11.0 RADIOACTIVE WASTE MANAGEMENT

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

The fission product inventory in the reactor core and the diffusion to the fuel pellet/cladding gap are presented in chapter 15. In this section two source terms are presented. The first is a conservative design base that utilizes a conventional fuel clad defect model. This design model serves as a basis for system and shielding requirements and calculations of the maximum offsite doses resulting from credible accidents.

The second source term is a realistic model used to predict expected long-term average concentrations of radionuclides in the primary and secondary fluid stream and an average plant's releases over its lifetime. This realistic model, based on available measured nuclide concentrations during normal operation, was formulated as a standard for the American National Standard Source Term Specifications, ANSI N237,[1] and is the source term model used in NUREG-0017.[2][3]

11.1.1 REACTOR COOLANT AND SECONDARY SIDE ACTIVITY

11.1.1.1 Design Basis Model

The parameters used in the calculation of the reactor coolant fission product inventories, together with the pertinent information concerning the expected coolant cleanup flowrate and demineralizer effectiveness, are summarized in table 11.1-1. The results of the calculations are presented in table 11.1-2. In these calculations the defective fuel rods are assumed to be present at the initial core loading. The fission product escape rate coefficients are based upon average fuel temperature and are further based on fuel defect tests performed at the Saxton reactor. Recent experience at two plants operating with fuel rod defects has verified the escape rate coefficients listed in table 11.1-1.

For further information on core fission product calculations see subsection 15.1.7.

For fuel failure and burnup experience see chapter 4.

The fission product activities in the reactor coolant during operation with small cladding defects (fuel rods containing pinholes or fine cracks) are computed using the following differential equations.

For parent nuclides in the coolant:

$$\frac{dN_{wi}}{dt} = D\nu_i N_{Ci} - (\lambda_i + R\eta_i + \frac{B'}{B_o - tB'}) N_{wi}$$

For daughter nuclides in the coolant:

$$\frac{dN_{wj}}{dt} = D\nu_j N_{Cj} - (\lambda_j + R\eta_j + \frac{B'}{B_o - tB'}) N_{wj} + \lambda_i N_{wi}$$

where:

N_o	=	Nuclide concentration.
D	=	Clad defects, as a fraction of rated core thermal power being generated by rods with clad defects.
R	=	Purification flow (coolant system volumes/s).
B_o	=	Initial boron concentration (ppm).
B'	=	Boron concentration reduction rate by feed and bleed (ppm/s).
η	=	Removal efficiency of purification cycle for nuclide.
λ	=	Radioactive decay constant (s^{-1}).
ν	=	Escape rate coefficient for diffusion into coolant.
t	=	Time (s).
C	=	Refers to core.
w	=	Refers to coolant.
i	=	Refers to parent nuclide.
j	=	Refers to daughter nuclide.

Table 11.1-3 lists the activities in the volume control tank using the assumptions summarized in table 11.1-1.

The activities in the pressurizer are separated between the liquid and the steam phase, and the results obtained are given in table 11.1-4 using the assumptions summarized in table 11.1-1.

The activities to be found in the gaseous waste processing system are given in table 11.1-5.

As a necessary part of the effort to reduce effluent of radioactive liquid wastes, Westinghouse has been surveying various pressurized water reactor (PWR) facilities which are in operation to identify design and operating problems influencing reactor coolant and nonreactor grade leakage and hence the load on the waste processing system.

Leakage sources have been identified in connection with pump shaft seals and valve stem leakage.

When packed glands are provided a leakage problem may be anticipated, while mechanical shaft seals provide essentially zero leakage. Valve stem leakage was experienced when the originally specified packing was used. A combination of graphite filament yarn packing sandwiched with asbestos sheet packing is used with improved results in several plants. A bellows seal being utilized in later plants eliminates all stem leakage.

In addition, seat leakage was experienced on some pressurizer power-operated relief valves. However, this was found to be due to a manufacturing error and has been corrected.

Current PWR design is based on a reactor coolant leakage of 20 gal/day/unit into the floor drain tank. Nonreactor grade leakage entering the floor drain tank from the containment and the auxiliary building is assumed to be 40 gal/day. In addition, an excessive reactor coolant leakage of 1 gal/min can be handled under abnormal operating conditions. Although leakage from the primary system to the secondary side in the steam generator is unlikely, secondary side fission product activity is evaluated by applying an expected 0.2 percent defective fuel and 20 gal/day primary to secondary side leakage.

Table 11.3-1 gives conservatively estimated leakages from the gaseous waste processing system with corresponding design activity discharges from the plant vent stack. The activity releases are based on a leakage of 100 sf^3/year with cladding defects in fuel rods generating 1 percent of the rated core thermal power. The leak rate is based on the sensitivity of commercially available portable leak detectors.

11.1.1.2 Realistic Model

The parameters used to describe the realistic model are given in table 11.1-6 together with the range of values utilized by ANSI N237-1976. Corrections have been made according to the ANSI N237-1976 standard formulas. Operation of a Westinghouse gaseous waste management system is assumed.

Regulatory Guide 1.112[4], appendix B, recommends input parameters needed to execute the gaseous and liquid effluents (GALE) computer code[2] for pressurized water reactors. These values are listed in table 11.1-7.

11.1.2 RADIOACTIVE RELEASE SOURCES

Total plant liquid and gaseous releases are discussed in subsections 11.2.6 and 11.3.6, respectively. Release pathways for gaseous effluents are described in subsection 12.2.2. Liquid release paths include the steam generator blowdown processing system, as described in subsection 10.4.8, and the turbine building drains. The turbine building sumps are periodically pumped out at a rate of approximately 800 gal/min per each pump. This flowrate is utilized to determine the total volume released from the turbine building sumps. The liquid release paths are shown in figure 9.2-1.

REFERENCES

1. American National Standard Source Term Specification, ANSI N237-1976/ANS-18.1, approved May 11, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," NUREG-0017, Office of Standard Development, April 1976.

3. Alabama Power Company letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," U.S. NRC Docket Nos. 50-348 and 50-364, June 3, 1976.

4. U.S. NRC Regulatory Guide 1.112, Revision O-R, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," April 1976.

TABLE 11.1-1 (SHEET 1 OF 3)

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES[c]

Parameter	Value
Reactor power (MWt)	2774
Clad defects, as a percent of rated core thermal power being generated by rods with clad defects	1.0[d]
Reactor coolant liquid volume (ft^3)	9723.3
Reactor coolant full power average temperature (°F)	577
Normal purification flowrate (gal/min)	60[d]
Effective cation demineralizer flow (gal/min)	6.0[d]
Volume tank volumes	
Vapor (ft^3)	175
Liquid (ft^3)	125
Fission product escape rate coefficients	
Noble gas isotopes (s^{-1})	6.5×10^{-8}
Br, I, and Cs isotopes (s^{-1})	1.3×10^{-8}
Te isotopes (s^{-1})	1.0×10^{-9}
Mo isotopes (s^{-1})	2.0×10^{-9}
Sr and Ba isotopes (s^{-1})	1.0×10^{-11}
Y, La, Ce, and Pr isotopes (s^{-1})	1.6×10^{-12}
Mixed bed demineralizer decontamination factors	
Noble gases and Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	1.0
All other isotopes including corrosion products	10.0

TABLE 11.1-1 (SHEET 2 OF 3)

Parameter	Value
Cation bed demineralizer decontamination factor for Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	10.0
Boron concentration and reduction rates	
B_o, initial cycle (ppm)	700
B', initial cycle (ppm/day)	2.5
B_o, equilibrium cycle (ppm)	1000
B', equilibrium cycle (ppm/day)	3.5
Pressurizer volumes	
Vapor (ft^3)	560
Liquid (ft^3)	840
Spray line flow (gal/min)	1.0
Pressurizer stripping fractions	
Noble gases	1.0
All other elements	0
Volume control tank noble gas and iodine stripping fractions	

Isotope	Stripping Fraction[a]
Kr-85	2.5×10^{-1}
Kr-85m	2.9×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}
Xe-131m	2.5×10^{-1}
Xe-133	2.5×10^{-1}
Xe-133m	2.6×10^{-1}
Xe-135	2.8×10^{-1}
Xe-138	8.0×10^{-1}

TABLE 11.1-1 (SHEET 3 OF 3)

Isotope	Partition Coefficient[(b)]
I-131	100
I-132	100
I-133	100
I-134	100
I-135	100

a. Fraction of inlet isotope concentration stripped off in the volume control tank.

b. Ratio between isotopic concentration in the liquid phase and isotope concentration in the vapor phase.

c. Reviewed for power uprate, RCS activities in table 11.1-2 remain bounding for power uprate

d. Evaluations of the impact of increasing letdown flow to 145 gpm (and cation demineralizer flow to 14.5 gpm) indicate this original combination of parameters continues to give conservatively high design basis source terms.

TABLE 11.1-2 (SHEET 1 OF 2)

REACTOR COOLANT EQUILIBRIUM FISSION AND CORROSION PRODUCT ACTIVITIES[a]

Isotope	Activity (μCi/g)
Br-84	4.4×10^{-2}
Rb-88	3.8
Rb-89	0.10
Sr-89	4.1×10^{-3}
Sr-90	1.4×10^{-4}
Sr-91	2.0×10^{-3}
Y-90	1.7×10^{-4}
Y-91	6.1×10^{-3}
Y-92	7.3×10^{-4}
Zr-95	7.0×10^{-4}
Nb-95	6.9×10^{-4}
Mo-99	5.5
I-131	2.5
I-132	0.9
I-133	4.0
I-134	0.6
I-135	2.2
Te-132	0.3
Te-134	3.0×10^{-2}
Cs-134	0.26
Cs-136	0.15
Cs-137	1.3
Cs-138	0.96
Ba-140	4.2×10^{-3}
La-140	1.4×10^{-3}
Ce-144	3.3×10^{-4}
Pr-144	3.2×10^{-4}
Kr-85	0.14
Kr-85m	2.1
Kr-87	1.3
Kr-88	3.6
Xe-131m	0.21
Xe-133	7.98×10^{1}
Xe-133m	1.5
Xe-135	5.7
Xe-135m	0.19
Xe-138	0.68
Mn-54[b]	7.8×10^{-4}
Mn-56[b]	2.9×10^{-2}
Co-58[b]	1.3×10^{-1}
Co-60[b]	7.5×10^{-4}

TABLE 11.1-2 (SHEET 2 OF 2)

Isotope	Activity (μCi/g)
Fe-59[b]	1.0×10^{-3}
Cr-51[b]	9.5×10^{-4}
Zn-65	8.0×10^{-3}

a. Based on parameters given in table 11.1-1.

b. Corrosion product activities based on activity levels measured at operating reactors.

TABLE 11.1-3

EQUILIBRIUM VOLUME CONTROL TANK ACTIVITIES[(a)]

Isotope	Vapor Activity (Ci)
Kr-85	4.3
Kr-85m	3.0×10^1
Kr-87	1.5×10^1
Kr-88	6.0×10^1
Xe-131m	4.5
Xe-133	2.24×10^3
Xe-133m	4.2×10^1
Xe-135	1.2×10^2
Xe-135m	0.35
Xe-138	1.3
I-131	0.0122
I-132	0.00448
I-133	0.0196
I-134	0.00266
I-135	0.0105

Isotope	Liquid Activity (Ci)
I-131	0.87
I-132	0.32
I-133	1.4
I-134	0.19
I-135	0.75

a. Based on parameters given in table 11.1-1.

TABLE 11.1-4

PRESSURIZER ACTIVITIES[a]

Isotope	Vapor Activity (μCi/cm^3)
Kr-85	1.5
Kr-85m	1.4×10^{-1}
Kr-87	2.6×10^{-2}
Kr-88	1.6×10^{-1}
Xe-131m	3.6×10^{-1}
Xe-133	1.4×10^{2}
Xe-133m	1.2
Xe-135	7.9×10^{-1}
Xe-135m	8.0×10^{-4}
Xe-138	3.2×10^{-3}

Isotope	Liquid Activity (μ)Ci/cm^3)
N-16 (max)	1.5
Rb-88	1.7×10^{-2}
Mo-99	2.7
I-131	1.8
I-132	2.9×10^{-2}
I-133	0.9
I-134	7.5×10^{-3}
I-135	0.17
Cs-134	2.6×10^{-1}
Cs-136	1.4×10^{-1}
Cs-137	1.3
Cs-138	7.7×10^{-3}

a. Based on parameters given in table 11.1-1.

TABLE 11.1-5

TOTAL WASTE GAS PROCESSING SYSTEM INVENTORY[a][b]

Isotope	Activity (Ci)
Kr-85	4.75×10^4
Kr-85m	2.2×10^1
Kr-87	3.0
Kr-88	2.8×10^1
Xe-131m	4.3×10^2
Xe-133	4.8×10^4
Xe-133m	3.9×10^2
Xe-135	1.8×10^2
Xe-135m	0.03
Xe-138	0.13
I-131	0.6624
I-132	0.00273
I-133	0.1036
I-134	0.000576
I-135	0.02016

a. Assuming 40 years of full power operation and no releases or leakages from the system.

b. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge (and holdup) mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year inventory of the GWPS is bounding for the period of extended operation. Refer to section 11.3 for a description of GWPS operation.

TABLE 11.1-6

PARAMETERS USED TO DESCRIBE THE REACTOR SYSTEM-REALISTIC BASIS

				ANSI N237 Range	
Parameter	Symbol	Units	Value	Maximum	Minimum
Thermal power	P	MWt	3565	3800	3000
Steam flowrate	FS	lb/h	1.2×10^7	1.7×10^7	1.3×10^7
Weight of water in reactor coolant system	WP	lb	4.0×10^5	6.0×10^5	5.0×10^5
Weight of water in all steam generators	WS	lb	2.7×10^5	5.0×10^5	4.0×10^5
Reactor coolant letdown flow (purification)	FD	lb/h	3.0×10^4	4.2×10^4	3.2×10^4
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/h	538	1.0×10^3	2.5 x 10
Steam generator blowdown flow (total)	FBD	lb/h	3.8×10^4	1.0×10^5	5.0×10^4
Fraction of radioactivity in blowdown steam that is not returned to the secondary coolant system	NBD	-	1.0	1.0	0.9
Flow through the purification system cation demineralizer	FA	lb/h	3.0×10^3	7.5×10^4	0.0
Ratio of condensate demineralizer flowrate to the total stream flowrate	NC	-	0.0	0.01	0.0
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system (not including the boron recycle system)	Y	-	See table 11.1-1	0.01	0.0
Primary-to-secondary leak rate	-	lb/day	100	-	100

TABLE 11.1-7 (SHEET 1 OF 3)

PARAMETERS SPECIFIED BY REGULATORY GUIDE 1.112 APPENDIX B (INPUT PARAMETERS FOR THE GALE COMPUTER CODE)[3]

Description	Value
Thermal power level (MWt)	2785
Mass of primary coolant (lb)	4.2×10^5
Primary system letdown rate (gal/min)	60[a]
Letdown cation demineralizer flowrate (gal/min)	6.0[a]
Number of steam generators	3
Total steam flow (lb/h)	12.3×10^6
Mass of steam in each steam generator (lb)	6.5×10^3
Mass of liquid in each steam generator (lb)	1.0×10^5
Total mass of secondary coolant	1.52×10^6
Total blowdown rate (lb/h)	3.75×10^4
Condensate demineralizer regeneration time	0.0
Condensate demineralizer flow fraction	0.0
Maximum radwaste dilution flow (gal/min)	16.0×10^3
Shim Bleed	
Shim bleed flowrate (gal/day)	1.56×10^3
Decontamination factor for I	10^5
Decontamination factor for Cs and Rb	4×10^4
Decontamination factor for others	10^6
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Equipment Drains	
Equipment drains flowrate (gal/day)	250
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Clean Waste	
Clean waste input flowrate (gal/day)	1.64×10^2
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4

TABLE 11.1-7 (SHEET 2 OF 3)

Description	Value
Collection time (day)	24.4
Process and discharge time (day)	0.37
Fraction discharged	1.0
Dirty Waste	
Dirty waste input flowrate (gal/day)	1.38×10^2
Fraction of reactor coolant activity	0.002
Decontamination factor for I	10
Decontamination factor for Cs and Rb	2.0
Decontamination factor for others	10
Collection time (day)	0
Process and discharge time (day)	15.0
Fraction discharged	1.0
Blowdown Waste	
Blowdown fraction processed	1.0
Decontamination factor for I	10^2
Decontamination factor for Cs and Rb	4.0×10^2
Decontamination factor for others	10^4
Collection time	0.0
Process and discharge time	0.0
Fraction discharged	1.0
Regenerant flowrate	
Decontamination factor for I	N/A
Decontamination factor for Cs and Rb	N/A
Decontamination factor for others	N/A
Collection time	N/A
Process and discharge time	N/A
Fraction discharged	N/A

TABLE 11.1-7 (SHEET 3 OF 3)

Description	Value
Gaseous Waste System	
Holdup time for xenon (day)	90
Holdup time for krypton (day)	90
Fill time of decay tanks for gas stripper	0.0
Gas waste system: HEPA?	Yes
Auxiliary building: charcoal?	Yes
Auxiliary building: HEPA?	Yes
Containment volume (ft^3)	2.15×10^6
Containment atmosphere cleanup rate (ft^3/min)	20×10^3
Containment shutdown purge: charcoal?, HEPA?	Yes, Yes
Number purge per year	8
Containment normal purge rate (ft^3/min); charcoal?, HEPA?	0; yes, yes
Fraction of iodine released from blowdown tank vent	0.05
Fraction of iodine released from main condenser air ejector	0.1
Detergent waste decontamination factor	0.1

a. Evaluation of the impact of increasing letdown flow to 145 gpm and cation demineralizer flow to 14.5 gpm indicate the releases shown in Tables 11.2-7, 11.2-8, 11.3-9, and 11.3-10 and the resultant doses shown in Tables 11.2-9 and 11.3-11 remain bounding.

11.2 LIQUID WASTE SYSTEMS

11.2.1 DESIGN OBJECTIVES

The liquid waste processing system (LWPS) is designed to receive, segregate, process, recycle, and discharge liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from radionuclides leaving the LWPS does not exceed a small fraction of the discharge limits defined in column 2, Table II, Appendix B to 10 CFR 20.

Further, overall radioactive release limits are established as a basis for controlling plant discharges during operation with the occurrence of a combination of equipment faults of moderate frequency. A combination of equipment faults that could occur with moderate frequency include operation with fuel cladding defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in LWPS.

C. Excessive leakage in reactor coolant system equipment.

D. Excessive leakage in auxiliary system equipment.

The radioactive releases from the plant resulting from equipment faults of moderate frequency are within the column 2, Table II, Appendix B to 10 CFR 20 limits on a short-term basis and do not exceed four to eight times the limits stated previously for normal operation on an annual average basis.

11.2.2 SYSTEM DESCRIPTIONS

The LWPS collects and processes potentially radioactive wastes for recycle or for discharge. Provisions are made to sample and analyze fluids before they are recycled or discharged. Based on the laboratory analysis, these wastes are either released under controlled conditions via the cooling water system or retained for further processing. A permanent record of liquid releases is provided by analyses of known volumes of waste.

The radioactive liquids discharged from the reactor coolant system can be processed by the boron recycle system or the portable demineralizer system described in paragraph 11.2.3.1.8. The limited amount of fuel leakage experienced in the plant operating history has enabled the use of the portable demineralizer system to process the bulk of the reactor coolant system radioactive liquid discharges. The operation of the demineralizer system results in a smaller volume of waste to be shipped offsite for disposal. The permanently installed boron recycle system remains available for use to ensure that the technical specification limits are met. The use of the portable demineralizer system or the boron recycle system limits input to the LWPS and results in processing of relatively small quantities of generally low activity wastes.

The LWPS is arranged to recycle reactor grade water if possible. This is implemented by the segregation of equipment drains and waste streams, which prevents the intermixing of liquid wastes. The LWPS consists of two main subsystems designated as drain channel A and drain channel B. Drain channel A processes water which can be recycled, and drain channel B processes water which is to be discharged. A drain system is also provided inside the containment to collect drains and leaks and transfer them to an appropriate tank. Capability for handling and storage of spent demineralizer resins is also provided.

Additionally, the plant has been equipped with a portable demineralizer system described in paragraph 11.2.3.1.8. This system is capable of processing water from any of the waste streams and producing a very low activity effluent. Water processed through the disposable demineralizer system is routed to the waste monitor tank for analysis prior to release.

Instrumentation and controls necessary for the operation of the LWPS are located on a control board in the auxiliary building. Any alarm on this control board is relayed to the main control board in the control room.

Process flow diagrams and piping and instrumentation diagrams are shown in figure 11.2-1 and drawings D-175042, sheets 1, 2, 3, 4, and 7, and D-205042, sheets 1, 2, 3, 4, and 7. All lines in the LWPS, including field run, are considered potential carriers of significant radioactivity.

Table 11.2-1 also gives process parameters for key locations in the system. Expected volumes to be processed by the LWPS are given in table 11.2-2. Assuming the volumes presented in table 11.2-2 are processed at a uniform rate, the input to the waste evaporator will be approximately 0.2 gal/min, while the evaporator is designed to handle 15 gal/min. Hence, excess capacity is available to handle abnormal operating conditions. This will only change the load on the system; otherwise the operating features will not change. Component failures in the LWPS are taken care of during system shutdown. The system is designed so that interchange of components is possible.

