1.8 and 3.2.5)

Allowable LOCA LHR Limits						
Core Elevation, ft		**LHR Limit for 0 MWd/mtU, kW/ft**		**LHR Limit for 28,000 MWd/mtU, kW/ft**		**LHR Limit for 62,000 MWd/mtU, kW/ft**
0.000		16.6		16.6		12.3
2.506		17.5		17.5		13.0
4.264		17.5		17.5		13.0
6.021		17.3		17.3		13.0
7.779		17.5		17.5		13.0
9.536		17.3		17.3		13.0
11.000		15.3		15.3		12.5
12.000		14.5		14.5		12.3

Allowable LOCA LHR Limits – Chrome-Coated Fuel Rods Only						
Core Elevation, ft		**LHR Limit for 0 MWd/mtU, kW/ft**		**LHR Limit for 28,000 MWd/mtU, kW/ft**		**LHR Limit for 62,000 MWd/mtU, kW/ft**
0.000		15.7		15.7		11.6
2.506		16.6		16.6		12.3
4.264		16.6		16.6		12.3
6.021		16.4		16.4		12.3
7.779		16.6		16.6		12.3
9.536		15.1		15.1		12.3
11.000		12.5		12.5		11.5
12.000		11.8		11.8		10.9

Note: The LOCA LHR limits may be linearly interpolated as a function of burnup between 0 MWd/mtU and 28,000 MWd/mtU and between 28,000 MWd/mtU and 62,000 MWd/mtU and as a function of core elevation.

Table 3A-10

AXIAL POWER SHAPING ROD INSERTION LIMITS AND ALARM SETPOINTS

(Limits are referred to by Technical Specification 3.2.2)

Up to 455 ± 10 EFPD, the APSRs may be positioned as necessary for transient imbalance control; however, the APSRs shall be fully withdrawn by 465 EFPD. After the APSRs withdrawal at 455 ± 10 EFPD, the APSRs shall not be reinserted, except during the end-of-cycle shutdown when the reactor power is equal to, or less than, 30% FP.

Table 3A-11

ANO-1 CYCLE 30 QUADRANT POWER TILT LIMITS AND ALARM SETPOINTS

(Referred to by Technical Specifications 3.2.4)

From 0 EFPD to EOC

Measurement System	Steady State Value (%)		Maximum Value (%)
	≤ 60 % FP	> 60 % FP	
Full Incore Detector System Setpoint	6.83	4.47	25.00
Minimum In-core Detector System Setpoint	2.78*	1.90*	25.00
Ex-core Power Range NI Channel Setpoint	4.05	1.96	25.00
Measurement System-Independent Limit	7.50	4.92	25.00

* Based on the condition that no individual long emitter detector affecting the minimum incore tilt calculation exceeds 73% sensitivity depletion. The setpoint must be reduced to 1.50% (power levels > 60% FP) and to 2.19% (power levels ≤ 60% FP) at the earliest time-in-life that this condition is no longer valid.

Table 3A-12

DNB POWER PEAKING FACTORS

(Limit is referred to by Technical Specification 3.1.8 and 3.2.5)

The following total power peaking factors define the Maximum Allowable Peaking (MAP) limits to protect the initial conditions assumed in the DNB Loss of Flow transient analysis. The total power peaking factors for both the Mark-B9 and the Mark-B-HTP fuels are provided. The total power peaking factors for IC-DNB 4-pump and 3-pump are identical; hence one set of IC-DNB values are provided for both 4-pump and 3-pump operation.

IC-DNB Total Power Peaking Factors

Mark-B-HTP								
Axial Peak	**x/L**	**IC MAP Limits**	**Axial Peak**	**x/L**	**IC MAP Limits**	**Axial Peak**	**x/L**	**IC MAP Limits**
	0.01	2.04426		0.01	2.94000		0.01	3.20469
	0.14	2.04515		0.14	2.94000		0.14	3.20469
	0.20	2.04535		0.20	2.93545		0.20	3.15423
	0.30	2.04551		0.30	2.84715		0.30	3.08083
	0.40	2.04470		0.40	2.76077		0.40	2.98064
1.1	0.50	2.04437	1.4	0.50	2.66671	1.7	0.50	2.89369
	0.60	2.04415		0.60	2.55808		0.60	2.78037
	0.70	2.04400		0.70	2.46508		0.70	2.68552
	0.80	2.04329		0.80	2.34973		0.80	2.56207
	0.89	2.00109		0.89	2.27714		0.89	2.49021
	0.99	1.90427		0.99	2.18525		0.99	2.39515
	0.01	2.33088		0.01	3.08066		0.01	3.24949
	0.14	2.33287		0.14	3.08066		0.14	3.24949
	0.20	2.33339		0.20	3.03513		0.20	3.20303
	0.30	2.33352		0.30	2.93856		0.30	3.13047
	0.40	2.33338		0.40	2.84115		0.40	3.04037
1.2	0.50	2.33285	1.5	0.50	2.75216	1.8	0.50	2.95027
	0.60	2.33232		0.60	2.63946		0.60	2.84225
	0.70	2.26721		0.70	2.54429		0.70	2.74696
	0.80	2.16931		0.80	2.42655		0.80	2.62489
	0.89	2.10460		0.89	2.35382		0.89	2.55373
	0.99	2.00767		0.99	2.26040		0.99	2.45882
	0.01	2.64464		0.01	3.14861		0.01	3.28611
	0.14	2.64863		0.14	3.14861		0.14	3.28611
	0.20	2.64909		0.20	3.09918		0.20	3.24461
	0.30	2.64997		0.30	3.01573		0.30	3.17163
	0.40	2.64949		0.40	2.91490		0.40	3.08589
1.3	0.50	2.56272	1.6	0.50	2.82718	1.9	0.50	3.00025
	0.60	2.46600		0.60	2.71210		0.60	2.89826
	0.70	2.37484		0.70	2.61653		0.70	2.80288
	0.80	2.26452		0.80	2.49634		0.80	2.68386
	0.89	2.19471		0.89	2.42370		0.89	2.61261
	0.99	2.09988		0.99	2.32955		0.99	2.51792

Table 3A-12 (continued)

Notes:

1. The values in Table 3A-12 are not error corrected.
2. The values in Table 3A-12 were generated using SCD methods which incorporate a 3.8% radial peak uncertainty in the DNBR design limit. Therefore, the above IC MAP limits can be compared to predicted peaks without the addition of up to 3.8% in radial peak calculation uncertainty. These limits, however, do not incorporate grid bias uncertainty.
3. The present T-H methodology allows for an increase in the design radial-local peak for power levels under 100% full power. The equations defining the multipliers are as follows:

	$P/P_m = 1.00$	$P/P_m < 1.00$
MAP Multiplier	1.0	$1 + 0.3(1 - P/P_m)$

Where P = core power fraction, and
P_m = 1.00 for 4-pump operation, or
= 0.75 for 3-pump operation

Table 3A-13

SHUTDOWN MARGIN (SDM) REQUIREMENTS

(Limit is referred to by Technical Specification 3.1.4, 3.1.5, 3.3.9)

Applicability	Required Shutdown Margin	Technical Specification Reference
MODE 1	≥ 1 %Δk/k	3.1.4, 3.1.5
MODE 2	≥ 1 %Δk/k	3.1.4, 3.1.5, 3.3.9

The required Shutdown Margin capability of 1 %Δk/k in Mode 1 and Mode 2 is preserved by the Regulating Rod Insertion Limits specified in the COLR as required by Technical Specification 3.2.1.

Table 3A-14

RCS LOOPS – MODE 1 AND MODE 2

(Limit is referred to by Technical Specification 3.4.4)

	Nominal Operating Power Level (% Power)
Four Pump Operation	100
Three Pump Operation	75
Two Pump Operation*	49

* Technical Specification 3.4.4 does not allow indefinite operation in Modes 1 and 2 with only two pumps operating.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						30A L04	30A C07	30A H11	30A C09	30A L12					
B				30A E07	30C L02	31C N13	32A F	31B O08	32A F	31C N03	30C L14	30A E09			
C			30B B08	32C F	31A P07	32A F	31A G10	32B F	31A G06	32A F	31A P09	32C F	30B H14		
D		30A G05	32C F	31A N07	32A F	31A E10	32A F	30A H09	32A F	31A E06	32A F	31A G04	32C F	30A G11	
E		30C B10	31A G14	32A F	30A G09	32A F	31A M04	31A N09	31A M12	32A F	30A K09	32A F	31A G02	30C B06	
F	30A D10	31C O12	32A F	31A L05	32A F	30A O05	32A F	30A M03	32A F	30A E03	32A F	31A L11	32A F	31C O04	30A D06
G	30A G03	32A F	31A L07	32A F	31A D11	32A F	31A L13	31A O10	31A O06	32A F	31A D05	32A F	31A L09	32A F	30A G13
H	30A E08	31B H13	32B F	30A G08	31A G12	30A O11	31A F13	26A5 F12 25	31A L03	30A C05	31A K04	30A K08	32B F	31B H03	30A M08
K	30A K03	32A F	31A F07	32A F	31A N11	32A F	31A C10	31A C06	31A F03	32A F	31A N05	32A F	31A F09	32A F	30A K13
L	30A N10	31C C12	32A F	31A F05	32A F	30A M13	32A F	30A E13	32A F	30A C11	32A F	31A F11	32A F	31C C04	30A N06
M		30C P10	31A K14	32A F	30A G07	32A F	31A E04	31A D07	31A E12	32A F	30A K07	32A F	31A K02	30C P06	
N		30A K05	32C F	31A K12	32A F	31A M10	32A F	30A H07	32A F	31A M06	32A F	31A D09	32C F	30A K11	
O			30B H02	32C F	31A B07	32A F	31A K10	32B F	31A K06	32A F	31A B09	32C F	30B P08		
P				30A M07	30C F02	31C D13	32A F	31B C08	32A F	31C D03	30C F14	30A M09			
R						30A F04	30A O07	30A H05	30A O09	30A F12					

Key

xxxx	Batch ID
yyy	Previous Location
zz	Reinsert Cycle Number

● SS Rod

● c-c Fuel Rod

North

FAID

SAR FIGURE NO. 3A-1

AMENDMENT 30

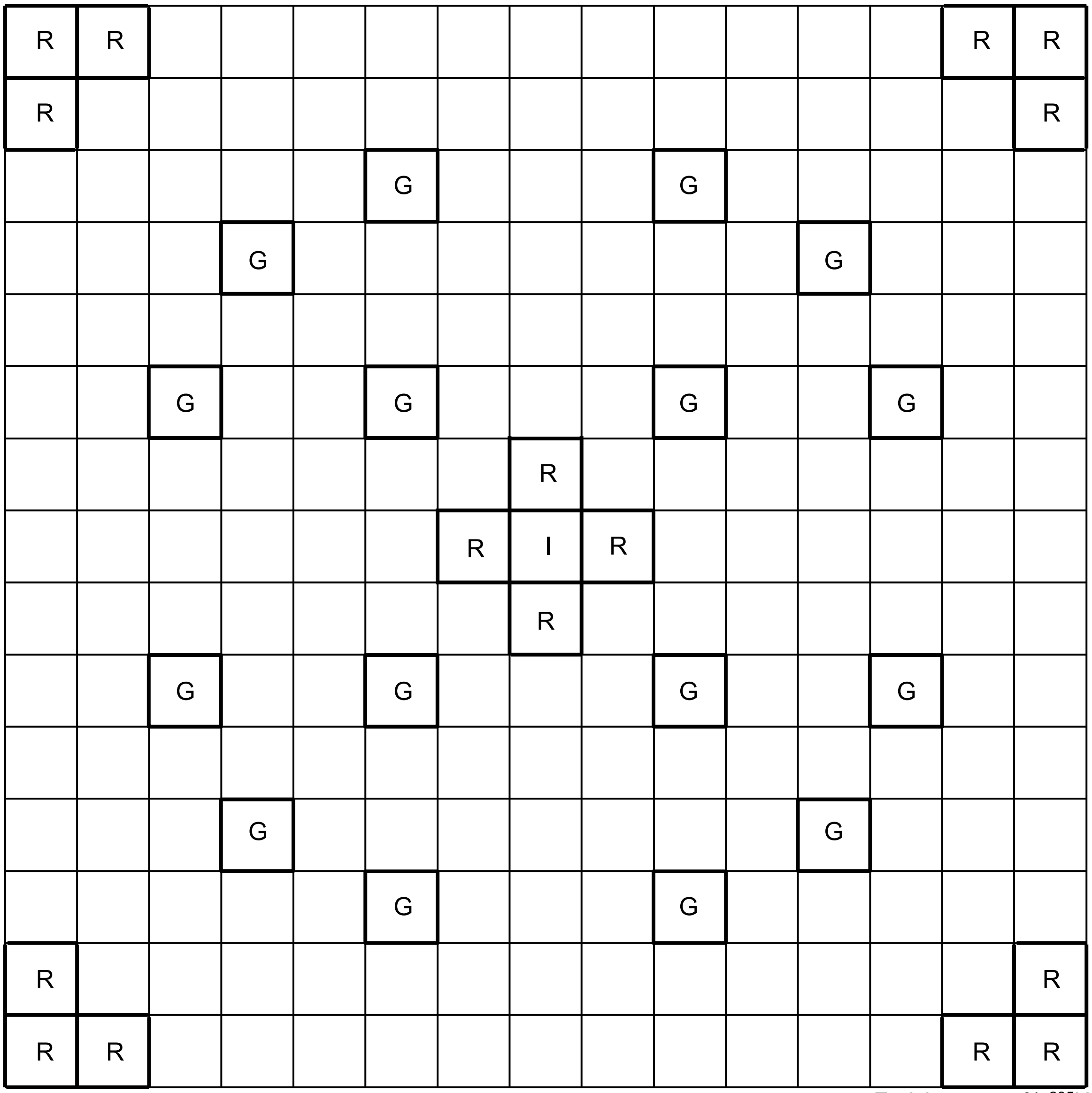
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

CORE LOADING FOR ANO-1 CYCLE 30

BASED ON DRAWING NO	SHEET	REV.



☐ = Fuel Rod at Nominal Enrichment (192 rods)

R = Fuel Rod at Reduced Enrichment (16 rods)

I = Instrument Cell

G = Guide Tube Cell

	Enrichment, wt% ^{235}U	
Batch	Nominal	Reduced
26A5	3.78	3.48
30A	3.51	3.21
30B1-30B4*	3.32	3.02
30C1-30C8*	3.32	3.02
31A	3.80	3.50
31B1-31B4*	3.67	3.37/3.07
31C1-31C8*	3.67	3.37/3.07
32A	3.72	3.42
32B1-32B4*	3.50	3.20
32C1-32C8*	3.50	3.20

* Batches 30B, 31B, and 32B fuel assemblies have four SSRs each, and batches 30C and 32C fuel assemblies have three SSRs each. Additionally, the batch 31B fuel assemblies have two zone loading enrichments. The four zone loaded pins around the center instrument cell are loaded at 3.37 wt% while the remaining zone loaded pins are loaded to 3.07 wt%.

The batch 31C fuel assemblies have one SSR each and two c-c rods. The batch 31C fuel assemblies have the same two zone loading enrichments and zone loading scheme as the batch 31B fuel assemblies. All SSRs and c-c rods replace reduced enrichment rods on the periphery of each assembly; see Figure 3A-1 for their approximate locations.

SAR FIGURE NO. 3A-2

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ANO-1 CYCLE 30 ZONE LOADING PATTERN	BASED ON DRAWING NO	SHEET / REV.

	8	9	10	11	12	13	14	15
H	3.76 38,755	3.78 19,563	3.49 30,267	3.78 20,155	3.49 33.552	3.48 0	3.64 19,739	3.49 36,102
K	3.78 19,563	3.78 19,573	3.70 0	3.78 20,107	3.70 0	3.78 20,539	3.70 0	3.49 36,926
L	3.49 30,267	3.70 0	3.49 30,262	3.70 0	3.78 20,754	3.70 0	3.63 13,868	3.49 37,134
M	3.78 20,155	3.78 20,103	3.70 0	3.49 34,801	3.70 0	3.78 16,928	3.30 26,021	
N	3.49 33,552	3.70 0	3.78 20,740	3.70 0	3.78 20,151	3.48 0	3.49 37,168	
O	3.48 0	3.78 20,656	3.70 0	3.78 16,942	3.48 0	3.30 31,823		
P	3.64 19,739	3.70 0	3.63 13,888	3.30 25,984	3.49 37,231			
R	3.49 36,102	3.49 36,973	3.49 37,152					

X.XX	Initial Enrichment (Fuel Assembly Average), wt% ^{235}U
XX,XXX	BOC Burnup, MWd/mtU

SAR FIGURE NO. 3A-3

AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

ANO-1 CYCLE 30 ENRICHMENT AND BOC BURNUP DISTRIBUTION

BASED ON DRAWING NO	SHEET	REV.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						1		6		1					
C					3		5		5		3				
D				7		8		7		8		7			
E			3		5		4		4		5		3		
F		1		8		6		2		6		8		1	
G			5		4		2		2		4		5		
H		6		7		2				2		7		6	
K			5		4		2		2		4		5		
L		1		8		6		2		6		8		1	
M			3		5		4		4		5		3		
N				7		8		7		8		7			
O					3		5		5		3				
P						1		6		1					
R															

X Group Number

Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Control
6	8	Control
7	8	Control
8	8	APSRs
TOTAL	68	

SAR FIGURE NO. 3A-4

AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

CONTROL ROD AND APSR LOCATIONS AND GROUP DESIGNATIONS FOR ANO-1 CYCLE 30

BASED ON DRAWING NO	SHEET	REV.

	8	9	10	11	12	13	14	15
H						0.80		
K			0.80		1.70		None	
L		0.80		1.40		1.40		
M			1.40		0.80			
N		1.70		0.80		None		
O	0.80		1.40		None			
P		None						
R								

X.XX — LBP Concentration (wt% B_4C in Al_2O_3)

	No. of BPRAs	Concentration wt% B_4C
	20	0.80
	16	1.40
	8	1.70
TOTAL	44	

SAR FIGURE NO. 3A-5

AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ANO-1 CYCLE 30 LBP CONCENTRATION AND DISTRIBUTION

BASED ON DRAWING NO	SHEET	REV.

	8	9	10	11	12	13	14	15
H	0.838	1.121	1.031	1.200	7 1.003	1.342	0.983	0.332
K	1.121	1.226	1.415	1.297	1.365	1.272	1.202	0.327
L	1.031	1.422	1.112	1.394	8 1.278	1.283	0.839	0.230
M	1.200	1.299	1.383	1.045	1.400	1.080	0.450	
N	7 1.003	1.367	8 1.273	1.396	7 1.160	0.977	0.261	
O	1.342	1.270	1.281	1.079	0.978	0.360		
P	0.983	1.201	0.837	0.450	0.261			
R	0.332	0.327	0.230					

X	Inserted Rod Group Number
X.XXX	Relative Power Density

SAR FIGURE NO. 3A-6

AMENDMENT 30

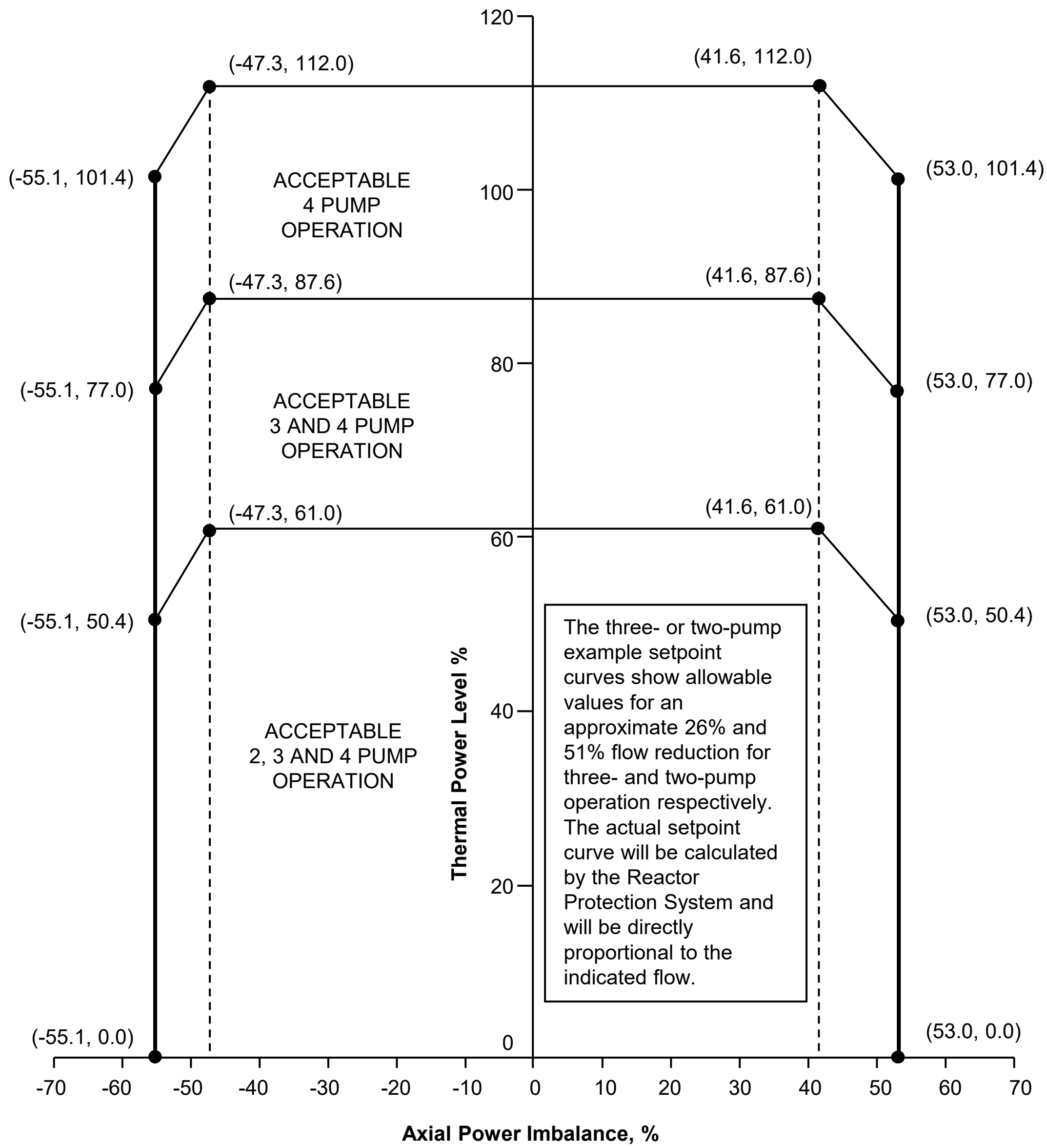
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ANO-1 CYCLE 30 BOC (4 EFPD) TWO-DIMENSIONAL RELATIVE POWER DISTRIBUTION – FP, EQUILIBRIUM XENON, GROUP 7 @ 90% WD, GROUP 8 AT 30% WD

BASED ON DRAWING NO	SHEET	REV.

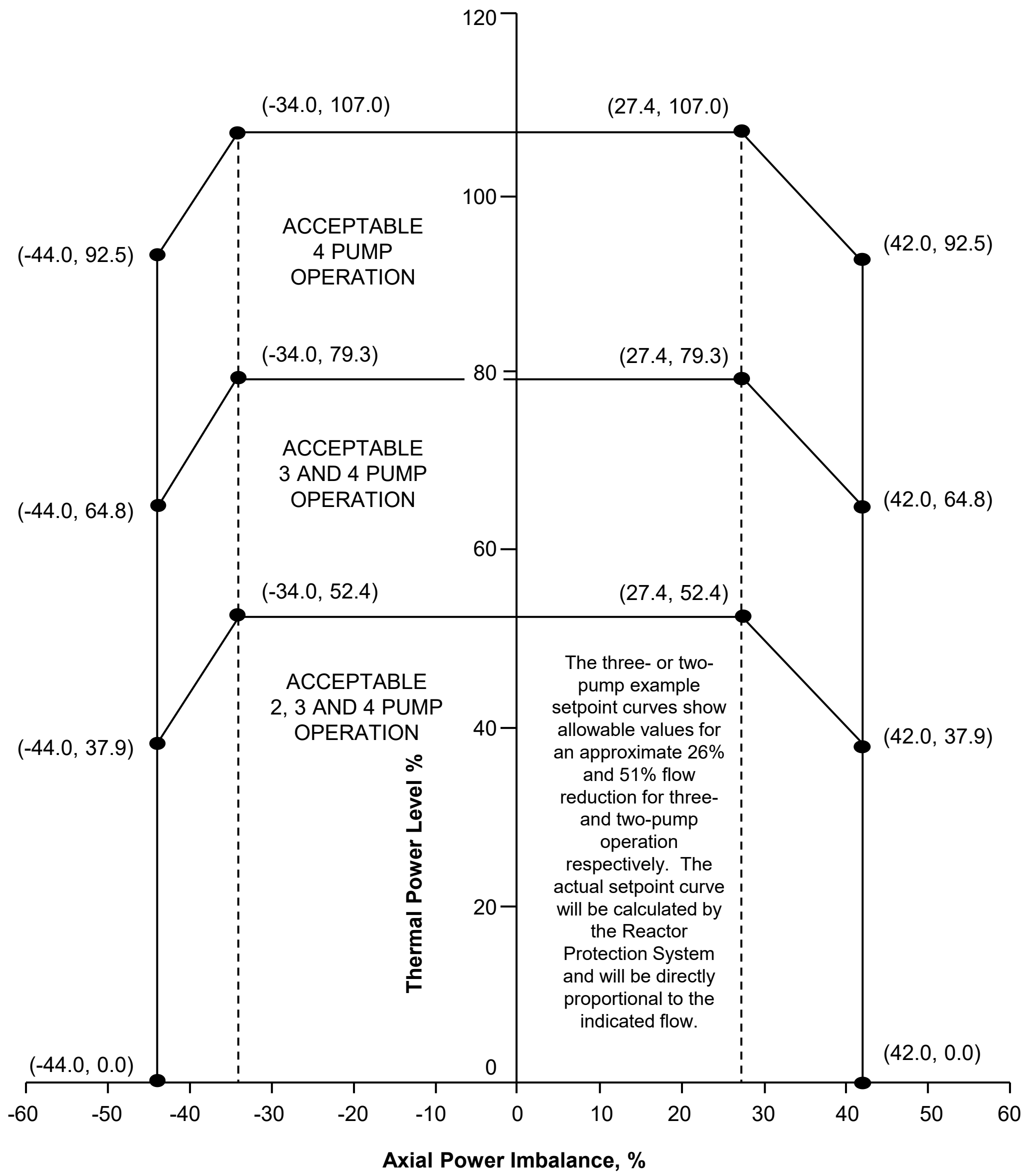


SAR FIGURE NO. 3A-7

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

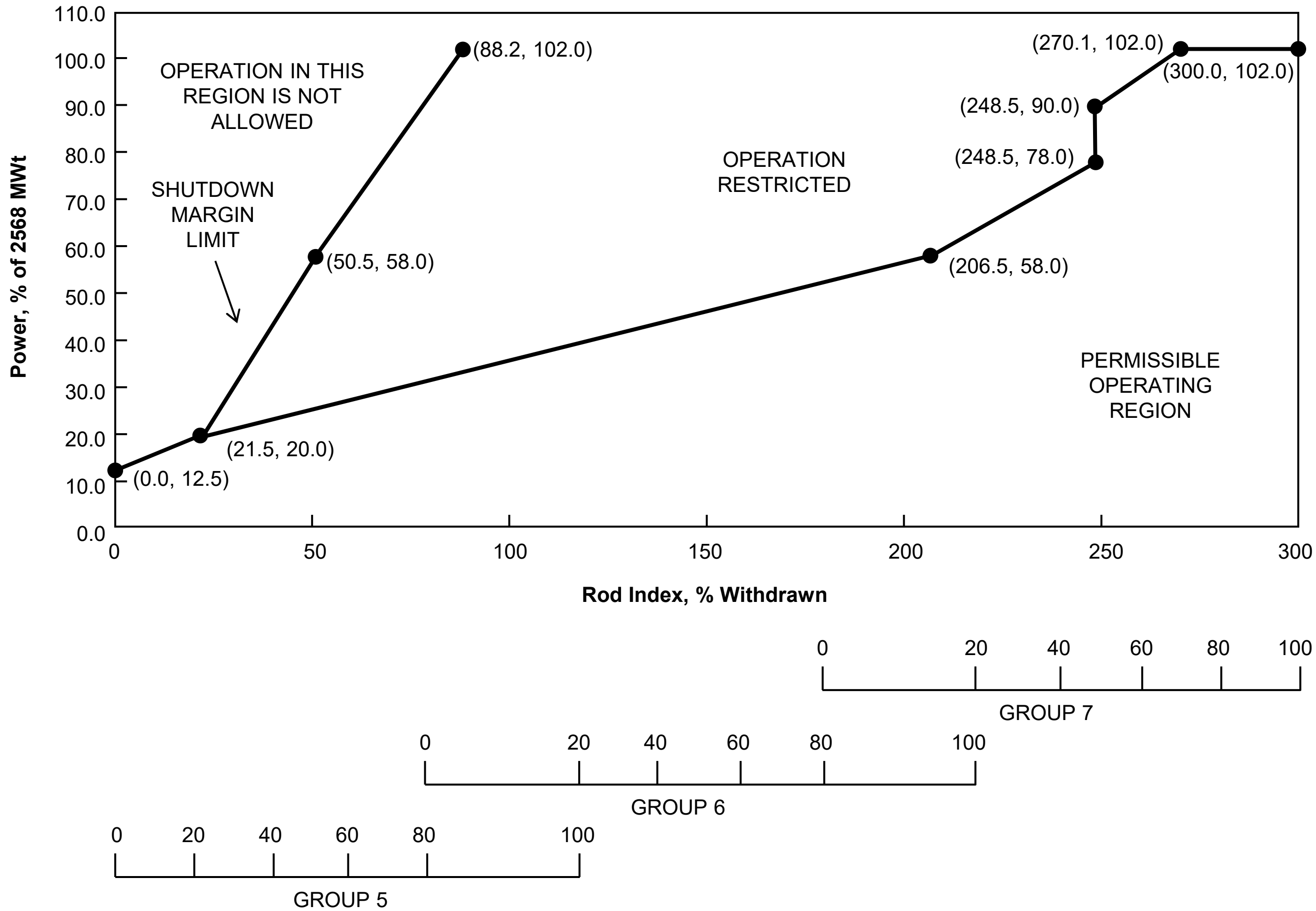
AXIAL POWER IMBALANCE PROTECTIVE LIMITS (Referred to by Tech Spec Bases 2.1.1)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3A-8

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PROTECTION SYSTEM MAXIMUM ALLOWABLE SETPOINTS FOR AXIAL POWER IMBALANCE (Referred to by Tech Spec 3.3.1 and Tech Spec Bases 2.1.1)	BASED ON DRAWING NO	SHEET / REV.



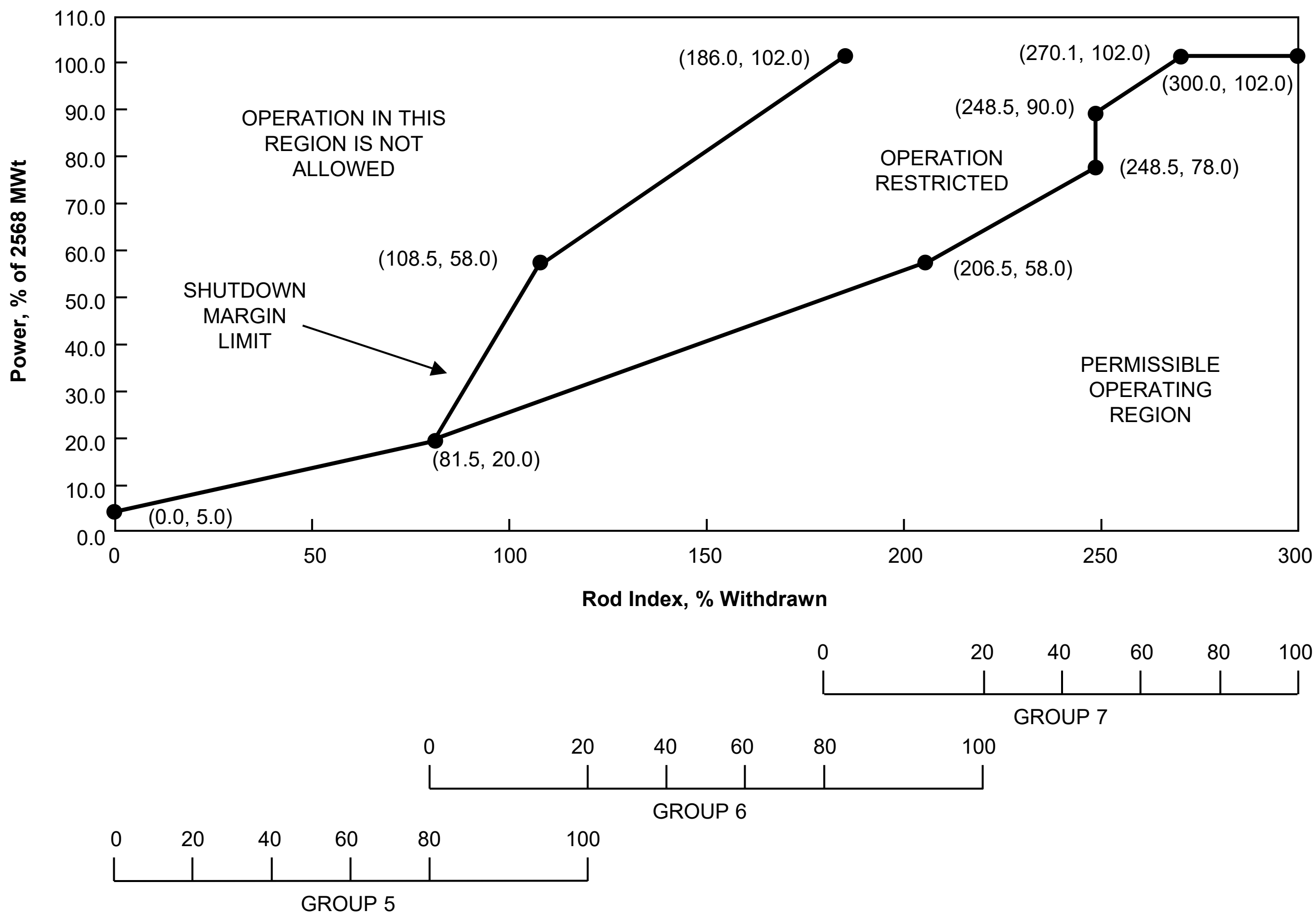
ROD POSITION ALARM SETPOINTS FOR FOUR-PUMP OPERATION FROM 0 TO 200 +10/-10 EFPD FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

SAR FIGURE NO. 3A-9

AMENDMENT 30

BASED ON DRAWING NO	SHEET	REV.



ROD POSITION ALARM SETPOINTS FOR FOUR-PUMP OPERATION
FROM 200 +10/-10 EFPD TO EOC FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)

SAR FIGURE NO. 3A-10

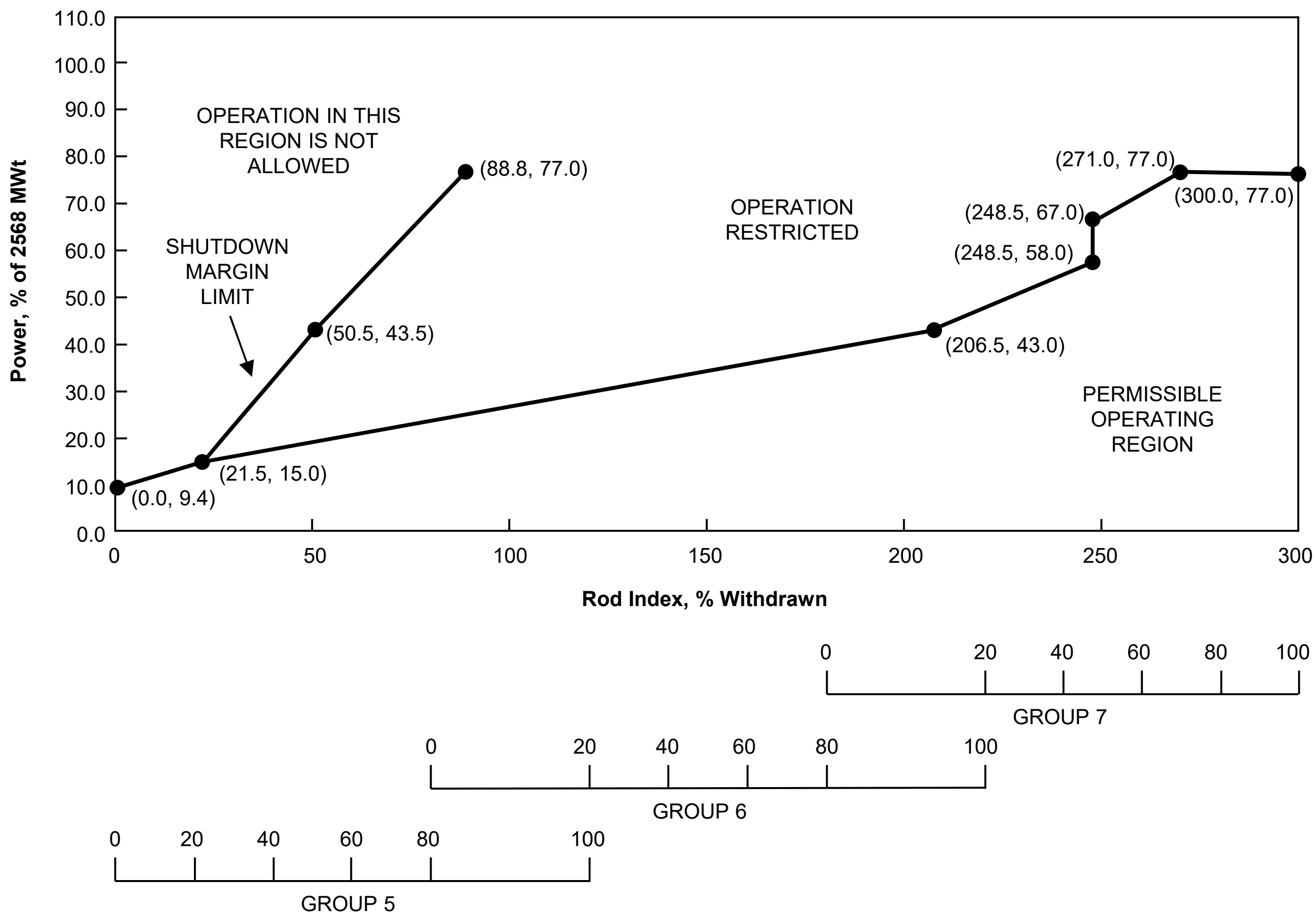
AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



ROD POSITION ALARM SETPOINTS FOR THREE-PUMP OPERATION FROM
0 TO 200 +10/-10 EFPD FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)

SAR FIGURE NO. 3A-11

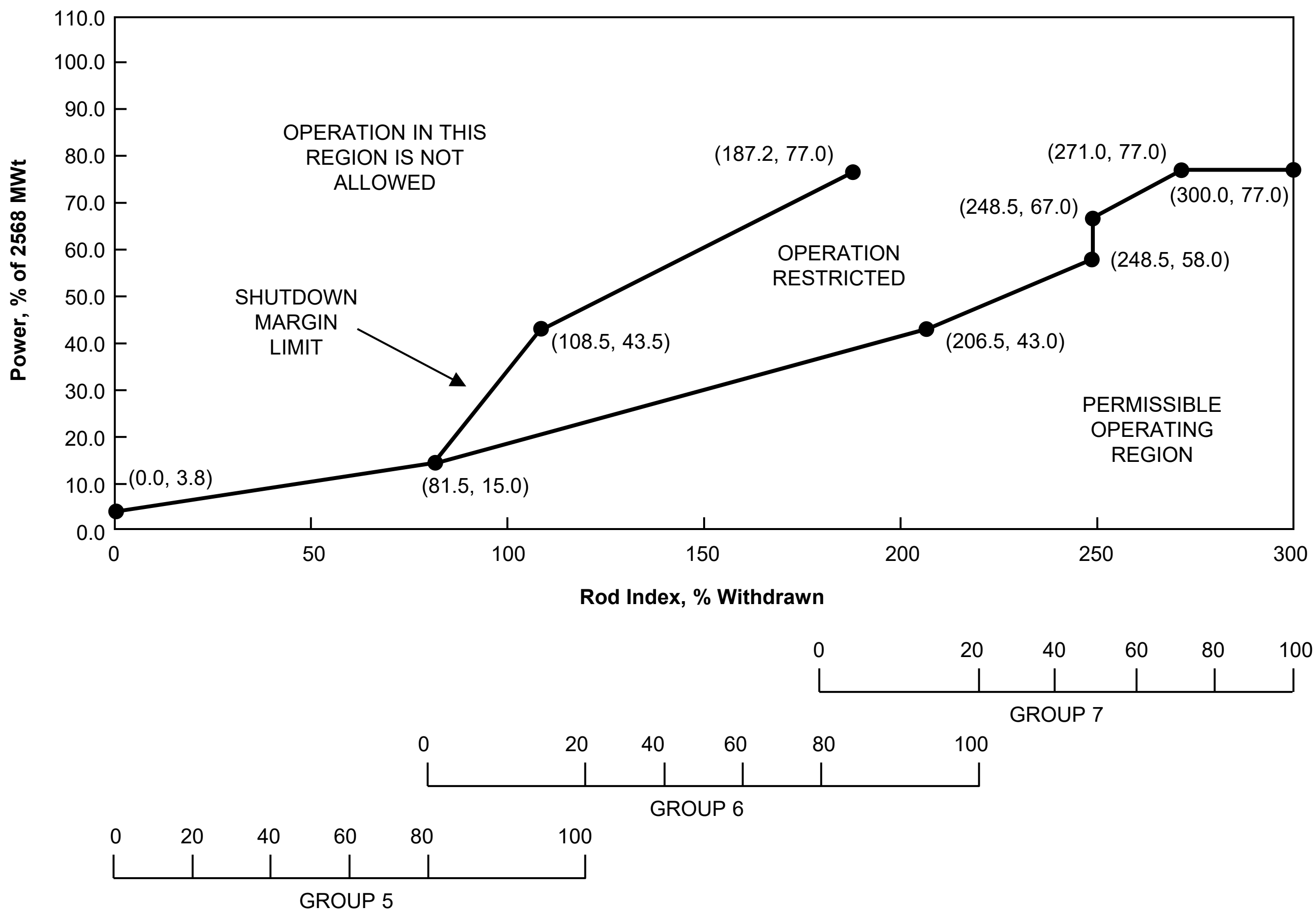
AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



ROD POSITION ALARM SETPOINTS FOR THREE-PUMP OPERATION FROM
200 +10/-10 EFPD TO EOC FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)

SAR FIGURE NO. 3A-12

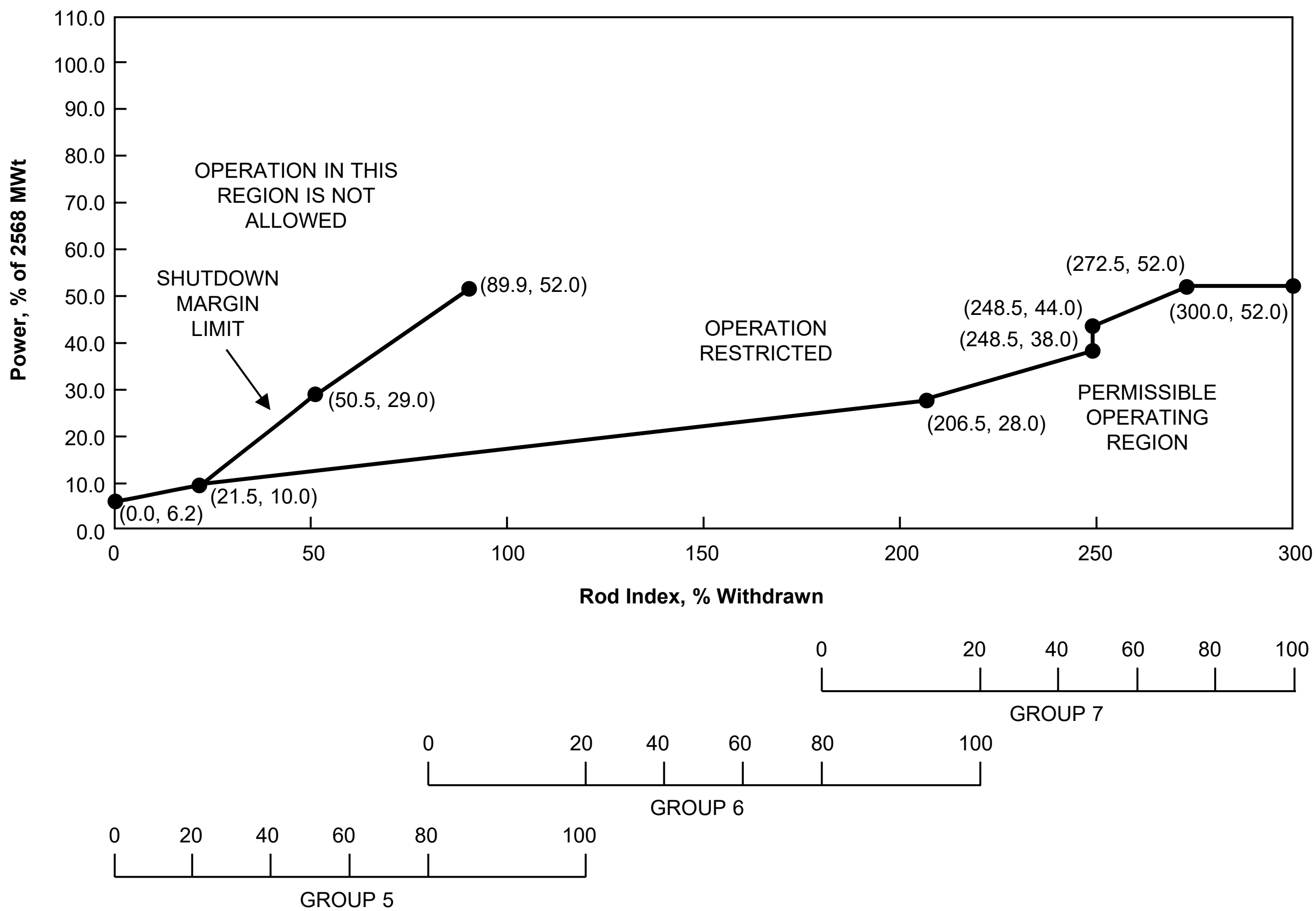
AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



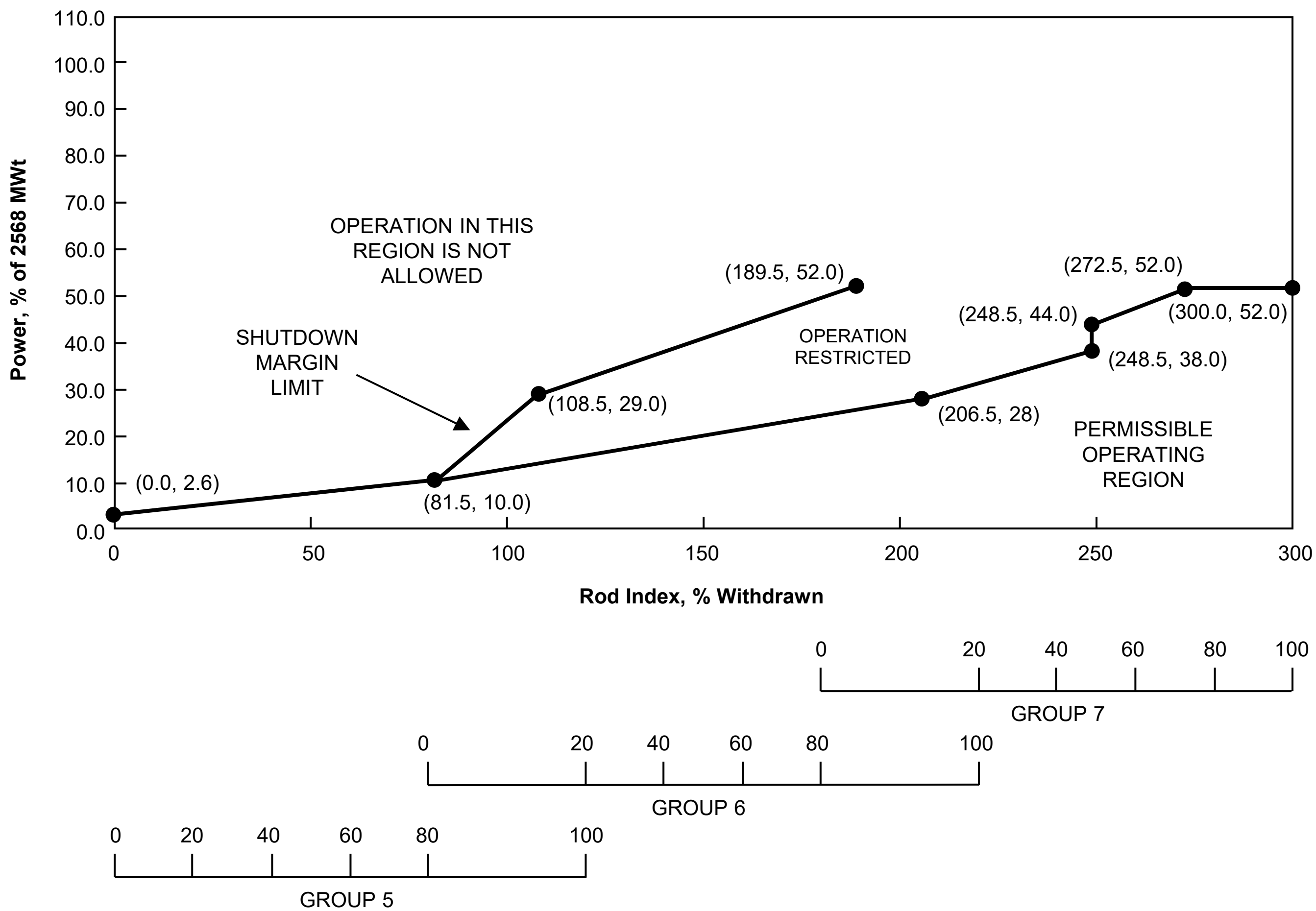
ROD POSITION ALARM SETPOINTS FOR TWO-PUMP OPERATION FROM 0 TO 200 +10/-10 EFPD FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)	SAR FIGURE NO. 3A-13
	AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



ROD POSITION ALARM SETPOINTS FOR TWO-PUMP OPERATION FROM
200 +10/-10 EFPD TO EOC FOR ANO-1 CYCLE 30 (Referred to by Tech Spec 3.2.1)

SAR FIGURE NO. 3A-14

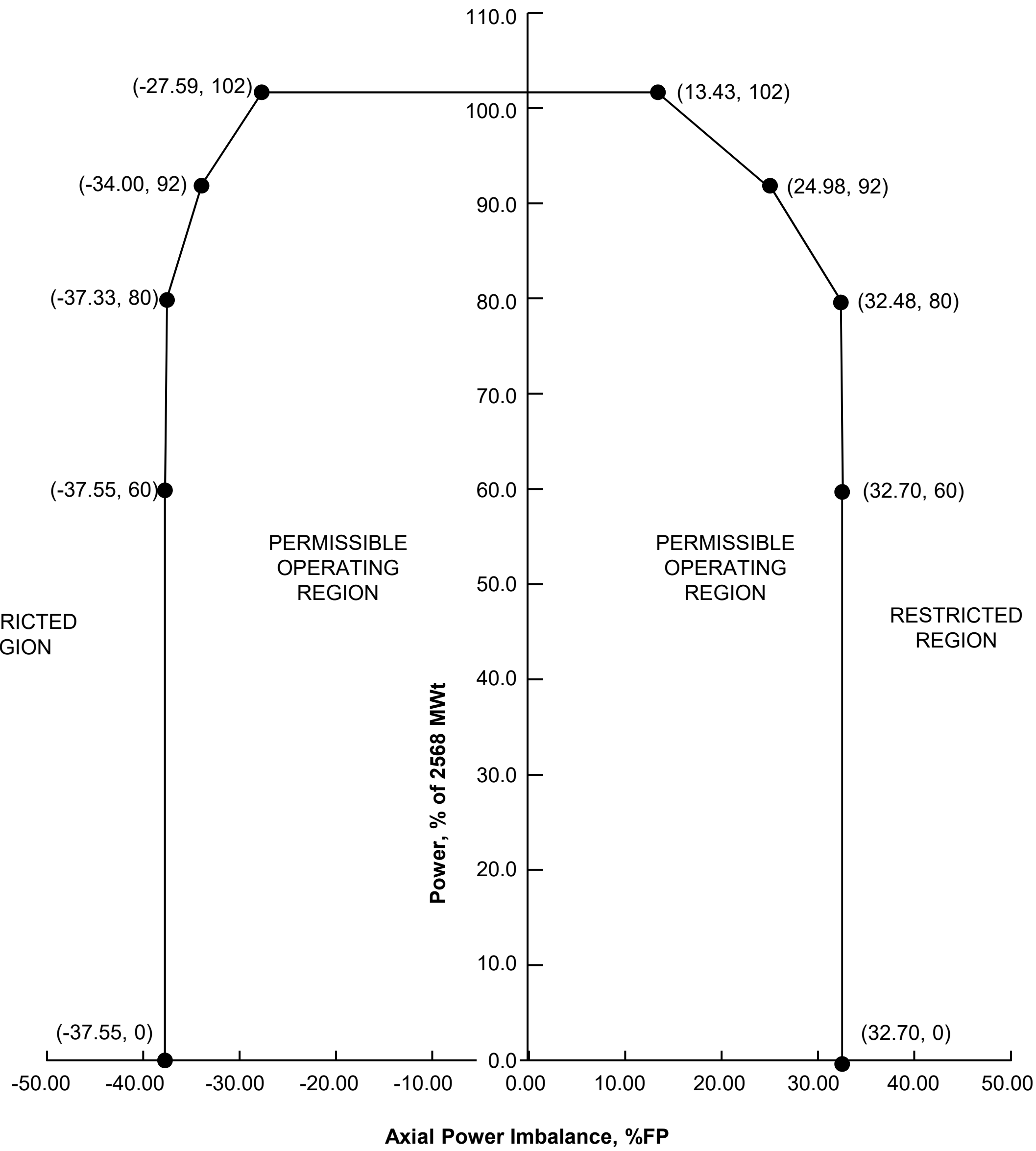
AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3A-15A

AMENDMENT 30

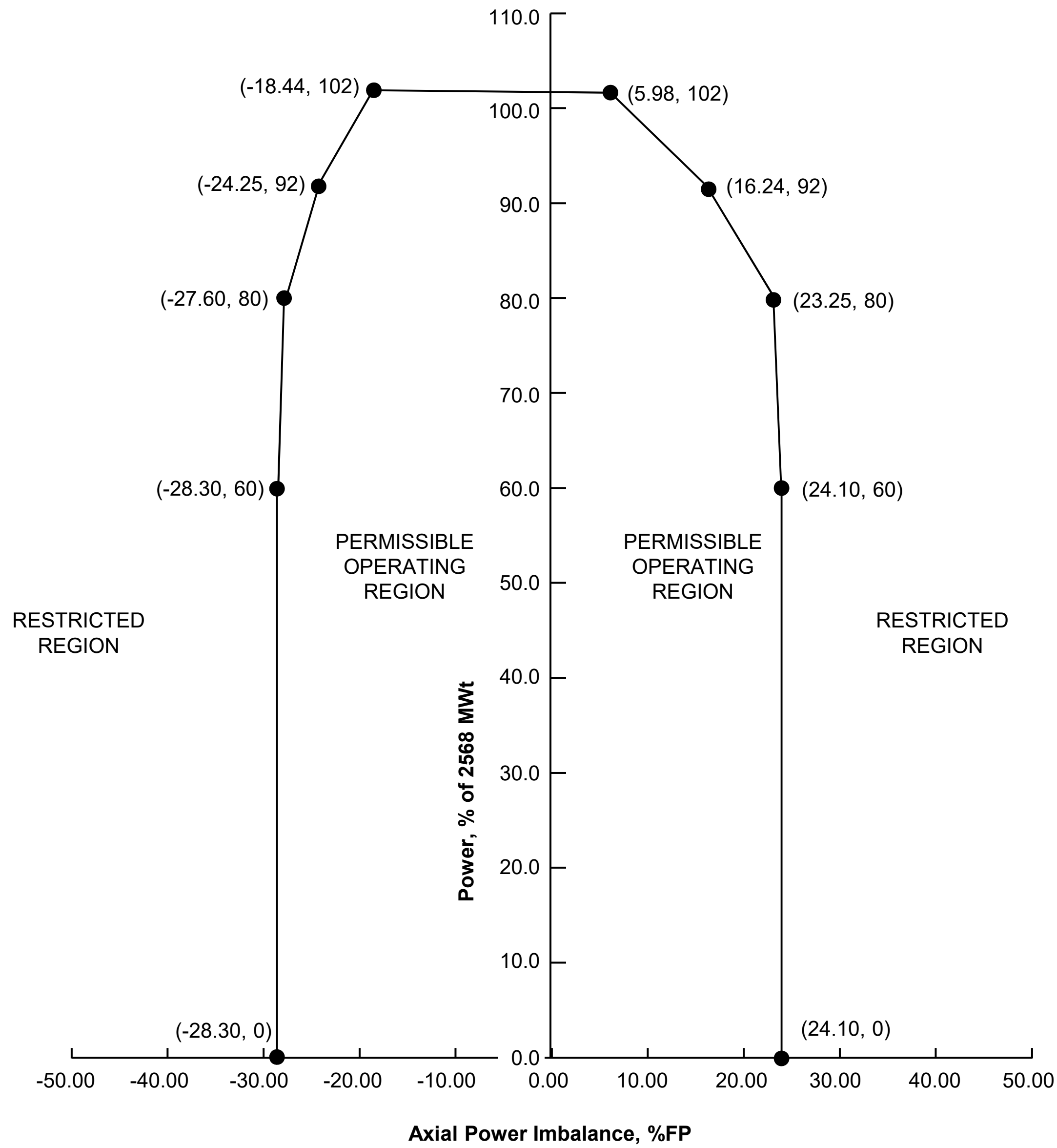
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR FOUR-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 FULL IN-CORE SYSTEM (Referred to by Tech Spec 3.2.3)

BASED ON DRAWING NO	SHEET	REV.

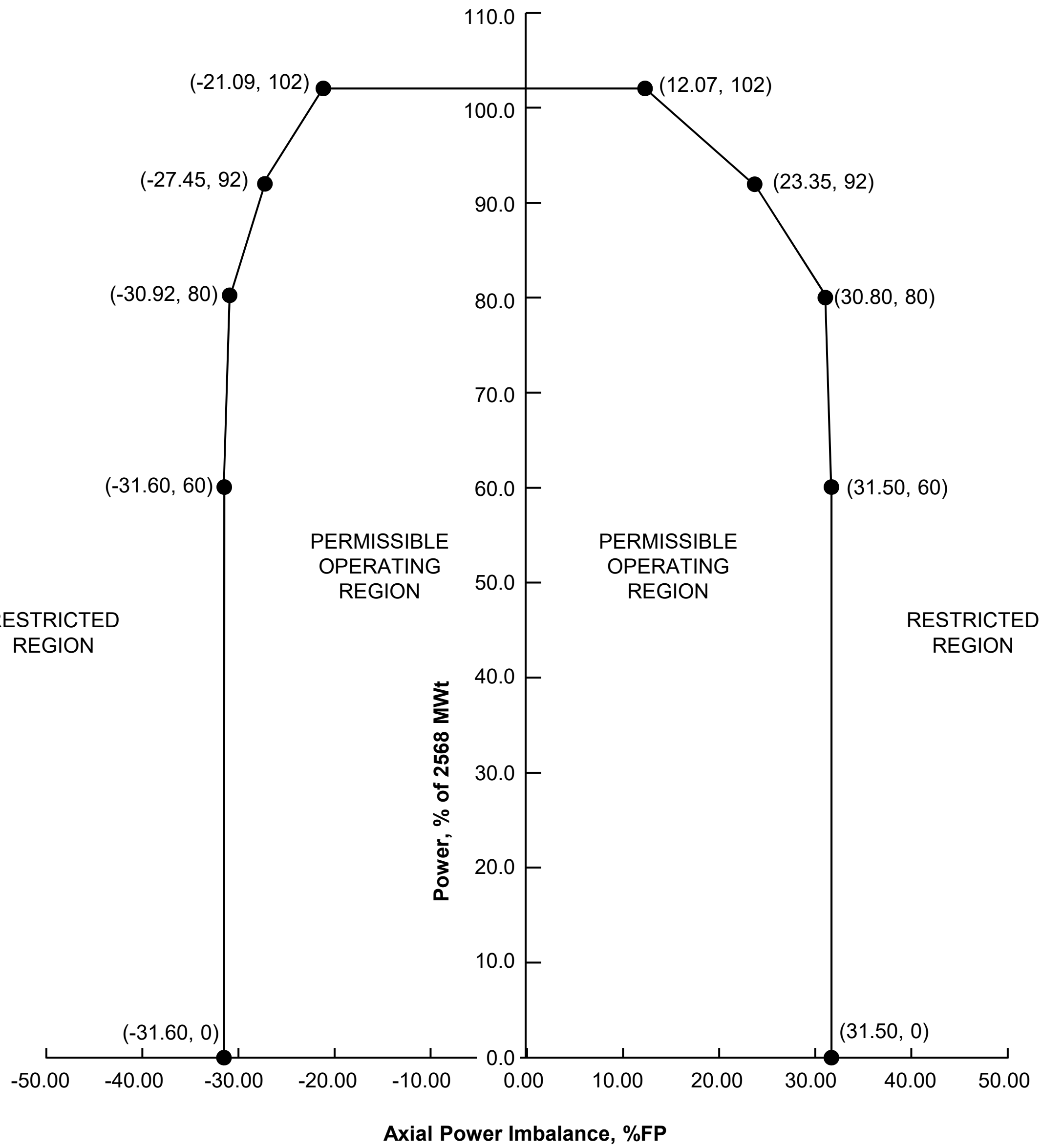


* Based on the condition that no individual long emitter detector affecting the minimum in-core imbalance calculations exceeds 73% sensitivity depletion. The imbalance alarm setpoints for the minimum in-core system must be reduced by 2.8% FP at the earliest time-in-life this condition is no longer valid.

SAR FIGURE NO. 3A-15B

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR FOUR PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 MINIMUM IN-CORE SYSTEM* (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET / REV.

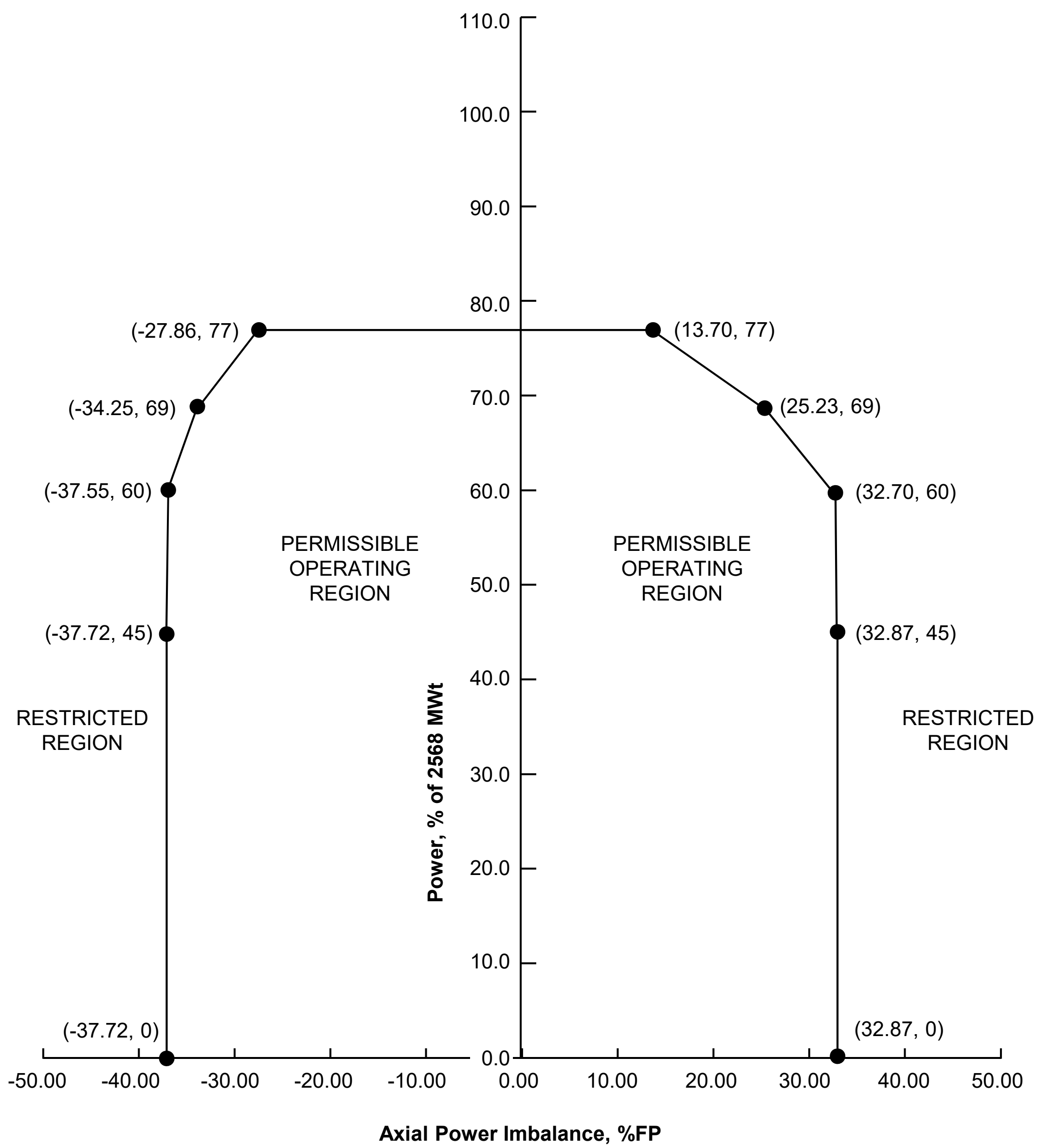


SAR FIGURE NO. 3A-15C

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

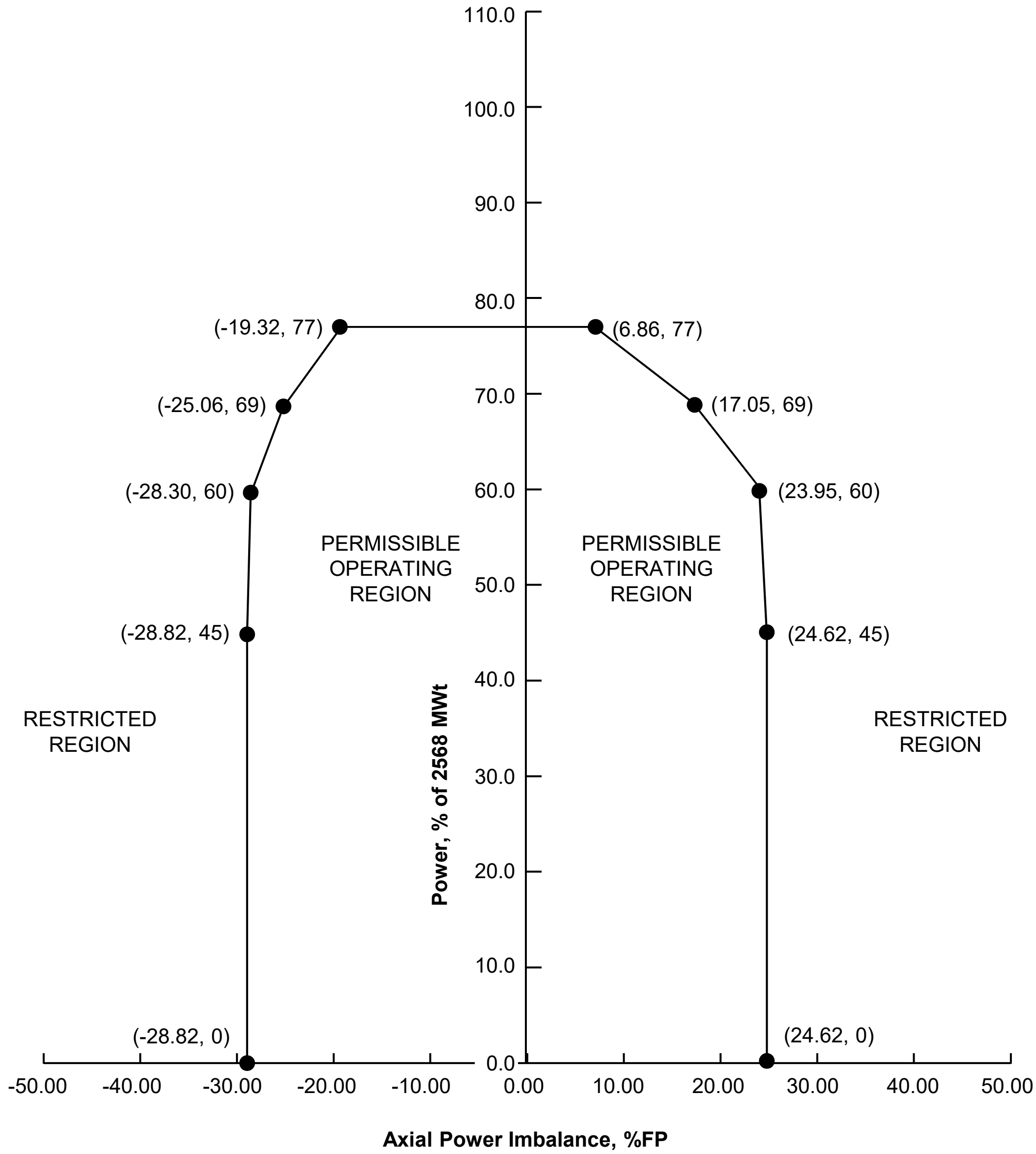
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR FOUR PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 EX-CORE DETECTOR SYSTEM (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3A-16A

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR THREE-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 FULL IN-CORE SYSTEM (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET / REV.

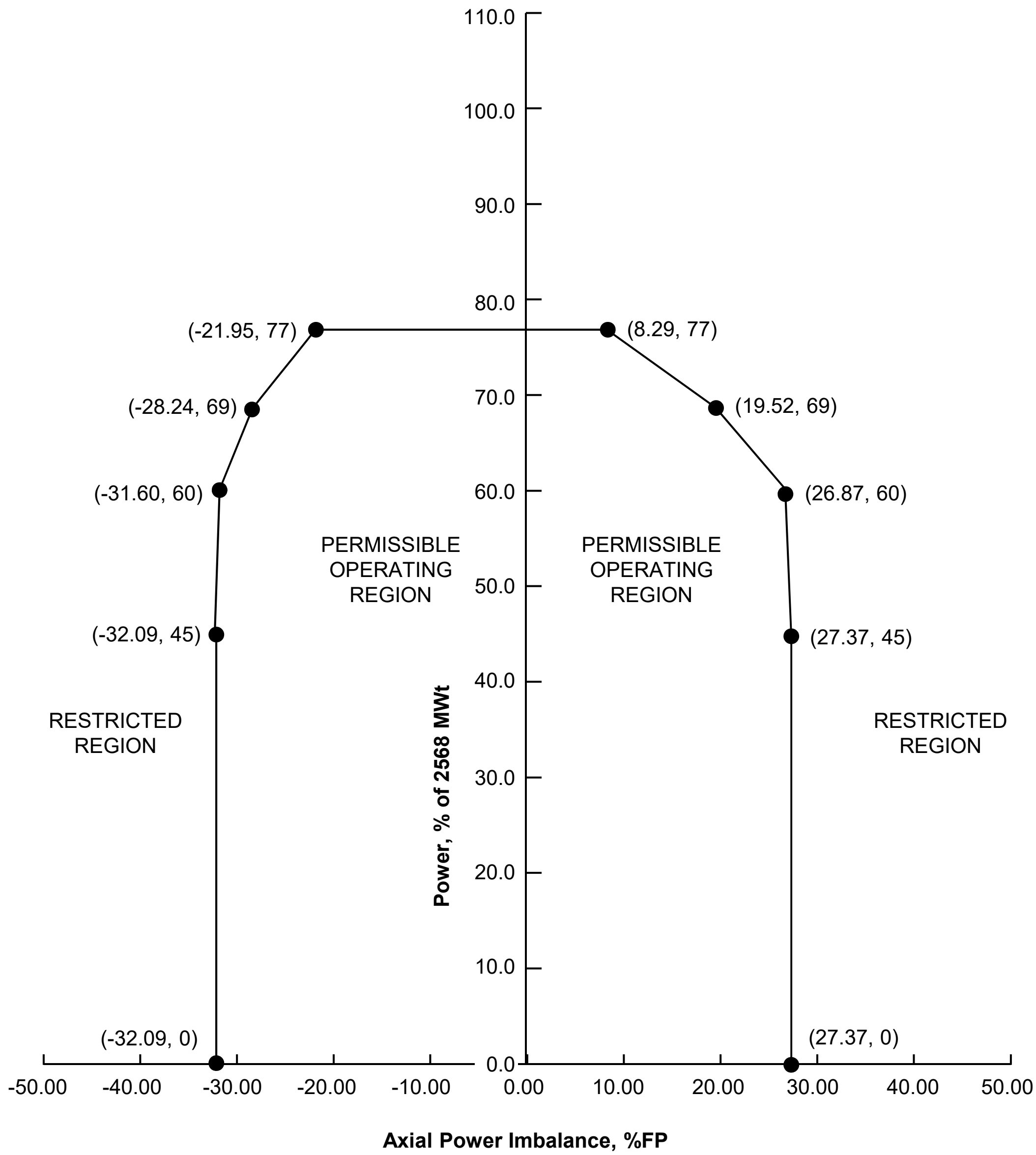


* Based on the condition that no individual long emitter detector affecting the minimum in-core imbalance calculations exceeds 73% sensitivity depletion. The imbalance alarm setpoints for the minimum in-core system must be reduced by 2.8% FP at the earliest time-in-life this condition is no longer valid.

SAR FIGURE NO. 3A-16B

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR THREE-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 MINIMUM IN-CORE SYSTEM* (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET / REV.

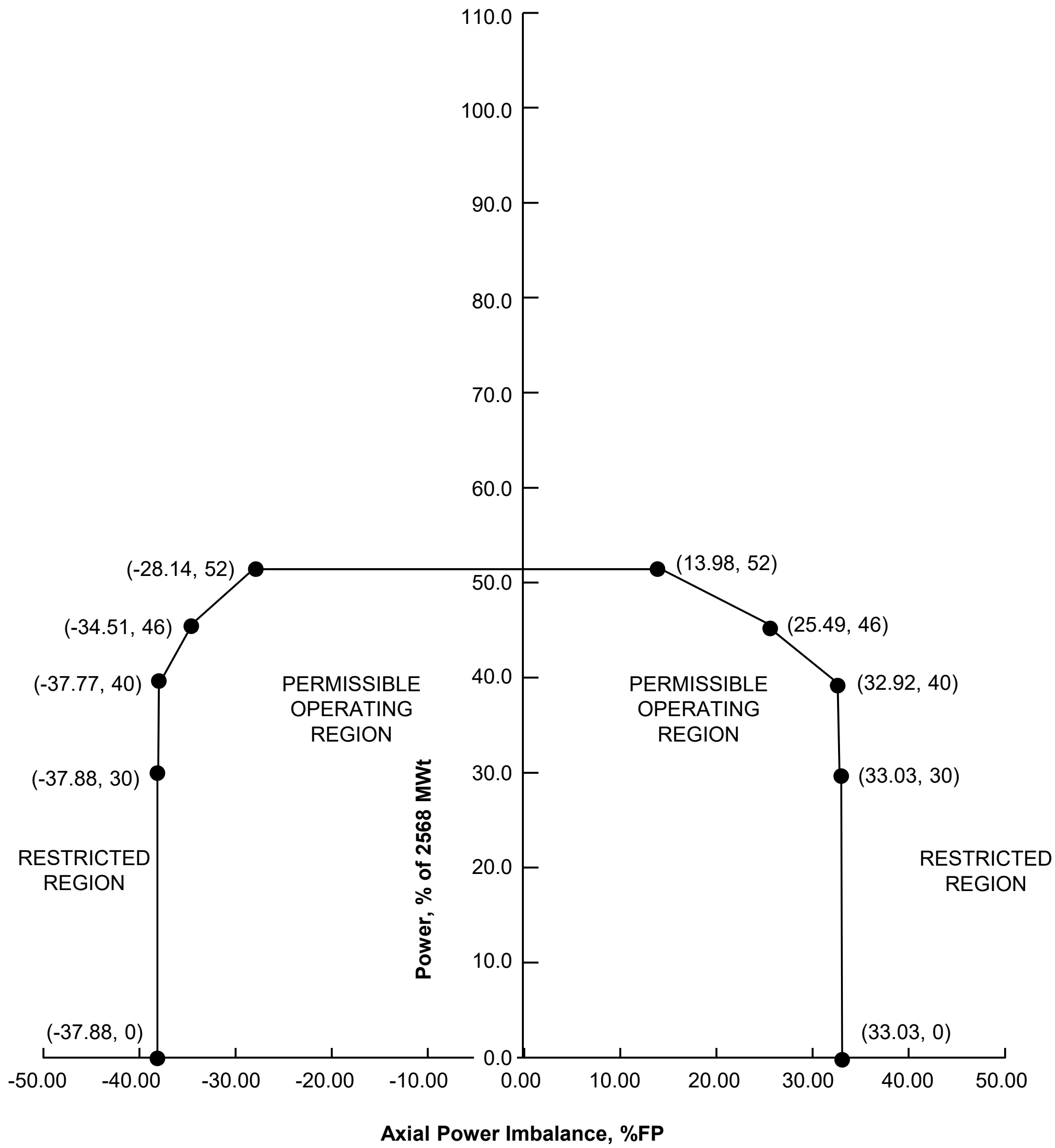


SAR FIGURE NO. 3A-16C

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR THREE-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 EX-CORE DETECTOR SYSTEM (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET	REV.

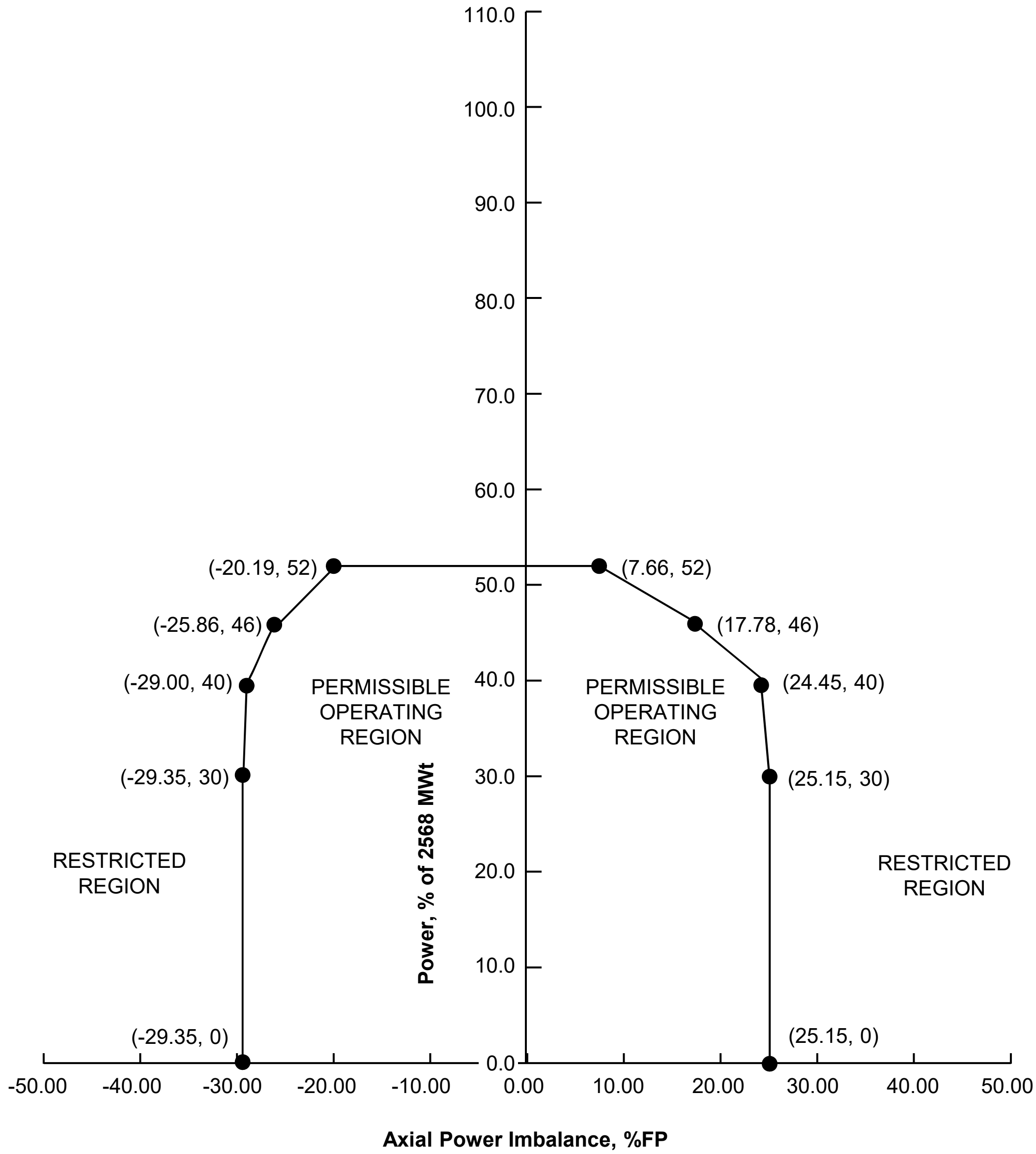


SAR FIGURE NO. 3A-17A

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR TWO-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 FULL IN-CORE SYSTEM (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET	REV.

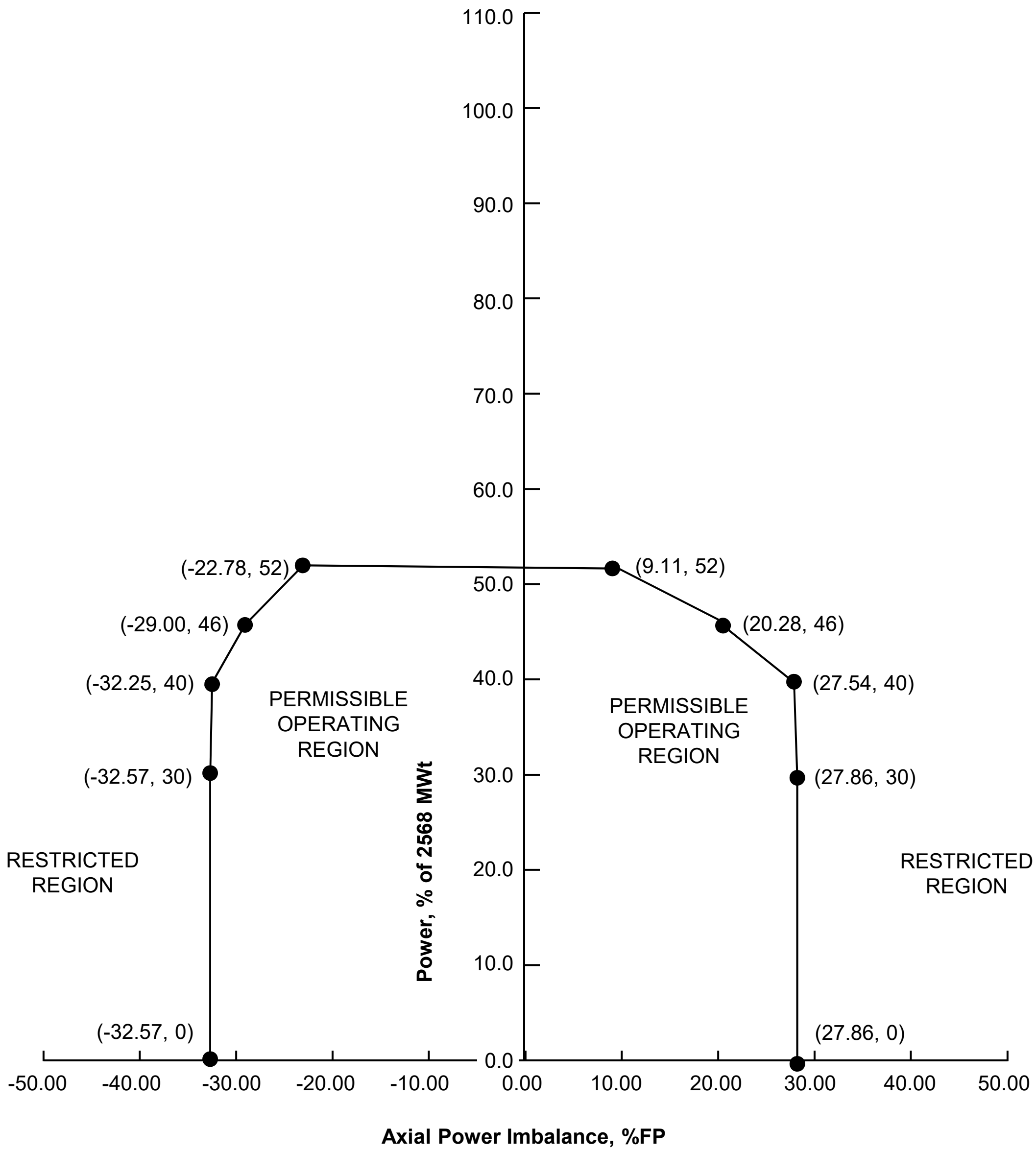


* Based on the condition that no individual long emitter detector affecting the minimum in-core imbalance calculations exceeds 73% sensitivity depletion. The imbalance alarm setpoints for the minimum in-core system must be reduced by 2.8% FP at the earliest time-in-life this condition is no longer valid.

SAR FIGURE NO. 3A-17B

AMENDMENT 30

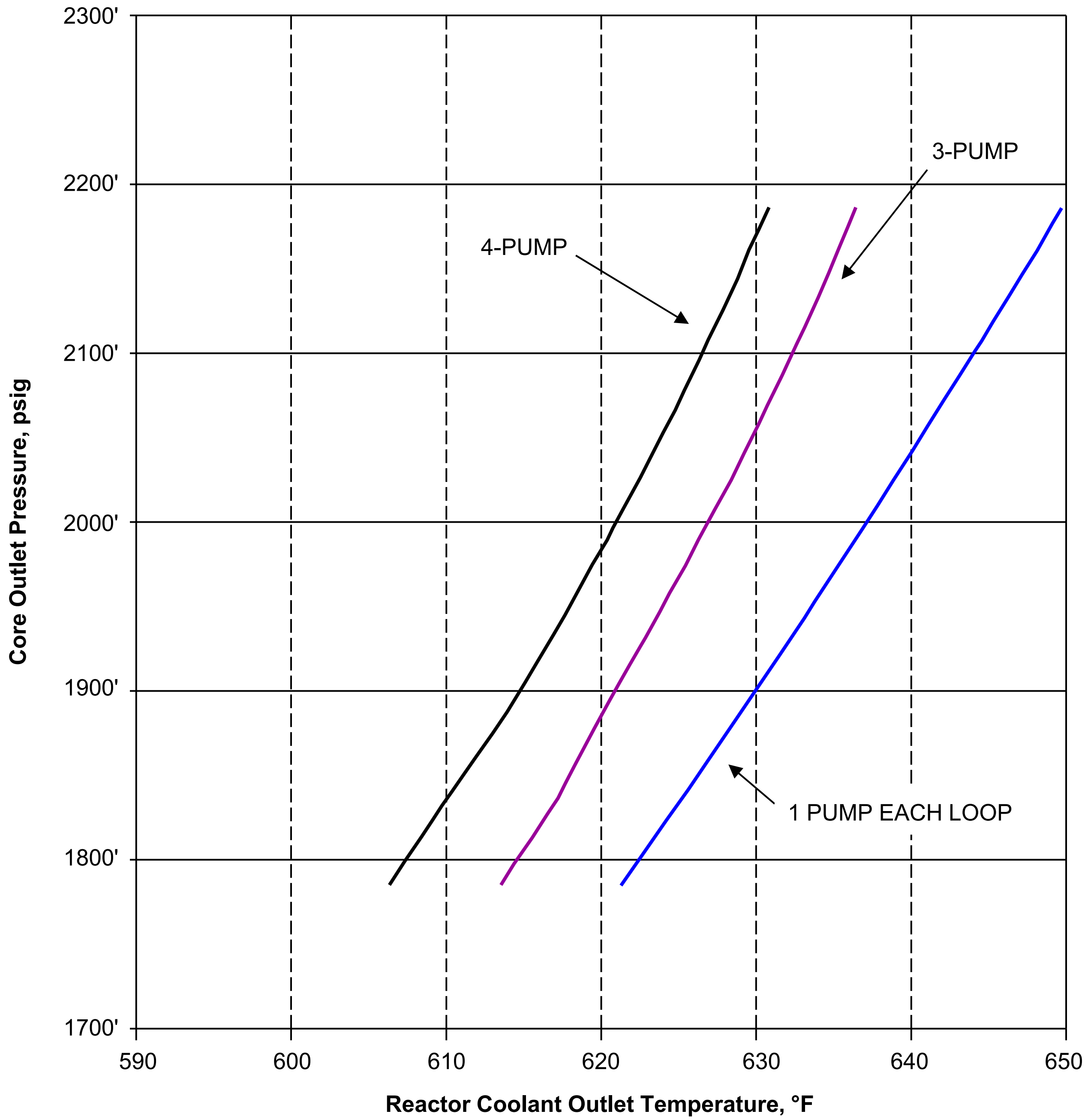
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR TWO-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 MINIMUM IN-CORE SYSTEM* (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3A-17C

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
OPERATIONAL POWER IMBALANCE ALARM SETPOINTS FOR TWO-PUMP OPERATION FROM 0 EFPD TO EOC FOR ANO-1 CYCLE 30 EX-CORE DETECTOR SYSTEM (Referred to by Tech Spec 3.2.3)	BASED ON DRAWING NO	SHEET / REV.



PUMPS OPERATING (TYPE OF LIMIT)	GPM*	POWER**
FOUR PUMPS (DNBR LIMIT)	383,680 (100%)	110%
THREE PUMPS (DNBR LIMIT)	284,307 (74.1%)	89%
ONE PUMP IN EACH LOOP (DNBR LIMIT)	188,003 (49%)	62.2%

* 109% OF DESIGN FLOW (2.5% UNCERTAINTY INCLUDED IN STATISTICAL DESIGN LIMIT)

** AN ADDITIONAL 2% POWER UNCERTAINTY IS INCLUDED IN STATISTICAL DESIGN LIMIT

SAR FIGURE NO. 3A-18

AMENDMENT 30

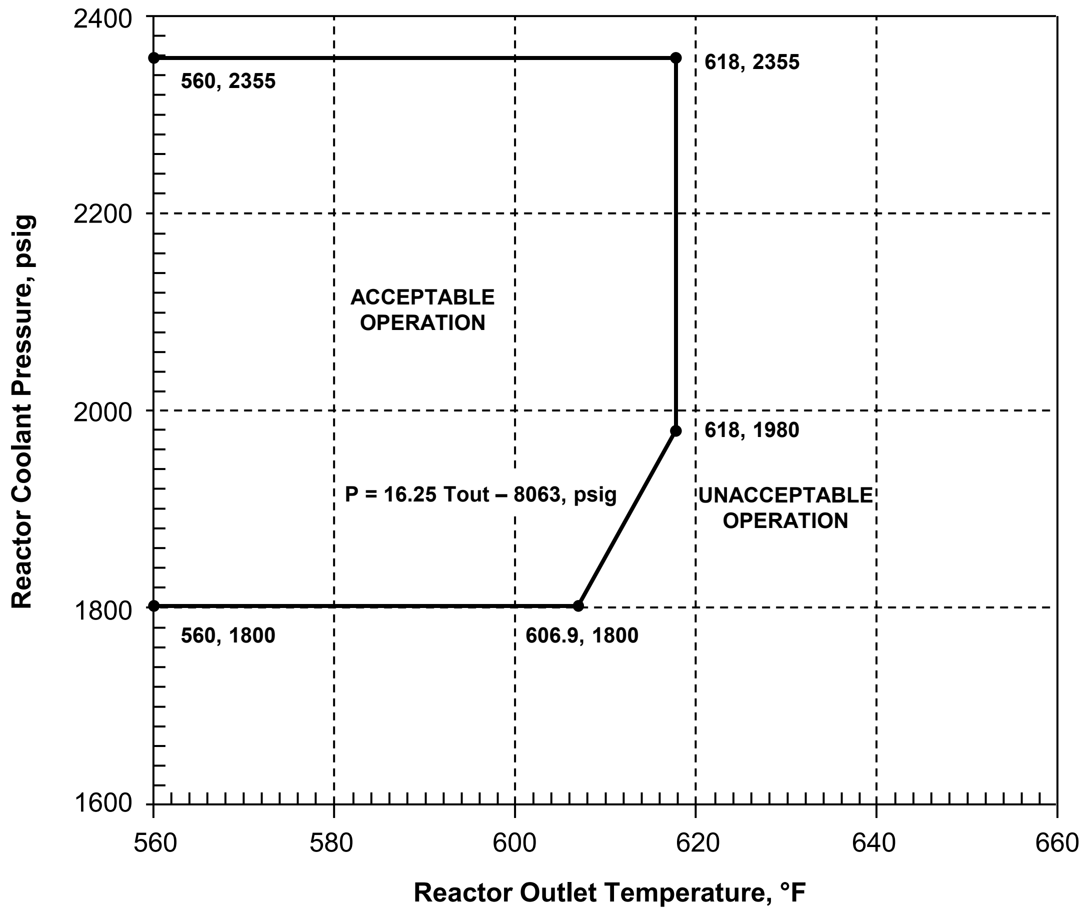
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

VARIABLE LOW PRESSURE – TEMPERATURE PROTECTIVE LIMITS (Referred to by Tech Spec 2.1.1.3)

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 3A-19

AMENDMENT 30

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
RCS PRESSURE – TEMPERATURE PROTECTIVE MAXIMUM ALLOWABLE LIMITS (Referred to by Tech Spec 3.3.1)	BASED ON DRAWING NO	SHEET / REV.

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
4.2.3.8	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated January 5, 1981. (0CAN018105)
4.2.2.5	Design Change Package 206, "Reactor Coolant Pump Protective Instrumentation, Seal Leakage Flow (Drinking Bird)," 1974. (1DCP740206)
4.2.3.8	Design Change Package 285, "RCS Leak Detection System Modification," 1975. (1DCP750285)
4.2.2.1 4.4.5	Design Change Package 432, "Reactor Surveillance Specimens, Removal of Specimen Holder Tubes," 1976. (1DCP760432)
4.2.3.6.3	Design Change Package 568, "High Purity H_2 Supply to Unit 2 Volume Control Tank," 1977. (0DCP770568)
4.2.5.2	Design Change Package 657, "Rewire ESF Pump Motor Starting Scheme to Allow Sequencing of Pumps," 1978. (1DCP780657)
4.2.4	Design Change Package 1050, "RCS Relief Valve Position Indication," 1979. (1DCP791050)
4.2.3.2 4.2.4	Design Change Package 1051, "RCS TSAT Meter," 1979. (1DCP791051)
4.2.4	Design Change Package 1062, "TMI Related Work, NUREG-0578, Block Valve Motive Power," 1979. (1DCP791062)
4.2.3.8	Design Change Package 1065, "Containment Water Level Instrumentation Upgrade," 1979. (1DCP791065)
4.2.4.4	Design Change Package 1078, "NUREG-0578 TMI-Related Work (Pressurizer Heaters)," 1979. (1DCP791078)
4.2.4.3	Design Change Package 1031, "Pressurizer Spray MOV.," 1980. (1DCP801031)
4.2.4.3	Design Change Package 1056 & 1056A, "Core Flood Tank MOVs on Emergency Diesel Power," 1980. (1DCP801056)
4.1	Design Change Packages 80-1180B & C, "Spool Pieces, Design Valves and Piping Support."
Figure 4-1	Design Change Package 80-1004B, "Remote Operated RCS High Point Vents."
Figure 4-1	Design Change Package 80-2043F, "Construction of Post Accident Sampling System."

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4.1.3.1 4.6 Table 4-2 Table 4-4	Design Change Packages 81-1057 and 81-1057A, "MFW Nozzles Replacement."
4.2 4.6	Design Change Package 80-1183, "Emergency Feedwater Initiation."
4.2	Design Change Package 83-1012, "ISO Device for Pressurizer ERV Block Valve."
4.6	Design Change Package 83-1075, "T9-01B Replacement of PZR Level and NNI Steam Generator Level."
4.2.38	Correspondence from Trimble, AP&L to Seyfrit, NRC, dated January 2, 1981. (0CAN018105)
4.2.4.2	Correspondence from Marshall, AP&L, to Eisenhut, NRC, dated November 1, 1982. (0CAN118203)
4.1.3.2 4.1.3.7 4.2.2.4 Table 4-6	Design Change Package 84-1014D, "Hot Leg Level Taps."
4.2.3.6.2	Correspondence from Tom Baker, ANO, to J. Ted Enos, NEL, (ANO-86-06935).
4.2.4 Table 4-1	Design Change Package 86-1016, "ERV Position Indicator."
4.2.6.6 Table 4-9	Design Change Package 85-1025B, "RCP Removal Documentation."
Table 4-5	Design Change Package 83-1075, "T9-01B Replacement of Pressurizer Level and NNI Pressurizer Level."
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4 REACTOR COOLANT SYSTEM

4.1 DESIGN BASES

4.1.1 PERFORMANCE OBJECTIVES

4.1.1.1 Steam Output

The Reactor Coolant System (RCS) design information presented in this section is for a design reactor power level of 2,568 MWt with an additional 21 MWt input from the Reactor Coolant Pumps, transferring a total of 2,589 MWt to the steam generators (see Section 1.2.2). The system is capable of producing a total steam flow of 11.2 million lb/hr.

4.1.1.2 Transient Performance

The RCS will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

A. Step load changes - increasing load steps of 10 percent of full power in the range between approximately 22 percent and 90 percent full power and decreasing load steps of 10 percent of full power between 100 percent and approximately 22 percent full power.

B. Ramp load changes - increasing load ramps of 10 percent per minute in the range between approximately 22 percent and 90 percent full power are acceptable, or decreasing load ramps of 10 percent per minute from 100 percent to approximately 22 percent full power. From 90 percent to 100 percent full power, increasing ramp load changes of five percent per minute are acceptable.

The combined actions of the control system, the turbine bypass valves, and the main steam safety valves are designed to accept separation of the generator from the transmission system without reactor trip.

However, this original design included a RPS high RCS pressure trip setpoint above the Pressurizer Electromatic Relief Valve (ERV) setpoint. As a result of the TMI-2 event in 1979, the NRC ordered a reversal of these setpoints such that a high RCS pressure trip is reached before the ERV is challenged. This arrangement effectively eliminated the unit's ability to survive a load rejection from full power without a reactor trip unless operator action is taken.

Also as a result of the TMI-2 event, the NRC ordered the installation of the Anticipatory Reactor Trip System (ARTS). The ARTS will reduce the energy input to the RCS by a prompt trip upon loss of the OTSG secondary heat sink. This system trips the reactor upon either loss of main feedwater or turbine trip, actuating about eight seconds sooner than the RCS high pressure trip. The ARTS is redundant, safety grade, and meets the requirements of IE Bulletin 79-05B and NUREG-0578. The ARTS hydraulic pressure sensing switches, however, are located near the main turbine and main feedwater pumps in the non-seismic (non type-I) turbine building.

As required by NUREG-0737, Supplement 1, system-oriented Emergency Operating Procedures (EOPs) were implemented based on the Abnormal Transients Operational Guidelines (ATOG) developed by the B&W Owners Group. This program utilized event trees, safety sequence, and system auxiliary diagrams to develop operator guidelines.

4.1.1.3 Partial Loop Operation

The RCS will permit operation with less than four Reactor Coolant Pumps in operation. The steady state operating power levels for combinations of Reactor Coolant Pumps operating are as follows:

Reactor Coolant Pumps Operating	Rated Power, %
4	100
3	75
1 pump each loop	49

Section 7.1.2 describes the protective system that allows partial loop operation.

4.1.2 DESIGN CONDITIONS

4.1.2.1 Pressure

The RCS components are designed structurally for an internal pressure of 2,500 psig.

4.1.2.2 Temperature

The RCS components are designed for a temperature of 650 °F with the exception of the pressurizer, surge line, and a portion of the spray line piping which are designed for 670 °F.

4.1.2.3 Reaction Loads

All components in the RCS are supported and interconnected so that piping reaction forces result in combined mechanical and thermal stresses in equipment nozzles and structural walls within established code limits. Equipment and pipe supports were originally designed to absorb piping rupture reaction loads for elimination of secondary accident effects such as pipe motion and equipment foundation shifting. Later, credit was taken for the "Leak-Before-Break" criterion, (B&W Owners Group reports BAW-1847, Rev. 1, and BAW-1889-P) specifically for the removal of the wire tie ropes on the RCPs and for the elimination of the dynamic loads from a hot leg or cold leg break on the steam generator upper lateral restraints and lower base support.

As part of the OTSG replacement activities, a structural analysis of the ANO reactor coolant system was performed. A three-dimensional static (thermal expansion, dead load) and dynamic (seismic, pipe rupture) analysis was performed on the reactor coolant system to determine piping and component nozzle stresses. An idealized mathematical model for both loops (A and B) consisting of distributed and lumped masses connected by elastic members has been employed. The model includes the reactor vessel, steam generators, the pressurizer, all four coolant pumps with associated piping, and the secondary shield wall with any attached coolant system restraints or supports. The pipe rupture analyses relied upon Leak-Before-Break methodology to determine the appropriate pipe rupture locations, which included guillotine breaks in the decay heat line, the surge line, the core flood line, the main feedwater line, and the main steam line. Theses analyses demonstrated that the ANO-1 reactor coolant system piping, component nozzles, and component supports were adequate to withstand the induced loadings, when considering the current design basis allowable stresses (Reference 21).

4.1.2.4 Cyclic Loads

All RCS components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles are shown in Table 4-8.

Flow induced vibration analyses have been performed for the fuel assembly, including fuel rods, and for the reactor internals components. The analyses and design criteria for the thermal shield, flow distributor assembly, surveillance holder tubes and shroud tubes, and the U-baffles are given in B&W Topical Report BAW-10051, Revision 1, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations" (see Section 4.4.6 for the modified surveillance tube program).

Components subjected to cross flow are checked for response during design so that the fundamental frequencies associated with cross flow are above the vortex shedding frequencies. It has also been conservatively determined that the flow induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc. Emergency operational modes are covered in Topical Report BAW-10008, Part 1, Revision 1, and BAW-10035, Revision 1.

4.1.2.5 Seismic Loads and Loss of Coolant Loads

RCS components are designated as Seismic Category 1 equipment and are designed to maintain their functional integrity during an earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals, vessels, integral support attachments, and piping are given in this section. A discussion of each of the cases of loading combinations follows.

4.1.2.5.1 Seismic Loads

Case I - Design Loads Plus Design Earthquake Loads - For this combination, the reactor must be capable of continued operation; therefore, all components, excluding piping, are designed to Section III of the ASME Code for Reactor Vessels (see Reference 15). The primary piping is designed according to the requirements of ANSI B31.7. However, the use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled. The S_m values for all components are specified in the ASME Code. Values for bolts are those specified in Table N-421 of the ASME B&PV Code. Details are reported in Topical Report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake."

Case II - Design Loads Plus Maximum Hypothetical Earthquake Loads - In establishing stress levels for this case, a "no-loss-of-function" criterion applies and higher stress values than in Case I can be allowed. The multiplying factor of 1.2 has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e. 304 SS, 316 SS, SA533B, SA516GR.70, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in B&W Topical Report BAW-10008, Part 1, Revision 1. S_m values for all components are those specified in Table N-421 of the ASME B&PV Code.

4.1.2.5.2 Loss-of-Coolant Loads

A Loss of Coolant Accident (LOCA) coincident with a seismic disturbance case was originally analyzed to assure the ability to initiate and maintain reactor shutdown and emergency core cooling. The following cases (III & IV), which were originally considered, are not entirely applicable due to the leak-before-break option being used to qualify the RCS for pipe rupture loads without the tie ropes on the RCPs and to qualify the steam generator upper lateral restraints and lower base support following the steam generator replacement.

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling. (Note: All the piping was originally qualified to the above criteria. Without the pipe rupture loads being applied due to the leak-before-break criterion being used in the specific area around the RCPs and replacement steam generators, and due to the removal of the wire tie ropes, this is now just a qualification of the design loads where the design load allowables apply (ASME Code)).

Case IV - Design Loads Plus Maximum Hypothetical Earthquake Loads Plus Pipe Rupture Loads - Two-thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of maximum hypothetical earthquake and reactor coolant pipe rupture. The stress compared to the 2/3 S_u allowable was calculated using elastic equations. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist. (Note: All piping was originally qualified to the above criteria. Without pipe rupture loads being applied due to the leak-before-break criterion now being used in the specific area around the RCPs and replacement steam generators, and due to the removal of the wire tie ropes, the loads qualified here are the same as the loads in Case II and the allowables in Case II apply.)

In addition to the Reactor Coolant Pressure Boundary components, the stress limits for design Case IV have been applied to the reactor internals and to the internal supports of the reactor vessel, steam generators, and pressurizer. The bases for such application are as follows:

A. All integral supports are manufactured and inspected in accordance with the same requirements as specified for pressure boundary components. Consequently the same stress limits are applicable.

B. Where applicable, reactor internals are manufactured and inspected to the same standards as pressure boundary components. Thus the same allowable stresses are used.

C. Since stress limits established for Case IV are conservative, components designed to them will have a substantial margin to failure (50 percent). Therefore the criteria for a safe and orderly shutdown and the maintenance of open coolant passages will be met.

In Cases II, III, and IV, secondary stresses were neglected since they are self-limiting. Design stress limits in most cases are in the plastic region and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. Topical Report BAW-10008, Part 1, contains a more extensive discussion of the stresses, the deflections, the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals.

The applicable design stress intensity limits associated with Cases I through IV of Section 4.1.2.5 above are as follows:

SB-163	S Ultimate	=	80,000 psi
	S Yield	=	35,000 psi at 100 °F
	S_m	=	23,300 psi

Case	Stress Limit	Limit Value
I	P_m equal to or less than 1.0 S_m	23,300 psi
	P_l plus P_b equal to or less than 1.5 S_m	35,000 psi
II	P_m equal to or less than 1.2 S_m	27,960 psi
	P_l plus P_b equal to or less than 1.2(1.5 S_m)	42,000 psi
III	P_m equal to or less than 1.2 S_m	27,960 psi
	P_l plus P_b equal to or less than 1.2(1.5 S_m)	42,000 psi
IV	P_m equal to or less than 2/3 S_u	42,600 psi at 600 °F
	P_l plus P_b equal to or less than 2/3 S_u	42,600 psi at 600 °F

The design conditions are conservative values of the maximum expected pressures, temperatures and operating transients.

The steam generator tubes are sized for the reactor coolant side pressures and temperatures acting alone using methods specified in ASME Section III, Subsection NB. The design pressure is 2,500 psig and the design temperature is 650 °F. Other design data is specified in Table 4-4, Steam Generator Design Data, and Table 4-9, Materials of Construction. The supporting effect of secondary side pressure is not considered in sizing the tubes. The tubes are designed for the system transients specified in Table 4-8, Transient Cycles.

STRESS LIMITS FOR
SEISMIC, PIPE RUPTURE AND COMBINED LOADS

Case	Loading Combination	Stress Limits
I	Design loads + design earthquake loads	$P_m \leq 1.0\ S_m$
		$P_l + P_b \leq 1.5\ S_m$
II	Design loads + maximum hypothetical earthquake loads	$P_m \leq 1.2\ S_m$
		$P_l + P_b \leq 1.2\ (1.5\ S_m)$
III	Design loads + pipe rupture loads (Originally qualified using these stress limits. Currently this does apply to the specific area of the RCPs and replacement steam generators, since leak-before-break option was used for the deletion of the RCP wire tie ropes and elimination of hot leg and cold leg break loads on the steam generator upper lateral restraints and lower base support.)	$P_m \leq 1.2\ S_m$ $P_l + P_b \leq 1.2\ (1.5\ S_m)$

IV	Design loads + maximum hypothetical (Originally qualified using these stress limits. Currently this does apply to the specific area of the RCPs and replacement steam generators, since leak-before-break option was used for the deletion of the the RCP wire tie ropes and elimination of hot leg and cold leg break loads on the steam generator upper lateral restraints and lower base support.)	$P_m \leq 2/3\ S_u$ $P_l + P_b \leq 2/3\ S_u$

where*
P_l = Primary local membrane stress intensity

P_m = Primary general membrane stress intensity

P_b = Primary bending stress intensity

S_m = Allowable membrane stress intensity

S_u = Ultimate stress for unirradiated material at operating temperature

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*(1) All symbols have the same definition or connotation as those in ASME B&PV Code Section III, Nuclear Vessels.

(2) All components will be designed to insure against structural instabilities regardless of stress levels.

4.1.2.6 Service Lifetime

The original design service lifetime for the major RCS components was 40 years. The number of cyclic system temperature, pressure, and operational changes (Table 4-8) was based on operation for this design lifetime. The commencement date for the original design service life was the date of the Construction Permit which approved the PSAR for Unit 1, which is December 6, 1968. However, in 1990, per License Amendment 131, an extension was granted to allow the operating license term to be changed to start at the issuance of the operating license to allow a 40-year service life that does not include the construction time period and end on May 20, 2014. A new operating license has been granted to extend the licensed term an additional 20 years to May 20, 2034. This was justified based on design transient cycles. The reactor coolant system was originally qualified using a conservative estimate of design cycles for a 40-year life. The design life is not dependent on years of service. The design life is dependent on fatigue cycles. In evaluations performed by the NSSS vendor, the actual cycles were extrapolated to 60 years. For the major RCS components, the design cycles exceeded the estimated cycles for a 60-year life. The original OTSGs were replaced and the replacement OTSGs have a design life of 40 years.

The actual transient cycles are tracked and documented to ensure they are maintained below the allowable number of design cycles as further discussed in Section 16.3.2. Table 4-8 shows the complete listing of transients from the RCS functional specification used in the design of components within the Reactor Coolant Pressure Boundary. Table 4-8A contains a listing of transients used by the Transient Cycle Logging Program. It includes additional transients contributing to component fatigue and eliminates logging transients having little or no effect on fatigue stresses. Records of significant transients are available from the periodic reviews of the Operations log. A record of all significant transients is maintained by System Engineering.

4.1.2.7 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the RCS surfaces. The design water quality is listed in Table 9-4. The reactor coolant chemistry is discussed in further detail in Section 9.2.

4.1.2.8 Vessel Radiation Exposure

The reactor vessel is the only RCS component exposed to a significant level of neutron irradiation and is therefore the only component subject to material radiation damage. The maximum exposure from fast neutrons (E > 1.0 MeV) was previously computed to be less than 3.0×10^{19} neut/cm^2 over a 40-year life with an 80 percent load factor. Revised calculations to support the extended 60-year life with a 90 percent load factor, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum exposures will be less than half of this value. The maximum inside surface fast neutron fluence at 54 EFPY is projected to be 1.34×10^{19} neut/cm^2. Reactor vessel irradiation calculations are described in Section 4.3.3.

4.1.3 CODES AND CLASSIFICATIONS

The codes listed in this section (as tabulated in Table 4-2) include the code addenda and case interpretations issued through Summer 1967 (June 30, 1967) unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in Section 4.3.12. Code case interpretations issued through January 21, 1972, which have been used in implementing codes delineated in Appendix A, are given in Table 4-2A.

4.1.3.1 Vessels

The design, fabrication, inspection, and testing of the reactor vessel and closure head, and pressurizer is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels.

The steam generator primary side pressure boundary is classified, designed, fabricated, and stamped as ASME Section III Code Class 1. The steam generator secondary side pressure boundary is classified and stamped as ASME Section III Code Class 2. The temperature sensing nozzles and their connecting welds are designed and fabricated to ASME Section III Class 2 requirements. The remainder of the secondary pressure boundary is designed and fabricated to ASME Section III Code Class 1 requirements.

Repairs or modifications made to the steam generator pressure boundary shall meet all of the requirements for a Class 1 vessel of Article IWA-4000 of ASME B&PV Code, Section XI. Repairs or modification made to the steam generator secondary pressure boundary shall meet at least the requirements for a Class 2 vessel of Article IWA-4000 of ASME B&PV Code, Section XI. Repairs or modifications made to non-pressure boundary parts of the steam generator shall be consistent with the intent of the requirements for Class 1 hardware contained in ASME B&PV Code, Section XI, Article IWA-4000 and good engineering practices.

4.1.3.2 Piping

The design, fabrication, inspection, and testing of the reactor coolant piping as listed in Table 4-6 is in accordance with ANSI B31.7, "Code for Pressure Piping, Nuclear Power Piping," dated February 1968 and as corrected for Errata under date of June 1968, or later appropriate ASME Section III Code sections provided that they have been reconciled.

Repairs or modifications made to the RCS piping, including the Class 1 portions of the decay heat line, shall meet the requirements for Class 1 components in accordance with IWA-4000 of ASME B&PV Code, Section XI.

4.1.3.3 Reactor Coolant Pump

The Reactor Coolant Pump casings are designed, fabricated, inspected, and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels but are not code stamped.

4.1.3.4 Safety Valves

The pressurizer code safety valves comply with Article 9, Section III, of the ASME Boiler and Pressure Vessel Code. The flanged inlets and outlets of these valves are designed in accordance with ANSI B16.5. Each nozzle, which is retained by the valve body, is designed to limit the stress due to internal pressure to ASME allowables.

4.1.3.5 Attachments to Loop

Nozzles on the reactor coolant piping comply with Section 4.1.3.2 and nozzles on the vessels comply with Section 4.1.3.1.

4.1.3.6 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessel Code, Section III and Section IX, as applicable.

4.1.3.7 Valves

All valves within the Reactor Coolant Pressure Boundary which were originally furnished by Bechtel were purchased after January 1, 1970 and were specified to meet Paragraph 452.1a of the BP&V code.

All valves within the Reactor Coolant Pressure Boundary which were originally furnished by B&W were purchased prior to January 1, 1970. These valves were specified to meet ANSI B16.5 or MSS-SP66 with material control and inspection to ANSI B31.7 with the exception of the pressurizer safety valves which were specified to meet ASME Section III, Article 9.

The valves which have been added to the RCS as a part of the hot leg level instrumentation system are 1" Kerotest globe valves and were designed, fabricated, and tested to the applicable requirements of subsection NB of the ASME Code, Section III.

Valves within the reactor coolant boundary can be procured from any manufacturer who has a QA program which meets the requirements of 10 CFR 50, Appendix B and ASME Section III, Division 1, Subsection NCA.

4.2 SYSTEM DESCRIPTION AND OPERATION

4.2.1 GENERAL

4.2.1.1 System

The Reactor Coolant System (RCS) consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed Reactor Coolant Pumps, one electrically heated pressurizer, and interconnecting piping. The system is arranged in two heat transport loops, each with two Reactor Coolant Pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a Reactor Coolant Pump. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector and a solvent for the soluble poison (boron in the form of boric acid). The RCS schematic is shown in Figure 4-1.

4.2.1.2 System Protection

A. Reaction Loads

All components in the RCS are supported and interconnected so that piping reaction forces result in combined mechanical and thermal stresses in equipment nozzles and structural walls within established code limits. Equipment and pipe supports were originally designed to absorb piping rupture reaction loads for elimination of secondary accident effects and equipment foundation shifting. Later, credit was taken for the "Leak-Before-Break" criterion, (B&W Owners Group reports BAW-1847, Rev. 1, and BAW-1889-P), specifically for the removal of the wire tie ropes on the RCPs and elimination of hot leg and cold leg break loads on the steam generator upper lateral restraints and lower base supports.

The RCS is surrounded by concrete shield walls. These walls provide shielding to permit access into the reactor building for inspection and maintenance of miscellaneous rotating equipment during rated power operation and for periodic calibration of the Incore Monitoring System. These shielding walls act as missile protection for the reactor building liner plate.

A lateral support structure is provided near the steam generator upper tubesheet elevation to resist lateral loads, including those resulting from seismic forces, and pipe rupture due to small break LOCAs. The RCS hot leg and cold leg pipe break loads are no longer required due to credit for leak-before-break. Barriers are provided over certain portions of the RCS for shielding and missile damage protection.

As part of the OTSG replacement activities, a structural analysis of the ANO reactor coolant system was performed. A three-dimensional static (thermal expansion, dead load) and dynamic (seismic, pipe rupture) analysis was performed on the reactor coolant system to determine piping and component nozzle stresses. An idealized mathematical model for both loops (A and B) consisting of distributed and lumped masses connected by elastic members has been employed. The model includes the reactor vessel, steam generators, the pressurizer, all four coolant pumps with

associated piping, and the secondary shield wall with any attached coolant system restraints or supports. The pipe rupture analyses relied upon Leak-Before-Break methodology to determine the appropriate pipe rupture locations, which included guillotine breaks in the decay heat line, the surge line, the core flood line, the main feedwater line, and the main steam line. Theses analyses demonstrated that the ANO-1 reactor coolant system piping, component nozzles, and component supports were adequate to withstand the induced loadings, when considering the current design basis allowable stresses (Reference 21).

B. Seismic Loads

RCS components are designated as Seismic Category 1 equipment and are designed to maintain their functional integrity during an earthquake. The basic design guide for the seismic analysis is the AEC publication TID-7024, "Nuclear Reactors and Earthquakes." Structures and equipment are designed in accordance with Section 5.1.

4.2.1.3 System Arrangement

The arrangement of the RCS is shown in Figures 4-2 and 4-3. Figures in Chapter 1 depict the system arrangement in relation to shielding walls, the reactor building, and other equipment in the building.

4.2.2 MAJOR COMPONENTS

4.2.2.1 Reactor Vessel

The reactor vessel consists of a cylindrical shell, a cylindrical support skirt, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor closure head is a one piece forging, which consists of a spherically dished head with a matching ring flange. The reactor vessel general arrangement is shown in Figure 4-4. The general arrangement of the reactor vessel with internals is shown in Figures 3-59 and 3-60. Reactor vessel design data are listed in Table 4-3.

The number and size of reactor vessel nozzles are also shown in Table 4-3. All coolant inlet, coolant outlet, core flooding, and Control Rod Drive nozzles are located above the level of the top of the core. The reactor vessel is vented through the Control Rod Drives.

The core flood nozzle contains a venturi type flow restrictor as shown in Figure 3-61. The restrictor and attachment weld are designed in accordance with ASME Section III. The significant transients which affect the restrictor and weld are RCS heatup and cooldown including the Core Flooding System periodic test transient and Decay Heat Removal initiation. All transients are considered as normal operating conditions and are considered in determining thermal stresses and the fatigue usage factor. The fatigue analysis includes a strength reduction factor of two on the weld per ASME Section III. The weld has been designed to withstand a differential pressure of up to 2,250 psi which may occur because of a core flooding line LOCA. A dynamic magnification factor of two was applied to the pressure to account for instantaneous application. Based on these criteria, the average shear stress in the weld yields a safety margin of 1.4. These assumptions and the safety margin are sufficient to insure the structural integrity of the nozzle, restrictor, and weld for all operating and faulted conditions. During the core flooding transient, the maximum ΔP across the nozzle is expected to be

approximately 200 psi. This is a factor of greater than 20 less than the design loading assumptions. During operation of the Decay Heat Removal System, the ΔP loads on the restrictor are insignificant. All major reactor vessel nozzles are installed with full penetration welds. All Control Rod Drive and incore instrument nozzles are installed with partial penetration welds. The gasket leakage tap is installed in each reactor vessel flange with a partial penetration weld.

The reactor closure head flange and the reactor vessel flange are joined by 60 6½-inch diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. Leakoff taps in the annulus between the two O-rings are provided to dispose of leakage. To ensure uniform loading of the closure seal, the studs are hydraulically tensioned.

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, Control Rod Assemblies (CRAs), Axial Power Shaping Rod Assemblies, and incore instrumentation. Guide lugs welded to the inside of the reactor vessel wall limit reactor internals and core to a vertical drop of one-half inch or less and prevent rotation of the core and internals about the vertical axis in the unlikely event of a major core barrel or core support shield failure. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods throughout their full stroke. The internals and the core are supported from the reactor vessel flange. The Control Rod Drive Mechanisms (CRDMs) are supported by the nozzles in the reactor vessel head.

Surveillance specimens, made from appropriately selected specimens of reactor vessel steel, have been retained for core fast neutron irradiation and subsequent study. These specimens will be examined at appropriate intervals to evaluate reactor vessel material Nil Ductility Transition Temperature (NDTT) changes. A full description of the modified surveillance program, including utilization of other B&W reactor plants and data sharing, is provided in Section 4.4.6.

4.2.2.2 <u>Steam Generator</u>

The steam generator general arrangement is shown in Figure 4-5. Principal design data are tabulated in Table 4-4.

The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the steam system. The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at an approximately constant pressure over the power range (see Figure 4-9). Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the upper and lower heads, the tube sheets, and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head.

The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet while the lower part is the feedwater inlet heating zone.

Vent, drain, instrumentation nozzles, and inspection handholes are provided on the shell side of the unit. The instrumentation nozzles provide upper and lower taps for steam generator level control and indication transmitters. Thirteen level transmitters are used to measure steam generator level. Eight of these are safety grade transmitters used for emergency feedwater. The remaining five transmitters are control grade. Of these, four are used for the ICS main feedwater control and one provides the operator with wide range level indication. The safety grade transmitters have taps located at 6", 102", 156", and 500". No taps are shared by the safety grade transmitters, among themselves, or with the non-safety grade transmitters. To prevent interference with level control transmitters LT-2664 and LT-2614, taps for the wide range level indicator transmitters LT-2651 and LT-2601 have been relocated from elevation 0" to elevation 272". In addition, a condensate pot, restrictor, and snubber have been added to each of the upper taps to help stabilize the system. The reactor coolant side has manway openings in both the top and bottom heads. The design of the lower head is such that it drains into the cold legs. Venting of the reactor coolant side of the unit is accomplished by a vent connection on the reactor coolant inlet pipe to each unit.

Retaining rings have been installed in the lower plenum of both steam generators in the area of the cold leg outlet nozzles to allow use of nozzle dams during refueling operations.

Nozzle dams would be used to prevent water from entering the steam generator lower head when the primary piping and refueling canal are full of water, thus allowing concurrent inspection and maintenance of steam generators while other refueling and reactor maintenance operations are taking place. The rings are installed for dam structural support and to hold the dams in place during use.

Nozzle dams would be used only during cold shutdown or refueling operations. During operations above cold shutdown, only the retaining rings are in place. Nozzle dams and retaining rings serve no safety functions.

If required above cold shutdown, emergency feedwater is supplied to the steam generator through an emergency feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all Reactor Coolant Pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet, these are:

A. Feedwater Heating

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration into the downcomer and heats the feedwater to saturation temperature.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section and the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling sections, and the bypass steam flow to the feedwater heating region. A constant minimum downcomer level is maintained up to approximately 22 percent load. From 22 to 100 percent load, the water level increases with steam flow.

B. Nucleate Boiling

The saturated water enters the tube bundle just above the lower tube sheet and the steam water mixture flows upward on the outside of the tubes countercurrent to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until Departure from Nucleate Boiling (DNB) is reached and then film boiling and superheating occurs.

C. Film Boiling

Dry saturated steam is produced in the film boiling region of the tube bundle.

D. Superheated Steam

Saturated steam is raised to final temperature in the superheater region. The amount of surface available for superheat varies inversely with load. As load decreases the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer regions vary. Changes in temperature, pressure, and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the Integrated Control System.

Historical data removed - To review exact wording please refer to Section 4.2.2.2 of the FSAR.

4.2.2.3 Pressurizer

The pressurizer general arrangement is shown in Figure 4-6 and principal design data are tabulated in Table 4-5. Pressurizer usage factors are listed in Table 4-21. Points of stress analysis are shown in Figure 4-14.

The electrically heated pressurizer establishes and maintains the RCS pressure within prescribed limits and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section to maintain RCS pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as RCS pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system (core, pressurizer heaters, and Reactor Coolant Pumps) are interconnected by the reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates approximately 1 gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for determining boric acid concentration. A steam space sampling line provides capability for sampling and/or venting accumulated gases.

During cooldown and after the Decay Heat Removal System is placed in service, the pressurizer can be cooled by circulating water through a connection from the discharge of the Decay Heat Removal pump to the pressurizer spray line. A connection has also been provided from the makeup pump discharge to provide a high pressure spray source during an extended loss of offsite power, when a natural circulation cooldown is required, or following an OTSG tube rupture with Reactor Coolant Pumps off.

Historical data removed - To review exact wording please refer to Section 4.2.2.3 of the FSAR.

4.2.2.4 Reactor Coolant Piping

The general arrangement of the reactor coolant piping is shown in Figures 4-2 and 4-3. Principal design data are tabulated in Table 4-6.

The major piping components in this system are the 28-inch ID cold leg piping from the steam generator to the reactor vessel and the 36-inch ID hot leg piping from the reactor vessel to the steam generator. Also included in this system are the 10-inch surge line and the 2½-inch spray line to the pressurizer. The system piping also incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, level taps, injection, Decay Heat Removal, and temperature element connections, there is a flow meter section in each 36-inch line to the steam generators to provide a means of determining the flow in each loop.

The 28- and 36-inch piping is carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casing and the 28-inch carbon steel lines. Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic stainless steel. All piping connections in the RCS are welded except for the flanged connections on the pressurizer for the relief valves.

Additional level taps have been field installed in the RCS hot leg piping. The taps consist of an Inconel sleeve and an Inconel nozzle. The sleeves have been roll expanded in the carbon steel penetration and seal welded to the hot leg pipe stainless steel cladding. The Inconel nozzles have been installed by using a partial penetration weld on the outside of the hot leg piping. In cases where the nozzle has been replaced after initial installation, the weld on the outside of the hot leg does not attach to the Inconel sleeve. This repair method results in a small gap between the end of the Inconel sleeve and the hot leg level tap nozzle.

Level taps previously field installed in the hot leg elbow at each steam generator inlet were replaced during steam generator replacement. Each steam generator hot leg inlet elbow has a high point vent/nitrogen supply nozzle and a level tap nozzle fabricated from Alloy 690 that were attached to the elbow with full penetration welds.

Vent and drain lines are shown on the system diagram, Figure 4-1. They are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer.

The reactor vessel cannot be drained below the reactor outlet nozzles using these drain lines. Each vent and drain line contains two manual valves in series, or one manual valve followed by a socket welded cap, or other acceptable system boundary closure means. (Note: A threaded cap is not permitted for this purpose, per the ASME Codes.) Drain lines are routed to reactor building floor drains or to the auxiliary building equipment drain tank, via temporary hoses. Some vents are routed to a vent header connected to the quench tank. Infrequently used vent lines are not routed to the vent header.

B&W has eliminated the use of wrought, furnace sensitized stainless steel in the component parts of the Reactor Coolant Pressure Boundary. Whenever stainless steel components were welded to ferritic materials, Inconel buttering of the ferritic material followed by a stress-relief heat treatment preceded joining of the two components with Inconel weld metal. Due to issues with primary water stress corrosion cracking (PWSCC) in Inconel (Alloy 600) material, the butt welded surge line connection to the “A” hot leg and connections to pressurizer piping nozzles were structurally overlayed with Alloy 690 material during 1R20. The butt welded Decay Heat piping connection to the “A” hot leg Decay Heat nozzle was structurally overlayed with Alloy 690 material during 1R21. Alloy 690 material has been shown to be highly resistant to the effects of PWSCC.

Welding of stainless steel components was accomplished using low heat input procedures. Maximum values for submerged arc welding and manual processes are 85 and 60 kilojoules per inch, respectively. Approved procedures for welding were used and periodic checks were made by quality control personnel to assure that the proper procedures were followed.

Core structural load bearing members were heat treated at 850 °F for 48 hours so that dimensional stability was maintained. Stainless steel, held at 850 °F and 950 °F for 24 hours, was subjected to the acidified copper sulfate test as specified in ASTM A-393. Subsequent examination did not reveal any fissures or cracks, thereby indicating freedom from intergranular attack. Based on the above facts, the core structural members, heat treated as described, are not sensitized.

Stainless steel types 304 and 316 with nitrogen added for strength enhancement were not specified and not utilized in construction.

Therefore, any susceptibility to stress corrosion cracking under sensitized conditions, which would be due to nitrogen addition, has been eliminated.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes and the surge line and spray line nozzles on the pressurizer.

There are temperature sensing elements attached to the top and bottom portions of the pressurizer auxiliary spray piping downstream of check valve DH-12. The temperature sensing elements will provide the information necessary to determine if thermal stratification is occurring on the pressurizer auxiliary spray line and associated piping nozzle due to the leakage of the upstream auxiliary spray line isolation valve from the Makeup System into the spray line.

4.2.2.5 Reactor Coolant Pumps

The Reactor Coolant Pumps are single stage, single suction, constant speed, vertical centrifugal pumps. A view of the pump is shown in Figure 4-7 and the principal design parameters are listed in Table 4-7. The Reactor Coolant Pump performance characteristics are shown in Figure 4-8.

The pump casing consists of a bottom suction inlet passage which delivers the reactor coolant to the pump impeller, a multivaned diffuser, and a collecting scroll which directs the fluid out through a horizontal discharge nozzle. A water lubricated radial hydrostatic bearing is located in the pump casing just above the main impeller. The pump casing is welded into the piping system and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

Each pump has a separate, single speed, top-mounted electric drive motor, which is connected to the pump by a removable shaft coupling. A driver mount is used as the transition piece between the motor and the pump which, in conjunction with the removable shaft coupling, allows pump seal replacement without removal of the drive motor.

Shaft sealing is accomplished in the upper part of the pump housing using a removable seal cartridge which contains three mechanical seals in series. A standpipe is mounted above the top seal to collect the last seals' lubricating and leakage water. The standpipe features an overflow drain to a closed collection system and limits leakage to the reactor building in the event of excessive deterioration of the mechanical seal performance. Seal leakage flow measurement instrumentation provides an indication of excessive leakage. Water for lubricating and cooling the seals is injected below the bottom seal at a pressure slightly above reactor system pressure. Part of the seal flow passes into the pump casing through a radial restriction bushing. The remainder flows up through the seal cartridge where it removes heat generated by the seal faces and is then returned to the seal water supply system. This controlled bleedoff flow passes through pressure breakdown orifices which provide the differential pressure between the three stages.

Intermediate cooling water is also furnished to the pump as a backup to the Seal Injection Water System. The cooling water flows to a heat exchanger mounted integral to the pump. The heat exchanger will provide, upon loss of Seal Injection Water, enough cooling capacity to prevent excessive heating of the mechanical seals. The upper drain chamber, which features an overflow drain to a closed disposal system, further prevents leakage to the reactor building if excessive deterioration of the mechanical seal performance should occur.

4.2.2.6 Reactor Coolant Pump Motors

The Reactor Coolant Pump motors are large, vertical, squirrel cage induction machines. The motors have flywheels to increase the rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost.

Reactor coolant pump motors are manufactured by Allis-Chalmers (A-C) and Jeumont Industrie (JI). In the A-C motor, two flywheel assemblies are provided, a large assembly at the upper end of the rotor and a smaller flywheel at the lower end of the motor. The JI motor uses one large flywheel mounted at the top of the motor above the upper bearing assembly. An anti-reverse device is included to prevent back rotation, thereby reducing starting time and rotor heating.

Historical data removed - To review exact wording please refer to Section 4.2.2.6 of the FSAR.

The motors are totally enclosed with air-to-water heat exchangers so as to provide a closed circuit cooling air flow through the motor. Radial bearings are of the pivoted pad type and of segmented or split construction. The thrust bearing is a double-acting Kingsbury type of self-equalizing pivoted pad construction designed to carry the full thrust of the pump. A High Pressure Oil System with separate pumps is provided with each motor to establish an oil film on the thrust bearings before starting. Once started, the bearings carry their own oil film and the motor provides its own oil circulation through the oil cooler.

4.2.2.7 Reactor Coolant Equipment Insulation

The insulation on the reactor vessel is shown in Figure 4-4. The insulation units on all components are designed for ease of removal and installation in such areas as seam welds, nozzles, and bolted closures.

Originally, reflective metallic insulation was installed on the reactor vessel, pressurizer, steam generators, primary coolant pumps, and primary coolant piping. These original insulation units permitted free drainage of any condensate or moisture from within the insulation unit. Subsequently, some of this original insulation was replaced with fiberglass insulation materials. These additions have been assessed with respect to fire protection concerns (oil leakage or misting) and free drainage and are considered acceptable.

4.2.3 SYSTEM PARAMETERS

4.2.3.1 Flow

The RCS is designed on the basis of 176,000 gpm flow rate in each heat transport loop. Actual Reactor Coolant flow rate per loop is determined periodically as required by Technical Specifications.

Reactor coolant flow rate is measured for each heat transport loop by a flow tube welded into the reactor outlet pipe. The power/flow monitor of the Reactor Protective System (RPS) utilizes this flow measurement to prevent reactor power from exceeding a permissible level for the measured flow. This is discussed in further detail in Section 7.3.2.

4.2.3.2 Temperatures

RCS system nominal temperatures as a function of power are shown in Figure 4-9. The system is controlled to a constant average temperature throughout the power range from approximately 22 to 100 percent full power. The average system temperature is decreased between approximately 22 and 0 percent of full power to the saturation temperature at approximately 910 psia.

4.2.3.3 Heatup

All RCS components are structurally designed for a continuous heatup rate of 100 °F per hour. However, reduced heatup rates may be utilized to reduce the thermal stresses in the RCS components and thereby minimize the restrictions that the Technical Specifications RCS pressure/temperature limits could impose on plant operations.

4.2.3.4 Cooldown

All RCS components are structurally designed for an initial cooldown rate of 100 °F per hour. However, reduced cooldown rates may be utilized to reduce the thermal stresses in the RCS components and thereby minimize the restrictions that the Technical Specification RCS pressure/temperature limits could impose on plant operations. System cooldown to less than 280 °F is normally accomplished by use of the steam generators and by bypassing steam to the condenser with the turbine bypass system. The Decay Heat Removal System provides the heat removal for system cooldown below approximately 280 °F.

4.2.3.5 Volume Control

4.2.3.5.1 Letdown

Reactor coolant is removed from the RCS as letdown to the Makeup and Purification System. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

4.2.3.5.2 Makeup

To maintain a constant pressurizer water level, total makeup to the RCS must equal that which is letdown from the system. Total makeup consists of the Seal Injection Water through the Reactor Coolant Pump shaft seals and makeup returned to the system through the reactor coolant pressurizer level control valve. The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady state conditions.

4.2.3.6 Chemical Control

Control of the reactor coolant chemistry is a function of the Chemical Addition and Sampling System. Sampling lines from the pressurizer and the letdown line of the Makeup and Purification System provide samples of the reactor coolant for chemical analysis. All chemical addition is made from the Chemical Addition and Sampling System to the Makeup and Purification System. See Chapter 9 for detailed information concerning the Chemical Addition and Sampling System, and the Makeup and Purification System.

4.2.3.6.1 Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the boric acid addition tank and is transported to the RCS in the same manner as described above for chemical addition. All bleed and feed operations for changing

the boric acid concentration of the reactor coolant are made between the Makeup and Purification System and the Liquid Waste System. Chapters 9 and 11 contain detailed information concerning these two systems.

4.2.3.6.2 Lithium

Lithium hydroxide is used in the RCS to maintain an alkaline condition during operation.

The lithium hydroxide concentration is controlled in the Reactor Coolant System by use of a coordinated lithium/boron control program in accordance with B&W recommendations. Under these conditions, the generation of activated corrosion products within the Reactor Coolant System is minimized. The specified range of lithium for a fuel cycle during normal "steady state" power operations is given in Table 9-4.

4.2.3.6.3 Zinc

Zinc injection for radiation source term reduction generally involves the addition of small concentrations of zinc into the plant RCS. The zinc is added in the form of zinc acetate, and depleted zinc (< 4% Zn-64) is used to maximize the dose reduction benefits. Limits of zinc in the primary water chemistry are included in Table 9-4.

4.2.3.6.4 Water Quality

The reactor coolant water chemistry specifications, listed in Table 9-4, have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification of the letdown stream of reactor coolant in the letdown filter and purification demineralizer of the Makeup and Purification System. A hydrogen overpressure is maintained in the makeup tank to insure that a predetermined amount of dissolved hydrogen remains in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water. High purity hydrogen is used for the makeup tank overpressure because of its low ^{40}Ar content. Higher ^{40}Ar concentrations found in low purity hydrogen cause high radiation levels at the tank and associated piping, because ^{40}Ar becomes core-irradiated to highly radioactive ^{41}Ar and is returned to the tank via the recirculation line. Hydrogen bottles for makeup overpressurization are housed in the gas storage room in a portion of the Unit 1 turbine auxiliary building.

In addition to water chemistry specifications and control, the reactor coolant is regularly sampled and analyzed during plant operation for the specific radioactivities listed below:

Analysis	Frequency
Gross Activity	3 times/wk; at least every third day*
Gamma Isotopic Analysis	Bi-weekly
Gross Radioiodine	Weekly
Dissolved Gases	Weekly

* This frequency also applies to the water chemistry and boron concentration analyses.

These analyses, limits, and procedures are discussed further in Technical Specifications 3.4.12 and TRM 3.4.6.

4.2.3.7 RCS Level and Inventory

To assist in detecting the effects of low reactor coolant level and inadequate core cooling, instrumentation is installed to measure primary coolant saturation conditions and level in the reactor vessel and hot legs. The Inadequate Core Cooling (ICC) Monitoring System measures and displays information on core exit thermocouple conditions including subcooled margin and on vessel and hot leg level.

An Inadequate Core Cooling Monitoring and Display System (ICCMDS) is provided to perform all of the required functions of the ICC Monitoring System, including display and annunciation of required system variables. The ICCMDS consists of two (2) functionally and physically independent processing and display cabinets designed to Class 1E standards and one (1) non-1E dual channel, remote Mimic Display indicator. Each of the cabinets contains all necessary data acquisition/processing system (DAS) hardware and software required for proper system operation. The Mimic Display, with its associated devices, is designed to accommodate two (2) instrumentation display channels.

The purpose of the ICCMDS is to provide control room operators with an integrated and unambiguous indication which provides:

- advance warning and annunciation of approach to ICC,
- tracking of coolant inventory during an ICC event,
- tracking of coolant inventory during accident recovery, and
- coolant temperature measurements at core exit.

In addition to the above functions, the ICCMDS provides for:

- RCS level indication under the following conditions; during normal decay heat removal during fill and vent and drain operations during RCS maintenance
- Calculation of subcooling margin based on CET inputs
- Indication of reactor vessel head fluid temperature and level to aid during natural circulation cooldown operation

In order to accomplish this, the ICCMDS monitors a variety of temperatures and pressures from various sensors within the reactor vessel and hot legs to provide input to calculations performed by ICCMDS system software. Specific plant variables are monitored and conditioned within each channel of ICCMDS. Required information is then made available for display in the control room on the SPDS, the PMS computer, a dual channel ICCMDS Mimic Display, and a safety grade local display unit provided on each ICCMDS cabinet. The following variables are monitored:

a. Core Exit Thermocouple (CET) Monitoring Subsystem

   CETs are monitored within the reactor vessel in order to display temperature at the core exit. Each channel of ICCMDS receives and monitors twelve (12) CET (Type K) inputs, and converts these inputs to engineering units representing the appropriate linearized temperature range. Individual as well as average CET temperature conditions are provided via ICCMDS.

b. Subcooling Margin Indication

Each channel of ICCMDS receives and conditions twelve (12) CET inputs and one (1) RCS pressure input. Utilizing the indicated inputs, i.e. average CET temperature, and saturation data extracted from steam tables, each channel of ICCMDS calculates subcooling margin and superheat temperature. In addition to the above display indications, margin to saturation conditions from each channel of ICCMDS is continuously displayed on a 2-channel digital indicator located on the main control board.

c. Hot Leg Level Monitoring Subsystem (HLLMS)

Each channel of ICCMDS receives and conditions five (5) differential pressure inputs, five (5) 100 ohm Plat, RTD Ref. leg temperature inputs, two (2) process temperature inputs, and one (1) process pressure input. Each channel also converts the above inputs to appropriate engineering units representing applicable component operating or electrical ranges.

d. Reactor Vessel Level Monitoring System (RVLMS)

Each channel of ICCMDS receives and conditions nine (9) heated differential thermocouple (Type K) inputs, one (1) absolute thermocouple (Type K) input, one (1) heater current input, and one (1) input that provides status of the four (4) reactor coolant pumps. Each channel also converts the above inputs to appropriate engineering units representing the applicable component operating or electrical ranges.

Reactor Coolant Pump current is available on the SPDS. The current indication provides information on the pump and motor integrity and on RCS inventory conditions.

4.2.3.8 Leak Detection

The RCS is located within the secondary shielding and is normally inaccessible during reactor operation. Any coolant leakage to the reactor building atmosphere will be in the form of fluid and vapor. The fluid will drain to the reactor building sump and the vapor will be condensed in the reactor building coolers and also reach the sump via a drain line from the coolers.

There are three methods of detecting reactor coolant leakage. These methods are as follows:

A. Sump Level

B. Inventory Balance

C. Radiation Monitoring

The above methods are described in the following paragraphs.

A. Sump Level

Changes in reactor building sump water level are an indication of RCS leakage. The Reactor Building Sump Level Detection System consists of two separate instrumentation loops. Sump level is detected by LE-1405B and LT-1405, installed in the sump. LE-1405B is designed to survive total submergence. Each loop has a control room indicator and a SPDS point. LT-1405 provides input to a control room annunciator alarm. The reactor building sump capacity varies with height, but averages about 66 gallons per inch. A 1 gpm leak would be detected in less than one hour. Standard factory calibration and field installation checks have been run to insure sensitivity and operability.

The Reactor Building Water Level Detection System is a redundant Q-level system indicating water levels up to 144 inches, which is the equivalent of 600,000 gallons. The system design meets the requirement of NUREG-0578 and is based on Regulatory Guide 1.97, Revision 1, and on IEEE 323-1974 and IEEE 344-1975. (0CAN018105).

B. Inventory Balance

Makeup to the RCS as a result of leakage is initially supplied from the makeup tank inventory. Monitoring of the makeup tank level will provide a direct indication of reactor coolant leakage. Each increment of the makeup tank level represents two inches of tank height. Each inch of tank height is equivalent to approximately 30.86 gallons of reactor coolant. With the reactor coolant average temperature and pressurizer water level held constant, a 1 gpm leak can be detected within 1.1 hours. Standard factory calibration and field installation checks have been run to insure sensitivity and operability.

C. Radiation Monitoring

This is the primary method of preventing abnormal radiation buildup within the reactor building. Such buildup could potentially result in serious decontamination difficulties and potentially abnormal releases to the environment.

Changes in the reactor coolant leakage rate in the reactor building may cause changes in the control room indication of the reactor building atmosphere gas radioactivities. The minimum detectable concentration of the reactor building radioactive gas detector is 5.5 x 10^{-7} μC/cc of ^{133}Xe in a 2.5 mrem/hr background. For a leak rate of 1 gpm, the following response times are obtained:

For 1% failed fuel, the response time	=	2 seconds
For 0.5% failed fuel, the response time	=	4 seconds
For 0.1% failed fuel, the response time	=	20 seconds
For 0.01% failed fuel, the response time	=	200 seconds
For tramp uranium (no failed fuel), response time	=	17.5 hours.

This assumes very little existing radioactivity in the reactor building atmosphere before the leak starts, that the radiation monitor is well shielded (background dose rate at the detector of 2.5 mrem/hr), that the gas in the leakage is instantaneously released and mixed in the reactor building atmosphere, and that there is no appreciable removal of the gas from the reactor building. If the background rate is significantly higher due to airborne gases or any other source (such as highly radioactive components nearby) then the sensitivity of the radiation monitor will suffer. Also, if there is a significant reactor building purge rate or if the reactor building mixing air flow rates are not high, the concentration at the monitor will not increase as quickly. These response times also do not account for radioactive decay, travel time in the sample tubing nor fill time of the sample chamber.

Additional analysis, which does not assume instantaneous mixing in the reactor building, has shown that the detector can "see" a 1 gpm leak within one hour for 0.001% failed fuel. The analysis assumes the Xe is completely released from the failed fuel to the RCS, all Xe in the leaking RCS fluid is released to the reactor building, uniform mixing occurs within 10 minutes, RCS Purification Systems provide no significant removal of noble gas fission products, background activity for the monitor is 150 mrem/hr, 50 minutes of leakage occurs, the initial reactor building activity has no impact, and activity occurs solely from ^{133}Xe.

The radioactive gas detector will be calibrated at the factory prior to shipment and its sensitivity will be determined by the following method. Two different concentrations of ^{133}Xe will be introduced into the volume of the sampler and the resultant average gross count rates recorded. A 2.5 mrem/hr background will be established using a ^{60}Co source and the gross counting rate recorded. From this information the background at 1 MeV will be calculated. Then, based on Minimum Detectable Concentration = ($2.6 \times \sqrt{\mathrm{BKG}}$ @ 1.0 MeV / SEN_{ave}), the sensitivity will be calculated. At the time of these tests, the average count rate of a solenoid operated check source will be established. Then during normal plant operation, the solenoid operated check source will be introduced into the sampler chamber by means of a control switch located in the main control room, thereby checking the sensitivity of the detector. This test will be performed at time intervals consistent with good preventive maintenance practice.

As noted above, this detector is least sensitive when the plant is operating with no fuel failures. However, in this case, the radiation buildup within the reactor building is least serious. Adequate detection of leakage without fuel failure will be accomplished by the methods of sump level and inventory balance.

An airborne particulate activity detector is also provided for additional radiation monitoring of the air in the reactor building. This detector is located outside the reactor building, and is used in conjunction with an air particulate sample line. This configuration prevents detector exposure to high background radiation, high temperatures, and hot moist air, minimizing detector failures. In addition, it provides sample collection capability from the occupied zone of the reactor building without requiring reactor building entry.

4.2.4 PRESSURE CONTROL AND PROTECTION

Normal RCS pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray and heaters. The system is protected against overpressure by RPS circuits such as the high pressure trip and by pressurizer safety valves located on the top head of the pressurizer. The Diverse Reactor Overpressure Prevention System (DROPS) also provides RCS overpressure protection in compliance with 10CFR50.62 (refer to Section 7.2.4 for the DROPS). The schematic arrangement of the safety valves is shown in Figure 4-1. Since all sources of heat in the system, i.e., core, Reactor Coolant Pumps, and pressurizer heaters, are interconnected by the reactor coolant piping with no intervening isolation valves, all safety valves are located on the pressurizer. RCS pressure settings and safety valve capacities are listed in Table 4-1.

The pressurizer water level indicators, pressurizer electromatic relief and block valves, and 126 kW of pressurizer heaters are supplied by emergency power. The block valve power is green AC, 480 volts and electromatic relief valve power is DC - red.

To provide additional information during abnormal events, an Acoustical Valve Monitoring System is installed to provide direct indication of relief valve position. Redundant sensors are mounted on all three relief valves and the control room annunciator panel will alarm when any relief is opened, in addition to providing open/closed position indication for all three valves. This system also provides positive position indication lights on C04 for the electromatic relief valve.

4.2.4.1 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The primary function of the code safety valves is to protect the RCS pressure boundary. The nozzles are designed to transfer full discharge loads to the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or setpoint testing.

4.2.4.2 Pressurizer Electromatic Relief Valve (ERV)

A separate nozzle on the top head of the pressurizer is used for the pressurizer electromatic relief valve. A motor-operated block valve is located between the pressurizer flange and the electromatic relief valve. The remote control of the ERV block valve can be isolated by a local selector switch to allow manual operation outside the control room per NFPA 805 requirements. This nozzle is designed to transfer full discharge load to the pressurizer.

The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure setpoints. The relief valve is equipped with a dual setpoint capability to preclude exceeding vessel pressurization limits. Flanged inlet and outlet connections provide ease of removal for maintenance purposes. Isolation capability is provided to permit local control of the ERV block valve in the event of an emergency situation in the control room. (0CAN118203)

The primary function of the pressurizer electromatic relief valve is to control operational transients and to provide low temperature overpressure protection. The ERV setpoint is 2,450 psig during operation. The analytical ERV setpoint of 508 psig, in conjunction with preserving an appropriate pressurizer level, provides low temperature overpressure protection.

4.2.4.3 Pressurizer Spray

The pressurizer spray line originates at the discharge of a Reactor Coolant Pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by an electric motor operated valve using on-off control in response to the opening and closing pressure setpoints. An electric motor operated valve in series with the spray valve provides a backup means of securing flow in the event the spray valve should stick open. In addition, the Non-Nuclear Instrumentation (NNI) System cabinet control circuitry was modified to prevent the loss of either the + 24 volt or - 24 volt buses from causing the spray valve to open and latch open until it is closed manually. Upon loss of ± 24-volt power, the modified circuitry minimizes issuance of an "open" command and issues a "close" command.

4.2.4.4 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during RCS heatup from a cooled down condition, and restore system pressure following transients. The heaters are arranged in 14 groups and are controlled by the pressure controller. Two groups utilize modulating control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the setpoint. On-off control is used for the remaining 12 groups. A low level interlock prevents the heaters from being energized with the heaters uncovered.

The pressurizer proportional heater feeders, RUB14 and RUB15, are furnished with current transducers providing an analog value of the circuit current to the SPDS. The magnitude of the current indicates operational status and integrity of the heater. Each diesel generator emergency bus supplies emergency power to 168 kW of pressurizer heaters. Two separate sets of heaters totaling 84 kW each are supplied by the diesel generators. The remaining 84 kW is powered from a swing bus which can be supplied by either diesel generator. This insures emergency power redundancy to sufficient heater capacity to establish and maintain pressure control at hot standby during a loss of offsite power.

The total heater design capacity of the pressurizer is 1,638 kW per Table 4-5. This 1638 kW of pressurizer heater capacity consists of three (3) separate heater bundles (upper, middle, and lower) each containing thirty-nine (39) individual non-replaceable heater elements rated nominal 14 kW per element.

Babcock & Wilcox Co. Instruction Manual Unit 1 Pressurizer Vessel, document number 620-0008-59 dated September 1975 (ANO Vendor Document TDB015 0210), states the following regarding the heater bundles:

> Each of the three heater bundles contains 39 heaters connected in a series of three heaters per circuit to permit using selected heaters to obtain desired water temperatures without energizing the whole bundle. The three bundles contain a total of 117 heaters. A heater bundle should be considered for replacement when 19-20 elements are not functional. However, operation with a greater outage is not anticipated to cause any damage, and will only increase the amount of time to recover after a pressure drop transient, or to heat up after a shutdown.

All of the pressurizer heaters are original ANO Unit 1 plant equipment. At the beginning of Cycle 26, the status of the pressurizer heater bundles was as follows:

M-310 (Upper Heater Bundle): 12 non-functional heaters

M-309 (Middle Heater Bundle): 6 non-functional heaters

M-308 (Lower Heater Bundle): 19 non-functional heaters

As heaters become non-functional during the life of the heater bundle, they are removed from the circuit when identified to maintain functionality of the remaining heaters in the circuit; therefore, each circuit may contain three, two, one, or zero electric heater elements.

4.2.4.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank, which condenses and collects the relief valve effluent. This is shown schematically in Figure 4-1. The tank contents are cooled by dilution with condensate.

The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank is vented to the Gaseous Radioactive Waste Disposal System (Chapter 11).

4.2.4.6 Cooldown

Reduction of pressure during RCS cooldown is accomplished by the pressurizer spray provided by the Reactor Coolant Pump. Below a system temperature of approximately 280 °F, the Decay Heat Removal System is used for system heat removal and the steam generators and Reactor Coolant Pumps are removed from service. During this period, spray flow is provided by a branch line from Decay Heat Removal line to the pressurizer spray line for further pressure reduction or complete depressurization of the RCS. In the event of branch line failure, continued plant depressurization and cooldown could be accomplished three alternate ways.

A. Discharge steam through the electromatic relief valve to the reactor coolant quench tank.

B. Drain pressurizer water through the sample line, thereby causing reactor coolant to flow through the surge line.

C. Perform a feed and bleed operation with the Makeup System, causing reactor coolant to flow through the surge line as the pressurizer level is increased.

Thus, the only effect of a failure in the auxiliary spray line would be increased cooldown time.

4.2.4.7 Control of Non-condensible Gases

A sample line from the pressurizer steam space to the Chemical Addition and Sampling Systems permits detection of non-condensible gases in the steam space. This sample line also permits a bleeding operation from the vapor space to the letdown line of the Makeup and Purification System to transport accumulated non-condensible gases in the pressurizer to the makeup tank.

Vents have been installed on the reactor head, both hot legs at the highest point, and the pressurizer specifically to allow removal of non-condensible gases from the reactor and primary system after a LOCA. These vents are operated from the control room by control switches located on C486-1&2. The system is designed to assure opening and closing of the vents with a single failure of a valve, switch, or power supply. During normal operation these vents are closed. The hot leg and reactor high point vents discharge to the reactor building atmosphere. The pressurizer high point vent discharges to the quench tank.

4.2.5 INTERCONNECTED SYSTEMS

4.2.5.1 Decay Heat Removal

The Decay Heat Removal System (DHR) provides the capability for cooling the RCS below about 280 °F during plant cooldown. During this mode of operation, coolant is drawn from the RCS through a nozzle on the reactor outlet pipe, circulated through the Decay Heat Removal coolers by the decay heat removal pumps, and then injected back in to the RCS through two nozzles on the reactor vessel into the inlet side of the core. The heat received by this system is rejected to the Service Water System. These two systems are described in Chapter 9.

The DHR also performs an emergency low pressure injection function for a LOCA and provides long-term emergency core cooling. This is described in Chapter 6.

To assure the DHR is isolated from the RCS when the RCS pressure exceeds DHR design pressure, setpoints have been developed for DHR isolation valve closure consistent with the DHR relief valve opening setpoint. These setpoints are given in Technical Specification Surveillance SR 3.4.14.3, along with the testing and calibration requirements and testing frequencies.

4.2.5.2 Makeup and Purification

The Makeup and Purification System controls the RCS coolant inventory, provides the seal water for the Reactor Coolant Pumps, and recirculates RCS letdown for water quality maintenance and reactor coolant boric acid concentration control. Letdown of reactor coolant is through a nozzle on the outlet coolant pipe from one steam generator. The reactor coolant which is letdown is returned to the RCS through a nozzle in a different heat transport loop from the heat transport loop containing the letdown line.

The Makeup and Purification System utilizes four injection nozzles in carrying out the high pressure emergency injection function after a LOCA. This is described in Section 6.1.2.1.1.

4.2.5.3 Core Flooding System

The Core Flooding System floods the core in the event of a LOCA. The Decay Heat Removal and core flooding lines tie together and connect to the same nozzles on the reactor vessel. To increase the operating safety margin, the motor operated vent isolation valve and the motor operated isolation valve to the RCS for each Core Flooding Tank are supplied from emergency power. The Core Flooding System is described in Section 6.1.2.1.3.

4.2.5.4 Secondary System

The principal method of Decay Heat Removal when the reactor coolant temperature is above 280 °F is by use of the Steam and Power Conversion System. This system is discussed in detail in Chapter 10.

The turbine bypass system routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. When the condenser is unavailable, the atmospheric dump valves are utilized. Overpressure protection for the secondary side of the steam generators is provided by the Turbine Bypass System and by safety valves mounted on the main steam lines outside of the reactor building. The Emergency Feedwater System will supply water to the steam generators in the event that the Main Feedwater System is inoperative. The physical layout of the RCS supports natural circulation of reactor coolant to insure adequate core cooling following a loss of all Reactor Coolant Pumps.

4.2.6 COMPONENT FOUNDATIONS AND SUPPORTS

The supports for all major components listed in this section are analyzed in detail to insure adequate structural integrity for its intended function during normal operating, seismic, and accident conditions. The design of the principal RCS component supports was performed in close cooperation between the Nuclear Steam Supply System (NSSS) supplier and the architect-engineer.

Pursuant to the source of loading, stresses and motions at significant locations are computed and compared to applicable criteria. The design bases of the principal RCS support, including the foundations and anchor bolts, snubbers, and guides are covered in Section 5.1. Discussion on pipe restraints design requirements is presented in Section 4.2.6.6. The seismic analysis of the system used an appropriate model of lumped masses and spring constants and is in accordance with Section 5.1.

As part of the OTSG replacement activities, a structural analysis of the ANO reactor coolant system was performed. A three-dimensional static (thermal expansion, dead load) and dynamic (seismic, pipe rupture) analysis was performed on the reactor coolant system to determine piping and component nozzle stresses. An idealized mathematical model for both loops (A and B) consisting of distributed and lumped masses connected by elastic members has been employed. The model includes the reactor vessel, steam generators, the pressurizer, all four coolant pumps with associated piping, and the secondary shield wall with any attached coolant system restraints or supports. The pipe rupture analyses relied upon Leak-Before-Break methodology to determine the appropriate pipe rupture locations, which included guillotine breaks in the decay heat line, the surge line, the core flood line, the main feedwater line, and the main steam line. Theses analyses demonstrated that the ANO-1 reactor coolant system piping, component nozzles, and component supports were adequate to withstand the induced loadings, when considering the current design basis allowable stresses (Reference 21).

4.2.6.1 Reactor Vessel

The reactor vessel is bolted to a reinforced concrete foundation designed to support and position the vessel and to withstand the forces imposed on it by a combination of loads, including the weight of vessel and internals, thermal expansion of the piping, Design Basis Earthquake (DBE), and dynamic load following reactor trip.

The foundation, in addition, restrains the vessel during the combined forces imposed by the circumferential rupture of 36-inch reactor outlet line and a simultaneous Maximum Hypothetical Earthquake.

The vessel foundation, further, is designed to provide accessibility for the installation and later inspection of incore instrumentation, piping, and nozzles; to contain ductwork and vent space for cooling air to remove heat losses from the vessel insulation; and to provide a sump and drainage line for leak detection.

4.2.6.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight lugs welded to the side of the vessel.

The foundation and supports are designed to withstand the loads imposed by thermal expansion of the pressurizer, the weight of the pressurizer including its contents and attached piping, relief valve reaction forces, and forces imposed by the Design Basis Earthquake. In addition, the foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10-inch surge line coupled with the Maximum Hypothetical Earthquake (MHE).

The foundation is also designed to permit accessibility to pressurizer surfaces for inspection.

4.2.6.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators and feedwater piping filled with water, the attached reactor coolant piping also filled with water, and steam lines under the DBE and the MHE.

The foundation designed to restrain the steam generator under the combined forces due to a circumferential rupture of the surge line, the decay heat line, or the core flood line. Lateral forces imposed on the generator by the rupture of these lines are transferred to the shielding walls by a support structure located near the top of the generator.

4.2.6.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the DBE and the MHE.

4.2.6.5 Pump and Motor

The Reactor Coolant Pump casing, internals, and motor weight are supported by the 28-inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping can support the coolant pump and motor without the hangers.

4.2.6.6 Prevention of Damage from Pipe Rupture Effects

To prevent damage of piping and systems due to pipe whip after a postulated rupture, the following design methods were employed.

A. Physical separation between critical lines and equipment has been effected whenever possible by routing the piping so that a ruptured line cannot reach and damage a critical line or vital piece of equipment including redundant Engineered Safeguards Systems. Generally, separation is the preferred method to protect vital components.

B. Routing of pressurized piping adjacent to walls, floors, ceilings, columns, abutments, and foundations was employed to limit movement.

C. In the original design, pipe restraints were used to prevent whip damage when the pipe whip could not be controlled by the other means enumerated above. The restraints were located at points where the displacement of the ruptured pipe could damage critical piping and system components and do not interfere with normal supporting elements which are designed for other conditions. As justified by B&W Topical Report BAW-1847, which provided a leak-before-break evaluation of RCS piping 28" and larger, pipe whip restraints are no longer required on the main RCS piping and may be modified to be inactive or removed as needed to facilitate maintenance or other activities. These restraints are not reinstalled once they have been removed.

The pipe protection design criteria are based on the following general requirements:

A. Prevention of simultaneous pipe break in the Reactor Coolant Pressure Boundary and the secondary pressure boundary.

B. Prevention of damage to the reactor building liner plate leaktight integrity during LOCA.

C. A failure of the Reactor Coolant Pressure Boundary shall not reduce the minimum performance capability of the Engineered Safeguards Systems below that specified in the safety analysis.

These requirements are expanded in the following table.

REQUIREMENTS FOR PREVENTION OF DAMAGE FROM PIPE RUPTURE EFFECTS

	Categories				
Reactor Coolant Pressure Boundary	A	B	C	D	E
Reactor Coolant Loop Piping		X	X	X	X
Emergency Core Cooling System Piping		X	X	X	X
Other Piping		X	X	X	
Secondary Pressure Boundary	A	B	C	D	E
Main Steam Line Piping	X				
Feedwater Piping	X				
Other Piping	X				

	Categories				
	A	B	C	D	E
Other Piping in Reactor Building	X				

LEGEND: Failure in this piping must not damage:

A. Piping in the Reactor Coolant Pressure Boundary

B. Reactor Building Liner Plate

C. Engineered Safeguards Systems

D. Piping in the Secondary Pressure Boundary

E. Parallel Redundant Piping

The following requirements are also satisfied.

A. Each line penetrating the containment and containing high pressure and high temperature fluids (main steam and feedwater) shall have structural restraints such that: the fracture of any one line will not jeopardize any other line whose fracture would result in a loss of coolant accident and electrical penetrations shall be protected or separated such that they would not be damaged by the whipping of any pipe, including the main steam lines.

B. A lateral support system near the steam generator upper tube-sheet elevation shall resist lateral loads, including those resulting from seismic forces, pipe rupture, and thermal expansion. The RCS hot leg and cold leg pipe rupture loading has been eliminated through credit for leak-before-break.

   The reactor building liner plate is protected during a LOCA as described in the table above. In addition, although it is not a design requirement, significant protection of the liner plate is afforded due to pipe restraints provided to protect the Reactor Coolant Pressure Boundary and electrical penetrations. Further protection is not required since reactor building liner damage by a main steam line rupture alone would not have a radiological environmental effect as great as that associated with a main steam line failure outside the reactor building (see Section 14.2.2.1). Hence, only a negligible activity release would result from the failure causing breaching of the reactor building liner plate.

In the design of the pipe restraints, the following design criteria were used.

A. Only piping systems operating at 300 psig or greater were considered.

B. Two basic types of pipe ruptures were generally considered to occur at any location within the Reactor Coolant Pressure Boundary:

   1. The guillotine type failure with complete pipe severance.

   2. The elongated slot type failure with the area of the slot equal to the cross-sectional area of the pipe.

C. Rupture of one inch lines or smaller were considered not to cause damage to other larger piping or systems as is the generally accepted design practice. It was also assumed that piping in this range does not possess sufficient energy to significantly damage the reactor building liner plate.

D. Pipe restraints are designed to absorb piping rupture reaction loads. For each type, the restraint design force was established using an appropriate dynamic consideration with the equivalent static load. In the structural design of the pipe restraints, the maximum flexural stress in each pipe restraint element was limited to $0.9f_y$ for the governing loading combination listed in Section 5.1. Spacing of the pipe restraints on the piping precludes the formation of a plastic hinge anywhere in the unsupported span. The forces are factored and combined in accordance with the factored loading combinations given in Section 5.1. The methods also described in Section 5.1 are used to size the individual members in the structural system.

4.2.7 MISSILE PROTECTION

In addition to static load analyses, a dynamic analysis was performed and demonstrated the structural adequacy of the highest stressed pipe whip restraint. The approach and techniques employed are those described in Bechtel Topical Report BN-TOP-2. To assure compatibility of inelastic stress analyses, elastic design techniques in accordance with Division 1.705, ANSI B31.7-1969 are used, or later appropriate ASME Section III Code sections provided that they have been reconciled. The major components including reactor vessel, reactor coolant piping, Reactor Coolant Pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contains one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle, or primary shield. The reactor vessel head and Control Rod Drives extend into the fuel transfer canal.

Penetrations for piping into the shielded cubicles for the steam generators, piping, and the pressurizer are located such that missiles which may be generated, such as valves, valve bonnets, valve stems, or reactor coolant temperature sensors will not escape the cubicles or possess sufficient energy to damage the reactor building liner plate.

Pipe lines carrying makeup and purification (high pressure injection) water are routed outside the shield walls entering only when connecting to the loop. Missiles which may be generated in one cubicle cannot rupture makeup and purification (high pressure injection) lines for the other loop.

Decay Heat Removal (low pressure injection) lines and core flooding lines connect outside the secondary shield walls. The decay heat and core flooding lines penetrate the secondary shield walls at opposite sides of the primary shield and enter opposite nozzles of the reactor vessel through penetrations in the primary shield. Thus, the Decay Heat Removal/core flooding lines are protected from potential missiles which might jeopardize their integrity.

A missile shield is located above the Control Rod Drives to stop a Control Rod Drive should it become a missile. The shield is removed during refueling.

4.3 SYSTEM DESIGN EVALUATION

4.3.1 DESIGN MARGIN

The Reactor Coolant System (RCS) is designed structurally for 2,500 psig and 650 °F. The system will normally operate at 2,240 psig at the reactor coolant pump outlet corresponding to 2155 psig at the RCS pressure tap on the hot leg and 604 °F at the reactor vessel outlet.

In the event of a complete loss of power to all reactor coolant pumps, reactor coolant flow, coastdown and subsequent natural circulation flow is more than adequate for core cooling and decay heat removal as shown by the analysis in Section 14.1.

The number of transient cycles specified in Table 4-8 for the fatigue analysis is conservative. Six heatup and cooldown cycles per year are specified, where the system may not be required to complete more than one or two cycles per year in actual operation. A heatup rate of 100 °F per hour was used in the analysis of Transients 1 and 2 in Table 4-8.

4.3.2 MATERIAL SELECTION

Each of the materials used in the RCS has been selected for the expected environment and service conditions. The major component materials are listed in Table 4-9. All RCS materials normally exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 stainless steel, Inconel, Alloy 690, 17-4PH(H1100), Zircaloy, or weld deposits with corrosion-resistant properties equivalent to or better than those of 304 stainless steel. Exceptions to the above statement involve RCS instrumentation nozzles that are modified after their initial installation. The repair methodology used for some of the instrumentation nozzles results in a small gap between the end of the existing Inconel sleeve or nozzle remnant and the new partial nozzle that penetrates the reactor coolant component. This process leaves gaps, which allow some contact of RCS with the carbon steel base material. Corrosion rates for this configuration have been evaluated and found acceptable. These materials were chosen for specific uses at various locations within the system because of their compatibility with the reactor coolant. There are no novel material applications in the RCS.

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained within the specifications given in the EPRI Secondary Water Chemistry Guidelines in order to prevent deposits and corrosion inside the steam generator. These required feedwater specifications have been successfully used in comparable once-through non-nuclear steam generators.

The selection of materials and the manufacturing sequence for the RCS components is arranged to ensure that no RCS pressure boundary material is furnace-sensitized stainless steel. Safe ends are provided on those carbon steel nozzles of the system vessels which connect the stainless steel piping. All dissimilar metal welds, with the exception of Inconel to stainless steel pipe welds, are made in the manufacturer's shops. Due to issues with primary water stress corrosion cracking (PWSCC) in Inconel (Alloy 600) material, the butt welded surge line connection to the “A” hot leg and connections to pressurizer piping nozzles were structurally overlayed with Alloy 690 material during 1R20. The butt welded Decay Heat piping connection to the “A” hot leg Decay Heat nozzle was structurally overlayed with Alloy 690 material during 1R21. Alloy 690 material has been shown to be highly resistant to the effects of PWSCC.

Susceptible Reactor Vessel (RV) Bottom Mounted Instrument Alloy 600 nozzle ID and OD surfaces and Alloy 82/182 welds have been laser peened to mitigate the potential for Primary Water Stress Corrosion Cracking.

4.3.3 REACTOR VESSEL

Stress Evaluation

A stress evaluation of the reactor vessel has been performed in accordance with Section III of the ASME Code. The evaluation shows that stress levels are within the Code limits. Table 4-14 lists the reactor vessel steady state stresses at various load points. The results of the transient analysis and the determination of the fatigue usage factor at the same load points are listed in Table 4-15. As specified in the ASME Code, Section III, Paragraph 415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 4-8. Figure 4-12 illustrates the points of stress analysis for the stresses listed in Table 4-14 and the fatigue usage factors listed in Table 4-15. These stress summaries demonstrate that all of the requirements for stress limits and fatigue required by ASME Section III for all of the operational requirements imposed by the design specifications have been met. The values tabulated in these summaries are the maximum values obtained in each region. They are calculated by methods based on conservative assumptions. The imposed transient descriptions are based on conservative description of all the realistic transient behavior that might be expected for this plant. Transients such as loss of flow and load that cause temperature and pressure excursions were considered in developing the specification transients. Their affect on accumulated usage factor is included in the stress analysis and in the values reported in summary Table 4-15. Table 4-16 presents a summary of the major reactor vessel material physical properties including the results of impact tests. Table 4-17 lists the chemical analysis results for the same materials.

Nil Ductility Transition Temperature (NDTT)

The reactor vessel plate material opposite the core was purchased to mill practices which improve material toughness and result in a lower range of NDTT values for heavy sections. The raw material was purchased to be capable of meeting Charpy V-notch values of 30 ft-lb or greater at a temperature of 40 °F. The material was tested during vessel fabrication after forming to show conformity to specified requirements or to determine the actual temperature at which the specified 30 ft-lb Charpy V-notch value was met.

The unirradiated or initial NDTT of pressure vessel base plate material was measured by the Charpy V-notch impact test (Type A) given in ASTM E23. Using the Charpy V-notch test, the NDTT is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA-533B steel, the ASME Section III Table N-332 specifies an energy value of 30 ft-lb. A curve of the temperature versus the energy absorbed in breaking the specimen is plotted. To obtain this curve at least two specimens are tested at a minimum of five different temperatures. Available data indicate NDTT differences as great as 40 °F between curves plotted through the minimum and average values respectively. The intersection of the minimum energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT. The determination of NDTT from the average curve is considered representative of the material and is consistent with procedures specified in ASTM E23. The material for these tests was treated by the methods outlined in ASME III Paragraph N-313. These tests were performed by the material supplier or B&W, in accordance with ASME Section III, Paragraphs N-313 and N-330.

The basic determination of vessel operation from cold startup and shutdown to full pressure and temperature operation is performed in accordance with a "Fracture Analysis Diagram" as published by Pellini and Puzak (see Reference 14).

At temperatures below the Design Transition Temperature (DTT), which is equal to NDTT + 60 °F, the pressure vessels will be operated so that the stress levels will be restricted to a value that will prevent brittle failure. These levels are:

A. Below the temperature of DTT minus 200 °F, a maximum stress of 10 percent yield strength.

B. From the temperature of DTT, minus 200 °F to DTT, a maximum stress which will increase from 10 to 20 percent yield strength.

C. At the temperature of DTT, a maximum stress of 20 percent yield strength.

With the stresses held within the above limits in "A" through "C", brittle fracture will not occur. This statement is based on data reported by Robertson (see Reference 16) and Kihara and Masubichi (see Reference 12) in published literature. These stress values are interpreted in terms of operating temperatures and pressures, and it can be shown that stress limits can be controlled by imposing operating procedure limits through control of pressure and temperature during heatup and cooldown (see References 15 & 16). This procedure assures that the stress levels do not exceed those specified in "A" through "C" above.

To confirm adequate fracture toughness of ferritic pressure retaining components of the reactor coolant pressure boundary, additional testing incorporating Drop Weight Tests (DWT) were performed. The resulting Nil Ductility Temperatures (NDT's) and upper shelf fracture energy levels are discussed below.

All toughness testing was performed in accordance with the number, direction and locations specified in Section III of the ASME Code for the given product form. Charpy V-notch tests were used to meet the fracture toughness requirements imposed by Section III.

The following estimates are made for pressure boundary material toughness properties:

The maximum estimated NDT temperature as obtained from DWT specimens is expected to be plus 50 °F or lower. The maximum temperature corresponding to the 50 ft-lb value of the C_V fracture energy is expected to be plus 60 °F or lower. The minimum upper shelf C_V energy value for the transverse (R.W.) quarter thickness is 85 ft-lbs or higher.

The pressure boundary materials were ordered and tested in accordance with the requirements of the 1965 Edition of Section III of the ASME Code including all addenda through Summer, 1967. The 1965 Edition does not require the determination of the NDT as obtained by drop weight test, nor the Charpy V-notch energy levels for specimens oriented in the "weak" direction. All base materials meet the Charpy V-notch energy value requirements listed in Section III at a temperature of plus 40 °F or lower. For the weld deposits the transition temperature was obtained by performing Charpy V-notch impact tests during procedure qualifications on weld deposits using the same flux and filler wire combinations as the production welds. All weld deposits meet Charpy V-notch energy values required by Section III at a temperature of plus 40 °F or lower.

Top and Bottom Shell Course

	Heat Number	Specimen Description	Drop Weight NDT	Temp at Avg Cv 50 ft-lb Energy Absorbed	Temp at Avg Cv 35 mils Lateral Expansion
Top Shell Course	C5114-2	R.W. 1/4	-10 °F	42 °F	43 °F
Forging	C5120-2	R.W. 1/4	-10 °F	12 °F	10 °F
Bottom Shell Course	C5114-1	R.W. 1/4	0	40 °F	40 °F
Forging	C5120-1	R.W. 1/4	-10 °F	26 °F	15 °F

Weld Deposits

Weld Description	Weld Number	Specimen Description	Charpy V-Notch Energy at +10 °F (ft-lb)
Top, Center and Bottom Circumferential Weld Deposits	WR-1	WL (Surface)*	30, 35, 40
Longitudinal Weld Deposits	WR-2-A1 and WR-2-A2	WL (Surface)*	38, 45, 46

* Specimen oriented in the weld width direction perpendicular to the weld centerline and parallel to the weld surface with the notch normal to surface. Specimens were taken 1/16 inch beneath the surface of the weld.

In addition to the impact test required by the ASME Code, the NDT and Charpy V-notch energy levels at several temperatures were obtained for the four plates that comprise the core region of the reactor vessel. The plate material is ASTM A533 Grade B, Class 1. The impact test specimens were taken from $^{1}/_{4}$ of the plate thickness, and oriented in the longitudinal direction with the length of the notch perpendicular to the surface of the material (R.W. direction). The weld deposits of the core region were impact tested at plus 10 °F using Charpy V-notch specimens oriented perpendicular to the direction of welding with the notch normal to the surface.

Specimen Description	Charpy V-notch Average Ft-Lb	Drop Weight Temperature	NDT Temperature
R.W. ¼ T	34	+ 10 °F	- 10 °F
R.W. ¼ T	63	+ 10 °F	- 10 °F
R.W. ¼ T	33	+ 10 °F	- 0 °F
R.W. ¼ T	37	+ 10 °F	- 10 °F

The approximate Charpy V-notch upper shelf fracture energy levels for the plates of the reactor vessel beltline region are given below. Presently, this information is not available for the weld deposits; however, the reactor vessel material surveillance program will determine such information. For the plates, the Charpy V-notch impact test specimens were taken from 1/4 of the plate thickness and oriented in the longitudinal direction with the notch perpendicular to the surface of the material (R.W. direction).

Approx Upper

	Heat Number	Specimen Description	Shelf Cv Energy (ft-lb)
Top Shell	C5114-2	R.W. 1/4	134
Course	C5120-2	R.W. 1/4	133
Bottom Shell	C5114-1	R.W. 1/4	128
Course	C5120-1	R.W. 1/4	123

The following estimates are made for the reactor vessel beltline materials, including welds:

A. The highest predicted End of Life (EOL) transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for specimens in the transverse (W.R.) direction and will be 310 °F when irradiated to the projected fluence level of 3 x 10^{19} n/cm^2 $\geq$ 1 Mev fission spectrum.

B. The minimum upper shelf energy value which will be acceptable for continued reactor operation without additional analysis is 50 ft-lb.

Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence value will be less than half of the originally estimated 3 x 10^{19} n/cm^2. The maximum inside surface fast neutron fluence at 48 EFPY is projected to be 1.44 x 10^{19} n/cm^2. The corresponding EOL transition temperature is similarly reduced while the minimum upper shelf energy value is increased.

The reactor vessel design has been reviewed to determine the feasibility of in-place annealing. The review indicates that the design does not preclude in-place annealing, if it should be necessary. The maximum reactor vessel temperature that can be obtained during in-place annealing has not been determined for the conditions above the design temperature of 650 °F.

Flux and Total Integrated Flux (nvt) at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is 3.0 x 10^{10} n/cm^2-s at the design power of 2,568 MWt. The corresponding calculated maximum fast neutron flux at the vessel wall is 2.2 x 10^{10} n/cm^2-s. This calculated value includes a lifetime average axial peaking factor of 1.3 and an azimuthal peaking factor of 1.29. For 40 years at 80 percent load this corresponds to an nvt of 2.2 x 10^{19} n/cm^2 (maximum) for the vessel wall.

Historical data removed - To review exact wording please refer to Section 4.3.3 of the FSAR.

The fluence calculations have been updated to 60 years at 90 percent load (54 Effective Full Power Years (EFPY) of operation). The fluence calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence values will be less than the originally calculated values. Reactor vessel materials within the scope of this analysis include forgings, plates, and welds that are expected to exceed a neutron fluence of 1E+17 neutrons per square centimeter (n/cm^2) with energies greater than 1.0 MeV projected to 54 EFPY. The calculated maximum fast neutron flux at the cladding to base metal interface is 7.18 x 10^9 neutrons per square centimeter per second (n/cm^2-s). The projected maximum fast fluence value for 54 EFPY has been determined to be

1.34 x 10^{19} n/cm^2 at the cladding to base metal interface. The maximum projected fluence at each reactor vessel material within the scope of the analysis was used to respond to the Fracture Toughness Requirements for protection against Pressurized Thermal Shock, 10 CFR 50.61.

Expected NDTT Shift

As a result of fast neutron bombardment of vessel metal in the region surrounding the core, the reactor vessel material ductility will change. The effect is an increase in the NDTT. Historical data removed - To review exact wording please refer to Section 4.3.3 of the FSAR.

Revised values of calculated vessel exposure have been used to determine the increase in the NDTT in accordance with 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock." For the 60-year exposure at 90 percent load (54 EFPY), the calculated value of the controlling material NDTT is 197.4 °F.

The NDTT shift is factored into the plant startup and shutdown procedures so that full operating pressure is not attained until the reactor vessel temperature is about DTT. The heatup and cooldown curves are given in Technical Specification 3.4.3, “Pressurization, Heatup and Cooldown Limitations.” Pressure-temperature limits are developed using an analytical approach that is in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. Additional requirements are contained in Table 1 of Appendix G to Title 10, Code of Federal Regulations, Part 50. The analytical techniques used to calculate P-T limits are based on approved linear elastic fracture mechanics methodology described in topical report BAW-10046A, Revision 2.

Permanent dosimetry hardware, installed in the reactor vessel cavity, is used to obtain neutron fluence measurements through the reactor vessel outer shell. This data will be correlated to the neutron fluence data obtained through the integrated specimen surveillance program which will permit an evaluation of the actual neutron exposure-induced shift for material NDTT.

Test coupons of welds, heat-affected zones, and base material for the material used in the reactor vessel, are incorporated in the reactor vessel surveillance program, as described in Section 4.4.6.

Fracture Mode Evaluation

Analyses have been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with postulated operation of the Emergency Core Cooling System (ECCS) at end of vessel design life.

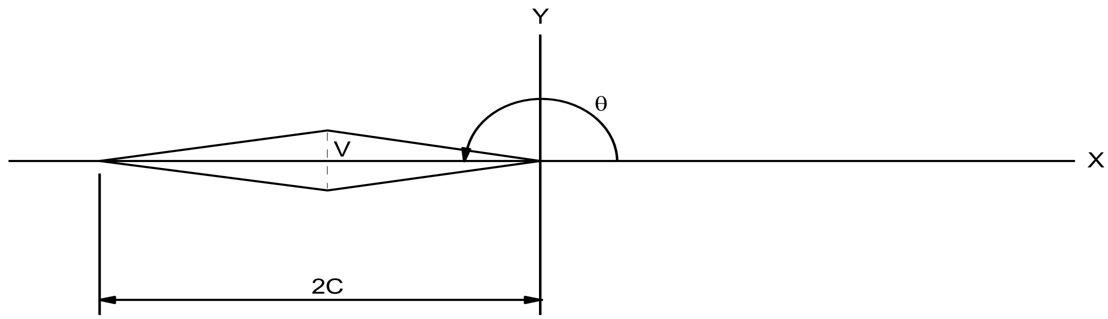
The state of stress in the reactor vessel during the Loss of Coolant Accident (LOCA) was evaluated for an initial vessel temperature of 603 °F. The inside of the vessel wall was rapidly subjected to 90 °F injection water of the maximum flow rate obtainable. The results of these analyses show that the integrity of the vessel is not violated.

Babcock & Wilcox Topical Report, BAW-10018, "Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock," provides the details of the analyses.

Crack Analysis

The design maximum allowable leakage rate for unidentified sources within the reactor pressure boundary is 3 gpm. The following fracture mechanics analysis derives the crack sizes as a function of wall thickness which would produce this flow.

In order to determine the crack size required for a leakage rate of 3 gpm, the following crack form was assumed:



According to ASTM Special Technical Publication, 381, "Fracture Toughness Testing and its Applications," the displacement (v) in the y-direction can be found to be:

$$V = \frac{K_I}{G}\left[\frac{(r)\frac{1}{2}}{2\Pi}\sin\frac{\theta}{2}\right]\left[2 - 2\gamma - \cos^2\frac{\theta}{2}\right]$$

where:

K_I = $\sigma_y \sqrt{\pi C}$

σ_y = yield strentgh

γ = Poisson's ratio

G = shear modulus

$\therefore$ For θ = 180°, r = C

v = $(\sigma_y G\sqrt{\underline{2}})(2\text{-}2\gamma)(C)$

From Victor Streeter's Fluid Mechanics, 1966, the velocity (U) out of the crack may be shown to be:

$$U = \sqrt{\frac{(P_1 - P_2)\ (2g)\ (144)}{\rho\left[\frac{ft}{d} + K\right]}}$$

where:

P_1 = operating pressure (2,200 psi)

P_2 = containment pressure (15 psi)

ρ = density (lbm/ft^3)

f = friction factor (assumed .06)

t = thickness of shell or pipe considered (in.)

d = equiv. diameter of crack (in.)

K = shock factor for expansion and contraction (1.5)

g = gravitational acceleration (32.2 ft/sec^2)

For the 28-inch ID cold leg piping:

T = 2.25 in.

ρ = 46.5 lbm/ft^3 • 555 °F

σ_y = 30.4 @ 10^3 psi • 570° (106 Grade C)

G = 11.5 x 10^6 psi

γ = 0.3

Assuming a crack length of 2-$^3/_8$ in. (C - 1-$^3/_{16}$")

V = .00312 in.

flow area (A) = (4)(C)(V/2) = 2CV = .00739 in^2

$$d = \frac{4A}{W.P.} = \frac{4\,(2CV)}{4\sqrt{C^2 + V^2}} \cong 2V = .00624 \text{ in}$$

$$U = \sqrt{\frac{(2200-15)\,(2)\,(32.2)\,(144)}{(46.5)\left[\frac{(.06)\;(2.25)}{.00624}+1.5\right]}} = 137.0 \text{ ft / sec}$$

flow rate in gal/min = 3.0 gpm

Therefore, a crack length of 2$^3/_8$ inches in the cold leg piping may be expected to leak at the rate of 3 gpm. Similarly, for the reactor vessel a crack length of 2.5 inches was determined.

Based on the answer above and the analysis described below, the ratios of through-wall crack length for the proposed leak limit to the critical through-wall crack are:

1) Irradiated Reactor Vessel Shell

$$\text{ratio} = \frac{2.5\text{ in.}}{10.9\text{ in.}} = 0.23$$

2) Primary Piping Cold Leg

$$\text{ratio} = \frac{2.375\text{ in.}}{8.7\text{ in.}} = 0.27$$

Through-wall crack in the irradiated reactor vessel shell. Because of the large diameter of the shell and wall thickness, the mathematical model considered most reasonable is for a through-wall crack in a flat plate. The equation for this crack, taken from ASTM Special Technical Publication 381, "Fracture Toughness Testing and its Applications," is:

$$K = \sigma\ ^1\Pi C$$

where:

K = stress intensity factor

C = 1/2 crack length

σ = membrane stress

When K reaches a critical value called K_{IC} the fracture toughness of the material, the crack length (2C) is the critical crack length.

For the reactor vessel the conditions are:

Material SA-533GR-B-CL1

Temperature 570 °F σ_y (570°) = 41.4 ksi = 29.0 ksi (from contract design reports)

K_{IC} - 120 ksi$\sqrt{\text{in}}$ (from Westinghouse Research Laboratories "Fracture Mechanics Technology as Applied To Thick-Walled Nuclear Pressure Vessels" by Wessel and Mager.)

Rearranging the above equation and substituting K_{IC} for

$$C = \frac{(K_{IC})^2}{\pi\sigma 2}$$

$$= \frac{(120)^2}{\pi(29)^2} = 5.45$$

Therefore, the critical crack length = 2C = 10.9"

Through-wall crack in the cold leg of the reactor coolant pipe. The equation proposed by Hahn in BMI-1883, "Review of Through-Wall Critical Crack Formulations for Piping and Cylindrical Vessels," for high toughness materials is applicable. The equation is:

$$\sigma^* = \sigma_h M$$

where:

σ^* = flow stress of the material = $1.04\sigma_y + 10$ ksi

σ_h = membrane loop stress

M = stress magnification factor = $\left(1+1.61\,\frac{C^2}{Rt}\right)^{1/2}$

σ_y = yield stress

C = ½ critical crack length

R = average radius of pipe

t = pipe wall thickness

For the reactor coolant piping the conditions are:

Material SA-106-GR C

Temperature 570 °F, σ_y (570°) = 30.425 ksi

ID = 28 5/8" max.

T = 2 1/4" min.

R = 15 7/16"

σ_h = σ_y = 30.425 (assumed as worst case)

Rearranging the above equation and making appropriate substitutions -

$$M = \frac{\sigma^*}{\sigma_h}$$

$$1 + 1.61\,\frac{C^2}{Rt} = \frac{(1.04^{\sigma y} + 10)^2}{(\sigma_y)}$$

$$C = 4.35$$

Therefore, the critical crack length = 2C = 8.7"

Report BMI-1883 stipulates the flow stress (σ^*) equation is valid if:

$$\frac{(K_C/\sigma_y)^2}{C} \geq 5$$

yielding

$$K_C \geq \sigma_y(5C)^{1/2}$$

$$\geq 30.425\ \ 5(4.35)^{1/2} = 142$$

where:

K_C = critical stress intensity factor (ksi in) as identified in BMI-1883.

K_C data for SA-106-GR C material is not available. However, BMI-1866, "Investigation of the Initiation and Event of Ductile Pipe Rupture," represents pipe rupture tests on SA-106-GR B material, some of which could be classified as SA-106 GR C. The lowest K_C value for the SA-106 GR B material that could be classified as GR C is 168 ksi $\sqrt{in}$. Therefore, the equation is valid.

Closure

The reactor closure head is bolted to a ring flange on the reactor vessel. The vessel closure seal is formed by two concentric metal O-ring seals with provisions for leak-off between the O-rings. Reactor closure head leakage will be negligible from the annulus between the metallic O-ring seals during vessel steady state and virtually all transient operating conditions. Only in the event of a rapid transient operation, such as an emergency cooldown, would there be some leakage past the inner O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal.

The reactor closure head is attached to the reactor vessel with 60 6½-inch diameter studs. The stud material is A-540, Grade B23 (ASME III, Case 1335) which has a minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, have a tensile stress of approximately 46,000 psi.

The studs are an integral part of the closure geometry and are evaluated for temperature and pressure effects considering the transient conditions identified in the RCS functional specifications. All normal, upset, emergency, and faulted conditions are considered in the analysis. The stress analysis of the studs includes a fatigue evaluation. The stress analysis of the closure verifies the structural integrity of the studs in accordance with ASME Section III criteria. The analysis also evaluates the sealing ability of the gaskets, thereby providing the assurance that a leak will not occur during design and normal operating conditions.

The fatigue evaluation results for the studs are included in Table 4-15.

Control Rod Drive Service Structure

The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined LOCA and Maximum Hypothetical Earthquake (MHE). Requirements for rigidity and space requirements for service routing results in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

4.3.4 STEAM GENERATORS

4.3.4.1 Research and Development

Historical data removed - To review exact wording please refer to Section 4.3.4.1 of the FSAR.

4.3.4.2 Stress Evaluation

A. General

Because the steam generator is of a straight tube-straight shell design and because of a minor difference in the coefficient of thermal expansion between Inconel and carbon steel, there exists a structural limitation on the mean temperature difference between the tubes

and the shell. During normal operation of the steam generator, the tube to shell mean ΔT poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant or a main steam line break would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred (some tubes may be initially in compression), there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a LOCA. The basis for this analysis is a hypothetical rupture of a pressurizer surge line resulting in a maximum pressure from the secondary side of 1,066 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet) and the tubes satisfy the applicable buckling criteria when subject to external pressure.

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 41.1 ksi across the center ligaments, which is well below the ASME Section III allowable limit of 94.5 ksi at 650 °F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress maximum intensity anywhere in the head including head-to-tubesheet juncture and head-to-support skirt juncture is 68.3 ksi, which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of $^5/_8$ in. OD/0.034-inch wall Inconel tubing show collapse under an external pressure of 4,950 psi. This is a factor of safety of 4.6 against collapse under the 1,066 psi accidental application of external pressure to the tubes. This means that failure of steam generator tubes as a consequence of design basis pipe break in the RCS is not a credible failure mechanism. In addition, the B&W Topical Report BAW-10027 notes features of the steam generator design that preclude the possibility of initiation of this failure mechanism.

Excess material of at least 0.0065 inch over minimum required pressure thickness is available in the tube wall from normal tolerances in material size specifications.

Collapse pressure tests on representative tube specimens proved that the mode of collapse was ductile without subsequent leakage. The minimum collapse pressure on a virgin tube sample was 4,950 psig at a temperature of 650 °F. The specimens were hydrotested with internal pressure of 3,600 psig after collapse and no leaks were detected. Other tube samples were dye penetrant examined after collapse and none were found to have cracks.

An examination of stresses under the conditions predicted to occur during a main steam line break show that the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4-18.

The basic design criterion for the tubes assumes a pressure differential of 2,500 psig in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effects of secondary side pressure loss and maximum hypothetical earthquake have been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated and considered. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

The ratio of allowable stresses to the computed stresses are summarized in Table 4-19.

B. Stress Intensities and Cumulative Usage Factors

The results of the transient analysis and the determination of the fatigue usage factor at various load points are listed in Table 4-20.

4.3.4.3 Steam Generator Tube Surveillance

The steam generator tubes are examined periodically under a systematic surveillance program in accordance with NEI 97-06, “Steam Generator Program Guidelines.” This program is fully described in Technical Specification 5.5.9.

4.3.5 REACTOR COOLANT PUMPS

The reactor coolant pumps make up one of the two active components in the RCS which are classified as Class 1 mechanical components. The other is the control rod drive mechanisms. Reactor coolant pumps are modeled in the RCS seismic analysis, and the appropriate response spectra are imposed at the support points of the major contributing system components. Resulting seismic acceleration of mass points are used to check bearing loadings and shaft deflections. Reactor Coolant Pumps and motor bearings are also designed to meet seismic design criteria.

Descriptions of characteristics and operations for the reactor coolant pumps and the pump motors are given in Sections 4.2.2.5 and 4.2.2.6, respectively.

4.3.5.1 Reactor Coolant Pump Motor Flywheels (A-C Motor)

Each Allis-Chalmers reactor coolant pump motor has two flywheel assemblies, a large assembly at the upper end of the motor and a small assembly at the lower end. These flywheels increase rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. An anti-reverse device is included to prevent back-rotation, thereby reducing starting time and rotor heating.

The flywheels are made from rolled plate per ASTM A516, Grade 65. The minimum yield strength specified is 35,000 psi. The NDTT as determined by Charpy V-Notch tests is less than + 10 °F.

Drop weight tests to determine the NDTT were not required at the time the flywheels were purchased and therefore were not performed. The minimum acceptable Charpy V-notch upper shelf energy level in the weak direction and the fracture toughness of the material at normal operating temperature were not determined.

The primary stress in the flywheels at normal operating speed (1,200 rpm) is due to centrifugal force. The following design stresses (specified as a percentage of the minimum specified yield strength) result from centrifugal forces:

		Normal Speed	125 Percent Overspeed
A.	Large Flywheel -		
	Bore Stress	36.9%	57.8%
	Average Stress	18.45%	28.9%
B.	Small Flywheel -		
	Bore Stress	18.3%	28.6%
	Average Stress	9.15%	14.3%

Rotating disc theory shows that failure does not occur until the average stress exceeds the yield stress of the material.

The stresses due to interference fit were considered in the design of the flywheel. The flywheels are installed on the shaft with an interference fit that results in a hoop stress in the flywheel inner bore of 63.2 percent of the specified yield strength. At the designed operating speed (1,200 rpm), the hoop stress in the flywheel inner bore remains 63.2 percent of yield strength, with 26.3 percent of the 63.2 percent attributed to the interference fit. Similarly, at the 125 percent overspeed condition, only 5.4 percent of the 63.2 percent is attributed to interference fit. The remaining stresses result from centrifugal forces on the rotating flywheel disc.

The motors are designed per NEMA standards which specify a maximum overspeed of 125 percent. The maximum anticipated overspeed of the flywheel is anticipated to be 101.5 percent based on the maximum variance in pump power frequency.

A speed of 1,800 rpm is estimated in the event that a double-ended rupture of the 28-inch pump discharge piping occurs at the same time as a loss of power to the pump motor. This is the estimated maximum speed obtainable in the event of a reactor coolant piping break in either the suction or discharge side of the pump. This speed is calculated on the basis of a free-spinning pump resistance, with the pump acting as a turbine increasing the flywheel speed until stabilized by the inertia of the mass of the pump impeller, motor and flywheel.

Factors tending to limit the maximum rotational speed of the flywheel were omitted for conservatism in the calculation of the estimated 1,800 rpm maximum attainable rotational speed of the flywheel. At this speed the maximum primary stress at the bore of the flywheel is calculated to be within the elastic range of the material (approximately 80 percent of the yield strength). This stress is well below the limit of $^2/_3$ of the ultimate strength of the material that is considered as an acceptable limit for faulted conditions for components of the primary coolant system. There was no preoperational overspeed test specified or run on any of the flywheels.

Limited surface and limited volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10-year intervals all four reactor coolant pump flywheels will be examined. The results of each examination will be compared to the original acceptance criteria (ASME Boiler and Pressure Vessel Code Section III NB 5330 Requirements for Class 1 Components). An evaluation will be made to assure the absence of unacceptable defects.

Detailed Description of Reactor Coolant Pump Motor Flywheels

A. The geometry of the flywheel and backstop device is shown in Figures 4-17 and 4-18.

B. Physical Dimensions:

1. Bore Diameter - 30.400" to 30.402"

2. Outer Diameter - 72.0" to 49.6"

3. Keyway Radial Depth - 0.39"

4. Corner Radius in Keyway - 0.03R

5. Size and Location of Holes in Flywheel - refer to Figure 4-17.

C. Material:

1. ASTM A516, Grade 65 Carbon Steel

2. Minimum Yield Stress - 35,000 psi @ room temperature (120 °F Operating Temp.)

3. Specific Weight - 0.283 lbs. per cubic inch

4. Poisson's Ratio - 0.3

5. Young Modulus - 30 x 10^6

6. NDTT as determined by Charpy V-notch test is less than +10 °F

7. Fracture Toughness

All flywheel segments are made from ultrasonic quality rolled plate steel per ASTM A516, Grade 65 with a minimum yield strength per specification of 35,000 psi in.

The rotating assembly of the A-C motor has a total inertia (WK^2) of 73,500 lb/ft^2 exclusive of the WK^2 for the pump rotating assembly. The flywheel mass is designed as required to obtain the required amount of inertia. The flywheel is designed not to interfere with removal of the rotor from the top of the stator.

All flywheel segments were inspected in accordance with B&W Specification S-102E, Ultrasonic Inspection of Plate by Straight and Angle Beams, April 18, I969 issue. This specification is taken from ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, Class A N-321 and N-322. For the angle beam inspection, an indication three percent of the wall thickness is the criterion for rejection. Instrument sensitivity is adjusted to display from at least 50 percent but less than 100 percent of full screen amplitude. The smallest radial crack that could exist in a plate having passed the angle beam ultrasonic inspection is less than 0.24 inch, based on a plate thickness of eight inches making the smallest notch possible of 0.24 inch (8" x 0.03 = 0.24").

For the straight beam inspection, any area where one or more discontinuities produce a continuous total loss of back reflection accompanied by continuous indications on the same plane that cannot be encompassed within a 3-inch or one-half of the plate thickness is unacceptable. In addition, two or more smaller indications are unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect or unless they may be collectively encompassed by a 3-inch circle.

All flywheel segments are subjected to magnetic particle examination of the finished bore and of other surfaces adjacent to the bore out to a distance equal to one-fourth of the distance from the bore out to the outer circumference. The following relevant indications are unacceptable:

A. Radial cracks.

B. Rounded indications with dimensions greater than $^{3}/_{16}$ of an inch.

C. Ten or more rounded indications in any six square inches of surface with the major dimension of this area taken in the most unfavorable location relative to the indications being evaluated.

D. Four or more rounded indications in a line separated by $^{1}/_{16}$ inch or less edge to edge.

The radial fit of the flywheel to the rotor spider is 0.0115 ± 0.001 inches (interference fit of 0.0125 inch is used for calculations).

The bolt stresses and compressive stresses on the flywheel plates due to the bolt load of the bolts used to assemble the three upper flywheels are not considered as significant with regard to the tangential stresses that are used for the fracture mechanics analysis.

Stress Analysis of Reactor Coolant Pump Flywheel

A stress analysis of the flywheel is performed to determine areas of stress concentration, stress magnitudes and the most likely flawed configurations to consider in a fracture mechanics analyses.

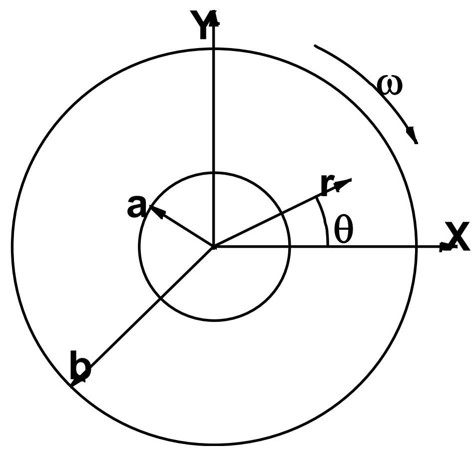
Figure 4-19 shows the finite element model of a quarter of the RCP flywheel face plate. The face plate of the flywheel assembly is considered to be the most critical component and is, thus, the only component modeled for a finite element stress analysis. The model is composed of triangular and quadrilateral constant strain plate elements which are arranged in a fine mesh pattern around spoke keyways and bolt holes.

Figure 4-20 shows the computed angular variation of the tangential stress σ_θ at the inner bore. Unit loads were applied at the location of each spoke to determine the stress variation shown. Inertia forces were omitted for this analysis. The stresses for a given shrink fit force can be determined by multiplying the magnitude of the force times the ordinates of the curve in Figure 4-21.

Figure 4-21 shows the variation of the tangential stress σ_θ across a section of the flywheel containing a bolt hole. The concentration of stress at the sides of the hole are shown.

Figure 4-22 shows the variation of the tangential stress σ_θ across a section of the flywheel containing a keyway. The tensile stress in this area is relieved by the bearing of the spoke load into the keyway and surrounding area.

The effect of inertia forces due to rotational speed ω can be assessed with sufficient accuracy by superimposing the tangential stress in a solid rotating disk on the stresses given in the previous three figures. The tangential stress in a solid rotating disk can be computed as described in Timoshenko's "Theory of Elasticity," as follows:



a = 15.20"
b = 36.00"
v = 0.3
ρ = 0.283 lb/in^3
ω = 0.732x10^{-3} lb-sec^2/in^4

$$\sigma_\theta = \frac{3+v}{8}\rho\omega^2\left(b^2+a^2+\frac{a^2b^2}{r^2}-\frac{1+3v}{3+v}r^2\right)$$

$$\frac{\sigma_\theta}{\omega^2} = \frac{3.3}{8}x.732x10^{-3}\left(36.0^2+15.20^2+\frac{(36.0)^2(15.20)^2}{r^2}-\frac{1.9r^2}{3.3}\right)$$

$$\frac{\sigma_\theta}{\omega^2} = \left(.3020x10^{-3}\right)\left(1527.04+\frac{297727.84-.58r^2}{r^2}\right)$$

Figure 4-23 shows the variation of the tangential stress σ_θ across a section of the flywheel. For a given rotational speed ω the ordinates of this curve need only be multiplied by ω^2 (rad/sec^2) to determine the stress distribution for inertia forces only. These stresses can then be superimposed on those obtained using Figures 4-19 through 4-21 to obtain the complete stress distribution for inertia and shrink-fit forces.

A. Fracture Mechanics Analysis

The method of analysis used for determining crack tip stress intensity factors for a hypothetical flawed flywheel subjected to shrink-fit and rotational stresses is the one suggested by Klecher in "Flywheel Fracture Mechanics Analysis," March 6, 1972. In order to utilize this method, the reactor coolant pump flywheels, which are mounted on spoked shafts, are assumed to be mounted on a solid shaft.

B. Fracture Toughness Prediction

In spite of the relatively common usage of ASTM A-516 steel plates under moderate or lower temperature services, few fracture toughness values appear in the literature. Most of the available "valid" fracture toughness values are for low temperatures, below -100 °F. The fracture toughness values are highly temperature dependent with a rapid increase in toughness near the NDTT. Generally, ASTM A-516 steel plates have a NDTT below 0 °F.

James in "Fracture Toughness of A-516, Grade 60 Steel," reported the fracture toughness values, K_{IC}, of a 1-inch thick plate of A-516, Grade 60 steel, for the temperature range of -300 °F to -100 °F. The data was obtained under a semi-dynamic loading condition and was reported in terms of the parameter T in (A/ε), where T is the absolute temperature and A is a constant equal to 108 sec^{-1}. Reference 14 also contains similar data for A-516, Grade 70 and for ABS-C steels. Figure 4-24 illustrates all the collected values for the three steels. Except where indicated otherwise, all the values in Figure 4-24 are "valid" per ASTM E-399. The two A-526, Grade 60 specimens tested at -100 °F failed to meet the thickness criterion $B > 2.5 (K_{IC}/\sigma oys)^2$, however, their fracture surfaces and the load-displacement recorders were similar to those for the "valid" tests conducted at lower temperatures. The NDTT of the 1-inch A-516, Grade 60 steel plate was found to be -40 °F. The significance of Figure 4-24 is that for A-516 steel plates the rapid up-swing of the fracture toughness, K_{IC} occurs at a temperature of approximately -100 °F.

The Charpy V-notch impact properties and the NDTT for ASTM A-516 steel plates are well known. The literature contains empirical correlations between transition temperature type data and fracture mechanics data. For the brittle-to-ductile transition range, the static fracture toughness has been shown to be related to the Charpy V-notch impact values by the expression

$$K_{IC} = 15{,}550\ (C_y)^{1/2}$$

where K_{IC} is the static fracture toughness in psi-$\sqrt{\text{in}}$, and C_y is the Charpy V-notch impact value in ft-lb. The range of C_V values for this correlation is limited to 5 ft-lb $\leq C_y \leq$ 50 ft-lbs. (see Reference 16.)

Charpy V-notch average (of three specimens performed in accordance with ASTM SA-379) test curves for the transverse and longitudinal direction are available for all the flywheel plates of this plant. The following illustrates the temperatures corresponding to the C_y 20 ft-lb and 50 ft-lb energy values for the transverse and longitudinal directions for each of the flywheels. The temperature values were extracted from the test curves (line through points representing average of test value).

Based on the above empirical relationship between C_y and K_{IC} 20 ft-lb of Charpy V-notch absorbed energy corresponds to a fracture toughness of 69.3 ksi-in and 50 ft-lb corresponds to 109.6 ksi-in.

	Temperatures (°F) at			
	20 ft-lb		50 ft-lb	
	Transverse	Longitudinal	Transverse	Longitudinal
Lower Flywheel Heat B-2010	-75	-130	-25	-25
Top Flywheel Heat B-2003	-85	-95	+15	-40
Large Top Fly- wheel Heat B-1972	-95	-115	-5	-70
Top Flywheel Heat B-7361	-45	-120	+30	-70

Calculations:

Flywheel 1

Given: Outer radius, b = 36.0

Inner radius, a = 15.2

$\rho = 490 \text{ lb/ft}^3 = 7.34 \times 10^{-4} \text{ lb sec}^2/\text{in}^4$

Normal operating speed, N = 1,200 rpm

Radial interference fit, U = 0.0125 inch

Young's modulus, $E = 30 \times 10^6$ psi

Fracture Toughness

$K_{IC} = 69.3 \; K_{si} \sqrt{\text{in}}$ @ -45 °F

$K_{IC} = 109.6 \; K_{si} \sqrt{\text{in}}$ +30 °F

Calculation:

$\alpha = a/b = 0.4222$

$$\left.\begin{aligned} \rho\omega^2 b^2 &= 15{,}030 \text{ psi @ 1,200 rpm} \\ \rho\omega^2 b^{5/2} &= 90.18 \; K_{si} \sqrt{\text{in}} \text{ @ 1,200 rpm} \end{aligned}\right\} \begin{gathered}\text{@ other speeds}\\ \text{multiply by}\\ (N/1200)^2\end{gathered}$$

$P(W) = P_o - (3 + v)/8 \; \rho\omega^2 b^2 \, (1 - \alpha^2)$

where:

P_o = $(EU/2b)(1 - \alpha^2)/\alpha$

P_o = 10,136 psi

so:

$P(W) = 10{,}136 - 0.339\,\rho\omega^2 b^2 = 5{,}041.4$ psi

K_{IC} uncertain estimated

$K_{IC_1} = 69.3\ K_{si}\sqrt{\text{in}}$ @ -45 °F

$K_{IC_2} = 109.6\ K_{si}\sqrt{\text{in}}$ @ 30 °F

From the above table the maximum temperature out of the four heats corresponding to the 20 ft-lb is -45 °F and to the 50 ft-lb is +30 °F. No credit is taken for the fact that the flywheels operate normally at 120 °F with brief periods of operation at 70 °F during startup.

Figure 4-25 represents the stress intensity factor (K_1) vs crack length (L) for the large top flywheel for various overspeeds. Only the large top flywheel was considered because it has the largest α (α = a/b) of the four flywheels.

Figure 4-26 shows the crack length (L) vs percent overspeed for the predicted fracture toughness values of 69.3 and 109.6 $K_{SI}\sqrt{\text{in}}$, which correspond to the temperatures of -45 °F and +30 °F, respectively.

4.3.5.2 Reactor Coolant Pump Clutch Anti-Reverse Rotation Device (A-C)

The reactor coolant pump clutch anti-reverse rotation device is shown in Figure 4-18. The clutch consists of three basic elements: a cylindrical outer race, solid cylindrical rollers uniformly spaced in a cage, and a cam unit with uniformly spaced ground incline planes equal in number to the number of rollers. When relative motion between the cam unit and the outer race is in the overrunning or freewheeling direction the lubricant wedge moves the rollers into the deeper cam zone. When the direction of rotation is reversed the rollers move up the incline plane and lock between the cam and the outer race, thereby preventing rotation.

The clutch is rated at a torque equal to that which would occur upon energizing the motor at full voltage in the reverse direction. In holding tests conducted at the manufacturer's facility a reverse torque of 1.5 times rated torque was applied to a clutch for a total of 1,650 applications with no slippage. An identical clutch did not slip when tested to three times rated torque for approximately two dozen applications.

The primary purpose of the anti-reversing device is to prevent reverse rotation of the pump motor during normal operation with reverse flow through a non-operating pump. As noted above, various tests of this anti-reversing device have been made to assure the adequacy of the device for its intended service.

There have been no tests conducted with conditions simulating a guillotine break at the pump suction. However, calculations have been made to determine the maximum theoretical speed and torque that the flywheel would attain assuming an inoperable anti-reversing device and a guillotine break at the pump suction. These calculations show that the theoretical maximum torque for this condition far exceeds the torque necessary to shear the pump shaft. The torque necessary to shear the pump shaft has been calculated to be 2.5 x 10^5 ft-lbs. At the time that this torque is reached the absolute speed of the flywheel calculated for the pump suction break is less than the speed used in the fracture analysis previously described. Therefore, it is concluded for this postulated accident that the pump shaft would shear prior to attaining speeds which could result in flywheel damage.

4.3.5.3 Jeumont Industrie Reactor Coolant Pump Motor Flywheel

The JI reactor coolant pump motor has one flywheel assembly at the top of the motor above the upper bearing assembly. This flywheel increases rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. An anti-reverse device is included to prevent back-rotation, thereby reducing starting time and rotor heating.

The flywheel is made from rolled plate per ASTM A533-82, Grade B, Class 1. The minimum yield strength specified is 50,000 lbs./sq. inch. The NDTT as determined by Charpy V-Notch tests is less than +10 °F (-12 °C).

The primary stress in the flywheel at normal operating speed (1,200 rpm) is due to centrifugal force. The following design stresses (specified as a percentage of the minimum specified yield strength) result from centrifugal forces:

Flywheel -

	Normal Speed	125 Percent Overspeed
Bore Stress	22.2%	34.6%
Average Stress	11.1%	17.3%

Rotating disc theory shows that failure does not occur until the average stress exceeds the yield stress of the material.

The stresses due to interference fit where considered in the design of the flywheel. The flywheel is installed on the shaft with an interference fit that results in a hoop stress in the flywheel inner bore of 8.24 percent of the specified yield strength. At the designed operating speed (1,200 rpm), the hoop stress in the flywheel inner bore increases to 10.69 percent of yield strength, with 0.0 percent of the 10.69 percent attributed to the interference fit. In other words, the interference fit stress component has disappeared at the operating speed of 1200 rpm. At the 125% percent overspeed condition, the stresses resulting from centrifugal forces on the rotating flywheel disc have increased to 16.71 percent of yield strength. Similarly, at the 125% percent overspeed condition, 0.0 percent of the 16.71 percent is attributed to interference fit. All of the stresses result from centrifugal forces on the rotating flywheel disc.

The motor is designed per NEMA standards which specify a maximum overspeed of 125 percent. The maximum anticipated overspeed of the flywheel is anticipated to be 101.5 percent based on the maximum variance in pump power frequency.

A speed of 1,800 rpm is estimated in the event that a double-ended rupture of the 28-inch pump discharge piping occurs at the same time as a loss of power to the pump motor. This is the estimated maximum speed obtainable in the event of a reactor coolant piping break in either the suction or discharge side of the pump. This speed is calculated on the basis of a free-spinning pump resistance, with the pump acting as a turbine increasing the flywheel speed until stabilized by the inertia of the mass of the pump impeller, motor, and flywheel.

Factors tending to limit the maximum rotational speed of the flywheel were omitted for conservatism in the calculation of the estimated 1,800 rpm maximum attainable rotational speed of the flywheel. At this speed the maximum primary stress at the bore of the flywheel is calculated to be within the elastic range of the material (approximately 24 percent of the yield strength). This stress is well below the limit of 2/3 of the ultimate strength of the material that is considered as an acceptable limit for faulted conditions for components of the primary coolant system. There was no preoperational overspeed test specified or run on the flywheel.

Limited surface and limited volumetric examination of the reactor coolant pump flywheel will be conducted coincident with refueling or maintenance shutdowns such that within a 10-year period after startup all four reactor coolant pump flywheels will be examined. The results of each examination will be compared to the original acceptance criteria (ASME Boiler and Pressure Vessel Code Section III NB 5330 Requirements for Class 1 Components). An evaluation will be made to assure the absence of unacceptable defects.

Detailed Description of Reactor Coolant Pump Motor Flywheel

A. The geometry of the flywheel and backstop device is shown in Figures 4-17a and 4-30.

B. Physical Dimensions:

   1. Bore Diameter - upper end = 11.504"
      lower end = 11.551"

   2. Outer Diameter - 67.874"

   3. Keyway Radial Depth - 0.835"

   4. Corner Radius in Keyway - 0.315"

C. Material:

   1. ASTM A533-82, Grade B, Class 1, Carbon Steel

   2. Minimum Yield Stress - 50,000 psi @ room temperature (120 °F Operating Temperature)

   3. Specific Weight - 0.283 lbs. per cubic inch

   4. Poisson's Ratio - 0.3

   5. Young's Modulus - 30.48 x 10^6

   6. NDTT as determined by Charpy V-notch test is less than +10 °F (-12 °C)

   7. Fracture Toughness (see page 4.3-27)

All flywheel segments are made from ultrasonic quality rolled plate steel per ASTM A533-82, Grade B, Class 1, with a minimum yield strength per specification of 50,000 lbs/sq. in.

The rotating assembly of the motor has a total inertia (WK^2) of 68,605 lb/ft^2 exclusive of the WK^2 for the pump rotating assembly. The flywheel mass is designed to obtain the required amount of inertia. The flywheel is designed not to interfere with removal of the rotor from the top of the stator.

The flywheel segments were inspected in accordance with ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, and Section IV Nondestructive Examination, and ANO Equipment Specification ANO-E-108. For the angle beam inspection, one or more discontinuities that produce indications exceeding in amplitude the indication from the appropriate calibration notches is criterion for rejection. Instrument sensitivity was adjusted to display from at least 50 percent but less than 100 percent of full screen amplitude.

For the straight beam inspection, all discontinuities producing a continuous total loss of back reflection accompanied by continuous indications on the same plant that cannot be encompassed within a circle whose diameter is 3 inches or one-half the thickness, whichever is greater, is the criterion for rejection. In addition, two or more defects smaller than described above, shall be unacceptable unless separated by a minimum distance equal to the greatest size of the largest defect unless they can be collectively encompassed by the circle described above.

The flywheel was subjected to dye penetrant examination of all flat surfaces as well as junction radii inside a 35 inch diameter circle centered from the flywheel center axis. Also examined were the finished bore, keyways, and gage holes to a diameter of 1.38 inches (35mm). The following relevant indications are unacceptable:

A. Linear indications with major dimensions greater than 3/16 of an inch.

B. Rounded indications with major dimension greater than 3/16 of an inch.

C. Four or more rounded indications in a line separated by 1/16 inch or less edge to edge.

D. Ten or more indications in any 6 square inches of area whose major dimension is no more than 6 inches with the dimensions taken in the most unfavorable location relative to the indications being evaluated.

The radial interference fit is given as 0.0016 inches (this dimension is used for calculations). The bolt stresses and compressive stresses on the flywheel plates due to the bolt load of the bolts used to assemble the flywheel is not considered as significant with regard to the circumferential stresses that are used for the stress analysis.

<u>**Stress Analysis of Reactor Coolant Pump Flywheel**</u>

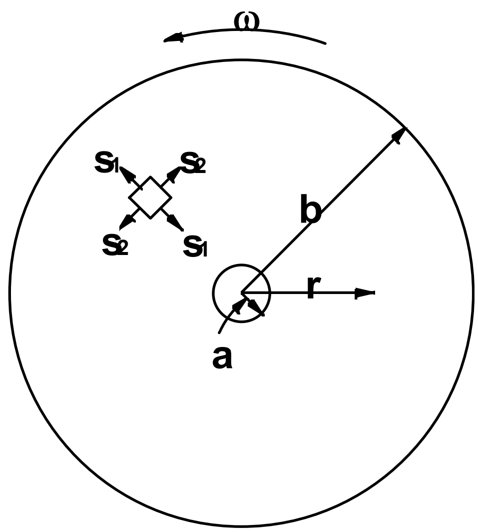
A stress analysis of the flywheel was performed to ensure that the combined primary stresses due to the centrifugal forces and the interference fit of the flywheel on the shaft do not exceed:

(1) either 1/3 of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction at the normal operating speed (1200 rpm)

(2) either 2/3 of the minimum specified yield strength or 2/3 of the measured yield strength in the weak direction at the design overspeed condition (1500 rpm)

The figure below shows the RCP flywheel face plate geometry used in the stress analysis. The face plate of the flywheel assembly is considered to be the most critical component and is, thus, the only component for which stress analysis calculations were performed. Indicated on this figure are the radial (S1) and circumferential (S2) stress directions and relevant geometries.

The stress analysis showed that, at nominal operating speed (1200 rpm), the shrink-fit forces went to zero. This was shown based on the calculation that the differential bore radius change exceeded the maximum interference value at 1200 rpm. Since the magnitude of the shrink-fit forces is zero at slightly less than nominal operating speed, the shrink-fit force was not considered a significant factor in the stress analysis. Any increase in rotational speed above the speed where shrink-fit forces go to zero increases the magnitude of the stresses due to inertia forces. The tangential stress in a solid rotating disk can be computed as described in Timoshenko's "Theory of Elasticity," as follows:



a = 5.752”
b = 33.93”
v = 0.3
ρ = 0.283 lb/in^3
ω = 0.732 x 10^{-3}lb - sec^2/in^4

$$\sigma_\theta = \frac{3+v}{8}\rho\omega^2\left(b^2 + a^2 + \frac{a^2b^2}{r^2} - \frac{1+3v}{3+v}r^2\right)$$

$$\frac{\sigma_\theta}{\omega^2} = \frac{3.3}{8} \times .732 \times 10^{-3}\left[33.93^2 + 5.752^2 + \frac{(33.93)^2(5.752)^2}{\mathrm{r}^2} - \frac{1.9\mathrm{r}^2}{3.3}\right]$$

$$\frac{\sigma_\theta}{\omega^2} = (.3020x10^{-3})\left(1184.33 + \frac{38089.52 - .58r^2}{r^2}\right)$$

As noted previously, the shrink fit forces are zero at the operating speed of 1200 RPM and all the stresses are due to inertia forces. For this reason, a figure with the two forces superimposed on each other is not given. For a given rotational speed ω the ordinates of Figure 4-28 need only be multiplied by $(N/1200)^2$ (when N is greater than or equal to 1055 RPM) to determine the stress distribution for inertia forces only.

Figure 4-28 shows the stress analysis calculation results with respect to the radius for the radial and tangential stress for 1200 rpm.

Figure 4-29 shows the stress analysis calculation results with respect to the radius for the radial and tangential stress for 1500 rpm.

A fracture toughness prediction was not performed as this is not specifically required in NUREG-0800 if the normal operating temperature of the material is 100 °F above the RT_{NDT}. The RT_{NDT} of the flywheel was tested and determined to be < 10 °F (documented in BWNT Document QCSR 92-446). The normal operating temperature for the flywheel is 120 °F. Therefore, the flywheel material meets the fracture toughness compliance criteria as identified in the Standard Review Plan, NUREG-0800.

4.3.5.4 Jeumont Industrie Reactor Coolant Pump Clutch Anti-Reverse Rotation Device

The JI reactor coolant pump motor anti-reverse rotation device is shown in Figure 4-30. The anti-reverse rotation device consists of four basic elements: five equidistant pawls fastened on the lower part of the flywheel, one ratchet plate with seventy two notches positioned such that the pawls can slide over the notches if the pawls turn in the normal direction of rotation, hydraulic shock absorbers attached to the ratchet plate, and retaining springs connected to the ratchet plate. As soon as the flywheel rotates in the reverse direction, one of the pawls drops into a notch in the ratchet plate, producing a rotation of the ratchet plate. Three hydraulic shock absorbers attached to the ratchet plate provide braking until the plate (and flywheel) come to a complete stop, thereby preventing rotation. Three retaining springs connected to the ratchet plate set back the assembly into the starting position after an operation. The shock absorbers are also equipped with an internal retaining spring for their piston.

The anti-reverse rotation device is rated at a torque equal to that which would occur upon energizing the motor at full voltage in the reverse direction plus the hydrostatic torque exerted on the pump at zero speed as per Equipment spec. ANO-E-108. The manufacturer has experience with this device in over 220 RCP motors and has also provided analysis which verifies the holding capability of the anti-reverse rotation device under these conditions.

The primary purpose of the anti-reverse rotation device is to prevent reverse rotation of the pump motor during normal operation with reverse flow through a non-operating pump. As noted above, analysis of this anti-reverse rotation device and experience assure the adequacy of the device for its intended service. There have been no tests conducted with conditions simulating a guillotine break at the pump suction. However, calculations have been made to determine the maximum theoretical speed and torque that the flywheel would attain assuming an inoperable anti-reverse rotation device and a guillotine break at the pump suction. These calculations show that the theoretical maximum torque for this condition far exceeds the torque necessary to shear the pump shaft. The torque necessary to shear the pump shaft has been calculated to be 2.5×10^3 ft-lbs. At the time that this torque is reached the absolute speed of the flywheel calculated for the pump suction break is less than the speed used in the fracture analysis previously described. Therefore, it is concluded for this postulated accident that the pump shaft would shear prior to attaining speeds which could result in flywheel damage.

4.3.6 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal system interconnected with the RCS is the steam and power conversion system. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the RCS. The turbine or electric driven emergency feedwater pump supplies feedwater to the steam generators and the resultant steam is vented to the atmosphere to provide required cooling. The analysis for this unlikely condition of total loss of station electric power is presented in Chapter 14.

4.3.7 SYSTEM INTEGRITY

The reactor protective system and the DROPS Diverse Scram System (Chapter 7) monitor parameters related to safe operation and trip the reactor to protect against RCS damage caused by high system pressure. The pressurizer code safety valves prevent RCS overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the RCS and the secondary system.

The integrity of the RCS is ensured by proper materials selection, fabrication quality control, design and operation. A summary of fabrication inspections for the components is given in Table 4-12. Components in the system are fabricated from materials initially having a low NDTT to eliminate the possibility of crack propagation. The reactor vessel material surveillance program described in Section 4.4.6 will provide a check on the predicted shift in NDTT. A complete stress analysis has been prepared for all design loadings specified in the design specification. The analysis shows that the reactor vessel, steam generator, pressurizer and pump casings comply with the allowable stress limits of Section III of the ASME Code and the requirements of the design specification. A similar analysis of the piping shows that it complies with the allowable stress limits of ANSI B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled.

As a further assurance of system integrity, the completed RCS will be hydrotested at 3,125 psig before initial operation. The primary and secondary sides of the replacement OTSGs are hydrostatically tested in the shop prior to shipment in accordance with the requirements of the ASME Code paragraphs NB-6200 and NC-6200, respectively.

The active components in the RCS which are classified as Class 1 mechanical components consist of the reactor coolant pumps with motors and the control rod drive mechanisms. These components are modeled in the seismic analysis of the RCS and the appropriate response spectra imposed at the support points of the major contributing components of the system. The resulting seismic accelerations of mass points are used to check bearing loadings and shaft deflections.

As a pump-motor shaft is designed to have a natural frequency at least 20 percent above the critical speed, the shaft is too stiff to respond to any of the lower seismic frequencies. The pump and motor bearings are designed to be capable of meeting the seismic design criteria.

The design specification for the control rod drives requires that the drives be capable of withstanding the seismic loadings within the stress limits for Class 1 equipment.

4.3.8 OVERPRESSURE PROTECTION

The RCS is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the reactor protective system, (2) pressure drop (static and dynamic) between the point of highest pressure in the RCS and the pressurizer, and (3) accident or transient overpressure conditions.

The combined capacity of the pressurizer code safety valves is based on the hypothetical case of withdrawal of a regulating control rod assembly bank from a relatively low initial power. The accident is terminated by high pressure reactor trip with resulting turbine trip. This accident condition produces a power mismatch between the RCS and secondary system larger than that caused by a turbine trip without immediate reactor trip, or by a partial load rejection from full load.

The steam generators are protected against overpressure by the main steam safety valves that are mounted on the steam lines. The capacity of these valves along with the required lift setpoints ensures that the maximum pressure on the secondary side of the plant does not exceed the allowable value. The limiting overpressure case is based on a turbine trip from full power conditions. No credit is taken for the ARTS and the reactor is tripped on high RCS pressure. This scenario produces the greatest pressure increase on the secondary plant. Consideration is also given for power operation with a number of MSSVs out of service.

In support of steam generator replacement, the turbine trip events were re-analyzed. The results of these analyses concluded that the peak pressure for the OTSGs and the main steam lines remained below 110% of their respective design pressures.

The results of the turbine trip analyses and that of the limiting RCS overpressure events are summarized in the AREVA/FANP Document 77-5018959-01, “ANO-1 Overpressure Protection Report Addendum.” This report has been prepared in accordance with the requirements of ASME Section III, NB-7220, 1989 Edition with no Addenda.

4.3.9 SYSTEM INCIDENT POTENTIAL

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Chapter 14.

The pressurizer spray line contains an electric motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this would prevent depressurization of the system to the saturation pressure of the reactor coolant. An electric motor-operated valve located between the pressurizer and the pressurizer electromatic relief valve can be closed to prevent pressurizer steam blowdown in the unlikely event the electromatic relief valve fails to re-close after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analysis in Section 14.2 indicates that one high pressure injection pump is sufficient to protect the core for an opening in the system considerably larger than one pressurizer code safety valve in the open position.

4.3.10 REDUNDANCY

Each heat transport loop of the RCS contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses (Chapter 8). The two pumps in each loop are fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the decay heat removal system. The high pressure injection lines are connected to the RCS on each of the four reactor coolant inlet pipes.

4.3.11 SAFETY LIMITS AND CONDITIONS

4.3.11.1 Maximum Pressure

The RCS serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the reactor building atmosphere. The safety limit of 2,750 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the RCS under the ASME Code, Section III.

4.3.11.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the RCS resulting from one percent defective fuel. Activity would be released to the environment if the reactor coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous activity could then be released to the environment by the vacuum pump on the main condenser through the Radwaste Area Vent or through the safety relief valves, if the plant should trip. In 10CFR20, Maximum Permissible Concentrations (MPC's) for continuous exposure to gaseous activity have been established. These MPC's will be used as the basis for maximum release of activity to the environment which has unrestricted access.

4.3.11.3 Leakage

RCS leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level and makeup tank water level over a time interval. All of these indications are recorded. The makeup tank capacity is 30.86 gallons per each inch of height and each graduation on the level recorder represents two inches of tank height. Instrumentation is also provided to measure the leakage from the reactor coolant pump seals.

The upper limit of reactor coolant identified leakage is 10 gpm as specified in Technical Specification 3.4.13.

The normal capacity of the reactor coolant makeup system is 300 gpm. The ratio of the maximum allowable leakage to the makeup capacity is 10/300 = 0.033. The normal capacity of containment water removal using an Auxiliary Building Sump Pump is 75 gpm. The ratio of the proposed maximum allowable leakage to the containment water removal normal capacity is 10/75 = 0.133.

Natural circulation can be maintained in the RCS for decay heat removal following a complete loss of unit power even if the system has been operating with an equipment leak. The natural circulation path will be maintained solid with water until the pressurizer has emptied, which is in excess of 6,000 gallons of coolant at normal power operating conditions. A 30 gpm leakage rate in conjunction with a complete loss of unit power and subsequent cooldown of the RCS by the atmospheric dump system and steam driven emergency feedwater pump would require a minimum of 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. Sixty minutes is ample time to restore electrical power to the plant and makeup flow to the RCS.

4.3.11.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources, i.e. pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280 °F, as the steam generator is the means for normal decay heat removal at temperatures above 280 °F.

A reactor coolant pump or decay heat removal pump is required to be in operation prior to reducing boron concentration by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

4.3.11.5 Combined Heatup, Cooldown, and Pressure Limitations

The stress level of the material in a reactor vessel, or any other component of the coolant system is a combination of stresses caused by internal pressures and temperature gradients. The maximum steady state stress resulting from gamma heating in the vessel is a relatively low value, and no problems are anticipated from thermal stresses in the reactor vessel wall. The initial NDTT for all reactor vessel material is based on Charpy V-notch tests. The DTT is defined as NDTT + 60 °F. NDTT, and subsequently DTT, increases as a function of cumulative neutron exposure.

To prevent brittle fracture during operation, stress levels are restricted to the following levels: (1) at DTT, the maximum stress permissible is 20 percent of the yield strength; and (2) this stress limit tapers off to 10 percent of design yield strength for DTT-200 °F, and remains at 10 percent below DTT-200 °F. Curves in the plant operating procedures define the operating limitation of pressure versus temperature to maintain stresses within the above levels. The predicted DTT shift due to irradiation exposure is monitored by the surveillance program. The operating limit curves will be revised if required by the results of the surveillance program testing.

4.3.12 QUALITY ASSURANCE

Assurance that the RCS will meet its design bases insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing. The quality program is described in Section 1.6.

4.3.12.1 Stress Analyses

Detailed stress analyses of the individual RCS components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the design bases.

A three-dimensional seismic, thermal, and dead load analysis has been performed on the primary coolant system to determine piping and component nozzle stresses. An idealized mathematical model for both loops (A and B) consisting of distributed and lumped masses connected by elastic members has been employed. The model includes the reactor, both steam generators, the pressurizer, all four coolant pumps with associated piping, and the secondary shield wall with any attached coolant system restraints or supports.

All flexibility calculations include the effects of torsional, shearing, bending, and axial deformations as well as changes in flexibility due to curved members and internal pressure. Flexibility factors have been calculated in accordance with ANSI B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The seismic analysis utilized the normal mode, response spectra approach with the frequencies and mode shapes being calculated by the modified Jacob technique. The inertia forces determined for each mode have been applied mathematically to the model and the resultant displacements, nozzle reactions, and internal forces and moments have been obtained by taking the square root of the sum of the squares of each mode's contribution. Closely spaced modes, those within 10%, are accounted for using the “Grouping Method” from Regulatory Guide 1.92.

A thermal expansion and dead load analysis has been performed using the same mathematical model as was used for the seismic analysis. At selected points within the loop the internal forces and moments from thermal expansion, dead loads, and seismic loadings have been combined appropriately in accordance with ANSI B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled. The resulting stresses have been combined with those resulting from pressure loading and thermal gradients. ANSI B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled, stress indices have been used on all applicable points. In addition, the effects of pressure and thermal cycling have been evaluated.

Independent thermal and dynamic analysis have been performed to ensure that piping connecting to the RCS is of the proper schedule and that it will not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA Schedule 160. The reactor coolant pump casing has been completely analyzed, separate from the loop, to ensure that the stresses throughout the casing are below the allowable stresses for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization.

4.3.12.2 Shop Inspection

Inspection and non-destructive testing of materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer are carried out for all of the RCS components and piping. The extent of these inspections and testing is listed in Table 4-12 for each for the components in the system. Shop testing of major components culminates with a hydrostatic test of each component followed by magnetic particle or liquid penetrant inspection of the component external surface. (Piping is hydrostatically tested in the field and will undergo the final inspection described in Section 4.3.12.3.)

Components are cleaned, packaged to prevent contamination, and shipped over a pre-selected route to the site. For materials purchased or manufactured outside of B&W, the results of the material inspection and testing program have been observed or audited by B&W. In addition there is an independent audit by B&W's Nuclear Power Generation Department Quality Assurance Section. For the replacement steam generators, for materials purchased or manufactured outside of AREVA, the results of the material and testing program are reported to AREVA on the supplier's Certified Material Test Report (CMTR) or End of Manufacturing Report, as applicable. The results are reviewed by AREVA to ensure compliance with the requirements stated on the purchase order for the material.

4.3.12.3 Field Inspection

Field welding of reactor coolant piping and piping connecting to nozzles is performed using procedures which will result in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds is identical to that performed on similar welds in the shop and is shown in Table 4-12.

Accessible shop and field welds and weld repairs in the reactor coolant piping are inspected by magnetic particle or liquid penetrant tests following the system hydrostatic test. For replacement reactor coolant piping, MT and PT examinations were performed prior to the RCS inservice leak tests.

4.4 TESTS AND INSPECTIONS

4.4.1 COMPONENT INSERVICE INSPECTION

The Reactor Coolant Pressure Boundary was designed so that provisions for access, as required by Sections IS-141 and IS-142 of Section XI of the ASME Boiler and Pressure Vessel Code (1970 Edition), could be followed as completely as possible.

Consideration has been given to the inspectability of the Reactor Coolant System (RCS) in the design of the components, in the equipment layout, and in the support structures to permit access for inspection. Access for inspection is defined as access for visual examination by direct or remote means and/or by contacting vessel surfaces during nuclear unit shutdown.

The reactor vessel is provided with stand off metal reflective insulation and removable internals, such that both outside and inside surfaces of the core region are accessible for inspection. The balance of the reactor vessel and the RCS components are provided with removable insulation to make the exterior surfaces accessible.

Reactor Vessel - Access for inspection of the reactor vessel will be as follows:

A. The vessel flange area and head can be inspected during refueling operations. All reactor internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the complete reactor vessel internal surface.

B. The closure head is stored dry during refueling to facilitate direct inspection. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling.

C. Inner surfaces of the vessel outlet nozzles can be inspected visually by remote means during refueling periods. The complete internal surface can be inspected by remote visual means following removal of the reactor core and vessel internals.

D. External surfaces of the vessel nozzle-to-piping welds can be inspected by remote visual means following removal of shielding and vessel insulation. All insulation on primary system components and piping is removable.

E. The external surface of the reactor vessel can be inspected during reactor lifetime if it should become necessary. An annulus has been provided between the reactor vessel and the primary shield to accommodate inspection requirements.

Pressurizer - The external surface will be accessible for surface and volumetric inspection. Manways allow access for internal inspection.

Steam Generator - The external surfaces of the steam generator are accessible for surface and volumetric inspection. The reactor coolant side of the steam generator can be inspected internally by remote visual means by removing the manway covers in the steam generator heads. Manways allow access for internal inspection.

Reactor Coolant Pumps - The external surfaces of the pump casings are accessible for inspection. The internal surface of the pump suction is available for inspection by removing the pump internals. The “C” RCP has a bore hole in the pump shaft for UT examination.

Piping - The reactor coolant piping, fittings, and attachments to the piping external to the primary shield are accessible for external surface and volumetric inspection.

Dissimilar Metal and Representative Welds - All dissimilar metal welds with the exception of Inconel-to-stainless steel pipe welds are made in the manufacturer's shops. All dissimilar metal welds are accessible for inspection during the service life of the nuclear unit.

Dissimilar metal welds on the reactor vessel include only the core flooding lines, incore instrumentation guide tubes, and control rod drive housings. Dissimilar metal welds in the piping include only attachments and extensions for welding to the Reactor Coolant Pump suction and discharge nozzles. Dissimilar metal welds in the pressurizer include the surge line, the relief valve, and spray line connections. Dissimilar metal welds on the steam generator occur only at the small drain lines and instrument attachments and between the steam outlet, main feedwater, and emergency feedwater nozzles at their safe ends.

Representative longitudinal and circumferential welds on the piping, pressurizer, and pump casing are inspectable as described above. Representative welds on the reactor vessel closure head are inspectable. Longitudinal and circumferential weld areas on the reactor vessel interior surfaces are inspectable. Circumferential welds on the steam generator are inspectable.

Valves - Testing of RCS valves is conducted for active Category 1 valves required for a safe reactor shutdown and for valves which perform a pressure isolation function under the definition of WASH-1400 Event V valve configuration valves. Testing of RCS valves required for a safe reactor shutdown is described in Section 4.4.4.1, "Active Category 1 Valve Tests."

The Reactor Safety Study, WASH-1400, identified in a PWR an intersystem Loss of Coolant Accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The designs examined in the WASH-1400 study were for cases where in-series check valves isolate the high pressure RCS piping from safety systems connected to the Reactor Coolant Pressure Boundary which have lower design pressures. Redundant isolation or check valves within the Class 1 boundary, which form the interface between the high and the low pressure systems, are therefore performing a pressure isolation function. The scenario which leads to the Event V accident is initiated by the failure of these valves to function as a pressure isolation barrier, causing an overpressurization and rupture of the low pressure piping which results in a LOCA that bypasses the containment. Identification of these valves, required tests, periodic surveillance and required records are specified in the Technical Specifications Section 3.4.14.

4.4.2 CONSTRUCTION INSPECTION

Historical data removed - To review exact wording please refer to Section 4.4.2 of the FSAR.

4.4.3 INSTALLATION TESTING

Historical data removed - To review exact wording please refer to Section 4.4.3 of the FSAR.

4.4.4 FUNCTIONAL TESTING

Prior to initial fuel loading, the functional capabilities of the RCS components were demonstrated at operating pressures and temperatures. Measurement of pressures, flows, and temperatures were recorded for various system conditions. Operation of Reactor Coolant Pumps, pressurizer heaters, pressurizer spray system, Control Rod Drive Mechanism, and other RCS equipment was demonstrated. For descriptions of the various functional tests performed, refer to Section 13.1.

4.4.4.1 Active Category 1 Valve Tests

The valves in the plant that are necessary for a safe shutdown and that are subjected to the most severe of operating conditions are the active, Category 1, Reactor Coolant Pressure Boundary valves, as defined by 10 CFR 50. In Unit 1, there are 15 such valves that meet this definition. They are required to actuate upon an engineered safeguards signal and to either isolate the reactor building or to open an engineered safeguard system flow path. These valves and their design conditions are listed in Table 4-23. A summary of these valves is as follows:

Valve	Size	Quantity	System
Letdown Cooler Outlet	2½"	2	Makeup and Purification
Letdown Isolation	2½"	1	Makeup and Purification
High Pressure Injection	2½"	8	Makeup and Purification

Valve	Size	Quantity	System
Makeup Isolation	2½"	2	Makeup and Purification
Low Pressure Injection	10"	2	Decay Heat Removal

NOTE: The following is historical information concerning active, category 1 RCS pressure boundary valves installed during initial construction. The four high pressure injection valves subsequently added by design change were manufactured in accordance with the latest NRC-approved edition of ASME Section III, 1986 edition with no addenda.

All of these valves receive extensive preoperational testing prior to initial fuel loading. The electric motor operators for these valves are Limitorque SMB series and Rotork NA1 series and have a history of qualification testing used to verify their reliability and operability. It is to be noted that, while they are all designed to RCS pressures and temperatures, their actual operating conditions are less and in the case of the low pressure injection valves an order of magnitude less. Extensive testing has been carried out by the valve operator manufacturer and by an independent institute. The testing has been done over a number of years and the most recent testing in Summer 1972 bears out the operability confirmed initially.

A. Manufacturer's Tests

1. Electric Motor Operator Qualification Testing Shock and Vibration Testing

   In August 1970, a Limitorque SMB Series operator was mounted on a test stand having a threaded valve stem driven by the operator simulating opening and closing a valve. The operator was electrically connected so as to stop at the full close position by means of a torque switch and stop at the full open position by means of a geared limit switch. The operator had a 4-train geared limit switch installed and all contacts not being used for motor control were wired to electric indicating lights at a remote panel.

The unit successfully completed a 5.3 G shock level at 32 Hz with no discrepancies noted. An exploratory scan of 5 Hz to 35 Hz was made and no critical resonant frequencies were noted on the operator. The unit was shocked and vibrated in each of three different axes a total of two minutes on, one minute off, three times per axis. The unit was operated electrically to both the full open and full close position and all torque switches and limit switches functioned properly. None of the auxiliary limit switches wired to indicating lights ever flickered or indicated they were opening. All electrical and mechanical devices on the operator worked successfully.

2. Heat Testing

In January 1969, a completely assembled and operational SMB series operator was placed in an oven where the temperature was maintained at approximately 325 °F for a duration of 12 hours.

The unit was electrically operated every 30 minutes for a period of approximately two minutes per cycle and the geared limit switches were used to stop the actuator at the full span and full closed position of travel. Indicating light circuits were also wired to the geared limit switches.

The test was successful in every respect. There were no malfunctions of the operator and upon inspection of the component parts used, there was no noticeable deterioration or wear.

3. Live Steam Testing

In January 1969, a complete SMB series operator was set up for electrical operation and live steam was piped into the conduit taps on the top of the limit switch compartment. One of the bottom conduit taps was left open to drain off any condensate. The operator was set up on a timer basis for operation over a period of approximately nine hours and operating every 30 minutes for two minutes per cycle. During this test, the live steam in the switch compartment seemed to have no effect whatever on the function of the limit switches in their control of the operator at the full open and full closed position of travel. In addition, the limit switches were wired up to indicating lights which operated satisfactorily.

The test was successful and there was no noticeable effect on the function of any of the parts in the limit switch compartment.

4. Life Cycle Testing

In January, 1969, the operator was mounted on a stand inside a test chamber and a 150-cycle load test was made on the unit. This test cycle consisted of stroking a $2^3/_8$ inch diameter valve stem at a speed of six inches per minute for a total of approximately 12 inches in two minutes. The valve stem in the full closed position produced a thrust of 16,500 pounds on a rigid plate securely bolted to the test chamber. The unit was wired so that the closing direction and the open position geared limit switch stopped the unit in the full open position.

After the life cycle testing was completed, the unit was inspected and found to be in excellent condition. There was no noticeable wear on any of the parts.

5. Simulated Accident Environment Testing

An electric motor operator was tested under conditions which simulated the temperature, humidity, and chemical environments that could be expected to exist in the containment following some postulated accident, such as the rupture of a major reactor coolant pipe.

The operator was placed in an Autoclave type chamber and subjected to 90 psig saturated steam. At specified intervals, the operator was cycled to assure proper operation. Forty minutes after the introduction of steam, a 1.5 percent boric acid solution was sprayed on the operator assembly. The operator continued to operate satisfactorily. Later, the steam pressure was periodically reduced to simulate post accident conditions. The boric acid spray was allowed to continue for four hours. The steam pressure was eventually reduced to 15 psig. The test continued for seven days. During this time, the operation of the operator became erratic. The corrosive effects of the steam and boric acid spray caused electrical contact malfunctions which were bypassed by the use of an appropriate jumper. The valve continued to cycle during the 7-day period.

A design change was made to the limit switch in order to correct the erratic operation and it was tested under similar accident conditions and found to operate satisfactorily. This design change has been incorporated into all subsequent applicable models of this operator.

B. Independent Testing

1. Franklin Institute Tests

More recent tests on a Limitorque SMB series operator in a simulated reactor containment post-accident environment were conducted during the summer of 1972 by the Franklin Institute Research Laboratories. In these tests, an operator was exposed to gamma radiation (200 megarads), a steam/chemical environment (for 12 days), a steam environment at temperatures going as high as 340 °F during the first day (test consisted of a 30-day exposure), and a seismic test similar to those in August, 1970. During all of these tests, as appropriate, the operator was periodically cycled and was found to operate satisfactorily.

C. Valve Specification

1. Valve Purchase Specification

In addition to a proven record as verified by the previous testing, the valve vendor must also comply with the purchase specification requirements. The purchase specifications for valves installed during initial construction required that they be hydrostatically tested, leak tested and cycled from the extremes of allowed movement of opened or closed. The hydrostatic test is in accordance with the Standard for Steel Pipe Flanges and Flanged Fittings (USAS B16.5). The leak test requires that, with the disc closed tight, hydrostatic pressure will be applied alternately on each side of the closed disc with the side opposite the pressure open for inspection. Valves will not show a leakage greater than 10 cubic centimeters per hour per inch of seat diameter or permanent deformation when

the valves are submitted to two times design pressure, except that the stress developed at test pressure will not exceed 90 percent of the specified minimum yield strength based on the minimum specified wall thickness.

The valve vendor is also required to submit calculations which show that, when the valve assembly is subjected to a 3g horizontal force and a 2g vertical force, stresses incurred are within the code allowable stresses. Verification has been provided which demonstrates that the first natural frequency is above 20 cps for all valves installed during initial construction.

2. Design Requirements for Installed Valves with Extended Top Works

a. Frequency of Top Works

The design requirements for installed valves are that the lowest natural frequency of the top works should be in the rigid range of the applicable spectrum response curve or greater than 20 cps. If the lowest natural frequency of the top works is not in the rigid range, then an appropriate safety factor should be applied to the seismic acceleration for the direction that the frequency is low in the valve seismic qualification calculation. The valve may also be physically changed to raise the lowest natural frequency above 20 cps.

b. Applied Acceleration

Seismic Category I valves shall be capable of performing their safety function during and after loading due to seismic forces. Specifically, valves having operators or similar features of extended proportions shall be able to withstand the loading associated with actual seismic accelerations in each direction as calculated in the piping stress calculations in addition to normal or accident operating loads as required by the valve design basis.

c. Design Stress Limits

The design stress limits for pressure retaining components shall be, as a minimum, in accordance with the code or record for Unit 1. The design stress limits for non pressure retaining components shall be designated by the analyst as to ensure that the component stays within the elastic region or be analyzed and qualified using accepted engineering practices.

D. Preoperational Testing

Testing procedures for valves that require operation to meet engineered safeguards requirements are quite extensive during the preoperational testing program. These tests demonstrate proper installation, strength, and functional performance of valves. Subsequent to satisfactory preoperational testing, surveillance testing requirements have been established to assure continued satisfactory operation of these valves. Furthermore, if maintenance or repair of these valves is required, appropriate functional testing will be accomplished to assure proper operation subsequent to the maintenance or repair.

1. System Electrical Test

   The purpose of these tests is to verify electrical characteristics of valve operators in performing their function. Preliminary checkout of the operator valve assembly requires that the valve be free to move and that if the motor operated valve travels in the wrong direction from its mid-travel position, its breaker must be tripped immediately as there would be no torque limit protection. The valve can be operated manually with a handwheel to ascertain its freedom of movement.

   The phase rotation of the operator is checked. During valve operation, verification that the valve travel and motor are stopped is done by closing the torque limit switch. Similarly, the opening of the valve is terminated by the opening of the limit switch.

   In checking the valve for engineered safeguards actuation, the valve is placed in the position opposite to its engineered safeguards position and then an engineered safeguards signal is simulated. The valve moves to its engineered safeguards position. Then the control room switch is turned to the opposite position of engineered safeguards operation and the valve is verified as remaining in its engineered safeguards position. Similarly, turning the circuit breaker panel switch to the opposite position of engineered safeguards operation has no effect on the valve.

   Acceptance criteria for these electrical tests are:

   a. Valves must open, close, and travel in the proper direction in response to control and engineered safeguards signals.

   b. The valve open and closed indicating lights must indicate correctly.

   c. Valve motor resistance-to-ground readings must be within specification.

   d. The specified valve travel time is within specification requirements.

2. System Engineered Safeguards Test

   The purpose of these tests is to demonstrate actual valve performance for its intended engineered safeguard use. Initially, all valves are placed in their non-engineered safeguard position prior to simulating an engineered safeguard signal. Upon initiation of an engineered safeguards signal, the tests for the subject valve demonstrate containment isolation and also emergency injection flow capability to the RCS from the Decay Heat Removal System's Low Pressure Injection (LPI) and the Makeup and Purification System's High Pressure Injection (HPI).

3. System Functional Testing

   The purpose of this testing is to verify that the valves perform as intended for normal operation. Cycling the valves under conditions of specified differential pressure and/or flow that may be encountered during plant operation will verify that the valve motor does not exceed maximum operating current and cycle time.

4. Integrated Engineered Safeguards Actuation Test

   The purpose of this test is to demonstrate the full operational sequence that would bring the Emergency Core Cooling Systems and the containment pressure reducing systems into action, including the transfer to alternate power sources.

   General acceptance criteria for this test are:

   a. The engineered safeguards systems operate as described in the FSAR.

   b. Upon actuation of an engineered safeguards signal, High Pressure and Low Pressure Injection to the RCS are supplied in accordance with FSAR requirements.

   c. Upon loss of normal station power, the engineered safeguards systems continue to perform their designed functions.

   Following completion of the preoperational test program and issuance of an operating license for the facility, these valves are functionally tested as required by the FSAR.

E. Technical Specifications

The Technical Specification testing requires that these valves be operated during an interval no longer than every three months to assure their continued availability. During the life of the facility, these valves will require maintenance, inspection, repair and possibly replacement or modification.

As a result of performing dynamic seismic analyses of piping systems which connect to the RCS, specific displacements and accelerations were determined for each of the 11 active, Category 1 RCS pressure boundary valves which are installed in the systems. The resultant displacements and accelerations for each of the active, Category 1 pressure boundary valve bodies are provided in Table 4-23 in addition to resultant displacements and accelerations for the valve operators. The maximum acceleration of 1.27g indicated for valve CV-1227 is considerably less than the maximum g force in either the horizontal or vertical direction that the 11 valves are required to withstand during testing. The entire scope of testing verifies the valves' operability from conditions of extreme duress to normal operation and the results of the earliest environmental, vibratory, and load testing have been repeatedly verified in later testing by independent research.

With respect to all safety related active valves in Unit 1, if any results of valve operability assurance programs developed by industry indicate design deficiencies in these valves resulting in an active valve not performing its intended design safety function, Entergy Operations, Inc. will incorporate appropriate design changes.

4.4.5 VIBRATION TESTING

To verify the integrity of the Reactor Coolant Pressure Boundary, it was monitored during preoperational functional testing of systems and components included within the boundary. Similarly, operation of systems or components not included within the pressure boundary, but which could induce a dynamic response, were observed. This testing confirms that any deflection or dynamic response is within acceptable levels and that the design of the piping restraints is satisfactory.

The following is a list of system operations which were monitored for this purpose:

A. Reactor Coolant System

Reactor coolant pump - start and stop
Pressurizer spray valves - open, shut during flow

B. Safety Injection System

High pressure and low pressure pumps - start and stop
Safety injection valves - open/shut while delivering and transfer from injection to recirculation mode.

C. Makeup and Purification System

Makeup pumps - start and stop
Letdown isolation valves - open, shut during flow

D. Decay Heat Removal System

Low pressure injection pumps - start and stop (system aligned for shutdown cooling)

Shutdown cooling line isolation valves and low pressure header (safety injection) valves - open/shut (system in operation).

Acceptance criteria for the piping requires that actual vibration displacements be compared with allowable displacements. Allowable displacements are those which produce stresses whose amplitudes are less than one-half the endurance limit of the material. The ASME B&PV Code Section III, 1971 Edition, Page 99, defines the endurance limit as "two times the Sa value at 10^6 cycles in the applicable fatigue curve."

In addition to detailed observations of system operations made by experienced startup engineers, direct displacement measurements were taken during main steam turbine stop valve closure and relief valve opening to verify acceptable displacements. Where past practice indicated the likelihood of transients, analyses were made to determine their effect on the system.

4.4.6 MATERIAL IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material were installed in the reactor vessel in accordance with ASTM Specification E-185, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. The types of specimens included in the irradiation capsules were Charpy V-notch (Type A) and tensile specimens which have been machined from welds, heat-affected zones, and base material used in the reactor vessel. Certified mill test reports for all heats of base metal were retained for the chemical elements as given in Table 4-17, "Reactor Vessel - Chemical Properties." There were a total of three Surveillance Specimen Holder Tubes (SSHTs) installed between the core and the inside wall of the reactor vessel shell. Refer to BAW-10006, "Reactor Vessel Material Surveillance Program," for a complete description of the initial surveillance program.

Following failure of the SSHTs at other Babcock & Wilcox designed plants, Arkansas Nuclear One - Unit 1 was shut down by the licensee on March 19, 1976 for a SSHT inspection. The inspection revealed that all of the SSHTs had suffered severe damage and that portions of two SSHTs had fallen to the bottom of the reactor vessel. To prevent further damage, all surveillance capsules and all parts of the SSHTs that had failed or were deemed likely to fail during the remainder of that operating cycle (Cycle 1) were removed from the vessel.

Since the discovery of the damage to the SSHTs, B&W, the reactor supplier, has undertaken the design, manufacture, and testing of an improved SSHT. SSHTs of this improved design are presently installed in Davis-Besse Unit 1 and Crystal River Unit 3. In addition, both of these reactors are of the same basic B&W 177 fuel assembly vessel design as Arkansas Nuclear One - Unit 1. The acceptability of the redesigned SSHTs has been demonstrated by a test program reviewed and approved by the staff and performed in conjunction with the hot functional test performed at Davis-Besse Unit 1.

Primarily to minimize the personnel radiation exposure which would be required to install the redesigned SSHTs in Arkansas Nuclear One - Unit 1, a modified and expanded specimen surveillance program was proposed and approved. This program utilizes the SSHT capacity at Davis-Besse Unit 1 and Crystal River Unit 3 and includes data sharing between B&W and other B&W reactor owners. Predictions of irradiation damage in B&W reactors, including Arkansas Nuclear One - Unit 1, will be based on Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”, May 1988.

The modified specimen surveillance program meets the requirements of 10 CFR 50, Appendices G and H. The program is described in detail in the latest revision of B&W Topical Report BAW-1543, Integrated Reactor Vessel Material Surveillance Program.

Permanent dosimetry hardware has been added to the reactor vessel cavity to further satisfy the requirements of 10CFR50, Appendix H, which requires each reactor in the integrated surveillance program to have an adequate dosimetry program.

4.5 REFERENCES

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BAW-10008, Part 1, Revision 1	Reactor Internals Stress and Deflection Due to Loss--of-Coolant Accident and Maximum Hypothetical Earthquake
BAW-10018	Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock
BAW-10027	Once-Through Steam Generator Research and Development Report
BAW-10035, Revision 1	Fuel Assembly Stress and Deflection Analysis Due to LOCA and Seismic Excitation
77-5018959-01	ANO-1 Overpressure Protection Report Addendum
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BAW-2075 Revision 1 (October 1989)	Analysis of Capsule ANI-C, Arkansas Power & Light Company, Arkansas Nuclear One, Unit 1, Reactor Vessel Material Surveillance Program
BAW-2308 Revision 1A and 2A	Initial RTNDT of Linde 80 Weld Materials

Table 4-1

REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure (psig)	Design Capacity lb/hr, total
Design Pressure	2,500	
Pressurizer Code Safety Valves	2,500*	600,000***
High Pressure Reactor Trip[(a)]	2,355*	
DROPS Diverse Scram System (DSS) High Pressure Reactor Trip	2,430**	
Pressurizer Electromatic Relief Valve[(a)]		
Normal Operation		
Open	2,450	
Close	2,395	
Low Temperature Overpressure Protection		
Open	400	
Close	350	
High Pressure Alarm[(a)]	2,255	
Pressurizer Spray Valve[(a)]		
Open	2,205	
Close	2,155	
Operating Pressure[(a)]	2,155	
Low Pressure Alarm[(a)]	2,055	
Low Low Pressure Alarm[(a)]	1,640	
Low Pressure Reactor Trip[(a)]	1,800*	
Hydrotest Pressure	3,125	

(a) At sensing nozzle on reactor outlet pipe.

* These valves are nominal; for exact values see the ANO Unit 1 Technical Specifications.

** Nominal value only

*** The Pressurizer Code Safety Valve Design Capacity was changed in the early 1980s by modification of the valve and ring settings. This valve modification and change in ring settings justified the change in the design capacity for each valve from 300,000 lb/hr to 324,000 lb/hr for a total design capacity of 648,000 lb/hr.

Table 4-2

REACTOR COOLANT SYSTEM COMPONENT CODES

Component	Codes	Addendum
Reactor Vessel	ASME III Class A	Summer 1967*
Reactor Vessel Closure Head	ASME III Class A	1989 Edition with no Addenda
Pressurizer	ASME III Class A	Summer 1967
Reactor Coolant System Piping	ANSI B31.7***	Errata through June 1968
R.C. Pump Casings	ASME III Class A (not code stamped)	1968 Edition with no Addenda
Safety and Relief Valves	ASME III Art. 9	Summer 1968
Welding Qualifications	ASME III and IX	Summer 1967
Steam Generator		
Primary Side	ASME III, Class 1	1989 Edition, no Addenda**,****
Secondary Side	ASME III, Class 2	1989 Edition, no Addenda**,****

* Welded joints tested in accordance with requirements of Article 7, Summer, 1966 Addenda.

** Repairs or modifications made to the steam generator primary pressure boundary shall meet all of the requirements for a Class 1 vessel of Article IWA-4000 of ASME B&PV Code, Section XI. Repairs or modifications made to the steam generator secondary pressure boundary shall meet at least the requirements for a Class 2 vessel of Article IWA-4000 of ASME B&PV Code, Section XI. Repairs or modifications made to non-pressure boundary parts of the steam generator shall be consistent with the intent of the requirements for Class 1 hardware contained in ASME B&PV Code, Section XI, Article IWA-4000 and good engineering practices.

*** Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

**** ASME Code Year for Alloy 690 material is 1998 Edition with 2000 Addenda.

Table 4-2A

CODE CASE INTERPRETATIONS

The following code case interpretations (special rulings) have been used in implementing the steam generator component codes delineated in Appendix A of the FSAR.

CODE CASE		TITLE
1.	N-725	Design Stress Values for UNS N06690 With a Minimum Specified Yield Strength of 35 ksi (240 MPa), Class 2 and 3 Components Section III, Division 1

The following code case interpretations have been used in implementing codes delineated in Appendix A of the FSAR.

CODE CASE		TITLE
1.	1340	Alternative Performance Qualification Section IX
2.	1425	Requirements for Piping Which Forms an Extension of a Containment Vessel Section III
3.	1426	Requirements for Electrical Penetration Assemblies and Other Appurtenances and Their Installation for Section III Class MC Vessels
4.	1427	Requirements for Valves Which Form an Extension of a Containment Vessel
5.	1451-1	Ultrasonic Examination for Electrical Penetrations and Attachment of Other Appurtenances to Section III Class MC Vessels
6.	N-416-1	Alternate Pressure Test Requirements and Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3. Section XI, Division 1.
7.	2142-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX (Applicable to all Sections, including Section III, Division 1, and Section XI).

Table 4-3

REACTOR VESSEL DESIGN DATA

Design Pressure, psig	2,500
Design Temperature, °F	650
Coolant Operating Temperature, Inlet/Outlet, °F	554/602**
Hydrotest Pressure, psig	3,125
Coolant Volume (Hot, Core and Internals in Place), ft^3	4,058
Reactor Coolant Flow, lb/hr	131.32×10^6**
Number of Reactor Closure Head Studs	60
Diameter of Reactor Closure Head Studs, in.	6-1/2
Vessel Dimensions	
Overall Height of Vessel and Closure Head, ft-in.	40-8-3/4
Shell ID, in.	171
Flange ID, in.	165
Straight Shell Minimum Thickness, in.	8-7/16
Shell Cladding Minimum Thickness, in.	1/8
Shell Cladding Nominal Thickness, in.	3/16
Vessel Insulation Thickness, in.	3
Closure Head Insulation Thickness, in.	5, except for center region of closure head dome insulation is 3¼
Closure Head Minimum Thickness, in.	6-5/8
Lower Head Minimum Thickness, in.	5

Vessel Nozzles

Function	No.	ID, in.	Material
Coolant Inlet	4	28	Carbon Steel, SS Clad
Coolant Outlet	2	36	Carbon Steel, SS Clad
Core Flooding - LPI	2	14 Sch 140	Carbon Steel[a], SS Clad

Table 4-3 (continued)

Vessel Nozzles (continued)

Function	No.	ID, in.	Material
Control Rod Drive	60	2.76	*Inconel[(b)]
Axial Power Shaping Rod Drive	8	2.76	Inconel[(b)]
In-Core Instrumentation	52	3/4 Sch 160	Inconel[(b)]
Vessel Level Probe	1	2.76	Inconel[(b)]/.2*

Dry Weight, lb.

Vessel	646,000
Closure Head	172,674
Studs, Nuts, and Washers	37,578

(a) With stainless steel safe end added after stress relief.

(b) With stainless steel flanges.

* Inconel is the registered trade name of an alloy manufactured by the International Nickel Company.

** Design values do not represent nominal operating parameters.

Table 4-4

STEAM GENERATOR DESIGN DATA
(Data Per Steam Generator)

Steam Conditions at Full Load, Outlet Nozzles	
Steam Flow, lb/hr	5.6×10^{6}*
Steam Temperature, °F	570*
Steam Pressure, psig	910
Feedwater Temperature, °F	457
Reactor Coolant Flow, lb/hr	65.66×10^{6}
Reactor Coolant Side	
Design Pressure, psig	2,500
Design Temperature, °F	650
Hydrotest Pressure, psig	3,125
Coolant Volume (hot), ft^3	2,026
Full Load Temperature Inlet/Outlet, °F	604/554
Secondary Side	
Design Pressure, psig	1,150
Design Temperature, °F	605
Hydrotest Pressure, psig	1,438
Net Volume, ft^3	3,666
Dimensions	
Tubes, OD/min Wall, in.	0.625/0.0368
Overall Height (Including Skirt), in.	881
Shell OD, in.	150.62
Shell Minimum Thickness, in.	3.31
Shell Minimum Thickness (at Tube Sheets and Feedwater Connect), in.	5.31
Tube Sheet Thicknesses, in.	24
Dry Weight, lb	1,005,000
Tube Length, ft.-in.	52-1-3/8

* Design values do not represent nominal operating parameters.

Table 4-4 (continued)

Nozzles - Reactor Coolant Side

Function	No.	ID, in.	Material
Inlet	1	36	Carbon Steel - SS Clad
Outlet	2	28	Carbon Steel - SS Clad
Manways	2	18	Carbon Steel - SS Clad
Handholes	6	8	Carbon Steel - SS Clad

Nozzles - Secondary Side

Function	No.	ID, in.	Material
Steam	2	24****	Carbon Steel
Vent	1	1-1/2 Sch 80	Carbon Steel
Drains	6	1-1/2 Sch 80	Carbon Steel
Drain	2	1 Sch 80	Carbon Steel
Level Sensing	24	1 Sch 80	Carbon Steel
Temperature Well	3	3/8	Inconel
Manways	2	16	Carbon Steel
Feedwater Connect	32	3 Sch 80	Carbon Steel or Inconel
Emergency Feedwater Connect	7***	3 Sch 80	Carbon Steel
Handholes	6	8	Carbon Steel
Orifice Plates	2 2	8 3	Carbon Steel
Inspection Ports	15	3	Carbon Steel

*** See Section 10.4.8.

**** Venturi throat diameter of 7.82 inches. Three venturi per nozzle for total venturi flow area of 1.0 ft^2.

Table 4-4A

INSTALLATIONS UTILIZING VELAN ENGINEERING CO. VALVES

Utility Company	Installation	Location
Carolina Power & Light Co.	Robinson	South Carolina
Conn. Yankee A.P. Co.	Yankee	Connecticut
Consolidated Edison	Indian Pt. 2	New York
Commonwealth Edison	Dresden	Illinois
Consumers Power Co.	Palisades	Michigan
Florida Power & Lt.	Turkey Pt. 3 & 4	Florida
Jersey Central Pwr. & Lt.	Oyster Creek	New Jersey
Niagara Mohawk	Nine Mile Point	New York
So. California Edison	San Onofre	California
T.V.A.	Browns Ferry 1 & 2	Alabama

TEST & RESEARCH FACILITIES	LOCATION
Aerojet General Corp.	San Ramon, California
Aerojet General Corp.	Sacramento, California
Argonne Research Corp.	Palos Park, Illinois
Argonne Research Corp.	Temont, Illinois
AEC Idaho Nuclear Corp.	Idaho Falls, Idaho
AEC Idaho	Scoville, Idaho
AEC & Tennessee Valley Authority	Oak Ridge, Tenn.
General Electric Co.	Pleasanton, California
General Electric, Atomic Power E.D.	San Jose, California
Knolls Atomic Power Lab.	Schenectady, New York
Westinghouse Electric Corp., Bettis Div.	Pittsburgh, Pa.
Westinghouse Electric Corp.	Ruffsdale, Pa.

Table 4-5

PRESSURIZER DESIGN DATA

Design/Operating Pressure, psig	2,500/2,166
Design/Operating Temperature, °F	670/648
Steam Volume, ft³	700
Water Volume, ft³	800
Hydrotest Pressure, psig	3,125
Electric Heater Total Design Capacity, kW (Includes non-vital, non-diesel backed heaters)	1,638
Dimensions	
Overall Height, ft.-in.	44-11-3/4
Shell OD, in.	96-3/8
Shell Minimum Thickness, in.	6.188
Dry Weight, lb.	291,000

Nozzles

Function	No.	ID, in.	Material
Surge Line	1	10 Sch 140	Carbon Steel, SS Clad[a]
Spray Line	1	4 Sch 120	Carbon Steel, SS Clad[b]
Safety Valve	2	3	Carbon Steel, SS Clad[a]
Relief Valve	1	2-1/2	Carbon Steel, SS Clad[a]
Vent	1	1 Sch 160	Inconel[d]
Sample	1	1 Sch 160	Inconel[d]
Temperature Well	1	1-1/2	Inconel[d]
Level Sensing	4	1 Sch 160	Inconel[c]
Heater Bundle	3	19-1/8	Carbon Steel, SS Clad
Manway	1	16	Carbon Steel, SS Clad
Temperature Well	1	1-1/2	Inconel[d]
Capped Tap	1	1 Sch 160	Inconel[d]

[a] With stainless steel safe end added after stress relief.

[b] With Inconel safe end.

[c] One of the upper Level Sensing nozzles has had a “Weld Buildup” nozzle repair (Reference LCP 90-5059 and Eng. Report 91-R-1017-01).

[d] Nozzles proactively repaired with Alloy 690 materials, OD weld pad, and groove weld per EC 12490.

Table 4-6

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping

Pipe ID, in.	28
Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
Minimum Thickness, in.	2-1/16
Coolant Volume (Hot - System Total), ft^3	1,085
Dry Weight, System Total, lb	214,000

Reactor Outlet Piping

Pipe ID, in.	36
Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
Minimum Thickness, in.	2-5/8
Coolant Volume (Hot - System Total), ft^3	979
Dry Weight, System Total, lb	200,000

Pressurizer Surge Piping

Pipe Size, in.	10, Sch 140
Design Pressure/Temperature, psig/°F	2,500/670
Hydrotest Pressure, psig	3,125
Coolant Volume, Hot, ft^3	20
Dry Weight, lb	5,000

Pressurizer Spray Piping

Pipe Size, in.	2-1/2, Sch 160
Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
Coolant Volume, Hot, ft^3	2
Dry Weight, lb	650

Nozzles

Function	No.	Nom. Dia., in.	Material
On Reactor Inlet Piping			
High Pressure Injection	4	2-1/2	Carbon Steel, SS Clad[(a)]
Pressurizer Spray	1	2-1/2	Stainless Steel
Drain/Letdown	1	2-1/2	Carbon Steel, SS Clad
Drain	3	1-1/2	Inconel
Pressure Sensing	1	1	Inconel
Temperature Well	4	1.125	Inconel
Mounting Boss	4	0.623	Inconel

Table 4-6 (continued)

Nozzles (continued)

Function	No.	Nom. Dia., in.	Material
On Reactor Outlet Piping			
Decay Heat	1	12	Carbon Steel, SS Clad
Vent	2	1	Alloy 690
Conn. on Flow Meters	4	1	Inconel
Pressure Sensing	4	1	Inconel
Temperature Well	2	1.40	Inconel
Mounting Boss	4	0.623	Inconel
Surge Line	1	10	Carbon Steel, SS Clad
Level Sensing	7	3/4	Inconel/Alloy 690
On Pressurizer Surge Piping			
Drain	1	1	Stainless Steel
On Pressurizer Spray Piping			
Auxiliary Spray	1	1-1/2	Stainless Steel
Spray Valve Bypass	1	1	Stainless Steel

[a] With stainless steel safe end added after stress relief.

Table 4-7

REACTOR COOLANT PUMP DESIGN DATA
(Data Per Pump)

Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	Per ASME III
rpm at Nameplate Rating	1,190
Developed Head, ft.	362
Capacity, gpm	88,000
Seal Water Injection, gpm	10
Controlled Bleedoff, gpm	1.5
Injection Water Temperature, °F	125
Cooling Water Temperature, °F	95
Pump Discharge Nozzle ID, in.	28
Pump Suction Nozzle ID, in.	28
Overall Height (Pump-Motor), ft.-in.	26-8 3/4
Dry Weight Without Motor, lb.	104,200
Coolant Volume, pump casing ft^3	98
Cooling Water Volume, ft^3	2.4
Pump-Motor Moment of Inertia, lb/ft^2	73,500 (A-C Motor) 68,605 (JI Motor)
Motor Data	
Type	Squirrel Cage Induction, Single-Speed, Water-Cooled
Voltage	6,900
Phase	3
Frequency, Hz	60
Insulation Class	F
Starting Current, amperes	3,800 (A-C Motor) 3,700 (JI Motor)
Power, HP (Nameplate)	9,000

Table 4-8

TRANSIENT CYCLES

Transient Number	Transient Description	(ASME Category)	Design Cycles
1A	Heatup from 70 °F to 8% Full Power	Normal	240[1]
1B	Cooldown from 8% Full Power	Normal	240[1]
1C	Technical Specification Cooldown	Emergency	10
2	Power Change 0 to 15% and 15 to 0%	Normal	1,440[2]
3	Power Loading 8% to 100% Power	Normal	18,000[3]
4	Power Unloading 100% to 8% Power	Normal	18,000[3]
5	10% Step Load Increase	Normal	8,000
6	10% Step Load Decrease	Upset	8,000
7	Step Load Reduction 100% to 8% Power	Upset	
	Resulting from Turbine Trip		160
	Resulting from Electrical Load Rejection		150
	Total		310
8	Reactor Trip	Upset	
	Type A		40
	Type B		160
	Type C		88
	Type D Trips Included in Transient Numbers 11, 15, 16, 17, and 21		112
	Total		400
9	Rapid Depressurization	Upset	40
10	Change of Flow	Upset	20
11	Rod Withdrawal Accident	Upset	40
12A	Hydrostatic Test of RCS (Except SGs)	Test	20
12B	Hydrostatic Test of Steam Generator A	Test	35
12C	Hydrostatic Test of Steam Generator B	Test	35
13	Deleted	-	-
14	Control Rod Drop	Upset	40
15	Loss of Station Power	Upset	40
16	Steam Line Break	Faulted	1
17A	Loss of Feedwater to One SG (per OTSG)	Upset	20
17B	Stuck Open Turbine Bypass Valve (per SG)	Emergency	10
18	Loss of Feedwater Heater	Upset	40
19	Feed and Bleed Operations	Normal	4,000
20	Miscellaneous A	Normal	30,000
	Miscellaneous B		20,000
	Miscellaneous C		4×10^6
	Miscellaneous D		40
	Miscellaneous E		1274[4]

Table 4-8 (continued)

Transient Number	Transient Description	(ASME Category)	Design Cycles
21	Loss of Coolant	Faulted	1
22A	High Pressure Injection	Upset	
	HPI Nozzle A (Note 1)		40
	HPI Nozzle B (Note 1)		40
	HPI Nozzle C (Note 1)		40
	HPI Nozzle D (Note 1)		40
22B	Core Flood Check Valve Test	Test	240
23	Steam Generator Filling, Draining, Flushing, and Cleaning	Normal	
	Steam Generator Secondary Side Filing		
	Condition 1		120
	Condition 2		120
	Steam Generator Primary Side Filling		
	Condition 1		120
	Condition 2		120
	Flushing		40
	Chemical Cleaning		20
24	RCP Restart with voids in RCS	Emergency	5
25	Refill of a Hot, Dry, Depressurized SG	Upset	50
26	Setting of Pressurizer Bypass Spray Flow	Test	10
27	Natural Circulation Cooldown	Upset	20

Notes

1. The replacement steam generators tube-to-tubesheet weld and the primary manway studs were designed for 180 and 140 cycles, respectively, but this is acceptable since they were installed in 2005 and were designed for more allowable cycles than remain for the original RCS components.
2. The replacement steam generators tube-to-tubesheet weld and the primary manway studs were designed for 720 cycles. This is acceptable since they were installed in 2005 and this is many more cycles than are actually expected.
3. The replacement steam generators tube-to-tubesheet weld and the primary manway studs were designed for 1800 and 700 cycles, respectively. This is acceptable since they were installed in 2005 and this is many more cycles than are actually expected.
4. The EFW nozzles for the replacement steam generators were designed for 1265 cycles of EFW actuation with > 40 °F EFW (Upset) and 9 cycles of 32 °F EFW (Emergency) for a total of 1274 cycles.

Table 4-8A

**TRANSIENT CYCLES
FOR TRANSIENT CYCLE LOGGING PROGRAM**

Transient Number	Transient Description	(ASME Category)	Design Cycles
	TESTS		
1	HPI Nozzle Test	Normal	40/Nozzle
2	Core Flood Check Valve Test	Normal	240
3	Bypass Spray Flow Test	Normal	10
4	Other RCS Test	Normal	None
	HEATUPS/COOLDOWNS		
5	Heatup	Normal	240[1]
6	Cooldown	Normal	240[1]
6A	Technical Specification Cooldown	Emergency	10
7	Rapid Cooldown	Upset	40
8	Natural Circulation Cooldown	Upset	20
9	SG Fill-Drain/Flush-Clean	Normal	240/40-20
	POWER LOADINGS & UNLOADINGS		
10	Reactor Runback	Upset	370
	SECONDARY SIDE TRANSIENTS		
11	Loss of MFW Heaters	Upset	40
	MISCELLANEOUS TRANSIENTS		
12	Hot, Dry, Depressurized SG Refill	Upset	25
13	RC Pump Restart with RCS Voids	Emergency	5
14	HPI Actuation	Upset	110/Nozzle
15A	EFW Actuation	Upset	1265
15B	EFW Actuation with 32 °F Water	Emergency	9
16	Other Non-Trip Transient	Upset	None
	REACTOR TRIPS		
17	Reactor Trip: Loss of RC Flow	Upset	80
18	Reactor Trip: Post Overcooling	Emergency	10
19	Reactor Trip: All Others	Upset	308

Note 1: The replacement steam generators tube-to-tubesheet weld and the primary manway studs were designed for 180 and 140 cycles, respectively, but this is acceptable since they were installed in 2005 and were designed for more allowable cycles than remain for the original RCS components.

Table 4-9

MATERIALS OF CONSTRUCTION

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-533, Grade B, Class 1[(a)]
	Pressure Forgings	A-508-64, Class 2 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel or Ni-Cr-Fe
	Studs, Nuts, and Washers	A-540, Grade B23 (Code Case 1335-2)
	Thermal Shield and Internals	SA-240, Type 304
	Guide Lugs	Ni-Cr-Fe, SB-168 (Code Case 1336)
Reactor Vessel Closure Head	Pressure Forging	SA-508 Class 3
	Cladding	308L, 309L Stainless Steel
Steam Generator	Pressure Forgings	SA-508, Class 3A; SA-105, (Safe ends, small nozzles)
	Cladding for Heads	309L/308L Stainless Steel
	Cladding for Tube Sheets	Ni-Cr-Fe UNS N06052, SFA-5.14
	Tubes	Ni-Cr-Fe, SB-163, Alloy 690
	Studs - Reactor Coolant Side	SA-193, Grade B7
	Nuts - Reactor Coolant Side	SA-194, Grade 7
	Studs - Secondary Side	SA-193, Grade B7
	Nuts - Secondary Side	SA-194, Grade 7
	Feedwater Headers	SA-182 F22
	Feedwater Risers and Elbows	SB-564 Alloy 690 SB-167 Alloy 690
Pressurizer	Shell, Heads, and External Plate	SA-212, Grade B
	Forgings	A-508-64, Class 1 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel
	Studs and Nuts	SA-320, Grade L43
	Internals Plate	SA-240, Type 304
	Internal Piping	SA-312, Type 304

Table 4-9 (continued)

Component	Section	Materials
Reactor Coolant Piping	28 in. and 36 in.	SA-516, Grade 70 (Elbows[b]); A-106, Grade C (Straights)
	Cladding	18-8 Stainless Steel or equivalent
	10 in. Surge Line and 2-1/2 in. Spray Line	A-403, Grade WP 316 (Elbows); A-376, Type 316 (Straights)
	Piping Safe Ends	A-376, Type 316 and Ni-Cr-Fe, SB-166
Reactor Coolant Pumps	Castings	
	Casing	A-351, Grade CF8M
	Stuffing Box	A-351, Grade CF8M
	Forgings	
	Shaft	A-461, Grade 660 Cond H1300, SA-638, Grade 660, or A182, FXM19
	Bolting	
	Casing Studs	A-540, Grade B23
	Casing Nuts	A-194, Grade 7
Valves	Pressure-Containing Parts	A-351, Grade CF8M A-182, F316, and F347

[a] This material is metallurgically identical to SA-302, Grade B, as modified by Code Case 1339.

[b] Straight section of pipe between the OTSG and hot leg elbow material was changed to SA-516 Gr. 70 during ANO-1 SG Replacement Project.

Table 4-10

DELETED

Table 4-11

DELETED

Table 4-12

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT
1. Reactor Vessel				
1.1 Forgings				
1.1.1 Flanges		X[(a)]		X
1.1.2 Studs, Bar		X		
1.1.3 Studs After Final Machining				X
1.1.4 Skirt Adapter		X[(a)]		
1.1.5 Nozzle Shell Forgings		X		X
1.1.6 Main Nozzle Forgings		X		X
1.1.7 Dutchman Forging		X[(a)]		X
1.1.8 CRD Mechanism Adapter		X	X	
1.1.9 CRD Mechanism Housing		X	X	
1.2 Plates				
1.2.1 Head and Shell Plate		X[(a)]		X[(+)]
1.2.2 Support Skirt		X[(a)]		X[(+)]
1.3 Instrumentation Tubes		X	X	
1.4 Closure O-Rings		X	X	
1.5 Weldments				
1.5.1 Longitudinal and Circumferential Main Seams	X			X
1.5.2 CRD Mechanism Adapter to Shell			X	
1.5.3 CRD Mechanism Adapter to Flange	X			
1.5.4 Main Nozzles	X			X
1.5.5 Instrumentation Nozzle Connection			X	
1.5.6 Nozzle Safe-Ends, Weld Deposit		X	X	

Table 4-12 (continued)

Component	RT	UT	PT		MT
1.5.7 Temporary Attachment After Removal					X
1.5.8 All Accessible Welds After Hydrotest			X	or	X
1.5.9 O-Ring Closure Weld	X		X		
1.5.10 Cladding, Sealing Surfaces		X(b)(+)	X		
1.5.11 Cladding, All Other		X(c)(+)	X		
1.5.12 Insulation Support Lugs					X
2. Steam Generator					
2.1 Tube Sheet					
2.1.1 Forging		X			X
2.1.2 Cladding		X(b)(g)	X		
2.2 Heads					
2.2.1 Forging		X			X
2.2.2 Cladding		X(b)(g)	X		
2.3 Shell					
2.3.1 Forging		X			X
2.4 Tubes*		X			
2.5 Nozzles and Elbows (Forgings)		X			X
2.6 Studs, Bar		X(f)	X(h)		
2.7 Inspection Ports and Secondary Manway (Forgings)		X			X
2.8 Weldments					
2.8.1 Shell, Circumferential	X	X	X		
2.8.2 Cladding		X(b)	X		
2.8.3 Nozzle to Shell	X	X	X		X
2.8.4 Level Sensing and Drain Connections, Partial Penetration and Drain			X		

Table 4-12 (continued)

Component	RT	UT	PT	MT
2.8.5 Instrument and Drain Connections, Full Penetration		X	X	
2.8.6 Support Skirt		X	X	X
2.8.7 Tube-to-Tube Sheet(d)			X	
2.8.8 Temporary Attachment After Removal				X
2.8.9 After Hydrostatic Test (All Accessible Welds)			X	
2.8.10 Inspection Port & Secondary Manway to Shell	X	X	X	
2.8.11 Feedwater Header/Riser Welds	X		X	
2.9 Feedwater Headers & Branch Connections (Forgings)			X	X
2.10 Feedwater Risers and Elbows			X	
2.11 Feedwater Nozzles		X	X	
3. Pressurizer				
3.1 Heads				
3.1.1 Plate		X(a)	X	
3.1.2 Cladding		X	X(c)(+)	
3.2 Shell				
3.2.1 Forging		X(a)		X
3.2.2 Plate		X(a)		X(+)
3.2.3 Cladding		X(c)(+)		X
3.3 Heater Bundles				
3.3.1 Cover Plate		X		X
3.3.2 Diaphragm and Spacer Plate		X	X	
3.3.3 Studs, Bar		X		
3.3.4 Studs and Nuts After Final Machining				X

Table 4-12 (continued)

Component	RT	UT	PT	MT
3.3.5 Heaters				
3.3.5.1 Tubing		X	X[+]	
3.3.5.2 Positioning of Heater Element in Tube	X			
3.4 Nozzle (Forgings)		X		X
3.5 Weldments				
3.5.1 Shell, Longitudinal as Deposited by Submerged Arc	X			X
3.5.2 Shell, Longitudinal as Deposited by Electroslag	X	X		X
3.5.3 Shell, Circumferential	X			X
3.5.4 Cladding, Sealing Surfaces		X[b][+]	X	
3.5.5 Cladding, All Other	X	X[c][+]	X	
3.5.6 Nozzle to Shell	X			X
3.5.7 Nozzle Safe-Ends (If Weld Deposit)	X	X		
3.5.8 Nozzle Safe-End (If Forging or Bar)	X			X
3.5.9 Instrumentation and Vent Connections			X	
3.5.10 Support Brackets				X
3.5.11 Heater Guide Tube Pad		X	X	
3.5.12 Temporary Attachment After Removal				X
3.5.13 All Accessible Welds After Hydrotest				X
3.5.14 Nozzle to Safe-End Weld Overlays		X[j]	X[j]	
4. Piping				
4.1 Pipe				
4.1.1 Forgings		X[a]		X
4.1.2 Cladding		X[c][+]	X	
4.1.3 Plate[i]		X[a]		X[+]
4.2 Bends				
4.2.1 Plate		X[a]		X[+]
4.2.2 Cladding		X[c][+]	X	
4.3 Nozzle Forgings		X		X

Table 4-12 (continued)

Component	RT	UT	PT	MT
4.4 Weldments				
4.4.1 Longitudinal	X			X
4.4.2 Circumferential	X	X[j]	X[j]	X
4.4.3 Cladding, Elbows		X[c][+]	X	
4.4.4 Cladding, Straight		X[c][+]	X	
4.4.5 Nozzles to Run Pipe	X			X
4.4.6 Thermowell Connections			X	
5. Reactor Coolant Pumps				
5.1 Castings	X		X	
5.2 Forgings		X	X	
5.3 Weldments				
5.3.1 Circumferential	X			X
5.3.2 Piping Connections				X

* An additional eddy current test is also performed which meets the requirements of ASME Code NB-2500 and the EPRI Guidelines for PWR Steam Generator Tubing Specification and Repair for Alloy 690 Steam Generator Tubing (EPRI No. TR-106743-V2R1).

(a) 100% scanning for longitudinal wave technique and 100% shear wave technique.

(b) UT of cladding defects and bond to base metal.

(c) UT of cladding bond to base metal (spot check).

(d) Also gas leak test – Framatome ANP requirement.

(e) Over 12-inch length on each end.

(f) UT not required for studs 2" and under. PT/MT not required for studs 1" and under. Either MT or PT may be used.

(g) UT for underclad cracking.

(h) Performed on finished diameter prior to threading.

(i) Straight section of pipe between the OTSG and hot leg elbow material was changed to SA-516 Gr. 70 during ANO-1 SG Replacement Project.

(j) UT and PT for structural weld overlays performed during 1R20 and 1R21.

(+) Additional B&W requirement.

Note: RT: Radiographic
UT: Ultrasonic
PT: Dye Penetrant
MT: Magnetic Particle

Table 4-13

Historical data removed

To review exact wording please refer to Table 4-13 of the FSAR.

Table 4-14

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL

Area		Stress Intensity, psi	Allowable Stress 3 S_m, psi (Operating Temperature)
Control Rod Housing	Low Alloy:	49,300	80,100
	High Alloy:	45,800	69,900
Head Flange		54,600	80,100
Vessel Flange		43,000	80,000
Closure Studs		89,400	107,400
Primary Nozzles			
Inlet		24,000	80,000
Outlet		24,000	80,000
Bottom Head to Shell		23,300	80,000
Bottom Instrumentation		10,100	69,900
Nozzle Belt to Shell		32,300	80,000
Core Flooding Nozzle		23,660	80,000
Support Skirt		88,000	93,700

Note: Locations or points of stress analysis are illustrated in Figure 4-14.

The actual stress intensity information contained in the table above is historical. For current stress information, see the latest qualifying stress analysis calculation(s).

Table 4-15

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL (Figure 4-12)

Item	Usage Factor[(a)]
RV Head Flange	0.03
RV Closure Studs	0.813
CRDM Housing	0.802
CRDM Flange	0.00
CRDM Motor Tube Flange	0.00
CRDM Mtg Flange Bolts	0.13
CRDM Nozzles	0.659
RV Inlet Nozzle	0.109
RV Outlet Nozzle	0.518
RV Lower Head	0.007
RV Support Skirt	0.05
Core Flood Nozzle	
Safe End/Nozzle Weld	0.012
Nozzle/Shell Juncture	0.502
CFN Venturi	0.802
RV Instrument Nozzles	0.18
RV Instrument Nozzle Welds	0.57
SSS Mounting FLG SEG	0.16
RV Shell	0.023

[(a)] As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

"The actual resultant cumulative fatigue usage information contained in the table above is historical. For current fatigue usage information, see the latest qualifying fatigue usage analysis calculation(s)."

Table 4-16

REACTOR VESSEL - PHYSICAL PROPERTIES

Item	Heat No.	Ultimate Strength, 10^3psi	Yield Strength, 10^3 psi	Elongation Range, %	Impact Test Temp, °F	Impact Values
Upper Shell Course	C-5114-2	92.75	72.0	24.2	+10	34-34-35
Upper Shell Course	C-5120-2	91.0	71.0	25.0	+10	67-63-59
Bottom Shell Course	C-5114-1	94.75	73.5	25.8	+10	44-34-25
Bottom Shell Course	C-5120-1	90.25	70.5	25.8	+10	49-41-21
Bottom Head	C-5025-3	94.0	70.0	28	+10	55-55-45
Upper Shell Flange	2V-986	91.0	66.5	26	+10	85-82-66
Core Flood Nozzle	7-9186	91.75	69.25	25	+10	51-56-43
Core Flood Nozzle	7-9186	86.78	65.0	25	+10	41-35-57
Inlet Nozzle	121WO15VA1	87.5	77.0	25.7	+10	93-100-94
Inlet Nozzle	121W015VA2	92.0	69.5	26.5	+10	110-105-100
Inlet Nozzle	122W119VA2	84.75	66.5	27.3	+10	110-115-95
Inlet Nozzle	122W119VA1	86.0	60.0	24.2	+10	94-138-94
Outlet Nozzle	124W479VA1	91.5	73.0	26.6	+10	97-122-115
Outlet Nozzle	124W479VA2	92.0	74.5	26.5	+10	96-114-111
Closure Head	02W28-1-1	88.2	67.6	28.8	+10	73-87-86

Table 4-16 (continued)

Item	Heat No.	Ultimate Strength, 10^3psi	Yield Strength, 10^3 psi	Elongation Range, %	Impact Test Temp, °F	Impact Values
Closure Studs	129848	154.265	136.9	17	+10	35-40-50
Head Transition Piece	25W609	99.5	75.5	24.5	+10	61-95-66
Upper Nozzle Shell Course	528420	90.0	71.3	26.0	+10	65-76-80
Lower Nozzle Shell Course	C-5120	94.4	75.0	24	+10	55-48-54

Table 4-17

REACTOR VESSEL - CHEMICAL PROPERTIES
(Element)

Heat No.	C	Mn	P	S	Si	Ni	Mo	Co	V	Cr	Cu
C-5114-2	0.21	1.32	0.010	0.016	0.20	0.52	0.57	0.012	--	--	0.15
C-5120-2	.22	1.41	.014	.013	.18	.55	.53	.012	--	--	.17
C-5114-1	.21	1.32	.010	.016	.20	.52	.57	.012	--	--	.15
C-5120-1	.22	1.41	.014	.013	.18	.55	.53	.012	--	--	.17
C-5025-3	.22	1.32	.010	.016	.19	.53	.55	--	--	--	--
2V-986	.27	0.68	.009	.017	.22	.74	.61	.01	0.05	0.29	.17
7-9186	.20	.68	.012	.011	.35	.80	.57	.01	--	.35	--
121W015VA1	.22	.65	.01	.010	.25	.69	.57	.014	.03	.40	--
121W015VA2	.18	.64	.01	.008	.21	.69	.57	.012	.01	.39	--
122W119VA2	.23	.64	.01	.009	.34	.75	.59	.010	--	.38	--
122W119VA1	.22	.64	.01	.009	.31	.75	.59	.01	--	.38	--
124W479VA1	.14	.67	.01	.008	.22	.76	.56	.015	.01	.36	.09
124W479VA2	.19	.71	.01	.013	.18	.78	.58	.015	.01	.35	.06
02W28-1-1	.18	1.48	0.004	0.001	.18	.88	.49	0.02	< 0.003	0.12	0.04
129848	.38	0.72	.014	.018	.20	1.69	.29	--	--	--	--
25W609	.23	.67	.01	.010	.29	0.76	.59	.010	0.02	.36	--
528420	.22	0.64	.01	.02	.22	.74	.65	.02	--	0.35	--
C-5120	0.22	1.41	0.014	0.013	0.18	0.55	0.53	0.012	--	--	0.17

Table 4-18

STRESSES DUE TO THE MAXIMUM FAULTED STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL

Stress	Computer Value	Allowable Value
Primary Membrane	24,000 psi	63,000 psi 0.7 S_u
Primary Membrane Plus Primary Bending	41,000 psi	94,500 psi 1.05 S_u

The actual stress intensity information contained in the table above is historical. For current stress information, see the latest qualifying stress analysis calculation(s).

Table 4-19

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES FOR THE MAXIMUM PREDICTED STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL

Component Parts	Stress Ratio
Primary Head including Primary Head Tube Sheet Joint	1.49
Tubes	1.7
Tube Sheet Ligament	1.26

The actual stress intensity information contained in the table above is historical. For current stress information, see the latest qualifying stress analysis calculation(s).

Table 4-20

STEAM GENERATOR USAGE FACTORS

Item	Usage Factor
SG Support Skirt-to-Head	0.72
Tubesheet Analysis	
Nominal Ligament	0.3
Minimum Ligament	0.73
SG Primary Inlet Nozzle	0.2
SG Primary Outlet Nozzle	0.63
SG Primary Outlet Nozzle Dam Ring	
Fillet Weld/Cladding	0.854
MFW Nozzle	0.68
Shell at MFW Nozzle	0.46
Studs	0.16
EFW Nozzle	0.55
Shell at EFW Nozzle	0.34
EFW Nozzle Studs	0.06
Steam Outlet Nozzle	0.69
SG Shell	0.72

Table 4-21

PRESSURIZER USAGE FACTORS (Figure 4-14)

Item	Usage Factor
PZR External Supports	0.02
PZR Shell	0.1
PZR Surge Nozzle Safe End-to-Elbow Weld	0.41
PZR Surge Nozzle-to-Head Juncture	0.35
PZR Spray Nozzle	0.053
Head	0.0
Nozzle Safe End	0.002
Internal Piping - Bimetallic Weld	0.354
Heater Bundle Closure	
PZR HBC Cover Plate	0.057
Stud	0.3
Heater-to-Tube Weld	0.05
Diaphragm Plate	0.60
Seal Weld	0.86
Level-Sensing Nozzle	E
Vent Nozzle	E
Sampling Nozzle	E
Manway	E
Thermowell Nozzle	E
2 1/2" Pressure Relief Nozzles	E

E - Exemption from fatigue analysis requirements are met.

Note: Points of stress analysis are illustrated in Figure 4-14.

"The actual resultant cumulative fatigue usage information contained in the table above is historical. For current fatigue usage information, see the latest qualifying fatigue usage analysis calculation(s)."

Table 4-21a

REACTOR COOLANT PIPING USAGE FACTORS (Figures 4-2 and 4-3)

Item	Usage Factor
Hot Leg Piping	0.145
Cold Leg Piping	0.2621
Decay Heat Nozzle	
Nozzle-to-Hot Leg	0.701
Pipe-to-Nozzle End	0.356
Surge Line	
Non-Elbows	0.38
Elbows	0.40
Hot Leg	
Nozzle-to-Surge Line Weld	0.198
End of Nozzle Taper	0.618
HPI Nozzles	
Nozzle A	0.76
Nozzle B	0.802
Nozzle C	0.802
Nozzle D	0.582
Cold Leg Spray Nozzle	0.007
Spray Line	0.40
Aux. Spray Check Valve	0.50
Sch. 160 Drain Nozzles	
P32C	0.323
P32D	0.499
Let Down Nozzle	
Pipe Weld SS	0.7606
Pipe Weld CS	0.4464
Branch Connection in RC Cold Leg	0.4727
Hot Leg Level Tap Ass.	
Welds	0.013
Branch Intersections	0.15
RC Pump Casing	0.3158
RC Pump Cover	0.56

“The actual resultant cumulative fatigue usage information contained in the table above is historical. For current fatigue usage information, see the latest qualifying fatigue usage analysis calculation(s).”

Table 4-22

REACTOR COOLANT TRAMP URANIUM ACTIVITY

Isotope	μCi/ml
Kr-85m	3.20(-3)
Kr-85	1.28(-4)
Kr-87	6.05(-3)
Kr-88	9.11(-3)
Xe-131m	4.52(-6)
Xe-133m	1.76(-4)
Xe-133	7.35(-3)
Xe-135m	3.24(-3)
Xe-135	1.15(-2)
Xe-138	1.14(-2)
I-131	5.66(-4)
I-132	9.50(-3)
I-133	7.35(-3)
I-134	1.88(-2)
I-135	1.08(-2)

One percent of the above listed iodine activities can be assumed to be in the particulate form.

Table 4-23

ACTIVE - CATEGORY I - REACTOR COOLANT PRESSURE BOUNDARY VALVES

Bechtel Number	System	Service	Size	Purchased by	Nuclear Class	System Design Rating	Seismic Class	System Conditions During Operation	Type	Motor Operator Type	Valve Mfr.
CV-1214	Make-up & Purification	Letdown Cooler Outlet	2 1/2"	B&W	I CCA	2500 psig 650 °F	I	2170 psig 135 °F 40-160 gpm	Gate	Rotork 14NA1	Anchor-Darling
CV-1216	Make-up & Purification	Letdown Cooler Outlet	2 1/2"	B&W	I CCA	2500 psig 650 °F	I	2170 psig 135 °F 40-160 gpm	Gate	Rotork 14NA1	Anchor-Darling
CV-1221	Make-up & Purification	Letdown line RB isolation	2 1/2"	EOI	II CCB	2500 psig 600 °F	I	2170 psig 120 °F 40-160 gpm	Gate	Limitorque SMB-00-10	Anchor-Darling
CV-1219	Make-up & Purification	High Press Inj. RB isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2220-2950 psig 120-245 °F 300 gpm	Globe	Limitorque SMB-0-25	Anchor-Darling
CV-1220	Make-up & Purification	High Pres Inj. RB isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200-2950 psig 120-245 °F 300 gpm	Globe	Limitorque SMB-0-25	Anchor-Darling
CV-1227	Make-up & Purification	High Pres Inj. RB isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200-2950 psig 120-245 °F 300 gpm	Globe	Limitorque SMB-0-25	Anchor-Darling

Table 4-23 (continued)

Bechtel Number	System	Service	Size	Purchased by	Nuclear Class	System Design Rating	Seismic Class	System Conditions During Operation	Type	Motor Operator Type	Valve Mfr.
CV-1228	Make-up & Purification	High Pres Inj. RB isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200-2950 psig 120-245 °F 300 gpm	Globe	Limitorque SMB-0-25	Anchor-Darling
CV-1234	Make-up & Purification	M-U RB Isolation	2 1/2"	B&W	II CCB	3050 psig 250 °F	I	2200-2950 psig 120-245 °F 13-50 gpm	Gate	Limitorque SMB-00-10	Anchor Darling
CV-1400	Decay Heat	L.P. Inj. RB isolation	10"	B&W	II CCB	2500 psig 300 °F	I	255 psig 280 °F 3000 gpm	Gate	Limitorque SMB-3	Velan
CV-1401	Decay Heat	L.P. Inj. RB isolation	10"	B&W	II CCB	2500 psig 300 °F	I	255 psig 280 °F 3000 gpm	Gate	Limitorque SMB-3	Velan
CV-1233	Make-up & Purification	M-U RB Isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2170 psig 155 °F	Gate	Limitorque SMB-00	Anchor-Darling
CV-1278	Make-up & Purification	HPI RB isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200 psig 120 °F	Globe	Limitorque SMB-0	Anchor-Darling
CV-1279	Make-up & Purification	HPI RB Isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200 psig 120 °F	Globe	Limitorque SMB-0	Anchor-Darling
CV-1284	Make-up & Purification	HPI RB Isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200 psig 120 °F	Globe	Limitorque SMB-0	Anchor-Darling
CV-1285	Make-up & Purification	HPI RB Isolation	2 1/2"	EOI	II CCB	3050 psig 250 °F	I	2200 psig 120 °F	Globe	Limitorque SMB-0-25	Anchor-Darling

Table 4-23 (continued)

**SEISMIC INFORMATION FOR ACTIVE, CATEGORY 1
REACTOR COOLANT PRESSURE BOUNDARY VALVES**

VALVE OPERATOR

	Displacement (inches)						Effective Accelerations ("G")					
	X + Y - EQ			Y + Z - EQ			X + Y - EQ			Y + Z - EQ		
VALVE NO.	Dx	Dy	Dz	Dx	Dy	Dz	Ax	Ay	Az	Ax	Ay	Az
CV-1214	.581	.361	.262	[See	Note	4]	.609	.535	.593	[See	Note	4]
CV-1216	.573	.266	.244	[See	Note	4]	.597	.472	.576	[See	Note	4]
CV-1221	.024	.010	.074	[See	Note	4]	.553	.192	1.174	[See	Note	4]
CV-1219	.110	.014	.212	[See	Note	4]	.968	.236	1.363	[See	Note	4]
CV-1220	.232	.033	.358	[See	Note	4]	1.945	.328	1.423	[See	Note	4]
CV-1227	.214	.069	.328	[See	Note	4]	.747	.212	1.401	[See	Note	4]
CV-1228	.029	.000	.037	[See	Note	4]	.299	.146	.299	[See	Note	4]
CV-1233	2.051	.038	.020	[See	Note	4]	1.151	.570	.499	[See	Note	4]
CV-1234	Displacements for CV-1234 are given in Calculation 89-D-1005-36											
CV-1278	.268	.129	.437	[See	Note	4]	1.052	.657	.921	[See	Note	4]
CV-1279	.214	.121	.967	[See	Note	4]	1.012	.714	1.260	[See	Note	4]
CV-1284	.129	.166	.134	[See	Note	4]	.802	1.083	1.019	[See	Note	4]
CV-1285	.097	.089	.146	[See	Note	4]	.929	1.105	1.533	[See	Note	4]
CV-1400	.107	.278	.122	.102	.277	.113	.374	.209	.408	.334	.178	.495
CV-1401	.008	.017	.029	.008	.021	.098	.253	.237	.156	.182	.210	.566

Table 4-23 (continued)

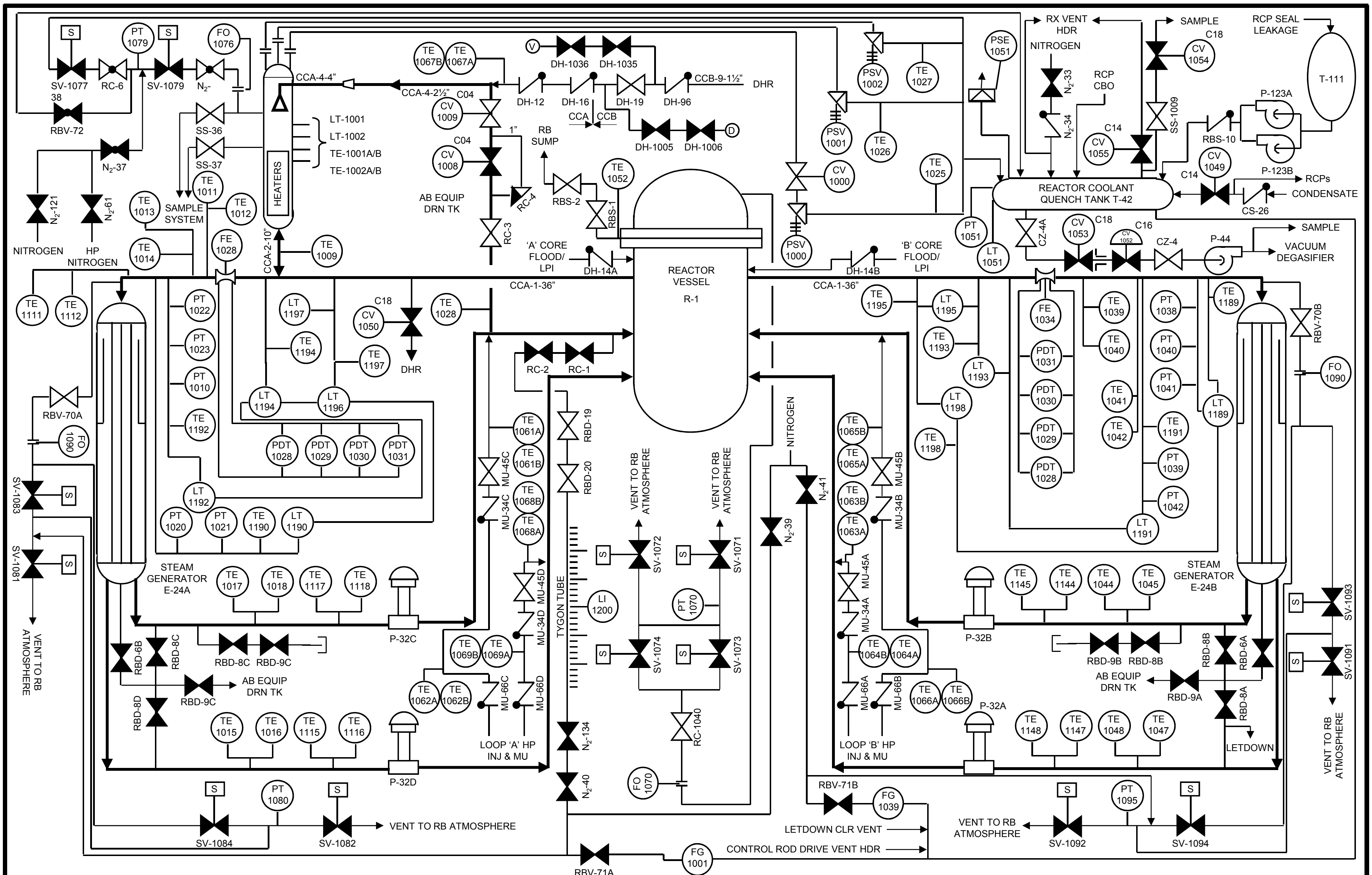
VALVE BODY

	Displacement (inches)						Effective Accelerations ("G")					
	X + Y - EQ			Y + Z - EQ			X + Y - EQ			Y + Z - EQ		
VALVE NO.	Dx	Dy	Dz	Dx	Dy	Dz	Ax	Ay	Az	Ax	Ay	Az
CV-1214	.554	.365	.214	[See	Note	4]	.560	.561	.352	[See	Note	4]
CV-1216	.555	.269	.115	[See	Note	4]	.554	.467	.243	[See	Note	4]
CV-1221	.015	.007	.060	[See	Note	4]	.418	.158	.849	[See	Note	4]
CV-1219	.052	.024	.102	[See	Note	4]	.301	.394	.606	[See	Note	4]
CV-1220	.122	.053	.181	[See	Note	4]	1.032	.537	.809	[See	Note	4]
CV-1227	.086	.117	.106	[See	Note	4]	.303	.270	.427	[See	Note	4]
CV-1228	.028	.000	.030	[See	Note	4]	.299	.146	.300	[See	Note	4]
CV-1233	2.050	.138	.118	[See	Note	4]	1.180	.359	.661	[See	Note	4]
CV-1234	.110	.023	.176	[See	Note	4]	1.665	.490	1.227	[See	Note	4]
CV-1278	.135	.121	.225	[See	Note	4]	.903	.596	1.068	[See	Note	4]
CV-1279	.107	.107	.802	[See	Note	4]	.790	.627	1.012	[See	Note	4]
CV-1284	.000	.053	.100	[See	Note	4]	.299	.698	.978	[See	Note	4]
CV-1285	.000	.041	.120	[See	Note	4]	.299	.600	.956	[See	Note	4]
CV-1400	.025	.278	.038	.02	.277	.031	.077	.209	.115	.611	.179	.087
CV-1401	.0001	.017	.018	.0001	.021	.039	.006	.236	.153	.005	.209	225

Table 4-23 (continued)

Notes:

1. Dx = Seismic displacement in X-direction (TYP)
2. Ax = Seismic Acceleration in X-direction (TYP)
3. Analyses are based on the "design earthquake" as defined in Section 5.1.4.1.
4. Where only one set of "X, Y, & Z" values are given above, they are the safe shutdown earthquake (SSE) values from a 3-D seismic analysis and include SAM (seismic anchor movements) displacements (if applicable).
5. Displacements and accelerations contained in the table above are historical. For current information, refer to the latest component qualification calculations.



REACTOR COOLANT SYSTEM		SAR FIGURE NO. 4-1		
		AMENDMENT 26		
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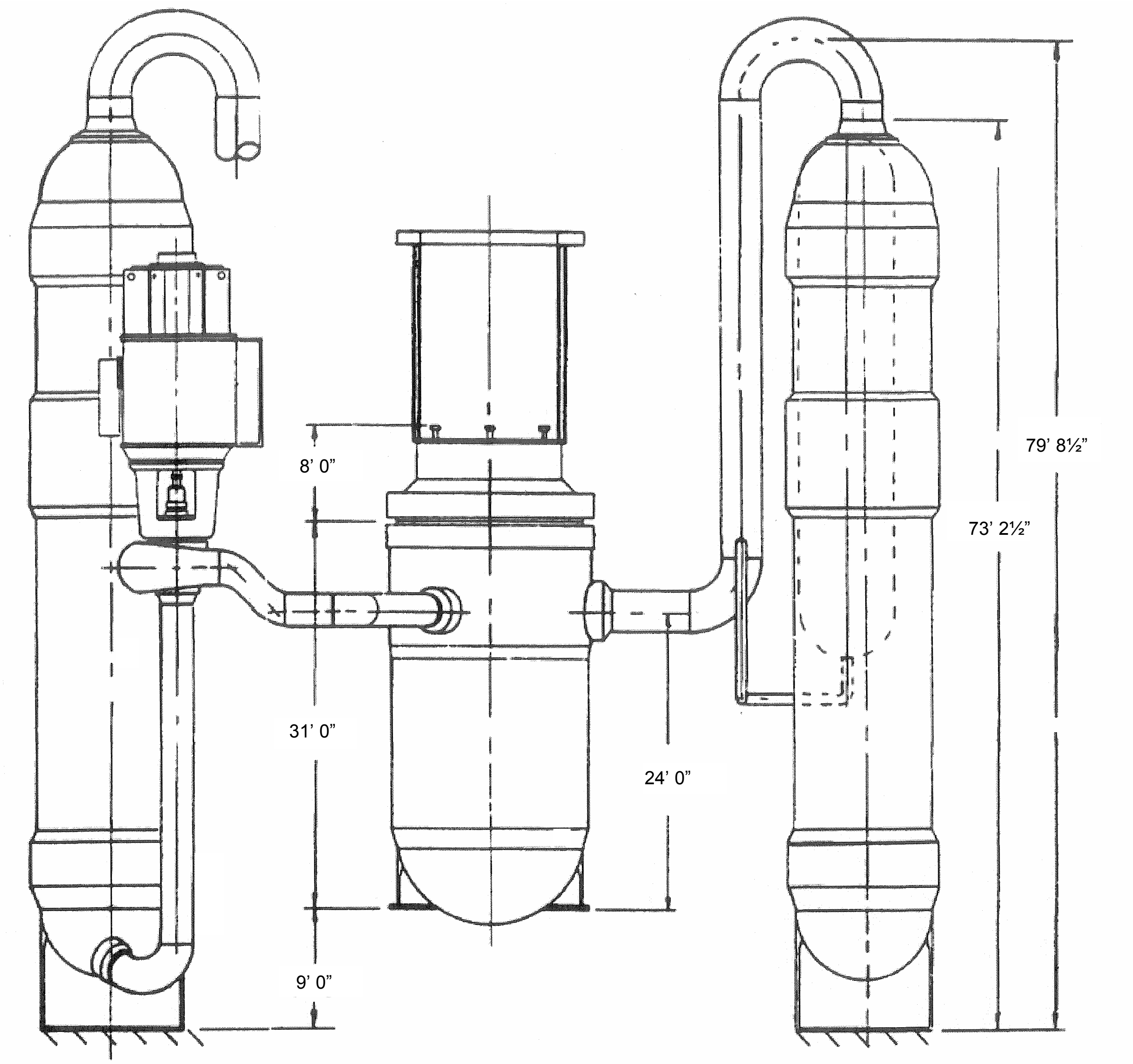
		DESIGN: ENTERGY		
		CAD NO: NONE		



SAR FIGURE NO. 4-2

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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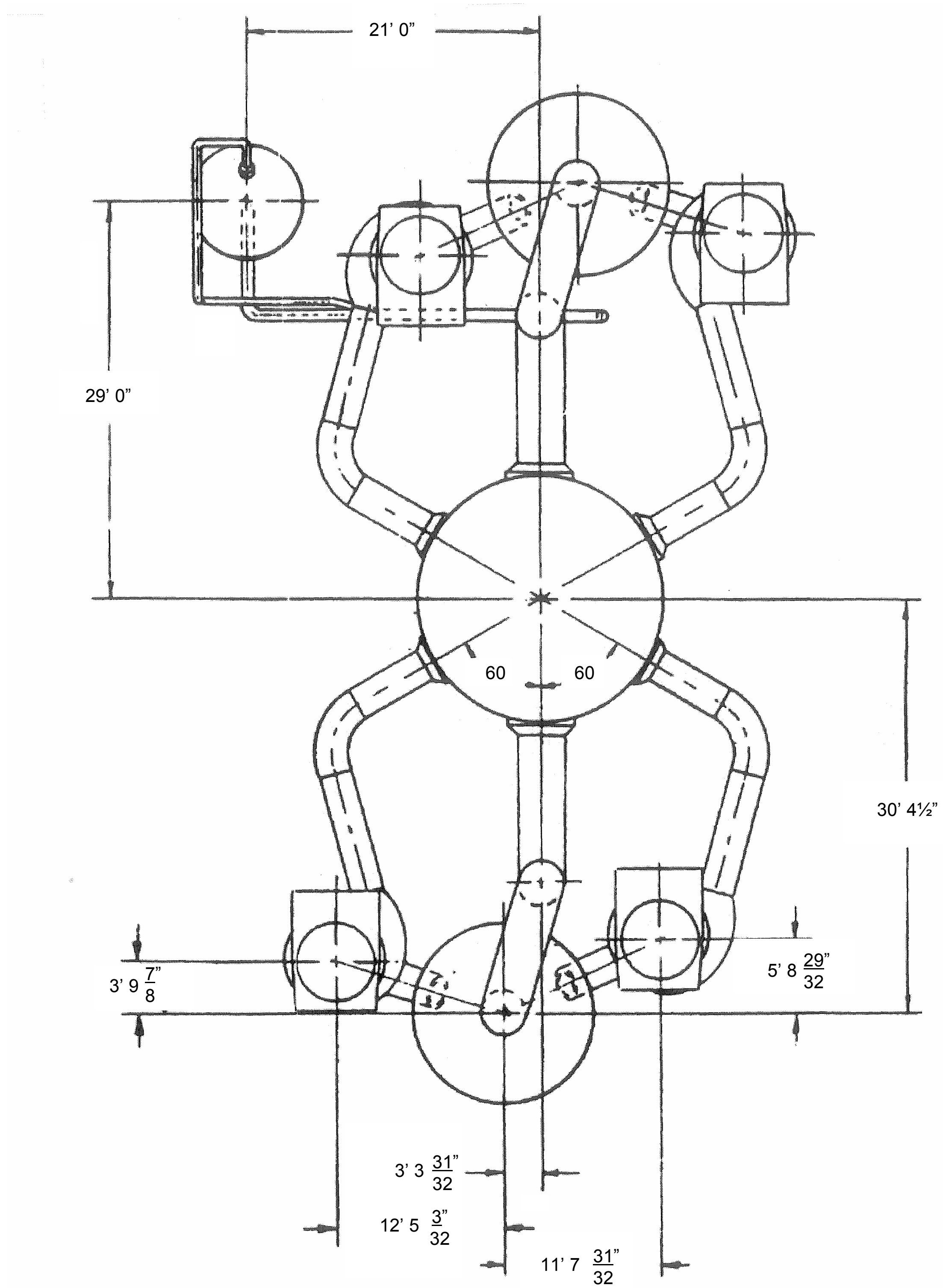
		DESIGN:	ENTERGY
		CAD NO:	

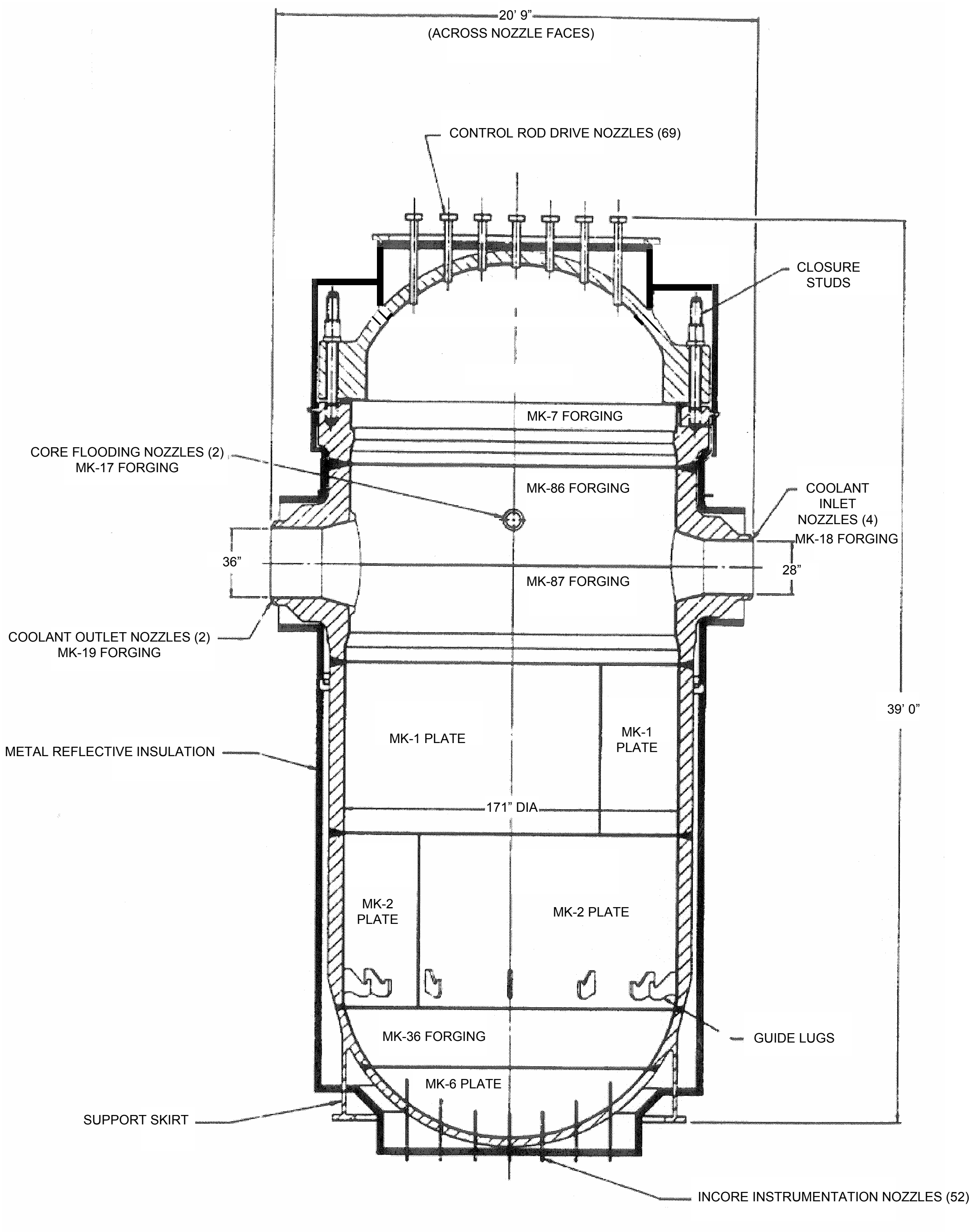
REACTOR COOLANT SYSTEM ARRANGEMENT - ELEVATION	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 4-3

AMENDMENT 20

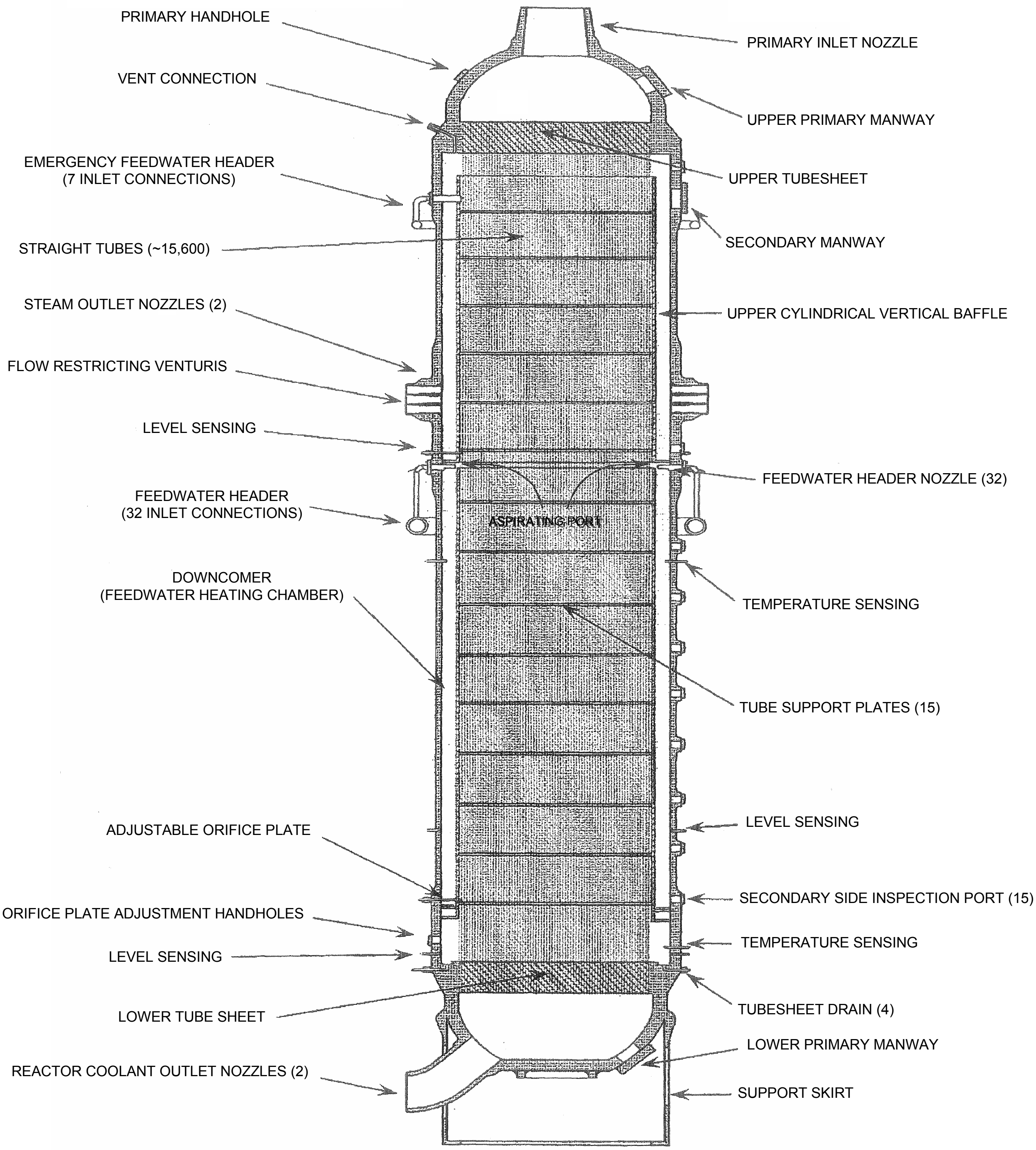
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR COOLANT SYSTEM ARRANGEMENT - PLAN	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-4

AMENDMENT 21

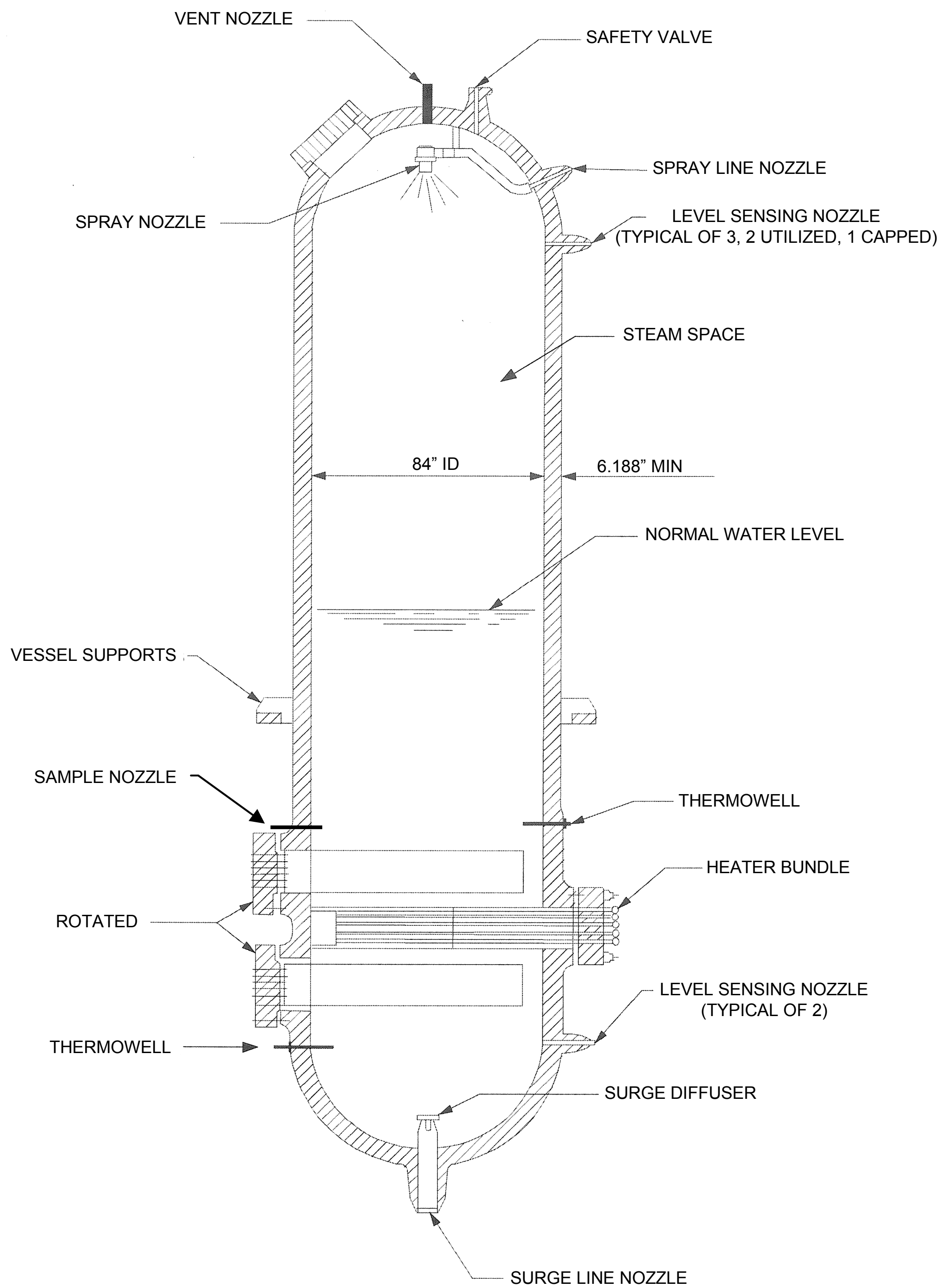
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR VESSEL OUTLINE	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-5

AMENDMENT 20

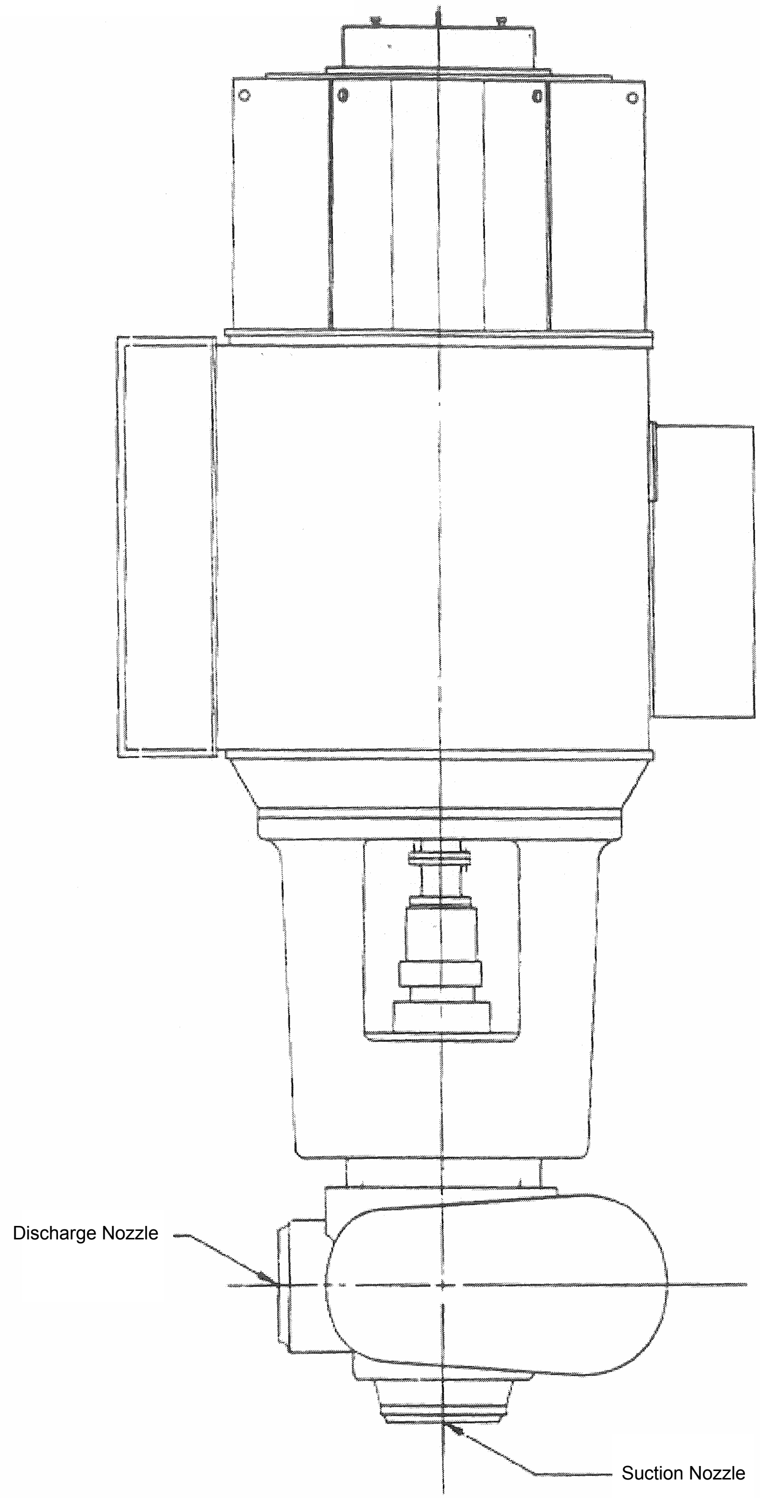
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STEAM GENERATOR OUTLINE	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-6

AMENDMENT 23

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PRESSURIZER OUTLINE	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-7

AMENDMENT 20

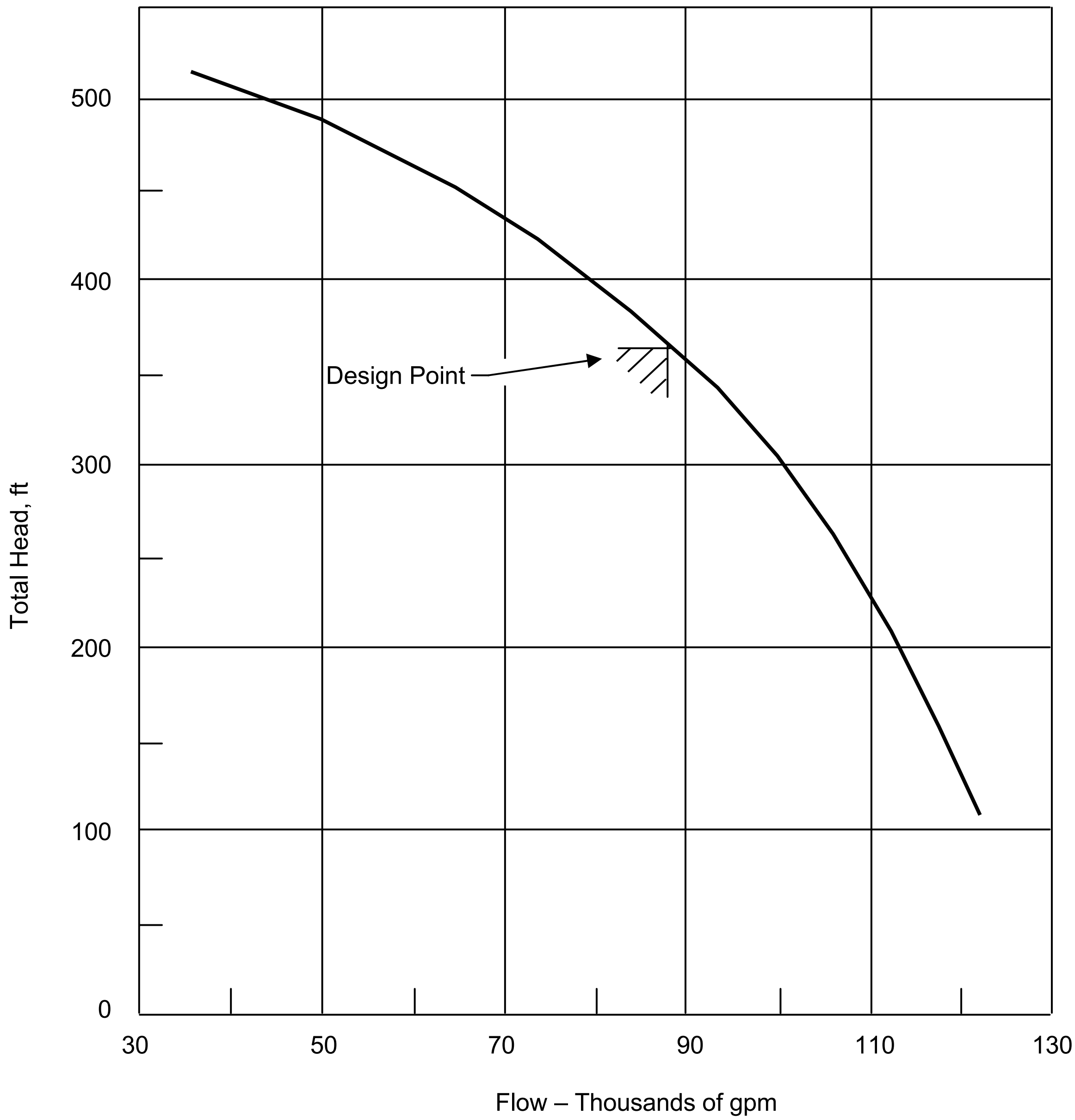
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

REACTOR COOLANT PUMP

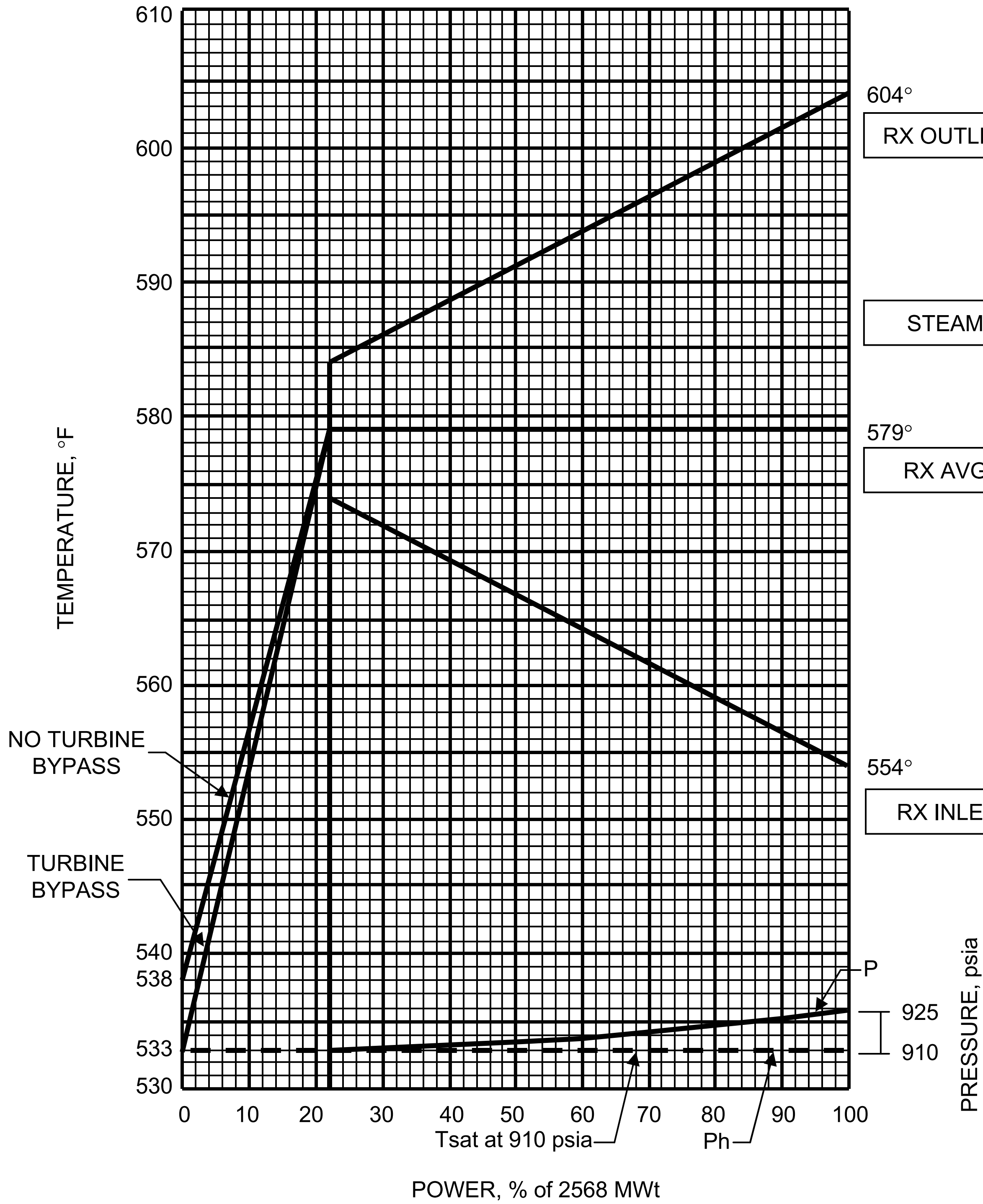
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 4-8

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR COOLANT PUMPS ESTIMATED PERFORMANCE CHARACTERISTICS	BASED ON DRAWING NO	SHEET / REV.



P = STEAM GENERATOR OUTLET PRESSURE
Ph = TURBINE HEADER PRESSURE

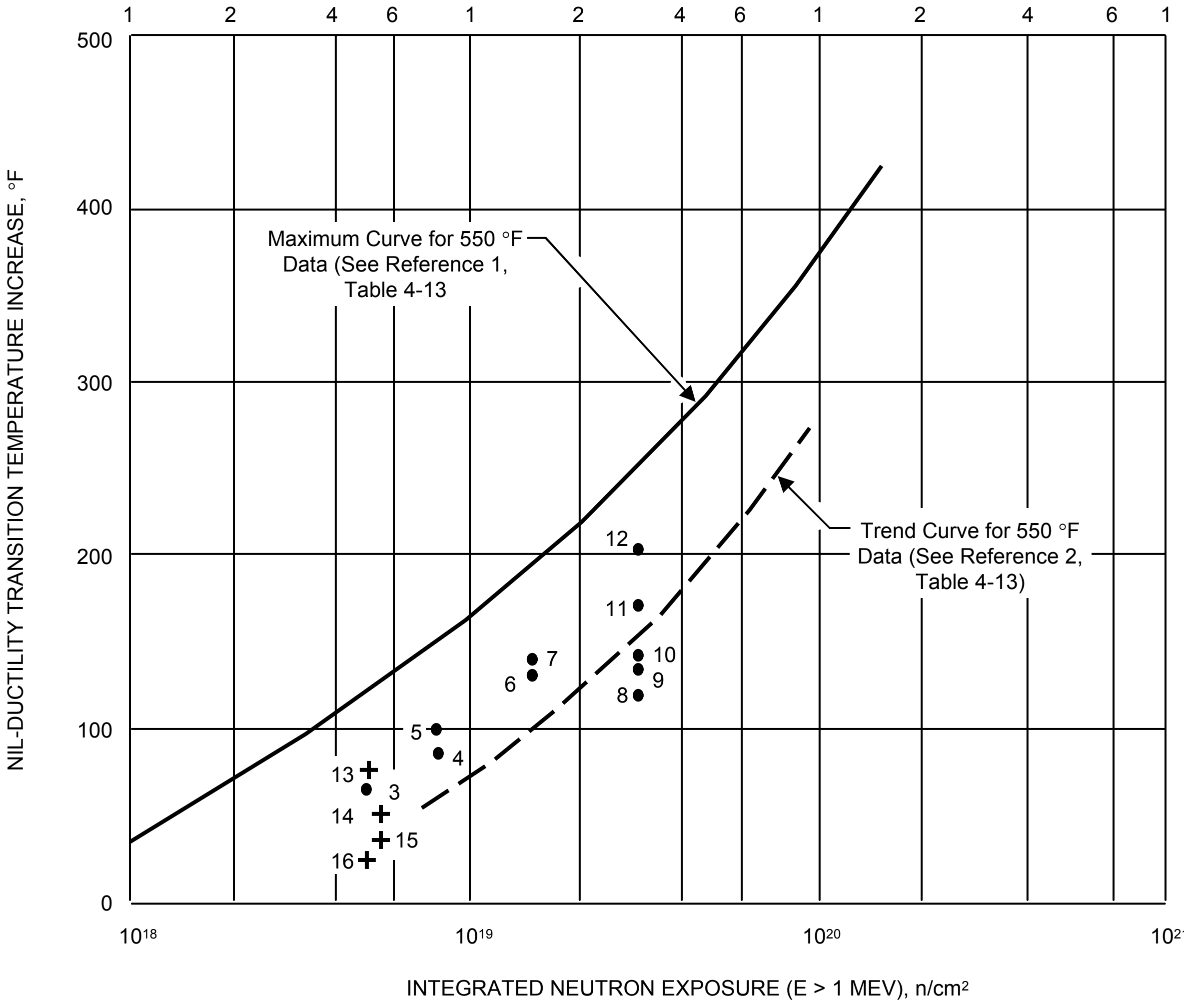
SAR FIGURE NO. 4-9

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
SYSTEM TEMPERATURE VERSUS LOAD	BASED ON DRAWING NO	SHEET / REV.

(FIGURE DELETED)

PREDICTED NDTT SHIFT VERSUS REACTOR VESSEL IRRADIATION			SAR FIGURE NO. 4-10		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 26		
			BASED ON DRAWING NO	SHEET	REV.



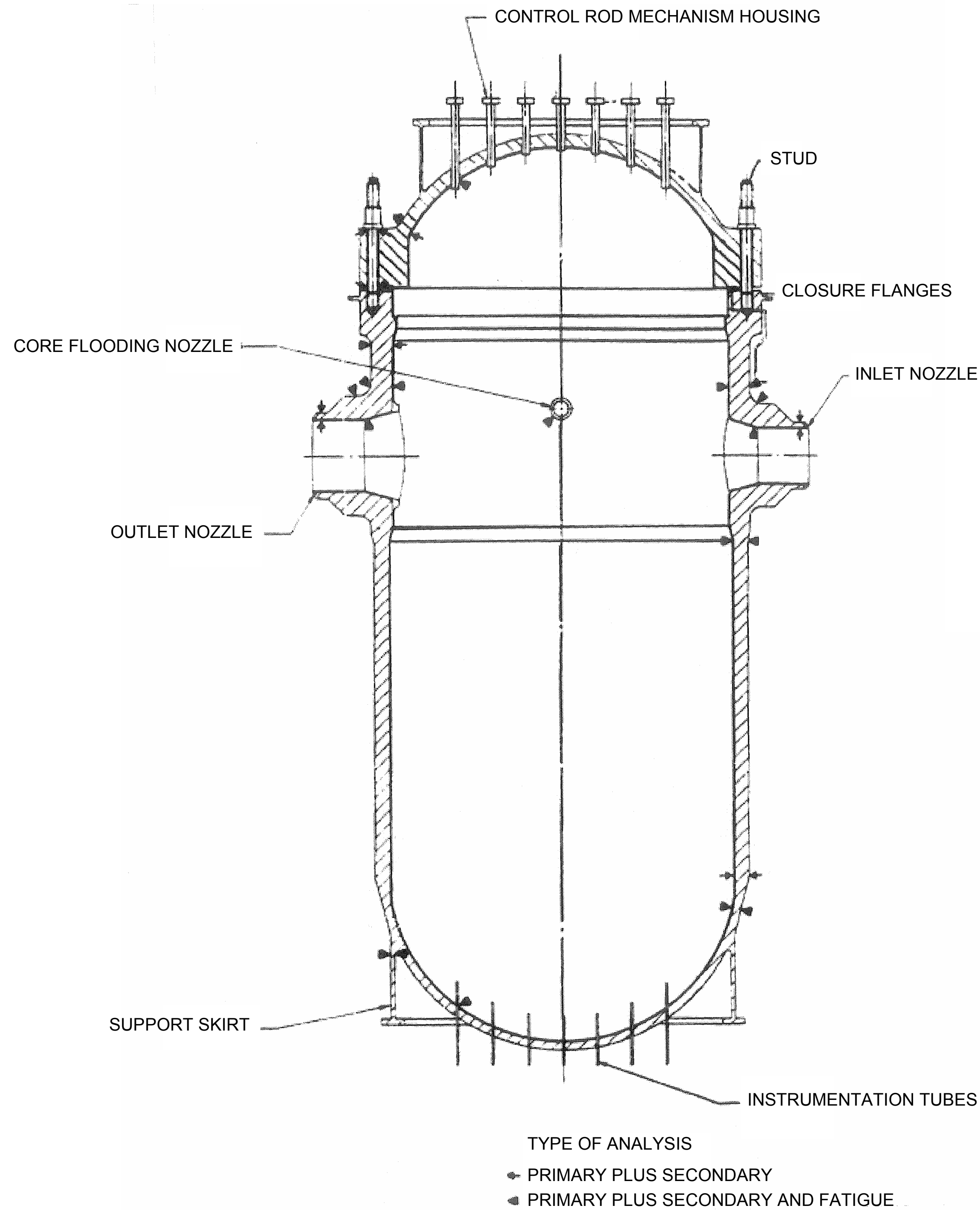
NOTES:

1. All data is for 30 ft-lb "Fix"
2. Curve from ASME Paper No. 62-WA-100. Numbers indicate references in Table 4-13

SAR FIGURE NO. 4-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
NDTT VERSUS INTEGRATED NEUTRON EXPOSURE FOR A302B STEEL	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-12

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
POINTS OF STRESS ANALYSIS – REACTOR VESSEL	BASED ON DRAWING NO	SHEET / REV.