11.2.2.1 Recycle Portion (Drain Channel A - Tritiated and Aerated Water Sources)

Drain channel A is provided to process borated water which enters the LWPS via equipment leaks and drains, valve leakoffs, pump seal leakoffs, tank overflows, and other tritiated and aerated water sources.

Deaerated, tritiated water inside the reactor containment (from sources such as valve leakoffs), which is collected in the reactor coolant drain tank, need not enter drain channel A. These may be routed directly to the boron recycle holdup tanks for processing and/or reuse.

Administratively controlled equipment drains are the major contributor of water that may be recycled. Valve and pump leakoffs outside the reactor containment are also collected in the recycle holdup tank for processing and recycle. Abnormal liquid sources include leaks that may develop in the reactor coolant and auxiliary systems. Considerable surge and processing capacity is incorporated in the recycle portion of the LWPS to accommodate abnormal operations.

The basic composition of the liquid collected in the recycle holdup tank is boric acid and water with some radioactivity. Liquid collected in this tank is sampled and recycled or drained to the waste holdup tank for processing via the LWPS. Evaporator bottoms are normally processed at a low boron concentration to the waste holdup tank unless found acceptable for boric acid recycle. The condensate leaving the waste evaporator may pass through the waste condensate demineralizer and then enter the condensate tank. When a sufficient quantity of water has collected in the waste condensate tank, it is normally transferred to the reactor makeup water storage tank for reuse. Samples are taken at sufficiently frequent intervals to ensure proper operation of the system to minimize the need for reprocessing. If a sample indicates that further processing is required, the condensate may be passed through the waste condensate demineralizer or, if necessary, returned to the recycle holdup tank for additional evaporation.

The water collected in the recycle holdup tank may be routed to the portable demineralizer for processing rather than processing the water through the evaporator. Water processed through the demineralizer is not normally recycled.

11.2.2.2 Waste Portion (Drain Channel B - Nonreactor Grade Water Sources)

Drain channel B is provided to collect and process nonreactor grade liquid wastes. These include floor drains, equipment drains containing nonreactor grade water, laundry and hot shower drains, and other nonreactor grade sources. Drain channel B equipment includes a floor drain tank and filter, laundry and hot shower tank and filter, chemical drain tank, waste monitor tank demineralizer and filter, disposable demineralizer system, and two waste monitor tanks.

Nonrecyclable reactor coolant leakage enters the waste holdup tank from system leaks inside the containment via the containment sump and enters the floor drain tank from system leaks in the auxiliary building via the floor drains. Unless an extremely large leak develops, this liquid would not be recycled because it is diluted and contaminated by water entering the floor drain tank from other sources, e.g., laboratory equipment rinses, hose water, component cooling leaks, etc. Nonreactor grade leakage enters the floor drain tank from the auxiliary building floor drains. Sources of water to the drains are fan cooler leaks, secondary side steam and feedwater leaks, component cooling water, and hose water. This leakage is assumed not to contribute significantly to activity release. The activity level is normally much less than 10^{-7} μCi/cm^3.

Normally, the activity of the floor drain tank contents is well below permissible levels. Hence the contents may be transferred directly to the waste monitor tanks after sampling. Following analysis to confirm the acceptable low level, the tank contents are discharged without further treatment. However, should spills, leaks, or equipment failures cause radioactive water to enter the floor drain tank, this water is processed through the waste evaporator or disposable demineralizer.

In general, if the activity in the floor drain tank is greater than 10^{-5} μCi/cm^3, the liquids should be processed. If such a case should occur, the waste evaporator concentrate is drummed or directed to the waste holdup tank and the condensate returned to drain channel B for ultimate discharge, or the waste is processed via the disposable demineralizer system and the effluent returned to drain channel B for ultimate discharge.

Laundry and hot shower drains are the largest source of liquid wastes and normally need no treatment for removal of radioactivity. This water is transferred to one of the waste monitor tanks via the laundry and hot shower filter. The laundry and hot shower water may be processed through the disposable demineralizer if required. A sample is taken, the results logged after analysis, and the water discharged if the activity level is below acceptable limits.

The basic criterion for the laboratory drain subsystem is that strict segregation of radioactive and nonradioactive liquid wastes be maintained. Two separate drains are provided for this control. One is used to dispose of spent and excess radioactive samples directly to the chemical drain tank for later processing via the waste holdup tank. The second drain is provided for normal laboratory equipment decontamination and rinsing. This liquid waste is directed to the floor drain tank. The sampling room contains two sinks. Excess sample purges of reactor grade coolant are drained from one sink to the waste holdup tank for recycle. The other sink is used for draining nonreactor grade excess samples to the floor drain tank.

Liquid wastes are released from the waste monitor tanks through a discharge valve interlocked with a process radiation monitor. The valve closes automatically and annunciates in the control room when the radioactivity concentration in the liquid discharge exceeds a preset limit. Liquid waste discharge flow volume is recorded.

11.2.2.3 <u>Waste from Spent Resin</u>

The spent resin sluice portion of the LWPS consists of a spent resin storage tank, a spent resin sluice pump, and a spent resin sluice filter. The equipment is arranged in such a way that the resin sluice water, after entering a demineralizer vessel, returns to the spent resin storage tank for reuse. The basic criterion for the system is to transport spent resin to the spent resin storage tank without generating large volumes of waste liquid. This is accomplished by reusing the sluice water for subsequent resin sluicing operations. Spent resins are sluiced from the portable demineralizer system to a container for disposal or offsite processing.

11.2.3 SYSTEM DESIGN

11.2.3.1 <u>Component Design</u>

In accordance with its safety classification, the LWPS components are designed to meet the design requirements of the codes and standards listed in table 3.2-1. However, it should be noted that the components of the LWPS listed in table 3.2-1, as a minimum, are designed and manufactured to the requirements given in Regulatory Guide 1.143, Revision 1, with the exception of the seismic design criteria given in Regulatory Position C.5. The components, systems, and structures of the LWPS are designed to the seismic design criteria given in section 3.7. For further information on the safety and seismic classification of this system, see chapter 3.

The materials of construction along with the essential design parameters are given in table 11.2-3. All parts of components in contact with borated water are fabricated or clad withaustenitic stainless steel. In addition, all pumps are provided with vent and drain connections.

Paragraph 3.9.2.7 gives the general design criteria for field run piping.

11.2.3.1.1 Pumps

A. Reactor Coolant Drain Tank Pumps

Due to the relative inaccessibility of the containment and the loop drain requirement, two pumps are provided. One pump provides sufficient flow for normal tank operation, with one pump for standby.

B. Waste Evaporator Feed Pump

One standard pump is used. The waste evaporator feed pump supplies feed to the evaporator based on level control in the waste holdup tank.

C. Waste Evaporator Condensate Tank Pump

The waste evaporator condensate tank pump is a transfer pump. One standard pump is used to transfer the contents of the waste condensate tank to reactor makeup water storage tanks or the boron recycle system holdup tank.

D. Deleted.

E. Spent Resin Sluice Pump

This pump is similar, with regards to performance characteristics, to the reactor coolant drain tank pumps. Its delivery flow is based on the required velocity to sluice resin.

F. Laundry and Hot Shower Tank Pump (Unit 1 only)

One standard pump is used to transfer the water to the waste monitor tank.

G. Floor Drain Tank Pump

One standard pump is used to transfer water normally to the waste monitor tank. The pump can also be used to supply the waste evaporator or for pumping the waste back to the waste holdup tank.

H. Waste Monitor Tank Pumps

One standard pump is used for each tank to discharge water or to recycle water if further processing is required. The pump may also be used for circulating the water in the waste monitor tank in order to obtain uniform tank contents and hence a representative sample before discharge. The pump can be throttled to achieve the desired flowrate.

11.2.3.1.2 Reactor Coolant Drain Tank Heat Exchangers

The reactor coolant drain tank heat exchanger is a U-tube type with one shell pass and two tube passes. Although the heat exchanger is normally used in conjunction with the reactor coolant drain tank, it can also cool the pressurizer relief tank from 200°F to 120°F in < 8 h.

11.2.3.1.3 Tanks

A. Reactor Coolant Drain Tank

One tank is provided for each unit. The purpose of the reactor coolant drain tank is to collect leakoff type drains inside the containment at a central collection point for further disposition through a single containment penetration via the reactor coolant drain tank pumps. The tank provides surge and net positive suction head requirements to the pumps.

The water entering the reactor coolant drain tank may be of adequate purity to allow direct recycling to the boron recycle system holdup tank. If this water is not compatible or if it contains dissolved air or nitrogen, it must be processed in the LWPS channel A.

Sources of water entering the reactor coolant drain tank include the reactor vessel flange leakoff, valve leakoffs, reactor coolant pump 2 and 3 seal leakoffs, and the excess letdown heat exchanger flow. No continuous leakage is expected from the reactor vessel flange during operation.

The system is designed to maintain a constant level in the tank to minimize the amount of gas sent to the waste gas processing system and also to minimize the amount of hydrogen required. One pump runs continuously. The level in the tank is maintained by a control valve in the discharge line. The valve operates on signals from a level controller connected to the tank and regulates flow fractions back to the tank and out of the system, respectively.
As an alternate mode of operation, the reactor coolant drain tank (RCDT) pumps may both be secured and manually actuated at the necessary intervals. Reactor coolant drain tank parameters are provided at the system control station. If the manual mode of operation is chosen, procedural requirements will ensure that the reactor coolant drain tank level is monitored at regular intervals and the pumps are actuated as needed.

With the waste gas system out of service, the RCDT can be vented via polytubing through the sample connection on the RCDT vent line to the waste gas decay tank sample station and into a waste gas decay tank.

B. Waste Holdup Tank

One atmospheric pressure tank is provided outside the containment to collect equipment drains, valve and pump seal leakoffs, recycle holdup tank overflows, and other water from tritiated, aerated sources.

C. Waste Evaporator Condensate Tank

One tank is provided to collect condensate from the waste evaporator.

D. Chemical Drain Tank

One tank is provided to collect chemically and radiologically contaminated water from the laboratories and decontamination room wastes.

E. Spent Resin Storage Tanks

The purpose of the spent resin storage tanks (one for primary spent resins and one for secondary spent resin) is to provide a collection point for spent resin to allow for decay of short lived radionuclides before drumming. The tank serves also as a head tank for the spent resin sluice pump. Vertical, cylindrical tanks are used because the symmetrical bottom facilitates the removal of resin. The tank is designed so that sufficient pressure can be applied in the gas space of the tank to push resin out and to the drumming station or solidification and dewatering building, which may be at a higher elevation than the spent resin storage tank.

The spent resin storage tank and associated equipment that may contain radioactive material are shielded to limit the dose to personnel.

F. Laundry and Hot Shower Tank (Unit 1 only)

One atmospheric pressure tank is used to collect laundry and hot shower drains within the controlled areas.

G. Floor Drain Tank

One atmospheric pressure tank is used to collect floor drains from the controlled areas of the reactor plant.

H. Waste Monitor Tanks

The two atmospheric pressure waste monitor tanks are provided for monitoring liquid discharges from the LWPS. Each tank is sized to hold a volume large enough so that sampling requirements are minimized, thus minimizing further laboratory effluent.

I. Waste Evaporator Reagent Tank

One tank is used for adding chemicals to the plant for such things as cleaning of the waste evaporator tubes.

J. Concentrated Waste Storage Tank

One atmospheric pressure tank (serving both Units 1 and 2) may be used to old evaporator bottoms prior to drumming or solidification. The tank can accept concentrated wastes from either unit and discharge to either the solidification and dewatering or the drumming station.

To ensure the concentrates do not crystallize, the tank is insulated and heated.

11.2.3.1.4 Demineralizers

As part of a continuous pressurized water reactor (PWR) operating plant following, Westinghouse has obtained operational data on demineralizer decontamination factors for selected isotopes. The measured range of decontamination factors for these isotopes is given in table 11.2-4.

These values were observed across mixed-bed demineralizers containing cation resin in the Li-7 form and anion resin in the borated form.

In considering the waste evaporator condensate demineralizer and the waste monitor tank demineralizer, it can be assumed that greater decontamination factors would be realized because the resin in both the demineralizers are in the hydrogen hydroxyl form. The minimum values in table 11.2-4 were generally observed just prior to resin flushing and recharging, while during the operating life of the demineralizer, decontamination factors were consistently closer to the maximum values.

Although specific operating decontamination factors have not as yet been measured for other isotopes, their behavior in a mixed-bed demineralizer may be inferred from this data. One would anticipate, for example, tellurium and bromine to have decontamination factors similar to those given above for the iodine and fluorine.
The process decontamination factors used for design are given in table 11.2-4.

A. Waste Evaporator Condensate Demineralizer

One mixed-bed demineralizer in the hydrogen hydroxyl form is provided to remove ionic contaminants from the waste condensate which is intended to be recycled to the reactor coolant system.

B. Waste Monitor Tank Demineralizer

One mixed-bed demineralizer is provided to remove trace contaminants from the water if evaporation is not deemed necessary.

11.2.3.1.5 Filters

Filters are provided with easily disposable filter media.

Each filter has a capacity to pass 100 gal/min of water at pressure ratings compatible with filter design and not to exceed a pressure differential across the filter of 200 ft head.

The filters provided will be of material and construction that will meet all system parameters as listed in FSAR tables 9.1-2, 9.3-6, and 11.2-3.

The methods employed to change filters and screens are dependent on activity levels. Filters are valved out of service with a pressure indicator between the isolation valves to ensure that the valves are not leaking and that the filter is not at system pressure. The filter is drained to the appropriate tank and vented locally. If the radiation level of the filter is low enough, it is changed manually. If activity levels do not permit manual change, the spent cartridge is removed remotely with temporary shielding to protect personnel. The spent cartridge is placed in a shielded drum for removal to the solid waste disposal area. A new cartridge is installed, the housing is reassembled, vent and drain valves are closed, and the filter is valved into service. Filters are normally changed because of high ΔP rather than high radiation levels.

11.2.3.1.6 Strainers

Strainers are provided in two different types: basket and Johnson screen. The basket type is a mesh or screen construction, and the Johnson screen is a wound wire construction. The basket type strainers are not given an absolute rating because the tolerances in the size range of particles are not critical; it would not be feasible to put meaningful absolute rating on these strainers. The Johnson screen type strainer also does not have an absolute rating but a nominal rating with a plus/minus tolerance. Actually, the largest absolute particle that can pass through the screen is the rating plus the tolerance.

The basket type laundry and hot shower strainer is not replaced after use but is cleaned and put back in service. Because this screen traps only large particles, it contains only negligible activity and provides no hazard to personnel. It is cleaned as necessary.
The drumming header strainer is a Johnson screen type and is backflushable. The system in which it is installed provides an easy method for backflushing without removal of the strainer.

11.2.3.1.7 Waste Evaporator

One waste evaporator is used. The waste and the boron recycle evaporators are identical units and are interconnected so that they serve as a standby for each other under abnormal conditions.

11.2.3.1.8 Portable Demineralizer System(s)

Units 1 and 2 are served by a LWPS capable of processing the same liquid streams as the evaporator and the chemical drains. Therefore, when the portable demineralizer system(s) is in use, the waste evaporators are not needed.

Presently, the plant portable demineralizer system consists of an atmosphere demineralizer system (ADS) and a pressurized demineralizer system (PDS). The systems are interfaced via air operated three-way valves such that it is the only system that can be operated at a time.

The effluent from the disposable demineralizer system is routed to the waste monitor tank for sampling and analysis prior to release. The liquid released is subject to the same release restrictions as evaporator distillate, so the offsite dose is accounted for in the same manner for either system.

The expended media (resin, charcoal, filters) can be transferred to appropriate shipping containers and dewatered or solidified as necessary and shipped to a licensed burial ground for ultimate disposal.

The volume of waste to be shipped offsite for burial is significantly lower for the disposable demineralizer system than for the solidified evaporator concentrates, therefore, the system acts as an effective volume reduction device.

11.2.3.2 <u>Instrumentation Design</u>

The system instrumentation is described in table 11.2-5 and shown in the flow and piping diagrams in drawings D-175042, sheets 1, 2, 3, 4, and 7, and D-205042, sheets 1, 2, 3, 4, and 7.

The instrumentation readout is located chiefly on the waste processing system panel in the auxiliary building. Some instruments are read where the equipment is located.

All alarms are shown separately on the waste processing system panel and further relayed to one common system annunciator on the main control board of the plant.

All pumps are protected against loss of suction pressure by a control setpoint on the level instrumentation for the respective vessels feeding the pumps. The reactor coolant drain tank pumps are interlocked with flowrate instrumentation and stop operating when the delivery flows reach minimum setpoints.

Pressure indicators upstream and downstream of filters, strainers, and demineralizers provide local indications of pressure drops across each component.

All liquid releases from the LWPS are monitored for radioactivity by a scintillation counter. This instrumentation is further described in section 11.4.

11.2.4 OPERATING PROCEDURES

The LWPS is manually operated except for some functions of the RCDT circuit and the disposable demineralizer system. The system includes adequate control equipment to protect the system components and adequate instrumentation and alarm functions to provide operator information to ensure proper system operation. All pumps in the system have low level shutoffs, and all filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

11.2.4.1 Normal Operation

Operation of the LWPS is essentially the same during all phases of normal reactor plant operation; the only differences are in the load on the system. The following sections discuss the operation of the system in performing its various functions. In this discussion, the term "normal operation" should be taken to mean all phases of operation except operation under emergency or accident conditions. The LWPS is not regarded as an engineered safety feature system.

11.2.4.1.1 Recycle Portion

Water is accumulated in the waste holdup tank until sufficient quantity exists to warrant an evaporator startup, to switch the evaporator operation from the floor drain tank to the waste holdup tank, or to process through the disposable demineralizer system.

During evaporation the distillate is checked for boron and activity concentration, and if the analysis shows compatibility with reactor makeup grade water, it is transferred to the reactor makeup water storage tank. If the distillate is high in boron concentration or activity, it may be passed through the waste evaporator condensate demineralizer before being transferred to the reactor makeup water storage tank. If reevaporation is required and the waste evaporator is not available, then the distillate can be transferred to the boron recycle holdup tanks for processing by the boron recycle evaporator. The bottoms from the waste evaporator may be concentrated to approximately 12-percent boric acid but are normally concentrated to a low boric acid concentration and dumped to the waste holdup tank. Should the bottoms be acceptable for recycle, they are concentrated to approximately 4-percent boric acid and transferred to the boric acid tanks.
During normal operation, the reactor coolant drain tank level regulation and pressure control are automatic and require no operator action.

Operation of the recycle portion of the LWPS during refueling is the same as for power operation, although the load on the system may be increased when refueling is complete. The water remaining in the canal following normal drain down is pumped to the suction of the refueling water purification pump by the RCDT pumps.

11.2.4.1.2 Waste Portion

The waste portion of the LWPS consists of two subsystems: the laundry and hot shower system and the floor drain tank system.

Laundry and hot shower water enters the laundry and hot shower tank for holdup. The water is filtered and transferred to the waste monitor tank where it is sampled and discharged.

The water in the floor drain tank is sampled to determine the degree of processing required. It can be sent directly to the waste monitor tank provided for floor drain tank water or to the waste monitor tank via the waste monitor tank demineralizer, or it can be processed through the waste evaporator or the disposable demineralizer system. If the water is evaporated, the distillate is sent to the waste monitor tank and the concentrate is recycled or solidified. The water in the waste monitor tank is again sampled and can be recirculated through the waste monitor tank demineralizer if further processing is required. When this water has been sufficiently processed, it is discharged into the plant discharge line at a rate so as not to exceed a small fraction of Technical Specification limits. A process decontamination factor in the range of 7.2×10^3 to 7.2×10^5 is expected for the evaporator demineralizer combination. Chemical trace element tests as well as operating experience on similar evaporators have justified this process decontamination factor.

Water leaving this system to the discharge canal is monitored for radiation. Should the radiation monitor close the discharge valve, it must be cleared before the valve can be reopened. The monitor element can be cleared by flushing it with demineralized water from the temporary connection back to the waste monitor tank. During refueling the load on the waste portion of the LWPS is increased, but there is no change in operation.

11.2.4.1.3 Laboratory Drain Portion

The laboratory drain portion consists of three sinks in the laboratory (one main sink and two sinks in the fume hoods), one sink in the gas analysis room fume hood, and one sink in the chemical drain tank and pump room. Spent and excess reactor coolant samples which cannot be recycled are disposed of via the chemical drain tank sink. When sufficient waste is collected in the tank, the waste is sent to the waste holdup tank. Equipment rinse water and other nonreactor grade water is disposed of via the floor drain tank sink.

11.2.4.1.4 Spent Resin Handling Portion

This portion of the system sluices resin from the demineralizers and transports resin from the spent resin storage tank to the drumming room or bulk shipping facility.

A. Resin Sluicing

   Before resin sluicing begins, the demineralizer is valved out of service and the flow path is aligned from the resin sluice pump through the process line of the demineralizer, through the screen at the top of the demineralizer, and back to the spent resin storage tank. This process loosens the bed in preparation for sluicing.

After about 10 min of backflushing, the pump is shut off and the valves in the backflush circuit are closed. The sluice line is opened, and the resin sluice pump is restarted. The resin then flows to the spent resin storage tank. After the resin sluice pump is shut off, the fresh resin is added via the resin fill line and the valve is then closed. The valves are then realigned for normal process operation. Resins are never sluiced through the spent resin sluice pump.

B. Resin Drumming

When sufficient resin is accumulated for processing, the valves in the line to the solidification and dewatering facility are opened and all other valves are closed. The tank is then pressurized with N_2. The appropriate valves are then opened and the resin is forced into the disposal container. During drumming, N_2 is forced through the spargers in the tank bottom to level the resin and maintain the tank pressure. When the containers are full, the valves are closed and the tank pressure is relieved to the plant vent. The valves are then flushed using reactor makeup water.

C. Disposable Demineralizer

The disposable demineralizers and filters consist of shipping cask liners equipped with underdrains and filled with the appropriate resin or filter material. The disposable demineralizer system consists of an activated carbon filter and two mixed-bed demineralizers or three demineralizers connected in series. When the demineralizer or filter is depleted it is dewatered and/or solidified, placed in a shipping cask, and shipped off site for disposal. This allows expended units to be disposed of without rehandling the resins or filter media. This procedure results in reduced radiation exposure to operating personnel.

The level indicating system in the spent resin storage tank is a conventional system, the only difference being that there is a bellows to keep resin fines away from the instrument. The method of operation limits the resin to a maximum level and the water to a minimum level, the minimum water level being some distance above the resin level. The lower level tap is located in the area which contains water and no resin. This arrangement minimizes the possibility of plugging the level tap. Because the system indicates only total level and not the amount of resin and the amount of water, an inventory of spent resins in the tank is maintained. Since the resin volumes flushed from demineralizers are known and the resin volumes drummed are known, the resin level in the tank is also known.

11.2.4.2 Faults of Moderate Frequency

The system can handle the occurrence of equipment faults of moderate frequency such as:

A. Malfunction in the LWPS

Malfunction in this system could include such things as pump or valve failures or evaporator failure. Because of pump standardization throughout the system, a spare pump can be used to replace most pumps in the system. There is sufficient surge capacity in the system to accommodate waste until the failures can be fixed and normal plant operation resumed. Also, the disposable demineralizer system provides additional surge capacity for the LWPS.

B. Excessive Leakage in Reactor Coolant System Equipment

The system can handle a 1-gal/min reactor coolant leak in addition to the expected leakage during normal operation. Operation of the system is almost the same as for normal operation except that the load on the system is increased. A 1-gal/min leak into the RCDT is handled automatically but may increase the load factor of the recycle evaporator. If the 1-gal/min leak enters the waste holdup tank, operation is the same as normal except for the increased load on the evaporator or disposable demineralizer system. Abnormal liquid volumes of reactor coolant resulting from excessive reactor coolant or auxiliary building equipment leakage (1 gal/min) can also be accommodated by the floor drain tank and processed by the LWPS.

C. Excessive Leakage in Auxiliary System Equipment

Leakage of this type could include water from steam side leaks and fan cooler leaks inside the containment which are collected in the containment sump and sent to the waste holdup tank. Other sources could be component cooling water leaks, service water leaks, and secondary side leaks. This water would enter the waste holdup tank and would be processed and discharged as during normal operation.

D. Steam Generator Tube Leaks

During periods of operating with fuel defects coincident with steam generator tube leaks, radioactive liquid is processed via the steam generator blowdown system. This system is described in subsection 10.4.8.

11.2.4.3 <u>Operating Experience</u>

Different processing systems with evaporators have been tested for feed to distillate decontamination factors. A 2-gal/min evaporator was operated with the feed in the pH range of 5.2 to approximately 11.6. Gross beta and gamma activity as well as I-131 activity were measured in the feed and the distillate. The decontamination factors obtained were in the range of 5×10^3 to 5×10^4. The same evaporator was also tested with sodium; the decontamination factors obtained for sodium were, in general, 10^5 or higher. A second evaporator was tested at different pH levels with sodium for gross beta and gamma activity. The test confirmed the results previously obtained. A Westinghouse-designed evaporator similar to the one to be used for Farley Nuclear Plant (FNP) has been shop tested measuring decontamination factor between bottom and distillate for sodium. The decontamination factors obtained were in the range of 10^5 to 10^6. Hence, a feed to distillate evaporator decontamination of 10^3 used for design is considered conservative.

For operational decontamination factors obtained on demineralizers, see paragraph 11.2.3.1.4.

11.2.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation and control equipment, and applicable alarms and control setpoints.

Operability testing of the LWPS is conducted periodically in accordance with the plant Technical Specifications to determine the reliability of components designed to reduce liquid activity levels. A decontamination factor for the waste evaporator is obtained by measuring the concentrations of I-131, Cs-137, and Co-60 before and after processing to monitor the efficiency of the LWPS. Demineralizer efficiency will be monitored in a similar manner. The waste evaporator output may be recycled in order to reduce contamination to the design levels. The disposable demineralizer system may be used to process the liquid waste to reduce the load on the LWPS.

The radiological analyses conducted to assess the performance of the LWPS and to determine discharge concentrations are discussed in detail in section 11.4.

11.2.6 ESTIMATED RELEASES

11.2.6.1 Nuclear Regulatory Commission Requirements

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for release of radioactive liquids:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

D. Regulatory Guide 1.113, Revision 1, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

During plant operations, radioactive liquid releases will be controlled in accordance with Technical Specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, it is acceptable that the limits associated with the release rate Technical Specifications are based on ten times the effluent concentration limits given in column 2, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, since operational history at Farley Nuclear Plant has demonstrated that the calculated maximum individual doses to members of the public are small percentages of the values given in 10 CFR 50, Appendix I, and 40 CFR 190.

11.2.6.2 Westinghouse PWR Experience Releases

The liquid releases are highly dependent upon administrative activities which control the use of water for decontamination, equipment and floor rinsing, and other uses in the controlled areas.

The plants operating at the time of Unit 1 licensing were reporting liquid discharges as shown in table 11.2-6 for years 1970 and 1971.

11.2.6.3 Expected Liquid Waste Processing System Releases

The equipment utilized during liquid waste processing is at the discretion of the operator; therefore, the calculated releases do not address all possible treatment processes but only the process which was the basis for the original plant design. Liquid releases from FNP were calculated using the PWR-GALE computer code[(2)] and parameters listed in table 11.1-7, which are discussed in more detail below. Releases calculated assuming operation with expected levels of fuel cladding defects of 0.12 percent are presented in table 11.2-7. Primary and secondary coolant activity levels are discussed in section 11.1 for the realistic case. In agreement with reference 2, the total releases include an adjustment factor of 0.15 Ci/year, using the same isotopic distribution as the calculated release, to account for anticipated operational occurrences.

The tables list the calculated annual release from each of the process paths discussed below as well as the total annual release. A comparison of annual average effluent concentrations with values stated in column 2, Table II, Appendix B to 10 CFR 20 is provided in table 11.2-8 for operation with expected fuel leakage.

The releases are calculated for one unit, assuming that both units are operating. This is done to reflect the impact of the second unit's operation on the operation of systems and components shared between the two units. To obtain the combined releases of the two units, simply double the values listed in table 11.2-7.

A survey has been performed of liquid discharges from different Westinghouse pressurized water reactor plants, with results presented in table 11.2-6 for years 1970 and 1971. The data include radionuclides released on an unidentified basis and are all within the permissible concentration for release of liquid containing an unidentified radionuclide mixture. The data in table 11.2-6 clearly indicate that actual releases are highly dependent upon the actual operation of the plant and can vary significantly from year to year for a given plant as well as from plant to plant.

The LWPS is assumed to operate as described in subsection 11.2.4.

11.2.6.4 Steam Generator Blowdown System

The secondary side activity used in the offsite release analysis is given in table 11.2-7.

The blowdown from the secondary side is normally released to the environment; however, the liquid may be recycled to the main condenser if required.

The estimated activity released per unit to the environment from such discharges is given in table 11.2-7. The system is further described in subsection 10.4.8.

11.2.6.5 Turbine Building Drains

The concentration of isotopes in steam or liquid leaked to the turbine building is considered a factor of 100 lower than secondary side concentration in table 11.2-8 for all isotopes except tritium. Tritium concentration in leakage is assumed to be the same as in the secondary side. The factor of 100 accounts for limited carryover in steam. Steam leakage of 5 gal/min (condensed) and liquid leakage of 12 gal/min is assumed to be discharged through turbine building drains. Discharge rates for each isotope are given in table 11.2-7.

11.2.6.6 Estimated Total Releases

The potential releases from each source have been evaluated as indicated in above sections. As shown in table 11.2-8, the total expected liquid release from one unit of the plant is a small fraction of the regulations as outlined in paragraph 11.2.6.1. It is further shown that the expected liquid releases from FNP are well below releases in presently-operating plants, as shown in table 11.2-6 for years 1970 and 1971. Hence, the releases from the plant are in accordance with the design objectives as outlined in subsection 11.2.1 and the plant Technical Specifications.

11.2.7 RELEASE POINTS

The LWPS is designed to minimize the total radioactive fluid released to the environment by processing and recycling as much water as possible. This design allows only one release point, as shown in drawings D-175042, sheet 4 and D-205042, sheet 4. Drawing D-170180, sheet 1 shows the physical location of this release point.

11.2.8 DILUTION FACTORS(1)

[Historical]
[*The volume of the mixing zone will be small and will contain concentrations ranging from full dilution to concentrations approaching those in the discharge pipe. The discharge pipe concentrations, less than 161 times higher than full dilution concentrations, would be confined to an extremely small volume.*

For doses received near the plant, the annual average flowrate of the Chattahoochee River (11,500 ft^3/s) is used. For performance of the 10 CFR 50 Appendix I analysis,[(1)] liquid effluent isotopes from the FNP were assumed to be diluted by a flow of approximately 16,000 gal/min (71.4 ft^3/s for 2 units) from each unit prior to discharge. Dilution flow of less than 16,000 gal/min may occur during certain operational conditions such as plant outages. Compliance with the FNP ODCM and Technical Specifications ensures that release concentrations are within acceptable limits.

The mixing ratio (inverse of the dilution factor) was taken as 0.2 for fish ingestion and recreational pathways in accordance with recommendations in Regulatory Guide 1.109, Table A-1.[(3)] The resultant concentrations roughly approximate those at the edge of the initial mixing zone.

The mixing ratio for other pathways was taken as 3.1×10^{-3}, which is equivalent to dilution of the 16,000 gal/min effluent stream in the full flow of the river (11,500 ft^3/s).[(4)]

Full dilution is warranted because the nearest downstream water usage for the other pathways is at least 20 miles downstream.

Mixing near the discharge structure and discharge characteristics are discussed further in subsection 2.4.12 and OLSER subsection 3.4.3.]

The historical information above was utilized in the dose calculations to confirm FNP compliance with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).

11.2.9 ESTIMATED DOSES[(1)]

[HISTORICAL]

[Dose models and values for usage rates, holdup times, and other parameters used to estimate maximum doses to individuals from discharges to the hydrosphere are those described in Regulatory Guide 1.109.[(3)]

Pathways evaluated include fish ingestion, shoreline recreation, boating, and swimming. Freshwater invertebrates are not normally consumed by humans. At present, there are no public water supply intakes closer than 50 miles downstream of the plant, and there are no known plans to construct any. Separate evaluation of a hypothetical drinking water pathway, based on full river flow dilution of plant effluents, indicates that no organ dose or total body dose would exceed 0.03 mrem/year, if the pathway existed.

River water is occasionally used for irrigation at several points 20 miles or more downstream from the plant. It is extremely unlikely that any of the product reaches the individual with the maximum doses from plant liquid effluents, so the pathway was not included in the estimation of *maximum doses to individuals. A separate analysis of the ingestion pathway, assuming full river flow dilution of plant liquid effluents and 5-inches-per-month irrigation (comparable to normal summer rainfall), indicates that no organ or total body dose in any age group would exceed 0.006 mrem/year from consumption of irrigated vegetables (fresh or stored), milk, or meat.*

The historical dose estimates above were calculated to confirm that the Farley Nuclear Plant conforms with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).]

The estimated doses for the appropriate pathway are outlined in table 11.2-9 along with the Appendix I design objective doses for comparison. It is clear that the estimated doses follow the design objective doses in each case. Actual plant releases during normal operation are governed by the Farley Nuclear Plant Technical Specifications and Offsite Dose Calculations Manual.

REFERENCES

1. Alabama Power Company Letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," USNRC Docket Nos. 50-348, 50-364, June 3, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

3. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

4. Regulatory Guide 1.113, Revision 1, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

TABLE 11.2-1

PROCESS FLOW DIAGRAM - LWPS – TABULATED ACTIVITIES

Line Number	Location of Line		Temp (°F)	Press. (PSIG)	Flow Rate (GPM)	Isotopic Concentration (µCi/gram)																										
						Rb-88	Sr-89	Sr-90	Sr-91	Y-90	Y-91	Zr-95	Nb-95	Mo-99	I-131	I-132	I-133	I-135	Cs-134	Cs-136	Cs-137	Te-132	Ba-140	La-140	Ce-144	Pr-144	Cr-51	Mn-54	Fe-59	Co-58	Co-60	H-3
1	CVCS RCP SEAL LEAKOFF INPUT TO RC DRAIN TANK		130	5	0.156	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	RC DRAIN TANK LOOP AFTER HX - INPUT TO BRS RH TANK		170	120	0.156	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
3	DRAIN HEADER INPUT TO WASTE HOLDUP TANK		AMB	ATM	0.139	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4	SAMPLE ROOM SINK - INPUT TO WASTE HOLDUP TANK		AMB	ATM	0.008	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
5	SUM OF INPUTS 3 AND 4		AMB	ATM	0.144	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
6	AFTER RH TANK - BEFORE WASTE EVAP. PACKAGE		AMB	ATM	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7	EVAPORATOR CONCENTRATES	A	156	50	0.006	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
		B	156	50	0.0007	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
8	AFTER WASTE EVAPORATOR PACKAGE	A	120	45	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
		B	20	45	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
9	AFTER WASTE EVAPORATOR CONDENSATE DEMINERALIZER	A	20	VAR	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
		B	120	VAR	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
10	TREATED WATER AFTER WASTE EVAP. CONDENSATE TANK		120	VAR	0.138	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
11	INPUT FROM FLOOR HEADER TO FLOOR DRAIN TANK		AMB	ATM	0.032	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
12	INPUT FROM AUXILIARY BLDG. SUMPS TO FLOOR DRAIN TANK		AMB	ATM	0.032	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
13	SUM OF INPUTS 11 AND 12		AMB	ATM	0.064	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
14	INPUT FROM LAB RINSE WATER TO FLOOR DRAIN TANK		AMB	ATM	0.052	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
15	OUTPUT FROM FLOOR DRAIN TANK		AMB	VAR	15	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
16	OUTPUT FROM WASTE MONITOR TANK 1		AMB	VAR	0.094	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
17	INPUT FROM SPENT SAMPLES TO CHEMICAL DRAIN TANK		AMB	ATM	0.002	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
18	OUTPUT FROM CHEMICAL DRAIN TANK		AMB	VAR	0.002	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
19	LAUNDRY AND HOT SHOWER DRAINS INPUT TO LAUNDRY AND HOT SHOWER TANK		160	ATM	0.229	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	[illegible]	[illegible]	N.D.	N.D.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
20	OUTPUT FROM WASTE MONITOR TANK 2		AMB	VAR	0.229	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	[illegible]	[illegible]	N.D.	N.D.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
21	SUM OF INPUTS 16 AND 20		AMB	ATM	0.323	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Vent Number	Vent Description	Temp (°F)	Press. (PSIG)	Flow Rate (CC/MIN)	Isotopic Discharge Rate (µCi/MIN)								
					Kr-85m	Kr-85	Kr-87	Kr-88	Xe-133m	Xe-133	Xe-135	I-131	I-133
1	REACTOR COOLANT DRAIN TANK TO GWPS	130	5	6.6	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	WASTE HOLDUP TANK VENT TO PLANT VENT	AMB	ATM	431	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
3	WASTE EVAPORATOR PACKAGE VENT TO PLANT VENT	150	3	6.0	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG
4	FLOOR DRAIN TANK VENT TO PLANT VENT	AMB	ATM	381	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
5	WASTE MONITOR TANK 1 LOCAL VENT	AMB	ATM	354	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
6	CHEMICAL DRAIN TANK VENT TO PLANT VENT	AMB	ATM	7.56	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
7	LAUNDRY AND HOT SHOWER TANK LOCAL VENT	VAR	ATM	866	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG
8	WASTE MONITOR TANK 2 LOCAL VENT	AMB	ATM	866	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG

NEG - NEGLIGIBLE
AMB - AMBIENT
VAR - VARIABLE
N.D. - NO DATA AVAILABLE

Tank Number	Tank Identification	Temp. (°F)	Press. (PSIG)	Tank Volume (GAL.)	Tank Inventories (Curies)																										
					Rb-88	Sr-89	Sr-90	Sr-91	Y-90	Y-91	Zr-95	Nb-95	Mo-99	I-131	I-132	I-133	I-135	Cs-134	Cs-136	Cs-137	Te-132	Ba-140	La-140	Ce-144	Pr-144	Cr-51	Mn-54	Fe-59	Co-58	Co-60	H-3
1	REACTOR COOLANT DRAIN TANK	130	5	350	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	WASTE HOLDUP TANK	AMB	ATM	10000	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
3	WASTE EVAPORATOR CONDENSATE TANK	120	ATM	5000	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4	FLOOR DRAIN TANK	AMB	ATM	10000	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
5	WASTE MONITOR TANK 1	AMB	ATM	5000	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
6	CHEMICAL DRAIN TANK	AMB	ATM	600	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7	LAUNDRY AND HOT SHOWER TANK	VAR	ATM	15000	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	[illegible]	[illegible]	N.D.	N.D.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
8	WASTE MONITOR TANK 2	AMB	ATM	5000	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	[illegible]	[illegible]	N.D.	N.D.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
9	SPENT RESIN STORAGE TANK	AMB	VAR	2618	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	[illegible]	[illegible]	N.D.	N.D.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

TABLE 11.2-2

PARAMETERS USED IN THE CALCULATION OF ESTIMATED ACTIVITY IN LIQUID WASTES

Collector Tank with Sources [a]	Volume of Liquid Wastes	Basis	Collection (Period Assumed Before Processing)	Comments
Reactor coolant drain tank	225 gal/day	0.05 gal/min/ reactor coolant pump 2 seal leak; 0.002 gal/min/reactor coolant pump 3 seal leak	Feed and bleed	Recycled to BRS
Waste holdup tank				
Equipment drains	57,000 gal/year	Filter drains, heat exchanger drains, tank drains, demineralizer drains		
Excess samples	3000 gal/year	3000 samples/year at 1 gal/sample		
Total	60,000 gal/year		20 days	Recycled to RMW
Floor drain tank				
Decontamination water	15,000 gal/year	40,000 ft^2 section once per week with 20 gal of water per 5000 ft^2 and remainder for fuel cask, vessel head, etc.		
Laboratory equipment	16,000 gal/year	60 gal/day for 5 day/week		
Nonrecyclable	7000 gal/year	20 gal/day		
Nonreactor grade leaks	13,000 gal/year	40 gal/day		
Total	51,000 gal/year		30 days	Discharged
Chemical drain tank	1000 gal/year	3000 samples/year at 1/8 gal/sample plus rinse water	90 days	Sent to the waste holdup tank
Laundry and hot shower tank	120,000 gal/year	300 gal/day with remainder for abnormal and refueling operations	7 days	Discharged

(a) These sources may be processed via disposable demineralizer system.