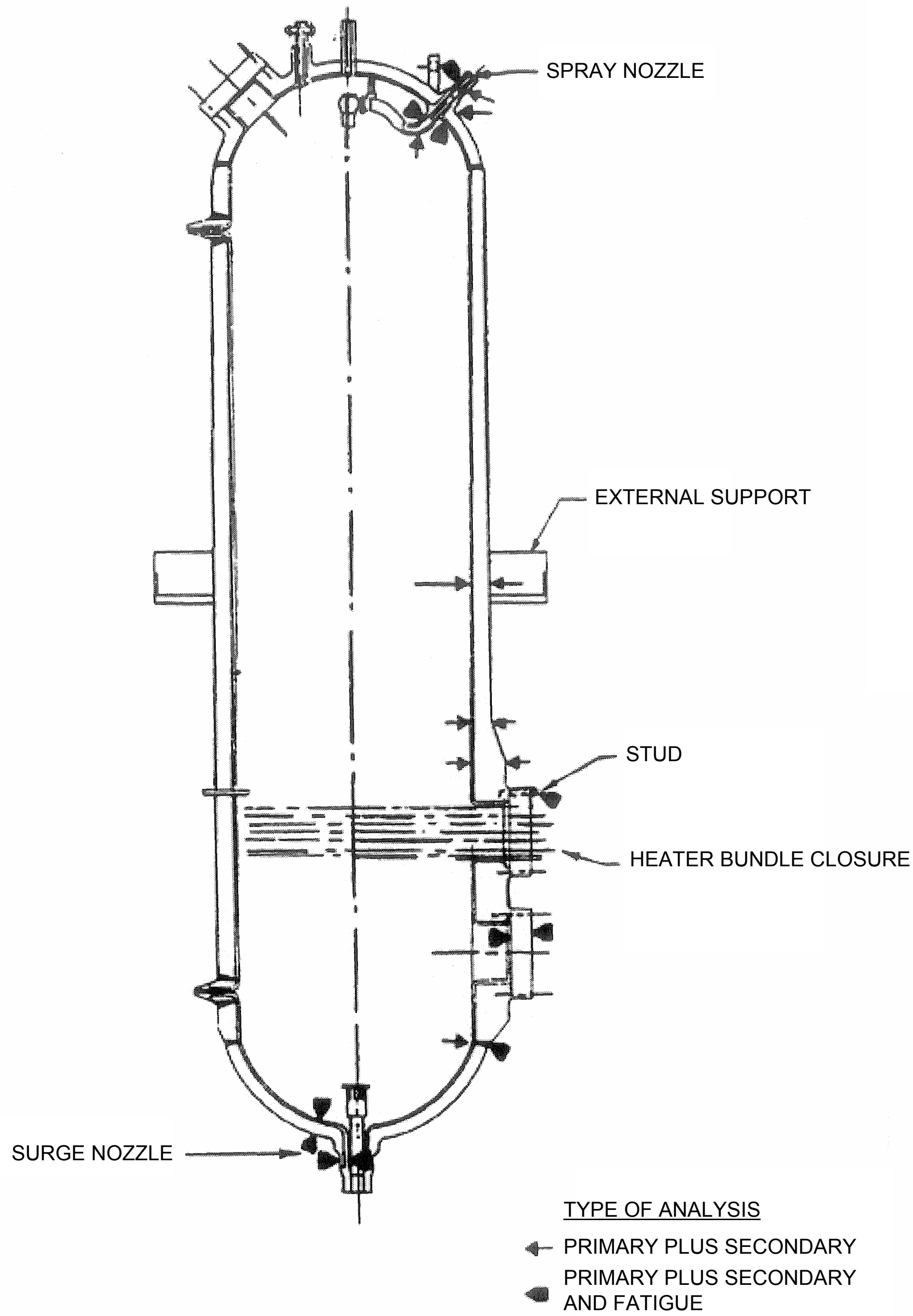
DELETED

SAR FIGURE NO. 4-13

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

POINTS OF STRESS ANALYSIS – STEAM GENERATOR	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 4-14

AMENDMENT 20

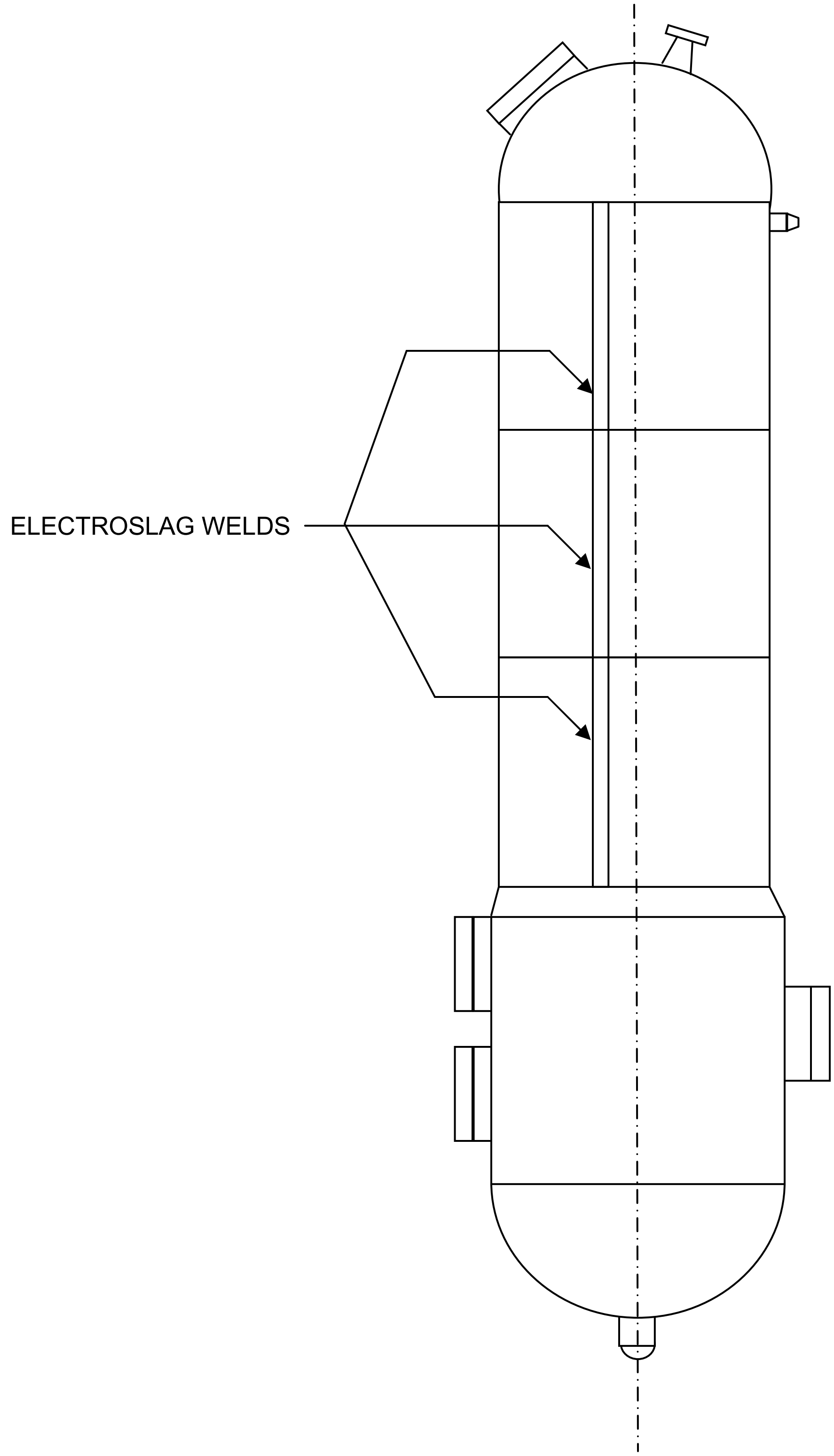
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
POINTS OF STRESS ANALYSIS – PRESSURIZER	BASED ON DRAWING NO	SHEET / REV.

DELETED

SAR FIGURE NO. 4-15

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STEAM GENERATOR ELECTROSLAG WELDS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 4-16

AMENDMENT 20

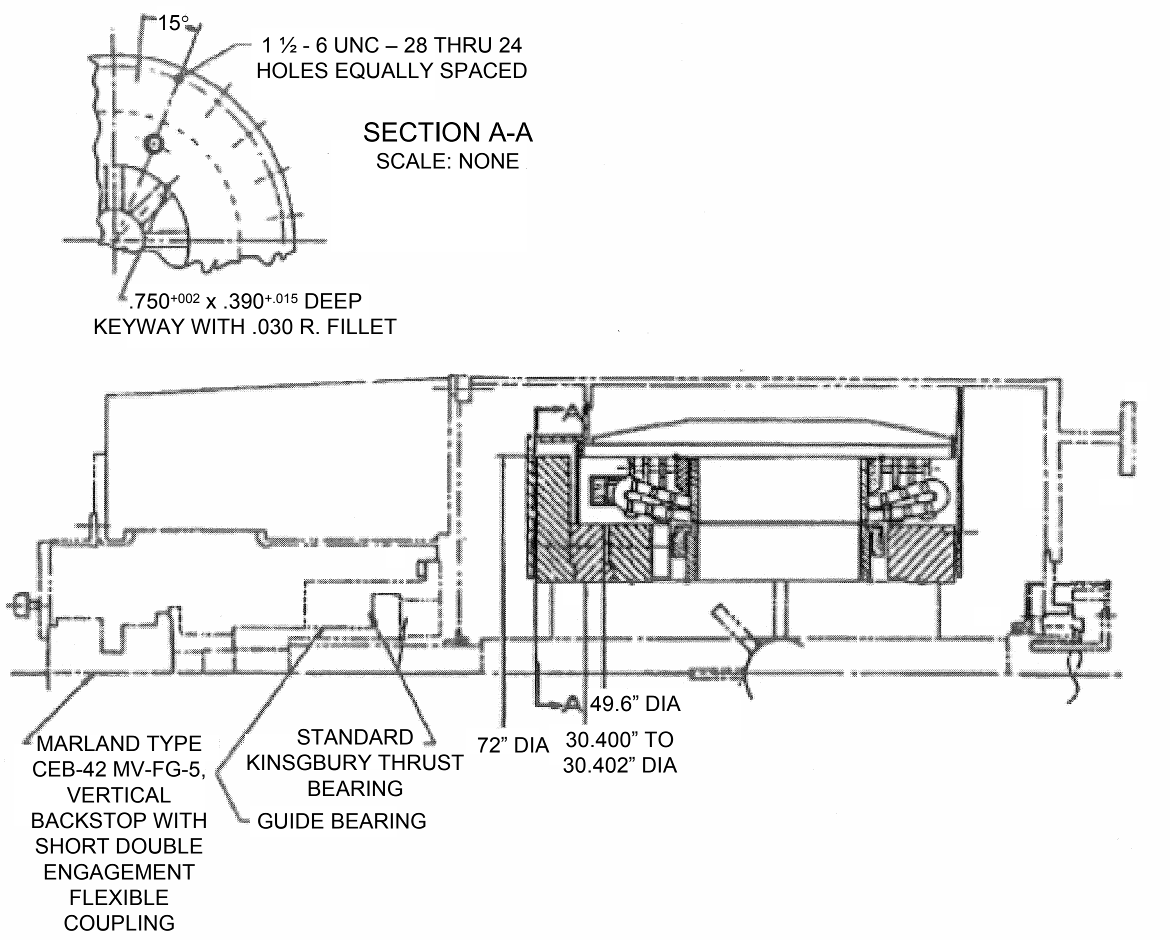
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

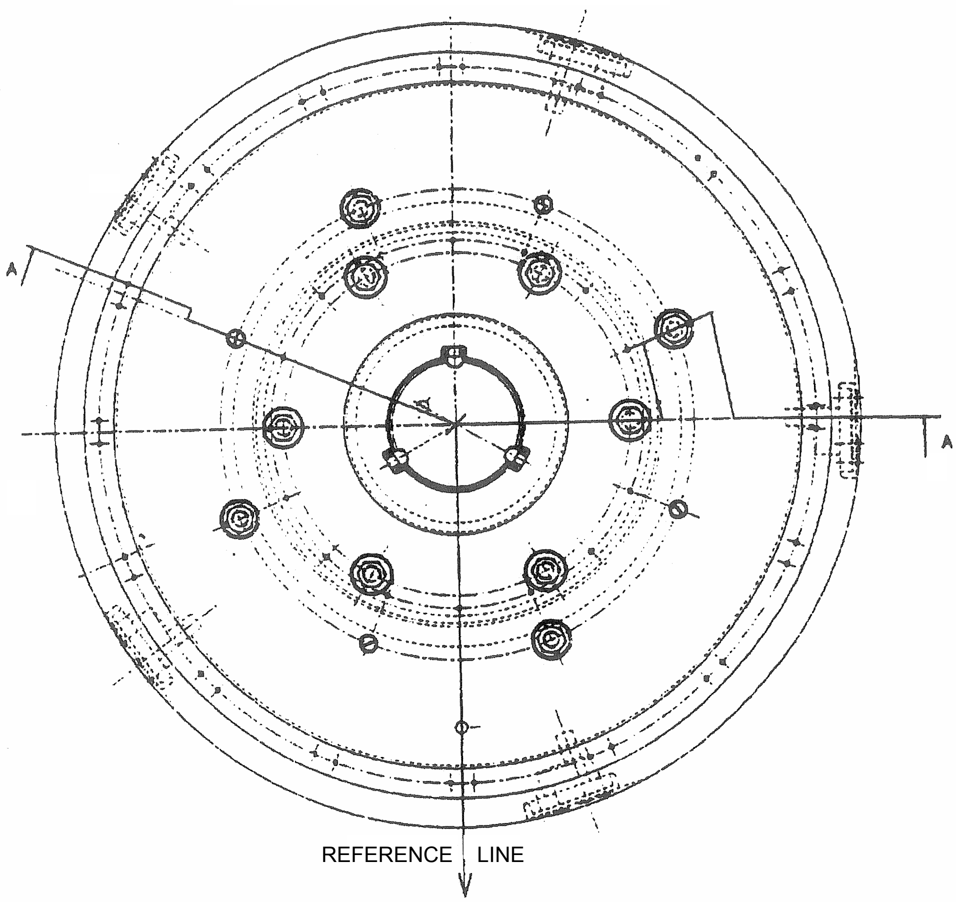
PRESSURIZER ELECTROSLAG WELDS

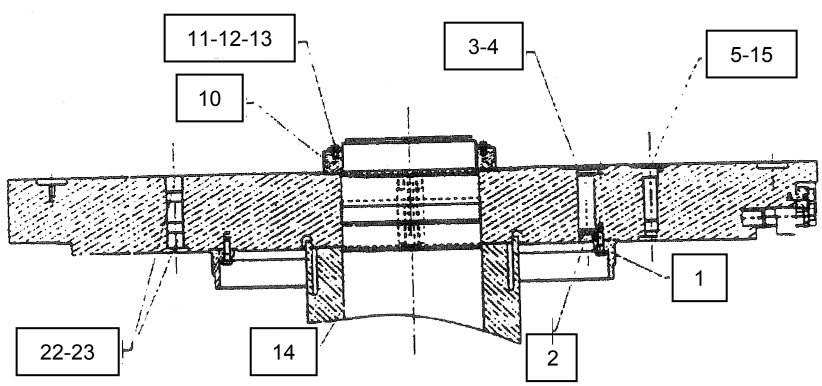
BASED ON DRAWING NO	SHEET	REV.



A-C REACTOR COOLANT PUMP FLYWHEEL GEOMETRY		SAR FIGURE NO. 4-17		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20	
			BASED ON DRAWING NO	SHEET	REV.

VIEW FROM ABOVE WITHOUT NOTCHED NUT





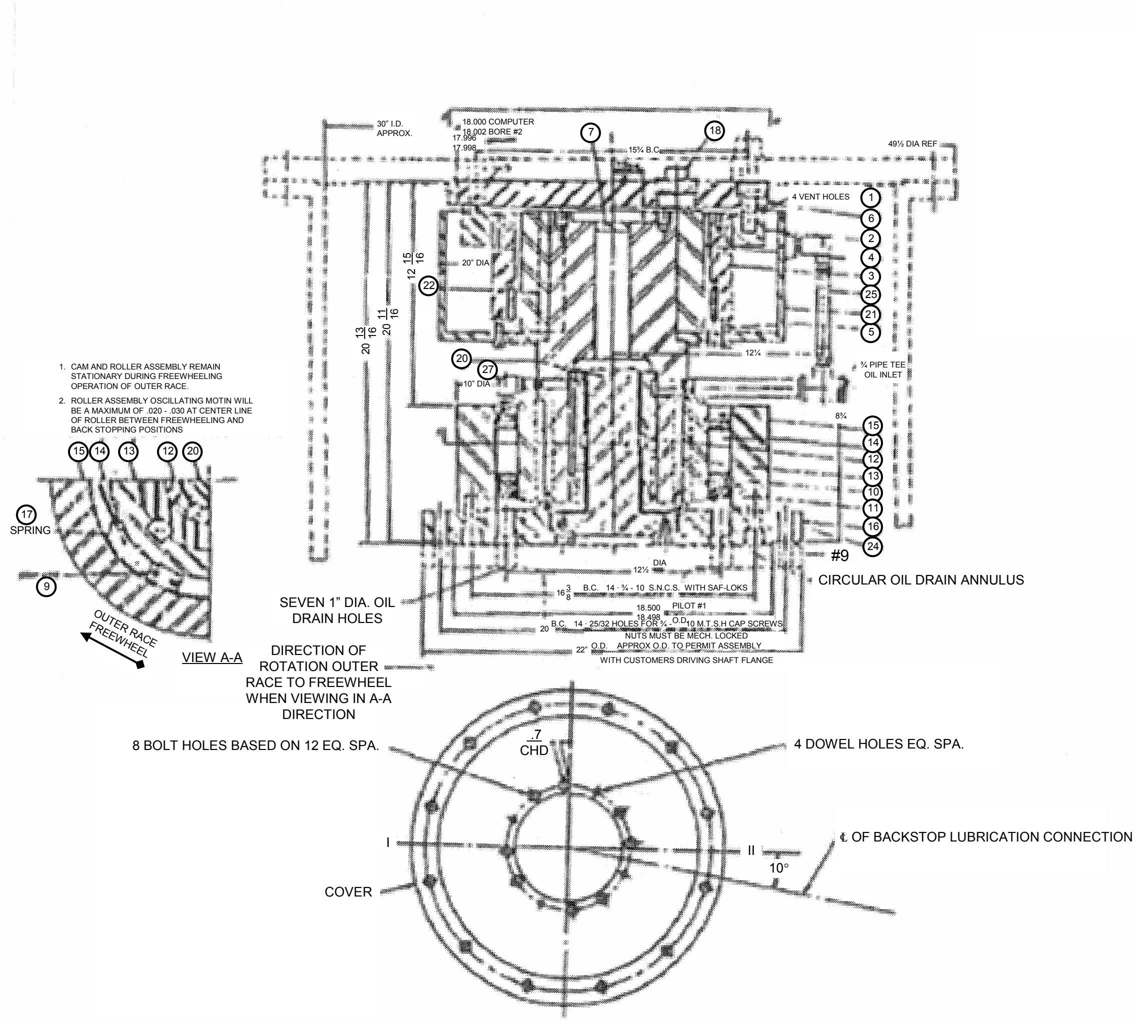
SECTION A-A

SAR FIGURE NO. 4-17A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

JI REACTOR COOLANT PUMP FLYWHEEL GEOMETRY	BASED ON DRAWING NO	SHEET	REV.



NO.	DESCRIPTION
1	Coverplate (or customer)
2	Ring Gear
3	Floating Gear
4	Geared Hub
5	Support Springs with Pins
6	Mounting Plate
7	Nut
8	
9	Stop Log
10	Roller Assem Ball Brg
11	Roller Support Ring
12	Cam Bearing
13	Cam Shaft Assembly
14	Roller Assembly
15	Outer Race
16	Drive Plate
17	Spring
18	Inspection Plug
19	
20	Pilot Brg Shaft
21	Coupling Oil Collector
22	Cam
23	
24	Ring # RS-1074
25	Flexible Hose
26	
27	Oil Jet Fitting – 5 @ 120°

A-C REACTOR COOLANT PUMP ANTI-REVERSE ROTATION DEVICE

SAR FIGURE NO. 4-18

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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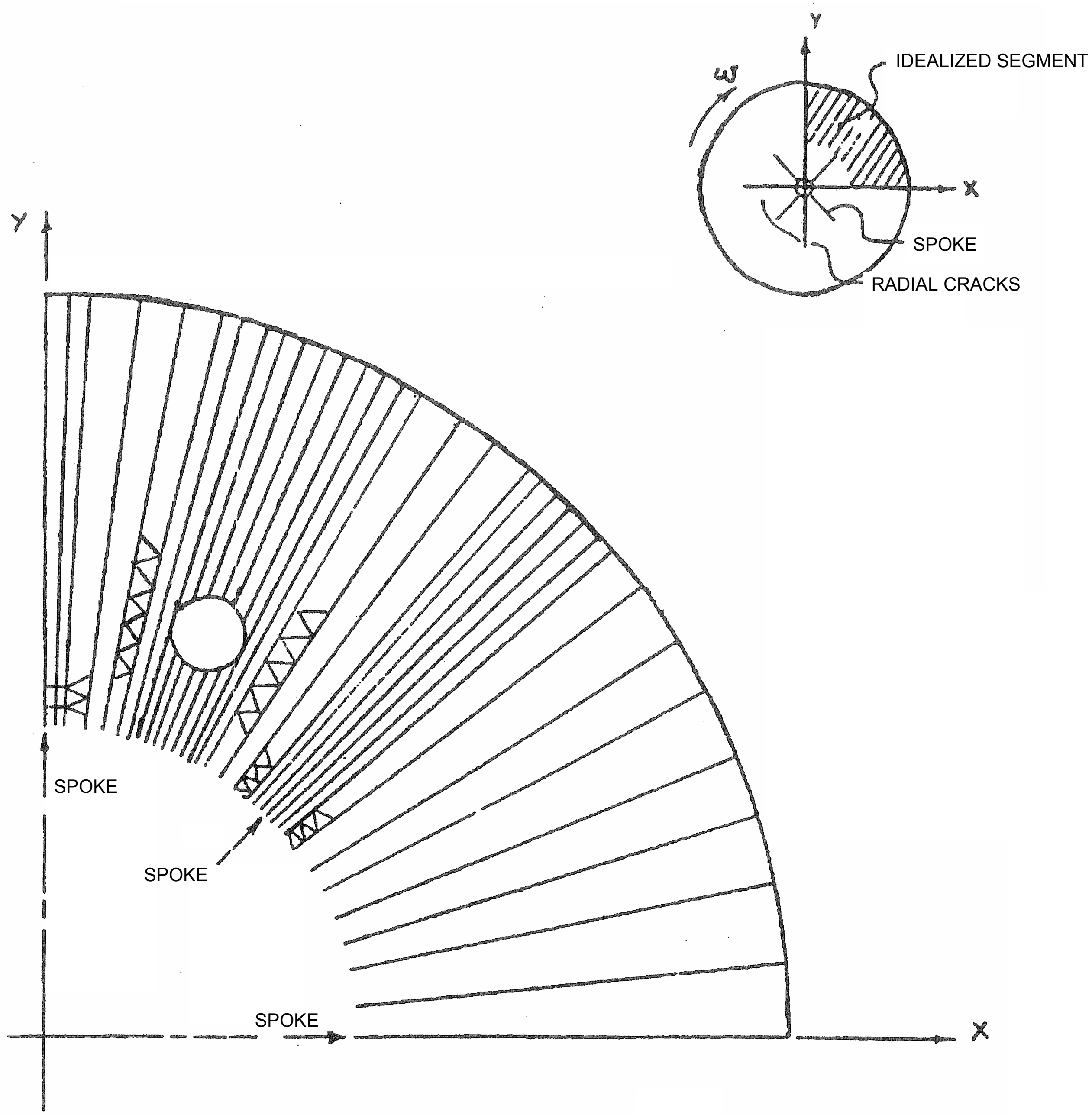
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

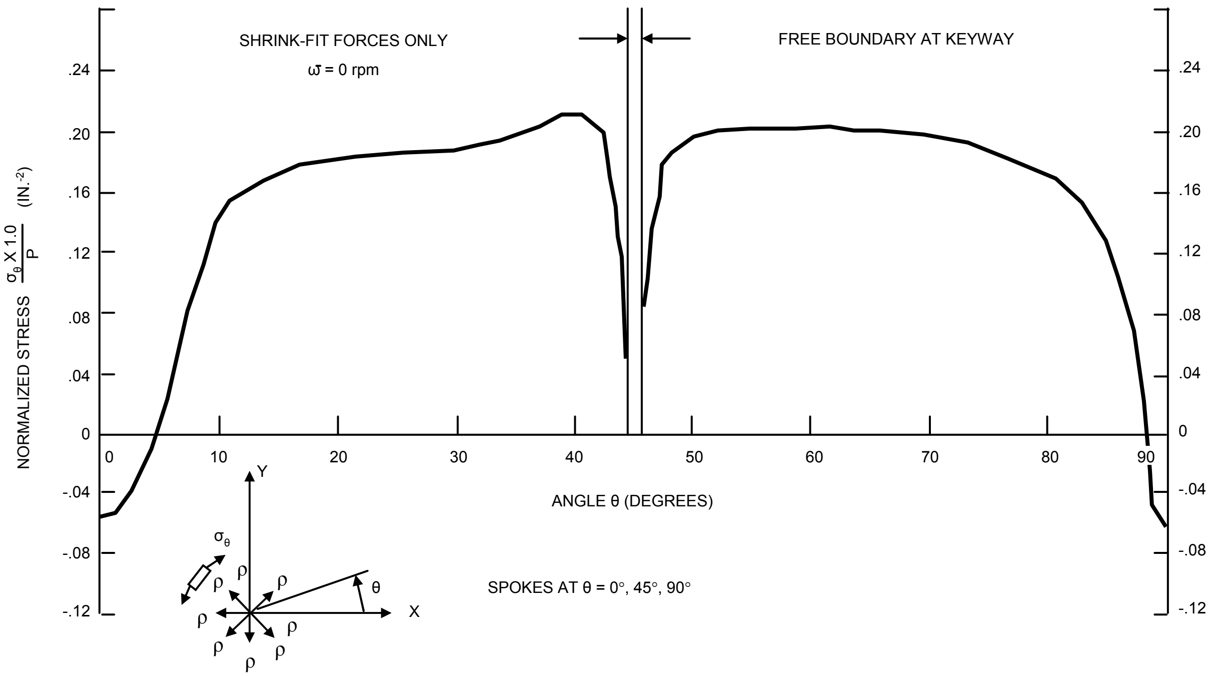
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 4-19

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
FINITE ELEMENT IDEALIZATION OF A-C RCP FLYWHEEL	BASED ON DRAWING NO	SHEET	REV.



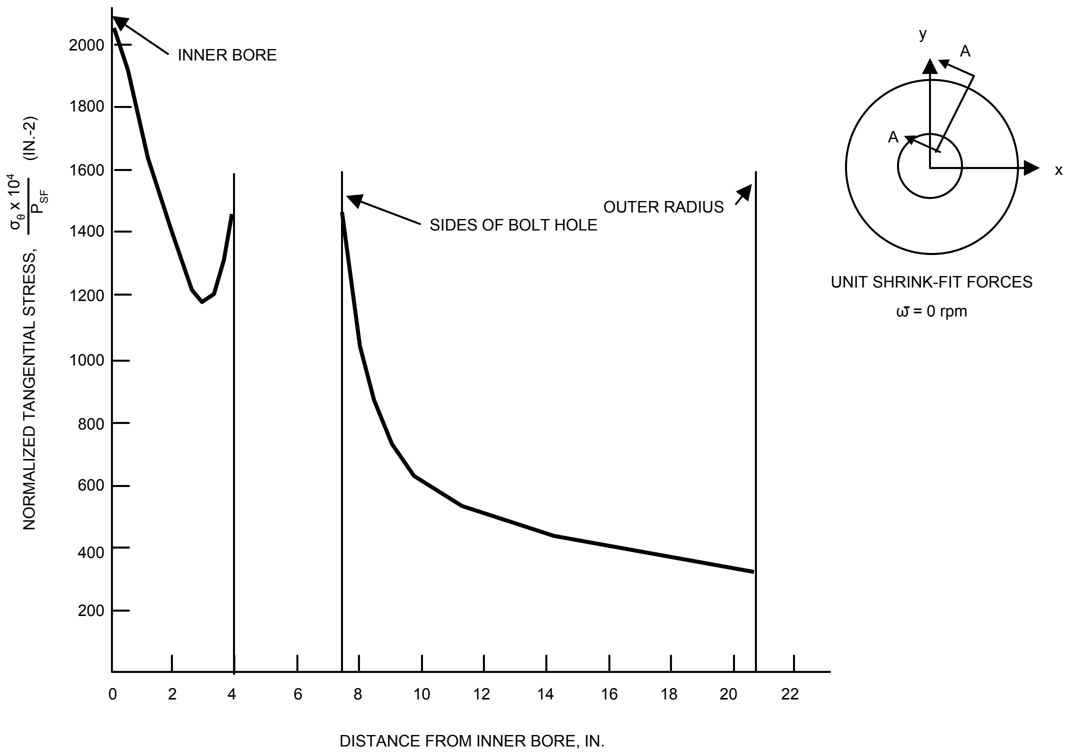
ANGULAR VARIATION OF TANGENTIAL STRESS $\sigma\theta$ IN A-C FLYWHEEL – INNER BORE	SAR FIGURE NO. 4-20
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

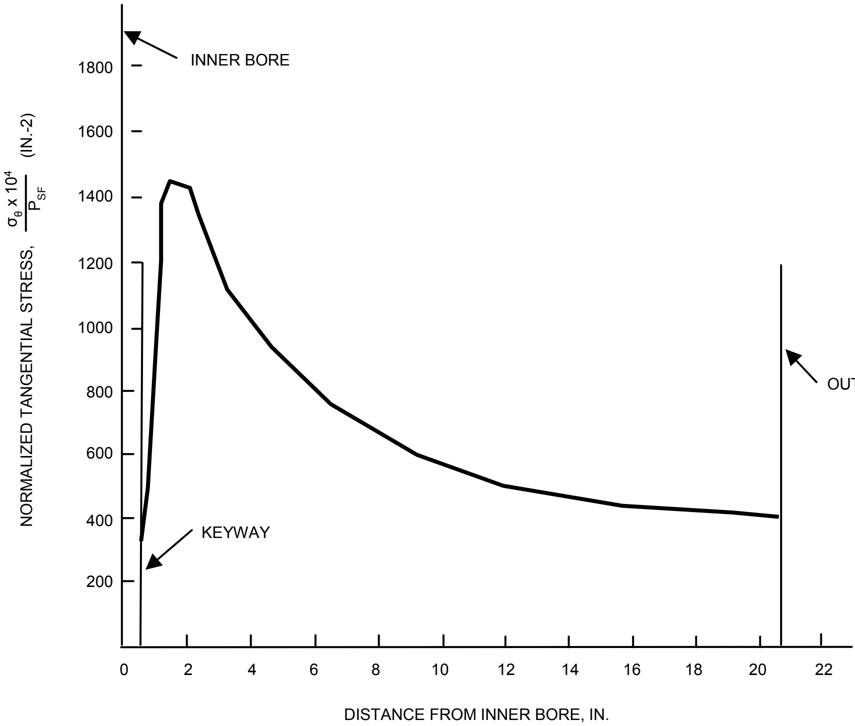
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

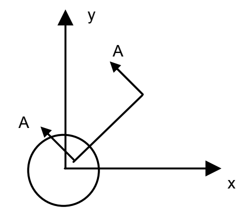
BASED ON DRAWING NO	SHEET	REV.



VARIATIONS OF TANGENTIAL STRESS FROM INNER BORE TO OUTER BORE – SECTION AA

VARIATION OF TANGENTIAL STRESS FROM INNER BORE TO OUTER BORE – SECTION AA A-C FLYWHEEL			SAR FIGURE NO. 4-21
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.





UNIT SHRINK-FIT FORCES

$\omega = 0$ rpm

VARIATION OF TANGENTIAL STRESS FROM INNER BORE TO OUTER BORE – SECTION AA A-C FLYWHEEL	SAR FIGURE NO. 4-22
	AMENDMENT 20

ARKANSAS NUCLEAR ONE

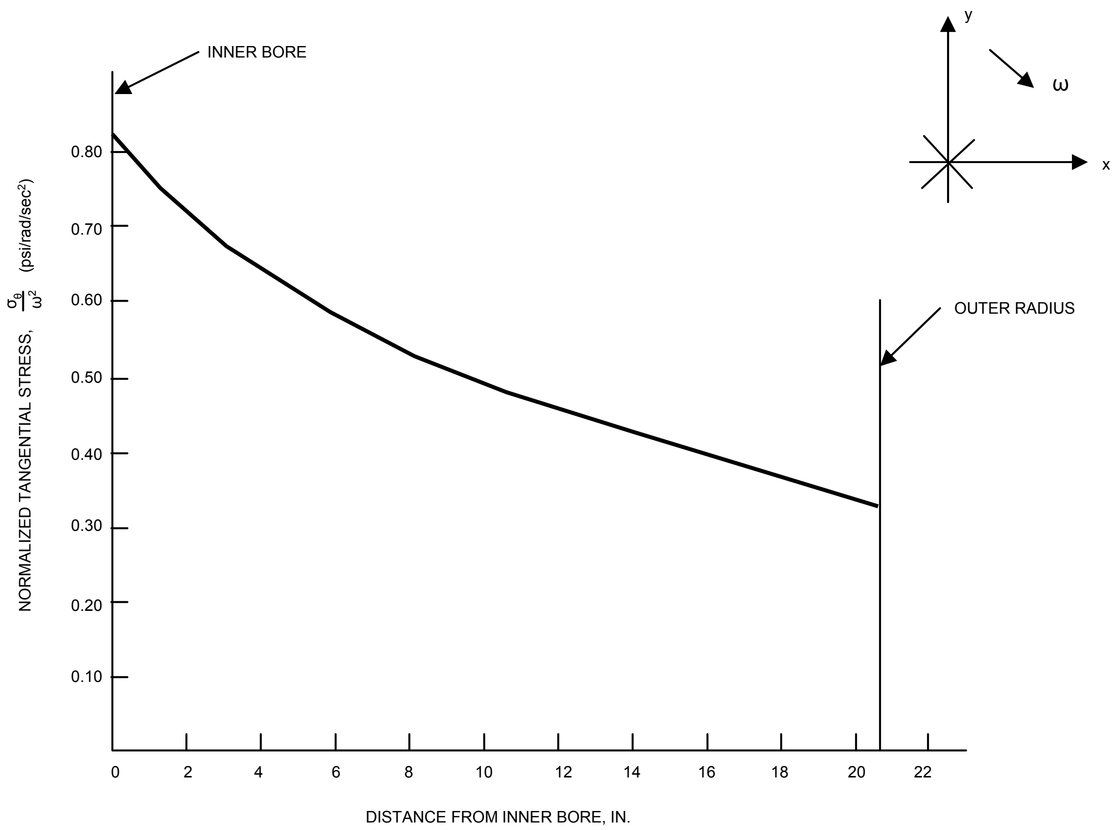
UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

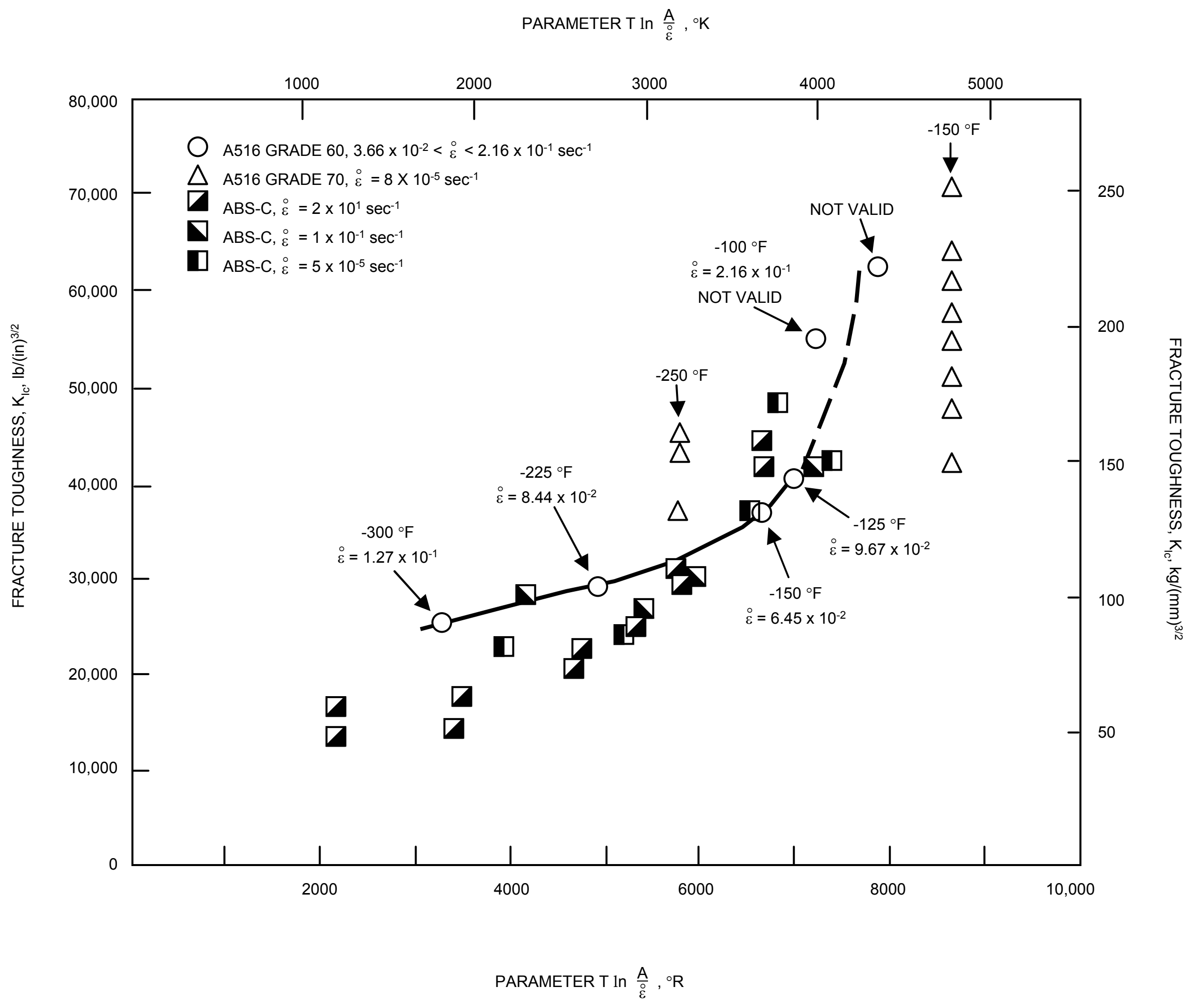
BASED ON DRAWING NO	SHEET	REV.



VARIATION OF TANGENTIAL STRESS $^{\sigma}\theta$ FROM INNER BORE TO OUTER BORE – INERTIA FORCES A-C FLYWHEEL

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	SAR FIGURE NO. 4-23 AMENDMENT 20	

BASED ON DRAWING NO	SHEET	REV.



FRACTURE TOUGHNESS BEHAVIOR OF ASTM 5_{16} GRADE 60, 5_{16} GRADE 70, AND ABS-C STEELS

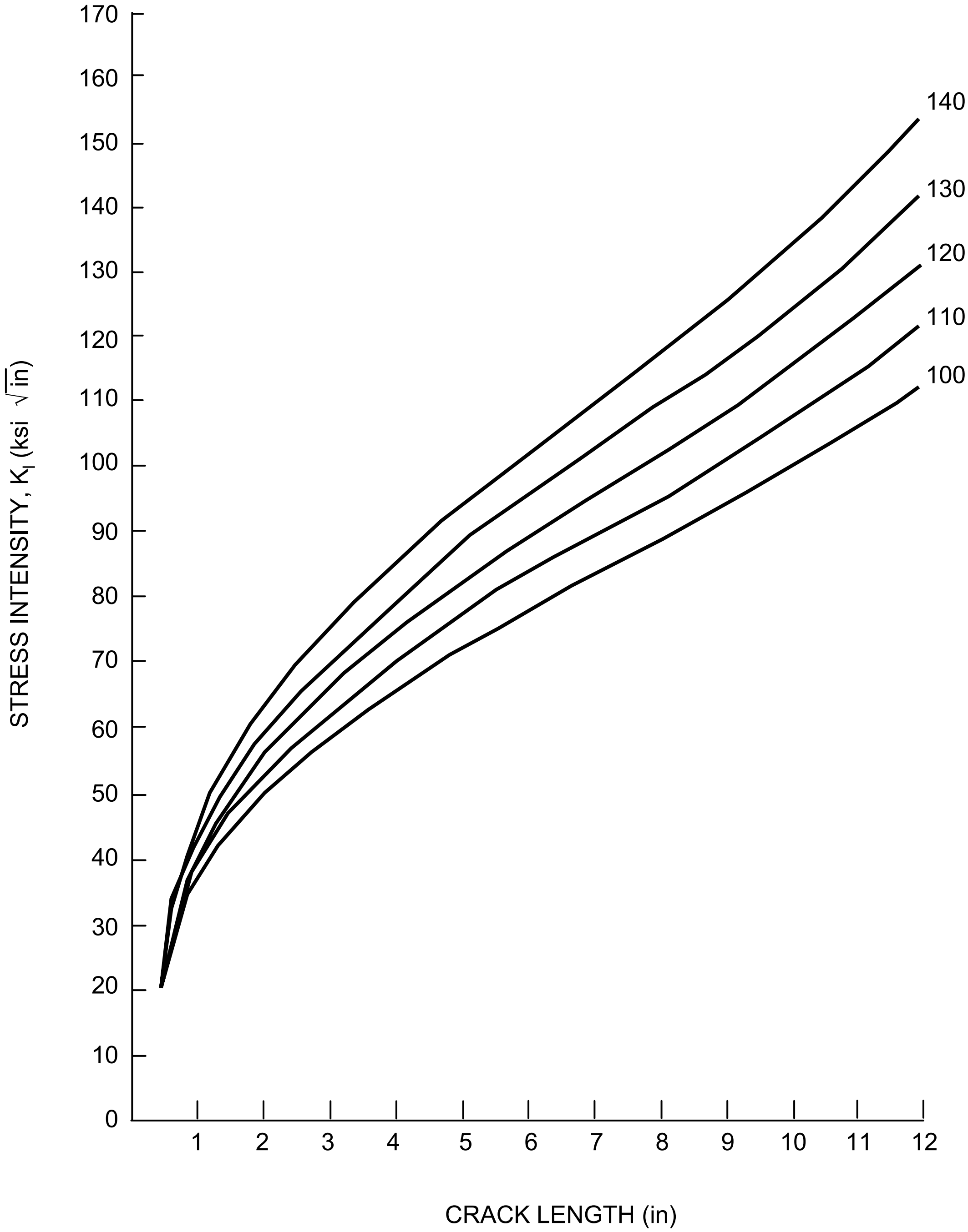
SAR FIGURE NO. 4-24	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

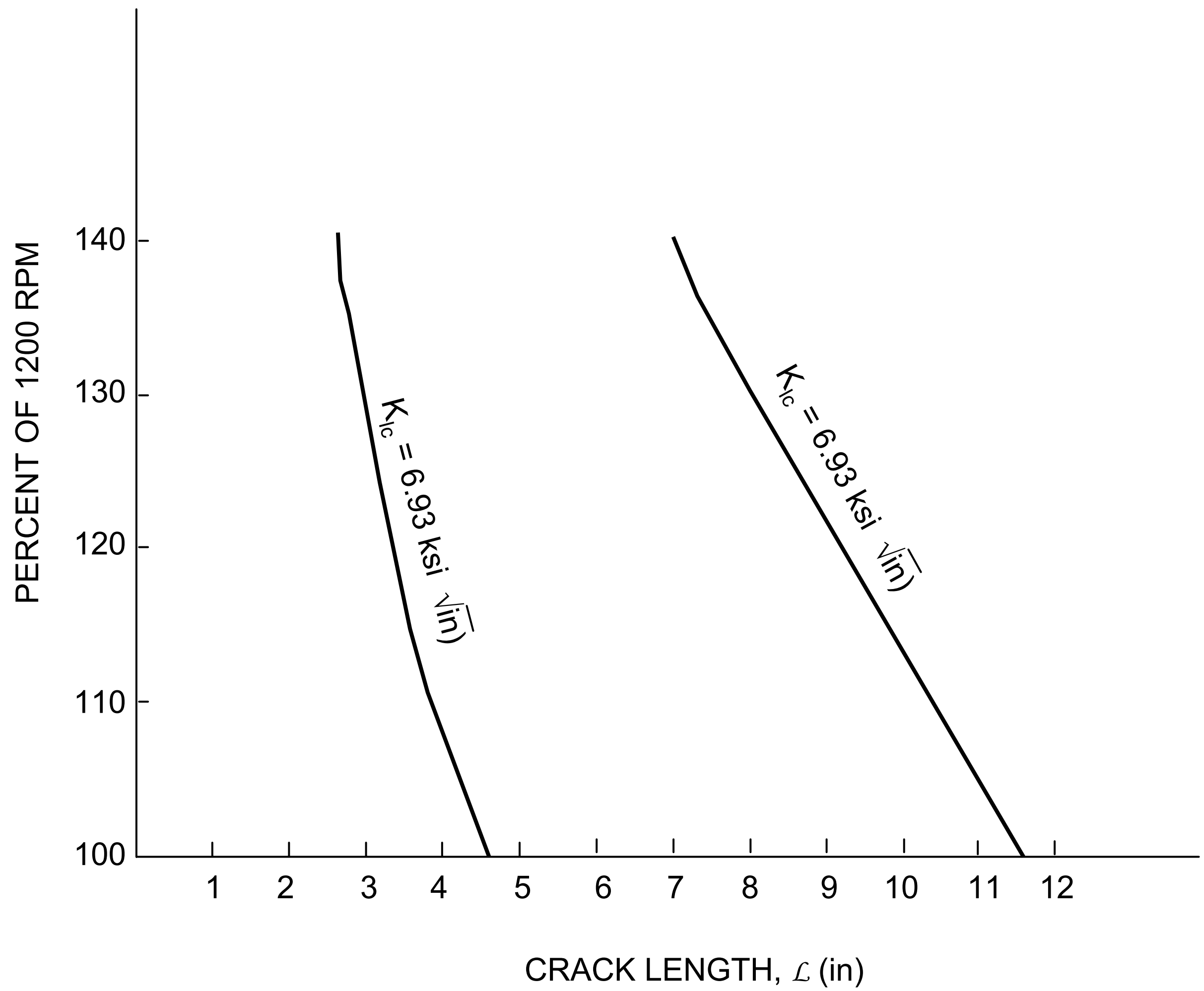
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 4-25

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
STRESS INTENSITY VS CRACK LENGTH A-C FLYWHEEL	BASED ON DRAWING NO	SHEET	REV.



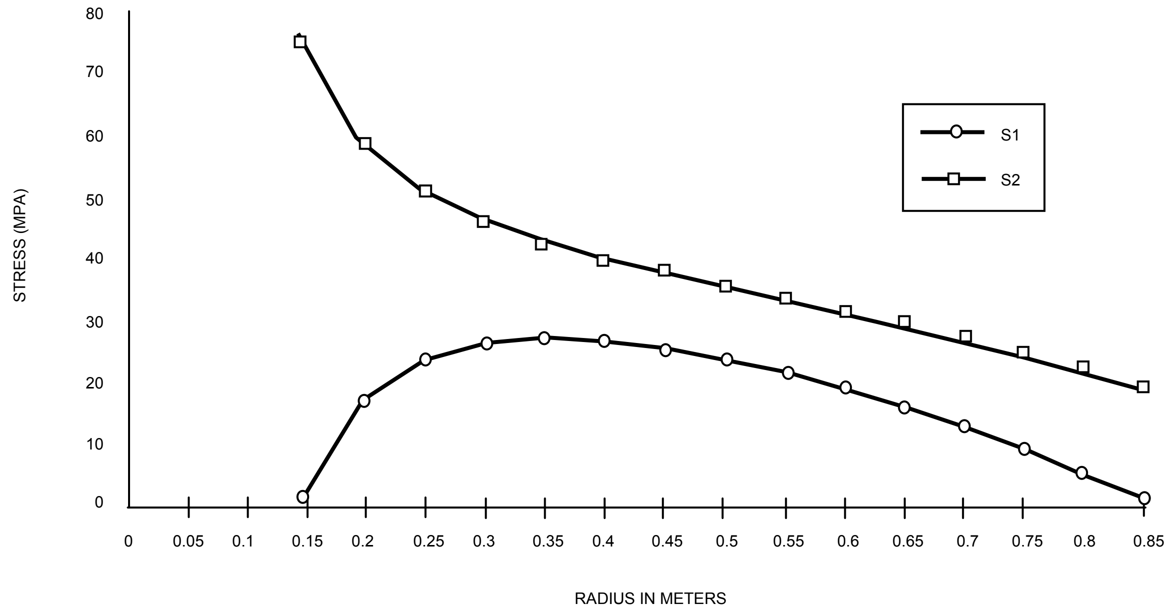
PERCENT SPEED VS CRACK LENGTH A-C FLYWHEEL			SAR FIGURE NO. 4-26
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20
			BASED ON DRAWING NO / SHEET / REV.

DELETED

SAR FIGURE NO. 4-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LITHIUM CONTROL FOR STEADY STATE POWER OPERATION	BASED ON DRAWING NO	SHEET 1 / REV.

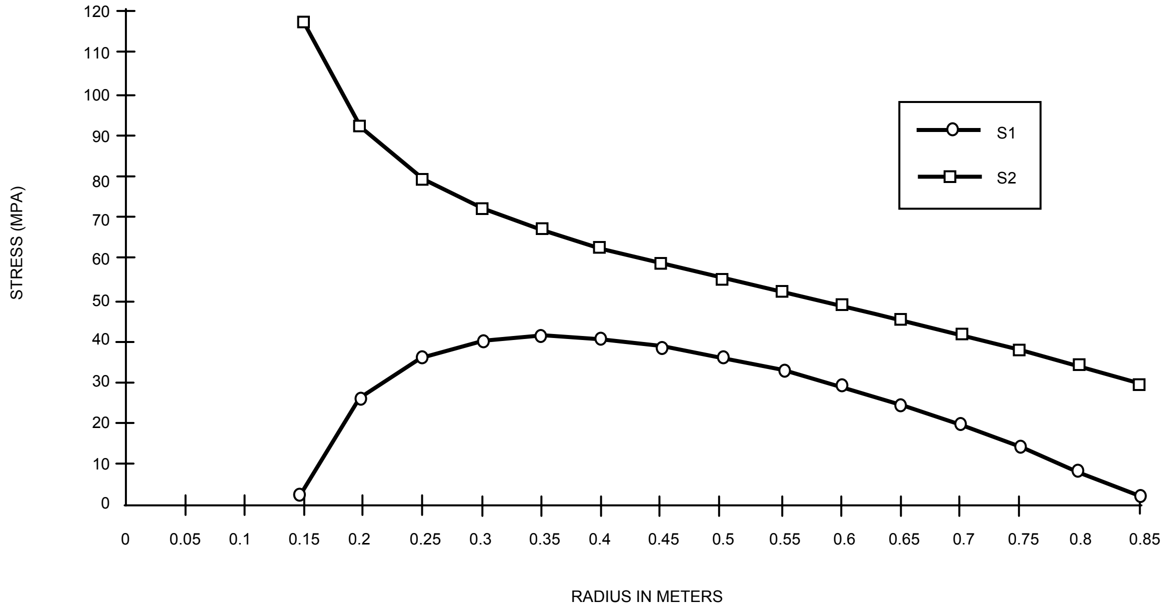


JI MOTOR FLYWHEEL STRESS VARIATION WITH RESPECT TO RADIUS (1200 RPM)

VARIATION OF JI FLYWHEEL RADIAL AND CIRCUMFERENTIAL STRESSES	SAR FIGURE NO. 4-28
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



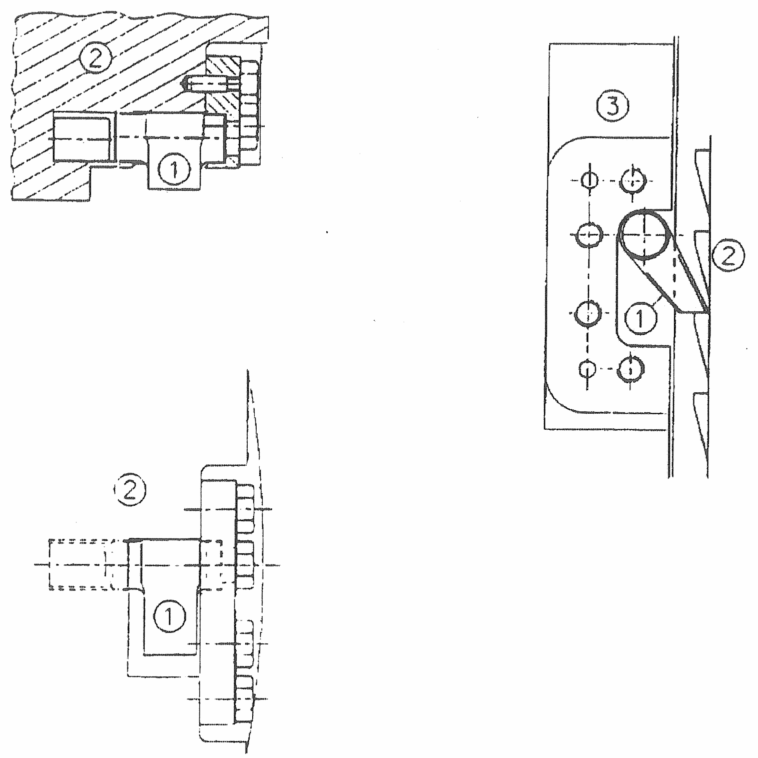
JI MOTOR FLYWHEEL STRESS VARIATION WITH RESPECT TO RADIUS (1500 RPM)

VARIATION OF JI FLYWHEEL RADIAL AND CIRCUMFERENTIAL STRESSES	SAR FIGURE NO. 4-29
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



1 PAWL
2 FLYWHEEL
3 RATCHET PLATE

SAR FIGURE NO. 4-30

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
JI REACTOR COOLANT PUMP MOTOR ANTI-REVERSE ROTATION DEVICE	BASED ON DRAWING NO	SHEET / REV.

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
5.3.2	Correspondence from NRC to AP&L dated May 1980. (0CNA058088)
5.2.2.1.1	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated July 13, 1979. (0CAN077907)
5.1.4.2.1	Correspondence from Seyfrit, NRC, to Cavanaugh, AP&L, dated July 2, 1979. (0CNA077929)
5.1.4.2.3 5.3.2	Correspondence from Cavanaugh, AP&L, to Seyfrit, NRC, dated July 3, 1980. (0CAN078007)
5.3.2	Correspondence from Cavanaugh, AP&L to Seyfrit, NRC, dated September 15, 1980. (0CAN098012)
5.2.6.1.1 5.2.7	Correspondence from Trimble, AP&L, to Reid, NRC, dated December 12, 1979. (1CAN127908)
5.2.2.1.3	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated November 19, 1980. (1CAN118018)
5.2 5.5	Design Change Package 80-1083, "Emergency Feedwater Initiation."
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5.2.4	Unit 1 Technical Specification Amendment 215

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5.2.2.1.3	Technical Specification Amendment 242, "Low Pressure Test of Reactor Building Air Lock"

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5.1.6	Condition Report CR-ANO-2-2016-4267, "Clarification of Auxiliary Building Boundary and Passive/Non-Passive Flood Seals"

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5 STRUCTURES

5.1 DESIGN BASES FOR STRUCTURES, SYSTEMS AND EQUIPMENT

5.1.1 GENERAL

The design bases for structures for normal operating conditions are governed by the applicable building design codes. The design bases for specific systems and equipment are stated in the appropriate SAR section. The basic design criterion for Loss of Coolant Accident (LOCA) and seismic conditions is that there be no loss of function if that function is related to public safety.

5.1.2 SEISMIC CLASSES OF STRUCTURES, SYSTEMS AND EQUIPMENT

5.1.2.1 Seismic Class 1

Seismic Class 1 structures, systems, and equipment are defined as structures, systems, and equipment whose failure could cause uncontrolled release of radioactivity or as those essential for safe reactor shutdown and the immediate and long-term operation following a LOCA. When a system as a whole is referred to as Seismic Class 1, portions not associated with loss of function of the system may be designated as Seismic Class 2.

5.1.2.1.1 Seismic Class 1 Structures

- Reactor building (including steel liner, prestressing system penetrations, air locks, interior structures, and shielding elements directly related to nuclear safety).
- Auxiliary building housing the engineered safeguards systems, control room, spent fuel pool (excluding liner plate), diesel generators, and radioactive materials.
- Portions of intake structure housing service water pumps.
- Supports for Seismic Class 1 system components.
- Emergency reservoir and pipelines.
- Emergency diesel fuel storage vault.
- Post Accident Sampling System Building

5.1.2.1.2 Seismic Class 1 Systems and Equipment

- Reactor vessel and internals including fuel assemblies, Control Rod Assemblies (CRAs) and Control Rod Drive Mechanisms (CRDMs).
- Reactor Coolant System (RCS) including steam generators, pressurizer, Reactor Coolant Pumps, piping, and valves.
- Reactor building penetration piping up to and including the first external isolation valve.

- Main steam piping from steam generators through main steam isolation valves, including atmospheric dump branch headers out to and including the first isolation valves, and safety valve branch headers out to and including the safety valves.

- Feedwater piping from steam generators through external reactor building isolation valves.

- Safety injection system including high and low pressure injection pumps, Core Flooding Tanks, Borated Water Storage Tank, decay heat removal heat exchangers, valves, and connecting piping.

- Reactor Building Spray System including reactor building spray pumps, chemical tanks, spray headers, valves, and connecting piping.

- Reactor Building Emergency Air Recirculation and Cooling System.

- Emergency Diesel Generators including the day tank.

- Control boards, switchgear, load centers, batteries, transformers and cable runs serving Seismic Class 1 equipment.

- Gaseous Radioactive Waste System including waste gas decay and surge tanks, vacuum degasifier, filters, piping, and valves.

- Reactor Building Polar Crane (L-2) is qualified for the following conditions:

  1. Seismic Class I when unloaded in its parked position.

  2. Seismic Class I when unloaded with the trolley and bridge parked in any position. It is also Seismic Class I with a load of up to 10,000 lbs on the main or auxiliary hook; however, during modes other than 5, 6, and defueled, ANO's commitments to NUREG-0612 shall be met and load drop analyses must be performed by Design Engineering.

  3. Seismic Class II/I with loads up to 190 tons in any position in Modes 5, 6, and defueled; however, ANO's commitments to NUREG-0612 shall be met.

- Emergency Feedwater System including Condensate Storage Tank (T41B), electric motor driven and steam turbine driven pumps, valves, and connecting piping.

- Service Water System including service water pumps, valves, connecting piping, and intake structure sluice gates.

- Control Room Emergency Air Conditioning and Air Filtration System.

- New and spent fuel storage racks.

- Process instrumentation and controls connected to and serving Seismic Class 1 systems and equipment.

- Reactor Protection and Engineered Safeguard Actuation Systems.

- Radiation Monitoring System Components: associated with reactor coolant leak detection or directly attached to other seismic Class I systems.
- Makeup and Purification System
- Fuel handling equipment
- Liquid Radwaste System
- Impingement and watertight doors

5.1.2.2 Seismic Class 2

Seismic Class 2 structures, systems, and equipment are defined as structures, systems, and equipment whose failure would not result in the uncontrolled release of radioactivity and would not prevent a safe reactor shutdown or the immediate and long-term operation following a LOCA. The failure of Seismic Class 2 structures, systems, and equipment may interrupt power generation.

5.1.2.2.1 Seismic Class 2 Structures

- Turbine building
- Administration building
- Emergency response facility
- Sodium bromide & sodium hypochlorite building
- Intake structure (excluding Service Water Systems)
- Transformer yard
- Turbine maintenance facility
- Alternate AC Power Source Building
- Original Steam Generator Storage Facilities (see Section 5.3.5.7 and the Unit 2 SAR for more detail)

5.1.2.2.2 Seismic Class 2 Systems and Equipment

- Pressurizer relief tank
- Reactor auxiliary systems
- Condensate Storage Tank
- Waste Disposal System (all elements not listed under Seismic Class 1)

- Reactor makeup water tank
- Turbine building crane
- Conventional equipment, tanks, and piping (not listed as Seismic Class 1)
- Alternate AC Power Source

5.1.3 DESIGN BASES

5.1.3.1 Seismic Class 1 Structures Design (Excluding Reactor Building)

Normal Operation - For loads to be encountered during normal plant operation, Seismic Class 1 structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

Accident, Seismic and Tornado Loads - The Seismic Class 1 structures are generally proportioned to maintain elastic behavior when subjected to various combinations of dead loads, thermal loads, accident loads, seismic and tornado loads. The upper limit of elastic behavior for structural steel is considered to be the yield strength of the structural steel member. The yield strength (Y) for steel (including reinforcing steel) is considered to be the guaranteed minimum given in appropriate American Society of Testing Materials (ASTM) specifications. The yield strength (Y) for reinforced concrete structures is considered to be the ultimate resisting capacity as calculated from the "Ultimate Strength Design" portion of the ACI-318-63 code, but with Ø (as defined in the ACI-318-63 Code) taken as unity. The allowable shear and axial stresses are also in accordance with the "Ultimate Strength Design" portion of the above code.

The final design of Seismic Class 1 structures satisfied the most severe of the following load combination equations. (Design equations for the reactor building are given in Section 5.2.1 of Chapter 5.)

$$Y = 1/\varnothing\ (1.25D + 1.0R + 1.25E)$$

$$Y = 1/\varnothing\ (1.25D + 1.25H + 1.25E)$$

$$Y = 1/\varnothing\ (1.25D + 1.25H + 1.25W)$$

$$Y = 1/\varnothing\ (1.0D + 1.8E)$$ (For structural elements carrying mainly earthquake forces.)

$$Y = 1/\varnothing\ (1.0D + 1.0R + 1.0E')$$

$$Y = 1/\varnothing\ (1.0D + 1.0H + 1.0E')$$

(0.90 D is used where dead load subtracts for critical stress in the first three equations.)

However, limited yielding of structures is allowed under missile forces. The ductilities assumed in the flexural design of the missile barriers are based on the values recommended in Chapter 7 of the "Air Force Design Manual - Principles and Practices for Design of Hardened Structures - Technical Documentary Report Number AFSWC - TDR - 62-138," published in December 1962. The minimum margin of safety is approximately 3.0, based on the ratio of ductility

recommended by the document and the design ductility. In all cases, the deflections are checked to insure that the affected Seismic Class 1 systems and equipment do not suffer loss of function and the structures retain their required integrities.

Y = required yield strength of the structure.

D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. In addition, a portion of "live load" is added when such load is expected to be present when the plant is operating. An allowance is also made for future permanent loads.

R = force or pressure on structure due to rupture of any one pipe.

H = force on structure due to thermal expansion of pipes under operating conditions.

E = "design earthquake load" resulting from ground acceleration of 0.1g.

E' = "maximum earthquake load" resulting from ground acceleration of 0.2g.

W = tornado load.

Ø = 0.90 for reinforced concrete in flexure.

Ø = 0.85 for shear, bond, and anchorage in reinforced concrete.

Ø = 0.75 for spirally reinforced concrete compression members.

Ø = 0.70 for tied compression members.

Ø = 0.90 for fabricated structural steel.

Ø = 0.90 for reinforcing steel (not prestressed) in direct tension.

The Reactor Building Containment and Engineered Safeguards Systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or crushing of barriers is permissible due to jet or missile impact, provided there is no general failure.

The final design of the missile barrier and equipment support structures inside the reactor building assures that they can withstand applicable pressure loads, jet forces, pipe reactions, and earthquake loads without loss of function. The deflections or deformations of structures and supports are checked to insure that the functions of the reactor building and engineered safeguards equipment are not impaired.

Tie-down arrangements for all Seismic Class 1 equipment against seismic and jet forces (as applicable), consist of anchoring the equipment to structural supports capable of withstanding these loads. Under normal operating conditions, in general, there are no gaps between seismic restraints and equipment in this plant. Seismic piping supports incorporate nominal gaps for thermal expansion per standard industry practice. See Appendix A.

The majority of Seismic Class 1 equipment is primarily housed in Seismic Class 1 structures designed to withstand tornado loading. Therefore, no special tie-down arrangements of such Seismic Class 1 equipment are considered necessary to resist tornado forces.

5.1.3.2 Seismic Class 1 Systems and Equipment Design

Components and systems classified as Seismic Class 1 are designed, where appropriate, in accordance with the provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Codes and American National Standards Institute (ANSI) B31.7, "Nuclear Power Piping."

Required control measures and interface procedures for design of all Seismic Class 1 items are outlined in the Quality Assurance Program Manual (QAPM).

The design of the reactor internals is discussed in Chapter 3. See Chapter 4 for the design of the RCS components.

5.1.3.3 Seismic Class 2 Structures, Systems and Equipment Design

Seismic Class 2 structures, systems, and equipment are designed in accordance with design methods of accepted codes and standards. Wind and earthquake loads, where applicable, conform to the requirements of the Uniform Building Code, Section 2314, 1967 Edition, Zone 1.

5.1.4 SEISMIC ANALYSES

5.1.4.1 General Criteria for Seismic Class 1 Structures, Systems and Equipment

The "design earthquake" used for this plant consists of 0.10g horizontal ground acceleration and 0.067g vertical ground acceleration acting simultaneously. The "maximum earthquake" consists of 0.20g horizontal ground acceleration and 0.133g vertical ground acceleration acting simultaneously. The design spectrum response curves are shown on Figures 5-10 and 5-11.

Earthquake analyses of the Seismic Class 1 structures, systems, and equipment are accomplished, where applicable, using the spectrum response or time history approaches, which utilizes the natural period, mode shapes, and appropriate damping values of the particular system, as discussed in Section 5.1.4.2 and 5.2.1.5.7, are shown on Figures 5-12, 5-13, and 5-14.

Details of the procedures used to incorporate soil-structure interaction in the mathematical models for seismic analyses of Seismic Class 1 structures are described in the following:

E	=	Modulus of Elasticity
μ	=	Poisson's Ratio
G	=	$\frac{E}{2(1+\mu)}$; Shear Modulus
R	=	Radius of Foundation
B	=	Width of Foundation
L	=	Length of Foundation
ϕ	=	Rotation of Rotational Spring
z	=	Deflection of Vertical Spring
βx & $\beta\phi$	=	Coefficients for Rectangular Footings
Kvs	=	Vertical Spring Coefficient
$K\phi s$	=	Rotational Spring Coefficient
Khs	=	Translational Spring Coefficient

To simulate soil properties, springs were introduced into the mathematical models for seismic analyses. The procedures used to develop these springs were different between circular bases (reactor building) and rectangular bases (auxiliary building and intake structure). Each procedure will be discussed separately. In both cases, an infinitely rigid structure in the vertical direction was assumed.

Circular Bases

Reference to the soil-structure interaction relationship may be found in the publication "Building Foundation Interaction Effects" by R. A. Parmelee from the ASEC Engineering Division Specifications Conference, October 12, 1966.

From this paper, we found:

$$K\phi s = - \frac{8GR^3}{3(1-\mu)}$$

$$Khs = \frac{4GR(1-\mu)}{(1-\mu)-0.125}$$

The rotational spring constant may be replaced by two vertical springs which may be expressed as follows:

$$K\phi s = RKvs\ 2R$$

$$Kvs = \frac{4GR}{3(1-\mu)} = \frac{2ER}{3(1-\mu^2)}$$

and

$$Khs = \frac{2ER(1-\mu)}{(0.875\mu)(1+\mu)}$$

Rectangular Bases

Reference to the soil-structure interaction relationship may be found in the publication "Design Procedures for Dynamically Loaded Foundations" by R. V. Whitman and F. E. Richart, Jr. presented in the Proceedings of ASCE Journal of the Soil Mechanics and Foundation Division, November 1967.

From this paper, we found:

$$K\phi s = \frac{G}{1-\mu} \beta\phi BL^2$$

$$Khs = 2(1+\mu)G\beta \times \sqrt{BL}$$

The rotational spring constant may be replaced by two vertical springs which may be expressed as follows:

$$K\phi s = \frac{L}{2} Kvs\ L$$

$$K\phi s = \frac{2G\beta\phi B}{1-\mu} = \frac{E\beta\phi B}{(1-\mu^2)}$$

and

$$Khs = E\,\beta \times \sqrt{BL}$$

After the spring constants have been determined, these springs are modeled by axial members and incorporated in the mathematical model by the following relationship: K = AE/Length.

The soil properties were used to determine the soil spring constants only. Rocking, vertical, translational, torsional damping effects due to soil-structure interaction were conservatively neglected.

The soil-structure interaction was investigated for the reactor building and the auxiliary building Seismic Class 1 structures, during an earthquake, with the following significant results:

A. Both structures are founded on good foundation rock at approximately the same elevations, with a 1-inch separation between the foundations and the structures.

B. Both structures show that the maximum predicted foundation bearing stresses, calculated under the critical loading combinations given in Section 5.2.1.4.5, are lower than the allowable bearing stresses.

Based on the above, soil-structure interaction for the two main Seismic Class 1 structures during an earthquake is predicted to be within allowable design limits.

Listed below are structural material and soil properties used in the seismic design analysis of Seismic Class 1 structures. The material properties were based on laboratory test results of samples, with the concrete in the containment using an applicable dynamic modulus of elasticity value. However, the soil properties were used to determine the soil spring constants only, and the recognized higher values of damping, due to soil-structure interaction, have been conservatively neglected.

CONCRETE AND SOIL PROPERTIES USED IN THE SEISMIC ANALYSIS

	CONCRETE			ROCK			
Structure	E, psi	μ	γ, pcf	E, psi	μ	K_H, kip/in	$K_{\phi S}$, kip/in
Reactor Building	6.7E10[6]	.18	150	.72E10[6]	.18	$\frac{4GR(1-\mu)}{(1-\mu)-.15}$	$\frac{8GR^3}{3(1-\mu)}$
	4.8E10[6]	.18	150	.72E10[6]	.18	$E\beta \times \sqrt{BL}$	$\frac{E}{2(1-\mu^2)\beta\phi BL^2}$

where:
E = Modulus of Elasticity
μ = Poisson's Ratio
γ = Weight Per Unit Volume
G = $\frac{E}{2(1+\mu)}$

R = Radius of Foundation
B = Width of Foundation
L = Length of Foundation
K_H = Horizontal Spring Constant
$K_{\phi S}$ = Rotational Spring Constant
β = Coefficients for Rectangular Footings

For comparison purposes, responses of Seismic Class 1 structures at selected points, obtained by the response spectrum and time history methods for 0.1 g ground acceleration and two percent critical damping are listed below:

	Responses at the Base of Structures			
	Shear, kips		Moment, kip-ft	
Structure	Response Spectrum Method	Time History Method	Response Spectrum Method	Time History Method
Reactor Building	18,035	19,832	1,810,000	2,414,000
Auxiliary Building	6,512	7,878	252,000	277,390
Intake Structure	2,250	2,961	66,798	85,757

	Responses at the Top of Structures			
	Acceleration, g		Deflection, in	
Structure	Response Spectrum Method	Time History Method	Response Spectrum Method	Time History Method
Reactor Building	.41	.44	.4	.400
Reactor Building Internals	.34	.27	.1	.09
Auxiliary Building	.25	.20	.21	.028
Intake Structure	.29	.28	.021	.028

It is noted that the response spectrum method results are obtained as the sum of the absolute modal values and not as the square root of the sum of the squares of modal values.

The seismic stress contributions were investigated at several critical locations of the Seismic Class 1 structures with the following results:

A. Reactor Building Base Slab (maximum degree of contributions)

Location of Reinforcing Steel	Stress Contribution	Governing Load Combination
Top	75%	$1.05D + 1.25P + 1.0T_A + 1.25E + 1.0F$
Bottom	85%	$1.0(D + P + T_A + E' + F)$

B. Reactor Building Wall

Location of Reinforcing Steel	Stress Contribution	Governing Load Combination
Outside Face	85%	$1.0(D + P + T_A + E' + F)$
Inside Face	75%	$1.05D + 1.25P + 1.0T_A + 1.25E + 1.0F$

The above stress allocations in percent are representative for the wall directly above the base slab. In general, seismic tensile stress contributions in the reactor building wall reinforcing steel are less than 50 percent.

C. Reactor Building Dome.

Earthquake forces in this region are small and the resulting stress contributions are nominal.

D. Auxiliary Building Exterior Walls at the Base.

The maximum predicted concrete compression stress due to seismic loads represents approximately 70 percent of the total stress.

All applicable Seismic Class 1 structures can adequately resist the combined effects of the postulated pipe rupture and seismic loads. The applicable structures also will have adequate capacity to resist further seismic loads following the decay of the postulated jet loads.

Material damping values are shown in the table below. These damping values were based on a conservative evaluation of values presented in the publication by Dr. N. M. Newmark, "Design Criteria for Nuclear Reactors Subject to Earthquake Hazards," Proceedings of the International Atomic Energy Agency Panel on a Seismic Design and Testing of Nuclear Facilities Japan Earthquake Engineering Promotion Society, Tokyo, Japan, 1967. These values were also reviewed and concurred with by Professor George W. Housner on November 3, I967.

	% Critical Damping	
	"Design Earthquake" (E) (0.1g ground acceleration)	"Maximum Earthquake" (E') (0.2g ground acceleration)
Welded steel plate assemblies	1	1
Welded steel framed structures	2	2
Bolted or riveted steel framed structures	2.5	2.5
Reinforced concrete equipment supports	2	3
Reinforced concrete frames and buildings	3	5
Prestressed concrete structures	2	5
Steel piping	0.5	0.5

The selection of three percent damping value for reinforced concrete structures was based on the following considerations:

A. The damping value for reinforced concrete frames and buildings in nuclear plants generally varies between two and five percent. The choice of three percent critical damping value for reinforced concrete buildings is consistent with several other nuclear power plants designed and built during the same period as Unit 1.

B. The selection of the three percent critical damping value is conservative because a composite damping value was not applied. In considering the foundation and structure interaction, the composite damping value would be higher than two percent even if damping of the fixed base reinforced concrete structure were assigned to be two percent critical. The increase in the composite damping value over the fixed base structural damping value is due to the combined effect of wave radiation and material damping in the foundation.

C. Even though the foundation damping is neglected, the difference in structural response between using two or three percent critical in the analysis is only about 10 to 15 percent. (Note that the fundamental frequency is about 11 cps for the auxiliary building and 11.5 cps for the intake structure).

D. While the application for construction permit was accepted on the basis of utilizing three percent damping value for all Seismic Class 1 reinforced concrete structures, only the designs of the auxiliary building, intake structure, and fuel storage vault incorporated this value. Internal reinforced concrete structures in the reactor building have been designed with two percent critical damping value.

5.1.4.2 <u>Seismic Class 1 Systems and Equipment</u>

5.1.4.2.1 <u>General</u>

All Seismic Class 1 systems and equipment are analyzed or test proven, and accordingly designed to withstand simultaneous seismic loadings in horizontal and vertical directions and the combination of gravity loads, operating loads, and applicable operating temperatures and pressure, without loss of function. Therefore, the operating mode is included in the analyses or tests. This includes Seismic Class 1 mechanical equipment such as pumps, heat exchangers, and electrical components such as cable trays, battery racks, instrument racks, and control consoles.

The general approach employed in the dynamic analysis of Seismic Class 1 systems and equipment design is based upon the spectrum response technique where applicable.

Basically, the dynamic analytical models used to predict the behavior of the structures are used to predict the behavior of the related systems and equipment.

Input data to this analysis is a closely related time history of acceleration or velocity based upon historical data and other simulated ground motions. The spectrum response curve generated, based upon this time history normalized for the site, is greater than or equal to the "design spectrum response curve."

This input ground spectrum response curve based upon time history, along with the basic analytical models, is used to produce a series of time histories and spectrum response curves at various support elevations for subsequent use in the analysis of systems and equipment throughout all required points.

The seismic analysis of systems and equipment is performed using one of the following methods:

A. If the equipment is structurally simple, the dynamic model may consist of one mass and one spring. Using the values of the mass and the spring constant, the natural frequency of the equipment may be determined. Acceleration response spectrum which corresponds to the location of the equipment produces the equipment acceleration when used in conjunction with the natural frequency and the equipment's damping. The accelerations, which are in terms of g's, are used in obtaining the inertia force by multiplying the equipment weight times the acceleration value.

B. In lieu of the detailed dynamic analysis, for those items of equipment which can be adequately represented by a single degree of freedom system the peak acceleration of the spectrum response curve may be applied. This acceleration value is then used to calculate the inertia force as outlined above.

C. If the equipment is structurally complex, the following method of analysis is used:

    1. The equipment is modeled by using lumped masses and spring constants.

    2. The natural frequencies and mode shapes for the equipment are determined.

    3. The damping value of the equipment is determined.

    4. Using the natural frequencies, damping, and the appropriate acceleration response spectrum, the spectral accelerations per mode are determined.

    5. The effective forces and weights per mode are calculated from the acceleration values obtained above.

    6. The resulting response values per mode of inertia forces, shears, and moments are determined from these effective forces and weights. The response values per mode are combined by using the square root of the sum of the squares.

D. For rigid equipment having natural frequencies greater than 30 cps, the response accelerations are equal to the accelerations of the supporting structure at the appropriate elevation. The design criteria for both structures and equipment required that the calculated stresses from seismic, combined with other loads, be below yield of the material. Since the calculated stresses were below yield, no cycle loading was considered in the design. For the small number of cycles from an earthquake, fatigue modes of failure of material are not applicable for structures and equipment. Seismic Class 1 piping systems are designed according to the following:

    1. ANSI B31.7 Nuclear Class 2, 3 and B31.1 piping systems do not require the use of the number of earthquake stress cycles in the analysis.

    2. Cyclic data for ANSI B31.7 Nuclear Class 1 piping systems are based on the El Centro spectrum and general data complied for Southern California earthquake histories. One maximum earthquake is assumed to occur during the life of the plant. This is a 20-second duration earthquake which, for a 10-cycle oscillation, amounts to 200 cycles. For the design earthquake, data collected in Southern

California for 100 years indicates that an earthquake of greater than a 5.0 Richter magnitude amounted to a total of 130 seconds of earthquake. From this, we derive 60 2.2-second duration earthquakes for the life of the plant. At 22 cycles per earthquake this yields 1,300 cycles total. However, Southern California is a UBC area 3 earthquake zone whereas Arkansas is only a UBC area 1. Therefore, these design cycles are divided by two yielding 650 total cycles.

Where Class 1 seismic structures are directly connected to or in close proximity to Class 2 seismic equipment and piping systems, the failure or excessive movement of the Class 2 seismic systems are restrained in such a way as not to cause a failure of Class 1 structures. Where Class 1 piping is directly connected to Class 2 piping, the entire system is analyzed to the first Class 1 seismic anchor (even if located in a Class 2 structure) as an integral Class 1 piping system.

E. An alternative method for the seismic design and verification of new, modified and replacement equipment (e. g., seismic equipment qualification) is to use earthquake and seismic testing experience.

Peak recording instruments mounted on selected Category 1 equipment will be located so as to insure a truly representative account of the response of vital equipment under seismic excitation. Following a seismic event, data from these peak recording instruments will be tabulated and compared with the corresponding maximum design values, which were based on conservative assumptions concerning damping characteristics, in order to verify that equipment integrity has not been compromised.

5.1.4.2.2 Special Conditions

For certain Seismic Class 1 systems and equipment, where analytical models and normal theory do not produce results of a significant confidence level, dynamic testing of prototypes or similar equipment is substituted to insure functional integrity. Test data, when selected, conforms to one of the following:

A. Performance data of equipment which, under the specified conditions, has been subjected to equal or greater dynamic loads than those to be experienced under the specified seismic conditions.

B. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to equal or greater dynamic loads than those specified.

C. Actual testing of equipment in accordance with one of the following methods:

   1. The equipment is subjected to an artificial time history response at the elevation of interest.

   2. The equipment is subjected to a sinusoidal excitation, sweeping through the desired range of significant frequencies, using input acceleration amplitudes for the forcing function which simulates the specified seismic conditions.

   3. The equipment is subjected to a transient sinusoidal motion synthesized by a pulse exciting a group of octave filters such that the response of the shaking table and the duration of loading simulates the artificial response spectrum curve at the elevation of interest.

The equipment design specifications provided for dynamic testing such as those outlined above. Since these rules may apply to different types of equipment, a dynamic testing procedure was not issued. Instead, the manufacturer of specific equipment must select the proper procedures to comply with specified seismic and operational requirements.

5.1.4.2.3 Seismic Supports and Restraints

All Seismic Class 1 supports and restraints, including those required for mechanical equipment and electrical components, were located by Bechtel project engineering personnel after an appropriate seismic analysis determined the requirements. The required location and type of restraint was then sent to the restraint supplier for design details. These details were carefully reviewed by the architect engineering staff personnel to assure that the location, function, and overall design was compatible with the seismic analysis. Seismic Class 1 pipe anchors and pipe supports were analyzed taking into account the flexibility of the base plate and bolt stiffness. The effects of dynamic and cyclic loads on expansion anchors installed in concrete were also taken into account.

Subsequent to installation, an engineer from the Bechtel engineering staff reviewed the installation at field location to insure compliance with the restraint detail and the assumptions in the analysis.

5.1.4.3 Seismic Class 2 Structures, Systems and Equipment

Seismic design of Class 2 structures is in accordance with the Uniform Building Code (UBC). The area is in Zone 1 in the seismic risk map of the UBC.

The Class 2 systems and equipment are designed to withstand normal design loads combined with a horizontal acceleration of 0.05g.

5.1.5 WIND AND TORNADO LOADS

Design wind loads are based on ASCE Paper 3269, "Wind Forces on Structures."

A. "Design" Wind Load (except tornado) is in accordance with the following tabulated values determined from ASCE 3269 paper for a basic wind velocity of 67 mph, 1.1 gust factor, and 1.3 shape factor:

Height Above Ground (Feet)	Design Pressure PSF
0-30	20
30-50	20
50-100	26
100-150	26

B. Tornado. Class 1 structures are analyzed for tornado loading (not coincident with accident or earthquake) on the following basis:

1. Differential bursting pressure between the inside and the outside of the structures – three psi positive pressure.

2. Lateral force on the structures will be assumed as the force caused by a tornado funnel having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph. The tornado resistant design of the ANO Units was completed prior to the issuance of Regulatory Guide 1.76. As a result, the radius of maximum rotational speed is not significant to the ANO design, and Category I structures were designed considering a uniform pressure resulting from the 300 mph wind velocity (Reference 0CAN059609), except for casks which are governed by Holtec HI-Storm and VSC-24 CFSARs. The applicable portions of wind design methods described in ASCE paper 3269 are used, particularly for shape factors. The provisions for gust factors and variations of wind velocity with height do not apply.

3. Tornado driven missiles equivalent to an airborne 4" x 12" x 12' plank traveling end-on at 300 mph or a 4,000-pound automobile flying through the air at 50 mph and at not more than 25 feet above the ground is assumed.

The NRC approved a license amendment request for ANO-1 that authorized the use of the Tornado Missile Risk Evaluator (TMRE) methodology (Reference letter 0CNA062003). TMRE is a risk-informed methodology based on NEI 17-02, "Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document," Revision 1B, for identifying and evaluating the safety significance associated with SSCs that are exposed to potential tornado-generated missiles and may be used as an alternative methodology for determining whether protection from tornado-generated missiles is required. The methodology can only be applied to discovered conditions where tornado missile protection was not provided and cannot be used to avoid providing tornado missile protection in the plant modification process. Should the result of a future change/modification exceed those assumptions and the risk metric thresholds in the TMRE methodology, NRC approval would be required before staging material that would create a missile count in excess of 240,000 missiles.

The TMRE evaluation demonstrated that tornado-generated missile protection is not required for the following:

- Green train safety-related electrical raceways that are vulnerable to a northern bound tornado-generated missile through small fire damper openings FD-0135-02-0009, FD-0135-02-0022, and FD-0141-02-0011 in the south wall of Controlled Access on Elevation (EL) 386’ (reference CR-ANO-1-2016-2752).
- Unprotected conduits in the demineralizer area (Room 73, Unit 1 “Bowling Alley,” EL 354’) on the east side of the hallway near Column Line 5.9 (reference CR-ANO-1-2017-1171).
- A cable tray opening in the reinforced concrete wall 3-S-23 between Electrical Equipment Rooms 103 and 104 on EL 368’ (reference CR-ANO-1-2017-1171).
- P-7A Emergency Feed Water (EFW) Pump Steam Supply Line located above EL 404’ (reference CR-ANO-1-2015-3940).
- Conduit EC1493 which includes the reactor head vent solenoid valve (reference CR-ANO-1-2016-02752).
- Small bore service water piping (HCD-65-2” and HCD-66-2”) to VCH- 4A and 4B pumps (reference CR-ANO-1-2017-3702).

The design bases for tornado loads for the reactor building are presented in Section 5.2.1.2.6. Components in Class 1 buildings, such as penetrations, locks, doors, large openings, spent fuel pool, emergency diesel generator rooms, etc., are inherently tornado protected by virtue of their being housed in tornado resistant structures. Penetrations which are not located within a tornado resistant structure or protected by a missile shield are the main steam line and purge line penetrations. These penetrations will withstand tornado winds and pressure drop loadings and it is highly improbable that they will be pierced by tornado missiles. Further analysis also indicates that, in the unlikely event these penetrations are impinged by tornado missiles, the incident will not cause a LOCA or prevent a safe shutdown of the plant. The spent fuel pool walls are inherently resistant to missiles. A discussion on the enclosure over the new and spent fuel pool storage facilities is included in Section 5.3.2. Investigations of previous tornadoes demonstrate that no funneling effect of the tornado winds is to be expected between the Unit 1 and Unit 2 containment structures.

The possibility of the generation of secondary missiles within the structure from tornado missiles was analyzed. The 3-inch diameter pipe, as described in Section 5.2.1.2.6 was considered as potentially damaging due to its high density. However, based on results of U. S. Army Corps of Engineers investigation (5), only 13 inches of concrete thickness is required to prevent spalling of 3,000 psi concrete from an impact by the 3-inch diameter pipe missile. Since 18 inches is the minimum exterior wall and slab thickness provided for Class 1 structures, the safety related equipment and systems will not be affected by secondary missiles.

Intervening SSCs, including large equipment and structural members, may be utilized as missile barriers provided they are qualified for the applicable design loads including tornado winds and differential pressure. For missile barrier design, BC-TOP-9A in conjunction with EPRI NP-440 may be utilized. EPRI NP-440 provides a basis for missile deformation and impact characteristics for designing missile barriers.

5.1.6 FLOODING

The Seismic Class 1 structures are designed for the flood level at elevation 361 feet (MSL). All Seismic Class 1 systems and equipment susceptible to the effects of flooding that are located within Seismic Class 1 structures are either located on floors above elevation 361 feet or protected by the following measures:

A. Wall thicknesses in Seismic Class 1 structures below flood level are a minimum of two feet with certain exceptions, which have been evaluated to ensure their design is adequate to withstand the hydrostatic forces associated with external flooding.

B. Potential flood paths through construction joints are addressed through the use of waterstops or other seal materials to maintain the required flood protection for Class 1 structures.

C. The number of openings in walls and slabs below flood level are kept to a minimum.

D. Watertight doors, watertight equipment hatches, and flood seals were installed as required.

E. Possible local seepage through the walls will be controlled by sumps and sump pumps.

Exterior openings and penetrations below the 361-foot flood level and the type of qualified flood protection features provided in the Seismic Class 1 structures, are identified and described as follows:

Structure	Opening or Penetration	Type of Flooding Protection
Reactor Building	Equipment hatch	Double seal in hatch cover
	Escape lock	Double seal in lock doors
	Tendon gallery access hatch	Water tight hatch cover
Auxiliary Building[1]	Door openings	Watertight doors
	Floor openings	Watertight hatch covers
	Roof openings over underground vaults	Concrete plugs with neoprene seals
	Pipe, conduit, or HVAC penetrations	Silicone-based seals, cementitious-based seals, urethane-based seals, boot seals, mechanical seals, or closure plates
Emergency Diesel Fuel Storage Vaults	Door opening	Watertight door
	Roof opening	Concrete plug with neoprene seals seal weld
	Pipe, conduit, or HVAC penetrations	Silicone-based seals, cementitious-based seals, urethane-based seals, boot seals, mechanical seals, or closure plates

[1] Includes the PASS annex

Seismic Class 2 structures are designed for a design flood level at elevation 338 feet (MSL). Waterstops in construction joints are installed below this elevation.

5.1.7 DESIGN LOADS

5.1.7.1 Normal Operating Loads

A. Dead Load (D.L.) includes weight of framing, roof, floors, walls, partitions, platforms, and all permanent equipment and material.

B. Live Load (L.L.) includes all vertical loads except D.L. The following are the stipulated minimum values in various area:

Roof snow load	20 psf
Offices	50 psf
Assembly rooms,	100 psf
Laboratories, locker rooms and laundry rooms	100 psf
Stairways and walkways	100 psf
Platforms with grating floors (except as local use dictates)	100 psf

Operating floor in turbine building (except for designated laydown)	350 psf (or 5 kip wheel-load)
Laydown area so designated on turbine building operating floor	1000 psf
Control room	250 psf
Cable spreading room	250 psf
Fuel cask wash down area	Cask load 130 tons dimension 6' square in plan x 19'
Spent fuel storage pit	Hydrostatic load, cast load, fuel assembly load, and equipment load
New fuel storage area (fuel assembly load)	1000 psf
Refueling floor in containment	700 psf
Ground floors and elevated concrete floors, unless otherwise noted on building	250 psf (or 5 kip wheel-loads)
Surcharge load outside and adjacent to structure	250 psf

Note: Above unit live loads include normal piping and conduit weight. Main piping must be added. A concentrated load shall be considered where the equipment load divided by the area of equipment plus a one-foot wide perimeter exceed the unit live load.

C. Design of the crane support structures will be based upon the following crane hook load capacities.

1.	Turbine Building Crane	150 ton
2.	Reactor Building Crane	190 ton
3.	Fuel Handling Crane	130 ton
4.	Machine Shop Crane	5 ton
5.	Intake Structure Gantry Crane	25 ton

Note: See Section 9.6.1.7 for the Design Load Rating of the Reactor Building and Fuel Handling Cranes.

The vertical impact force for traveling crane support girders and their connections will be 25 percent of the total vertical load.

The lateral impact force on the crane runways will be 20 percent of the sum of the weights of the lifted load and crane trolley applied at the top of the rail, one half on each side of the runway, and will be considered as acting in either direction normal to the crane rail.

The longitudinal impact force shall be 10 percent of the maximum wheel load of the crane applied at the top of the rail. The weights of the crane, wheel spacings, and maximum wheel loads to be used for the design of the support structure will be determined on the basis of past experience. However, the design for the crane support structure will be checked after the final drawings and loadings are available from the crane manufacturer.

D. In addition to the live loads given above, beams over equipment shall be designed for an additional 2_k anywhere on the beam to account for lifting. This load need not be carried to the columns.

E. Railroad--------------------- Cooper E72

F. Road Bridge------------------------- H-20

G. Live Load Reductions:

1. For live loads greater than 100 psf no reduction is allowed.

2. For loads of 100 psf or less, the provisions of the UBC will be followed.

H. Lateral Earth and Groundwater Pressures.

Lateral earth pressure will be assumed to act as an equivalent fluid.

1. For cantilevered walls and walls not rigidly supported at its crest, the "Active" condition will govern.

2. For rigid walls and walls rigidly supported at its crest, the "At Rest" condition will govern.

Water table at Elevation 338' feet will be assumed for normal operating condition. However, Seismic Class 1 structures will be checked against failure for maximum probable flood.

Weight of backfill material will be assumed as 120 p.c.f.

I. Pressure and Thermal Loads

1. <u>Reactor Containment Structure</u>

Design internal pressure (accident)	59
Design external pressure	2.5 psig
Design internal temperature (normal)	110 °F
Design internal temperature (accident)	286 °F

2. <u>Spent Fuel Pit</u>

Design and internal temperature	142 °F

J. Missile and Jet Loads

Individual missile sources will be checked and adequate missile protection installed as required. Missile barriers will be designed on the basis of absorbing energy by plastic yielding; the only criteria for design being that they withstand the missile design conditions postulated.

5.1.8 CODES AND STANDARDS

The civil and structural design conforms to the applicable portions of the following codes:

A. General — Uniform Building Code (UBC), 1967 Edition.

B. Steel — American Institute of Steel Construction (AISC), Manual of Steel Construction, 6th Edition, April 1963.

C. Concrete — American Concrete Institute (ACI-318-63), Building Code Requirements for Reinforced Concrete.

- Mechanical splicing of reinforcing steel using "swaged" splices associated with the Once Through Steam Generator/Reactor Vessel Closure Head (OTSG/RVCH) replacement was in accordance with ASME Section III, Division 2, 1989 Edition with No Addenda.

D. Welding — American Welding Society (AWS) D1.1-92 (or later approved Edition), AWS D1.4-92 (or later approved Edition). AWS D1.0 is no longer an active welding code and has been superseded by AWS D1.1. Likewise, AWS D12.1 has been superseded by AWS D1.4. Alternatively, Welding Procedure Specifications (WPS's) and/or Welder Performance Qualification (WPQ) to the ASME Section IX Code requirements may be permitted for structural steel applications in lieu of AWS D1.1 WPS's and WPQ's provided all other provisions of AWS D1.1 are met unrelated to WPS's and WPQ's

General welding associated with the OTSG/RVCH replacement was in accordance with AWS D1.1-92. Reinforcement steel (rebar) welding was in accordance with AWS D1.4-98.

E. Containment Liner and Penetration Welding — American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Repair/replacement of reactor building liner plate associated with OTSG/RVCH replacement was in accordance with ASME Section XI, IWA/IWE-4000, 1992 Edition with 1992 Addenda, and with ASME Section VIII, 1989 Edition with No Addenda.

F. Railroad — American Railway Engineering Association (AREA), Manual of Recommended Practice.

G. Circulating Water Pumps — American Water Works Association (AWWA), AWWA M-11 Steel Pipe Manual.

H. Turbine Pedestal — Bechtel Standard C-331, 11-10-67, and Westinghouse Manual AD 1001.01.

I. Wind — American Society of Civil Engineers (ASCE), Paper 3269, Wind Forces on Structures.

J. Seismic — AEC Publication TID 7024, Nuclear Reactors and Earthquakes, and UBC, 1967 Edition.

K. Road Bridge — American Association of State Highway Officials (AASHO), Standard Specification for Highway Bridges, 1965.

5.1.9 MATERIALS AND ALLOWABLE STRESS

5.1.9.1 General

Applicable codes will be used for load combinations and allowable stresses for normal design conditions.

5.1.9.2 Containment Vessel

For load combination and allowable stresses of containment design, see Section 5.2.

5.1.9.3 Materials

5.1.9.3.1 Concrete

Concrete design strengths for the various structures will be based upon the following table:

	Strength P.S.I.		
Item	3-Day	28-Day	90-Day
Reactor Building - Base Slab and Gallery		-	4000
Reactor Building - Walls, Dome and Internals		-	5500
Auxiliary Building		3000	-
Turbine Building		3000	-
Administration Building		3000	-
Intake and Discharge Structures		3000	-
Circulating Water Canal		3000	-
Reactor Building – OTSG/RVCH Construction Opening	5500		

Design concrete strengths (fc) are specified for various structures. Design strength values are less than measured concrete strength values. Documents such as Engineering Standard SES-11 record measured concrete strength values for the containment building, auxiliary building, turbine building, and other structures. Measured concrete strength values are usable input in future design, operability evaluations, and reconciliation. Prior to extrapolating concrete strength values, consideration should be given to Information Notice 2012-17 and requirements in SES-11.

5.1.9.3.2 Reinforcing Steel

Reinforcing Steel will conform to ASTM A615-68 Grades 40 and 60 as follows.

5.1.9.3.2.1 Grade 40

A. Turbine building foundations.

B. Auxiliary building foundations.

C. Reactor building (except as below).

5.1.9.3.2.2 Grade 60

A. Discharge flume (outside the turbine building) and the intake and discharge structures.

B. Reactor building base slab and openings.

C. New reinforcing steel for OTSG/RVCH construction opening conforms to ASTM A615-03, Grade 60.

5.1.9.3.3 Structural Steel

Structural steel will conform to ASTM A36 or better.

5.1.9.4 Allowable Stresses

Structural steel as specified in AISC Code Reinforced Concrete as specified in ACI-318.63 Reinforcing Steel as specified in ACI-318.63.

5.1.10 MISCELLANEOUS DATA

A. Elevations

Plant datum is U.S.C. & G.S. datum. Nominal plant site elevation 353 feet.

B. Rainfall

Average annual	52.5"
Maximum annual	69.3"
Maximum for 24-hour period (10 years storm)	7.3"
Maximum for 1-hour period (10 years storm)	2.5"

C. Snow

Approximately 0.6 feet per year with a negligible amount recorded in 24 hours.

D. Frost

18" below grade

E. Temperature

Extremes approximately ±

Average of 93 days per year greater than 90 °F

Average of 74 days per year less than 32 °F

Average of 2 days per year less than 0 °F

F. Wind

Fastest mile of wind - 67 mph with both 50 years and 100 years recurrences.

G. Groundwater Table

Piezometers installed in the plant area indicate that the groundwater table is at Elevation 337'. However, the groundwater table may be expected to ultimately rise to Elevation 338'.

H. Soil Condition (Generalized)

The overburden in the general plant area consists of one to four feet of mainly soft sandy clays and clayey silts, underlain by stiff to very stiff highly plastic clays. Bedrock, encountered at depths varying from 13 to 23 feet, consists of two to five feet of weathered dark gray shale underlain with a hard and dense dark gray, horizontally bedded shale.

I. Lake and Flood Levels

Low water level of the reservoir is elevation 336 feet. Normal water level is Elevation 338'. "Design" flood does not rise above the normal water level. "Maximum probable flood", combined with an upstream dam failure, could raise the water level at the plant site to elevation 361 feet.

J. Allowable Soil Bearing Pressure

Unweathered shale (Normal Operating Conditions)	12 Tons/S.F.
Unweathered shale (Earthquake)	19 Tons/S.F.
Compact Backfill (Normal Operating Conditions)	2.5 Tons/S.F.
Compacted Backfill (Earthquake)	3.0 Tons/S.F.

K. Foundations

The reactor building, auxiliary building, turbine pedestal, turbine building, intake structure, and the transformer foundations are founded on rock. The foundations for the administration building, tube pulling pit, other yard structures, including storage tanks and a portion of the ground floor in the turbine and auxiliary building are founded on compacted backfill.

L. Area Drainage

In general, surface water drains in a southerly direction, flowing through open ditches, road culverts, catch basins, or storm sewers, finally discharging in the intake and discharge canals. Domestic water supply is obtained from the Russellville water supply system and stored in the raw water storage tank.

5.2 REACTOR BUILDING

5.2.1 STRUCTURAL DESIGN

5.2.1.1 General Description

The reactor building is a Seismic Class 1, fully continuous, reinforced prestressed concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab.

The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a 3-way post-tensioning system. Hoop tendons are placed in 3 - 240 degree systems using three buttresses as anchorages. The foundation slab is conventionally reinforced with high strength reinforcing steel and is founded on bedrock. A continuous access gallery is provided beneath the base slab for installation of vertical tendons. A welded steel liner is attached to the inside face of the concrete shell to insure a high degree of leak tightness. The base liner is installed on top of the structural slab and is covered with concrete. The structure provides shielding for both normal and accident conditions.

The reactor building completely encloses the entire reactor and Reactor Coolant System (RCS). This insures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the RCS occurs. The building also encloses portions of the auxiliary and engineered safeguards systems. The dimensions of the reactor building are: inside diameter, 116 feet; inside height, 208-1/2 feet; vertical wall thickness, 3-3/4 feet; dome thickness, 3-1/4 feet; and the foundation slab, 9 feet. The internal net free volume is approximately 1.81×10^6 cubic feet.

A personnel access lock and a personnel escape lock are located at the northeast and southwest sides of the building. The inner and outer doors are interlocked to maintain integrity of the containment during normal plant operation. An equipment hatch on the west side of the building permits passage of large items. All the penetrations through the cylindrical wall for the passage of pipes, ducts, and electrical conduits are designed and sealed to maintain containment integrity.

Full advantage was taken in the design of this reactor building from the experience gained in similar designs for the Florida Power and Light Company's Turkey Point Plant, Consumers Power Company's Palisades Plants, Wisconsin-Michigan Power Company's Point Beach Plant, and the Duke Power Company's Oconee Nuclear Station.

Mathematical symbols used in these sections are generally those used in the ACI 318-63 and the AISC Manual.

5.2.1.1.1 Description of OTSG/RVCH Replacement Construction Opening

During the Once Through Steam Generators/Reactor Vessel Closure Head (OTSG/RVCH) replacement a temporary construction opening was made in the wall of the reactor building. The opening was at azimuth 270°, approximately 23'-4" wide by 24'-0" high. The bottom was at approximate elevation 401'-6". The opening was sized to allow moving the OTSGs and the RVCH into and out of the reactor building.

As part of the modification, 6 vertical tendons (V75 to V80) and 16 hoop tendons (21H21 to 21H28 and 31H21 to 31H28) were removed and replaced with new tendons. Another 30 vertical (V60 to V74 and V81 to V95) and 18 hoop tendons (21H16 to 21H20, 21H29 to 21H32, 31H17 to 31H20, and 31H29 to 31H33) around the opening were detensioned and retensioned. Tendons that were degreased or partially degreased were regreased.

Existing reinforcement in the opening was removed and either reinstalled or replaced with new equivalent reinforcement. Additional new reinforcement was added on the interior face of the wall. New or replaced reinforcement was installed using mechanical splices to attach to existing rebar projections with limited fusion welding.

Existing spiral wound tendon sheaths in the opening were removed and replaced with pipe tendon sheaths.

The existing concrete in the opening was removed by means of hydro-demolition. The replacement concrete is a specially designed mix to achieve required strength at an early age (3 days) with no shrinkage and low creep characteristic. The concrete was produced on site with a portable automated concrete batch plant.

A section of the existing liner plate (approximately 17'-8" x 21'-6 3/8") was removed and reinstalled. While removed, it was repaired as necessary and prepared for reinstallation. Additional C5 x 6.7 stiffeners were added to strengthen the liner plate. All welds to or in the liner plate were to ASME Section VIII requirements.

The restored reactor building with the opening replaced was analyzed and checked to meet the requirements of FSAR Section 5.2. The functionality of the reactor building in the restored condition is the same as in the pre-opening condition.

5.2.1.2 Basis for Design Loads

The reactor building is designed for all credible conditions of loading, including normal loads, loads during a LOCA, test loads, and loads due to adverse environmental conditions. The following loadings are covered:

A. The loadings caused by the pressure and temperature transients of the Design Basis Accident (DBA)

B. Thermal loads (operating and accident)

C. Dead loads

D. Live loads

E. Earthquake loads

F. Wind force and tornado loads

G. Uplift due to buoyant forces

H. External pressure load

I. Prestressing loads

The critical loading conditions are those caused by a LOCA resulting from failure of the RCS and those caused by an earthquake.

5.2.1.2.1 Design Basis Accident Load

The design pressure and temperature of the reactor building exceeds the peak pressure and temperature occurring as a result of any rupture of the reactor coolant pipe.

Transients resulting from the LOCA and other lesser accidents are presented in Chapter 14 and serve as the basis for the reactor building design pressure of 59 psig.

5.2.1.2.2 Thermal Load

The temperature gradient through the wall during operation and during the LOCA is shown in Figure 5-5. Both summer and winter conditions are considered. The temperature at the inside of the wall is conservatively increased to the design temperature of 286 °F to accommodate future DBA considerations. The variation of temperature with time and the expansion of the liner plate was considered in designing for the thermal loads associated with the DBA.

5.2.1.2.3 Dead Load

Dead loads consist of the weight of the concrete wall, dome, base slab, and any internal concrete. Weights used for dead load calculations are as follows:

A. Concrete — 148 lb/ft^3

B. Steel Reinforcing — 489 lb/ft^3, using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes and nominal cross-sectional areas of prestressing wires.

C. Steel Liner — 489 lb/ft^3, using nominal cross sectional area of liner.

5.2.1.2.4 Live Loads

Live loads include snow loads on the roof of the reactor building dome. The design roof load is 20 pounds per horizontal square foot.

Equipment loads are those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

Live loads used in the design of internal slabs are consistent with the intended use of the slabs.

5.2.1.2.5 Earthquake Loads

Earthquake loading is predicated upon a design earthquake at the site having a horizontal ground acceleration of 0.10g. In addition, a maximum earthquake having a horizontal ground acceleration of 0.20g is used to check the design to insure no loss of function. Seismic design spectrum curves for horizontal ground acceleration are given in Section 5.1. A simultaneous vertical component two-thirds of the magnitude of the horizontal acceleration was applied. A dynamic analysis was used to determine equivalent static loads for design.

5.2.1.2.6 Wind and Tornado Loads

The structure has been analyzed for tornado loading (not coincident with accident or earthquake) on the following basis:

A. Differential bursting pressure between the inside and outside of the reactor building structure is assumed to be three pounds per square inch positive pressure occurring in three seconds (one psi per second), followed by a calm for two seconds and a repressurization.

B. The wind force on the reactor building structure is considered as a uniform static load caused by a tornado funnel having a peripheral tangential velocity of 300 mph and a forward progress of 60 mph. The tornado resistant design of the ANO Units was completed prior to the issuance of Regulatory Guide 1.76. As a result, the radius of maximum rotational speed is not significant to the ANO design, and Category I structures were designed considering a uniform pressure resulting from the 300 mph wind velocity (Reference 0CAN059609), except for casks which are governed by Holtec HI-Storm and VSC-24 CFSARs. The applicable portions of wind design methods described in ASCE Paper 3269 are used, particularly for shape factors. The provisions in ASCE Paper 3269 for gust factors and variation of wind velocity with height do not apply.

C. Tornado driven missiles equivalent to an airborne 4" x 12" x 12' wooden plank traveling end-on at 300 mph or a 4,000-pound automobile flying through the air at 50 mph and at not more than 25 feet above the ground with a contact area of 20 square feet were assumed. In addition, a tornado missile equivalent to a 3-inch diameter schedule 40 pipe, 10 feet long, traveling end-on at 100 mph, striking anywhere over the full height of the structure was considered.

A discussion of the frequency of tornado occurrence is presented in Chapter 2.

5.2.1.2.7 Uplift Due to Buoyant Forces

Uplift forces which are created by the displacement of groundwater by the structure have been accounted for in the design of the structure.

5.2.1.2.8 External Pressure Load

External pressure loading with the outside pressure 2½ psi greater than inside is considered.

The external design pressure is adequate to permit the reactor building internal atmosphere to be cooled to 50 °F from an initial maximum operating condition of 110 °F.

5.2.1.2.9 Prestressing Loads

Prestressing forces are considered in all loading combinations.

5.2.1.3 Construction Materials

Basically five materials were used for the reactor building structure. These are:

A. Concrete

B. Reinforcing steel

C. Steel prestressing tendons

D. Steel liner plate

E. Interior coatings

Detailed specifications and working drawings for these materials and their installation are of such scope as to assure that the quality of work is commensurate with the necessary integrity of the reactor building structure.

Basic specifications follow in Sections 5.2.1.3.1 through 5.2.1.3.5.

5.2.1.3.1 Concrete

All concrete work is done in accordance with ACI 318-63, "Building Code Requirements for Reinforced Concrete" and to ACI-301, "Specifications for Structural Concrete for Buildings." Concrete consists of a dense, durable mixture of round, coarse aggregate, fine aggregate, cement, and water. Admixtures were added to improve the quality and workability of the fluid concrete during placement and to retard the set of the concrete. Maximum practical size aggregate, water reducing additives, and low slumps were used to minimize shrinkage and creep. Aggregates conform to ASTM Designation C33, "Standard Specifications for Concrete Aggregate." Fine aggregate consists of sharp, hard, strong, and durable sand, free from adherent coating, clay loam, alkali, organic material, or other deleterious substances.

Inspection, testing and acceptability of aggregates are based on the ASTM tests discussed in Section 5.2.3.3. These tests have been performed by a qualified commercial testing laboratory.

Cement is Type II, low alkali cement as specified in ASTM Designation C150, "Standard Specifications for Portland Cement," and has been tested to comply with ASTM C-114.

Water for mixing concrete is clean and free from deleterious amounts of acid, alkali, salts, oil, sediment, or organic matter. A water reducing agent is employed to reduce shrinkage and creep of concrete. Admixtures containing chlorides have not been used. The following types of agents were tested with the concrete materials selected for the reactor building structure: Pozzolith No. 8, Doratard, Plastiment, HPSR Green Label, and Placewel LS.

The agent selected was Placewel LS, the one providing the smallest shrinkage as determined by ASTM C-494, Type "D" Specifications for Chemical Admixtures for Concrete.

Concrete mixes are designed in accordance with ACI-613 using materials qualified and accepted for this work. Only mixes meeting the design requirements specified for reactor building concrete were used. Trial mixes were tested in accordance with applicable ASTM Codes as indicated below:

Test	ASTM
Making and Curing Cylinder in Laboratory	C-192
Air Content	C-231
Slump	C-143
Bleeding	C-232
Compressive Strength Tests	C-39

Eight cylinders were cast from each design mix for two tests on each of the following days: 3, 7, 28, and 90 days. The concrete has a design compressive strength of 5,500 psi at 90 days for the reactor building wall and dome and 4,000 psi at 90 days for the reactor building base slab.

Test cylinders were cast from the mix proportions selected for construction and the following concrete properties were determined.

- Uniaxial creep
- Modulus of elasticity and Poisson's ratio
- Autogenous shrinkage
- Thermal diffusivity
- Thermal coefficient of expansion
- Compressive strength

A qualified organization familiar with determining creep and corresponding factors was retained to perform the above test.

A concrete technologist participated in the preparation of concrete specifications, placement procedures, and field quality control and testing programs. The technologist made periodic site visits during construction.

An independent laboratory tested the concrete mixes. To maintain the quality of the mix used in the structure, the workability and other characteristics of the mixes were ascertained before placement. An onsite concrete control laboratory was set up. A batch plant inspector was assigned and testing, as shown below, was performed. Field control was in accordance with the ACI Manual of Concrete Inspection as reported by Committee 611.

Aggregate testing was carried out as follows:

A. Sand Sample for Gradation (ASTM C-33 Fine Aggregate)

B. Organic Test on Sand (ASTM C-40)

C. 3/4-Inch Sample for Gradation (ASTM C-33 Size No. 67)

D. 1 1/2-Inch Sample for Gradation (ASTM C-33 Size No. 4)

E. Check for Proportion of Flat and Elongated Particles

Concrete samples were taken from the mix every 100 cubic yards according to ASTM C-172, "Sampling Fresh Concrete." From these samples, cylinders for compression testing were made. They were stripped within 24 hours after casting and marked and stored in the curing room. These cylinders were made in accordance with ASTM C-31, "Tentative Method of Marking and Curing Concrete Compression and Flexure Test Specimens in Field."

Slump, air content, temperature, and weight measurements were taken when cylinders were cast. Slump tests were performed in accordance with ASTM C-143, "Standard Method of Test for Slump of Portland Cement Concrete." Air content tests were performed in accordance with ASTM C-231, "Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive strength tests were made in accordance with ASTM C-39, "Method of Test for Compressive Strength of Molded Concrete Cylinders."

Evaluation of compression tests was performed in accordance with ACI 214-65.

Cracking due to drying of the hardened concrete and/or thermal stresses was minimal due to the concrete materials and mixes selected for use, along with the limitations applied on placement dimensions. The minimal cracking resulting from these causes is not sufficient to result in deterioration of the concrete.

5.2.1.3.1.1 Concrete – OTSG/RVCH Construction Opening

Concrete for the OTSG/RVCH construction opening is a specially designed mix to achieve high early strength (5500 psi @ 3 days, and 6140 psi @ 7 days) with no shrinkage and low creep characteristics. The cement for the concrete mix is Type III Portland Cement manufactured by “Holcim” at the Artesia, Mississippi Plant. An expansive additive “Komponents” by CTS Cement Manufacturing, conforming to ASTM C845 Type K was added to reduce shrinkage. Fine aggregate in accordance with ASTM C33 was furnished by Foster Dixiana Corporation, Columbia, S.C. Coarse aggregate in accordance with ASTM C33 meeting the gradation requirements of Number 67 stone was furnished by Vulcan Materials, Maryville, Tenn. Water for the mix was from the ANO potable water supply. Air entraining admixture, Master Builders MBAE 90, is in accordance with ASTM C260. Water reducing admixture, Master Builders Glenium 3030 NS is in accordance with ASTM C494. Concrete Mix Design, Testing, Production, and Placing, were in accordance with Specification ANO-C-513.

During placement of concrete for repair of the reactor building construction opening following the replacement of the OSTGs/RVCH, samples were taken in accordance with ASTM C172. Samples were taken every 30 cubic yards of placement. These samples were used to perform slump tests in accordance with ASTM C143, temperature tests in accordance with ASTM C1064, air content tests in accordance with ASTM C231, unit weight tests in accordance with ASTM C138, and for making cylinders for compressive strength tests in accordance with ASTM C31 and ASTM C172.

Placement of concrete for repair of the reactor building construction opening was done in accordance with ACI 301 and ACI 304R. Concrete was consolidated by mechanical, internal type vibrators in accordance with ACI 309R. Form pressure was measured using load cells during concrete placement to assure design limits were not exceeded. Placement of concrete in hot or cold weather was performed in accordance with ACI 305R or ACI 306R, respectively. Concrete placement inspection activities were performed in accordance with SGT Quality Assurance Program and Specification ANO-C-513.

5.2.1.3.2 Reinforcing Steel

All reinforcing steel, except the spiral reinforcing steel at the trumpet of the dome tendon anchorages, is deformed billet steel conforming to ASTM Designation A-615. Spiral reinforcing steel conforms to ASTM Designation A-82. Grade 60 is used throughout the reactor building structure.

Mill test results were obtained from the reinforcing steel supplier for each heat of steel to show proof that the reinforcing steel had the specified composition, strength, and ductility. Splices in reinforcing for bar sizes No. 11 and smaller were lapped in accordance with ACI 318-63. For bars larger than No. 11, Cadweld splices were used.

In addition, user tests were conducted through random sampling. Full size rebars were tested. Therefore, the method of testing complied with Safety Guide 15. However, the frequency of sampling was less than required by Safety Guide 15. Since user tests showed good correlation to mill test results, the adequacy of rebar strength was sufficiently confirmed. Therefore, the intent of Safety Guide 15 has been met.

Radial reinforcing steel is provided throughout the reactor building dome. The following are the criteria used in the design of the radial reinforcing steel:

A. The ties shall enclose both the top reinforcing steel and the lower layer of prestress tendons.

B. The area of ties shall be sufficient to resist at .5fy the prestress forces caused by the following:

   1. The radial components of all prestress forces multiplied by the distance from the outside surface to the centroid of the tendon system and divided by the total shell thickness.

   2. The radial components of the prestress forces in the innermost layer of tendons.

C. The maximum tie spacing shall not exceed the thickness of the shell. Based on the above criteria, #6 ties at nine inches were provided along all top layer of tendons.

   The approximate spacing of the top layer of tendons is three feet. The thickness of the dome concrete is three feet, three inches.

   For the typical hoop tendons in the reactor building wall, the theoretical curvature, based on the centerline of the hoop tendons, is 60 feet, 10 inches. The predicted maximum principal tensile stress in concrete generated by the typical tendon at $0.7f'_s$ (where f'_s is the ultimate strength of the tendon) is approximately 35 psi. This is considerably lower than the allowable concrete tension of $1.0f'_c$ or 74 psi specified in Section 5.2.1.4.3. Therefore, no radial ties were provided in the reactor building wall to resist the tensile stresses in concrete caused by the typical hoop tendons.

For prestressed cylindrical shells with vertical axis, the "Criteria for Reinforced Concrete Nuclear Power Containment Structures, "ACI 349, Section 2.5.6.5 requires that radial ties be provided if the tensile stress in the net area of concrete exceeds $2\sqrt{f'_c}$ or 148 psi.

The predicted average tensile stress in concrete in the reactor building wall is 35 psi. The stress concentration factor at the tendon ducts is predicted to be approximately four times the average tensile stress of concrete or approximately 140 psi. In addition, the radial stress is predicted to reduce greatly a few inches away from the face of the tendon duct. Based on the above, the requirements of ACI-349 are satisfied.

5.2.1.3.2.1 Reinforcing Steel – OTSG/RVCH Construction Opening

New reinforcing bars used in repair of the OTSG/RVCH construction opening following replacement of the OTSGs/RVCH conform to ASTM A615 Grade 60. Mechanical property tests for yield strength, tensile strength, and percentage elongation as well as bend tests required by ASTM A615 were performed per ASTM A370. Placement of rebar was in accordance with ACI 301. For replacement of rebar in the OTSG/RVCH construction opening either lapped splices, mechanical splices, or limited fusion welding of rebar was used. Mechanical splices, either “Cadweld” or “Swaged” were used to splice replaced rebar to existing rebar projection. Fusion welding was only used where physical limitations prohibited mechanical splices. “Cadweld” mechanical splice testing per Specification ANO-C-514 was in accordance with ANSI N45.2.5. “Swaged” mechanical splice testing per Specification ANO-C-517, was in accordance with ASME, Section III, Division 2, Section CC-4333.

5.2.1.3.3 Steel Prestressing Tendons

The tendon is composed of 186 stress relieved wires of ¼-inch diameter with an ultimate strength of 240,000 psi furnished in accordance with ASTM A-421.

The basic performance requirements for the end anchors of the tendons are stated qualitatively by the Seismic Committee of the Prestressed Concrete Institute and published in their Journal of June 1966 as follows:

> "All anchors of unbonded tendons should develop at least 100 percent of the guaranteed ultimate strength of the tendon. The anchorage gripping should function in such a way that no harmful notching would occur on the tendon. Any such anchorage system used in earthquake areas must be capable of maintaining the prestressing force under sustained and fluctuating load and the effect of shock. Anchors should also possess adequate reserve strength to withstand any overstress to which they may be subjected during the most severe probable earthquake. Particular care should be directed to accurate positioning and alignment of end anchors."

It is a performance requirement that the end anchors used develop 100 percent of the minimum ultimate strength of the prestressing steel as defined by the appropriate ASTM standards. The test results in Section 5.2.1.3.3.1 give evidence that those end anchors can meet the performance requirement.

Another performance requirement is that the end anchor be capable of maintaining integrity for 500 cycles of loads corresponding to an average axial stress variation between 0.7 and $0.75f'_s$ at a repetition rate of one cycle in 0.1 second. This requirement, of course, sets minimum acceptable limits on fatigue effects due to notching by the end anchor and tendon performance in response to earthquake loads. The number of cycles is set by increasing to 500 from the 100 predicted. The stress variation is increased from a conservatively predicted 0.6 to $0.64f'_s$ to the 0.7 to $0.75f'_s$. Further, the number of cycles caused by earthquake loads is predicted as only 30 of the total of 100 and by using all those strong ground motions which exceed one half of the peak ground motion for the earthquake. The predicted stress variation due to earthquake motion alone is predicted as being 10 percent of the total of $0.04f'_s$ stress variation predicted. The $0.04f'_s$ predicted stress variation in turn results from combinations of earthquake, wind, and accident loadings. Analyses made during the investigation included consideration of tendon excitation parallel and perpendicular to the tendon axis.

Another tendon performance requirement is that suitable atmospheric corrosion protection be maintained for all exposure environments from the point of tendon manufacture to and including the installed locations. The corrosion protection provides assurance that the tendon integrity is not impaired by atmospheric corrosion.

Prior to final greasing, the tendon is protected with factory applied micro-crystalline wax (No. OXID 500) and a hand-applied field coat of tendon sheathing filler (Visconcorust 2090P) incorporating the following characteristics:

A. High dielectric strength

B. Corrosion inhibitors

C. Wetting agents

During the interim period, subsequent to placement and before final greasing, the characteristics of the corrosion preventatives will inhibit the effects of welding currents. After tendon insertion in the sheathing and tensioning, the interior of the spiral wrapped, semi-rigid, galvanized, corrugated sheathing is pumped full of a corrosion protection material which consists of a modified, thixotropic, refined petroleum oil base product. The tendons and end anchors are surrounded by the corrosion protection material which in turn are encapsulated by the sheathing and gasketed end caps which are sealed against the bearing plate. On filling the sheaths and end caps, the filler displaces air and water vapor before solidifying to a soft gel which surrounds the end anchors and the bottom heads. Once the filler has cooled, contracted, and jelled, the vertical tendons are "topped-off" by injecting filler material through the upper grease cap fittings. Any contraction resulting after the initial installation leaves a residual film 1/4- to 3/8-inch thick on the post tensioning wires and trumpet within this void. This film is adequate for corrosion protection during those short periods of void formation.

The sheathing and couplers are ferrous metals conforming to ASTM A-366-66T, 24 gauge cold rolled carbon steel. The 5¼-inch outside diameter sheathing is a galvanized, spiral wrapped, semi-rigid, corrugated tubing which is made continuous by attaching a coupler to the ends of the sheathing. The coupler is a galvanized, semi-rigid, corrugated tubing of approximately one foot long. The internal diameter of the coupler is equal in size to the outside diameter of the sheathing. The corrugations of the sheathing and coupler act as threads. The two are attached by turning the coupler while the installed sheathing is prevented from rotating. Leak tightness during concrete pour and greasing operations is maintained by wrapping each joint, formed by the intersection of the coupler and the sheaths, by plastic tape. Drains are provided at all low points of the sheathing to prevent accumulation of water from condensation in ducts.

Replacement tendon sheaths within the OTSG/RVCH replacement opening are 5“ diameter ASTM A53 Type S Grade B pipe.

Cathodic protection is not provided for corrosion protection of the steel containment liner, reinforcing bars, or tendon sheathing. Steel within such monolithic concrete structures is inherently protected by the high pH afforded by the concrete. A zinc electrode monitoring system is provided within the containment base slab to allow periodic observation of corrosion currents in the remote possibility that physical changes occur in the foundation.

A cathodic protection system would have no effect on post-tensioning tendons since they are surrounded by a high dielectric casing filler and shielded by metallic sheathing and reinforcing steel. In fact, cathodic protection is not advisable for such high strength steel tendons due to their susceptibility to hydrogen embrittlement.

Testing of the permanent corrosion protection material indicates that there are no significant amounts of chlorides, sulfides, or nitrates. However, to verify that none are present in the material used, samples are taken from each shipment with at least one sample per factory batch. The samples are analyzed as follows:

Water soluble chlorides (Cl) are determined by ASTM Method D512-67 or equivalent with a limit of accuracy of 0.5 ppm.

Water soluble nitrates (NO_3) are determined by the water and sewage analysis procedure of the Hach Chemical Company or equivalent. Water soluble sulfides (S) are determined by ASTM Method D-1255 or equivalent with a limit of accuracy of 1 ppm.

No significant traces of the impurities are allowed. The chemical composition of the filler material, being about 98 percent microcystalline wax, indicates that it possesses the normal stability of linear hydrocarbons for the site temperature ranges.

5.2.1.3.3.1 Information for the Reactor Building Post-Tensioning Hardware

5.2.1.3.3.1.1 Components

The major components of the post-tensioning system used for the reactor building structure are listed below:

A. Tendon

B. Bearing Plate

C. Fixed end anchor plate

D. Shims

E. Stressing washer

The geometries and the properties of the components are shown in Figure 5-27.

5.2.1.3.3.1.1.1 Tendon

A. Description

The tendon is composed of 186 stress relieved, high strength wires of 1/4-inch diameter furnished in accordance with ASTM A-421. Button heads are provided at the ends of each wire to transfer the wire tensile force to the anchor plates.

B. Efficiency of the Curved Tendon vs. Straight Tendon

Efficiency may be defined as the ratio between the ultimate strength of the entire tendon and the ultimate strength derived by multiplying the mean ultimate strength of a wire by the total number of wires. The mean ultimate strength is determined from a large number of samples.

The efficiencies of the straight and curved axes large tendons were evaluated in connection with the construction of the Dungeness B nuclear power station. A slight reduction of 99.7 percent in the case of straight tendons to approximately 96.5 percent for the curved tendons resulted.

C. Fatigue Strength of the Tendon

When a body is alternatively stressed between two limits $\sigma\ell$ and $\sigma\mu$ and if the upper limit σ_μ is so selected that rupture takes place after two million stress cycles, the difference is defined as fatigue strength. The fatigue strength of the button head ranges from 15,000 to 27,000 psi.

The adequate fatigue strength of the tendon and its anchorages (based on the performance criteria in Section 5.2.1.3.3) were demonstrated when the dynamic test results indicated no loss of stress based on 500 cycles of rapid loading (0.10 second per cycle) from a stress level of $0.70f'_s$ to stress level of $0.75f'_s$ and return to $0.70f'_s$.

5.2.1.3.3.1.1.2 Bearing Plate

The bearing plate is a high strength steel plate located at the end of each tendon to transfer the prestress force from the fixed end anchor plate (at nonstressing end) and from the shim (at stressing end) to the concrete.

To insure good concrete bearing of the tendon bearing plate, the following procedures were followed:

A. Bearing plates were cleaned of dirt and grease.

B. Bearing plates were accurately placed into position and securely anchored.

C. Placement of concrete was in accordance with Section 5.2.3.1 and applicable construction codes.

D. Work quality control was in accordance with Section 5.2.3.2 regarding the quality assurance program.

E. Quality assurance documentations on concrete placement check lists and field inspection reports were prepared for each pour to insure high standards of concrete placement.

F. Visual inspections were conducted after removal of forms.

5.2.1.3.3.1.1.3 Fixed End Anchor Plate

The fixed end anchor plate is a high strength steel plate located at the non-stressing end of the tendon and transfers the tendon tensile force to the bearing plate by means of the button head.

5.2.1.3.3.1.1.4 Shims

Shims are semi-circular steel rings inserted between the bearing plate and the stressing washer to obtain the desired prestress force.

5.2.1.3.3.1.1.5 Stressing Washer

The stressing washer is a round piece of steel plate threaded both internally to provide attachment for stressing the tendons. It also provides the transfer of the tendon tensile force to the shim.

The stepped holes in the stressing washers should not damage the wires during erection based on the following:

A. The stressing washers used in the post-tensioning system of other reactor buildings also contain stepped holes and have not damaged the tendon wires.

B. During the post-tensioning hardware tests, the tendons utilizing this stress washer were able to sustain forces greater than the guaranteed ultimate tensile strength of the tendon.

C. All stressed tendons in the reactor building structure have been stressed to $.80f'_s$ (.8x guaranteed ultimate strength of the tendon) without wire failure.

5.2.1.3.3.1.2 Vendor Testing of the Post Tensioning Hardware

To assure the adequate performances of the tendons and their anchorages, vendor testing of each component was performed. The test arrangement for all tests is shown in Figure 5-23.

5.2.1.3.3.1.2.1 Tendon Test

In the tendon tensile test, the tendon was found to develop the minimum guaranteed ultimate strength of 2,192 kips with more than three percent elongation in a test stand length of 10 feet. The above test demonstrates the satisfactory ductility and strength of the tendon.

5.2.1.3.3.1.2.2 End Anchorage Tests

In the individual testing of the bearing plates, fixed end anchor plate, and stressing washer, the ability of the end anchorages to develop the ultimate strength of the tendon and to distribute the ultimate force (2,192 kips) from the button heads of the prestressing wires to the concrete without concrete or end anchorage failure was satisfactorily demonstrated. The tests incorporated a tendon load eccentricity of 1/4-inch representing possible erection inaccuracy. The test results showed no adverse effect on the ultimate strength of the end anchorages. An analysis was made after the fixed end anchor and bearing plate tests by using the finite element computer program. This program considers the bilinear properties of each material, the reinforcing steel and the cracking of concrete. The analytical model of the finite element mesh is shown in Figures 5-24 and 5-25. The material properties used for analysis are shown in Figure 5-26. The predicted deflection contours of each component was compared to the experimental deflection contours. The results were in close agreement.

5.2.1.3.3.1.3 Effect of Twisted Tendon on End Anchorage

To equalize the wire lengths in a hoop or dome tendon, the tendon is twisted at a rate of approximately one revolution in 40 feet of tendon length. It was shown by analysis that friction alone between the components of the tendon anchor assembly can resist the torsional moment generated by the twisting of tendons. Also the effect of twisting on the ultimate strength of the anchor and the prestressing wires is predicted to be negligible.

5.2.1.3.3.2 Steel Prestressing Tendons – OTSG/RVCH Construction Opening

As part of the creation of the OTSG/RVCH construction opening for replacement of the OTSGs/RVCH, the tendons in the area of the opening were removed. 16 horizontal tendons (21H21 through 21H28 and 31H21 through 31H28) and 6 vertical tendons (V75 through V80) were removed. The replacement horizontal tendons were 186 wire (¼" diameter wires) ASTM A421, Type BA, with relaxation losses not exceeding those of the original tendons. All replacement tendons and existing tendons that were detensioned for the OTSG/RVCH Replacement Project were stressed sufficiently to ensure that the 60-year minimum effective prestress levels are met. The new replacement tendons were protected from corrosion after fabrication with Visconorust 1601 Amber. New replacement tendons and tendons that were detensioned and retensioned were regreased with Visconorust 2090P-4 tendon sheath filler. The replacement tendon sheaths for the OTSG/RVCH construction opening are 5 inch diameter pipe. The connection to existing sheaths is sealed with Belzona 2211MP Hi-Build Elastomer to prevent concrete intrusion into the sheath.

5.2.1.3.4 Steel Liner Plate

The reactor building is lined with welded steel plate conforming to ASTM A-516, Grade 60, to insure low leakage.

The design, construction, inspection, and testing of the liner plate, which acts as a leak tight membrane and is not a pressure vessel, is not covered by any recognized code or specification.

The A-516 material is selected on the basis that it has sufficient strength as well as ductility to resist the expected stresses from design criteria loading and at the same time preserve the required leak tightness of the reactor building. In specific areas, liner plate degradation, such as corrosion, pitting, or gouges, has thinned local areas below the nominal plate thickness of ¼-inch. These areas may be evaluated for use-as-is if an evaluation documents that the liner plate retains capacity to perform its design function.

The structural integrity of the reactor building is maintained by the prestressed, post-tensioned concrete without dependence on the liner for structural stability. The liner plate functions as a leak tight membrane and has been designed to withstand forces and strains imposed by the reactor building. Since the applied force to the liner plate membrane from shrinkage and creep of the concrete is compressive and no significant applied tensile forces are expected from internal pressure loading, there is no need to apply special Nil Ductility Transition Temperature (NDTT) criteria to the liner plate material. With the prestress level, this is true even for the condition created by 40 °F minimum Reactor Building Spray impinging on the liner plate. On the other hand, all material for reactor building parts which must resist applied internal pressure, such as penetrations, is impact tested in accordance with the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, of the ASME Code.

As the liner plate and the concrete adjacent to the liner plate are predicted to be in compression under prestress, concrete creep and shrinkage, and thermal loads, the liner face which is in contact with the concrete is not predicted to be exposed to the outside environment. Based on the above reasoning, no provisions are made to prevent liner corrosion. However, in the unlikely event that localized corrosion is suspected on the liner face, ultrasonic tests to detect corrosion may be performed from the inside of the reactor building.

At an elevation of 493 feet, 9 3/4 inches, crane brackets, which penetrate the thickened liner plate, are provided around the periphery of the reactor building to support the polar crane. The crane bracket is attached to the thickened liner plate by full penetration welds around the entire periphery of the bracket section. The leak tightness is checked by performing magnetic particle testing. In addition, all welds were vacuum box tested.

5.2.1.3.4.1 Steel Liner Plate – OTSG/RVCH Construction Opening

During replacement of the OTSG/RVCH a construction opening was created. As part of the process for creating this construction opening, an approximately 17'-8" wide x 21'-6 3/8" high section of the liner plate was removed. Following completion of the OTSG/RVCH replacement this section of the plate was reinstalled. Welding, visual examinations, magnetic particle and/or radiographic weld examination, and vacuum box leak testing of the new liner plate welds were performed in accordance with ANO Specification 6600-C-030 and SGT Quality Assurance Program Procedures (which are based on ANO's commitments to the requirements of ASME Section VIII, Division 1, and Subsection IWE of ASME Section XI) to ensure weld acceptability and leak-tightness. Dye penetrant inspection was used to confirm the complete removal of all defects from areas which were prepared for repair welding. Leak testing in accordance with 10 CFR 50 Appendix J was also performed. Additional stiffeners were added to strengthen the replaced section. All liner plate welding was in accordance with ASME Section VIII and Section XI.

5.2.1.3.5 Reactor Building Interior Coatings

Surfaces of the liner plate in contact with concrete are not coated.

All exposed carbon steel surfaces of the liner plate, structural steel, grating, toe plates, stair stringers, handrails, ladders, and cages were sandblasted and coated with an inorganic zinc primer. Damaged areas are repaired such that they are equivalent to the original condition. The liner plate above an elevation of 424 feet, 6 inches was finish coated with an inorganic topcoat. The liner plate below an elevation of 424 feet, 6 inches and all aforementioned carbon steel surfaces were finish coated with an organic topcoat.

Floor slabs and certain other concrete surfaces were finished with an epoxy surfacer and an epoxy topcoat to minimize possible absorption of contaminated fluids and to facilitate decontamination.

Coatings were selected from suitable compounds which are resistant to radiation and successfully withstand tests in an environment simulating post-DBA conditions, including exposure to the proposed spray. Current references in the selection of coating include the ANSI standards "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," N101.2-1972, N101.4-1972, N45.2-1971, N5.12-1974 and "Applicability of Conventional Protective Coatings to Reactor Containment Buildings," IN-1169, June 1968 by B. J. Newby.

Criteria for selection of coatings applied to surfaces within the containment are based on their necessary performance during both a LOCA and the normal environment. Coatings are selected to withstand radiation, boric acid sprays, steam, and jet impingement. Repairs of damaged coatings are generally equivalent to the original coating. Reformulated replacement coatings may contain materials different than the previous coatings, may not require a primer, and may not be the same type (e.g., inorganic). Coatings and repairs are governed by Specification ANO-A-2437.

Simulated LOCA tests performed by an independent laboratory, The Franklin Research Institute Laboratories, Philadelphia, Pennsylvania, using boric acid spray solution with adjusted pH to the required design conditions, showed these coatings to pass on the basis that no delamination, peeling, or flaking occurred. The radiation resistance of the coating exceeds 1 x 10^7 rads (^{60}Co source) without breakdown and the coating systems have good properties for decontamination. Resistance to boric acid spray washdown and steam environment is good.

Portions of coating could be removed by steam jet impingement per analysis in the sump debris generation calculation.

5.2.1.3.5.1 <u>Liner Plate Interior Coating – OTSG/RVCH Construction Opening</u>

To accommodate cutting and welding activities on the liner plate, portions of the existing coating on the interior surface of the liner plate were removed prior to cutting the liner plate section. The affected area of the liner plate was recoated in accordance with Specification ANO-A-2437 prior to resumption of power operations.

5.2.1.4 <u>Reactor Building Design Criteria</u>

Safety and performance of the structure under extraordinary circumstances at various loading stages are the main considerations in establishing the structural design criteria.

The two basic criteria are:

A. The integrity of the liner plate is guaranteed under all loading conditions and

B. The structure has a low strain, essentially elastic response such that its behavior is predictable under all design loadings.

The strength of the reactor building at working stress and overall yielding is compared to various loading combinations to insure safety. The reactor building was examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to insure proper performance. The structure is designed to meet the performance and strength requirements under the following conditions:

A. Prior to prestressing

B. At transfer of prestress

C. Under sustained prestress

D. At design loads

E. At factored loads

The stresses for the design loading conditions in the working stress method are specified under Sections 5.2.1.4.5 and 5.2.1.4.6.

No special design bases are required for the design and checking of the base slab. It acts primarily in bending rather than membrane stress. This condition is covered by the ACI Code 318-63. The loads and stresses in the cylinder and dome are determined as described below.

5.2.1.4.1 Design Method

The design and checking of the reactor building structure have been made in accordance with the applicable sections of the ACI 318-63 Code and the ACI Committee 334 publication, "Concrete Shell Structures and Commentary."

The reactor building shell is analyzed using a finite element computer program for individual loading cases of dead load, live load, prestress, temperature, and pressure. The computer output includes normal stresses, shear stresses, principal stresses, and displacement of each nodal point. The concrete and reinforcement stress due to flexure for loadings other than thermal are calculated by the conventional working stress method in accordance with the ACI 318-63 Code (see Section 5.2.1.5.9.1). The stresses in the concrete and reinforcing steel across any wall section due to the thermal gradient are calculated by considering the equilibrium of forces in the stress tabulations for each element (given by the computer analysis) across the applicable wall section (refer to Section 5.2.1.5.9.1).

Stress plots which show the total stresses from appropriate combinations of loading cases are made and areas of high stress are identified.

In order to consider creep deformation, the modulus of elasticity of concrete under sustained loads such as dead load and prestress is differentiated from the modulus of elasticity of concrete under instantaneous loads such as internal pressure and earthquake loads.

The forces and shears are added over the cross section and the total moment, axial force, and shear are determined. From these values, the straight line elastic stresses are computed and compared to the allowable values. The ACI design methods and allowable stresses are used for concrete and reinforcing steel except as noted in these criteria.

5.2.1.4.2 Loads Prior to Prestressing

Under this condition the structure is designed as a conventionally reinforced concrete structure. It is designed for dead load, and live loads (including construction loads). Allowable stresses are according to ACI 318-63 Code.

5.2.1.4.3 Loads at Transfer of Prestress

The reactor building is checked for prestress loads and the stresses compared with those allowed by the ACI 318-63 Code with the following exceptions: ACI 318-63, Section 26 allows concrete stress of $0.60f'_c$ at initial transfer. In order to limit creep deformation, the membrane compression stress is limited to $0.30f'_c$ whereas in combination with flexural compression the maximum allowable stress is limited to $0.60f'_c$ per the ACI 318-63.

For predicted local stress concentrations with nonlinear stress distribution, as predicted by the finite element analysis, $0.75f'_c$ was permitted when local reinforcing is included to distribute and control these localized strains. These high local stresses are present in every structure, but they are seldom identified because of simplifications made in design analysis. These high stresses are allowed because they occur in a very small percentage of the cross section, are confined by material at lower stress, and would have to be considerably greater than the values allowed before significant local plastic yielding results. Nonprestressed reinforcing is added to distribute and control these local strains.

Membrane tension and flexural tension are permitted provided they do not jeopardize the integrity of liner plate. Predicted membrane tension is permitted to occur during the post-tensioning sequence but is limited to $1.0f'_c$. When there is flexural tension, but no membrane tension, the section is designed in accordance with Section 2605 (a) of the ACI Code. The stress in the liner plate due to combined membrane tension and flexural tension, is limited to $0.5f_y$. The effects of the prestressing sequences are considered.

Shear criteria are in accordance with the ACI 318-63 Code, Chapter 26 as modified by the equations shown in the Section 5.2.1.4.6, using a load factor of 1.5 for shear loads.

5.2.1.4.4 Loads Under Sustained Prestress

The conditions for design and the allowable stresses for this case are the same as stated in Section 5.2.1.4.3 except that the allowable tensile stress in nonprestressed reinforcing is limited to $0.5f_y$ and no membrane concrete tensile stresses are permitted. The ACI limits the concrete compression to $0.45f'_c$ for sustained prestress load. Values of $0.30f'_c$ and $0.60f'_c$ which bracket the ACI allowable value are used as described above. However, with these same limits for concrete stress at transfer of prestress, the predicted stresses under sustained load are reduced due to creep.

5.2.1.4.5 At Design Loads

This loading case is the basic "working stress" design. The reactor building is designed for the following specific loading cases:

A $D + F + L + T_O$

B. $D + F + L + P + T_A$

where:

D = Dead Load

L = Appropriate Live Load

F = Appropriate Prestressing Load

P = Pressure Load

T_O = Thermal loads due to operating temperature (See Figure 5-5.)

T_A = Thermal loads based on temperature corresponding to pressure P (see Figure 5-5)

Sufficient prestressing is provided in the cylinder and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking is permitted but is controlled by bonded un-prestressed reinforcing steel.

Under the design loads, the same performance criteria, as specified in "Loads at Transfer of Prestress," are applied with the following exceptions:

A. If the predicted net membrane compression of concrete is below 100 psi, it is neglected and a cracked section is assumed in the computation of flexural, non-prestressed, reinforcing steel. Flexural tensile stresses in nonprestressed reinforcing of $0.5f_y$ are allowed.

B. When the maximum predicted flexural tensile stress of concrete does not exceed $6.0\sqrt{f'_c}$ and the extent of the tension zone is no more that one-third the depth of the section, non-prestressed reinforcing is provided to carry the entire tension force in the tension block. Otherwise, the non-prestressed, reinforcing steel is designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel is $0.5f_y$.

C. The problem of shear and diagonal tension in the prestressed concrete structure is considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is provided to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined membrane tension and membrane shear is considered under Section 5.2.1.4.6.

Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). The present ACI 318-63 provisions of Chapter 26 for shear are considered adequate for the design purpose with proper modifications as discussed under Section 5.2.1.4.6, using a load factor of 1.5 for shear loads.

Crack control in the concrete is accomplished by adhering to the ACI-ASCE Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute and are as follows:

- 0.25 percent reinforcing shall be provided at the tension face for small members
- 0.20 percent for medium size members
- 0.15 percent for large members
- 0.15 percent mild steel reinforcing will be provided on the exterior faces of the wall and dome for proper crack control.

The liner plate is attached on the inside faces of the wall and dome. Since there is no predicted tensile stress on the inside faces, mild steel reinforcing is not provided.

5.2.1.4.6 Factored Load Combinations

The reactor building is checked for the factored load combinations given below.

The load factors are ratios by which loads are multiplied for design purposes to assure that the load/deformation behavior of the structure is essentially elastic, low strain behavior. The load factor approach is used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permitted the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also placed minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or tornado loads.

The final design of the reactor building satisfies the following factored load combinations.

A. $Y = 1/\phi\ (1.05D + 1.5P + 1.0\ T_A + 1.0F)$

B. $Y = 1/\phi\ (1.05D + 1.25P + 1.0\ T_A + 1.25H + 1.25E + 1.0F)$

C. $Y = I/\phi\ (1.05D + 1.25H + 1.0R + 1.0F + 1.25E + 1.0T_O)$

D. $Y = 1/\phi\ (1.05D + 1.0F + 1.25H + 1.25W + 1.0\ T_O)$

E. $Y = 1/\phi\ (1.0D + 1.0P + 1.0\ T_A + 1.0H + I.0E' + 1.0F)$

F. $Y = 1/\phi\ (1.0D + 1.0H + 1.0R + 1.0E' + 1.0F + 1.0\ T_O)$

where:

Y = Required yield strength of the structure to resist factored loads

ϕ = Capacity reduction factor (defined later in this section)

D = Dead loads of structures and equipment plus any other permanent loading contributing stress, such as hydrostatic or soil. In addition, a portion of the live load is added when it includes items such as piping and cable trays suspended from floors. An allowance is made for future additional permanent loads.

P = Design Basis Accident (DBA) pressure load

F = Effective prestress loads

R = Force or pressure on structure due to rupture of any one pipe

H = Force on structure due to thermal expansion of pipes design conditions

T_O = Thermal loads due to the temperature gradient through wall during operating conditions

T_A = Thermal load due to the temperature gradient through the wall and expansion of the liner. It is based on a temperature corresponding to the factored design accident pressure.

E = Design earthquake load

E = Maximum earthquake load

W = Tornado load

Equation A assures that the reactor building has the capacity to withstand pressure loadings at least 50 percent greater than those calculated for the postulated LOCA alone.

Equation B assures that the reactor building has the capacity to withstand loadings at least 25 percent greater than those calculated for the postulated DBA coincident with a design earthquake.

Equation C assures that the reactor building has the capacity to withstand earthquake loadings 25 percent greater than those calculated for the design earthquake coincident with rupture of any attached piping due to that earthquake.

Equation D assures that the reactor building has the capacity to withstand tornado loading 25 percent greater than the design tornado loading.

Equations E and F assure that the reactor building has the capacity to withstand either the postulated LOCA or the rupture of any attached piping coincident with the maximum earthquake.

The load combinations, considering load factors given above, will be less than the yield strength of the structure. The yield strength of the structure is defined as the upper limit of elastic behavior of the effective load carrying structural materials. For steels (both prestress and non-prestress) this limit is taken to be the guaranteed minimum yield given in the appropriate ASTM specification. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 and the 90-day ultimate compressive strength for concrete in flexure (f'_c). The ultimate strength assumptions of the ACI Code for concrete beams in flexure is not allowed; that is, the concrete stress is not allowed to go beyond yield and redistribute at a strain of three to four times that which causes yielding.

The maximum strain due to secondary moments, membrane loads, and local loads exclusive of thermal loads is limited to that corresponding to the ultimate stress divided by the modulus of elasticity (f'_c/E_c) and a straight line distribution from there to the neutral axis assumed. For the above loads combined with thermal loads, the peak strain is limited to 0.003 inch per inch. For concrete membrane compression, the yield strength is assumed to be $0.85f'_c$ to allow for local irregularities, in accordance with the ACI approach. The reinforcing steel forming part of the load carrying system will be allowed to go to, but not exceed, yield as is allowed for ACI ultimate strength design.

The predicted strain in the liner plate for all loading combinations does not exceed 0.005 inch per inch.

Principal concrete tension due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3\sqrt{f'_c}$.

As an example, the effect of the earthquake shear force will contribute to the membrane tension in the reactor building walls. The maximum predicted principal membrane tension in concrete where shear reinforcing is not provided, from the governing factored loading combination including earthquake forces, is 187 psi. This is less than the allowable concrete membrane tensile stress of 222 psi as specified above. The maximum predicted principal membrane stress in the concrete at the same location and using the same loading combination but with load factors equaling 1.0 is 28 psi in compression.

Principal concrete tension due to combined membrane tension, membrane shear, and flexural tension due to bending moments or thermal gradients is limited to $\sqrt{f'_c}$. When the principal concrete tension exceeds the limit of $6\sqrt{f'_c}$ bonded reinforcing steel is provided in the following manner.

A. Thermal Flexural Tension - Bonded reinforcing steel is provided in accordance with the methods of ACI 505. The minimum area of steel provided is 0.15 percent in each direction.

B. Bending Moment Tension - Sufficient bonded reinforcing steel is provided to resist the moment by the conventional working stress method. The bending moment tension is additive to the thermal tension which is determined in accordance with the methods of ACI 505. Shear stress limits and shear reinforcing for radial shear are in accordance with Chapter 26 of ACI 318-63 with the following exceptions

Formula 26-12 of the Code is replaced by:

$$V_{ci} = Kb'd\sqrt{f'_c} + M_{cr}\frac{V}{M'} + V_i$$

where:

$$K = \left[1.75 - \frac{0.036}{np'} + 4.0np'\right]$$ but not less than 0.6 for $p' \geq 0.003$. For $p' < 0.003$, the value of K is zero.

M_{cr} = $\frac{I}{Y}(6\sqrt{f'_c} + f_{pe} + f_n + f_i)$

f_{pe} = Compressive stress in concrete due to prestress applied normal to the cross section after all losses (including the stress due to any secondary moment) at the extreme fibre of a section at which tension stresses are caused by live loads.

f_n = Stress due to axial applied loads (f_n was negative for tension stress and positive for compression stress).

f_i = Stress due to initial loads, at the extreme fibre section at which tension stresses are caused by applied loads (including the stress due to any secondary moment). f_i is negative for tension stress and positive for compression stress.

n = $\frac{505}{\sqrt{f'_c}}$

p' = $\frac{As'}{bd}$

V = Shear at the section under consideration due to the applied loads.

M' = Moment at a distance d/2 from the section under consideration measured in the direction of decreasing moment, due to applied loads.

V_i = Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads).

Lower limit placed by ACI 318-63 on V_{ci} as 1.7 b'd $\sqrt{f'_c}$ is not applied.

Formula 26-13 of the Code is replaced by:

$$V_{cw} = 3.5b'd\sqrt{f'_c}\left[\sqrt{\frac{(1+f_{pe}+f_n)}{3.5\sqrt{f'_c}}}\right]$$

The term f_n is as defined above. All other notations are in accordance with Chapter 26, ACI 318-63.

A. This formula is based on tests and work done by Dr. A. H. Mattock of the University of Washington.

B. This formula is based on the commentary for the Proposal Redraft of Section 2610-ACI-318 by Dr. A. H. Mattock, dated December 1962.

When the above mentioned equations indicated that the allowable shear force in concrete is zero, radial shear ties are provided to resist all the calculated shear.

The load factors and load combinations in the design criteria represent the consensus of the individual judgments of a group of Bechtel Engineers and consultants who are experienced in both structural and nuclear plant design. Their judgment has been influenced by current and past practice, by the degree of conservativeness inherent in the basic loads, and particularly by the probabilities of coincident occurrences in the case of accident, tornado, and seismic loads.

The following discussions explain the justification for the individual factors, particularly as they applied to the reactor building.

A. Dead Load

Dead load in a large structure such as this is easily identified and its effect can be accurately predicted at each point in the vessel. For dead load in combination with accident and design earthquake or wind loads, a load factor representing a tolerance of five percent is chosen to account for dead load inaccuracies. ACI 318-63 allows a tolerance of plus 25 percent and minus 10 percent, but the code is written to cover a variety of conditions where weights and configurations of materials in and on the structure may not be clearly defined and are subject to change during the life of the structure.

B. Live Load

The live load that would be present (along with accident, seismic, flood, and wind loads) would produce a very small portion of the stress at any point. Also, it is extremely unlikely that the full live load would be present over a large area at the time of an unusual occurrence. For these reasons, a low load factor is felt to be justified and live load is considered together with dead load at load factor 1.05.

C. Seismic

The design earthquake is considered to be the strongest possible earthquake which could occur during the life of the plant. In addition to the design earthquake, a maximum earthquake which defines the maximum credible earthquake that could occur at the site is considered in the design. Seismic Class 1 structures are designed so that no loss of function would result from the maximum earthquake. The probability of an earthquake causing the maximum credible accident is very small. For this reason, the two events, seismic and accident, are considered together but at much lower load factors than those applied to the events separately. The earthquake load factors of 1.25 for design earthquake and 1.0 for maximum earthquake are conservative for both earthquakes in combination with the factored LOCA.

D. Wind and Tornado

Loads are determined from the design tornado wind speed. With the reactor building structure designed for this extreme wind, it is inconceivable that the wind would cause an accident. Therefore, wind load is not considered with accident loads. However, a load factor of 1.25 is applied to the tornado load to provide assurance of the structure performing satisfactorily.

E. Accident

The design pressure and temperature are based on the Reactor Building design basis accident analysis described in Section 14.2.2.5.5. Accepted practice has been to use a load factor of 1.5 on the design pressure. Refer to T. C. Waters and N. T. Barrett, "Prestressed Concrete Pressure Vessels for Nuclear Reactors," J. Brit. Nucl. Sec. 2, 1963. This factor is considered reasonable and is adopted in factored load equations excluding earthquake. The probability of a maximum accident occurring simultaneously with a design earthquake is very small; therefore, a reduced load factor of 1.25 for both accident and design earthquake is used for the combination of the two events.

The design accident temperature is defined as that temperature corresponding to the design accident pressure. At 1.5 P, the temperature inside the reactor building will be somewhat higher than the temperature at P. It would be unrealistic to apply a corresponding temperature factor of 1.5 since this could only occur with a pressure much greater than a pressure of 1.5 P.

<u>Yield Capacity Reduction Factors</u>

The yield capacity of all load carrying structural elements is reduced by a yield capacity reduction factor, Ø. This factor provides for "the possibility that small adverse variation in material strengths, workmanship, dimensions, control, and degree of supervision, while individually within required tolerances and the limits of good practice, occasionally may combine to result in undercapacity" (refer to footnote on page 66 of ACI 318-63 Code).

<u>Justification for Yield Reduction Factors (Ø-Factors) Used In Determining Yield Strength of the Reactor Building Structure</u>

The Ø factors are provided to allow for variations in materials and workmanship.

In the ACI Code 318-63, Ø varies with the type of stress or member considered; that is, with flexure, bond or shear stress, or compression.

The Ø factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength, assuming materials are as strong as specified, sizes are as shown on the drawings, the workmanship is excellent, and the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables except as noted by using these Ø factors:

Ø = 0.90 for concrete in flexure.

Ø = 0.85 for diagonal tension, bond, and anchorage in concrete.

Ø = 0.90 for reinforcing steel in direct tension.

Ø = 0.90 for welded or mechanical splices of reinforcing steel.

Ø = 0.95 for prestressed tendons in direct tension.

Ø is larger for flexure because the variability of steel strain is less than that of concrete and the concrete as designed in compression has a fail-safe mode of behavior; that is, material understrength without failure. Also, it is possible that the analysis might not combine the worst combination of axial load and moment. Since the member is critical in the gross collapse of the structure, a lower value is used.

Conventional ultimate strength design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI uses Ø = 0.90. The same reasoning is applied in assigning a value of Ø = 0.90 to reinforcing steel in tension, which includes axial tension. However, the code recognizes the possibility of reduced bond of bars at the laps by specifying a Ø of 0.85. Mechanical and welded splices develop at least 135 percent of the yield strength of the reinforcing steel. Therefore, Ø = 0.90 is recommended for the mechanical type of splice and Ø = 0.85 is recommended for the lap type.

The only significant new value introduced is Ø = 0.95 for prestressed tendons in direct tension. A higher Ø value than for conventional reinforcing was allowed because, during installation, the tendons are each jacked to about 90 percent of their yield strength so, in effect, each tendon has been proof tested and (cold drawing and stress relieving) assures a higher quality product than for conventional reinforcing steel.

Prestress Losses

In accordance with ACI Code 318-63, the design provides for prestress losses caused by the following effects:

A. Seating anchorage

B. Elastic shortening of concrete

C. Creep of concrete

D. Shrinkage of concrete

E. Relaxation of prestressing steel stress

F. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses have been predicted with a reasonable degree of accuracy.

The environment of the prestress system and concrete is not appreciably different in this case from that found in numerous bridge and building applications. Considerable research was done to evaluate the above items and is available to designers in assigning the allowances.

5.2.1.4.7 Reactor Building Liner Plate Criteria

5.2.1.4.7.1 General

The reactor building liner is designed to serve only as the primary leakage barrier. The design analysis considers the composite effects of the liner and the concrete structure and included the transient effects on the liner due to normal operation, accident (as a result of rupture in the primary reactor system), and extreme environmental conditions.

The liner is designed to follow the major strain pattern of the surrounding concrete.

In the design of plate attachments in which loads are applied normal to the plane of the plate, consideration is given to the possible effects of possible laminations in the plate and also to the reduced strength when loading through the thickness.

The stability of the liner plate is assured by anchoring of the liner plate to the concrete structure.

The maximum compressive strains are caused by accident thermal loading, prestress when present, shrinkage and creep, earthquake, etc.

The liner plate is anchored at all major discontinuities to eliminate excessive strains. The anchors have sufficient strength and ductility to resist the applied load.

In isolated areas, the liner plate may have initial inward curvature due to construction. The anchors are designed to resist the forces and moments induced when a section of the liner plate between anchors has initial inward curvature. Under both operating and accident conditions, inward deformation of the liner between anchors may occur. Under these conditions, the anchorages are designed to have sufficient ductility to undergo displacement to relieve the load without rupture.

At all penetrations, the liner plate is thickened to reduce stress concentrations in accordance with the ASME Code, Section III, for Class B vessels. The thickened portion of the liner plate is then anchored to the concrete by use of anchors around the penetrations. When the nozzle is anchored, it constitutes the thickened liner place anchor. The nozzle and all welds associated with the penetrations are designed to resist all the specified applied loads.

All components of the liner which must resist the full design pressure, such as penetrations, are selected to meet the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, of the ASME Code. ASTM A-516 Grade 60 made to ASTM A-300 is typical of a steel which meets these requirements and is used as a plate material for penetrations. This material has excellent weldability characteristics and as much ductility as is obtainable in any commercially available pressure vessel quality steel.

In accordance with ASME Code Case 1247, allowable stresses for A-516 Grade 60 are the same as those permitted for A-201 Grade B.

5.2.1.4.7.2 Liner Design Criteria Under Accident Conditions

The design criteria which are applied to the reactor building liner to meet the specified leak rate under accident conditions are as follows:

A. The liner shall be protected against damage by missiles generated from a LOCA.

B. The liner plate strains shall be limited to those values that have been shown by past experience to result in leak tight pressure vessels.

C. The liner plate shall be prevented from developing distortions sufficient to impair leak tightness.

5.2.1.4.7.3 Loads

All load combinations are considered in the above analysis. The following fatigue loads are considered in liner design:

A. Thermal cycling due to annual outdoor temperature variations - Daily temperature variations will not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner plate. The number of cycles for this loading is 60 cycles for the plant life of 60 years.

B. Thermal cycling due to reactor building interior temperature varying during the heatup and cooldown of the reactor system - The number of cycles for this loading is assumed to be 500.

C. Thermal cycling due to the DBA was assumed to be one cycle.

5.2.1.4.7.4 Stresses and Permissible Strains

The best basis for establishing allowable liner plate strains is considered to be that portion of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically, the following sections are adopted as guides in establishing allowable strain limits:

Paragraph N-412 (m)(2) Thermal Stress (2)

Paragraph N-412 (n)

Table N-413

Figure N-414, N-415 (A)

Paragraph 414.5 Peak Stress Intensity

Paragraph N-415.1

Implementation of the ASME design criteria requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint (Paragraph N-412 (m)(2)).

The thermal stresses in the liner plate are considered to fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME Boiler and Pressure Vessel Code. The allowable stresses in Figure N-415 (A) are for alternating stress intensity for carbon steels and temperatures not exceeding 700 °F. In addition, the ASME Code further requires that significant distortion of the material be prevented.

In accordance with ASME Code Paragraph 412 (m)(2), the liner plate is restrained against significant distortion by anchors and never exceeds the temperature limitation of 700 °F and also satisfies the criteria for limiting strains on the basis of fatigue consideration. Paragraph 412 (n) Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels. Because of the conservative factors it contains on both stress intensity and stress cycles, and its being a part of a recognized design code, Figure N-415 (A) and its appropriate limitations are used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below 10 cycles, 10 cycles are used for a LOCA instead of one cycle.

Establishing an allowable strain based on the one significant thermal cycle of the accident conditions would permit an allowable strain (from Figure N-415A) of approximately two percent. The strain in the liner plate at the proportional limit is approximately 0.1 percent. The liner plate is allowed to go beyond proportional limit strains during the accident condition. The maximum allowable membrane compressive strain is conservatively set at 0.5 percent and membrane strain combined with flexural strains does not exceed one percent (compared to two percent shown above).

For the preceding, the material properties are essentially the same as those used in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

The predicted radial strains in the liner plate relative to this centerline of the penetration and adjacent to the thickened liner plate approach twice the liner strains at a typical location away from the penetrations. The predicted circumferential strains relative to the penetration approach zero as a limiting condition.

These strains are always in compression. The strains at the penetration reinforcing plate - liner intersection are very low relative to the ultimate strain capacity of the material and satisfy the criteria given above.

5.2.1.4.7.5 Weld Design

Paragraphs UW-8 to UW-19 of Subsection B of Section VIII of the ASME Code are used as a guide in design of welds. Particular attention is given in the design of welds to the anticipated behavior of the structure under accident conditions.

5.2.1.4.7.6 Design of Liner Plate Anchors

5.2.1.4.7.6.1 Design Criteria

The anchors are designed to preclude failure when subjected to the worst specified loading combinations. The anchors are also designed such that in the event of a missing or failed anchor the total integrity of the anchorage system is not jeopardized by the failure of adjacent anchors. The liner plate anchor design is identical to the liner plate anchor designs of Palisades, Point Beach, and Oconee Nuclear Plants.

A. Loading Conditions

The following loading conditions are considered in the design of the anchorage system:

1. Prestress
2. Pressure
3. Shrinkage and creep of concrete
4. Thermal gradients
5. Dead load
6. Earthquake
7. Wind or tornado
8. Flood
9. Vacuum

B. Factors Affecting Anchors

The following factors are considered in the design of the anchorage system:

1. Initial inward curvature of the liner plate between anchors due to fabrication and erection inaccuracies.
2. Variation of anchor spacing.
3. Misalignment of liner plate seams.
4. Variation of plate thickness.
5. Variation of liner plate material yield stress.
6. Variation of Poisson's ratio for liner plate material.
7. Cracking of concrete in anchor zone.
8. Variation of the anchor stiffness.

C. Design Conditions

The anchorage system satisfies the following conditions:

1. The anchor has sufficient strength and ductility so that its energy absorbing capability is sufficient to restrain the maximum force and displacement resulting from the condition where a panel with initial outward curvature is adjacent to a panel with initial inward curvature.

2. The anchor has sufficient strength to resist the bending moment which will result from Condition 1.

3. The anchor has sufficient strength to resist radial pullout force.

5.2.1.4.7.6.2 <u>Mathematical Method of Analysis</u>

When the liner plate moves inward radially as shown in Figure 5-28, the sections develop membrane stress due to the fact that the anchors move closer together. Due to initial inward curvature, the section between one and four deflect inward giving a longer length than adjacent sections and some relaxation of membrane stress occurs. It should be noted here that Section 1-4 cannot reach an unstable condition due to the manner in which it is loaded providing that the anchors do not fail.

The first part of the solution for the liner plate and anchorage system is to calculate the amount of relaxation that occurs in Section 1-4, since this value is also to be the force across Anchor 1 if it is infinitely stiff. The above solution can be obtained by solving the general differential equation for beams and the use of calculus to simulate relaxation or the lengthening of Section 1-4. Figure 5-28 shows the symbols for the forces that result from the first step in the solution.

Using the model shown in Figure 5-29 and evaluating the necessary spring constants, the anchor can now be allowed to displace.

The solution produces a force and displacement at Anchor 1, but the force in Section 1-2 is now (N) - $K_{RPL1} \cdot \delta_1$ and Anchor 2 is not in force equilibrium.

The model shown in Figure 5-29 may be used to allow Anchor 2 to displace and then find the effects on Anchor 1.

The displacement of Anchor 1 is now $\delta_1 + \delta_1'$ and the force on Anchor 1 is $K_c (\delta_1 + \delta_1')$. Now Anchor 3 is not in force equilibrium and the solution may continue to the next anchor.

After the solution is found for displacing Anchor 2 and Anchor 3, the pattern is established with respect to the effect on Anchor 1 and, by inspection, the solution considering an infinite amount of anchors is obtained in a form of a series solution.

The preceding solution produces all necessary results. The most important results are the displacement and force on the Anchor 1.

5.2.1.4.7.6.3 Final Method of Analysis

The method outlined in Section 5.2.1.4.7.6.2 produces an equation that is a very useful tool in designing the anchorage system. By varying the different variables which are contained in the equation, their effect on the design may be determined. If the conservative assumption is made that the spring constant K_{Bl} is small relative to the anchor spring constants K_c and K_c then the solution is fully dependent on the stiffness of the anchor.

Since the capacity of the anchor is both a function of its displacement and the applied force, the design must be based on energy considerations.

References 1, 2, and 8 can be used to evaluate an anchor spring constant. By using the equation obtained previously with the chosen spring constant, the amount of energy to be absorbed by the anchor may be evaluated.

By applying reasonable variations to the anchor spring constant, the most probable maximum energy may be found.

References 1, 2, and 8 may also be used to conservatively evaluate the amount of energy that the anchor system can absorb.

The factor of safety may be obtained by dividing the amount of energy that the system can absorb by the most probable maximum energy.

5.2.1.4.7.6.4 Results of Analysis

By considering the worst possible loading condition and the conditions stated below, Table 5-3 is obtained.

- Case I Simulates the perfect plate with a yield stress of 32 ksi and no variation of any other parameters.
- Case II Simulates a 1.25 increase in yield stress and no variation of any other parameters.
- Case III Simulates a 1.25 increase in yield stress, a 1.16 increase in plate thickness and a 1.08 increase for all other parameters. This case should adequately simulate the worst condition on the liner plate.
- Case IV Simulates a 1.88 increase in yield stress with no variation of any other parameter. The occurrence of this situation is considered highly unlikely since the maximum ultimate strength of the liner plate material is 72 ksi. This case is not considered as a design situation, but the anchor is still adequate.
- Case V Is the same as Case III except the anchor spacing is doubled to simulate what happened if an anchor is missing or has failed.

5.2.1.4.7.6.5 Liner Plate Anchor Tests

A. Test Description

On March 19, 1969 eight pieces of 2 x 3 x 1/4-inch angle (ASTM-A36) fillet welded to 1/4-inch plate (ASTM A-36) were cast in concrete. During placement, the concrete was thoroughly vibrated. 24 hours after placement, the forms were stripped and the blocks coated with clear seal and covered with burlap to simulate actual jobsite curing conditions. Eight cylinders were also taken at this time.

The concrete used is high early strength type with an f'_c = 5,000 psi @ 16 days. The fillet welds connecting the 2 x 3 x 1/4-inch angle to the liner are made by Advanced Heli Welders in accordance with AWS D 1.0-63. The welding rod is E6010 cellulose electrode. There were eight different weld and angle positions used. The specified welds are 3/16-inch fillet welds.

After seven days, two concrete cylinders were compression tested to failure with an average strength of 3,944 psi and a modulus of elasticity of 2.52 x 10^6 psi. On the basis of these tests it was decided that the concrete would reach its desired strength (f'_c = 5,000 psi) at 16 days. The Unit 1 reactor building uses concrete with f'_c = 5,500 psi @ 90 days. On April 4, 1969 (16th day) two cylinders were compression tested to failure with an average strength of 5,040 psi and a modulus of elasticity of 2.67 x 10^6 psi. Concrete cylinders were also compression tested at 28 and 187 days with the average results of 5,475 psi and 5,410 psi strength and 3.54 x 10^6 psi and 2.67 x 10^6 psi modulus of elasticity, respectively. The purpose of these latter tests was to determine the ultimate strength and modulus of elasticity of the concrete.

On April 4, 1969, seven of the eight liner plate concrete test blocks were compression tested to failure. These tests were performed on a 120,000 Tinius Olson Hydraulic Testing Machine. Measurement curves were made at loads of applied forces of 5,000, 10,000, 20,000 and in 10,000 increments thereafter. Additional measurements were made at specified loads or deflections after yielding occurred. The deflection readings were taken on eight dial gauges placed at different locations on the specimen.

The eight different specimens represented the possible configurations of two basic 3/16-inch intermittent fillet welds (4-12 and 6-12 applied to the different angle and weld orientations).

The eighth specimen was tested on September 22, 1969 (187th day) in the same manner as the other seven tested on the 16th day.

The results of the tests for both concrete cylinders and the liner plate anchor blocks are tabulated and plotted as load versus displacement curves.

B. Purpose and Scope of Tests

The tests were performed to obtain information that would substantiate the predicted load displacement relationships for the liner plate anchorage system. The loadings considered were in the plane of the liner plate and the displacements considered were parallel to the liner plate surface.

The tests and data taking apparatus were set up so that the following information could be obtained:

1. Load vs. displacement curve of the anchorage system.
2. Spring constant.
3. Total energy available in the anchorage system when subjected to shearing type loads.
4. Maximum load capacity.
5. Maximum displacement capability.

C. Summary and Conclusions

The test results as shown in Table 5-4 indicate the great load carrying capacity and displacement capability of the liner anchorage system being used. The high strength of the system can be attributed to the fact that the fillet weld is being loaded in one of its strongest directions and that the failure of the weld is not through the throat but in the face where the weld attaches to the liner plate. The specimen failures under applied shear loading indicated that the angle to plate weld underwent negligible rotation but was subjected to almost pure shear. The large displacement capacity of the anchorage system is primarily due to the ability of the concrete to yield locally at the angle to plate connection.

In a liner plate anchorage system of this type, it is not important that both displacement capability and load capacity be maximum values at the same time. The main criterion, rather, is that the system has sufficient available energy (which is defined as the area under the load displacement curve) to absorb the energy from the applied loading conditions.

Some of the tests, however, showed less load or displacement capacity than predicted values. This is attributable to various possibilities inherent in such type of testing:

1. The size of the specimens was very small compared to the real structure. This tended to make the system quite sensitive to dimensional variations. Since the specimens could not truly simulate prestressed mass concrete, the loads could not distribute as they would in a large structure. The cracking that was observed at the inner toe at the vertical angle leg should not occur in the real structure due to the effect of the additional channels which attach to the angles, geometry (cylindrical plate instead of one-way plate), and the prestressed mass concrete.

2. The major spalling of the concrete occurred at the outer edges where the concrete was unconfined. Also at the outer edges, the concrete under the weld could not distribute the load as it could in the center portion. In the real structure, the concrete will not be in an unconfined condition.

3. In the test condition, the displacement was increased as the load increased; but in the real structure, the primary membrane loads are self-relieving in nature and as the displacement increases, the loads will reduce.

4. The concrete used in the specimens had a high early strength (5,040 psi at 16 days after placement but a very low modulus of elasticity, 2.67 million psi) relative to the concrete that will exist in the real structure. The real anchorage system with aged concrete will have a higher spring constant and higher load capacity.

5. Due to rather large differences in the dial gauge readings on the opposite sides of the specimen, it is apparent that the load was not uniformly over the width of the specimen, rather it was distributed with considerable eccentricity. Attempts were made to eliminate this condition, but they were unsuccessful.

Even though some load, displacement, and energy absorbing values are slightly lower than predicted values, the overall test results indicate the suitability of such an anchorage system. Based on the factors indicated above, the anchorage system in the actual structure should have more than sufficient capacity to withstand the anticipated load conditions. These test results indicate the predictability of the liner plate anchorage system load versus displacement characteristics and the suitability of such a system for use in the actual structure.

5.2.1.4.8 <u>Missile Protection Criteria</u>

High pressure RCS equipment, which could be a source of missiles, is suitably screened either by the concrete shield wall enclosing the reactor coolant loops, by the concrete operating floor, or by special missile shields to block passage of missiles to the containment walls. Potential missile sources are oriented so that the missile is intercepted by the shields and structures provided. A structure is provided over the Control Rod Drive Mechanism to block any missile generated from fracture of the mechanisms.

Missile protection is provided to comply with the following criteria:

A. The reactor building liner plate is protected from loss of function due to damage by such missiles as might be generated in a DBA for break sizes up to and including the double ended severance of a main coolant pipe.

B. The engineered safeguards systems and components required to maintain reactor building integrity are protected against loss of function due to damage by the missiles defined later.

During the detailed plant design, the missile protection necessary to meet the above criteria is developed and implemented using the following methods:

A. Components of the RCS are examined to identify and classify missiles according to size, shape, and kinetic energy for purposes of analyzing their effects.

B. Missile velocities are calculated considering both fluid and mechanical driving forces which can act during missile generation.

C. The RCS is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double ended rupture of a main coolant pipe and designed to stop the missiles.

D. The structural design of the missile shielding of Seismic Class 1 structures, including the spent fuel pool, took into account both static and impact loads and is based upon the state of the art of the published missile penetration data. The design considered the most severe striking angle and missile orientation (end-on impact on the target surface). It was assumed that the total kinetic energy of the missile would be absorbed by the missile barrier and the deformation and frangibility of the missile was conservatively neglected. The penetration of missiles into concrete was estimated by the modified Petry formula (see Reference 7) and the Corps of Engineers formula (see Reference 3). The structural integrity and energy absorption capacity of the missile barriers were established using the energy methods. Further discussion pertaining to the missile barrier design is incorporated in Section 5.1.3.1.

The types of missiles for which missile protection is provided are:

A. Valve stems

B. Valve bonnets

C. Instrument thimbles

D. Various types and sizes of nuts and bolts

E. External missiles defined in Section 5.2.1.2.6

A list of typical missiles and their characteristics is presented in Table 5-2.

Certain postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, inservice inspection, and conservative design as applied to the particular component and, therefore, protection is not provided for missiles in this category, such as missiles caused by massive, rapid failure of the Reactor Coolant Pump casings and flywheels. It is also considered incredible that the reactor building polar crane would be a source of missiles that may endanger the containment structure since it is designed to maintain its integrity and not damage equipment during a Design Basis Earthquake (DBE).

5.2.1.5 Structural Design Analysis

The reactor building is analyzed for individual loading cases as described in Section 5.2.1.4.1. To find the stresses or moments and forces in these places, the computer program described later is employed. The results of the program are carefully coordinated in the framework of the overall design.

The ACI 318-63 Code design methods and allowable stresses are used for concrete and prestressed and non-prestressed reinforcing steel except as noted herein.

5.2.1.5.1 Important Design Areas

The main areas considered for design analysis of prestressed concrete reactor building are:

A. The restraints at the top and bottom of the cylinder

B. The restraints at the edge of the dome

C. The stresses and strains around the large penetrations

D. The behavior of the base slab relative to an elastic foundation

E. The stresses due to transient temperature gradients in the liner plate and concrete

F. Stresses within the ring girder

G. Penetrations and locations of concentrated loads

H. Seismic loads

I. Tendon anchorage zones

5.2.1.5.2 Analytical Techniques

The reactor building analysis is performed by the finite element method developed by E. L. Wilson, under sponsorship of National Science Foundation Research Grant G18986. This method is now further developed to apply to axisymmetric structures. Such a method of analysis is normally used only for thick walled structures where conventional shell analysis yields inaccurate results. Good correlation can be demonstrated between the finite element analysis method and the test results for thick wall model vessels.

For analysis of the reactor building under axisymmetric loads such as dead load, live load, temperature, pressure, and post-tensioning, the finite element computer program is used.

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints).

In the application of the method to the axisymmetric reactor building, the continuous structure is replaced by a system of rings of triangular cross sections which are interconnected along circumferential joints. Based on energy principles, equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are unknowns of the system. A solution of this set of equations is inherent in the solution of the finite element system.

The finite element grid of the structure's base slab is extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab.

The use of a finite element analysis permits the prediction of the stress pattern at any location on the structure.

Demonstration of the use of finite element analysis exists for structures such as:

A. Arch dams (including a portion of the foundation)

B. Thick walled prestressed concrete vessels

C. Spacecraft heat shields

D. Rocket nozzles

The computer program used in the analysis is capable of handling the following inputs:

A. Seven different materials

B. Nonlinear stress-strain curves for each material

C. Any shape transient temperature curves

D. Any shape axisymmetric loadings

The program outputs are:

A. Direct and shear stresses for each element

B. The principal stresses and their directions for each element

C. The deflections for each nodal point

D. Resultant forces, shears, and moments at specified sections

An auxiliary computer program plotted stress curves based on the above analysis program outputs.

Equilibrium is checked at critical locations such as the reactor building wall and dome. Table 5-5 contains an equilibrium check of the reactor building wall (vertical direction) for unfactored loading cases of dead load, prestress, and pressure.

5.2.1.5.3 <u>Thermal Loads</u>

The thermal loads are results of the temperature differential within the structure. The design temperature gradients for the reactor building are shown in Figure 5-5. The finite element analysis is prepared by specifying temperatures at every nodal point in the finite element mesh. Thermal stresses are calculated at the center of each element. The liner plate is handled as an integral part of the structure, having different material properties and not as a mechanism which would act as an outside source to produce loading on the concrete portion of the structure.

The liner plate is designed to have plastic deformation as a result of prestressing and high thermal stresses.

The finite element method includes this analysis too, by successive approximations, changing the modulus of elasticity of those elements which are subject to stresses higher than the proportional limit.

The output of the computer analysis indicates the effect of the thermal loads on liner plate and concrete. The liner plate and the inside of the concrete are subject to compressive stress and the outside of the concrete section is subject to tension. These tension stresses balance the compressive stresses so that, except close to any discontinuity, there is no resultant membrane force. That is, all the compressive forces in the liner plate are carried by the prestressed concrete and reinforcement near the outside surface of the structure.

The compressive stresses in the liner plate exceed the proportional limit in the case of the DBA. An increased temperature could keep the liner plate in plastic condition but only a negligible additional stress could develop and thermal stresses would stay unchanged.

5.2.1.5.4 Analysis of Reactor Building Post-Tensioning System Based on As-Built Conditions

Two horizontal tendons in the Unit 1 reactor building were not installed because of obstructions in the tendon sheathing. The object of this analysis is to determine the effects of omitting these tendons from the post-tensioning system. A description of the tendon layout and the location of the two omitted tendons is shown in Figure 5-30. The numbers of the omitted tendons are 31H12 and 32H8.

During prestressing of the reactor building tendons, the stressing jacks broke. The stressing jack failure per se has no influence on the capability of the installed and post-tensioned prestressing system whose adequacy has been determined and reported within this chapter. Tendons were removed and replaced if the stressing washer was damaged or the wires broken or deformed as a result of jack failure. Accordingly, the expected service of the tendons is the same as though no jack failure had occurred.

Glancing impact of the stressing washer on the edges of the hole in the bearing plate caused upsetting of the bearing plate. These small bearing plate deformations were ground smooth to allow post-tensioning of tendons for that location. Examinations were made of trumpet deformations and potential grease leaks were sealed. The concrete and liner plate adjacent to impact areas were examined and no evidence of damage was found.

5.2.1.5.4.1 Analysis Procedure

The finite element method was used to analyze the structure. The finite element mesh used in the analyses was essentially the same as that used in the design of the reactor building. Modifications were made to account for the omitted tendons. A section of the mesh near these tendons is shown in Figure 5-31.

The results from the finite element analyses were combined with those from seismic analyses according to the loading combinations given in Section 5.2.1.4.6. A summary of the wall forces and moments from the various loadings and the governing load combinations is given in Tables 5-6 through 5-11.

For the governing loading condition of 1.05D + 1.0F + 1.0T + 1.5P, plots were made comparing the forces and moments in the wall when the two tendons are omitted to the forces and moments in the wall under full prestress. These are shown in Figures 5-32, 5-33 and 5-34.

The wall forces from the finite element analyses are used to compute the radial shear stress and the membrane tension stress, as well as the liner plate, tendon, and rebar stresses in both the hoop and meridional directions. These stresses are compared with allowable values.

Two additional calculations were made as a check on the results obtained from the finite element analysis.

5.2.1.5.4.2 Analysis Results

The results of the finite element analyses and the seismic analyses are listed in Tables 5-6 through 5-11, in the form of resultant forces and moments at various sections through the wall.

Examination of these results has shown that the most critical of the loading combinations given in Section 5.2.1.4.6 are

A. $1.05D + 1.0F + 1.0T_A + 1.5P$

B. $1.05D + 1.0F + 1.25P + 1.25E$

The wall forces, moments, and the corresponding stresses under these loading combinations were computed. The following conditions were observed:

A. The meridional force and the tangential shear were not affected by the omission of the two tendons.

B. The change in hoop moment when the two tendons were omitted was less than one percent.

C. The radial shear, which is small under full prestress, increased somewhat when the two tendons were omitted, but it remained relatively insignificant.

D. The change in meridional moment when the two tendons were omitted was less than five percent.

E. The hoop force was noticeably changed when the two tendons were omitted.

F. Analysis showed that all stresses remained within allowable limits.

   1. The principal membrane tension stress was 187 psi compared to an allowable of 222 psi.

   2. The maximum stress in the hoop rebar was 28 ksi compared to an allowable of 54 ksi.

   3. The maximum stress in the hoop tendons was 148 ksi compared to an allowable of 192 ksi.

   4. The maximum stress in the vertical rebar was 18 ksi compared to an allowable of 54 ksi.

   5. The maximum stress in the vertical tendons was 138 ksi compared to an allowable of 192 ksi.

G. Approximate hand computations showed good agreement with finite element results.

5.2.1.5.4.3 <u>Conclusion</u>

Despite the omission of tendons 31H12 and 32H8 from the post-tensioning system, the reactor building still meets the design criteria as specified in this chapter.

5.2.1.5.4.4 Design Conservatism From As-Built Conditions

The original design analyses were based on conservative design assumptions. In some cases, the as-built conditions provide even more conservatism in the design. A few examples of this are:

A. The average prestressing force in the tendons after losses was computed on the basis of a friction factor of $\mu = 0.14$. Field data on tendon elongation has shown that this value is much lower than the original design assumption. As a result, the average prestressing force in the tendons is actually about seven percent greater than the assumed design value.

B. In many of the design loading conditions, thermal accident is combined with pressure loading. Under thermal accident, a thermal gradient is assumed across the concrete wall. However, since heat transfer through concrete is a relatively slow process, the pressure loads will have dissipated before the thermal gradient is established. Thus, pressure and thermal gradient would not occur simultaneously.

C. The concrete strength assumed in design was 5,500 psi. Compression tests on concrete cylinders from the jobsite have shown that an average concrete strength is 6,700 psi.

5.2.1.5.4.5 Stress Calculations From Finite Element Analyses

5.2.1.5.4.5.1 Shear

In the calculation of allowable shear stress, the prestressing forces are neglected. This is conservative. The allowable shear stress on an unreinforced wall section according to ACI 318-63 Section 1201 C is

$$v = 1.1\sqrt{f'_c} = 1.1\sqrt{5500} = 82 \text{ psi}$$

The maximum radial shear under the governing loading combinations is 12 kips/ft (see Table 5-8). The corresponding shear stress is

$$v = 12{,}000/(12 \times 45) = 22.2 \text{ psi}$$

This is less than allowable. The tangential shear did not change from the original design conditions.

5.2.1.5.4.5.2 Membrane Stress

The allowable principal concrete tension as given in Section 5.2.1.4.6 is $3\sqrt{f'_c} = 222$ psi. The governing wall section is Section 71-77 (see Figure 5-31). Two loading conditions are examined. Under the loading condition 1.0D + 1.0F + 1.0T_A + 1.5P, the wall forces are

Hoop Force = F_h = 101 kips/ft.

Normal Force = F_n = -191 kips/ft.

Tangential Shear = V_t = 0 kips/ft.

The corresponding stresses are

Hoop Stress = σ_h = 101,000/(12x45) = 187 psi

Normal Stress = σ_n = -191,000/(12x45) = -354 psi

Tangential Shear Stress = τ_t = 0 psi

Under the loading condition 1.05D + 1.0F + 1.0T_A + 1.25P + 1.25E, the wall forces are

F_h = -20 kips/ft.

F_n = -103 kips/ft.

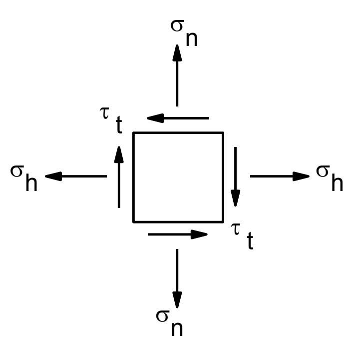
V_t = 76 kips/ft.

The corresponding stresses are

σ_h = - 20,000/(12x45) = -37 psi

σ_n = - 103,000/(12x45) = -191 psi

τ_t = 76,000/(12x45) = 141 psi



The principal stresses are

$\sigma_p = (\sigma_h + \sigma_n)/2 \pm \sqrt{(((\sigma_h - \sigma_n)/2)^2 + \tau_t^2)}$

For loading conditions 1.05D + 1.0T_A + 1.5P, the principal concrete tensile stress is

$\sigma_p = \sigma_h$ = 187 psi

For loading condition 1.05D + 1.0F + 1.0T_A + 1.25P + 1.25E, the principal concrete tensile stress is

$\sigma_p = (-37-191)/2 + \sqrt{((-37 + 191)/2)^2 + 141^2)}$ = 47 psi

For both these loading conditions, the calculated stress is less than allowable.

5.2.1.5.4.5.3 Hoop Reinforcement

The governing case for forces in the hoop direction is at Section 71-77 under loading condition 1.05D + 1.0F + 1.0T_A + 1.5P. The hoop force is 101 kips/ft and the hoop moment is -843 ft. kips/ft. All of the hoop moment is due to thermal loading.

The stresses caused by thermal moment are computed according to the method outlined in Section 5.2.1.5.1.1. The tendon area in one foot is 2/3 x 9.128 = 6.08 inches^2/feet. The hoop reinforcing steel is #10 bars at 12 inches spacing, or 1.27 inches^2/feet. The liner plate area is 1/4 in. x 12 in. = 3 in^2/ft.

The stresses across the wall from the finite element analyses for loading condition 1.05D + 1.0F = 1.0T_A + 1.5P are

In order to compute the steel stresses and take advantage of the self relieving of thermal moment due to concrete cracking, the neutral axis is shifted by an amount Δ. The new forces are

Tendon and rebar tension force = 8.6(7.35)(1779 + Δ)

Liner plate compression force = 8.6(3)(28,965/8.6 - Δ)

Concrete compression force = 7.5(12)(2082 - Δ)

Σ F = 101,000 lbs.

8.6(7.35)(1779 + Δ) - 8.6(3)(28,965/8.6 - Δ) - 7.5(12)(2082 - Δ) = 101,000

179 Δ = 262,824

Δ = 1,468 psi

Rebar stress = (1779 + 1468)(8.6) = 27,924 psi (tension)

Tendon stress = 120,000 + (1779 + 1468)(8.6) = 147,924 psi (tension)

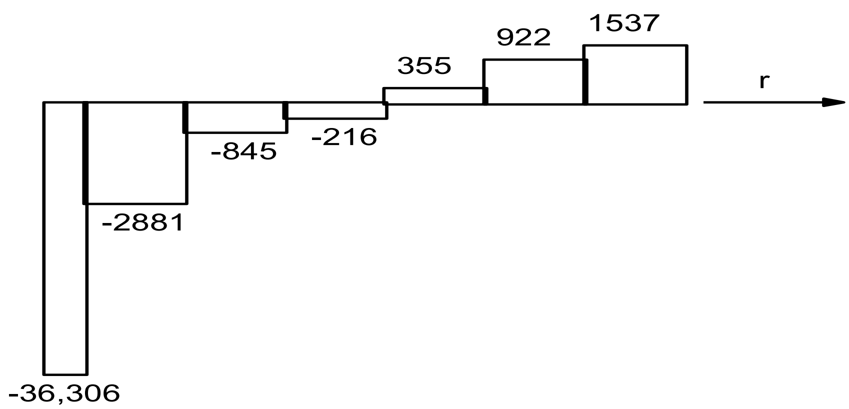
Liner plate stress = (28,965 - 8.6 x 1468) = 16,340 psi (compression)

5.2.1.5.4.5.4 Meridional Reinforcement

The governing case for forces in the meridional direction is at Section 85-91 under loading condition 1.05D + 1.0F + 1.5P. The meridional force is - 194 kips/ft and the meridional moment is - 978 kip ft/ft. The meridional moment due to thermal loading only is - 912 ft. kips/ft.

The stresses caused by thermal moment are computed according to the method outlined in Section 5.2.1.5.1.1. The tendon area in a 1-foot section is 102 x 9.128/60 x Π = 4.94 inches^2/feet. The meridional reinforcing steel is #10 bars at 12-inch spacing, or 1.27 inches^2/feet. The liner plate area is 1/4 inch by 12 inches = 3 inches^2/feet.

The stresses across the wall from the finite element analyses for loading condition 1.05D + 1.0F + 1.0T_A + 1.5P are



In order to compute the steel stresses and take advantage of the self relieving of thermal moment due to concrete cracking, the neutral axis is shifted by an amount Δ.

The new forces are

Tendon and rebar tension force = 8.6(6.31)(1537 + Δ)

Liner plate compression force = 8.6(3)(36,306/8.6 - Δ)

Concrete compression force = 7.5(12)(2881 - Δ)

Σ F= -194,000 lbs.

8.6(6.31)(1537 + Δ) - 8.6(3)(36,306/8.6 - Δ) - 7.5(12)(2881 - Δ) = -194,000

170 Δ = 90,801

Δ = 534 psi

Rebar stress = (1537 + 534)(8.6) = 17,810 psi (tension)

Tendon stress = 120,000 + (1537 + 534)(8.6) = 137,810 psi (tension)

Liner plate stress = (36,306 - 8.6 x 534) = 31,714 psi (compression)

5.2.1.5.4.6 Verification of Finite Element Results

5.2.1.5.4.6.1 Equilibrium Check

The prestressing force in the two omitted tendons is

Average Prestressing Pressure = 82.8 psi

Average Stress in Tendon = 119.7 ksi

Tendon Area = 9.128 in^2

T = 2 x 9.128 x 119.7 = 2,185 kips

The net difference in hoop force between the omitted tendon case and the full prestress case for each section is given in Table 5-12. The summation of the forces from each section gives a total difference in hoop force of 2,184.25 kips.

5.2.1.5.4.6.2 Check Using Classical Theory of Cylindrical Shells

The wall forces midway between the two omitted tendons are calculated using the classical theory of cylindrical shells outlined in Timoshenko's Theory of Plates and Shells, Chapter 15, Articles 114 & 115.

The forces calculated are those which would be created by omission of the two tendons. This is done by considering a long, thin cylinder with two outward ring loads at the locations of the two omitted tendons.

The equations for computation of wall forces are

Meridional Moment = $M_x = (P/4\beta) \times \Psi(\beta x)$

Radial Shear = $Q_x = (P/2) \times \theta(\beta x)$

Hoop Force = $N_\Phi\ (EhP/8a\ \beta^3 D) \times \phi(\beta x)$

where:

P = Ring Load = 82.8 x 18 x 1/1000 = 1.49 kips/in.

E = Modulus of Elasticity of Wall = 4.5 x 10^3 ksi

a = Radius of Wall = 718.5 in.

h = Wall Thickness = 45 in.

ν = Poisson's Ratio = 0.25

$= \sqrt[4]{(3(1-\nu^2)/a^2h^2}$ = 0.0072 1/in.

D = $Eh^3/12(1-\nu^2) = 2.237 \times 10^8$ in. kips.

$\Psi(\beta x) = e^{-\beta x} (\cos\beta x - \sin\beta x)$

$\theta(\beta x) = e^{-\beta x} \cos\beta x$

$\phi(\beta x) = e^{-\beta x} (\cos\beta x + \sin\beta x)$

The forces are checked midway between the two tendons at x = 78 in.

At x = 0 in.	At x = 78 in.
$\beta x = 0$	$\beta x = 0.5616$
$\Psi (\beta x) = 1$	$\Psi (\beta x) = 0.1809$
$\theta (\beta x) = 1$	$\theta (\beta x) = 0.4835$
$\phi (\beta x) = 1$	$\phi (\beta x) = 0.7866$
$M_x = 51.7$	$M_x = 9.35$
$Q_x = 0.745$	$Q_x = 0.36$
$N_\Phi = 4.05$	$N_\Phi = 3.19$

The forces from both tendons are summed to give the following totals at x = 78 in.

N_Φ = 3.19 + 6.38 = kips/in. = 76.6 kips/ ft.

M_x = 9.35 + 9.35 = 18.7 kip in./in. = 18.7 kip ft./ft.

Q_x = 0.36 + 0.36 = 0.72 kips/in. = 8.6 kips/ft.

The finite element results show

N_Φ = 720 - 653 = 67 kips/ft.

M_x = 30 - 14 = 16 kip ft./ft.

Q_x = 0 kips/ft.

The agreement between the two analyses is good with the exception of radial shear. However, the radial shear is small and is influenced by the presence of the base slab.

5.2.1.5.5 Tendon Failure Analysis

There are 102 vertical tendons, 171 hoop tendons, and 90 dome tendons, using the 186-wire system. The hoop tendons are placed in three 240-degree sections around the cylinder using three buttresses as anchorages. Therefore, failure of a hoop tendon or a series of adjoining tendons or spaced hoop tendons is limited between 240-degree segments of the reactor building.

All prestressed tendons are subjected to the most critical stresses predicted for design during initial tensioning. There is a loss of prestress of the order of 15 percent, which reduces the stress level in the tendons. Even at the factored loads, the stress in the tendons is not as high as during initial tensioning. Each tendon is pretested at the time of initial jacking and the stress in the tendons under accident loading is approximately 80 percent of this jacking stress. This means that the possibility of tendon failure under design accident loading is quite remote.

The integrity of the reactor building and the liner plate is not affected by the postulated loss of two or three tendons during accident conditions. The thick concrete walls of the reactor building are sufficient to transmit the force from the adjacent tendons without resulting in serious local stresses.

5.2.1.5.6 Stresses Near Equipment Openings

Analytical solutions for the determination of state of stress in the vicinity of equipment openings are obtained from reference to the following articles:

A. A. C. Eringen, A. K. Naghdi, and C. C. Thiel, "State of Stress in a Circular Cylindrical Shell with a Circular Hole," Welding Research Council Bulletin No. 102, January, 1965.

B. Samuel Levy, A. E. McPherson and F. C. Smith, "Reinforcement of a Small Circular Hole in the Plane Sheet Under Tension," Journal of Applied Mechanics, June, 1948.

The analysis of the reactor building as a whole was first carried out without considering the openings in it. This analysis has been done by using the finite element program.

The reactor building structure with the opening in it is analyzed in the following steps:

A. Formulation of differential equations for the shell in complex variable form with the center of the hole as the origin (see Reference A above).

B. Solution of the differential equations (see Reference A above).

C. Evaluation of parameters in the solution (see Reference A above).

D. Formulation of the boundary conditions based on the stresses obtained from the vessel analysis above, without the hole.

E. Calculation of membrane forces, moments, and shears around and at the edge of the opening.

F. The wall thickness around the opening is then increased and reinforced to carry the higher forces, moments, and shears. The effect of the thickening on the stress concentration factors is considered using Reference B above.

G. Evaluation of some of the prestressing effects.

H. Finally, the design is checked to insure that the strength of the reinforcement provided replaced the strength removed by the opening. This check is used to maintain a good degree of compatibility between the general vessel shell and the area around the opening.

The deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis and does not include the effect of the curvature of the shell. However, it gives assurance to the correctness of the assumed membrane stress pattern caused by the prestressing around the opening.

The seismic load produces vertical membrane stress in the structure based on a cantilevered circular beam subjected to base accelerations. The membrane stresses at the opening are modified by appropriate stress concentration factors.

The temperature variation through the concrete wall produces a stress condition like one caused by a moment: constant in all directions on the continuous cylindrical or spherical surface. However, at any discontinuity, such as an opening, stress concentrations occur.

Using the center of the opening as the reference point to relate the directions of moments, the radial moment is zero at the edge of the opening, there being no resistance against radial rotation. The hoop moment is highly increased, the outside fiber being forced to take the shape of a larger circle while the inside fiber takes the shape of a smaller circle.

Away from the edge of the opening both moments gradually reach the constant value on the undisturbed portion of the cylinder.

In case of $1.5P + 1.0T_A$ (accident temperature), the cracked concrete with highly strained tension reinforcement constitutes a shell with stiffness decreased but still constant in all directions.

In the case of accident temperature combined with low internal pressure, very small or no tension develops on the outside, so the thermal strains are built up without the relieving effect of the cracks. However, as has already been stated elsewhere, the liner plate reaches its yield stress, and so does the concrete at the inside corner of the penetration, thereby relieving once again the very high stresses but still carrying a high moment in the state of redistributed stresses.

For the analysis of the thermal stresses around the opening the following method is used.

At the edge of the opening a uniformly distributed moment equal but opposite to the moment existing on the rest of the shell is applied and evaluated using the methods of the preceding reference and the effects are superimposed on the stresses calculated by the computer using the finite element method for axisymmetric solids.

The membrane stresses at design accident loading are not expected to be significantly different from the pattern of membrane stress during the proof test.

5.2.1.5.7 Seismic Analysis

5.2.1.5.7.1 General

The loads on the reactor building caused by earthquake are determined as a result of a dynamic analysis of the structure. The dynamic analysis is made on a model consisting of lumped masses and weightless elastic columns acting as spring restraints. The analysis is performed in two stages: the determination of the natural frequencies of the structure and its mode shapes and the modal response of these modes to the earthquake as by the response spectrum method.

This method uses an earthquake described by response spectrum curves shown in Figures 5-10 and 5-11.

5.2.1.5.7.2 Criteria Used to Determine Seismic Response of Structures

A. Determination of Natural Frequencies and Mode Shapes

The structure is converted into a mathematical model in terms of lumped masses and stiffness coefficients. Within the building, points are chosen to lump the weights of the structure and between these locations, properties are calculated for moments of inertia, cross-sectional areas effective shear areas and lengths. These properties are used in a computer program to obtain the flexibility coefficients of the building at the mass locations.

The natural frequencies and mode shapes are computed from the equations of motion of the lumped masses and stiffness properties.

The equations of motion in the form of

$$[K]\ [\Delta] = \omega^2\ [M]\ [\Delta]$$

are solved by iteration techniques in a computer program.

[K] = Matrix of stiffness coefficients including the combined effects of shear, flexure, rotation, and horizontal translation.

[M] = Matrix of concentrated masses.

[Δ] = Matrix of mode shape.

ω = Natural frequency of vibration (rad/sec).

The computation results in several values of ω_n and mode shapes Δ_n for n = 1,2,3,....n where n is the number of degrees of freedom, i.e. lumped masses, assumed in the idealized structure.

B. Damping

The following values of damping and ground acceleration are used in the analysis together with the natural period to obtain spectral acceleration:

Ground Acceleration	% Damping	Ground Acceleration	% Damping
0.20	5%	0.10	2%

5.2.1.5.7.3 Criteria Used to Determine Forces, Deflections and Accelerations

The response of each mode of vibration to the design earthquake is computed by the response spectrum technique as follows:

A. The effective force of the nth mode

where

$F_n = W_n S_{an_n} (\omega_n, \gamma_n)$

W_n = Effective weight of the structure in the nth mode computed from:

$$W_n = \frac{\left[\sum_{x=1}^{x=m} {}_x\Delta_{xn} W_x\right]^2}{\sum_{x=1}^{x=m} (\Delta_{xn})^2 W_x}$$

where the subscript x refers to levels throughout the height of the structure

ω_n = Angular frequence of the nth mode.

$S_{an} (\omega_n, \gamma_n)$ = Spectral acceleration of a single degree of freedom system with a damping coefficient of γ_n, obtained from the response spectrum.

m = Number of lumped masses.

B. The horizontal load distribution for the nth mode is then computed as:

$$F_{xn} = \frac{F_n (\Delta_{xn} W_x)}{\sum_{x=1}^{x=m} {}_x\Delta_{xn} W_x}$$

Accelerations a_{xn} for each mode at each point are determined by:

$$A_{xn} = \frac{F_{xn}}{W_x/g}$$

Deflections d_{xn} for each mode at each point are determined by:

$$d_{xn} = \frac{A_{xn}}{W_n^2}$$

As shown above, modal inertial forces of the structure are generated at each point for each seismic excited mode. These inertial forces are used to compute shears and moments per mode. The several mode contributions are then combined to give the final response of the structure to the design earthquake on the basis of the sum of the absolute values.

Evaluation of the modal frequencies obtained by the seismic analysis of the structure indicates that there are no closely spaced modal frequencies.

5.2.1.5.7.4 Criteria Used to Determine Seismic Response of Systems and Equipment

To determine the seismic response of equipment, a time history analysis was made of the structural model using an earthquake as the input ground motion.

This analysis generated the floor acceleration time histories at the various mass points at which the equipment is located. The equipment response spectrum was then generated for each of the floor acceleration time histories at various damping values for use in the design of the equipment. The amplification and frequency spectra of the structure was considered. The reactor core seismic analysis was performed using time histories as described in Chapter 3.

To determine the piping and instrumentation response to an earthquake, Seismic Class 1 piping systems are analyzed dynamically by means of a three-dimensional model using two-thirds of the horizontal ground response spectra for the vertical spectra. For each piping system, a horizontal response spectrum analysis is performed for the North-South and for the East-West directions. The results of each analysis are combined with the results of excitation in the vertical direction. The design internal force of moment, restraining force or moment, or displacement is the larger number obtained from either of these analyses. The possible combined vertical and horizontal amplified response load for the design of piping and instrumentation includes the effect of the response of building, floors, supports, equipment, and components.

5.2.1.5.7.5 Additional Considerations

A. Variations in Structural Response

   The equipment response spectrum curves have been broadened by a smooth curve in the vicinity of the peaks which are clearly associated with the natural frequencies of structures. As an example, in the containment structure the fundamental natural frequency is approximately 4.8 cps with a rigid base and 3.3 cps with an assumed flexible base. Both of these values, which constitute a considerable frequency range, are either on a peak or within 10 percent of the peak of the ground response spectrum curve. This means that the most probable range of frequencies essentially bounds the peaks of the ground response spectrum curve and the response will be approximately equal for any frequency between the bounds. For the containment structure, the average response at the various levels will decrease approximately 25 percent if the natural frequency is varied from 3.3 cps for the assumed flexible base to 4.8 cps for the rigid base.

B. Determination of Combined Vertical and Horizontal Response

The vertical ground design spectrum curves were derived as two-thirds of the horizontal values. This 2/3 value is considered to be conservative by the strong motion records from both the United States and foreign countries.

Analyses for both the horizontal and vertical directions were performed using the ground design spectrum curves. The forces, moments, and resulting stresses were combined directly assuming a simultaneous occurrence of the vertical and horizontal motions. Vertical structural elements were considered vertically rigid. Horizontal structural elements of the Seismic Class 1 structures were further investigated for vertical responses.

In the reactor building, the columns and floor supporting Seismic Class 1 equipment were found to be rigid. In the other Seismic Class 1 structures, these floors, due to relatively short spans and large depths, fell generally in the rigid category. Special cases were investigated where applicable.

The vertical ground response spectrum curve was used for equipment since it is attached to rigid portions of the structure. Rigid portions have high natural frequencies and the ground motion would not be appreciably altered. The equipment forces, moments, and stresses from both the vertical and horizontal motion were considered as acting simultaneously.

C. Torsional Effect Considerations

The structure was originally considered to be torsionally rigid and not to respond in torsional modes. But, the analysis techniques did consider additional forces due to the center of rigidity and the center of mass having different locations, where applicable.

However, a study was made for a symmetrical containment structure 200 feet high, 120 feet diameter with walls three feet thick in accord with N. M. Newmark's publication, "Torsion in Symmetrical Building" (Proceedings of Fourth World Conference on Earthquake Engineering, 1969). The results showed that, under seismic conditions, the increase in the membrane shear stress due to torsion was about 10 percent of that obtained in the original analysis. Based on this study, the seismic analysis of Unit 1 containment was reviewed and the total resulting stresses found to be within allowable limits.

Another study made in accord with the above mentioned publication was applied to the reinforced concrete intake structure, which is the only other symmetrical Seismic Class 1 structure. The structural system has shear walls and horizontal slabs acting as rigid diaphragms. The results indicated a 22 percent increase in the concrete shear stresses at the end walls under maximum earthquake conditions. This increase in stresses was found to be well within the allowable limits.

Seismic stresses were investigated in the reactor building base slab and the reactor building wall, then compared with the governing load combination.

A. Reactor Building Base Slab: (Max. Predicted Stresses)

Location	Material	Seismic Stress	Combined Stress	Governing Load Combination
Top	Reinforced Steel	37,400 psi	51,700 psi	$1.05D+1.25P+1.05T_A$ $+1.25E+1.0F$
	Concrete	-1,480 psi	-2,040 psi	
Bottom	Reinforced Steel	36,800 psi	43,500 psi	$1.0(D+P+T_A+E'+F)$
	Concrete	-1,420 psi	-1,720 psi	

B. Reactor Building Wall:

Location	Material	Seismic Stress	Combined Stress	Governing Load Combination
Outside Face	Reinforced Steel	30,200 psi*	38,000 psi*	$1.0(D+P+T_A+E'+F)$
	Concrete	-1,770 psi**	-2,550 psi**	
Inside Face	Reinforced Steel	28,400 psi*	44,000 psi*	$1.05D+1.25P+1.05T_A+$ $1.25E+1.0F$
Concrete		-326 psi**	-4,250 psi**	$1.0D+1.0E'+1.05T_A+1.0F$

* At the top of base slab
** 10 feet above base slab

In general, seismic stress contributions in the reactor building wall are less than 50 percent.

5.2.1.5.8 Safety Margins of Containment Structure

The predicted state of stress in the reactor building is essentially biaxial since the radial stress is, in general, of small magnitude. Biaxial stresses in concrete are considered in Appendix A (Design of Two Way Slabs) of the ACI 318-63 Code. To demonstrate that the reactor building is designed and built with sufficient safety margins, the allowable concrete stress of $0.45\sqrt{f'_c}$; $0.60\sqrt{f'_c}$; $0.85\sqrt{f'_c}$; $6\sqrt{f'_c}$; $3.5\sqrt{f'_c}$; $2\sqrt{f'_c}$; 1.1, will be considered in the following paragraphs:

$.45f'_c$ and $.60f'_c$ (ACI 2605(b) - Allowable Concrete Compression). The allowable concrete stresses of $0.30f'_c$ for membrane compression and $0.60f'_c$ for combined flexural and membrane compression as defined in Section 5.1.1.4.4. "At Design Loads" were established for a structure which is subjected to a detailed analysis that includes thermal effects.

In the reactor building design, the prestress is primarily used to induce external loads to balance internal forces. Since the induced loads cause membrane concrete compression, it was necessary to place a limit on predicted concrete compressive stress that related to creep effects as well as strength capability. The value of $0.30f'_c$ as opposed to $0.60f'_c$ defined in the ACI Code is considered to be a conservative limit for concrete membrane compressive stress and was imposed to insure that the membrane creep losses will be generally small.

The limit of $0.60f'_c$ as opposed to $0.45f'_c$ as given in the ACI Code applies to the combination of predicted membrane and flexural compression. This value is considered to be conservative in view of the fact that thermal loads are included.

$0.85f'_c$ (ACI 2608 - Allowable Concrete Compression). In the design of the post-tensioned members, in addition to the working stress design, each member must be investigated for its ultimate strength. This requirement is incorporated in Section 5.2.1.4.6, "Factored Load Combinations." $1.0f'_c$ is used as the allowable concrete compressive stress at ultimate strength as opposed to $.85f'_c$ specified by the ACI Code. The concrete stresses in the reactor building are conservatively calculated from the factored loading equations by the modified working stress method which assumes a straight line stress distribution. The predicted maximum concrete stress in the reactor building less than $0.75f'_c$.

$6\sqrt{f'_c}$ (ACI 2605(b) - Allowable Concrete Tension). A maximum of $6\sqrt{f'_c}$ tension in the concrete is permitted by the ACI 318-63 Code for post-tensioned beams subjected to unfactored loads. For the same condition, an allowable maximum flexural concrete tension of $3.0\sqrt{f'_c}$ (Section 5.2.1.4.3) is used in the reactor building design. Reinforcing steel is provided in all predicted tension zones caused by flexure.

$3.5\sqrt{f'_c}$, $2\sqrt{f'_c}$, and $1.1\sqrt{f'_c}$ (ACI 1701(d), 1207(c), 1201(c) - Allowable Concrete Shear). The shear design of the reactor building wall and dome is in accordance with the prestressed concrete Section 26 of the ACI 318-63 Code with modifications as described in Section 5.2.1.4.6.

The reactor building base slab is not prestressed and acts primarily in bending. This condition is covered by the ACI Code. Therefore, the same concrete shear values, as supplied by the ACI Code, are used in the base slab design.

5.2.1.5.8.1 At End of Service Life

During normal operation and postulated accident conditions, the concrete adjacent to the liner plate is predicted to be in compression. Only in the case of reactor shutdowns and startups coincident with an infrequent ambient temperature of 100 °F outside the reactor building, concrete adjacent to the liner plate is predicted to have a negligible amount of tension. The following example predicts the magnitude of concrete tension adjacent to the liner plate in the vertical direction of the reactor building wall for the above condition:

σ_T = Predicted concrete stress due to the thermal gradient

α = Thermal coefficient of concrete = 6.5×10^{-6}

ΔT = Thermal gradient (100 °F - 55 °F = 45 °F)

U = Poisson's ratio of concrete

E_c = Modulus of elasticity of concrete

σ_p = Predicted compressive stress in concrete due to post-tensioning after all losses

F_p = Vertical prestress force per foot horizontally

$\Sigma\sigma_c$ = Predicted compression in concrete

$\Sigma\sigma_T$ = Predicted tension in concrete

$$\sigma_T = \frac{\pm\ \alpha\ E_c\ \Delta T}{2(1 - U)}$$

$$\sigma_T = \frac{\pm\ 6.5 \times 10^{-6} \times 4.3 \times 10^{6} \times 45 = \pm\ 740\ PSI}{2(1 - .15)}$$

$$F_p = \frac{19880}{60} = 331\ \text{k/foot}$$

$$\sigma_p = \frac{331 \times 1000}{12 \times 45} = -613\ \text{psi}$$

$\Sigma\sigma_c$ = -740 - 613 = -1353 psi (outside face, compression)

$\Sigma\sigma_T$ = +740 - 613 = + 127 psi (inside face, tension)

The above indicates that under the most critical gradient, the maximum predicted concrete tension of 127 psi is approximately 23 percent of the ultimate concrete tensile strength of 550 psi or $0.1f'_e$. As this condition exists only for a short period of time during the life of the reactor building, reinforcing steel on the inside face of the reactor building walls is not necessary for maintaining the structural integrity of the building.

5.2.1.5.8.2 At End of Service Life – OTSG/RVCH Replacement Opening

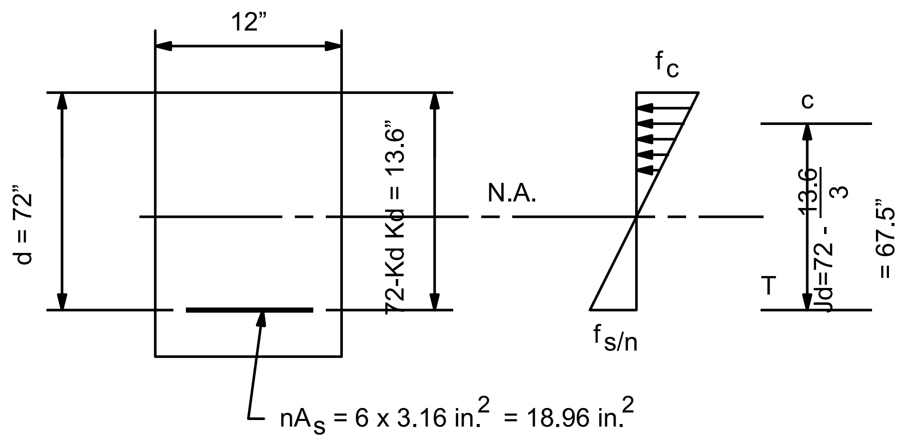
In the vicinity of the OTSG/RVCH replacement opening, the analysis indicated that the maximum predicted flexural tensile stress of concrete exceeded $6\sqrt{f'_c}$. Therefore, reinforcement was added to the inside face of the reactor building concrete wall to ensure that the stresses did not exceed code allowables. The reinforcing is designed assuming a cracked section and considers a coefficient of thermal expansion for the replaced concrete of 4.77×10^{-6} in./in.

5.2.1.5.9 Reactor Building Anchorage Zone Reinforcement Design

5.2.1.5.9.1 The Ring Girder

The concrete and reinforcing steel stresses in the ring girder were examined for all the design loading combinations by the use of the finite element program for an uncracked concrete section. The concrete and reinforcement stresses due to flexure for loadings other than thermal are calculated by the conventional working stress method in accordance with the ACI Code 318-63.

Example



where:

$$(12Kd)\left(\frac{Kd}{2}\right) = 18.96(72 - Kd)$$

$$6Kd^2 + 18.96Kd - 1363 = 0$$

$$Kd = \frac{-18.96 \pm \sqrt{\left(18.96^2 - 4 \times 6(-1363)\right)}}{12}$$

$$Kd = 13.6"$$

$$fs = \frac{M}{A_s\, Jd}$$

$$fs = \frac{200 \times 12}{3.16 \times 67.5} = 11.2 \text{ ksi}$$

Using similar triangles from the stress block and solving for fc.

$$f_c / f_s/n = \frac{Kd}{d - Kd}$$

$$fc = \frac{11.2}{6} \times \frac{13.6}{58.4} = 0.435 \text{ksi}$$

where:

M = Moment Due to Loading Combination = 200 K ft.

T = Tension Force in Reinforced Steel

C = Compression Force in Comp. Block

fs = Steel Stress

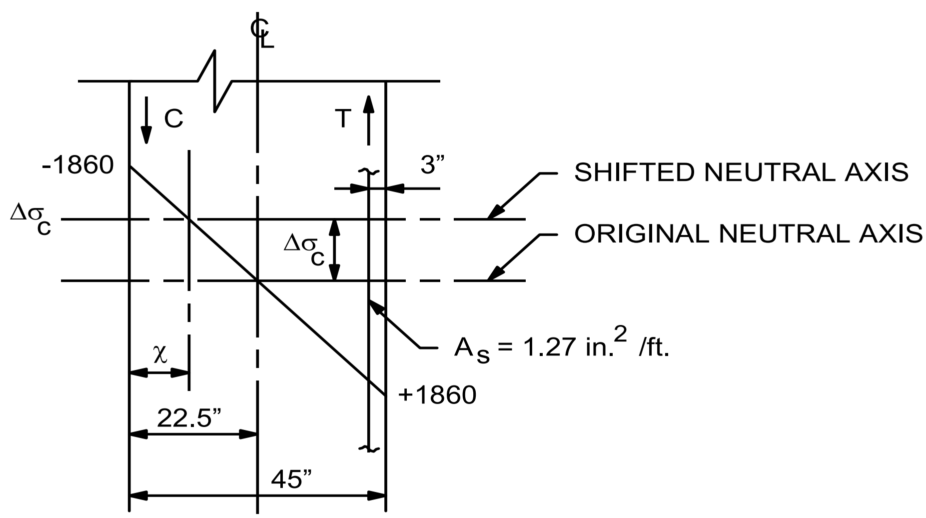
fc = Concrete Compressive Stress

n = Modular Ratio = 6

A_s = Area of Reinforced Steel. = 3.16 in^2

To derive the thermal stresses in concrete and reinforcing steel, the effect of cracking in concrete is simulated by an analysis similar to the example shown below. The example considers the equilibrium of forces in a homogenous section due to a 100 °F thermal gradient:

Example



$$\sigma = \frac{\alpha E_c \Delta T}{2(1 - v)} \quad (1)$$

$$= \frac{6.5 \times 10^{-6} \times 4.3 \times 10^{6} \times 100}{2(1 - .25)} = \pm 1860 \text{ psi}$$

Using similar triangles on the compression side

$$\frac{1860}{22.5} = \frac{1860 - \Delta\sigma_c}{\chi}$$

$$\chi = (1860 - \Delta\sigma_c)\frac{22.5}{1860}$$

(Equation of Equilibrium)

$$\underbrace{\frac{\sigma-\Delta\sigma_c}{2}(\chi)(12'')}_{C} - \underbrace{\text{nAs}\frac{19.5\sigma}{22.5}+\Delta\sigma_c}_{T} = 0 \qquad C - T = C_{net}$$

$$\frac{1860-\Delta\sigma_c}{2}(1860-\Delta\sigma_c)\frac{22.5\times12}{1860}-7\times1.27[1610+\Delta\sigma_c]=0$$

$$251{,}000-270\Delta\sigma_c+.0726\Delta\sigma_c^2-14{,}300-8.9\Delta\sigma_c=0$$

So, $0.0726\Delta\sigma_c^2-279\Delta\sigma_c+236{,}700=0$

$$\Delta\sigma_c=\frac{+279\pm\left[279^2-4\times.0726\times236{,}700\right]^{1/2}}{2\times.0726}$$

$$\Delta\sigma_c=\frac{376}{.1452}=2600>1860\ \text{N.G.}$$

$$\Delta\sigma_c=\frac{182}{.1452}=1250\ \text{psi}$$

$$\text{fT}=\left[\frac{19.25}{22.5}\times1860+1250\right]7$$

$$\text{fT}=20{,}000\ \text{psi}$$

where:

ν = Poisson's Ratio

E_c = Modulus of Elasticity of Concrete

α = Coefficient of Expansion of Conc.

ΔT = Thermal Gradient

C = Compression Force Due to Shifted N.A.

T = Tension Force Due to Shifted N.A.

C_{net} = Net Compression Force on Section

n = Modular Ratio

σ = Thermal Stress of Uncracked Concrete Section

fT = Tensile Stress in Reinforcement

The thermal stresses in the reinforcing steel and concrete are also calculated by using the Chimney Code ACE-505. Local stresses in the reinforcing steel due to the bursting forces caused by the dome and vertical end anchorages are analyzed by using Leonhardt's formula R = .3 F (1-a_1/a) for post-tensioned beams. These stresses are added to the reinforcing steel stresses derived from the governing factored loading combination where the latter reinforcing steel is used to resist the bursting force. Listed below are the maximum predicted stresses for all the groups of reinforcing steel in the anchorage zone of the ring girder.

Dome Tendon Anchorages

Re-Steel Type	Load Condition	Governing Stresses (PSI)
Spiral (5/8"Ø)	Bursting Force	13,200
Hoop (#11)	D + F + 1.5P + T_A	11,000
	Bursting Force	21,000
		32,000
Vertical (#11)	D + F + 1.5P + T_A	15,600
	Bursting Force	26,800
		42,400

Vertical Tendon Anchorages

Re-Steel Type	Load Condition	Governing Stresses (PSI)
Hoop (#11)	D + F + 1.5P + T_A	13,100
	Bursting Force	26,300
		39,400
Radial (#11)	D + F + 1.5P + T_A	11,700
	Bursting Force	30,700
		42,400

(See Section 5.2.1.4.6 for definitions of D, F, P and T_A). The ring girder is checked for equilibrium by postulating some crack patterns as shown in Figure 5-15.

In Figure 5-15, Type I, a diagonal crack, extending from the inside edge of the vertical tendon bearing plate to the top of the dome anchorages, is assumed. The shear friction theory is used to analyze the adequacy of the reinforcing steel in the ring girder. The vertical prestress force is resisted by 16 #11 hoop, 6 #11 diagonals, and 2 #11 radials reinforcing steel.

$$V = .7f_s' = .7 \times 2192^K = 1540\,\text{kips}$$

$$V_u = 1.5 \times 1540 = 2310\,\text{kips}$$

$$\upsilon_u = \frac{V_u}{bd} = \frac{2310^K}{24 \times 132} = 730\,\text{psi}$$

$$730\,\text{psi} < .2f'_c = 1100\,\text{psi}$$

$$730\,\text{psi} < 800\,\text{psi}$$

$$A_s = 24 \times 1.56\,\text{in}^2 = 37.4\,\text{in}^2$$

$$f'_s = \frac{Vu}{\theta \mu A_s} = \frac{2310\,\text{kips}}{.85 \times 1.4 \times 37.4} = 52{,}000\,\text{psi}$$

where:

V = Applied Prestress Force

V_u = Factored Load of Prestress Force

b = Width of Crack = 24"

d = Depth of Crack = 132"

υ_u = Nominal Design Shear Stress

f'_s = Ultimate Strength of Tendon

f'_c = Concrete Strength

f_s = Reinforcement Stress

μ = Coefficient of Friction

θ = Capacity Reduction Factor

A_s = Area of Reinforcing Steel

The 24 #11 reinforcing bars resist the factored load of 2,310 kips with a predicted stress of 52,000 psi.

The allowable stresses and the ultimate strength of the concrete blocks are derived from the shear friction section of Chapter 11, Revision of ACI 318-63 building code requirements for reinforced concrete published in the Journal of the ACI, February 1970. The new section in the proposed code came as a result of extensive tests by Hofbeck, Ibrahim, and Mattock who were more interested in the direct shear failure as opposed to diagonal tension (see Reference 5).

In Figure 5-15, Type II, the simulated cracks extend from the bottom edges of the dome tendon bearing plates to the vertical tendon anchorages. This postulated condition assumes the top of the ring girder corner to be displaced by the wedging action of the dome anchorages. Assuming that this wedging force is equivalent to the bursting force, an upper limit of 383 kips is predicted. The wedging force is resisted by 5 #11 hoops and 3 #11 diagonals reinforcing steel (A_s = 12.48 in^2) with a predicted stress of 30,700 psi.

5.2.1.5.9.2 The Buttress

The concrete and reinforcing steel stresses in the buttress are examined for all the design loading combinations. The method used for the buttress design is similar to that for the ring girder. All axisymmetric loadings are analyzed by using the finite element computer program. Local stresses caused by the end anchors of the hoop tendons are derived from Leonhardt's formula for post-tensioned beams.

The following table summarizes the predicted stresses in the reinforcing steel.

Hoop Tendon Anchorage

Re-Steel Type	Load Condition	Governing Stresses (PSI)
Hoop (#10)	$D + F + 1.5P + T_A$	19,700
Vertical (#11)	$D + F + 1.5P + T_A$	16,300
	Bursting Force	30,800
		47,100
Radial (#8)	Bursting Force	33,700

To verify the adequacy of the anchorage zone, a cracked buttress is assumed and a separate analysis is made assuming a wedge shaped failure mechanism such as has been observed in concrete cylinder tests by Taylor. Figure 5-16 shows some postulated crack patterns on the outside face of the buttress.

Type II of Figure 5-16 shows the postulated crack patterns inside the buttress. As the force F is applied on the central wedge, it has the tendency to create forces perpendicular to the applied force. The forces may be considered as the bursting forces. The reinforcing steel confines all the outer blocks. Section A, as shown in Figure 5-16, provides one view of the simple system formed by the postulated cracks.

The equilibrium equations below establish the magnitude of the forces perpendicular to the applied force.

For the equilibrium of the system:

$$\Sigma_{FX} = 0$$

$$.25\,F = N\sin\theta + f\cos\theta$$

$$.25\,F = N(\sin\theta + \mu\cos\theta) \qquad \text{(EQUATION 1)}$$

$$\Sigma_{FY} = 0$$

$$R = N\cos\theta - f\sin\theta$$

$$R = N(\cos\theta - \mu\sin\theta) \qquad \text{(EQUATION 2)}$$

SOLVING FOR N IN EQUATION 1 AND SUBSTITUTING INTO EQUATION 2

$$R = .25\,F\,\frac{(\cos\theta - \mu\sin\theta)}{(\sin\theta + \mu\cos\theta)} \qquad \text{(EQUATION 3)}$$

R = BURSTING FORCE

F/4 = APPLIED PRESTRESS FORCE IN QUARTER SECTION

F = μN = FRICTIONAL FORCE

N = NORMAL FORCE EXERTED BY THE WEDGE

μ = COEFFICIENT OF FRICTION

θ = INCLUDED ANGLE OF THE POSTULATED FAILURE PLANE

The same solution results if a similar solution is performed for Type I of Figure 5-16.

Equation 3 shows that the bursting force (R) is a function of the coefficient of friction (μ), the inclined angle (θ) of the postulated failure, and the applied prestress force. The expected values of $\mu = .75$ and $\theta = 45°$ results in a calculated bursting force of .0348 F (55 kips). It is expected that the actual value is less than 55 kips since the coefficient of friction is likely to be larger than .75 for the rough failure surfaces observed in the tests mentioned. As a very conservative limit, however, values of $\mu = .50$ and $\theta = 30°$ are assumed. This produces a bursting force of .165 F (252 kips). The upper limit value of 252 kips is less than the predicted bursting forces calculated by the Leonhardt's formula for post-tensioned beams.

5.2.1.5.9.3 The Base Slab

The concrete and reinforcing steel in the base slab are designed to resist all design load combinations.

The predicted concrete and reinforcing steel stresses are derived by methods shown in Section 5.2.1.5.9.1. Local reinforcing steel stresses caused by the bursting forces due to the bottom vertical tendon end anchorages are calculated from Leonhardt's formula.

Following is the maximum predicted stress for the bottom reinforcing steel in the base slab.

Base Slab Anchorages

Re-Steel Type	Load Condition	Governing Stresses (PSI)
Radial (#18)	$D + F + T_A + E' + H + R$	28,500
	Bursting Force	17,400
		45,900

5.2.2 DESIGN, CONSTRUCTION, AND TESTING OF PENETRATIONS

5.2.2.1 Types of Penetrations

All penetrations are pressure resistant, leak tight, welded assemblies designed, fabricated, and tested in accordance with the ASME Nuclear Vessel Code, Section III, Class B vessels.

5.2.2.1.1 Electrical Penetrations

Canister type penetrations are used for all electrical conductors passing through the reactor building. The penetration canister is a hollow cylinder closed on both ends with an insulated bushing or connector assembly. This canister is provided with a test plug to allow test pressurization of the penetration assembly between the header plates. All materials of completed penetration assemblies are self-fire extinguishing in accordance with ASTM-D35. Electrical penetration assemblies are qualified to withstand a LOCA environment as documented by the Environmental Qualification Program and defined in Engineering Standard NES-01, Environmental Qualification Program. Subsequently, electrical penetration assemblies pass a leak test.

The method used to seal the joint between the canister end plates and the conductor depended upon the type of cable and connector assembly involved. In general, there are three types used:

A. Type 1 - High voltage power, 4160 volts and 6900 volts

B. Type 2 - Power and control, 600 volts and below

C. Type 3 - Instrumentation, thermocouple leads, coaxial, and other special wires.

Type 1 penetrations are high voltage insulated copper rods with copper braid. These insulated rods are connected between the insulated bushings at each end plate of the canister. High voltage insulating bushings and seals may be used to provide the barrier.

Type 2 penetrations are single or multiconductor insulated cable. This cable is connected between the connector assembly or terminal blocks at each end of the penetration canister. A hermetically sealed contact assembly which is sealed in the canister end plates is used.

Type 3 penetrations are the same as Type 2 except that the conductors are thermocouple material, coaxial cable, or special wires. The sealing methods are the same as for Type 2 penetrations. Cables are connected at each end of the penetration canister between the connector assemblies.

5.2.2.1.2 Piping Penetrations

Single barrier piping penetrations are provided for all piping passing through the reactor building exterior wall. The closure of the pipe to the liner plate is accomplished with pipe caps and flued heads welded to the pipe and to the liner plate reinforcement.

In the case of piping carrying hot fluid, the pipe is insulated. The modes of isolating the piping penetrations are covered in Section 5.2.5.

The anchorages of penetration closure connecting pipes to the reactor building wall are designed as Seismic Class 1 structures to resist all forces and moments caused by a postulated pipe rupture, and thermal and seismic loads. The design conditions include the maximum pipe reactions and pipe rupture forces.

The following criteria are for typical piping penetrations to insure the integrity of the liner penetration junction at the piping.

A. The penetration assembly consisting of pipe cap and flued head and pipe sleeve section and the assembly welds and welds to the liner plate are full penetration welds. The assembly is anchored into the wall concrete and designed to accommodate all forces and moments due to pipe rupture and thermal expansion.

B. The design criterion is that the pipe penetration is the strongest point in the system when a pipe break is postulated. Pipe stops, increased pipe thickness, or other means are used to attain this. Part of this criterion is also that the operation of closure valves is not impaired by any postulated pipe break.

5.2.2.1.3 Equipment and Personnel Access Hatches

An equipment hatch 15 feet in diameter is provided. The hatch is fabricated from welded steel and furnished with a double gasketed flange and bolted, dished door. Equipment up to and including the size of the reactor vessel O-ring seal can be transferred into and out of the reactor building through this hatch.

During outages, a temporary equipment hatch cover may be used in lieu of the permanent hatch cover. The temporary cover is designed to meet the requirements for reactor building closure during cold shutdown and refueling conditions.

Two personnel locks are provided on opposite sides of the reactor building. One of these is for emergency access only. Each personnel lock is a double door, welded steel assembly. A quick-acting type, equalizing valve connects the personnel lock with the interior and exterior of the reactor building for the purpose of equalizing pressure in the two systems when entering or leaving the reactor building.

The two doors in each personnel lock are interlocked to avoid the possibility of both being opened simultaneously and to insure that one door is completely closed before the opposite door can be opened. Provisions have been made to permit bypassing the door interlocking system to allow doors to be left open during plant cold shutdown. Each door lock hinge is capable of independent, three-dimensional adjustment to assist in proper seating. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior.

To permit testing of the personnel and escape locks at full accident pressure, a test connection is installed in the outer bulkhead of the locks. In addition, a test connection incorporated into the construction of each door is designed to allow testing of the seals at 68 psig (1.15 design pressure). Routine periodic testing is performed in accordance with requirements of 10 CFR 50, Appendix J at a minimum pressure of 54 psig (Pa) except for air lock door seals, which may be tested at the lower pressure stated in the Technical Specifications. Plant preheatup containment integrity checklists include verification of isolation of local leak rate connections. (1CAN118018)

The seismic design of the equipment hatch, personnel and escape locks is performed as described in Section 5.1.4.2.1. A conservative value of 0.5 percent of critical damping was used. The maximum differential deflection at the reactor building equipment hatch due to the "maximum earthquake" is 0.041 inch. This differential deflection will not interfere with the proper operation and sealing of the reactor building equipment hatch.

5.2.2.1.4 Special Penetrations

A. Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor building and the spent fuel pit in the auxiliary building. The penetration consists of a 30-inch stainless steel pipe installed inside a casing pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange with a pressure test connection. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for test pressurizing all welds essential to the integrity of the penetration during plant operation.

Expansion joint bellows at the fuel transfer tube provide for the relative movement between the reactor building, internals, and the auxiliary building. The bellows consider all loading conditions including the combination of maximum earthquake and maximum hydraulic pressure. The design of the expansion joint bellows considered the maximum computed, relative axial, and lateral displacement of the fuel transfer tube occurring simultaneously. The selected type of bellows for these design conditions was certified by the manufacturer. Access for inspection and maintenance of the bellows is provided through the refueling canal and the fuel tilt pit when empty.

B. Reactor Building Supply and Exhaust Purge Ducts

The ventilation system purge duct is equipped with two valves to be used for isolation purposes. The valves are remotely operated for reactor building purging as described in Chapter 9.

5.2.2.2 Design of Penetrations

5.2.2.2.1 Design Criteria

Penetrations conform to the applicable sections of ANSI N6.2-1965, "Safety Standard for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors." The penetration is considered as an anchorage; therefore, no provisions are made for expansion. All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME Section III, Nuclear Vessels Code.

Each line which penetrates the reactor building and contains high pressure and high temperature fluids (main steam and feedwater) has structural restraints such that the fracture of any one line will not jeopardize any other line whose fracture would then constitute a LOCA. Safeguard electrical penetrations are protected or separated such that their functions are not impaired by the whipping of any pipe.

Further protection of each line, necessary to preclude pipe rupture between penetration and first valve, is accomplished by shortening the exposed length of pipe and installing the first valve as close as possible to the internal or external wall of the reactor building, depending upon valve operating and maintenance clearances. Criteria which apply to the provision of automatic and manual isolation valves in the penetration lines are contained in Section 5.2.5.

To insure the safe transfer of membrane and bending stresses into the reactor building wall, the penetrations are, in general, designed as pressure vessels; however, the sizes and thicknesses of the penetration nozzles are generally governed by pipe rupture loads. The penetration nozzles are designed to resist elastically the ultimate capacities of the applicable pipes. The following example shows the determination of a typical penetration nozzle.

$$Mu_p = fu_p\ Sp_p = fu_p\ 4R_p^2t_p$$

$$Me_n = .9fy_n\ Se_n = .9fy_n\ \Pi R_n^2t_n$$

Setting equations for "Me_n" and "Mu_p" equal and solve for "t_n",

$$t_n = \frac{fu_p\ 4R_p^2 t_p}{.9fy_n \Pi R_n^2}$$

where:

Mu_p = Ultimate moment capacity of pipe

fu_p = Guaranteed ultimate stress of pipe

Sp_p = Plastic section modulus

Me_n = Yield moment of nozzle

fy_n = Guaranteed yield stress of nozzle

Se_n = Elastic section modulus of nozzle

t_p = pipe thickness

t_n = nozzle thickness

R_p = Mean radius of pipe

R_n = Mean radius of nozzle.

Based on this design, a conservative factor of safety is achieved against the nozzle failure due to pipe rupture. The nozzle has also been investigated for the LOCA. For this condition, the pressure and temperature loads resulting from LOCA are applied to the nozzle. The primary stresses resulting from the pressure are negligible relative to the pipe rupture loads. Thermal stresses are considered to be secondary stresses due to the self-limiting nature of this condition. For this reason, the ASME Code has assigned allowable thermal stress values of 80,000 psi. This value is conservatively based on approximately 1,000 cycles of loads. The maximum predicted thermal stress in the nozzle, based on a differential temperature of 100 °F, is 35,400 psi.

Based on the above analyses of the penetration nozzles, the nozzles are predicted to safely transfer the membrane and bending stresses into the rest of the structure.

5.2.2.2.2 Design of High Temperature Penetrations

The main high temperature piping consists of two penetrations for feedwater and two penetrations for main steam which have a maximum operating temperature of 465 °F and 600 °F, respectively. Thermal insulation has been provided on the outside of each line. The insulation is designed to restrict maximum temperature rise in the concrete to 200 °F.

Under certain conditions, the 4-inch diameter steam generator drainline may be used as a temporary blowdown line during initial plant startup and/or during subsequent startups following refueling shutdowns. Under this mode of operation, the steady state temperature of the concrete adjacent to the drain line penetration will reach 200 °F when the fluid temperature reaches 350 °F and may go up to a maximum of 250 °F when the fluid reaches its maximum anticipated temperature of 556 °F.

This temperature is limited to a very short time period (considerably less than one percent of the lifetime of the plant) and consequently its effect on the concrete will be negligible.

The basis for limiting strains in the penetration steel is in accordance with the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4, 1965 and, therefore, the penetration structural and leak tightness integrity is maintained. Local heating of the concrete immediately around the penetration develops compressive stress in the concrete adjacent to the penetration and a negligible amount of tensile stress over a large area. The mild steel reinforcing added around penetrations distributes local compressive stresses for overall structural integrity.

5.2.2.2.3 Penetration Materials

The materials for penetrations including the personnel and equipment access hatches together with the mechanical and electrical penetrations are carbon steel and conform to the requirements of the ASME Nuclear Vessel Code. As required by the Nuclear Vessel Code, the penetration materials meet the necessary Charpy V-notch impact values at temperatures 30 °F below the lowest service temperature.

A. Piping Penetration Materials

Materials and corresponding specifications are listed below:

Piping Penetration Material	Specification
Penetration Sleeve	ASTM-A333
Penetration Reinforcing Rings	ASTM-A516 made to ASTM-A300
Penetration Sleeve Reinforcing	ASTM-A516 made to ASTM-A300
Bar Anchoring Rings and plates	ASTM-A36
Rolled Shapes	ASTM-A36

B. Electrical Penetration Materials

The penetration sleeves used to accommodate the electrical penetration assembly canisters are 12-inch I.D. and 32-inch I.D., Schedule 80, carbon steel pipe, except where otherwise noted.

C. Access Penetration Materials

Materials of pressure retaining parts and corresponding specifications are listed below:

Access Penetration Material	Material Specification
Equipment Hatch Insert	ASME SA-516 Grade 70 made to SA300
Equipment Hatch Flanges	ASME SA-516 Grade 70 made to SA300
Equipment Hatch Head	ASME SA-516 Grade 70 made to SA300
Personnel Locks	ASME SA-516 Grade 70 made to SA300

5.2.2.3 Installation of Penetrations

The qualification of welding procedures and welders are in accordance with Section IX, "Welding Qualification," of the ASME Boiler and Pressure Vessel Code. The repair of defective welds is in accordance with Paragraph N-528 of Section III, "Nuclear Vessels" of the above Code.

5.2.2.4 Testability of Penetrations

Some lines which penetrate the reactor building are not open to the reactor building atmosphere. Where these lines are located outside the missile barrier, they are considered to be closed systems, not subject to rupture following a LOCA. The main steam lines, feedwater lines, and service water lines which provide cooling water to the coolers in the ventilation air handling units fall within this category. Since the containment barrier integrity is not breached during LOCA conditions, these penetrations are not subject to leak rate testing.

Any leakage through these closed lines is detected as part of the integrated leak test of the reactor building or normal monitoring of the reactor building sump fill rate.

Steam generator secondary side drain and sample lines are also considered closed systems inside the reactor building. Portions of these lines are less than 2 1/2" and therefore the seismic design requirements of GDC 2 do not apply to these sections as detailed in Regulatory Guide 1.29.

Some lines which penetrate the reactor building are part of the closed piping systems located outside the reactor building. Containment atmospheric leakage through lines of this type is not possible and thus leak rate testing is not required. Typical lines falling within this category are:

A. Reactor building spray.

B. The Reactor Coolant Pump seal water supply.

C. The decay heat removal loop inlet and outlet.

D. High pressure injection.

E. ESF suction from the Reactor Building sump.

Chapter 6 discusses, in detail, the above systems.

5.2.2.4.1 Provisions for Isolation Valves

The reactor building isolation valve arrangements comply with the principal architectural and engineering criteria in effect at the time the plant was designed (See PSAR Section 1.4 and Supplement No. 1). In accordance with these criteria the use of a check valve external to the reactor building is acceptable in a line that is not directly connected to the RCS or the reactor building atmosphere and through which the direction of flow is into the reactor building.

Isolation valving for all containment penetrations satisfies General Design Criteria 54, 55, 56, and 57 with the following exceptions:

Penetration	System	Comments
P-1	Main Steam	This penetration meets the requirements of GDC-57 with the exception of HV-158 and MS-29A, which are maintained in the open position during power operations. These valves provide a path to a steam trap and closing the valves during power operation potentially challenges the operability of the steam driven EFW pump and the atmospheric dump valve (ADV). The NRC has approved this configuration per 0CNA040508.
P-2	Main Steam	This penetration meets the requirements of GDC-57 with the exception of HV-157, which is maintained in the open position during power operations. This valve provides a path to a steam trap and closing it during power operation potentially challenges the operability of the steam driven EFW pump. The NRC has approved this configuration per 0CNA040508.
P-5 & P-23	RB Spray	The Reactor Building Spray lines do not explicitly meet Criterion 56. The design is acceptable on the basis that the system is used post LOCA to mitigate the consequences of an accident. The check valve inside the Reactor Building is automatic and the MOV outside the Reactor Building is capable of remote manual operation.
P-7	Pressurizer	The pressurizer and RCS hot leg sample line does not explicitly meet Criterion 55. The design is acceptable on the basis that the system is used infrequently and the remotely operated valves inside and outside the Reactor Building are normally closed. In addition to the physical design and infrequent use, administrative control is exercised by requiring the sampling station operator to request the control room operator to open the inner and outer Reactor Building isolation valve.

Penetration	System	Comments
P-8, P-13, P-15, & P-34	HPI	The HPI lines do not explicitly meet Criterion 55. The design is acceptable on the basis that the system is used post LOCA to mitigate the consequences of an accident. The check valve inside the Reactor Building is automatic and the MOV's outside the Reactor Building are cable of remote manual operation. The makeup isolation MOV's on P-34 are automatic.
P-10	Steam Generator Sample	The steam generator sample lines do not explicitly meet Criterion 57 when the system is in use. When the system is in use, the design is acceptable on the basis that the system is infrequently used, the piping is closed inside the Reactor Building, and the manual valve outside the Reactor Building is normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure the manual isolation valve will be closed in a timely manner. The closed system inside the reactor building associated with this penetration is not seismically qualified. However, this is consistent with Regulatory Guide 1.29, which does not require seismic qualification of secondary system piping less than 2½".
P-12	Core Flooding Tank Sample	The Core Flooding Tank sample lines do not explicitly meet Criterion 56. The design is acceptable on the basis that the system is used infrequently and both the remotely operated valves inside the Reactor Building and the manual valve outside the Reactor Building are normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure that the manual isolation valve will be closed in a timely manner.
P-14	Makeup and Purification	This penetration supports the normal operation of the reactor coolant letdown to the makeup and purification system. The penetration does not explicitly meet the requirements of GDC 55 when it is in use. This is acceptable since the penetration is also required post-accident to sample the RCS due to the elimination of PASS sampling requirements. The valves which may be opened for RCS sampling are capable of remote manual closure.
P-16	RCP Seal Supply	The RCP seal supply line does not explicitly meet Criterion 55. The design is acceptable on the basis that the system is used post LOCA to support continued RCP operation. The check valves inside the reactor building are automatic and the MOV outside the Reactor Building is capable of remote manual operation.

Penetration	System	Comments
P-24A & B	RB Hydrogen Sampling, APM	These lines do not explicitly meet Criterion 56. The design is acceptable on the basis that these systems are used post LOCA. The two inside MOV's and two outside SOV's which are open post LOCA for hydrogen sampling use are capable of remote manual operation. PASS piping for SOV's SV-7510 and SV-7512 has been permanently capped.
P-26 & P-36	LPI	The LPI lines do not explicitly meet Criterion 55. The design is acceptable on the basis that the system is used post LOCA to mitigate the consequences of an accident. The check valves inside the Reactor Building are automatic and the MOV outside the Reactor Building is capable of remote manual operation.
P-27	DHR Letdown	The DHR letdown line does not explicitly meet Criterion 55. The design is acceptable on the basis that the MOV's inside and outside the Reactor Building are normally closed during power operations. The MOV inside and the MOV outside the Reactor Building are capable of remote manual closure.
P-31, P-32, & P-41	CFT Fill and Makeup/Nitrogen Supply	The CFT Fill and Makeup/Nitrogen Supply lines do not explicitly meet Criterion 56 when the supply systems are in use. The design is acceptable on the basis that the system is used infrequently and both the check valve inside and the manual valves outside the Reactor Building are normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure the manual isolation valves are closed in a timely manner.
P-33	Auxiliary Pressurizer Spray	The Pressurizer auxiliary spray line does not explicitly meet Criterion 55 when in use. The design is acceptable on the basis that the system is infrequently used. The check valve inside the Reactor Building is automatic and the MOV outside the Reactor Building is capable of remote manual operation.
P-43 & P-46	Breathing Air Instrument Air	The breathing and instrument air lines do not explicitly meet Criterion 56 when the systems are in use. The design is acceptable on the basis that each system is used infrequently and both the manual valve inside and the manual valves outside the Reactor Building are normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure the manual isolation valves are closed in a timely manner.
P-53	RB Hydrogen Sampling	These lines do not explicitly meet Criterion 56. The design is acceptable on the basis that this system is used post LOCA. The two inside MOV's and two outside SOV's which are open post LOCA for hydrogen sampling use are capable of remote manual operation.

Penetration	System	Comments
P-58 & P-64	Steam Generator Drains	The steam generator drain lines do not explicitly meet Criterion 56 when the system is in use. When the system is in use, the design is acceptable on the basis that the system is infrequently used, and the inboard manual isolation valve and the manual valve outside the Reactor Building are normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure the manual isolation valves will be closed in a timely manner.
P-66 & P-67	RB Sump to ES Pumps	These lines do not explicitly meet Criterion 56. The design is acceptable on the basis that the system is used post LOCA to mitigate the consequences of an accident. The MOV's inside and outside the Reactor Building and the SOV outside the Reactor Building are capable of remote manual operation. PASS piping for SOV's SV-1440 and SV-1443 has been permanently capped.
P-68	Reactor Building Sump Drain	The Reactor Building sump drain and sample lines do not explicitly meet Criterion 56 during sampling operations. The design is acceptable on the basis the system is used infrequently and both the inboard automatic motor operated isolation valve and the outboard manual globe isolation valve are normally closed. In addition to the physical design and infrequent use, administrative control is exercised to insure the manual isolation valve is closed in a timely manner.

The emergency closing time of the main steam isolation valves is set at approximately five seconds and, consequently, no significant shockwave formation is expected to result from their closure. The turbine stop valves, however, are set to close at a rate much faster than the MSIVs and, as a result, the main steam piping system has been analyzed for the effect of the shockwave, associated with the fast closure of these valves. The piping supports are designed to withstand the dynamic forces associated with this shockwave, while the main steam isolation valves are designed to withstand the resultant pressure from normal operation plus dynamic shock loading. The stresses of pipe at valve connections due to dynamic effect are within ANSI B31.7 code allowable.

5.2.3 CONSTRUCTION PRACTICES AND QUALITY ASSURANCE

5.2.3.1 Applicable Construction Codes

The following codes of practice are used to establish standards of construction procedure:

- ACI 301 - Specification for Structural Concrete for Buildings
- ACI 318 - Building Code Requirements for Reinforced Concrete
- ACI 347 - Recommended Practice for Concrete Formwork
- ACI 605 - Recommended Practice for Hot Weather Concreting
- ACI 613 - Recommended Practice for Selecting Proportions for Concrete
- ACI 614 - Recommended Practice for Measuring, Mixing and Placing Concrete

- ACI 615 - Manual of Standard Practice for Detailing Reinforced Concrete Structures
- Part UW - Requirements for Unfired Pressure Vessels Fabricated by Welding of Section III of the ASME Boiler and Pressure Vessel Code.
- AISC - Steel Manual, Code of Standard Practice
- ACI - Inspection Manual
- AWS - Code for Welding in Building Construction (D 1.0-66)

The following additional codes were used during the OTSG/RVCH replacement:

- AWS - Structural Welding Code (D 1.1-92)
- AWS - Structural Welding Code – (Reinforcing Steel) (D1.4-98)
- ASME - Section III, Division 2, 1989 Edition, No Addenda
  - Section VIII, 1989 Edition, No Addenda
  - Section XI, Subsections IWA, IWE, IWL, 1992 Edition, 1992 Addenda

In every instance, the construction procedures for the reactor building were equal to or exceeded the recommendations set forth in the foregoing publications. The extent to which each detailed process exceeded standard requirements cannot be described without incorporating all applicable job specifications. In general, however, whenever the applicable codes specify minimum and ideal criteria, the ideal is incorporated into the specifications.

5.2.3.2 Quality Assurance Program

A formal quality control organization and reporting system was employed to assure that critical structures were built in accordance with the specifications. This is generally explained in Section 1.6.

The senior engineers of the engineering design groups and engineering specialists such as the supervising materials engineer and senior metallurgical engineers were at the site during critical periods of construction. They were also in constant contact with the QA engineer.

Bechtel's rotational program assigned responsible field personnel to the engineering office to gain experience in the design of facilities. This procedure developed field engineers who were well rounded in design and construction practices and were well qualified to perform inspection assignments. Moreover, they had at their command Bechtel's technical specialists who were called upon to support their organization. Among these were included the soils mechanics and geology, concrete, metallurgical, and welding sections.

Independent testing laboratories were used for quality control testing and reporting of the concrete materials.

5.2.3.3 Construction Materials Inspection and Installation

Materials used in the reactor building include concrete, reinforcing steel, prestressing system materials, and steel liner plate. The inspection and testing of each material are as follows.

5.2.3.3.1 Concrete Materials

A. Cement

In addition to the tests required by the cement manufacturers, the following tests are performed:

ASTM C 114 - Chemical Analysis

ASTM C 115 - Fineness of Portland Cement (Turbidimeter)

or

ASTM C 204 - Fineness of Portland Cement (Air Permeability)

ASTM C 151 - Autoclave Expansion

ASTM C 191 - Time of Set (Vicat Needle)

or

ASTM C 266 - Time of Set (Gillmore Needle)

ASTM C 109 - Compressive Strength

ASTM C 190 - Tensile Strength

The purpose of the above tests is to ascertain conformance with ASTM Specification C150. In addition, tests ASTM C 191 or ASTM C 266 and ASTM C 109 are repeated periodically during construction to check storage environmental effects on cement characteristics. The tests supplement visual inspection of material storage procedures.

Cement for the OTSG/RVCH replacement concrete is Type III Portland cement. The cement was manufactured by Holcim at their Artesia, Mississippi plant conforming to ASTM C150. Qualification testing was performed by an independent testing laboratory to verify the cement conformed to standard physical and chemical requirements of ASTM C150 for Type III Portland cement.

B. Fly Ash

Chemical tests of fly ash were performed in accordance with the ASTM C311-68. A typical chemical analysis of fly ash used in concrete work is presented below. Sulfates and chlorines are not present and sulfur trioxide (SO_3) content is well within the limit set by the ASTM C 618-68T.

Typical Chemical Analysis of Fly Ash Used

Silicon Dioxide (SiO_2), percent	46.9%
Iron & Aluminum Oxides, R_2O_3, percent	34.3%
Magnesium Oxide, MgO, percent	0.12%
Sulfur Trioxide, SO_3, percent	0.25%
Available Alkalies, percent	0.32%
Loss on Ignition	0.09%
Moisture content, percent	0.05%

No fly ash was used in the replacement concrete for the OTSG/RVCH replacement construction opening.

C. Admixtures

A concrete testing laboratory was engaged to perform the necessary strength and shrinkage tests on various water reducing agents to establish the particular additive with the most desirable characteristics for this application.

For the OTSG/RVCH replacement concrete, an air-entraining admixture was added conforming to ASTM C260. Additionally, water reducing, retarding, and accelerating admixtures were added to the concrete conforming to ASTM C494.

D. Aggregates

Tests of concrete aggregate include the following:

C 131	Los Angeles Abrasion	ASTM Spec C-33	To conform with specification
C 142	Clay Lumps and Friable Particles in Aggregate	ASTM Spec C-33	To conform with specification
C 117	Material Finer than No. 200 Sieve	ASTM Spec C-33	To conform with specification
C 40	Organic Impurities	ASTM Spec C-33	To conform with specification
C 289	Potential Reactivity (Chemical)	ASTM Spec C-33	To conform with specification
C 136	Sieve Analysis	ASTM Spec C-33	To conform with specification
C 88	Soundness	ASTM Spec C-33	To conform with specification
C 127	Specific Gravity and Absorption	ASTM Spec C-33	Mix Design Calculations
C 128	Specific Gravity and Absorption	ASTM Spec C-33	Mix Design Calculations
C 295	Petrographic	ASTM Spec C-33	To conform withspecification

In addition to the above mentioned tests, a daily inspection control program was carried out during construction to verify consistency in potentially variable characteristics such as graduation and organic content.

Aggregate for the OTSG/RVCH replacement concrete was tested per the above standards except that Potential Reactivity was tested per ASTM C1260 rather than ASTM C289. In addition, aggregates were tested for “moisture content” per ASTM C566 and for “lightweight pieces-coal and lignite” per ASTM C123.

E. Water

Water used in concrete mixing was sampled and analyzed by a qualified testing laboratory to assure conformance with specifications.

F. Expansive Component

An expansive additive was added to the concrete for the OTSG/RVCH replacement to ensure that the concrete would have no shrinkage or be slightly expansive. The expansive additive is Komponent by CTS Cement Manufacturing Corporation and conforms to ASTM C845, Type K.

5.2.3.3.2 Concrete

A. Design Mix

Design mixes and the associated tests were provided by a qualified concrete testing laboratory. The design of mixes were in accordance with ACI 613 to obtain material proportions for the specified concrete. During construction, the field inspection personnel made any minor modifications that were necessitated by variations in aggregate gradation or moisture content.

B. Compressive Strength

Concrete strength, slump, and temperature inspections are performed. The purposes of the tests and inspections are to verify conformance to specifications. The basis for the proposed inspection procedure is ACI Manual of Concrete Inspection with upgraded modifications to meet the more stringent requirements of this application.

Concrete slump test samples were taken at the discharge of the batch plant stationary mixer and at points of placement. Concrete compression test samples during construction were taken at the discharge of the batch plant stationary mixer. Additional correlation cylinders were taken at the point of placement as required.

5.2.3.3.2.1 Concrete – OTSG/RVCH Replacement

The concrete mix for the OTSG/RVCH replacement was developed by an independent testing laboratory to meet specific physical and mechanical properties. The developed mix was tested for meeting the requirements per the referenced ASTM Standards. A pumpable mix was developed with proportions such that the concrete is plastic and workable and can be placed without segregation of aggregates and without honeycombing. The developed mix meets the following physical and mechanical properties:

1. Restrained volume change (shrinkage) zero or slightly positive at 3 days (ASTM C878)
2. 72-hour cylinder strength of 5,500 psi or greater (ASTM C39).
3. Density of at least 148 pounds per cubic foot (ASTM C138)
4. Coefficient of thermal expansion of 4.77 x 10^{-6} in./in./°F at an age of 28 days (ASTM C531-as a guide only).
5. Poisson’s ratio of 0.20 or less at an age of 3 days (ASTM C469).

6. Young's Modulus of Elasticity of $4.0x10^6$psi or greater at an age of 3 days (ASTM C469).
7. Design slump of 8 inches +/- 1.5 inches at the point of placement (ASTM C143).
8. Air content of 2.5% to 5% at the point of placement (ASTM C231).

5.2.3.3.3 Reinforcing Steel

5.2.3.3.3.1 Material

All reinforcing steel was tested in accordance with ASTM specifications. Tests included one tension and one bend test per heat or per mill shipment, whichever was less, for each diameter bar. High strength bars were clearly identified prior to shipment. Additionally, user's tests have also been conducted.

5.2.3.3.3.2 Mechanical Splices

5.2.3.3.3.2.1 Specification for Splicing Reinforcing Bar Using The Cadweld Process

5.2.3.3.3.2.1.1 Scope

This specification covers the mechanical splicing of deformed concrete reinforcing bar for full tensile loading. The average tensile strength of the Cadweld joints was equal to or greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in the appropriate ASTM standard. The minimum tensile strength of the splices was equal to or exceeded 125 percent of the minimum yield strength for each grade of reinforcing steel, as specified in the appropriate ASTM standard.

5.2.3.3.3.2.1.2 Process

All splices were made by the Cadweld process using clamping devices, sleeves, charges, and so on, as specified by the Cadweld instruction sheets for "B" and "T" series connections. "C" series materials were not permitted.

5.2.3.3.3.2.1.3 Qualification of Operators

Prior to the production splicing of reinforcing bars, each operator or crew, including the foreman or supervisor for that crew, prepared and tested a joint for each of the positions to be used in production work. These splices were made and tested in strict accordance with this specification using the same ASTM grade and size of bar to be spliced in the production work. To qualify, the completed splices met the acceptance standards of Section 5.2.3.3.3.2.1.6 for workmanship, visual quality, and minimum tensile strength. A list containing the names of qualified operators and their qualification test results was maintained at the jobsite.

5.2.3.3.3.2.1.4 Procedure Specification

All joints were made in accordance with the manufacturer's instruction sheets, "Rebar Instructions for Vertical Column Joints," plus the following additional requirements:

A. A manufacturer's representative, experienced in Cadweld splicing of reinforcing bar, shall be present at jobsite at the outset of the work to demonstrate the equipment and techniques used for making quality splices. He shall also be present for at least the first 50 production splices to observe and verify that the equipment is being used correctly and that quality splices are being obtained.

B. The splice sleeves, exothermic powder, and graphite molds shall be stored in a clear dry area with adequate protection from the elements to prevent absorption of moisture.

C. Each splice sleeve shall be visually examined immediately prior to use to insure the absence of rust and other foreign material on the inside surface.

D. The graphite molds shall be preheated with an oxyacetylene or propane torch to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

E. Bar ends to be spliced shall be in good condition with full size, undamaged deformations. The bar ends shall be power brushed or sand blasted to remove all loose mill scale, rust, concrete, and other foreign material. Prior to power brushing, all water, grease, and paint shall be removed by heating the bar ends with an oxyacetylene or propane torch.

F. A permanent line shall be marked 12 inches back from the end of each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

G. Immediately before the splice sleeve is placed into final position, the previously cleaned bar ends shall be preheated with an oxyacetylene or propane torch to insure complete absence of moisture.

H. Special attention shall be given to maintaining the alignment of sleeve and guide tube to insure a proper fill.

I. When the temperature is below freezing or the relative humidity is above 65 percent, the splice sleeve shall be externally preheated with an oxyacetylene or propane torch after all materials and equipment are in position.

J. The reinforcing bar deformations which become engaged in the Cadweld splice shall not be ground, flame-cut, or altered in any way except the longitudinal ribs which may be ground to a diameter not less than the other bar deformations.

K. A hairpin piece of soft, twisted wire may be inserted at the top of the horizontal splices between the rebar and the sleeve to provide an escape route for the gases generated during the casting of the filler material.

L. The packing material at the ends of the horizontal splices and at the top of the vertical splices should not be hard packed. The material should be firmly in place but loose enough to allow the escape of gases.

5.2.3.3.3.2.1.5 Joint Testing

A. All completed splices shall be visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve.

B. Selected splices shall be tensile tested in accordance with the following schedule for each position, bar size, and grade of bar:

1 out of first 10 splices

3 out of the next 100 splices

2 out of the next and subsequent units of 100 splices

C. Splices for testing shall be selected by Quality Control personnel on a random selection basis directly from the production work. If splices are made before the bars are placed, splices can be selected on the site of splice production.

D. Under the extreme condition, when the proper replacement of a production splice specimen slices is extremely difficult, sister splices may be used with the approval of the project engineer.

5.2.3.3.3.2.1.6 Joint Acceptance Standards

A. Sound, nonporous filler metal shall be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed 1/4 inch from the end of a sleeve due to the packing material and is not considered a poor fill.

B. Splices which contain slag or porous metal in the rise, tap hole, or at the ends of the sleeves (general porosity) shall be rejected except as covered in "E" below. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from porosity as described above.

C. There shall be evidence of filler material between the sleeve and bar for the full 360 degrees.

D. The splice sleeves need not be exactly concentric or axially aligned with the bars. However, there shall be a minimum of 1/16 of an inch between the splice sleeve and deformations of the reinforcing bar. This condition of maximum acceptable eccentricity or concentricity may be measured using a stiff 1/16-inch diameter wire.

E. The Cadweld splices, both horizontal and vertical, may contain voids at either or both ends of the Cadweld splice sleeve. At the end of the Cadweld splice sleeves, the acceptable size void for an 18S splice shall not exceed three square inches per end of splice sleeve. The area of the void shall be assumed to be the circumferential length as measured at the inside face of the sleeve times the maximum depth of wire probe minus 3/16 inch.

F. The average tensile strength of the Cadweld joints shall be equal to or greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in the appropriate ASTM standard. The minimum strength of the Cadweld joints shall be equal to or greater than 125 percent of the specified minimum yield strength for the particular bar.

5.2.3.3.3.2.1.7 Repairs

A. Joints which do not meet the visual quality acceptance standards of Section 5.2.3.3.3.2.1.6 shall be rejected and completely removed. The bars shall then be rejoined with new splice(s) made in accordance with this specification.

B. No failure of Cadweld splices below the required minimum tensile strength is expected. However, in the unlikely event that one does occur, the sample shall be sent to an independent testing laboratory for analysis of failure. Based on the test lab's report, additional samples shall be taken to insure that there are no other defective welds.

5.2.3.3.3.2.2 Splicing Reinforcing Bar Using "Swaged" Mechanical Splices

During the OTSG/RVCH replacement, "swaged" mechanical splices for concrete reinforcing steel (rebar) were used in lieu of Cadwelds. The average tensile strength of the "swaged" joints is equal to or greater than the minimum tensile strength of the ASTM A615 Grade 60 reinforcing steel used in the OTSG/RVCH replacement. The minimum tensile strength of the splices is equal to or exceeds 125 percent of the minimum yield strength for the ASTM A615 Grade 60 reinforcing steel.

All "swaged" splices were made using Barsplice Products BPI-GRIP XL series sleeves and in accordance with Barsplice *"Splice Qualification System"* as required by ASME Section III, Division 2, Subsection CC-4333, "Mechanical Splices", Specification ANO-C-517, "Swaged Mechanical Splices", and the manufacturer's instructions.

All completed splices were visually inspected. Selected splices were tensile tested in accordance with ASME Section III, Division 2, Subsection CC-4333.5.3, "Testing Frequency." Production testing was performed using sister splices.

5.2.3.3.3.3 Fabrication

Visual inspection of fabricated reinforcement is performed to verify dimensional conformance with specifications and drawings.

5.2.3.3.3.4 Placement

Visual inspection of in-place reinforcement is performed by the placing inspector to assure dimensional and location conformance with drawings and specifications.

5.2.3.3.3.5 Welding

Generally, welding was not used for splicing of reinforcement. It was applied only to some bars at temporary openings in the auxiliary building. Welding was performed in accordance with publication AWS D12.1-61 and by qualified welders with approved welding procedures. The welding engineer performed inspections on completed welds. No welding was performed in the vicinity of prestressing tendons before or during their erection. To insure that there is no accidental welding performed adjacent to prestressing tendons, a warning stencil has been attached to each stressing jack, stating, "No welding or weld ground permitted near this stressing ram."

Welding of reinforcing steel was permitted on a limited basis for the OTSG/RVCH replacement. Welding of rebars was used when mechanical splices could not be made due to space restrictions or physical limitations of the rebar projection. Rebar welding was performed and inspected in accordance with AWS D1.4, “Structural Welding Code-Reinforcing Steel”.

5.2.3.3.4 Prestress System

A. Wires

Sampling and testing of the tendon material used on construction conforms to ASTM Standard A-421. The following procedure is used:

1. Button head rupture tests from each reel of wire are made.
2. Each size of wire from each mill heat and all strands from each manufactured reel shipped to the site are assigned an individual lot number and tagged in such a manner that each such lot can be accurately identified at the job site. Anchorage assemblies are identified. All unidentified prestressing steel or anchorage assemblies received at the job site are subject to rejection.
3. Random samples, as specified in the ASTM Standards stated above, are taken from each lot of prestressing steel used in the work. With each sample of prestressing steel wire or strand that is tested, a certificate is submitted stating the manufacturer's minimum guaranteed ultimate tensile strength of the sample tested. Stress-strain curves are plotted and the yield and tensile strength verified. The anchorages develop the minimum guaranteed ultimate strength of the tendon and the minimum elongation of the tendon material as required by the applicable ASTM specification.

Field inspection insures that there are no visible mechanical or metallurgical notches or pits in the tendon material.

B. Installation

All prestressing installation work is continuously inspected by qualified inspectors. All measuring equipment used for installation are calibrated and certified by an approved independent testing laboratory. During stressing operations, records are kept by Bechtel for use in comparing force measurements with elongation for all tendons. The resultant cross reference provides a final check on measurement accuracy. Measurement accuracy and rejection allowances are in accordance with ACI 318, Chapter 26.

C. Grease

Grease is sampled after delivery and submitted to a qualified testing laboratory for chemical analysis to establish conformance with specifications.

5.2.3.3.5 Reactor Building Liner

A. Steel Plate

Steel plate is tested at the mill in full conformance with the applicable ASTM specifications. Certified mill test reports are supplied for review and approval by the design group in the project engineer's office.

B. Fabrication and Installation

Welding inspection conforms to the quality control inspection procedure described in detail by Section 5.2.4.1.1.

Dimensional tolerances are checked by installation inspectors to prevent unacceptable installation deformations.

5.2.3.4 Specific Construction Topics

5.2.3.4.1 Bonding of Concrete Between Lifts

Horizontal construction joints are prepared for receiving the next lift by wet sandblasting or by air water cutting. Surface set retardant compounds are not used.

When wet sandblasting is employed, it is continued until all laitance, coating, stains, debris, and other foreign materials are removed. The surface of the concrete is washed thoroughly to remove all laitance and to expose clean, sound aggregate, but not so as to undercut the edges of the larger particles of aggregate. After cutting, the surface is washed and rinsed as long as there is any trace of cloudiness of the wash water.

Horizontal surfaces are wetted and covered with one-quarter inch to one-half inch of mortar of the same cement-sand ratio as used in the concrete, immediately before the concrete is placed.

Vertical construction joints are waterblasted, cleaned, and wetted before placing adjacent concrete.

5.2.3.4.2 Prestressing Sequence

The detailed stressing sequence is developed to minimize unbalanced loads and differential stresses in the structure.

The procedure for prestressing is carefully worked out with the post-tensioning vendor.

Bechtel provides the vendor with the prestressing force requirements, estimated concrete elastic shrinkage, and the maximum prestress forces for each stage of prestressing. The vendor then incorporates this information along with any steel relaxation, friction, and anchorage losses to establish the initial jacking force for each sequential operation.

Force and stress measurements are made by measuring the elongation of the prestressing steel and comparing it with the force indicated by the jack dynamometer or pressure gauge. The gauge indicates the pressure in the jack within plus or minus two percent. Force-jack pressure gauge or dynamometer combinations are calibrated against known precise standards just before application of prestressing forces began and all calibrations were so certified prior to use. Pressure gauges and jacks so calibrated are always used together.

During stressing, records are kept of elongations as well as pressure obtained. Jack dynamometer or gauge combinations are checked against elongation of the tendons. Any discrepancy exceeding plus or minus five percent of that predicted by calculations (using average load elongation curves) is corrected. Differences in load elongation from averages are documented. Calibration of the jack dynamometer or pressure gauge combinations is maintained accurately within above limits.

5.2.3.4.3 Concrete Pumping

A portion of concrete in Seismic Class 1 structures has been placed by pumping; however, aluminum pipe was not used during pumping operation.

5.2.4 REACTOR BUILDING INSPECTION, TESTING AND SURVEILLANCE

The program for the surveillance of the reactor building during the lifetime of the plant is described in Technical Specification 5.5.16.

5.2.4.1 Tests to Insure Liner Integrity

During reactor building construction and after it is complete with liner, concrete, and all electrical and piping penetrations, equipment hatch, and personnel locks in place, the following tests are performed.

A. Construction Tests - These take place during the erection of the reactor building liner.

B. Preoperational Tests - These are performed after the erection of structure is complete but before reactor operation.

5.2.4.1.1 Tests on Liner During Original Construction

This procedure outlines the general requirements for welding quality control to assure that all field welding of liner plate is performed in full compliance with the applicable job specifications.

5.2.4.1.1.1 Qualifications for Welding Inspections

All welding inspectors who inspect welds covered by this specification are qualified by meeting the following minimum requirements:

A. Inspectors shall have a thorough knowledge of the various welding processes and techniques employed in field construction and shall be able to demonstrate the proper methods for shielded metal arc welding, gas tungsten arc welding, and gas metal arc welding;

B. Inspectors shall have a minimum of two years previous welding experience or equivalent experience and training in welding fabrication and nondestructive testing; and,

C. Inspectors shall be required to demonstrate, to the satisfaction of the Bechtel Material and Fabrication Quality Control Department, their knowledge of the fundamentals, techniques, and application of the inspection methods set forth in this standard, i.e., visual magnetic particle, liquid penetrant, vacuum box, and radiographic inspection.

5.2.4.1.1.2 Welding Performed By Bechtel Construction Personnel

5.2.4.1.1.2.1 Welding Procedure Specifications

All welding performed by Bechtel Construction personnel is in strict accordance with approved Bechtel welding procedure specifications. The appropriate Bechtel welding procedure specifications for field welds have been prepared, qualified, and issued to the field by the Bechtel Material and Fabrication Quality Control Department as directed by Power Division Engineering.

5.2.4.1.1.2.2 Welder Qualification

All welders who make welds under Bechtel Welding Procedure specifications have been qualified by performing the tests required by the applicable Bechtel welder performance specification WQ-F-1 for ferrous materials, and WQ-NF-1 for nonferrous materials. These Bechtel specifications encompass the requirements of Section IX of the ASME Code. No welder has been permitted to perform production welding until he has passed the necessary tests and has the appropriate welder performance qualification test record (Form WR-1) on file at jobsite. All testing of welders has been under the direction of the field welding inspector.

5.2.4.1.1.3 Instructions For Bechtel Field Welding Inspectors

5.2.4.1.1.3.1 Welding Procedure Specifications

It is the responsibility of the field welding inspector to assure that all welding is performed in strict accordance with the appropriate qualified welding procedure specifications. Specific items to be checked follow:

A. Determine that the proper welding procedure specification has been selected to match the base materials being welded and the welding processes being employed.

B. Permit only welders properly qualified under the essential variables of each welding procedure specification to make welds under that procedure.

C. Check to see that the welding electrodes, bare filler rod, consumable insert rings, and backing rings all match that which has been specified.

D. Inspect weld joints as necessary prior to welding to insure proper edge penetration, cleaning, and fit up.

E. Check to see that the welding machine settings are correct and fall within the range of current and voltage specified.

F. Check for proper preheat and interpass temperature.

G. Inspect the in-process welding for proper technique, cleaning between passes, and appearance of individual weld beads.

5.2.4.1.1.3.2 Visual Inspection of Welds

The field welding inspectors have been responsible for carrying out the necessary welding surveillance to insure that all welding meets the following requirements for visual quality and general workmanship. Visual inspection has been performed prior to, during, and after welding.

A. All weld beads, passes, and completed welds shall be free of slag, cracks, porosity, incomplete penetration, and lack of fusion.

B. Cover passes shall be free of coarse ripples, irregular surface, non-uniform bead pattern, high crown, deep ridges, or valleys between beads and shall blend smoothly and gradually into the surface of the base metal.

C. Butt welds shall be slightly convex, of uniform height, and shall have full penetration.

D. Fillet welds shall be of specified size with full throat and, unless otherwise specified, the legs shall be of approximately equal length.

E. Repair, chipping, or grinding of welds shall be done in such a manner as not to gouge, groove, or reduce the base metal thicknesses.

F. Where different base metal thicknesses are jointed by welding, the finished joint shall have a taper no steeper than 1:4 between the thick and thin sections.

5.2.4.1.1.4 Testing and Inspection

5.2.4.1.1.4.1 General

All personnel performing radiographic testing have been qualified in accordance with the Society of Non-Destructive Testing Document SNT-TC-1A.

5.2.4.1.1.4.2 Radiographic Testing and Inspection

A dual film technique is used and each radiograph has been reviewed by the welding inspector. For quality control purposes, at least one spot radiograph 12 inches in length has been taken in the first 10 feet of welding completed in the flat, vertical, horizontal, and overhead positions by each welder on liner plate butt welds. No further welding is permitted until initial radiographic inspection has been satisfactorily completed and the welding found to be acceptable by the welding inspector. Thereafter, a minimum of 10 percent of the welding (to at least include one-third of the locations where there are welded backing strips, welded splices, and intersections of joints) has been progressively spot examined as welding is performed using 12-inch length film. This has been done on a random basis (except as indicated otherwise above) with the location specified by the welding inspector in such a manner that an approximately equal number of spot radiographs can be taken from the work of each welder. Under conditions where two or more welders make weld layers in a joint or on the two sides of a double welded butt joint, one spot examination represented the work for both welders.

The techniques for radiographic examination of butt welds are in accordance with Bechtel Corporation's Non-Destructive Testing Standard RT-XG-1, Latest Revision, and Paragraph UW-51 of Section VIII of the ASME Code and ASTM Class 1 or Class 2 film.

The criterion for radiographic acceptance of welds is in accordance with paragraph UW-51, Section VIII of the ASME Code, except that the maximum acceptable length of slag inclusion in the seam welds of the 1/4-inch liner plate is not to exceed 0.125 inch.

Where a radiograph discloses welding which does not comply with the minimum quality requirements of this specification, two additional spots 12 inches in length are examined in the same weld seam at locations away from the original spot. The locations of these additional spots are determined by the welding inspector.

If the two additional spots examined shows welding which meets the minimum quality requirements of this specification, the entire weld represented by the three radiographs is judged acceptable. The defective welding disclosed by the first of the three radiographs is removed and repaired by welding.

If either of the two additional spots examined showed welding which does not comply with the quality requirements of this specification, the entire portion of the seam represented is rejected or, at the welding inspector's option, the entire weld represented is completely radiographed and defective welding corrected to meet the requirements of this specification.

The rewelded joints or weld repaired areas are completely re-radiographed and meet the quality requirements of this specification.

5.2.4.1.1.4.3 Dye Penetrant Testing

Welds which, on the basis of visual examination, are judged to be of questionable quality have been additionally inspected by dye penetrant testing. Welds which cannot be vacuum box tested because of configuration and space limitations are dye penetrant tested. Dye penetrant inspection is used to confirm the complete removal of all defects from areas which have been prepared for repair welding. All dye penetrant inspection has been done in accordance with Bechtel Corporation's Non-Destructive Testing Standard PT-SR-1, 2, Revision 2, which encompasses the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

5.2.4.1.1.4.4 Magnetic Particle Testing

Where nonradiographable butt welds were used, such as in the floor and dome liner plate, magnetic particle testing has been substituted for radiography. A minimum of 10 percent of all such welding (to at least include one-third of the locations where there are welded backing strip splices and intersections of joints) has been progressively examined as the welding is performed. Where magnetic particle inspection disclosed welding which does not comply with the requirements of this specification, additional magnetic particle testing is performed to the same extent as required for radiography.

Where unwelded backing strip splices and intersections have been permitted in the liner plate welds, such locations are examined by magnetic particle testing.

All magnetic particle testing has been done in accordance with Bechtel Corporation's Non-Destructive Testing Standard MT-P-1, 2, Revision 2, which encompasses the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

5.2.4.1.1.4.5 Vacuum Box Testing

5.2.4.1.1.4.5.1 General

Generally, liner plate, welds which must maintain leak tight integrity, including floor plates, vertical plates and dome plates, are tested as the work proceeds using a vacuum box which can be placed over the test area and evacuated. The testing equipment and procedure is discussed in the following sections.

5.2.4.1.1.4.5.2 Vacuum Box

The vacuum box is portable and has a viewing window large enough to view the complete test area and to allow sufficient light to enter the box for proper examination. The box is capable of producing and holding a pressure differential of at least 8 psi. A gauge approved by the welding inspector is placed in the system to verify the required pressure differential and to detect vacuum seal or other type leaks in the inner plate.

5.2.4.1.1.4.5.3 Leak Detecting Solution

The solution is Leaktec No. 577-V (American Gas and Chemicals, Inc.) or other soap-free, specially prepared solution as approved by the project engineer. Bubble formation properties are checked against a standard sample with a known leak path.

5.2.4.1.1.4.5.4 Test Procedures

The test area is cleaned and free of slag, scale, grease, paint, or any other materials which interfere with the test procedure or the interpretation of the test. The vacuum box is then put in place and evacuated to at least a 5 psi pressure differential with respect to the atmospheric pressure.

5.2.4.1.1.4.6 Leak Chase System Testing

5.2.4.1.1.4.6.1 General

After the liner plate welds have been inspected by vacuum testing and the leak chase has been installed in the areas shown on the design drawings, the isolated portions of the leak chase system are leak tested by the tests indicated below.

5.2.4.1.1.4.6.1.1 Leak Detecting Bubble Test

Each isolated portion of the leak chase channel is pressurized to a test pressure of 80 psi. All seal welds are coated with the leak detecting solution. Any bubble appearing within 20 seconds is cause for repairing that portion of the weld.

The leak test is repeated after repairs until no further bubbles are detected as approved by the welding inspector.

5.2.4.1.1.4.6.1.2 Pressure Decay Test

During the leak detecting bubble test, the pressure is monitored by the valving off of the air supply and measuring any pressure decay. Any pressure decay within 15 minutes indicated by a duly calibrated, 3-inch minimum diameter, dial pressure gauge is cause for rejection of that portion of the liner plate seam welds and the leak chase system. The gauge is approved by the welding inspector. All leaks are repaired. After repair, the leak chase system is completely retested.

5.2.4.1.1.5 Quality Control

5.2.4.1.1.5.1 General

A quality control system assures that the following materials, including purchased items, are erected in accordance with the applicable specification:

A. All liner plates.

B. All penetration assembly reinforcing plates and nozzle material.

C. All thickened liner plate at brackets, electrical ground rods, and floor insert plates.

D. All welding filler materials used in fabricating items "A", "B", and "C"

All members have been designated by individual permanent piece marks which are cross referenced with the piece marks shown on the fabrication drawings. Inspection reports reflected these designated piece marks.

All jobsite testing and inspection has been performed in the presence of the welding inspector.

5.2.4.1.1.5.2 Documentation Requirements

The following documentation is maintained:

A. Reports of mill tests, impact tests, ultrasonic tests, and heat treatment,

B. Manufacturer's certification for welding filler materials,

C. Qualified welding procedures,

D. Welder qualification records,

E. Records of welds required for leak tight integrity with name of welder, date, and procedure,

F. Copies of radiographs identified as to location and record of passing or failing result,

G. The vacuum box test reports with passing or failing result,

H. Dye penetrant test reports with passing or failing result,

I. Magnetic particle test reports with passing or failing result and magnetic particle testing procedures,

J. Records of testing leak chase system with passing or failing result, and

K. Records of tolerance checks.

5.2.4.1.2 Tests on Liner During OTSG/RVCH Replacement

During the OTSG/RVCH Replacement a section of the liner plate was removed and reinstalled to allow moving the OTSGs and RVCH into and out of the reactor building. After reinstallation, the liner plate welds at the perimeter of the restored opening were examined by visual examinations, 100% magnetic particle and/or random radiography, and vacuum box leak testing in accordance with the original code and specifications. Dye penetrant inspection was used to confirm the complete removal of all defects from areas which were prepared for repair welding. Both surfaces of the liner plate were visually inspected for damage prior to its reinstallation and repaired as required. After completion of the construction code inspections, a pre-service examination/inspection of the liner plate was performed per the requirements of subsection IWE-2200 of ASME Section XI.

5.2.4.1.3 Preoperational Integrated Leak Test

The design leak rate is not more than 0.2 percent by weight of the contained atmosphere in 24 hours at design pressure. It has been demonstrated that, with good quality control during erection, this is a reasonable requirement.

The absolute method was used for the test. The objectives of the test were:

A. To determine the initial integrated leakage rate for comparison with the design leakage rate.

B. To establish representative leakage characteristics of the containment system to permit retesting at reduced pressures.

C. To establish a test method and the equipment to be used for subsequent retesting.

The leakage rate was measured by integrating the leakage rate for a period of 8 3/4 hours. The necessary instrumentation was installed to provide accurate data for calculating the leakage rate.

5.2.4.1.4 OTSG/RVCH Replacement – Integrated Leak Rate Test (ILRT) and Containment Pressure Test (CPT)

After the OTSG/RVCH replacement, the restored structure was subjected to a post-repair/pre-service ILRT per Article IWE 5000 of ASME Section XI. This test also satisfied all requirements of the Containment Pressure Test (CPT) required by Article IWL 5000 of ASME Section XI. The detailed procedure for the CPT was developed under the direction of the Responsible Engineer (RE) in accordance with Article IWL 5250 of ASME Section XI. The ILRT procedure was reviewed by the RE to assure all applicable CPT requirements included in the ILRT meet Article IWL 5000 requirements. The ILRT/CPT was conducted based on the DBA pressure of 59 psi.

5.2.4.2 Strength Test

The basic purpose of the reactor building structural integrity test is to verify: 1) the structure can safely carry the design pressure load and 2) the structural behavior is similar to that predicted by analysis. Since the test pressure exceeds the design pressure, the test will verify the reactor building structure has reserve strength. The displacement measurements during the test provide verification of structural behavior predicted by analysis.

The number of points in the reactor building structure that will be monitored during the structural integrity test is not significantly different from that recommended by Safety Guide 18. However, the distribution of points presently planned provides more coverage in monitoring of the structural response, therefore, providing additional data for analysis of the reactor building structural response.

The development of the testing program is based on past experiences gained in testing of earlier post-tensioned containments (as described in Section 5.2.4.2.1). Specifically, the following considerations were given:

A. During the structural acceptance test of the previous containments, the results indicated that: 1) the buttress and wall strains and radial displacements were similar; 2) the relative displacement between the wall and buttress was not appreciable; and 3) there was close agreement between observed response and design calculations. Based on these results, the buttress may be predicted to have a negligible effect on the adjacent walls. Therefore, extensive measurement at the buttress is not necessary.

B. In addition to the results obtained from past prestressed containments, full scale buttress test and tendon end anchor reinforcement tests were performed to verify the structural adequacy of the buttress. The test reports for the above tests are included in the Topical Report, BC-TOP-7, "Full Scale Buttress Test for Prestressed Nuclear Containment Structures" and the Topical Report, BC-TOP-8, "Tendon End Anchor Reinforcement Test." Numerous strain gauges were incorporated in the test structures which were subjected to more extensive and severe loading conditions relative to the applied pressure test required by Safety Guide 18. In the buttress test, the results indicated that the buttress, with identical and reduced reinforcing, can adequately withstand loads equivalent to the ultimate capacity of the tendon. This is approximately 1.4 times the tendon anchoring load. In the case of the tendon end anchor reinforcement tests, the results indicated the buttress can carry loads, of 3,000 kips, or approximately two times the tendon anchoring load without reaching its ultimate capacity. Therefore, it may be concluded that the buttresses are not predicted to adversely affect the structural integrity of the reactor building.

C. The tendons to be used for the post-tensioning of the reactor building structure are composed of 186 stress relieved, high strength wires of 1/4-inch diameter furnished in accordance with ASTM A-421-65. The only difference relative to those tendons used for earlier containments is in the number of wires. The tendon anchorages for both tendons, large and small, are designed to conform to the same design criteria, since the prestress level for any unit length in the reactor building structure is approximately the same as in past projects. Therefore, the structural integrity of the reactor building structure is predicted to be influenced by the increase in tendon capacity.

In consideration of the factors described above, it is felt that the structural acceptance test is adequate as performed in accordance with Section 5.2.4.2.1. Accordingly, the following aspects of Safety Guide 18 are not incorporated in the reactor building structural acceptance test:

A. Two meridians, as opposed to the six specified in Paragraph C.2 of the safety guide, will be used to measure the radial displacement of the reactor building wall and buttress.

B. The strain measurements required for a prototype containment, as specified in Paragraph C.5 of the safety guide, were not provided.

5.2.4.2.1 Reactor Building Proof Tests

A pressure test is made on the completed building using air at 115 percent of design pressure. This pressure is maintained on the building for a period of one hour. During this proof testing, measurements and observations are made to verify the adequacy of the structure design. Periodic structural integrity tests are not planned after the initial proof test as there is no reason to anticipate deterioration during the life of the reactor building structure. However, as in the case for the Integrated Leak Rate Test (see Section 5.2.4.4), periodic inspections will be conducted in accessible areas. In addition, a tendon surveillance program, as described in Section 5.2.4.3, will be performed to insure structural integrity of the reactor building.

The basic purpose of a structural proof test is to substantiate that the structure can, in fact, carry the load for which it is designed. By subjecting the structure to some degree of overpressure, the test can show that the vessel has that margin over design pressure and thus cannot be at incipient failure at design pressure.

As a minimum, the test pressure used must exceed design pressure by the accuracy tolerance of the pressure measuring instruments. As a maximum, it must not endanger or damage the vessel (by reducing the reserve capacity of concrete in tension or creating permanent deformation in the vessel) by encroaching too close to the actual factored loads for which the vessel is designed. Since the overall structural integrity of the vessel depends primarily on the integrity of the tendons, it is appropriate to test the vessel to at least that pressure which creates a stress in the tendons equivalent to the stress at design conditions due to dead load, pressure, and temperature at end of plant life. The predicted stresses in the tendons due to the factored accident pressure load (1.5P) is 64 percent of the ultimate strength. It is felt that a test pressure of 110 percent of design pressure adequately demonstrates the overall structural integrity. However, in order to conform to past practices, it has been decided to proof test the reactor building at 115 percent of design pressure.

The safety margin of the prestressed concrete reactor building at test, based on ultimate strength, can be compared to a steel vessel by reviewing safety margins on various types of stresses and the significance of the stresses in the failure mode of the respective vessels.

The prestressed concrete reactor building relies upon the tensile strength of the tendons for its strength. The secondary stresses of the vessel are isolated from the tendons. At ultimate capacity of the vessel, the secondary stresses and the thermal stresses for concrete have been relieved by local cracking of the concrete and the tendons are subjected to internal pressure and dead load only. Dead load stresses are insignificant and tend to reduce the tendon stresses.

The engineered margin of safety for ultimate structural integrity of a steel containment vessel should be based on the ultimate stress as related to stress at test pressure for various combinations of stresses. Based on the ASME Boiler and Pressure Vessel Code, Section III, the margin of safety for the steel containment relative to its ultimate strength, is as follows:

Type of Stress	Stress at Test (1.25 allowable, S_M)	Margin of Safety
Membrane	21,900	3.2
Membrane plus Bending	32,800	2.13
Membrane plus Bending plus Secondary	65,600	0.92

The prestressed concrete reactor building has various material elements contributing to the structural integrity of the vessel. The margin of safety at test pressure of the tendons, which are the most critical elements of the structure, is 1.60. This margin of safety for the prestressed concrete reactor building is lower than that of a steel containment when it is based on a comparison of membrane stresses. However, the margin of safety of 3.2 shown for membrane stress in a steel vessel neglects the effect of secondary and thermal stresses and their ability to propagate failure. Since the membrane integrity at ultimate strength is controlled by the secondary stress concentrations, the margin of safety for this case forms a more reasonable basis for comparison with the prestressed vessel. Certainly, the margin of safety at ultimate failure is larger than 0.92 and must lie between 0.92 and 2.13, depending on the significance of the secondary stresses. An exact value for this margin of safety is virtually impossible to evaluate.

Based on the ultimate capacities, the margin of safety of a steel vessel and a prestressed vessel, at test pressure, are roughly comparable.

The prestressed concrete reactor building is a ligament type vessel where the failure of a single ligament results in a load redistribution to adjacent ligaments. This type of gradual progressive failure of isolated ligaments gives ample warning of distress during tests rather than a possible sudden catastrophic failure of a biaxially stressed steel membrane.

The selected test pressure cannot be considered as proof of tendon strength, but rather the safe design of the other important components, mainly the concrete and, to some limited extent, the liner plate. There is no other significant similarity between the test pressure and the design pressure. The design pressure is not considered to act before thermal stresses have been developed in the concrete shell. The uneven temperature in the shell produces compressive stresses in the inside and tensile stresses in the outside fibers. The test pressure produces tension across the whole concrete section. This membrane stress does not exceed the compressive stresses developed by the prestressing. However, some localized, flexural tensile stresses exist at test pressure. Therefore, further increase of the test pressure is not advisable.

In the case of accident, the liner plate is under high compressive stresses. The design pressure decreases the compressive stresses, but, unlike the test pressure, does not cause any tension stresses. The high compressive stresses in the liner plate result from the prestressing and thermal conditions. This condition cannot be reproduced without heating up the inside of the vessel to the accident temperature while the outside ambient temperature also happens to coincide with the design conditions.

5.2.4.2.3 Reactor Building Structure Instrumentation

The purpose of instrumenting and testing the prestressed concrete reactor building structure is to provide a means for comparing the actual response of the structure due to the loads induced during post-tensioning and pressure testing with the predicted responses. If the actual response is within the range predicted, the design techniques are assumed to have been verified.

The reactor building structure is very similar to the Turkey Point, Palisades, and Oconee structures. The design and construction are similar. The structures for Turkey Point and Palisades are completely instrumented. The Turkey Point instruments provided approximately 400 strain measurements at 55 locations throughout the structure and liner and deformation measurements were made. The Palisades and Oconee instrumentation are comparable. This amount of data permitted a detailed comparison between design calculations and observed response. The basic structural response and the accuracy of the calculation procedures used by Bechtel are, therefore, verified by these tests.

Since the above detailed confirmation of the design techniques are available, strain measurements are not provided for this structure. However, measurements of the reactor building structural response are made during post-tensioning and pressure testing.

Prior to the reactor fuel loading and operation, the integrity of the reactor building structure is demonstrated by a pressure proof test. The post-tensioning and the pressure test provide verification that the structural response due to the induced loads is consistent with the predicted behavior. This is accomplished by measurements of the structure's deformation.

From present knowledge of the analytical uncertainties, it is expected that agreement will be found between test results and analytical predictions that are within the ranges of the following table:

Agreement Between Measures and Predicted Stresses

Cylinder at Equator	15%
Dome	15%
Bottom Slab	25%
Bottom Slab Wall Junction	25%
Dome-Wall Junction	20%
Around Opening	30%
Localized Stress Concentration	100%

However, test values falling outside of the ranges will not be considered as indicative of a lack of adequate structural integrity since the test results could indicate a greater than anticipated conservatism for the loads used in the test or other load conditions. Also, the structural integrity cannot be judged on the data acquired from only one sensor, since such precise devices can malfunction. In any event, the multiform character of the structural system decreases the sensitivity of the structure to local variation from predicted values.

The measurement technique used during post-tensioning requires the attaching of targets to the surface of the reactor building structure at appropriate elevations and azimuths and around the access openings. Measurements of displacement to within 0.01 inch can be made. Figure 5-17, "Reactor Building Instrumentation for Post-Tensioning," indicates the targets for measuring shell deflections, both vertical and radial, and at the opening. The system for measuring these deflections is also shown.

The measurement technique used during pressure proof testing requires the attaching of taut wire extensometers from points on the containment liner plate to other points on the liner plate or to internal concrete structures. These extensometers are located at appropriate elevations and azimuths and around access openings as shown in Figure 5-18, "Reactor Building Instrumentation for Pressure Proof Test." Measurements of displacements to within 0.001 inch can be made.

The proof testing of each tendon is achieved at the time of post-tensioning when the tendon is stressed to the maximum value occurring during its life.

5.2.4.3 <u>Inservice Tendon Surveillance Program</u>

The inservice surveillance program for the reactor building consists of evaluating the tendon system performance and the corrosion protection. Further, the reactor building is a passive type system where mechanical operational failures are nonexistent, thus only requiring that the system remain in status quo and be available to perform its function in the unlikely event that it is required. It is the intent of the surveillance programs to provide sufficient inservice historical evidence necessary to maintain the confidence that the integrity of the reactor building is being preserved.

To accomplish the surveillance program, the following quantity of tendons were selected for inspection and lift off readings for inspections 1, 3, and 5 years following the initial containment structural integrity test.

- Hoop - Ten tendons, at least three in each of the three 240-degree sectors of the reactor building.
- Vertical - Five tendons, located at approximately equally spaced intervals.
- Dome - Six tendons, two located in each of the three groups of dome tendons.

The surveillance program for structural integrity and corrosion protection consists of the following operations being performed during each inspection.

A. Lift off readings are taken for all tendons being inspected.

B. A minimum of three surveillance tendons, one from each of the hoop, vertical, and dome families, are relaxed and one wire from each tendon is removed as a sample for inspection.

C. After the inspection, the tendons are retensioned to the stress level measured at the lift off reading and then checked by a final lift off reading.

D. Should the inspection reveal any significant physical change (pitting or loss of area), additional wires from each surveillance tendon will be removed and further inspection will be made to determine the extent and cause of the change. Samples of corroded wires will be tested to evaluate the effects of any corrosion.

Inspection requirements for reactor building, as well as those for other systems, are a part of the Technical Specifications for the plant and are included in the operating license application. Changes in these inspection requirements or their elimination at any time during plant life are subject to NRC regulations governing Technical Specifications and require review and approval by the NRC after justification by the applicant.

A conservative testing frequency has been established and nothing in the design of the reactor building precludes the testing of tendons or pneumatic testing of the reactor building at any time in its life. Temporary inspection structures are used to obtain access to the reactor building tendons for periodic inspection and testing.

As a result of acceptable inspections during the 1, 3, and 5 year surveillances, Technical Specifications were revised to reduce the total sample population from 21 to 9 tendons (3 hoop, 3 dome, 3 vertical) for the ten year surveillance and all remaining surveillances.

5.2.4.4 Leakage Monitoring System

No continuous reactor building leakage monitoring system has been provided. Leak chase channels are provided at the liner plate seams where the liner plate is not readily accessible for inspection.

The barrier to leakage from the reactor building is the 1/4-inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. These penetrations are an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete and is an integral part of this structure. The reactor building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable DBA, as well as for all other types of loading conditions which could be experienced. Thorough control is maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetration and during construction of the entire reactor building.

The comprehensive program for preoperational testing, inspection, and post-operational surveillance is described in detail in Section 5.2.4 and is summarized in the following paragraphs.

During construction, the entire length of every seam weld in the liner plate is leak tested. Individual penetration assemblies are shop tested. Welded connections between penetration assemblies and the liner plate are individually leak tested after installation.

Following completion of construction, the entire reactor building, the liner, and all its penetrations are tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate test of the entire reactor building is conducted at 100 percent of the design pressure and at successively lower pressures to demonstrate vapor tightness and to establish a reference for periodic leak testing for the life of the station. Multiple and redundant systems based on different engineering principles are provided as described in Chapter 6 to provide a very high degree of assurance that the accident conditions will never be exceeded and that the vapor barrier of the reactor building will never be jeopardized.

Under all normal operating conditions and under accident conditions including the DBA, no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls will be enforced to minimize the possibility of human error. Station operators will be trained and licensed in accordance with current regulations. Safety analyses are presented in Chapter 14.

Penetrations such as the permanent equipment and personnel access hatches cannot be opened except by deliberate action and are interlocked and alarmed by fail-safe devices (i.e. monitored and controlled by plant security via Central Alarm Station/Secondary Alarm Station systems) such that the reactor building will not be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate is blocked by interior concrete, means have been provided so that weld seams can be tested for leakage. The liner plate is protected against corrosion by suitable coatings. Walls and floors for radiation and missile shielding, and for access and operating purposes, also provide compartmentation which constitutes protection for the liner during operating as well as accident conditions.

Once the adequacy of the liner has been established initially, there is no reason to anticipate progressive deterioration during the life of the station which would reduce the effectiveness of the liner as a vapor barrier. Inside the reactor building, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by 3-3/4 feet of prestressed concrete which is exceptionally resistant to all weather conditions.

Most penetrations are grouped and are in penetration areas. Any leakage that might occur into the penetration areas from these penetrations will be collected and exhausted through the station vent. See Section 6.5 for information on the Reactor Building Penetration Room Ventilation System (PRVS). Section 6.5.2.1 contains a list of penetrations not served by the PRVS and a discussion of the acceptability of the design.

Should there be any indications of abnormal leakage, the source of excessive leakage will be located and such corrective action as necessary will be taken. This will consist of repair or replacement. Appropriate action will also be taken to minimize the possibility of recurrence of excessive leakage, including such redesign as might prove to be necessary to protect public health and safety. Leak testing will be continued until a satisfactory leak rate has again be demonstrated. The Integrated Leakage Rate Test can be performed at calculated peak pressure anytime during the plant life, if required, providing that the enclosed systems have been prepared for the test. The proposed reactor building leakage testing program for Unit 1 complies with Appendix J to 10 CFR 50.

The method used to conduct the ILRT is the absolute method. Additional information concerning testing criteria, testing methods, and expected accuracy is found in Bechtel Topical Report BN-TOP-1, "Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants," and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements." The Type A leakage rate is calculated using the definition contained in NEI 94-01, Revision 2-A, Section 5.0 (Reference 1CNA021501).

5.2.5 ISOLATION SYSTEM

5.2.5.1 <u>Design Bases</u>

The general design basis governing isolation valve requirements for reactor building penetrating lines is:

> Leakage through all fluid penetrations not serving engineered safeguards systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building, and various types of isolation valves.

Reactor building isolation occurs automatically on a signal of high pressure in the reactor building or a signal of low pressure in the Reactor Coolant System.

The isolation system closes all fluid penetrations not required for operation of the engineered safeguards in order to prevent leakage of radioactive materials to the environment. Fluid penetrations serving engineered safeguards also meet this design basis.

All remotely operated reactor building isolation valves are provided with position limit indicators in the control room.

5.2.5.2 <u>System Design</u>

All piping in reactor building isolation valve systems is designed, fabricated, and inspected to ANSI B31.7 Class 1 or 2, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The accident analysis for failure or malfunction of each valve is presented with the respective system evaluation of which that valve is a part, for example, Chemical Addition and Sampling System, etc.

5.2.6 VENTILATION SYSTEM

5.2.6.1 Design Bases

5.2.6.1.1 Governing Conditions

The reactor building normal ventilation system utilizes internal cooling units that remove normal heat loads from equipment and piping in the reactor building. Originally designed for purging of the reactor building with fresh air whenever desired, operation of this system for normal purging is restricted to cold shutdown or refueling shutdown conditions due to the concern for containment integrity during other operating conditions. (1CAN127908) (NUREG-0737, Item II.E.4.2)

5.2.6.1.2 Sizing

To provide for access to the reactor building, the normal ventilation system is sized to control the interior air temperature to 110 °F during operation and a minimum of 50 °F during shutdown.

5.2.6.2 System Design

A flow diagram of the Reactor Building Cooling and Purge Systems is shown in Figure 5-7.

The normal cooling system utilizes the five fan cooler units, VSF1A, B, C, D, E, located in the building outside the secondary shielding. The coolers use chilled water supplied by a water chiller as the heat removal medium and discharge the cooled air through ducts to provide adequate distribution for the equipment and areas including reactor cavity. During any accident, including the DBA, four coolers, VSF1A, B, C, D, use low pressure service water through separate cooling coils to reduce the pressure by removing heat from the reactor building atmosphere. The details of the units and their special features are described in Section 6.3.

The Reactor Building Purge System consists of a supply fan unit with a heater and a filter, and an exhaust fan filter unit with a prefilter, absolute filter, and charcoal filter. All of the Reactor Building Purge System, except interior ducts, and two isolation valves are located outside the reactor building. Ducts are provided inside the reactor building for adequate distribution.

The Reactor Building Purge System discharge to the atmosphere through the reactor building flue (Figure 11-4) is monitored and alarmed to prevent release exceeding acceptable limits.

5.2.6.2.1 Isolation Valves

Service water lines are provided with isolation valves to allow access to the cooling units for maintenance. Chilled water lines and the Reactor Building Purge System are provided with double automatic isolation valves. The purge valves will be normally closed and will be opened only for the reactor building purging operation.

5.2.7 REACTOR BUILDING ACCESSIBILITY CRITERIA

The reactor building is designed for purging during operation (See Sections 5.2.6.1.1 and 5.2.6.2) to allow accessibility into certain areas of the reactor building for necessary maintenance. The purging rate will be limited by monitors and, together with the filters shown in Figure 5-7, will reduce radioactive release to as low as practicable. Particulate matter will be removed from the purge gas before gas is released to the atmosphere through the purge vent.

With the 1986 installation of redundant internal hydrogen recombiners instead of the original hydrogen purge system (with capability to install an external hydrogen recombiner) for combustible gas control, engineering analysis found that the Reactor Building Purge System supplied sufficient controlled purge capability to meet the recommendation of Regulatory Guide 1.7, Rev. 2. Regulatory Guide 1.7, Rev. 2 suggests that there be an installed capacity for a controlled purge of the containment atmosphere to aid in cleanup:

> "The purge or ventilation system may be a separate system or part of an existing system. It need not be redundant or be designated Seismic Category I (see Regulatory Guide 1.29), except insofar as portions of the system constitute part of the primary containment boundary or contain filters."

5.2.8 SYSTEM DESIGN EVALUATION

The reactor building with the appurtenant engineered safeguards systems will prevent uncontrolled release of radioactivity to the plant and surrounding areas during normal operation and a DBA. The reactor building is designed for the pressure and temperature resulting from a DBA concurrent with other design loads.

5.3 OTHER MAJOR PLANT STRUCTURES

5.3.1 REACTOR BUILDING INTERNALS

The reactor building internals are designed as a Seismic Class 1 structure. The design basis is discussed in detail in Section 5.1.

The reactor building internals are a reinforced concrete structure. The internals consist of the reactor cavity, two steam generator compartments, and a fuel transfer canal which is located between the two steam generator compartments and above the reactor cavity. The reactor cavity houses the reactor vessel and serves as a shield. The steam generator compartments houses the steam generators and other components of the Reactor Coolant System (RCS). The primary function of the steam generator compartment walls is to serve as secondary shield walls and to resist the pressure and jet loads due to pipe rupture. The jet force, caused by a break in a high pressure pipe in the form of a guillotine or slot failure, is calculated as a product of the cross-sectional area and the maximum pressure in the pipe. The dynamic effect caused by the impingement of the jet force on the internal walls is also considered in the design.

Results of differential and absolute pressures calculated for Loss of Coolant Accident (LOCA) are listed below.

Cavity (Compartment)	Differential Pressure (psid)
A. Reactor cavity below level of refueling canal reactor vessel seal	230
B. Steam generator and pressurizer compartment	16
C. Operating floor	0.8

The jet force is calculated as 3,360,000 pounds for the hot leg break and 2,000,000 pounds for the cold leg break.

The cavity walls are designed to withstand a jet force coincident with the pressure load resulting from pipe rupture. Loading combinations and allowable material stresses listed in Section 5.1 are used. Local yielding under pipe rupture loads is allowed with the ductility factor limited to "3." Maximum stresses occur at those locations subjected to pipe rupture loads.

The reactor building primary concrete shield design considered the effects of radiation generated heat. The calculations of the amount of heat generated by radiation were based on the analytical models outlined in Atomic Energy Research Establishment R/R 1963 entitled, "Heat Release in Concrete Reactor Shields," by D. B. Halliday. The design of the reinforcement at this location was based on the results of a computer finite element program and is in accord with the design criteria presented in Section 5.1.3.1.

The reactor building polar crane is Seismic Class 1 equipment and is designed in accordance with the seismic considerations associated with Seismic Class 1 structures (refer to Section 5.1.4). In addition, two seismic restraints are provided at each end of the polar crane girders to preclude dislodgement during a seismic disturbance.

The reactor building crane is supported on the cylindrical shell by structural steel brackets. The main and auxiliary fuel handling bridges and Control Rod Drive crane are supported on the secondary shield walls. The fuel tilt machine is a stainless steel transfer tube through the cylindrical wall. It permits moving new fuel in and spent fuel out of the containment and is designed to maintain containment isolation criteria.

The major structural materials used in the design of the reactor building internals are as follows:

Structural Steel: ASTM-A36

High Strength Bolts: ASTM-A325 and ASTM-A490

Reinforced Concrete: f'_c = 3,000 psi @ 28 days for steel support slabs and
f'_c = 5,500 psi @ 90 days for primary and secondary shield walls and supports

Reinforcing Steel: ASTM-A6l5 Grades 40 and 60

Reinforcing arrangements and anchoring details for the more critical parts of the internals are presented in Figures 5-8 and 5-9.

The above structural materials were also used in the design of structural supports for the reactor, steam generators, pressurizer, main coolant pumps, and Core Flooding Tanks. The design criteria, allowable stresses and strains, and the load combinations used in the design of these supports are in accordance with Section 5.1. Section 4.2.6 contains descriptions of these systems.

5.3.2 AUXILIARY BUILDING

The auxiliary building is basically a Seismic Class 1 structure, housing the fuel pool, safeguard system, radwaste system, and other Seismic Class 1 equipment. It is designed as specified in Section 5.1. The following additional requirements, where applicable, are also considered:

A. Internal flooding due to pipe rupture and the resulting hydrostatic flood.

B. High velocity missiles, except for the steel superstructure and other structures not housing Seismic Class 1 equipment.

C. Special requirements to prevent criticality of new and spent fuel bundles.

D. External flooding as described in Section 5.1.6.

The following significant facilities related to plant safety are located in the auxiliary building:

A. New and spent fuel handling storage and shipment facilities.

B. Control room and related facilities.

C. Decontamination facilities.

D. Radioactive waste disposal facilities.

E. Access control station.

F. Engineered safeguards equipment.

The auxiliary building is designed in accordance with "Building Code Requirements for Reinforced Concrete, ACI-318-63" for concrete and AISC Manual of Steel Construction, 1963 edition for structural steel unless otherwise noted. Tornado, wind, earthquake loadings, load factors, and load combinations as specified in Section 5.1 are used in the design of the auxiliary building. To proportion steel reinforcement for Seismic Class 1 reinforced concrete structures other than the reactor building, the U.S.D. technique was used.

The major structural materials used in the design of the auxiliary building are as follows:

Concrete:	f_c' = 3000 psi @ 28 days
Reinforcing Bars:	ASTM-A6l5 Grade 40
Structural Steel:	ASTM-A36
Stainless Steel Pool Liners:	ASTM-A167, Type 304L

All Seismic Class 1 portions of the auxiliary building are founded on rock.

The reinforced concrete slabs and walls have been designed for dead, live, and lateral loads. Seismic, wind, and other lateral loads have been assumed to be carried to the foundations by diaphragm action in slabs and shear wall action in walls.

Concrete block walls within the Seismic Class 1 boundaries of the auxiliary building are considered to be in proximity to safety related systems and could be considered to have potential impact upon the operability of safety related systems. A detailed, structural analysis was performed during 1980 in response to IE Bulletin 80-11 on those concrete block walls which were determined to meet the following criteria: (1) The wall had Seismic Class 1 piping, or (2) The wall had other seismic class 1 components attached to it. Also, if the wall was in proximity to safety-related equipment, it was evaluated. The results of this review were summarized in letter 0CAN018120, dated January 20, 1981. The dynamic loads due to Seismic Class 1 piping or equipment attached to the walls were included in this analysis. The spent fuel pool walls are inherently resistant to missiles. The spent fuel pool can withstand the maximum heat load of 212 °F inside the pool without failure. Reinforcing steel is provided in the structure to control cracking. The 212 °F temperature results from the loss of all cooling from the two pumps and coolers. This condition is described in Chapter 9 as highly improbable and exists only under the most unusual circumstances.

The emergency diesel generator rooms are also a part of the auxiliary building. The layout is between column lines A & C and column lines 1 & 3. A structural wall and a fireproof door along column line B will prevent missiles, explosions and fires in one diesel generator unit from affecting the other unit. The emergency diesel generator rooms are above the 361’ flood level.

Reinforcing arrangements for the spent fuel pool in the auxiliary building are shown in Figure 5-9.

The floor area at Elevation 317’ containing engineered safeguard equipment is partitioned into separate rooms to provide protection in the event of flooding due to a pipe rupture. The partition walls are designed to withstand hydrostatic loading over their full height.

The auxiliary building also houses the fuel handling crane, which is supported overhead from the structural steel above the fuel handling floor. The fuel handling bridge, which travels on rails at the fuel handling floor level, runs along the spent fuel pool. The travel of the overhead crane is restricted over the fuel pool by means of limit switches and administrative controls to allow only the movement of fuel over the storage racks.

The control room is on the northeast side of the auxiliary building at Elevation 386'. It contains all the automatic control equipment required to maintain safe plant operation. Adequate radiation protection is provided for the personnel working inside, even under accident conditions. It is fireproof and is protected by a missile barrier at its entrance.

The superstructure over the spent fuel pool consists of a steel frame clad with light gauge metal siding and roof decking. Portions of siding and roof deck will be blown off but the steel frame is designed to assure that it will not collapse or distort so as to allow the bridge crane to fall. Steel frames supporting the roof and crane have been checked to assure that stresses do not exceed 90 percent of minimum yield stress if one-third of the siding remains in place. The assumption that only one-third of the siding remains in place is quite reasonable, considering that applied force is 185 psf on the windward side and 110 psf on the leeward side and the normal allowable design pressure is in the order of 25 psf.

Mechanical anti-derailing devices mounted on the wheel axles of the overhead crane bridge and trolley and the fuel handling bridges prevent them from being dislodged during an earthquake.

Fire protection is provided by firewalls and fire doors which divide vital areas into fire zones. Security doors are provided to limit access.

5.3.3 TURBINE BUILDING

The turbine building is a Seismic Class 2 structure which houses the turbine generator and other Seismic Class 2 equipment, piping, and systems. See Section 5.1 for design basis.

Below grade, the building consists of reinforced concrete walls, slabs, piers and footings. All structural supporting members are founded on bedrock. The superstructure consists of structural steel, reinforced concrete slab, and steel grating floors. The building is enclosed with metal siding.

The major structural materials used in the design of the turbine building are as follows:

Structural Steel:	ASTM-A36 or better
High Strength Bolts:	ASTM-A325 or better
Reinforced Concrete:	f_c' = 3,000 psi @ 28 days or better
Reinforcing Bars:	ASTM-A6l5 Grade 40 or better

The turbine building is analyzed for seismic loads, wind loads, and operating loads as outlined in Section 5.1.

The reinforced concrete turbine generator pedestal is designed for static and dynamic load conditions.

5.3.4 INTAKE STRUCTURE

The intake structure houses the circulating water, fire, screen wash, sodium bromide/sodium hypochlorite, and service water pumps as well as motor control centers and traveling screens. While, generally, the intake structure is a Seismic Class 2 structure, portions housing Seismic Class 1 equipment are designed as Seismic Class 1 structures. The design bases are discussed in detail in Section 5.1. The intake structure is a reinforced concrete structure founded on bedrock. It is serviced by a 25-ton gantry crane which is supported by steel crane rails and girders on reinforced concrete piers.

The intake structure has been designed to Seismic Class 1 standards as outlined in Section 5.1 to ensure safe operation of service water pumps which are Seismic Class 1 equipment. The structure has been designed to withstand tornado, flood, live, and dead loads.

The major structural materials used in the design of the intake structure are as follows:

Structural Steel:	ASTM-A36
High Strength Bolts:	ASTM-A325
Reinforced Concrete:	f'_c = 3,000 psi @ 28 days
Reinforcing Bars:	ASTM-A6l5 Grade 40

Reinforcing arrangements of the intake structure are shown in Figure 5-9.

5.3.5 MISCELLANEOUS STRUCTURES

5.3.5.1 <u>Administration Building</u>

The administration building is a Seismic Class 2 structure on the south side of the turbine building. The floor and roof are precast, reinforced concrete deck supported on structural steel framing. The foundations are reinforced concrete spread footings and are founded on compacted backfill.

5.3.5.2 <u>Emergency Diesel Fuel Storage Vault</u>

The emergency diesel fuel storage vault is a Seismic Class 1 structure on the northwest side of the reactor building. It is a rigid, reinforced concrete box structure. It contains four diesel fuel storage tanks, partitioned into separate rooms to provide protection if fire or flooding occur. The walls are designed to withstand hydrostatic loading over their full height. The structure has a mat foundation founded on rock.

5.3.5.3 <u>Sodium Bromide and Sodium Hypochlorite Building</u>

The sodium bromide & sodium hypochlorite building is on the southeast side of the turbine building, adjacent to the Unit 2 intake structure. It is a Seismic Class 2 structure with structural steel columns and roof framing. The foundation is reinforced concrete mat and is founded on compacted backfill.

5.3.5.4 Transformer Yard

The transformer yard is on the east side of the turbine building. It contains three main transformers, one spare main transformer, one auxiliary transformer, and two startup transformers. The transformers are founded on reinforced concrete pads supported by caissons. The transformers are separated by precast concrete fire walls. The buses serving the transformers are supported by a structural steel framing system founded on spread footings.

5.3.5.5 Condensate Storage Tank (T41B)

The Condensate Storage Tank (T41B) is a Seismic Class 1 structure located on the west side of the reactor building. The tank is supported on a 2½ foot thick by 52 foot reinforced concrete octagon mat supported on 42" diameter drilled concrete piers. Two 11.5 X 12.5 X 8.5 foot valve pits are located partially underneath and on opposite sides of the foundation. A 5-foot high reinforced concrete wall designed for missile protection surrounds the tank.

5.3.5.6 Alternate AC Power Source Building

The Alternate AC Power Source Building is north of and adjacent to the north side berm of the Fuel Oil Storage Tank (T-25). It is a Seismic Class 2 structure. It is of steel framed pre-cast concrete construction with a steel framed roof and reinforced concrete slab and founded on grade beams supported by drilled in piers (caissons).

5.3.5.7 Original Steam Generator Storage Facilities (OSGSFs)

The Original Steam Generator Storage Facilities (OSGSFs) are located north of Unit 2, outside the protected area and adjacent to the north access road. They are reinforced concrete structures designed for long-term storage of contaminated equipment (i.e., the steam generators originally stored in Units 1 and 2, the Unit 1 RCS hot leg elbows, and the reactor vessel closure head originally installed in Unit 1) and are not intended for onsite storage of radwaste materials. The shielding provided by the concrete walls and roofs is sufficient to keep doses external to the OSGSFs below the requirements of 10 CFR 20 and 40 CFR 190.

5.4 REFERENCES

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4. "Nuclear Reactors and Earthquakes," AEC Publication TID-7024

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6. "Wind Forces on Structures," Task Committee on Wind Forces, ASCE Paper No. 3269

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8. Hardingham, R. P., J. V. Parker, and T. W. Spruce, "Liner Design and Development for the Oldsbury Vessels", Group J., Paper 56, London Conference on Prestressed Concrete Pressure Vessels.

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11. U.S. Army Corps of Engineers, "Fundamentals of Protective Design," Report AT1207821, Office of the Chief of Engineers, 1946.

12. BC-TOP-9A, “Design of Structures for Missile Impact,” Rev. 2, September 1974

13. EPRI NP-440, “Full-Scale Tornado-Missile Impact Tests,” July 1977

Table 5-1

REACTOR BUILDING ISOLATION VALVES

PENETRATION DATA					COMPONENT DATA					VALVE POSITION(3)					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria(4)	System/ Service	Line Size (in.)	Component(6)	Valve Type(1)	Act Type(2)	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
1	7-22	57	Main	36	CV-2691	GLB	AO	OUT	OUT	EFIC	O	C	C	YES	N
	10-1	ODB	Stean from							A/B					
	M206-1,2		SG E24A	6	PSV-2699	REL	SA	OUT	OUT		C				N
	M204-6		to EFW &	6	PSV-2698	REL	SA	OUT	OUT		C				N
			Main	6	PSV-2697	REL	SA	OUT	OUT		C				N
			Turbine,	6	PSV-2696	REL	SA	OUT	OUT		C				N
			LPN to SG	6	PSV-2695	REL	SA	OUT	OUT		C				N
			E24A	6	PSV-2694	REL	SA	OUT	OUT		C				N
				6	PSV-2693	REL	SA	OUT	OUT		C				N
				6	PSV-2692	REL	SA	OUT	OUT		C				N
				4	CV-2666	GAT	MO	OUT	OUT		LC	LC	FAI	NO	N
				4	CV-2667	GAT	MO	OUT	OUT	EFIC A	O	O	FAI	YES	N
				1	N2-54	GLB	M	OUT	IN		LC	LC		NO	N
				8	CV-2676	GAT	MO	OUT	OUT	EFIC A	C	O	FAI	YES	N
				1	HV-158	GAT	M	OUT	OUT		O	O		NO	N
				1	MS-29A	GAT	M	OUT	OUT		O	O		NO	N
					Closed System			IN							A
2	7-22	57	Main	36	CV-2692	GLB	AO	OUT	OUT	EFIC	O	C	C	YES	N
	10-1	ODB	Stean from							A/B					
	M206-1,2		SG E24B	6	PSV-2684	REL	SA	OUT	OUT		C				N
	M204-6		to EFW &	6	PSV-2685	REL	SA	OUT	OUT		C				N
			Main	6	PSV-2686	REL	SA	OUT	OUT		C				N
			Turbine,	6	PSV-2687	REL	SA	OUT	OUT		C				N
			LPN to SG	6	PSV-2688	REL	SA	OUT	OUT		C				N
			E24B	6	PSV-2689	REL	SA	OUT	OUT		C				N
				6	PSV-2690	REL	SA	OUT	OUT		C				N
				6	PSV-2691	REL	SA	OUT	OUT		C				N
				1	N2-51	GLB	M	OUT	IN		LC	LC		NO	N
				4	CV-2617	GAT	MO	OUT	OUT	EFIC B	O	O	FAI	YES	N
				8	CV-2619	GAT	MO	OUT	OUT	EFIC B	C	O	FAI	YES	N
				1	HV-157	GAT	M	OUT	OUT		O	O		NO	N
					Closed System			IN							A

Table 5-1 (continued)

PENETRATION DATA				COMPONENT DATA						VALVE POSITION[3]					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria[4]	System/ Service	Line Size (in.)	Component[6]	Valve Type[1]	Act Type[2]	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
3	7-22	57	Main	18	CV-2680	GAT	MO	OUT	IN	EFIC A	O	C	FAI	YES	N
	10-1		Feedwater		Closed System			IN							A
	M206-1		to SG												
			E24A												
4	7-22	57	Main	18	CV-2630	GAT	MO	OUT	IN	EFIC B	O	C	FAI	YES	N
	10-1		Feedwater		Closed System			IN							A
	M206-1		to SG												
			E24B												
5	6-1	56 OBD	Rx Bldg	1 1/2	SA-20	GAT	M	OUT	IN	-	LC	LC	-	NO -	N
	M236		Spray from	8	CV-2401	GLB	MO	OUT	IN	ES-7	C	O	FAI	YES	N
			Pump A	8	BS-4A	CHK	SA	IN	IN	-	-	-	-		N
6.			Spare												A
7A	9-5	55 ODB	Pzr & RCS	3/4	SV-1818	GLB	SO	OUT	OUT	-	C	C[7]	C	YES	A,C
	M237-1		Sampling	3/4	CV-1814	GAT	MO	IN	OUT	-	C	C	FAI	YES	A,C
				3/4	CV-1816	GAT	MO	IN	OUT	-	C	C	FAI	YES	A,C
				3/4	SV-1840	GLB	SO	IN	OUT	-	C	C[7]	C	YES	A,C
				3/4	PSV-1800	REL	SA	IN	OUT	-	-	-	-	-	A,C
7B	4-1	56	Quench	1/2	CV-1845	GLB	AO	OUT	OUT	ES-2	C	C	C	YES	A,C
	9-5		Tank	1	CV-1054	GAT	MO	IN	OUT	ES-1	C	C	FAI	YES	A,C
	M230-2		Sampling												
	M237-1														
8	4-1	55 ODB	HPI	2 1/2	CV-1228	GLB	MO	OUT	IN	ES-2	C	O	FAI	YES	N
	6-1		Pumps to	2 1/2	CV-1278	GLB	MO	OUT	IN	ES-1	C	O	FAI	YES	N
	M231-3		RC Loop B	2 1/2	MU-66A	SCHK	SA	IN	IN	-	-	-	-	-	N
	M230-1		– RCP A												
9	9-3	55	MU&P	1	CV-1274	GLB	MO	OUT	OUT	ES-2	O	C	FAI	YES	A,C
	M231-2		RCP	1	CV-1273	GAT	MO	IN	OUT	ES-1	O	C	FAI	YES	A,C
			Controlled	1	CV-1272	GAT	MO	IN	OUT	ES-1	O	C	FAI	YES	A,C
			Bleed-off	1	CV-1271	GAT	MO	IN	OUT	ES-1	O	C	FAI	YES	A,C
				1	CV-1270	GAT	MO	IN	OUT	ES-1	O	C	FAI	YES	A,C
				3/4	PSV-1201	REL	SA	IN	OUT	-	-	-	-	-	A,C

Table 5-1 (continued)

PENETRATION DATA				COMPONENT DATA						VALVE POSITION[3]					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria[4]	System/ Service	Line Size (in.)	Component[6]	Valve Type[1]	Act Type[2]	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
10	9-5 M237-1	57 ODB	Stm Gen Sampling	3/4	SS-146 Closed System	GAT	M	OUT IN	OUT		LC	LC		YES	N A
11	11-3 M215	56	Gas Rdwst Vent Hdr	2 2	CV-4804 CV-4803	GLB GAT	AO MO	OUT IN	OUT OUT	ES-2 ES-1	C C	C C	C FAI	YES YES	A,C A,C
12	M-236	56 ODB	Core Flood Tank Sampling	1 1 1 3/4	CF-2 CV-2418 CV-2416 PSV-2400	GAT GLB GLB REL	M MO MO SA	OUT IN IN IN	OUT OUT OUT OUT	- - - -	LC C C -	LC C C -	- FAI FAI -	YES YES YES -	C[5] C[5] C[5] C[5]
13	4-1 6-1 M231-3 M230-1	55 ODB	HPI Pumps to RC Loop A – RCP C	2 1/2 2 1/2 2 1/2	CV-1219 CV-1284 MU-66C	GLB GLB SCHK	MO MO SA	OUT OUT IN	IN IN IN	ES-1 ES-2 -	C C -	O O -	FAI FAI -	YES YES -	N N N
14	9-3 M231-2	55 ODB	MU&P Letdown to Demin	2 1/2 2 1/2 2 1/2 3/4	CV-1221 CV-1214 CV-1216 PSV-1200	GAT GAT GAT REL	MO MO MO SA	OUT IN IN IN	OUT OUT OUT OUT	ES-2 ES-1 ES-1 -	O O O -	C[7] C[7] C[7] -	FAI FAI FAI -	YES YES YES -	A,C A,C A,C A,C
15	4-1 6-1 M231-3 M230-1	55 ODB	HPI Pumps to RC Loop B – RCP B	2 1/2 2 1/2 2 1/2	CV-1227 CV-1279 MU-66B	GLB GLB SCHK	MO MO SA	OUT OUT IN	IN IN IN	ES-2 ES-1 -	C C -	O O -	FAI FAI -	YES YES -	N N N
16	M231-1	55 ODB	MU&P RCP Seal Supply	2 1 1/2 1 1/2 1 1/2 1 1/2	CV-1206 MU-29A MU-29B MU-29C MU-29D	GAT SCHK SCHK SCHK SCHK	MO SA SA SA SA	OUT IN IN IN IN	IN IN IN IN IN	- - - - -	O - - - -	O - - - -	FAI - - - -	YES - - - -	N N N N N
17	10-1 M204-3	57	EFW to S/G A	4 4 4	CV-2670 CV-2627 CV-2660A Closed System	GAT GAT GLB	MO MO MO	OUT OUT OUT IN	IN IN IN	EFIC D EFIC C -	O O LC	O O LC	FAI FAI FAI	YES YES YES	N N N A

Table 5-1 (continued)

PENETRATION DATA					COMPONENT DATA					VALVE POSITION(3)					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria(4)	System/ Service	Line Size (in.)	Component(6)	Valve Type(1)	Act Type(2)	Loca- tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
18	-		Spare												A
19	9-47	56	SFP	8	SF-42	GAT	M	OUT	-	-	LC	LC	-	NO	A,C
	M235		Cooling	8	M-307	-	-	IN (Spectacle Blind Flange)	-	-	-	-	-	-	A,B
20	-		Spare												A
21	6-1	57	SW from	10	CV-3815	BTY	MO	OUT	OUT	ES-6	C	O	FAI	YES	N
	9-6		Bldg	-	Closed Sys	-	-	IN	-	-	-	-	-	-	N
	M210		Emerg Clg												
			Coils C&D												
22	6-1	57	SW to Bldg	10	CV-3813	GAT	MO	OUT	IN	ES-6	C	O	FAI	YES	N
	9-6		Emerg Clg	1	SV-3813	GAT	SO	OUT	IN	-	O	O	O	YES	N
	M210		Coils C&D		Closed Sys	-	-	IN	-	-	-	-	-	-	-
23	6-1	56 ODB	Rx Bldg	1 1/2	SA-21	GAT	M	OUT	IN	-	LC	LC	-	NO	N
	6-3		Spray from	8	CV-2400	GLB	MO	OUT	IN	ES-8	C	O	FAI	YES	N
	M236		Pump B	8	BS-4B	CHK	SA	IN	IN	-	-	-	-	-	N
24A	M261-3	56 ODB	RB	2	CV-7446	GLB	MO	IN	IN	-	O	O	FAI	YES	A,C
			Hydrogen	2	CV-7445	GLB	MO	OUT	IN	-	LC	LC	-	NO	A,C
			Sampling,	3/4	SV-7469	GLB	SO	OUT	IN	-	C	C(7)	C	YES	A,C
			APM	3/4	SV-7456	GLB	SO	OUT	IN	ES-4	O	C	C	YES	A,C
				-	Pipe Cap	-	-	OUT	-	-	-	-	-	-	-
				3/4	SV-7479	GLB	SO	OUT	IN	ES-3	O	C	C	YES	A,C
				-	Closed Sys	-	-	OUT	-	-	-	-	-	-	B
24B	M261-3	56 ODB	RB	2	CV-7450	GLB	MO	IN	OUT	-	O	O	FAI	YES	A,C
			Hydrogen	2	CV-7449	GLB	MO	OUT	OUT	-	LC	LC	-	NO	A,C
			Sampling	3/4	SV-7459	GLB	SO	OUT	OUT	-	C	C(7)	C	YES	A,C
				-	Closed Sys	-	-	OUT	-	-	-	-	-	-	B
25	M261-3	56	RB APM	1	CV-7453	GLB	MO	IN	OUT	ES-3	O	C	FAI	YES	A,C
				1	SV-7454	GLB	SO	OUT	OUT	ES-4	O	C	C	YES	A,C
				-	Pipe Cap	-	-	OUT	-	-	-	-	-	-	-

Table 5-1 (continued)

PENETRATION DATA				COMPONENT DATA						VALVE POSITION(3)					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria(4)	System/ Service	Line Size (in.)	Component(6)	Valve Type(1)	Act Type(2)	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
26	6-1	55 ODB	LPI from	10	CV-1401	GAT	MO	OUT	IN	ES-3	C	O	FAI	YES	N
	M232		DHP A	12	DH-13A	CHK	SA	IN	IN	-	-	-	-	-	N
	M230-1			8	DH-17	CHK	SA	IN	IN	-	-	-	-	-	N
27	6-1	55 ODB	DHR	12	CV-1404	GAT	MO	OUT	OUT	-	C	C(7)	FAI	YES	N
	M232		Letdown	3/4	PSV-1403	REL	SA	IN	IN	-	-	-	-	-	N
				12	CV-1410	GAT	MO	IN	OUT	-	C	C(7)	FAI	YES	N
28	-		Spare												A
29	-		Spare												A
30	-		Spare												A
31	4-1	56 ODB	MU&P and	1	MU-35A	GLB	M	OUT	IN	-	LC	LC	-	NO	A,C
	M236		Nitrogen	1	N2-3	GLB	M	OUT	IN	-	LC	LC	-	NO	A,C
	M230-1		Supply to	1	N2-61	GAT	M	IN	IN	-	LC	LC	-	NO	A,C
			CFT T2A	1	MU-36A	CHK	SA	IN	IN	-	-	-	-	-	A,C
32	M236	56 ODB	MU&P and	1	MU-35B	GLB	M	OUT	IN	-	LC	LC	-	NO	A,C
			Nitrogen	1	N2-5	GLB	M	OUT	IN	-	LC	LC	-	NO	A,C
			Supply to	1	MU-36B	CHK	SA	IN	IN	-	-	-	-	-	A,C
			CFT T2B												
33	4-1	55 ODB	Decay	1 1/2	CV-1416	GLB	MO	OUT	IN	-	LC	LC(7)	FAI	YES	A,C
	M232-1		Heat	1 1/2	DH-96	CHK	SA	IN	IN	-	-	-	-	-	A,C
	M230-1		Pressurize												
			r Spray												
			Line												
34	6-1	55 ODB	HPI	2 1/2	CV-1220	GLB	MO	OUT	IN	ES-1	C	O	FAI	YES	N
	M231-1,3		Pumps to	2 1/2	CV-1233	GAT	MO	OUT	IN	ES-1	O	C	FAI	YES	N
			RC Loop A	2 1/2	CV-1234	GAT	MO	OUT	IN	ES-2	O	C	FAI	YES	N
			– RCP D	2 1/2	MU-66D	SCHK	SA	IN	IN	-	-	-	-	-	N
				2 1/2	CV-1285	GLB	MO	OUT	IN	ES-2	C	O	FAI	YES	N
35	-		Spare												A

Table 5-1 (continued)

PENETRATION DATA					COMPONENT DATA					VALVE POSITION[3]					
Pen No.	SAR Fig No. P&ID No.	General Design Criteria[4]	System/ Service	Line Size (in.)	Component[6]	Valve Type[1]	Act Type[2]	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
36	6-1	55 ODB	LPI from	10	CV-1400	GAT	MO	OUT	IN	ES-4	C	O	FAI	YES	N
	M232-1		DHP B	12	DH-13B	CHK	SA	IN	IN	-	-	-	-	-	N
	M230-1			8	DH-18	CHK	SA	IN	IN	-	-	-	-	-	N
37	-		Spare												A
38	-		Spare												A
39	4-1	56	Condensate to	3	CV-1065	GAT	AO	OUT	IN	ES-5/6	C	C	C	YES	A,C
	M230-2		Quench Tank	3	CS-26	CHK	SA	IN	IN	-	-	-	-	-	A,C
40	9-16	56	Fire Water	3	CV-5612	GAT	MO	IN	IN	ES-3	C	C	FAI	YES	C[5]
	M219-1		to Rx Bldg	3	CV-5611	GAT	MO	OUT	IN	ES-4	C	C	FAI	YES	C[5]
41	9-4	56 ODB	Nitrogen	1	N2-32	CHK	SA	IN	IN	-	-	-	-	-	A,C
	M233		Supply	1	N2-47	GAT	M	OUT	IN	-	LC	LC	-	NO	A,C
42	M220-3	56	Plant	3	PH-19	GAT	M	IN	OUT	-	LC	LC	-	NO	A,C
			Heating	3	PH-20	GAT	M	OUT	OUT	-	LC	LC	-	NO	A,C
43	9-14	56 ODB	Breathing	1	BA-140	GAT	M	IN	IN	-	LC	LC	-	NO	C[5]
	M218-5		Air to Rx	1	BA-141	GAT	M	OUT	IN	-	LC	LC	-	NO	C[5]
			Bldg												
44	-		Spare												A
45	-		Spare												A
46	9-14	56 ODB	Instrument	1 1/2	IA-15	GAT	M	IN	IN	-	LC	LC	-	NO	C[5]
	M218-4		Air	1	IA-37	GAT	M	OUT	IN	-	LC	LC	-	NO	C[5]
47	9-7	56	ICW to	3	CV-2235	GAT	MO	OUT	IN	ES-5/6	O	C	FAI	YES	A,C
	M234-1		CRDs	3	ICW-30	CHK	SA	IN	IN	-	-	-	-	-	A,C
48	M220-3	56	Plant	3	PH-17	GAT	M	OUT	IN	-	LC	LC	-	NO	A,C
			Heating	3	PH-18	GAT	M	IN	IN	-	LC	LC	-	NO	A,C

Table 5-1 (continued)

PENETRATION DATA					COMPONENT DATA						VALVE POSITION[3]				
Pen No.	SAR Fig No. P&ID No.	General Design Criteria[4]	System/ Service	Line Size (in.)	Component[6]	Valve Type[1]	Act Type[2]	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
49	M236	-	ILRT Test	3	Blind Flange	-	-	IN	-	-	-	-	-	-	B[5]
			Connectio	-	Closed Sys	-	-	OUT	-	-	-	-	-	-	B[5]
			n												
50	-		Spare												A
51	M222-1	56	CW to RB	6	CV-6202	GAT	AO	OUT	IN	ES-5/6	O	C	C	YES	A,C
			Clg Coils	6	AC-60	CHK	SA	IN	IN	-	-	-	-	-	A,C
52	9-7	56	ICW to	8	CV-2234	GAT	AO	OUT	IN	ES-5/6	O	C	FAI	YES	A,C
	M234-1		RCPs	8	ICW-26	CHK	SA	IN	IN	-	-	-	-	-	A,C
53A	M261-3	56 ODB	RB	3/4	SV-7467	GLB	SO	OUT	IN	-	C	C[7]	C	YES	A,C
			Hydrogen	2	CV-7444	GLB	MO	IN	IN	-	O	O	FAI	YES	A,C
			Sampling	-	Closed Sys	-	-	OUT	-	-	-	-	-	-	B
53B	M261-3	56 ODB	RB	2	CV-7448	GLB	MO	IN	OUT	-	O	O	FAI	YES	A,C
			Hydrogen	3/4	SV-7457	GLB	SO	OUT	OUT	-	C	C[7]	C	YES	A,C
			Sampling	-	Closed Sys	-	-	OUT	-	-	-	-	-	-	B
54	9-7	56	ICW to	8	ICW-114	CHK	SA	IN	IN	-	-	-	-	-	A,C
	M234-2		Letdown &	8	CV-2233	GAT	AO	OUT	IN	ES-5/6	O	C	FAI	YES	A,C
			Seal Clrs												
55	6-1	57	SW to RB	10	CV-3812	GAT	MO	OUT	IN	ES-5	C	O	FAI	YES	N
	9-6		Cooling	1	SV-3812	GAT	SO	OUT	IN	-	O	O	O	YES	N
	M210		Coils A&B	-	Closed Sys	-	-	IN	-	-	-	-	-	-	N
56	-		Spare												A
57	-		Spare												A
58	7-22	56 ODB	SG A	4	HV-150	GAT	M	IN	OUT	-	LC	LC	-	NO	A,C
	M206-1		Blowdown	4	HV-151	GAT	M	OUT	OUT	-	LC	LC	-	NO	A,C
				3/4	PSV-2600	REL	SA	IN	OUT	-	-	-	-	-	A,C

Table 5-1 (continued)

PENETRATION DATA					COMPONENT DATA					VALVE POSITION(3)						
Pen No.	SAR Fig No. P&ID No.	General Design Criteria(4)	System/ Service	Line Size (in.)	Component(6)	Valve Type(1)	Act Type(2)	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
59	M222-1	56	CW from RB Cooling Coils	6 6 3/4	CV-6205 CV-6203 PSV-6205	GAT GAT REL	MO AO SA	IN OUT IN	OUT OUT OUT	ES-5 ES-6 -	O O -	C C -	FAI C -	YES YES -	A,C A,C A,C
60	9-7 M234-1	56	ICW from CRDs and RCPs	8 8 3/4	CV-2221 CV-2220 PSV-2203	GAT GAT REL	MO MO SA	IN OUT IN	OUT OUT OUT	ES-6 ES-5 -	O O -	C C -	FAI FAI -	YES YES -	A,C A,C A,C
61	-		Spare												A
62	9-7 M234-2	56	ICW from LD & Seal Coolers	8 8 3/4	CV-2215 CV-2214 PSV-2206	GAT GAT REL	MO AO SA	IN OUT IN	OUT OUT OUT	ES-6 ES-5 -	O O -	C C -	FAI FAI -	YES YES -	A,C A,C A,C
63	6-1 9-6 M210	57	SW from Clg Coils A&B	10 -	CV-3814 Closed Sys	BTY -	MO -	OUT IN	OUT -	ES-5 -	C -	O -	FAI -	YES -	N N
64	7-22 M206-1	56 ODB	SG B Blowdown	4 4 3/4	HV-139 HV-140 PSV-2601	GAT GAT REL	M M SA	IN OUT IN	OUT OUT OUT	- - -	LC LC -	LC LC -	- - -	NO NO -	A,C A,C A,C
65	10-1 M204-3	57	EFW to SG E24B	4 4 4	CV-2620 CV-2626 CV-2660B Closed System	GAT GAT GLB	MO MO MO	OUT OUT OUT IN	IN IN IN	EFIC C EFIC D -	O O LC	O O LC	FAI FAI FAI	YES YES YES	N N N A
66	6-1 M232-1	56 ODB	DHR/LPI Sump Recirc to Pump B	14 14 -	CV-1415 CV-1406 Pipe Cap	GAT GAT -	MO MO -	IN OUT OUT	OUT OUT -	- - -	O C -	O C(7) -	FAI FAI -	YES YES -	N N -
67	6-1 M232-1	56 ODB	DHR/LPI Sump Recirc to Pump A	14 14 -	CV-1414 CV-1405 Pipe Cap	GAT GAT -	MO MO -	IN OUT OUT	OUT OUT -	- - -	O C -	O C(7) -	FAI FAI -	YES YES -	N N -
68	11-2 M213-2	56 ODB	Rx Bldg Sump Drn	4 4	CV-4446 CV-4400	GAT GAT	MO AO	IN OUT	OUT OUT	ES-1 ES-4	C C	C C	FAI C	YES YES	A,C A,C

Table 5-1 (continued)

PENETRATION DATA					COMPONENT DATA						VALVE POSITION(3)				
Pen No.	SAR Fig No. P&ID No.	General Design Criteria(4)	System/ Service	Line Size (in.)	Component(6)	Valve Type(1)	Act Type(2)	Loca-tion	Flow Dir.	Isol. Sig.	Norm	Post LOCA	Pwr Flr	Pos. Ind	App J Type Test
69	11-1 M214-3	56	RCS Drains	4 4	RBD-23 RBD-24	GAT GAT	M M	IN OUT	OUT OUT	- -	LC LC	LC LC	- -	NO NO	A,C A,C
70	4-1 M230-2	56	Quench Tank Drain	3 3 3/4	CV-1053 CV-1052 PSV-1004	GAT GLB REL	MO AO SA	IN OUT IN	OUT OUT OUT	ES-3 ES-2 -	C C -	C C -	FAI C -	YES YES -	A,C A,C A,C
V1	5-7 M261-1 M218-1	56	RB HVAC Supply Instrument Air Supply	24 24 1 1	CV-7404 CV-7402 IA-823 IA-824	BTY BTY GAT CHK	AO AO M SA	IN OUT OUT IN	IN IN IN IN	ES-3 ES-4 - -	LC LC LC -	LC LC LC -	C C - -	YES YES - -	A,C A,C C C
V2	5-7 M261-1	56	RB HVAC Return	24 24	CV-7403 CV-7401	BTY BTY	AO AO	IN OUT	OUT OUT	ES-3 ES-4	LC LC	LC LC	C C	YES YES	A,C A,C
C1		56	Rx Bldg Access	15'6"	Equipment Hatch	-	-	-	-	-	-	-	-	-	A,B
C2		56	Rx Bldg Access	6'0"	Emergency Escape	-	-	-	-	-	-	-	-	-	A,B
C3	9-47 M235	56	Fuel Transfer	48	-	-	-	-	-	-	-	-	-	-	A,B
C4		56	Rx Bldg Acess		Personnel Hatch	-	-	-	-	-	-	-	-	-	A,B
C5			Spare												A
C6			Spare												A
E1,3-14, 21,23-29, 33-36,50- 55,57-63, 66,67		56	Electrical Penetration												A,B
E2,22,30- 32,41-45, 56,64,65,6 8-73		56	Spares												A
E74		56	Ground Rod												A

Table 5-1 (continued)

Codes and Symbols

(1) Valve Type

The following codes are used to identify value type:

CHK - Check
GAT - Gate
GLB - Globe
BTY - Butterfly
BAL - Ball
REL - Relief
SCHK - Stop Check
BCK - Ball Check

(2) Actuator Type

The following codes are used to identify valve actuator types:

AO - Air Operator
SO - Solenoid Operator
MO - Motor Operator
M - Manual
SA - Self-Actuated

(3) Valve Position

The following codes are used to identify different valve positions:

O - Open
C - Closed
FAI - Fail As Is
LC - Locked Closed

Notes

(4) Design Criteria

This column references the appropriate General Design Criteria of 10 CFR 50, Appendix A. ANO-1's construction permit was issued prior to May 21, 1971 and compliance with the explicit requirements of the GDCs is not required as long as the intent of the GDCs are met (0CNA040508).

(5) These components are not tested during the Type A test. A penalty is accessed against the Type A test results using LLRT data.

(6) Vents, drains and other connections that are not connected to another process system are not included as isolation barriers although they are a part of the isolation boundary. Where closed systems are identified as a barrier, the individual valves that form part of the closed system boundary are not listed separately.