TABLE 11.2-3 (SHEET 1 OF 10)

EQUIPMENT PRINCIPAL DESIGN PARAMETERS

Component	Value
Pumps	
Reactor coolant drain tank pumps	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	100
Design point 2	140 (150 for N2G21P001B-N)
Design head (ft)	
Design point 1	300
Design point 2	250
Material	Stainless steel
Waste evaporator feed pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
Waste evaporator condensate tank pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel

TABLE 11.2-3 (SHEET 2 OF 10)

Component	Value
Spent resin sluice pump	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	100
Design point 2	140
Design head (ft)	
Design point 1	300
Design point 2	250
Material	Stainless steel
Laundry and hot shower tank pump	
Number (per unit)	1 (Unit 1)
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
Floor drain tank pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure, (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel

TABLE 11.2-3 (SHEET 3 OF 10)

Component	Value
Waste monitor tank pumps	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
Heat Exchangers	
Reactor coolant drain tank heat exchanger	
Number (per unit)	1
Type	U-tube
Est. UA (Btu/h)(°F)	70,000
Design pressure (psi)	
Shell	150
Tube	150
Design temperature (°F)	
Shell	250
Tube	250
Design flow (lb/h)	
Shell	112,000
Tube	44,600
Temperature in (°F)	
Shell	105
Tube	180
Temperature out (°F)	
Shell	125
Tube	130
Material	
Shell	Carbon steel
Tube	Stainless steel

TABLE 11.2-3 (SHEET 4 OF 10)

Component	Value
Tanks	
Reactor coolant drain tank	
Number (per unit)	1
Usable volume (gal)	350
Type	Horizontal
Design pressure (psig)[(a)]	100
Design temperature (°F)	250
Material	Stainless steel
Diaphragm	No
Waste holdup tank	
Number (per unit)	1
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Waste evaporator condensate tank	
Number (per unit)	1
Usable volume (gal)	5000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	Optional
Chemical drain tank	
Number (per unit)	1
Usable volume (gal)	600
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No

TABLE 11.2-3 (SHEET 5 OF 10)

Component	Value
Spent resin storage tank	
Number (per unit)	2
Usable volume (ft^3)[(b)]	350
Type	Vertical
Design pressure (psig)	100
Design temperature (°F)	200
Radiation level inside compartment (R/h)	1000
Material	Stainless steel
Diaphragm	No
Laundry and hot shower tank	
Number	1 (Unit 1)
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Floor drain tank	
Number (per unit)	1
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Waste monitor tank	
Number (per unit)	2
Usable volume (gal)	5000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No

TABLE 11.2-3 (SHEET 6 OF 10)

Component	Value
Waste evaporator reagent tank	
Number (per unit)	1
Usable volume (gal)	5
Type	Vertical
Design pressure (psig)	150
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Demineralizers	
Waste evaporator condensate demineralizer	
Number (per unit)	1
Type	Flushable
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
Resin volume (ft^3)	30
Material	Stainless steel
Resin type	IRN-150[(C)]
Design process decontamination factor	100
Waste monitor tank demineralizer	
Number (per unit)	1
Type	Flushable
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
Resin volume (ft^3)	30
Material	Stainless steel
Resin type	IRN-150[(C)]
Design process decontamination factor	100
Disposable demineralizer system	
Number (shared)	1
Design flow (gal/min)	32
Design pressure (psig)	
Piping	250
Feed concentrations (ppm boron)	10-2500
Effluent concentrations (ppm boron)	10-2500
Average process decontamination factor for all isotopes	100

TABLE 11.2-3 (SHEET 7 OF 10)

Component	Value
<u>Filters</u>	
Waste evaporator feed filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles 98 percent retained (μm, nominal)	25
Radiation level (R/h)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Waste evaporator condensate filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	20
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/h)	<1
Materials	
Housing	Stainless steel
Filter Element	Polypropylene
Spent resin sluice filter	
Number (per unit)	2
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	150
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25[d]/6[e]
Radiation level (R/h)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene

TABLE 11.2-3 (SHEET 8 OF 10)

Component	Value
Laundry and hot shower tank filter	
Number	1 (Unit 1)
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (mR/h)	<100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Floor drain tank filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/hr)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Waste monitor tank filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/h)	90
Materials	
Housing	Stainless steel
Filter element	Polypropylene

TABLE 11.2-3 (SHEET 9 OF 10)

Component	Value
Strainers	
Laundry and hot shower tank strainer	
Number	1 (Unit 1)
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	Negative
Mesh number	40
Nominal rating (in.)	1/16
Radiation level	Negative
Materials	Stainless steel
Floor drain tank strainer	
Number (per unit)	1
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	Negative
Mesh number	40
Nominal rating (in.)	1/16
Radiation level	Negative
Materials	Stainless steel
Drumming header strainer	
Number	1 (Unit 1)
Type	Johnson
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	40
ΔP at design flow (psi)	Negative
Mesh number	100
Nominal rating (in.)	0.003 $\pm$ 0.001
Radiation level	Negative
Materials	Stainless steel

TABLE 11.2-3 (SHEET 10 OF 10)

Component	Value
<u>Evaporators</u>	
Waste evaporator	
Number (per unit)	1
Design flow (gal/min)	15
Steam design pressure (psig)	50
Feed concentrations (ppm boron)	10-2500
Bottoms concentrations (ppm boron)	7000-21,000
Design process decontamination factor	1000

a. External design pressure - 60 psig.
b. Total for resin and liquid.
c. Rohm and Haas Amberlite or equivalent.
d. Applies to Cuno filters.
e. Applies to ultipor GF Plus filter.

TABLE 11.2-4

RANGE OF MEASURED DECONTAMINATION FACTORS FOR SELECTED ISOTOPES[(a)]

Isotope	Minimum	Maximum
F-18	1.73×10^{1}	1.5×10^{3}
Mn-54	$>2.5 \times 10^{1}$	$>1.3 \times 10^{2}$
Co-58	3.2×10^{1}	8.2×10^{3}
I-131	1.1×10^{1}	1.6×10^{4}
I-133	1.1×10^{1}	1.8×10^{4}
I-135	1.4×10^{1}	2.0×10^{4}
Cs-137	2.4	1.3×10^{3}

a. These values were observed across mixed bed demineralizers containing cation resin in the Li-7 form and anion resin in the borated form.

TABLE 11.2-5 (SHEET 1 OF 5)

LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
Flow Instrumentation							
FI-1007	Waste evaporator feed pump discharge flow	150	200	0-30 gal/min			Local
FIC-1008	Reactor coolant drain tank pump discharge flow	150	250	0-250 gal/min		Low, 85 gal/min	WPS panel
FIA-1009	Reactor coolant drain tank recirculation flow	150	250	0-250 gal/min	Low, 85 gal/min		WPS panel
FICA-1011	Spent resin sluice pump discharge flow	150	200	0-150 gal/min	Low, 50 gal/min	Low, 50 gal/min	WPS panel
FI-1085A	Waste monitor tank pump 1 discharge flow	150	200	0-100 gal/min			WPS panel
FI-1085B	Waste monitor tank pump 2 discharge flow	150	200	0-100 gal/min			WPS panel
Pressure Instrumentation							
PIA-1004	Reactor coolant drain tank	150	250	0-100 psig	High, 8 psig		WPS panel
PIA-1006	Spent resin storage tank	150	200	0-100 psig	High, 90 psig		WPS and drumming panel
PI-1016	Waste evaporator feed pump discharge pressure	150	200	0-150 psig			Local
PI-1017	Waste evaporator feed header pressure	150	200	0-150 psig			Local

TABLE 11.2-5 (SHEET 2 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PI-1018A	Reactor coolant drain tank pump 1 discharge pressure	150	250	0-150 psig			Local
PI-1018B	Reactor coolant drain tank pump discharge pressure	150	250	0-150 psig			Local
PI-1018C	Laundry and hot shower tank pump discharge pressure	150	200	0-150 psig			Local
PI-1018G	Waste evaporator condensate pump discharge pressure	150	200	0-150 psig			Local
PI-1074	Waste evaporator outlet pressure	150	200	0-150 psig			Local
PI-1075	Waste evaporator condensate demineralizer outlet pressure	150	200	0-150 psig			Local
PI-1076	Waste evaporator condensate filter outlet pressure	150	200	0-150 psig			Local
PI-1078	Floor drain tank filter inlet pressure	150	200	0-150 psig			Local
PI-1079	Floor drain tank filter outlet pressure	150	200	0-150 psig			Local
PI-1080	Laundry and hot shower tank filter inlet pressure	150	200	0-150 psig			Local
PI-1081	Laundry and hot shower tank filter outlet pressure	150	200	0-150 psig			Local
PI-1084A	Waste monitor tank pump 1 discharge pressure	150	200	0-150 psig			Local

TABLE 11.2-5 (SHEET 3 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PI-1084B	Waste monitor tank pump 2 discharge pressure	150	200	0-150 psig			Local
PI-1086	Resin sluice filter inlet pressure	150	200	0-150 psig			Local
PI-1087	Resin sluice filter outlet pressure	150	200	0-150 psig			Local
PI-1088	Waste monitor tank filter inlet pressure	150	200	0-150 psig			Local
PI-1089	Waste monitor tank filter outlet pressure	150	200	0-150 psig			Local
PI-1090	Floor drain tank pump discharge pressure	150	200	0-150 psig			Local
<u>Level Instrumentation</u>							
LICA-1001	Waste holdup tank	150	200	0-100%	High-high, 90% High, 30% Low, 10%	Low, 15%	Local and WPS panel
LICA-1002	Chemical drain tank	150	200	0-100%	High, 90%	Low, 10% Low, 5%	Local, WPS, and drumming panels
LICA-1003	Reactor coolant drain tank	150	250	0-100%	High, 75% Low, 5%	Low, 30%	WPS panel
LICA-1005	Spent resin storage	150	200	0-100%	High, 60%	Low, 35% Low, 30%	WPS and drumming panels
LICA-1010	Laundry and hot shower tank	150	200	0-100%	High, 90% Low, 10%	Low, 15%	Local and WPS panels
LICA-1012	Waste evaporator condensate	150	250	0-100%	High, 68%	Low, 15%	Local and WPS

TABLE 11.2-5 (SHEET 4 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
	tank				Low, 10%		panels
LICA-1077	Floor drain tank	150	200	0-100%	High, 55% Low, 10%	Low, 15%	Local and WPS panels
LICA-1082	Waste monitor tank 1	150	200	0-100%	High, 90% Low, 10%	Low, 10%	Local and WPS panels
LICA-1083	Waste monitor tank 2	150	200	0-100%	High, 90% Low, 10%	Low, 10%	Local and WPS panels
Temperature Instrumentation							
TIA-1058	Reactor coolant drain tank	150	250	50-250°F	High, 170°F		WPS panel
Radiation Instrumentation							
RICA-18	Waste discharge line	150	200	10-1 million counts/min	High	Variable	WPS and radiation monitor panels
Disposable Demineralizer Instrumentation							
	Influent conductivity	250	200	0-1000 µmho/cm	-	-	Disposable demineralizer control panel (N2G21L001-N)
	Effluent conductivity	250	200	0-100 µmho/cm	-	-	Disposable demineralizer control panel (N2G21L001-N)
	Flowrate	250	200	0-72 gal/min	-	-	Disposable demineralizer control panel (N2G21L001-N)

TABLE 11.2-5 (SHEET 5 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
	Flow integrator	250	200	0-10 million gal	-	-	Disposable demineralizer control panel (N2G21L001-N)

a. The following abbreviations are used:
- F - flow
- P - pressure
- L - level
- t - temperature
- R - radiation
- I - indication
- C - control
- A - alarm

TABLE 11.2-6

RADIOACTIVE LIQUID RELEASES FROM WESTINGHOUSE-DESIGNED PWR PLANTS

Plant	Year	Cladding	Average Percentage Fuel Defects	Total Released (B+Y) Ci	Average Discharge Concentration (μCi/ml)	Fraction of Column 2, Table II, Appendix B to 10 CFR 20.1-20.601 Concentration
Yankee Rowe	1970	Stainless steel	Negligible	0.036	1.5×10^{-10}	1.5×10^{-3}
	1971	Stainless steel	0.001	0.00034	1.25×10^{-12}	1.25×10^{-5}
Connecticut Yankee	1970	Stainless steel	0.01	29.5	4.02×10^{-8}	4.02×10^{-1}
	1971	Stainless steel	0.03	5.85	7.75×10^{-9}	7.75×10^{-2}
San Onofre	1970	Stainless steel	0.007	3.41	6.1×10^{-9}	6.1×10^{-2}
	1971	Stainless steel	0.015	9.21	1.34×10^{-8}	1.34×10^{-1}
R. E. Ginna	1970	Zircaloy	0.4	9.35	1.43×10^{-8}	1.43×10^{-1}
	1971	Zircaloy	0.26	0.96	1.45×10^{-9}	1.45×10^{-2}
H. B. Robinson Unit 2	1970	Zircaloy				
	1971	Zircaloy	<0.001	0.74	1.01×10^{-9}	1.01×10^{-2}
Point Beach Unit 1	1970	Zircaloy				
	1971	Zircaloy	<0.01	0.14	2.48×10^{-10}	2.48×10^{-3}

TABLE 11.2-7 (SHEET 1 OF 2)

CALCULATED RELEASES OF RADIOACTIVE MATERIAL IN LIQUID EFFLUENTS (PER UNIT) ASSUMING EXPECTED FUEL LEAKAGE

FNP – 1 & 2 UPRATE CASE LIQUID EFFLUENTS ANNUAL RELEASE TO DICHARGE CANAL

		Coolant Concentrations									
Nuclide	Half-Life (Days)	Primary (Micro Ci/ML)	Secondary (Micro Ci/ML)	Boron RS (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/Yr)	Detergent Wastes (Ci/Yr)	Total (Ci/Yr)
CORROSION AND ACTIVATION PRODUCTS											
CP-51	2.78E+01	1.92E-03	2.02E-07	0.00000	0.00005	0.00000	0.00000	0.00006	0.00011	0.00000	0.00011
MN 54	3.03E+02	3.13E-04	4.86E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00010	0.00012
FE 55	9.50E+02	1.61E-03	1.70E-07	0.00000	0.00006	0.00000	0.00000	0.00007	0.00012	0.00000	0.00012
FE 59	4.50E+01	1.01E-03	1.24E-07	0.00000	0.00003	0.00000	0.00000	0.00003	0.00006	0.00000	0.00006
CO 58	7.13E+01	1.61E-02	1.72E-06	0.00001	0.00053	0.00003	0.00002	0.00058	0.00109	0.00040	0.00150
CO 60	1.92E+03	2.02E-03	2.18E-07	0.00000	0.00008	0.00000	0.00000	0.00008	0.00015	0.00087	0.00100
ZR 95	6.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00014	0.00014
NB 95	3.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00020	0.00020
FISSION PRODUCTS											
BR 83	1.00E-01	5.30E-03	1.98E-07	0.00000	0.00000	0.00029	0.00000	0.00030	0.00056	0.00000	0.00056
BR 84	2.21E-02	2.91E-03	3.39E-08	0.00000	0.00000	0.00005	0.00000	0.00005	0.00009	0.00000	0.00010
RB 86	1.87E+01	8.58E-05	1.03E-08	0.00000	0.00001	0.00000	0.00000	0.00002	0.00003	0.00000	0.00003
RB 88	1.24E-02	2.25E-01	1.41E-06	0.00000	0.00000	0.00053	0.00000	0.00053	0.00098	0.00000	0.00098
SR 89	5.20E+01	3.53E-04	4.96E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002
MO 99	2.79E+00	8.60E-02	1.04E-05	0.00003	0.00008	0.00016	0.00010	0.00036	0.00068	0.00000	0.00068
TC 99M	2.50E-01	5.19E-02	2.87E-05	0.00002	0.00008	0.00043	0.00019	0.00072	0.00134	0.00000	0.00130
RU103	3.96E+01	4.54E-05	4.99E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001	0.00002
RU106	3.67E+02	1.01E-05	1.21E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00024	0.00024
AG110M	2.53E+02	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00004	0.00004
TE127M	1.09E+02	2.82E-04	2.20E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002

TABLE 11.2-7 (SHEET 2 OF 2)

FNP – 1 & 2 UPRATE CASE LIQUID EFFLUENTS ANNUAL RELEASE TO DICHARGE CANAL

		Coolant Concentrations									
Nuclide	Half-Life (Days)	Primary (Micro Ci/ML)	Secondary (Micro Ci/ML)	Boron RS (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/Yr)	Detergent Wastes (Ci/Yr)	Total (Ci/Yr)
TE127	3.92E-01	9.09E-04	1.57E-07	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002
TE129M	3.40E+01	1.41E-03	1.51E-07	0.00000	0.00004	0.00000	0.00000	0.00004	0.00008	0.00000	0.00008
TE129	4.79E-02	1.78E-03	9.27E-07	0.00000	0.00003	0.00001	0.00000	0.00004	0.00008	0.00000	0.00008
I130	5.17E-01	2.23E-03	1.72E-07	0.00000	0.00000	0.00026	0.00001	0.00027	0.00051	0.00000	0.00051
TE131M	1.25E+00	2.59E-03	2.30E-07	0.00000	0.00000	0.00000	0.00000	0.00001	0.00001	0.00000	0.00001
TE131	1.74E-02	1.23E-03	8.64E-07	0.00000	0.00000	0.00001	0.00000	0.00001	0.00003	0.00000	0.00002
I131	8.05E+00	2.74E-01	3.08E-05	0.00100	0.00300	0.04590	0.00300	0.05290	0.09883	0.00001	0.09900
TE132	3.25E+00	2.76E-02	2.66E-06	0.00001	0.00004	0.00004	0.00003	0.00012	0.00022	0.00000	0.00022
I132	9.58E-02	1.10E-01	1.47E-05	0.00006	0.00005	0.02188	0.00026	0.02225	0.04156	0.00000	0.04200
I133	8.75E-01	3.98E-01	3.49E-05	0.00094	0.00002	0.05205	0.00285	0.05586	0.10435	0.00000	0.10000
I134	3.67E-02	5.25E-02	9.16E-07	0.00000	0.00000	0.00137	0.00000	0.00137	0.00256	0.00000	0.00260

a. From GALE-CODE calculations.

TABLE 11.2-8 (SHEET 1 OF 2)

COMPARISON OF CALCULATED CONCENTRATIONS IN EFFLUENT WATER DISCHARGE WITH CONCENTRATION VALUES STATED IN COLUMN 2, TABLE II, APPENDIX B TO 10 CFR 20

ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cr-51	1.10E+02	4.32E-12	5.00E-04	8.64E-09
Mn-54	1. 20E+02	4. 71E-12	3.00E-05	1.57E-07
Fe-55	1.20E+02	4.71E-12	1.00E-04	4.71E-08
Fe-59	6.00E+01	2.36E-12	1.00E-05	2.36E-07
Co-58	1.50E+03	5.89E-11	2.00E-05	2.95E-06
Co-60	1.00E+03	3.93E-11	3.00E-06	1.31E-05
Zr-95	1.40E+02	5.50E-12	2.00E-05	2.75E-07
Nb-95	2.00E+02	7.86E-12	3.00E-05	2.62E-07
Br-83	5.60E+02	2.20E-11	9.00E-04	2.44E-08
Br-84	1.00E+02	3.93E-12	4.00E-04	9.82E-09
Rb-86	3.00E+01	1.18E-12	7.00E-06	1.68E-07
Rb-88	9.80E+02	3.85E-11	4.00E-04	9.62E-08
Sr-89	2.00E+01	7.86E-13	8.00E-06	9.82E-08
Mo-99	6.80E+02	2.67E-11	8.00E-06	3.34E-06
Tc-99m	1.30E+03	5.11E-11	1.00E-03	5.11E-08
Ru-103	2.00E+01	7.86E-13	3.00E-05	2.62E-08
Ru-106	2.40E+02	9.43E-12	3.00E-06	3.14E-06
Ag-110m	4.00E+01	1.57E-12	6.00E-06	2.62E-07
Te-127m	2.00E+01	7.86E-13	9.00E-06	8.73E-08
Te-127	2.00E+01	7.86E-13	1.00E-04	7.86E-09
Te-129m	8.00E+01	3.14E-12	7.00E-06	4. 49E-07
Te-129	8.00E+01	3.14E-12	4.00E-04	7.86E-09
I-130	5.10E+02	2.00E-11	2.00E-05	1.00E-06
Te-131m	1.00E+01	3.93E-13	8.00E-05	4.91E-09
Te-131	2.00E+01	7.86E-13	8.00E-05	9.82E-10
I-131	9.90E+04	3.89E-09	1.00E-06	3.89E-03
Te-132	2.20E+02	8.64E-12	9.00E-06	9.60E-07
I-132	4.20E+04	1.65E-09	1.00E-04	1.65E-05
I-133	1.00E+05	3.93E-09	7.00E-06	5.61E-04
I-134	2.60E+03	1.02E-10	4.00E-04	2.55E-07
Cs-134	1.30E+04	5.11E-10	9.00E-07	5.67E-04
I-135	3.70E+04	1.45E-09	3.00E-05	4.84E-05

TABLE 11.2-8 (SHEET 2 OF 2)

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cs-136	3.70E+03	1.45E-10	6.00E-06	2.42E-05
Cs-137	1.10E+04	4.32E-10	1.00E-06	4.32E-04
Ba-137m	7.30E+03	2.87E-10	--	--
Ce-144	5.20E+02	2.04E-11	3.00E-06	6.81E-06
All Others	6.00E+01	2.36E-12	2.00E-09	1.18E-03
TOTAL	3.24E+05	1.27E-08	--	6.75E-03
H-3	5.50E+08	2.16E-05	1.00E-03	2.16E-02
Total+H-3	5.50E+08	2.16E-05	--	2.84E-02

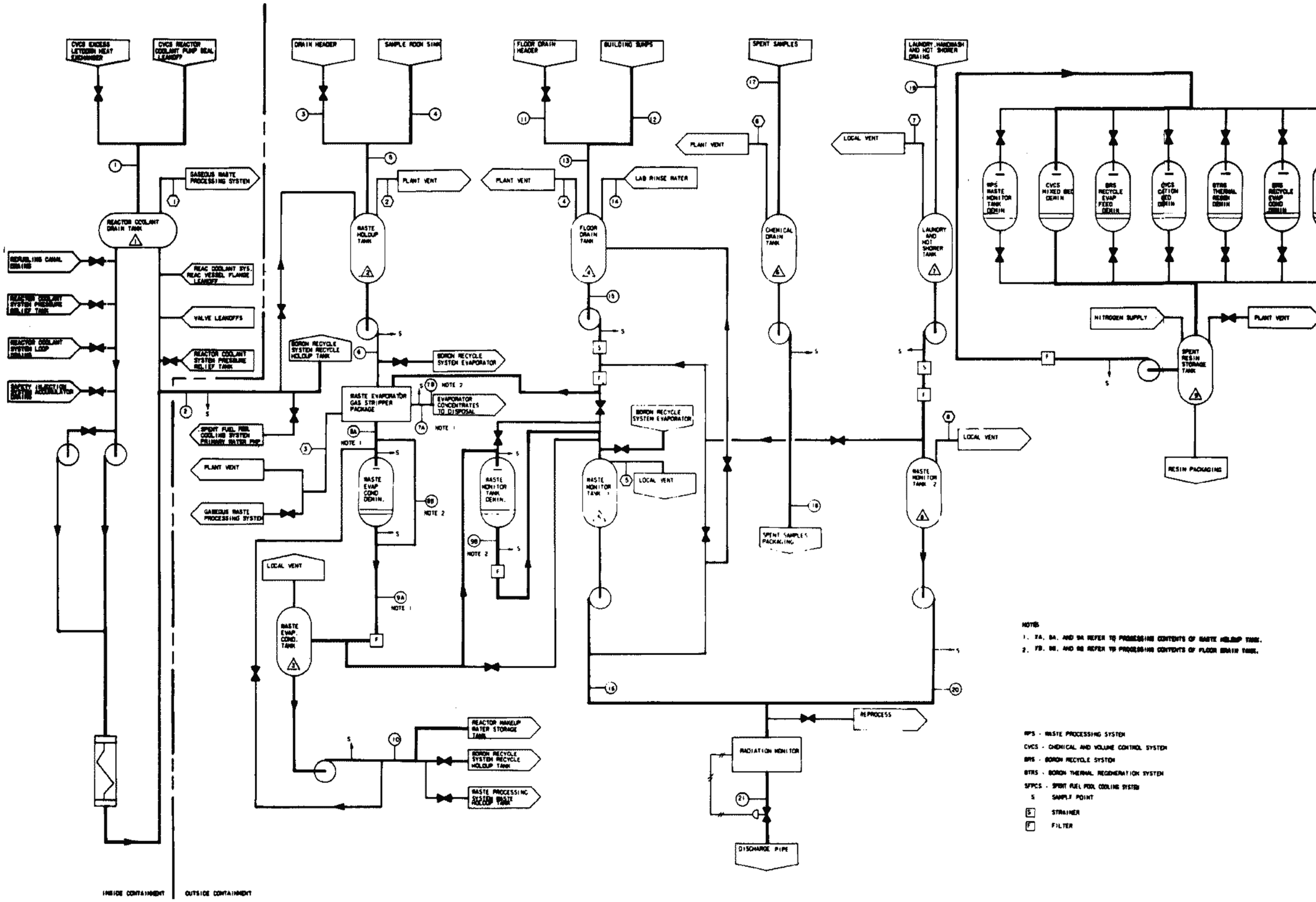
a. Based on the estimated isotopic liquid effluents in Table 11.2-7.
b. Based on the 16,000 gal/min/unit discharge in FSAR subsection 11.2.8.
c. Column 2, Table II, Appendix B to 10 CFR 20.

TABLE 11.2-9

ESTIMATED MAXIMUM INDIVIDUAL DOSE FROM LIQUID EFFLUENTS (mrem/year)[(a)]

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Drinking water	Adult								
	Teen				Not Applicable (See Text)				
	Child								
	Infant								
Fish Ingestion	Adult	6.45E-01	4.30E-02	4.51E-01	8.54E-01	2.94E-01	3.89E-01	9.87E-02	0.00E+00
	Teen	3.75E-01	3.14E-02	4.72E-01	8.73E-01	2.96E-01	3.65E-01	1.13E-01	0.00E+00
	Child	1.53E-01	1.39E-02	5.83E-01	7.55E-01	2.48E-01	3.82E-01	8.88E-02	0.00E+00
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Shoreline Recreation	Adult	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	5.14E-04
	Teen	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.87E-03
	Child	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.99E-04
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Boating	Adult	-	-	-	-	-	-	-	-
	Teen	-	-	-	-	-	-	-	-
	Child	-	-	-	-	-	-	-	-
	Infant	-	-	-	-	-	-	-	-
Swimming	Adult	-	-	-	-	-	-	-	-
	Teen	-	-	-	-	-	-	-	-
	Child	-	-	-	-	-	-	-	-
	Infant	-	-	-	-	-	-	-	-
Totals	Adult	6.45E-01	4.34E-02	4.51E-01	8.54E-01	2.94E-01	3.89E-01	9.91E-02	5.14E-04
	Teen	3.77E-01	3.39E-02	4.74E-01	8.75E-01	2.98E-01	3.67E-01	1.15E-01	2.87E-03
	Child	1.54E-01	1.44E-02	5.84E-01	7.56E-01	2.49E-01	3.83E-01	8.93E-02	5.99E-04
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

a. Appendix I Design Objectives for Liquid Effluents: total body dose = 3 mrem/year per unit from all pathways; dose to any organ = 10 mrem/year per unit from all pathways. Docket RM-50-2 Annex to Appendix I Design Objectives: total body dose = 5 mrem/year per site from all pathways; dose to any organ = 5 mrem/year per site from all pathways; nontritium releases = 5 Ci/year per unit.b. Dash (-) indicates dose less than 0.001 mrem/year.



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAM
LWPS TABULATED ACTIVITIES

FIGURE 11.2-1

11.3 GASEOUS WASTE SYSTEMS

11.3.1 DESIGN OBJECTIVES

The gaseous waste processing system (GWPS) was designed to remove fission product gases from the reactor coolant and have the capacity to contain these throughout the 40-year plant life. This was based on continuous operation with reactor coolant system activities associated with operation, with cladding defects in the fuel rods generating 1% of the rated core thermal power. The system was also designed to collect and store expected fission gases from the boron recycle evaporator and reactor coolant drain tank throughout the plant life.[a]

Although the system has the design capacity to contain fission product gases for the life of the plant, operating experience has demonstrated that the waste gas decay tanks must be released periodically due to nitrogen buildup. These releases are necessary following degassification of the reactor coolant system (RCS) prior to each RCS maintenance or refueling outage and again following deoxygenation at the end of each RCS maintenance or refueling outage. These releases are monitored and quantified in accordance with the radiological effluent technical specifications. At a preset level in the plant vent stack, waste gas decay tank releases are automatically terminated, as described in paragraph 11.4.2.2.5.

Experience has also shown that fuel defect levels have been extremely low. This greatly reduces the benefit of continuous purging of the volume control tank (VCT). Thus, during normal operation with low fuel defect levels, the VCT purge may not be performed. Without this continuous input of hydrogen with trace fission gases, there is no need to continuously operate the GWPS. When the GWPS is required during periods of high RCS fission gas concentrations and the VCT purge is initiated, the compressor and gas decay tanks can be utilized in a compressed storage mode of operation.

Gaseous activity released due to equipment leakage during normal operation of the plant is mixed with ventilation exhaust and is further diluted due to atmospheric dilutions. Table 11.3-9 gives estimated activity discharges from the plant vent stack.

The plant design considers potential personnel exposure and ensures that quantities of gaseous radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from gaseous radionuclides leaving the GWPS does

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

not exceed a small fraction of the discharge limits as defined in column 1, Table II, Appendix B to 10 CFR 20. Although plant operating procedures, equipment inspection, and preventive maintenance are performed during plant operations to minimize equipment malfunction, overall radioactive release limits have been established as a basis for controlling plant discharges during operation with the occurrence of a combination of equipment faults of moderate frequency. A combination of equipment faults which could occur with moderate frequency include operation with fuel defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in liquid waste processing system.

C. Malfunction of GWPS.

D. Excessive leakage in reactor coolant system equipment.

E. Excessive leakage in auxiliary system equipment.

The radioactive releases from the plant resulting from equipment faults of moderate frequency are within column 1, Table II, Appendix B to 10 CFR 20 limits on the short-term basis and do not exceed four to eight times the limits stated previously for normal operation.

11.3.2 SYSTEM DESCRIPTION

The GWPS consists mainly of a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, and gas decay tanks to accumulate the fission product gases.

The major input to the GWPS during normal operation is taken from the gas space in the volume control tank. The volume control tank gas space may be purged at a rate of 0.7 sf^3/min. Table 11.3-2 lists the rate of activity input to the GWPS during normal operation. There are no liquid seals in the system. The system is designed to preclude explosions by keeping the concentration of hydrogen and oxygen below the explosive limits. The Technical Requirements Manual contains limits for the concentration of hydrogen and oxygen.

Process flow diagrams and piping and instrumentation diagrams are shown in figure 11.3-1 and drawings D-175042, sheets 5, 6, 11, and 12, and D-205042, sheets 5, 6, 9, and 10. All lines in the gaseous waste system, including field run, are considered potential carriers of significant radioactivity. Only non-Category I pipe of Class B31.1 of the American National Standards Institute (size 2 in. and under) will be field run. This piping is shown on the piping and instrumentation drawings and is designated as Safety Class NNS. Table 11.3-3 gives process parameters for key locations in the system.

The GWPS includes two waste gas compressors and eight gas decay tanks to accumulate fission gases. Seven of these gas decay tanks are used during operation in the compressed gas storage mode, while the eighth one is normally used to accept relief valve discharges and administratively approved inputs.

11.3.3 SYSTEM DESIGN

11.3.3.1 Component Design

Gaseous waste processing equipment parameters are given in Table 11.3-4. For further information on safety and seismic classification see chapter 3. Paragraph 3.9.2.7 gives the general design criteria for field run piping.

11.3.3.1.1 Waste Gas Compressors

Two waste gas compressor packages are provided. One unit is normally used, with the other on a standby basis.

The units are centrifugal displacement machines which are skid-mounted in a self-contained package. Construction is primarily of carbon steel. Mechanical seals are provided to minimize the outleakage of seal water. The compressor has been used in Westinghouse pressurized water reactor (PWR) plants with excellent experience.

11.3.3.1.2 Recombiners

Two catalytic hydrogen recombiners are provided. As shown on drawing D-175042, sheets 11 and 12, one of the two recombiners is normally used to remove hydrogen from the hydrogen-nitrogen fission gas mixtures by oxidation to water vapor, which is removed by condensation. The other recombiner is available on a standby basis. Both units are self-contained and designed for continuous operation. In the compressed storage operating mode, the recombination function is not used. However, the recombiners and H_2 and O_2 analyzers remain installed.

11.3.3.1.3 Gas Decay Tanks

Gas decay tanks are provided as described in table 11.3-4. The tanks used during power operation are of vertical cylindrical type, and the shutdown tanks are horizontal cylindrical. All the gas decay tanks are constructed of carbon steel.

11.3.3.1.4 Valves

Each valve in the system is designed to meet the temperature, pressure, and code requirements for the specific application in which it is used. Special consideration is given to leaktightness. The recombiner circuits use manual valves provided with a diaphragm to prevent stem leakage and control valves with leakoffs returned to the gas system. Other parts of the gas system use Saunders valves and control valves with bellows seal.

11.3.3.2 Instrumentation Design

The main system instrumentation is described in table 11.3-5 and shown in the flow and piping diagrams of drawings D-175042, sheets 5, 6, 11, and 12, and D-205042, sheets 5, 6, 9, and 10.

The instrumentation readout is located mainly on the waste processing system panel in the auxiliary building. Some instruments are read where the equipment is located.

All alarms are shown separately on the waste processing system panel and further relayed to one common system annunciator on the main control board of the plant.

When used, the catalytic recombiner system is designed for automatic operation with a minimum of operator attention. Each package includes four online gas analyzers which are the primary means of recombiner control. A multipoint temperature recorder monitors temperatures at several locations in the recombiner packages.

Process gas flowrate is measured by an orifice located upstream of the recombiner preheater. Local pressure gauges indicate pressures at the recombiner inlet and the oxygen supply pressure.

11.3.4 OPERATING PROCEDURES

The gaseous waste processing system at Farley Nuclear Plant and systems of similar design have been operating for several years with excellent experience, as far as components and overall system performance are concerned.

Systems constructed from carbon steel have been in service for many years, and no failure due to corrosion damage has been reported.

Components of identical design to those used for the FNP are in use on several Westinghouse-designed GWPS. The performance and operating history of the compressors have been excellent.

11.3.4.1 Operation with Continuous VCT Purge

Prior to the system being put into operation, the GWPS is flushed free of air and filled with nitrogen. During normal power operation, nitrogen gas is continuously circulated around the loop by one of the two compressors. Fresh hydrogen gas is charged to the volume control tank, where it is mixed with fission gases which are stripped from the reactor coolant into the tank gas space. The contaminated hydrogen gas is then vented from the tank into the circulating nitrogen stream to transport the fission gases into the GWPS. The resulting mixture of nitrogen-hydrogen fission gas is pumped by the compressor to the recombiner where enough oxygen is added to reduce the hydrogen to a low residual level by oxidation to water vapor on a catalytic surface. After the water vapor is removed, the resulting gas stream is circulated to the gas decay tanks and back to the compressor suction to complete the loop circuit.

Each gas decay tank is capable of being isolated, and the number of tanks valved into operation at any time is restricted to diminish the amount of radioactive gases which could be released as a consequence of any single failure, such as the rupture of any single tank or connected piping. By alternating use of these tanks, the accumulated activity is contained in approximately equal parts.

When the hydrogen contained in the reactor coolant must be removed in preparation for a cold shutdown, two methods are available for removal. In the first method, the gas decay tanks are valved out of service and one of the two shutdown tanks is placed in service between the compressor discharge and the recombiner suction. The first degassing operation will require that the shutdown tank be pressurized with nitrogen before degassing begins. In addition, the flow of hydrogen to the volume control tank is stopped, the bypass on the volume control tank vent line is opened, and purge flow from the shutdown tank to the volume control tank is initiated, thus establishing a recirculation path between the GWPS and the volume control tank. The flow of gas through the volume control tank is controlled at the flowrate required to support RCS, CVCS, and GWPS operational requirements.

Initially, the flow will be predominantly hydrogen, but as degassing progresses the gas will become primarily nitrogen. Because of the difference in density between the gases, the throttle valve in the bypass line may require adjustment during the degassing operation to maintain a constant flowrate.

The alternative method of dissolved hydrogen removal consists of the controlled addition of hydrogen peroxide, which reacts with the dissolved hydrogen to form water. In this method, the reactor coolant system is sampled and analyzed for the dissolved hydrogen concentration. The gas spaces of various tanks which may contain hydrogen are sampled to establish initial hydrogen or oxygen concentration. Then, the stoichiometric amount of hydrogen peroxide is calculated and added to the charging pump suction via the chemical addition tank. Sampling is again performed to confirm that the reactor coolant dissolved hydrogen concentration is less than 5 cc/kg and that no hazardous mixtures of hydrogen and oxygen have been created.

11.3.4.2 Operation Without Continuous VCT Purge

Although the GWPS was designed for continuous purge of the VCT and 40-year holdup of fission gases, operating experience at FNP has shown that the GWPS can be operated without a continuous purge while maintaining personnel exposure within limits and maintaining releases within concentration and offsite dose limits. Many other operating PWRs are not designed with continuous purge capability and have operated for many years with gaseous releases from the GWPS well within MPC and offsite dose limits.

Fission gas production is directly related to fuel integrity. Fuel defects have been minimal at plants with Westinghouse fuel and, therefore, fission gas RCS concentrations are normally well below design limits (1-percent fuel defects).

The purpose of the VCT purge is to strip fission gases from the reactor coolant to reduce the exposure to personnel from fission gases which escape with reactor coolant leakage. However, the primary contributor to exposure from leakage is Co-60, which is dissolved in the liquid. The only fission gases with significant half-lives are Xe-133 and Kr-85. Without a continuous VCT purge, Xe-133 will reach an equilibrium value in the RCS in about 30 days. Without fuel defects, the equilibrium concentration will be orders of magnitude less than design values and will not contribute to doses as a result of reactor coolant leakage. Kr-85 will accumulate in the RCS because of its long half-life. However, being a beta-emitter, it is not expected to contribute significantly to personnel exposure. Therefore, the benefit of the continuous purge is limited when fission gas concentrations are already low.

When RCS fission gas concentrations are low, there is no need for continuous inputs to the GWPS in the compressed gas storage mode. Therefore, the system is shut down. When it is necessary to vent gases from other systems to the GWPS, a waste gas compressor and gas decay tank are aligned in a recirculation loop or remain shutdown. These small inputs are normally accumulated in a gas decay tank, isolated for a period of decay, then released. Refer to plant procedures for information regarding the transfer of gases to the GWPS.

When RCS fission gas concentrations are high, the VCT purge may be initiated in the compressed gas mode to allow the hydrogen steam to carry fission gases into the GWPS recirculation loop. Each gas decay tank is aligned prior to the previous tank exceeding its pressure limit. Filled GDTs are isolated for decay of the fission gases. VCT purge operations may be planned such that by the time the seventh tank is filled, the contents of the first tank are released and the tank is available to collect gases again. This process may continue until RCS fission gas concentrations reach an acceptable level.

Doses resulting from a steam generator tube rupture accident have been evaluated without taking credit for VCT purge (see section 15.4). Regarding the gas decay tank rupture accident, the offsite dose will be limited by maintaining the activity in each gas decay tank within the Technical Requirements Manual limit of 70,500 Ci. Without the VCT purge, gas decay tank activity accumulation will be drastically reduced during normal operation but will increase during plant shutdown, due to RCS fission gas activity buildup. In spite of this buildup, the gas decay tank curie limit can still be maintained by the normal practice of spreading the activity among two or more tanks during shutdown degassing.

Therefore, the VCT purge is not required during normal power operation and can be aligned in or out of service at the operator's discretion. This philosophy allows for a simpler and more reliable GWPS operation as described below.

11.3.4.2.1 Plant Startup

This operation remains the same as with a VCT purge. The nitrogen gas space in the VCT is replaced with hydrogen by burping the gas space to the GWPS until the required RCS dissolved hydrogen concentration is achieved.

11.3.4.2.2 Normal Power Operation

The VCT purge is normally isolated. Without the major hydrogen input, the need for the recombiner operation is reduced. One compressor and one gas decay tank may be placed in service as necessary to accommodate the very small flow and volumetric inputs from the reactor coolant drain tank vent, the recycle evaporator vent, and the recycle holdup tank diaphragm vent. The VCT purge can be initiated as required to reduce RCS radiogas inventory at the discretion of the plant operators. At a nominal flowrate of 0.7 sf^3/min, a gas decay tank will be filled in 4 days. Therefore, a recombiner must eventually be placed into service, depending upon the available capacity in the remaining gas decay tanks.

11.3.4.2.3 Plant Shutdown

If shutdown is required such that the RCS is opened to the containment atmosphere (e.g., refueling or maintenance), the RCS fission gas and hydrogen concentration must be reduced to required levels. The VCT gas space is burped to the GWPS and fresh nitrogen is aligned. With the reactor shut down, no additional fission gases are produced. Therefore, the VCT burping needs to remove only residual fission gases, if their concentration is above the required shutdown limit. In this mode, the GWPS compressor and gas decay tank must be available to receive and collect the VCT burp volume. The gases can be stored for decay as required, then released as during normal plant operation.

11.3.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation, and control equipment.

Periodic testing of the oxygen monitors and hydrogen monitors is conducted in accordance with the Technical Requirements Manual.

11.3.6 ESTIMATED RELEASES

11.3.6.1 <u>Nuclear Regulatory Commission Requirements</u>

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for radioactive releases:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.42, Revision 1, “Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light Water Cooled Nuclear Power Reactors,” March 1974.

Regulatory Guide 1.42, which was in effect during plant design, was withdrawn by the NRC in 1976 and replaced by guidance presented in the following regulatory guides:

A. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I,” October 1977.

B. Regulatory Guide 1.111, Revision 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” July 1977.

C. Regulatory Guide 1.112, Revision O-R, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Reactors,” April 1976.

D. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications Nuclear Power Plants,” October 1978.

During plant operations, radioactive gaseous releases will be controlled in accordance with technical specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, the use of dose rate values based on the guidance contained in NUREG-0133[(4)] is acceptable for use as a technical specification limit for gaseous effluent release rates since operational history at Farley Nuclear Plant has demonstrated that, with these dose rate limits in effect, the calculated maximum individual doses to members of the public are small percentages of the limits of 10 CFR 50, Appendix I and 40 CFR 190.

11.3.6.2 Radioactive Noble Gas Releases From FNP

A summary of gaseous discharges from FNP for years 1977 to 1983 is presented in table 11.3-6.

11.3.6.3 Expected GWPS Releases (Recombiner or Compressed Gas Mode)

The GWPS is designed to remove fission product gases from the volume control tank and recycle evaporator and was designed with the capacity to contain them through the lifetime of the plant. Since the VCT purge to the GWPS reduces fission product gas concentrations in the reactor coolant during unit operation, it reduces the escape of radioactive gases arising from any possible reactor coolant leakage. Design is based on continuous operation, with reactor coolant system activities associated with operation, with cladding defects in fuel rods generating

1% of the rated core thermal power. Table 11.3-7 shows the maximum fission product inventory in the GWPS over the 40-year plant life based on a 1-unit plant.[a]

Figure 11.3-2 shows that, for a given power rating, the quantity of fission gas activity accumulated in the gas system after 40 continuous years of operation is only twice the activity accumulated after 30 days of operation.[a] This is because most of the accumulated activity arises from short-lived isotopes reaching equilibrium in 1 month or less.

The difference between the 30-day and 40-year accumulations is essentially all Kr-85. This accumulation of Kr-85 is not a hazard to the plant operator because:

A. Radiation background levels in the plant are not noticeably affected by the accumulation of Kr-85, which is a beta emitter for which the tanks themselves provide adequate shielding.

B. The system activity inventory is distributed in several tanks so that the maximum permissible inventory in any single tank is actually less than that of earlier GWPS designs.