(7) Valve positions shown are immediately following automatic actuation/isolation signals from Engineered Safeguards. These valves will/may be subsequently opened by the operator

(8) DELETED

(9) DELETED

Table 5-2

TYPICAL MISSILES

Missiles	Nature	Weight lbs	Mass/Area $(lb\text{-}sec^2)/in^3 \times 10^{-2}$	Cross Sectional Shape	Point of Impact	Impact Velocity ft/sec	Where Originated	How Originated
1) Plank 4”x12”x12’-0”	Exterior	108	0.58	Rectangular 4” x 12”	Exterior surfaces	440		By tornado
2) Automobile	Exterior	4000	0.36	Rectangular 3’-0 x 6’-6”	Exterior surfaces (25’ above ground max)	73.5		By tornado
3) Control rod assembly	Interior	1000	4.05	D = 9”	Removable slab @ El. 424’6” (above reactor)	90	Reactor	By accident
4) 1 ½” valve bonnet stud	Interior	2	6.48	D = 1”	Secondary shield wall	73.5	Steam generator	By accident
5) Heater bundle stud	Interior	25	0.93	D = 3”	Secondary shield wall	73.5	Pressurizer	By accident
6) 10” electric motor operated valve stem	Interior	50	4.18	D = 2”	Concrete slab @ El. 357’-0” (above vessel)	130.0	Reactor vessel outlet line to L.P. System	By accident
7) 3” steel pipe schd. 40, 10’ long	Exterior	75.8	2.04	D = 3.5”	Exterior Surfaces	147		By tornado

Table 5-3

LINER PLATE CALCULATIONS - RESULTS

Case	Nominal Plate Thickness (in)	Initial Inward Displacement (in)	Anchor Spacing L_1 (in)	Anchor Spacing L_1 (in)	Factor of Safety Against Failure
I	.25	.125	15	15	37.0
II	.25	.125	15	15	19.4
III	.25	.125	15	15	9.9
IV	.25	.125	15	15	6.28
V	.25	.25	30	15	4.25

Table 5-4

LINER PLATE ANCHOR TESTS - RESULTS

Specimen Test No.	Spring Constant K	Max. Load K	Maximum Displacement (in)	Total Energy Available (in-lbs)
I	131.7	4.94	.313	1263
II	123.5	4.45	.625	900
III	151.5	6.16	.250	722
IV	132.7	5.97	.125	541
Average	134.9	5.38	.328	857
Predicted	82.5	5.00	.180	700

Table 5-5

RESULTS FROM FINITE ELEMENT ANALYSIS

Results From Hand Computation

Loading	Avg. Stress (psi) Conc	Avg. Stress (psi) Liner Plate	F Resultant (kips/ft)	Stress in Element 22 Concrete (psi)	23 Concrete (psi)	24 Concrete (psi)	25 Concrete (psi)	26 Concrete (psi)	27 Concrete (psi)	28 Liner Plate (psi)	28 F Resultant (kips/ft)
Dead Load	-150	-1570	-110	-188	-188	-188	-187	-187	-187	-1600	-107
Eff. Prestress	-583	-4670	-329	-587	-586	-585	-585	-584	-583	-5011	-331
Int. Pressure	-436	-3480	-246	-439	-441	-442	-444	-446	-448	-2035	-245

Table 5-6

SUMMARY OF MERIDIONAL FORCE IN WALL

LOADINGS (Kips)

A = SECTION

B = DEAD LOAD (D)

C = THERMAL OPERATING (T_O)

D = THERMAL ACCIDENT (T_A)

E = INTERNAL PRESSURE (P)

F = FULL PRESTRESS (F_1)

LOAD COMBINATIONS (Kips)

G = REDUCED PRESTRESS (F_2)

H = DESIGN EARTHQUAKE (1.25 E)

I = MAXIMUM EARTHQUAKE (E')

J = $1.05\ D + 1.0\ F_1 + 1.0\ T_A + 1.5\ P$

K = $1.05\ D + 1.0\ F_2 + 1.0\ T_A + 1.5\ P$

L = $1.05\ D + 1.0\ F_1 + 1.0\ T_A + 1.25\ P + 1.25\ E$

M = $1.05\ D + 1.0\ F_2 + 1.0\ T_A + 1.25\ P + 1.25\ E$

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	- 92	-51	-106	250	-331	-331	±95	±109	-159	-159	-316	-316
15-21	-103	-51	-106	250	-331	-331	±114	±128	-170	-170	-347	-347
29-35	-108	-51	-106	250	-331	-331	±124	±138	-175	-175	-362	-362
36-42	-111	-51	-106	250	-331	-331	±129	±144	-179	-179	-370	-370
43-49	-114	-51	-106	250	-331	-331	±134	±153	-182	-182	-378	-378
50-56	-117	-51	-106	250	-331	-331	±140	±156	-185	-185	-387	-387
57-63	-119	-51	-106	250	-331	-331	±143	±159	-187	-187	-392	-392
64-70	-120	-51	-106	250	-331	-331	±146	±162	-188	-188	-396	-396
71-77	-123	-51	-106	250	-331	-331	±150	±166	-191	-191	-404	-404
78-84	-125	-51	-106	250	-331	-331	±154	±171	-193	-193	-410	-410
85-91	-126	-51	-106	250	-331	-331	±157	±174	-194	-194	-414	-414
92-98	-128	-51	-106	250	-331	-331	±160	±176	-196	-196	-419	-419
99-105	-129	-51	-106	250	-331	-331	±162	±180	-197	-197	-422	-422
106-112	-131	-51	-106	250	-331	-331	±165	±183	-200	-200	-427	-427
113-119	-133	-51	-106	250	-331	-331	±168	±186	-202	-202	-432	-432
120-126	-135	-51	-106	250	-331	-331	±171	±189	-204	-204	-437	-437
127-133	-136	-51	-106	248	-332	-332	±173	±192	-209	-209	-444	-444
134-140	-138	-51	-106	224	-333	-333	±175	±196	-218	-218	-454	-454
141-147	-141	-51	-107	241	-334	-334	±176	±200	-228	-228	-464	-464
148-154	-193	-52	-107	237	-333	-333	±179	±204	-235	-235	-473	-473
155-161	-146	-52	-108	233	-330	-330	±180	±206	-242	-242	-480	-480

Table 5-7

SUMMARY OF MERIDIONAL MOMENT IN WALL

LOADINGS (Kips)

A = SECTION

B = DEAD LOAD (D)

C = THERMAL OPERATING (T_O)

D = THERMAL ACCIDENT (T_A)

E = INTERNAL PRESSURE (P)

F = FULL PRESTRESS (F_1)

LOAD COMBINATIONS (Kips)

G = REDUCED PRESTRESS (F_2)

H = DESIGN EARTHQUAKE (1.25 E)

I = MAXIMUM EARTHQUAKE (E')

J = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.5 P

K = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.5 P

L = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.25 P + 1.25 E

M = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.25 P + 1.25 E

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	-5	-431	-855	15	-22	-22	-	-	-860	-860	-864	-864
15-21	-5	-432	-850	17	-25	-18	-	-	-855	-848	-859	-852
29-35	-5	-434	-842	15	-24	-12	-	-	-849	-837	-853	-841
36-42	-5	-434	-840	12	-20	-10	-	-	-849	-839	-850	-840
43-49	-6	-434	-840	6	-13	-9	-	-	-850	-846	-852	-848
50-56	-6	-434	-843	-2	-2	-14	-	-	-854	-866	-854	-866
57-63	-7	-431	-847	-27	30	-9	-	-	-865	-904	-858	-897
64-70	-7	-434	-852	-26	30	3	-	-	-868	-895	-862	-889
71-77	-9	-438	-861	-25	30	14	±25	±27	-878	-894	-897	-913
78-84	-12	-445	-885	-52	68	41	±49	±53	-908	-935	-944	-971
85-91	-16	-454	-912	-75	102	63	±54	±59	-939	-978	-975	-1014
92-98	-16	-458	-913	-56	79	58	±58	±64	-935	-956	-979	-1000
99-105	-19	-472	-938	-49	75	71	±63	±70	-956	-960	-1007	-1011
106-112	-22	-487	-959	-27	51	58	±83	±93	-972	-965	-1048	-1041
113-119	-25	-505	-977	15	3	17	±104	±115	-978	-965	-1086	-1073
120-126	-26	-522	-983	81	-77	-60	±42	±46	-966	-949	-1028	-1011
127-133	-12	-554	-1006	145	-155	-129	±20	±22	-956	-930	-972	-946
134-140	25	-569	-1001	171	-198	-170	±91	±100	-916	-888	-868	-840
141-147	68	-572	-961	216	-283	-253	±162	±179	-849	-819	-741	-711
148-154	114	-560	-885	283	-416	-383	±346	±480	-757	-724	-348	-315
155-161	165	-524	-756	371	-579	-545	±525	±580	-605	-571	-173	-139

Table 5-8

SUMMARY OF RADIAL SHEAR FORCE IN WALL

LOADINGS (Kips)	LOAD COMBINATIONS (Kips)
A = SECTION	G = REDUCED PRESTRESS (F_2)
B = DEAD LOAD (D)	H = DESIGN EARTHQUAKE (1.25 E)
C = THERMAL OPERATING (T_O)	I = MAXIMUM EARTHQUAKE (E')
D = THERMAL ACCIDENT (T_A)	J = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.5 P
E = INTERNAL PRESSURE (P)	K = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.5 P
F = FULL PRESTRESS (F_1)	L = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.25 P + 1.25 E
	M = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.25 P + 1.25 E

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	0	-1	1	0	0	0	-	-	1	1	1	1
15-21	0	-1	2	0	0	1	-	-	2	3	2	3
29-35	0	0	1	-1	1	1	-	-	0	2	1	1
36-42	0	0	0	-2	2	1	-	-	-1	-2	-1	-2
43-49	0	0	0	-3	3	0	-	-	-2	-5	-1	-4
50-56	0	0	-2	-4	5	-3	-	-	-3	-11	-2	-10
57-63	0	0	-3	-5	6	4	-	-	-5	-7	-3	-5
64-70	-1	-1	-4	-5	7	1	-	-	-6	-12	-4	-10
71-77	-1	-2	-5	-5	7	7	-	-	-7	-7	-5	-5
78-84	-1	-3	-7	-3	5	1	-	-	-8	-12	-7	-11
85-91	-1	-4	-8	-1	3	5	-	-	-8	-6	-7	-5
92-98	-1	-5	-9	1	0	9	-	-	-8	1	-9	0
99-105	-1	-6	-10	7	-6	0	-	-	-13	-7	-8	-2
106-112	-1	-8	-9	-14	-16	-11	-	-	-6	-1	-8	-3
113-119	-1	-8	-7	24	-28	-26	±3	±4	0	2	-9	-7
120-126	0	-8	-2	37	-45	-43	±8	± 9	9	11	-9	-7
127-133	0	-7	5	49	-63	-62	±13	±14	16	17	-10	-9
134-140	1	-5	14	60	-80	-79	±24	±26	25	26	-14	-13
141-147	2	1	27	71	-110	-110	±35	±38	26	26	-27	-27
148-154	4	9	45	83	-127	-126	±53	±58	47	48	-27	-26
155-161	6	21	66	95	-142	-142	±70	±77	73	73	-21	-21

Table 5-9

SUMMARY OF HOOP FORCE IN WALL

LOADINGS (Kips)	LOAD COMBINATIONS (Kips)
A = SECTION	G = REDUCED PRESTRESS (F_2)
B = DEAD LOAD (D)	H = DESIGN EARTHQUAKE (1.25 E)
C = THERMAL OPERATING (T_O)	I = MAXIMUM EARTHQUAKE (E')
D = THERMAL ACCIDENT (T_A)	J = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.5 P
E = INTERNAL PRESSURE (P)	K = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.5 P
F = FULL PRESTRESS (F_1)	L = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.25 P + 1.25 E
	M = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.25 P + 1.25 E

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	0	7	-12	491	-726	-730	-	-	-2	-6	-124	-128
15-21	0	-4	4	496	-733	-732	-	-	15	16	-109	-108
29-35	0	-4	6	501	-740	-727	-	-	18	31	-108	-95
36-42	1	-2	9	504	-744	-721	-	-	22	45	-104	-81
43-49	1	1	12	506	-747	-710	-	-	25	62	-101	-65
50-56	2	5	17	505	-748	-694	-	-	29	83	-98	-44
57-63	3	8	21	503	-746	-681	-	-	33	98	-93	-28
64-70	3	11	24	497	-740	-673	-	-	33	100	-92	-25
71-77	4	17	28	481	-720	-653	-	-	34	101	-87	-20
78-84	4	22	28	457	-690	-624	-	-	28	94	-87	-21
85-91	3	24	25	438	-666	-602	±10	±11	19	83	-80	-16
92-98	2	26	17	411	-631	-575	±21	±23	5	61	-77	-21
99-105	2	25	-6	369	-575	-531	±31	±34	-25	19	-87	-43
106-112	-3	20	-29	320	-509	-476	±57	±63	-61	-28	-84	-51
113-119	-9	9	-70	267	-435	-412	±83	±92	-114	-91	-98	-75
120-126	-16	-11	-131	215	-359	-344	±91	±100	-184	-169	-147	-132
127-133	-23	-43	-209	185	-317	-307	±99	±109	-273	-263	-220	-210
134-140	-29	-114	-316	168	-297	-289	±169	±187	-391	-383	-264	-256
141-147	-36	-207	-454	148	-266	-261	±240	±266	-536	-531	-333	-328
148-154	-45	-289	-584	131	-231	-227	±295	±325	-666	-662	-403	-399
155-161	-56	-376	-664	123	-194	-192	±350	±385	-732	-730	-413	-411

Table 5-10

SUMMARY OF HOOP MOMENT IN WALL

LOADINGS (Kips)	LOAD COMBINATIONS (Kips)
A = SECTION	G = REDUCED PRESTRESS (F_2)
B = DEAD LOAD (D)	H = DESIGN EARTHQUAKE (1.25 E)
C = THERMAL OPERATING (T_O)	I = MAXIMUM EARTHQUAKE (E')
D = THERMAL ACCIDENT (T_A)	J = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.5 P
E = INTERNAL PRESSURE (P)	K = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.5 P
F = FULL PRESTRESS (F_1)	L = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.25 P + 1.25 E
	M = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.25 P + 1.25 E

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	0	-432	-846	49	-74	-74	-	-	-847	-847	-859	-859
15-21	0	-434	-843	50	-75	-73	-	-	-843	-841	-855	-853
29-35	0	-434	-841	50	-76	-71	-	-	-842	-837	-854	-849
36-42	0	-434	-840	49	-75	-70	-	-	-841	-836	-854	-849
43-49	0	-434	-840	48	-74	-69	-	-	-842	-837	-854	-849
50-56	0	-433	-841	46	-71	-69	-	-	-843	-841	-854	-852
57-63	0	-432	-841	38	-61	-7	-	-	-845	-851	-854	-860
64-70	0	-433	-842	39	-62	-62	-	-	-845	-845	-855	-855
71-77	0	-433	-843	38	-61	-57	-	-	-847	-843	-856	-852
78-84	-1	-434	-850	28	-47	-47	-	-	-856	-856	-863	-863
85-91	-2	-437	-857	19	-35	-40	±5	±6	-865	-870	-875	-880
92-98	-2	-437	-857	23	-39	-39	±10	±12	-863	-863	-879	-879
99-105	-3	-441	-866	20	-34	-30	±16	±18	-873	-869	-894	-890
106-112	-4	-446	-875	21	-33	-28	±21	±24	-880	-875	-886	-881
113-119	-5	-451	-884	25	-37	-31	±27	±30	-890	-884	-922	-916
120-126	-7	-460	-896	38	-51	-45	±11	±12	-897	-891	-917	-911
127-133	-5	-514	-1003	54	-71	-63	±6	±6	-998	-990	-1005	-997
134-140	4	-604	-1198	61	-84	-76	±24	±26	-1186	-1178	-1178	-1170
141-147	13	-699	-1400	72	-109	-100	±42	±47	-1387	-1378	-1363	-1354
148-154	23	-793	-1567	90	-145	-136	±89	±98	-1552	-1543	-1446	-1437
155-161	35	-855	-1579	124	-198	-188	±136	±150	-1554	-1544	-1450	-1440

Table 5-11

SUMMARY OF TANGENTIAL SHEAR IN WALL

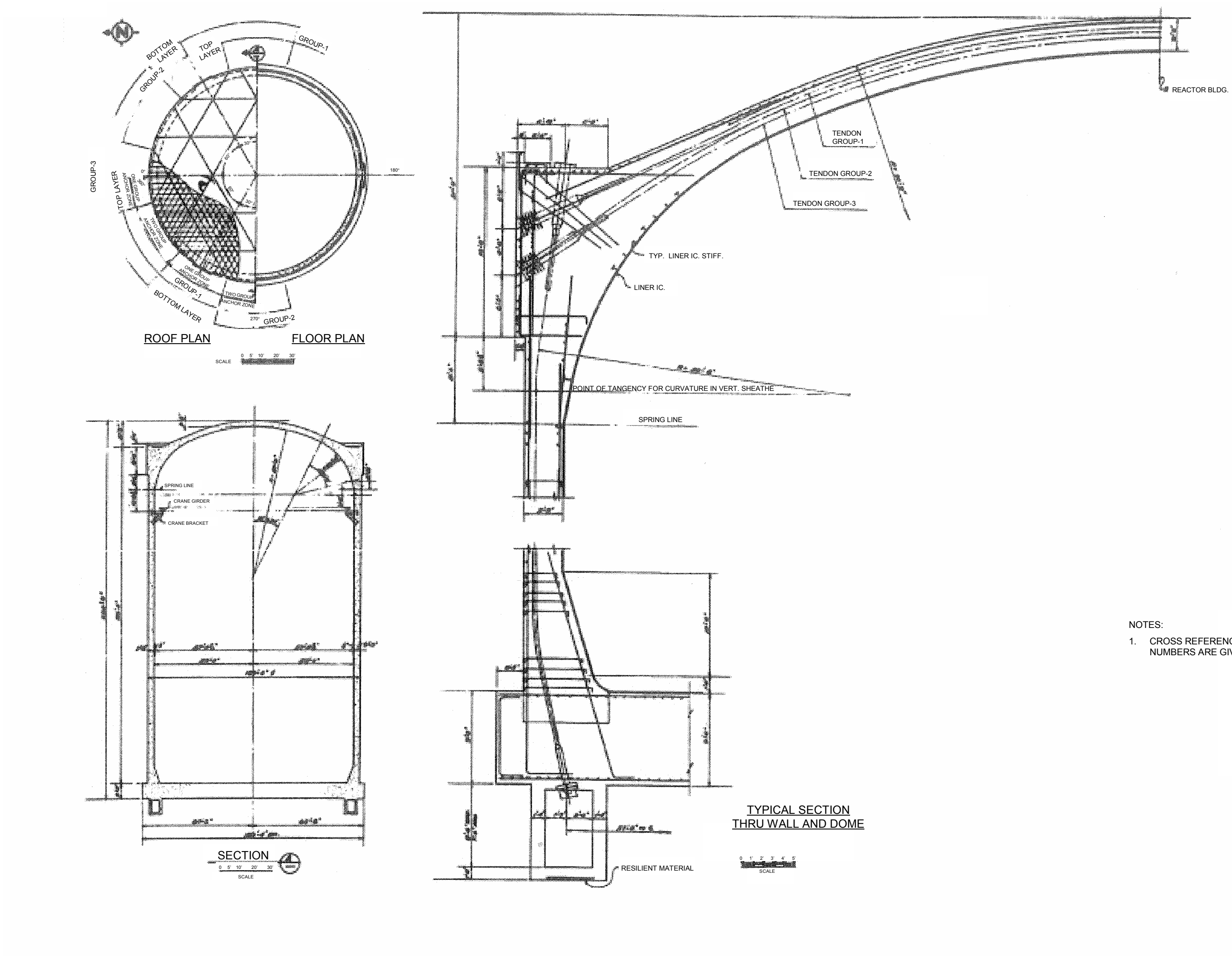
LOADINGS (Kips)	LOAD COMBINATIONS (Kips)
A = SECTION	G = REDUCED PRESTRESS (F_2)
B = DEAD LOAD (D)	H = DESIGN EARTHQUAKE (1.25 E)
C = THERMAL OPERATING (T_O)	I = MAXIMUM EARTHQUAKE (E')
D = THERMAL ACCIDENT (T_A)	J = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.5 P
E = INTERNAL PRESSURE (P)	K = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.5 P
F = FULL PRESTRESS (F_1)	L = 1.05 D + 1.0 F_1 + 1.0 T_A + 1.25 P + 1.25 E
	M = 1.05 D + 1.0 F_2 + 1.0 T_A + 1.25 P + 1.25 E

A	B	C	D	E	F	G	H	I	J	K	L	M
1-7	-	-	-	-	-	-	±66	±75	-	-	±66	±66
15-21	-	-	-	-	-	-	±66	±75	-	-	±66	±66
29-35	-	-	-	-	-	-	±76	±87	-	-	±76	±76
36-42	-	-	-	-	-	-	±76	±87	-	-	±76	±76
43-49	-	-	-	-	-	-	±76	±87	-	-	±76	±76
50-56	-	-	-	-	-	-	±76	±87	-	-	±76	±76
57-63	-	-	-	-	-	-	±76	±87	-	-	±76	±76
64-70	-	-	-	-	-	-	±76	±87	-	-	±76	±76
71-77	-	-	-	-	-	-	±76	±87	-	-	±76	±76
78-84	-	-	-	-	-	-	±76	±87	-	-	±76	±76
85-91	-	-	-	-	-	-	±76	±87	-	-	±76	±76
92-98	-	-	-	-	-	-	±76	±87	-	-	±76	±76
99-105	-	-	-	-	-	-	±76	±87	-	-	±76	±76
106-112	-	-	-	-	-	-	±76	±87	-	-	±76	±76
113-119	-	-	-	-	-	-	±79	±91	-	-	±79	±79
120-126	-	-	-	-	-	-	±79	±91	-	-	±79	±79
127-133	-	-	-	-	-	-	±79	±91	-	-	±79	±79
134-140	-	-	-	-	-	-	±79	±91	-	-	±79	±79
141-147	-	-	-	-	-	-	±79	±91	-	-	±79	±79
148-154	-	-	-	-	-	-	±79	±91	-	-	±79	±79
155-161	-	-	-	-	-	-	±79	±91	-	-	±79	±79

Table 5-12

CHANGES IN HOOP FORCE IN WALL WHEN TWO TENDONS ARE OMITTED

Section	Element Width	Change in Hoop Force per Foot (kips)	Total Change In Hoop Force (kips)
1-7	20'	±4	±80
15-21	5'	-1	-5
29-35	10'	-13	-130
36-42	5'	-23	-115
43-49	5'	-37	-185
50-56	5.25'	-54	-283.5
57-63	1.5'	-65	-97.5
64-70	3.25'	-67	-217.75
71-77	5'	-67	-335
78-84	3.25'	-66	-214.5
85-91	1.5'	-64	-96
92-98	3.25'	-56	-182
99-105	3'	-44	-132
106-112	3'	-33	-99
113-119	3'	-23	-69
120-126	3'	-15	-45
127-133	2'	-10	-20
134-140	2'	-8	-16
141-147	2'	-5	-10
148-154	2'	-4	- 8
155-161	2'	-2	- 4
Total			2184.25



Unit 1

NOTES:

1. CROSS REFERENCES M-XXXX TO FSAR FIGURE NUMBERS ARE GIVEN IN TABLE 1-5

REACTOR BUILDING STRUCTURE – TYPICAL DETAILS

SAR FIGURE NO. 5-1

ARKANSAS NUCLEAR ONE

UNIT 1

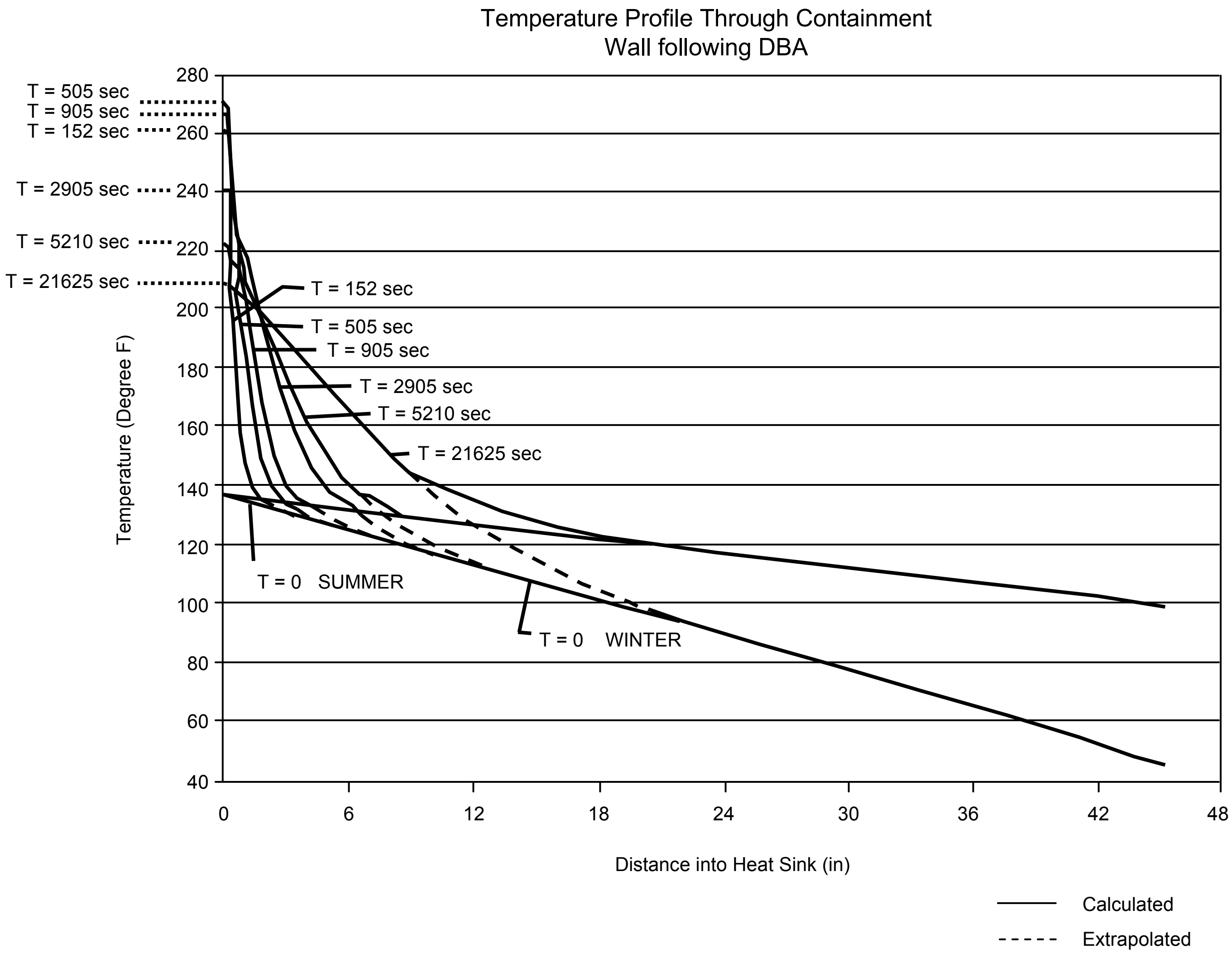
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



THERMAL TRANSIENTS DURING DESIGN BASIS ACCIDENT

SAR FIGURE NO. 5-5

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 5-6

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

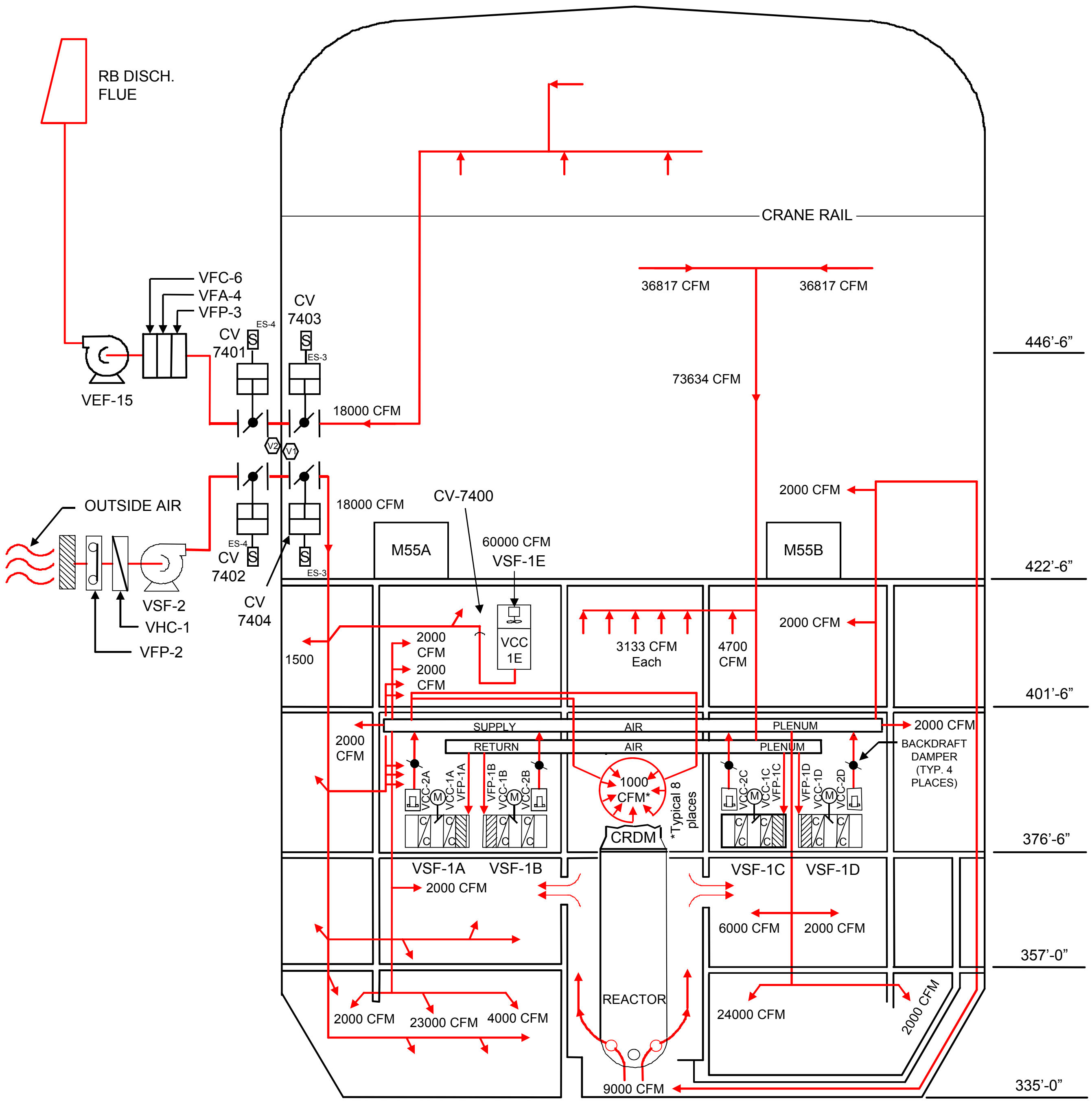
REACTOR BUILDING ISOLATION VALVE ARRANGEMENT	BASED ON DRAWING NO	SHEET	REV.
		1	

DELETED

SAR FIGURE NO. 5-6A

AMENDMENT 20

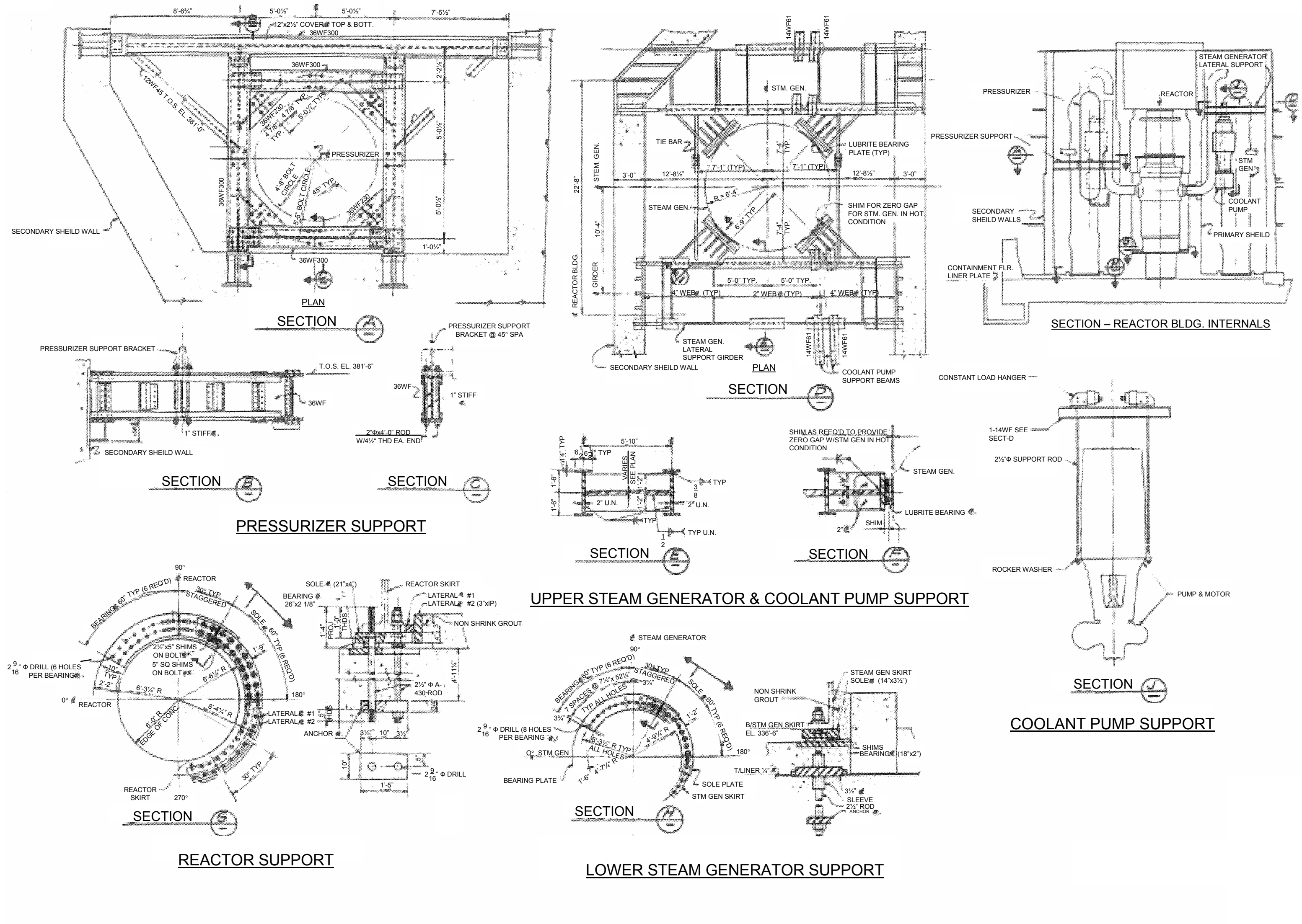
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING ISOLATION VALVE ARRANGEMENT	BASED ON DRAWING NO	SHEET 1 / REV.



SAR FIGURE NO. 5-7

AMENDMENT 28

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING HVAC	BASED ON DRAWING NO	SHEET 1 / REV.



REACTOR BUILDING INTERNALS PRIMARY EQUIPMENT SUPPORTS TYPICAL DETAILS

SAR FIGURE NO. 5-8

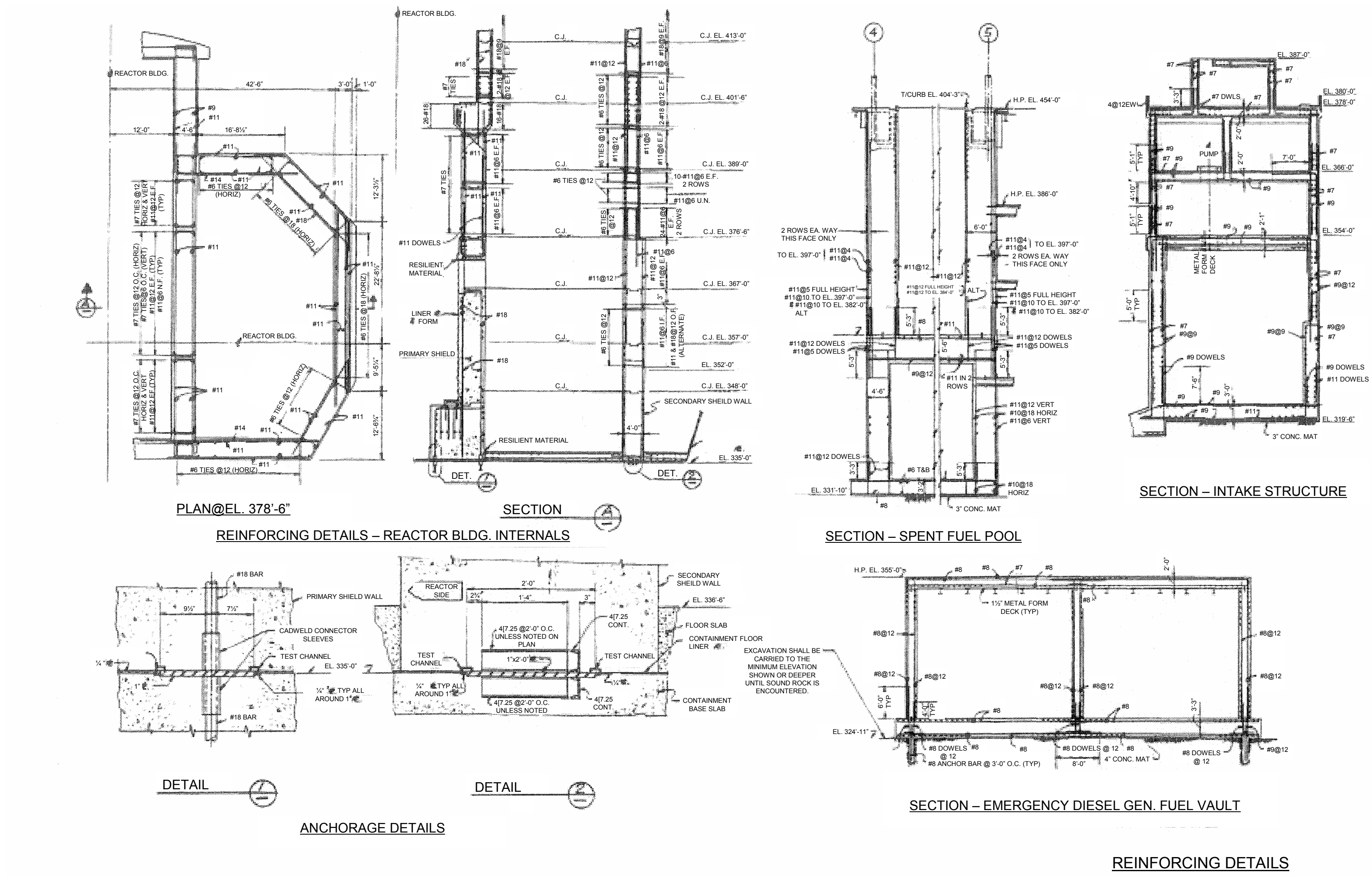
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



CLASS 1 STRUCTURES REINFORCING & ANCHORAGE TYPICAL DETAILS

SAR FIGURE NO. 5-9

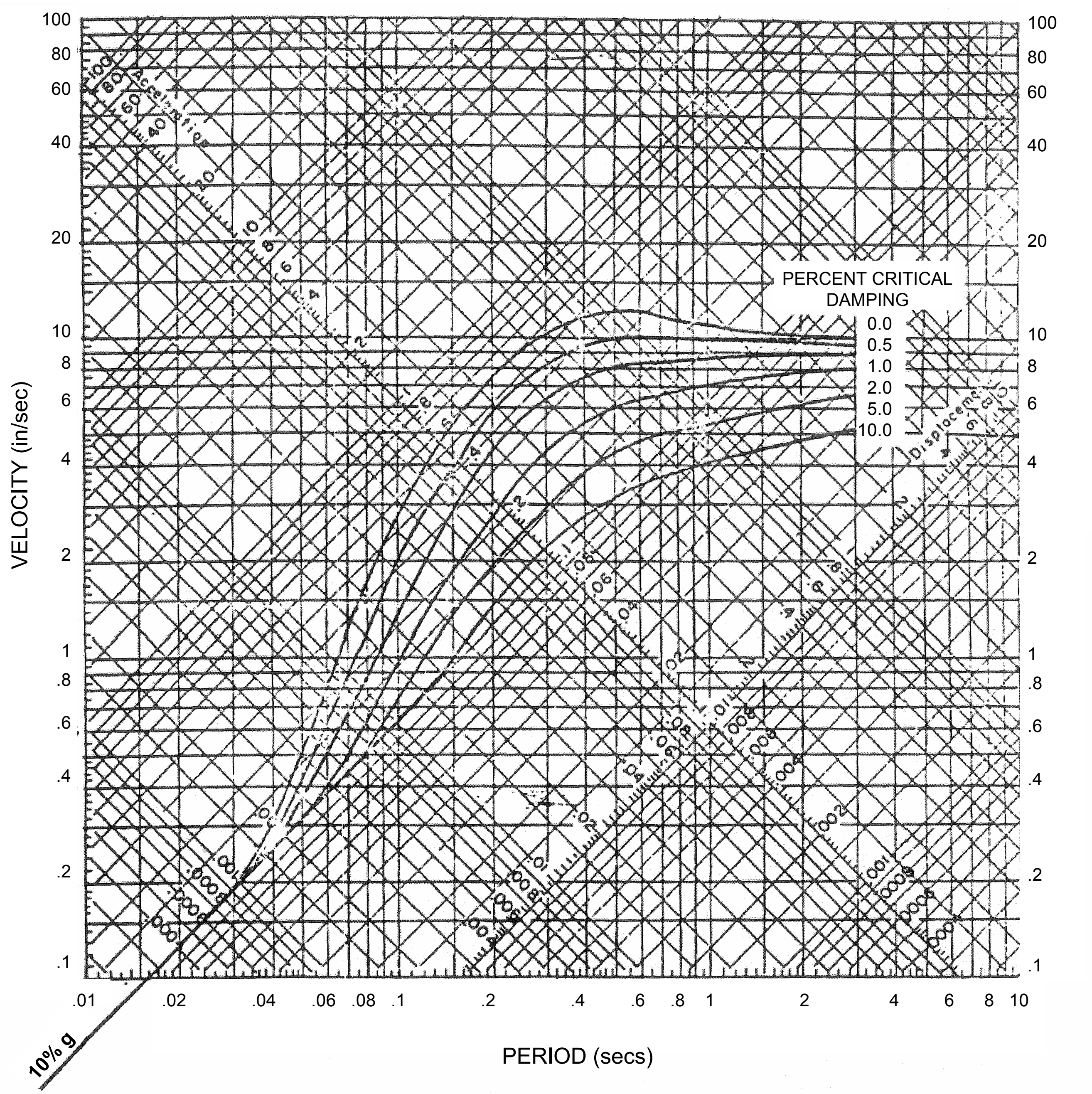
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

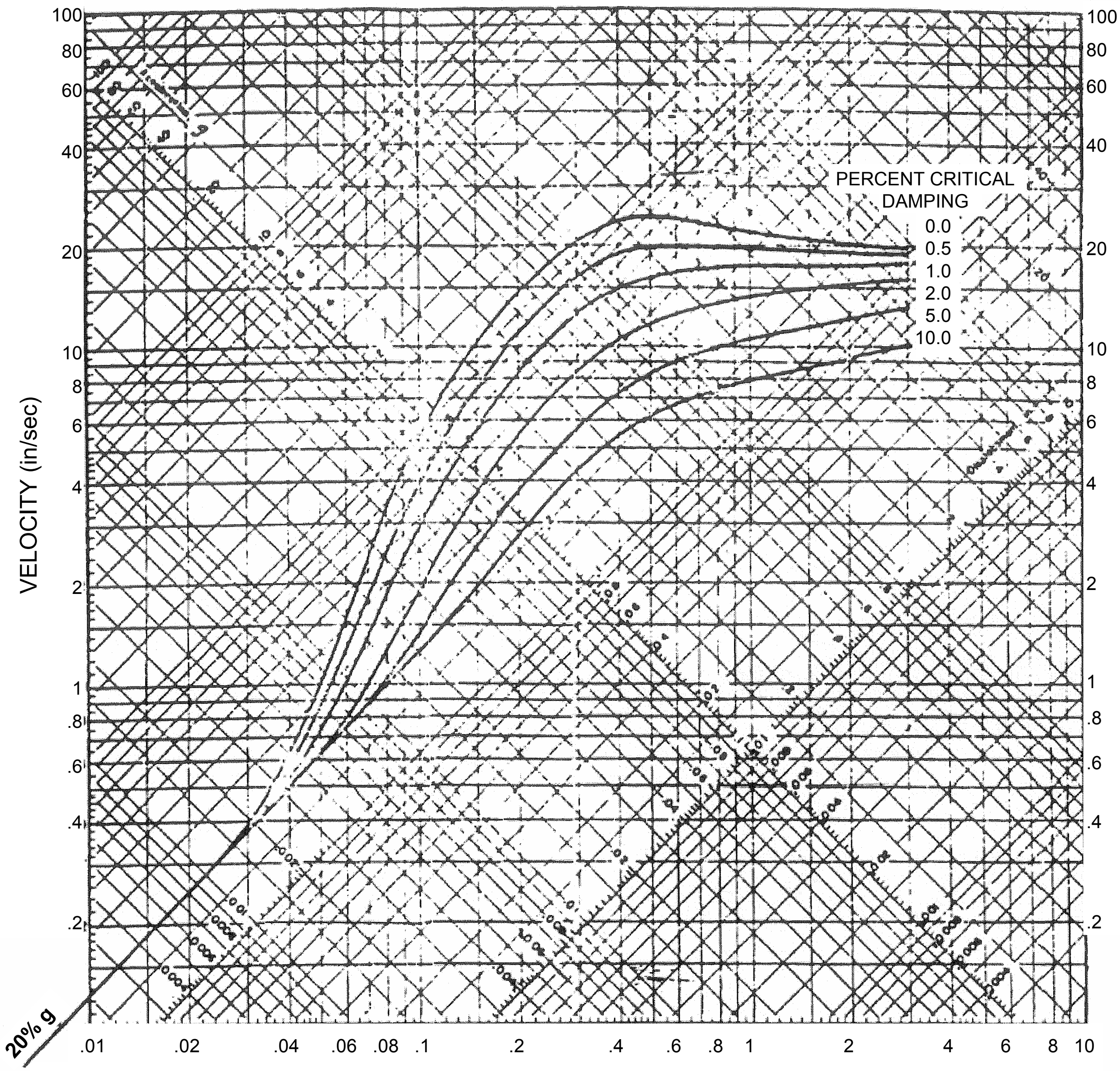
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 5-10

AMENDMENT 20

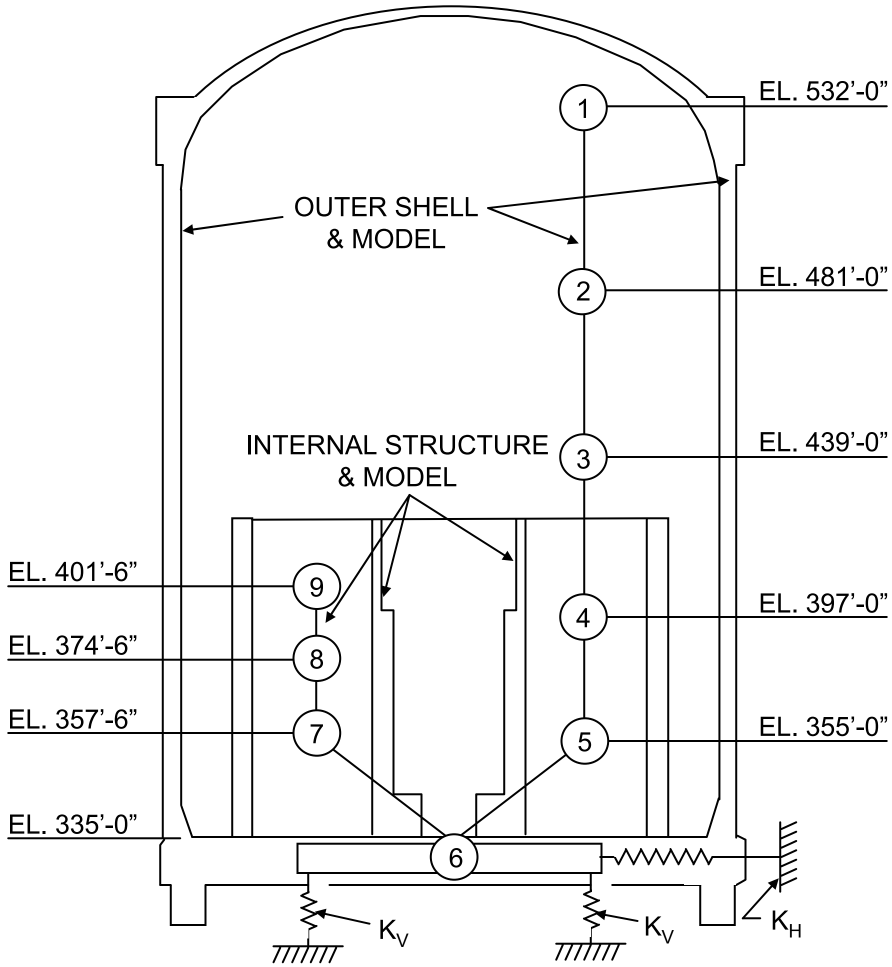
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
SPECTRUM RESPONSE CURVE DESIGN EARTHQUAKE	BASED ON DRAWING NO	SHEET 1	REV.



SAR FIGURE NO. 5-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
SPECTRUM RESPONSE CURVE MAXIMUM EARTHQUAKE	BASED ON DRAWING NO	SHEET 1 REV.



REACTOR BUILDING

MASS POINT	WEIGHT (KIPS)	MOMENT OF INERTIA (FT^4)	CROSS SECTION AREA (FT^2)	SHEAR AREA (FT^2)
1	16918			
		2530000	1410	946
2	8914			
		2530000	1410	946
3	8914			
		2530000	1410	946
4	8914			
		2530000	1410	946
5	8914			
		2530000	1410	946
6	20194			
		927000	1730	843
7	7075			
		915000	1790	742
8	7808			
		841000	1690	1040
9	9588			

$K_H = 11.82 \times 10^6$ K/FT

$K_V = 4.10 \times 10^6$ K/FT

SPRINGS SIMULATE SOIL RESPONSE

MATHEMATICAL MODEL OF THE REACTOR BUILDING

SAR FIGURE NO. 5-12

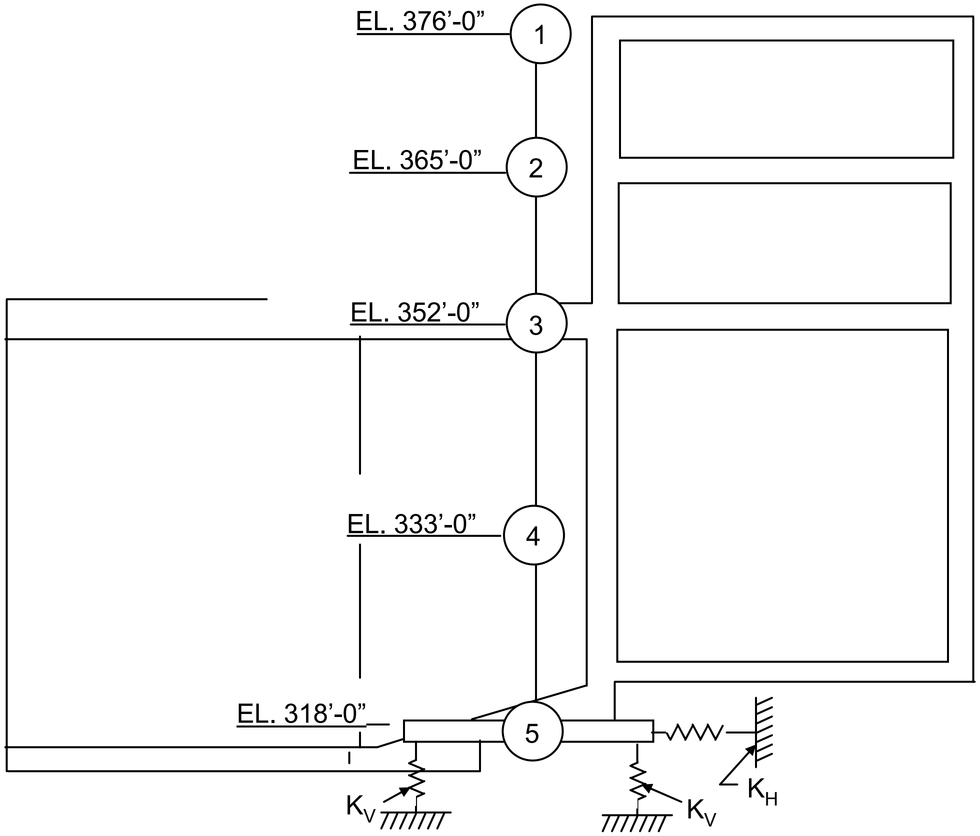
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



MASS POINT	WEIGHT (KIPS)	MOMENT OF INERTIA (FT^4)	CROSS SECTION AREA (FT^2)	SHEAR AREA (FT^2)
1	1106			
		81800	492	268
2	1310			
		81800	492	268
3	5850			
		680000	1354	370
4	2850			
		680000	1354	370
5	5060			

$K_H = 8.3 \times 10^6$ K/FT

$K_V = 3.84 \times 10^6$ K/FT

SPRINGS SIMULATE SOIL RESPONSE

INTAKE STRUCTURE

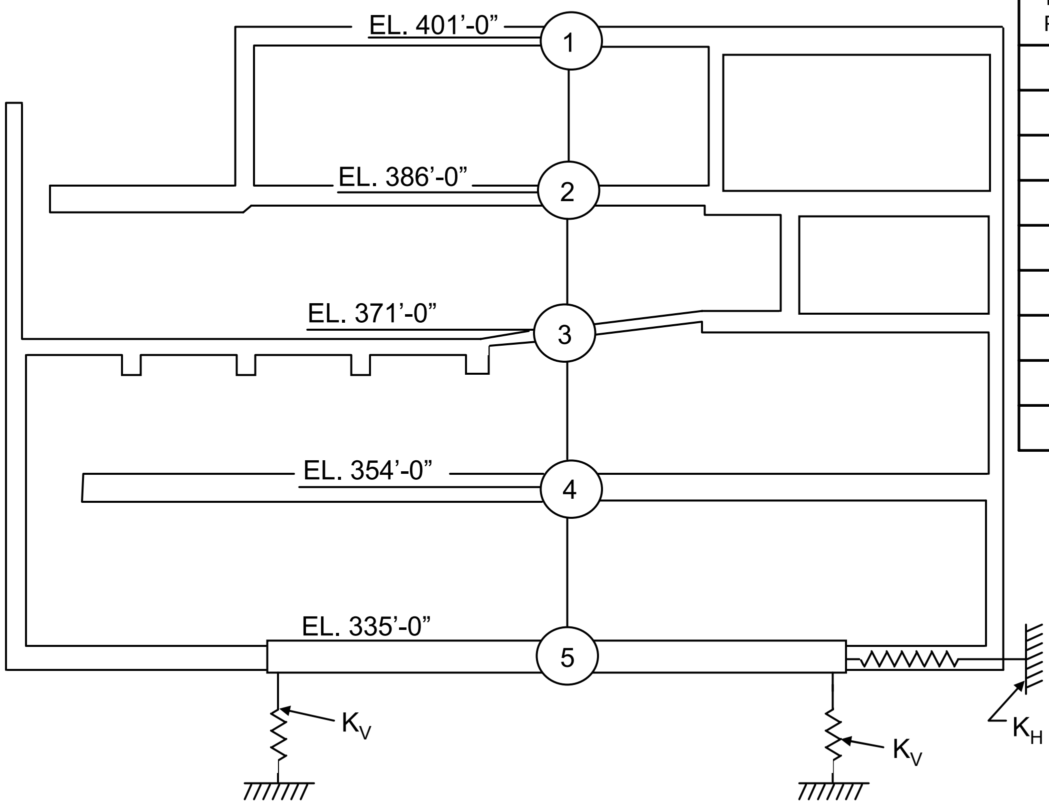
MATHEMATICAL MODEL OF THE INTAKE STRUCTURE	SAR FIGURE NO. 5-13
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



MASS POINT	WEIGHT (KIPS)	MOMENT OF INERTIA (FT^4)	CROSS SECTION AREA (FT^2)	SHEAR AREA (FT^2)
1	5790			
		1799000	2233	1049
2	9700			
		3281000	2924	1432
3	10340			
		2952000	2533	1072
4	15930			
		3691000	2362	948
5	7900			

$K_H = 13.7 \times 10^6$ K/FT

$K_V = 9.0 \times 10^6$ K/FT

SPRINGS SIMULATE SOIL RESPONSE

NOTE: MASS POINT 5 INCLUDES AREA @ EL. 317'-0"

AUXILIARY BUILDING

MATHEMATICAL MODEL OF THE AUXILIARY BUILDING

SAR FIGURE NO. 5-14

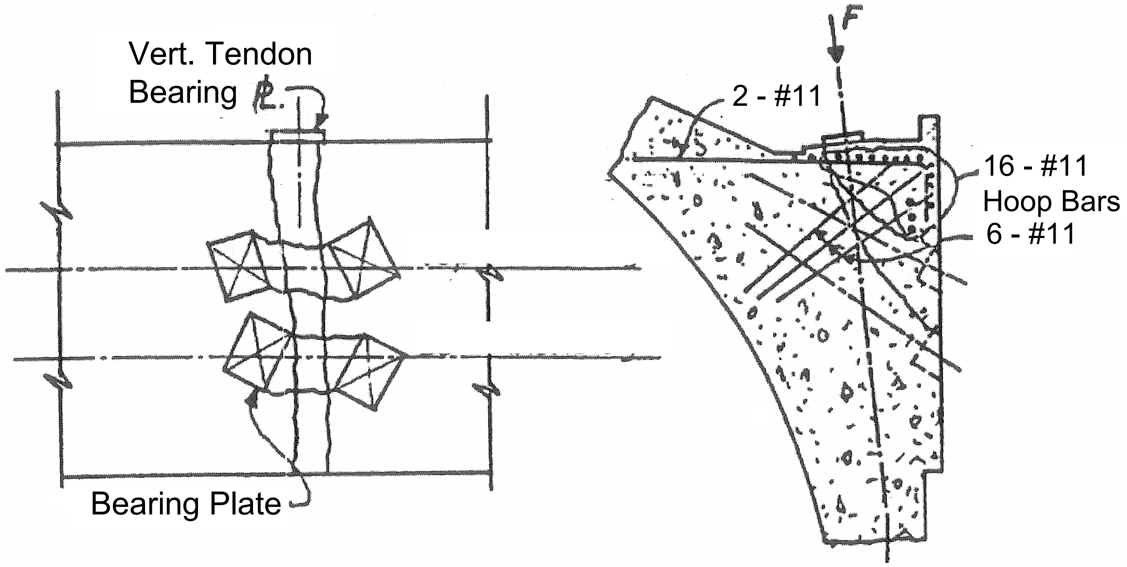
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

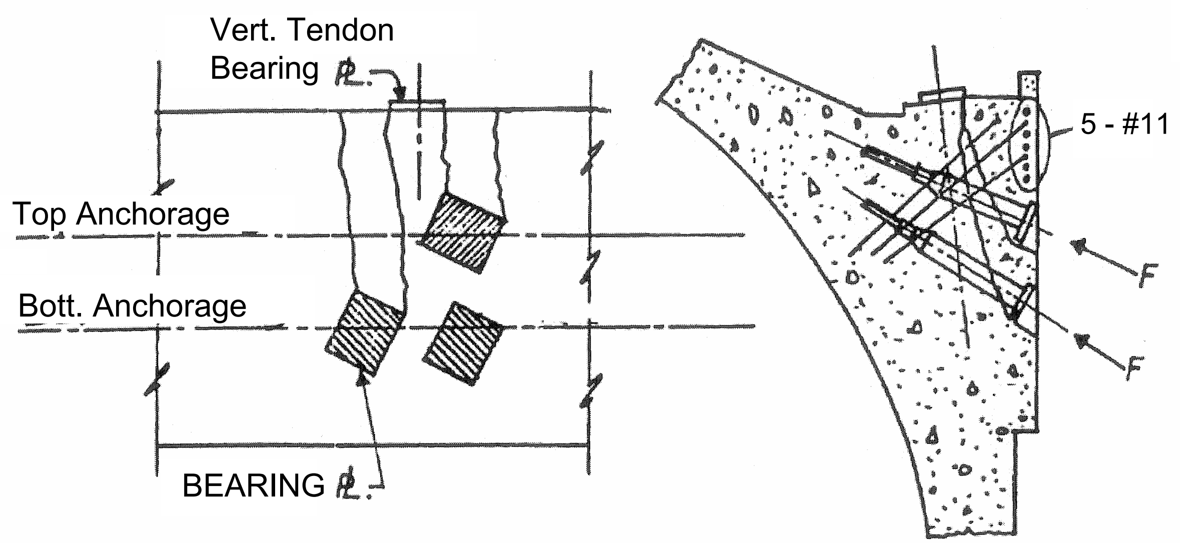
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



RING GIRDER ELEV. SECTION

POSTULATED RING GIRDER CRACK PATTERN TYPE I



RING GIRDER ELEV. SECTION

POSTULATED RING GIRDER CRACK PATTERN TYPE II

SAR FIGURE NO. 5-15

AMENDMENT 20

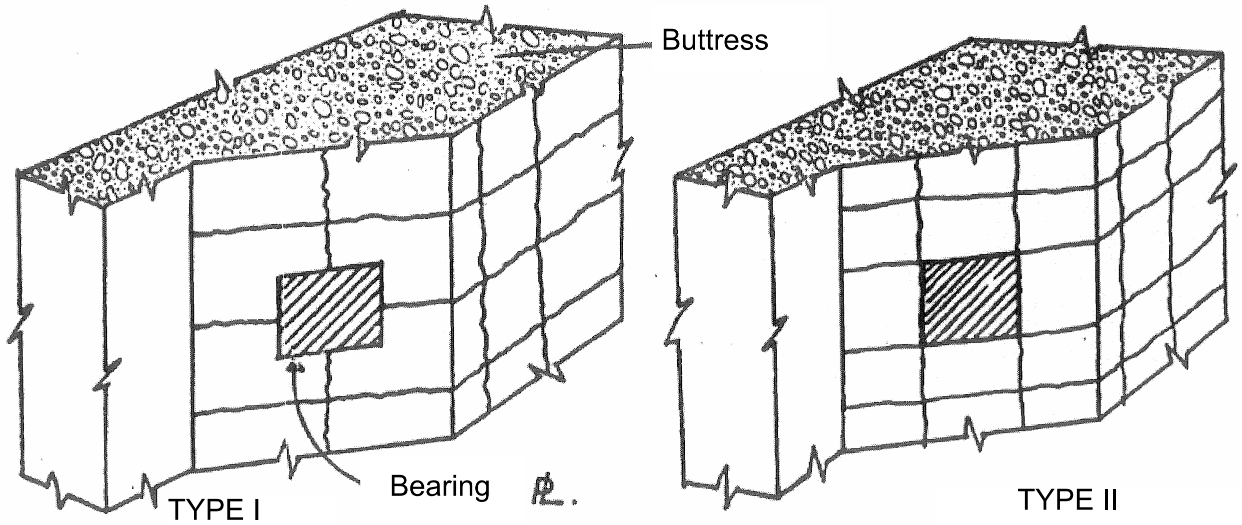
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

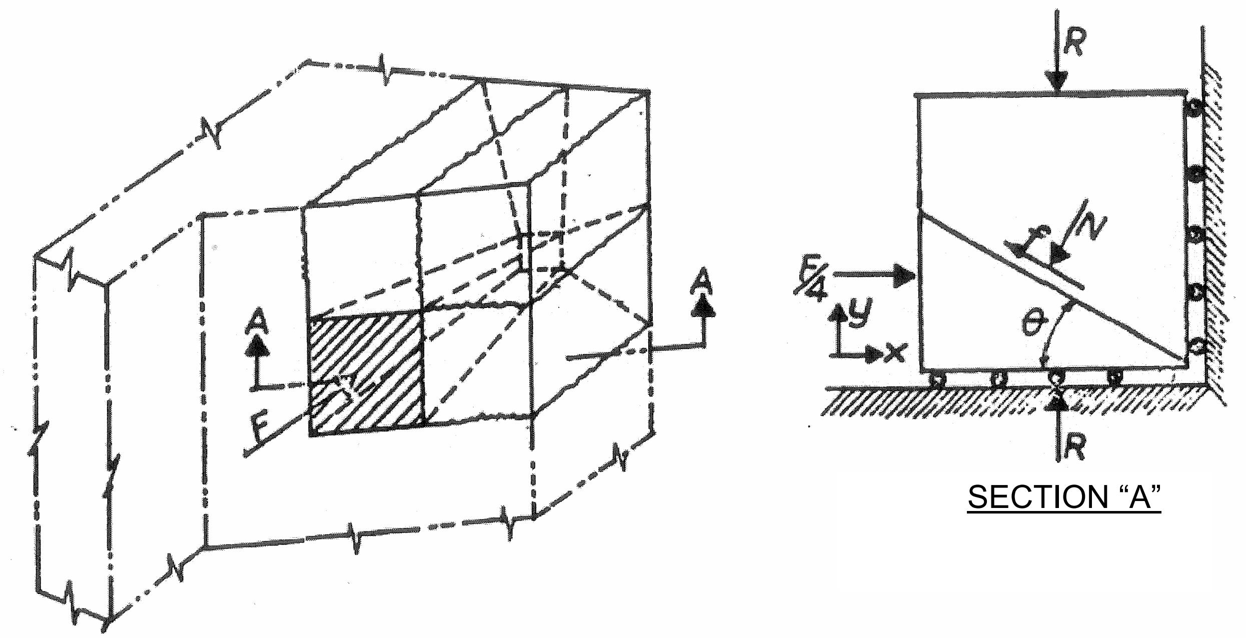
SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

POSTULATED RING GIRDER CRACK PATTERNS

BASED ON DRAWING NO	SHEET	REV.
	1	



POSTULATED BUTTRESS CRACK PATTERNS

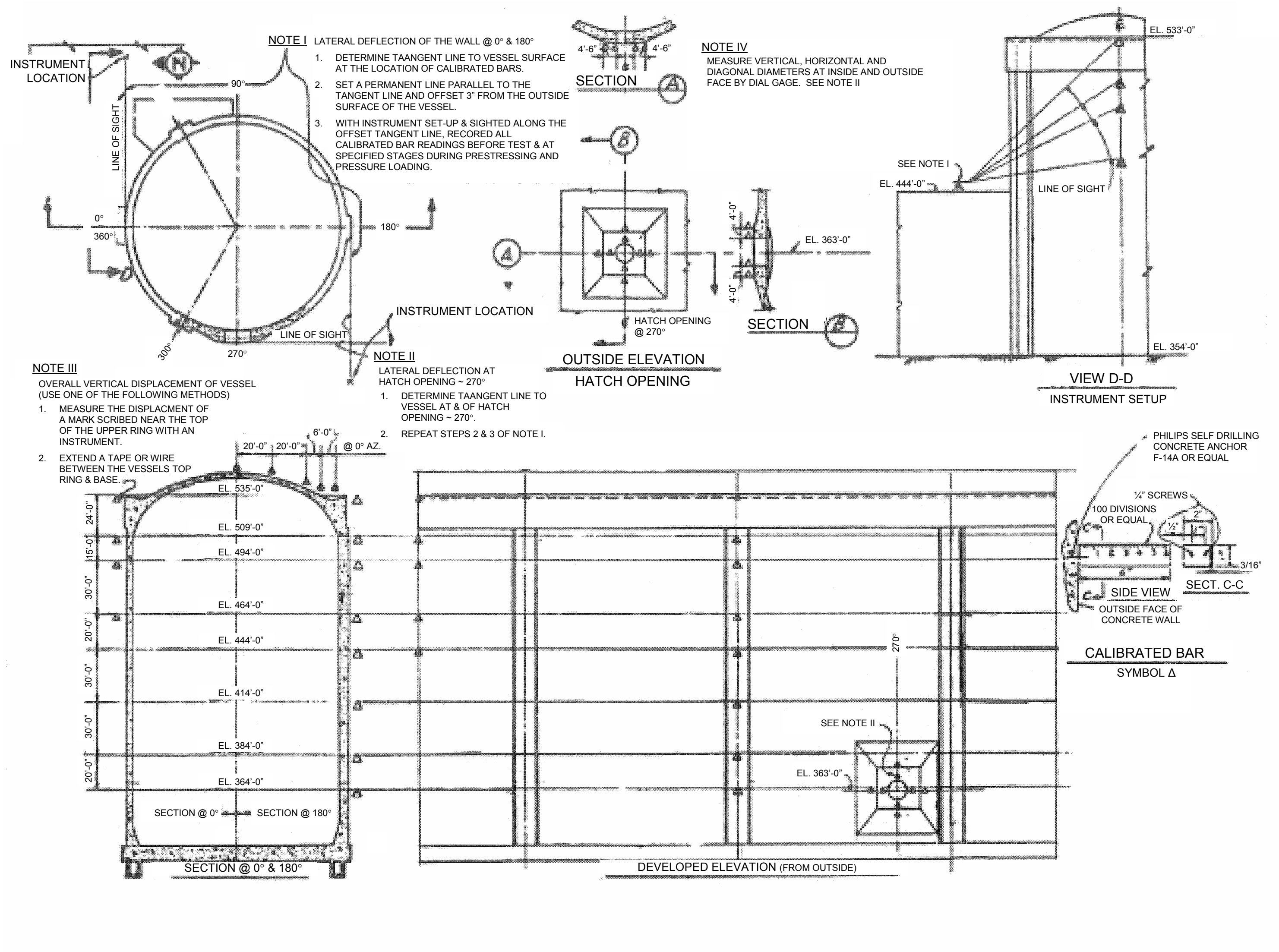


TYPE II POSTULATED CRACK PATTERN INSIDE BUTTRESS

SAR FIGURE NO. 5-16

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
POSTULATED BUTTRESS CRACK PATTERNS	BASED ON DRAWING NO	SHEET 1 — REV.



REACTOR BUILDING INSTRUMENTATION FOR POST TENSIONING

SAR FIGURE NO. 5-17

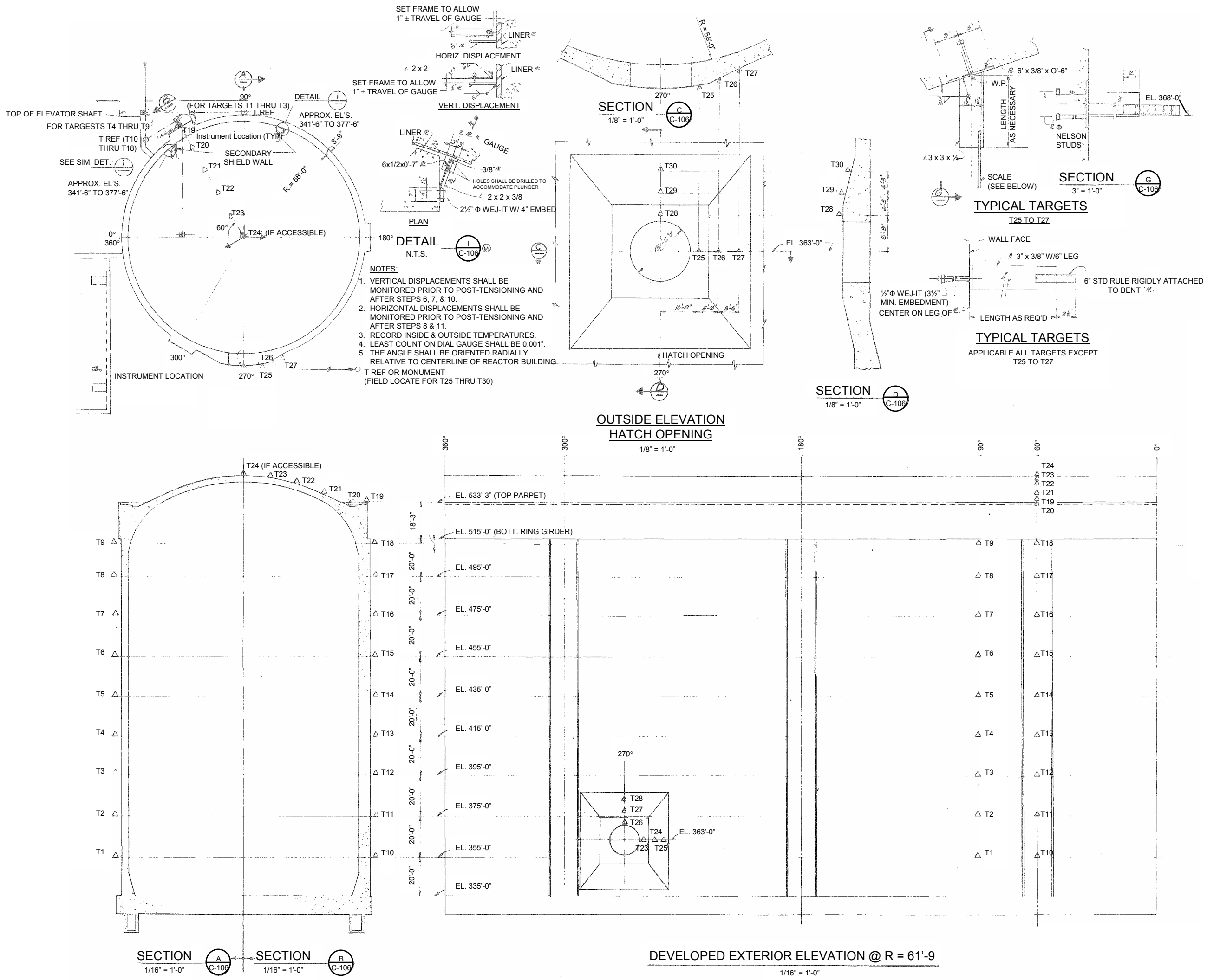
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



NOTES

I. SCALE TARGETS NO T17 THRU T18 (HORIZ. DISPLACEMENTS)
1. INSTALL BULLS EYE TARGETS UNDER EACH INSTRUMENT LOCATION.
2. INSTALL REFERENCE TARGETS (T-REF) AS HOWN ON PLAN
3. INSTALL SCALE TARGETS (T1 THRU T18) AT LOCATIONS MARKED Δ.
4. FOR MEASUREMENTS, THE FOLLOWING PROCEDURES SHALL BE APPLIED:
   a. CENTER INSTRUMENT (OPTICAL PLUMMET) OVER BULLS EYE TARGET AND LEVEL.
   b. BACK-SIGHT THE INDICATED REFERENCE TARGET TO ESTABLISH LINE OF SIGHT AND LOCK THE HORIZONTAL SCALE.
   c. RECORD SCALE TARGET READINGS IN THE ESTABLISHED VERTICAL PLANE.
5. FOR TARGETS T25 THRU T30 SIMILAR.

II. TARGETS NO T19 THRU T24 (VERTICAL DISPLACEMENTS)
1. INSTALL:
   a. PERMANENT MONUMENT ON TOP OF THE AUXILIARY BLDG. SLAB @ EL. 335'-0".
   b. BENCH MARK AT T19 LOCATED AT TOP OF RING GIRDER PARAPET.
   c. PERMANENT BULLS EYE TARGET UNDER TARGETS T20 THRU T24 AND RECORD VERTICAL HEIGHT READINGS.
2. MEASURE THE VERTICAL DISTANCE BETWEEN THE MONUMENT ON THE SLAB @ EL. 335'-0" & THE BENCH MARK
3. SET UP AND LEVEL INSTRUMENT AT INDICATED LOCATIONS.
4. SWING TO LEVEL RODS HELD AT TARGET T19 THRU T26 AND RECORD VERTICAL HEIGH READINGS.

III. OTHER REQUIREMENTS
1. ALL READINGS SHALL BE DONE WITH AN ACCURACY OF ± 0.0625".
2. RECORD INSIDE & OUTSIDE TEMPERATURES (REACTOR BLDG) AT EVERY READING IN ALL CASES.
3. RECORD:.
   a. LOAD ON TAPE.
   b. IF TAPE WAS SUPPORTED AND HOW.

IV. MEASUREMENT SHALL BE TAKEN IN ACCORDANCE WITH THE FOLLOWING SCHEDULE

TARGETS TO BE MEASURED / STATUS PRESTRESSING	T1 TO T18	T19 TO T24	T25 TO T30
PRIOR POST-TENSIONING	√	√	√
#AFTER COMPETION OF 50% STEP 4 & STEP 4		√	
#AFTER COMPETION OF 50% STEP 9 & STEP 9		√	
#AFTER COMPETION OF STEP 8	√		√
AFTER COMPETION OF POST-TENSIONING	√	√	√

STEPS ARE DEFINED ON DWG. C-105

V. ALL MEASUREMENTS SHALL BE SUBMITTED TO ENGINEERING FOR EVALUATION FOLLOWING EACH SURVEY.

REACTOR BUILDING INSTRUMENTATION MEASUREMENT

SAR FIGURE NO. 5-18

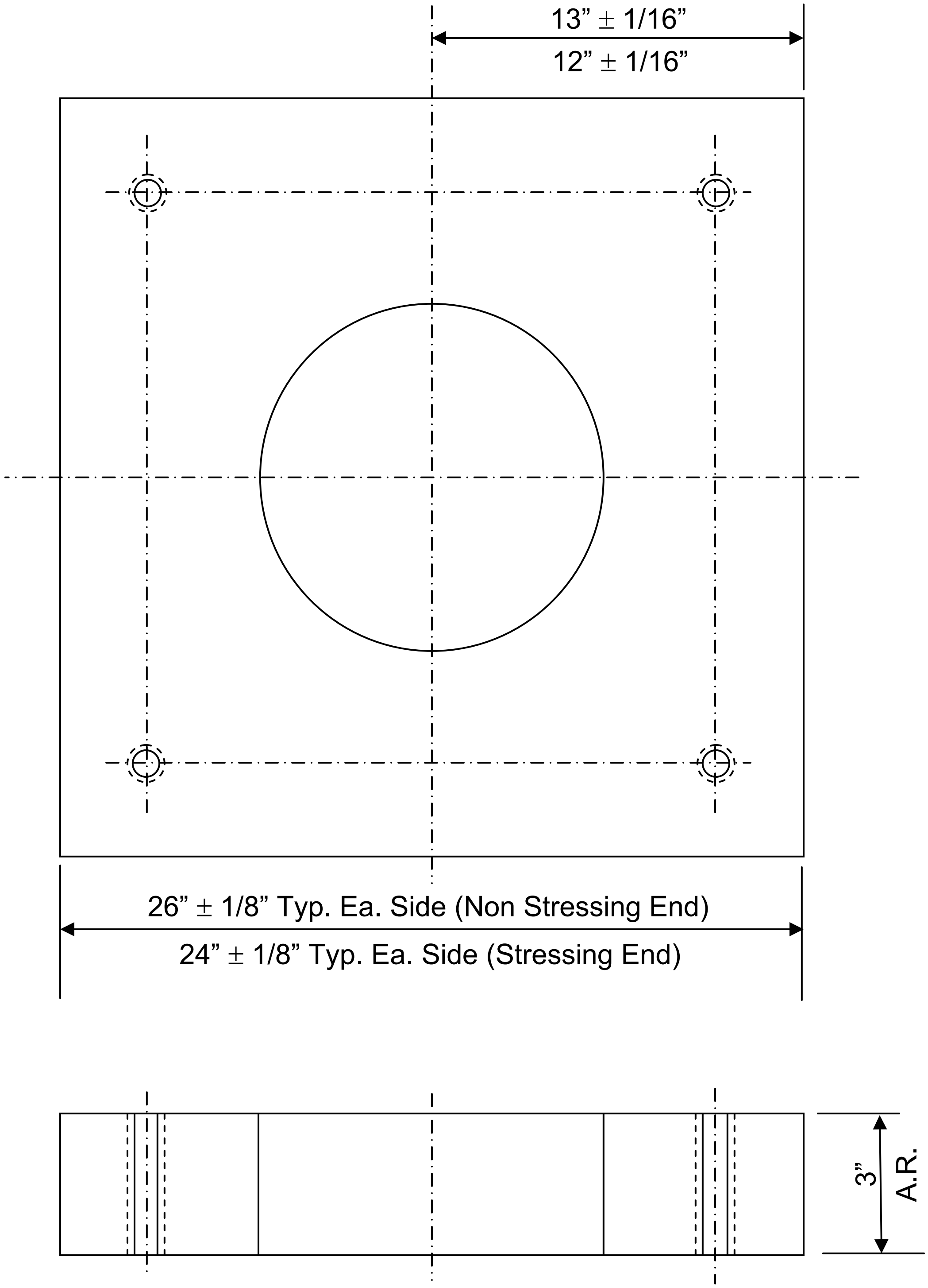
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

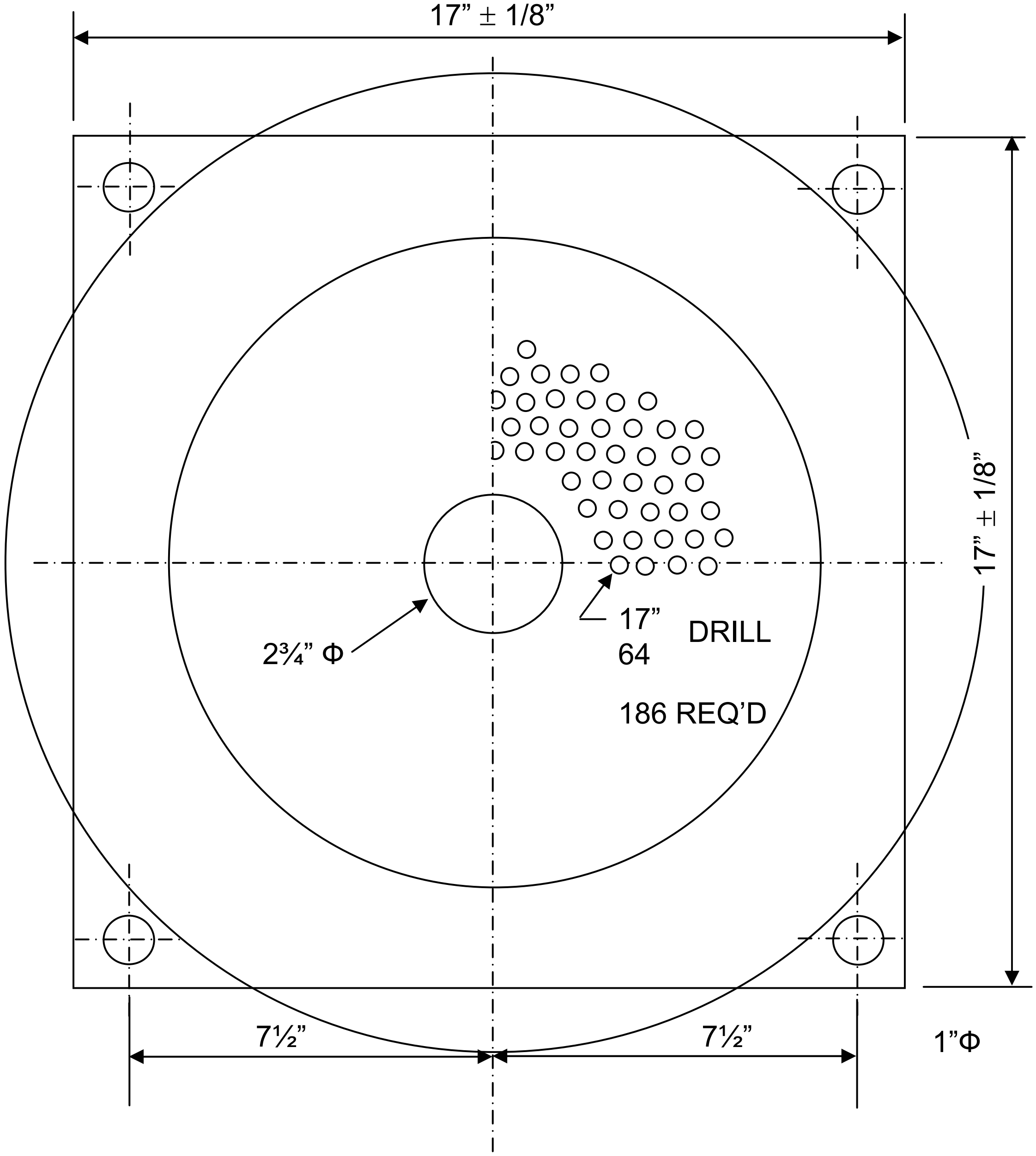
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 5-19

AMENDMENT 20

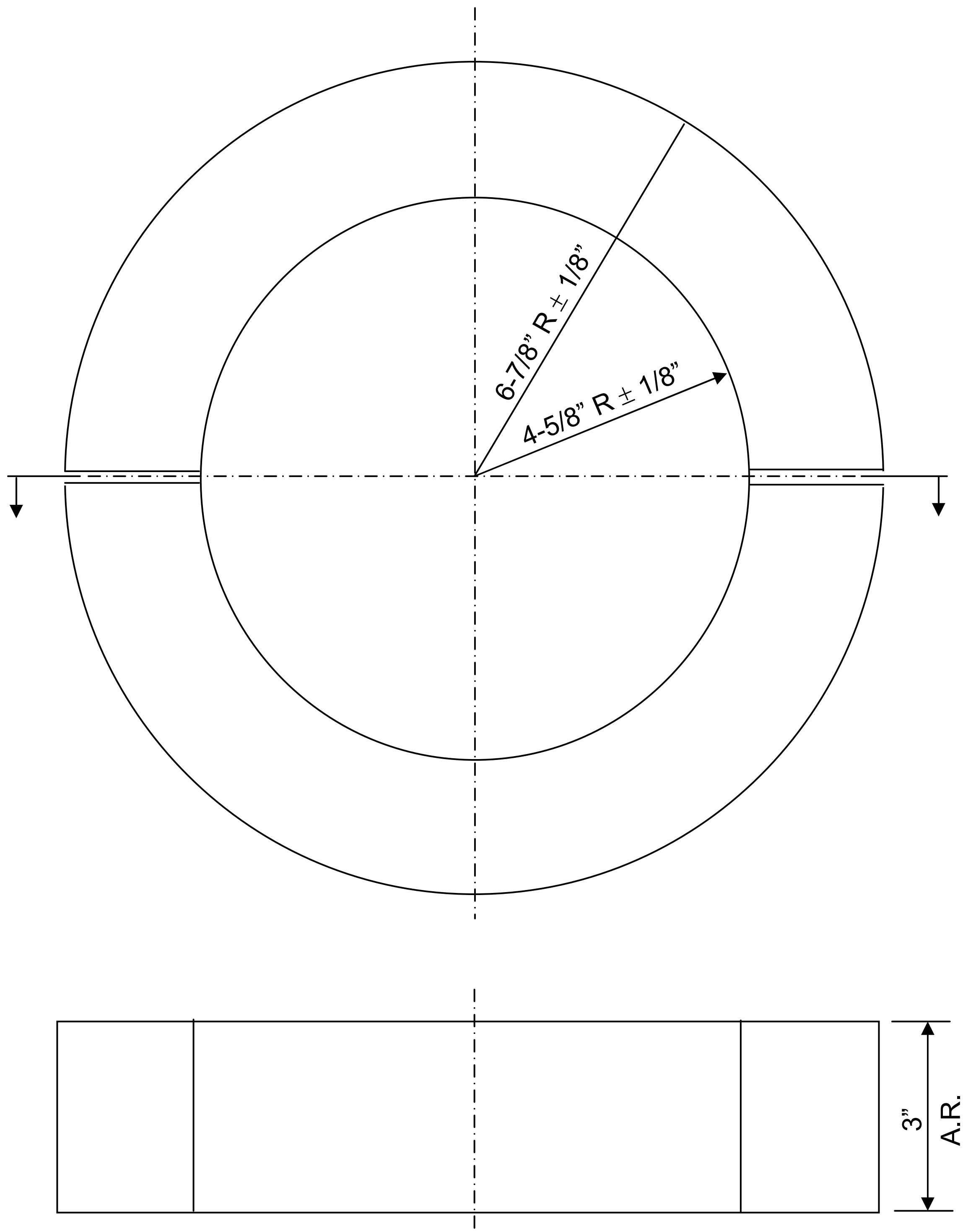
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
186 – WIRE BEARING PLATE	BASED ON DRAWING NO	SHEET 1	REV.



SAR FIGURE NO. 5-20

AMENDMENT 20

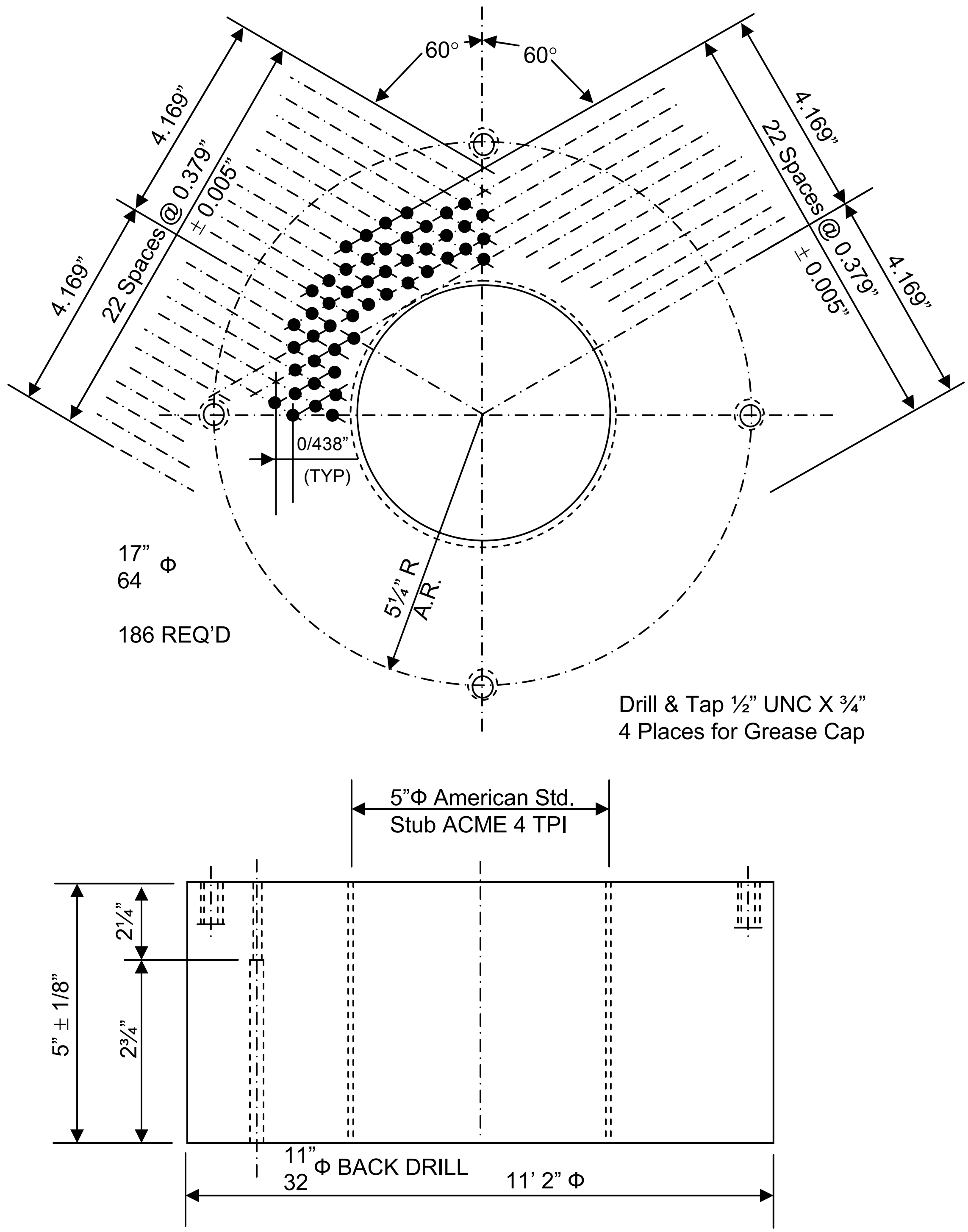
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
FIXED END PLATE	BASED ON DRAWING NO		SHEET 1	REV.



SAR FIGURE NO. 5-21

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
SPLIT SHIMS	BASED ON DRAWING NO	SHEET 1 REV.



SAR FIGURE NO. 5-22

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

STRESSING WASHER

BASED ON DRAWING NO	SHEET	REV.
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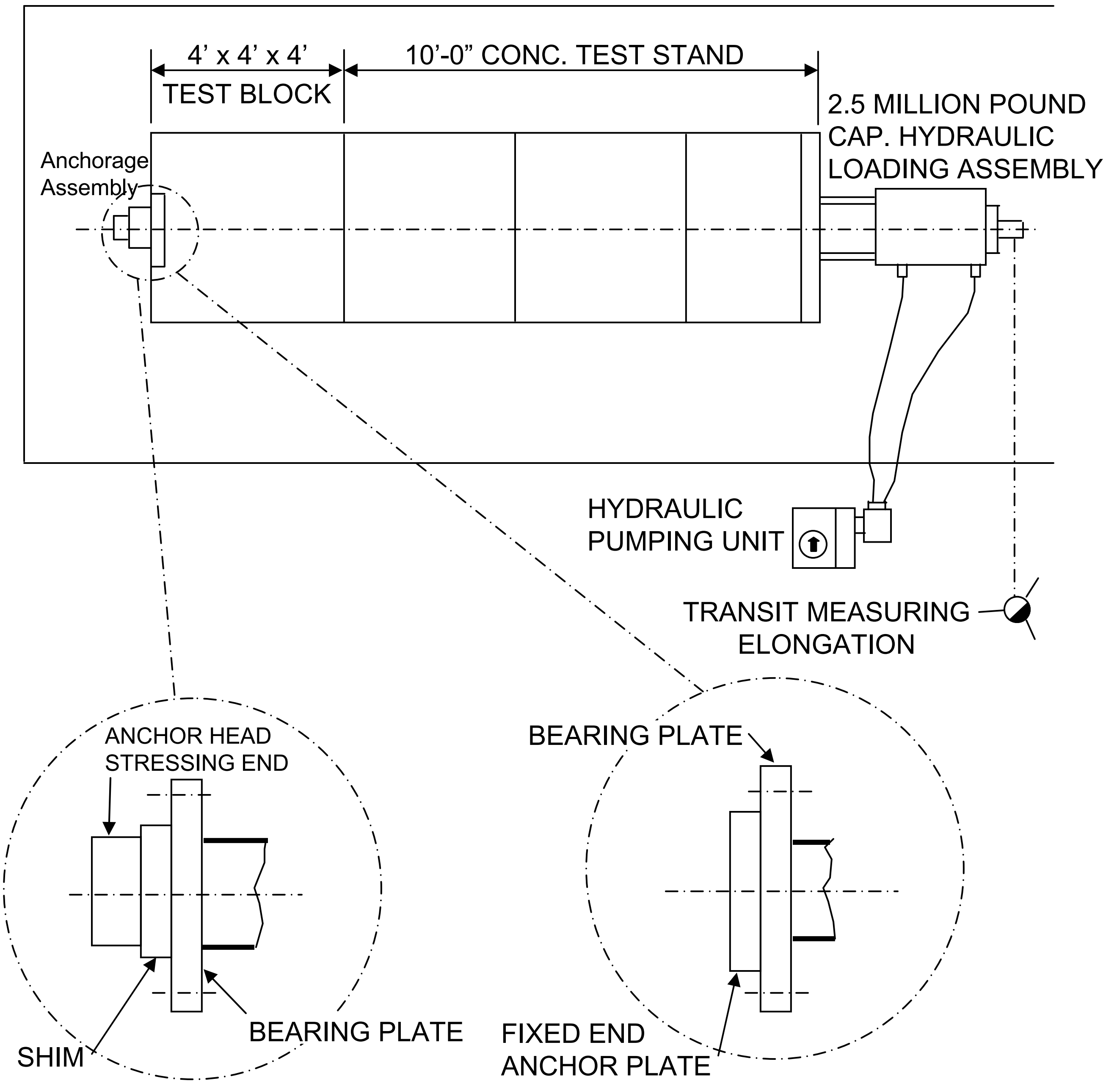


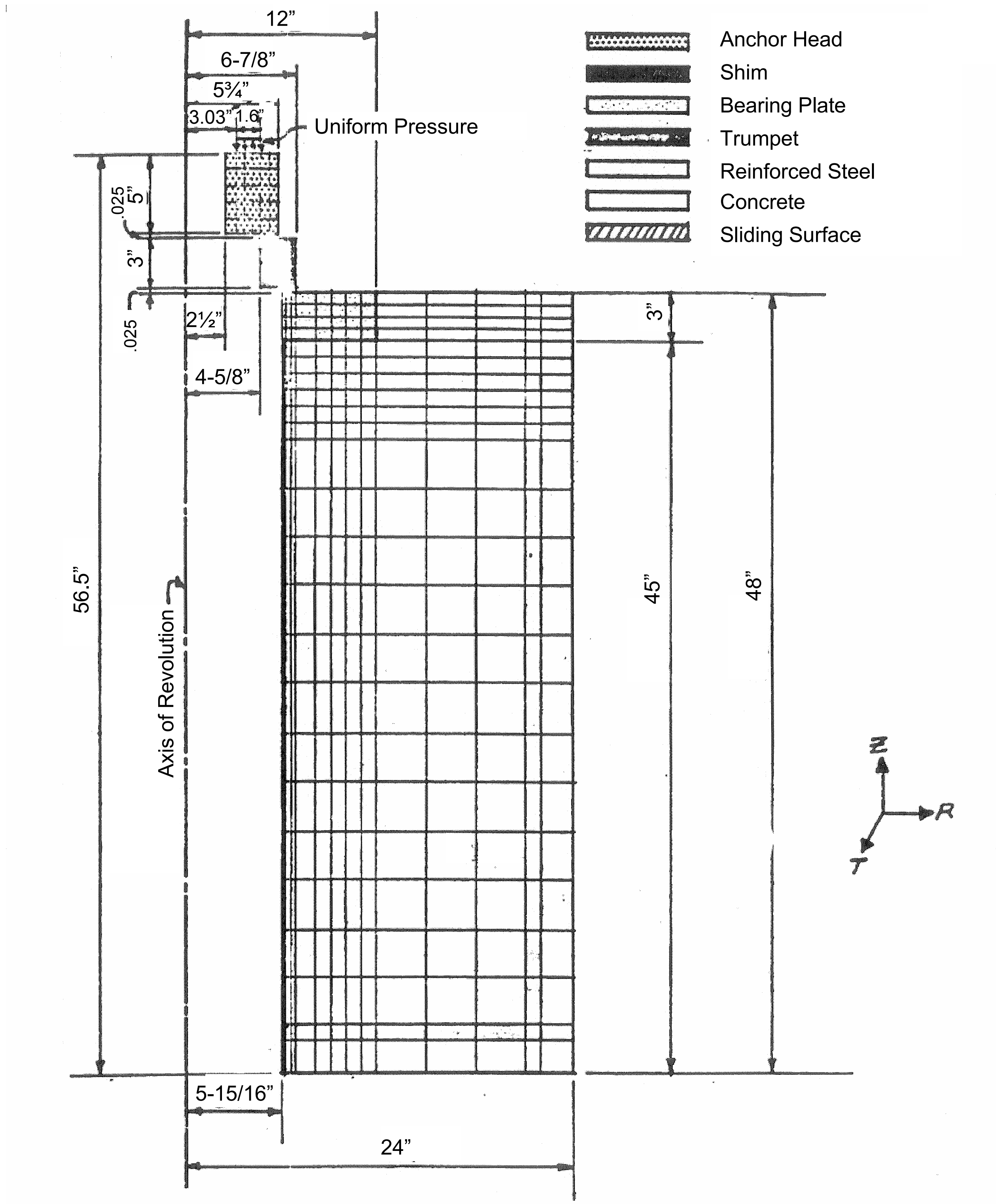
STRESSING END

NON-STRESSING END

SAR FIGURE NO. 5-23

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
TEST SET-UP	BASED ON DRAWING NO	SHEET 1 / REV.



SAR FIGURE NO. 5-24

AMENDMENT 20

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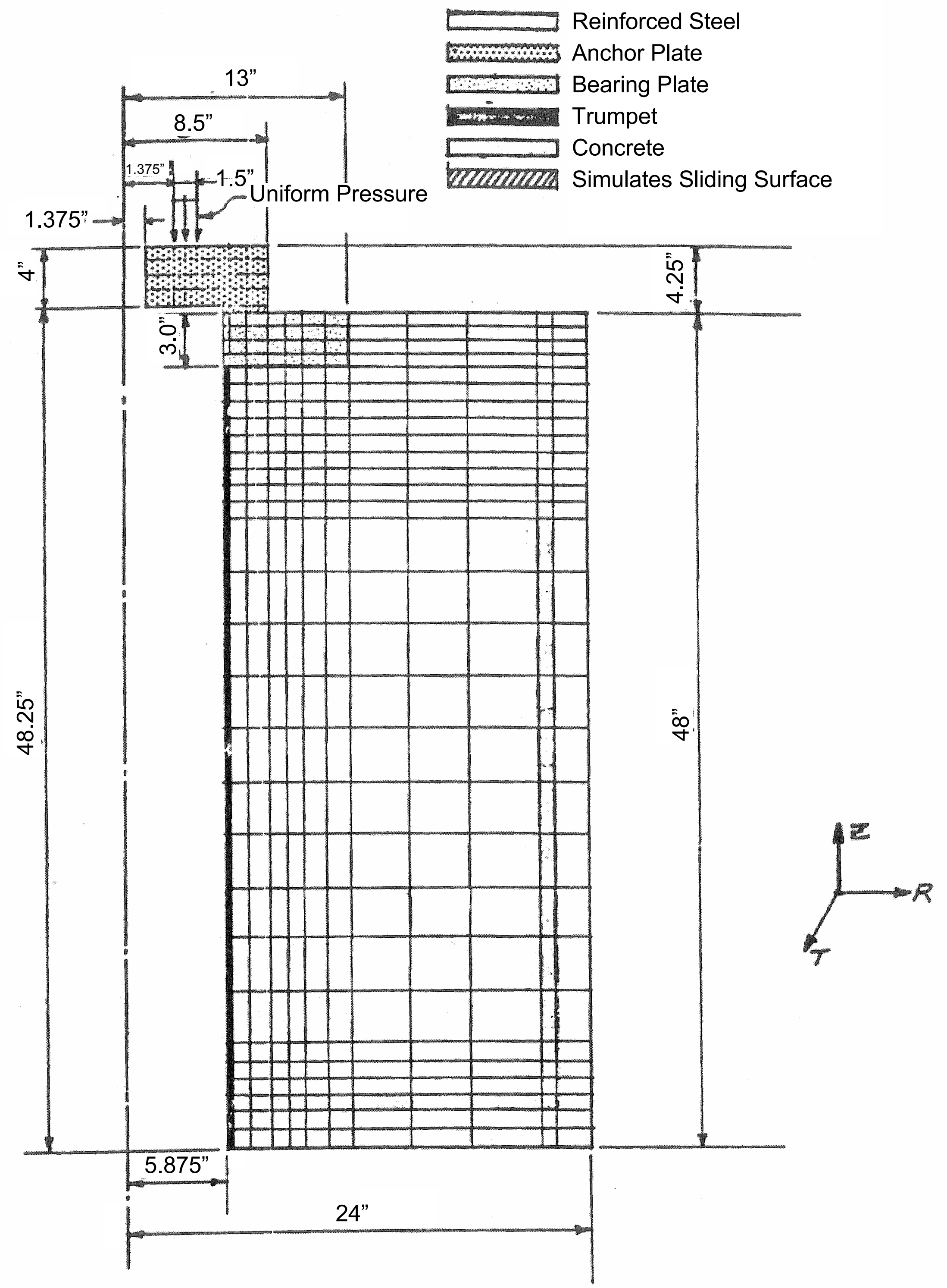
FINITE ELEMENT MESH FOR STRESSING END ANCHOR BLOCK	BASED ON DRAWING NO	SHEET 1	REV.



SAR FIGURE NO. 5-25

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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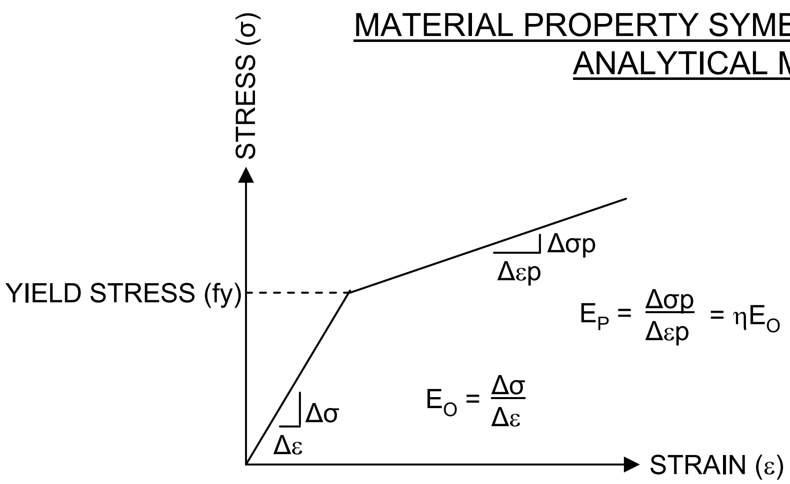
		DESIGN:	ENTERGY
		CAD NO:	

FINITE ELEMENT MESH FOR NON-STRESSING END ANCHOR BLOCK	BASED ON DRAWING NO	SHEET	REV.
		1	

ASSUMED MATERIAL PROPERTIES

PROPERTY / MATERIAL	BEARING PLATE	ANCHOR PLATE	TRUMPET	CONCRETE	REINF STEEL
RZ and T PLANE TENSION					
E_O psi	30×10^6	30×10^6	30×10^6	4.59×10^6	30×10^6
V	0.3	0.3	0.3	0.2	0.3
fy psi	60,000	60,000	60,000	CRACK@200	40,000
η psi	0.05	0.05	0.05	---	0.05
COMPRESSION					
E_O psi	30×10^6	30×10^6	30×10^6	4.59×10^6	30×10^6
V	0.3	0.3	0.3	0.2	0.3
fy psi	60,000	60,000	60,000	3,000	40,000
η psi	0.05	0.05	0.05	0.1	0.05

MATERIAL PROPERTY SYMBOLS USED FOR THE ANALYTICAL MODEL



$$E_P = \frac{\Delta\sigma p}{\Delta\varepsilon p} = \eta E_O$$

$$E_O = \frac{\Delta\sigma}{\Delta\varepsilon}$$

$\Delta\sigma$ - CHANGE IN STRESS IN ELASTIC ZONE

$\Delta\varepsilon$ - CHANGE IN STRAIN IN ELASTIC ZONE

$\Delta\sigma p$ - CHANGE IN STRESS IN PLASTIC ZONE

$\Delta\varepsilon p$ - CHANGE IN STRAIN IN PLASTIC ZONE

Ep - PLASTIC MODULUS OF ELASTICITY

E_O - ELASTIC MODULUS OF ELASTICITY

η - MODULAR RATIO (RATIO OF PLASTIC MODULUS TO ELASTIC MODULUS

V - POISSON'S RATIO

fu - YEILD STRESS

SAR FIGURE NO. 5-26

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

MATERIAL PROPERTIES FOR THE ANALYTICAL MODEL

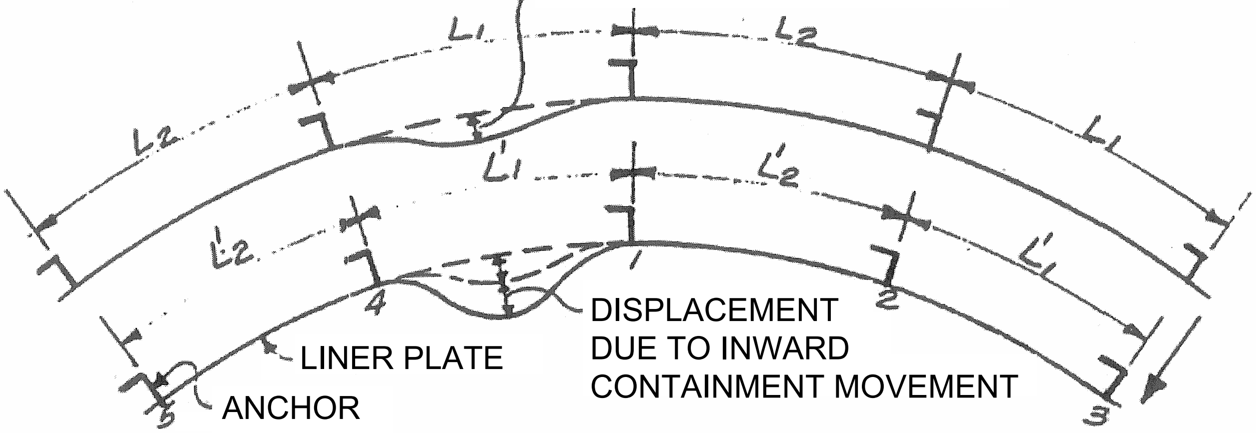
BASED ON DRAWING NO	SHEET	REV.
	1	

COMPONENT	MATERIAL	YIELD STRENGTH P.S.I.	ULTIMATE STRENGTH P.S.I.	DIMENSION (INCHES)	NDT CHARACTERISTIC
BEARING PLATE	HIGH STRENGTH VNT	60,000	80,000	24x24x3	NO BREAK AT -30° F
FIXED END ANCHOR PLATE	HIGH STRENGTH VNT	60,000	80,000	17x17x4	NO BREAK AT -30° F
STRESSING WASHER	FORGING – CHEMISTRY MADE TO ASTM A-514 TYPE H	80,000	100,000	11½"Φx5"	NO BREAK AT -50° F
SPLIT SHIMS	HIGH STRENGTH VNT	60,000	80,000	13¾"Φx3"	————
TENDON (186 WIRES)	ASTM A-421	192,000	240,000	¼Φ	————

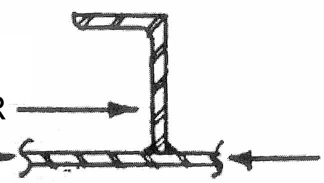
SAR FIGURE NO. 5-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MATERIAL DESCRIPTION FOR THE POST-TENSIONING HARDWARE	BASED ON DRAWING NO	SHEET 1 REV.



5K-1a

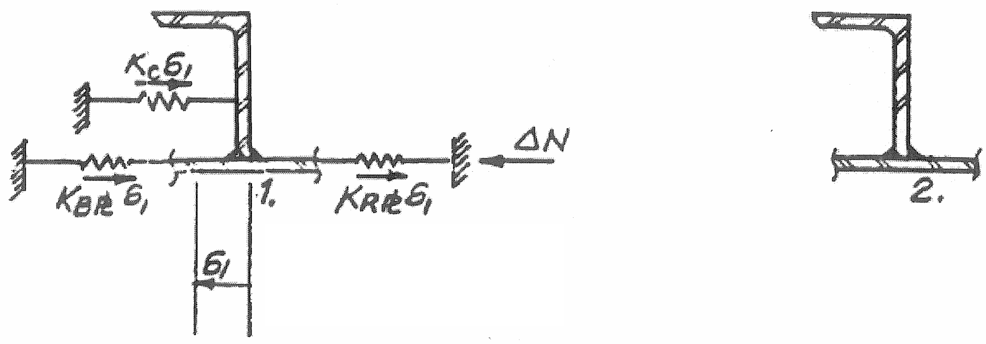


5K-1b

SAR FIGURE NO. 5-28

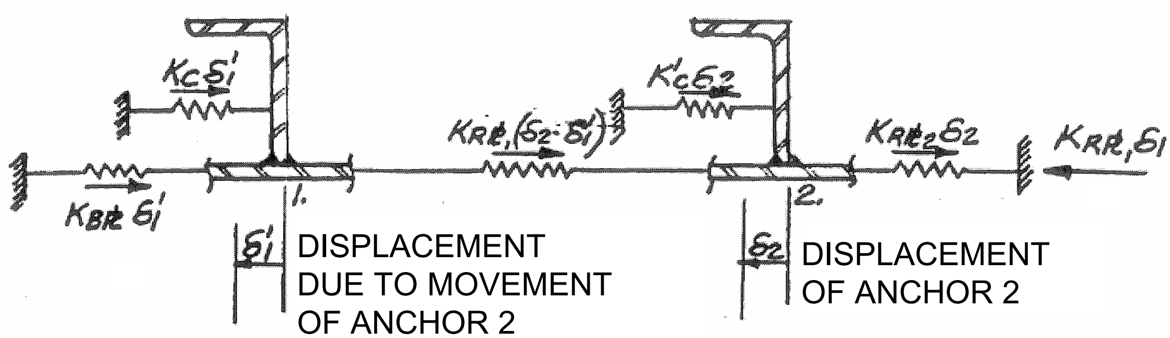
AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LINER PLATE ANCHOR DESIGN	BASED ON DRAWING NO	SHEET 1 / REV.



K_C	Spring constant of anchor
K_{BR}	Spring constant of deformed plate
$K_{RR,}$	Relaxation of section 1-2 due to δ_1

<u>5K-2a</u>

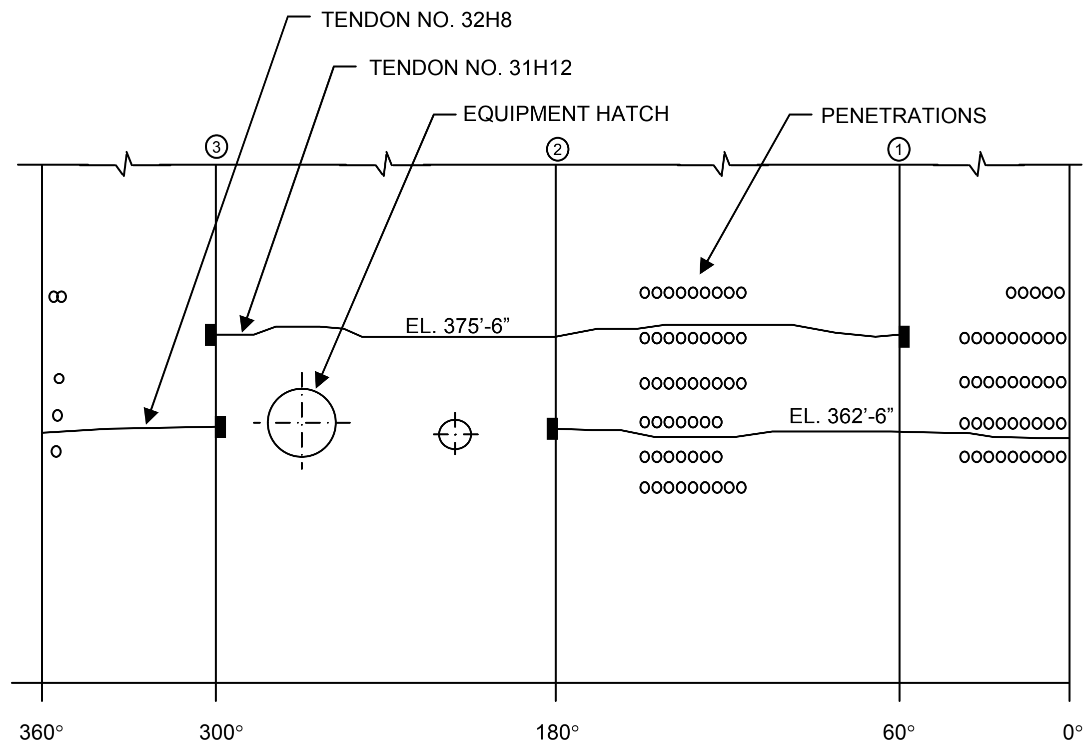


<u>5K-2b</u>

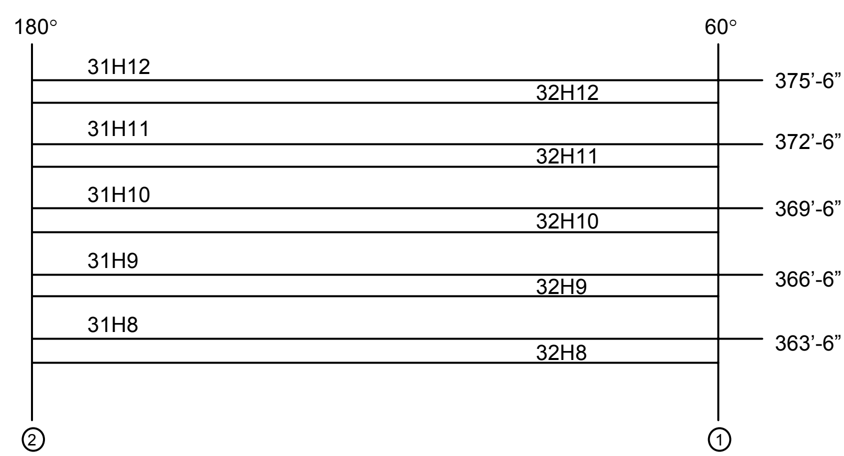
SAR FIGURE NO. 5-29

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
LINER PLATE ANCHOR DESIGN	BASED ON DRAWING NO	SHEET 1	REV.



DEVELOPED ELEVATION OF OMITTED TENDONS
(NO SCALE)



TENDON LAYOUT BETWEEN BUTTRESSES ① & ②
(NO SCALE)

SAR FIGURE NO. 5-30

AMENDMENT 20

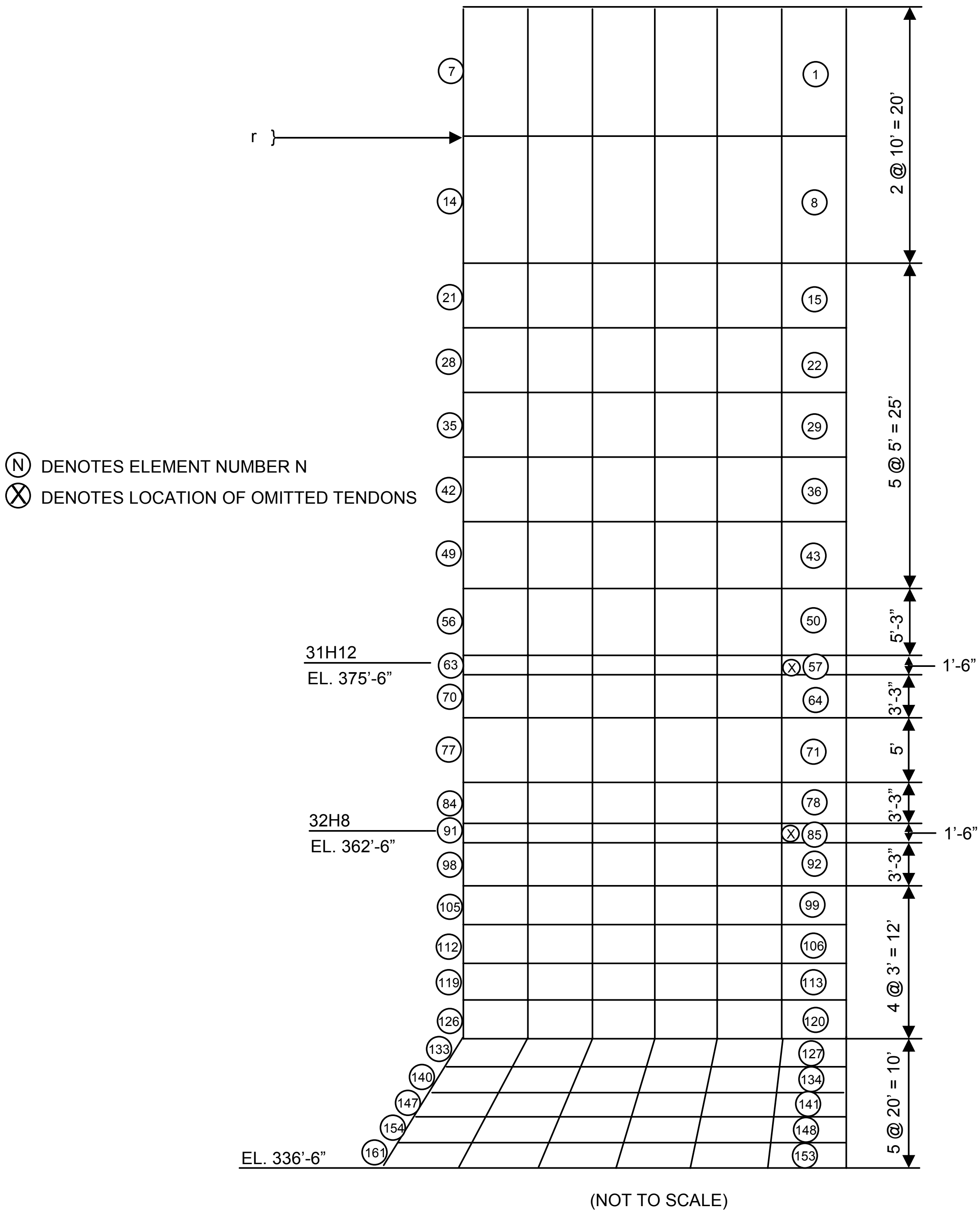
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

LOCATONS OF OMITTED TENDONS

BASED ON DRAWING NO	SHEET	REV.
	1	



SAR FIGURE NO. 5-31

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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		DESIGN:	ENTERGY
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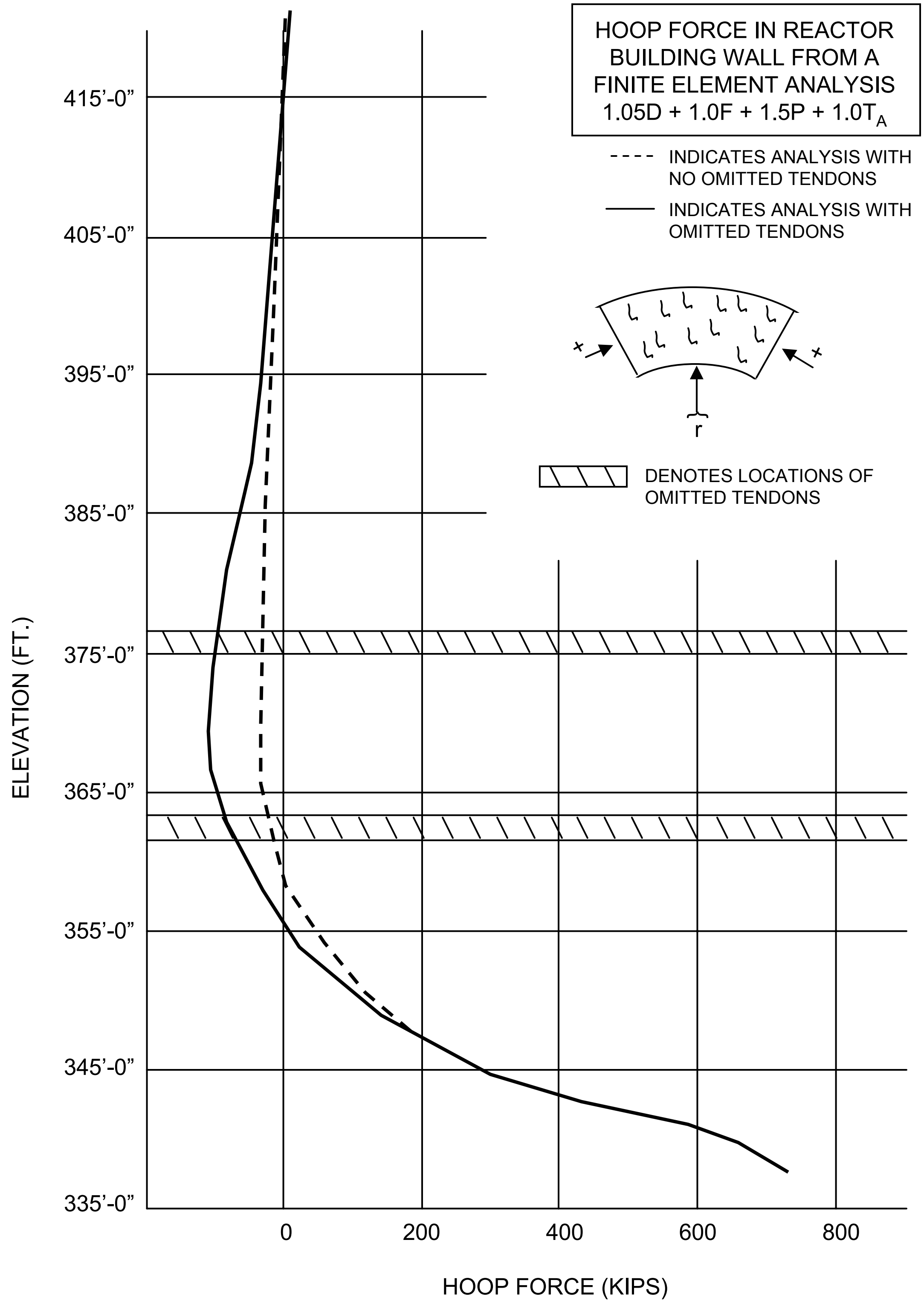

SAR FIGURE NO. 5-32

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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		DESIGN:	ENTERGY
		CAD NO:	

HOOP FORCE FROM FINITE ELEMENT ANALYSES	BASED ON DRAWING NO	SHEET	REV.
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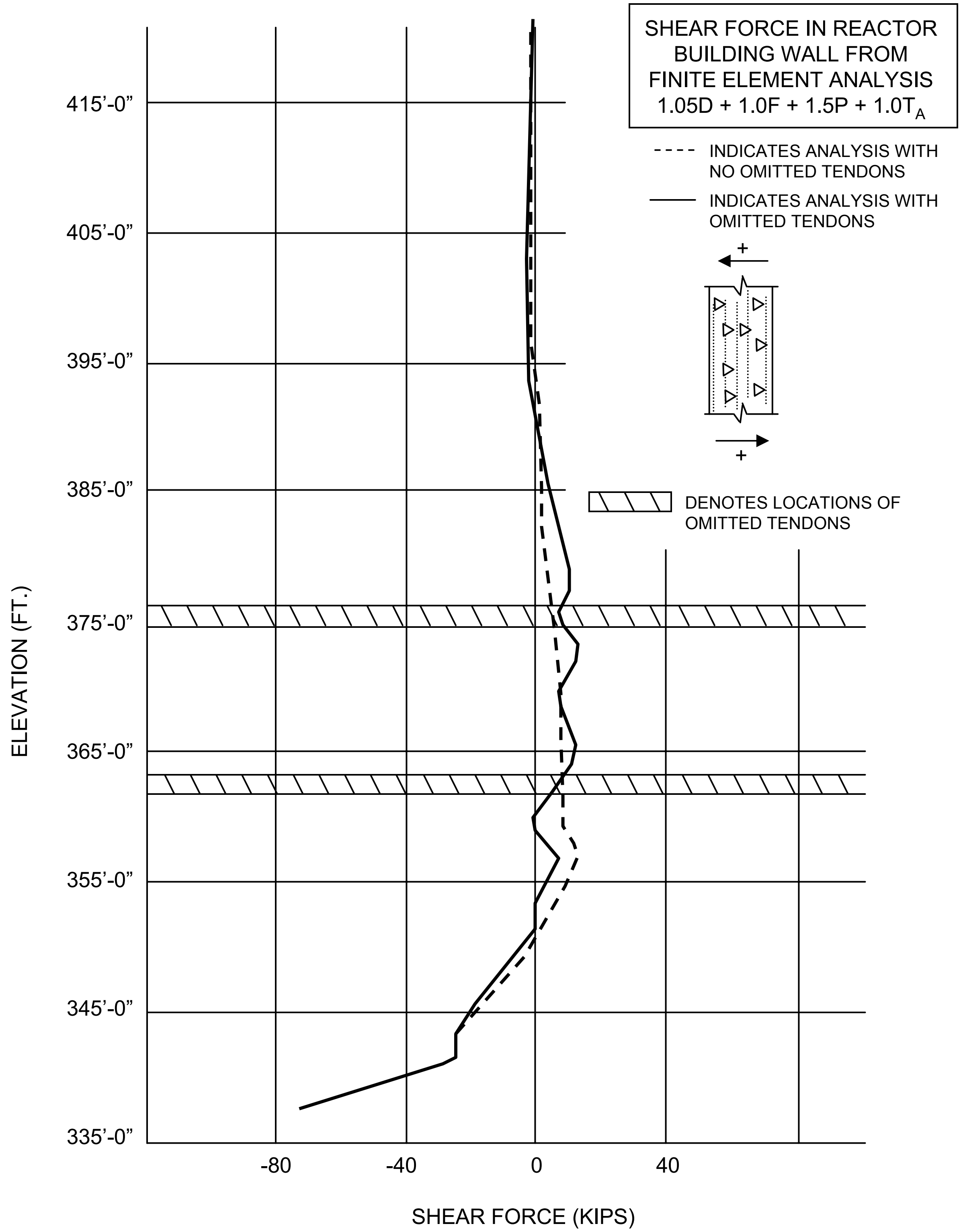

SAR FIGURE NO. 5-33

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
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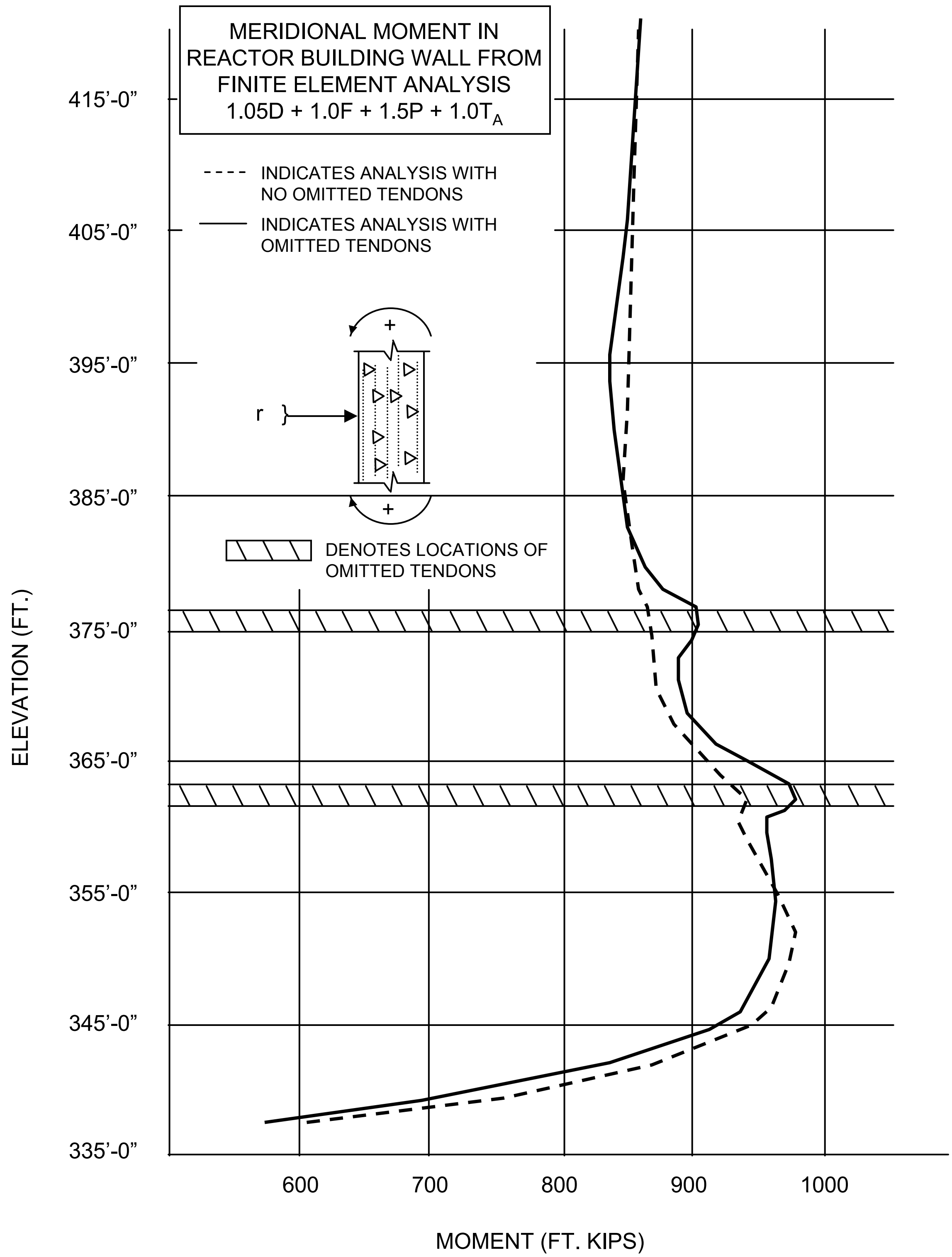

SAR FIGURE NO. 5-34

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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		DESIGN:	ENTERGY
		CAD NO:	

MERIDIONAL MOVEMENT FROM FINITE ELEMENT ANALYSES	BASED ON DRAWING NO	SHEET	REV.
		1	

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6-4	Deleted
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6-13	Deleted
6-14	POST LOCA H_2 FROM METAL CORROSION
6-15	POST LOCA HYDROGEN SOURCES
6-16	DELETED
6-17	DELETED
6-18	DELETED

UPDATE REFERENCE LIST

Section and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross Reference
Table 6.4	Correspondence from Phillips, AP&L, to Giambusso, NRC, dated July 9, 1975. (1CAN077507)
Table 6.4	Correspondence from Woodward, AP&L, to Ziemann, NRC, dated December 31, 1975. (1CAN127529)
6.1.1	Correspondence from Cavanaugh, AP&L to Ziemann, NRC, dated August 3, 1976. (1CAN087605)
6.1.1	Correspondence from Phillips, AP&L to Ziemann, NRC, dated December 1, 1976. (1CAN127602)
6.6.1	Correspondence from Denton, NRC, to AP&L dated October 30, 1979. (0CNA107965)
6.6.1 6.6.2.1	Correspondence from Trimble, AP&L, to Eisenhut, NRC, dated January 31, 1980. (0CAN018032)
6.1.2.3	Correspondence from Howard, AP&L, to NRC Document Control Desk, Dated May 27, 1988. (0CNA058813)
6.4.3	Correspondence from Steel, AP&L, to Eisenhut, NRC, dated March 24, 1980. (0CAN038003)
6.6	Correspondence from Trimble, AP&L, to Eisenhut, NRC, dated July 10, 1980. 6.6.1 (0CAN078013)
6.6.2.1	Correspondence from Cavanaugh, AP&L, to Eisenhut, NRC, dated December 17, 1980. (0CAN128010)
6.1.2.1.3	Design Change Package 412, "Add Provisions for Batch Make-Up to Core Flood Tank," 1976. (1DCP760412)
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6.6.2.1	Design Change Package 81-1071, "H_2O Removal System for H_2 Purge."

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Table 6-10	Design Change Package 83-1116A, "Power for Backdraft Damper Valves."
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6.2.2.1	Correspondence from Trimble, AP&L to Seyfrit, NRC, dated June 10, 1980. (0CAN068004)
6.1.1 6.1.2.1.1 6.1.2.1.2 6.1.3.1 6.4.2	AP&L Engineering Report 89E-0018-02, Engineering Evaluation of ANO-1 High Pressure Injection System Backflow Condition.
6.4.2 Table 6-11	AP&L Engineering Report 89E-0039-02, HPI Seal Integrity Evaluation
6.1.3.1	Babcock & Wilcox Document BAW-1976, "Small Break Loss of Coolant Accident Analysis for B&W 177FA Lowered Loop Plants in Response to NUREG-0737, Item II.K.3.31", September 1986.
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Figure 6-8	AP&L Calculation 89E-0018-06, Decay Heat Removal Heat Exchanger Performance.
6.3.2	AP&L Engineering Report 88E-0115-01, ANO-1 Service Water Flow Degradation DBA Analysis Impact.
6.1.3.1 6.1.3.2	AP&L Calculation 89E-0040-01, "ANO-1 SFAS Response Time Evaluation"

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6.1.3.1	AP&L Calculation 88E-0083-01, "Evaluation of HPI Coolant Reduction Due to a 35 Second Stroke Time for CV1300 and CV1301.
Figure 6-3	Design Change Package 88-1081A, "Piping and Support Code Compliance for Reactor Building Spray Piping [outside containment]."
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6.1.2.2.4	Clarification of Relief Valves Description. Ref: 86D-1129-01, "Decay Heat Removal System Overpressure Protection Design Base Document."
6.2.2.4.5	Calculation 89E-0040-01, "Correction to the RB Spray and Containment Cooler Response Times."
Figure 6-10	Design Change Package 88-1057, "Containment Penetration Room Ventilation System Interlock."
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Figure 6-18	Deleted
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6 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. In the event of a Loss of Coolant Accident (LOCA), the engineered safeguards act to provide emergency coolant to assure structural integrity of the core, to maintain the integrity of the reactor building, and to reduce the fission products expelled to the reactor building. Special precautions are taken to assure high quality in the components and in system design and to assure reliable and dependable operation.

The engineered safeguards include provisions for:

A. High Pressure Injection (HPI) of borated water by the Makeup and Purification System;

B. Low Pressure Injection (LPI) of borated water by the Decay Heat Removal System;

C. Core flooding with borated water by the Core Flooding System;

D. Reactor building cooling by the Reactor Building Cooling System;

E. Reactor building cooling with borated water spray by the Reactor Building Spray System;

F. Reactor building isolation by the Containment Isolation System and filtration of containment building leakage by the Penetration Room Ventilation System;

G. Removal of iodine fission products in the reactor building atmosphere by the Reactor Building Spray System and passive Containment Sump Buffering Agent System using NaTB to balance sump pH.

Figure 6-1 schematically depicts the major engineered safeguards systems related to core and building protection. A general description of the engineered safeguards provisions is presented below and a more detailed description is presented in the latter portion of this section.

The systems above fulfill the functions ascribed to engineered safeguards in the FSAR. The Emergency Feedwater (EFW) System was subsequently upgraded to the standards of engineered safeguards equipment (see CLAPNR 1-120-02 dated 12/3/1980). Although the EFW is considered an ES system by design, the description of ES systems in this chapter and throughout the SAR generally refer to those systems listed above.

The HPI and LPI and the Core Flooding System are collectively designed as an Emergency Core Cooling System (ECCS) which, for the entire spectrum of Reactor Coolant System (RCS) break sizes, terminates the core thermal transient, limits the amount of zirconium-water reaction, and assures that the core integrity is maintained. Figure 6-1 shows the ECCS.

The HPI System is an integral part of the Makeup and Purification System which uses two of the three makeup pumps (P36A, P36B, P36C) for injection of coolant from the Borated Water Storage Tank (BWST) (T-3). The LPI system is an integral part of the Decay Heat Removal System which uses the two decay heat pumps (P34A, P34B) and two decay heat coolers (E35A, E35B) and has provision for coolant injection from the BWST or recirculation from the reactor building sump. (See Chapter 9 for a description of the Makeup and Purification and Decay Heat Removal Systems.) The Core Flooding System is composed of two separate pressurized tanks (T2A, T2B) containing borated water at reactor building ambient temperature. This passive system

automatically discharges its contents directly into the reactor vessel at a preset RCS pressure without reliance on any actuating signal, or on any externally actuated component.

Reactor building integrity is insured by two independent pressure reducing systems operating on different principles; the Reactor Building Spray System and the Reactor Building Cooling System (refer to Figure 6-1). These systems have the redundancy required to meet the single failure criterion. These systems operate over the entire spectrum of RCS break sizes to rapidly reduce the driving force for leakage of radioactive materials from the reactor building. The Reactor Building Spray System also reduces the iodine fission product concentration in the reactor building atmosphere following a LOCA. The passive Containment Sump Buffering Agent System using NaTB balances sump pH such that the iodine fission products remain in solution in the containment sump in the reactor building following a LOCA. The Penetration Room Ventilation System further reduces post LOCA fission product releases by filtration of the penetration room atmosphere. This processes leakage from the containment into the penetration rooms.

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems function during normal reactor operation thus providing a constant check on operational status. Where equipment is used for emergency functions only, such as in the Reactor Building Spray System, the systems have been designed to permit meaningful periodic tests. Operational reliability has been achieved by using proven component designs wherever possible and/or by conducting tests. Quality control and assurance requirements are implemented during the design, manufacture, and installation of the engineered safeguards components and systems to assure that a high quality level is maintained. The quality program is based upon the use of accepted industry codes and standards as well as supplementary test and inspections. The resultant high quality level of the components gives assurance that they will perform their intended function under the worst anticipated conditions following a LOCA. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a Design Basis Accident (DBA). All equipment must remain functional throughout the life of the plant. Certain safety-related equipment must operate during the design plant life as well as function as required during and following a DBA at the end of plant life.

This chapter describes the physical arrangement, design, and operation of the engineered systems as related to their safety function. Reactor building isolation is described in Chapter 5. Chapter 7 describes the actuation instrumentation for engineered safeguards systems. Table 6-12 gives actuation setpoints for all systems discussed. Chapter 14 describes the analysis of the engineered safeguards systems' capability to provide adequate protection during accident conditions. Chapter 9 discusses functions performed by these systems during normal operation and gives further design details and descriptive information.

6.1 EMERGENCY CORE COOLING SYSTEM

6.1.1 DESIGN BASES

The principal design for the Emergency Core Cooling System (ECCS) as described in the NRC General Design Criterion 35, "Emergency Core Cooling," is met. Protection for the entire spectrum of RCS break sizes is provided. Separate and independent flow paths are provided in the ECCS and redundancy in active components insures that the required functions are performed if a single failure occurs. Separate emergency power sources are supplied to the redundant active components and separate instrument channels are used to actuate the systems. Actuation pressures for the ECCS systems are shown in Table 6-12. The adequacy of the installed ECCS to prevent fuel and clad damage is discussed in Chapter 14.

The ECCS is designed to operate in the following modes:

A. Injection of borated water from the BWST by the HPI System.

B. Injection of borated water by the Core Flooding System.

C. Injection of borated water from the BWST by the LPI system.

D. Long-term core cooling by recirculation of injection water from the reactor building sump to the core through the LPI system for large breaks or through the LPI system and the HPI System in series for small breaks where primary pressure remains above the shutoff head of the LPI pumps.

Although the HPI and the LPI Systems operate to provide full protection across the entire spectrum of break sizes, each system may operate individually and each is initiated independently. The HPI System prevents uncovering of the core for small coolant piping leaks where high RCS pressure is maintained and delays uncovering of the core for intermediate sized breaks. The Core Flooding and LPI Systems are designed to recover the core at intermediate-to-low RCS pressures, and to insure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The LPI System is also designed to permit long-term core cooling in the recirculation mode after a LOCA. The LPI System and the HPI System are designed to permit the recirculation mode at various system pressures following a LOCA. This is accomplished using LPI directly to the core for the low RCS pressures that exist following a large break LOCA. The LPI discharge provides suction to the HPI in the "piggyback" mode of operation for higher RCS pressures which may exist following a small break. Pumped injection includes both HPI and LPI, each with separate and independent flow paths. One flow path from HPI System and one flow path from LPI System and the Core Flooding Tanks are capable of providing 100 percent of the necessary core injection. The redundant protection afforded by the ECCS components, subsystems, and systems for the spectrum of reactor coolant pipe break sizes is discussed in Chapter 14.

6.1.2 SYSTEM DESIGN

6.1.2.1 System Description

The schematic diagram for the ECCS is shown in Figure 6-1. The supporting systems are described in Chapter 9.

6.1.2.1.1 High Pressure Injection

During normal reactor operation, the Makeup and Purification System recirculates reactor coolant for purification and for supply of seal water to the Reactor Coolant Pumps. This normal operating mode and component data are described in Chapter 9 (After HPI is initiated, the makeup pumps are referred to as the HPI pumps).

HPI of borated water is initiated by a low RCS pressure or high reactor building pressure (Table 6-12). Automatic actuation of the valves and pumps by the engineered safeguards actuation signals switches the system from its normal operating mode to the emergency operating mode to deliver water from the BWST into the reactor vessel through the reactor coolant lines. The following actions accomplish this change.

A. The isolation valves in the purification letdown line (CV1216, CV1214, CV1221), the controlled bleed off line (CV1270, CV1271, CV1272, CV1273, CV1274), the normal makeup line (CV1234, CV1233), and the BWST purification recirculation (CV1438, CV1441) close.

B. The valves in the lines to the BWST outlet headers (CV1407, CV1408) open.

C. The inlet valve in each HPI line (CV1219, CV1220, CV1227, CV1228, CV1278, CV1279, CV1284, CV1285) opens.

D. The HPI recirculation valves (CV1300, CV1301) close.

E. The auxiliary lube oil pumps start.

F. The idle HPI pump starts.

In addition to the automatic action described, the HPI pumps and valves may be remotely actuated by the operator from the control room.

During a small break LOCA, if system pressure remains above the LPI shutoff head, upon depletion of the BWST, the LPI suction is manually realigned to the containment sump and the discharge is aligned to the suction of the HPI pumps (piggyback operation).

Operation of the HPI System continues until the system operation is manually terminated.

6.1.2.1.2 <u>Low Pressure Injection System</u>

The LPI System is designed to maintain core cooling for large break sizes and operates independently of and in addition to the HPI System. The normal operating mode and component data for this system are described in Chapter 9.

Automatic actuation of LPI is initiated by low RCS pressure or high reactor building pressure. Initiation provides the following actions:

A. The valves in the lines connecting the BWST to the LPI pump suction headers (CV1407, CV1408) open.

B. The valve in each LPI line opens (CV1400, CV1401).

C. Decay heat removal pumps start.

D. Decay heat removal cooler service water isolation valves (CV3821, CV3822) open in response to the pump start.

E. Decay heat removal cooler outlet valves (CV1428, CV1429) open and bypass valves (CV1432, CV1433) close.

F. The isolation valves in the BWST purification recirculation line (CV1438, CV1441) close.

LPI is accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

A cross connection between the two Decay Heat Removal lines (as shown in Figure 6-1) provides the capability of injecting an adequate supply of borated water for core cooling even in the event of a core flood line rupture.

The initial injection of water by the LPI System involves pumping water from the BWST into the reactor vessel. With all engineered safeguards pumps operating and assuming the maximum break size, this mode of operation lasts for a minimum of about 25 minutes. When the BWST reaches an indicated level of 6 feet, the operator opens the suction valves from the reactor building sump (CV-1405, CV-1406) permitting recirculation of the spilled reactor coolant and injection water and closes the BWST outlet valves (CV-1407, CV-1408). An alarm is also annunciated in the control room to indicate a low-low level in the BWST. Check valves (BW-4A, BW-4B) and the closed BWST outlet valves (CV-1407, CV-1408) in the line from the BWST provide redundant isolation to prevent backflow into the BWST.

Leakage gaps between the reactor vessel outlet nozzles and the core support shield have been determined to be sufficient by themselves to preclude post-LOCA boron precipitation in the reactor core. As a secondary measure, Emergency Operating Procedures direct establishing another dilution flow path in the LPI System. The LPI System flowpaths used for this purpose are:

A. Establishing a suction through the normal decay heat drop line with one decay heat pump.

B. Gravity draining to the reactor building emergency sump.

C. Adding dilution water through the pressurizer auxiliary spray line.

Any one of these flowpaths has the capability to limit the maximum boron concentration in the reactor vessel following a LOCA well below the concentration that would allow precipitation.

6.1.2.1.3 <u>Core Flooding System</u>

The Core Flooding System provides core protection continuity for intermediate and large RCS pipe failures. It automatically floods the core when the RCS pressure drops below approximately 600 psig. The Core Flooding System is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks is sufficient to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA.

The discharge pipe from each Core Flooding Tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFTs contains an electric motor operated stop valve (CV2415, CV2419) adjacent to the tank and two in-line check valves (DH14A, DH14B, CF1A, CF1B) in series. The stop valves at the CFT outlet are open during the reactor power operation. Position switches on each valve actuate separate open and closed valve position indicators. These indicators are located in the main control room. Two separate alarms based on isolation valve position are provided. If either valve is closed while reactor coolant pressure increases during plant startup, an alarm is actuated. If either valve is open while reactor coolant pressure decreases to a value which could cause emptying of the Core Flooding Tanks,

a second alarm is actuated. The second alarm helps preclude the discharging of the CFTs during planned shutdowns. During power operation, when the RCS pressure is higher than the core flooding pressure, the two check valves in the line to the CFTs prevent high pressure reactor coolant from entering the CFTs.

The breakers which supply power to the motor operators for the isolation valves are open under administrative control. The Technical Specifications require this action as a prerequisite for operation of the RCS above 800 psig.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one-third of the CFT volume. Connections are provided for adding or removing both borated water and nitrogen during power operation so that the proper level and pressure may be maintained. A mixing tank and recirculation pump is provided for making up to the CFTs during operation. The recirculation system affords effective mixing and thus more accurate sampling capability of the CFT contents in the event that additions have been made to the tank or the contents have stratified to the extent that recirculation is necessary.

Each CFT is protected from overpressurization by a relief valve (PSV2415, PSV2421). Level and pressure indicators and alarms are provided on the Engineered Safeguards vertical boards (C16/C18) in the Control Room.

Design data for major system components are shown in Table 6-1.

6.1.2.2 Quality Control Codes and Standards

The HPI, LPI, and the Core Flooding Systems are designed and manufactured to the codes and standards in Table 6-2 and in Chapter 9.

Components and piping in the ECCS are designated as Seismic Category 1 equipment and are designed to maintain their functional integrity during an earthquake (Chapter 5 defines the acceptable stress limits for Seismic Category 1 equipment).

6.1.2.3 Material Compatibility

All components with surfaces in contact with water containing boric acid are protected from corrosion and deterioration. In addition, Entergy Operations, Inc. has initiated an extensive program of RCS leak evaluation coupled with RCS walkdowns to rapidly identify and correct any leaks which could potentially cause boric acid corrosion (0CAN058813). The HPI System, which continuously pumps borated water, is constructed entirely of stainless steel. With the exception of the BWST, which is carbon steel with an interior protective coating, the major wetted components in the LPI System are constructed of stainless steel. The core flooding piping and valves are stainless steel and the tanks are constructed of stainless clad carbon steel.

The major active components of the Emergency Core Cooling System are external to the reactor building and not exposed to the post-accident building environment. The safeguards transmitters and cables used to supply the required actuating signals located in the reactor building can withstand the post-accident environment long enough to provide the actuating signals. Information on the qualification of electrical equipment to function in post-accident environments is provided in Chapter 7.

6.1.2.4 Component Design

6.1.2.4.1 Piping

The HPI System piping takes into consideration normal and emergency operating conditions. The LPI System piping and valves are subjected to more severe conditions during normal operation during decay heat removal than during emergency operation. The design pressures and temperatures of these systems are shown in Table 6-3. To assure system integrity, major piping has welded connections except where flanges are dictated for maintenance reasons. The piping shown on Figure 6-1 is designed, fabricated, and inspected in accordance with ANSI Nuclear Power Piping Code B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The cold leg HPI nozzles have temperature sensing elements attached to the top and bottom portions of the end of the nozzles. The locations of the temperature sensing elements are used to determine if temperature abnormalities, i.e. thermal stratification, are occurring at the nozzle end where the HPI piping is welded to the nozzle. This might occur if leakage occurred from the HPI isolation valves.

There are temperature sensing elements attached to the top and bottom portions of the piping upstream of HPI check valves MU-34A, MU-34B, MU-34C and MU-34D. The temperature sensing elements are used to determine if the check valves are leaking, i.e. allowing RCS to backflow through the HPI piping. The temperature sensing elements will also provide the information necessary to determine the temperature effect, i.e. thermal stratification, on the HPI piping.

6.1.2.4.2 Pumps

The pumps used in the emergency injection systems are of proven design and are used in many other applications. Periodic testing is performed on the pumps in accordance with ASME OM Code to insure operability and ready availability. Historical data removed - to review exact wording please refer to Section 6.1.2.4 of the SAR.

Curves of total dynamic head (TDH), net positive suction head (NPSH), and brake horsepower (BHP) versus flow are shown in Figure 6-6 for the HPI pumps and in Figure 6-7 for the LPI pumps.

6.1.2.4.3 Heat Exchangers

The decay heat removal coolers are designed and manufactured to the requirements of the ASME VIII and III-C and the TEMA-R (Tubular Exchanger Manufacturers Associations) standards. Historical data removed - To review exact wording please refer to Section 6.1.2.4.3 of the SAR.

The decay heat removal coolers are designed to remove sensible heat and the decay heat generated during a normal shutdown. Each decay heat removal cooler is capable of cooling the injection water during the recirculation mode following a LOCA to remove decay heat and thus provide adequate core cooling. The predicted heat transfer capability of the decay heat removal cooler as a function of recirculated water temperature is illustrated in Figure 6-8.

6.1.2.4.4 Valves

All valves in the emergency injection systems (Figure 6-1) were inspected in accordance with the code as designated in Table 6-2 and are reinspected as part of the ASME Section XI inservice inspection program. Electric motor operators for valves are designed to operate under the emergency service conditions and are capable of operating at 80 percent rated voltage.

The seats and discs of these valves are manufactured from materials that prevent galling and seizing. All remotely operated valves shown on Figure 6-1 are of the backseating type.

Low capacity relief valves are provided to protect the low pressure piping and components from overpressure.

6.1.2.4.5 Instrumentation

The engineered safeguards actuation instrumentation for the emergency injection systems employs redundant channels and signals as described in Chapter 7. The control room layout is arranged so that all indicators and alarms are grouped in one sector at a convenient location for viewing. The status of all ECCS equipment can be monitored from this area both during the accident and in the post-accident cooling mode.

6.1.2.4.6 Coolant Storage

The makeup tank (T4) has a total volume of 600 ft^3 and normally contains approximately 300 ft^3 of water. This tank provides water to the operating HPI pump lined up as a makeup pump until the BWST outlet valve opens. The makeup tank is designed and inspected in accordance with the requirements of ASME III-C.

The BWST is described in Chapter 9. Provisions are made for sampling the water and for adding concentrated boric acid solution or demineralized water. Boron in the tank is maintained at a concentration specified in the Technical Specifications. When the borated water storage tank is required to be operable, the manual valve on the discharge line from the borated water storage tank is locked open.

Each CFT contains the required volume and concentration of borated water at the concentration of boron as specified in the Technical Specifications.

6.1.2.4.7 Motors

All ECCS engineered safeguards pump motors are air cooled and have service factor of 1.15 of design rating. Insulation for these motors is rated at least to Class B. The motors are capable of starting at 80 percent rated voltage and are designed to be able to accelerate to full speed within six seconds after receiving a signal to start.

6.1.2.5 Reliability Considerations

System reliability is assured by the system functional design, including the use of normally operating equipment for safety functions, testability provisions, and equipment redundancy; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the General Design Criteria. There is sufficient redundancy in the ECCSs to protect against any credible single failure. This is demonstrated by the single failure

analysis presented in Table 6-4. This analysis is based on the assumption that a major LOCA occurs coincidentally with a malfunction or failure in the engineered safeguards systems. For example, the analysis includes realistic malfunctions or failures such as electrical circuit or motor failures, valve operator failures, etc. Table 6-4 also presents an analysis of possible malfunctions of the Core Flooding System that could reduce its post-accident availability. It is shown that these malfunctions result in indications that are obvious to the operators so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable since a program of periodic testing is incorporated in the station operating procedures. The adequacy of equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in Chapter 14. This analysis shows that only one HPI pump, one LPI pump, and one decay heat removal cooler (two of each are normally available) in combination with the Core Flooding System are required to protect against the full spectrum of break sizes.

6.1.2.6 Missile Protection

Protection against missile (tornado generated, internally generated, etc.) damage is provided by either direct shielding or physical separation of duplicate equipment. Most of the ECCS piping within the reactor building is located outside the primary and secondary shielding and hence is protected from missiles originating within these areas. The portions of the injection lines located between the reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area.

The HPI lines enter the reactor building via penetrations with physical separation. Each injection line splits into two lines outside the reactor building to provide four injection paths to the RCS. The four connections to the RCS are located between the Reactor Coolant Pump discharge and the reactor inlet nozzles. There are four injection lines penetrating the missile shield, minimizing the effect on injection flow in the unlikely event of missile damage to an injection line inside the secondary shield.

Protection from missiles is given to the LPI lines within the reactor building. The portion of the system located in the reactor building consists of two redundant injection lines which are connected to injection nozzles located on opposite sides of the vessel. Both redundant suction lines from the sump are missile protected. The sump suction is located outside of the secondary shielding.

The entire Core Flooding System is located within the reactor building. The CFTs and two of the three valves in each core flooding line are located outside the secondary shield.

6.1.2.7 Actuation

A description of the engineered safeguards protective system is given in Chapter 7. Table 6-12 gives design ECCS actuation setpoints. ECCS actuation setpoints are provided in the Technical Specifications.

6.1.3 DESIGN EVALUATION

6.1.3.1 High Pressure Injection System

Two pumps are actuated upon receipt of an emergency safeguards initiation signal. The safety analysis in Chapter 14 shows that one HPI pump is sufficient to prevent core damage for those smaller leak sizes which do not allow the RCS pressure to decrease rapidly to the point at which

LPI is operative. The HPI System is fully operational within 35 seconds of receiving an actuation signal, except for the make-up pump recirculation isolation valves CV-1300 and CV-1301. The recirculation isolation valves meet an HPI System requirement of closing within 50 seconds after receiving an actuation signal. The 35 and 50 second HPI response times assume a loss of offsite power coincident with the actuation signal and include diesel load times of 20 seconds for the primary make-up pumps and 15 seconds for the motor operated valves.

In addition to meeting the requirements of providing operational makeup flow, each nominal HPI pump meets the requirement of delivering 480-500 gpm at a vessel pressure of 600 psig. The HPI flow is reduced by 50 gpm at this pressure until the recirculation isolation valves close. This temporary reduced flow (430-450 gpm at 600 psig) is within the bounds of the accident analysis. One of the HPI pumps is normally in operation and a positive static head of water assures that all pipe lines are filled with coolant. The injection lines contain thermal sleeves at their connections into the reactor coolant piping to prevent overstressing the pipe juncture.

To assure adequate makeup capability for the full range of small break LOCAs, each HPI pump is piped to all four injection lines. Redundant flow instrumentation and throttling globe valves were also added to prevent pump runout and allow flow balancing between the four injection paths.

Operation of this system does not normally depend on the operation of any other engineered safeguard. However, the system can be operated in series with the LPI System, if needed, in the recirculation mode.

6.1.3.2 Low Pressure Injection System

Two pumps can deliver 6,000 gpm to the reactor vessel through two separate injection lines when the vessel pressure is at approximately 100 psig. Following a LOCA, assuming a simultaneous loss of normal power sources, the emergency power source and the LPI System can be in full operation within the required 35 seconds after actuation. The analysis of the LOCA in Chapter 14 shows that half of this system (assumed to inject below a pressure of 203 psia with a maximum modeled injection flow of 3150 gpm at 15 psia) is adequate to maintain core cooling.

The LPI System is connected with other safeguards systems in four respects: (1) the HPI, the LPI, and the Reactor Building Spray Systems take their suction from the BWST, (2) the LPI pumps and the reactor building spray pumps share common suction lines from the reactor building sump during the coolant recirculation mode, (3) the LPI System and the Core Flooding System utilize common injection nozzles on the reactor vessel, and (4) the HPI System may be operated through the LPI System if recirculation through the HPI pumps is required.

Following a LOCA, if one Decay Heat Removal (DHR) (LPI) pump fails to start, the cross connection shown in Figure 6-1 and associated cavitating venturies insures that at a vessel pressure of approximately 100 psig: (1) 2,700 gpm of borated water is injected into the reactor vessel for core cooling in the event of a large reactor coolant pipe rupture and (2) approximately 800 gpm of borated water is injected into the reactor vessel for core cooling in the event of a core flooding line break. The cross connection shown in Figure 6-1 and associated cavitating venturies also insure that no greater than 3647 gpm flow (3547 gpm LPI flow + 100 gpm LPI pump recirculation flow) will occur through a DHR pump, while on sump recirculation.

The NPSH available to the DHR pumps during recirculation from the reactor building emergency sump are calculated based on:

A. Verified isometric piping drawings;

B. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410;

C. An entrance loss of 0.78 velocity heads (K = 0.78);

D. Constant maximum recirculation flow rates of 3547 and 1320 gpm for the decay heat and building spray pumps, respectively;

E. Reactor building total pressure (partial pressures of air plus vapor) equal to the minimum TS allowed pressure;

F. Density of the sump water (varies with Reactor Building Sump temperature);

G. Reactor building sump water level at elevation 341.39 feet;

H. Decay heat removal pump suction at elevation 319.83 feet.

The results of the NPSH calculation are shown below for the decay heat pumps based on the above-mentioned assumptions. The NPSH required values were obtained directly from certified pump curves at a flow rate of 3647 gpm for the DHR pumps.

	Maximum Design Flow Rate Including Recirculation (gpm)	NPSH Available (ft)	NPSH Required (ft)
Decay Heat Removal Pump A	3647	14.5	9.2
Decay Heat Removal Pump B	3647	13.8	9.0

NPSH Available above for the DHR pumps is calculated based upon clean strainer head loss (no debris present). The RB Sump Strainer head losses are compared against NPSH Margin (NPSH Available – NPSH Required) to determine the acceptability of RB Sump and DHR Pumps.

6.1.3.3 Core Flooding System

Injection response of the Core Flooding System is dependent upon the rate of reduction of RCS pressure. The capability of the Core Flooding System to reflood the core is described in Chapter 14.

The core flooding nozzles and lines are designed to assure that they will accommodate the differential temperature which occurs between the injection mode and the recirculation mode.

6.1.4 TESTS AND INSPECTIONS

All active components, listed in Table 6-5, of the emergency injection system are tested periodically to demonstrate system readiness. Performance of active systems is tested by establishing flow and observing pressures and flows. The HPI System is inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. During operational testing of the LPI pumps, the portion of the system subjected to pump pressure is inspected for leaks. Items for inspection are pump seals, valve packing, flange gaskets, heat exchangers, and safety valves for leaks to atmosphere.

6.2 REACTOR BUILDING SPRAY SYSTEM

6.2.1 DESIGN BASIS

The Reactor Building Spray (RBS) System is designed to furnish building atmosphere cooling to reduce the post-accident building pressure to nearly atmospheric pressure. Containment Sump Buffering is provided to ensure iodine fission products remain in solution and is accomplished via the Containment Sump Buffering Agent System, which employs sodium tetraborate in baskets located throughout containment.

6.2.2 SYSTEM DESIGN

6.2.2.1 System Description

A schematic diagram of the system and Corresponding interfaces are shown on Figure 6-1.

The Reactor Building Spray System serves only as an engineered safeguard and performs no normal operating function. In the event of a major LOCA, the system sprays a chemical spray solution into the reactor building atmosphere to reduce the post-accident energy and to reduce fission product inventory in the reactor building atmosphere. The system consists of two pumps (P35A, P35B), two reactor building spray headers, and the necessary piping, valves, instrumentation, and controls. The pumps and remotely operated valves may be operated from the control room. The Reactor Building Spray System is sized to furnish 100 percent of the design cooling capacity with both of the spray paths in operation. Both paths operate independently. The Reactor Building Spray System also operates separately from the reactor building cooling units.

A high reactor building pressure signal from the Engineered Safeguards Actuation System (ESAS) initiates Reactor Building Spray operation. Table 6-12 shows the reactor building pressure actuation level. (For further reference, see the Technical Specifications.) The two pumps start and take suction initially from the Borated Water Storage Tank (BWST) through the Low Pressure Injection System piping. The chemical spray solution is injected in the building atmosphere through the spray headers and nozzles.

The Containment Sump Buffering Agent System is a passive system that uses the dry form of sodium tetraborate placed in three (3) baskets located near the sump and anchored to the Reactor Building floor. The baskets are 6 feet by 6 feet and 4.5 feet high and have perforations to allow coolant into the baskets from the sides. Following a LOCA event, coolant and spray from the RBS system flood the reactor building floor. During flood-up, the coolant mixes with the sodium tetraborate increasing its pH into the alkaline range (7.0 - 9.0) which retains iodine in solution in the sump water.

After Reactor Building Spray operation has been initiated the operator is directed to throttle the building spray flow to establish acceptable flows for recirculation from the reactor building sump. Throttled Reactor Building Spray flow conditions have been assessed at flows as low as 1000 gpm and as high as 1320 gpm (to account for potential instrument error) and as early as immediately upon spray initiation (to account for all possible operator action times). After the water in the BWST reaches a low-low level, the spray pump suction is transferred to the reactor building sump when the operator places the Low Pressure Injection (LPI) System in the recirculation mode.

The operator's primary source of information directing this manual transfer of the spray system pump suction is the level indication on the BWST. Both low level and low-low level alarms sound as the water is pumped from the tank. Secondary annunciation comes from low flow alarms on the discharge of the reactor building spray pumps and the DHR pumps.

The spray system is an Engineered Safeguards System and the electrical supply is designed in accordance with Section 8.1.4 which states "No single component failure will prevent operation of the required engineered safeguards."

Flow into the recirculation suction is totally screened. Debris larger than 1/16-inch in diameter cannot enter the piping. The sump strainer assembly is designed so that debris cannot fall into or block the common suction piping for the reactor building spray and low pressure injection systems. The strainer assembly is permanently installed and structurally supported such that a hole in a portion of the strainer cannot bring about a failure or collapse of the entire assembly. Strainer cartridges were fabricated from perforated stainless steel sheet metal. A partition is installed between the two intakes to prevent common failure. The sump strainer will not generate vortices during its design basis support function.

There are three, four-inch and one, two-inch Reactor Building drain lines that directly drain into the sump from various levels. These drain lines provide a common header that drain the 335' elevation floor drains and three headers from the upper elevation. These drain lines are provided with screens or caps to prevent debris larger than 1/16-inch from entering the sump. Other basement area floor drains are routed directly to the sump. Drain lines directly entering the sump also feature debris strainers. The total RB Sump Strainer surface area is 2715 ft^2.

The Reactor Building Sump (RBS) analysis during post accident conditions complies with the requirements of Generic Letter 2004-02 for addressing Generic Safety Issue 191, as detailed in ANO's compliance response to the Generic Letter. The updated design basis includes the following:

1. Debris quantities generated by a LOCA are determined by analysis that establishes the distance from a break at which various insulation, coatings, and other detrimental materials are released. A limiting break is determined from this analysis based on the debris mixture that produces the most detrimental strainer head loss.

2. Chemical effects associated with the precipitation of aluminum compounds are evaluated for their impact on sump strainer head loss and reactor fuel assemblies.

3. Strainer head loss due to debris and chemical precipitates was determined via scaled testing of strainer modules. The head loss test results were compared against available NPSH margin or strainer structural limits for acceptability.

4. The NPSH margin for the LPI and Reactor Building Spray pumps were revised to include available margins as sump temperatures become sub-cooled. This allows determination of the sump water temperature range at which NPSH becomes less limiting to strainer allowable head loss than the design structural limitation of the strainer of approximately 10.8 feet.

5. The sump strainer assemblies are analyzed and credited for vortex protection of the suction lines to the LPI and RBS pumps in the sump.

6. Down Stream Effects are evaluated for components in the sump recirculation flow path to ensure the debris generated by a LOCA that can pass through the strainer's 0.0625-inch openings will not result in blockage or unacceptable wear.

7. The reactor vessel internals, including fuel assemblies, are evaluated for potential detrimental effects from the debris generated by a LOCA, including chemical effects precipitates, that can pass through the strainer. Large-scale fiber bypass testing, model development, and in-core fiber calculations were utilized to satisfactorily evaluate that in-vessel debris will not prevent long term core cooling.

6.2.2.2 Quality Control Codes and Standards

The equipment is designed to the applicable codes and standards given in Chapter 9 and Tables 6-2 and 6-7.

Components and piping in the Reactor Building Spray System are designated as Seismic Category 1 equipment and are designed to maintain their functional integrity during an earthquake. Chapter 5 defines the acceptable stress limits for Category 1 equipment; Chapter 1 describes the quality program.

6.2.2.3 Material Compatibility

All wetted materials are compatible with the reactor coolant and chemical additions. The major components of the system are constructed of stainless steel. Minor parts such as pump seals utilize other corrosion resistant materials.

None of the active components of the Reactor Building Spray System are located within the reactor building, so none are required to operate in the steam-air environment produced by the DBA LOCA. Information on the environmental qualification of electrical equipment to function in accident environments is provided in Chapter 7.

6.2.2.4 Component Design

6.2.2.4.1 Piping

Except for the section of lines requiring flanged connections for maintenance, the entire system is of welded construction. Table 6-3 lists the design conditions for this system. The piping is designed, fabricated, and inspected in accordance with ANSI Nuclear Power Piping Code B31.7, or later appropriate ASME Section III Code sections provided that they have been reconciled.

6.2.2.4.2 Pumps

The reactor building spray pumps are similar to those used in refinery service. Periodic testing is performed on the pumps in accordance with ASME OM Code to insure operability and ready availability. Historical data removed - to review exact wording please refer to Section 6.2.2.4 of the FSAR.

The net positive suction head (NPSH) available to the reactor building spray (RBS) pumps during recirculation from the reactor building emergency sump are calculated based on:

A. Verified isometric piping drawings;

B. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410;

C. An entrance loss of 0.78 velocity heads (K = 0.78);

D. Constant maximum recirculation flow rates of 3547 and 1320 gpm for the decay heat and building spray pumps respectively;

E. Reactor building total pressure (partial pressures of air plus vapor) equal to the minimum TS allowed pressure;

F. Density of the sump water (varies with Reactor Building Sump temperature);

G. Reactor building sump water level at elevation 341.39 feet;

H. RBS pump suction at elevation 319.42 feet; and

The results of the NPSH calculation are shown below for the spray pumps based on the above-mentioned assumptions. The NPSH required values are obtained directly from pump test curves at a flow rate of 1,320 gpm for the building spray pumps.

	Recirculation Maximum Flow Rate (gpm)	NPSH Available (ft)	NPSH Required (ft)
RB Spray Pump A	1320	17.0	11.0
RB Spray Pump B	1320	15.1	11.0

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-9.

6.2.2.4.3 Valves

The valves of the RBS System are designed and inspected to the same code requirements as the valves in the Emergency Core Cooling Systems (see Section 6.1.2.4).

6.2.2.4.4 Spray Headers and Nozzles

The reactor building spray nozzles are arranged on each of the two sets of concentric reactor building spray headers. The spray nozzles are spaced in the headers to give uniform spray coverage of the reactor building volume above the operating floor with one or both of the spray header systems in operation. Nozzles are located no closer than eight feet and no further than 10 feet from any adjacent nozzle. The nozzles are 1-piece construction Spraco 1713A type, which feature a ramp bottom design that gives a uniform hollow cone spray.

The potential for nozzle clogging is minimal. The nozzles have large, free, unobstructed passages with a pressure drop across the spray nozzles of 40 psi under design basis conditions. The average drop size is approximately 400 microns with the largest size being 700 microns at design basis conditions. The flow rates when the RBS system is aligned to the borated water storage tank are dependent on the reactor building pressure and borated water storage tank level. The initial flow rate per header is expected to be greater than 1500 gpm and higher flowrates are possible as the reactor building pressure decreases. Following operator action to throttle flow, the flow rate per header for the injection and recirculation phases of system operation is 1050-1200 gpm. The fall height of the spray is approximately 115 feet. The spray system heat removal efficiency is 100 percent.

6.2.2.4.5 Instrumentation

The engineered safeguards actuation instrumentation for the RBS System employs redundant channels and is described in Chapter 7.

The reactor building high pressure setpoint signal, which initiates the RBS pumps to provide spray flow for reactor building cooling, is set at 30 psig. The setting is sufficiently high to prevent spurious actuation of the RBS System and yet low enough to provide a signal in the event of large LOCAs where high building pressure is a concern. The reactor building design pressure transient evaluation, which uses this setpoint plus an arbitrary 300-second delay, shows in Chapter 14 that the reactor building design pressure limits are not exceeded.

The inherent delay times are the (1) times required to reach the 30 psig setpoint following a LOCA, (2) the time required for diesel start and the engineered safeguards loading sequence, (3) the time for the reactor building spray valves to open and pumps to come up to speed, and (4) the time it takes to fill the reactor spray piping with spray solution. The times to reach the 30-psig setpoint for the break sizes in Figure 14-58 are as follows:

Break Size, ft^2	Time, sec	Peak Pressure, psig
14.0	3	51.3
8.5	4	52.6
5.0	7	53.1
3.0	10	52.5
1.0	30	49.8

The RBS pumps deliver full flow within 40.2 seconds after the actuation signal setpoint is reached. The time required to fill the spray line from the isolation valves to the nozzles at 1,500 gpm is 56 seconds.

6.2.2.4.6 Coolant Storage

The RBS System shares BWST capacity with the HPI and LPI Systems. The BWST provides a source of water to the RBS pumps until the pump suction is transferred to the reactor building sump.

6.2.2.4.7 Motors

The RBS pump motors are designed to the same requirements as the Emergency Core Cooling System motors. Refer to Section 6.1.2.4.

6.2.2.5 Reliability Considerations

A failure analysis was made on all active components of the system to show that the failure of any single active component does not prevent fulfilling the design function. This analysis is shown in Table 6-6.

6.2.2.6 Missile Protection

Protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary and secondary concrete shield.

6.2.2.7 Actuation

A description of the engineered safeguards protective system is given in Section 7.1.3. Table 6-12 gives spray actuation setpoints.

6.2.3 DESIGN EVALUATION

The RBS system, acting independently of the Reactor Building Cooling System, can reduce the building pressure to near atmospheric level after a LOCA. In combination with the reactor building cooling units, it provides redundant alternative methods to maintain containment pressure at a level below design pressure. Either of the following combinations of equipment can provide sufficient heat removal capability to accomplish this:

A. The RBS System.

B. One reactor building cooling unit and the RBS System operating at one-half capacity.

6.2.4 TESTS AND INSPECTION

The components of the RBS System are tested as shown in Table 6-8.

During these tests, the equipment is visually inspected for leaks. Valves and pumps are operated and inspected following maintenance on the system to assure proper operation.

6.3 REACTOR BUILDING COOLING SYSTEM

6.3.1 DESIGN BASES

Reactor building cooling is provided to maintain reactor building temperature within design limits during normal operation as well as to maintain the post-accident reactor building pressure and temperature to values shown to be acceptable by the accident analysis (See Section 14.2.2.5.5).

6.3.2 SYSTEM DESCRIPTION

The schematic flow diagram of the Reactor Building Cooling System and associated instrumentation is shown in Figure 6-4.

The normal cooling system consists of reactor building cooling fans (VSF-1A, VSF-1B, VSF-1C, VSF-1D, VSF-1E) and their associated chilled water cooling coils (VCC-1A, VCC-1B, VCC-1C, VCC-1D, VCC-1E). During normal plant operation, chilled water from the plant main water chillers is circulated through the normal cooling coils (VCC-1A, VCC-1B, VCC-1C, VCC-1D, VCC-1E) in each of the five units (VSF-1A, VSF-1B, VSF-1C, VSF-1D, VSF-1E).

The post-accident cooling system consists of reactor building cooling fans (VSF-1A, VSF-1B, VSF-1C, VSF-1D) and their associated service water cooling coils (VCC-2A, VCC-2B, VCC-2C, VCC-2D), thus accident heat loads are rejected to the Service Water System. The post-accident cooling system is arranged in two redundant trains of reactor building cooling. Each train consists of two fan-cooler units (VSF-1A/VCC-2A and VSF-1B/VCC-2B; and VSF-1C/VCC-2C and VSF-1D/VCC-2D) and their respective service water supplies.

During ESAS operation, VSF-1A, VSF-1B, VSF-1C, and VSF-1D receive start signals, the chilled water supply to cooling coils VCC-1A, VCC-1B, VCC-1C, VCC-1D and VCC-1E is isolated and service water flow is initiated to the emergency cooling coils VCC-2A, VCC-2B, VCC-2C, and VCC-2D. Under design service water flow conditions, the manufacturer’s cooler performance prediction model predicts that each of these units can remove 56.5 x 10^6 Btu/hr at design reactor building temperature conditions. Figure 6-5 shows the manufacturer’s predicted cooling coil characteristics versus building ambient conditions at design service water flow conditions. (See Note 4, Table 6-9 regarding heat removal rate assumed in the DBA analysis.) The design data for the cooling units are shown in Table 6-9.*

* The 95° F service water (cooling water inlet temperature) shown on Figure 6-5 and Table 6-9 is based upon the maximum expected temperature of the Lake Dardanelle Reservoir.

At times one fan-cooler unit in a train may be isolated for maintenance or other reasons. Service water flow to the unit not being utilized must be physically isolated by blind flanges since the individual cooler units are not provided with isolation valves. This action ensures that the remaining unit receives the total service water design flow. In this configuration, each individual fan-cooler unit has sufficient capacity, with the assumption that the total design flow is applied to that unit, to provide the required cooling should a loss of coolant accident occur.

6.3.3 DESIGN EVALUATION

The Reactor Building Cooling System provides adequate long-term cooling of the reactor building atmosphere and supports the Reactor Building Spray System to assure sufficient heat removal capability to maintain the post-accident reactor building pressure and temperature to

values shown to be acceptable by the accident analysis (See Section 14.2.2.5.5). The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils to transfer heat from the reactor building to the Service Water System.

The distribution of the supply air to the compartments and areas in the reactor building is based on the heat gains in the respective compartments and areas. The major air supply is from the lower levels and is ducted into the compartments and distributed within these compartments in such a manner as to thoroughly mix the air. The air within the reactor building has forced movement from the compartments and areas to the upper areas which adequately mix the atmosphere before being returned to the cooling system.

As a result of degraded service water piping conditions, design flow is not currently available to all four units. Design flow is available to one unit per train when the opposite unit in that train is isolated. Evaluation of the Reactor Building Cooling System and Reactor Building Spray System response to the Design Basis Accident, presented in Chapter 14, has demonstrated that one cooler unit with one spray train will provide sufficient heat removal capability to maintain the post accident reactor building pressure and temperature below acceptable values. Operation of two air coolers in one train at reduced service water flow provides more heat removal capability than one unit at design flow.

6.3.3.1 Reliability Considerations

A single failure analysis for all components of the systems as shown in Table 6-10 is used to show that the failure of any single component does not prevent fulfilling of the design functions. This analysis is based on the assumption that a major LOCA occurred. It is then assumed that an additional malfunction or failure occurs either in the process of actuating the reactor building emergency cooling system or as a secondary accident effect. The analysis includes malfunctions or failures such as electrical circuit or motor failures, stuck valves, etc. In general, failures of the type assumed in this analysis are unlikely because periodic testing and normal operation would identify deficiencies.

The fan motors and power cables which are required for proper functioning of the Reactor Building Cooling System are tested and proven to operate under the post-accident environmental conditions, or a prototype of the same is tested and proven to operate under these conditions. Chapter 7 includes additional information on the environmental qualification of electrical components to function as required during and after the Design Basis Accident (DBA).

In order to insure that critical ducts remain intact following a DBA, all ducts are designed and supported for a Seismic Class 1 earthquake. The supply ducts from the fan discharge on the cooling system are provided with relief valves to prevent their collapse. A detailed schematic of the reactor building ventilation units and ducts is shown in Figure 5-7.

6.3.3.2 Reactor Building Cooling Response

During normal operation, air recirculation is established through the chilled water coils of the five reactor building cooling units. Under accident conditions the chilled water coils are bypassed and air is recirculated through the service water coils in four units (VCC 2A, B, C and D).

This action is initiated through an Engineered Safeguards (ES) signal when the reactor building pressure increases to approximately 4 psig.

6.3.3.3 Special Features

The casing design for the ventilation units is conventional in nature but is designed to withstand an exterior pressure of 2 psig. The housing of the units is equipped with pressure equalization devices to preclude collapse in the event of high pressure. These devices open at 10 inches water gauge pressure. The ventilation units are located outside the concrete shield for the reactor vessel, steam generators, and Reactor Coolant Pumps at an elevation above the water level in the bottom of the reactor building at post-accident conditions. In this location, the systems in the reactor building are protected from credible missiles and from flooding during post-accident operations. This location also provides shielding so that the design radiation level allows for maintenance, repair, and inspections to be performed during power operation.

The units are approximately 9' x 16' x 11'. Each unit consists of one set of chilled water cooling coils to be used for normal cooling, one set of service water coils to be used for emergency cooling, followed by an axial type fan. The fan is mounted vertically on top of the unit. The fan motors are of the type that are tested in accordance with IEEE Report No. NSF/TCS/SC2-A, "Proposed Guide for Qualification Tests for Class 1E Motors Installed Within the Containment of Nuclear Fueled Generating Stations." A bypass damper is provided on top of the unit between the chilled water coils and the service water coils. The damper opens on a high (4 psig) reactor building pressure ES signal allowing the saturated air in the reactor building to bypass the suction ducts and the chilled water cooling coils and to go through only the service water coils. The damper permits the internal pressure in the unit to equalize with the external pressure. The decrease in pressure drop due to the bypassing of the chilled water cooling coils permits the fan in the unit to handle the necessary quantity of air for emergency cooling purposes at the same speed as required for normal operation, thereby precluding the necessity of two speed motors with their additional controls and wiring. The units are provided with condensate drains that will remove condensate at a rate which will preclude reduction in cooling capacity of the coolers.

6.3.3.4 Actuation

A description of the engineered safeguards protective system is given in Chapter 7. Table 6-12 gives emergency cooling actuation setpoints. (For further reference, see the Technical Specifications.)

6.3.4 TESTS AND INSPECTIONS

Active components of the ventilation units are normally in service. Valving on the chilled water cooling coils is periodically cycled from the control room. Additionally, the bypass damper is tripped open by a signal from the control room. However, since this is a fail-safe (open) damper, it must be reset manually. The service water coils are periodically tested by opening the containment isolation valves in the Service Water System.

6.4 ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS

6.4.1 INTRODUCTION

The use of normally operating equipment for engineered safeguards functions and the location of some of this equipment outside the reactor building requires that consideration be given to direct radiation levels from the fluids circulating in these systems and the leakage from these systems after fission products accumulate.

The shielding for components of the Engineered Safeguards System is designed to meet the following objectives in the event of a maximum hypothetical accident.

A. To provide protection for personnel to perform all operations necessary for mitigation of the consequences of the accident.

B. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the required equipment.

6.4.2 SUMMARY OF POST-ACCIDENT RECIRCULATION

Following a Loss of Coolant Accident (LOCA), flow is initiated in the High Pressure Injection (HPI) and Low Pressure Injection (LPI) Systems from the Borated Water Storage Tank (BWST) to the reactor vessel. Flow is also initiated by the Reactor Building Spray (RBS) System to the building spray headers. When the BWST inventory is depleted, recirculation from the reactor building sump is initiated by the operator for both the LPI flow and the reactor building sprays. If elevated RCS pressure requires piggyback operation, recirculation through the HPI System will also exist. The post-accident recirculation flow paths include all piping and equipment external to the reactor building as shown on Figure 6-1 up to the valves leading from the BWST.

6.4.3 BASES OF LEAKAGE ESTIMATE

While the reactor auxiliary systems involved in the recirculation complex are closed to the auxiliary building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

A. Valves

  1. Disc leakage when valve is on recirculation system boundary.

  2. Stem leakage.

  3. Bonnet flange leakage.

B. Flanges.

C. Pump shaft seals.

While leakage rates are assumed for these sources, maintenance and periodic testing of these systems precludes all but a small percentage of the assumed amounts. With the exception of the interfacing system isolation valve discs, all of the potential leakage paths are examined during periodic tests or normal operation. In order to minimize the probability of leakage from the RCS to low pressure system piping during accident conditions, operating procedures require that an operator observe, during hot shutdown following each cold shutdown or refueling, local pressure indication between the decay heat isolation check valves and between the outer check valves and the motor operated valves. If leakage is indicated, it is quantified. The check valve is declared inoperable if leakage exceeds 5 gpm. The interfacing system isolation valve disc leakage is retained in the other closed systems and, therefore, is not released to the auxiliary building. While valve stem leakage is assumed, the manual valves in the recirculation complex are of the backseating type and do not rely on packing alone to prevent stem leakage.

6.4.4 DESIGN BASIS LEAKAGE

Values assumed for the total leakage quantities of the various components which would be in contact with the recirculated fluid appear in Table 6-11.

Section 14.2.2.5 presents an analysis of the effects associated with the release of the radioactive fluid.

6.5 REACTOR BUILDING PENETRATION ROOM VENTILATION SYSTEM

6.5.1 DESIGN BASES

This system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post-accident reactor building leaks. Experience has shown that reactor building leakage is more likely at penetrations than through the liner plates or weld joints.

6.5.1.1 Design Criteria

The Reactor Building Penetration Room Ventilation System is designed to meet the following criteria:

A. The system is a Seismic Category 1 system.

B. The system includes the capability to withstand a single failure without loss of function.

C. Equipment is powered from the engineered safeguards buses.

D. The system is designed such that sufficient remote instrumentation and controls are available to permit the operator to monitor and control equipment operation from the control room.

6.5.2 SYSTEM DESIGN

6.5.2.1 System Description

The system schematic and characteristics are shown on Figure 6-10. Penetration rooms are formed adjacent to the outside surface of the reactor building by enclosing the area around the majority of the penetrations. The only penetrations which do not pass through this area are:

A. Two main steam lines

B. Permanent equipment hatch which contains a double gasketed closure

C. Emergency personnel access lock

D. Refueling tube

E. Purge lines

F. Containment Sump Penetrations

G. Quench Tank Drain

H. Reactor Building Drain Header

The purge lines and Containment Sump penetrations are not considered a source of significant leakage because they are welded to the liner plate. The Quench Tank drain and Reactor Building drain header are not considered a source of significant leakage because they are embedded in the base slab. The main steam lines are not considered a source of significant

leakage because they are connected to the liner plate by welding through flued heads. The access openings are tested during normal operation and are not considered sources of significant leakage. There are double seals at each of these access openings. The refueling tube opening is covered with a gasketed blind flange which is removed only during plant shutdown for transfer of fuel to the spent fuel pool. Also, the Containment Sump penetrations, Quench Tank drain, and Reactor Building drain header penetrations are located in the Decay Heat Removal Pump Rooms that are isolated post-LOCA. The main function of the system is to control and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions. When the system is in operation, a negative pressure is maintained in the penetration room to assure that any leakage goes into the penetration room.

Leakage into either of the penetration rooms is discharged to a vent through a pair of filter assemblies (one redundant), each consisting of a prefilter, high efficiency particulate filter and a charcoal filter in series, and two full-size constant speed fans (one redundant) designed to operate under negative pressures up to the fan discharge downstream of the filters. The required negative pressure in the penetration rooms can be maintained with system flow equal to 1800 scfm $\pm$ 10%.

The design evacuation rate from the penetration room far exceeds the maximum anticipated reactor building leakage. The design leak rate of 0.1 percent per day of reactor building free volume into the penetration rooms on the first day of the Maximum Hypothetical Accident amounts to a volume transfer rate of 1.25 scfm averaged over the entire day. The total leakage rate on the first day is assumed to be 0.2 percent, of which 50% is processed by the Penetration Room Ventilation System with the balance assumed to escape to the environment. This average leak rate into the penetration rooms can be compared to a design evacuation rate of 1800 scfm for the total of six penetration rooms.

Each filter assembly and each fan is designed to handle 2,000 scfm; the second filter assembly and second fan being a full-size standby (redundant) unit. Normal system flow is approximately 1800 cfm.

Following a Loss of Coolant Accident (LOCA), a reactor building isolation signal places the lead system in operation by starting the fan and opening the power operated butterfly valve on the outlet of the filter assembly. If the lead system does not achieve proper flow within 20 seconds, the lead system is automatically stopped and five seconds later the standby system automatically starts.

In the event of an excessive pressure drop across any filter or a high radiation reading on the filter assembly discharge, the standby system can be remotely started. Original studies showed that some airflow is required across an iodine-exposed charcoal filter to prevent the filter from reaching a combustion temperature of 650 °F and igniting. However, a conservative ANO calculation shows a margin of greater than 90 °F below the ignition temperature with no cooling flow to the idle loaded filter. To resolve a single failure issue, the cooling flow was permanently isolated.

The communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals. Therefore, it can be assumed that no pressure differentials exist in the room so that an instrument string sensing pressure at a single point can be used. Penetration room vacuum is displayed in the control room and low vacuum is annunciated. Fan operating status and the radiation level of filter effluent are displayed in the control room and high radiation is annunciated. Filter ΔP is

displayed locally and annunciated in the control room. Filter high temperature is annunciated in the control room. System flow rate is displayed adjacent to the remote control valve stations. The system may be manually actuated from the control room.

Particulate filtration is achieved by a medium efficiency prefilter and a high efficiency (HEPA) filter. Adsorption filtration is accomplished by an activated charcoal filter. The design basis requirement for each charcoal filter is to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during a Design Basis Accident (DBA) 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the reactor building.

6.5.2.2 Quality Control Codes and Standards

The equipment in this system is designed to Seismic Category 1 requirements and is in accordance with the following standards:

Pre-Filter (VFP-22A, VFP-22B) - Filter efficiency is determined by the "NBS Dust Spot Test" utilizing atmospheric dust for original filters. Replacement filter efficiency is determined by ASHRAE Standard for atmospheric dust spot efficiency test.

Absolute Filter (VFA-6A, VFA-6B) - The basic design criteria for this filter is set forth in AEC Health and Safety Bulletin 212 (6-25-65), which incorporates U.S. Military Specification MIL-F-51068A captioned "Filter, Particulate, High Efficiency, Fire Resistant." In addition, the dust holding capacity is determined by utilizing the test procedures of NBS and ASHRAE Standard.

Adsorptive Charcoal Filter (VFC-5A, VFC-5B) - The specified ignition temperature of the charcoal is checked using the procedure set forth in USAEC Report DP-1075, "High Temperature Absorbents of Iodine," by R. C. Milhaus. This test is conducted on one sample from each lot of charcoal.

Fans (VEF38A, VEF38B) - Fan performance is determined by prototype test according to procedures set forth by the Air Moving and Conditioning Association (AMCA) 1960 Standard Test Code.

6.5.2.3 Component Design

Components are designed to Seismic Category 1 standards. Each system is powered from a separate engineered safeguards bus. Appropriate alarms are annunciated in the control room which indicate high radioactivity releases as determined by dedicated process monitors described in Chapter 11, fan or filter failure, or high filter ∆P or temperature.

6.5.3 DESIGN EVALUATION

The system is provided with two fans and two filter assemblies. The fans discharge to the unit vent and are controlled from the main control room.

During normal plant operation this system is not running. An Engineered Safeguards Actuation System (ESAS) Signal actuates the lead system. Control room instrumentation monitors operation and the system can be tested during normal operation.

The single failure analysis shown on Table 6-13 has been made on all components of the system to show that the failure of any single component will not prevent fulfilling the design functions. This analysis assumed that a major LOCA has occurred. It was then assumed that an additional malfunction or failure occurred either in the process of actuating the Penetration Room Ventilation System or as a secondary accident effect. The analysis includes malfunctions or failures such as electrical circuit or motor failures, stuck valves, etc. In general, failures of the type assumed in this analysis are unlikely because of a program of periodic testing.

6.5.4 TESTS AND INSPECTION

The ventilating equipment is designed and constructed in accordance with accepted industry standards for power station equipment and is accessible for periodic testing and inspection during normal plant operation.

The units are exterior to the reactor building and may be visually inspected and/or repaired at anytime except when in actual use subsequent to a LOCA.

Specific tests and inspections are performed on the HEPA filters, charcoal filters, and fan units to assure system operability. Periodic testing is also conducted to verify proper response to an actuation signal. Control board indication is used to verify that all components have responded properly to the actuation signal. The Technical Specifications include the detailed testing and inspection requirements.

6.6 POST-LOSS OF COOLANT ACCIDENT HYDROGEN CONTROL

6.6.1 DESIGN BASES

Following a design basis Loss of Coolant Accident (LOCA), hydrogen is generated from radiolytic decomposition of water, from metal-to-water reaction, and from corrosion of various metals and metal-containing materials. The rate of hydrogen generation was originally computed using the assumptions listed in Safety Guide 7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident." More recent analyses were performed using COGAP (NUREG/CR-2847) which is based upon Regulatory Guide 1.7. This hydrogen generation rate is used to calculate the hydrogen concentration inside the reactor building assuming uniform mixing within the entire reactor building free volume.

The specific sources of hydrogen generation considered inside the reactor building are as follows:

A. Radiolytic decomposition of water. Radiolytic decomposition of water in the reactor building sump and the reactor core occurs throughout the post-accident period.

   Values for parameters used as assumptions in calculating the energy released in the core and sump solution are the same as those listed in Regulatory Guide 1.7. The beta and gamma energy production rates are calculated by the COGAP computer code (NUREG/CR-2847) and are taken from "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", Branch Technical Position ASB 9-2. A value of 1000 days of reactor operation was assumed as input for calculating the energy release. The amount of hydrogen produced by radiolytic decomposition of water is presented in Figure 6-15 along with other sources of hydrogen.

B. Metal-to-water reaction. The amount of hydrogen generated by metal-water reaction will be five times the maximum amount calculated in accordance with 10 CFR 50.46, but no less than the amount that would result from reaction of all the metal on the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inch.

C. Reactor coolant water. It is assumed that the Reactor Coolant System water, including the pressurizer, instantaneously evolves its entrained hydrogen. Nine hundred and fifty scf of hydrogen is assumed in the reactor coolant and pressurizer.

D. Corrosion of aluminum. The integrated hydrogen production within the reactor building as a function of time for corrosion of aluminum is given in Figure 6-14. For purposes of the hydrogen generation analysis, all aluminum (Reactor Building sump analysis assumptions with respect to aluminum are different – see SAR Section 6.2.2.1) in the reactor building is included in one of the following four categories:

   I. Completely exposed with a thickness conservatively assumed to be 0.002137 inches. This category of aluminum is assumed to react very quickly. Assumed reaction rates are conservatively higher than the Regulatory Guide 1.7 values. Components assumed to react almost instantaneously include, but are not limited to, the reactor coolant system paint, and miscellaneous small parts. The total surface area assumed for this category is 10,000 ft^2.

II. Completely exposed with a thickness of more than 14 inches which translates to an almost infinite source of aluminum. Components assumed to react almost indefinitely include, but are not limited to equipment tags and instrument transmitters without qualified coating. The total surface area assumed for this category is 1,500 ft^2.

III. Painted with a qualified coating and assumed not to react following the Design Basis Accident (DBA).

IV. Sealed or otherwise protected from the building spray following the DBA and assumed not to react. Components that are not in contact with building spray include, but are not limited to, nuclear instrument housing (sealed in instrument thimbles) and reactor coolant motor air cooler fins (sealed in air box).

E. Zinc base paint. Approximately 162,000 ft^2 of inorganic zinc base paint with an inorganic top coat was applied during construction on the reactor building liner plate and on various equipment inside the reactor building. These coatings are assumed to react with the boric acid sprays and evolve hydrogen as an infinite source. (This is extremely conservative since the hydrogen evolution ceases when the sprays terminate). A total mass of 104,197 lb of zinc is assumed to corrode at conservatively high reaction rates dependent on reactor building spray pH (adjusted to room temperature conditions) and reactor building temperature. The amount of hydrogen generated from zinc paint is shown in Figure 6-14. Repairs of damaged coatings may use reformulated replacement coatings that contain materials different than the previous coatings, may not require a primer, and may not be the same type (e.g., inorganic). Although replacement coatings can decrease the zinc mass in the containment, analyses for post-accident conditions using previous zinc mass estimates are conservative. Coatings and repairs conform to the requirements of the standards specified in Section 5.2.1.3.5.

F. Zinc metal. There is approximately 56,000 ft^2 of galvanized steel in the reactor building. Analyses assume that 65,000 ft^2 of galvanized steel corrodes at conservatively high reaction rates dependent on reactor building spray pH (adjusted to room temperature conditions) and reactor building temperature. The amount of hydrogen generated from zinc metal as a function of time is presented in Figure 6-14.

G. Copper. There is approximately 500 ft^2 of exposed copper in the reactor building which is assumed to corrode at conservatively high reaction rates dependent on reactor building spray pH (adjusted to room temperature conditions) and reactor building temperature. Since copper is not very reactive, its contribution to the hydrogen generation is insignificant as shown in Figure 6-14.

Figure 6-11 shows computed hydrogen concentration versus time. As can be seen from the figure, using the above assumptions, hydrogen reaches the average lower flammability limit (four percent) in approximately 9 days assuming no pre-existing hydrogen concentration. To preclude the possibility of buildup of hydrogen inside the reactor building, redundant internal hydrogen recombiners are installed (1CAN048604). However, the NRC determined in its October 2003 change to 10 CFR 50.44 that hydrogen mitigation during postulated accident events is not required.

The internal hydrogen recombiners provided for combustible gas control in containment are designed to:

A. Limit the buildup of hydrogen in the reactor building to a concentration of 4.0 percent. This concentration represents the lower flammability limit of hydrogen in air or steam-air atmospheres and is well established and adequately conservative.

B. Provide redundancy and the capability to withstand a single failure.

6.6.2 SYSTEM DESIGN

6.6.2.1 System Description

The Combustible Gas Control System consists of two, physically separated, independent hydrogen recombiners. Each hydrogen recombiner is powered from a separate engineered safeguards bus and is designed to Seismic Category 1 standards. The hydrogen recombiners are Westinghouse flameless thermal hydrogen recombiners. The hydrogen recombiners are located within the reactor building and are environmentally qualified for operation following a Loss of Coolant Accident.

Two hydrogen samplers, which function independently of the hydrogen recombiner system, are each located on lines which were originally hydrogen purge lines prior to the 1986 installation of the internal hydrogen recombiners; one at the inlet and one at the outlet. The reactor building atmosphere is mixed by the spray system attached to the reactor building dome and by the emergency cooling units, providing adequate mixing to prevent formation of local hydrogen pockets.

10 CFR 50.44 and Regulatory Guide 1.7 (Rev. 2) recommends that reactor licensees should have the installed capability for a controlled purge of the containment atmosphere to aid in cleanup. This capability is found in the Reactor Building Purge System.

The containment penetrations to the hydrogen samplers meet the single-failure criteria for containment isolation and operation of the sampling system.

6.6.2.2 Codes and Standards

The internal hydrogen recombiner installation is designed to Seismic Category 1 requirements. The remaining components of the original hydrogen purge system are designed to Seismic Category 1 requirements and are in accordance with the following standards:

Reactor building isolation valves	NP&VC Class 2
Penetration piping	B31.7 Class 2*
Piping external to reactor building isolation valves	B31.7 Class 3*
Valves external to reactor building isolation valves	NP&VC Class 3
Filter container	ASME III - 3
Blowers/Compressors	ASME III - 3

*Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

6.6.2.3 Component Design

The Combustible Gas Control System, shown in Figure 5-7, consists of two Westinghouse flameless thermal hydrogen recombiners located within the reactor building at Elevation 426'. The hydrogen recombiners are convection flow, 100 scfm electric units rated at 75 kW each. They are designed to operate at 95 percent efficiency and each unit is capable of maintaining the hydrogen concentration in the reactor building below 4.0% following any postulated LOCA.

The internal hydrogen recombiners are basically passive systems which thermally recombine hydrogen and oxygen by the use of electric heating coils. The environmental qualification program consisted of a series of tests, some of which were run on a prototype recombiner and the balance of which were run on production components which demonstrated the equipment's ability to perform its required safety function.

The hydrogen recombiners are designed to Seismic Category 1 standards. Each recombiner is powered from a separate engineered safeguards bus. Appropriate instrumentation in the control room indicates when the system is operating. The initiation and control circuits conform to IEEE 279-1968 requirements.

6.6.2.3.1 Hydrogen Samplers

The hydrogen sampling capability consists of redundant hydrogen samplers which are provided to analyze hydrogen concentration in the reactor building atmosphere.

The hydrogen samplers are installed outside of the reactor building on lines which were originally part of the hydrogen purge, which was subsequently replaced by hydrogen recombiners. Reactor building penetrations for the original hydrogen purge continue to serve the hydrogen samplers.

The detector consists of a block with appropriately arranged cavities and gas passages. Four elements, usually in the form of a helix, are mounted in the cavities. Two elements are exposed to the sample gas. The other two elements are exposed to a reference gas of constant composition. The elements are electrically connected to form a bridge through which current is passed to heat the elements. The two elements exposed to the reference gas lose heat at a constant rate to the block and, consequently, have a constant temperature. The two elements exposed to the sample dissipate heat at rates which vary with the sample composition and, consequently, vary in temperature with sample variations. The element resistance changes with temperature so that the bridge is electrically unbalanced by difference in composition between the sample and reference gas. Through passive circuitry this unbalance is exhibited as a voltage change which is indicated or recorded as a measure of gas composition. The hydrogen monitoring equipment readouts are available locally and repeated in the control room.

The critical parameters for reliable thermal conductivity measurement are the block temperature control, the power supply stability with temperature and power source variations, the insensitivity of the bridge to power supply variations (bridge balance), and the lack of zero shift with gas composition. In the sampler, the block, power supply, and measuring circuit components are mounted on a thermostated heat sink in a separate compartment in the instrument. The sample passages go through the heat sink. The effects of ambient temperature changes, sample gas temperature changes, junction and potentiometer contact "thermocouples" are thus all controlled and minimized. The zener regulated power supply, which is also temperature controlled, attenuates input power variations so that they do not affect

stability. The bridge elements are selected and matched so that the detector is insensitive to power supply variations and the bridge zero is insensitive to gas composition. The calibrated range is zero to ten percent.

Devices for pretreatment of the sample include a cooler, a moisture trap, and a pressure regulator which maintains a constant pressure drop across the analyzer. Radiation levels in the sample gas do not affect the readout accuracy. The sample is pretreated to an acceptable moisture level and reactor building pressure reduced in order to protect the sampler from surges. There are no restrictions on sample quality.

The hydrogen sampler in the north piping penetration room, Elevation 360′0″, is shielded from both the activated charcoal filter on the exhaust side of the currently inactive hydrogen purge system and recirculation piping in the north room. This shielding keeps the dose rate absorbed by the hydrogen sampler below the criterion of 1×10^4 rads following a LOCA. The charcoal filter in the south piping penetration room was removed.

6.6.3 DESIGN EVALUATION

The hydrogen recombiners in the Combustible Gas Control System are under manual control only, actuated and monitored by an operator in the control room. It is placed in operation before hydrogen concentration reaches 4.0 percent, not expected to be reached before approximately 9 days after a DBA (LOCA) if there is no pre-existing hydrogen concentration.

The hydrogen recombiners are sized to recombine hydrogen and air at a rate of 100 scfm at 95% efficiency. One recombiner, operating at this 95% efficiency, is sufficient to prevent the hydrogen concentration from increasing above 4.0 v/o. There is sufficient excess capacity to quickly reduce the hydrogen concentration, as shown in Figure 6-11, even if the recombiner is not started until the hydrogen concentration reaches 4.0 v/o. The recombiner efficiency can be affected by the density of the air. To assure that a 4% hydrogen concentration is not reached prior to the time that reactor building post LOCA pressure drops to a value at which 95% recombiner efficiency can be assured, operational hydrogen concentrations in the reactor building that may develop over time as a result of leakage from the RCS are monitored and appropriately controlled.

Since the Combustible Gas Control System consists of two physically separated, 100 percent capacity trains, a single failure does not prevent adequate control of hydrogen.

6.6.4 TESTS AND INSPECTION

Specific tests and inspections are performed on the recombiners to assure system operability. The units are inside the reactor building and may be visually inspected and/or repaired during shutdown except when in actual use.

Table 6-1

CORE FLOODING SYSTEM COMPONENT DESIGN VALUES

Core Flooding Tanks	
Number	2
Design Pressure, psig	700
Operating Pressure, psig	600 ± 20
Design Temperature, °F	300
Operating Temperature, °F	110
Total Volume per Tank, ft^3	1,410*
Normal Water Volume per Tank, ft^3	1,040
Minimum Water Volume per Tank, ft^3	1,010
Material of Construction	Carbon Steel Lined with SS
Check Valves	
Number per Flood Line	2
Size, in.	14
Material	316 SS
Design Pressure, psig	2,500
Design Temperature, °F	
Valve nearest reactor	650
Valve nearest tank	300
Isolation Valves	
Number per Flood Line	1
Size, in.	14
Material	316 SS
Design Pressure, psig	2,500
Design Temperature, °F	300
Piping	
(See Table 6-3)	

*1410 ft^3 is original design; Total volume has been calculated to be 1421 ft^3.

Table 6-2

QUALITY CONTROL STANDARDS FOR ENGINEERED SAFEGUARDS SYSTEMS

Summary of Requirements for Core Flooding Tanks

CLASSIFICATION: ASME III, Class C, Paragraph N-2113 and the requirements of ASME VIII, Paragraph UW-2(a) (Lethal substances)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME III
2. Hydro test	ASME III
3. Radiograph	ASME VIII

Summary of Requirements for Decay Heat Removal Cooler

CLASSIFICATION: Shell ASME VIII, Tube ASME III, Class C (lethal)

Inspection Requirements	Acceptance Standard
1. Inspection of raw materials and review of material certificates	ASME II, III, VIII
2. Seal weld on tube-to-tube sheet	TEMA-R-7 and additional requirements
3. Liquid penetrant inspection on tube-to-tube sheet weld	ASME III, N-627 and additional requirements
4. Hydro test	ASME III, VIII, TEMA-R
5. Leak test and seal weld (air)	--

Summary of Requirements for Valves

Inspection Requirements	Acceptance Standard
Class I and II Valves per FSAR Section A.3	
1. Radiographic inspection of the body casting and discs	ANSI B31.7/NP + VC/ASME III '71
2. Inspection of material and review of material certificates	ANSI B31.7/NP + VC/ASME III '71
3. Liquid penetrant inspection of the valve body casting	ANSI B31.7/NP + VC/ASME III '71
4. Hydro test of valve assembly	ANSI B16.5/NP + VC/ASME III '71
5. Seat leakage test	NSS-SP-61

Table 6-2 (continued)

Inspection Requirements	Acceptance Standard
Class III Valves per FSAR Section A.3	
1. Inspection of material and review of material certificates	ANSI B31.7/NP + VC/ASME III '71
2. Hydro test of valve assembly	ANSI B16.5/NP + VC/ASME III '71
3. Seat leakage test	MSS-SP-61

In addition to these inspections listed above, all valve pressure part materials must meet the ASTM material specification.

Summary of Requirements for Engineered Safeguards Systems Pumps

Inspection Requirements	Acceptance Standard
1. Inspection of materials and review of materials certificates	ASTM (applicable specification)
2. Liquid-penetrant inspection of castings	ASME VIII
3. Performance test	ASME Power Test Code (PTC 8.2)

Additional Requirements

Low Pressure Injection Pumps

1. Hydrotest casing to at least 675 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This meets the hydrotest requirements of ASME VIII Paragraph UG-99 (≥ 1.5 x design pressure).

Reactor Building Spray Pumps

1. Hydrotest casing to at least 525 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This meets the hydrotest requirements of ASME VIII Paragraph UG-99 (≥ 1.5 x design pressure).

High Pressure Injection Pumps

1. Hydrotest pump casing to 4,500 psig. Test pressure is held for 30 minutes per inch of thickness with a minimum holding time of 30 minutes. This meets the hydrotest requirements of ASME VIII Paragraph UG-99 (≥ 1.5 x design pressure).

Table 6-2 (continued)

Summary of Requirements for Borated Water Storage Tank

Inspection Requirements	Acceptance Standard
1. Welds	AWWA-D-100

Summary of Requirements for Reactor Building Coolers

Inspection Requirements	Acceptance Standard
1. Welds	AWWA-D-100
2. Fan Performance	AMCA
3. Pressure Test	
a. Housing	2 psi external pressure with atmospheric pressure inside
b. Coils	Hydrostatic test to 300 psig
4. Leakage Test	Need not be "Leak Tight," but should be sufficiently tight to meet a test requiring housing to be placed under an air pressure of 4" water column and retain at least 2-1/2" water column after a period of 5 minutes.

Table 6-3

ENGINEERED SAFEGUARDS PIPING DESIGN CONDITIONS

	Temp, °F	Pressure, psig
1. High Pressure Injection System		
a. Normal HPI		
From BWST to Pump (P36)	110	75
From Pumps to RCS	115	3050
b. Piggy-Back Mode		
From Sump to Pumps (P34)	250	75
From Pump to Coolers	250	450
From the Decay Heat Coolers (E35) to the Pumps (P36)	212	450
From the Pumps to the RCS	250	3050
2. Low Pressure Injection System		
a. LPI with suction from BWST		
From the BWST to Pumps (P34)	110	75
From the Pumps to the RCS	110	450
b. LPI with suction from sump		
From sump to the Pump (P34)	280	75
From Pumps to RCS	280	450
3. Core Flooding System		
a. From the reactor vessel to the first valve in the core flooding line	650	2500
b. From the first check valve to the isolation valve in the core flooding line	300	2500
c. From the isolation valve in the core flooding line to the CFT	300	700
4. Reactor Building Spray System		
a. From the connection with the decay heat removal system at the isolation valve to the reactor building spray pumps	280	75
b. From the reactor building spray pumps	300	350
c. Ring header to the nozzles	300	180
d. From the BWST to the pumps	110	75

Table 6-4

SINGLE FAILURE ANALYSIS - EMERGENCY CORE COOLING SYSTEM

Component	Malfunction	Comments
A. High Pressure Injection System		
1. Suction valve for makeup pump from BWST	Fails to open	The parallel valve will supply the required flow to one pump string
2. High pressure injection valve	Fails to open	The alternate line will provide the flow required for protection
3. High pressure injection pump (operating)	Fails (stops)	Adequate injection is provided by the redundant pump
4. High pressure injection pump	Fails to start	Adequate injection is provided by the redundant pump
5. Controlled bleedoff line isolation valve	Fails to close on ES signal	Depending on which valve is assumed to fail, either the isolation valves on each controlled bleedoff line or the isolation valve on the controlled bleedoff return header will close eliminating this flow path
6. Letdown line isolation valve	Fails to close on ES signal	Depending on which valve is assumed to fail, either the letdown cooler outlet valves will both close or the isolation valve outside containment will close eliminating the flow path.
7. HPI pump recirculation line valve	Fails to close on ES signal	The other valve will close the flow path
8. HPI stopcheck valve inside containment	Fails to close on backflow from RCS	Backflow will be prevented by a redundant check valve
9. BWST purification Recirculation line	Fails to close on ES signal	The other valve will close
B. Low Pressure Injection System		
(Injection From Borated Water Storage Tank)		
1. Low pressure injection pump .	Fails to start	Adequate injection is provided is provided by redundant pump
2. Emergency injection valve	Fails to open	Other line admits necessary flow Pump protection is provided by the minimum recirculation line
3. Valve in suction line from BWST	Fails to open	Other line admits necessary flow

Table 6-4 (continued)

Component	Malfunction	Comments
(Recirculation from Reactor Building Sump)		
1. Valve in suction line from sump	Fails to open	Other line admits necessary flow
2. Valve in suction line from BWST	Fails to close after initiating recirculation	Check valve prevents flow into BWST
3. Low pressure injection pump	Loss of pump	Reactor core protection will be maintained by alternate pump and LPI string
4. Service Water System common return valve to circulating water discharge flume	Closes during normal operation	Results in the loss of cooling by both LPI strings and to all ECCS components by providing no discharge for service water. The breaker for this valve is to be locked open during normal operation
C. Core Flooding System		
		Isolation valve is open and breaker open in accordance with the Technical Specifications
1. Isolation valve in discharge line	Closes during normal operation	If the valve cannot be manually opened, the reactor must be shut down or operations limited as specified in Technical Specifications
2. Tank relief valve	Opens during normal operation.	Loss of nitrogen pressure and consequent loss of ability of tank to perform. Reactor must be shut down or operations adjusted to Technical Specification limits until the relief valve is repaired.
3. Check valves in discharge line	Excessive leak detected during normal reactor operation	It is extremely unlikely that both check valves would permit excessive leakage. Leakage would be indicated by CFT pressure and level changes. If leak becomes progressively worse or is unacceptably high, reactor must be shut down while the check valves are repaired

Table 6-4 (continued)

Component	Malfunction	Comments
4. Core Flood Tank vent valves.	Opens before or during CFT discharge	CFT discharge flow rate could be less than that used in ECCS analysis. Due to reducing orifice in discharge line, pressure would bleed off at a slow rate. Alarm at 585 psig would be received approximately 15 minutes following valve failure. Approximately 30 total minutes would elapse before tank pressure would reach Technical Specification limit of 575 psig, allowing sufficient time for corrective action.

Table 6-5

EMERGENCY INJECTION SYSTEMS COMPONENT TESTING

System/Component	Test
High Pressure Injection System	A test signal is applied at a frequency specified in the Surveillance Frequency Control Program (SFCP) to actuate system for emergency core cooling operation. Appropriate pump breakers are opened or closed. Valves are monitored for travel completion. Control board indications verify component response to actuation signal.
Low Pressure Injection System	A test signal is applied at a frequency specified in the SFCP to actuate system for emergency core cooling operation. Engineered safeguard function of service water system is verified to assure adequate cooling water supply for decay heat removal cooler operability. Appropriate pump breakers are opened or closed. Valves are monitored for travel completion. Control board indications verify component response to actuation signal.
Core Flooding System	A test is conducted at a frequency specified in the SFCP to demonstrate operability of core flooding system. Control board indications of core flood tank level verify opening of all check valves.
High Pressure Injection and Low Pressure Injection Pumps	Approximately quarterly, pumps are started, operated for 15 minutes, and observed for discharge pressure and flow within the acceptable range per ASME OM Code.
Power Operated Valves	At intervals not to exceed three months, appropriate signals actuate power operated valves. Valves must indicate full travel within allowable time.

Table 6-6

SINGLE FAILURE ANALYSIS - REACTOR BUILDING SPRAY SYSTEM

Component	Malfunction	Comments
1. Reactor building spray pump	Fails to start	The remaining string will provide heat removal capability at a reduced rate. In combination with the Reactor Building Cooling System, heat removal capability In excess of requirements will be provided. Iodine removal is adequate with one string operating
2. Building isolation valve	Fails to open	(Same as above)
3. Check valve in suction or discharge line	Fails to open	(Same as above)

Table 6-7

REACTOR BUILDING SPRAY SYSTEM TANK DATA

	Borated Water Storage Tank
Design Temperature, °F	150
Design Pressure, ft. H_2O	10
Code	AWWA-D-100
Chemical Concentration H_3BO_3,ppm boron	2470 ± 200
Capacity, gal.	405,090
Material	Carbon Steel with Placite 7155 Liner

Table 6-8

REACTOR BUILDING SPRAY SYSTEM COMPONENT TESTING*

Component	Test
Reactor Building Spray Pump	Each pump is periodically tested to demonstrate performance.
Reactor Building Spray	With the pumps shut down and the borated injection valves water storage tank outlet valves closed, each of these valves is opened and closed by operator action to verify operability.
Reactor Building Spray Nozzles	With the reactor building spray inlet closed, low pressure air or smoke is blown through the test connections in each spray header. Visual observation of smoke or the effect of air on balloons or streamers, or air flow verification using thermographic equipment, confirms that nozzle flow paths are open.

*Specific testing requirements are provided in the Technical Specifications.

Table 6-9

REACTOR BUILDING COOLING UNIT PERFORMANCE AND EQUIPMENT DATA
(capacities are on a per unit basis)

	Emergency	Normal	(Note 3) Supplemental Normal
Number Installed	4	4	1
Number Required	Note 1	3	1
Type Coil	Finned Tube	Finned Tube	Finned Tube
Heat Removal Capacity, BTU/hr	60×10^6 (Note 4)	1.17×10^6 (Note 5)	2.5×10^6
Fan Capacity, cfm	30,000	30,000	60,000
Containment Atmosphere Inlet Conditions:			
Temperature, °F	286	110	120
Steam Partial Pressure, psia	54	---	---
Air Partial Pressure, psia	20	---	---
Total Pressure, psig	59	Atmospheric	Atmospheric
Cooling Water Flow, gpm	1,200 (Note 2)	130	225
Cooling Water Inlet Temperature, °F	95	50	48
Cooling Water Outlet Temperature, °F	192	68	70

Note 1: With one train of the Reactor Building Spray (RBS) System inoperable, one reactor building emergency cooling train is required to be operable.

Note 2: 1200 gpm is the design service water flow rate assumed by the manufacturer for each cooling unit. This flow can be provided to a single unit if the other unit in that train is isolated. Heat removal capabilities of two units in one train sharing available flow will exceed the capability of a single unit at design flow.

Note 3: VSF1E was added during 1R9 for supplemental cooling during normal operation.

Note 4: A peak value of 53.7×10^6 Btu/hr is assumed in the DBA Analysis. This value represents a 5% reduction from the peak predicted design value for the air coolers. Variations in any of the design parameters presented in this table are acceptable provided the heat removal capacity assumed in the DBA analysis is maintained.

Note 5: The chill water coils in VSF-1A, 1B, and 1C were replaced with longer coils having a heat capacity of 1.41×10^6 BTU/hr. The heat capacity of cooler VSF-1D remains as shown in the table.

Table 6-10

SINGLE FAILURE ANALYSIS - REACTOR BUILDING COOLING SYSTEM

Component	Malfunction	Comments
1. Cooling Fan	Fails to start	Required cooling can be provided by the unaffected train of cooling units and one spray train or by two spray trains
2. Service Water Coil	Ruptures	(Same as above)
3. Bypass Damper	Fails to open	(Same as above)
4. Motor Operated Backdraft Damper	Fails to open	(Same as above)
5. R.B. Isolation Valves on Service Water Line for its Respective Two Cooling Units	Fails to open.	(Same as above)

Table 6-11

LEAKAGE QUANTITIES TO AUXILIARY BUILDING

SYSTEM	COMPONENT TYPES	TOTAL LEAKAGE cc/hr
Low Pressure Injection System	Pump Seals, Flanges, Process Valves, Instrumentation Valves, Boundary Valves	1516
Reactor Building Spray System	Pump Seals, Flanges, Process Valves, Instrumentation Valves, Boundary Valves	773
High Pressure Injection System*	Pump Seals, Flanges, Process Valves, Instrumentation Valves, Boundary Valves	4660

*Operates in recirculation only for certain small break LOCAs. 100,000 cc/hr used in the analysis in SAR Section 14.2.2.5.7.

Table 6-12

ENGINEERED SAFEGUARDS EMERGENCY ACTUATION

	Reactor Coolant System Pressure ~psig	Reactor Building Pressure ~psig
High Pressure Injection	1585*	4*
Low Pressure Injection	1585*	4*
Core Flooding System	600*	--
Reactor Building Spray System	--	30*
Reactor Building Cooling System	--	4*
Penetration Room Ventilation System	--	4*
Combustible Gas Control System (Manual Actuation)	--	--

*Values are nominal. Refer to Technical Specification for current values.

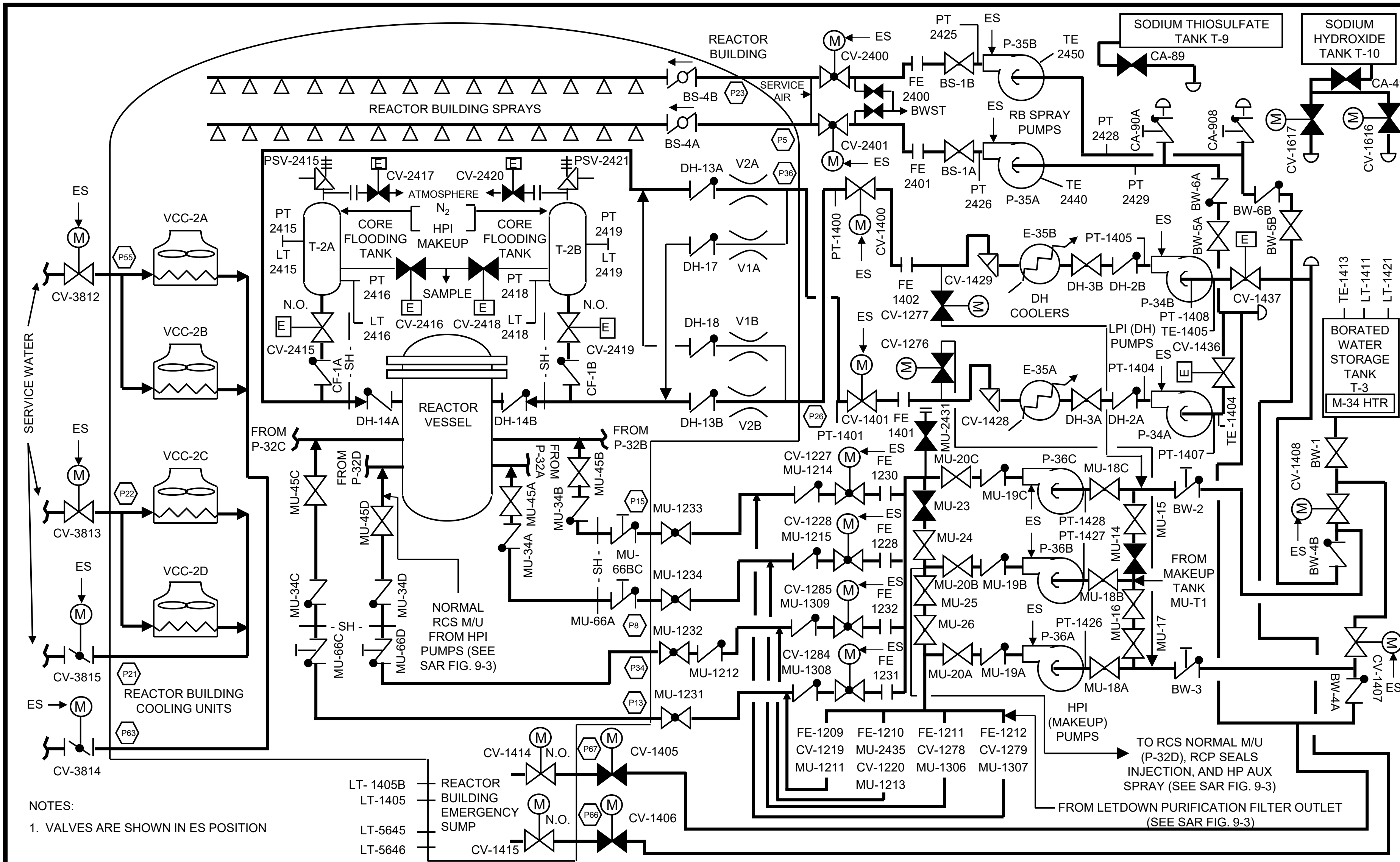
Table 6-13

SINGLE FAILURE ANALYSIS OF PENETRATION ROOM VENTILATION SYSTEM

Incident#	Component	Postulated Failure	Automatic Failure Indication	Other Automatic Action	Required Operator Action	Comments
1	Suction duct-work-lead system	Breaks	Loss of penetration room vacuum – alarm in control room sounds on loss of vacuum	Check valve in penetration room lead system suction duct-work closes	Stop lead fan. Close lead filter discharge valve. Open standby filter discharge valve. Start standby fan.	Lead and standby suction ducts are physically separated in penetration rooms and simultaneous failure of both systems is not postulated.
2	Pneumatically operated penetration room duct isolation valves	Loss of Air pressure	Indicating lights in control room	Valves fail closed	Check if ESAS had occurred. If so, this is normal valve action & lead system is automatically activated. If ESAS has not occurred, check source of loss of air pressure.	During an ESAS, air pressure is lost and duct valves automatically close sealing boundaries of penetration room.
3	Lead system filter outlet valve (motor-operated)	Loss of electric power	Indicating lights in control room	Valve fails closed	None	If valve fails closed on initial start signal, standby system will automatically start after 25 sec.
4	DELETED					

Table 6-13 (continued)

Incident#	Component	Postulated Failure	Automatic Failure Indication	Other Automatic Action	Required Operator Action	Comments
5	Carbon Filter	Static press. drop exceeds 1.20 WC	High pressure drop alarm in control room	None	Stop lead fan. Close lead filter discharge valve. Open standby filter discharge valve. Start standby fan.	
6	HEPA filter	Static press. drop exceeds 2.00 WC	"	None	"	
7	Prefilter	Static press. drop exceeds 0.80 WC	"	None	"	
8	High radiation on filter discharge	Filter train ineffective in removing radioactive elements	High radiation alarm in control room	None	"	
9	Lead fan	Fails	Flow trouble alarm in control room	In "hand" none. In "auto" – standby system will start	In "hand" – same as above. In "auto" - none	
10	Lead fan	Fails on receipt of ES signal	Flow trouble alarm in Control room	If proper flow is not achieved within 20 seconds, the lead system is automatically stopped and the standby system is started 5 seconds later.	None	



NOTES:

1. VALVES ARE SHOWN IN ES POSITION

SIMPLIFIED SCHEMATIC DIAGRAM OF ENGINEERED SAFEGUARDS SYSTEM FOR CORE AND BUILDING PROTECTION

SAR FIGURE NO. 6-1

ARKANSAS NUCLEAR ONE

UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 30

BASED ON DRAWING NO	SHEET	REV.

DELETED

EMERGENCY CORE COOLING SYSTEM			SAR FIGURE NO. 6-2
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

DELETED

REACTOR BUILDING SPRAY AND CORE FLOODING SYSTEMS	SAR FIGURE NO. 6-3
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

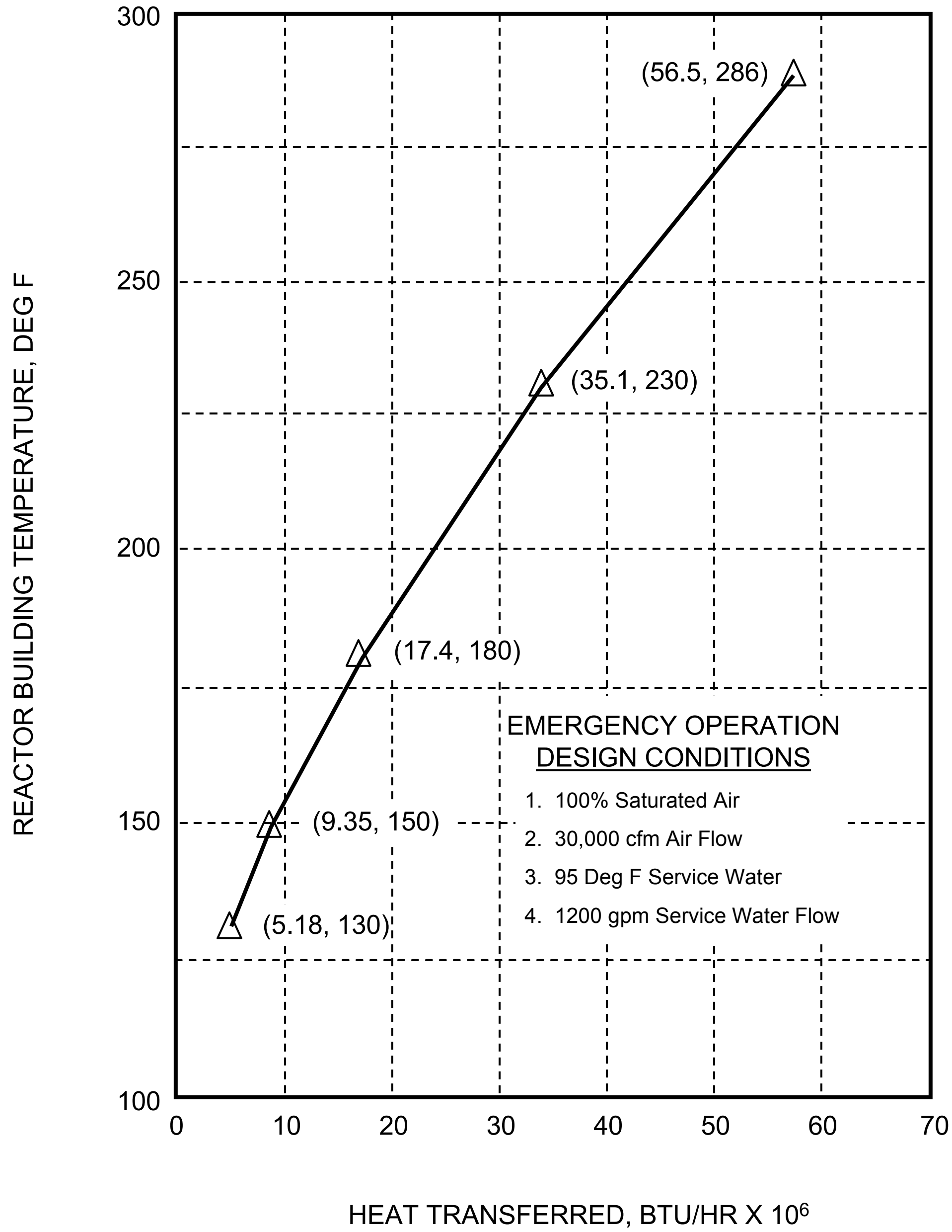
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

DELETED

REACTOR BUILDING COOLING SYSTEM		SAR FIGURE NO. 6-4
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

AMENDMENT 20		
BASED ON DRAWING NO	SHEET	REV.



REACTOR BUILDING COOLER CHARACTERISTICS

SAR FIGURE NO. 6-5

AMENDMENT 23

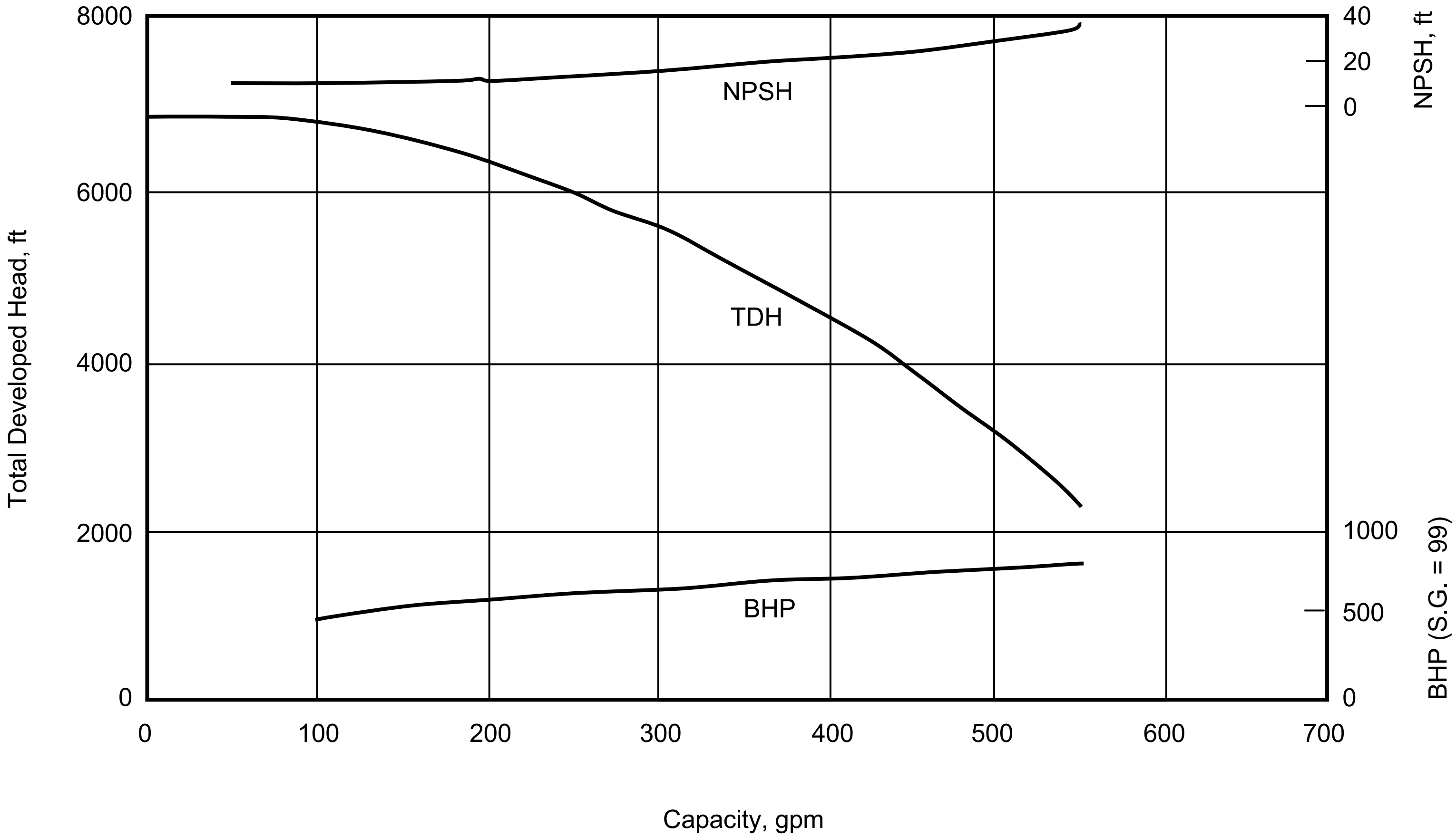
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING COOLER CHARACTERISTICS ASSUMED IN THE COLD LEG BREAK CONTAINMENT ANALYSIS	BASED ON DRAWING NO	SHEET: 1 REV.:

DELETED

SAR FIGURE NO. 6-5A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING COOLER CHARACTERISTICS	BASED ON DRAWING NO	SHEET 1 REV.



HIGH PRESSURE INJECTION PUMP CHARACTERISTICS	SAR FIGURE NO. 6-6
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

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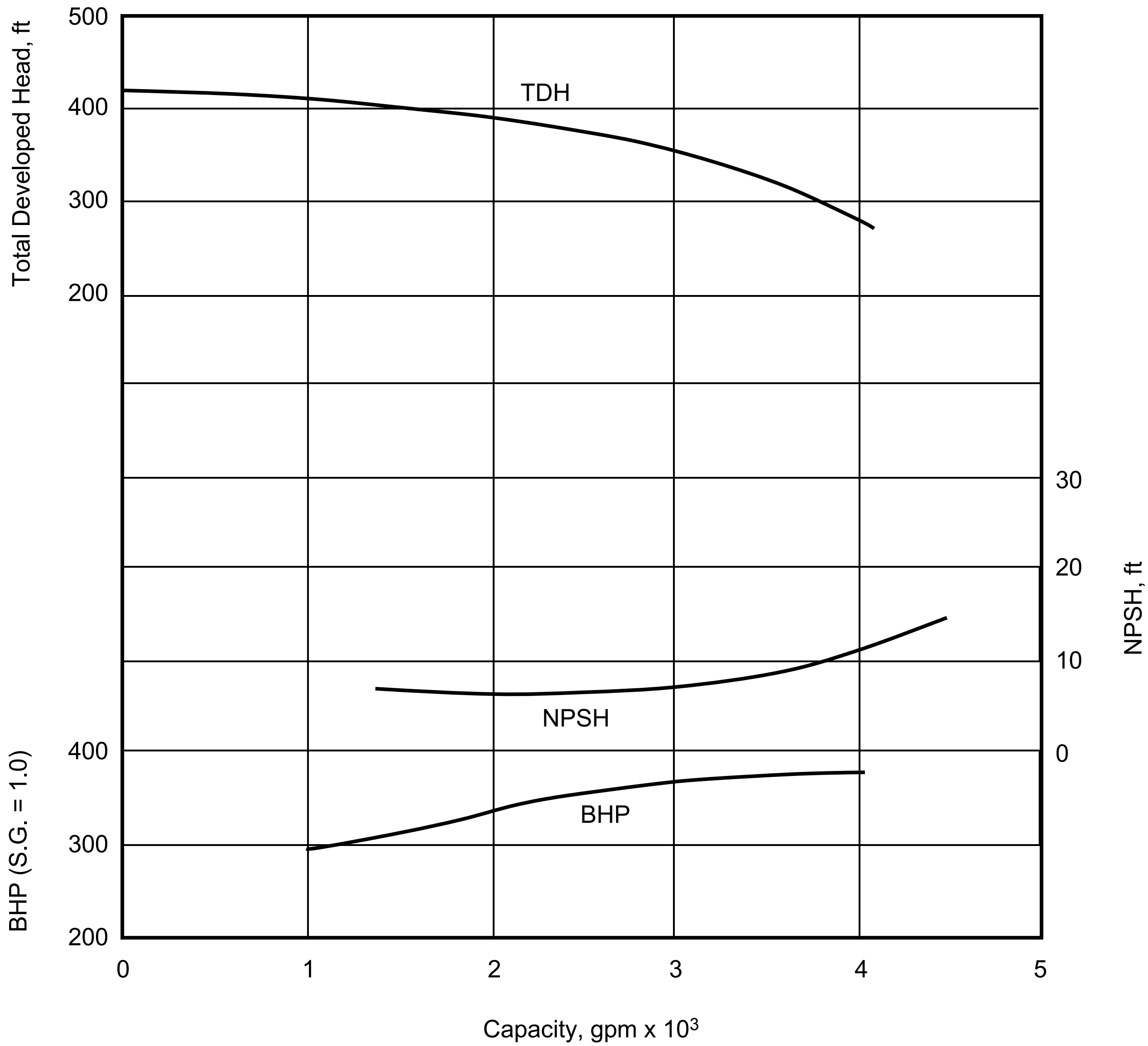
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 6-7

AMENDMENT 23

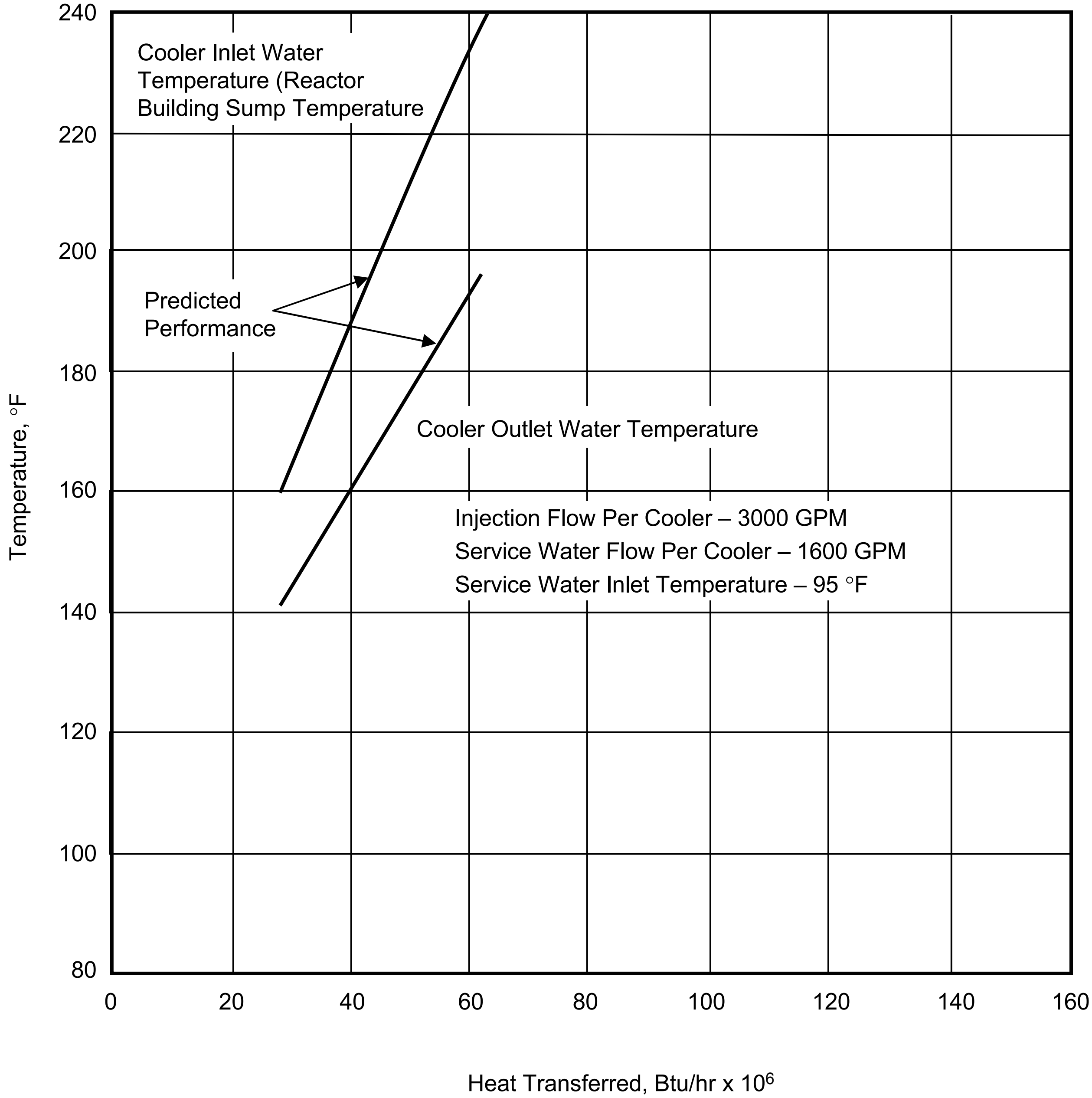
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

LOW PRESSURE INJECTION PUMP CHARACTERISTICS

BASED ON DRAWING NO	SHEET	REV.
	1	

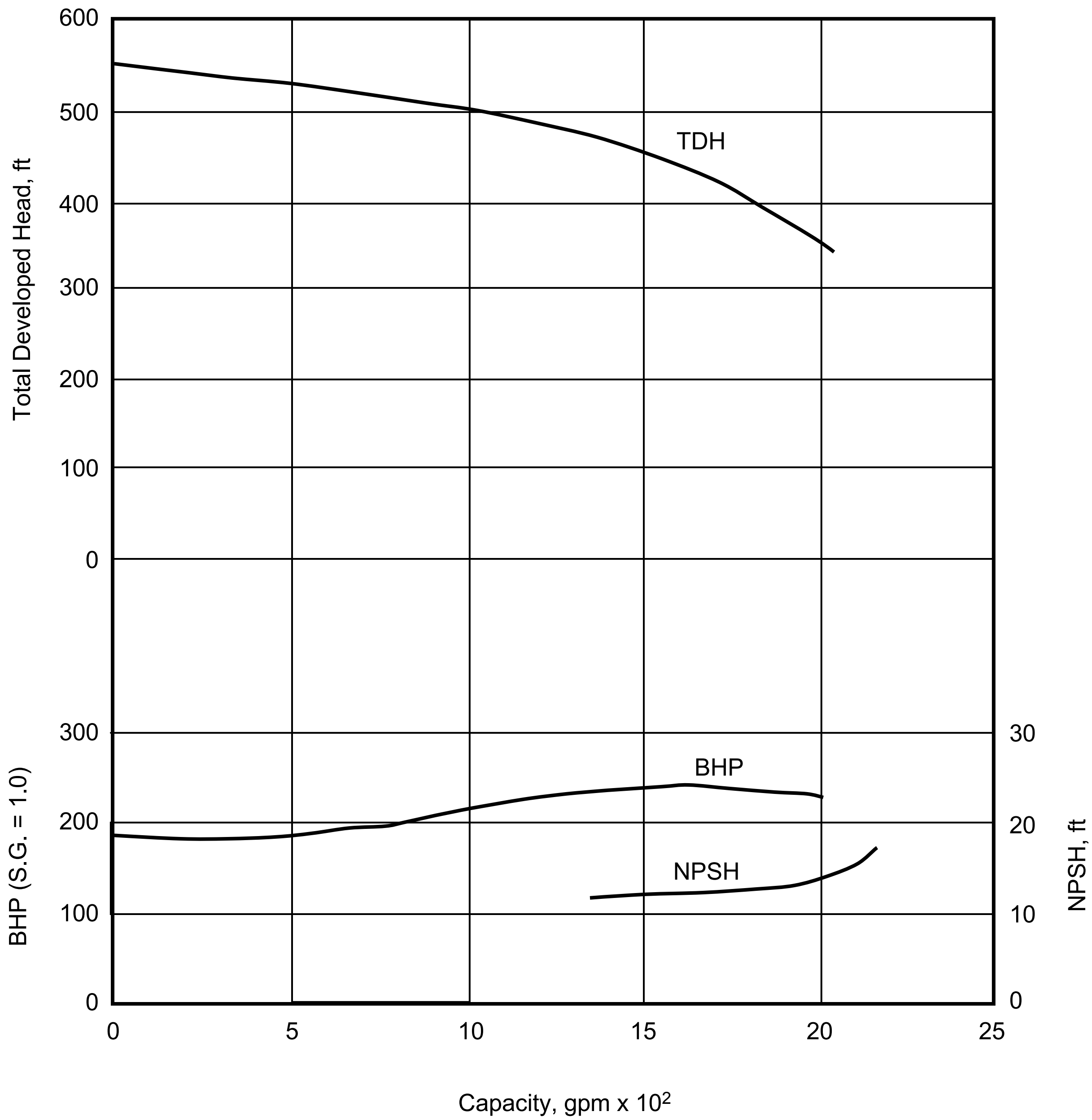


SAR FIGURE NO. 6-8

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

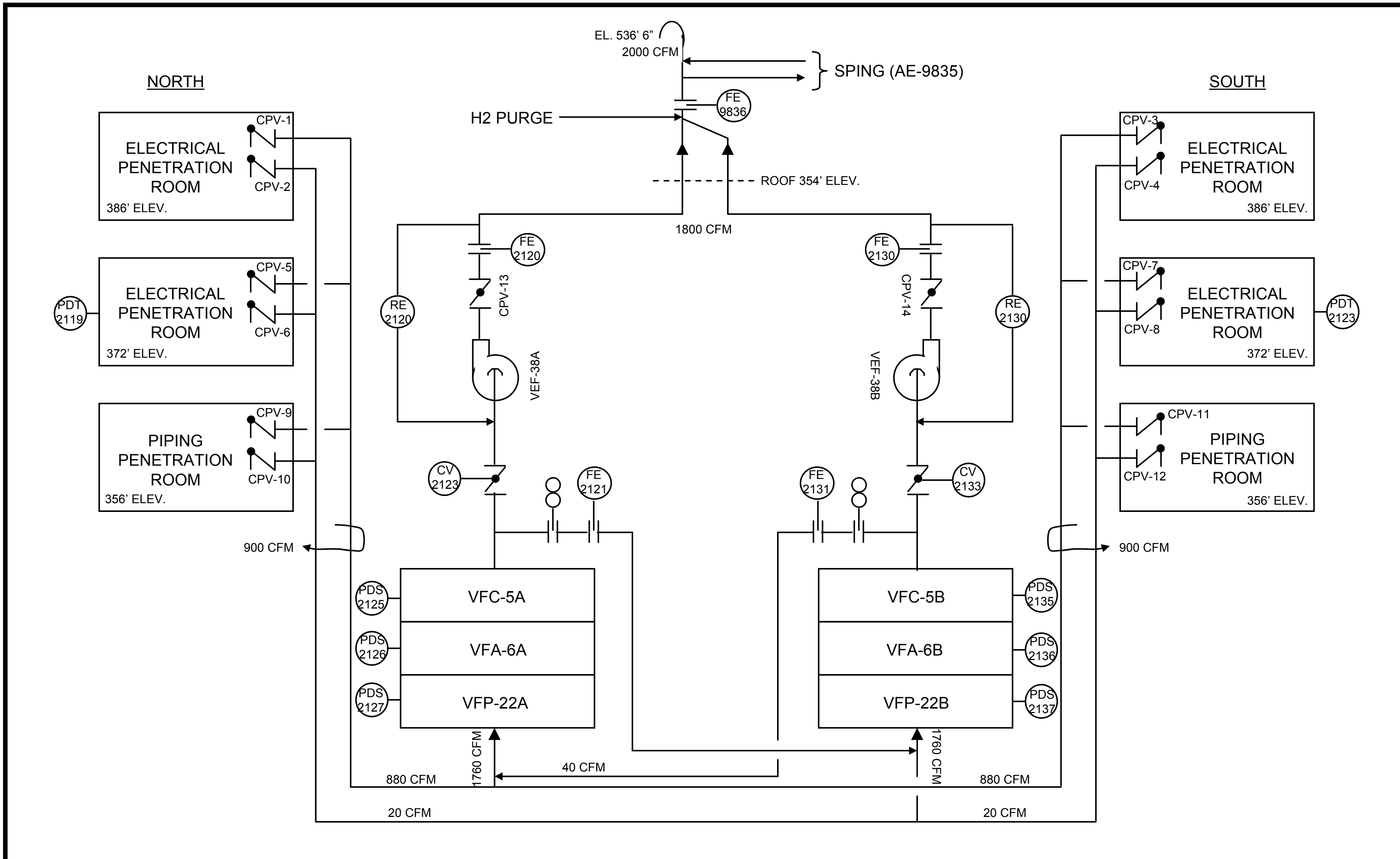
DECAY HEAT REMOVAL COOLER CHARACTERISTICS	BASED ON DRAWING NO	SHEET 1	REV.



SAR FIGURE NO. 6-9

AMENDMENT 23

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
REACTOR BUILDING SPRAY PUMP CHARACTERISTICS	BASED ON DRAWING NO	SHEET 1	REV.



VENTILATION SYSTEM AIR FLOW CONTAINMENT PENETRATION ROOM	SAR FIGURE NO. 6-10
	AMENDMENT 20

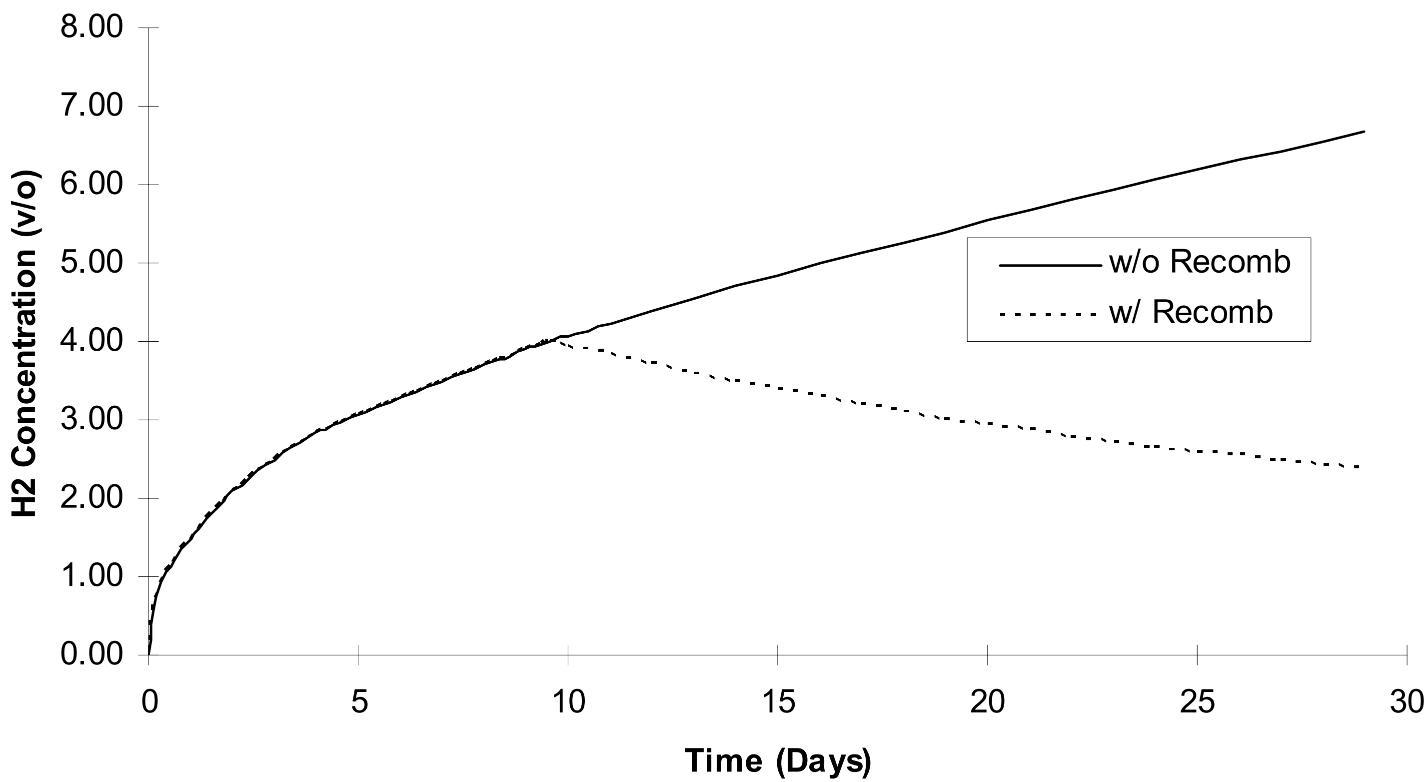
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

Figure 1 H2 Concentration With and Without 1 Recombiner



POST LOCA HYDROGEN			SAR FIGURE NO. 6-11		
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			BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 6-12

AMENDMENT 21

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.
	1	

DELETED

(REFER TO FIGURE 5-7)

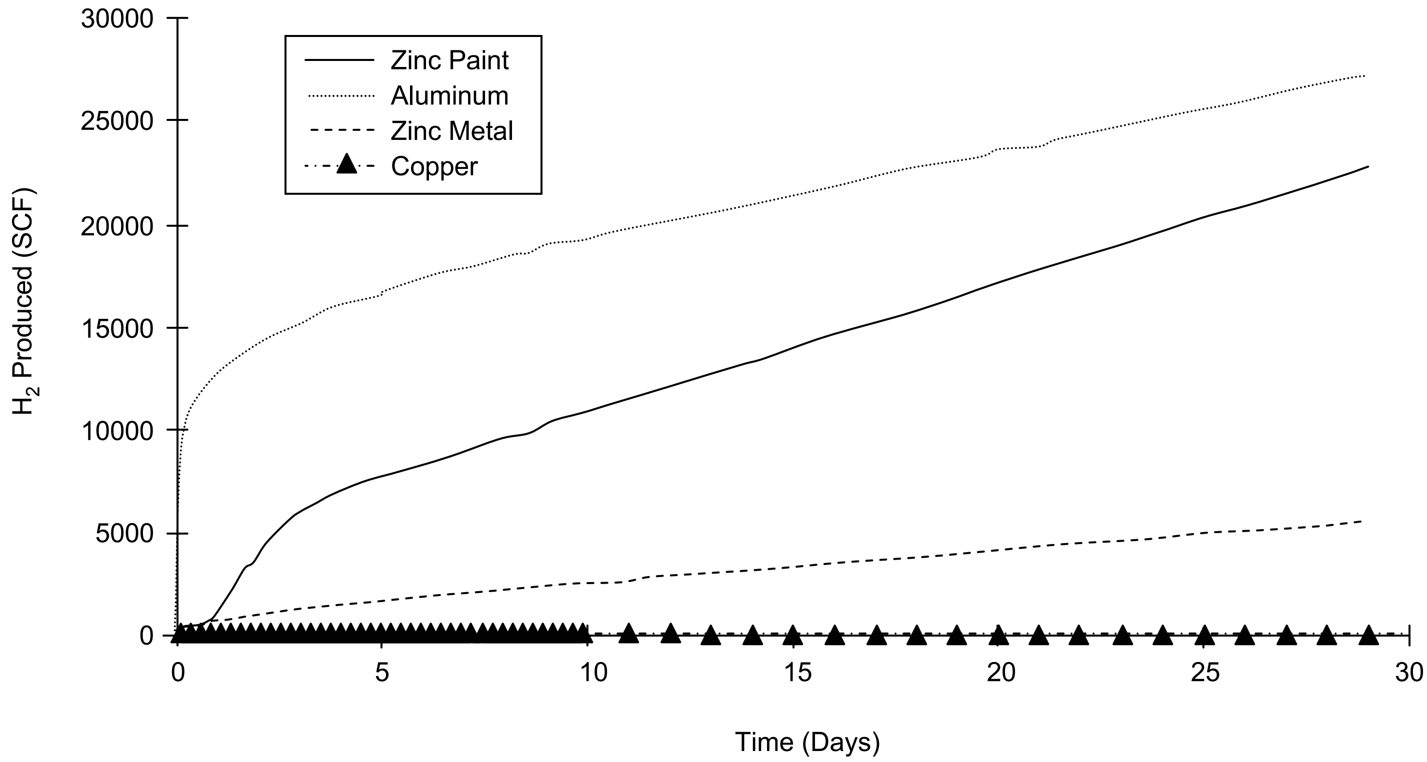
SAR FIGURE NO. 6-13

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

REACTOR BUILDING VENTILATION SYSTEM	BASED ON DRAWING NO	SHEET	REV.
		1	

Figure 3 H_2 Production From Metal Corrosion

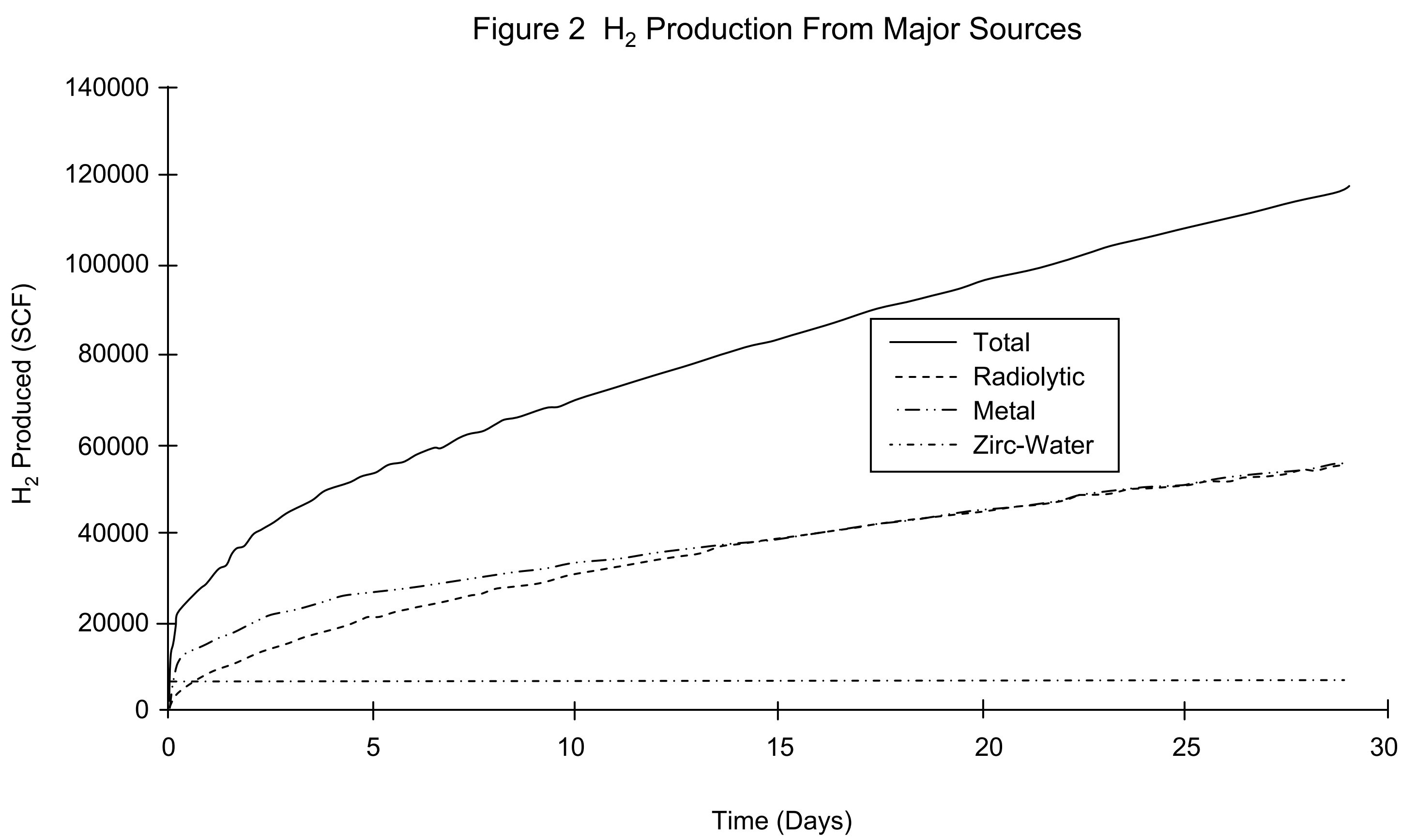


POST LOCA H2 FROM METAL CORROSION			
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	

SAR FIGURE NO. 6-14

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



POST LOCA HYDROGEN SOURCES			SAR FIGURE NO. 6-15		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
			BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 6-16

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
HYDROGEN EVOLUTION FROM CORROSION	BASED ON DRAWING NO	SHEET 1 REV.

DELETED

SAR FIGURE NO. 6-17

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ENERGY RELEASE IN THE CORE	BASED ON DRAWING NO	SHEET 1 REV.

DELETED

SAR FIGURE NO. 6-18

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ENERGY RELEASE IN THE SUMP	BASED ON DRAWING NO	SHEET 1 / REV.

ARKANSAS NUCLEAR ONE
Unit 1

CHAPTER 7

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7 INSTRUMENTATION AND CONTROL

Instrumentation and control systems include the Reactor Protection System (RPS), the Engineered Safeguards Actuation System (ESAS), Emergency Feedwater Initiation and Control (EFIC) System, the Rod Drive Control System (RDCS), the Integrated Control System (ICS), the Diverse Reactor Overpressure Prevention System (DROPS), the Nuclear Instrumentation (NI) System, the Non-Nuclear Instrumentation (NNI) System, and the Incore Monitoring System. Major portions of the RPS (including the NI, NNI, and trip actuation equipment), RDCS, ESAS, and EFIC are supplied by Babcock and Wilcox (B&W).

7.1 PROTECTION SYSTEMS

The protection and engineered safeguards systems consisting of the RPS, the ESAS, and EFIC perform important control and safety functions. These protection systems consist of the sensing instrumentation, associated logic circuitry, and the final actuating devices, such as circuit breakers and pump or valve motor contactors and solenoids.

7.1.1 DESIGN BASES

The protection systems are designed to sense plant parameters and actuate emergency actions in the event of abnormal plant parameter values. They meet the requirements of the proposed IEEE "Criteria for Nuclear Power Plant Protection Systems," dated August 1968 (IEEE 279) with the exception of the EFIC, which meets IEEE 279-1971.

7.1.1.1 Single Failure

The protection systems meet the single failure criterion of IEEE 279-1968 to the extent that:

A. No single failure will prevent a protection system from fulfilling its protective functions when action is required.

B. No single failure will initiate unnecessary protection system action where implementation does not conflict with the criterion above.

7.1.1.2 Redundancy

All protection system functions are implemented by redundant sensors, measuring channels, logic, and actuation devices. These elements combine to form the protection channels as defined in Section 7.3.

7.1.1.3 Independence

Redundant protection channels are electrically independent and packaged to provide physical separation.

7.1.1.4 Separation

Protection channels are physically separated and are electrically isolated from Non-Class 1E instrumentation.

7.1.1.5 Manual Trip

Manual trip switches, independent of the automatic trip instrumentation, are provided.

7.1.1.6 Testing

Manual testing facilities are built into the protection systems to provide for:

A. Pre-critical testing to give assurance that a protection system can fulfill its required protective functions.

B. On-line testing to prove operability and to demonstrate reliability.

7.1.1.7 Environment

Protection system equipment within the reactor building, but outside the primary shield, is designed for continuous operation in ambient conditions as shown in Table 7-1. This equipment can also withstand the superimposed accident dose given in Table 7-1 with the associated environment for the length of time the equipment is required to operate following a LOCA or steam line break.

Safety related equipment which are required to remain functional during and following an accident have been demonstrated to be qualified to withstand the environment resulting from these accidents. The program is described in Engineering Standard NES-01, "EQ Program."

7.1.1.8 Seismic Qualification

The protection system, engineered safeguards circuits, and emergency power system are designed to maintain their functional capabilities during and after a Maximum Hypothetical Earthquake (MHE). The seismic testing program meets the requirements of IEEE 344-1971, "Seismic Qualification for Class I Electric Equipment for Nuclear Power Generating Stations." The seismic tests are designed to demonstrate that (1) the equipment being tested is structurally capable of enduring a MHE and (2) the individual modules contained in the cabinets and the individual circuits and components can function during and after a MHE.

To ensure compliance with the seismic design criteria, detailed seismic qualification requirements are imposed upon the equipment supplier by means of the system procurement specification.

Results of seismic testing and analysis for the RPS and ESAS are contained in B&W Topical Report BAW-10003 Revision 2, "Qualification Testing of Protection System Instrumentation" and Vitro Corporation Report #3801-1310 summarizes the results of seismic testing and analysis for the EFIC. The results in these reports indicate that these protection systems in the configuration used in Arkansas Nuclear One - Unit 1 will perform their intended functions during and after a MHE.

Results of seismic testing or analyses for the emergency power system and those portions of the protection systems not covered by the above reports are contained in various seismic documentation requested in the specification for each order. The results indicate that the protection system circuits and the emergency power system components (diesel generator, batteries, control and distribution panels, battery chargers, and inverters) will perform their intended function during and after a MHE.

7.1.2 REACTOR PROTECTION SYSTEM

The RPS monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against Reactor Coolant System(RCS) damage caused by high system pressure by limiting energy input to the system through reactor trip action.

The RPS is designed to provide the necessary protection to insure that certain safety limits and accident analysis acceptance criteria are not violated. Three safety limits are of importance to the RPS. These include the high Reactor Coolant System pressure, Departure from Nucleate Boiling Ratio (DNBR), and linear heat rate (kW/ft).

7.1.2.1 Design Basis

The RPS includes all design basis features of Section 7.1.1 with the following additions.

7.1.2.1.1 Loss of Power

A loss of power to an RPS channel will cause the affected protection channel to trip.

7.1.2.1.2 Equipment Removal

The RPS initiates a protection channel trip whenever a module or subassembly is removed from the equipment cabinet. Provisions are made in each protection channel to supply a bypass signal (Section 7.1.2.3.8) which leaves the channel in a non-tripped condition for testing and maintenance. It is not possible to bypass more than one channel. A channel bypass is indicated by a plant annunciator in the control room.

7.1.2.1.3 Diversity and Independence from ATWS Systems

The RPS must be diverse and independent of all ATWS systems in accordance with 10 CFR 50.62. The Diverse Reactor Overpressure Prevention System (DROPS) is the ANO-1 ATWS system and consists of the Diverse Scram System (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC). Refer to Section 7.2.4 for the DROPS.

7.1.2.2 System Design

7.1.2.2.1 System Logic

The RPS consists of four identical protection channels, each terminating in a trip relay within a reactor trip (RT) module. In the normal, untripped state, each protection channel passes current to the terminating trip relay and holds it energized as long as all inputs are in the normal energized (untripped) state. Should any one or more inputs become de-energized (tripped) the terminating relay in that protective channel de-energizes (trips).

Each protection channel trip relay has four logic controlling contacts, each controlling a logic relay in one RT module. Therefore, each RT module has four logic relays controlled by the four protection channels. The four logic relays combine to form a 2-out-of-4 coincidence network in each RT module. The coincidence logics in all RT modules trip whenever any two of the four protection channels trip.

The RT modules are given the same designation as the protection channel whose trip relay they contain and in whose cabinet they are physically located. Thus, the protection channel "A" RT module is located in protection channel "A" cabinet, etc. The coincidence logic in each RT module controls one or more breakers in the Control Rod Drive power system.

The coincidence logic contained in the RPS channel "A" RT module controls AC breaker "A" in the Control Rod Drive System, channel "B" RT module controls AC breaker "B", channel "C" RT module controls DC breakers "C" and contactor "E", and channel "D" RT module controls DC breakers "D" and contactor "F". Breakers "A" and "B" control all the 3-phase primary power to the rod drives; breakers "C" and "D" control the DC power to rod groups 1 through 4; and contactors "E" & "F" control the gating power to rod groups 5 through 8.

The Control Rod Drive circuit breaker and contactor combinations that initiate a reactor trip can best be stated in logic notation as "AB" + "ADF" + "BCE" + "CDEF". This is a 1-out-of-2 logic used twice and is referred to as a 1-out-of-2 x 2 logic. When any 2-out-of-4 protection channels trip, all RT module logics trip, commanding all Control Rod Drive breakers to trip.

The undervoltage coils of the Control Rod Drive breakers receive their power from the protection channel associated with each breaker. A redundant shunt trip coil will also receive a breaker trip input from its associated RPS channel. The manual reactor trip switch is interposed in series between each RT module logic and the assigned breakers undervoltage coil.

7.1.2.2.2 Summary of Protective Functions

The four RPS protection channels are identical in their functions. The channels combine in the system logic to trip the reactor automatically to protect the reactor core and RCS. The conditions that initiate a reactor trip are tabulated in Table 7-2.

7.1.2.2.3 Description of Protection Channel Functions

The functions of the RPS described below apply to each protection channel.

A. Overpower Trip

The nuclear instrumentation provides a linear neutron flux signal in the power range as an indication of reactor power to a protection system bistable trip module.

When the neutron flux signal exceeds the trip point of the bistable, the bistable trips, de-energizing the associated protection channel trip relay.

B. Power Trip Based on Imbalance and Flow Functions

Neutron flux and reactor coolant flow are continuously monitored by a power-imbalance-flow bistable. The bistable trips, de-energizing the protection channel trip relay, whenever the power (ø) exceeds the variable limit set by the summation of the imbalance function f(Δø) and the flow function f(F). That is, a channel trip occurs when $ø > f(Δø) + f(F)$.

The power signal is the summation of the individual linear neutron flux signals from the top and bottom neutron detectors. The imbalance signal is the difference of the individual linear neutron flux signals from the top and bottom neutron detectors. The flow signal is the total reactor coolant flow rate derived from the flow tube dP cells.

An example of a power trip based on imbalance and flow will be described. This description is representative only, with the actual setpoint for the trip defined in the current cycle Core Operating Limits Report. The imbalance function f(Δø) in percent is shown on Figure 7-2 and the breakpoint values B_1, B_2, B_3, and B_4 are given in Table 7-3. The flow function f(F) shown on Figure 7-3 is the percentage reactor coolant flow rate multiplied by a power-to-flow ratio. Power trip based on imbalance and flow functions is shown on Figure 7-4 where the solid line represents the imbalance function plus the flow function operating envelope with a 4-pump 100 percent flow. The dashed line of Figure 7-4 represents the imbalance function and flow function operating envelope with a reduced flow rate.

All flow dP cells for a single loop are connected to common 1-inch low pressure and high pressure lines from the flow tube in that loop. Severance of the low pressure line will result in maximum indicated flow for the loop. Severance of the high pressure line will result in zero indicated flow for the loop and possibly a power-imbalance-flow reactor trip.

C. Reactor Outlet Temperature Trip

The reactor outlet temperature is measured by resistance elements. The bridge for each resistance element is considered a part of, and is located within, its associated protection system channel.

The reactor outlet temperature signal from the temperature bridge passes through a signal converter and then is applied to a bistable trip module. When the temperature exceeds the trip point of the bistable, the bistable trips, de-energizing the channel trip relay.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that, in fact, the temperature channel is fully operational approximately 10 percent above the calibrated range.

The instrument design was determined to be acceptable since the channel will trip at a value of reactor coolant outlet temperature no higher than 620 °F even in the worst case and since the channel is fully operational approximately 10 percent above the calibrated range and exhibits no hysteresis or foldover characteristics.

D. Pressure-Temperature Trip

Figure 7-5 shows an example of the operating reactor coolant pressure-temperature boundaries formed by the combined reactor high temperature, high pressure, low pressure, and the pressure-temperature comparator trip settings. The actual trip settings are provided in the Technical Specifications. The pressure-temperature comparator bistable trips whenever the specified reactor pressure-outlet temperature relationship is exceeded. The comparator forms the boundary line A-B in Figure 7-5.

E. Reactor Pressure Trip

The reactor coolant pressure signal from the pressure transmitter is received by a buffer amplifier module in the associated protection channel. This module acts as a signal conditioner and isolation unit. Pressure signals go to a high pressure bistable trip module and a low pressure trip module. When the pressure exceeds the trip point of the high pressure bistable, the bistable trips, de-energizing the protection channel trip relay.

The low pressure bistable trips when the pressure falls below the trip point, tripping the protection channel trip relay.

F. Power/Reactor Coolant Pumps Trip

The Reactor Coolant Pump power breakers and the motor underpower relays are monitored to determine that the pumps are running. No RT setpoint adjustments must be made when reducing the number of operating pumps. Loss of a single pump initiates four independent signals, one to each protection channel. This information is received by a contact monitor, which counts the number of RCPs in operation and identifies the coolant loop in which the pumps are operating. The contact monitor output controls the trip point of a power/pump comparator which initiates a protection channel trip for the conditions in Table 7-2.

G. Reactor Building Pressure Trip

Each of the four protection channels receives reactor building pressure information from an independent pressure switch. The pressure sensors are located inside the reactor building and do not need instrument sensing lines penetrating the building. A contact buffer in each protection channel continuously monitors the state of the associated pressure switch. When the state of the pressure switch changes to that corresponding to a reactor building pressure exceeding the trip point specified in Table 7-2, the contact buffer de-energizes the protection channel's trip relay.

H. Anticipatory Reactor Trip System (ARTS)

The Anticipatory Reactor Trip System (ARTS) provides for a reactor trip on a turbine trip or a loss of both main feedwater pumps.

Each of the four protection channels receives input based on the status of the turbine (tripped/not tripped) and both main feedwater pumps (tripped/not tripped). The information is supplied by pressure switches which monitor the hydraulic control oil pressure for the main turbine and the main feed pump turbines. The main turbine or both main feed pump turbines must trip to initiate an ARTS trip.

A contact buffer in each protection channel continuously monitors the state of the associated pressure switches. When the state of the pressure switch changes to a tripped condition, the contact buffer de-energizes the protection channel's trip relay.

Operating bypasses are provided to allow normal startup and shutdown of the plant. Automatic removal of the bypasses occurs based on increasing flux level (greater than 45 percent full power for turbine trip and greater than 10 percent full power for loss of both main feedwater pumps).

The anticipatory trips were added to meet the requirements of the TMI-2 Action Plan, NUREG-0737. The purpose of these trips is to provide a reactor trip signal in those cases where a loss of secondary heat sink would likely challenge the RPS (on some other parameter) or the Pressurizer Electromatic Relief Valve (ERV). The anticipatory trips limit the heat input to the system after a loss of heat sink, reducing the amount of heat that must be removed after the trip.

7.1.2.2.4 Availability of Information

The modules, logic, and analog equipment associated with a single protection channel are contained wholly within a two-bay RPS cabinet. Within these cabinets, there is a meter for every analog signal employed by the protection channel and a visual indication of the state of every logic element. At the top of one cabinet mounted externally is a protection channel status panel. Lamps on this panel give a quick visual indication of the trip status of the particular protection channel and of the RT module associated with it. Additional lamps on the panel give visual indication of a channel bypass or a fan failure.

In addition to the visual indications and readouts within the protection channel cabinets, trip functions, power supply, and analog signals are monitored by the plant computer.

Instrumentation including neutron power, neutron power imbalance, flow, temperature, and pressure is indicated on the main control console.

7.1.2.3 System Evaluation

7.1.2.3.1 System Logic

The RPS is a 4-channel, redundant system in which the four protection channels are brought together in four identical 2-out-of-4 logic networks of the RT modules. A trip in any two of the four protection channels initiates a trip of all four logic networks. The system to this point has the reliability and advantages of a pure 2-out-of-4 system.

Each of the RT modules (2-out-of-4 logic networks) controls a Control Rod Drive breaker, set of breakers, or contactor. Thus, a trip in any two of the four protection channels initiates a trip of all the breakers and contactors. The power breakers and contactors, however, are arranged in what is effectively a 1-out-of-2 x 2 logic. This system combines the advantages of the 2-out-of-4 and the 1-out-of-2 x 2 systems, while eliminating some disadvantages of the 1-out-of-2 x 2 system alone. The combination results in a system that is considered superior to either of the basic systems alone.

In evaluating system performance, it is arbitrarily assumed that "failure" can either prevent a trip from occurring or can initiate trip action. The RPS can tolerate several input function failures without a reduction in performance capability, provided the failures occur in unlike variables in different protection channels or all occur within one protection channel. When a single protection channel fails, the system is left in either a 2-out-of-3 or 1-out-of-3 logic mode as explained below.

The protection channel trip relay of each channel is located in a RT module associated with each channel. Within each RT module is a logic relay for each protection channel. These combine in each module to form the 2-out-of-4 logic. A failure mode and effects analysis of the RT module has demonstrated that single failures within the module or in its interconnections can produce one or more of the following effects:

A. Trip the breaker(s) associated with the module.

B. Place the system in a 2-out-of-3 mode as if the associated protection channel had been bypassed or had suffered a cannot trip type of failure.

C. Place the system in a 1-out-of-3 mode as if the associated protection channel had tripped. Physical removal of a component associated with a single protection channel will leave the remaining components and protective channel operational in a 1-out-of-3 system.

The combination of RT modules and Control Rod Drive breakers and contactors form a 1-out-of-2 x 2 logic. At this level, the system will tolerate a “cannot trip” type of failure of one RT module or of the breaker(s) and/or contactors associated with one RT module without degrading the system's ability to trip all control rods. The failure analysis demonstrates that no single failure involving a RT module will prevent its associated breakers and contactors from opening.

7.1.2.3.2 Redundancy

The protection system logic provides enough redundancy such that failure of one channel can neither prevent a trip function from occurring or cause a trip action. The system is designed such that physical removal of components from a single protection channel would leave the remaining components and protective channels operational in a 1-out-of-3 system.

7.1.2.3.3 Electrical Isolation

All signals leaving the RPS are isolated from the system either by the use of isolation amplifiers for analog signals or by relay contacts in the case of digital signals. The effect of this isolation is to prevent faults occurring to signal lines outside of the RPS cabinets from being reflected into more than one protection channel. The isolation thus provided also assures that two or more protection channels cannot interact through the cross-coupling or faulting of related signal lines.

Faults such as short, open, or grounded circuits and cross-connections of analog output signals from two or more channels have no effect upon the protection channels or their functions.

7.1.2.3.4 Periodic Testing and Reliability

The use of 2-out-of-4 logic between protection channels permits a channel to be tested on-line without initiating a reactor trip. Maintenance to the extent of removing and replacing any module within a protection channel may also be accomplished in the on-line state without a reactor trip.

To prevent either the on-line testing or maintenance features from creating a means for unintentionally negating protection action, a system of interlocks initiates a protection channel trip whenever a module is placed in the test mode or is removed from the system unless the channel is bypassed (Section 7.1.2.3.8).

The test scheme for the RPS is based upon the use of comparative measurements between like variables in the four protection channels, and the substitution of externally introduced digital and analog signals as required, together with measurements of actual protective function trip points.

On-line testing may be performed at different intervals and levels within the system consistent with satisfactory system reliability characteristics. The reliability of the system for random failures has been assured by careful selection of components, testing of logic elements, environmental testing of the system's modules, and long-term prototype proof-testing with the Babcock & Wilcox Test Reactor (BAWTR), as described in BAW-10003, Revision 2, "Qualification Testing of Protection System Instrumentation."

The reliability of the system logic, primarily the relays and coincidence networks in the RT modules, has been made very high so as to eliminate the need for frequent tests of the logic. The logic relays are of two classes: one class designed for high speed, light electrical loads and more than 10^6 operations under load; and the other class of switching electric loads of up to 10 amperes and more than 10^5 operations.

The system test scheme includes frequent visual checks and comparisons within the system on a regular schedule in which all protection channels are compared at one time together with less frequent electrical tests conducted on a rotational plan in which tests are conducted on different protection channels at different times.

A regular check of all RPS indications is required. The check includes such things as comparing the value of the analog variables between protection channels and observing that the equipment status is normal. In addition, power-range protection channel readings are compared with a heat balance calculation of power. These checks are designed to detect the majority of failures that might occur in the analog portions of the system as well as the self-annunciating type of failure in the digital portions of the system. The electrical tests are designed to detect more subtle failures that are not self-evident or self-annunciating and are detectable only by testing.

Electrical tests are conducted on a rotational basis, periodically, in accordance with the Technical Specifications.

Rotational testing has several advantages. It significantly reduces the probability of system failure as compared to testing all protection channels at one time. It also reduces the chance of systematic errors entering the system.

The design of the RPS permits testing during plant shutdown to verify the trip response times of the RPS trip variables. These tests can be performed by plant maintenance personnel using auxiliary testing equipment.

RPS trip response time tests are accomplished by inserting a test signal at the input of an instrument channel and then varying this signal to a point which exceeds the setpoint of the trip variable. Trip response time is determined by measuring the time from the point at which the RPS test input signal reached the setpoint of the trip variable to the time of generation of the RPS trip output signal.

For RPS trip actions which are based on the monitoring of more than one variable, several trip response time tests are performed. The variable of interest is changed while the others are held constant, and a trip response time for each independent variable is measured.

The relationship between tested trip response times and the trip response times assumed in the safety analysis is shown in Table 7-13.

7.1.2.3.5 Physical Isolation

Isolation criteria are met in the physical arrangement of the protection channels within separate cabinets and wiring within the cabinets separating power and signal wiring to reduce the possibility of some physical event impairing system functions. The system's sensors are separated from each other. There are four pressure taps for the reactor coolant pressure measurements to reduce the likelihood of a single event affecting more than one sensor. Outside the RPS cabinets, vital signals and wiring are separated to preserve protection channel independence and maintain system redundancy.

Isolation of non-1E circuits from the 1E circuits in RPS is discussed in B&W Topical Report BAW-10003 Revision 2.

7.1.2.3.6 Primary Power

The primary source of 120-volt, AC power for the RPS comes from four vital buses, one for each protection channel, as described in Chapter 8.

7.1.2.3.7 Manual Trip

Manual trip may be accomplished from the main control console by a trip switch. This trip is independent of the automatic trip system. Power for the Control Rod Drive breakers' undervoltage coils and contactor coils comes from the RT modules. The manual trip switches are between the RT module output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. There is a separate switch in series with the output of each RT module. All switches are actuated through a mechanical linkage from a single pushbutton.

7.1.2.3.8 Bypassing

Each protection channel is provided with two key-operated bypass switches, a channel bypass switch, and a shutdown bypass switch. The channel bypass switch enables a protection channel to be bypassed without initiating a trip. Actuation of the switch initiates a visual alarm on the main console, cabinet alarm lamp panel, and internally in the cabinet. The alarm remains in effect during any channel bypass. The key switch will be used to bypass one protection channel during on-line testing. Thus, during on-line testing, the system will operate in 2-out-of-3 coincidence. The coincidence meets the requirements of IEEE 279-1971. The key switches are interlocked in such a way that, if one is in the bypass position, placing another in its bypass position will have no effect.

Unless the channel is bypassed, as would normally be the case during periodic testing, RPS equipment testing or module removal produces a channel trip which is indicated on the plant annunciator, cabinet alarm lamp panel, cabinet internal indicator, and plant computer alarm. A channel trip results from the testing or removal of only those modules which process information used for protection functions.

The shutdown bypass switch enables the power-imbalance-flow, power-pump, low pressure, and pressure-temperature trips to be bypassed allowing Control Rod Drive tests to be performed after the reactor has been shut down and depressurized below the low reactor coolant pressure trip point. This bypass is indicated on the plant annunciator, the cabinet alarm lamp panel, and internally in the cabinet. Before the bypass may be initiated, a high pressure trip bistable, which is incorporated in the shutdown bypass circuitry, must be manually reset.

The setpoint of the high pressure bistable (associated with shutdown bypass) is set below the low pressure trip point. If pressure is increased with the bypass initiated, the channel will trip when the high pressure bistable associated with the shutdown bypass trips. Additionally, trip protection is provided while in shutdown bypass for reactivity addition accidents at low system temperature and pressure. During any such accident, any safety or regulating rods that are withdrawn will be automatically inserted in the core if the flux level exceeds the bistable setpoint. The use of the shutdown bypass key switch is under administrative control.

Operating bypass of the ARTS are provided to allow normal startup and shutdown. Automatic removal of the bypasses occurs based on increasing flux level.

7.1.3 ENGINEERED SAFEGUARDS ACTUATION SYSTEM

The Engineered Safeguards Actuation System (ESAS) monitors variables to detect loss of RCS boundary integrity. Upon detection of "out of limit" conditions of these variables, it initiates High Pressure Injection (HPI) and Low Pressure Injection (LPI) of the Emergency Core Cooling System (ECCS). It also initiates Reactor Building Cooling and Isolation and Reactor Building Spray Systems. Additionally, it starts the Emergency Diesel Generators DG1 and DG2 which are in standby redundancy with the engineered safeguards 4160-volt buses A3 and A4.

The Arkansas ESAS is basically identical to the Oconee Nuclear Station with the following exceptions:

A. High and Low Pressure Injections are initiated at the same low reactor coolant pressure setpoint since the decay heat removal pumps can operate against a closed head (minimum recirc) for an extended period of time. (Reference Engineering Calculation 88-E-0086-02.)

B. Each of the redundant Reactor Building Spray Systems is actuated by two actuation channels. The channels which actuate a given spray system are located in the same digital subsystem. Since the single failure criteria of IEEE 279 is applied at the subsystem level, this arrangement meets the single failure criteria.

C. Reactor building pressure, for spray actuation, is sensed by pressure transmitters. This change has no functional significance.

The ESAS, as illustrated in Figure 7-6, is organized into subsystems: ANALOG - containing analog instrument strings, and DIGITAL - containing actuation channels.

The ESAS philosophy is that each protective system is redundantly actuated by the digital subsystems when required by the analog subsystems.

7.1.3.1 <u>Design Basis</u>

The design basis of this system includes the items referred to in Section 7.1.1 and the following items.

7.1.3.1.1 <u>Loss of Power</u>

A. The loss of vital bus power to the analog instrument strings and bistables will initiate a trip output of the affected string.

B. Loss of power to actuation channels will not result in protective action. The engineered safeguards equipment is divided between the redundant actuation channels in such a way that the loss of one of the vital power buses does not inhibit the systems' intended actuation function.

7.1.3.1.2 Module Withdrawal

A. Withdrawal of ESAS analog instrument string modules in the ESAS cabinets results in a trip output of the affected analog instrument string.

B. Withdrawal of actuation channel modules does not result in a trip; however, withdrawal of these modules is annunciated.

7.1.3.2 System Design

7.1.3.2.1 System Logic

The ESAS is a basic 2-out-of-3 coincidence logic system. Each input variable is measured three times and the three redundant signals terminate in bistables as shown in Figure 7-6.

The ESAS consists of 10, 2-out-of-3 coincidence logic networks for actuating the equipment in the engineered safeguards systems as described in Table 7-4. The odd numbered networks are located in one digital subsystem while the even numbered networks are located in the other digital subsystem. See Figure 7-6 for details.

The coincidence logic output is normally de-energized. Trip action consists of closing the electrical path through the logic thereby energizing a trip bus. Unit Controls (UCs) are employed to couple the trip signal from the trip bus to the safeguards devices (pump, valve, etc.). There is one UC for every safeguards device or group of related devices controlled by an actuation channel. The trip signal follows a normally closed path in each UC, finally terminating in an output relay, R_O, within each UC. The R_O relays of an actuation channel's UC's are connected to a common trip bus.

A normally open R_O relay contact is connected across a control line terminating in a Control Relay (CR) in the controller of the safeguards device assigned to the individual UC. Power for operating the CR relay is supplied by the equipment controller in series with the R_O relay contact. Trip action involves energizing the R_O relay, closing its contact which in turn energizes the CR relay actuating the assigned safeguards device.

Each actuation channel is equipped with a Logic Test (LT) module. The LT module, together with the UC's, provides the necessary circuitry to permit trip testing individual safeguards devices without tripping an entire engineered safeguards system or actuation channel.

The UC also provides a means whereby, following a system trip, an individual safeguards device may be removed from the control of the ESAS and returned to manual control. This block action cannot be initiated prior to a system trip.

The design of the ESAS logic can be summarized in terms of the systems operation as follows:

A. Each protective action is initiated by either of two actuation channels with 2-out-of-3 coincidence between input signals.

B. Protective action is initiated by applying power from the actuation channel coincidence logic to the individual R_O relays in the UC's which in turn energize the CR relays in each safeguards device controller.

C. There is a UC for every safeguards device or group of related devices (valve, pump, etc.).

7.1.3.2.2 High and Low Pressure Emergency Core Cooling

There are three independent reactor coolant pressure sensors. The output of each sensor terminates in an isolation amplifier which provides individually isolated outputs. One output of each pressure measurement goes to the plant computer for monitoring. One output goes to a bistable, for initiating HPI and LPI for emergency core cooling. Bistable action is initiated when the low reactor coolant pressure setpoint is reached.

Three reactor building pressure transmitters provide signals to the ESAS, each pressure transmitter providing a signal to one of the three analog cabinets through a buffer amplifier. The output signal from the buffer amplifier is provided to three reactor building pressure bistables. Bistable action is initiated when the high reactor building pressure setpoint is reached.

The output of the three redundant reactor coolant pressure bistables is combined in an OR arrangement with the trip outputs of three reactor building pressure bistables. The combination of reactor coolant pressure and reactor building pressure bistables outputs allows either variable to initiate HPI and LPI.

These OR signals are brought together in four identical 2-out-of-3 coincidence logics which form four actuation channels. Either of two actuation channels is independently capable of initiating HPI through redundant HPI equipment. Either of the other two actuation channels is independently capable of initiating LPI through redundant LPI equipment.

Certain containment isolation valves also receive closure signals on low RCS pressure in order to provide diverse actuation. Table 7-5 provides a list of associated systems and valves.

7.1.3.2.3 Reactor Building Cooling and Reactor Building Isolation Systems

There are three reactor building pressure sensors. The output of each sensor terminates in an isolation amplifier which provides individually isolated outputs. One isolated output of each pressure measurement goes to the plant computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each of the three bistables has contact outputs that are combined in parallel with the output of the reactor coolant pressure bistables as previously described.

The outputs of the three bistables are brought together in two identical 2-out-of-3 coincidence logics which provide two engineered safeguards actuation channels. Either of the two channels is independently capable of initiating reactor building isolation and emergency cooling through redundant safeguards equipment. In addition, certain isolation valves also receive closure signals on low RCS pressure in order to provide diverse actuation. Table 7-5 provides a list of systems and valves affected.

7.1.3.2.4 Reactor Building and Reactor Building Penetration Room Ventilation Systems

The design of the Engineered Safety Features (ESF) actuation system includes the Reactor Building Ventilation System and the Reactor Building Penetration Room Ventilation System.

The fans, control, and ducting in the design of the Penetration Room Ventilation System meet the requirements of IEEE 279-1968.

The shut-off damper valves, which isolate these rooms from the rest of the plant upon engineered safeguards actuation, are designed to close upon loss of air supply or electrical power. Each ESF air-operated valve also automatically goes to its safeguards position upon loss of air.

Torque switches, which are provided on motor operated valves, enable the full closure of the valve and the prevention of damage to the valve body. (Damage could result if all of the available torque, required to unseat a valve, could open the torque switch at a time when it is not desired.) It is for this reason that the torque switch is bypassed by a position switch during the first five percent of travel. After this amount of travel the valve moves freely. Should the valve stop at an intermediate position, the torque associated with restarting the valve is well below the setting of the torque switch. The control circuits were examined and it was determined that all engineered safeguards actuated valves would continue their travel if power was lost in the intermediate position. Non-engineered safeguards valves would require re-actuation to enable continuation of valve travel.

7.1.3.2.5 Reactor Building Spray System

Each isolation amplifier, described in Section 7.1.3.2.3, supplies input signals to two additional bistables which trip on high reactor building pressure. The trip output of these six bistables form four identical 2-out-of-3 coincidence logics which form four actuation channels. To minimize the possibility of inadvertent spray actuation, each of the two redundant Reactor Building Spray Systems requires actuating signals from two actuation channels in order for all components in a particular train to operate.

Actuation Channel 7 actuates the Reactor Building Spray pump, P35A, and its attendant spray valve, CV-2401. The suction of P35A is connected to the borated water storage tank by CV-1407 which is actuated by actuation Channels 1 or 3. The tripping of actuation Channels 1 or 3 and 7, which are located in the same digital subsystem, is required to establish spray via this path. The second spray path, which utilizes P35B, CV-2400 and CV-1408, is actuated by actuation channels 2 or 4 and 8, which are located in the redundant digital subsystem, in a similar fashion (see Figure 6-3, Reactor Building Spray System and Table 7-5, Engineered Safeguards Actuation Devices). Actuation meets the requirements of IEEE 279-1968.

7.1.3.2.6 Availability of Information

All system analog signals are indicated within the system cabinets and are monitored by the plant computer. All bistable states are indicated within the cabinets. Logic outputs are indicated within the cabinets and their state monitored by the plant computer. Points monitored by the plant computer are illustrated in Figure 7-6.

Plant annunciators provide the operator with immediate indication of changes in the status of the ESAS. Points annunciated are illustrated in Figure 7-6. All test switches and ESAS channel trips are included.

7.1.3.2.7 Summary of Protective Action

Actions initiated by the ESAS are tabulated in Table 7-4. The devices actuated by the ESAS are listed in Table 7-5. ESF actuated equipment remain in the emergency mode upon removal of the actuating signal. Deliberate operator action is required to return the individual equipment to its normal mode. Channels indicated may be referred to applicable systems as shown in Figure 7-6.

7.1.3.3 System Evaluation

The ESAS is a 3-channel, redundant system employing 2-out-of-3 coincidence between measured variables.

The system will tolerate the failure of one of three reactor coolant pressure measurements or reactor building pressure measurements without losing its ability to perform its intended functions. HPI and LPI for emergency core cooling is actuated by either reactor coolant pressure or reactor building pressure, thus providing diversity in actuation. The system will tolerate single or multiple failures within an actuation channel without affecting the operation of its redundant counterpart. This is the result of keeping even numbered actuation channel logics independent of the odd numbered actuation channel logics. The independence is carried through the actuating channel logic and up to the final actuating (CR) control relay. This is best illustrated by considering the actuation arrangement for the makeup pumps shown in Figure 7-7.

There are two makeup pumps which operate in the event of an accident and one standby makeup pump. P36A is under the control of actuation channel 1 and P36C is under the control of actuation channel 2 while P36B is under the control of both channels and operates only if either P36A or P36C fails to operate. Motor control of P36B is provided by either of electrically independent 4.16 kV breakers A307 (for ESAS channel 1 actuation) and A407 (for ESAS channel 2 actuation). Within the motor controllers of P36A and P36C there is a single CR control relay controlled by the R_O relay in the pump's associated unit control. The operation of the actuation channel logic, the R_O relay in relation to the CR relay, was described previously. Should any two of the three reactor coolant pressure measurements drop below their bistable setpoint, both actuation channel 1 and 2 logics will trip, energizing the appropriate CR relays, and start the pumps. Within each 4.16 kV breaker control circuit for P36B, there is an independent CR relay, controlled by a separate R_O relay (channel 1 for A307 and channel 2 for A407). The arrangement is identical to the way a channel would control any device since all elements are independent and duplicated through the CR relay.

On an engineered safeguards channel 1 reactor building high pressure or Reactor Coolant System low pressure condition, the P36A starts if P36B is not running on bus A3. If P36A is not running or does not start on bus A3, P36B automatically starts if its disconnect switch to bus A3 is closed. Similarly, on a channel 2 actuation P36C starts if P36B is not running on bus A4. If P36C is not running on bus A4 and P36B disconnect to bus A4 is closed, P36B would automatically start. The provision and operation of the CR and R_O relays in each of the control circuits for the P36B pump circuitry provides the desired redundancy and separation for P36B.

The example just presented shows the independence and redundancy of the system. There is redundancy of sensors, logic, and equipment. The redundancy is preserved and kept effective by independence of sensors, instrument strings, logic, and control elements in the final actuator. These characteristics enable the system to tolerate single failures at all levels.

The safeguards devices (pumps, valves, etc.) require electrical power in order to operate and perform their functions. The power for operating the CR relays is taken from the power source of the associated device. Loss of power to a CR relay or device does not impair engineered safeguards functions since there is a second redundant device for each required function. The power for the R_O relays, logic, and instruments is taken from the plant's system of battery backed vital buses since loss of power at this level could affect the performance capability of the system. The system will tolerate the loss of one vital bus without loss of protective capability.

7.1.3.3.1 Redundancy and Diversity

The system as evaluated is shown to have sufficient redundancy and diversity to withstand single failures at every level.

7.1.3.3.2 Electrical Isolation

The use of isolation amplifiers will effectively prevent any faults (shorts, grounds, or cross connection of signals) on any analog signal leaving the system from being reflected into or propagating through the system. The direct connection of any analog signal to a source of electrical power can, at worst, negate information from the measured variable involved. The use of individual R_O relays for each controlled device effectively preserves the isolation of each device, and of elements of one actuation channel from another. Faults in the control wiring between an R_O relay and its CR relay in the controller of a safeguards device will not affect any other device or actuation channel.

Separation of redundant engineered safeguards functions is accomplished by assigning the actuation channels to two groups (Table 7-5). Isolation for power, control, equipment location, and cable routing is maintained throughout. Channels 1, 3, 5, and 7 are assigned to one group (odd actuation channels). Channels 2, 4, 6, and 8 are assigned to a second group (even actuation channels). All equipment required to perform a specific function is assigned to the same group. For example, a pump motor and all valves required for that pump to perform its function are assigned to the same group.

AC power for equipment controlled by the odd numbered actuation channels is supplied from odd numbered switchgear group A3, motor control centers B51, B52, and B53, actuation power from vital bus RS1, and control power from bus D1. Engineered safeguards functions which are redundant to those controlled by the odd numbered actuation channels are controlled by the even numbered actuation channels. AC power for this equipment is supplied from even numbered switchgear group A4, motor control centers B61, B62 and B63, vital bus RS2, and DC control power from bus D2. Where a third unit of ES equipment is used to provide additional redundancy, it is actuated by both the even and odd actuation channels. AC power for this equipment is supplied from either switchgear A3 or A4 and motor control center B55 or B56.

7.1.3.3.3 Physical Isolation

The arrangement of modules within the ESAS cabinets is designed to reduce the chance of physical events impairing ESAS operation. Control wiring between the unit controls and the safeguards devices is physically separated and protected against damage which could impair system operation.

Separation between redundant channels of equipment, control cables, and power cables provides independence of redundant engineered safeguards functions. Power and control cables for each group of engineered safeguards equipment are routed in cable trays that

contain no cable for redundant equipment. The only active components located in the reactor building which must operate during the accident are the Reactor Building Cooling units. Cables for Reactor Building Cooling units each enter the reactor building through two separate penetrations and are routed in two different directions to the cooling units.

7.1.3.3.4 Periodic Testing and Reliability

The ESAS is designed to be tested at any time during plant operation or while shutdown without requiring any defeat of its protective functions. The ESAS is tested in overlapping segments from the sensor input to each analog channel through to the actuation of each final device, e.g. valve or pump. This system fully complies with the requirements of IEEE 279-1968.

The system coincidence logic, 2-out-of-3, is not complex, thus the number of elements which can fail in a single instrument string is small. The redundancy of the logic and the division of safeguards devices between logics forms a system having two parallel protective channels, either of which is capable of performing the required functions. The logic elements are relays which have been selected for reliability and subjected to confirming tests under loads identical to those encountered in the system. The resultant calculated probability of logic failure is several orders of magnitude less than the known or estimated probability of failure of all other system elements.

The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

To avoid a full actuation channel or system trip, the 2-out-of-3 logics are tested by sequentially tripping their inputs and confirming that the proper logic elements respond. The continuity of the electrical connections from the output of the coincidence logic up to each R_O relay then is tested by means of the Logic Test (LT) modules and the unit controls. An LT module can neither prevent a trip of the associated actuation channel when protective action is called for nor initiate a trip of the associated actuation channel.

An individual safeguards device may be actuated by means of the unit control manual switch. Operating this switch energizes the R_O relay as if the actuation channel has tripped, actuating the associated safeguards device. A lamp associated with the unit control confirms that the R_O relay returned to its normal state upon release of the manual switch.

On-line checks of the system will confirm the normal state of the system, principally by comparative readings of similar analog indications between redundant measurements and by the status lamps on bistables and logic modules.

7.1.3.3.5 Manual Trip

A manual trip switch is provided in each ESAS channel. There are 8 manual trip pushbuttons on the control console, one for each actuation channel.

7.1.3.3.6 Bypassing

The ESAS emergency core cooling injection bypass is provided so that HPI and LPI will not be initiated when the plant is undergoing a normal reduction in pressure. Presence of a bypassed condition is indicated on the plant annunciator, cabinet alarm light panel, and internally in the cabinet.

If any portion of the ESAS is required to perform its protection functions when placed in test, indications are available on the plant annunciator, cabinet alarm lamp panel, and internally in the cabinet. Removal of modules in the digital subsystem used in processing information required for protection action is also indicated on the plant annunciator. The digital subsystems cannot be bypassed. Removal of modules in an analog subsystem will result in an analog subsystem trip output that is indicated on the plant computer, cabinet alarm lamp panel, and internally in the cabinet.

The Emergency Core Cooling System (ECCS) trip function (HPI and LPI) can be bypassed whenever the reactor is depressurized below the bypass bistable setpoint. Bypassing must be initiated manually within a fixed pressure band above the trip point of the RCS low pressure bistable. The ECCS may be bypassed only when the reactor pressure is 1,750 psi or less. The bypass is automatically removed when the reactor pressure exceeds the 1,750 psi value. This is in accordance with IEEE 279-1968, Section 4.12. The bypass setpoint is above the trip point in order to obtain a pressure band in which the trip may be bypassed during the normal cooldown. The bypass does not prevent actuation of the ECCS on high reactor building pressure. Bypassing is under administrative control. Since the ESAS incorporates triple redundant analog subsystems, there are three emergency core cooling bypass switches. Two of the three switches must be operated to initiate a bypass.

7.1.4 EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM

The Emergency Feedwater Initiation and Control (EFIC) System is designed to protect against the consequences of a simultaneous blowdown of both steam generators. Upon detection of a steam line break, EFIC automatically initiates action to isolate the faulted steam generator by closing the Main Steam Isolation Valve (MSIV), the Main Feedwater Isolation Valve (MFIV), startup and low load feedwater control valves, and Emergency Feedwater Isolation Valves (EFIV) on each line to the degraded OTSG. Any MSIV or MFIV can be manually closed from the control room at any time.

There is an interlock in the Emergency Feedwater Initiation and Control (EFIC) such that, when the EFIC actuates on low steam generator pressure, an open signal is sent to the EFW turbine steam admission valves and the electric EFW pump is started. This addition assures timely EFW initiation following an EFIC steam line break trip (real or spurious).

The EFIC System is an instrumentation system designed to provide the following:

A. Initiation of EFW,

B. Control of EFW flow rate to the steam generators to control level at appropriate setpoints (at steam generator levels of approximately 31, 312, and 378 inches above the lower tube sheet),

C. Steam generator level rate control when required to minimize overcooling,

D. The selection of appropriate steam generator(s) under conditions of steam line break or main feedwater or emergency feedwater line break downstream of the last check valve,

E. Control of atmospheric dump valves to a predetermined steam generator pressure setpoint, and

F. Signals for isolation of the main steam and main feedwater lines of a depressurized steam generator.

In the achievement of above functions, EFIC serves to provide the mitigating functions for a main steam line break accident.

7.1.4.1 Functions of EFIC

The EFIC System monitors OTSG level and pressure, MFW pump status, DROPS/AMSAC actuation on loss of MFW flow, RCP status, and ESAS channels 3 and 4 in order to initiate EFW or OTSG isolation should an actuation setpoint be reached. EFW is actuated to protect the core against the consequences of an overheating condition upon a loss of main feedwater or a loss of primary side forced circulation (loss of all four RCPs). OTSG isolation is actuated to protect the core against the consequences of an overcooling condition upon a main steam or feedwater line rupture.

The EFW protective function is actuated by any of the following eight conditions:

1. "A" OTSG Low Level
2. "B" OTSG Low Level
3. "A" OTSG Low Pressure
4. "B" OTSG Low Pressure
5. Loss of both Main Feedwater Pumps
6. Loss of all four RCPs
7. ESAS Actuation
8. DROPS/AMSAC Actuation

Note: Some of the input conditions may be manually or automatically bypassed during lower power levels to permit startup and shutdown without initiating EFIC.

7.1.4.2 System Logic

The EFIC System consists of four channels (A,B,C, and D). Each of the four channels are provided with input, initiate, and vector logics. Channels A and B also contain train A and train B trip logics and control logics, respectively. Each channel monitors inputs by means of the input logic, ascertains whether action should be initiated by means of the initiate logic, and determines which steam generators should be fed by means of the vector logic.

Trains A and B monitor initiate signals from each of the four initiate logics by means of the trip logics and transmit trip signals when required. Train A and B also exercise control of emergency feedwater flow to the steam generators by means of control logics to maintain level at prescribed values once EFW has been initiated.

7.1.4.2.1 Initiate Logic

The initiate logic derives its inputs from the input logic and provides signals which result in the issuance of trip signals via the trip logics in trains A and B located in channels A and B cabinets, respectively.

The initiate logic issues a call for EFW actuation when:

A. All four Reactor Coolant Pumps are tripped,

B. Both main feedwater pumps are tripped,

C. The level of either steam generator is low,

D. Either steam generator pressure is low,

E. ECCS actuation (high reactor building pressure or low RCS pressure), or

F. DROPS/AMSAC actuation (loss of MFW flow).

Another function of the initiate logic is to issue a call for steam generator main feedwater and main steamline isolation when steam generator pressure is low.

Note: Some of the input conditions may be manually or automatically bypassed during lower power levels to permit startup and shutdown without initiating EFIC.

7.1.4.2.2 Trip Logic

Each train (A and B) is provided with three 1-out-of-2 taken twice trip logic networks. These networks monitor the appropriate outputs of the initiate logics in each of the channels, ESAS channels 3 and 4, DROPS/AMSAC channels 1 and 2, and output signals for initiating:

A. Emergency feedwater,

B. Steam generator A main steam line isolation, and

C. Steam generator B main steam line isolation.

For each trip function, the trip logic is provided with two manual trip switches. This affords the operator with a means of manually tripping a selected function by depressing both switches. The use of two trip switches allows for testing the trip switches and also reduces the possibility of accidental manual initiation.

Once a trip occurs, the trip can only be removed by manual reset action following return of the initiating parameter to an non-tripping value except as described in the next paragraph.

So that the operator may resume manual control of EFIC initiated devices following a trip, each trip logic is provided with a manual reset pushbutton. Operation of the manual reset pushbutton:

A. Will have no effect on the trip logic so long as a trip condition does not exist;

B. Will remove the trip from the trip bus only so long as the switch is depressed in the case of a one half trip (either bus but not both tripped). This allows for testing the manual function; and,

C. Will remove the trip from both buses so long as a full trip (both buses are tripped) exists. This is accomplished by means of manual latching logic. Institution of the manual function also breaks the latch so that, if the initiating stimuli clears, the trip logic will revert to the automatic trip mode in preparation for tripping if a parameter returns to the trip region.

7.1.4.2.3 Vector Logic

The vector logic monitors:

A. Steam generator pressure signals and

B. EFW trip signals (vector enable) originating in Channels A and B trip logics.

The vector logic outputs are in a neutral state until enabled by trip signals (vector enable) from the Channel A or B trip logics.

When enabled, the valve open/close commands are determined by the relative values of steam generator pressures as follows:

Pressure Status	Steam Generator A Valve Command	Steam Generator B Valve Command
SG A & B > Setpoint	Open	Open
SG A > Setpoint & SG B < Setpoint	Open	Closed
SG A < Setpoint & SG B > Setpoint	Closed	Open
SG A < Setpoint & SG B < Setpoint and SG A & B within 100 psig	Open	Open
SG A < Setpoint & SG B < Setpoint and SG A 100 psi > SG B	Open	Closed
SG A < setpoint & SG B < Setpoint and SG B 100 psi > SG A	Closed	Open

7.1.4.3 EFW Signal Application

Figure 10-2 illustrates the application of EFIC signals to the Emergency Feedwater System.

Figure 10-2 illustrates, in block diagram form, the path taken by EFIC signals during their operation.

Figure 7-22, shows the application of EFIC signals to the steam generator secondary system.

7.1.4.4 Interface with Valve and Pump Controllers

Valve and pump controllers are generally designed such that signals from the EFIC system will override any other control signals. An exception is the MSLI signal to ICS. In this case, certain control signals have precedence over the EFIC input. Also, when an EFIC signal is removed, the controller design is such that valves (other than the EFW control valve and valves controlled by ICS) will not change position and pumps will not change state without a specific manual command. When the vector logic close command to the EFW control valve is removed, the control valve shall be positioned as required by the EFW control system or the manual control as selected.

7.1.4.5 Manual Control of EFW Components

Manual override capability of EFIC control signals exists in the control room on cabinet C09 for the following valves:

A. CV-2670, CV-2627, CV-2620, CV-2626 (EFW isolation valves);

B. CV-2667, CV-2617, CV-2613, CV-2663 (EFW turbine steam valves);

C. CV-2619, CV-2676 (atmospheric dump isolation valves);

D. CV-2630, CV-2680 (main feedwater isolation valves); and,

E. CV-2645, CV-2646, CV-2647, CV-2648 (EFW flow control valves).

Position of the override control in any other than the "AUTO" position is continuously annunciated in the control room for each EFW train.

The EFW motor driven pump P7B may be stopped by placing its control room handswitch on C09 to the pull-to-lock position. This is annunciated as part of EFW train B.

7.1.4.6 Actuation Requirements

EFW shall be automatically initiated after the occurrence of any of the following conditions:

- Loss of all main feedwater as indicated by the loss of both main feedwater pumps or by
- actuation of the DROPS/AMSAC;
- Low level in either steam generator;
- Loss of all four Reactor Coolant Pumps (RCPs);
- Low pressure in either steam generator; and
- ESAS ECCS actuation (high reactor building pressure or low RCS pressure).

Note: Some of the input conditions may be manually or automatically bypassed during lower power levels to permit startup and shutdown without initiating EFIC.

<u>Level* Requirements</u>

Three adjustable level setpoints are required.

A. Following EFW actuation, the level setpoint shall be automatically selected to be approximately 31 inches, if any of the four RCPs are operating.

B. Following EFW actuation, the level setpoint shall be automatically selected to be approximately 312 inches if all four RCPs are tripped. The transition from the 31 inches to 312-inch setpoint, is controlled by the "rate limited follower" control circuit. This circuit will minimize overcooling by restricting rapid level increases by maintaining steam generator pressure between 800 and 1050 psig. Upon reaching the required 26 feet natural circulation setpoint flow is controlled as required to maintain level regardless of SG pressure (see 7.1.4.7)

C. Provision has been made for manual selection of a reflux boiling level control setpoint of approximately 378 inches. This setpoint will be selected by the operator in accordance with operating guidelines and is intended for use during conditions involving loss of RCS subcooling margin. Operation during transition to 378 inches is controlled by the rate limited follower as in "B" above.

* For the purpose of EFW design, "LEVEL" refers to the equivalent height of a saturated liquid column (1,065 psia) referenced from the top of the lower tube sheet.

7.1.4.7 Flowrate Requirement

The objective of flowrate control is to minimize RCS overcooling for low decay heat conditions. The EFW flow rate is controlled by the rate of level increase**. A level rate of two to eight inches per minute has been estimated to provide adequate RCS cooling. This fill rate is varied automatically as a function of steam generator pressure in the range of 800 to 1,050 psig for the transient conditions which require EFW.

The level rate limit is adjustable under administrative control.

** When RCPs are in service, EFW flow rate is not limited.

7.1.4.8 Steamline Break/Feedwater Line Break

A steam line break or feedwater line break that depressurizes a steam generator will cause the isolation of the main steam line and main feedwater line on the depressurized steam generator. If isolation of the steam generator does not isolate the break, EFW will be provided only to the intact steam generator. No single active failure in the system will prevent EFW from being supplied to the intact steam generator nor allow EFW to be supplied to the broken steam generator.

To meet these requirements the following design was implemented:

A. Isolation - low steam pressure (below 584.2 psig) in either steam generator will isolate the main steam line and main feedwater line to the affected steam generator.

B. Steam Generator Selection

- If both steam generators are above 584.2 psi - supply EFW to both OTSGs.
- If one steam generator is below 584.2 psi and one above, supply EFW to the steam generator with pressure above 584.2 psi.
- If both steam generators are below 584.2 psi and within 100 psi of each other, supply EFW to both.
- If both steam generators are below 584.2 psi and one steam generator pressure is 100 psi below the other, supply EFW to the generator with the higher pressure.

7.1.4.9 Heatup from Cold Shutdown to Hot Standby

Before heating up from cold shutdown, the operator will verify the status of the EFIC. All signals should be bypassed with the exception of the low OTSG level initiation of EFW, which is required to be operable when RCS temperature is at or above 280 °F. The loss of MFW pumps trip will be bypassed based on a logic function located in the RPS. The DROPS/AMSAC loss of MFW flow trip will be bypassed based on a logic function located in the DROPS/AMSAC.

When the first Reactor Coolant Pump is started, the "Loss of Four Reactor Coolant Pumps" initiation signal may be manually reset. This is accomplished by depressing the "Bypass Reset" button located in each of the EFIC cabinets. If the bypass is not manually reset, it will automatically reset when the plant reaches 10 percent power. As the plant begins heating up, the bypass of the low OTSG pressure signal will be reset. This bypass reset will automatically occur before both steam generators exceed 750 psig.

7.1.4.10 Hot Standby to Full Power

At hot standby conditions, all trip functions should be active except the MFW pump trip and the DROPS/AMSAC MFW flow trip. As power is increased, the MFW pump trip will automatically become active at about 10 percent power. The logic for this function is located in the NI/RPS. The DROPS/AMSAC MFW flow trip will automatically become active at about 45 percent power.

When reducing power from full power to hot standby, the MFW pump trip will be automatically bypassed in the NI/RPS when power is reduced below approximately 10 percent. Likewise, the DROPS/AMSAC MFW flow trip will automatically be bypassed in the DROPS/AMSAC when power is reduced below approximately 45 percent.

7.1.4.11 Cooldown from Hot Standby to Cold Shutdown

During the cooldown, two shutdown bypasses must be implemented. The first is the low steam pressure shutdown bypass. When either steam generator is below 750 psig, this bypass may be implemented by depressing the low steam pressure shutdown bypass buttons located in the EFIC cabinets. One button in each of the four channels must be depressed. This action must be taken before either OTSG pressure reaches 584.2 psig.

The second shutdown bypass is for the "Loss of four Reactor Coolant Pumps" trip. This must be done before the system is ready for the last RCP to be tripped. It can be implemented any time after power has been reduced below 10%. This bypass may be actuated by depressing the loss of four Reactor Coolant Pumps shutdown bypass switches on the EFIC cabinets.

7.1.4.12 Testing

The number of components which can fail in a single instrument channel is small. The redundancy of the logic and the division of the devices forms a system having two parallel protective channels, either of which is capable of performing the required functions. These characteristics are basic to an inherently reliable system. All components have been selected for reliability and are subjected to tests under identical loads encountered in the system.

The built-in test facilities permit complete electrical testing of each channel. During a test, the redundant components are capable of providing a complete protective signal. Contact monitoring circuits were added to the test scheme to verify test contact positions during and after testing.

The EFIC System was designed to be tested from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the trip interface equipment. (Activation of both relays is required in order to actuate the controllers.) The two relays are tested individually to prevent automatic actuation of the component. Testing of the sensor inputs to the EFIC will normally be accomplished with the plant at cold shutdown.

The main steam line isolation portion is tested as part of the EFIC test. In addition, MFIV and MSIV exercise is conducted as described below.

Exercise MSIV in Channel 1 Test Mode

Exercising a MSIV is performed by energizing an exercise solenoid valve on the MSIV, thus allowing air to bleed-off and the valve to begin to close. A limit switch controls the distance through which the valve can move. Exercising cannot occur until the channel 1 test logic has been powered and the operator has manually initiated the test.

Exercise MFIV in Channel 1 Test Mode

Exercising a MFIV is performed by momentarily providing a close signal to the valve control circuit. A limit switch and backup time-delay relay control valve travel. Exercising cannot occur until the channel 1 test logic has been powered and the operator has manually initiated the test.

Emergency Feedwater Flow Test Mode

An EFW flow test loop was included to allow flow test of each EFW pump and flow control valves.

7.1.4.13 EFW Indications

7.1.4.13.1 Emergency Feedwater Flow Indication

Qualified flow indication is provided in each flowpath to each steam generator. The flow indication is powered from vital power. This design provides qualified flow indication for operator information.

7.1.4.13.2 Emergency Feedwater Pump Suction Pressure

Class 1E pressure switches are provided at the suction to each emergency feedwater pump. The switches supply low suction pressure annunciation in the control room for operator information. Upon indication of this alarm, the operator manually transfers the suction of emergency feedwater pumps to the Service Water System, which is the assured source of water. Although the Service Water System, is the assured source of water, procedures direct operators to utilize all available condensate sources prior to switching to service water.

7.1.4.13.3 Condensate Storage Tank T41B Level

Two redundant Class 1E channels of level instrumentation are provided for the qualified CST T41B tank. One channel feeds a qualified indicator on C09 in the control room. The other channel feeds a qualified recorder, also on C09. Low and low-low level alarms are also provided in the control room.

7.2 CONTROL SYSTEMS

7.2.1 DESIGN BASES

Reactor power is regulated by the use of movable Control Rod Assemblies (CRAs) and soluble boron dissolved in the coolant. Control of relatively fast reactivity effects, including Doppler, xenon, and moderator temperature effects, is accomplished by the control rods. The control response speed is designed to overcome these reactivity effects. Relatively slow reactivity effects, such as fuel burnup, fission product buildup, samarium buildup, and hot-to-cold moderator reactivity deficit, are controlled by soluble boron.

Control rods are normally used for control of xenon transients associated with normal reactor power changes. Chemical shim will be used in conjunction with control rods to compensate for equilibrium xenon conditions. Reactivity control may be exchanged between rods and soluble boron consistent with limitations on power peaking. Reactor regulation is a composite function of the Integrated Control System (ICS) and Rod Drive Control System (RDCS). For historical information see the FSAR.

7.2.2 ROD DRIVE CONTROL SYSTEM

The RDCS includes drive control, power supplies, position indicators, operating panels and indicators, safety devices, and enclosures.

7.2.2.1 Design Bases

The RDCS design bases are categorized into safety considerations, reactivity rate limits, startup considerations, and operational considerations.

7.2.2.1.1 Safety Considerations

A. The CRAs are inserted into the core upon receipt of protection system trip signals. Trip command has priority over all other commands.

B. No single failure will inhibit the protective action of the RDCS.

7.2.2.1.2 Reactivity Rate Limits

The speed of the mechanism and group rod worth provide the reactivity change rates required. For design purposes the maximum rate of change of reactivity that can be inserted by any group of rods has been set in Section 14.1.2.3.3. The drive controls, i.e. the drive mechanism and rods combination, have an inherent speed-limiting feature.

7.2.2.1.3 Startup Considerations

The RDCS design bases for startup are as follows:

A. Reactor regulation during startup is a manual operation.

B. Control rod "out" motion is inhibited when a high startup rate is detected.

C. An interlock is provided to prevent energization of the Control Rod Drive Mechanisms if sufficient cooling water flow rate does not exist.

7.2.2.1.4 Operational Considerations

For operation of the reactor, functional criteria related to the RDCS are:

A. CRA Positioning

The RDCS provides for controlled withdrawal, controlled insertion, and holding of the CRAs to establish the power level required for a given reactor coolant boron concentration.

B. Position Indication

Continuous rod position indication, as well as full-in and full-out position indication, is provided for each control rod drive.

C. System Monitoring

The RDCS design includes provisions for routinely monitoring conditions that are important to safety and reliability.

7.2.2.2 System Design

The RDCS provides for withdrawal and insertion of the CRAs to maintain the desired reactor output. This is achieved either through automatic control by the ICS discussed in Section 7.2.3 or through manual control by the operator. As noted previously, this control compensates for short-term reactivity changes. It is achieved through the positioning in the core of 60 CRAs and eight Axial Power Shaping Rod assemblies. The 60 rods are grouped for control and safety purposes into seven groups. Four groups function as safety rods and three groups serve as regulating rods. An eighth group serves to regulate axial power peaking due to xenon poisoning. Seven of the eight groups may be assigned from four to twelve CRAs. Eight rod assemblies are used in Group 8. Control rods are arranged into groups at the control RDCS patch panel. Typically, 36 rods, including the Axial Power Shaping Rods, are assigned to the regulating groups, and 32 rods are assigned to the safety rod groups. The most recent rod grouping arrangement for the Unit 1 core may be found in the Reload Report (Chapter 3A) and its supporting documentation.

A typical rod grouping arrangement is shown below:

Safety Rods	Regulating Rods	Axial Power Shaping Rods
Group 1 - 8	Group 5 - 8	Group 8 - 8
Group 2 - 8	Group 6 - 8	
Group 3 - 8	Group 7 - 12	
Group 4 - 8		

During startup, the safety rod groups are withdrawn first, enabling withdrawal of the regulating control groups. The sequence allows operation of only one regulating rod group at a time except where reactivity insertion rates are low (first and last 20 ± 5 percent of stroke), at which time two adjacent groups are operated simultaneously in overlapped fashion. These insertion rates are shown in Figure 7-8.

As fuel is used, dilution of soluble boron in the reactor coolant is necessary. When Group 6 is more than 95 percent withdrawn, interlocks permit dilution. The reactor controls insert Group 6 to compensate for the reduction in boron concentration by dilution. The dilution is automatically terminated by a preset volume measuring device. Interlocks are also provided on Group 6 rod position to terminate dilution at a preset insert limit (see Section 9.1.2.6.C).

7.2.2.2.1 System Equipment

The RDCS consists of three basic components: (1) control rod drive motor power supplies, (2) system logic, and (3) trip breakers. The power supplies consist of four group power supplies, an auxiliary power supply, and two holding power supplies. The group power supplies are of a redundant, 6-phase, half-wave rectifier design. In each half of a group power supply, rectification and switching of power is accomplished through the use of Silicon Controlled Rectifiers (SCRs). This switching sequentially energizes first two, then three, then two of the six CRA motor stator windings in stepping motor fashion to produce a rotating magnetic field for the CRA motor to position the CRA. Switching is achieved by gating the six SCRs on for the period each winding must be energized. As each of the six windings utilize SCRs to supply power, six gating signals are required.

Gating signals for the group power supplies are generated by a solid state programmer consisting of input interfacing relays, a programmed microcomputer, and output drivers. The insert and withdraw commands and the driver outputs are redundant, thus providing separate but synchronized gating signals to the dual power supply units.

Identical type power supplies are used for the regulating (control) groups and for the auxiliary power supply. Each half of each group power supply is capable of driving up to 12 drive mechanisms, the maximum number that may be in any one group. The power supplies have dual power inputs, each half fed from separate power sources and capable of carrying the full load.

Unlike the control group power supplies, the holding power supply is used to maintain the safety rods fully withdrawn; consequently, switching is not required. A 6-phase rectified power supply is used for this purpose. Two holding power supplies are provided. Each is rated to furnish power to one winding of 48 mechanisms.

The auxiliary power supply is used to position the safety rod groups and to provide single rod control. The safety rod groups are maneuvered with the auxiliary power supply and then, when fully positioned, are transferred to the holding buses described above. After positioning the safety rods, the auxiliary power supply is available to the regulating groups, through transfer relays, to serve either as a single rod controller, should repositioning of a single rod be necessary, or as a spare group controller, should one of the group control power supplies require maintenance. The auxiliary power supply cannot be used to control more than one group at a time.

The system logic encompasses those functions which command control rod motion in the manual or automatic modes of operation, including Control Rod Drive (CRD) sequencing, safety and protection features, and the manual trip function. Major components of the logic system are the operator's control panel, CRA position indication panels, automatic sequencer, and relay logic.

Switches are provided at the operator's control panel for selection of the desired rod control mode. Control modes are: (1) automatic mode--where rod motion is commanded by the ICS; and, (2) manual mode--where rod motion is commanded by the operator. Manual control permits operation of a single rod or a group of rods. Group 8 control rods can only be controlled manually even when the remainder of the system is in automatic. Indicators on the operator's control panel alert the operator to the systems status at all times.

The sequence section of the logic system utilizes rod position signals to generate control interlocks which regulate rod group withdrawal and insertion. The sequencer operates in both automatic and manual modes of reactor control and controls the regulating groups only. Analog position signals are generated by the reed switch matrix on the CRA and an average group position is generated by an averaging network. This average signal serves as input to electronic setpoint trip units which are activated at approximately 20 percent and at approximately 80 percent of group rod withdrawal. Two bistable units are provided for each regulating group. Outputs of these bistables actuate "enable" relays which permit the rod groups to be commanded in automatic or manual mode.

The automatic sequencer circuit can control only Rod Groups 5, 6, and 7. The safety rod groups, Groups 1 - 4, are controlled manually, one group at a time. In addition, the operator must select the safety group to be controlled and transfer it to the auxiliary power supply before control is possible. There is no way in which the automatic sequencer can affect the operations required to move the safety rods.

In addition to the sequencer, relay logic monitors are provided in the "enable" circuits which prohibit out-of-sequence conditions. The selection of manual control mode and sequence bypass mode functions permit intentional out-of-sequence conditions. "Sequence" operation may be bypassed at any time if the manual control mode has been selected. If automatic control is selected, "sequence" operation cannot be bypassed. This condition is indicated to the operator.

"Sequence bypass" operation permits selection of any rod group or any single rod for control. It will not permit selection of more than one rod group at any given time. Motion of more than one group at any given time is also not possible when this operation is selected.

Inputs to the system logic from the RPS and ICS provide interlock control over rod motion. These interlocks cause rod motion command lines and/or automatic control mode selection to be inhibited.

Under certain conditions, the nuclear instrumentation generates an "out inhibit" signal. When this signal is received by the CRD system, all out command circuits are disabled, thus preventing withdrawal of all rods in either automatic or manual control.

Automatic insertion of rods can only be commanded by the ICS when the CRD system is in the automatic mode. These commands can only affect Rod Groups 5, 6, and 7.

Insertion of control rod groups 5, 6, and 7 will be initiated by the DROPS Diverse Scram System (DSS) 2-out-of-2 actuation signals. The DROPS/DSS will accomplish control rod insertion by removing 24V DC power from the programmable controllers and gate drives of groups 5, 6, and 7 and the auxiliary group (reference Section 7.2.4 for the DROPS/DSS).

In the RDCS, two methods of position indication are provided: an absolute position indicator and a relative position indicator. Either position signal is available to the control board indicator through a selector switch. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity.

As the leadscrew (and the CRA) moves, the switches operate sequentially producing an analog voltage proportional to position. The position indicator assemblies have been replaced with a new R4C type position indicator tube. The resolution of these position indicators under normal conditions is ± 2.5 inches. The R4C allows continued operation of the position indicator system with a faulty reed switch. Resolution under these conditions is ± 3.5 inches. For Group 8 only, there is an additional + 3/8" bias or offset error which affects signal accuracy of the R4C assembly. Also for Group 8, a one rod misalignment of up to an additional 5/8" applies. Other reed switches are included in the same tube with the position indicator matrix to provide full-in and full-out limit indications.

The relative position transducer is a small pulse-stepping motor driven from the power supply for the rod drive motor. This small motor is coupled to a potentiometer with an output signal accuracy of ± 0.97 inch, producing a readout with an accuracy of ± 2.4 inches.

RDCS trip breakers and contactors are provided to interrupt power to the control rod drive motors. When power is removed, the roller nuts disengage from the leadscrew and a gravity free-fall trip of the CRA occurs. Two series trip methods are provided for removal of power to the CRA motors. First, a trip is initiated when RPS logic interrupts power to the undervoltage trip circuit of the main AC feeder breakers. Secondly, a trip is initiated when the SCR gating power and the DC holding power are interrupted. As parallel power feeds are provided on both holding and gating power, interruption of both feeds is required for trip action in either method of trip. Trip circuitry is shown in Figure 7-9. The Group 8 drive mechanisms are modified to prevent rod drop into the core when power is removed from the stators. In this type of mechanism, the roller nuts are mechanically restrained to remain engaged with the lead screw at all times. Thus, the mechanical "trip" action has been removed from these ASPRs and they remain at the position they occupied immediately before trip was initiated. Following each refueling outage prior to return to power, this design feature of the APSRs is demonstrated to prevent movement on a loss of power (reactor trip). When a reactor trip is initiated, power to the Group 8 power supply is interrupted in the same manner as for the other regulating power supplies.

AC power feed breakers are of the 3-pole, stored energy type, each housed in a separate metal clad enclosure. Each breaker trip mechanism provides diverse tripping action by utilizing an instantaneous undervoltage trip device and a backup shunt trip actuated by an undervoltage relay. The secondary trip breakers, mechanically gauged for DC service, are also of the stored energy type and utilize the same type diverse trip circuitry parallel connected to each breaker pair of common supply. The undervoltage relays and trip devices are operated by the RPS as are the 2-pole gating power contactors.

7.2.2.3 System Evaluation

7.2.2.3.1 Safety Considerations

A reactor trip occurs whenever power has been removed from the rod drive motors. The design provides two stored energy breakers to interrupt the electrical feeds to rod drive control power supplies and a second set of circuit-interrupting devices in series on the output of the power supplies. All devices have interrupting capacity of sufficient rating to open under any group load configuration. Reactor trip is further assured by providing series trip devices, split buses, and provisions for periodic testing. Trip redundancy is provided by series breakers while availability and testability are provided through dual power sources. Redundant power supplies permit testing of the trip action of each power-interrupting device without loss of plant availability.

Reactivity shutdown margin provided by the safety rods is assured by diversification of their power buses. This feature, as shown in Figure 7-9, utilizes four separate buses, each having a separate trip device, to power the safety rods. A failure in one bus does not reflect into the other buses, therefore, a single failure in the distribution system for the safety rods does not prevent a plant shutdown.

The probability of an external DC source being applied to the control rod drive mechanisms downstream from the reactor trip points such that the CRAs are held in their withdrawn positions after a trip is not considered credible for the following reasons.

A. The secondary trip devices in the Control Rod Drive System remove all DC power from the drives.

B. Control rod drive power cables are terminated at only three points between the control rod drive cabinets and the drive mechanisms.

   Two of these terminations are made outside and inside the reactor building electrical penetrations inside junction boxes containing only control rod drive power cables. The third termination is made in bulkhead connectors (one per drive) in the area of the reactor. The only other cables terminated in this area are the control rod drive instrumentation cables. The instrumentation cables are terminated in bulkhead connectors of a different size and configuration, therefore, mismating of connectors could not be accomplished. Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection is performed.

C. No cable splices are permitted between termination points described.

D. DC systems from the batteries are not grounded and are equipped with ground detecting circuitry.

In summary, series redundant trip devices having adequate rating, testability, and a "split-bus" arrangement insure safety of reactor trip circuits.

7.2.2.3.2 Reactivity Rate Limits

The desired rate of change of CRA reactivity insertion and uniform reactivity distribution over the core are provided for by the control rod drive and power supply design and the selection of rods in a group. The motor, leadscrew, and power supply designs are fixed to provide a uniform rate

of speed of 30 in/min. The reactivity change is then controlled by the rod group size. To insure flexibility in this area, a patch panel has been included in the power supply to enable the interchange of rod worth between rod groups. Any rod may be patched into any group with the exception of Group 8. Rod patch configurations are administratively confirmed by comparison of commanded rod motion to indicated rod motion. This verification is performed whenever the control rod drive patch panel is reconnected after testing, reprogramming, or maintenance. This verification is acceptable when the proper rod has responded and movement is indicated on the plant computer printout or on the input to the plant computer for that rod. An improperly programmed rod is considered inoperable until properly programmed. The control rod drive patch panels are locked with access limited to those authorized by Operations.

A linear reactivity insertion rate is provided for by the withdraw-insert sequence of rod groups. As described in Section 7.2.2.2.1, the sequence and sequence monitor are interlocked to provide redundancy in the sequencing of Groups 5, 6, and 7. Control interlocks are in the sequencing equipment and protection against total rod withdrawal accident is provided by the RPS as analyzed in Section 14.1.2.2.

Uniform and symmetrical group insertion rate is provided for by synchronous withdrawal of all rods in that group. Such synchronous withdrawal is achieved by the design of the power supply. A group power supply operates synchronously by having its load (four to 12 CRA motor windings) connected in parallel on the output of the SCRs. As the programmer gates on the SCRs, all rods in a group have the same motor winding energized simultaneously, producing synchronous motion of the entire group.

Speed control is accomplished through the use of an internal quartz crystal and counters to set the cycle time for RUN and JOG speed, resulting in highly accurate rod movement signals. Speed and direction commands are reread after each motor phase step to detect any change in status. Speed limiting is provided with the accuracy and reliability of the quartz crystal and solid state counters.

Each control rod is provided with a rod position indication monitor (Section 7.2.2.3.4) to sense asymmetric rod patterns by comparing the individual rod position with its group average position. When the rod moves out of step from its group by a preset amount, the monitor alarms the condition to the operator, computer, and the ICS. Depending on the power setting and the control mode, action is initiated by the ICS, if an in-limit is also present, to insert rods and reduce power.

7.2.2.3.3 Startup Considerations

The rod drive controls receive interlock signals from the ICS and Nuclear Instrumentation (NI). These inputs are used to inhibit automatic mode selection if a large error exists in the ICS reactor controls and to inhibit out motion for high startup rates, respectively.

In addition to the startup considerations, dilution controls, to permit removal of reactor shutdown concentrations of boron in the reactor coolant, are provided. This control bypasses the normal reactor coolant dilution controls, described in Section 7.2.2.2, providing all safety rods are withdrawn from the core and the operator initiates a continuous feed and bleed cycle. An additional interlock on Rod Group 5 inhibits the use of this circuit when Rod Group 5 is more than 80 percent withdrawn.

7.2.2.3.4 Operational Considerations

The CRA positioning system provides the ability to move any rod to any position required consistent with reactor safety. As noted in Section 7.2.2.3.2, a uniform speed is provided by the drive system. A fixed rod position, when motion is not required, is obtained by the power supply ability to energize two adjacent windings of the CRA motor stator. This static energizing of the windings maintains a latched stator and fixed rod position.

Position Indication

As previously described, two separate position indication signals are provided. The absolute position sensing system produces signals proportional to CRA position from the reed switch matrix located on each CRA mechanism. The relative position indication system produces a signal proportional to the number of CRA motor power pulses from a stepping motor and precision potentiometer for each CRA mechanism.

Position indicating readout devices mounted on the operator's console consist of 68 single rod position meters and four control group average position meters. Accuracy of all meters is to plus or minus one percent of full scale. The operation of a selector switch permits either relative or absolute position information to be displayed on the single rod meters.

The control group average meters display the arithmetic average of the absolute position signals of all CRAs in a group. A selector switch on the operator's console permits the group meters to display either the positions of all safety rod groups (Groups 1 - 4) or the positions of all regulating rod groups (Groups 5 - 7) and the Axial Power Shaping Rod groups (Group 8).

Indicator lights are provided on the single-rod meter panel to indicate when each rod is (1) fully inserted, (2) fully withdrawn, (3) under control, and (4) whether a fault is present. Indicators on the operator's console show full insertion, full withdrawal, under control, and fault indication for each of the eight control rod groups.

The plant computer monitors the status of each trip device in the CRD system and will alarm a trip condition.

Failures which could result in unplanned control rod withdrawal are continuously monitored by fault detection circuits. When failures are detected, indicator lights and alarms alert the operator. Fault indicator lights remain on until the fault condition is cleared by the operator. A list of indicated faults is shown below:

A. Asymmetric rod pattern (indicator and alarm).

B. Sequence faults (indicator and alarm).

C. Rod position sensor faults.

D. Safety rods not withdrawn (indicator only).

Faults serious enough to warrant immediate action produce automatic correction commands from the fault detection circuits. Status indicators on the operator's console provide monitoring of control modes. A description of each fault detector follows.

A. Asymmetric Rod Monitor

1. Design Basis - To detect and alarm if any rod deviates from its group reference position by more than a maximum of 13 inches true position.

2. Circuit Operation - There are 68 asymmetric rod pattern monitors; one assigned to each control rod. These monitors continuously compare the individual rod absolute position signal with the absolute group reference (average) signal. The absolute value of the difference between the two signals is computed and if this difference is less than the maximum value set by the circuit calibration, no output results. If, however, the difference is greater than the setpoint, a relay is actuated which alarms the asymmetric condition. Two alarm channels are provided in each monitor which are identical except for the setpoints. One setpoint is calibrated for a 7-inch signal differential (maximum 11-inch true position separation) and initiates an alarm. The other setpoint is a 9-inch signal differential (maximum 13-inch true position separation) and initiates the action described below.

3. Corrective Action - Action taken upon detection of an asymmetric rod fault depends upon the control mode and the power level in effect at the time the fault is detected. Corrective action is the same for any asymmetric condition including "stuck-in," "stuck-out," or dropped control rods.

   Detection of a 7-inch signal differential is defined as an "asymmetric rods alarm." Actuation of this alarm causes the fault indicator lamp for that rod to be energized and an alarm signal to be sent to the plant computer and annunciator.

   If the condition is not corrected and the separation increases to a 9-inch signal difference, the following actions occur.

   a. "Asymmetric fault" lamp on the operator's console is energized. If operation is in the manual control mode, operator action is required by administrative control.

   b. If operation is in the automatic mode and an in-limit is detected also, a "runback fault" signal is sent to the ICS. The ICS will impose a maximum reactor power level of 40 percent of rated power if power is initially less than 40 percent. When reactor power is greater than 40 percent of rated power, the Control Rod Drive System generates an "Out Inhibit" signal which disables the "Out" command circuits to all drives and the ICS initiates a runback to 40 percent reactor power. "Out Inhibit" alarms are sent to the ICS, plant annunciator, and plant computer. Reactor power remains limited to 40 percent maximum in automatic control until the fault is corrected.

B. Sequence Monitor

1. Design Basis - To detect any motion of the regulating rod groups outside of the predetermined automatic sequence patterns and to prevent further automatic motion when such conditions occur.

2. Circuit Operation - The sequence monitor continuously compares the group average (reference) signals for each regulating rod group with the allowable sequence patterns. Bistable amplifiers and digital logic are used for this purpose.

The safety rod groups' "Out" limit signals serve as a permissive to automatic sequencing; the sequence monitor prevents automatic control until the safety rods are fully withdrawn.

3. Corrective Action - When an out-of-sequence condition is detected and operation is in the automatic control mode, the automatic mode disengages and an alarm lamp alerts the operator to the malfunction. Control reverts to manual and remains in manual until the fault is corrected and the system is reset by the operator.

C. Rod Position Sensor Faults

All rod position sensor faults lead to false asymmetric, stuck, or dropped rod symptoms which are acted upon by the asymmetric rod monitor described in Item A above.

The consequences of rod position sensor faults have been mitigated by the replacement of position indicator assemblies with new, redundant "R4C" position indicator assemblies. The R4C provides channel isolation capability through the use of a switch on the amplifier card.

D. Safety Rods Not Withdrawn

1. Design Basis - To prevent, on plant startup, withdrawal of the regulating rods until the safety rods are fully withdrawn.

2. Circuit Operation - The circuit continuously monitors the group "out" limit for the four safety rod groups. When the four groups are all fully withdrawn, signals are sent to the sequencer and the sequence monitor which permit automatic control.

3. Corrective Action - Alarms are provided.

7.2.3 INTEGRATED CONTROL SYSTEM

7.2.3.1 Design Basis

The Integrated Control System (ICS) provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator feedwater system, and turbine. When any single portion of the plant is at an operating limit or a control section is on manual, the ICS design uses the limited or manual section as a load reference.

The ICS maintains constant average reactor coolant temperature when permitted by reactor and steam generator operating limits and constant steam pressure at all loads. Optimum unit performance is maintained by (1) limiting steam pressure variations, (2) limiting the imbalance between the steam generator, turbine, and the reactor, and (3) limiting the total unit load demand upon loss of capability of the steam generator feed system, the reactor, or the turbine generator. The control system provides limiting actions to assure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power.

The response of the Nuclear Steam Supply System (NSSS) to increasing and decreasing power transients is limited by the ICS as indicated in Table 7-6.

7.2.3.2 System Design

7.2.3.2.1 General Description

The ICS includes four independent subsystems, as shown in Figure 7-11. The four subsystems are the Unit Load Demand, the integrated master control, the steam generator control, and the reactor control. The system philosophy is that control of the plant is achieved through feed-forward control from the unit load demand. The unit load demand produces demands for parallel control of the turbine, reactor, and feedwater system through respective subsystems.

The steam generator control is capable of automatic or manual feedwater control from startup to full output. The integrated master control is capable of automatic or manual turbine valve control from minimum turbine load to full output and of manual control below minimum turbine load.

The reactor control subsystem of the ICS is designed for manual operation when steam generators are low level limited and for automatic or manual operation when steam generator levels are allowed to vary.

The basic function of the ICS is matching megawatt generation to unit load demand. The ICS does this by coordinating the steam flow to the turbine with the rate of steam generation. To accomplish this efficiently, the following basic reactor/steam generator requirements are satisfied.

A. The ratios of feedwater flow and Btu input to the steam generator are balanced as required to obtain desired steam conditions.

B. Btu input and feedwater flow are controlled:

   1. To compensate for changes in fluid and energy inventory requirements at each load.

   2. To compensate for temporary deviations in feedwater temperature resulting from load change, feedwater heating system upsets, or final steam pressure changes.

7.2.3.2.2 Unit Load Demand Control

The Unit Load Demand (ULD) is designed to accomplish two objectives related to the operation of the plant. First, the ULD conditions the load demand signal to make it compatible with the power level of the plant and its ability to change load. Second, the ULD initiates load limiting and runback functions to restrict operation within prescribed limits. Figure 7-12 illustrates the functions incorporated in the subsystem.

The ULD obtains a load demand signal from the operator or from the plant computer. The load demand is restrained by a maximum load limiter, a minimum load limiter, and a rate limiter.

Rate limiting is designed as a function of load, so transients are limited as shown in Table 7-6.

The maximum load and rate limiters also act to run back and/or limit the load demand under any of the following conditions.

A. Loss of any number of coolant pumps; runback at 50 percent per minute to the power corresponding to the remaining pumping capability.

B. Loss of one feedwater pump or two out of three condensate pumps; runback at 50 percent per minute to the remaining pump capability.

C. Asymmetric rod withdrawal pattern exists in reactor; runback at 30 percent per minute to 40 percent of rated power.

The output of the limiter is a megawatt demand signal which is applied to the turbine control, steam generator feedwater control, and reactor control in parallel.

The controlling subsystems of the ICS (turbine control, steam generator feedwater control, and reactor control) normally operate in the automatic mode in response to a demand signal from the ULD. The subsystems control function is kept within pre-established bounds under other than normal automatic operation by a "load tracking" feature built into the ICS. The system will switch to the load tracking mode if any of the following conditions exists.

A. One or more of the subsystems are in manual.

B. Errors greater than preset limits develop between the reactor control and steam generator feedwater control.

C. Errors greater than preset limits develop between the demand and the variable in the turbine control subsystem.

D. A reactor trip occurs.

E. A turbine trip occurs.

F. The generator separates from the 500 kV bus.

In this mode, the load demand is made to follow the manual or limited control subsystem by using the actual generator output as the demand input to the ULD. Load tracking continues until the limiting condition is brought back to within the pre-established deadband or the subsystem is returned to automatic operation, at which time control is transferred back to the ULD.

7.2.3.2.3 Integrated Master Control

The integrated master control has been designed to receive the megawatt demand signal from the ULD subsystem and convert this signal into a demand for the feedwater, turbine, and reactor control. A functional diagram of the integrated master control is shown in Figure 7-13. The megawatt demand is compared with the generator megawatt output and the resulting megawatt error signal is used to change the steam pressure setpoint. The turbine valves then change position to control steam pressure. As the megawatt error reduces to zero, the steam pressure setpoint is returned to the steady-state value. By limiting the effect of megawatt error on the steam pressure setpoint, the system can be adjusted to permit controlled variations in steam pressure to achieve the desired rate of turbine response to megawatt demand.

ULD is also utilized as the feed-forward demand to the steam generator and reactor while operating in the integrated control mode. This demand is compensated for deviations in the steam header pressure from its setpoint. The pressure error increases the steam generator and reactor demands if the pressure is low. It decreases the steam generator and reactor demands if the pressure is high.

Before the main turbine is placed in operation, steam generator pressure is controlled by the turbine bypass system with pressure error for each steam generator based on that generator's outlet pressure. As the throttle valves are opened during turbine startup, turbine bypass control is transferred from the outlet pressure signals to selected turbine header pressure.

Once the turbine is on line, pressure control is transferred from the bypass system to the turbine. The turbine bypass setpoint is then biased up, with the bypass system providing pressure relief for the turbine header.

Following a reactor trip, the pressure setpoint is biased up an additional amount to limit reactor coolant temperature and pressurizer level reduction. Steam generator outlet pressures are used by the bypass system for control after a reactor trip.

7.2.3.2.4 <u>Steam Generator Control</u>

Control of the steam generator is based on matching feedwater flow to the feedwater demand produced in the integrated master control. Figure 7-14 illustrates the steam generator feedwater controls.

The basic control actions for parallel steam generator operation are:

A. Steam pressure compared to set pressure, and the pressure error applied to megawatt demand.

B. Megawatt demand converted to feedwater demand.

C. Total feedwater flow demand split into feedwater flow demand for each steam generator.

D. Feedwater demand compared to feedwater flow for each steam generator. The resulting error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

At low power levels, the steam generator control acts to maintain a preset minimum downcomer water level. The conversion to level control is automatic and is introduced into the feedwater control train through an auctioneer.

The steam generator control also provides ratio, limit alarm, and runback actions, as shown in Figure 7-14, which include:

A. Steam Generator Load Ratio Control

Under normal conditions the steam generators will each produce one-half of the total load established by a simulated balanced reactor coolant flow signal. Steam generator load ratio control is provided to balance reactor inlet coolant temperatures during operation with more reactor coolant pumps in one loop than in the other. Under this condition, the simulated balanced reactor coolant flow is replaced with the actual reactor coolant flow signals.

B. Water Level Limits

A maximum water level limit prevents overpumping of feedwater and assures superheated steam under all operating conditions.

A minimum water limit in the downcomer section is provided for low load control.

C. Reactor Coolant Flow Limiters

These limiters restrict feedwater demand to match reactor coolant pumping capability. For example, if one Reactor Coolant Pump is not operating, the maximum feedwater demand to the steam generator in the loop with the inoperative pump is limited to one-half normal.

D. Btu Limit Alarm

The Btu limit calculation provides an alarm to notify the operator to reduce feedwater demand. The Btu limit is a maximum feedwater demand calculated as a function of reactor outlet temperature, feedwater temperature, reactor coolant flow, and steam generator pressure for the loop.

E. Feedwater Cross Limits

A feedwater demand signal is limited to maintain the feedwater demand always within five percent of the reactor power. Feedwater demand is limited to within about five percent of the reactor power demand both in the increase and decrease feedwater demand directions.

F. Feedwater Valve and Pump Control

At lower power levels, steam generator level is maintained at a low level limit through modulation of the startup and low load feedwater control valves. The control valves are sequenced in operation so that the startup valve opens first followed by the low load valve. Differential pressure across the control valves is maintained through modulation of feedpump speed.

As reactor power and feedwater demand increase, feedwater demand eventually exceeds the flow required to maintain minimum level, at which point valve control transfers from steam generator level to steam generator flow control.

When loop feedwater demand increases to 50%, the main feedwater block valve opens, the startup and low load valve demands are held constant, and the main feedpump transfers from controlling valve differential pressure to directly controlling feedwater flow.

This sequence is reversed for power reductions except that the block valve closes at 45% Loop Feedwater demand.

In the event of a trip of a single feedwater pump, power demand will automatically run back to approximately 40 percent of unit load demand. The main feedwater cross-tie valve immediately opens. When reactor power is greater than 80%, ICS will subtract a

bias from the reactor demand signal. The magnitude of this bias is a function of reactor power, and it varies from a 30% subtraction at 100% power to 0% at 80% power. This will cause immediate insertion of rods and crosslimit Feedwater demand to increase demand to the remaining feedpump. The main feedwater block valves will remain open until reactor power drops below 80 percent. These features help to limit RCS pressure increase upon a loss of a main feedpump, thereby reducing the probability of a reactor trip due to high RCS pressure.

G. Rapid Feedwater Runback

A Rapid Feedwater Runback (RFR) is provided to improve the control of feedwater after a reactor trip in order to minimize reduction in pressurizer pressure and level.

Upon reactor trip the following occurs:

A. A zero feedwater demand is introduced into each feedwater loop string to aid in coordinating control actions during recovery following RFR action.

B. The RFR runs the feedwater pump turbine speed demand down to minimum speed until the steam generator startup level drops to 45 inches at which point they are restored to auto control for runup to the feedwater pump speed required for feedwater valve ΔP control. Return to ΔP control in a loop is prevented until its main block valve is closed.

C. The low load and startup feedwater valves are forced closed by a negative feedwater error signal applied to both feedwater loop strings. If an EFIC MSLI signal is not present, the negative error terminates when the level in the steam generator drops to 40 inches. The valves remain shut until level reaches the low level setpoint, at which point the startup valve assumes control of steam generator level. If an EFIC MSLI signal is present, the valves receive a maintain close command from ICS which provides a backup to the main feedwater isolation valve closure.

7.2.3.2.5 Reactor Control

The reactor control is designed to maintain a constant average reactor coolant temperature over the load range from approximately 22 to 100 percent of rated power. The steam system operates at an approximately constant pressure at all loads. The average reactor coolant temperature decreases over the range from approximately 22 percent to zero load. Figure 4-9 shows the nominal reactor coolant and steam temperatures and the steam pressure over the entire load range.

The reactor control consists of analog computing equipment with inputs of megawatt demand, core power, and reactor coolant average temperature. The output of the controller is an error signal that causes the control rod drive to be positioned until the error signal is within a deadband. A block diagram of the reactor control is shown on Figure 7-16.

First, reactor power level demand (N_d) is computed as a function of the megawatt demand (MW_d) and the RCS average temperature deviation ($\bar{\Delta}T$) from the setpoint, according to the following equation:

$$N_d = K_1 MW_d + K_2(\bar{\Delta}T + \frac{1}{\tau} \int \bar{\Delta}T\, dt)$$

Megawatt demand is introduced as a part of the demand signal through a proportional unit having an adjustable gain factor (K_1). The temperature deviation is introduced as a part of the demand signal after proportional plus reset (integral) action is applied. For the temperature deviation, K_2 is the adjustable gain and τ is the adjustable integration factor.

The reactor power level demand (N_d) is then compared with the reactor power level signal (N_i), which is derived from the nuclear instrumentation. The resultant error signal (N_d - N_i) is the reactor power level error signal (E_N).

When the reactor power level error signal (E_N) exceeds the deadband settings, the control rod drive receives a command that withdraws or inserts rods depending upon the polarity of the power error signal.

The following additional features are provided with the reactor power controller:

A. A high limit on reactor power level demand (N_d).

B. A low limit on reactor power level demand (N_d).

C. A megawatt demand limit imposed by lack of feedwater flow capability from the steam generator controls.

The reactor control incorporates automatic or manual rod control above approximately 22 percent of rated power and manual rod control below approximately 22 percent of rated power.

The reactor control subsystem also generates the following interlock signals:

A. A signal to the RDCS to prevent placing the rod drive controls in the automatic mode if a large error exists in the ICS.

B. A signal to the RDCS to cause the rod drive controls to revert to the manual mode if power for automatic operation of the ICS is lost.

C. A signal to the RDCS indicating that reactor power is greater than 40 percent which is used to generate the "Out Inhibit" signal.

D. A signal to the Reactor Coolant Pump motor controls which prevents starting an idle pump when reactor power is greater than 22 percent.

E. A signal to an ICS module to subtract a bias from the reactor demand signal which will vary from 30% at 100% power to 0% at 80% reactor power when a main feedpump is tripped and reactor power is greater than 80%.

F. A signal to prevent the main feedwater block valves from closing when a main feedpump is tripped with reactor power greater than 80%.

G. A signal to the pressurizer pressure controls to lower pressurizer spray setpoint when a main feedpump is tripped with reactor power greater than 80%.

Features E., F., and G. help to limit RCS pressure increase upon a loss of a main feedpump, thereby reducing the probability of a reactor trip due to high RCS pressure.

7.2.3.3 System Evaluation

7.2.3.3.1 System Failure Considerations

Redundant sensors for major system parameters are available to the ICS. Automatic signal selection of the good signal protects the inputs in the event of a sensor failure. The operator can manually select any of the redundant sensors from the control room. Manual reactivity control is available at all power levels. Loss of electrical power to automatic control stations reverts the control system to manual. The unit master is designed to minimize the impact of hardware or software failures on the rest of the ICS.

7.2.3.3.2 System Limits

Maximum and minimum limits on the reactor power level demand signal (N_d) prevent the automatic reactor controls from initiating undesired power excursions.

Maximum and minimum levels on the megawatt demand signal (MW_d) prevent the reactor controls from initiating undesired power excursions.

Cross limiting between the steam generators and the reactor prevents reactor power excursions that may result in a reactor trip from reactor coolant pressure or temperature.

7.2.3.3.3 Modes of Control

The ICS is designed to revert to a "Load Tracking" mode of control to tie the unit to the subsystem on manual or to the subsystem being limited.

In startup control mode, the controls are arranged so that the steam system follows reactor power rather than turbine system power demand. The controls will limit steam bypass to the condenser when condenser vacuum is inadequate.

The ICS provides a fully automatic control mode based on core thermal power (from the plant computer). This mode provides fine control (± 1% of setpoint maximum) for maintaining unit load near 100%.

7.2.3.3.4 Loss-of-Load Considerations

The nuclear unit is designed to accept a 10 percent step load rejection without safety valve action or turbine bypass valve action. The controls will prevent steam dump to the condenser when condenser vacuum is inadequate, in which case the safety valves may operate.

The features that permit continued operation under load rejection conditions include:

A. Integrated Control System

During normal operation the ICS controls the unit load in response to load demand from the operator. During normal load changes and small frequency changes, turbine control is through the speed changer to maintain constant steam pressure.

During large load and frequency upsets, the turbine governor takes control to regulate frequency.

B. 100 Percent Relief Capacity in the Steam System

This provision acts to reduce the effect of large load drops on the reactor system.

Consider, for example, a sudden load rejection greater than 10 percent. When the turbine generator starts accelerating, the governor valves and the intercept valves begin to close to maintain set frequency. In addition, the megawatt demand signal is reduced, which reduces the governor speed changer setting, feedwater flow demand, and reactor power level demand. As the governor valves close, the steam pressure rises and acts through the control system to reinforce the feedwater flow demand reduction already initiated by the reduced megawatt demand signal. In addition, when the load rejection is of sufficient magnitude, the turbine bypass valves open to reject excess steam to the condenser and the safety valves open to exhaust steam to the atmosphere. The rise in steam pressure and the reduction in feedwater flow cause the average reactor coolant temperature to rise which reinforces the reactor power level demand reduction, already established by reduced megawatt demand, to restore reactor coolant temperature to the set value.

As the turbine generator returns to set frequency, the turbine controls revert to steam pressure control rather than frequency control. This feature holds steam pressure within relatively narrow limits and prevents further large steam pressure changes.

7.2.4 ATWS - DIVERSE REACTOR OVERPRESSURE PREVENTION SYSTEM (DROPS)

7.2.4.1 Design Bases

The DROPS is designed in compliance with 10 CFR 50.62, known as the ATWS Rule. The DROPS is diverse and independent from the existing Reactor Trip System and will trip the reactor, trip the turbine, and initiate Emergency Feedwater during conditions indicative of an ATWS event. An ATWS event is any RCS overpressurization concurrent with a common mode RPS failure that could lead to the over-stressing of RCS components beyond their ASME Section III, Service Level C stress limits, thereby resulting in a challenge to the RCS integrity. For ANO-1, the RCS overpressure limit for an ATWS event has been established to be 3250 psig.

The design bases for the DROPS are categorized into Safety Considerations, Operational Considerations, and Start-up Considerations.

7.2.4.1.1 Safety Considerations

A. The DROPS Diverse Scram System (DSS) will remove the redundant power to the Control Rod Drive Control System programmable controller and gate drives for control rod groups 5, 6, and 7 and the auxiliary group for conditions indicative of an Anticipated Transient Without Scram (ATWS).

B. The DROPS ATWS Mitigation System Actuation Circuitry (AMSAC) will trip the turbine and initiate Emergency Feedwater (EFW) upon the loss of Main Feedwater (MFW) flow when reactor power is greater than 45 percent.

7.2.4.1.2 Operational Considerations

A. A single failure will not cause a spurious actuation.

B. The DROPS provides Control Room annunciation of system actuation, testing, bypassing, and problems.

C. An automatic turbine trip will be initiated upon a DSS reactor trip.

7.2.4.1.3 Start-up Considerations

A. The DROPS AMSAC trip is automatically bypassed when the plant is below 45 percent power. This bypass is automatically removed when the plant is above 45 percent power.

B. A start-up bypass for the DSS trip is not required since this trip is initiated by high RCS pressure.

7.2.4.2 System Design

The DROPS is a two channel logic system which monitors plant process parameters and compares them to pre-selected trip values to determine if conditions exist that are indicative of an ATWS event. The monitored parameters are diverse and independent from those which are used by the RPS. The parameters monitored are Wide Range RCS Pressure, Linear Reactor Power, and MFW Flow for Loop A and Loop B.

The DROPS generates a DSS reactor trip when the Wide Range RCS Pressure exceeds its setpoint of 2430 psig. Each DSS channel trip will remove 24 VDC power from one of two CRDCS programmable controller channels for control rod groups 5, 6, and 7 and the auxiliary group. Loss of both programmable controller channels is required to trip the reactor, thus providing the 2-out-of-2 logic. In addition, a 2-out-2 logic is provided to automatically trip the turbine upon a DSS reactor trip.

The AMSAC turbine trip and EFW initiation signals are generated when MFW flow is less than 15 percent in both loops and when reactor power is greater than 45 percent. The turbine trip signals are summed in the existing turbine trip circuitry and upon receipt of both DROPS channel signals, the auto-stop oil trip solenoid and the auto-stop back-up trip solenoid will be energized, either of which will trip the turbine. The summation of the two DROPS channel signals provides for a 2-out-of-2 logic actuation. The EFW initiation signals are sent to the EFIC System with DROPS Channel 1 actuating the Anticipatory Trip of EFIC Channel A and DROPS Channel 2 actuating the Anticipatory Trip of EFIC Channel D. Actuation of EFIC Channels A and D will result in full EFW initiation.

The DROPS includes a battery back-up module for each channel which will provide a minimum of 15 minutes of continued operation in the event of a loss of the 120 VAC power source.

The DROPS may be tested at power by the use of front panel controls. The system test will remove the process inputs and internally simulate the inputs. The setpoints and the trip signals can be monitored at the front panel for verification during testing. Since the DROPS is a 2-out-of-2 logic system, both channels will be disabled to avoid spurious operation while testing. Channel testing and bypassing will be alarmed in the Control Room.

7.2.4.3 System Evaluation

The DROPS is a highly reliable 2-out-of-2 logic system designed to prevent spurious actuation due to a single failure. The system input parameters are diverse and independent from those used by the RPS. The DROPS provides signals diverse from the existing Reactor Trip System to trip the reactor, trip the turbine, and initiate EFW. The system has a minimum period of operation of 15 minutes after a loss of the 120 VAC power source.

7.3 INSTRUMENTATION

7.3.1 NUCLEAR INSTRUMENTATION

The Nuclear Instrumentation System is shown in Figure 7-17. The system meets the intent of the proposed IEEE 279, "Criteria for Nuclear Power Plant Protection Systems," dated August 1968, for those elements associated with the Reactor Protection System (RPS).

7.3.1.1 Design Basis

The Nuclear Instrumentation (NI) System is designed to supply the reactor operator with neutron information over the full operating range of the reactor and to supply reactor power information to the RPS and to the Integrated Control System (ICS).

The system sensors and instrument strings are redundant in each range of measurement. Measurement ranges are designed to overlap providing complete and continuous information over the full operating range of the reactor.

7.3.1.2 System Design

The NI has eight channels of neutron information divided into three ranges of sensitivity: source, intermediate, and power range. The three ranges combine to give a continuous measurement of reactor power from source level to approximately 125 percent of rated power, or 10 decades of information. A minimum of one decade of overlapping information is provided between successive ranges of instrumentation. The relationship between instrument ranges is shown in Figure 7-18.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity fission chambers. These channels are used over a counting range of 0.1 to 10^5 counts/sec and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -1 to +7 decades/min. An interlock is provided, i.e. a control rod withdraw inhibit, on a high startup rate of +2 decades/min in either channel.

The intermediate range instrumentation has two log N channels originating in two electrically identical gamma-compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from 10^{-11} to 10^{-3} amperes. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 to +5 decades/min. A high startup rate of +3 decades/min in either channel will initiate a control rod withdraw inhibit.

The power range instrumentation has four linear level channels originating in four detector assemblies with each assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and the bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the power imbalance of the reactor core. The channel outputs are directly proportional to reactor power and covers the range of 0 to 125 percent for the total power and -62.5 to +62.5 percent for the power imbalance. The gain of each channel is adjustable, providing a means of calibrating the output against a reactor coolant heat balance.

The four power range channels, NI-5, 6, 7, and 8, supply reactor power level and imbalance information continuously to the RPS. Isolation amplifiers are used to buffer every signal leaving the system cabinets. The isolation amplifiers prevent the reflection of faults on external signal lines back into the systems (Section 7.1.2.3.3).

The neutron flux signals from the power range channels go through individual isolation buffer amplifiers. Power range channels NI-5 and NI-6 power signals are fed to a high auctioneer and power range channels NI-7 and NI-8 power signals are fed to a separate high auctioneer. The output signals of the auctioneers are taken from the isolation amplifiers, which are similar to the isolation buffer amplifiers. The isolated outputs are fed to the NNI Smart Automatic Signal Selector (SASS). The SASS selects the preferred signal channel as the input signal, to the ICS and power range recorder. In the event of a failure of the selected signal the good signal is automatically selected.

Destructive tests of the buffer amplifiers used as isolation devices establish that they will block the passage to the input of ±400-volt DC or AC (peak to peak) when applied to their output. Other tests demonstrate that the output of any isolation buffer amplifier can be open, shorted, or grounded in any manner without effect upon the input.

The auctioneer isolation amplifier has similar isolation characteristics to the isolation buffer amplifiers feeding it. Both the auctioneer isolation amplifier and isolation buffer amplifier will prevent faults occurring at the output of one amplifier from propagating back into the two power range channels. Shorts, opens, or grounds at the amplifier output cannot affect the power range signals. Calculations show that a voltage on the order of ±10,000 volts AC (peak to peak) applied across the output of the amplifier would be prevented from propagating back to the signal source. Therefore, the signal line leaving the RPS and going to the control system is adequately isolated from the power range channels so that no credible fault can affect the power range channels.

Since the isolation and auctioneer modules are active amplifiers, no fault which would destroy or degrade their isolation capability can occur without a pronounced effect upon their output which would be detected when routine on-line system tests were conducted on a single protection channel.

This portion of the reactor protection system complies with Section 4.7.4 of IEEE 279-1971 by employing a diversity of variables. Reactor coolant pressure is the variable diverse from neutron flux which meets the requirements of Section 4.7.4.1 of IEEE 279-1971.

7.3.1.2.1 Neutron Detectors

The original source range channels consisted of a BF_3 proportional counter, a preamplifier, a log count rate amplifier, a rate-of-change amplifier and associated power supplies. These detectors provide no safety function. Two Gamma-Metrics source range channels were added during 1R6. These new channels are based on Gamma Metrics fission chambers. These fission chambers were installed in the existing spare excore wells. Each Gamma Metrics channel consists of a fission chamber detector, cable junction box, signal amplifier, signal processor, and various displays. The channels provide neutron flux level information in three different ranges from .1 cps to 200% of full power. A rod withdrawal interlock inhibits control rod out commands if SR start up rate exceeds 2 dpm. This interlock is bypassed by IRM and PRM contact at 10^{-9} amps or 10% power.

The intermediate range compensated ion chambers are of the electrically adjustable gamma-compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

Uncompensated ion chambers are used in the power range channels. Each power range detector consists of two 72-inch sections with a single high voltage connection and two separate signal connections. The outputs of the two sections are summed and amplified by the linear amplifiers in the associated power range channel. A signal proportional to the difference in percent full power between the top and bottom halves of the core is derived from the difference in currents from the top and bottom sections of the detector. The difference signal is displayed on the control console to permit the operator to maintain proper axial power distribution. The manual test and calibration facilities provide a means for reading the output of the individual sections of the detector. Each detector has a combined sensitive volume extending approximately from the bottom to the top of the reactor core.

The physical locations of the neutron detectors are shown in Figure 7-19. A power range detector is located external to each quadrant of the reactor core. The source range detectors are located on opposite sides of the core, 180 degrees apart. The two intermediate range detectors are also located on opposite sides of the core, but rotated approximately 55 degrees from the source range detectors.

Table 7-7 provides pertinent characteristics of the out-of-core neutron detectors. The flux ranges illustrated in Figure 7-18 are compatible with these characteristics. Identical type out-of-core detectors have been or are being used at power reactors listed in Table 7-8.

7.3.1.2.2 <u>Test and Calibration</u>

Test and calibration facilities are built into the system. The facilities permit an accurate calibration of the system and the detection of system failures in accordance with the requirements of the RPS design and IEEE 279-1968.

7.3.1.3 <u>System Evaluation</u>

The nuclear instrumentation will monitor the reactor over a minimum 10-decade range from source to 125 percent of rated power. The system design full power neutron flux level at the power range detectors was determined to be approximately 3.2×10^9 nv for initial operation. The actual value of neutron flux will vary as a function of core life and core design. The detectors employed will provide a linear response up to approximately 2.5×10^{10} nv before they are saturated.

The intermediate range channels overlap the source range and the power range channels as shown in Figure 7-18, providing the continuity of information needed during startup.

The radial flux distribution within the reactor core will be measured by the incore neutron detectors described in Section 7.3.3. The out-of-core detectors may also be used to indicate radial flux tilts. Both out-of-core detectors will be used to obtain the axial power distribution. The sum of the outputs from the two sections of each power range detector will be calibrated to a heat balance. The sum will be recalibrated whenever it disagrees with the heat balance by two percent or more, except during the power escalation test program and up to the commercial operation date plus three months, when NI calibration is required on a deviation of +4, -2 percent between the NI indication and the heat balance.

The difference signal will be unaffected by calibration of the sum. Periodically the operator will compare the difference indication from the power range channels with the difference obtained from the incore detectors.

7.3.1.3.1 Primary Power

The NI draws its primary power from the vital buses as described in Section 8.3.1.1.6. Redundant channels are supplied by different vital buses.

7.3.1.3.2 Reliability and Component Failure

The requirements established for the RPS apply to the NI. All channel functions are independent of every other channel and, where signals are used for safety and/or control, electrical isolation is employed to meet the criteria of Section 7.1.1.

7.3.1.3.3 Relationship to Reactor Protection System

The relation of the NI to the RPS is described in Section 7.1. Each power range channel provides level information to a different RPS channel. The higher powers of Channels NI-5 and NI-6 and channels NI-7 and NI-8 are continuously monitored by SASS and one of these signals is used to supply information to the ICS.

7.3.2 NON-NUCLEAR PROCESS INSTRUMENTATION

7.3.2.1 Design Bases

The non-nuclear process instrumentation provides the required input signals of process variables for the reactor protection, regulating, and auxiliary systems. It performs the required process control functions in response to those systems and provides instrumentation for startup, operation and shutdown of the reactor system under normal and emergency conditions.

7.3.2.2 System Design

The Non-Nuclear Instrumentation provides measurements used to indicate, record, alarm, interlock, and control process variables such as pressure, temperature, level, and flow in the Reactor Coolant System (RCS), secondary system, and auxiliary reactor systems as shown in Figures 7-21 and 7-22. Process variables required on a continuous basis for the startup, operation and shutdown of the unit are indicated, recorded and controlled at the control room. Alternate essential indicators are provided at other locations to maintain the reactor in a hot shutdown condition if the control room has to be evacuated.

Significant criteria for the design of Non-Nuclear Instrumentation (NNI) include:

A. Instrumentation supplying information to protection systems complies with IEEE 279-1968.

B. Response time and accuracy of measurements are adequate for reactor protection and regulating systems and other control functions to be performed.

C. Instrumentation in the protection systems is provided to operate as required under the environmental conditions specified in Section 7.1.

D. Whenever both precise control and the measurement of wide process ranges are required, wide range and narrow range instrumentation are provided.

E. Process parameters used by the ICS during power operation are derived from selectable redundant transmitters.

F. Recorders are used where it is desirable to monitor process trends.

7.3.2.2.1 Non-Nuclear Process Instrumentation in Protection Systems

Four independent measurement channels are provided for each process parameter for input to the RPS.

Three independent measurement channels are provided for each process parameter for input to the ESAS.

A. Reactor Outlet Temperature

Reactor outlet temperature inputs to the RPS are provided by two fast-response resistance elements and associated transmitters in each loop.

B. Reactor Coolant Flow

Reactor coolant flow inputs to the RPS are provided by eight high-accuracy differential pressure transmitters which measure flow through calibrated flow tubes. Operation of each Reactor Coolant Pump breaker is also monitored as an indication of flow.

One outlet temperature compensated differential pressure transmitter signal for each loop is utilized as the normal flow measurement providing input to the ICS.

C. Reactor Coolant Pressure

RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each loop. One pressure transmitter signal will be utilized for pressurizer pressure control.

ESAS inputs of reactor coolant pressure in each loop are provided by redundant pressure transmitters with two transmitters in loop A and one in loop B. One pressure signal is utilized for high pressure alarm, low pressure alarm, and interlock to decay heat removal suction isolation valves.

D. Reactor Building Pressure

Reactor building pressure input to the RPS is provided by four independent pressure transmitters driving electronic switches to provide switch contacts for the RPS. Reactor building pressure input to the ESAS is provided by three redundant pressure transmitters.

Tables 7-9 and 7-10 provide pertinent information concerning the NNI sensors supplying inputs to the RPS and ESAS, respectively.

7.3.2.2.2 Non-Nuclear Process Instrumentation in Regulating Systems

Selective redundant measurements and input signals are provided for the process variables required for critical control functions. To insure reliability of the signals, the redundant measurements are monitored by the Smart Automatic Signal Selector (SASS). The SASS provides the capability of manually selecting one of the redundant signals from the control room for use by the ICS or allowing automatic selection. In the automatic mode the preferred signal is selected and fed to the ICS for control and is displayed in the control room. The redundant signal is fed to the plant computer. In the event of a failure of the preferred signal, the SASS transfers the redundant (good) signal to the ICS for control and display.

The following inputs to the ICS are provided:

A. Reactor Outlet Temperatures

   Selected loop or unit average outlet temperature input is provided in each loop by two fast-response resistance elements and associated transmitters.

B. Reactor Controlling Average Temperature

   Instrumentation separate from the RPS supplies input to the ICS.

   Loop or unit average temperature signals are selected for indication and input as controlling average temperature. Automatic selection determined by loop flows is provided for input of the appropriate signal.

   Reactor inlet temperature signals required for loop and unit average and differential temperatures, are provided in each loop by two fast response resistance elements and associated transmitters.

C. Reactor Inlet Differential Temperature

   Reactor inlet differential temperature is indicated and provided for input to the ICS.

D. Reactor Coolant Flow

   Reactor coolant flow signals are provided for each loop and summed for total flow. Total flow is recorded and "low" total flow is alarmed.

   Flow measurement of each loop is provided by selection from two redundant loop flow transmitters provided within the RPS.

   Loop "low" flow signals provide the logic for automatic selection of reactor controlling average temperature.

   Contacts from Reactor Coolant Pump motor breakers provide fast indication to the ICS that a pump has tripped.

E. Feedwater Temperature

Feedwater temperature input from each loop is provided by two resistance elements and associated transmitters. The selected input also provides signals for indication and feedwater temperature compensation.

F. Feedwater Flow

Feedwater flow input is provided from startup and redundant main feedwater flow transmitters for each loop. The main feedwater flow measurement in each loop is provided by redundant differential pressure transmitters that measure flow through a flow nozzle. Startup feedwater flow measurement in each loop is provided by a differential pressure transmitter that measures flow through a flow nozzle. Startup feedwater signals are indicated and the full-range flow signal from the ICS is recorded for each loop.

G. Steam Generator Level

Selected "startup" level and "operate" level inputs are provided from each steam generator. Startup level indication is used during shutdown and low power operations while the operate level indications are used during normal power operations. Redundant measurements of each level are provided by differential pressure transmitters. Temperature compensation for "operate" level to augment the predetermined compensation for normal operating temperature is provided by two fast response resistance elements per generator and associated transmitters which measure steam generator lower downcomer temperature.

The selected "operate" level input is recorded and "high" level alarmed. The selected "startup" level input is indicated and "low" level alarmed. A redundant indicator is provided for "startup" level in each loop for essential indication.

A full range level measurement is provided for indication of each steam generator level but does not provide protective or regulating systems input.

H. Steam Generator Outlet Pressure

Selected outlet pressure input is provided from each steam generator. Measurement is made by pressure transmitters in both outlet lines of each steam generator. The selected input is also indicated.

I. Turbine Header Pressure

Turbine header pressure measurement is provided for input by a pressure transmitter in each header line from the steam generators. The selected pressure signal is also recorded and high and low pressures alarmed.

7.3.2.2.3 Other Non-Nuclear Process Instrumentation

The following instrumentation is provided for measurement and control of process variables necessary for proper reactor operation.

A. Pressurizer Temperature

Pressurizer temperature is measured by two resistance elements. Each signal is sent to a separate cabinet for buffering and isolation. Outputs from each cabinet's isolators are sent to that cabinet's temperature level compensation network, the SPDS computer, and the NNI system. One of the NNI signals is manually selected by the operator to provide front panel indication. The non-selected temperature goes to the plant computer.

B. Pressurizer Level Control

Pressurizer level is measured by two differential pressure transmitters. Each signal is sent to a separate cabinet for buffering and isolation. Outputs from each cabinet's isolators are sent to that cabinet's temperature level compensation network and the SPDS computer. The compensated level signal from each channel is sent to the NNI. One of the NNI signals is selected for level control, alarms, and interlock to de-energize the pressurizer electric heaters on low level. The level controller output positions the makeup control valve in the Makeup and Purification System to maintain a minimum preset level. Pressurizer level is lowered by reactor coolant letdown manually controlled at the control room. Channel number 1 (red) compensated pressurizer level is recorded on the control panel. Channel number 2 (green) compensated pressurizer level is indicated on the control panel. Both channels provide high and low alarming in the control room.

C. Reactor Coolant Pressure Control

The reactor coolant pressure signal used for pressure control is also monitored by SASS, which selects from one of two pressure transmitters used in the RPS. The signal selected is used as input for automatic control of:

1. Pressurizer electric heaters.

2. Pressurizer spray control valve.

3. Pressurizer electromatic relief valve.

The heaters are grouped in banks which are energized below preset pressures.

The spray and relief valves are opened above preset pressures. In addition, the electromatic relief valve has a dual setpoint feature to prevent reactor vessel overpressurization as specified in the Technical Specifications. The selected signal also provides input to a pressure controller which automatically modulates the output of two banks of heaters to maintain a preset pressure. One pressure transmitter signal is recorded and high and low pressure alarms are provided for each loop. (1CAN127601)

Provisions are included to help insure that, upon loss of power, the electromatic relief valve and pressurizer spray valve will remain closed. (1CAN038012)

D. Reactor Coolant Pump Start Interlocks

Interlock signals are provided to the Reactor Coolant Pump switching logics to prevent: (a) starting a pump without seal injection flow, (b) starting a pump without cooling water, and (c) starting the fourth pump until a preset reactor coolant inlet temperature is reached.

Although a pump interlock system prevents startup of a Reactor Coolant Pump in an idle loop when operating above 22 percent power, the cold water accident is analyzed without consideration of pump interlock operation. The most conservative accident in this case is stated in Section 14.1.2.5.3 and is the startup of two Reactor Coolant Pumps when the plant is operating with two pumps at 60 percent of rated power and 49 percent flow. As shown in Section 14.1.2.5.4, the reactor protection criteria of a minimum Departure from Nucleate Boiling Ratio (DNBR) greater than 1.3 and a RCS pressure less than code pressure limits are satisfied in the analysis of this accident. Thus, the pump interlock system, although it is useful in preventing a cold water accident, is not necessary for reactor protection and does not need to meet the requirements of IEEE 279.

E. Feed and Bleed Control

The feed and bleed control instrumentation in the Makeup and Purification System provides control and interlocks to permit continuous letdown of reactor coolant and makeup to the system to adjust the reactor coolant boron concentration.

7.3.2.3 System Evaluation

The quantity and types of process instrumentation have been selected to provide assurance of safe and orderly operation of all systems and processes over the full operating range of the plant. Some of the criteria for design are:

A. Separate instrumentation has been provided for the protective systems and vital control circuits.

B. Time of response and accuracy of measurements are adequate for protective and control functions to be performed.

C. Where wide process variable ranges are required and precise control is involved, both wide range and narrow range instrumentation are provided.

D. All electrical and electronic instrumentation required for operation is supplied from vital or regulated instrumentation buses.

E. A transfer switch provides an alternate diesel generator backed source of AC power to the NNI upon loss of 120 vital AC inverter power. (1CAN038012)

7.3.3 INCORE MONITORING SYSTEM

7.3.3.1 Design Basis

The Incore Monitoring System provides neutron flux detectors to monitor core performance. Incore, self-powered neutron detectors measure the neutron flux in the core to provide a history of power distribution during power operation. Data obtained provides power distribution information and fuel burnup data to assist in fuel management. The plant computer provides normal system readout and a backup readout system is provided for selected detectors.

7.3.3.2 System Design

7.3.3.2.1 System Description

The Incore Monitoring System consists of assemblies of self-powered neutron detectors located at 52 positions within the core. The incore detector locations are shown in Figure 3-60. In this arrangement, an incore detector assembly consisting of seven local flux detectors and one background detector is installed in the instrumentation tube of each of 52 fuel assemblies as shown in Figures 3-66A and 3-66B. The local detectors are positioned at seven different axial elevations to provide the axial flux gradient. The outputs of the local flux detectors are referenced to the background detector output so that the differential signal is a true measure of neutron flux.

Readout for the incore detectors is performed by the plant computer. Multi-point recorder readouts of selected detectors are provided independent of the computer.

When the reactor is depressurized, the incore detector assemblies can be inserted or withdrawn through guide tubes which originate in the incore instrument tank located in a shielded area in the reactor building as shown in Figure 7-24. These guide tubes enter the bottom head of the reactor vessel where internal guides extend up to the instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube serves as the guide for the incore detector assembly. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new locations, the incore detector assemblies are returned to their fully inserted positions.

7.3.3.2.2 Calibration Techniques

The nature of the detectors permits the manufacture of nearly identical detectors, which produces a high relative accuracy between individual detectors. The detector signals are compensated continuously by the plant computer for burnup of the neutron sensitive material.

Calibration of detectors is not required. The self-powered detectors, are controlled to precise levels of initial sensitivity by quality control during the manufacturing stage. The sensitivity of the detector changes over its lifetime due to such factors as detector burnup, control rod positions, fuel burnup, etc. The results of experimental programs to determine the magnitude of these factors have been incorporated into calculations and are used to correct the output of the incore detectors for these factors. Operation of detectors in both power and test reactors has demonstrated that this compensation program, when coupled with the initial sensitivity, provides detector readout accuracies sufficient to eliminate the need for a detector calibration system.

7.3.3.3 System Evaluation

7.3.3.3.1 Operating Experience

Self-powered incore neutron detectors have been operated since 1962. Such detectors have been assembled and irradiated in a Babcock & Wilcox development program that began in 1964.

Historical data removed - To review exact wording please refer to Section 7.3.3.3.1 of the FSAR.

For Incore Monitoring System development program results and conclusions, refer to B&W Topical Report BAW-10001, "In-Core Instrumentation Test Program."

7.3.3.3.2 Detection of Power Distribution

Under normal operating conditions, the incore detectors supply information to the operator in the control room.

Each individual detector measures the neutron flux at its vicinity and is used to determine the local power density. The individual power densities are then averaged and a peak-to-average power ratio calculated. This information can be used to indicate possible power oscillations (see Section 3.2.2.2.1.G).

Historical data removed - To review exact wording please refer to Section 7.3.3.3.2 of the FSAR. Upon initial installation, the self-powered detector has the capability to measure the relative flux with an accuracy of five percent when used in conjunction with an adjacent background detector. The sensitivity of the detector will decrease with exposure to neutron flux due to transmutation of the emitter in the detector. However, by use of integrated current inventories, it is felt that the additional inaccuracies will be no more than one percent per year for the average flux conditions.

The use of the Incore Monitoring System to detect xenon oscillations is described in B&W Topical Report BAW-10010, Part 1, "Stability Margins for Xenon Oscillations Modal Analysis."

7.3.4 OTHER INSTRUMENTATION FOR OPERATOR INFORMATION

7.3.4.1 Post Accident Monitoring Instrumentation

Indications of plant variables are required by the control room operating personnel during accident situations to (1) provide information required to permit the operator to take preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, safety-feature systems, and manually initiated safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining Reactor Coolant System integrity, and maintaining containment integrity); and (3) provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, Reactor Coolant Pressure Boundary, and containment) and to determine if a gross breach of a barrier has occurred. In addition to the above, indications of plant variables that provide information on operation of plant safety systems and other systems important to safety are required by the control room operating personnel during an accident to (1) furnish data regarding the operation of plant systems in

order that the operators can make appropriate decisions as to their use and (2) provide information regarding the release of radioactive materials to allow for early indications of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

7.3.4.1.1 Table Format

Table 7-11A lists the variables committed to by Entergy Operations, Inc. and recommended by Regulatory Guide 1.97, Revision 3. The subsequent columns include the assigned category, range, redundancy, power supply, type of control room display, availability on SPDS, and comments.

The power supply column specified the type of available power as follows:

- 1E - instrument is powered from a qualified 1E power source.
- UPS - instrument is powered from a battery backed uninterruptable power source.
- DG - instrument is powered from a source that is backed by the Emergency Diesel Generators.
- OP - instrument is powered from the normal offsite power source.

The SPDS column specifies whether the variable is available on the SPDS display. The SPDS display is located in both the Unit 1 and Unit 2 control rooms, the Technical Support Center and Emergency Offsite Facility.

Another type of control room display is the Radiological Dose Assessment Computer System (RDACS). The RDACS is a computerized dose projection system which combines effluent release data with real time meteorological data. RDACS terminals are located in the same facilities as the SPDS.

7.3.4.1.2 Definition of Variables

The variables identified in Table 7-11A are divided into five types in accordance with the Regulatory Guide 1.97. The definition for each type of variable is as follows:

Type A Those variables which provide the primary information required to permit the control room operators to take specific manual actions for which no automatic control is provided and that are required for a safety system to accomplish its safety function for design basis accident scenarios. Type A variables are not specified in Regulatory Guide 1.97. They are plant specific and were selected based on a review of Emergency Operating Procedures to identify information essential for the direct accomplishment of specific safety functions. As a result of a review of the ANO-1 Emergency Operating Procedures, the following variables were identified as Type A:

RCS Hot Leg Water Temperature
RCS Pressure
Steam Generator Level
Steam Generator Pressure
Condensate Storage Tank Level
Borated Water Storage Tank Level
Flow in HPI System
Reactor Building Spray Flow
Low Pressure Injection Flow

Type B These variables provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are defined as: reactivity control, core cooling, maintaining Reactor Coolant System integrity, and maintaining containment integrity.

Type C These variables provide information to indicate the potential for breach of the barriers to fission product release. The barriers are defined as: fuel cladding, primary coolant pressure boundary, and containment.

Type D These variables provide information to indicate the operation of individual safety systems and other systems important to safety.

These variables help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Type E These variables provide information for use in determining the magnitude of the release of radioactive materials and for use in assessing the consequences of such releases.

7.3.4.1.3 Evaluation Criteria

As recommended by Regulatory Guide 1.97, each variable type was evaluated based on the importance to safety of the measurement of the specific variable. The criteria are therefore separated into three categories for a graded approach as follows:

Category 1: provides the most stringent requirements and is intended for key variables. Type A, B, and C key variables fall into this category.

Category 2: provides less stringent requirements and applies to instrumentation designated for indicating system operating status. Type D and E key variables fall into this category.

Category 3: provides requirements that will ensure that high quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. This category is also used when the state-of-the-art will not support requirements for higher qualified instrumentation. All backup variables fall into this category.

The specific design and qualification criteria used to evaluate each variable, based on the category classification, are presented below:

Category 1:

Environmental Qualification - Currently installed instrumentation was evaluated to determine if, as a minimum, the equipment meets the requirements of IE Bulletin 79-01B and 10 CFR 50.49. This determination was based on either having actual environmental qualification documentation available or documentation on similar equipment available.

Seismic Qualification - Currently installed instrumentation was evaluated against the seismic qualification criteria used as a basis for the plant operating license. The criteria are described in Section 7.1.1.8. The ANO-1 seismic criteria are synonymous with the requirements for Class 1 equipment as defined in IEEE Standard 344-1971. New instrumentation will be installed in accordance with the criteria specified in the SAR.

Redundancy and Sensor Location - A response of "yes" in the redundancy column indicates that redundant channels are available up to and including any isolation device and that the channels are both electrically independent and physically separate from each other and from non-safety equipment in accordance with IEEE Standard 279-1971. This standard was used as the basis for the ANO-1 operating license and meets the intent but not all the strict requirements for physical separation of redundant channels as defined in Regulatory Guide 1.75. Where applicable, the general sensor location is listed.

Power Supply - All Category 1 instruments are supplied with power from a Class 1E power supply. The ANO-1 Class 1E power system is designed to meet the requirements of IEEE 279-1971, IEEE 308-1971, 10 CFR 50 including Appendices A and B, and Safety Guides 6 and 9.

Quality Assurance - All instrumentation was, and will continue to be, purchased and installed in accordance with the provisions of the NRC approved Quality Assurance Program.

Control Room Display and Recording - Continuous real-time display of at least one channel is provided in the control room. Recording of the instrument readout information is provided for at least one of the redundant channels, although this recording may be "Non-Q." (It may also be used for the continuous display mentioned above, if qualified.) Variables which input to the SPDS may be displayed and/or trended on demand. Where it has been determined that direct and immediate trend or transient information is essential for operator information or action (Type A variables), a continuous dedicated recorder is provided with redundant backup recording and trending available on SPDS and redundant dedicated indicators in the control room that can be utilized for trend information if necessary. (One of the redundant dedicated indicators may be the continuous dedicated recorder listed above.)

Category 2:

Environmental Qualification - Same as Category 1.

Seismic Qualification - No specific provision.

Redundancy - Not required.

Power Supply - Powered by DG or UPS, both considered to be highly reliable.

Quality Assurance - Same as Category 1.

Control Room Display - "On-demand" or continuous display is provided in the control room. No direct or immediate trend or transient information was determined to be essential for operator information or action.

Category 3:

Environmental Qualification - Not required.

Seismic Qualification - Not required.

Redundancy - Not required.

Power Supply - Powered by an available source of power.

Quality Assurance - Same as Category 1.

Control Room Display - Same as Category 2.

7.3.4.2 Safety Parameter Display System (SPDS)

The SPDS is a computer-based system designed to monitor and display to the operator a concise set of parameters from which the safety status of the plant can be readily and reliably ascertained. The system functions as the SPDS for both the ANO-1 and ANO-2 control rooms and provides plant status information for the Technical Support Center (TSC) and Emergency Operations Facility (EOF).

7.3.4.2.1 Design Overview

AP&L began the development of a SPDS in 1979 as part of an in-house initiated EOP upgrade program and expanded the development early in 1980 in response to the "Lessons Learned" NUREGs 0578 and 0585. NUREG-0737, issued in October 1980, required the implementations of the plant SPDS. NUREG-0737 referenced NUREG-0696 for use as the criteria for design of the SPDS and Technical Support Center (TSC)/Emergency Response Facility (ERF) instrumentation systems. However, NUREG-0696 was not issued until March 1981 so several modifications were required to the original computer system design as a result of the new guidance. Supplement 1 of NUREG-0737 was issued in December 1982 to provide additional information and clarification to certain NUREG-0737 requirements.

Supplement 1 also promoted an integrated approach for the implementation of the SPDS, upgraded Emergency Operating Procedure (EOP), control room design reviews, Emergency Response Facilities, and Regulatory Guide 1.97 instrumentation reviews. The present SPDS computer system is designed to meet the intent of the above referenced NRC documents, as described in the ANO-1 SPDS Safety Analysis Report. (1CAN068403)

The SPDS concept was to display a small but critical subset of the information already presented by control room instrumentation in order to minimize information overload. Critical safety functions identified by NUREG-0737, Supplement 1, were carefully identified and parameters for their display selected in order to provide the operators with a concise set of data to aid them in rapidly and reliably determining the safety status of the plant. Selection of the parameters was also coordinated with the development of the upgraded ANO-1 Emergency Operating Procedure (EOP).

7.3.4.2.2 Design Basis

In accordance with the NUREG-0737, Supplement 1, the SPDS was designed to assist the operator in implementing the upgraded EOP. The ANO-1 EOP was developed to achieve timely and accurate safety status assessment either with or without the SPDS. The design of the specific SPDS graphic displays correspond to specific sections of the upgraded EOP, so the SPDS will complement the use of the upgraded EOP.

The basic configuration of the SPDS consists of redundant data acquisition, processing, and display devices. The SPDS computer accesses the necessary input parameters from sensors in ANO-1 and ANO processes the signals, and provides displays to each control room. The SPDS performs no plant control action, but serves as a human-engineered data display system to aid the operator in rapidly and reliably determining plant safety status. The SPDS design should provide enhanced capabilities for responding properly to both anticipated and unanticipated plant conditions.

The SPDS design also includes displays in the TSC and EOF. This improves the operational aids available to the plant technical staff to assist them in evaluating transient conditions and providing guidance and direction to the operations staff.

The TSC and EOF both have access to the large-scale data storage and retrieval capabilities of the SPDS to assist in event diagnosis and historical documentation. Considerable effort has been expended during the initial design of the SPDS to incorporate human factor principles. In addition, the ANO operations staff has played a vital role in the SPDS design and implementation to insure that the system will be responsive to the needs of the operators during normal and emergency conditions. Entergy included the SPDS in the scope of the control room design review (CRDR) program to formally evaluate the proper incorporation of human factors principles including equipment location, display formats and characteristics, operator interfaces, and compatibility with the EOP.

The SPDS was designed to be isolated from electrical and electronic interference with equipment and sensors that are in use for safety systems and was reviewed with respect to IEEE Standard 384-1977, Section 6.2, and found to be in compliance with the isolation criteria. The specific methods of isolation include current transformers for the analog signals and optical couples and relays for the digital signals requiring interference isolation.

The SPDS was subjected to a verification and validation process to insure that applicable requirements were met. This verification included a system requirements review and a design review based on the system requirements. Validation included testing and evaluation of the completed system, hardware, and software to insure compliance with design, function, performance, and interface requirements. This testing confirmed field input calibration; input source to computer point identification relationship; software programs for the acquisition, conversion, manipulation, and display of data from field inputs; proper operation of the central processors related circuits and memory; and peripheral devices. The SPDS verification and validation process was performed and documented in accordance with NRC guidelines in NUREG-0737, Supplement 1.

7.3.4.2.3 Basis For Selection and Displays

The Emergency Operating Procedures (EOPs) for ANO-1 and 2 both have their origin in the Babcock & Wilcox Abnormal Transient Operating Guideline (ATOG) program. From this program it was determined, following a reactor trip and verification of shutdown, that there are three symptoms of primary interest to a pressurized water reactor operator to prevent core and Reactor Coolant System damage: (A) inadequate subcooling of the primary system inventory, (B) inadequate primary to secondary heat transfer, and (C) excessive primary to secondary heat transfer. These symptoms are important for the following reasons:

A. Inadequate primary inventory subcooling: If the operator knows the primary fluid is in a liquid state, he is assured that it is available and capable of removing heat from the core. If subcooling is lost, these issues are in doubt and he is therefore directed to make every effort to regain subcooling.

B. Inadequate primary to secondary heat transfer: This symptom addresses the heat transfer coupling across the steam generator. It describes the ability of the system to keep the flow of energy moving from the Reactor Coolant System to the ultimate heat sink.

C. Excessive primary to secondary heat transfer: In this case, the symptom is indicative of a secondary side malfunction (e.g. loss of steam pressure control or steam generator overfill). The heat transfer is again unbalanced and the operator's attention is directed toward generic actions to restore this balance.

The ATOG pressure-temperature diagram (ATOG Diagram) was developed to provide the above described information to the plant operator in a timely fashion with little or no effort on his part. The ATOG Diagram is the top level display for the ANO-1 SPDS for this reason.

To further enhance the operators' ability to assess the plant's response to transients and to more precisely monitor specific safety functions, additional displays were developed for the ANO-1 SPDS. These additional displays were carefully created to be used in conjunction with the ANO-1 EOP and will aid the operator in the implementation of this procedure as well as some select abnormal operating procedures. A description of the ATOG display and these additional displays is provided below. Implicit in the B&W ATOG program was the consideration of the five critical safety functions identified in NUREG-0737, Supplement 1. Correlation between the following described SPDS displays and the five critical safety functions are also discussed below.

ATOG Pressure-Temperature (ATOG)

The ATOG basic diagram features a grid of RCS pressure versus RCS temperature with fixed curves showing the saturation lines, a minimum adequate margin to saturation line and the RCS NDTT limits for normal operation. During power operation the Reactor Protection System (RPS) pressure-temperature trip limits are shown along with a normal transient window showing the minimum and maximum pressures and temperatures expected immediately following a trip. After a reactor trip, a small box will appear inside this window showing the expected pressure-temperature relationship for normal hot shutdown conditions.

The dynamic elements of the display include a bar graph representation of OTSG levels, digital values of selected parameters displayed below the P-T grid, and points identifying T_{cold} and T_{hot} versus pressure. Following a reactor trip, the past plotted values of RCS temperature versus pressure remain on the screen showing the trajectory which they are following. The values shown at the bottom of the screen are reactor building temperature and pressure, A and B OTSG pressures, and the average of the five highest core exit thermocouple temperatures.

During inadequate core cooling situations, pressure versus temperature is plotted on a high range temperature scale. On this display, core exit temperatures versus pressure corresponding to 1400 °F and 1800 °F cladding temperatures are indicated. These curves and the high range temperature scale are displayed when the average of the five highest core exit temperatures indicates greater than 50 °F superheat. The ATOG display is returned when the RCS is returned to saturation.

To provide the operator with indications to monitor the approach to the inadequate core cooling situation, a graphic display of RCS inventory parameters is provided. This is a software representation if the ICCMDS with additional inputs from pressurizer level, steam generator level, and relief valve positions.

If the RCS is less than adequately subcooled after a reactor trip, a "Loss of Subcooling Margin Elapsed Time HH:MM:SS" is displayed in the lower right area of the P-T grid. This will disappear and reset when adequate margin-to-saturation is regained.

A typical plant response to a reactor trip is shown in Figure 7-28. T_{cold} should merge with T_{hot} as the decay heat rapidly drops. Both temperatures should then move toward normal hot shutdown pressure and temperature conditions. If they do not, a departure from normal is indicated. If they move outside a larger normal transient limit area, the definite need for operator action is indicated.

Each of the three basic symptoms discussed earlier leave their unique signature on the ATOG diagram as displayed in Figures 7-29, 7-30, and 7-31. Use of this tool enables an operator's priority to be fixed on controlling the plotted parameters within target bounds. If successful, he will be able to bring the reactor to a safe condition. This will be the case regardless of whether or not he has properly diagnosed (or diagnosed at all) the event which has occurred. However, use of the SPDS in conjunction with the EOP does not discourage an operator from diagnosing the cause of the transient. The SPDS and EOP are based on directing the operator to take proper actions without diagnosis or with misdiagnosis.

Primary to Secondary Heat Transfer (PSHT)

The primary to secondary heat transfer display was designed to provide more detailed historical data on some of the parameters shown on the ATOG diagram. This display specifically addresses the reactor core cooling and heat removal critical safety function. This display (and several of the others described following) presents data in a scrolling trend versus time format, similar to a strip chart recorder, which results in a familiar, easily readable and understandable display.

Page one has trends for A and B loops of the average of the five highest core exit thermocouples, loop average hot leg temperature, loop average cold leg temperature, and saturation temperature for OTSG pressure.

Page two displays trends of feedwater flow, OTSG level, and OTSG shell-to-tube temperature differential for both loops.

This display will be particularly valuable in natural circulation conditions where RCS temperature can be closely monitored. Verification of natural circulation can be readily accomplished by noting that core exit and hot leg temperatures are tracking and trending down. Also, cold leg temperatures should be tracking the saturation temperature for that loop's OTSG pressure.

Reactivity Control (RHO)

The reactivity control display was provided to aid the operator in immediate verification that the reactor is indeed tripped and remains shut down. This display specifically addresses the reactivity control critical safety function. Trends of regulating control rod group index and average hot leg temperature are shown on the top half of this display and intermediate and source range nuclear instrumentation outputs are shown on the bottom half.

Steam Generator Tube Rupture (SGTR)

A steam generator tube rupture is treated as a unique event in both the Abnormal Transient Operating Guidelines and the EOP. This event is unique in that it has the potential for a direct release of radiation to the environment. The steam generator tube rupture display is designed to aid the operator in diagnosing a tube rupture, determining the affected generator, and cooling the plant down to a condition where the primary to secondary leakage can be terminated. This display shows trends for each loop of RCS average temperature and shell-to-tube temperature differential on the top half and condenser off-gas radiation and A and B N-16 monitors on the bottom half.

RCS Inventory (RCSI)

The RCS inventory display is provided to aid the operator in assessment and maintenance of RCS inventory. This display specifically addresses the RCS integrity critical safety function. Pressurizer level, RCS average temperature, and RCS pressure are trended on the top half of this display while reactor building sump level, letdown flow, and makeup tank level are on the bottom half.

Trends shown on this display may also aid in the diagnosis of the initiating event. For example, the behavior of RCS temperature while pressurizer level and pressure are decreasing is the key to differentiating between a small break loss of coolant accident and an overcooling event. Both events will show decreasing pressure and pressurizer level; however, rapidly decreasing temperature would indicate an overcooling event while a constant or very slight decrease in temperature would indicate a loss of coolant.

Reactor Building Condition (RB)

The reactor building conditions display is provided to show reactor building parameters which may be useful in assessing and maintaining reactor building integrity. This display specifically addresses the containment conditions critical safety function. This display provides trends of reactor building atmosphere hydrogen concentration and high-range radiation monitor indications on the top half and reactor building pressure, temperature, and sump (flood) level on the bottom half.

The radioactivity control critical safety function is not specifically addressed by any single SPDS display; however, as discussed above, various radiation monitor indications are included on the appropriate displays.

Auxiliary Displays

In addition to the dedicated SPDS displays described above, there are auxiliary displays, which may be selected. Examples of auxiliary displays include heatup and cooldown displays, the core exit thermocouple map, graphic trend and history displays, inadequate core cooling display and a low-range pressure-temperature display. Historical pressure versus temperature data from the previous 20 hours is saved and displayed on the heatup and cooldown displays when these displays are requested. These displays were designed for operator convenience only and are not considered necessary to address any of the critical safety functions and therefore are not addressed by this Safety Analysis Report.

Alternate and Safe Shutdown Displays

During the 1R6 refueling outage, two additional displays were added to address the new Appendix R requirements concerning fire protection and safe shutdown capabilities. These displays are the alternate shutdown green channel inputs and the safe shutdown red channel inputs. These safe shutdown displays are two page displays with sensor inputs that are independent of the plant Non-Nuclear Instrumentation (NNI) System. The first page is a P-T diagram similar to the ATOG display and the second is similar to the trend format displays. These displays are used during an event which requires the use of the Alternate Shutdown Abnormal Operations Procedure, such as a fire in the control room or cable spreading room. The inputs to these displays are source range flux, pressurizer level, RCS pressure and A and B hot and cold leg temperature, OTSG levels and pressures.

7.3.4.2.4 Hardware Description

The computer system was selected primarily because of its flexibility, reliability, and maintainability. Flexibility is needed to permit the incorporation of future modifications without unwarranted difficulty. The computer hardware selected is similar to existing hardware already in use at ANO. This hardware has been proven to be reliable. Furthermore, Entergy Operations, Inc. personnel have considerable experience in maintaining this equipment which should improve the overall reliability of the system.

The computer system is an integrated network which is designed to perform the functions required for the ANO-1 SPDS, the ANO-2 SPDS, the Technical Support Center data display system, and the Emergency Operations Facility data display system. To achieve these functions, the SPDS computer accesses necessary input parameters from sensors in ANO-1 and ANO-2, processes these signals, and provides displays to each control room as well as to the TSC and the secondary TSC portion of the EOF. Suitable isolation is provided between the SPDS data acquisition system and Class IE sensors. In each control room, color graphic displays are provided for the operators. Color graphic displays are also provided in the TSC as well as the EOF. "Touch screen" controls are utilized on the color graphic displays on C09 and C19 to allow for rapid access of the information necessary to determine safety status of the plant.

Several features have been incorporated into the SPDS design in order to approach the availability goals specified in the NRC guidance and to allow for incorporation of future modifications while the system is operating. These features include redundant CPUs, redundant data acquisition hardware, redundant networks, and redundant color graphic displays in each control room. Except for the TSC equipment, which is powered from a diesel generator backed panel, the SPDS power is supplied from an uninterruptable power supply (UPS). The SPDS computer room is also provided with its own air conditioning system.

7.3.4.2.5 Software Description

The primary operating system software for the SPDS is similar to that used for the replaced SEL 8600 plant monitoring computer. Parts of this operating system and some of the associated application programs were modified by the ANO Computer Support Department to facilitate the special requirements of the SPDS.

The SPDS software provides applicable data over redundant networks to display units in the control room, EOF, and TSC. Each display unit generates appropriate screens and processes user commands.

7.4 OPERATING CONTROL STATIONS

Following proven power station design philosophy, control stations, switches, controllers, and indicators necessary to start up, operate, and shut down the plant are located in the control room. Control functions necessary to maintain safe conditions after a Loss of Coolant Accident (LOCA) are initiated from the centrally located control room. Controls for certain auxiliary systems are located at remote control stations when the system controlled does not involve power generation control or emergency functions.

The flammable materials in the control room consist primarily of charts, records, and some of the electronic equipment such that a fire resulting in loss of redundant control systems is not considered possible. Shielding around the control room is provided to ensure operator safety during any Design Basis Accident (DBA). No design basis accident or credible combination of accidents will result in loss of control room functions. As discussed in Section 1.4.15, the reactor can be shut down and maintained in a safe shutdown condition in the unlikely event that the control room need be abandoned.

7.4.1 GENERAL LAYOUT

The control room is designed so that one man can supervise operation during normal steady-state conditions. During other than normal operating conditions other personnel can assist the reactor operator without mutual interference. Figure 7-25 shows the control room layout. Corresponding control console and vertical panel areas are aligned, thus operator(s) face control and instrumentation equipment on a system basis. This feature has been extended to include the annunciator groups mounted on the vertical panels. The physical separation of duplicate engineered safeguards equipment is ensured by means of fire barriers, separate raceways, and careful layout.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTION

The necessary information for most routine monitoring of the nuclear unit and the plant is displayed on the control room consoles and vertical panels in the immediate vicinity of the operator. Information display and control equipment frequently employed on a routine basis, or protective equipment quickly needed in case of an emergency, are mounted primarily on the console sections. Recorders and radiation monitoring equipment are mounted primarily on the vertical panels. Infrequently used equipment, such as indicators and controllers used primarily during startup or shutdown, are generally mounted on vertical panels.

A plant computer is provided in the control room for alarming, post-trip review, sequence of events analysis, performance monitoring, display of information related to the Anticipatory Transients Operating Guidelines (ATOG), data logging, and other auxiliary functions. On-demand printout is available to the operator at his discretion in addition to the computer periodic logging of plant variables. No vital alarm function is accomplished through the computer.

Information displays are designed to provide the operator with sufficient information to make proper evaluations under the full range of plant operating conditions. The displays are arranged to facilitate evaluation and to avoid the possibility of confusing the operator.

All parameters monitored by the Reactor Protection System (RPS) are indicated on the system cabinets, the control consoles, and as plant computer outputs. Instrumentation strings associated with the RPS are designed in accordance with IEEE 279-1968. The design criteria enumerated in Section 7.3.2.2 apply to all displays of non-nuclear parameters.

Design features of the instrumentation used to monitor the reactor, Reactor Coolant System (RCS), and the containment are contained in Table 7-11. The design ranges of these indicators are derived by a conservative analysis of the parameter involved. In this way the operator is assured that he will have accurate information over the full range of each parameter. The number of sensors or channels used is based on the requirements for systems which monitor parameters.

Table 7-11 identifies the information readouts or indicators provided for the operator to monitor conditions in the RCS and the containment. For specific post-accident monitoring instrumentation see Section 7.3.4.

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units are incorporated into the control room to warn the operator if unsafe conditions are being approached by plant systems. Standard annunciator equipment extensively used in modern power plants is used. Audible evacuation alarms are initiated manually by the operator from the Radiation Monitoring System panel. Audible alarms will be sounded in appropriate areas throughout the plant if high radiation conditions are present.

Annunciator panels are mounted on the vertical control boards on a systems basis as mentioned in Section 7.4.1. Main alarms correspond to the following systems:

A. Electrical System

B. Turbine Generator System

C. Feedwater and Condensate Systems and pumps

D. Integrated Control System

E. Reactor Coolant Pumps and System, Makeup and Purification System

F. Nuclear Instrumentation, Protection and Engineered Safeguards Systems

G. Chemical Addition, Decay Heat Removal and Intermediate Cooling Water Systems

H. Plant Auxiliary and Miscellaneous Systems

7.4.4 COMMUNICATION

Plant telephone and paging systems are provided with redundant power supplies to provide the control room operator with constant communication with plant areas. Two plant telephone systems are installed. Each plant telephone extension has direct dialing access to the whole paging system so that loss of one call station will not substantially decrease the paging function. Acoustic booths are supplied in areas where the background noise level is high. Communication outside the plant is accomplished through the commercial telephone system. Communication to the Nuclear Regulatory Commission is through the Emergency Notification System (ENS) to the Incident Response Center and also through the Health Physics Network (Health Physics phone).

Communications system tests (including noise level tests) have been conducted to demonstrate the adequacy of the in-plant communications system to provide communications between vital plant areas following accidents or incidents. The communications system has been modified as necessary to meet the acceptance criteria of these tests.

In-plant radio systems are provided for use with the remote shutdown panel.

7.4.5 OCCUPANCY

Safe occupancy of the control room during abnormal conditions is provided for in the design of the auxiliary building. Adequate shielding is used to maintain tolerable radiation levels in the control room for DBA conditions. The Control Room Ventilation System is provided with radiation detectors and appropriate alarms. The Emergency Air Conditioning and Filtration Systems provided for the Control Room are described in Section 9.7.2.1. Emergency lighting is provided via automatic switching by DC and diesel generator fed AC circuits and by battery-powered lights.

The potential magnitude of a fire in the control room is limited by the following factors:

A. The control room construction is of noncombustible materials.

B. Control cables and switchboard wiring is fabricated and assembled to pass the flame test as described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.

C. Furniture used in the control room is primarily of metal construction.

D. Combustible supplies such as logs, records, procedures, drawings, and manuals are limited to amounts required for plant operations.

E. A Halon fire suppression system is utilized in the control room.

F. All areas of the control room are readily accessible for fire extinguishing.

G. Adequate fire extinguishers are provided.

H. The control room is occupied at all times by qualified personnel who have been trained in fire extinguishing techniques.

The flammable materials inside the control room consist primarily of:

A. Paper in the form of logs, records, procedures, manuals, and diagrams.

B. Small amounts of combustible materials contained in various electronic equipment. The above list indicates that the flammable materials are distributed to the extent that a fire would be unlikely to spread. Therefore, a fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

Essential auxiliary equipment is controlled by stored energy, closing-type, air-circuit breakers which are accessible and can be manually closed in the event DC control power is lost.

7.4.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations are provided where their use simplifies control of auxiliary systems equipment such as sample valve selectors, chemical addition, and radwaste.

7.4.7 SAFETY FEATURES

The control room layout provides the necessary controls to start, operate, and shut down the nuclear unit with sufficient information display and alarm monitoring ensuring safe and reliable operation under normal and accident conditions. The layout of the engineered safeguards section of the control board was intended to minimize the time required for the operator to evaluate the system performance under accident conditions. Deviations from normal conditions are alarmed so that the operator may take corrective action using the controls provided on the control panel.

7.5 REFERENCES

Babcock & Wilcox Topical Report References

BAW-10001	Incore Instrumentation Test Program
BAW-10003, Revision 2	Qualification Testing of Protection System Instrumentation
BAW-10003A, Revision 4	Qualification Testing of Protective System Instrumentation
BAW-10010	Part 1 - Stability Margin for Xenon Oscillations - Modal Analysis

Table 7-1

**ENVIRONMENTAL OPERATING CONDITIONS
SPECIFIED FOR PROCUREMENT OF EQUIPMENT**

Equipment	Conditions	
Protection System Equipment (continuous operation)	Ambient Conditions:	40 - 120 °F, 100% humidity, and 25 mRem/hr up to a total radiation exposure of 1 x 10^4 roentgens
Protection System Equipment (24 hours of operation by essential equipment)	Following LOCA:	286 °F, 59 psig, 100% humidity, and total radiation exposure of 2 x 10^4 roentgens (1CAN127711)
Neutron Detectors	Design Conditions:	150 °F, 40% humidity 212 °F, 90% humidity, and 150 psig
Fission Chambers (Source range)	Normal Conditions: Following LOCA:	120 °F, 20 - 90% humidity 53.1 psig peak 280 °F 100% humidity*

* This equipment has been environmentally qualified for post-LOCA conditions. For current qualification, see the EQ program documentation.

Table 7-2

REACTOR TRIP SUMMARY

NOTE: For trip value or condition for trip, see ANO-1 Technical Specifications Table 3.3.1-1.

Trip Variable	No. of Sensors	Steady-State Normal Range
Over Power	4 Flux Sensors	2-100%
Power-Imbalance- Flow	4 Flux Sensors 8 $\Delta\rho$ Flow Transmitters 2 Flow Nozzles	Variable
Power/RC pumps	4 Pump Monitors with 16 Contacts 4 Flux Sensors	2-4 Pumps
Reactor Outlet Temperature	4 Temperature Sensors	532 - 604 °F
Pressure/Temperature	4 Pressure Sensors 4 Temperature Sensors	Variable
Reactor Coolant Pressure	4 Pressure Sensors	2,090 - 2,220 psig
Reactor Building Pressure	4 Pressure Sensors	0 psig
Turbine Trip	4 Pressure Sensors	Normally Closed
Loss of Both Main Feedwater Pumps	4 Pressure Sensors (Each Pump)	Normally Closed

Table 7-3

IMBALANCE OPERATING ENVELOPE PARAMETERS

Parameter	Range of Adjustment
$f(\Delta\theta)$	0 to 125%
B_1	$-62.5 \leq \Delta\theta \leq B_2$
B_2	$B_1 \leq \Delta\theta \leq B_3$
B_3	$B_2 \leq \Delta\theta \leq B_4$
B_4	$B_3 \leq \Delta\theta \leq 62.5$

Table 7-4

ENGINEERED SAFEGUARDS ACTUATION CONDITIONS

NOTE: For actual values, see ANO-1 Technical Specifications.

Channel No.	Action	Trip Condition	Steady State Normal Value
1,2	High Pressure Injection	Low Reactor Coolant Pressure or	2,090 - 2,220 psig
	Diverse Containment Isolation	High Reactor Building Pressure	Atmospheric
3,4	Low Pressure Injection	Low Reactor Coolant Pressure or	2,090 - 2,220 psig
	Diverse Containment Isolation EFIC	High Reactor Building Pressure	Atmospheric
5,6	Reactor Building Cooling and Reactor Building Isolation	High Reactor Building Pressure	Atmospheric
7,8	Reactor Building Spray System	High Reactor Building Pressure	Atmospheric

Table 7-5

ENGINEERED SAFEGUARDS ACTUATED DEVICES

Channel 1	Channel 2	Channels 1 & 2	Channel 3	Channel 4	Channel 5
P36A	P36C	P36B	P34A	P34B	CV6205
CV1219	CV1227	CV3643	CV1401	CV1400	VSF1A
CV1220	CV1228	K4A	CV1407	CV1408	VSF1B
CV1278	CV1284	K4B	CV7404	CV4400	CV3812
CV1279	CV1285	P64B	CV7403	CV7401	CV3814
CV1214	CV1221		CV1053	CV7402	CV2214
CV1216	CV1408		CV5612	CV7454	CV2220
CV1407	CV1274		CV1667*	CV5611	SV7410
CV1270	CV3811		CV7453	CV1667*	SV7411
CV1271	CV3644		EFIC-A	SV7510**	CV7470
CV1272	CV1234		CV3840	SV7512**	CV7471
CV1273	CV3642		CV3822	SV7456	VEF38A
CV1233	CV1300		CV1428	SV7454	CV2123
CV3820	P64C		CV1433	EFIC-B	
CV3640	CV1845		SV7479	CV3821	
CV3646	CV4804		CV1441	CV3841	
CV1301	CV1052			CV1429	
P64A	CV1438			CV1432	
CV1054				CV1438	
CV4446					
CV4803					
CV1441					

Channel 6	Channel 5&6	Channels 7	Channel 8
VSF1C	CV2234	P35A	P35B
VSF1D	CV6202	CV2401	CV2400
CV3813	CV2100		
CV3815	CV2101		
CV2215	CV2102		
CV2221	CV2103		
CV6203	CV2104		
SV7412	CV2105		
SV7413	CV2106		
CV7472	CV2107		
CV7473	CV2108		
VEF38B	CV2111		
CV2133	CV2112		
	CV2113		
	CV2114		
	CV2115		
	CV2116		
	CV2233		
	CV2235		
	CV1065		

* Credit is not taken for CV-1667 since N2-47 performs the containment isolation function.
** Physically isolated from boundary, but still receives an Engineered Safeguards Actuation Signal.

Table 7-6

INTEGRATED CONTROL SYSTEM TRANSIENT LIMITS

Transient	Power Range (% Full Power)	Ramp Input Limit (% Power/min)	Step Input Limit (% Power)
Power Increase	0 - 15	NA	NA
	15 - 20	5	0
	20 - 90	10	10
	90 - 100	5	0
Power Decrease	100 - 20	10	10
	20 - 15	5	0
	15 - 0	NA	NA

Table 7-7

CHARACTERISTICS OF OUT-OF-CORE NEUTRON DETECTOR ASSEMBLIES

Characteristic	Source	Source	Intermediate	Power
Tube Type	PC	Fission Chamber	CIC	UCIC
Sensitivity (Assembly)				
Thermal Neutron Flux	50 cps/nv	27 cps/mv (typical)	7.6×10^{-14} A/nv	1.62×10^{-13} A/nv
Gamma Flux	NA	(Ref V53 Item 3, page 2)	2.3×10^{-13} A/R/h (compensated)	3×10^{-10} A/R/h
Maximum Ratings				
External Pressure	150 psig	70 psig	150 psig	150 psig
Temperature (Assembly)	212 °F	466 °F (peak)	212 °F	212 °F
Thermal Neutron Flux				
Operating	4×10^{4} nv	2×10^{10} nv	2.5×10^{10} nv	2.5×10^{10} nv
Non-Operating	1×10^{10} nv		2.5×10^{11} nv	2.5×100^{11} nv
Gamma Flux	1×10^{5} R/hr	10^{6} R/hr	5×10^{5} R/hr	5×10^{5} R/hr
Integrated Exposure Before 10% Reduction In Sensitivity				
Neutron	10^{19} nvt		10^{19} nvt	10^{19} nvt
Gamma	3×10^{9} R		3×10^{9} R	3×10^{9} R

* Original detection system; the high sensitivity fission chambers are the primary operating source range system.

Table 7-8

POWER REACTORS WHERE IDENTICAL TYPE DETECTORS ARE EMPLOYED (HISTORICAL INFORMATION)

Tube Type	Reactors	Utility
PC,CIC	Beznau-I	NOK
	R.E. Ginna	Rochester Gas & Electric
	Oconee I - II	Duke Power
	Three Mile Island-I	Metropolitan Edison
	Crystal River	Florida Power
UCIC	Connecticut Yankee	Connecticut Yankee Power
	Oconee I - II	Duke Power
	Three Mile Island-I	Metropolitan Edison
	Crystal River	Florida Power

Table 7-9

NNI INPUTS TO THE REACTOR PROTECTION SYSTEM

Characteristic	Reactor Outlet Pressure (NR)[a]	Reactor Outlet Temperature (NR)[a]	Reactor Coolant Flow
Component Item Number	PT-1021 PT-1023 PT-1038 PT-1039	TE-1012 TE-1013 TE-1040 TE-1041	PDT-1028 PDT-1029 PDT-1030 PDT-1031 PDT-1034 PDT-1035 PDT-1036 PDT-1037
Reactor Protection Channel	A,B,C,D	A,B,C,D	A,B,C,D[b]
Sensor Type	Pressure Transmitter	RTD	Differential Pressure Transmitter
Accuracy[d]	± 0.25% of Span	± 1.0 °F	± 0.25% of Span
Expected Failure Mode	Low	High	Low
Type Readout	All Indicating	All Indicating	All Indicating
Power Required	External	External	External
Sensors Connected to Common Taps	PT-1021 and PT-1020[c] PT-1023 and PT-1022[c] PT-1038 and PT-1040[c] PT-1039 and PT-1042[c]	Not Applicable	All Sensors for Same Loop are Connected to Common Taps

[a] NR = Narrow Range.

[b] Each channel has an input from each loop.

[c] Pressure taps for each RPS channel are independent. A RPS channel and an ESAS channel may have common pressure sensing taps.

[d] Manufacturer's guaranteed accuracy for non-accident reference conditions.

Table 7-10

NNI INPUTS TO THE ENGINEERED SAFEGUARDS SYSTEM

Characteristic	Reactor Outlet Pressure	Reactor Building Pressure
Component Item Number	PT-1020 PT-1022 PT-1040	PT-2405 PT-2406 PT-2407
ESAS Channel	A,B,C	A,B,C
Sensor Type	Pressure Transmitter	Pressure Transmitter
Accuracy[a]	± 0.25% of Span	± 0.25% of Span
Expected Failure Mode	Low	Low
Type Readout	All Indicating	All Indicating
Power Required	External	External
Sensors Connected to Common Taps[b]	PT-1020 and PT-1021 PT-1022 and PT-1023 PT-1040 and PT-1038	All Separate Building Penetrations

[a] Channel has an input from each loop.

[b] Pressure taps for each ESAS channel are independent. A RPS channel and an ESAS channel may have common pressure sensing taps.

Table 7-11

INFORMATION READOUTS AVAILABLE TO THE OPERATOR FOR MONITORING CONDITIONS IN THE REACTOR, REACTOR COOLANT SYSTEM, AND IN THE REACTOR BUILDING

Measured Parameter	Type of Readout	Number of Channels Or Sensors	Indicator Range	Indicator Location
Area Radiation Level	B,F,G	4	0.01 to 1000 rem	B,C
Gaseous Radiation Level	B,F,G	1	0 to 10^8 cpm	B,C
Reactor Building Sump Level	A,G	1	0 to 100%	B
Reactor Building Pressure	A,F,G	4	0 to 65 psi	A,B
Source Range Neutron Level	B,F	2	10^{-1} to 10^5 cps	A,B,C
Source Range Start Up Rate	A,F	2	-1 to 7 dpm	A,B,C
Intermediate Range Neutron Level	B,F	2	10^{-11} to 10^{-3} amp	A,B,C
Intermediate Range Start Up Rate	A,F	2	-0.5 to 5 dpm	A,B,C Power
Range Neutron Level	A,F,E	4	0 to 125% FP	A,B,C
Power Range Neutron Level Imbalance	A,F	4	62.5 to +62.5% FP	A,B,C
Incore Neutron Detector Response**	F	416	0 to 2000 na (Short Emitter) 0 to 6000 na (Long Emitter)	C
RC Loop Outlet Temperature	A,F	4 in Each Loop	520 to 620 °F	A,B,C,D
RC Loop Outlet Temperature	A,E,F,D	3 [2 in A loop; 1 in B loop]	50 to 700 °F	B,C
RC Unit Outlet Temperature	A,E	*	520 to 620 °F	B
RC Loop Inlet Temperature (Wide)	A,F	2 in Each Loop	50 to 650 °F	B,C
RC Loop Inlet Temperature (Narrow)	A,F	2 in Each Loop	520 to 620 °F	B,C
RC Unit Inlet Temperature	A	*	520 to 620 °F	B
RC Loop Average Temperature	A	*	520 to 620 °F	B
RC Unit Average Temperature	A,E	*	520 to 620 °F	B
RC Loop Temperature Difference	A	*	0 to 70 °F	B
RC Unit Temperature Difference	A	*	0 to 70 °F	B
RC Loop Pressure	A,E,F	7	0 to 2500/1700 to 2500 psig	A,B,C,D
Pressurizer Level	A,E,F	2	0 to 320 in	B,D,C

Table 7-11 (continued)

Measured Parameter	Type of Readout	Number of Channels Or Sensors	Indicator Range	Indicator Location
Pressurizer Temperature	A,F	2	0 to 700 °F	B,C
RC Loop Flow	A,F	4 Each Loop	0 to 110% Full Flow	A,B,C
RC Total Flow	A,E,F	*	0 to 110% Full Flow	A,B,C
Steam Generator Lower Range Level	A,F,E	2 in Each Loop	6 - 156 in	B,C
Steam Generator Operator Range Level	E,F,A	2 in Each Loop	102 - 500 in	B,C
RC Hot Leg Level	A,D,F,G	1 Wide Range per Loop 4 Narrow Range per Loop	0 - 588 in	A,B,C
Steam Generator Outlet Pressure	A,F		0 - 1200 psig	B,D,C
Steam Temperature	A,F	2	100 - 650 °F	B,D,C
Turbine Throttle Pressure	D,E,F	2	600 - 1200 psig	B,D,C
Startup Feedwater Flow	A,F,E***	2	0 - 1.5 x 10^6 lb/hr	B,D,C
Main Feedwater Flow	A,F,E***	4	0 - 6.0 x 10^6 lb/hr	B,D,C
Feedwater Temperature	A,F	4	0 - 500 °F	B,D,C
Pressurizer Safety/Relief Valve Position	C,G	2	Open/Close	B
Reactor coolant T_{SAT}	D,G,F	2	-100 - 200 °F	B,C
High Range Containment Radiation	B,E,F	2	1 – 10^8 R/hr	A,C
Wide Range Containment Pressure	A,E,F	2	0 - 210 psi	B,C
Containment Water Level	A,E,F	2	0 - 144 in	B,C
Containment Hydrogen Concentration	A,E,F	2	0 - 10%	A,C

Location

A - System cabinets
B - Control console
C - Plant computer output
D - Other locations outside of control room

Legends: Type of Readout

A - Linear scale indicator
B - Log scale indicator
C - Indicator light
D - Digital indicator
E - Recorder
F - Plant computer output
G - Alarm

Notes:
* Two or more signals are combined to produce the indicated parameter.
** The incore monitoring system provides information on core power distribution as described in Section 7.3.3.
*** Recorder used for both main and startup feedwater flow.

Table 7-11A

R.G. 1.97 POST ACCIDENT MONITORING VARIABLES

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "A" VARIABLES							
RCS Pressure	1	0 - 3000 psig	Yes (2 Channels)	1E	1 Indicator 1 Recorder	Yes	
RCS Hot Leg Water Temp	1	50 - 700 °F	Yes (2 Channels)	1E	1 Indicator 1 Recorder	Yes	
Steam Generator Pressure	1	0 - 1200 psig	Yes (2 Channels/SG)	1E	2 Indicators (1/SG) 2 Single Pen Recorders (1/SG)	Yes	
Steam Generator Level	1	6" - 156" H_2O 102" - 500" H_2O	Yes (2 Channels/SG/ range)	1E	4 Dual Indicators (1/SG Channel with both ranges/ Indicator) 2 Dual Pen Recorders (1/SG Range/Pen)	Yes	
Borated Water Storage Tank Level	1	0 - 45 ft	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	
Condensate Storage Tank Level	1	0 - 30 ft	Yes (2 Channels)	1E	1 Indicator 1 Recorder	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "A" VARIABLES (continued)							
Reactivity Control							
Flow in HPI System	1	0 - 200 gpm	Yes (2 Channels)	1E	2 Indicating Recorders	Yes	
Reactor Building Spray Flow	1	0 - 2000 gpm	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	
Low Pressure Injection Flow	1	0 - 4500 gpm	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	
TYPE "B" VARIABLES							
Reactivity Control							
Neutron Flux	1	10^{-8}% to 100% F.P.	Yes (2 Channels)	1E	1 Recorder	Yes	
Control Rod Position	3	Full in or Not Full in	N/A	UPS	SPDS	Yes	
RCS Soluble Boron Conc.	3	--	N/A	N/A	N/A	No	See Note 1
RCS Cold Leg Water Temp	3	50 - 650 °F	N/A	DG/UPS	1 Dual Indicator (Selectable to any cold leg)	Yes	
Core Cooling							
RCS Hot Leg Water Temp	1	---	---	---	---	---	See previous listing

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "B" VARIABLES (continued)							
Core Cooling (continued)							
RCS Cold Leg Water Temp.	3	---	---	---	---	---	See previous listing
Reactivity Control							
RCS Pressure	1	---	---	---	---	---	See previous listing
Core Exit Temp.	3	---	---	---	---	---	See listing below
Coolant Inventory							
Hot Leg Level	1	368'6" - 417'6"	Yes (2 Channels)	1E	2 Display Devices	Yes	
Reactor Vessel Level	1	Fuel Alignment Plate to Top of Dome	Yes (2 Channels)	1E	2 Display Devices	Yes	
Degrees of Subcooling	2	0 - 200 °F	Yes (2 Channels)	1E	2 Display Devices 1 Indicator	Yes	The given range is for the Bargraph on the SMM* Indicator. The range of the digital indication is -100 to 220 °F

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "B" VARIABLES (continued)							
Maintaining RCS Integrity							
RCS Pressure	1	---	---	---	---	---	See previous listing
Reactor Bldg Sump Water Level (Narrow Range-Sump)	2	0 - 100%	N/A	1E	1 Indicator	Yes	
Reactor Bldg Water Level (Wide Range)	1	0 - 144"	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	
Reactor Bldg Pressure	1	0 - 210 psia (-15 - 195 psig) Design = 74 psia (59 psig)	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	
Maintaining Containment Integrity							
Containment Isolation Valve Position	1	Closed/Not Closed	Yes	1E	Lights (2/Valve)	No	See Note 2
Containment Pressure	1	---	---	---	---	---	See previous listing
TYPE "C" VARIABLES							
Fuel Cladding							
Core Exit temp.	1	50°-2300 °F	Yes (2 Channels)	1E	2 Display Devices	Yes	CET Average Temperature

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "C" VARIABLES (continued)							
Fuel Cladding (continued)							
Radioactivity Concentration or Radiation level in Circulating Primary Coolant	3	10^{-4} μCi/gm to 10 Ci/gm	N/A	OP	N/A	No	See Note 1
Analysis of Primary Coolant (Gamma Spectrum)	3	10^{-4} μCi/ml to 10 Ci/ml	N/A	OP	N/A	No	See Note 1
Reactor Coolant Pressure Boundary							
RCS Pressure	1	---	---	---	---	---	See previous listing
Containment Pressure	1	---	---	---	---	---	See previous listing
Containment Sump Water Water Level-Narrow Range Wide Range	--- 2 1	---	---	---	---	---	See previous listing
Containment Area Radiation Monitors	1	---	---	---	---	---	See listing Type "E"
Reactor Coolant Pressure Boundary							
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust	3	0 to 10^{8} cpm	N/A	DG	1 Indicator 1 Recorder	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "C" VARIABLES (continued)							
Containment							
RCS Pressure	1	---	---	---	---	---	See previous listing
Containment Hydrogen Concentration	3	0 – 10% Vol.	Yes (2 Channels)	1E	2 Indicators 1 Recorder	Yes	See Note 3
Containment Pressure	1	---	---	---	---	---	See previous listing
Containment Effluent Radioactivity-Noble Gases from Identified Release Points	2	$1.0E^{-6}$ µCi/cc to $1.0E^{5}$ µCi/cc 0 - 110% Vent Design Flow	N/A	OP	CRT (RDACS)	No	
Effluent Radioactivity-Noble Gases from Building	2	$1.0E^{-6}$ µCi/cc to $1.0E^{5}$ µCi/cc 0 - 110% Vent Design Flow	N/A	OP	CRT (RDACS)	No	
TYPE "D" VARIABLES							
Residual Heat Removal (RHR) or Decay Heat Removal System							
RHR System Flow	2	0 - 4500 gpm (Design = 4000 gpm					See "LPI Flow" (Type A)

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "D" VARIABLES (continued)							
Residual Heat Removal (RHR) or Decay Heat Removal System (continued)							
RHR Heat Exchanger Outlet Temp	2	0 - 400 °F	N/A	UPS/DG	2 Indicators (1/Heat Exchanger)	Yes	
Core Flood Tank (Accumulator) Level	3	(0 - 14 ft) 11% - 78%	N/A	UPS/DG	4 Indicators (2/Tank)	Yes	
Core Flood Tank Pressure	3	0 - 800 psig	N/A	UPS/DG	4 Indicators (2/Tank)	No	
Core Flood Tank (Accumulator) Isolation Valve Position	2	Closed/ Not Closed	N/A	DG	4 Lights (2/Valve)	No	
Flow in HPI System	2	---	---	---	---	---	See previous listing
Flow in LPI System	2	---	---	---	---	---	See previous listing
Refueling Water Storage Tank Level	2	---	---	---	---	---	See previous listing
Primary Coolant System							
RCP Status (Motor Current)	3	0 - 600 amps	N/A	UPS	SPDS	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "D" VARIABLES (continued)							
Primary Coolant System (continued)							
Primary System Safety Relief Valve Position (Including PORVs and Code Valves) or Flow Through or Pressure in Relief Valve Lines	2	Closed/ Not Closed	N/A	DG	3 Indicators (1/Safety Valve)	Yes	
Pressurizer Level	1	87" - 407"	Yes (2 Channels)	1E	1 Recorder 1 Indicator	Yes	
Pressurizer Heater Status	2	Current	N/A	DG	SPDS	Yes	
Quench Tank Level	3	400 - 9600 gals	N/A	DG	1 Indicator	No	
Quench Tank Temperature	3	0 - 400 °F	N/A	DG	1 Indicator	No	
Quench Tank Pressure	3	0 - 100 psig Design = 100 psig	N/A	DG	1 Indicator	No	
Secondary System (Steam Generator)							
Steam Generator Level	1	---	---	---	---	---	See previous listing
Steam Generator Pressure	1	---	---	---	---	---	See previous listing
Safety/Relief Valve Position or Main Steam Flow	2	Closed/Not Closed	N/A	DG	2 Lights/Valve	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "D" VARIABLES (continued)							
Secondary System (Steam Generator) (continued)							
Main Feedwater Flow	3	0 - 6.0 x 10^6 lb/hr (Design = 5.5 x 10^6 lb/hr)	N/A	UPS/DG	2 Indicators (1/MF Pump) 2 Recorders (1/MF Pump)	Yes	
Emergency Feedwater System							
Emergency Feedwater Flow	1	0 - 900 gpm (Design = 700 gpm)	Yes (4 Channels) (1/Leg)	1E	4 Indicators (1/Channel)	Yes	
Condensate Storage Tank Level	1	---	---	---	---	---	See previous listing
Containment Cooling Systems							
Containment Spray Flow	1	---	---	---	---	---	See previous listing
Heat Removal by the Containment Fan Heat Removal System	3	On-Off	N/A	DG	SPDS	Yes	Cooler Service Water Flow
Containment Atmosphere Temperature	3	0 - 300 °F	N/A	OP/UPS	SPDS	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE “D” VARIABLES (continued)							
Chemical & Volume Control System							
Makeup Flow-In	3	0 - 200 gpm (Design = 149 gpm)	N/A	UPS/DG	1 Indicator	No	
Letdown Flow-Out	3	0 - 160 gpm (Design = 45 gpm)	N/A	UPS/DG	1 Indicator	Yes	
Makeup Tank Level	3	0 - 100"	N/A	UPS/DG	1 Recorder	Yes	
Cooling Water System							
Component Cooling System Flow to ESF System	2	0 - 150 psig	N/A	1E	2 Indicators	Yes	Service Water Header Pressure
	2	Closed/Not Closed	N/A	1E	2 Lights/Valve		Service Water Valve Positions
Radwaste System							
High Level Radioactive Liquid Tank Level	3	0 - 100%	N/A	DG	4 Indicators (1/Tank)	No	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE “D” VARIABLES (continued)							
Ventilation Systems							
Emergency Ventilation Damper Position	2	Closed/Not Closed	N/A	DG	15 Lights	No	1/ea Penetration Vent Valve position. Unit 1 and Unit 2 CRs share a common habitability zone. Refer to Unit 2 RG 1.97 Table for combined information
		Open/Not Open	N/A	DG	4 Lights	No	1/ea containment chiller bypass damper. Valve position
		Open/Not Open	N/A	DG	4 Lights	No	1/ea containment chiller backdraft damper. Valve position
EDG Room Ventilation Damper Position	2	Fully Open/ Not Fully Open	N/A	DG	8 Lights		2 Dampers per EDG 2 Lights per Damper

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE “D” VARIABLES (continued)							
Power Supplies							
Status of Standby Power and Other Energy Sources Important to Safety	2	Voltages; Breaker Position etc.	N/A	Various	SPDS	Yes	
TYPE "E" VARIABLES							
Containment Area Radiation-High Range	1	1R/hr to 10^8 R/hr Gamma	Yes	1E	2 Indicators 1 Recorder	Yes	
Area Radiation							
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)	3	10^{-2} R/hr to 10^3 R/hr; 10^{-4} R/hr to 10 R/hr;	N/A	DG	20 Indicators 2 Displays (Plant Computer and SPDS)	Yes	
Airborne Radioactive Materials Released From Plant							
Containment or Purge Effluent	2	---	---	---	---	---	See previous listing
Auxiliary Bldg (including any bldg containing primary system gases, e.g., waste gas decay tank)	2	---	---	---	---	---	See previous listing

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "E" VARIABLES (continued)							
Airborne Radioactive Materials Released From Plant (continued)							
Condenser Air Removal System Exhaust	2	$1.1E^{-7}$ µCi/cc to $1.3E^{5}$ µCI/cc 0 - 110% Vent Design Flow	N/A	OP	CRT (RDACS)	No	
Common Plant Vent Discharging any of the above Release (if containment purge is included)	2	$1.0E^{-6}$ µCi/cc to $1.0E^{5}$ µCi/cc 0 - 100% Vent Design Flow	N/A	OP	CRT (RDACS)	No	
Vent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves	2	0.1 to 10^{4} mR/hr	N/A	1E	2 Indicators	Yes	
Particulates and Halogens							
All identified plant release points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust)	3	10^{-3} µCi/cc to 10^{2} µCi/cc 0 - 110% Vent Design flow	N/A	OP	CRT (RDACS)	No	See Note 4
Environs Radiation and Radioactivity							
Airborne Radiohalogens and Particulates	3	10^{-9} µCi/cc to 10^{-3} µCi/cc	N/A	N/A	N/A	N/A	Portable Sampling

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "E" VARIABLES (continued)							
Environs Radiation and Radioactivity (continued)							
Plant and Environs Radiation	3	$10E^{-3}$ R/hr to $10E^{3}$ R/hr Photons; $10E^{-3}$ R/hr to 50 R/hr	N/A	N/A	N/A	N/A	Portable Inst
Plant and Environs Radioactivity	3	Isotopic Analysis	N/A	N/A	N/A	N/A	
Meteorology							
Wind Direction	3	0 - 360° ± 1/2% full scale; starting speed - .75 mph; damping radio - .6; distance constant -1m.	N/A	OP	1 Recorder CRT (RDACS)	Yes	
Wind Speed	3	0 - 100 mph accuracy greater of ± 1% or ± 0.15 mph; starting threshold 0.6 mph; distance constant 5 ft.	N/A	OP	1 Recorder CRT (RDACS)	Yes	

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE “E” VARIABLES (continued)							
Meteorology (continued)							
Estimation of Atmospheric Stability	3	-5 to 5 °C (Temp. Difference;) 0 - 100 Degrees Wind Direction Sigma Theta	N/A	OP	1 Dual Pen Recorder CRT (RDACS)	Yes	
Accident Sampling Capability (Analysis Capability Onsite)							
Primary Coolant and Sump • Gross Activity 1 μCi/ml to 10 Ci/ml • Gamma Spectrum (isotopic analysis) • Boron Concentration 0 - 6000 ppm • Chloride content • Dissolved hydrogen or Total Gas • Dissolved Oxygen	3	---	N/A	N/A	N/A	N/A	See Note 1

Table 7-11A (continued)

	CATEGORY	RANGE	REDUNDANCY	POWER SUPPLY	CR DISPLAY	SPDS	COMMENTS
TYPE "E" VARIABLES (continued)							
Accident Sampling Capability (Analysis Capability Onsite) (continued)							
Primary Coolant and Sump (continued)							
• pH 1-13 (Analysis Capability Offsite, 0CNA019306)							
Containment Air Grab Sample	3	--	N/A	N/A	N/A	N/A	See Note 1
• Hydrogen Content 0 - 10% Vol.							
• Oxygen Content							
• Gamma Spectrum (Isotopic Analysis)							

* Subcooled Margin Monitor

NOTE 1: Parameter not required during events involving > 5% clad failure. Relief from maintaining a post accident sampling system (PASS) was obtained in ANO-1 Technical Specification Amendment 208 (2CNA080005). As part of the relief approval, RG 1.97 and NUREG 0737 requirements concerning PASS capability are no longer applicable. However, sampling capability must continue to be maintained for those fuel accident events involving ≤ 5% clad failure. This capability exists via non-PASS sampling locations and systems. Contingency plans must also be maintained to support RCS and containment atmosphere sampling efforts under post-accident conditions, assuming sample location dose rates are such that samples can be safely obtained and analyzed. These contingency plans are contained within various Emergency Plan, Chemistry, Radiological and Operating procedures, including the Severe Accident Management Guidelines. It should be noted, however, that sampling intended to support fuel damage assessment is no longer required for RCS chlorides, RCS oxygen, RCS dissolved hydrogen (total gas), or containment atmosphere oxygen content.

Table 7-11A (continued)

NOTE 2: Reactor building isolation valves listed in Table 5-1 of the ANO-1 SAR were evaluated. This evaluation excluded check valves, locked closed manual valves which are part of a passive boundary, and valves which are locked closed and administratively controlled shut. Redundancy is satisfied by GDC 55, 56, or 57.

NOTE 3: Category 3 instrumentation for beyond design basis accidents only (reference October 2003 revision to 10 CFR 50.44, ANO Commitment #17973).

NOTE 4: Each Super Particulate Iodine Noble Gas (SPING) monitor which inputs to the Radiological Dose Assessment Computer System has the capability to measure halogen and particulate activities as it is accumulated on a sample media. The SPING microcomputer then calculates the gross radiohalogen and particulate sample concentration based on the rate of increase of activity on the filter media. If necessary, to further define the analysis or to extend the range, an isotopic analysis of the filter media can be performed by plant radiochemistry personnel.

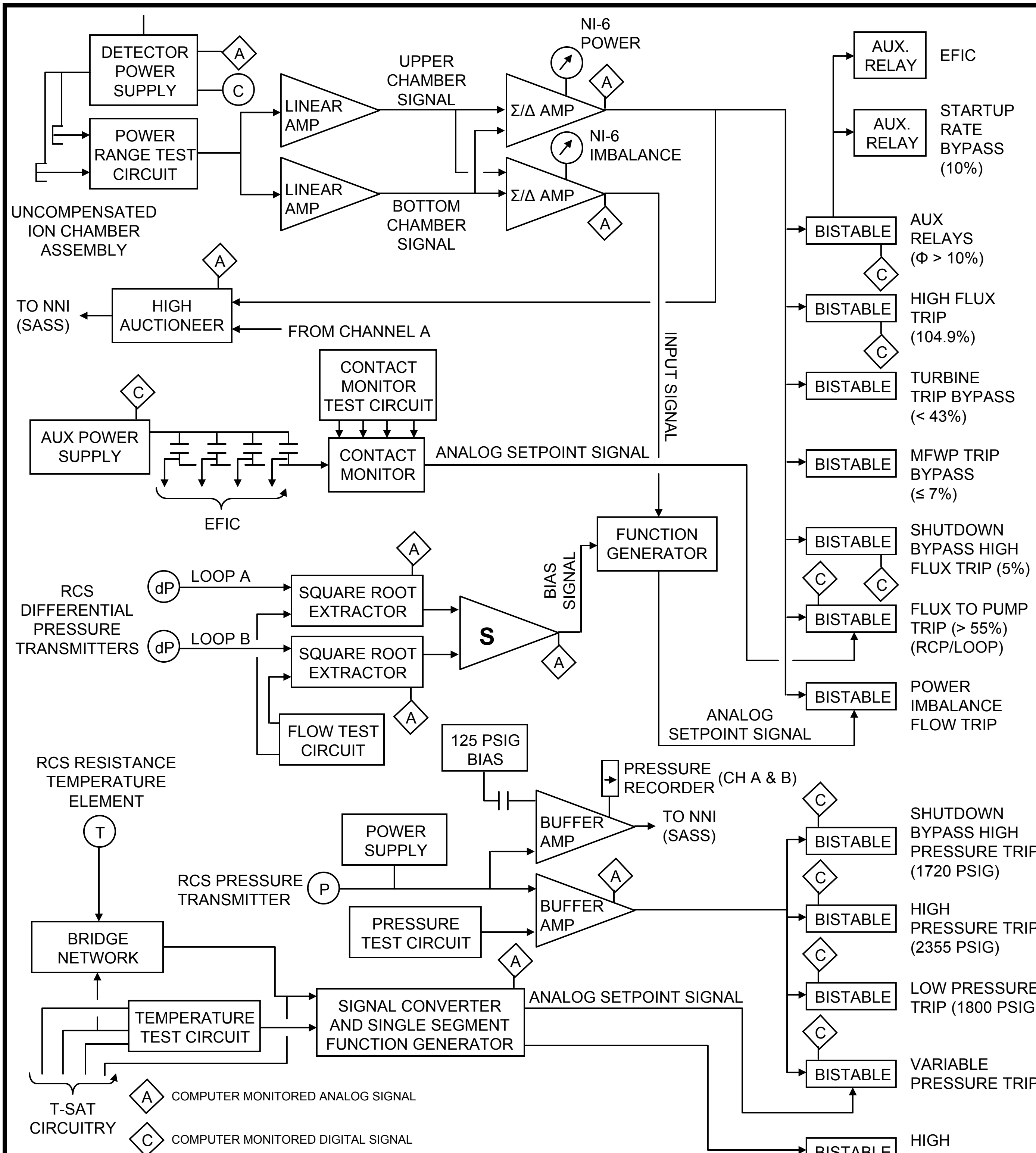
NOTE 5: DELETED

NOTE 6: DELETED

Table 7-13

RELATIONSHIP BETWEEN TESTED TRIP RESPONSE TIMES AND TRIP RESPONSE TIMES ASSUMED IN THE SAFETY ANALYSIS

	RPS Trip Response Time Assumed in Safety Analysis (msec)	Tested Trip Response Time (msec)	Margin (msec)
High Reactor Coolant Pressure	150	84	66
Low Reactor Coolant Pressure	150	81	69
Pressure/Temperature (decreasing pressure)	150	82	68
Reactor Building Pressure	150	50	100
Reactor Outlet Temperature	150	130	20
Overpower	150	75	75
Power/Imbalance/Flow (increasing power)	240	80	160
Power/Imbalance/Flow (decreasing flow)	240	145	95
Power/Imbalance/Flow (increasing imbalance)	240	80	160
Power/RC Pumps	150	40	110



ANALOG LOGIC CHANNEL B
(TYPICAL OF CHANNEL B & C)

SAR FIGURE NO. 7-1

AMENDMENT 20

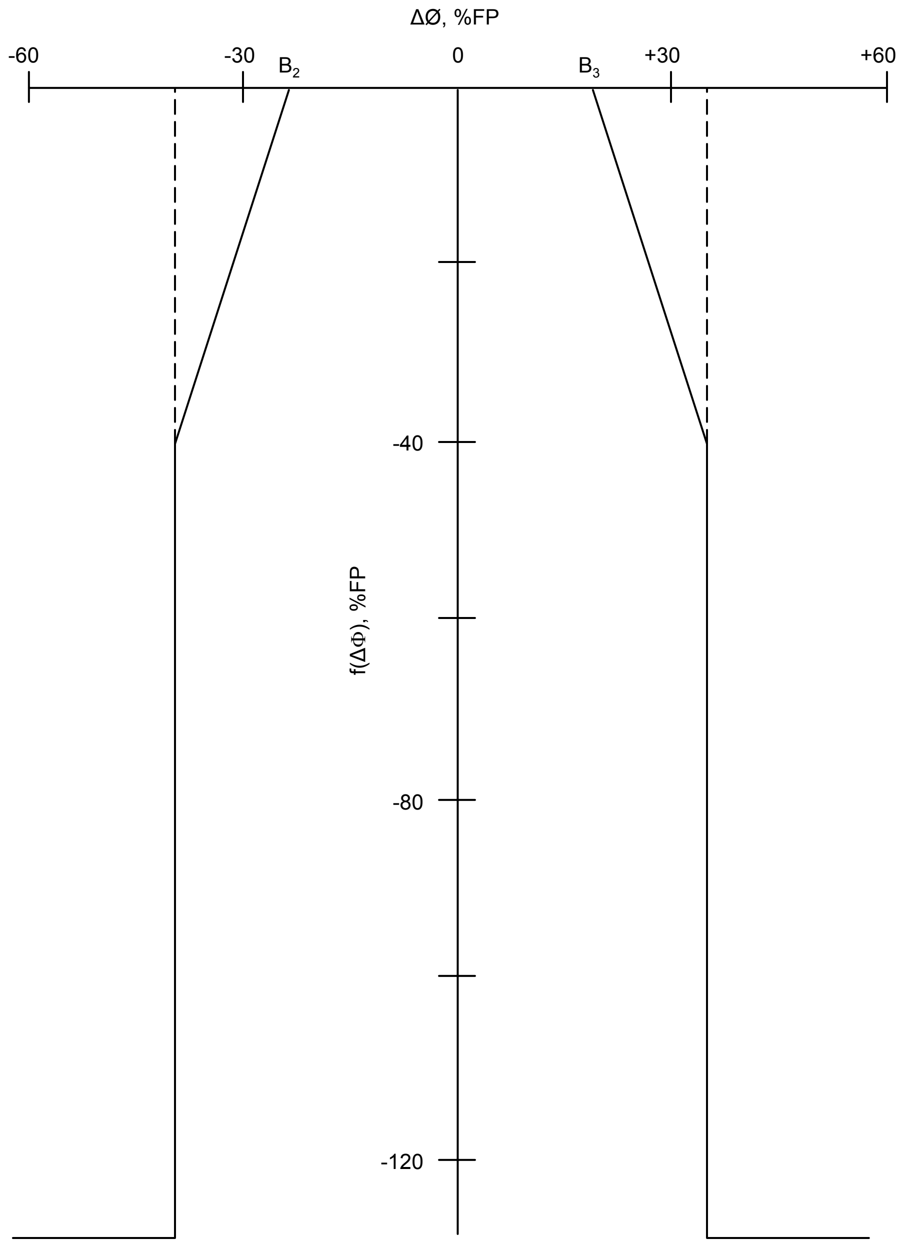
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

REACTOR PROTECTION SYSTEM

BASED ON DRAWING NO	SHEET	REV.



NOTE:

THIS IS AN EXAMPLE OF THE IMBALANCE FUNCTION. THIS EXAMPLE IS REPRESENTATIVE ONLY, WITH THE ACTUAL SETPOINT FOR THE TRIP DEFINED BY THE CURRENT CYCLE CORE OPERATING LIMITS REPORT

SAR FIGURE NO. 7-2

AMENDMENT 20

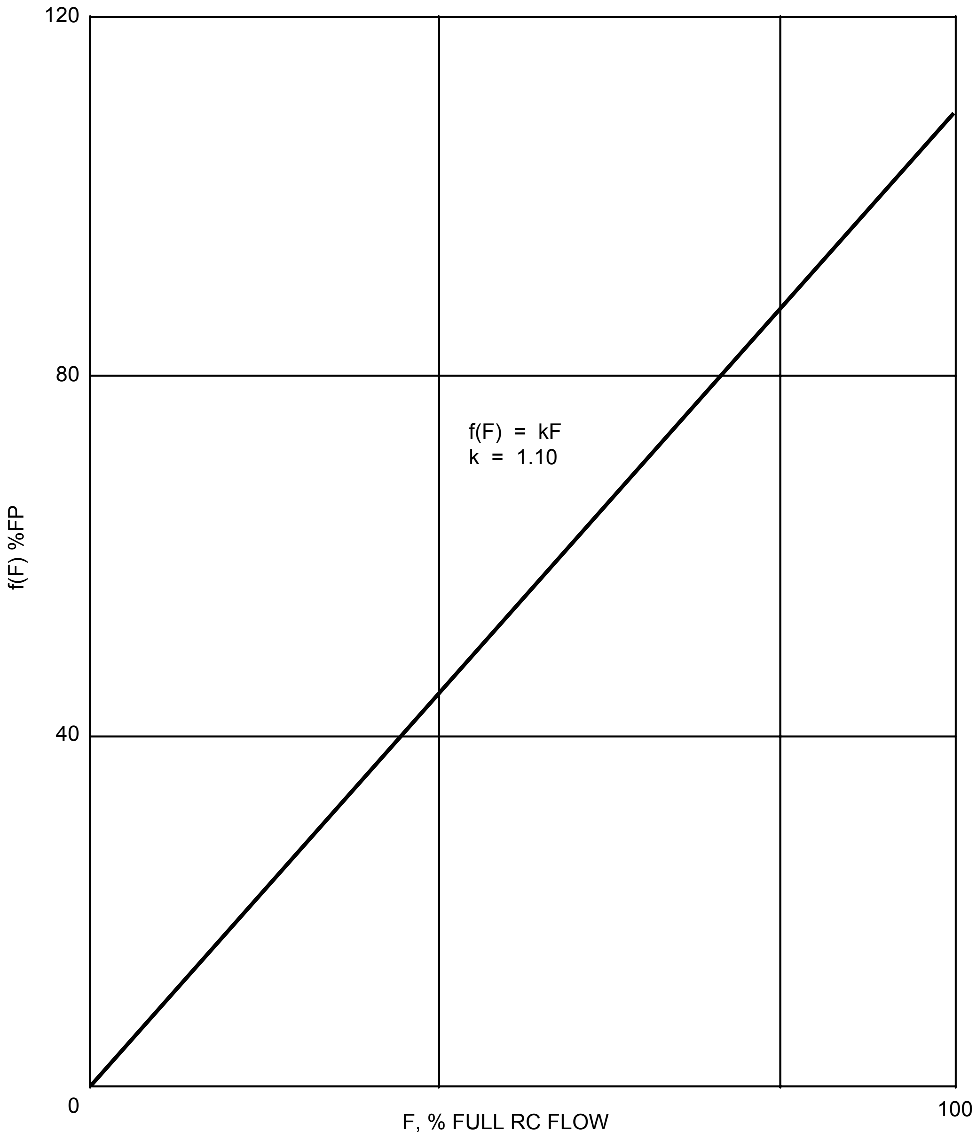
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

IMBALANCE FUNCTION f($\Delta\Phi$)

BASED ON DRAWING NO	SHEET	REV.
	1	



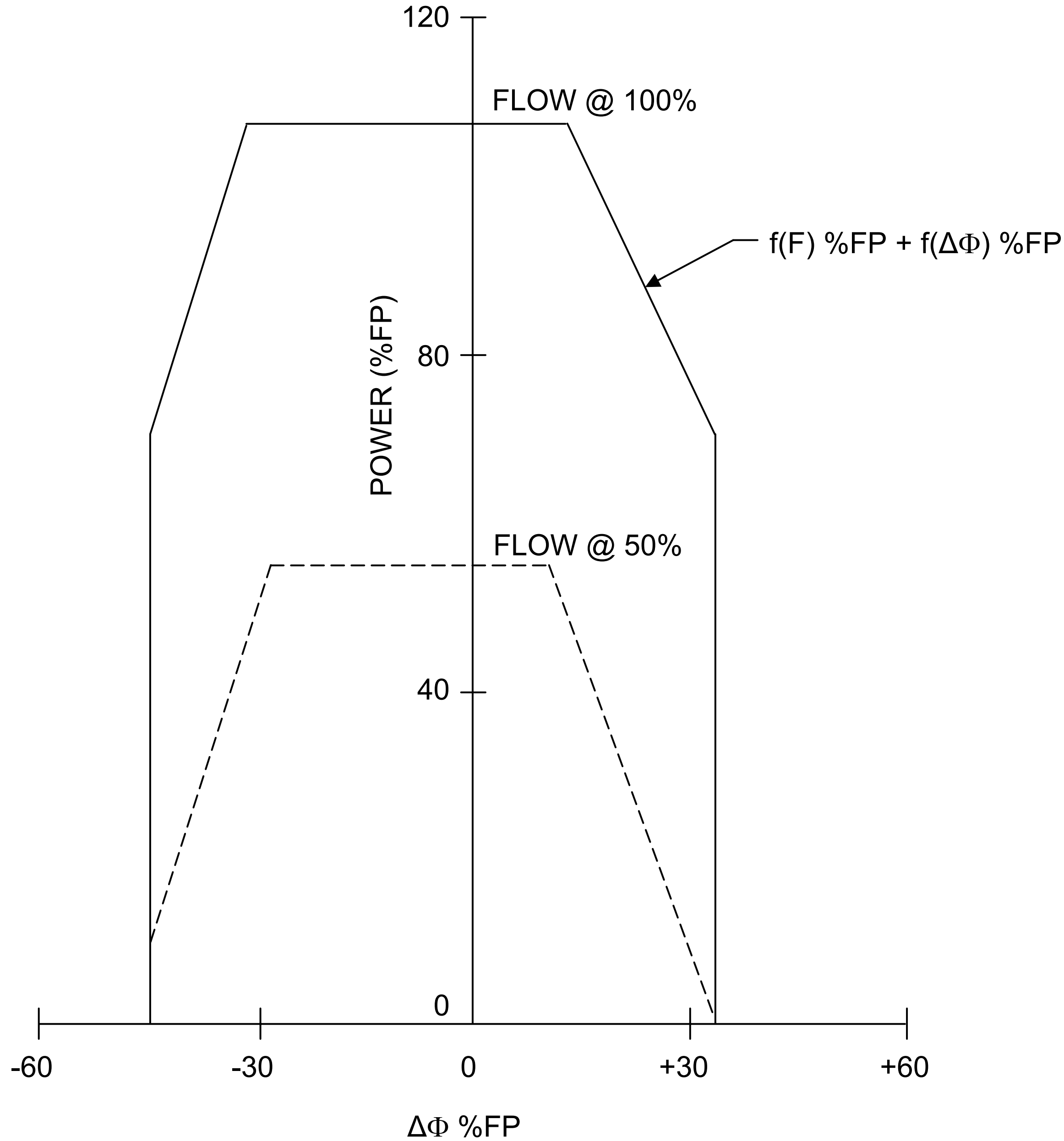
NOTE:

THIS IS AN EXAMPLE OF THE FLOW FUNCTION. THIS EXAMPLE IS REPRESENTATIVE ONLY, WITH THE ACTUAL SETPOINT FOR THE TRIP DEFINED BY THE CURRENT CYCLE CORE OPERATING LIMITS REPORT.

SAR FIGURE NO. 7-3

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
FLOW FUNCTION f(F)	BASED ON DRAWING NO	SHEET 1 REV.



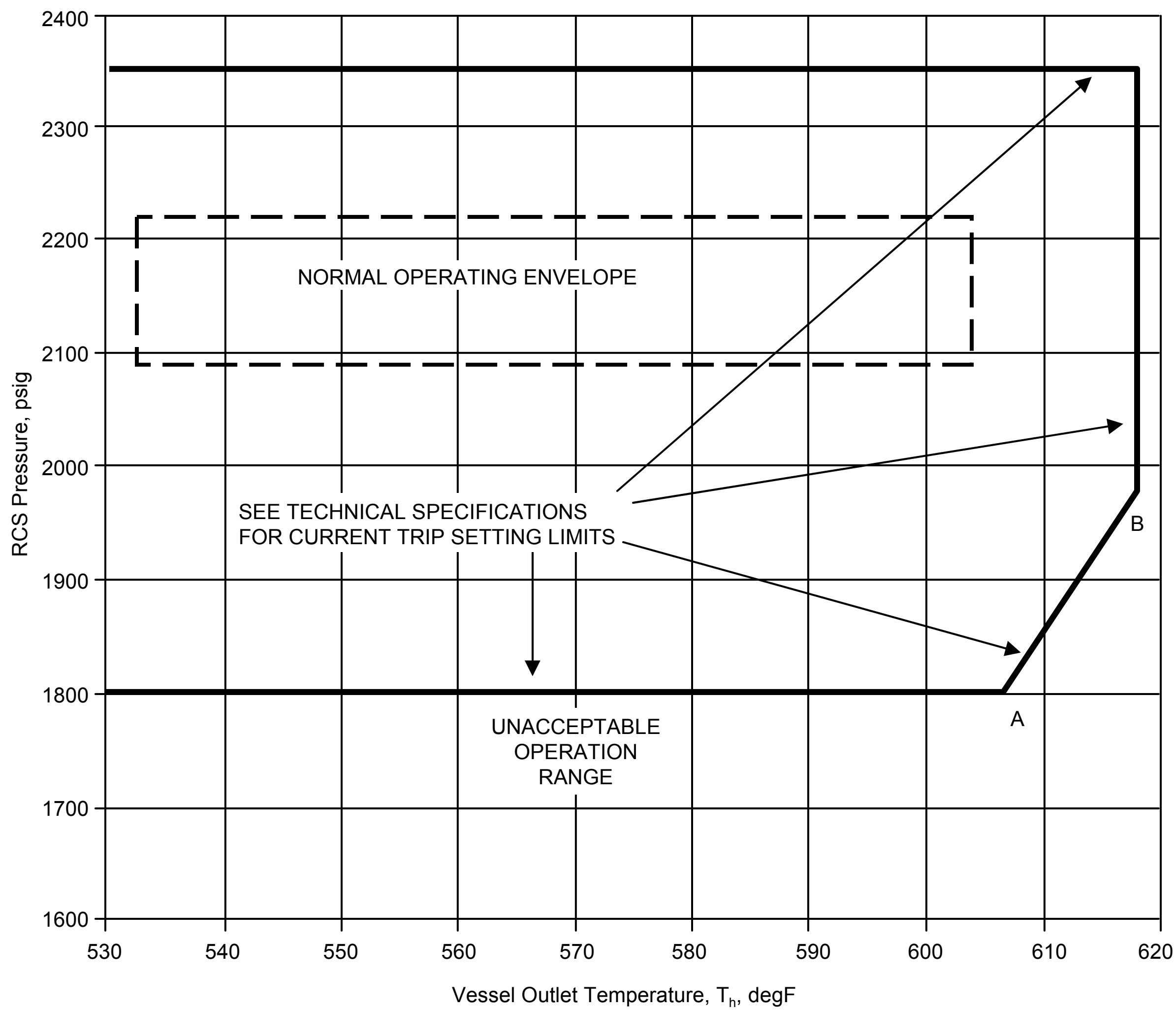
NOTE:

THIS IS AN EXAMPLE OF A POWER TRIP BASED ON IMBALANCE AND FLOW. THIS EXAMPLE IS REPRESENTATIVE ONLY, WITH THE ACTUAL SETPOINT FOR THE TRIP DEFINED BY THE CURRENT CYLCE CORE OPERATING LIMITS REPORT.

SAR FIGURE NO. 7-4

AMENDMENT 20

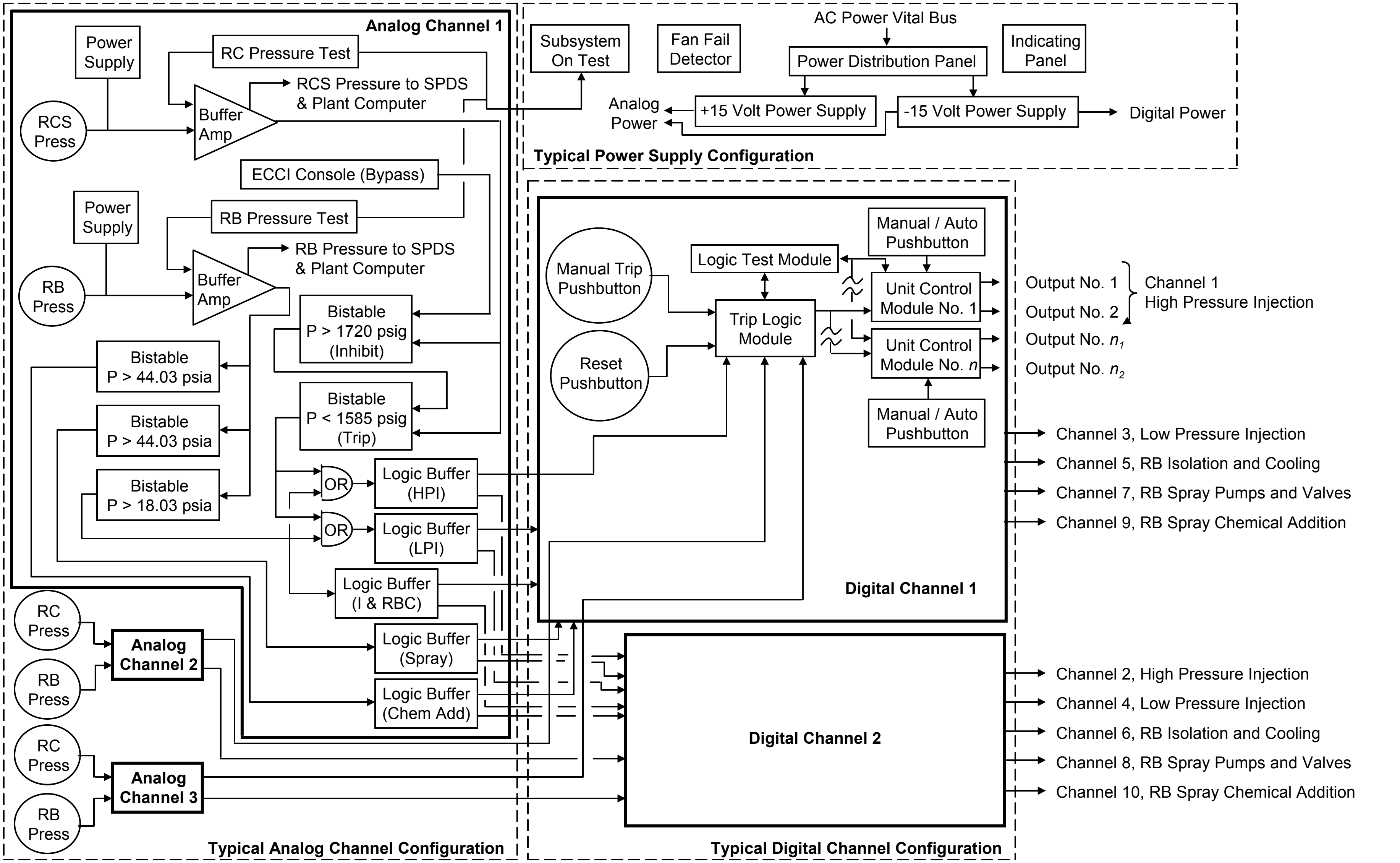
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
POWER TRIP BASED ON IMBALANCE AND FLOW FUNCTIONS	BASED ON DRAWING NO	SHEET: 1	REV.



SAR FIGURE NO. 7-5

AMENDMENT 20

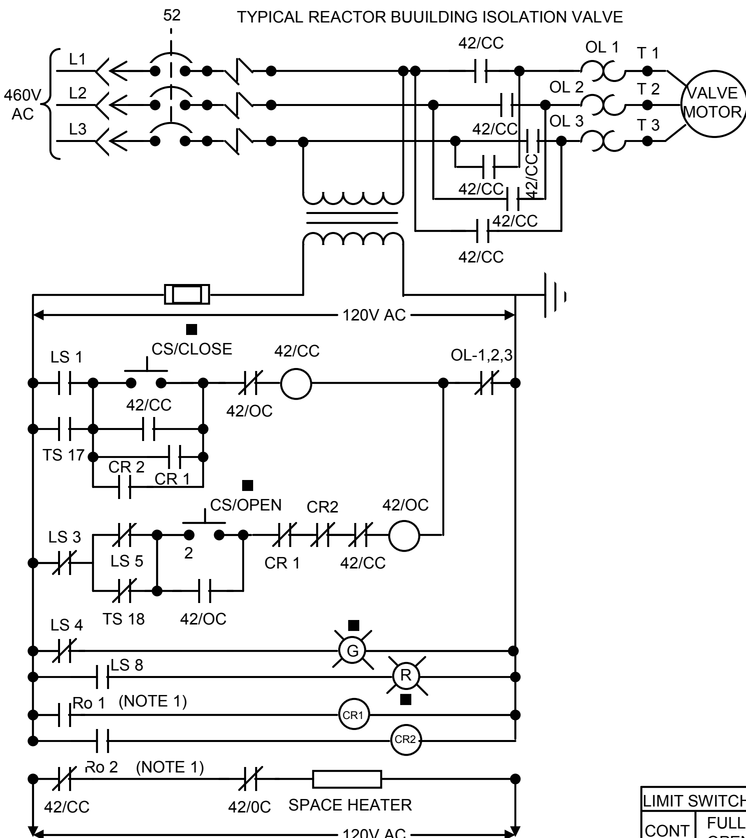
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
RCS PRESSURE – TEMPERATURE BOUNDARIES FOR ANO-1 (SEE TECHNICAL SPECIFICATIONS FOR CURRENT TRIP SETTING LIMITS)	BASED ON DRAWING NO	SHEET 1 / REV.

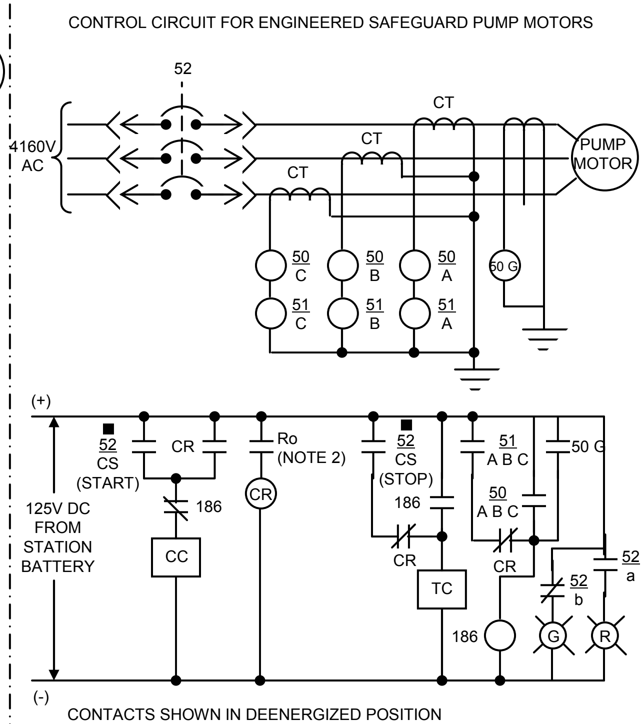


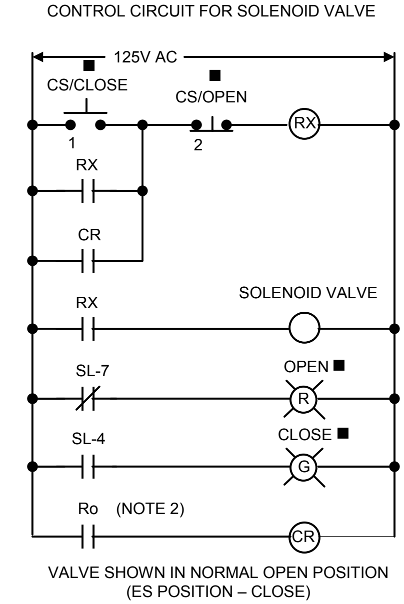
ENGINEERED SAFEGUARDS ACTUATION SYSTEM BLOCK DIAGRAM	SAR FIGURE NO. 7-6
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

SCALE:	NONE	BASED ON DRAWING NO	SHEET	REV.
DRAWN:	ENTERGY			
DESIGN:	ENTERGY			
CAD NO:				

Entergy







■ CONTROL SWITCH (ES)

OPEN CLOSE — SPRING RETURN TO NEUTRAL FROM OPEN AND CLOSE POSITION

CONTACTS	POSITION		
	OPEN	NEU.	CLOSE
1			X
2	X		
3			
4			

X – INDICATES CONTACT IS CLOSED

LIMIT SWITCH CONTACT DEVELOPMENT

CONT	FULL OPEN	FULL CLOSE	FUNCTION
LS1			TS17 BYPASS
LS2			ES IND.
LS3			OPEN LIMIT
LS4			INDICATION
LS5			TS18 BYPASS
LS6			ES IND.
LS7			SPARE
LS8			INDICATION

LIMIT SWITCH CONTACT DEVELOPMENT

CONT	FULL OPEN	FULL CLOSE	FUNCTION
LS9			SPARE
LS10			SPARE
LS11			SPARE
LS12			SPARE
LS13			SPARE
LS14			SPARE
LS15			SPARE
LS16			ISPARE
TS17			CLOSE TORQ SW.
TS18			OPEN TORQ SW.

DENOTES CONTACT IS CLOSED

CR – CONTROL RELAY
Ro – FROM SAFEGUARDS ACTUATION SYSTEM
CC – CIRCUIT BREAKER CLOSING CIRCUIT
TC – CIRCUIT BREAKER TRIP CIRCUIT
50 – INSTANTANEOUS OVERCURRENT RELAY
51 – TIME OVERCURRENT RELAY
52 CS – CIRCUIT BREAKER CONTROL SWITCH
186 – AUXILIARY TRIPPING RELAY
50G – GROUND DETECTOR RELAY

NOTES:
1. Ro1 and Ro2 from Safeguards Actuation System
2. Ro from Safeguards Actuation System
3. ■ Device Located at Main Console

■ CONTROL SWITCH (ES)

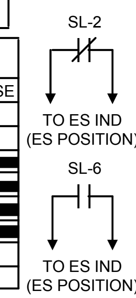
OPEN CLOSE — SPRING RETURN TO NEUTRAL FROM OPEN AND CLOSE POSITION

CONTACTS	POSITION		
	OPEN	NEU.	CLOSE
1			X
2		X	X

AIR OPERATED VALVE LIMIT SWITCH POSITION

SW	OPEN	INTERMEDIATE	CLOSE
SL1			
SL2			
SL3			
SL4			
SL5			
SL6			
SL7			
SL8			

DENOTES CONTACT IS CLOSED



TYPICAL CONTROL CIRCUITS ENGINEERED SAFEGUARDS EQUIPMENT

SAR FIGURE NO. 7-7

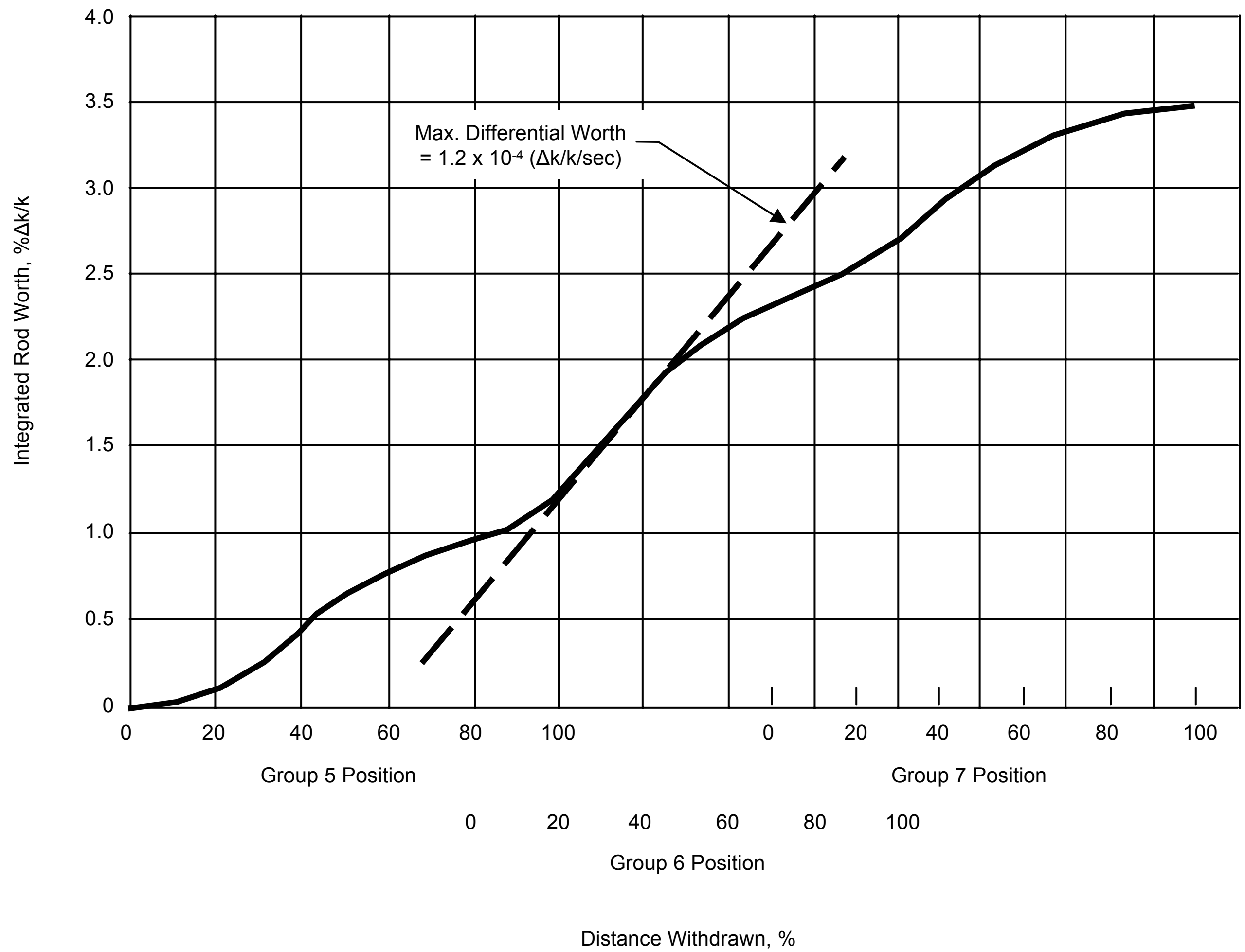
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

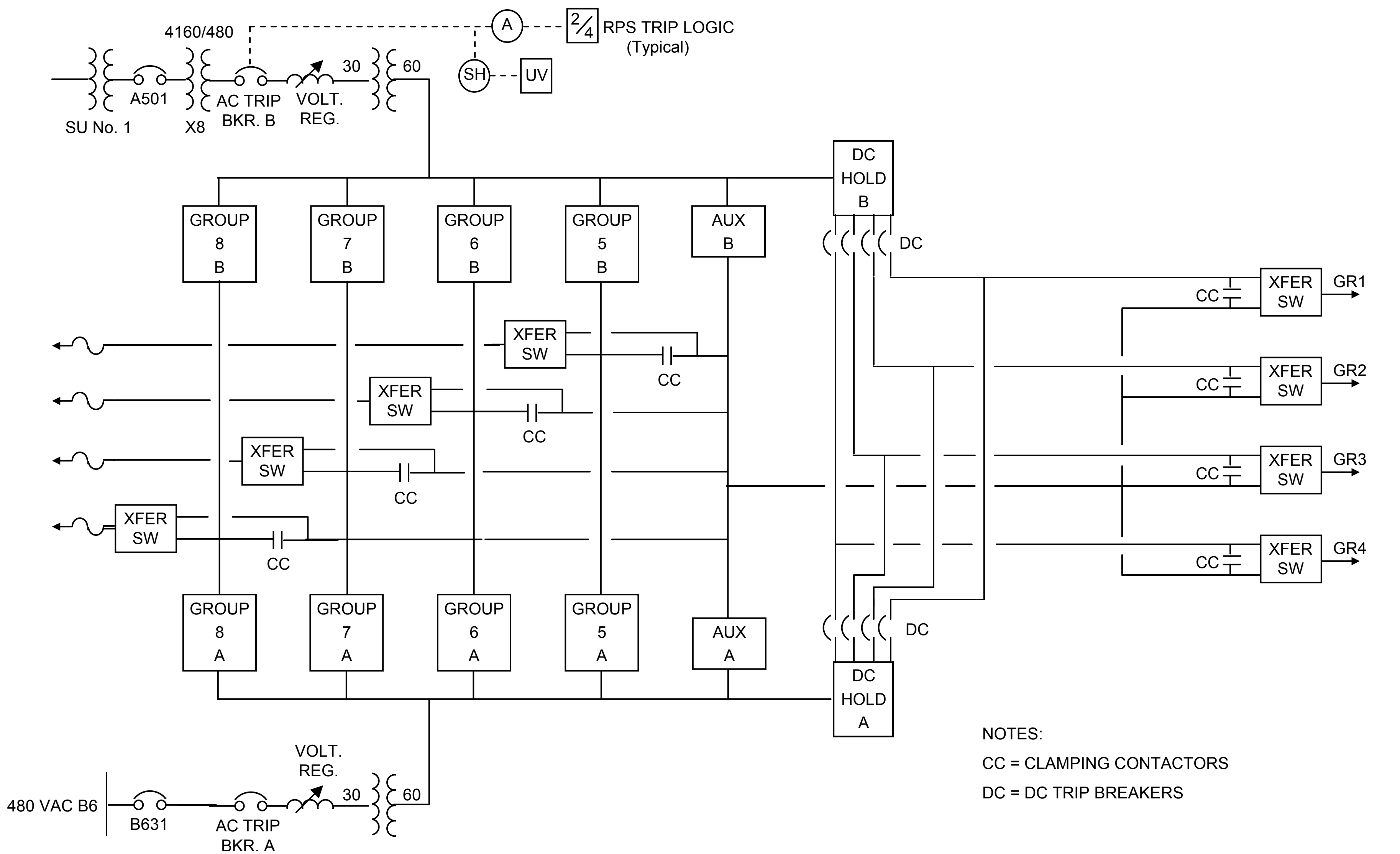
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



AUTOMATIC CONTROL ROD GROUP – TYPICAL WORTH CURVE VERSUS DISTANCE WITHDRAWN			SAR FIGURE NO. 7-8
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



NOTES:

CC = CLAMPING CONTACTORS

DC = DC TRIP BREAKERS

ROD DRIVE CONTROLS		SAR FIGURE NO. 7-9
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

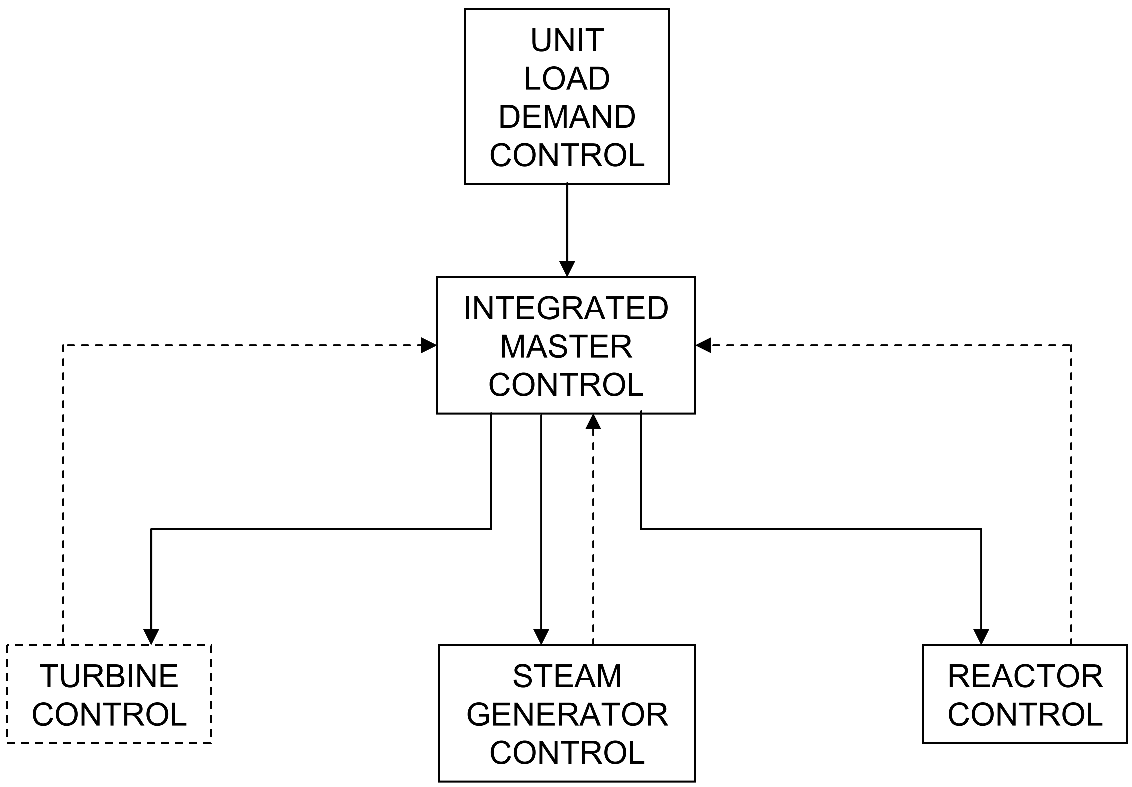
DELETED

(See Figure 7-9)

SAR FIGURE NO. 7-10

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
CONTROL ROD DRIVE SYSTEM AND TRIP BLOCK DIAGRAM	BASED ON DRAWING NO	SHEET 1 / REV.



NOTES:

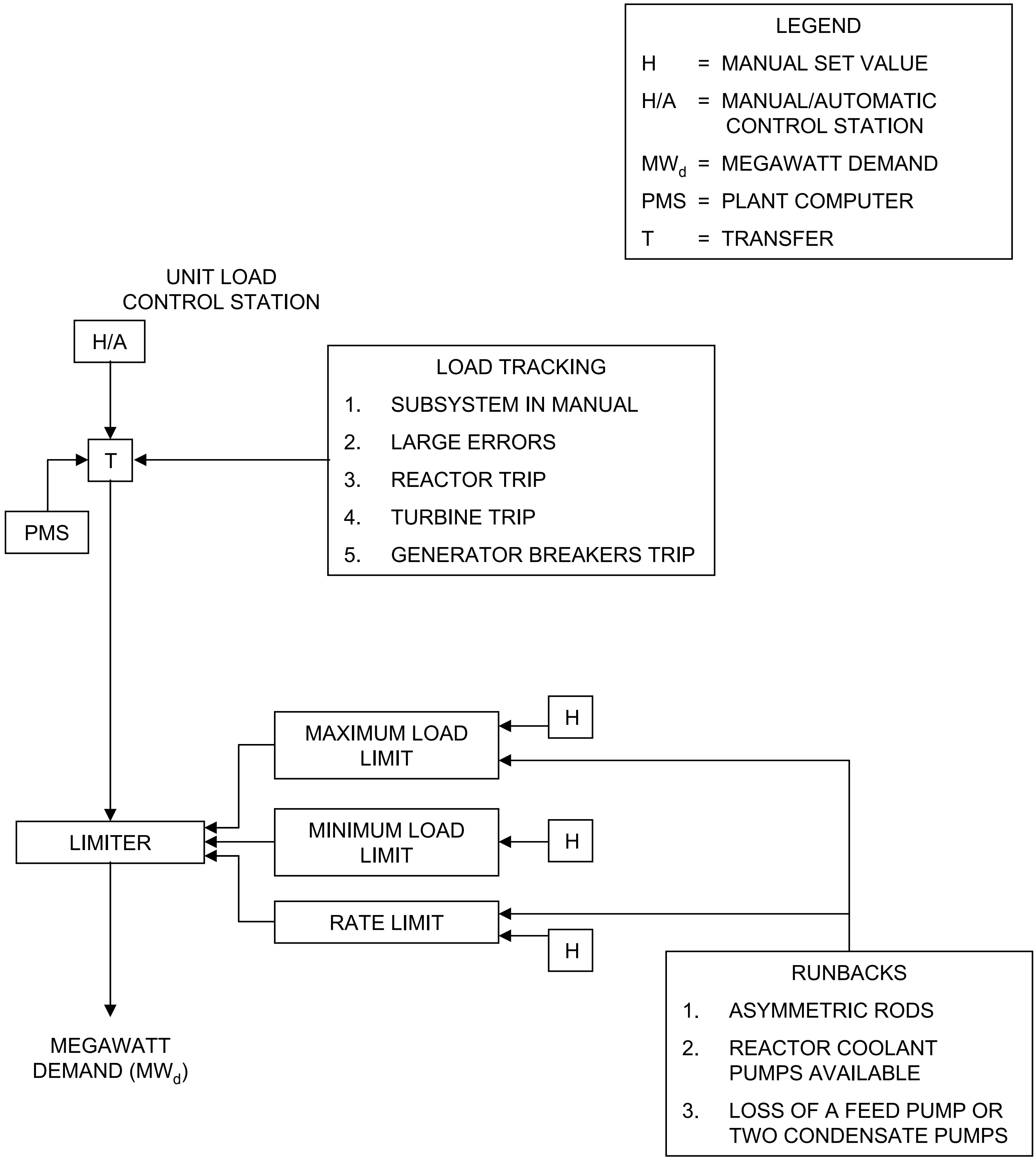
1. TURBINE CONTROL IS FURNISHED WITH THE TURBINE EQUIPMENT

SAR FIGURE NO. 7-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

INTEGRATED CONTROL SYSTEM	BASED ON DRAWING NO	SHEET	REV.
		1	



SAR FIGURE NO. 7-12

AMENDMENT 20

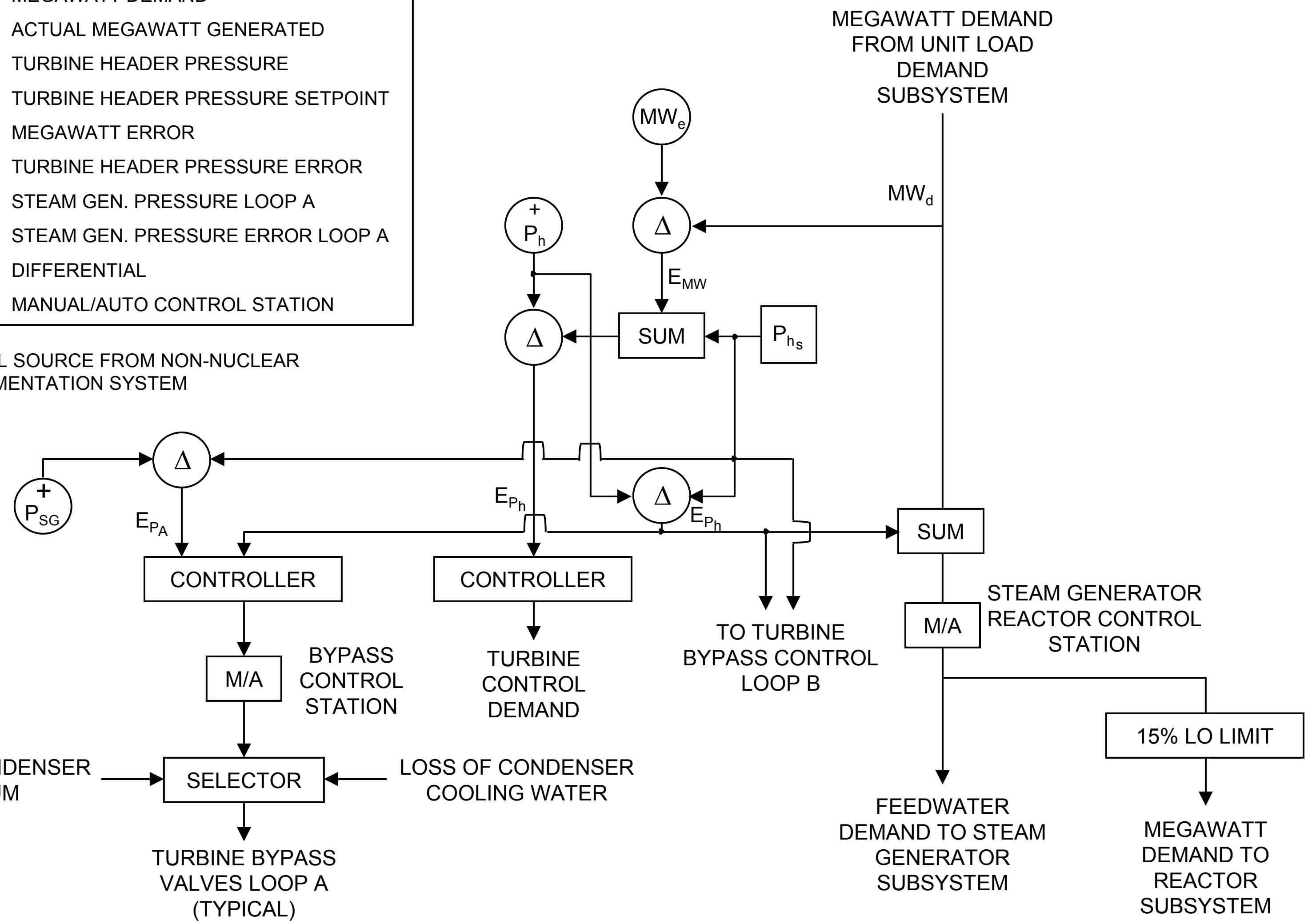
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

UNIT LOAD DEMAND CONTROL INTEGRATED CONTROL SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

LEGEND

MW_d	=	MEGAWATT DEMAND
MW_e	=	ACTUAL MEGAWATT GENERATED
P_h	=	TURBINE HEADER PRESSURE
P_{h_s}	=	TURBINE HEADER PRESSURE SETPOINT
E_{MW}	=	MEGAWATT ERROR
E_{P_h}	=	TURBINE HEADER PRESSURE ERROR
P_{SG}	=	STEAM GEN. PRESSURE LOOP A
E_{P_A}	=	STEAM GEN. PRESSURE ERROR LOOP A
Δ	=	DIFFERENTIAL
M/A	=	MANUAL/AUTO CONTROL STATION

1. + SIGNAL SOURCE FROM NON-NUCLEAR INSTRUMENTATION SYSTEM



INTEGRATED MASTER CONTROL INTEGRATED CONTROL SYSTEM

SAR FIGURE NO. 7-13

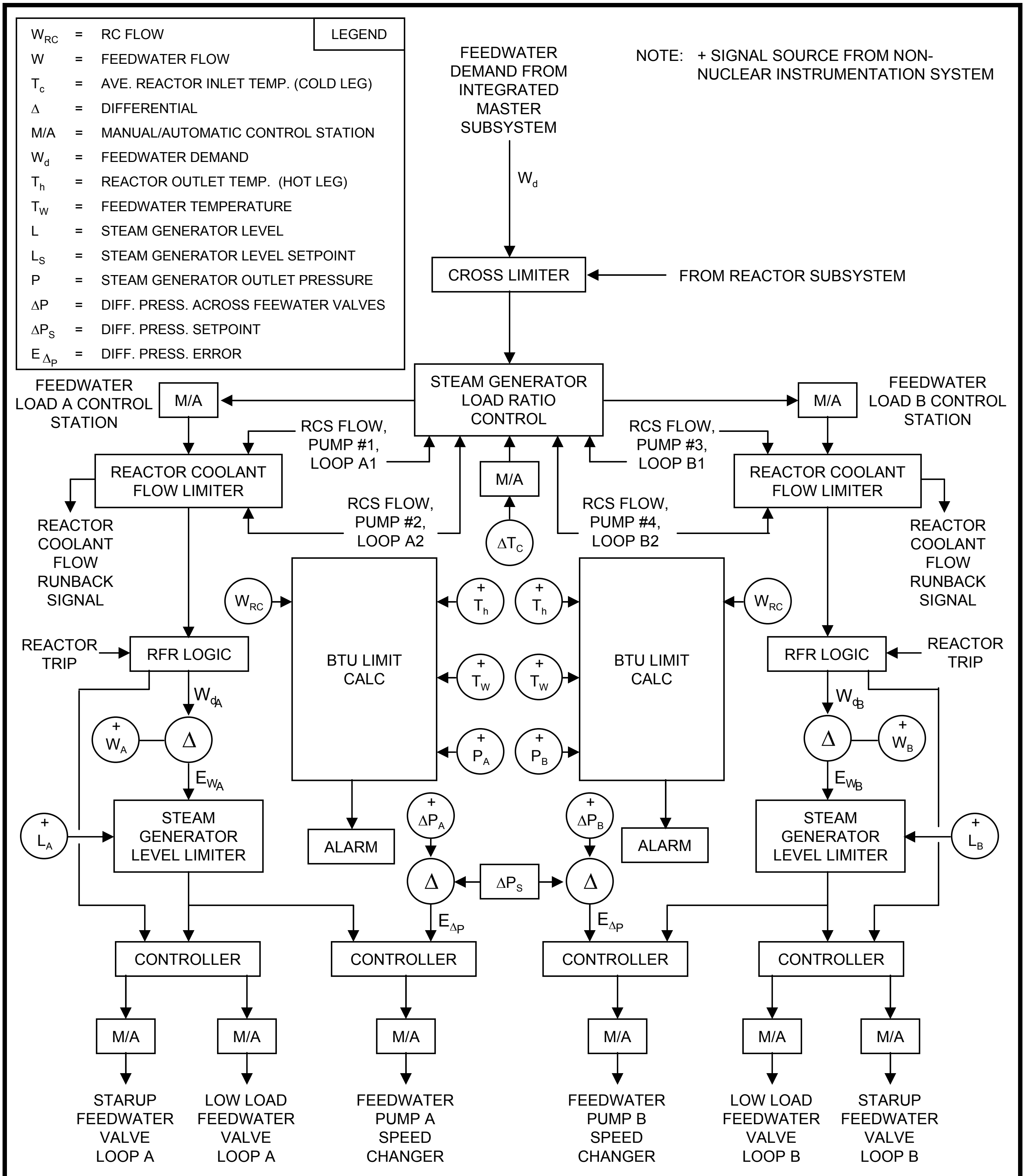
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 7-14

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STEAM GENERATOR CONTROL INTEGRATED CONTROL SYSTEM	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

SAR FIGURE NO. 7-15

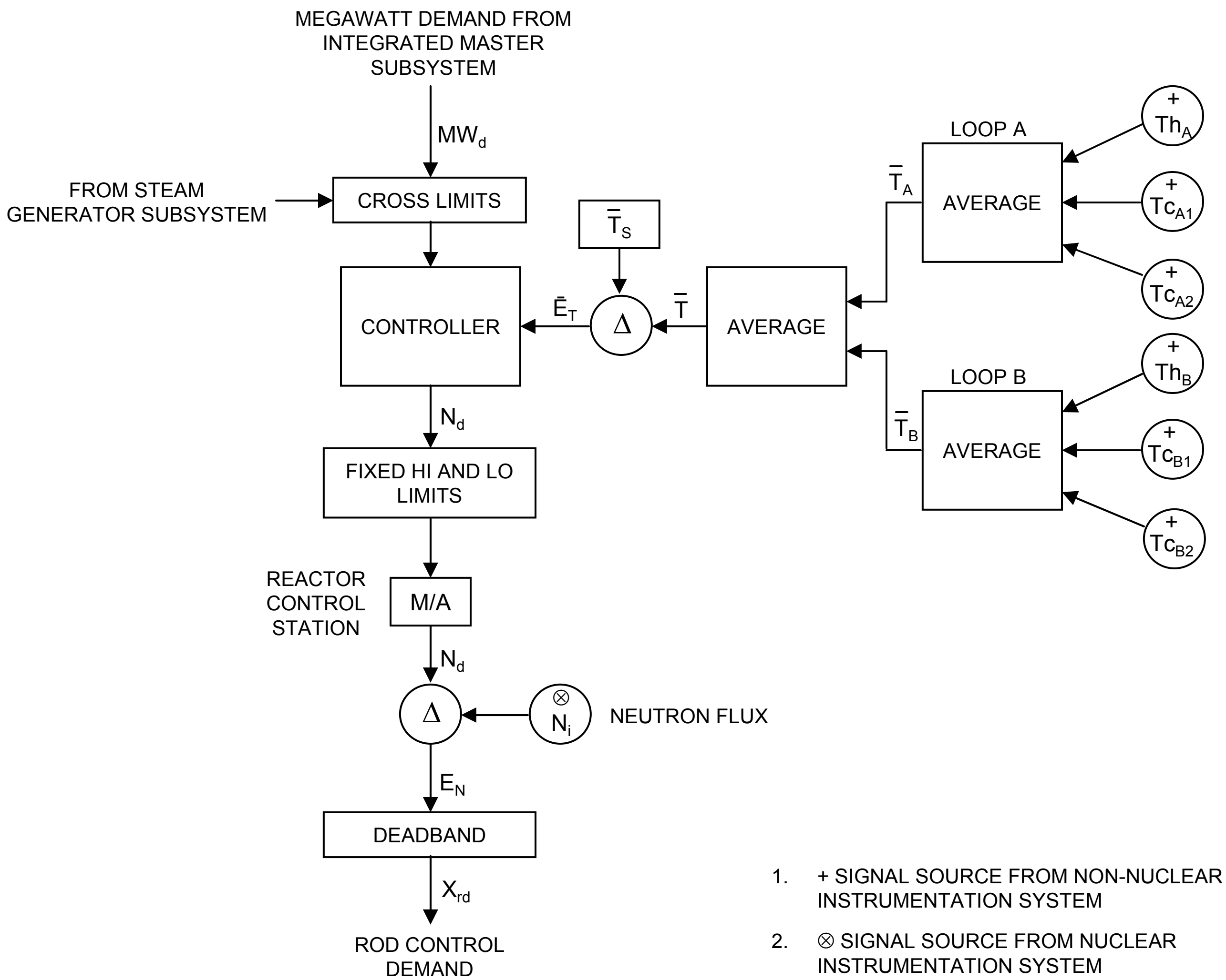
AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

REACTOR AND STEAM TEMPERATURES VS. REACTOR POWER	BASED ON DRAWING NO	SHEET 1	REV.

LEGEND

$\overline{T}_S$	=	AVERGE TEMPERATURE SETPOINT
Tc	=	RCS COLD LEG TEMPERATURE
Th	=	RCS HOT LEG TEMPERATURE
$\overline{T}$	=	REACTOR AVERAGE COOLANT TEMPERATURE
$E_{\overline{T}}$	=	DEVIATION OF AVERAGE TEMPERATURE FROM SETPOINT
Δ	=	DIFFERENTIAL
M/A	=	MANUAL AUTOMATIC STATION
N_{dl}	=	REACTOR POWER LEVEL DEMAND
N_i	=	REACTOR POWER LEVEL
E_N	=	REACTOR POWER LEVEL ERROR
X_{rd}	=	ROD CONTROL DEMAND



SAR FIGURE NO. 7-16

AMENDMENT 20

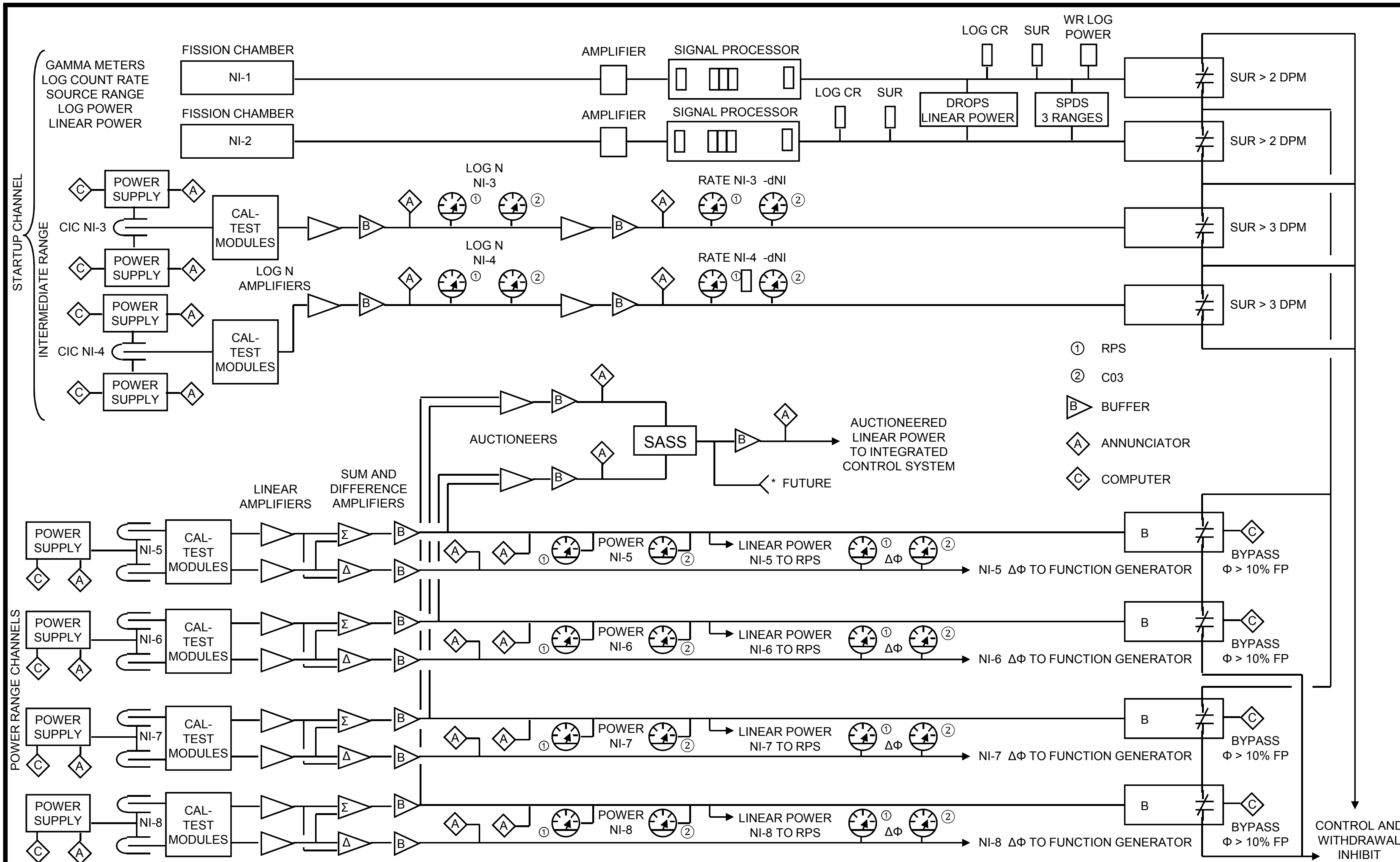
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

REACTOR CONTROL – INTEGRATED CONTROL SYSTEM

BASED ON DRAWING NO	SHEET	REV.
	1	



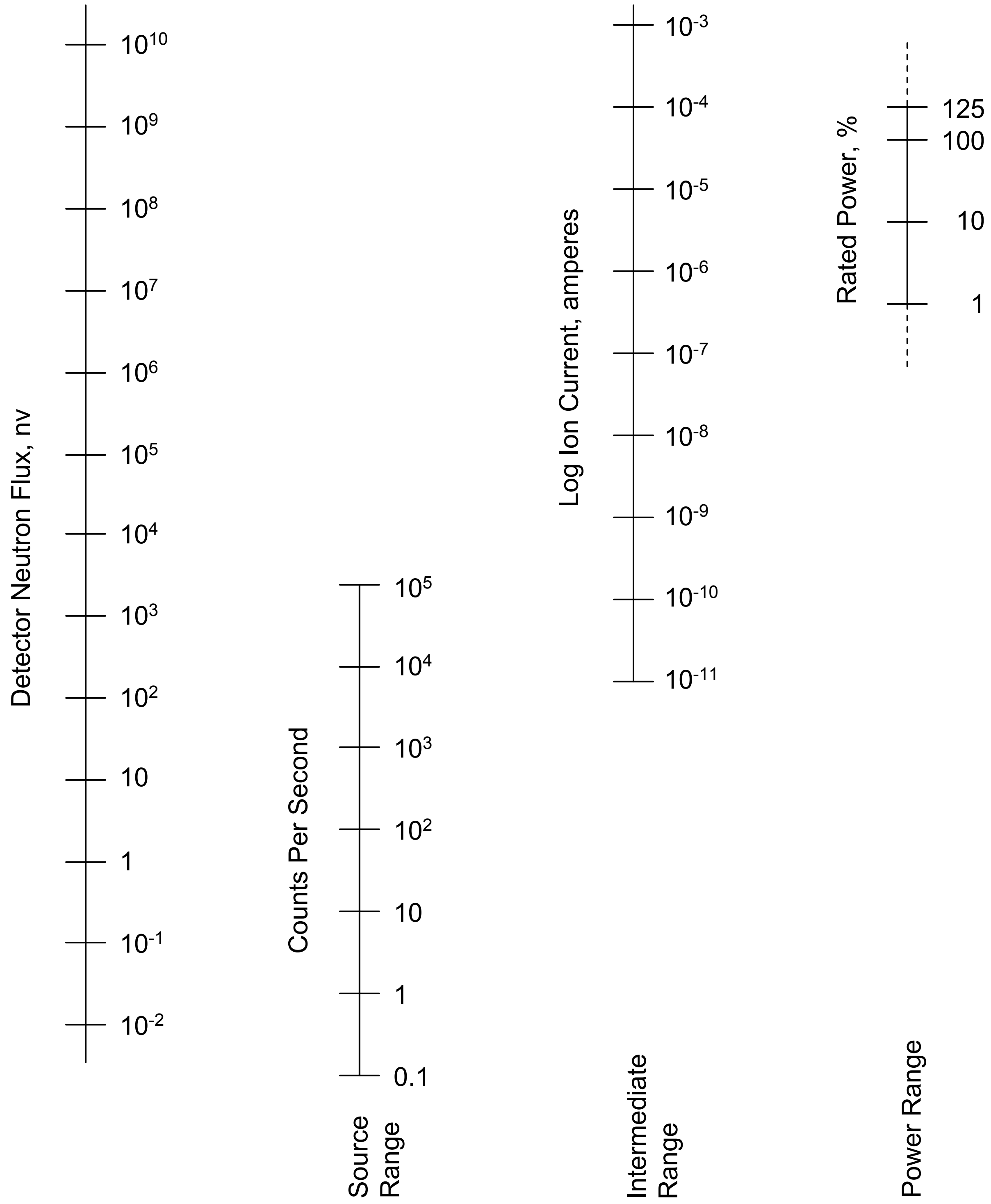
NUCLEAR INSTRUMENTATION SYSTEM	SAR FIGURE NO. 7-17
	AMENDMENT 28

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

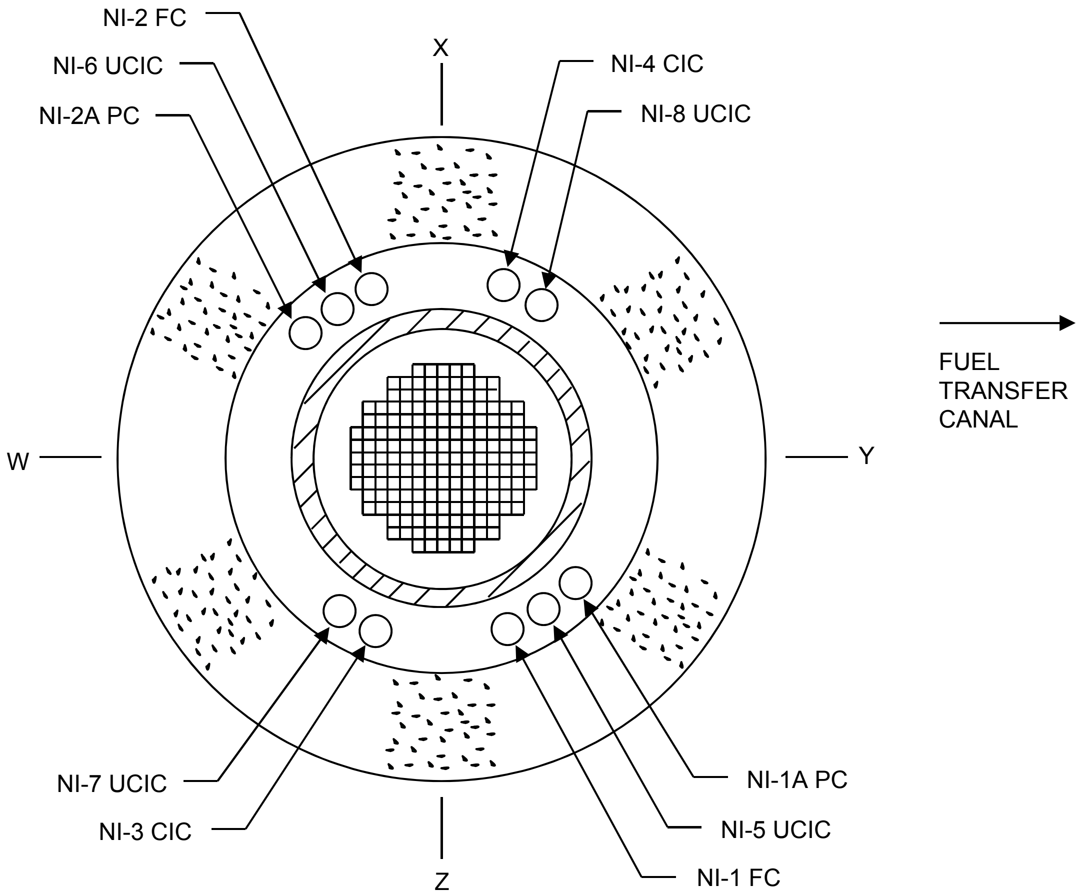
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 7-18

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
NUCLEAR INSTRUMENTATION FLUX RANGES	BASED ON DRAWING NO	SHEET 1 / REV.



LEGEND

PC* - PROPORTIONAL COUNTER – SOURCE RANGE DETECTOR

CIC - COMPENSATED ION CHAMBER – INTERMEDIATE RANGE DETECTOR

UCIC - UNCOMPENSATED ION CHAMBER – POWER RANGE DETECTOR

FC - SOURCE RANGE FISSION CHAMBER

* Original detection system; the high sensitivity fission chambers are the primary operating source range system.

SAR FIGURE NO. 7-19

AMENDMENT 28

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
NUCLEAR INSTRUMENTATION DETECTOR LOCATION	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

(REFER TO FIGURE 4-1 Sht 1)

SAR FIGURE NO. 7-20
AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
PIPING & INSTRUMENT DIAGRAM REACTOR COOLANT SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

DELETED

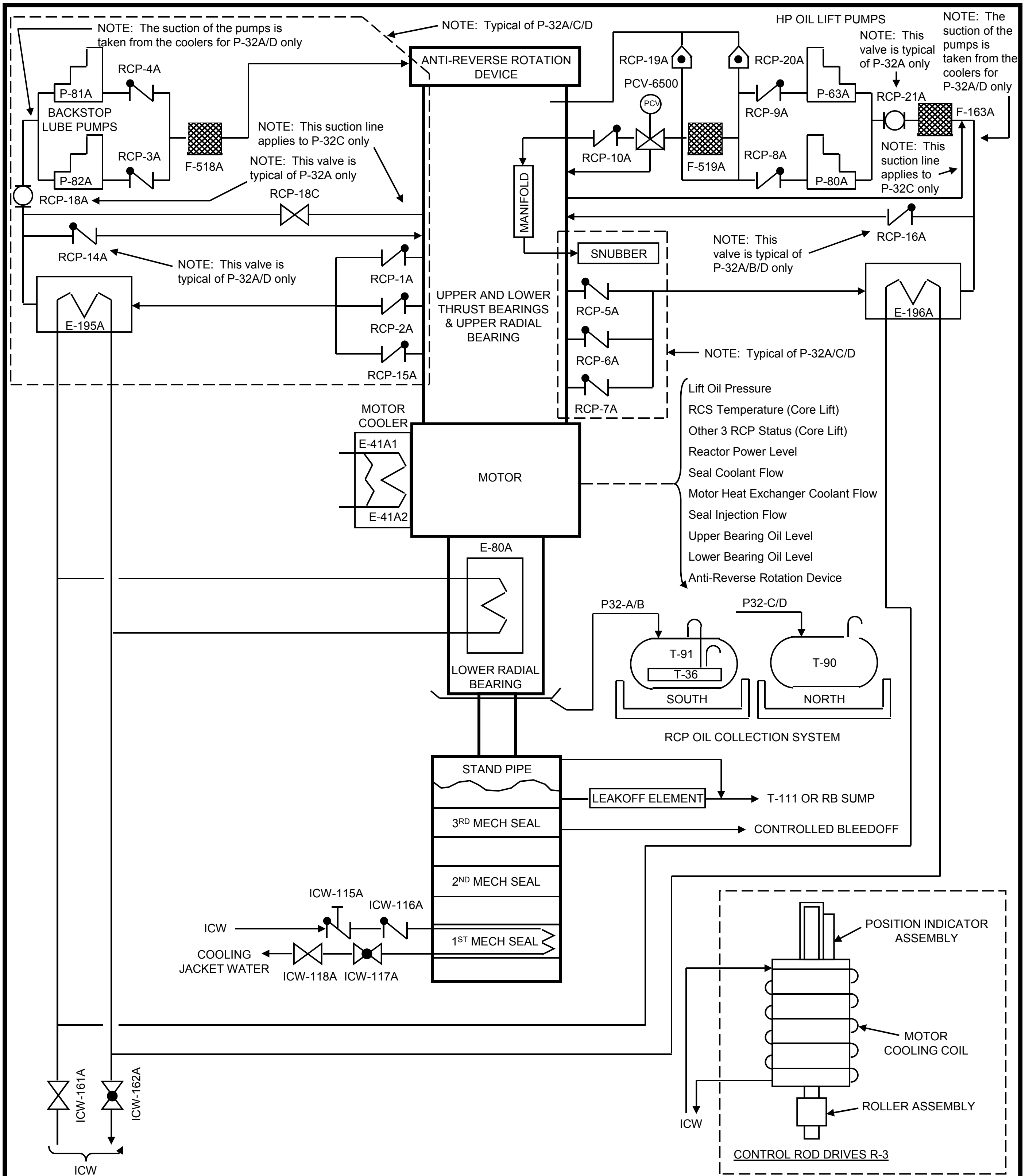
(REFER TO FIGURE 4-1 Sht 1)

SAR FIGURE NO. 7-20

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

PIPING & INSTRUMENT DIAGRAM REACTOR COOLANT SYSTEM	BASED ON DRAWING NO	SHEET 2	REV.



SAR FIGURE NO. 7-21

AMENDMENT 28

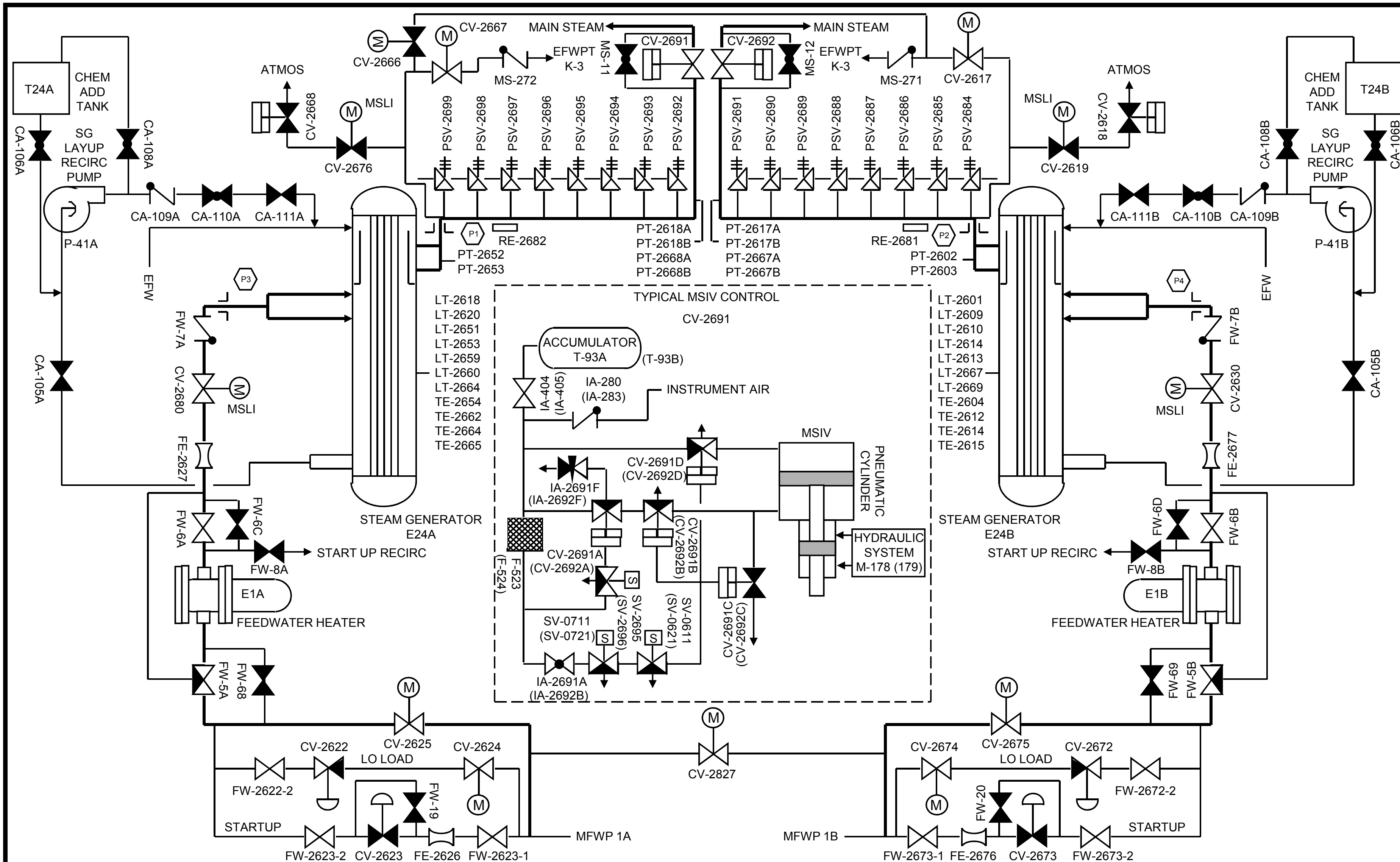
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

CONTROL ROD DRIVE & MISC. REACTOR COOLANT PUMP CONNECTIONS

BASED ON DRAWING NO	SHEET	REV.
	1	



STEAM GENERATOR SECONDARY SYSTEM			SAR FIGURE NO. 7-22
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE	AMENDMENT 20
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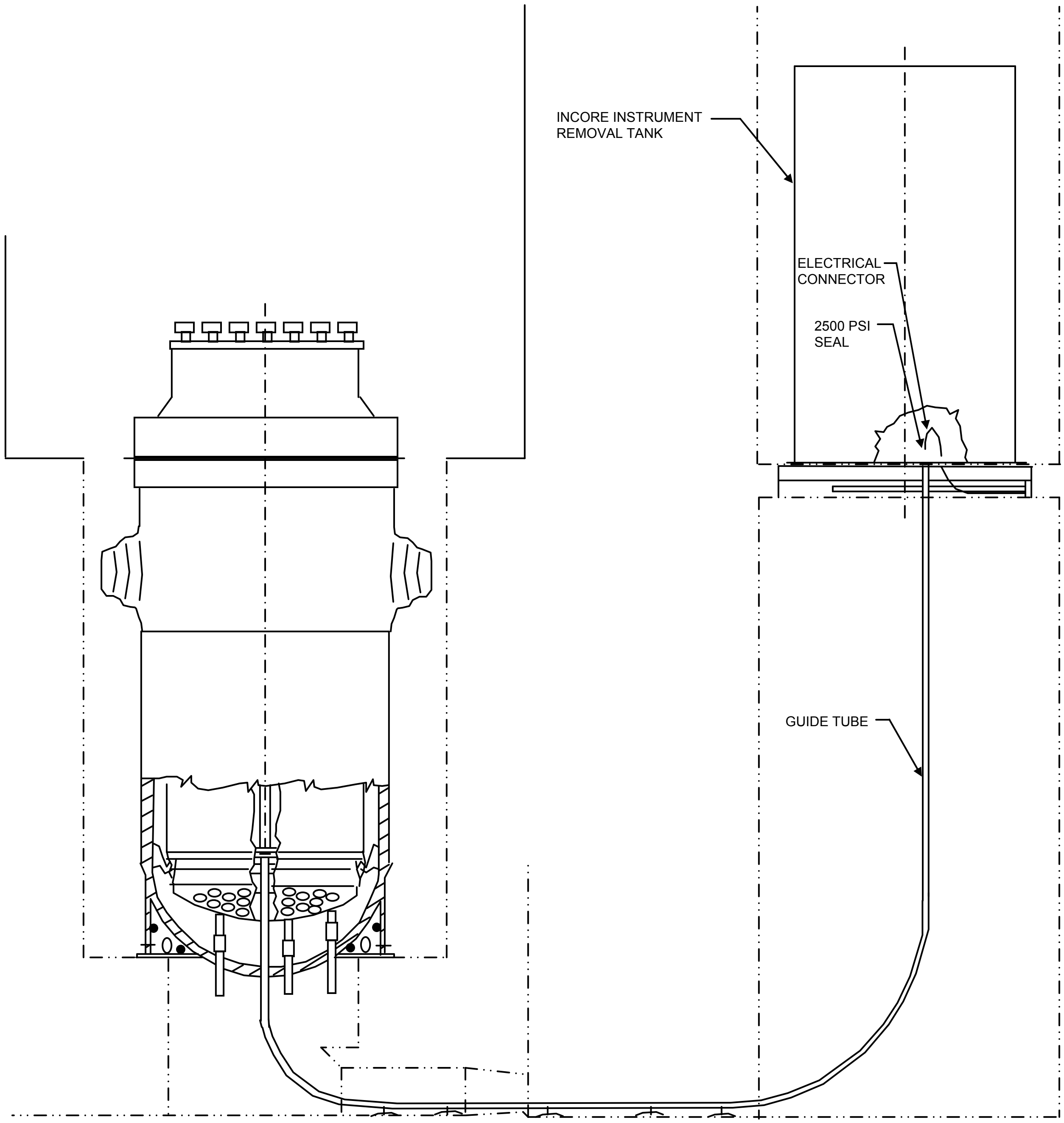
		DESIGN: ENTERGY	
		CAD NO:	

DELETED

SAR FIGURE NO. 7-23

AMENDMENT 20

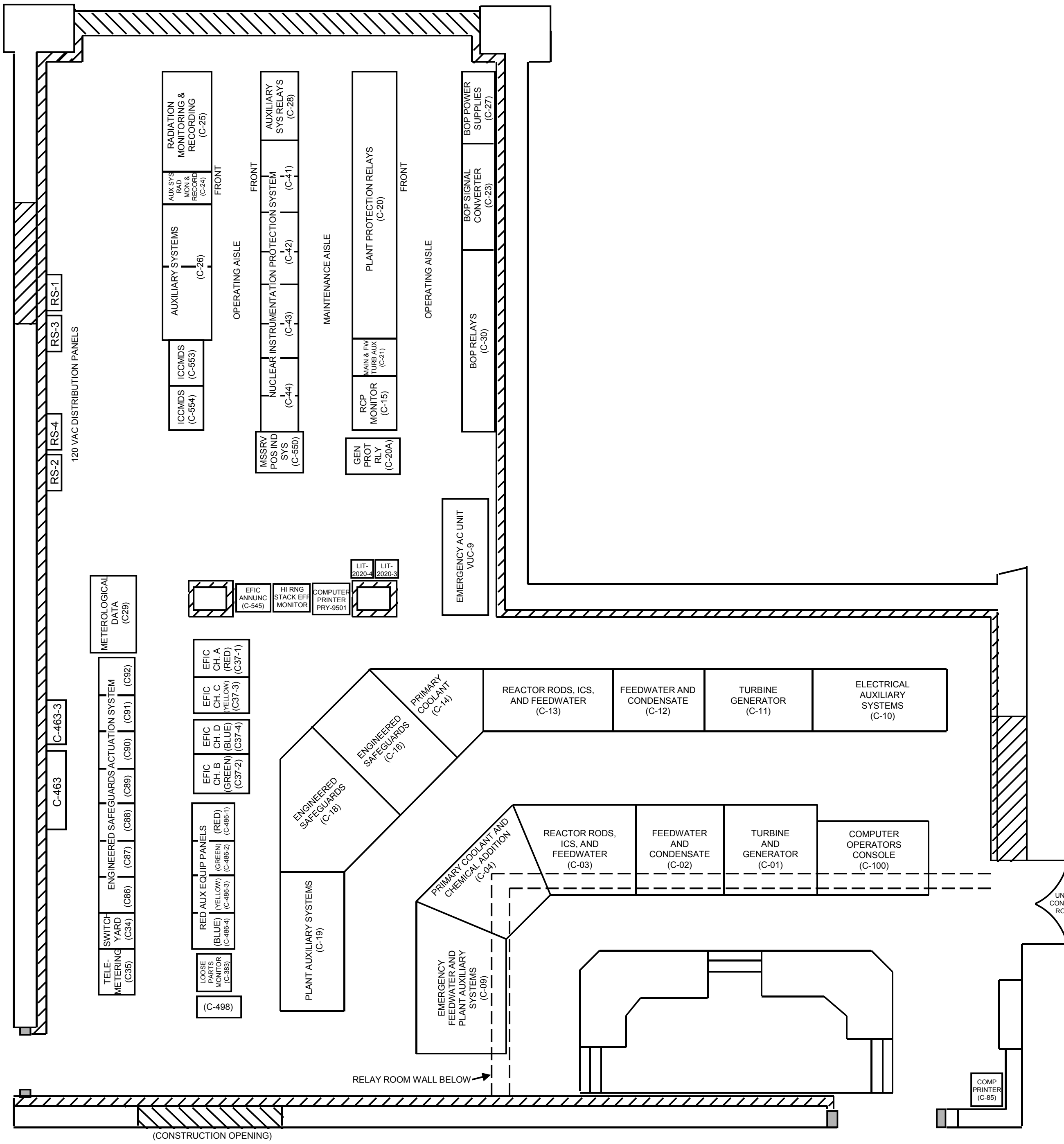
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
INCORE DETECTOR LOCATION	BASED ON DRAWING NO	SHEET 1 / REV.



SAR FIGURE NO. 7-24

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
INCORE MONITORING CHANNEL	BASED ON DRAWING NO	SHEET 1 REV.



SAR FIGURE NO. 7-25

AMENDMENT 26

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

MAIN CONTROL ROOM

BASED ON DRAWING NO	SHEET	REV.
	1	

DELETED

SAR FIGURE NO. 7-26

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.
	1	

DELETED

SAR FIGURE NO. 7-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.
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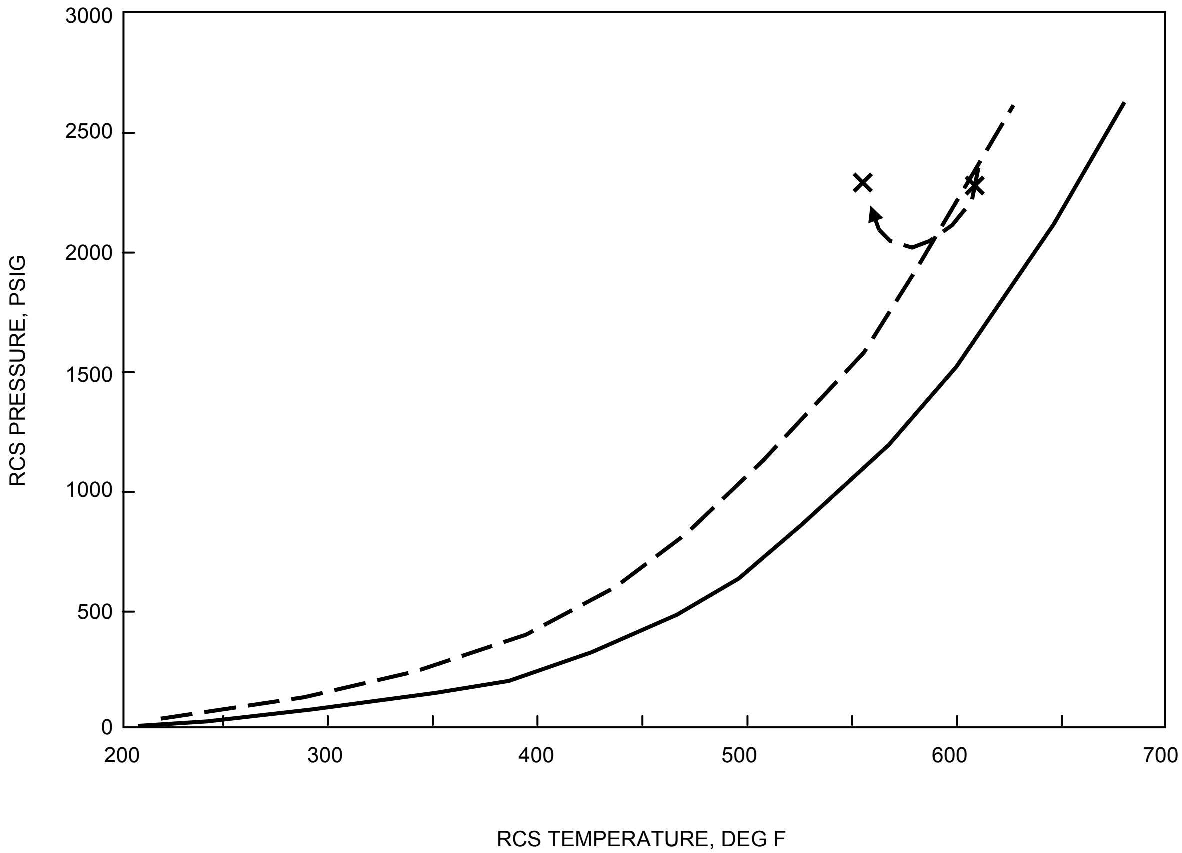
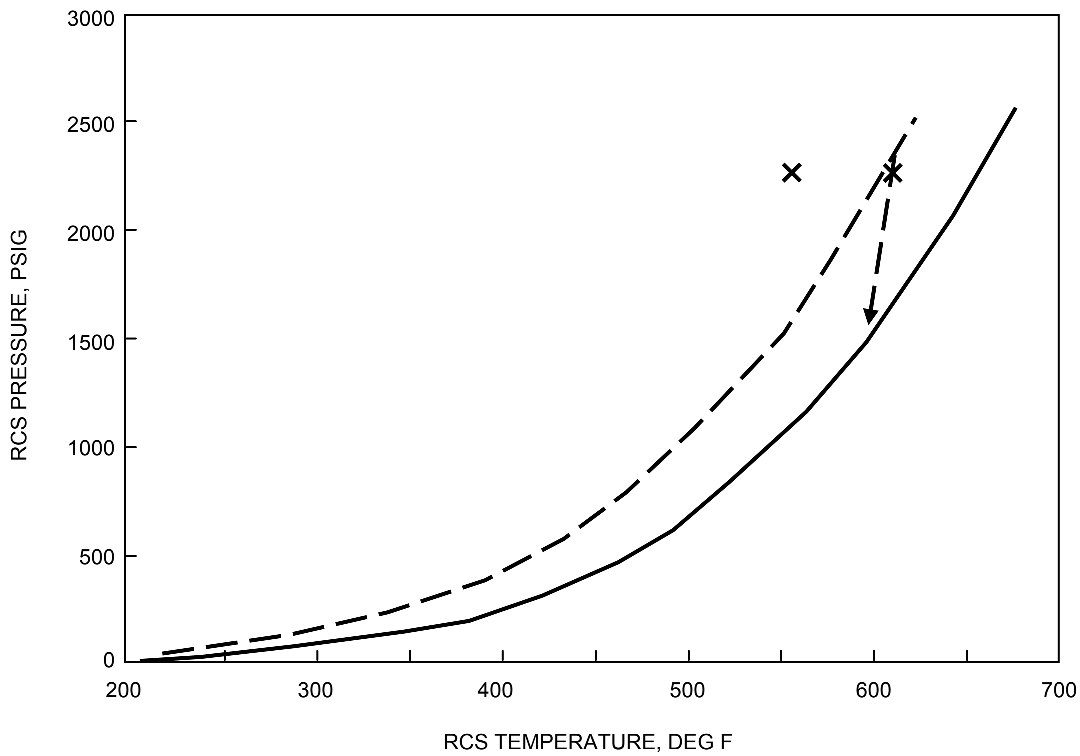
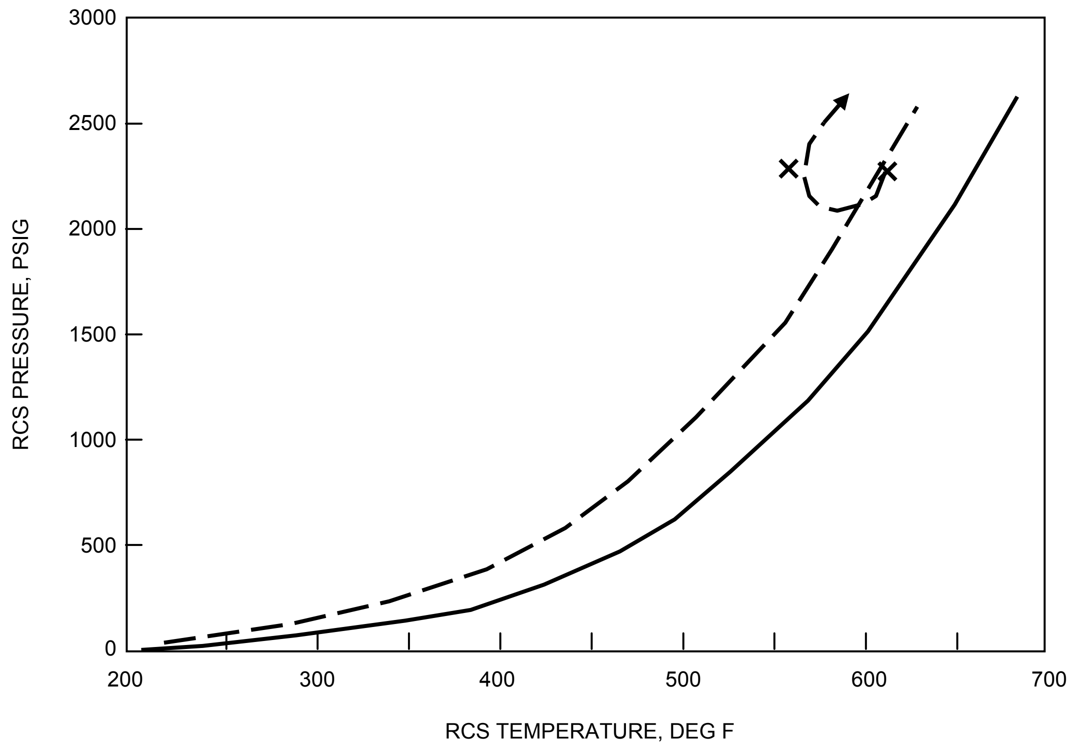


TYPICAL POST TRIP RESPONSE			SAR FIGURE NO. 7-28
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20
			BASED ON DRAWING NO / SHEET / REV.



Inadequate Subcooling Margin: T_{hot} is not progressing toward its target value; in fact, it has rapidly dropped through the subcooled margin line. This condition is diagnosed as loss of adequate primary inventory, or simply "inadequate subcooling margin."

INADEQUATE SUBCOOLING MARGIN		SAR FIGURE NO. 7-29
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	AMENDMENT 20
SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:		BASED ON DRAWING NO / SHEET / REV.



Loss of Primary-to-Secondary Heat Transfer: T_{hot} is increasing as steam generator T_{sat} is decreasing. A ΔT between the two is growing larger. The secondary is no longer removing heat and has lost coupling with the primary. This condition is diagnosed and treated as loss of (inadequate) primary-to-secondary heat transfer.

LOSS OF (INADEQUATE) PRIMARY-TO-SECONDARY HEAT TRANSFER

SAR FIGURE NO. 7-30

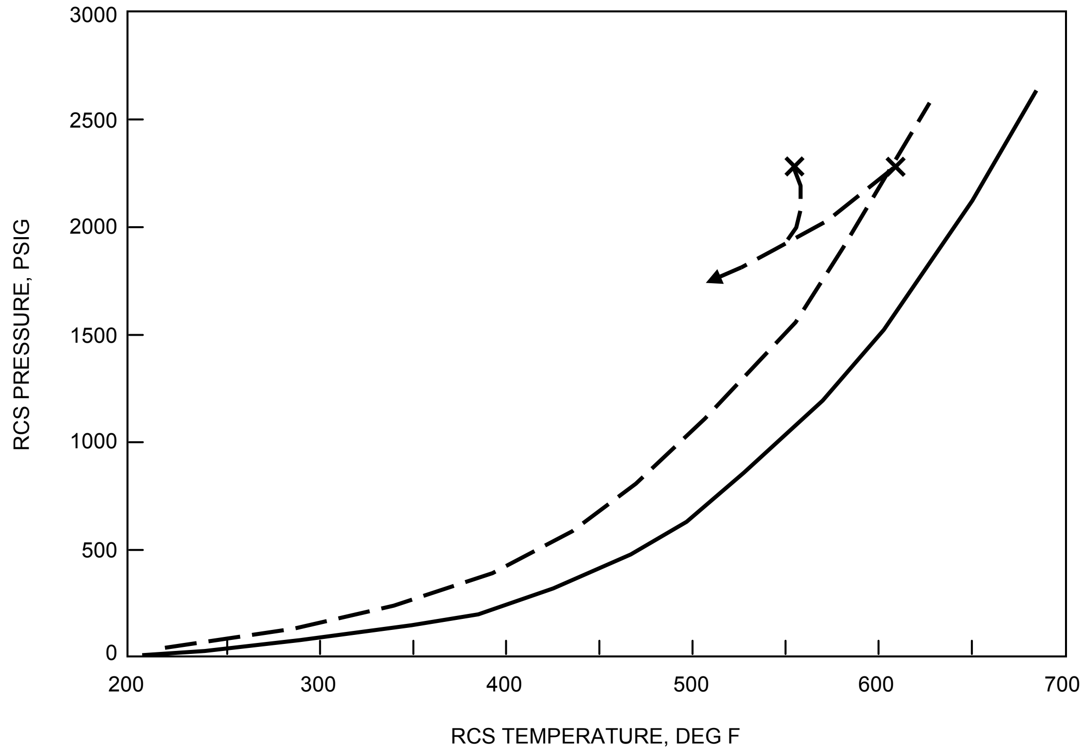
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



Excessive Primary-to-Secondary Heat Transfer: Steam generator T_{sat} has decreased below its established limit. T_{hot} and T_{cold} have reached equal values, but both have gone out of the post-trip window following steam generator T_{sat}. This condition is diagnosed and treated as excessive primary-to-secondary heat transfer.

EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER		SAR FIGURE NO. 7-31
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

CHAPTER 8

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8-6	DELETED
8-7	DELETED
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8-9	500 KV SWITCHYARD AUXILIARY POWER

UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
8.3.2	Correspondence from Phillips, AP&L, to Giambusso, NRC, dated March 13, 1973. (1CAT037304)
8.3.1.6	Correspondence from Ziemann, NRC, to Phillips, AP&L, dated August 13, 1976. (1CNA087616)
8.3.1.6 Table 8-2	Correspondence from Rueter, AP&L, to Ziemann, NRC, dated September 17, 1976. (1CAN097616)
8.3.1.4	Correspondence from Rueter, AP&L, to Stello, NRC, dated October 5, 1976. (1CAN107604)
8.3.1.6	Correspondence from Williams, AP&L, to Reid, NRC, dated August 23, 1978. (1CAN087815)
8.3.1.6	Correspondence from Williams, AP&L, to Reid, NRC, dated October 27, 1978. (1CAN107816)
8.3.1.6	Correspondence from Williams, AP&L, to Reid, NRC, dated December 28, 1978. (0CAN127808)
8.3.1.6	Correspondence from Williams, AP&L, to Stolz, NRC, dated January 19, 1979. (0CAN017911)
8.3.1.6 Table 8-3 Table 8-4 Table 8-5 Table 8-6 Table 8-7	Correspondence from Rueter, AP&L, to Reid & Stolz, NRC, dated March 13, 1979. (0CAN037907)
8.3.1.6 Table 8-8 Table 8-9 Table 8-10	Correspondence from Trimble, AP&L to Reid, NRC, dated May 21, 1979. (1CAN057927)
8.3.1.6	Correspondence from Trimble, AP&L, to Reid, NRC, dated July 12, 1979. (1CAN077904)
8.3.1.6	Correspondence from Trimble, AP&L, to Reid, NRC, dated September 10, 1979. (1CAN097904)
Table 8-1	Correspondence from Trimble, AP&L, to Reid, NRC, dated October 22, 1979. (1CAN107921)

UPDATE REFERENCE LIST (continued)

Section	Cross References
8.3.1.6	Correspondence from Reid, NRC, to Cavanaugh, AP&L, dated December 17, 1979. (1CNA127969)
8.3.1.6	Correspondence from Cavanaugh, AP&L to Reid, NRC, dated February 20, 1981. (1CAN018122)
8.3.1.6	Correspondence from Trimble, AP&L, to Stolz, NRC, dated October 19, 1981. (1CAN108107)
8.3.1.6	Correspondence from Trimble, AP&L, to Stolz, NRC, dated October 30, 1981. (1CAN108110)
8.3.1.4	Design Change Package 589, "Tie-Ins for Emergency Cooling Electrical Equipment Room (Perm. Fix)," 1978. (1DCP780589)
8.3.1.1	Design Change Package 658, "Fire Detection System Modification," 1979. (1DCP790658)
8.3.1.1	Design Change Package 663, "Millstone Undervoltage," 1979. (1DCP790663)
Table 8-1	Design Change Package 1024, "Automatic Vital Power to EFW Pump P7B," 1979. (1DCP791024)
Table 8-1	Design Change Package 1078, "NUREG-0578 TMI Related Work (Pressurizer Heaters)," 1979. (1DCP791078)
8.3.1.6	Correspondence from Williams, AP&L, to Reid, NRC, dated August 22, 1978. (1CAN087815)
8.3.1.6	Correspondence from Williams, AP&L, to Reid, NRC, dated October 25, 1978. (1CAN107809)
8.3.1.6	Correspondence from Trimble, AP&L, to Reid, NRC, dated August 24, 1979. (1CAN087913)
8.3.1.3	Correspondence from Trimble, AP&L, to Madsen, NRC, dated February 22, 1980. (1CAN028016)
8.3.1.1	Design Change Package 81-1080, "Modification of Diesel Generator Voltage 5D Device."
8.3.2	Design Change Package 81-1021A, "Remove Handswitch from SU Transformer #2 (C-10)."
8.3.2	Design Change Package 83-1119, "Modify D01 & D02 Bus Under Voltage Alarm."
8.3.1.1	Design Change Package 84-1061, "Diesel Generator Exhaust Fans."

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8.4	Design Change Package 80-1083, "Emergency Feedwater Initiation."
8.3.1.4	EAR-85-332, "Evaluation on Safety Significance of tray covers and barriers, per guidelines of IEEE 384." (Regulatory Guide 1.75 "Cable Tray Covers")
8.3.1.1	Design Change Package 85-1032, "ANO-1 Emergency Diesel Generator Alarm Upgrade."
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Figure 8-3	Design Change Package 82-2079, "Install EDG Cross-Connect in EDG Fuel Storage Building."
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8.3.1.2	NRC Inspection 92-16, Unit 1 EDSFI, Question 65
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Table 8-1	Engineering Calculation 86-E-0002-1, “Diesel Generator Load Study.”
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Figure 8-3	Plant Change 93-7093, “Removal of Emergency Diesel Generator Fuel Oil Sample Valves.”
Figure 8-3	Plant Change 963146P101, “Bulk Diesel Storage Tank Filter Sample Valves.”
Figure 8-3	Plant Change 96-7019, “Oily Water Separator Discharge Isolation.”
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8.3.2.1.8	Technical Specification Amendment 200, "Service and Testing of Switchyard Battery."
Figure 8-1	Design Change 963089D102, "Magnablast Breaker Replacement on A3."
Figure 8-1	Design Change 991812N101, "Administration Building Chiller Upgrade."
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8.3.1.5.1 8.3.1.5	Nuclear Change Package 974450N101, “Setpoint Changes for Degraded Voltage & Loss-of-Voltage Relays”
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Table 8-1	Calculation 92-E-0003-01, “Clarification of EDG Loading Restrictions During the First Three Minutes”
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Table 8-1	Engineering Request ER-ANO-2003-0042-000, “Installation of Alternate Power Supply for SPDS Computer Room Cooler 2VUC-30”
Figure 8-4	License Document Change Request 1-1.1-0008, “Reformatting of Various Drawing Labels”
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UPDATE REFERENCE LIST (continued)

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8 ELECTRIC POWER

8.1 INTRODUCTION

The electrical system for Arkansas Nuclear One-Unit 1 (ANO-1) is designed to be electrically self-sufficient and provide adequately reliable power sources for all electrical equipment for startup, normal operation, safe shutdown, and handling of all emergency situations. The electrical system of Unit 1 is independent of Unit 2, except for Startup Transformer 2 and the Alternate AC Power Source which are common to both units.

8.1.1 UTILITY GRID AND ITS INTERCONNECTIONS

The electric system for Entergy Arkansas, Inc. (formerly Arkansas Power and Light Company) extends over an area which provides service to over 610,000 customers in 62 of Arkansas' 75 counties. The system consists of over 4,690 miles of transmission lines ranging from 115 kV to 500 kV.

Entergy Arkansas, Inc. is a wholly-owned subsidiary of the Entergy system, which operates as an entity with a highly integrated system consisting of hydro, fossil-fired, and nuclear fueled generating plants.

Entergy Arkansas, Inc. benefits from over 50 points of interconnection either directly or through the Entergy system. At the time Unit 1 was licensed, these points of interconnection were with the following systems: Oklahoma Gas & Electric Company; Southwestern Electric Power Company; Gulf States Utilities Company; Central Louisiana Electric Company; Mississippi Power Company; Union Electric Company; Empire District Electric Company; Missouri Utilities; Tennessee Valley Authority; Southwestern Power Administration; Arkansas Electric Cooperative Corporation; and Associated Electric Cooperatives, Inc. The current Entergy interconnections with the grid are similar although the names of the interconnecting systems may have changed.

8.1.2 ONSITE ELECTRIC SYSTEM

Figure 8-1 shows the single line electrical diagram arrangement of the station power systems. ANO-1 generates electrical power at 22 kV which is fed through an isolated phase bus to the Unit 1 main transformer bank, consisting of three single-phase transformers, where it is stepped up to 500 kV transmission voltage and delivered to the station switchyard.

A bus tie autotransformer bank, consisting of three single phase autotransformers, interconnects the 500 kV and 161 kV systems in the station switchyard. The 161 kV switchyard at the generating station is a ring bus design. The 22 kV tertiary winding of the autotransformer bank supplies Startup Transformer 1, which is identical to the Unit Auxiliary Transformer (UAT). Startup Transformer 2, which will serve both units, is supplied from the 161 kV ring bus.

Auxiliary power for normal plant operation is supplied by the main generator through the Unit Auxiliary Transformer. Power for plant startup and shutdown can be supplied by either of the two startup transformers. In the event of non-availability of these two power sources, power to the Engineered Safety Features (ESF) buses can be furnished by the two fully redundant Emergency Diesel Generator sets.

Upon loss of all onsite and offsite AC power, i.e., station blackout, one of the ESF buses A3 or A4 can be manually energized from the Alternate AC Power Source through ties to Unit 2 switchgear bus 2A9.

For a description of the Alternate AC Power Source, refer to Unit 2 SAR Section 8.3.3.

A 120-volt uninterruptible AC power system has been provided for reactor protection and ESF control channels. The AC power system consists of four channelized distribution panels. Each distribution panel is fed from a power supply that includes an inverter, static switch, manual load switches, and alternate source transformer. See Section 8.3.1.1.6 for further detail.

Two independent and physically separated Class 1E 125-volt batteries along with their own control panels and four battery chargers and a Non-Class 1E battery and its control panel and battery charger provide the necessary DC power sources for the plant.

8.1.3 REACTOR PROTECTION AND ENGINEERED SAFETY FEATURE LOADS

The ESF loads are shown on Table 8-1.

8.1.4 DESIGN BASIS FOR SAFETY RELATED ELECTRIC SYSTEMS

The electrical system for ANO-1 is designed to be electrically self-sufficient and provide adequately reliable power sources for all electrical equipment for startup, normal operation, safe shutdown, and handling of all emergency situations. The following criteria are used in the system and equipment design:

A. Components of the system are sized for operation under normal and emergency conditions.

B. No single component failure will prevent operation of the required engineered safeguards.

C. Redundant sources of power and/or automatic transfer of loads are provided to insure continuous operation of equipment as required under emergency conditions.

D. As far as practical, the system is arranged to make it possible to test equipment with the plant in operation.

E. The relevant American National Standards Institute (ANSI), National Electrical Manufacturers Association (NEMA), Institute of Electrical and Electronics Engineers (IEEE), and the National Electrical Code recommendations are used as a guide in the design.

F. Class 1 electrical equipment is seismically qualified in accordance with the IEEE 344-1971 "Seismic Qualification for Class 1 Electric Equipment for Nuclear Power Generating Stations." See Section 7.1.1.8 for further information.

G. Electrical and physical separation of cables and equipment associated with redundant elements of the engineered safeguards is provided. All safety related portions of the standby electrical power system conform to Safety Guides 6 and 9 and IEEE 308.

H. The electrical system of Unit 1 is independent of Unit 2 except Startup Transformer 2 and the Alternate AC Power Source which are common to both units.

I. The electrical power system is designed to conform to General Design Criteria (GDC) 17 and 18 of Appendix A to 10 CFR 50.

J. The electrical system is provided with Grid Undervoltage Protection (Millstone 2 Event).

K. The electrical system is designed to conform to Regulation 10 CFR 50.63, “Loss of All Alternating Current Power”, and Regulatory Guide 1.155, “Station Blackout”. See ANO Unit 2 SAR Section 8.3.3 for additional details.

Additionally, monitoring of certain parameters of the electrical power system is designed to conform to Regulatory Guide 1.97.

Finally, to prevent the concurrent loss of all engineered safeguards power, the various electrical sources of power are independent of and isolated from each other. The power supplies and controls of equipment providing engineered safeguards functions are arranged to minimize the possibility of physical damage and subsequent loss of equipment operating functions.

8.2 OFFSITE POWER SYSTEM

8.2.1 DESCRIPTION

ANO-1 generates power which is fed through an isolated phase bus to the Unit 1 main transformer bank where it is stepped up to 500 kV transmission voltage and delivered to the station switchyard.

The 500 kV switchyard is a 2-bus design consisting of two breakers each for the Unit 1 and Unit 2 generators and a breaker and one-half for the lines. The 500 kV switchyard includes three outgoing 500 kV lines: one 500 kv line 86.74 miles in length to Mabelvale EHV Substation, one 500 kV line 93.82 miles in length to the Fort Smith (O.G.&E.) EHV Substation, and one 500 kV line 32.62 miles in length to the Pleasant Hill EHV Substation.

The 161 kV switchyard at the generating station is a ring bus design and includes one line to the Russellville East 161 kV Substation and one line to the Pleasant Hill 161 KV substation.

One spare single-phase main transformer and one spare single-phase autotransformer are provided to replace any single-phase unit in the main transformer bank or the autotransformer bank, respectively, in case of a transformer failure.

8.2.1.1 Single Line Diagrams

Figure 8-1 is a single line diagram of the station electrical distribution system.

8.2.1.2 Transmission Lines

The extra high voltage and high voltage transmission system which connects to the switchyard and to the plant includes the following:

A. Approximately 24 miles of two parallel 500 kV single circuit lines with triple bundle phase conductors on metal towers from the ANO extra high voltage substation to a junction with the existing Mabelvale 500 kV line to a point four miles east of Danville, Arkansas, including crossings of the McClellan-Kerr Arkansas River Navigation System. The two lines are parallel to each other and separated by 140 feet on a common 320 foot wide right-of-way, except at the Arkansas River crossing where the parallel lines are separated by 220 feet due to the 190 foot tall river crossing structures.

B. Approximately 32.62 miles of 500 KV single circuit line with triple-bundle phase conductors on metal towers on 180 foot wide right-of-way from the ANO EHV Substation, southeast to the Pleasant Hill EHV Substation north of Morrilton, Arkansas.

C. Approximately 12 miles of single circuit shielded 161 kV line on steel towers and wooden H-frame structures from the ANO extra high voltage substation, southeasterly to the existing Russellville East 161 kV Substation, east of Russellville, Arkansas. The 1.83 mile steel tower portion is constructed on double circuit structures together with "D" below.

D. Approximately 34 miles of single circuit shielded 161 kV line on steel towers and wooden H-frame structures from the ANO extra high voltage substation easterly to the Pleasant Hill Substation.

E. One span (622 feet) of single circuit shielded, 500 kV tie line with triple-bundle phase conductors on metal towers from the 500 kV switchyard to the plant transformer yard for Unit 1. The conductor is a three-bundle 954 MCM 45/7 strand ACSR and the shield wire 7 #7 alumoweld cable.

F. Two spans (920 feet) of single circuit shielded, 500 kV tie line with triple-bundle phase conductors on metal towers from the 500 kV switchyard to the plant transformer yard for Unit 2. The conductor is a three-bundle 954 MCM 45/7 strand ACSR and the shield wire 7 #7 alumoweld cable.

G. Five spans (1237 feet) of single circuit shielded, 161 kV tie line on steel single pole structures from the 161 kV switchyard to the plant transformer yard. The conductor is 336.4 MCM 26/7 strand ACSR and the shield wire 3/8-inch steel cable.

Single circuit metal towers of the two 500,000 volt lines on the same right-of-way afford the advantages of added reliability, least overall height, quicker restoration in the event of failure, and safer conditions during maintenance. Self-supporting steel structures of the single circuit 500 kV design are installed in both lines north of McClellan-Kerr Arkansas River Navigation System. Guyed aluminum towers of the single circuit 500 kV design are installed south of the McClellan-Kerr Arkansas River Navigation System to the point of interconnection with the Mabelvale (O.G.&E.) 500 kV line. The conductor is a 3-bundle 954 MCM 45/7 strand ACSR and the shield wire 7 #7 alumoweld cable.

The two 161,000 volt lines are double-circuited on a common single steel tower line north of the ANO switchyard for a distance of 1.83 miles. The two lines then separate toward their respective destinations.

The southernmost ANO-Russellville East 161 kV line extends easterly for 10.17 miles to the existing Russellville East Substation where the line connects to the 161 kV system. The structures of the 161 kV line beyond the double-circuited section are of wooden H-frame design. The conductor is 1534 MCM 42/19 ACAR and the shield wire is 3/8-inch 7 strand galvanized steel.

The northernmost ANO-Pleasant Hill 161 kV line, leaving the double-circuited steel tower line north of the ANO switchyard, traverses easterly for a distance of 32.02 miles to the site of the Pleasant Hill 161 kV substation where the line interconnects with AP&L's 161 kV system. The structures are wooden H-frame. The conductor is 1024 MCM 24/13 strand ACAR and the shield wire 3/8-inch galvanized steel strand.

The rights-of-way of the 500 kV and the 161 kV transmission lines are not common and do not cross. They are separated by a distance, at the closest point of approach, that will ensure a single structural failure on any of the three 500 kV lines would not affect operation of the 161 kV circuits. A single structural failure on either or both of the 161 kV lines would not affect operation of the 500 kV circuits.

8.2.1.2.1 Crossover

There are six specific incidents of transmission lines crossing other transmission lines. Five of these transmission lines are owned and operated by Entergy Arkansas, Inc. and one by Southwestern Power Administration. Three of these crossings involve lines owned and operated by Entergy Arkansas, Inc. and Southwestern Power Administration and three of these crossings involve lines wholly owned and operated by Entergy Arkansas, Inc. These crossings are discussed as follows:

A. The Southwestern Power Administration double circuit 161 kV line crosses over the ANO - Russellville East 161 kV line northeast of ANO (Unit 2 SAR Figure 8.2-4, Detail 1A). A shield wire, conductor, or structure failure on the Southwestern Power Administration line in this vicinity could cause an outage to this offsite power source, however, four alternate sources would still be in service to feed the ANO switchyard.

B. The Southwestern Power Administration double circuit 161 kV line crosses over the ANO - Pleasant Hill 161 kV line northeast of ANO (Unit 2 SAR Figure 8.2-4, Detail 2A). A shield wire, conductor, or structure failure on the Southwestern Power Administration line in this vicinity could cause an outage to this offsite power source, however, four alternate sources would still be in service to feed the ANO switchyard.

C. The Southwestern Power Administration double circuit 161 kV line crosses under the ANO – Pleasant Hill 500 KV line northeast of ANO (Unit 2 SAR Figure 8.2-4, Detail 3A). A shield wire, conductor or structure failure on the Southwestern Power Administration line in this vicinity would not cause an outage to this offsite power source.

D. The ANO – Pleasant Hill 500 KV line crosses over the ANO - Pleasant Hill 161 kV line northeast of Russellville (Unit 2 SAR Figure 8.2-4, Detail 4A). A shield wire, conductor, or structure failure on the 500 kV line in the vicinity of this crossing could cause an outage to both of these offsite power sources, however, three alternate sources would still be in service to feed the ANO switchyard.

E. The two parallel ANO - Jct. Mabelvale O.G.&E. 500 kV lines cross over the Dardanelle Dam - Dardanelle-Danville 161 kV line northeast of Danville, Arkansas (Unit 2 SAR Figure 8.2-4, Detail 5A). A shield wire, conductor, or structure failure on either of the 500 kV lines at this crossing would operate switches at the Dardanelle Dam and Danville Substations, thereby isolating the loss of service to the 161 kV system and, assuming the parallel 500 kV circuit was intact, there would be four offsite power sources in service to feed the ANO switchyard.

F. The Pleasant Hill-Mayflower 500 KV Line crosses over the ANO – Pleasant Hill 161 KV line north of Morrilton (see Figure 8.2-4, Detail 4A). A shield wire, conductor or structure failure on the 500 KV line in the vicinity of this crossing probably would cause an outage to this 161 KV off-site power source, however, four alternative sources would still be in service to feed the ANO switchyard.

8.2.1.2.2 Shared Rights of Way

There are three separate examples where two offsite lines are parallel to each other on separate structures sharing the same right-of-way. These examples are individually discussed below:

A. The 24 miles of two parallel 500 kV single circuit transmission lines on metal towers exit the ANO switchyard north, then turn west and south to a point northeast of Danville, Arkansas. The two lines are parallel to each other and separated by 140 feet on a common 320 foot wide right-of-way except at the Arkansas River crossing where separation is 220 feet and right-of-way is 460 feet wide due to the 190 foot tall structures. Both circuits of the approximately 5.57 mile parallel section north of the river are composed of steel structures which vary in height from 70 to 110 feet. The structures are placed essentially opposite each other with a maximum differential elevation at the base of approximately 36 feet. Should one of these parallel lines go down it would not affect either the towers or the conductors of the parallel line. On the 17.67 mile parallel section south of the river, guyed aluminum towers are installed on a 140 foot separation. Tower heights vary from 75 to 115 feet on both circuits. The towers are offset slightly longitudinally. Four guy cables on each structure extend toward the parallel line and toward the edge of the right-of-way. Failure in the anchorage, guy cables, or loss due to vandalism could result in structural failure to either line and could cause a structure to fall into the structure guys or conductors of the parallel line. This could result in an outage to both of these parallel offsite power sources. However, there still would be three offsite power sources to feed the ANO switchyard.

B. The ANO – Pleasant Hill single circuit transmission line exits the plant north on metal towers and parallels the twin lines of the ANO - Jct. Mabelvale - OG&E 500 kV line a distance of 1.51 miles. The towers vary in height from 70 to 110 feet with a maximum differential elevation at the base of approximately 46 feet. Tower separation is 140 feet east of the more easterly of the twin lines exiting the plant. Structures are essentially opposite of the structures of the east twin line. In the event of a structure failure, only one circuit would be damaged, leaving four offsite power sources in service feeding the ANO switchyard.

C. The ANO – Pleasant Hill line parallels the ANO - Pleasant Hill 161 kV line northeast of Russellville, Arkansas. These lines are parallel on a 250 foot common right-of-way with a line separation of 110 feet for a distance of approximately 4.44 miles. The 500 kV line is on metal towers with a phase-to-phase separation of 30 feet, 3 inches and varies in height from 70 to 150 feet. The 161 kV line is on wooden structures with a phase-to-phase separation of 14 feet, 6 inches and the poles vary in height from 55 to 85 feet. A total of 37 wooden structures and 23 metal structures are involved in this parallel section. A structure failure on the wooden 161 kV line would not cause an outage of the 500 kV line but a structural failure on the 500 kV line could cause an outage to both circuits. In this event, three offsite power sources would still be in service to feed the ANO switchyard.

8.2.1.3 Switchyard

The ANO switchyard consists of a 500 kV yard and a 161 kV yard connected by a 600 MVA autotransformer bank with a 22 kV tertiary winding.

The 500 kV portion of the yard has two main buses. Both the Unit 1 and Unit 2 generators are connected to the main buses by double breaker bays. The three 500 kV transmission lines and the autotransformer are connected to the main buses by two breaker-and-a-half bays. The Mabelvale and Fort Smith transmission lines are on one bay and the Pleasant Hill line and the autotransformer are on the other bay.

The 161 kV portion of the switchyard is a 4-element ring bus. The four elements connected to this 161 kV ring bus include the ANO-Russellville East 161 kV transmission line, the ANO - Pleasant Hill 161 kV transmission line, the ANO 500/161 kV autotransformer, and the 161 kV line which feeds an automatic voltage regulator in the switchyard which then feeds Start-up Transformer 2 in the plant. The ring bus is arranged so the autotransformer and the lines to Startup Transformer 2 are not connected to a common 161 kV breaker. Likewise, the two 161 kV transmission lines are not connected to a common 161 kV breaker.

The 22 kV tertiary winding of the autotransformer is connected to two 22 kV breakers. One breaker feeds an automatic voltage regulator in the switchyard which then feeds Startup Transformer 1 in the plant via underground 22 kV cable. The other breaker feeds an automatic voltage regulator in the switchyard which then feeds Startup Transformer 3 in the plant, also via underground 22 kV cable.

The 161 kV yard is separated from the autotransformer by two 161 kV circuit breakers. Therefore, the failure of any one of the 161 kV breakers will trip the adjacent breakers and interrupt only one of the plant offsite power sources. The 500 kV lines and the autotransformer will remain available. Conversely, the failure of a 500 kV breaker which feeds the autotransformer will trip the two 161 kV breakers connected to the autotransformer, but will not interrupt the 161 kV circuit to the plant.

Each breaker in the switchyard has a separate control circuit. A failure of one circuit to operate properly will not affect any other breaker control circuit.

Protective relaying on the autotransformer consists of a primary and a backup relaying scheme. The primary relaying scheme uses transformer differential relays. The backup relaying scheme uses two sets of distance and directional ground overcurrent relays. One set looks into the autotransformer from the 161 kV side. The other set looks into the autotransformer from the 500 kV side. The primary and backup schemes do not have in common any components such as protective relays, current transformers, potential transformers, trip coils, or DC thermal breakers. Therefore, the failure of one scheme to operate properly will not affect the operation of the other scheme. Both schemes initiate the local breaker failure scheme.

Protective relaying on the 161 kV circuit to the plant also consists of a primary and a backup relaying scheme. The primary scheme uses bus differential relays and the backup scheme uses distance and directional ground overcurrent relays. Again, the two schemes do not have any components in common so that a failure in one scheme will not affect the operation of the other scheme. Both schemes initiate the local breaker failure scheme. Additional protective relaying on the 161 kV circuit to the plant includes overvoltage protection and sudden pressure relaying for the 161 kV automatic voltage regulator. All of these protective relaying schemes on the 161 kV circuit to the plant operate lockout relays in the switchyard which operate Startup Transformer 2 lockout relays in the plant. This results in the opening of the 161 kV breakers to de-energize the Startup Transformer 2 circuit to the plant and results in the automatic transfer of plant auxiliary loads from Startup Transformer 2 to Startup Transformer 1 if the plant auxiliary loads were being served from Startup Transformer 2.

Protective relaying on Startup Transformer 1 and 2 also consists of Open Phase Detection. NRC Bulletin 2012-01 discusses the possibility that an open phase condition, with or without accompanying ground faults, located on the high-voltage side of a transformer connecting a GDC 17 offsite power circuit to the plant electrical system, could result in a degraded condition in the on-site power system. To address this issue, redundant Open Phase Detection systems monitor the power supply circuits to Startup Transformers 1 and 2. Startup Transformers 1 and 2 are monitored for open phase conditions affecting one or two phases, with or without a ground fault. The relays initiate alarms in the main control room when an open phase condition is detected. In the event of an open phase condition, the Open Phase Detection system will initiate a trip of the affected transformer to isolate the distribution system buses from the open phase condition.

The control power for the 500 kV and 161 kV switchyard breakers is supplied from two 125 volt DC battery banks and their respective battery chargers located in the switchyard control building. Manual switches are provided to allow either battery bank to carry the entire switchyard load if the other battery is unavailable. The battery charger for Battery Bank #2 is also a battery eliminator and can carry the entire switchyard load if either or both battery banks are unavailable. Three sources of AC power are available for the battery charger: 1) the auxiliary power transformers on the 22 kV bus; 2) the plant 480 volt load center bus "B3"; and 3) the plant 480 volt engineered safeguard bus "B6." Figure 8-9 shows how these three sources may be connected to the battery chargers. The switchyard DC bus is a non-1E power supply.

The switchyard DC control bus is isolated from ground and detectors are provided to alarm when a ground exists on either the positive or negative bus. All equipment will function properly when one side of the DC bus is grounded. If either battery charger fails, its associated battery will carry its switchyard DC load without interruption for a minimum of eight hours.

8.2.1.4 <u>Reliability</u>

Reliability considerations to minimize the possibility of power failure due to faults in the network interconnections and associated substations are as follows:

A. Each 500 kV line is capable of carrying the full Unit 1 output under normal grid conditions.

B. The 500 kV transmission lines are single circuit and the towers are designed as recommended by AECE paper No. 3269.

C. 500 kV and 161 kV system stability will be maintained on simultaneous tripping of the Unit 1 and Unit 2 generators.

D. The 161 kV Pleasant Hill line may be directly connected to the Startup Transformer 2 in the event of loss of switchyard control during the maximum flood condition.

E. The high voltage systems are protected from lightning and switching surges by lightning protection equipment and by overhead electrostatic shield wires.

F. Primary and backup relaying are provided for each circuit along with local circuit breaker backup switching. These provisions permit the following:

1. Any circuit can be switched under normal or fault switching without affecting another circuit.

2. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.

3. Short circuits of a section of the main bus will be isolated without interrupting service to any circuit other than that connected to the faulty bus section.

4. Circuit protection is provided by redundant relaying.

5. Offsite power to the engineered safety features is available assuming a single failure in the control power supplies, control circuits, and protective relaying.

G. The electrical power systems conform to General Design Criterion 17 as discussed in Section 8.3.1.2.

Electric power is supplied to the station switchyard by five separate transmission lines. Three lines, one from the Mabelvale substation, one from the Ft. Smith substation, and one from the Pleasant Hill substation, feed the 500 kV ring bus. The remaining two lines, one line from the Russellville East substation and the other from the Pleasant Hill substation, feed the 161 kV ring bus. Two physically independent circuits with startup transformers are provided from the station switchyard to the onsite electrical distribution system. Startup Transformer 1 is supplied by the autotransformer bank through underground cables and Startup Transformer 2 is supplied by the 161 kV ring bus. Startup Transformer 2 serves as a second source of power for both ANO units.

Both Unit 1 and Unit 2 are prevented from automatic transfer to Startup Transformer 2 during normal power operation for all buses except Buses A1/A3 (Unit 1) and 2A1/2A3 (Unit 2). Procedures administratively control Unit 1 and Unit 2 access to Startup Transformer 2. Procedures may allow other fast transfer capabilities to Startup Transformer 2 for specifically analyzed conditions and restrictions defined in approved Engineering Calculations and Evaluations.

The DC system is designed to provide continuous power for control, instrumentation, Reactor Protection and Engineered Safeguard Systems, safeguard actuation control systems, and other loads for normal operation and orderly shutdown. Two independent Class 1E 125-volt batteries and DC control centers are provided for the essential instrumentation, distribution panels, and motors. Each battery is sized up to carry the continuous emergency and vital AC load for a minimum period of two hours in addition to supplying power for the operation of momentary loads during the 2-hour period. A Non-Class 1E 125 VDC battery and DC control center provide power for non-1E loads such as emergency lighting, instrumentation, and motors.

H. De-energizing of the 500-161 kV switchyard will be required at a flood elevation of 356.5 feet. At this flood elevation, it will additionally be necessary to de-energize and bypass the 161 kV voltage regulator to Startup Transformer 2. In order to maintain a source of offsite power at flood elevations above 356.5 feet, it is necessary to install temporary connections over the 161 kV switchyard to connect the Startup Transformer 2 directly to the 161 kV Pleasant Hill transmission line. These temporary connections are jumpers which are sized and stored onsite. In the event of a flood, these jumpers would be installed by local office or plant personnel prior to flood waters reaching the site. This operation would take two men approximately three hours. All the equipment connections necessary to maintain offsite power are above the 361 foot design flood elevation or procedural controls have been established to ensure appropriate flood protection is verified prior to an external flood impacting site-specific SSCs. This offsite power source is provided with lightning arrestor protection and shielding. This Pleasant Hill line has a minimum clearance of 17 feet above the 361 foot elevation.

I. Additional redundant relaying is provided for each Startup Transformer to detect open phase conditions and protect the transformers.

8.2.2 ANALYSIS

System studies have been conducted to test the performance of the Arkansas Power & Light Company/Entergy System in both the steady-state mode and under transient conditions. The conditions studied included, but were not limited to, outages of multiple circuit lines using common towers; coincidental, but not simultaneous, loss of one transmission line and one generator or two generators; or of two transmission lines.

The results of these studies indicated that under steady-state conditions no loss of power would occur and adequate system voltage and acceptable loading of equipment would be maintained. Under the transient conditions, the studies indicated that the system is transiently stable and no loss of power or cascading type conditions would occur.

The availability of the three 500 kV lines at the Arkansas Nuclear One switchyard can be predicted by examining the history of the existing 500 kV transmission grid.

Using data from the 1966 - 1972 period on six 500 kV lines with a cumulative mileage of 740 miles, the frequency, duration, and causes of outages on the three 500 kV lines to Arkansas Nuclear One should be as follows: 1) the Arkansas Nuclear One - Ft. Smith line will have an average of 0.98 outages per year; 2) the Arkansas Nuclear One - Mabelvale line will have an average of 0.93 outages per year; and 3) the Arkansas Nuclear One - Pleasant Hill line will have an average of 0.34 outages per year. The following table gives the causes and duration of outages:

Cause	% of Outages	Avg. Duration
Tornado or High Wind	11	189 hours
Lightning	40	1.0 hour
Equipment Failure	35	5.79 hours
Unknown	13	1.14 hours
Operator Error	1	0.13 hours

8.3 ONSITE POWER SYSTEMS

8.3.1 AC POWER SYSTEMS

8.3.1.1 Description

The onsite AC power system of Unit 1 consists of the various auxiliary electrical systems designed to provide reliable electrical power during all reactor operating and shutdown conditions. The systems are designed with sufficient power sources and redundant buses with required switching to accomplish this. Engineered safeguards auxiliaries are arranged so that the loss of a single bus section results in only single losses of auxiliaries, leaving redundant auxiliaries to perform the same function.

Unit 1 generates electrical power at 22 kV which is fed through an isolated phase bus to the main transformer bank. The main transformer consists of three single-phase transformers, which step up the voltage to 500 kV and deliver it to the station switchyard. During normal operation the switchyard is used for the transmission of power generated at the station to the offsite distribution system. During startup and shutdown conditions, the switchyard is used as a means of supplying power to station equipment from the grid.

The onsite AC distribution system is a network of transformers and buses used to distribute electrical power to various components. The major power buses are the following:

A. 6900 Volt System - consists of two buses and supplies power to the Reactor Coolant Pump motors.

B. 4160 Volt System - consists of four buses; A1, A2, A3, and A4. Buses A1 and A2 supply power to auxiliary components. Buses A3 and A4 supply power to the engineered safeguards loads.

C. 480 Volt System - consists of load centers and motor control centers which supply power to station auxiliary loads and engineered safeguards loads.

D. Station DC System - consists of batteries, battery chargers, and buses to provide control power to switchgear, essential inverters, and other auxiliary power.

E. 120/208 Volt Instrument AC System - consists of instrument transformers and distribution panels providing power to non-vital instrumentation.

F. 120 Volt Vital AC System - consists of inverters and distribution panels providing power to Reactor Protection System and other vital instrumentation system.

The onsite AC system is capable of starting the largest required load with the remainder of the connected motor loads in service. The system will transfer to a Startup Transformer following a turbine generator or reactor trip without a loss of auxiliary load.

Protective relaying is arranged for selective tripping of circuit breakers, thus limiting the loss of power to the affected area.

8.3.1.1.1 Main Generator and Main Transformer

The steam turbine driven main generator delivers AC power at 22 kV through an isolated phase bus to the main transformer bank and the Unit Auxiliary Transformer (UAT). Power is stepped up to 500 kV for transmission to the switchyard and stepped down to 6.9 kV and 4.16 kV for the station auxiliaries during normal operation. The main generator produces a 3 phase, 22 kV, 60 Hz, 902 MW output at rated conditions.

A description and rating of the main generator and the isolated phase bus are as follows:

Main Generator

Output, MVA	1002.6 @ 0.9 pf
Voltage, kV	22
Speed, rpm	1800
Phases	3
Frequency, Hz	60
Cooling	Hydrogen @ 75 psig

Isolated Phase Bus-Rated Current @ 22 kV

Main generator bus	28,000A
Auxiliary Transformer bus	4,000A

The main transformer bank consists of three single-phase transformers along with a single phase spare of compatible size and rating. The main transformer bank is located outdoors, east of the turbine building. There are firewalls separating each of the transformers along with an automatic deluge system. The isolated phase bus connects the main generator output to the main power transformer.

The main transformers are equipped with sampling valves for periodic testing of transformer oil. Test switches are provided for testing sudden pressure relays on the transformer and all protective relays and instruments associated with the transformers have provisions for inservice testing and calibration.

8.3.1.1.2 Unit Auxiliary Transformer, Startup Transformers, and 6900 Volt Systems

During normal operation of the plant, the full size Unit 1 Auxiliary Transformer connected to the generator isolated phase bus provides the primary source of electrical power for the plant auxiliaries. Startup Transformer 1, identical to the Unit 1 Auxiliary Transformer, is connected to the 22 kV tertiary of the bus tie autotransformer bank. Startup Transformer 2 is connected to the 161 kV ring bus.

Each of the two Startup Transformers, supplied from separate sources, will be capable of providing a source of power for startup, shutdown, and after shutdown requirements. Startup Transformer 1 will serve as a complete standby source for the plant auxiliaries of Unit 1. Startup Transformer 2 serves as another source of standby power for ANO - Units 1 and 2.

<u>6900-Volt Auxiliary System</u>

Two 6900-volt buses, designated as 1H1 and 1H2, are provided for the operation of the four Reactor Coolant Pumps. This provides the necessary flexibility and reliability during normal plant operation, startup, and shutdown. During normal operation, each bus is fed from the 6.9 kV winding of the Unit 1 Auxiliary Transformer. During startup and shutdown, the buses are fed from the 6.9 kV secondary winding of Startup Transformer 1.

Manual transfer of a 6900-volt bus between the three sources (Unit Auxiliary Transformer, Startup Transformer 1, and Startup Transformer 2) is initiated by the operator from the control room and automatic transfer between the three sources is initiated by protective relay action.

Manual 6900-volt bus transfers used on startup or shutdown of a unit are "live bus" transfers, i.e. the incoming source feeder circuit breaker closes onto the energized bus section and its interlocks trip the outgoing source feeder circuit breaker, resulting in transfer without power interruption. After the circuit breaker of the incoming transformer closes, the supply breaker of the other transformer that was connected to the bus before the manual transfer was initiated, is automatically tripped when the operator releases the control switch handle of the incoming breaker.

Automatic fast transfer of a 6900-volt bus from the UAT to an available Startup Transformer is initiated by the generator protective lockout relay. If both Startup Transformers are available, transfer is made to the transformer which has been selected as "preferred standby" by means of selector switches.

A Startup Transformer is "available" to a bus if:

A. It has normal secondary voltage.

B. Its protective lockout relay is not tripped.

C. The bus feeder breaker control switch is not in the pull-to-lock position.

D. The bus feeder breaker closing DC potential is available.

Automatic transfer of a 6900-volt bus from one Startup Transformer to the other Startup Transformer is initiated by the in service transformer protective lockout relay.

If the automatic transfer is not executed due to unavailability of both Startup Transformers at the instant that transfer is required, then the 6900-volt bus will be re-energized after all the motor services have been tripped due to undervoltage. The 6900-volt bus cannot be re-energized until incoming voltage has been restored.

8.3.1.1.3 <u>4160-Volt Auxiliary System</u>

Four 4160-volt buses have been provided. Each of the two main buses designated as A1 and A2 provides power to non-ESF, 4kV auxiliary motors and feeders to 480-volt non-ESF load centers. The two 4160 volt ESF buses designated as A3 and A4 supply equipment essential for the safe shutdown of the plant.

Normally, the two main buses are fed from the 4160-volt winding of the Unit 1 Auxiliary Transformer. During startup the main buses are fed from the 4160-volt winding of Startup Transformer 1. During shutdown, the main buses are fed from the 4160-volt winding of Startup Transformer 1 or Startup Transformer 2. Manual and automatic bus transfers for A1 and A2 are provided and are similar to those described in Section 8.3.1.1.2 for the 6900-volt buses. Non-ESF bus A1 can be connected to the Alternate AC Power Source (Alternate AC Generator System) through a connection to Unit 2 4160 volt bus 2A9.

The two 4160-volt engineered safeguards buses are normally fed on a one-for-one basis from the main buses. Upon loss of normal and standby power sources, each of the two 4160-volt engineered safeguard buses are energized from its respective diesel generator. Bus load shedding, bus transfer to the diesel generator, and pickup of critical loads is automatic. These features are described in detail in Section 8.3.1.1.7.

Two diesel generators, each connected to one of the 4160-volt engineered safeguards buses, are provided. Each diesel generator unit is rated at 4160 volts, 2600 kW continuous, with an intended service rating of 2750 kW, which is sufficient to carry the engineered safeguard bus loads following a Design Basis Accident (DBA) and still drive additional motor operated valves. The diesel generators can be manually started by using an air start solenoid override on a loss of DC power. Detailed operation of the diesel generator is described in Section 8.3.1.1.7. During station blackout, either of the two ESF buses A3 or A4 can be manually connected to the Alternate AC Generator System through a connection to Unit 2 4160 volt bus 2A9. This connection terminates at the A3 bus end of the A3-A4 bus tie. The Alternate AC Generator is rated 4400 kW continuous at 4160 volts.

Arrangement of the 6900-volt and the 4160-volt system buses is shown in Figures 8-1.

Power cables for the 4.16 kV system are shielded copper conductors rated 5 kV, 90 °C or greater with flame resistant insulation and jacketing materials capable of withstanding the worst case environment conditions to which the cable could be exposed. The conductors are sized to carry the maximum available short circuit current for the time required for the circuit breaker to clear the fault. The conductors are also sized for continuous operation at 125 percent of full load current.

All of the safety related distribution equipment, including raceway systems, were designed to meet the seismic requirements for Class 1E electric equipment as discussed in Section 5.1. The 4.16 kV switchgear is not seismically qualified in the racked-down configuration.

The 4160-volt engineered safeguard switchgear is located within a Seismic Category 1 structure which minimizes exposure to mechanical, fire, and water damage and protects from potential missile hazards. This equipment is properly coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions. Physical separation and isolation of each train has been maintained.

8.3.1.1.4 <u>480-Volt Auxiliary System</u>

The 480-volt system, which is normally supplied from 4160 V buses A1 and A2, contains eight load centers, B1, B2, B3, B4, B5, B6, B7, and B9 each consisting of a 4160/480 volt transformer and a 480-volt switchgear bus. The load centers are arranged in pairs with the exception of load centers B7 and B9. Four of the load centers, which supply non-ESF loads, are paired into double-ended configuration with a tie breaker between buses. Two load centers, which supply

ESF loads, have two tie breakers between buses. Redundant loads are fed from the same load center pair, one per bus. The capacity of load center transformers and tie breakers in paired load centers are sufficient, under certain loading configurations, to permit operation with one of the transformers out of service.

The 480-volt auxiliary system also contains motor control centers, eight of which serve engineered safeguard or essential loads. Two are designated preferred emergency motor control centers and can be manually transferred to be fed from either engineered safeguard subsystem. These two motor control centers provide such loads as emergency lighting and the turbine-generator emergency bearing oil pump.

Arrangement of the 480-volt AC distribution system and ratings of circuit breakers, are shown in Figure 8-1.

The engineered safeguards 480-volt load centers are located within a Class 1 structure (Seismic Category 1) area to minimize exposure to mechanical, fire, and water damage. This equipment is properly coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions.

Separation of load centers is accomplished through provision of fire proof compartments and/or barriers and separate and protected cable runs to and from each load center.

The 480-volt motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located below the turbine-generator operating floor level. Those associated with the Nuclear Steam Supply System (NSSS) are located in areas to minimize their exposure to mechanical, fire, and water damage.

Engineered safeguards motor control centers are located within a Class 1 structure. Details for heating, ventilating, and cooling systems for the electrical systems are provided in Chapter 9.

The main incoming breakers and motor starter breakers of the load centers can be controlled from the control room and the switchgear. This is also true of all of the load center cross-tie breakers except the B2-B7 cross-tie. It can only be operated at the switchgear and is used only during cold or refueling shutdown. All other load center breakers can be controlled only at the switchgear.

All of the ESF 480-volt distribution equipment is designed to meet the seismic requirements for Class 1E electric equipment as discussed in Section 5.1. All load center transformers, load center buses, and motor control center buses have adequate capacity to supply the momentary and continuous loads connected to the 480-volt buses.

The redundant vital MCC buses have no inter-connecting tie breakers with the exception of one common MCC which receives power from the load center redundant buses through breakers and a transfer switch. Assurance against interconnection of redundant emergency buses via the common MCC is accomplished by the manually operated transfer switch, which enables power supply from only one load center emergency bus.

As a result of NUREG-0578, Item 2.1.1, an additional 42 kW of pressurizer heater capacity was added. These heaters are powered from the 480-volt swing bus using a safety grade breaker.

8.3.1.1.5 120/208-Volt Instrument AC System

The 120/208-volt instrument AC system is designed to provide power for normal station service reactor auxiliary control instrumentation and the instrument fan cooling power supply. This system includes four independent and physically separated instrument panels for the normal station service and reactor auxiliary control instrumentation. These instrument panels are supplied from separate 480-volt engineered safeguards motor control centers in a 480-120/208-volt step-down transformers.

The step-down transformers for panels Y1 and Y2 are rated to carry the full load of both instrument panels. A bus tie with key locked breakers is provided between these instrument panels to supply power to either instrument panel in the event of loss of one source or transformer.

The step-down transformers for panels Y3 and Y4 are rated to carry the full load of their associated instrument panel. There are no Kirk-Key interlock breakers between these panels and no intertie capabilities.

8.3.1.1.6 120-Volt Vital AC System

Four redundant 120-volt vital AC distribution panels are provided to supply power for Nuclear Instrumentation, Reactor Protection Systems, Engineered Safeguards Systems and other vital loads.

Each distribution panel is supplied separately from a static inverter connected to one of the two DC control centers. One swing or standby inverter is provided for each power train. To eliminate circuits for synchronization, load transfer between two inverters requires both units to be in manual bypass (alternate source) during the transfer.

Upon loss of the DC supply, or in the event of an inverter failure, a static transfer switch automatically transfers the 120-volt vital AC load to an AC supply from an engineered safeguards motor control center. After energization of the standby inverter, the panel load can be transferred to the standby inverter.

Each of the four redundant channels of the Nuclear Instrumentation and the Reactor Protection System is supplied from one of the four redundant distribution panels. Also, each of the three redundant channels of the engineered actuation system actuating devices and the associated logic cabinets are supplied from one of the four redundant distribution panels. The system is arranged such that any type of single failure or fault will not prevent proper RPS or ESAS protective action at the system level.

The plant computer, which is non-essential for the Reactor Protection System, is supplied from a Non-Class 1E inverter. This inverter is normally fed from the Non-Class 1E 125 VDC system and from a Class 1E motor control center for alternate power.

8.3.1.1.7 Emergency Power Supply System

8.3.1.1.7.1 Description

Two diesel generators (EDGs), each connected to one of the 4160-volt engineered safeguards buses, are provided. Each diesel generator unit is rated 4160 volts, 2600 kW continuous, with an intended service rating of 2750 kW, which is sufficient to carry the engineered safeguards bus loads following a Design Basis Accident (DBA) and still drive additional motor operated valves.

The "intended service" rating is the ANO specified and EDG manufacturer provided loading value considered as the EDGs' "continuous" rating for the specified operational requirements. This rating is also to be used for purposes of periodic surveillance testing and analysis of postulated DBA loading scenarios. The "intended service" rating (2750 kW) is conservative in that it is less than the EDGs' 2000 hour rating (2850 kW) while the required operating duration during a postulated DBA is only 30 days (720 hours). The diesel generators can be manually started either from the control room, locally, or, on a loss of DC power by using an air start solenoid override at the air start solenoid valve. The field shorting device is removed from the circuit on loss of 125-volt DC power to allow emergency operation of each diesel generator unit.

A sync-check relay is provided to prevent remote manual closing of diesel generator breaker A308 (A408) if: (1) there is no synchronization between the diesel generator DG1 (DG2) and energized bus A3 (A4) or (2) diesel is not running. The relay will permit manual closing of the diesel generator breaker if: (1) the bus A3 (A4) is dead and diesel generator DG1 (DG2) voltage is established or (2) there is synchronization between the diesel generator and bus A3 (A4). The same type of feature, i.e. sync-check, is provided in the bus A3 (A4) normal supply breakers. This relay is provided to prevent remote manual closing of breaker A309 (A409) if there is no synchronization between bus A1 (A2) and bus A3 (A4) and both buses are energized. This relay will permit manual closing of breaker A309 (A409) if: (1) there is synchronization between bus A1 (A2) and bus A3 (A4); (2) bus A3 (A4) is dead and bus A1 (A2) is live; or (3) bus A1 (A2) is dead and bus A3 (A4) is live.

A load profile indicating the starting kVA in addition to the connected loads is shown in Figure 8-2. The starting kW are not shown since the power factor is expected to vary between approximately 0.2 and 0.8 during starting. The computed duration of the transient is shown on Figure 8-2.

Each diesel generator is equipped with a static excitation system. The voltage regulator has a response time of 50 milliseconds. The diesel generators are not used for peaking.

The diesel generators are of such a size that, during the incremental adding of loads, the recommendations of Safety Guide 9 will not be exceeded.

The following events are automatically initiated by the trip of channels 1 and 2 Engineered Safeguards Actuation System (ESAS) channel.

A. Each diesel generator, if not already running and if available, is automatically started.

B. If the associated 4160-volt engineered safeguards bus or 480-volt engineered safeguards bus tie breaker is closed, it is automatically tripped.

C. If the associated 4160/480-volt engineered safeguards load center transformer primary or secondary breaker is open, it is automatically closed.

D. If the associated 4160-volt engineered safeguards bus is being fed from the main bus and bus voltage is normal, the ESAS operated loads on that bus are applied sequentially. The diesel generator breaker is prevented from closing yet the diesel remains running in case normal power is lost later.

E. If the associated 4160-volt engineered safeguards bus is being fed only from the diesel generator and bus voltage remains normal, the ESAS operated loads are applied sequentially and in the order shown in Table 8-1.

The following events trip a diesel generator and, unless corrected or reset, render the diesel generator unavailable:

A. Electrical protection lockout relay trip caused by:

   1. Diesel generator differential relay operation

   2. Diesel generator voltage restraint overcurrent

   3. Diesel generator motoring

   4. Diesel generator loss-of-field

B. Overspeed

C. Low lube oil pressure

D. Overcrank (cranking without starting beyond preset time)

E. Control mode selector switch at diesel in "maintenance" position

F. Diesel stop-start control switch in control room in "lockout" position

G. Crankcase overpressure (if pressure switch is not isolated)

Specifically:

A. The following conditions that render the diesel generator incapable of responding to an automatic emergency start signal are inputs to alarms which warn the operators that the EDG is not available for auto start.

   1. Trouble shutdown relay not reset

   2. Engine control switch in "maintenance" position

   3. Loss of 125-volt DC

   4. Diesel generator control switch in "lockout"

   5. Generator breaker control switch in "pull to lock"

   6. Insufficient starting air pressure

   7. Insufficient fuel in tank

   8. Emergency fuel oil shutoff valve not fully open

B. Control room annunciator window wording for conditions identified in A. include:

D1: "EDG1 not available"

C2: "EDG1 critical trouble"

D2: "EDG1 non-critical trouble"

F4: "EDG fuel oil trouble"

C. Other alarm signals on annunciator:

D1: 1) Diesel generator lockout

C2: 1) Low oil pressure - trip 17
2) Overspeed
3) Low oil pressure - alarm 26
4) High water temperature
5) Ground overcurrent
6) High oil temperature
7) Jacket cooling water low pressure
8) Fuel day tank level low
9) Turbo bearing oil pressure low

D2: 1) Fuel transfer pump auto-start failure
2) Fuel transfer pump handswitch - off
3) Soak back pump (P106A3) - off
4) Soak back pump (P106A2) - off
5) Soak back pump (P106A2) - oil pressure low
6) Water heater auto-start failure
7) Compressor (C4A1)/dryer/aftercooler trouble
8) Compressor (C4A2)/dryer/aftercooler trouble
9) Cooling water expansion tank low level
10) Starting air pressure low

F4: 1) Emergency diesel fuel tank T57A level low
2) Emergency diesel fuel tank T57B level low
3) Emergency diesel trans. pump P16A – disch pressure high
4) Emergency diesel trans. pump P16B – disch pressure high

A1: 1) "EDG1 autostart command"

B2: 1) "EDG1 overcrank"

B1: 1) "EDG1 bkr autoclose failure"

C1: 1) "EDG1 PT fuse blown"

D. There is no unalarmed condition that renders the diesel generator incapable of responding to an automatic emergency start signal.

E. If not specifically alarmed in control room, conditions are alarmed locally (except for T57 A/B and P16 A/B alarms).

F. The crankcase overpressure trip bleeds pressure from the lube oil sensing line (if pressure switch is not isolated), thereby simulating a low lube oil pressure trip. A crankcase overpressure trip is confirmed with local indication.

The diesel generators are housed in Emergency Diesel Generator rooms inside the Seismic Class 1, tornado proof portions of the auxiliary building. The combustion air inlets are positioned such that external missiles cannot penetrate to the diesel generator. Quality assured, 100 percent redundant, auxiliary building Emergency Diesel Generator room exhaust fans protect against excess room temperature.

The bases for the design and location of the diesel generator support systems have been that a single event will not disable both diesel generators. Each diesel generator is protected from missiles generated from outside or from the other engine. Combustion air inlets are protected such that external missiles cannot penetrate to the diesel generator. The exhaust pipe from each engine has Seismic Class 1 hangers.

The combustion air intake is shown in Figure 8-3 and Figure 8-4.
The lubrication system, cooling water system, fuel oil system, and air starting system are shown in Figure 8-3.

Should a fire occur in one Emergency Diesel Generator room that cannot be controlled by hand operated extinguishers, the sprinklers would be used. It would then be necessary to abandon that engine and use the other. Since the engine rooms are separated by fireproof partitions and a double fire door, one engine will be operable even if there is a fire in the other room. Sufficient drainage capability is provided in the Emergency Diesel Generator rooms to maintain the level of flooding water below that at which damage to the equipment could result. A barrier is installed between the redundant rooms of sufficient height to prevent flood water from one room reaching the other room. The drain systems are separated by backflow preventors such that flooding in one room will not cause flooding in the other room through the drain system.

The Fire Suppression Systems for both of the Emergency Diesel Generator rooms are automatic pre-action sprinkler systems. Flame and smoke detectors are provided for fire detection. These detectors automatically open isolation valves to pressurize the Emergency Diesel Generator room fire protection header. The fire protection headers are equipped with fusible heads with water to the headers controlled by remote-manual valves located external to the rooms. Since there is no normally pressurized header in the rooms, it is not considered credible that the rooms would be flooded.

8.3.1.1.7.2 <u>Diesel Fuel Transfer and Storage</u>

The diesel oil supply system for the two diesel generators consists of two large Seismic Class 1 emergency storage tanks located in underground vaults in the plant yard and two smaller Seismic Class 1 "day tanks" located within the supporting frame of the diesel generators. The emergency storage tank inventory is continuously replenished from the conventional above ground fuel oil bulk storage tank by gravity feed through a buried pipeline.

Figure 8-3 shows the components and arrangement for the diesel fuel transfer and storage system.

Two emergency diesel storage fuel tanks and two transfer pumps are contained in a flood proof, Seismic Class 1, excavated vault. The discharge of the transfer pump is connected to a fuel oil day tank. Manual valving is provided to enable either transfer pump to take suction from either fuel tank and discharge to either day tank.

The emergency storage tank vaults are of Seismic Class 1 design and, in addition, have been specifically designed to resist the loadings imposed by the design flood. This includes anchoring the vault to rock and providing ventilation openings above flood elevation. The outside door is of watertight construction.

When a flood is imminent, the inlet to the fuel storage tanks, which is inside the vault, can be closed. There is then an assured inventory of fuel available for at least seven days at full load for one diesel engine. The emergency storage tank vaults are equipped with an automatic deluge sprinkler-type fire protection system which is actuated by smoke detectors in the area or by manual actuation from the vault pull stations located just inside the vault's water tight doors. In addition, each storage tank vault is separated from the other vaults by a 3-hour fireproof door. The volume of the vault below the door elevation is sufficient to contain the oil in the unlikely event of a ruptured fuel tank. Each emergency storage tank vault has a sump which, in turn, is connected to a main sump equipped with a sump pump. The four small sumps are kept isolated from the main sump by normally closed valves. Each emergency storage tank will be gravity connected to Seismic Class 1 transfer pumps located in the underground vault. The transfer pumps discharge to the diesel "day tanks" through buried, Seismic Class 1 pipelines. The transfer pumps are capable of being cross connected at their suctions and discharges with double valving and each is capable of drawing diesel oil from either emergency storage tank and supplying it to either diesel engine. In performing the Appendix R analysis and review of July 1982, "red" and "green" power cables for these pumps were found routed together in a manner that is not in compliance with the Fire Protection separation requirements. Therefore, a cross-connection of the Unit 1 and Unit 2 transfer pumps was made in order that pumps from either unit may supply fuel oil to the diesel generator of the other unit. The deterministic requirements for NFPA 805 are satisfied with this configuration. The diesel oil transfer system has been analyzed for a single failure and satisfactorily meets the criteria. Each emergency storage tank is vented by a Seismic Class 1 vent pipe which penetrates that portion of the vault roof which is above the maximum flood elevation.

The vault is also fireproof. Ventilation supply and exhaust ducts have fire dampers with fusible links which will close in case of fire. In addition, these openings are all located well above the 372-foot floor. When the diesel engines are running at full load, the fuel consumed does not exceed 4 gpm each. Each fuel oil day tank holds about 275 gallons and each emergency diesel fuel tank can hold over 20,000 gallons. Each fuel transfer pump is rated 10 gpm at 58 ft. head.

The diesel fuel oil transfer piping from the emergency storage tanks is designed to ANSI B31.1.0 and in accordance with the additional requirements of Seismic Class 1 and the Quality Assurance Program discussed in Chapter 1.

The total capacity of one underground emergency storage tank plus one diesel day tank will be sufficient for not less than three and one-half days operation of one diesel generator loaded to full capacity. Thus, at least a 7-day total diesel oil inventory will be available onsite in the emergency storage tanks for operation of one diesel generator operation during complete loss of electric power conditions.

Under maximum flood conditions (no nuclear accident), the plant will be shut down and the diesel generator loaded to about 50 percent of the rating of a single unit. Therefore, the emergency fuel oil capacity is expected to last for about seven days of operation, even if one emergency storage tank is considered unavailable. Because station power during a flood event can be supplied by either an EDG or the Startup #2 Transformer, any single failure assumed would not prevent the unit from meeting post-flood safe shutdown requirements. After the Maximum Hypothetical Earthquake (MHE) and a simultaneous nuclear accident, three and one-half days emergency supply of diesel oil is assumed to be available even in the unlikely event one emergency storage tank has failed. Within this period, additional fuel could be delivered to the plant site by any one of three methods: truck delivery, rail car delivery, or delivery by barge from the river. In the highly unlikely event that all three of these normal supply routes are unavailable because of the earthquake, fuel could be airlifted to the plant site via helicopter. It is expected that the maximum probable flood would be above plant grade (Elevation 353') for about two to five days.

All other systems requiring fuel oil, such as the plant heating boiler and the startup boiler, take oil directly or indirectly from the common fuel oil bulk storage tank. See Figure 8-3. Thus, the inventory of diesel fuel oil in the emergency diesel fuel tanks is unaffected by other demands on the fuel oil bulk storage tank.

Because the bulk storage tank is not an essential fuel supply to the diesels, it is not designed to tornado, flood, or single failure criteria.

Testing of the diesel fuel is in accordance with Technical Specification 5.5.13.

8.3.1.1.8 Specific Details of Onsite AC Power Systems

Degraded voltage protection features and the interaction of onsite power sources with load shed features are described in sections 8.3.1.5 and 8.3.1.6.

8.3.1.1.8.1 Interlocks Against Interconnection of Redundant Emergency Buses

The interlocks provided to insure against interconnections of redundant emergency buses are as follows and meet the intent of Safety Guide No. 6:

1. 4160-Volt Redundant Bus

   The two tie breakers between the two 4160-volt redundant buses are interlocked so that both breakers cannot be closed when both emergency buses are supplied from the non-redundant 4160-volt buses (condition during normal operation) or when the two diesel generators are supplying the 4160-volt emergency buses. These interlocks consist of auxiliary contacts on the 4160-volt circuit breakers and interconnecting wiring.

2. 480-Volt Load Center Redundant Breakers

   The two tie breakers between the redundant load center buses are interlocked so that both breakers cannot be closed when the two redundant load center buses receive power simultaneous through the incoming circuit breakers. These interlocks consist of auxiliary contacts on the 480-volt load center breakers and interconnecting wiring.

3. 480-Volt Motor Control Center Redundant Buses

   The redundant motor control center buses have no interconnecting tie breakers, with the exception of common sets of MCC B56 and B55, which receive power from the load center redundant buses through breakers and a transfer switch. Assurance against interconnection of redundant emergency buses via the common motor control center is accomplished by the manually operated transfer switch which can only be in one or the other position enabling power supply from one or the other load center emergency bus.

4. 120-Volt AC Redundant Buses

   Power supply to the 120-volt instrument AC distribution panels Y1 and Y2 are accomplished from either motor control center redundant bus. The tie breakers between the two instrument AC distribution panels are normally open and are provided with keyed locks. Power supply to panel Y3 is accomplished from MCC B57 while panel Y4's supply is accomplished from MCC B65. There are no tie breakers provided between these two instrument panels.

5. 125-Volt DC Redundant Buses

   Interconnection of the two 125-volt DC redundant buses is prevented as described in Section 8.3.2.1.2 and 8.3.2.1.3.

6. The three inverters per power train are arranged for the swing inverter to feed either of the distribution panels associated with the power train. Each inverter has a manual load selector switch. These switches are arranged to ensure contact isolation between the two distribution panels.

8.3.1.2 Analysis

Conformance to General Design Criteria 17, "Electrical Power Systems"

The onsite electrical distribution system arrangement minimizes the vulnerability of vital circuits to physical damage. Two independent circuits can supply power from the different offsite transmission lines through the corresponding transformer to safety oriented components during operating and postulated accident and environmental conditions. Each of these circuits (Startup Transformer 1 and 2) are designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the Reactor Coolant Pressure Boundary are not be exceeded. One of these circuits is designed to be available within a few seconds following a loss-of coolant accident to assure that core cooling, containment integrity, and other vital safety function are maintained.

Provisions are included to minimize the probability of losing electrical power from the remaining sources as a result of, or coincident with, a loss of the nuclear power unit, the transmission network, or the onsite power sources. With a loss of the electrical power generated by the nuclear power unit, auxiliary plant loads will be shifted automatically by fast acting bus transfer devices to the offsite power source fed through the Startup Transformer 1, or either automatically or manually to the Startup Transformer 2 source. Both ANO-1 and ANO-2 are prevented from automatic transfer to Startup Transformer 2 during normal power operations for all buses except Buses A1/A3 (Unit 1) and 2A1/2A3 (Unit 2). Procedures administratively

control Unit 1 and Unit 2 access to Startup Transformer 2. Procedures may allow other fast transfer capabilities to Startup Transformer 2 for specifically analyzed conditions and restrictions defined in approved Engineering Calculations and Evaluations.

The electrical power systems conform to Criterion 17 through provision of an offsite 500 kV/161 kV transmission system and onsite diesel engine generators and batteries. Each of these systems provides sufficient capacity and capability to assure operation of the necessary safety functions during anticipated operational occurrences and postulated accidents assuming the other system is not functioning.

The onsite power supplies, including the diesel generators, batteries, and distribution system, have sufficient independence, redundancy, and testability to perform their safety functions. Both offsite power sources are designed for non-interrupted availability during a LOCA and the two diesel generators are designed for a maximum starting of 15 seconds from admission of starting air to being ready for loading. Normally, the two 4160-volt engineered safeguards buses are fed on a one-for-one basis from the main buses. Upon loss of normal (Auxiliary Transformer) and standby (offsite) power sources, the two 4160-volt engineered safeguards buses are each energized from their respective diesels. With the loss of all offsite power to the engineered safeguards buses, the associated bus is cleared of all auxiliaries and ties prior to application of the associated diesel generator. This prevents loss of the diesel generator as a result of an offsite power fault. Because of the design of the fault protection system, there is a low probability that loss of the onsite diesel generator power sources could cause loss of either of the offsite nuclear unit electrical power sources.

Conformance to General Design Criterion 18, "Inspection and Testing of Electrical Power Systems"

All important passive components of the emergency power system such as wiring, insulation, connections, and switchboards are designed to permit appropriate, periodic inspection and testing to assess their continuity and condition.

System design provides for the following periodic Emergency Diesel Generator electrical tests:

A. Each diesel generator is manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel is varied from one test to another to verify all starting circuits are operable. The generator is synchronized from the control room and loaded.

B. A test is conducted as directed by the Surveillance Frequency Control Program to demonstrate the overall automatic operation of the emergency power system. The test is initiated by a simulated simultaneous loss of normal and standby power sources and simulated ES signal. Proper operations are verified by bus load shedding and automatic starting of selected motors and equipment to establish that restoration with emergency power has been accomplished within a limited time interval, approximately 70 seconds.

The tests specified are designed to demonstrate that the diesel generators will provide adequate power for operation of equipment. They also assure that the diesel generator and bus load sequencing control systems for safeguard equipment will function automatically in the event of a loss of all normal 4160-volt outside power and the presence of an Engineered Safeguards (ES) signal.

Frequent tests will be made to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The control components are in dust tight enclosures having space heaters for humidity control. The fuel supply and starting circuits and control are continuously monitored and any faults are annunciated. An abnormal condition in these systems will be signaled without having to place the diesel generators on test.

Conformance with IEEE 279-1971

A brief description of the conformance of the undervoltage protection and load shedding features to the requirements of IEEE 279-1971, is provided in Section 8.3.1.6.

Conformance with Regulation 10CFR50.63, "Loss of All Alternating Current Power"

The Alternate AC Power Source, which is common to both Unit 1 and Unit 2, complies with the requirements of Section C(2) of the Criteria in that the Alternate AC Power Source is independent of other onsite or offsite power sources and can be started by Unit 2 Control Room operators when needed. After closure of the appropriate bus 2A9 breaker, the final connection to bus A3 or A4 is performed by Unit 1 Control Room operators.

Conformance with Regulatory Guide 1.155, "Station Blackout"

The Alternate AC Power Source conforms to the requirements in Section 3.3.5 of Regulatory Guide 1.155.

The Alternate AC Power Source can be started from the Unit 2 Control Room or locally at the engine/generator control panel. Unit 2 4160 volt switchgear 2A9 is dedicated to the Alternate AC Power Source and is available to make the proper connection to the Unit 1 buses.

8.3.1.3 Compliance with Appropriate QA Requirements

The routing of all cables is verified by an engineer or craft supervisor. Routed cables are selected and independently verified by Quality Control to assure proper routing and adherence to the separation criteria.

Procedures are provided for the scheduling and performance of relay inspection and testing. Surveillance is performed in accordance with Technical Specifications and the procedure acceptance criteria.

8.3.1.4 Independence of Redundant Systems

The redundant systems are designed to be physically independent of each other so that failure of one train or channel will not jeopardize safe shutdown of the reactor.

The Class 1E electric systems are designed to insure that any of the design basis events listed in IEEE 308-1971 will not prevent operation of the minimum number of ESF equipments required to safely shutdown the reactor and to maintain a safe post-shutdown condition.

The Class 1E power system is designed to meet the requirements of IEEE 279-1971, IEEE 308-1971, 10 CFR 50, including Appendices A and B, and Safety Guide 1.6. ESF loads are separated into two completely redundant load groups. Each load group has adequate capacity to start and operate a sufficient number of ESF loads to safely shut down the reactor, without

exceeding fuel design limits or Reactor Coolant Pressure Boundary limits, during normal operation or design basis event. As required by IEEE 308 and 10 CFR 50, General Design Criterion 17, each redundant ESF load can be powered by both onsite and offsite power supplies. Two diesel generators, one on each ESF bus, will furnish the required onsite emergency ESF power supply requirements. Consistent with Regulatory Guide 1.6, no provision exists for automatically transferring loads between the redundant power sources. Further, the redundant load groups cannot be automatically connected to each other nor can the two emergency power sources be paralleled automatically. Separation and independence have been maintained between all redundant systems, including the raceways, so that any component failure in one ESF channel will not disable the other ESF channel.

8.3.1.4.2 Design Criteria and Bases for Cable

The application and routing of control, instrumentation, and power cables minimizes their vulnerability to damage from any source. All cables are sized with conservative margins for their current carrying capacities, insulation properties, and mechanical construction. Cable insulation in the reactor building was selected to minimize the harmful effects of radiation, heat, and humidity.

Appropriate instrumentation cables are shielded to minimize induced voltages and magnetic interferences. Cables related to Engineered Safeguards and Reactor Protective Systems have color coded identification and have been routed and installed to maintain the integrity of their respective redundant channels. Power and control cables for redundant auxiliaries or services are run by different routes to reduce any probability of disabling more than one piece of equipment.

8.3.1.4.2.1 Cable Derating and Cable Tray Fill

Base capacity rating of cable is as conservative as or more conservative than that established in published IPCEA standards and in accordance with the manufacturer's standards. Motor feeder cables are selected based on 100 percent load factor. Cables for inside the reactor building are derated to allow for the higher ambient temperature.

Tray fill limitations are 30 percent. This is not applicable for trays installed in restricted areas where it is physically impossible to adhere to the guideline, such as in raised floors in the computer room and in the control room. In these locations, the raised floors contain only electrical, low-energy cables and trays merely serve as mechanical protection during installation of these cables to the control panels mounted on the raised floor. Also, exception to the fill guideline is made for circuits that normally carry electrical current in some, but not all, conductors of a cable. Tray filling in excess of the guidelines is only permitted upon review of each case by engineering with regards to the nature of circuits, physical location, heating effect, and seismic loading. Conduit fill limitation is 40 percent in general. This is only exceeded if the conduit contains cables in which, by design, only some, but not all, conductors carry electrical current or upon review of each case by engineering with regard to the nature of the circuits, physical location, heating effects, and the potential for cable damage during installation. Sleeves used for wall or floor penetrations are not considered a conduit and therefore not subject to fill limitations.

8.3.1.4.2.2 Cable Routing in Congested Areas and/or Hostile Environment

8.3.1.4.2.2.1 Cable Routing Criteria

A. Separate cable tray, conduit, and penetration systems are installed for the following classes of cable; 8 kV, 5 kV, 600-volt power and control, and instrumentation cable. In general, power cables are run in the top trays with control and instrumentation in the lower trays.

B. Shielded instrumentation cables and thermocouple cables are not run in the same trays as the control cables.

C. The Fire Hazards Analysis, described in Section 9.8, considered the potential effects on safe shutdown capability where redundant cables in the same fire zone or compartment were separated by less than 20 feet.

D. Fire stops are provided where cable trays pass through wall or floor openings as described in Section 9.8.

E. Spacing of trays containing only non-redundant electrical cables is determined by the access required for installation.

F. No electrical equipment, such as power distribution panels, transformers, motors, etc., carrying heavy electrical currents and no equipment or systems containing flammable fluids are installed in the cable spreading room or underneath the raised floors. This room, and the space underneath these raised floors, is used solely for the routing of electrical cables to control panels.

   120-volt vital AC control power distribution panels furnishing power to redundant logic circuits are mounted in the control room. Also, redundant air conditioning units are installed inside the control rooms and are provided with redundant 480-volt, 3-phase power.

G. Redundant cables are routed only in conduits inside the cable spreading room and under the raised floor in the control room.

H. Separation that was maintained during initial construction for trays containing redundant cables.

   1. In rooms containing heavy rotating machinery or high pressure pipelines, a minimum separation of 20 feet or a 6-inch thick reinforced concrete wall was provided. In other areas, a separation of three feet or a barrier equivalent to one inch of transite covered with a sheet of 16 gauge steel was generally provided.

   2. Generally, 3 feet horizontal and 5 feet vertical separation distances were maintained between redundant cable trays. Where these separation distances were not maintained, tray covers or barriers were provided using IEEE 384-1974 as guidance. An evaluation concluded that the omissions of tray covers or barriers is not safety significant when the separation distance is greater than 1 inch. Although their omission is not safety significant, the use of cable tray covers and barriers per IEEE 384-1974 is a good engineering practice and should continue to be used as guidance or recommendations.

3. Since this separation and these type barriers may not provide adequate protection from a large exposure fire, the Fire Hazards Analysis described in Section 9.8 considered the effects on safe shutdown capability if redundant cables that are separated by less than 20 feet are lost.

8.3.1.4.2.2.2 Cable Reliability

Cable reliability for circuits in the reactor building was insured by selecting cable insulation to minimize the harmful effects of radiation, heat, and humidity.

All electrical conductor insulation is flame resistant and specified to be qualification tested for the service required.

8.3.1.4.3 Sharing Cable Trays for Safety and Non-Safety Related Cables

Non-engineered safeguards circuits and engineered safeguards circuits may share the same cable trays but the routing is controlled such that the non-engineered safeguards circuits will not cross over into other redundant engineered safeguards cable trays. Where a non-vital cable from a channel "2" redundant source (AC or DC) is routed with the vital cables that are supplied from the other redundant source, channel (1), and vice versa, this non-vital cable is run in steel conduit between the channel "2" source and the channel "1" cable tray and vice versa.

8.3.1.4.4 Cable and Cable Tray Markings

Cables and raceways are identified by numbered and colored markers. Non-critical cables are identified by black jackets and critical cables by color coded jackets. Additionally, each cable is identified by permanently affixed cable markers. Markers have channel number plus a color mark corresponding to the cable jacket color. Non-critical cable markers have no color mark.

Cable raceways are identified by adhesive vinyl tray markers. Trays or conduits used for reactor protection or engineered safety features channels have color spots corresponding to the channel color code. For example, a channel 1 tray contains only red-marked cables with red-marked cable markers. Although other cables with black jackets that are not part of reactor protection or engineered safety features may share a color-marked tray, in no case will these cables with black jackets cross from one tray channel to another.

The above identification schemes comply with Section 4.22 of IEEE 279-1971.

8.3.1.4.5 Spacing of Wiring and Components Associated with Class 1E Electric System in Control Boards, Etc.

Protection system, safety feature system, and Class 1 electrical system components mounted on control boards, panels, and relay racks are designed for operator convenience and physical separation between redundant wiring and components.

Generally, redundant channel wiring enters the control panels in conduits. The bulk of redundant wiring inside control panels are separated by a steel barrier.

However, wiring which is common to two different redundant channels exists. These common wires are color coded green. For example, this necessitates that in certain cases green wires must pass through red compartments. However these wires are routed in conduits inside the panels except where physically impossible.

Redundant wiring and components in control boards, panels, and racks are either separated by a steel barrier or at least six inches of space. Some wiring and components are common to the two redundant safeguards channels. Also, in many cases interlocks are required between equipment of different channels to perform certain safety functions. The cables containing these common or interlocking conductors are color coded green when they are routed through the raceway system between panels belonging to different safeguards channels. These cables are run in flexible steel conduits inside the control panels. The flexible steel conduit is extended as close to the terminals as physically possible.

Since this separation may not be adequate for a large fire in a control panel, the Fire Hazards Analysis described in Section 9.8 considered the effects on safe shutdown capability for a fire that damages all wiring within a control panel and the risk of fire spread between them. Protective features were provided as described in Section 9.8.

8.3.1.4.6 Additional Criteria

A. Fire protection systems (smoke detectors and sprinklers) are provided near the electrical penetrations inside and outside of the containment building. Cable spreading rooms for Units 1 and 2 are separate and each is surrounded by 3-hour fire rated walls. Each room has its own fire detection and suppression system. A discussion of the fire detection and suppression systems can be found in Chapter 9.

B. The physical locations of electrical distribution system equipment are to minimize vulnerability of vital circuits to physical damage. Additional detail on fire protection feature is contained in Section 9.8.

   1. The UAT and Startup Transformers are located out of doors with sufficient physical separation from each other or separated by fire walls. Lightning arresters are used where applicable for lightning protection. All transformers are protected by automatic water spray systems to extinguish oil fires quickly and prevent the spread of fire.

   2. The engineered safeguards 4160-volt switchgear and 480-volt load centers are located within a Class 1 structure area to minimize exposure to mechanical, fire, and water damage. This equipment is properly coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions.

   3. 480-volt motor control centers are located in the areas of electrical load concentration. Those associated with the turbine generator auxiliary system in general are located below the turbine generator operating floor level. Those associated with the Nuclear Steam Supply System (NSSS) are located in areas to minimize their exposure to mechanical, fire, and water damage. Engineered safeguards motor control centers are located within a Class 1 structure.

4. Coolers, which maintain electrical equipment within acceptable operating temperature bands, are provided in the following: battery/charger rooms, electrical switchgear rooms, and electrical equipment rooms. Details of associated ventilation systems for electrical systems are described in Chapter 9.

5. Within practical limits, metal-enclosed 6900-volt and 4160-volt buses are used for all major bus runs where large blocks of power are to be carried. The routing of this metal-enclosed bus minimizes its exposure to mechanical, fire, and water damage.

8.3.1.4.7 Conformance to Separation Criteria

A. The separation of redundant cables of the Reactor Protection System and Engineered Safeguards System circuits is accomplished by spatial separation where cables are installed in trays.

B. Historical data removed - To review the exact wording, please refer to Section 8.2.2.8.f(10) of the FSAR.

C. The installation of all cables is verified as described in Section 8.3.1.3.

8.3.1.5 Grid Undervoltage Protection (Millstone 2 and ANO Events)

On July 20, 1976, an event occurred at Millstone 2 which involved power system undervoltage problems. The following discussion is a summary of the event and how it affected ANO-1 (1CAN087815, 1CAN107809, 1CAN087913). Based upon the Millstone event, each licensee was requested to supply information regarding the design of the Class 1E electrical distribution system, including a description of the load shedding features, definition of the facility operating limits, and a description of any proposed actions or modifications. AP&L responded to the request for information by providing analyses which demonstrated the adequacy of the ANO-1 design. Subsequently, the NRC developed a number of positions with respect to this subject. During the time frame of the NRC questions/positions and AP&L's responses, a significant related event occurred at ANO. This event, along with the Millstone event, led to several modifications of the ANO-1 electrical design. The following lists these positions along with the ANO-1 responses and subsequent design changes.

8.3.1.5.1 NRC Positions/ANO Responses

Position 1: Second Level of Under- or Over-Voltage Protection With a Time Delay

Requirement that a second level of voltage protection for the onsite power system be provided and that this second level satisfy the following criteria:

A. The selection of voltage and time setpoints should be determined from an analysis of the voltage requirements of the safety related loads at all onsite system distribution levels;

B. The voltage protection should include coincidence logic to preclude spurious trips of the offsite power source;

C. The time delay selected should be based on the following conditions:

1. The allowable time delay, including margin, should not exceed the maximum time delay that is assumed in the FSAR accident analyses;

2. The time delay should minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and,

3. The allowable time duration of a degraded voltage condition at all distribution system levels should not result in failure of safety systems or components.

D. The voltage monitors should automatically initiate the disconnection of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded;

E. The voltage monitors should be designed to satisfy the requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Generating Stations;" and

F. The Technical Specifications should include limiting conditions for operation, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second level voltage protection monitors.

Position 1: Response

* Note: These values are historical in nature, refer to the Unit 1 Technical Specifications for allowable values.

A. Three levels of voltage protection for the onsite power system will be provided as follows:

1. Two definite-time undervoltage relays on each 4160-volt safety bus with a nominal voltage setting of 78* percent (of motor base voltage) and time dial setting of 1.0.

   Upon loss of power, these undervoltage relays will, in approximately 2.6 seconds, initiate load shedding and starting of the associated diesel generator. Isolation of the safety related buses will be delayed approximately 2.5 seconds to allow a fast transfer to offsite power. The safety related bus will be isolated only if the fast transfer is unsuccessful.

2. Two instantaneous undervoltage relays on each safety-related 480-volt load center bus with a nominal setting of 92* percent of 460 volts.

   Upon voltage degradation to 92* percent of 460 volts and after a delay of eight seconds, the relay will isolate the associated safety related 4160-volt bus from offsite power, start and connect the associated diesel generator. To prevent spurious actuation, this circuit is blocked when large motors are being started.

3. Two definite-time undervoltage relays on each safety-related 480-volt load center bus with a nominal setting of 88.6 percent of 460 volts. With an Engineered Safeguards Actuation System (ESAS) signal present, upon voltage degradation to 88.6 percent of 460 volts and after a delay of 3.3 seconds, the relay will isolate the

associated safety related 4160-volt bus from offsite power, and start and connect the associated diesel generator.

B. The loss of power (78* percent) relays described in (A.1) will be connected in parallel so either relay will isolate the associated safety bus from the offsite source.

The degraded voltage (92* percent) relays described in (A.2) will be connected in series to provide the required coincidence logic.

The motor start protection (88.6 percent) relays described in (A.3) will be connected in series to provide the required coincidence logic.

C. As stated in (A.1), the maximum time between total loss of power and power restoration from an offsite source is approximately 5.1 seconds. The diesel generator is independently started during this interval and will start sequencing Engineered Safeguards (ES) loads in 15 seconds.

1. The above intervals are within the limits assumed in the FSAR accident analysis.

2. The 92* percent undervoltage relays are delayed 8.0 seconds as stated in (A.2) above. This delay is adequate to prevent spurious operation of the relay when motors start on the safety related 4160- or 480-volt buses. An interlock will be added to prevent operation of the timer when large non-safety related motors are being started.

3. The 88.6 percent undervoltage relays are delayed 3.3 seconds as stated in (A.3) above. This delay is adequate to prevent spurious operation of the relay when motors start on the safety related 4160- or 480-volt buses. These relays are wired in parallel with the 92* percent undervoltage relays.

4. Under the conditions identified by the system analysis, the safety related equipment will function satisfactorily.

D. The voltage monitors will automatically initiate the disconnecting of the offsite power sources whenever the voltage and time setpoints have been exceeded.

E. The voltage monitors and the associated circuitry are designed as Class 1E circuits in accordance with IEEE 279-1971.

F. The Technical Specifications will address these requirements.

Position 2: <u>Interaction of Onsite Power Sources with Load Shed Feature</u>

<u>Requirement that the current system design automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses.</u> This design also includes the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement features will be verified during the periodic testing identified in Position 3.

Position 2: <u>Response</u>

The design will be modified to prevent load shedding from the emergency buses when a safety signal is present and the diesel generators are supplying power.

Position 3: Onsite Power Source Testing

Requirement that the Technical Specifications include a test requirement to demonstrate the full functional operability and independence of the onsite power at least once per 18 months during shutdown. The Technical Specifications will include a requirement for tests: (1) simulating loss of offsite power in conjunction with a safety injection actuation signal and (2) simulating interruption and subsequent reconnection of onsite power sources to their respective buses. Proper operation will be determined by:

A. Verifying that upon loss of offsite power the emergency buses have been de-energized and that the loads have been shed from the emergency buses in accordance with design requirements;

B. Verifying that upon loss of offsite power the diesel generators start from ambient condition on the autostart signal, the emergency buses are energized with permanently connected loads, the autoconnected emergency loads are energized through the load sequencer and the system operates for five minutes while the generators are loaded with the emergency loads; and

C. Verifying that upon interruption of the onsite sources the loads are shed from the emergency buses in accordance with design requirements and that subsequent loading of the onsite sources is through the load sequencer.

Position 3: Response

The Technical Specifications (TSs) address this requirement with a test performed during outages. Originally, this test was performed every 18 months. Later, ANO-1 implemented the Surveillance Frequency Control Program (SFCP) which relocated the surveillance frequencies of several TS surveillance requirements to licensee control. The frequency for this test is now controlled by the SFCP.

A. Simulating loss of offsite power in conjunction with an ESAS and

B. Simulating interruption and subsequent reconnection of onsite power sources to their respective buses.

8.3.1.6 Description of Changes

Based on the analyses performed and in response to NRC positions, the following changes were incorporated:

A. Circuit Changes

1. Installation of second level undervoltage relays, two on each 480-volt Class 1E bus, with drop out setting at 92* percent (motor base voltage) and 8 second time delay in coincident trip logic (2-out-of-2) degraded voltage protection.

Upon installation, the relays will be calibrated using instruments of 0.02 percent accuracy. Calibration will be repeated quarterly until the next refueling and one datum will be used to determine if the setpoint should be changed to accommodate the drift. Test-in-service pushbuttons are provided in the circuit for periodic testing of the undervoltage trip logic.

2. Blocking 92* percent relays trip during start of Reactor Coolant Pumps.

   The blocking circuitry will be Class 1E and will function for 20 seconds during the starting of the Reactor Coolant Pump (starting of condensate pump does not drop bus voltage below acceptable level and thus no blocking of 92 percent trip relay is required when condensate pump starts). A Class 1E timer will be added to annunciate a delayed alarm in the control room in case the blocking circuit does not automatically reset.

3. Delete slow automatic transfer from UAT to Startup Transformer 1.

4. Install two definite-time undervoltage relays on each 4160-volt Class 1E bus with 78* percent (motor voltage base) drop out setting in parallel trip logic (1-out-of-2) for loss of offsite power protection.

5. For fast transfer to offsite source with a safety signal:

   a. Shed selected non-ESF loads.

   b. Sequence idle ESF loads.

   c. Delay selected 480-volt ESF loads.

6. For offsite source available but fast-transfer fails:

Note: If an ESAS signal is present, depending on conditions at the time, it is possible for the MSP relays to actuate and tie safety-related loads onto the respective EDG before a transfer to offsite power can complete.

   a. Shed selected non-ESF and ESF loads

   b. Sequence ESF loads to offsite source.

   c. Delay selected 480-volt ESF loads.

   d. When a fast transfer from the UAT is to Startup Transformer 1, all loads except ESF loads are transferred in bulk and ESF loads are subsequently sequenced if needed. Shedding of non-ESF loads is not required.

   e. When a fast transfer from the UAT is to Startup Transformer 2 (in case Startup Transformer 1 is not available), certain selected Non-Class 1E loads will be shed after transfer takes place.

7. Block load shedding feature on the 4160-volt safety buses when the diesel generators are supplying these buses. Automatically reinstate the feature when the onsite source supply breakers are tripped.

8. Add two definite-time undervoltage relays on each 480-volt Class 1E bus with 88.6 percent (motor voltage base) drop out setting in series trip logic (2-out-of-2) for motor start protection.

* Note: The assumptions in B.1 are historical in nature; refer to CALC-95-E-0001-06 for current assumptions.

B. Equipment Changes

1. Increase the size of control transformers as required based on the following voltage study.

   The calculations used in the study were based on these assumptions:

   a. Assume 90 percent of motor rated voltage 460 (0.9 x 460 = 414) is available at the input side of the Control Power Transformer (CPT).

   b. Calculation of the CPT voltage drop is made during the in-rush current prior to initial movement of contactor armature.

   c. Next, calculate the output voltage at the CPT.

   d. Calculate the difference between output voltage at the CPT terminal and minimum voltage required to successfully close the contactor coil. The difference is the maximum allowable voltage drop for control wiring.

   e. Compute the length of control cables that would give the allowable drop mentioned above.

   f. For those starters that have long control cables that did not meet the above criterion, replace the CPT with a larger size.

2. Add auxiliary interposing relays in starters as required.

CONFORMANCE TO IEEE 279

The following briefly describes how the ANO-1 Millstone Modifications conform to the requirements of each section of IEEE 279.

A. General Functional Requirements

   The devices and equipment used are qualified for Class 1E applications and the performance of the devices is highly reliable. The system is designed so that the protective action is automatically initiated as the system reaches preset levels.

B. Single Failure Criterion

The two load groups are provided with redundant protective actuation control systems. Also, wiring for each of the two control systems is routed in separate Class 1E raceways.

C. Quality of Components and Modules

The devices used for the two protection systems are qualified for Class 1E application.

D. Equipment Qualifications

The available type test data for Class 1E components qualifications confirms the required satisfactory performance of the protection equipment under the environmental conditions stated in Sections 3.7 & 3.9 of IEEE 279-1971. Further, the components have been demonstrated qualified in accordance with the ANO EQ Program Manual, NES-01, which meets the requirements of 10 CFR 50.49.

E. Channel Integrity

The protective systems proposed have been designed for fail-safe operation. The devices and circuitry used are Class 1E and, therefore, will remain operational under extreme environmental, energy supply, and accident conditions.

F. Channel Independence

The equipment, devices, and raceways for one Class 1E system are independent and physically separated from the other system. The circuits for the two Class 1E protective systems are also routed in separate raceways.

G. Control and Protection System Interactions

All the equipment is considered part of the protection system which is designed to meet the requirements of IEEE 279 with the exception of the RCP starting bypass initiation signal, i.e. the contact from which the bypass signal is initiated. The signal, however, is isolated from the Class 1E portions of the system by a Class 1E buffer relay.

H. Isolation Devices

Isolation devices are not used as the systems are completely Class 1E.

I. Single Random Failure and Multiple Failures Resulting From a Credible Single Event.

There are no single failure points as the systems are completely Class 1E, separate, and redundant.

J. Derivation of System Inputs.

The undervoltage relays proposed at the 480-volt ESF buses will measure the system degraded conditions directly and initiate the protective action at the system level within its respective load group.

K. Capability for Sensor Checks

The 92* percent undervoltage relays have been provided with functional test switches. Periodic testing will insure the sensors operational capability.

L. Capability for Test and Calibration

The system has the capability for testing.

M. Channel Bypass or Removal From Operation

Systems have the capability to be tested in service without initiating a protective action at the system level and also continue to meet the single failure criterion.

N. Operating Bypass

The protective action to the two systems is automatically bypassed manually during starting of the RCP motors. The operating bypasses are Class 1E.

O. Indication of Bypass

The bypasses will be alarmed in the control room to indicate that the bypass failed to reset. Test bypasses are not alarmed in the control room.

P. Access to Means for Bypassing

Manual bypass of protective action is provided through test switches. The access to the test switches will be under the administrative control of Operations.

Q. Multiple Setpoints

The protective devices are set at one setpoint only.

R. Completion of Protective Action Once it is Initiated

Once the protective action has been initiated, the offsite source is automatically disconnected and the onsite source (diesel) is made available within a short time. The protective action will go to completion once initiated.

S. Manual Initiation

Manual control is provided on each of the two breakers for connecting or disconnecting the offsite and onsite sources to the auxiliary power systems. Manual initiation requires operation of a minimum number of switches.

T. Access to Setpoints Adjustment, Calibration and Test Points

Access to setpoint adjustment, calibration, and test points is controlled administratively by Operations and is limited to qualified personnel.

U. Identification of Protective Action

The breakers for the offsite and onsite sources have close and open indications in the control room to identify the protective actions.

V. Information Readout

Sufficient monitoring has been provided in the control room which will enable the operator to know the deteriorating conditions of the system.

W. System Repair

Periodic testing of the system will insure the detection of the malfunction of components or modules. Plug in type of components are used where possible so that the faulty units can be replaced, repaired, or adjusted expeditiously. Also, the system protective action is designed such that the failure of one undervoltage relay will not disable protective action.

X. Identification

The equipment and wiring of the two protective systems have been identified as red and green channels.

TECHNICAL SPECIFICATIONS

As part of the resolution of this issue, the Technical Specifications were revised to include the undervoltage relay settings and requirements for surveillance testing and limiting conditions for operation. See the Technical Specifications for further information.

8.3.2 DC POWER SYSTEM

8.3.2.1 Description

The Station DC system consists of two 1E 125-volt batteries, each with two battery chargers, DC control center, distribution panels, and motor control center in addition to a Non-1E 125-volt battery with its own charger, DC control center, and distribution panel.

The system is designed to provide a continuous, reliable, and redundant 125 VDC power source for control, instrumentation, and DC loads required for normal operation and orderly shutdown and control of the station.

The 1E 125 VDC system consists of two independent, physically and electrically separated 125 volt batteries designated D06 and D07 which provide DC power to the 125 VDC control centers and distribution panels. Four battery chargers are supplied with two serving as normal supplies to the DC control centers with the associated battery floating on the bus. The second battery charger on each bus serves as a standby battery charger. The battery chargers are supplied from separate 480-volt engineered safeguard motor control centers. The batteries supply the load without interruption should the battery chargers fail.

The Non-1E 125 VDC system consists of a single (black) train 125 VDC battery designated D45 which provides power to control center D41. Battery charger D42 serves as the normal supply to D41 with battery D45 maintained in a float condition. Charger D42 is fed from Non-1E 480 VAC MCC B11. The battery supplies the load without interruption should the battery charger fail.

8.3.2.1.1 Batteries

Each of the 1E batteries, designated D06 and D07 consists of cells assembled in heat and shock resisting, polycarbonate, noncombustible jars. They are of the lead-calcium type, consisting of 58 cells each and are rated at 1423 ampere-hours for an 8-hour rate of discharge to a cell voltage of 1.81 volts per cell. Each battery is sized to carry the continuous emergency DC and vital AC load for a minimum period of two hours in addition to supplying power for the operation of momentary loads during the 2-hour period. According to the battery manufacture, a float current of less than 2 amps indicates that the battery is at least 98% charged. Since only a 98% charge can be assumed, a 2% design margin factor is maintained in the battery sizing calculations to ensure 100% battery capacity is maintained.

The safety related batteries are sized per IEEE 485-1983. The minimum design limit for electrolyte temperature is 60 °F. The batteries are sized to include a temperature correction factor to compensate for the possibility of a 60 °F operating temperature. The batteries are also sized to include an aging factor to ensure adequate end of life battery capacity exists. In addition, the batteries are sized to include a design margin factor to account for factors other than aging and temperature. These margins are maintained under the Margin Management program.

Non-1E battery D45 consists of 60 cells contained in polycarbonate, noncombustible jars. The battery cells are of the lead calcium type and are sized to carry the battery loads for a minimum period of 2 hours.

In normal operation, the batteries are floated on the buses and assume load without interruption on loss of a battery charger. Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 127.6 V (calculated based on 2.20 V * 58 cells) at the battery terminals, or 2.20 Vpc. This is the minimum design limit for battery terminal float voltage established by the battery manufacturer and provides adequate over-potential, limiting the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. The ungrounded DC systems have detectors which alarm when there is a ground existing on either polarity. A ground on one side of the DC system will not cause any equipment to malfunction.

Monitoring features are provided to monitor battery discharge and a failure of the charger (see Section 8.3.2.2.1). The subsequent alarm in the control room cannot be removed until proper battery charging conditions are restored.

The adequacy of the components of the 1E 125 VDC system (except the battery cells, battery chargers, and battery racks) to withstand the forces experienced during a design earthquake is evaluated by means of seismic analysis. These analyses consist of calculations performed by the vendor of the equipment and are checked by Engineering. The analyses show that the equipment is adequate to withstand these forces.

A prototype of the battery cells, battery chargers, and racks has been tested and has demonstrated the ability to successfully withstand the seismic forces experienced during a design earthquake.

8.3.2.1.2 <u>Battery Chargers</u>

The 1E 125 VDC chargers, designated D03A, D03B, D04A, and D04B (one standby backup per power train) are of the thyristor full-wave rectifier, constant average voltage, type. They are of the convection cooled type, rated for continuous operation at 122 °F ambient. Each charger is rated at 400 A and suitable for float charging or equalizing the 125 volt DC battery. The chargers operate from 460-volt, 3-phase, 60 Hz supplies from 480-volt Class 1E MCCs. Each charger is adequate to supply the normal continuous DC load connected to its respective control center and keep the associated battery in a fully charged condition. Each charger is designed to prevent the battery from discharging back into any internal charger load in case of AC power supply failure or charger malfunction. The chargers may be used as battery eliminators so that they can supply 125 volt DC power if the associated battery has to be taken out of service for testing or becomes unavailable for any reason.

Each power train has a standby normally de-energized battery charger. Load sharing and transfer controls are provided for load transfer between the two battery chargers. Other than during load transfers or maintenance only one battery charger per train will be energized.

The Non-1E battery charger D42 is of the full wave constant voltage, regulated and filtered type with SCR control. It is convection cooled and rated for continuous operation at 104 °F. The housing is a free standing NEMA-1 enclosure. D42 is rated for an output of 400 A. The charger maintains a nominal float voltage of 130 volts and a nominal equalize voltage of 140 volts. Both are adjustable ± 5 percent. The charger maintains a ± 1 percent DC output voltage regulation and will accept an AC input of 480 VAC, ± 10 percent, and a frequency variation of 60 Hz, ± 5 percent. D42 is adequately sized to provide power to the normally connected loads on D41 and charge battery D45.

8.3.2.1.3 DC Control Centers

The 1E 125 VDC system is provided with two redundant 125VDC control centers. Each control center has an independent, physically and electrically separated DC bus arrangement. Each bus is connected to a corresponding 125 volt battery, a normal battery charger source, and a manual standby battery charger.

The two redundant DC buses, supplied from separate batteries, have tie connections through manually operated breakers and transfer switches. The redundant buses cannot be interconnected since the manually operated transfer switch has one contact open when the other contact is closed.

The Non-1E battery D45 provides power to a single DC control center D41.

8.3.2.1.4 Distribution Panels and Motor Control Centers

Four 125 VDC distribution panels fed from the 1E buses designated RA1, RA2, D11, and D21, have been provided. Two 125 VDC motor control centers (D15, D25) have been provided. Panels D11 and D21 are provided with a transfer switch allowing selection of an emergency power source from either DC control center, as described in Section 8.3.2.1.3. The two engineered safeguards actuation control distribution panels, RA1 and RA2, are supplied from the two 125-volt DC control centers.

One 125 VDC distribution panel, D46, is fed from the Non-1E DC bus.

8.3.2.1.5 System Operation

Each DC system, operated ungrounded with a ground fault detector relay, is set to annunciate the first ground on either the positive or the negative leg of the system. The annunciator provides an opportunity for corrective action before a second ground might occur. Single ground fault will not cause a DC circuit failure. With an arrangement of this type, two grounds are required before any of the circuit protective devices operate.

One undervoltage relay is provided on each bus section to initiate an alarm if voltage on the bus drops to a preset value. A charger failure relay provided on each charger detects and annunciates failures in AC power input and DC power output.

Cables for the DC power and control systems are rated 600 volts with flame resistant insulation and jacketing materials capable of withstanding the worst case environmental conditions to which the cable could be exposed and copper conductors sized to carry the maximum available short circuit current for the time required by the primary circuit breaker to clear the fault. Additional information on cables and raceway for the DC power supply system is provided in Section 8.3.1.

8.3.2.1.6 Equipment Separation and Redundancy

The 1E 125-volt DC system is designed to meet the Seismic Category 1 requirement. The two redundant batteries and their related accessories are located in separate rooms in the auxiliary building which is designed as a Seismic Category 1 structure and they are protected from potential missile hazards. The safety related DC loads have been grouped into two redundant

load groups such that the loss of either group will not prevent the minimum safety function from being performed. Complete separation and independence have been maintained between components and circuits of the two 125-volt ESF DC systems, including the raceways. For the raceway separation criteria, see Section 8.3.1. Because of the physical and electrical separation provided for the batteries, chargers, distribution equipment, and wiring for the 125-volt DC ESF systems, a single failure at any point in either system will not disable both 125-volt DC systems.

The Non-1E 125 VDC system is non-seismic and is physically located separate from any 1E component or system.

8.3.2.1.7 Ventilation

Each 1E 125-volt DC system battery and its related charger and the distribution equipment are located in a separate, ventilated room. Exhaust fans (VEF 33, VEF 34) insure a buildup of hydrogen does not occur and emergency cooling units (VUC-14A VUC-14C) provide cooling if normal cooling is lost. This equipment is powered from ESF MCCs of the respective channels.

The Non-1E battery D45 is housed in a dedicated room located in a non-seismic area of ANO-1. This room has a dedicated Non-1E HVAC system and exhaust fan.

8.3.2.1.8 Inspection, Servicing, Testing, Installation and Qualification

The station batteries and their associated equipment are easily accessible for inspection, servicing, and testing. Service and testing will be performed on a routine basis in accordance with the manufacturer's recommendations for all station batteries and Technical Specifications for the Class 1E station batteries. Typical inspection includes visual inspection for leaks, corrosion, or other deterioration and checking all batteries for voltage, specific gravity, level of electrolyte, and temperature. As part of the purchase order for batteries D06 and D07, the vendor performed an acceptance test on these batteries, in accordance with IEEE-450, to demonstrate that the batteries are capable of supplying their rated capacity.

The 1E 125-volt DC, system components were purchased and installed under a strict quality assurance program. Certified records of quality assurance inspection and tests performed during production were obtained from the equipment manufacturer. The equipment has been qualified by both tests and satisfactory operation at other operating stations.

The qualification of the 1E 125-volt DC system equipment meets the requirements of IEEE 323-1971.

The Non-1E DC system is provided with metering to accurately measure the charging current when the battery is fully charged. The battery charging current can be used in lieu of the specific gravity parameter to declare the battery operable following a discharge test.

Each battery cell case is marked with a high and low electrolyte level line. The minimum electrolyte level is determined by the low electrolyte level line on the battery case.

8.3.2.2 Analysis

The 125-volt DC Class 1E electrical systems are designed to meet the requirements of IEEE 279-1971, IEEE 308-1971, 10 CFR 50 General Design Criteria 17 and 18, and Regulatory Guide 1.6. They will also meet the requirements for the design basis events described and evaluated in Chapter 14.

8.3.2.2.1 Compliance with Design Criteria and Guides

The following analysis of the Class 1E DC system demonstrates compliance with General Design Criteria 17 and 18, Regulatory Guide 1.6, and IEEE 308-1971.

Criterion 17

The two systems which supply the 125-volt DC power to redundant Class 1E load groups from the two separate 125-volt DC buses are electrically independent and physically separated from each other. Each of the two systems has adequate capacity to supply the 125-volt DC power for the safety-related loads required to safely shut down the reactor.

Criterion 18

As described in Section 8.3.2.1.8, the Class 1E DC system is designed to permit appropriate periodic inspection and testing.

Regulatory Guide 1.6

As described in Section 8.3.2.1.6, the Class 1E DC system is designed with sufficient independence to perform its safety functions assuming a single failure.

IEEE Standard 308-1971

The following presents an analysis per the Supplementary Design Criteria of IEEE 308-1971 as applicable to the Class 1E DC system.

The Class 1E DC system provides DC electric power to the Class 1E DC loads and for control and switching of the Class 1E systems. Physical separation, electrical isolation, and redundancy have been provided to prevent the occurrence of common failure modes. The design of the Class 1E DC system includes the following features.

A. The DC system is separated into two main redundant systems.

B. The safety actions by each group of loads are independent of the safety actions provided by its redundant counterpart.

C. Each DC redundant subsystem includes power supplies that consist of one battery and one battery charger.

D. Redundant batteries cannot be interconnected.

E. The batteries are arranged to prevent a common mode failure.

Each distribution circuit is capable of transmitting sufficient energy to start and operate all required loads in that circuit. Distribution circuits to redundant equipment are independent of each other. The distribution system is monitored to the extent that it is shown to be ready to perform its intended function. The DC auxiliary devices required to operate equipment of a specific AC load group have been supplied from the same load group.

Each battery supply is continuously available during normal operation and following a loss of power from the AC system to start and operate all required loads.

Instrumentation is provided to monitor the status of the battery supply as follows:

A. Battery output voltage indication;

B. Battery output current indication;

C. Battery ground detection annunciation (control room);

D. Battery ground reference DC voltmeters (125 VDC distribution panel);

E. No charge alarm relay indicating alarm (control room);

F. Battery main fuse blown annunciation (control room); and

G. Open battery safety disconnect switch annunciation (control room)

The batteries will be maintained in a fully charged condition and will have sufficient stored energy to operate all necessary circuit breakers and provide an adequate amount of energy for all required emergency loads.

The battery chargers of one redundant subsystem are independent of the battery chargers for the other redundant system. Instrumentation has been provided to monitor the status of the battery charger as follows:

A. Output voltage at charger;

B. Output current at charger;

C. Breaker position indication at charger; and

D. Controls in the charger to control floating voltage and equalizing charge voltage, and acting as a current limiting device.

Each battery charger has an input AC and output DC circuit breaker for isolation of the charger. Each battery charger power supply has been designed to prevent the AC supply from becoming a load on the battery due to a power feedback as the result of the loss of AC power to the chargers.

Equipment of the Class 1E DC system is protected and isolated by fuses or circuit breakers in case of short-circuit or overload conditions. Indications have been provided in the control room to identify equipment that is made unavailable per the following:

	Event	Available Indication
A.	Battery charger AC input breaker trip	Charger malfunction alarm
B.	Battery charger DC output breaker trip	Charger malfunction alarm
C.	DC control center feeder trip	Breaker trip alarm
D.	Distribution circuit breaker trip	Individual equipment alarm
E.	Inverter DC feeder breaker trip	Inverter malfunction alarm
F.	Inverter output AC breaker trip	120 volt AC vital bus undervoltage alarm
G.	Blown battery fuses or open battery disconnect	Battery not available

Dependable power supplies have been provided for the RPS and ESAS. Two independent DC and four independent AC power supplies have been provided for control and instrumentation of these systems. The independent DC supplies have been provided by distribution circuits from each of two redundant DC distribution panels. Independent AC supplies have been provided by the four inverters and associated 120-volt AC vital buses. Refer to Section 8.3.1 for further description of these vital instrument AC power supplies.

Since the inverters are directly fed from the DC bus, the failure of a battery or battery charger that does not disable the bus will not in any way affect the operation of the required AC loads from the inverter. Failures that disable the DC bus result in automatic transfer of the associated inverters to their alternate source.

TABLE 8-1

EMERGENCY DIESEL GENERATOR LOADS

	Total No.	H.P. Each	*Auto-Connected Load With One DG Operating Connected For MHA Diesel A No.	Diesel A H.P.	Diesel B No.	Diesel B H.P.	Starting Sequence	Time from Safety Injection Signal
Load Cntr Transf. Supplying:							1	15
Motor Operated Valves		-	50	127	40	106		
DG Oil Pumps	2	1	1	1	1	1		
Cont. Rm Air Conditioner	1	17	1	17	1	17		
M/U Pp Rm Unit Coolers	3	7.5	2	15	2	15		
M/U Pp Aux. Oil Pumps	3	1/2	2	1	2	1		
EDG Room Exhaust Fans	4	15	2	30	2	30		
Diesel Fuel Transf. Pumps	2	1/2	1	1/2	1	1/2		
NSSS I&C Rod Cont**	-	-	-	46	-	46		
Battery Chgr Emerg Ltg** and Misc.	-	-	-	321	-	444		
Primary Makeup Pumps***	3	700	1	760	1	760	2	20
SPDS Room Cooler**	1	51	1	51	1	51	3	25
Decay Heat Pumps	2	350	1	380	1	380	3	25
Decay Heat Unit Coolers	4	10	1	10	1	10	3	25
Decay Heat Rad Monitors	2	3	1	3	1	3	3	25
Service Water Pumps	3	350	1	350	1	350	4	30
Emergency Feedwater Pump	1	700	1	700	-	-	5	35
Reactor Bldg Spray Pumps	2	250	1	250	1	250	6	50
Reactor Bldg Cooler Fans	4	125	2	250	2	250	7	65
Total				3298		2700		

* For the first 3 minutes of operation, the EDGs' loading capability is limited to a single step load of 2650 KW due to EDG turbocharger loading (addressed in EDG Transient Calculation # 92-E-0003-01). Therefore, specific automatically or manually started loads that are delayed for longer than 3 minutes by certain processes are not included in the table (e.g. Turbine Turning gear and related T.G. oil pumps are not included since these components are not loaded onto the diesel until the turbine coasts down). Per the ESI letter, load may be applied in smaller steps (100 – 200 KW increments) as the load approaches 2650 KW up to 2850 KW.

** kW/kVA rated loads were converted to H.P. via the formula H.P. = kW / 0.7456. Component efficiency was assumed to be 100% for conservatism.

*** The Swing Primary Make-up Pump start time from safety injection signal is 22 seconds.

Note: The above load profile represents an approximation of each components actual operating load on the diesel generators during an accident. The loadings shown for each diesel is based on data from Calculation No. 86E-0002-01. For actual load configuration, see Calculation No. 86E-0002-01.

Table 8-2

VOLTAGE/LOAD STUDY

The following excerpts from the Bus Voltage Study (Calculation No. E-1-800, issued August 1979) define ANO-1 plant conditions with respect to the distribution system voltage profile in study and transient conditions.

Description	Case 1	Case 2	Case 3	Case 4
Reactor Status	C.S.	62.5%	99.2%	
Generator Status	T-G	510 MWe	855	
No. Reactor Coolant Pumps Running	0	4	4	
No. Condensate	0	2	2	
No. Circ. Water	2	4	4	
Other:				

Parameter	Inst	Loc	Units			
161 kV Grid Volts				167	169	168
500 kV Grid Volts				533	528	528
Generator Volts	V/Gen	C01	Vab	__	22.5	22.5
			Vbc	__	22.4	22.5
			Vca	__	22.5	22.5
Amps	A/@A		Aa	__	13.3	22.4
	A/@B		Ab	__	13.5	22.5
	A/@C		Ac	__	13.4	22.0
Freq.	F/Gen		Hz	__	60.02	60.02
Gross MW	WR/G	C11	MW	__	510	855
Net MW	WR/N	C11	MW	__	475	820
Reac. Pwr.	VAR/Gen	C01	VARS	__	+65	+90
Diesel generators not on line						
22kV System						
ST #1 "H" Line	C10	Vab	21.4	21.3	21.25	
		Vbc	21.3	21.1	21.0	
ST #1 "Y" Load	A-113	A113	Aa	300	0	0
			Ab	290	0	0
			Ac	300	0	0
	A-213	A213	Aa	500	0	0
			Ab	500	0	0
			Ac	500	0	0

ST #2 Not Loaded

Table 8-2 (continued)

4160 V System (ES)						
Tie Bkr Load	V-A3	A309	Vab	4300	4200	4190
			Vbc	4300	4190	4190
			Vca	4300	4200	4200
	A-309		Aa	110	90	80
			Ab	110	90	80
			Ac	110	90	80
	V-A4	A409	Vab	4350	4250	4250
			Vbc	4350	4210	4250
			Vca	4350	4250	4250
	A-409		Aa	0	160	160
			Ab	0	160	160
			Ac	0	170	170
Cross-Connect Not Loaded						
Diesel Generators Not on Line						
480 V System						
Non ES Bus Volts	V-B1	B111	Vab	498	482	478
			Vbc	498	478	476
			Vca	500	480	479
	V-B2	B211	Vab	505	489	485
			Vbc	503	487	482
			Vca	505	488	485
	V-B3	B311	Vab	510	488	488
			Vbc	510	484	484
			Vca	510	488	488
	V-B4	B411	Vab	508	490	490
			Vbc	503	488	488
			Vca	508	490	490
	V-B7	B711	Vab	510	496	495
			Vbc	510	492	490
			Vca	510	492	495
ES Bus Volts	V-B5	B511	Vab	503	492	490
			Vbc	500	490	488
			Vca	503	490	490
	V-B6	B611	Vab	503	490	487
			Vbc	503	488	483
			Vac	503	490	485

Table 8-3

INITIAL CONDITIONS FOR VOLTAGE STUDY

1. AUTOTRANSFORMER IS OUT

2. TIE BETWEEN 500 kV AND 161 kV SYSTEMS IS OUT

3. 161 kV SYSTEM IS INTACT

4. FAST TRANSFER TO STARTUP TRANSFORMER 1/STARTUP TRANSFORMER 3 IS NOT SUCCESSFUL

5. MINIMUM VOLTAGE ON 161 kV SYSTEM OF 1 P.U.

6. INTERLOCKS TO KEEP BOTH UNITS FROM ACCESSING STARTUP TRANSFORMER 2 AUTOMATICALLY IN OPERATION

MINIMUM ACCEPTABLE VOLTAGES ON BUSES

SYSTEM	MOTOR BASE	RUNNING LOAD	STARTING LOAD
6.9 kV BUS	6.6 kV	0.91 P.U.	0.82 P.U.
4.16 kV BUS	4.0 kV	0.91 P.U.	0.82 P.U.
480-VOLT BUS	460-VOLT	0.92 P.U.	0.86 P.U.

NOTE: MINIMUM ACCEPTABLE VOLTAGE AT MOTOR TERMINALS

RUNNING -- 90% (MOTOR BASE)

STARTING -- 80% (MOTOR BASE)

Table 8-4

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Table 8-5

THIS TABLE INTENTIONALLY DELETED PER AMENDMENT 6

Table 8-6

THIS TABLE INTENTIONALLY DELETED PER AMENDMENT 6

Table 8-7

BUS VOLTAGES (MOTOR BASE) FOR ANO-1

	6900 V	4160 V	4160 V	480 V	480 V
FULL HOUSE - S.S. (NO ES)	1.032	1.006	1.006	1.009	1.009
ES					
T = 5	1.028	0.977	0.977	0.877	0.919
T = 10	1.028	0.978	0.978	0.917	0.930
T = 15	1.031	1.000	1.000	0.941	0.954
T = 20	1.029	0.9828	0.9828	0.927	0.940
T = S.S.	1.031	1.000	1.000	0.945	0.962

Table 8-8

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Table 8-9

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Table 8-10

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Table 8-11

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Table 8-12

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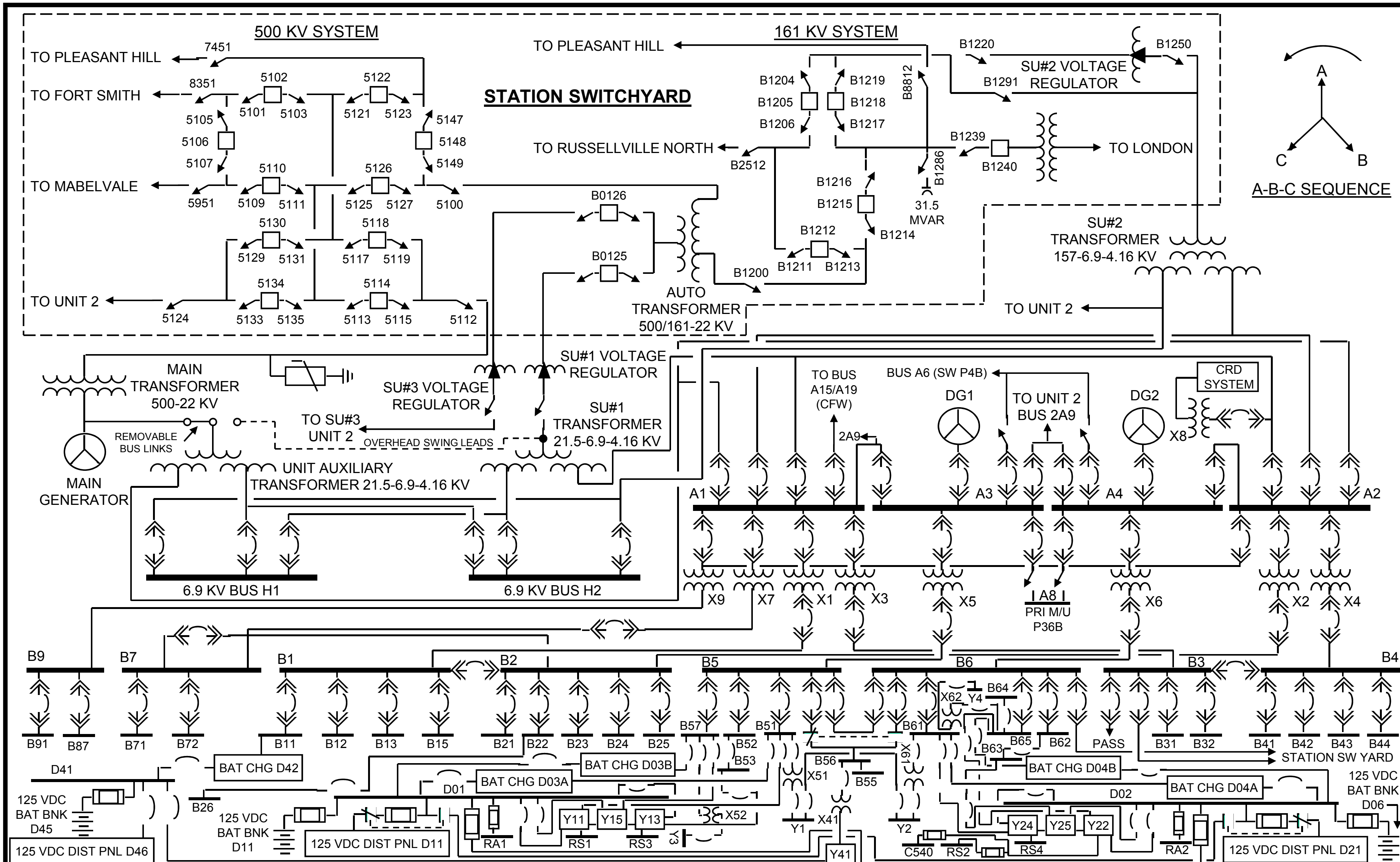


STATION SINGLE LINE DIAGRAM

SAR FIGURE NO. 8-1

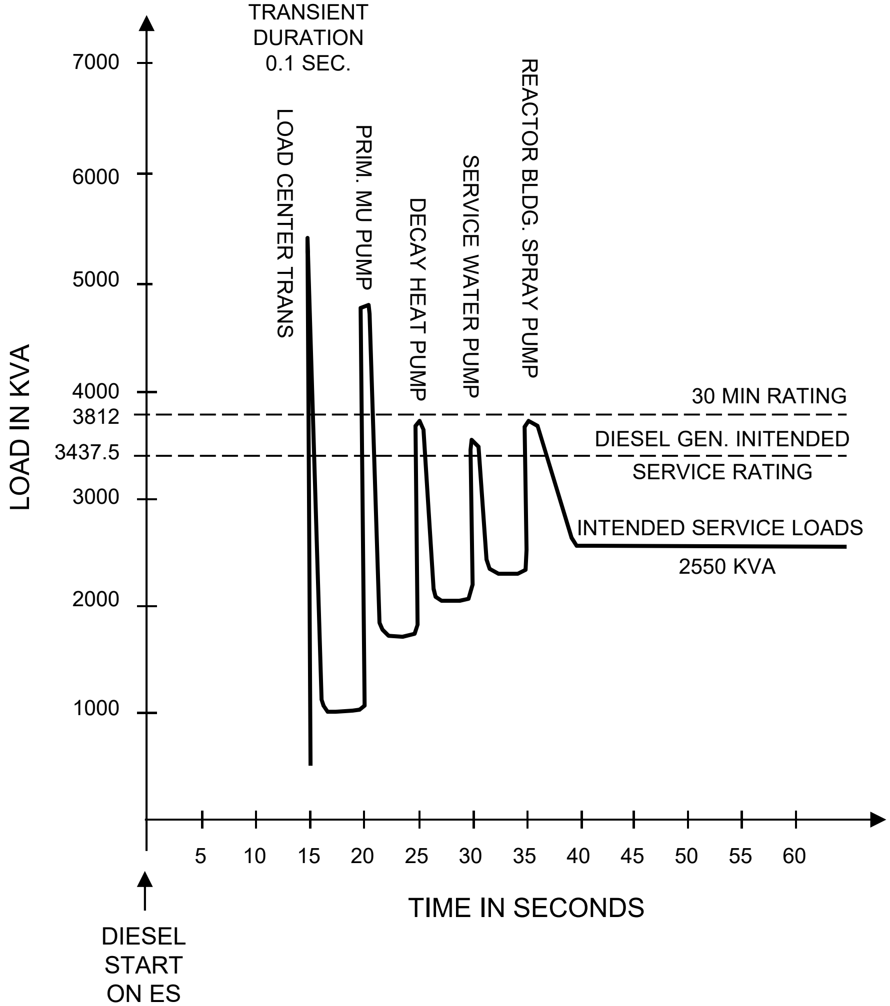
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

AMENDMENT 28

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



DIESEL RATING:

2600 KW FOR 8750h = CONTINUOUS RATING
2750 KW = INTENDED SERVICE
2850 KW FOR 2000h
2860 KW FOR 2h OUT OF 24h
2950 KW FOR 7 DAYS
3000 KW FOR 4h
3050 KW FOR 30 MIN

WK^2 = 23950 $LBFT^2$ – GEN. ROTOR
4495 $LBFT^2$ – ENGINE WHEN TURBOCHARGER ON
2380 $LBFT^2$ – ENGINE WHEN TURBOCHARGER OFF

GEN. RATING:

2750 KW
3437.5 KVA
0.8 PF
X'd = .149 P.U.
Xd = .711 P.U.
X'd = .086 P.U.
SCR = 1.555

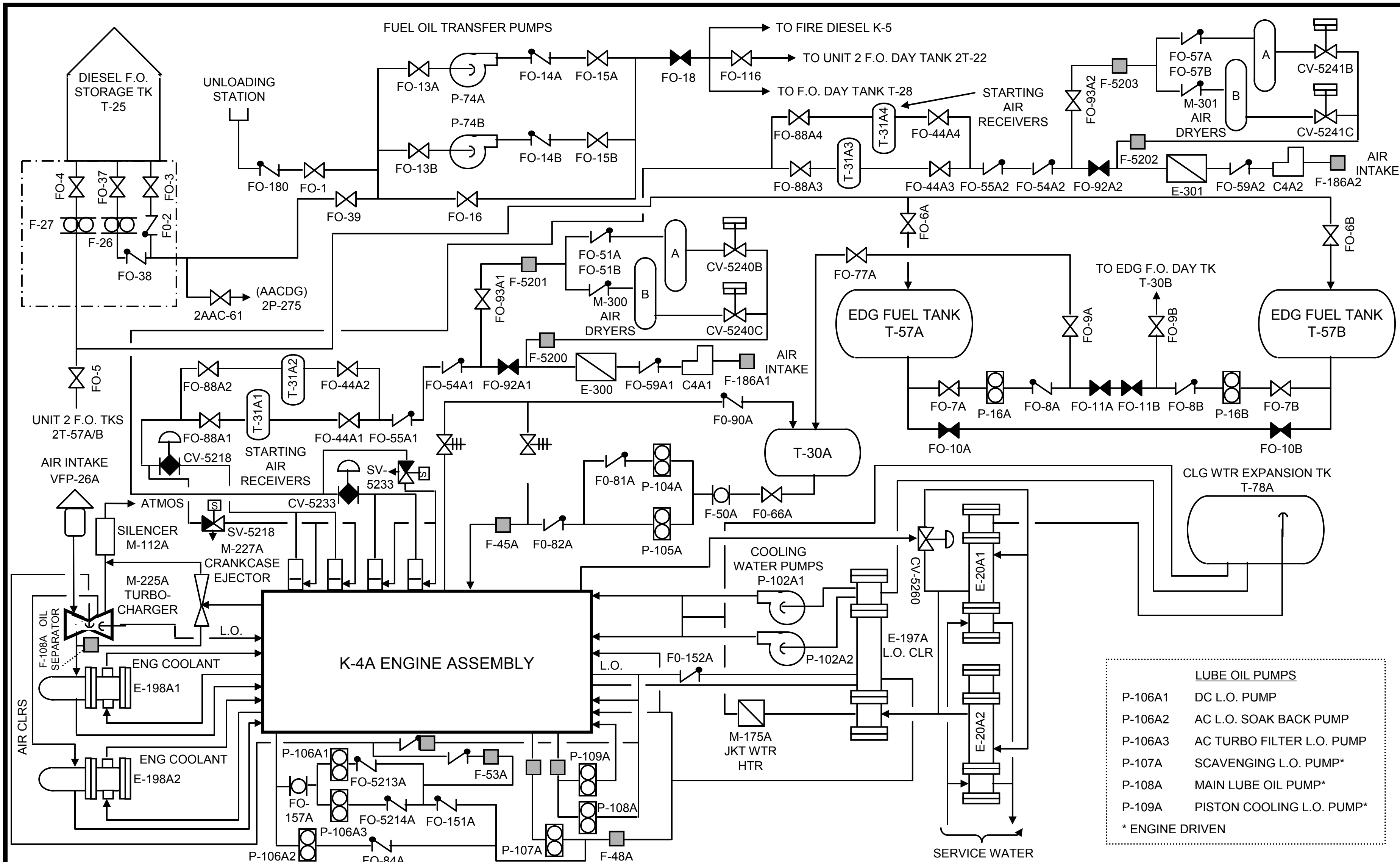
EXCITATION SYSTEM: STATIC VOLTAGE REGULATOR
RESPONSE TIME = 50 msec.

NOTES:

1. THE SWING PRIMARY MAKE-UP PUMP STARTS AT 22 SECONDS
2. THIS LOAD PROFILE REPRESENTS EDG LOADING WHEN ANO-1 BEGAN COMMERCIAL OPERATION
3. THE 1992 LOAD PROFILE IS TO BE BASED ON EDG TRANSIENT LOAD STUDY PROVIDED BY MKW POWER SYSTEMS, INC. (SEE CALC. #92-E-0003-01 FOR CURRENT TRANSIENT RESPONSE)

EMERGENCY DIESEL GENERATOR LOAD PROFILE	SAR FIGURE NO. 8-2
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 29
Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.



EMERGENCY DIESEL GENERATOR K-4A, FUEL OIL STORAGE, AND STARTING AIR

SAR FIGURE NO. 8-3

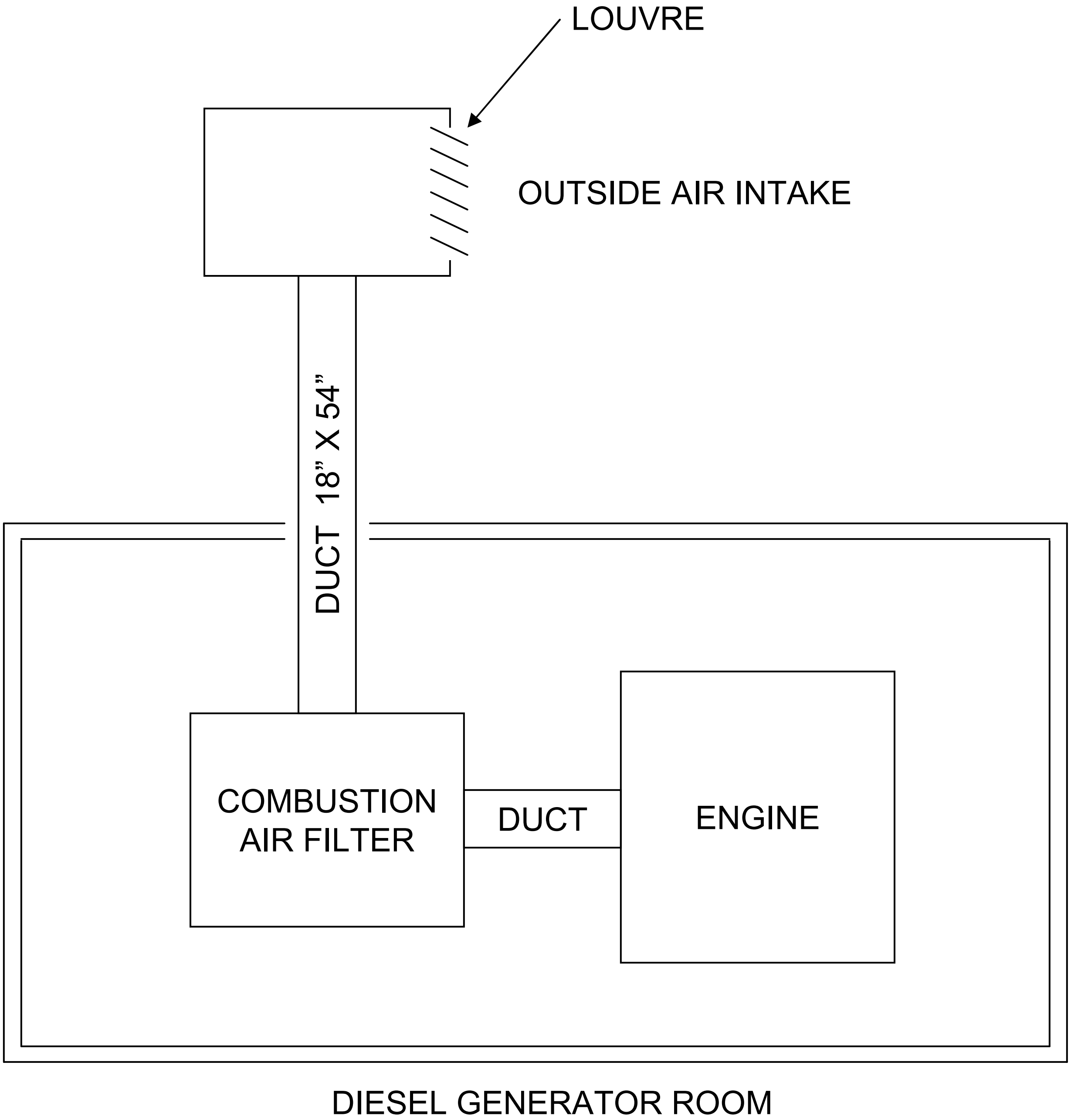
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN: ENTERGY
DESIGN: ENTERGY
CAD NO:

BASED ON DRAWING NO | SHEET | REV.



NOTE: THE COMBUSTION AIR INLET LOUVRE AND THE DUCT ARE PROTECTED FROM PROJECTILES TRAVELING HORIZONTALLY.

SAR FIGURE NO. 8-4

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
EMERGENCY DIESEL GENERATOR COMBUSTION AIR	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

ALL INFORMATION FROM THIS
FIGURE IS CONTAINED IN FIGURE 8-3

SAR FIGURE NO. 8-5

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
EMERGENCY DIESEL GENERATOR LUBE OIL SYSTEM	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

SAR FIGURE NO. 8-6			
AMENDMENT 20			
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
EMERGENCY DIESEL GENERATOR ENGINE COOLING SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

DELETED

SAR FIGURE NO. 8-7

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

EMERGENCY DIESEL GENERATOR FUEL OIL SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

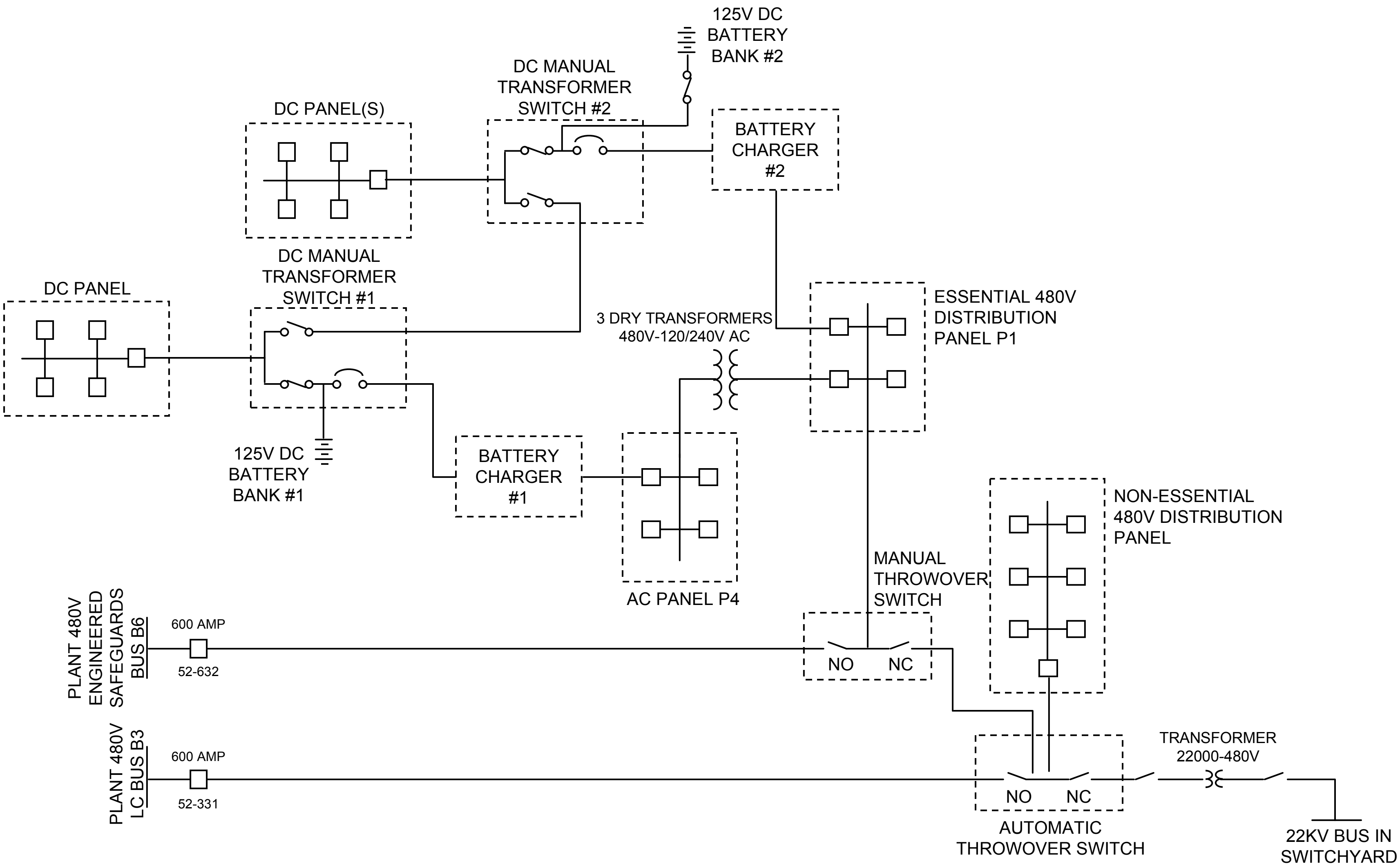
DELETED

SAR FIGURE NO. 8-8

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

	BASED ON DRAWING NO	SHEET	REV.
EMERGENCY DIESEL GENERATOR AIR START SYSTEM		1	



500 KV SWITCHYARD AUXILIARY POWER	SAR FIGURE NO. 8-9
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 22
Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:
BASED ON DRAWING NO	SHEET / REV.

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
9.3.2.4	Correspondence from Phillips, AP&L, to Giambusso, NRC, dated May 11, 1973. (1CAN057303)
9.6.2.3 Table 9-11	Correspondence from Pettit, AP&L, to Ziemann, NRC, dated October 7, 1976. (1CAN107601)
9.6.2.3 Table 9-11	Correspondence from Murphey, AP&L to Ziemann, NRC, dated October 18, 1976. (1CAN107609)
9.6.2.3 Table 9-11	Correspondence from Rueter, AP&L to Ziemann, NRC, dated November 11, 1976. (1CAN117607)
9.8	Correspondence from Williams, AP&L, to Stolz & Reid, NRC, dated June 15, 1978. (0CAN067803)
9.7.2.1	Correspondence from Trimble, AP&L, to Reid, NRC, dated January 10, 1980. (1CAN018008)
9.1.2.1	Design Change Package 206, "Reactor Coolant Pump Protective Instrumentation, Seal Leakage Flow (Drinking Bird)," 1974. (1DCP740206)
9.3.2.4	Design Change Package 307, "Complete Revision of DCR #294 (Same Subject)", 1975. (1DCP750307)
9.6.1.2 9.6.1.3	Design Change Package 478, "Spent Fuel Storage Racks, Installation of New Spent Fuel Storage Racks," 1976. (1DCP760478)
9.7.2.1	Design Change Package 593, "Reliability Intertie Unit 1 & Unit 2 Control Room," 1978. (0DCP780593)
9.2.2	Design Change Package 614, "Adding of Unit 1 to Unit 2 Intertie on the Boric Acid Mixed Tanks," 1978. (0DCP780614)
9.3.2.4	Design Change Package 644, "Emergency Cooling Pond Modifications," 1978. (0DCP780644)
9.8	Design Change Package 655, "Provide Fire Hoses & Station in the Containment [(10) Ten Stations]," 1978. (1DCP780655)
9.8.1	Design Change Package 658, "Fire Detection System Modification," 1979. (1DCP790658)
9.8.2.2	Correspondence from Marshall, AP&L, to Eisenhut, NRC, dated July 1, 1982 (0CAN078202)

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9.8.2.2	Correspondence from Marshall, AP&L, to Stolz/Miller, NRC, dated August 15, 1984. (0CAN088404)
9.8.2.3 9.5.1.2.2	Correspondence from Rueter, AP&L, to Davis, NRC, dated August 30, 1977. (1CAN087710)
9.8.1	Design Change Package 84-1017, "Replacement of H1 Volt Smoke Detectors."
9.1.2.4	Correspondence from Phillips, AP&L, to Zieman, NRC, dated December 3, 1976. (1CAN127601)
9.7	Correspondence from Rueter, AP&L, to Howard, NRC, dated September 20, 1977. (1CAN097710)
9.2.2	Correspondence from Trimble, AP&L, to Eisenhut, NRC, dated March 31, 1980. (1CAN038022)
9.5.2	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated June 10, 1980. (0CAN068004)
9.1	Design Change Package 81-1066, "RCP Controlled Bleed-off Bypass to Sump."
9.3.2.1	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated January 2, 1981. (0CAN018185)
9.1.2.1	Correspondence from Marshall, AP&L, to Seidle, NRC, dated September 29, 1982. (0CAN098219)
9.5.2.4	Design Change Package 82-1067, "Decay Heat Isolation Valve Figure 9-12 Verification for CV-1050 and CV-1410."
9.5.2	Correspondence from Marshall, AP&L, to Seidle, NRC, dated July 29, 1983. (1CAN078315)
9.9	Design Change Package 83-153, (Title Not Available)
9.11	Design Change Package 84-1035, "Emergency Lighting and Communication System."
9.11	Design Change Package 84-1060, "Emergency Lighting Upgrade, Appendix R."
9.9.2.2	Design Change Package 80-1126 A&B, "Service Air/Breathing Air."
9.1.2.1	Design Change Package 81-1066, "RCP Seal Controlled Bleed off By Pass."
9.1	"Spent Fuel Pool Rerack Licensing Submittal," 0CAN118205, Arkansas Power & Light Company - Arkansas Nuclear One - Unit 1 and Unit 2 November 1, 1982.

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Figure 9-5	Design Change Package 82-1060, "PASS Solenoid Valve Replacement."
Figure 9-5	Design Change Package 83-1078, "Replacement of H_2 and O_2 Analyzers in Panel C-119."
Figure 9-5	Design Change Package 85-2104, "Removal of PASS pH/Boron and Chloride Analyzers."

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9 AUXILIARY AND EMERGENCY SYSTEMS

The auxiliary and emergency systems required to support the reactor and Reactor Coolant Systems (RCSs) during normal operation and servicing are described in this section. Some of these systems are also described and discussed in Chapter 6 since they also serve as engineered safeguards. The information in this section deals primarily with the functions served by these systems during normal operation.

The majority of the components in these systems are located within the auxiliary building. Those systems connected by piping between the reactor building and the auxiliary building are equipped with reactor building isolation valves as described in Chapter 5.

The systems considered in this section are:

A. Makeup and Purification System.

B. Chemical Addition and Sampling System.

C. Cooling Water Systems.

D. Spent Fuel Pool Cooling System.

E. Dry Spent Fuel Storage.

F. Decay Heat Removal System.

G. Fuel Handling System.

H. Plant Ventilation System.

I. Fire Protection System.

J. Compressed Air Systems.

The majority of rotating equipment and valves are located outside the reactor building to facilitate maintenance and periodic operational testing and inspection.

9.1 MAKEUP AND PURIFICATION SYSTEM

9.1.1 DESIGN BASES

The Makeup and Purification System is designed to accommodate the following functions during normal reactor operation:

A. Supply the RCS with fill and operational makeup water.

B. Provide seal injection water for the Reactor Coolant Pumps.

C. Provide for purification of the reactor coolant to remove corrosion and fission products.

D. Control the boric acid concentration in the reactor coolant.

E. In conjunction with the pressurizer, accommodate temporary changes in reactor coolant volume due to small temperature changes.

F. Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the RCS.

G. Supply borated water to the Core Flooding Tanks.

The portions of the Makeup and Purification System necessary for safe shutdown are:

A. Makeup pumps.

B. Suction valves from Borated Water Storage Tank (BWST).

C. Injection valves on the discharge side of the makeup pumps.

D. Interconnecting piping for the above equipment.

The Makeup and Purification System serves to control the reactor coolant inventory and the boric acid concentration in the RCS through the processes of letdown and makeup. Purification of the letdown fluid by removal of impurities is also accomplished. Based upon a capacity of 80 gpm for each makeup filter, the maximum letdown flow rate permitted at full reactor pressure is 160 gpm. The normal letdown flow rate of 45 gpm, provided by the letdown orifice, permits recirculation of one RCS volume per day through the purification demineralizers and the makeup filters. Each letdown cooler, makeup filter, and demineralizer is sized for at least half of the maximum letdown flow rate. In addition to the normal purification scheme, a prefilter with a capacity of 140 gpm can be placed in service upstream of the purification demineralizer. The letdown and makeup processes also accommodate thermal expansion and contraction of the reactor coolant water during startup and shutdown transients.

9.1.2 SYSTEM DESCRIPTION AND EVALUATION

The Makeup and Purification System is shown schematically on Figure 9-3. Tables 9-1 and 9-2 list the system performance requirements and data for individual components. The following is a brief functional description of system components.

<u>Letdown Coolers</u>

The letdown coolers reduce the temperature of the letdown flow from the RCS to a temperature suitable for demineralization and injection to the Reactor Coolant Pump seals. Heat in the letdown coolers is rejected to the Intermediate Cooling Water System.

<u>Letdown Flow Control</u>

The letdown flow rate at reactor operating pressures can be controlled by a fixed block orifice or a parallel remotely operated valve can be opened to obtain flowrates up to the maximum letdown capability to accommodate more rapid reactor coolant boron concentration adjustment and increase the purification flow rate. This valve is also used to maintain the desired letdown rate at reduced reactor coolant pressures, i.e. during startup and shutdown. In addition, there is a second valve in parallel with the orifice which is normally closed and which may be manually positioned for flow control.

The letdown orifice is sized to allow 45 gpm flow at the normal operating pressure of the Reactor Coolant and Makeup Systems. The parallel remotely operated valve is capable of increasing the letdown flow up to 170 gpm over that passing through the orifice. Increasing the letdown flow up to the 160 gpm maximum flow may be required when the RCS pressure is less than normal. Under this condition, the reduced flow through the letdown orifice may be insufficient to meet the requirements for purification, feed and bleed, or reactor coolant expansion during heatup. At these reduced pressures the remotely operated valve is used to increase the flow.

The manual valve is a backup for either the letdown orifice or the control valve. This valve, when fully open, passes as much flow as the control valve. The manual valve is opened only if the control valve is shut and secured. Thus, the maximum flow is 170 gpm through the control valve or the manual valve and 45 gpm through the orifice. This yields a total of 215 gpm. The relief valve downstream of the letdown orifice is set for 150 psig and can pass 257 gpm at 10 percent accumulation. Even in the very unlikely situation that all three paths are open simultaneously (which would require multiple operator error), the flow capacity of the Makeup System combined with the relief valve prevents overpressurizing the letdown line.

<u>Letdown Flow Radiation Monitor</u>

Upon leaving the letdown coolers, the letdown flow is continuously monitored for gross and iodine gamma activity. The monitoring is accomplished by a gamma sensitive scintillation detector.

<u>Makeup Prefilter</u>

The makeup flow (or part of the Decay Heat Removal flow during shutdown) can be diverted through a prefilter prior to normal demineralization and filtration in the event extra filtering is deemed necessary.

RCS transient changes in pressure, temperature, pH, and flow during startup, cooldown, and refueling cause the release of crud within the RCS. In order to keep crud buildup in the demineralizers at a minimum, the makeup prefilter can be put into operation prior to scheduled system transients.

The makeup prefilter normally is not used during steady state operation. Reactor coolant water quality specifications are listed in Table 9-4. When additional filtration is desired to meet these specifications, the makeup prefilter can be used.

<u>Purification Demineralizers</u>

The mixed-bed demineralizers are boric acid saturated and are used to remove reactor coolant impurities. Since the reactor coolant may be contaminated with fission and corrosion products, the resins remove certain radioactive impurities. Chapter 11 describes coolant activities, coolant handling and storage, and expected limits on activity discharge. The Purification System is capable of processing up to the maximum letdown flow rate with no bypass required.

Three types of impurities exist in the reactor coolant and the water which flows through the Purification System: (1) corrosion products, either soluble or insoluble; (2) fission products; and (3) other normal ionic impurities. To cope with these impurities, the Purification System incorporates one prefilter, two demineralizers, and two post-filters. The primary purpose of the

prefilter is to remove corrosion products during transient conditions when "crud bursts" occur. However, the filter can also be used during normal steady-state situations. The demineralizers remove ionic impurities but have some filtering capability for suspended materials. Each demineralizer contains a mixed bed of cation and anion resins. One unit is normally operating in the Lithium-7 borate form while the second unit is normally in standby in the hydrogen-borate form. Although both forms of resin are capable of removing soluble impurities of the three types listed above, the hydrogen-form cation resin has better removal ability than the Lithium-7 form for some positive ion fission products, particularly cesium, molybdenum, and yttrium. The hydrogen form resin in the standby unit is used to remove these three fission products when the need arises. Operating experience has demonstrated that mixed bed demineralizers have higher actual decontamination factors than those assumed in Chapter 11. The post-filters are used to remove suspended material that passes through the demineralizer and to prevent resin fines from entering the remainder of the Purification System and the RCS.

Makeup Filters

Two makeup filters of the disposable cartridge type are installed in parallel to remove particulates from the effluent of the purification demineralizers. This prevents solids from entering the Reactor Coolant Pump seals and the Primary Coolant System.

Makeup Pumps

The makeup pumps are designed to return the purified letdown fluid to the RCS and to supply the seal water to the Reactor Coolant Pumps. One pump can provide normal operating makeup and seal water flow.

Reactor Coolant Pump Seal Return Coolers

The seal return coolers are sized to remove the heat added by the makeup and purification pumps and the heat picked up in passage through the Reactor Coolant Pump seals. Heat from these coolers is rejected to the Intermediate Cooling Water System. Two coolers are provided and one is normally in operation.

Makeup Tank

The makeup tank serves as a receiver for letdown, seal return, chemical addition, and system makeup. The tank also accommodates temporary changes in system coolant volume. The volume of the tank is such that the useful tank volume is equal to the maximum expected expansion and contraction of the RCS during power transients.

9.1.2.1 Mode of Operation

During normal operation of the RCS, one makeup pump continuously supplies high pressure water from the makeup tank to the seals of each of the reactor coolant pumps and to a makeup line to one of the reactor inlet lines. Makeup flow to the RCS is regulated by the makeup control valve which operates on signals from the pressurizer level controller.

Seal injection flow is automatically controlled to the desired rate. A portion of the water supplied to the pump seals leaks off as controlled bleed-off and returns to the makeup tank after passing through one of the two Reactor Coolant Pump seal return coolers. The remainder is injected into the RCS.

Upon initiation of a containment isolation signal to terminate RCP controlled bleed off, an alternate flow path to the quench tank will be automatically maintained by the opening of solenoid valves SV-1273, SV-1272, SV-1271, and SV-1270, thus ensuring continuation of controlled bleedoff from the RCP seals. Manual operation of the valves can be accomplished via individual hand switches in the control room.

Seal water inleakage to the RCS makes necessary a continuous letdown flow of reactor coolant. Letdown flow is also required for removal of impurities and boric acid from the reactor coolant. The letdown flow is cooled by the letdown coolers, reduced in pressure by the letdown orifice, and then passed through a prefilter, if necessary, and a purification demineralizer to a 3-way valve which directs the coolant to the makeup tank or to the Waste Disposal System.

Normally, the 3-way valve is positioned to direct the letdown flow to the makeup tank. If the boric acid concentration in the reactor coolant is to be reduced, the 3-way valve is positioned to divert the letdown flow to the Waste Disposal System and boric acid is removed by the feed and bleed method. Feed and bleed is the process of directing the letdown flow to the Waste Disposal System and maintaining the level in the makeup tank by the addition of demineralized water pumped from a supply of unborated water. The flow of demineralized water is measured and totaled by an inline flow integrator and associated instrumentation. The flow of demineralized water to the makeup tank is controlled remotely by the makeup control valve. During normal operation the flow integrator (batch controller), the Control Rod Drive interlock, or the operator terminates the dilution.

The makeup tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the tank to ensure that a predetermined amount of dissolved hydrogen remains in the reactor coolant. Chemicals in solution are injected into the letdown flow upstream of the letdown filters and then pass into the makeup tank, which serves as a final mixing location. The letdown flow can also be diverted to the vacuum degasifier for the purpose of dissolved gas removal. After being degassified, the coolant can be returned to the makeup tank or diverted to the Clean Waste System.

System control is accomplished remotely from the control room with the exception of periodic switching of the Reactor Coolant Pump seal return coolers. The letdown flow rate is set for flow rates other than normal by remotely positioning the letdown flow control valve to pass the desired flow rate.

The letdown flow rate can be determined by various indications including a console mounted flow indicator provided with a calibrated range of 0 to 160 gpm. The plant computer also monitors this flow and indicates the rate to the operator.

The operator can divert flow to the Waste Disposal System as required by the operating procedures during the following situations:

Heatup - the expansion volume change is diverted to the Waste Disposal System. Control of the diverted flow is based on the pressurizer and makeup tank levels.

Boration or Dilution - changes in boron concentration are controlled by the batch controller which is set by the operator based on operating procedures and control rod position.

The letdown flow rate is essentially controlled by the RCS pressure when the pressure is near normal. During shutdown, startup, or other abnormal RCS conditions, the operator controls the flow using the letdown orifice bypass valve. The flow rate is determined by the operator based on factors such as makeup flow rate at the given reactor coolant pressure, bleed rate required to compensate for reactor coolant expansion, and the pump seal leakage.

The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. The letdown flow to the Waste Disposal System is diverted by remote positioning of the 3-way valve and the valves in the Waste Disposal System. The control valve in the injection line to the Reactor Coolant Pump seals is automatically controlled to maintain the desired flow rate to the seals. The reactor coolant volume control valve is automatically controlled by the pressurizer level controller.

During accident conditions, only the High Pressure Injection pumps in the Makeup and Purification System are required to function as described in Sections 6.1.2.1.1 and 6.1.3.1. These pumps are fed by gravity from the BWST and from the Low Pressure Injection (LPI) pumps in the Decay Heat Removal (DHR) system. The Emergency Core Cooling System (ECCS) does not rely on any Makeup and Purification System function except the operation of the isolation valves described in Section 6.1.2.1.1. Emergency operating procedures prescribe the manual isolation of letdown and diversion of seal return flows to the quench tank (if not previously performed automatically) whenever conditions that create the potential for core damage exist.

In the event of an extended loss of offsite power or an OTSG tube rupture when Reactor Coolant Pumps are not available to provide pressurizer spray, the makeup pumps will provide a source of water to the auxiliary pressurizer spray system.

9.1.2.2 <u>Reliability Considerations</u>

This system provides essential functions for the normal operation of the unit. Redundant components and alternate flow paths have been provided to improve system reliability.

In addition to the letdown orifice, the system has two full-capacity control valves in parallel with the orifice. One of these control valves is manually operated and one is remotely operated.

The unit has three makeup pumps, each capable of supplying the required Reactor Coolant Pump seal and makeup flow. One is normally in operation.

The part of the system associated with the ECCS (the High Pressure Injection System) is tested in accordance with Section 3.5.2 of the Technical Specification. The Makeup and Purification System is in continuous operation during plant operation. The makeup pumps are alternated so that the operability of the entire system can be demonstrated in the normal course of plant operation.

The Makeup System is supplied with equipment designed to accommodate letdown flow up to 160 gpm. Equipment consists of redundant letdown coolers, purification demineralizers, and purification filters and one 140 gpm prefilter. When letdown flow is less than 80 gpm, one train of the redundant equipment is sufficient to perform the required duty.

9.1.2.3 <u>Codes and Standards</u>

Each component of this system is designed, fabricated, and inspected to the code or standard, as applicable, noted in Tables 9-2 and 6-2.

The part of the Makeup and Purification System within the Reactor Coolant Pressure Boundary receives inservice inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. The remainder of the Makeup and Purification System is inspected in accordance with the manufacturers' recommendations for specific components and the plant preventative maintenance program.

9.1.2.4 System Isolation

The letdown line and the Reactor Coolant Pump controlled bleedoff line are outflow lines which penetrate the reactor building. These lines contain electric motor-operated isolation valves inside and outside the reactor building which are automatically closed by an engineered safeguards signal. The injection line to the Reactor Coolant Pump seals is an inflow line penetrating the reactor building. This line contains stop-check valves inside the reactor building and a remotely operated valve on the outside of the reactor building. The makeup line joins one of the High Pressure Injection lines outside the reactor building. Makeup line isolation is provided by the injection line check valve in the reactor building and a motor operated valve in the makeup line outside the reactor building. Check valves in the discharge of each makeup pump provide further backup for reactor building isolation. The emergency coolant injection lines are used for injecting coolant to the reactor vessel after a LOCA. After use of the lines for emergency injection is discontinued, the electric motor-operated isolation valves in each line outside the reactor building may be closed remotely by the control room operators for isolation. At cold conditions, the controls to the motor operators of the HPI injection valves will be disabled to preclude the possibility of High Pressure Injection into the RCS at cold conditions and causing the system to go solid (1CAN127601). Use of the HPI valves is allowed as a RCS inventory makeup method during loss of DHR events. The Technical Specifications define the precise requirements for disabling the motor operators.

9.1.2.5 Leakage Considerations

Design and installation of the components and piping in the Makeup and Purification System considers radioactive service. Except where flanged connections are installed for ease of maintenance, the system is of all-welded construction.

The effects of failures and malfunctions in the Makeup and Purification System concurrent with a LOCA are presented in Chapter 6.

Section 9.1.2.4 describes system isolation. These analyses show that redundant safety features are provided where required.

For pipe failures in the Makeup and Purification System, the consequences depend upon the location of the rupture. If the rupture occurs between the reactor coolant loop and the first isolation valve or check valve, it would lead to an uncontrolled loss of coolant from the RCS. This LOCA is included in Chapter 14. If the rupture occurs beyond the first isolation valve (check valves for HPI lines) or outside the reactor building, the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve.

The piping in the Makeup and Purification System between the letdown orifice and the makeup tank is classified Seismic Category 2. If there is a failure in the Category 2 portion of the system, there would be no adverse affect on the Makeup and High Pressure Injection (HPI) portions of the system. The flow from the RCS and the break would be limited by the letdown orifice to approximately 45 gpm. The normally operating makeup pump can continue taking suction from the makeup tank until the BWST valves are remote manually opened. The Makeup System can continue to control the pressurizer level until the letdown line is manually isolated. The makeup valve would then shut as no further makeup would be needed. Because of limited inleakage to the RCS, there is ample time to take action before the pressurizer level exceeds the operating range and the plant can be shutdown using the BWST to supply boration.

If a break occurs in the Makeup and Purification System concurrently with a break in the RCS, the HPI system automatically initiates core injection with the BWST valves opening and the letdown line being isolated. No loss of HPI function can occur due to a failure of the Category 2 seismic portions of the Purification System.

A single failure cannot prevent boration when desired for reactivity control since several alternate paths are available for adding boron to the RCS. These are: (a) through the normal makeup line, (b) through the Reactor Coolant Pump seals, and (c) through the HPI connections. If a pump suction is unavailable from the makeup tank, a source of borated water is available from the Borated Water Storage Tank to one or more makeup pumps.

Boron is normally added to the reactor coolant through the makeup line and the Reactor Coolant Pump seals.

Should a failure of the makeup line prevent its use, any one of the four HPI lines can be opened to provide the required flow. Each HPI line has the capability of up to 250 gpm with one makeup pump operating and greater flow at reduced reactor coolant pressures. Each HPI line has throttling capabilities to control flow as required. There is no limiting condition for boron addition more restrictive than the normal means of makeup.

9.1.2.6 <u>Operational Limits</u>

Alarms or interlocks are provided to limit variables or conditions of operation that could cause system malfunctions. The variables or conditions of operation that are limited are as follows:

A. Makeup Tank Level

High and low water level in the makeup tank are alarmed. Low-low water level is interlocked to the 3-way bleed valve and switches the 3-way valve from the bleed position to its normal position.

B. Letdown Line Temperature

A high letdown temperature in the letdown line downstream of the letdown coolers is alarmed and interlocked to close the letdown isolation valve located outside the building, thus protecting the purification demineralizer resins.

The high temperature alarm and isolation valve interlocks on the letdown line are provided to protect the purification demineralizers. Excessive temperature would cause damage to the resins. The alarmed temperature is not indicative of the system pressure but, rather, is indicative of improper letdown cooler operation. The Makeup System pressure is maintained by the letdown orifice and the associate parallel valves. The high temperature interlock is backed up by the high temperature alarm, a temperature indicator, and a computer record. Failure of the interlock combined with failure of the operator to respond to the alarm would result in overheating the purification resins and possibly flow blockage through the demineralizer. The relief valve downstream of the letdown orifice is set to open at 150 psig and is sized to allow 257 gpm flow at 10 percent accumulation, thus preventing overpressure of the Makeup System. Bypass around the demineralizers or manual isolation of the letdown line can then be accomplished.

C. Dilution Control

The dilution cycle is initiated by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

1. The dilution valves are interlocked so that the operator must preset the desired dilution batch size before initiating the dilution cycle. The dilution cycle automatically terminates when the dilution flow has integrated to the preset batch size.

2. Interlocks on the regulating control rod bank automatically terminate the dilution cycle if the regulating rod group (Group 6) is inserted into the core to the 75 percent withdrawn position.

3. The operator may manually terminate the dilution cycle at any time.

4. Continuous dilution of the moderator can be performed only with the control rods withdrawn to preset bounds to insure that there is adequate negative reactivity available to render the core subcritical in the event of inadvertent moderator over dilution. When these conditions are met, the operator can place the 3-way valve in the "bleed" position and, simultaneously, add dilution water to the makeup tanks. If the above conditions are not met, the 3-way valve cannot be placed in the "bleed" position when adding dilution water.

9.2 CHEMICAL ADDITION AND SAMPLING SYSTEMS

9.2.1 DESIGN BASES

Chemical addition and sampling operations are required to change and monitor the concentration of various chemicals in the Reactor Coolant System (RCS) and auxiliary systems. The Chemical Addition System is designed to add boric acid to the RCS for reactivity control, lithium hydroxide for pH control, hydrazine for oxygen control, and zinc for lowering radiation source terms. The system also provides boric acid for other plant components and is sized to be able to add sufficient boric acid to maintain the core one percent $\Delta k/k$ subcritical at any time during life. The Sampling System is designed to sample reactor coolant, steam generator effluent, and liquids and gases from the auxiliary systems. The Waste Gas Analyzer System (WGAS) monitors the gases for free oxygen and hydrogen.

9.2.2 SYSTEM DESCRIPTION AND EVALUATION

The Chemical Addition and Sampling Systems are shown schematically on Figures 9-4 and 9-5. The Sampling System has separate sampling stations for reactor coolant and steam generator sampling. The Chemical Addition System permits chemical addition to the RCS and reactor auxiliary systems during normal reactor operation.

The Chemical Addition and Sampling Systems are required to function as identified in the Technical Specifications. The steam generator feedwater quality is maintained within the limits given in the EPRI Secondary Water Chemistry Guidelines, and the reactor coolant quality is maintained by this system within the limits given in Table 9-4. A brief functional description of the major systems components follows.

Boric Acid Addition Tank

The boric acid addition tank is provided as a source of concentrated boric acid solution. The volume of the tank provides sufficient boric acid solution to increase the RCS boron concentration to that required for cold shutdown with no xenon. Tank trace heating and electrically heat traced transfer lines maintain the fluid temperature above that required to assure solubility of the boric acid.

Boric Acid Mix Tank

The boric acid mix tank is used to mix and supply boric acid for the boric acid addition tank. The tank is equipped with an agitator for mixing and electrical heaters to maintain the temperature for boric acid solubility.

A cross-connection between the Unit 1 boric acid mix tank (T-7) and the Unit 2 boric acid batching tank (2T-7) exists. This interconnection allows use of both tanks on either system and also prevents delays which might occur due to limited boric acid batching capacities.

Boric Acid Pumps

Two boric acid pumps are provided to transfer the concentrated boric acid solution from the boric acid addition tank to the Borated Water Storage Tank, makeup tank, the spent fuel storage pool, or to the Core Flood Tank (Figure 9-4).

Lithium Hydroxide Mix Tank

Lithium hydroxide is mixed in this tank and added to the Makeup and Purification System for pH control of the reactor coolant. The tank is equipped with an agitator for mixing.

Lithium Hydroxide Pump

The lithium hydroxide pump transfers lithium hydroxide from the lithium hydroxide mix tank to the letdown line upstream of the makeup filters. If this pump is unavailable, the hydrazine pump can be used to transfer lithium hydroxide.

Hydrazine Drum

A drum supplies hydrazine to the RCS; hydrazine is used to scavenge dissolved oxygen.