C. Since this system permits fission gas removal from the reactor coolant during normal operation, it is expected to reduce plant activity levels caused by a leakage of reactor coolant.

The capability to release a waste gas decay tank directly to the plant vent stack was provided as part of the original design of each unit. Automatic shutoff for such release occurs at a preset vent stack radiogas monitor setpoint as described in paragraph 11.4.2.2.5.

To further ensure design basis releases in accordance with the "as low as reasonably achievable" philosophy, the plant Offsite Dose Calculation Manual establishes limits for the releases. The quantity of radioactivity contained in each waste gas storage tank is limited by the Technical Requirements Manual to 70,500 Ci.

11.3.6.4 Releases from Ventilation Systems

A detailed analysis of one unit of the plant has been made to ascertain those items that could possibly contribute to airborne radioactive releases. Results of the analysis are presented in Section 11.3.9.

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

During normal plant operations, airborne noble gases and/or iodines can originate from reactor coolant leakage, equipment drains, venting and sampling, secondary side leakage, condenser air ejector and gland seal condenser exhausts, GWPS leakage, refueling operations, and evaporations from the spent fuel pool.

11.3.6.5 Estimated Total Releases

The potential release from the sources discussed in subsections 11.3.6.3 and 11.3.6.4 has been evaluated. Radioactive effluent releases from the plant for normal operation are given in table 11.3-9. These release rates were calculated using a composite of the PWR-GALE code[1] and plant operating parameters referenced in paragraph 11.1.1.2 and table 11.1-7 for operation in the continuous purge mode (section 11.3.4.1) or in the compressed storage mode (section 11.3.4.2). The releases are calculated for one unit; to obtain the combined releases for the two units, double the values listed in table 11.3-9.

The dose calculations, based on the estimated total plant releases, show that the releases are in accordance with the design objectives in subsection 11.3.1 and meet the regulations and guidelines as outlined in subsection 11.3.6.1. Further, the total plant releases, noted in table 11.3-10, are within the plant technical specifications, which are developed to be consistent with the "as low as reasonably achievable" criterion and the concentration limits specified in column 1, Table II, Appendix B to 10 CFR 20.

11.3.7 RELEASE POINTS

The GWPS is designed to contain all fission product gases generated during the plant lifetime. Any gases that do leak from the system are swept up by the radwaste area ventilation system, which discharges the gas to the plant vent stack.

The vent stack is the principal release point of gaseous waste to the environment. However, in the event of primary to secondary leakage, the power-operated atmospheric relief valve vents and the turbine building vent could become release points.

The vent stack is shown as part of the ventilation system in drawings D-175045 and D-205045. The physical location of the stack, shown in the plant general arrangement drawings of figures 1.2-1 through 1.2-9, exhausts at a height of 145 ft 9 in. above grade.

The main exhaust line from the radwaste area ventilation system to the vent stack is a 6-ft diameter duct which is flanged into the vent stack.

The vent stack parameters are as follows:

A. Base elevation - 155 ft (same as ground elevation).

B. Orifice elevation - not applicable.

C. Orifice inside diameter - not applicable.

D. Effluent velocity at flange of main exhaust line 2650 ft/min.

E. Heat input to stack - 39,078 Btu/h.

11.3.8 DILUTION FACTORS

Gaseous and particulate radioactive effluents may be normally released from the plant vent and turbine building vent as discussed in 11.3.7. Subsection 2.3.5 outlines the methodology and information used to determine the long-term atmospheric dilution (X/Q) and deposition (D/Q) for these release points. Effluent sources and associated vents are listed in table 2.3-15. Vent design information and input assumptions utilized for the long-term diffusion estimates are given in tables 2.3-16 and 2.3-17.

11.3.9 ESTIMATED DOSES[1]

[HISTORICAL]

[*Dose models, usage factors, and other parameters used to estimate the maximum doses to individuals from discharges of gaseous and particulate radioactive effluents are those described in Regulatory Guide 1.109.[3] These models were applied to the FNP using the source terms and atmospheric dilution factors discussed in subsections 11.3.6.5 and 11.3.8.*

Pathways included are plume exposure, ground shine, inhalation, and food ingestion (cow or goat milk, vegetation, and meat). Beta and gamma radiation doses to air were determined for the offsite location with the highest potential dose.

Receptor locations were selected by inspection of dispersion parameter values at locations tabulated in Tables 2.3-21 and 2.3-22. Doses were evaluated for a number of locations at which real receptors exist. Results were reviewed to identify the maximally exposed individual. This process is considered to yield doses which are unlikely to be substantially underestimated.

*The historical dose estimates above were calculated to confirm that the Farley Nuclear Plant conforms with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).****]***

Detailed results are presented in Table 11.3-11 for the maximally exposed individual. This table provides a breakdown by organ and pathway, and doses include the summation from both assumed plant discharge points given in Table 2.3-16. Furthermore, the total dose to each organ is given in Table 11.3-11 along with the Appendix I design objective doses for comparison. It is clear that the estimated doses follow the design objective dose in each case. Actual plant releases during normal operation are governed by the Farley Nuclear Plant Technical Specifications and Offsite Dose Calculations Manual.

REFERENCES

1. Alabama Power Company letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," USNRC Docket Nos. 50-348, 50-364, June 3, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

3. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

4. U.S. Nuclear Regulatory Commission, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," NUREG-0133, October 1978.

TABLE 11.3-1

(This table has been intentionally deleted.)

TABLE 11.3-2

ACTIVITY INPUT TO THE GASEOUS WASTE PROCESSING SYSTEM DURING NORMAL OPERATION[a]

Isotope	Input (Ci/year)
Kr-85	3.21×10^{3}
Kr-85m	3.16×10^{4}
Kr-87	1.58×10^{4}
Kr-88	6.31×10^{4}
I-131	20.5
I-132	7.53
I-133	33
I-134	4.47
I-135	17.66
Xe-131m	4.73×10^{3}
Xe-133	2.36×10^{6}
Xe-133m	4.42×10^{4}
Xe-135	1.26×10^{5}
Xe-135m	3.7×10^{2}
Xe-138	1.37×10^{3}

a. The table is based on 1 percent fuel defects and no decay.

TABLE 11.3-3

PROCESS FLOW DIAGRAM - GWPS – TABULATED ACTIVITIES

WASTE PROCESSING SYSTEM (GAS) - OPERATING PARAMETERS

BASIS: FUEL DEFECTS - 0.25% | GAS DECAY TANKS (NOTE 4) - 8
POWER LEVEL - 2774 | OPERATING INTERVAL - 2 DAYS
NO. OF UNITS - 1 | STRIPPING EFFICIENCY - 0.4

ITEM	DESCRIPTION	TEMP	PRESS	FLOW	N_2	H_2	ISOTOPIC CONCENTRATION, μC/CC (NOTE 1)						
	(GAS STREAMS)	°F	PSIG	SCFM	%	%	KR85 (NOTE 3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
1	VOLUME CONTROL TANK PURGE	130	15	0.7	0	100	0.11	0.73	0.38	1.45	54.3	1.35	2.65
2	GAS DECAY TANK DISCH. TO COMP.	AMB	0.5	40	99.9	0.1	29	0.29	0.04	0.37	225	5.10	2.38
3	COMPRESSOR SUCTION	AMB	0.5	40.7	98.2	1.8	24.5	0.30	0.05	0.39	221	5.03	2.38
4	COMP. DISCH. TO RECOMBINER	140	45	40.7	98.2	1.8	24.5	0.30	0.05	0.39	22	5.03	2.38
5	RECOMBINER DISCH. TO GAS DECAY TANKS	140	30	40	99.9	0.1	25	0.29	0.04	0.32	225	5.10	2.38
6	MISC. VENTS - EVAPS. RCDT, RHUTEDUCTOR	140	0.5		0	100							
7	RECOMBINER OXYGEN SUPPLY	AMB	50	0.35	0	0	0	0	0	0	0	0	0
8	RECOMBINER CALIBRATING GAS	AMB	15	0.004	94	6	0	0	0	0	0	0	0
9	RECOMBINER CALIBRATING GAS	AMB	ATM	0.004	94	6	0	0	0	0	0	0	0
10	WASTE GAS SYSTEM NITROGEN SUPPLY	AMB	100	0	100	0	0	0	0	0	0	0	0
11	NSSS NITROGEN SUPPLY	AMB	100	0	100	0	0	0	0	0	0	0	0
12	NITROGEN RELIEF TO PLANT VENT	AMB	100	0	100	0	0	0	0	0	0	0	0
13	NSSS HYDROGEN SUPPLY	AMB	100	0.7	0	100	0	0	0	0	0	0	0
14	VOLUME CONTROL TANK HYDROGEN	AMB	100	0.7	0	100	0	0	0	0	0	0	0
15	HYDROGEN RELIEF TO PLANT VENT	AMB	100	0	0	100	0	0	0	0	0	0	0
16	WASTE GAS DISCH. TO PLANT VENT	AMB	ATM	0	100	0	25	0	0	0	0	0	0
17	RECYCLE GAS TO VOLUME CONTROL TANK	AMB	100	0	100	0	0	0	0	0	0	0	0
18	PRESSURIZER RELIEF TANK VENT AND RETURN	120	3	0	100	0							
19	SHUTDOWN TANK RELIEF	AMB	ATM	0	100	0	0	0	0	0	0	0	0

ITEM	DESCRIPTION	TEMP	PRESS	FLOW	ISOTOPIC CONCENTRATION, μC/CC (NOTE 2)						
	LIQUID STREAMS	°F	PSIG	GPD	KR85 (NOTE 3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
1	WASTE GAS COMPRESSOR DRAIN	140	45	0	5.96	.073	.01	.096	44.5	1.0	0.48
2	RECOMBINER DRAIN	140	30	6	4.33	.053	.008	.068	32.5	.75	0.35
3	GAS DECAY TANK DRAINS	AMB	40	36	1.78	.023	.003	.028	15.3	0.3	0.15
4	SYSTEM DRAINS TO VOL. CONTROL TANK	140	30-45	42	2.28	.028	.005	.035	17.0	0.38	0.18
5	RECOMBINER REACTOR MAKEUP WATER	AMB		0	0	0	0	0	0	0	0
6	COMPRESSOR MAKEUP WATER	AMB		36	0	0	0	0	0	0	0

ITEM	COMPONENT	TEMP	PRESS	N_2	H_2	COMPONENT INVENTORY, CURIES						
		°F	PSIG	%	%	KR85 (3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
A	COMPRESSOR	140	45	98.2	1.8	11.25	0.13	0.018	0.17	101.8	2.53	1.08
B	RECOMBINER	140	30	99.9	0.1	8.5	0.098	0.014	0.13	78	1.73	0.80
C	GAS DECAY TANK	AMB	1.0	99.9	0.1	1981	5.25	0.73	6.8	4875	92.3	45.3
	TOTAL SYSTEM					11885	5.50	0.75	7.0	12000	96.3	46.0

NOTES:

1. CONCENTRATIONS IN μC PER CC OF GAS AT ATMOSPHERIC PRESSURE AND 140°F.
2. CONCENTRATIONS IN μC PER CC LIQUID AT ROOM TEMPERATURE.
3. KR - 85 CONCENTRATIONS ARE MAX VALUES BUT DO NOT OCCUR SIMULTANEOUSLY WITH OTHER ISOTOPE MAX CONCENTRATIONS.
4. INCLUDES TWO SHUTDOWN TANKS

TABLE 11.3-4

GASEOUS WASTE PROCESSING SYSTEM COMPONENT DATA[a]

Waste gas compressors	
Type	Centrifugal
Quantity	2
Design pressure (psig)	150
Design temperature (°F)	180
Operating temperature (°F)	70-140
Operating suction pressure	0.5
Design discharge pressure (psig)	100
Design flow, N_2 at 140°F (sf^3/min)	40
Gas decay tanks	
Type	Vertical, horizontal
Quantity	8
Design pressure (psig)	150
Design temperature (°F)	180
Volume, each (ft^3)	600
Material of construction	Carbon steel
Recombiners	
Type	Catalytic
Quantity	2
Design pressure (psig)	150
Design temperature (°F)	(b)
Design flowrate (sf^3/min)	50
Operating hydrogen recombiner rate (sf^3/min)	0.7
Material of construction	Stainless steel

a. The above components are designed and manufactured to the requirements given in Regulatory Guide 1.143, Revision 1, with the exception of the seismic design criteria given in Regulatory Position C.5. The components, systems, and structures are designed to the seismic design criteria given in section 3.7.

b. Varies by component, but exceeds component operating temperature by 100°F.

TABLE 11.3-5 (SHEET 1 OF 2)

GASEOUS WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
<u>Flow Instrumentation</u>							
QIA-*1091	Gas decay tank water flush	150	180	0-6000 gal	Adjustable 3000-6000 gal		Local
<u>Pressure Instrumentation</u>							
PIA-1036	Gas decay tank 1	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1037	Gas decay tank 2	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1038	Gas decay tank 3	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1039	Gas decay tank 4	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1052	Gas decay tank 5	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1053	Gas decay tank 6	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1054	Gas decay tank 7	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel

*Unit 2 only.

TABLE 11.3-5 (SHEET 2 OF 2)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PIA-1055	Gas decay tank 8	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1065	Hydrogen supply header	150	180	0-150 psig	90 psig		WPS panel
PIA-1066	Nitrogen supply header	150	180	0-150 psig	90 psig		WPS panel
PICA-1092	Compressor suction header	150	180	2 psi vacuum 2 psig	0.5 psi	0.5 psi	WPS panel
PI-1094	Volume control tank discharge pressure	150	250	0-20 psig			Local
PI-1047	Gas decay tank inlet nitrogen pressure						
Level Instrumentation							
LICA-1030	Compressor moisture separator	150	180	0-30 in. water	15 in. water	15, -10, 8, -5, -1 in. water	WPS panel and local
LICA-1032	Compressor moisture separator	150	180	0-30 in. water	15 in. water	15, -10, 8, -5, -1 in. water	WPS panel and local

a. The following abbreviations are used:

F	flow	T	temperature	I	indication
Q	water integrator	L	level	C	control
P	pressure	R	radiation	A	alarm

TABLE 11.3-6

RADIOACTIVE NOBLE GAS RELEASES FROM FNP

Year	Total Release (Ci)	Annual Boundary Dose (mR/year)	Annual Tech Spec Fraction %
		UNIT 1	
1977	1.00×10^{3}	7.85×10^{-3}	0.08
1978	3.53×10^{3}	9.89×10^{-2}	0.99
1979	3.18×10^{3}	5.08×10^{-2}	0.51
1980	1.96×10^{4}	2.54	25.40
1981	2.21×10^{2}	1.26×10^{-1}	1.26
1982	3.81×10^{4}	1.13×10^{-2}	0.11
1983	2.20×10^{4}	5.00×10^{-2}	5.00
		UNIT 2	
1977	-	-	-
1978	-	-	-
1979	-	-	-
1980	-	-	-
1981	2.60	2.03×10^{-3}	0.02
1982	3.54×10^{3}	4.24×10^{-1}	4.24
1983	8.47×10^{2}	1.18×10^{-1}	1.18

TABLE 11.3-7

ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER 40 YEARS OPERATION[a] [b]

	Activity (Ci) Following Plant Shutdown		
Isotope	Zero Decay	30 Days	50 Days
Kr-85	11,890	11,820	11,780
Kr-85m	5.5	~0	~0
Kr-87	0.75	~0	~0
Kr-88	7.0	~0	~0
I-131	0.1656	0.0126	0.00226
I-132	0.000684	~0	~0
I-133	0.0259	~0	~0
I-134	0.000144	~0	~0
I-135	0.00504	~0	~0
Xe-131m	108	19	5.8
Xe-133	12,000	232	17
Xe-133m	96	0.01	~0
Xe-135	45	~0	~0

a. The table is based on 40 years continuous operation with 0.25% fuel defect.

b. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 ears. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

TABLE 11.3-8

REDUCTION IN REACTOR COOLANT SYSTEM GASEOUS FISSION PRODUCTS RESULTING FROM NORMAL OPERATION OF THE GASEOUS WASTE PROCESSING SYSTEM[(a)]

Reactor Coolant Gaseous Fission Product Activities (µCi/g)		
Isotope	GWPS Operating	GWPS Not Operating
Kr-85m	2.1	2.2
Kr-85	0.14	5.5
Kr-87	1.3	1.3
Kr-88	3.6	3.8
Xe-131m	0.21	1.8
Xe-133m	1.5	3.2
Xe-133	79.8	290
Xe-135	5.7	6.1

a. Based on operation with cladding defects in fuel generating 1 percent of the rated core thermal power; power - 2774 MWt; purification letdown rate - 60 gal/min; purge rate -0.7 sf^3/min.

TABLE 11.3-9 (SHEET 1 OF 2)

EXPECTED ANNUAL AVERAGE RELEASE OF AIRBORNE RADIONUCLIDES[(a)(b)]

GASEOUS RELEASE RATE (CURIES PER YEAR)[(c)]

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)	Gas Stripping Shutdown	Gas Stripping Continuous	Building Ventilation Reactor	Building Ventilation Auxiliary	Building Ventilation Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
KR-83m	1.973E-02	6.646E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
KR-85m	8.438E-02	2.900E-08	0.0E+00	0.0E+00	2.0E+00	4.0E+00	0.0E+00	0.0E+00	2.0E+00	8.0E+00
KR-85	2.064E-03	7.048E-10	5.5E+01	5.6E+02	2.4E+02	3.0E+00	0.0E+00	0.0E+00	1.0E+00	8.6E+02
KR-87	5.952E-02	1.936E-08	0.0E+00	0.0E+00	0.0E+00	4.0E+00	0.0E+00	0.0E+00	2.0E+00	6.0E+00
KR-88	1.735E-01	5.818E-08	0.0E+00	0.0E+00	2.0E+00	7.0E+00	0.0E+00	0.0E+00	3.0E+00	1.2E+00
KR-89	5.600E-03	1.912E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
XE-131M	5.366E-03	1.844E-09	1.0E+01	6.2E+02	3.8E+02	1.5E+01	0.0E+00	0.0E+00	7.0E+00	1.0E+03
XE-133M	3.859E-02	1.326E-08	0.0E+00	0.0E+00	8.0E+00	2.0E+00	0.0E+00	0.0E+00	0.0E+00	1.0E+01
XE-133	1.606E+00	5.441E-07	1.0E+01	5.1E+02	6.7E+02	5.8E+01	0.0E+00	0.0E+00	2.7E+01	1.3E+03
XE-135M	1.424E-02	4.810E-09	0.0E+00	0.0E+00	0.0E+00	3.0E+00	0.0E+00	0.0E+00	1.0E+00	4.0E+00
XE-135	1.997E-01	6.749E-08	0.0E+00	0.0E+00	1.7E+01	2.0E+01	0.0E+00	0.0E+00	1.0E+01	4.7E+01
XE-137	1.007E-02	3.410E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
XE-138	4.810E-02	1.600E-08	0.0E+00	0.0E+00	0.0E+00	3.0E+00	0.0E+00	0.0E+00	1.0E+00	4.0E+00
TOTAL NOBLE GASES										3.3E+03

IODINE RELEASE RATE - CURIES PER YEAR

Nuclide	PrimaryCoolant (µCi/gm)	Secondary Coolant (µCi/gm)	Building Ventilation Fuel Handling	Building Ventilation Reactor	Building Ventilation Auxiliary	Building Ventilation Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
I-131	2.739E-01	3.079E-05	6.0E-04	1.4E-03	1.4E-02	1.7E-03	1.8E-01	2.7E-03	2.0E-01
I-133	3.979E-01	3. 491E-05	1.9E-03	4.6E-03	4.6E-02	1.9E-03	2.1E-01	4.0E-03	2.7E-01

TRITIUM GASEOUS RELEASE 560 CURIES/YR

TABLE 11.3-9 (SHEET 2 OF 2)

AIRBORNE PARTICULATE RELEASE RATE - CURIES PER YEAR

Nuclide	Waste Gas System	Building Ventilation: Fuel Handling	Building Ventilation: Reactor	Building Ventilation: Auxiliary	Total
CR-51	1.4E-07	1.8E-06	3.2E-07	3.2E-06	5.5E-06
MN-54	4.5E-05	(e)	9.2E-07	1.8E-04	2.3E-04
FE-59	1.5E-05	(e)	3.2E-07	6.0E-05	7.5E-05
CO-57	0.0E+00	0.0E+00	2.9E-08	0.0E+00	2.9E-08
CO-58	1.5E-04	2.1E-04	3.2E-06	6.0E-04	9.6E-04
CO-60	7.0E-05	(e)	1.4E-06	2.7E-04	3.4E-04
SR-89	3.3E-06	2.1E-05	4.6E-07	1.3E-05	3.8E-05
SR-90	6.0E-07	8.0E-06	1.8E-07	2.9E-06	1.2E-05
ZR-95	4.8E-08	3.6E-08	0.0E+00	1.0E-05	1.0E-05
NB-95	3.7E-08	2.4E-05	6.3E-08	3.0E-07	2.4E-05
RU-103	3.2E-08	3.8E-07	5.6E-08	2.3E-07	7.0E-07
RU-106	2.7E-08	6.9E-07	0.0E+00	6.0E-08	7.8E-07
SB-125	0.0E-05	5.7E-07	0.0E+00	3.9E-08	6.1E-07
CS-134	4.5E-05	1.7E-05	9.2E-07	1.8E-04	2.4E-04
CS-136	5.3E-08	0.0E+00	1.1E-07	4.8E-07	6.4E-07
CS-137	7.5E-05	2.7E-05	1.6E-06	3.0E-04	4.0E-04
BA-140	2.3E-07	0.0E+00	0.0E+00	4.0E-06	4.2E-06
CE-141	2.2E-08	4.4E-09	4.6E-08	2.6E-07	3.3E-07

(a) For one unit.