Hydrazine Pump

The hydrazine pump transfers hydrazine from the hydrazine drum to the letdown line upstream of the makeup filters. If this pump is unavailable, the lithium hydroxide pump can be used to transfer hydrazine.

Zinc Mixing Tank

The zinc is mixed in this tank before being injected into the RCS. The tank is equipped with a recirculation pump for mixing.

Zinc Injection Metering Pumps

Two 100% zinc injection metering pumps are provided to transfer the zinc acetate from the zinc mixing tank to the RCS through the lithium hydroxide pump discharge line.

Pressurizer Sample Cooler

This cooler cools the samples taken from the pressurizer steam or water space.

Steam Generator Sample Cooler

This cooler cools the sample taken from the secondary side of either steam generator.

Note: NRC letter (0CNA080005) dated 8/17/2000, eliminated the requirement for PASS from the ANO Units 1 and 2 Technical Specifications (TSs). ER-ANO-2003-0221-000 isolated PASS components by maintaining closed PASS boundary valves and de-energizing Unit 1 liquid solenoid valves SV-1440 and SV-1443. The PASS is no longer used and will be isolated from the plant, but remains part of the plant configuration management process. Therefore, PASS components still appear in LBDs. The design bases of PASS have not changed. Should the PASS be put back into service, the PASS would perform its design function as long as EC 18779 is uninstalled and Reg. Guide 1.97 compliant PASS building ventilation monitoring is restored. SV-1440 and SV-1443 are isolated with permanent caps.

Post-Accident Sampling System (PASS-0CAN038013)

The PASS building and equipment was constructed to permit prompt, safe, remote sampling of primary systems and containment air during accident (and in some cases normal) conditions. Shielded sample lines and remotely controlled analyzers, MOVs, and solenoid valves are provided to minimize dose and reduce the time required for sampling. Samples are available from the following locations: letdown, hotleg, containment sump, and containment air. Pressurizer gas and liquid samples can be aligned to the post accident sampling system, however, the samples are not aligned due to code compliance issues regarding associated PASS sample valves. Sample lines for containment sump and containment air space have been permanently capped.

Sample lines leading to (and in some cases from) PASS analyzers and equipment have been insulated and in the case of containment air, heat traced, in an effort to reduce temperature depression. Liquid samples remain at or near system pressure and temperature until exiting the area of the radionuclide detector. These liquid samples are then cooled (not depressurized) and pass through an inline hydrogen detector. After exiting the hydrogen detector, the samples are depressurized and flow is then diverted to the desired destination (i.e. containment building during accident conditions).

During design and construction, special consideration was given to radii and bends in sample lines to reduce the potential for crud traps and line losses.

Several shielded, portable, grab sample vessels have also been provided in the event that it becomes necessary for analytical tests to be performed offsite. The capability exists to obtain grab samples of either containment air or reactor coolant liquid samples.

Flushing water is provided for liquid sample lines to reduce radiation levels in the analyzer rooms. Purge air is available to reducing radiation levels in containment air sample lines. Therefore, should it become necessary to perform maintenance, the radiation levels can be reduced significantly.

The PASS facility is equipped with a separate, packaged air conditioning system which includes a dedicated exhaust system. Air exiting the building passes through filters to remove particulates and radioactive iodines. Radioactive material that may penetrate these filters is monitored prior to exiting the PASS building stack.

Waste Gas Analyzer System (WGAS)

The WGAS, which consists of two redundant instrument panels, provides the capability to monitor the tanks and equipment in the Waste Gas System for free oxygen and hydrogen. The monitoring capability is provided by either panel.

9.2.2.1 Mode of Operation

The Chemical Addition System delivers the necessary chemicals to other systems as required. Boric acid is supplied to the spent fuel pool, the Core Flood Tank, Borated Water Storage Tank (BWST), and makeup tank as makeup for leakage or to change the concentration of boric acid in the associated systems.

The Sampling System is used to take samples to assure that desired water qualities, gas concentrations, and boric acid concentration are maintained. Sampling locations and types of samples taken at each location are shown in Table 9-5.

9.2.2.2 Reliability Considerations

The Chemical Addition and Sampling Systems are required to function as identified in the Technical Specifications. Redundant boric acid pumps and independent boric acid addition lines are provided to guard against a single component failure rendering the system inadequate for boron addition. Boric acid is available for boration from the boric acid addition tank. To prevent precipitation, heat tracing is installed on components and lines used to transfer concentrated boric acid. The pumps, tanks, coolers, and instrumentation are located in the auxiliary building and are accessible for inspection and maintenance. All components of the Chemical Addition System are available for inspection at all times. Detection and control of leakage is to be visual and manual with minimum possibility existing for these systems to becoming contaminated radiologically.

There are redundant electric power sources, heating elements, thermostats, and switches for heat tracing in the Chemical Addition System.

Sample line locations are selected to insure that a representative sample is withdrawn. The sample lines have provisions for circulation to the Makeup System to allow complete purging of the Sampling System. The procedures for obtaining a sample state the time required for purging. The sample lines are small to insure a high velocity. The transit time from sampling location to sample bottle are short to preclude loss of the sample constituents. Pressure and temperature compensations are computed for the samples to relate the concentrations to the pressure and temperature conditions at the sampling source.

The above design features and operating procedures insure the samples taken are representative.

All sample lines are shielded under the same criteria as the system from which they sample. Sampling of the Makeup System letdown line is accomplished downstream of the delay pipe in the letdown line to minimize the ^{16}N contribution to plant personnel radiation exposure. All other primary coolant sample lines are located at points in the system where the ^{16}N contribution is expected to be insignificant. Provisions, i.e. protective clothing, face masks, etc., are made at the sample stations to minimize hazard to the safety of plant personnel due to radiological or chemical contamination during handling of samples.

The Combustible Gas Control System contains the capability to sample the reactor building atmosphere for hydrogen concentration during accident conditions.

9.2.2.3 Codes and Standards

Sample lines connected to Seismic Category 1 systems with a pressure rating of greater than 600 psi are designed as Seismic Category 1 up to and including the second isolation valve. Sample lines connected to Seismic Category 1 systems with a 600 psi or lower pressure rating are designed as Seismic Category 1 up to and including the first isolation valve. The sample line up to and including the specified isolation valve is designed to the same standards as the process piping to which they connect or later appropriate ASME Section III Code sections provided that they have been reconciled. The piping downstream of the specified isolation valve

is not Seismic Category 1 and is designed to piping Code B31.1. This arrangement assures that any break in the non-Seismic Category 1 portion of the sample line can be isolated and the integrity of the Seismic Category 1 systems maintained.

One exception to this rule occurred on small bore connections to the secondary side of the steam generator and associated main steam and main feedwater piping. Consistent with guidance found in Regulatory Guide 1.29, a distinction was made that connecting piping 2″ and smaller was not required to be designed to Seismic Category 1 standards. In addition, this piping or tubing was designed to ANSI B31.1.0 and the associated isolation valve was also designed to standards or methods permitted by ANSI B31.1.0.

It should be noted that for some newer designs, piping or tubing downstream of the isolation valve(s) may be designed to the same codes and standards as the upstream piping.

For the latest information in this regard, refer to the current qualification documentation for the components of interest.

9.2.2.4 System Isolation

Isolation of this system from the reactor building is accomplished by signals from the safeguards actuation system and is described in Chapter 5.

9.2.2.5 Leakage Considerations

Leakage of radioactive reactor coolant from this system within the reactor building is collected in the reactor building sump. The sample stations are located under hoods which exhaust to the unit vent to collect any leakage of radioactive gases which might occur during sampling operations.

9.2.2.6 Failure Considerations

To evaluate system safety, the following failures or malfunctions are assumed concurrent with a LOCA and the consequences are analyzed. As a result of this evaluation, it is concluded that proper consideration is given to plant safety in the design of the systems.

The failure analysis of the Sampling System is as follows:

Component	Failure	Comments and Consequences
Pressurizer Sample	Electrically operated sampling valve inside reactor building fails to close	Manual valve outside the reactor building can be closed by sample operator
Steam Generator Sample	Electrically operated sampling valve inside reactor building fails to close	Sample line is not connected directly to Reactor Coolant System, and steam generator therefore provides first barrier
Core Flooding	Electrically operated sampling valve inside reactor building fails to close	Sample line is not connected directly to Reactor Coolant System, and core flooding check valves provide first barrier

Quench Tank Sample	Electrically operated sampling valve inside reactor building fails to close on ES signal	Diaphragm-operated valve outside the reactor building closes
Sample Line From Any of the Preceding Components	Line breaks, inside reactor building downstream of EMO valves	Manual valve outside reactor building is closed by sample operator, except for quench tank in which case remotely operated valve outside reactor building closes on signal from ES system

An analysis of chemical addition malfunction and the effect of the malfunction related to the control chemistry and shutdown margin is summarized below. It is concluded that there are no safety related functions which would be affected adversely by failures in the Chemical Addition System.

The failure analysis of the Chemical Addition (CA) System is as follows:

Component	Failure	Comments and Consequences
1. Boric Acid Mix Tank Mixer	Fails to operate	Utilize concentrated boric acid from BA addition tank or from clean waste receiver tank or BWST
2. Piping from BA Mix Tank to BA Addition Tank	Break in line	Same as above
3. BA Addition Tank and Line to BA Addition Pumps	Break	Reactor coolant contraction taken up by the pressurizer fluid, makeup tank contents, and BWST contents
4. BA Pump	Failure to operate	The parallel pump supplies the required flow. If the parallel pump is inoperable, utilize the boric acid solution from the BWST
5. 1½-inch Stainless Steel Line Connecting the Discharge of the Two Pumps	Break	Same as 3
6. Heat Tracing	Failure to heat tanks or lines	Same as 3

As indicated by the above analysis, makeup or borated water from the BWST may be required during cooldown after a malfunction of the CA system. Typical 18 month cycle RCS boron concentration requirements for shutdown are as follows:

Nominal Full Power Concentration	Hot Standby > 525 °F Requirement	Cold Shutdown Requirement
BOL 1423 ppm	1468 ppm	1632 ppm
EOL 25 ppm	209 ppm	651 ppm

Considering the contraction volume available and the utilization of only the BWST for makeup, the following boron concentrations exist in the Reactor Coolant System:

	Hot Standby > 525 °F (579 °F 532 °F)	Cold Shutdown (579 °F 70 °F)
BOL	1448 ppm	1625 ppm
EOL	92 ppm	561 ppm

Thus, the feed and bleed volumes that are required to further raise the boron concentrations to the one percent hot and cold shutdown margin requirements are as follows:

	Hot Standby > 525 °F	Cold Shutdown
BOL	229 ft^3 (1,713 gal)	103 ft^3 (770 gal)
EOL	516 ft^3 (3,860 gal)	510 ft^3 (3,815 gal)

The four 7,760 ft^3 (58,000 gal) clean waste receiver tanks can accept the required reactor coolant bleed.

With the minimum tripped rod worth, the BWST concentration alone is sufficient to borate without feed and bleed and achieve a shutdown margin in excess of -1 percent $\Delta k/k$.

Other chemicals added by the CA System are hydrazine and lithium hydroxide, which are used to inhibit corrosion in the RCS and are not required for safe shutdown. Zinc is added for radiation reduction and is not required for safe shutdown.

9.2.2.7 Operational Limits

The Chemical Addition and Sampling Systems provide certain chemicals to several systems in proper quantities and concentrations and provide a capability to sample fluids in various systems. The limits that must be maintained on these operations are contained in Technical Specifications.

9.2.2.8 Uncontrolled Releases

In the event of uncontrolled releases due to pipe or tank failure, the following considerations are made relative to the safety of personnel, equipment, and systems.

All tanks and pumps in the Chemical Addition System are provided with manually operated isolation valves. With the exception of the boric acid addition tank and zinc mixing tank, valves downstream of the tanks in the Chemical Addition System are normally closed to minimize releases and reduce the hazard to the safety of plant personnel in the unlikely event of pipe or tank failure. The boric acid mix tank is normally operated on a batch basis and is empty a significant portion of the time.

The Chemical Addition System is not required to function during an emergency condition. Therefore, a failure in this system does not directly affect plant safety. In addition, any adverse effects on safety related systems due to chemical release, i.e. corrosion, are significant only if allowed to continue over a long period of time. Since administrative controls require that any chemical spillage be cleaned up as soon as possible following an accidental release, there can be no effects of a pipe or tank failure on safety related systems.

Personnel are evacuated immediately from areas in which high radiation level (as indicated by area radiation monitors and alarms or portable samples) or chemical spillage exist. Necessary safety operating procedures, e.g. area isolation, protective clothing, etc., are then implemented to eliminate the hazard to the safety of plant personnel.

9.3 COOLING WATER SYSTEMS

9.3.1 DESIGN BASIS

The Cooling Water Systems are arranged in three separate pumping systems.

A. The Service Water System cools some of the nuclear auxiliary components. It consists of two independent but interconnected, full capacity, 100 percent redundant systems to insure continuous heat removal. During an emergency when outside power is lost, the service water pumps are powered from the diesel generators. In the unlikely event of a complete loss of water from the Dardanelle Reservoir as a result of a loss of the dam, water is supplied by gravity flow from the Emergency Cooling Pond through the supply line to the service water compartment in the intake structure. After being circulated through the Service Water System, the cooling water is returned to the Emergency Cooling Pond.

B. The Auxiliary Cooling Water System supplies non-nuclear related cooling water requirements. It is a subsystem of the Service Water System in that it receives its water supply from the service water pumps. The water supply is isolated in the event of an ES actuation.

   An Intermediate Cooling Water System cools some nuclear auxiliary components and some non-nuclear components, using a closed condensate loop on one side and service water on the other side of shell and tube heat exchangers. During an emergency this system is isolated and shut down.

C. The Condenser Circulating Water System provides cool water to the main surface condensers under normal operation. Upon loss of normal electrical power, this system stops functioning.

These systems are sized to ensure adequate heat removal based on highest expected cooling water temperatures, maximum loadings, and leakage allowances. The equipment in these systems is designed to applicable codes and standards tabulated in Chapter 9.

All cooling water systems are designed to prevent any component failure from curtailing emergency plant operation. It is possible to isolate all heat exchangers and pumps on an individual basis.

All systems are monitored and operated from the control room. All cooling water lines penetrating the reactor building have isolation valves mounted on one or both sides of the penetrations. Electrical power requirements for the Service Water System are supplied by any of the redundant power sources described in Chapter 8. The Intermediate Cooling Water System, the Auxiliary Cooling Water System, and the Condenser Circulating Water System are not normally intended to be powered by the diesel generators.

All system components, except the condenser circulating water pipes, are hydrostatically tested to 150 percent of their design pressure prior to unit startup. Subsequent hydrostatic tests associated with replacement/repair to portions of the service water system will be conducted in accordance with the rules for Class III piping per ASME, Section XI, the edition and addendum in effect when the replacements are installed. Electrical components, such as switchgear and starting controls, are tested periodically. Design parameters for system components are listed in Tables 9-7 and 9-8.

9.3.2 SYSTEMS DESCRIPTION AND EVALUATION

9.3.2.1 Service Water and Intermediate Cooling Water Systems

The Service Water (SW) System provides cooling water for the following equipment (Figures 9-6 and 9-7).

Normal operation

General Notes:

a. Normal operation is taken to be operation with the plant at normal operating temperature and pressure supplying electrical power to the grid.

b. Normal operation (typical) is depicted in Figures 9-6 and 9-10; for specific operation refer to OP 1104.029.

c. Emergency operation is taken to be two-loop operation with both redundant trains supplying cooling water to the Engineered Safeguards (ES) equipment.

d. Specific emergency operation is dependent upon operating pump configuration for normal operation.

Specific Notes:

1. SW isolation valves close on ES actuation.
2. SW isolation valves open on ES actuation or when their respective components are actuated by an ES actuation.
3. SW isolation valves open whenever the diesel generator is in operation.
4. Alternate water source supplied by remote-manual valve operation.
5. Service water supplied at all times, but not needed for normal operation.

Equipment (SW Loads)	Normal Operation	Normal Cooldown	Emerg. Operation	Specific Notes
Intermediate Cooling HX	X	X		1
Reactor Bldg. SW Cooling Coils			X	2
DHR Pump Room Coolers		X	X	5
DHR Pump Bearing Coolers		X	X	2
Decay Heat Coolers		X	X	2
Emergency DG Jacket HX			X	2,3
Primary Makeup Pump L.O. Clrs	X	X	X	
Primary Makeup Pump Rm Clrs	X	X	X	
Control Room Emerg. A.C. Unit	X	X	X	
Rx Bldg. Spray Pump L.O. Clrs			X	2
ES Switchgear Room Chillers	X	X	X	
CW Pump Bearing Lub./Cool/Seal	X	X	X	
Water Source to EFW Pumps				4

The intermediate cooling HXs service the following equipment:

Letdown coolers

Reactor Coolant Pump cooling water

Reactor Coolant Pump seal return coolers

Pressurizer sample cooler

Steam generator sample cooler

Spent fuel pool coolers

Waste gas compressor aftercoolers

Isophase bus coolers

Reactor Coolant Pump motor and lube oil coolers

Main feedwater pump lube oil coolers

Control Rod Drive cooling coils

Redundancy is obtained by dividing the Service Water System into two independent circuits arranged such that the failure of any single vital component does not affect the required performance of the remaining system. Revised logic for service water crossover valves (CV-3640, CV-3642, CV-3644, CV-3646) permit ACW to be shared by both SW loops with any combination of SW pumps while maintaining single failure criteria and loop separation during emergency operation.

The service water pumps are sized to provide sufficient flow and head characteristics to meet the requirements of normal operation, shutdown cooling, and DBA cooling requirements. Two service water pumps and the two essential loops provide two systems, each of which can provide 100 percent of the required cooling flow for the above situations. A third pump is provided so that maintenance can be performed on one of the pumps and the above requirements can still be met. The service water pumps are powered from the engineered safeguards buses, therefore operating with or without off-site power. The service water piping is sized such that adequate flow is delivered to all components under all modes of operation. Table 9-15 summarizes the line sizes, flow paths, temperatures, NPSH, and flows for the limiting modes of operation. Each portion of the system is designed to the maximum temperature expected in that portion of the system during worst case accident conditions.

The sluice gates, shown in Figure 9-6A, are motor operated with controls and indication in the control room. The motors and controls are powered from the engineered safeguards buses.

The service water pumps are located in the intake structure which is designed to withstand flood loading.

The motor mount sits on floor elevation 366 feet. Normal pool elevation is 338 feet and flood level is 361 feet.

Failure of any Seismic Category 2 equipment adjacent to any component of the Service Water System does not, in any way, affect the latter's integrity.

This criterion is implemented by physical separation between the Category 1 service water equipment and other Category 2 equipment, shields, or protection around the Category 1 equipment and/or complete elimination of Category 2 equipment from Category 1 service water equipment areas. A specific example of the application of this criterion is the service water pump motors discussed above.

To help limit biological fouling such as flow blockage from bivalve mollusks and Corbicula (Asiatic clams), a biocide is added at the intake structure in sufficient concentration to kill the mollusks. The service water intake bays are inspected and cleaned periodically to prevent clam buildup and fouling. In addition, flow measurement orifices and instrumentation have been added to the service water headers, to the reactor building coolers, and to most of the auxiliary building coolers. Where permitted by cooler fittings design, many of the 1/2-inch and 3/4-inch cooling lines have been replaced with 1-inch lines. Flow measurements are being periodically taken and trended to detect any possible developing flow blockage from biological fouling.

The entire Service Water System is redundant except that there are three pumps, a single supply line from the cooling pond, and a common discharge (return) header which can be aligned to either the lake or the cooling pond. The system is therefore testable in any manner including full design flow on the unused portion of the system. Components of the Service Water System are tested to ensure proper functioning at a frequency specified in the Surveillance Frequency Control Program (SFCP).

Submergence required at the pump inlet bell is 3.46 feet at pump design flow of 6500 gpm and 95 °F water temperature, which is the peak lake temperature. At the low lake elevation of 336 feet, which would be the lake elevation during severe drought, 12.5 feet is available at the suction bell elevation of 323.5 feet. Figure 9-21 shows the original service water pump flow against head and NPSH required.

Submergence required at the pump inlet bell is also 3.46 feet at pump design flow of 6500 gpm and 116 F water temperature, which is the peak ECP temperature. Line losses from the ECP are accounted for in the design calculations related to Service Water pump submergence and NPSH (Net Positive Suction Head).

It should be observed that NPSH required does not consider the manner of pump installation. For a wet pit pump, required submergence is the governing parameter. For the above conditions, adequate NPSH is provided by the submergence available for both cases.

Raw water is circulated to the coolers located throughout the plant. Normally, two pumps are in service on independent power sources. One pump is a standby pump. During an emergency, the auxiliary cooling load is shed and all pumps are available for service water cooling duty.

The performance requirements of the Service Water System during a post-accident situation are described in Chapter 6. Normal operational requirements are listed in Table 9-8.

The Intermediate Cooling Water pumps and heat exchangers are located in the auxiliary building. This closed loop system provides an additional barrier between high pressure reactor coolant and service water to prevent an accidental release of radioactivity to the environment.

The Service Water System and intermediate cooling loops are continuously monitored for radioactivity as described below and in Chapter 11 and can be isolated as required.

Each of the four reactor building cooling units contains two separate and independent cooling coils. One coil is for normal operation and one is for emergency operation. The coils used during normal operation are served by chilled water to remove the reactor building heat load. Service water is circulated through the emergency cooling coils to cool the reactor building atmosphere after a LOCA.

The Service Water System is protected against severe water hammers caused by a loss of offsite power, with or without a coincident LOCA.

The operation of the Service Water and Intermediate Cooling Water Systems is monitored by the following instrumentation:

A. Pressure indicators on pump discharge lines.

B. High differential pressure alarms on basket strainers in pump discharge lines.

C. Pressure indicators and temperature indicators in supply headers to the Service Water System.

D. Radiation monitors and alarms on each set of two reactor building cooling units.

E. Radiation monitors and alarms on each outlet line from each decay heat cooler.

F. Motor bearing high temperature alarm.

G. Remotely operated valve position indication.

H. Pump trip alarms in control room.

I. Hour counters on the pump breakers.

J. Flow instrumentation on the reactor building coolers.

Service water flow to the reactor building service water cooling coils (VCC-2A through 2D) is monitored outside of the reactor building by means of two flow indicators (0CAN018105). One flow indicator monitors service water flow to reactor building cooling coils VCC-2A and 2B and one flow indicator monitors service water flow to cooling coils VCC-2C and 2D. The flow elements associated with each transmitter are flow orifices located inside of the reactor building on the combined header on the discharge side of the reactor building service water cooling coils.

Throttle valves SW-814 and SW-815 can be used to prevent excessive flashing in VCC-2A/2B and VCC-2C/2D, respectively.

<u>Intermediate Cooling Water System</u>

A. Pressure indicators and low pressure alarms on pump discharge lines.

B. Local level indicators on the Intermediate Cooling Water System surge tanks.

C. Pressure indicators on Control Rod Drive pump discharge lines.

D. Temperature indicator and alarm on Control Rod Drive cooling coils supply line.

E. Flow element and temperature indicator and alarm on Control Rod Drive cooling coils outlet line.

F. Flow elements on Reactor Coolant Pumps cooling water heat exchangers.

G. Radiation monitors and low flow alarms upstream of intermediate heat exchangers.

H. Temperature alarms downstream of intermediate heat exchangers.

The reliability considerations for these systems are primarily based on the considerations for post-accident operations which are discussed in Chapter 6.

System Isolation

Service Water System

Each inlet and outlet cooling water line serves two reactor building cooling water coils and has a power operated valve outside the reactor building wall that can be closed manually from the control room. There are no direct connections to any other system or to the reactor building atmosphere.

Intermediate Cooling Water System

All Intermediate Cooling Water System inlet lines which penetrate the reactor building have power operated isolation valves outside and check valves inside the reactor building. All outlet lines which penetrate the reactor building have power operated stop valves inside and power operated valves outside the reactor building. All equipment vent and drain lines are equipped with normally closed, manually operated valves.

Leakage Considerations

Water leakage from piping, valves, and equipment in the Intermediate Cooling Water System inside the reactor building is not generally considered detrimental unless the leakage exceeds the makeup capability. Service Water System leaks within the reactor building are considered a degradation of the system. To reduce any possibility of water leakage both inside and outside the reactor building from piping, valves, and equipment, welded connections are used where possible. Service water could become contaminated with radioactive water due to one of the following:

A. A leak in a letdown cooler tube or in the heat exchanger for the mechanical seal on a Reactor Coolant Pump simultaneously with a leak in the intermediate heat exchanger tubes. Tube or coil leaks in components being cooled by the Intermediate Cooling Water System are detected by radiation monitors located in both main outlet headers upstream of the intermediate heat exchangers. A leak in a letdown cooler can be isolated by closing the motor-operated isolation valves.

B. Leakage of radioactive products into the cooling water from the reactor building atmosphere during post-accident operation of the air coolers. Coil leaks in the reactor building air coolers are detected by radiation monitors located in each of the cooling water outlet headers. A single train of reactor building cooling units (VSF-1A/VCC-2A and VSF-1B/VCC-2B or VSF-1C/VCC-2C and VSF-1D/VCC-2D), which comprises 50% of the emergency cooling capacity, may be isolated if necessary.

C. A tube leak in the DHR coolers, either during normal use for DHR or during post-accident usage, is detected by the radiation monitors located in the cooling water outlet lines. The coolers are then isolated by closing the motor operated valves upstream of the coolers and the manually operated valves downstream of the coolers. The redundant cooler could then be used to continue operations in a limited manner.

Failure Considerations

The consequences of equipment, piping, valve, instrumentation, and other failures in this system are presented in Chapter 6 and Table 9-11. All operating conditions permitted by the Technical Specifications are listed in Table 9-16.

9.3.2.2 Auxiliary Cooling Water System

The Auxiliary Cooling Water System is normally operated with one of the three service water pumps available to pump raw water from the intake structure to heat exchangers located in the auxiliary and turbine buildings. The system flow diagram is shown on Figure 9-9.

All equipment served by this system is expendable in the event of an accident. The system is isolated at the time of an accident. No nuclear or essential equipment is served by the auxiliary cooling system.

9.3.2.3 Condenser Circulating Water System

Raw water from the Dardanelle Reservoir is used as the source of water for the Condenser Circulating Water System. The system flow diagram is shown on Figure 9-10.

The four circulating water pumps are located in the intake structure. They are protected from flotsam and debris by a skimmer, trash rack, and traveling screens, all in tandem. The Condenser Circulating Water System is designed to take advantage of the siphon effect to reduce the otherwise required pump discharge head.

9.3.2.4 Emergency Pond

The emergency pond serves as a heat sink for simultaneously shutting down both Unit 1 and Unit 2 in the unlikely event of a loss of the Dardanelle Reservoir water inventory. The capability of the Service Water System to take suction from the Emergency Cooling Pond (ECP) and to establish a discharge flow path to the pond is demonstrated by performance of various periodic functional tests. It is sized to contain sufficient water for dissipating the total combined heat transferred to the Unit 1 and Unit 2 Service Water Systems as a result of a postulated DBA limiting the returning plant cooling water temperature to 121 °F. This was originally calculated to be 120 °F then revised up to 129.5 °F. Further investigation with more accurate calculational methods yielded a peak temperature of 120.8 °F or 121 °F. The most recent analysis integrates ECP temperature observations, Little Rock and Russellville meteorological data, and ANO

meteorological tower data to determine a new peak ECP temperature more representative of actual site conditions. This analysis determined a peak temperature of approximately 116 °F (115.8 °F). The pond has a surface area of approximately 14 acres and is maintained between an average depth of 5.2 to approximately 6.0 feet for a total water volume of 70 to approximately 82 acre-feet.

During any normal or abnormal plant shutdown operation, the design objectives of the plant ultimate heat sink are assured through the availability of the ECP, together with the associated intake and discharge piping. The Dardanelle Reservoir provides the primary heat sink during normal plant operation while the ECP may provide the seismic category 1 backup source for plant safe shutdown if necessary under normal or accident conditions.

A valved connection is available for the supply of raw water to the Emergency Cooling Pond as makeup water. This connection is provided with a 6-inch reverse flow valve and is cross-connected between an 8-inch and a 10-inch supply line.

The factor most affecting the evaporative loss of the cooling pond for a given set of climatological conditions is the heat load rejected by the cooling systems. In general, the higher the heat load, the greater the evaporation. Differences in the quantities of water flow for different modes of operation do not affect this evaporation. For example, increasing the water flow rate through the pond does not significantly decrease the water's residency time in the pond and, hence, the evaporation is unaffected.

The shutdown of either unit individually imposes less heat load on the pond than a simultaneous shutdown of both units. Similarly, it can be shown that the heat load is less for a simultaneous shutdown of both units than for a DBA on one unit and a concurrent shutdown of the other unit. Since there is slightly more stored energy in Unit 2, the operating condition which results in the minimum margin to Technical Specification limits is a Unit 2 DBA and a concurrent Unit 1 shutdown.

The average water depth of the pond is monitored daily to ensure that it is greater than or equal to the minimum depth specified in the Technical Specifications. The depth is read from a permanently installed device in the pond and recorded in a log by plant personnel. Since changes should be small from day to day, more than sufficient time is available to observe dangerous trends, e.g. decreasing water depth, and take appropriate action. Consequently, high and low level alarms are not necessary. To ensure that the required volume of water is available, soundings are made annually to determine if the pond bottom elevation has changed.

The Emergency Cooling Pond is conservatively sized using proven analytical methods to handle the highly unlikely event of a DBA in one unit and a shutdown in the other unit under adverse climatological conditions. Thus, a special operational test program to confirm predicted performance is judged to be unnecessary.

The kidney shape design of the pond and design of the overflow weir prevent hydraulic short circuits. It is considered that possible layers of ice on the pond surface would not cause flow blockage of the cooling water system. Since the use of oil-base insect sprays in the area surrounding the pond is minimal, any small amounts of this spray which might settle on the surface of the cooling pond have a negligible effect on its performance.

Should it become necessary to use fire fighting equipment while the pond is being used concurrently for plant shutdown, a 2,500 gpm fire pump may be operated for two hours without reducing the normal pond water inventory by more than 1.1 percent (a decrease in pond level of approximately 0.79 inches with an initial water level of 6 feet).

Natural surface drainage (drainage area, 225 acres[Note 1]) is used for initial fill as well as makeup for evaporation losses. In the event of a prolonged drought period, the pond minimum water level can be maintained by supplying makeup water from the Russellville water supply line to the plant site or from Lake Dardanelle using the service water pumps. A suitable spillway is also provided for the overflow of excess water. The profile of the cooling pond spillway exit channel to the Dardanelle Reservoir is shown on Figure 9-29. The existing channel is an unlined channel with a bottom slope of 0.000625. A 20-foot long concrete apron with baffle blocks and a sill are provided immediately downstream of the ogee spillway to adequately dissipate the energy from the maximum discharge of 1063 cfs before it reaches the discharge channel.

Note 1: The natural surface drainage area during site construction was approximately 225 acres. No credit is taken for drainage area in the ECP inventory analyses; therefore, it is not required to maintain a minimum 225 acre drainage area (Reference CR-ANO-C-2008-1728).

Effect of the channel sill on the spillway capacity

The hydraulic analysis on the channel downstream from the spillway and the stilling basin showed that at a discharge of 1,063 cfs the expected water surface in the basin would be at elevation 344.6 feet which is 2.73 feet below the spillway crest. Therefore, the channel sill would not cause a reduction in the spillway capacity.

The analysis is based on a channel slope of 0.000625 with a bed width of 60 feet and side slopes of 2.5:1. The water surface profile between the channel and the pond is shown in Figure 9-30.

Spillway rating curve

The spillway rating curve is shown in Figure 9-39. The curve was developed following the methodology of the US Bureau of Reclamation "Design of Small Dams."

Pond stage hydrograph

The pond level hydrograph, which is shown in Figure 9-31, is obtained from routing the flood, which is equivalent to one-half the Probable Maximum Flood (PMF), through the pond storage using the developed spillway rating curve. The inflow and outflow hydrographs are shown on Figure 9-38.

A topographic map of the Emergency Cooling Pond showing the inlet and outlet structures is shown in Figure 9-32. The normal drainage flow into the cooling pond is routed through by means of open ditches. No diversion dike is required for this purpose. Plan and elevation views of all structures, including spillway, intake and discharge structures, and a topographical map of the area surrounding the pond are provided in Figures 9-33 through 9-35.

The pond performance is analyzed using a numerical cooling pond model which simulates the steady state operation of a flow-through cooling pond. The rainfall distribution, runoff hydrograph, pond routing hydrographs, and rating curves are provided in Figures 9-36 through 9-39.

The small drainage area and the short concentration time make it impossible to derive a unit hydrograph to define the runoff pattern with sufficient accuracy. Therefore, the Rational formula is used to determine the peak discharge. This formula is generally acceptable for small areas.

The spillway is designed to accommodate one-half the local maximum precipitation at maximum flow capacity. Since it is conservatively assumed that there is no storage upstream of the railroad embankment, the computed values are somewhat higher than the actual values and no additional design margin is considered necessary. The remainder of the pond remains functional for conditions corresponding to the spillway design condition.

An initial dilution of 1.5 (entrained flow is one-half circulating water flow) is used for the initial mixing at the point of discharge into the pond. A low overflow weir is provided to spread the discharge across the surface and inhibit the initial mixing. It is assumed that the pond water surface was at elevation 347 feet and the effective surface area is 14 acres. An effective depth of three feet (Total pond depth is 5.8 feet.) was used for the original Unit 1 ECP performance analysis. This depth represents that of the warm water layer, and with the effective area of 14 acres, results in a travel time of a parcel of water through the pond of about 12 hours. The steady state model is used to determine the temperature drop of a parcel of water as it traveled across the pond using average imposed heat loads for each 12-hour period. The Lake Hefner evaporation equation is used for the pond performance calculation using the average July climatology condition listed below:

dry-bulb temperature	82 °F
relative humidity	72 percent
solar radiation	555 Langley's/day
wind speed	4 mph

It is also assumed that no rain falls during the cooling period to restore the water that is lost by evaporation.

The wind speed used is the average summer value determined from measurements made at the site in 1968 and 1969. The remaining values are based on average July data for the years 1951 through 1967 from the Little Rock Weather Bureau station. The Lake Hefner evaporation equation is adopted as a conservative model for the pond and the computed equilibrium temperature for the above climatic conditions is calculated to be 94 °F. This equilibrium temperature is then used as the initial pond temperature.

The meteorological data used represents an average July situation. The 94 °F equilibrium temperature computed using this data, and the pond outlet temperatures, are believed to be conservative for the conditions studied because of the use of the Lake Hefner evaporation equation. Recent studies indicate that the Lake Hefner equation underestimates the evaporation rate at low wind speeds. Therefore, it can be expected that actual pond outlet temperatures are lower than the calculated temperatures.

As part of the licensing of Unit 2, pond temperatures were re-analyzed using the worst 1-day average meteorology followed by the worst 30-day average meteorology for Little Rock for the period of record (1930-1973). Worst meteorology is defined as those conditions resulting in maximum pond outlet temperature. To determine whether the Little Rock long-term meteorological data is representative of conditions at the plant site, a comparison was made of simultaneous data taken by the onsite instrumentation and the Little Rock data. Since maximum pond temperatures are expected only during summer months, the period April 1 through September 30, 1972 was used for this study. A digital computer program was used to compute the daily and 30-day running average equilibrium temperatures and evaporation rates for each location. Because only temperature and wind speed data were taken at the site, the Little Rock relative humidity, solar radiation, and cloud cover were also used in the site calculations. Pond performance was simulated using the worst case meteorology for both the site and Little Rock. It was found that the peak pond outlet temperature, based on site data, was 6.5 °F higher than the peak pond outlet temperature based on Little Rock data. Brady's wind speed function was used for this temperature analysis.

The original DBA maximum pond outlet temperature, based on worst case Little Rock meteorology for the period 1930-1973 and Brady's windspeed function, was computed to be 123 °F, including the effects of diurnal fluctuations. The worst case 1-day and 30-day meteorology conditions are summarized in Table 9-17. Adjustment for the difference between site and Little Rock data results in an estimated peak pond outlet temperature of 129.5 °F.

The 129.5 °F peak pond temperature, determined in a revised analysis performed in accordance with Revision 1 of Regulatory Guide 1.27, is 9.5 degrees higher than the temperature determined in the previous analysis. As a result of this temperature increase, a study was conducted to determine the impact of these higher temperatures on the safety-related equipment which is cooled by the Service Water System and required to function throughout the post-LOCA recovery period. Both Unit 1 and Unit 2 equipment were included in this study. The pond temperature vs. time curve shown in Figure 9.2-19 (ANO-2 SAR) was sent to the suppliers of each of the equipment involved.

As a result of this study, some of the equipment was re-rated for the higher temperatures based on existing test data. For the remainder of the equipment, additional testing is being performed and/or equipment modifications are being made to ensure that this equipment will function properly throughout the period of elevated pond temperatures.

Required modifications to the Control Room Emergency Air Conditioning System which were made to meet operating requirements with the above maximum pond temperature are described in Section 9.7.2.1.

Further analysis of ECP response, consistent with Regulatory Guide 1.27, Revision 1, with a computer model which has been benchmarked against an operating cooling pond and more accurately reflects pond behavior, has yielded a much lower peak temperature of 120.8 °F or 121 °F (Reference 14). This computer model maintains the assumptions utilized in the previous model except the discharge water is assumed to be pulled from all layers of the pond, not just the top layer. The meteorology was reviewed to ensure that compliance with the regulatory guide was maintained. Changes to the meteorological data used pertained to the inclusion of diurnal fluctuations for the first five days of the event to verify the peak pond temperature had been attained.

The most recent analysis was performed in accordance with Regulatory Guide 1.27, Revision 2, accepted methodology using a meteorological data record that also includes local and site observed data. This analysis has yielded a lower peak temperature of approximately 116 °F (115.8 °F) (Reference 19). This analysis uses the record of observed ECP temperatures (approximately 33 years) to identify the period of highest ECP temperature (July, 1980). Regional meteorological data supplemented by local data was then used to demonstrate that the surface heat exchange model reproduces the ECP temperature observations. Wind speed reduction calibration methods of local (met tower) and regional (Little Rock) wind speed data were used to reproduce local temperature observations to eliminate the need for a temperature correction of 6.5 °F between the site and Little Rock data.

The temperature profile from the most recent analysis is shown in Figure 9-22A and the heat loads utilized for this analysis are shown in Figures 9-23 through 9-25.

In the event of a LOCA and the single failure of the Lake Dardanelle Reservoir, Unit 1 can be safely shut down on the ECP.

Pond evaporation losses were analyzed using the worst 30-day meteorology with respect to natural evaporation rate and dew point depression. Total pond evaporative water loss during the 30 days of operation after the postulated ANO-2 LOCA was computed to be 24 inches, or 28.2 acre-feet. The emergency cooling pond inventory loss analysis includes the effect of evaporative losses, safe shutdown unit condensate requirements, spent fuel pool makeup requirements, sluice gate leakage, system boundary valve leakage, two hours fire pump usage, the impact on pond inventory due to transferring service water system discharge and suction from the lake to the pond, piping leakage, and seepage (Ref. 19). Margins are maintained such that the level 30 days after the start of the event is maintained above the acceptable minimum. The minimum pond depth for proper hydraulic performance (cooling capability) is computed at 18 inches. Operator action is credited, as allowed by Regulatory Guide 1.27, to increase pond inventory analysis during the transfer of the service water system to the pond. Specifically, pump returns are transferred to the pond shortly after a loss of lake event and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the pond, lake water is pumped into the pond increasing level. This additional water is required, along with that maintained by Technical Specifications to ensure a 66.9 inch pond depth which corresponds to a 30 day supply of cooling water (Ref. 19).

Separate suction and discharge water lines are used for supplying pond water to the Unit 1 and Unit 2 Service Water Systems. The ends of the lines terminating at the cooling pond are housed in Category 1 structures to prevent blockage of the pipe entrance and outlet. In addition, screens are provided on the intake structure to prevent the inclusion of soil or foreign objects in the water delivered from the pond. The intake screens consist of 12 gauge wire with 3/8-inch openings. Provisions are made for access to clean these screens periodically. However, because of the low fluid velocity and relatively little turbulence around the intake structure, it is not anticipated that significant amounts of soil or foreign objects are included in the pond water.

The cooling pond, in general, is excavated in an impervious clay strata with the bottom of the pond about 4 to 16 feet above rock. The pond location and the logs of the pond exploration holes indicate that depth to sound bedrock varies from about 10 to 22 feet below the present ground surface. Weathered shale which extends to or above the pond bottom is excavated to a depth of two feet below the pond bottom and replaced with well compacted impervious clay material. The pond sides are handled in a similar manner as required.

The pond side slopes are protected against wave action by at least 18 inches of rip-rap placed all around the pond to prevent erosion.

The slope of the pond is a Category 1 structure and is only a shallow excavation in the overburden. The stability of the pond side slopes was investigated under normal operating conditions under a seismic loading of 0.2g ground acceleration. The added embankment slopes for the reinforced concrete spillway were investigated for an effective seismic loading of 0.355g.

For normal operating conditions a factor of safety of 1.5 is required and for the above-mentioned seismic condition a factor of safety of 1.0 is considered acceptable.

The varying depth to bedrock below the bottom of the pond is taken into consideration in the analyses.

Stability analysis design strength parameters for the pond slopes are based on residual shear strengths obtained from drained direct shear tests on undisturbed samples of overburden soils.

These are as follows:

Unit weight of soil above groundwater table:	124 lb./cu.ft.
Submerged unit weight of soil:	62 lb./cu.ft.
Consolidated drained direct shear strength:	s = 200 + ptan (23°)

where s is the shear strength in pounds per square foot and p is the normal pressure in pounds per square foot on the plane on which the shear strength is measured.

The analyses for the original pond embankments are made in accordance with the modified Swedish Slip Circle method, known as the "Method of Slices." It is assumed that the water and earth pressures on the sides of each slice are in balance. The earthquake load used in the analysis is the DBE with 0.2g ground acceleration acting on the mass of soil and water in the slope.

For the slopes added with construction of the reinforced concrete spillway for the ECP, the stability analyses were made in accordance with the Simplified Bishop Method, the Simplified Janbu Method, and the Spencer Method. These methods are also part of the family of "The Method of Slices," and provide equivalent or conservative results compared to the Modified Swedish Method (see Army Corps of Engineers, "Slope Stability," EM 1110-2-1902, 2003). For the seismic analysis of these added embankment slopes, the equivalent seismic DBE acceleration of 0.355g was used, which was obtained from an updated soil-structure interaction analysis. The same load conditions as used for the original ECP slopes were used for the analysis of these added embankments. Laboratory test results were used for the shear strength properties for the clay for the new embankments, with all other soil shear strength properties the same as for the original embankments.

The results of the above analyses indicate that soil slopes of two and one-half horizontal to one vertical are adequately stable under the conditions stated above. The results of the analyses are listed in Table 9-19.

Earth fill is placed along a limited portion of the pond perimeter to maintain a crest elevation of 351 feet. The inside face (water side) of the dike on the sides of the spillway is protected with an 18-inch thick layer of dump rip-rap on top of a 6-inch layer of bedding material. The rip-rap is brought up to an elevation of 353 feet on approximately 700 feet of dike; thus providing a 3-foot freeboard for the design condition of one-half the local probable maximum precipitation. This provides a normal freeboard of six feet. This fill and rip-rap construction begins above an elevation of 346 feet, which is one foot below normal pool level. Because of the very minor amount of fill extending below the normal pool level (one foot maximum) no separate stability analyses are provided for this condition. This is because the 2.5 horizontal to one vertical was previously shown to be adequately stable for slopes higher than the nine feet at those places where fill is required. The added embankments for the reinforced concrete spillway are constructed one-foot higher in elevation than the original embankments.

The specification for selection and placement of the earth fill requires that impervious material for embankments and backfill shall consist of red silty or sandy clay from required excavation, free of gravel, cobbles, boulders, or pockets of silt, sand, or other non-plastic material. Embankment material shall be placed in 8-inch loose layers, moisture conditioned, and compacted to 95 percent of maximum dry density as determined by the Modified Compaction Procedure ASTM Test Designation D-1557. Per EC 443, for the embankments added with the reinforced concrete ogee spillway, it was demonstrated that the embankment material placed in 8-inch loose lifts, compacted to 90 percent of maximum dry density provided adequate stability and met permeability requirements.

Since the pond sides and bottom consist of an impervious material and field and laboratory permeability test results indicate little or no water losses due to seepage, loss of water through these areas should be expected to be minimal. Similarly the leakage due to buried pipe and pump compartments is assumed to be negligible. Natural runoff above the pond is diverted so as to pass through the pond before discharging to the Dardanelle Reservoir. Any excess runoff is discharged by use of a spillway, which is designed to accommodate one-half the probable maximum precipitation at maximum flow capacity, based on a total rainfall of 19.5 inches in two hours. The spillway is a reinforced concrete ogee shaped spillway, founded on and keyed into the weathered shale layer. It is shown to be stable against sliding and overturning for the DBE and hence, is also stable under an OBE condition. Minor losses at the spillway as well as minor leakage and normal evaporation should be more than offset by the average annual rainfall over the pond.

For the flood condition due to one-half the local probable maximum precipitation, the maximum water surface for the design flow rate of 1,063 cfs for the new spillway has been determined by a hydraulic analysis to be approximately Elevation 350 feet 5¼ inches. The indicated maximum water surface relative to the original spillway was estimated to be at Elevation 350 feet. The increased maximum height of the pond during the design storm results from abandoning the original spillway in-place, with no modifications. The new spillway was designed with the consideration that the top layers of articulated blocks would be removed from the original spillway. The blocks were left intact to alleviate future potential erosion concerns, with the result of a reduction in the efficiency of the new spillway. The three foot freeboard originally provided was based on standard practice at the time of original construction with no consideration for actual wave-height and wind setup calculations for the ECP. From EC443, a maximum rise in the water surface from the design wind of 90 mph was conservatively estimated to be 0.13 feet and the significant wave height was conservatively estimated to be 1.88 feet. Combined, this gives an estimated maximum wave height to be considered of 2.01 feet. This is less than the nominal 2 feet 6¾ inch freeboard provided with the pond level at 350 feet 5¼ inch.

The pond was excavated using conventional earth moving equipment. The concrete Category 1 intake and discharge structures are keyed into the existing clay and backfilled using an impervious material to form a seal at these points.

9.3.2.5 Potable Water and Sanitary Water

Neither the Potable nor the Sanitary Water System has any connection with any safety-related item in the plant.

Figure 9-40 is a flow diagram of the Domestic Water System. Russellville water supply to the raw water storage tank is rated at not less than 100 gallons per minute, continuous. Since there is no communication between the Potable and Sanitary Water Systems and any contaminated plant water system, there is no way to contaminate either system with radioactive material, flammable material, or chemicals through system cross-connection.

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 DESIGN BASES

The Spent Fuel Cooling (SFC) System is designed to maintain the water quality and clarity and to remove the decay heat from the stored fuel in the spent fuel pool. It is designed to maintain the spent fuel pool water at less than or equal to approximately 150 °F during full core offload conditions, while removing a decay heat load of less than or equal to 31.0 MBTU/hr.

In meeting the foregoing design bases, the system has the capability of maintaining the spent fuel pool water at approximately 120 °F with a heat load based on the decay heat generated from approximately one-third of the core fuel assemblies discharged at the end of any given cycle.

In addition to its primary function, the system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the Borated Water Storage Tank (BWST) in order to remove fission and corrosion products and to maintain water clarity for fuel handling operations. The system also provides for filling the fuel transfer canal, the incore instrumentation tank, and the cask loading area from the BWST.

9.4.2 SYSTEM DESCRIPTION

The Spent Fuel Cooling System, shown in Figure 9-47, provides cooling for the spent fuel pool to remove fission product decay heat energy. Design data are shown in Table 9-9. Major components of the system are summarized in Table 9-9 and briefly described below.

Spent Fuel Coolers

The spent fuel coolers are designed to maintain the temperature of the spent fuel pool as noted in Section 9.4.1. The spent fuel pool coolers reject heat to the nuclear intermediate cooling water system which subsequently rejects its heat to the service water system.

Spent Fuel Pool Circulating Pumps

The two spent fuel pool circulating pumps take suction from the spent fuel pool and recirculate the fluid back to the pool after passing through the coolers. Cold water is discharged into the pool through two nozzles simultaneously. One of these nozzles is located near the water surface while the other one is near the bottom of the pool. The suction nozzle for the spent fuel pool circulating pumps is located on the opposite end of the pool from the discharge nozzles and is near the water surface. This arrangement provides thermal mixing and insures uniform water temperature. The system provides purification of the spent fuel pool water through the spent fuel pool demineralizer and spent fuel pool filters in order to remove fission and corrosion products and to maintain water clarity for fuel handling operations. During refueling operations these pumps are also used for filling the fuel transfer canal and incore instrumentation tank with borated water from the BWST.

Complete filling of the fuel transfer canal is accomplished using a spent fuel pool circulating pump taking suction from the BWST.

The connection between the SFC system and the decay heat letdown line within the reactor building is used to partially empty the fuel transfer canal using the decay heat removal pump. The canal water is pumped to the BWST via the test line. When the water level reaches the reactor vessel flange, final drainage of the lower fuel transfer canal is accomplished using the spent fuel pool circulating pumps.

Spent Fuel Coolant Demineralizer

The spent fuel coolant demineralizer processes approximately one-half of the spent fuel pool volume in 24 hours.

Spent Fuel Coolant Filters

The spent fuel coolant filters are designed to remove particulate matter from the spent fuel pool water. They are sized for the same flow rate as the demineralizer.

Borated Water Recirculation Pump

This pump removes water from the BWST for demineralization and filtering. The pump may also be used for demineralizing and filtering the water in the fuel transfer canal during a transfer of fuel. It can also provide for purification of the spent fuel pool.

During refueling operations the water in the transfer canal and the water in the Reactor Coolant System (RCS) come in contact. The borated water recirculation pump and the spent fuel demineralizers are used to purify this water during refueling and prior to draining the transfer canal back into the BWST.

The purification and filtering of the water in the BWST is performed when further purification of the water drained from the transfer canal into the BWST is desired or when the level in the spent fuel pool is reduced by pumping the water into the BWST.

9.4.2.1 Modes of Operation

9.4.2.1.1 Normal Operation

The normal operation of the Spent Fuel Pool Cooling System serves two main functions. The first is to maintain the pool water at temperature less than or equal to approximately 150°F for the full range of fuel pool heat loads, although the system can typically maintain the pool water at or below a temperature of approximately 120 °F for a normal offload. The second function is to provide purification of the spent fuel pool coolant for clarity during fuel handling operations.

The first function is accomplished by recirculating spent fuel coolant water from the spent fuel pool through the pumps and coolers and back to the pool. The spent fuel coolant pumps take suction from the spent fuel pool and deliver the water through the tube side of two coolers arranged in parallel back to the pool.

A bypass purification loop is provided to maintain the purity of the water in the spent fuel pool. This loop is also utilized to purify the water in the BWST following refueling and to maintain clarity in the fuel transfer canal during refueling. Water from the BWST or fuel transfer canal can be purified by using the borated water recirculation pump.

9.4.2.1.2 Thermal-Hydraulic Considerations

The ANO spent fuel pool cooling and purification systems are designed to maintain water quality and clarity and to remove the decay heat from the stored fuel.

9.4.2.1.2.1 Spent Fuel Pool Cooling System Design Bases

The ANO-1 cooling system is designed to maintain the spent fuel pool water at or below approximately 150 °F with the maximum theoretical heat load resident in the pool for the range of service water temperatures and corresponding nuclear intermediate cooling water temperatures anticipated during the cooler months of the year when refueling is typically scheduled. Refueling operations are administratively controlled in order to minimize the potential of exceeding a pool temperature of 150 °F during a full core discharge whenever service water and nuclear intermediate cooling water temperatures are elevated. The administrative controls include evaluation of decay heat loads using the guidelines of ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling. Station administrative controls ensure that the heat removal capacity of the pool is sufficient for future core offloads given the type and enrichment of fuel, and the number of assemblies to be offloaded.

9.4.2.1.2.2 Local Fuel Bundle Thermal-Hydraulic Analysis

A local fuel bundle thermal-hydraulic analysis is performed to determine the bounding peak local water temperature in the storage rack cell containing the hottest spent fuel assembly and the bounding peak fuel cladding temperature for the hottest spent fuel assembly in the racks. Conservative assumptions were incorporated into the bounding local thermal hydraulic analysis, such that actual decay heat loads are lower than calculated. The local thermal hydraulic analysis predicated decay heat loads using ORIGEN2 based methodology.

9.4.2.1.2.2.1 Assumptions and Considerations

Included among the key assumptions used in the analysis are:

1. All passive losses (i.e., conduction through walls and slab or losses from the surface) are completely neglected. This conservatively maximizes the net heat load, thereby maximizing both global and local temperatures.

2. All calculations are conservatively performed by assigning the highest hydraulic resistance parameters, from those calculated for the different rack types in the spent fuel pool, to all the racks in the SFP.

3. The calculated fuel assembly hydraulic resistance parameters are worsened to ensure an analysis that bounds any small deviations in fuel assembly and rack geometry. The two calculated parameters, permeability and inertial resistance factor, are conservatively worsened by 10%.

4. The bottom plenum gap (i.e., between the floor and the rack base plate) is conservatively modeled as less than the actual value. This ensures that the effects of additional flow restrictions around rack pedestals and bearing pads are bounded in the model.

5. No downcomer flow is assumed to exist between the rack modules.

6. Flow through the flux trap gaps are conservatively neglected in the analysis.

7. The model credits conservatively lower rack-to-wall gaps instead of the actual values. This conservatively maximized the downcomer flow resistance and ensures that the calculations will encompass slight positional deviations in the as-installed rack configuration.

8. At each pedestal location, the rack baseplate has 4 holes of 2 inches diameter each and 1 large hole 4 inches in diameter. The pedestal feet block the flow through the large hole. Therefore, water flow into the rack cell at the pedestal locations is through the small holes. The non-pedestal locations have 1 hole of 4 inches diameter on the baseplate performing the same function. In the analysis, water flow area into every rack cell is taken to be equal to the cross-sectional area of 4 holes of 2 inches diameter each. This area is the same as that of 1 hole of 4 inches diameter.

9. The hydraulic resistance of every rack cell in the spent fuel storage rack includes the inertial resistance that would result from a dropped fuel assembly lying across the top of the rack. This conservatively increases the total rack cell hydraulic resistance and bounds the thermal-hydraulic effects of a fuel assembly dropped anywhere in the spent fuel storage area.

10. All the freshly discharged fuel assemblies are assumed to be located together at the center of the spent fuel pool, conservatively maximizing the local decay heat generation rates.

11. For the bounding peak fuel cladding computations, the maximum local water temperature (at the fuel rack cell exit) and peak heat flux (typically near the mid-height of the active fuel region) are considered to occur co-incidentally. The superposition of these two maximum values ensures that the calculated peak fuel cladding temperature bounds the fuel cladding temperature anywhere along the length of the fuel assembly.

9.4.2.1.2.2.2 Description of Analytical Method

The decay heat generated by the fuel assemblies stored in the rack induces a buoyancy-driven water flow upward through the fuel rack cells. Cooler water is supplied to the bottom of the rack cells through the downcomers and the bottom plenum. An elevation view of a typical model is sketched in Figure 9-51 where the flow paths are indicated by arrows. Each cell shown actually corresponds to a row of cells that are located at the same distance from the pool walls. Quantification of the coupled flow and temperature fields in the spent fuel storage rack is accomplished through the use of a Computational Fluid Dynamics (CFD) analysis.

A three-dimensional model of the spent fuel pool area is made using FLUENT. The regions in the SFP occupied by the fuel racks loaded with heat generating fuel assemblies are modeled as a porous media region. Flow through the narrow fuel assembly passages is laminar and governed by Darcy’s Law.

Navier-Stokes equations of fluid motion along with the energy conservation equation are solved to obtain the local flow field and the steady-state temperature distribution in the SFP. Buoyancy effects and turbulence effects are included in the CFD analysis. Turbulence effects are modeled by relating time-varying “Reynolds Stresses” to the mean bulk flow quantities using the standard k-ε model. Again, flow through the fuel rack will be laminar and turbulence effects are “turned off” in the porous media region.

The decay heat generated by the spent fuel assemblies stored in the fuel racks is included in the model as a volumetric decay heat generation in the porous media region.

Once the spatial temperature distribution in the spent fuel storage area is obtained from the CFD solution, the difference between the cladding surface temperature and the local water temperature (also called the cladding superheat) is conservatively calculated from the principles of laminar flow heat transfer. The cladding superheat is added to the CFD computed peak local water temperature to obtain a bounding cladding surface temperature.

9.4.2.1.2.2.3 Types of Calculations

Several calculations are performed that generate data used to construct the three-dimensional CFD model. First, the overall spent fuel pool and fuel rack dimensions are converted into a set of x-, y-, and z-coordinates. These coordinates are used to rapidly construct the CFD model geometry, including the positions of the porous media region used to model the loaded fuel racks. Second, the hydraulic resistance parameters (i.e., permeability and the inertial resistance factor) are calculated for the loaded fuel racks cells. All the different rack types in the SFP are evaluated and the resistance of the most limiting rack type is used in the CFD model. Third, the volumetric decay heat generation rates for the fuel rack regions of the CFD model are determined along with the inlet water conditions.

The FLUENT program is used to compute the coupled temperature and velocity profiles that exist in the spent fuel pool. A single bounding scenario is evaluated that includes the highest permissible bulk SFP temperature and highest calculated decay heat loads, the highest fuel assembly hydraulic resistance and the additional resistance of an assumed dropped fuel assembly laying across every rack cell in the spent fuel storage racks. The peak local water temperature is determined using the FLUENT post-processing functions.

The bounding fuel cladding superheat is determined from fuel assembly geometry data and the bounding assembly decay heat. Finally, a bounding value for the peak cladding temperature is determined by adding the peak fuel cladding superheat value to the peak local water temperature computed using the three dimensional CFD model.

9.4.2.1.2.2.4 Acceptance Criteria

The criteria used to determine the acceptability of the design from a thermal-hydraulic viewpoint are summarized as follows:

1. The bounding peak local water temperature in the spent fuel pool storage cell containing the hottest spent fuel assembly must be less than the local saturation temperature of water.

2. The bounding peak fuel cladding temperature for the hottest spent fuel assembly, determined by adding the bounding temperature difference between cladding material and water in the rack to the maximum local water temperature, should be less than the local saturation temperature of water. If the fuel cladding temperature exceeds the local saturation temperature of water, it must be shown that departure from nucleate boiling (DNB) will not occur.

9.4.2.1.2.3 Rack/Pool Thermal Hydraulics - Design Bases

The ANO-1 cooling system is designed to maintain the spent fuel pool water at or below approximately 150 °F with the maximum theoretical heat load resident in the pool at the maximum service water temperature and corresponding nuclear intermediate cooling water temperature anticipated during the cooler months of the year when refueling is typically scheduled. Refueling operations are administratively controlled in order to minimize the potential of exceeding a pool temperature of 150 °F during a full core discharge whenever service system temperature and nuclear intermediate cooling water temperature are elevated. The administrative controls include evaluation of decay heat loads using the guidelines of ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling. The maximum theoretical heat load was historically based on ASB 9-2 based decay heat calculations using actual or typical spent fuel operating and cooling times. Since bounding calculations have been performed for the local thermal hydraulic analysis and bulk temperature limits are satisfied through a combination of administrative controls and cooling system performance curves, a maximum theoretical heat load of up to 31.0 MBTU/hr can now be accommodated. The cooling system performance curves are included in the Technical Requirements Manual. Station administrative controls ensure that the heat removal capacity of the pool is sufficient for future core offloads given the type and enrichment of fuel, and the number of assemblies to be offloaded.

9.4.2.2 Reliability Considerations

The Spent Fuel Pool Cooling System provides adequate capacity and component redundancy to assure the cooling of stored spent fuel. Ample time is available to assure that cooling can be effected even in the unlikely event of multiple component failures or complete cooling loss. The system is so arranged that no uncontrolled loss of water from the pool is possible by piping or component failure. The system performs no emergency function and is not directly connected to the RCS.

9.4.2.3 Codes and Standards

Each component of this system is designed to the code or standard, as applicable, as noted in Table 9-9.

9.4.2.4 Leakage Considerations

If a leaking fuel assembly is transferred from the fuel transfer canal, a small quantity of fission product activity may enter the spent fuel pool cooling water, even though the assembly's cladding temperature is lowered and leakage may reasonably be expected to be minimized. The purification loop removes these fission products and other contaminants from the water.

The fuel handling and storage area housing the spent fuel pool is monitored for gaseous activity and gases are exhausted to the environment through the unit vent. Provisions are made in the design to air-test the flanged end of the fuel transfer tube for leak tightness after use.

9.4.2.5 System Isolation

The Spent Fuel Cooling System has no process lines connecting to the RCS. Its major penetration to the reactor building is through the fuel transfer tube. The fuel transfer tube is isolated inside the reactor building by a blind flange connection in the fuel transfer canal. The

Decay Heat Removal System may be used under reactor defueled conditions for spent fuel cooling. In this mode of operation, the fuel transfer tube is used to transfer water from the spent fuel pool to the reactor vessel for transfer to the decay heat pump suction. For details, see SAR Section 9.5.

9.4.2.6 Failure Considerations

Failure of a single component in this system does not permit uncovering of the stored spent fuel under normal operating conditions since the system is designed with redundant components. With a complete loss of cooling, considerable time is required to raise the temperature of the pool water to boiling.

The Spent Fuel Cooling System is designed with redundant components to prevent loss of cooling and subsequent evaporative loss of coolant due to failure of a single component. In the event of such a failure, the loop containing the component would be isolated and cooling continued with the other loop. Coolant loss due to boiling, in the unlikely event of a complete loss of cooling, would not be significant because of the considerable time required to raise the pool water to boiling temperature. Fuel pool indicators and alarms provide prompt warning of any loss of coolant.

9.4.2.7 Operational Limits

A level transmitter provides Control Room indication and alarm~~s~~ on high (+ 0.5') or low (- 0.5') storage pool level. Normal level is indicated in the Control Room at 0.0' which corresponds to a pool level of 3.5 feet below the Spent Fuel Pool (SFP) deck.

NRC Order EA-12-051, "Issuance of Order to Modify Licenses with regard to Reliable Spent Fuel Pool Instrumentation," required plants to provide reliable SFP instrumentation in partial response to the March 2011 Fukushima accident. NRC interim staff guidance JLD-ISG-2012-03 endorsed NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051," as an acceptable approach for satisfying the requirements of Order EA-12-051. Primary and backup SFP level instruments have been installed which output in the Control Room. The level instruments aid in the monitoring and maintenance of SFP level to support operation of the SFP cooling system, provide radiation shielding for personnel on the SFP operating deck, and to ensure the fuel remains covered. The instruments are powered from battery-backed vital 120 VAC power and each have a backup battery source installed.

Three pool temperature indications are available, two providing computer point alarm in the Control Room on high pool temperature (140 °F) and high SFP demineralizer inlet temperature (120 °F), and one providing local indication.

9.5 DECAY HEAT REMOVAL SYSTEM

9.5.1 DESIGN BASES

The Decay Heat Removal (DHR) System removes decay heat from the core and sensible heat from the Reactor Coolant System (RCS) during the latter stages of cooldown. The system also provides auxiliary spray to the pressurizer for complete depressurization, maintains the reactor coolant temperature during refueling, and provides a means for partial draining of the fuel transfer canal. Under special defueled conditions, the system can provide an alternate means of Spent Fuel Cooling. In the event of a LOCA, the system injects borated water into the reactor vessel for long-term emergency cooling. The emergency functions of this system are described in Chapter 6.

9.5.2 SYSTEM DESCRIPTION AND EVALUATION

The DHR System is shown schematically in Figure 6-1. The DHR System normally takes suction from the reactor coolant outlet line and delivers the water back to the reactor through the core flooding nozzles after passing through the DHR pumps and coolers. The DHR System may be operated when the reactor pressure is within the system design pressure for cooldown of the system to refueling temperatures. During this operation, the pressurizer is cooled by auxiliary spray from the decay heat system in accordance with operating procedures which prescribe the required cooldown rate limitations (1CAN078315). The decay heat is transferred to the Service Water System by the DHR coolers. Performance and equipment data are shown in Table 9-10.

A description of the major system components follows.

Decay Heat Removal Pumps

Two pumps are arranged in parallel and are designed for continuous operation during the period required for removal of decay heat for refueling. If two pumps are in service, the design flow can cool the RCS from approximately 280 °F to 140 °F in 14 hours. The emergency functions of these pumps for Low Pressure Injection are described in Chapter 6.

Decay Heat Removal Coolers

The steam generators are used to cool the RCS from operating temperature to less than 280 °F. The coolers remove the decay heat from the circulated reactor coolant during a routine shutdown. Both coolers can be operated to cool the circulated reactor coolant from approximately 280 °F to 140 °F in 14 hours. If there is only one cooler and pump available, the cooldown period to 140 °F is approximately five days. The emergency function of these coolers is described in Chapter 6.

Borated Water Storage Tank

The BWST is located outside the reactor building and the auxiliary building. It contains a minimum of 2,270 ppm boron in solution and is used both for emergency core injection and filling the fuel transfer canal during refueling. The BWST also supplies borated water for emergency cooling to the Reactor Building Spray System, the LPI System, and the HPI System. It also supplies makeup water to the Spent Fuel Cooling System.

The bulk water temperature in the Borated Water Storage Tank (BWST) must be maintained above 40 °F under all weather conditions. For this purpose, four electrical immersion heaters with 180 kW total capacity are provided. Each of these four heaters consists of three elements of 15 kW capacity each and each of these elements has its own individual circuit breaker. As a result, the failure of any one element reduces the total capacity by only 15 kW. Since the insulation on the vessel limits the heat losses to below the capacity of one heater, the effect of a BWST heater failure is insignificant. A tank vent is provided to assure proper tank draining. The heating of the tank is considered to be a protection against freezing for the tank (0CAN068004).

9.5.2.1 Mode of Operation

One pump and cooler normally performs the decay heat cooling function. After the steam generators reduce the reactor coolant temperature to less than 280 °F, decay heat cooling is initiated with the pumps taking suction from the reactor outlet line and discharging through the cooler into the reactor vessel. To reduce the cooling time, both decay heat removal pumps and coolers may be operated in parallel. The equipment utilized for decay heat cooling is also used for Low Pressure Injection (LPI) during LOCA conditions.

During refueling, the decay heat from the reactor core is rejected to the decay heat removal coolers in the same manner as it is during the cooldown to 140 °F. At the beginning of the refueling period, both coolers and both pumps may be required to maintain 140 °F in the core and fuel transfer canal. Later, as core decay heat decreases, one cooler and pump can maintain the required 140 °F. When only one pump and cooler is in use, the other pump and cooler may be aligned in the Low Pressure Injection mode to allow rapid recovery of level in case of inadvertent loss of inventory.

Figure 9-45 shows the reactor coolant and fuel transfer canal temperature from approximately one to five days after shutdown, assuming only one DHR pump and cooler is available. Under these conditions, refueling operations would be delayed by four days until the reactor coolant temperature is reduced to 140 °F.

The decay heat pumps and coolers are required to maintain reactor coolant temperature at or below 140 °F during refueling. The DHR System is designed to remove the core decay heat load at 20 hours after shutdown with two coolers at the following design conditions:

Shell side flow:	3,000 gpm
Shell side inlet temp:	140 °F
Tube side flow:	3,000 gpm
Tube side inlet temp:	85 °F

The DHR System can be used to drain and fill the fuel transfer canal during refueling.

The DHR System may be used for Spent Fuel Cooling under defueled conditions when all fuel is offloaded to the Spent Fuel Pool. Cooling water is supplied utilizing existing piping and valve arrangements. Cooling water is returned via the fuel transfer tube and drop leg back to the decay heat pumps suction.

9.5.2.2 Reliability Considerations

Since the equipment is designed to perform both normal and emergency functions, separate and redundant flow paths downstream of the drop line and equipment are provided to prevent a single component failure from reducing the system performance below a safe level.

The reactor building sump is covered with a strainer fabricated of perforated stainless steel sheet metal with 1/16 inch diameter holes. All of the components in the Decay Heat Removal System which are used when the system is in the recirculation mode are capable of operating in the presence of any debris which may pass through this screen without plugging.

9.5.2.3 Codes and Standards

Each component of this system is designed, fabricated, and inspected to the code or standard, as applicable, as noted in Table 9-10 and Table 6-5.

9.5.2.4 System Isolation

The DHR System is connected to the reactor outlet line on the suction side and to the reactor vessel through the core flooding line on the discharge side. The system is isolated from the reactor building on the suction side by two electric motor operated valves located inside the reactor building and one electric motor operated valve located outside the reactor building.

On the discharge side, each line is isolated from the reactor building by an electric motor operated valve and a check valve. All of these valves are normally closed whenever the reactor is in the operating condition. In the event of a LOCA, the valves on the system discharge side open and the valves between the reactor vessel and the suction side of the pumps may be manually opened to establish a long-term boron dilution flow path.

Pressure transmitters have been installed downstream of the two isolation valves in the DHR System to provide indication to the control room operators to aid them in monitoring fluid leakage past the valves.

9.5.2.5 Leakage Considerations

During reactor power operation, all equipment of the DHR System is idle and all isolation valves are closed. Under LOCA conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks were evaluated to obtain the total radiation dose due to leakage from this system. The evaluation is discussed in Chapters 6 and 14.

Leak detection for the BWST is by visual observation supplemented by level indicators and alarms and leakage control is accomplished manually. Although, no "inservice inspection" requirements are specified, the tank is visually inspected internally to the extent possible during refueling outages and inspected externally for deterioration or leakage.

9.5.2.6 Failure Considerations

Failures and malfunctions in the DHR System in conjunction with a LOCA are discussed in Chapter 6.

Use of DHR for Spent Fuel Cooling can result in pumped draindown of the spent fuel pool due to rupture in the spent fuel piping portion of the flow path. Fuel pool indicators and alarms provide prompt warning of any loss of pool level. In the event of such failure, the drop leg would be isolated and the decay heat pump in use would be secured. As noted in Section A.7.6, a break in the piping normally used for decay heat removal is not considered credible.

9.5.2.7 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or plant safety. These variables or conditions of operation are as follows:

Decay Heat Removal Flow Rate

Low flow from the decay heat pumps during the DHR mode of operation is alarmed to signify a reduction or stoppage of flow and cooling of the core. High flow from the decay heat pumps during the DHR mode of operation is alarmed to signify possible vortexing conditions that could result in a loss of suction to the decay heat pumps.

Reactor Coolant Pressure Interlock

The DHR System insures automatic closure of both decay heat suction isolation valves whenever the reactor coolant pressure exceeds the interlock setpoint or when Core Flooding System isolation valves are not closed. The system design also prevents opening of the valves when these conditions exist. This is accomplished by a DHR System suction isolation valve Automatic Closure and Interlock (ACI) System which extends from the sensing instruments to the final actuated devices (valve motors). The purpose of the ACI system is to prevent gross overpressurization in order that a rupture of the DHR System pressure boundary does not occur. The interlock setpoints are set such that the pressures will not exceed the limits of the DHR System. The ACI is designed to comply with the applicable requirements of IEEE 279-1968 and is based on diverse principles.

The ACI consists of two subsystems which are electrically and physically separate from each other and are redundant. Each subsystem consists of pressure monitoring instrumentation, logic instrumentation, contact buffering, a bypass switch, and a valve motor controller and controls the operation of one decay heat suction isolation valve.

To minimize the potential for loss of DHR, both ACI subsystems are bypassed whenever an open RCS assures overpressurization cannot occur. One subsystem at a time may be briefly bypassed for testing or maintenance during DHR operation when the RCS is not open. Each bypass is annunciated in the control room.

Facilities are provided for manual testing of the ACI both prior to pressurization of the RCS and during operation at high pressure. However, the system design precludes opening the decay heat suction isolation valves while testing with RCS pressure above the interlock setpoint. The bypasses are not used for testing under high pressure conditions.

The ACI, including the power supply to the valve motors, is connected to assured redundant power sources and the wiring for these systems is routed in Seismic Category 1 redundant raceways.

Reactor Coolant Leaving the DHR Coolers

High temperature of the reactor coolant discharging from the coolers is alarmed on the plant computer to signal a loss of cooling capability in the respective cooler.

RCS Level Low

RCS level low is alarmed during the DHR mode of operation to warn the operator of a loss of primary water which is indicative of a degradation of loss of decay heat removal capacity from the core.

CET Temperature High

The average temperature of the connected CETs during the DHR mode of operation is alarmed on high temperature to warn the operator of an unacceptable increase in CET temperature which is indicative of a degradation or loss of decay heat removal capacity from the core.

Decay Heat Vortex Warning

High/low conditions of the decay heat pumps motor current is input to a decay heat vortex warning alarm. This alarm is provided to warn the operator of potential vortexing conditions that could result in a loss of suction to the pumps.

9.6 FUEL HANDLING SYSTEM

9.6.1 DESIGN BASES

9.6.1.1 General System Function

The Fuel Handling System is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release. The ventilation filtration system meets the intent of Safety Guide 13. The fuel handling ventilation system operates continuously when fuel is being handled in the fuel handling area. The reactor building purge system must be either operable or isolated when fuel is being handled inside containment.

The reactor is refueled with equipment designed to handle the spent fuel assemblies under water from the time they leave the reactor vessel until they are placed in a cask for shipment from the pool. Underwater transfer of spent fuel assemblies provides an effective, economical, transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Borated water ensures subcritical conditions during refueling.

9.6.1.2 New Fuel Storage Area

The new fuel storage area is a separate and protected area for the dry storage of new fuel assemblies in the fuel storage and handling area. New fuel storage consists of a nine by eight array with a 21-inch pitch in both directions. Depending on the maximum fuel enrichment, eight or ten interior storage cells, as shown in Figures 9-56a and 9-56b, are precluded from use and will be physically blocked prior to any storage in the new fuel rack.

The new fuel storage racks are structural frames consisting of beams, columns and diagonal bracing acting as a unit to provide both vertical support at the bottom of the fuel element and lateral support at the top and bottom of the element.

A. The racks are constructed of aluminum.

B. The racks are designed for gravity loads from the racks and fuel elements as well as the Design Basis Earthquake (DBE).

C. The racks are designed in accordance with ASCE Paper 3341.

D. The new fuel rack vault is located in the auxiliary building. The floor of the vault is at elevation 386 feet, well above the flood elevation of 361 feet. The top of the vault is open to the operating floor at elevation 404 feet. The vault is a Seismic Category 1 structure and is separated from all other equipment to minimize the effect of a failure of this equipment on the new fuel storage area.

9.6.1.3 Spent Fuel Storage Pool

The spent fuel storage pool, located in the fuel storage and handling area, is a Seismic Category 1 reinforced concrete structure lined with stainless steel. It is provided with a makeup system design which meets the intent of Safety Guide 13. That is, the Borated Water Storage

Tank (BWST) is a Seismic Category 1 vessel; all connecting piping is located in a Seismic Category 1 structure; a backup system for supplying water to the pool is provided through a temporary connection to the Seismic Category 1 service water system; and, there is sufficient time to rig a temporary makeup water supply to the pool in the event of failure of the normal source.

The racks are arranged in 44 rows of 22 elements for an actual capacity of 968 spent fuel assemblies. This allows the concurrent storage of a full core of irradiated fuel assemblies (177) and twelve batches of spent fuel assemblies. The spent fuel assemblies are stored in racks in parallel rows having a center-to-center distance of 10.65 inches in both horizontal directions. Control Rod Assemblies (CRAs) requiring removal from the reactor are stored in the spent fuel assemblies.

Non-fuel irradiated components are stored in the Spent Fuel Pool for radioactive decay until such time as they can be shipped offsite. An inventory of these items is maintained.

Adjacent to the spent fuel pool are two smaller pools. One pool contains the fuel transfer mechanism, the other pool is a loading area for a spent fuel shipping cask. These two pools are connected to the main pool by short water channels and may be isolated from the spent fuel pool by water tight gates. These smaller pools may then be pumped down to allow dry handling of the spent fuel shipping cask or maintenance of the fuel transfer mechanism.

Makeup water for the spent fuel pool can be supplied from the Seismic Category 1 BWST. The flow path of the water is through the connecting piping to the spent fuel pool circulating pumps suction line and into the spent fuel pool. This flow path is shown in Figure 9-47.

However, if the BWST is not available, a temporary hose connection is provided on the Seismic Category 1 service water system piping near the spent fuel pool. Using this connection, a hose can be temporarily installed to transfer service water to the spent fuel pool for makeup. The service water system can be supplied from either the Dardanelle Reservoir or the emergency cooling pond.

It is considered that the spent fuel pool structural design meets the intent of Safety Guide 13:

A. The steel frame of the superstructure is designed to ensure that it does not collapse or distort so as to allow the bridge crane to fall.

B. The fuel pool walls are inherently resistant to missiles.

C. The spent fuel pool is flanked by the reactor and auxiliary buildings which would prevent the entry of the predominantly horizontal missiles generated by cyclonic winds.

9.6.1.3.A <u>VSC-24 Dry Fuel Storage</u>

The Independent Spent Fuel Storage Installation (ISFSI) is a passive, dry cask storage system consisting of a concrete cask storage pad located within a separate fence inside the site controlled area, portions of the site rail system for movement of components between the pad and the plant train bay, and the support equipment to lift and transport the cask from the spent fuel pool to its assigned storage location. The ISFSI is expandable to meet future site storage requirements. One dry cask storage system utilized is Sierra Nuclear Corporation's Ventilated Storage Cask (VSC) system. The VSC is comprised of a metal basket, the Multi-assembly Sealed Basket (MSB) that contains the fuel, a transfer cask, the MSB Transfer Cask (MTC), and

a concrete outer shell, the Ventilated Concrete Cask (VCC). These components are Safety Related as described in the NRC approved SAR and Certificate of Compliance (C of C) for Dry Spent Fuel Storage Casks under 10CFR72.214. Since the dry fuel storage cask system has been independently reviewed and approved for use by the NRC, apart from the reactor licenses, the full description of the system is in other documents. These include the VSC SAR, the VSC C of C, the ANO site specific VSC Specifications and drawings, the NRC Safety Evaluation Report (SER), and ANO Engineering Report 95-R-0015-01. Equipment used with the VSC system include non-safety related impact limiters that are positioned at the bottom of the Cask Loading Pit (CLP), safety related impact limiters that attach to the bottom of the MTC, a work platform that holds the MTC over the CLP for welding of the MSB following fuel loading, a MTC dolly, a VCC rail car and an air pallet system that is used to move the VCC from the storage pad to the VCC rail car and back with a loaded MSB. The MTC dolly is used to transport the MTC to and from the Auxiliary Building to the Turbine Building in the train bay.

The ANO ISFSI VSC-24 system cask storage pad is not seismically qualified, but has been analyzed and designed to support the postulated worst case static casks loading conditions. The storage pad is a two (2) foot thick reinforced concrete slab with reinforced 8" deep by 2'-0" wide haunches around the perimeter of the pad. The pad is 45'-0" wide by 135′-0" long. The pad is located within the ANO security fence approximately 500 feet east of the Auxiliary Building. The pad area was undercut four (4) feet and filled with compacted low plasticity fill material to significantly reduce any potential differential movement of the pad due to plasticity characteristics of the soil. The design bases earthquake for the site is specified at ground level (elevation 354'), and the top of the pad is at 356′8″, therefore amplification of the seismic loading on the cask by the pad is not significant.

9.6.1.3.B Holtec Dry Fuel Storage

The other dry fuel storage system utilized is Holtec International's HI-STORM system. The HI-STORM system is comprised of a metal basket, the Multi-Purpose Canister (MPC) that contains the fuel, an on-site transfer cask (HI-TRAC 125D), and a concrete and steel storage container (HI-STORM 100S). These components are classified as important to safety (ITS) as described in the NRC approved SAR and Certificate of Compliance (CoC) for the Holtec dry fuel storage cask system licensed under 10 CFR 72.214. These components are treated as safety related (Q) at ANO. Since the dry fuel storage cask system has been independently reviewed and approved for use by the NRC apart from the site reactor licenses, the full description of the system is in other documents. These documents include the Holtec CoC, Holtec SAR, NRC Safety Evaluation Report (SER) and ANO Engineering Request ER-ANO-2000-3333-010.

Ancillary equipment used with the Holtec system include: 1) a work platform that holds the HI-TRAC over the spent fuel pool cask loading pit for welding of the MPC closure lids following fuel loading, 2) a HI-TRAC Yoke for lifting and moving the HI-TRAC, 3) MPC lifting cleats for lowering the MPC into the HI-STORM, and 4) an on-site cask transporter (split railcar), an air pallet system that is used to move the HI-STORM assembly on the cask storage pad and the ISFSI cask storage pad.

The ANO ISFSI cask storage pad for the Holtec system, which is classified as not important to safety (non-Q), was seismically designed and qualified to support the worst case casks loading conditions. The storage pad is an approximately 3 feet thick reinforced concrete slab. One section is 55 feet wide by 190 feet long and a second section is 41 feet wide by 95 feet long. The pad is located north of the VSC-24 ISFSI pad and west of the eastern most security fence and bounded by railroad tracks on the west.

An ISFSI Facility was installed at the north-end of the ANO Site, immediately north of the existing Low Level Radwaste Building. The ISFSI Facility includes (4) passive reinforced Concrete Storage Pads, which will provide (98) storage locations for the fully-loaded Holtec HISTORM 100S(C) Overpacks. The storage pads are classified as "Safety-Related" and "Seismic I" structures.

The ISFSI Facility includes (2) passive reinforced Concrete Turning Pads with an embedded Rail Spur, which are located east of the Concrete Storage Pads in the vicinity of the existing Rail Haul Path. The east turning pad is classified as a "Non-Safety-Related" and "Non-Seismic" structure, whereas, the west turning pad is classified as a "Safety-Related" and "Non-Seismic" structure. The turning pads will facilitate 1) unloading of the fully-loaded HI-TRAC/Multi-Purpose Canister (MPC) from the Holtec Railcar, 2) transfer of the fully-loaded HI-TRAC/MPC to the Cask Transfer Facility (CTF) for placement of the MPC within the HI-STORM Overpack and 3) transfer of the fully-loaded HI-STORM Overpack from the CTF to the storage pad.

9.6.1.3.C Cask Transfer Facility

The ANO Cask Transfer Facility (CTF) Component Number DFS-CTF-1 is a passive reinforced concrete structure, which will support the stack-up configuration of a fully-loaded HI-TRAC, mating device, and HI-STORM during MPC transfer operations from the HI-TRAC into the HI-STORM. In addition, the CTF may be used to load an empty MPC into an empty HI-TRAC. The CTF is basically a 12'-8" x 12'-8" by approximately 14 feet deep cavity in the ground. The cavity is constructed from 4,500 psi at 28 days concrete. Lateral seismic restraints are installed in the cavity to assist in energy absorption and transferring loads to the surrounding soil. The CTF also has a concrete apron around the CTF cavity structure.

A Vertical Cask Transporter (VCT) is used to carry the HI-STORM overpack and straddle the CTF while lowering the overpack into the CTF cavity as well as carry the HI-STORM overpack to the new or existing ISFSI Pads. The Low Profile Transporter (LPT) is used to carry the loaded HI-TRAC from Auxiliary Train Bay to the loading dock, or the turning pads (TP-PAD-1/2) from where the VCT will carry the HI-TRAC to the CTF and position it for stack-up. When transitioning the VCT across the approach slab to the top slab of the CTF, the site may choose to place down a 3/4” thick plate. The VCT is capable of traversing over this plate.

The ANO CTF (DFS-CTF-1) supports the stack-up configuration of a fully-loaded HI-TRAC, mating device, and HI-STORM during MPC transfer operations. The CTF consists of a below grade passive reinforced concrete structure, (4) removable upper lateral restraints, and (4) removable lower lateral restraints. The lateral restraints (wedges) are used to secure the HI-STORM in position and transfer lateral loads, which may result from a seismic event into the surrounding soil.

The CTF also includes a removable handrail system, which was installed around the perimeter of the CTF Cavity for personnel protection. The CTF includes a removable work platform, which provides access for personnel to facilitate the stack-up configuration of a fully-loaded HI-TRAC, mating device, and HI-STORM during MPC transfer operations. In addition, the CTF includes a personnel access ladder, which facilitates installation of the lower lateral restraints. The CTF is protected from inclement weather by the Non-Safety Related CTF Building which consists of a metal building and concrete shallow foundation. The CTF Building envelopes and encloses the stack-up configuration of a fully-loaded HI-TRAC, mating device, and HI-STORM during MPC transfer operations.

The CTF has been designed to accommodate both the HI-STORM 100S, Version C, and HI-STORM 100S (232), which are currently in use at ANO. The CTF will accommodate the combined loads of a fully-loaded HI-STORM 100S, Version C with high density concrete shielding during both placement and retrieval.

9.6.1.4 Fuel Transfer Tube

A horizontal tube is provided to convey fuel between the reactor building and the fuel storage pool. This tube contains tracks for the fuel transfer carriage, a gate valve on the spent fuel storage pool side, and a flanged closure on the reactor building side. The fuel transfer tube penetrates the fuel transfer canal near the bottom of the pool, where space is provided for upending the fuel transfer carriage basket containing a fuel assembly.

9.6.1.5 Fuel Transfer Canal

The fuel transfer canal is a passageway in the reactor building extending eastward from the reactor vessel to near the reactor building wall. It is formed by an upward extension of the primary shield walls. The enclosure is a reinforced concrete structure lined with stainless steel. It forms a canal above the reactor vessel, which is filled with borated water for refueling.

Space is available in the fuel transfer canal for underwater storage of the reactor vessel internals plenum assembly.

A deeper fuel transfer station portion of the fuel transfer canal can be used for storage of the reactor vessel internals core support assembly. A fuel rack accommodating six fuel assemblies and a failed fuel detection container is also located at the deep end of the canal.

9.6.1.6 Miscellaneous Fuel Handling Equipment

The miscellaneous fuel handling equipment consists of fuel handling bridges, fuel handling tools, new fuel storage racks, spent fuel storage racks, new fuel elevator, control rod handling tools, viewing equipment, a fuel transfer mechanism and shipping casks. In addition to the equipment directly associated with the handling of fuel, equipment is provided to handle the reactor closure head and the plenum assembly to expose the core for refueling. All the reactor internals can be removed as necessary using the available equipment. Handling equipment used in conjunction with the reactor building and fuel handling cranes meets ASTM material specifications.

Figure 9-48 shows a trimetric view of a typical refueling system.

The following refueling components are seismically qualified:

Main Fuel Handling Bridge (in the parked and locked position)

Auxiliary Fuel Handling Bridge (in the parked and locked position)

Fuel Storage Handling Bridge (in the parked and locked position)

Reactor Building Polar Crane (L-2) is qualified for the following conditions:

1. Seismic Class I when unloaded in its parked position.

2. Seismic Class I when unloaded with the trolley and bridge parked in any position. It is also Seismic Class I with a load of up to 10,000 lbs on the main or auxiliary hook; however, during modes other than 5, 6, and defueled, ANO's commitments to NUREG-0612 shall be met and load drop analyses must be performed by Design Engineering.

3. Seismic Class II/I with loads up to 190 tons in any position in Modes 5, 6, and defueled; however, ANO's commitments to NUREG-0612 shall be met.

Fuel Storage Building Crane

New and Spent Fuel Storage Racks

Fuel Transfer Tube

The following major tools and service equipment are used in refueling and servicing the reactor:

Main Fuel Handling Bridge

The main fuel handling bridge, located in the reactor building, operates between the reactor core and the fuel transfer station (or fuel storage racks and the failed fuel detection can in the deep portion of the fuel transfer canal near the transfer station). One mast assembly is provided for fuel assembly handling, and one mast assembly is provided for control component (control rods, orifice rods, axial power shaping rods and burnable poison rods) handling. EC-84159 disabled the control component mast controls. All control component handling is performed using the Control Rod Assembly Handling Tool in the SFP. Provision is made for installing a closed-circuit television system between the fuel handling mast and control component handling mast on the trolley of the main bridge.

A self-contained closed circuit television system for underwater viewing is installed on the main fuel handling bridge, although it can be installed on the auxiliary or the fuel storage bridge. The system includes a radiation resistant camera in a stainless steel waterproof container, high resolution monitor, telescoping stainless steel mast, motorized mast rotation (approximately 300 degrees), control box with pendant switches, all necessary cable and cable reels, remote optical focus, integral lighting, interlock of telescoping mast with bridge and trolley (with override switch), and other items necessary for submerged operation. The camera can be tilted 37 degrees from the vertical centerline to permit reading the control component identification numbers and to view internals vent valves.

Auxiliary Fuel Handling Bridge

The auxiliary fuel handling bridge is no longer capable of moving fuel assemblies and has been converted into a work platform. The auxiliary bridge may be positioned along the refueling canal, either by manual means or driven by an electric motor, to serve as a platform from which to conduct various refueling outage activities.

Fuel Storage Handling Bridge

The fuel storage handling bridge, located in the fuel storage building, operates over the fuel storage pool and is used to move fuel assemblies from the new fuel elevator to the fuel storage racks; from the storage racks to the transfer system basket; from the transfer system basket to the storage racks; and from the storage racks to the spent fuel shipping cask.

Closed circuit television may be installed on the fuel storage handling bridge.

Fuel Transfer System

The fuel transfer system is used to transport fuel assemblies and control components between the fuel storage building and the reactor building. The complete fuel transfer system consists of two rotating frames, one carriage and fuel basket, a fuel transfer tube with a gate valve in the fuel storage building and a manually installed and removed gasketed cover plate in the reactor building, and control consoles in the reactor building and fuel storage building.

Fuel Transfer Tube and Gate Valves

The fuel transfer tube serves as a passageway between the reactor building and the fuel storage building during refueling. The portion of the fuel transfer tube inside the reactor building is an integral part of the reactor containment during reactor operation.

A tube support in the fuel storage building supports the transfer tube and gate valve which is bolted to the flanged end of the tube. The support is designed to accommodate transfer tube thermal expansion and seismic loads. The manually operated gate valve is closed during reactor operation.

The fuel transfer carriage assembly moves through the transfer tube on rails welded to the inside of the transfer tube.

The means for driving the fuel transfer carriage is provided by a cable & sheave system powered by electric winches. The cable and sheave system utilizes two winches, one permanently mounted in the reactor building and one permanently mounted in the auxiliary building. Each winch attaches a cable to either end of the fuel transfer carriage and pulls the fuel transfer carriage towards its intended location. Upon retrieval, the fuel transfer carriage is totally in the fuel tilt pit of the auxiliary building. The reactor building winch cable is disconnected and fully retracted into the reactor building so no additional storage preparation is needed prior to the next refueling outage. An electrical interlock prevents the fuel transfer carriage from being moved until the gate valve is fully open.

Internals Storage Stand

The internals storage stand is a stainless steel weldment designed to support the upper plenum assembly for temporary storage during each refueling period and to support the core support and upper plenum assembly for temporary storage during reactor vessel internal surface inspection.

Head Storage Stand

The head storage stand is a carbon steel weldment designed to support the reactor vessel closure head assembly for temporary storage and maintenance during each refueling period.

Internals Indexing Fixture

The internals indexing fixture is an aluminum weldment designed to radially orient the plenum assembly and the core support assembly for insertion into and removal from the reactor vessel.

Stud Tensioners

The stud tensioner is a hydraulically operated jacking device which elongates and preloads the reactor closure head studs at cold shutdown conditions to permit tightening or loosening of the stud nuts without the use of torque wrenches or bolt heaters. Relief valves are provided on each stud tensioner to prevent over-tensioning of the studs due to excessive hydrostatic pressure.

Stud Handling Tool

The stud handling tool is an assembly consisting of an air motor, motor reducer assembly, motor and spring support assembly, drive screw, stud hook and transfer yoke, and stud balance springs.

This tool removes the 650-pound closure studs from the reactor vessel flange tapped holes after the stud nuts are loosened and re-inserts the studs into the tapped holes after the closure head assembly has been set onto the reactor vessel.

Stud Tensioner & Tooling Hoists

Three stud tensioner hoists are mounted on a circular monorail attached to the Control Rod Drive Service Structure for supporting and positioning the stud tensioners over each of the sixty (60) closure head studs. The stud tensioner hoists are also used for supporting the stud handing tools which remove the stud(s) from the Reactor Vessel Flange tapped holes.

Head and Internals Handling Fixture

The head and internals handling fixture is a carbon steel weldment designed for handling the closure head assembly (reactor vessel closure head, closure head service structure, and control rod drives) and the reactor vessel internals (internals indexing fixture and internals handling adapter which are used for insertion and removal of the plenum assembly and core support assembly).

Tripod Storage Stand (Handling Fixture Storage)

The tripod storage stand (Figure 9-58) is a carbon steel structure designed for supporting and providing a storage area for the tripod (handling fixture), internals indexing fixture, and internals handling adapters.

Head Lifting Device

The head lifting device is a carbon steel structure with solid fixed lift pendants designed for handling the closure head assembly (reactor vessel head, closure service structure, and control rod drives).

Internals Handling Adapter

The internals handling adapter is an assembly designed for handling the two reactor vessel internals assemblies--the core barrel assembly during initial insertion into the reactor vessel, the plenum assembly for refueling, and the combined plenum and core barrel assembly for periodic reactor vessel inspection. This adapter is used in conjunction with the head and internals

handling fixture, the internals handling extension and the internals indexing fixture for internals handling. Additionally, this adapter is used in conjunction with the head handling fixture for handling of the closure head assembly.

Internals Handling Extension

The internals handling extension is a carbon steel assembly designed to connect the head and internals handling fixture to the main hook of the reactor building crane.

Failed Fuel Container

The failed fuel container is an aluminum receptacle designed primarily for storage in the spent fuel storage pool of a gross failed fuel assembly to contain loose pieces.

Rod Assembly Handling Container

The rod assembly handling container is a stainless steel component having the same external physical size and shape as a fuel assembly and consists of a modified fuel assembly lower end fitting and upper end fitting connected by four corner angles. Sixteen guide tubes are positioned between the end fittings to provide guidance for the rod assembly rods. It accepts an orifice rod, control rod, Burnable Poison Rod or Axial Power Shaping Rod Assembly for temporary storage or for handling and transfer between the reactor building and the fuel storage building.

Long-Handled Tools

The long-handled tools consist of a segmented handle to which can be attached various types and shapes of tool heads. The square aluminum handle consists of a main section with lifting strap for attachment to a hoist and an adapter section connected to the main section by two ball-lok pins. The removable stainless steel tool heads supplied with the handle include a hook tool, a guide tool, and several sizes of socket or wrench tools. An extension section can be added between the main and adapter handle sections to produce a tool to permit reaching to the top of the core or to the bottom of the transfer canal or the storage pool.

Internals Vent Valve Exercise Tool

The internals vent valve exercise tool is a long stainless steel rod with a lifting eye at one end and a hook at the other end. This tool is used to ascertain that the internal vent valve disc is free. This is accomplished by manually engaging the hook into the bracket provided on the back of the valve disc and lifting up on the tool.

Internals Vent Valve Handling Tool

The internals vent valve handling tool is a tool used for removal and replacement of an internal vent valve assembly if this should ever be necessary.

New Fuel Handling Tool

The new fuel handling tool is a manually-operated grapple used primarily for handling of new fuel assemblies. This tool is required for removing and transferring new fuel assemblies from the shipping containers to the new fuel storage racks, the new fuel transfer container or the handling elevator for insertion into the spent fuel pool.

New Control Rod Assembly Handling Tool

The new control rod assembly handling tool is a manually-operated grapple used primarily for handling of new control and orifice rod assemblies. It can also be used for handling axial power shaping rod assemblies or burnable poison rod assemblies. The tool is required for removing and transferring new rod assemblies from their shipping containers to new fuel assemblies or into the "rod assembly handling container" for transfer into the storage pool.

Control Rod Assembly Handling Tool

The control rod assembly handling tool is a manually-operated grapple used primarily for handling of irradiated control rod assemblies. It can also be used for handling axial power shaping rod assemblies or burnable poison rod assemblies. The tool is required for removing and transferring control rod assemblies under water in the SFP.

New Fuel Elevator

The new fuel elevator is an electrically operated elevator with a manual backup and it is used to lower new fuel elements one at a time into the fuel pool. Once in the lowered position the fuel storage handling bridge moves into position over the new fuel element and removes it from the elevator.

New and Spent Fuel Racks

The new fuel racks are for dry storage of the new fuel assemblies and are of aluminum construction. New fuel is transferred from the new fuel racks to the spent fuel pool by the fuel handling crane.

The spent fuel racks are of stainless steel construction and are for underwater storage of spent fuel elements. New fuel elements may be temporarily stored in the spent fuel racks prior to transfer to the reactor building.

Reactor Building Polar Crane

The reactor building polar crane is an electric, circular traveling bridge crane with a single trolley. It consists of a main hoist used to lift the reactor vessel closure head, the reactor upper plenum assembly, and the reactor core barrel assembly, and an auxiliary hoist used to perform lifts of refueling equipment. The reactor building polar crane (L-2) normally operates during routine refueling shutdown and is also qualified to operate with loads of up to 10,000 lbs on the main or auxiliary hook. However, during modes other than 5, 6, and defueled, the ANO commitments to NUREG-0612 shall also be met.

Fuel Handling Crane

The fuel handling crane is an electric overhead traveling bridge crane with a single trolley. The crane consists of three hoists, one single failure proof main hoist to handle the spent fuel shipping cask, one non-single failure proof hoist to handle new fuel elements, and a small non-single failure proof hoist for miscellaneous lifts. The crane is used during reactor refueling operations and at various times throughout the year. The crane is located in the auxiliary building over the fuel pool.

New Fuel Handling Crane

New fuel can also be handled with the 2L35 New Fuel Handling Crane.

9.6.1.7 Reactor Building and Fuel Handling Crane

The reactor building crane is designed and built in accordance with the minimum requirements for Class C cranes of the Electric Overhead Crane Institute (EOCI) Specification 61.

The fuel handling crane, L-3, is designed in accordance with the minimum requirements of the Crane Manufacturers Association of America (CMAA Specification #70), ASME NOG-1 (Rules for Construction of Overhead and Gantry Cranes) and NUREG-0554 for single failure proof cranes. Table 9.1-30 provides site specific information regarding compliance with Ederer's Generic Topical Report for single failure proof cranes.

Original welding on the cranes was done in accordance with American Welding Society (AWS) D2.0, Specification for Welded Highway and Railroad Bridges. AWS D2.0 is no longer an active welding code and has been superseded by AWS D1.1, Structural Welding Code – Steel. New welding on the reactor building polar crane will be to AWS D1.1. As an alternative to AWS requirements, Welding Procedure Specifications (WPS's) and/or Welder Performance Qualification (WPQ), meeting the ASME Section IX Code requirements may be utilized for the reactor building polar crane applications provided all other applicable provisions of AWS D1.1 are met unrelated to WPS's and WPQ's.

In addition to EOCI 61, CMAA 70, NOG-1, and AWS D2.0 the units and accessories are designed and constructed in accordance with sections of applicable standards including: ASTM, AISE, AGMA, AFBMA, IEEE, NEMA, NEC, NFPA, USASI, and any laws and regulations of the State of Arkansas or other local regulatory bodies having jurisdiction over such apparatus.

The reactor building containment crane (rated load capacity = 190 tons) is equipped with dual electric and Magnetorque regenerative braking. The electric brakes are mounted on the hoist motor shaft and are automatically applied in the event of power interruption. In the event of a mechanical failure the Magnetorque regenerative braking is automatically applied.

The fuel handling crane (rated load capacity = 130 tons) hoist is equipped with an emergency drum brake system as designed by Ederer Cranes (X-SAM®Hoists). This system provides an independent means for reliably and safely stopping and holding the load following a failure in the hoist machinery between the high speed shafting and the hoist drum. The brake pads are released by an externally supplied force and are spring set, to provide fail safe operation, i.e., the emergency drum brake system sets, controlling load if all power is removed from the hoist and the load starts to lower. This system is normally not set during normal duty cycles.

9.6.1.7.1 Control of Heavy Loads Requirements

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" contains the NRC guidance to ensure that load handling systems are designed and operated such that their probability of failure is low and appropriate for the critical tasks in which they are employed. There are eleven (11) cranes installed in ANO-1 that fall under the guidelines of NUREG-0612. They are (1) the Reactor Building Polar Crane (Equipment No. L2); (2) the Fuel Handling Crane (L3); (3) the Auxiliary Fuel Handling Crane (a 2-ton monorail attached to L3); (4) the Intake Structure Gantry

Crane (L7); (5 - 8) four (4) Reactor Coolant Pump Maintenance Jib Cranes (L32, L32B, L32C, L32D); the Reactor Building General Maintenance Crane (L37); and the Turbine Building Cranes (L-1 and 2L-1) when used to lift loaded spent fuel casks. ANO-1 has implemented the NUREG-0612 Phase 1 guidance for these load handling systems. In a Safety Evaluation issued in October 1984 (0CNA108406), the NRC concluded that the Phase I requirements of NUREG-0612 had been satisfied for ANO-1 for the Reactor Building Polar Crane, the Fuel Handling and Auxiliary Fuel Handling Cranes, and the Intake Structure Gantry Crane. In a June 26, 1985 generic letter (Generic Letter 85-11) (0CNA068520), the NRC further concluded that the NUREG-0612 Phase II effort was considered complete. Further evaluation was performed pursuant to NEI 08-05, Industry Initiative on Control of Heavy Loads, which verified that the Unit 1 Reactor Vessel Closure Head Load Drop Analysis demonstrates allowable stress limits per ASME Section III, Appendix F.

The four Reactor Coolant Pump Maintenance Jib Cranes are installed on top of the Steam Generator Cavities and are used to service the Reactor Coolant Pumps, Steam Generators, and Pressurizer. These cranes are 2-ton capacity wire rope hoists which are designed, operated and maintained in accordance with ANSI/ASME B30.11c-1992 and B30.16c-1992. The Reactor Building General Maintenance Crane is a 10-ton capacity, electromechanical boom crane which is designed in accordance with the requirements of API-2c, 4th Ed. and the applicable sections of ANSI/ASME B30.5b-1992. These cranes are seismically designed in their unloaded condition. The jib crane hoists are removed and stored outside the Reactor Building during power operations and the jib crane booms remain installed and are secured to withstand an SSE during power operations. The boom crane is stored in a secure position inside the Reactor Building during power operations with its boom secured to a boom storage structure.

The reactor vessel head and internals lifting tripod is classified as a special lifting device in accordance with NUREG-0612, which requires special lifting devices to meet ANSI N14.6-1978. ANSI N14.6-1978 specifies that minimum yield strength be used for meeting a factor of 3.0 to yield and 5.0 to ultimate. The lifting tripod was evaluated and is acceptable using a reduced safety factor of 2.78 to yield.

ANSI N14.6 specifies either a Magnetic Particle or a Dye Penetrant Test as part of the inspections for this tripod. These NDE methods have been revised to include Acoustic Emission (AE). ANSI N14.6 specifies the NDE to be performed prior to use. The AE monitoring method begins as the load is applied to the tripod. The recorded ultrasonic sound provides a warning of an existing flaw to allow the safe removal of the load from the tripod prior to failure. The application of the AE methodology has been determined to be acceptable in accordance with the requirements of 10 CFR 50.59.

9.6.2 SYSTEM DESCRIPTION AND EVALUATION

9.6.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers and then transferred to dry storage racks or spent fuel storage racks which can then be transferred to the reactor building through the transfer canal. This is accomplished by lifting a new fuel assembly from the new fuel racks and placing it in the new fuel elevator which is attached to the side of the spent fuel pool. The elevator is in its highest position, such that the top of the fuel extends out of the water. When the elevator receives a new fuel assembly, it is lowered to the bottom of the pool where the fuel storage handling bridge can lift it from the container and place it in the spent fuel storage racks.

9.6.2.2 Loading and Removing Fuel

Following the reactor shutdown and reactor building entry, the refueling procedure is begun by removing the control rod drive mechanisms and reactor closure head. Head removal and replacement time is minimized by the use of up to three stud tensioners simultaneously. A stud tensioner is a hydraulically operated device that permits preloading and unloading of the reactor closure studs at cold shutdown conditions. The studs are tensioned to their preoperational load in two steps in a predetermined sequence. Required stud elongation after tensioning is verified by a micrometer or other comparable measuring device.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in the closure head bolt holes by specially designed spacers. Removal of the studs with the reactor closure head minimizes handling time and reduces the chance of thread damage. However, the lift weight and height of the reactor vessel head and attached components are controlled by procedures due to NUREG 0612 load lift analyses and crane capacity limitations.

The annular space between the reactor vessel flange and the bottom of the fuel transfer canal is sealed off prior to removal of the reactor closure head by placement of six gasketed seal plate access and ventilation covers. These covers are bolted to the permanent seal plate which is welded to the shield ring embedded in the annulus shield wall and to the reactor vessel flange seal ledge.

The reactor closure head assembly is handled by a lifting fixture supported from the reactor building crane. The head is lifted out of the canal onto a storage stand located on the operating floor. The stand is designed to protect the gasket surface of the closure head. The lift is guided by two closure head alignment guide pins installed in two of the stud holes. These pins also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling. The studs and nuts can be removed from the reactor closure head at the storage location for inspection and cleaning using special stud and nut handling fixtures. A stud and alignment pin storage rack is provided.

Once the reactor vessel head is removed, the reactor building crane is only used to insert and remove the reactor internals indexing fixture and to remove and insert the reactor upper plenum assembly. The internals handling extension and the internals handling fixture are designed so that all pins, bolts, nuts, and removable parts are safety locked to prevent them from dropping into the reactor vessel.

The maximum drop height of the reactor vessel head over an opened reactor vessel is determined in the heavy load lift analysis for the L2 polar crane. Upon clearing the reactor vessel guide pins with an acceptable margin, the reactor vessel head is moved laterally along the canal and away from the opened reactor vessel. Once cleared of the reactor vessel to the extent practical, it is then lifted vertically to its storage area.

The lifting rig is connected to three separate points on the reactor vessel head as shown in Figure 9-49. During or immediately after the initial lift the unlikely event of a failure of the lifting equipment at any one point could result in impact damage to the reactor vessel, the reactor vessel head seal surface, the alignment studs, the core support shield, or the outlet nozzle seal surfaces. Depending on the angle and the position of the reactor vessel closure head upon impact, it is possible that damage to the control rod guide tubes and the plenum assembly, particularly the plenum cover, could be incurred. Impact loads on the control rod drive guide

tubes and the plenum assembly could then be transmitted to the fuel assemblies. The compressive loads on the fuel bundles could conceivably be severe enough to cause fuel rod distortion and bending.

The upper core barrel assembly is taken to refer to the plenum assembly, which is removed during refueling operations. Using a special lifting fixture, the plenum assembly is removed from the reactor by the reactor building crane and stored under water on a stand on the fuel transfer canal floor.

The maximum lift required before the plenum assembly is moved laterally to its designated storage area is approximately 14 feet. In the unlikely event that the plenum assembly lifting fixture shown in Figure 9-50, or the lifting lugs should fail with the plenum assembly completely removed from the vessel, the water shield in the reactor vessel slows its descent and substantially decreases the impact energy transmitted to the core support shield and the fuel assemblies.

Based on an additional B&W analysis assuming a 16-foot drop in air, it was calculated that the stress in the support skirt resulting from a plenum drop in air (35,115 psi) is less than that calculated for the reactor vessel head drop. Based on this evaluation, there should be no restriction on lifting the plenum without flooding the canal.

For the plenum assembly to fall and impact the core barrel and fuel assemblies after it is lifted clear of the reactor vessel would require that the lifting rig fail completely. If a failure should occur at any one of the three lifting connection points on the plenum assembly, it is highly improbable that any serious damage would result to the fuel assemblies. Damage may result to the vessel seal surfaces or to the guide studs.

Depending on the angle and position of the plenum assembly at impact, damage could result to the core support shield. If the plenum assembly, in its fall, reached the top of the core barrel, damage could result to the core barrel and the fuel assemblies. It is possible that an impact load on the fuel assemblies could cause fuel rod damage.

Both the reactor vessel closure head and internals handling fixture and the internals handling adapter are based on design standards such as the American Institute of Steel Construction Manual - Part 5, specification and codes.

Design loads are approximately 5.0 times maximum operating loads, and test loads are approximately 1-1/2 times maximum operating loads. The operating loads are equal to the weight of the internals minus the core, all in a dry condition. The design of each of these rigs precludes the possibility of either a closure head or internal plenum assembly drop.

Because the CRAs, except for those being transferred, are normally fully inserted in the core during handling operations, and refueling boron concentration is sufficient to maintain the reactor subcritical if all CRAs were withdrawn there is no possibility of a criticality accident.

The removable missile shielding consists of four similar concrete segments, which measure 31' x 7' x 2.5'. At a density of 150 lb/ft^3, each segment weighs approximately 82,000 pounds. The normal elevation of the segments is 424 feet, 6 inches, while the top of the service structure is at approximately 401 feet, 6 inches elevation. A possible 23-foot drop could result.

The reactor closure head cannot be lifted before the missile shielding is removed. In the event that a shielding segment should be dropped during its removal, the type of damage resulting from the collision would depend greatly on the orientation of the shielding during impact and the path and manner of its fall. It is conceivable that a shielding segment could damage the service structure and that the resulting deformation could damage the Control Rod drive Mechanisms located inside the service structure cylinder.

This kind of occurrence could result in control rod and reactor internal component damage. However, because the reactor would be depressurized and the closure head would still be in place, only limited contamination could result. The fuel transfer canal is then filled with borated water (Figure 9-47).

The fuel transfer canal and the incore instrumentation tank are filled with borated water from the BWST. The Decay Heat Removal System and Spent Fuel Cooling System are used for the filling operation.

Once the Reactor Coolant System (RCS) is vented and filled to the reactor vessel flange level, filling of the transfer canal may begin. Water is drawn from the BWST by the decay heat removal pumps or the spent fuel pool circulating pump(s) and pumped into the transfer canal and incore instrumentation tank. When the refueling level is reached, the decay heat removal pump(s) or the spent fuel pump(s) are stopped.

Refueling operations are carried out from the main fuel handling bridge which spans the fuel transfer canal. The main bridge is used to shuttle spent fuel assemblies from the core to the transfer station and new fuel assemblies from the transfer station to the core. The main bridge may also be used to relocate partially spent fuel assemblies in the core as specified in the fuel management program or move fuel assemblies to or from the fuel storage racks.

Fuel assemblies are handled by a pneumatically operated fuel grapple attached to a telescoping and rotating mast which moves laterally with the main bridge. CRAs are handled by a Control Rod Assembly Handling Tool located in the SFP.

The main (2-mast) bridge moves a spent fuel assembly from the core under water to the transfer station where the fuel assembly is lowered into the fuel transfer basket on the tilt mechanism. After being tilted to a horizontal position, the basket with the spent fuel assembly is then transported to the spent fuel storage pool from the reactor building via a fuel transfer tube by means of a fuel transfer carriage. The spent fuel assemblies are removed from the fuel transfer basket using the fuel storage handling bridge. This motor-driven bridge spans the spent fuel storage pool and permits the refueling crew to store new fuel assemblies in any of the many vertical storage rack positions. New fuel assemblies are transferred to the reactor building by the reverse operation.

The fuel transfer mechanism is an underwater carriage that runs on tracks extending from the spent fuel storage pool through the transfer tube and into the reactor building. A rotating fuel basket is mounted on one end of the fuel transfer carriage to receive fuel assemblies in a vertical position. The hydraulically operated fuel basket on the end of the carriage is rotated to a horizontal position for passage through the transfer tube, and then rotated back to a vertical position in the spent fuel storage pool for vertical removal of the fuel assembly. The fuel transfer mechanism is designed to permit initiation of the carriage fuel basket rotation from the building in which the carriage fuel basket is being loaded or unloaded.

Control rods may be transferred from one fuel assembly to another using the following procedure in the SFP. Position the Control Rod Assembly Handling Tool (CRA Tool) on the fuel assembly containing a control rod to be moved. Use the CRA Tool to remove the control rod assembly. Reposition the CRA Tool over the fuel assembly intended to receive the control rod. Use the CRA Tool to insert the control rod assembly. This same procedure applies to BPRAs and APSRs.

Once refueling is completed, the fuel transfer canal is sealed from the spent fuel pool by the gate valve on the transfer tube. Most of the fuel transfer canal water is then drained by suction through a pipe located in the deep transfer station area. Either one or two Decay Heat Removal (DHR) pumps may be used to drain the canal depending upon the requirements for decay heat removal. If decay heat removal is required, the flow from one pump and cooler is directed back to the reactor, and the flow from the other pump and cooler is diverted to the BWST. If decay heat cooling is not required, the flow from both pumps and coolers is diverted to the BWST. This water is pumped to the BWST to be available for the next refueling or for emergency cooling following a LOCA. The water remaining in the fuel transfer canal (approximately eight inches) after this evolution can be drained to either the auxiliary building equipment drain tank or the reactor building sump by utilizing a temporary hose connection. The incore instrumentation tank bottoms can also be drained to either of these locations using a similar hose connection.

During operation of the reactor, the carriage is stored in the spent fuel storage pool, thus permitting the gate valve on the spent fuel storage pool side of the transfer tube to remain closed and a blind flange to be installed on the reactor building side of the tube.

The spent fuel storage pool has space for a spent fuel shipping cask, as well as for required fuel storage. Following a sufficient decay period, the spent fuel assemblies can be removed from storage and loaded into the spent fuel shipping cask under water for removal from the site or transfer to approved onsite storage.

A decontamination area is located in the building adjacent to the spent fuel storage pool. In this area the outside surfaces of the shipping casks can be decontaminated before shipment by using steam, water, or detergent solutions, and manual scrubbing to the extent required.

9.6.2.3 Safety Provisions

Safety provisions are designed into the Fuel Handling System to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations.

New and spent fuel assembly storage facilities in the Auxiliary Building maintain a safe geometric spacing of 21 inches and 10.65 inches, respectively, center-to-center spacing for stored assemblies. The new fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby ensuring the necessary spacing between assemblies. Spent fuel storage outside the Auxiliary Building will be in an approved array with safety provisions addressed for each type of specific application.

SFP storage racks are divided into three regions. Region 1 storage racks (220 storage cells) are composed of stainless steel walls separated by a gap with degraded Boraflex neutron absorber panels (attached by stainless steel sheathing). The steel walls define the storage cells and the stainless steel sheathing defines the boundary of the flux-trap water gap used to augment reactivity control. Region 2 storage racks (484 storage cells) are composed of stainless steel

walls with no neutron absorber panels. The stainless steel walls are formed in such a way as to create a water gap between adjacent cells used to augment reactivity control. Region 1 and 2 storage racks are qualified for storage of spent fuel assemblies with a maximum initial enrichment of 4.95 wt% ^{235}U that have accumulated a minimum burnup with credit for cooling times between 0 and 20 years. Placement of fuel in Regions 1 and 2 is determined by burnup calculations and controlled administratively. Additionally, the Region 1 and 2 racks are qualified for storage of fresh fuel assemblies, in a checkerboard pattern with empty storage cells, at a maximum enrichment of 4.95 wt% ^{235}U.

Region 3 storage racks (264 storage cells) are identical to the Region 2 storage racks, with the single exception of Metamic poison insert assemblies used to augment reactivity control being installed in the flux-trap water gaps between the storage cells. Region 3 storage racks are qualified for unrestricted storage of new unburned fuel assemblies with a maximum nominal enrichment of 4.35 wt% ^{235}U. Additionally, the Region 3 storage racks are qualified for storage of a "3 of 4" configuration of 3 fresh fuel assemblies with a maximum enrichment of 4.95 wt% ^{235}U and 1 spent fuel assembly with a maximum initial enrichment of 4.95 wt% ^{235}U that have accumulated a minimum specified burnup, which is controlled administratively.

The Region 1, 2 and 3 storage racks meet the requirements of the NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, dated April 14, 1978, and modified January 18, 1979, with the exception that credit is taken for fuel burnup, cooling time, and soluble boron in accordance with 10 CFR 50.68, proposed revision 2 to Regulatory Guide 1.13, and L.I. Kopp memorandum "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", dated August 19, 1998.

Rack module data for Regions 1, 2 and 3 are described in Table 9-29.

Region 1 Mechanical Design

The Region 1 storage racks are composed of individual storage cells made of type 304 stainless steel and conform to the requirements of ASME B&PV Code, Section III, Subsection NF. These racks contain a neutron absorbing material, Boraflex, which is attached to each cell. However, due to excessive degradation of the Boraflex material, no credit is taken for the neutron absorbing characteristics of Boraflex in the latest criticality analyses. The cells within a module are interconnected by grid assemblies to form an integral structure. Each rack module is provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The spent fuel storage rack assembly consists of three major sections which are the leveling pad assembly, the lower and upper grid assembly, and the cell assembly.

The major components of the leveling pad assembly are the leveling pad and the leveling pad screw. The top of the support plate is welded to the base plate. The leveling pad assemblies transmit the load to the pool floor and provide a sliding contact. The leveling pad screw permits the leveling adjustment of the rack.

The lower grid attaches the cell assembly to the base plate. The lower grid consists of box-beam members and the base plate. The bottom of the cell assembly is welded to the lower grid. The upper grid consists of the box-beam members. The upper part of the cell assembly is

welded to the upper grid. The upper and lower grid assemblies maintain the center-line to center-line spacing between the cells and provide the structural connections between the cells to form a fuel rack assembly.

The major components of the cell assembly are the fuel assembly cell, the degraded Boraflex material, and the wrapper.

The wrappers are attached to the outside of the cells by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of the Boraflex to the pool environment. Based on the original rack design requirements, some cells have wrappers on all four sides, some on three sides, and some on two sides.

Region 2 Mechanical Design

The Region 2 storage racks consist of type 304 stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure. Each cell has attached to its outer wall a stainless steel wrapper plate creating a pocket opened at the top and bottom. This is referred to as a "spacer pocket" design. The spacer pockets are designed to accept poison inserts if future need arises.

This design is also provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are neither anchored to the floor nor braced to the pool walls.

The spent fuel storage rack assembly consists of two major sections which are the base support assembly and the cell assembly.

The major components of the base support assembly are the leveling pad, the leveling pad screw, and the support plate. The top of the support plate is welded to the fuel rack base plate. The leveling pads transmit the loads to the pool floor and provide a sliding contact. The leveling pad screw permits the leveling adjustment of the rack.

The components of the cell assembly are the cell enclosure and the wrapper plates. The wrapper plates are attached to the outside of the cell by spot welding along the entire length of the wrapper. The wrapper forms the pocket and establishes the size of the non-cell locations.

Region 3 Mechanical Design

The Region 3 storage racks are identical to the Region 2 storage racks, with the exception that a neutron absorbing Metamic poison insert assembly (PIA) is contained within the spacer pocket and an optional lead-in device made from SA240-304L may be installed on top of the spacer pockets. The PIA consists of two Metamic poison panels separated by a welded SA240-304 stainless steel frame to maintain the water gap specified by criticality considerations. Metamic consists of a high purity Type 6061 aluminum alloy matrix reinforced with Type 1 ASTM C-750 isotopically graded boron carbide.

Spent Fuel Pool Rack Design Basis

The spent fuel rack designs described herein employ three separate and different arrays which will be considered as three separate spent fuel racks. All three storage arrays are designed on the basis of the currently accepted NRC guidance on spent fuel rack design, with consideration

of the changes in fuel and fission product inventory resulting from depletion in the reactor core. Although all three storage racks types differ in design, they take credit for the reduction in reactivity associated with fuel burnup. Region 3 racks additionally contain neutron absorbing material (Metamic) inserts, which allow for the storage of fresh (new) fuel assemblies. Criticality safety is assured by restricting the fuel burnup to a minimum acceptable value which depends upon the initial fuel enrichment. Because of the differences in design between the three rack types, different limiting burnup requirements are specified for each region. These limiting burnup requirements assure that the racks will remain subcritical, including the bias and reactivity effects of manufacturing tolerances, evaluated at the 95% probability with a 95% confidence level, even in the absence of soluble boron in the pool water.

All fuel handling and transfer containers are also designed to maintain a safe geometric array. Mechanical damage to the fuel assemblies during transfer operations is possible, although remote. Since the fission product release would occur under water, the amount of radiological activity reaching the environment would present no appreciable hazard. A fuel handling accident analysis is included in Chapter 14.

All spent fuel assembly transfer operations are conducted under water except with the loaded dry fuel system cask. The water level in the fuel transfer canal will be maintained to ensure a nominal depth of 8 feet of water over the active fuel line of the spent fuel assemblies while the fuel is shielded by a minimum of 1.75 inches of steel in fuel moving equipment or 9 feet of water with no other shielding during movement from the core into storage. This limits radiation due to spent fuel at the surface of the water to less than 10 mrem/hr. The depth of the water over the assemblies, as well as the thickness of the concrete walls of the transfer canal, is sufficient to limit the maximum continuous radiation levels in the working area due to spent fuel to 2.5 mrem/hr. The spent fuel pool is covered by a superstructure described in Section 5.3.2.

Table 9-12 establishes the radionuclide concentrations of the spent fuel pool. The following dose rates above the pool are as follows:

A. At the handrail around the pool - 1.88 mrem/hr

B. On the fuel handling bridge - 7.48 mrem/hr (includes 0.21 mrem/hr contribution from fuel element in transfer mast).

These dose rates are highly conservative in that they are based on the maximum expected concentration of radionuclides in the pool assuming one percent failed fuel and a crud burst model. Using these dose rates to estimate occupational doses includes the additional conservative assumption that these dose rates are present at all times while in reality they would be reduced as the pool is cleaned by the purification system.

It is estimated based on the above that the occupational dose received from activities associated with fuel handling in the spent fuel pool is less than 2.44 man-rem per year, with exposure by activity as specified below:

Activity	Dose
80 man-hours cleanup and inspection @ 1.88 mrem/hr	0.15 man-rem
700 man-hours new fuel receipt and inspection @1.88 mrem/hr	1.31 man-rem

120 man-hours equipment checkout prior to refueling @ 1.88 mrem/hr	0.23 man-rem
100 man-hours fuel handling during refueling @ 7.48 mrem/hr	0.75 man-rem
Total Annual Dose	2.44 man-rem

Additional dose is expected during transfer of fuel to dry storage. A breakdown of the fuel handing sequence and associated predicted dose can be found in the dry fuel system SARs.

In addition to the above, some dose is received during filter and spent resin change operations associated with the spent fuel pool purification system. Based on experience with these operations, it is estimated that the annual dose equivalent from these operations is < 0.2 man-rem.

The water level in the spent fuel pool will be maintained to ensure a nominal depth of 8 feet of water over the active fuel line of the spent fuel assemblies while the fuel is shielded by a minimum of 1.75 inches of steel in fuel moving equipment or 9 feet of water with no other shielding during fuel handling and 24 feet of water over the active fuel line when the assemblies are stored in the spent fuel storage racks. This depth of water is calculated to limit radiation at the surface of the water to less than 10 mrem/hr. The walls of the spent fuel pool are designed to limit radiation in surrounding areas in accordance with Chapter 11.

The spent fuel pool is a Seismic Class 1 structure and is completely lined with stainless steel for leak tightness. In addition, the suction nozzle for the spent fuel pool circulating pumps is located three feet below the normal water level in the pool to prevent inadvertent drainage of the pool in the unlikely event of a non-isolable leak in the spent fuel pool cooling system.

The spent fuel pool area radiation monitor activates an alarm if the dose rate in the area exceeds a value based on normal background value. In this event, the area would be secured and maximum flow would be diverted to the purification system. Procedures would be undertaken to determine the cause of the increased radiation level.

Variation in the water level of the spent fuel pool is monitored by a level indicator, and alarms if the level increases to more than one foot above, or decreases to more than six inches below normal water level. Procedures would then be undertaken to determine and secure the source of water flow. In the case of decreasing water level, provisions would be made to supply water to the pool from the most readily available source.

Component failures would be detected by the level and temperature indicators on the spent fuel pool. The loop containing the failed component would be isolated and cooling would be continued with the other loop. Maintenance would be performed on the component, and the component returned to service as soon as possible.

Level indicators and alarms are provided on the spent fuel pool to detect leakage.

The spent fuel pool is completely lined with stainless steel to prevent leakage. However, a system of drainage channels is provided behind the liner plate to collect water in the unlikely event that a small amount of leakage does occur.

The spent fuel pool cooling system is designed to prevent inadvertent drainage of the spent fuel pool due to a failure in the cooling system as described in Sections 9.4.2.2 and 9.4.2.6.

Water in the reactor vessel is cooled during shutdown and refueling by the DHR system described in Section 9.5. In case of a power failure, this system is operated by the emergency power supply. The spent fuel storage pool water is cooled by the spent fuel cooling system as described in Section 9.4. A power failure during the refueling cycle creates no immediate hazardous condition owing to the large water volume in both the fuel transfer canal and spent fuel storage pool. With a normal quantity of spent fuel assemblies in the storage pool and no cooling available, the water temperature in the spent fuel storage pool would increase as discussed in Section 9.4.

During the refueling period the water level in both the fuel transfer canal and the spent fuel storage pool is the same, and the fuel transfer tube valve is open. The operating mechanisms of the fuel transfer carriage consist of two winches, one located in the auxiliary building and one located in the reactor building. The auxiliary building winch is easily accessible for maintenance and inspection before the start of refueling operations.

During reactor operation, a bolted and gasketed closure plate, located on the reactor building end of the fuel transfer tube, prevents leakage of water from the spent fuel storage pool into the transfer canal in the event of a leak through the fuel transfer tube valve. Both the spent fuel storage pool and the fuel transfer canal are completely lined with stainless steel for leak tightness and ease of decontamination. The fuel transfer tube is appropriately attached to these liners to maintain leak integrity. The spent fuel storage pool cannot be accidentally drained due to pipe rupture, since water must be pumped out through a suction pipe.

All electrical gear except some limit switches is located above water for greater integrity and ease of maintenance. The hydraulic system that actuates the rotating fuel basket uses demineralized water for operation to eliminate contamination.

The fuel transfer canal and storage pool are provided fill and makeup water, respectively, from the BWST, whose boron concentration is maintained at a minimum value of 2270 ppm (nominal 2470 ± 200) (for exact values see the ANO Unit 1 Technical Specifications). The minimum boron concentration required in the fuel transfer canal and storage pool is > 2000 ppm. Although this concentration is sufficient to maintain core shutdown if all of the control rod assemblies are removed from the core, only a few control rods are removed at any one time during the fuel shuffling and replacement. Although not required for safe storage of spent fuel assemblies, the spent fuel storage pool water is also borated so that the transfer canal water is not diluted during fuel transfer operations.

Each fuel handling bridge mast travel is designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Gross failures of fuel are prevented by applying appropriate safety margins in the design and control of the core.

9.6.2.4 Safety and Environmental Analysis

9.6.2.4.1 General Description

The number of spent fuel storage racks was modified to 968 fuel assemblies to increase storage volume. These replacement spent fuel storage racks provide storage capacity for a full core of irradiated fuel assemblies and twelve batches of spent fuel assemblies. Therefore, 10.5 annual discharges may be accommodated or seven annual discharges may be accommodated while still maintaining the capability for a full core discharge.

The replacement spent fuel storage racks were fabricated from approximately 347,000 pounds of type 304 stainless steel and 9,600 pounds of Boraflex. Approximately 23,600 pounds of Metamic PIAs have been inserted into the newly designated Region 3 storage racks. A description of the spent fuel storage racks is provided in Section 9.6.2.3.

9.6.2.4.2 Environmental Aspects

Currently solid wastes are collected from the spent fuel pool and concentrated in two areas: (1) The spent fuel pool ion exchanger which removes ionic material from the pool water and (2) The particulate filters which remove particulates larger than 5 microns. The ANO-1 ion exchanger resin volume is approximately 20 ft^3 and is currently changed about once per year. ANO-1 particulate filter system contains two filters which are changed approximately once per year.

The ANO-1 spent fuel pool demineralizer (ion exchanger) resin volume of approximately 20 ft^3 represent $\leq$ 10% of the total ANO-1 annual solid waste in the form of spent resins.

These resins are designed to be changed based on an increase in differential pressure rather than on the lack of ability to remove radioactive ionic material. Neither the frequency of fuel addition nor the annual amount of fuel to be added to the spent fuel pool will change due to the rack modifications. Therefore, the annual amounts of contaminants added to the pools are not expected to increase significantly. Since the resins are designed to be changed annually, and the annual amount of contaminants is not expected to increase, no appreciable increase in solid waste in the form of spent fuel pool ion exchanger resins is expected.

Similarly, the particulate filters are designed to be changed annually, based on differential pressure. As in the case of the ion exchanger resins, no significant change in the frequency of replacement due to the modified racks is expected.

The number of ANO-1 storage locations was increased in 1976 and again in 1982, and no significant increase in radioactive solid waste generation in the form of resins or filters has resulted.

The radionuclide concentrations in the spent fuel pools as presented are based on one percent failed fuel and a crud burst model. Exposures based on this are presented in Section 9.6.2.3. No significant increase in personnel doses was made due to the increase in storage capacity.

In regard to personnel exposure received during filter and resin changes, based on experience it is estimated that the annual exposure is $\leq$ 0.2 man-rem/unit. This annual exposure is not expected to increase, since the frequency of change is not expected to be altered.

The total volume of water in the spent fuel pool, cask pit and tilt pit at normal pool level is 368,000 gallons. As mentioned previously, the spent fuel pool purification loop utilizes two filters and a 20 ft^3 nonregenerative mixed bed demineralizer with a flow rate of 180 gpm. One volume of the spent fuel pool can thus be processed in approximately 34 hours.

In view of the above, the radioactive solid waste generated from the facility will not increase as a result of this modification. Table 9-12 presents the maximum concentrations of radionuclides in the spent fuel pool. These concentrations are the same as those used in the spent fuel pool and are highly conservative since they are based on one percent failed fuel and a crud burst

model. They are also appropriate for the increased storage capacity since the primary contribution to the activity comes from the crud burst assumption which is not affected by the increased storage capacity.

No significant contribution to dose is made by the fuel covered to a depth of 24 feet with water. Fuel being transferred is the controlling contributor to the basic dose rates, not the stored fuel. Thus, there should be no significant increase in annual man-rem exposure over that which would be experienced with the existing spent fuel pool storage facility.

Radioactive gases may be released from the spent fuel pool directly into the atmosphere of the auxiliary building. This air is exhausted through particulate and charcoal filters. The major radioactive gas that may be released during fuel pool storage is ^{85}Kr with a half-life of 10.76 years.

The design of the Unit 1 facility permits measurement of radioactive gases released from individual ventilation systems. Data is also available for releases from the overall plant. The data for ^{85}Kr, ^{131}I and tritium for part of 1974, all of 1975, and the first half of 1976 are provided in Table 9-13.

The initial estimate of ^{85}Kr release from the auxiliary building was 6 curies/year. Increasing the fuel pool storage capacity by a factor of 2.33 does not necessarily increase the ^{85}Kr release rate. The fuel discharge from the reactor continues on a one-third core per year rate and the release of ^{85}Kr is most likely to occur during the initial handling and the first year of storage. Nevertheless, a conservative approach is to assume that the ^{85}Kr yearly release increases by a factor of 2.33. Therefore, the maximum ^{85}Kr release from the auxiliary building is 14 Curies/year, an increase of 8 Curies/year. The total plant release of ^{85}Kr initially projected was 710 Curies/year; thus, the maximum percentage increase due to fuel pool storage expansion would be less than 1.13 percent.

9.6.2.4.3 <u>Safety Analysis</u>

9.6.2.4.3.1 <u>Criticality Considerations</u>

9.6.2.4.3.1.1 Criticality Analysis for New Fuel Storage

The new fuel storage vault was originally sized to accommodate 72 new fuel assemblies. The new fuel assemblies are stored in parallel rows having a center-to-center distance of 21 inches in both horizontal directions. The criticality analysis restricts the use of 8 or 10 interior locations, depending on initial enrichment, so only 62 or 64 locations may contain fuel assemblies.

The design method which determines the criticality safety of fuel assemblies in the new fuel storage rack uses the NITAWL/KENO5a computer codes for cross section generation and reactivity determination. Originally developed by Oak Ridge National Laboratory for criticality safety analyses, NITAWL uses the 238 group SCALE cross-section library to derive weighted cross-sections for the U-238 in the resonance region with the Nordheim resonance integral treatment. Output from NITAWL is supplied to the KENO5a program, a three dimensional Monte-Carlo code that calculates the system multiplication factor, k_{eff}. Critical experiments have been analyzed using the KENO methodology to demonstrate applicability to criticality analysis and to establish method bias and uncertainty which are included in the reactivity analysis of the rack. The NITAWL/KENO5a code system was also used to determine the reactivity effects of fuel pellet density and enrichment and storage cell lattice spacing.

The criticality of fuel assemblies in the fresh fuel storage rack is prevented by limiting the U-235 enrichment in the fuel rods to 4.95 wt%, by maintaining a minimum separation of 21 inches between assemblies, and by precluding the placement of ten (10) interior storage cell locations as illustrated in Figure 9-56a. If the enrichment of the fuel is limited to 4.2 wt% ^{235}U or less, then only eight (8) locations need to be restricted in the positions shown in Figure 9-56b. The criticality models use a B&W 15x15 base assembly. No burnable poison, control rod assembly, axial power shaping rod, or other fixed poisons were credited in the criticality analysis though they could be stored in the pit. Rack structures and assembly grid spacer materials are conservatively neglected. The maximum k_{eff} is calculated to be less than 0.98 under conditions of optimum moderation, and less than 0.95 under conditions of flooding by unborated room temperature water.

9.6.2.4.3.1.2 Criticality Analysis for Spent Fuel Storage

9.6.2.4.3.1.2.1 Methodology

The design method which determines the criticality safety of fuel assemblies in the spent fuel storage racks uses the CASMO-4 and MCNP4a system of codes for generation of the isotopic composition of spent fuel and reactivity determination. Three dimensional MCNP4a models of fuel assemblies are utilized.

Benchmark calculations of more than 50 critical experiments were used to establish a bias and bias uncertainty of 0.0009 and 0.0011, respectively for MCNP4a. The CASMO-4 two-dimensional integral transport code is used to evaluate the rack and fuel design tolerances for the spent fuel storage rack calculations and to calculate the isotopic composition of the spent fuel.

Fuel assembly and rack geometry mechanical tolerances are evaluated through a statistical combination of the reactivity increment associated with each tolerances at its most reactive condition. The statistically treated tolerances include variability in fuel enrichment, density, as well as dimensional uncertainties. Dimensional uncertainties include variations in pellet and clad radii, fuel rod pitch, guide tube thickness, rack structure thickness, rack cell and fuel assembly eccentric positioning. Therefore, the maximum keff is determined from the MCNP4a calculated k_{eff}, the calculational bias, the temperature bias, and the applicable uncertainties and tolerances (bias uncertainty, calculational uncertainty, rack tolerances, fuel tolerances, depletion uncertainty). Since these tolerances are independent, they are statistically combined and applied as a Δk bias to the MCNP4a calculated k_{eff} in the determination of the 95/95 value.

The high-density spent fuel PWR storage racks for ANO are analyzed in accordance with the applicable codes and standards listed below. The objective of this analysis is to ensure that the effective neutron multiplication factor (keff) is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity. In addition, it is demonstrated that keff is less than 1.0 under the assumed loss of soluble boron in the pool water, i.e. assuming unborated water in the spent fuel pool. The maximum calculated reactivities include a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level.

Applicable codes, standard, and regulations or pertinent sections thereof, include the following:

- *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3 - July 1981.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- L. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- Code of Federal Regulation 10CFR50.68, Criticality Accident Requirements (for soluble boron)

9.6.2.4.3.1.2.2 Region 1 and Region 2 Storage Racks

Region 1 was originally designed with credit for Boraflex as a fixed neutron absorber. In the present analysis, the Region 1 racks have been reanalyzed with no credit for the Boraflex in the racks.

Two separate analyses were performed for the Region 1 and Region 2 storage racks. The first analysis considers fresh fuel assemblies at 4.95 wt% ^{235}U loaded into two of every four cells in a checkerboard pattern. The second analysis determines the minimum allowable burnup for a range of enrichments between 2.0 and 4.95 wt% ^{235}U. The analysis of spent fuel reactivity includes consideration of the axial distribution in burnup. CASMO calculations were used to determine the isotopic composition, including actinide and fission products, in each of the 10 axial zones of the active fuel length. These spent fuel compositions were then used in a three-dimensional reactivity calculation using the MCNP4a code.

Depletion calculations were made for various post shutdown cooling times and used to determine the reactivity of the racks for 0, 5, 10, 15 and 20 years cooling time.

For normal fuel storage, a model is developed based on the following assumptions:

1. The moderator is borated or unborated water at a temperature in the operating range that results in the highest reactivity.

2. Structural components (such as the fuel assembly end fittings, the rack grid plates, and the rack support plates) are not included in the model.

3. The base fuel assembly analyzed is a B&W 15 x 15 assembly. Radial enrichment zoning is not credited, all enriched fuel is modeled at 4.95 wt% ^{235}U.

4. No credit is taken for the presence of integral fixed poisons, such as BPRAs or APSRs in the spent fuel storage rack calculations. BPRAs and APSRs were considered in evaluating the spent fuel isotopic composition from in-core depletion calculations.

5. APSRs and BPRAs are assumed to extent the entire active fuel length.

6. The effective multiplication factor of an infinite radial array of fuel assemblies was used in the analyses, except for the assessment of certain abnormal/accident conditions where neutron leakage is inherent.

The criticality analyses for Region 1 and Region 2 of the spent fuel storage pool demonstrate that for the defined acceptance criteria, the maximum keff is less than 1.0 without credit for soluble boron and less than 0.95 with credit taken for soluble boron. Under accident conditions, it is demonstrated that soluble boron is required to assure that the maximum keff is less than or equal to 0.95. The minimum soluble boron concentration under normal and accident conditions required to maintain k_{eff} below 0.95, including all manufacturing and calculational tolerances, for the storage of spent fuel in the Region 1 and Region 2 racks is less than the Technical Specification minimum boron concentration of > 2000 ppm.

The criticality analysis also demonstrates that for storage of fresh fuel assemblies in the Region 1 and Region 2 racks in a 2-of-4 checkerboard pattern with the alternate cells remaining empty, the maximum k_{eff} is below the regulatory limit of 0.95 without requiring any soluble boron.

Based on these calculations, the minimum burnup requirements for various enrichments and post-shutdown cooling period is depicted in Table 9-31 for Region 1 and Table 9-32 for Region 2. Linear interpolation of the minimum burnup requirements between enrichments is allowed, however, linear interpolation between cooling times is not allowed. Therefore, the cooling time of an assembly to be stored in Region 1 or Region 2 must be rounded down to the nearest cooling time.

9.6.2.4.3.1.2.3 Region 3 Storage Racks

The Region 3 storage cells are identical to the Region 2 storage cells, with the single exception of Metamic PIAs centrally located in the water gaps between storage cells.

Two separate analyses are performed for the Region 3 storage racks. The first analysis considers fresh, non-irradiated fuel assemblies at 4.35 wt% ^{235}U loaded into every storage cell. The second analysis considers a 3 of 4 loading pattern, which contains 3 fresh, non-irradiated fuel assemblies at 4.95 wt% ^{235}U, and 1 irradiated fuel assembly with a maximum nominal enrichment of 4.95 wt% ^{235}U and a minimum burnup of 20.1 GWD/MTU. No credit was taken for longer cooling times in the 3 of 4 loading pattern.

The criticality analyses for Region 3 of the spent fuel storage pool demonstrate that for the defined acceptance criteria, the maximum k_{eff} is less than 0.95 with credit taken for soluble boron. Under accident conditions, it is demonstrated that soluble boron is required to assure that the maximum k_{eff} is less than or equal to 0.95. The minimum soluble boron concentration under normal and accident conditions required in the Region 3 racks to maintain k_{eff} below 0.95, including all manufacturing and calculational tolerances, is less than the Technical Specification minimum boron concentration of > 2000 ppm. Additionally, it is demonstrated that the maximum k_{eff} is less than 1.0 assuming unborated water in the spent fuel pool.

9.6.2.4.3.1.2.4 Interaction Between Adjacent Storage Racks And Different Loading Patterns

In addition to the calculations for each style of rack in the ANO Unit 1 spent fuel pool, the reactivity effect of potential interaction between adjacent racks and between different loading patterns were determined. Interfaces within the rack include spent and fresh fuel loading patterns within the same rack to determine acceptability. Interface calculations between racks include Region 1- Region 1, Region 2-Region 2, Region 3-Region 3, Region 1-Region 3 and Region 2-Region 3. The calculated reactivity from the interface calculations were then compared to the calculated reactivity from the reference infinite array calculations. The criticality analysis pertaining to the reactivity effect of the interfaces concluded that:

- In the Region 1 and Region 2 Racks, a fresh fuel checkerboard and uniform spent fuel loading may be placed in the same rack.

- In Region 1 and Region 2 racks, if adjacent racks contain a checkerboard of fresh fuel assemblies, the checkerboard must be maintained across the gap, i.e., fresh fuel assemblies may not face each other across a gap.

- In Region 3, uniform loading of fresh fuel at 4.35wt% ^{235}U may be combined with 3 of 4 loading in the same rack as long as a row of fresh and spent fuel in the 3 of 4 loading pattern faces the uniform loading of all fresh fuel at 4.35 wt% ^{235}U.

- If adjacent Region 3 racks contain different loading patterns (one rack contains all fresh fuel at 4.35 wt% and the other rack contains a 3 of 4 loading pattern), both fresh and spent fuel must be in the outer row of the rack containing the 3 of 4 pattern facing the all fresh fuel pattern.

- If adjacent Region 3 racks both contain 3 of 4 loading patterns, both racks may not have fresh fuel facing the other rack. Calculations with both Region 3 racks containing 3 of 4 patterns with all fresh fuel in the outer row of one rack and fresh and spent fuel in the outer row of the second rack is allowed.

- All interfaces between dissimilar racks (Region 1-Region 3 and Region 2-Region 3) are permitted.

9.6.2.4.3.1.2.5 Dry Casks

Prior to loading or unloading a cask in the SFP, a SFP criticality analysis must be performed to ensure the SFP requirements are satisfied.

<u>Holtec HI-STORM 100 MPC-24</u>

The methods used in the analysis of the MPC-24 include MCNP4a, CASMO-4 and KENO5a. The base assembly analyzed was AVERA Mk B-HTP 15 x 15 fuel assemblies. The ANO-1 SFP also contains B&W 15 x 15 Mark B fuel, however, the AVERA fuel represents more reactive fuel over the full range of burn-ups.

The analyses assume the most reactive fuel and moderator temperature, no credit for the presence of control rod assemblies or burnable poison rod assemblies, and no credit was taken for soluble boron present in the water for normal operations. The analyses also conservatively assumed the ANO-1 spent fuel with initial enrichments up to 4.4 wt%. Credit was taken for soluble boron under accident conditions.

The result of these analyses is a 95 percent probability/95 percent confidence level of K_{eff} less than 0.95 for storage of spent fuel in the MPC-24 while it is in the SFP cask loading pit with the cask loading pit gate open.

9.6.2.4.3.2 Abnormal and Accident Considerations

New Fuel Storage

In the consideration of the fuel handling accident where an assembly drops on top of the fresh fuel rack, 'dry' conditions are assumed. For this case, K_{eff} was determined to be well below the regulatory guidelines.

Spent Fuel Storage

The effects on reactivity of credible abnormal and accident conditions are analyzed. This analysis identifies which of the credible abnormal or accident conditions (abnormal temperature, horizontal and vertical fuel assembly drops, misloaded fresh fuel assemblies and mislocated fresh fuel assemblies) result in exceeding the limiting reactivity value (keff ≤ 0.95). Most accidents will not result in K_{eff} of the rack exceeding the regulatory limit. However, for those accident or abnormal conditions that result in exceeding the limiting reactivity, a minimum soluble boron concentration is determined to ensure that keff < 0.95. The double contingency principal of ANSI N16.1-1975 (and the USNRC letter of April 1978) specifies that it shall require at least two unlikely independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

For all the abnormal conditions and accidents analyzed, mislocated fuel assemblies in Regions 1, 2 and 3 required the highest level of soluble boron in the SFP water to ensure the regulatory reactivity limit of $k_{eff} \leq 0.95$ is not exceeded. The maximum SFP soluble boron concentrations, required to maintain k_{eff} less than 0.95, including all biases and uncertainties for this postulated accident condition, were calculated to be 889 ppm for Region 1, 875 ppm for Region 2 and 843 ppm for Region 3. These boron concentrations are well below the required Tech Spec boron concentration of > 2000 ppm in the spent fuel pool.

For Regions 1 and 2, the temperature coefficient of reactivity is positive, and temperatures above the maximum would cause an increase in the reactivity, and therefore are treated as accidents. Credit for the soluble boron in the pool water is allowed and will maintain a low and acceptable reactivity below k_{eff} limits. For Region 3, the temperature coefficient of reactivity is negative, therefore any increase in temperature above the minimum would cause a reduction in the reactivity.

For the case in which a fuel assembly is assumed to be dropped on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 12 inches, which is sufficient to preclude neutron coupling. Consequently, the horizontal fuel assembly drop accident will not result in a

significant increase in reactivity. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

For the case in which a fuel assembly is assumed to be dropped into a location that might be occupied by another assembly or that might be empty, a vertical impact would at most cause a small compression of the stored assembly, if present, or result in a small deformation of the baseplate for an empty cell. These deformations could potentially increase reactivity. However, the reactivity increase would be small compared to the reactivity increase created by the abnormal location of a fresh assembly. The vertical drop is therefore bounded by this abnormal location accident.

The misloading of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (keff of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment were to be inadvertently misloaded into a storage cell intended to be used for spent fuel or misloaded into a storage cell intended to be empty. Based on the criticality analysis, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this misloaded fuel assembly accident.

The mislocation of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (keff of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (4.95 wt%) were to be accidentally mislocated outside of a storage rack adjacent to other fuel assemblies. Based on the criticality analysis, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this mislocated fuel assembly accident.

An additional analysis was performed to determine the reactivity impact of the Region 3 rack in the unlikely event that the neutron absorber was to be completely absent. Credit was taken for 1600 ppm soluble boron and the Metamic material was replaced with water. Additionally, it was conservatively assumed that all assemblies in the Region 3 rack were fresh fuel assemblies with a maximum initial enrichment. The results of this postulated accident condition show that the maximum keff, including all bias and uncertainties, is below the regulatory limit (keff of 0.95).

<u>Dry Spent Fuel Storage Casks</u>

When evaluating postulated accidents associated with loading/unloading any design of dry casks, the double contingency principle of ANS N16.1-1975 is applied, which specifies that at least two unlikely independent and concurrent events are required to produce a criticality accident. Therefore, for accident conditions, the presence of soluble boron up to 1600 ppm in the SFP water can be assumed as a realistic initial condition since its absence would be a second unlikely event. The cask is analyzed such that $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

<u>Holtec HI-STORM 100 MPC-24</u>

The effects of credible abnormal and accident reactivity conditions previously analyzed for pool racks were evaluated for the HI-STORM 100 MPC-24. None of the abnormal or accident conditions identified cause the reactivity of the MPC-24 to exceed the limiting reactivity value ($K_{eff} \leq 0.95$) considering the presence of the Unit 1 Technical Specification minimum soluble boron.

9.6.2.4.3.3 Cask Drop Consequences

The spent fuel shipping cask storage area is acceptably designed to minimize the consequences of an accidental drop of a spent fuel shipping cask. The spent fuel rack modification does not involve the spent fuel shipping cask area. Therefore, the modification does not affect the original cask drop evaluation.

9.6.2.4.3.4 Material Considerations

All permanent structural material used in the fabrication of the new spent fuel storage racks and Metamic PIAs is 300 series stainless steel, mostly 304. This material was chosen for compatibility with the spent fuel pool water which contains boric acid at a maximum concentration of 3,500 ppm boron.

At the normal operating temperature of 120 °F, there is no deterioration or corrosion of stainless steel in this environment. There is also no corrosion problem at temperatures up to and including pool boiling. All other structural components in the spent fuel pool system, such as the pool liner, cooling system pipe connections, etc., are made of stainless steel.

The 304 stainless steel utilized at fabrication of the racks complies with ASTM Specifications A276-71 or A167-74.

The neutron absorber material used in the Region 3 racks is Metamic, an extruded and hot-rolled composite of aluminum and boron carbide, B_4C. Boron carbide is a very hard, strong and chemically stable component, compatible with the spent fuel pool environment. Aluminum is a highly corrosion resistant metal in water containing boric acid. The composite is therefore very stable and compatible with the spent fuel pool environment.

In summary, the material used in the current spent fuel storage racks is similar to the original components and does not affect or alter previous evaluations.

9.6.2.4.3.5 Thermal Considerations

The discussion that previously resided in this section has been removed since it was essentially a duplicate of Section 9.4.2.1.2, “Thermal-Hydraulic Considerations” within the spent fuel cooling system chapter. Refer to that section for the discussion related to this heading.

9.6.2.5 Seismic and Structural Analysis

9.6.2.5.1 Mechanical Considerations

Comprehensive structural evaluations of the Unit 1 modified spent fuel storage racks were performed. These evaluations included calculation of static and dynamic seismic loads, stress analysis for all applicable loading combinations, and determination of structural adequacy of all load carrying members including the spent fuel pool structure.

Rack analyses were performed at the time of the original rerack project (considering Region 1 and Region 2 racks). Subsequent analyses of the racks were performed to include the addition of the Metamic poison panels (inserts) to the flux traps of two of the Region 2 racks (now referred to as the Region 3 racks).

Design Criteria

Structural design criteria were developed to assure conformance with recognized codes and applicable NRC Regulatory Guides as follows:

A. The fuel storage racks are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class III Linear Supports, 1980 Edition through Winter 1981 Addenda.

B. Regulatory Guide 1.13 - The design is in conformance with the stated provisions for spent fuel equipment.

C. Regulatory Guide 1.29 - The spent fuel storage racks are designed as Category I structures.

D. Regulatory Guide 1.92 - Seismic load combinations of vibrational modes and three orthogonal component motions (two horizontal and one vertical) are in compliance with the requirements of Regulatory Guide 1.92.

E. NRC - Standard Review Plan, Section 3.8.4, Other Seismic Category I Structures - Load combinations and structural acceptance criteria for steel structures are used.

Loading conditions considered included dead load, live load, operating basis earthquake, safe shutdown earthquake, thermal and hydraulic loads. Evaluation of the effects of the maximum crane uplift force and a dropped fuel assembly accident are included. For the seismic load combinations, consideration is given to the worst possible fuel loading condition of the fuel storage racks, varying from empty to fully loaded. The Unit 1 floor response spectra (horizontal and vertical) for the auxiliary building at the elevation of the bottom and at the top of the spent fuel storage pool are used as the basis for the seismic analysis.

Seismic Analysis

The dynamic response of the fuel rack assembly during a seismic event is the condition which produces the governing loads and stresses on the fuel rack structures. The dynamic response, internal stresses, and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. For the analyses performed including the added Metamic poison panels to two of the racks, the nonlinear models included representation of all eight racks together as well as single rack models. The second phase is a static analysis of detailed rack assembly finite element models to obtain stresses, using forces obtained from the nonlinear analyses. The damping values used in the seismic analysis are two percent damping for OBE and four percent damping for SSE for the analyses for the original rerack project, and two percent damping for both OBE and SSE for the subsequent reanalyses including addition of the Metamic poison panels.

The simplified nonlinear finite element models are used to determine the fuel rack response for the structural characteristics of a submerged rack assembly. The nonlinearities of the fuel rack assembly which are accounted for in the models are the gaps between the fuel cell and the fuel assembly, the gaps between the added Metamic poison panels and the fuel cell walls, the boundary conditions of the fuel rack support locations (that is, allowance for potential uplift and sliding), and energy losses at the support locations.

The WECAN computer program was used to determine the nonlinear time history response of the fuel assembly/fuel rack system for the analyses for the original rerack project, and the SOLVIA computer program was used for the subsequent reanalysis including the added Metamic poison panels. The fuel assembly to cell impact loads, and overall rack response are obtained from the nonlinear time history results.

For the analysis for the original rerack project, the detail model is a three-dimensional finite element representation of a rack assembly consisting of discrete three-dimensional beams interconnected at a finite number of model points. The results of the nonlinear time history model are incorporated in the detail model. Since the detail model does not account for the nonlinear effect of a fuel rack assembly, the internal loads and stresses for the rack assembly obtained from this model are corrected by load correction factors. The load correction factor is derived from the nonlinear model results and is applied to the components in the stress analysis. The responses of the model from accelerations in three directions are combined by the Square Root Sum of the Squares method in the stress analysis. The loads in two major components (support pad assembly and fuel cell) are examined, and the maximum loaded section of each of these components is found. These maximum loads from the detail model are corrected by the nonlinear load correction factors and used in the stress analysis to obtain the stresses within the rack assembly.

For the subsequent analyses including the Metamic poison panels, detailed models consisted of three-dimensional finite element representations of portions of the racks (single and multiple cells) consisting of shell and beam elements. Forces from the nonlinear time history analyses were applied to these models directly to obtain stress results.

Fuel Rack Structural Analysis

The stress analysis for the racks is performed using the load combinations specified in the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications." These load combinations are consistent with the load combinations from NUREG-0800, Standard Review Plan, Section 3.8.4.

The thermal loads due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction. The major seismic loads are produced by the operational basis earthquake (OBE) and safe shutdown earthquake (SSE) events.

It is noted from the seismic analysis that the magnitude of stresses vary considerably from one geometrical location to the other in the model. Consequently, the maximum loaded cell assembly, grid assembly, and the leveling pad assembly are analyzed. Such an analysis envelopes the other areas of the rack assembly.

The computed stresses are below the allowable stresses as required by the NRC Position Paper.

Fuel Handling Crane Uplift Analysis

A fuel handling crane uplift analysis is performed to demonstrate that the rack can withstand the maximum 3000 pound uplift load of the fuel handling crane without violating the criticality acceptance criteria. In this analysis the uplift load is assumed to be applied to a fuel cell. Resulting stresses are within acceptable limits, and there is no change in rack geometry of a magnitude which causes the criticality acceptance criteria to be violated.

Fuel Bundle/Module Impact Evaluation

An analysis is performed to evaluate the effect of an impact load due to fuel assembly and fuel storage cell interaction during a seismic event. The fuel rack system consists of an array of cells which form the fuel rack structure and fuel assemblies. The fuel rack system is located in the spent fuel pool and is submerged in water.

Since the fuel assembly is stored within the cell, the gap between the fuel assembly grid and cell changes (i.e., opens and closes) during a seismic event. From the equation of motion for such a system it is evident that the fuel rack system is nonlinear. This condition necessitates the performance of a transient dynamic analysis.

The mathematical features of the nonlinear fuel rack model facilitate the determination of the fuel assembly/cell interaction and hydrodynamic mass (fluid mass) effects on the fuel rack response during seismic excitation.

The effect of fuel assembly and fuel storage cell impact force on the rigid body displacements is obtained from the nonlinear analysis. The analyses were conducted with a minimum coefficient of friction of 0.2, and it is shown that the rigid body displacement is less than existing clearances between the racks and between the racks and pool walls. Thus, impact between adjacent rack modules or between a rack module and the pool wall is precluded.

The fuel assembly and fuel storage cell impact forces obtained from the nonlinear analyses are used to evaluate the effects on the fuel rack structure and fuel assembly structure. These loads are within the allowable limits of the fuel rack module materials and fuel assembly materials. Therefore, there is no damage to the fuel assembly or fuel rack module due to impact loads.

Acceptance Criteria

The results of spent fuel rack analysis for normal and faulted load combinations are noted in the Spent Fuel Pool Rerack Analysis Licensing Submittal, Section 3.2, in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

The major normal and upset conditions loads are produced by the operational basis earthquakes (OBE). The thermal stresses due to rack expansion relative to the pool floor are negligible since the support pads are not structurally restrained in the lateral direction.

The faulted condition loads are produced by the safe shutdown earthquakes (SSE) and a postulated fuel assembly drop accident.

The stresses are below the allowable stresses as required by the ASME B&PV Code, Section III, Subsection NF, 1980 Edition through Winter 1981 Addenda.

In summary, the results of the seismic and structural analyses show that the spent fuel storage racks meet all the structural acceptance criteria adequately.

Pool Structural Analysis

The existing spent fuel pool structures (floor slab, walls, and supporting walls) were analyzed for the modified fuel rack loads using the STARDYNE finite element computer program. The finite element model consisted of the pool walls and floor, foundation walls, cask laydown area, and fuel transfer canal area with the lowest elevation of 335'-0" assumed as a fixed boundary. The pool walls and floor slabs are modeled utilizing two layers of three dimensional solid "brick" elements with three degrees of freedom per node. In order to permit recovery of cube surface stresses, extremely thin quadrilateral membrane elements coincident with the nodes forming the corresponding surface of the solid elements were utilized.

Recovery of these stresses along with cube centroidal stresses permitted the calculation of the required resultant forces and moments for an American Concrete Institute Code evaluation. The foundation walls were modeled using only one layer of brick elements through the thickness.

The pool liner was not modeled since it is not acceptable to count on its stiffness contribution to the overall pool capacity. However, stress evaluation of this component was conducted using results obtained from the computer analysis.

In order to correctly represent boundary conditions, modeling of the floor diaphragms and shear walls attached to the spent fuel pool was accomplished utilizing STARDYNE matrix elements.

The pool structure analysis was updated using the change in loads from the racks (deadweight and seismic) obtained from the subsequent analyses of the racks including the addition of Metamic poison panels to two of the racks. This was accomplished by factoring the pool structure analysis load cases for the rack loads appropriately, and recalculating the resulting forces and moments for the pool structure.

Fuel Storage Rack Load Combinations

The following are load combinations specified for the modified racks:

Elastic Analysis	Acceptance Limits
(1) D + L	Normal Limits of NF 3231.1a
(2) D + L + E	Normal Limits of NF 3231.1a
(3) D + L + T_o	Lesser of 2 S_y or S_u Stress Range
(4) D + L + T_o + E	Lesser of 2 S_y or S_u Stress Range
(5) D + L + T_a + E	Lesser of 2 S_y or S_u Stress Range
(6) D + L + T_a + E'	Faulted Condition Limits of NF 3231.1c

Definitions:

D - Dead loads or their related internal moments and forces including any permanent equipment and hydrostatic loads.

L - Live loads or their related internal moments and forces including any movable equipment loads.

E - Operational Basis Earthquake Loads

E' - Design Basis Earthquake Loads

T_o - Operating Thermal Loads

T_a - Accident Thermal Loads

Notes: (1) Reference code sections for to the ASME B&VP Code, Section III Division 1, 1980 Edition through Winter 1981 Addenda.

Loads and Loading Combinations

Table 9-14 lists the individual load components to which the spent fuel pool facilities were subjected. In all cases, except for fuel rack loads, the loads are consistent with the original loads used in the analyses performed and documented in the FSAR for each unit.

Table 9-14a describes the final load combinations investigated for both normal and accident conditions. These combinations are based on the strength design method.

Seismic Loading

Original plant response spectra and damping values were used to analyze the spent fuel facilities. For components that are classified as Seismic Category 2, methodologies and acceptance criteria may be different than for Category 1 components, but are sufficiently conservative to meet the intent of Regulatory Guide 1.29, C.2. Since only one horizontal and one vertical spectra were originally generated, the horizontal spectra was assumed to act in two horizontal directions. NUREG-0800 states that the three directions of the earthquake must be combined in an SRSS fashion; however, a strict SRSS procedure does not maintain the signs of the force components which are of obvious importance for concrete structures. Therefore, the requirements of NUREG-0800 were met by formation of resultant earthquakes based on the three directions of earthquake applied simultaneously.

Components of seismic loads include vertical earthquake effects, which are factored dead weight, plus vertical rack forces and the horizontal effects, which are pool wall inertia forces, pool hydrodynamic forces, building seismic forces, and fuel rack horizontal forces.

Structural Acceptance Criteria

The resulting forces and moments for the controlling load combinations were compared against the ultimate strength of the concrete sections. This comparison was initially carried out without thermal relaxation. All load combinations not involving thermal loads must pass this initial criteria check. Any load combination or element involving thermal loads, which meets the criteria without consideration of thermal relaxation, was eliminated from further processing. The load combinations failing to qualify on this first pass were selectively investigated by relieving the thermal moment as the section cracks. This procedure for post-processing ensured that the controlling load combinations were addressed in the most efficient manner possible.

Results

Using the above methods, models, loads, loading combinations, and acceptance criteria, the existing spent fuel facilities were determined to safely support the loads generated by the new fuel racks.

9.6.2.6 Handling

Layout of the fuel handling area in the auxiliary building is such that the spent fuel casks are never required to transverse the spent fuel pool during removal of spent fuel elements. Diverse electrical interlocks (a limit switch and a power disconnect from the main contact rails) are provided on the fuel handling crane to prevent an inadvertent transverse of the pool with a cask.

The electrical interlocks, crane, and crane rails are designed to stop the fuel handling crane from full speed, assuming that one electrical interlock fails and that no operator action is taken. At full speed, the crane travels a maximum of 25 fpm resulting in a minimal pendulum-like swing of the cask. The runway interlocks (the limit switch and power disconnect from the main contact rails) are positioned to interrupt the power and/or apply the brake to bring the crane to a stop with sufficient margin to ensure that the cask will not encroach on the spent fuel pool.

It is not therefore, considered credible that the spent fuel cask would ever fall into the spent fuel pool. This conclusion is based on the fact that it would require five independent concurrent failures to cause the cask to fall into the spent fuel pool (the five failures are as follows: 1 & 2) both, diverse, electrical interlocks must fail; 3) failure of the operator to stop the crane, 4) a mechanical device on the crane must fail, e.g. the crane hook; and, 5) administrative controls must have failed).

An analysis was performed on the 3-foot, 6-inch thick reinforced concrete relay room ceiling slab, located below the cask travel path between column lines A2 and C2. The analysis was performed to demonstrate that a postulated cask drop would not damage any safety-related equipment located in the relay room. The analysis followed an energy absorption method. The energy input to the relay room ceiling slab was based on a 260 kip cask weight, 92-inch cask diameter and a drop height of one inch. This considers that the main hoist is designed such that the maximum load motion following a single wire rope failure is less than 1.5 feet and the maximum kinetic energy of the load will be less than that resulting from one inch free fall of the maximum critical load.

The spent fuel cask arrives and is lifted from the train bay using the fuel handling crane until the crane operator stops the cask motion at or above the designed minimum height. The minimum raised height is based on the cranes required design features as discussed in Table 9-30.

The cask is then moved to the cask washdown area and if necessary, the cask is cleaned prior to lowering into the cask loading pit.

In order to transverse the floor area on Elevation 404’ just north of the Railcar Loading Hatch (Figure 9.1-14), the VSC-24 transfer cask with a loaded canister will utilize four slings. While not located over a safety related system, component, or structure, that section of spent fuel pool floor is 1 foot, 9 inches thick and not designed to withstand the static weight or drop of the cask. Instead of using a yoke to lift the loaded cask, the redundant slings are attached from the four trunnions of the transfer cask to the L-3 main hook. This dual path sling arrangement provides compliance with NUREG-0612 Section 5.1 regarding alternative measures to increase crane reliability by providing increased safety factor, redundant load paths, etc.

After the spent fuel is loaded into the cask, the cask is lifted from the loading pit until the crane operator stops the cask motion. If the crane operator is unable to stop the cask motion (either at this point or when the cask came up from the train bay) with the crane controls, the Failure Detection System of the X-SAM®Hoist will be initiated and subsequently will actuate the

Emergency Drum Brake, stopping cask motion. Crane operating procedures would require that all activities of the cask be immediately terminated and the cask lowered or held in position until the crane circuitry is evaluated or repaired. The “two blocking” interlock also provides an upper limit switch to stop cask motion.

The minimum height the cask may be above the spent fuel floor is eighteen inches. This is to ensure compliance with the cranes automatic actuation of the emergency drum brake system. The maximum amount of load motion and kinetic energy has been factored into the crane design to any potential load drop from impacting the spent fuel floor. The main hoist is designed such that the maximum load motion following a single wire rope failure is less than 1.5 feet and the maximum kinetic energy of the load will be less than that resulting from one inch of free fall of the maximum critical load.

For any heavy load over 29 tons, the single failure proof crane will be used to prevent a potential load drop from breaching the floor of the train bay at elevation 354′. While the possibility of a load drop is not considered credible, the off-site doses resulting from a fuel cask drop of 50 feet have been evaluated for a generic cask. The assumptions used in evaluation are as follows: (10 CFR 72 requires cask drop evaluation for specific cask system site application)

- Three years operating time.
- At design power on all assemblies.
- Cask movement occurs not sooner than 100 days after shutdown.
- Twenty-five percent of the noble gases is released.*
- Ten percent of the iodine is released.*
- Fifty percent of the iodine released plates out.**
- None of the solid fission products are released from the fuel.

$$\frac{X}{Q} = 1.2 \times 10^{-4} \text{ sec/m}^3 \text{ at site boundary}$$

* Based on Division of Reactor Licensing (DRL) staff evaluation of fuel handling accidents.

** The 50 percent plate from the iodine is a conservative value and is based on the fact that the door at the rail spur access to the plant is shut when the spent fuel cask is lowered. Iodine which escapes the cask resting area after the postulated drop would be plated-out on the walls, floors and ventilation piping.

No credit is taken for filtration from the auxiliary building ventilation system, since a portion of the ventilation flow path goes through the unfiltered turbine building. Further conservatism is evidenced in that an instantaneous release of the activity is assumed versus a "release-rate" approach.

The site boundary thyroid dose from this accident is 4.3 rem and the whole body dose is 0.0022 rem. These doses are below the limits of 10 CFR 100.

The dose to reactor operators in the control room due to a postulated 50-foot drop of the refueling cask is estimated, using the conservative fission product source term assumption described above, even though it is not expected that cask or fuel failure results from such a drop.

Due to the shielding provided by building concrete structures and the 1-foot, 6-inch concrete control room floor, ceiling, and walls, the dose contribution from refueling cask shine and airborne fission product shine external to the control room is negligible in comparison to the dose received from fission products postulated to enter the control room via its air intake duct.

Noble gas and iodine fission products escaping from the cask are assumed to escape instantaneously from the turbine building vent (without filtration) and to mix in the reactor building wake. A wind speed of 1.0 meters per second, a building cross section of 2,200 square meters, and a shape factor of 0.5 are used to determine the dilution factor. The control room intake air is conservatively assumed to be drawn from the building wake for the 2-hour incident duration, and instantaneous fission product mixing is assumed. Very conservatively, no credit is taken for control room isolation or recirculating high efficiency iodine and particulate filtration which are available.

The resulting estimated operator doses are approximately 17 millirem whole body and 30 rem thyroid for the duration of the postulated incident. [Note: The assumptions and offsite and control room dose results reported above for the hypothetical dropped fuel shipping cask are historical. ANO-1 does not presently ship spent fuel offsite and there are no plans to do so. Future transport of spent fuel for shipment offsite will require evaluation of the specific type and quantity of spent fuel being transported and NRC approval of the shipping cask used.]

The off-site doses which could result from the breach of a VSC-24 transfer cask dropped 50 feet was also evaluated. Assumptions used in this evaluation were the same as the previous analysis with exception of:

- Reg. Guide 1.25 Source Terms and Methodology
- 24 fuel assemblies
- Cask movement occurs not sooner than five years after the fuel is removed from the reactor (10 CFR 72 license condition)
- $X/Q = 6.5 \times 10^{-4}$ sec/m^3

The result of this analysis is a calculated site boundary thyroid dose of zero due to the short half-life of radioactive iodine. The estimated whole body dose for 24 assemblies with the stated accident conditions is 0.0119 rem site boundary and less than 0.002 rem in the control room. These values are well within the requirements of 10 CFR 100, Criterion 19 of Appendix A to the 10 CFR 50 General Design Criteria and the guidelines of NUREG-0612.

9.6.2.7 Operational Limits

Certain manipulations of the fuel assemblies and reactor internals during refueling may result in short-term exposures with radiation levels greater than 2.5 mrem/hr. The exposure time is limited so that the integrated doses to operating personnel do not exceed the limits of 10 CFR 20.

The fuel handling bridges are limited to handling of fuel, control rod and orifice rod assemblies and failed fuel transfer containers only. All lifts for handling the reactor head and reactor internals use the reactor building crane.

Travel speeds for the fuel handling bridges, masts, and the fuel transfer carriage are controlled to ensure safe handling conditions.

9.7 PLANT VENTILATION SYSTEMS

9.7.1 DESIGN BASES

The plant is designed to provide maximum safety and convenience for operating personnel with equipment arranged in zones, so that potentially contaminated areas are separated from clean areas. The heating, ventilating, and air conditioning systems for the plant are designed to provide a suitable environment for equipment and personnel. The path of ventilating air in the auxiliary building is from areas of low activity toward areas of higher activity. Ventilating air is recirculated in clean areas only. All normal systems design is based on an outside air temperature of 95 °F DB, 78 °F W.B. in summer and 6 °F D.B. in winter. The equipment heat gains, lighting load, adequate air changes, comfort conditions, etc., are considered.

Radiation monitoring sensors are provided in the control room as well as in the other areas of the auxiliary building. The basis for locating the sensors is:

A. Recognizing potentially radioactive areas.

B. Considering areas where personnel traffic is most likely to occur.

9.7.2 SYSTEM DESCRIPTION AND EVALUATION

The reactor building ventilation system is discussed in Chapter 5 and shown on Figure 5-7. The remaining ventilation systems for the plant are discussed in this section and are independent of those used in any other area. The systems handling potentially contaminated air are all discharged to the reactor building flues (plant vents).

9.7.2.1 Auxiliary Building

The auxiliary building systems primarily utilize all outside air. The radwaste area air supply is partially cooled to achieve smaller equipment sizes, and to maintain a 105 °F ambient temperature. Air to the fuel handling floor area is supplied without any cooling in order to provide for a sufficient number of air changes during the fuel handling at a maximum temperature of 109 °F. Supply air to both systems is tempered during the winter to maintain a minimum room ambient temperature of 60 °F. Although the radwaste area ventilation system and fuel handling area ventilation system are not required to meet the single failure criterion, an analysis is provided in Tables 9-21 through 9-24.

The following are the auxiliary building ventilation systems seismic design classifications:

Seismic Category 1 Design

1. Control Room Emergency Air Conditioning and Air Filtration System
2. Makeup Pump Rooms Unit Coolers
3. Decay Heat Removal Rooms Unit Coolers
4. Electrical Equipment Room Emergency Chillers, VCH-4A and VCH-4B
5. Switchgear Rooms Unit Coolers, VUC-2B and VUC-2D

6. North and South Electrical Room Unit Coolers, VUC-14B and VUC-14D

7. North and South Battery/DC Room Unit Coolers, VUC-14A and VUC-14C

8. Emergency Diesel Generator Rooms Ventilation System

9. Reactor Building Penetration Rooms Emergency Ventilation System

10. Battery Room Exhaust Fan VEF-33

11. Switchgear Room Exhaust Fan VEF-35

<u>Seismic Category 2 Design</u>

1. Radwaste Area Ventilation System including the Reactor Building Penetration Rooms Normal Ventilation System

2. Fuel Handling Floor Area Ventilation System

3. Battery Room Ventilation System

4. Heating and Ventilating Equipment Rooms Ventilation System

5. Cable Spreading Room Cooling System

6. Relay Room Cooling System

7. Control & Computer Room Normal Air Conditioning System

8. Boiler Room Exhaust Fans

9. Switchgear Room Unit Coolers, VUC-2A and VUC-2C

10. Battery/DC Room Unit Coolers, VUC-13A and VUC-13B

11. South Electrical Room Unit Cooler, VUC-11

Seismic Category 1 design is provided to ensure that the equipment in the areas which must remain operable during a Design Basis Accident (DBA) is capable of functioning after a Design Basis Earthquake (DBE).

The auxiliary building is served by separate ventilation systems for the fuel handling area, the radwaste area, the non-radioactive area, and the control room area. These systems are shown on Figure 9-13.

The capacity of the fuel handling floor area ventilation system is based on the equipment heat gains and lighting loads and is designed to provide a minimum of 15 air changes per hour up to 15 feet above floor level.

The capacity of the radwaste area ventilation system is based on the equipment heat gains and is designed to provide a flow pattern which maintains movement of air from low to high activity areas in order to inhibit the spread of any radioactive contamination. The filter banks are sized in accordance with manufacturers' recommendations for maximum efficiency.

One supply unit serves the fuel handling area, and two supply units serve the radwaste area. The ventilation air from these areas is discharged to the reactor building flues (plant vents) through multi-filter units. The fuel handling area exhaust unit is approximately 23 feet, 10 inches long by 16 feet, 10 inches wide by 16 feet, 4 inches high. The radwaste area exhaust unit is approximately 23 feet, 10 inches long by 19 feet, 2 inches wide by 18 feet, 8 inches high. These exhaust units include roughing filters, high efficiency filters, and charcoal filters.

The ventilation air from the fuel handling and radwaste areas is continuously discharged through the exhaust filters. Since the supply air to these areas passes through roughing filters, clogging of the exhaust system filters is not anticipated. The fuel handling and radwaste ventilation exhaust systems each have redundant fans. If one fan fails, the other fan starts automatically, ensuring continuous ventilation of the areas and precluding any overheating of the charcoal filters. Failure of the single supply fan does not prevent adequate ventilation air flow as other supply paths may be used.

Figure 9-13 shows the reactor building flues (plant vents) for the fuel handling floor area and the radwaste area. Figure 5-7 shows the reactor building flue (plant vent) for the reactor purge air exhaust. The reactor building penetration rooms emergency exhaust air discharges through filters into the atmosphere through a common exhaust stack. The penetration room ventilation system consists of two fully redundant trains. The system operation is based on a lead and standby design. Upon receipt of an engineered safeguard system signal the lead equipment starts. Failure of this equipment to start or perform properly stops and isolates the train. These conditions, in conjunction with the engineered safeguard actuation signal, start the standby system. Both systems can be operated manually and their performance monitored in the control room ensuring full testability. It is considered that this system fully meets its intended purpose and provides a highly reliable, redundant system capable of meeting the single failure criteria. Refer to Chapter 6 for engineered safeguards.

The penetration rooms, fuel handling, radwaste, and reactor building purge exhaust lines are monitored separately for radiation by an isokinetic sample taken downstream of each filtering unit. Figure 11-4 provides an illustration of the exhaust flues with reference to the SPING radiation monitor which monitors each path.

<u>DECAY HEAT REMOVAL ROOMS</u> (ENGINEERED SAFEGUARD ROOMS)

Each decay heat removal room is cooled by two 100 percent capacity fan coil unit coolers (one redundant) located in the room. The unit coolers utilize service water and, therefore, the loss of the cooling system is not anticipated. The coolers are designed to maintain room temperature below 110 °F DB under normal conditions. The lead unit cooler is automatically energized when a decay heat removal or RB Spray pump is started. The standby cooler will automatically start if flow from the lead cooler is below its setpoint. Service water is supplied to these unit coolers on a continuous basis. Purge systems for each room are in use normally and are automatically secured if an ES actuation occurs. The purge air is discharged through the radwaste exhaust system. Ventilation dampers in the room boundaries are manually isolated during initiation of the recirculation phase of post-LOCA cooling.

MAKEUP PUMP ROOMS (ENGINEERED SAFEGUARD ROOMS)

A 100 percent capacity fan coil unit cooler is installed in each of the three interconnected makeup pump rooms. The makeup pumps remain operable without any of the unit coolers in operation. The unit coolers utilize service water.

ELECTRICAL EQUIPMENT ROOM EMERGENCY CHILLERS

The Class IE Electrical Equipment Room Emergency Chillers (EERECs) provide emergency cooling for the north and south emergency ES 4160V switchgear rooms, Battery/DC rooms, and the north and south electrical equipment rooms. The EERECs were not part of original plant design, but were installed after initial startup to address post-accident cooling for the associated electrical equipment areas. Because design and installation of the EERECs required substantial time to complete, two non-seismic, vital-powered cooling units (VUC-13A/B) were installed to temporarily offset the lack of qualified cooling for these areas, which remain available to support area cooling.

Chiller VCH-4A (green train) is located in Corridor 98. Chiller VCH-4B (red train) is located in the A3 4160V switchgear room. VCH-4A serves the northern rooms and the VCH-4B chiller serves the southern rooms. Due to the importance of maintaining switchgear temperatures, the EERECs are controlled as TS support equipment. Contingencies are provided in procedures for equipment failures or maintenance. The EERECs are currently designed to maintain a maximum temperature of 120 °F in the ES 4160V switchgear rooms, the electrical equipment rooms, and the battery/DC rooms during a LOCA. Post accident long term operator actions may be required to open doors to the South Electrical room to maintain temperatures within limits.

The EERECs are normally in standby. In the event that the temperature in any of the electrical rooms exceeds 95 °F, the associated chiller and fan coil unit coolers for the affected rooms will start. System operation can also be manually initiated at the local control cabinets.

VCH-4A and VCH-4B are York reciprocating, 9 cylinder water chillers having a nominal capacity of 18.3 tons each. The chillers operate on a conventional compression cycle. The electric motor driving the compressor is a 480 volt, 3 phase motor, powered from vital power sources. Cooling water for the condenser is supplied by the associated essential Service Water (SW) loop. Each chiller unit is designed to reject 304,610 btu/hr to 40 gpm of SW entering at 121 °F. Chilled water pumps VP5A/B are Crane-Deming, centrifugal, 1.5 horsepower, 31 gpm capacity pumps, which develop 50 feet of head. The pumps are powered from the same breakers as the compressors.

During an accident, several, if not all, normal cooling/ventilation systems may remain available. Based on the normal cooling systems that may be available, the EERECs may or may not be required during post-accident conditions. Even if offsite power is lost, the station Alternate AC Diesel Generator (AACDG) may be aligned to restore power to non-vital cooling/ventilation systems. Notwithstanding the AACDG capability, the safety-related Penetration Room Ventilation fans (VEF-38A/B) are expected to remain available during post-accident conditions. Although not seismically qualified, the South and North Battery Charger Room Normal Cooling units (VUC-13A/B) and North battery exhaust fan (VEF-34) may also be available post-accident. South battery exhaust fan VEF-33, normally used for battery room ventilation, has been upgraded to have the capability to provide safety related alternate cooling of electrical areas when VCH-4A or VCH-4B are out of service. VEF-33 is seismically qualified.

Calculations performed conclude that the required electrical equipment in the associated rooms would remain below qualified temperature limits with no normal or emergency room cooling available for a pre-accident condition of red train and green train EEREC inoperability, respectively. In other words, if the red train EEREC were removed from service to perform maintenance and an accident subsequently occurred thereafter, required electrical equipment would remain within qualified temperature limits for the entire post-accident mission time (30 days for most equipment) even if the green train EEREC is lost. The calculations rely, in part, on proceduralized compensatory measures associated with opening doors, aligning dampers, starting switchgear exhaust fan VEF-35 for alternate cooling, and verifying unnecessary equipment is de-energized. The most restrictive time for establishing any of these measures is within 2 hours post-accident, affording Operators sufficient opportunity to complete necessary actions.

With a loss of all cooling, rate of heat rise evaluations also indicate that sufficient time is available to restore cooling equipment post-accident prior to electrical equipment in the associated rooms exceeding qualified temperature limits. In addition, if room cooling is not restored, the room temperature limits are not exceeded by a significant margin.

4160V SWITCHGEAR ROOMS (ENGINEERED SAFEGUARD ROOMS)

Each switchgear room is normally cooled by air circulated through fan coil unit coolers (VUC-2A and VUC-2C) supplied with chilled water from the control room chillers. Emergency cooling is provided by separate fan coil unit coolers (VUC-2B and VUC-2D) supplied by emergency chillers VCH 4A and 4B.

NORTH AND SOUTH ELECTRICAL ROOMS (ENGINEERED SAFEGUARD ROOMS)

The north vital electrical equipment room is normally cooled by the radwaste auxiliary building ventilation system and the south vital electrical equipment room is normally cooled by an air-cooled refrigeration unit. Emergency cooling is provided by separate fan coil unit coolers (VUC-14B and VUC-14D) supplied by emergency chillers VCH-4A and VCH-4B.

BATTERY/DC ROOMS

The redundant battery rooms have independent vital power backed exhaust fans VEF 33 and VEF 34. VEF-34 is not safety related, but VEF-33 has been upgraded for safety related service. These battery and DC rooms are normally cooled by vital power backed non-safety related air-cooled refrigeration units VUC-13A and VUC-13B. Emergency cooling is provided by separate fan coil unit coolers (VUC-14A and VUC-14C) supplied by emergency chillers VCH-4A and VCH-4B.

EMERGENCY DIESEL GENERATOR ROOMS

The emergency diesel generator rooms are each ventilated by two exhaust fan units. Makeup air to these rooms is 100 percent outside air. The capacity of each fan is based on the equipment heat gains within the room. With two fans running the room ambient temperature should be 110 °F DB maximum. In the event of failure of one fan the room is maintained at a temperature not exceeding 120 °F maximum by the other fan. The emergency diesel electrical equipment and controls can operate at 100 percent diesel load and will not be affected by an ambient design temperature of 120 °F.

CONTROL ROOM AIR CONDITIONING

NORMAL VENTILATION SYSTEM

The control room normal ventilation system recirculates air in addition to the outside air which is supplied by the air conditioning unit. The mixed air is cooled to eliminate excess moisture, latent heat and sensible heat gains, and then is tempered to the needs of the individual room, as necessary.

The normal control room air conditioning system serves the control room and computer room only.

The control room and computer room air is recirculated in order to maintain cleaner air and to economize on cooling and heating loads. Adequate fresh air makeup is provided in the system. The control room design is based on the space load, lighting load and equipment load with a room ambient temperature of 75 °F DB and 43 percent relative humidity.

The control room and computer room are normally air conditioned by two 100 percent capacity air conditioning units receiving chilled water from two 100 percent capacity chillers. One air conditioning unit is normally running, with the other in standby status isolated from the system by shutoff dampers. The standby unit is available for manual actuation in the event of failure of the operating unit. The fan failure is monitored by a flow switch at the unit with an indicating light in the control room.

CONTROL ROOM ISOLATION SYSTEM

The Control room air is continuously monitored with alarms for high radiation. The control room inlet air radiation monitor system consists of two identical monitor strings each having an auto ranging digital ratemeter, pre-amplifier, and Beta-Gamma sensitive scintillation detector. These monitors have a minimum detectable level of 1E-5 μCi/cc of Cs-137 with no lead shield. A variable setpoint for the monitor is set slightly above equilibrium background level and alarms are provided for high radiation and circuit failure. The configuration is such that either monitor can isolate the control room for high radiation or circuit failure conditions. Redundant quick acting chlorine detectors are presently in place in the control room fresh air inlets which institute closing of the isolation dampers upon attaining 5 ppm Cl_2 levels in the incoming air. The detectors are able to detect and signal a step increase in chlorine concentration of 0 to 15 ppm within 5 seconds (based on 66% response) (1CAN018008).

Chlorine detection system design features are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975. However, since elemental chlorine is no longer stored or used on site or within a 5-mile radius of the plant site, the regulatory guide recommendation for seismic category I designation is not necessary for ANO design requirements. A postulated seismic event concurrent with transport failure and release of chlorine or other toxic gas offsite is considered an incredible event.

In the event of high radiation or reading a chlorine level of 5 ppm, the normal air conditioning system is automatically de-energized and the normal control room ventilation system is completely isolated from both the outside air and the rest of the building within 5 seconds after the detector trip signal is received. The actuation level for high radiation is sufficiently below hazardous radiation levels to minimize operator dose during an accident and is sufficiently above

normally experienced background levels to minimize spurious actuations. The control room isolation dampers in the supply and return ductwork are spring loaded such that they fail closed upon loss of air or power. The single supply and single return isolation dampers are each actuated by either of two solenoid valves. In the event of an actuation due to chlorine level, the control room emergency air filtration system is manually initiated. Under these conditions control room air is recirculated by the emergency air filtering system. The emergency air filtering system consists of two redundant filter trains, VSF-9 and 2VSF-9. Due to space limitations, the two trains are designed differently. One filter train (2VSF-9) consists of a fan, roughing filters, HEPA filters, and a 4-inch deep bed charcoal adsorber rated for 2000 cfm. The other train (VSF-9) consists of a fan, one filter unit assembly rated for 2000 cfm with an outside air filter unit rated for 333 cfm, each with the necessary roughing filters, HEPA filters and 2-inch charcoal tray adsorber. Both VSF-9 and 2VSF-9 were originally designed to provide ~333 cfm outside air to minimize unfiltered air inleakage to the combined control room envelope (CRE), which was in turn based upon providing greater than or equal to 0.5 volume changes per hour based upon Standard Review Plan 6.4, Rev.2 (July 1981). However, the actual outside air drawn by 2VSF-9 is ~465 cfm, as measured during control room tracer gas testing. Calculations have been performed that indicate that even with the higher 2VSF-9 makeup air flow rate, operation of VSF-9 with 333 cfm makeup air is still limiting in terms of control room radiation dose (1CAN060302). For either train outside air will be filtered through four inches of charcoal adsorber and the recirculation air will go through at least two inches of charcoal bed. Fan failure is monitored by a flow switch with an indicating light in the Control Room. On an indication of fan failure, the standby unit will be manually started.

EMERGENCY VENTILATION SYSTEM

The emergency control room air conditioning system design is based on heat gains from equipment, occupancy and lighting load. The space load is included based on 105 °F DB temperature for areas surrounding the control room.

The original DBA maximum emergency cooling pond temperature of 129.5 °F discussed in Section 9.3.2.4 exceeded earlier estimates of past ECP temperature. This change required the replacement of Unit 2 emergency air conditioning units with larger units capable of operating with 129.5 °F cooling water. The new control room emergency air conditioning units are located in the Unit 2 control room and provide emergency air conditioning to both Unit 1 and Unit 2 control rooms and provide for air mixing during a control room isolation condition. Seismically supported duct work has been added for air distribution to the Unit 1 control room.

In conjunction with installation of new control room emergency air conditioning units, an intertie has been provided from Loop 2 of the Unit 1 service water system to the air conditioning condensing units, to provide an alternate source of service water to the emergency air conditioning system.

The worst case outside environment assumed for this analysis is 103 °F DB, 83 °F WB and 43 percent relative humidity maximum and is based on records of ambient conditions at the site. The control room emergency recirculation system is based on a minimum of three room air changes per hour for the combined control room volume (1CAN018008). The filter banks are sized in accordance with manufacturer's recommendations for maximum efficiency. The control room operator has manual control for selecting fan, filter, and air conditioning unit operations in order to ensure satisfactory control room conditions following an accident. Self-contained breathing apparatus (SCBA) are available for use following a toxic gas release.

All portions of the reactor protection and engineered safeguards actuation systems located in the control room are designed to operate in ambient conditions of 110 °F and 80 percent relative humidity.

The heat loads within the reactor protection system and engineered safeguards actuation system cabinets are low and the cabinets do not require specific air conditioning ducting. Individual component and modules of the reactor protection system are specified to require factory testing at design temperatures and humidity conditions.

Fire or smoke in the control room could be detected visually by the operator. The valves that isolate the control room from the other areas close in five seconds. This prevents significant quantities of smoke from entering from outside the control room. In the unlikely event of a fire in the control room the smoke is exhausted from the room to the outside of the building and makeup air is supplied to the room by the normal ventilation system. The system is sized such that it provides 15 air changes per hour. This ventilation rate rapidly dissipates any smoke generated or admitted to the control room.

A failure analysis was performed to demonstrate the ability of the control room emergency air conditioning system to meet single failure criterion. The analysis is shown in Table 9-20.

COMPUTER ROOM NORMAL AIR CONDITIONING

Computer equipment cooling is achieved by two out of three packaged air conditioning units (one standby) which are located in the computer room and circulate air through the false floor. The computer room space air conditioning is performed by a supply air and a return air zone from the control room normal air conditioning system.

The design of the room air conditioning is based on the space load and lighting load with room ambient temperature of 75 °F DB and 43 percent relative humidity. When the control room normal air conditioning system is de-energized for control room isolation, the computer room has no air conditioning.

ELECTRICAL ROOMS

During operation, the cable spreading room is cooled by a recirculation unit (VUC-3) located outside the room. The relay room has two recirculation type cooling units (VUC-4A, VUC-4B), one of which is a standby unit. The rooms are designed for an ambient temperature of 85 °F during normal operation. Also, a small portion of air is supplied to the relay room from Unit-2 supply fan 2VSF-6 for pressurization of the relay and cable spreading room to prevent any in-leakage from the turbine building. The air leaks to the turbine building area and is not recirculated in the system.

GENERAL

The ventilating equipment is designed and constructed in accordance with accepted industry codes and standards for power station equipment and is accessible for periodic testing and inspection during normal operation. Where redundant equipment is provided, it is operated alternately to provide assurance of operability.

The safety related high efficiency particulate filters and the charcoal filters are tested in accordance with the guidance in USNRC Regulatory Guide 1.52 as specified in the ANO-1 Technical Specification.

9.7.2.2 Turbine Building

The turbine area ventilation system recirculates air with provisions for makeup as required from fresh air louvers. Exhaust air is discharged directly to the atmosphere through roof ventilators. The building is designed for 105 °F maximum and 60 °F minimum. The non-safety related battery room has a dedicated HVAC system designed to maintain a nominal 77 °F ambient temperature during normal plant operating conditions.

9.7.2.3 Intake Structure

The intake structure ventilation system utilizes 100 percent outside air. A smaller capacity exhaust fan is used for ventilation during the winter than is used during the summer in order to minimize electric heating. Emergency (DBA) cooling utilizes natural convection via outside air entering and exiting through architectural openings.

9.8 FIRE PROTECTION SYSTEM

9.8.1 DESIGN BASIS

Fire protection for ANO-1 was originally designed and installed to satisfy the requirements of building codes, OSHA, NFPA and NEPIA. This resulted in significant use of fire barriers, fire doors, sprinkler systems and hose stations. As a result of the Browns Ferry fire of March 1975, the ANO Fire Protection Program was improved, including many changes to satisfy NRC criteria in BTP9.5-1, Appendix A to BTP9.5-1, Appendix R, Sections G, J, O, and L by reference from G.3 to 10 CFR 50, and other staff positions. Subsequently, a license amendment request was submitted and approved by the NRC to transition the ANO-1 fire protection license basis to General Design Criteria 3 of 10 CFR 50, Appendix A, and NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, in accordance with 10 CFR 50.48(c). The ANO-1 Fire Protection Program includes the following:

A. Structures, systems and components important to safety are designed and located to minimize, consistent with other safety requirements, the fire hazard. Noncombustible and heat resistant materials are used wherever practical throughout the unit. This requirement is in compliance with 10 CFR 50, Appendix A, General Design Criterion 3, Fire Protection.

B. The Fire Protection System (FPS) is designed to minimize the effects of fires on structures, systems and components important to safety, in accordance with 10 CFR 50, Appendix A, General Design Criterion 3, Fire Protection. It is designed to provide adequate capability to fight the fire hazard encountered in all plant areas.

C. The FPS is designed so that pipe rupture or inadvertent operation does not cause loss of function of plant structures, systems and components important to safety, in compliance with 10 CFR 50, Appendix A, General Design Criterion 3, Fire Protection.

D. Procedural controls are established to limit the use of combustible materials and to prevent potentially hazardous situations. A description of these controls is contained in Section 9.8.3.4.

E. Hydraulically balanced automatic sprinkler systems, ordinary or extra hazard automatic sprinkler systems, and hydraulically designed automatic water spray systems are installed in all areas with a high fire potential, or where such protection is required due to a high concentration of safe shutdown related cabling.

F. Inside hose connections, hose reels and manual fire-fighting equipment are provided. A fire brigade is staffed and trained in order to provide prompt response to fires and to backup automatic suppression systems. A description of the fire brigade training and staffing is contained in Section 9.8.3.

G. Portable fire extinguishers are provided throughout normally accessible areas of the plant in accordance with applicable NFPA, OSHA, property insurer, and NRC regulations and recommendations.

H. Alarms are provided in the control room upon activation of automatic fire protection systems. Fire and smoke monitoring and detection systems are installed in safety-related areas and in the supply and return ventilation ductwork of the Control Room/ Computer Room. These systems alarm in the control room, and if personnel can be in the vicinity, locally.

I. A fire protection water supply system jockey pump is provided to minimize cycling of the main fire pumps.

J. In areas where redundant safe shutdown related cabling are in close proximity to each other such that they could both be damaged in a fire, additional protective measures are provided such as thermal barriers, cable coating, or extensive fast-acting directed water spray systems.

Fire Protection System piping was designed to meet NFPA construction requirements at the time of installation. In addition, much of the fire protection piping in the Seismic Category I structures has been designed and constructed in accordance with the more stringent requirements of the ANSI B31.1 Power Piping Code. Further, some of the piping near Seismic Category I equipment, such as the fire hoses and associated standpipes inside containment, has been analyzed to the requirements of Seismic Category I piping to provide reasonable assurance that they are capable of withstanding a seismic event.

9.8.2 SYSTEM DESCRIPTION AND EVALUATION

GENERAL DESCRIPTION

The plant fire protection system is comprised of diversified monitoring, detection, alarm, suppression, and extinguishment facilities particularly selected to protect the area or equipment from damage by fire, and include, among other things, the following major features:

A. Fire protection water supplies, yard mains and hydrants.

B. Automatic wet sprinklers, hydraulically designed, ordinary and extra hazard.

C. Deluge and pre-action systems, hydraulically designed.

D. Water spray systems, hydraulically designed.

E. Standpipes and hose reels.

F. Automatic CO_2 extinguishing system and automatic Halon suppression system.

G. Portable CO_2, dry chemical, water spray, and Halon extinguishers.

H. Fire and smoke monitoring, detection and alarm systems.

I. Fire barriers, seals, and penetrations.

A simplified schematic is provided in Figure 9-16.

Equipment and components that are required for fire protection have been designated as ″F Listed″ in the controlling plant database (such as equipment database/component database).

COMPONENT DESCRIPTION

The components of the plant fire protection system are selected to provide comprehensive protection against fire hazards throughout the plant with greatest emphasis placed on the risk of fire in the component's immediate location. The following provides a general description of the plant fire protection systems that may be credited to support a basis for engineering equivalency evaluations (EEEE), defense-in-depth (DID), or risk. Detection and suppression systems required for EEEE, DID, or risk are specifically noted in the Fire Hazards Analysis (FHA).

FIRE PROTECTION WATER SUPPLIES, YARD MAINS, AND HYDRANTS – The fire pumps and accessories are located in the Category 1 portion of the intake structure. The main electric motor driven fire pump and the diesel engine driven fire pump are each designed to supply 2,500 gpm at 125 psig. The jockey pump is designed for 90 gpm at 125 psig. The jockey pump starts and stops automatically to maintain approximately 118 to 134 psig pressure in the fire loop. The electric motor driven fire pump starts if the line pressure drops to approximately 110 psi and the diesel engine driven fire pump starts if the line pressure drops to approximately 90 psi.

The yard area of the plant is protected by a 12-inch fire water loop around the area and is provided with Underwriters Laboratories listed hydrants and hose houses located at approximately 250-foot intervals. Sectional control valves are provided on each side of the pump connections to the loop and at approximately quarter points of the loop to provide isolation of sections of the yard piping should it require maintenance or be taken out of service.

Isolation valves in supply lines to suppression systems are provided with seals or tamper switches to alarm if the valve is closed.

DELUGE AND PRE-ACTION SYSTEMS – The Deluge/Water Spray Systems which protect the diesel fuel storage vaults, the cable spreading room, and the corridor south of the cable spreading room consist of dry pipe open head sprinkler systems which are actuated by the opening of a fire water control valve associated with each system. Each of these valves can be manually actuated at the valve or in the control room, with the exception of those valves in the Diesel Fuel Storage Vault. In addition, the valves may be automatically actuated by the following detection scheme:

Diesel fuel storage vaults	Two out of two smoke detectors
Cable spreading room	Any one line heat detector and one smoke detector
Corridor south of cable spreading room	Any one line heat detector and one smoke detector

The Preaction Fire Suppression Systems which protect portions of the auxiliary building and electrical penetration rooms consist of fusible head dry pipe sprinkler systems. The associated control valves can be manually activated at the valve or from the control room. In addition, the valves may be automatically actuated by the following detection scheme:

Emergency diesel generator rooms	Any one smoke detector and one flame detector
Auxiliary building electrical penetration rooms	Any single smoke detection zone

For the Preaction Suppression Systems, water will only spray when sufficient heat is generated to melt the fusible heads.

HALON SUPPRESSION SYSTEMS – The Halon Systems which protect the control room ceiling, the auxiliary control room ceiling, and the false floor consist of piping and distribution nozzles which are activated by the opening of a system control valve for each system. These valves can be manually activated at the valve or in the control room. In addition, the valves may be automatically actuated by the following detection scheme:

Control room ceiling (main control room area)	One smoke detector on each of two loops
Auxiliary control room ceiling	One smoke detector on each of two loops
Auxiliary control room floor	One smoke detector on each of two loops

WET PIPE SYSTEMS – Reactor building electrical penetration areas are provided with fusible head wet pipe sprinkler systems. The outside and inside motor operated valves must be actuated remotely from the control room. An interlock is provided for those wet pipe sprinklers inside the reactor building to prevent the flow of fire water upon an ES channels 3 or 4 actuation signal to avoid diluting the borated containment sump inventory with unborated water.

A wet pipe suppression system is provided at Elevation 354' to protect redundant cables in the Intake Structure Building. Also, a wet pipe suppression system is provided in fire pump room area, Elevation 366' to protect the Diesel Fire Pump in the Intake Structure Building.

A wet pipe sprinkler system is provided to protect redundant cables in the condensate demineralizer area, Turbine Building, el. 354′.

In addition to the above, sprinkler systems are provided throughout the plant for fire protection of non-safety-related equipment. For example, the transformers and switchyard are protected by an automatic deluge system which receives water from the Fire Protection System. The turbine lubricating oil storage tanks, the turbine lubricating oil reservoir and the building wall next to the transformers are protected by approved automatic deluge systems. The floor area under the turbine on the lubricating oil piping side of the turbine pedestal, the oil piping, the fuel oil storage tank, the intermediate floor and grating floors are protected with fusible head sprinkler systems.

FIRE AND SMOKE MONITORING, DETECTION AND ALARM SYSTEMS – Smoke detectors are also located throughout the plant where safety-related equipment and cabling are located. These additional detectors and sprinkler systems alarm in the control room when activated. Required flame, heat, or smoke detectors are listed in the Technical Requirements Manual (TRM). Smoke and heat detectors inside containment have been provided with smoke/heat collector panels where the detectors are located below grating. (1CAN047821)

STANDPIPES AND HOSE REELS – The interior of the buildings is protected by hose stations located so that the entire area can be reached by a hose stream with 100-foot lengths of the fire hose.

The reactor building contains 10 hose stations having 100 feet of one and one-half inch CRL U.L. Listed fire hose with adjustable nozzles. These stations are fed through two (series) isolation supply valves tied into the existing reactor building sprinkler system. Each hose station is individually valved. This is, in effect a dry system. (1CAN017910)

PORTABLE FIRE EXTINGUISHERS – Additional protection is available from strategically located portable fire extinguishers that meet the requirements of NFPA 10, “Standard for Portable Fire Extinguishers” (2007 Edition).

MANUAL FIREFIGHTING EQUIPMENT – In addition to hose stations and portable fire extinguishers, manual fire-fighting equipment has been provided to assist fire brigade fire-fighting activities. This equipment includes:

A. Portable radio communication equipment;

B. Portable smoke exhaust units with flexible ductwork;

C. Emergency self-contained breathing apparatus;

D. Battery powered sealed beam portable lights; and,

E. Fire-fighting foam and foam nozzle.

The use of this equipment in specific fire areas/zones is contained in the ANO Pre-Fire Plan.

OTHER DESIGN CONSIDERATIONS FOR MINIMIZING THE POTENTIAL FOR FIRES

In accordance with NFPA 805, a risk-informed, performance-based analysis was performed that ensures key safety functions (i.e., reactivity control, inventory and pressure control, decay heat removal, and vital auxiliaries) will be maintained in the event of a fire. See the FHA for more detail.

The design of the FPS was considered in the analysis as a means to minimize the effects of fires and to provide the capability to extinguish a fire encountered in any of the plant fire hazard areas. The basis fire prevention and protection is achieved by physical separation of systems or by fire barriers between such installations. The functional requirements for fire barriers are covered in the TRM.

Fire doors which separate redundant safe shutdown equipment, or which separate safe shutdown equipment from large oil hazards, are either locked, provided with electrical supervision, or inspected to ensure that they are closed (1CAN047903). The functional requirements for fire doors are covered in the TRM.

The reactor coolant pump oil collection system provides collection capability at all potential leakage points by the addition of drip pans, deflectors and drain lines with the use of a holding tank, overflow tank, and curbed area surrounding the collection system to collect and drain oil that might leak from reactor coolant oil reservoirs, external piping, flanged connections, oil gauges, or motor mounted oil fill connections.

9.8.2.1 Codes and Standards

Fire Protection Systems consist of Underwriter’s Laboratories Listed or FM Approved equipment unless otherwise approved by Fire Protection Engineering. The Fire Protection Systems are designed considering the requirements of the National Fire Protection Association (NFPA) and the site property loss insurer. As part of the implementation of NFPA 805, the NRC approved a deviation from NFPA 20, Sections 457 and 511c, which requires the electrical fire pump motor and electric fire pump controller to be UL Listed / Approved for fire pump service. In addition, the

NRC approved a deviation from the requirements of NFPA 20, Sections 626a, 626d.e2, and 626d.e5 for the Cummins Diesel Engine controller, since vendor documents do not identify a certification for the batteries and do not identify the discharge rate of the lead acid batteries.

9.8.2.2 Fire Hazard Analysis

The purpose of the FHA is to evaluate potential fire hazards and appropriate fire protection systems and features used to mitigate the effects of fire in any plant location.

To evaluate the potential effects of major fires on redundant safe shutdown equipment, the plant was divided into fire areas that are separated by 3-hour or adequate for the hazard fire-rated barriers. The fire areas were further divided into fire zones or physical analysis units by masonry partitions, non-combustible partitions or clear space. Penetrations of fire barriers, such as doorways, cable tray or conduit penetrations, and ventilation penetrations are protected to a rating equivalent to that of the barrier or evaluated using the guidance specified in NFPA 805. Three-hour rated dampers and fire doors have been installed in ventilation duct and doorway penetrations of fire barriers. Cable tray penetrations of fire barriers have been upgraded to a design tested configuration to demonstrate a 3-hour fire rating. Piping and conduit penetrations are sealed around the piping and conduit to prevent smoke transmittal; large conduits are also sealed internally.

All safety-related circuitry and equipment within each fire zone were analyzed in the fire hazards analysis along with the potential fire loading within the zone. The potential for a fire to cause loss of safe shutdown functions was evaluated.

In the initial Appendix R analysis, where it was determined that redundant safe shutdown cables were in sufficient proximity that they could be lost in a fire, modifications were made to assure that at least one of the redundant cables would remain functional for a fire that did not receive prompt suppression. Subsequently, when transitioning to NFPA 805 additional modifications were implemented to ensure the plant could be maintained in a safe and stable condition (i.e., Mode 3).

An evaluation was also performed with regard to the potential for a fire to degrade safe shutdown systems through associated circuits. There are two routing configurations of associated circuits (circuits connected to safety systems but which perform non-safety functions) that could possibly cause concurrent loss of redundant associated circuits by fire: routing in adjacent raceways or routing in a common raceway. Neither configuration is precluded by the separation criteria applied to ANO-1 for Class 1E circuits.

The failure modes of circuits considered in the evaluation were open circuits, hot shorts, short circuits and grounds. Hot shorts are defined as shorts which take place when conductors come into physical contact without operating an overcurrent protective device. The evaluation found that although several cases were identified where associated circuits for redundant divisions are located in adjacent raceways or in a common raceway, in all cases the associated circuits are electrically isolated from the Class 1E position of the circuit by contacts. Therefore, failures of associated circuits, including hot shorts, will not cause loss of safety system function.

As allowed by 10 CFR 50.48(c), ANO-1 revised the fire protection licensing basis to comply with NFPA 805 (1CAN011401). The NRC issued a Safety Evaluation Report approving the revised license (1CNA101601). The methods of assumptions used in performing these analyses are described in the FHA, SAR Appendix 9B.

9.8.2.3 Failure Considerations

The electric motor driven fire pump and the diesel driven fire pump each take 2,500 gpm of water separately from the intake structure. Each pump has a separate compartment in the intake structure and they are separated from the Category 1 service water pumps by a Category 1 wall. The fire system piping is physically separated from critical systems to preclude damage to these systems except as discussed below. Water to the intake structure comes from either the Dardanelle Reservoir or, in an emergency, from the emergency cooling pond.

The Fire Protection System can operate with any single failure. Failure of the jockey pump has no effect on the Fire Protection System as the main fire pump starts when the line pressure decreases to a pre-set pressure point. Failure of the main electric driven 2,500 gpm pump starts the diesel driven 2,500 gpm fire pump. All branches off the main fire piping yard loop are equipped with block valves which can be closed in case of failure of a branch line. The yard loop is equipped with sectional control valves to provide isolation of sections of the yard piping in case of failure of any section.

In the event of failure of the piping in the intake structure and the continued operation of the fire pumps, the water drains through openings in the floor back into the intake water pool, thereby precluding flooding of the Category 1 equipment. The jockey pump starts and stops, but the quantity of water handled by this pump is only 90 gpm and rapidly drains away. Failure of any non-seismic part of the Fire Protection System cannot damage a Category 1 structure. In case of a complete failure of the Fire Protection System, Category 1 structures can be protected from fire by service water, portable fire extinguishers, and by fire equipment called from the local fire department.

To decrease the probability that a fire protection sprinkler or piping system malfunction or failure affects vital equipment, fire protection sprinkler and piping systems are generally located in areas where there is no safety-related equipment which can be damaged by water (see Figure 9-16). Fire protection sprinkler and piping systems are located, however, in the reactor building, emergency diesel generator rooms, electrical penetration areas, diesel fuel storage tank vaults, cable spreading room, emergency feedwater pump room, and the corridor south of the cable spreading room. Sprinkler nozzles are located where they will not directly impinge on equipment required for safe shutdown.

In areas that are provided with water spray systems, the floor drains have been upgraded to be able to prevent buildup of water that could damage vital equipment. Areas where these larger drain systems have been provided are the electrical equipment room (Elevation 368 of the auxiliary building), the cable spreading room, the auxiliary building corridor (Elevation 372), and the upper north electrical penetration room.

In addition, the main Fire Protection Header in the intake structure has been analyzed to the Seismic Class 1 requirements, minimizing the probability of a major failure in this area. Piping near P-6A and P-11 was analyzed to an even more stringent stress requirement to provide assurance that vital equipment located in the area would be protected from pipe cracks and pipe ruptures.

9.8.3 FIRE PROTECTION PROGRAM

The Arkansas Nuclear One Fire Protection Program was developed to define the organizational responsibilities, procedural controls, fire brigade staffing and training, and the quality assurance provisions that have been established for both Unit 1 and Unit 2. The overall objective of the fire protection program is to minimize both the probability and consequences of postulated fires and to maintain the capability to safely shut down the plant if a fire should occur. The following sections comprise the elements of the Arkansas Nuclear One Fire Protection Program.

Entergy Operations' organizational responsibilities for fire protection are provided in the SAR Section 12.1.1.

9.8.3.1 Organizational Responsibilities

The quality program for fire protection is designed to comply with the requirements of the Quality Assurance Program Manual (QAPM) and with the quality assurance guidelines identified in BTP-APCSB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants, subject to exceptions noted in this Appendix.

This quality program is to ensure that the fire protection systems for safety-related areas are controlled in accordance with applicable NRC regulations, industrial standards and codes, policies, rules, procedures and licensing documents. The quality program is implemented through approved procedures. The effectiveness of the fire protection program is verified through surveillances and scheduled audits conducted by the Quality Organization, under the cognizance of the SRC. General requirements for this program are also described in Section G.1 of the QAPM. Personnel performing inspections need not be certified to ANSI N45.2.6, when inspections are performed on equipment not listed on the Q-list.

9.8.3.1.1 Design Control

Sections B.2 and B.3 of the QAPM are applicable for design control activities pertaining to fire protection systems.

9.8.3.1.2 Procurement Document Control

Sections B.4 and B.5 of the QAPM are applicable for procurement control activities pertaining to fire protection systems. When these items are not associated with the Q-list, the procurement document is to include the requirement that items be U.L. listed or F.M. approved for fire protection use, where applicable, in accordance with approved procedures. In addition, new fire protection systems are subject to the most current code or standard.

9.8.3.1.3 Instructions, Procedures, and Drawings

Sections A.1.d and B.1.c of the QAPM are applicable to instructions, procedures and drawings pertaining to fire protection systems.

9.8.3.1.4 Document Control

Section B.14 of the QAPM is applicable for the control of quality program documents related to fire protection.

9.8.3.1.5 Control of Purchased Material, Equipment, and Services

The control of purchased fire protection-related materials, equipment and services is described in Sections B.4 and B.5 of the QAPM, with the following exceptions. For the procurement of fire protection-related items or services not associated with the Q-list, the vendor/contractor qualification criteria (including periodic reassessment of their program) is not required. Nonconformances dispositioned repair or use-as-is by the vendor are to be submitted to and accepted by ANO only when so designated on the procurement document.

9.8.3.1.6 Identification and Control of Materials, Parts, and Components

Section B.6 of the QAPM is not applicable to the identification and control of materials, parts and components pertaining to fire protection systems. The identification and control of materials, parts and components are conducted in accordance with existing procurement and materials management procedures and practices.

9.8.3.1.7 Control of Special Processes

Section B.11 of the QAPM is not applicable for the control of special processes, as applicable to fire protection systems. The control of special processes for the maintenance of the fire protection system is performed in accordance with applicable approved procedures and practices.

9.8.3.1.8 Inspection

Inspection activities applicable to the fire protection system are described in Section B.12 of the QAPM with the exception of Section G.1, when inspections are performed on equipment not associated with the Q-list. Personnel performing inspections need not be certified to ANSI N45.2.6.

In addition to the provisions of the QAPM, inspections and surveillances are addressed in applicable portions of the SAR for each nuclear unit.

9.8.3.1.9 Test Control

A test program is to be established and implemented to ensure that testing is performed and verified on applicable systems and components to demonstrate conformance with design and system readiness requirements. The tests are to be performed in accordance with written test procedures and test results evaluated for conformance to the test objectives.

The control of testing activities is described in Section B.8 of the QAPM. Surveillance testing requirements are identified in the Safety Analysis Report for each nuclear unit.

9.8.3.1.10 Control of Measuring and Test Equipment

Section B.9 of the QAPM is not applicable for the control of measuring and test equipment. Measuring and test equipment is controlled in accordance with applicable approved procedures and practices.

9.8.3.1.11 Handling Storage and Shipping

Section B.7 of the QAPM is not applicable for the handling, storage and shipping of fire protection-related materials and equipment. Handling, storage and shipping activities are controlled in accordance with applicable approved procedures and practices.

9.8.3.1.12 Inspection, Test, and Operating Status

Measures are established to provide for the identification of items that have satisfactorily passed required inspections and tests and are documented per approved instructions or procedures.

Section B.10 of the QAPM is applicable for identifying the inspection, test and operating status of the fire protection system.

9.8.3.1.13 Nonconforming Material, Parts, and Components

The control of nonconforming materials, parts and components related to the fire protection system is described in Sections A.6, B.6 and B.13 of the QAPM. Vendor nonconformances are to be submitted to ANO only when so designated on the procurement document.

9.8.3.1.14 Corrective Actions

A corrective action system is established to ensure that conditions adverse to fire protection, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible materials and nonconformances, are promptly identified, reported and corrected.

Corrective action activities are controlled as described in Sections A.6 and B.13 of the QAPM, with the exception that, vendors furnishing fire protection items not associated with the Q-list are not required to be listed on the QSL.

9.8.3.1.15 Quality Assurance Records

Records which furnish evidence that the criteria enumerated in this program are being met for activities affecting the fire protection program are to be prepared and maintained as described in Section B.15 of the QAPM.

9.8.3.2 Fire Brigade Training

The Arkansas Nuclear One Fire Brigade Training Course insures that the capability to fight potential fires is established and maintained. The program consists of initial classroom instruction outlined in the Plant Procedures and periodic instruction in the form of classroom sessions, fire drills and practice in fire-fighting. This training is intended for all Fire Brigade members.

9.8.3.2.1 Classroom Instruction

A. The initial classroom instruction includes the following:

1. The course identifies the various classes of fires and types of combustibles involved, citing examples of specific materials in each class. This identification and location of all such hazards cannot be realistically covered in the initial classroom training. The training does cover some specific hazard locations but its thrust is to generate an

awareness in the training of what types of materials constitute a hazard. The trainee can then relate this information to materials he sees in the plant during his normal work.

2. The course covers the types of fire fighting equipment utilized onsite and explains the use of each type. The general locations of the equipment are also discussed. Familiarization of the trainees with plant layout including access and egress routes to each area are discussed.

3. The course discusses or demonstrates the use of available fire-fighting equipment and each trainee has the opportunity to utilize the equipment. The curriculum includes training in fighting structural fires and flammable liquid fires. The principles utilized in planning a fire attack are discussed also. The instruction includes a discussion of utilizing water in combatting electrical fires. The proper extinguishing methods and agents to be utilized in combatting all four classes of fires are covered and specific examples are cited where appropriate to illustrate techniques.

4. The course discusses in detail the action to be taken when a fire is discovered, the reporting requirements for fires, the actions to be taken by the individuals or groups on site at the time and provides information on contacting appropriate outside fire-fighting organizations, as outlined in the Plant Procedures.

5. The proper use of the installed communications equipment is either common knowledge, e.g. telephone use, or is covered during the new employee's initial orientation, e.g. public address system use. The installed plant lighting system is normally on and in the event it is lost, the emergency lights activate automatically. Portable handheld lights are available for firefighting use. The installed ventilation system is operated by qualified operators.

   The initial training addresses the principle of ventilation in the fire-fighting. The discussion includes vertical and horizontal ventilation, where to ventilate and distribution of personnel when ventilating.

   If the trainee does not routinely receive instruction outside the fire protection course, the following will be presented in the fire protection course:

   a. Hands-on training in the use of self-contained breathing equipment.

   b. Nomenclature, normal use, emergency use, donning, and storage of the breathing equipment.

   c. Changing out of air bottles on breathing equipment.

      The course discusses methods of escaping from a fire zone should the air supply become expended.

6. The initial training course discusses the need for organization and coordination in fire-fighting operations. More effectual leadership is developed through presentations on the tactics utilized in fire-fighting. This is done by discussing several different fire scenarios with the trainees once the concepts and considerations of making a fire attack have been covered. The pre-fire plans and procedures are used as a basis.

7. One part of the course consists of a discussion regarding smoke and toxic products of combustion. The use of self-contained breathing apparatus is emphasized.

8. Training is provided in fighting fires in windowless structures such as those that could occur in ANO.

9. Leadership training is provided such that any individual completing the course is capable of taking charge at the scene of the fire.

10. One part of the presentation includes the principles and techniques involved in structural fire-fighting. This includes types of fires: incipient, free burning, smoldering; thermal balance; heat and smoke movements; types of attack; visibility; need for breathing apparatus; and, tactics. Utilizing the principles learned in this part of the course, the brigade members can apply them to an actual fire inside a structure.

11. The course covers in detail the provisions of the procedure on fire brigade organization and training in the general fire plan. The provisions of those procedures not related to fire-fighting are not discussed nor are the various pre-fire plans. The periodic training program is the form utilized for familiarization with the Pre-fire plan.

12. The initial fire brigade training course is maintained up-to-date with regard to modifications to the fire-fighting equipment provided and to the installed Fire Protection Systems. Changes in pre-fire plans are covered in the periodic training course.

B. The instruction is provided by individuals who are knowledgeable, experienced and trained in the subject matter presented.

C. The initial training course is intended for Fire Brigade members. No individual will be placed on the Fire Brigade unless he has completed the Initial Classroom Training.

D. Training is conducted in accordance with Technical Specifications and will repeat the content of the Fire Brigade Training Course over a 2-year period.

9.8.3.2.2 <u>Practice</u>

Practice sessions are held for fire brigade members on the proper method of fighting various types of fires similar to those which could occur in Arkansas Nuclear One. These sessions provide fire brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions. These practice sessions are provided at intervals not exceeding one year for each fire brigade member.

9.8.3.2.3 <u>Drills</u>

Fire brigade drills are performed at the plant.

A. Each drill is assessed for effectiveness by a person knowledgeable in fire-fighting tactics. Among items considered are:

1. Time elapse from start of drill to announcing of fire, as appropriate.

2. Time required for minimum expected fire brigade members to respond.

3. Time required for full fire brigade response.

4. What equipment was broken out for use at the fire scene.

B. The drills also include observation of the brigade's performance in a discussion with the brigade members. The observation and discussions serve to provide the evaluator with an indication of each brigade member's knowledge in his role in a fire, conformance with plant procedures, and use of equipment.

C. The drills include simulated use of appropriate equipment in the areas, and types of fires or conditions are varied from drill to drill such that brigade members are trained in fighting fires in safety-related areas containing significant fire hazards (moderate or greater heat loads) with the exception of the reactor building. The Pre-fire plans and procedures are used as a basis for these drills. Table top drills are not utilized in lieu of actual drills.

The situation selected for drills simulates the size and arrangement of a fire which could reasonably occur in the zone selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming loss of the automatic protection system within the zone, if any.

D. Assessment of the Fire Brigade Leader's direction of the fire-fighting effort is performed as described in "B" above.

E. Fire drills are performed at varying intervals such that all fire brigade members participate in a drill at least semi-annually to the requirements of NFPA 500, “Standard for Industrial Fire Brigade.” At least one drill per shift is scheduled to be performed on a "back shift." At least one drill per year is unannounced for each shift. At least once per year an attempt is made to hold a drill with the Fire Brigade and the local fire department.

F. The drills conducted at Arkansas Nuclear One are pre-planned to accomplish certain objectives. A critique of each drill is generated by an individual knowledgeable in fire-fighting tactics and copies are distributed to appropriate management personnel. Drills are normally observed by an individual knowledgeable in fire-fighting tactics with the assistance of other selected personnel as required.

9.8.3.2.4 Records

Records of all formal training provided to each fire brigade member are maintained and are available for review.

9.8.3.2.5 Local Fire Department

Information is available to the local fire department for use in their training programs. This information includes basic radiation principles, typical radiation hazards, and precautions to be taken in a fire involving radioactive materials in the plant.

9.8.3.3 General Employee Fire Prevention Training

Personnel are instructed on the proper handling of accidental events such as leaks or spills of flammable materials that are related to fire prevention. In addition, all plant contractor personnel are indoctrinated in appropriate administrative procedures which implement the Fire Protection Program and the emergency procedures relative to fire protection.

9.8.3.4 Control of Combustibles

Administrative controls minimize the amount of combustibles that a safety-related area may be exposed to. The following discussion describes the controls developed.

A. Plant Procedures govern the handling, use and storage of combustibles in the plant. They specify the quantities of flammable and combustible liquids and flammable gases which may be stored in various buildings and set requirements regarding the handling, storage, and use of all combustible materials.

B. The Plant Procedures contain guidance regarding location, allowable quantities, types of materials allowed, and other factors affecting the fire loading in the safety-related zones of the plant. All work in the NFPA 805 defined power block is reviewed to assess the impact of fire protection by the Cognizant Supervisor and/or Operations. Where any questions arise regarding additional fire protection measures needed, the suitability of materials, storage arrangement, etc., Fire Protection personnel are consulted.

C. The Plant Procedures contain specific requirements to minimize the fire hazard associated with waste, scrap, rags, oil spills, or combustibles resulting from work activity in the NFPA 805 defined power block. These requirements consider the type of hazards involved and specify one or more of the following actions as appropriate:

    1. Timely removal of all waste, debris, scrap, rags, oil spills or other combustibles resulting from the work activity. Removal is affected upon the completion of the work activity, or end of the shift, whichever is sooner (NFPA 805, Section 3.3.1.2(3)).

    2. Properly storing the material to minimize the hazard.

    3. Insuring that the zone is continually manned or frequently inspected while the hazard is present.

    4. Periodic surveillance is made throughout the plant in an effort to control the accumulation of combustibles.

    5. Wood used in the plant is treated with a fire retardant, pressure treated fire retardant lumber, or approved by Design Engineering / Fire Protection personnel.

    6. Plastic sheeting materials used in the NFPA 805 power block shall be fire-retardant types as approved by Fire Protection personnel.

9.8.3.5 Control of Ignition Sources

9.8.3.5.1 Administrative Controls

Administrative Controls are instituted to protect safety-related equipment from fire damage or loss resulting from work involving ignition sources, such as welding, cutting, grinding or open flame work. Controls prohibit the use of open flame or combustion smoke for leak testing. Smoking is prohibited in the plant.

9.8.3.5.2 Control of Welding, Cutting, Grinding and Open Flame Work

A. The Plant Procedures control all cutting, welding, grinding, and other open flame operations. Prior to beginning any cutting, welding, burning, or other open flame operations, the workers must have a Hot Work Permit issued by the Cognizant Supervisor and/or Operations.

B. The Plant Procedures require that the Cognizant Supervisor and/or Fire Protection Personnel be responsible for having the work site inspected prior to work commencement and determine the precautions necessary for the safe performance of the work. Where any questions arise in regard to the fire protection, the Fire Protection personnel should be contacted.

   The Plant Procedures assure that the following precautions are accomplished:

   1. The Cognizant Supervisor and/or Operations assures that all movable combustible material below and within a 35-foot radius of the cutting, welding, grinding, or open flame work is removed, if feasible to do so.

   2. Where the above requirements cannot be met, the Cognizant Supervisor and/or Operations assures that all combustible materials within a radius of 35 feet and below the cutting, welding, grinding, or open flame work are protected in accordance with NFPA-51B, "Standard for Fire Protection During Welding, Cutting, and Other Hot Work," Item 3-3.2.

      The Cognizant Supervisor has the option of requiring a fire watch, if needed, to meet the requirements of NFPA-51B or to satisfy himself that the job can be accomplished without undue hazard to equipment or personnel.

      In the event the Cognizant Supervisor determines a fire watch is not needed, concurrence must be obtained from the Shift Manager or Fire Protection personnel.

      If the fire watch is deemed necessary by the above paragraph, the duties of that person(s) are defined in accordance with NFPA-51B.

   3. In accordance with the provisions of NFPA-51B, the Cognizant Supervisor assures himself that the cutting and welding equipment to be used is in satisfactory operating condition and in good repair.

   4. Welding, cutting, grinding, or open flame work to be performed on any pressurized system or any tank, vessel, or piping, which may contain residual combustible vapors, is performed in accordance with NFPA-51B. Where appropriate, the use of suitable instrumentation in verifying a safe atmosphere is required.

9.8.3.5.3 Leak Testing

The Plant Procedures provide for the strict control of open flame work, regardless of the purpose. Thus, no special procedure for a specific application is needed.

9.8.3.5.4 Smoking and Ignition Source Restrictions

Smoking is prohibited in all Entergy enclosed facilities.

9.8.3.6 Plant Procedures (Fire Fighting)

The Plant Procedures cover such items as notification of the fire, fire emergency procedures, and coordination of fire-fighting activities with off-site fire departments. The Plant Procedures identify:

A. Actions to be taken by individuals discovering the fire, such as, notification of control room, attempt to extinguish fire and actuation of local fire protection systems.

B. The specific actions required of the control room operator when a fire is reported.

C. The people required to report to the scene of the fire when a fire is announced.

D. Means of immediately alerting the Fire Brigade upon report of a fire or receipt of an alarm on the control room annunciator panel.

E. Actions to be taken by the Fire Brigade after notification by the control room of a fire, including location to assemble.

   A set of Pre-fire plans is available which provide the Fire Brigade Leader with information, which with his training, allows him to make intelligent fire ground decisions and specify the responsibilities to the fire brigade members for the selection of fire-fighting equipment and transportation to the scene of the fire.

   The operation of installed fire protection systems is covered in the Fire Brigade Training Course. Pre-planned strategies, incorporated in the Pre-fire plans, are available for all safe shutdown zones and zones presenting a hazard to safe shutdown equipment. Copies of the Pre-fire plans are provided in both unit control rooms, both unit simulators, and the fire brigade lockers. Other copies may be distributed as needed.

F. Pre-fire plans

   1. Each plan identifies any hazardous combustibles located in the zone concerned or any combustibles of an unusual nature which might be encountered.

   2. The plan lists fire-fighting equipment available in the vicinity of the fire zone and a plan of the area is incorporated to identify their location. No attempt is made to identify such extinguishers thus fitted for the particular material in the area, unless a material with unique extinguishing requirements is present. The fire brigade members receive training in the selection of extinguishing agents for various types of fires. The Pre-fire plan should contain no more information than is absolutely necessary. To load it with information with which the plant personnel are quite familiar detracts from the usability of the document.

3. The Pre-fire plans provide guidance in selecting the most favorable direction from which to attack fires in the particular zone. Specific instructions, as to which way to attack from, are not provided, due to the many different types of fire situations which may occur in a given zone.

   Many factors are considered when providing guidance regarding directions from which to attack the fire. Among these are accessibility, protection of personnel, drainage, ventilation, availability of equipment, potential fire configurations within the zone, etc.

   Each Pre-fire plan includes a plan view of the fire area on which all means of access and egress are marked. Fire-fighting equipment locations are identified, major items of equipment are shown, and other significant features are depicted.

   Where locked doors are involved, they are specifically identified in the pre-fire plans and means of accessing them is specified.

4. The Pre-fire plans include the location of the various systems and equipment controls which might be of value during a fire attack.

5. The Pre-fire plans include a listing of exposures which may need to be protected or they are shown on attached arrangement drawings. In addition, the plan discusses hazards which might be present, such as drums of combustible liquids, and state what action is required to negate or minimize the hazards. For example, solvent drums should be kept cool to prevent overpressure rupture or an internal vapor air explosion.

6. The plant Pre-fire plans provide the Fire Brigade Leader with the type of radiological hazards present. Any personnel hazards of toxic nature not included in the combustion gases or ordinary materials are either listed in the pre-fire plans or shown on attached drawings. In addition, the plant utilizes the hazardous materials identification system developed by NFPA, and as described in the NFPA Code 704M, "Standard System for Identification of the Hazards of Materials for Emergency Response," where significant quantities of toxic materials are normally present.

7. The Pre-fire plans discuss the means available for ventilating the zones concerned.

G. General instructions for operators and general plant personnel are set forth in the Plant Procedures.

   Instructions for the control room operators have been discussed in "B" above. General plant personnel are required to report to their work center, or if that area is involved, to an alternate assembly area.

H. The validity of the Pre-fire plans is tested by drills and at such drills all aspects of the plan are reviewed with the fire brigade and the discussion of the plan is concluded.

I. The Plant Procedures define the organization and outline coordination with outside fire departments.

J. The Plant Procedures assure that the responsibilities of each Fire Brigade position corresponds with actions required as outlined in the procedures.

K. The Plant Procedures assure that the responsibility of the Fire Brigade to the operation of the plant do not conflict with their responsibilities during a fire emergency.

9.8.3.7 <u>Fire Brigade (Organization and Composition)</u>

The Fire Brigade consists of five trained individuals established each shift and is onsite 24 hours a day, seven days a week. The Fire Brigade consists of five personnel who have received the full training. Upon arrival at the scene, the Fire Brigade Leader assesses the situation, instructs his brigade members and extinguishes the fire. He also reports his assessment to the control room. He requests the control room to supply him assistance if he does not think his team can control the situation. In this event, the control room contacts the local fire department and any other outside agency he believes should respond.

At the beginning of each shift, a Fire Brigade is selected. Each position of the brigade corresponds to a position title. Specific instructions as to command control, fire hose-laying, applying extinguishant, advancing support supplies to the fire, communication with the control room coordination with outside fire department are covered in general in the initial classroom training course. Those fire zones which have pre-fire plans contain specific instruction in the areas above applicable for that specific zone.

The Fire Brigade Leader communicates directly with the Control Room. Response activities requiring coordination or authorization by Operations are provided in the Plant Procedures and/or Pre-fire Plans.

Each Fire Brigade member, at least annually, must satisfactorily complete a physical examination for performing strenuous duties. This requirement is enforced to effectively screen out personnel with heart or respiratory disorders and preclude these personnel from Fire Brigade duties.

The recommendation in NFPA 27 and the standards contained in the associated Appendix were considered in the organization, training, and functioning of the Fire Brigade.

9.9 COMPRESSED AIR SYSTEMS

The Instrument Air System is designed to provide a reliable, continuous supply of dry, oil-free, compressed air for pneumatic instrument and valve operation. The IA System can supply job-site filtration units which provide OSHA Grade D breathable air throughout the plant.

The Service Air System supplies compressed air to service outlets throughout the plant for operation of pneumatic tools or other requirements.

The Breathing Air System is designed to provide a reliable supply of dry, oil-free, OSHA grade D quality compressed air for use in air-fed respirators in controlled access areas. The Breathing Air System consists of one compressor, two receiver tanks, and two air purification units.

9.9.1 DESIGN BASES

9.9.1.1 Performance Objective and Design Criteria

The air compressors are designed to supply air at a nominal pressure of 100 psig, with pressure reduced as necessary, at the point of use for various service requirements.

The capacity of each of the C-28A and C-28B instrument air compressors and their associated equipment is sufficient to meet the average instrument air demand for the Nuclear Steam Supply System (NSSS) and the balance of the plant.

The breathing air compressor is designed to supply air at a nominal pressure of 100 psig, with pressure reduced, as necessary, at the point of use for respirator requirements. A cross-connect to the instrument air cross-connect between Units 1 & 2 is provided to enable Breathing Air to backup Instrument Air.

9.9.1.2 Codes and Standards

The air receivers and the instrument air dryers are built in accordance with ASME Section VIII requirements. Piping, fittings, and valves are in accordance with ANSI B31.1 code.

The Breathing Air System is shown schematically on Figure 9-14.

9.9.2 SYSTEM DESCRIPTION AND OPERATION

9.9.2.1 System Description

The Compressed Air System is shown schematically on Figure 9-14. A continuous supply of instrument air is required to hold various pneumatically operated valve actuators in their operating positions. In case of a loss of the Instrument Air supply, all pneumatically operated valves assume a fail-safe position or retain the capability to be moved to the safety function position by normal and ES control inputs.

9.9.2.2 System Operation

The Instrument Air System powers all pneumatic valves in the plant. The safe position for each valve is its shelf position except for some double acting piston operated valves which have individual air receivers to assist in moving them to the safety function position.

There are two instrument air compressors (C-28A and C-28B) provided with air cooled aftercoolers which discharge to a common receiver tank. For normal service, the operation of these compressors will maintain system pressure. Each compressor is rated at greater than 340 inlet cfm at a discharge pressure of 110 psig. The compressors will be operated at the nominal design pressure of 100 psig. The heaterless desiccant dryer towers provided with these compressors are designed to maintain a dew point of at least 18 °F below the minimum ambient temperature on site (Ref. Quality Standard For Instrument Air ANSI/ISA-S7.0.0.1-1996). Instrument Air is filtered downstream of the dryers through filter mechanisms to the required purity. Instrument Air from these compressors is oil free.

If the instrument air compressors are inoperable, a backup tie from the service air header is provided to supply air automatically to the Instrument Air System. A check valve in the tie line prevents instrument air from entering the station service air header. Instrument Air can be cross-connected between Unit 1 and Unit 2 either as dry air or without moisture removal. Instrument air can also be provided by the breathing air compressors as previously described.

The Service Air System is used for pneumatic tools, plant maintenance, and miscellaneous services. The Unit 1 and Unit 2 Service Air Systems are normally cross-connected. System pressure is normally maintained by the Unit 2 yard air compressor (2C-43). This system has no effect on nuclear safety-related equipment.

The Breathing Air System consists of compressor C-30 which is designed to supply compressed air to be used in breathing air respirators in ANO-1 & 2 controlled access areas. The system also has connecting piping for supplying instrument air if needed. Air enters C-30 through an intake filter. The compressed air is controlled with a Variable Frequency Drive (VFD). The compressed air is discharged through an intercooler, an aftercooler, moisture separators, and then to Breathing Air receivers T-118 and T-120. The compressed air from T-118 and T-120 then enters the air prefilters, then the air dryer tanks. Downstream of the dryer tanks is a 2" connection that ties into the Instrument Air System piping. A filter is installed upstream of the Instrument Air to Breathing Air cross-connect to minimize the potential of introducing desiccant into the Instrument Air System. Thus, this compressor can be used as backup to the Instrument Air System when there are no requirements for the use of Breathing Air. With the instrument air connection valves shut, the air leaves the dryer tanks and then enters the breathing air catalyst, which will remove the carbon-monoxide in the compressed air. After leaving the catalyst, the air passes through a final filter and is sampled by the carbon monoxide monitor/alarm before flowing into the Breathing Air System piping in the Unit 1 and 2 auxiliary and reactor buildings. In addition, carbon monoxide monitors have been provided to determine carbon monoxide concentrations.

9.9.2.3 Safety Functions

Although the Compressed Air System is not required for safe shutdown of the plant, responses of air operated valves do serve some safety function to engineered safeguards signals. These safety functions fall into two categories. In the first case, where a valve is normally in the safeguards position, the solenoids serve a passive function by preventing the valve position from being changed. If the valve is in an abnormal, non-safeguards position, the engineered safeguards signal returns the valve to its safeguard position and then prevents any further position change. In the second case, where a valve is normally in the non-safeguards position, the engineered safeguards signals energize the solenoids to change the valve to its safeguards position. If the valve is in its abnormal, safeguards position, the engineered safeguards signal prevents any position change.

All safety-related, pneumatically operated valves fail in the safe position except for three reactor building isolation valves. These three valves are provided with integral accumulators, instrument air check valves, and associated piping (ICW isolation valves CV-2214, CV-2233, CV-2234), and have a safety function to close. These valves may be reopened utilizing manual operation. All accumulators are designed to Seismic Category I and are equipped with local pressure indicators to insure performance of their functions. All other air leakage detection is to be local aural. Every instrument has local air isolating valves. In cases where there is one pneumatically operated isolation valve outside containment and a check valve inside containment, the air operated valve controls are designed such that the isolation valve can be closed by either of two independent signals. In most cases, where process lines are duplicated, one engineered safeguard signal is connected to one solenoid valve for each pneumatic isolation valve.

Tables 9-25 and 9-26 list all the air operated Seismic Category 1 valves and ventilation system dampers.

The Breathing Air System is not required for the safe shutdown of the plant.

9.9.2.4 Testing and Inspection

Each component is inspected and cleaned prior to installation into the system. Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability and limits during preoperational testing. The system is operated and tested initially with regard to flow paths, flow capacity, and mechanical operability. There is no common station vent plenum system.

No testing of the automatic transfer of standby compressors is required. Inspection of inservice compressors will be in accordance with manufacturer's recommendations.

9.10 NITROGEN PRESSURIZATION SYSTEM

The Nitrogen Addition System distributes nitrogen from the supply area, where it is stored in bottles, to the systems as shown in Figure 9-4. The bottles are automatically recharged from a liquid nitrogen storage tank via a high pressure pump and vaporization system. Two independent pressure reducing stations located immediately downstream of the supply area regulate the high pressure system and the low pressure system. The low pressure system also ties into the liquid nitrogen storage tank vapor space to utilize tank evaporation to minimize nitrogen losses.

During normal operation, the isolation valve to the high pressure system is closed and nitrogen is supplied only to the low pressure system. However, the core flooding tanks are repressurized with nitrogen from the high pressure system following refueling and periodically during operation. In addition to the normally closed isolation valve on the high pressure nitrogen system, the core flooding tanks are further protected against inadvertent depressurization by a locked closed "quality assured" isolation valve.

The Nitrogen Pressurization System is also used for the dry fuel storage project for drying of the dry fuel system fuel canisters.

9.11 EMERGENCY LIGHTING AND COMMUNICATIONS SYSTEMS

This section provides a discussion of lighting and communications systems' operations during an emergency, and requirements for their inspection and testing. A description of the communication systems under normal operating conditions is in Chapter 7.

An emergency lighting system is provided which receives its power from the non-class 1E 125-volt DC power bus. This DC emergency lighting system is energized upon loss of AC power. The bulbs of the emergency lighting system are normally not energized. Testing and inspection can be accomplished by depressing a test pushbutton.

Two separate in-plant communication systems are provided, a Computerized Branch Exchange (CBX) and public address system and a plant intercom system. The ANO CBX system is a direct dial, private, automatic telephone exchange consisting of three independent nodes interconnected by fiber optics. Each node is powered by a 48 volt battery bank capable of operating the switch for up to 8 hours. Each battery bank is kept charged by dual parallel battery chargers powered by commercial mains. The Emergency Operating Facility (EOF) CBX system is powered by a 48 volt battery bank capable of operating the switch for up to 8 hours. The battery bank is kept charged by a battery charger that is powered by commercial mains and is backed by the EOF diesel generator. The public address system is powered by a battery charger with a battery backup. The following additional features have been incorporated:

A. Paging public address and page answer lines

B. Conference lines; and

C. "Meet-me" conference line.

The in-plant 4-channel intercommunication system is composed of local amplifier-loudspeaker-handset combinations. The entire system is solid-state.

A direct connection between the off-site public telephone system and telephone stations in the control room and the administration building is provided.

A detailed description of communication methods and capabilities during an emergency is given in sections E and F of the Emergency Plan.

9.12 REFERENCES

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Table 9-1

MAKEUP AND PURIFICATION SYSTEM PERFORMANCE DATA

Normal Letdown Flow, gpm	45
Maximum Allowable Letdown Flow, gpm	160
Total Seal Flow to Each Reactor Coolant Pump, gpm	10
Seal Inleakage to Reactor Coolant System per Reactor Coolant Pump, gpm	8.5
Temperature to Seals, normal/maximum, °F	125/150
Purification Letdown Fluid Temperature, normal/maximum, °F	120/125
Makeup Tank Normal Operating Pressure Range, psig	15-35
Makeup Tank Water Volume, nominal, ft^3	300

Table 9-2

MAKEUP AND PURIFICATION SYSTEM PERFORMANCE DATA

<u>Makeup Pumps</u>	
Number	3
Type	Horizontal, Multistage, Centrifugal, Mechanical Seal
Rated Capacity, gpm	300 (Also see Figure 6-6)
Rated Head, ft H_2O	5,545 (Also see Figure 6-6)
Motor Size, horsepower	700
Pump Material	SS Wetted Parts
Design Pressure, psig	3,000
Design Temperature, °F	200
<u>Letdown Coolers</u>	
Number	2
Type	Shell and Spiral Tube
Heat Transferred, Btu/hr	16.2×10^6
Letdown Flow, lb/hr	3.5×10^4
Letdown Cooler Inlet/Outlet Temperature, °F	558/125
Material, shell/tube	CS/SS
Design Pressure (Shell/Tube), psig	200/2,500
Design Temperature (Shell/Tube), °F	350/650
Component Cooling Water Flow (Each), lb/h	2×10^5
Code, tube side/shell side	ASME III-3/VIII, Division 1
<u>Reactor Coolant Pump Seal Return Coolers</u>	
Number	2
Type	Shell and Tube
Heat Transferred, Btu/hr	7.74×10^5

Table 9-2 (continued)

Reactor Coolant Pump Seal Return Coolers (continued)	
Seal Return Flow, lb/hr	1.1×10^5
Seal Return Temperature Change, °F	127 to 120
Material, (shell/tube)	CS/SS
Design Pressure (Shell/Tube), psig	150/150
Design Temperature (Shell/Tube), °F	300/200
Recirculated Cooling Water Flow (Each), lb/hr	1.1×10^5
Code Tube Side/Shell Side	ASME III-C/VIII
Makeup Tank	
Volume, ft^3	600
Design Pressure, psig	100
Design Temperature, °F	200
Material	SS
Code	ASME III-C
Purification Demineralizers	
Number	2
Type	Mixed Bed, HOH
Material	SS
Resin Volume, ft^3	50
Flow, gpm	70*
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Code	ASME III-C

Table 9-2 (continued)

Makeup Filters

Number	2
Flow Rate, gpm	80
Material	SS
Design Pressure, psig	300
Design Temperature, °F	250
Code	ASME III-C

* Original Design. Subsequent analysis increased the limit to 125 gpm each. The procedure limit is 123 gpm.

Table 9-3

DELETED

Table 9-4

REACTOR COOLANT QUALITY

Boron, ppm	See Figure 3-1 and Chapter 3A, "Reload Report" and its supporting documentation
Lithium as ^{7}Li, ppm (where required for pH adjustment)	Consistent with Station Lithium Program
Dissolved Oxygen as O_2 (max), ppm	0.1*
Chlorides as Cl^- (max), ppm	0.15**
Hydrogen as H_2, cc/kg H_2O, (STP)	25-50**
Fluorides as F^- (max), ppm	0.15**
Total Dissolved Gas (max), cc/kg H_2O, (STP)	100**
Zinc (maximum)	< 10 ppb***
Zinc (operating)	< 8 ppb***

* Above 250 °F
** Above cold shutdown conditions.
*** Normal limits under non-transient conditions.

Table 9-5

CHEMICAL ADDITION AND SAMPLING SYSTEM SAMPLING LOCATIONS

Steam Generator Samples

1. Secondary Side of Steam Generator

Reactor Coolant Samples

1. Pressurizer Water Space
2. Pressurizer Steam Space
3. Decay Heat Cooler Outlet
4. Core Flooding Tanks
5. RCS Hot Leg

Auxiliary Systems Samples

1. Purification Demineralizer Outlet
2. Purification Demineralizer Inlet
3. Makeup Tank Water Space
4. Reactor Coolant Letdown
5. Clean Waste Receiver Tanks
6. Spent Fuel Pool Filter Outlet
7. Spent Fuel Pool Cooler Outlet
8. Primary Coolant Quench Tank
9. Makeup Tank Gas Space

Hydrogen and Oxygen Gas Analyzer

1. Primary Coolant Quench Tank
2. Clean Waste Receiver Tank
3. Dirty Waste Drain Tank
4. Auxiliary Building Equipment Drain Tank
5. Waste Gas Decay Tanks
6. Waste Gas Surge Tank

Sample Containers (to be analyzed for a variety of substances)

1. Makeup Tank Gas Space
2. Pressurizer Steam Space
3. Letdown Line Upstream of Purification Demineralizer
4. In addition, all gas analyzer samples can be containerized for further analysis

Table 9-6

CHEMICAL ADDITION AND SAMPLING SYSTEM COMPONENT DATA

<u>Boric Acid Mix Tank</u>	
Type	Vertical Cylindrical
Volume, ft^3	134
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Material	SS
<u>Lithium Hydroxide Mix Tank</u>	
Type	Vertical Cylindrical
Volume, gal.	50
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Material	SS
<u>Boric Acid Addition Tank</u>	
Type	Horizontal Cylindrical
Volume, ft3	1000
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Material	SS
<u>Zinc Mixing Tank</u>	
Type	Vertical Cylindrical
Volume, gal	50
Design Pressure, psig	Atmospheric
Design Temperature, °F	500
Material	SS
<u>Pressurizer Sample Cooler</u>	
Type	Shell and Spiral Tube
Rated Capacity, Btu/hr	2.1 x 105
Sample Flow Rate, lb/hr	200
Maximum Sample Inlet Temperature, °F	650
Sample Outlet Temperature, °F	120
Cooling Water Flow, lb/hr	5,000
Design Temperature Shell/Tube, °F	250/670
Design Pressure Shell/Tube, psig	150/2,500
Code Tube Side, Shell Side	ASME III-C/VIII

Table 9-6 (continued)

Steam Generator Sample Cooler	
Type	Shell and Spiral Tube
Sample Flow Rate, lb/hr	260
Sample Inlet/Outlet Temperature, °F	535/100
Cooling Water Flow, lb/hr	1,350
Design Temperature Shell/Tube, °F	125/800
Design Pressure Shell/Tube, psig	250/5,000
Boric Acid Pumps	
Number	2
Type	Centrifugal
Capacity, gpm	25
Rated Head, ft	140
Design Pressure, psig	100
Design Temperature, °F	200
Pump Material	SS
Motor Size, hp	4.1
Hydrazine Pump	
Type	Reciprocating, variable stroke
Capacity, gph	10
Maximum Discharge Pressure, psig	100
Hydrazine Pump (continued)	
Design Pressure, psig	150
Design Temperature, °F	200
Pump Material	SS
Motor Size, hp	1/3
Lithium Hydroxide Pump	
Type	Reciprocating, variable stroke
Capacity, gph	10
Maximum Discharge Pressure, psig	100
Design Pressure, psig	150
Design Temperature, °F	200
Pump Material	SS
Motor Size, hp	1/3
Zinc Mixing Tank Recirc. Pump	
Type	Centrifugal
Capacity, gpm	12
Maximum Discharge Pressure, psig	22

Table 9-6 (continued)

Zinc Mixing Tank Recirc. Pump (continued)	
Design Pressure, psig	150
Design Temperature, °F	250
Pump Material	SS
Motor Size, hp	1/2
Zinc Injection Metering Pump	
Type	Positive Displacement Dual Diaphragm
Capacity, gph	0.3 to 2.95
Maximum Discharge Pressure, psig	150
Design Pressure, psig	300
Design Temperature, °F	180
Pump Material	SS
Motor Size, hp	1/2

Table 9-7

REACTOR BUILDING COOLING UNITS PERFORMANCE AND EQUIPMENT DATA

(capacities are for single components)

Number	4 (Note 1)
Number Normally Operating	4
Normal air flow, cfm	30,000
Fan Static Pressure Head, in. water gage	3
Emergency Duty	
Entering Air Conditions, °F/saturated	286
Heat Rejected to Service Water, Btu/hr	56.5×10^6

Note: Other design parameters for post-accident operations are listed in Section 6.

Note 1: The fifth cooling unit, VSF1E, is for normal cooling only.

Table 9-8

INTERMEDIATE COOLING AND SERVICE WATER SYSTEM DESIGN PARAMETERS AND MAJOR EQUIPMENT DATA

<u>Intermediate Cooling Water Pumps</u>	
Number	3
Number Normally Operating	2
Type	Horiz. Centrifugal
Rated Capacity, gpm	2,500
Rated Head, ft	125
Motor horsepower	100
Material Casing/Trim	Cast Iron/Bronze
<u>Intermediate Cooling Water Booster Pumps</u>	
Number	2
Number Normally Operating	1
Type	Vertical Inline
Rated Capacity, gpm	260
Rated Head, ft.	130
Motor Horsepower	15
Material	Stainless Steel
<u>Intermediate Coolers</u>	
Number	3
Number Normally Operating	2
Type	Shell and Tube
Heat Transferred, Btu/hr	30×10^6
Shell Side (intermediate cooling system)	
Normal Inlet/Outlet Temperatures, °F	125/95
Flow Rate, gpm	2,000

Table 9-8 (continued)

Intermediate Coolers (continued)	
Tube Side (service water system)	
Normal Inlet/Outlet Temperatures, °F	85/106.4
Flow Rate, gpm	2,800
Material Tubes/Shell	Admiralty/Carbon Steel (E28A/B) Titanium/Carbon Steel (E-28C)
Service Water Pumps	
Number	3 (full capacity)
Number Normally Operating	2
Type	Vertical Turbine
Design Capacity, gpm	6500
Design head, ft	167
Motor horsepower	350
Material Casing/Trim	Stainless Steel

Table 9-9

SPENT FUEL POOL COOLING SYSTEM PERFORMANCE AND EQUIPMENT DATA

<u>Spent Fuel Heat Load</u>	
Normal (1/3-core), Btu/hr	13.07×10^6
Maximum (Full core), Btu/hr	31.00×10^6
System Design Pressure, psig	125
System Design Temperature, °F	250
<u>Spent Fuel Coolers</u>	
Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/cooler	8.75×10^6 (a)(b)
Cooling Water Flow, lb/hr/cooler	5.0×10^5
Code Shell/Tube	ASME VIII/III-C
<u>Spent Fuel Coolant Pumps</u>	
Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	1,000
Head, ft H_2O	100
Motor Size, hp	40
Spent Fuel Pool Volume, ft^3	39,000
<u>Spent Fuel Coolant Filters</u>	
Number	2
Design Flow Rate, gpm	200
Design Pressure, psig	150

Table 9-9 (continued)

Spent Fuel Coolant Filters (continued)	
Design Temperature, °F	250
Material	SS
Code	ASME III-C
Borated Water Recirculation Pump	
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	180
Head ft H_2O	140
Motor Size, hp	15
Spent Fuel Demineralizer	
Type	Mixed bed
Material	SS
Resin Volume, ft^3	21
Flow, gpm	180
Code	ASME III-C

(a) Assumes pool water to cooler at 131 °F and cooling water to cooler at 95 °F.

(b) Capacity listed is based on original purchase specification data and does not reflect current or required capacity.

Table 9-10

DECAY HEAT REMOVAL SYSTEM PERFORMANCE AND EQUIPMENT DATA

Reactor Coolant Temperature at Startup of Decay Heat Removal, °F	< 280
Reactor Coolant System Time to Cool From ~280 °F to 140 °F, hr	14
Refueling Temperature, °F	140
Fuel Transfer Canal Fill Time, hr	6
Fuel Transfer Canal Drain Time, hr	6
Minimum Boron Concentration in BWST, ppm	2,270
<u>Decay Heat Removal Pumps</u>	
Number	2
Type	Single stage, centrifugal
Rated Capacity, gpm	3,000
Rated Head at Rated Capacity, ft H_2O	350
Motor Horsepower	350
Material	SS (wetted parts)
Design Pressure, psig	450
Design Temperature, °F	300
<u>Decay Heat Removal Coolers</u>	
Number	2
Type	Shell and tube
Capacity (at 140 °F), Btu/hr	30×10^6
Reactor Coolant Flow, gpm	3,000
Service Water Flow, gpm (Design Flow Rate)	3,000
Service Water Inlet Temp, °F	85
Material, Shell/Tube	SS/CS-SS
Design Pressure, Shell/Tube, psig	450/120

Table 9-10 (continued)

Decay Heat Removal Coolers (continued)	
Design Temperature, Shell/Tube, °F	300/300
Code	ASME III-C/VIII and TEMA-R
Borated Water Storage Tank	
Capacity, gal.	405,090
Material	Carbon Steel w/Placite 7155 Liner
Design Pressure, ft H_2O	10
Design Temperature, °F	150
Code	AWWA-D-100

Table 9-11

SINGLE FAILURE ANALYSIS - SERVICE WATER SYSTEM

Component	Failure	Comments
Dardanelle Dam	Fails	Alarmed in Control Room. Service Water Supply Shifted to Emergency Pond. Plant Shutdown Required.
Sluice Gate 1*	Fails to Close	Shut 2,3,4,5. Open 6,7. Two Service Water Bays Full
Sluice Gate 2*	Fails to Close	Shut 1,3,4,7. Open 5,6. Two Bays Full
Sluice Gate 5*	Fails to Open	Shut 1,2; Open 3,4,6,7. All Bays Full
Sluice Gate 6*	Fails to Open	Shut 1,2; Open 3,4,5,7. All Bays Full
Sluice Gate 7*	Fails to Open	Shut 1,2; Open 3,4,5,6. All Bays Full
Service Water Pump	Fails to Start	Other Two Pumps Provide Necessary Redundancy
Valves	Fail to Position Correctly	Redundant Loop Provides Adequate Cooling
Piping	Fails	(See Above)
Component	Fails	(See Above)

* See Figure 9-6A

Table 9-12

SPENT FUEL POOL ACTIVITY

Isotope	Activity (µCi/cc)
Sr-89	6.39(-10)
Sr-90	5.77(-11)
Sr-91	4.52(-12)
Y-89M	6.39(-14)
Y-90	6.51(-9)
Y-91	8.33(-10)
Y-91M	2.92(-12)
Mo-99	3.40(-8)
I-131	4.06(-8)
I-133	5.49(-9)
I-135	2.59(-11)
Cs-134	9.26(-4)
Cs-136	3.87(-4)
Cs-137	9.15(-3)
Da-140	9.08(-10)
La-140	7.86(-10)
Pr-143	4.52(-11)
Ce-144	4.52(-11)
Cs-135	3.31(-12)
Fe-59	4.99(-6)
Mn-54	6.68(-6)
Cr-51	8.63(-4)
Zr-95	2.22(-7)
Co-58	1.11(-3)
Co-60	1.27(-4)
Nb-95	1.28(-8)
Rb-95M	1.91(-9)

Table 9-13

GASEOUS RADIOACTIVITY RELEASES FROM ARKANSAS NUCLEAR ONE - UNIT 1

	Quantity Released (Ci)		
Nuclide	1974*	1975	1976**
Kr-85	5.6E-2	3.15	2.23
I-131	5.3E-2	< 7.28E-3	3.44E-2
H-3	3.0E-2	5.2E-1	3.4

* ANO-1 began operation in August 1974. Data reflects the releases from August 1974 through December 1974.

** Through June 30, 1976.

Table 9-14

**SPENT FUEL POOL STORAGE FACILITY STRUCTURAL EVALUATION
SUMMARY OF COMPOSITE LOADS**

COMPOSITE LOADS	LOAD FACTORS												DESCRIPTION
	D_c	H	F	T_o	T_a	E_{ew}	E_{ns}	D_{fr}	FR_v	FR_{ew}	FR_{ns}	C[1]	
D	1.0	1.0	-	-	-	-	-	-	-	-	-	-	Dead Load
L	-	-	-	-	-	-	-	1.0	-	-	-	1.0	Live Load
T_o	-	-	-	1.0	-	-		-	-	-	-	-	Operating Thermal Load
T_a	-	-	-	-	1.0	-	-	-	-	-	-	-	Accident Thermal Load
E_1	0.067	0.067	-	-	-	0.1	0.1	-	1.0	1.0	1.0	-	Loads Generated by
E_2	0.067	0.067	-	-	-	0.1	-0.1	-	1.0	1.0	-1.0	-	0.1g Operating
E_3	-0.067	0.067	-	-	-	0.1	0.1	-	-	1.0	1.0	-	Basis Earthquake
E_4	-0.067	0.067	-	-	-	0.1	-0.1	-	-	1.0	-1.0	-	
F	-	-	1.0	-	-	-	-	-	-	-	-	-	Flood Load

(1) Crane load will be considered only if it is determined to contributed to live load in a conservative manner.

(2) DBE (E') taken as 2xOBE

Table 9-14a

SPENT FUEL POOL STORAGE FACILITY STRUCTURAL EVALUATION LOAD COMBINATION SUMMARY TABLE

No.	Load Combination	Reference[1]
1	$1.4D + 1.7L + 1.9E$	Load Case 2
2	$.75(1.4D + 1.7L + 1.7T_o)$	Load Case 4
3	$.75(1.4D + 1.7L + 1.7T_o + 1.9E)$	Load Case 5
4	$D + L + T_o + F$	Load Case a
5	$D + L + T_o + F$	Load Case b
6	$D + L + T_A$	Load Case c
7	$D + L + T_A + 1.25E'$	Load Case d

Notes: (1) NUREG-0800, Standard Review Plan, Section 3.8.4.

(2) E' = Represents a load, generated by .20g safe shutdown earthquake (SSE).

E' = 2.0 E

Table 9-15

SERVICE WATER SUMMARY

Mode of Operation	Water Source	Line Sizes	Flow Paths
Normal	Dardanelle	Open Suction 18" Disch	Dardanelle Pump Component Dardanelle
Shutdown (without lake)	Emergency Cooling Pond	36" Suction 18" Disch	Pond Pump Component Pond
DBA	Dardanelle	Open Suction 18" Disch	Dardanelle Pump Component Dardanelle

Mode of Operation	Submergence Above Pump Suction[1] (required)(ft)/(avail)(ft)	Nominal Flow (gpm)
Normal	2.94/14.5	5692
Shutdown (with outside) Power	4.28/14.5	7692
DBA	3.30/14.5	6255

Note 1: Submergence based upon lake operation at normal level, at peak lake temperature of 95 °F.

Table 9-16

COOLING POND OPERATING MODES

Operation	No. of Service Water Loops Operating on the ECP	
	Unit 1	Unit 2
Unit 1 shutdown	1	0
	2	0
Unit 1 LOCA	1	0
	2	0
Concurrent Unit 1 - Unit 2 shutdowns	1	1
	1	2
	2	1
	2	2
Unit 1 LOCA, Concurrent Unit 2 shutdown	1	1
	1	2
	2	1
	2	2
Unit 2 LOCA, Concurrent Unit 1 Shutdown	1	1
	1	2
	2	1
	2	2

Table 9-17
LITTLE ROCK METEOROLOGY - TEMPERATURE ANALYSIS
(Historical Data – not used in updated analysis)

Worst Case 1 day Meteorology - July 17, 1934
Worst Case 30 day Meteorology - June 26 - July 25, 1934

Date	Wind Speed (mph)	Dry Bulb Temp.(°F)	Relative Hum. (%)	Cloud Cover (Frac.)	Solar Radiation (Langleys/Day)
July 17, 1934	1.4	87	64	.25	691
June 26 – July 25, 1934	4.4	84	65	.25	685

Table 9-18
LITTLE ROCK METEOROLOGY - EVAPORATION ANALYSIS
(Historical Data – not used in updated analysis)

Highest 30-day Dew Point Depression - July 16 - August 14, 1930
Highest Daily Average Windspeed - July 16 - August 14, 1930

Date	Wind Speed (mph)	Dry Bulb Temp.(°F)	Relative Hum. (%)	Cloud Cover (Frac.)	Solar Radiation (Langleys/Day)
July 16 – Aug. 14, 1930	10.0	87	41	.25	653

Table 9-19
LONG-TERM STABILITY EMERGENCY COOLING WATER POND

Excavation Slope	Loading Condition	Minimum Computed Factor of Safety
2-1/2 to 1	Normal	2.3
2-1/2 to 1	Seismic	1.1

Consolidation test results on the plant area soils indicate that the permeability is on the order of $1x10^{-6}$ centimeters per second or lower.

Table 9-20

SINGLE FAILURE ANALYSIS - CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

Component	Failure	Comments and Consequences
1. Offsite Power	Not available	Emergency diesel start and supply electrical load to system.
2. Emergency diesels	One not available	The operative diesel supplies necessary power to one of the redundant system flow paths.
3. Control Room Emergency Air Conditioning Systems	One not available	The standby, full-capacity, control room emergency air conditioning system is available to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment.
4. Control Room Emergency Air Conditioning Systems	Rupture of equipment casings	The system is designed as a Seismic Category 1 System. The equipment and components are also inspectable and protected against credible missiles.
5. Control Room Emergency Air Conditioning Systems	Rupture of one loop of the Service Water System	The standby, full-capacity, Control Room Emergency Air Conditioning System is supplied by the second loop of the Service Water System piping.

Table 9-21

SINGLE FAILURE ANALYSIS
DECAY HEAT REMOVAL ROOMS COOLING
SWITCH GEAR ROOMS COOLING*

Component	Malfunction	Comments
Fan	Fails	Either one of the two fans will adequately cool.
Cooling Coil	Leaks	The cooling coil and fan in the second unit will cool.

* The normal switchgear room unit coolers (VUC-2A and VUC-2C) are not Seismic Category 1.

Table 9-22

SINGLE FAILURE ANALYSIS - MAKEUP PUMP ROOMS COOLING

(Based on two out of three pumps operating and the rooms having partial separation walls.)

Component	Malfunction	Comments
Fan	Fails	One out of three fans will cool.
Cooling coil	Leaks	One out of three cooling coils and their respective fans will cool.

Note: Makeup pumps remain operable without any of the unit coolers in operation.

Table 9-23

SINGLE FAILURE ANALYSIS - RADWASTE AREA VENTILATION

Component	Malfunction	Comments
Exhaust fan	Fails	Either one of the two fans will ventilate
Shut off damper on exhaust fan	Fails	The other damper and fan will ventilate
Supply fan	Fails	The other supply fan plus some in leakage from the turbine areas will ventilate.

NOTE - Although the exhaust filters are not redundant, it is felt that they are highly reliable and a failure is not credible. Since supply air passes through roughing filters, clogging of the exhaust filters is not anticipated. Pressure drop across the filters is measured and alarmed and preventive maintenance will be performed prior to excessive filter loading.

Table 9-24

SINGLE FAILURE ANALYSIS - FUEL HANDLING AREA VENTILATION

Component	Malfunction	Comments
Exhaust fan	Fails	Either one of the two fans will ventilate
Shut off damper on exhaust fan	Fails to open	The shut off damper on the second fan will operate.
Supply fan	Fails	The air will be made up through the non-insulated metal siding and through the door at the railroad track and up through the equipment hatch.

NOTE: - Although the exhaust filters are not redundant, it is felt that they are highly reliable and a failure is not credible. Since supply air passed through roughing filters, clogging of the exhaust filters is not anticipated. Pressure drop across the filters is measured and alarmed and preventive maintenance will be performed prior to excessive filter loading.

Table 9-25

**SCHEDULE OF AIR OPERATED VALVES
FOR SYSTEMS ASSOCIATED WITH SAFETY SYSTEMS**

Line size (in)	Valve No.	Valve Description	Valve Position: Normal Operation	Valve Position: Air Failure	Valve Position: During LOCA
3	CV-1052	Quench Tank Drain	C	C	C
3	CV-1065	Condensate for Quench Tank	C	C	C
6	CV-1432	DH Removal Cooler Bypass	C	C	C
6	CV-1433	DH Removal Cooler Bypass	C	C	C
½	CV-1845	Primary Coolant to Sampling	C	C	C
8	CV-2214	ICW from letdown coolers	O	*	C
8	CV-2233	ICW to letdown coolers	O	*	C
8	CV-2234	ICW to RCP motor coolers	O	*	C
36	CV-2691	Main steam from steam generator A	O	C	C
36	CV-2692	Main steam from steam generator B	O	C	C
8	CV-2668	Atmospheric Dump Valve for SG A	C	C	C
8	CV-2618	Atmospheric Dump Valve for SG B	C	C	C
1½	CV-3804	RB Spray Pump (P-35A) LO Coolers	C	O	O
1½	CV-3805	RB Spray Pump (P-35B) LO Coolers	C	O	O
1½	CV-3840	DH Removal pump cooler	C	O	O
1½	CV-3841	DH Removal pump cooler	C	O	O
4	CV-4400	RB sump drain	C	C	C
2	CV-4804	Vent header	C	C	C
6	CV-6202	Chilled water to RB cooling coils	O	*	C
6	CV-6203	Chilled water to RB cooling coils	O	*	C
24	CV-7401	RB purge valve	C	C	C

Table 9-25(continued)

Line size (in)	Valve No.	Valve Description	Valve Position Normal Operation	Air Failure	During LOCA
24	CV-7402	RB purge valve	C	C	C
24	CV-7403	RB purge valve	C	C	C
24	CV-7404	RB purge valve	C	C	C
12	CV-7621	Decay heat removal room ventilation	O	C	C
12	CV-7622	Decay heat removal room ventilation	O	C	C
½	CV-7635	Air to CV-7622 and CV-7638	O	C	C
½	CV-7636	Air to CV-7621 and CV-7637	O	C	C
12	CV-7637	Decay heat removal room ventilation	O	C	C
12	CV-7638	Decay heat removal room ventilation	O	C	C

* Remains open until closed by remote manual or ES signal, then remains closed.

Table 9-26

SCHEDULE OF AIR OPERATED VENTILATION SYSTEM DAMPERS FOR SYSTEMS ASSOCIATED WITH SAFETY SYSTEMS

Line Size (in)	Valve No.	Valve Description	Valve Position: Normal Operation	Air Failure	During LOCA
18x12	CV-2100	Penetration Rooms Ventilation	O	C	C
84x16	CV-2101	Penetration Rooms Ventilation	O	C	C
35x16	CV-2102	Penetration Rooms Ventilation	O	C	C
65x27	CV-2103	Penetration Rooms Ventilation	O	C	C
65x27	CV-2104	Penetration Rooms Ventilation	O	C	C
48x30	CV-2105	Penetration Rooms Ventilation	O	C	C
60x30	CV-2106	Penetration Rooms Ventilation	O	C	C
18x42	CV-2107	Penetration Rooms Ventilation	O	C	C
26x38	CV-2108	Penetration Rooms Ventilation	O	C	C
22x16	CV-2111	Penetration Rooms Ventilation	O	C	C
35x16	CV-2112	Penetration Rooms Ventilation	O	C	C
62x45	CV-2113	Penetration Rooms Ventilation	O	C	C
30x12	CV-2114	Penetration Rooms Ventilation	O	C	C
30x12	CV-2115	Penetration Rooms Ventilation	O	C	C
80x50	CV-2116	Penetration Rooms Ventilation	O	C	C
30x28	CV-7905	CR Ventilation Supply	O	C	C
30x30	CV-7907	CR Ventilation Return	O	C	C
16x16	CV-7910	VSF-9 Inlet From Computer Room	(1)	O	(2)

Notes:

(1) Damper position is intermediate during normal operation.

(2) Damper has two safety positions; open with VSF-9 operation, closed with 2VSF-9 operation. With air failure, safety related reserve air bottle is used to close as necessary.

Table 9-27

DELETED

Table 9-28

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Table 9-29

SPENT FUEL POOL STORAGE RACK ARRANGEMENT

Unit 1

	Region 1	Region 2	Region 3
Number of Storage Locations	220	484	264
Number of Rack Arrays	2 (10x11)	4 (11x11)	2 (12x11)
Center-to-Center Spacing (Inches)	10.65	10.65	10.65
Cell I.D. (Inches)	8.97	8.97	8.97
Type of Fuel	B&W 15x15 B&W HTP 15x15	B&W 15x15 B&W HTP 15x15	B&W 15x15 B&W HTP 15x15
Rack Assembly Dimensions (inches)	(10x11) 106.7x117.4x169.0	(11x11) 118.2x118.2x169.0	(12x11) 128.8x118.2x169.0
Dry Weights (lbs)	27,500 (10x11 with Boraflex panels)	18,000 (11x11)	31,309 (12x11 with Metamic PIAs)

Table 9-30

APPENDIX B SUPPLEMENT TO GENERIC LICENSING TOPICAL REPORT EDR-1 FOR SPENT FUEL HANDLING CRANE L-3 SUMMARY OF PLANT SPECIFIC CRANE DATA

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.1.a	III.C(C.1.a)	1. The actual crane duty classification of the crane specified by the applicant.	1. The trolley has a Class “C” crane duty classification in accordance with CMAA Specification #70. The bridge has a Class A crane duty classification.
C.1.b	III.C(C.1.b)	1. The minimum operating temperature of the crane specified by the applicant.	1. The trolley was designed and fabricated for a minimum operating temperature of 30 °F.
C.2.b	III.C(C.2.b) III.E.4	1. The maximum extent of load motion and the peak kinetic energy of the load following a drive train failure. 2. Provisions for actuating the emergency drum brake prior to traversing with the load, when required to accommodate the load motion following a drive train failure.	1. The Main Hoist was designed such that the maximum vertical load motion following a drive train failure does not exceed 1.5 feet and the maximum kinetic energy of the load is less than that resulting from one inch of free fall of the maximum critical load. 2. Provisions for automatically actuating the emergency drum brake prior to traversing with the load are not required since the maximum amount of load motion and kinetic energy has been factored into the facility design of the spent fuel pool floor (Elevation 404’-0”). This elevation can accommodate the load motion and the load will be administratively controlled by Procedure 1402.133 to maintain greater than or equal to 1.5 feet when traversing the Elevation 404’-0” floor.
C.3.e	III.C(C.3.e)	1. The maximum cable loading following a wire rope failure in terms of the acceptance criteria established in Section III.C(C.3.e).	1. The maximum cable loading following a wire rope failure in the Main Hoist meets the maximum allowed by the acceptance criteria established in Section III.C(C.3.e).

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.3.f	III.C(C.3.f)	1. Maximum fleet angle 2. Number of reverse bends 3. Sheave diameter	1. 3.5 degrees, Main Hoist. 2. None, other than the one between the wire rope drum and the first sheave in the load block. 3. 18 x wire rope diameter, Main Hoist.
C.3.h	III.C(C.3.h) III.E.11	1. The maximum extent of motion and peak kinetic energy of the load following a single wire rope failure.	1. The Main Hoist was designed such that the maximum load motion following a single wire rope failure is less than 1.5 feet and the maximum kinetic energy of the load does not exceed that resulting from one inch of free fall of the maximum critical load.
C.3.i	III.C(C.3.i)	1. The type of load control system specified by the applicant. 2. Whether interlocks are recommended by Regulatory Guide 1.13 to prevent trolley and bridge movements while fuel elements are being lifted and whether they are provided for this application.	1. A. Ederer AC flux vector, Main Hoist. B. Shepard Niles mechanical load brake, Auxiliary Hoist. 2. The crane will not be used to lift fuel elements from the spent fuel racks. Therefore, interlocks to prevent trolley and bridge movements while hoisting have not been provided.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.3.j	III.C(C.3.j)	1. The maximum cable and machinery loading that would result in the event of a high speed two blocking, assuming a control system malfunction that would allow the full breakdown torque of the motor to be applied to the drive motor shaft. 2. Means of preventing two blocking of auxiliary hoist, if provided.	1. The energy absorbing torque limiter (EATL) was designed such that the maximum machinery load, which would result in the event a two-blocking occurs while lifting the rated load at the rated speed. This allows the full breakdown torque of the motor to be applied to the drive shaft and it will not exceed 3 times the design rated loading. In addition, the EATL design does not allow the maximum cable loading to exceed the acceptance criteria established in Section III.C(C.3.e) during the above described two-blockings. 2. The 15 Ton Auxiliary Hoist has a geared upper limit switch and an arm actuated up over-travel switch.
C.3.k	III.C(C.3.k)	1. Type of drum safety support provided.	1. The alternate design drum safety restraint shown in Figure III.D.4 of EDR-1 is arranged to counter gear and brake forces as well as downward loads. These brackets act on the diameter of the ends of the drum on the Main Hoist.
C.3.o	NA	1. Type of hoist drive to provide incremental motion.	1. AC flux vector.
C.3.p	NA	1. Maximum trolley speed. 2. Maximum bridge speed. 3. Type of overspeed protection for the trolley and bridge drives.	1. 28 fpm. 2. 25 fpm. 3. Overspeed switches which actuate the brakes are provided for the trolley and bridge drives.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.3.q	NA	1. Control station location.	1. The complete operating control system, including an emergency stop button, is located on the remote radio control station. An additional emergency stop button is located on the pendant station, permitting de-energization of the crane independent of the control station.
NA	III.D.1	1. The type of emergency drum brake used, including type of release mechanism. 2. The relative location of the emergency drum brake. 3. Emergency drum brake capacity.	1. A pneumatically released band brake is used for the Main Hoist. 2. The emergency drum brake engages the wire rope drum of the Main Hoist. 3. The Main Hoist emergency drum brake has a minimum capacity of 125% of that required to hold the design rated load.
NA	III.D.2	1. Number of friction surfaces in EATL. 2. EATL torque setting.	1. The main surface of the EATL has 21 friction surfaces. 2. The specified EATL torque setting is approximately 130% of the Main Hoist design rated load.
NA	III.D.3	1. Type of failure detection system.	1. A totally mechanical drive train continuity detector and emergency drum brake actuator have been provided in accordance with Appendix G of Revision 3 of EDR 1 for the Main Hoist.
NA	III.D.5	1. Type of hydraulic load equalization system.	1. The Main Hoist hydraulic load equalization system includes both features described in Section III.D.5.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
NA	III.D.6	1. Type of hook.	1. Both the Main and Auxiliary Hooks have a single load path.
		2. Hook design load.	2. A. The Main Hook design critical lift load is 130 tons with a 10:1 factor of safety on ultimate. B. The Auxiliary Hook design lift load is 15 tons with a 5:1 factor of safety on ultimate.
		3. Hook load test.	3. The test load for each load path of the Main Hook is 260 tons.
NA	III.F.1	1. Design rated load.	1. Main Hoist – 130 Ton Auxiliary Hoist – 15 Ton
		2. Maximum Critical Load (MCL) rating.	2. Main Hoist – 130 Ton Auxiliary Hoist – NA
		3. Trolley weight (net).	3. 74,000 lb (incl. Hooks)
		4. Trolley weight (with load).	4. 334,000 lb
		5. Hook lift.	5. Main Hook – 80 feet, 0 inches Auxiliary Hook – 80 feet, 0 inches
		6. Number of wire rope drums.	6. The Main and Auxiliary Hoist each have one wire rope drum.
		7. Number of parts of wire.	7. Main Hoist – 4 parts per wire rope, 2 ropes, with (2) ropes off drum.
		8. Drum size (pitch diameter).	8. Main Hoist – 33 inches Auxiliary Hoist – 16 inches
		9. Wire rope diameter.	9. Main Hoist – 1-3/8 inches Auxiliary Hoist – 5/8 inch

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
	III.F.1 (continued)	10. Wire rope type.	10. Main Hoist – Spelter Socket Williamsport Wire Ropes Works Royal Purple Plus Triple PAC Auxiliary Hoist – 6 x 37 EIPS/IWRC
		11. Wire rope material.	11. Carbon steel Main and Auxiliary Hoist.
		12. Wire rope breaking strength.	12. Main Hoist – 259,200 lb Auxiliary Hoist – 41,200 lb
		13. Wire rope yield strength.	13. Main Hoist – 207,360 lb Auxiliary Hoist – NA
		14. Wire rope reserve strength.	14. Main Hoist – 0.5777 Auxiliary Hoist – NA
		15. Number of wire ropes.	15. The Main Hoist has 2 ropes. The Auxiliary Hoist has one rope.
	III.C(C.1.b(1))	1. The extent of venting of closed box sections.	1. Closed box sections are not vented since the auxiliary building that houses the crane will not be pressurized.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.1.b(3) C.1.b(4) C.4.d	III.C(C.1.b(3)) III.C(C.1.b(4)) III.C.(C.4.d)	1. The nondestructive and cold proof testing to be performed on existing structural members for which satisfactory impact test data is not available.	1. The procurement documents for the modified bridge structure did not invoke 10 CFR 50 Appendix B. An installation plan, to capture all of the critical characteristics of the structural members, is being used to ensure the structural members meet the requirements for the crane's intended function. Cold proof testing has been performed on the modified bridge, followed by a visual inspection of all accessible welds whose failure would result in the drop of a load. Visual inspections of structural degradation of the modified bridge were performed and no degradation was identified. Additional NDE of the critical welds on the bridge girder was performed. The ambient temperature when the 125% (+0%, -5%) static load test is performed is the minimum operating temperature for the crane. The minimum crane operating temperature has been established at 65 °F. In the event that the crane must be operated at a lower temperature, another 125% (+0%, -5%) static proof test will be performed at the lower temperature.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.1.c	III.C.(C.1.c)	1. The extent the crane's structures which are not being replaced are capable of meeting the seismic requirements of Regulatory Guide 1.29.	1. The modified bridge structure and new trolley have been analyzed. Existing steel and concrete support structures have been analyzed for the design basis earthquake while supporting the maximum critical loads. This is documented in Entergy Calculation 61, Bechtel Book 21. Analysis methods and acceptance criteria are consistent with the ANO design basis for the support structure and runway girder, and meet the intent of Reg Guide 1.29, C.2.
C.1.d	III.C(C.1.d)	1. The extent welds joints in the crane's structures, which are not being replaced, were nondestructively examined 2. The extent the base material, at joints susceptible to lamellar tearing, was nondestructively examined.	1. Nondestructive examinations of the existing bridge structure were not required by existing regulations at the time of construction. However, the X-SAM® system provides additional overload protection, and the inspections of the existing structure described in C.1.b(3) above are adequate to ensure the structural integrity of the existing bridge. 2. The weld geometries used in (a) the existing bridge structure and (b) the replacement trolley structure are not considered to be susceptible to lamellar tearing.
C.1.e	III.C(C.1.e)	1. The extent the crane's structures, which are not being replaced are capable of withstanding the fatigue effects of cyclic loading from previous and projected usage, including any construction usage.	1. All past and projected uses of the modified structural components were assessed to ensure the crane is within the cyclic loading capability of the modified bridge structure and welds at 130 Tons for CMAA Class "A" service.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.1.f	III.C(C.1.f)	1. The extent the crane's structures which are not being replaced, were post-weld heat treated in accordance with Sub article 3.9 of AWS D1.1, "structural welding code".	1. The material thickness of the modified bridge components are such that paragraph III.C(C.1.f) of EDR-1 does not require post weld heat treatment.
C.2.b	III.C(C.2.b) III.E.4	1. Provisions for accommodating the load motion and kinetic energy following a drive train failure when the load is being traversed and when it is being raised or lowered.	1. Administrative procedures will be used to assure that a minimum of 1.5 feet of clearance is maintained between the load and surfaces that cannot withstand the kinetic energy associated with one inch free fall of the load involved. The spent fuel pool floor laydown area (Elevation 404'-0") can withstand the kinetic energy associated with one inch free fall of the MCL, documented in Calculation 92-D-2001-62.
C.2.c	III.C(C.2.c)	1. Location of safe laydown areas for use in the event repairs to the crane are required that cannot be made with the load suspended.	1. The laydown areas that can be used in the event that repairs to the crane are required that cannot be made with the load suspended are the spent fuel pool floor and Elevation 404'-0" laydown area, documented in Procedure 1402.133.
C.2.d	III.C(C.2.d)	1. Size of modified components that can be brought into the building for repair of the crane without having to break the building integrity. 2. Location of area where repair work can be accomplished on the crane without affecting the safe shut-down capability. 3. Any limitations on operations that would result from crane repairs.	1. The X-SAM® trolley and modified bridge components can be brought in through the auxiliary building floor opening. The opening is 12'-0" x 24'-0". 2. Area is identified in Procedures 1402.133 and 1402.135. 3. No limitations for normal operations.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.3.b	III.C(C.3.b)	1. The design margin and type of lifting devices that are attached to the hook to carry critical loads.	1. As an alternative to a dual load path system, the normal stress design factors have been doubled. The maximum critical load lifting device attached to the hook to carry critical loads will support a load six times the static plus dynamic load being handled without permanent deformation. The safety factor is 10:1 when compared to ultimate. This is in accordance with NUREG-0612, Section 5.1.6, Paragraph 1(a) and ANSI N14.6, Section 7.2.1.
C.3.t	III.C(C.3.t)	1. The extent construction requirements for the crane's structures, which will not be replaced, are more severe than those for permanent plant service. 2. The modifications and inspections to be accomplished on the crane following construction use, which was more severe than those for permanent plant service.	1. Prior use and load histories have been documented and reviewed for the modified bridge components as part of the final closeout information documented in Entergy MAI's and ER-ANO-2000-2688-02 associated with the spent fuel crane modifications. 2. Nondestructive examination of the accessible load bearing weld seams, and justification that the fatigue life of the modified components has not been compromised, were completed prior to the 125% design load test.
C.3.u	NA	1. The extent of installation and operating instructions.	1. The installation and operating instructions will be updated to fully comply with the requirements of Section C.3.u of Regulatory Guide 1.104 and Sections 7.1 and 9 of NUREG-0554.

Table 9-30 (continued)

Reg. Guide 1.104 Position	EDR-1 Topical Report Section	Information to be Provided	Specific Crane Data
C.4.a C.4.b C.4.c C.4.d	NA	1. The extent of assembly checkout, test procedures, load testing and rated load marking of the crane.	1. Prior to handling critical loads, the crane was given a complete assembly checkout, and then given a no-load test of all motions in accordance with updated procedures provided by Ederer. A 125% static load test and 100% performance test were also performed at this time in accordance with updated test procedures provided by Ederer. A no-load test of all motions and a two blocking test were performed by Ederer prior to delivery of the crane per Topical Report EDR 1. The maximum critical load is plainly marked on each side of the crane.
C.5.a	III.C(C.5.a)	1. The extent the procurement documents for the crane's structure's, which will not be replaced, required the crane manufacturer to provide a quality assurance program consistent with the pertinent provisions of Regulatory Guide 1.28.	1. The procurement documents for the components of the modified bridge structure did not invoke 10 CFR 50 Appendix B. However, these components were built to the manufacturer's quality control processes. Quality assurance provisions denoted in the procurement documents covered such items as design control, material selection and inspection and testing. The installation of the trolley and any structural modifications to the existing bridge is controlled by the Arkansas Nuclear One quality assurance plan and design change package ER-ANO-2000-2688.

Table 9-31

MINIMUM BURNUP VERSUS ENRICHMENT VAUES FOR REGION 1 RACKS WITH SPENT FUEL

Enrichment	2.0	2.5	3.0	3.5	4.0	4.5	5.0
Cooling Time	Minimum Burnup (GWD/MTU)						
0	2.3	9.2	15.5	22.1	27.7	33.0	39.0
5	2.2	8.7	14.8	21.1	26.7	31.1	37.1
10	2.1	8.3	14.0	20.0	25.6	29.8	35.3
15	2.0	8.1	13.6	19.4	25.3	29.1	34.0
20	2.0	8.0	13.5	19.0	24.6	28.6	33.3

Table 9-32

MINIMUM BURNUP VERSUS ENRICHMENT VAUES FOR REGION 2 RACKS WITH SPENT FUEL

Enrichment	2.0	2.5	3.0	3.5	4.0	4.5	5.0
Cooling Time	Minimum Burnup (GWD/MTU)						
0	4.5	11.7	18.7	25.7	30.6	36.9	42.8
5	4.2	11.0	17.6	24.2	29.1	34.4	40.7
10	4.0	10.6	16.7	23.0	28.1	33.0	38.6
15	4.0	10.1	15.9	22.4	27.4	31.8	37.4
20	4.0	9.8	15.7	21.8	26.8	31.2	36.4

DELETED

SAR FIGURE NO. 9-1

AMENDMENT 20

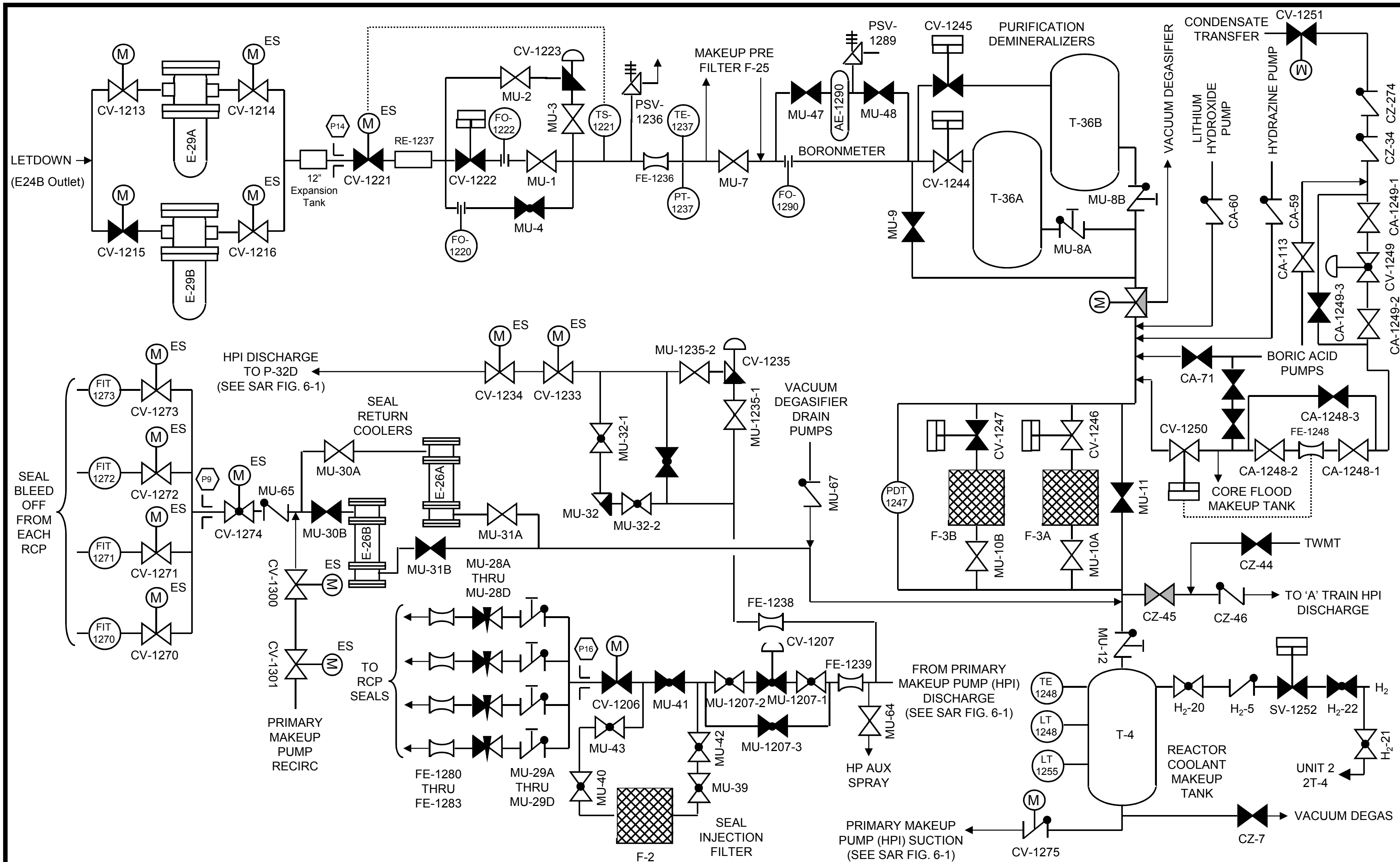
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PIPING SYMBOLS & DWG'S INDEX	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

SAR FIGURE NO. 9-2

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
INSTRUMENTATION SYMBOLS	BASED ON DRAWING NO	SHEET 1 / REV.



MAKEUP AND PURIFICATION SYSTEM

SAR FIGURE NO. 9-3

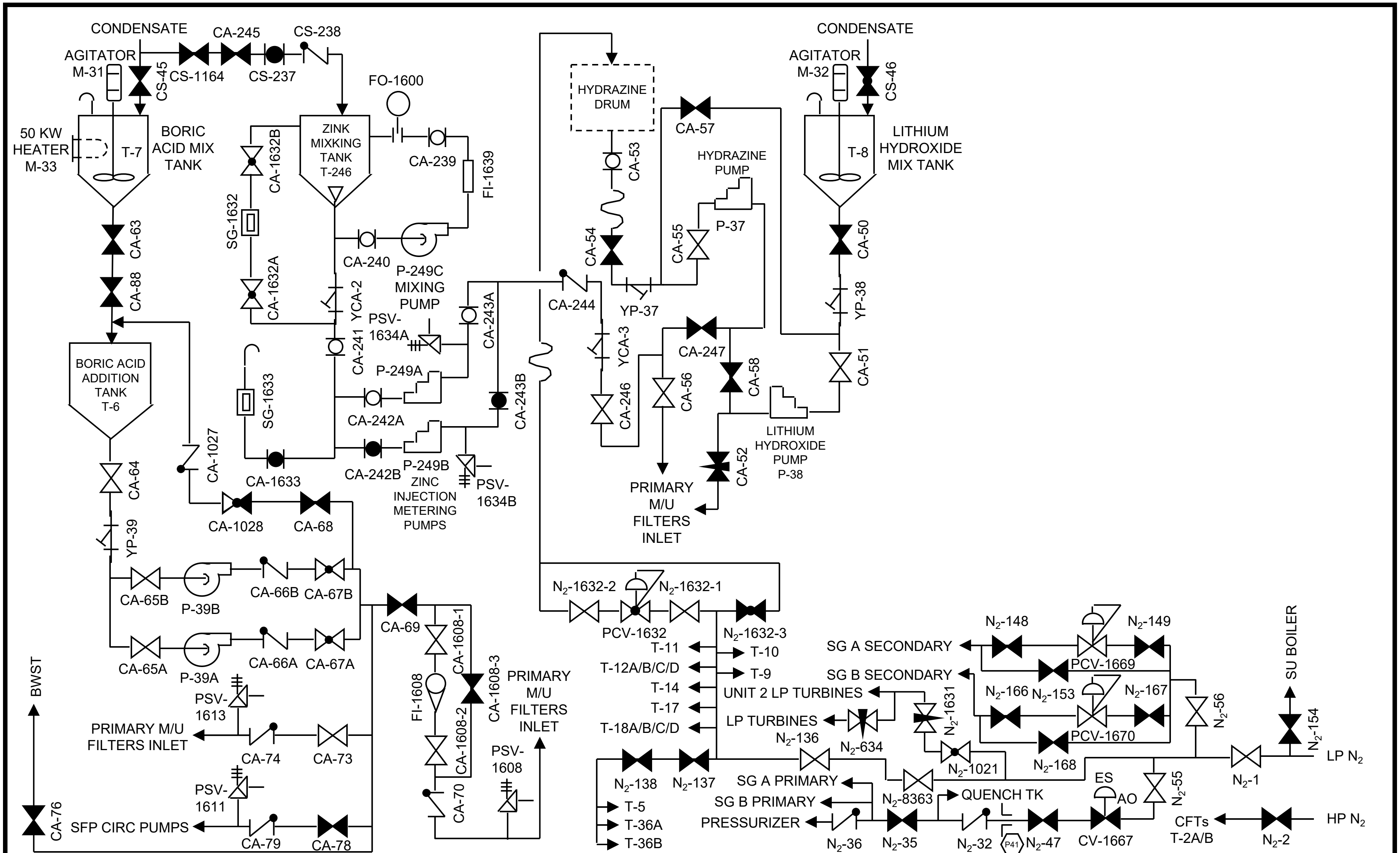
AMENDMENT 24

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



CHEMICAL ADDITION SYSTEM

SAR FIGURE NO. 9-4

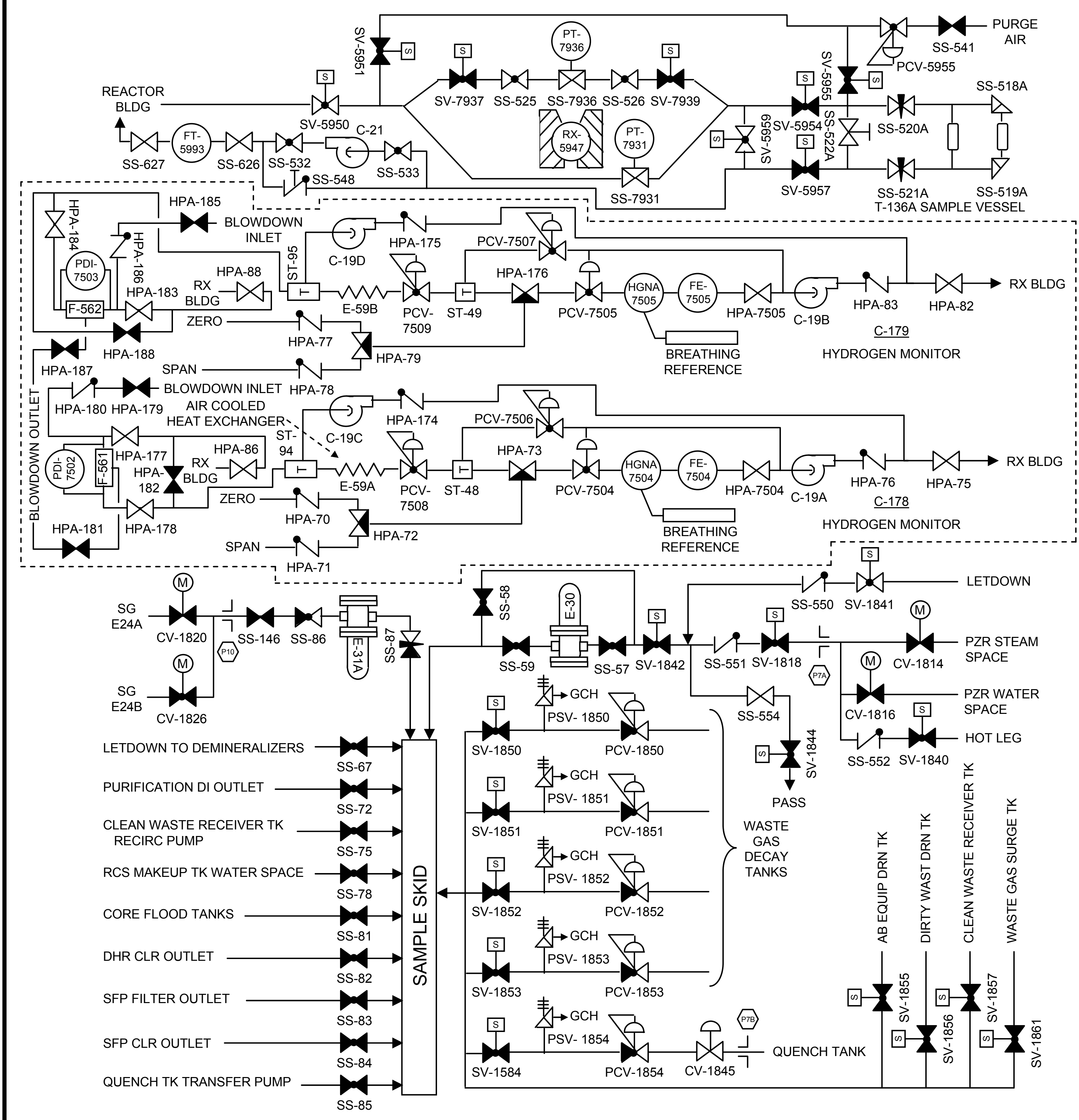
ARKANSAS NUCLEAR ONE

UNIT 1
RUSSELLVILLE, ARKANSAS

AMENDMENT 30

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

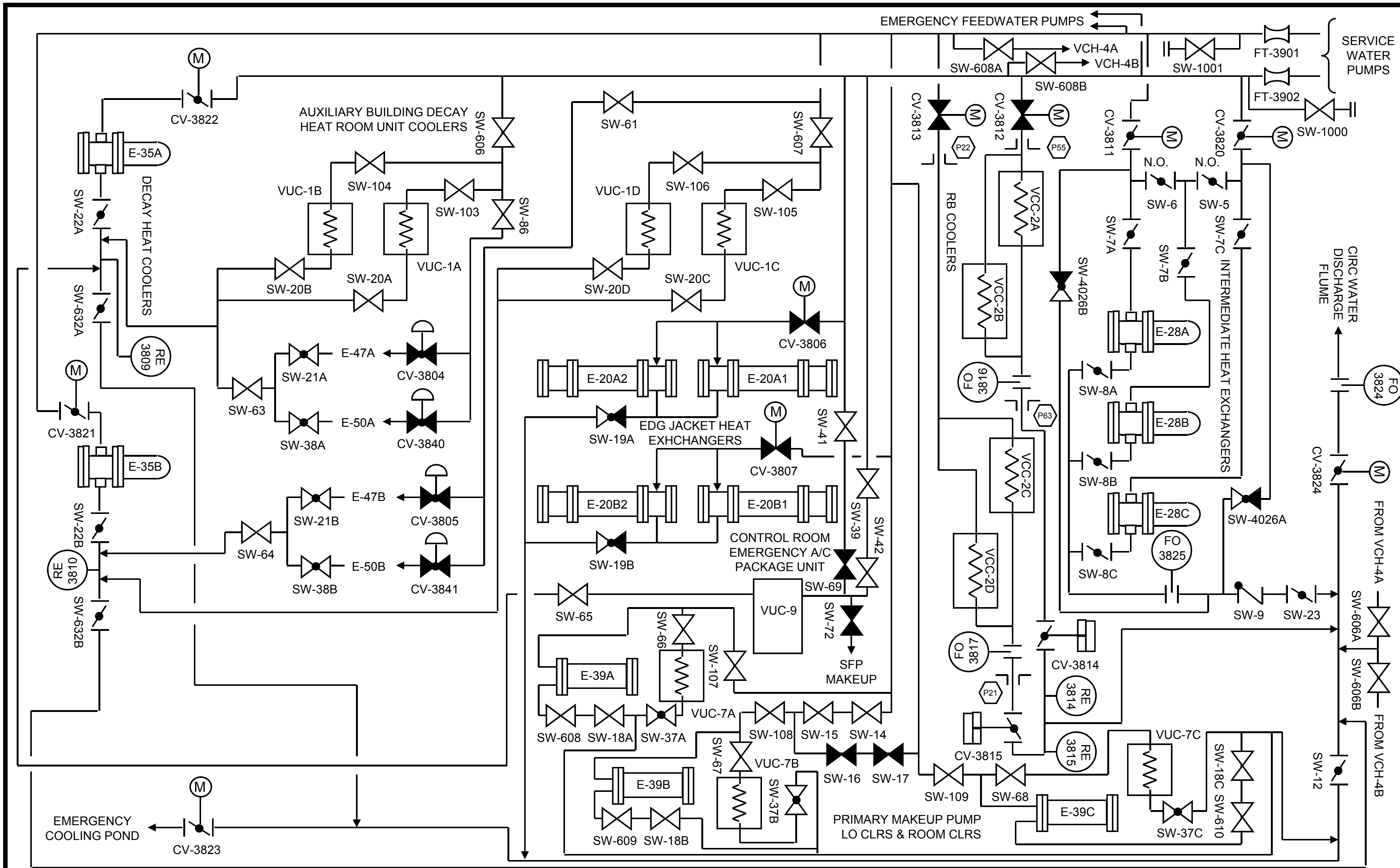
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 9-5

AMENDMENT 22

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PRIMARY & CONTAINMENT ATMOSPHERE SAMPLING SYSTEM	BASED ON DRAWING NO	SHEET 1 / REV.



SERVICE WATER

SAR FIGURE NO. 9-6

AMENDMENT 26

ARKANSAS NUCLEAR ONE

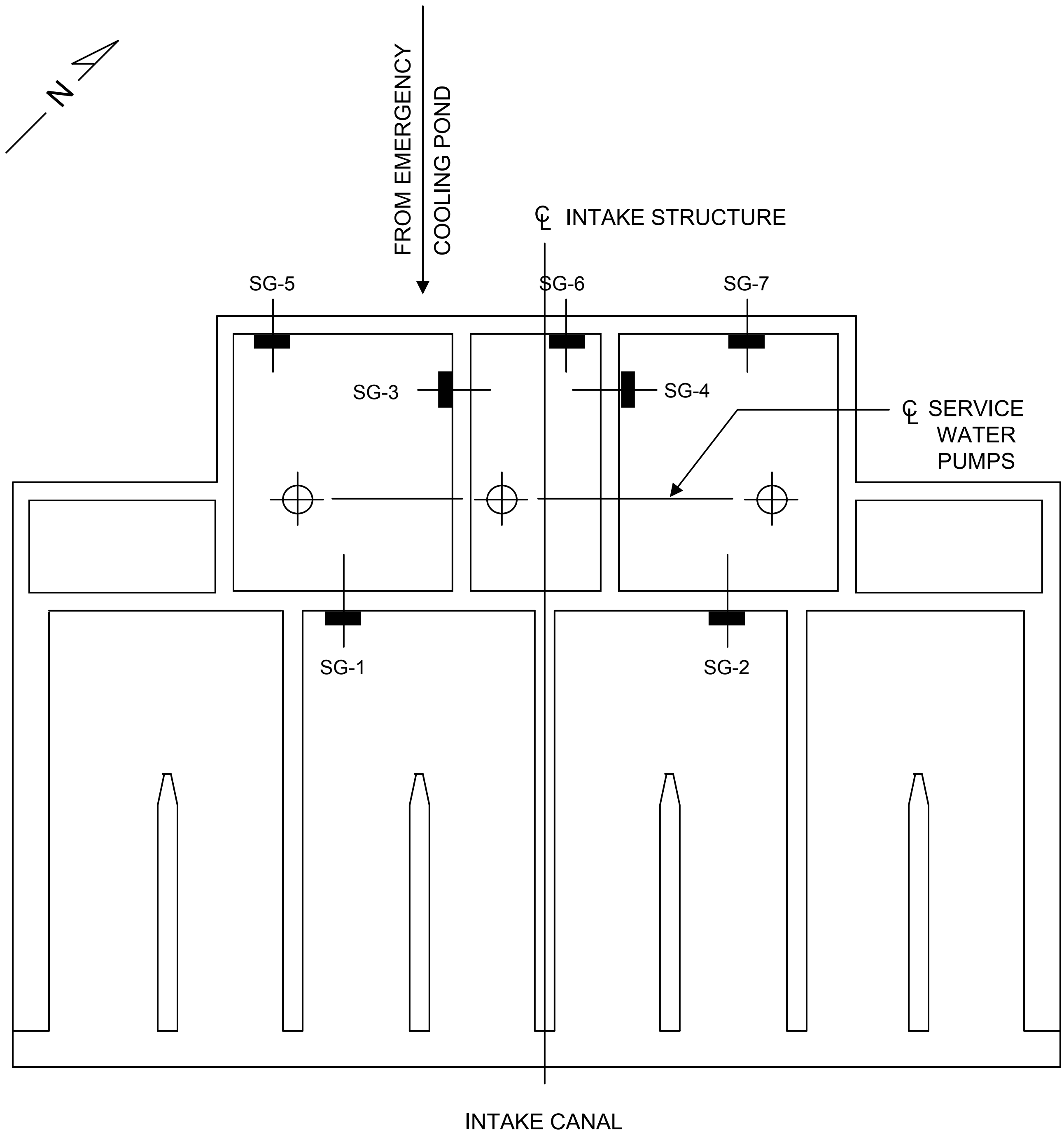
UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



PLAN BELOW EL. 354' – 0"

SAR FIGURE NO. 9-6A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
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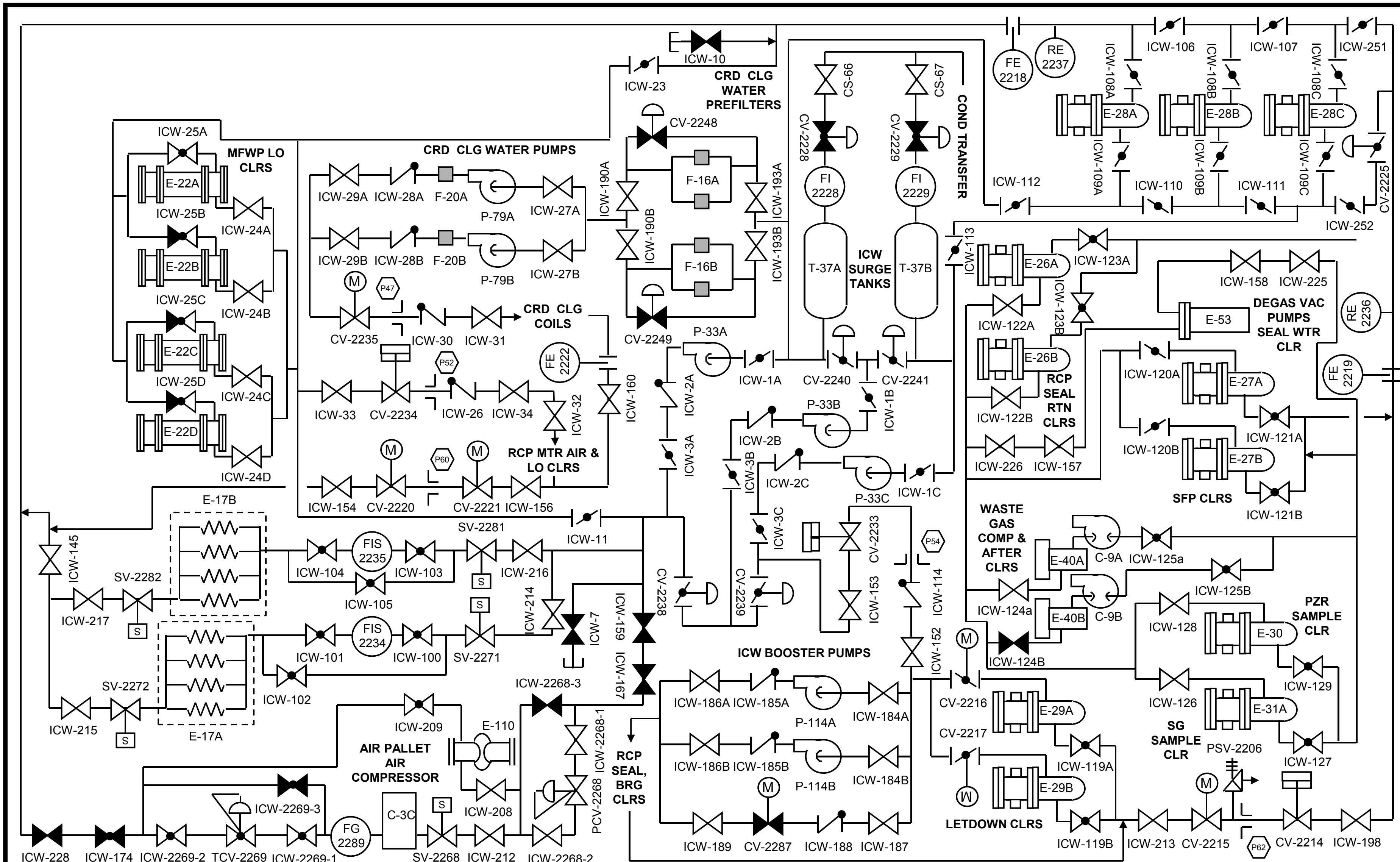
		DESIGN:	ENTERGY
		CAD NO:	
SLUICE GATE ARRANGEMENT	BASED ON DRAWING NO	SHEET 1	REV.



INTERMEDIATE COOLING SYSTEM

SAR FIGURE NO. 9-7

ARKANSAS NUCLEAR ONE

UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 27

BASED ON DRAWING NO	SHEET	REV.

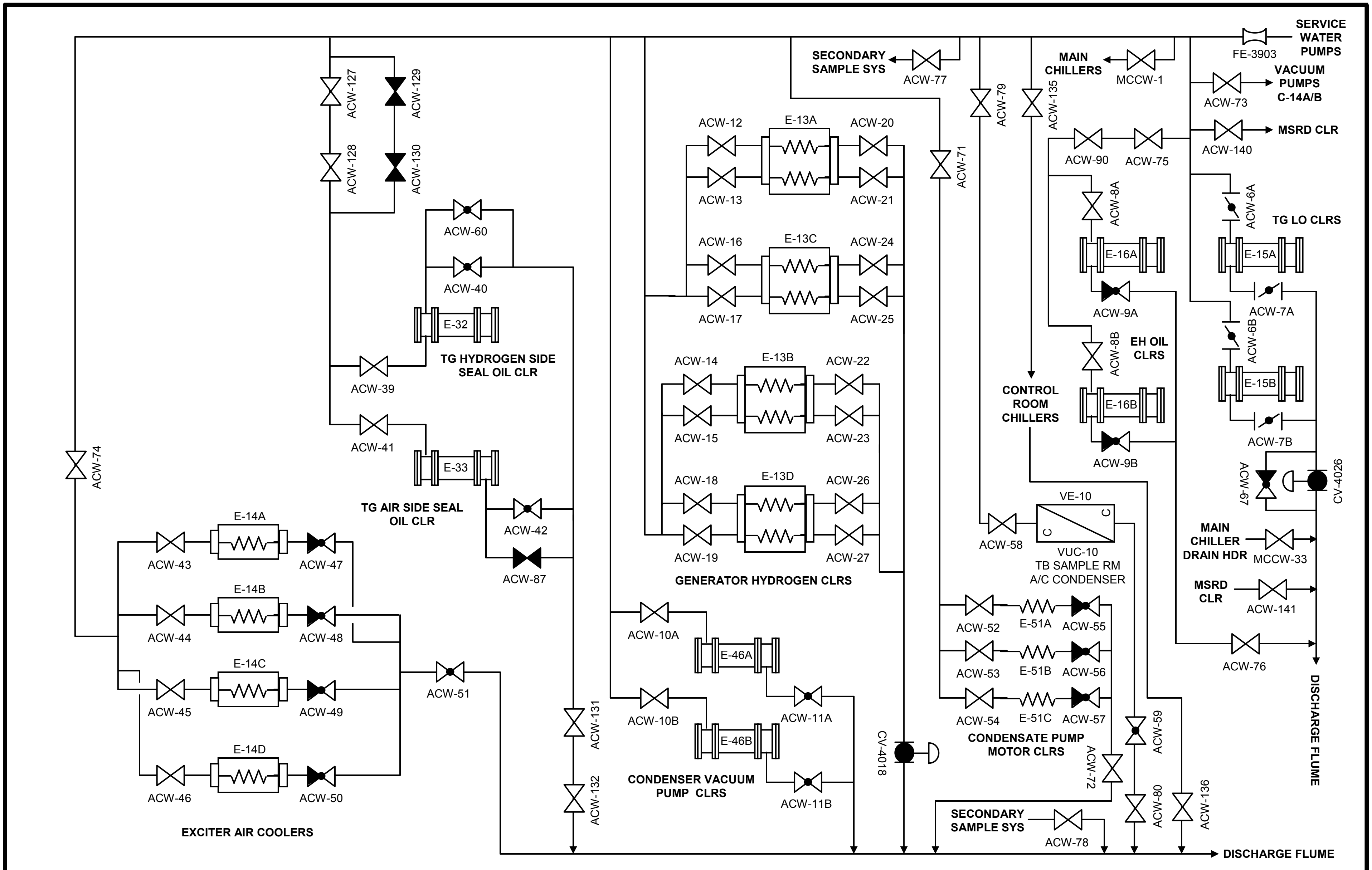
DELETED

SAR FIGURE NO. 9-8

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

INTERMEDIATE COOLING SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.



AUXILIARY COOLING WATER

SAR FIGURE NO. 9-9

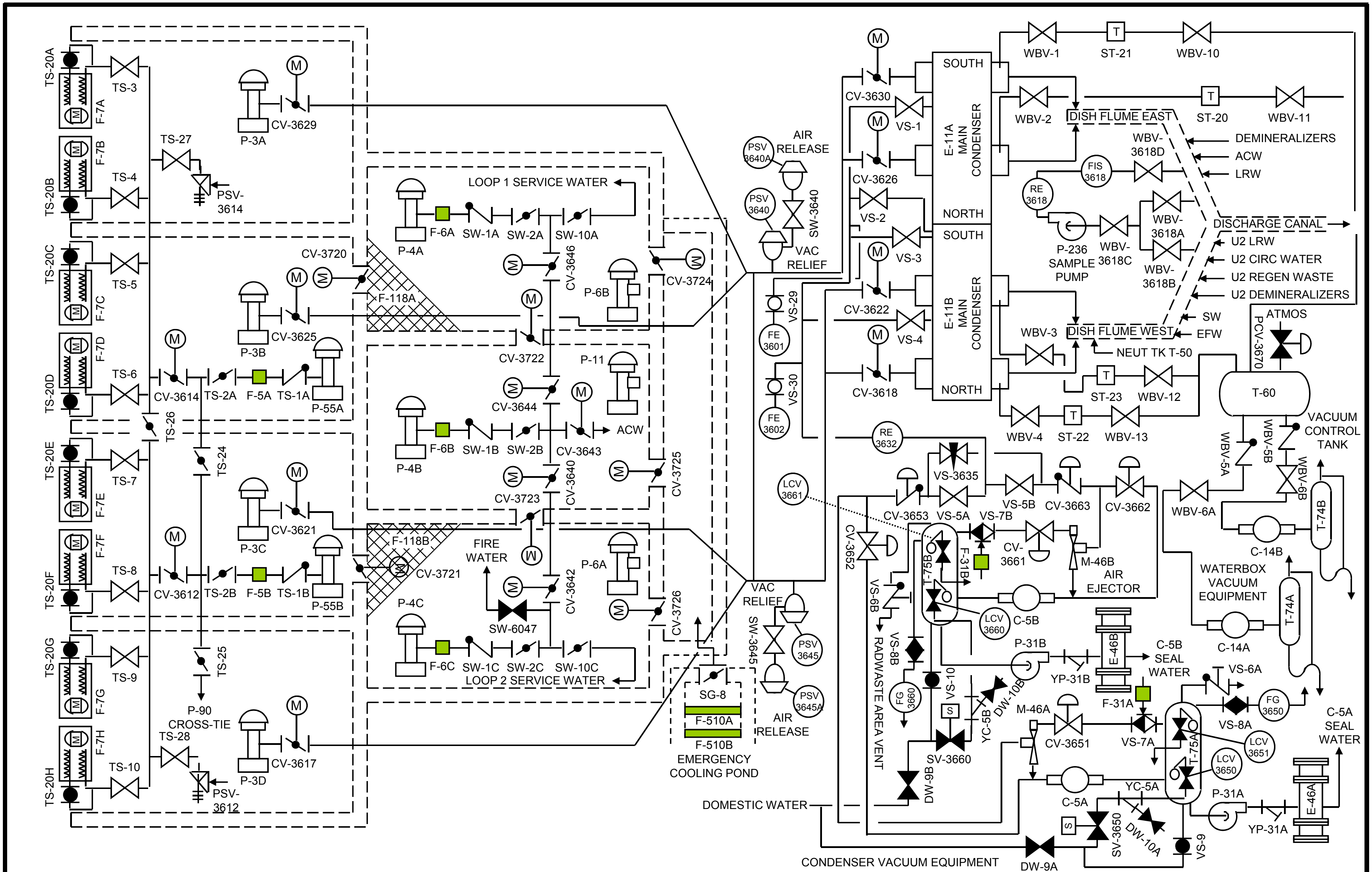
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	NONE

BASED ON DRAWING NO	SHEET	REV.



CIRC. WATER, SERVICE WATER, FIRE WATER, CONDENSER VACUUM, SCREEN WASH, INTAKE STRUCTURE, AND DISCHARGE STRUCTURE

SAR FIGURE NO. 9-10

AMENDMENT 26

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	NONE

BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 9-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
SPENT FUEL POOL COOLING SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

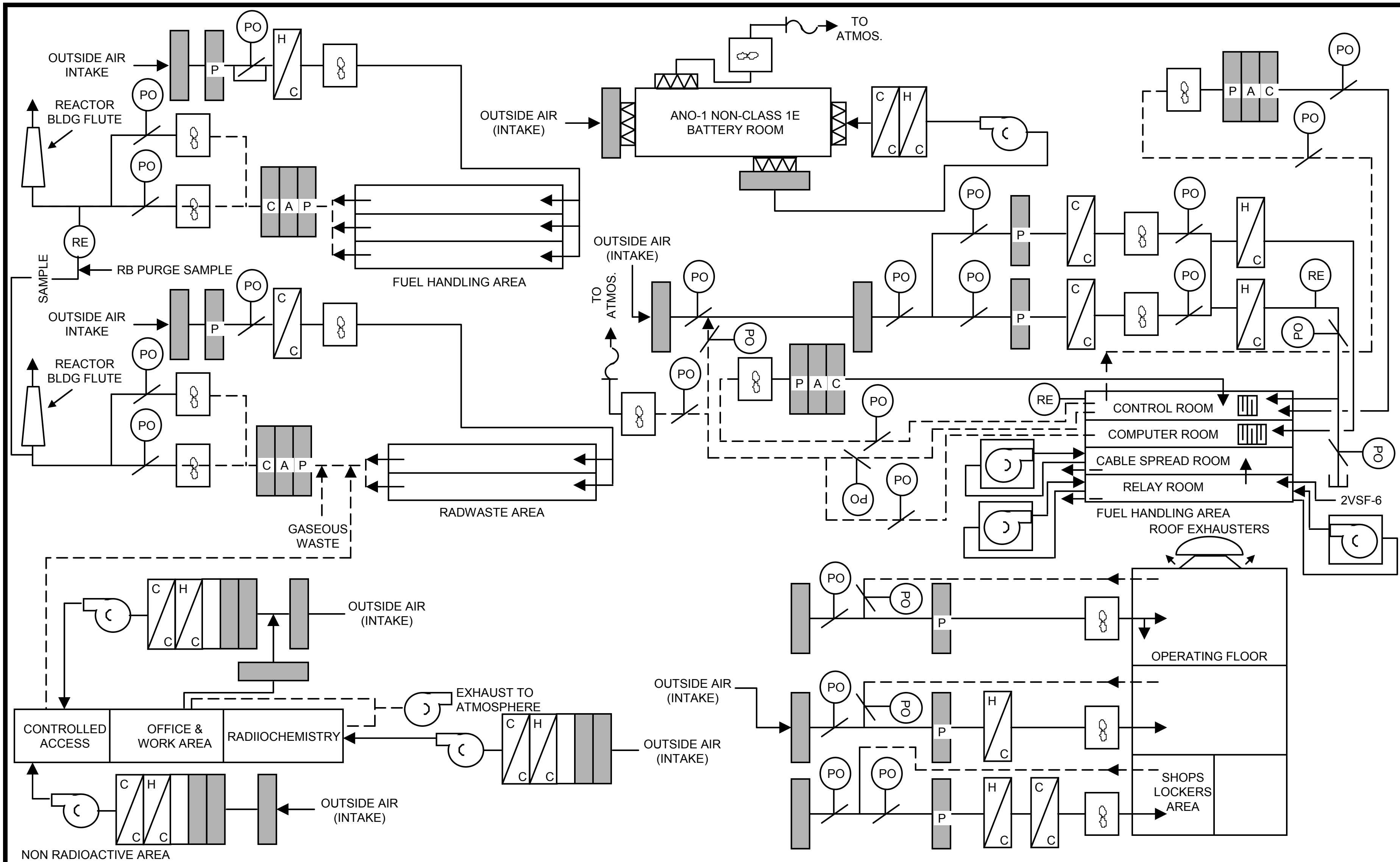
DELETED

SAR FIGURE NO. 9-12

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

DECAY HEAT REMOVAL SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.



TURBINE & AUXILIARY BUILDING VENTILATION SYSTEM

SAR FIGURE NO. 9-13

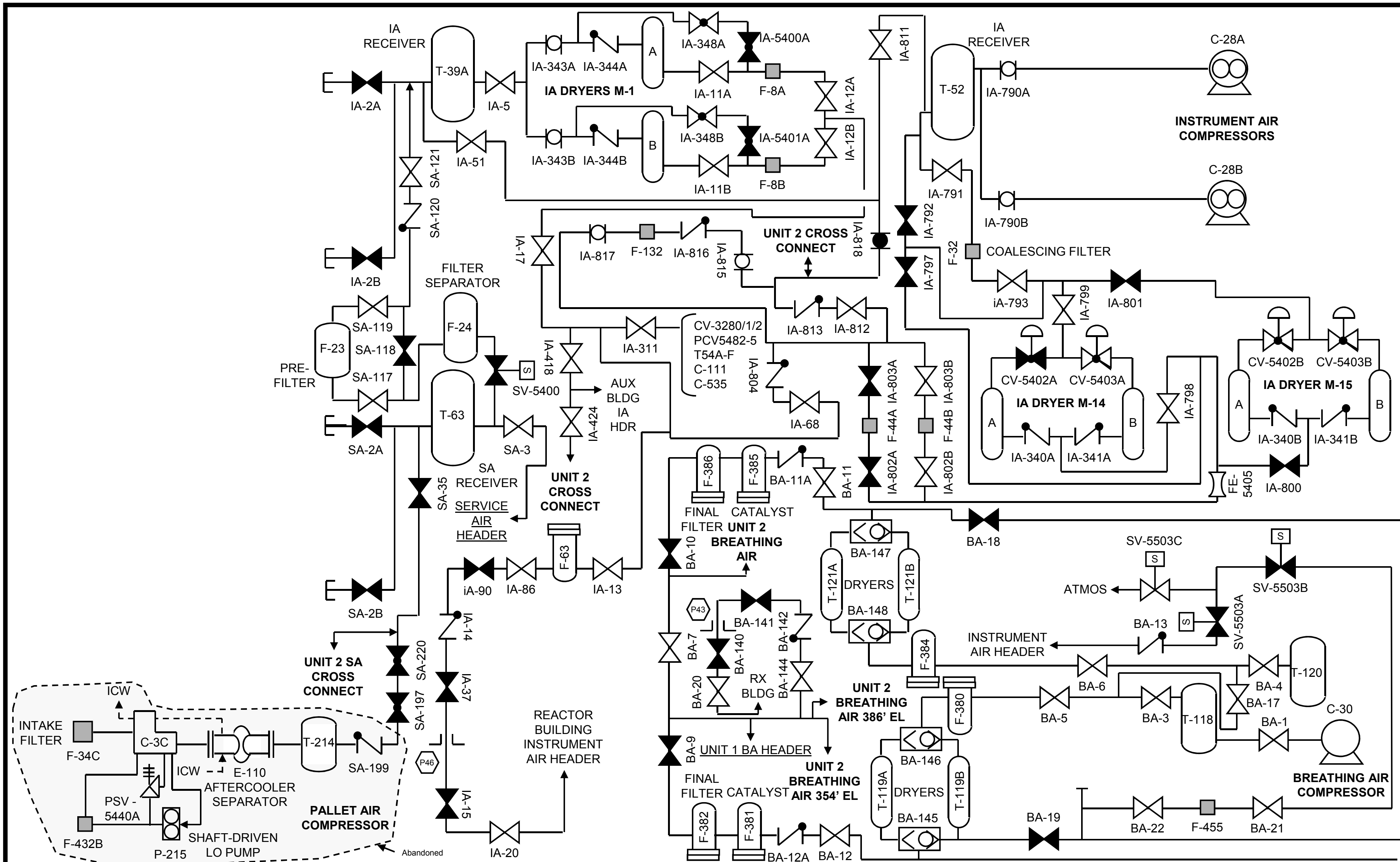
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



INSTRUMENT, SERVICE, & BREATHING AIR SYSTEM

SAR FIGURE NO. 9-14

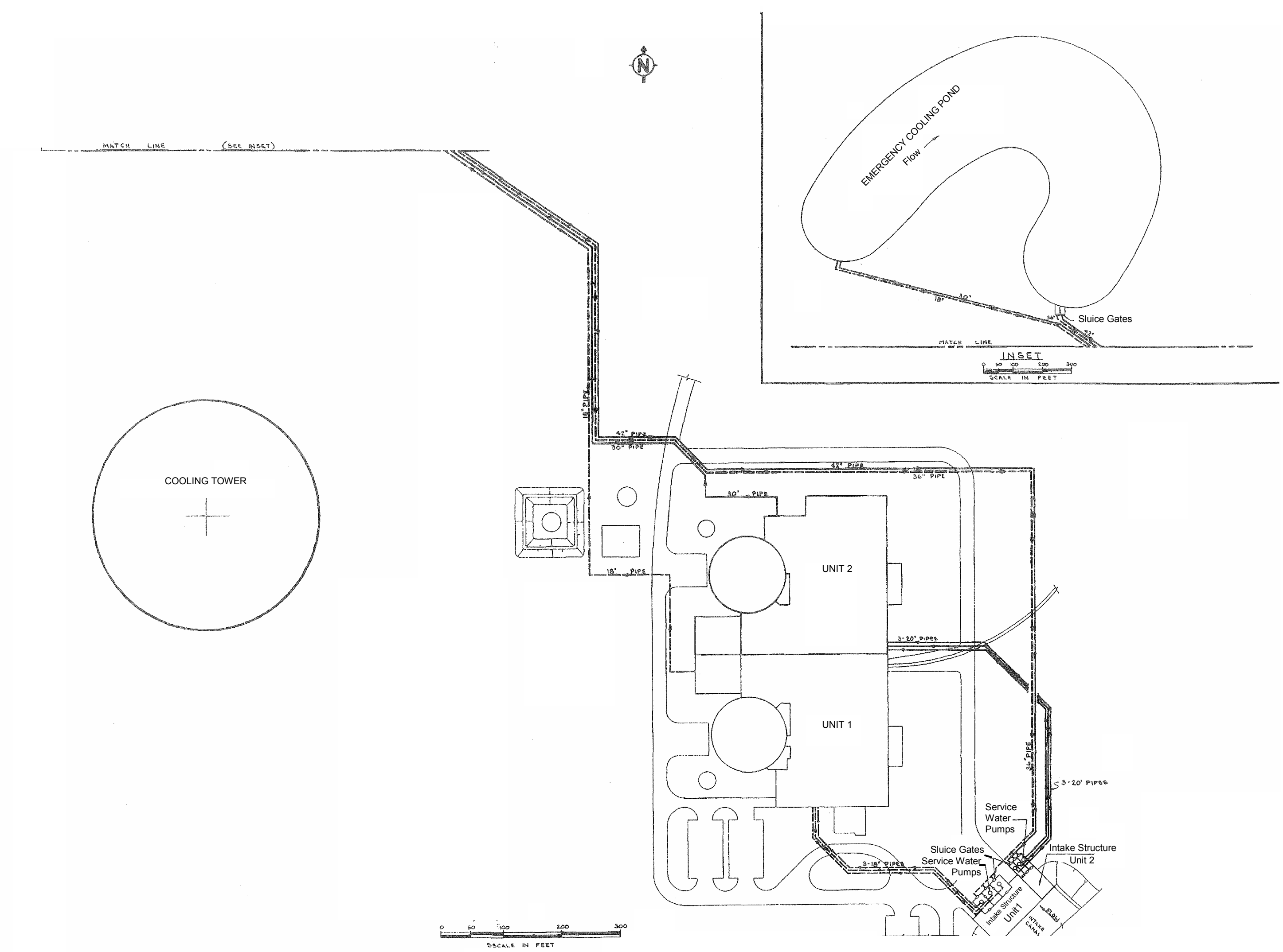
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

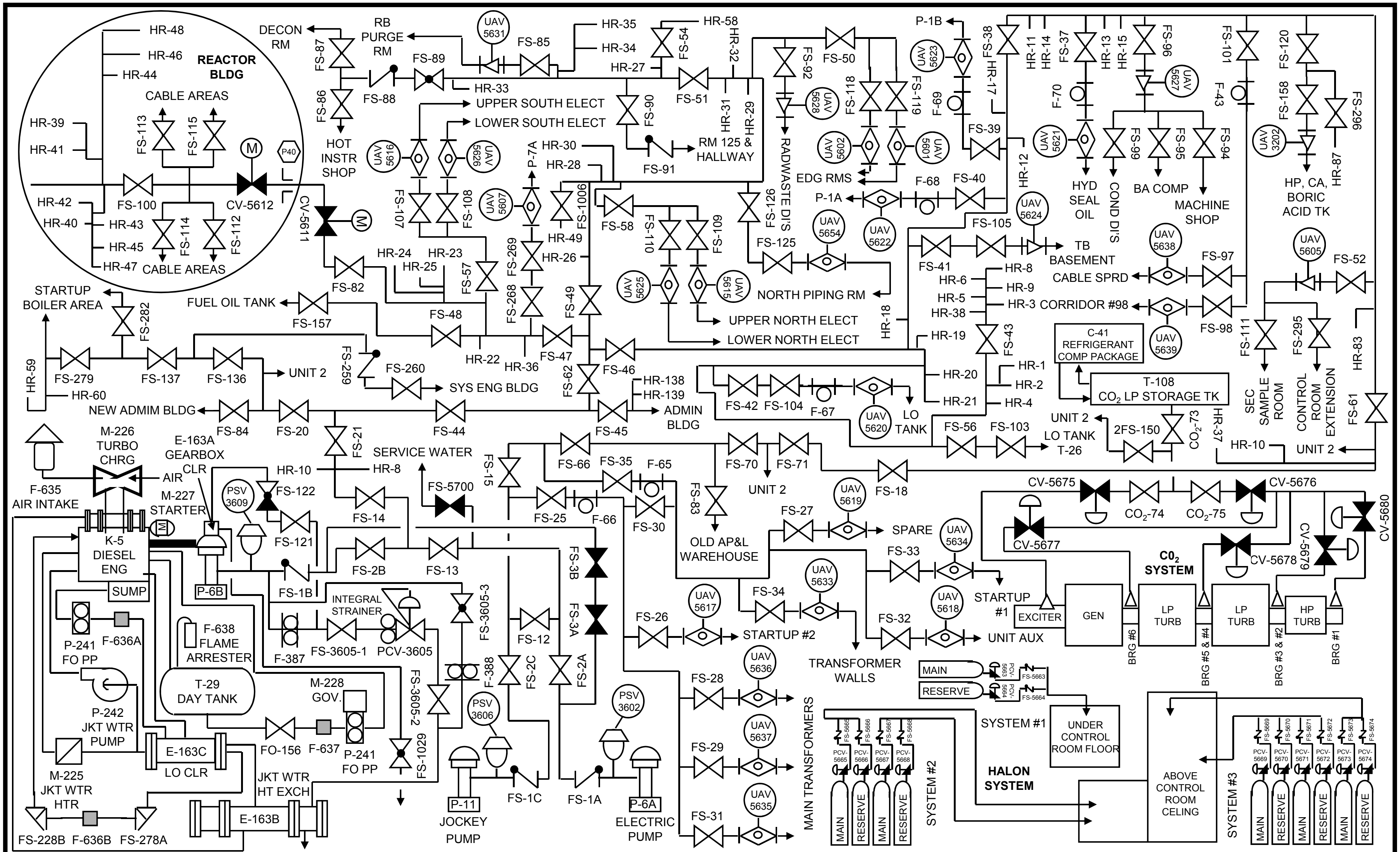
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 27

BASED ON DRAWING NO	SHEET	REV.



SERVICE WATER SYSTEM WATER SOURCES			SAR FIGURE NO. 9-15		
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20		
			BASED ON DRAWING NO	SHEET	REV.



FIRE WATER

SAR FIGURE NO. 9-16

AMENDMENT 26

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

SCALE:	NONE	BASED ON DRAWING NO	SHEET	REV.
DRAWN:	ENTERGY			
DESIGN:	ENTERGY			
CAD NO:				

DELETED

(REFER TO FIGURE 9-10)

SAR FIGURE NO. 9-17

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PIPING & INSTRUMENT DIAGRAM CIRC. WATER, SERVICE WATER & FIRE WATER INTAKE STRUCTURE EQUIPMENT	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

(REFER TO FIGURE 9-6)

SAR FIGURE NO. 9-18

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

CASE 3, LEVEL 10 – HEAT TRANSFER COEFFICIENT VERSUS TIME	BASED ON DRAWING NO	SHEET	REV.
		1	

DELETED

(REFER TO FIGURE 9-10)

SAR FIGURE NO. 9-19

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PIPING & INSTRUMENT DIAGRAM CIRC. WATER, SERVICE WATER & FIRE WATER INTAKE STRUCTURE EQUIPMENT	BASED ON DRAWING NO	SHEET 1 / REV.

DELETED

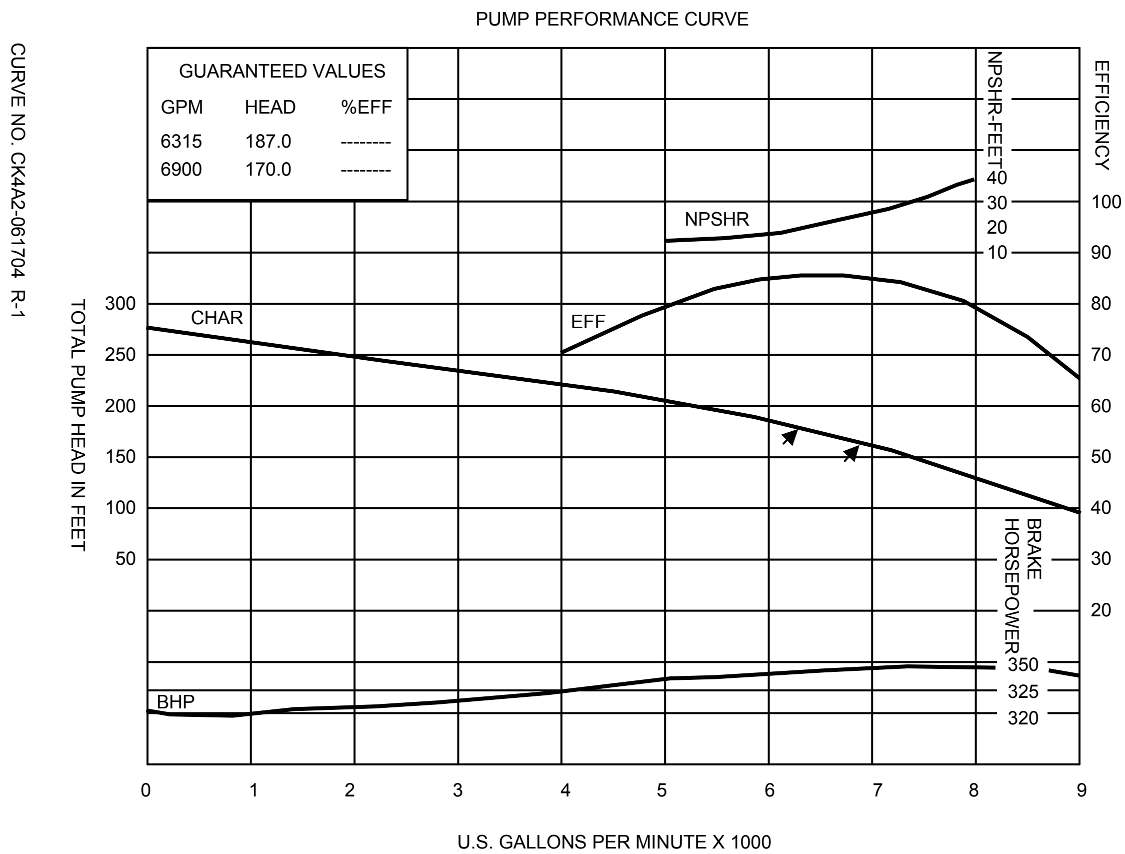
(REFER TO FIGURE 9-6)

SAR FIGURE NO. 9-20

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

	BASED ON DRAWING NO	SHEET	REV.
PIPING & INSTRUMENT DIAGRAM SERVICE WATER		1	



THIS CURVE IS BASED ON ACTUAL TEST PERFORMANCE OF A SIMILAR PUMP. ONLY THE INDICATED POINT(S) IS GUARANTEED.

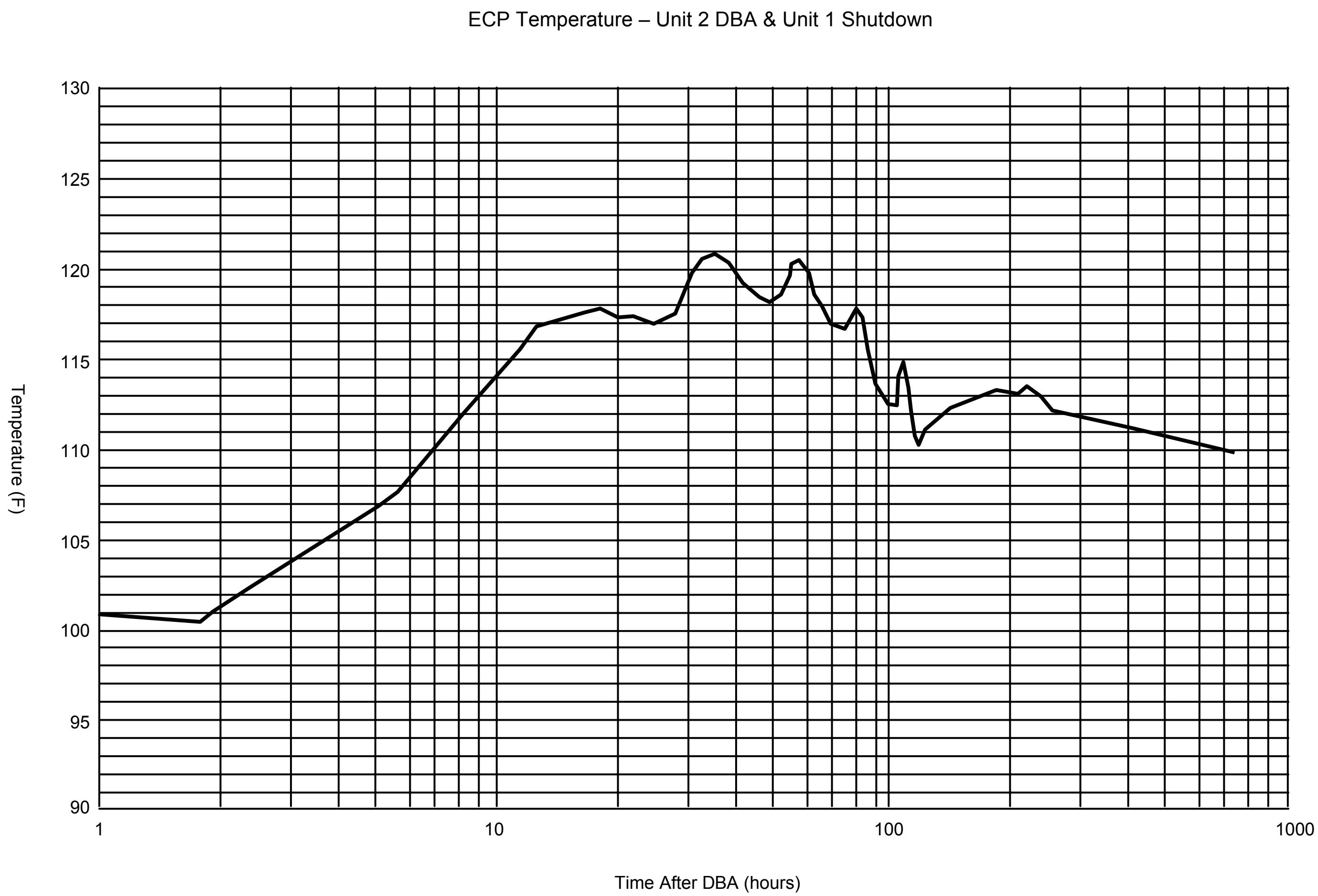
NO. STAGES: TWO
REFERENCE: YJ8
PLOTTED BY: PTR
DATE: 12/16/92

SIZE – MODEL: 24H-7000
IMPELLER DES.: T7MAA99A
IMPELLER DIA.: AS REQUIRED
RPM(S): 1185

SERVICE WATER PUMP CURVES	SAR FIGURE NO. 9-21
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



EMERGENCY COOLING POND TEMPERATURE

SAR FIGURE NO. 9-22

ARKANSAS NUCLEAR ONE

UNIT 1

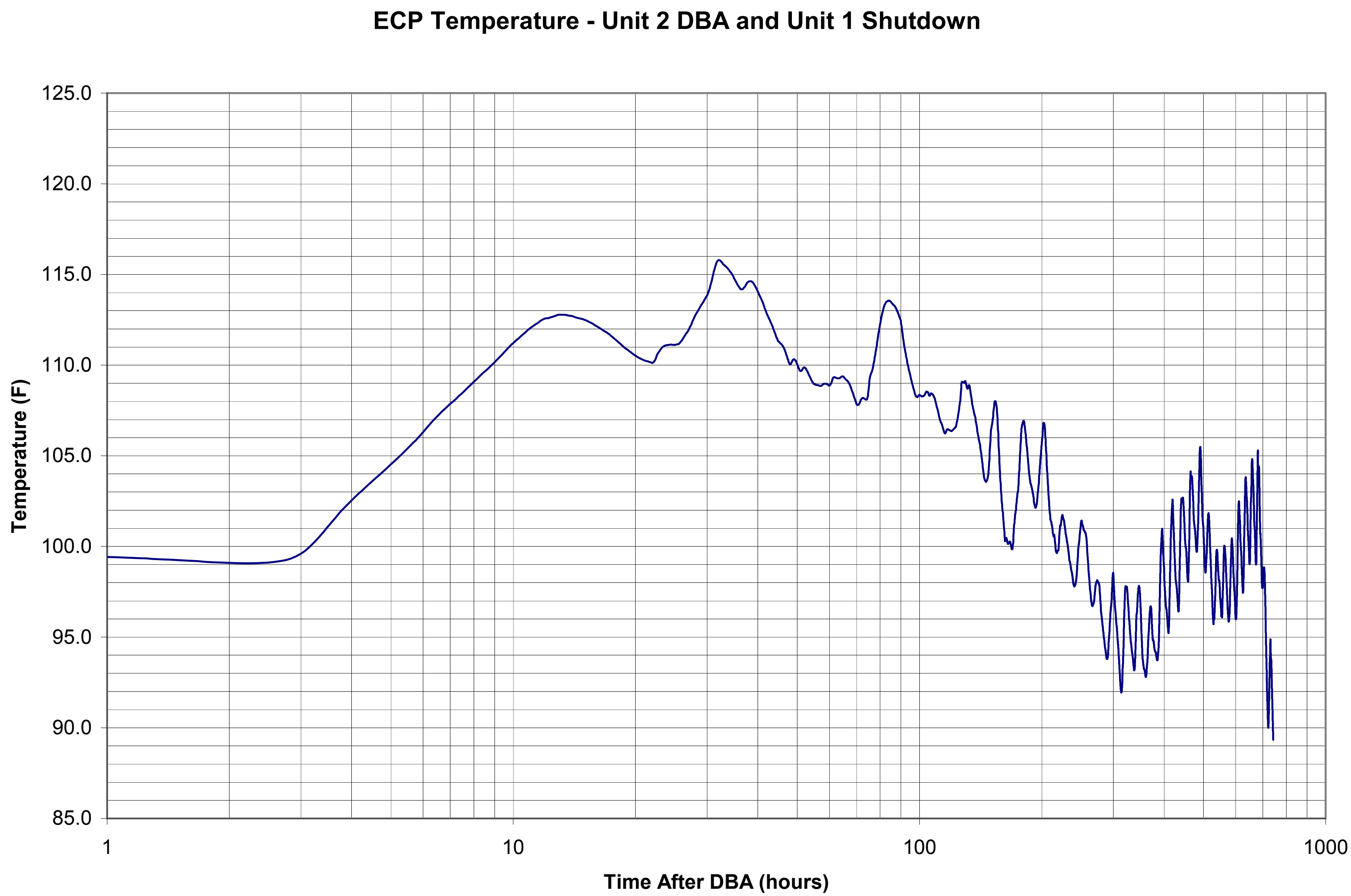
RUSSELLVILLE, ARKANSAS

Entergy

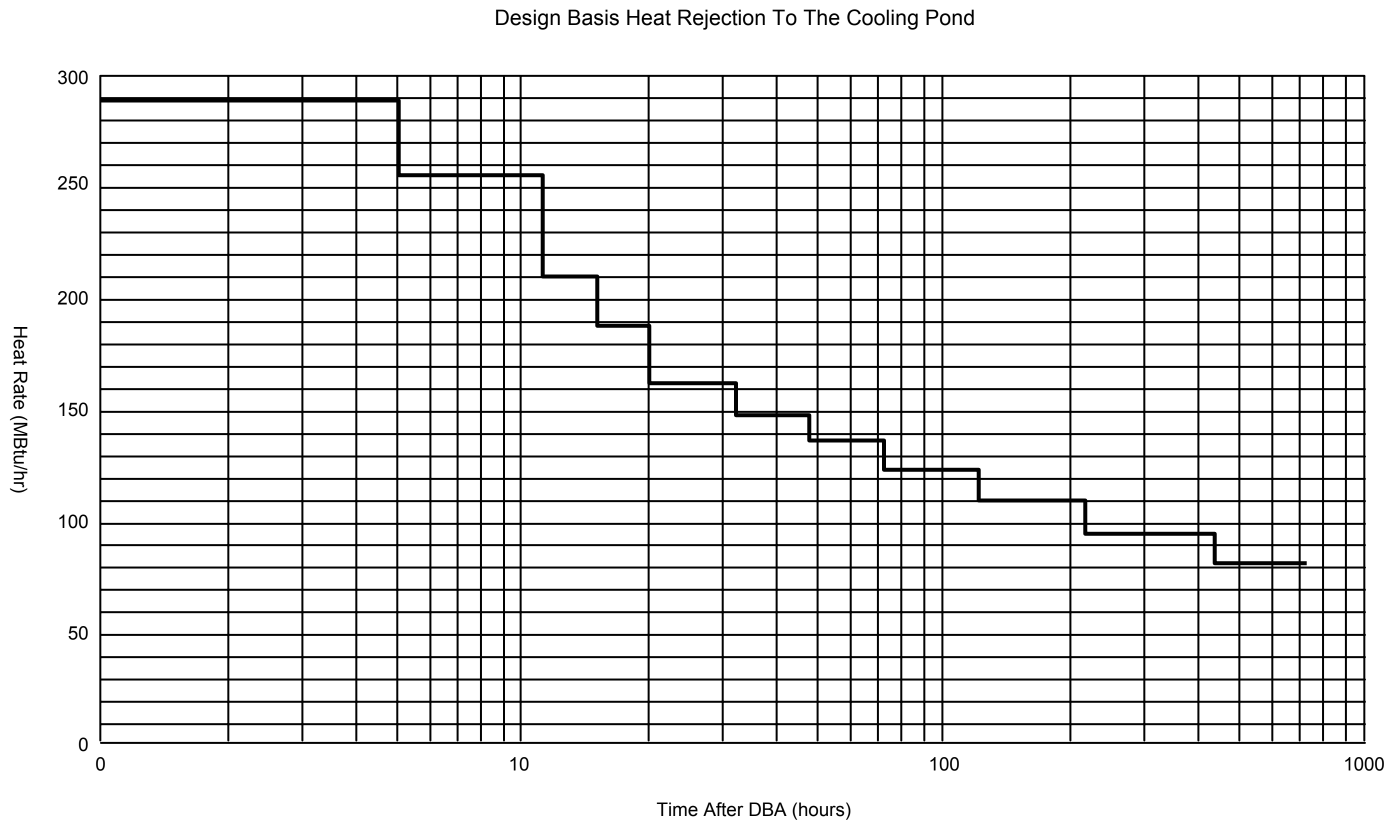
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



EMERGENCY COOLING POND TEMPERATURE – UPDATED ANALYSIS			SAR FIGURE NO. 9-22A
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 24 BASED ON DRAWING NO / SHEET / REV.