(b) Twenty-five Ci/year of argon-41 are released from the containment, and 8 Ci/year of carbon-14 are released from the waste gas processing system (from PWR-GALE code, section 11.1, reference 1). 710 Ci/year of tritium are released via vapor pathways.

(c) The appearance of 0.0 in the table indicates release is less than 1.0 Ci/year for noble gas and 0.0001 Ci/year for I.

(d) Composite highest value GALE data as described in section 11.3.6.

(e) Nuclide amount insignificant; does not contribute to nuclide total value.

TABLE 11.3-10 (SHEET 1 OF 2)

COMPARISON OF CALCULATED MAXIMUM OFFSITE AIRBORNE CONCENTRATION WITH CONCENTRATION VALUES STATED IN COLUMN 1, TABLE II, APPENDIX B TO 10 CFR 20 ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (μCi/ml)	Maximum Permissible Concentration (MPC)[c] (μCi/ml)	Fraction of MPC
Kr-83m	0.00E+00	0.00E+00	5.00E-05	0.00E+00
Kr-85m	8.00E+00	3.17E-12	1.00E-07	3.17E-05
Kr-85	8.60E+02	3.41E-10	7.00E-07	4.87E-04
Kr-87	6.00E+00	2.38E-12	2.00E-08	1.19E-04
Kr-88	1.20E+01	4.76E-12	9.00E-09	5.29E-04
Kr-89	0.00E+00	0.00E+00	N/A	--
Xe-131m	1.00E+03	3.96E-10	2.00E-06	1.98E-04
Xe-133m	1.00E+01	3.96E-12	6.00E-07	6.61E-06
Xe-133	1.30E+03	5.15E-10	5.00E-07	1.03E-03
Xe-135m	4.00E+00	1.59E-12	4.00E-08	3.96E-05
Xe-135	4.70E+01	1.86E-11	7.00E-08	2.66E-04
Xe-137	0.00E+00	0.00E+00	N/A	--
Xe-138	4.00E+00	1.59E-12	2.00E-08	7.93E-05
I-131	2.00E-01	7.93E-14	2.00E-10	3.96E-04
I-133	2.70E-01	1.07E-13	1.00E-09	1.07E-04
C-14	8.00E+00	3.17E-12	3.00E-07	1.06E-05
Ar-41	3.40E+01	1.35E-11	1.00E-08	1.35E-03
H-3	5.60E+02	2.22E-10	1.00E-07	2.22E-03
Cr-51	5.50E-06	2.18E-18	3.00E-08	7.27E-11
Mn-54	2.30E-04	9.12E-17	1.00E-09	9.12E-08
Fe-59	7.50E-05	2.97E-17	5.00E-10	5.95E-08
Co-57	2.80E-08	1.11E-20	4.00E-09	2.77E-12
Co-58	9.60E-04	3.81E-16	2.00E-09	1.90E-07
Co-60	3.40E-04	1.35E-16	2.00E-10	6.74E-07
Sr-89	3.80E-05	1.51E-17	1.00E-09	1.51E-08
Sr-90	1.20E-05	4.76E-18	3.00E-11	1.59E-07
Zr-95	1.00E-05	3.96E-18	5.00E-10	7.93E-09
Nb-95	2.40E-05	9.51E-18	2.00E-09	4.76E-09
Ru-103	7.00E-07	2.77E-19	2.00E-09	1.39E-10

TABLE 11.3-10 (SHEET 2 OF 2)

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (μCi/ml)	Maximum Permissible Concentration (MPC)[c] (μCi/ml)	Fraction of MPC
Ru-106	7.80E-07	3.09E-19	8.00E-11	3.86E-09
Sb-125	6.10E-07	2.42E-19	3.00E-09	8.06E-11
Cs-134	2.40E-04	9.51E-17	2.00E-10	4.76E-07
Cs-136	6.40E-07	2.54E-19	9.00E-10	2.82E-10
Cs-137	4.00E-04	1.59E-16	2.00E-10	7.93E-07
Ba-140	4.20E-06	1.66E-18	2.00E-09	8.32E-10
Ce-141	3.30E-07	1.31E-19	1.00E-09	1.31E-10
TOTAL				6.82E-03

a. Total Ci/year from table 11.3-9.

b. Based on the sum of contributions from the plant vent and turbine building vent using a ground release dilution factor (X/Q) of 1.0 x 10^{-5} s/m^3.

c. From column 1, Table II, Appendix B to 10 CFR 20.

d. 0.0 indicates release <1.0 Ci/yr for noble gas.

TABLE 11.3-11 (SHEET 1 OF 2)

ESTIMATED ANNUAL DOSES TO A MAXIMUM EXPOSED INDIVIDUAL FROM GASEOUS AND PARTICULATE EFFLUENTS (MREM/YR)[a][b]

Pathway	Group	Body	Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume (.9 mi WSW)	All	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.22E-02	1.04E-01
Ground	Adult	9.49E-02	9.21E-02	1.90E-01	9.67E-02	9.85E-02	1.55E+00	9.02E-02	9.10E-02
Shine + Inhal +Veg.	Teen	1.21E-01	1.18E-01	3.01E-01	1.25E-01	1.25E-01	1.36E+00	1.17E-01	1.18E-01
(0.9 Mile WSW)	Child	2.21E-01	2.17E-01	7.13E-01	2.27E-01	2.26E-01	2.07E+00	2.16E-01	2.16E-01
	Infant	1.51E-02	1.50E-02	6.37E-03	1.53E-02	1.53E-02	1.01E-01	1.50E-02	1.61E-02
Cow or Goat	Adult								
	Teen								
	Child				None within 5 miles				
	Infant								
Meat	Adult	8.36E-03	8.30E-03	3.02E-02	8.55E-03	8.69E-03	1.29E-01	8.02E-03	8.00E-03
(1.1 Miles WSW)	Teen	6.49E-03	6.42E-03	2.55E-02	6.70E-03	6.81E-03	9.38E-02	6.27E-03	6.25E-03
	Child	1.12E-02	1.10E-02	4.79E-02	1.15E-02	1.17E-02	1.43E-01	1.10E-02	1.09E-02
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Totals[c]	Adult	1.03E-01	1.00E-01	2.20E-01	1.05E-01	1.07E-01	1.68E+00	9.82E-02	9.90E-02
(excluding plume)	Teen	1.28E-01	1.25E-01	3.27E-01	1.32E-01	1.32E-01	1.45E+00	1.23E-01	1.24E-01
	Child	2.32E-01	2.28E-01	7.61E-01	2.38E-01	2.38E-01	2.21E+00	2.27E-01	2.27E-01
	Infant	1.51E-02	1.50E-02	6.37E-03	1.53E-02	1.53E-02	1.01E-01	1.50E-02	1.61E-02

TABLE 11.3-11 (SHEET 2 OF 2)

a. Highest offsite annual Beta air dose = 0.13 mrad[e]
Highest offsite annual Gamma air dose = 0.05 mrad[e]

b. All data are on a per unit basis.

c. Evaluated at a location where an exposure pathway and dose receptor actually exist at the time of licensing.

Appendix I design objectives - radioiodines and particulates:
Dose to any organ from all pathways - 15 mrem/year per unit

Annex to Appendix I Docket RM-50-2 design objectives:

Dose to any organ from all pathways - 15 mrem/year per site
I-131 releases - 1 Ci/year per unit (reference table 11.3-9)

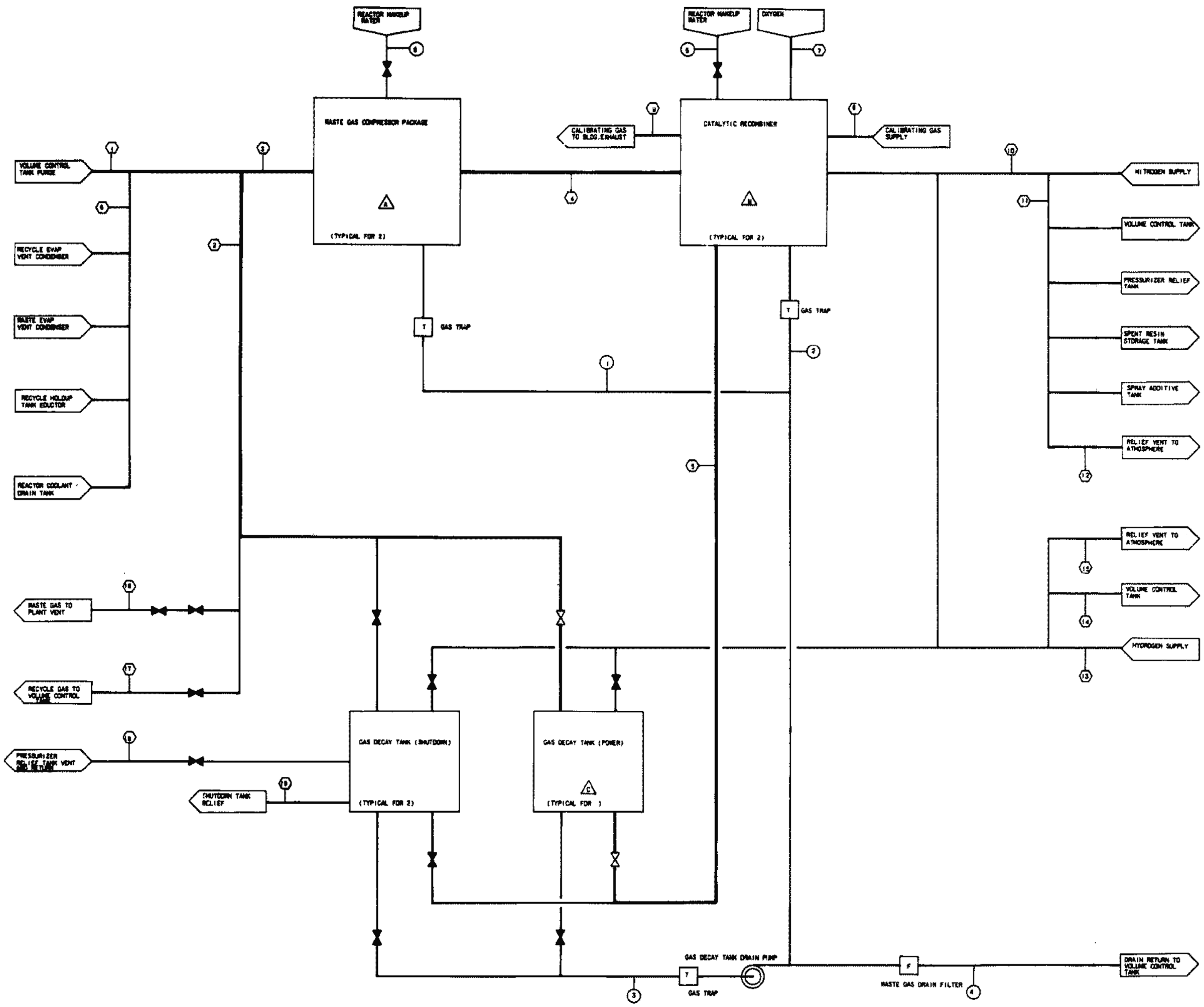
d. Evaluated at a location that could be occupied during the term of plant operation.

Appendix I design objectives - gaseous effluents (noble gases only):

Gamma dose in air - 10 mrad/year per unit
Beta dose in air - 20 mrad/year per unit
Dose to total body of individual - 5 mrem/year per unit
Dose to skin of individual - 15 mrem/year per unit

Annex to Appendix I, Docket RM-50-2. Design objectives are the same as Appendix I except on a per-site basis; therefore, calculated doses should be multiplied by 2.

e. Provided for information only; a receptor is assumed present at the location of a potential pathway.
This evaluation is based on the worst case X/Q at the site boundary.



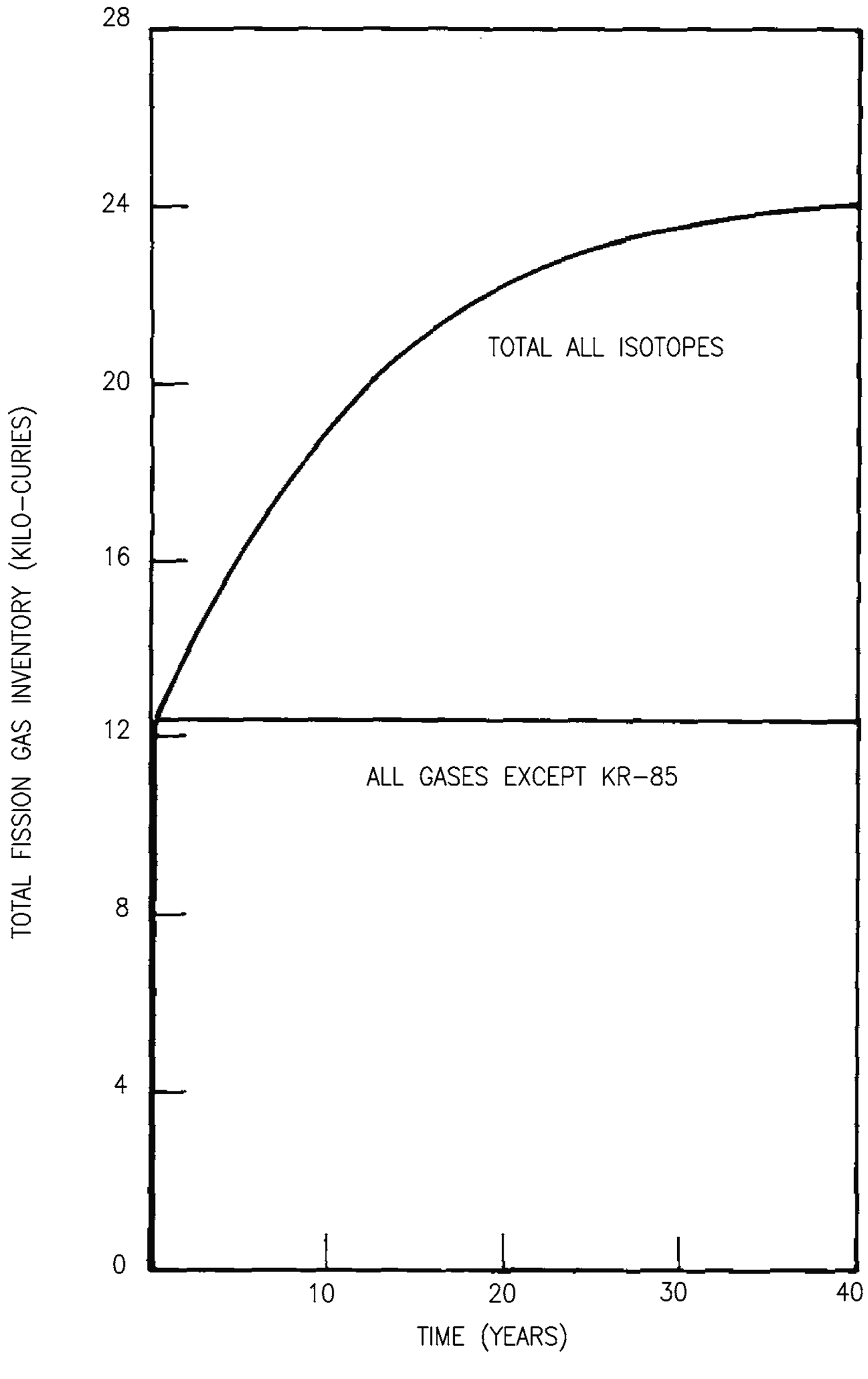
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAMS FOR GASEOUS WASTE PROCESSING SYSTEM

FIGURE 11.3-1



(BASED ON CONTAINMENT CORE OPERATION WITH 0.25% FUEL DEFECTS)

REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ESTIMATED WASTE GAS PROCESSING SYSTEM FISSION GAS ACCUMULATION FIGURE 11.3-2

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 DESIGN OBJECTIVES

The radiation monitoring system consists of the following subsystems:

A. The process and effluent radiological monitoring system (PERMS), which includes both continuous process and periodic sampling systems.

B. The area radiation monitoring system, which monitors radiation fields in various areas within the plant. This system is further described in subsection 12.1.4.

C. The airborne radioactivity monitoring system, which is described in subsection 12.2.4.

The PERMS is designed to enable plant operation to be in compliance with Table II, Appendix B to 10 CFR 20.1 - 20.601, the low as reasonably achievable criterion of 10 CFR 50, and the Technical Specification limits of the Farley Nuclear Plant, in addition to being in accordance with the NRC acceptance criteria contained in 1971 General Design Criteria 60, 63, and 64 as documented in section 11.5 of NUREG 75/034, dated May 2, 1975.

The radiation monitoring system does not meet the guidelines of NRC Regulatory Guide 1.21 in its entirety. Specifically, continuous isotopic analysis and measurement of radionuclides to exceedingly low sensitivities and monitoring of all potential paths for radioactive release are not within the current state of the art and are therefore not addressed in the design of this system.

The general design objectives for the PERMS are to:

A. Warn of any radiation health hazard to operating personnel.

B. Warn of leakage from process systems containing radioactivity.

C. Monitor activity released in effluents and provide alarm and termination of the release when radiation levels exceed setpoint limits. Where the termination of release is not feasible, the monitors provide a continuous indication of the magnitude of activity released.

The accomplishment of these general objectives by the PERMS will provide assurance that exposures to individuals in restricted and unrestricted areas are as low as reasonably achievable during all modes of plant operation and during accidents.

Except for the containment purge exhaust line monitors, the spent fuel pool exhaust flow gas monitors, and the containment air particulate and noble gas monitor, the PERMS is not designed to Seismic Category I or to Institute of Electrical and Electronics Engineers (IEEE) accident grade standards and, therefore, no credit is taken for these monitors in the accident evaluations in chapter 15. The containment purge exhaust line monitors and the spent fuel pool exhaust flow gas monitors are designed to Seismic Category I and to IEEE accident grade standards. The accident evaluations in chapter 15 take credit for the containment purge exhaust line monitors to mitigate the consequences of a fuel handling accident inside the containment by isolating the containment purge and mini-purge lines and for the spent fuel pool exhaust flow gas monitors to mitigate the consequences of a fuel handling accident in the spent fuel pool area by realigning the spent fuel pool ventilation exhaust to the penetration room filtration (PRF) unit. The containment air particulate and noble gas monitor is part of the nonsafety-related reactor coolant pressure boundary leakage detection system and is procured under a quality assurance program consistent with Seismic Category I and IEEE Class 1E requirements. The containment air particulate and noble gas monitor does not perform a safeguard function and is not required to operate during or after a seismic event.

The design objectives for PERMS periodic sampling include the following:

A. Enable manual collection of representative samples of planned gaseous and liquid effluents prior to discharge to unrestricted areas during normal reactor operation and during anticipated operational occurrences in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present in these discharges as required by 10 CFR 50.36a.

B. Enable manual collection of representative samples of gaseous and liquid process streams during normal reactor operation and during anticipated operational occurrences in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present to verify and supplement the continuous process system monitors.

Written procedures shall be established, implemented, and maintained covering the quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, dated February 1979.

11.4.2 PROCESS AND EFFLUENT RADIATION MONITORING

11.4.2.1 General Description

The components of the PERMS are designed for the following environmental conditions:

A. Temperature - An ambient temperature range of 40°F to 120°F.

B. Humidity - Relative humidity of 0 to 95 percent.

C. Pressure - Components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand containment test pressure.

D. Radiation - Process and effluent radiation monitors are of a nonsaturating design so that they will register full scale if exposed to radiation levels up to 100 times full-scale indication.

E. Radiation monitoring equipment is designed and located in such a way that radiation damage to electrical insulation and other materials will not affect its usefulness over the life of the plant. The only components of this system that are located in the containment are the detectors and associated local alarm and indication equipment for certain area monitoring channels. They are not expected to operate following a major loss-of-coolant accident (LOCA) and are not designed for this purpose.

Figure 11.4-1 contains the overall function block diagram for the radiation monitoring system.

All continuous process radiation monitors are indicated in the control room. They provide instrument malfunction and high activity annunciation in the main control room.

High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to completely remove the various chassis from the cabinet after disconnecting the cable connectors from the rear of these units.

Radiation monitoring system cabinet alarms consist of a red indicator light for high radiation and detector or circuit failure. Except for the R-10, R-11/12, Unit 1 R-29B & C, and Unit 2 R-29B & C monitors, the local meter and alarm assembly at the area monitor detector locations contains a red indicator light and a buzzer type alarm annunciator which are actuated on high radiation. The monitors R-10, R-11/12, Unit 1 R-29B & C, and Unit 2 R-29B & C have a local display which contains a red indicator light which is actuated on high radiation. [See figures 11.4-2, 11.4-3, 11.4-4, 11.4-5, and 11.4-6 and drawings U-167650, U-167651, U-167652, B-507156 (Unit 1 only), B-507157 (Unit 1 only), B-356750 (Unit 2 only), and B-356751 (Unit 2 only)]. An indicating light is provided on each drawer and a common annunciator is provided on the control board to indicate a channel placed in the test mode.