DESIGN BASIS HEAT REJECTION TO THE COOLING POND

SAR FIGURE NO. 9-23

AMENDMENT 20

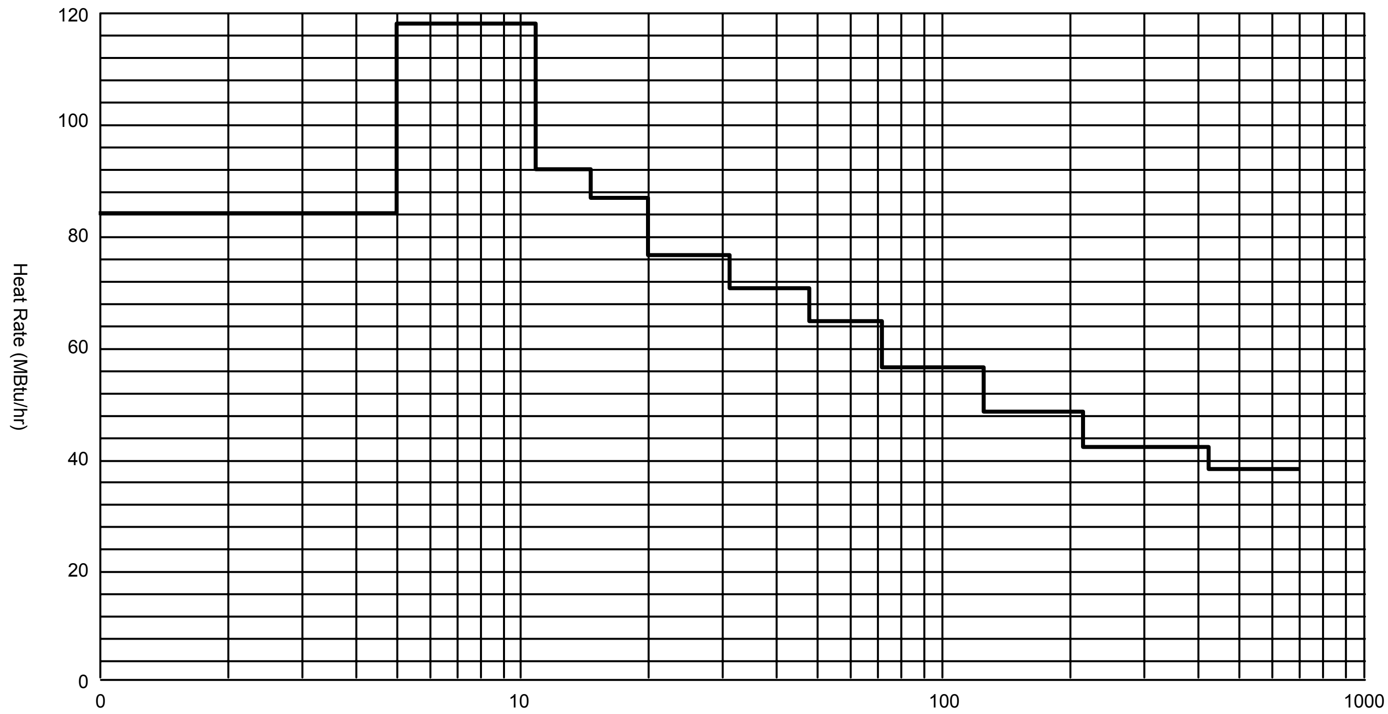
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

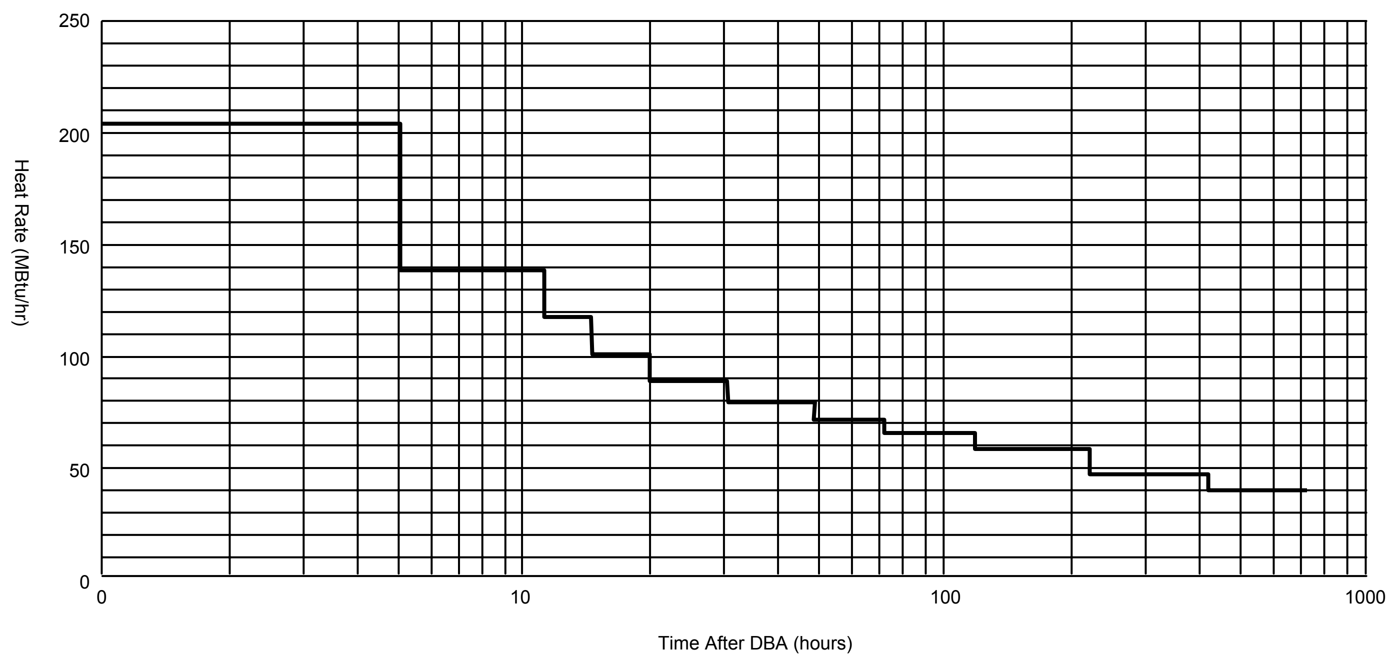
BASED ON DRAWING NO	SHEET	REV.

HEAT LOAD – UNIT 1 SAFE SHUTDOWN



HEAT LOAD UNIT 1 SAFE SHUTDOWN			SAR FIGURE NO. 9-24
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.

HEAT LOAD – UNIT 2 DBA



HEAT LOAD UNIT 2 DBA	SAR FIGURE NO. 9-25
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 9-26

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

HEAT LOAD UNIT 2 DBA	BASED ON DRAWING NO	SHEET 1	REV.

DELETED

SAR FIGURE NO. 9-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

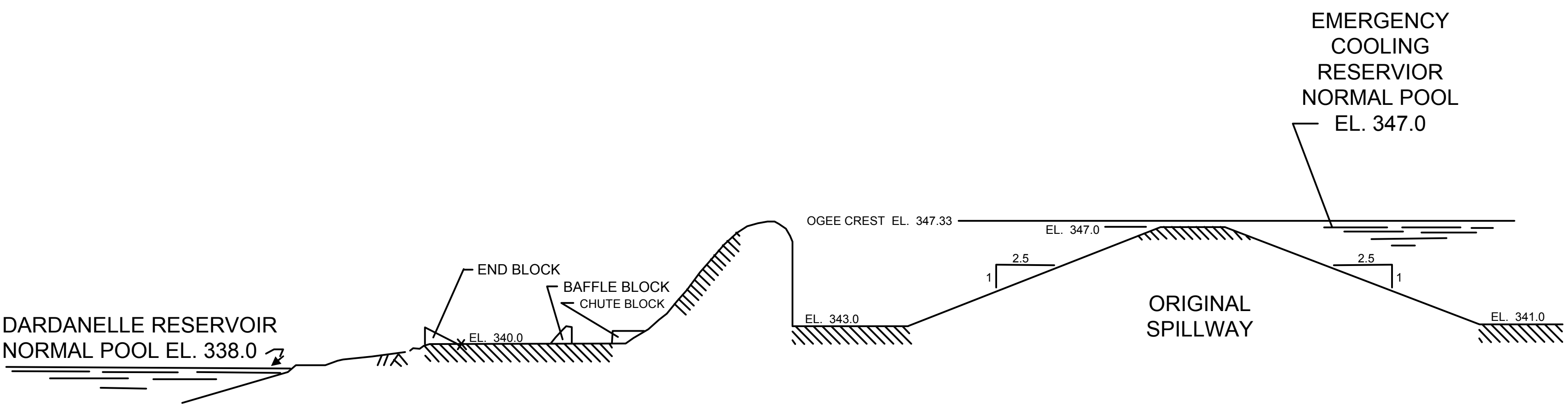
	BASED ON DRAWING NO	SHEET	REV.
HEAT LOAD UNIT 2 ONE LOOP SHUTDOWN		1	

DELETED

SAR FIGURE NO. 9-28

AMENDMENT 20

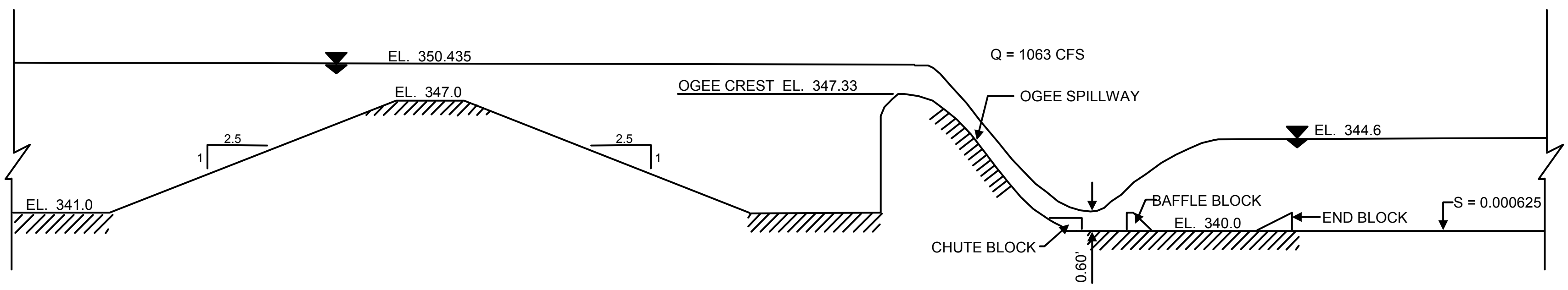
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
HEAT LOAD – UNIT 1 DBA AND UNIT 2 SHUTDOWN	BASED ON DRAWING NO	SHEET 1 REV.



SPILLWAY PROFILE

SPILLWAY PROFILE		SAR FIGURE NO. 9-29
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 23

BASED ON DRAWING NO	SHEET	REV.



WATER SURFACE PROFILE THROUGH STILLING BASIN

EMERGENCY COOLING POND STILLING BASIN

SAR FIGURE NO. 9-30

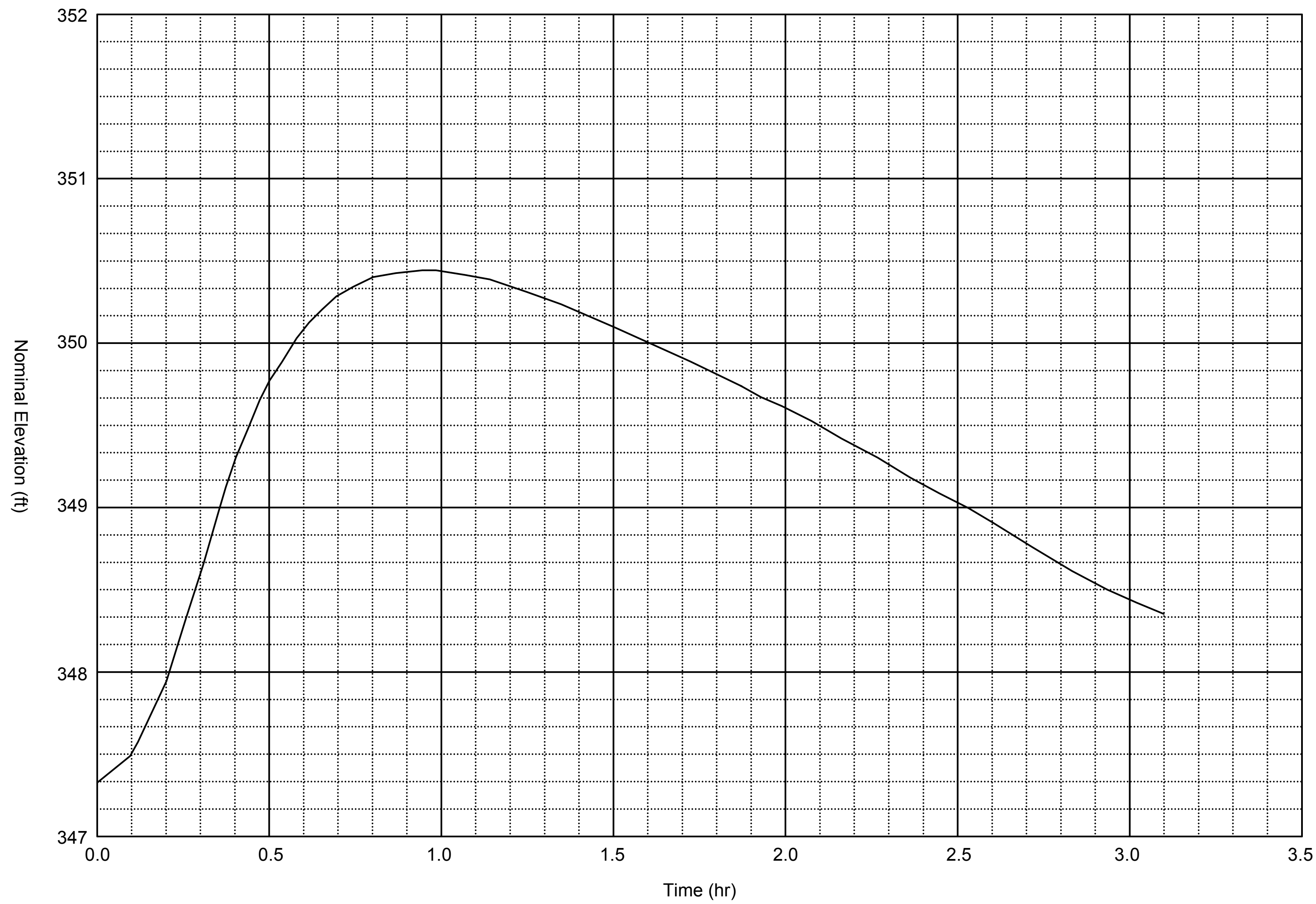
AMENDMENT 23

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



POND LEVEL HYDROGRAPH FOR DESIGN STORM FOR NEW R/C OGEE SPILLWAY

SAR FIGURE NO. 9-31

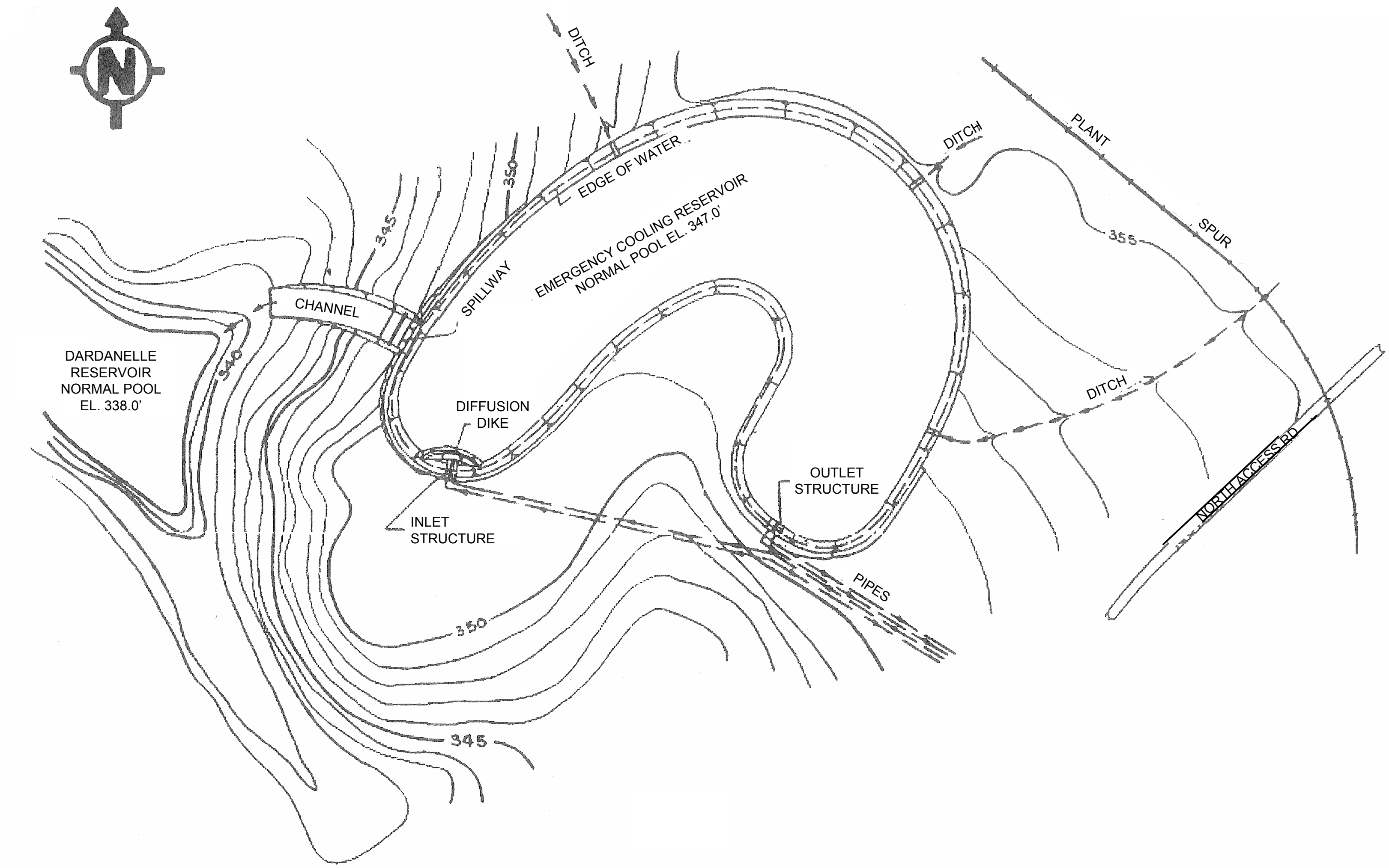
AMENDMENT 23

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



TOPOGRAPHIC MAP

SAR FIGURE NO. 9-32

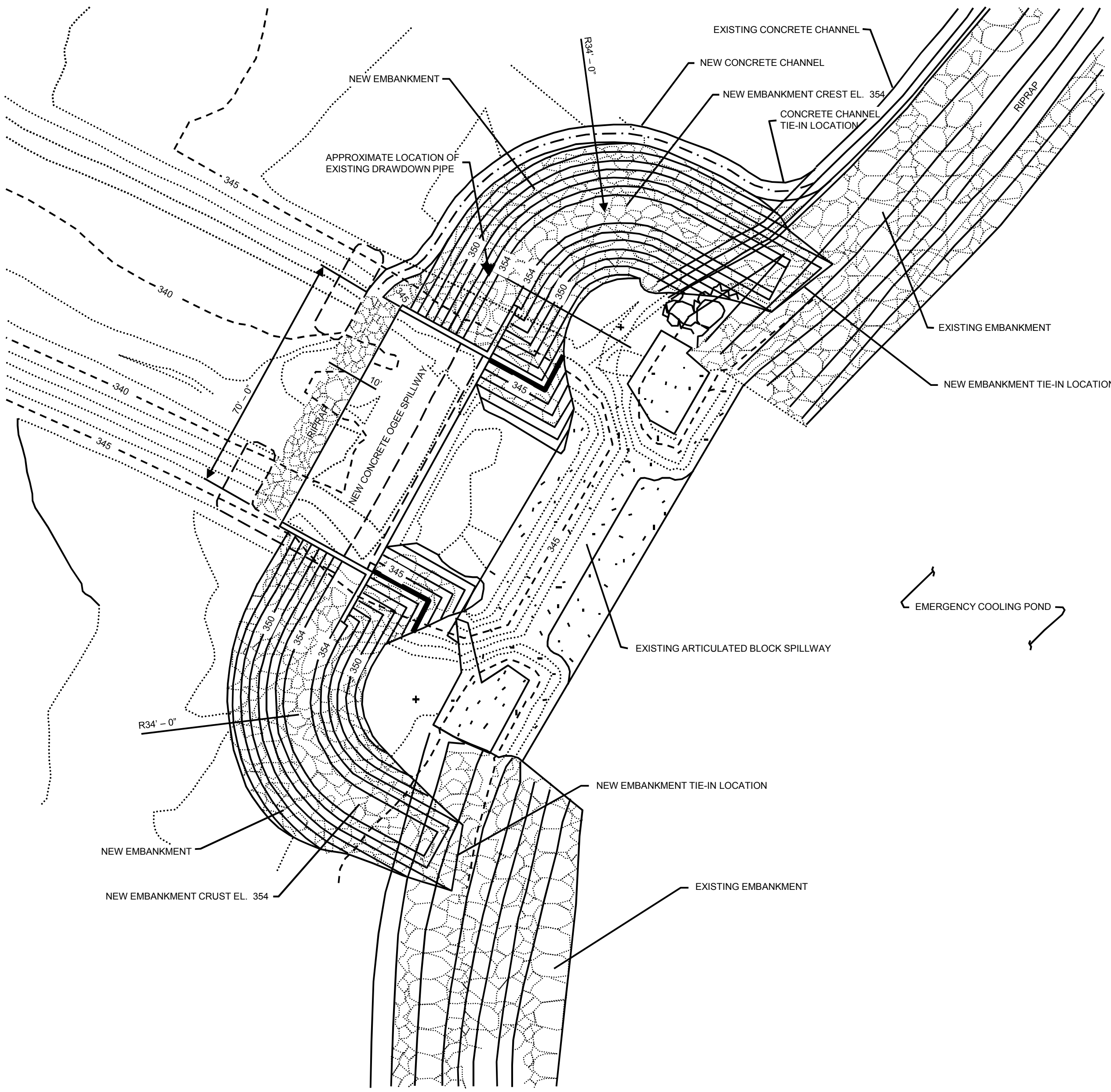
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

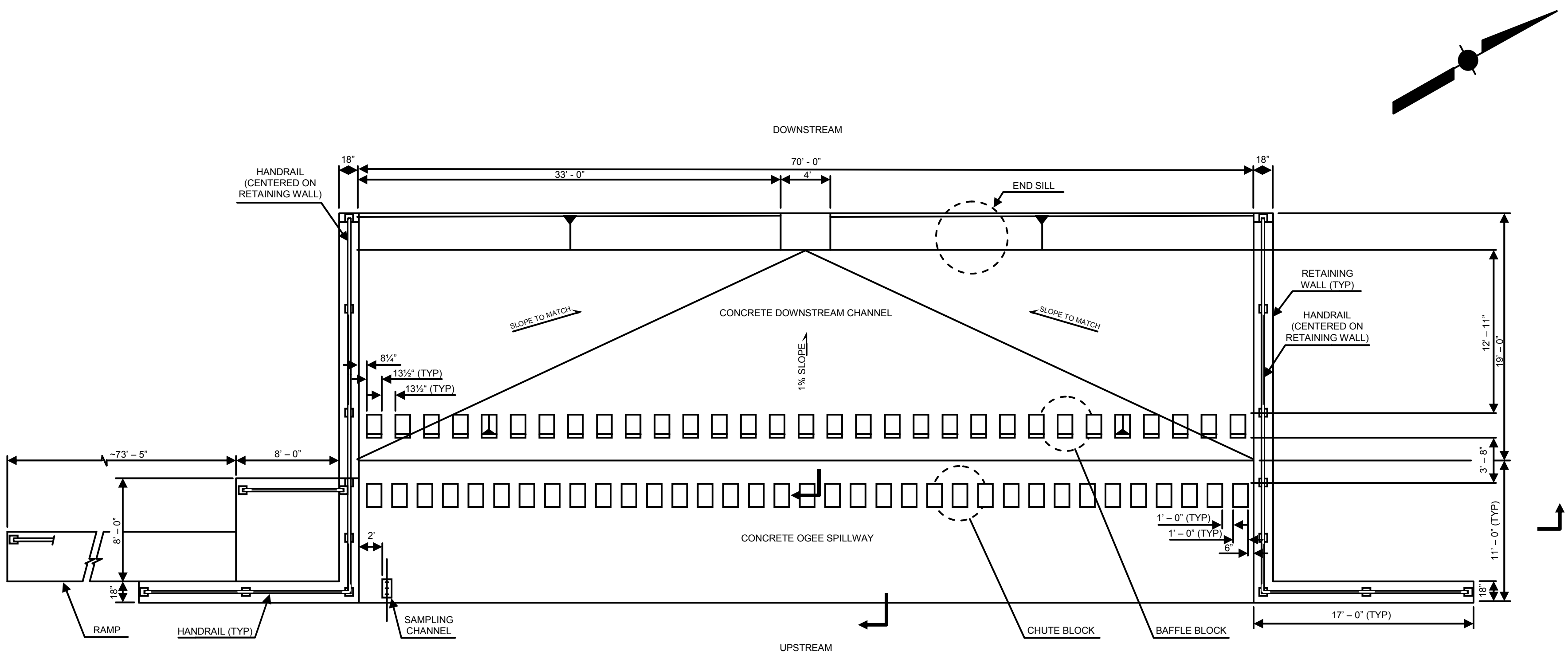
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 9-33A

AMENDMENT 27

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
EMERGENCY POND SPILLWAY	BASED ON DRAWING NO	SHEET	REV.
		1	



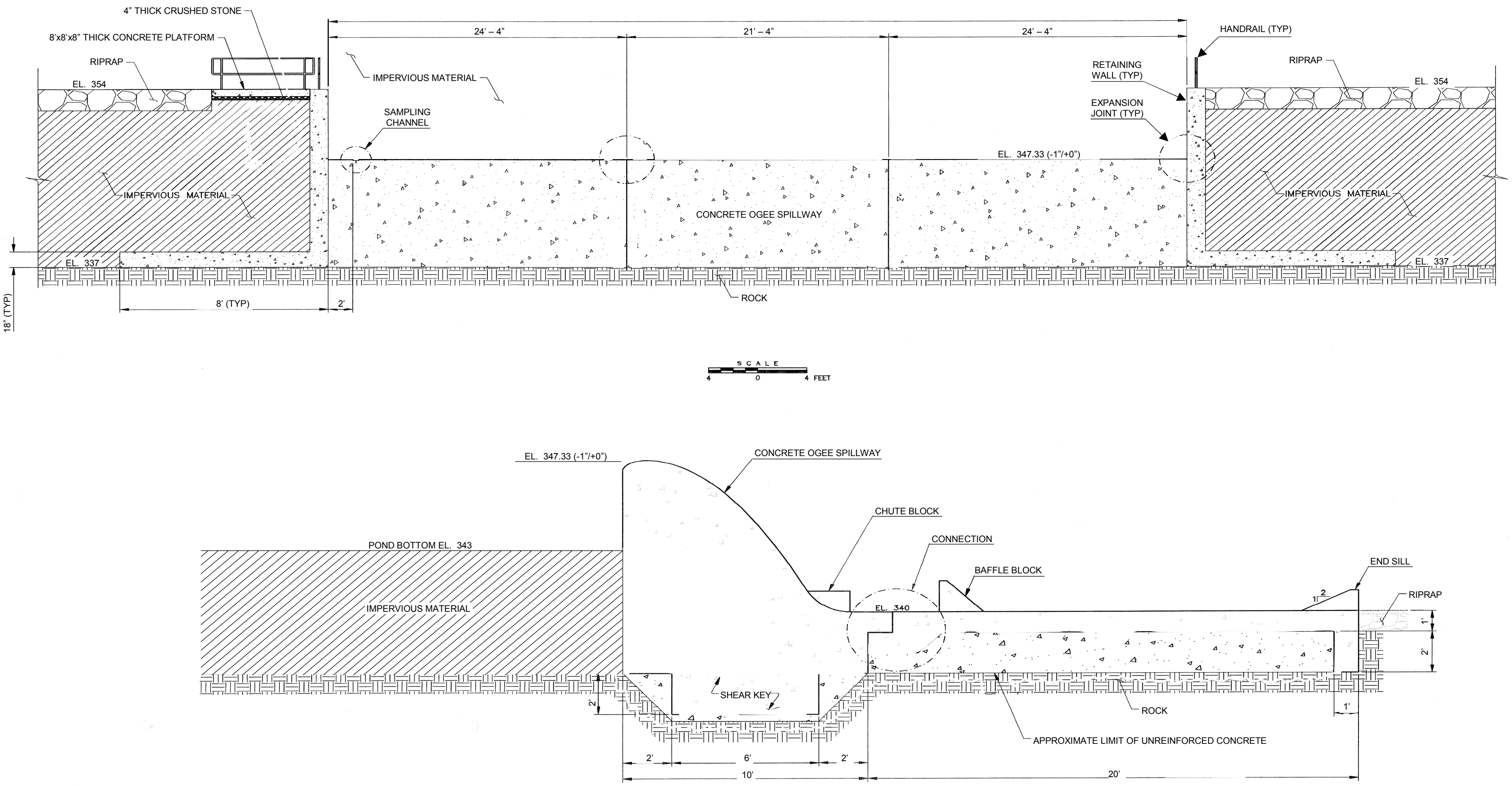
EMERGENCY POND SPILLWAY	SAR FIGURE NO. 9-33B
	AMENDMENT 23

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



EMERGENCY POND SPILLWAY

SAR FIGURE NO. 9-33C

ARKANSAS NUCLEAR ONE

UNIT 1

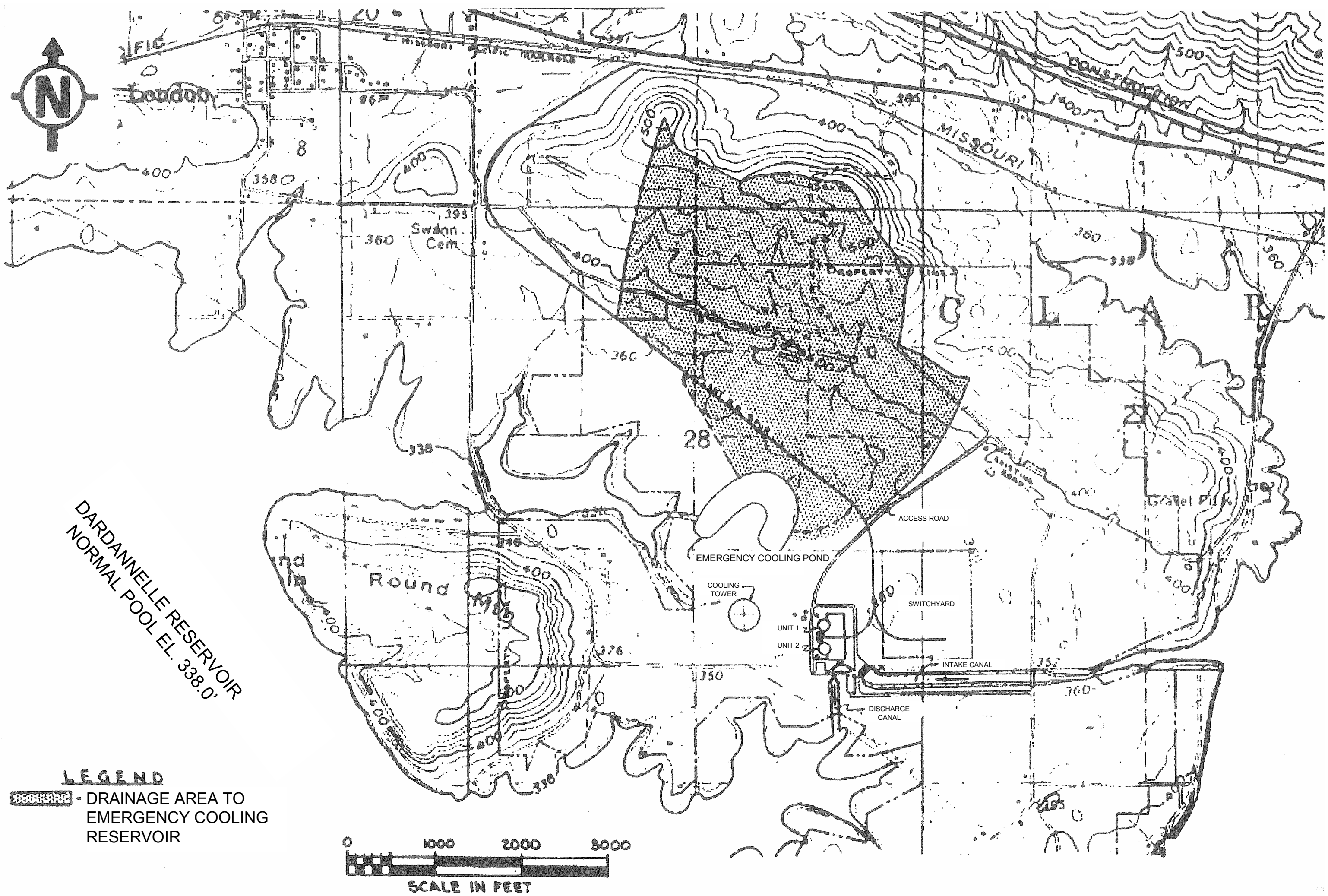
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 23

BASED ON DRAWING NO	SHEET	REV.



COOLING POND SURRONDING AREA TOPOGRAPHICAL MAP

SAR FIGURE NO. 9-35

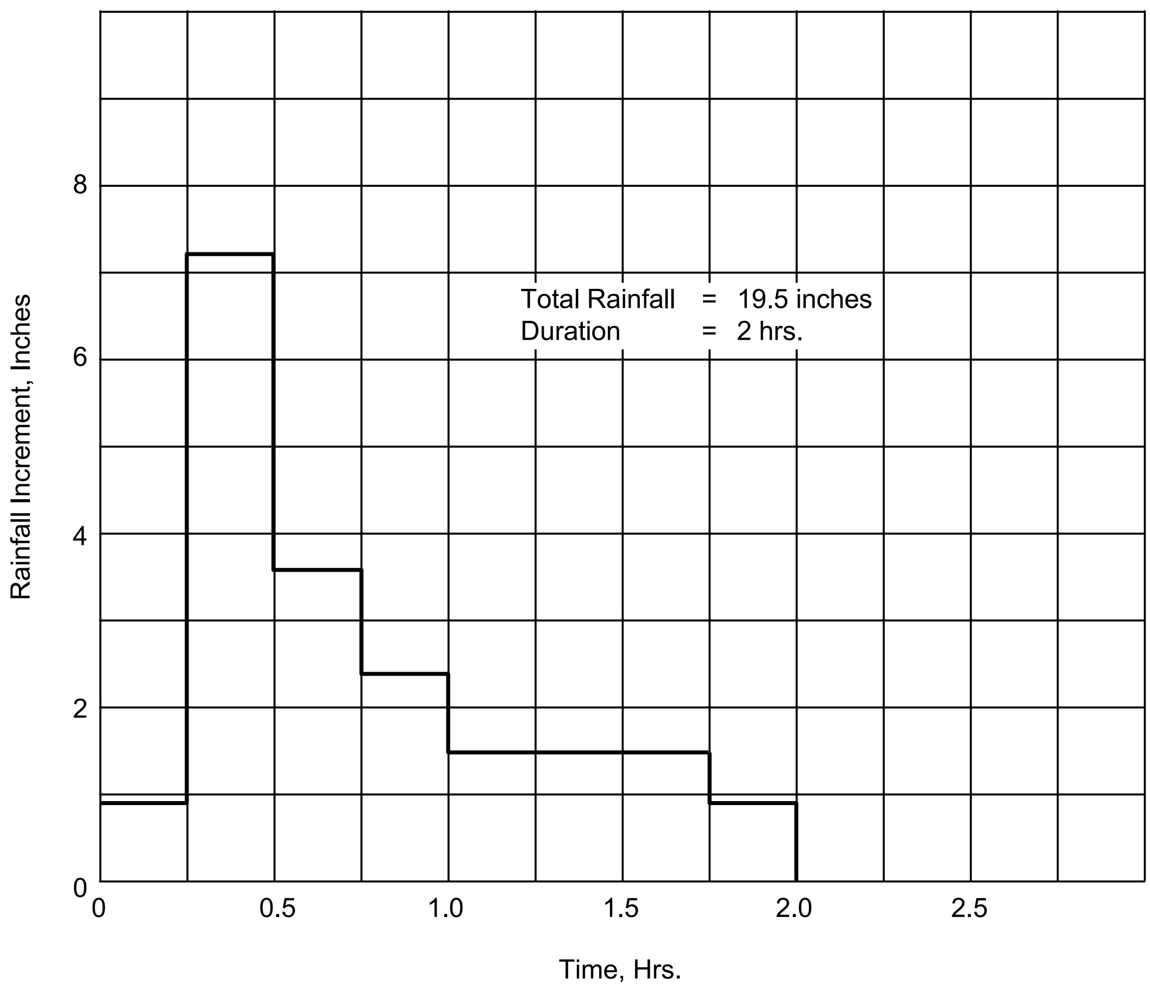
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

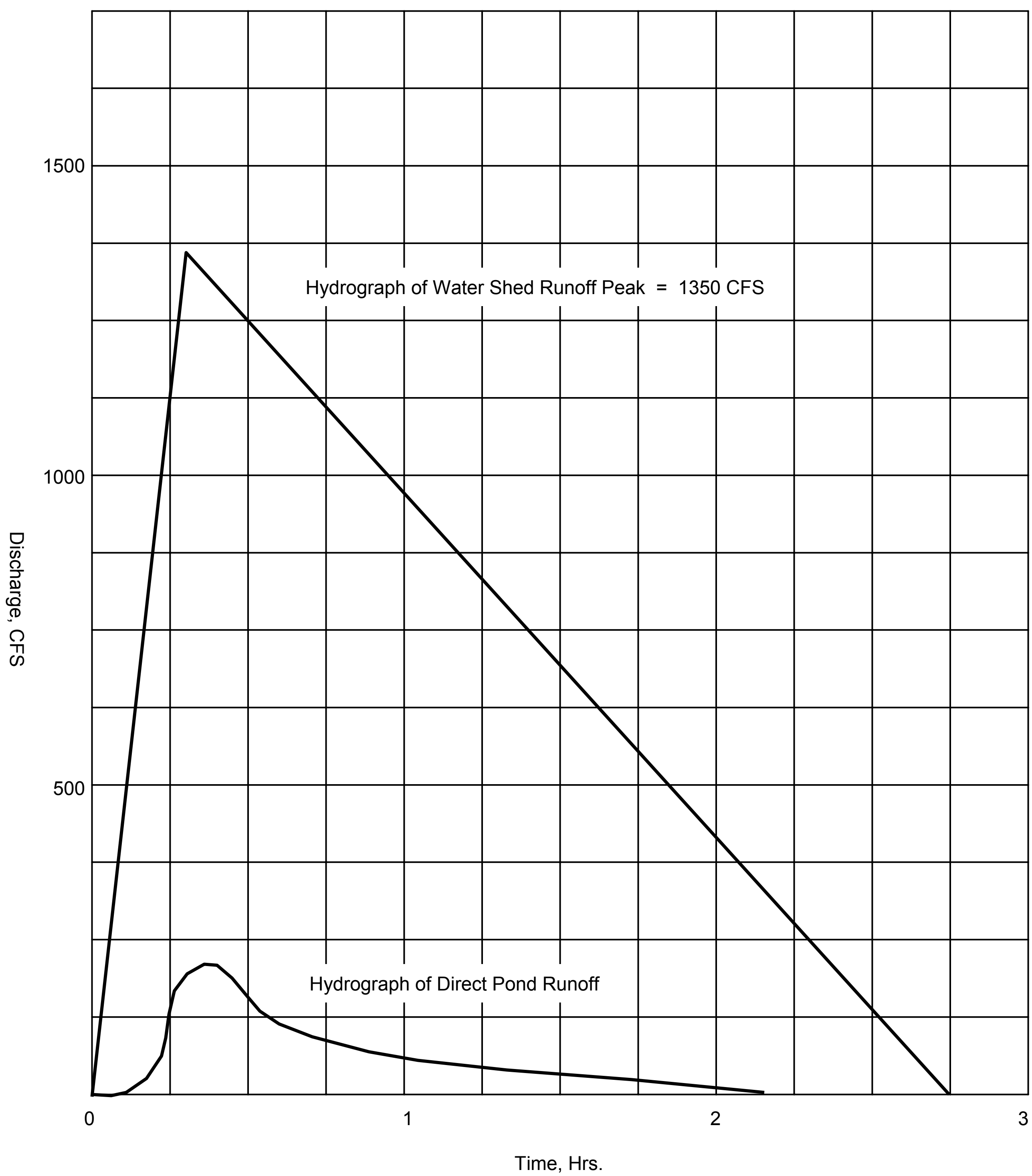
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



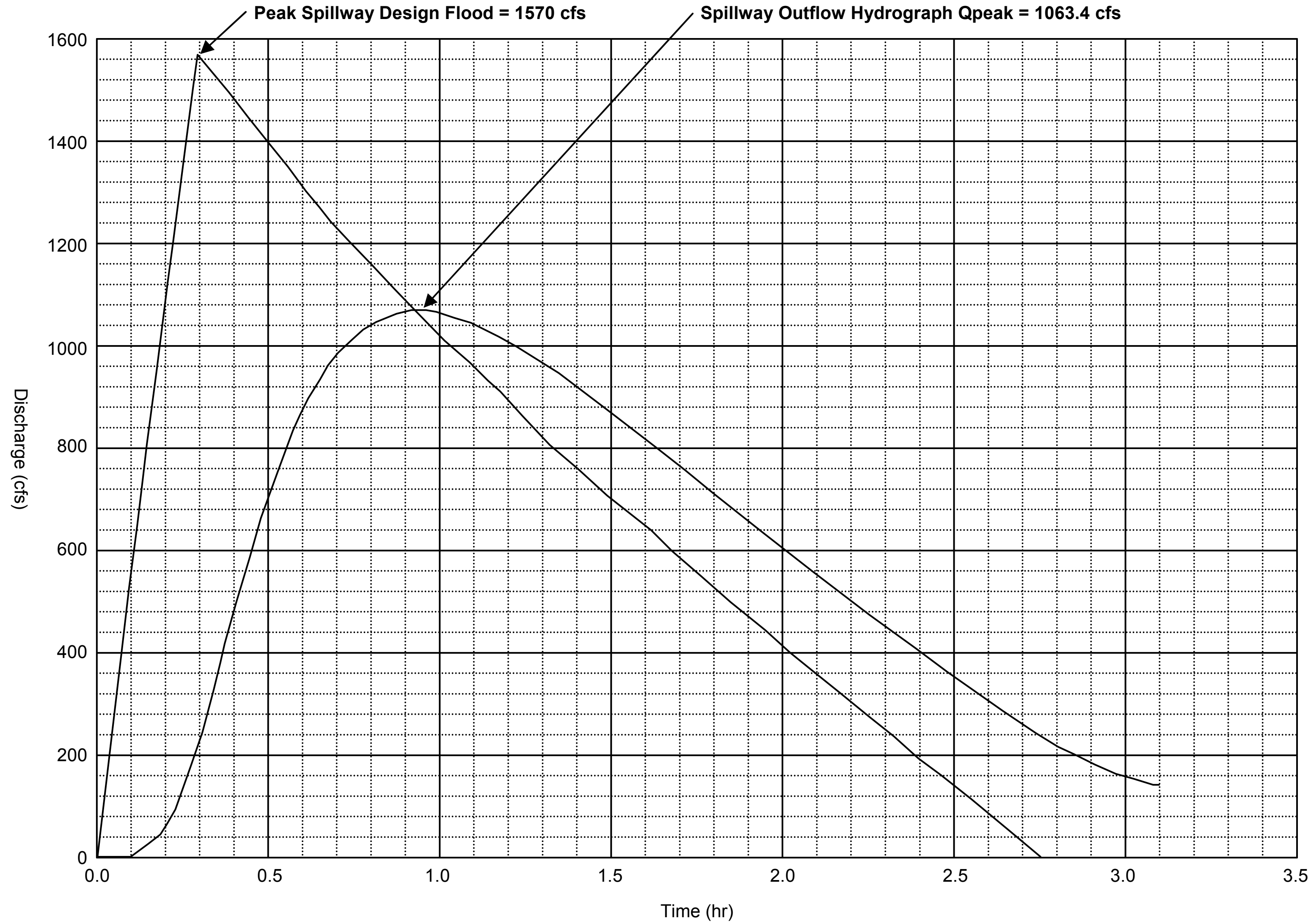
HISTOGRAM OF PROBABLE MAXIMUM PRECIPITATION		SAR FIGURE NO. 9-36
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	AMENDMENT 20
SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:		BASED ON DRAWING NO / SHEET / REV.



SAR FIGURE NO. 9-37

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ONE HALF PROBABLE MAXIMUM FLOOD	BASED ON DRAWING NO	SHEET 1 / REV.



FLOOD ROUTING HYDROGRAPHS

SAR FIGURE NO. 9-38

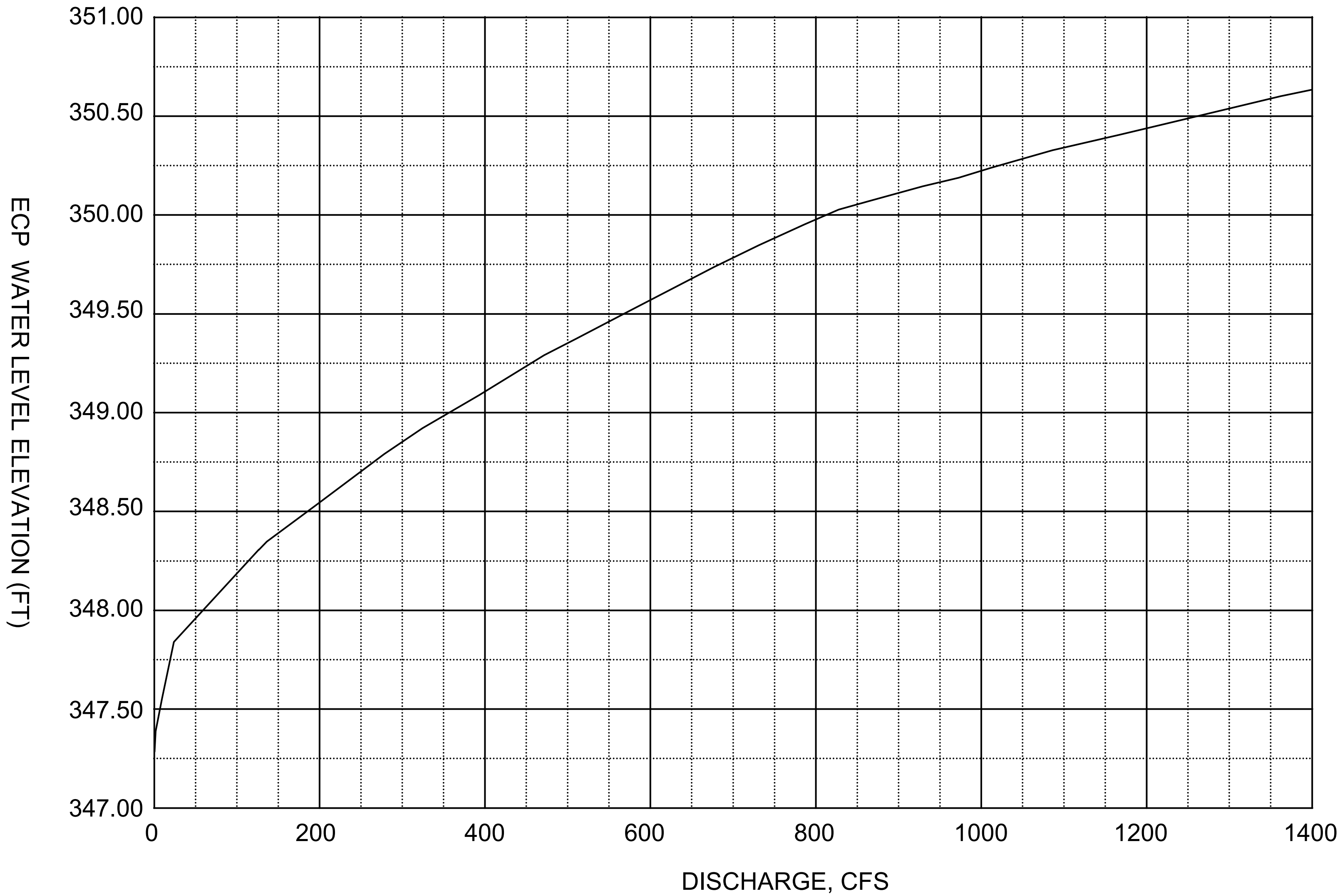
AMENDMENT 23

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

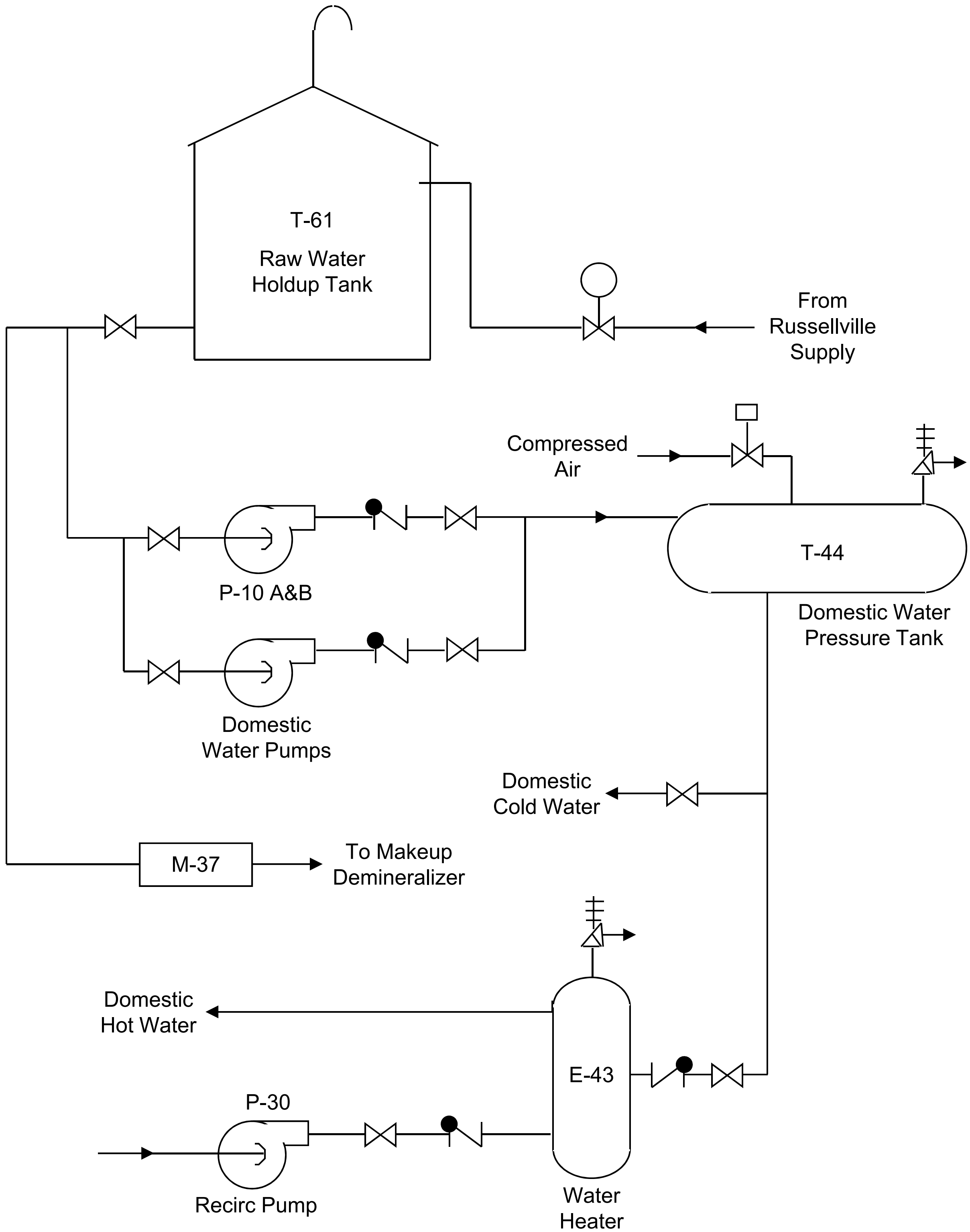
Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



ECP OGEE SPILLWAY RATING CURVE			SAR FIGURE NO. 9-39
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 23 BASED ON DRAWING NO / SHEET / REV.



SAR FIGURE NO. 9-40

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
DOMESTIC WATER SYSTEM	BASED ON DRAWING NO	SHEET 1	REV.

FIGURES

9-41

THROUGH

9-44

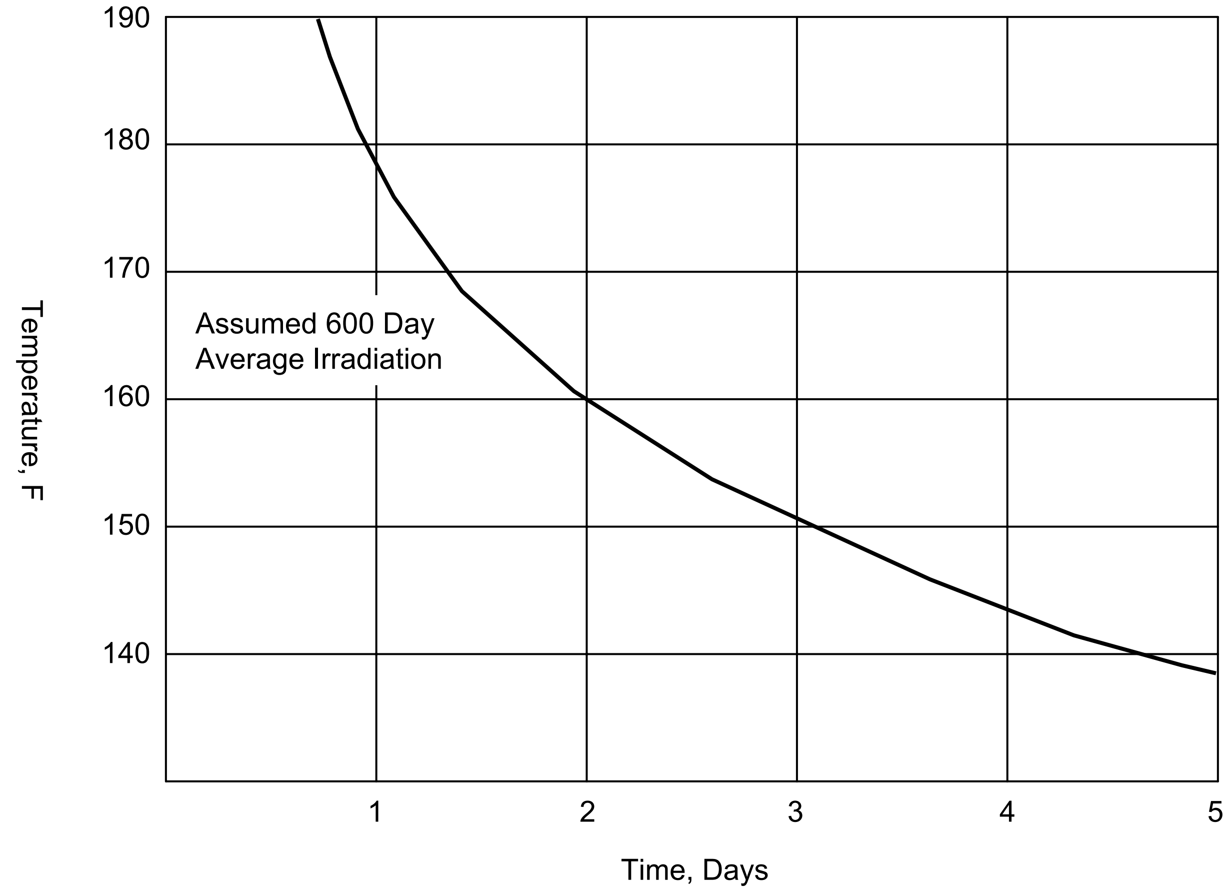
DELETED

SAR FIGURE NO. 9-41

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

	BASED ON DRAWING NO	SHEET	REV.
		1	



ASSUMPTIONS:

1. Constant UA
2. Constant Balanced Flow (3000 GPM)
3. Service Water Inlet Temperature
4. Only One Pump & Cooler Available

SAR FIGURE NO. 9-45

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR COOLANT AND FUEL TRANSFER CANAL TEMPERATURE VS TIME AFTER SHUTDOWN	BASED ON DRAWING NO	SHEET 1 REV.

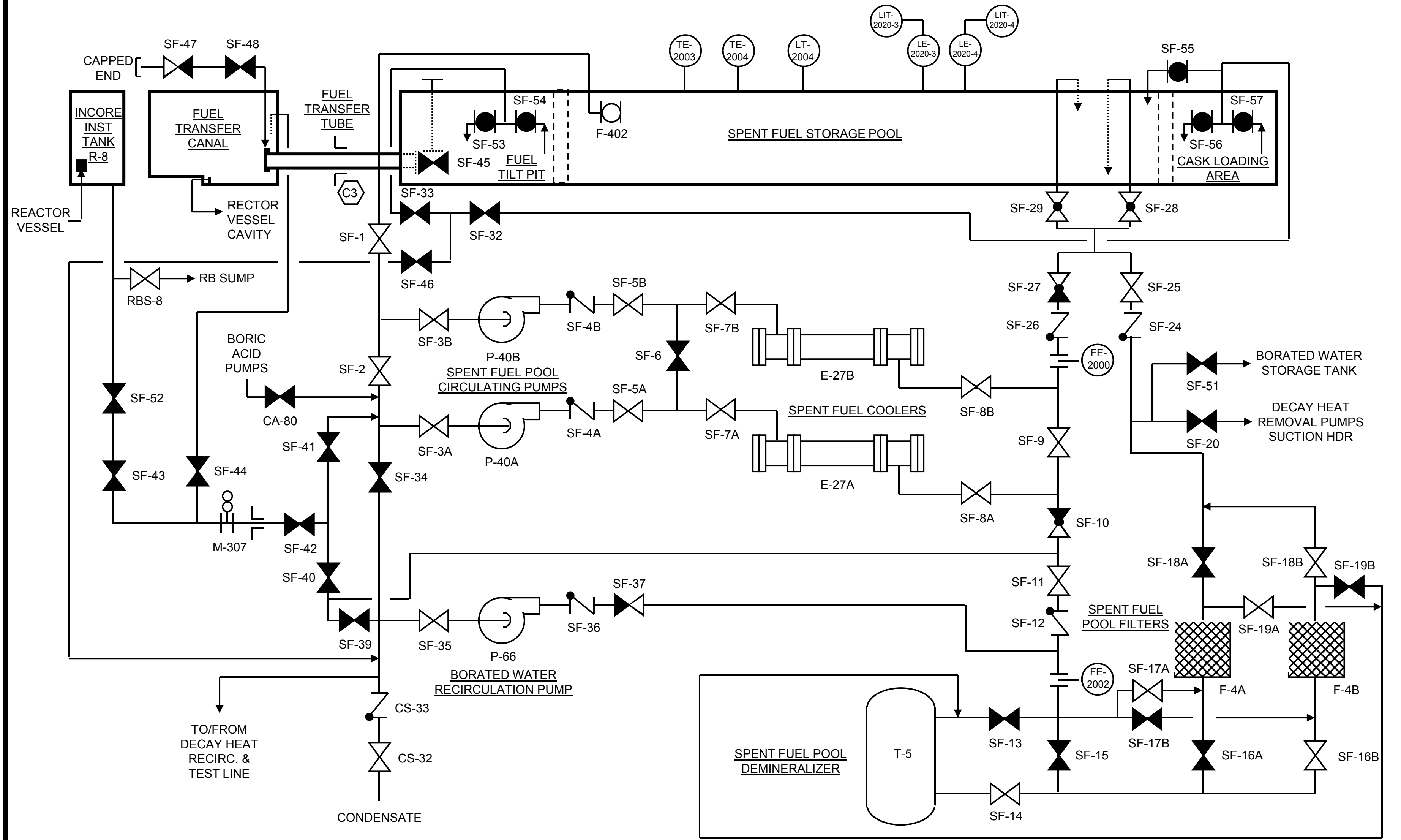
DELETED

SAR FIGURE NO. 9-46

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MAKEUP FROM BORATED WATER STORAGE TANK	BASED ON DRAWING NO	SHEET 1

REV.



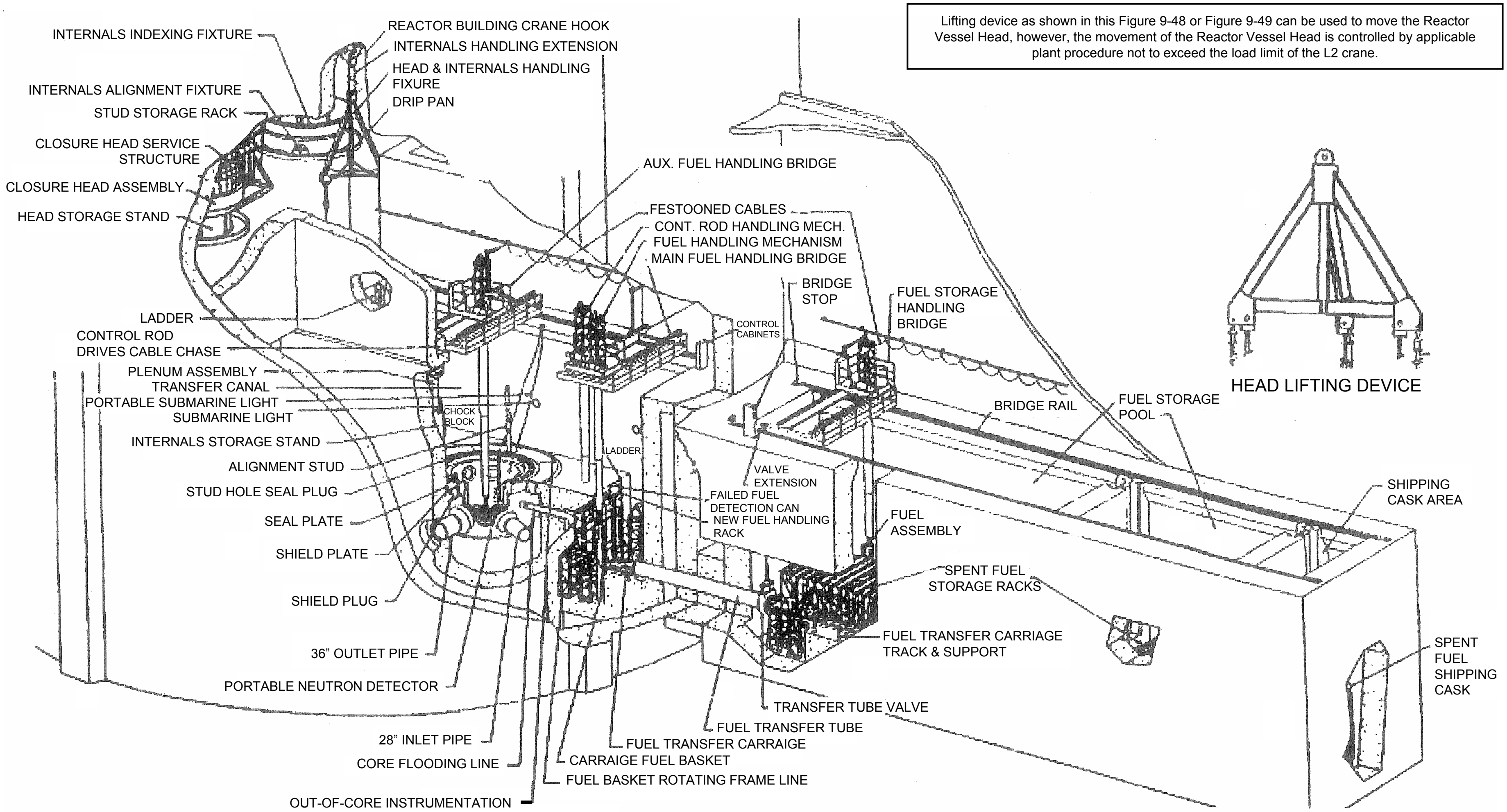
SPENT FUEL COOLING SYSTEM	SAR FIGURE NO. 9-47
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 26

Entergy

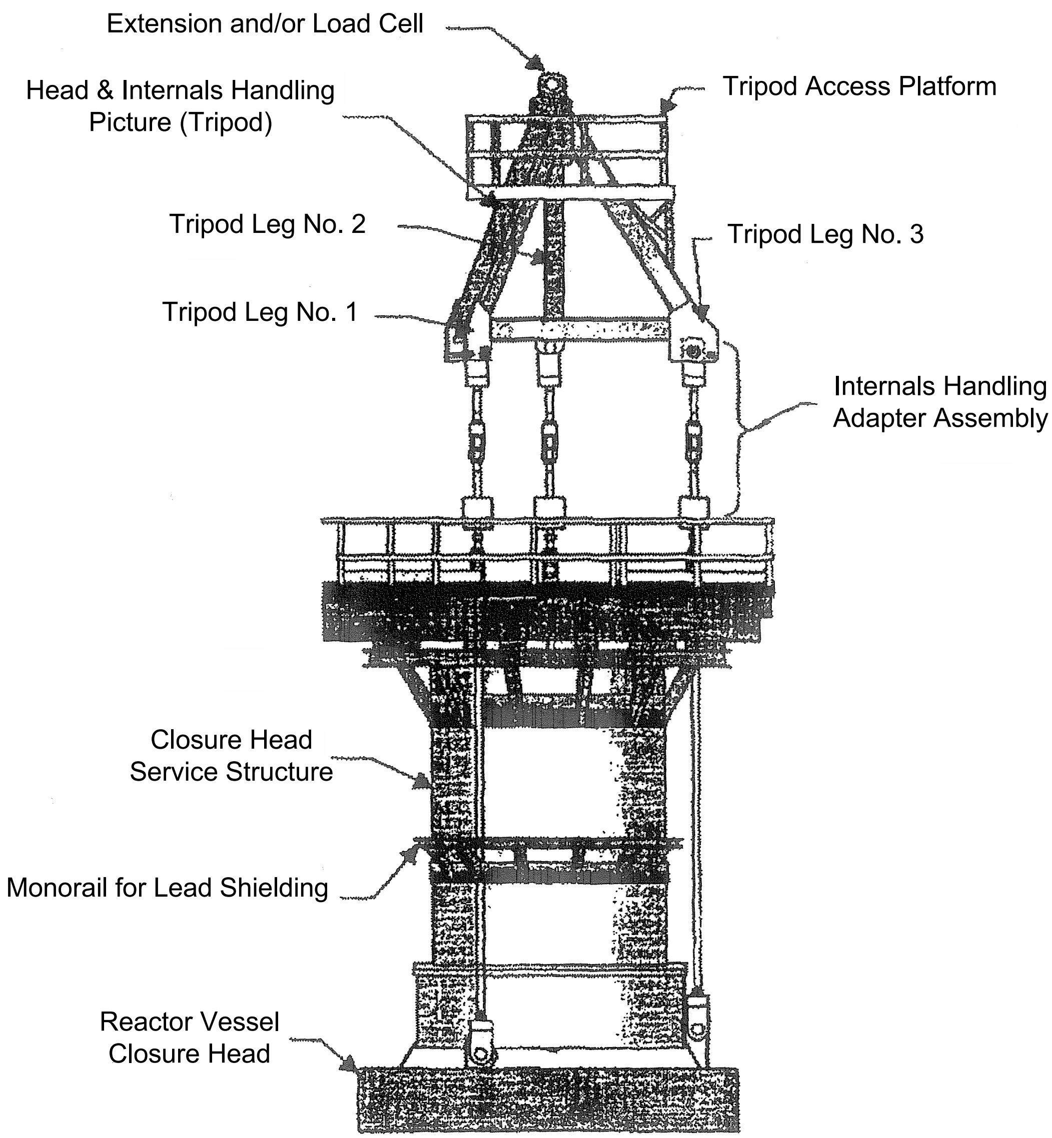
SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

Lifting device as shown in this Figure 9-48 or Figure 9-49 can be used to move the Reactor Vessel Head, however, the movement of the Reactor Vessel Head is controlled by applicable plant procedure not to exceed the load limit of the L2 crane.



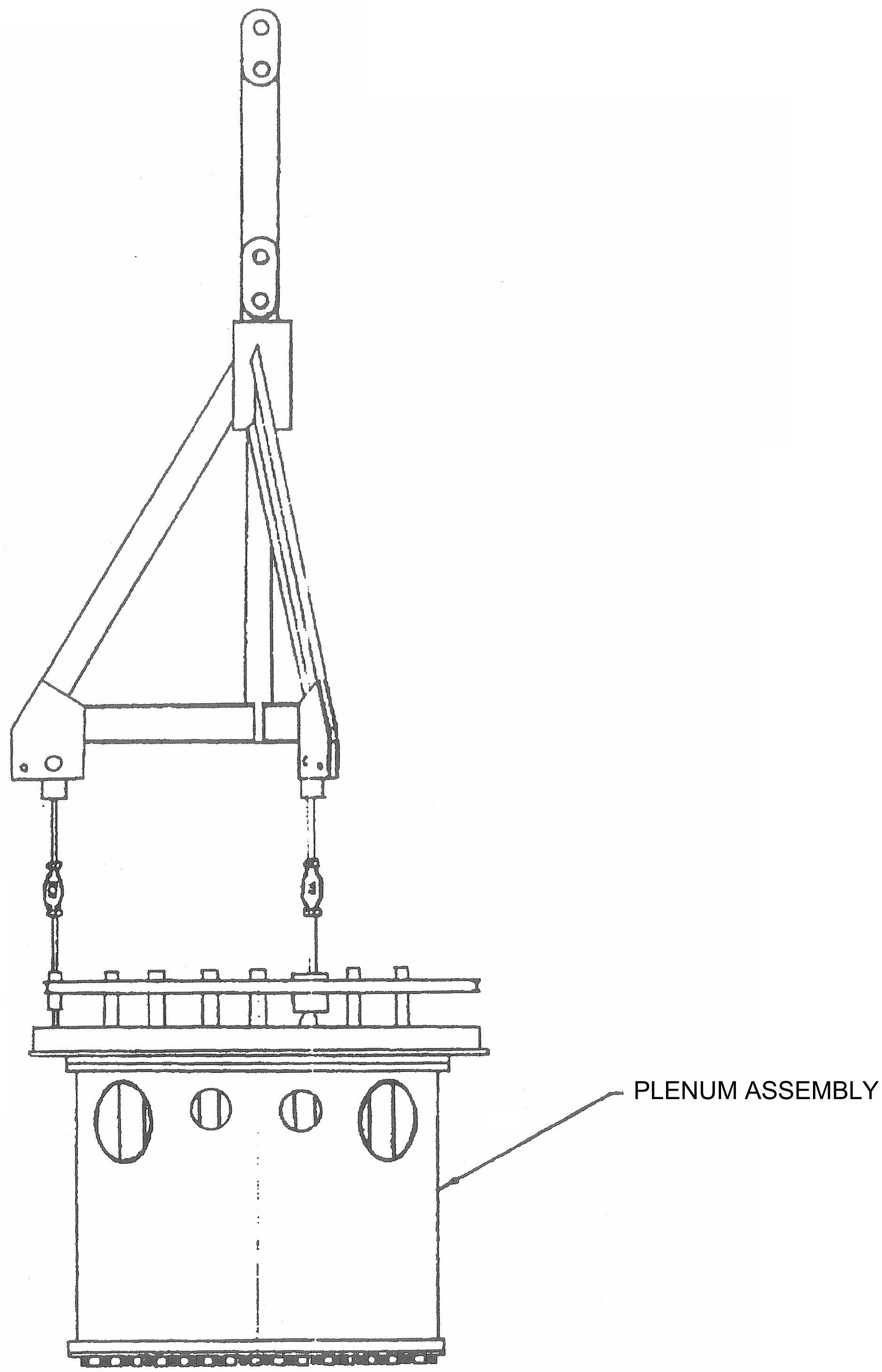
TYPICAL REFUELING SYSTEM		SAR FIGURE NO. 9-48
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	AMENDMENT 21
	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	BASED ON DRAWING NO / SHEET / REV.



SAR FIGURE NO. 9-49

AMENDMENT 21

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
CLOSURE HEAD REMOVAL	BASED ON DRAWING NO		SHEET 1	REV.



SAR FIGURE NO. 9-50

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
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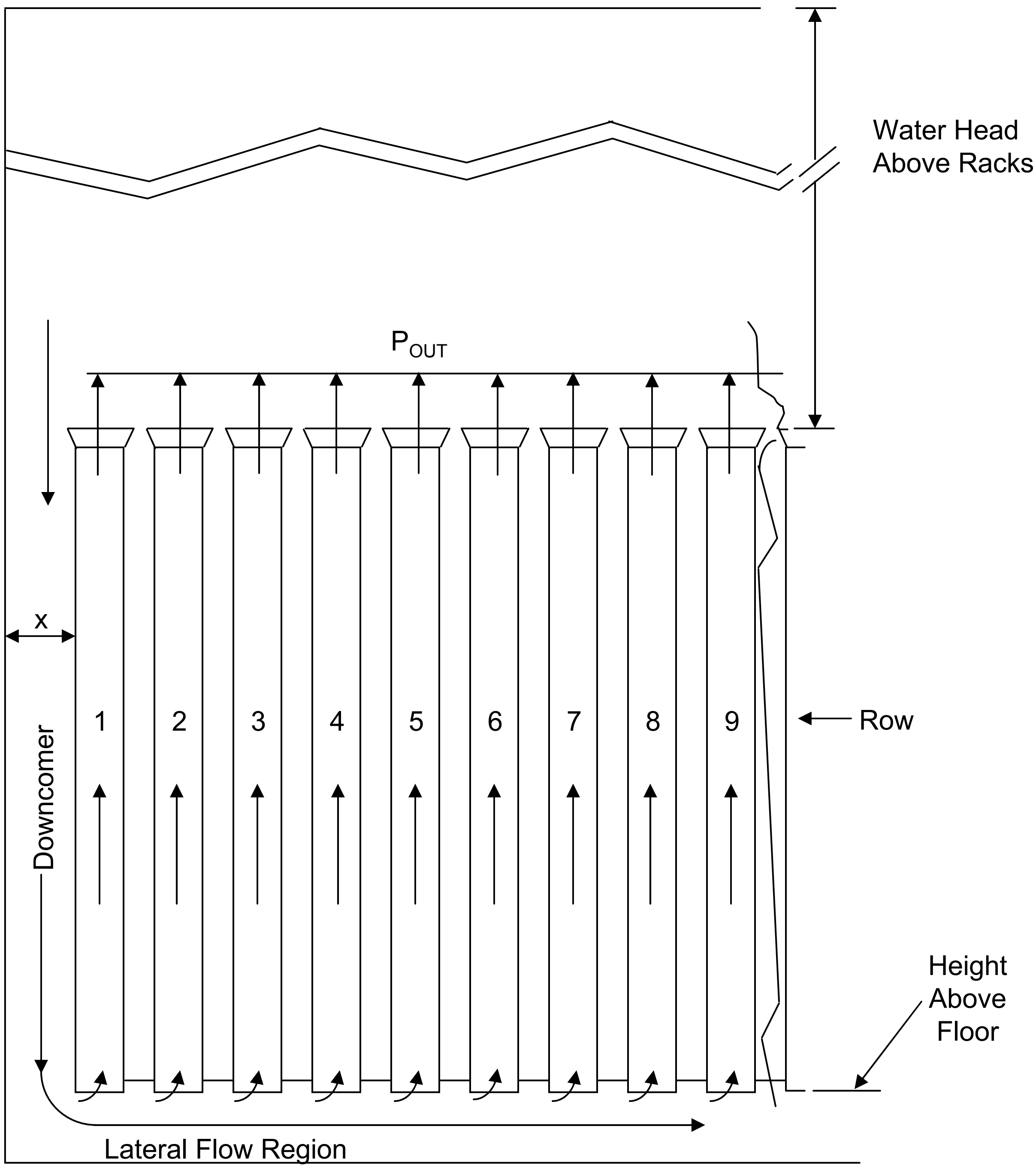


SAR FIGURE NO. 9-51

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

SPENT FUEL POOL WATER SURFACE PATH

BASED ON DRAWING NO	SHEET	REV.
	1	

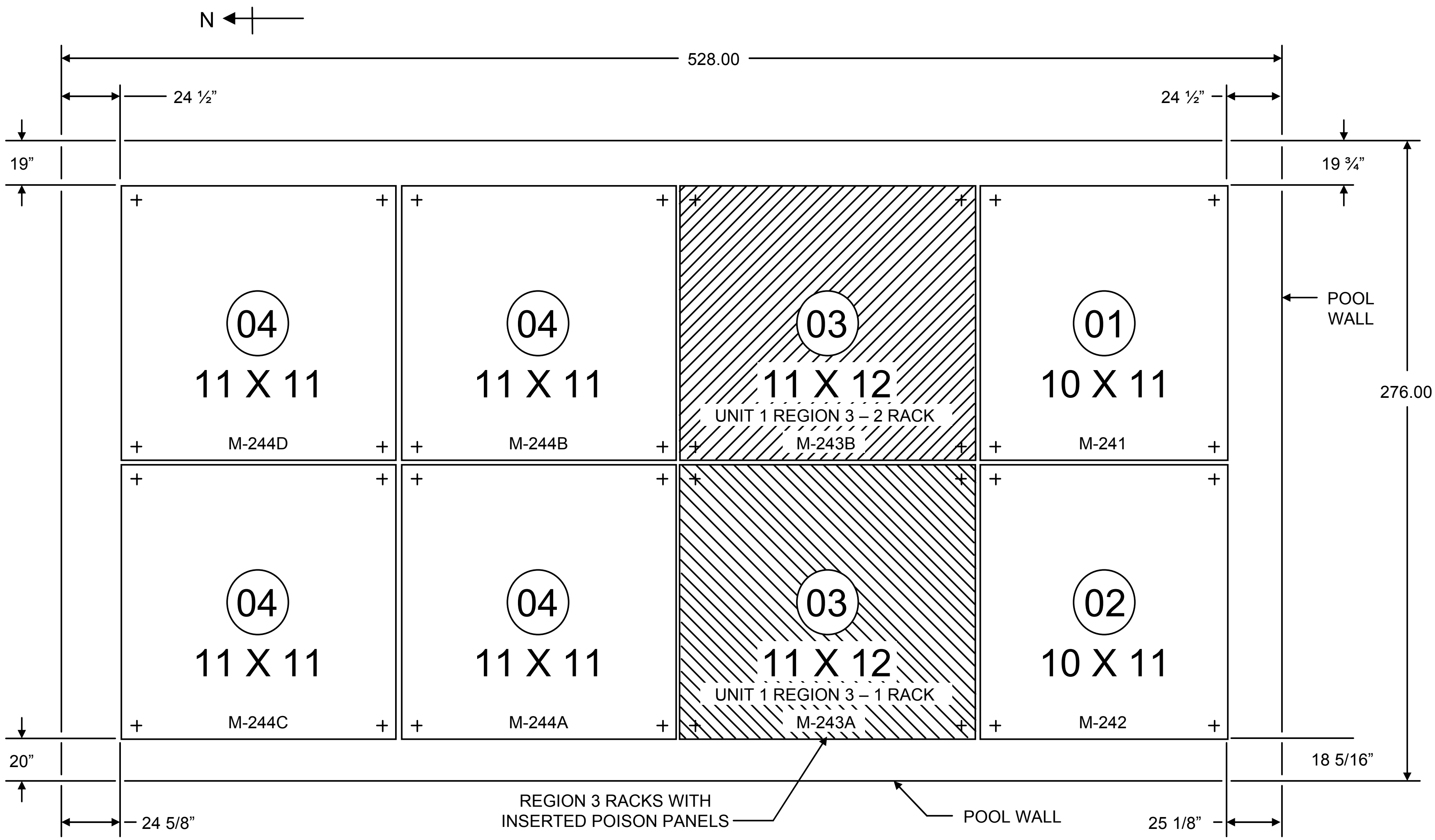
DELETED

SAR FIGURE NO. 9-52

AMENDMENT 21

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

SPENT FUEL POOL WATER SURFACE PATH (PLAN VIEW)	BASED ON DRAWING NO	SHEET 1	REV.



UNIT 1 SPENT FUEL POOL RACKS ARRANGEMENT

SAR FIGURE NO. 9-53

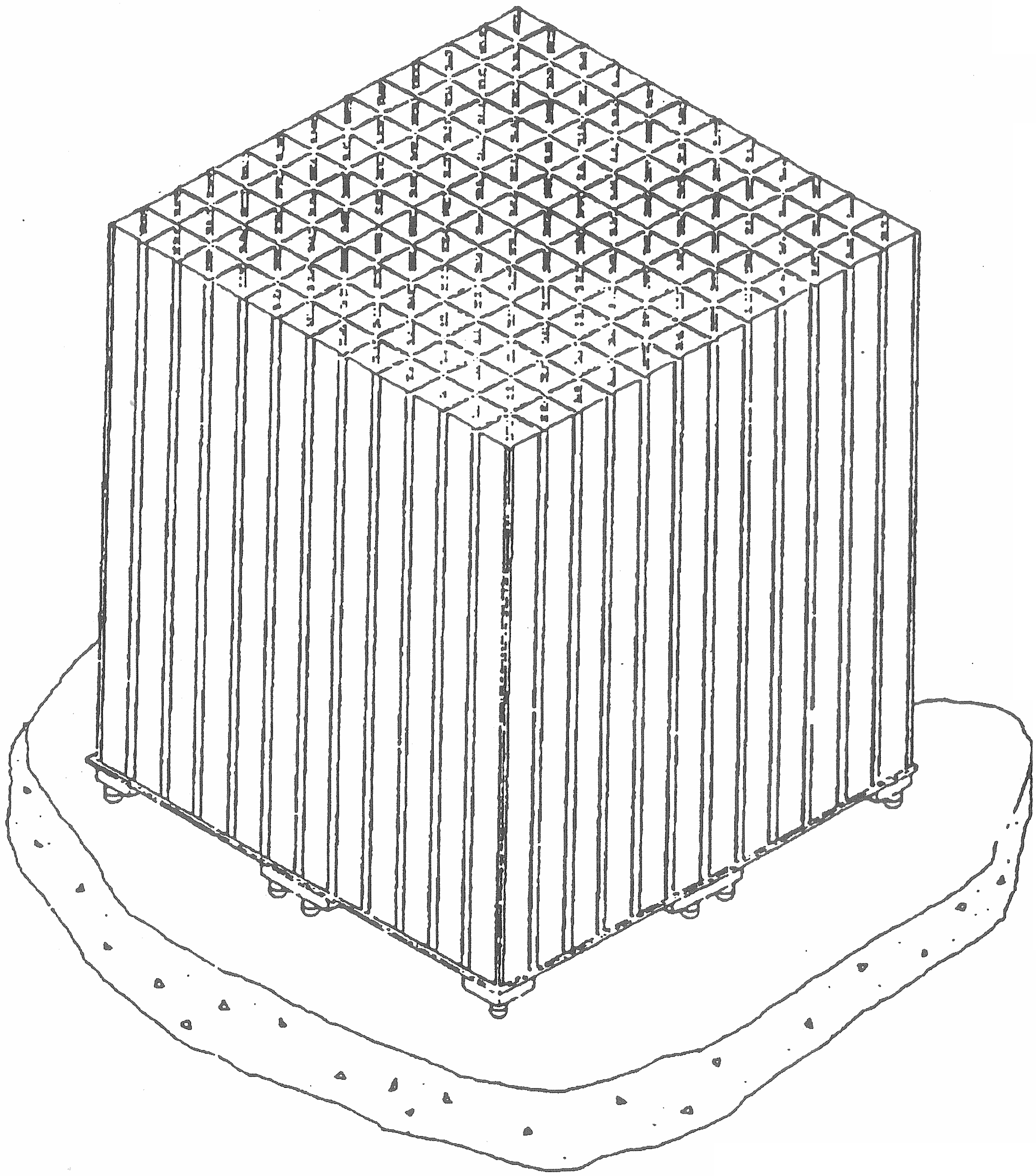
AMENDMENT 21

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

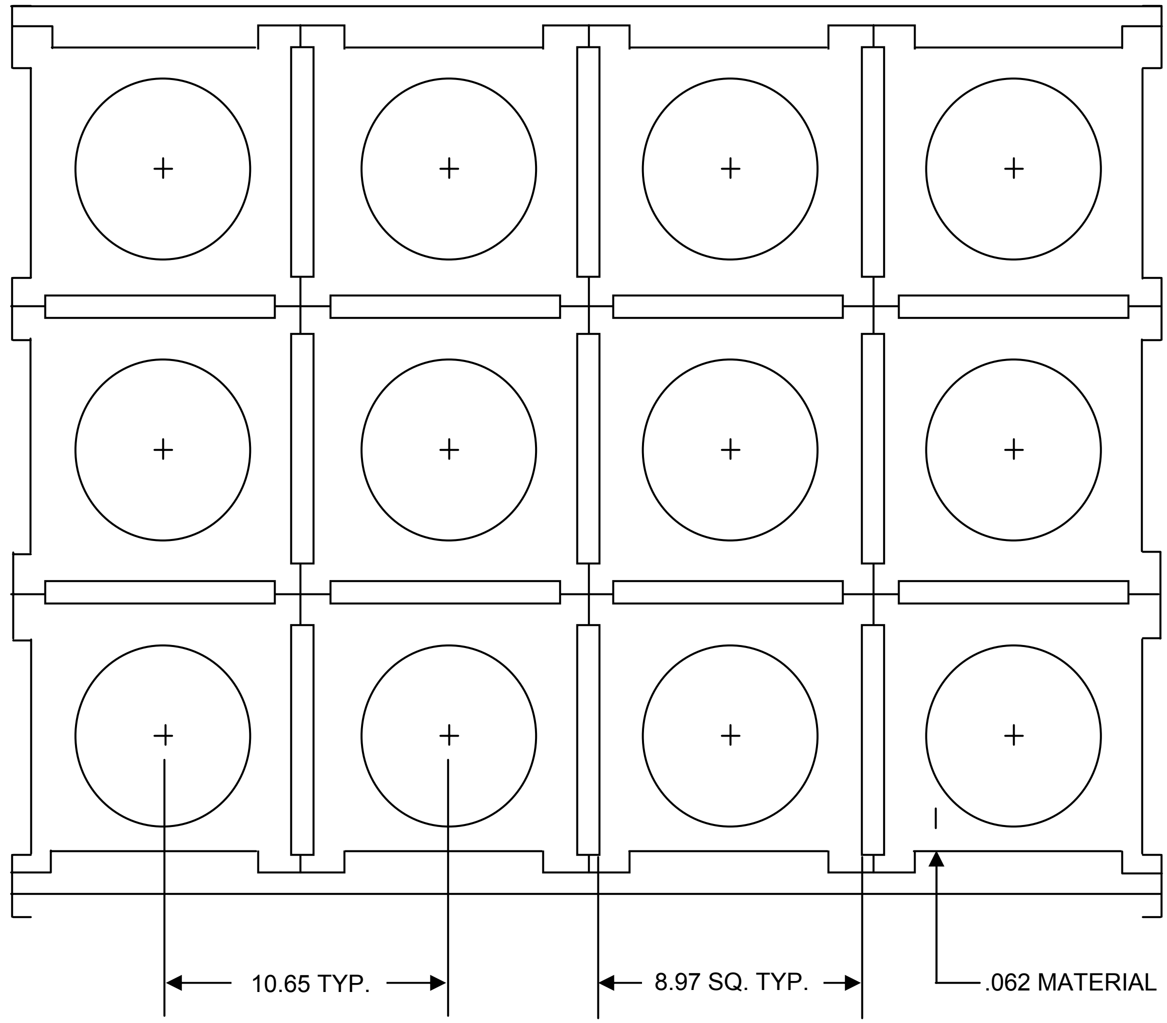
BASED ON DRAWING NO	SHEET	REV.



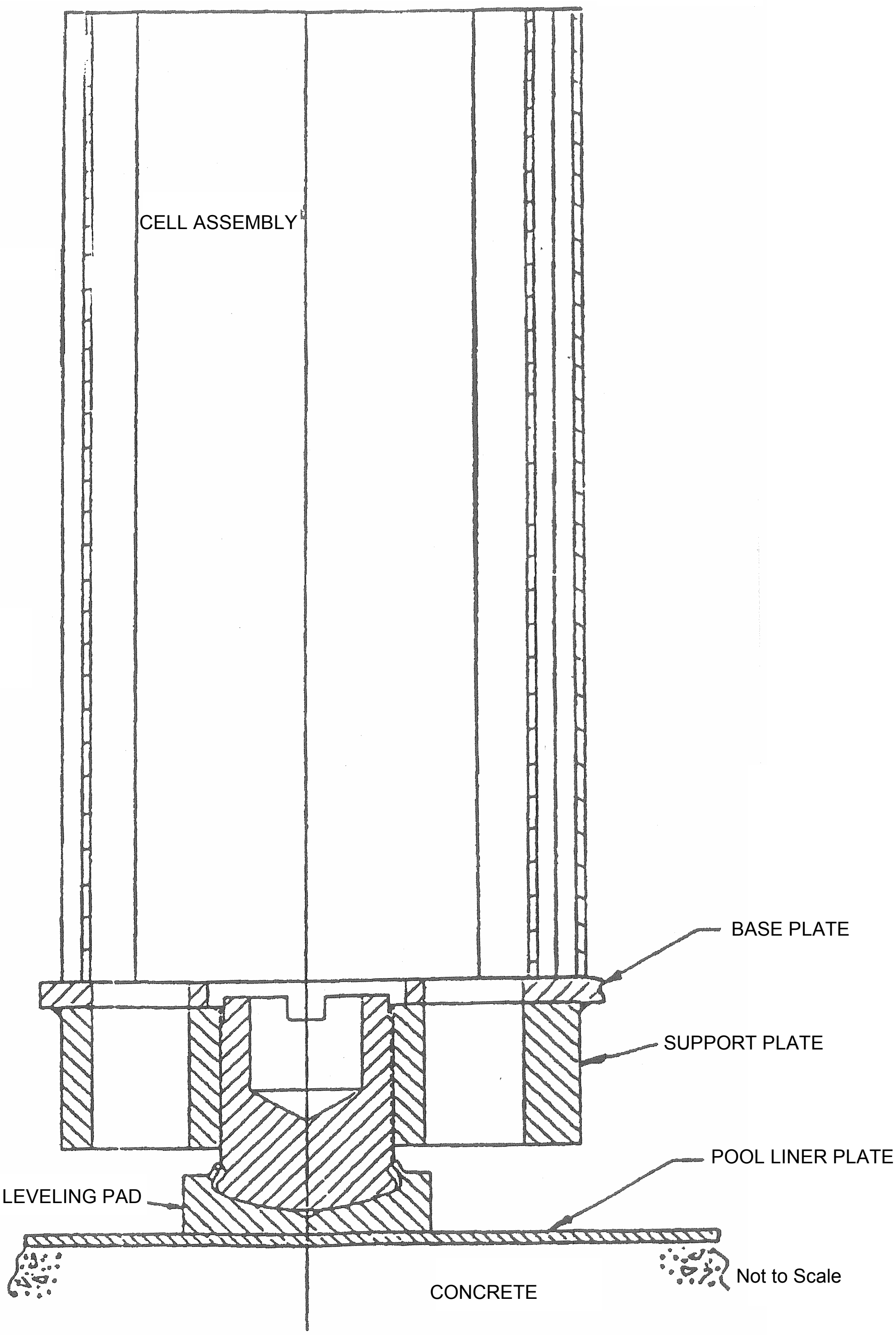
SAR FIGURE NO. 9-54

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
SPENT FUEL POOL REGION 2 STORAGE RACK "SPACER POCKET" DESIGN	BASED ON DRAWING NO	SHEET 1	REV.



SPENT FUEL POOL REGION 2 STORAGE RACK "SPACER POCKET" DESIGN PLAN VIEW			SAR FIGURE NO. 9-54A
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



SAR FIGURE NO. 9-55

AMENDMENT 20

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		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

FUEL RACK BASE SUPPORT ASSEMBLY AND CELL ASSEMBLY	BASED ON DRAWING NO	SHEET	REV.
		1	

8									
7									
6									
5			X	X	X	X	X		
4			X	X	X	X	X		
3									
2									
1									
	A	B	C	D	E	F	G	H	K

NOTE: X's SHOW THE LOCATIONS THAT CANNOT STORE FUEL ASSEMBLIES.

SAR FIGURE NO. 9-56a

AMENDMENT 21

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN:
DESIGN: ENTERGY
CAD NO:

ACCEPTABLE NEW FUEL STORAGE VAULT CONFIGURATION FOR 4.95 WT% ENRICHMENT FRESH FUEL

BASED ON DRAWING NO	SHEET	REV.
	1	

8									
7									
6					X				
5				X	X	X			
4				X	X	X			
3					X				
2									
1									
	A	B	C	D	E	F	G	H	K

NOTE: X's SHOW THE LOCATIONS THAT CANNOT STORE FUEL ASSEMBLIES.

SAR FIGURE NO. 9-56b

AMENDMENT 21

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ACCEPTABLE NEW FUEL STORAGE VAULT CONFIGURATION FOR 4.2 WT% ENRICHMENT FRESH FUEL

BASED ON DRAWING NO	SHEET	REV.
	1	

DELETED

SAR FIGURE NO. 9-57

AMENDMENT 21

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.
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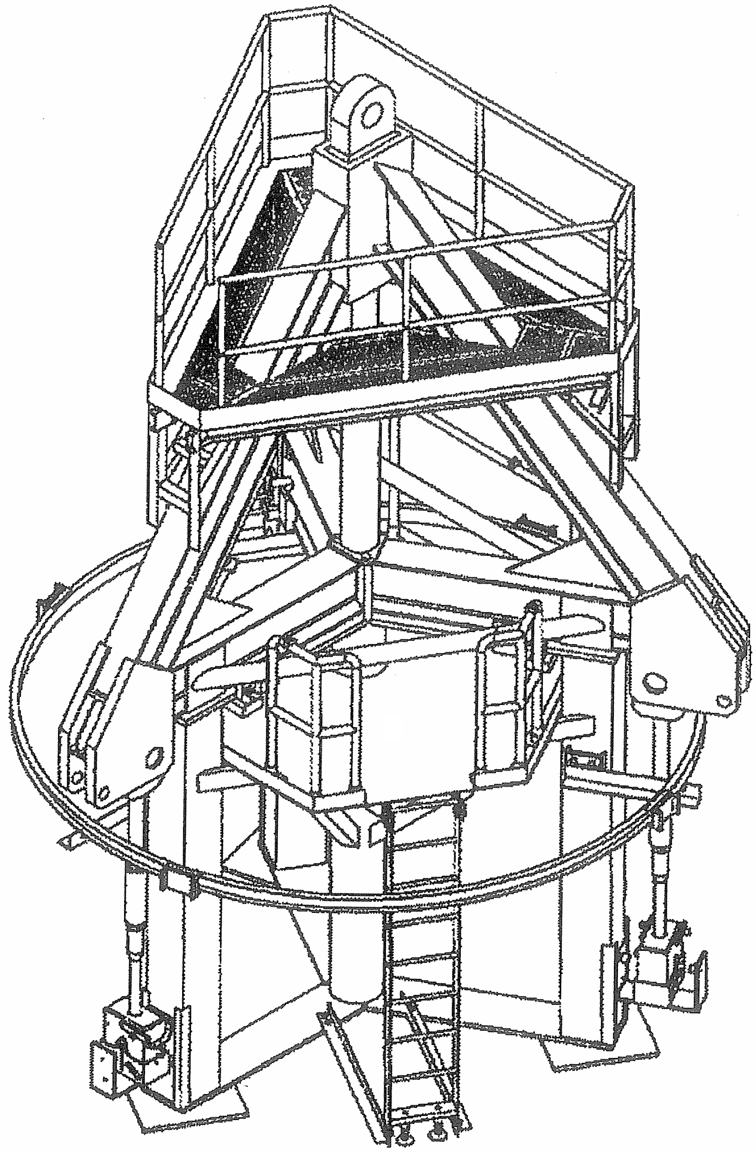

SAR FIGURE NO. 9-58

AMENDMENT 21

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
TRIPOD STORAGE STAND	BASED ON DRAWING NO	SHEET / REV.

APPENDIX 9A

FIRE PROTECTION PROGRAM

9A.1 PROGRAM DESCRIPTION

The Arkansas Nuclear One Fire Protection Program is defined as the following documents:

A. Section 9.8 of the Safety Analysis Report (SAR),

B. The Operability, Surveillance, and Administrative Requirements (located in the SAR Appendix 9D),

C. The 2014, 2015, and 2016 NFPA 805 submittals, and

D. The 2016 NFPA 805 fire protection Safety Evaluation Report.

9A.2 SCOPE AND APPLICABILITY

The ANO Fire Protection Program has been incorporated into the SAR per Generic Letters 86-10 and 88-12. The Program is described in Section 9.8 and Appendices 9A through 9D.

The purpose of the Fire Protection Program is to extend the concept of defense-in-depth to fire protection in fire areas important to safety with the following objectives:

A. Preventing fires from starting,

B. Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage, and

C. Providing an adequate level of fire protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

The Fire Protection Program also delineates the responsibilities and the methods to be used to accomplish the objectives stated above. The Fire Protection Program will interface with other ANO manuals, plans, and procedures to provide an effective and coordinated Fire Protection program that encompasses all phases of operation, administration, and maintenance.

9A.3 ORGANIZATION AND RESPONSIBILITY

The personnel and organizations responsible for the oversight, implementation, and maintenance of the ANO Fire Protection Program are described in SAR Section 12.1.1.

9A.4 ADMINISTRATIVE CONTROLS AND PROCEDURES

The ANO Fire Protection Program is controlled and maintained by various plant procedures that includes, but not limited to, implementing procedures, operational procedures, maintenance procedures, surveillance procedures, etc.

9A.5 FIRE HAZARDS ANALYSIS

The ANO Fire Hazards Analysis (FHA) is described in the SAR Appendix 9B.

9A.6 SAFETY EVALUATION REPORTS

ANO received a Safety Evaluation Report for the Fire Protection Program in 1978 and another one for Appendix R exemptions in 1988. These documents are numbered 1CNA087810 (1N-78-91) for 1978 and 1CNA108806 for 1988. Subsequently, ANO requested NRC approval to transition to NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The NFPA 805 Safety Evaluation Report was received in 2016 (1CNA101601).

9A.7 SAFE SHUTDOWN SYSTEMS

A listing of safe shutdown systems and the methodology used to insure that modifications do not affect the ability to achieve a safe shutdown following a fire are included in the FHA (SAR Appendix 9B).

9A.8 FIRE PROTECTION SYSTEMS

The ANO fire protection systems consists of fire pumps and distribution header, sprinkler systems, detection systems, hose reels, portable extinguishers, barriers, etc. A description of the fire protection systems may be found in the SAR Section 9.8.

9A.9 QUALITY ASSURANCE

Appropriate QA program controls insure that the Fire Protection Program is maintained in accordance with SAR Section 12.1.1.

9A.10 FIRE BRIGADE

Fire Brigade staffing, training, etc. may be found in the SAR Section 9.8.

APPENDIX 9B

FIRE HAZARDS ANALYSIS REPORT

The ANO Fire Hazards Analysis (FHA), maintained under separate cover, is considered part of the Fire Protection Program described in Appendix 9A of the SAR and as such is subject to the provisions of ANO Operating Licensing Condition 2.C.(8).

APPENDIX 9C

DELETED

APPENDIX 9D

FIRE PROTECTION SYSTEM ADMINISTRATIVE REQUIREMENTS

Appendix 9D contains the administrative requirements related to Fire Protection Systems. These requirements were relocated from the ANO-1 Technical Specifications in accordance with Generic Letter 88-12. Other Fire Protection System requirements have been relocated to the ANO-1 Technical Requirements Manual.

9D.1 ADMINISTRATIVE REQUIREMENTS

9D.1.1 FIRE BRIGADE

At least 5 individuals with fire protection training shall be maintained onsite at all times. These individuals shall not include the minimum shift crew necessary for safe shutdown of the unit (2 members) or any personnel required for other essential functions during a fire emergency.

9D.1.2 TRAINING

A training program for the Fire Brigade shall be maintained and shall meet or exceed the requirements of Section 3.4.3 of NFPA 805, as approved in the NRC Safety Evaluation Report associated with the transition to NFPA 805 (1CNA101601).

CHAPTER 10

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10-2	CONDENSATE AND FEEDWATER, EMERGENCY FEEDWATER
10-3	COMMON FEEDWATER SYSTEM

UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
10.3	Correspondence from Phillips, AP&L, to Giambusso, NRC, dated April 29, 1974. (1CAN047423)
10.3	Correspondence from Phillips, AP&L, to Giambusso, NRC, dated January 6, 1975. (1CAN017502)
10.1.1 10.2 10.4.7	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated May 21, 1979. (1CAN057925)
10.4.8	Correspondence from Cavanaugh, AP&L, to Denton, NRC, dated May 11, 1979. (1CAN057920)
10.4.8	Correspondence from Cavanaugh, AP&L, to Denton, NRC, dated May 17, 1979. (1CAN057924)
10.4.8	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated December 18, 1979. (1CAN127913)
10.1 10.3 10.4 10.5	Design Change Package 80-1083, "Emergency Feedwater Initiation."
10.3	Correspondence from Trimble, AP&L, to Reid, NRC, dated March 13, 1981. (1CAN038104)
10.4.8	Correspondence from Trimble, AP&L, to Stolz, NRC, dated April 30, 1981. (1CAN048110)
10.4.8	Correspondence from Marshall, AP&L, to Stolz, NRC, dated May 27, 1982. (1CAN058210)
10.4.8	Correspondence from Marshall, AP&L, to Stolz, NRC dated May 31, 1983. (1CAN058307)
10.3	Correspondence from Cavanaugh, AP&L, to Denton, NRC, dated October 7, 1979. (1CAN107907)
10.4.7	Correspondence from Trimble, AP&L, to Reid, NRC, dated January 16, 1981. (1CAN018114)
10.1.1 10.3	Design Change Package 187, "SLBIC," 1974. (1DCP740187)

UPDATE REFERENCE LIST (continued)

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10.1.1 10.2 10.4.7	Design Change Package 1032, "Reactor Trip on Loss of Feedwater or Turbine Trip," 1979. (1DCP791032)
10.3	Design Change Package 85-1012, "EFW Modifications."
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Figure 10-2	Design Change Package 80-2123L, "ANO-1 SPDS."
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10.4.8	AP&L Engineering Report, 89E-0054-01, "ANO-1 EFW Delay Justification."
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Figure 10-2	ANO Procedure 1106.16, "Condensate, Feedwater, and Steam System Operation."
Figure 10-2	Design Change Package 86-1091, "EFW Pump Recirculation Indicators."
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10.1.1	Design Change Package 88-1092, "OTSG Level Setpoints & ICS Known Safe States."
Figure 10-2	ANO Procedure 1106.016, Rev. 23, "Condensate, Feedwater and Steam Systems Operation."
Figure 10-2	Design Change Package 89-1063, "Heater Drain Tank Dump Valve Upgrade."
Figure 10-2	Plant Change 89-0790, "Remove Cross-Connect Between the Domestic Water and Condensate Systems."
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Figure 10-2	Design Change Package 86-1006, "Safety Systems Diagnostic Instrumentation."
Figure 10-2	Design Change Package 90-1019, "Auxiliary Feedwater System Modifications."

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Figure 10-2	Limited Change Package 89-5017, "Main Feed Water Recirculation Controllers."
10.4.5	ANO-1 Technical Specifications Amendment 132, dated July 6, 1990. (0CNA079009)
10.4.7	ANO Procedure 1000.042, Rev. 7, "Steam Generator Water Chemistry Monitoring - Unit One."

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Section	Cross References
10.4.7	Design Change Package 86-1121, "Main Feedwater Turbine Trip Modifications."
10.4.8 Figure 10-2	Design Change Package 84-1063, "Pressure Gauge on Discharge of P7A."
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Figure 10-2	Plant Change 85-0200, "Add Resonant Reed Tachometer to EFW Pump Turbine."
Figure 10-2	Design Change Package 88-1009A, "ANO-1 Secondary System Morpholine Injection."
Figure 10-2	Plant Change 88-1332, "Removal of Sensing Lines, Pressure Indicators, and Valves on Heater Drain Pumps, P8A & B."
Figure 10-2	Design Change Package 90-1025, "MFW Pump Recirculation."
Figure 10-2	Design Change Package 90-1035, "Startup Boiler Control Valves."
Figure 10-2	Plant Change 91-7015, "P-1A Suction Pipe Drain Valve Removal."

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10.4.5	Design Change Package 90-2023, "Sodium Bromide/Sodium Hypochlorite System Addition."
10.4.8 Table 10-1	Limited Change Package 90-5009, "CLQL Replacement of CS-99."
Figure 10-2	Design Change Package 90-1067, "Unit 1 Plant Computer Replacement."
Figure 10-2	Design Change Package 88-1070, "Upgrading of the Heater Drain Tanks."
Figure 10-2	Plant Engineering Action Request 91-7250, "Addition of Sample Valves to Pump Discharges on P&ID."

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Figure 10-2	ANO Procedure 1106.06, Rev. 37, "Emergency Feed Pump Operation."
Figure 10-2	Plant Change 91-7027, "Addition of EFW Pump/Turbine Speed Indication to PPAS."
Figure 10-2	Plant Change 91-7053, "EFW Turbine Exhaust Line Steam Trap Replacement."
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Figure 10-2	Plant Change 92-7103, "Corrosion Product Monitors."
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10.3	Licensing Document Change Request to Reflect Changes and Results of Cycle 12 Reanalysis.
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10.2	Condition Report 1-96-0155, “Guard Pipe on Hydrogen Supply Line.”
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Figure 10-2	Design Change Package 95-1015, “Atmospheric Dump Valve Replacement.”
Figure 10-2	Plant Change 95-7007, “Conversion of Emergency Feedwater Pump Discharge Pressure Transmitters.”
Figure 10-2	Plant Change 95-7081, “Modification of Valve FW-1016.”
Figure 10-2	Plant Change 95-7103, “Modification to Emergency Feedwater Isolation Valves to Prevent Pressure Locking and Thermal Binding.”

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Figure 10-2	Plant Change 963408P101, “Isolation of Abandoned Piping from Condenser E-11B.”

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10.2 Figure 10-2	Design Change Package 951018D101, "Main Condenser Tube Bundle and Waterbox Replacement."
10.2	Plant Change 974396P101, "Main Turbine LP-B Rotor Installation."
10.4.6	Procedure 1102.004, "Partial Bypass of Demineralizers."
Figure 10-2	Design Change Package 95-1010, "Main Feedpump Turbine Supervisory Instrumentation."
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Figure 10-2	Plant Change 94-8026, "Permanent N_2 Supply to Condensers."

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Nuclear Change 963314N101, "Main Steam Isolation Valve Upgrade."

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10.4.5	ANO Procedure 1628.014, “Operation of the Oxidizing Biocide System.”
Figure 10-2	Engineering Request 002475N101, "MSR Demineralization Tie-Ins."
Figure 10-2	Engineering Request 991642N101, "ANO-1 EFW Steam Supply Check Valve Replacement."

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Figure 10-2	Engineering Request ER-ANO-2001-0745-002, “Hater Drain Pump P-8A/B Seal Cooling Water Supply System Evaluation”.

Amendment 19

10.4.6	Engineering Request ER-ANO-2000-2475-003, “Installation of Moisture Separator Reheater System”

UPDATE REFERENCE LIST (continued)

Section	Cross References
10.4.3	Engineering Request ER-ANO-2003-0920-000, “Addition of Reference to the ANO-2 Steam Supply to ANO-1”
10.1.6	SAR Discrepancy 1-97-0586, “Clarification of Main Steam Safety Valve Design Basis”
Figure 10-2 Sht 1 & 3	Drawing Revision Notice DRN 04-1887, "Corrects P&ID M-204 Sheet 1 By Depicting Four Condensate Cross Under Lines That Exist Beneath E11A And E11B Main Condenser Hotwells"
Figure 10-1	License Document Change Request 1-1.1-0008, “Reformatting of Various Drawing Labels”
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10.4.8	Engineering Request ER-ANO-2002-1381-000, “Replacement Steam Generator Project”
Figures – ALL	License Document Change Request 1-1.1-0009, “Deletion/replacement of Excessive Detailed Drawings from the SAR”
10.4.8	Engineering Request ER-ANO-2005-0871-000, ERCN 1, "Revision of EFW Delivery Time"
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10.4.5	Engineering Request ER-ANO-2002-1083-000, “Modification of Unit 1 and Unit 2 Service Water System Chemical Addition System”
10.4.7	Engineering Request ER-ANO-2000-2294-001, “Replacement of Feedwater Heaters E-4A/B and E-5A/B”
Figure 10-1	Engineering Request ER-ANO-2006-0389-000, “Q-CST Alignment to ANO-2 EFW System”
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Figure 10-1	Engineering Change EC-44043, "ANO-1 FLEX Strategy"
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10.1	Condition Report CR-ANO-1-2017-1230, "Clarification of Loss of Load Events"
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10 STEAM AND POWER CONVERSION SYSTEM

10.1 SUMMARY DESCRIPTION

The steam and power conversion system is designed to remove heat energy from the reactor coolant in two steam generators and converts a portion of this energy into electrical energy via the turbine generator. This system is shown schematically in Figure 10-1. The condenser transfers the heat which is unusable in the cycle to the condenser cooling water and deaerates the condensate. The closed regenerative turbine cycle heats the condensate and returns it to the steam generators. The entire system is designed for the maximum expected energy from the Nuclear Steam Supply System (NSSS).

Upon loss of full load, the system will dissipate all the energy existent or produced in the Reactor Coolant System (RCS) either through bypass valves to the condenser (CV 6687, CV 6688, CV 6689, and CV 6690), the power operated atmospheric dump valves (CV 2668 and CV 2618), or the main steam safety valves (PSV 2684, PSV 2685, PSV 2686, PSV 2687, PSV 2688, PSV 2689, PSV 2690, PSV 2691, PSV 2692, PSV 2693, PSV 2694, PSV 2695, PSV 2696, PSV 2697, PSV 2698 and PSV 2699). The unit is designed to maintain station auxiliary load without a reactor trip upon loss of full load due to faults occurring on the system side of the 500 kV switchyard breakers. This loss of full load assumes the condition results in the opening of the main generator output breakers B5114 and B5118. This may be caused by protective features that result in a turbine/generator lockout or protective features in the switchyard that result in the opening of the generator output breakers. As described in SAR Sections 7.2.3.3.4 and 14.1.2.8.1, the ICS will attempt to maintain the unit on-line during a loss of load event which does not involve a turbine/generator lockout by initiating a runback to ~15% full power (note that a generator lockout also locks out the main turbine, defeating the need for a runback). However, the runback may not be successful in all cases and is not credited in the accident analyses. The steam bypass to the condenser and atmospheric dump valves will be utilized as necessary to achieve this load reduction. Note that conditions where the main generator output is channeled only through the Autotransformer may lead to a loss of load event (depending on power level), but initial plant response may be more complicated than an initiating loss of load event (see Section 14.1.2.8.1).

The steam and power conversion system provides steam for driving two steam generator feedwater pumps, and building heating as required. These are completely closed systems and have no direct atmospheric interfaces. Steam used for the turbine gland sealing system does not leak outward. Leakage, if any, will be air leakage inwards. Steam used for the emergency feedwater pump turbine is exhausted to atmosphere. This exhausted steam could be radioactive only if a steam generator tube leak exists. If this unlikely event were to occur and radioactivity leakage to the atmosphere should occur, the turbine driven pump would be secured or isolated from the affected steam generator and the backup electrical driven emergency feedwater pump would be available.

In the absence of minor primary to secondary leaks, there are no radioactive contaminants present in the steam and power conversion system. It is possible for this system to become contaminated through steam generator tube leaks. In these events, monitoring of the steam generator shell side sample points and the condenser vacuum pumps suction air will detect any contamination. Additionally, radiation monitoring instrumentation is located on each main steam line. These instruments, having a measurement range from 10^{-1} to 10^{4} mR/hr, are available whenever the reactor is above the cold shutdown condition.

10.1.1 TRIPS, AUTOMATIC CONTROL ACTIONS AND ALARMS

Trips, automatic control actions and alarms will be initiated by deviations of system variables within the steam and power conversion system. In the case of automatic corrective action in the steam and power conversion system, appropriate corrective action will be taken to protect the RCS. The more significant malfunctions or faults which cause trips, automatic actions or alarms in the steam and power conversion system are:

A. Main Turbine Trips

1. Generator faults.
2. Electrical faults (Loss of ± 15 volt or 48 volt DC in EHC cabinet).
3. Loss of condenser vacuum.
4. Thrust bearing wear.
5. Low bearing oil pressure.
6. Turbine overspeed.
7. Reactor trip.
8. Manual (Emergency) Trip.
9. Rotor long alarm.
10. Rotor short alarm.
11. Trip of both main feedwater pump turbines.

B. Automatic Control Actions

1. Feedwater flow lagging feedwater demand results in a reduction in power demand.
2. Low feedwater temperature results in a reduction in demand.
3. High level in steam generator results in a reduction in feedwater flow.
4. Low level in steam generator results in an increase in feedwater flow.
5. Loss of main feedwater pump results in plant runback to 40 percent of unit load demand.
6. Loss of two main feedwater pumps will initiate emergency feedwater if reactor power is greater than nominal 10 percent (for exact value see the ANO Unit 1 Technical Specification).

C. Principal Annunciator Alarms

1. Low pressure at each feedwater pump suction.
2. Low vacuum in condenser.
3. High or low water level in condenser hotwell.
4. High or low water level in steam generator.
5. High or low pressure in steam generator.
6. Unit load demand in excess of reactor coolant flow (power to flow rates), maximum load limit, feedwater pumps limit, reactor coolant flow, reactor coolant pump limits, steam generator low level limit or asymmetric rod limit.
7. Feedwater reactor limited or reactor feedwater limited.
8. High hydrogen gas temperature in the generator.
9. Low differential pressure across feedwater control valves.

Anticipatory safety grade reactor trips are provided for loss of both main feedwater pumps and for turbine trip. These reactor trips are bypassed below a predetermined flux level, typically nominal 10 percent of full power for the loss of feedwater pumps and nominal 45 percent of full power for turbine trip (for exact values see ANO Unit 1 Technical Specification), to allow for normal startup.

A part of the Emergency Feedwater Initiation and Control (EFIC) is installed to provide quick response to steam line ruptures. The EFIC is designed to detect a main steam line rupture and isolate the affected steam generator by closing the respective feedwater isolation valve and main steam isolation valve. Upon main steam line isolation a signal also initiates emergency feedwater.

10.1.2 TRANSIENT CONDITIONS

The analysis of the effects of loss of full load on the RCS is discussed in Chapter 14. Analysis of the effects of partial loss of load on the RCS is discussed in Chapter 7.

10.1.3 MALFUNCTIONS

The effects of inadvertent steam relief or steam bypass are covered by the analysis of the steam line failure given in Chapter 14. The effects of an inadvertent rapid throttle valve closure are also covered by the loss of full load discussion in Chapter 14.

10.1.4 FUNCTIONAL LIMITATIONS

The rate of change of reactor power is limited to values consistent with the characteristics of the RCS and its control systems. Further limitation in the steam and power conversion system may reduce the RCS functional limits as given in Chapter 7.

10.1.5 IMPURITIES CONTROL

Impurities in the steam and power conversion system are controlled to maintain specified steam generator water purity as given in the EPRI Secondary Water Chemistry Guidelines. The condensate is polished by a full flow demineralizer system.

10.1.6 OVERPRESSURE PROTECTION

The Main Steam system design pressure of 1,050 psig. The first safety valve in each of two safety valve banks, located outside the reactor building, is set to relieve at this pressure. The design pressure is based on the operating pressure of 910 psia plus a 10 percent allowance for transients and a four percent allowance for blowdown. Additional safety valve banks are set at pressures up to 1,100 psig, as allowed by the ASME Code, Section III, Class 2.

The design basis of the Main Steam Safety Valves is to limit secondary system pressure to $\leq$ 110% of design pressure when passing 102% of design steam flow (100% plus 2% heat balance error). The main steam pipe wall thickness is increased in the area of the pressure relief valves to safely transmit full discharge loads to the main steam system anchors. For the case of the main steam relief valves, the design criteria, which have been used to take into account full discharge loads imposed on valves and connected piping, considers the eight valves on each side discharging concurrently.

10.1.7 TESTS AND INSPECTIONS

As is essential in successful operation of any modern power station, frequent functional operational checks will be made on vital valves, control systems and protective equipment.

10.2 TURBINE - GENERATOR

The turbine is a tandem compound, 3-cylinder, 4-flow, impulse reaction, condensing, 1,800 rpm unit with 44-inch last row blades. It has external moisture separation and two stage steam reheaters between the High Pressure (HP) and the Low Pressure (LP) cylinders.

Note: During the 1R8 Refueling Outage a "fully integral" Low Pressure turbine rotor was installed to replace the "built up" rotor design in the "1" section of the Low Pressure turbine. During the 1R14 refueling outage, the ANO-1 ″2″ LP turbine rotor was replaced with a partially integral rotor design.

Connected to the turbine shaft is the AC generator. Coupled to the generator is the excitation system.

Steam enters the HP turbine through four stop valves and four governing control valves. After expanding through the HP turbine, the exhaust steam flows through moisture separators and reheaters to the LP turbines. Reheated steam leaves the moisture separator vessels and flows to the LP turbines through reheat stop and intercept valves in the reheat steam lines. The moisture from the separators is returned to the feedwater system.

In the absence of or in the unlikely event of a minor primary to secondary leak, no radiation shielding is required for components of the steam and power conversion system, and continuous access to components of this system is possible during normal operation. If a leak were to occur between the primary and secondary systems, then the plant staff will implement normal radiation protection control procedures to protect plant personnel during access to these systems.

Following a turbine trip, the safety valves will relieve excess steam until the output is reduced to the point at which the steam bypass to the condenser or, if necessary, the atmospheric dump valves can handle all the steam generated. A turbine trip causes an anticipatory safety grade reactor trip. This reactor trip is bypassed below a predetermined flux level, typically nominal 45 percent of full power (for exact value see ANO Unit 1 Technical Specification), to allow for normal startup.

The turbine generator equipment conforms to the applicable ANSI, ASME, and IEEE Standards. Protective measures incorporated in the design to prevent the creation of turbine missiles are discussed in Section 14.1.2.9.

The turbine generator field windings are hydrogen cooled. Turbine generator hydrogen is stored in 24 permanently installed cylinders outside the turbine building in the gas storage building, approximately 660' east and 150' south of the reactor building centerline. This storage is for both Units 1 and 2 and also serves to house carbon dioxide bottles. Hydrogen is piped to the turbine building.

10.3 MAIN STEAM SUPPLY SYSTEM

The steam and power conversion system is shown in Figure 10-1. Steam generated in each of the two steam generators is conducted through the reactor building wall in separate lines. The penetration design for the main steam lines is discussed in Section 5.2.2.1.2 and is shown in Figure 5-3 sheet 2. Each main steam line, between the reactor building penetration and the corresponding block valve, is provided with spring-loaded safety valves and power-operated dump valves which discharge to atmosphere. Motor operated block valves are provided to isolate the atmospheric dump valves if either should fail to reseat during a Main Steam Line Break (MSLB), Main Feedwater Line Break (MFWLB), or Loss of Coolant Accident (LOCA) condition. Each main steam line contains a main steam isolation valve (MSIV) just downstream of the main steam safety valves. The MSIVs are air and spring actuated and are designed to fail closed on loss of instrument air. The MSIV air operators function to maintain the valves in the open position and to allow the MSIVs to close rapidly upon receipt of a Main Steam Line Isolation (MSLI) signal from EFIC. The MSIVs have a hydraulic speed control cylinder mounted in tandem with the air cylinder to slow the stem travel rate of the MSIVs. Each MSIV has a 2″ bypass line that can be used to reduce DP and warm the downstream main steam piping. DP must be reduced to 50 psi or less and the steam lines must be warmed to prevent water/steam hammer when opening the MSIV to allow use of the Turbine Bypass system for heat removal. Each MSIV bypass line has a 2″ manual globe valve that is maintained locked closed during normal operations. The MSIVs and Atmospheric Dump Valves (ADVs) are served by shared accumulators that act to provide an air reserve in the event the instrument air supply is lost. The reserve air source is relied upon to maintain the MSIVs open and to allow initial control of the ADVs, but these are not safety related functions. Upon receiving Channel A and/or Channel B MSLI signal(s), one or both of the MSIVs' solenoid operated valves are energized to dump air from the bottom side of the air operator piston and to supply air to the top side of the piston. No credit is taken for the air operator assisting the springs to close the MSIVs. The main steam isolation valves are designed to serve as containment isolation valves, and meet Criterion 57 of 10 CFR 50, Appendix A.

Each main steam line supplies two turbine leads. Each turbine lead has an automatic turbine stop valve and a governing-control valve ahead of the turbine. The main steam lines are cross-connected downstream of the turbine stop valves.

The Main Steam Safety Valve (MSSV) nominal relief capabilities are as follows:

Quantity	Pressure Setting (psig)	Capacity lb/hr flow	% Accumulation*
2	1050	845,759 each	10
2	1060	845,759 each	9
4	1070	845,759 each	8
4	1090	845,759 each	6
4	1100	845,759 each	5

* % Accum = $\frac{P_{DF} - P_{LIFT}}{P_{LIFT}} * 100$

As designed, purchased and installed, valves located on the main steam lines were specified to have zero leakage to the atmosphere. At six percent under the set pressure, there was to be no visually detectable leakage and no condensation observed on a test metal rod placed across

the outlet. Minor leakage during normal operation is not uncommon and is acceptable. Should leakage occur, considerations of personnel safety would determine whether the whole system has to be secured for maintenance work to be performed.

Downstream of the main steam isolation valves are turbine bypass valves which discharge to the condenser. There are four turbine bypass valves, each with a rated capacity of 418,500 lb per hour. Similarly, these valves are designed for tight shutoff and zero leakage may be expected.

A portion of the Emergency Feedwater Initiation and Control (EFIC) system is designed to detect a main steam line rupture and isolate the affected steam generator by closing the respective Main Feedwater Isolation Valve (MFIV) and Main Steam Isolation Valve (MSIV). Upon initiation, EFIC also provides a signal to initiate emergency feedwater. Manual closure of the MSIV or MFIV can be performed from the control room at any time.

During normal operation steam generator outlet pressure varies from 895 psig ± 9 psig at zero percent power to 910 psig ± 9 psig at 100 percent power. During plant heatup and cooldown, steam pressures may reach atmospheric values. During expected transients, main steam pressure may decrease to values as low as 700 psig.

Six hundred psig was selected as the setpoint for initiating protective action because it represents a significant degradation in system pressure, outside the range of expected transients, and indicates a high probability of loss of main steam or main feedwater system integrity. Concurrently, actuation at 600 psig assures timely response of protective systems to design basis events and maintains reactor core and Reactor Coolant System (RCS) parameters within allowable limits.

System response time is limited by the closing rate of the main steam isolation valves and the main feedwater isolation valves. Assumptions for closure times of these valves for the analyzed accident scenarios are provided in Section 14.2.2.1. Response times of various electronic components will (in total) not exceed 0.25 - 0.75 seconds and are, therefore, insignificant. Pressure sensing devices in the system are accurate to within two percent.

During normal plant operation, the MSIVs and the MFIVs should never have to close. Inadvertent closure of these valves will likely result in a plant trip. To help prevent an inadvertent closure a 1-out-of-2 taken twice coincidence logic has been engineered into EFIC. In addition, EFIC has been designed to meet the criteria set forth in IEEE Std. 279-1971.

The built-in test facilities of the EFIC system permit complete electrical testing of each channel from its input terminals to the trip interface equipment (see Section 7.1.4.12), while the plant is operational. The MSIVs and the MFIVs can be exercised while the plant is operational. Complete testing of all inputs, logics and components will normally be accomplished during cold shutdown.

The EFIC system logic provides redundant main steam and EFW isolation signals (see Section 7.1.4). Upon receipt of a main steam isolation trip signal, each EFIC cabinet will act to close the appropriate MSIV. Either cabinet can control both MSIVs. The MSIV can be manually closed from the control room at anytime. The MSIV will remain closed until the trip condition goes away and the valve manually opened from the control room. Since EFIC will not discriminate between steam line rupture and normal low pressure, a manual bypass to the EFIC trips is performed from the control room during start-up and normal shutdown. The bypass is operational for steam generator pressure less than 750 psi. It can be manually reset, or it will be automatically reset before pressure rises above 750 psi during startup.

Following a steam line break, steam pressure will decrease at a rate commensurate with the size of the break until it reaches 600 psig. At this point, the main steam isolation valve and main feedwater isolation valve will begin to close and the steam pressure will begin to rise (assuming the break has been isolated). EFIC will initiate EFW and feed either or both steam generators according to EFW vector logic. EFIC will isolate Emergency Feedwater to the de-pressurized loop according to its vector logic criteria. The atmospheric dump valve block valves begin opening and the Atmospheric Dump Valves (ADV) will begin to relieve pressure to the atmosphere. The pressure will rise to 1,050 psig at which point the first group of main steam safety valves will lift. Upon resetting of the safety valves, EFIC will control steam generator pressure at the selected setpoint by modulating the ADV. If the break has not been isolated in either loop, the steam pressure in that loop will continue to drop.

The EFIC cabinets control the operation of the EFW turbine steam isolation valves. There is one valve per steam generator, and it is normally open. An EFW trip would send an open signal to these valves, in order to supply steam to the EFW turbine driven pump. Manual bypass capability is provided in the control room over all automatic control of these valves. Position in any other than 'Auto' will be continuously annunciated in the control room.

The atmospheric dump valve block valves have safety grade power to allow opening if needed in the event of loss of the normal power. (1CAN038104).

A single failure analysis of the EFIC system is provided in Table 10-1 in Chapter 10.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSER

The Main Condenser System is supplied with cooling water by the Circulating Water System described in Chapter 9. The condenser is of the deaerating type and is sized to condense exhaust steam from both the main turbine and the main feedwater pump driving turbines under full-load conditions. It also provides for condensing turbine bypass steam from the steam generator.

The main condenser evacuation system removes the non-condensable gases from the condenser and serves to maintain the condenser vacuum.

The condenser hotwell is a storage reservoir for the deaerated condensate which supplies the condensate and main feedwater pumps, and normally contains five minutes storage (approximately 50,000 gallons) at full load. This supply of condensate is backed up by a larger Condensate Storage Tank from which condensate may be admitted by gravity and vacuum drag into the condenser for deaeration.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

The main condenser evacuation is accomplished by two 100 percent capacity electric motor driven exhausters. For fast evacuation of the non-condensable gases and establishment of the condenser vacuum, both exhausters can be used initially and one will cut out at a predetermined back pressure. During normal operation, should one exhauster not be able to maintain vacuum, the second unit will automatically cut in.

The main condenser vacuum pump air is monitored for radioactivity, and safe operating limits will be established for the station.

All parts of the main condenser evacuation system are accessible for inspection at all times. Historical data removed - to review the exact wording please refer to Section 10.4.2 of the FSAR.

10.4.3 TURBINE GLAND SEALING SYSTEM

During startup, a separate startup boiler or ANO Unit 2 provides steam for turbine gland sealing. Once the Main Steam System is available, startup steam is discontinued. Under normal operating conditions, no radioactive contaminants are present in this system. Additionally, leakage at the glands, if any, will be of air inward. This air will be removed by the gland steam exhauster. In the event of a failure of the pressure regulator, it may be possible for a small amount of steam to leak outward.

All parts of the turbine gland sealing system are accessible for inspection at all times. Historical data removed - to review the exact wording please refer to Section 10.4.3 of the FSAR.

10.4.4 TURBINE BYPASS SYSTEM

The Turbine Bypass System is designed for a total steam flow equivalent to 15 percent of main feedwater flow. The steam flows through four 6-inch lines (two lines off each main steam line) with pneumatically operated pressure reducing control valves. A manual isolation valve is

provided upstream of each of these control valves for isolation in case of malfunction. Quality assured main steam isolation valves are available to isolate either main steam line in case of pipe failure downstream. An interlock insures that condenser vacuum is established before the bypass valves can be opened. Piping is designed to ANSI B31.1.0 standards.

The steam that is condensed during the bypass operation will remain within the closed system consisting of the condenser and the Condensate Storage Tank, and thus will not affect the environment. Since the atmospheric dump system operates in parallel with the turbine bypass, a failure of one system or a part of it will not affect the functional ability of the other system or the remaining part of it.

10.4.5 CIRCULATING WATER SYSTEM

Four circulating water pumps normally supply cooling water to four condenser water boxes. The common header on the pump discharges permit any one pump to be directed to any one condenser water box. Although the circulating water pumps supply cooling water for emergency steam dumping to the condenser, this is not essential for safe plant shutdown.

The circulating water at the intake structure has a biocide added periodically by the Sodium Bromide/Sodium Hypochlorite System. Biocide may also be added manually to the circulating water system from portable containers of sodium hypochlorite and/or sodium bromide.

10.4.6 CONDENSATE CLEANUP SYSTEM

The condensate cleanup system consists of six mixed bed demineralizers and a mixed bed Moisture Separator Reheater drain demineralizer. The mixed bed demineralizer regeneration process is initiated by high sodium ion concentration, individual filter excess pressure drop, or a predetermined cumulative flow.

It is possible for condensate to become contaminated if steam generator tube leaks occur. In this event, monitoring of the steam generator shell side sample points and the condenser vacuum pumps suction air will detect any contaminants. The regeneration wastes are normally processed in Unit 1, but if necessary can be piped to Unit 2 for processing in the Regenerative Waste System.

10.4.7 CONDENSATE AND FEEDWATER SYSTEM

The feedwater cycle is a closed system with deaeration accomplished in the main condenser. Condensate from the hotwell is pumped by the condensate pumps, through the condensate demineralizers, and through the low pressure feedwater heaters to the suction of the main feedwater pumps. The feedwater is then pumped through the high pressure feedwater heating stage to the steam generators.

The feedwater heaters are arranged in two parallel trains, with each train carrying half of the feedwater flow. There are bypasses around the low and high pressure heaters to allow some heaters to be taken out of service, while the others continue functioning.

Two of the three one-half capacity condensate pumps are in normal use. Each of the two turbine driven feedwater pumps is rated at 60 percent full plant capacity. The motor driven auxiliary feedwater pump has five percent full load capacity. It is connected in parallel with the main feedwater pumps and is used for startup and shutdown.

The design, materials and details of construction of the feedwater heaters are in accordance with both the ASME Code, Section VIII, Unified Pressure Vessel and the Standards of Feedwater Heater Manufacturers Association, Inc. Feedwater Heaters E-4A/B and the bundles for Heaters E-5A/B meet the requirements of ASME Section VIII Division I and the requirements of the Heat Exchange Institute (HEI) Standards for Closed Feedwater Heaters.

Hydrazine is added to the feedwater for oxygen control, and morpholine is used to maintain the pH at the optimum value for the materials of construction for the system.

Historical data removed - To review the exact wording please refer to 10.4.7 of the FSAR.

In the event of a trip of a single feedwater pump, power demand will be automatically run back to approximately 40% of unit load demand. When reactor power is greater than 80%, ICS will subtract a bias from the reactor demand signal. The magnitude of this bias is a function of reactor power, and it varies from a 30% subtraction at 100% power to 0% at 80% power. This will cause an immediate insertion of rods and crosslimit feedwater demand to increase demand to the remaining feedpump. The main feedwater block valves will remain open until reactor power drops below 80%. (Also, the pressurizer spray setpoint will be decreased until power drops below 80% to begin anticipatory pressurizer spray.) These features help to limit RCS pressure increase upon a loss of a main feedpump, thereby reducing the probability of a reactor trip due to high RCS pressure. If both feedwater pumps fail, (when reactor power is greater than 10 percent full power) emergency feedwater will be initiated. To improve the ability to isolate feedwater to the steam generators, and to improve operational reliability, the isolation valves for the main feedwater system receive safety grade power.

Loss of both feedwater pumps will also cause an anticipatory safety grade reactor trip and initiate emergency feedwater. These trips are bypassed below a predetermined flux level, typically nominal 10 percent of full power (for exact value see ANO Unit 1 Technical Specification), to allow for normal startup.

Loss of flow in both main feedwater loops will cause a DROPS/AMSAC initiation of the EFIC System, resulting in the actuation of EFW. This trip is bypassed below approximately 45 percent reactor power.

Under normal operating conditions, the spare condensate pump will start on failure of one of the other condensate pumps.

Historical data removed, to review the exact wording please refer to Section 10.4.7 of the FSAR.

10.4.8 EMERGENCY FEEDWATER SYSTEM

The design requirement of the Emergency Feedwater (EFW) System is to provide a minimum flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage, and pump wear. The EFW System is designed to deliver water to the SG(s) within 80 seconds of SG water level dropping to the EFIC low level setpoint.

The EFW System pumps, valves and connecting piping are designed, constructed, and maintained in accordance with Seismic Category 1 requirements.

EFW System initiation is automatic upon any one of the following: loss of both main feedwater pumps; DROPS/AMSAC actuation; loss of all four reactor coolant pumps; low steam generator pressure; ECCS actuation; or, low OTSG level. The automatic initiation on loss of the main feedwater pumps is automatically bypassed when reactor power is less than 10 percent full power and the bypass will be automatically removed when reactor power is greater than 10 percent full power. An annunciator output is provided to annunciate the bypass condition. The automatic initiation of the DROPS/AMSAC on loss of MFW flow is automatically bypassed when reactor power is less than 45 percent of full power. This bypass will be automatically removed when reactor power is greater than 45 percent of full power. In the event of failure of automatic initiating circuitry, the system is capable of manual actuation from the control room. The EFW System automatic initiation is a safety grade system, which utilizes single failure design criteria and safety grade logic.

The EFW System has two pumps; one steam turbine driven (P7A) and one electrically motor driven (P7B). Each pump has the flow capacity to remove heat load equal to 3½ percent full power operation, and is piped up to supply the total required feedwater to either steam generator from either pump.

The EFW pumps are located in the turbine auxiliary building on the 335-foot elevation. The turbine exhausts to the atmosphere at an elevation of 394 feet. A protective wall is provided between the turbine driven and electric motor driven emergency feed pumps. This wall will serve as a missile barrier as well as a shield to prevent grounding of the electric motor driven pump due to water spray from a failure of the turbine driven pump. Since the compartment which houses these pumps contains open door doorways, failure of the pumps by flooding is not considered credible.

There are two safety-grade sources of emergency feedwater for P-7A and P-7B. First, normal pump suction alignment is to the safety grade condensate tank (T-41B). The second assured source is the Service Water System. The non-safety grade CST (T-41) is normally isolated from the EFW System by gate valve CS-275. For extended EFW operation, operators have the ability to align T-41 to the EFW System. Safety grade check valves CS-99 and CS-262, which have passive safety functions, would prevent loss of EFW should the non-safety grade CST and its associated pressure boundaries be breached. This arrangement is made in order to normally supply the best quality water to the steam generators. Each EFW pump suction has supply path from the Condensate Storage Tanks and from the Service Water System. All control valves presently in the system will be maintained to keep the condensate supply separate from the service water supply until such time as service water is required. Failure of the EFW System supply is indicated by alarms for low-low level in condensate tanks T-41B, and low pump suction pressure. The safety grade condensate tank (T-41B) has a 30 minute (minimum) supply of EFW protected by a tornado missile shield wall, giving operators ample time to align the EFW pump suctions to the Service Water System. Low-low level alarm for condensate tank T-41B and the low EFW pump suction pressure alarm are in the control room. In addition, EFW flow indicators which conform to safety-grade requirements are installed in the control room (1CAN048110). There are two EFW flow indicators for each steam generator (one from each pump). There is also a control room annunciator which indicates EFW System actuation. In the event of loss of condensate supply, suction is manually transferred to the Service Water System.

A local mechanical discharge pressure gauge (PI-2811A) is located on the discharge of EFW pump P7A. This pressure gauge will facilitate manual control of turbine speed by monitoring pump discharge pressure.

Manual valves in the EFW System pump suction and others that could interrupt EFW flow are locked in their correct positions for EFW supply to the steam generators. EFW tests and operability demonstration requirements are given in the Technical Specifications and Plant Operating Procedures provide necessary guidance for operational verification and actions with events of loss of OTSG flow. (1CAN127913, 1CAN057924, 1CAN058307)

EFW flow is supplied to the steam generators through piping independent of the main feedwater piping and via a separate feed ring located near the top of each steam generator. Each EFW feed ring has seven feed nozzles around its perimeter.

A flow diagram of the EFW System is provided in Figure 10-1. The capability of this system to adequately perform its function under the single failure criterion is provided in Table 10-1.

10.4.9 STEAM GENERATOR BLOWDOWN SYSTEM

ANO-1 uses an AREVA/Framatome-ANP replacement once-through steam generator with straight tubes. During normal operation the quality of feedwater is high and is very closely controlled, no provision is made or required for steam generator blowdown.

Under certain startup conditions, the steam generator drain line may be used as a blowdown line. In this instance, the blowdown will be routed directly to the condenser.

10.4.10 Common Feed Water System

Design Bases

The Common Feed Water (CFW) system provides a non-safety related backup source of water to either or both Steam Generators (SGs) on either Unit 1 or Unit 2 in the event that the Emergency Feed Water (EFW) system is not available during certain fire scenarios.

System Description

The CFW system has two motor driven pumps (P-805A and P-805B) equipped with minimum and full flow recirculation orifices to provide protection from overheating and to allow for pump testing. The pumps discharge into a common header with branch lines that connect into the EFW piping. Each branch line has: a motor operated valve (MOV) and check valve that are an extension of the EFW system. MOVs CV-2660A and CV-2660B control the injection rate to each Once Through Steam Generator (OTSG) from the CFW pumps. In addition to the CFW function, CV-2660A and CV-2660B provide the containment (Reactor Building) isolation function. The valves are maintained locked closed and de-energized unless the CFW system is needed to feed the OTSGs. The system is designed to allow the valves to be energized from the Control Room through the use of disconnect switches. The CFW piping system in the Auxiliary Building is Seismic Class 1.

The preferred pump suction alignment is from the Unit 1 CST (T-41). Alternately, the suction may be aligned from Unit 2 CSTs (2T-41A and 2T-41B).

Power to each CFW pump motor is supplied normally from 4.16 kV Switchgear A15 (P-805A) and A19 (P-805B). The power sources that are available to supply both A15 and A19 include Unit 1 4.16 kV AC Bus A1 and the “London Line” through 13.8 kV/4.16 KV Transformer X-150.

A15 and A19 feed 4160 V/480 VAC transformers X-155 and X-195 respectively. The output of these transformers supply two automatic transfer switches which allow either transformer to supply either 480 V AC Panel PP51 or PP91. PP51 supplies power to one CFW MOV on Unit 1 and one CFW MOV on Unit 2. Likewise PP91 feeds the opposite CFW MOV on both units.

The CFW control system consists of two main Programmable Logic Cabinets (PLC) cabinets and a termination cabinet located in the Startup Boiler Control Room and a remote Input/Output (I/O) cabinet in the Safety Parameter Display System (SPDS) computer room. The CFW control system monitors and controls CFW parameters and equipment including CFW pumps, MOVs, flow and pressure transmitters, switchgear breakers, and associated system status inputs.

Three Human Machine Interface (HMI) terminals are utilized by the CFW control system; one located in the Unit 1 Control Room, one in the Unit 2 Control Room, and one in the Startup Boiler Control Room. The CFW control system provides a common alarm output to each unit via the SPDS programmable alarms.

Safety Evaluation

The CFW system is a NFPA 805 based fire protection program system shared by ANO Units 1 and 2. The CFW system is non-safety related augmented quality and is not credited to mitigate any accident identified in the SAR.

Tests and Inspections

The CFW system is tested and inspected at a regular frequency to ensure system availability.

Instrumentation

Instrumentation is installed to allow monitoring of the pumps, motors, switchgear, and control system. Indication of system status and alarms are displayed on the HMI terminals in the Control Room and in the Start-up Boiler Control Room. Flow elements are installed to provide indication of pump total flow and flow to each OTSG.

A flow diagram of the CFW System is provided in Figure 10-3.

Table 10-1

SINGLE FAILURE ANALYSIS - EMERGENCY FEEDWATER SYSTEM

Component	Failure	Comments
EFIC Cabinets (C3701, C37-2, C37-3, C37-4)	Failure of one cabinet	All the control functions of the EFIC system will be performed by the other cabinet.
MSIV Relay Cabinets (C186, C187)	Failure of one cabinet	MSIV operation for both steam generators will be controlled by the other cabinet. In the event where isolating the steam generator associated with the failed cabinet is required, this can be accomplished by the operation of valves that are controlled by the other cabinet.
TIE Cabinets (C511, C512, C513, C514)	Failure of one cabinet	All necessary functions will be performed by the redundant EFW train.
EFW Pumps	Electric motor driven pump fails	Emergency feedwater will be provided by the turbine driven pump in order to cooldown the primary system to the Decay Heat Removal System (DHR) initiation conditions. (For calculation details see Calc. Nos. 32-1159704-02 & 03, and 86-1159705-02, 03, & 04).
	Turbine driven pumps fail	This might happen for a number of reasons: turbine failure, steam supply failure, pump failure, piping failure, etc. In any case, emergency feedwater will be provided by the motor driven pump. A protective wall is provided between the turbine-driven and the electric motor driven EFW pumps. This wall will serve as a missile barrier and as a shield to prevent grounding of the electric motor driven pump due to water spray from a failure of the turbine driven pump.

Table 10-1 (continued)

Component	Failure	Comments
EFW Pumps (continued)	Turbine driven pumps fail (continued)	All the control functions of the EFIC system will be performed by the other cabinet. Since the room which houses these pumps contains open doorways, failure of the pumps by flooding is not considered credible.
Bus A3 or its Associated "Red" Diesel Generator.	Loss of all A.C. "Red" power	In the event of both EFW and MSLI trips, emergency feedwater will be supplied to the unaffected steam generator(s)[1] via the turbine driven EFW pump and the D.C. powered control and isolation valves. The assured supply of EFW is the service water loop and the supply valves to the turbine driven pump are supplied by "Green" power sources. The supply of steam to the turbine will be from the unaffected steam generator(s)[1] via the D.C. powered steam admission valves. The MSIV Relay cabinets are D.C. powered, and will operate under this failure mode. In the event of a loop A MSLI, MFW will be isolated via automatic closure of MFW block, low load control, and startup control valves (in lieu of failed MFIV).

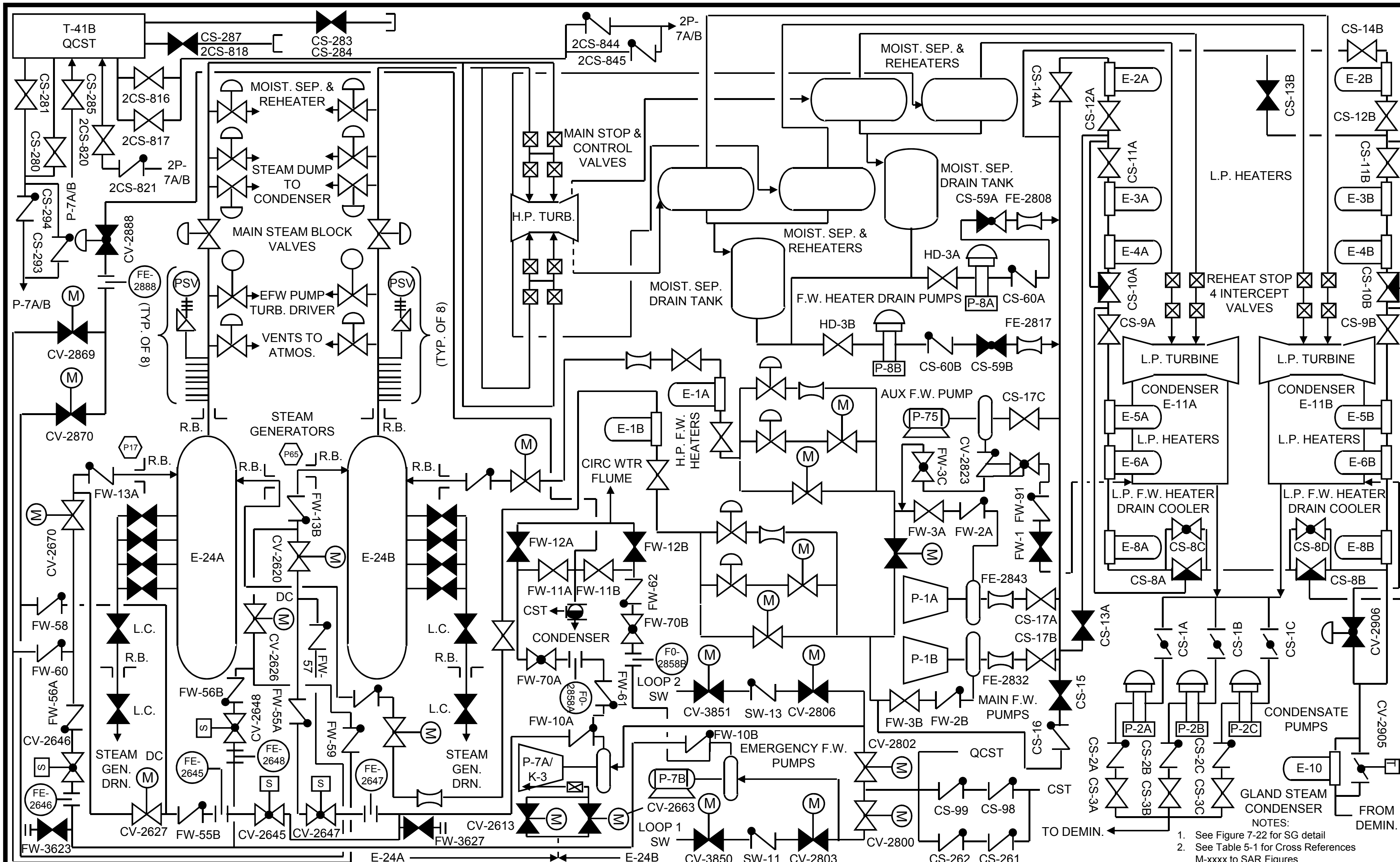
Table 10-1 (continued)

Component	Failure	Comments
Bus A4 or its Associated "Green" Diesel Generator	Loss of all A.C. "Green" power	Emergency feedwater will be supplied to the unaffected steam generator(s)[1] via the "Red" motor driven pump and the DC powered flow control valves. The EFW isolation valves on this train are normally open, and upon loss of the "Green" bus would remain open. The assured supply of EFW to the motor driven pump is the service water loop, and its supply valves are powered by "Red" power sources. The MSIV Relay cabinets are DC powered, and will operate under this failure mode. In the event of a loop B MSLI, MFW will be isolated via automatic closure of MFW block, low load control, and startup control valves (in lieu of failed MFIV).
Service Water Supply to the EFW Pumps	Loss of either the supply valve, or the suction valve or pipe failure, which would lead to loss of service water to the emergency pump.	Each EFW pump is supplied by independent service water loop. The loss of service water supply to one pump would still leave service water supply to the other pump which will provide emergency feedwater as required.
EFW Discharge Piping to System	One of the two series valves in the discharge line from each pump to a steam generator fails to close during steam generator isolation.	The other valve will close provide requisite isolation.
	One of the two series valves fails to open.	The affected steam generator will be supplied by the discharge line from the other EFW pump.

Table 10-1 (continued)

Component	Failure	Comments
Condensate supply from T-41	Loss of tank or piping and valves upstream of parallel check valves CS-99 and CS-262 resulting in loss of suction to EFW pumps.	Condensate will remain available from EFW storage tank T-41B. CS-275 is normally closed; CS-99 and CS-262 have passive safety functions.
Condensate supply T-41B	Loss of inventory function T-41B due to use or by tornado missile above shield wall around T-41B.	Low-low level alarm will sound in control room giving operators a minimum of 30 minutes warning to manually open SW supply valves to EFW pumps.

Note 1: In the event of a steam line break between the steam generator and main steam isolation valves, the affected generator will be isolated and only one steam generator (the unaffected one) will be used for decay heat removal. If the break occurs downstream of the block valves, both steam generators will be used for cooling.



STEAM AND POWER CONVERSION SYSTEM

SAR FIGURE NO. 10-1

ARKANSAS NUCLEAR ONE

UNIT 1

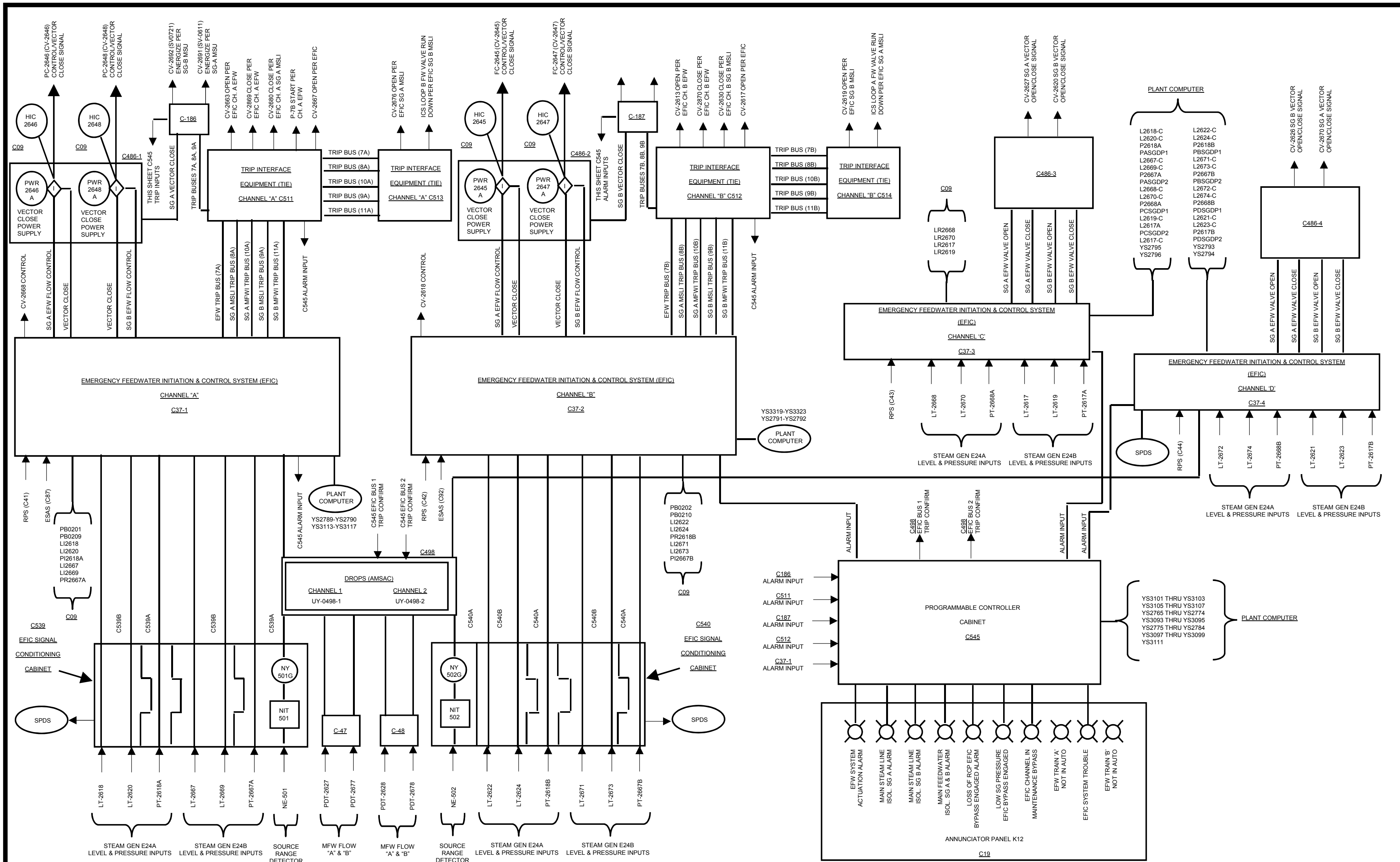
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 26

BASED ON DRAWING NO	SHEET	REV.



EMERGENCY FEEDWATER INITIATION AND CONTROL SIMPLIFIED DIAGRAM

SAR FIGURE NO. 10-2

AMENDMENT 20

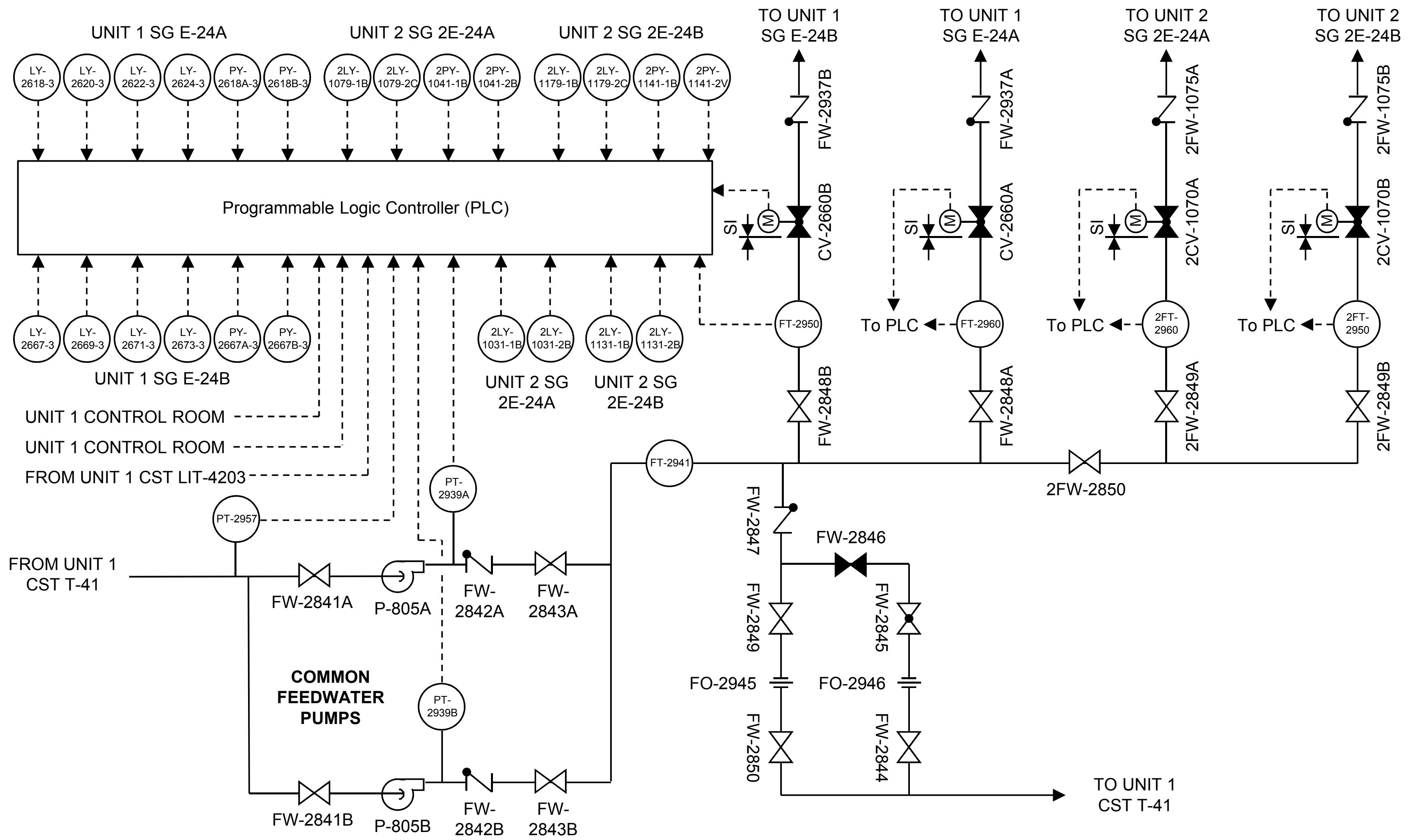
ARKANSAS NUCLEAR ONE

UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	NONE

BASED ON DRAWING NO	SHEET	REV.



COMMON FEEDWATER SYSTEM

SAR FIGURE NO. 10-3

AMENDMENT 28

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

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Unit 1

UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

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11.1.3.1	Design Change Package 452, "Treated Waste Monitor Tank, Addition of Bypass Line Around Tank," 1976. (1DCP760452)
11.1.3.3	Design Change Package 493, "Addition of Air & Water Lance to Spent Resin Storage Tank," 1977. (0DCP770493)
11.1.3.3	Design Change Package 515, "Enlarge Flush Connection Line & Valve from a 1" to 2". Add Y", 1977. (1DCP770515)
11.2.6.2	Design Change Package 1061, "In Containment Radiation Monitors (Bechtel DCP #894 Revision 0)," 1979. (1DCP791061)
11.3.1.1	Correspondence from Trimble, AP&L, to Norelius, NRC, dated May 19, 1981. (1CAN058106)
11.2.2	Correspondence from Marshall, AP&L, to Seidle, NRC, dated April 1, 1983. (0CAN048301)
11.1.3.3.1	Correspondence from Griffin, AP&L, to ES Department Heads, AP&L, dated December 16, 1985. (EDS-85-10)
11.1.3.3.4	Arkansas Nuclear One - Effluent and Waste Disposal Report (Regulatory Guide 1.21) for the period January 1, 1985 to December 31, 1985.
11.1.3.3.6	Arkansas Nuclear One - Unit 1&2 Low Level Radioactive Waste Storage Building Design and Licensing Reports - prepared by NUTECH Engineers, San Jose, California.
11.1.3 Table 11-7	Design Change Package 84-1069, "Hydrogen Recombiner."
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Figure 11-1	Design Change Package 79-1082, "Vacuum Degasifier Replacement."
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Figure 11-3	Design Change Package 79-1066, "Replacement of Diaphram Valves in Gaseous Radwaste System."
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11.2.5.2	Plant Change 90-7041, "Removal of Local Annunciator Horn on the Control Room Area Rad Monitor RE-8001."
Table 11-7	Design Change Package 88-1092, "Removal of Service Water Radiation Monitor Interlocks."
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Figure 11-8	Limited Change Package 89-5001, "Nurse's Station Relocation."

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Section	Cross References
11.1.3.6.2 11.1.3.7.2 11.1.3.7.3	Design Change Package 83-1077, "Hydrogen and Oxygen Waste Gas Analyzers."
11.2.6.1 11.2.6.5 11.3.2	ANO-1 Technical Specifications Amendment 124, dated June 21, 1989. (0CNA068917)
11.2.6.1	ANO-1 Technical Specifications Amendment 147, dated June 21, 1989. (0CNA069115)

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Figure 11-2	Procedure 1104.014, Rev. 14, "Dirty Liquid Waste and Drain Processing."
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Table 11-6	Condition Report 2-89-0053, "Pall-Trinity Filter Data."
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Figure 11-1	Limited Change Package 93-5030, "Installation of Manual Valve RBD-25 for HSD-5-4" Line".

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11.1.3.8.A	Design Change Package 951018D101, "Condenser Tube Bundle and Waterbox Replacement."
11.2.6.3	Procedure 1601.205, "Personnel Dose Assignment."

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11.2.4.1	Condition Report 2-97-0301, "Update of Discussion Regarding Control Room Dose."
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Figure 11-1	Engineering Request 981005N101, "Treated Waste Monitor Tank Filter Installation."

* NOTE: Neutralizing Tank Filtration System is abandoned in place by ER-ANO-2001-0896-002

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Figure 11-2	Engineering Request 980791E101, “Upgrade in Line Class of P-113 Discharge Piping.”
Figure 11-8	Engineering Request 992205E101, “Evaluate the Removal of Various Walls Inside CA-1 for the SGRP.”

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11.1.3.1.3	Engineering Request ER-ANO-2002-0077-000, “Sodium Hydroxide Condensate Flush to Laundry Tanks”

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Figure 11-1	Procedure 1104.020, Revision 04-01-0, “Clean Waste System Operation”
Figure 11-1	Condition Report CR-ANO-1-2002-0959, “Correction to the Line Class Designation on Drawing M-214, Sheet 3”
Figure 11-2	Engineering Request ER-ANO-2000-2477-009, “1R17 RB Cooler Mods for VSF-1 A & B (red train)”
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Table 11-6	Engineering Request ER-ANO-2000-2425-004, “Installation of Stainless Steel Liner in the Containment Building Sump”
Figures 11-4, 11-5, 11-7, 11-9	Condition Report CR-ANO-C-1991-0073, “Upgrade of LBD Change Process”
Figure 11-3	Engineering Request ER-ANO-2003-0857-000, “Re-classification of Spent Fuel Pool Piping 1HCC-10 (1HCC-33)”
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Figure 11-2	Engineering Request ER-ANO-2003-0063-000, "Addition of Zinc Injection System"
11.2.2 11.2.3 Figures – ALL	License Document Change Request 1-1.1-0009, “Deletion/replacement of Excessive Detailed Drawings from the SAR”
11.2.6.3.1 11.2.6.4	License Document Change Request 1-11.2-0034, “Replacement of TLDs with Optically-Simulated Luminescent Dosimeters (OSLDs)”

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11.1.3.3.8	License Document Change Request 1-11.1-0039, "Revision to Low Level Radwaste Building Dose Rate Criteria"
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Table 11-6	Engineering Request ER-ANO-2001-1205-001, “Reactor Building Sump Modifications”
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11.2.6.2.6	License Document Change Request 07-061, "Replacement of Obsolete Continuous Air Monitors (PINGs)”
11.2.6.1 11.2.6.2.5 11.3.2	Condition Report CR-HNQ-2007-0151, “Employee Alignment-Related Organizational Changes”
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11 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.1 RADIOACTIVE WASTES

11.1.1 PERFORMANCE OBJECTIVE

The radioactive waste disposal systems are designed to collect, store, process, monitor and safely dispose all liquids, gases and solids which are potentially radioactive. The controlled release of radioactivity to the environment is in accordance with 10 CFR 20 radioactive effluent Technical Specifications (RETS), offsite dose calculation manual (ODCM), requirements for normal operations and anticipated transient conditions; 10 CFR 100, guidelines for potential reactor accidents; and, the Arkansas Department of Health. The cycle specific information, such as cycle length, is considered historical and is not updated throughout this section. The RETS and ODCM ensure that radioactive waste storage and release is controlled in accordance with applicable regulations.

11.1.2 DESIGN BASES

11.1.2.1 Waste Sources

The major sources of liquid wastes are the expansion of the reactor coolant during heatup of the primary system, and the bleed-off of the reactor coolant to decrease boron concentration as fuel burns up and poisons build up during normal operations.

Other sources of liquid waste are laundry operations, resin bed flushing operations, decontamination operations, laboratory drains, equipment drains, low point drains, and miscellaneous leakage from pumps and valves.

Under certain startup conditions the steam generator drain line may be used as a blowdown line. In this instance the blowdown will be routed directly to the condenser and will be processed with the condensate system. Secondary side liquids and gases can be monitored for radioactive content.

Under normal conditions the once-through steam generator design does not require a blowdown system, and no steam generator blowdown system is installed.

Gaseous activity consists primarily of gases removed from the bleed flow in the vacuum degasifier, and gases purged from various tanks. Gaseous waste also results from the purging of the sampling system, and the venting of various primary and auxiliary systems equipment.

Solid radioactive waste consists of spent demineralizer resins, filter elements, contaminated clothing, contaminated equipment, and paper, rags and plastics used in decontamination and contamination control.

11.1.2.2 Waste Quantities

The following mode of operation was postulated for the equilibrium fuel cycle to conservatively estimate liquid and gaseous waste volumes, activity levels, and activity inventories released through normal station operation.

A. Startup of plant from refueling boron concentration of approximately 2,270 ppm.

B. Startup from cold shutdown condition at 77.5, 155, and 232.5 days in the cycle. Shutdown boron concentration is sufficient to maintain one percent subcritical margin.

C. Equilibrium boron concentration is continuously decreased by bleed and feed to a value of 17 ppm at the end of the cycle.

D. The primary system is drained for maintenance at the 232.5-day and 310-day cold shutdown periods.

The estimated volumes and maximum rates of accumulation of the various forms of radioactive wastes generated in the station are listed in Tables 11-1, 11-2 and 11-3.

11.1.2.3 Waste Activity

Activity accumulation in the Reactor Coolant System (RCS) and associated waste handling equipment has been determined on the basis of fission product leakage through clad defects in one percent of the fuel. The activity levels were computed assuming ultimate core power operation of 2,568 MWt for two core cycles with no defective fuel, followed by operation during the third core cycle with one percent defective fuel. Continuous reactor purification at a rate of one RCS volume per day was used with a zero removal efficiency for krypton, xenon, cesium, molybdenum and yttrium, and a 99-percent removal efficiency for all other nuclides except tellurium. All tellurium is assumed to plate out on system surfaces. Activity reduction by lifetime shim bleed was also considered, since none of the coolant bleed is normally returned to the system. Reuse of reactor coolant is controlled by evaluating each reuse to ensure compatibility with the system.

The quantity of fission products released to the reactor coolant during steady state operation is based on the use of "escape rate coefficients" as determined from experiments involving purposely defective fuel elements (2, 3, 4, 5). Values of the escape rate coefficients used in the calculations are shown in Table 11-4.

Calculations of activity released from the fuel were performed with a digital computer code which solves the differential equations for a 5-member radioactive chain for buildup in the fuel, release to the coolant, removal from the coolant by purification and leakage, and collection on resin or in a holdup tank. The activity levels in the reactor coolant which occur during the third core cycle are shown in Table 11-5. The tritium activity listed is the expected concentration after the third core cycle obtained by adding the activities produced by ternary fission with one percent diffusion through the clad, boron activation by neutrons, and lithium activation by neutrons. All values of tritium activity at any time in the core cycle will be less than or equal to this value.

The liquid waste generated by leakage, sampling, and demineralizer sluice or rinse is assumed to have an activity concentration equal to the concentration in the reactor coolant. Reactor coolant bleed is taken from the downstream side of the purification demineralizer. It is assumed to have the same activity concentration as the reactor coolant reduced by the decontamination factor of the purification demineralizer. Laundry and shower wastes are assumed to contain relatively small amounts of radioactivity. Dirty wastes from floor and area drains, sump discharges, decontamination area drains, and low activity equipment drains are conservatively assumed to have one percent of the maximum coolant activities listed in Table 11-5.

The major part of the gaseous activity is generated by the degasification of radioactive gases from liquids prior to their storage in the clean waste receiver tanks. Therefore, the activity of the gases is dependent upon the liquid activity. The assumptions for liquid activity are described above.

11.1.2.4 Methods of Disposal

Procedures are available and used when their associated equipment or process is used. All of the equipment and processes listed are available for use, but not necessarily used. Due to technological changes in waste monitoring and processing, alternative vendor supplied specifically authorized methods approved by ANO may be used.

Liquid wastes from the station are disposed of, under continuous radiation monitoring and control, in the following ways, depending on the concentration of radioactivity and quantities involved:

A. Collected, processed, sampled, analyzed, and released to the plant discharge canal at a controlled rate.

B. Collected, processed, sampled, analyzed, held up for decay, resampled, reanalyzed, and released to the plant discharge canal at a controlled rate.

Liquid waste effluent is diluted in the discharge canal to concentrations far below those stipulated in 10 CFR 20.

Gaseous wastes are disposed of by:

A. Continuous dilution and discharge through high efficiency filters to the station vent, with sweep gas being drawn through tank voids.

B. Diversion to waste gas holdup tanks for decay, with subsequent sampling and controlled release through high efficiency filters to the station vent.

Gaseous wastes are released from the station so that actual effluent concentrations are far below permissible concentration limits for unrestricted areas at the exclusion area boundary when averaged over a year in accordance with the requirements of 10 CFR 20 RETS, and ODCM. The concentrations at the boundary are determined after applying appropriate dilution factors derived from onsite meteorological studies (See Section 2.3.).

Solid wastes are collected and prepared for either transportation to an off-site disposal facility or temporary storage in the on-site Low Level Radioactive Waste Storage Building (LLRWSB).

11.1.2.5 Turbine Building Trench

An additional source of liquid waste is the Turbine Building Trench due to seepage of radioactive contaminated liquid which collects in the Trench. The Trench collects liquid from a system of floor drains in the Turbine Building. The trench liquid is sampled, analyzed, and released by Nuclear Chemistry. During the release, the liquid is monitored by Turbine Building Drain Radiation Monitor (RE-5641), if operable. Trench releases have auto or manual capabilities. The trench level indication is located at the SE corner of the trench. It has a capacity of 327.25 gallons/inch.

11.1.3 SYSTEM DESIGN AND EVALUATION

11.1.3.1 Liquid Radwaste System

Liquid waste is stored, processed, and monitored according to its source and expected activity.

11.1.3.1.1 Clean Liquid Radioactive Waste System

The liquid processed by this system comes from RCS bleed and drains, reactor coolant auxiliary system reliefs and drains, and the radwaste system reliefs and drains. As such it has the greatest quantity of radioactivity to process. The system is designed to remove fission products, corrosion and wear products, and radioactive gases from the reactor coolant to ensure safe discharges to the environment (see Figure 11-1).

Procedures are available and used when their associated equipment or process is used. All of the equipment and processes listed are available for use, but not necessarily used. Due to technological changes in waste monitoring and processing, alternative vendor supplied specifically authorized methods approved by ANO may be used.

Liquid waste is generated primarily as a result of boron dilution throughout the operating cycle. The letdown flow is passed through the vacuum degasifier to remove hydrogen and fission product gases, then pumped into the clean waste receiver tanks for holdup during decay. As liquid wastes fill any of these tanks, the contents may be mixed by recirculation with the receiver tank recirculation pump and may be sampled for radioactivity. The liquid in the tank is then discharged through a clean waste filter to remove particulate activity, passed through two radwaste demineralizer systems in series to remove soluble activity, and put into the treated waste monitor tank for recirculation and sampling prior to discharge. After analyzing activity in the waste to be discharged, and verifying the proper circulating water flow rate for the waste dilution, desired waste flow rates will be set on the waste effluent control valve. If the waste activity reaches a preset activity level, an alarm will sound and the effluent control valve will be closed automatically. Both waste activity and waste discharge flow rates are indicated and recorded to ensure that proper dilution of discharged waste is being attained.

If, after sampling in the treated waste monitor tank, it is determined that the activity is too high for discharge, the contents of the monitor tanks are returned to the receiver tanks for additional decay time and reprocessing. A bypass system around the treated waste monitor tank can be used to allow continuous cleanup of the clean waste receiver tanks through the radwaste demineralizer.

Normal drainage of reactor coolant from components inside the reactor building passes into the auxiliary building equipment drain tank, via temporary hoses connected to the reactor building drain header, or T-55 reactor building sump by way of temporary hoses into floor drain, and drainage of radioactive liquid from the auxiliary building passes into the auxiliary building equipment drain tank by way of the auxiliary building equipment drain header. Since these liquids have a negligible dissolved gas content, they bypass the degasifier and are pumped directly to the clean waste receiver tanks to be processed as above.

All piping and equipment in contact with reactor coolant is constructed of corrosion-resistant material or lined. This equipment is arranged and located to permit detection and collection of system losses and to prevent escape of any unmonitored radioactive liquid to the environment. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-1. Component data are shown in Table 11-6.

Table 11-18 indicates pressure ratings, materials, design codes and seismic class of the piping and valves.

All waste tanks are vented to the gas collection header to provide for filling and emptying without creating a overpressurization or vacuum. Nitrogen is supplied to each waste collection tank for purging accumulated gases if needed.

With the exception of the devices listed below, all the instruments and controls necessary for the operation of the clean liquid waste system are located on the clean liquid radioactive waste control panel near the equipment.

<u>Exceptions</u>

A. Level indication for the auxiliary building equipment drain tank is also located on the tank.

B. Radiation recording and flow indication equipment for the liquid discharge is located in the control room.

C. Level indication for the system tanks is also located in the control room. However, level indication for the new laundry tank (T109) is local only.

All alarms are annunciated locally with the exception of the high radiation alarm in the circulating water discharge header which sounds in the control room. A common alarm for the clean liquid radwaste system will be annunciated in the control room if critical alarms exist at the local panel.

11.1.3.1.2 Dirty Liquid Radioactive Waste System

The liquid processed by this system comes from various floor and area drains, auxiliary building sump discharge, reactor building sump drain, and drains from auxiliary building equipment of very low expected activity.

Procedures are available and used when their associated equipment or process is used. All of the equipment and processes listed are available for use, but not necessarily used. Due to technological changes in waste monitoring and processing, alternative vendor supplied specifically authorized methods approved by ANO may be used.

All the liquid from this system is collected in the dirty waste drain tank. This tank is divided into two equal sections so that liquid in the section which is full can be processed on a batch basis, while the other section is used for collection. This provides uninterrupted system operation. When full, the liquid is recirculated, mixed, sampled, and pumped through one dirty waste filter where particulates are removed. If additional filtration is required, two filters in series can be used. After filtering, the liquid enters the filtered waste monitor tank for recirculation, sampling, and discharge to the circulating water flume upstream of the circulating water discharge valve. This tank is also divided to provide greater flexibility in waste processing. Should the activity in the tank be too high, means are provided for transferring this dirty waste to the clean waste receiver tanks for additional processing through the clean waste system.

All waste tanks are vented to the gas collection header to provide for filling and emptying without creating a overpressurization or vacuum.

All piping and equipment in this system are constructed of corrosion-resistant material or lined. This equipment is arranged and located to permit detection and collection of system wastes and to prevent escape of any unmonitored radioactive liquid to the environment. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-2. Component data are shown in Table 11-6.

Table 11-16 indicates pressure ratings, materials, design codes and seismic class of the piping.

With the exception of the devices listed below, all the instruments and controls necessary for the operation of the dirty liquid waste system are located on the dirty liquid and laundry radioactive waste control panel near the equipment.

Exceptions

A. Level indication for the system tanks are also located in the control room.

B. Level indication for the auxiliary building sump is also located in the control room.

C. Hand switches for actuating the reactor building sump drain line reactor building isolation valves are located in the control room.

All alarms are annunciated locally. A common alarm for the dirty liquid radioactive waste system will be annunciated in the control room if critical alarms exist at the local panel.

11.1.3.1.3 Laundry Radioactive Waste System

Laundering of contaminated protective clothing is normally accomplished by shipping the radioactive laundry to an authorized off-site processing facility in approved containers. Vendor services are normally employed to clean, repair or replace damaged/worn items, and remove items from service which exceed acceptable radioactivity levels. Installed laundry equipment is available for processing small volumes of reusable towels and other cleaning materials used in the radiologically controlled area. Water used for cleaning floors is also drained to the Laundry Waste Tanks, T-19A/B and T-109 via the floor drain in Room 125. The capability to process large volumes of radioactive laundry on-site, utilizing portable equipment, is available as an alternative method of providing cleaned laundry.

11.1.3.2 Gaseous Radwaste System

The purpose of the system is to: (1) collect and discharge, under monitored conditions, radioactive gases vented from auxiliary system equipment and tanks, and (2) collect, compress, store, and discharge under controlled conditions radioactive gases released from RCS equipment and other primary systems. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-3. Component data are shown in Table 11-6. Table 11-19 indicates pressure ratings, materials, design codes and seismic class of the piping and valves.

Aerated low activity gases are collected in the gas collection header, and are vented directly to the station vent plenum through the gas collection header filter. The sources of these gases are the various equipment vents in the auxiliary building which may contain air. These gases are therefore separated from all the gases associated with primary system water (which contains hydrogen gas) to prevent the formation of an explosive mixture anywhere in the gaseous waste system.

Primary system vents and the discharge of the vacuum degasifier contribute the largest quantity of gaseous radwaste of all the components processed through the system. These gases are normally filtered, monitored, and discharged to the environment. Should the activity in the discharge header be too high, the gaseous waste discharge valve will close and the gas flow diverted to and stored in the surge tank. When a high surge tank pressure signal is generated one compressor starts providing additional compressed storage capability. The gas is compressed and stored in the waste gas decay tanks until the radioactivity level drops sufficiently to be discharged through the discharge header to the environment. The decay tanks are conservatively sized to provide a 30-day holdup time for decay. The two compressors are alternately started by a mechanical sequencer. High surge tank pressure starts one compressor; 30 seconds later the second compressor will start and both compressors will continue to operate until a surge tank pressure signal stops them. Either compressor may be started manually.

Sample lines are provided in each of the gas decay tanks to verify low activity prior to discharge, and the flow through the gaseous waste discharge header is constantly monitored both for activity and flow rate. Nitrogen purging of the gas decay tanks is available to prevent formation of an explosive mixture and to dilute the radioactive gas at discharge.

With the exception of supplementary switches for the actuation of the reactor building vent header, reactor building isolation valves, and the recording equipment for the discharge activity in the gaseous waste discharge header (both located in control room), instruments and controls necessary for the operation of the gaseous waste system are located on the gaseous radioactive waste control panel near the equipment. Alarms are annunciated locally except for the gaseous discharge high radiation alarm which sounds in the control room. A common alarm for the gaseous radioactive waste system will be annunciated in the control room if any alarm exists at the local panel.

11.1.3.3 Solid Radioactive Waste Program

The previously installed Radioactive Waste Concentration and Solidification Systems which were designed to process boric acid and other highly radioactive liquid wastes are obsolete due to technical advances. Therefore use of these systems has been abandoned. Boric acid and other highly radioactive liquid wastes are presently processed similarly to lower level liquid radioactive wastes. The description of the previous system has been replaced with a description of the current Solid Radioactive Waste Program.

11.1.3.3.1 Program Objectives

The purpose of the Solid Radioactive Waste Program is to solidify, stabilize, or encapsulate solid radioactive waste materials for storage, off-site shipment and disposal in accordance with applicable regulations. Solid radioactive waste processing is controlled under a single program at ANO which serves Units 1 and 2. Processed wastes typically include spent ion exchanger resins, used filter elements, and miscellaneous refuse classified as radioactive waste.

Considerations for keeping radiation exposure As Low As Reasonably Achievable (ALARA) are factored into the management of solid radioactive wastes at ANO. These considerations include items such as shielding of waste packages to reduce dose rates, remote operations, use of quick connect fittings, and use of temporary shielding when required.

Another objective of the Solid Radioactive Waste Program is to reduce the volume of radioactive waste generated at ANO. This objective is met by incorporating, practices which:

A. recommend, review and evaluate radioactive waste operations and volume reduction techniques to minimize further radioactive waste volumes,

B. implement policies and procedures which eliminate or minimize the number of contaminated areas within the ANO facility and which also control the amount of material and equipment introduced into contaminated areas,

C. implement waste segregation and sorting procedures,

D. incorporate volume reduction requirements, techniques and practices into the ANO General Employee Training Program.

11.1.3.3.2 Spent Resins

Spent ion exchanger resins are disposed of after being collected and held for decay in spent resin storage tanks. Disposal is accomplished by shipping the resins to an authorized offsite processing facility or disposal facility in approved containers. The capability to solidify spent resins is available as an alternate method of disposal processing. Vendor services may be employed either to dewater or solidify spent resins. All vendor services are performed in accordance with the Radioactive Waste Process Control Program, as specified in the Offsite Dose Calculation Manual. Shipment occurs after the resins have been solidified or dewatered.

11.1.3.3 3 Expended Filter Cartridges

Spent radioactive filters may be removed and transported to the work area by means of a remotely operated shielded cask or by other approved means. The filters are placed in approved shipping containers and shipped to an offsite processing facility or disposal facility in accordance with the Radioactive Waste Process Control Program.

11.1.3.3.4 Dry Active Waste

Low level, or potentially contaminated, wastes are processed in a low level waste work area. Solid wastes such as anti-contamination clothing, rags, paper, gloves, shoe coverings, glove bags and other items are packaged in approved containers. Filled containers are monitored for radiation and may be temporarily stored prior to shipment to an offsite processing facility or disposal facility.

11.1.3.3.5 Equipment Description

Solid radioactive waste processing is not vital to plant protection and has no accident prevention functions. The drumming station, waste storage tanks and permanent waste handling equipment are all located in Seismic Category 1 structures. Portable vendor waste handling equipment is typically staged and operated in the west end of the trainbay which is Seismic Category 1 qualified.

11.1.3.3.6 Expected Solid Waste Quantities

The quantity of solid waste generated and processed at ANO will vary with plant conditions. Operating conditions, maintenance activities, outages, repairs, and installation of new equipment influence the volume of waste generated.

11.1.3.3.7 Packaging and Shipping

Radioactive wastes shipped from the ANO site are packaged and transported in accordance with applicable regulations.

11.1.3.3.8 Storage Facilities

11.1.3.3.8.1 Low Level Radioactive Waste Storage Building

A Low Level Radioactive Waste Storage Building (LLRWSB) is provided consistent with the guidance of Generic Letter (GL) 81-38, “Storage of Low-Level Radioactive Wastes at Reactor Power Sites” and is located northeast of Unit 2, adjacent to the switchyard. The facility is designed to provide a controlled environment for receiving and shipping, inspection, equipment sorting, compaction, and decontamination activities associated with on-site storage and off-site shipment of LLRW (reference Figure 11-6). In addition to the GL 81-38 recommendations, area gamma detectors are provided as an additional personnel protective feature. The LLRWSB is a reinforced concrete structure containing separate storage areas for Dry Active Waste (DAW) and other low activity waste, and High Specific Activity Waste (HSAW). Reusable radioactive material may also be stored in the LLRWSB. A truck bay, a waste segregation and compaction area, and an office area are also included in the building design. The LLRWSB is used to temporarily store packaged waste prior to shipment to an offsite processing facility or disposal facility and other (non-waste) low level radioactive material. The building design includes storage capacity for five years of waste generation.

The DAW and other waste are generally characterized by relatively low activities and the ability to handle by direct contact using appropriate radiological controls. Typically, DAW consists of general contaminated trash items such as protective clothing, laboratory equipment, small tools, mops, brooms, rags, HEPA filters, and other miscellaneous items including various wood, metal, plastic, and rubber objects. Other waste typically consists of absorbed or solidified oil and sludge, damp trash and scintillation vials. HSAW, consisting primarily of filters and resins, is considered LLRW, but is characterized by higher activities than DAW and the necessity of remote handling and are maintained in a locked high radiation area within the LLRWSB, which is surrounded on all sides by concrete shield walls. The HSAW containers are transported to the LLRWSB in shielded transportation casks and remotely moved to a dedicated storage area. The total annual generation of LLRW for ANO is estimated to be approximately 24,000 ft^3.

The overall building design incorporates systems for remote handling of HSAW containers via overhead crane and camera system, handling of DAW and other waste, drainage collection and sampling, fire detection/protection, HVAC, electrical power supply and lighting, radiation detection and decontamination, and building utilities such as water and sewer. A remotely operated overhead crane, equipped with closed circuit television cameras, is used to handle HSAW containers within the LLRWSB. DAW containers are typically handled by use of an electric forklift.

The LLRWSB is designed to permit unrestricted access to the outside of the building at any time (except near the truck bay entrances during handling of resin or filter liners). Dose rates around the LLRWSB should not exceed 2.0 mrem/hr (reference 10 CFR 20.1302(b)(2)(ii)) at the vertical plane of the inner security fence forming the protected area boundary. Dose rates beyond the site boundary should not exceed 0.9 mrem/year. The LLRWSB office area is designed to have dose rates less than 0.5 mrem/hr. Administrative controls associated with the stacking of waste containers near the office area boundary wall may be required to maintain this design objective.

Calculated dose rates and radioactive energies conservatively assumed the HSAW full of resin containers. The Curie content was calculated using the isotopic content (reference Calc 83-0-2228-01) in order to eventually determine expected dose rates. An estimation of facility dose rates is provided in Figure 12.1-13 for informational purposes. The total estimated Curie content in the facility is 1.96E3 Ci. The dose rate at the Exclusion Area Boundary (EAB) per Curie of stored material is estimated at 8.8E-10 mR/hr per Curie. See Table 11-22 for radioactive isotope inventory.

The only potential release of radioactivity would occur during compacting operations; however, this process is not currently used. The LLRWSB is equipped with an HVAC system which maintains a negative pressure within the building. HEPA filters are installed to reduce the release of airborne particulates to the atmosphere. The HVAC system is normally in operation. Radiation Protection personnel perform air sampling based on work activities and the potential for the generation of airborne radioactive contamination.

The radiation monitoring system in the LLRWSB consists of area radiation detectors with local and control room monitors and alarms. The exhaust from the LLRWSB is monitored by 2RX-9850 (SPING-11), which is a SPING-type monitor. The SPING provides on-line monitoring of particulates, iodines and noble gases in the effluent exhaust stream.

Removable filters are provided along with sample ports upstream of the filters permitting plant personnel to obtain representative grab samples of the effluent being released. Sample stream concentrations of up to 10^{-4} µCi/cc of gaseous radioiodine and particulate can be monitored by use of filters and grab samples.

The SPING has the capability to remotely purge the monitor assembly as well as remotely actuating a check source. Readouts of each detector channel and the status of each monitor are available in both control rooms.

Fire protection systems within the LLRWSB include an automatic deluge system for the HSAW area and a portion of the DAW storage area, automatic wet pipe sprinkler systems for the office area, truck bay and remaining DAW storage areas. Other fire protection equipment includes portable fire extinguishers, smoke/heat detectors and alarms, a fire hose station and a nearby fire hydrant.

11.1.3.3.8.2 Original Steam Generator Storage Facilities

The Original Steam Generator Storage Facilities (OSGSFs) are located north of Unit 2 within the Security Owner Controlled Area (SOCA), outside the protected area, and adjacent to the north access road. They are reinforced concrete structures designed for long-term storage of contaminated equipment. The Unit 1 OSGSF consists of two bays, each containing one of the original steam generators (OSGs) with the Unit 1 RCS hot leg elbows stored between them, and a separate compartment housing the original Unit 1 reactor vessel closure head (RVCH), service structure, and Control Rod Drive Mechanisms (CRDMs). The Unit 2 OSGSF contains both OSGs. Based on a response to a Commissioner's comment in SECY-80-511 and NRC memorandum from L. Cunningham/Po Lohaus to directors of the regional headquarters, these components are not treated as radwaste, but contaminated equipment to be retained on site until the plant is decommissioned (reference EC 4372). The facilities are not intended for onsite storage of radwaste materials such as resins, solid radwaste shipping casks, contaminated clothing, or other waste products. Further detail of the Unit 1 OSGSF is included below. For further information related to the Unit 2 OSGSF, refer to Section 11.5.6.2 of the ANO-2 SAR.

The Unit 1 OSGSF is a permanent reinforced concrete and steel structure of approximately 5200 ft^2 with nominal 2-foot thick walls and roof, and a 3-foot thick mat foundation, designed to the Uniform Building Code (UBC). No ventilation or internal utilities are provided. The sides of the building originally contained construction openings to allow access for placement of the OSGs and RVCH. After placement, the openings were closed with pre-cast concrete panels and sealed. With the openings sealed, the OSGSF has no normally open penetrations.

A vestibule is provided at the two personnel entrances to the OSGSF. Water collection sumps are provided in the OSGSF floor slab, one in the OSG bay area and one in the RVCH area. The sump access/monitoring ports are located outside the OSGSF and designed to accommodate checking the collection sumps without entry into the facility and to allow access for radiological survey of the facility sumps. The OSGs were drained and the nozzle openings sealed prior to storage in the OSGSF. Likewise, the RVCH was drained, lead shielding installed at various locations, and a shield plate with gasket bolted to the bottom of the RVCH. Therefore, the normal source of water collected in the sumps, if any, will be condensation. In the unlikely event the sump fills with water and requires draining, the sump can be pumped by inserting a hose through the access port. The OSGSF walls are sealed to a minimum of 361 ft (probably maximum flood). In addition, the entrance into the OSGSF from the vestibule is sealed to a minimum of 361 ft by a removable vertical steel plate.

The OSGSF is a non-safety-related structure. The facility is designed for dead, live, wind, seismic, and flood loads which meet or exceed the Unit 1 SAR requirements for Seismic Category II structures. The applicable recommendations of RG 1.69, Concrete Radiation Shields for Nuclear Power Plants, were used in the design and construction of the OSGSF. The

OSGSF is a stand-alone facility, having no interface with other permanent plant SSCs, onsite or offsite power supplies, and is not equipped with lighting or electrical convenience outlets. The OSGSF location is illustrated in Figure 2-32.

The shielding provided by the concrete walls and roofs is sufficient to keep cumulative external doses external to both the Unit 1 and Unit 2 OSGSFs below the requirements of 10 CFR 20 and 40 CFR 190. The OSGSF has been designed such that the exterior dose rate will not exceed 0.05 mR/hr (roof limit of 0.1 mR/hr). Cumulative dose has been calculated based on a 100% occupancy factor at the nearest occupied building and the EAB. The shielding (building) was designed to ensure ≤ 1 mR/year at the EAB. The calculated dose rate for the OSGSF exterior classifies it as a radiation Zone I area, although the roof is designated as a Zone II area, since the roof will not be accessed by non-plant personnel or visitors to the site and only accessed infrequently by plant personnel. For radiological considerations, both the Unit 1 and Unit 2 OSGSFs are assumed to fail during a seismic event. A design basis event (fire, tornado, seismic event, flood) which causes the simultaneous collapse of the Unit 1 and Unit 2 OSGSF and the subsequent release of contamination from each OSGSF will not result in doses that exceed a small fraction of 10 CFR 100 dose limits for accidental releases, or the maximum predicted dose that a member of the public at the EAB could receive as the result of a waste gas tank rupture as evaluated in the SAR.

Total estimated Curie content for both Unit 1 OSGs in the mausoleum at the time of internment (November, 2005) is (2615 Ci)(2) = 5230 Ci. The estimated total Curie content of the RVCH in the mausoleum at the time of internment (November 2005) is 31.4 Ci. The dose rate at EAB per Curie of stored material is estimated at 8.8 E-10 mR/hr per Curie. Tables 11-20 and 11-21 contain the Curie content with respect to each nuclide for the OSGs and RVCH, respectively. Note that Curie concentrations decrease over time as a result of radioisotope decay. The east vestibule of the OSGSF is designated Radiation Zone III, the west vestibule Radiation Zone I, and the building interior as Radiation Zone IV. Facility design and radiation zoning is illustrated in Figure 11-5.

11.1.3.3.8.3 Old Radwaste Storage Building

A second storage facility is located east of the Unit 1 turbine building and will be used for storage of reusable, contaminated tools. The standard storage container used is a B-25 box. The building is a metal and concrete structure with 6’ high 0.64’ thick concrete shield walls provided along the inside of the north and south walls as necessary to comply with applicable regulations. Building drains are collected in a trench that is directed to a sump located in its north-west region. Based on dimensions, the facility could house up to 1799 B-25 boxes (reference CR-ANO-C-2014-1356 and SAR Figure 11-8). The total estimated Curie content in the facility is 150 Ci. The dose rate at EAB per Curie of stored material is estimated at 1.0E-6 mR/hr per Curie. See Table 11-23 for radioactive isotope inventory.

11.1.3.3.8.4 Pole Barn

The pole barn is a metal building with approximate dimensions of 100’ x 200’ x 50’ which contains a posted radioactive material storage area. The area contains low activity level material that is logistically challenging (size, origin, plan for storage and disposal) to the site. Based on dimensions, the facility could house material such that the total estimated Curie content would be 0.218 Ci (reference CR-ANO-C-2014-1356 and SAR Figure 11-9). The dose rate at EAB per Curie of stored material is estimated at 1.2E-6 mR/hr per Curie. See Table 11-24 for radioactive isotope inventory.

11.1.3.3.8.5 Cask Transfer Facility

A Cask Transfer Facility (CTF) commercial building has been constructed to enclose the CTF structure (where dry fuel storage activities will transfer canister of spent fuel assemblies from a metal transfer storage cask to a concrete storage cask) located on the northeastern part of the plant protected area just northeast of the Low Level Radwaste Storage facility. Radioactive / contaminated dry fuel transfer equipment and tools will be kept and stored in the CTF building. No radwaste will be permanently stored in the CTF building. Storage of the radioactive / contaminated dry fuel equipment and tools will be controlled per ANO site and Entergy Radiological Protection (RP) procedures and ANO dry fuel procedures.

11.1.3.4 Process Radiation Monitoring System

11.1.3.4.1 Description

The process radiation monitoring system is designed to assure that ionizing radiation levels are recorded, indicated and alarmed so that action, either automatic or manual, can be taken to prevent radioactive release from exceeding the limits of 10 CFR 20 and RETS.

Devices are located in the various process systems to monitor radiation levels and annunciate any abnormally high activity. Instrument ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20 and RETS. The plant process monitoring system is capable of meeting the liquid release requirements of Section C.4 of Safety Guide 21, and the gas release requirements of Sections C.1, C.2, and C.3 for releases from the waste gas decay tanks. The reactor building purge releases and main condenser vacuum pump releases comply with this safety guide:

A. The combustible gas control system contains provisions for sampling the reactor building atmosphere prior to purging in accordance with C.1.

B. Sampling capability has been provided for the main condenser vacuum pumps exhaust in accordance with C.1.

C. The exhaust from the main condenser vacuum pumps is routed through the filter in the sample line for the radwaste area to provide iodine and particulate sampling capability for the condenser vacuum pump releases in accordance with C.2 and C.3.

All electronic circuitry (except for photomultipliers and geiger tubes) is solid-state and each circuit has its own DC power supply and provisions for instrument operational testing while in service.

All monitors are supplied with check sources. The check source simulates a radioactive sample and serves as a check for both the readout and detection equipment.

Radiation monitoring equipment panels are located inside the main control room. These panels provide mounting for indicators, recorders, power supplies, and alarms for the radiation monitoring systems. The radiation monitoring panels are fed by the instrument AC bus which, in the event of a loss of normal power, is fed by the diesel generator bus.

The type of detectors used and the information displayed on these panels are listed in Table 11-7. The sensitivity and alarm conditions for each instrument are also listed.

Gases released from the waste gas decay tanks pass through the decay tank effluent control valve which terminates flow from the tank on a high radiation signal. Gases are filtered in the gaseous waste discharge filter, and monitored for flow and activity prior to discharge into the radwaste area ventilation system exhaust ducting upstream of the exhaust filter unit described in Section 9.7.2. As shown in Figures 9-13 and 11-4, the ventilation flow is then sampled prior to release from one of the reactor building flues.

The gaseous waste discharge filter and the gas collection header filter are HEPA-charcoal cartridge type filters. The significant design parameters of each are listed below.

	Gaseous Waste Discharge Filter	Gas Collection Header Filter
Design Code	ASME III Class 3	ASME VIII
Design P/T:	125 psig/150 °F	200 psig/150 °F
Rated Flow:	11 scfm	11 scfm
Filter Element Rating	0.03 micron DOP @ 11 scfm & 0.86" W.G.	Same
HEPA Material	Glass per mil-F-51079A	Same
HEPA Thickness:	3-1/16"	Same
Charcoal Mat'l:	NACAR G617	Same
Charcoal Thickness	2"	Same
Tested Efficiency (mil-std-282)	99.97%	Same

The reactor building purge, radwaste area, fuel handling area, and emergency penetration room exhaust stacks are monitored by means of an isokinetic sampling system. A SPING provides on-line monitoring of particulates, iodines and noble gases in the effluent. The noble gas is monitored by two detectors of different ranges.

The readings taken by the SPINGs are shielded from background radiation effects by integral shielding. Removable filters are provided along with sample ports upstream of the filters permitting plant personnel to obtain representative grab samples of the effluent being released. Sample stream concentrations of up to 10^{-4} μCi/cc of gaseous radioiodine and particulate can be monitored by use of filters and grab samples.

The SPING has the capability to remotely purge the monitor assembly as well as remotely actuating a check source. Readouts of each detector channel and the status of each monitor are available in both control rooms.

11.1.3.4.2 System Evaluation

All process systems which contribute to plant discharges are monitored prior to entering the various discharge systems. Each discharge system is also monitored providing redundancy of radiation detection for plant effluents. The reactor building air, waste gas, and the main condenser air discharge radiation monitoring systems are backed up by the stack gas monitoring system. The service water, radwaste liquids discharge, intermediate cooling, and decay heat radiation monitoring systems are backed up by the liquid discharge monitor system.

Testing and maintenance for all systems and circuit testing of readout equipment and power supplies can be performed from the panels located in the control room. The circuit being tested or repaired indicates and alarms this condition in the control room. Faulty or inoperative circuits during operation are also alarmed and indicated in this manner.

11.1.3.4.3 Reliability

The radiation monitoring system has been designed to the 99 percent statistical confidence level. To release liquid or gaseous waste effluents to the environment, two valves in series must be opened. As these effluents leave the waste disposal system, they are continuously monitored. At a set radioactivity level a signal from the line monitor automatically closes the downstream discharge valve.

Backup monitoring of gases is provided by the station vent monitors. Backup monitoring of liquids is provided by a continuous circulating water discharge monitor. Excessive radiation levels at either of the monitoring stations is alarmed in the control room.

The radioactive gas detectors have been calibrated at the factory prior to shipment and their sensitivity has been determined by the following method: A 2.5 mRem/hr background is established using a Co-60 source and the gross counting rate recorded. From this information the background at 1 Mev is calculated. Two different concentrations of Xe-133 are introduced into a volume of the sampler and each of the resultant average gross count rates recorded. Then based on Minimum Detectable Concentration (= 2.6 BKG @ 1.0 MEV/SEN_{ave}) the sensitivity is calculated. The radioactive liquid detectors use the same procedure except two different concentrations of Cs-137 are used.

11.1.3.5 Non-Monitored Releases

All significant releases of activity from the plant are monitored prior to discharge to the environment. The following releases are not directly monitored:

A. Gland steam condenser exhauster releases.

B. Emergency feedwater pump turbine discharge.

C. Main steam dump valves.

D. Main steam relief valves.

Each of these releases will discharge activity to the environment only if one of the steam generators leaks reactor coolant into the steam system.

In addition, releases from the relief and dump valves are very infrequent, occurring only in the event of unusual operations. Although the emergency feedwater pump turbine is tested monthly, activity releases result only from steam generator leakage, as stated above, and thus are infrequent.

Releases from the gland steam condenser are expected to be insignificant. The total design steam flow to this condenser has been estimated to be approximately 4,600 lb per hour. Since this is only .0413 percent of the rated steam flow, it can be seen that any gaseous releases as the result of tube leaks will be more effectively monitored at the main condenser air ejector discharge.

11.1.3.6 Design Evaluation

All analyses on the liquid and gaseous waste disposal systems were performed on the basis of one percent defective fuel. Although it is not expected that the number of clad defects will ever approach one percent of the total fuel, or that the plant would ever be operated for an extended period with this defect, the object is to demonstrate the capability of safe operation far below the limits specified in 10 CFR 20.

A summary of the various operations considered in the analysis, and the total fraction of MPC in the station effluents from operations with the maximum one percent failed fuel activities is listed in Table 11-8. The activity concentrations resulting are given as fractions of the MPC for unrestricted areas. The concentration of each radioactive nuclide has been divided by its respective maximum permissible concentration as set forth in 10 CFR 20, and the total fraction of MPC obtained by summation over all isotopes considered. The reactor coolant activity and bleed rates used for this calculation are given in Tables 11-5 and 11-8. Dirty waste activity was conservatively estimated to be one percent of reactor coolant activity, and laundry waste activity was neglected.

11.1.3.6.1 Liquid Wastes

The accumulated reactor coolant bleed over a core cycle is approximately 82,000 ft³ including startup expansion and dilution, and let down from boron concentration adjustment. Since there is normally no boron reuse, it was assumed that this total quantity is discharged annually into the circulating water discharge canal.

The maximum short-term waste volume produced has been conservatively estimated to be 16,600 ft^3 normally (initial startup from refueling boron concentration), and 23,000 ft^3 emergency (cold startup at Day 300). The four clean waste receiver tanks (Volume, each = 7,760 ft^3) have reserve capacities of 46.6 percent and 25.8 percent, respectively, for these conditions. The treated waste monitor tanks (Volume, each = 1,970 ft^3) are sized such that these maximum volumes may safely be processed and discharged in 2 to 2.5 days.

It is extremely unlikely that operating conditions would occur such that the volume generated as a result of coolant draining prior to shutdown would require simultaneous storage with the above liquid volumes. However, even if simultaneous storage were required, it could be accommodated by 85 percent of the available storage capacity of the clean waste system. This demonstrates the ability of the system to safely accommodate any foreseen waste volume requirement, as well as to provide extra capacity for liquid storage when desired.

The activity level in the station effluent from feed and bleed was determined by conservatively assuming a purification demineralizer activity reduction factor of 100 (except for those isotopes noted in Section 11.1.2.3), a reduction factor of 1000 for the two radwaste demineralizer systems in series for the same isotopes, a 20-day decay time, and dilution in the discharge canal by a stream flow of 1,700 cfs with four circulating water pumps in operation.

Dirty waste system tanks were conservatively sized to process, on a batch basis, the water volumes generated as a result of draining and flushing operations after refueling (4,300 gal.). This is by far the largest expected single volume to be processed. The dirty waste drain tank and the filtered waste monitor tank each have a capacity of 5,530 gallons, which leaves an approximate 7,300-gallon system storage capability for other draining and flushing operations.

The dirty waste activity was determined by conservatively assuming one percent of equilibrium reactor coolant activity for the maximum one percent failed fuel reactor coolant concentrations listed in Table 11-5. The station effluent activity from dirty waste was determined by using the above and by assuming a 1-day decay time and dilution in the discharge canal by a stream flow of 1,700 cfs. All particulate activity in the auxiliary building and reactor building sumps was assumed removed in the dirty waste filters.

The above analysis demonstrates that the capacity and the processing of the liquid waste disposal system are sufficient to permit flexibility in station operation while providing a means for safe disposal with activity well below the acceptable limits, even in the remote case of one percent failed fuel.

11.1.3.6.2 Gaseous Wastes

Gaseous wastes produced in the operating cycle from degasification of reactor coolant in the vacuum degasifier, venting the pressurizer, venting the reactor coolant makeup tank, and sampling are conservatively estimated to be 25,000 scf. This volume assumes that: (1) all of the hydrogen in solution in the reactor coolant at a concentration of 40 cc/l is totally removed from normal bleed flow in the vacuum degasifier, and from the reactor coolant in the makeup tank during normal system degasifying, and (2) all nitrogen blanketing gas, having practically zero activity, is processed as a potentially harmful gas. By assumption, all of these gases are stored in the waste gas decay tanks for decay prior to release to the environment. The four decay tanks have a total volume of 1,084 ft^3, and an operating pressure of 123 psig, which provides a total gas storage capacity of 10,000 scf. Assuming a 30-day decay period for the gas held in the tanks, and a 15-day release period, this total capacity is sufficient to enable three of the four tanks to process all the waste generated in the postulated operating cycle. Maximum rates of accumulation of waste gases listed in Table 11-2 are easily accommodated by the four tanks, providing a great deal of flexibility in plant operation.

The total average annual concentration at the exclusion distance resulting from the discharge of the waste gas tanks, using an atmospheric dilution in conformance with Section 2.3 (Meteorology) is .0153 mpc for unrestricted areas.

An analysis was made to determine the consequence of reactor operation with a continuous steam generator tube leak of 1 gpm at the maximum one percent failed fuel activities listed in Table 11-5. All the gases and iodine from the 1 gpm steam generator tube leak were assumed to pass directly to the condenser. All the gases and 1/1000 of the iodine were assumed to be removed by the vacuum pumps and released to the plant vent with no radioactive decay.

Using the same atmospheric dilution as above, the annual average concentration at the exclusion distance was computed to be .27 mpc for unrestricted areas.

An analysis was also done to evaluate the effects of periodic purging of the containment. Assuming maximum one percent failed fuel coolant concentrations, a continuous leak of .1 gpm, purging every 90 days, the release of all inert gases and 25 percent of the iodine into the containment, and charcoal filtering of the purge flow resulted in a concentration of .00266 mpc at the exclusion distance.

This evaluation demonstrates that the gaseous waste system is capable of storing and processing the conservatively estimated maximum gaseous waste volumes generated per operating cycle and safely discharging them with activities far below the requirements of 10 CFR 20.

Potentially contaminated gases are collected in the gas collection header, passed through the gas collection header filter and are discharged into the radwaste area ventilation ducting upstream of the radwaste area ventilation discharge filter (Figure 9-13). The gas collection header filter is a HEPA-charcoal cartridge type filter. The radwaste area filter is described in Section 9.7.2.

Because of the nature and source of the waste gases released to the gas collection header, it is improbable that sufficient activity could be released by this mode to produce a condition which would require an alarm. For this reason, no instrument is provided for direct monitoring of these header releases, and no provisions have been made for termination of these releases in the event of high activity. Indirect monitoring of all potentially radioactive ventilation system releases is provided as shown in Figure 9-13.

11.1.3.6.2.1 Sources of Potentially Radioactive Gas Releases From the Collection Header

A. Pump, heat exchanger, and filter vents from reactor coolant auxiliary systems which have the possibility of containing radioactive gases: These vents will be used very infrequently during startup and maintenance operations and are expected to release little, if any, radioactive gas.

B. Dirty liquid and laundry radioactive waste system tank vents: Little, if any, gaseous activity is expected to accumulate in these tanks and be released to the gas collection header. Most of the wastes treated by the dirty waste system come from floor drains, decontamination operations, resin sluicing, and infrequent draining of auxiliary systems equipment for maintenance.

C. Clean liquid radioactive waste system tank vents: Although these tanks provide the only sources of gaseous release to the gas collection header, these releases are expected to be small. The auxiliary building equipment drain tank is used primarily for draining of the RCS components after degassing for refueling and maintenance. Since the liquid wastes entering this tank are at ambient temperatures, and the release of gases while in the tank are governed by molecular diffusion, and the "stay time" in the tank is small, the gas releases from this tank to the gas collection header are expected to be small, if not insignificant. The clean waste receiver tanks and the treated waste monitor tanks are located downstream of the vacuum degasifier and are used for the accumulation and monitoring of bleed wastes from the RCS. Since the vacuum degasifier is expected to have a df = 400 for inert gases and the system is provided with redundant vacuum pumps and drain pumps (which makes bypassing of the degasifier unlikely), the concentration of inert gases in the liquid entering these tanks will be a small fraction of the concentrations of these gases in the RCS. Since the temperature of the liquid in the tanks is low (120 °F) and the release of gases to the gas collection header is governed by molecular diffusion, the annual release of gases from these tanks is expected to be small.

11.1.3.7 Maximum Storage and Discharge Activities

Listed in Tables 11-9, 11-10, and 11-11 are the maximum expected activity levels in the radwaste system decay tanks and the maximum expected annual discharges from the unit. The reactor coolant activity concentrations from which these values were calculated are in accordance with the assumptions stated in Section 11.1.2.3, and listed in Table 11-5. In addition, no credit was taken for decay while filling or discharging the tanks.

11.1.3.7.1 Clean Waste Receiver Tanks

Each of the four clean waste receiver tanks has a capacity of 58,000 gallons. Assuming one percent defective fuel and that the letdown flow from the RCS bypasses the purification demineralizers, the activity concentrations of the liquid will be those listed in Table 11-5. On this basis the maximum quantity of each isotope that would be contained in a receiver tank is given in Table 11-9, Column I. Column II lists the maximum quantity of each isotope in a tank with the purification demineralizer in use.

11.1.3.7.2 Waste Gas Decay Tanks

Each of the four waste gas decay tanks has a capacity of 271 ft^3. Waste gas is stored in the tanks at a maximum pressure of 123 psig, which gives each tank a 2,500 scf storage capacity. The reactor coolant being let down contains up to 55 cc/liter hydrogen and it is conservatively assumed that all of this gas, including the xenon and krypton nuclides, will be stripped off in the vacuum degasifier and pumped to a decay tank. Using the reactor coolant concentrations as listed in Table 11-5, the maximum quantity of each isotope that would be contained in a gas decay tank is given in Table 11-10A.

11.1.3.7.3 Maximum Design Annual Discharges

The annual discharges listed in Table 11-11 are based on the maximum activity values listed in Table 11-5. The clean liquid waste activities take into account cleanup through both the purification and radwaste demineralizers and a 20-day holdup for decay in the clean waste receiver tanks. Dirty waste activities are conservatively estimated to be one percent of the clean waste activities and corrosion product activity is assumed negligible and not included in the table. For an estimate of corrosion product releases, see Table F.4-4 of Appendix F of the Environmental Report.

Based on an operating cycle involving no reactor shutdowns and no reclaiming of liquid wastes, it is expected that the tritium activity in the reactor coolant will build up to a maximum of .343 microcuries per cubic centimeter after three core cycles assuming one percent ternary fission diffusion through the fuel cladding. It is assumed that no decay or dilution of this activity concentration occurs until the liquid waste is discharged into the circulating water discharge. At this point based on full circulating water flow the tritium activity will be diluted to 6.23×10^{-7} microcuries per cubic centimeter based on an annual average. This concentration is far below the maximum permissible concentration of 10 CFR 20. Any reuse of reactor coolant is monitored to ensure that the reactor coolant tritium level does not exceed 0.343 microcuries per milliliter.

11.1.3.8 Tritium Considerations

All tanks in the liquid and gaseous radioactive waste systems which discharge to the environment will be sampled and analyzed for tritium using a liquid scintillation counter prior to discharge.

A. Uncertainties in the amount of tritium generated may be attributable to uncertainties which may exist in the basic nuclear data, the parameters (such as neutron fluxes and flux spectrum) used to calculate the tritium production rate, the model used to account for the changing reactor coolant boron concentration, and the assumed amount of the tritium produced by ternary fission that reaches the reactor coolant. The calculational

uncertainty in the amount of tritium produced in the fuel by ternary fission is estimated to be less than a factor of 1.3. The calculational uncertainty in the amount of tritium produced by boron and lithium activation in the reactor coolant is estimated to be less than a factor of 2.8.

With 766,000 gpm of circulating water available, the plant could discharge more than 7,600 curies of tritium per year and remain within the limits in 10 CFR 50, Appendix I. Since the expected tritium discharge per year is 503 curies (Section F.4.12, Supplement 1 to the Environmental Report), the uncertainty in estimating the amount of tritium produced in the RCS could be as high as a factor of 15 and the plant would not exceed allowable discharge limits. The ANO-1 Condenser Replacement resulted in a 2.4% increase in circulating water flow rate. This increase in flow rate provides more dilution water for radiological releases. Therefore, the flow rate stated above is bounding and conservative.

B. During refueling, with the RCS in a degassed condition, the only possible hazard to refueling personnel was originally expected to be from the evaporation of refueling canal and fuel pool water with the concurrent release of tritium to the reactor building and auxiliary building atmospheres. However, analysis has shown that this is not the case. The fuel handling area ventilation system will be in use throughout the refueling operation and the reactor building purge system will be either operable or isolated, as required by Technical Specifications.

C. Due to the fact that Unit 1 utilized a zirconium based clad and waste water will not normally be recycled, it is not expected that reactor coolant tritium levels will ever reach levels requiring dilution prior to pulling the reactor head. Assuming one percent clad failure and taking into account tritium production from boron and lithium activation, tritium concentrations in the reactor coolant are not expected to exceed .343 μc/ml. Any reuse of reactor coolant is monitored to ensure that maximum tritium concentrations in the reactor coolant do not exceed this value.

D. The air above the spent fuel pool will not be monitored for tritium unless tritium activities in the spent fuel pool indicate that a significant exposure from the evaporation of water could result to refueling and operating personnel. Should measurement of tritium levels over the fuel pool be required, a sample will be collected and the sample analyzed in the lab for tritium activity.

11.1.4 WASTE DISPOSAL SYSTEM FAILURES

The possibility of a significant activity release off the site from accidents in either the solid or the liquid waste disposal equipment is extremely remote. The installed plant systems are located in shielded, controlled-access areas with provisions for maintaining contamination control in the event of spills or leakage. Accidental release from monitoring tanks is unlikely, since to initiate discharge, pumps must be started manually and valves must be set manually such that the radiation and flow sensing circuitry will not close the discharge valve due to insufficient dilution. Loss of either air or power to the discharge valve causes closure. Should equipment fail, means are provided to contain or process activity as shown in Table 11-12.

Inadvertent discharge of activity from the gaseous waste system during normal operations is unlikely, since to initiate discharge from the waste gas decay tanks, a manual discharge valve at the tank and the decay tank effluent control valve must be manually opened from the gaseous

radwaste control panel. Each of the control valves in the system will fail closed on loss of air or electrical power. The effect of the failure of a waste gas decay tank has been conservatively evaluated and the released activity found to be effectively controlled; the area surrounding the waste gas decay tanks is ventilated and discharges to the atmosphere through the plant vent where sufficient dilution occurs such as to not exceed 10 CFR 20 at the exclusion distance. The effects of failure of other equipment are also tabulated in Table 11-12. Inadvertent discharge of gases which are discharged to the environment is expected to be detected by monitoring as described in 11.1.3.2, with alarms to alert Operations personnel so that the release can be identified and terminated.

11.2 NUCLEAR RADIATION PROTECTION

11.2.1 DESIGN BASIS

11.2.1.1 Radiation Exposure of Individuals

The basis for the radiation shielding design for normal plant operation is the "Code of Federal Regulations," Title 10, Chapter 1, Part 20, entitled "Standards for Protection Against Radiation."

Allowable design dose rates for all controlled access areas of the plant are a maximum whole body dose of 1.25 rem per calendar quarter. For all areas outside the plant, the allowable dose rate is not more than 0.5 rem for one calendar year.

All areas of the plant are subject to these regulations. The areas are zoned according to their expected occupancy by plant personnel and their designed radiation exposure level under normal operating conditions.

A program is in place which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program includes the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment

For the Design Basis Accident (DBA) the plant shielding design is based on the Code of Federal Regulations, Title 10, Chapter 1, Part 100 Reactor Site Criteria."

The exposure of individuals to concentrations of radioactive materials in air or water, above the contributions from natural background, is limited to values in Appendix B, of 10 CFR 20.

11.2.1.2 Radiation Exposure of Materials and Components

Materials and components are selected on the basis that their design radiation exposure will not cause significant changes in their physical properties which adversely affect operation of equipment during the design life of the plant. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a DBA. The approximate radiation damage threshold for various materials is shown in Table 11-13.

11.2.2 RADIATION ZONING AND ACCESS CONTROL

The areas inside the plant structures, as well as the general plant yard areas, are all identified by individual radiation zones.

Table 11-14 gives the individual radiation zones which are used for Arkansas Nuclear One Unit 1 (ANO-1).

Areas within the radiological controlled area are posted and controlled in accordance with 10 CFR 20 and the technical specifications.

11.2.3 GENERAL SHIELDING DESIGN CONSIDERATIONS

The shielding is designed to ensure that during normal operations the radiation dose to operating personnel is within 10 CFR 20 limits, and to insure that operating personnel are adequately protected in the event of a reactor accident so that the accident can be terminated without undue hazard to the general public.

The shielding design considers three conditions:

A. Normal full power operation.

   This also includes shielding requirements for certain off-normal conditions such as the release of fission products from leaking fuel elements.

B. Shutdown.

   This condition deals mainly with the radioactivity from the subcritical reactor core, with radiation from spent fuel bundles during onsite transfer, and with the residual activity in the reactor coolant and neutron-activated materials.

C. Design Basis Accidents.

   Several accidents have been investigated including the maximum hypothetical accident which releases the maximum amount of fission products to the general free space of the reactor building.

A list of referenced publications and computer programs which have been used in the design of the radiation shielding is provided in Section 11.4.

All waste treatment components are shielded to ensure that under the design condition of one percent failed fuel, doses to individuals in the controlled access areas of the plant adjacent to the equipment will not exceed the doses listed in Table 11-14. In most cases, manual valves required for equipment operation are located behind shield walls, and in-place shielding is provided for liquid filters to minimize the dose during element replacement.

The Original Steam Generator Storage Facilities (OSGSFs) are located north of Unit 2, outside the protected area and adjacent to the north access road. They are reinforced concrete structures designed for long-term storage of contaminated equipment (i.e., the steam generators originally stored in Units 1 and 2, the Unit 1 RCS hot leg elbows, and the reactor vessel closure head originally installed in Unit 1) and are not intended for onsite storage of radwaste materials. The shielding provided by the concrete walls and roofs is sufficient to keep doses external to the OSGSFs below the requirements of 10 CFR 20 and 40 CFR 190.

11.2.3.1 <u>Specific Design Values</u>

The materials used for most of the plant shielding is ordinary concrete with an average bulk density of 147 lbs/ft^3. Wherever cast-in-place concrete has been replaced by concrete blocks the design assures protection on an equivalent shielding basis. Only in a very few instances have steel or water been utilized as primary shielding materials.

Normal full power operation design conditions assume that the core is operating with an average power density of 83.3 watts/cc and a power level of 2,568 MWt. At this power level the N-16 coolant activity leaving the pressure vessel is 4.0 x 10^7 MeV/cc-sec. Reactor water fission product concentrations were assumed to be the maximum values expected with one percent failed fuel (see Table 11-5).

These reactor coolant activities yield maximum shielding conditions in the areas housing the demineralizers, filters and other associated equipment.

The shutdown condition assumes that the reactor core has been operating at 2,568 MWt for at least 1,000 hours. At 1,000 hours the fission product inventory approaches the infinite operation case.

11.2.4 SHIELDING DESIGN DESCRIPTIONS

The different areas of radiation protection are listed by specific location or building for convenience.

11.2.4.1 Main Control Room

The original DBA defining the protection required for the plant main control room was the Maximum Hypothetical Accident (MHA). ANO-1 has since completed alternate source term dose analyses using the guidance of NRC Regulatory Guide 1.183. These analyses show the dose to the control room operators following any postulated event will be below the 5 rem TEDE acceptance criteria of 10 CFR 50.67. Except as noted, the following discussion continues to describe the original MHA analysis and is retained for historical purposes. The main control room design was based on the airborne fission product inventory in the reactor building following an MHA. Containment and control room shielding have been designed such that the doses to operating personnel for the duration of the MHA are less than 5 rem whole body or its equivalent to any part of the body.

The dose to personnel occupying the control room continuously for 30 days from the onset of the accident derives primarily from two sources:

A. Direct radiation from radioactive fission products inside the containment.

B. Radiation from the cloud formed by the leakage of fission products out of the containment.

The dose computed for direct containment shine to the control room personnel makes several assumptions:

A. The reactor has been operating for a long time such that the fission product inventory is the saturation inventory given by $q_s = (0.865 \times 10^6)(P_0)(V_0)$ Curies

where:

q_s = the saturation inventory of isotope i

P_0 = reactor power (Mwt) = 2,568 Mwt

V_0 = fission yield for isotope i

B. One hundred percent of the noble gases, 50 percent of the halogens and one percent of the solid fission products are released to the containment.

C. The isotopes in "B" above are uniformly distributed in the containment, taken to be a cylinder with free volume of 1.9×10^6 ft^3. No credit is taken for shielding by the internal structures in the containment. Credit is taken for the 3-foot, 9-inch containment wall.

D. No credit is taken for containment leakage, plateout of iodines or effectiveness of the containment spray system in removing fission products from the containment atmosphere. The only decrease in source strength is decay.

The 30-day integrated dose from containment shine is 210 mrem. The 30-day integrated containment shine dose calculated using alternate source terms is 80 mrem following a LOCA.

In determining the dose from cloud shine all of the above assumptions apply except that the containment leakage is conservatively assumed to be 0.2%/day for the duration of the accident.

The leakage is taken to be at the control room roof level and passed directly over the control room, continuously for 30 days. The dose to personnel from cloud shine is 950 mrem. The 30-day integrated cloud shine dose calculated using alternate source terms is 48 mrem following a LOCA.

The total 30-day dose from containment shine and cloud shine is 1160 mrem. The total 30-day integrated dose from containment shine and cloud shine using alternate source term is 128 mrem.

The Emergency Air Conditioning and Filtration Systems provided for the control room are described in Section 9.7.2.

11.2.4.2 Reactor Building Shell

The reactor building shell is a reinforced prestressed concrete structure which serves two main shielding purposes:

A. During normal operation, it shields the surrounding plant structures and yard areas from radiation originating at the reactor vessel and the primary loop components. Together with additional shielding inside the containment, the concrete shell will reduce radiation levels outside the shell to below 1.0 mrem/hr in those areas which are occupied by personnel either on a permanent or routine basis.

B. In the event of an accident, the shell shielding will reduce plant and offsite radiation intensities emitted directly from released fission products below levels as defined by: (1) onsite occupancy limits of 5 rem TEDE and, (2) exclusion distance acceptance criteria of 10 CFR 50.67. The concrete roof of the reactor building has been specifically designed to reduce radiation contributions from sky-shine. Activities inside the reactor building following an accident and the off-site doses associated with the accident are given in Chapter 14.

11.2.4.3 Reactor Building Interior

During reactor operation, access to most areas inside the containment will be prohibited due to high radiation levels.

Large sections over the steam generator compartments and the refueling canal are open and unshielded. These openings cause a high dose rate at the refueling floor during reactor operation. Neutrons streaming out of these areas increase the containment internal dose rate. The reactor vessel which is the major radiation source is surrounded by a concrete shield.

A concrete shield surrounds the steam generators, reactor coolant pumps, pressurizer and associated piping. The shield supplements the shield surrounding the reactor vessel by further attenuating neutrons that escape that shield. It also attenuates high energy gamma radiation from the reactor coolant system allowing limited access in containment during full power operation.

After shutdown, most of the containment is accessible for limited time periods; all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from one to 1,000 mrem/hr depending on the location inside the containment (excluding the reactor cavity). These dose rates results from residual fission products, neutron-activated materials, and corrosion products in the RCS.

For the transport of spent fuel elements, concrete shielding provides protection for the areas close to the transfer route of the fuel. Shielding is provided around the reactor internals storage location and the steam generator compartments. This shielding is designed for personnel protection during storage of activated reactor internals and for protection during the refueling operation.

As a general design practice, the reactor internals and RCS have been designed to preclude crevices, cracks, and other local crud traps as a means of minimizing radiation sources during maintenance operations.

The design air flow in the space between the standoff insulation and the reactor cavity shield wall is assumed to be 9,000 cfm, and the thermal neutron flux is 1.8×10^{9} m/cm^{2}-sec. An AR-41 activity of 1.2×10^{-4} μCi/cc would be present in the reactor building due to the activation of this cooling air. This activity would result in a whole body dose rate inside the reactor building of 37 mrem/hr during plant operation. Personnel will enter the reactor building periodically for inspection during plant operation. These entries will generally be for short periods and normal administrative radiological procedures would be observed to minimize exposure.

If the highly enriched Ar-41 air within the standoff insulation were instantaneously released into the cooling stream the incremental dose rate would be 0.94 mrem/hr.

During a normal purge, the large dilution factor and the short isotopic half-life will reduce the contribution of Ar-41 to an insignificant value.

11.2.4.4 Auxiliary Building

All room and areas containing equipment and piping that process radioactive fluids and material are accessible through service corridors which are entered from the access control station by way of the elevator or stairs. For normal equipment operation, none of the areas or rooms containing radioactive equipment are designed to be continuously occupied. Manually operated valves, necessary for plant operation, are provided with remote operation capability. Gauges and other instruments which require routine visual observation are located in the access corridors or are capable of remote observation in the main control room. Plant personnel are thus able to perform duties necessary for normal operation of the plant while maintaining their radiation exposures as low as reasonably achievable.

Most equipment and systems are located in individually shielded compartments so they maybe isolated for maintenance or repair without significant radiation exposure from other systems or equipment.

During operation, the major radiation sources are the tanks, pumps, and piping containing contaminated drainage. The concentrations of radioactive fission and activation products in these systems are expected to be generally of the low to medium type (10^{-3} μCi/cc to 10 μCi/cc). Concrete shielding walls provide protection for the adjacent basement operation areas. Concrete shielding is provided around the waste gas decay tanks.

Figure 9-3 shows the routing of the reactor coolant letdown line into the purification system in the auxiliary building. The N-16 concentration in this line is .483 μCi/cc at the reactor building penetration, decreasing to 4.2×10^{-2} μCi/cc at the purification demineralizers, 5.7×10^{-3} μCi/cc at the primary system makeup water filters, and 1.6×10^{-3} μCi/cc at the reactor coolant makeup tank. These values represent the maximum possible N-16 activities in this line. Shielding of the letdown line in the auxiliary building is based on the above values. There is no N-16 in any piping or equipment outside of the reactor building except that associated with the letdown line.

The spent resin holdup tank compartment is normally not accessible. All filters and demineralizers are shielded by concrete to reduce the radiation dose rates in adjacent areas.

Other equipment compartments contain volume control equipment, treated and filtered waste, laundry and radiation chemistry drain tanks and pumps, radwaste fan and filters, etc. A special area is designated and designed for the decontamination of equipment. This decontamination room can be sealed off by doors in order to prevent the spreading of contamination during the cleaning operations.

After shutdown of the plant, the Decay Heat Removal (DHR) system heat exchangers will become radiation sources. Fission product activity and neutron-activated corrosion products present in the reactor coolant are the radiation emitters. For these reasons this equipment has been located in shielded compartments.

The transfer of spent fuel is the primary source of radiation during periods of refueling. Because of the relative closeness of fully accessible and uncontrolled areas heavy shielding has been provided for the areas surrounding the spent fuel pool.

11.2.4.5 Administration Building

The dose rate from all sources to general areas in the administration building is conservatively estimated to be a maximum of 6.5 mrem/yr. The two primary contributors to this dose rate are N-16 gamma radiation from the RCS components and the noble gas releases from the expected 0.1 percent failed fuel. Other sources are negligible.

No occupancy factors are applied. The above doses assume 365-day per year and 24-hour per day exposure with the plant operating continuously at 100 percent power.

11.2.4.6 Guard Area

The guard area is in the Administration Area. For more information, see the Security Plan.

11.2.4.7 OSGSF Shielding

The shielding for the Original Steam Generator Storage Facilities (OSGSFs) is designed to keep the dose rate outside of the OSGSFs less than 1 mrem/hr. The OSGSFs are further discussed in SAR Section 11.1.3.3.8.2 and Section 11.5.6.2 of the ANO-2 SAR. Radiation zoning is illustrated in SAR Figure 11-5 (ANO-2 SAR Figure 12.1-15).

11.2.4.8 Outside Plant Areas

The radiation fields at all areas outside the auxiliary building and the containment are much the same during both operation and shutdown. For each of these cases, the dose rate at the building outside surface is normally less than 1 mrem/hr. One exception is the Old Radwaste Storage Building if assumed to be loaded to full capacity; however, normal radioactive material stored is of less curie content than that assumed in calculations and area loading is minimal in comparison to that which the facility is capable of storing. Refer to SAR Section 11.1.3.3.8.3 for further radiological information.

11.2.4.9 LLRWSB

A Low Level Radioactive Waste Storage Building (LLRWSB) provides a controlled environment for receiving and shipping, inspection, equipment sorting, compaction, and decontamination activities associated with on-site storage and off-site shipment of LLRW (see Figures 11-6 and 11-7). HSAW, consisting primarily of filters and resins, is considered LLRW, but is characterized by higher activities than DAW and the necessity of remote handling and are maintained in a locked high radiation area within the LLRWSB, which is surrounded on all sides by concrete shield walls. The LLRWSB is further discussed in SAR Section 11.1.3.3.8.1.

11.2.4.10 ORWSB

The Old Radwaste Storage Building (ORWSB) is primarily used to store reusable radioactive material. The standard storage container used is a B-25 box. The building is a metal and concrete structure with 6' high 0.64' thick concrete shield walls extending along north and south sides (see Figure 11-8). The ORWSB is further discussed in SAR Section 11.1.3.3.8.3.

11.2.4.11 Radwaste and Source Area Shielding

The shielding of the radwaste area includes exterior and interior shield walls. The radwaste area exterior shield walls surround all equipment in the area. The interior shield walls provide shielding for each piece of equipment consistent with its postulated maximum radiation level and with the access requirements of the area and adjacent areas. The sources of radiation in the radwaste area are gases and liquids from the RCS, leakages from other systems, and solids from the spent resins, filters, liquid waste solidification, and shipping container storage area. See Section 11.1 for a discussion of the radwaste systems.

The ANO-2 Auxiliary Building Extension Elev. 335' (east and west concrete racket ball courts) are locked by a chain-link gate and are used for dosimeter irradiations and historically contain a 3 Ci (original activity) Cs-137 source and a 1.2 Ci (original activity) Cs-137 source, respectively. In addition, the ANO-1 Turbine Building Elev. 335' (Pu-Be storage room) is accessed via a locked door and is used to response check neutron instrumentation. This area historically contains a 4.71 Ci (original activity) PuBe source, a 0.72 Ci (original activity) AmBe source, and a 0.120 Ci (original activity) Cs-137 source. The aforementioned areas are posted and controlled in accordance with procedure at all times.

11.2.4.12 Pole Barn

The pole barn is a metal building with approximate dimensions of 100’ x 200’ x 50’ which contains a posted radioactive material storage area. The area contains low activity level material that is logistically challenging (size, origin, plan for storage and disposal) to the site (see Figure 11-9). The Pole Barn is further discussed in SAR Section 11.1.3.3.8.4.

11.2.5 AREA RADIATION MONITORING SYSTEMS

11.2.5.1 Design Basis

The Area Radiation Monitoring System (ARMS) is provided to supplement personnel and area radiation monitoring provisions of the plant health physics program described in Section 11.2.6. Included in this system are 20 permanently located radiation detectors which provided continuous local and remote indication and alarm of direct radiation dose rate. The primary objectives of the ARMS are:

A. To immediately alert plant personnel prior to entering or working in areas of increasing or abnormally high radiation levels, which, if unnoticed might possibly result in inadvertent overexposures;

B. To inform control room operators of the occurrence and approximate location of unusual conditions resulting in the release of radioactive materials or the degradation of shielding structures;

C. To provide, in the event of many types of hypothetical accidents leading to the contamination of the plant, a means of remotely determining external dose rates in those areas most likely to be contaminated, prior to entry by personnel; and

D. To provide a continuous record of external dose rates at selected locations, thereby ensuring detection of transient increases in dose which are attributable to rapid changes in the radioactivity content of equipment and process streams.

The ARMS has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment.

11.2.5.2 System Description

The Area Radiation Monitoring System (ARMS) consists of 20 low range area monitors. Each channel consists of a detector, remote and local alarms, remote and local indicators and a remote power supply. Two high range channels are provided to give an indication of containment radiation levels for post accident conditions in compliance to various regulations and licensing conditions. The containment high range monitors have a range of 1 R/hr to 10^8 R/hr.

The 20 low range ARMS detectors are wall mounted coaxial Geiger Mueller tubes. The associated meters are designed with range and sensitivity suitable for their location. High radiation levels and individual circuit failures are alarmed both visually and audibly on the ARMS panel.

Alarm levels are set to indicate transient radiation level increases. On occasion other alarm points may be selected depending upon measured radiation levels in the area. Some monitors are expected to have different alarm points when the reactor is critical than they have when the reactor is shutdown. Setpoints for warning alarms and high alarms are established in accordance with a plant procedure. Setpoints are adjusted to a value high enough to prevent alarm actuations due to electrical noise deflections, but low enough to detect increasing radiation levels as early as possible. Alarms initiate continuous audible and visual alarms at the detector, except the control room detector which has a visual alarm only. The control room annunciator provides a single window which alarms for any channel detecting high radiation levels. Visual verification of the channel that has alarmed is done at the Area Radiation Monitoring Panel in the main control room.

Table 11-15 lists the locations, sensitivities and ranges of the Area Radiation Monitoring System Detectors. The criterion for selecting these particular locations is to provide a minimum of one ARM on each elevation of the auxiliary building where equipment handling normally radioactive material is located. On each elevation, the monitors are placed in those areas having 40 hour per week occupancy (radiation zone II) so as to be seen and/or heard, upon alarming, from both ingress and points of highest occupancy. Reading from monitors inside containment will permit estimation of personnel occupancy time prior to entry.

11.2.5.3 Testing and Maintenance

Circuit testing of readout equipment and the power source can be performed from the control room. The circuit being tested or repaired is inoperative during this time and acts as a tripped channel. Installed radioactive sources are used to periodically response check the detectors and circuits.

11.2.6 HEALTH PHYSICS PROGRAM

11.2.6.1 Program Objectives

Rules and procedures have been established for protection against radiation and contamination. All personnel assigned to the plant and all visitors are required to follow these rules and procedures.

The ANO Radiation Protection Manual and Health Physics procedures specify administrative controls designed to provide protection of personnel from exposure to radiation in excess of the limits specified in 10CFR20 and further to maintain personnel radiation exposure As Low As Reasonably Achievable (ALARA).

The procedures concerning radiation exposure are subject to the same review and approval process as all other plant procedures. All work in Controlled Access, or other radiation or contamination areas, requires an appropriate Radiological Work Permit (RWP). An RWP must be approved by Health Physics. The radiological hazards associated with the jobs will typically be determined and evaluated prior to approval of the RWPs. The RWP identifies the precautions to be taken, the protective clothing to be worn, and any other radiological safety precautions that are required. In order to control access to radiological areas, warning signs, audible and visual indicators, barricades, and locked doors are used as appropriate.

It is the policy of Entergy Operations, Inc. to keep personnel radiation exposure within the limits set forth in applicable regulations. In addition, Entergy Operations, Inc. is dedicated to the philosophy of ALARA and has established an ALARA Program at the Arkansas Nuclear One Facility. The ALARA program is designed to ensure that each person will keep their radiation exposure as low as is reasonably achievable, consistent with performing assigned work. Each individual is responsible for observing rules adopted for the individual's safety and the safety of others.

Procedures, work plans, Radiological Work Permits, and other documents required for jobs involving substantial radiation hazards are reviewed by the ALARA Committee. Items considered in this review may include the use of remote or special tools, temporary shielding, preliminary job-site decontamination, special ventilation containment areas, respiratory protection, special anti-contamination clothing, continuous air monitoring, or any other procedures or devices that will serve to reduce the dose received during the job performance.

11.2.6.2 Facilities and Equipment

11.2.6.2.1 Controlled Access Facilities

Most of the ANO radiologically controlled areas exist inside the Unit 1 and Unit 2 Reactor and Auxiliary Buildings. (Notable exceptions are the Radioactive Waste Storage Building and the Contaminated Materials Storage Building.) Normal ingress to the Unit 1 and Unit 2 Reactor and Auxiliary Buildings is through a single controlled access portal located on the 386 foot elevation of the Unit 2 side of the plant. Normal egress from these areas is through a similar controlled access portal located on the 386 foot elevation of the Unit 1 side of the plant. The Unit 2 controlled access point contains offices for Health Physics personnel and the ingress control point. The Unit 1 controlled egress point contains a frisking station, personnel decontamination station, and the egress control point.

11.2.6.2.2 Nuclear Chemistry Laboratory Facilities

Nuclear chemistry facilities consist of two conventional chemistry laboratories, a nuclear chemistry laboratory and nuclear chemistry counting room, sample rooms and sample preparation room.

The nuclear chemistry laboratory has been constructed for ease of decontamination. The lab is designed for the preparation and analysis of reactor coolant. There is a stainless steel lined fume hood in the nuclear chemistry laboratory with a designed exhaust flow of 800 cfm. Two exhaust fans are capable of drawing air from the fume hood, nuclear chemistry laboratory, and other areas in the auxiliary building. One fan will start automatically if the other fan stops. If both fans fail, an alarm will sound in the control room. Drains from the nuclear chemistry laboratory including the fume hood, floor drains and sinks are connected to the radioactive liquid collection system.

The counting room contains laboratory equipment needed to implement the Nuclear Chemistry Program.

11.2.6.2.3 Decontamination Facilities

The personnel decontamination station is equipped with a shower, sink, hampers for contaminated clothing, and monitoring equipment for personnel decontamination. Detergents and other cleansing agents along with brushes and towels are also available for this purpose.

Large pieces of equipment can be decontaminated in the cask washdown area on the fuel handling floor, if necessary.

11.2.6.2.4 Medical Facilities

Medical observation and treatment are available in case of overexposure and contamination. Local physicians and hospital facilities are used in the initial care and treatment of injuries received at ANO.

11.2.6.2.5 Portable Health Physics Instruments and Equipment

Available portable instruments are of sufficient variety and function to measure all types of radiation of concern and in all ranges expected during normal and emergency operations. Health Physics instrument procedures are available and training is conducted to assist qualified personnel in the proper selection and use of instrumentation.

Typical types of portable Health Physics survey and monitoring instruments available at ANO include:

A. Battery powered GM count rate meters,

B. High range GM survey meters,

C. AC powered GM count rate meters,

D. Scintillation detectors,

E. High and low range ionization chamber survey meters,

F. Neutron REM meters.

Calibration and maintenance of Health Physics instrumentation is performed in accordance with applicable procedures using National Institute of Standards and Technology traceable sources and equipment.

11.2.6.2.6 Air Monitoring Equipment

Continuous air monitors consist of air samplers capable of continuously sampling and monitoring airborne particulate and iodine radioactivity. The monitors can be set to identify airborne activity levels, alarm locally, and transmit data to a remote station for display and recording. When continuous air monitors indicate airborne activity, portable air samplers equipped with collection media for particulate, radioiodine, and noble gas are used to obtain air samples for analysis.

11.2.6.2.7 Personnel Protective Equipment

Anti-contamination clothing, such as shoe covers, gloves, coveralls, and head covers, is available for use. The protective clothing, respiratory protective equipment and other equipment required for use in contaminated areas is determined by Health Physics and is typically specified on the required Radiological Work Permit.

Respiratory protection equipment used at Arkansas Nuclear One for radiological protection consists of full face respirators, air-line breathing units, and self contained breathing apparatus. Procedures governing the use and maintenance of this equipment comply with the requirements of 10 CFR 20, other applicable regulations, and rules of good practice such as those published by the American Industrial Hygiene Association, "Respiratory Protective Devices Manual."

11.2.6.2.8 Special Equipment and Devices

Temporary shielding, remote and special tools, and mock-ups are used as applicable to reduce personnel radiation exposures. Temporary shielding in various forms, such as bricks, blankets and sheets, is available at ANO. Standard types of remote handling tools are also available. Other remote or special handling tools may be purchased or fabricated as the need arises.

11.2.6.3 Personnel Dosimetry

11.2.6.3.1 External Dosimetry

Personnel monitoring devices such as electronic alarming dosimeters (EADs) and primary dosimeters of legal record (DLR) are used for measuring personnel dose equivalent due to beta-gamma and neutron radiation from external sources. External dosimetry measurements, records, and reporting systems comply with the requirements set forth in 10 CFR 19 and 10 CFR 20.

Dose of legal record from external radiation exposure is based upon primary dosimeter results. Primary dosimeters are processed by a laboratory certified by the National Voluntary Laboratory Accreditation Program (NVLAP) as required by NRC regulations. If valid primary dosimeter results are not available, external dose of record will be assigned in accordance with applicable procedures.

11.2.6.3.2 Internal Dosimetry

Arkansas Nuclear One has the capability to assess the presence of radionuclides and perform in-vivo isotopic analysis. The presence of internally deposited radionuclides is assessed periodically at ANO. The reasons for monitoring include initiation of dose monitoring, assessment of the effectiveness of the respiratory protection program, evaluation of suspected

intakes of radioactive material, and termination of dose monitoring. At ANO this monitoring is performed by personnel monitoring equipment. In-vivo isotopic analyses are normally performed when the assessment indicates internally deposited radionuclides in quantities that exceed the monitor setpoint.

Certain isotopes, such as tritium (H-3) which are difficult to assess in-vivo are assessed by in-vitro analyses. These measurements may be made using Entergy facilities or contracted to vendor services. The need for in-vitro bioassay measurement is evaluated by Health Physics. Any calculated doses, due to internal exposure, are assigned in accordance with the requirements of 10 CFR 20.

11.2.6.4 Procedures

Procedures are an integral portion of any program. Our Radiological Waste Management and Health Physics Programs are no exception. The procedures which are part of these programs deal with many topics. As a minimum procedures are maintained to address the following areas:

A. Solid Radioactive Waste Program

   1. Minimize the number of contaminated areas and reduce the introduction of materials into contaminated areas.

   2. Waste segregation methods and procedures.

B. Health Physics Program

   1. Protection against radiation and contamination.

   2. Use of survey and monitoring instruments.

   3. Calibration of survey and monitoring instruments.

   4. Use and care of respiratory protective equipment.

   5. Assignment of dose received if dosimeter of legal record (DLR) results are not available.

C. Radioactive Materials Safety

   1. Control, use and storage of Special Nuclear Material.

   2. Fuel handling.

   3. Sealed source inventory.

11.3 RADIOACTIVE MATERIALS SAFETY

11.3.1 RADIOACTIVE MATERIALS SAFETY PROGRAM

The Radioactive Materials Safety Program is administered at ANO as a single program for both Unit 1 and Unit 2.

11.3.1.1 Special Nuclear Material

Entergy Operations, Inc. has implemented a program to control the special nuclear material under title to or in possession of the company. The use, storage, and control of special nuclear material is accomplished as specified in the operating license and as specifically implemented in applicable plant procedures. These procedures are part of the company's fuel accountability program and are used in receiving, handling, and storing the fuel assemblies containing the special nuclear material.

Special nuclear material contained in unirradiated fuel assemblies is stored in noncritical arrays in either the new fuel pit or spent fuel pool, both of which are Seismic Class I structures as described in Section 9.6, "Fuel Handling System." Fuel assemblies are inserted into and removed from the reactor according to approved fuel handling procedures using fuel handling equipment as described in Section 9.6.2.2, "Loading and Removing Fuel." Irradiated fuel assemblies are handled and stored underwater, as described in Section 9.6.2.2. Licensed storage or shipping casks will be used to move spent fuel from the spent fuel pool.

11.3.1.2 Other Sealed Sources

Other sealed sources of radioactive material, including specified quantities of special nuclear material, are controlled through the Health Physics Program described in Section 11.2.6. Sealed sources containing a sufficient quantity of radioactive material to create a High Radiation Area are locked in a shield or stored in the shielded position when not in use. The rooms in which these sources are used are posted and locked when the sources are in an unshielded position and left unattended.

11.3.2 RADIOACTIVE MATERIALS CONTROL

Reactor Engineering is responsible for the physical accountability of special nuclear material at ANO. Reactor Engineering is also responsible for providing the storage, movement, and loading sequences to minimize the handling of fuel assemblies. Operations is responsible for the safe movement and handling of fuel assemblies according to written plant procedures. Personnel trained and qualified in the use of fuel handling equipment perform material movement operations. Onsite shift management is responsible for fuel handling operations on the respective shift.

Other sealed sources of radioactive material are controlled through the Health Physics Program. These sealed sources are used primarily for instrument and out-of-core detector calibrations and may typically include materials such as radiocesium, radiocobalt, or plutonium-beryllium of various activities. Sealed sources are accounted for through inventories performed as specified by procedure. Leak testing of sealed sources is performed in accordance with plant Technical Specifications.

11.3.3 RADIOACTIVE MATERIALS

A complete inventory of licensable quantities of sealed radioactive sources is maintained. At present there are no plans to have any radioactive materials on site that exceed the limits specified below:

	Material	Form and Use	Possession Limit
A.	Any byproduct, source and special nuclear material	As reactor fuel; as sealed sources for reactor startup; as sealed sources for reactor instrument and radiation monitoring equipment calibration; and as fission detectors	As required for reactor operation
B.	Any byproduct, source or special nuclear material	Any form for sample analysis or instrument calibration excluding Krypton-85	100 milliCuries each isotope; any by-product material
			100 milligrams each isotope any source or special nuclear material
			*2 Curies Krypton-85 as instrument calibration source

* A maximum of 2 curies of Krypton-85 will be on site for use as an instrument calibration source for both plant process and area radiation monitors. The isotope is kept in pressurized bottles. The maximum total activity for any single bottle is not expected to exceed 950 milliCuries.

11.4 REFERENCES

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List of Computer Codes Used for the Shielding Design

GRACE I - For computing gamma-ray attenuation and heat generation.
GRACE II

FAIM - For computing neutron attenuation and heat generation.
ANISN

GHT - For computing heat conduction problems due to gamma-ray and neutron attenuation.

MCNP4C – A generalized geometry, time-dependent, Monte Carlo transport code that can be used to model complex radiological shielding models.

Review of Shielding Design

An extensive design review of plant shielding and sampling capabilities was performed in 1979. The results of the review, which was conducted by NUS Corporation, are published in report NUS-3504, Design Review of Plant Shielding and Sampling Capabilities in Response to NUREG-0578.

Table 11-1

ESTIMATED RADIOACTIVE WASTE QUANTITIES

Waste Source	Quantity/Year (Ft^3)	Assumptions and Comments
Reactor Coolant System		
Startup Expansion	15,850	4 cold startups
Startup Dilution	22,100	One startup from cold condition at beginning of cycle 77.5, 155 and 232.5 full power days
Lifetime Boron Feed & Bleed	33,500	Dilution from 1070 to 17 ppm boron
System Drain	12,200	2 Drains: on day 232.5 and 310
Liquid Waste		
Sampling and Laboratory Drains	2,930	12 Samples per day at 5 gal per sample
Demineralizer Sluice	650	4 ft^3/ft^3 resin
Laundry	5,200	150 gal/day
Showers	11,700	8 Showers/day 30 gal/shower
Miscellaneous System Leakage	11,700	10 gal/hr
Refueling Canal Draining & Flushing	575	Last 1 ft depth (3300 gal) & 1000 gallon flush
Gaseous Waste		
Off-Gas From Reactor Coolant System	3,500	Degas at 40 cc H_2 per liter
Off-Gas From Liquid Sampling	138	12 Samples per day at 5 gallons per sample & 55 cc H_2 per liter
Make-Up Tank	15,000	Degas coolant prior to shutdown.
Pressurizer	3,600	Venting H_2 once per week from 1.5 gpm spray flow with 40 cc H_2 per liter

Table 11-2

ESTIMATED MAXIMUM RATE OF ACCUMULATION OF RADIOACTIVE WASTES

Waste Source	Max Rate of Accumulation	Assumptions and Comments
Reactor Coolant System		
Startup Expansion & Dilution, ft^3/30 hrs	23,000	Cold startup at Day 300 (last day startup prior to refueling)
Lifetime Shim Bleed, ft^3/10 days	4,950	
System Drain, ft^3/Refueling	6,200	
Liquid Waste		
Demineralizer Sluice, ft^3/Change	368	1 Change of Makeup, Spent Fuel, and Radwaste Demineralizers
Gaseous Waste		
Off-Gas From Reactor Coolant Bleed, scfm	1.05	Maximum Letdown at End of Life
Make-Up Tank, ft^3/Purge	5,000	From degassing prior to Refueling
Pressurizer, ft^3/Purge	72	
Solid Waste		
Demineralizer Resin Change, ft^3/Change	196	1 Change of Makeup, Spent Fuel, and Radwaste Demineralizers

Table 11-3

MAXIMUM ESTIMATED ANNUAL ACCUMULATIONS OF RADIOACTIVE WASTES

Forms of Waste	Accumulations
Liquid, gals	867,000
Gaseous, scf	25,000

Table 11-4

ESCAPE RATE COEFFICIENTS FOR FISSION PRODUCT RELEASE

Element	Escape Rate Coefficient, Sec^{-1}
Xe	1.0×10^{-7}
Kr	1.0×10^{-7}
I	2.0×10^{-8}
Br	2.0×10^{-8}
Cs	2.0×10^{-8}
Rb	2.0×10^{-8}
Mo	4.0×10^{-9}
Te	4.0×10^{-9}
Sr	2.0×10^{-10}
Ba	2.0×10^{-10}
Zr	1.0×10^{-11}
Ce and other rare earths	1.0×10^{-11}

Table 11-5

REACTOR COOLANT ACTIVITY

(Predicted Reactor Coolant Activity for the Third Core Cycle, Based on 1% Defective Fuel Elements)

		Time, Full Power Days, (µCi/ml)						
Isotope	Half Life	100	200	260	280	290	300	310
Kr 85m*	4.4h	1.5	--					1.5
Kr 85	10.4y	9.8	9.7	6.3	4.5	3.3	2.1	1.1
Kr 87	78.m	.84	--					.84
Kr 88	2.8h	2.7	--					2.7
Rb 88	18.m	2.7	--					2.7
Sr 89	54.d	.04l	.041	.040	.040	.039	.038	.038
Sr 90	28.y	.0029	.0033	.0034	.0034	.0034	.0034	.0033
Sr 91	9.7h	.046	.046	.046	.046	.046	.045	.045
Sr 92	2.7h	.017	--					.017
Xe 131m	12.d	2.0	1.9	1.7	1.5	1.4	1.1	.92
Xe l33m	2.3d	2.7	2.7	2.6	2.5	2.4	2.3	2.1
Xe 133	5.27d	243	238	227	211	205	173	154
Xe 135m	15.m	.94	.94	.94	.94	.93	.92	.92
Xe 135	9.2h	5.6	5.6	5.6	5.5	5.5	5.4	5.3
Xe 138	17.m	.51	--					.51
I 131	8.05d	3.2	3.2	3.2	3.2	3.1	3.0	3.0
I 132	2.3h	4.7	4.7	4.6	4.4	4.2	3.9	3.6
I 133	21.h	3.8	3.8	3.8	3.8	3.7	3.6	3.6
I 134	52.m	.50	--					.50
I 135	6.7h	2.7	2.7	2.7	2.7	2.6	2.6	2.6
Cs 136	13.d	.76	.73	.66	.58	.51	.42	.34
Cs 137	30.y	26	36	29	22	18	12	6.9
Cs 138	32.m	.74	--					.74
Mo 99	67.h	5.4	5.3	5.2	5.0	4.8	4.4	4.1
Ba l39	85.m	.081	--					.081
Ba 140	12.8d	.065	.065	.065	.064	.063	.062	.061
La 140	40.2h	.021	.021	.021	.021	.020	.020	.019
Y 90	64.h	.26	.57	.75	.82	.84	.85	.85
Y 91	58.d	.18	.19	.15	.12	.10	.072	.049
Ce 144	285.d	.0027	.0027	.0027	.0027	.0027	.0027	.0026
Cs 134	2.05.y	2.5	3.6	3.0	2.2	1.8	1.2	7.0
H_3	.343 µc/ml							

Average Bleed Rates:

Day 0 – 2:	1.5×10^{-6} Vol/sec	Day 260 – 280: 2.8×10^{-7} Vol/sec
Day 2 – 100:	4.4×10^{-8} Vol/sec	Day 280 – 290: 4.4×10^{-7} Vol/sec
Day 100 – 200:	7.5×10^{-8} Vol/sec	Day 290 – 300: 7.2×10^{-7} Vol/sec
Day 200 – 260:	1.5×10^{-7} Vol/sec	Day 300 – 310: 9.8×10^{-7} Vol/sec

*m Metastable Isotope

Table 11-6

WASTE DISPOSAL SYSTEMS COMPONENT DATA

Clean Waste Receiver Tank	
Number	4
Volume Each, ft^3/gal	7,760/58,000
Material/Lining	Carbon Steel/Plasite 7122
Design Pressure, psig	2
Operating Temp, °F	60 - 170
Design Code/Standard	API 620
Seismic Category	2
Treated Waste Monitor Tank	
Number	2
Volume Each, ft^3/gal	1,970/14,850
Material/Lining	Carbon Steel/Plasite 7122
Design Pressure, psig	5
Operating Temp, °F	120
Design Code/Standard	API 620
Seismic Category	2
Auxiliary Building Equip. Drain Tank	
Number	1
Volume, ft^3/gal	400/3,000
Material/Lining	Carbon Steel/Phenoline 372
Design Pressure, psig	10
Operating Temp, °F	100 - 140
Design Code/Standard	API 620
Seismic Category	2
Clean Resin Mix Tank	
Number	1
Volume, ft^3/gal	90/725
Material/Lining	Carbon Steel/Plastisol
Design Pressure, psig	150
Operating Temp, °F	85
Design Code/Standard	ASME VIII
Seismic Category	2
Spent Resin Storage Tank	
Number	1
Volume, ft^3/gal	655/4,300
Material/Lining	Stainless Steel/None
Design Pressure, psig	100
Operating Temp, °F	150
Design Code/Standard	ASME III Class C
Seismic Category	2

Table 11-6 (continued)

Dirty Waste Drain Tank

Number	1
Volume, ft^3/gal	740/5,530
Material/Lining	Carbon Steel/Phenoline 372
Design Pressure, psig	5
Operating Temp, °F	70 - 140
Design Code/Standard	API 620
Seismic Category	2

Laundry Drain Tank 1

Number	1
Volume, ft^3/gal	189/1,340
Material/Lining	Carbon Steel/None
Design Pressure, psig	5
Operating Temp, °F	160
Design Code/Standard	API 620
Seismic Category	2

Laundry Drain Tank 2

Number	1
Volume, ft^3/gal	268/2,000
Material/Lining	Fiberglass
Design Pressure, psig	Atmospheric
Operating Temp, °F	160
Design Code/Standard	ASTM D3299-74
Seismic Category	2

Filtered Waste Monitor Tank

Number	1
Volume, ft^3/gal	740/5,530
Material/Lining	Carbon Steel/Phenoline 372
Design Pressure, psig	5
Operating Temp, °F	100
Design Code/Standard	API 620
Seismic Category	2

Waste Gas Surge Tank

Number	1
Volume, ft^3/gal	26.4/197
Material	Carbon Steel
Design Pressure, psig	Full Vac to 50
Operating Temp, °F	120 - 170
Design Code/Standard	ASME III Class C
Seismic Category	1

Table 11-6 (continued)

Waste Gas Decay Tanks

Number	4
Volume Each, ft^3/scf @123 psig	271/2,500
Material	Carbon Steel
Design Pressure, psig	132
Operating Temp, °F	120 - 140
Design Code/Standard	ASME III Class C
Seismic Category	1

Reactor Building Sump

Number	1
Volume, ft^3, free	436
Material	Concrete with SS Floor Liner

Auxiliary Building Sump

Number	1
Volume, ft^3	120
Material	Concrete Steel Lined

Auxiliary Building Drain Transfer Pump

Number	1
Capacity, gpm	100
Diff. Head, ft.	64
Operating Temp, °F	100 (140 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Vacuum Degasifier Drain Pump

Number	2
Capacity each, gpm	140
Diff. Head, ft.	85
Operating Temp, °F	120 (170 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Clean Waste Receiver Tank Recirculation Pump

Number	1
Capacity, gpm	225
Diff. Head, ft.	123
Operating Temp, °F	100 (140 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Table 11-6 (continued)

Clean Waste Receiver Tank Transfer Pump

Number	2
Capacity each, gpm	100
Diff. Head, ft.	150
Operating Temp, °F	100 (140 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Treated Waste Monitor Pump

Number	2
Capacity each, gpm	85
Diff. Head, ft.	72
Operating Temp, °F	70 (120 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Dirty Waste Drain Pump

Number	2
Capacity each, gpm	75
Diff. Head, ft.	140
Operating Temp, °F	60 (150 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Auxiliary Building Sump Pump

Number	2
Capacity each, gpm	75
Diff. Head, ft.	40

Laundry Drain Pump

Number	1
Capacity, gpm	50
Diff. Head, ft.	125
Operating Temp, °F	75 - 150
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Filtered Waste Pump

Number	2
Capacity each, gpm	38
Diff. Head, ft	58
Operating Temp, °F	60 (120 max)
Design Code/Standard	Hydraulic Institute
Seismic Category	2

Table 11-6 (continued)

Clean Waste Filter

Number	2
Capacity, gpm	100
Design Pressure, psig	300
Nominal Micron Rating	0.45 Nom.
Material	Stainless Steel
Operating Temp, °F	100 – 120
Design Code/Standard	ASME III Class C
Seismic Category	2

Dirty Waste Filter

Number	2
Capacity, gpm	75
Design Pressure, psig	300
Nominal Micron Rating	10
Material	Stainless Steel
Operating Temp, °F	60 - 150
Design Code/Standard	ASME III Class C
Seismic Category	2

Laundry Clean-Up Filter

Number	1
Capacity, gpm	50
Design Pressure, psig	150
Nominal Micron Rating	20 ABS.
Material	Carbon Steel
Operating Temp, °F	140
Design Code/Standard	ASME III Class C
Seismic Category	2

Gaseous Waste Discharge Filter

Number	1
Rated Flow, scfm	11
Type	HEPA-Charcoal
Design Pressure, psig	125
DOP Micron Rating	.03
Material	Stainless Steel
Operating Temp, °F	120 - 140
Design Code/Standard	ASME III Class 3
Seismic Category	1

Table 11-6 (continued)

Gas Collection Header Filter

Number	1
Rated Flow, scfm	11
Type	HEPA-Charcoal
Design Pressure, psig	5
DOP Micron Rating	.03
Material	Carbon Steel
Operation Temp, °F	120
Design Code/Standard	ASME VIII
Seismic Category	1

Radwaste Demineralizer

Number	3
Design pressure, psig	150
Capacity Normal/Max each, gpm	50/100
Resin Volume each, ft^3	32
Material	Stainless Steel
Operating Temp, °F	120
Design Code/Standard	ASME III Class C
Seismic Category	2

Vacuum Degasifier

Number	1
Design pressure, psig	125
Capacity, gpm	140
Material	Stainless Steel
Operating Temp, °F	120 (170 max)
Design Code/Standard	ASME III Class C
Seismic Category	1

Degasifier Vacuum Pump

Number	2
Type	2 Stage Elliptical Casing, Water Seal
Capacity each, cfm @ 27”Hg	240
Operating Temp, °F	120 (170 max)
Design Code/Standard	ASME III Class 3
Seismic Category	1

Waste Gas Compressors

Number	2
Type	Single Stage Diaphragm
Capacity, each, scfm	4.4
Discharge Pressure, psig	123
Operating Temp, °F	120 (170 max)
Design Code/Standard	ASME Pump & Valve Code for Class III eq't
Seismic Category	1

Table 11-7

SERVICE AND EQUIPMENT

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
Liquid Service Water	Scintillation detector/piping, valves, sample pump, and detector housing, water line to discharge structure, two systems	Log count rate meter, 0 – 10^8 cpm, trended on plant computer	1.9×10^{-6}(µCi/cc) of CS-137	Alarm on high radiation, circuit failure
Radwaste Liquid Discharge	Scintillation detector/ in well in radwaste liquid line to discharge structure	Log count rate meter, 0 – 10^8 cpm, recorded on chart recorder, trended on plant computer	5×10^{-6}(µCi/cc) of Cs-137	Alarm on high radiation, circuit failure, high radiation prohibits radwaste discharge
Intermediate Cooling Water	Scintillation detector/piping, valves, sample pump, and detector housing, inlet to coolers, two systems	Linear rate meter, 0 - 10^8 cpm, trended on plant computer	3.8×10^{-6}(µCi/cc) of Cs-137	Alarm on high radiation, circuit failure
Flumes Liquid	Scintillation detector/piping, valves, sample pump, and detector housing, circulating water to discharge structure	Linear rate meter, 0 - 10^8 cpm, recorded on chart recorder, trended on plant computer	1×10^{-6}(µCi/cc) of Cs-137	Alarm on high radiation, circuit failure
Stack Gas	Scintillation detector/piping, valves, filters, sample pump, and detector housing and sample nozzle, discharge to atmosphere	Spectrometer-Log, 0 - 10^8 cpm, recorded; stack flow recorder; sample flow recorded	1×10^{-5} (µCi/cc) of Xe-133	Alarm on high radiation and circuit failure
High Range Reactor Building Area Monitor	Ion Chambers	Log Rate Meter	10^0 to 10^8 r/hr	No alarm or control funcitons

Table 11-7 (continued)

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
Main Condenser Air	Scintillation detector/well in line, main condenser vacuum pumps non-condensables	Spectrometer-Log, 1 - 10^8 cpm, recorded on chart recorder, trended on plant computer	1×10^{-5} (µCi/cc) of Xe-133	Alarm on high radiation and circuit failure
Waste Gas Radiation	Scintillation detector/well in line from the waste gas surge tank and waste gas decay tanks to plant vent	Linear rate meter, 1 - 10^8 cpm, recorded on chart recorder, trended on plant computer	2×10^{-6}(µCi/cc) of Xe-133	Alarm on high radiation and circuit failure; isolates waste gas surge tank and decay tanks
Decay Heat Water	Scintillation detector/piping, valves, sample pump, and detector housing, water line to discharge structure, two systems	Linear rate meter, 0 - 10^8 cpm, trended on plant computer	2.6×10^{-6}(µCi/cc) of Cs-137	Alarm on high radiation and circuit failure
Failed Fuel	Scintillation detector/ External to Letdown Line	Log and Spectrometer-Log rate meter, 0 - 10^8 cpm, gross and iodine trended on plant computer	4×10^9 dis/sec of I-131; 1×10^{12} Gammas/sec of Gross Gamma	Alarm on high radiation and circuit failure
Turbine Building Drain Line	Scintillation Detector/Off line	Five decade log scale 10 - 10^6 cpm	3.7×10^{-7}(µCi/cc) of Cs-137	Alarm on Hi radiation, circuit failure
Neutralizing Tank T-50 Contents	Scintillation detector/piping, valves, sample pump, and detector housing, neutralizing tank T-50 contents	Digital display in cpm	3.7×10^{-7}(µCi/cc) of Cs-137	Alarm on high radiation, low sample flow

Table 11-8

MAXIMUM ACTIVITY CONCENTRATIONS IN THE STATION EFFLUENT OPERATING WITH 1% FAILED FUEL

Operation	Yearly Average Concentration In Discharge, Fraction of MPC
Liquid Waste	
Lifetime Shim Bleed Including Startup Expansions and Dilutions	.00443
Discharge of Dirty Waste	.0738
Gaseous Waste	
Lifetime Shim Bleed Including Startup Expansions and Dilutions, and Venting of Letdown Storage Tank	.0153
Steam Generator Tube Leakage of 1 gpm	.27
Containment Purge (4 Times per Year)	.00266

Table 11-9

MAXIMUM CLEAN WASTE RECEIVER TANK ACTIVITY

Isotope	Activity in Receiver Tank, Curies*	
	Column I	Column II
Rb 88	593	5.93
Sr 89	9.0	.090
Sr 90	.747	.00747
Sr 91	10.1	.101
Sr 92	3.73	.0373
I 131	703	7.03
I 132	1030	I0.30
I 133	835	8.35
I 134	110	1.10
I 135	593	5.93
Cs 134	791	791
Cs 136	167	167
Cs 137	7,910	7,910
Cs 138	162	162
Mo 99	1,185	1,185
Ba 139	17.8	.178
Ba 140	14.3	.143
La 140	4.61	.0461
Ce 144	.593	.00593
Y 90	186.7	186.7
Y 91	41.7	41.7
H 3	75.5	75.5
Total	14,444	10,558

* Full Tank with Reactor Coolant at Maximum 1% (no Decay) Failed Fuel Concentration

Column I: Purifications Demineralizer By Passed

Column II: Purification Demineralizer Used Prior to Entry to Tanks

Table 11-10A

MAXIMUM WASTE GAS DECAY TANK ACTIVITY

(Assuming Maximum 1% Failed Fuel Reactor Coolant Concentrations)

Isotope	Activity in Decay Tank, Curies**
Kr 85 m*	1,9l0
Kr 85	12,400
Kr 87	1,180
Kr 88	3,420
Xe 131 m	2,800
Xe 133 m	3,420
Xe 133	308,000
Xe 135 m	1,160
Xe 135	7,100
Xe 138	646
Total	342,036

*m indicates metastable isotope

**no decay assumed in filling

Table 11-10B

MAXIMUM WASTE GAS DECAY TANK ACTIVITY AT DISCHARGE

(Assuming Maximum 1% Failed Fuel Reactor Coolant Concentrations)

Isotope	Activity in Decay Tank, Curies*
Kr 85	12,600
Xe 131 m	455
Xe 133 m	0.416
Xe 133	5,930
Total	18,985

*30-day decay after fill.

Table 11-11

MAXIMUM EXPECTED ANNUAL DISCHARGES

(Based on 20-Day Decay)

Isotope	Discharge (mc)	Isotope	Discharge (mc)
Rb 88	0	Cs 136	6.3l x 10^{2}
Sr 89	.722	Cs 137	8.51 x 10^{4}
Sr 90	.0803	Cs 138	0
Sr 91	0	Mo 99	9.10 x 10^{4}
Sr 92	0	Ba 139	0
I 131	13.6	Ba 140	5.24 x 10^{-1}
I 132	0	La 140	1.29 x 10^{-4}
I 133	.0000129	Ce 144	6.07 x 10^{-2}
I 134	0	Y 90	3.50 x 10^{3}
I 135	0	Y 91	3.36 x 10^{5}
Cs 134	8,500	H 3	9.45 x 10^{5}

Table 11-11A

MAXIMUM DESIGN ANNUAL CLEAN WASTE DISCHARGES

(Based on 20-Day Decay)

Isotope	Discharge (Ci)	Isotope	Discharge (Ci)
Rb 88	0	Cs 136	6.31×10^{-1}
Sr 89	7.22×10^{-4}	Cs 137	8.51×10^{1}
Sr 90	8.03×10^{-5}	Cs 138	0
Sr 91	0	Mo 99	9.04
Sr 92	0	Ba 139	0
I 131	1.36×10^{-2}	Ba 140	5.24×10^{-4}
I 132	0	La 140	1.29×10^{-7}
I 133	1.29×10^{-8}	Ce 144	6.07×10^{-5}
I 134	0	Y 90	3.5×10^{-1}
I 135	0	Y 91	3.36×10^{1}
Cs 134	8.50	H 3	9.45×10^{2}

Table 11-11B

MAXIMUM DESIGN ANNUAL DIRTY WASTE DISCHARGES

(Based on 1-Day Decay)

Isotope	Discharge (Ci)	Isotope	Discharge (Ci)
Rb 88	5.78×10^{-6}	Cs 136	6.45
Sr 89	3.48×10^{-1}	Cs 137	3.06×10^{2}
Sr 90	2.89×10^{-2}	Cs 138	4.36
Sr 91	7.07×10^{-2}	Mo 99	4.58×10^{1}
Sr 92	3.10×10^{-4}	Ba 139	6.83×10^{-1}
I 131	2.72×10^{1}	Ba 140	5.52×10^{-1}
I 132	2.96×10^{-4}	La 140	$I.78 \times 10^{-1}$
I 133	1.46×10^{-1}	Ce 144	2.29×10^{-2}
I 134	1.32×10^{-3}	Y 90	5.57
I 135	1.93	Y 90	1.61
Cs 134	3.06×10^{1}	H 3	3.40

Table 11-12

WASTE DISPOSAL FAILURE ANALYSIS

Component	Failure	Comments
Waste Gas Compressor	Gas or Water Leak, Failure to Operate	Isolate for maintenance. Use other compressor.
Waste Gas Decay Tank	Leak or Rupture Waste Gas Surge Tank	Tanks protected by relief valves. Area purged to station vent through filters.
Gaseous Waste Discharge Filter	Rupture or Loss of Efficiency	High activity level monitored. Alarmed if sufficient vent dilution not available. Header closed. Gases diverted gas decay tanks.
Gas Collection Header Filter	Rupture or Loss of Efficiency	Differential pressure indicated and alarmed. Isolate header and repair.
Clean Waste Receiver Tanks	Leak	Tanks in air tight area. Any leakage may be drained to auxiliary building sump. Gases purged through filter To station vent.
Radwaste Liquid Filters	Housing Rupture or Loss of Efficiency	Protected by relief valves. Differential pressure indicated and alarmed. Continuous operation not required. Isolate and repair.
Radwaste Demineralizers	Shell Rupture, Resin Exhaustion	Protected by relief valves. Use Use installed spare. Continuous operation not required. Receiver tanks provide for collection during maintenance.
Vacuum Degasifier	Leak or rupture	Protected by relief valve. Liquid bleed may bypass degasifier to clean waste receiver tanks for maintenance. Gas removed by purging and diluting with N_2 to gas collection header.
Reactor Building Inside (Outside) Sump Drain Valves Sump Drain Valves	Fails to Close	Backup isolation is provided by outside (inside) reactor building isolation valve.
Reactor Building Sump Drain Valve	Fails to Operate	Continuous drainage not required Located for maintenance after shutdown.

Table 11-12 (continued)

Component	Failure	Comments
Auxiliary Building Sump Pumps	Fail to Operate	Continuous operation not required. Located for maintenance.
Radwaste Pumps (10)	Fail to Operate	Redundant pumps are provided for all pumps which are used for frequent operation and are duplicated such that the loss of one pump will not impair system operation.
Auxiliary Building Drain Transfer Pump	Fails to Operate	Tested sufficiently prior to infrequent use to ensure proper operation.
Clean Waste Receiver Tank Recirculation Pump	Fails to Operate	Sufficient holdup available in tanks for maintenance.

Table 11-13

RADIATION DAMAGE THRESHOLD

Material	Approximate Damage Threshold
Teflon	1.0 x 10^4 Rads
Most thermoplastic resins and elastomers	1.0 x 10^6 Rads
Some thermoplastics	1.0 x 10^7 Rads
Ceramics	1.0 x 10^{10} Rads
Metals	1.0 x 10^{11} Rads

Table 11-14

RADIATION ZONES

(Design Dose Rate for Radiation Workers)

Zone Designation	Mrem/Hr (on a 40 Hr/Week Basis)	Description
I	≤ 1.0	Uncontrolled, unlimited occupancy
II	> 1.0 ≤ 2.5	Controlled, limited occupancy. 40 hrs/week.
III	> 2.5 ≤ 15	Controlled access, limited occupancy between 6 - 40 hr/wk.
IV	> 15 ≤ 100	Controlled access, limited occupancy for short period: 1 - 6 hr/wk.
V	> 100	Normally inaccessible. Controlled access, limited occupancy for short periods during emergencies.

Controlled Access Areas - where higher radiation levels and/or radioactive contamination, which have a greater probability of radiation health hazard to individuals, can be expected. Only individuals directly involved in operation of the plant will, in general, be allowed to enter these areas. Ingress and egress are under the direct supervision and authorization by the plant health physics staff.

Occupancy - The time spent by an individual in a particular area. For zones III - V, occupancy is to be determined on an area-by-area basis and individual-by-individual basis by the plant health physics staff.

Table 11-15

AREA RADIATION MONITORS

Monitor Number	Location	Range	Sensitivity
RE - 8001	Main Control Room	0.1 mrem – 10 rem	0.1 mrem
RE - 8002	Relay Room at El. 372' - 0"	"	"
RE - 8003	Machine Shop at El. 354' - 0"	"	"
RE - 8004	Outside Stairway at El. 3l7' - 0"	"	"
RE - 8005	Sample Room Hall at El. 354' - 0"	"	"
RE - 8006	Radio-Chem Laboratory at El. 354' - 0"	"	"
RE - 8007	Outside Stairway at El. 372' - 0"	"	"
RE - 8008	Decontamination Room at El. 386' - 0"	"	"
RE - 8009	Spent Fuel Pool at El. 404' - 0"	"	"
RE - 8010	Controlled Access Entry at El. 386' - 0"	"	"
RE - 8011	Make Up Pump Area at El. 335' - 0"	"	"
RE - 8012	Elevator at Feed Pump at El. 335' - 0"	"	"
RE - 8013	Emergency Feed Pump at El. 335' - 0"	"	"
RE - 8014	CZ Pump at El. 326' - 0"	"	"
RE - 8015	Condensate Demineralizer at El. 354' - 0"	"	"
RE - 8016	Spent Fuel Filter Area at El. 404' - 0"	"	"
RE - 8017	Fuel Handling at El. 406' - 0"	10 mrem - 1000 rem	10 mrem
RE - 8018	Personnel Access Hatch at El. 386' - 0"	"	"
RE - 8019	Incore Instrument Tank at El. 340' - 0"	"	"
RE - 8020	Equipment Hatch at El. 354' - 0"	"	"

Table 11-16

DIRTY LIQUID RADIOACTIVE WASTE SYSTEM PIPING

Piping

USE	PRIMARY RATING	MATERIAL	DESIGN CODE	SEISMIC CATEGORY
Floor Drains, Eq't Drains and Drain Headers	150# @ 500 °F	SS	B31.1	2
Main Process Piping	150# @ 500 °F	SS	B31.7 Class III[1]	2
Reactor Building Penetration	150# @ 500 °F	SS	B31.7 Class II[1]	1
Circ Water Discharge	150# @ 500 °F	SS	B31.1	2
Rx Bldg Sump to Aux Bldg Sump	150# @ 500 °F	SS	B31.1	2
Equipment Vents	150# @ 500 °F	CS	B31.1	2
Service Air	125# WSP	CS	B31.1	2

Valves

1. All valves for B31.7 Class 2 piping were specified to the Pump and Valve Code for Class 2 valves.

2. All valves for B31.7 Class 3 piping were specified to the Pump and Valve Code for Class 3 valves.

3. All valves for B31.1 piping were specified to USAS B 16.5.

Note 1: Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

Table 11-17

LAUNDRY RADIOACTIVE WASTE SYSTEMS

Piping

USE	PRIMARY RATING	MATERIAL	DESIGN CODE	SEISMIC CATEGORY
Main Process Piping	150# @ 500 °F	CS	B31.1	2
Circ Water Discharge Header	150# @ 500 °F	SS	B31.1	2
Vents	150# @ 500 °F	CS	B31.1	2
Drains	150# @ 500 °F	SS CS	B31.1	2

Valves

See "Dirty Liquid Radioactive Waste System"

Table 11-18

CLEAN LIQUID RADIOACTIVE WASTE SYSTEM

Piping

USE	PRIMARY RATING	MATERIAL	DESIGN CODE	SEISMIC CATEGORY
Floor Drains, Eq't Drains & Drain Headers	150 # @ 500 °F	SS	B31.1	2
Main Process Piping	150 # @ 500 °F	SS	B31.7 Class III[1]	2
Resin Sluice Piping	150 # @ 500 °F	SS	B31.7 Class III[1]	2
Clean Waste Receiver Tank Vents	150 # @ 500 °F	CS	B31.7 Class III[1,2]	2
All Other Vents	150 # @ 500 °F	CS	B31.1	2
Circ Water Discharge Discharge	150 # @ 500 °F	SS	B31.1	2
Demineralized Water	150 # @ 500 °F	CS	B31.1	2
Vacuum Degasifier Gas Piping	150 # @ 500 °F	SS CS	B31.7 Class III[1]	1
Air	125# WSP	CS	B31.1	2
Nitrogen	150# @ 500 °F	CS	B31.1	2

Valves

See "Dirty Liquid Radioactive Waste System"

Note 1 Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

Note 2 The vent piping from each tank up to the first tee and the vacuum breaker is B31.7. The remaining piping and valves are B31.1.

Table 11-19

GASEOUS RADIOACTIVE WASTE SYSTEM

Piping

USE	PRIMARY RATING	MATERIAL	DESIGN CODE	SEISMIC CATEGORY
Reactor Building Vent Header	150# @ 500 °F	CS	B31.7 Class III[1]	2
Reactor Building Penetration	150# @ 500 °F	CS	B31.7 Class II[1]	1
Auxiliary Bldg Vent Header & Surge Tank Inlet	150# @ 500 °F	CS	B31.7 Class III[1]	2
Main Processing	150# @ 500 °F	CS	B31.7* Class III[1]	1
Gas Discharge to Ventilation System & GCH	150# @ 500 °F	CS	B31.1	2
Equipment Drains	150# @ 500 °F	SS	B31.1	2
Nitrogen	150# @ 500 °F	CS	B31.1	2

Valves*

See "Dirty Liquid Radioactive Waste System"

* Piping and valves in the Main Processing portion have been downgraded to the B31.1 design code per DCP-87-1055.

Note 1: Use of later appropriate ASME Section III Code(s) is acceptable, provided the Code section used is reconciled.

Table 11-20

ORIGINAL STEAM GENERATOR STORAGE FACILITY (OSGSF) TOTAL RADIOACTIVE ISOTOPE INVENTORY OF STEAM GENERATORS

Nuclide	Curies	uCi/cc
H-3	1.752E+02	1.068E+00
Cr-51	1.190E+03	7.258E+00
Mn-54	1.181E+02	7.200E-01
Co-57	6.878E+00	4.194E-02
Co-58	1.630E+03	9.940E+00
Fe-59	4.412E+01	2.690E-01
Co-60	1.991E+02	1.214E+00
Nb-95	2.704E+02	1.649E+00
Zr-95	6.868E+01	4.188E-01
Ag-110m	1.607E+01	9.796E-02
Sn-113	9.300E+00	5.670E-02
Cs-134	4.700E+00	2.866E-02
Cs-137	5.506E+01	3.356E-01
Fe-55	1.128E+03	6.874E+00
Ni-63	2.888E+02	1.761E+00
Sr-90	7.276E-04	4.436E-06
Pu-238	1.669E-04	1.017E-06
Pu-239	5.156E-05	3.144E-07
Pu-240	5.156E-05	3.144E-07
Pu-241	9.018E-03	5.498E-05
Am-241	1.350E-04	8.230E-07
Cm-242	3.544E-04	2.160E-06
Cm-243	2.568E-04	1.566E-06
Cm-244	2.568E-04	1.566E-06
Sb-124	2.102E+01	1.282E-01
Sb-125	4.030E+00	2.456E-02
TOTALS	**5.230E+03**	**3.189E+01**

- Modeling of the SG to assess curie content utilized WMG Megashield, Version 3.0.
- Isotopic proportions based on previous samples.
- The composite curie content mixture was derived from gamma spectrum analyses performed in the years 2000, 2002, and 2004. Analyses were performed by Duke Engineering and Services Environmental Laboratory (DE&S), Framatome ANP DE&S Laboratory, and Framatome ANP Environmental Laboratory in support of 10 CFR 61 waste characterization requirements.
- The estimate is conservative to arrive at a value that bounds component storage. The SG channel heads and reactor coolant system pipe surface contamination contributions are considered to be absorbed within the conservatism of the estimate. Reference ANO-2014-0052.

Table 11-21

ORIGINAL STEAM GENERATOR STORAGE FACILITY (OSGSF) RADIOACTIVE ISOTOPE INVENTORY OF REACTOR VESSEL CLOSURE HEAD

Nuclide	Curies	uCi/cc
H-3	1.051E+00	9.647E-03
Cr-51	7.142E+00	6.555E-02
Mn-54	7.086E-01	6.504E-03
Co-57	4.127E-02	3.788E-04
Co-58	9.782E+00	8.978E-02
Fe-59	2.647E-01	2.430E-03
Co-60	1.195E+00	1.097E-02
Nb-95	1.623E+00	1.490E-02
Zr-95	4.121E-01	3.782E-03
Ag-110m	9.640E-02	8.848E-04
Sn-113	5.580E-02	5.122E-04
Cs-134	2.820E-02	2.588E-04
Cs-137	3.303E-01	3.032E-03
Fe-55	6.766E+00	6.210E-02
Ni-63	1.733E+00	1.591E-02
Sr-90	4.365E-06	4.006E-08
Pu-238	1.001E-06	9.188E-09
Pu-239	3.094E-07	2.840E-09
Pu-240	3.094E-07	2.840E-09
Pu-241	5.411E-05	4.967E-07
Am-241	8.100E-07	7.435E-09
Cm-242	2.126E-06	1.951E-08
Cm-243	1.541E-06	1.414E-08
Cm-244	1.541E-06	1.414E-08
Sb-124	1.262E-01	1.158E-03
Sb-125	2.418E-02	2.219E-04
TOTALS	3.138E+01	2.880E-01

- Modeling of the Reactor Vessel Closure Head (RVCH) to assess curie content utilized WMG Megashield, Version 3.0.
- Isotopic proportions based on previous samples.
- The composite curie content mixture was derived from gamma spectrum analyses performed in the years 2000, 2002, and 2004. Analyses were performed by Duke Engineering and Services Environmental Laboratory (DE&S), Framatome ANP DE&S Laboratory, and Framatome ANP Environmental Laboratory in support of 10 CFR 61 waste characterization requirements.
- The estimate is conservative to arrive at a value that bounds component storage. The SG channel heads and reactor coolant system pipe surface contamination contributions are considered to be absorbed within the conservatism of the estimate. Reference ANO-2014-0052.

Table 11-22

LOW LEVEL RADIOACTIVE WASTE STORAGE BUILDING (LLRWSB)

TOTAL RADIOACTIVE ISOTOPE INVENTORY BASED ON CAPACITY

(Reference CR-ANO-C-2014-1356)

B-25 Boxes

Nuclide	mCi per B-25 Box	Total Ci
Cs-137	36.8	23.552
Co-58	7.66	4.9024
Co-60	17.4	11.136
Mn-54	2.32	1.4848
Cs-134	2.78	1.7792
Sb-125	16.4	10.496

Total	29.7984

Maximum B-25 box storage capacity estimated at 640 boxes.

55 Gallon Drums

Nuclide	mCi per Drum	Total Ci
C-14	40.6	31.1808
Co-58	0.221	0.169728
Co-60	0.719	0.552192
Cs-137	5.55	4.2624
H-3	61.4	47.1552
I-129	15.6	11.9808
Mn-54	986	757.248
Nb-95	0.00188	0.001444
Sb-125	0.00687	0.005276
Tc-99	38.4	29.4912

Total	882.047

Maximum 55 Gallon Drum storage capacity estimated at 768 drums.

Table 11-22 (continued)

Filter Liners

Nuclide	mCi per Filter Liner	Total Ci
Be-7	4.14E-03	0.00019
C-14	112	5.152
Cr-51	1.07E-09	4.92E-11
Co-57	1.92E+00	8.83E-02
Co-58	0.127	0.005842
Co-60	653	30.038
Cs-134	16.3	0.7498
Cs-137	39.3	1.8078
Fe-55	1.13E+03	51.98
Fe-59	3.89E-06	1.79E-07
H-3	61.4	2.8244
Hf-181	3.19E-07	1.47E-08
Mn-54	138	6.348
Nb-95	1.41E-06	6.49E-08
Ni-63	322	14.812
Sb-124	4.60E-05	2.12E-06
Sb-125	63.2	2.9072
Sn-113	0.424	0.019504
Sr-89	1.23E-05	5.66E-07
Sr-90	3.36	0.15456
Tc-99	0.816	0.037536
Zn-65	2.15	0.0989
Zr-95	2.79E-02	0.001283

Total	117.0253

Maximum Filter Liner storage capacity estimated at 46 liners.

Table 11-22 (continued)

Resin Liners

Nuclide	mCi per Resin Liner	Total Ci
Am-241	5.64E-02	2.54E-03
C-14	1.09E+03	49.05
Ce-144	9.24E+01	4.16E+00
Co-57	1.22E+01	5.50E-01
Co-58	38.6	1.737
Co-60	6000	270
Cs-134	78.8	3.546
Cs-137	749	33.705
Fe-55	4.68E+03	210.6
H-3	13.9	0.6255
I-129	0.0969	0.004361
Mn-54	476	21.42
Ni-59	39.2	1.764
Ni-63	6390	287.55
Sb-125	798	35.91
Sr-90	6.26	0.2817
Tc-99	2.58	0.1161
Zn-65	89.4	4.023

Total	925.04

Maximum Resin Liner storage capacity estimated at 45 liners.

Sea Land (One – Located in Truck Bay)

Nuclide	mCi per Sea Land	Total Ci
C-14	4.94E+01	0.0494
Co-58	26.8	0.0268
Co-60	60.9	0.0609
Cs-134	9.74	0.00974
Cs-137	129	0.129
H-3	298	0.298
I-129	12.4	0.0124
Mn-54	8.12	0.00812
Tc-99	67.2	0.0672
Sb-125	57.4	0.0574

Total	0.71896

Table 11-23

OLD RADWASTE STORAGE BUILDING
TOTAL RADIOACTIVE ISOTOPE INVENTORY BASED ON CAPACITY

Nuclide	mCi / B-25 Box	Total Ci at ORWSB Capacity
Co-58	7.66	13.78
Cs-137	36.8	66.21
Co-60	17.4	31.31
Mn-54	2.32	4.17
Cs-134	2.78	5.00
Sb-125	16.4	29.51
Total		149.99

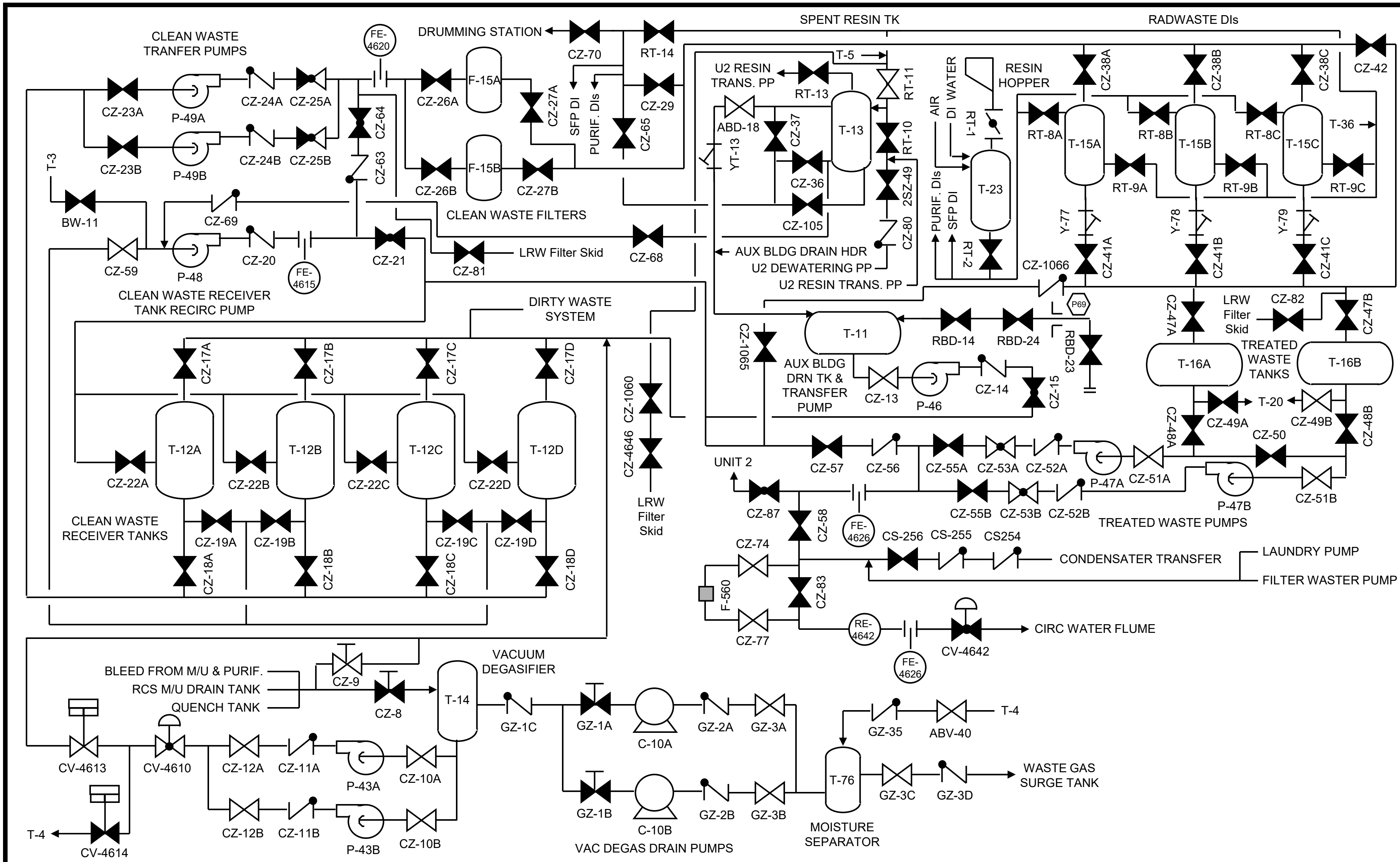
Maximum B-25 box storage capacity stacked 4 high is estimated at 1799 boxes assuming a total storage volume of 202444 ft^3.

Table 11-24

POLE BARN
TOTAL RADIOACTIVE ISOTOPE INVENTORY BASED ON CAPACITY

Nuclide	mCi*
C-14	2.44
Co-57	0.68
Co-58	1.06
Co-60	52.35
Cr-51	0.62
Cs-134	0.20
Cs-137	21.24
Fe-55	70.15
Mn-54	2.49
Nb-95	1.96
Ni-63	63.39
Sb-125	0.61
Zr-95	1.10
Total	218.3

* Activity estimate based on material that could be stored given building dimensions.



CLEAN LIQUID RADWASTE

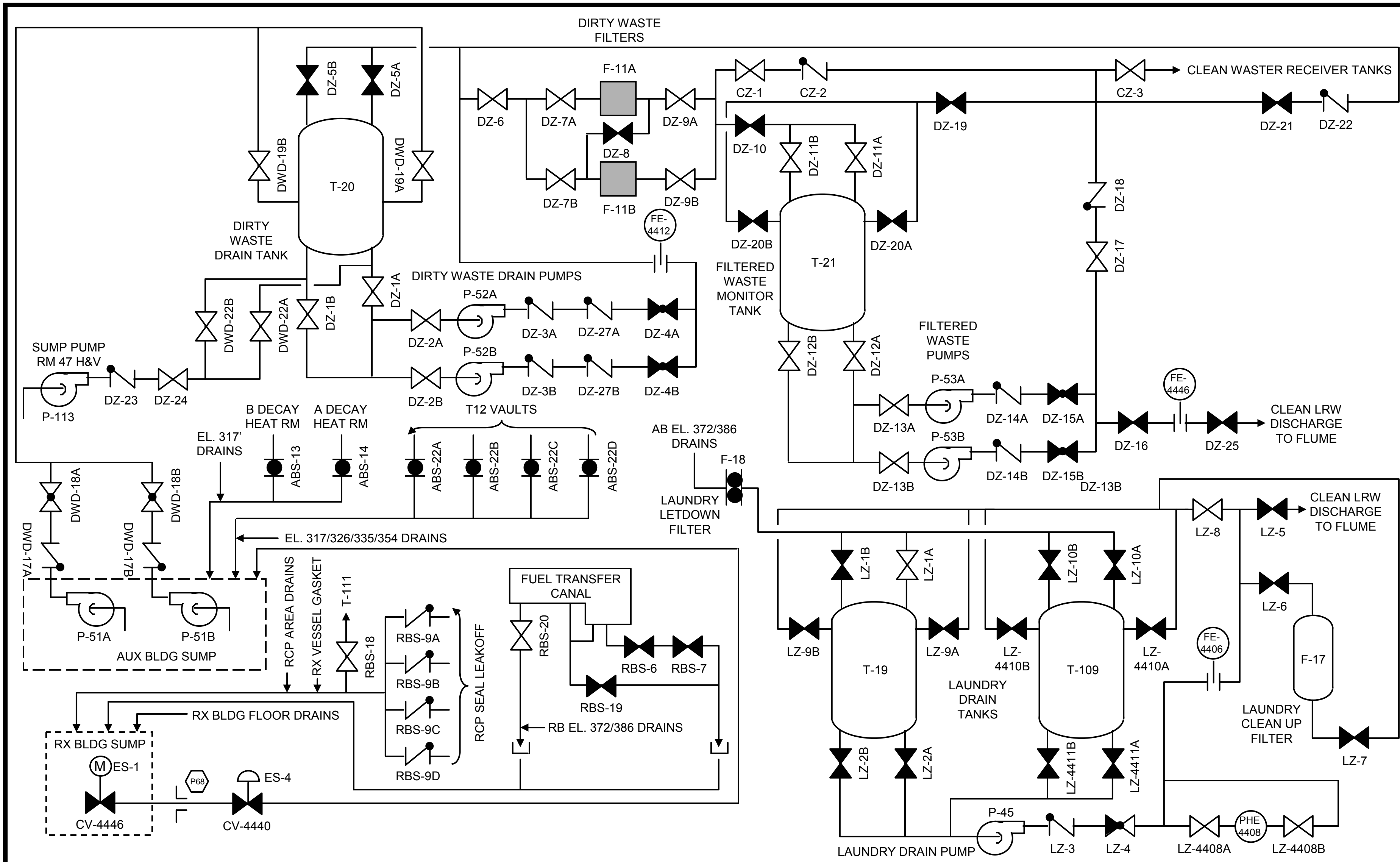
SAR FIGURE NO. 11-1

AMENDMENT 30

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



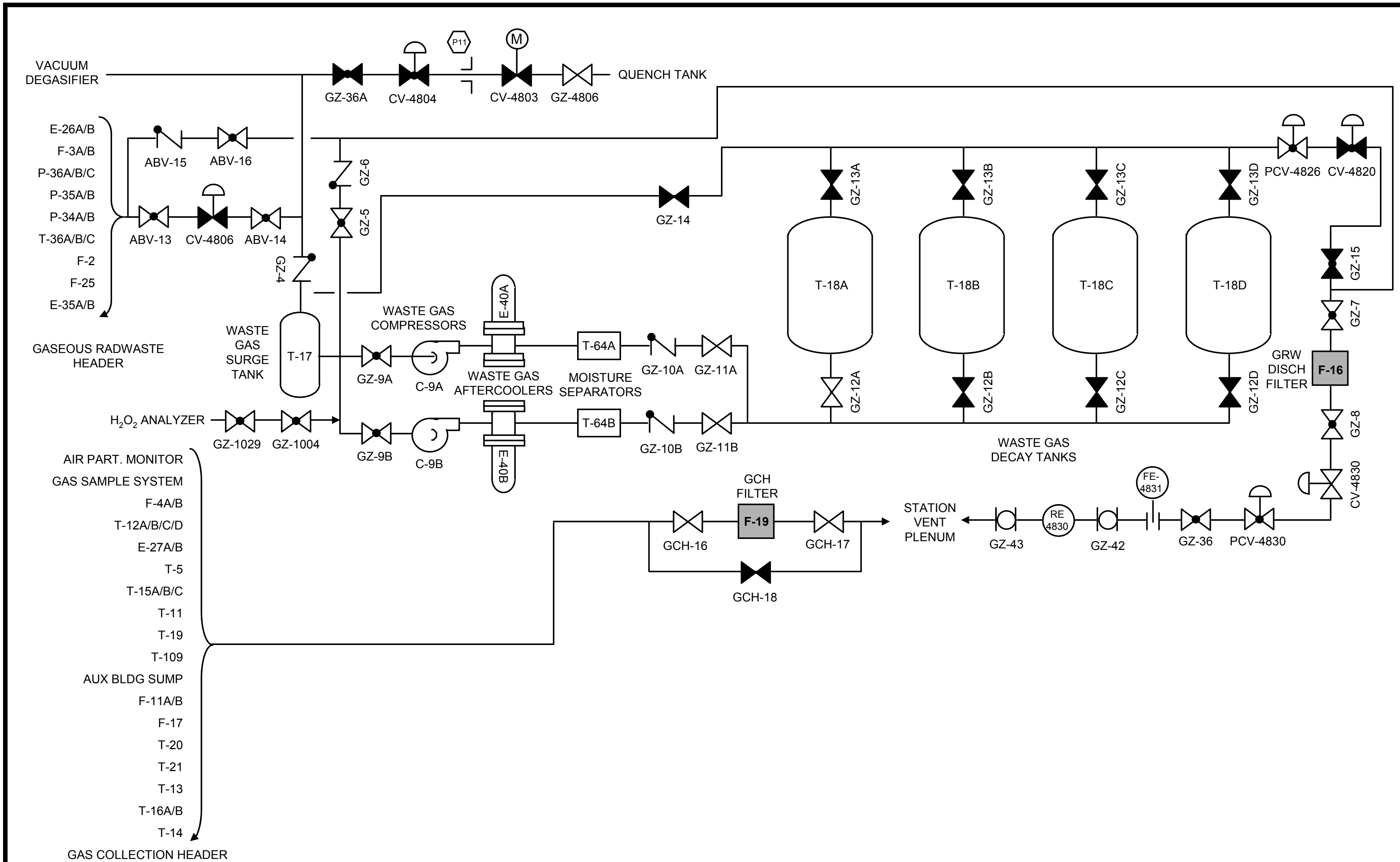
DIRTY LIQUID RADWASTE	SAR FIGURE NO. 11-2
	AMENDMENT 22

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



GASEOUS RADWASTE

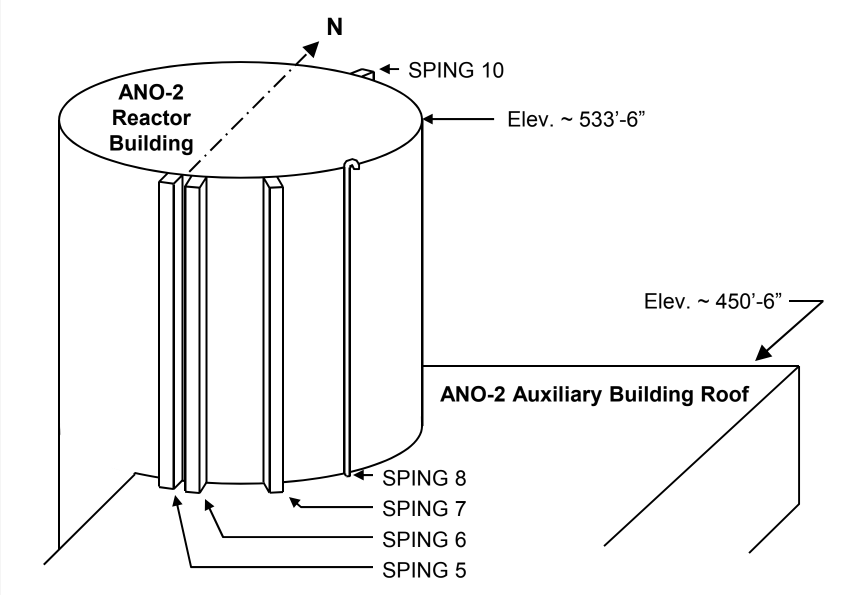
SAR FIGURE NO. 11-3

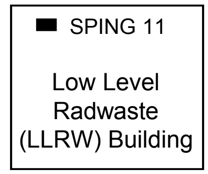
AMENDMENT 20

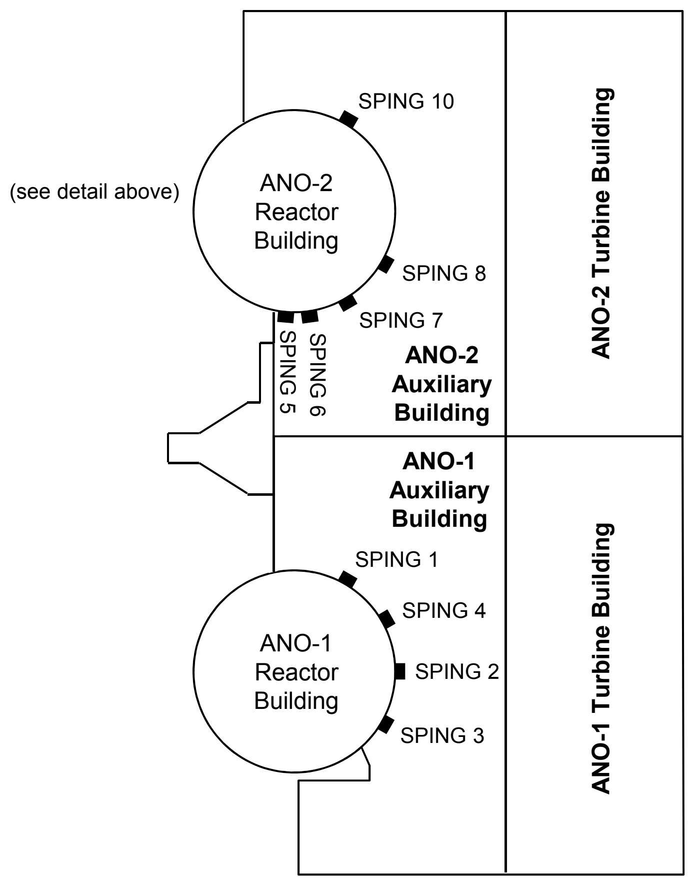
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.







NOTES:

1. PASS release point at approximate Elev. 404'
2. LLRW release point at approximate Elev. 392'
3. Ground level approximate Elev. 354'
4. SPING monitors are as follows:

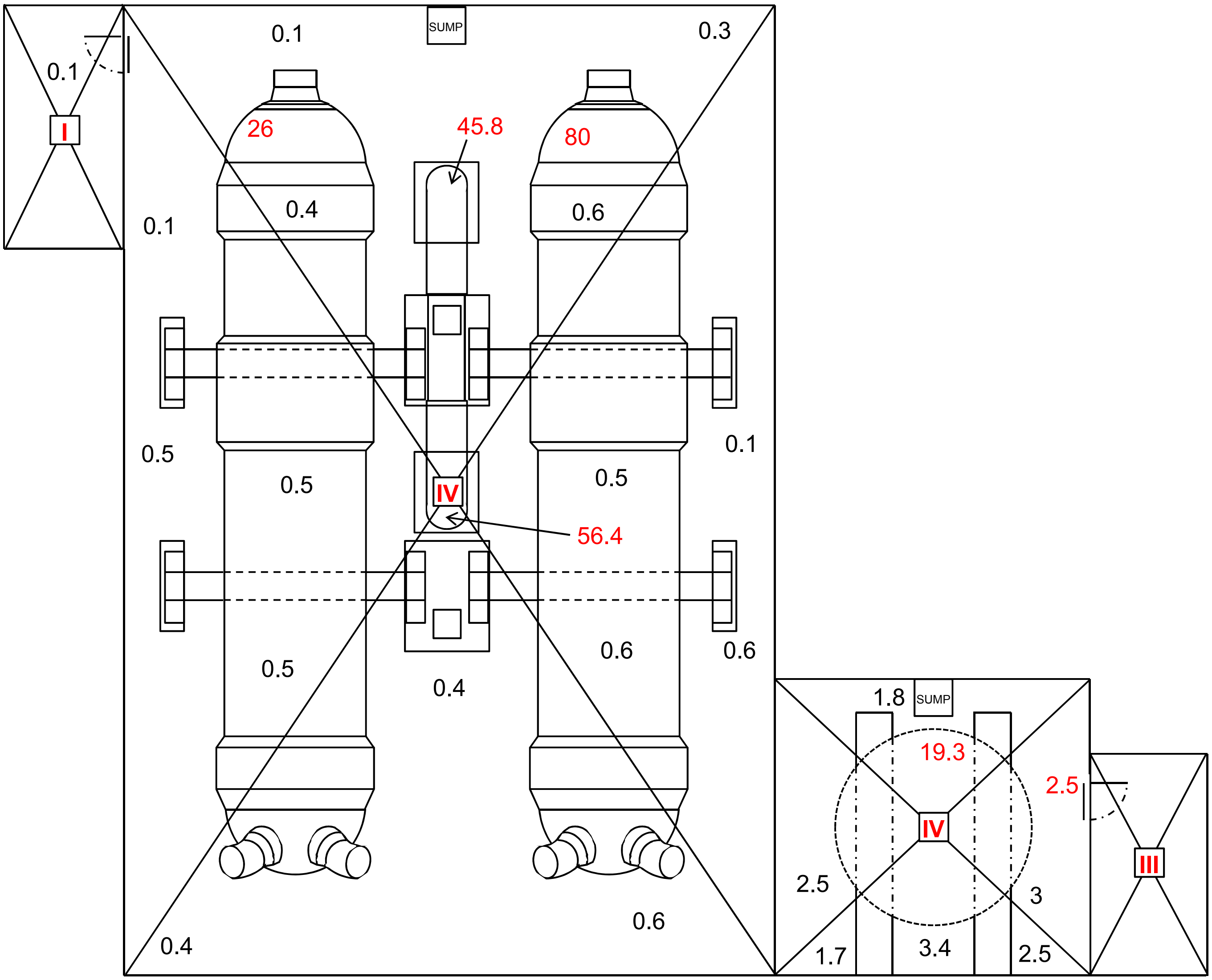
SPING 1	RX-9820	U1 Reactor Building Purge
SPING 2	RX-9825	U1 Radwaste Area
SPING 3	RX-9830	U1 Fuel Handling Area
SPING 4	RX-9835	U1 Penetration Rooms
SPING 5	2RX-9820	U2 Containment Purge
SPING 6	2RX-9825	U2 Radwaste Area
SPING 7	2RX-9830	U2 Fuel Handling Area
SPING 8	2RX-9835	U2 Penetration Rooms
SPING 10	2RX-9845	U2 Aux Building Extension
SPING 11	2RX-9850	LLRW Building

5. Gaseous Radwaste System vents and Condenser Vacuum off-gas discharge into the respective unit's Radwaste Area exhaust ducting upstream of the installed HEPA/Charcoal filter units
6. Other potential release points are included in ANO-2 SAR Tables 11.3-5A and 11.3-5B
7. SPING sample points are downstream of each exhaust fan and HEPA/Charcoal filter unit

SAR FIGURE NO. 11-4

AMENDMENT 29

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MONITORED VENTILATION EXHAUST RELEASE POINTS	BASED ON DRAWING NO	SHEET 1 REV.



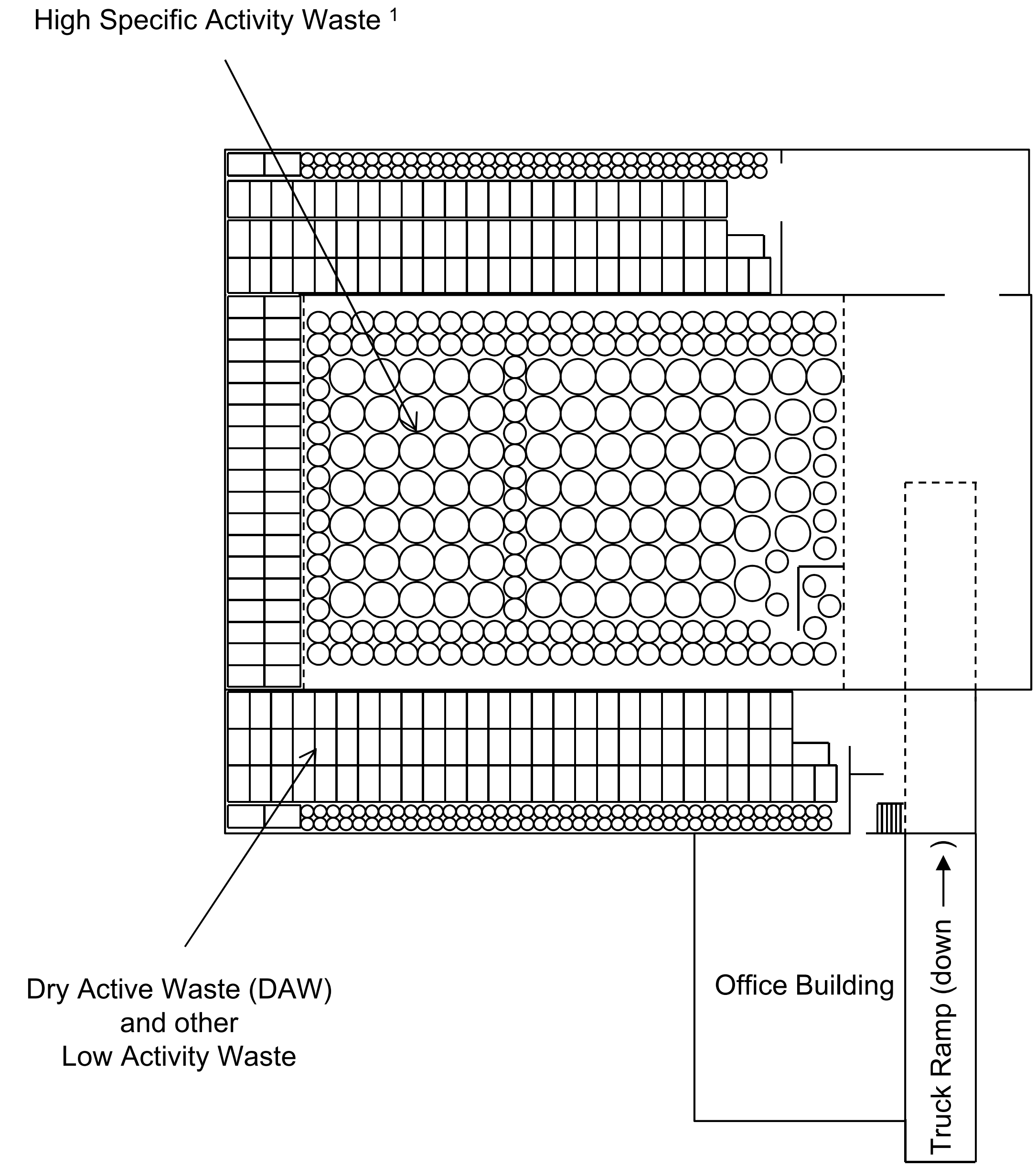
Notes

1. Radiation Zones I – V are defined in SAR Table 11-14
2. OSGSF Roof is a Zone II Area
3. General area dose rates (mR/hr) in black
4. Contact dose rates (mR/hr) in red
5. Dose rates based on 2014 survey

SAR FIGURE NO. 11-5

AMENDMENT 26

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
RADIATION ZONING & ACCESS CONTROL PLAN AT OSGSF	BASED ON DRAWING NO	SHEET 1 REV.



[1] High Specific Activity Waste (HSAW) consists of primarily filters and resins, and is considered Low Level Radwaste (LLRW), but is characterized by higher activities than DAW and the necessity of remote handling.

SAR FIGURE NO. 11-6

AMENDMENT 26

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
LOW LEVEL RADWASTE BUILDING STORAGE	BASED ON DRAWING NO	SHEET	REV.

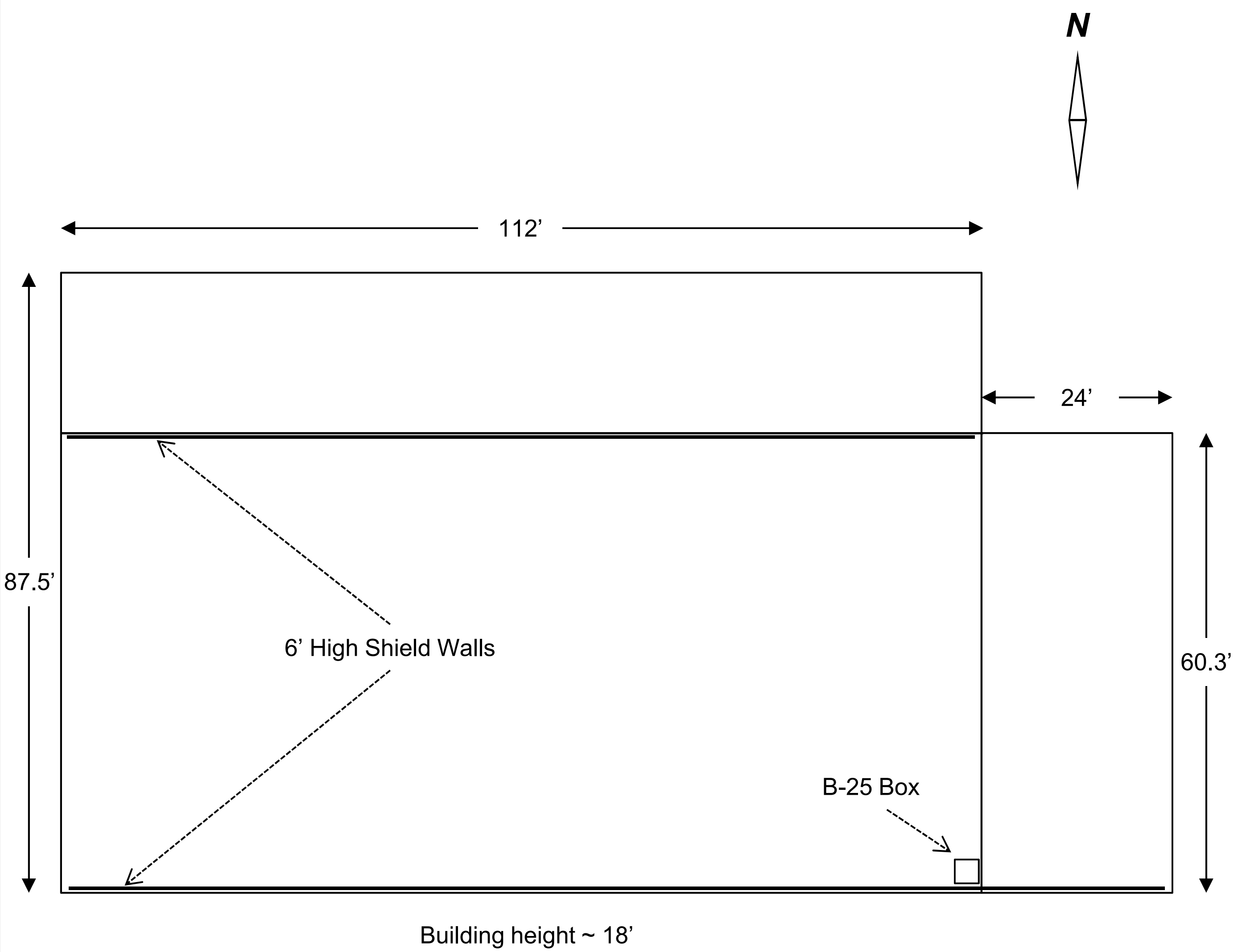
(dose rates based on maximum storage (mrem/hr))

[1] High Specific Activity Waste (HSAW) consists of primarily filters and resins, and is considered Low Level Radwaste (LLRW), but is characterized by higher activities than DAW and the necessity of remote handling.

SAR FIGURE NO. 11-7

AMENDMENT 26

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
LOW LEVEL RADWASTE BUILDING ROOF ESTIMATED DOSE RATES	BASED ON DRAWING NO	SHEET	REV.



NOTES:

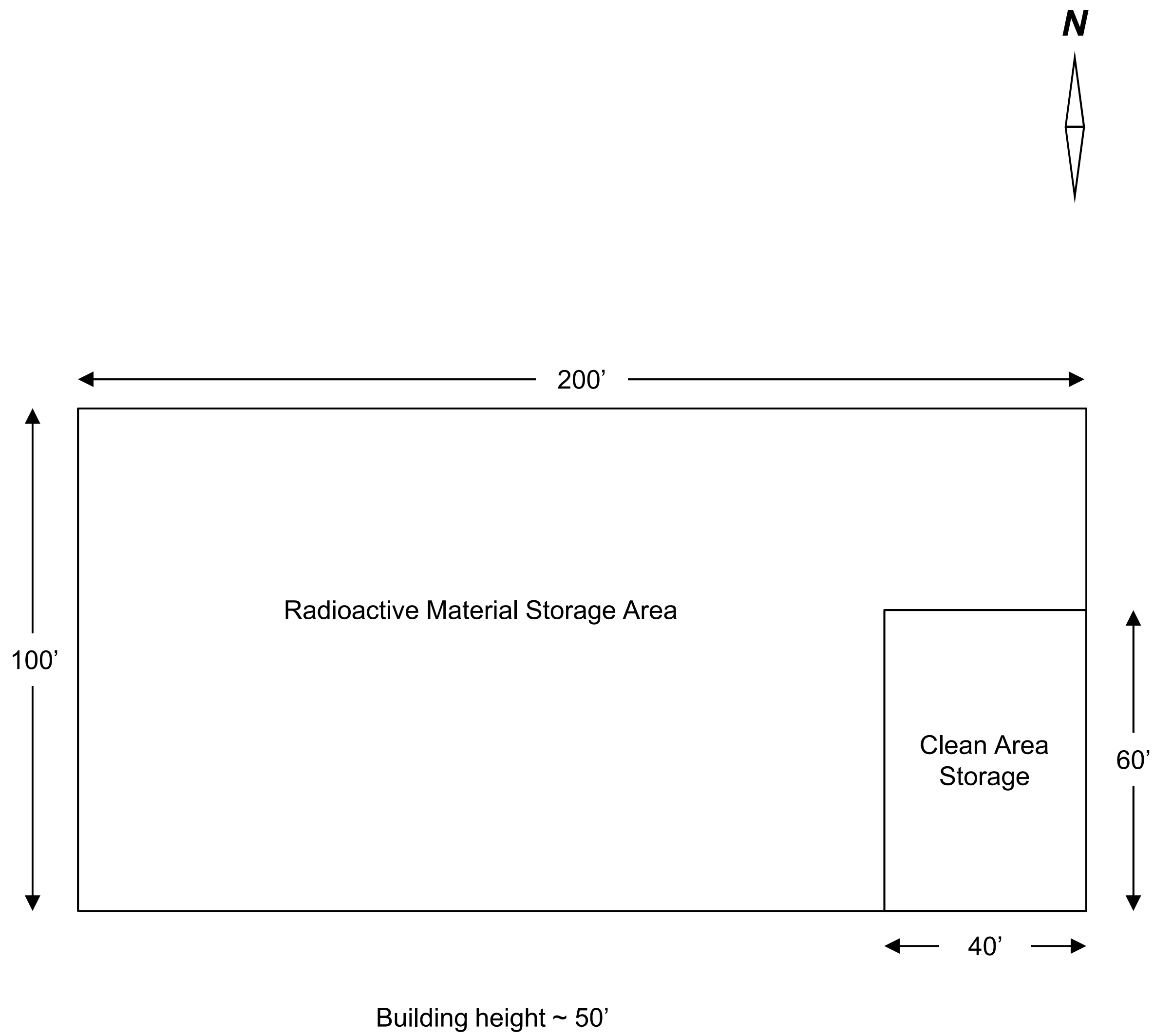
1. Based on estimated volumetric footprint, a total of 1799 B-25 boxes can be stored if stacked 4 high.

2. Conservative estimated dose rate per B-25 box ~ 80 mR/hr.

SAR FIGURE NO. 11-8

AMENDMENT 26

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
OLD RADWASTE STORAGE FACILITY	BASED ON DRAWING NO	SHEET	REV.



NOTE: Hypothetical source term geometry estimated to be a right circular cylinder of 50’ radius and 30’ height with total content of 0.218 Curies.

SAR FIGURE NO. 11-9

AMENDMENT 26

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

POLE BARN

BASED ON DRAWING NO	SHEET	REV.

ARKANSAS NUCLEAR ONE
Unit 1

Chapter 12

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12-2	DELETED
12-3	DELETED
12-4	DELETED
12-5	DELETED
12-6	DELETED

UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross Reference
12.1.2.1	Correspondence from Williams, AP&L, to Heltemes, NRC, dated April 17, 1978. (0CAN047801)
12.3	Correspondence from Cavanaugh, AP&L to Reid and Stolz, NRC, dated April 6, 1978. (0CAN047802)
12.3	Correspondence from Trimble, AP&L, to Reid, NRC, dated December 28, 1979. (0CAN127917)
12.1.2.2	Correspondence from Cavanaugh, AP&L, to Reid and Clark, NRC, dated October 20, 1980. (0CAN108017)
12.3	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated October 22, 1980. (0CAN108021)
12.2.1	Correspondence from Trimble, AP&L, to Eisenhut, NRC, dated December 31, 1980. (0CAN128017)
12.3	Correspondence from Trimble, AP&L, to Denton, NRC, dated January 30, 1981. (0CAN018131)
12.1.1	Correspondence from Trimble, AP&L, to Clark and Stolz, NRC, dated May 27, 1981. (0CAN058116)
12.3	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated July 23, 1981. (0CAN078109)
12.1.1 12.1.2.1	Correspondence from Cavanaugh, AP&L, to Stello, NRC, dated November 12, 1980. (0CAN118012)

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12.1.2.2	Correspondence from Cavanaugh, AP&L, to Seyfrit, NRC, dated April 16, 1981. (0CAN048112)

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12.1.1	ANO-1 Technical Specifications Amendment 143, dated February 4, 1991. (0CNA029101)
12.2.2.1 12.2.3.1	Changes per Generic Letter 87-07
12.2.2.2.C	Procedure 1063.024, Rev. 9, "Shift Engineer/Shift Technical Advisor Training Program."

UPDATE REFERENCE LIST (continued)

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12.5.2.1.F.12	ANO Procedure 1203.004, Rev. 5, "Reactor High Startup Rate."
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12.2.2.2.C	Licensing Document Change Request to Reflect Requirements of NUREG-0737, Unit Specific Program, and Course Summaries for Shift Engineer and STA Training.
Amendment 15	
12.8 12.8.1 12.8.2	ANO-1 Technical Specification Amendment 192, "Addition of Technical Requirements Manual."
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12.1.1.4.2.2	Procedure AD-101, Revision 8, “Nuclear Management Manual Process.”
12.1.1.5.2.3	Procedure DC-120, Revision 0, “Entergy Code Programs.”
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12.1.1.3.2.1 12.1.1.3.2.1.1	License Document Change Request 1-12.1-0018, “Clarification of Authority”
12.1.1.3.2.1	License Document Change Request 1-12.1-0019, “QAPM Approval Revision”

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12.1.1.3.2.1 Figure 12-6	License Document Change Request 1-12.1-0020, “Quality Assurance Organizational Changes”
12.1.1.5.2.1.1	License Document Change Request 1-12.1-0021, “Division of Responsibility”
12.2.2.1 12.2.3.1	License Document Change Request 1-12.2-0003, “Editorial Corrections”
12.1.1.3.2.1 12.1.1.3.2.1.2 12.1.1.3.5 12.1.1.5.2.1.1	License Document Change Request 1-12.1-0022, "Site Organizational Changes"
Figures 12-1 thru 12-5	License Document Change Request 1-1.1-0008, “Reformatting of Various Drawing Labels”

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Section	Cross Reference
12.2.2.2	License Document Change Request 1-12.2-0004, "Clarification of Shift Engineer Training Requirements"
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Section	Cross Reference
12.1.1.3.2.1	Licensing Document Change Request 06-047, "Revision to QAPM”
12.1.1.3.1.2 12.1.1.3.2.1 12.1.1.3.2.1.1 12.1.1.3.2.1.2	Licensing Document Change Request 06-048, "Inspection Program Revision”
12.1.1.3.4	License Document Change Request 07-070, “Employee Alignment-Related Organizational Changes”

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12.1.1	License Document Change Request 13-024, "Remove Specific Corporate Management Titles"
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UPDATE REFERENCE LIST (continued)

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Amendment 27	
12.2.2.2	Licensing Basis Document Change LBDC 16-046, "General Employee Refresher Training Frequency"
12.1.1 Figure 12-6	Licensing Basis Document Change LBDC 17-013, "Removal of Obsolete or Redundant SAR Figures"

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12 CONDUCT OF OPERATIONS

12.1 ORGANIZATIONAL STRUCTURES OF APPLICANT

12.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATIONS

The Entergy organization is described in the Quality Assurance Program Manual (QAPM). The role of the corporate organization is to monitor, assess and provide support to the Entergy Operations' nuclear sites and help minimize the activity of the nuclear sites staffs that are not directly related to day-to-day site management.

Plant specific titles for generic titles in the Technical Specifications are located in Figure 12-1.

12.1.2 ORGANIZATIONAL INTERFACES AND RESPONSIBILITIES

Entergy Arkansas, Inc. and Entergy Operations, Inc. are joint licensees under the facility operating license condition, each responsible for specific areas and jointly responsible for regulatory compliance and response. Entergy Arkansas, Inc. is licensed to possess the facility and Entergy Operations, Inc. is licensed to possess, use and operate the facility.

Each supplier of equipment, material or services and each maintenance or modification contractor is responsible for administering the applicable quality assurance/quality control functions as required by ANO. The Quality Organization is responsible for assuring by surveillance, inspection, audit or review of objective evidence that onsite functions are accomplished for systems, structures and services that affect the safety and integrity of the plant.

12.1.3 QUALIFICATIONS OF NUCLEAR OPERATIONS PERSONNEL

The education and experience of the plant operation, technical, and maintenance support personnel meet the requirements set forth in ANSI/ANS 3.1-1978. In addition, the facility staff qualifications and training programs conform with Regulatory Guide 1.8.

The personnel assigned to Unit 1 have an aggregate of many years experience in conventional stations and have been given special nuclear training. Section 12.2 gives the details of this training program and methods for qualifying station personnel.

Qualification and educational backgrounds of Arkansas Nuclear One management and supervisory personnel are maintained on file at Arkansas Nuclear One.

12.1.4 TECHNICAL CONSULTANTS

Historical data removed - To review exact wording please refer to Section 12.1 of the FSAR.

12.2 TRAINING

Section 12.2.1 describes the initial training developed and conducted for personnel at ANO-1 during the time up to and including initial licensing. Current training is described in Section 12.2.2.

12.2.1 STATION STAFF – INITIAL TRAINING

Historical data removed - to review the exact wording please refer to Section 12.2.1 of the FSAR.

12.2.2 REPLACEMENT TRAINING

12.2.2.1 Replacement Training - Licensed Personnel

The replacement training program for licensed personnel is based on a systems approach to training and is accredited by the National Academy of Nuclear Training. The initial accreditation of the training program took place in January 1984. The new training program, which incorporated revisions required by changes to 10 CFR 55, was audited by the National Academy of Nuclear Training in January, 1988. The current program supercedes the previously existing NRC approved training program (certified to the Commission in accordance with Generic Letter No. 87-07 and the guidance given in NUREG-1262 in May, 1988).

12.2.2.2 Replacement Training - Non-Licensed Personnel

Training for personnel not requiring NRC licenses is provided based upon the individual’s background experience and duties and responsibilities of his or her position.

A. General Employee Training

All employees who enter the protected area at ANO are required to attend the General Employee Training program or be accompanied by someone who has completed the training. The program provides a general plant physical and safety orientation to insure safe execution of their duties. Refresher training is provided to personnel on a periodic basis.

B. Non-Licensed Operator Training

Non-Licensed Operator Training consists of two separate programs. The first program is the Auxiliary Operator Program which includes on-the-job training and classroom instruction in the categories of shift administration, plant design, and plant auxiliary systems.

The second program is the Waste Control Operator Program which includes classroom presentation and on-the-job training in shift administration, plant systems and design, and advanced radiation worker training.

C. Shift Engineer/Shift Technical Advisor Training

The Shift Engineer/Shift Technical Advisor (SE/STA) Training Program is a comprehensive program of study based on NUREG-0737 designed to provide an SE/STA candidate with a Bachelor's Degree or equivalent in a scientific or engineering

discipline with the necessary knowledge and training to perform the duties of an SE/STA. The program is divided into classroom presentations and on-the-job training in the following courses: General Employee Training, Plant Systems/Instrumentation and Controls, Plant Response and Analysis, Operating Procedures, Shift Administration, and Simulator Training.

Continuing training for Shift Engineers/Shift Technical Advisors consists of attending the Licensed Operator Continuing Training Program on the unit involved. A minimum of 40 hours of simulator training per year is provided. This 40 hours will include 20 hours of classroom training and 20 hours of training on the simulator.

D. Other Non-Licensed Training

Other areas of training are provided for non-licensed personnel as needed and include such areas as maintenance training (electrical, mechanical, and I&C) health physics training, radioactive waste training, chemistry training, and radiochemistry training.

12.2.3 RETRAINING PROGRAM

12.2.3.1 <u>Licensed Operator Continuing Training Program</u>

This program was established for the purpose of maintaining Licensed Operators at a level of knowledge and proficiency which is necessary for continued safe operation of the plant.

The Licensed Operator Continuing Training program for licensed personnel is based on a systems approach to training and is accredited by the National Academy of Nuclear Training. The initial accreditation of the training program took place in January, 1984. The new training program, which incorporated revisions required by changes to 10 CFR 55, was audited by the National Academy of Nuclear Training in January, 1988. The current program supercedes the previously existing NRC approved training program (certified to the Commission in accordance with Generic Letter No. 87-07 and the guidance given in NUREG-1262 in May, 1988).

12.3 EMERGENCY PLANNING

As part of the overall program of developing station procedures, an emergency plan, entitled the Arkansas Nuclear One Emergency Plan, has been developed. For further information see the Emergency Plan.

12.4 REVIEW AND AUDIT

A program for reviews, including in-plant and independent corporate reviews and audits, is established to provide for independent and objective evaluation of matters affecting the safe operation of ANO.

A complete description of this program is discussed within the EOI QA Program Manual.

ANO facility review and audit procedures are consistent with the requirements of Regulatory Guide 1.33.

12.5 PLANT PROCEDURES

12.5.1 ADMINISTRATIVE PROCEDURES

Detailed written procedures, covering administrative areas listed below, will be adhered to for the administrative control of safety-related activities of the plant staff.

A. Administrative procedures describing operator authority and responsibilities and definition of "Control Room."

B. Quality Control procedures, including, but not limited to, design equipment and material control.

C. Surveillance and testing requirements.

ANO administrative procedures are consistent with the requirements of Regulatory Guide 1.33.

EOI Headquarters administrative and section procedures provide guidance to the site to ensure consistent management direction on programs and processes. They also convey information to the site for specific requirements based on multi-site development of standardized programs and processes. The site maintains alignment with these procedures through direct implementation or implementation through site administrative and section procedures. All procedures issued from corporate headquarters that are implemented directly at the site and provide instructions in the areas of significant safety or management administrative controls are prepared and reviewed as delineated in Section 12.5.3.

12.5.2 OPERATING AND MAINTENANCE PROCEDURES

Detailed written procedures, covering operating and maintenance activities listed below, are adhered to for the operation and maintenance of all systems and components involving nuclear safety.

12.5.2.1 Operating Procedures

A. Plant Operation Procedures, including, but not limited to, the following activities:

   1. Plant Preheat and Precritical Check

   2. Plant Startup

   3. Power Operation

   4. Plant Shutdown and Cooldown

   5. Reactor Trip Recovery

B. Reactor Coolant System Operating Procedures, including, but not limited to, the following activities:

   1. Filling and Venting Reactor Coolant System

   2. Soluble Poison Control

3. Pressurizer Operation
4. Reactor Coolant Pump Operation
5. Quench Tank Operation
6. Draining and Blanketing Reactor Coolant System
7. Reactor Coolant Leak Detection
8. Reactivity and Heat Balance Calculation

C. Auxiliary System Operating Procedures, including, but not limited to, the following activities:

1. Volume Control System Operation
2. Decay Heat System Operation
3. Core Flooding System Operation
4. Reactor Building Spray Operation
5. Spent Fuel Cooling Operation
6. Circulating Water System Operation
7. Waste Control
8. Instrument and Service Air System Operation
9. Cooling Water System Operation
10. Diesel Generator Operations
11. Communications System Operation
12. Building Atmosphere Control

D. Instrumentation and Control Systems, including, but not limited to, the following activities:

1. Nuclear Instrumentation and Reactor Protection System Operation
2. Vibration and Loose Parts Monitor Operation
3. Safeguards Actuation System
4. Reactor and Feedwater Regulating System Operation
5. Non-Nuclear Instrumentation Operation

6. Control Rod Drive System

7. Computer Operation

E. Steam System Operating Procedures, including, but not limited to, the following activities:

1. Turbine Startup

2. Condenser and Vacuum System Operation

3. Condensate, Feedwater and Steam System Operation

4. Startup Boiler

5. Condensate Demineralizer System

F. Abnormal Operating Procedures, including, but not limited to, the following conditions:

1. Loss of Turbine Load

2. Loss of Reactor Coolant

3. Loss of Instrument Air

4. Loss of Service Water

5. Reactor Coolant Pump Emergencies

6. Loss of Feedwater

7. Remote Shutdown

8. Radiological Incidents

9. Evacuation

10. Natural Emergencies

11. Reactor Regulating System Abnormal Operation

12. Control Rod Drive Mechanism (CRDM) Malfunction

13. Loss of Containment Integrity

14. Waste Discharge Line High Radiation

15. Annunciator Corrective Action

16. Refueling Procedures

12.5.2.2 Maintenance and Radiation Protection Procedures

A. Radiation Protection Procedures.

B. Plant Instrumentation Calibration and Test Procedures.

C. Plant Maintenance Procedures, including, but not limited to:

   1. Pressurizer Maintenance

   2. Reactor Coolant Pump Maintenance

   3. OTSG Maintenance

   4. Reactor, Internals, Head and Rod Maintenance

   5. Auxiliary Systems Maintenance

   6. Refueling Maintenance Procedures

D. Security Procedures

12.5.3 PROCEDURE CONTROL

A description of requirements for procedure review and changes to procedures is given in Section 5.4.1 of the Unit 1 Technical Specifications.

12.6 INDUSTRIAL SECURITY

The Arkansas Nuclear One Industrial Security Plan was developed to accommodate the security of both Unit 1 and Unit 2. This plan is designed to provide the information described in Regulatory Guide 1.17 and covers basically three objectives:

A. To describe the physical security protection of Arkansas Nuclear One;

B. To define actions taken by individuals to prevent a security hazard; and,

C. To describe the preventive measures beyond physical barriers taken to minimize hazards.

Further details are given in the Arkansas Nuclear One Industrial Security Plan (Docket No. 50-313).

12.7 FIRE PROTECTION PLAN

(See Section 9.8.3)

12.8 TECHNICAL REQUIREMENTS MANUAL

The NRC's Final Policy Statement on Technical Specification Improvements for nuclear power plants and 10 CFR 50.36 allow certain requirements to be relocated from the Technical Specifications (TS) to other licensee controlled documents such as the SAR, ODCM, or administrative procedures. To provide an alternative central location for the requirements relocated from the TS and ensure that the necessary administrative controls are applied to these requirements, ANO developed a Technical Requirements Manual (TRM).

The TRM is an operator aid that provides a central location for relocated items. In addition to using the TS numbering and format for relocated items, the TRM provides a reference to the TS, when appropriate, to connect the relocated information to the applicable TS.

12.8.1 REGULATORY STATUS/REQUIREMENTS

The requirements contained in the TRM are part of the licensing basis. Failure to comply with a requirement of the TRM will be evaluated in accordance with the ANO corrective action program. This evaluation will consider effects upon equipment operability and applicable reporting requirements.

12.8.2 CHANGES TO THE TRM

The TRM is administered as part of the SAR. Changes to the TRM are subject to the criteria of 10 CFR 50.59. Administrative controls for processing TRM changes are included in the site procedures.

Table 12-1

MINIMUM SHIFT CREW COMPOSITION

This table has been intentionally deleted.

Technical Specification Generic Title	Corresponding Plant Specific Title
Shift Manager	Manager, Shift Ops
Plant Manager Operations	Sr. Manager, Operations
Operations Manager	Sr. Manager, Operations
Assistant Operations Manager	Manager, Ops - Support
	Manager, Ops - Shift
Radiation Protection Manager	Manager, Radiation Protection
General Manager	General Manager, Plant Operations

SAR FIGURE NO. 12-1

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ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

Plant Specific Titles for Generic Titles Located in the Technical Specifications

BASED ON DRAWING NO	SHEET	REV.

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13 INITIAL TESTS AND OPERATIONS

A comprehensive initial testing and operation program was conducted on Unit 1. The purpose of this program was: (1) to assure that the equipment and systems perform in accordance with design criteria, (2) to effect initial fuel loading in a safe, efficient manner, (3) to determine the nuclear parameters, and (4) to bring the unit to rated capacity.

The test program began as installation of individual components and systems was completed. The individual components and systems were tested and evaluated according to written test procedures. An analysis of the test results verified that each component and system performed satisfactorily.

The written procedures for the initial tests and operation included the purpose, conditions, precautions, limitations, prerequisites, and the acceptance criteria.

The NRC publication "Guide for the Planning of Preoperational Test Programs" and "Guide for the Planning of Initial Startup" were complied with in the preparation of these programs, with the exceptions as noted in Tables 13-3 and 13-4.

13.1 ORGANIZATION OF TEST PROGRAM

13.1.1 GENERAL ORGANIZATION

The organization for development and execution of the test program included participants from the Unit 1 operating personnel, the Production Department General Office staff, Babcock & Wilcox Company (B&W) Site Operations, and Bechtel Startup Group. Figure 13-1 shows the organization diagram and the relationship of the vendors and other support personnel to the AP&L operation organization.

The Unit 1 organization for the test program is shown in Figure 13-1.

The B&W Site Operations organization for the test program consisted of the Site Operations Manager, Site Operations Engineer, Consulting Nuclear Engineer, and Site Service Engineers who work in the specific areas of test procedures, testing, startup, operations, maintenance, fueling, field analysis, and reports. B&W Service Engineers with prior nuclear plant startup experience were on duty on each shift during the physics test program to provide support in physics measurements. The test program had technical support from B&W Nuclear Power Generation Division engineers. This support included technical analysis of the results of certain tests with the resulting analyses transmitted to Unit 1 through normal channels of communication. Special rapid channels of communication were utilized where results were needed as soon as possible for other operations to proceed. The qualifications for the B&W Site Operations organization are listed below:

A. The minimum qualifications for the B&W Site Operations Manager are:

   1. Graduate in engineering, or related physical science, or equivalent experience. (Two years experience for one year of college.)

   2. Four years of responsible power plant experience or two years of responsible nuclear reactor experience.

   3. One year engineering or test program preparation experience for this or similar nuclear plant.

B. The minimum qualifications for the B&W Site Operations Engineer are:

1. Graduate in engineering, or related physical science, or equivalent experience. (Two years experience for one year of college.)

2. Two years of responsible power plant experience or one year of responsible nuclear reactor experience.

3. One year engineering or test program preparation experience for this or similar nuclear plant.

Either the Site Operations Manager or Site Operations Engineer met the nuclear experience requirements of A.2 or B.2 above.

C. The minimum qualifications for the B&W Consulting Nuclear Engineer are:

1. A degree in nuclear engineering or physics.

2. A minimum of 10 real reactor startups, or past or current NRC operator (or senior) license or equivalent.

3. A minimum of 60 hours of simulator time including at least two plant startups from one percent $\Delta k/k$ shutdown or greater. Some portions of the above will be associated with simulation of typical startup (physics) tests.

4. A minimum of six months work on test preparation and planning for a B&W Nuclear Steam Supply System (NSSS). A minimum of 18 months experience in such areas as reactor physics, core measurement, core heat transfer, and core physics testing programs.

5. He will have either participated in the Oconee 1 startup testing (or some other large NSSS) and/or will have participated in the complete planning phases of low power and power escalation test programs.

Various individuals from the Arkansas Power & Light Company Production Department were to have furnished technical support, if needed, in specific areas. Similarly, individuals in the Bechtel Corporation were to have furnished technical support, as needed.

This support principally applied to the review of test procedures prior to approval, analysis of test results as requested by the Test Working Group, and the development and installation of modifications to the equipment and systems as required and identified during the test program. Lines of responsibility and authority of each member of the initial operation organization are indicated on Figure 13-1. Qualifications for Arkansas Power & Light Company personnel are contained in Appendix 1 of the FSAR.

During the initial criticality (including fuel loading) and post-criticality phases of the test program, the nuclear physics and thermal hydraulics aspects of the reactor operation were under the responsibility of the Operations Supervisor, with technical support from the Nuclear Engineer and the Technical Support Engineer, and with assistance from B&W Site Operations, B&W Nuclear Power Generation Division, Bechtel Startup Group, and Arkansas Power & Light Production Department as needed. A very close coordination between these groups existed with the appropriate support available when needed.

Prior to testing each system, the normal operating force was involved in operational procedure writing for each system. Upon release of the systems from construction, the operating force was used during the test to perform various operator duties. Following a successful test and acceptance of the system by AP&L, the operators assumed full responsibility for that system for the remainder of the test program and into commercial operation.

13.1.1.1 Qualifications

A. The qualifications of the Arkansas Power & Light Company personnel involved in the test program are contained in Appendices 1.1.5 and 1.1.6 of the FSAR.

B. The qualifications of the Bechtel Startup Group are as follows:

Robert J. Glover, Lead Startup Engineer, attended Yakima Valley College for two years and Washington State University for one year, majoring in Business Administration. He then attended J. M. Perry Institute for two years and graduated from that institution in 1950, majoring in Electronics, Instrumentation and Controls. Mr. Glover also completed three semesters at San Jose City College, studying Transistor Physics. While employed by General Electric from 1962-1967, he completed four GE-sponsored courses: 1) GE Reactor Technology, 2) Foxboro Instrument Training Course, 3) Leeds and Northrup Instrument Training Course, and 4) Geometric Dimensioning and Tolerancing.

Mr. Glover's pre-Bechtel experience includes nine years (from 1953-1962) with the Industrial Instrument Lab, where he owned and managed the Aircraft Instruments Service Company. From 1962-1967 he was employed by General Electric Company as a Nuclear Instrument Technician at Richland, Washington for three years and San Jose, California for two years.

His Bechtel experience includes: Hanna Mining Company for one-half year; Union Carbide Company for one-quarter year; Rochester Gas & Electric Co., Ginna Plant - 490 MWe, as an Instrument and Control Startup Engineer for one year and 10 months; Wisconsin Michigan Power Co., Point Beach Plant - 490 Mwe (Westinghouse Reactor) as Lead Startup Engineer for 18 months.

William McMahon, Electrical Startup Engineer, graduated from Specerian Business College, Newburg, New York in 1940 in Business Administration. From 1941-1948, he served in the U. S. Navy where he attended the U. S. Navy Diesel School qualifying in submarines. He also attended the Gyro Compass Electronics School while in the Navy. In 1952, Mr. McMahon graduated from Purdue University at Lafayette, Indiana with a Bachelor of Science degree in Electrical Engineering.

Mr. McMahon's pre-Bechtel experience includes the following positions with Allis Chalmers: Naval Engineering - Assistant Field Superintendant and Startup Engineer for 10,000 HP turbine; Corps of Engineers - Transformer Field Superintendent and Startup Engineer; Ohio Edison - Field Superintendent and Startup Engineer for 3,000 HP motor installation; Commonwealth Edison (Fisk Station) - Field Superintendent and Startup Engineer for Flooded Switchgear and Turbine Generator Rehabilitation; Air Force Wind Tunnel - Field Superintendent and Startup Engineer for 16,000 HP Synchronous Motors, Driving Fans, and Controls; Air Force - Stead, Beale and Norton Air Force Base - Field Superintendent and Startup Engineer for six

1,000 HP diesels. Other corporations with whom he has been employed are Chemtron Corporation, as Plant Engineer in Electrical Instrumentation and Control; Harvey Aluminum as Electrical Engineer for Solid State Controls; and Hunter Engineering Company, as Project Engineer and Plant Debugging Supervisor.

Since joining Bechtel in 1971, Mr. McMahon has been with the Power and Industrial Startup Division assigned to the Pilgrim Nuclear Power Station at Boston, Massachusetts, before being assigned to this job as Electrical Startup Engineer.

Donald B. Brimmer, Mechanical Startup Engineer, graduated from Trenton Technical Institute in 1964 with a degree in Mechanical Engineering. He did graduate work at Temple University from 1964-1966.

His pre-Bechtel experience includes part-time employment with Dubrow Electronics, Inc., 1962-1964. From 1964-1971, Mr. Brimmer was employed by DeLaval Turbine, Inc. as Lab Technician, Methods Engineer, and Field Service Engineer. Since joining Bechtel in 1971, he has been assigned to the Power and Industrial Startup Division as Mechanical Startup Engineer.

Mark J. Okey, Instrumentation and Control Engineer, graduated from Southern Colorado State College in 1964 with an Associate of Arts degree in Mathematics. He then attended the University of Colorado where he graduated in 1966 with a Bachelor of Science degree in Electrical Engineering. He attended the University of Idaho Graduate School for one semester in 1967. Mr. Okey also completed a 13-course program entitled "Fundamental Operations of Nuclear Test Reactors."

His pre-Bechtel experience includes Reactor Engineer for over 2 years with Idaho Nuclear Corporation. He was the Instrumentation and Control Startup Engineer for Westinghouse Electric Corporation, assigned to the Rochester Gas & Electric Company, Ginna Nuclear Project - 490 MWe - for 14 months and to Carolina Power & Light, Robinson Nuclear Unit - 760 MWe - for 15 months. Since joining Bechtel in 1970, Mr. Okey has worked in startup, assigned to Florida Power & Light, Turkey Point Units 3 & 4 - 760 MWe - where he served as Electrical Startup Engineer.

The qualifications of the B&W Startup Group are as follows:

J. N. Kaelin, Site Operations Manager, graduated from the University of Louisville in 1960 with a Bachelor of Science degree in Chemical Engineering. In 1965, he graduated from the U. S. Navy Officers Nuclear Training Program and qualified as engineering officer-of-the-watch for S1W and S5W Navy Nuclear Propulsion Plant. Mr. Kaelin's Nuclear Reactor experience includes 21 months - service (from 1965-1966) as an officer in the Engineering Department aboard the nuclear submarine USS John Marshall SSB(N)-611. Since joining B&W in 1967, he has also served as Startup Engineer, assigned to Oconee Nuclear Station, Unit 1, for one year.

Mr. Kaelin's test program experience includes 21 months (from 1967-1968) in the safety analysis work associated with Oconee Units 1 and 2 and Three Mile Island. Also, from 1968-1971 he participated in the test program phase of construction training associated with: Arkansas Nuclear One, Unit 1; Oconee, Unit 1; and, Three Mile Island, Unit 1.

R. K. Cothren, B&W Site Operations Engineer, has a Bachelor of Science Degree in Nuclear Engineering from North Carolina State University. In 1969, he received a Master of Science degree in Nuclear Engineering from North Carolina State University. His responsible nuclear reactor experience includes: 24 months Reactor Operator/Reactor Technician/Instructor experience at Nuclear Power Training Unit (AIW) Nuclear Reactor Testing Station, Arco, Idaho (1961-1963) and 11 months Reactor Operator experience aboard USS Enterprise (CVA(N)-65) (1963-1964). He also held an AEC operator's license for the North Carolina State University Research Reactor from 1967-1969.

Mr. Cothren has had 2½ years (1969-1971) of test program preparation and customer training associated with Oconee Unit 1, Rancho Seco Unit 1, and Arkansas Nuclear One, Unit 1. He is also a graduate of U. S. Navy Nuclear Power Training Program (1960-1961).

13.1.2 RESPONSIBILITIES

Plant Superintendent

The Plant Superintendent or his authorized representative had final responsibility for the overall test program which included the approval of the test procedures, major modification of test procedures, scheduling completion of the tests, and approval of the test results. Approval of test procedures, major modifications of test procedures, and approval of test results were not made without giving proper consideration to Babcock and Wilcox in their areas of interest through the Test Working Group.

Test Working Group

The Test Working Group (TWG) evaluated the test program and test procedures periodically and recommended to the Plant Superintendent any changes they felt should be made in order to fulfill requirements. The TWG reviewed and approved all test procedures before testing. After testing, the TWG reviewed test results and recommended to the Plant Superintendent the action the committee felt should be taken: (1) in resolving discrepancies or deficiencies that were found to exist, (2) in determining if acceptance criteria were met when this determination was not within the capabilities of the Station Test Coordinator to the Test Administrator, and (3) when, for any reason, the test was found to be unacceptable.

Representatives were from Unit 1 (Technical Support Engineer and Assistant Superintendent), Babcock & Wilcox (Site Operations Engineer and Test Program Coordinator) and Bechtel (Startup Engineer). Bechtel Engineering and Construction Departments had representatives participate as required. The Technical Support Engineer or the Assistant Plant Superintendent was chairman of the TWG. The TWG met at regular intervals during the most active phases of the program.

Station Test Coordinator

A Station Test Coordinator was assigned by the Station Superintendent for each test and coordinated by the Test Administrator. His responsibility was to coordinate the preparation for the performance of the test, to maintain a chronological test log of significant events and unusual conditions occurring during the test, to identify discrepancies between test results and acceptance criteria, and to have test results analyzed.

General Office Safety Review Committee

An audit of safety-related tests and their results was performed by the General Office Safety Review Committee.

13.1.3 RESOLUTIONS OF DISCREPANCIES

Any discrepancies or deficiencies found to exist during testing or upon review of test data after testing was completed were documented by itemizing discrepancies and deficiencies on the proper form and attaching the completed form to the official copy of the test procedure. The TWG reviewed the form and recommended to the Plant Superintendent the action the committee felt should be taken in resolving the deficiency. Retests were performed on systems and components, as necessary, to verify the adequacy of the corrective action.

Prior to any revisions relating to the health and safety of the public or plant personnel, structural integrity of plant components and systems, or items covered by codes and nuclear standards, review and approval by the Arkansas Power & Light Company Production Department, with assistance from vendors or consultants as necessary, were obtained.

13.2 TESTS PRIOR TO REACTOR FUEL LOADING

The tests prior to reactor fuel loading assured that systems were complete and operated in accordance with design. The test program was divided into two phases: Preheatup Test Phase and Hot Functional Test Phase. In many instances, systems were tested during both the Preheatup Test Phase and the Hot Functional Test Phase. A summary of the test program prior to fuel loading is listed in Table 13-I.

The types of tests are classified as hydro/leak, operational, electrical, and functional, with the following definitions for each classification:

Hydro/Leak Test - Structural integrity leak test of the various systems and components at the appropriate pressure.

Operational Test - Operation of systems and equipment under operating conditions.

Electrical Test - Consists of: grounding, megger, continuity, and phasing checks; circuit breaker operation and control checks; potential measurement and energizing of buses and equipment to insure continuity, circuit integrity, and proper functioning of electrical apparatus.

Functional Test - Tests to verify that systems and equipment will function as intended.

Instruments and controls of each system or component were also subjected to a preoperational instrumentation and controls calibration test prior to the initial operation of that system or component to assure proper operation.

An Engineered Safeguard Actuation System test was performed to assure actuation and proper operation of the Engineered Safeguards System and to evaluate the test method and frequency for future testing.

13.2.1 PREHEATUP TEST PHASE

The objective of the Preheatup Test Phase was to assure that the equipment and systems perform as required for hot functional testing. This phase of the testing included certain preoperational calibration, hydro/leak, operational, electrical, and functional tests as required. The reactor building containment system underwent a structural integrity and integrated leakage rate test to verify the building design and to insure that leakage was within the design limit.

13.2.2 HOT FUNCTIONAL TEST PHASE

The Hot Functional Test Phase was a period of hot operation of the Reactor Coolant System (RCS) and the associated auxiliary systems prior to the initial fueling of the reactor. The RCS was heated up to no-load operating pressure and temperature.

The Hot Functional Test Phase continued the preparation toward the initial fuel loading. The objectives of this phase of the test program are:

A. Operational test of systems, components, and non-nuclear instrumentation and controls at no-load operating pressure and temperature.

B. Operator training.

C. Verification of normal operating procedures, where practicable.

13.3 INITIAL CRITICALITY TEST PROGRAM

The initial criticality test program consisted of the initial fuel loading followed by initial criticality.

13.3.1 INITIAL FUEL LOADING

Fuel was loaded into the reactor in accordance with a step-by-step written procedure. This procedure contained a number of safety precautions and operating limitations.

The fuel loading procedure included:

A. A sequence of loading temporary detectors, sources, control rods, and fuel assemblies in order to maintain shutdown margin requirements.

B. The conditions under which fuel loading may continue after any given step.

C. An identification of responsibility and authority.

D. During any reactivity changes, a minimum of two detectors will be operating and indicating neutron level after the source has been inserted. At all other times, at least one detector will be indicating neutron level.

E. Two completely independent plots of reciprocal neutron multiplications as a function of the parameter causing reactivity change will be maintained.

F. Reactivity effects for each additional fuel assembly will be checked prior to the release of the fuel assembly by the fuel handling grapple.

G. An estimate of the reactivity effect for the next fuel addition will be made prior to insertion of the next fuel assembly.

H. The boron concentration in the reactor vessel, spent fuel pool, and Reactor Coolant System (RCS) will be maintained at a value to assure the required subcritical margin at all times.

I. The valve alignment of the auxiliary systems connected to the RCS will be checked periodically to prevent dilution of the reactor coolant boron concentration.

J. Chemical analysis and water level monitoring will be used to assure that inadvertent dilution of the reactor coolant boron concentration has not occurred.

K. Communication between control room and fuel handling areas will be maintained.

L. The plant Radiation Monitoring Systems will be in operation.

M. Health physics and chemistry monitoring services will be provided.

13.3.2 PREPARATION FOR INITIAL CRITICALITY

Upon completion of the initial fuel loading, pre-startup checks were completed prior to the approach to initial criticality. The pre-startup checks included:

A. Control rod trip test

B. Reactor coolant flow test

C. Reactor coolant flow coastdown test

A reactor coolant flow test and a reactor coolant flow coastdown test were conducted under cold reactor conditions to assure that the flow characteristics of the RCS were not materially changed as a result of the reactor core installation.

13.3.3 INITIAL CRITICALITY

A written procedure was followed during the approach to initial criticality. This procedure detailed the sequence to be followed, the limitations and precautions, the required plant status, and the prerequisite system conditions. This procedure also specified the alignment of fluid systems to assure controlled boron dilution and core conditions under which the approach to initial criticality proceeded.

Initial criticality was achieved by one of the following two methods:

A. Method one.

   1. Withdrawal of safety groups to insure at least one percent $\Delta k/k$ shutdown margin, including allowance for stuck rod.

   2. Boron dilution until criticality prediction indicates that initial criticality can be achieved by control rod group withdrawal with the controlling group between 25 percent and 75 percent during Step 3 below.

   3. Withdrawal of control rod groups in incremental steps until criticality is attained.

B. Method two.

   1. Begin with boron concentration sufficient for shutdown with all rods out.

   2. Withdraw rods in sequence until Rod Group 7 is between 25 percent and 75 percent withdrawn.

   3. Deborate until criticality is achieved.

Permissible rod group withdrawal and deboration were based on calculated reactivity effects. Two independent inverse neutron multiplication curves were maintained during rod group withdrawal and deboration. A predicted rod group position or boron concentration for criticality was determined before the next rod group withdrawal or deboration was started.

13.4 POST-CRITICALITY TEST PROGRAM

The post-criticality test program was performed to provide assurance that the plant was operating in a safe and efficient manner. Systems and components which could not be operationally tested prior to initial criticality were tested during the post-criticality test program to verify reactor parameters and to obtain information required for plant operation. A summary of the post-criticality testing is included in Table 13-2.

13.4.1 ZERO POWER PHYSICS TESTS

Following initial criticality, a program of reactor physics measurements was undertaken to verify the physics parameters. Measurements were made under zero power condition to verify calculated worths of control rod groups, moderator temperature coefficient, boron worth, and excess reactivity of the core. In addition, the response of the source and intermediate range nuclear instrumentation was verified.

Detailed written procedures were followed specifying the sequence of tests, parameters to be measured, and conditions under which each test was to be performed. These tests involved a series of prescribed control rod configurations and boron concentrations with intervening measurements of control rod and/or boron worth during boron dilution or boron injection.

13.4.2 POWER ESCALATION TEST PROGRAM

Following determination of the operating characteristics and physics parameters of the reactor at zero power, a detailed power escalation test program was conducted. This program consisted of specified incremental increases in power levels up to full power, with appropriate testing conducted at each power level. An analysis of the significant parameters at each step was made prior to initiating an additional power escalation.

At selected power levels, the following tests were performed:

A. Unit heat balance test

B. Power coefficient measurement

C. Core power distribution measurement

D. Unit load steady state test

E. Unit transient test

Other power escalation tests were performed at one or more power levels in the test sequence.

13.5 OPERATING RESTRICTIONS

During initial operations and associated testing, the normal plant safety procedures and Technical Specifications were in effect. In addition, special safety precautions and limitations were included in the test procedures and more restrictive operating limitations than those in the Technical Specifications were imposed where required from initial criticality through the power escalation program. The Reactor Protective System power level trip point was initially set at a low value and raised as the power escalation program progressed.

13.6 SUMMARY OF TESTS

SUMMARY OF REACTOR BUILDING STRUCTURAL INTEGRITY TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To verify the structural integrity of the reactor building at 115 percent of design pressure.

2. Prerequisites

   2.1 Completion of reactor building penetration leak tests.

   2.2 Verification of the leak tightness of the reference system.

3. Test Method

   The reactor building will be pressurized in steps up to a maximum of 115 percent of design pressure. At selected pressure levels, data will be recorded and inspections will be made to verify the structural integrity. After remaining at 115 percent of design pressure for approximately one hour, depressurization will begin. Data will be recorded at selected pressure levels during depressurization. During the test, reactor building, hatches, penetrations, and gaskets will be visually inspected.

4. Data Required

   4.1 Record readings of strain gauges, load cells, and deflection rods at selected pressure level.

   4.2 Record temperature and pressure of reactor building air and the temperature of the reactor building liner plate.

5. Acceptance Criteria

   Structure is capable of withstanding internal pressure of 1.15 times the design pressure described in Section 5.2.4.2.

SUMMARY OF REACTOR BUILDING LEAK TESTS FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To leak test each reactor building penetration and seal at calculated peak accident pressure or above.

   1.2 To measure integrated reactor building leakage rate at calculated peak accident pressure and at one-half calculated peak accident pressure.

2. Prerequisites

2.1 Penetrations are installed.

2.2. Penetrations tests completed before Integrated Leak Rate Test.

2.3 Reactor Building Cooling System operable.

2.4 Reactor building recirculation fans in operation with temporary filters installed.

2.5 Containment ventilation system, personnel lock, and isolation valves are all operable.

3. Test Methods

3.1 The reactor building leak rate will be determined at calculated peak accident pressure and at one-half calculated peak accident pressure. The reference vessel method and/or absolute pressure method will be used. The "known leak" and "pump back" techniques may be used to verify the results.

3.2 The penetration leak rates will be determined by pressurizing the volume between the appropriate boundaries to one-half the calculated peak accident pressure and observing the pressure decay.

4. Data Required

4.1 Integrated Leak Rate Test.

- Reactor building temperature, pressure, and humidity.
- Reference vessel temperature and pressure.
- Atmospheric pressure and temperature.
- "Known leak" and "pump back" air flow.
- Time.

4.2 Penetrations Leak Test.

- Pressure.
- Atmospheric pressure and temperature.
- Time.

5. Acceptance Criteria

The leak rates must not exceed the allowable limits given in Technical Specifications 5.5.16

SUMMARY OF REACTOR BUILDING COOLING SYSTEM ENGINEERED SAFEGUARDS TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To verify the engineered safeguards function of the reactor building coolers and related equipment.

2. Prerequisites

   The low pressure Service Water System shall be operable.

3. Test Method

   With the system in normal operation, an engineered safeguards signal will be inserted to initiate engineered safeguards operation.

4. Data Required

   4.1 Fan speed and air flow.

   4.2 Service water pressure drop across coolers.

5. Acceptance Criteria

   5.1 Idle fans start from auto position.

   5.2 Valves assume proper position.

SUMMARY OF REACTOR INTERNALS VENT VALVE INSPECTION TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To demonstrate use of internals vent valve exercise tool.

2. Prerequisites

   The reactor vessel closure head and the internals upper plenum will be removed.

3. Test Method

   For the exercise test, each valve disc will be cycled using the normal exercise tool.

5. Data Required

   5.1 The vent valve exercise tool can be used to successfully complete the inspection.

   5.2 Accessibility conditions are acceptable.

   5.3 Maximum opening force does not exceed 120 lbs. and the maximum force to hold wide open does not exceed 540 lbs.

SUMMARY OF CORE FLOODING SYSTEM FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To demonstrate that the flow path from the Core Flood Tanks to the reactor vessel is free from obstruction by discharging water from the Core Flood Tanks to the empty reactor vessel or through temporary piping. To demonstrate that the check valves in the Core Flooding System piping are operable subsequent to the Reactor Coolant System (RCS) hydrostatic test. To functionally test the Core Flood Tank level and pressure instrumentation. To obtain maximum valve performance data.

2. Prerequisites

   2.1 Core Flooding Tanks hydrostatic tests.

   2.2 Core Flood System instrumentation and preoperational calibration test.

   2.3 Core Flooding System valves electrical test.

3. Test Method

   Check level alarm actuations. Discharge tanks into reactor vessel to verify valve operability. Obtain stop and check valves maximum performance data. Record electrical data on electrically operated valves.

4. Data Required

   4.1 Record by hand the Core Flood Tank pressure, Core Flood Tank level, pressurizer level, and RCS pressure prior to the test and following the test.

   4.2 Readings of valve cycle time and valve full load amps will be recorded and retained as baseline data.

5. Acceptance Criteria

   5.1 Flow paths open to and from Core Flooding System.

   5.2 Core flood valves operable.

   5.3 Core Flood Tank level alarms actuate at setpoint ± 2 percent.

   5.4 Core Flood Tank pressure alarms actuate at setpoint ± 2 percent.

SUMMARY OF MAKEUP AND PURIFICATION SYSTEM ENGINEERED SAFEGUARDS TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To demonstrate emergency makeup (MU) injection flow to the RCS through each MU pump.

   1.2 To verify that operating times are within values of Chapters 6 and 15.

2. Prerequisites

   2.1 MU system functional test.

   2.2 Decay heat system hydrostatic test.

   2.3 Pressurizer relief valve test.

   2.4 RCS Hydrostatic Test.

   2.5 Reactor Coolant Pump initial operation test.

3. Test Method

   Each MU pump will be tested individually. RCS and MU system initial condition will be established. Engineered safeguard valves for flow path to be used will be assured to be in normal operating position. Upon simulation of an engineered safeguard signal to the MU system, the required valves and pumps responses will be observed and data will be recorded. When the RCS pressure reaches a prescribed limit or MU flow drops below a prescribed limit, the pumps will be stopped.

4. Data Required

   4.1 Record readings of emergency MU flow to loops A and B, pressurizer water level, and Borated Water Storage Tank (BWST) level.

   4.2 Record readings of RCS temperature for loops A and B, pressurizer temperature, BWST temperature, and makeup tank level.

5. Acceptance Criteria

   5.1 MU flow to the RCS $\geq$ 500 gpm at 600 psig RCS pressure.

   5.2 No abnormal conditions detected in MU pumps and engineered safeguards valves during tests.

SUMMARY OF DECAY HEAT REMOVAL SYSTEM FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To functionally test the operation of the components within the Decay Heat Removal (DHR) System.

2. Prerequisites

   2.1 BWST available for use and filled with water.

   2.2 RCS filled and fuel transfer canal available for filling.

   2.3 MU pumps available for operation.

3. Test Method

   Decay heat pumps will be operated: (a) in the DHR mode and (b) supplying Low Pressure Injection (LPI) water to the Makeup System.

4. Data Required

   4.1 RCS pressure.

   4.2 Decay heat and High Pressure Injection (HPI) flow.

   4.3 Decay heat and MU pump discharge pressure and suction pressure.

5. Acceptance Criteria

   DHR pumps meet normal system design flow requirements for the various flow paths and supply MU pumps at pressure above MU pumps NPSH requirements, as described in Section 9.5.

SUMMARY OF DECAY HEAT REMOVAL SYSTEM ENGINEERED SAFEGUARDS TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   Demonstrate emergency injection flow capability to the RCS from the Decay Heat Removal System.

2. Prerequisites

   2.1 DHR system functional test.

   2.2 Reactor coolant filled with approximately 50 psig nitrogen overpressure.

   2.3 Pressurizer level near 50 inches.

3. Test Method

All motor control breakers for engineered safeguards pumps and valves not under test will be opened. Initial conditions will be established and the decay heat pump will be lined up for injection. Data will be recorded for throttle flow and pressurizer level. The procedure will be repeated for the other decay heat pump.

4. Data Required

Recorders will record the following:

4.1 Pump flows.

4.2 Engineered safeguards valve positions.

4.3 Time of simulated engineered safeguards signal to either decay heat pump.

4.4 RCS pressure.

4.5 Pressurizer level.

5. Acceptance Criteria

5.1 DHR system components function properly.

5.2 Decay heat flow greater than or equal to 3,000 gpm to the reactor vessel with 100 psig pressure in the RCS.

SUMMARY OF REACTOR BUILDING SPRAY SYSTEM FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To functionally test the operation of the components within the Reactor Building Spray System.

2. Prerequisites

2.1 BWST available.

2.2 DHR suction header operable.

2.3 Reactor building vented to atmosphere.

2.4 Flow indicating sensors available for all spray nozzles.

3. Test Method

Flow will be established with each spray pump by recirculating water from the BWST, high and low alarms will be checked, and eductor performance from the spray additive tank will be verified.

4. Data Required

   4.1 Suction pressure and discharge pressure of each pump at different flow rates.

   4.2 Observe flow through each spray nozzle.

5. Acceptance Criteria

   5.1 Pump characteristic curve is in accordance with Figure 6-9.

   5.2 Spray nozzle paths are open.

SUMMARY OF SOLUBLE POISON CONCENTRATION CONTROL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To operationally test boric acid, change predictions used to control RCS boron level.

   1.2 To test the equipment used to change the boron concentration in the RCS.

   1.3 To verify the soluble poison concentration control process.

2. Prerequisites

   RCS and HPI system in operation.

3. Test Method

   Increase and decrease the boron concentration in the RCS using the required boron control equipment.

4. Data Required

   4.1 Boron concentration versus time in the RCS.

   4.2 Volumes and concentrations of required liquids.

5. Acceptance Criteria

   Boron concentration can be controlled, as required, to meet operational requirements.

SUMMARY OF SERVICE WATER SYSTEM FUNCTIONAL & OPERATIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To test the normal operation and engineered safeguards function of the Service Water System.

2. Prerequisites

   Service Water System operable.

3. Test Method

   Operate the system in normal and emergency modes.

4. Data Required

   Record differential pressure across the reactor building coolers and DHR coolers.

5. Acceptance Criteria

   Flow through reactor building coolers and DHR coolers, as measured by differential pressure, is equal to or greater than minimum required for engineered safeguards operation, as described by Section 9.3.

SUMMARY OF STEAM GENERATOR - SECONDARY SYSTEM HYDROSTATIC TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To hydrostatically test the non-isolable feed and steam lines connected to the secondary side of the steam generator.

2. Prerequisites

   2.1 Steam generator secondary side filled and vented.

   2.2 RCS filled and vented above design transition temperature.

3. Test Method

   Using hydro pumps, raise system pressure to code requirements, check system for leakage, and depressurize.

4. Data Required

   Pressure, temperature, and results of visual inspection.

5. Acceptance Criteria

   Conformance to USAS B31.1.0 and ASME III.

SUMMARY OF INTEGRATED CONTROL ROD DRIVE SYSTEM TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. <u>Objective</u>

   To ascertain that the Control Rod Drive System, with all related inputs, outputs, and system responses, is operating satisfactorily.

2. <u>Prerequisites</u>

   2.1 The Control Rod Drive System has been electrically tested.

   2.2 Reactivity control meets Technical Specifications limits if fuel is loaded.

3. <u>Test Method</u>

   The Control Rod Drive System will be exercised in all operating modes.

4. <u>Data Required</u>

   A detailed record will be maintained for each operational mode of the Control Rod Drive System and external equipment.

5. <u>Acceptance Criteria</u>

   System operates satisfactorily in all modes of control, as shown in Section 15.4.6 of the FSAR.

SUMMARY OF CONTROL ROD DRIVE TRIP TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. <u>Objective</u>

   To measure Control Rod Drive trip time from fully withdrawn to 3/4 control rod drive insertion and to verify Control Rod Drive attains fully inserted position.

2. <u>Prerequisites</u>

   2.1 Control Rod Drive System shall be operable.

   2.2 Reactor shutdown margin in accordance with Technical Specifications.

3. <u>Test Method</u>

   Withdraw each control rod or group to upper limit and trip.

4. <u>Data Required</u>

   4.1 Reactor coolant temperature, pressure, flow, and trip time for each Control Rod Drive.

   4.2 After each rod is tripped, verify that rods bottom.

5. Acceptance Criteria

Trip times do not exceed the value in the Technical Specifications, Section 15.4.6 of the FSAR.

SUMMARY OF PROCESS RADIATION MONITORING SYSTEM CALIBRATION & FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To adjust the system, fluid flows, alarms, self-checking features and interlock functions. To calibrate the system to primary radioactivity calibration data.

2. Prerequisites

2.1 Radiation monitoring system operable.

2.2 Process systems being monitored must be in operation before final adjustments can be made.

3. Test Method

3.1 Calibration of detectors - using secondary standard radioactivity sources, calibrate output to primary radioactivity calibration data.

3.2 Adjust alarms, interlocks, and recorders.

3.3 Adjust sampling parameters, including fluid flow rates and tape speeds. Verify sampling valve sequences and self-checking functions.

4. Data Required

4.1 Output from each channel for secondary standard radioactivity sources and installed operational check sources.

4.2 Alarm and interlock settings.

4.3 Fluid flow rates and flow alarm settings.

5. Acceptance Criteria

5.1 Outputs in agreement with primary calibration data.

5.2 Proper sampling, alarm, interlock, and self-checking functions.

SUMMARY OF AREA RADIATION MONITORING SYSTEM CALIBRATION & FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To adjust the system alarms, self-checking features, and interlock functions. To verify manufacturer's initial calibration.

2. Prerequisites

   Radiation Monitoring System operable.

3. Test Method

   3.1 Using standard radiation source, calibrate detectors.

   3.2 Adjust alarms, self-checking features, interlocks, and recorders.

4. Data Required

   4.1 Output from each channel for standard radiation source and installed operational check source.

   4.2 Alarm and interlock settings.

   4.3 Verification of self-checking features.

5. Acceptance Criteria

   5.1 Outputs in agreement with standard source radiation levels.

   5.2 Proper alarm, interlock, and self-checking functions.

SUMMARY OF REACTOR COOLANT HOT LEAKAGE TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To demonstrate that reactor coolant leakage, at hot pressurized system conditions, is within the limits set in the Technical Specifications.

2. Prerequisites

   2.1 Hydrostatic tests of systems which form the reactor coolant boundary are complete.

   2.2 The Makeup and Purification System and the RCS are operating as a closed system.

   2.3 The RCS is hot and pressurized.

3. Test Method

Calculate reactor coolant inventory change by measuring changes in pressurizer and letdown storage tank levels during a specified time interval, with correction for reactor coolant temperature change. The RCS will be visually inspected.

4. Data Required

Record RCS average temperature, pressurizer level, and MU tank level at the beginning and end of the time interval.

5. Acceptance Criteria

Reactor coolant leakage does not exceed limits set by the Technical Specifications.

SUMMARY OF CONTROL ROD DRIVE OPERATIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To demonstrate operation of the Control Rod Drive System in all modes of control at hot functional test conditions.

2. Prerequisites

2.1 Control Rod Drive System operable.

2.2 RCS at hot conditions.

2.3 Dummy drive guide assemblies in reactor vessel.

3. Test Method

3.1 Exercise Control Rod Drive System in all mechanisms' mode of control.

3.2 Continue periodic operation of Control Rod Drive Mechanism during hot functional test period. Verify trip functions after sustained mechanism operation.

4. Data Required

4.1 Control rod positions, stator temperature, and mode of control.

4.2 Record operational cycles imposed on each Control Rod Drive.

5. Acceptance Criteria

Control Rod Drive System operates satisfactorily at hot operational conditions outlined in the Technical Specifications.

SUMMARY OF ELECTRICAL SYSTEM NORMAL AND EMERGENCY OPERATION TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To prove normal operation of the station electrical system and to verify automatic switchover to emergency sources when required.

2. Prerequisites

   2.1 All normal and emergency power sources complete.

   2.2 Unit is on normal power source.

3. Test Method

   Simulate signals to the relays to cause transfer to each emergency power source.

4. Data Required

   4.1 Bus transfer time for each mode of transfer.

   4.2 Emergency Diesel Generator start.

   4.3 Sequential load pickup and load shedding.

5. Acceptance Criteria

   Successful automatic switchover to each emergency power source.

SUMMARY OF INTEGRATED ENGINEERED SAFEGUARDS ACTUATION TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To demonstrate the full operational sequence of the Engineered Safeguards Actuation System and to demonstrate the transfer to alternate power sources.

2. Prerequisites

   2.1 All systems actuated by the Engineered Safeguards System should have their individual functional tests completed.

   2.2 Engineered safeguards preoperational and calibration tests completed.

3. Test Method

   3.1 Actuate all engineered safeguards channels on the normal engineered safeguards power sources.

3.2 Actuate all engineered safeguards channels with concurrent simulated failure of all normal engineered safeguards power sources.

4. Data Required

4.1 Initial status.

4.2 Final status.

4.3 Engineered Safeguards System flows.

4.4 Diesel generator output.

5. Acceptance Criteria

The Engineered Safeguards System and electrical power systems shall respond as described in Chapter 6.

SUMMARY OF REACTOR PROTECTIVE SYSTEM FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To demonstrate functional performance of the Reactor Protective System.

2. Prerequisites

2.1 Reactor Protective System is operable.

2.2 Reactor shutdown margin in accordance with Technical Specifications.

3. Test Method

Utilizing installed test equipment, vary each trip parameter until channel trips.

4. Data Required

Trip value for each parameter.

5. Acceptance Criteria

5.1 Trip setting for each parameter is within tolerances of Technical Specifications.

5.2 Reactor Protective System functions as described in Chapter 7.

SUMMARY OF STATION EMERGENCY POWER FUNCTIONAL TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To verify that the emergency diesel units supply emergency power to Units 1 and 2.

2. Prerequisites

   2.1 Plant and diesel unit switchgear is set up for normal operation.

   2.2 Diesel units operable.

3. Test Method

   Each diesel unit is started and brought up to speed in the following way:

   3.1 Automatically from control room.

   3.2 Automatically from the diesel remote switchgear.

   3.3 Automatically by simulated engineered safeguards signal.

4. Data Required

   Time to synchronous speed and bus voltage at both units.

5. Acceptance Criteria

   Successful startup and automatic synchronization of both units in each mode.

SUMMARY OF REACTOR BUILDING PURGE SYSTEM ISOLATION TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To verify the isolation capability of the Reactor Building Purge System.

2. Prerequisites

   Reactor Building Purge System operable.

3. Test Method

   3.1 Open valves in purge line.

   3.2 Supply an engineered safeguards signal to isolate purge system.

4. Required Data

   Valve positions.

5. Acceptance Criteria

Upon engineered safeguards signal, correct valves close in purge line.

SUMMARY OF ZERO POWER PHYSICS TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To verify design physics parameters.

2. Prerequisites

2.1 The reactor core is installed.

2.2 The RCS is operable.

2.3 Reactor building integrity is established.

2.4 Precritical check list is complete.

3. Test Method

The reactivity effect of Control Rod Assembly (CRA), boron, and temperature changes will be measured.

4. Data Required

4.1 Initial criticality data.

4.2 Differential CRA group worths as a function of positions.

4.3 Differential boron worths as a function of boron concentration.

4.4 The "all Control Rod Assemblies out" boron concentration for excess reactivity data.

4.5 The temperature coefficients of reactivity.

4.6 The worth of single CRAs simulating an ejected CRA in the hot condition.

4.7 Verify stuck rod margin shutdown capability.

4.8 The total CRA worth for shutdown margin verification in the hot condition.

5. Acceptance Criteria

5.1 Applicable design features in Chapter 3 are verified.

5.2 Parameters in Technical Specifications are demonstrated.

SUMMARY OF NUCLEAR INSTRUMENTATION POWER CALIBRATION FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To calibrate the power range nuclear instrumentation indication to the thermal power level of the reactor, as determined by heat balance.

2. Prerequisites

   Nuclear instrumentation preoperational calibration and testing complete.

3. Test Method

   3.1 Obtain heat balance.

   3.2 Adjust channel gain until indicated power level agrees with heat balance.

   3.3 Repeat at various power levels up to full power.

4. Data Required

   4.1 Indicated power level.

   4.2 Channel gain.

   4.3 Heat balance data.

5. Acceptance Criteria

   The power range nuclear instrumentation indicates that the power calculated by the heat balance is within the limits of Section 7.3.1.3.

SUMMARY OF BIOLOGICAL SHIELD SURVEY TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To measure radiation in all accessible locations of the plant outside of biological shields.

   1.2 To obtain baseline radiation levels for comparison with future measurements of activity buildup with operation.

2. Prerequisites

   2.1 Radiation survey instruments calibrated.

   2.2 Background radiation levels measured in designated locations prior to initial criticality.

3. Test Method

Measure gamma, and neutron dose rates at low, intermediate, and full power.

4. Data Required

Power level, gamma, and neutron dose rates at each specified location.

5. Acceptance Criteria

Accessible areas and occupancy times during power operation are defined.

SUMMARY OF REACTIVITY COEFFICIENTS AT POWER FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

To measure the power Doppler and moderator coefficients of reactivity at selected power levels.

2. Prerequisites

Reactor is operating at power.

3. Test Method

3.1 Measure the power Doppler coefficient by determining the reactivity change associated with a change in power level at constant reactor coolant average temperature.

3.2 Measure the moderator coefficient by determining the reactivity change associated with a change in reactor coolant average temperature at constant power level.

4. Data Required

4.1 Power changes.

4.2 Temperature changes.

4.3 Reactivity changes.

5. Acceptance Criteria

The power Doppler and moderator coefficients are consistent with the limits specified in Section 3.2.2.1.5.

SUMMARY OF CORE POWER DISTRIBUTION TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To measure core power distribution using incore instrumentation.

2. Prerequisites

   2.1 Reactor at power.

   2.2 Incore Monitor System operable.

3. Test Method

   Read required data at selected conditions of reactor power level, xenon, and control rod position.

4. Data Required

   4.1 Incore instruments data.

   4.2 Reactor power.

   4.3 CRA positions.

   4.4 Power history.

   4.5 Reactor coolant pressure, temperature, and flows.

5. Acceptance Criteria

   Core power distributions are within design limits. Linear heat rate, Departure from Nucleate Boiling Ratio (DNBR), and quadrant power tilt are within limits of Technical Specifications.

SUMMARY OF REACTOR/TURBINE TRIP TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To measure unit response to a turbine trip.

   1.2 To measure unit response to a reactor trip.

2. Prerequisites

   Unit operating at power.

3. Test Method

   Trip reactor and turbine and record transient response from a steady state condition.

4. Data Required

   4.1 RCS flow, pressure, temperatures, pressurizer level, and power versus time.

   4.2 Steam temperature, pressure, steam generator water level, and feedwater flow versus time.

5. Acceptance Criteria

   No specified equipment or system limits exceeded.

SUMMARY OF ROD REACTIVITY WORTH TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   1.1 To measure differential CRA reactivity worths during zero power operations.

   1.2 To measure differential CRA reactivity worths during power operation.

   1.3 To determine differential boron worth during CRA measurements.

2. Prerequisites

   Reactor in critical mode.

3. Test Method

   3.1 Measure zero power differential CRA reactivity worths by withdrawals or insertions of specified CRAs and concurrently recording reactivity.

   3.2 Measure differential CRA reactivity worth at power by using withdrawal and immediate insertion technique to compensate for temperature feedback.

4. Data Required

   4.1 CRA positions.

   4.2 Reactivity.

   4.3 Temperature.

   4.4 Boron concentration.

5. Acceptance Criteria

   Maximum and minimum differential rod worths are within Section 3.2.2.1.6 limits.

SUMMARY OF NUCLEAR STEAM SUPPLY SYSTEM HEAT BALANCE FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To measure core thermal power.

2. Prerequisites

   Unit operating at power.

3. Test Method

   Core thermal power will be calculated from a best estimate heat balance using a weighted combination of primary and secondary heat balances at each stable power level during power escalation.

4. Data Required

   4.1 Feedwater flow, temperature, and pressure.

   4.2 Steam generator outlet pressure and temperature.

   4.3 Letdown flow, temperature, and pressure.

   4.4 Reactor Coolant Pump power.

   4.5 RCS flow, temperature, and pressure.

5. Acceptance Criteria

   The calculated core thermal power agrees with that calculated by computer within specified limits.

SUMMARY OF UNIT LOAD STEADY STATE TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To measure RCS and steam generator steady-state parameters as a function of reactor power, as described below.

   1.1 Determine turbine steam temperature versus reactor power.

   1.2 Measure RCS and steam generator parameters with selected Reactor Coolant Pumps operating.

2. Prerequisites

   All systems ready for power operation.

3. Test Method

RCS performance will be measured at selected power levels, as described below.

3.1 Establishing steady state at test power level with:

T_{ave} constant ± 2 °F

RCS pressure constant ± 50 psig

Reactor power constant ± 2 percent full power

3.2 Measuring power by heat balance.

3.3 Predicting trend of Nuclear Steam Supply System parameters from all measured steady state data for next selected power level.

4. Data Required

4.1 Reactor coolant pressure, temperatures, flows, pressurizer level, and reactor power.

4.2 Steam generator steam temperatures, pressures, levels.

4.3 Turbine header pressure.

4.4 Feedwater temperature and flow.

4.5 Valve positions in feedwater system and steam system.

4.6 Control rod positions.

4.7 Generated megawatts.

5. Acceptance Criteria

5.1 Results of measurements at each power level are sufficient to predict performance at the next power level.

5.2 Reactor outlet temperature increases and reactor inlet temperature decreases monotonically with increasing power level.

5.3 Specified unit parameters do not exceed equipment or safety limits.

SUMMARY OF THE UNIT LOAD TRANSIENT TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objectives

   1.1 To demonstrate plant capability to change power at rates less than or equal to design ramp rates after optimization of the Integrated Control System (ICS) at 100 percent of rated reactor power.

   1.2 To demonstrate plant capability to operate at 49 percent of rated power with one Reactor Coolant Pump in each loop operating.

   1.3 To demonstrate plant capability to operate at 75 percent of rated power with three Reactor Coolant Pumps operating.

2. Prerequisites

   Unit operating at power.

3. Test Method

   3.1 Perform load reduction and load increase at selected power levels between 15 and 100 percent power.

   3.2 Stop and restart Reactor Coolant Pumps to verify plant capability to operate at 49 percent full power with one pump in each loop and at 75 percent full power with three pumps running.

4. Data Required

   4.1 Reactor power.

   4.2 RCS pressure, pressurizer level, flow, and temperatures.

   4.3 Steam generator levels, temperature, and pressure.

   4.4 Steam header pressure.

   4.5 Feedwater flow and temperature.

5. Acceptance Criteria

   No equipment or system limits exceeded.

SUMMARY OF REACTOR COOLANT FLOW TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To demonstrate the flow characteristics of the RCS and the Reactor Coolant Pumps.

2. Prerequisites

   The reactor is shut down and all RCS parameters are satisfied for Reactor Coolant Pump operation.

3. Test Method

   Operate various combinations of Reactor Coolant Pumps during cold and hot conditions with the core installed.

4. Data Required

   Reactor coolant flow, pressure, and temperature.

5. Acceptance Criteria

   Flow under the various combinations must reach acceptable values for the power levels shown in Chapter 4.

SUMMARY OF REACTOR COOLANT FLOW COASTDOWN TEST FOR ARKANSAS POWER & LIGHT COMPANY, ARKANSAS NUCLEAR ONE

1. Objective

   To determine flow versus time after trip for various Reactor Coolant Pump combinations.

2. Prerequisites

   2.1 Reactor coolant flow is established with desired pumps.

   2.2 The reactor is shut down.

3. Test Method

   For each desired operating pump combination, trip pumps and measure reactor coolant flow versus time.

4. Data Required

   4.1 Record reactor coolant initial temperature, flow, and pressure.

   4.2 Record time of Reactor Coolant Pump trip and flow during coast down.

5. Acceptance Criteria

   Flow during coastdown is equal to or greater than the calculated minimum allowable flow required to preserve the minimum allowable DNBR.

Table 13-1

TESTS PRIOR TO INITIAL FUEL LOADING

Test Title:		Summary Page No.
1.	Reactor Building Structural Integrity Test	13.6-1
2.	Reactor Building Leak Tests	13.6-1
3.	Reactor Building Cooling System Engineered Safeguards Tests	13.6-3
4.	Reactor Internals Vent Valve Inspection Test	13.6-3
5.	Core Flooding System Functional Test	13.6-4
6.	Makeup and Purification System Engineered Safeguards Test	13.6-5
7.	Decay Heat Removal System Functional Test	13.6-6
8.	Decay Heat Removal System Engineered Safeguards Test	13.6-6
9.	Reactor Building Spray System Functional Test	13.6-7
10.	Soluble Poison Concentration Control Test	13.6-8
11.	Service Water System Functional and Operations Tests	13.6-9
12.	Steam Generator Secondary System Hydrostatic Test	13.6-9
13.	Integrated Control Rod Drive System Test	13.6-10
14.	Control Rod Drive Trip Test	13.6-10
15.	Process Radiation Monitoring System Calibration and Functional Test	13.6-11
16.	Area Radiation Monitoring System Calibration and Functional Test	13.6-12
17.	Reactor Coolant Hot Leakage Test	13.6-12
18.	Control Rod Drive Operational Test	13.6-13
19.	Electrical System Normal and Emergency Operation Test	13.6-14
20.	Integrated Engineered Safeguards Actuation Test	13.6-14
21.	Reactor Protective System Functional Test	13.6-15
22.	Station Emergency Power Functional Test	13.6-16
23.	Reactor Building Purge System Isolation Test	13.6-16

Table 13-2

POST-CRITICALITY TESTING SUMMARY

		Summary Page No.
1.	Zero Power Physics Test	13.6-17
2.	Nuclear Instrumentation Power Calibration Test	13.6-18
3.	Biological Shield Survey Test	13.6-18
4.	Reactivity Coefficients at Power Test	13.6-19
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6.	Reactor/Turbine Trip Test	13.6-20
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9.	Unit Load Steady State Test	13.6-22
10.	Unit Load Transient Test	13.6-24
11.	Reactor Coolant Flow Test	13.6-25
12.	Reactor Coolant Flow Coastdown Test	13.6-25

Table 13-3

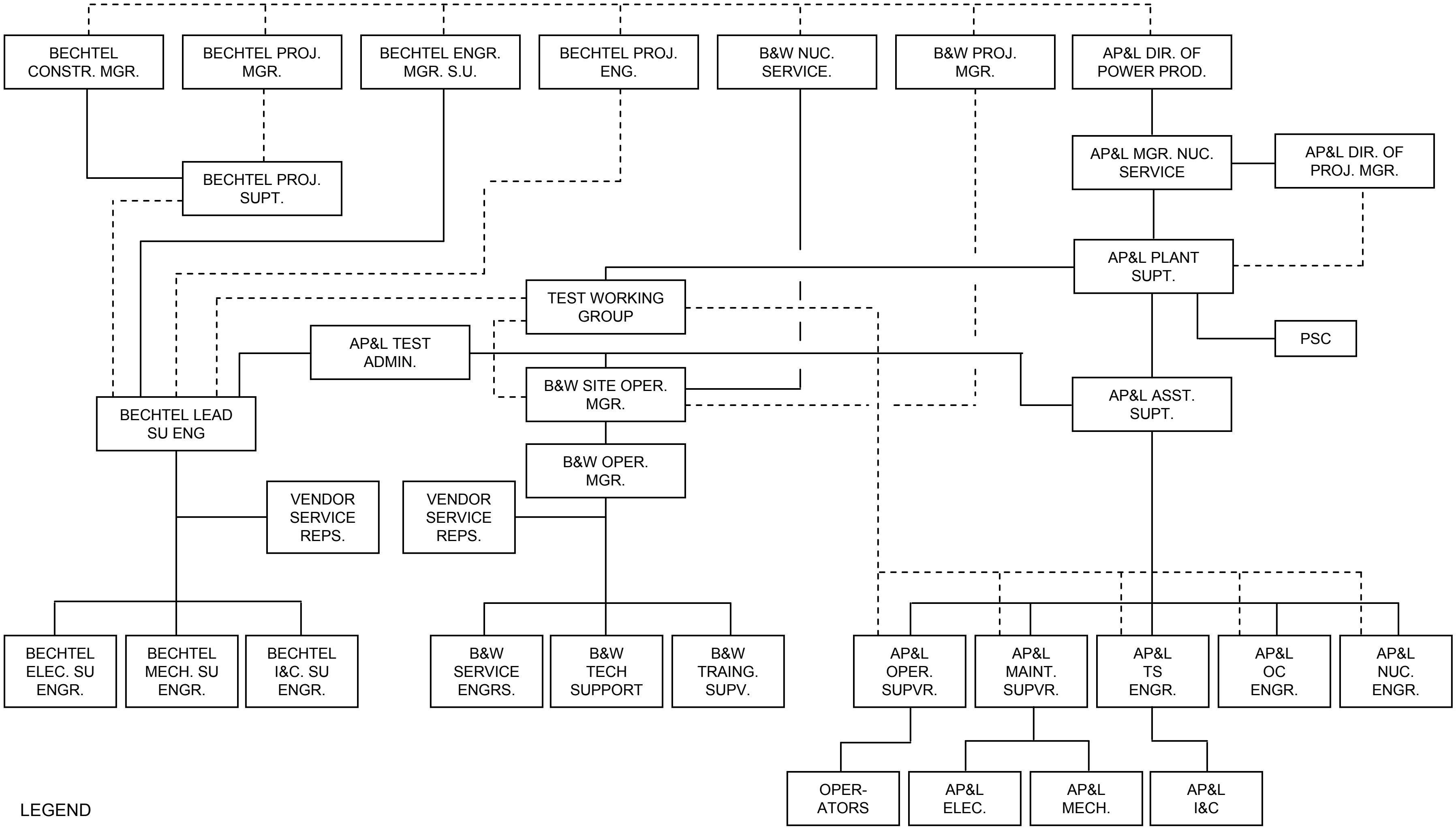
EXCEPTIONS TO THE GUIDE FOR THE PLANNING OF PREOPERATIONAL TEST PROGRAMS

Guide Section	Exception
I.A.1	No formal test; components will be tested during system functional tests.
I.C.1	No formal test; intent covered by vendor tests.
II.E.3	Not applicable; for BWRs only
IV.A.1	No formal test
IV.A.2	No formal test
V.L.1	No formal test; each individual air operated valve listed in Tables 9.71-1 and 9.71-2 of the FSAR is tested for loss of instrument air in the test of the system under which the valve is scoped.
V.M	Not applicable; for BWRs only
V.N	Not applicable; no reactor vessel head cooling system
V.O	Not applicable; no shield cooling system
VII.B.1	Not applicable to plant design
VII.D	Not applicable; for BWRs only
VII.E.4	Not applicable; no ice condenser
A.2	No formal test; components will be tested during system functional tests
XII.B, XII.C	No formal tests; calibration procedures will be written and data sheets for each instrument will be maintained by health physics personnel.

Table 13-4

EXCEPTIONS TO THE GUIDE FOR THE PLANNING OF INITIAL STARTUP

Guide Section	Exception
General	Major test plateaus are to be 15% FP, 40% FP, 75% FP, and 100% FP, rather than 25% FP, 50% FP, 75% FP and 100% FP.
F.1.j	No formal test; prototype testing done at Oconee I; internals will be visually inspected after hot functional testing.
H.1.e	No formal test; prototype testing done at Oconee I.
H.1.f	Control rod reactivity worth determinations will be done on rod groups only, not on individual rods.
J.1.a	No formal test; prototype testing done at Oconee I.
J.1.h	No formal test; prototype testing done at Oconee I.
J.1.p	No formal test; prototype testing done at Oconee I.
J.1.b	No formal test at the 15% FP (25% in Guide) plateau.
J.1.j	No formal test at the 40% FP (50% in Guide) plateau.
J.1.d	To be conducted at 40% FP rather than 15% FP (25% in Guide).
J.1.k	To be performed at the 15% FP plateau.
J.1.1	To be performed at the 15% FP plateau.
J.1.q	To be performed at the 40% FP plateau.
J.1.o	To be performed at the 40% FP plateau.



LEGEND

- - - - ADVISORY INFORMATIN

——— DIRECT RESPONSIBILITY

STARTUP ORGANIZATION CHART

SAR FIGURE NO. 13-1

AMENDMENT 20

ARKANSAS NUCLEAR ONE

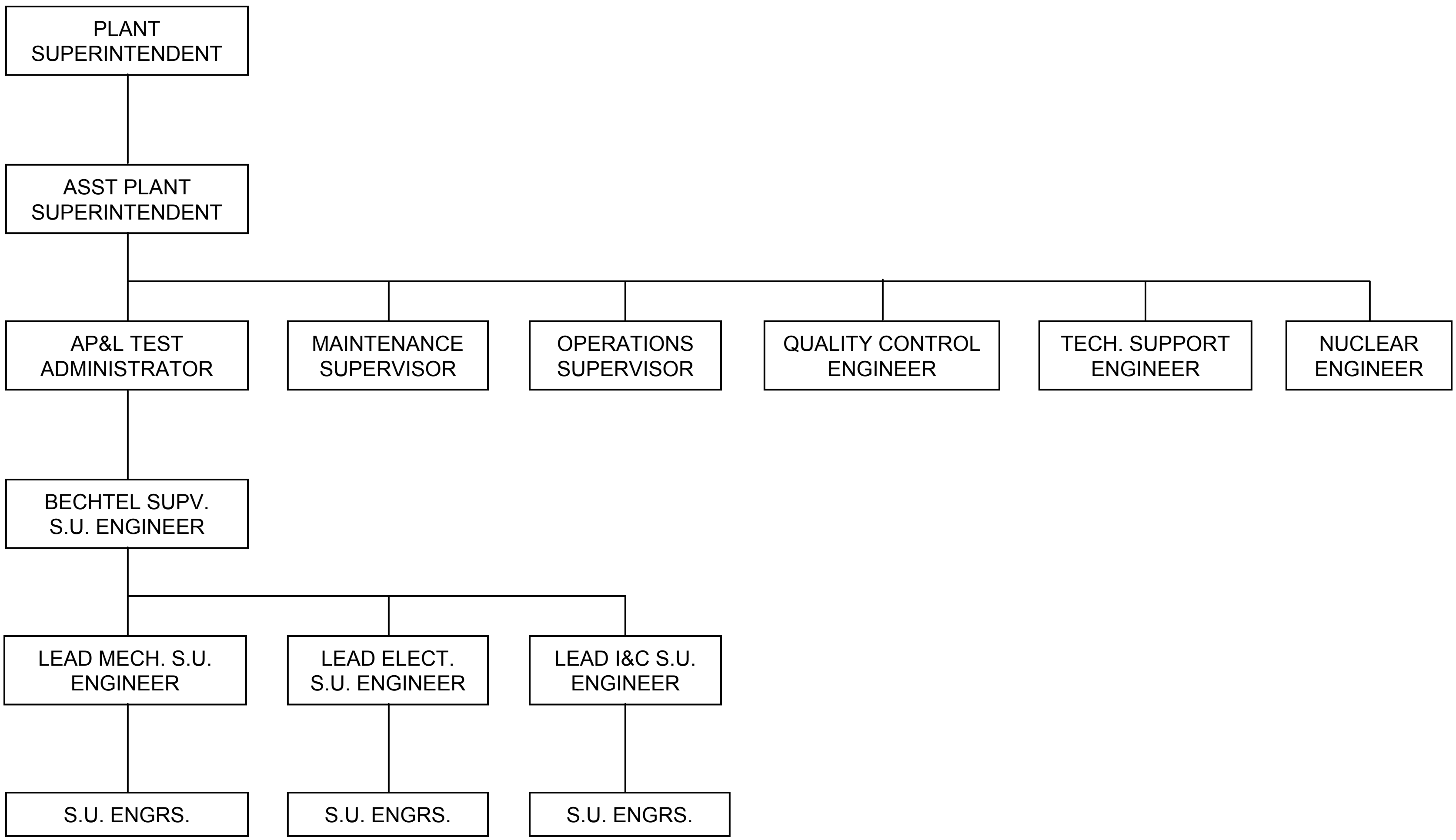
UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



PLANT ORGANIZATION – PRE-OPERATIONAL TESTING

SAR FIGURE NO. 13-2

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

CHAPTER 14

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UPDATE REFERENCE LIST

Sections and reference listed below denote documents that contain additional cross reference information used to update the SAR.

Sections	Cross Reference
14.2.2.5.2	Correspondence from Trimble, AP&L, to Reid, NRC, dated December 12, 1979. (1CAN127908)
14.1.2.9.5	Correspondence from Trimble, AP&L, to Stolz, NRC, dated September 24, 1981. (1CAN098109)
14.1 14.2	Design Change Package 80-1083, "Emergency Feedwater Initiation.”
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14.2.2.5.5	AP&L Engineering Report 88E-0105, Effects of Revised LOCA Curves on the Environmental Qualifications of Equipment Located Inside Containment.
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14.2.2.5.4.2	Design Change Package 89-1012B, "Redundant High Pressure Injection Lines"
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14 SAFETY ANALYSIS

Since the performance of the original analyses in this chapter, additional analyses were performed to demonstrate plant safety with regard to response to postulated events. These analyses caused the Technical Specifications, operating procedures, and, in some cases, the design to be changed in an effort to continue to confirm the results of this chapter. Chapter 3A, “Reload Report”, documents the effective cycle for the analysis of record for each analyzed event.

At the end of Cycle 19, the original OTSGs were replaced. In support of Cycle 20 operation, evaluations of the FSAR Chapter 14 events were performed to verify that the LOCA and non-LOCA analyses of record remain bounding for ANO-1 following steam generator replacement. The results of these evaluations as well as other safety analyses affecting the licensing basis are provided in References 94 and 95. It was concluded that all analyses that represent the current licensing basis in Chapter 14 of the ANO-1 FSAR remain valid and bounding following steam generator replacement.

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1 ABNORMALITIES

Both normal and abnormal operations of the various systems and components are discussed in prior sections of this report. This section summarizes and further explores abnormalities that are inherently terminated or require operation of the normal protection systems to maintain the integrity of the fuel and/or the Reactor Coolant System (RCS). These abnormalities have been evaluated for a rated core power of 2,568 MWt. Fission product dispersion in the atmosphere is assumed to occur as predicted by the dispersion models developed in Section 2.3. Table 14-1 summarizes the potential abnormalities studied.

14.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.1.2.1 Uncompensated Operating Reactivity Changes

The UORC (Uncompensated Operating Reactivity Changes) analysis was suggested for inclusion into the FSAR through AEC Regulatory Staff guidance when ANO-1 was licensed for operation. Subsequent Regulatory Staff guidance dropped this type of "abnormality" from the list of those events which should be analyzed. These normal operating phenomena analyzed as operating reactivity changes result in benign effects, especially in light of the normal plant response. Due to the Regulatory Staff's discontinued interest in the UORC analysis and the capability of the plant to compensate for greater reactivity perturbations than those in the scope of this analysis without the initiation of protection systems, the UORC should not be looked upon as having the same safety significance as the remainder of events analyzed in the SAR. It is provided in this section of the SAR, however, for historical information.

14.1.2.1.1 Identification of Cause

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion, xenon burnout, and changes in fission product poison concentration and burnable poison depletion. These slow reactivity changes, if left uncompensated, could cause the operating limits to be exceeded. In all cases, however, the normal operation of the ICS (Integrated Control System) would compensate for any reactivity perturbations due to these

normal core operating phenomena. If the ICS failed to operate properly and the operator failed to respond to the reactivity perturbations, the RPS (Reactor Protection System) would prevent the safety limits from being exceeded.

14.1.2.1.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are that the rate of reactivity addition or temperature change will be much less than the rate at which the automatic control system can compensate for the change.

14.1.2.1.3 Methods of Analysis

During normal operation, the Integrated Control System (ICS) senses any reactivity change in the reactor. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the average temperature of the RCS increases or decreases and the ICS acts to restore reactor power to the power demand level and to re-establish this temperature at its setpoint. If manual corrective action is not taken, or if the automatic control system malfunctions, then the RCS's average temperature changes to compensate for the reactivity disturbance. In the analysis it is assumed that the secondary system follows the temperature changes in the RCS.

14.1.2.1.4 Results of Analysis

Table 14-2 summarizes the reactivity changes and the corresponding change in the average moderator temperature. These reactivity changes are extremely slow and allow time for the operator or the ICS to detect and compensate for the change.

14.1.2.2 Startup Accident

14.1.2.2.1 Identification of Cause

The objective of a normal startup is to bring a subcritical reactor to the critical or slightly supercritical condition and then to increase power in a controlled manner until the desired power level and system operating temperature are obtained. During a startup, an uncontrolled reactivity addition could cause a nuclear excursion. This excursion is terminated by the strong negative Doppler effect if no other protective action operates.

The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential power excursions.

A. The control system is designed so that only one control rod group can be withdrawn at a time, except that there is a 20 +/-5 percent overlap in travel between two regulating rod groups successively withdrawn. This overlap occurs near the minimum worth positions for each group since one group is near the end of travel and the other is near the beginning of travel. The maximum total calculated worth of the control rods when the reactor is critical is specified in Table 14-3. See Chapter 3A, “Reload Report”, and its supporting documentation for the most current calculations.

B. Control rod withdrawal rate is limited.

C. A short-period withdrawal stop and alarm are provided in the source range.

D. A short-period withdrawal stop and alarm are provided in the intermediate range.

E. A high flux level and a high pressure trip are provided in the power range.

14.1.2.2.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Reactor thermal power shall not exceed 112 percent of rated power.

B. RCS pressure shall not exceed code pressure limits.

14.1.2.2.3 Methods of Analysis

A B&W digital computer model of the reactor core and RCS was used to determine the characteristics of this accident. This model used 102.67% of design reactor coolant flow but no sprays in the pressurizer. Steady-state heat transfer out of the RCS is established for the initiation of the transient, and it is held constant throughout the transient. A conservative (less negative) Doppler coefficient was used such that the Doppler coefficient is much larger (more negative) than this for actual plant operation. The rods were assumed to be moving out along the steepest part of the rod worth versus rod travel curve. The values of the principal parameters used in this analysis are listed in Table 14-3.

In addition, the criterion for minimum movable control rod worth is that a shutdown margin of one percent $\Delta k/k$ at the hot standby condition is required (Section 3.1.2.2). The startup accident has been analyzed using the minimum tripped rod worth with the maximum worth stuck rod as part of the analysis. The startup accident was analyzed from the hot, pressurized condition.

14.1.2.2.4 Results of Analysis

Figure 14-1 shows the results of the reactivity addition rate that results in the peak pressure and thermal power. The Doppler effect terminates the neutron power (neutron power is defined as the total energy release from fission) rise, but the heat input to the reactor coolant increases the pressure past the trip point and the transient is terminated by the high pressure trip.

Figure 14-2 shows the results of withdrawing all Control Rod Assemblies (CRAs) at the maximum speed from 10^{-7} percent of rated power. This results in a maximum possible reactivity addition rate. The total rod worth used in this analysis is slightly greater than the calculated worth (Table 3-5). The power rise is terminated by the negative Doppler effect. The high neutron flux trip takes effect after the peak neutron power is reached and terminates the transient. The peak thermal heat flux is significantly less than the rated power heat flux.

A sensitivity analysis was performed for the peak pressure and thermal power reached during a startup accident (protected by the high pressure trip) to determine the effect of varying several key parameters.

Variation of the high pressure trip delay time from 0.55 to 0.7 second resulted in a change in peak thermal power of less than one percent of rated power and a change in the peak pressure of 10 psi.

Figure 14-3 shows the effect of varying the reactivity addition rate on the peak thermal power. This reactivity rate was varied from more than an order of magnitude below the single rod group rate to a rate slightly above that for simultaneous withdrawal of all rods. The slower rates will result in the pressure trip being actuated. Only the very fast rates actuate the high neutron flux level trip.

Figure 14-6 shows the peak thermal power and peak pressure variation as a function of moderator coefficients for a range of reactivity addition rates that result in worst case peak pressure and thermal power.

The peak RCS pressure was found to be dependent on the initial pressurizer level. Higher initial pressurizer levels result in less volume to accommodate the expansion of the RCS volume due to the heat input caused during the startup event. An analytical limit on initial pressurizer level of 320 inches was found to produce acceptable RCS pressure results, provided that the lift tolerance on the PSV setpoint is +1% versus the +3% assumed in the base startup analysis.

Figure 14-8 shows the effect of varying the reactivity addition rate on peak pressure.

Table 14-4 summarizes the results of the postulated startup accidents.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach the design overpower condition and the peak pressure never exceeds code allowable limits.

14.1.2.3 Rod Withdrawal Accident at Rated Power Operation

14.1.2.3.1 Identification of Cause

A rod withdrawal accident pre-supposes an operator error or equipment failure resulting in accidental withdrawal of a control rod group while the reactor is at rated power. As a result, the power level increases, the reactor coolant and fuel rod temperatures increase, and, if the withdrawal is not terminated by the operator or the protection system, core damage would eventually occur.

The following provisions are made in the design for the indication and termination of this accident.

A. High reactor coolant outlet temperature alarms.

B. High RCS pressure alarms.

C. High pressurizer level alarms.

D. High reactor coolant outlet temperature trip.

E. High RCS pressure trip.

F. High power level, i.e., neutron flux level, trip.

14.1.2.3.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Reactor thermal power shall not exceed 112 percent of rated power.

B. RCS pressure shall not exceed code pressure limits.

14.1.2.3.3 Methods of Analysis

A B&W digital computer code was used to determine the characteristics of this accident. Included were a complete kinetics model, pressure model, average fuel rod model, steam demand model with secondary coastdown to decay heat level, coolant transport model, and a simulation of the instrumentation for pressure and flux trip. The initial conditions were normal rated power operations without automatic control. Only the Doppler and moderator coefficients of reactivity were used as feedback. The nominal values used for the main parameters in evaluating the accident are given in Table 14-5. For trip, the minimum control rod worth that satisfies the criterion for a shutdown margin of one percent $\Delta k/k$ at the hot standby condition is used throughout the analysis.

14.1.2.3.4 Results of Analysis

Figure 14-9 shows the results of the nominal rod group withdrawal from rated power. The transient is terminated by a high neutron flux level trip and the reactor thermal power is limited to well below the design overpower. The changes in the parameters are quite small, as shown in Table 14-6.

A sensitivity analysis of important parameters was performed around this nominal case and the resultant RCS pressure responses are shown in Figures 14-10 through 14-13.

Figure 14-10 shows the pressure variation for a very wide range of rod withdrawal rates, about an order of magnitude smaller and greater than the nominal case. For the very rapid rates, the neutron flux trip is the primary protective device for the reactor core. It also protects the system against high pressure during fast rod withdrawal accidents. The high pressure trip is relied upon for the slower transients. In no case does the thermal power exceed the design overpower.

Figures 14-11 through 14-13 show the pressure response to variations in the trip delay time, Doppler coefficient, and moderator coefficient. In all cases the neutron flux level trip is actuated.

An analysis has been performed extending the evaluation of the rod withdrawal accident for various fractional initial power levels up to rated power. In this evaluation, simulated withdrawal of all control rods was assumed, giving the maximum possible reactivity addition rate. This rate is significantly higher than that used for the cases evaluated for withdrawal of a single group (Table 14-5). The results of this analysis are shown in Figures 14-14 and 14-15.

As seen in Figure 14-14, the peak thermal power occurs for the rated power case and is below the design overpower. The peak neutron power for all cases slightly overshoots the assumed trip level. Figure 14-15 shows that the maximum fuel temperatures reached in the average rod and the hot spot are well below melting. Even in the most severe case at rated power, the average fuel temperature increases only a few degrees. Therefore, it is readily concluded that no fuel damage will result from the simultaneous withdrawal of all rods from any initial power level.

This analysis demonstrates that the high flux level trip and the high pressure trip adequately protect the reactor against any rod withdrawal accident from power.

14.1.2.4 Moderator Dilution Accident

14.1.2.4.1 Identification of Cause

Boron, in the form of boric acid, in the reactor coolant controls excess reactivity. During power operation the boron content of the reactor coolant is periodically reduced to compensate for fuel burnup. The dilution water is supplied to the RCS by the Makeup and Purification System. This system has several interlocks and alarms to prevent improper operation as described in Sections 7.2.2.2 and 9.1.2.6. These are as follows:

A. Flow of dilution water to the makeup tank must be initiated by the operator. During power operation, continuous dilution of the moderator can be performed only with the control rods withdrawn to preset bands to insure that there is adequate negative reactivity available to render the core subcritical in the event of inadvertent moderator overdilution. When these conditions are met, the operator can place the 3-way valve in the "bleed" position and, simultaneously, add dilution water to the makeup tank.

B. If the above conditions are not met, the 3-way valve cannot be placed in the "bleed" position while adding dilution water. Either of the valves may be operated separately, i.e. feed or makeup may be added or inventory may be bled from the system to compensate for temperature increase, but they may not be operated simultaneously. Dilution of the moderator by this method is a very slow process due to the relatively small quantities of dilution water that could be added at a time.

C. The operator must preset a batch size and start the batch controller before the makeup valve will open to admit water to the Makeup and Purification System. Dilution water is added at flow rates up to the maximum makeup rate at operating pressure.

D. Flow of dilution water is automatically stopped when the flow has integrated to a preset value or when the rods have been inserted to a preset position (at 75 percent full stroke).

E. A continuous dilute permit light and 3-way valve position lights on the console are on whenever continuous dilution is in progress. Information concerning dilution is available to the operator and the dilution can be terminated by the operator at any time.

The values of system parameters used in evaluating this accident from power operation conditions are listed in Table 14-7. The Makeup and Purification System normally has one pump in operation, which supplies makeup to the RCS and the required seal flow to the Reactor Coolant Pumps. Thus, the total makeup flow available is normally limited by pump capacity. When the makeup rate is greater than the letdown rate, the net water increase will cause the pressurizer level controller to close the makeup valves. The nominal moderator dilution event considered is the pumping of water with zero boron concentration from the makeup tank to the RCS.

It is possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal.

Furthermore, with a combination of multiple valve failures or maloperations plus more than one makeup pump operating with reduced RCS pressure, the resultant inflow rate could be much higher than the normal rate (Table 14-7). This constitutes the maximum dilution accident. A reactor trip would terminate unborated water addition to the makeup tank and makeup flow into the coolant system would be terminated by high pressurizer level.

During refueling or maintenance operations when the reactor closure head has been removed, the RCS boron concentration is procedurally controlled to assure a minimum SDM that is greater than the change in reactivity that would result from a dilution event. In these conditions, the sources of dilution water to the makeup tank and therefore to the RCS are isolated, and the makeup pumps are not operating. To ensure the ability of the reactor to tolerate a moderator dilution during shutdown, the consequences of accidentally filling the makeup tank with dilution water and starting the makeup pumps are evaluated. The results of this evaluation demonstrate that the Technical Specification required refueling boron concentration is sufficient to protect the reactor from a dilution event. Otherwise, the evaluation establishes a new lower limit, above the Technical Specification required minimum, for the refueling concentration (other criteria for a higher minimum concentration may also be applied).

14.1.2.4.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Reactor thermal power shall not exceed 112 percent of rated power.

B. RCS pressure shall not exceed the code pressure limit of 2750 psig.

C. The dilution transient shall be terminated before the shutdown margin is eliminated. For a dilution event during refueling, the core shall remain subcritical.

14.1.2.4.3 Methods of Analysis

The reactor is assumed to be operating at rated power with a maximum initial boron concentration in the RCS, as given in Table 14-7. The dilution water is uniformly distributed throughout the reactor coolant volume because of the very small discharge rate of makeup flow into the very large reactor coolant flow. The analysis assumes beginning of life conditions to maximize the initial boron concentration and provide a conservative core power response. The moderator coefficient is assumed to be positive although plant operations preclude full power operation with a positive moderator temperature coefficient.

For the cases at power, the rate of boron concentration change in the RCS was calculated for each flow rate. Coupling these results with the inverse boron worth, the reactivity insertion rates for each flow rate were determined. The reactivity insertion rates were used with a digital computer code to predict the plant response to a dilution event. The code is a simplified model of the RCS that can model reactivity changes and predict the expected changes in reactor coolant temperatures and the pressurizer pressure.

The evaluation of the dilution during a refueling accident demonstrates that the Technical Specification required refueling boron concentration is sufficient to protect the reactor from a dilution event. This evaluation is performed for each new fuel cycle. The Technical Specification refueling concentration, is that concentration required to keep the reactor subcritical by at least 1% Δk/k with all control rods removed from the core. A dilution event from this concentration is shown to result in a reduced boron concentration that is still higher than the critical boron concentration for the refueled core configuration with the two most reactive rods withdrawn. If the post dilution event concentration is calculated to be less than the critical concentration, the minimum allowed refueling concentration is procedurally controlled at a higher level that will assure the core remains subcritical for a dilution event.

The refueling evaluation assumes a conservatively small volume of RCS water will be diluted by the injection of a makeup tank full of deborated water.

The volume of water assumed to be diluted corresponds to the minimum reactor vessel level allowed for maintenance activities with the fuel in the core, plus the volume of the smaller of the two decay heat removal loops (since one of the loops must be in operation to allow the dilution water to mix with the vessel water). Water in the refueling canal is conservatively assumed not to be diluted. The change in concentration caused by the dilution is independent of the rate at which the dilution occurs.

14.1.2.4.4 <u>Results of Analysis</u>

The highest rate of dilution can be handled by the ICS, which inserts rods to maintain the power level and the RCS temperature. Rod insertion beyond the allowable band will close the makeup valve to terminate the addition of water to the Makeup System.

If the reactor is under manual control with no control rod insertion, these reactivity additions will cause a high RCS pressure trip, which will cause the makeup valve to close, terminating the addition of dilution water to the Makeup System. Peak pressures and thermal powers for this case are shown in Table 14-8 for the normal and the maximum dilution flow rates. The thermal power does not exceed the design overpower limit of 112 percent of rated power and the system pressure does not exceed code allowable limits. Therefore, moderator dilution accidents will not cause any damage to the RCS.

The reactivity insertion prior to the reactor trip for the maximum RCS dilution flow rate, plus the reactivity insertion due to the emptying of a full makeup tank after the reactor trip, was shown to be slightly less than one percent Δk/k even at the beginning of core life when the initial boron concentration is high. Limits on the position of control rods at power assure a minimum shutdown margin of 1% Δk/k. Thus, sufficient rod worth is available to maintain the reactor subcritical following the trip even with the highest worth rod stuck out of the core. Emptying a full makeup tank of dilution water into the RCS would require nine minutes even at the maximum dilution rate considered.

The minimum refueling boron concentration required by the Technical Specifications normally assures more than enough post event concentration to keep the core subcritical. For example, the required refueling concentration for a new core may be 2300 ppm boron. For this refueling concentration, the injection of a full makeup tank of 0 ppm water will result in a final concentration of approximately 1900 ppm. This is much higher than the critical concentration for the same new core configuration with the two most reactive rods withdrawn, of approximately 1750 ppm. Expected new core configurations are unlikely to ever require an

initial concentration higher than the refueling concentration to maintain the core subcritical for this event. If necessary, however, the RCS concentration would be procedurally controlled at the higher level.

14.1.2.5 Cold Water Accident

14.1.2.5.1 Identification of Cause

The classical cold water accident is not possible in this reactor because the reactor coolant flow path contains no check or isolation valves.

However, when the reactor is operated with one or more idle pumps and these pumps are then started, the increased flow rate will cause the average core temperature to decrease. If the moderator coefficient is negative, then reactivity will be introduced into the core and a power rise will occur.

When operating with one or more idle pumps, the pump control circuitry has an interlock to prevent starting a pump if the reactor power is above 22 percent. This interlock insures that the trip setpoint will not be reached when restarting an idle pump.

14.1.2.5.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. The minimum DNB ratio shall not be less than 1.3.

B. The RCS pressure shall not exceed code pressure limits.

14.1.2.5.3 Methods of Analysis

A detailed digital simulation of the plant was used to evaluate the transient response to this accident. The model includes point kinetics and a multi-region fuel pin model connected through time delays to a pressurizer model. A steam generator model was included.

It was assumed that the plant was operating with two pumps at 60 percent of rated power when the remaining two pumps were started. The startup of the idle pumps caused the system flow to increase from 49 percent to 100 percent of design flow in nine seconds. It was found that the maximum temperature decrease for this case occurs in less than one loop time and amounts to about 6 °F. Conservative values of the moderator coefficient and the Doppler coefficient were assumed to exist at the time of the accident. The minimum tripped rod worth was used. The conditions used in this analysis are shown in Table 14-9.

14.1.2.5.4 Results of Analysis

The results are shown in Figure 14-16. It is seen that the maximum neutron power is reached several seconds after the pumps are started, and the pressure peaks well below its trip point several seconds later. The mismatch between the heat removal in the steam generator and the power generation causes this pressure rise. The thermal power lags the neutron power and reaches its maximum value several seconds after initiation of the accident. The results of this analysis are shown in Table 14-10.

Based on the analysis it is concluded that the pump control circuitry is adequate to prevent adverse transients in that it prevents starting an idle pump if power is above 22 percent of rated power. Additionally, the ICS would serve to limit the imbalance between the reactor and steam generator powers and reduce the severity of the accident. Since the thermal power does not exceed about 75 percent of rated power at full flow, the DNBR will be greater than 1.3 and since the pressure does not exceed the trip setpoint, the protection criteria are met.

14.1.2.6 Loss of Coolant Flow

14.1.2.6.1 Identification of Cause

The reactor coolant flow rate is reduced if one or more of the Reactor Coolant Pumps fail. A pumping failure can occur from mechanical failures or from a loss of electric power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others. The mechanical failure considered in the analyses is the locked pump rotor in which a pump stops instantaneously, with a very rapid reduction in flow.

Each Reactor Coolant Pump receives electric power from one of the two electrically separate buses, and, as discussed in Chapter 8, one pump in each loop is connected to each bus. Loss of the Unit Auxiliary Transformer (UAT), to which the 6,900-volt bus is normally connected, will initiate a rapid transfer to the second transformer source without loss of coolant flow.

Faults in an individual pump motor or its power supply could cause the loss of one pump with a corresponding reduction in flow as the pump coasted to a stop. A failure of a 6,900-volt bus could cause the loss of two pumps. A complete loss of forced flow is unlikely and would occur only if the primary source of offsite power were lost simultaneously with loss of the UAT. A complete power loss would cause immediate reactor trip independent of protection system actuation.

The original ANO-1 accident analyses specifically addressed the four pump coastdown, demonstrating that core damage would not result from this low probability event. Although not part of the original accident analyses, one and two pump coastdown events have also been addressed in the determination of limiting safety system settings for the RPS overpower trip based on flow and imbalance.

14.1.2.6.2 Reactor Protection Criterion

The criterion for reactor protection for the pump coastdown loss-of-flow event is that the minimum DNB ratio shall not be less than the applicable critical heat flux correlation limit (e.g., 1.3 for BAW-2 correlation, 1.18 for the BWC correlation, or 1.132 for the BHTP correlation). For the locked rotor accident, the minimum DNB ratio shall not be less than applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions.

14.1.2.6.3 Methods of Analysis

The original loss of coolant flow analyses reported in the ANO-1 FSAR included the locked rotor event and several variations of the four pump coastdown. Near the end of the original licensing process, the effects of fuel densification and associated power spiking were considered. Revised loss of coolant flow analyses, first presented in B&W Topical Report, BAW-1391, Arkansas Nuclear One Unit 1 Fuel Densification Report, superseded the FSAR analyses.

The four pump coastdown and locked rotor analyses presented in BAW-1391, are summarized here. These analyses have subsequently been performed with improved codes and updated inputs. However, the conservative results of the analyses presented in BAW-1391 remain limiting, and have been the basis of comparison for acceptability of later analyses.

The analysis methodology for the four pump coastdown and locked rotor events is similar. The major differences between the events are the rate of core flow reduction and the RPS trip response used as analysis inputs. Significant analysis input parameters used for the four pump coastdown and locked rotor events are presented in Tables 14-11 and 14-12 respectively.

The original four pump coastdown flow profile was calculated by an analog computer model that simulated the pump characteristics determined from manufacturer's data, the pump/motor inertia, and the flow/pressure drop relations of the flow path components around the system loops. The original four pump flow coastdown profile is presented in Figure 14-17. The applicable time segment of the profile is also presented in Figure 14-18a. This profile, from the fuel densification report, included the effects of an open internals vent valve which reduced the assumed flow to 95% of the original design flow.

The original locked rotor flow profile was a simple step change in flow from the initial flow rate to 75% design flow in 0.1 seconds. For the fuel densification report analyses, the locked rotor flow profile was more realistically characterized by a reduction from the initial flow rate to 75% design flow in less than two seconds. The locked rotor flow profile is presented in Figure 14-19a. An open internals vent valve was not assumed for the locked rotor analysis.

A separate digital computer code was used to calculate the core power, core inlet temperature and system pressure as a function of time for the original analyses.

The core power profile used in the four pump coastdown analyses was based on the profile presented in Figure 3-13 of the FSAR, corrected for the trip delay time corresponding to that of the power/pump monitor trip. This same profile was used in the coastdown analysis performed for the fuel densification report and is presented in Figure 14-18a. The core power profile for the locked rotor analysis, presented in Figure 14-19a, is very similar except that the profile included a pre-trip power increase due to an assumed positive moderator temperature coefficient.

The original four pump coastdown and locked rotor analyses accounted for the slight rise in system pressure that would result from the transients. The locked rotor analysis performed for the fuel densification report conservatively assumed system pressure remains constant. The core inlet temperature remains essentially constant for the most critical first seconds of the transients.

Once the flow, power, pressure, and inlet pressure profiles were established, a digital computer code was used to determine the core hot channel thermal-hydraulic conditions during the transient. The code used these inputs to calculate the pressure drop over an average fuel channel, with the resulting thermal-hydraulic conditions imposed on the hot channel. The hot channel flow and heat flux were characterized by radial and axial peaking factors and mechanical flow area reduction factors. The hot channel flow and heat flux characteristics were used to calculate the DNBR in accordance with the approved heat transfer correlation, and fuel and cladding temperatures as a function of time for the locked rotor analysis.

The two pump coastdown loss of flow analyses have not traditionally been considered a part of the accident analyses. The two pump coastdown analysis, which determines the setpoint for the power to flow portion of the power/power imbalance/flow RPS trip, is performed as a part of the cycle reload analyses. The two pump coastdown is more limiting than the four pump coastdown because the RPS trip credited in the two pump coastdown event (power to flow trip) is slower than the trip credited for the four pump coastdown event (pump monitor trip). Methodology similar to that described above, is used for the two pump coastdown analyses.

14.1.2.6.4 Results of Analysis

The results of this analysis show that the reactor can sustain a loss of coolant flow accident without damage to the fuel. The results of the four pump coastdown analysis, with effects of fuel densification, are presented in Figure 14-18b. The minimum DNBR remained well above the 1.3 limit for the critical heat flux correlation in use at the time of the analysis (W-3).

The RCS is capable of providing natural circulation flow after the pumps have stopped. The natural circulation characteristics of the RCS have been calculated with conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or the reactor outlet piping. Table 14-14 shows the natural circulation flow capability as a function of the decay heat generation. These flows provide more than adequate heat transfer capability for core cooling and decay heat removal by the RCS.

The reactor is protected from the consequences of Reactor Coolant Pump failure(s) by the Reactor Protection System and the ICS. The ICS initiates a power reduction upon pump failure to prevent reactor power from exceeding that permissible for the available flow. The reactor is tripped if insufficient reactor coolant flow exists for the power level.

The results of the locked rotor event, analyzed to consider densification effects, are presented in Figure 14-19b. As the figure shows, the DNBR rapidly fell as flow decreased. The critical heat flux correlation limit was reached at about 1 second. The figure shows the calculated increase in cladding temperature as flow continued to decrease. Core heat flux began to decrease after the reactor tripped, allowing the cladding temperature to peak at about 1350 °F and then begin to fall. This peak cladding temperature was well below the 2300 °F (subsequently reduced to 2200 °F) then accepted as the allowable limit of the interim ECCS analysis criteria.

One additional loss of coolant flow mechanism has been analyzed in the reactor design evaluation. This involves possible flow or leakage past the seat of a reactor internals vent valve. These valves are designed to be closed during all normal operations and during all normal and accident transients except for those accident transients involving reverse flow through the core. The design provides for positive closure even with no flow, and several rotational clearances are provided to ensure free motion and to prevent any tendency to stick. The valves also have a self-alignment feature to prevent reactor coolant leakage. Hydrostatic and vibrational tests have been made to demonstrate that the valves will operate as designed. However, the case of a reactor internals vent valve remaining open has been analyzed. This malfunction reduces the effective core flow and results in a reduced steady state and flow coastdown transient DNBR. To support Cycle 2, a Technical Specification surveillance requirement on the internals vent valves was incorporated, permitting the deletion of the flow penalty for a stuck open valve.

The results of the two pump coastdown analyses are used to determine the trip setpoint for the power to flow RPS trip. The event is examined with each reload analysis to assure that the power to flow RPS trip setpoint will provide a reactor trip early enough to prevent DNB.

14.1.2.7 Stuck-Out, Stuck-In, or Dropped Control Rod Accident

14.1.2.7.1 Identification of Cause

In the event that a control rod cannot be moved, localized power peaking and shutdown margin must be considered. If a control rod is dropped into the core while operating, the resulting transient must be examined.

Adequate hot standby, > 525 °F, shutdown margin is assured by requiring a subcriticality of one percent $\Delta k/k$ with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in Section 3.2.2 demonstrates that this criterion can be satisfied. This criterion has been analyzed in terms of the minimum tripped rod worth available in the loss of coolant flow, rod ejection, startup, rod withdrawal, and steam line failure accidents. In all cases the available rod worth is sufficient to provide margins below any damage threshold.

If a control rod deviates from its group reference position by more than an indicated nine inches and an in-limit is detected, the control rod drive system and ICS will inhibit all rod-out motion, and the steam generator load demand is run back to 40 percent of rated load. The details of these actions, which occur for both a dropped or stuck rod, are described in Sections 7.2.2 and 7.2.3.

14.1.2.7.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. The minimum DNB ratio shall not be less than 1.3.

B. The RCS pressure shall not exceed code pressure limits.

14.1.2.7.3 Methods of Analysis

A detailed B&W digital model has been used to analyze the transient response to a dropped control rod. This program includes fuel point, point kinetics, pressurizer, and loop models, including the steam generators.

The reactor is assumed to be operating at rated power when the control rod is dropped. To achieve the most adverse response, the most negative values of the moderator and Doppler coefficients were used along with the maximum calculated rod worth for rated power operation. In addition, no control rod drive system or ICS action was assumed to occur. The parameters used in this analysis are shown in Table 14-15.

14.1.2.7.4 Results of Analysis

The results are presented in Figure 14-20. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial

power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power; however, the thermal power level never exceeds the initial value by more than 0.1% thermal power.

Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial value. Since the thermal power never exceeds the initial value by more than 0.1% thermal power, only exceeding the initial thermal power between 50 and 65 seconds into the transient, and the pressure decreases during the transient, the protection criteria are met.

Cases have been run for rod drops at Beginning of Life (BOL) conditions and lower rod worths. These transients are not included in this discussion because they represent less severe conditions than the End of Life (EOL) conditions and the maximum calculated rod worth.

14.1.2.7.5 Reload Evaluation

Margins to departure from nucleate boiling (DNB), centerline fuel melt (CFM), and cladding strain linear heat rate (LHR) limits for the dropped rod event are verified for each reload core. A bounding, generic evaluation of the dropped rod accident for 177-fuel-assembly core designs was performed to verify that DNB, CFM, and cladding strain LHR limits are not violated. As part of each reload core design process, beginning with the reload core design of Cycle 15, parameters are checked to ensure that the generic evaluation remains valid. These checks include:

A. Verify maximum allowable steady-state and design power peaks remain applicable.

B. Verify CFM and cladding strain LHR limits remain applicable.

C. Verify that the dropped rod maximum axial peaking (MAP) limit relative to the steady-state MAP limit remains bounded.

D. Maintain radial pin peak to a value less than the design limit given in BAW-10187P-A (Reference 88a).

E. Verify that the maximum dropped rod worth does not exceed 0.2 percent $\Delta k/k$.

The generic evaluation of the dropped rod accident was performed on a core design with the greatest steady-state peaking factors allowed by implementation of statistical core design (SCD). A dropped rod worth of 0.2 percent $\Delta k/k$ was used. Dropped rod power distributions and corresponding peaking increases were initiated from the limits of normal operation, including transient xenon, rod insertion, and imbalance. Effects due to quadrant power tilt were also accounted for. The RCS system and core average power responses to a dropped rod, without power runback, were generated using the CADDS computer code and a lowered-loop, 177-fuel-assembly plant model. A sensitivity study was performed to determine the most adverse combination of Doppler and moderator reactivity coefficients that would produce the limiting conditions with respect to minimum DNBR (i.e., maximum core power and minimum core exit pressure). The bounding case represents conditions near middle-of-cycle and bounds all times in core life. This generic evaluation of the rod drop accident showed that departure from nucleate boiling, centerline fuel melt, and cladding strain linear heat rate limits are not violated.

14.1.2.8 Loss of Electric Power

14.1.2.8.1 Identification of Cause

The unit is designed to withstand the effects of loss of electric load or electric power. Emergency power systems are described in Chapter 8. Two types of power losses are considered:

A. A loss of load condition caused by separation of the unit from the transmission system.

B. A hypothetical condition resulting in a complete loss of all system and unit power except the unit batteries.

Transients that do not result in an immediate loss of load condition can occur, such as due to faults on the 500 kV system where the main generator remains tied only to the 161 kV system. This occurred in 2017 which resulted in increased reactive power demand when the main generator voltage began to drop. This drop in generator voltage caused the voltages on in-house load centers and MCCs to drop. Some control rods dropped into the core due to the voltage drop seen on B6. The autotransformer locked out soon after event onset, at which time a loss of load condition did occur, as evaluated in the following sections.

Unlike an initiating loss of load event, the aforementioned reactive demand resulted in control systems reacting as if an increase in load was occurring. The control systems attempted to increase steam flow demand to the main turbine generator (MTG). This, along with the dropped control rods, led to a drop in RCS temperature and pressure (an initiating loss of load event would result in rising RCS pressure). Absent the eventual generator lockout and Operator action to manually trip the reactor, a reactor trip would have occurred on low RCS pressure due to the rod insertions. The reactor was effectively shut down prior to the manual trip initiated by Operators when the control rods inserted as a result of degraded voltage, reaching a K_{eff} of approximately 0.972, mitigating any potential power peaking or thermal limit concern that might result from the control rod groups remaining withdrawn or partially inserted. If rods had not inserted, the plant would have responded to the eventual loss of load condition, as discussed in the following sections.

14.1.2.8.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Fuel damage must not occur.

B. RCS pressure shall not exceed code pressure limits.

C. Resultant doses for loss of load shall not exceed 10 CFR 20 limits.

14.1.2.8.3 Results of Loss-of-Load Conditions Analysis

The unit has been designed to accommodate a loss-of-load condition without a reactor or turbine trip.

Note that if generator output breakers B5114 and B5118 are opened by the primary tripping mechanism (286-G1-3 lockout relay), then the unit cannot maintain station auxiliary load without reactor trip because relay 286-G1-3 also trips the generator field breaker and generator lockout relays 286-G1-1 and 286-G1-2, which in turn trip the turbine lockout relay 286/T. In such an event, the plant will respond similar to the response following a turbine trip.

It is possible for the generator output breakers to trip based on backup protective schemes such as protective relaying in the switchyard, without directly causing a generator lockout. Under such circumstances, a runback signal causes an automatic power reduction to 15 percent reactor power. The runback may not be successful if the reactor high pressure setpoint is reached, at which time a reactor trip would occur (see SAR Sections 7.2.3.3.4 and 10.1). If successful, other actions that occur include:

A. All vital electrical loads, including the Reactor Coolant Pumps, condenser circulating water pumps, condensate pumps, and other auxiliary equipment, will continue to obtain power from the unit generator. Feedwater is supplied to the steam generators by the steam-driven feedwater pumps.

B. As the electric load is dropped, the electro-hydraulic system closes the governor valves. The unit frequency will change momentarily, but the governor will rapidly restore the set frequency.

C. During closure of the turbine governor valves, steam pressure increases to the turbine bypass valve setpoint and may increase to the steam system safety valve setpoint. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for a brief period following loss of load from 100 percent initial power until the turbine bypass can handle all excess steam generated. The amount of steam relieved to the atmosphere is shown in Table 14-16. Steam relief permits energy removal from the RCS to prevent a high pressure reactor trip. The initial power runback is to 15 percent reactor power, which is a higher power level than needed for the unit auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess steam above unit auxiliary load requirements is rejected to the condenser by the turbine bypass valves.

D. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increases in proportion to the generator's frequency increase. All motors and electrical gear so connected will withstand the increased frequency.

E. After the turbine generator has been stabilized at auxiliary load and set frequency, the station operator may reduce reactor power to the auxiliary load as desired.

The loss-of-load accident does not result in fuel damage or excessive pressures on the RCS. There is no resultant radiological hazard to station operating personnel or to the public from this accident, since only secondary system steam is discharged to the atmosphere.

Unit operation with one percent defective fuel and a 1 gpm primary-to-secondary tube leak has also been evaluated for this transient. The steam relief accompanying a loss-of-load accident would not change the whole body dose. The whole body dose is primarily due to the release of xenon and krypton and is considered to be negligible. Release of these gases is not increased by the steam relief because, even without relief, all of these gases are assumed to be released

to the atmosphere through the condenser vacuum pumps. The rate of release of iodine during relief would increase because the iodine is released in steam vented directly to the atmosphere rather than through the condenser and unit vent. The iodine contained in the 1 gpm primary coolant leakage is assumed to be carried off in the secondary steam flow of 5.56 x 10^6 lbs/hour at the rate at which it enters the secondary system. Table 14-16 gives the quantity of steam released, the activity of the iodine contained in the steam, and the resulting site boundary thyroid dose. The relative concentration factor from Section 2.3 is based on mixing of the discharge in the wake of the reactor building and a wind speed averaged over the height of the reactor building.

14.1.2.8.4 Results of Complete Loss of All Unit AC Power

The second power loss considered is the hypothetical case where all unit power except the unit batteries is lost. Loss of all AC power is regarded as an incredible occurrence and was not a basis for the original plant design since in addition to the normal AC power supplies, redundant fast starting emergency diesels are provided. Addition of the Alternate AC Power Source per 10 CFR 50.63 makes a loss of all AC power even more incredible. However, analysis of this hypothetical event demonstrates that even in the absence of all sources of AC power, decay heat can be removed. The sequence of events and the evaluation of consequences for this accident are:

A. A loss of power results in gravity insertion of the control rods and trip of the turbine valves.

B. After the turbine stop valves trip, excessive temperatures and pressures in the RCS are prevented by excess steam relief through the main steam line safety valves and the atmospheric dump valves (turbine bypass valve steam relief is lost due to loss of power to the condenser cooling water circulating pumps). Excess steam is relieved until the RCS temperature is below the saturation temperature for the steam generator corresponding to the pressure setpoint of the atmospheric dump valves. Thereafter, the atmospheric dump valves are used to remove decay heat.

C. The RCS flow decays without the occurrence of fuel damage. Decay heat removal after coastdown of the Reactor Coolant Pumps is provided by the natural circulation characteristics of the system. This capability is discussed in the loss-of-coolant-flow evaluation (Section 14.1.2.6).

D. The Condensate Storage Tank provides emergency feedwater to the steam generators with the EFIC system raising the water level in the steam generators at a controlled rate until the 26 foot natural circulation setpoint is reached. The Condensate Storage Tank minimum inventory is 107,000 gallons, when only Unit 1 is aligned, or 267,000 gallons when both units are aligned.

E. The turbine-driven emergency feed pump normally takes suction from the Condensate Storage Tank and is driven by steam from either or both steam generators. The Emergency Feedwater System is discussed in Section 10.4.8. All required valves in the Emergency Feedwater System can be operated automatically since they are DC powered.

The following is a description of the necessary loads which are connected to the station batteries and which would be operable following a loss of AC power:

A. Four inverters supplying the necessary Nuclear and Non-Nuclear Instrumentation, Reactor Protection System, Engineered Safeguards Actuation System, and Emergency Feedwater Initiation and Control system.

B. Emergency lighting panels.

C. DC distribution panels.

D. DC Motor Control Centers

The above loads provide sufficient power for indication and control to maintain the reactor in a safe shutdown condition for a minimum period of two hours with an expected period of four hours. In the event a longer battery life is required, certain redundant loads can be disconnected.

In view of the foregoing sequence, the loss of all unit power does not result in fuel damage or excessive pressure in the RCS.

14.1.2.9 Turbine Overspeed

14.1.2.9.1 Background

There is a very low probability that the turbines used at ANO-1 will experience a major structural failure of a rotating part resulting in missile-like pieces leaving the turbine casing (see SAR References 1 through 13).

This is based upon:

A. Present manufacturing techniques - factory inspection and test procedures ensure sound discs with mechanical properties equal to or exceeding the specified levels.

B. Redundant control system - the main speed governing system will normally hold the turbine speed within set limits. An overspeed trip device backed by a redundant overspeed trip device provides three lines of protection in all.

C. Routine testing - testing of the main steam valves and the overspeed trip devices while the unit is carrying load.

D. Turbine Disc Inspection consisting of: (1CAN098109)

   1. Inspection of new discs at the first refueling outage or before any postulated crack would grow to more than ½ the critical depth;

   2. Discs previously inspected to be free of cracks or that have been repaired to eliminate all indications should be per Item 1 above, calculating crack growth from the time of the last inspection; and

3. Discs operating with known and measured cracks should be reinspected before ½ the time calculated for any crack to grow to ½ the critical crack depth.

NOTE: Inspection schedules may be varied to coincide with scheduled outages.

E. Use of fully integrated LP rotor - During the 1R8 Refueling Outage, a fully integrated type rotor was installed in the "1" section of the low pressure turbine. This rotor differs from the original "2" section of the low pressure turbine in that the original "2" rotor was a built up design with shrunk on "discs" and the fully integrated design is a one piece forging design (see section F). Westinghouse Electric Corporation's position is that the missile generation criteria for the shrunk on wheels does not apply to the fully integrated design because a failure for the fully integrated rotor would assume a situation where the rotor reaches a high enough overspeed to cause the centrifugal stresses to exceed the material strength. Calculations performed by Westinghouse show that the required overspeed cannot be reached even if loss of load occurs at full load conditions. The amount of steam entering the turbine from the time the load is lost to the time the stop valves close is insufficient to drive the turbine to the required overspeed. If the valves didn't close, other turbine components would fail (such as last stage blades, generator wedges, bearings if high vibration occurs, etc.) at speeds below the rotor burst speed and again eliminate the potential for any turbine missiles.

   Therefore, the installation of the fully integrated rotor in the "1" low pressure turbine reduces the probability of missile generation from this section of the turbine.

F. Use of partially integral LP rotor - During the 1R14 Refueling Outage, a partially integral (PI) type rotor was installed in the "2" section of the low pressure turbine. This rotor differs from the originally installed "2" section rotor of the low pressure turbine in that the original "2" section rotor was a built up design with shrunk on “discs” and the partially integral design is a combination of forged and shrunk on discs. The partially integral rotor has the first three discs integrally forged and the last two discs shrunk on. Westinghouse Electric Corporation performed a “Low Pressure Turbine Missile Generation Study” (97-R-1020-02) to calculate the probability for disc rupture and missile generation resulting from stress corrosion for the partially integral rotor versus the originally installed rotor with shrunk on discs. The new PI rotor has a lower probability of failure than the original rotor and has been documented in the referenced study (See Chart 1). The accident analysis contained in this SAR section was based on the original rotor and the new PI rotor has a lower probability of failure as documented in the referenced study. Therefore, the installation of the partially integral rotor in the "2" low pressure turbine reduces the probability of missile generation from this section of the turbine.

14.1.2.9.2 Energy Considerations

A turbine disc has a total energy which is purely rotational of ½ ($I\omega^2$), where I is the mass moment of inertia of the disc about its axis of rotation and ω is the angular velocity of the turbine at the postulated failure speed. After failure the mass center of each fragment translates at a velocity of ω_r, where r is the distance from the rotation axis of the disc to the mass center of the fragment. In addition, the fragment rotates about its center of mass with an angular velocity of ω. The initial rotational energy of the disc is partitioned into both the translational and rotational kinetic energy of the fragments.

Test results and analytical considerations indicate that the translational kinetic energy of a fragment is of much greater importance than the rotational kinetic energy in predicting the ability of the fragment to penetrate the turbine casing. Rotational kinetic energy tends to be dissipated as a result of friction forces developed between the fragment surface and stationary parts.

These principles apply to fragments which would be generated by failure of the high pressure turbine rotor and the low pressure turbine discs.

This analysis applies to the original LP turbine design. During the 1R8 refueling outage, AP&L replaced the "1" LP turbine with a fully integrated design. During the 1R14 refueling outage, the ANO-1 "2" LP turbine rotor was replaced with a partially integral rotor design. A description of these designs and their missile generation capacity is provided in Section 14.1.2.9.1.

14.1.2.9.3 Hypothetical Rotating Element Failure

A study has been made of the probable sequence of events that might occur as a result of a hypothetical failure of the control devices to control the admission of steam to the turbine after a loss-of-load. As speed progressively increases above 1,800 rpm, the expected sequence of events is as follows:

Approximately 101 Percent:

Speed governor valves start to shut and will be fully shut at 103 percent speed. This will limit speed to less than 110 percent of rated speed. Assume these fail.

Approximately 111 Percent:

Emergency governor trips shut all control, stop intercept and reheat stop valves. Assume these fail.

At 111.5 Percent:

The backup overspeed trip device also trips all valves shut. Assume these fail.

Approximately 120 Percent:

Maximum speed expected after tripping stop control and interceptor valves. More highly stressed turbine discs lose their shrink fits.

120-150 Percent:

Shrunk-on discs (and the generator fan ring) continue to loosen until at 150 percent speed the last stage discs are eccentric by more than 15 mils, creating a large rotating unbalance. Some of the low pressure turbine critical speeds are expected to be in this range, thus producing a very large magnification factor on the unbalance force. At near constant speed, this would produce enormous bearing shaking forces which could cause bearing failure, disruption of the steam path, and braking of the machine. The actual shaking forces would depend on the rate of speed change through the critical speed region.

160-170 Percent:

Severe rubbing of the last stage blades would wear off portions of blade length, further increasing rotor unbalance. Generator rotor teeth and coil slot wedges would begin to yield plastically. The retaining ring would burst and probably separate into a few large high energy fragments. Based on the documented three such failures, it is not expected that these fragments would escape the generator casing with sufficient velocity to constitute hazards comparable to those resulting from turbine parts. Such failures tend to brake the rotors.

170-180 Percent:

Generator coil slot wedges begin to fly out permitting the field windings to be thrown into the air gap. Rotor teeth begin to fail and also add to the air gap debris, which should tend to brake the rotor to a standstill. All blades from the last stage disc will have failed at their roots with each such loss creating a rotating unbalance force of several hundred thousand pounds. Forces of this magnitude would probably cause immediate bearing failure, violent rubbing, and disruption of the steam path. It is possible that discs would fail. Heavy fragments could pierce the exhaust hood and would cause station damage. The vacuum would be broken and increased windage rotation loss would add to the many forces tending to stop the rotating shaft.

186 Percent:

Maximum runaway speed possible assuming the heat generated by friction and vibration have not braked the unit to standstill. At this speed the highest stressed low pressure turbine disc will fracture. Upon failure these disc fragments will puncture the turbine casing to the extent that additional overspeed is not possible since steam will leak to the atmosphere.

This analysis applies to the original LP turbine design. During the 1R8 refueling outage, AP&L replaced the "1" LP turbine with a fully integrated design. During the 1R14 refueling outage, the ANO-1 "2" LP turbine rotor was replaced with a partially integral rotor design. A description of these designs and their missile generation capacity is provided in Section 14.1.2.9.1.

14.1.2.9.4 Discussion of Rotating Element Fragments Following Failure

A. Components not contributing dangerous missiles.

   The preceding discussion has shown that many turbine and generator parts would be exposed to extreme stress conditions during the hypothetical runaway accident. Most of these will not result in significant missile problems. Examples are listed below.

   1. Generator field and retaining ring parts are expected to be retained by the generator housing, which by its construction is an ideal absorber.

   2. Small exciter components may be thrown during a hypothetical failure but of much smaller mass and velocity than the low pressure turbine parts listed below.

   3. Small rotating parts may be thrown during a hypothetical failure but they are also of smaller mass and have less stored energy than do corresponding parts of the low pressure turbine.

4. High pressure turbine rotors would not be expected to fail at runaway speed but, even if failure did occur, the fragments would be retained by the heavy-section, bolted, high pressure shells.

5. Couplings are designed to withstand overspeed higher than the maximum speed at runaway.

B. Components that could produce high energy missiles.

1. Low pressure turbine discs. The disc capable of producing the most dangerous missile is that of the last stage. Using the analysis techniques described in Reference 50, it has been shown that a 90° fragment is the most dangerous in terms of a concrete slab which can be perforated after leaving the turbine casing.

2. Low pressure turbine blade. The last stage blade is both the heaviest and most energetic of the blades capable of escaping from the turbine casing.

   Favorably oriented, i.e., head-on, the blade could conceivably penetrate approximately 10 inches of steel. Such orientation is unlikely, since the tip of the blade will quickly strike the inner cylinder cover tangentially. There will also be interference with the other blades. Depending on the ensuing angle, there is a good probability that the inner casing and hood structure would contain the blade.

   If the worst were to happen, the blade could ricochet and escape through the approximately 1¼-inch thick plate toward the end of the hood. It is judged that, in any case, not more than one-half of the initial energy is retained. In which case it is expected that the intervening building concrete structures in the missile path will absorb the remaining energy so that the steel containment liner would not be breached.

This analysis applies to the original LP turbine design. During the 1R8 refueling outage, AP&L replaced the "1" LP turbine with a fully integrated design. During the 1R14 refueling outage, the ANO-1 "2" LP turbine rotor was replaced with a partially integral rotor design. A description of these designs and their missile generation capacity is provided in Section 14.1.2.9.1.

14.1.2.9.5 Conclusions

Because of the redundancy and reliability of the plant turbine control and protection system, the close control of oil purity, the periodic check of steam admission valve freedom, the periodic turbine disc inspections, (1CAN098109), and the high value of the bursting overspeed, any missile resulting from a turbine-generator overspeed incident is hypothetical only and not considered credible.

14.1.2.10 Fuel Loading Errors

Enrichment errors in fuel pellets or pins beyond normal tolerances will result in local power shapes which vary from those calculated with nominal enrichments. Assembly enrichment errors or loading errors may cause gross power shapes which are peaked in excess of reference design values. The incore instrumentation system is designed to monitor assembly power distributions as discussed in Section 7.3.3, and is capable of detecting assembly misplacement. Fuel pellet and pin enrichment loading errors in excess of manufacturing

tolerances, are prevented by extensive loading controls and procedures. One such manufacturing process to assure that fuel pins are placed in the proper assembly is the "campaigning" of enrichments so that only a single enrichment is handled at a time during fuel fabrication. Gross fuel assembly misplacement in the core is prevented by administrative loading procedures and the prominent display of identification markings on each fuel assembly upper end fitting. During fuel loading, these identification numbers are compared to the loading diagram by at least two persons working independently.

Following each refueling, an incore power distribution is taken during startup testing and compared to calculated power distributions. Gross fuel assembly misplacement would be detected by the incore detectors during this phase of testing. Even for an assembly out of position, the incores can determine if a radial power tilt is present or developing. Similarly, the out-of-core detectors will also indicate quadrant tilt conditions.

Analysis

Power distributions resulting from enrichment loading errors in pellets, rods, and assemblies have been analyzed. The thermal-hydraulic conditions resulting from the perturbed power shapes have been determined and compared to design values.

The following cases have been analyzed:

Case 1:

A 3.40 wt percent ^{235}U fuel pellet was loaded in the center of a 2.70 wt percent ^{235}U fuel rod. The nuclear analysis was performed using a one-dimensional axial representation of the fuel rod.

Case 2:

A 3.40 wt percent ^{235}U fuel rod was loaded into the high flux region of a 2.70 wt percent assembly. The nuclear analysis was performed in two dimensions.

Case 3A:

The center assembly of an equilibrium fuel cycle core was replaced by a 3.40 wt percent assembly. This was an enrichment increase of 0.55 wt percent ^{235}U.

Case 3B:

An equilibrium cycle symmetrical assembly (near the outer edge of the core) was replaced by a 3.40 wt percent ^{235}U assembly. This was an enrichment increase of 0.55 wt percent ^{235}U.

The power distributions from Cases 3A and 3B were obtained from a two-dimensional, x-y plane, PDQ07 analysis.

The power distributions for Case 3A are presented in Figure 14-108. Only power peaks in the central region of the core are appreciably altered by a misloaded center assembly.

Similarly, misplacement of the center assembly by a higher enriched assembly does not cause a radial power tilt. The maximum radial-local power peak occurs in the center assembly which is a detector assembly. The incore instrumentation would detect an assembly power increase of this magnitude.

The power distribution for Case 3B is presented in Figure 14-109. A significant power tilt results which would be detected by the incore instrumentation. Misplaced assemblies in other core orientations would introduce radial power tilts which would be more easily detected than this case.

The thermal analysis of Cases 1 and 2 (misloaded pellet and pin) resulted in localized DNBR reductions which are limited to the misloaded pellet or pin.

Conclusion

Strict administrative controls will prevent enrichment errors during fuel fabrication and during fuel loading. In the unlikely event that gross core loading errors occur, the incore instrumentation is designed to detect it.

The analysis of fuel loading errors was initially completed for the expected enrichments of up to 3.4 wt percent. Fuel enrichments of up to 4.1 wt percent were evaluated prior to Cycle 12 and found to not invalidate the analysis conclusions. A wider range of fuel enrichments, lump burnable poison concentration, and burnup would result in an equal or higher probability of detections of a power distribution change from a core loading error. (1CAN069108)

14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 SITUATIONS ANALYZED AND CAUSES

This section presents an analysis of accidents in which one or more of the protective barriers are not effective and standby safeguards are required. All accidents evaluated are based on the rated core power level of 2,568 MWt. Table 14-18 summarizes the potential accidents studied.

14.2.2 ACCIDENT ANALYSIS

14.2.2.1 Steam Line Failure

14.2.2.1.1 Identification of Cause

Analyses were performed to determine the effects and consequences of loss of secondary coolant due to a double-ended steam line rupture. The main steam header piping (24″ and 36″) between the main steam block valves and the high pressure turbine stop valves, and the B31.1.0 portion of the main steam piping system (8″) between the atmospheric dump valve isolation valves and the dump valves themselves, is designed and constructed to meet ANSI B31.1.0 requirements with 100 percent volumetric examination of welds. The portion that penetrates the reactor building out through the main steam block valves is designed and constructed to meet the requirements of ANSI B31.7, Class II, or later appropriate ASME Section III Code sections provided that they have been reconciled. In addition, the main steam line from the steam generator outlet to the turbine has been analyzed and found adequate to withstand seismic loadings. Consequently the probability of a break in these lines is considered very low.

14.2.2.1.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. The core will remain intact for effective core cooling.

B. Loss of reactor coolant boundary integrity resulting from steam generator tube failure due to the loss of secondary side pressure and resultant temperature gradients will not occur.

C. Resultant doses will not exceed 10 CFR 50.67 limits.

14.2.2.1.3 Analysis and Results

14.2.2.1.3.1 Accident Dynamics

The loss of secondary coolant due to a failure of a steam line between the steam generator and the turbine causes a decrease in steam pressure and thus places a demand on the control system for increased feedwater flow. Increased feedwater flow, accompanied by steam flow through the turbine stop valves and the break, lowers the average reactor coolant temperature. The Emergency Feedwater Instrumentation and Control (EFIC) system (see Sections 7.1.4 and 7.2.4 is designed to protect against the consequences of a simultaneous blowdown of both steam generators. Upon detection of a steam line break, the EFIC system automatically initiates action to isolate each affected steam generator by closing its main steam isolation valve (MSIV) and its main feedwater isolation valve (MFIV).

(Note: Reactor power will also increase due to Integrated Control System (ICS) pulling rods to increase T_{ave}).

A steam line rupture of small area causes a slow decrease in steam pressure. The reactor power will increase with decreasing average reactor coolant temperature as a result of the negative moderator coefficient. This coefficient is used based on the fact that the steam line failure accident results in a decrease in the Reactor Coolant System (RCS) temperature so that the largest negative moderator temperature coefficient results in the maximum positive reactivity insertion. As shown in Table 3-8, the largest negative moderator temperature coefficient (MTC) during the core lifetime occurs at the end of the equilibrium fuel cycle. The most negative MTC predicted for Cycle 1 was -3.0×10^{-4} (Δk/k)/°F, later cycles have resulted in more negative values; hence -3.5×10^{-4} (Δk/k)/°F has been used in this analysis. The ICS will then cause control rod insertion in an attempt to limit reactor power to 102 percent. A reactor trip occurs due to low reactor coolant pressure or high neutron flux. Following reactor trip, the turbine trips and the turbine stop valves and feedwater control valves close. EFIC will be actuated on low steam generator pressure initiating closure of the main steam isolation valves and the feedwater isolation valves. The steam generator in the steam loop associated with the rupture blows dry after feedwater isolation. Decay heat is removed by the EFW flow supplied to the unaffected steam generator steaming through the atmospheric vent control valve and main steam safety valves.

The results of the analysis of the maximum break size at rated power are similar to those discussed above, but the maximum break represents the worst condition for a steam line rupture accident. A controlled cooldown rate can be established by feedwater isolation.

The reactor is assumed to be operating at rated power before the accident. Other parameters used in the analysis are shown in Table 14-19a. TRAP2, Babcock and Wilcox's once-through steam generator transient digital simulation FORTRAN code, was used to model the RCS response to the main steam line break. The following sequence of events occurs following a steam line rupture:

A. Reactor trip on low RCS pressure (1830.446 psig) or high flux (112 %) will occur following the break. Due to the size of break assumed an initial decrease in pressure will be seen in the core before the cold water from the steam generator reaches the core; hence a reduction in power occurs initially followed by an increase in power. A trip on low pressure is expected before the high flux trip is reached.

B. As a result of reactor trip the turbine stop valves close, however these valves are conservatively not modeled. The steam lines are assumed to be isolated by the main steam isolation valves which are modeled to close over a 9 second period starting one second after the EFIC low steam generator pressure setpoint (580 psig) is reached.

C. Following reactor trip, offsite power is still assumed available due to the greater primary to secondary heat removal capabilities of the RCPs.

D. Feedwater flow will continue until reactor trip occurs. The trip causes positive closing of the feedwater low load, startup, and control valves. Feedwater is conservatively modeled to continue to flow until EFIC is actuated on low steam generator pressure. Additionally, a single failure of the main feedwater isolation valve for the affected steam generator is assumed. The feedwater addition rate is predicted by the TRAP2

code based on the dynamics of the system. One second after the EFIC signal is received the main feedwater isolation valve for the unaffected and the feedwater control valve for the affected steam generator are modeled as closing over a 29 second time frame.

E. The Emergency Feedwater System will be initiated upon low steam generator pressure or loss of both main feedwater pumps. This portion of the EFIC system has two trains and each train will be able to supply and control the flow of EFW to either steam generator as required. No EFW flow is assumed to go to the affected steam generator based on the feed good generator only logic. EFW flow to the unaffected steam generator is not initiated until the low level setpoint of 30 inches is reached.

F. ESAS will be actuated on low RCS pressure (1480 psig). The HPI pumps are assumed to start injecting 25 seconds after the setpoint is reached. Two HPI pumps are assumed available due to the single failure of the feedwater isolation valve already assumed.

G. Normal post-trip response of the feedwater startup valve is to follow the control system requirements for steam generator minimum level control. An EFIC signal sent to the startup valves is assumed to prevent the valves from reopening following a steam line break.

14.2.2.1.3.2 <u>Evaluation of Results</u>

The event begins with the double-ended rupture of the main steam line upstream of the main steam isolation valve on steam generator "B" (line not associated with the pressurizer). The loss of secondary coolant causes the steam pressure and saturation temperature to decrease and places a demand on the control system for increased feedwater flow. Heat transfer between the primary and secondary systems increases because of the higher feedwater flow and greater primary to secondary temperature differential. This causes the primary system pressure and temperature to decrease until the reactor trips on low RCS pressure at 2.6 seconds, transient time (Table 14-19c contains a sequence of events and Figures 14-21 a-g indicate the RCS response to the double-ended rupture of the main steam line).

The EFIC low steam generator pressure setpoint is reached 3.28 seconds after rupture. Following a one-second delay, the main feedwater isolation valve to the unaffected steam generator, main feedwater control valve (MFCV), startup valve, and low load valve to the affected steam generator (assuming a single failure of the MFIV), and both main steam isolation valves are modeled to close. The MSIVs close over 9 seconds and the MFIV and MFCV close over 29 seconds. The EFW system receives a signal from EFIC to actuate, but flow is not initiated unless the low steam generator level of 30 inches is reached. In this case the low level setpoint was not reached and no EFW flow was modeled. ESAS actuates on low RCS pressure 6.69 seconds following the break. Assuming that offsite power is available (no diesel generator start), HPI flow begins following a 25 second delay. The Core Flood Tanks were also modeled but the system pressure did not drop below 560 psig; hence the tanks did not inject.

Once the MSIV closes, the unaffected steam generator begins to re-pressurize and becomes a heat source at approximately 20 seconds when the core exit temperature decreases below the unaffected steam generator saturation temperature. The unaffected steam generator begins to refill once the MSIV closes and continues to fill throughout the transient. This is because feedwater flow continues until the MFIV closes and because of condensation of the steam once

the steam generator becomes a heat source for the primary system. The affected steam generator continues to boil off inventory and cool the primary system. Negative moderator feedback due to temperature and density changes causes a positive reactivity insertion. This decreases the subcritical margin. The increased moderator density causes subcritical multiplication leading to an increase in the core power.

At 32 seconds, the affected steam generator is approaching dry out. However, shortly thereafter the water in the feedwater piping downstream of the MFCV begins to flash. Normally, this liquid is unavailable for cooling, since the upward orientation of the piping prevents it from draining into the steam generator. In this transient, however, voids form in the feedwater piping and displace the liquid into the steam generator. This additional inventory continues the overcooling by the affected steam generator throughout the transient.

The primary system continues to cool and depressurize until the heat removal in the affected steam generator becomes less than the combined heat additions to the RCS from the unaffected steam generator, primary metal, and decay heat. At this time, the core inlet temperature reaches a minimum. By 46 seconds, the minimum subcritical margin of 8.389×10^{-3} % $\Delta k/k$ occurs. The maximum return to power of 33% is reached at 50 seconds. Subsequently, the primary system then begins to re-pressurize, and core power decreases while subcritical margin increases.

The analysis predicted rapid reactor trip on low Reactor Coolant System pressure. The rod insertion was modeled to occur and ensured that a 1 percent subcritical condition was met at hot zero power conditions. Since the Reactor Protection System ensures a reactor trip prior to reaching adverse core conditions, the power response prior to reactor trip will be acceptable independent of the (hot full power) MTC. In fact, a more negative MTC will result in more power production and lead to a lessening of the overcooling transient. Once the reactor has tripped and the rods are in the core, the overcooling continues to reduce the fuel and moderator temperatures. Under these conditions a negative MTC will add positive reactivity to the core which may result in a return to criticality, thereby making the negative hot zero power MTC a limiting input for the steam line break. The analysis assumed a -3.5×10^{-4} $(\Delta k/k)/°F$ MTC at hot zero power conditions, and demonstrated acceptable plant and core response to the event with these assumptions. Bounding Departure from Nucleate Boiling (DNB) analyses were performed using detailed kinetic and core thermal-hydraulic models to determine the hot pin fuel response during a return to power resulting from an SLB. The analyses showed the margin to DNB to be sufficiently large that it can be concluded that DNB will not occur with the predicted return to power in this case.

During the first minute following the break, the average tube temperature in the affected steam generator remains above the shell temperature. Since thermal equilibrium is established, the average reactor coolant temperature will remain near the saturated temperature corresponding to the pressure at which the turbine bypass valve is set. Therefore, the tube-to-shell temperature difference will approach zero. The resultant tube stresses will remain less than the stresses corresponding to a pressure difference of 2,200 psi. (See Table 14-20.)

14.2.2.1.3.3 <u>Single Failure Postulations</u>

A limiting single failure of the main feedwater isolation valve to the affected steam generator to close was assumed. Additional information regarding potential single failures in the emergency feedwater during recovery from a steam line break is included in Table 10-1.

Offsite power was assumed to be available in this analysis due to the additional primary to secondary heat removal that will result from the RCPs running. The heat from the RCPs was not modeled; but 1.0 times the 1979 ANS 5.1 decay heat curve was assumed.

14.2.2.1.3.4 Subsequent Evaluations

The following sub-sections will describe subsequent SLB re-evaluation input parameter changes and results. Only analysis efforts which result in more limiting input parameters or results are outlined below.

14.2.2.1.3.4.1 Main Steam Line Diameter Change

Various changes have been made to the steam line rupture analysis defined above and in Tables 14-19a-d, Table 14-20, and Figures 14-21a-i. The following issues have been addressed in the re-evaluation of the steam line rupture accident:

A. The largest main steam line pipe inside diameter is 33.89 inches. An inside diameter of 32.5 inches is used in the above analysis. To bound the largest pipe break, a new break area associated with the larger pipe diameter is assumed.

B. An EFIC low pressure setpoint of 595 psia is assumed in the above analysis. In the re-evaluation, a setpoint of 585 psia is assumed. The lower setpoint is more conservative due to the time delay associated with steam generator pressure reaching the lower setpoint and the delay of isolating the steam generator.

C. A very conservative 25 second delay before HPI pump flow delivery is assumed in the above analysis. Due to the boron addition from the HPI pumps being credited in reactivity control, it is conservative to delay the addition from the pumps. With offsite power available, the HPI system is designed to deliver flow in less than 20 seconds from when the low RCS pressure setpoint is reached. Therefore, the re-evaluation credited the faster response time.

D. Isolation of both steam generators was initiated when the affected steam generator pressure dropped below the low pressure setpoint. The EFIC logic will only isolate the respective steam generator as the pressure in that generator goes below the setpoint. Hence, the above analysis modeled a premature isolation of the unaffected steam generator. In the re-evaluation, isolation of each generator is initiated as the pressure in that generator goes below the setpoint.

E. To offset the consequences of the model change defined in item D above, the MSIV stroke time is reduced from 9 seconds to 7 seconds in the re-evaluation. The EFIC signal delay is maintained at 1 second making the total MSIV response time 8 seconds.

F. An MTC of -3.5×10^{-4} $\Delta k/k/°F$ at hot zero power (532 °F) is assumed in the above analysis and the re-evaluation. However, the MTC curve over the range of RCS temperatures and pressures that the steam line rupture analysis transverses was shifted. An average MTC of -3.05×10^{-4} $\Delta k/k/°F$ over the temperature range is modeled in the re-evaluation versus -2.88×10^{-4} $\Delta k/k/°F$.

As a result of the above changes, a peak return to power of 34.2 percent and a subcritical margin of 3.203 x 10^{-2} %Δk/k are calculated with a blowdown mass and energy of 169,742 lbm and 1.919 x 10^{8} Btu. These results are bounded by those given in Table 14-19b and Table 14-20 except for the return to power, which is higher due to the more negative average MTC. The blowdown mass and energy are bounded by that given in Table 14-19b, which is used in the containment analysis. Due to the bounding nature of the blowdown data used in the containment analysis, no re-evaluation was performed. The peak return to power of 34.2 percent is slightly above the previous peak of 33 percent. However, the conclusions for the above analysis are still valid.

14.2.2.1.3.4.2 <u>RELAP Model</u>

As described in Section 14.2.2.1.3.2, a 29 second MFCV stroke time was assumed in the main steam line break analysis. The stroke of the valve was modeled at a constant rate, which is a conservative assumption due to the valve actually having a two speed operator. A constant rate was assumed so as to bound the response of the MFIV, which was assumed to fail. Additionally, the main steam line break analysis did not credit the RFR signal to run back the main feedwater pumps. A RELAP model of the feedwater system was developed to justify a 32 second stroke time of the MFCV by crediting the two speed operation of the valve and the feedwater pump run back. The results of the RELAP analysis indicate that the feedwater mass delivered to the affected steam generator is less when the MFCV two speed operator with a 32 second response time and feedwater pump run back are assumed, as opposed to the main steam line break assumptions of no feedwater pump run back and a 29 second constant valve closure time.

14.2.2.1.3.4.3 <u>Reduced MFIV Flow Resistance</u>

As described above a RELAP5/MOD2 model of the main feedwater system was developed to justify a longer stroke time for the main feedwater control valves. The RELAP5/MOD2 results were compared to those calculated by the TRAP2 computer code. The comparison showed the main feedwater (MFW) flow calculations in the TRAP2 simulation were significantly lower. A review of the methods used to model the main feedwater system in the TRAP2 simulation determined that the technique used to model the system flow resistance was incorrect and yielded a significant under prediction of the main feedwater flow during a main steam line break (MSLB). To offset the significant amount of feedwater addition, two analyses were performed; one with a single failure of the MFIV to the affected steam generator and the other with a failure of the non-Q equipment that would be expected to assist in the mitigation of the MSLB.

Case 1 (non-Q equipment failures) used the RELAP5/MOD2 feedwater system model to determine the feedwater addition into the steam generator following a MSLB. This feedwater data was input into the TRAP2 code with the feedwater model disabled. For Case 2, the feedwater model in TRAP2 was corrected; hence, only the TRAP2 code was used. All of the input assumptions and parameters defined above in Section 14.2.2.1.3.1 with the changes defined in Section 14.2.2.1.3.4.1 and 14.2.2.1.3.4.2 were used in these analyses with the following changes:

Case 1

1. No credit was taken for non-Q equipment operation following normal reactor trip (no credit was taken for closure of the TSVs and the MFCVs, nor ramp back of the MFPs).

2. The MFIV stroke time was modeled as a linear closure in 26.5 seconds rather than 29 seconds.

3. The MSIVs were assumed to have a stroke time of 6 seconds rather than 7 seconds.

4. The RELAP5/MOD2 model was used to predict the feedwater addition to the steam generators.

Case 2

1. A single failure of the MFIV to the affected steam generator was assumed.

2. The MFCV for the affected steam generator was assumed to have a stroke time of 32 seconds with credit taken for the two speed operator.

3. The main feedwater pumps were assumed to run back to 3200 rpm.

4. The MFIV for the unaffected steam generator was assumed to have a stroke time of 26.5 seconds.

5. The MSIVs were assumed to have a stroke time of 6 seconds.

The first analysis resulted in a peak return to power of 27.31 percent and a minimum subcritical margin of 0.05558 %Δk/k. A peak return to power of 19.92 percent and a subcritical margin of 0.1842 %Δk/k was predicted for the second case. The results from both analyses are bounded by the results in Sections 14.2.2.1.3.2 and 14.2.2.1.3.4.1 with the no single failure case (Case 1) being more limiting. The blowdown mass and energy for Case 1 are 164,265 lbm and 1.999×10^8 Btu. Table 14-19b blowdown mass and energy used in the reactor building pressure analysis bounds the Case 1 results.

Based on the MTC model and reactivity calculations in TRAP2 a maximum reactivity feedback, termed reactivity deficit, of 0.93702 %Δk/k was determined for the most limiting temperature and pressure range transversed in Case 1 below HZP conditions (532 °F and 2200 psia). The most limiting moderator temperature and pressure were 477.51 °F and 735.87 psia with a fuel temperature of 650.7 °F.

14.2.2.1.3.4.4 Feedwater Heater Replacement

Replacement of the E-1 high pressure feedwater heaters, which were explicitly modeled in the RELAP feedwater model, necessitated the reevaluation of the Main Steam Line Break (Case 1 in 14.2.2.1.3.4.3) because the replacement heaters had a larger volume, smaller flow resistance, and different heat transfer characteristics than the original heaters. The reevaluation was performed using the methods described in BAW-10193P, August 1995. In addition to properly modeling the replacement heaters, some assumed parameters were changed to yield a more conservative analysis to minimize the need for future reevaluations should the actual values change. These included the following changes, all made to the secondary side: 1) elevated steam generator outlet pressure, 2) elevated feedwater initial flow, 3) elevated initial feedwater temperature, 4) MFIV stroke time of 29 seconds, and 5) MSIV stroke time of 9 seconds.

The results were compared to the peak return to power and minimum subcritical margin and found to be bounded by the previous results reported in section 14.2.2.1.3.2 and 14.2.2.1.3.4.1. The mass and energy blowdown results were compared to the mass and energy blowdown used in the MSLB containment analysis and found to be bounded as well.

14.2.2.1.3.4.5 Steam Generator Replacement

At the end of Cycle 19, the original OTSGs were replaced. In support of Cycle 20 operation, a specific steam line break analysis was performed to determine if the MSLB analysis of record would remain bounding (Reference 94). The RELAP5 model described above was used and the steam generator model details were updated to reflect the replacement steam generators.

No other changes were made to the model. The results of the core-response analysis were compared to the analysis of record with the original OTSGs. It was concluded that the existing analysis will still be bounding for ANO-1 with the replacement OTSGs.

14.2.2.1.3.4.6 Boron Inject Credit Removed

During Cycle 23, it was decided to remove credit for the injection of boron following a MSLB. The RELAP5 model described above, as modified to reflect the replacement Steam Generators, was revised by reducing the boron concentration in the BWST to 0 ppm. Other changes included reducing the BWST water temperature to the lowest allowed by the plant Technical Specifications (i.e., 40 °F), and modifying the ESAS low pressure setpoint and the initial pressurizer level in order to initiate HPI sooner. These additional revisions are conservative for a MSLB, due to the negative MTC. The results of the new core-response analysis were compared to the analysis of record with the original OTSGs and it was concluded that the existing analysis will still be bounding for ANO-1 with no credit for boron.

14.2.2.1.4 Resultant Doses

The resultant doses from this accident are calculated by assuming that:

A. The unit has been operating with a maximum of 1 gpm steam generator tube leakage.

B. The unit has been operating at the maximum primary and secondary activity limits allowed by Technical Specifications.

C. The steam line break occurs between the reactor building and a turbine stop valve.

D. Reactor coolant leakage into the faulted steam generator continues for 251.8 hours until the RCS can be cooled down and depressurized and the leakage terminated.

E. Either an accident-initiated iodine spike (GIS) occurs or a pre-existing iodine spike (PIS) exists.

The steam line failure is assumed to result in the release of activity contained in the steam generator inventory, activity contained in the feedwater, and activity contained in reactor coolant leakage. (See Table 14-21.) The iodine, primarily resulting from reactor coolant leakage in the cooldown period following the steam line break, is assumed to be released directly to the atmosphere. Atmospheric dilution is calculated using the relative concentration developed in Section 2.3. Using these assumptions, TEDE doses have been calculated (see Table 14-21). These doses are less than the acceptance criteria of 10 CFR 50.67.

14.2.2.1.5 Building Pressure

The resultant mass and energy release to containment are taken from the above analysis using TRAP2 results (see Figures 14-21F and 14-21G) and are summed in Table 14-19b. A detailed thermal-hydraulics analysis was performed using the blowdown data in the ANO-1 DBA COPATTA model described in Section 14.2.2.5.5. The results from this analysis are shown in Figures 14-21H and 14-21I, and summarized in Table 14-20. A peak reactor building pressure of 51.1 psig occurs at 74 seconds. The peak pressure is within the DBA pressure of 54.0 psig and the reactor building design pressure of 59 psig. The MSLB temperature profile, although it exceeds the DBA temperature profile for less than 3 minutes early in the transient, is short lived and is considered bounded by the DBA profile.

Parameters used in the MSLB containment analysis along with those which are different from that assumed in DBA analysis are given in Table 14-19d. A high-high containment pressure setpoint of 36.7 psig is assumed for reactor building spray actuation. The spray response time is assumed to be 105.868 seconds based on offsite power being available as was determined above to be the most limiting case (see Section 14.2.2.1.3.3). Reactor building coolers are assumed to start at 300 seconds with the performance curve given in Figure 14-110. Only one train of sprays and coolers are modeled. This is conservative due to the single failure of the MFIV already assumed in developing the blowdown data.

At the end of Cycle 19, the original OTSGs were replaced. In support of Cycle 20 operation, an evaluation of the containment pressure/temperature response with the replacement OTSGs for LOCAs and MSLBs was performed and is documented in Reference 94. It was concluded that the current post-LOCA response would remain bounding for the replacement OTSGs. For the steam line break, the containment pressure response with the replacement OTSGs was also bounded by the current analysis. The post-MSLB temperature response with the replacement OTSGs would be worse. EOI has adopted NUREG-0458 into the ANO-1 licensing basis which recognizes that the post-MSLB atmosphere may become superheated, but the temperature spike is of such short duration that the thermal lag of any SSC inside containment will not increase significantly. Consequently, the initial temperature peak does not define operating limits on any system, structure, or component (SSC) and the long-term containment temperature (which is essentially the saturation temperature) dominates the temperature response of SSCs. Therefore, as long as the peak MSLB pressure is less than the peak pressure following a LOCA, the temperature response of SSCs will still be defined by the LOCA.

14.2.2.1.6 Conclusions

The ANO-1 plant response to a double-ended steam line break with a failure of the main feedwater isolation valve on the affected side has been shown to be acceptable. The analysis has shown the acceptability of a hot zero power moderator temperature coefficient of -3.5×10^{-4} (Δ k/k)/°F. The predicted maximum return to power assuming a conservative core kinetics model is below that necessary to exceed fuel design limits. The maximum temperature differential that occurs in the steam generator does not produce excessive stresses, and the integrity of the steam generator is maintained. The resultant doses are within acceptable limits.

14.2.2.2 Steam Generator Tube Failure

14.2.2.2.1 Identification of Cause

The resultant doses associated with steam generator tube leakage and subsequent release to the environment are evaluated in the preceding sections. The complete severance of a steam generator tube has also been evaluated. For this occurrence, activity contained in the reactor coolant would be released to the secondary system. Some of the radioactive noble gases and iodine would be released to the atmosphere through the main steam safety valves, atmospheric dump valves, and the condenser air removal system.

14.2.2.2.2 Reactor Protection Criteria

The criteria for reactor protection for this accident are:

A. Resultant doses shall not exceed 10 CFR 50.67 limits.

B. Additional loss of reactor coolant boundary integrity shall not occur due to resultant temperature gradients.

14.2.2.2.3 Analysis and Results

In analyzing the consequences of this failure, the following sequence of events is assumed to occur (input parameters are shown in Table 14-22 and results are summarized in Table 14-23):

A. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end.

B. The initial leak rate exceeds the normal makeup to the RCS, and system pressure decreases. No initial operator action is assumed, and a low RCS pressure trip will occur.

C. Following reactor trip, the RCS pressure continues to decrease until HPI is actuated. The capacity of the HPI is sufficient to compensate for the leakage and maintains both pressure and volume control of the RCS. Thereafter, the reactor is assumed to be cooled down and depressurized at 100 °F per hour until isolation of the affected steam generator can be achieved.

D. Following reactor trip, offsite power is lost and, since a reactor coolant-to-secondary system leak has occurred, steam line pressure will increase, opening the main steam safety valves. The fission products escaping from the main steam safety valves are released to the atmosphere.

E. After the RCS temperature has decreased to a value that corresponds to a saturation pressure which is below the main steam line safety valves setpoint, the affected steam generator can be isolated. Cooldown on the unaffected steam generator continues using an atmospheric dump valve until the temperature is reduced to less than 280 °F. Thereafter, cooldown to ambient conditions is continued using the Decay Heat Removal (DHR) system. The initial leak rate is conservatively assumed to continue during the entire depressurization time.

F. The operator will receive early notification that a primary to secondary leak has occurred by the radiation alarm on the reactor console which is initiated by the radiation monitor on the air ejector. The operator does not have to make a judgment quickly since the analyses assumes that no action is taken until 20 minutes after the tube rupture. Thus there is sufficient time available before starting cooldown and depressurization for the operator to obtain samples from the steam generator and to definitely determine from radioactivity and chemical analyses which generator contains the leak. For tube leaks smaller than the 435 gpm leak analyzed in the steam generator tube rupture accident, the operator has even more time to identify that a tube leak has occurred and to determine the affected steam generator.

G. Either an accident-initiated iodine spike (GIS) occurs or a pre-existing iodine spike (PIS) exists.

The radioactivity released during this accident is assumed to be discharged both through the main steam safety valves and atmospheric dump valves to the environment and through the turbine bypass to the condenser and then out the condensate vacuum pump exhaust. A gas-to-liquid partition factor of 10^4 is assumed for the iodine in the condenser (See References 28 and 33), but noble gases are assumed to be released directly to the atmosphere. TEDE dose results are given in Table 14-23. The atmospheric dilution is calculated using the 2-hour relative concentrations developed in Section 2.3.

14.2.2.3 Fuel Handling Accident

14.2.2.3.1 Identification of Cause

Spent fuel assemblies are handled entirely under water. Before refueling, the boron concentrations of the reactor coolant and the fuel transfer canal water above the reactor are increased so that, with all control rods removed, the k_{eff} of the core would not exceed 0.99. The fuel assemblies are stored under water in the spent fuel storage pool; the storage racks have a safe geometric spacing. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. A mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

14.2.2.3.2 Reactor Protection Criterion

The criterion for reactor protection for this accident is that resultant doses shall not exceed the acceptance critieria of 10 CFR 50.67.

14.2.2.3.3 Methods of Analysis

The assumptions made for this analysis are shown in Table 14-24. The reactor is assumed to have been shut down for 72 hours, since Technical Specifications prohibit fuel handling operations prior to this time. It is further assumed that the cladding of six rows of fuel rods in the assembly, 82 of 208, suffers mechanical damage.

Since the fuel pellets are cold, only the gap activity is released, and consists of 30 percent of the Kr^{85}, 12 percent of the I^{131}, and 10% of all isotopes in the damaged rods. Radioactive decay of the fission product inventory during the interval since shutdown and commencement of fuel handling operations is considered.

14.2.2.3.4 Results of Analysis

The gases released from the fuel assembly pass upward through the spent fuel storage pool water before reaching the atmosphere of the fuel handling building. The gas is assumed to pass through 23 feet of water, and 99.5 percent of the iodine released from the fuel assembly is assumed to remain in the water. No retention of the noble gases is assumed. The radionuclides released during the fuel handling accident are assumed to enter the atmosphere directly without filtration. The atmospheric dilution is calculated using the 2-hour relative concentration developed in Section 2.3. Dose conversion factors consistent with FGR 11 and FGR 12 were utilized.

The parameters used to analyze the fuel handling accident are given in Table 14-24. Table 14-25 gives the TEDE doses at the exclusion distance and low population zone.

14.2.2.4 Rod Ejection Accident

14.2.2.4.1 Identification of Cause

Reactivity excursions initiated by uncontrolled rod withdrawal (See Section 14.1.) were shown to be safely terminated without harming the reactor core or the integrity of the RCS. In order for reactivity to be added to the core at a more rapid rate, physical failure of a pressure barrier component in the Control Rod Drive assembly must occur. Such a failure could cause a pressure differential to act on a CRA and rapidly eject the assembly from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by Reactor Protection System trips.

Since CRAs are used to control load variations only and boron dilution is used to compensate for fuel depletion, only a few CRAs are inserted (some only partially) at rated power level. Thus, the severity of a rod ejection accident is inherently limited because the amount of reactivity available in the form of control rod worth is relatively small.

14.2.2.4.1.1 Accident Bases

Using an analytical method based on diffusion theory (See Section 3.2.2.2.1.), the worth of the most reactive CRA in each rod group was determined for different control rod configurations. The maximum rod worths and other important parameters used in the study are shown in Table 14-26. The tripped rod worth corresponds to the minimum worth available with the maximum-worth rod stuck out at Beginning of Life (BOL) and End of Life (EOL).

The severity of the rod ejection accident depends on the worth of the ejected rod and the reactor power level. The control rod group of greatest worth is the first in the entire rod pattern to be withdrawn. The maximum worth of a rod in this group can be as high as 2.5 percent $\Delta k/k$, but a rod would have this worth only when the reactor was subcritical. The details of the control rod worth calculations and the methods of selecting the number of control rods in each group are presented in Sections 3.2.2, 3A, and 7.2.2.2.

When the reactor is subcritical, the boron concentration is maintained at a level that ensures that the reactor is at least one percent subcritical with the control rod of greatest worth fully withdrawn from the core. Thus, a rod ejection will not cause a nuclear excursion when the reactor is subcritical and all the other rods are in the core.

As criticality is approached, the worth of the remaining rods decreases so that at criticality the calculated reactivity addition from a rod ejection would be 0.23 percent $\Delta k/k$.

At BOL, the calculated rod worth at the rated power is 0.40 percent $\Delta k/k$. However, a detailed analysis has been performed at worths up to 0.7 percent $\Delta k/k$ to show the large margin that exists between the calculated rod worths and those worths needed to approach any failure thresholds. A maximum rod worth of 0.65 percent $\Delta k/k$ at rated power has been considered as a limiting value to demonstrate the inherent ability of the system to safely terminate this postulated transient. A rod must be fully inserted in the core to have the foregoing reactivity worth values. Assuming that the failure occurs so that the pressure barrier no longer offers any restriction to the ejection and that there is no viscous drag force limiting the rate of ejection, the control rod travel time to the top of the active region of the core is calculated to be 0.176 second. Since most of the reactivity is added during the central 75 percent of this travel, only this distance is used in the analysis, resulting in an ejection time of 0.15 second for the analysis.

14.2.2.4.1.2 Fuel Rod Damage

The consequences of a rod ejection accident depend largely on the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and by reactor trip, then the energy release rate is limited by a relatively low surface-to-volume ratio for heat transfer. The energy stored in the fuel rods will then be gradually released to the coolant (over a period of several seconds) at a rate that poses no threat to the integrity of the RCS. However, if the magnitude of the nuclear excursion is so great that the fuel rod cladding does not remain intact, then both fuel and cladding may be dispersed into the coolant to such an extent that the heat transfer rate increases significantly.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod ejection accidents could lead to three potential modes of fuel rod failure. Failure by the first mode occurs when internal pressures developed in the fuel rod are insufficient to cause cladding rupture, but subsequent heat transfer from fuel to cladding raises the temperature of the cladding and weakens it until local failure occurs. Departure from Nucleate Boiling (DNB) usually accompanies and contributes to this mode of failure, and little or no fuel melting would be expected. In this mode of failure, fuel fragmentation is usually only minor, and any dispersal of fuel to the coolant would occur very gradually; system contamination would be the worst probable consequence.

The second mode of failure occurs when significant fuel melting causes a rapid increase in internal fuel rod pressure which, combined with a loss of cladding strength at higher temperatures, causes the fuel rod cladding to rupture (See Reference 34). Some fuel vaporization may occur, contributing to the pressure buildup. Considerable fragmentation and dispersal of the fuel would be expected in this mode.

The third and most serious mode of fuel rod failure is the occurrence of extensive fuel melting and subsequent vaporization due to a very large and rapid reactivity transient in which there is insufficient time for heat to be transferred from the fuel to the cladding. In this mode, destructive internal pressures can be generated without increasing cladding temperatures significantly.

In evaluating the effects of these modes of failure, two failure thresholds are considered. The first, associated with a gradual and usually minor cladding failure, may be defined approximately by the minimum heat flux for DNB at the cladding surface. The second failure threshold, defined as the enthalpy threshold for prompt fuel failure with significant fragmentation and dispersal of fuel and cladding into the coolant, is used to describe the energy required to cause failure by either the second or the third mode of failure.

A correlation of the results of different experiments conducted on Zircaloy-2 clad UO_2 fuel rods at TREAT (See Reference 17.) has been interpreted by the experimenters to show a threshold at 280 cal/g of fission energy input. That is, below this value the fuel rod can be expected to remain intact, and above this value fragmentation can be expected. The enthalpy corresponding to the melting point of UO_2 is about 260 cal/g (See Reference 35) and the heat of fusion is at least 78 cal/g (See Reference 11). Thus, the 280 cal/g represents a condition in which only part of the fuel is molten. Also of interest as a probable indication of the degree and rapidity of fuel and cladding dispersal are the measurements of pressure rise rates in the autoclave in the TREAT experiments (See Reference 17). Preliminary analysis indicates that there is only a modest pressure rise up to an energy input of 400 cal/g. Above 500 cal/g, however, there is a very definite pressure pulse. Thus, between 400 and 500 cal/g there is a transition, which probably corresponds to the change from the second to the third failure mode discussed previously.

A fuel failure threshold of 280 cal/g, at the pellet radius corresponding to the average temperature of the hottest fuel pellet, has been used in this study to define the extent of fuel failure.

In computing the average enthalpy of the hottest fuel pellet during the excursion for the rated power cases, it is assumed that no heat is transferred from the fuel rod between the time the accident is initiated and the time when the neutron power returns to the rated power level. For the zero-power cases, the enthalpy increase was based on the peak value of the average fuel temperature. In all cases the average enthalpy rise from the integrated energy or the fuel temperature traces is multiplied by the maximum peaking factor to obtain the enthalpy increase in the hottest fuel pellet.

The latest correlation of the ANL TREAT (See Reference 17.) data for the meltdown experiments on Zircaloy-2-clad UO_2 fuel rods shows the threshold for the zirconium-water reaction to be 210 to 220 cal/g energy input. A conservative threshold value of 200 cal/g is used in this study.

In calculating the volume of the core experiencing burnout in a given rod ejection accident, it is assumed that any DNB condition results in burnout for each rod where the DNB occurs. DNB in a rod ejection transient is assumed to occur whenever the peak thermal power of a given fuel rod exceeds the peak at steady state conditions that could result in a DNB, which in turn is assumed to occur for a DNBR of 1.3 using the W-3 correlation.

In determining the environmental consequences from this accident, an even more conservative approach is taken in computing the extent of DNB experienced in the core. All fuel rods that undergo DNB to any extent are assumed to experience cladding failure with subsequent release of all the gap activity. Actually, most of the fuel rods will recover from DNB, and no fission product release will occur. The fuel rods that experience DNB at BOL are assumed to have EOL gap activities.

14.2.2.4.2 Reactor Protection Criterion

The criterion for reactor protection for this assumed accident is that the effects of a control rod ejection accident shall not further impair RCS integrity.

14.2.2.4.3 Method of Analysis

A B&W digital computer program has been used to analyze the rod ejection accident. This program agrees to within a few percent in all cases with CHIC-KIN (See Reference 23). The B&W program is a point kinetics model with a reactor coolant loop and pressurizer model. The core heat transfer model allows for up to 30 radial mesh points in the fuel and cladding, and the mesh size can be different in the two regions. The model accounts for the gap conductivity and film coefficient of heat transfer. Reactivity feedback is calculated in each mesh point and in the coolant and is weighted for inclusion in the kinetics simulation. The thermal properties are input separately for each mesh point but remain constant with time. The loop model includes a simulation of the steam generator, which can have a nonlinear heat demand input on the secondary side. Trip action is initiated on high or low RCS pressure or on high neutron flux. Decay heat can be taken into account as well. This code was used to calculate the neutron and thermal power, integrated energy, reactivity components, pressure, and fuel rod and loop temperatures. Six delayed neutron groups are considered. The control rod trip is represented by a 20-segment curve of reactivity insertion during trip versus time, obtained by combining the actual rod worth curve with a rod velocity curve. Nominal values for the various nuclear and physical parameters used as inputs are listed in Table 14-27.

As a check on the point kinetics calculation, the rod ejection accident was also analyzed for a limited number of cases in support of the Technical Specification rod worth using the exact, two-dimensional, space-and-time dependent TWIGL digital computer program (See Reference 38). The point kinetics model assumes that the flux shape remains constant during a transient. This flux shape contains peaking factors which reflect unusual rod patterns such as the flux adjacent to a position where a high worth rod has been removed. Therefore, these point kinetics peaking factors are much higher than any that would actually occur in the core during normal operation. The purpose of using an exact space-time calculation is to find the flux shape during a transient. But to have a transient where a rod is ejected from the core, one must start with a flux shape that is necessarily depressed in the region of the ejected rod. In fact, the higher the worth of the rod, the more severe becomes the depression. This flux depression also causes a fuel temperature depression. When the rod is ejected from this position, the flux quickly assumes a shape that shows some local peaking.

However, when this "exact" peaking is applied to a region initially at depressed fuel temperatures, as it is in the case of the regions adjacent to the ejected rod, the resultant energy deposited in these regions causes a lower peak temperature and peak thermal power than does applying an arbitrary maximum peaking factor to an undepressed peak power region. The results from TWIGL were used to calculate the maximum total energy deposited in each region of the core following a rod ejection; the highest energy is reported in Table 14-28. The result is that the hottest region simulated in the TWIGL code actually undergoes a less severe transient than the hottest fuel rod assumed in the point kinetics model. As seen in Table 14-28, this result is uniformly true for all rod worths. For certain cases where the ejected rod has a low worth, or where at least one reactivity coefficient is very negative, or the initial power level is low, there is considerable pressure buildup in the RCS because of the increased heat being added to the coolant with no increase in heat demand. Many of these transients never reach the overpower trip point. For this class of possibility, the high pressure trip must be relied on, and this is incorporated in the calculation.

14.2.2.4.4 Results of Analysis

14.2.2.4.4.1 Zero Power Level

The nominal BOL and EOL rod ejection analysis was performed at 0.1 percent of rated power, and the results can be seen in Table 14-29. No DNB and no fuel damage would result from these transients.

A sensitivity analysis has been performed around these two cases in which the Doppler and moderator coefficients, trip delay time, and rod worth were varied. Figure 14-22 shows the peak neutron power as a function of ejected rod worth from 0.2 to 0.7 percent $\Delta k/k$. The curve shows two distinct parts corresponding to worths less than and values near to and greater than β. Figure 14-23 shows the corresponding results for the peak thermal power. It is seen that for rod worth values near prompt critical, the period is small enough to carry the transient through the high neutron flux trip. For lower values the pressure trip is relied on. No DNB occurs for any of these parameter variations.

Figure 14-24 shows that the peak enthalpy in the fuel for the rod worths in the range being evaluated never exceeds 75 cal/g. Therefore, no threshold for damage is approached.

Figures 14-25 and 14-26 show the peak neutron and thermal power as a function of Doppler coefficient from -0.8 to -1.8 x 10^5 $(\Delta k/k)/°F$. It is seen that the variation is relatively small. Similar results are shown in Figures 14-27 and 14-28 for the variation of the moderator coefficient from -4.0 x 10^4 to +1.0 x 10^4 $(\Delta k/k)/°F$. The slope of the curve for 10^{-3} rated power at BOL is the greatest slope of any of the four curves because this case relies on the pressure trip, which makes it a longer transient. Figure 14-29 shows the effect of the trip delay time on the peak thermal power. It is seen that there is very little effect.

14.2.2.4.4.2 Rated Power

An analysis was performed for a BOL rod ejection at rated power. The results of this analysis are shown in Table 14-29. A sensitivity study was made around this case and around the same rod worth at EOL. Figures 14-22 through 14-29 show these results.

As seen in Figure 14-23, the peak thermal power shows relatively little change with increased rod worth. The peak neutron power in Figure 14-22 does show a marked change with increased worths, but the thermal effect is small because the transients are rapidly terminated by the Doppler effect. As further evidence of this small thermal effect, the peak fuel enthalpies are given in Figure 14-24. The threshold for the zirconium-water reaction is not reached until values of BOL and EOL ejected rod worths are above any that are considered feasible. The results of varying the Doppler and moderator coefficients and trip delay time show very little effect on the peak neutron and thermal powers.

The results of the DNB calculation for BOL are shown in Figure 14-30. For the BOL analysis, ejection of the maximum rod worth of 0.65 percent $\Delta k/k$ at rated power results in 2.6 percent of the core volume in DNB. This corresponds to 14 percent of the rods.

14.2.2.4.5 Energy Required to Produce Further Reactor Coolant System Damage

The reactor vessel has been analyzed to estimate the margin that exists between the rod worths assumed for the calculated rod ejection accident transients and those worths that could initiate RCS failure. The pressure vessel material is SA-533 Grade B steel. Table 14-30 lists the values used in this analysis. The radial deformation assumed to represent failure of the vessel is 50 percent of the total elongation, or 0.13 in./in. To calculate the weight of an explosive charge required to reach 50 percent elongation, the vessel is simulated by a single cylinder with the same OD as the actual vessel, but with an increased thickness to account for the thermal shield and core barrel.

Using this formula for the equivalent vessel, the required weight of explosive charge was calculated. The results indicate that 1,410 pounds of TNT would strain the mid-meridian ring up to the 50 percent ε_u, i.e. 0.13 in./in. The 1,410 pounds of TNT have an energy equivalent of 6.74×10^8 cal.

Ejected rod worths higher than those reported in the preceding sections were analyzed to estimate the transient required to obtain an energy release equivalent to 1,410 pounds of TNT. These cases were evaluated to find the amounts of fuel melting and zirconium-water reaction. Using the conservative assumption that all the fuel that exceeds the melting threshold is fragmented, dispersed into the coolant, and quenched to the coolant average temperature, a total thermal energy release can be determined. The conversion of this energy release to an equivalent deformation energy is dependent upon the duration of the release. TNT has an energy release in microseconds and a deformation conversion efficiency of about 50 percent. The energy generated during a reactor transient from the zirconium-water reaction and a molten fuel dispersal is in the range from milliseconds to seconds. Thus, the conversion efficiency to deformation energy would be considerably less and is assumed to be one-fifth that of TNT (See References 36 and 37). Using these figures, the reactor vessel's capability is 3.37×10^8 cal, and under the foregoing assumptions, a reactivity addition of 1.52 percent $\Delta k/k$ is required to release energy necessary to cause deformation of the vessel.

14.2.2.4.6 Conclusions

The hypothetical rod ejection accident has been investigated in detail at two different initial reactor power levels: rated power and zero power. Both BOL and EOL conditions were considered. The results of the analysis prove that the reactivity transient resulting from this accident will be limited by the Doppler effect and terminated by the Reactor Protection System with no serious core damage or additional loss of the coolant system integrity. Furthermore, it has been shown that an ejected rod worth greater than 1.52 percent $\Delta k/k$ would be required to cause a pressure pulse, due to prompt dispersal of fragmented fuel and zirconium-water reaction, of sufficient magnitude to cause rupture of the pressure vessel, whereas the maximum rod worth as shown in Table 14-26 is about a factor of two less.

As a result of the postulated pressure housing failure associated with the accident (see Section 14.2.2.4.1), reactor coolant is lost from the system. The rate of mass and energy input to the reactor building is considerably lower than that subsequently reported for the smallest rupture size considered in the loss of coolant analysis (see Section 14.2.2.5.5). The maximum hole size resulting from a rod ejection is approximately 2.76 inches. This lower rate of energy input results in a much lower reactor building pressure than those obtained for any rupture sizes considered in the Loss of Coolant Accident (LOCA). Reactor building leakage is conservatively assumed to occur at the rate associated with the peak calculated pressure for the design basis loss of coolant accident, 0.2% volume per day for the first 24 hours, and 0.1% per day thereafter.

The resultant doses from this accident are calculated assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Subsequently, this gap activity is released to the reactor building or the steam generators via primary-secondary leakage. For the case of a BOL rod ejection of the maximum rod worth of 0.65 percent $\Delta k/k$ at rated power, the fuel rods that experience DNB are assumed to fail, releasing activity as shown in Table 14-31.

Fission product activities released to the reactor building atmosphere for this accident are calculated using the methods described in NRC Regulatory Guide 1.183. The doses were also calculated per Regulatory Guide 1.183. The TEDE doses at the exclusion distance can be seen in Table 14-31. Also shown in Table 14-31 are the TEDE doses at the Low Population Zone (LPZ). No iodine removal by the spray or plateout on reactor building surfaces was assumed. These doses are less than the acceptance criteria of 10 CFR 50.67.

14.2.2.5 Loss of Coolant Accident

14.2.2.5.1 Identification of Accident

A Loss of Coolant Accident (LOCA) is defined as a failure of the RCS pressure boundary that would allow partial or complete release of reactor coolant into the reactor building, thereby interrupting the normal mechanism for removing heat from the reactor core. If all the coolant were not released immediately, residual heat, fission product heat, and possibly heat from chemical reactions would cause the remaining amount of coolant to be boiled off unless an alternate source of coolant were made available. Coolant from the Emergency Core Cooling System (ECCS) is available, and the fast cooling action provided by the Core Flooding Tank water prevents significant chemical reactions and destructive core heatup.

14.2.2.5.2 Accident Bases

14.2.2.5.2.1 ECCS Performance Criteria

The components of the RCS have been designed and fabricated to ensure high integrity and thereby minimize the possibility of their rupturing. The RCS, the safety factors used in its design, and the special provisions taken in its fabrication to ensure quality are described in Chapter 4. In addition, emergency core cooling is provided to ensure that the core will continue to be cooled and will not lose its geometric configuration even if the RCS boundary should fail and release coolant. This emergency core cooling standby safeguard is provided by the Core Flooding System and two independent, full capacity emergency core cooling strings. These systems are described in detail in Chapter 6.

The basic design objective for the ECCS equipment is to terminate the temperature transient and thus maintain core geometry for a LOCA. Specific criteria for ECCS performance have been established as regulatory requirements. The Final Acceptance Criteria for the performance of Emergency Core Cooling Systems are delineated in 10 CFR 50.46. The five Final Acceptance Criteria may be summarized as follows:

1. The calculated peak local (hotspot) cladding temperature shall not exceed 2200 °F.

2. The calculated total oxidation of the cladding shall not exceed 17 percent of the total cladding thickness at any location in the core. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation calculation beginning at the time of the rupture.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed that which would be generated by the oxidation of one percent of all reactor cladding.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. Long term core cooling shall be established such that the calculated core temperatures shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

ANO-1 meets the ECCS performance analysis requirements of 10 CFR 50.46.

14.2.2.5.2.2 ECCS Performance Analysis Assumptions

Compliance with the ECCS analysis requirements involves two separate, but related, analysis tasks. The first is the development of a valid evaluation model (described further in Section 14.2.2.5.3) and the second is the application of that model to a number of postulated loss of coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe LOCAs are assessed (presented in Section 14.2.2.5.4).

The Emergency Core Cooling System is designed such that its calculated cooling performance following postulated LOCAs conforms to the Final Acceptance Criteria. Breaks may be postulated in any pipe within the Reactor Coolant Pressure Boundary. The Reactor Coolant Pressure Boundary includes all piping connected to the Reactor Coolant System up to the outermost containment isolation valve in the system piping which penetrates primary reactor containment, or up to the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment (as defined by 10 CFR 50.2).

The analyses must demonstrate that for the postulated breaks and locations the results will be acceptable. The upper bound for the size of postulated LOCAs is twice the cross sectional area of the piping at the break location, which is equivalent to a double ended guillotine break allowing unimpeded flow from both pipe ends. The definition of the LOCA in 10 CFR 50.46 establishes the lower bound as a break which causes a leak rate in excess of the capability of the Reactor Coolant Makeup System. Because of the significant difference in the RCS response to the largest and smallest LOCAs, the analyses of large and small break LOCAs are treated slightly differently. The lower bound for a large break and the upper limit of a small break LOCA is 0.75 ft^2 (0.5 ft^2 for Mark-B-HTP fuel) (Reference 79).

The leak is assumed to occur instantaneously and may be located anywhere in the piping of the Reactor Coolant Pressure Boundary. Breaks may be double-ended with unobstructed flow from both ends or longitudinal splits.

For large breaks, offsite power is assumed to be lost at the time of the event. The single failure for the LBLOCA analysis is the loss of an EDG (Emergency Diesel Generator). For the limiting LBLOCA analysis, HPI flow is also not credited, due to the copious amount of ECCS available from the CFTs and LPI. Analyses have been performed for small breaks assuming both loss of offsite power and no loss of offsite power. Where offsite power is assumed available for small breaks, the worst case single failure is conservatively assumed to be a passive loss of a vital

bus, which will make one train of ECCS (one HPI pump and one LPI pump) unavailable. If the single failure were assumed to be an active failure only it would have been the loss of an HPI pump. Where the loss of offsite power is assumed, the worst case single active failure of an emergency diesel generator is assumed. For the analysis of reactor building backpressure, both trains of reactor building air coolers and sprays are assumed to operate which is inconsistent with the single failure, but conservative.

The reactor is assumed to trip on low RCS pressure; however, reactor shutdown due to partial rod insertion is credited for small breaks only during the blowdown phase for initial shutdown. For large breaks, moderator voiding alone is credited during the blowdown phase and a combination of moderator voiding in the core, boron injection and some of the negative reactivity from rod insertion is credited during the refill phase to maintain reactor shutdown.

Reactor Coolant Pumps are assumed to trip upon a Loss of Offsite Power for LBLOCA analyses. It was shown through sensitivity study that the RCS response with the pumps powered is less severe from a core cooling perspective than with pumps tripped (Ref. 79) for the limiting LBLOCA case. For small breaks, the pumps are assumed to trip and begin a coastdown coincident with reactor trip when the loss of offsite power is assumed. For the small breaks where the loss offsite power is not assumed, the reactor coolant pumps continue to run until 2 minutes after adequate vessel exit subcooling margin in lost, when they are assumed tripped by the operator.

Both Core Flood Tanks (CFTs) are assumed to provide injection flow for the cold leg pump discharge spectrum of breaks (small and large) and the HPI line break. Only one CFT is credited for the core flood tank line break, where one CFT’s contents is assumed spilled out the break. Where a loss of offsite power is assumed, with the assumed diesel generator single failure, only one train of High Pressure and Low Pressure Injection is assumed to operate. For the cold leg pump discharge small breaks where loss of offsite power is not assumed, the single failure has been conservatively assumed to the be the loss of a vital buss failing an HPI pump and an LPI pump in one ECCS loop, so that only one HPI pump, one LPI pump, and both CFTs are available. For conservatism, no HPI flow is credited in the analysis of large break LOCAs.

Both trains of reactor building air coolers and sprays are assumed to operate with the shortest response time consistent with no loss of offsite power. This is physically inconsistent with other assumptions, but the calculated results are conservatively higher with this equipment in operation.

For small break LOCAs, secondary cooling with feedwater from the Emergency Feedwater System and steam relief from the main steam safety valves is assumed. The turbine generator and main feedwater pumps are assumed to trip coincident with the reactor trip.

Operator action is credited to move the secondary water level in the steam generator from the natural circulation setpoint to the reflux boiling setpoint on the operate range to enhance boiler-condenser mode cooling in the steam generators once adequate vessel exit subcooling margin is lost.

Operator action is assumed for support of long-term core cooling. Operator action is required to switch LPI pump suction from the BWST to the reactor building emergency sump. Operator actions may also be required to maintain pumped injection flow by modifying ECCS lineups and to assure boron concentration control. Operator action is directed by Emergency Operating Procedures, as a backup to reactor internals leakage gaps, to establish LPI flowpaths that provide a secondary measure to preclude boron precipitation.

14.2.2.5.2.3 Additional Accident Considerations

In order to make an immediate identification of a small loss of reactor coolant, i.e. that which can be compensated for by normal makeup, the Reactor Coolant Leak Detection System (described in Section 4.2.3.8) is provided. The system consists of sump level monitoring, inventory balance checks, a radioactive gas detector, and an air particulate monitor. The reactor building area radiation monitors serve as a supplement to this system. The reactor coolant and reactor building pressure sensors of the Engineered Safeguards System provide immediate indication of a major loss of coolant and, in addition, activate the ECCS.

Immediately following a major LOCA, and up to a period of 24 hours following this accident, the pressure, temperature, and flow in the RCS can be monitored via the Reactor Protection System (described in Sections 7.1 and 7.3). These indicators verify that the performance of the ECCS in maintaining safe core conditions is within predicted values during the critical first 24 hours following the accident.

Following the first 24 hours, performance of the ECCS is verified by monitoring the operational status of ECCS components as well as ECCS flow rates. This information is available in the control room during the entire post-LOCA cooling period.

Performance of the Reactor Building Cooling System in maintaining the reactor building pressure within acceptable limits is monitored via the reactor building pressure sensors of the Engineered Safeguards System. This information, as well as the operational status of system components and system flow rates, are available in the control room throughout the entire post-LOCA cooling period.

The Primary Sampling System samples the reactor coolant at clad failure ≤ 5%. Spectroscopic analysis of the samples indicates the "trend" of the accident, thus assisting operator response.

Grab sample capabilities on the air particulate detector and the Combustible Gas Control System samplers allow for long-term monitoring of the reactor building atmosphere. These samples will be analyzed in the lab at the plant site which has the capability for spectroscopic analysis. The Combustible Gas Control System is also equipped with redundant hydrogen samplers which are discussed in Section 6.6.2.3. Any radioactive release to other plant buildings is monitored by portable samples which are analyzed in the lab, and various building area monitors (sensitivities and ranges for the area monitors are listed in Table 11-15). Radioactive releases to the atmosphere (offsite) are monitored by process radiation monitors in the Penetration Room Ventilation System and by stack radiation monitoring equipment. Indication of meteorological conditions is available to the operator at all times during these releases.

An emergency procedure is provided for the operators to follow in the event of an accident. This procedure instructs the operator to verify that the Engineered Safeguards Systems are all operating under accident conditions and are at normal flow conditions. If an abnormal indication appears on any of the instrumentation, the operator is instructed to take further corrective action to insure that the consequences of the accident are mitigated.

During plant operation the reactor building atmosphere is normally sealed in order to minimize the possible leakage paths from the building. Reactor building purge is closed in reactor operating conditions above cold shutdown per the Technical Specifications. However, prior to personnel entry, in either cold shutdown or refueling shutdown conditions (1CAN127908)

(NUREG-0737, Item II.E.4.2), the Reactor Building Purge System is operated to purge the reactor building atmosphere. If a release of radioactivity were to occur during this operation, a radiation detector located downstream of the reactor building purge exhaust filter (see Figure 9-13) would indicate the rate of release ($\mu Ci/cm^3$) to the control room operator.

Immediately following a design basis accident, the reactor building will be isolated. It is estimated that approximately 50 percent of any leakage will be through the Penetration Room Ventilation System which is monitored as shown in Figure 6-10.

In 1986, the internal hydrogen recombiners were installed in the Combustible Gas Control System to replace the hydrogen purge as the means of combustible gas control post-LOCA. (1CAN048604) The Hydrogen Purge System is, therefore, no longer considered a significant contributor to radioactive releases post-LOCA.

The Penetration Room Ventilation System radiation monitoring equipment is located downstream of the exhaust filters and is connected directly to the exhaust line. The equipment uses Geiger nuclear detectors and will detect up to 0.1 $\mu Ci/cm^3$ of ^{133}Xe.

Stack radiation monitoring equipment is designed to detect maximum permissible concentration release of radioactivity from the three stacks in accordance with 10 CFR 20. Absence of halogens is verified by lab analysis of particulate filters (99 percent efficient of 0.3 micron particles). The gas sampler consists of a scintillation detector and two ratemeters. One ratemeter provides a linear count of gross gamma with a 10^8 cpm indication range (± 3 percent linearity). The second ratemeter is a spectrometer and provides a log count for pulse height analysis. It also has an indication range of 10^8 (± 3 percent overall integral linearity and ± 10 percent linearity) when operating as a pulse height analyzer with a 50 keV wide differential energy discrimination.

The plant radiation monitors will be checked monthly to ensure that all components are functioning properly and will receive a full calibration at least once each 18 months. The site monitors will be checked and calibrated according to the same schedule. Calibration of the site monitors will be done by comparison of standards which are factory calibrated.

In the event that an accidental release of radiation occurs that is expected to affect offsite areas the State Radiological Response Team would be notified as outlined in the emergency plan. This team provides technical assistance to the State Department of Emergency Management (DEM) in determining the need for protective measures in offsite areas.

In addition, Arkansas Nuclear One personnel continue to evaluate the extent of the affected areas and provide input to the state agencies. The Arkansas Nuclear One personnel utilize meteorological data recorded in the control room (including temperature at 30- and 190-foot levels, wind velocity, and wind direction), plant radiation monitor readings, prepared diffusion map overlays, and input from Arkansas Nuclear One emergency teams doing radiation surveys in order to follow the course of the accident.

The following is a list of some types of instrumentation that are provided for use by the Arkansas Nuclear One emergency teams:

A. High range ionization chambers providing direct gamma dose radiation measurements. Two different chambers are used; one having six scales in the range 0 - 500 rem/hour and one having two scales in the range 1 - 1000 rem/hour.

B. Thin end window G-M count rate meters to check for contamination with a range of 0 - 280,000 cpm. This meter utilizes a shielded pancake probe and also has a sample holder attachment for smears and air particulate samples and a pulse integrator to convert it to a portable scaler.

C. Battery powered air samplers for obtaining air particulate samples.

D. Environmental monitoring program equipment including TLD chips and air samplers with charcoal cartridges for iodine readings.

E. Dosimeter of Legal Record (DLR) badges and dosimeters for team members.

The reactor building pressures have been evaluated for a complete spectrum of rupture sizes. Rupture sizes smaller than the 36-inch-ID break result in longer blowdown times, and the amount of energy transferred to the reactor building atmosphere is increased. As a result, intermediate break sizes result in reactor building pressures slightly greater than that produced by the 36 inch ID, double-ended pipe rupture. The peak reactor building pressure results from a 5 ft^2 break. The reactor building analysis and results are covered in Section 14.2.2.5.5. The analysis presented in Section 14.2.2.5.5 demonstrates that in the unlikely event of a failure of the RCS, the reactor building integrity is maintained.

14.2.2.5.3 ECCS Performance Evaluation Model

The LOCA evaluation model is a structured arrangement of several individual computer codes and supporting analyses used by B&W to analyze LOCAs specifically for B&W plants. The evaluation model describes how the individual computer codes and supporting analyses are used with one another to demonstrate that the Emergency Core Cooling Systems supplied with the B&W NSSSs are in compliance with each of the five Final Acceptance Criteria of 10 CFR 50.46. The evaluation model includes all of the codes and supporting analyses necessary to demonstrate compliance with the five criteria of 10 CFR 50.46.

The ECCS evaluation models are computerized mathematical representations of a Nuclear Steam Supply System which solve equations for heat transfer and fluid dynamics as a function of time over the course of an accident. The models are flexibly structured and can be modified to describe transients on NSSSs of various sizes and configurations. The phenomena associated with large and small break LOCAs are sufficiently different to warrant separate models.

The evaluation models predict the RCS response and fuel cladding conditions for each modeled event. By changing the input assumptions, the model can be applied to a spectrum of postulated breaks for a given plant. This series of application analyses must demonstrate compliance with regulatory acceptance criteria.

10 CFR 50.46 includes specific requirements for the features, allowable techniques, and characteristics of the evaluation models. The evaluation models may be developed as best estimate analyses supported by detailed evaluations of analysis uncertainties or they may contain the conservative features and techniques delineated in Appendix K of 10 CFR 50. The B&W model conforms to the Appendix K requirements.

Evaluation models for large and small break LOCAs have been described separately.

14.2.2.5.3.1 Large Break LOCA Evaluation Model

The current B&W large break LOCA evaluation model is documented in Volume 1 of BAW-10192, Rev. 0 (Reference 79). This document describes the basic functions and features of the individual computer codes used to calculate peak cladding temperatures and local cladding oxidation and the supporting analyses used to demonstrate compliance with the other acceptance criteria. The applicability and limitations of the model are defined. The documentation also demonstrates how the evaluation model complies with the requirements of Appendix K of 10 CFR 50. Supplement 1P-A of BAW-10192 describes the approved methodology to address fuel thermal conductivity degradation.

BAW-10192, Revision 0, includes references to many other topical reports, correspondence, and related documents that provide the details of the model and its use. For example, the individual computer codes used in the evaluation model are described in separate topical reports.

A. Peak Cladding Temperature/Local Cladding Oxidation Analysis

The majority of all LOCA analysis efforts have been focused on the calculation of post accident peak cladding temperatures. These analyses also calculate local cladding oxidation due to the metal/water reaction. The major codes used in large break LOCA cladding peak temperature/local oxidation analyses include RELAP5/MOD2-B&W, REFLOD3B, CONTEMPT, and BEACH. These codes are supported by other minor codes, such as TACO3, which have specialized functions.

For the analysis of a given break, the RELAP5/MOD2-B&W code (Reference 75) is run first to establish the core power generation and thermal--hydraulic behavior of the primary system during blowdown. RELAP5/MOD2-B&W uses initial fuel thermal condition input from the fuel performance code TACO3 (Reference 85). RCS and core parameters calculated by RELAP5/MOD2-B&W at the end of blowdown are input into REFLOD3B. Blowdown mass and energy release data from RELAP5/MOD2-B&W are input into CONTEMPT.

The REFLOD3B code (Reference 82) is a simplified thermal-hydraulics code used to calculate the time to refill the vessel to the bottom of the core and to calculate the rate of reflooding of the core. The code uses assumptions which assure that the calculated reflood rates are conservatively low. This in turn assures conservatively high calculated peak cladding temperatures during the reflood period. REFLOD3B develops mass and energy release data for the refill and reflood phases for input into CONTEMPT.

The CONTEMPT code (Reference 77) calculates the containment pressure following a LOCA. This calculation is not used for containment analysis purposes; it is used only for LOCA analyses. The CONTEMPT code uses assumptions which assure a conservatively low calculated pressure. This in turn assures a conservatively low calculated reflood rate and high peak cladding temperatures. CONTEMPT is run concurrently with RELAP5/MOD2-B&W and REFLOD3B using mass and energy release data calculated by these codes. The calculated containment back pressure is then compared to the assumed values in REFLOD3B and RELAP5/MOD2-B&W. If the values assumed by REFLOD3B and RELAP5/MOD2-B&W are less than those calculated by CONTEMPT, the REFLOD3B and RELAP5/MOD2-B&W results are acceptable.

With RCS and core conditions from RELAP5/MOD2-B&W during blowdown and from REFLOD3B during refill and reflood, BEACH (Reference 76) calculates the hot pin temperature response, the metal-water reaction, and the peak cladding temperature during all three phases.

B. LOCA kW/ft Limits Analysis

The Appendix K requirement for consideration of the possible range of core power distributions is satisfied by a series of analyses with power distribution peaks at five different core elevations (2, 4, 6, 8, and 10 feet). For each of these analyses the assumed pre-accident peak power, expressed in kW/ft, is adjusted such that post-accident core thermal conditions meet the Final Acceptance Criteria. These kW/ft values, calculated for each fuel cycle, are the LOCA limits maintained in the Core Operating Limits Report, which are protected by the operating limits on rod insertion and axial power imbalance.

When analyzing LOCA limits, the limiting break analysis is run once for each LOCA limit core elevation (2, 4, 6, 8, and 10 ft). The axial power distribution for each run is peaked at the elevation of interest. The peak linear heat rate (kW/ft) assumed in RELAP5/MOD2-B&W is greater than or equal to the anticipated LOCA limit linear heat rate for that elevation. The method used in Section A can be repeatedly run for each elevation, with a progressively lower peak linear heat rate, until peak cladding temperatures are within limits. Additional runs may be performed to evaluate the effects of cycle burnup. The final linear heat rates assumed become the LOCA limits.

C. Maximum Hydrogen Generation Analysis.

To determine the maximum quantity of hydrogen generated following a LOCA, the metal/water reaction rate over the entire core must be considered. The maximum hydrogen generation analysis, based on RELAP5/MOD2-B&W, uses very conservative, simplifying assumptions to estimate the maximum rates of cladding oxidation over the entire core. The whole-core oxidation increase is provided by one of two methods: 1) a comparison of the hot channel average oxidation increase with the 10 CFR 50.46 criteria of less than 1%, or 2) a more detailed calculation using the hot and average channel oxidation increases with a typical core power map. If the first method is successful, the second method is unnecessary and is not pursued, since the oxidation rate increases exponentially with temperature. The oxidation increases for each of the power shapes used in the LOCA limits study are evaluated.

D. Coolable Core Geometry

To demonstrate compliance with the Final Acceptance Criterion on a coolable core geometry, B&W (Framatome) has considered the effects of fuel rod swelling, rupture, and bowing (including NUREG-0630 considerations) and the effects of possible mechanical loads caused by a LOCA. As described in BAW-10192-A (Reference 79), analyses by B&W and other industry research demonstrates that no gross core blockage or disfiguration will occur to preclude adequate core cooling.

E. Long-Term Core Cooling

The analyses of peak cladding temperatures demonstrate the ability of the ECCS to cool the core during the complex initial thermal-hydraulic transient. Once this capability has been established, compliance with the long-term cooling criterion requires the ability to maintain a continuous supply of injection flow to the core.

The requirements for demonstrating compliance with the long-term cooling criterion are generically described in BAW-10192-A (Reference 79). The ANO-1 ECCS characteristics necessary to support long term cooling configurations are described in Chapter 6 of the SAR. Operator actions required to establish and maintain long-term cooling configurations are incorporated in the emergency operating procedures.

14.2.2.5.3.2 Small Break LOCA Evaluation Model

The current B&W small break LOCA evaluation model is documented in BAW-10192-A (Reference 79). This document describes the basic functions and features of the individual computer codes used to calculate peak cladding temperatures and local cladding oxidation. The applicability and limitations of the model are defined. The documentation also demonstrates how the evaluation model complies with the requirements of Appendix K of 10 CFR 50.

BAW-10192-A is the primary documentation of the small break LOCA evaluation model. It includes references to many other topical reports, correspondence, and related documents that provide the details of the model and its use. For example, the individual computer codes used in the evaluation model are described in separate topical reports.

The RELAP5/MOD2-B&W code (Ref. 75) is used to predict the reactor coolant system and core response including the hot channel and average channel cladding temperatures. The containment pressure is set conservatively high or provided by CONTEMPT (Ref. 77) analysis where the break could unchoke. Because of the difference between the large and small break RCS response, the noding schemes used by RELAP5/MOD2-B&W are different for the large and small break models. For example, the small break model has a more detailed representation of the steam generators than the large break model.

14.2.2.5.4 Accident Analysis Results

ANO-1's compliance with the Final Acceptance Criteria has been demonstrated by large and small break LOCA analyses performed with the B&W evaluation models. These analyses have been documented in Framatome Reports 86-5064591-02 (Ref. 78) and FS1-0039424 (Ref. 84).

The B&W 177 fuel assembly, lowered loop plants of both 2568 and 2772 MWt ratings have generally been treated as a single class of plant for ECCS analysis purposes. The analyses supporting ANO-1 in total are described in References 78 and 84.

14.2.2.5.4.1 Large Break LOCA Analyses

A. Large Break Spectrum Analyses

10 CFR 50.46 requires the analysis of a spectrum of postulated break sizes and locations to assure that the worst possible results are identified. To comply with this requirement, analyses of breaks at various locations were first performed to determine

the worst break location. A spectrum of break types and sizes was then analyzed at the worst break location to identify the specific break with the most limiting results. The limiting break analysis forms the basis for the subsequent analyses of the LOCA limits.

The spectrum of analyses which identified the most limiting break location, size, and type is documented in Reference 78. This study applies to both Mark-B9 and Mark-B-HTP fuel types. To determine the worst break location, analyses were performed for several break sizes up to 8.55 ft^2 in the cold leg break at the pump discharge, an 8.55 ft^2 break at the pump suction, and for the 14.14 ft^2 hot leg break. Because of the increased ECCS bypass flow and the more restricted steam venting path during reflood, the cold leg break at the pump discharge is the worst break location. A spectrum of analyses for various break sizes at this location demonstrated that the largest (8.55 ft^2) double ended break produced the highest peak cladding temperatures.

B. LOCA Limits Analyses

The LOCA kW/ft limits provide a useful means of compensating for subsequent changes in the evaluation model and in fuel cycle design which would otherwise have resulted in exceeding the criteria. The LOCA limits are the primary adjustable variable used to assure compliance with 10 CFR 50.46. The LOCA limits ECCS analyses complying with 10 CFR 50.46 are documented in References 78 and 84.

The fuel type-specific LOCA analyses are performed with the limiting set of inputs developed by the sensitivity study results to show compliance with 10 CFR 50.46. The base case is a full double-area, guillotine break in the cold leg pump discharge piping at the elevation of the reactor vessel inlet nozzle. A discharge coefficient of 1.0 is used to maximize the break flow. A loss of offsite power is assumed at the time of break opening, so the reactor coolant pumps and main feedwater pumps are not powered. Other major assumptions include:

1) Plant-average 20% tube plugging in the steam generators

2) Byron-Jackson homologous head flow curves with RELAP5 Semiscale two-phase head difference curves and head degradation using the M3-modified (minimum degradation) two-phase multiplier

3) Non-mechanistic ECCS bypass method is used during blowdown

4) Time delay of 35 seconds after the ESAS RCS low pressure trip setpoint

5) No HPI flow available for vessel refill

6) RCP (Reactor Coolant Pump) rotors are assumed to be in a fixed position during refill and reflood

7) Maximum BWST fluid temperature

8) Minimum (one train) of pumped ECCS fluid for reflood condition

9) Maximum (two trains) of ECCS (including HPI) available for reactor building pressure response

Other ANO-1 analyses assumptions and input are given in References 78 and 84.

The impact of changes to LOCA limits is summarized in the reload report for each fuel cycle which includes appropriate references to supporting analysis documentation describing these changes. The current ANO-1 LOCA limits are maintained in the Core Operating Limits Report.

C. Limiting Large Break Results

Figures 14-31 through 14-33 show typical RCS pressure, break mass flow rate and the PCT for an 8.55 ft^2 break at the cold leg pump discharge (the worst-case LBLOCA). The actual results for the current licensing basis were generated for the LBLOCA case that produced the highest PCT based on the worst time-in-life and the axial power peak occurring at a specific core axial position as documented in the FANP Report 86-5064591-02 (Ref. 78). The RCS/core response shown in these figures is typical of the worst-case break in a B&W 177 fuel assembly lowered loop plant, although subsequent evaluation model improvements would alter this response slightly. The responses given will vary slightly based on different times-in-life, linear heat rates and axial power distribution assumptions.

The five 10 CFR 50.46 ECCS acceptance criteria were determined to be met for the LBLOCA event as follows:

1) The LHR limits are chosen such that the PCT at the worst core elevation for the limiting LBLOCA is predicted to be less than 2200 °F. This includes the ruptured and the unruptured segments.

2) The local oxidation values were calculated using a conservative initial oxide thickness to maximize the cladding temperature response due to metal-water reaction. The calculated local cladding oxidation values were calculated to not exceed 4 percent for any case. Therefore there is sufficient margin to the 17 percent local oxidation limit to demonstrate that the local oxidation limit of 17 percent is met.

3) The maximum whole-core hydrogen generation for the Mark-B9 and Mark-B-HTP fuel was estimated to be less than 0.3 percent for all cases. Therefore the criterion that the total amount of hydrogen generated shall not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the fuel rod cladding reacted with steam or water is met.

4) The effects of the pin rupture and the fuel rod bowing are assessed during the LOCA. Deformation of the fuel pin lattice at the core periphery is allowed to occur from the combined mechanical loading of the LOCA and a seismic event. Using the (leak-before-break) LBB methodology, the spacer grid impact loads are within the spacer grid elastic load limit and no permanent grid deformation is predicted. The resulting consequences from both thermal and mechanical deformation of the fuel assemblies have been shown to maintain a coolable core geometry.

5) Long-term cooling is achieved by the pumped injection systems that are redundant and provide continuous flow of cooling water to the core fuel assemblies. Demonstration that the entire core has quenched and the cladding temperatures have returned to approximately the saturation temperature shows

successful initial operation of the ECCS. Thereafter, long-term cooling is achieved by maintaining the operation of the pumped injection systems while taking suction from the BWST or the reactor building sump. Therefore the capability of long-term cooling has been established.

LBLOCA analyses were performed for both the Mark-B9 and the Mark-B-HTP fuel. The fuel-type-specific LOCA limits are based on analysis results for a full core of either fuel type or a mixed core of both fuel types. The analysis for a full core of Mark-B9 fuel produced the limiting LOCA limits for that fuel type, while the analysis for a mixed core of Mark-B9 and Mark-B-HTP fuel produced the limiting LOCA limits for the Mark-B-HTP fuel. Therefore, the full-core Mark-B-HTP LOCA limits incur a penalty when Mark-B9 assemblies are present. The fuel-type-specific LOCA limits based on core configuration are presented in Reference 78 and are reported in the cycle-specific Reload Report, i.e. Chapter 3A of this document. LBLOCA limits based on the NRC-approved LBLOCA methodology, are presented in Reference 84 and will be reported in the cycle-specific Reload Report (Chapter 3A) starting the Cycle 29.

D. Large Break Analysis Documentation

ANO-1 compliance with the five Final Acceptance Criteria is documented in References 78 and 84 and those references identified in the cycle reload report as pertinent to the LOCA analyses.

Any subsequent changes to the LOCA analysis and to the LOCA kW/ft limits as documented in References 78 and 84 are contained in the cycle reload report. The LOCA limits themselves for the current cycle are maintained in the Core Operating Limits Report.

The approval of the use of the EM (Evaluation Model) (Ref. 79) was given in the SER dated February 18, 1997 (Ref. 68). The SER delineated the conditions upon which the EM was acceptable for reference in licensing applications. The following conditions have been addressed in the ANO-1 application of the EM as documented in References 78 and 84 relating to LBLOCA:

1) The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model.

2) The guidelines, code options and prescribed input specified in the EM documentation (Ref. 79) should be used.

3) The limiting LHR (Linear Heat Rate) for LOCA limits is determined by the power level and the product of the axial and radial peaking factors. An appropriate axial peaking factor for use in determining LOCA limits is one that is representative of the fuel and core design and that may occur over the core lifetime. The radial peaking factor is then set to determine the limiting linear heat rate. As future fuel or core designs evolve, the basic approach that was used to establish these conclusions may change. FANP must revalidate the acceptability of the evaluation model peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limits differ appreciably from those used to demonstrate Appendix K compliance.

4) The mechanistic ECCS bypass model is acceptable for cold leg transition (0.75 ft^2 to 2.0 ft^2 for Mark-B9 fuel and 0.5 ft^2 to 2.0 ft^2 for Mark-B-HPT fuel) and hot leg break calculations. The nonmechansitic ECCS bypass model must be used in the large cold leg break methodology since the demonstration calculations and sensitivities were run with this model.

5) Time-in-life LOCA limits must be determined with, or shown to be bounded by, a specific application of the NRC-approved evaluation model.

6) LOCA limits for three-pump operation must be established for each class of plants by application of the methodology described in this report. An acceptable approach is to demonstrate that three pump operation is bounded by four pump LHR limits.

7) The limiting ECCS configuration, including minimum versus maximum ECCS, must be determined for each plant or class of plants using this methodology.

8) B&W-designed plants have internal RVVV (reactor vessel vent valves) that provide a path for core steam venting directly to the cold legs. The BWNT LOCA evaluation model credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction seal to clear during blowdown and then reform during reflood before the evaluation model analyses predict average core quench. Since the REFLOOD3B code cannot predict this reformation of the loop seal, FTI is required to run the RELAP5/MOD2-B&W system model until the whole core quench, to confirm that the loop seal does not reform. This demonstration should be performed at least once for each plant type and be judged applicable for all LBLOCA break sizes.

The key approval documentation for the original ANO-1 licensing basis for LOCA analysis is the NRC Safety Evaluation Report for Technical Specification Amendment 21 and the NRC Safety Evaluation Report concerning the ECCS single failure analysis dated October 12, 1979 (1CNA107915). The current methodology as described in Reference 79 has replaced the original approved methodology, however the original methodology's approval documentation may form a partial licensing basis for the current methods. As such, the reference section to this SAR chapter provides a list of correspondence between ANO or B&W and the NRC under "Additional Correspondence" associated with the approval of the original licensed methodology and analysis. These references are provided for information only.

14.2.2.5.4.2 Small Break LOCA Analyses

The EM studies determined that the most limiting SBLOCA break location is in the bottom of the RCS cold leg piping between the reactor vessel inlet nozzle and the HPI nozzle, also described as at the cold leg pump discharge (CLPD). ANO-1's compliance with the Final Acceptance Criteria is demonstrated by performing analyses on the spectrum of a number of break sizes at the CLPD location, plus a core flood tank (CFT) line break and an HPI line break.

A. Small Break Spectrum, CFT Line Break and HPI Line Break Results

Several specific break areas, in square feet, were analyzed at the CLPD location ranging in size from 0.01 to 0.75. Based on similarities in transient response, SBLOCAs can be grouped into three categories: small, intermediate, and large.

The smaller LOCAs, between approximately 0.01 ft^2 and 0.08 ft^2, present the greatest challenge to the HPI system to replace lost liquid inventory before significant fuel or clad damage occurs. These break sizes are not large enough to rapidly depressurize the RCS to the CFT fill pressure. Consequently, the HPI flow reaching the reactor vessel must be sufficient to match the core decay heat.

The smallest break size analyzed was 0.01 ft^2. Breaks smaller than 0.01 ft^2 will produce a slower RCS inventory loss rate and will be less challenging to the capacity of the ECCS to provide adequate core cooling. This break size demonstrates the effectiveness of continuous SG heat removal via EFW preservation and level control.

Even for these relatively small break sizes, the RCS depressurization and voiding quickly interrupted loop flow. With primary-to-secondary heat transfer interrupted, an RCS repressurization was predicted for the 0.01- and 0.08-ft^2 breaks in this category. For all the break sizes in this category, the RCS pressure response caused delays in ESAS actuation, and, consequently, in the start of HPI flow.

Most of the small SBLOCA cases, experienced partial core uncovering. The duration of core uncovering was a direct result of the sustained high RCS pressure, which limited the HPI flow and delayed the start of CFT injection. For the small SBLOCA cases analyzed, the RCS pressure did not depressurize sufficiently to allow LPI flow to enter the reactor vessel. In each case, at the time the analysis was ended, the core was completely recovered, the downcomer level was increasing, and the HPI flow matched flashing plus core boil-off due to decay heat and wall metal heat contributions. These conditions confirmed that the HPI flow was adequate to prevent significant cladding temperature excursions and to ensure long-term cooling via preservation of HPI and EFW.

The intermediate small breaks were analyzed in the RCP discharge piping with break areas of above 0.08 up to about 0.15 ft^2. The intermediate break sizes caused the RCS to depressurize faster, and to enter into the boiling pot mode sooner, than the smaller breaks. The intermediate breaks continuously depressurized the RCS after the forced flow phase ended, although the depressurization rate slowed temporarily. The RCS depressurization rate increased after the steam venting path through the RVVVs was established.

ESAS was actuated relatively early in the transient. The RCS depressurization rate also was not significantly affected when the RCS pressure decreased below the secondary pressure and secondary-to-primary (reverse) heat transfer began. The intermediate breaks were able to discharge fluid mass, energy, and volume so effectively that they cooled and depressurized both the primary and the secondary systems. Even though the RCS depressurized fairly quickly, allowing the CFT injection to begin relatively early in the event, partial core uncovering did occur for the intermediate breaks. As a result, clad heatup occurred in the upper core regions.

When the analysis of the intermediate small breaks ended, the ESAS low RCS pressure setpoint had been reached but the RCS pressure remained above the LPI shutoff head. Furthermore, the pressure was decreasing and sufficient liquid remained in the CFTs to sustain CFT injection until LPI flow could begin. Therefore, the analyses confirmed that, for intermediate breaks, the ECCS capacity was sufficient to recover the core and to provide adequate long-term cooling.

The large SBLOCAs, with break areas of above 0.15 ft^2 up to 0.75 ft^2, were also analyzed. For these large SBLOCA sizes, the effects of the break dominated other factors (such as the timing and magnitude of EFW flow, BCM cooling, and reverse heat transfer) that could potentially affect the RCS depressurization rate.

For all the large SBLOCA cases, the low-pressure reactor trip setpoint and subsequently, the ESAS was actuated quite a bit earlier than for the intermediate SBLOCAs. In comparison with the smaller breaks, the HPI, CFT, and LPI delivery during the large SBLOCAs was enhanced by the lower transient RCS pressures, i.e. the actuation times were earlier and the flow rates were higher. CFT injection started soon after the onset of core uncovering in both cases. The rapid RCS depressurization rates caused by the large SBLOCAs produced significant core voiding. The improved ECCS performance significantly shortened the duration of core uncovering and produced lower PCTs than in the intermediate size SBLOCAs.

LPI flow was initiated for the large SBLOCA cases, but after the HPI and CFT flow had limited the clad temperature increase and completely refilled the core. However, while the HPI and CFTs had demonstrated adequate short-term core cooling, the availability of the LPI ensured that adequate long-term cooling would continue after the CFTs emptied.

Results of the limiting SBLOCA are presented in the next section.

For the CFT line break, a more severe degradation of the ECCS capacity must be considered than for a CLPD break. During a CFT line break, the break location prevents one CFT and one LPI train from injecting into the reactor vessel. The other LPI train and one HPI train are assumed to be unavailable due to a single failure. Therefore, only one CFT and one train of HPI remain available for core cooling. However, since the HPI piping remains completely intact, all of the flow from the available HPI pump is able to enter the RCS. For the CFT line break, the nominal CFT line resistance, minimum CFT inventory and maximum fill pressure were modeled. The break was modeled to occur at the connection of the CFT line nozzle to the reactor vessel. The break area was limited to 0.44 ft^2 by the cross-sectional area of the nozzle insert. This break was analyzed with both a LOOP (loss of offsite power) and no LOOP. The no-LOOP case was limiting.

The relatively large break size of the CFT line break with no LOOP caused a rapid RCS depressurization. The timing of CFT actuation and the available HPI flow were not sufficient to maintain core covering. However, the results of this case also demonstrated that one CFT and one HPI pump provide adequate ECCS flow to mitigate the consequences of a CFT line break.

The HPI mode has two independent trains of flow to supply borated coolant to the system. If the HPI piping were to rupture in one of the four lines between the attachment to the reactor coolant pipe and the check valve, then the other three lines of this system would have adequate capacity to protect the core against this small leak.

The HPI line break location was modeled in an HPI line just upstream of the HPI nozzle, and the thermal sleeve in the nozzle was assumed to be blown out coincident with the break opening. The break area is limited by the cross-sectional area of the pipe without the thermal sleeve (0.02463 ft^2). HPI flow from the broken line was assumed to spill into the containment and was, therefore, not modeled.

Manual valves in each of four HPI lines are throttled to allow a total HPI flow of between 480 and 500 gpm with 600 psig RCS pressure with a nominal pump. This resistance will limit the flow that would be lost if one of the HPI lines were broken, and will increase the flow injected through the intact lines. If an HPI line break occurs with only one train available, the operator will throttle the line with the highest flow until it is indicated to be within 20 gpm of the flow in line with the next highest flow. The prethrottling of the manual valves and the operator action to balance the HPI system will assure adequate flow in the intact lines even in the case of an HPI line break concurrent with the loss of one train of safety grade power.

The results of the HPI line break analysis indicate no core uncovering and no cladding heat up. The timing of CFT actuation and the available HPI flow were sufficient to maintain core covering throughout the event. Therefore, the HPI system has sufficient capacity to provide adequate short- and long-term core cooling for the HPI line break.

B. Limiting Small Break analysis

The SBLOCA CLPD spectrum was evaluated for both a LOOP (loss of offsite power) and no-LOOP condition. The no-LOOP condition credits operator action to trip the RCPs (Reactor Coolant Pumps) two minutes after the loss of subcooling margin and was found to be limiting.

The limiting SBLOCA in terms of PCT was evaluated to be a 0.15 ft^2 break at the CLPD with no LOOP as documented in the Reference 78. This break size produces the limiting PCT for both the Mark-B9 and Mark-B-HTP fuel types. This size SBLOCA's analysis results also bounded the CFT line break and the HPI line break analysis results. The PCT calculation includes conservatism in that a bounding power level and lesser ECCS flow is assumed compared to actual plant parameters. Other major input and assumptions for both the LOOP and no-LOOP cases are given in Reference 78.

Figures 14-34 through 14-36 depict several critical parameters' responses for a spectrum of SBLOCA sizes with a Mark-B-HTP/Mark-B9 mixed core, including the limiting break size of 0.15 ft^2. Figure 14-34 shows the RCS pressure to decrease very quickly to the RPS low pressure trip setpoint and subsequently to the ESAS low pressure trip setpoint. The pressure continued to decrease through the CFT injection setpoint and finally past the point at which LPI will start to enter the vessel. The transient was terminated after the core was completely quenched. Figure 14-35 shows that the mixture level uncovers the core (hot channel) for a period of time and recovered the core well before 400 seconds. Figure 14-36 demonstrates that the PCT is predicted to occur shortly after 200 seconds, with the core totally quenched a few seconds later. Though the core was uncovered for a short period of time, no fuel cladding rupture was predicted.

The five 10 CFR 50.46 ECCS acceptance criteria were determined to be met for the SBLOCA event as follows:

1) The PCT for the limiting SBLOCA was calculated to be well below the PCT limit of 2200 °F. In addition, no fuel cladding rupture was indicated in the analysis.

2) The local oxidation values were calculated using a conservative initial oxide thickness to maximize the cladding temperature response due to metal-water reaction. The calculated local cladding oxidation values were calculated to be less than 1 percent for any case. Therefore there is sufficient margin to the 17 percent local oxidation limit to demonstrate that the local oxidation limit of 17 percent is met.

3) The maximum whole-core hydrogen generation for the Mark-B9 fuel was estimated to be less than 0.1 percent for all cases. Therefore the criterion that the total amount of hydrogen generated shall not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the fuel rod cladding reacted with steam or water is met.

4) The effects of the pin rupture and the fuel rod bowing are assessed during the LOCA. Deformation of the fuel pin lattice at the core periphery is allowed to occur from the combined mechanical loading of the LOCA and a seismic event. Using the (leak-before-break) LBB methodology, the spacer grid impact loads are within the spacer grid elastic load limit and no permanent grid deformation is predicted. The resulting consequences from both thermal and mechanical deformation of the fuel assemblies have been shown to maintain a coolable core geometry.

5) Long-term cooling is achieved by the pumped injection systems that are redundant and provide continuous flow of cooling water to the core fuel assemblies. Demonstration that the entire core has quenched and the cladding temperatures have returned to approximately the saturation temperature shows successful initial operation of the ECCS. Thereafter, long-term cooling is achieved by maintaining the operation of the pumped injection systems while taking suction from the BWST or the reactor building sump. Therefore the capability of long-term cooling has been established

C. Small Break Analysis Documentation

The approval of the use of the EM (Evaluation Model) (Ref. 79) was given in the SER dated February 18, 1997 (Ref. 74a). The SER delineated the conditions upon which the EM was acceptable for reference in licensing applications. The following conditions have been addressed in the ANO-1 application of the EM as documented in Reference 78 relating to SBLOCA:

1) The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model.

2) The guidelines, code options and prescribed input specified in the EM documentation (Ref. 79) should be used.

3) The limiting ECCS configuration, including minimum versus maximum ECCS, must be determined for each plant or class of plants using this methodology.

4) For the small break model, the hot channel radial peaking factor to be used should correspond to that of the hottest rod in the core, and not to the radial peaking factor of the 12 hottest bundles.

5) The constant discharge coefficient model (discharge coefficient = 1.0) referred to as the “High or Low Break Voiding Normalized Value,” should be used for all small break analyses. The model which changes the discharge coefficient as a function of void fraction, i.e., the “Intermediate Break Voiding Normalized Value”, should not be used unless the transient is analyzed with both discharge models and the intermediate void method produces the more conservative result.

6) For a specific application of the FTI small break LOCA methodology, the break size which yields the local maximum PCT must be identified. In light of the different possible behaviors of the local maximum, FTI should justify its choice of break sizes in each application to assure that either there is no local maximum or the size yielding the maximum local PCT has been found. Break sizes down to 0.01 ft^2 should be considered.

The key approval documentation for the original ANO-1 licensing basis for LOCA analysis is the NRC Safety Evaluation Report for Technical Specification Amendment 21 and the NRC Safety Evaluation Report concerning the ECCS single failure analysis dated October 12, 1979 (1CNA107915). The current methodology as described in Reference 79 has replaced the original approved methodology, however the original methodology’s approval documentation may form a partial licensing basis for the current methods. As such, the reference section to this SAR chapter provides a list of correspondence between ANO or B&W and the NRC under “Additional Correspondence” associated with the approval of the original licensed methodology and analysis. These references are provided for information only.

14.2.2.5.5 Reactor Building Pressure Analyses

14.2.2.5.5.1 Introduction and Summary

In the event of a hypothetical LOCA, the release of the reactor coolant from the rupture area will cause the high pressure, high temperature liquid to rapidly flash to steam and water within the reactor building. The release of this mass and energy will result in a rise in the pressure and temperature of the reactor building atmosphere.

The design of the reactor building is based upon the mass and energy absorption capacity of the volume contained within the structure. The net free volume of the reactor building is a minimum of 1.81×10^6 ft^3. The design pressure is 59 psig and the design temperature is 286 °F. These design values have been determined, based upon the results of the pressure-temperature analyses given in this section, such that an adequate safety margin exists between the highest calculated result and the design values.

A spectrum of RCS break sizes have been considered in the evaluation of the reactor building design. Break sizes from 14.1 ft^2, corresponding to a double-ended guillotine rupture of a RCS hot leg, to a 1 ft^2 hot leg rupture were studied to determine the worst condition of RCS mass and energy releases in combination with sensible and shutdown heat sources during the blowdown phase of the LOCA. The reactor building responses to these break sizes were analyzed assuming a minimum of operable ECCS and without Reactor Building Cooling Systems (RBCS) in operation until blowdown is completed. This technique assures a conservative prediction of the response during the blowdown period of the LOCA.

Analyses of the pressure-temperature response to the LOCA were performed using the digital computer program COPATTA. COPATTA was developed by Bechtel Corporation and was derived from the CONTEMPT (See Reference 25.) code.

The results of the blowdown period pressure transient analyses show that the 5 ft^2 break case gives the maximum reactor building pressure and is therefore the containment design basis accident (DBA). The pressure resulting from the 5 ft^2 hot leg break is calculated using the mass and energy input obtained from a FLASH computer code analysis which assumes a positive moderator coefficient of $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ and assumes no credit for control rod insertion. The mass and energy calculated by FLASH is overly conservative since these assumptions result in appreciably more heat being generated in the core than would otherwise be expected.

For the spectrum of breaks greater in size than the 5 ft^2 case, core voids are created faster resulting in less heat being generated in the core prior to void shutdown. Therefore, building pressure calculations for these cases result in lower peak building pressures. For the analysis of breaks smaller in size than the 5 ft^2 case, rods are assumed to be inserted following reactor trip resulting in a faster core shutdown.

A conservative positive moderator coefficient of $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ is also assumed for these analyses. Since the core power is reduced much faster, the total heat generated by the core will be less than that for the 5 ft^2 case resulting in a lower peak building pressure.

The conservative mass and energy release values, for the most restrictive 5 ft^2 break case, are used as inputs to COPATTA. Given the ANO-1 design values for initial containment conditions and containment cooling system parameters, (see Tables 14-39 and 14-43 respectively) the containment DBA analysis was performed. This DBA analysis results in a peak reactor building pressure of 54.0 psig occurring at 20 seconds after the postulated rupture. The peak occurs eight seconds before the end of the blowdown of the RCS. The maximum vapor temperature reached was 284 °F.

The peak calculated pressure of 54.0 psig verifies that the passive mass and energy absorbing capabilities of the reactor building are adequate to maintain the pressure below the design value of 59 psig during the blowdown period of the LOCA.

After the blowdown peak pressure occurs, the ECCS and RBCS reduce the pressure and temperature within the reactor building atmosphere until sump water recirculation begins at about 4,257 seconds due to depletion of the BWST water supply. This time is dependent upon which emergency equipment is used. Thereafter, heat is removed from the reactor building by the Reactor Building Cooling System. During this sump water recirculation period the reactor building pressure reaches a secondary peak of 21.7 psig and then decreases continuously. Thus, the heat removal capability of the ECCS and RBCS adequately maintain the reactor building below the blowdown period maximum pressure and below the reactor building design pressure during the long-term post-blowdown transient.

The pressure transient results indicate that even with the conservative assumptions employed in the analyses, a margin of about 9.3 percent exists between the reactor building structure design pressure of 59 psig and the maximum calculated pressure of 54.0 psig (See Section 14.2.2.5.5.5 for a detailed discussion of the margin). This design margin has been used to evaluate the capability of the reactor building to withstand additional mass and/or energy releases. This evaluation shows that, at the time of peak pressure, about 19,230 lbm of additional blowdown containing 21.6×10^6 Btu may be added to the reactor building atmosphere to just reach the design conditions. The additional energy margin could also be associated with a 10 percent metal-water reaction up to the time of peak pressure.

The input assumptions to the reactor building DBA have changed since initial plant operation to include consideration of changes in operating conditions that are more limiting than originally assumed. Some specific changes now addressed include:

A. Reduction in the number of Technical Specification required operable reactor building coolers. In addition, the air cooler performance curve in Figure 14-110 was used as input to the analysis.

B. Reduction in the service water flow to the LPI coolers.

C. Consideration of the worst case service water temperature from Lake Dardanelle rather than the nominal value previously used.

D. Consideration of the increased initial reactor building bulk temperature.

E. Reduction in the reactor building net free volume.

F. Addition of heat loads due to the operation of the hydrogen recombiners.

G. Increase in the BWST temperature to worst case, rather than the nominal value used previously.

H. Reduction in the reactor building spray flow to ensure adequate NPSH during recirculation.

I. Increase in the BWST inventory, also to ensure adequate NPSH during recirculation.

Individually these changes have the following impact:

A. The reduction in the number of air coolers and the reduced cooler performance has no impact on the calculated peak pressures and temperatures because these peaks occur prior to the assumed start of the air coolers. These changes do, however, impact the long term rate of heat removal.

B. The reduction in the service water flow to the LPI coolers has no impact on peak reactor building conditions since the LPI coolers do not remove reactor building energy until recirculation starts, long after peak conditions are reached. As with the air coolers, this change impacts long term heat removal.

C. Consideration of worst case service water temperature from Lake Dardanelle reduces the effectiveness of the air coolers and LPI coolers. This further affects the long term heat removal rates.

D. The increased initial containment temperature has a minor impact on peak conditions. It has a correspondingly minor impact on the long term reactor building pressure and temperature.

E. The reduction in the reactor building net free volume results in an increase to the peak reactor building pressure and temperature.

F. Addition of hydrogen recombiner heat loads does not affect the peak building conditions, but has a minor impact on the long term reactor building pressure and temperature.

G. The consideration of a worst case BWST temperature does not impact the peak reactor building temperature. A minimal impact on the long term reactor building pressure and temperature profile results due to a higher BWST temperature.

H. A reduction in the spray flow does not affect the peak reactor building conditions due to the sprays being modeled after the peak conditions occur. A minimal long term effect on the reactor building temperature and pressure profile is apparent.

I. An increase in the BWST inventory affects the recirculation initiation time and long term reactor building conditions.

Collectively, these changes result in increased peak pressure and temperature and in an extended time to reduce pressure and temperature following an accident. These changes have been offset by the inclusion of more detailed input capabilities in the COPATTA code used in the DBA analysis. The peak pressure, predicted by the improved COPATTA code with the more detailed input assumptions, is 54.0 psig, which is greater than the original analysis, but well within the design pressure of 59 psig. The peak temperature of the revised DBA (284 °F) is also slightly higher than for the original analysis, but still within the design temperature of 286 °F.

These changes have had a more significant impact on the rate of reactor building cooldown and depressurization. The temperature and pressure profiles used to determine reactor building equipment environmental qualification requirements have been significantly altered by these changes. The DBA analysis, therefore, has had to be considered with respect to the equipment EQ requirements.

The results of a containment pressure analysis for cold leg pump suction breaks are provided in Table 14-55. This analysis was performed to show that the internals vent valves will preclude steam venting through the steam generators for a cold leg, pump suction break at a point close to a steam generator. Since the 7.0 ft^2 break results in the highest peak pressure, a pressure-time response curve for this particular break is provided in Figure 14-105. Mass release rates to the reactor building and the enthalpy of the mass released throughout the blowdown and reflood phases of the accident for the 7.0 ft^2 cold leg break are provided in Table 14-56. Core inlet velocity as a function of time for the reflood phase of the accident is shown in Figure 14-107. This analysis was based on the original design performance parameters of the reactor building heat removal systems. All calculations were made assuming that two air cooler units start operation 45 seconds after the rupture and one spray train begins injecting water through the spray nozzles at 91 seconds after the accident. Blowdown data was taken from the Oconee plant but is applicable to Arkansas Nuclear One - Unit 1. All other inputs were consistent with those used for the original DBA analysis at the time of initial plant licensing.

The 7.0 ft^2 cold leg pump suction break containment analysis has been reanalyzed assuming the air coolers start operation at 91 seconds and the spray starts injecting water at 121.5 seconds. Evaluation has shown that an air cooler start time of up to 112.7 seconds is acceptable. The revised analysis used the blowdown data given in Table 14-56 and was performed with the improved COPATTA code and revised inputs. The initial conditions and input assumptions for the revised analysis were identical to the original analysis with the following exceptions:

A. The number of coolers assumed to operate was reduced from 2 to 1.

B. The assumed service water temperature from Lake Dardanelle was increased from 85 °F to 95 °F which bounds the peak observed temperatures.

C. The air cooler performance curve shown in Figure 14-110 was used as input to COPATTA.

D. The initial bulk reactor building temperature was assumed to be 140 °F.

E. A reactor building net free volume of 1.81 x 10^6 ft^3 was assumed.

F. Reactor building spray flow was reduced from 1500 gpm to 1000 gpm.

G. The BWST temperature was taken as 110 °F, bounding the worst case temperature.

A peak pressure of 52.3 psig occurred, which is still below the DBA analysis.

An additional analysis of the core reflood model used following a primary coolant system blowdown was conducted utilizing mass and energy release rates determined by the use of a multiregion CRAFT code. The nodal arrangement of the CRAFT model is shown in Figure 14-106. The secondary side of the steam generators and the associated feedwater piping are represented by control volumes 6, 20, 17, and 21. Main feedwater was added directly into the secondary side of the steam generators over a period of 17 seconds as the feedwater control valve closed. At 15 seconds, the paths connecting the feedwater piping to the steam generators (which were assumed to be closed in the CRAFT model until this time) were opened so that the mass and energy trapped in the piping could enter the steam generator. Emergency Feedwater was modeled to start at 35 seconds. It should be noted that AP&L has subsequently justified a change to a 60 second initiation time. (Reference: AP&L Engineering Report 89E-0054-01 ANO-1 EFW Delay Justification.) Sensible heat stored in the steam generator tubes was modeled by slabs of metal in the secondary control volumes. Stored heat in the primary metal was simulated by slabs of metal in the control volumes which discharge their stored energy as the RCS temperature decreases.

To ensure a conservative calculation, the CRAFT code was run at 102 percent core power (2,619 MWt) and the mode of heat transfer in the core was assumed to be nucleate boiling until the quality of the coolant was approximately 1.0. Decay heat was calculated by using the ANS Standard times 1.2.

A further discussion of the hydraulic modeling of the reactor coolant and the method used to compute the reflood water boiloff is provided in B&W Topical Report BAW-10092, Revision 2, regarding the CRAFT code. The analyses of the LOCA have led to the conclusion that the reactor building design is adequate to withstand the postulated ruptures and release of the reactor coolant and associated energy sources without exceeding the design pressure. Furthermore, the reactor building design exceeds the consequences of the conservatively calculated DBA by an ample margin.

14.2.2.5.5.2 Analytical Methods and Conservatisms

A. Computer Program Description

The reactor building pressure transient analyses are performed using the COPATTA computer program. The COPATTA code is Bechtel's program to analyze the effects of a LOCA on a reactor building. COPATTA was derived from the original CONTEMPT (See Reference 25.) code written by the Phillips Petroleum Company for the LOFT program. The present COPATTA program is written in FORTRAN IV and uses the GE 635 computer.

1. COPATTA Model

The COPATTA model is concerned both with the pressure and temperature within the reactor building atmosphere and the temperatures in the reactor building structures. It is assumed that separate blowdown and core thermal behavior studies have been made, yielding mass and/or energy input rates from sources such as the release of reactor coolant, decay heat (causing heating or boiling of residual water in the reactor vessel), superheating of the steam as it passes through the core, and metal-water reactions. This information is provided by the Nuclear Steam Supply System (NSSS) manufacturer together with appropriate information on its use.

The basic reactor building model describes the reactor building and reactor vessel regions and the effects of heat sinks following a LOCA. Included in this basic model are features of engineered safeguards normally employed in power reactor plants and analytical techniques to enable calculation of leakage from the reactor building. Several options have been incorporated in the basic model to assist in using these features.

COPATTA calculates a pressure-time transient with stepwise iteration between the thermodynamic state points. The iterations are based on the laws of conservation of mass and energy together with their thermodynamic functions. Superposition of heat input functions is assumed so that any combination of blowdown, metal-water reaction, decay heat generation, and sensible heat can be used with appropriate engineered safeguards to determine the reactor building pressure-time history associated with a LOCA.

The program assumes a 3-region reactor building. The reactor building atmosphere is the vapor region; the sump is the liquid region; and, the liquid water contained in the reactor vessel is the third region. Energy is transferred between the liquid (sump) and vapor regions by boiling, but evaporation is neglected. A convective heat transfer coefficient can be specified between the sump and vapor regions. However, since any heat transfer in this mode is small, a conservative coefficient of zero is generally assumed. Each region is assumed homogeneous, but a temperature difference can exist between regions. Any moisture condensed in the vapor region during a time step is assumed to fall immediately into the liquid region. Noncondensable gases are included in the vapor region.

2. Thermodynamic Assumptions

During reactor system blowdown, the program calculates conditions in two separate regions of the reactor building: a water region in the sump, and an atmosphere region. Following completion of the RCS blowdown, the program also calculates conditions in a water region within the reactor vessel. The three regions are open systems in a thermodynamic sense, since the program permits mass flow across boundaries of all three regions. The appropriate expression of the first law of thermodynamics for such open systems is:

$$\frac{dU}{dt} = \sum_j \frac{dQ}{dt}_j + \sum_i h_i \frac{dm}{dt}_i$$

where:

U = the internal energy of the system (Btu)

Q = heat energy transferred to the system (Btu)

H = enthalpy of the mass entering the system (Btu/lbm)

m = mass entering the system (lbm)

t = time (h)

Integration of the above equation for each region, from the start of the transient to any later time, provides the thermodynamic conditions with which the state point properties of pressure and temperature can be determined. Numerical integration of the above equation and calculation of the properties within each region are based on the following assumptions:

a. At the rupture point, the discharge flow separates into a steam phase that is added to the atmosphere (vapor) region and a water phase that is added to the sump (liquid) region. The water phase is at the saturation temperature corresponding to the total reactor building pressure, while the steam phase is at the partial pressure of the steam in the reactor building.

b. The reactor building atmosphere pressure is also the sump pressure and, following blowdown, the RCS (reactor vessel region) pressure.

c. The steam-air mixture and the water phase are each assumed homogeneously mixed with uniform properties. Specifically, thermal equilibrium between the air and steam is assumed. However, a temperature difference may exist between the vapor region and the liquid region.

d. All the steam condensed from the atmosphere during any calculational time interval is added to the sump immediately at the end of the interval.

e. Mass and energy will be transferred from the liquid regions (sump and reactor vessel) to the reactor building atmosphere by boiling, if the calculation indicates that the reactor building pressure is less than the saturation pressure corresponding to the liquid temperature.

f. The sump region contains no water at the beginning of the transient.

g. Condensation of steam due to a vapor pressure gradient between the steam in the reactor building atmosphere and the water in the sump is neglected.

h. Condensation of steam on structural heat sinks occurs at the saturation temperature corresponding to the total pressure in the reactor building. Thus, during atmospheric superheat conditions, the condensing boundary layer is at saturation conditions.

3. Atmosphere and Sump Regions

Initially, the reactor building system is entirely composed of water vapor and air occupying the free volume of the reactor building. The water vapor and air partial pressures, masses, and internal energies are determined from the initial temperature, total pressure and relative humidity. During the first time advancement, a step input of mass and energy can be entered into the reactor building atmosphere.

The transient pressure and temperature calculations are made by considering the mass, volume, and energy equations for the water, steam, and air in the atmosphere and sump regions. These equations are:

$$M = m_w + m_s + m_a$$

$$V = m_w v_w + m_s v_s$$

$$U = m_w u_w + m_s u_s + m_a u_a$$

where:

M = total mass of water (w), steam (s), and air (a) (lbm)

m = individual constituent mass (lbm)

V = total free volume of system (ft³)

V = individual constituent specific volume (ft³/lbm)

U = total internal energy of water, steam and air (Btu)

u = individual constituent specific internal energy (Btu/lbm)

The above equations are solved iteratively for the vapor temperature each time advancement until a specified convergency criterion is met. The respective water (w), steam (s), and air (a) properties used in the equations are evaluated based upon the steam table values for water or steam at their respective temperatures, T_w or T_s, and the air temperature value, T_a. The steam temperature, T_s, and the air temperature, T_a, are equal, by assumption, to the vapor region temperature,

further designated by T_v. The specific volume of air, v_a, is calculated from the Ideal Gas Law and the air specific internal energy, u_a, is calculated from the equation:

$$u_a = C_v (T_v - T_o)$$

where:

C_v = .171 Btu/lbm-°F for the specific heat of air, and T_o is the initial reactor building atmosphere temperature.

Once the mass, volume, and energy equations are solved for the converged values of T_v and T_w, the reactor building pressure is calculated. The total reactor building pressure is computed from the sum of the partial pressures of steam and air at the reactor building atmosphere temperature, T_v. The steam pressure is taken from the steam table values and the air pressure is computed from the Ideal Gas Law relationship.

4. Reactor Vessel Region

A separate reactor vessel region is used in the COPATTA program to include the effects of decay heat, metal-water reaction, heat transfer from the RCS metal, and emergency core cooling. This region is not used during the blowdown phase when it is not in equilibrium with the reactor building atmosphere. During the blowdown phase, reactor coolant stored energy, decay heat, and other energy sources are normally added to the vapor in the reactor building atmosphere as blowdown energy releases furnished by the NSSS manufacturer.

At the end of the blowdown or the reactor vessel refilling phase, calculations are initialized for the reactor vessel region, and it is assumed that the reactor vessel region is in pressure equilibrium with the reactor building atmosphere. The reactor vessel region contains no air at the end of the blowdown and its water vapor partial pressure equals the total pressure of the reactor building atmosphere. Since the blowdown has just been completed, the liquid in the vessel is at thermal equilibrium with the vapor. Therefore, at the specified initialization time, the program sets the internal energy of the vessel water at the saturated liquid value corresponding to the total pressure of the reactor building. The volume available in the reactor vessel at initialization is specified as the vessel volume capable of retaining water. The program accepts water inventories less than this volume but sets the mass equal to the value corresponding to this volume and releases any excess if the volume is filled.

After initialization of the reactor vessel calculations, the program maintains mass and energy balances in the reactor vessel region in conjunction with the other reactor building regions. Since the vessel region is assumed to be in pressure equilibrium with the reactor building atmosphere at all times, its own vapor region is conservatively neglected in the mass and energy balances.

The internal energy of the vessel water is calculated and compared to the saturation internal energy corresponding to the total reactor building atmosphere pressure. If the vessel water internal energy is less than the saturated value, no

steam is transferred from the reactor vessel to the reactor building atmosphere. If it is greater than the saturated value, water is boiled off to reactor building until the vessel water internal energy is reduced to saturation conditions.

5. Heat Transfer Considerations

During a LOCA, heat transfer takes place between the reactor building atmosphere, the sump or reactor vessel water, and the exposed surfaces inside the reactor building. These surfaces may be building structures, reactor coolant or secondary system components, and equipment or other possible heat sinks or sources. The rate of heat transfer between the reactor building regions and these conducting masses is determined by the surface area, the surface temperature, the heat transfer coefficient, the physical arrangement of the conducting masses, and the thermal properties of these masses. All of the above parameters are considered by the COPATTA computer program during the transient analyses as described below.

6. Heat Conduction Calculations

The COPATTA program makes provision for the simulation of up to 20 heat conducting masses in the analytical model. These heat conducting masses are described by the following one-dimensional, multiregion, heat conduction equation:

$$\rho C \frac{dT}{dt} = \nabla \cdot (K \nabla T) + S$$

where:

T = temperature (°F)

T = time (hr)

K = thermal conductivity (Btu/hr-ft-°F)

ρC = volumetric heat capacity (Btu/ft^3-°F)

S = volumetric heat generation rate (Btu/hr-ft^3)

The spatial gradient operator, ∇, is applied in one of three coordinate systems in order to perform heat transfer calculations for rectangular, cylindrical, or spherical geometries. The input for the heat conduction calculations include provisions for specifying the geometry type, the surface area, the boundary conditions, and the number of and coordinates for different material regions in each heat conducting mass. The different regions in each mass are further described by specifying the calculation mesh point spacing, the material type, the magnitude of heat generation, and the thermal properties for each region.

Normally, each heat conducting mass specified will include all of the solid components comprising a structure in the reactor building. For example, the reactor building walls are specified by three materials, consisting of several mils of protective coating, a steel liner plate and several feet of concrete. Similar specifications are made for the other structures within the reactor building.

Boundary conditions are applied to the heat conducting mass external surfaces, as appropriate. These boundary conditions may indicate exposure to a constant temperature, a time dependent temperature, the reactor building atmosphere temperature, the sump temperature, the reactor vessel water temperature, or some combination of the above. Heat transfer coefficient control is similar, ranging from values of zero, values dependent on the steam/air ratio in the reactor building atmosphere, or the condensing steam value dependent upon a turbulence parameter inside the reactor building.

7. Heat Transfer Coefficient

Heat transfer in the conducting masses within the reactor building is dependent upon the assumed film coefficients between the reactor building region and the conducting mass surface. During the blowdown period of a LOCA, heat transfer to the heat sinks from the containment atmosphere region is determined by a "Modified Tagami" heat transfer coefficient function (See References 3 and 41). This function relates the condensing heat transfer coefficient to a "mixture turbulence factor" depending on the reactor building free volume, the integrated energy release to the reactor building up to the time of peak pressure, and the time from the accident initiation to the peak pressure. The "Modified Tagami" function correlates the experimentally measured results of Tagami, et al., (see References 29 and 30) and those of Kolflat and Chittenden (see Reference 14) on condensing heat transfer during the blowdown period.

Descriptive reports providing the method used to correlate the "Modified Tagami" heat transfer coefficient function with the Tagami and Kolflat data have been prepared and submitted to the AEC in References 3 and 42. These Bechtel topical reports describe the "Modified Tagami" function, the correlation method, the effects of surface materials on the heat transfer coefficient, and the effects of spatial variations in turbulence on the coefficient. Additional information has been reported in Reference 43, concerning surface conditions, reactor building size, reactor building height, and scale-up from the Tagami results to full-sized reactor buildings.

During the post-blowdown period of the transient, a quasi-steady state condition develops due to decreasing turbulence in the reactor building. Heat transfer under these conditions is dependent upon the steam-air mixture. Experimental work by Uchida, et al., (see Reference 32) has shown that during free convection cooling periods the condensing heat transfer coefficient is dependent on the ratio of noncondensable gas to steam masses. Application of the Uchida data during the long-term cooling period, and adoption of transition heat transfer coefficient between the "Modified Tagami" value and the Uchida value (based on the reduction of turbulence in the reactor building) completes the specification of the condensing heat transfer coefficient during LOCA transients.

The heat transfer coefficient between the water region of the sump and heat sinks is assumed to be 0.4 Btu/hr-ft^2-°F, a conservative value for liquid (see Reference 25). A natural convection coefficient of 2 Btu/hr-ft^2-°F is assumed for the outer surfaces of the reactor building structure which are exposed to the outside atmosphere. Zero heat transfer is specified at external surfaces exposed

to the earth and at the liquid-vapor interface between the reactor building sump and atmosphere regions. These conservative values assure maximum pressure and temperature conditions within the reactor building.

8. Heat Exchanger Calculations

Provision is made in the COPATTA code for shutdown heat removal with heat exchangers during the long-term cooling period of the LOCA. Four types of heat exchangers may be specified with seven optional combinations of reactor building spray and/or emergency core cooling heat exchangers (using either a single, two parallel, or two series units).

The available heat exchanger types are: shell and tube with single pass shell, shell and tube with crossflow shell, counterflow, or parallel flow. The physical and thermal characteristics of each heat exchanger are specified, as are the coolant inlet temperatures and flow rates.

The heat exchanger performance calculations are made using the method of heat transfer effectiveness as developed by Kays and London (see Reference 13). This method employs the number of exchanger transfer units (NTU) to evaluate an effectiveness for heat transfer. The effectiveness is defined as the ratio of the actual heat transfer to the maximum theoretical heat transfer. An effectiveness is calculated for any of the four heat exchanger types specified, and, from the effectiveness, the heat exchanger duty and system temperatures are determined.

B. Emergency Core Cooling Systems

The effect of the ECCS on the reactor building pressure analyses is simulated in the COPATTA code by a provision for pumped coolant injection into the reactor vessel region during the post-blowdown period. The accumulation of water in the reactor vessel region, the addition of heat to the region, and the effects of the pumped core cooling are all accounted for in the reactor vessel region calculations.

Core cooling water may be supplied from an external source or from recirculation from the reactor building sump. Variable injection flow rates and temperatures may be specified. A fractional part of the core cooling flow may be added directly to the reactor building sump, thereby permitting any combination of system ruptures to be simulated. Core cooling water which is supplied by sump recirculation may be passed through the heat exchangers previously described.

The effects of HPI or Core Flooding Systems during blowdown are normally included in the NSSS blowdown data as mass and energy entering the reactor building from the RCS.

C. Reactor Building Cooling Systems

The COPATTA program is capable of simulating the effects of reactor building engineered safeguards. These include the addition of either externally supplied or internally recirculated reactor building spray water and the cooling of the atmosphere region by fan cooling units.

The Reactor Building Spray System flow rate, inlet temperature, starting time, termination time, and efficiency may be specified. The water supply may be external or internal as a function of time. The spray system efficiency for heat removal as defined by (see Reference 25):

$$e = \frac{T_e - T_n}{T_v - T_n}$$

where:

e = spray system efficiency

T_e = spray water temperature entering liquid region (°F)

T_n = spray water temperature at spray nozzle exit (°F)

T_v = reactor building vapor region temperature (°F)

was taken from Reference 29 to calculate the heat removal or addition rate to the atmosphere region during spray operation. This spray efficiency is specified as a function of the steam-air mass ratio (specific humidity) in the reactor building atmosphere during a LOCA as shown in Figure 14-104. The heat removed from the atmosphere region due to heating of the spray water is added to the sump water. In a similar manner, any condensate due to this atmosphere cooling is also added to the sump.

The reactor building fan coolers may be described by the number of units operating, the unit starting and stopping times, and the heat removal rate. This heat removal rate may be dependent upon the reactor building atmosphere temperature. The fan coolers result in direct removal of energy and a corresponding condensation of water vapor from the reactor building atmosphere. The condensate is added to the reactor building sump.

Containment Pressure Differentials

Generally, the occurrence of a LOCA is postulated to result from a rupture of the RCS piping either within the reactor cavity or the steam generator compartments of the reactor building. (These subcompartments have thus been analyzed for such occurrences.) A pipe rupture of this type results in the expulsion of high enthalpy water out of the ruptured pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-air-water mixture flows through openings into the main reactor building. The maximum differential pressure achieved between the compartments can determine the design strength of walls and supports between the compartments. The maximum pressure differential depends on the number and shape of the openings leading between the compartments, the volume of each compartment, and the blowdown rate from the broken pipe.

Differential pressure analyses are made with the Bechtel computer program COPRA which calculates the transient pressure responses of two reactor building compartments during a LOCA. COPRA calculates a mass and energy balance of the 2-phase, 2-component, steam-water-air mixture as reactor coolant enters the compartment during the LOCA and exits through vents and openings into the main reactor building. There is no provision in the code for heat transfer to structures or for

engineered safeguards, since these options generally have a negligible effect on compartment pressures for the short time following the rupture within which peak differential pressures occur.

A COPRA Summary Document, (See Reference 41.) which lists and discusses the program assumptions, is contained in "Containment Pressure Analysis," NS731 TN, dated December 1968.

The pressure transient results for the reactor cavity subcompartment have been calculated for an instantaneous 28-inch diameter double-ended cold leg pipe rupture (8.55 ft^2 break area) at the reactor vessel nozzle to pipe weld. The maximum pressure differential developed between the reactor and cavity wall for this pipe rupture is 230 psid relative to the main containment volume and 235 psig (250 psia) relative to ambient outside pressure. The peak cavity pressure is reached 0.42 seconds after the pipe rupture. The cavity volume pressurized is about 5,000 ft^3 and the cavity vent area to the main containment volume is 176 ft^2.

Per approved B&W Topical Report BAW-1847 "Leak-Before-Break Evaluation of Margins Against Full Break for the RCS Primary Piping of B&W Designed NSS", LOCAs in main loop primary reactor coolant piping no longer need to be used as input into subcompartment pressurization analysis. (See 1CNA028604). The most limiting break in the reactor cavity would then be a core flood line break. Analysis of a core flood line break in the cavity indicates a maximum differential pressure of 91.3 psid.

The pressure transient for the two steam generator compartments is established based upon an instantaneous 36-inch diameter double-ended hot leg pipe rupture (14.14 sq. ft. break area) on the pressurizer side of the containment in the vertical section of the hot leg riser.

Mass release rates and enthalpy of mass released as functions of time for an 8.55 ft^2 double-ended cold leg break in the reactor cavity and 14.1 ft^2 double-ended hot leg break in the steam generator compartments are provided in Tables 14-57 and 14-58. A reactor blowdown rate discharge coefficient of 1.0 was assumed in the analysis.

The maximum pressure buildup is confined to a steam generator compartment volume of 33,000 ft^3 (pressurizer side) which vents both vertically upward to the main containment and downward through wall vents into the opposite steam generator compartment and the annular space between the compartments and containment wall which communicate vertically to the main containment volume. The total effective compartment vent area is 1,020 ft^2 (actual vent area of 1,990 ft^2 x a flow coefficient of 0.51). The maximum compartment pressure buildup is 16 psid, relative to the main compartment volume and 17 psig (32 psia) relative to initial ambient pressure. Peak pressure is reached at 0.08 second after pipe rupture occurrence.

The most limiting break in the steam generator cavity given the leak-before-break approval described above is the main steam line break. A comparison of Table 14-58 to Figure 14-21G reveals that the peak energy addition rate (BTU/sec) of the main steam line break is almost an order of magnitude smaller than that for the hot leg break. From this it can be concluded that the peak subcompartment pressure for the main steam line break will be significantly smaller than for the hot leg break. The main steam line break clearly bounds the feedwater line break because:

1. The main steam energy content per mass is higher because of the energy it has acquired in the steam generator,

2. The mass blowdown for the two breaks will be similar because the feedwater will flash before exiting the break and during the time frame of interest, both breaks will experience critical steam flow exiting the break area. Given the same size pipe, the feedwater line break might experience slightly higher mass flow rates due to carryover of moisture which flashes after exiting the break,

3. The feedwater pipe is half the diameter of the main steam pipe. This means that the break flow area of the feedwater line break is ¼ that of the main steam line break.

All other piping in the steam generator cavity not excluded by leak-before-break is significantly smaller than either the main steam line or the main feedwater line.

The structural design capability of these compartments to withstand the energy releases that might result from DBAs is discussed in Section 5.2.1.

14.2.2.5.5.3 <u>Initial Conditions and Input Data</u>

The reactor building pressure analysis input data is developed based upon the design of the plant. A thorough compilation of geometric, thermodynamic, design, and initial operating conditions prior to the hypothetical occurrence of a LOCA has been prepared. These physical and performance conditions are determined based upon conservative estimates of the most adverse design parameters with respect to maximizing reactor building pressure and temperature. A summary of the data developed for the analysis is given below.

The reactor building design parameters which determine the net free internal volume, the reactor building surface areas, and the design pressure and temperature are given in Table 14-39. The assumed reactor building operating temperature is 140 °F. The net free internal volume of 1.81×10^6 ft^3 was computed from a gross volume and occupied volume calculation. As an additional conservatism, the volume occupied by the reactor coolant prior to the LOCA is included as occupied volume inside the reactor building, rather than free volume.

The initial conditions within the RCS and the reactor building system prior to the accident initiation are given in Table 14-40. The RCS conditions correspond to a power level of 2,568 MWt with maximum coolant inventory. The reactor building system is assumed to be at atmospheric pressure and maximum inside and outside design operating temperatures.

The reactor building heat sink data specified for the heat transfer calculations during the LOCA are given in Table 14-41 and 14-42. Table 14-41 lists the geometric configuration of the heat sinks, including materials, thicknesses, and surface areas for concrete, steel, and steel lined structures. The carbon steel surfaces and the reactor building floor slab are assumed to be covered with a 10-mil thickness of appropriate protective coatings. The total surface area available for heat absorption during the accident is 236,900 ft^2, with 223,500 ft^2 exposed to the reactor building atmosphere and 13,400 ft^2 to the liquid on the reactor building floor. The conservatively determined thermodynamic properties and heat transfer coefficients used in the analyses are given in Table 14-42. This data plus the geometric data completely specify the necessary heat sink conditions for the calculation of heat transfer to and within the structures of the reactor building during the LOCA.

The performance of the engineered safeguards under design conditions for both 100 percent operation and for the reactor building DBA operation is given in Table 14-43. The 100 percent operation would correspond to full capacity while the accident operation corresponds to a degraded condition with only one HPI pump, one LPI pump, one spray pump, and one fan cooling unit. Only the ECCS Safeguards 1 and 2 in Table 14-43 were assumed available during the blowdown period of the accident. The other safeguards are the long-term RBCS. These systems were conservatively assumed to start at 300 seconds to verify the capability of the reactor building to withstand a LOCA with no active RBCS in operation. After 300 seconds the RBCS were operated at degraded levels to verify their ability to maintain lower pressures and temperatures in the reactor building.

The DBA analysis was performed with the improved COPATTA code and limiting inputs, as discussed in Section 14.2.2.5.5.1. The initial conditions and input assumptions for the DBA analysis are identical to those assumed in the original analysis with the following exceptions:

A. The number of coolers assumed to operate was reduced from 2 to 1.

B. The assumed service water flow to the LPI coolers was reduced from 3000 gpm to 1600 gpm.

C. The assumed service water temperature from Lake Dardanelle was increased from 85 °F to 95 °F which bounds the peak observed temperatures.

D. The air cooler performance curve shown in Figure 14-110 was used as input to COPATTA. The COPATTA code calculates the LPI cooler performance for the revised flow and service water temperature.

E. The initial bulk reactor building temperature was assumed to be 140 °F.

F. As an extra conservatism, the LPI cooler primary side flow during recirculation was reduced from a nominal 3500 gpm to 3000 gpm.

G. The reactor building net free volume was reduced from 1.865 x 10^6 ft^3 to 1.81 x 10^6 ft^3.

H. A heat load of 4.5 x 10^5 Btu/hr was added to the containment atmosphere starting at 30 minutes to account for the hydrogen recombiner operation.

I. The BWST temperature was increased from 85 °F to 110 °F, bounding the worst case temperatures.

J. Reactor building spray flow was reduced from 1500 gpm to 1000 gpm during both injection and recirculation.

K. BWST inventory of 312,210 gallons was assumed to be injected into the reactor building before a recirculation time of 4257 seconds.

14.2.2.5.5.4 <u>Accident Results</u>

A. Design Basis Accident Identification

The response of the reactor building for a spectrum of RCS hot leg pipe ruptures has been determined. Five break sizes were investigated: 14.1, 8.5, 5.0, 3.0, and 1.0 ft^2.

The blowdown and vessel refill mass and energy releases were calculated in a blowdown code which carries the transient out to the point where the core has come into equilibrium with the reactor building. The peak reactor building pressure occurs prior to the end of blowdown for all of the break sizes investigated. The blowdown analyses were conducted assuming a minimum of available ECCS are in operation. The reactor building pressure analyses were conducted assuming that the RBCS are not in operation for the blowdown period of the transient. This assumption assures a conservative prediction of the reactor building pressure response to RCS depressurization and blowdown.

The reactor building pressure-time responses for the five breaks considered are given in Figure 14-58. The pressure responses are shown for only 300 seconds since reactor building depressurization occurs for all five cases after about 90 seconds. The peak pressures, times of peak pressure, blowdown energy releases at the times of peak pressure, and heat transfer to the reactor building structures at the times of peak pressure are given in Table 14-44 for the five breaks considered. The maximum peak pressure is 53.1 psig, occurring for the 5 ft^2 break at 20 seconds. At this time, 305.6 x 10^6 Btu has entered the reactor building due to blowdown energy releases.

Based upon the results presented in Figure 14-58 and Table 14-44 the 5 ft^2 hot leg pipe rupture has been identified as the reactor building DBA. This identification results from a combination of maximum energy release with consideration of the effects of heat transfer to the reactor building structures.

The changes to the inputs used in the revised DBA analysis have not changed the mass and energy release values for the accident. Since these values completely characterize the RCS break size for the reactor building analysis, the 5 sq. ft. break is still the DBA.

B. Design Basis Accident Results

1. Blowdown Period

As previously described, the reactor building design basis LOCA has been identified as the 5 ft^2 break. This break gives a maximum pressure of 54.0 psig at 20 seconds after the pipe rupture and a maximum vapor temperature of 284 °F. The blowdown mass and energy release rates as a function of time for the 5 ft^2 break case are shown in Figure 14-59. The release data of Table 14-54 regarding post-blowdown mass and energy for 5.0 ft^2 DBA rupture are included in addition to the blowdown releases. Specifically, the integrated mass and energy quantities for these rates are 461,600 lbm and 307.9 x 10^6 Btu for the entire blowdown which ends at 28 seconds. At 20 seconds, the time of peak pressure, the integrated mass and energy releases are 459,727 lbm and 305.6 x 10^6 Btu.

At the time of peak pressure (20 seconds), a detailed mass and energy balance has been performed on the reactor and reactor building system. This data plus pressure, temperature, and heat transfer results are given in Tables 14-45 and 14-46 for the DBA. Table 14-45 lists the reactor building pressures, temperatures, and masses for the time of pipe rupture and the time of peak pressure with the 5 ft^2 break. Table 14-46 gives the energy distribution in the RCS and in the reactor building at the time of the break and at the time of peak pressure. This

table verifies the energy balance during the reactor building pressurization period (0 to 20 seconds), since the total NSS energy release equals the net gain in the reactor building system energy.

The condensing heat transfer coefficient is given in Figure 14-60 versus time for the DBA. The initial portion of the curve is the "Modified Tagami" (see References 29 and 30) value with a maximum of 272 Btu/hr-°F-ft^2 at 20 seconds. Following 25 seconds, the value decays (due to less reactor building turbulence) to the steady state Uchida (See Reference 32.) value of 140 Btu/hr-ft^2-°F at 35 seconds. Thereafter, the coefficient is the Uchida value which is dependent upon the slowly changing steam-air mass ratio in the reactor building.

2. Post-Blowdown Period

The long-term results of the DBA have been evaluated to verify the ability of the ECCS and RBCS to adequately cool the core and maintain the reactor building below the design conditions. This evaluation has been based upon conservatively assumed design performance of the engineered safeguards.

The reactor building pressure-time response for the DBA out beyond 10^6 seconds (278 hours or 11.6 days) is shown in Figure 14-61 for the safeguards performance mode outlined in Table 14-43. The maximum pressure of 54.0 psig occurs at 20 seconds, followed by reactor building depressurization until 4,257 seconds when sump water recirculation begins due to depletion of the BWST supply. During sump water recirculation, a second pressure peak of 21.7 psig occurs at 7,225 seconds due to steam evolution from the reactor caused by boiling of the hotter core injection water.

The reactor building cooling fans and Reactor Building Spray System were assumed to be actuated at 300 seconds (for the long-term cooldown period subsequent to RCS blowdown and refill) to reduce the reactor building pressure. The RBCS operation mode assumed one spray pump and one atmosphere cooling fan.

The reactor building atmosphere and sump temperature versus time are given in Figure 14-62 for the DBA. The peak atmosphere temperature of 284 °F was predicted at the time of peak pressure. However, the maximum sump water temperature of 281.8 °F was predicted at 1,300 seconds due to decay heat and sensible heat releases from the RCS.

The energy distribution in the reactor building versus time is shown in Figure 14-63. The energy components include: the vapor energy content (steam plus air), the liquid energy content, and the reactor building structures energy absorption. The total energy content of the above is also shown. The total energy increases to 791 x 10^6 BTU at about 7,225 seconds due to net energy addition to the system and thereafter decreases as the air coolers provide a net energy removal from the reactor building. For informational purposes, the accumulated energy removal from the reactor building by the reactor building fan cooling unit is also shown in Figure 14-63.

A chronology of events with time for the 5 ft^2 DBA case is given in Table 14-47 from the time of pipe rupture to 2.6 x 10^6 seconds when accident calculations were terminated. The time before the reactor building would reach atmospheric pressure is estimated as 3 x 10^6 seconds, or 35 days.

The temperature response of the reactor building structure is shown in Figure 14-64 at several points in time for the DBA. The reactor building wall liner plate reaches a maximum of 272 °F at 405 seconds, whereas the reactor building wall concrete remains below 150 °F for all locations in the concrete more than 12 inches away from the liner plate. An evaluation of these results shows that the resultant stresses for the concrete and steel reinforcement, and resultant strains for the liner plate, remain below the ASME allowable values (Reference: ANO Calculation 93-E-0001-01).

The pressure and temperature profiles of the DBA have been used as input in an evaluation of reactor building equipment environmental qualification requirements. Environmental qualifications of equipment have been demonstrated to meet the pressure and temperature conditions of the DBA.

14.2.2.5.5.5 Reactor Building Design Margins

This section provides a discussion of the margin available between the reactor building design pressure and the calculated peak pressure during a DBA. The DBA analysis assumes an initial reactor building pressure of 14.7 psia. Any pressure greater than this, as allowed by the plant Technical Specifications, would reduce the 5 psi margin reported below.

A. Design Capability

If the reactor building were to reach its design LOCA pressure of 59 psig, the corresponding pressures, masses, and energies would be those given in Table 14-48 assuming saturation conditions in the vapor region, no heat transfer to the reactor building structures, and the same average blowdown specific enthalpy as given in Section 14.2.2.5.5.4.B, i.e. 307.9 x 10^6 Btu/461,600 lbm = 667 Btu/lbm. With these conditions, the total energy capability of the reactor building is 328.0 x 10^6 Btu and the vapor region capability is 261.1 x 10^6 Btu.

Comparison of Table 14-48 with Table 14-45 shows that 5.0 psi or 9.3 percent margin exists between the maximum calculated and design pressure for the DBA. A 2 °F vapor temperature margin and a 25 °F liquid temperature margin is available at the time of peak pressure.

B. Blowdown Margins

At the time of peak pressure for the DBA, the adequacy of the reactor building design can be demonstrated by the mass and energy margins between the calculated peak pressure conditions of Tables 14-45 and 14-46 and the design capability values of Table 14-48. Thus, at the peak calculated reactor building pressure, the vapor region could absorb an additional 19,230 lbm and 21.6 x 10^6 Btu. These values are 4.2 percent of the mass and 7.0 percent of the energy input to the reactor building during the entire 28-second blowdown period. If this increase in vapor region capability were provided by additional blowdown at the average blowdown specific

enthalpy of 667 Btu/lbm, the total additional blowdown would be 40,654 lbm and 27.1 x 10^6 Btu, 8.8 percent of the reference DBA 28-second blowdown mass and energy integrals.

C. Post-Blowdown Margins

Following the time of peak pressure for the DBA, the adequacy of the reactor building design can be demonstrated by an energy margin defined as the difference between the energy capability of the reactor building (see Table 14-48) and the calculated energy content at any given time. Figure 14-65 shows this margin as a function of time for the vapor region only and for the sump plus vapor region. At the time of peak pressure the margins are 21.6 x 10^6 Btu for the vapor region and 27.1 x 10^6 for the entire reactor building. At 1,100 seconds the margins are 67.7 x 10^6 Btu and 84.9 x 10^6 Btu, respectively.

The vapor region energy margin can be related to the potential energy release of a hypothetical zirconium-water reaction. Using a reaction energy of 2,800 Btu/lbm zirconium (see Reference 19), reaction of 100 percent of the core zirconium would generate 140 x 10^6 Btu. If all the hydrogen liberated by this 100 percent metal-water reaction were burned and generated heat at a rate of 2,350 Btu/lbm zirconium, the total energy generated would be 258 x 10^6 Btu. Thus, the vapor region energy margin at the time of peak pressure could be associated with a 15 percent zirconium-water reaction or a 8 percent zirconium-water reaction with hydrogen combustion. At 1,100 seconds, the associated reactions would be 48 percent and 26 percent, respectively.

14.2.2.5.5.6 Conclusions

The pressure transient results indicate that, even with the conservative assumptions employed in the analyses, a margin of about 9.3 percent exists between the reactor building structure design pressure of 59 psig and the maximum calculated pressure of 54.0 psig. It may be concluded from the analyses of the LOCA that the reactor building design is adequate to withstand the postulated release of the reactor coolant and associated energy sources without exceeding the design pressure. Furthermore, the reactor building design has ample margin exceeding the energy releases considered.

Reactor building equipment environmental qualifications have been acceptably demonstrated for the long term conditions predicted by the DBA analysis.

14.2.2.5.6 Resultant Doses From a LOCA

The resultant doses from a LOCA are calculated by assuming that the activity associated with the gap of all fuel rods is released to the reactor building atmosphere. The timing of releases is modeled as specified in NRC Regulatory Guide 1.183. The activity in the coolant was evaluated to be less than one percent of the gap activity and therefore neglected.

The activity released to the reactor building from the gaps of all fuel rods is tabulated in Table 14-49.

ESF leakage equal to twice that described in Section 14.2.2.5.7 as outside the sealed rooms is assumed to occur in this analysis.

The maximum core power of Arkansas Nuclear One, Unit 1 is 2568 MWt. Fission product activities were calculated based on a power level of 2619.36 MWt to account for a 2% power uncertainty. A reactor building leak rate of 0.2 percent volume for the first day and 0.1 percent volume per day thereafter was assumed for containment leakage.

The TEDE offsite dose calculations were calculated using the RADTRAD code. Dose conversion factors were obtained from FGR 11 and FGR 12.

The sodium tetraborate in the reactor building spray provided by the Containment Sump Buffering Agent System reduces the airborne iodine. Table 14-51 lists the iodine removal constants used in this analysis. It is assumed that 0.15 percent of the iodine is organic, 95 percent is particulate, and 4.85 percent is elemental. These numbers are consistent with Regulatory Guide 1.183. Also, one spray header was assumed to be operating.

The following relative concentration factors, X/Qs, are based on windspeeds averaged over the 190-foot height of the reactor building as discussed in Section 2.3.6.2.

0-2 hr, exclusion distance	6.8E-4 sec/m^3
0-8 hr, LPZ	1.1E-4 sec/m^3
8-24 hr, LPZ	1.1E-5 sec/m^3
1-4 day, LPZ	4.0E-6 sec/m^3
4-30 day, LPZ	1.3E-6 sec/m^3

The resultant doses due to a LOCA are given in Table 14-49.

14.2.2.5.7 Effects of Engineered Safeguards Systems Leakage during the Loss of Coolant Accident

The Reactor Building Spray System pumps and LPI pumps are located in sealed rooms of the auxiliary building through which air does not circulate. Cooling is accomplished by a closed cycle ventilation system which blows room air over cooling water coils. Therefore iodine leaking from these pumps is not exhausted through the plant vent by the ventilation system. A flow path does exist from LPI and the Reactor Building Spray Pumps through the penetration rooms and into the Reactor Building. Leakage from portions of this flow path outside the sealed rooms has been evaluated to assess the dose impact. Offsite dose estimates from containment and ES leakage are included in the TEDE dose calculation results reported in Section 14.2.2.5.6.

Iodine leaking from the HPI pumps and portions of the HPI System flow path is not contained in sealed rooms. This leakage has been evaluated to assess the impact upon the doses even though recirculation through the HPI System in the piggyback mode is expected only for certain small break LOCAs. The additional dose from HPI System leakage, using source terms consistent with the minimal fuel damage expected during small break LOCAs, was determined to be less than 0.04 rem thyroid for both the 2 hour exclusion distance and 30 day low population zone dose. Therefore, no significant offsite doses result from these sources, and the radiation released is as low as practicable.

14.2.2.5.8 Control Room Doses

The dose to the control room operator from reactor building and ES leakage has been assessed. The Emergency Air Conditioning and Filtration Systems provided for the Control Room are described in Section 9.7.2.1. Iodine efficiencies of 95% for the recirculation filters

(99% for particulate) and 99% for the outside filtered air used for control room pressurization are assumed. Unfiltered inleakage is assumed to be 82 cfm. The 30 day integrated TEDE dose to the control room operator due to LOCA is 3.961 Rem.

14.2.2.6 Waste Gas Tank Rupture

In this accident, it is assumed that a waste gas tank ruptures releasing the waste gas it contains into the auxiliary building. The radioactive waste gas is then assumed to be carried out the plant vent by the Auxiliary Building Ventilation System. In this analysis, it is assumed the plant vent remains open. In addition, no decay of radioisotopes is assumed after the waste gas tank rupture occurs.

The maximum inert gas activity which could accumulate in a single waste gas tank is given in Table 11-10. In addition the tank would also contain traces of radioiodine. The quantity of radioiodine was calculated using the maximum coolant activities in Table 11-5, a partition factor of 10^{-4} and hydrogen removal from the coolant of 55 cc/l. The maximum amount of iodine that could be found in a waste gas tank is listed in Table 14-53.

The quantity of radioactivity contained in a single tank has been limited to a curie value which will prevent a member of the public at the exclusion area boundary from receiving a total body exposure exceeding 0.5 Rem in a 2-hour period. This is consistent with NUREG-0800, Branch Technical Position ETSB 11-5, July 1981.

14.3 ADDITIONAL ANALYSIS - EMERGENCY FEEDWATER SYSTEM SIZING

As part of the post-TMI review, the emergency feedwater system requirements were re-examined. As part of the emergency feedwater upgrade, the requirements for the minimum flow were re-examined.

A value of 500 gpm was established as a conservative minimum flow for the emergency feedwater system. This value took into account a single failure and allowed for pump recirculation flow, seal leakage and pump wear. This flow was then used to analyze the loss of main feedwater from 102 percent full power. Reactor coolant pump heat was included and a core decay heat which bounds ANS-5.1-1979 plus 2-sigma uncertainty was assumed. No credit was taken for the anticipatory reactor trip. The reactor was tripped on high reactor coolant pressure. The primary acceptance criteria were met for this analysis. These criteria included:

Reactor Coolant System Pressure	110 percent of design
Departure from Nucleate Boiling Ratio (DNBR)	Applicable correlation limit
Doses	10 CFR 100

Other Chapter 14 events, which require emergency feedwater include loss of electric power, main steamline failure and loss of coolant flow. The impact of a 500 gpm emergency feedwater flow on the results of these analyses was reviewed. It was determined that these results would not change with a 500 gpm emergency feedwater flow.

The emergency feedwater criteria assumed for a small break LOCA are described in Reference 78. This reference shows that an emergency feedwater flow of 500 gpm for Arkansas Nuclear One - Unit 1 will not lead to violation of any acceptance criteria.

14.3.1 EMERGENCY CORE COOLING SYSTEM ANALYSES

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14.3.1.1 Re-Evaluation of Emergency Core Cooling Performance

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14.3.1.2 ECCS Performance for Small Break LOCA in Discharge at Reactor Coolant Pump

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14.3.1.3 Post-TMI ECCS Analyses

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14.3.2 EMERGENCY FEEDWATER SYSTEM SIZING

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46. "Natural Circulation in B&W Operating Plants," transmitted on May 8, 1979.

47. "Operating Procedure Guidelines for Small Breaks," transmitted on May 5, 1979.

48. "Possible Zirconium-Water Reactions in Water Reactors," AEC Regulatory Staff Symposium, April 27, 1965.

49. "Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7208, Westinghouse (Proprietary), 1968.

50. "Report Covering the Effects of a High-Pressure Turbine Rotor Fracture and Low-Pressure Turbine Disc Fracture at Design Overspeed," Westinghouse (Proprietary).

51. "Report Covering the Effects of a Turbine Accelerating to Destructive Overspeed," Westinghouse (Proprietary).

52. Deleted

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53. Letter from Pettit, Arch P. to Harold R. Denton, 1-049-21, April 26, 1979, notifying of actions to be completed prior to startup of Unit 1 following refueling outage. Will also analyze small reactor coolant system breaks in order to write and implement procedures for accommodating a return to natural circulation cooling.

54. Letter from Cavanaugh, William to Harold R. Denton, 1-059-01, May 3, 1979, proposing certain actions to be completed prior to restart of Unit 1 in response to NRC Office of Nuclear Reactor Regulation's April 25, 1979 status report to the Commission.

55. Letter from Cavanaugh, William to Harold R. Denton, 1-059-20, May 11, 1979, superseding May 10, 1979 letter. Provides additional information regarding proposed actions in response to the Office of Nuclear Reactor Regulation's April 25, 1979 status report to the Commission.

56. Letter from Trimble, David C. to Robert W. Reid, 1-059-22, May 14, 1979, informing that the flow test of the ECCS long-term small break modification is complete.

57. Letter from Trimble, David C. to K. V. Seyfrit, 1-059-25, May 21, 1979, forwarding a report on the assessment of proposed trip signals which shows that a trip on low steam generator level would not be anticipatory. Includes Appendix A to report, proposed safety evaluation program for anticipatory trips, and Appendix B, proposed design for safety grade anticipatory trips on loss of main feedwater and/or on turbine trip.

58. Letter from Trimble, David C. to Robert W. Reid, 1-059-33, May 23, 1979, responding to NRC questions as a result of May 17, 1979 order with regard to savings realized from anticipatory trips based on past plant experience for feedwater and turbine trips resulting in reactor trips, and evolutions included in the emergency feedwater flow test that will be conducted immediately following startup.

59. Letter from Trimble, David C. to K. V. Seyfrit, 1-089-16, August 29, 1979, responding to IE Bulletin 79-05C requests with regard to short-term actions. "Analysis Summary in Support of an Early Reactor Coolant Pump Trip" is enclosed.

60. Letter from Trimble, David C. to K. V. Seyfrit, 1-099-12, September 14, 1979, forwarding a supplemental small break analysis and a revision to Section III of Attachment 1 to AP&L's August 29, 1979 submittal.

61. Letter from Trimble, David C. to Robert W. Reid, 1-109-25, October 31, 1979, forwarding information, requested in D. F. Ross's August 21, 1979 letter, regarding relief and safety valve flow model justification, OTSG 3-mode model justification, noncondensible gases in the primary system and CRAFT-2 simulation of TMI-2 accident.

62. Letter from Trimble, David C. to Robert W. Reid, 1-119-13, November 16, 1979, forwarding information, requested in D. F. Ross's August 21, 1979 letter, regarding a small break LOCA case which causes the reactor coolant system to repressurize to the PORV setpoint, and the effect of non-condensible gases on a small break LOCA analysis.

63. Letter from Trimble, David C. to Robert W. Reid, 1-020-05, February 11, 1980, forwarding information, requested in D. F. Ross's August 21, 1979 letter, regarding a benchmark analysis of sequential auxiliary (emergency) feedwater flow to the steam generators following loss of main feedwater using a 3-mode CRAFT-2 representation.

64. Letter from Trimble, David C. to Robert W. Reid, 1-050-14, May 23, 1980, responding to the NRC's November 21, 1979 request for additional information concerning small break loss-of-coolant accidents.

65. Letter from Trimble, David C. to Robert W. Reid, 1-070-02, July 1, 1980, forwarding information, requested in D. F. Ross's August 21, 1979 letter, regarding the lowered loop 177 fuel assembly B&W plant response to a 0.01 ft^2 cold leg break without feedwater and subsequent repressurization and sticking open of the pressurizer power operated relief valve.

66. Letter from Trimble, David C. to Darrell G. Eisenhut, GR-0481-20, April 30, 1981, submitting information, per commitment in AP&L February 5 & 27, 1981 letters, clarifying Item II.K.3.30 in NUREG-0737 with regard to small break LOCA methods.

67. Letter from Trimble, David C. to John F. Stolz, 1CAN078105, July 21, 1981, responding to NRC request for additional information relating to AP&L's submittal on asymmetric LOCA loads. References J.H.Taylor (B&W) July 9, 1981 letter to D.G. Eisenhut (NRC).

68. Letter from Griffin, John M. to Harold A. Denton, 1CAN118104, November 16, 1981, following up October 23, 1981 meeting on verification testing for small break LOCA methods.

B&W Correspondence to NRC

69. "Operating Procedure Guidelines for Small Breaks," transmitted on May 5, 1979.

70. "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," transmitted on May 7, 1979.

71. "Natural Circulation in B&W Operating Plants," transmitted on May 8, 1979.

72. "Steam Generator Tube Thermal Stress Evaluation," transmitted on May 10, 1979.

73. Letter from J. H. Taylor to T. M. Novak, providing background information on reactor coolant pump operation, May 10, 1979.

74. Letter from J. H. Taylor to R. J. Mattson, providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and on HPI Pump," identified as Supplements 1 and 2 to Section 6.0 of report in Item 1.2, May 12, 1979.

74a. "Acceptance for Referencing of Topical Report BAW-10192-P, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", February 18, 1997.

Babcock & Wilcox/Framatome-ANP Topical Reports References

75. Topical Report BAW-10164P-A, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis", Rev. 6.

76. BAW-10166P-A, Rev. 5, "A Computer Program for Reflood Heat Transfer During LOCA".

77. B&W Topical Report BAW-10095, "Babcock & Wilcox Revisions to CONTEMPT, Computer Program for Predicting Containment-Pressure Temperature Response to a Loss- of-Coolant Accident."

78. FANP Report 86-5064591-02, "ANO-1 Mark-B-HTP & Mark-B9 LOCA Summary Report (EOTSG)."

79. FANP Topical Report BAW-10192P-A, "BWNT Loss-of-Coolant Accident Evaluation Model for OTSG Plants," Rev. 0, and Supplement 1P-A, Rev. 0.

80. B&W Topical Report BAW-10227P-A, Rev. 0, "Evaluation of Abnormal Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.

81. B&W Topical Report BAW-10141, "TACO2, Fuel Pin Performance Analysis."

82. Topical Report BAW-10171, "REFLOD3B, Model for Multinode Core Reflooding Analysis."

83. Old Ref. 75.

84. Framatome Report FS1-0039424, "ANO-1 LBLOCA Summary Report."

85. B&W Topical Report BAW-10162P, "TACO3 Fuel Pin Thermal Analysis Computer Code."

86. Not Used.

87. Deleted.

88. "Arkansas Nuclear One - Unit 1, Fuel Densification Report" BAW - 1391, Babcock & Wilcox, Lynchburg, Virginia

88a. B&W Topical Report BAW-10187, "Statistical Core Design for B&W-Designed 177 FA Plants."

Westinghouse Electric Corporation Correspondence to AP&L

89. Letter from Davila, P.R., to Gaylon Hayes, LRC 26-49, July 15, 1987, giving AP&L new calculated P1 values for fully integrated Low Pressure turbine rotor at various inspection time intervals for both running speed and 120% overspeed. Letter also references NRC approved reports by Westinghouse on "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors."

Additional Correspondence

90. J.H. Taylor (B&W) to R.L. Baer (NRC), April 25, 1978.

91. J.H. Taylor (B&W) to R.L. Baer (NRC), May 1, 1978.

92. J.H. Taylor (B&W) to S.A. Varga (NRC), June 19, 1978.

93. J.H. Taylor (B&W) to S.A. Varga (NRC), July 18, 1978.

94. AREVA/FANP Document 51-5018851-01, “ANO-1 EOTSG Non-LOCA Safety Evaluation.”

95. AREVA/FANP Document 51-5018852-00, “ANO-1 EOTSG LOCA Safety Evaluation.”

Table 14-1

ABNORMALITIES AFFECTING CORE AND COOLANT BOUNDARY

Event	Analysis Assumptions	Effect
Uncompensated Operating Reactivity Changes	Automatic control system inoperative or unused	Change in reactor system average temperature. Automatic reactor trip if uncompensated. No equipment damage or radiological hazard.
Startup Accident	Uncontrolled single-group and all-group rod withdrawal from subcriticality with the reactor at zero power. Only high flux and high pressure trips were used to terminate the accident.	Power rise terminated by negative Doppler effect, control rod inhibit on short period, high Reactor Coolant System pressure, or overpower. No equipment damage or radiological hazard.
Rod Withdrawal Accident at Rated Power Operation	Uncontrolled single-group and all-group rod withdrawal with the reactor at rated power. Only high flux and high pressure trips were used to terminate the accident.	Power rise terminated by overpower trip or high pressure trip. No equipment damage or radiological hazard.
Moderator Dilution Accident	Uncontrolled addition of unborated water to the Reactor Coolant System due to failure of equipment designed to limit flow rate and total water addition.	Slow change of power terminated by reactor trip on high temperature or pressure. During shutdown a decrease in shutdown margin occurs, but criticality does not occur. No radiological hazard.
Cold Water Accident	Two Reactor Coolant Pumps started with reactor at 60% of rated power and end-of-life conditions.	Power and pressure transient produced by increase in flow does not result in a reactor trip. No equipment damage or radiological hazard.
Loss of Coolant Flow	Reactor Coolant System flow decreases because of mechanical or electrical failure in one or more Reactor Coolant Pumps.	None. Reactor is protected by the flux-imbalance-flow and power-pump trip. No radiological hazard.
Stuck-out, Stuck-in, or Dropped-in Control Rod	Maximum worth control rod dropped into core with the reactor at rated power, end-of-life condition.	None. Subcriticality can be achieved if any one rod is stuck out. If stuck in or dropped in, continued operation is permitted if effect on power peaking is not.severe. No radiological hazard.
Loss of Electric Power	A blackout condition or a complete loss of all station power is considered.	Possible power reduction or reactor trip, depending on condition. Redundancy provided for safe shutdown.

Table 14-2

UNCOMPENSATED REACTIVITY DISTURBANCES

Cause	Maximum Reactivity Rate, (Δk/k)/min	Rate of Average Temperature Change (uncorrected),°F/min
Fuel Depletion	-1.97×10^{-7}	-0.0007
Xenon Buildup	-1.91×10^{-5}	-0.052
Xenon Burnout	$+1.14 \times 10^{-3}$	+3.1

Note: The analysis is based on a Doppler coefficient of-1.17×10^{-5} (Δk/k)/°F and a moderator coefficient of 0.5×10^{-4} (Δk/k)°F.

Table 14-3

STARTUP ACCIDENT PARAMETERS

Maximum Rod Speed, in./min	30
Maximum Number of CRAs	60
Maximum Rod Worth, All Rods, % Δk/k	13.9
Maximum Reactivity Addition Rate, All 60 Rods at Max Speed, (Δk/k)/s	10×10^{-4}
Doppler Coefficient, (Δk/k)/°F	-1.3×10^{-5}
Moderator Coefficient, (Δk/k)/°F	$+0.9 \times 10^{-4}$
Peak Thermal Power Permitted (Design Overpower), % rated power	112
Trip Parameters	
High Pressure Trip Setpoint, psia	2400
Delay for High Pressure Trip, s	0.6
Delay for High Flux Trip, s	0.4
Control Rod Travel Time to 2/3 Insertion, s	1.4
Delayed Neutron Fraction (keff)	0.007
Number of PSVs	2
PSV Lift Tolerance (Accumulation)	+3% (75psi)
PSV Flow Rate, lbm/hr/valve	324,000
Initial Power, watts	2.568 (10^{-7} %-rated power)
Initial Pressurizer Level, inches	180
Number of RCPs in Operation	4
Core Flux Axial Peaking Factor	1.5

Table 14-4

SUMMARY OF STARTUP ACCIDENT ANALYSIS

1. Peak thermal power for withdrawal rates less than that corresponding to the withdrawal of all rods is always less than rated power.

2. Average fuel temperature in the average fuel rod never exceeds 1,000 °F.

3. The peak RCS pressure was assured to be less than 2750 psig for an actual pressurizer level of 180 inches with two pressurizer safety valves (PSVs) relieving at a 2590 psia setpoint and a flow rate of 324,000 lbm/hr/valve.

4. The peak RCS pressure was assured to be less than 2750 psig for an actual pressurizer level of 320 inches with two pressurizer safety valves (PSVs) relieving at a 2540 psia setpoint and a flow rate of 324,000 lbm/hr/valve.

Table 14-5

ROD WITHDRAWAL ACCIDENT PARAMETERS

High Pressure Trip Level, psia	2,430
High Flux Trip Level, %	112*
Trip Delay Time (High Pressure Trip),s	0.5
Trip Delay Time (High Flux Trip),s	0.3
CRA Insertion Time (2/3 Insertion),s	1.4
Doppler Coefficient, (Δk/k)°F	-1.17×10^{-5}
Moderator Coefficient, (Δk/k)°F	Zero
Control Rod Speed, in./min	30
Control Rod Group Worth, % Δk/k	1.71
One Rod Group Reactivity Addition Rate, (Δk/k)/s	1.23×10^{-4}
Maximum Reactivity Addition Rate of All 61 Control Rods, (Δk/k)/s	9.27×10^{-4}

* Note that the analysis results presented in Table 14-6 and Figures 14-9 through 14-15 were based on a design over power of 114%. Analyses based on a design over power of 112%, required for fuel densification considerations, demonstrate consistent results.

Table 14-6

SUMMARY OF ROD WITHDRAWAL ACCIDENT ANALYSIS

Reactivity Rate, $(\Delta k/k)/s \times 10^{-4}$	Peak Thermal[a] Power, % rated power	Peak System Pressure Increase, psi
1.23	108	88
9.27	103	15

[a] The criterion for this transient is to stay at or below the maximum design overpower of 112% of rated power.

Table 14-7

MODERATOR DILUTION ACCIDENT PARAMETERS

Flow Rates Considered:

Dilution Flow Rate Condition

Normal Dilution, gpm	70
Normal Dilution With Low Reactor Coolant System Pressure, gpm	100
Maximum Considered, gpm	500
Initial Boron Concentration in Reactor Coolant System, ppm (hot, clean, BOL)	2270
Boron Reactivity Worth, ppm/1% $\Delta k/k$	140
Moderator Coefficient, $(\Delta k/k)/°F$	0.5×10^{-4}
Doppler Coefficient, $(\Delta k/k)/°F$	-1.17×10^{-5}

Dilution Valve Interlock Setpoints

Dilution Valve May be Opened When:

1. Control rod group is 95% withdrawn and
2. Integrated flow timing device is set.

Dilution Valve Automatically Close When:

1. Control rod group is 75% inserted, or
2. Timing device has reached a preset value corresponding to a given integrated flow, or
3. Reactor trips.

Table 14-8

SUMMARY OF MODERATOR DILUTION ACCIDENT ANALYSIS

A. Dilution at Power:

Condition	Dilution Water Flow, gpm	Reactivity Rate, (Δk/k)/s	Average Reactor Coolant System Temp Change,°F/s
Normal	70	$+3.413 \times 10^{-6}$	0.0966
Low RCS Pressure	100	$+4.842 \times 10^{-6}$	0.1095
Maximum Considered	500	$+2.183 \times 10^{-5}$	0.2082

B. Dilution to Trip:

Dilution Water Flow, gpm	Peak Thermal Power, Percent of Rated	Peak Pressure, psia	Time to Trip, s
70	104.99	2533.0	72.30
100	105.5	2534.9	63.29
500	109.45	2555.0	34.27

Table 14-9

PUMP STARTUP ACCIDENT PARAMETERS

Moderator Coefficient, (Δk/k)/°F	-4.0×10^{-4}
Doppler Coefficient, (Δk/k)/°F	-1.3×10^{-5}

Table 14-10

SUMMARY OF PUMP STARTUP ACCIDENT ANALYSIS

Maximum Neutron Power (at 4 s)	92%
Maximum Thermal Power (at 6 s)	75%
Maximum Pressure Rise (at 11 s)	204 psi

Table 14-11

LOSS OF COOLANT FLOW ACCIDENT PARAMETERS FOR ORIGINAL FOUR PUMP COASTDOWN ANALYSIS

Initial Power, % rated power	102
Initial Flow, % design flow	95

System Characteristics:

The initial core inlet temperature for each power level is assumed to be plus 2 °F in error.

The initial system pressure is assumed to be minus 65 psi in error.

The pump monitor trip delay time is 620 ms.

The pump inertia is 73,500 lb-ft^2.

Table 14-12

LOCKED ROTOR ACCIDENT ORIGINAL ANALYSIS PARAMETERS

Initial Power, % rated power	102
Initial Flow, % design flow	100
Flux-Flow Trip Delay Time, ms	650

System Characteristics:

The initial pressure and inlet temperature are nominal values minus 65 psi and plus 2 °F, respectively.

Maximum design conditions were assumed for the thermal conditions.

The reactor is tripped by the flux-flow monitor.

Table 14-13

DELETED

Table 14-14

NATURAL CIRCULATION CAPABILITY

Time After Loss of Decay Heat Power, s	Decay Heat Core Power, %	Natural Circulation Core Flow Available, % full flow	Flow Required for Removal, % full flow
3.6×10^1	5	4.1	2.3
2.2×10^2	3	3.3	1.2
1.2×10^4	1	1.8	0.36
1.3×10^5	0.5	1.2	0.20

Table 14-15

DROPPED ROD ACCIDENT PARAMETERS

Moderator Coefficient, (Δk/k)°F	-4.0×10^{-4}
Doppler Coefficient, (Δk/k)°F	-1.3×10^{-5}
Control Rod Worth at Rated Power, % Δk/k	0.65
Control Rod Drop Time to Full Insertion, s (insertion rate = 0.325% Δk/k/s)	2.0

Table 14-16

LOSS-OF-LOAD ACCIDENT PARAMETERS AND RESULTS

Steam Relieved to the Atmosphere, lb	205,000
Steam Venting Time, min	3
Relative Concentration at Exclusion Distance, s/m^3	6.5×10^{-4}
Iodine Released During Relief (in Iodine-131 dose equivalent Curies)	3.8×10^{-2}
Total Integrated Thyroid Dose at Exclusion Distance, rem	1.2×10^{-2}

Table 14-18

SITUATIONS ANALYZED FOR STANDBY SAFEGUARDS ANALYSIS

Event	Analysis Assumptions	Effect
Steam Line	Reactor coolant leakage into the faulted steam generator continues for 251.8 hours following reactor operation at Technical Specification activity limits and 1 gpm total steam generator tube leakage.	Reactor trips following a large rupture. See Table 14-21 for resultant doses.
Steam Generator Tube Failure	Reactor coolant leakage into the faulted steam generator continues for 34 minutes following reactor operation at Technical Specification activity limits.	Reactor automatically trips if leakage exceeds normal makeup capacity to Reactor Coolant System. See Table 14-23 for resultant doses.
Fuel Handling Accident	Gap activity is released from six rows of fuel rods in one assembly while in spent fuel storage pool or fuel transfer canal. No retention of noble gases and only 99.5% retention of iodine is considered.	See Table 14-25 for resultant doses.
Rod Ejection Accident	All fuel rods that experience DNB are assumed to release their total gap activity to the reactor coolant.	Some fuel cladding failure. See Table14-31 for resultant doses.
Loss-of-Coolant Accident	The design of the ECCS is based on the double-ended rupture of the 36-in. diameter Reactor Coolant System pipe. The reactor building design is based on the 5.0 ft^2 rupture. Environmental effects are based on the release of all the gap activity.	Cladding temperature remains below 2,200 °F. See Table14-49 for resultant doses. See Table 4-43 for summary of reactor building pressure analysis.
Waste Gas Tank Rupture	A tank is assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel.	See Section14.2.2.7 for resultant doses.

Table 14-19a

STEAM LINE FAILURE PARAMETERS

Assumed Steam Generator Inventory (I) as a Function of Fractional Power Level (P) for Powers Greater than 15% of Rated Power, lbm	I = 41,200 P + 13,800
Maximum Pipe Break Size (ID) Assumed in Analysis, in	32.5
High Flux Trip Point (Assumed), %	112
Low Pressure Trip Point (Assumed), psia	1830.446
Doppler Coefficient (EOL), (Δk/k)/°F	-2.0×10^{-5}
Moderator Coefficient (EOL), (Δk/k)/°F	-3.5×10^{-4}
Trip Delay Time, s	0.3
Control Rod Movement Time to 2/3 Insertion During Trip, s	1.4
EFIC Low Steam Generator Setpoint, psig	580
EFIC Delay Time, s	1
MSIV Isolation Time, s	9
MFIV Isolation Time, s	29
ESAS Low Pressure Actuation, psig	1480
HPI Response Time, s	25

Table 14-19b

MASS AND ENERGY RELEASED FOR BUILDING PRESSURE ANALYSIS

	Mass lbm	Energy Btu x 10^{-6}
Blowdown from Unaffected Steam Generator	40,900	48.6
Blowdown from Affected Steam Generator	151,200	179.2
Total	192,100	227.8

Table 14-19c

SEQUENCE OF EVENTS

Event	Time After Rupture, (s)
Reactor Trip on Low Pressure(Setpoint Reached)	2.62
EFIC Actuation on Low Steam Generator Pressure (Setpoint Reached)	3.28
ESAS Actuation on Low Reactor Pressure (Setpoint Reached)	6.69
MSIV Fully Closed	13.28
Feedwater Heaters Off	30.28
HPI Initiates After 25 second Delay	31.69
MFIV Fully Closed	33.28
Minimum subcritical Margin	46
Maximum Return to Power	50

Table 14-19d

STEAM LINE BREAK REACTOR BUILDING PARAMETERS

Reactor Building Net Free Volume, ft^3	1.81×10^6
Initial Reactor Building Conditions Temperature, °F	140
Pressure, psia	14.7
Humidity, %	50
Reactor Building Coolers Number of Trains	1
Start Time, s	300
Reactor Building Sprays Number of Trains	1
High-high Containment Pressure Actuation Setpoint, psig	36.7
Spray Flow, gpm	1000
Response Time, s	105.868
BWST Temperature, °F	110

Table 14-20

SUMMARY OF STEAM LINE FAILURE ANALYSIS

Maximum Thermal Power During Transient, %	100
Maximum Return to Power After Trip, %	33
Minimum Subcritical Margin, % $\Delta k/k$	0.0084
Peak Reactor Building Pressure, psig (occurs at 74 s)	51.1
Peak Reactor Building Temperature, °F (occurs at 67 s)	386
Maximum Tube Stress, psi	7,350

Table 14-21

RESULTANT DOSES FROM A STEAM LINE FAILURE

Source Terms	See Table 14-50
Faulted Steam Generator Mass, lbm	60,000
Duration Primary-Secondary Leak Rate, gpm (per Steam Generator)	0.5
Relative Concentration at Exclusion Distance, s/m^3	6.8×10^{-4}
TEDE Doses, rem (GIS)	
Exclusion Distance	2.07
Low Population Zone	1.05
Control Room	3.72
TEDE Doses, rem (PIS)	
Exclusion Distance	0.45
Low Population Zone	0.19
Control Room	1.84

Table 14-22

STEAM GENERATOR TUBE FAILURE INPUT PARAMETERS

Initial Leak Rate, gpm (faulted steam generator)	435
Duration Leak Rate, gpd (intact steam generator)	150
Normal Makeup Rate, gpm	70
High-Pressure Injection Setpoint, psig	1,500

Table 14-23

SUMMARY OF STEAM GENERATOR TUBE FAILURE ANALYSIS

Low-Pressure Trip Occurs at, min	11
Time to Isolation of Faulted Steam Generators, min	34
Reactor Coolant Leakage through Faulted Steam Generator, ft^3	1,977
Time to Isolation of Intact Steam Generator, hr	237.8
Source Terms	See Table 14-50
TEDE Doses, rem (GIS)	
Exclusion Distance	1.26
Low Population Zone	0.23
Control Room	1.00
TEDE Doses, rem (PIS)	
Exclusion Distance	2.20
Low Population Zone	0.37
Control Room	2.33
Relative Concentration at Exclusion Distance, s/m^{-3}	6.8×10^{-4}

Table 14-24

FUEL HANDLING ACCIDENT PARAMETERS

Fuel Batch Average Burnup for Peak Assembly, Mwd/ton	61,050
Power Level During Operation, MW (including 2% uncertainty)	2619.36
Radial Peaking Factor	1.8
Decay Time, hrs	72
Relative Concentration at Exclusion Distance, sec/m^3	6.8×10^{-4}
Pool Decontamination Factors	
Organic Iodine	1
Inorganic Iodine	286
Noble Gases	1
Iodine GAP Composition	
Inorganic, %	99.85
Organic, %	0.15
Fraction of Assembly Activity in GAP	
Iodine-131, %	12
Krypton-85, %	30
Other Isotopes, %	10
Number of Damaged Pins	82

Table 14-25

FUEL HANDLING ACCIDENT DOSES

TEDE Doses for Fuel Handling Accident in Spent Fuel Pool or in Reactor Containment, rem (Unfiltered Release)

Exclusion Distance	1.40
Low Population zone	0.25
Control Room	1.00

Table 14-26

ROD EJECTION ACCIDENT PARAMETERS

Worth of Ejected Rod	
Rated Power, No Xenon, % Δk/k	0.40
Rated Power, With Xenon, % Δk/k	0.40
Hot, Zero Power, Critical, % Δk/k	0.23
Rated Power, Maximum Worth, % Δk/k	0.65
Rod Ejection Time, s	0.15
Rated Power Level, MWt	2,568
Reactor Trip Delay Time	
High Flux Trip, s	0.3
High-Pressure Trip, s	0.5
Control Rod Drive Trip Time to 2/3 Insertion, s	1.4

Table 14-27

NOMINAL VALUES OF INPUT PARAMETERS FOR ROD EJECTION ACCIDENT ANALYSIS

	BOL	EOL
Delayed Neutron Fraction, β_{eff}	0.0071	0.0053
Neutron Lifetime, μs	24.8	23.0
Moderator Coefficient, (Δk/k)/°F	Zero	-4.0×10^{-4}
Doppler Coefficient, (Δk/k)/°F	-1.17×10^{-5}	-1.30×10^{-5}
Reactor Coolant Inlet Temperature, °F	554	554
Initial System Pressure, psia	2,200	2,200
Total Nuclear Peaking Factor, $°F_q$	3.24	2.92
Average Fuel Temperature of Fuel Pellet, °F	1,540	1,670
Average Fuel Temperature of Hottest Pellet, °F	2,810	2,685

Table 14-28

COMPARISON OF SPACE-DEPENDENT AND POINT KINETICS RESULTS ON FUEL ENTHALPY

		Peak-to-Average Values		Fuel Enthalpy, cal/g	
	Ejected Rod Worth, %Δk/k	TWIGL	Point Kinetics	TWIGL	Point Kinetics
BOL Rated Power	0.38	3.04	3.24	125	150
	0.83	2.67	3.24	174	225
BOL Zero Power	0.56	4.1	3.24	38	60
	0.83	4.4	3.24	48	71

Table 14-29

SUMMARY OF ROD EJECTION ACCIDENT ANALYSIS

Initial Power Level, % rated power	Ejected Rod Worth, %Δk/k	Neutron	Peak Power, % rated Power, Thermal
0.1 (BOL)	0.23	10	9
0.1 (EOL)	0.23	4.1	3.9
100.0 (BOL)	0.40	227	115
100.0 (BOL)	0.65	680	138

Table 14-30

REACTOR VESSEL PARAMETERS

Vessel Temperature, °F	600
Yield Strength (0.2% offset), psi	55,000
Ultimate Strength, psi	80,000
Ultimate Strain, (μ_u), %	26
Strain Energy (E_s) per Unit Volume up to Strain Equal to 1/2 Ultimate Strain, in.-lb/in^3	8,000
Strain Energy (E_s) per Unit Volume up to Ultimate Strain, in.-lb./in^3	17,000
Equivalent Pressure Vessel Dimensions, in.	
OD	188.25
ID	166.69
Thickness	10.78

The expression[(a)] used for the weight of explosive required to strain the vessel a given amount is:

$$W = \left[\frac{1.407 E_S (3.41 + 0.117\, R_i / t)(R_e^2 R_i^2)^{1.85}}{10^5\, w^{-0.85} (1.47 + 0.0371\, R_i / t)^{0.15} (R_i)^{0.15}} \right]^{0.811}$$

where:

W = charge weight (TNT or Pentolite), lb,

w = weight density of vessel material, lb/ft^3,

R_i = initial internal radius of vessel, ft,

R_e = initial external radius of vessel, ft,

t = initial wall thickness of vessel wall, ft,

E_s = wall strain energy, in.-lb/in^3

(a) Section 14.4, Reference 36.

Table 14-31

RESULTANT DOSES FROM ANALYSIS OF THE ROD EJECTION ACCIDENT

Source Terms from fuel rods experiencing DNB

Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]
Kr-83m	8.77E x 10^6	I-130	1.36 x 10^6
Kr-85	9.61 x 10^5	I-131	7.22 x 10^7
Kr-85m	1.90 x 10^7	I-132	1.05 x 10^8
Kr-87	3.73 x 10^7	I-133	1.48 x 10^8
Kr-88	5.01 x 10^7	I-134	1.67 x 10^8
		I-135	1.41 x 10^8
Xe-131m	7.55 x 10^5		
Xe-133	1.48 x 10^8	Cs-134	1.46 x 10^7
Xe-133m	4.60 x 10^6	Cs-136	2.98 x 10^6
Xe-135	3.51 x 10^7	Cs-137	9.88 x 10^6
Xe-135m	3.09 x 10^7	Cs-138	1.38 x 10^8
Xe-138	1.27 x 10^8		
		Rb-86	1.29 x 10^5

Reactor Building Leak Rate
- 0.2%/day on the first day
- 0.1%/day thereafter

Relative Concentrations, sec/m^3	
0-2 hour, exclusion distance	6.8 x 10^{-4}
0-8 hour, low population zone	1.1 x 10^{-4}
8-24 hour, low population zone	1.1 x 10^{-5}
1-4 day, low population zone	4.0 x 10^{-6}
4-30 day, low population zone	1.3 x 10^{-6}

TEDE Doses, rem (containment release path)	
2-hour, Exclusion Distance	4.73
30-day, Low Population Zone	2.28
30-day, Control Room	3.40

TEDE Doses, rem (primary-secondary release path)	
2-hour, Exclusion Distance	3.03
30-day, Low Population Zone	1.64
30-day, Control Room	4.95

TABLES 14-32 THROUGH 14-38

DELETED IN AMENDMENT 12

Table 14-39

REACTOR BUILDING DESIGN PARAMETERS

Reactor Building Design Pressure, psig	59
Reactor Building Design Temperature, °F	286
Internal Dimensions	
Cylindrical Wall Diameter, ft	116.0
Cylindrical Wall Height, ft	171.6
Dome Height, ft	37.0
Gross Internal Volume, ft^3	2,087,000
Occupied Volume, ft^3	277,000
Net Free Internal Volume, ft^3	1,810,000

Table 14-40

INITIAL CONDITIONS FOR PRESSURE ANALYSES

Reactor Coolant System	
NSS Power Level, MWt	2,568
Coolant Pressure, psig	2,185
Average Coolant Temperature, °F	579
Coolant Mass Inventory, lbm	519,200
Coolant Energy Inventory, Btu	306.7×10^6
Internal Coolant Volume, ft^3	11,800 Reactor
Building System	
Pressure, psia	14.7
Relative Humidity, %	50
Inside Temperature, °F	140
Outside Temperature, °F	95
Service Water Temperature, °F	95
Borated Water Storage Tank	
(BWST) Water Temperature, °F	110
Core Flooding Tank Water Temperature, °F	110

Table 14-41

REACTOR BUILDING HEAT SINK GEOMETRIC DATA

Heat Sink		Thickness, in.	Surface Area, ft^2
1.	Reactor Building Walls	45.25	81,000
	Inorganic Protective Coating[(a)]	0.01	
	Steel Liner Plate	0.24	
	Concrete	45.00	
2.	Stainless Steel Lined Concrete Structures	24.24	9,500
	Steel Liner	0.24	
	Concrete	24.00	
3.	Uninsulated Steel Structures	0.24	55,000
	Inorganic Protective Coating[(a)]	0.01	
	Steel	0.23	
4.	Unlined Concrete Compartment Partitions	23.0	78,000
	Concrete	23.0	
5.	Reactor Building Floor and Sump	48.01	13,400
	Inorganic Protective Coating[(a)]	0.01	
	Concrete	48.00	
	Total		236,900

[(a)] This is original coating information. Reformulated replacement coatings may contain materials different than the previous coatings, may not require a primer, and may not be the same type (e.g., inorganic). Although replacement coatings can alter physical properties of the coating, analyses for post-accident containment heat sink considerations are conservative. Coatings and repairs conform to the requirements of the standards specified in Section 5.2.1.3.5.

Table 14-42

REACTOR BUILDING HEAT SINK THERMODYNAMIC DATA

	Material Property[a]			
	K, Btu/hr-ft°F	ρ, lbm/ft³	C_p, Btu/lbm-°F	ρC_p, Btu/ft³-°F
Materials				
Inorganic Protective Coatings[b]	1.0	170	0.12	20
Stainless Steel Liner Plate	10	490	0.11	54
Carbon Steel Liner Plate	25	490	0.11	54
Structural Concrete	0.8	143	0.21	30

Heat Transfer Coefficients	Value
Reactor Building Atmosphere to Heat Sink Surfaces	"Modified Tagami" (See Section 14.2.2.5.5b.1)
Reactor Building Atmosphere to Sump	0
Reactor Building Sump to Heat Sinks	0.4 Btu/hr-ft^2-°F
Outside Atmosphere to Heat Sinks	2.0 Btu/hr-ft^2-°F

[a] K = thermal conductivity, Btu/hr-ft-°F,

ρ = density, lbm/ft^3,

C_p = specific heat, Btu/lbm-°F,

ρC_p = volumetric heat capacity, Btu/ft^3-°F

[b] This is original coating information. Reformulated replacement coatings may contain materials different than the previous coatings, may not require a primer, and may not be the same type (e.g., inorganic). Although replacement coatings can alter physical properties of the coating, analyses for post-accident containment heat sink considerations are conservative. Coatings and repairs conform to the requirements of the standards specified in Section 5.2.1.3.5.

Table 14-43

ENGINEERED SAFEGUARDS PERFORMANCE FOR REACTOR BUILDING

System	Normally Available at 100% Operation	Design Basis Accident Operation
1. Passive Emergency Core Cooling[(b)]		
Water Source: CFT		
No. Tanks and Lines	2	2
Quantity Available (Water at 110 °F), ft^3	1,040	1,040
Operating Point (Reactor Pressure), psig	600	600
2. Active Emergency Core Cooling[(b)]		
Water Sources: BWST and Recirculation from Sump[(c)]		
No. Lines (HPI/LPI)	2/2	1/1
No. Pumps (HPI/LPI)	3/2	1/1
Flow Rate (HPI + LPI), gpm[(h)]	7,000	3,500
3. Reactor Building Spray		
Water Sources: BWST and Recirculation from Sump[(c)]		
No. Lines and Headers	2	1
No. Pumps	2	1
Flow Rate, gpm	3,000	1,000
4. Reactor Building Cooling Units		
No. Units	4	1
Flow Rate (Air Side), cfm	120,000	30,000
Total Heat Removal at 286 °F, Btu/hr	226×10^6	53.7×10^6
5. Decay Heat Removal Heat Exchanger		
Cooling Water Supply: Service Water System		
Type: Shell and U-tube; No. Units	2	1
Heat Transfer Area, ft^2	4,138	2,069
Heat Transfer Coefficient (Overall), Btu/hr-ft^2-°F	414	[(e)]
Flow Rate: Sump Water Side, gpm	6,000	3,000
Service Water Side, gpm	6,000	1600
Operating Point (Time), s[(c)]	1988	4257

Table 14-43 (continued)

[b] Emergency core cooling:

CFT - Core Flooding Tank

BWST - Borated Water Storage Tank (Quantity Available - 387,400 gal, Quantity Assumed Injected - 312,210 gal)

HPI - High Pressure Injection

LPI - Low Pressure Injection

[c] Based on the injection and spray actuation times given in Table 14-47, sump water recirculation begins at 1988 and 4257 s for the 100% and "reactor building design basis" operation modes, respectively.

[e] In the DBA analysis, the decay heat removal heat exchanger heat transfer coefficient was modified to account for the different flows and temperatures as a part of the analysis.

[h] The DBA analysis assumes no HPI flow during recirculation, i.e., HPI + LPI = 3000 gpm during recirculation.

Table 14-44

ORIGINAL REACTOR BUILDING PRESSURE ANALYSIS, RESULTS FOR A SPECTRUM OF BREAK SIZES[a]

	Break Size, ft^2				
	14.1	8.5	5.0[b]	3.0	1.0
Peak Pressure, psig	51.3	52.6	53.1	52.5	49.8
Time of Peak Pressure, s	10	13	20	28	86
Blowdown Energy Release at Time of Peak Pressure,10^6 Btu	301.0	304.6	305.6	299.7	286.7
Heat Transferred to Structures at Time of Peak Pressure, 10^6 Btu	11.1	12.2	14.6	16.1	30.0

[a] Assumptions: Reactor building design basis ECCS performance, no reactor building cooling systems in operation during blowdown period.

[b] This spectrum analysis established the 5.0 ft^2 break size as the reactor building DBA. See Tables 14-45 and 14-46 for the current reactor building DBA results.

Table 14-45

REACTOR BUILDING RESULTS FOR THE DESIGN BASIS ACCIDENT

	Prior to LOCA	At Peak Pressure
Time, s	0	20
Pressures		
Steam, psia	1.4	52.2
Air, psia	13.3	16.5
Total, psia	14.7	68.7
Total Gauge, psig	0	54.0
Temperatures		
Steam and Air, °F	140	283.9
Water in Sump, °F	---	260.6
Mass Release, lbm[a]	0	459,727
Reactor Building Masses[a]		
Air, lbm	108,130	108,130
Steam, lbm	7,366	220,573
Water in Sump, lbm	0	246,520
Total, lbm[a]	115,496	575,223
Heat Transfer Coefficient, Btu/hr-ft^2-°F[b]	2.0	272

[a] Reactor Building total mass change = total mass release.

[b] Between reactor building atmosphere and structures.

Table 14-46

ENERGY DISTRIBUTION IN REACTOR AND REACTOR BUILDING FOR THE DESIGN BASIS ACCIDENT

Energy (values in 10^6 Btu[a])

Reactor Coolant System	0 Seconds	Decrease	20 Seconds
Primary Coolant	306.7	276.4	30.3
Core Flood Tank Water	9.1	2.4	6.7
Reactor Core	50.3	13.4	36.9
Reactor Vessel	55.6	0.8	54.8
Reactor Vessel Internals	19.2	3.6	15.6
Secondary Coolant	72.0	0.0	72.0
Steam Generator Metal	57.3	0.0	57.3
Piping and Pumps	40.8	0.3	40.5
Total Stored Energy	611.0	296.9	314.1
Core Shutdown Heat Generated		8.7	
Blowdown Energy Release		305.6[b]	
Reactor Building	0 Seconds	Increase	20 Seconds
Air	2.0	0.6	2.6
Steam	5.8	236.0	241.8
Water in Sump	0.0	56.5	56.5
Structures	[c]	12.5	[c]
Total		305.6[b]	

[a] Energy reference temperature is 32 °F.

[b] Total stored energy decrease + core shutdown heat generated = blowdown energy release = total reactor building energy increase.

[c] Actual energy contained in the reactor building structures has not been calculated; only the increase in energy is available from the analysis.

Table 14-47

REACTOR BUILDING DESIGN BASIS ACCIDENT CHRONOLOGY OF EVENTS

Time, s	Event
0	Pipe ruptures (5 ft^2 hot leg), reactor begins depressurization
13.3	Core flood tanks begin injection
20	Reactor building reaches peak pressure
	Reactor vessel level falls below break and blowdown becomes super-heated
28	Blowdown ends as reactor vessel reaches pressure equilibrium with reactor building
35	HPI and LPI pumps begin injection
135	Liquid overflow begins as vessel level reaches break point
300	Reactor building spray (1) and fan cooler (1) begin operation
405	Liner plate reaches maximum temperature of 272 °F
1,300	Sump reaches maximum temperature of 282 °F
4,257	Spray and injection water begin recirculation from the sump as BWST reaches low level--injection water cooled by decay heat removal heat exchanger
7,225	Containment pressure reaches second peak of 21.7 psig
	Reactor building reaches maximum energy value, 791 x 10^6 Btu
10,800	Fan cooler duty + heat exchanger duty begins to exceed decay heat rate at 101 x 10^6 Btu/hr
3 x 10^6	Reactor building reaches atmospheric pressure (estimated)

Table 14-48

REACTOR BUILDING DESIGN CAPABILITY FOR A LOSS OF COOLANT ACCIDENT[a]

	Prior to LOCA	Change At	Design Pressure
Pressures			
Steam, psia	1.4	55.7	57.1
Air, psia	13.3	3.3	16.6
Total, psia	14.7	59.0	73.7
Total gauge, psig	0	59.0	59.0
Masses, lbm			
Air	108,130	0	108,130
Steam	7,366	232,437	239,803
Water in Sump	0	267,944	267,944
	115,496	500,381	615,877
Energies, 10^6 Btu[b]			
Steam	5.8	255.3	261.1
Air	2.0	2.9	4.9
Water in Sump	0	62.0	62.0
Total	7.8	320.2	328.0

[a] Assumptions: Saturation conditions in sump liquid, no heat transfer to reactor building structures, average blowdown specific energy = 667 Btu/lbm.

[b] Energy reference temperature is 32 °F.

Table 14-49

RESULTANT DOSES FROM LOCA

Core Power, MWt: 102% of 2,596

Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]	Isotope	Core Inventory [Curies]
Kr-85	9.61E+05	**Sb-127**	6.56E+06	**Ce-143**	1.12E+08
Kr-85m	1.90E+07	**Sb-129**	2.01E+07	**Ce-144**	1.05E+08
Kr-87	3.73E+07	**Te-127**	6.52E+06	**Np-239**	1.39E+09
Kr-88	5.01E+07	**Te-127m**	1.16E+06	**Pu-238**	1.93E+05
Xe-131m	7.55E+05	**Te-129**	1.88E+07	**Pu-239**	2.51E+04
Xe-133	1.48E+08	**Te-129m**	3.66E+06	**Pu-240**	3.88E+04
Xe-133m	4.60E+06	**Te-131m**	1.40E+07	**Pu-241**	9.82E+06
Xe-135	3.51E+07	**Te-132**	1.02E+08	**Am-241**	1.02E+04
Xe-135m	3.09E+07	**Sr-89**	7.25E+07	**Cm-242**	2.71E+06
Xe-138	1.27E+08	**Sr-90**	7.47E+06	**Cm-244**	1.99E+05
I-130	1.36E+06	**Sr-91**	8.78E+07	**La-140**	1.32E+08
I-131	7.22E+07	**Sr-92**	9.40E+07	**La-142**	1.15E+08
I-132	1.05E+08	**Ba-139**	1.32E+08	**Nb-95**	1.34E+08
I-133	1.48E+08	**Ba-140**	1.28E+08	**Nd-147**	4.70E+07
I-134	1.67E+08	**Mo-99**	1.35E+08	**Pr-143**	1.11E+08
I-135	1.41E+08	**Rh-105**	7.25E+07	**Y-90**	7.75E+06
Cs-134	1.46E+07	**Ru-103**	1.14E+08	**Y-91**	9.53E+07
Cs-136	2.98E+06	**Ru-105**	7.64E+07	**Y-92**	9.51E+07
Cs-137	9.88E+06	**Ru-106**	4.19E+07	**Y-93**	1.07E+08
Cs-138	1.38E+08	**Tc-99m**	1.18E+08	**Zr-95**	1.29E+08
Rb-86	1.29E+05	**Ce-141**	1.23E+08	**Zr-97**	1.23E+08

Table 14-49 (continued)

Reactor Building Leak Rate	
0.2%/day on the first day	
0.1%/day thereafter	
Relative Concentrations, sec/m^3	
0 - 2 hour, exclusion distance	6.8×10^{-4}
0 - 8 hour, low population zone	1.1×10^{-4}
8 - 24 hour, low population zone	1.1×10^{-5}
1 - 4 day, low population zone	4.0×10^{-6}
4 - 30 day, low population zone	1.3×10^{-6}
Iodine removal constant	(See Table 14-51)
*TEDE Doses, Rem	
2 hour, exclusion distance	11.091
30 day, low population zone	2.738
30 day, control room	3.961

* 50% of containment leakage is processed by the penetration ventilation system. Doses include the dose assessed from 782 cc/hr ESF leakage from components located outside sealed rooms for the EAB and LPZ, and 1178 cc/hr ESF leakage total from components located outside sealed rooms and the volumetric expansion of atmosphere inside sealed rooms for the CR.

Table 14-50

SOURCE TERMS FOR MSLB & SGTR

The RCS activity presented in the following table is based on an equilibrium 1 μCi/g dose equivalent I-131 and 72/E μCi/g total. The secondary activity is based on an equilibrium 0.1 μCi/g dose equivalent I-131.

Isotope	RCS Activity (Ci)	Secondary Activity (Ci)
Kr85	4.80E+02	0.00E+00
Kr85m	1.31E+03	0.00E+00
Kr87	2.09E+03	0.00E+00
Kr88	2.93E+03	0.00E+00
Xe131m	3.78E+02	0.00E+00
Xe133	3.02E+04	0.00E+00
Xe133m	7.64E+02	0.00E+00
Xe135	1.30E+04	0.00E+00
Xe135m	1.18E+03	0.00E+00
Xe-138	3.46E+03	0.00E+00
I130	6.82E+02	1.29E+01
I131	7.33E+01	1.39E+00
I132	1.00E+03	1.90E+01
I133	6.91E+02	1.31E+01
I134	1.48E+03	2.81E+01
I135	1.22E+03	2.31E+01
Cs-134	5.11E+02	9.70E+00
Cs-136	4.03E+01	7.66E-01
Cs-137	4.22E+02	8.01E+00
Cs-138	1.00E+04	1.90E+02
Rb-86	6.43E+01	1.22E+00

Table 14-51

REACTOR BUILDING SPRAY SYSTEM EFFECTIVENESS

Parameter	1 Spray Header Operates	2 Spray Headers Operate
Spray flow, gpm	1000	2000
Effective fall ht, ft	115	115
Rx Building Free volume, ft	1,810,000	1,810,000
Mass Median diameter, microns	786	786
Sprayed Volume Fraction	0.89	
Unsprayed to Sprayed Volume Mixing Rate, cfm	6270	
Average removal rate constant, hr^{-1}		
Elemental (4.85%)	20 (injection) 10 (recirculation	
Organic (0.15%)	0	
Partic. (95%)	2.6	
Recirculation Start Time (hr)	1.183	

Table 14-52

DELETED

Table 14-53

MAXIMUM IODINE ACTIVITY IN A WASTE GAS TANK

Isotope	Curies
^{131}I	0.41
^{132}I	0.60
^{133}I	0.48
^{134}I	0.06
^{135}I	0.34

Table 14-54

POST-BLOWDOWN MASS AND ENERGY RELEASE DATA FOR THE 5.0 FT^2 DBA RUPTURE

Time Interval (sec)	Mass Rate (lb/sec)	Energy Rate (10^5 Btu/sec)
0 - 28	0	0
28 - 40	523.6	6.147
40 - 50	344.6	4.046
50 - 58	268.0	3.145
58 - 68	212.4	2.494
68 - 88	170.2	1.998
88 - 108	145.0	1.703
108 - 135	130.0	1.588
135 - 160	484.0	2.188
160 - 210	484.0	2.184
210 - 260	484.0	1.943
260 - 300	484.0	1.239
300 - 400	484.0	1.174
400 - 500	484.0	1.127
500 - 800	484.0	1.035
800 - 1000	484.0	.984
1000 - 1500	484.0	.952
1500 - 2000	484.0	.912
2000 - 4000	484.0	.816
4000 - 4550	484.0	.747

Table 14-55

CONTAINMENT PRESSURE ANALYSIS FOR COLD LEG PUMP SUCTION BREAKS

	Blowdown Period		Reflood Period	
Break Area (ft^2)	Peak Pressure (psig)	Time (sec)	Peak Pressure (psig)	Time (sec)
8.56	49.1	16	49.1	125
7.00	49.4	20	50.7	120
5.13	49.2	24	49.5	140
3.00	47.8	40	47.3	165

Table 14-56

MASS RATE AND ENTHALPY TO THE REACTOR BUILDING FOR THE 7-FT^2 SPLIT AT THE PUMP SUCTION

Time Interval (sec.)	Average Mass Flow Rate (lb/sec)	Average Enthalpy (Btu/lb)
0 - 2	57300	558.333
2 - 4	53350	566.382
4 - 6	47035	583.019
6 - 8	34195	617.780
8 - 10	22187	689.503
10 - 12	12904	765.916
12 - 14	4768	1086.838
14 - 16	4630	735.501
16 - 18	5605	520.250
18 - 20	5416	457.903
20 - 24	2893	419.165
24 - 28	889	412.658
28 - 32	14	298.246
32 - 36	38	337.748
36 - 40	0.0	0.0
40 - 44	48	333.333
44 - 48	0.0	0.0
48 - 56	79	265.263
56 - 62	1427	416.472
62 - 68	656	1073.895
68 - 74	1477	604.153
74 - 80	2688	461.519
80 - 90	2479	477.207
90 - 100	1959	538.497
100 - 110	1131	692.002
110 - 120	561	888.632
120 - 140	157	1133.524
140 - 160	54	1180.801
160 - 180	50	1182.093
180 - 200	48	1148.691
200 - 240	323	431.353
240 - 280	673	474.770
280 - 320	556	438.506
320 - 360	340	344.001
360 - 400	413	322.518
400 - 440	540	316.633
440 - 480	391	300.288
480 - 520	383	280.439
520 - 560	359	275.384
560 - 600	345	273.663

Table 14-57

8.55 FT^2 COLD LEG BREAK, REACTOR VESSEL CAVITY

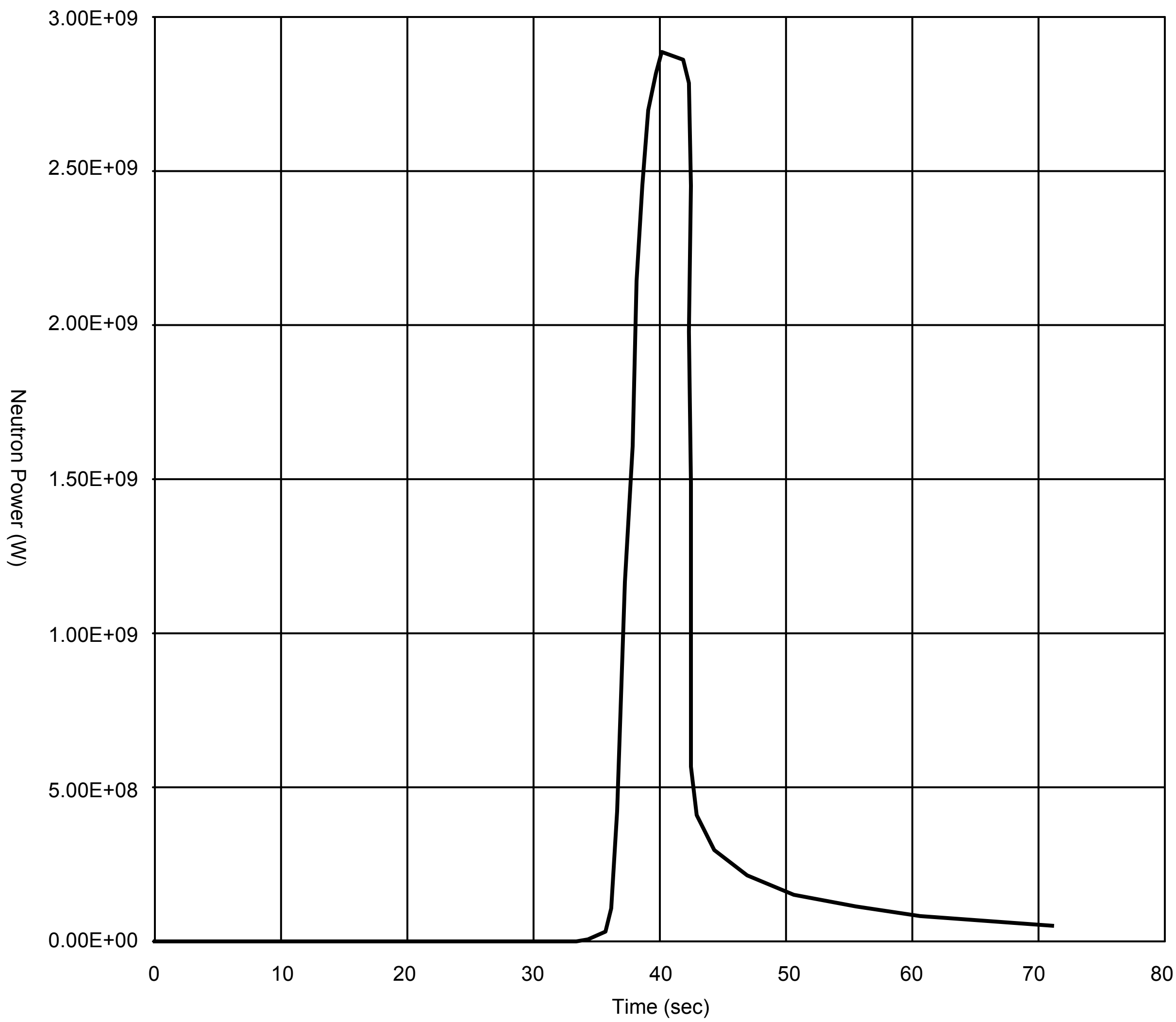
TIME (sec)	BLOWNDOWN RATE (lb/sec)	ENTHALPY (Btu/lb)
0	180000	557
0.005	163000	557
0.010	132000	557
0.015	130000	557
0.020	132000	557
0.025	127000	557
0.030	116000	557
0.035	108000	557
0.040	106000	557
0.045	108000	557
0.050	109000	557
0.055	111000	557
0.070	111000	557
0.075	110000	557
0.095	110000	558
0.10	109000	558
0.15	108000	558
0.20	106000	561
0.30	103000	563
0.40	100000	566
0.50	96000	570
0.60	93000	572
0.70	89000	573
0.80	86000	575
0.90	83000	578
1.0	80000	581
1.5	75000	587
2.0	73000	588

Table 14-58

14.14 FT^2 HOT LEG BREAK STEAM GENERATOR COMPARTMENT

TIME (sec)	BLOWNDOWN RATE (lb/sec)	ENTHALPY (Btu/lb)
0	151500	620
0.005	138200	620
0.010	136300	620
0.015	134300	620
0.020	132400	620
0.025	130400	619
0.030	128500	619
0.035	126600	619
0.040	124900	619
0.045	123000	619
0.050	121300	619
0.055	119500	619
0.060	117800	619
0.065	116000	619
0.070	114500	619
0.075	112800	619
0.080	111400	619
0.085	109800	619
0.090	108400	619
0.095	107100	619
0.10	106000	619
0.15	99899	619
0.20	103600	619
0.30	105800	619
0.40	96800	615
0.50	100300	605
0.60	107800	596
0.70	91200	606
0.80	86500	613
0.90	85300	604
1.0	83200	612
1.5	73000	605
2.0	71700	608
3.0	63700	590
4.0	62000	594
5.0	42100	611

Neutron Power Response for a Startup Accident From 10^{-7}% of Rated Power
Reactivity Addition Rate = 1.74×10^{-4} Δk/k/sec

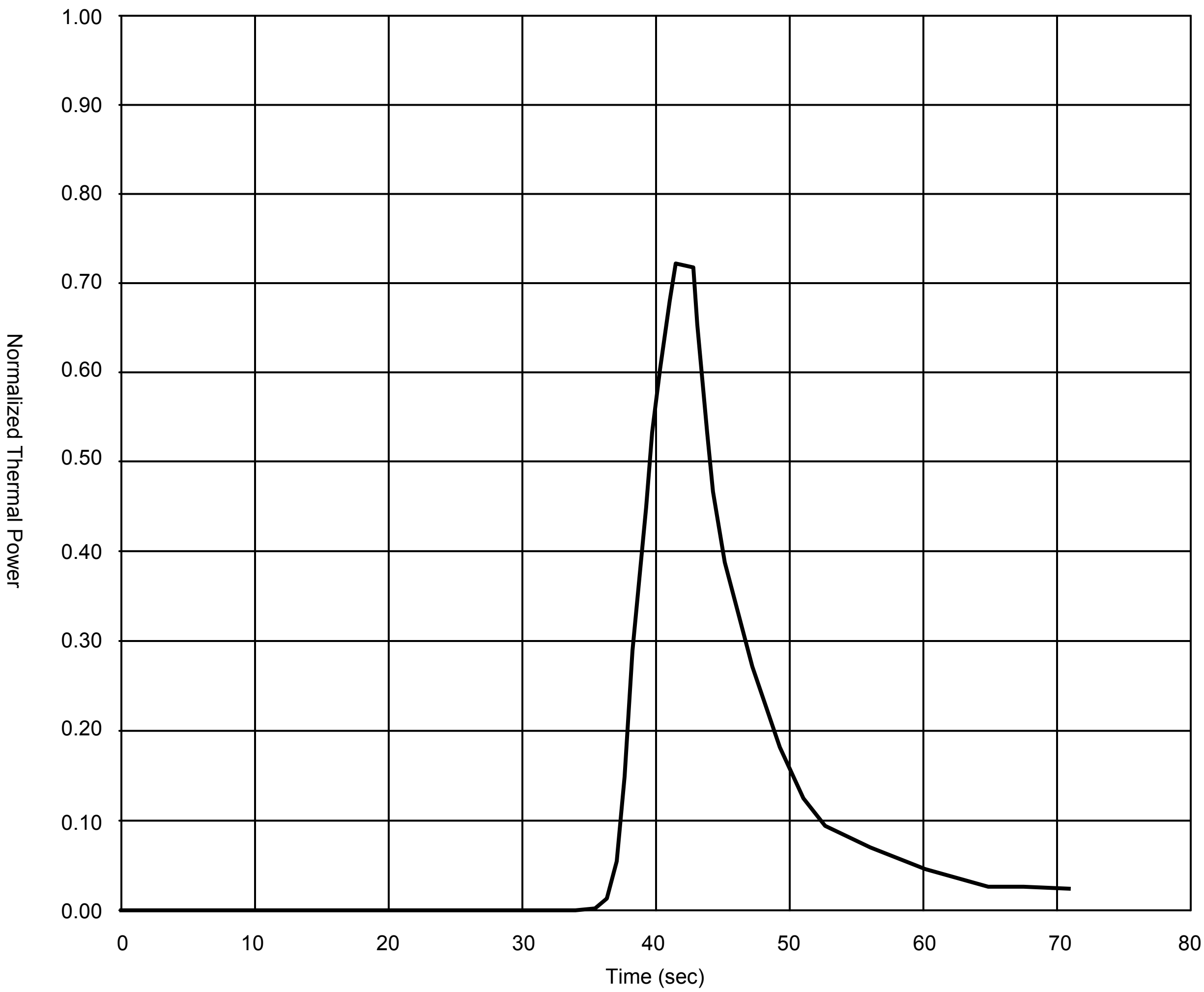


SAR FIGURE NO. 14-1A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.74×10^{-4} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.

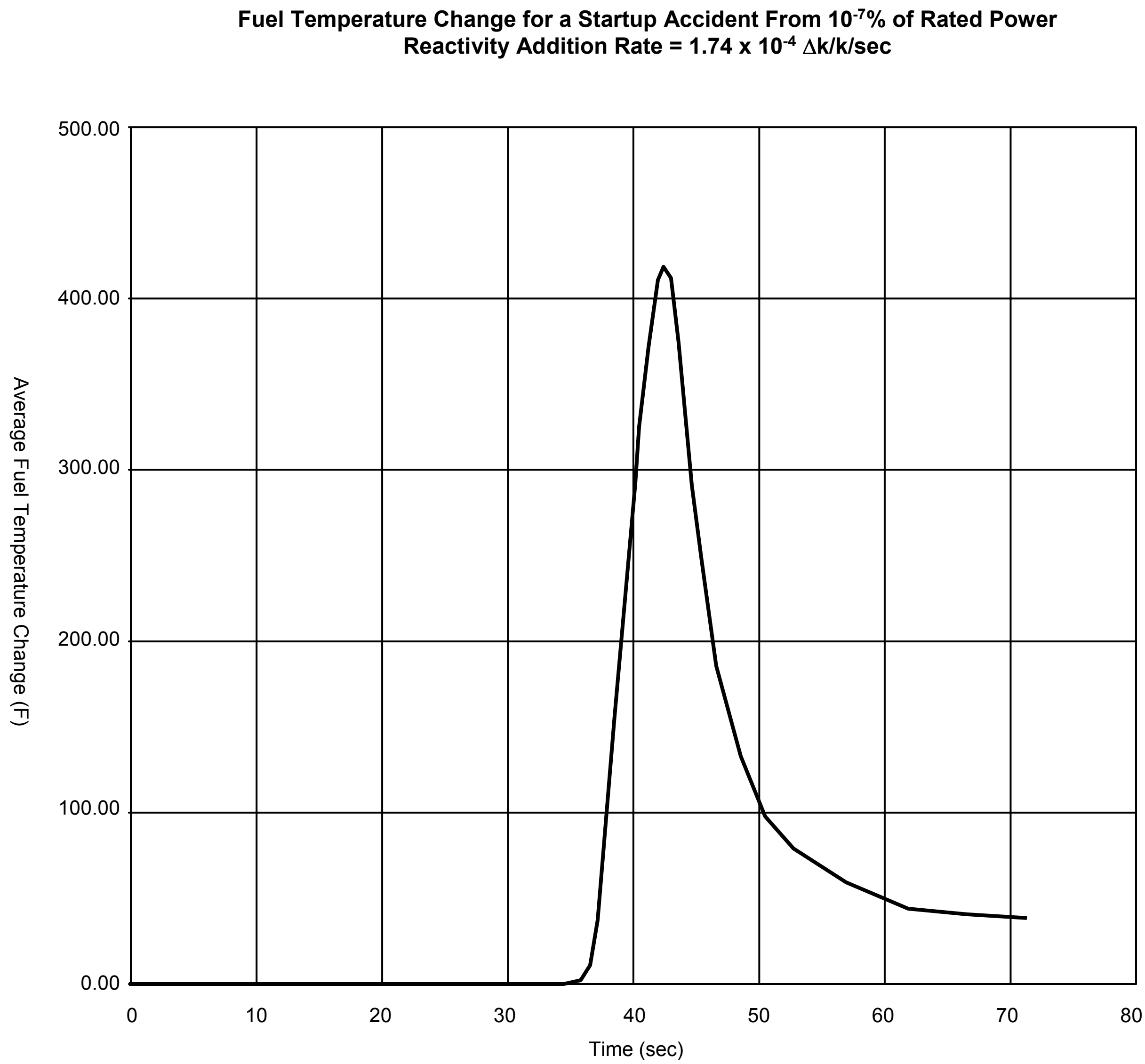
Thermal Power Response for a Startup Accident From 10^{-7}% of Rated Power
Reactivity Addition Rate = 1.74×10^{-4} Δk/k/sec



SAR FIGURE NO. 14-1B

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.74×10^{-4} ΔK/K/SEC	BASED ON DRAWING NO	SHEET / REV.

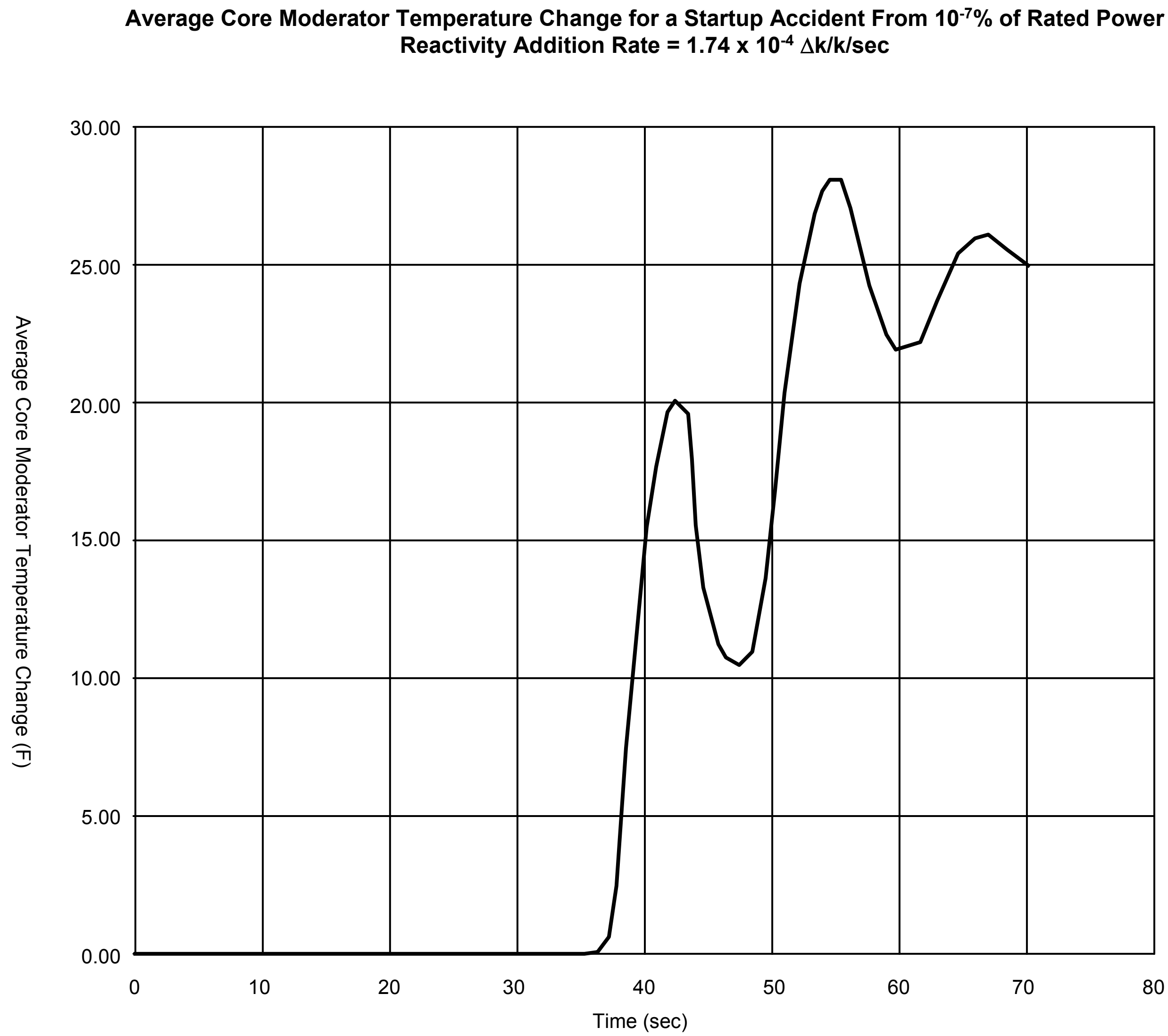


SAR FIGURE NO. 14-1C

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.74×10^{-4} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.

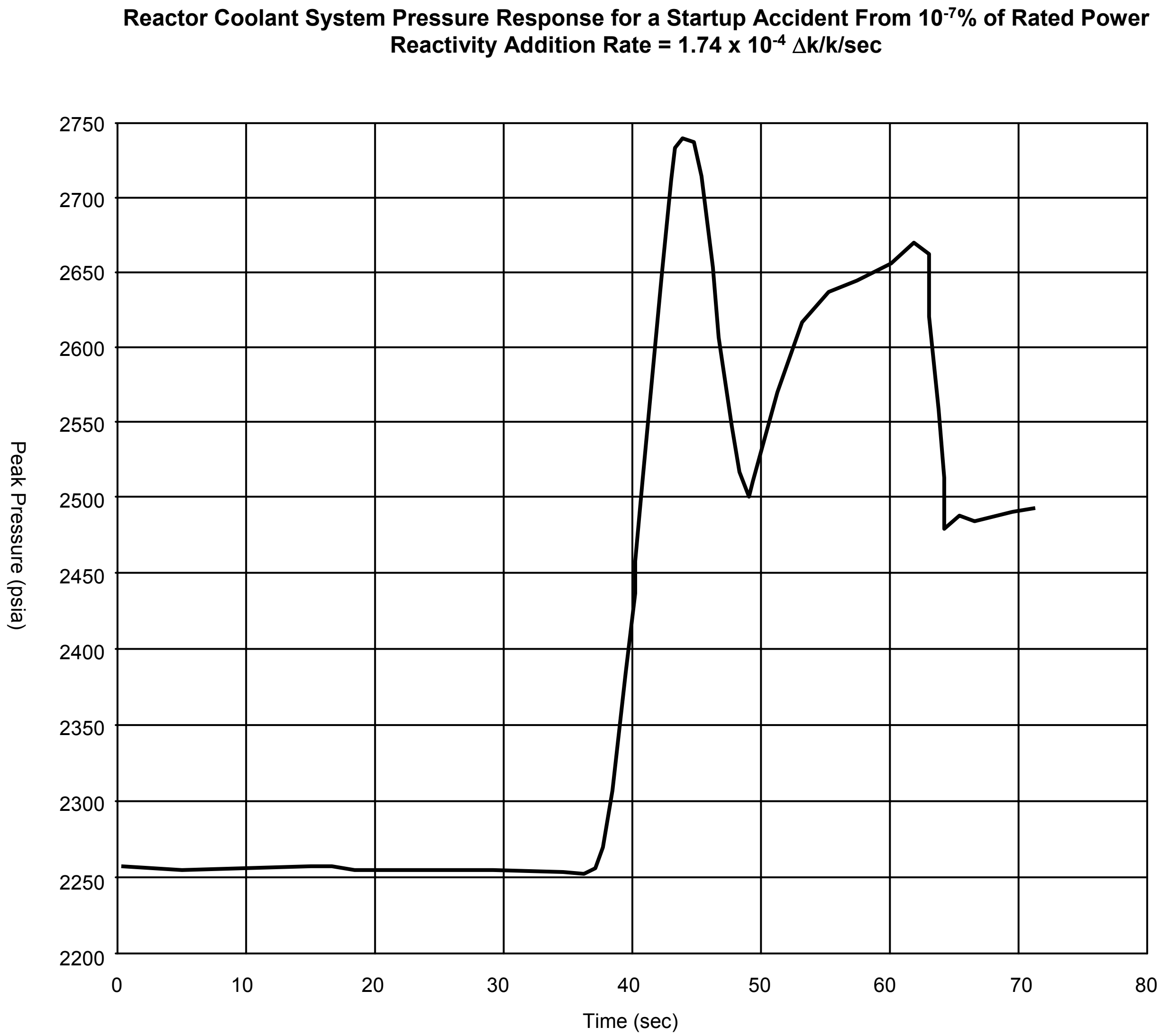


SAR FIGURE NO. 14-1D

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

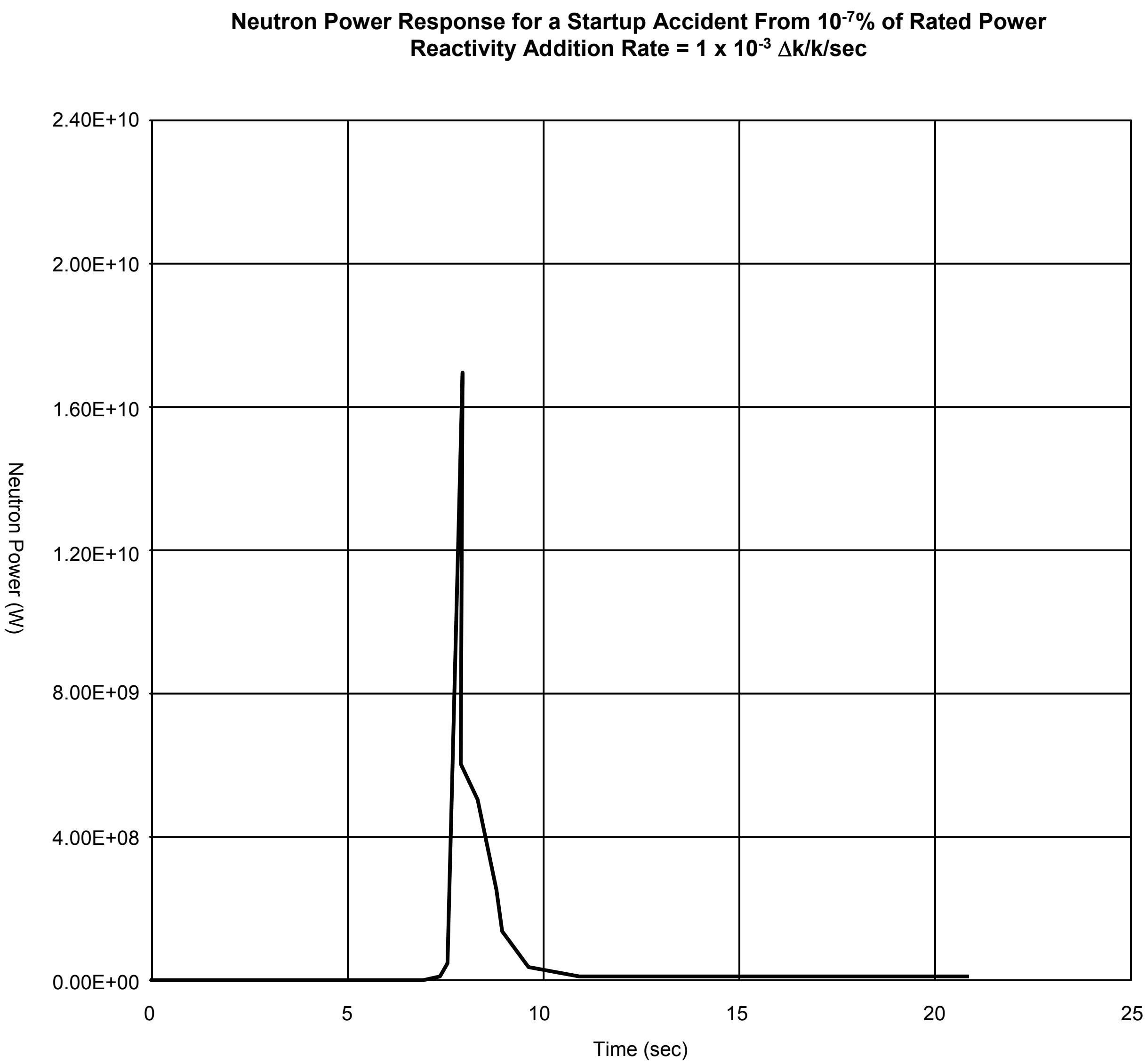
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.74×10^{-4} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-1E

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.74×10^{-4} ΔK/K/SEC	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-2A

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

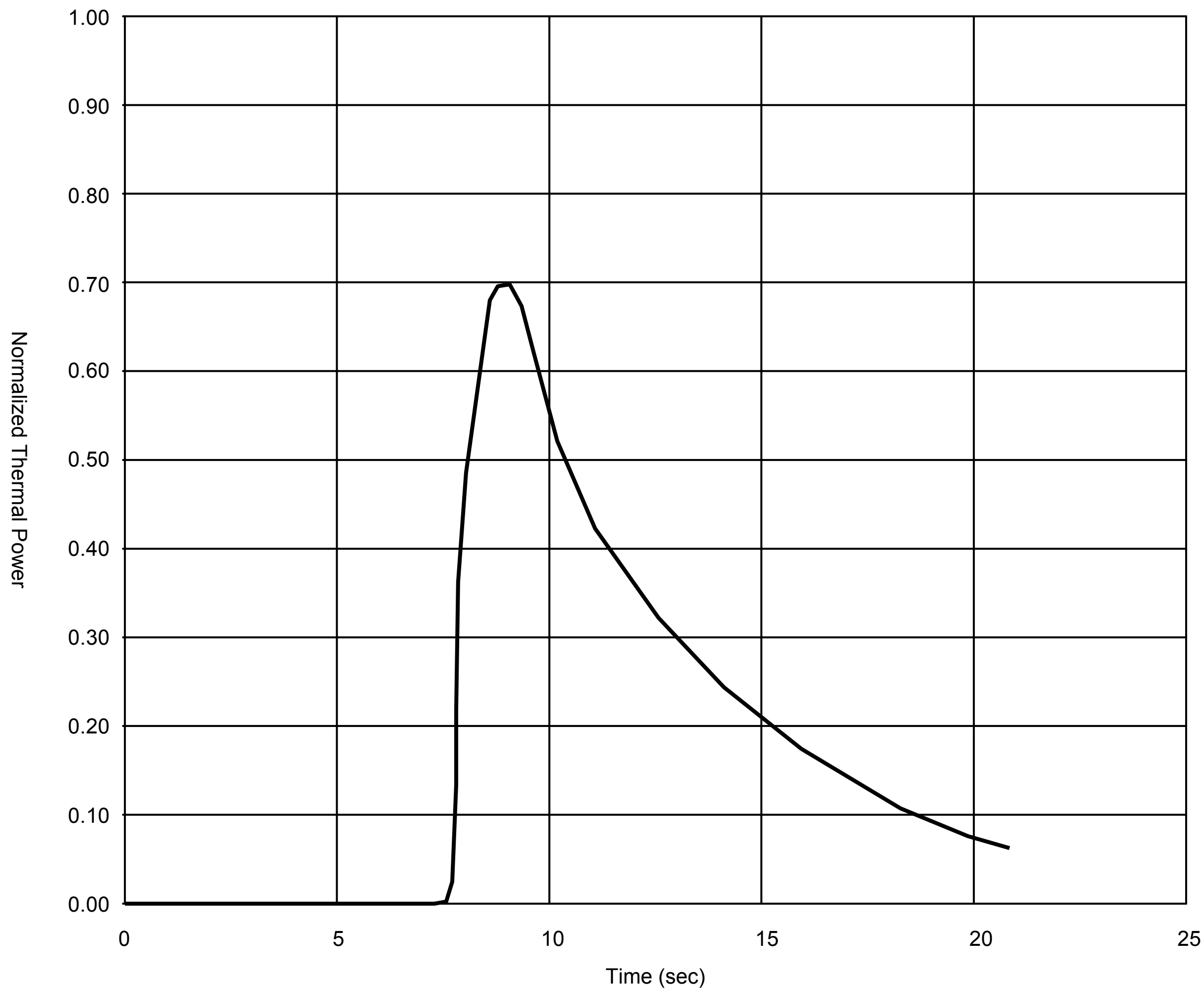
Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.0×10^{-3} ΔK/K/SEC

BASED ON DRAWING NO	SHEET	REV.

Thermal Power Response for a Startup Accident From 10^{-7}% of Rated Power
Reactivity Addition Rate = 1×10^{-3} Δk/k/sec

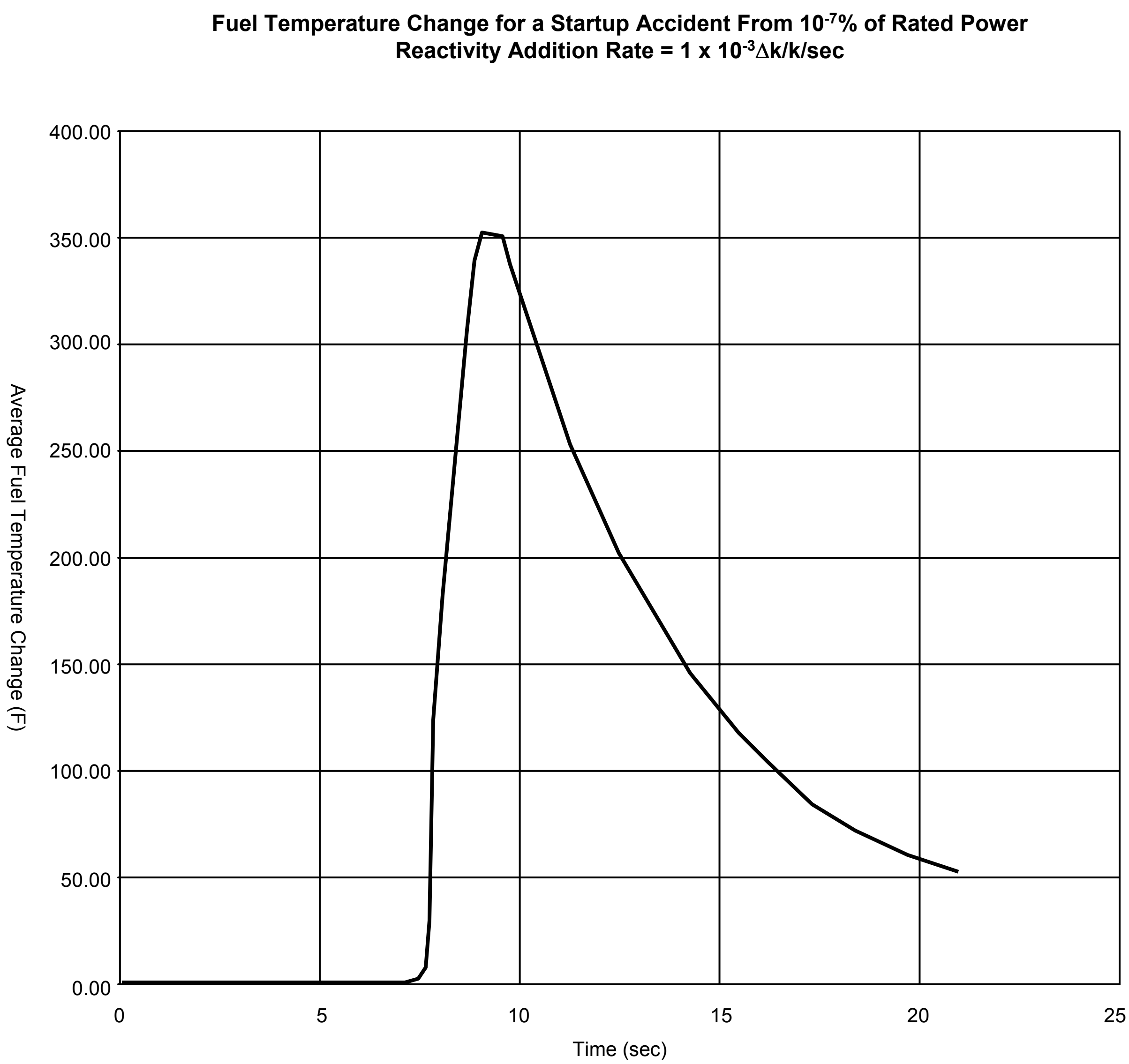


SAR FIGURE NO. 14-2B

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.0×10^{-3} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.



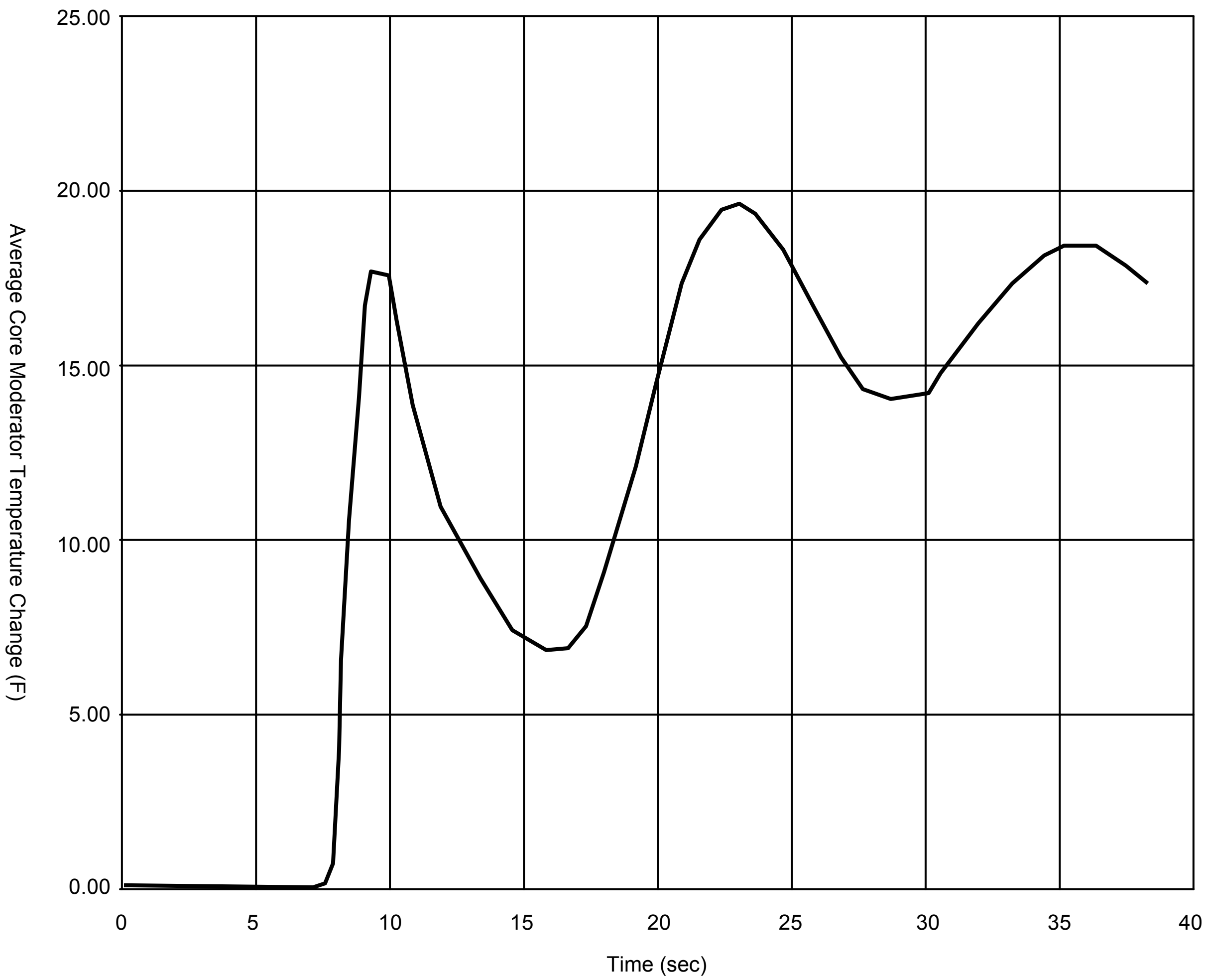
SAR FIGURE NO. 14-2C

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.0×10^{-3} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.

Average Core Moderator Temperature Change for a Startup Accident From 10^{-7}% of Rated Power
Reactivity Addition Rate = 1×10^{-3} Δk/k/sec

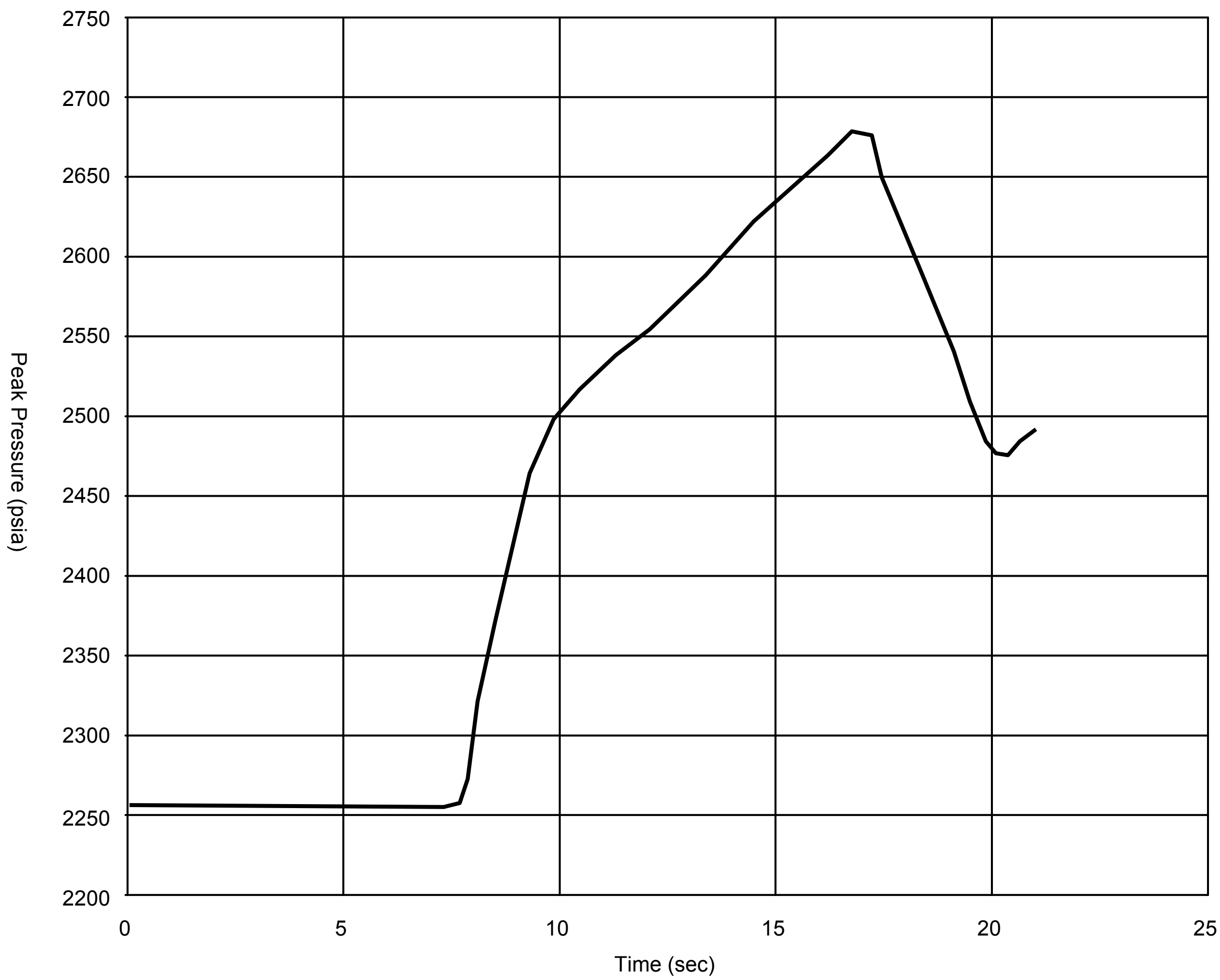


SAR FIGURE NO. 14-2D

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.0×10^{-3} ΔK/K/SEC	BASED ON DRAWING NO	SHEET / REV.

Reactor Coolant System Pressure Response for a Startup Accident From 10^{-7}% of Rated Power
Reactivity Addition Rate = 1×10^{-3} Δk/k/sec

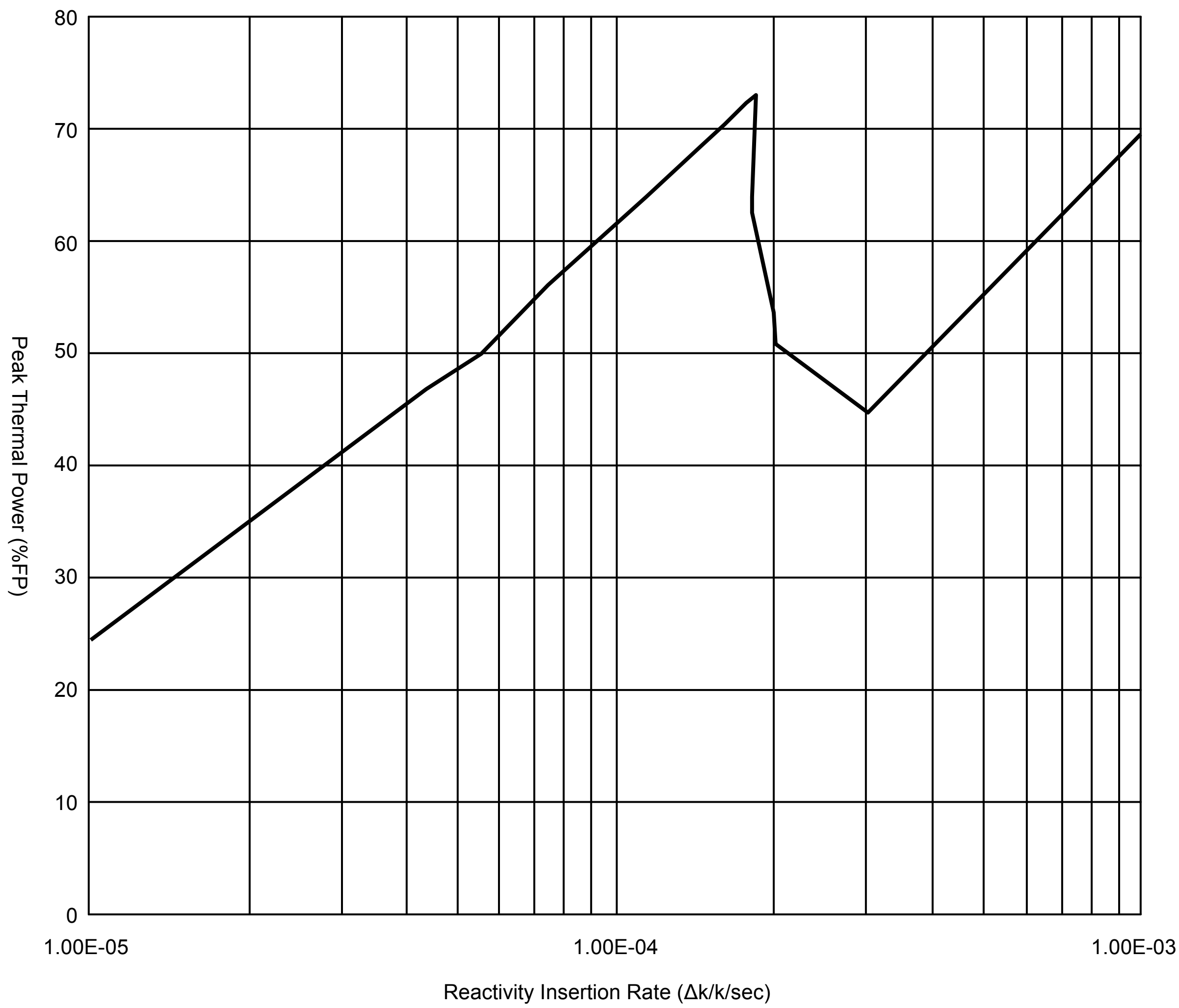


SAR FIGURE NO. 14-2E

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
STARTUP ACCIDENT FROM 10^{-7}% RATED POWER USING A REACTIVITY ADDITION RATE OF 1.0×10^{-3} ΔK/K/SEC	BASED ON DRAWING NO	SHEET	REV.

Peak Thermal Power vs. Reactivity Addition Rate for a Startup Accident
From 10^{-7}% of Rated Power



SAR FIGURE NO. 14-3

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK THERMAL POWER VS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 1.0 x 10^{-7}% RATED POWER	BASED ON DRAWING NO	SHEET / REV.

DELETED

SAR FIGURE NO. 14-4

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

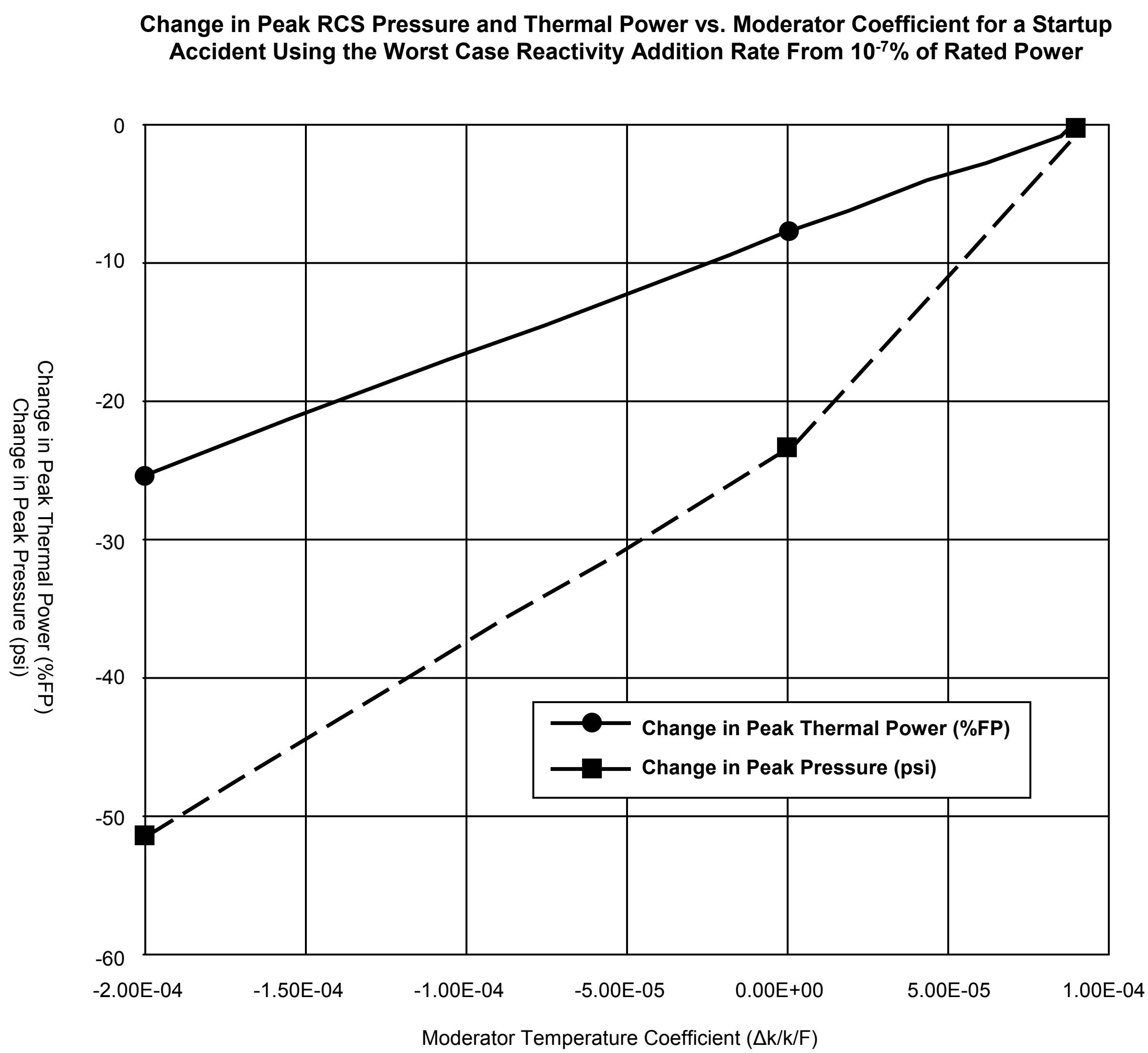
PEAK THERMAL POWER VS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER	BASED ON DRAWING NO	SHEET	REV.

DELETED

SAR FIGURE NO. 14-5

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK PRESSURE VS PSV FLOWRATE FOR A STARTUP ACCIDENT USING A REACTIVITY ADDITION RATE OF 1.73 x 10^{-4} ΔK/K/SEC FROM 10^{-9} RATED POWER	BASED ON DRAWING NO	SHEET | REV.



SAR FIGURE NO. 14-6

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK PRESSURE & THERMAL POWER VS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING THE WORST CASE REACTIVITY ADDITION RATE FROM 10^{-7}% RATED POWER	BASED ON DRAWING NO	SHEET / REV.

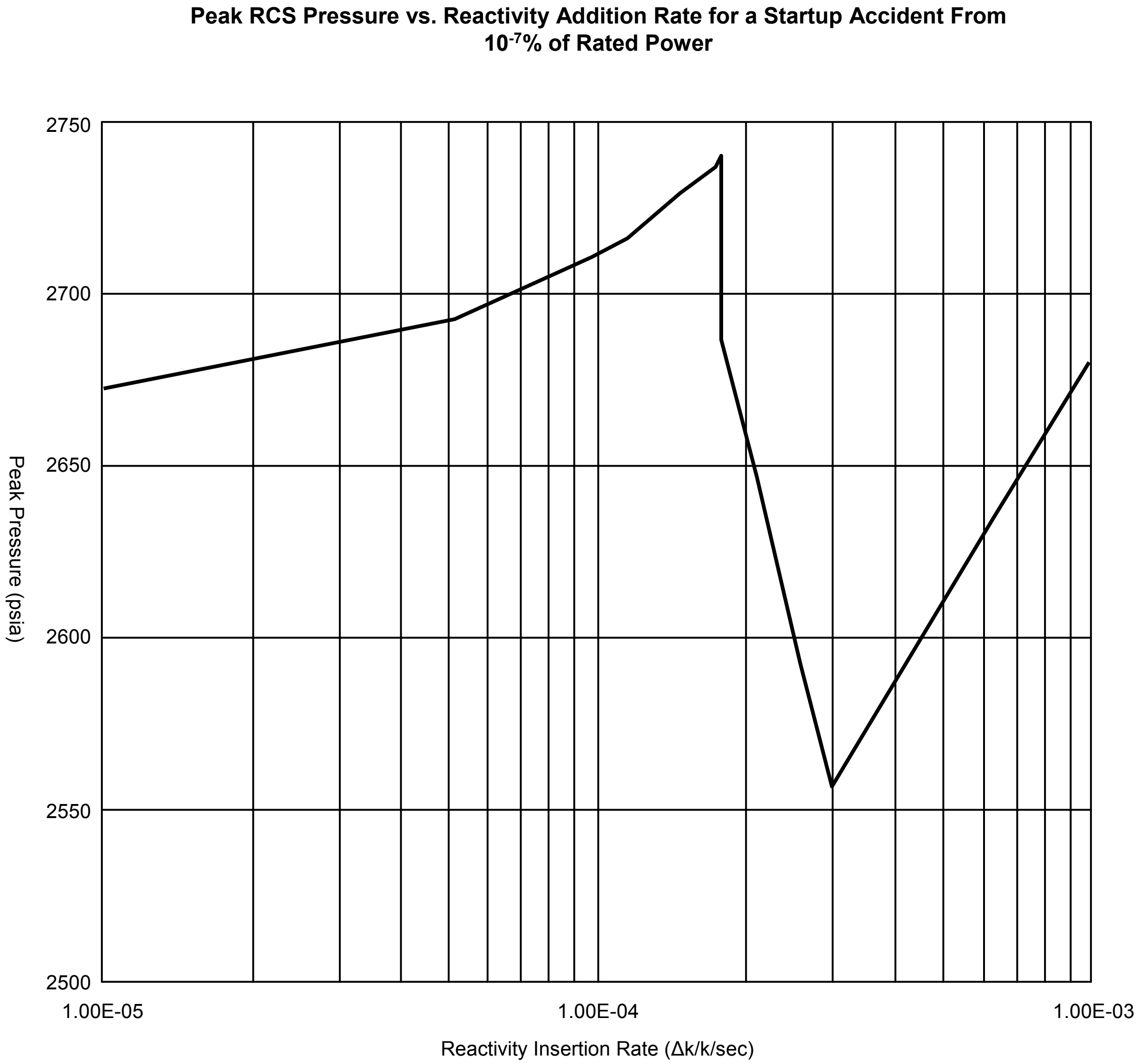
DELETED

SAR FIGURE NO. 14-7

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

PEAK RCS PRESSURE VS INITIAL PRESSURIZER LEVEL FOR A STARTUP ACCIDENT USING A REACTIVITY ADDITION RATE OF 1.73×10^{-4} $\Delta K/K/SEC$ FROM 10^{-9} RATED POWER	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-8

AMENDMENT 20

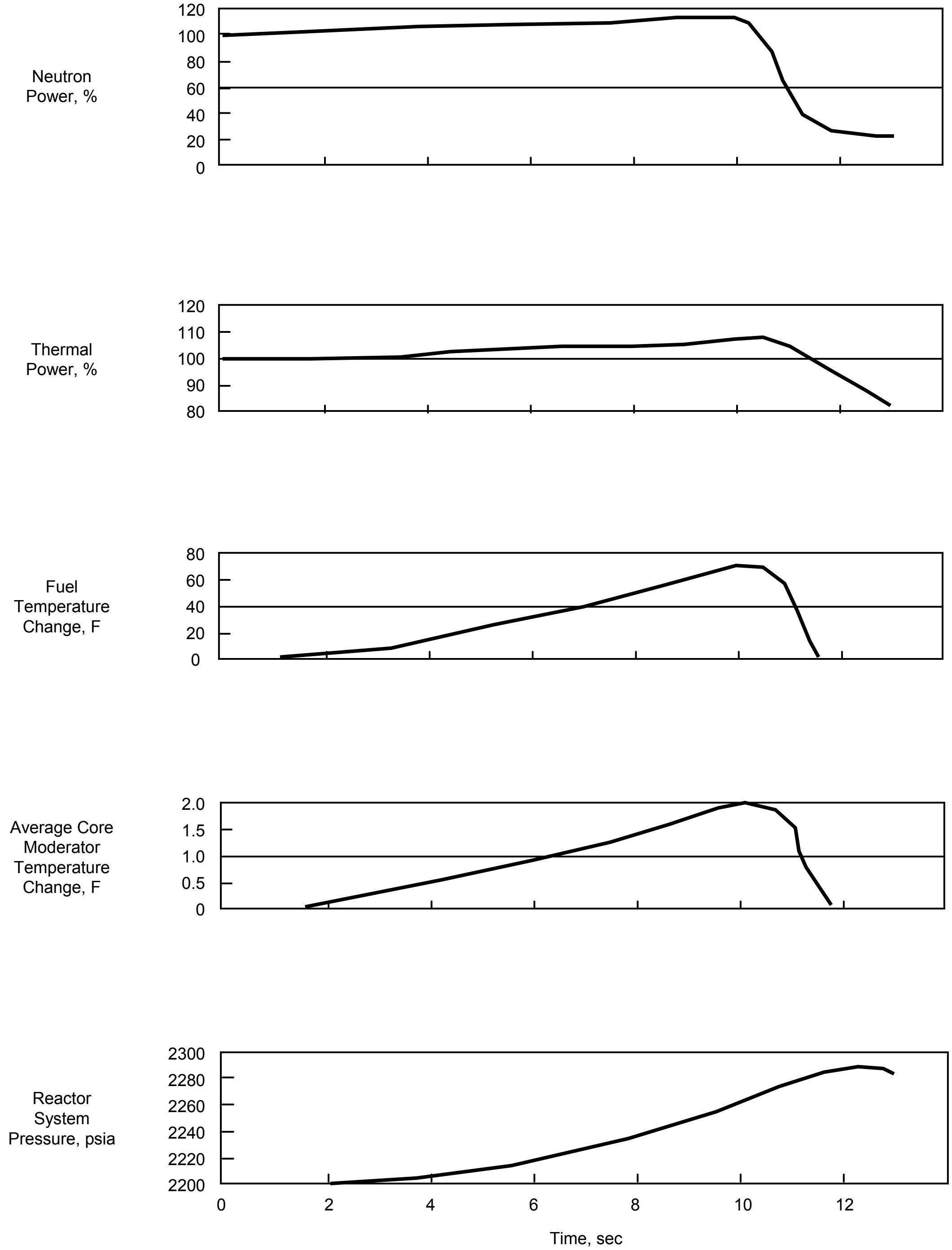
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

PEAK PRESSURE VS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-7}% RATED POWER

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-9

AMENDMENT 20

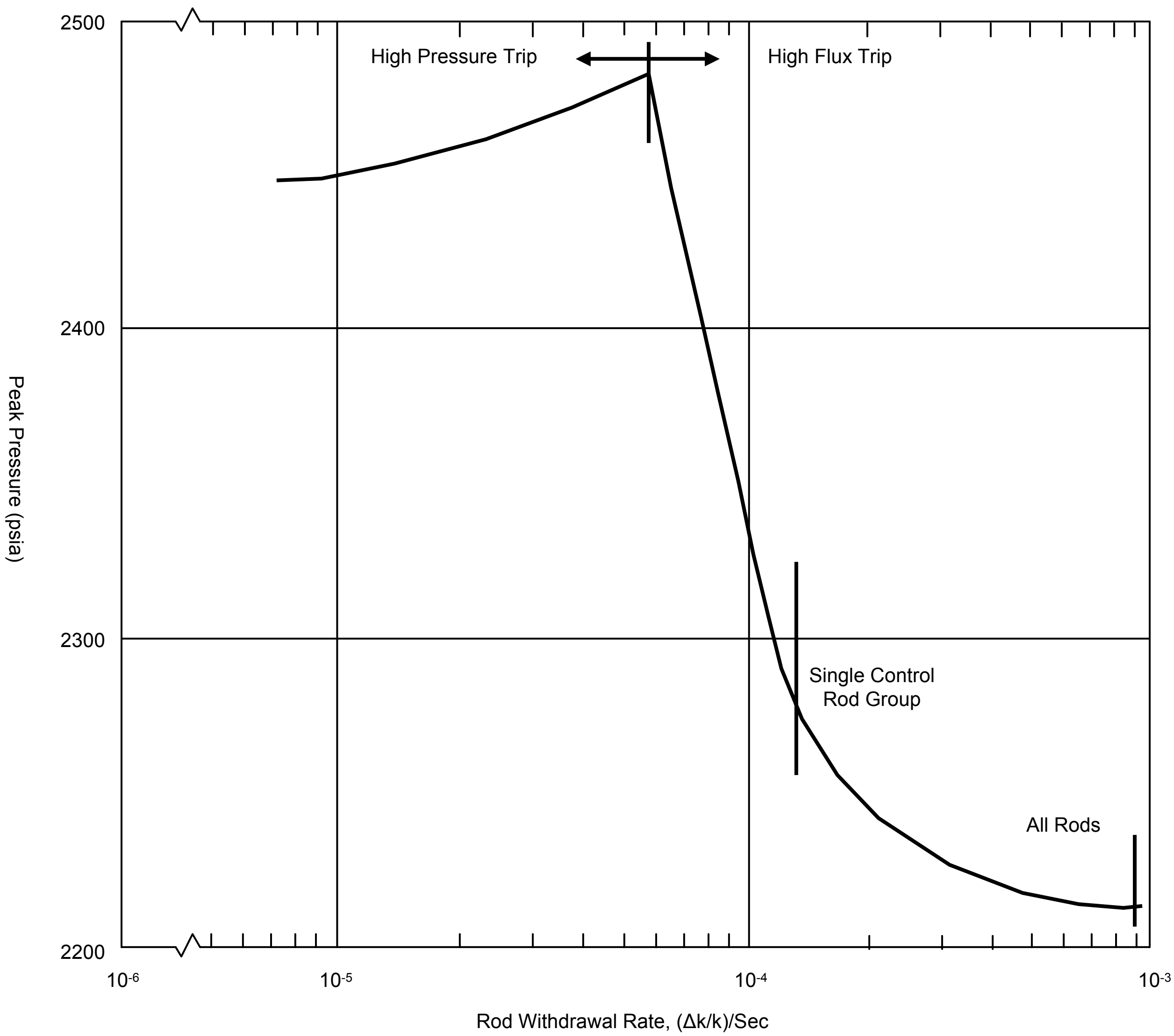
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.71% ΔK/K ROD GROUP AT 1.23 x 10^{-4} ΔK/K/S; HIGH FLUX REACTOR TRIP IS ACTUATED

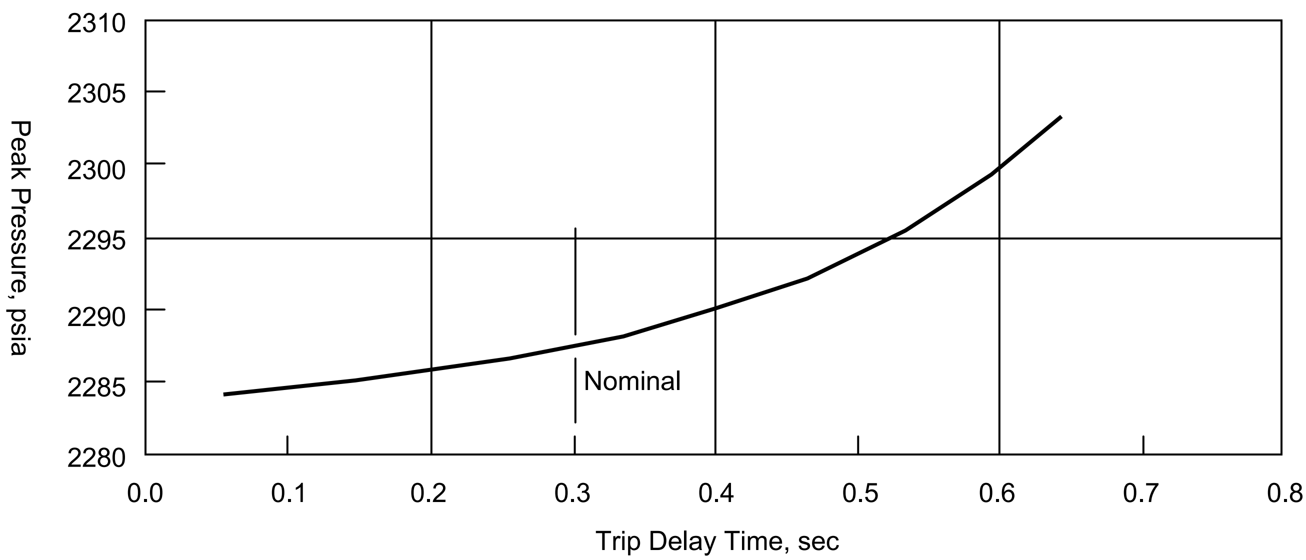
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-10

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK PRESSURE VS ROD WITHDRAWAL RATE FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER	BASED ON DRAWING NO	SHEET / REV.

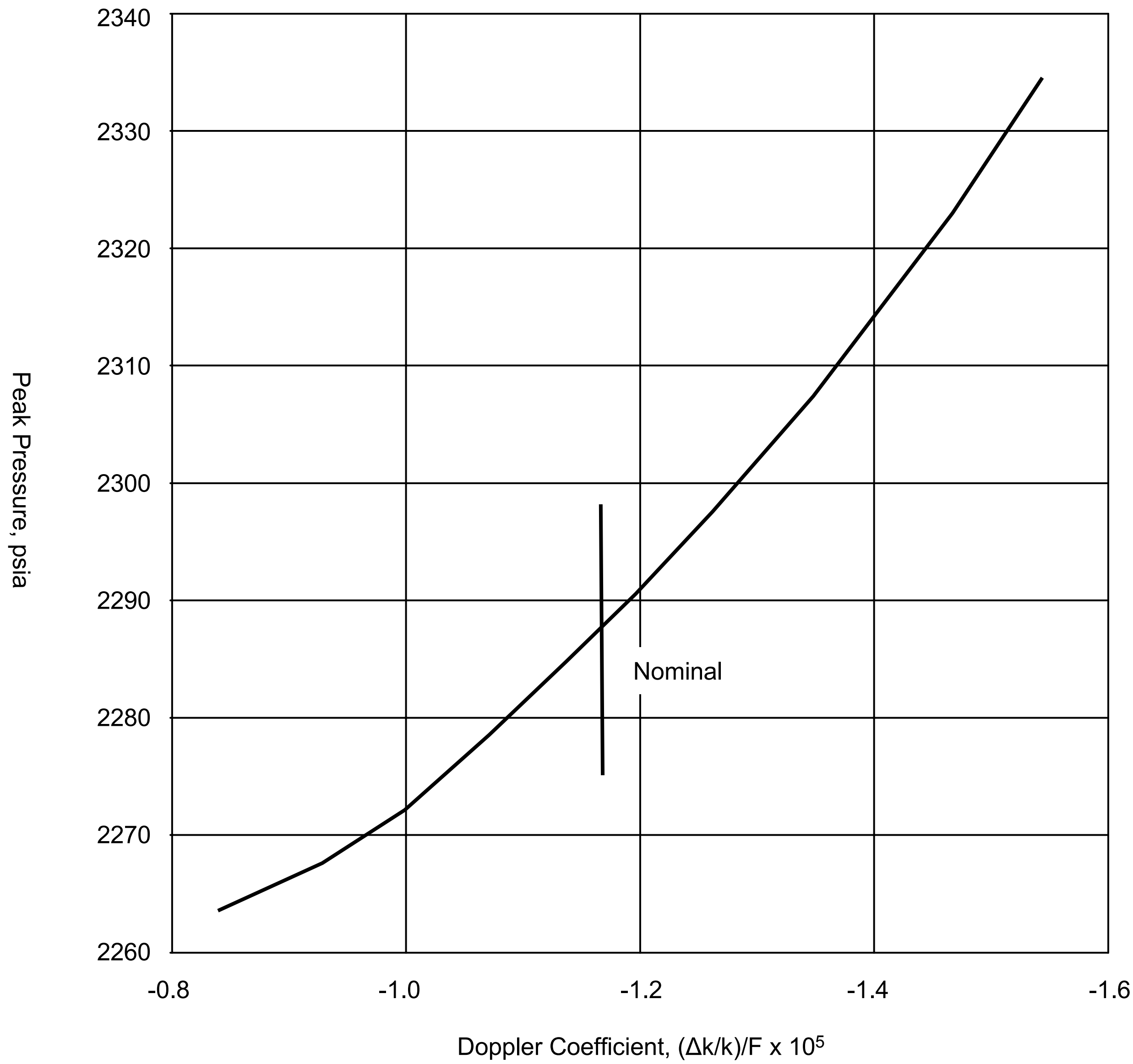


SAR FIGURE NO. 14-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

PEAK PRESSURE VS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING 1.71% ΔK/K ROD GROUP	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-12

AMENDMENT 20

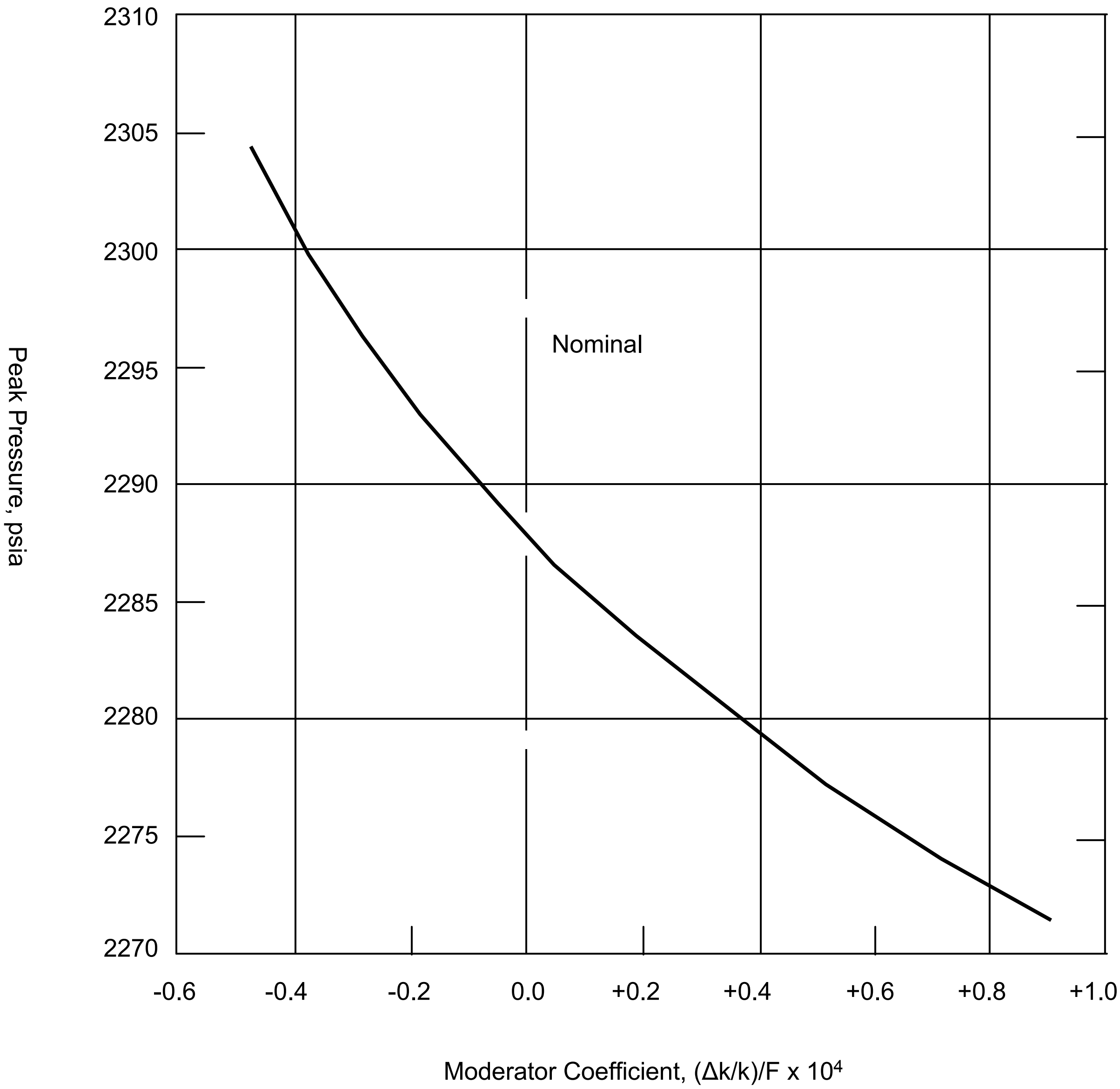
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

PEAK PRESSURE VS DOPPLER COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING 1.71% ΔK/K ROD GROUP

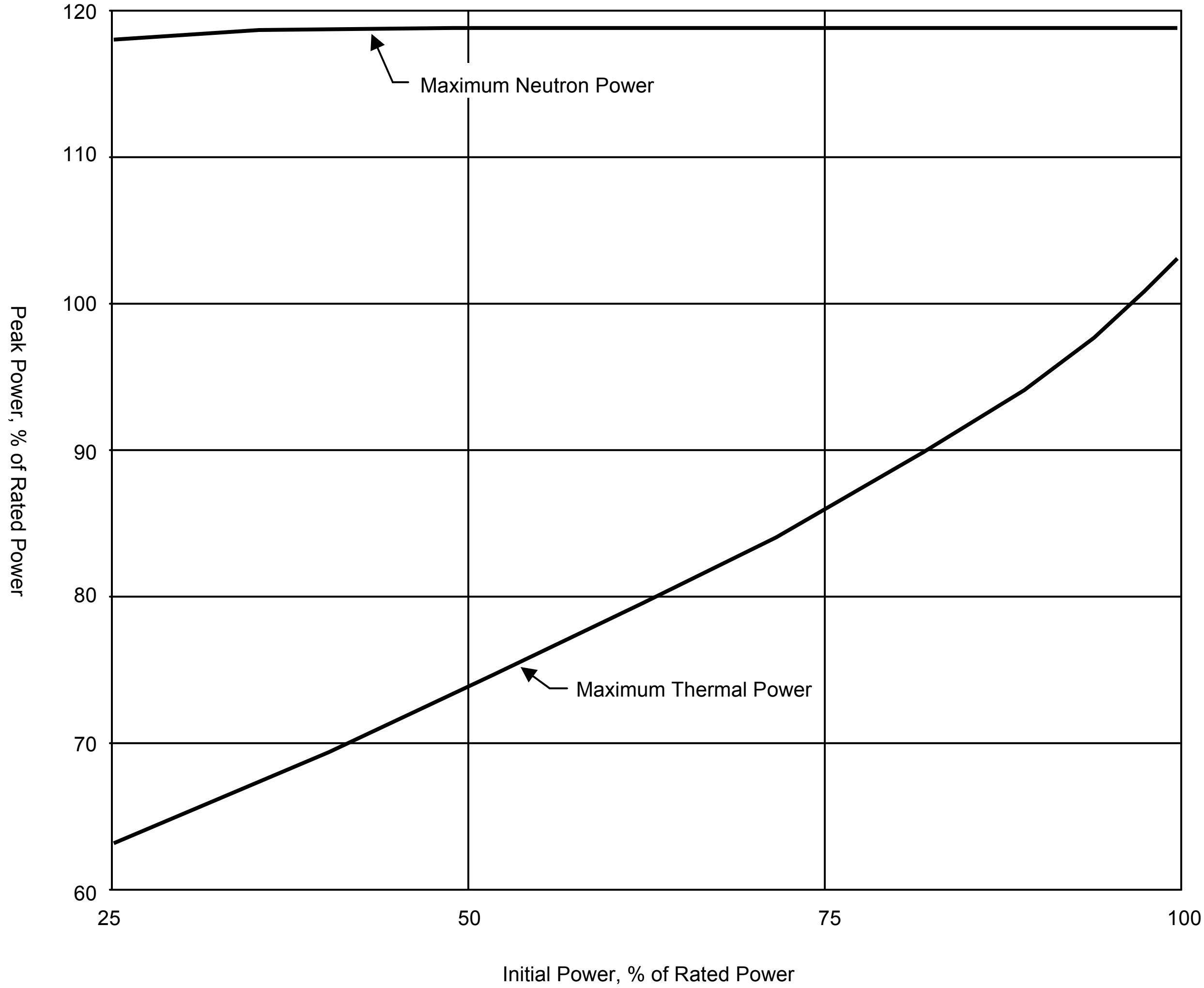
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-13

AMENDMENT 20

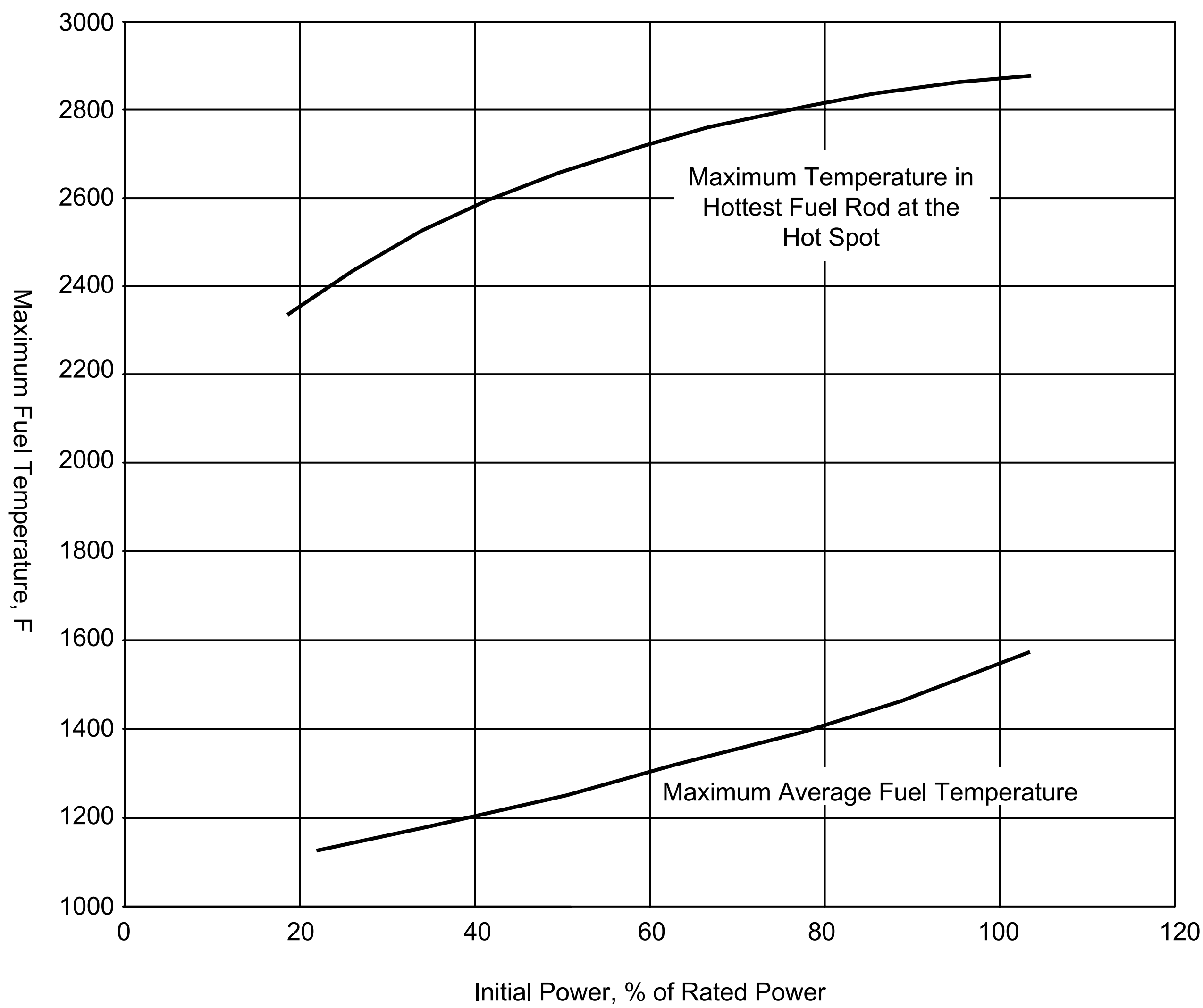
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK PRESSURE VS MODERATOR COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING 1.71% ΔK/K ROD GROUP	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-14

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
MAXIMUM NEUTRON AND THERMAL POWER FOR AN ALL- RODS WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-15

AMENDMENT 20

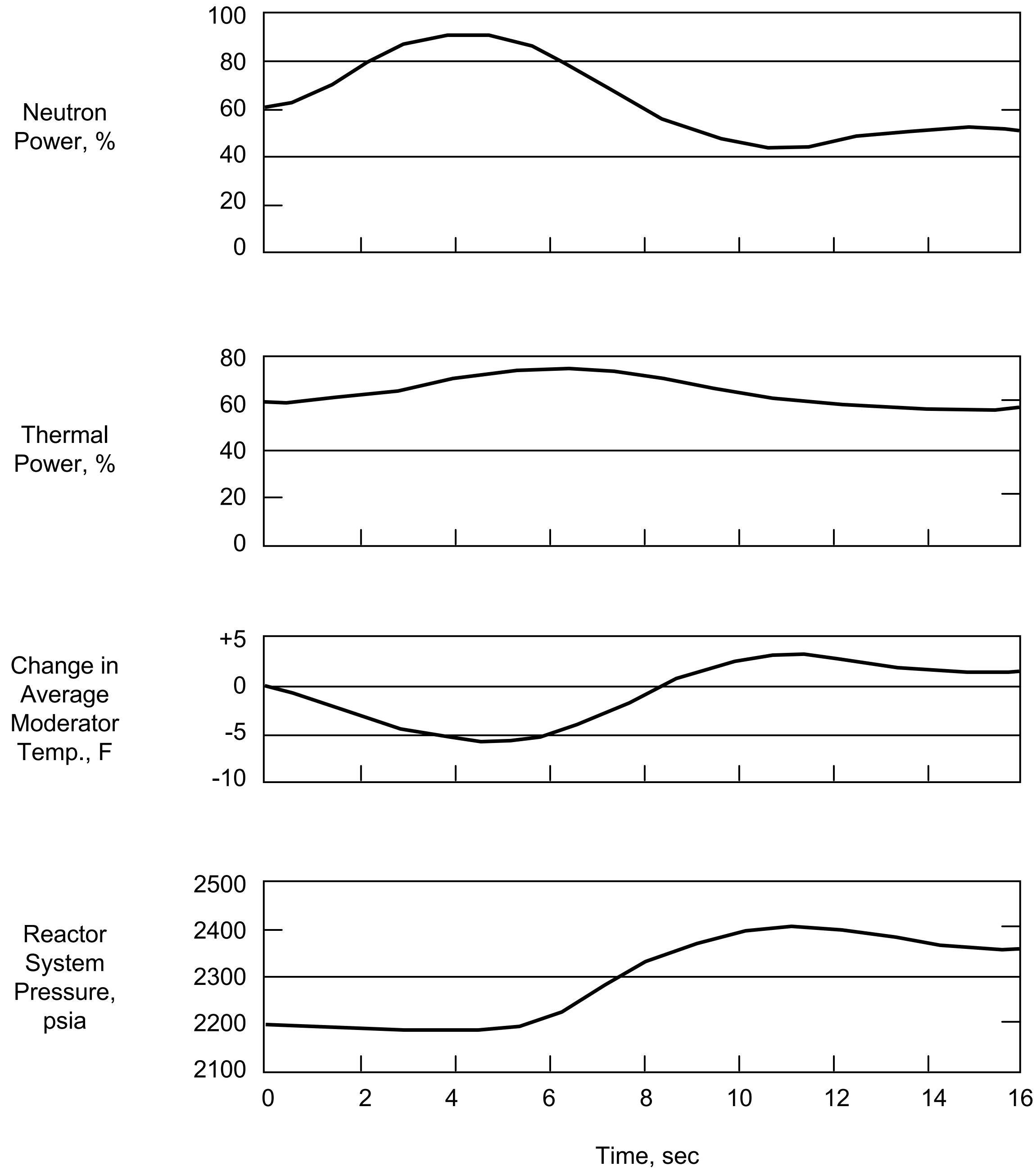
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

PEAK FUEL TEMPERATURE IN AVERAGE ROD AND HOT SPOT FOR AN ALL- RODS WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS

BASED ON DRAWING NO	SHEET	REV.

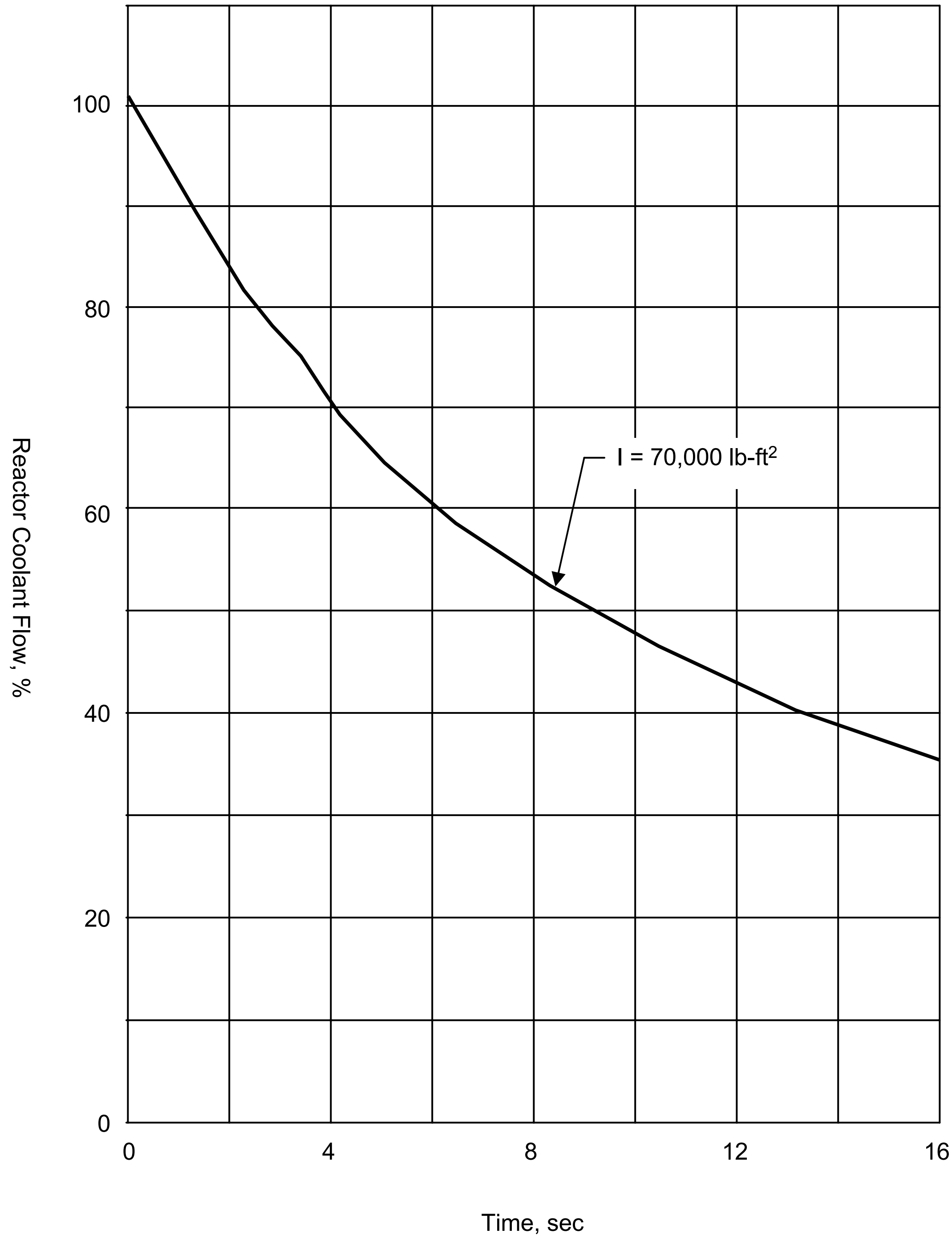


SAR FIGURE NO. 14-16

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

PUMP START-UP FROM 60% POWER AND 49% FLOW	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-17

AMENDMENT 20

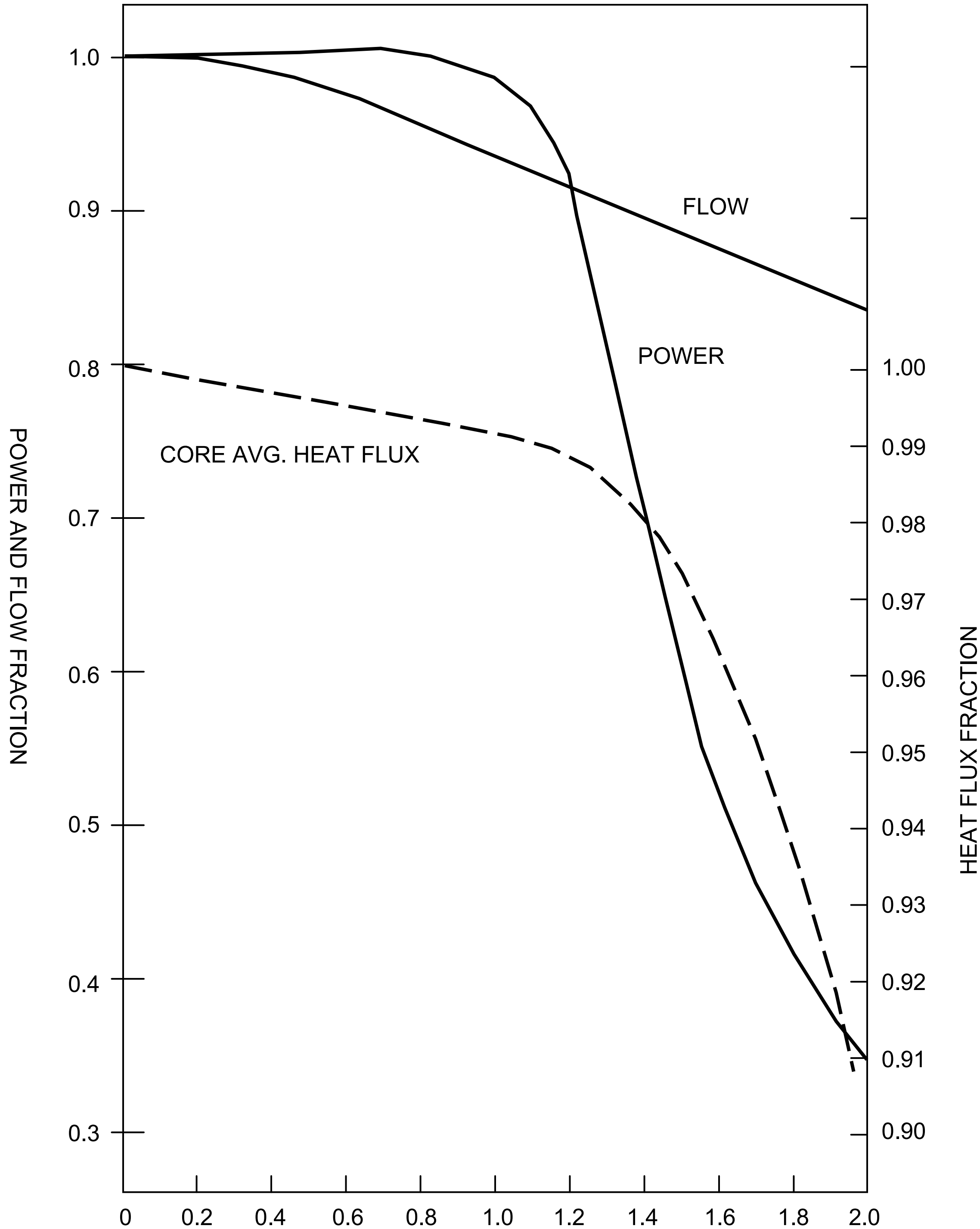
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

PERCENT REACTOR COOLANT FLOW AS A FUNCTION OF TIME AFTER LOSS OF PUMP POWER

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-18A

AMENDMENT 20

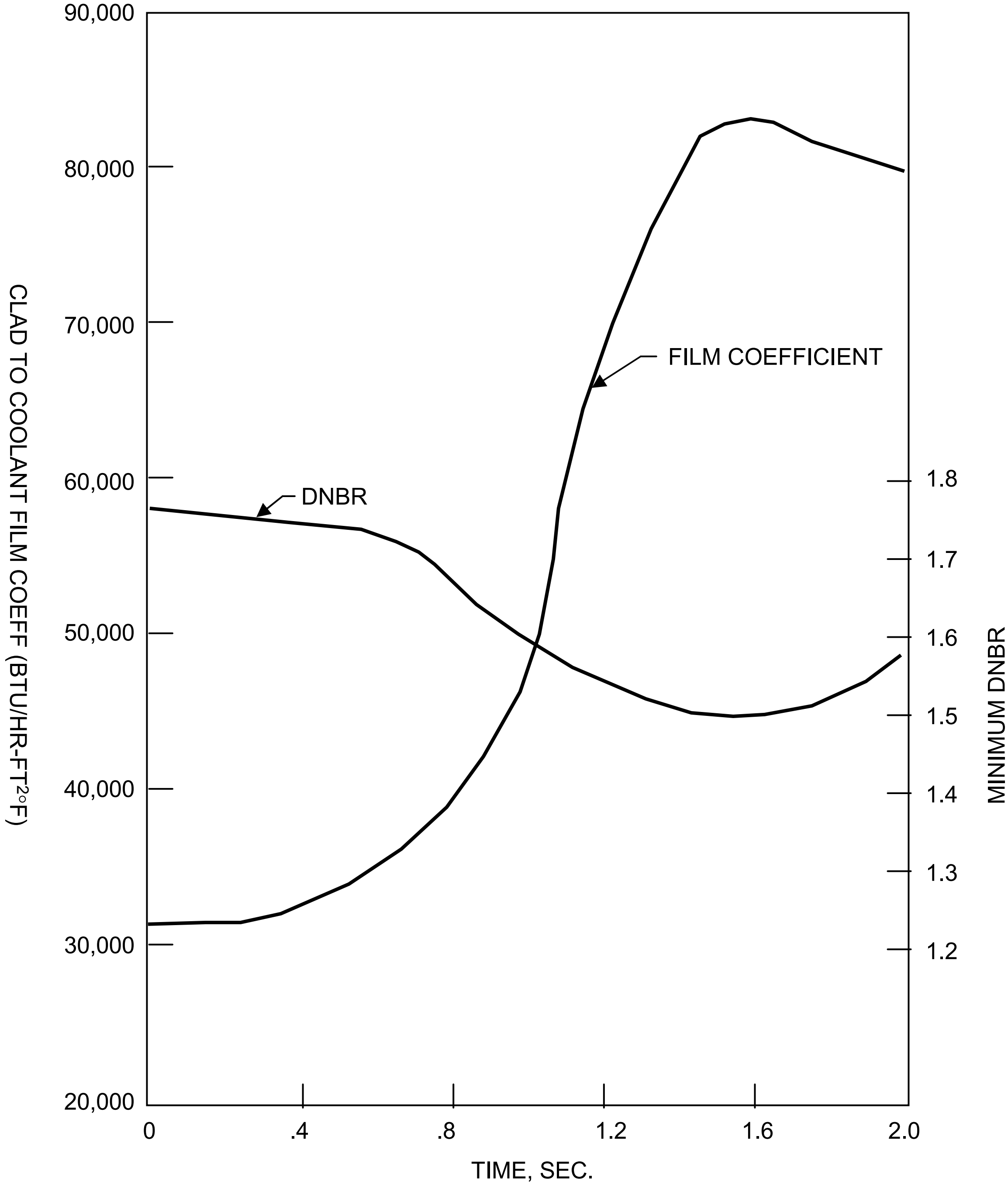
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

POWER, FLOW, AND HEAT FLUX VS TIME FOR DENSIFIED FUEL, FOUR-PUMP COASTDOWN

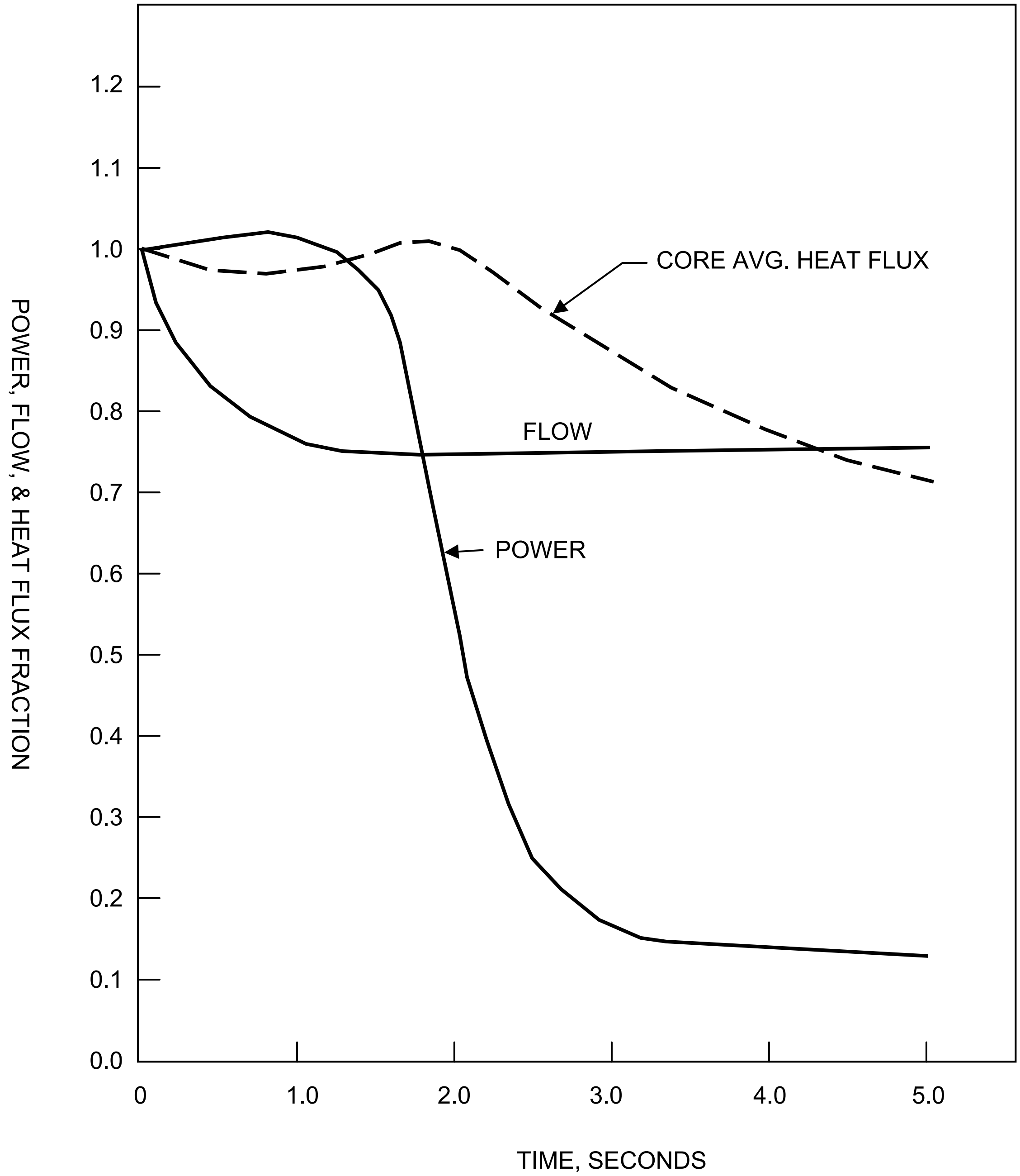
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-18B

AMENDMENT 20

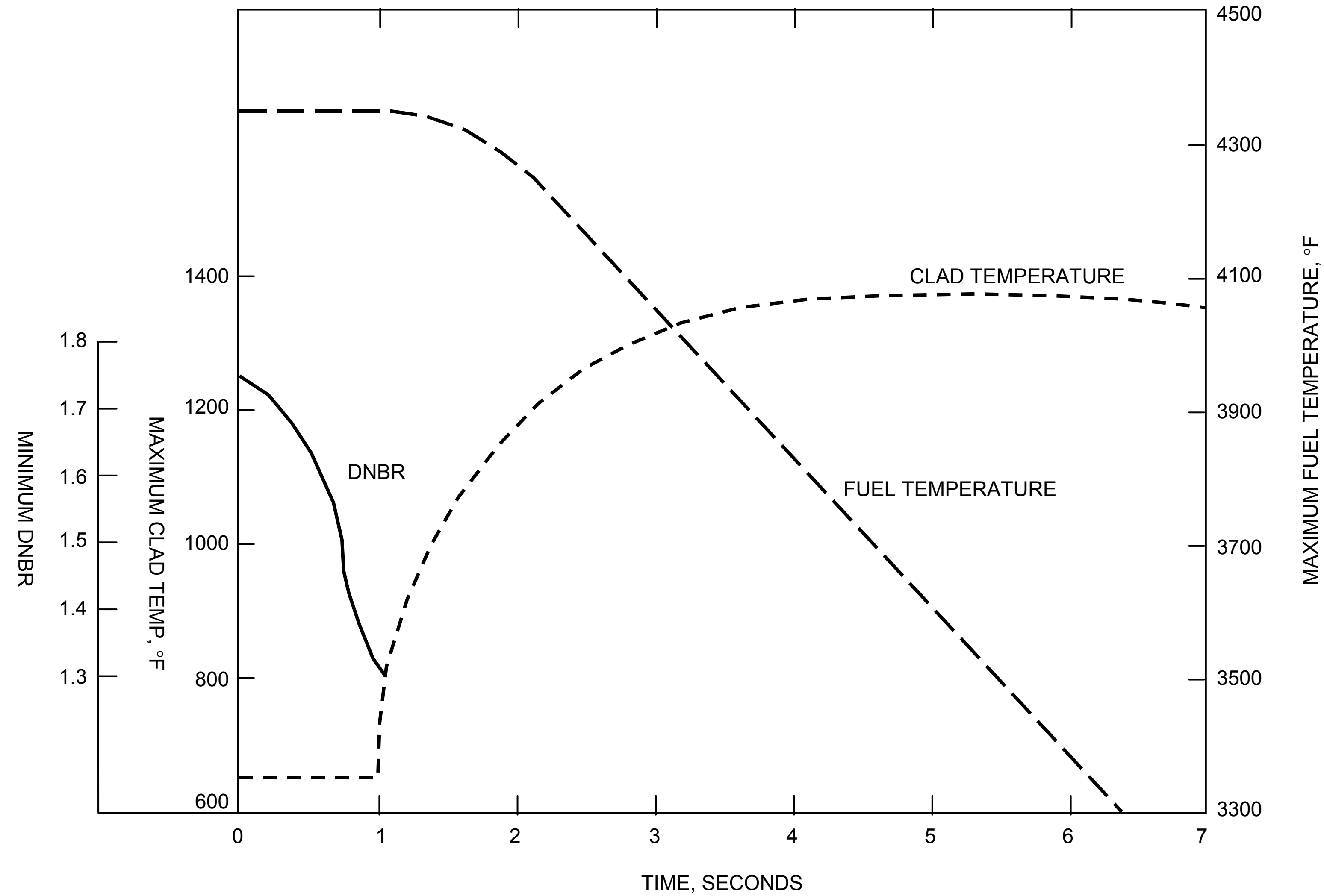
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
DNBR AND FILM COEFFICIENT VS TIME FOR DENSIFIED FUEL, FOUR-PUMP COASTDOWN	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-19A

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
POWER, FLOW, AND HEAT FLUX VS TIME FOR DENSIFIED FUEL, LOCKED ROTOR ACCIDENT	BASED ON DRAWING NO		SHEET	REV.



CLADDING TEMPERATURE, FUEL TEMPERATURE, & DNBR VS TIME FOR DENSIFIED FUEL, LOCKED ROTOR ACCIDENT

SAR FIGURE NO. 14-19B

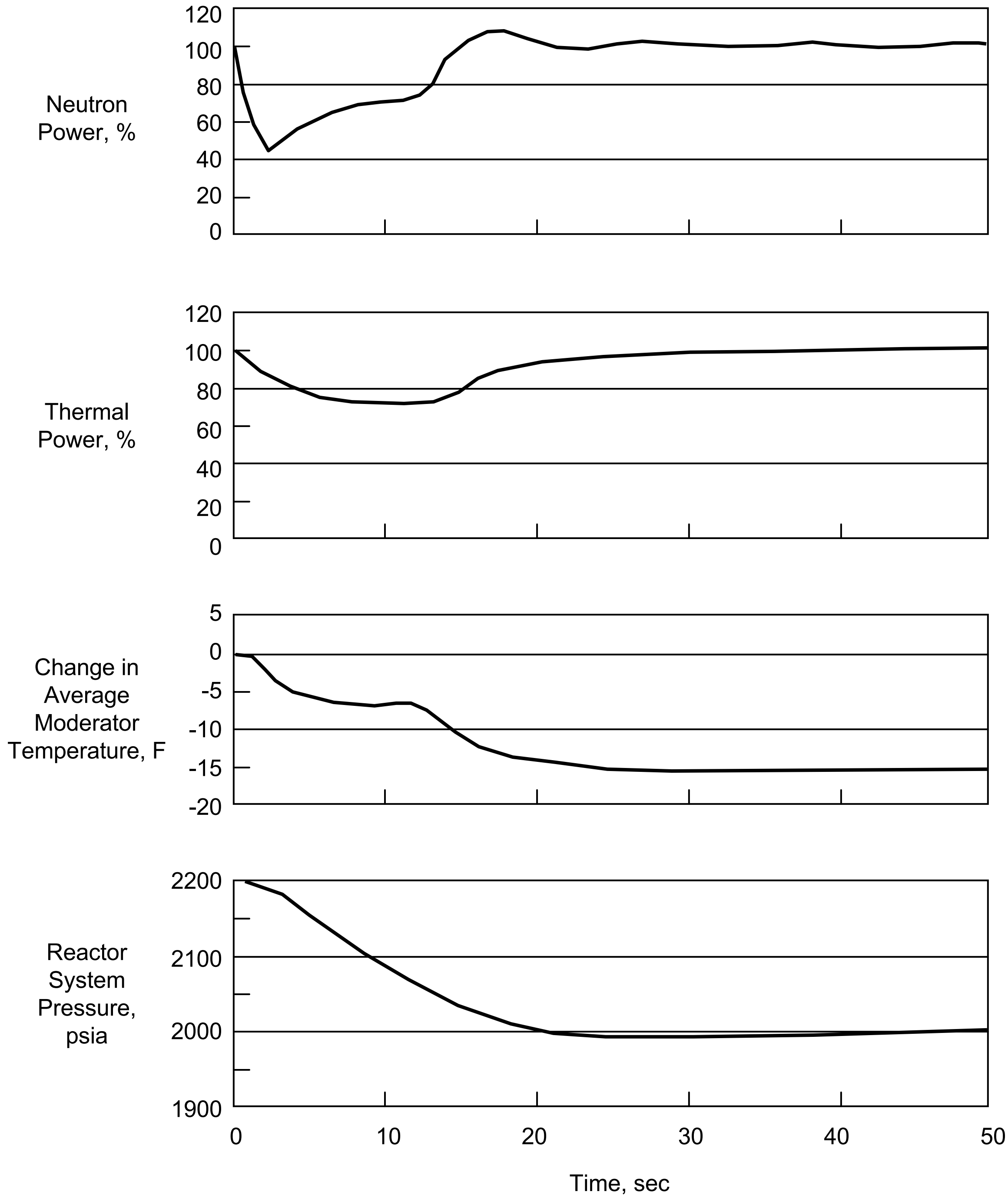
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

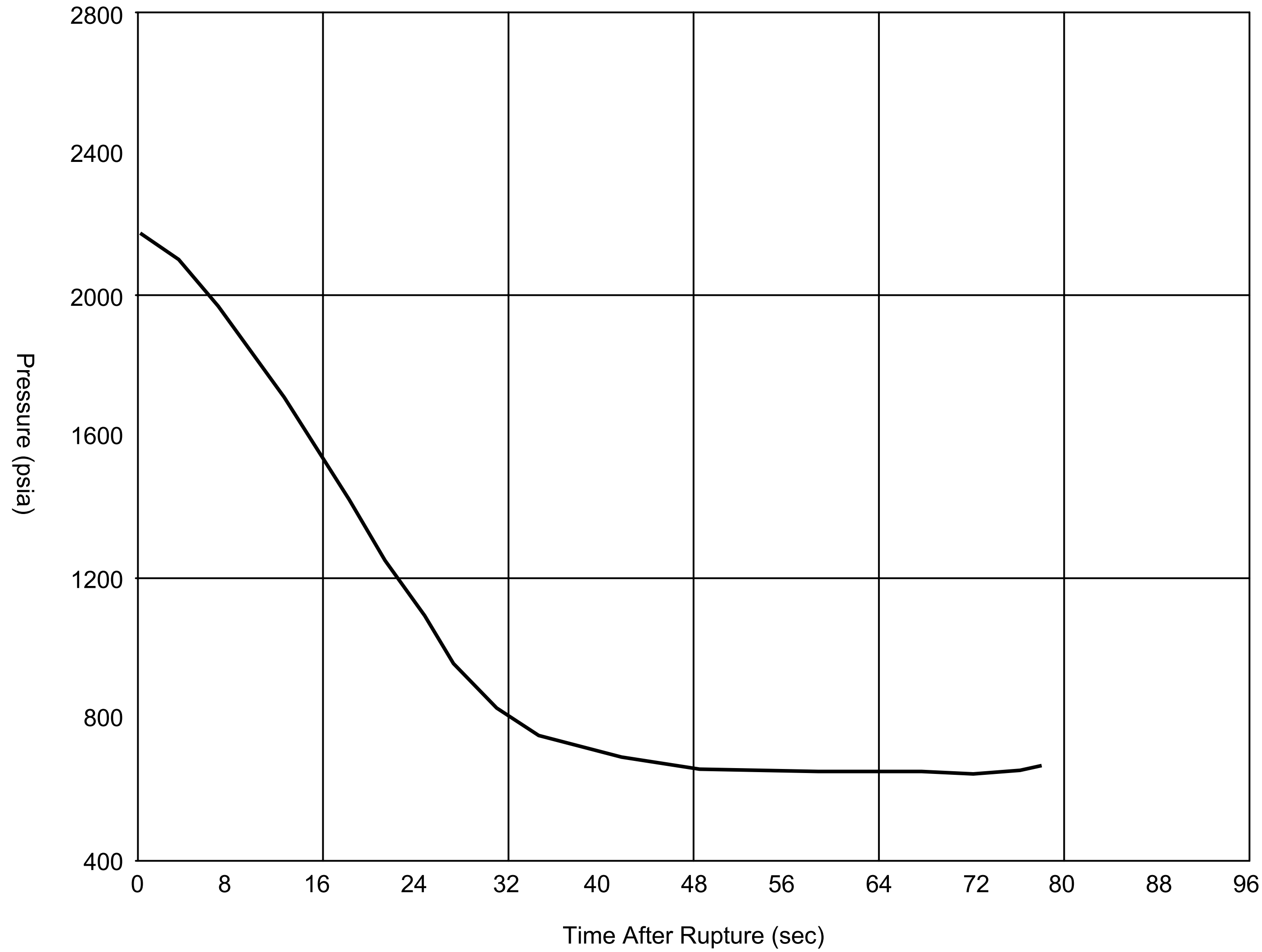
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-20

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
0.65% ΔK/K ROD DROP FROM RATED POWER AT EOL CONDITIONS	BASED ON DRAWING NO	SHEET / REV.



STEAM LINE BREAK TRAP ANALYSIS PRESSURIZER PRESSURES

SAR FIGURE NO. 14-21A

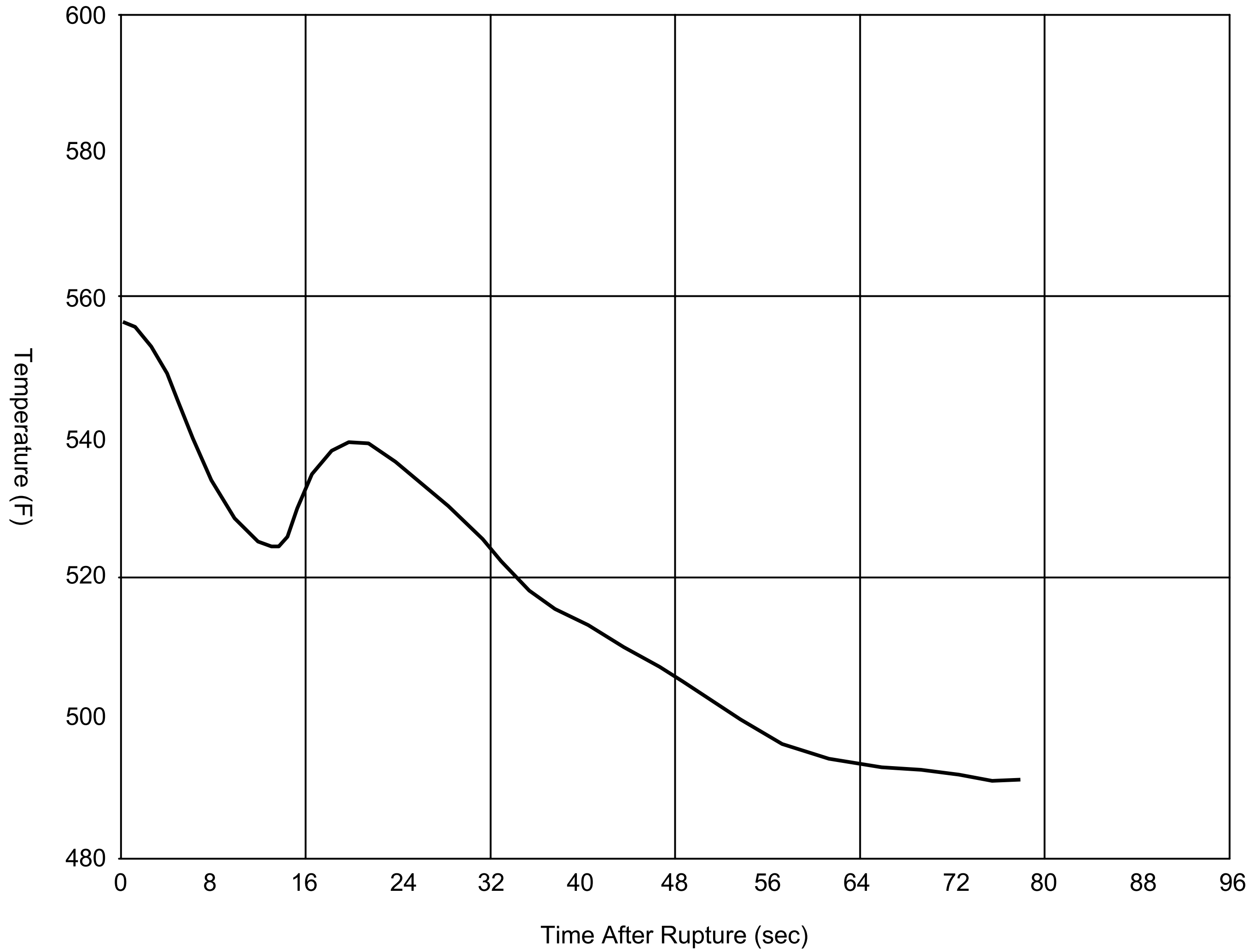
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE: NONE
DRAWN: ENTERGY
DESIGN: ENTERGY
CAD NO:

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



STEAM LINE BREAK TRAP ANALYSIS COLD LEG TEMPERATURE LOOP A

SAR FIGURE NO. 14-21B

AMENDMENT 20

ARKANSAS NUCLEAR ONE

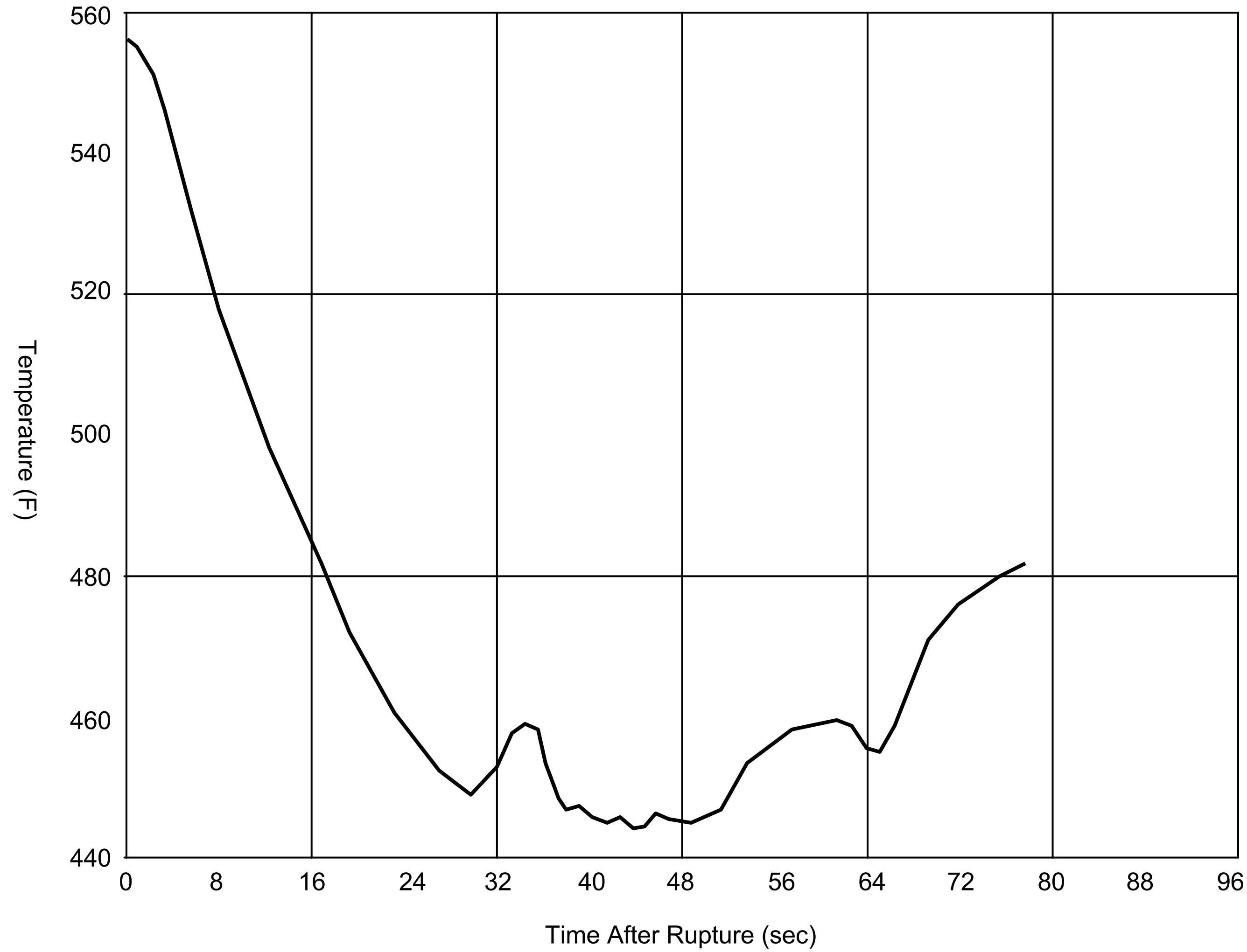
UNIT 1

RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



STEAM LINE BREAK TRAP ANALYSIS COLD LEG TEMPERATURE LOOP B

SAR FIGURE NO. 14-21C

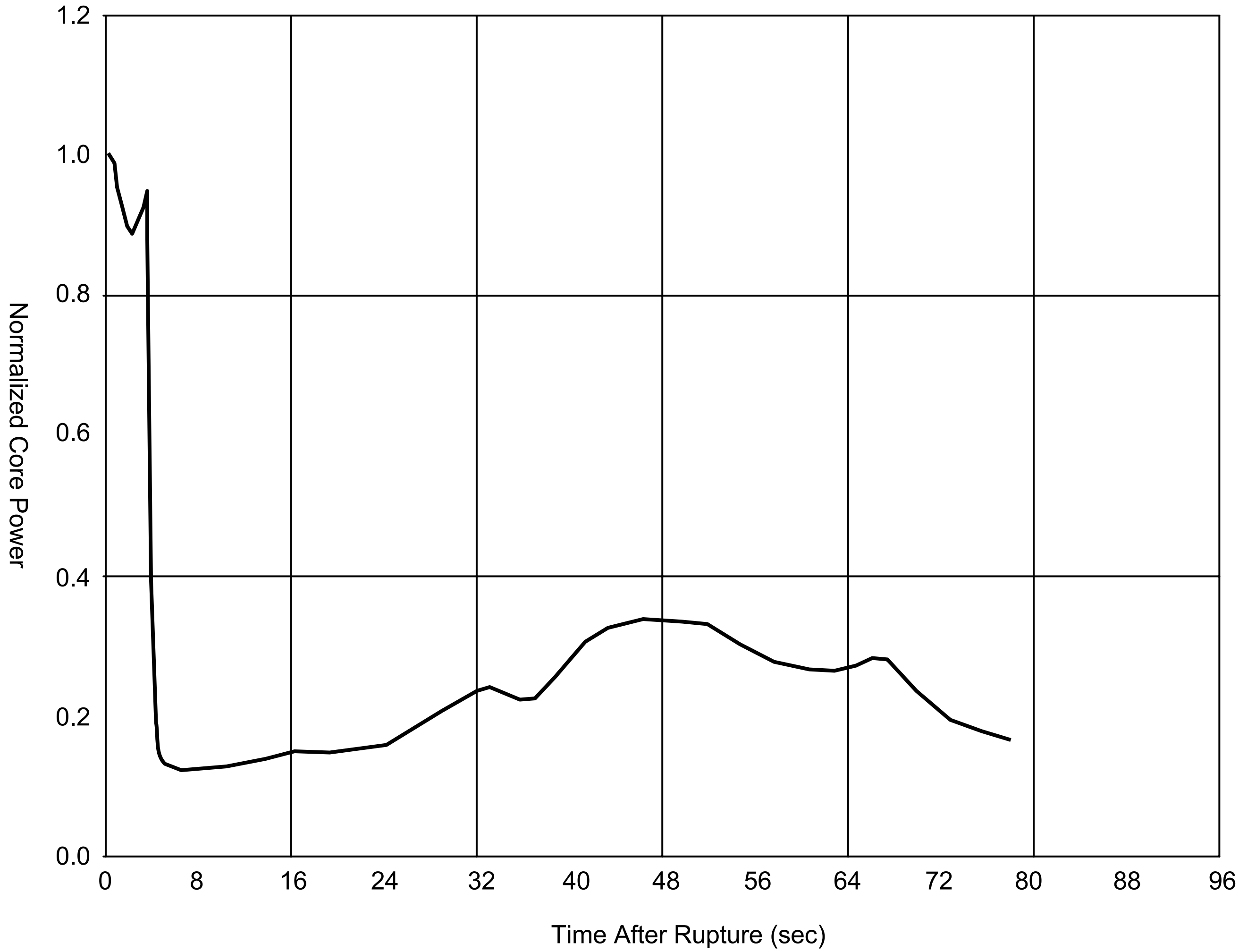
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



STEAM LINE BREAK TRAP ANALYSIS CORE POWER

SAR FIGURE NO. 14-21D

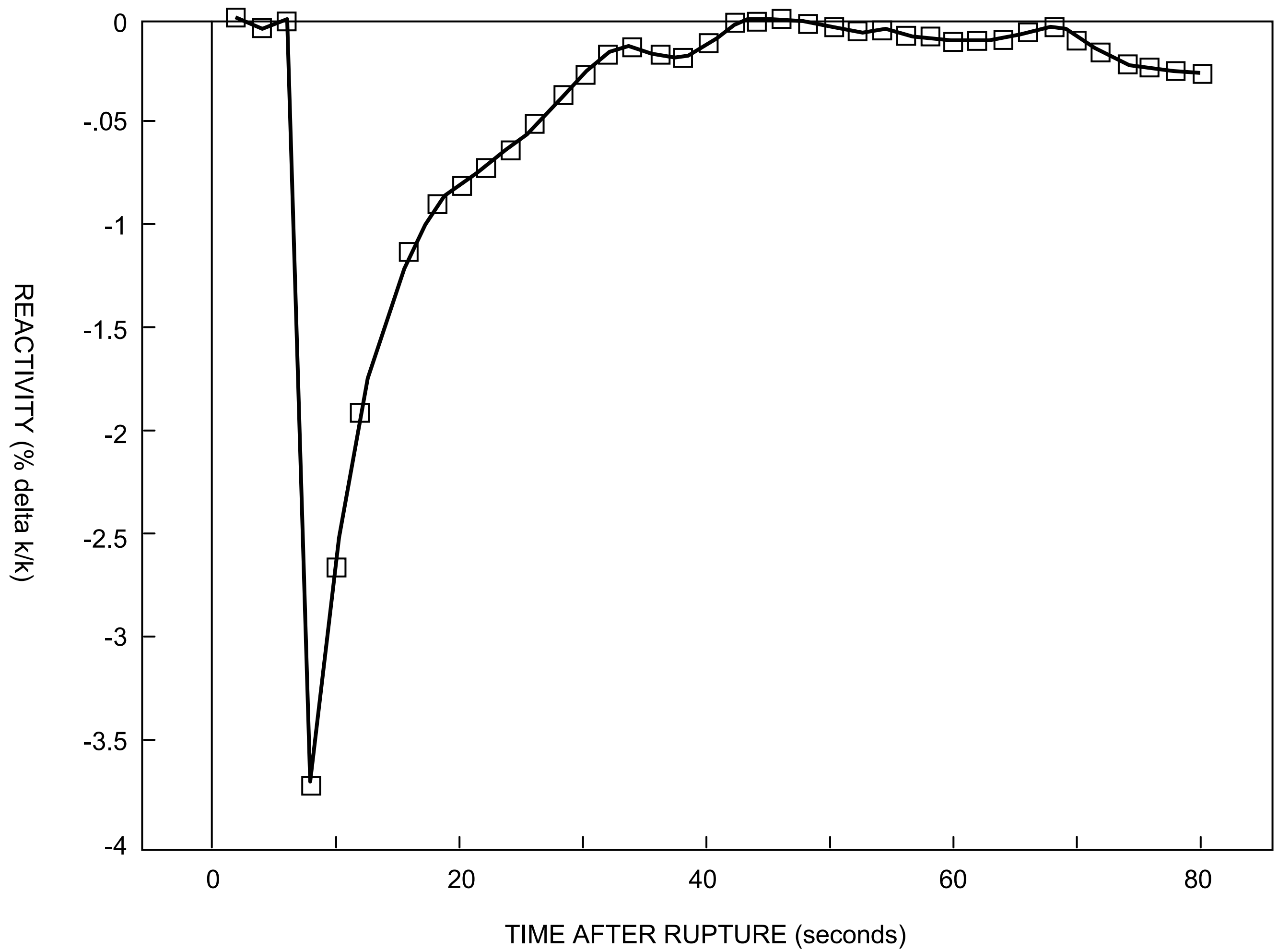
AMENDMENT 15

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



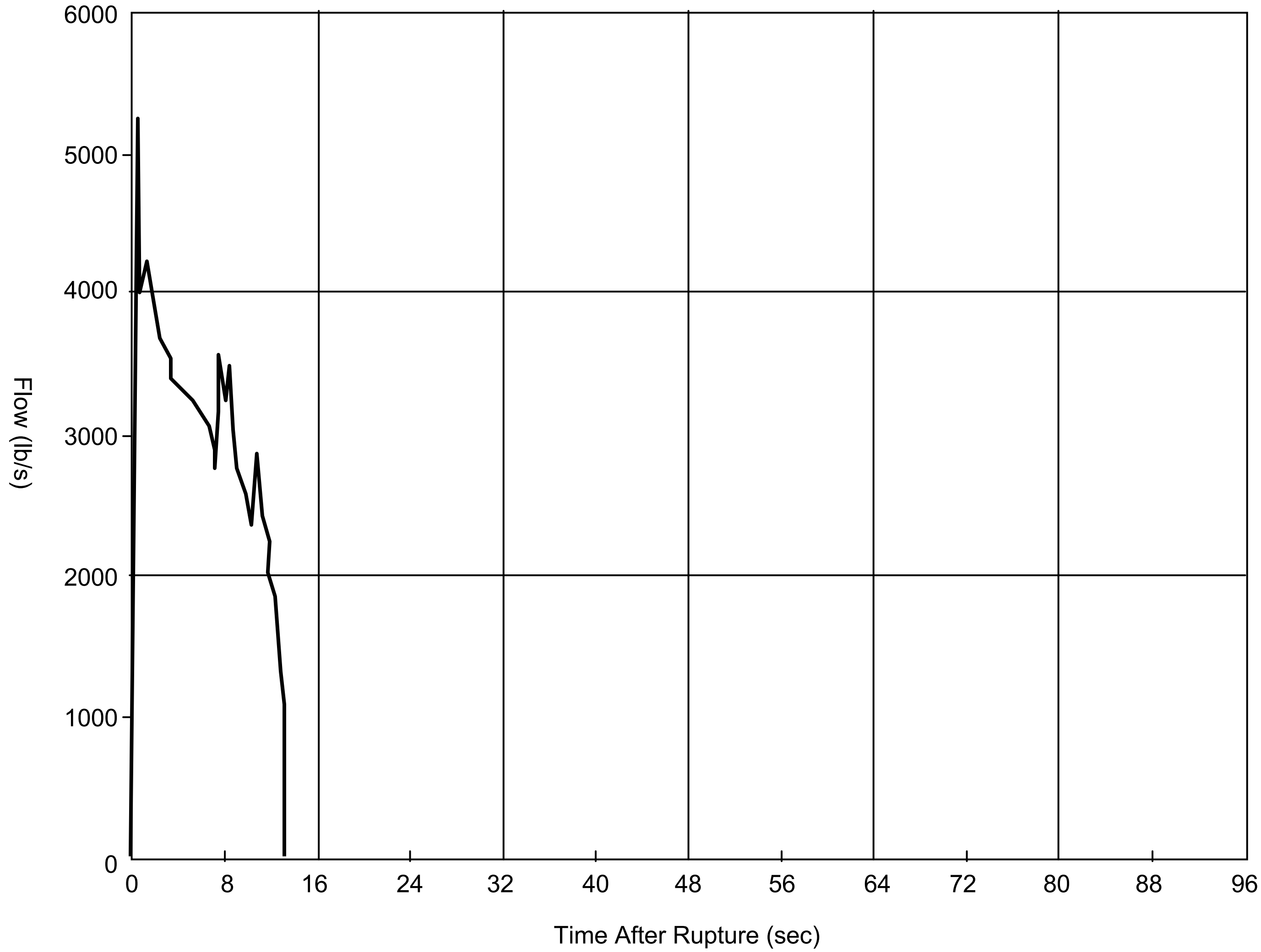
STEAM LINE BREAK TRAP ANALYSIS REACTIVITY VS TIME AFTER RUPTURE	SAR FIGURE NO. 14-21E
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



STEAM LINE BREAK TRAP ANALYSIS FLOW OUT OF BREAK FROM SG A

SAR FIGURE NO. 14-21F

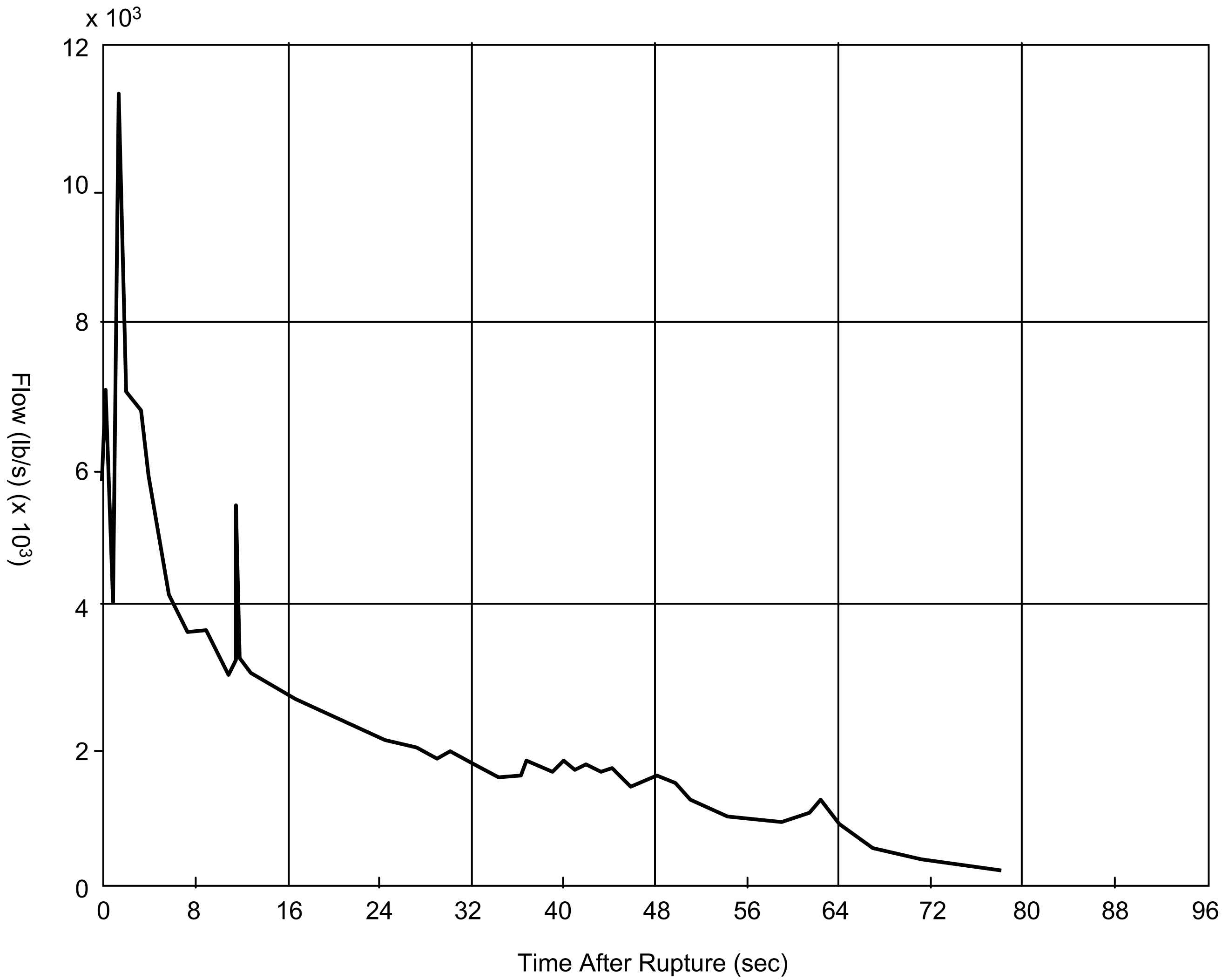
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



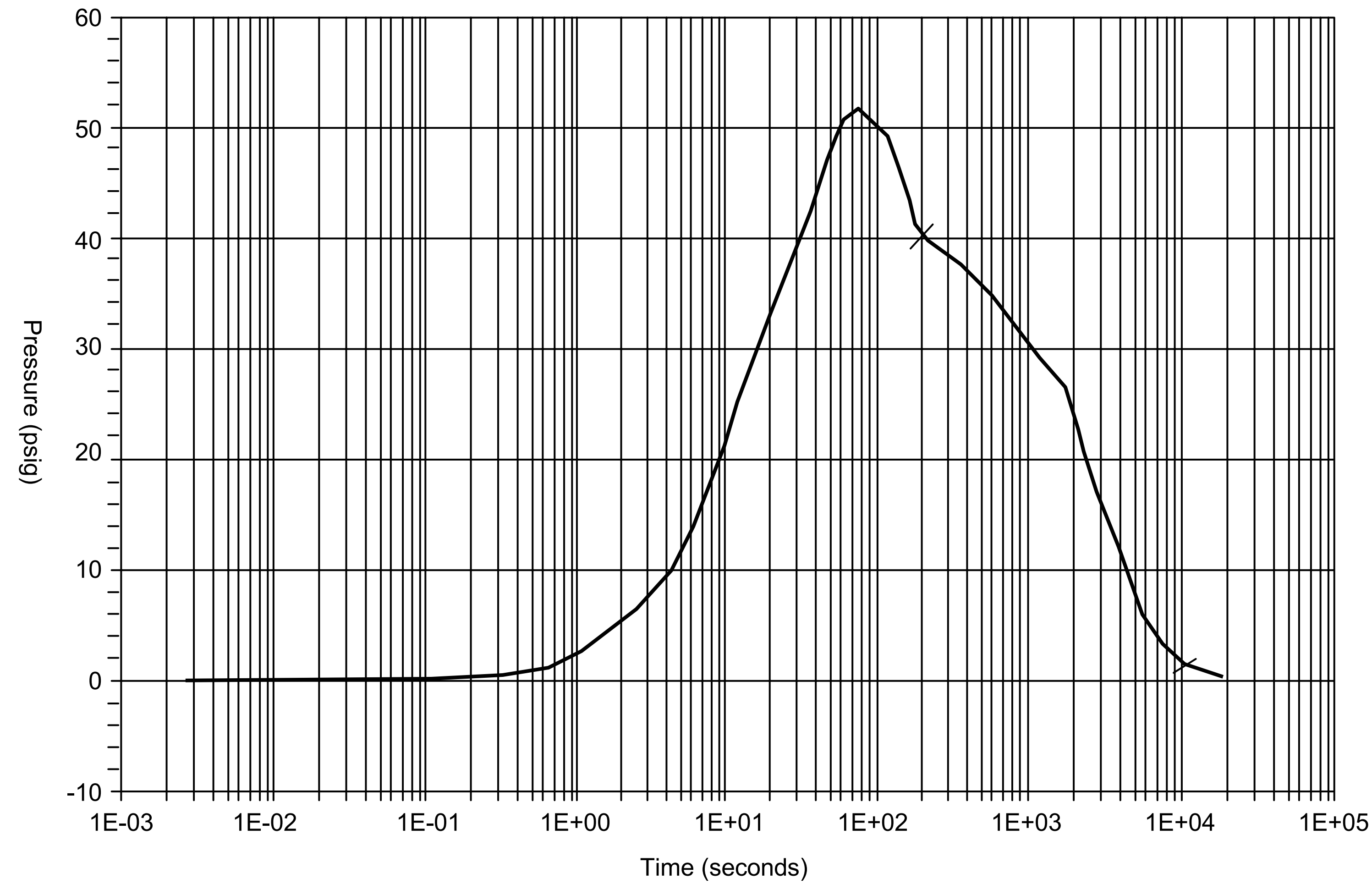
STEAM LINE BREAK TRAP ANALYSIS FLOW OUT OF BREAK FROM SG B	SAR FIGURE NO. 14-21G
	AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



MSLB CONTAINMENT PRESSURE PROFILE

SAR FIGURE NO. 14-21H

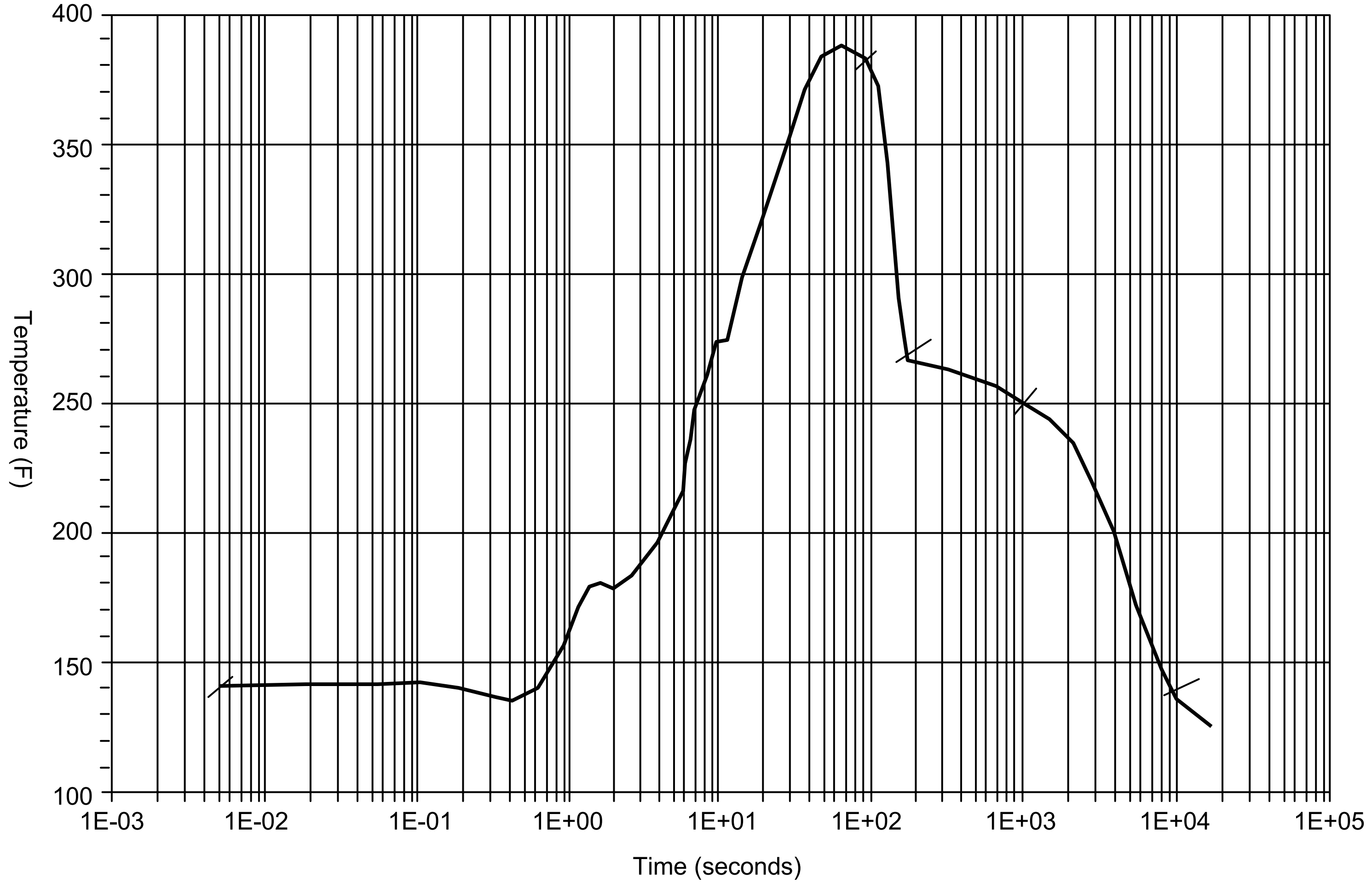
AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



MSLB CONTAINMENT VAPOR TEMPERATURE PROFILE

SAR FIGURE NO. 14-21I

AMENDMENT 20

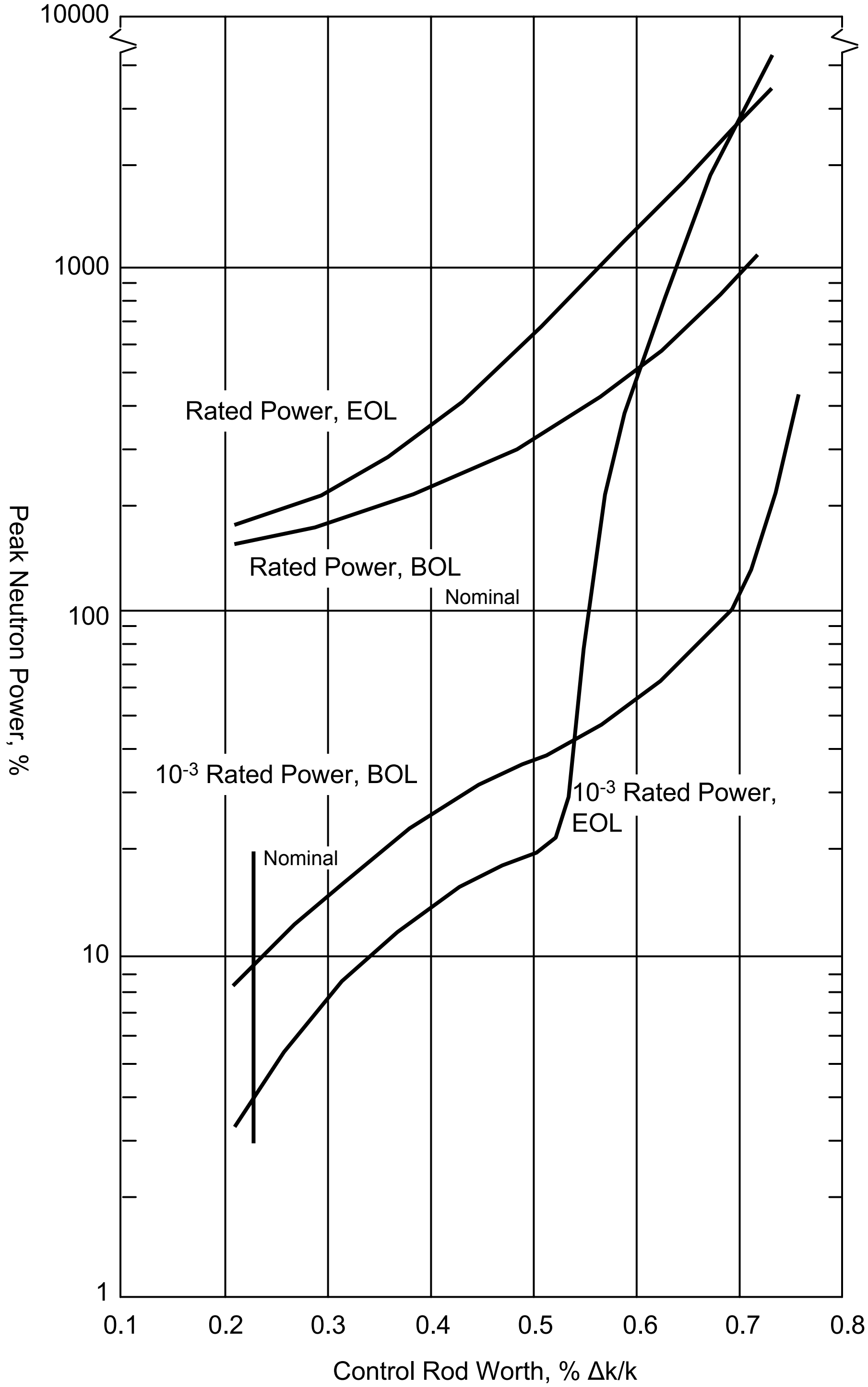
ARKANSAS NUCLEAR ONE

UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

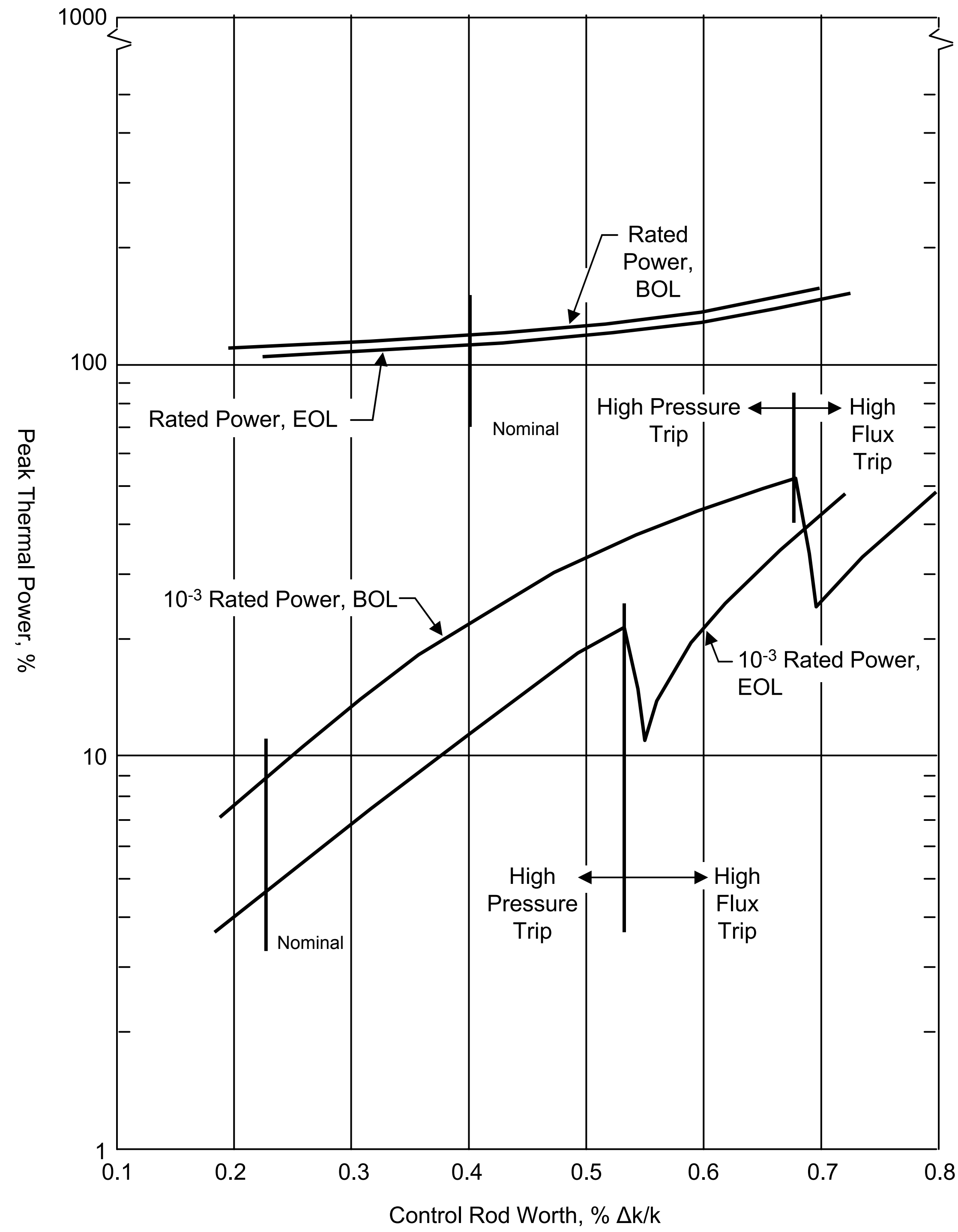


SAR FIGURE NO. 14-22

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

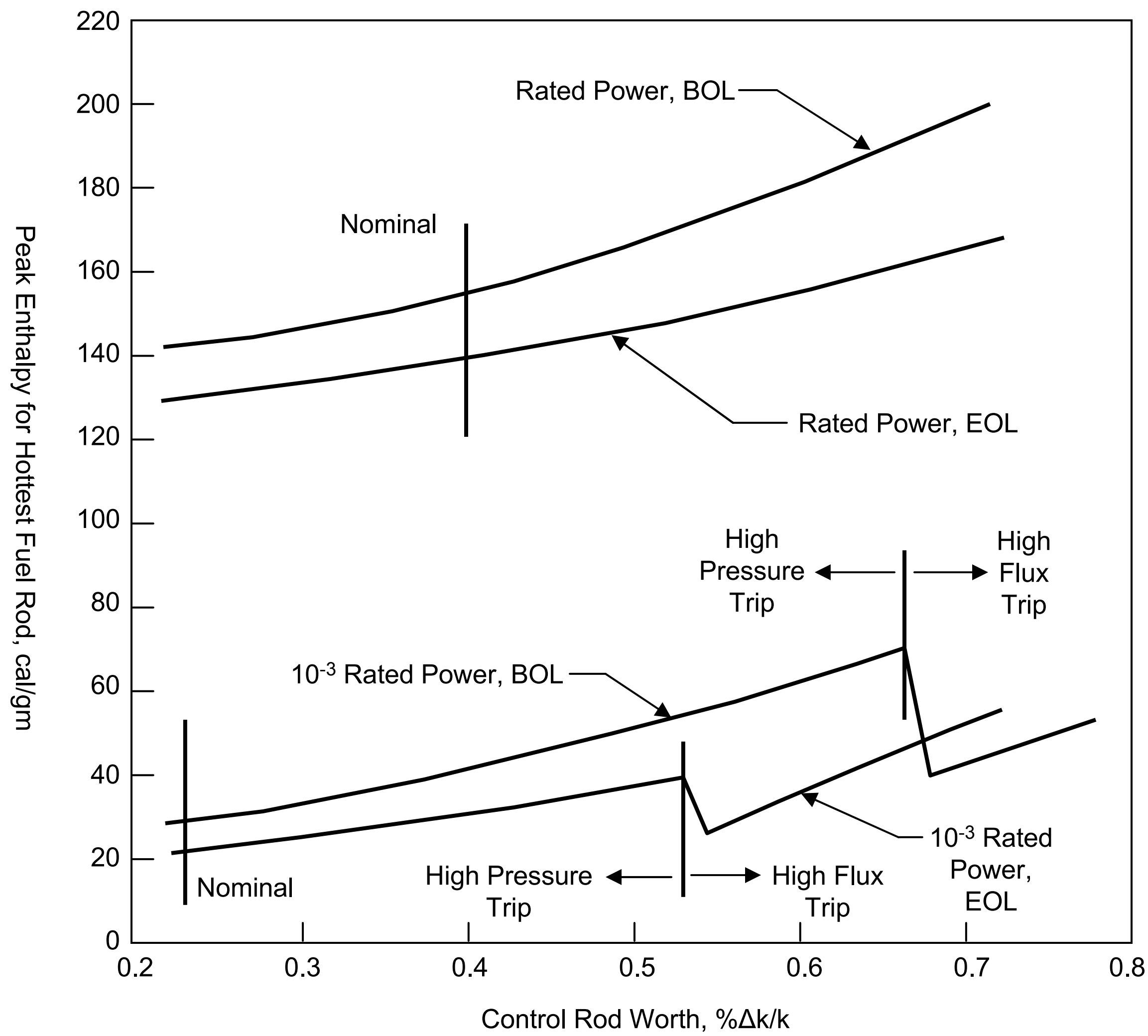
PEAK NEUTRON POWER VARIATION WITH EJECTED CONTROL ROD WORTH	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-23

AMENDMENT 20

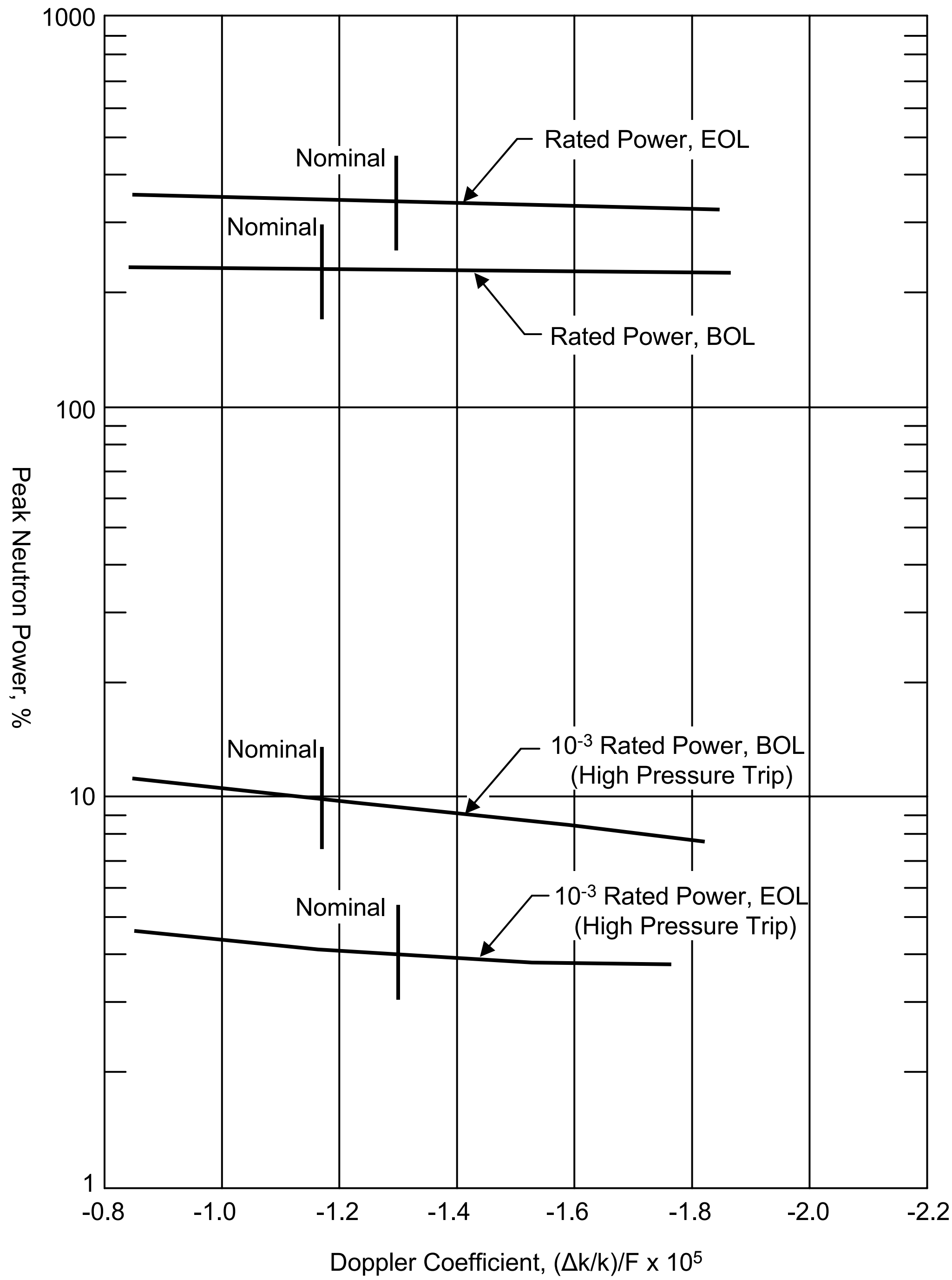
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
PEAK THERMAL POWER AS A FUNCTION OF EJECTED CONTROL ROD WORTH	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-24

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
PEAK ENTHALPY OF HOTTEST FUEL ROD VERSUS EJECTED CONTROL ROD WORTH	BASED ON DRAWING NO		SHEET	REV.



SAR FIGURE NO. 14-25

AMENDMENT 20

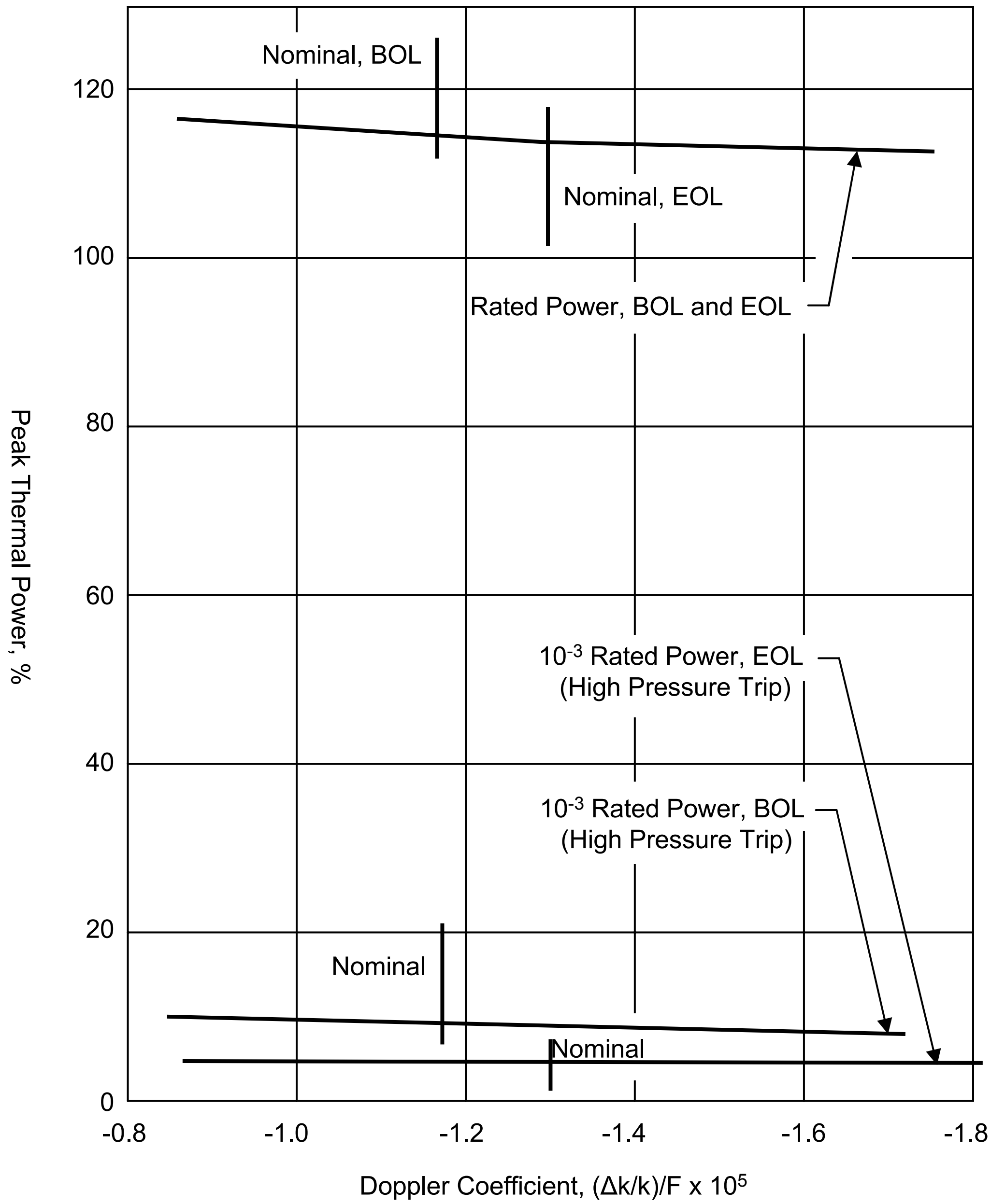
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

EFFECT ON PEAK NEUTRON POWER OF VARYING THE DOPPLER COEFF. FOR AN EJECTED ROD WORTH OF 0.23% $\Delta K/K$ AT 10^{-3} RATED POWER AND 0.40% $\Delta K/K$ AT RATED POWER

BASED ON DRAWING NO	SHEET	REV.

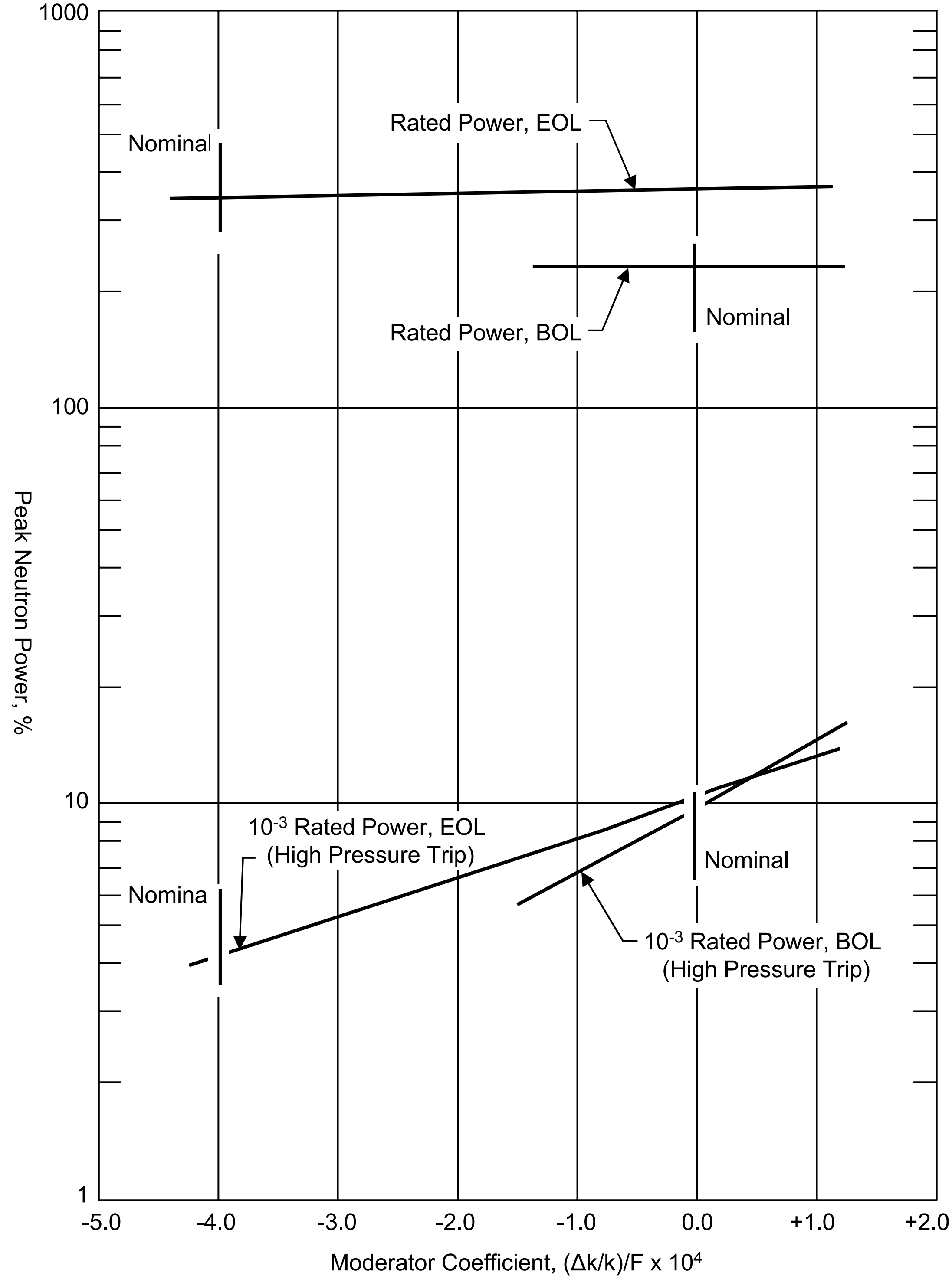


SAR FIGURE NO. 14-26

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

EFFECT ON PEAK THERMAL POWER OF VARYING THE DOPPLER COEFF. FOR AN EJECTED ROD WORTH OF 0.23% ΔK/K AT 10^{-3} RATED POWER AND 0.40% ΔK/K AT RATED POWER	BASED ON DRAWING NO	SHEET	REV.

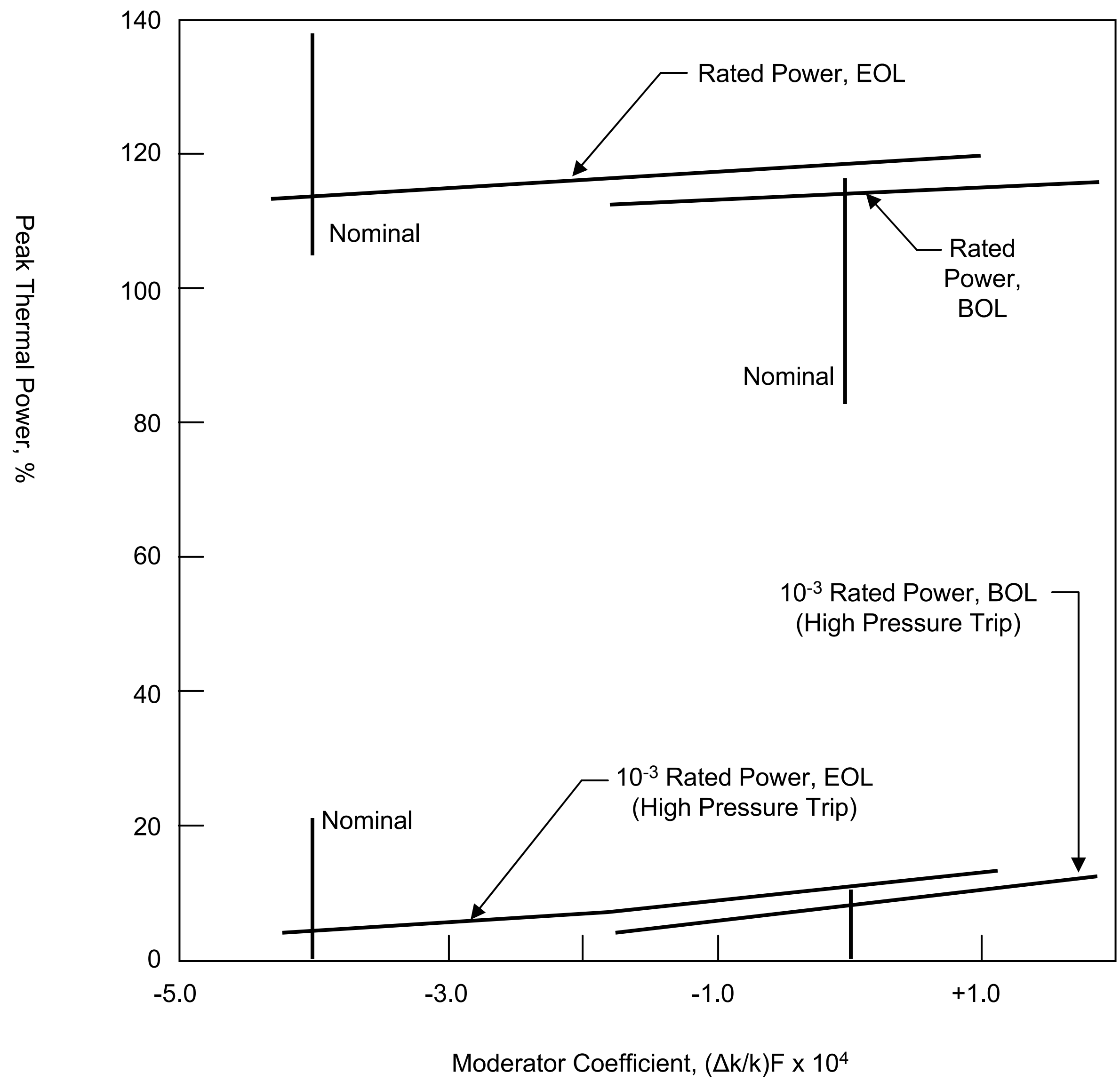


SAR FIGURE NO. 14-27

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

EFFECT ON PEAK NEUTRON POWER OF VARYING THE MODERATOR COEFF. FOR AN EJECTED ROD WORTH OF 0.23% ΔK/K AT 10^{-3} RATED POWER AND 0.40% ΔK/K AT RATED POWER	BASED ON DRAWING NO	SHEET	REV.

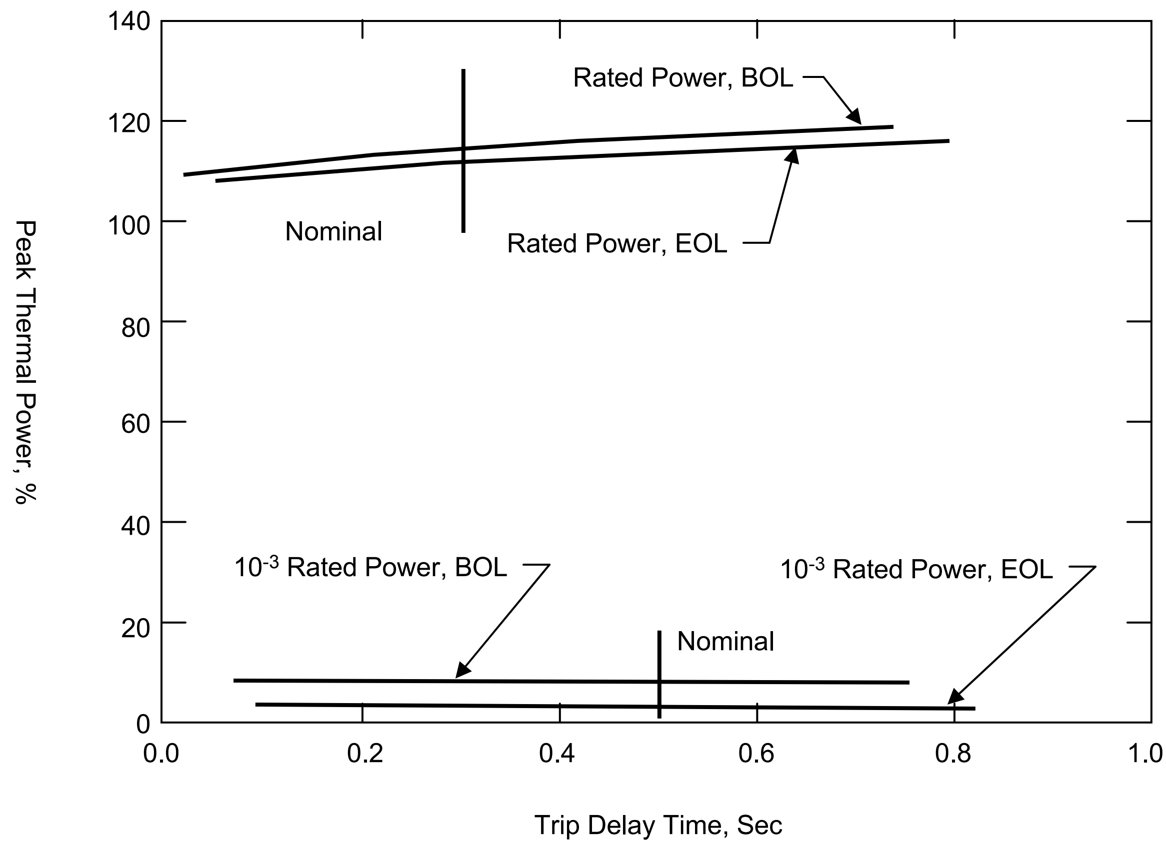


SAR FIGURE NO. 14-28

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

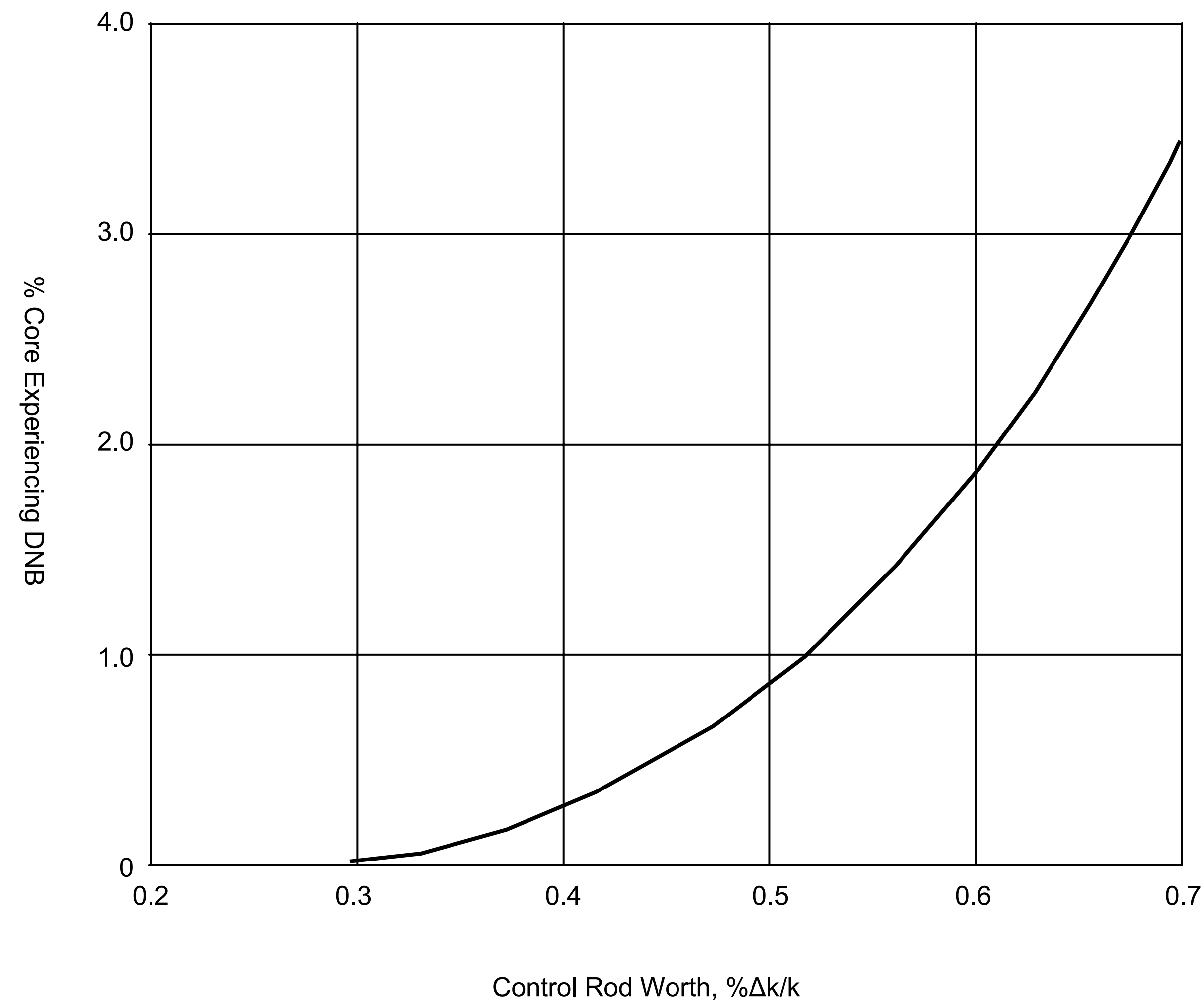
EFFECT ON PEAK THERMAL POWER OF VARYING THE MODERATOR COEFF. FOR AN EJECTED ROD WORTH OF 0.23% ΔK/K AT 10^{-3} RATED POWER AND 0.40% ΔK/K AT RATED POWER	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-29

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
EFFECT ON PEAK THERMAL POWER OF VARYING THE TRIP DELAY TIME FOR AN EJECTED ROD WORTH OF 0.23% ΔK/K AT 10^{-3} RATED POWER AND 0.40% ΔK/K AT RATED POWER	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-30

AMENDMENT 20

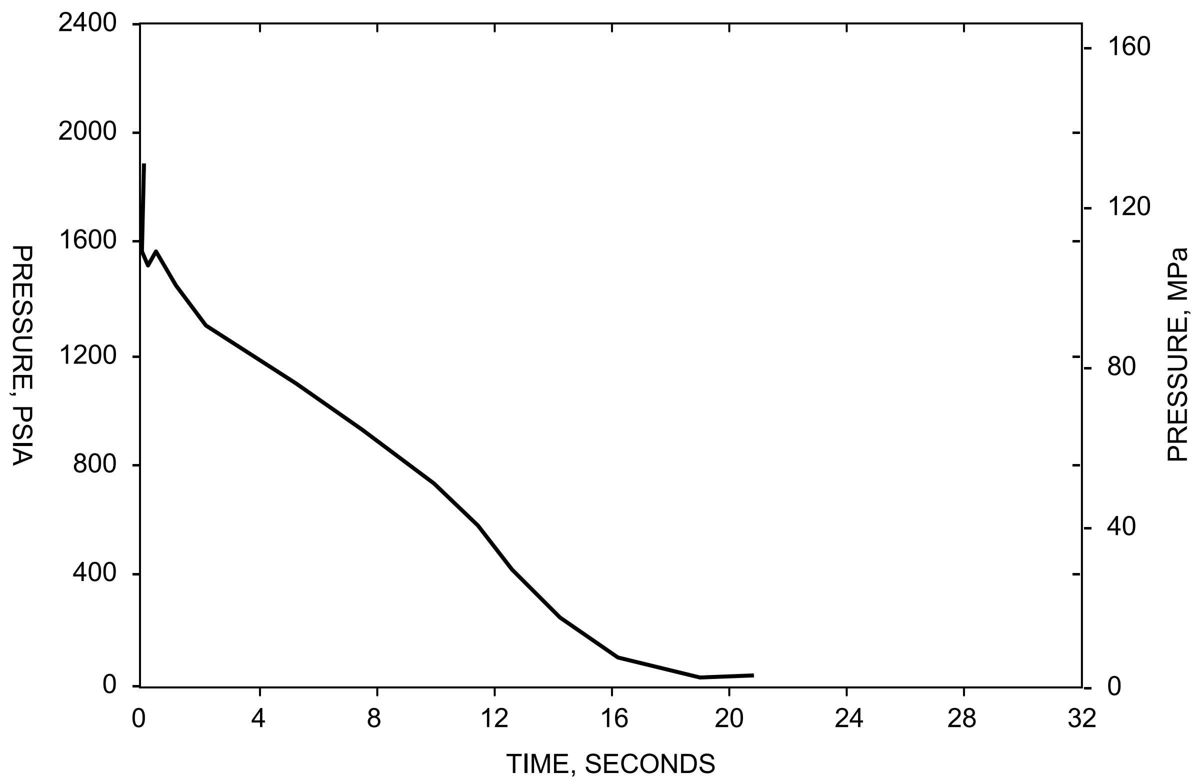
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

EFFECT CORE EXPERIENCING DNB AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT RATED POWER, BOL

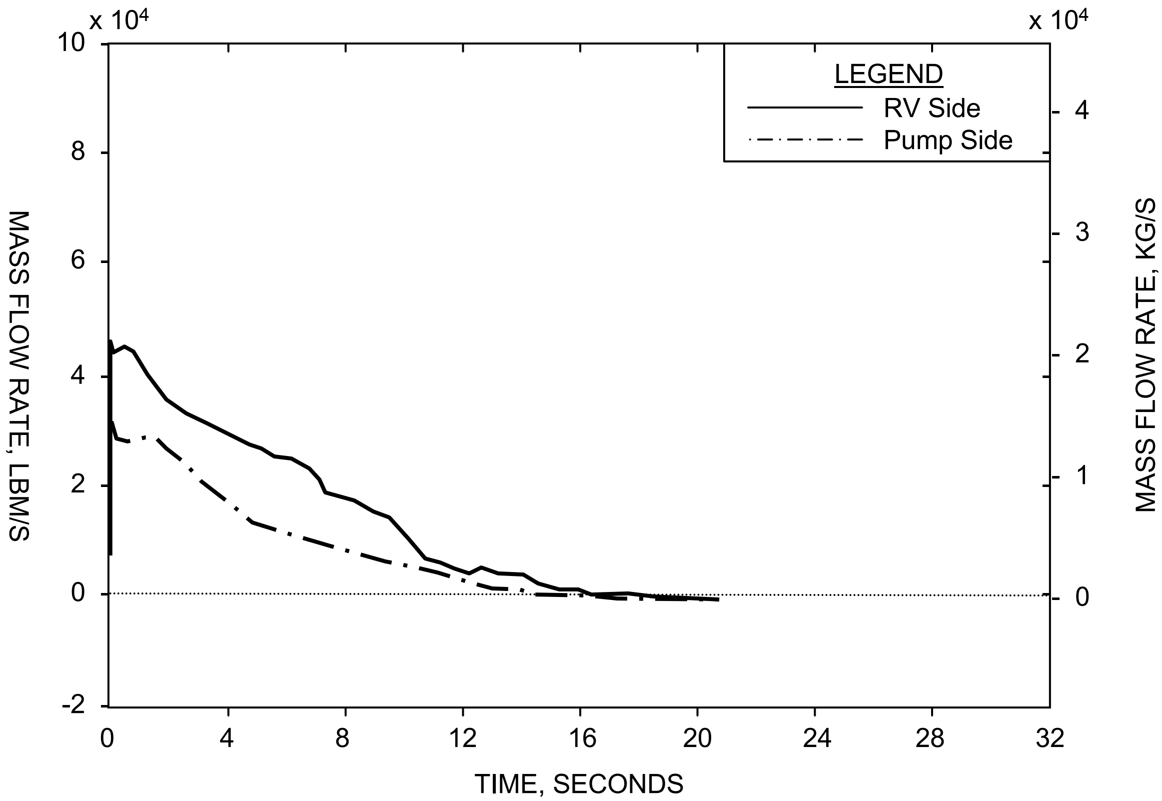
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-31

AMENDMENT 20

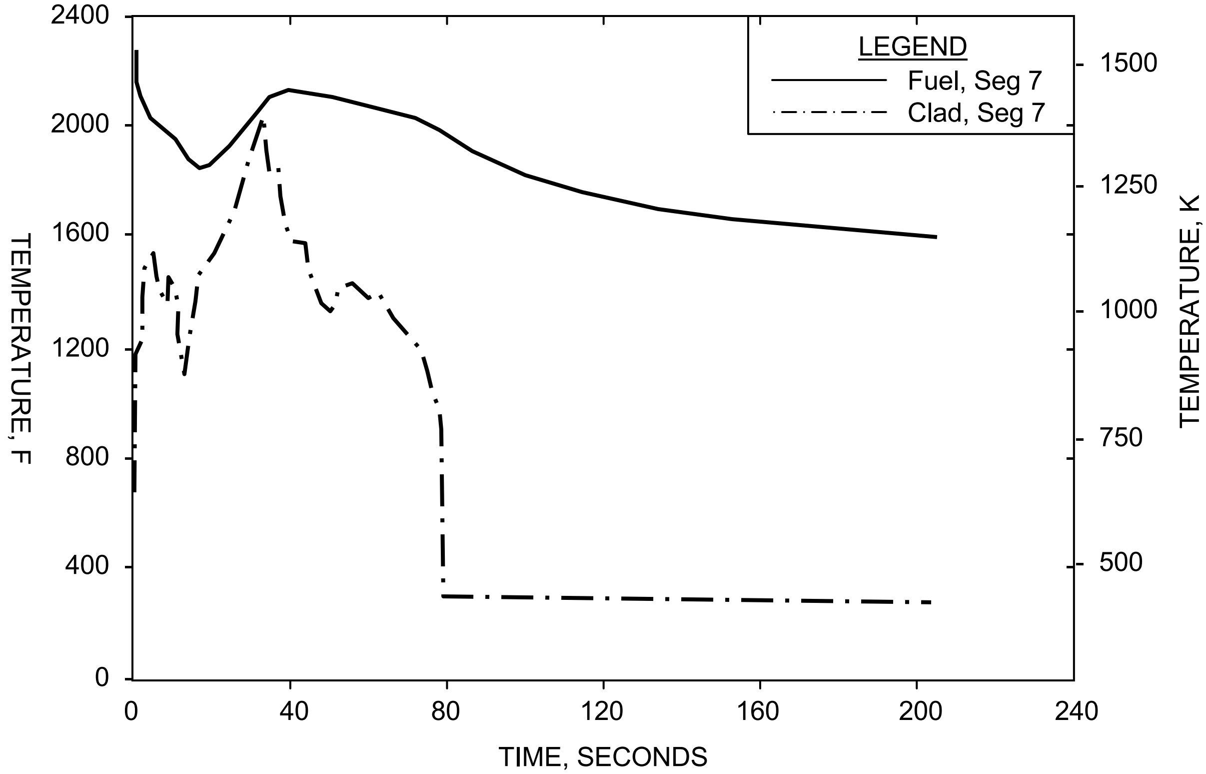
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
LBLOCA – 8.55 FT^2 CLPD BREAK REACTOR VESSEL UPPER PLENUM PRESSURE	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-32

AMENDMENT 20

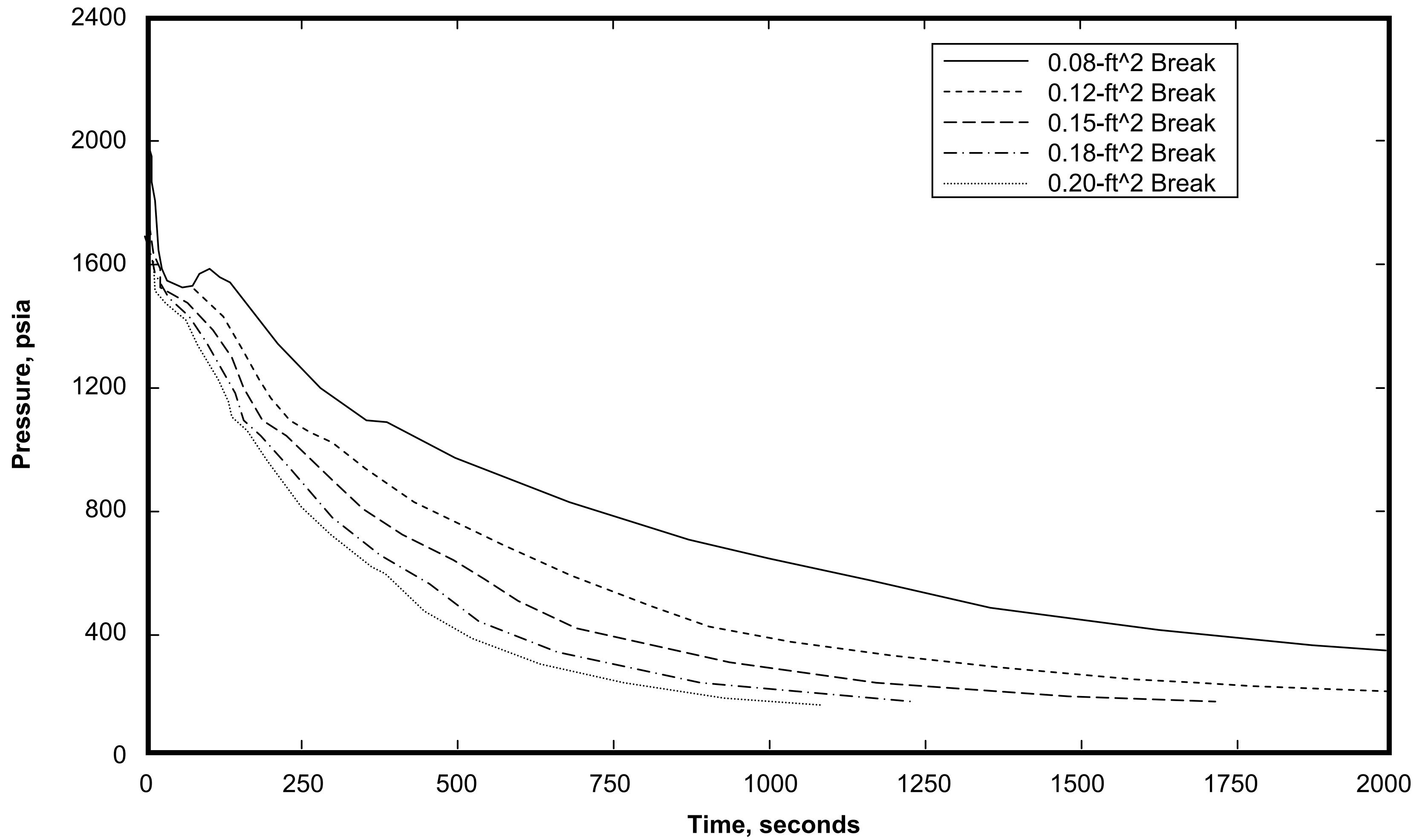
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
LBLOCA – 8.55 FT^2 CLPD BREAK MASS FLOW RATES	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-33

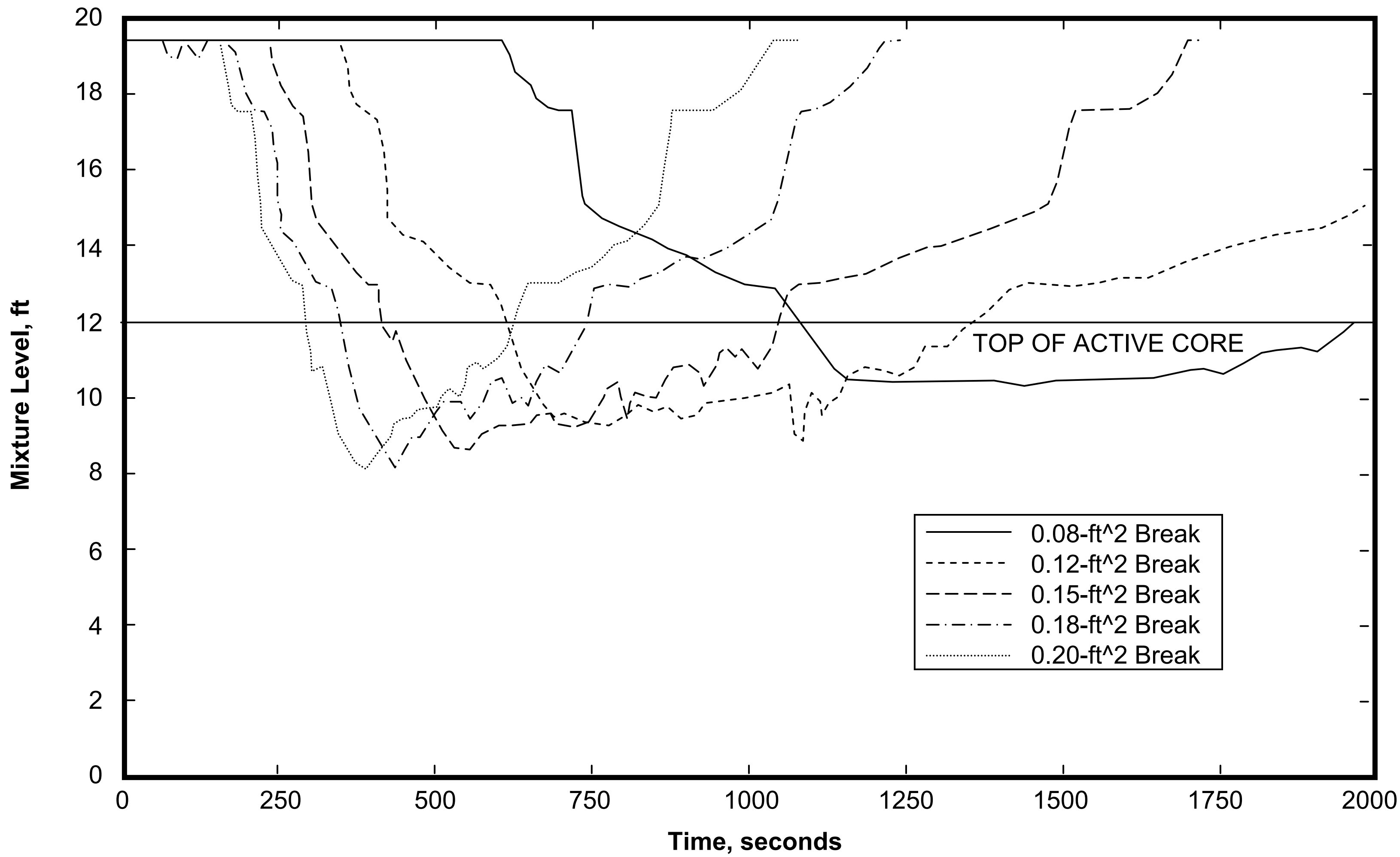
AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
LBLOCA – 8.55 FT^2 CLPD BREAK LIMITING HC FUEL & CLAD TEMPERATURES	BASED ON DRAWING NO	SHEET	REV.



ANO-1 MK-B-HTP MIXED CORE WITH EOTSG SBLOCA, CATEGORY 4 BREAK SIZES: PRESSURE

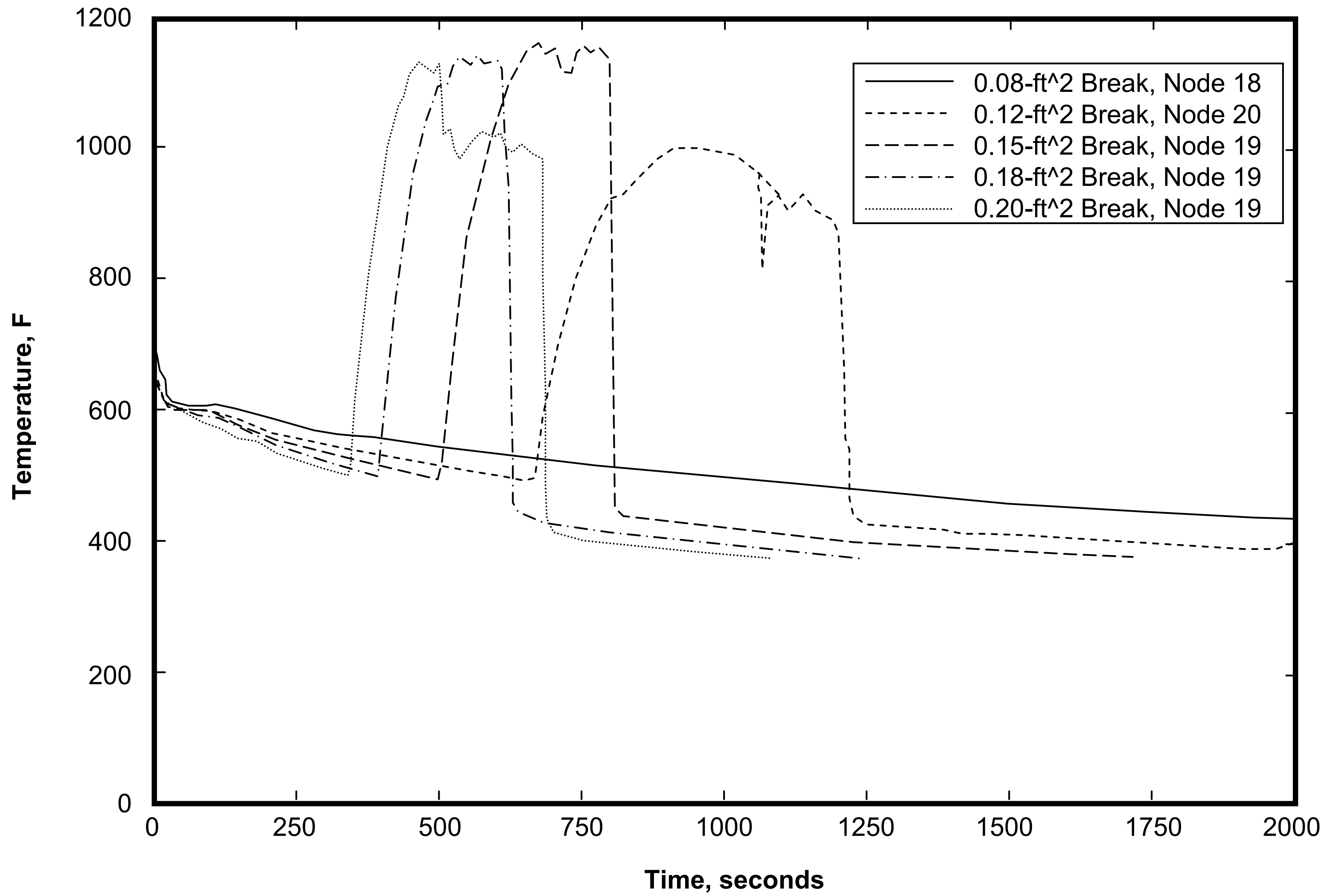
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	SAR FIGURE NO. 14-34 AMENDMENT 20 BASED ON DRAWING NO / SHEET / REV.



ANO-1 MK-B-HTP MIXED CORE WITH EOTSG SBLOCA, CATEGORY 4 BREAK SIZES:
CORE MIXTURE LEVEL

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:	SAR FIGURE NO. 14-35 AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.



ANO-1 MK-B-HTP MIXED CORE WITH EOTSG SBLOCA, CATEGORY 4 BREAK SIZES:
HOT CHANNEL PCT COMPARISON

SAR FIGURE NO. 14-36

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

FIGURES 14-37 THROUGH 14-57

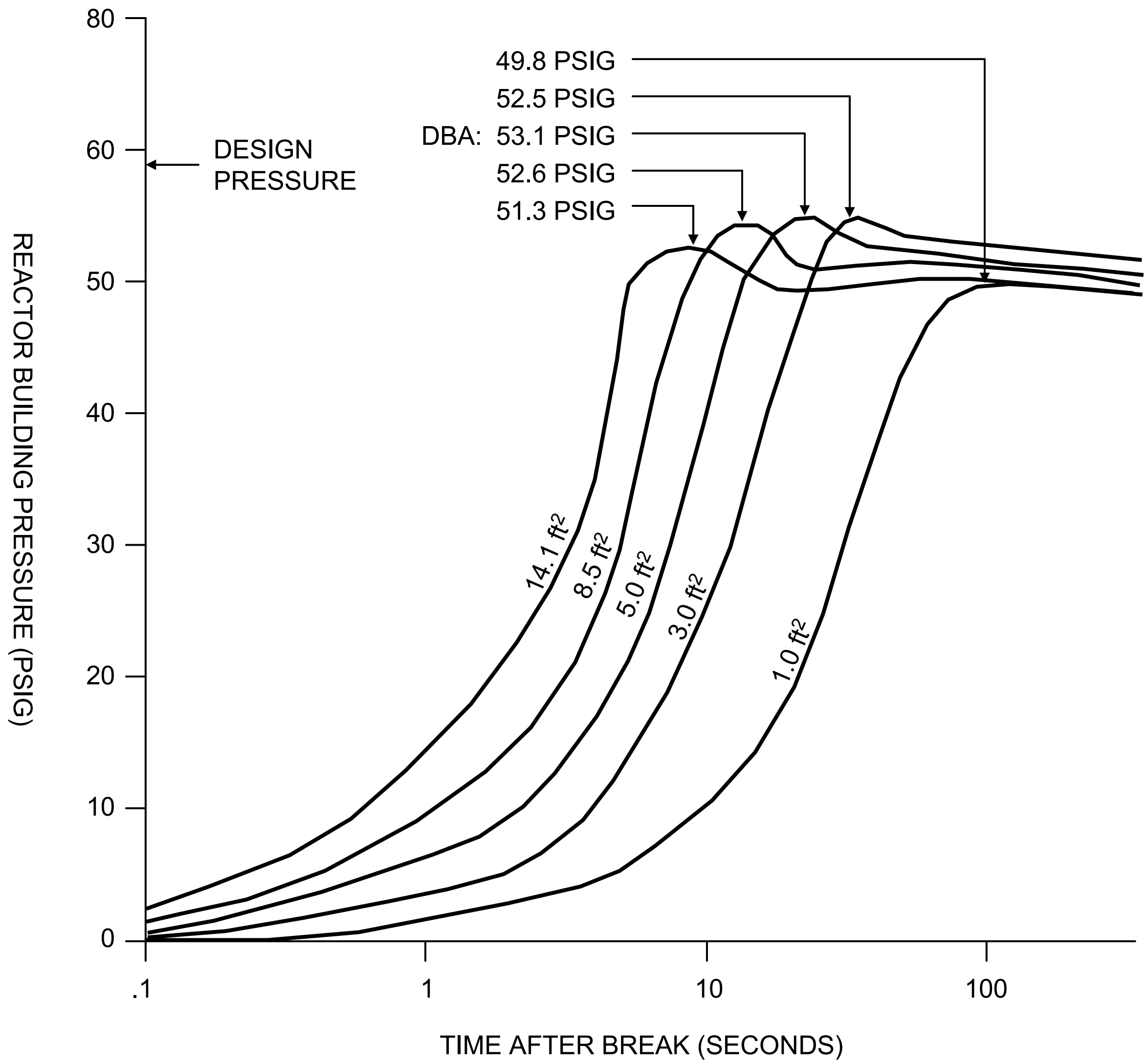
DELETED

SAR FIGURE NO. 14-37

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

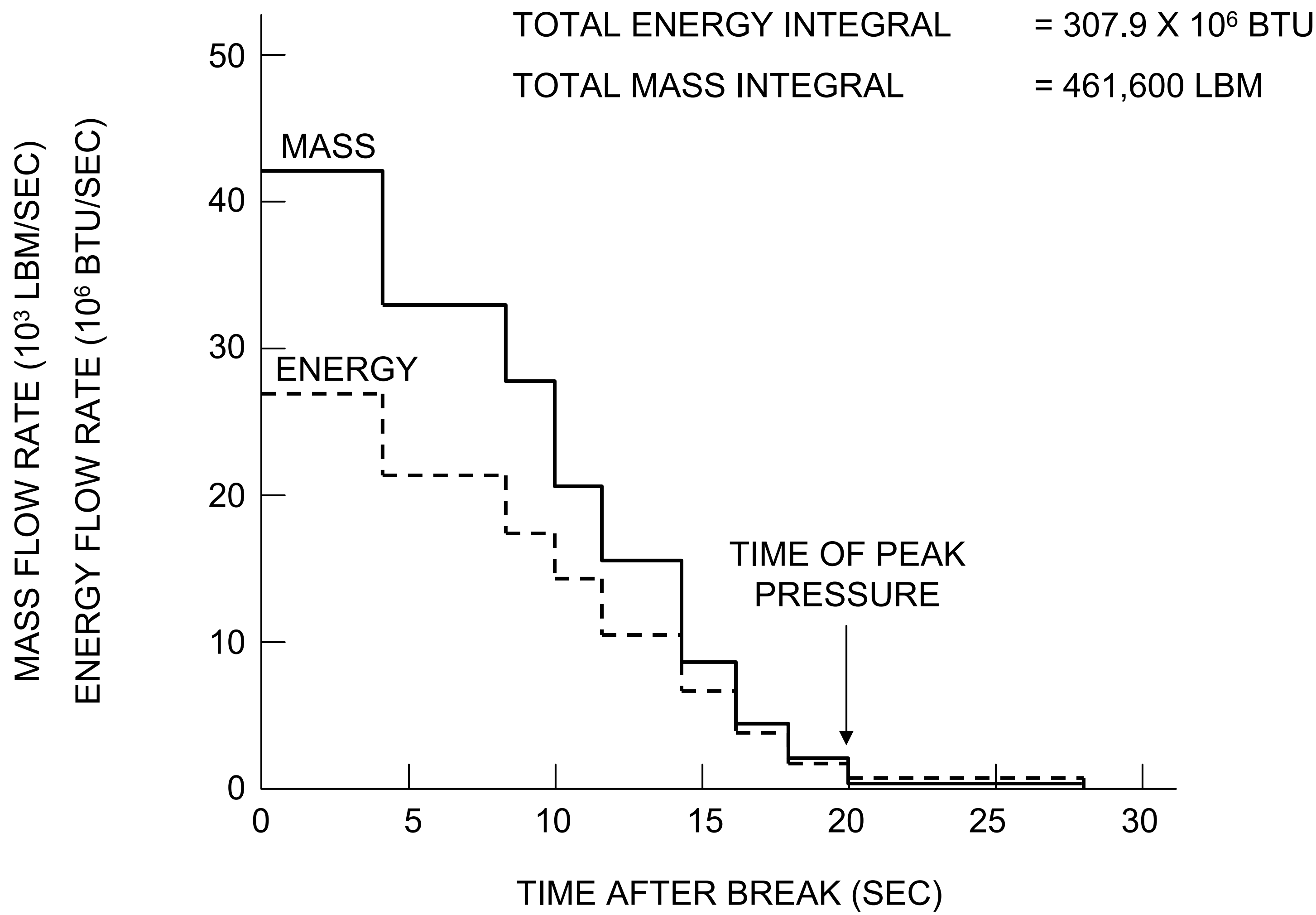
	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-58

AMENDMENT 20

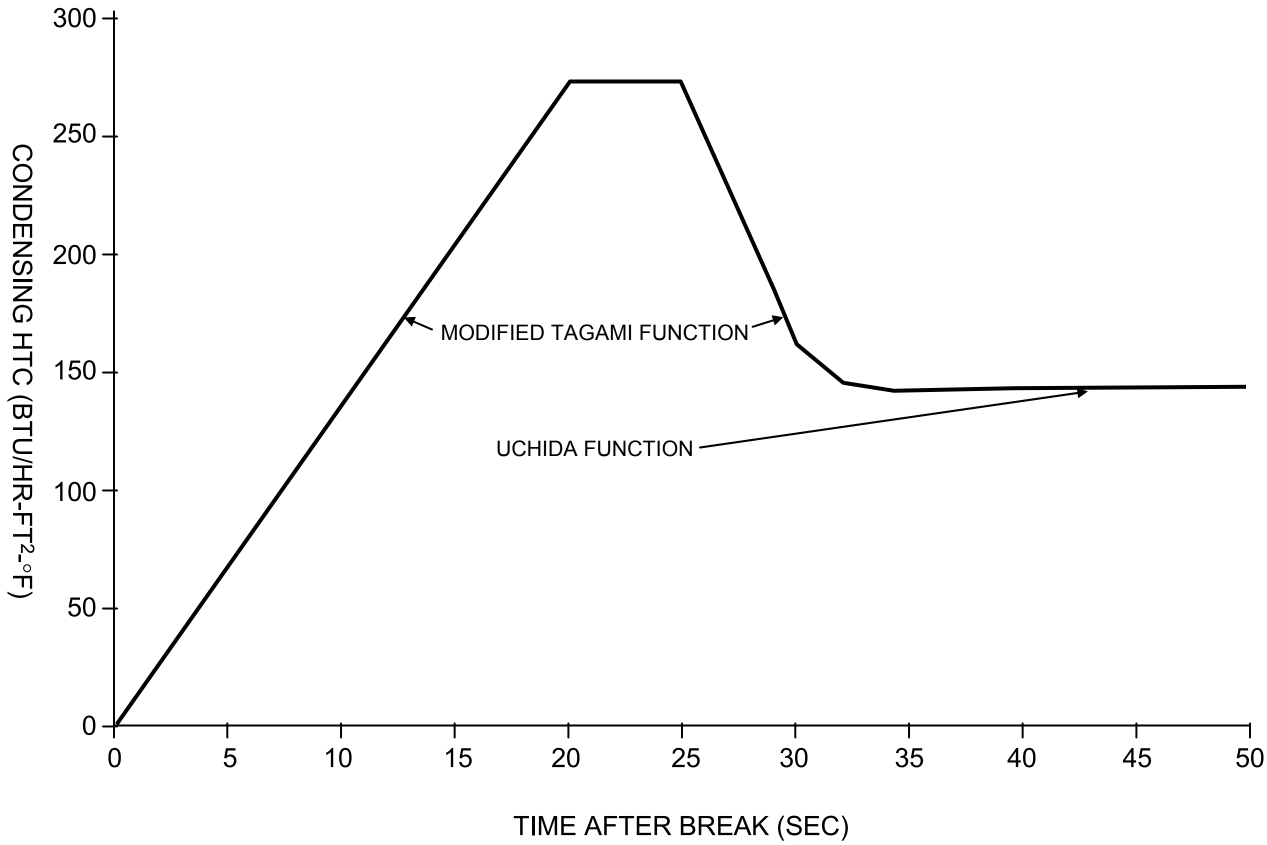
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ORIGINAL REACTOR BUILDING PRESSURE TRANSIENT FOR THE SPECTRUM OF BREAK SIZES	BASED ON DRAWING NO	SHEET / REV.



PRIMARY SYSTEM MASS AND ENERGY RELEASE FOR THE REACTOR BLDG DBA (5 FT^2)	SAR FIGURE NO. 14-59
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	AMENDMENT 20

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.

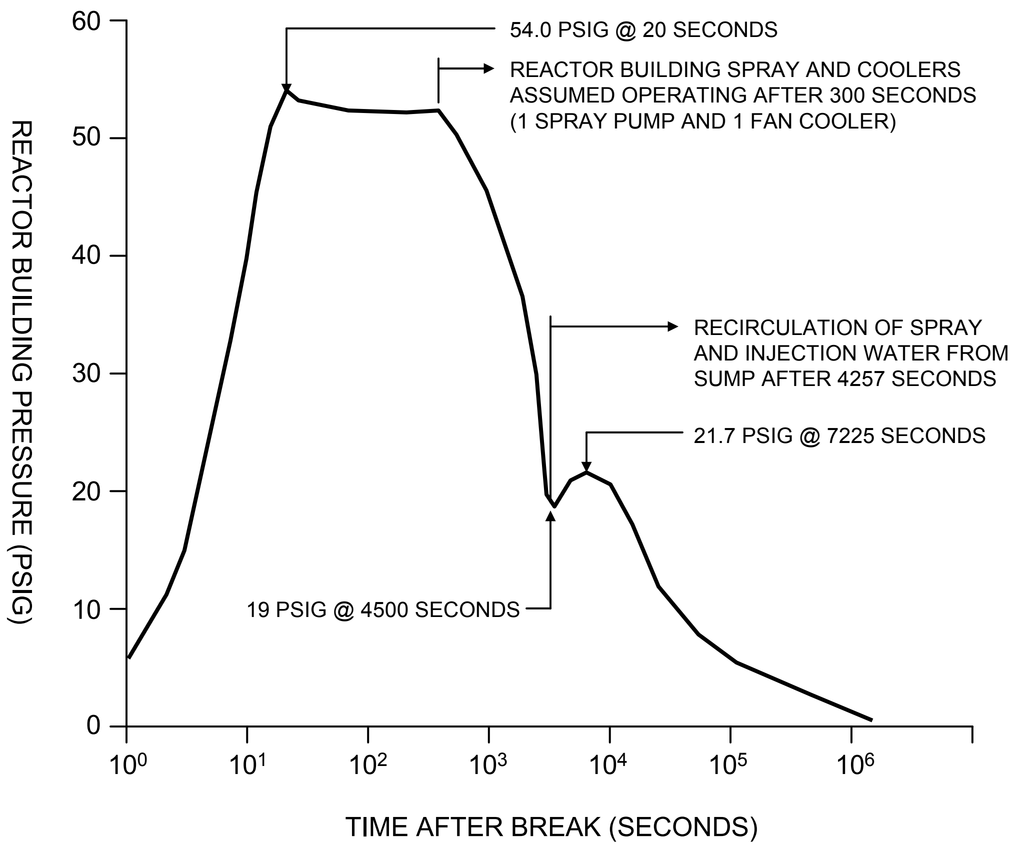


SAR FIGURE NO. 14-60

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

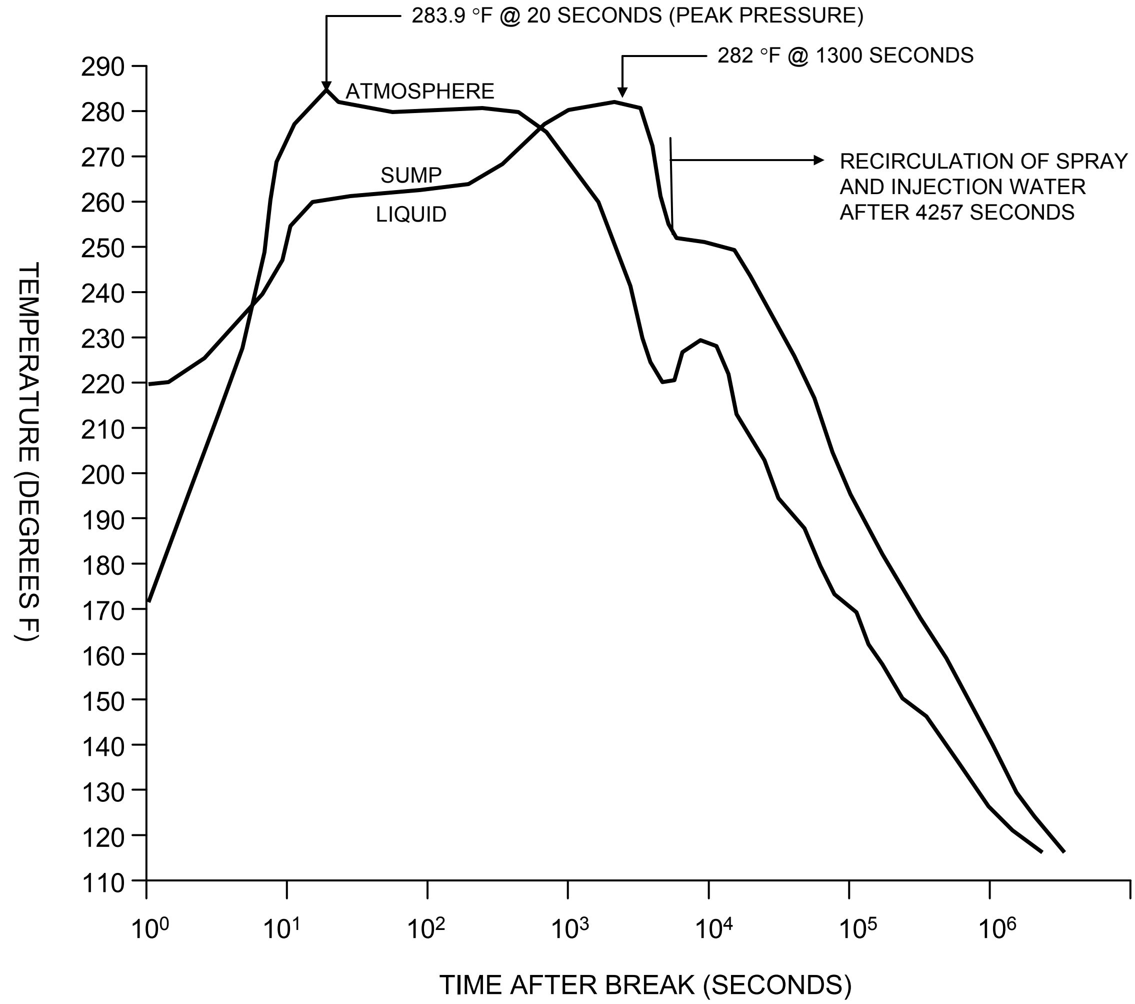
CONTAINMENT CONDENSING HEAT TRANSFER COEFFICIENT FOR THE REACTOR BUILDING DBA (5 FT²)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-61

AMENDMENT 20

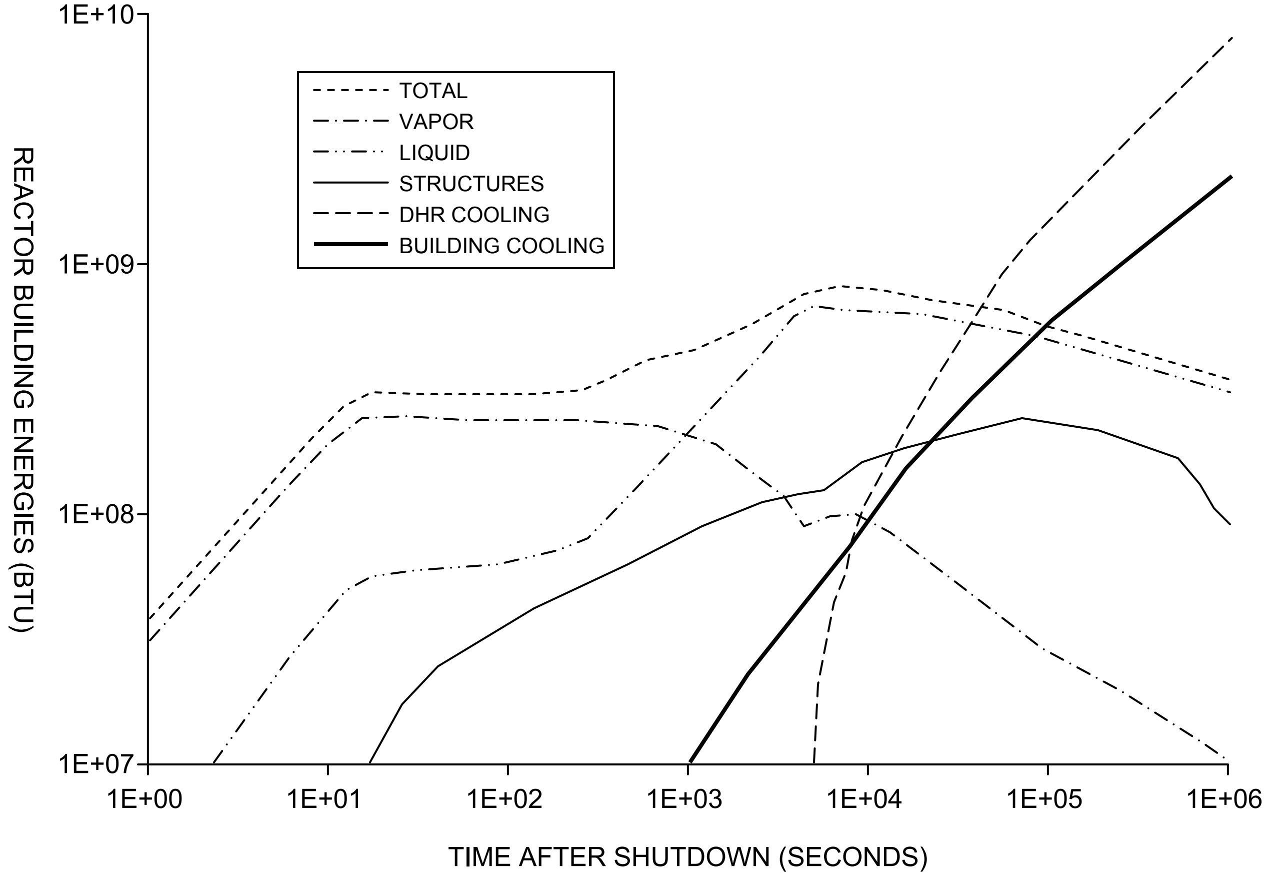
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING PRESSURE TRANSIENT FOR THE REACTOR BUILDING DBA (5 FT^2)	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-62

AMENDMENT 20

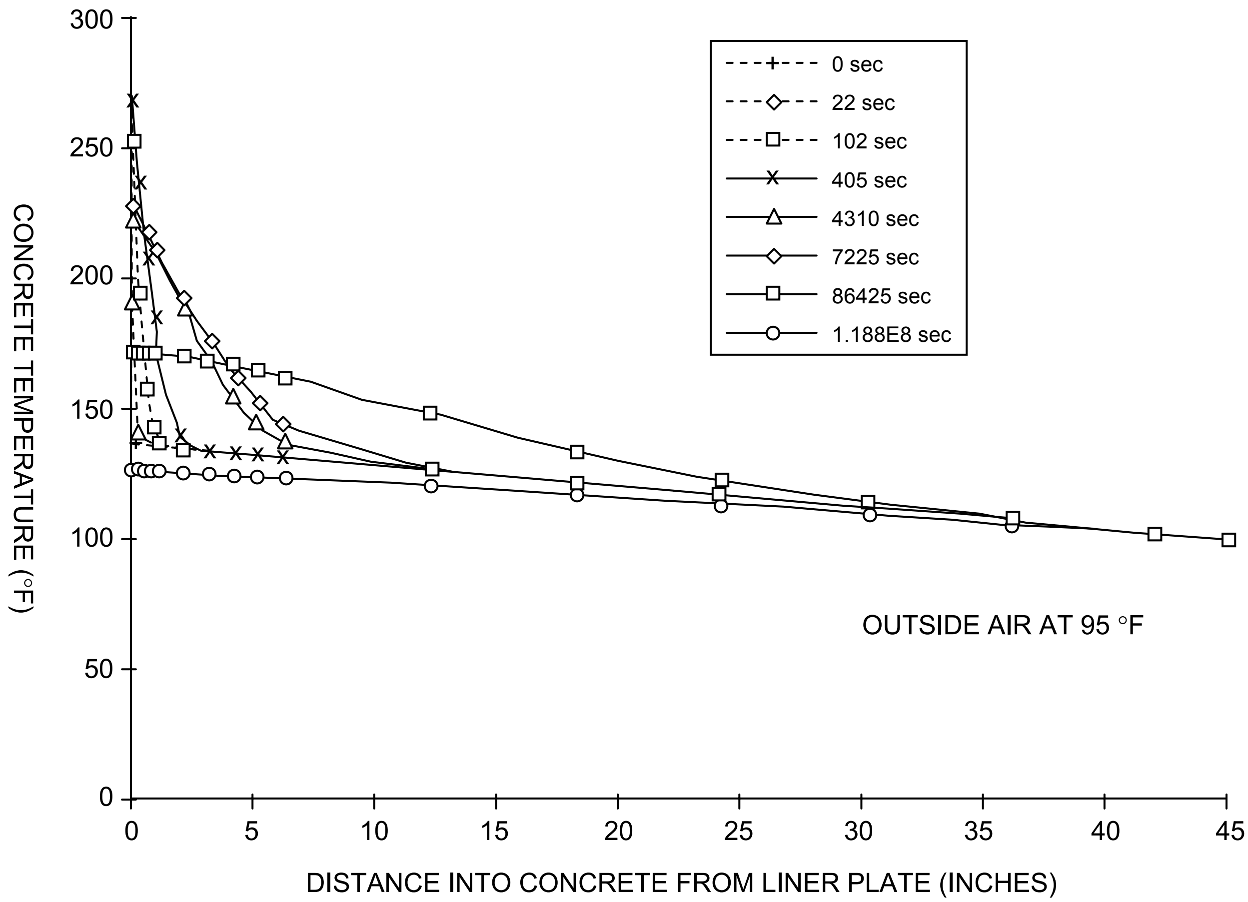
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ATMOSPHERE AND SUMP TEMPERATURE FOR THE REACTOR BUILDING DBA (5 FT^2)	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-63

AMENDMENT 20

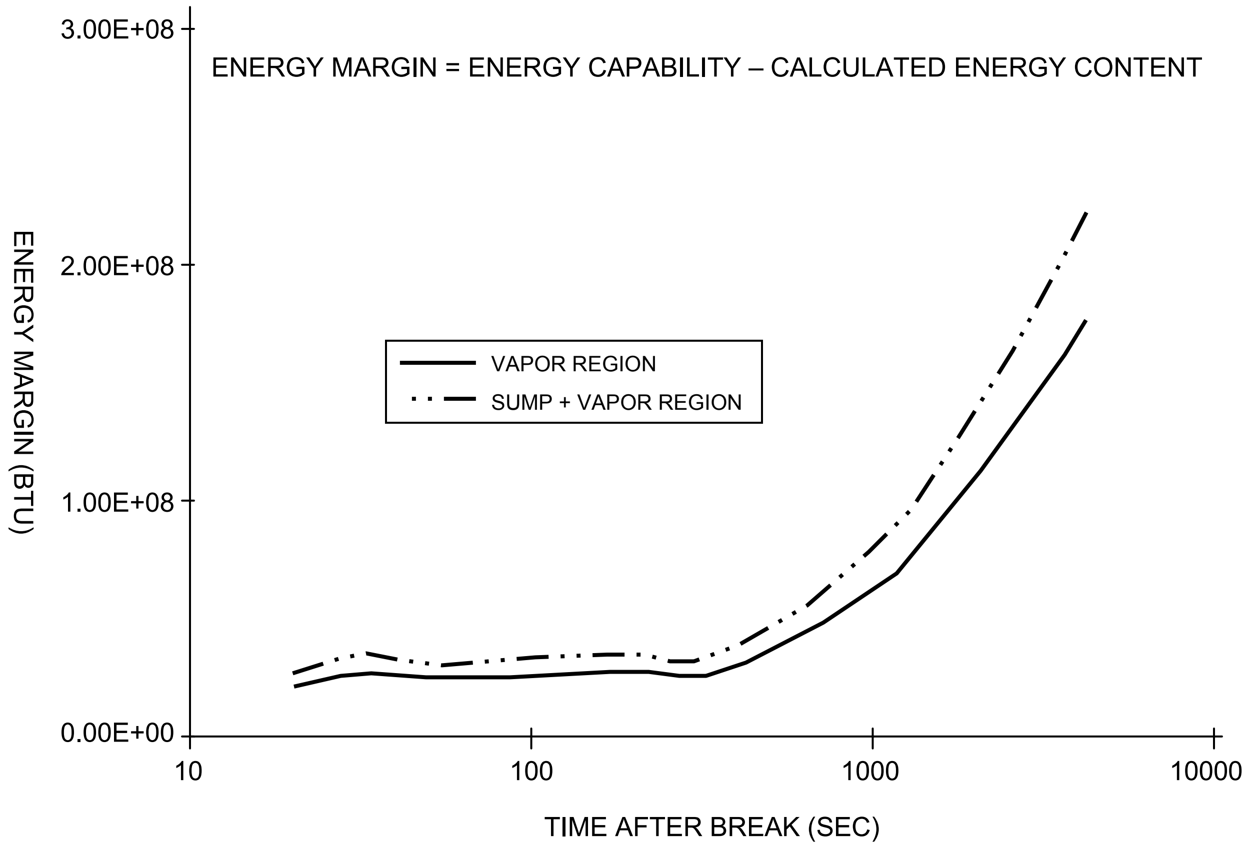
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
REACTOR BUILDING ENERGY DISTRIBUTION FOR THE REACTOR BUILDING DBA (5 FT^2)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-64

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
REACTOR BUILDING WALL TEMPERATURE PROFILES FOR THE REACTOR BUILDING DBA (5 FT²)	BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-65

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
ENERGY MARGIN VS TIME FOR THE REACTOR BUILDING DBA (5 FT^2)	BASED ON DRAWING NO	SHEET / REV.

FIGURES 14-66 THROUGH 14-103

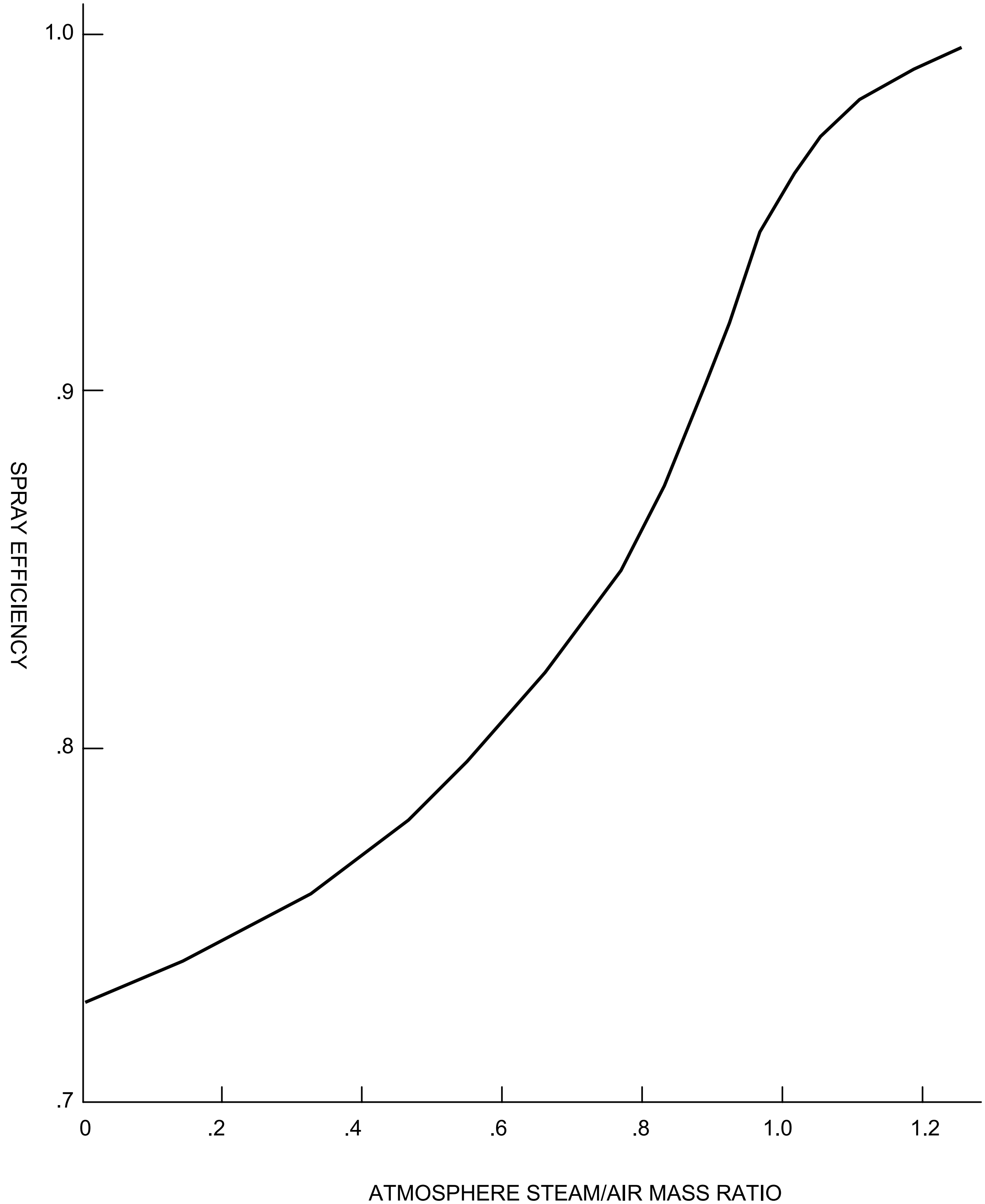
DELETED

SAR FIGURE NO. 14-66

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

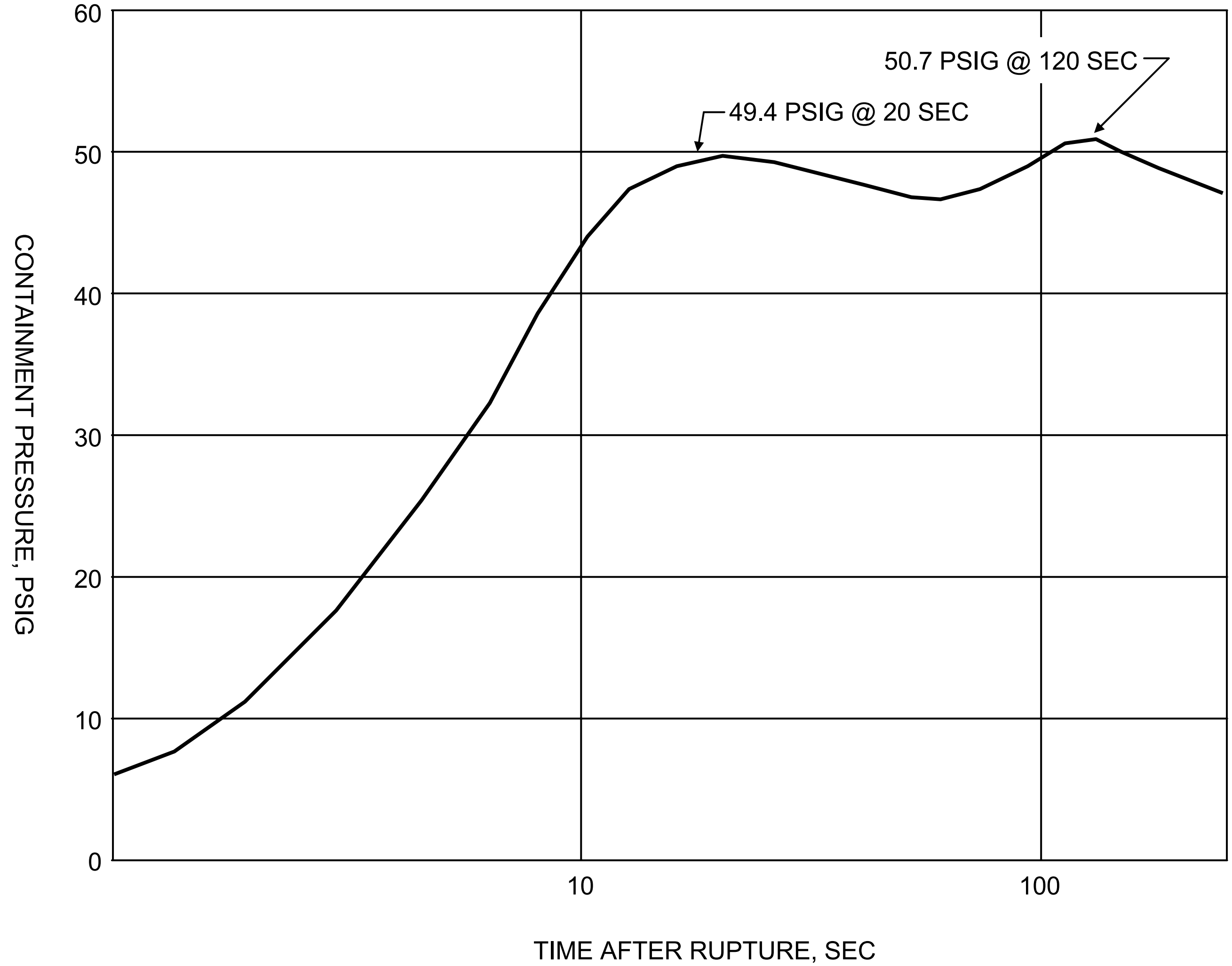
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-104

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
THERMAL HEAT REMOVAL EFFICIENCY OF REACTOR BUILDING SPRAYS	BASED ON DRAWING NO	SHEET / REV.



CONTAINMENT PRESSURE 7.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

SAR FIGURE NO. 14-105

AMENDMENT 20

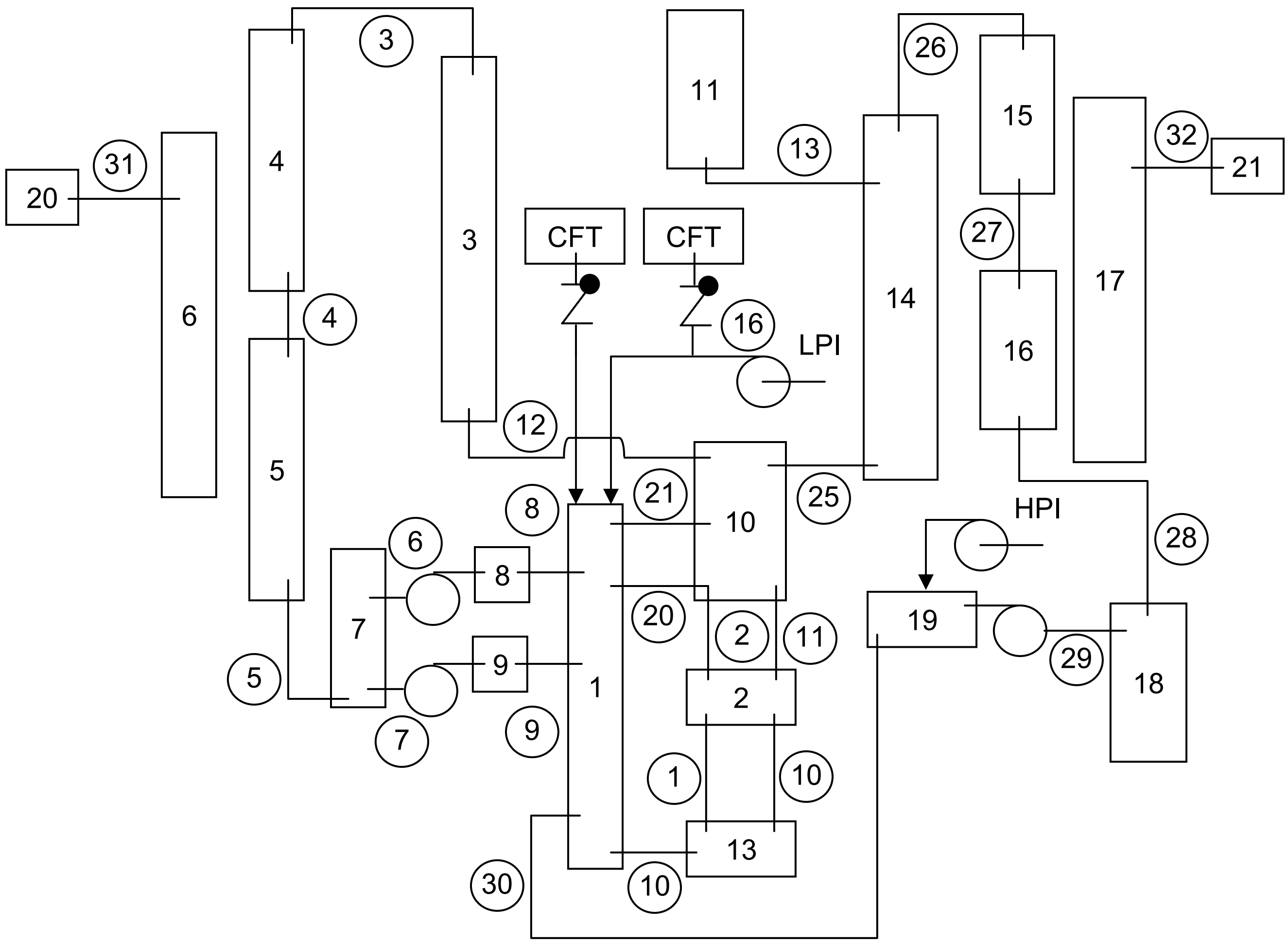
ARKANSAS NUCLEAR ONE

UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

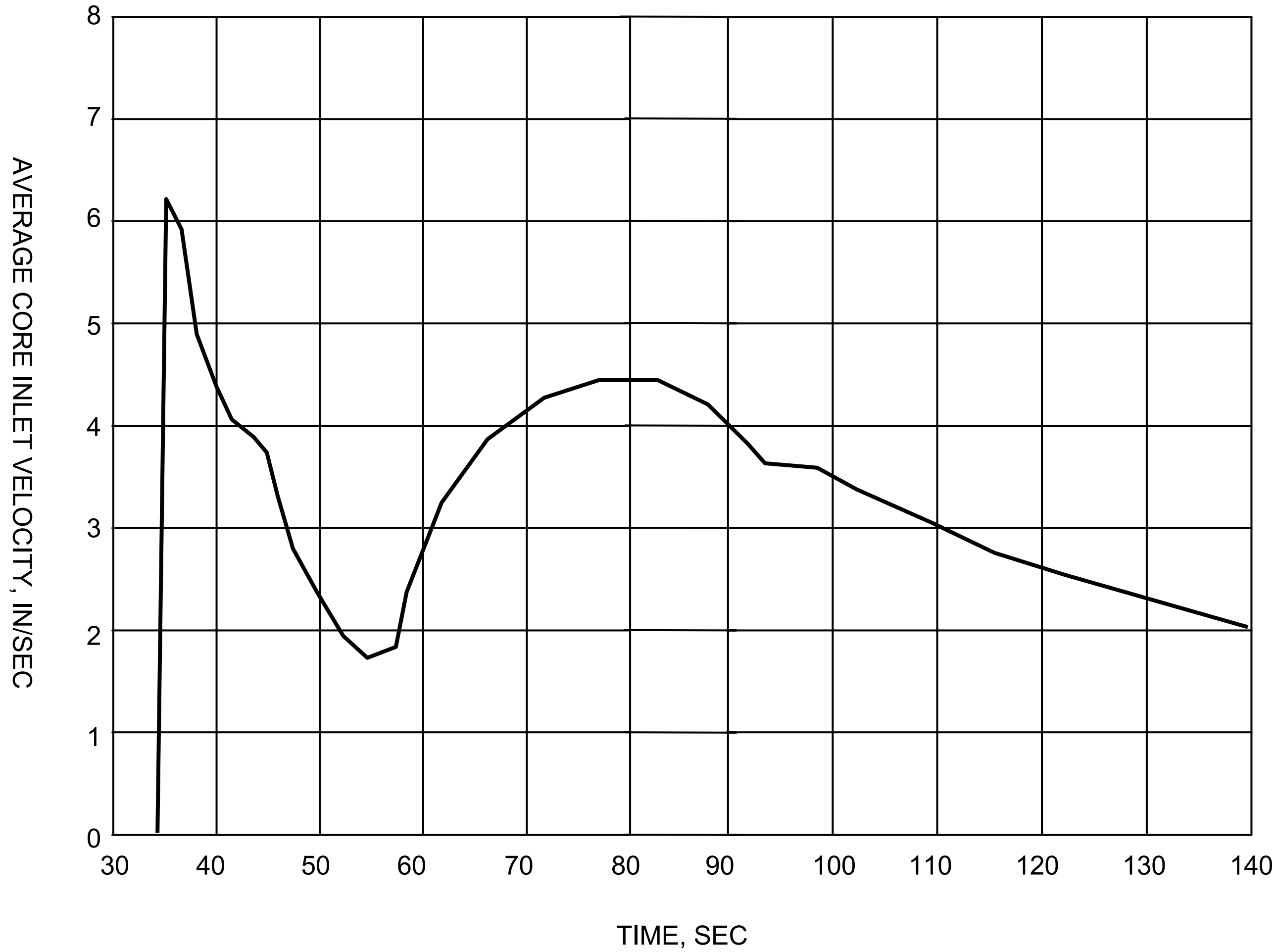
BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. 14-106

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	
MULTINODE REPRESENTATION OF NUCLEAR STEAM SUPPLY SYSTEM	BASED ON DRAWING NO	SHEET	REV.



AVERAGE CORE INLET VELOCITY 7.0 FT^2 COLD LEG BREAK (PUMP SUCTION)

SAR FIGURE NO. 14-107

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

NOMINAL MISLOADED	1.51 1.98	1.11 1.20	1.40 1.45	1.11 1.19	1.03 1.02	1.42 1.41	1.12 1.11	0.83 0.83
		1.29 1.33	1.63 1.68	1.48 1.50	1.05 1.05	1.14 1.13	0.92 0.92	0.83 0.83
			1.58 1.60	1.25 1.28	1.25 1.25	1.00 1.00	1.07 1.07	0.80 0.80
				1.00 0.90	1.06 1.03	1.18 1.17	1.03 1.03	
					1.25 1.16	1.27 1.24	1.03 1.02	
						1.14 1.12		

XXX	NOMINAL POWER DISTRIBUTION
XXX	MISLOADED ASSEMBLY POWER DISTRIBUTION

SAR FIGURE NO. 14-108

AMENDMENT 20

ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	
DESIGN:	ENTERGY
CAD NO:	

RADIAL X LOCAL ASSEMBLY POWER DISTRIBUTIONS – CASE 3A

BASED ON DRAWING NO	SHEET	REV.

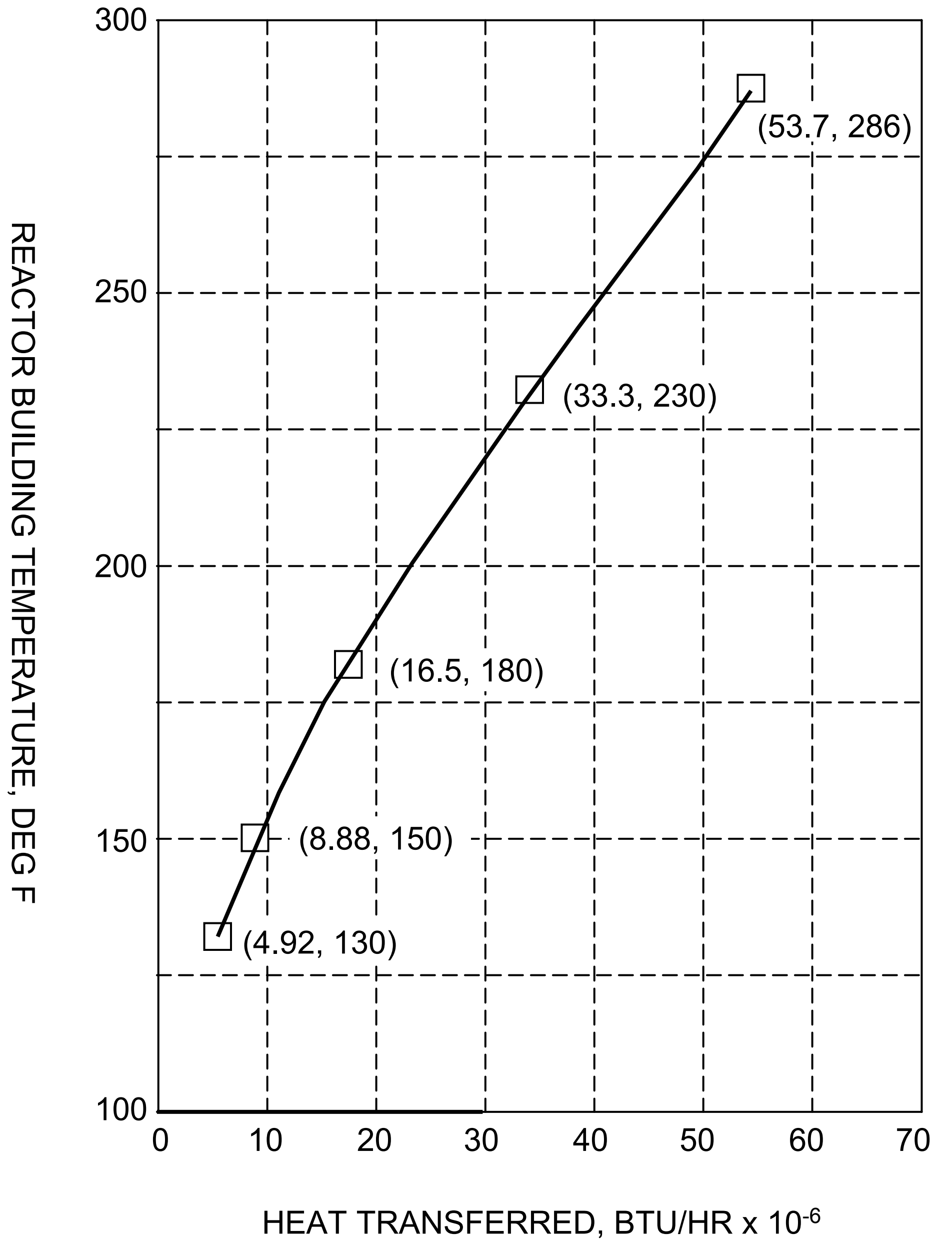
					1.11	1.11	1.06	1.11	1.11					
					1.76	1.78	1.79	1.78	1.77					
			.96	1.17	1.46	1.01	1.23	1.01	1.46	1.17	.96			
			.98	.99	1.01	.91	1.09	.92	1.06	1.04	1.04			
		.95	1.02	1.16	1.05	1.18	1.30	1.18	1.05	1.16	1.02	.95		
		1.06	1.20	1.10	1.93	1.13	1.38	1.15	1.99	1.20	1.30	1.19		
	.96	1.02	1.00	.95	1.14	1.00	.99	1.00	1.14	.95	1.00	1.02	.96	
	.98	1.19	1.17	1.01	1.20	1.02	1.02	1.05	1.26	1.08	1.31	1.34	1.11	
	1.17	1.16	.95	.88	1.01	1.21	1.01	1.21	1.01	.88	.95	1.16	1.17	
	.98	1.08	1.01	.95	1.13	1.43	1.15	1.48	1.27	.98	1.12	1.26	1.12	
1.11	1.46	1.05	1.14	1.01	1.26	1.35	1.26	1.35	1.26	1.01	1.14	1.05	1.46	1.11
.74	1.00	.92	1.19	1.19	1.51	1.57	1.38	1.63	1.62	1.24	1.36	1.07	1.23	.92
1.11	1.01	1.18	1.00	1.21	1.35	1.26	1.19	1.26	1.35	1.21	1.00	1.18	1.01	1.11
.76	.87	1.10	1.01	1.41	1.56	1.24	1.11	1.31	1.58	1.58	1.28	1.47	1.16	.99
1.06	1.23	1.30	.99	1.01	1.26	1.19	1.55	1.19	1.26	1.01	.99	1.30	1.23	1.06
1.76	1.04	1.33	.99	1.10	1.35	1.07	1.50	1.14	1.45	1.17	1.29	1.93	1.44	.99
1.11	1.01	1.18	1.00	1.21	1.35	1.26	1.19	1.26	1.35	1.21	1.00	1.18	1.01	1.11
.76	.87	1.10	1.01	1.41	1.56	1.24	1.11	1.31	1.68	1.58	1.28	1.47	1.16	.99
1.11	1.46	1.05	1.14	1.01	1.26	1.35	1.26	1.35	1.26	1.01	1.14	1.05	1.46	1.11
.74	1.00	.92	1.19	1.19	1.51	1.57	1.38	1.63	1.62	1.24	1.36	1.07	1.23	.92
	1.17	1.16	.95	.88	1.01	1.21	1.01	1.21	1.01	.88	.95	1.16	1.17	
	.98	1.09	1.01	.95	1.13	1.43	1.15	1.48	1.27	.98	1.12	1.26	1.12	
	.96	1.02	1.00	.95	1.14	1.00	.99	1.00	1.14	.95	1.00	1.02	1.96	
	.98	1.19	1.17	1.01	1.20	1.02	1.02	1.06	1.26	1.08	1.31	1.34	1.11	
		.95	1.02	1.16	1.05	1.18	1.30	1.18	1.05	1.16	1.02	.95		
		1.06	1.20	1.10	.93	1.13	1.38	1.15	.99	1.20	1.30	1.19		
			.96	1.17	1.46	1.01	1.23	1.01	1.46	1.17	1.96			
			.98	.99	1.01	.91	1.09	.92	1.06	1.04	1.04			
					1.11	1.11	1.06	1.11	1.11					
					.76	.78	.79	.78	.77					

XXX	NOMINAL POWER DISTRIBUTION
XXX	MISLOADED ASSEMBLY POWER DISTRIBUTION

SAR FIGURE NO. 14-109

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
RADIAL X LOCAL ASSEMBLY POWER DISTRIBUTIONS – CASE 3B	BASED ON DRAWING NO	SHEET / REV.



SAR FIGURE NO. 14-110

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:
REACTOR BUILDING COOLER CHARACTERISTICS	BASED ON DRAWING NO	SHEET / REV.

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UPDATE REFERENCE LIST

Section and references listed below denote documents that contain additional cross reference information used to update the SAR

Section	Cross Reference
Amendment 26	
15.1 15.1.1	TS Amendment 253, “Adoption of TSTF-431, “Change in Technical Specifications End States (BAW-2441)”

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15 TECHNICAL SPECIFICATIONS

The final Technical Specifications (TSs) were issued with the operating license and have been updated periodically through various license amendments. The TSs currently exist as updated in a separate binder/volume.

15.1 MODE 4 TS END STATES

NRC Safety Evaluation Report (SER) (1CNA031502, ML15023A147) approved a Mode 4 end state for several ANO-1 TSs. Provided risk is appropriately managed, the shutdown statements within the subject TSs no longer require placing the unit in Mode 5.

15.1.1 MODE 4 TS END STATE IMPLEMENTATION AND RISK ASSESSMENT

The aforementioned SER is associated with ANO-1 TS Amendmnet 253, which adopted TSTF-431, “Change in Technical Specifications End States (BAW-2441).” The NRC approved the Mode 4 end states for TSs specified in the SER provided the following implementation guidance is implemented:

- Entergy will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 4A, April 2011.

- Entergy will follow the guidance established in TSTF-IG-07-01, Revision 1, "Implementation Guidance for TSTF-431, Revision 3, 'Change in Technical Specifications End States,' BAW-2441-A," with the exception that Section 11 of NUMARC 93-01, Revision 4A, will be utilized to meet 10 CFR 50.65(a)(4) requirements in lieu of NUMARC 93-01, Revision 3.

The implementation guidance of TSTF-IG-07-01 has been implemented into a common Operations directive. The directive, along with procedures associated with risk assessments, were also updated to ensure risk assessments would be performed in accordance with NUMARC 93-01, Revision 4A, Section 11.

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UPDATE REFERENCE LIST

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Section	Cross References
Amendment 19	
16.3.2	SAR Discrepancy 1-97-0302, “Resolution to 1995 Revision of Transient Cycles”
Amendment 20	
16.2.20	Engineering Request ER-ANO-2002-1381-000, “Replacement Steam Generator Project”
Amendment 21	
16.2.7 16.3.7	Engineering Request ER-ANO-2002-0638-000, "Replacement of Reactor Vessel Closure Head"
16.3.9	Engineering Request ER-ANO-2002-1078-015, “Steam Generator and Reactor Vessel Head Closure Replacement – Removal and Replacement”
16.3.6	Engineering Request ER-ANO-2001-0168-024, “Implementation of Metamic Panel Inserts in the ANO-1 Spent Fuel Pool”
Amendment 21	
16.2.9	License Document Change Request 10-044, “Correction of Typographical Errors”
16.2.7	Condition Report CR-ANO-C-2010-0320, "Revise SAR to be Consistent with 10 CFR 50.55a"
Amendment 24	
16.3.2	Engineering Change EC-27035, "RCS Nozzles Environmentally-Assisted Fatigue"
Amendment 25	
16.3.4	Calculation CALC 99-E-0015-01, "Prestress Forces for Reactor Building Dome"
Amendment 26	
16.1.1 16.1.4.2 16.2.1 16.2.3.1 16.2.3.7 16.2.7	Licensing Basis Document Change LBDC 14-035, "License Renewal Commitment Changes"

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross References
16.2.18	Engineering Change EC-47351, Rev. 1, “ANO-1 PT Limits for 54 EFPY”

Amendment 28

16.2.8.1	Licensing Basis Document Change LBDC 18-027, "NFPA 805 Implementation – Removal of Fire Wrap Requirements”

Amendment 30

16.1.7 16.2.3.3 16.2.3.7 16.2.21.7	Engineering Change EC-84449, "Change from NaOH to NaTB"

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16 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment for license renewal identified several new programs and activities, modifications to existing programs, and existing programs, necessary to continue operation of ANO-1 during the additional 20 years beyond the initial license term. This chapter describes these programs and activities. The ANO-1 corrective action program, which includes the confirmation process to assure that the cause of the condition is determined and corrective action taken to preclude repetition, and the administrative (document) control program, which governs site procedures, were credited for license renewal. These programs are in accordance with the corporate quality assurance program pursuant to 10CFR Part 50, Appendix B. These programs apply to all of the programs credited for license renewal.

16.1 NEW ACTIVITIES

16.1.1 BURIED PIPE INSPECTION

Buried Pipe Inspections will be performed to ensure that a loss of material due to external surface corrosion of buried piping is adequately managed. This inspection program is based on random excavations associated with required maintenance activities, usually to facilitate repairs. The safety-related portions of underground carbon steel piping on the service water and fuel oil systems are within the scope of this inspection. The aging effect addressed by the Buried Pipe Inspection is a loss of material due to corrosion of the external surfaces of pipe caused by loss of the protective coating. This inspection will be initiated prior to the end of the initial 40-year license term.

16.1.2 ELECTRICAL COMPONENT INSPECTION

The Electrical Component Inspection Program will inspect splices, connectors, and cables within the scope of license renewal that are located in areas that may be conducive to accelerated aging. The scope of the inspection program includes cables exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperatures due to the current they are carrying, connectors used in impedance-sensitive circuits, and cable splices subject to aging-related stressors. The aging effect for cables and cable splices is a change of material properties, as evidenced by cracking or discoloration of the insulation or by degradation of a tested parameter. The aging effect for connectors in impedance-sensitive circuits is a change of material due to corrosion of connector pins. The Electrical Component Inspection Program will be formally implemented and the first inspection or test of in-scope cables, splices, and connectors will be completed prior to the expiration of the initial 40-year licensing term. Inaccessible medium-voltage cables exposed to significant moisture and voltage will either be tested for the presence of aging effects or a replacement program for these cables will be developed. If a periodic replacement of medium-voltage underground cables is determined to be the most effective action for this type of cable, ANO-1 will define the frequency for replacement prior to the expiration of the initial 40-year licensing term. The frequency will be based on site specific and industry operating experience.

16.1.3 HEAT EXCHANGER MONITORING

The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained. The Heat Exchanger Monitoring Program manages aging effects on the following safety-related systems and components: reactor building coolers, emergency diesel generator jacket cooling water heat exchangers, make-up pump lube oil

coolers, make-up pump room coolers, decay heat room coolers, decay heat system heat exchangers, electrical room chillers and coolers, control room chillers and coolers, emergency feedwater system heat exchangers, and chilled water condensers and evaporators. The aging effects addressed by the Heat Exchanger Monitoring Program are cracking and loss of material that could result in degradation in the seismic qualification of the heat exchangers. Inspection will be initiated prior to the end of the initial 40-year license term.

16.1.4 PRESSURIZER EXAMINATIONS

The Pressurizer Examinations include two specific examinations: the pressurizer cladding and the pressurizer heater penetration welds examination.

16.1.4.1 Pressurizer Cladding Examination

The pressurizer cladding examination will assess the condition of the pressurizer cladding. The scope of this activity will include the cladding and attachment welds to the cladding of the pressurizer. The aging effect is cracking of cladding by thermal fatigue, which may propagate to the underlying ferritic steel. The volumetric examinations of the pressurizer items that are most susceptible to thermal fatigue will provide assurance that cracking of cladding has not extended into the base metal of the pressurizer. These examinations are included in the ISI program and will be carried forward to the period of extended operation.

16.1.4.2 Pressurizer Heater Bundle Penetration Welds Examination

The pressurizer heater bundle penetration welds examination will be completed at ANO-1, or on the Oconee or Three Mile Island Unit 1 pressurizer heaters, and will assess the condition of the pressurizer heater penetration welds. This examination will be applicable to the heater sheath-to-diaphragm plate penetration welds inside the pressurizer. The aging effect is cracking at the heater bundle penetration welds, which may lead to reactor coolant leakage. The heater bundle examination may occur prior to entering the period of extended operation or during the period of extended operation.

16.1.5 REACTOR VESSEL INTERNALS AGING MANAGEMENT

Ongoing industry efforts are aimed at characterizing the aging effects requiring management associated with the reactor vessel internals. Further understanding of these aging effects will be developed by the industry over time and will provide additional bases for the inspections under this program. Entergy Operations will participate in the BWOG Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate, to continue investigation of aging effects requiring management for the reactor vessel internals. These activities will assist in establishing appropriate monitoring and inspection programs for the reactor vessel internals. Entergy Operations will provide periodic updates after the completion of significant milestones in the preparation of the Reactor Vessel Internals Inspection, commencing within one year of the issuance of the renewed license. Entergy Operations will submit a report to the NRC, at or about, the end of the initial 40-year operating license term. This report will summarize the current understanding of aging effects applicable to the reactor vessel internals and will contain the Entergy Operations' inspection plan, including methods for each inspection. Entergy Operations will perform the Reactor Vessel Internals Inspection. Should data or evaluations indicate that this inspection can be modified or eliminated, Entergy Operations will provide plant-specific justification to demonstrate the basis for the modification or elimination. The purpose of the Reactor Vessel Internals Inspection is to inspect and

examine the reactor vessel internals to assure the aging effects will not result in loss of the intended functions of the internals during the period of extended operation. The inspection applies to the reactor vessel internals. This inspection will begin during the period of extended operation. The aging effects for the reactor vessel internals include cracking due to either stress corrosion or irradiation assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, dimensional changes due to void swelling, and loss of bolted closure integrity due to stress relaxation.

16.1.6 SPENT FUEL POOL MONITORING

The Spent Fuel Pool Monitoring Program will manage the aging effects requiring management of the ANO-1 spent fuel pool liner. Stress corrosion cracking is possible from the external surface of the liner in weld heat-affected zones since this was not verified to be chloride-free during construction. This program will be initiated before the end of the initial 40-year license term.

16.1.7 WALL THINNING INSPECTION

Wall Thinning Inspections will be performed to ensure wall thickness is above the minimum required to avoid leaks or failures under normal conditions and postulated transient and accident conditions, including seismic events. Wall Thinning Inspections will cover the following safety-related systems and components: EFW pump casing and carbon steel discharge piping and valves, EFW steam supply components downstream of steam admission valves, EFW steam exhaust piping and valves, EFW carbon steel cooling water, seal water, and instrument piping and valves, EFW turbine lube oil cooler, carbon steel EFW supply header piping and valves (condensate supply), carbon steel piping and components of the main steam system, and carbon steel components of penetrations 11, 42, 43, 48, 49, 51, 52, 54, 58, 59, 60, 62, and 64. The aging effect to be addressed by Wall Thinning Inspection is a loss of material due to corrosion of the internal surfaces of carbon steel piping and components. This program will be initiated before the end of the initial 40-year license term.

16.2 <u>EXISTING ACTIVITIES</u>

16.2.1 ALLOY 600 AGING MANAGEMENT

The Alloy 600 Aging Management Program section has been incorporated into the ISI Program.

16.2.2 ALTERNATE AC DIESEL GENERATOR TESTING AND INSPECTIONS

The Alternate AC Diesel Generator Testing and Inspections ensures that the effects of aging are managed before the loss of the intended functions of the system. The Alternate AC Diesel Generator Testing and Inspections applies to the station blackout diesel and its components. The aging effects addressed by the Alternate AC Diesel Generator Testing and Inspections include: loss of material and loss of mechanical closure integrity for the starting air subsystem components; loss of material and loss of mechanical closure integrity for the intake combustion air subsystem components; loss of material, fouling, and loss of mechanical closure integrity for the intake air aftercooler; loss of material for carbon steel components, cracking of the stainless steel components and loss of mechanical closure integrity for the exhaust subsystem components; loss of mechanical closure integrity for the lube oil subsystem components; fouling, loss of material from wear, and loss of mechanical closure integrity for the lube oil cooler; loss of material and loss of mechanical closure integrity for the cooling water subsystem components; fouling and a loss of material for the AAC radiator; loss of material from wetted portions of the exhaust fan housings; and fouling of the fuel oil heat exchanger.

16.2.3 ASME SECTION XI INSERVICE INSPECTION

16.2.3.1 <u>IWB Inspections</u>

The ASME Section XI, Subsection IWB Inspections under the scope of the Inservice Inspection Plan identifies and corrects degradation of ASME Class 1 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWB Inspections, credited for license renewal, is identified specifically for each component and for applicable component features in the ISI Plan. The aging effects managed as part of the ASME Section XI, Subsection IWB Inspections include cracking, loss of mechanical closure integrity at bolted connections, and loss of material. A one-time augmented inspection of a reactor coolant pump (visual inspection of the pressure retaining surfaces, including the cover) is required to be performed prior to the end of the initial 40-year license term. The visual inspection of the RCP P-32C casing was performed during the 1R22 outage. In lieu of a visual inspeciton of the RCP P-32C cover, an inspection of an identical Davis-Besse plant RCP cover in 2002, in conjunction with evaluations of water chemistry data and flaw tolerance, is credited for aging management of the cover.

16.2.3.2 <u>IWC Inspections</u>

ASME Section XI, Subsection IWC Inspections under the scope of the ANO-1 Inservice Inspection Plan identify and correct degradation of ASME Class 2 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWC Inspections, credited for license renewal, includes components on the following systems: core flood, RBS, main feedwater, spent fuel, service water, HPI/makeup and purification, LPI/decay heat, EFW, main steam, reactor building isolation, and chilled water system. The aging effects managed as part of the ASME Section XI, Subsection IWC Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.3 IWD Inspections

ASME Section XI, Subsection IWD Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of ASME Class 3 pressure-retaining components. The scope of the ASME Section XI, Subsection IWD Inspections, credited for license renewal, includes components on the following systems: service water, spent fuel, main steam, EFW, and condensate storage. The aging effects managed as part of the ASME Section XI, Subsection IWD Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.4 IWE Inspections

ASME Section XI, Subsection IWE Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of Class MC pressure retaining components, their integral attachments, the metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. The scope of the ASME Section XI, Subsection IWE Inspections, credited for license renewal includes inspections of the reactor building liner plate. The aging effect managed as part of the ASME Section XI, Subsection IWE Inspections is a loss of material of the steel surfaces.

16.2.3.5 IWF Inspections

ASME Section XI Inservice Inspection Program, IWF Inspections identify and correct degradation of ASME Class 1, 2, 3, or MC component supports. The aging effects managed as part of the ASME Section XI, Subsection IWF include cracking, loss of material, and change in material properties.

16.2.3.6 IWL Inspections

ASME Section XI Inservice Inspection Program, IWL Inspections provides instructions and documentation requirements for assessing the quality and structural performance of the reactor building’s post-tensioning system and concrete components. The scope includes the reactor building's post-tensioning system and concrete components. The aging effects are loss of material for tendon wires and anchorage and cracking and change in material properties for prestressed concrete components. The IWL inspections and associated trending are performed in accordance with 10 CFR 50.55a(b)(2)(ix)(B). Acceptance criteria are in accordance with ASME Code, Section XI, Subsection IWL requirements. Corrective actions may include re-tensioning, replacing tendons, or reanalysis of the reactor building to assure adequate tendon prestress to meet design requirements.

16.2.3.7 Augmented Inspections

The ASME Section XI, Augmented Inspections identify and correct degradation of components outside of the jurisdiction of ASME Section XI. Augmented periodic inspections are completed for several main feedwater and main steam system welds, not in the Class 2 piping, to support the high energy line break analysis. Augmented inspections are completed for the BWST header including the lines from the reactor building sump. Augmented inspections that will be added to the program because of license renewal include a special augmented inspection on the welds of the piping wetted by the reactor building sump water, some supplemental inspections of the “Q” stainless piping of the main steam system, at least a one-time inspection of the penetration 68 piping and components and the decay heat pump room drain valves, and special inspections of penetrations 10, 47, 58, and 64. Other augmented inspections include a

visual inspection of pressure retaining surfaces of one RCP, and volumetric/non-destructive inspection of chilled water system stainless steel tubing and valves. The required visual inspection of one RCP casing, P-32C, was performed in the 1R22 outage. In lieu of a visual inspeciton of an RCP cover, inspection results of an identical pump cover from Davis-Besse plant, in conjunction with evaluations of water chemistry data and flaw tolerance, are credited for aging management of the cover. The aging effects managed by these inspections are cracking and loss of material. The new inspections will be initiated prior to the end of the initial 40-year license term.

16.2.3.8 Small Bore Piping and Small Bore Nozzles Inspections

The Small Bore Piping and Small Bore Nozzles Inspections identify aging effects on small bore piping and nozzles. The small bore piping and small bore nozzles, within the scope of this program, are RCS piping and nozzles less than 4-inch NPS that do not receive volumetric inspection in accordance with ASME Section XI. The aging effect managed by this program is cracking. A risk-informed ISI method has been implemented to select RCS piping welds for inspection. The risk-informed approach consists of two essential elements. A degradation mechanism evaluation is performed to assess the failure potential of the piping system under consideration, and a consequence evaluation is performed to assess the impact on plant safety in the event of a piping failure. The results from these two independent evaluations are coupled to determine the risk significance of piping segments within the system, and priority is then given to the most risk significant piping segments during the selection of RCS piping welds for inspection.

16.2.4 BOLTING AND TORQUING ACTIVITIES

Bolting and Torquing Activities prevent degradation of bolting or identify and correct degradation of bolting. The scope of Bolting and Torquing Activities applies to pressure boundary bolting applications associated with components within the scope of license renewal. Applications include bolted flange connects for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps or valves and miscellaneous process components. The aging effects addressed by Bolting and Torquing Activities are cracking, loss of material, and loss of mechanical closure integrity.

16.2.5 BORIC ACID CORROSION PREVENTION

The Boric Acid Corrosion Prevention Program prevents corrosion damage due to leakage from the borated water systems. The Boric Acid Corrosion Prevention Program is concerned with the RCS and other structures and components containing, or exposed to, borated water. This program is credited with monitoring the boric acid corrosion of carbon steel external surfaces of structures and components exposed to leakage from borated water. Carbon steel is utilized for bolting on many of the systems that contain borated water. This program manages the loss of material of bolts that could eventually result in a loss of mechanical closure integrity for bolted connections.

16.2.6 CHEMISTRY CONTROL

The following subsections address the individual ANO-specific chemistry control programs in more detail:

- Primary Chemistry Monitoring
- Secondary Chemistry Monitoring
- Auxiliary Systems Chemistry Monitoring
- Diesel Fuel Monitoring
- Service Water Chemical Control

16.2.6.1 Primary Chemistry Monitoring

The Primary Chemistry Monitoring Program maximizes long-term availability of primary systems by minimizing system corrosion, fuel corrosion, and radiation field build-up. The scope of the Primary Chemistry Monitoring Program includes sampling activities and analysis on the following systems: RCS, borated water storage tanks, spent fuel pool system, letdown purification demineralizers, and reactor makeup water. The Primary Chemistry Monitoring Program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems covered by the program. This prevents or minimizes the occurrence of cracking and other aging effects.

16.2.6.2 Secondary Chemistry Monitoring

The Secondary Chemistry Monitoring Program maximizes the availability and operating life of major components. The scope of the Secondary Chemistry Monitoring Program includes sampling activities and analysis on the main feedwater system, emergency feedwater system, main steam system, condensate storage system, and steam generators. The aging reviews for many of the safety-related, non-Class 1 systems also indirectly credit the Secondary Chemistry Monitoring Program since the condensate storage tanks are used as a source of makeup water to these systems. The Secondary Water Chemistry Monitoring Program ensures the levels of contaminants and oxygen are maintained within a range that prevents or minimizes the occurrence of loss of material and other aging effects.

16.2.6.3 Auxiliary Systems Chemistry Monitoring

The Auxiliary Systems Chemistry Monitoring Program maximizes the availability and operating life of the components used for the closed cooling loops. The scope of the Auxiliary Systems Chemistry Monitoring Program is limited to sampling activities and analysis on the ICW system, chilled water systems, emergency diesel generators, and the AAC diesel generator. The Auxiliary Systems Chemistry Monitoring Program is credited with minimizing the loss of material due to corrosion, cracking, fouling, and loss of mechanical closure integrity.

16.2.6.4 Diesel Fuel Monitoring

The Diesel Fuel Monitoring Program ensures that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of the fuel systems. The scope of the Diesel Fuel Monitoring Program is limited to sampling activities and analysis on the following

tanks: bulk fuel oil storage tank, EDG fuel tanks, EDG day tanks, fire pump diesel day tank, and the AAC diesel generator day tank. The aging management reviews credit the sampling and monitoring as providing an adequate control of the fuel oil to ensure water and contamination (including microbiological) are not present in the system. The Diesel Fuel Monitoring Program is credited with minimizing fouling, cracking, and loss of material.

16.2.6.5 Service Water Chemical Control

The Service Water Chemical Control Program maximizes the availability and operating life of the components in the service water system. The scope of the Service Water Chemical Control Program includes sampling activities and analysis on the service water system. The scope also includes chemical injection into the service water bays. The fire protection system takes suction from the service water bays. The Service Water Chemical Control Program has been credited for aging management in the service water system and the fire protection system since these systems draw suction from the intake structure. The chemical additions are credited only with reducing corrosion, not eliminating this mechanism.

16.2.7 REACTOR VESSEL CLOSURE HEAD (RVCH) PENETRATION INSPECTION

The CRDM Nozzle other vessel closure penetration inspections are in accordance with ASME IWB (SAR Section 16.2.3.1) and Code Case N-729-1.

16.2.8 FIRE PROTECTION

Fire Protection Program activities, with respect to aging management, include: fire barrier inspections, fire hose station inspections, fire suppression water supply system surveillance, fire suppression sprinkler system surveillance, fire water piping thickness evaluation, control room halon fire system inspection, NFPA 25 testing of sprinkler head components that are 50 years old, and RCP oil collection system visual inspection.

16.2.8.1 Fire Barrier Inspections

Fire Barrier Inspections provide for periodic surveillance of fire barriers separating redundant safe shutdown systems to assure that they perform their separation functions. The scope includes 10 CFR 50.48-required fire walls and fire floors as indicated on the fire protection drawings. Fire doors/hatches, fire damper mountings, and penetration fire stops associated with 10 CFR 50.48-required fire walls and fire floors are within the scope. The aging effects for fire barriers are cracking, loss of material, and change in material properties.

16.2.8.2 Fire Hose Station Inspections

The Fire Hose Station Inspections assure that manual fire suppression is available to safety-related equipment. Fire hose reels associated with 10 CFR 50.48-required fire hose stations are within the scope of license renewal. The aging effect for fire hose reels is a loss of material.

16.2.8.3 Fire Suppression Water Supply System Surveillance

This surveillance verifies operability of fire suppression water supply system components. The Fire Suppression Water Supply System Surveillance applies to fire water system supply piping and valves. The surveillance applies to several diesel fire pump subsystems including the

intake air, exhaust, lube oil, and cooling water. Fire protection system heat exchangers are also within the scope of this surveillance. This program verifies that loss of material due to internal surface corrosion of carbon steel, stainless steel, brass, or bronze components is managed. This program is credited with managing fouling of heat exchangers. Cracking of stainless steel, brass, or bronze components is also an aging effect being managed. This program also manages the loss of mechanical closure integrity and the loss of material from external surfaces for fire protection components.

16.2.8.4 Fire Suppression Sprinkler System Surveillance

This surveillance verifies operability of fire suppression sprinkler system components. Within the scope of license renewal, the Fire Suppression Sprinkler System Surveillance applies to fire suppression sprinkler system piping, valves, and nozzles. This surveillance manages loss of material due to internal surface corrosion of carbon steel, stainless steel, brass or bronze components. Cracking of stainless steel, brass, or bronze components is also an aging effect being managed.

16.2.8.5 Fire Water Piping Thickness Evaluation

The Fire Water Piping Thickness Evaluation provides a method for the examination and evaluation of pipe wall thickness changes in the fire water system. Within the scope of license renewal, the Fire Water Piping Thickness Evaluation applies to fire water system piping. A loss of material by internal surface corrosion of cast iron or carbon steel fire water system components is the aging effect managed by the Fire Water Piping Thickness Evaluation.

16.2.8.6 Control Room Halon Fire System Inspection

The Control Room Halon Fire System Inspection assures that frequently manipulated components are free of aging effects. The components within the scope of the Control Room Halon Fire System Inspection are the halon discharge nozzles, the halon discharge tube assembly and the halon pilot header flexible tubing and fittings and discharge tube assembly fittings. The aging effects addressed by the Control Room Halon Fire System Inspection are loss of material due to wear from frequent manipulations and cracking.

16.2.8.7 Reactor Coolant Pump Oil Collection System Visual Inspection

The Reactor Coolant Pump Oil Collection System Inspection ensures integrity of the reactor coolant pump oil leakage collection system. The Reactor Coolant Pump Oil Collection System Inspection applies to the shrouds, drip pans, dammed areas, accessible piping, collection tanks, and spray protection. The aging effects addressed by the Reactor Coolant Pump Oil Collection System Inspection are a loss of material and a loss of mechanical closure integrity. These aging effects would be caused by general corrosion of the carbon steel internal surfaces or external surfaces due to the potential for water leakage into the system.

16.2.9 FLOW ACCELERATED CORROSION PREVENTION

The Flow Accelerated Corrosion Prevention Program provides a programmatic approach for identifying, inspecting, and managing loss of material for components that are adversely affected by flow accelerated corrosion (also known as erosion/corrosion). Within the scope of license renewal, only the main feedwater and main steam systems are identified as susceptible to flow-accelerated corrosion. The aging mechanism is flow-accelerated corrosion, a

phenomenon that results in metal loss from components made of carbon steel, which occurs only under certain conditions of flow, chemistry, geometry, and material. The aging effect is loss of material. The aging management reviews credit this program with determining which systems are susceptible to flow-accelerated corrosion and monitoring the loss of material for those systems.

16.2.10 INSPECTION AND PREVENTIVE MAINTENANCE OF THE POLAR CRANE

The program provides for the inspection and preventive maintenance of the polar crane. The scope of this program includes the structural steel associated with the polar crane. The aging effect managed by the Inspection and Preventive Maintenance of the Polar Crane is a loss of material.

16.2.11 INSTRUMENT AIR QUALITY

With respect to license renewal, the Instrument Air Quality Program ensures that the instrument air supplied to components is maintained free of water and significant contaminants. The Instrument Air Quality Program applies to those components, within the scope of license renewal, supplied with instrument air where pressure boundary integrity is required for the component to perform its intended function. The aging effects requiring management addressed by the Instrument Air Quality Program are loss of material and cracking.

16.2.12 LEAKAGE DETECTION IN REACTOR BUILDING

Leakage detection in the reactor building monitors leakage to manage the consequences of cracking, loss of material, or loss of mechanical closure integrity. Leakage detection in the reactor building is focused on RCS leakage, but also includes other systems that have the potential to leak in the reactor building.

16.2.13 MAINTENANCE RULE

Maintenance Rule system and structural walk downs are conducted to detect and manage aging effects of structures and components within the scope of the license renewal. This includes coatings inspections of coated surfaces on structures and components. The Maintenance Rule is utilized to manage cracking, loss of material, loss of mechanical closure integrity, and change in material properties of structures and components within the scope of license renewal. This program applies only to external surfaces of the structures and components within the scope of the license renewal.

16.2.14 OIL ANALYSIS

The Oil Analysis Program ensures the oil environment in the mechanical systems is maintained to the quality required. Oil analysis program controls are credited as a program for maintaining oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to corrosion. The scope of the Oil Analysis Program, with respect to license renewal, is limited to sampling activities and analysis on the auxiliary building electrical room chillers, emergency diesel generators, decay heat pumps, reactor building spray pumps, primary makeup pumps, diesel driven fire pump and engine, EFW pumps and turbine, the AAC diesel generator, and the control room chiller compressor. The Oil Analysis Program has been credited for ensuring the oil is free of water or contaminants. This manages the aging effects of cracking and loss of material.

16.2.15 PREVENTIVE MAINTENANCE

The purpose of the Preventive Maintenance Program is to perform preplanned, repetitive maintenance tasks on plant components and systems to extend equipment operating-life and to minimize the possibility of in-service component failures. The scope of the Preventive Maintenance Program, with regard to license renewal, is the preventive maintenance tasks credited with managing the aging effects listed below:

Preventive Maintenance Activity	Aging Effect
BWST internal inspection	Loss of material
BWST external inspection	Loss of material and loss of mechanical closure integrity
Reactor building ventilation cooling coil cleaning and inspection	Fouling, cracking* and loss of material
Hydrogen sampling system cabinet/heat exchanger cleaning, inspection, and lubrication	Fouling
Emergency fire diesel cooling water quarterly sampling for corrosion inhibitor	Loss of material
Emergency fire diesel intake air, exhaust air, lube oil and cooling water.	Loss of mechanical closure integrity
Emergency fire diesel inspection.	Loss of material, cracking, loss of mechanical closure integrity and fouling
Penetration room floor drain check valves inspection	Loss of material and cracking
Decay heat room drain valves inspection	Loss of material, cracking, and loss of mechanical closure integrity
EDG fuel oil tank inspection	Loss of material
EDG HVAC components inspection	Loss of material and loss of mechanical closure integrity
Control room ventilation inspections	Fouling and loss of material
Battery Charger and Penetration Room Cooler Cleaning and Inspection	Loss of material, cracking* and loss of mechanical closure integrity
Auxiliary Building Switchgear Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, cracking* and fouling
Auxiliary Building Decay Heat Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling
HPI Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling

* cracking applies to inspection of expansion joint only

16.2.16 REACTOR BUILDING LEAK RATE TESTING

The Reactor Building Leak Rate Testing Program provides assurance that leakage from the reactor building does not exceed required maximum values for reactor building leakage. The Reactor Building Leak Rate Testing Program is comprised of Integrated Leak Rate Testing and Local Leak Rate Testing. The integrated leak rate test measures the primary reactor building overall integrated leakage rate. The scope of the integrated leak rate test is the entire reactor building. Integrated leak rate testing identifies loss of material or cracking. The local leak rate test measures the leakage across individual penetration components and determines the leakage of each penetration. Local leak rate testing identifies loss of material and cracking.

16.2.17 REACTOR BUILDING SUMP CLOSEOUT INSPECTION

The Reactor Building Sump Closeout Inspection detects significant degradation of the sump components and removes foreign objects that could impede suction from the sump. The scope of the Reactor Building Sump Closeout Inspection applies to reactor building sump, the area immediately surrounding the sump, the screening materials, and the equipment inside the sump. The aging effects addressed by the Reactor Building Sump Closeout Inspection are loss of material for the carbon steel components and cracking for stainless steel components.

16.2.18 REACTOR VESSEL INTEGRITY

The ANO-1 Reactor Vessel Integrity Program consists of the following five interrelated subprograms:

- Master Integrated Reactor Vessel Surveillance Program (MIRVP)
- Cavity Dosimetry Program
- Fluence and Uncertainty Calculations
- Pressure/Temperature Limits
- Monitoring Effective Full Power Years

The purpose of the MIRVP is to monitor reactor pressure vessel materials to determine the reduction of material toughness by neutron irradiation embrittlement. The purpose of the Cavity Dosimetry Program is to verify the accuracy of fluence calculations and to determine fluence uncertainty values. The purpose of the reactor vessel fluence and uncertainty calculations is to provide an accurate prediction of the actual reactor vessel accumulated neutron fast fluence value, for use in development of the pressure/temperature limit curves and pressurized thermal shock calculations. The purpose of the pressure/temperature limit curves is to establish the normal operating limits for the RCS. The pressure/temperature limit curves apply to the reactor vessel. The purpose of determining the EFPY is to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel. The reduction of material toughness by neutron irradiation embrittlement is the aging effect addressed by these five subprograms.

16.2.19 SERVICE WATER INTEGRITY

The service water integrity program ensures the service water system components continue to operate and perform their safety-related functions for the remaining life of ANO-1. The scope of the Service Water Integrity Program, with respect to license renewal, is limited to activities on service water system components and structures, including the emergency cooling pond. The Service Water Integrity Program has been credited with managing the following aging effects:

- The flow rate testing ensures the effects of fouling do not reduce flow rates below required values.
- The heat exchanger testing manages the aging effect of fouling by ensuring the heat exchangers can remove the necessary heat load.
- The thickness mapping and visual inspections manage the aging effects of loss of material from the service water components.
- Visual inspections of a sample of safety-related valves and heat exchangers manage the effect of cracking of the components.
- The service water bay inspection is credited for managing loss of material of the mechanical components in the service water bay.
- The ECP return line epoxy coating inspection manages loss of material in the ECP return line.

16.2.20 STEAM GENERATOR INTEGRITY

The Steam Generator Integrity Program ensures the steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The Steam Generator Integrity Program applies to the steam generator internals, tubing, and associated repair techniques and components. The aging effects addressed by the Steam Generator Integrity Program are loss of material, cracking, and fouling.

At the end of Cycle 19, the original OTSGs were replaced. The replacement OTSG tubes are made of thermally-treated Alloy 690 for improved resistance to flow induced erosion and flow assisted corrosion. Comparison of the Alloy 690 used in the replacement OTSGs with the Alloy 600 tube material in the original OTSGs shows that the replacement OTSG material strength characteristics are better than those of the original design.

16.2.21 SYSTEM AND COMPONENT MONITORING, INSPECTIONS, AND TESTING

16.2.21.1 Annual Emergency Cooling Pond Sounding

This program verifies the availability of a sufficient supply of cooling water in the emergency cooling pond to handle design basis accidents, with a concurrent loss of Lake Dardanelle. The scope includes the emergency cooling pond and surrounding structural components. The aging effect managed by this program is a loss of form of the emergency cooling pond due to sedimentation.

16.2.21.2 Battery Quarterly Surveillance

The battery rack inspections ensure their structural integrity. Seismically-qualified battery racks are within the scope. Battery racks and associated threaded fasteners are inspected for physical damage or abnormal deterioration including a loss of material.

16.2.21.3 Control Room Ventilation Testing

With respect to license renewal, the Control Room Ventilation Testing is credited as one of the programs to manage the aging effects. The control room ventilation testing applies to the control room emergency cooling coils. Fouling on the external surfaces of the cooling coil tubes is the aging effect managed by this program.

16.2.21.4 Core Flood Tank Monitoring

With respect to license renewal, the core flood tank monitoring provides a method to manage the aging effect of loss of material due to boric acid corrosion. The core flood tank monitoring applies to both core flood tanks. The loss of material due to boric acid corrosion on parts wetted by leaks from the core flood tanks may be detected through core flood tank monitoring.

16.2.21.5 Emergency Diesel Generator Testing and Inspections

With respect to license renewal, EDG Testing and Inspections provide a means of detecting aging effects associated with the various emergency diesel generator subsystems. The scope for these activities includes the emergency diesel generator assembly and associated support components. Loss of material is an aging effect for the carbon steel components in the EDG starting air system. Loss of material is identified as an aging effect for the unpainted carbon steel internal surfaces and the outer portion of the intake that could be wetted by rain. Loss of material and fouling are considered aging effects for the EDG intake air after coolers. Loss of material from the piping and muffler internal surfaces and from external surfaces exposed to the weather is an aging effect for the EDG exhaust components. Loss of material and fouling are aging effects for the lube oil coolers. The cooling water carbon steel components are susceptible to a limited loss of material from corrosion and the stainless steel components have the aging effect of cracking. Loss of material and fouling are aging effects for the cooling water heat exchangers. Since the portions of the subsystems on the engine are exposed to high vibration, loss of bolted closure integrity was identified as an aging effect for the skid mounted and connected components. The EDG day tanks are also drained and inspected internally for loss of material.

16.2.21.6 Emergency Feedwater Pump Testing

With respect to license renewal, the Emergency Feedwater Pump Testing is credited as one of the programs for managing the effects of aging. The scope includes the turbine and electric motor driven EFW pumps and associated components. Fouling in the system heat exchangers is the primary aging effect that this testing will identify. This testing also is credited with identifying the aging effects of loss of material and loss of mechanical closure integrity for system components.

16.2.21.7 Deleted

16.2.21.8 Spent Fuel Pool Level Monitoring

The spent fuel pool level monitoring provides a method of detecting changes in the spent fuel pool level that might indicate cracks in the spent fuel pool liner. The spent fuel pool level monitoring applies to the detection of leakage through the spent fuel pool liner. Cracking of the spent fuel pool liner is the aging effect addressed by spent fuel pool level monitoring.

16.3 TIME-LIMITED AGING ANALYSIS ACTIVITIES

16.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in Sections 4.3.3. Time-limited aging analyses applicable to the reactor vessel are:

- neutron embrittlement of the beltline region, including pressurized thermal shock and Charpy upper-shelf energy reduction; and

- intergranular separation in the heat affected zone of low alloy steel under austenitic stainless steel weld cladding is addressed in Section 16.3.7.

The Reactor Vessel Integrity Program as described in Section 16.2.18 is being utilized to ensure that the time dependent parameters used in the time-limited aging analysis evaluations for pressurized thermal shock and Charpy upper-shelf energy reduction are tracked such that the time-limited aging analysis remains valid through the period of extended operation.

16.3.2 METAL FATIGUE

Cyclic loads are described in Section 4.1.2.4. For the extension of plant service-life from 40 years to 60 years, metal fatigue resulting from thermal transient cyclic loads is considered a time-limited aging analysis as defined by 10 CFR 54.21(c). The time-limited aging analysis requires metal fatigue evaluations to remain valid for the period of extended operation. This is achieved by maintaining adequate documentation of fatigue stress analyses to show the allowable design cycles of the RCS components for the applicable transient events and monitoring or tracking the actual operating cycles to ensure the allowable cycles are not exceeded. Fatigue evaluations are performed based on the design-allowable cycles specified in Table 4-8 and the NSSS vendor functional specification. As such, the fatigue evaluations and the fatigue life of the RCS components are dependent on the actual operating cycles and independent of the service-life of the plant. As long as the number of applicable transient cycles are below the design-allowable cycles, the fatigue evaluations originally performed for a 40-year plant service-life are applicable for the 60-year plant service-life. Transient cycle logging will be maintained to ensure the fatigue analysis assumptions remain valid during the period of extended operation.

An assessment of the potential for fatigue cracking of the pressurizer surge line, the makeup/high pressure injection (HPI) nozzles, and the decay heat removal ASME Class 1 piping has been conducted using the method and environmental fatigue data provided in NUREG/CR-5704 and NUREG/CR-6260.

This assessment has concluded that, prior to entering the period of extended operation, for each surge line and HPI nozzle and safe-end location that may exceed a CUF of 1.0 when considering environmental effects, an approach will be developed to show that the effects of fatigue can be managed. The approach for addressing fatigue will include one or more of the following:

1. Further refinement of the fatigue analysis to lower the CUFs to below 1.0, or

2. Repair of the affected locations, or

3. Replacement of the affected locations, or

4. Management of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should Entergy Operations select Option 4 (i.e., inspection) to manage environmentally-assisted fatigue during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC prior to entering the period of extended operation.

Entergy Operations selected Option 1 to manage environmentally assisted fatigue (EAF) during the period of extended operation. The fatigue analysis performed determined that the fatigue usage for the pressurizer surge piping, hot leg surge nozzle, pressurizer surge nozzle, and high pressure injection nozzles to be acceptable (less than the allowable value of 1.0) even when the effect of EAF is considered.

16.3.3 ENVIRONMENTAL QUALIFICATION

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses. Equipment covered by the EQ Program has been evaluated to determine if existing EQ aging analyses can be projected to the end of the period of extended operation by re-analysis or additional analysis. Qualification into the license renewal period will be treated the same as equipment initially qualified for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding the qualified life in accordance with the EQ Program.

16.3.4 CONCRETE REACTOR BUILDING TENDON PRESTRESS

Loss of tendon prestress over the original license term is a time-limited aging analysis that was performed at the time of initial licensing and requires review for license renewal. Tendon lift off forces obtained during tendon surveillance are plotted and trended to assess the performance of the post-tensioning system. Acceptance criteria requires that actual forces during tendon surveillance be compared to the minimum required prestress forces of 1250 kips, 1274 kips, and 1164 kips for typical dome, vertical and hoop tendons, respectively. If trending indicates that minimum required prestress will not be met before the next tendon surveillance, corrective actions including re-tensioning, tendon replacement, or reanaylsis will be implemented. From the license renewal review, it was determined that the loss of prestress analysis is valid for the period of extended operation and will continue to be managed by tendon surveillance activities conducted under the ANO-1 ASME Section XI Inservice Inspection Program, IWL Inspections.

16.3.5 REACTOR BUILDING LINER PLATE FATIGUE ANALYSIS

Several thermal cycling conditions, which include annual outdoor temperature variations, changes in interior temperature during start-up and shutdown of the reactor coolant system, and DBA conditions were considered in the fatigue analysis of the liner plate. This analysis is a time-limited aging analysis. The projected number of cycles for these loadings for 60 years is bounded by the existing fatigue analysis. Therefore, the original design assumptions for addressing thermal fatigue of the liner plate and piping penetrations remain valid for the period of extended operation.

16.3.6 DELETED

16.3.7 REACTOR VESSEL UNDERCLAD CRACKING

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were detected in SA-508, Class-2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. BAW-10013 contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size, plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separation found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the AEC. To cover the period of extended operation, an analysis was performed using current ASME Code requirements. This analysis is fully described in BAW-2251A. The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is greater than the IWB-3612 acceptance criterion of 1.41. It is therefore concluded that the postulated intergranular separations in the ANO-1 reactor vessel 508 Class-2 forgings are acceptable for continued safe operation through the period of extended operation. Furthermore, the replacement closure head is constructed of a 508 Class-3 forging and is not susceptible to underclad cracking as specified in Regulatory Guide 1.43.

16.3.8 REACTOR VESSEL NOZZLES – FLOW-INDUCED VIBRATION ENDURANCE LIMIT

Report BAW-10051, “Flow Induced Vibration Endurance Limit Assumptions,” calculated stress values for the reactor vessel incore nozzles and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 1012 cycles for 40 years of operation. The number of fatigue cycles was extended for 60 years, and the component item stress values were compared to the recalculated endurance limit values and shown to be acceptable.

16.3.9 LEAK-BEFORE-BREAK

The successful application of leak-before-break to the main RCS piping is described in report BAW-1847 “The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS.” This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions. LBB LOCA loadings are considered for the faulted analyses of fuel assemblies as described in Section 3.3.3.3.2.1, and LBB is credited in the Section 4.2.6.6 to justify pipe whip restraints being no longer required on the main RCS piping and may be removed as needed to facilitate maintenance or other activities and in Section 4.1.2.3 for the elimination of the dynamic loads from a hot leg or cold leg break on the steam generator upper lateral restraints and lower base support. In addition, the analysis of reactor building internal pressure differentials following a LOCA applies LBB in the selection of breaks to be analyzed (see Section 14.2.2.5.5.2). The evaluation performed in FTI document 51-5000709-00, Assessment of TLAA Issues in LBB Analysis of RCS Primary Piping, demonstrates that the LBB analysis remains valid for the period of extended operation.

16.3.10 RCP MOTOR FLYWHEEL

Flaw growth analysis associated with the RCP motor flywheel is a time-limited aging analysis. The analysis for fatigue crack growth addresses the growth of pre-existing cracks. The crack growth analysis was performed for 4,000 startup/shutdown cycles for the RCP motors, which exceeds the number of design cycles by a factor of eight. Therefore, the existing crack growth analysis remains valid for the period of extended operation.

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UPDATE REFERENCE LIST

Sections and references listed below denote documents that contain additional cross reference information used to update the SAR.

Section	Cross Reference
A.7.1.13	Correspondence from Trimble, AP&L, to Seyfrit, NRC, dated July 13, 1979. (0CAN077907)
A.7.3 Table A-6	Design Change Package 85-1072, "HELB Analysis for MFW Piping."
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Figure A-1	Design Change Package 86-1032, "RCP Seal Maintenance Facility."
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A.7.4.4.4 A.7.4.13.1	Design Change Package 89-1012B, "Redundant High Pressure Injection Lines"; Calculation 89-D-1005-37, Rev. 0, "HELB/MELB Review for DCP 89-1012B Piping Additions & Modifications (HPI).
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* NOTE: Neutralizing Tank Filtration System is abandoned in place by ER-ANO-2001-0896-002

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A PIPING AND EQUIPMENT DESIGN

A.1 SCOPE

This appendix provides additional information pertinent to design, fabrication, testing, inspection, erection and assembly of pressure boundary components of equipment and piping systems classified Nuclear and/or Seismic Category 1.

Piping and equipment are classified according to service and location. It is expressly intended that this appendix be used as a reference to determine the applicability and/or jurisdiction of the codes and standards involved.

The requirements and provisions of this appendix are applicable to the Nuclear Power System components, such as pressure vessels, piping, pumps, and valves; equipment pressure parts such as fittings, flanges, bolts, valve bodies, pump casings; and, similar piping system parts which constitute a pressure boundary for the process fluid.

Specifically excluded from the scope of this appendix are non-pressure parts, such as pump motors, shafts, seals, impellers, wear rings, gland followers, seat rings, guides, and operators; any non-metallic material, such as packing and gaskets; fasteners not in pressure part joints; and, washers of any kind.

All work performed on the systems herein referenced will at least meet the requirements of the applicable codes (listed in A.3 below). In some instances, it may have been desirable to upgrade a component, i.e. incorporate more stringent requirements for examination and testing from a more restrictive code, without changing the classification of the component(s).

A.2 CODES AND INDUSTRIAL STANDARDS

The piping and equipment in this nuclear power plant are designed, fabricated, inspected and tested in accordance with recognized industrial codes and standards to the extent that these codes and standards can be applied. The application of industrial codes and standards is defined in this appendix, as well as the application of supplementary requirements. Piping systems, pumps, valves, heat exchangers and pressure vessels have been separated into classifications corresponding to the following codes and industrial standards (Note 1):

A. ANSI B31.7 Nuclear Piping Code, including the latest published addenda in force on the date of purchase.

B. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels," including the latest published addenda in force on the date of purchase.

C. ASME Boiler and Pressure Vessel Code, Section VIII, "Pressure Vessels, Division I," including the latest published addenda in force on the date of purchase.

D. ANSI B31.1.0 Power Piping Code, including the latest published addenda in force on the date of purchase.

E. ASME Nuclear Pump and Valve Code, including latest published addenda in force on the date of purchase.

The following code case interpretations (special rulings) have been used in implementing the steam generator component codes delineated herein.

CODE CASE	TITLE
N-725	Design Stress Values for UNS N06690 With a Minimum Specified Yield Strength of 35 ksi (240 MPa), Class 2 and 3 Components Section III, Division 1

The following code case interpretations have been used in implementing codes delineated herein.

CODE CASE	TITLE
1340	Alternative Performance Qualifications Section IX
1425	Requirements for piping Which Form Extensions of a Containment Vessel Section III
1426	Requirements for Electrical Penetration Assemblies and Other Appurtenances and Their Installation for Section III Class MC Vessels
1427	Requirements for Valving Which Form Extensions of a Containment Vessel
1451-1	Ultrasonic Examination for Electrical Penetrations and Attachment of Other Appurtenances to Section III Class MC Vessels

NOTES:

1. The Codes and Standards listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific equipment of interest.

A.3 CLASSIFICATION OF PIPING AND EQUIPMENT

Piping systems, valves, pumps, heat exchangers, pressure vessels, and equipment pressure parts have been separated into classifications corresponding to applicable codes and industrial standards. These classifications meet code requirements to the extent outlined in this appendix (Note 8).

CLASSIFICATION AND CODE COMPLIANCE REQUIREMENTS

Class	Piping	Valves	Heat Exchangers	Pressure Vessels
Nuclear I	B.31.7 Class I	NP&VC Class I (Note 5)	ASME-III Class A (Note 9)	ASME-III Class A (Note 1)
Nuclear II	B31.7 Class II	NP&VC Class II (Note 6)	ASME-III Class C	ASME-III Class C Class B(Note 2)
Nuclear III	B31.7 Class III	NP&VC Class III (Note 7)	ASME-III Class C	ASME-III Class C (Note 3)
Non-Nuclear Power	B31.1.0	ASME-VIII(Note 4) B31.1.0	ASME-VIII Div. 1	ASME-VIII Div. 1 (Note 3)

NOTES:

1. Refer to Chapter 4, "Reactor Coolant System," for reactor pressure vessel requirements.

2. Containment vessel only.

3. ASME Sections III and VIII do not apply to atmospheric tanks. These tanks are to be designed, fabricated, constructed and tested to meet the intent of API Standards 650 or 620. AWWA Standard D100 applies to water storage tanks.

4. Section VIII, Division 1, of the Boiler and Pressure Vessel Code will be used as a guide in calculating the thickness for pressure rated parts and sizing the cover bolting of pumps operating above 150 psig and 212 °F. Below 150 psig and 212 °F, use manufacturer's standard for service intended.

5. The Reactor Coolant System (RCS) boundary extends out through the second valve on each line off the RCS except in the case of the pressurizer relief valves, in which case the boundary will extend only through the relief valve.

   Within this boundary, the code classifications of each valve is dependent upon the purchase order date as follows:

   First Valve

Valves purchased after July 1, 1971	Constructed in accordance with ASME B&PV Code, Section III, 1971 Edition, Nuclear Power Plant Components
Manual valves purchased after January 1, 1970	Designed, fabricated, tested and inspected to ASME Pump & Valve Code, Class 1
Power operated valves purchased prior to January 1, 1970	Material and non-destructive testing to ANSI B31.7, Class 1
Relief Valves	Material and non-destructive testing equivalent to the requirements of ANSI B31.7, Class 1

<u>Second Valve</u>

Valves purchased after July 1, 1971	Constructed in accordance with ASME B&PV Code, Section III, 1971 Edition, Nuclear Power Plant Components
Manual valves purchased after January 1, 1970	Designed, fabricated, tested and inspected to ASME Pump & Valve Code, Class 1
Power operated valves purchased prior to January 1, 1970	Material and non-destructive testing to ANSI B31.7, Class 2

6. Power operated valves purchased prior to January 1, 1970 have material and nondestructive testing to ANSI B31.7, Class 2.

   Valves purchased after July 1, 1971 are constructed in accordance with ASME B&PV Code, Section III, 1971 Edition, Nuclear Power Plant Components.

7. Power operated valves purchased prior to January 1, 1970 have material and nondestructive testing to ANSI B31.7, Class 3.

   Valves purchased after July 1, 1971 are constructed in accordance with ASME B&PV Code, Section III, 1971 Edition, Nuclear Power Plant Components.

   All valves within the reactor coolant pressure boundary which are furnished by Bechtel were purchased after January 1, 1970, and were specified to meet Paragraph 452.1a of the BP&V code.

   All valves within the reactor coolant pressure boundary which are furnished by B&W were purchased prior to January 1, 1970. These valves were specified to meet ANSI B16.5 or MSS-SP66 with material control and inspection to ANSI B31.7 with the exception of the pressurizer safety valves which were specified to meet ASME Section III, Article 9.

8. The Codes and Standards listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific equipment of interest.

9. The letdown coolers, which were purchased prior to the availability of ANSI N18.2-1973, are ASME III.C on the tube side and ASME VIII on the shell side. Replacement letdown coolers are ASME III-3 on the tube side and ASME VIII Division 1 on the shell side.

A.4 INSTRUMENT/SAMPLE PIPING (NUCLEAR)

Instrument and sampling piping are of the same classification as the system piping to which they connect out to the system boundary valves. The instrument or sampling lines downstream of the boundary valve may be designed in accordance with Power Piping Code B31.1.0.

One exception to this rule occurred on small bore connections to the secondary side of the steam generator and associated main steam and main feedwater piping. Consistent with guidance found in Regulatory Guide 1.29, a distinction was made that connecting piping 2″ and smaller was not required to be designed to Seismic Category 1 standards. In addition, this piping or tubing was designed to ANSI B31.1.0 and the associated isolation valve was also designed to standards or methods permitted by ANSI B31.1.0.

It should be noted that for some newer designs, piping or tubing downstream of the isolation valve(s) may be designed to the same codes and standards as the upstream piping.

For the latest information in this regard, refer to the current qualification documentation for the components of interest.

A.5 SEISMIC DESIGN

A.5.1 CATEGORY 1 SEISMIC

This class includes piping systems and equipment whose failure could cause uncontrolled release of radioactivity or those essential for safe shutdown and immediate or long-term operation following a LOCA.

Category 1 seismic equipment is examined to assure its ability to withstand seismic requirements. Experienced designers examine the equipment using appropriate techniques to determine which specific portions of the systems and components require further examination. These techniques fit into two general categories. They are: 1) normal analytical techniques, consisting of empirical design methods as defined by appropriate design codes, and 2) special techniques, employed to supplement code calculations or to cover conditions not considered by existing codes.

A. Normal Design Techniques

All Category 1 seismic equipment, e.g. valve bodies and pump casings, is designed in accordance with applicable industrial codes & standards. Some codes utilize empirical design methods for equipment which cannot be sized by conventional rational stress analysis methods, and which do not require a detailed stress analysis for primary design work. This equipment is designed to meet a detailed functional requirement specification. The design is supported by test experience and service experience.

B. Special Supplemental Methods

Some complex equipment, e.g. reactor internals, are normally sized by rational stress analysis techniques and require supplemental criteria in areas where industrial codes do not apply.

Buried Seismic Category 1 pipes are laid in a prepared trench and backfilled with granular material. The backfill material is compacted to 90 percent of maximum dry density as determined by the Modified Compaction Procedure ASTM Designation D 1557, Method D. All Category 1 structures are founded on rock and backfilled in the same manner as Category 1 piping.

Where applicable, the analysis considered a simultaneous displacement of the structure and the pipe relative to the movement of the surrounding soil, during a seismic disturbance. The resulting pipe stresses combined with the internal pipe pressure did not exceed the allowable stress of the pipe.

Seismic Category 1 piping is designed to accommodate the maximum relative differential movement which could occur at the support points. Since the stresses resulting from this maximum relative differential movement are not predicted to occur in phase with the maximum stresses due to dynamic response of the pipe, if any, the two were combined on a root-mean-square basis. The seismic stresses will be combined in accordance with the design rules of Appendix A.

A.5.2 CATEGORY 2 SEISMIC

Category 2 equipment and piping systems are those whose failure would not result in the uncontrolled release of radioactivity and would not prevent reactor shutdown and the immediate and long-term operation following a LOCA. The failure of Category 2 equipment and piping systems may interrupt power generation.

A.5.3 PIPING

Category 1 seismic piping is classified as either rigid or flexible. Rigid piping is that which has fundamental frequency in the rigid range of the spectrum curves for the building locations. This corresponds to frequencies greater than 30 cps. These piping systems are analyzed with static loads corresponding to the acceleration in the rigid range of the spectrum curves.

The dynamic analysis of flexible Category 1 seismic piping systems for seismic loads is performed using the spectrum response method. The percentage of critical damping for all modes is 0.5 percent for the Design Basis Earthquake (DBE) and 0.5 percent for the Maximum Hypothetical Earthquake (MHE).

In lieu of the above procedure, some Category 1 seismic piping is analyzed for static load equivalent to the peak of the spectrum curve for the applicable floor elevation.

The horizontal acceleration spectrum curves applied to the piping systems are developed as part of the seismic analysis for the building in which the piping is located.

A.5.4 EQUIPMENT

Equipment is supported or restrained to accommodate seismic loading determined in accordance with the criteria defined in Section 5.1, "Design Bases for Structures, Systems and Equipment."

A.6 FABRICATION, ASSEMBLY AND ERECTION

Piping and equipment pressure parts are assembled and erected by welding. However, flanged and screwed joints may be used, where necessary and permissible by Code, to provide essential take-down capability for maintenance, testing, flushing or refueling operation. Welding is performed in accordance with the applicable codes.

A.6.1 CLEANING, PROTECTION AND SHIPMENT

To minimize the requirements for cleaning after erection, and to prevent damage during shipment, storage or handling, piping and equipment pressure parts are shop cleaned and sealed or otherwise protected from contamination or damage during shipping and handling.

A.7 AN EVALUATION OF PIPE BREAKS OUTSIDE THE REACTOR BUILDING

In accordance with criteria as outlined in "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," and to the Errata Sheet thereto, the following systems with changes as summarized below are considered high energy systems and are discussed in the following sections:

A.7.1 Main Steam (MS) - Added bumpers in steam line floor and wall penetrations to limit pipe movement in the event of a postulated rupture. Installed shields around the service water lines in the main steam pipe tunnel and the steam line to the “B” Train ADV and the turbine driver on the emergency feedwater pump. Evaluation of the 8 inch line to the atmospheric dump valves is included in this section.

A.7.2 MS to Emergency Feedwater Pump Turbine Driver - Added restraints at the isolation valve tee to prevent common failure of redundant steam supplies. Applied relaxation from arbitrary intermediate pipe rupture per NRC Generic Letter 87-11.

A.7.3 Main Feedwater - Added bumpers in the penetration room to prevent lines from "backing out" in the event of a guillotine break at the reactor building penetration. Installed flow restricting annuli at the reactor building penetrations to limit blowdown from a postulated pipe break at the reactor building penetration and maintain pressure below compartment allowable. Modified door in the south piping room to allow for quick opening to insure adequate vent area. Strengthened door to the electrical equipment room to protect safety related electrical equipment against pipe whip. Installed rupture disc in exhaust line from emergency feedwater pump turbine driver.

A.7.4 Reactor Coolant Letdown Line - Modified doors in the north piping room and north piping penetration rooms to allow for quick opening to insure adequate vent area in the event of a postulated break in this line.

A.7.5 Steam Generator and Pressurizer Sample Lines - Line sizes are less than one inch, therefore only a pipe crack is postulated.

A.7.6 Decay Heat Removal - Based on level of quality control, periodic inservice inspection, low usage factor, the short time the system contains high energy fluid, and the strict administrative control when system is in use, a break in this system is not considered credible.

The subsections of each of the following six sections correspond to the 21 questions in the above mentioned document. Several subsections in Section A.7.1 will serve as general discussions for the following respective subsections. Where required, those subsections will be expanded.

A.7.1 MAIN STEAM (MS)

Each steam generator is connected with a single 36-inch pipe line to the steam header near the turbine. The two steam lines penetrate the reactor building at elevations of 428 and 436 feet, respectively, in a non-Category 1 area of the auxiliary building. They make a short horizontal run in parallel and then turn downward. The vertical run is next to, but outside of, the Category 1 structure. At Elevation 345', the lines make a horizontal run out to the turbine building. Normal operating conditions for the main steam lines (and ADV lines up to the block valves) are 925 psia @ 598.2 °F at the OTSG outlet.

A.7.1.1 Pipe Whip

The main steam lines and ADV lines normally operate at conditions above 275 psig and 200 °F, and therefore an evaluation is provided for pipe whip following a longitudinal or circumferential break. The 8-inch ADV line is normally pressurized up to the ADV block valves and is therefore considered a high energy line requiring evaluation.

A.7.1.2 Criteria for Pipe Break Location

The postulation of arbitrary intermediate break locations may not be required if NRC Generic Letter 87-11 is implemented. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Pipe breaks are postulated, or alternatively addressed, to occur at the following locations:

A. Terminal ends.

B. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed $0.8(S_h + S_A)$ (S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code, Section III, Winter 1972 Addenda, or later appropriate ASME Section III Code sections provided that they have been reconciled. S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967, or later appropriate ASME Section III Code sections provided that they have been reconciled.) or the expansion stresses exceed $0.8\ S_A$.

C. Two additional intermediate locations are selected on the following bases when NRC Generic Letter 87-11 is not applied to the system piping:

   1. The points of highest stress or as needed to provide protection,

   2. No break in a short pipe run (up to five pipe diameters).

D. A critical crack, defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width, is postulated to occur at any location when NRC Generic Letter 87-11 is not applied to the system piping or Generic Letter 87-11 stress limits are exceeded.

A.7.1.3 Criteria for Pipe Break Orientation

A longitudinal pipe break is considered for lines four inches and larger. The break is assumed to be parallel to the pipe axis and oriented at any point around the pipe circumference. A circumferential break is considered for lines greater than one inch. The break is assumed to be oriented perpendicular to the pipe axis. A critical crack is assumed to be oriented at any point around the pipe circumference.

A.7.1.4 Summary of Pipe Whip Dynamic Analysis

A.7.1.4.1 Location and Number of Breaks

The locations and number of design basis breaks are chosen in accordance with the criteria in Section A.7.1.2. Two types of breaks, longitudinal and circumferential, are considered in accordance with the criteria in Section A.7.1.3.

The critical crack is considered to occur anywhere on the line.

A.7.1.4.2 The Postulated Rupture Orientation

The longitudinal break is parallel to the pipe axis and oriented at any point around the pipe circumference. The longitudinal break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction opposite the forcing function. The circumferential break is perpendicular to the pipe axis and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from a circumferential break are assumed to separate the piping axially and cause whipping in the direction opposite the forcing function.

The critical crack is oriented at any point around the pipe circumference.

A.7.1.4.3 Description of Forcing Function and Mathematical Model

The jet impingement force, caused by the momentum change of fluid flowing through the break, is a function of the upstream fluid conditions, fluid enthalpy, source pressure, pipe flow restrictional friction and dimensions, and target area.

The jet forces acting upon the pipe are computed using the method outlined in Bechtel's Topical Report, BN-TOP-2, Revision 2, "Design for Pipe Break Effects", or other commonly accepted methods such as those provided in ANSI/ANS-58.2, “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture”.

A large pipe break is assumed to be a one-time event, requiring a plant shutdown and necessary repairs. Permanent deformations of the pipe (and restraint) are allowed.

A.7.1.4.4 Analyses of Unrestrained Motion

If, following a single break in a required, evaluated pipe, the unrestrained movement of either end could impact required safety related equipment when rotated in the direction opposite the forcing function around a plastic hinge formed at the nearest suitable location, a potential hazard was assumed to exist for evaluation. The term “safety related equipment,” as appears in this evaluation, is defined as that equipment which is required to mitigate the consequences of a rupture in the evaluated pipe and bring the unit to a cold shutdown condition.

The probability of failure due to impact on safety related equipment was not considered credible if:

A. The safety related equipment was spatially separated from the break or hinge by a distance larger than the length of whipping pipe.

B. There was an interposition of suitable barriers which shielded the safety related equipment from the whipping pipe.

C. Piping of a certain size and wall thickness was impacted by a pipe of equal or smaller size and wall thickness.

D. The piping was suitably restrained.

E. It is impractical to erect whip restraints, and the weld at the postulated break location is required to be included in an augmented ISI inspection program providing 100% volumetric inspection during each inspection period as specified in Section XI of the ASME Code (reference section 3.6 of the Unit One SER).

F. The piping meets the break exclusion criteria of the NRC Branch Technical Position MEB 3-1 Contained in Standard Review Plan 3.6.2, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment”.

The main steam and ADV lines run outside the Class 1 areas and are separated from safety related equipment by Class 1 walls. The control room is separated by sufficient distance from the main steam lines to eliminate the possibility of damage due to pipe whip. No damage, therefore, can occur to structures, systems, or components important to plant safety due to a main steam line break.

A.7.1.5 Protection Against Pipe Whip, Jet Impingement and Reactive Force

A.7.1.5.1 Pipe Restraints

Based on the criteria for pipe break location and orientation (Sections A.7.1.2 and A.7.1.3) and the results of the analyses of unrestrained motion of a ruptured steam line, (Section A.7.1.4.4) no additional pipe restraints were considered necessary. However, bumpers are installed in the existing floor slab and wall penetrations of the main steam line to act as guides and to further limit motion of a ruptured steam line in the direction normal to the pipe axis.

A.7.1.5.2 Protective Provisions

Shields will be installed around the service water lines in the steam pipe tunnel and around the steam line to the “B” Train ADV and the turbine driver on the emergency feedwater pump to protect these lines from the jet impingement due to a postulated critical crack in the main steam line. These shields are shown in Figures A-9 and A-10.

A.7.1.5.3 Separation of Redundant Features

A postulated break in the main steam line will not cause any failure of redundant systems or components required to mitigate the consequences of that break. Consequently, no provisions are necessary to further separate piping and other components of redundant features.

A.7.1.5.4 Typical Pipe Whip Restraints

Not applicable.

A.7.1.5.5 Inservice Inspection Techniques

In accordance with previously accepted NRC criteria (reference section 3.6 of the Unit One SER, and NRC Branch Technical Position MEB 3-1 contained in Standard Review Plan 3.6.2), protection against postulated pipe ruptures is afforded in special cases by special inservice inspection (ISI) efforts on the critical welds. The specific requirements are 100% inspection of the welds during each inspection period as specified in ASME Code, Section XI.

A.7.1.6 Evaluation of Category 1 Structures

A.7.1.6.1 Method of Evaluating Stresses

Category 1 existing and added structures were evaluated for structural adequacy following a postulated rupture using the design bases shown in Section 5.1. Ultimate strength design method for concrete was used.

A.7.1.6.2 Allowable Design Stress

Design stresses are proportioned such that the combined stresses are within the limits established in Section 5.1.

A.7.1.6.3 Load Factors and Load Combinations

Load factors and load combinations are discussed in Section 5.1.

A.7.1.7 Structural Design Loads

The following design loads are used to evaluate the adequacy of Class 1 structures following a postulated rupture:

A. Dead Load - Actual weight of structural elements supported.

B. Live Load - Maximum expected live load in the area under consideration.

C. Equipment Load - Actual static load of equipment.

D. Pipe Load - Maximum calculated forces expected under normal operating and upset conditions. The forces include dead load, seismic forces, and thermal forces.

E. Pressurization - The maximum expected compartment pressure buildup that would result from a postulated rupture.

F. Jet Impingement - Jet impingement forces resulting from full pipe area breaks and critical cracks were considered.

G. Temperature - The effects of temperature increase from a pipe rupture are considered to be short-term increases and will not affect the structure adequacy.

H. Seismic Forces - Seismic forces as shown in Section 5.1.

The above loads are combined and evaluated in accordance with the criteria established in Section 5.1.

A.7.1.8 Reversal of Loads on the Structure

The forces which could cause reversal of loadings, and which may occur concurrently, due to the postulated accident on the Seismic Category 1 structural components are:

A. Compartment Pressurization

B. Jet Impingement Force

C. Pipe Whip

The main steam lines and ADV lines run outside the Category 1 structures in areas where large amounts of vent space are available except in the steam line tunnel where, in accordance with pipe break location criteria, only critical cracks are postulated to occur. Consequently, reversal of loads on Category 1 structural elements due to compartment pressurization will not occur.

The Category 1 wall, past which the steam lines run, has been analyzed and found capable of withstanding the jet impingement force from a postulated full area steam line break.

Based on the criteria for pipe break location and orientation (Sections A.7.1.2 and A.7.1.3) and the results of the analyses of unrestrained motion of a ruptured steam line (Section A.7.1.4.4), no reversal of loads on Category 1 structural elements due to pipe whip from a main steam line or an ADV line will occur.

A.7.1.9 Structural Effects of Openings

No additional openings are required to mitigate the consequences of a postulated rupture of a main steam line or ADV line.

A.7.1.10 Effect of Structural Failure

There will not be a failure of any structure, including Category 2 (non-Seismic Category 1) structures, due to the accident that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident and

B. Capability to bring the unit to a cold shutdown condition.

A.7.1.11 Verification That Pipe Rupture Will Not Affect Safety

The only equipment required to mitigate the consequences of a steam line break which is not separated from the main steam line by at least one Category 1 wall are the two AC-operated steam admission valves (and their associated conduits), which pass steam to the turbine driver on one of the emergency feedwater pumps, and the two main service water headers, which cross the main steam lines in the steam line tunnel. Several pressure sensing lines to the MSIS/EFIC pressure transmitters are also located in the area of these lines.

The 4-inch line, which passes steam to the turbine driver of one of the emergency feedwater pumps, are located in close proximity (approximately 20 feet below the nearest main steam line). It is located outside the path of a whipping steam line and sufficiently far away such that

the jet impingement force is negligible. The piping is restrained to prevent loss of steam supply from the unaffected steam line due to the broken steam line pulling both 4-inch lines. This is discussed in Section A.7.2.5.

The 8-inch line from the south steam generator main steam line passes within 18 inches of the main steam line from the north steam generator. A protective barrier has been installed around the 8-inch line to protect it from the jet impingement force due to a critical crack in the main steam line (See Figure A-10).

Postulated pipe ruptures of the main steam and atmospheric dump lines could conceivably result in failure of MSIS sensing lines; however, in each case emergency feedwater is available from the motor driven EFW pump or from the steam driven turbine through the unaffected steam supply line.

The steam pipe tunnel is crossed by a safeguards pipeway just before the pipe tunnel reaches the turbine building. This safeguards pipeway contains the two main service water headers. These service water headers are necessary for safe shutdown of the plant. Steel plate barriers have been installed to protect the service water lines from jet impingement due to a critical crack in the main steam lines (see Figure A-9).

In order to provide additional protection for postulated breaks in the main steam and ADV lines, the critical welds are included in the ANO-1 ISI program as described in Section A.7.1.5.5.

A.7.1.12 Effect on Control Room

The control room and its associated ventilation ductwork is located sufficiently far away from the steam lines so as to be unaffected by pipe whip, jet impingement, or steam environment resulting from a postulated break in either of the main steam lines. The control room will remain habitable and provide capability for safe shutdown and cooldown of the plant in the event of a main steam line break.

A.7.1.13 Environmental Qualification of Safety-Related Electrical Equipment Inside the Reactor Building

During the time frame in which ANO-1 was designed, equipment procured, and constructed, no general environmental qualification standards existed. Specifically, the bulk of ANO-1 design and procurement was completed previous to the issuance of IEEE 323-1971. As such, the licensing requirements for ANO-1 did not require conformance to any specific environmental qualification requirements or the qualification of designated post-accident monitors.

However, engineering practice during this period considered environmental qualification, although not necessarily to the degree of sophistication we now place on qualification based on our evolution on appropriate learning curves.

Environmental qualification requirements for ANO-1 equipment was specified on the purchase order specification when the equipment was procured. Engineering judgment was the basic rule which prevailed when determining qualification requirements, due to the lack of industry standards, etc. As such, we interpret the ANO-1 environmental qualification requirements as those which were required on the purchase order specification.

In responding to IE Bulletin 79-01 for ANO-1, AP&L researched their purchase orders and reported the specifications found there as the "Environmental Qualification Requirements."

Since that time, Entergy Operations, Inc. has implemented a detailed Environmental Qualification Program in accordance with the requirements of 10CFR50.49. The program is described in detail in Engineering Standard NES-01, "EQ Program." All components determined to be "Important-to-Safety" as stipulated by 10CFR50.49 have been evaluated and their environmental qualification has been documented.

A.7.1.14 Design Diagrams and Drawings

Figure 10-1 shows the main steam line system flow diagram. The physical routing of the main steam lines in relation to general plant arrangement is shown in plant design drawings (see Table 1-5). Isometrics of the main steam lines and ADV lines showing stress values at different locations are provided in plant design drawings (see Table 1-5) and Tables A-1, A-1A, A-2, and A-2A.

A.7.1.15 Flooding

The main steam lines and ADV lines are routed outside of Class 1 areas and separated from safety related equipment by Class 1 walls. Moisture, separated from escaping steam or formed by condensation on a cold surface, can be adequately handled by the plant drainage system. Any excess water produced in the steam line tunnel is channeled out into the turbine building. There is no possibility of flooding safety related equipment in the event of a main steam or ADV line failure.

A.7.1.16 Quality Control

A description of the quality control and inspection programs at Unit 1 is provided in Section 1.6.

A.7.1.17 Leak Detection

Following a steam line rupture the operator would note the following indications:

A. Rapid decrease in steam generator pressure.

B. Reactor trip.

C. Turbine trip.

D. Rapid drop in reactor coolant temperature and pressure.

E. Loud noise in affected area.

F. Low pressurizer level.

G. Safety injection system actuation.

H. Main steam and main feedwater isolation valves close on low SG pressure.

Depending on the size of the leak, the indications listed above may or may not be received. If these indications are not received after a small leak has occurred, the operator would note the leak on his rounds in the auxiliary building and turbine building. The operator would then evaluate the need to shut down the plant.

A.7.1.18 Emergency Procedures

Plant emergency procedures include appropriate actions to be taken by the operator in the event of a steam line supply system major break accident. These actions include those necessary to verify that automatic actuation of the Reactor Protection System (RPS), Turbine Trip System, Emergency Feedwater Initiation and Control (EFIC/MSIS), and ESAS have been accomplished and those necessary to place the plant in the normal post trip mode 3 condition and ultimately the cold shutdown condition.

A.7.1.19 Seismic and Quality Classification

The main steam header piping (24″ and 36″) between the main steam block valves and the high pressure turbine stop valves, and the B31.1.0 portion of the main steam piping system (8″) between the atmospheric dump valve isolation valves and the dump valves themselves, is designed and constructed to meet ANSI B31.1.0 requirements with 100 percent volumetric examination of welds. The portion that penetrates the reactor building out through the main steam block valves is designed and constructed to meet the requirements of ANSI B31.7, Class II, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The main steam line from the steam generator outlet to the turbine was analyzed and found adequate to withstand seismic loadings.

The ADV lines up to the normally closed block valves are designed and constructed to meet the requirements of ANSI B31.7, Class 2, or later appropriate ASME Section III Code sections provided that they have been reconciled. Downstream of the block valves, the piping is designed and constructed to meet the requirements of ANSI B31.1.0, though this portion is not a high energy line.

A.7.1.20 Description of Assumptions, Methods and Results of Analysis for Pressure and Temperature Transients in Compartments

For the purposes of environmental qualification the Main Steam Line Break was re-analyzed as described in ANO Environmental Qualification - Environmental Service Conditions, Engineering Standard NES-13. The following described analysis was performed in accordance with the original HELB analysis.

With the exception of the steam line tunnel, the main steam lines are routed outside of Category 1 structures and in regions where there is sufficient volume and vent area to preclude compartment overpressurization.

A critical crack can be postulated to occur in either main steam line inside of the steam pipe tunnel.

The mass and energy release data for the critical crack was computed using the Moody correlation for frictionless flow which yielded the following results:

Mass discharge rate, lbm/sec	–	24
Enthalpy (average), Btu/lbm	–	1223

This data was analyzed using the following technique, which was developed to study the pressure transients in compartments resulting from the depressurization of the primary coolant, steam, and feedwater systems. The analysis is based on the solution of the conservation equations for mass and energy, the continuity equation, and the equation of state for water.

The analysis permits the user to select the control volume and flow path configuration that results in the best representation of the pressure transients in the compartments along the flow paths from the break to the containment or the atmosphere. During each calculation time increment, the time dependent blowdown data is added to compartment 1. The program then solves the conservation equations and the state equations for water to determine the stagnation properties in the compartment. The compressible fluid flow equation is then solved for each flow path to determine the flow during the time increment from the compartment to each connected compartment. The conservation and state equations are again solved to determine the compartment state conditions for the beginning of the next time increment.

The results of the analysis show a peak pressure of 0.9 psig occurring at 0.5 seconds after the crack and a peak temperature in excess of 390 °F occurring at five seconds after the crack, assuming it occurs in the worst compartment. The concurrent jet loading is approximately 12.2 kips. The walls and ceiling of the pipe tunnel have been structurally analyzed and found able to withstand this internal pressure concurrent with jet impingement forces.

A.7.1.21 <u>Integrity of Containment Structure</u>

Since a LOCA concurrent with a steam line break is not considered credible, local reactor building structural integrity is not required for safe shutdown of the plant. A whipping main steam line could conceivably impact the reactor building in the area of the steam line penetrations. In this event, some cracking of the reactor building may be expected but significant structural failure will not occur.

A.7.2 MS TO EMERGENCY FEEDWATER PUMP TURBINE DRIVER

Two 4-inch lines, one from each train of main steam, provide steam as a driving force to the turbine driven emergency feedwater pump. The lines exit the main steam atmospheric dump lines, and run vertically parallel approximately 15 feet apart.

Near the floor slab at Elevation 404', each line has a normally open AC isolation valve, past which a check valve separates the line from a common header. There is a bypass valve around CV-2667 and its check valve. The bypass valve is normally locked closed and is not used for any normal operations. Two normally closed DC isolation valves and two normally closed DC bypass valves separate the high energy portion of the header from the single 4-inch pipe, which connects to the steam-turbine.

A.7.2.1 <u>Pipe Whip</u>

The 4-inch lines upstream of the DC-powered isolation valves normally operate at conditions above 275 psig and 200 °F and, therefore, evaluation is required for pipe whip following a longitudinal or circumferential break.

A.7.2.2 <u>Criteria For Pipe Break Location</u>

The criteria is discussed in Section A.7.1.2. Terminal end pipe break criteria are discussed in Section A.7.1.5.5.

A.7.2.3 Criteria for Pipe Break Orientation

The criteria is discussed in Section A.7.1.3.

A.7.2.4 Summary of Pipe Whip Dynamic Analysis

A.7.2.4.1 Location and Number of Breaks

The criteria for locating breaks and their orientations is discussed in Sections A.7.1.2 and A.7.1.3. The specific locations for breaks in these lines are discussed in Section A.7.2.2.

A.7.2.4.2 The Postulated Rupture Orientation

Same as that discussed in Section A.7.1.4.2.

A.7.2.4.3 Description of Forcing Function and Mathematical Model

Same as that discussed in Section A.7.1.4.3.

A.7.2.4.4 Analyses of Unrestrained Motion

The criteria for the analyses are the same as that discussed in Section A.7.1.4.4.

The 4-inch lines run in the area of the reactor building and the main steam lines and unrestrained motion of a broken 4-inch line will not damage either of these. One 4-inch line is considered incapable of damaging the other 4-inch line. The 4-inch line routed through the Auxiliary Building has no un-addressed regions of pipe rupture that would lead to unrestrained motion.

A.7.2.5 Protection Against Pipe Whip, Jet Impingement and Reactive Force

A.7.2.5.1 Pipe Restraints

Pipe restraints are installed upstream of the AC isolation valve in each line at Elevation 406'. These restraints will restrain motion of the common piping in the event of a main steam line whipping and pulling the 4-inch line with it.

A.7.2.5.2 Protective Provisions

These are discussed in Section A.7.1.5.2.

A.7.2.5.3 Separation of Redundant Features

As a result of installation of the restraints at Elevation 406', breaks in 36-inch main steam lines or 8-inch atmospheric dump lines will not cause a loss of the pressure boundary of the common piping. Breaks in the 4-inch lines are considered incapable of damaging the reactor building, the 8-inch atmospheric dump lines, or the 36-inch main steam lines. The only other safety related components located in this area are those related to the steam supply to the turbine driven EFW pump and MSIS instrument tubing.

A.7.2.5.4 Typical Pipe Whip Restraints

Pipe whip restraints are installed as discussed in Section A.7.1.5.1.

A.7.2.6 Evaluation of Category 1 Structures

A.7.2.6.1 Method of Evaluating Stresses

Same as that discussed in Section A.7.1.6.1.

A.7.2.6.2 Allowable Design Stress

Same as that discussed in Section A.7.1.6.2.

A.7.2.6.3 Load Factors and Load Combinations

Same as that discussed in Section A.7.1.6.3.

A.7.2.7 Structural Design Loads

Same as that discussed in Section A.7.1.7.

A.7.2.8 Reversal of Loads on Structures

The criteria is the same as that discussed in Section A.7.1.8. The 4-inch lines which pass main steam to the turbine driven emergency feedwater pump are relatively short lines which are located above the Category 1 auxiliary building in areas where large volumes and vent space are available. They are located in the same general area as the main steam lines. Consequently, any nearby structures are designed for the much more severe requirements imposed by the main steam lines.

A.7.2.9 Structural Effects of Openings

No additional openings have been created to mitigate the consequences of a break in either of these lines.

A.7.2.10 Effects of Structural Failure

There will not be a failure of any structure, including non-Seismic Category 1 structures, due to the accident that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident and

B. Capability to bring the unit to a cold shutdown condition.

A.7.2.11 Verification That Pipe Rupture Will Not Affect Safety

The only equipment required to mitigate the consequences of a break in the 4-inch lines from the main steam lines (and located near these lines) are the isolation valves and their associated cabling at Elevation 406' tee. These are protected from pipe whip by the installed restraints and are not considered susceptible to damage from a critical crack in the 4-inch lines. The terminal end welds located in the Auxiliary Building Room 46 are considered non-credible to pipe rupture through augmented ISI examination discussed in Section A. 7.1.5.5.

A break in the 4-inch line is within the capacity of the main feedwater pumps; therefore, safety systems actuation (i.e. reactor trip) is not required.

MSIS pressure sensing lines are also located in the vicinity of the postulated breaks; consequently, it is conceivable that these lines could fail, thus resulting in a reactor trip. The consequences of this failure have been evaluated and are considered acceptable since emergency feedwater is still assured from the motor driven EFW pump, which is unaffected by this scenario. Furthermore, safe shutdown can be accomplished via auxiliary feedwater or High Pressure Injection.

A.7.2.12 Effect on Control Room

The lines located in the area of the main steam lines above Elevation 406' are considered incapable, by virtue of their location, of affecting the control room or its supporting systems. The pipeline routed through the Auxiliary Building Rooms 83, 47, and 46 are located sufficiently far from the Control room to not affect it or its supporting systems.

A.7.2.13 Environmental Qualification of Affected Required Equipment

Same as that discussed in Section A.7.1.13 except that a postulated break in the 4-inch steam line produces a much less severe environment than a postulated break in the 36-inch main steam line.

A.7.2.14 Design Diagram and Drawings

The isometric and physical routing of the 4-inch lines are shown in plant drawings detailing pipe route. Refer to isometric drawings for current configuration.

A.7.2.15 Flooding

These lines are routed outside Category 1 areas and the potential for flooding of safety related equipment is non-existent. The terminal end of the piping is in the Auxiliary Building Room 46 and remains bounded by the Main Feedwater flooding model.

A.7.2.16 Quality Control

Same as that discussed in Section A.7.1.16.

A.7.2.17 Leak Detection

Same as that discussed in Section A.7.1.17.

A.7.2.18 Emergency Procedures

Plant emergency procedures include appropriate actions to be taken by the operator in the event of a break in steam supply lines from the Main Steam System to the emergency feedwater pump turbine.

A.7.2.19 Seismic and Quality Classification

The 4-inch steam lines to the turbine driven emergency feedwater pump down to valves CV-2666 and CV-2667 are designed and constructed to meet the requirements of ANSI B31.7, Class 2, or later appropriate ASME Section III Code sections provided that they have been reconciled and are designed as Category 1 (seismic) systems. The 4-inch steam lines from valves CV-2666 and CV-2667 to the turbine driven Emergency Feedwater pump are designed and constructed to meet the requirements of ANSI B31.1.0, Seismic Category 1. The seismic requirements for Category 1 (seismic) systems are described in Section 5.1.

A.7.2.20 Description of Assumptions, Methods and Results of Analyses for Pressure and Temperature Transients in Compartments

Because these lines are located in large, open areas with large amounts of vent space (Auxiliary Building Rooms 83 and 170), no compartment analysis was done for a postulated line break. In accordance with Generic Letter 87-11 limits for pipe stress allowables combined with augmented weld ISI examination for terminal end welds, no compartment analysis was done for Auxiliary Building Rooms 46 and 47 for a postulated line break.

A.7.2.21 Integrity of Containment Structure

The 4-inch lines from the main steam lines are not considered capable of damaging the reactor building.

A.7.3 MAIN FEEDWATER

The main feedwater lines carry water from the condenser through the Purification System, the low pressure heaters, the main feedwater pumps, the high pressure heaters, and through the south piping penetration room to the steam generators inside the reactor building. All piping upstream of the main feedwater pumps is located in the non-Category 1 turbine building. The piping downstream of the pumps is routed through non-Category 1 areas of the turbine auxiliary building except for the south piping penetration room which is a Category 1 area. The section of these lines from the high pressure heaters to the reactor building penetrations have been analyzed and found adequate to withstand seismic loadings.

A.7.3.1 Pipe Whip

The main feedwater lines normally operate at conditions above 275 psig and 200 °F and, therefore, protection is provided for pipe whip following a longitudinal or circumferential break.

A.7.3.2 Criteria for Pipe Break Location

Main feedwater (MFW) piping outside containment penetration is ASME Section III, Class 2, Seismic Class I up to the containment isolation valves (CV-2630 on loop A and CV-2680 on loop B). The MFW piping beyond these valves is B31.1, non-Seismic Category 1. In-line anchors are added on MFW piping upstream of isolation valves to provide seismic separation between safety related and non-safety related sections.

For the portions of MFW piping between containment penetrations and the new anchors, pipe breaks are postulated to occur at:

A. Terminal ends.

B. Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed $0.8(S_h + S_A)$ (S_h is the stress calculated by rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code, Section III, Winter 1972 Addenda. S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967) or the expansion stresses exceed 0.8 S_A.

Based on the NRC Generic Letter 87-11 and Entergy Operations Letter NED-90-0262, arbitrary intermediate breaks will not be postulated for this portion of the MFW piping.

For conservatism, piping between the containment penetrations and the isolation valves will follow the same criteria as stated above instead of the more relaxed criteria permitted by the Branch Technical Position MEB 3-1, Rev. 2.

Tables A-4 and A-5 list the stress values at different locations on the MFW lines up to the new anchors. Figures A-14 and A-15 identify the locations of stress points and the postulated pipe break locations.

For piping upstream of the new anchors, breaks are postulated to occur anywhere. This portion of the MFW piping is non-safety related, non-seismic B31.1 piping.

A critical crack, defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width, is postulated to occur at any location.

A.7.3.3 <u>Criteria for Pipe Break Orientation</u>

Same as that discussed in Section A.7.1.3.

A.7.3.4 <u>Summary of Pipe Whip Dynamic Analysis</u>

A.7.3.4.1 <u>Location and Number of Breaks</u>

The criteria for location of breaks and their orientation is discussed in Sections A.7.3.2 and A.7.3.3.

A.7.3.4.2 <u>The Postulated Rupture Orientation</u>

Same as that discussed in Section A.7.1.4.2.

A.7.3.4.3 <u>Description of Forcing Function and Mathematical Model</u>

Same as that discussed in Section A.7.1.4.3.

A.7.3.4.4 <u>Analyses of Unrestrained Motion</u>

The criteria for the analysis is the same as that discussed in Section A.7.1.4.4.

The MFW piping upstream of the new anchors is located in the turbine building and non-Seismic Category 1 sections of the turbine auxiliary building. As stated in A.7.3.2, breaks are postulated to occur anywhere for this portion of the MFW piping. The turbine building contains large volume and vent areas and no safety related equipment. No safety hazard would be created by a break in this section of the MFW piping. The main feedwater piping in the turbine auxiliary building is separated from the safety related equipment by at least one Category 1 wall. As described in Section A.7.3.5.2, the door to the electrical equipment room has been strengthened to protect safety related electrical equipment against a break in this section of the main feedwater piping.

Postulated break points for the main feedwater lines in the piping penetration room are located at the reactor building penetration in each line. As discussed in Section A.7.3.5, bumpers and flow restricting devices have been installed to prevent unrestricted motion of the main feedwater line in the event of a postulated pipe rupture.

The exhaust line from the emergency feedwater turbine driver passes within a few feet of the main feedwater lines at Elevation 380'. The possibility of the main feedwater lines impacting this exhaust line and pinching it shut has been considered. As discussed in Section A.7.3.5.2, a rupture disc has been installed in the exhaust line below the Elevation 368' floor slab to provide an alternate vent path in the unlikely event of this occurrence.

A.7.3.5 Protection Against Pipe Whip, Jet Impingement and Reactive Force

A.7.3.5.1 Pipe Restraints

As shown in Figure A-16, bumpers are installed on each main feedwater line at the first elbow away from the penetration in the south piping penetration room. These bumpers prevent the pipe from "backing out" in the event of a postulated circumferential break at the reactor building penetration. As further protection, each line is restrained outside of the penetration room to prevent the severed line from slipping off the bumper.

A.7.3.5.2 Protective Provisions

Flow restricting annuli are installed around the main feedwater line reactor building penetrations to limit the blowdown from a postulated pipe rupture at the penetration (circumferential or longitudinal) to that which would result from a critical crack. These annuli will also serve to limit pipe movement in the lateral directions following a postulated pipe rupture. These are shown in Figure A-16.

The piping penetration room is protected from overpressure by the installation of Door 493 which allows for a vent path to the main chiller room. Door 493 is installed to open under a slightly positive pressure to provide sufficient vent area to maintain the pressure and temperature in the penetration room within allowable limits. Door 19 into the EFW equipment room has to be latched to isolate this room from the direct steam flow of the MFW HELB. As an alternate, Door 594 into the penetration room isolates the EFW equipment room from direct steam flow or MFW HELB.

A rupture disc has been installed in the emergency feedwater pump turbine driver exhaust line just below the Elevation 368' floor slab. This will provide an alternate vent path in the unlikely event that the exhaust line is pinched closed due to impact by a main feedwater line. This rupture disc is located in a non-Class 1 area which contains sufficient volume and vent area to accept the steam exhaust. There is no safety related equipment located in this area.

The door to the electrical equipment room has been modified to open outward and has been strengthened to withstand the pipe whip and jet impingement force due to a postulated rupture in the main feedwater piping surrounding the high pressure heaters.

A.7.3.5.3 Separation of Redundant Features

Based on the analyses of unrestrained motion of the main feedwater piping (discussed in Section A.7.3.4.4) and the additional modifications installed (discussed in Sections A.7.3.5.1 and A.7.3.5.2), no further provisions for separation or redundant features are required.

A.7.3.5.4 Typical Pipe Whip Restraints

The bumpers and flow restricting annuli which have been installed to limit pipe movement are shown in Figure A-16.

A.7.3.6 Evaluation of Category 1 Structures

A.7.3.6.1 Method of Evaluating Stresses

Same as that discussed in Section A.7.1.6.1.

A.7.3.6.2 Allowable Design Stress

Same as that discussed in Section A.7.1.6.2.

A.7.3.6.3 Load Factors and Load Combinations

Same as that discussed in Section A.7.1.6.3.

A.7.3.7 Structural Design Loads

Same as that discussed in Section A.7.1.7.

A.7.3.8 Reversal of Loads on Structures

The criteria are the same as that discussed in Section A.7.1.8. The ceiling of the south piping penetration room forms the floor of the south electrical penetration room immediately above it. The south electrical penetration rooms contain cabling which is required to function following a main feedwater line break. The slab between the two rooms will begin to lift at an internal pressure of 1.29 psig in the piping penetration room. As discussed in Section A.7.3.20, the internal pressure in the piping penetration room will not exceed 0.95 psig in the event of a main feedwater line break.

A.7.3.9 Structural Effects of Openings

No additional openings have been created to mitigate the consequences of a postulated main feedwater line break.

A.7.3.10 Effect of Structural Failure

There will not be a failure of any structure, including non-Seismic Category 1 structures, due to the accident, that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident and

B. Capability to bring the unit to a cold shutdown condition.

A.7.3.11 Verification That Pipe Rupture Will Not Affect Safety

Equipment which must function following a main feedwater line break includes the emergency feedwater pump, the emergency feedwater isolation valves, and some electrical cabling in the electrical penetration room.

The emergency feedwater pumps located in Room 38 are not rendered inoperable by the main feedwater HELB since Room 38 is not in the direct steam vent path.

The emergency feedwater block valves utilize qualified motorized valve actuators.

As discussed in Sections A.7.3.8 and A.7.3.20, the pressure in the piping penetration room due to a main feedwater line break will not reach a level sufficient to lift the ceiling and allow steam into the electrical penetration room.

A.7.3.12 Effect on Control Room

The control room and associated ventilation ductwork is located sufficiently far away from the main feedwater lines so as to be unaffected by pipe whip, jet impingement, or steam environment resulting from a postulated break in any main feedwater system pipe. The control room will remain habitable and provide capability for safe shutdown and cooldown of the plant in the event of a main feedwater line break.

A.7.3.13 Environmental Qualification of Affected Required Equipment

A.7.3.13.1 List of Equipment Required to Function

This is discussed in Section A.7.3.11.

A.7.3.13.2 Testing

This is discussed in Section A.7.3.11.

A.7.3.13.3 Impingement Barriers

No impingement barriers are required to protect electrical equipment.

A.7.3.13.4 Control Room

A break at any of the postulated locations has no effect on the control room environment.

A.7.3.13.5 Onsite Power

The high temperature steam and water are prevented by walls from propagating into the switchgear, battery, and Emergency Diesel Generator rooms. Although some relatively lower temperature steam/air mixture enters the battery room area via ventilation ductwork, the onsite power sources and distribution systems will remain operable.

A.7.3.14 Design Diagrams and Drawings

Figure 10-1 shows the main feedwater system flow diagram. Isometrics of the main feedwater lines showing stress values at different locations are provided in Figures A-14 and A-15 and Tables A-4 and A-5.

A.7.3.15 Flooding

With the exception of the piping penetration room, the main feedwater lines are routed in open non-Category 1 areas and are separated from safety related equipment by Category 1 walls.

Water and steam released from a postulated break in the main feedwater line inside the piping penetration room will flow out through Door 493 in the piping penetration room. After 30 minutes of MFW blowdown there will be less than 2 inches of water on the floor of Room 77. A majority of the break flow is entrained in the steam/air/vapor mixture being exhausted. The less than 2 inches of water does not represent a flooding problem. Also, after 30 minutes of MFW blowdown, the room below the piping penetration room will have less than 2 inches of water on the floor via the spiral stair case separating Rooms 77 and 46. This amount of water does not represent a flooding problem for Room 46.

A.7.3.16 Quality Control

Same as discussed in Section A.7.1.16.

A.7.3.17 Leak Detection

The method of detecting a leak in the main feedwater line is the same as that discussed in Section A.7.1.17. A small gap exists between the insulation on the piping and the flow restricting annuli at the reactor building penetrations of the main feedwater piping. This will enable the operator to visually detect a small leak inside the main feedwater piping penetration.

A.7.3.18 Emergency Procedures

Same as those discussed in Section A.7.1.18.

A.7.3.19 Seismic and Quality Classification

The main feedwater lines are designed and constructed to meet ANSI B31.1.0 requirements, except for the portion that penetrates the reactor building out through the main feedwater isolation valves, which is designed and constructed to meet the requirements of ANSI B31.7, Class 2, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The main feedwater line from just downstream of the high pressure heaters to the steam generator inlet has been analyzed and found adequate to withstand seismic loadings.

A.7.3.20 Descriptions of Assumptions, Methods and Results of Analyses for Pressure and Temperature Transients in Compartments

With the exception of the south piping penetration room, the main feedwater lines are routed outside of Category 1 structures in regions where there is sufficient volume and vent area to preclude compartment overpressurization.

As seen in Tables A-4 and A-5 and Figures A-14 and A-15, a break can be postulated at the reactor building penetration in each line. These breaks would release high pressure water and steam into the penetration room.

Critical flow was used for the blowdown mass flow rate. The Henry-Fauske critical flow correlation was used. No credit was taken for frictional losses in the pipe that has ruptured. The flow area of the annulus is 3.1 square inches. The mass flow rate and statepoint data for the blowdown are presented in Table A-6.

These data were analyzed as described in Section A.7.1.20 using a model of the relevant portions of the Auxiliary Building.

The vent path through Door 493 located in the upper south piping penetration room allows the MFW HELB mass and energy to vent out of the auxiliary building into the main chiller room. This vent path assumes that Door 19 is latched, which prevents any direct steam flow into the EFW equipment room. As an alternate to Door 19, Door 594 prevents any direct steam flow into the EFW equipment room. The maximum temperature and pressure in the upper south piping penetration room is 215 °F and 0.95 psig. The structural integrity of Room 77 is qualified up to 1.29 psig. Therefore, the postulated MFW HELB event will not cause any structural failures.

A.7.3.21 Integrity of Containment Structures

As a result of modifications to the main feedwater piping in the penetration room, there is no possibility of the piping impacting the reactor building in the event of a postulated rupture. The reactor building penetrations are designed to withstand the effects of a full area break. Therefore, there will be no adverse effects on the reactor building due to a main feedwater line break.

A.7.4 REACTOR COOLANT LETDOWN LINE

The 2-1/2 inch reactor coolant letdown line runs from the reactor building into the north piping penetration room and then down to the north piping room. The pressure reducing orifice in this line is located in the piping room. The letdown line ceases to be a high energy line after the pressure reducing orifice.

A.7.4.1 Pipe Whip

The letdown line normally operates at 2,170 psig and 120 °F. However, in the event of a rupture in this line, flow would increase to such a rate that the letdown coolers would provide little cooling. The fluid temperature would increase to approximately 500 °F. Consequently, in the event of a rupture, the fluid in the letdown line would reach conditions above 275 psig and 200°F. Therefore, this line has been analyzed for the effects of pipe whip.

A.7.4.2 Criteria for Pipe Break Location

For the letdown line, breaks were postulated to occur anywhere.

A.7.4.3 Criteria for Pipe Break Orientation

Same as that discussed in Section A.7.1.3.

A.7.4.4 Summary of Pipe Whip Dynamic Analysis

A.7.4.4.1 Location and Number of Breaks

Breaks were postulated to occur anywhere in the piping between the reactor building penetration and the pressure reducing orifice.

A.7.4.4.2 The Postulated Rupture Orientation

In accordance with Section A.7.4.3, only circumferential breaks are postulated to occur. A discussion of circumferential breaks were postulated to occur anywhere and it was not necessary to consider critical cracks.

A.7.4.4.3 Description of Forcing Function and Mathematical Model

Same as that described in Section A.7.1.4.3.

A.7.4.4.4 Analysis of Unrestrained Motion

The criteria is the same as that discussed in Section A.7.1.4.4.

Because of its relatively small size, the letdown line is considered incapable of damaging any safety related piping in the north penetration room. It can, however, be considered capable of damaging valve operators in safety related situations. The only valve operators which could be damaged are one high pressure injection (HPI) isolation valve and one low pressure injection (LPI) valve. The loss of the operator on the HPI valve will not result in loss of redundancy in the HPI System. Damage to the valve operator in the LPI System will result in a momentary loss of redundancy in the LPI System. However, since the LPI System is not required to mitigate the consequences of a letdown line break until a considerable period of time has elapsed after the rupture, the operator will have more than enough time to bring the plant to a safe condition and manually adjust the valve before the LPI System is required to function.

DCP 89-1012B added four (4) new redundant lines in the lower north piping room. An evaluation of the effects of a letdown line break in that room indicates that only one valve operator could sustain damage as a potential target. The above mentioned evaluation documents that complete redundancy exists for that operator and the HPI safety function would still be provided in the event of a HELB on the letdown line. Therefore the effects of such a HELB in this area would be mitigated.

A.7.4.5 Protection Against Pipe Whip, Jet Impingement and Reactive Force

Based on the analysis for unrestrained motion presented in Section A.7.4.4.4, no pipe restraints are required.

A.7.4.5.1 Protective Provisions

The doors on the north piping room and piping penetration room have been modified for fast opening at a small positive pressure. These will provide sufficient vent area to maintain pressures within allowable limits in the event of a letdown line break.

A.7.4.5.2 Separation of Redundant Features

There is sufficient separation of redundant features as described in Section A.7.4.4.4.

A.7.4.5.3 Typical Pipe Whip Restraints

Not applicable.

A.7.4.6 Evaluation of Evaluating Stresses

A.7.4.6.1 Method of Evaluating Stresses

Same as discussed in Section A.7.1.6.1.

A.7.4.6.2 Allowable Design Stress

Same as that discussed in Section A.7.1.6.2.

A.7.4.6.3 Load Factors and Load Combinations

Same as that discussed in Section A.7.1.6.3.

A.7.4.7 Structural Design Loads

Same as discussed in Section A.7.1.7.

A.7.4.8 Reversal of Loads on Structure

The criteria is the same as that discussed in Section A.7.1.8. The structural components of the north piping room and the north piping penetration room have been analyzed for reversal of loads due to a postulated rupture in the reactor coolant letdown line. The results of the analyses show that structural integrity of these components will not be compromised.

A.7.4.9 Structural Effects of Openings

No new openings have been created to mitigate the consequences of a postulated reactor coolant letdown line break.

A.7.4.10 Effect of Structural Failure

There will not be a failure of any structure, including non-Seismic Category 1 structures, due to the accident that could cause failure of any other structure in a manner to adversely affect:

A. Mitigation of the consequences of the accident and

B. Capability to bring the unit to a cold shutdown condition.

A.7.4.11 Verification That Pipe Rupture Will Not Affect Safety

The analysis of unrestrained motion of the pipe, presented in Section A.7.4.4.4, and the analyses of structures and structural components, including the analysis for compartment over-pressurization presented in Section A.7.4.20, show that, in the event of a reactor coolant letdown line break, the ability of the plant to cope with that break will not be impaired.

A.7.4.12 Effect on Control Room

The control room is located sufficiently far away from the reactor coolant letdown line so as to be unaffected by pipe whip, jet impingement, or steam environment resulting from a postulated rupture of the letdown line. The control room will remain habitable and provide capability for safe shutdown and cooldown of the plant in the event of a letdown line rupture.

A.7.4.13 Environmental Qualification of Affected Required Equipment

A.7.4.13.1 List of Equipment Required to Function

The HPI and LPI System isolation valves located in the north piping penetration and the lower north piping rooms are required to function following a postulated rupture of the letdown line. Additional information on the letdown line break analysis is described in ANO Environmental Qualification - Environmental Service Conditions, Engineering Standard NES-13.

A.7.4.13.2 Testing

These valves are motor operated valves whose electric motor operators are qualified per the environmental qualification program.

A.7.4.13.3 Impingement Barriers

No impingement barriers are required to protect electrical equipment.

A.7.4.13.4 Control Room

A break at any of the postulated locations has no effect on the control room environment.

A.7.4.13.5 Onsite Power

The high temperature steam is prevented by walls from propagating into the switchgear, battery, and Emergency Diesel Generator rooms. Although some relatively lower temperature steam/air mixture enters the battery room area via ventilation ductwork, the onsite power sources and distribution systems will remain operable.

A.7.4.14 Design Diagrams and Drawings

Figure 9-3 shows the reactor coolant letdown line flow diagram.

A.7.4.15 Flooding

The letdown line is routed through the north piping penetration room and north piping room. Each of these doors is equipped with a fast opening door which will provide both vent area and relief for flooding water. In addition, there are floor drains in each of these rooms. In each case the water is channeled into larger areas capable of accommodating it. Under no circumstances will water levels rise high enough to impair the functions of safety related equipment.

A.7.4.16 Quality Control

Same as discussed in Section A.7.1.16.

A.7.4.17 Leak Detection

A temperature sensor, located downstream of the pressure reducing orifice, causes automatic closure of the outside reactor building isolation valve on high fluid temperature. Should this element fail to function, the operator would receive indications of low pressurizer level and low reactor coolant pressure. In the event of a full area break in the letdown line, an engineered safeguards signal automatic closure of both the inside and outside reactor building isolation valves occurs in approximately 30 to 40 seconds.

A.7.4.18 Emergency Procedures

In the event of a letdown line break, actions taken by the operator generally will include:

A. Manually trip the reactor and turbine.

B. Isolate the reactor coolant letdown line.

C. Manually initiate safety injection, if required.

D. Follow the emergency boration procedures, if required.

E. Conduct a cooldown using normal plant procedures, if required.

Note that if no operator action is taken, steps A, B, and C will occur automatically within 30 to 40 seconds after the break. These steps will be accomplished by the Engineered Safeguards (ES) and Integrated Control Systems (ICS).

A.7.4.19 Seismic and Quality Classification

The reactor coolant letdown line is designed and constructed to meet the requirements of ANSI B31.7, Class 2, or later appropriate ASME Section III Code sections provided that they have been reconciled. Seismic Category 1 for that section of pipe that is considered a high energy line.

A.7.4.20 Description of Assumptions, Methods and Results of Analyses for Pressure and Temperature Transients in Compartments

As discussed earlier, the high energy section of the letdown line is routed through the north piping and north piping penetration rooms where breaks are assumed to occur anywhere in the line. The postulated breaks would release high energy water and steam into either of these rooms.

The mass and energy release data for a circumferential break in the letdown line was analyzed using the Moody correlation and an fl/d value of 1.5 which yielded the following results:

Mass discharge rate, lbm/sec	-	374 Enthalpy (average),
Btu/lbm	-	555.8

This data was then analyzed as described in Section A.7.1.20. The volume of the piping room is 24,000 ft^3 and the volume of the penetration room is 17,100 ft^3. The two rooms are not connected. Each room contains a 21 ft^2 door which has been modified to provide fast opening at a small positive pressure. Each door provides vent area for its respective compartments.

Two cases were run for each room, one assuming no vent area and the other assuming that the door was open at the beginning of the transient. For the no vent area cases, the pressure was found to be steadily increasing at rates of 2.8 psi per second and 4.0 psi per second in the piping and penetration rooms, respectively. With the assumption that the doors are open at the beginning of the transient, the volumes of the two subcompartments will affect the rates at which each pressurizes but the steady state pressure of each compartment will be identical if identical vent conditions exist. The peak pressure in the penetration room (worst case) is 1.0 psig and occurs at 0.75 seconds. The maximum temperature at 1.0 psig will be 215 °F, the saturation pressure.

The doors to each of these rooms have been modified to provide for rapid opening at a small positive pressure (less than 0.25 psig). The piping room and piping penetration room have been structurally analyzed and found able to withstand these internal pressures and concurrent jet forces.

In addition to these analyses, the letdown line break was reanalyzed for environmental qualification purposes as described in ANO Environmental Qualification - Environmental Service Conditions, Engineering Standard NES-13. The re-analysis provides additional analytical detail not afforded by the original analysis described above.

A.7.4.21 <u>Integrity of Containment Structure</u>

Since the reactor coolant letdown line is only 2 1/2 inches in diameter, any forces acting on the reactor building structure will not be significant enough to affect its integrity.

A.7.5 STEAM GENERATOR AND PRESSURIZER SAMPLE LINES

The 3/4-inch steam generator and pressurizer sample lines exit the reactor building and pass through the north piping penetration room en route to the sample room. The sample room is located just to the north of the penetration room. Each sample line contains both an internal and external reactor building isolation valve. The external isolation valve is located in the sample room. All reactor building isolation valves can be operated from the control room.

The reactor building isolation valves are normally closed so that the sample lines can be considered high energy lines only when sampling is actually in progress. During this time, the sample lines are being constantly monitored.

A.7.5.1 <u>Pipe Whip</u>

In accordance with the AEC criteria, pipe breaks in lines one inch nominal pipe size and smaller need not be considered. Therefore no pipe whip will occur.

A.7.5.2 <u>Criteria for Pipe Break Locations</u>

A critical crack, defined as one-half the pipe diameter in length and one-half the pipe wall thickness in width, is postulated to occur at any location.

A.7.5.3 <u>Criteria for Pipe Break Orientation</u>

A critical crack is assumed to be oriented at any point around the pipe circumference.

A.7.5.4 Summary of Dynamic Analysis

Not applicable.

A.7.5.5 Protective Measures

No additional protective measures are necessary to protect safety related systems, structures, and components from the jet impingement resulting from a critical crack in either of the sample lines.

A.7.5.6 Evaluation of Category 1 Structures

Not applicable.

A.7.5.7 Structural Design Loads

Not applicable.

A.7.5.8 Load Reversal Analysis

Not applicable.

A.7.5.9 Structural Effects of Openings

No new openings are required.

A.7.5.10 Effect of Structural Failure

No structures will fail.

A.7.5.11 Verification That Pipe Rupture Will Not Affect Safety

There will be no significant environmental consequences as a result of a crack in the sample system piping.

The sample lines are considered high energy lines only when sampling is actually being done. During this time the sample lines are being constantly monitored. Both the inside and outside reactor building isolation valves are remotely operable from the control room.

A.7.5.12 Effect on Control Room

The control room is located sufficiently far away from the sample line so as to be unaffected by a critical crack in either of these lines.

A.7.5.13 Environmental Qualification of Affected Required Equipment

Any environmental consequences which may result from a critical crack in the sample lines will impose a much less severe design condition on safety related equipment than the consequences resulting from a full area break in the reactor coolant letdown line. (See Section A.7.4)

A.7.5.14 Design Drawings

The flow paths of the sample lines are shown in Figure 9-5.

A.7.5.15 Flooding

In view of the small blowdown rates associated with a critical crack in either sample line, no flooding of safety related equipment will occur.

A.7.5.16 Quality Control and Inspection Programs

Same as that discussed in Section A.7.1.16.

A.7.5.17 Leak Detection

Since the lines are being constantly monitored while in the conditions which qualify them as high energy lines, any leak in these lines will be immediately detected.

A.7.5.18 Emergency Procedures

In the event that a leak is detected in either of the sample lines the operator will be notified immediately and will close the isolation valves and isolate the leak.

A.7.5.19 Seismic and Quality Classification

The pressurizer sample lines are designed and constructed as Seismic Category 1 and meet all of the requirements of ANSI B31.7, Class 1, out to the outboard reactor building isolation valves. The steam generator secondary side sample lines, which are smaller than 2 1/2″, were constructed as Seismic Category II consistent with Regulatory Guide 1.29, paragraph 1f. This piping and the associated valves were designed in accordance with the requirements of ANSI B31.1.0. More recent modifications or additions may be conservatively designed as Seismic Category 1 or ASME Section III. For the latest information in this regard, refer to the current qualification documentation for the components of interest.

A.7.5.20 Descriptions of Assumptions, Methods and Results of Analyses for Pressure and Temperature Transients in Compartments

A critical crack in either sample line would impose a much less severe design condition on the north piping penetration room than a full area break in the reactor letdown line. (See Section A.7.4.20.)

A.7.5.21 Integrity of Containment Structures

Not applicable.

A.7.6 DECAY HEAT REMOVAL SYSTEM

In the case of the Decay Heat Removal System, the maximum temperature and pressure at which shutdown can be initiated is 300 psig and 280 °F. At this point, the reactor is in a safe condition, i.e., the Reactor Coolant System boron concentration is sufficient to preclude criticality. The Decay Heat Removal System is designed to withstand seismic loadings and is designed and constructed to meet the requirements of ANSI B31.7, Class 2, or later appropriate ASME Section III Code sections provided that they have been reconciled.

The decay heat removal mode is a manually initiated operation and is under strict administrative control during the entire cooldown period of approximately 25 hours. It is further noted that the system conditions are above 200 °F and 275 psig for less than 30 minutes.

Based on the level of quality control, periodic inservice inspection, the usage factor, the short time the system exceeds 200 °F and 275 psig, and the strict administration control of the decay heat removal operation, a break in this system is not considered credible.

Table A-1

STRESS VALUES - MAIN STEAM LINE FROM NORTH STEAM GENERATOR

DATA POINT	NEW DATA POINT	THERMAL STRESS (PSI)	OBE SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	WEIGHT STRESS (PSI)	TOTAL STRESS (PSI)
50**	5	1939	6893	7922	850	17604
51**	9	8622	11282	8178	1303	29385
52	10E	8284	7781	8178	1462	25705
53	55B	5653	4880	8178	760	19471
54	55E	5669	4701	8178	1166	19714
55	65B	7068	3863	8178	478	19587
56	65E	6997	4049	8178	673	19897
57**	69K	2623	3262	8178	711	14774
58	70E	2041	2770	8178	70	13059
59	70A	1877	3012	8178	309	13376
60	75E	2015	3033	8178	435	13661
61**	100	6206	3781	5725	483	16195
70**	85	5268	3704	5725	524	15221
71	89E	3385	2529	5725	757	12396
72	89B	1915	2098	5725	767	10505
73**	89B	1915	2098	5725	767	10505
74	90B	1375	2107	5725	868	10075
75**	98E	3063	1762	5725	1834	12384
76	98B	2005	2283	5725	461	10474
80**	105	3682	3171	5725	402	12980
81	110E	4366	5238	5725	586	15915
82	110B	4618	4467	5725	612	15422
83	115E	1308	3381	5725	1157	11571
84	115B	1655	3911	5725	616	11907
85	120	5386	4464	5725	655	16230

* INCLUDES STRESSES DUE TO RV THRUST FORCES AND SAM.

** POSTULATED BREAK-POINTS.

*** 0.8 (S_h + S_a) = 37,800. psi. for 36” piping; = 32,400. psi. for 24” piping.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

TABLE A-1A

STRESS VALUES - ADV LINE, NORTH STEAM GENERATOR

DATA POINT	THERMAL STRESS (PSI)	SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	WEIGHT STRESS (PSI)	TOTAL*** STRESS (PSI)
MS169	1,774	3,546	0	537	5,857
286	3,908	5,853	5,842	745	16,348*
283	684	4,147	5,842	760	11,433
282	1,336	6,123	5,842	1,309	14,610**
281	1,528	5,939	5,842	1,235	14,544*
MS164	4,585	1,645	5,842	372	12,444
276	9,963	2,696	5,842	657	19,158*
MS202	3,967	1,693	5,842	649	12,151
272	5,510	3,028	5,842	172	14,552*
MS171	2,689	2,078	5,842	123	10,732
MS163	4,700	2,946	5,842	86	13,574
268	6,825	3,826	5,842	141	16,634
266	5,882	3,243	5,842	148	15,115
264	3,981	2,517	5,842	179	12,519
262	3,009	4,166	5,842	213	13,230
260	3,421	7,511	5,842	395	17,169*
259	9,846	8,978	5,842	1,176	25,842*
MS2B	2,245	5,973	5,842	171	14,231**

* High stress points other than terminal ends.

** Postulated break-points.

*** $0.8 (1.2 S_h + S_A) = 32,400$ psi.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Table A-2

STRESS VALUES - MAIN STEAM LINE FROM SOUTH STEAM GENERATOR

DATA POINT	NEW DATA POINT	THERMAL STRESS (PSI)	OBE SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	WEIGHT STRESS (PSI)	TOTAL STRESS (PSI)
1**	P2	3496	4621	7922	1158	17196
2	9	7963	6005	8178	683	22829
3	10E	7283	7667	8178	861	23990
4	60	6053	3293	8178	510	18034
5	61E	7470	2703	8178	542	18893
6	74A	6679	3023	8178	523	18403
7	75E	6796	4367	8178	313	19654
8**	94A	2513	3827	8178	361	14878
9**	95E	1890	3294	8178	617	13978
10**	100	1796	2717	8178	490	13181
20**	105	650	4974	5725	544	11893
21**	115E	3989	3806	5725	253	13773
22**	115B	3880	3778	5725	200	13583
30**	135	5685	8477	5725	412	20300
31**	145E	916	3762	5725	387	10790
32**	145B	1838	2690	5725	118	10372
33**	160	2759	6648	5725	215	15348

* INCLUDES STRESSES DUE TO RV THRUST FORCES.

** POSTULATED BREAK-POINTS.

*** 0.8 (S_h + S_a) = 37,800. psi. for 36” piping; = 32,400. psi. for 24” piping.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

TABLE A-2A

STRESS VALUES - ADV LINE, SOUTH STEAM GENERATOR

DATA POINT	THERMAL STRESS (PSI)	SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	WEIGHT STRESS (PSI)	TOTAL*** STRESS (PSI)
78A	1,953	4,905	5,842	150	12,850**
76	4,669	8,142	5,842	168	18,821*
74	4,555	5,714	5,842	210	16,321
81	5,218	4,692	5,842	231	15,983
MS173	2,349	2,404	5,842	230	10,825
84	6,869	3,505	5,842	87	16,303
86	3,547	2,336	5,842	57	11,782
89	8,116	3,017	5,842	283	17,258*
91	7,512	3,162	5,842	559	17,075
MS203	3,227	2,839	5,842	610	12,518
94	7,204	3,740	5,842	309	17,095*
96	7,205	3,850	5,842	353	17,250*
MS168	3,986	4,865	5,842	294	14,987
98	3,112	4,845	5,842	835	14,634**
99	1,334	5,167	5,842	931	13,274
101	1,113	4,701	5,842	810	12,466
103	2,882	4,748	5,842	905	14,377
104	2,500	5,154	5,842	931	14,427**

* High stress points other than terminal ends.

** Postulated break-points.

*** $0.8 (1.2 S_h + S_A) = 32,400$ psi.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Table A-3

STRESS VALUES - 4-INCH LINES FROM MAIN STEAM LINES

DATA POINT	THERMAL STRESS (PSI)	SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	WEIGHT STRESS (PSI)	TOTAL*** STRESS (PSI)
255	6,169	7,228	5,222	829	19,448**
254	12,167	9,564	5,222	882	27,835*
253	11,321	7,850	5,222	663	25,056*
252	5,625	3,078	5,222	544	14,469
251	4,666	4,062	5,222	596	14,546
210	4,143	4,258	5,222	227	13,850
200	4,410	4,186	5,222	1,184	15,002*
170	8252	6,816	5,222	2,420	22,710
160	5,453	6,436	5,222	2,617	19,728
140	4,545	6,689	5,222	3,404	19,860*
130	6,261	7,322	5,222	3,930	22,735*
90	5,143	4,783	5,222	1,830	16,978*
80	6,681	3,940	5,222	455	16,298*
MS1	7,967	9,965	5,222	138	23,292**
500	10,923	6,462	5,222	1,380	23,987
510	12,009	7,151	5,222	1,466	23,848*
414	9,721	5,464	5,222	1,169	21,576
413	14,399	7,354	5,222	572	27,547*
411	11,534	5,943	5,222	785	23,484
410	10,551	7,613	5,222	2,283	24,903
760	10,241	4,252	5,222	585	20,300**
570	10,115	6,826	5,222	1,935	24,098**

* High stress points other than terminal ends.

** Postulated break-points.

*** 0.8 (S_h + S_A) = 32,400 psi.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Table A-4

STRESS VALUES - MAIN FEEDWATER TO NORTH STEAM GENERATOR

DATA POINT	WEIGHT* STRESS (PSI)	THERMAL STRESS (PSI)	OBE SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	TOTAL*** STRESS (PSI)
1	332	4,042	1,910	5,517	11,801**
2	308	11,328	1,632	5,517	18,785
3	1,023	10,587	2,603	5,517	19,730
4	627	14,818	680	7,292	23,417
5	100	12,948	653	7,292	20,993
6	99	12,832	652	7,292	20,875
7	215	5,127	517	7,292	13,151
8	144	9,746	610	7,292	17,792
9	127	16,226	1,100	7,292	24,745
10	235	19,283	1,356	7,292	28,166**

** Postulated break-points

*** $0.8 (S_h + S_A) = 30{,}000$ psi.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Table A-5

STRESS VALUES - MAIN FEEDWATER TO SOUTH STEAM GENERATOR

DATA POINT	WEIGHT* STRESS (PSI)	THERMAL STRESS (PSI)	OBE SEISMIC STRESS (PSI)	LONGITUDINAL PRESSURE STRESS (PSI)	TOTAL*** STRESS (PSI)
20	177	3,091	3,828	5,517	12,613**
21	262	7,226	4,778	5,517	17,783
22	470	7,014	4,258	5,517	17,259
23	859	15,235	2,961	7,292	12,347
24	1,225	14,261	1,913	7,292	24,691
25	342	12,536	1,833	7,292	22,003
26	395	9,813	2,536	7,292	20,036
27	370	9,183	4,130	7,292	20,975
28	286	11,293	4,822	7,292	23,693
29	397	10,628	5,747	7,292	24,064**

** Postulated break-points.

*** 0.8 (S_h + S_A) = 30,000 psi.

The stress values listed are historical in nature. For the latest information in this regard, see the current qualification documentation for the specific piping of interest.

Table A-6

BLOWDOWN DATA - MAIN FEEDWATER LINE BREAK

Critical-Crack Mass and Energy Release

Mass Flow	375 lbm/sec
Specific Enthalpy	446 Btu/lbm
Feedwater Pressure	1100 psia

A critical-crack of 30 minute duration was postulated to occur in a main feedwater line in Room 77. A critical-crack, as defined by the NRC is an opening in a pipe with an area equal to half the wall thickness times half the nominal pipe diameter.

Figures A-1 through A-8A

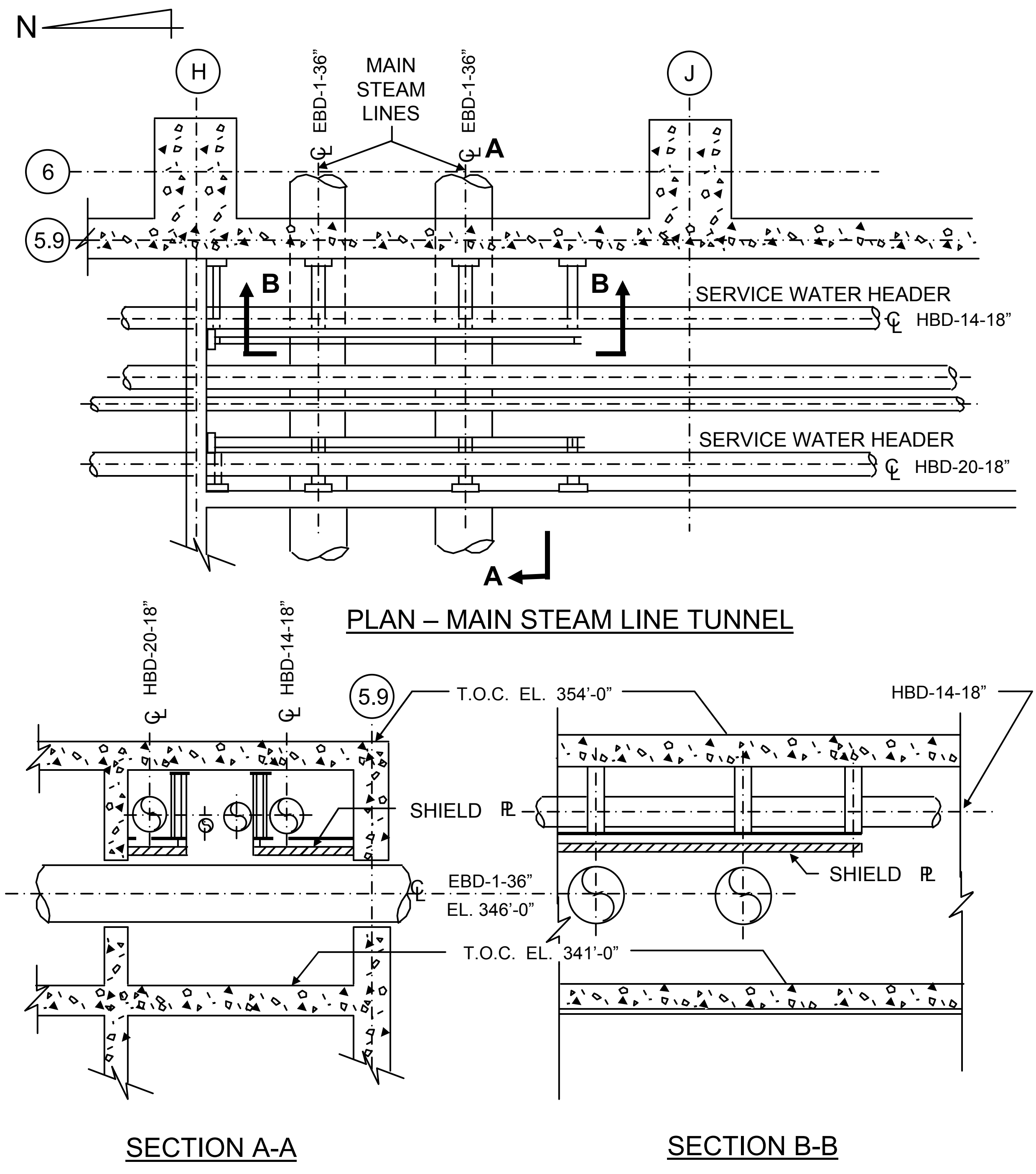
DELETED

SAR FIGURE NO. A-1

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

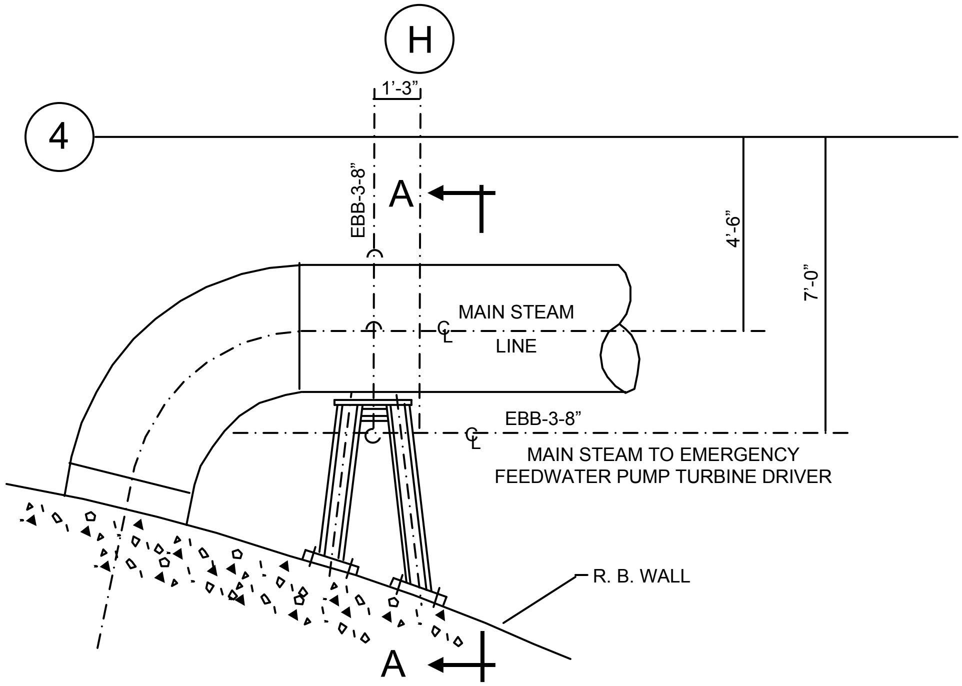
FUEL HANDLING FLOOR PLAN	BASED ON DRAWING NO	SHEET	REV.
		1	



SAR FIGURE NO. A-9

AMENDMENT 20

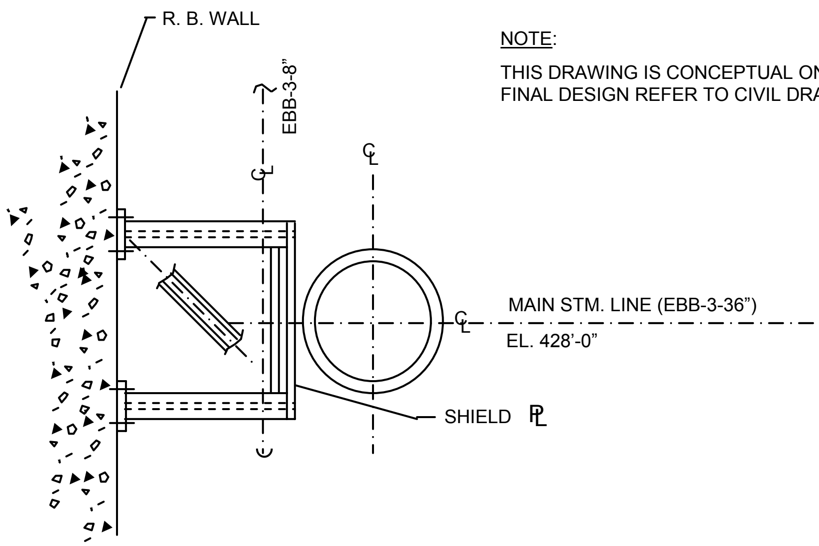
ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:		
PROTECTIVE BARRIER SERVICE WATER LINES	BASED ON DRAWING NO		SHEET 1	REV.



PLAN ABOVE EL. 428'-0"

NOTE:

THIS DRAWING IS CONCEPTUAL ONLY. FOR FINAL DESIGN REFER TO CIVIL DRAWING C-294.



SECTION A-A

SAR FIGURE NO. A-10

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:	
PROTECTIVE BARRIER – MAIN STEAM TO EMERGENCY FEEDWATER PUMP	BASED ON DRAWING NO	SHEET 1	REV.

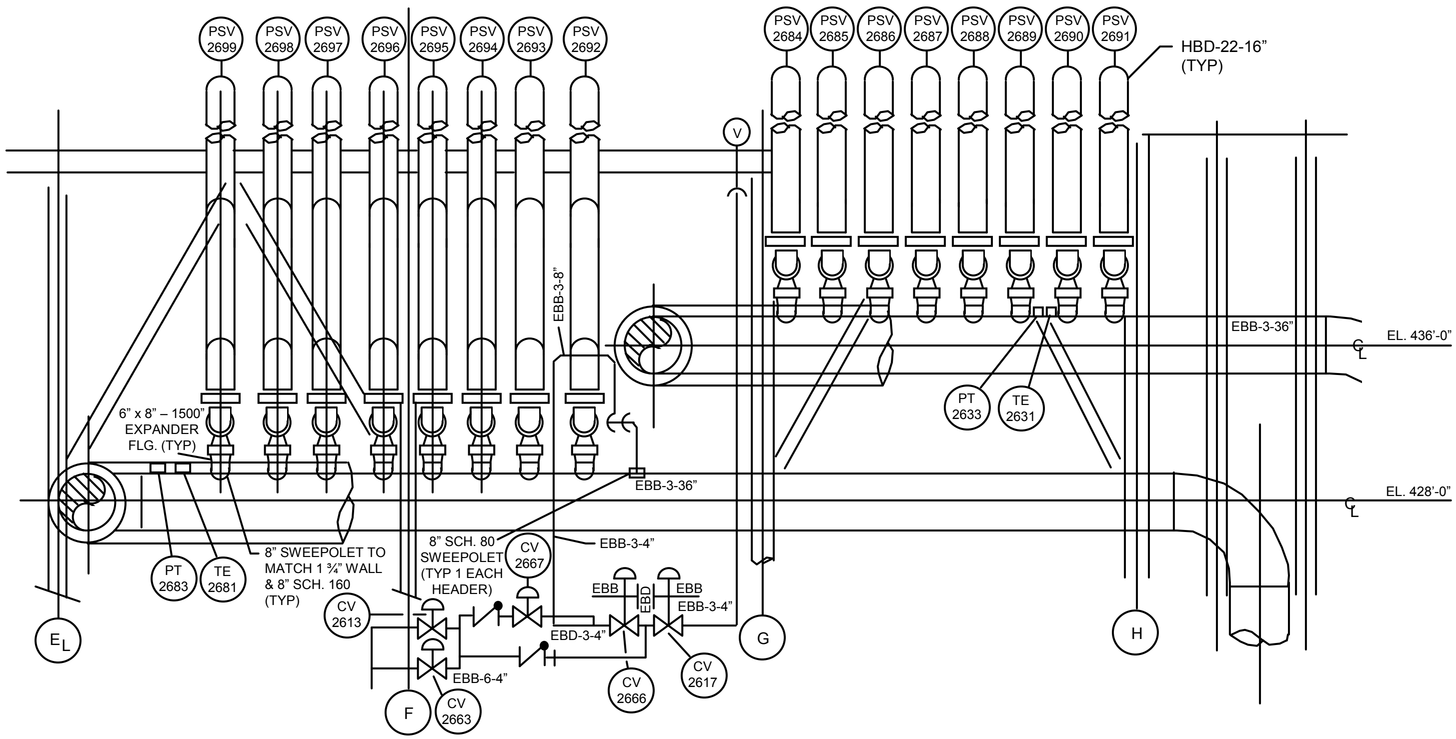
DELETED

SAR FIGURE NO. A-11

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

	BASED ON DRAWING NO	SHEET	REV.
MAIN STEAM TO ISOLATION VALVES UPSTREAM OF EMERGENCY FEEDWATER PUMP TURBINE K-3		1	



MAIN STEAM TO EMERGENCY FEEDWATER PUMP

SAR FIGURE NO. A-12

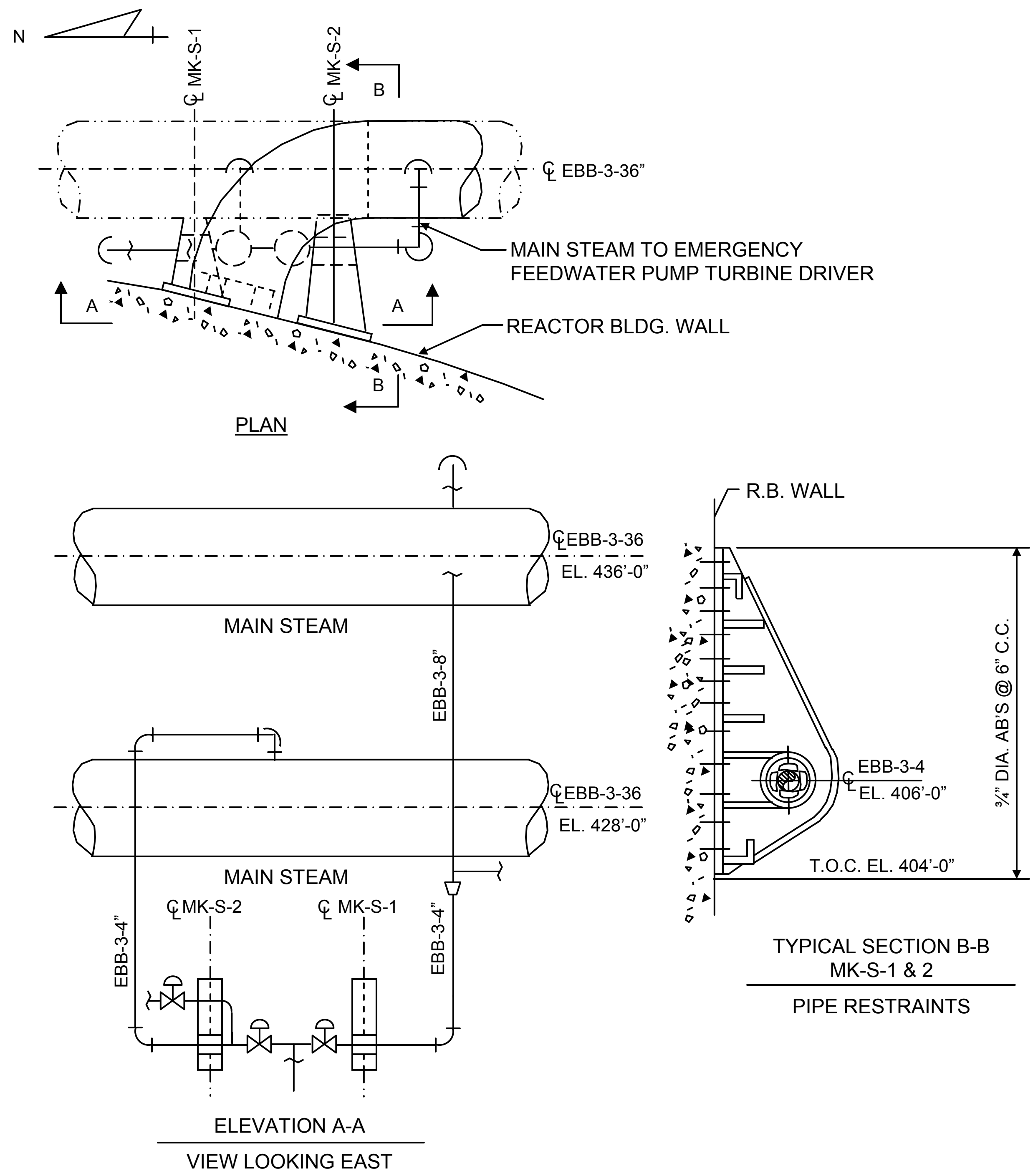
ARKANSAS NUCLEAR ONE
UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

AMENDMENT 20

BASED ON DRAWING NO	SHEET	REV.

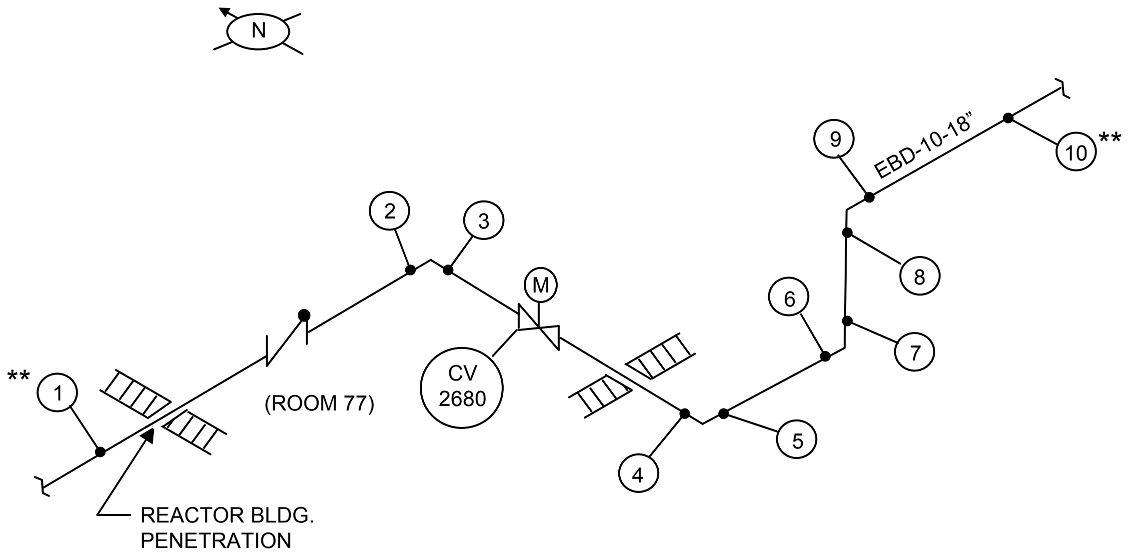


SAR FIGURE NO. A-13

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

RESTRAINTS ON 4 INCH STEAM LINES	BASED ON DRAWING NO	SHEET	REV.
		1	



** = Postulated Break Points

ISOMETRIC MAIN FEEDWATER TO NORTH STEAM GENERATOR

SAR FIGURE NO. A-14

AMENDMENT 20

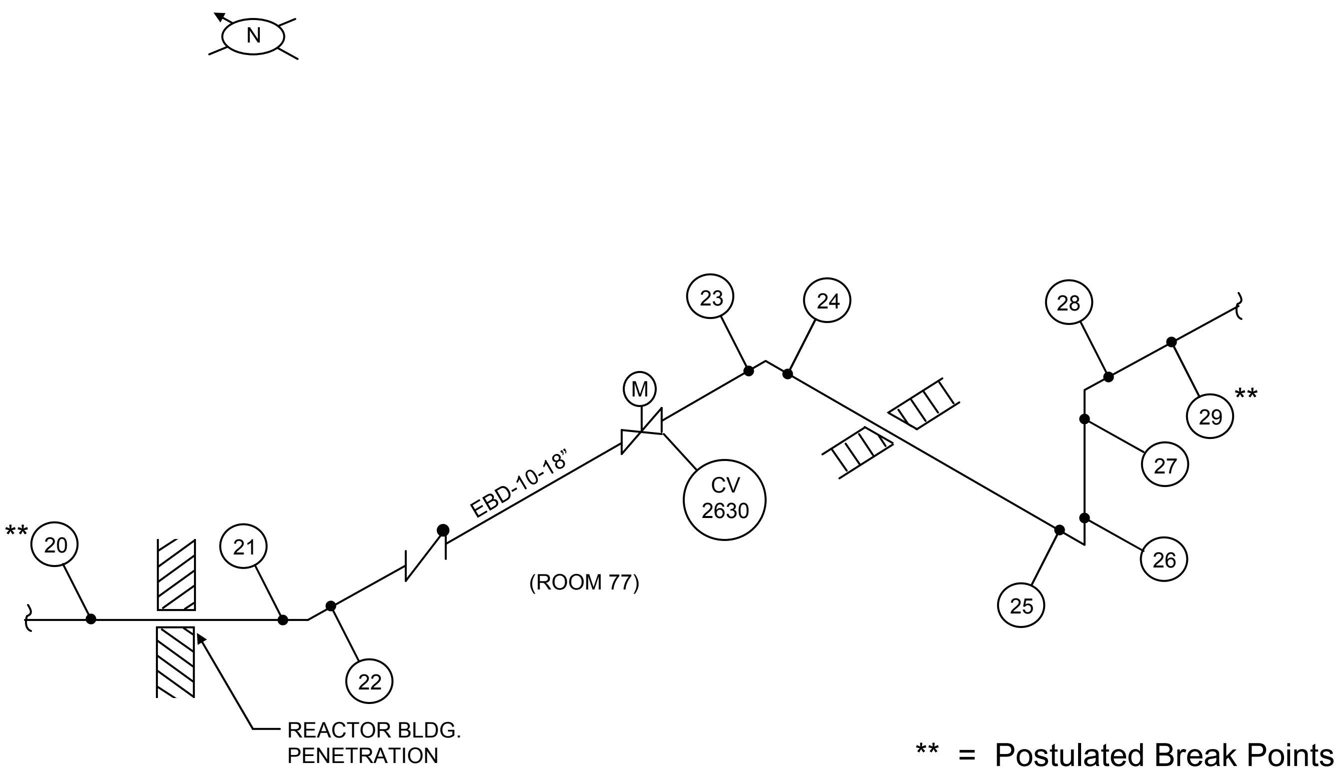
ARKANSAS NUCLEAR ONE

UNIT 1
RUSSELLVILLE, ARKANSAS

Entergy

SCALE:	NONE
DRAWN:	ENTERGY
DESIGN:	ENTERGY
CAD NO:	

BASED ON DRAWING NO	SHEET	REV.



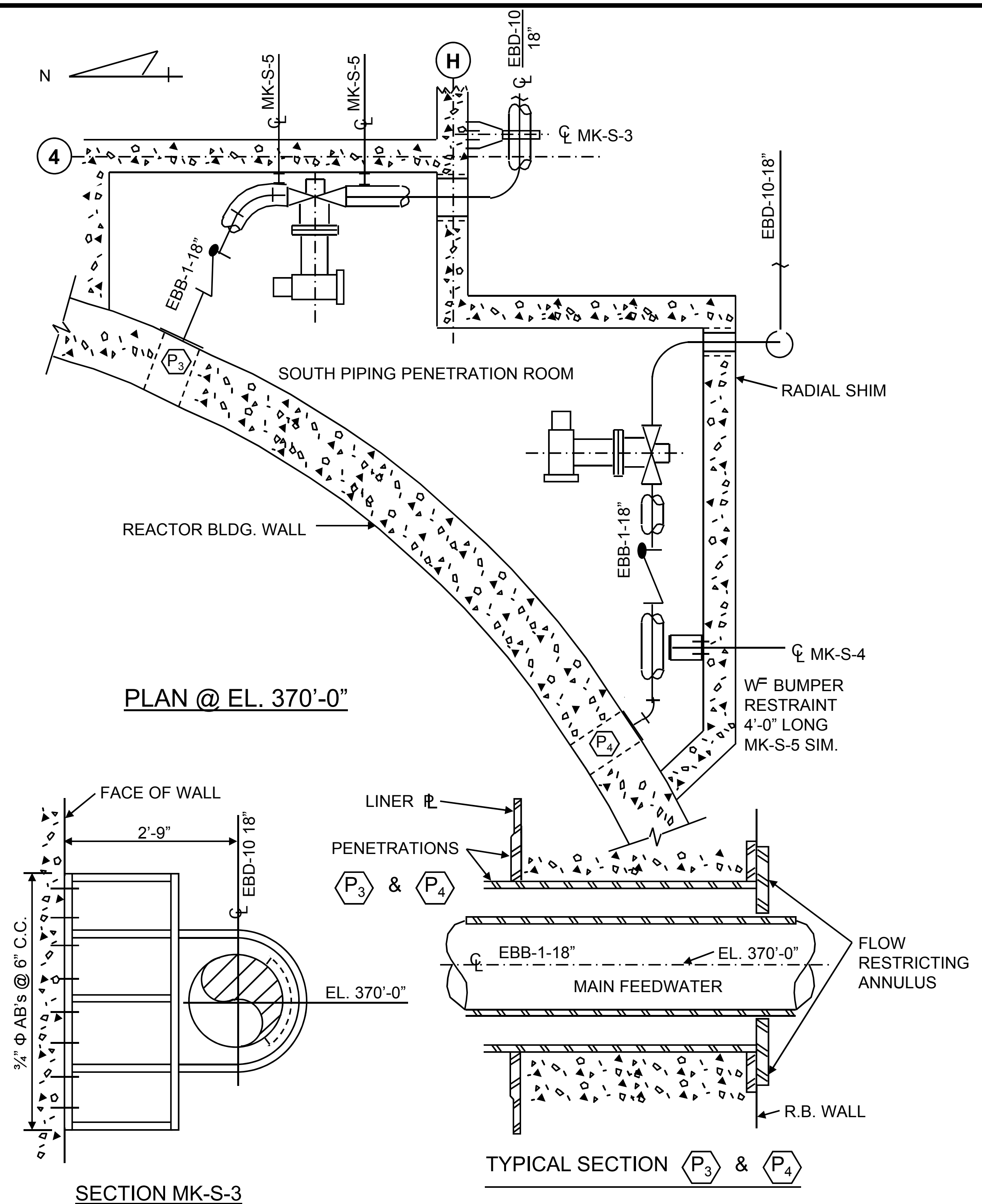
ISOMETRIC MAIN FEEDWATER TO SOUTH STEAM GENERATOR

SAR FIGURE NO. A-15

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: ENTERGY DESIGN: ENTERGY CAD NO:

BASED ON DRAWING NO	SHEET	REV.



SAR FIGURE NO. A-16

AMENDMENT 27

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE:	NONE
		DRAWN:	
		DESIGN:	ENTERGY
		CAD NO:	

MAIN FEEDWATER PIPING PROTECTIVE PROVISIONS	BASED ON DRAWING NO	SHEET	REV.
		1	

DELETED

SAR FIGURE NO. A-17

AMENDMENT 20

ARKANSAS NUCLEAR ONE UNIT 1 RUSSELLVILLE, ARKANSAS	Entergy	SCALE: NONE DRAWN: DESIGN: ENTERGY CAD NO:

GEOMETRIC MODEL OF SOUTH PIPING PENETRATION ROOM	BASED ON DRAWING NO	SHEET	REV.
		1	