Radiation levels are recorded by a data acquisition system computer which can display data, on demand, to the operator.

Table 11.4-1 indicates the detector medium and temperature conditions. The different operating temperature ranges are within the design limits of the system.

Sensitivity of a radiation monitor is defined as the minimum signal level which is discernible above environmental noise (background). The sensitivity of all process and effluent monitors is designed to be two times the environmental signal level.

Each channel monitors gross concentrations, and detector output is measured in counts per minute (cpm), microCuries per cubic centimeter (μCi/cc), milliRad per hour (mRad/h), or roentgens per hour (R/h). Each channel has a minimum range of three decades.

The anticipated concentrations of radionuclides in the various process and effluent streams will be normal background radiation. The sensitivity of the detectors monitoring these streams will ensure that abnormal conditions will be detected before they cause an undue hazard to the operators or the general public.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the section describing those systems. Periodic tests will ensure that the channels operate properly.

11.4.2.2 System Description

This system consists of multiple channels which monitor radiation levels in various plant operating systems. The output from each channel detector is transmitted to the radiation monitoring system cabinets located in the control room where the radiation level is indicated by a meter or dot-matrix display and recorded. High radiation level alarms are indicated on the radiation monitoring system cabinets with annunciation at the control board in the control room.

A main control board annunciator provides a single window which alarms for any channel (process or area) detecting high radiation. A second common main control board annunciator is actuated for any channel failure. A third common main control board annunciatior is actuated when any channel is placed in test mode. Verification of which channel has alarmed is made at the radiation monitoring system cabinets. (See figure 11.4-1.) Individual annunciators are located near the base of the radiation monitoring system cabinets.

A tabulation of the process radiation monitoring channels is found in table 11.4-2. The minimum sensitivity listed in the table is based on a Co-60 background level of at least 2 mR/h.

Each channel contains a completely integrated modular assembly, which includes the following:

A. Log Level Amplifier or Microprocessor Controller/Display

With the exception of R-10, R-11, R-12, Unit 1 R-29B & C, and Unit 2 R-29B & C, the log level amplifier accepts detector pulses, performs a log integration (converts total pulse rate to a logarithmic analog signal), and amplifies the resulting output for suitable indication and recording. For R-10, R-11, R-12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C) detector output is processed by the microprocessor and transmits the resulting output for indication and recording.

B. Power Supplies

Furnishes the positive and negative voltages for the circuits, relays, and alarm lights and provides the high voltage for the detector.

C. Test Calibration Circuitry

A precalibrated pulse signal to test channel electronics and a solenoid-operated radiation check source to verify channel operation are provided. A common annunciator on the main control board indicates when a channel is in the test mode as shown on table 11.4-2. Channels R-15 (B and C), R-66 (A through F), R-10, R-11, Unit 1 monitors R-29B & C, Unit 2 monitors R-29B & C, and Channel R-60 (A through C) have a solenoid-operated radiation checksource to verify channel operation and an alarm light that indicates when an abnormal detector signal or power conditions exist.

D. Radiation Level Meter, LCD, or Dot-Matrix Display

Provides level indication calibrated in either cpm, µCi/cc, mRad/h, or R/hr for the minimum range shown on table 11.4-2. The level signal is also recorded.

E. Indicating Lights

Indicate high radiation alarms, circuit failures, and for the R-10, R-11/12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C) radiation monitoring system, “OPER” fault.

F. Bistable Circuits

Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level within the range of the instruments) and one to alarm on loss of signal (circuit failure or system “OPER” fault).

G. Checksource

A remotely operated, long half-life radiation checksource is furnished in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause approximately 30 percent of full scale indication. For R-10, R-11/12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C), the checksource activity is compared with a stored value in the local skid-mounted microprocessor and must exceed 85 percent of that value to pass.

11.4.2.2.1 Penetration Room Air Particulate Monitor - Channel R-10

The penetration room air particulate monitor (figure 11.4-2) detects air particulate beta radioactivity in the penetration room ventilation system discharge ducting. This monitor is functionally similar to the containment air particulate monitor R-11.

11.4.2.2.2 Containment Air Particulate Monitor - Channel R-11

This monitor (figure 11.4-6) is provided to continuously monitor air particulate beta radioactivity in the containment. This channel takes a continuous air sample from the containment atmosphere. The sample is drawn outside the containment in a closed system monitored by a scintillation counter fixed filter paper detector assembly. The filter paper collects 99 percent of all particulate matter > 1 μm in size and is viewed by a photomultiplier beta scintillation combination. The air sample is returned to the containment after it passes through the series connected (channel R-12) gas monitor.

The detector assembly is in a completely enclosed housing. The detector is a photomultplier tube beta scintillation combination and the output is transmitted through the microprocessor to the radiation monitoring system control room cabinets.

Lead shielding is provided to reduce the effect of background radiation to where it does not interfere with the detector's sensitivity. A fixed filter paper system is part of the detector unit.

11.4.2.2.3 Containment Radioactive Gas Monitor - Channel R-12

This monitor (figure 11.4-6) is provided to continuously measure gaseous beta-gamma radioactivity in the containment.

This channel takes the continuous air sample from the containment atmosphere after it passes through the air particulate monitor and draws the sample through a closed system to the gas monitor assembly. The sample is monitored by a beta detector located in a fixed, shielded volume. The sample is then returned to the containment.

The detector assembly is in a completely enclosed housing containing a beta-gamma sensitive detector mounted in a constant gas volume container. Lead shielding is provided to reduce the effect of background radiation to a point where it does not interfere with the detector's sensitivity.

The signal is processed by a microprocessor. The output signal is transmitted to the radiation monitoring system cabinets in the control room.

The containment air particulate and radioactive gas monitors (channels R-11 and R-12) have assemblies that are common to both channels. They are described as follows:

A. The flow control assembly includes a pump unit and selector valves that provide containment sample or a clean sample to the detectors.

B. The pump unit consists of:

1. A pump to obtain the air sample.

2. A digital mass flow indicator/controller to adjust and indicate the flowrate.

3. A flow control valve to provide steady flow.

4. High and low flow alarms provided on the radiation monitoring system rack.

C. Selector valves are used to select the sample for monitoring and to block flow to and from the sampling area when the channel is in maintenance or "purging" condition.

D. A pressure sensor is used to protect the system from high pressure. This unit automatically closes the inlet and outlet valves upon a high pressure condition.

E. Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector "purged" with a clean sample.

F. The flow control panel in the control room radiation monitoring system racks permits operation of the flow control assembly. By operating a sample selector switch on the control panel, either the containment or "purge" sample may be monitored. Indicator lights are actuated by the following:

- Flow alarm assembly (low or high flow).

- The pressure sensor assembly (high pressure).

- The pump power control switch (pump motor on).

The containment air particulate and noble gas monitor (channels R-11 and R-12) is part of the nonsafety-related reactor coolant pressure boundary leakage detection system. These monitors provide continuous monitoring of the containment atmosphere to comply with the requirements of GDC 30 and GDC 64 for normal plant operation.

In the appendix 3A conformance statement for Regulatory Guide (RG) 1.45, it is noted that the reactor coolant pressure boundary leakage detection system does not perform a safeguard function and, contrary to the guidance of RG 1.45 for airborne particulate equipment, the system is not required to operate during or after a seismic event. The NRC accepted this position as noted in the original SER (NUREG-75/034, May 2, 1975, pages 5-6 and 5-7).

The R-11 and R-12 monitors do not serve a safety-related function and are not required to function during or after a seismic event. The monitors are Q-listed only for the purpose of documenting repairs and modifications. As permitted by FSAR subsection 17.3.5, the replacements can be purchased to meet the quality requirements of the original specification. These quality requirements are the appropriate quality assurance provisions for the purpose of meeting FSAR subsection 17.3.1 requirements for replacements.

Replacement of the existing Unit 1 R-11 and R-12 monitors with equipment procured under a quality assurance program consistent with a safety classification of safety-related and to Seismic Category I safe shutdown earthquake (SSE) requirements is advantageous for possible future changes to accident analysis requirements. Power to the skid-mounted equipment will be from Class 1E MCCs. The control panel in the control room will retain its current classification of nonsafety-related, Seismic II/I. An additional advantage would be to provide a higher assurance for the closure function of the monitor's isolation valves after an SSE, allowing the associated containment isolation valves to be opened for operation of post-accident grab sampler R-67.

Post-accident grab sampler R-67 taps off upstream of the R-11/12 monitor inlet and returns to the system downstream of the R-11/12 monitor outlet. The purpose of this grab sample point is to provide a means for sampling the atmosphere in containment post-accident for particulates, iodines, and gases with a minimum of exposure. This sample point consists of:

A. Remote-operated valves on the inlet line and the outlet line to the grab sample point.

B. Filter holder with quick disconnects (holds filter disk and silver zeolite cartridge).

C. Gaseous collection vessel.

D. Vacuum pump with throttle valve.

E. Flow indication.

F. Control panel to allow remote operation of the valves and vacuum pump.

As documented in NRC SERs[(1)(2)(3)], post-accident sampling of containment atmosphere either conforms to NRC acceptance criteria contained in NUREG-0578, NUREG-0737, and Regulatory Guide 1.97 or deviations have been justified.
As documented in NRC SER[(4)], dated May 22, 2002, issuance of amendments 156 and 148 to the Units 1 and 2, respectively, Facility Operating Licenses supersedes the post-accident sampling system (PASS) specific requirements imposed by post-TMI confirmatory orders. However, the capability to obtain grab samples of the containment atmosphere will be maintained.

11.4.2.2.4 Waste Gas Processing Monitor - Channel R-13

The input line to the waste gas system compressor (drawing U-167652) is monitored for gaseous activity by a Geiger-Mueller tube to ensure the Technical Specification limit for waste gas decay tank storage is not exceeded and to provide the capability to establish the radioactive gas inventory in the waste gas processing system. Remote indication and annunciation are provided on the waste processing system control panel. The alarm setpoint will be based upon the waste gas decay tank rupture accident such that the resulting dose at the site boundary will be limited to 0.5 rem.

11.4.2.2.5 Deleted

11.4.2.2.6 Condenser Air Ejector Gas Monitor - Channel R-I5

This channel (drawing U-167650) monitors the discharge from the air ejector exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine building vent. A beta-gamma sensitive Geiger-Mueller tube is used to monitor the gaseous radioactivity level. The detector is inserted into an inline fixed-volume container which includes adequate lead shielding to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

11.4.2.2.7 Intermediate Range Condenser Air Ejector Monitor - Channel R-15B

This channel utilizes a beta-gamma sensitive Geiger-Mueller tube to monitor the gaseous radioactivity level in the air ejector exhaust. The monitor is set to alarm at one-half decade below the upper scale limit of R-15, thus providing an indication of the severity of a primary to secondary system leak that may be out of the measurement range of R-15.

11.4.2.2.8 High Range Condenser Air Ejector Monitor - Channel R-15C

This channel utilizes an ion chamber to monitor the gaseous radioactivity level in the air ejector exhaust. The monitor is set to alarm at one-half decade below the upper scale limit of R-15B, thus providing an indication of the severity of a primary to secondary system leak that may be out of the measurement range of R-15 or R-15B.

11.4.2.2.9 Deleted

11.4.2.2.10 Component Cooling Liquid Monitor - Channel R-17A and B

This channel (drawing U-167651) continuously monitors the component cooling system for radiation indicative of a leak of reactor coolant from the reactor coolant system and/or the residual heat removal system.

Scintillation counters are located in inline wells. A high radiation level alarm signal initiates closure of the valve located in the component cooling surge tank vent line to prevent gaseous radiation release.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

11.4.2.2.11 Waste Processing System Liquid Effluent Monitor - Channel R-18

This channel (drawing U-167651) continuously monitors all waste processing system liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. A scintillation counter in an inline sampler assembly monitors these effluent discharges. Remote indication and annunciation are provided on the waste processing system control board.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

The monitor is located in the discharge line, prior to dilution, to provide superior measurement of radioactivity. Should the instrument fail during a release or should the activity exceed the instrument setpoint, the discharge valves will close and stop the release.

11.4.2.2.12 Steam Generator Liquid Sample Monitor - Channel R-19

This channel (figure 11.4-3) monitors the liquid phase of the secondary side of the steam generator for radioactivity that would indicate a primary to secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from the bottom or surface of each of the steam generators are combined in a common header and the resulting common sample is continuously monitored by a scintillation counter in an inline sampler assembly.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

High activity alarm indications are displayed at the radiation monitoring system cabinets and at the monitor location, with annunciation at the main control board.

In the event of a high activity alarm, isolation valves in the sample lines would close. Subsequent identification of the steam generator that is leaking would be made by manual override of sample line isolation and by drawing samples from each steam generator for analysis.

11.4.2.2.13 Service Water Liquid Monitor - Channel R-20A and B

A scintillation counter is located in an offline sampler assembly.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

Radiation monitors are provided in the discharge line from the containment air coolers, which are the main source of radioactivity discharged via the service water system to the environment (figure 11.4-6). Sensitivity of these instruments is given in table 11.4-2 (channel R-20). The only time there could be radioactive leakage into the service water system through the containment coolers is for a very short period of time when the containment pressure is higher than the service water pressure following a LOCA.

11.4.2.2.14 Deleted

11.4.2.2.15 Deleted

11.4.2.2.16 Steam Generator Blowdown Processing System Monitors – Channel R-23A and B

Steam generator blowdown process radiation monitor channel R-23A (figure 11.4-3) is provided to continuously monitor the liquid activity level of the blowdown fluid entering the surge tank. This full flow, inline monitor detects large fluctuations in activity concentration due to variations in steam generator inleakage conditions or to radioactive breakthrough of the system demineralizer train. A high signal from this instrument sounds an alarm and stops blowdown by closing the system's process controlled isolation valve.

Steam generator blowdown discharge radiation monitor channel R-23B (figure 11.4-3) is provided to continuously monitor the liquid activity level of the discharged or recycled fluid. This full flow, inline monitor provides the final radioactive control of the system's effluent. A high signal from this instrument sounds an alarm and closes the discharge valve if it should be open. Being downstream of channel R-23A, this instrument provides redundant protection against the discharge of excessive amounts of activity.

11.4.2.2.17 Containment Exhaust Flow Gas Monitors – Channel R-24A and B

A. Introduction

The containment exhaust flow gas monitors (figure 11.4-4) act to limit the radioactive releases associated with a fuel handling accident in the containment during purge operations. Design requirements were derived by analyses of the radioactive releases associated with the fuel handling accident discussed in chapter 15.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the containment exhaust flow gas monitors:

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic 11.4-14 Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The degree of conformance is given in paragraph 7.1.2.7.

I. Conformance to Regulatory Guide 1.22

The containment purge isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

As shown in figure 11.4-4, when the gaseous activity in the containment purge exhaust line reaches the high level setpoint, automatic isolation of the containment purge and exhaust lines is initiated. Following the isolation of the purge system, radiation monitoring of subsequent releases into the containment is provided by the nonprotection grade air particulate and gas monitors (channels R-11 and R-12).

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in subsection 6.2.3.

Q. Supporting Systems

Supporting systems for the redundant containment purge exhaust gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail-safe with respect to all of these supporting systems.

R. Nonsafety Systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity levels of the containment purge exhaust flow during containment purge operations.

A motor-driven positive displacement pump is used to draw a continuous sample from the containment purge exhaust flow line and direct the sample through a particulate removal prefilter. The sample is then routed to a 4π shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is monitored for radioactivity by a thin beta crystal scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the exhaust line to the stack.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs energy in the form of a pulse (current) which is fed directly into a preamplifier at the base of the detector assembly and in turn provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a five-decade log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits.

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the containment at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by Technical Specifications. The programmatic controls contained in the Technical Specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channels R-24A and B is provided from vital motor control centers A and B, respectively.

A "loss of power" and "channel failure" are monitored for each detector providing annunciation in the control room.

A channel performance test is available to the operator. An electronic pulse signal is used to verify the performance of the readout instrumentation.

A radioactive checksource, controlled from the readout instrument in the control room, can be actuated to check system integrity. This checksource is used as a convenient operational and gross calibration check of the detection system.

The checksource is of the same or similar energy and range of the isotopes to be monitored.

A three-way, solenoid-operated valve at the sample chamber inlet is operable from the control room. It is provided to permit air purging of the sample chamber to facilitate background activity checks.

Visual/audible indication of channel failure and/or high radiation is provided in the control room.

11.4.2.2.18 Spent Fuel Pool Exhaust Flow Gas Monitors - Channel R-25A and B

A. Introduction

The spent fuel pool exhaust flow gas monitors (figure 11.4-5) act to limit the radioactive releases associated with a fuel handling accident in the spent fuel pool. Design requirements were derived by analyses of the radioactive releases associated with the fuel handling accident discussed in chapter 15.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the spent fuel pool gas monitors.

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electronic Equipment During Construction of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The degree of conformance is given in paragraph 7.1.2.7.

I. Conformance to Regulatory Guide 1.22

The isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

As shown in figure 11.4-5, when the gaseous activity in the spent fuel handling building exhaust line reaches the high level setpoint, automatic isolation of the ventilation lines is initiated and the penetration room filtration system automatically starts, taking suction from the spent fuel area.

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in paragraph 9.4.2.2.2.

Q. Supporting Systems

Supporting systems for the gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail safe with respect to all of these supporting systems.

R. Nonsafety systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity releases to the environs by the ventilation fans exhausting the spent fuel pool area of the auxiliary building. The offline monitors incorporate a positive displacement pump that draws a continuous air sample from the spent fuel pool ventilation exhaust duct. This sample is then directed through a particulate removal prefilter. The sample is then routed to a 4π lead-shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is then monitored for radioactivity by a thin beta scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the common vent duct exhausting all spent fuel pool spaces.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs the energy photons and emits the absorbed energy in the form of a pulse (current) fed directly into a preamplifier at the base of the detector assembly and, in turn, provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits. The ratemeters have a five-decade range (10^1 to 10^6 cpm).

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the fuel handling area at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by Technical Specifications. The programmatic controls contained in the Technical Specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channel R-25A and B is provided from vital motor control centers A and B, respectively.

11.4.2.2.19 Deleted

11.4.2.2.20 Noble Gas Effluent Monitors

A1. **Deleted**

A2. Plant Vent Stack Monitor - GGG Monitor Skid Assembly R-29B & Sample Conditioning Skid Assembly R-29D

The GGG monitor skid assembly R-29B and sample conditioning skid assembly R-29D are located in the mechanical MCC room at el 155 ft of the auxiliary building. The GGG monitor system measures the radiation activity in the plant vent stack by drawing a representative sample via the monitor pumps, feeding it to the assembly's detectors, and exhausting it back to the plant vent. The system consists of a low- and mid/high-range gas monitoring with sample/filtering of particulate and iodine.

1. GGG Monitor - Low (R-29E)/Mid (R-29F)/High (R-29G) Noble Gas Channels with Composite Channel for R-29B:

   The GGG vent stack monitoring system has three noble gas detectors with a combined range of 10^{-7} to 10^5 μCi/cc. The GGG monitor has low- and mid/high-range gas detectors with two different sample paths, low (normal) and mid/high (accident). When the plant vent gas activity concentration is near the top of the range of the low-range gas detector, the mid/high-range sample path starts automatically. A fourth channel, the composite channel, represents the entire system combined range to cover the low-, mid-, and high-gas detector ranges and displays the current activity of the selected gas detector.

The low-range channel uses a beta scintillator/PhotoMultiplier tube (PMT) to detect beta radiation. The mid- and high-range channels use solid-state cadmium telluride (chlorine-doped) elements for detecting beta-gamma radiation. The radiation in the gas sample causes an electrical output that is in direct proportion to the level of the beta (low gas) and beta-gamma (mid/high gas) radioactivity present in the sample. The signal from these detectors is processed by a microprocessor. Radiation activity data, indication, and alarms are displayed locally at the skid and remotely in the main control room in the radiation monitoring system cabinets. Annunciation is provided on the main control board annunciation panel.

Remote indication and annunciation are provided on the waste gas processing system control board. A high alarm initiates the automatic closure of the gas decay tank discharge valve.

The assembly has lead shielding to minimize the effects of background radiation. The detectors have the fixed type of background subtract capability.

2. Sample Conditioning Skid Assembly 29D:

The sample conditioning skid (SCS) - The plant vent stack particulate/iodine grab samplers on the R-29D function to filter particulates and iodine from the vent stack sample air prior to entering the GGG gas detector chambers. There are two sets of filters, one set for the low-range (normal) sample path and another set for the mid/high-range (accident) sample path for a total of 4 filters. The two filters in each sample path allow the filters to be changed without stopping the monitors. The two mid/high-range filters are each enclosed in a 2-in. lead shield assembly to protect personnel from radiation exposure during filter removal. Grab sample ports are also provided in the GGG monitor assembly’s outlet sample return line to facilitate collection of required plant vent gaseous samples via a portable gas sample apparatus.

Calibration of all detector channels is by use of external calibration sources and is performed upon installation and at established intervals. The GGG monitor is capable of functioning both during and following an accident.

In addition, checksources are provided to verify channel operation. The following summarizes by channel number and type which checksources are provided:

- Channels R-29E - 0.1 µCi, Cl-36 Beta Checksource.
- Channels R-29F and R-29G - 50 µCi, Cs137 Gamma Checksource.

The plant vent noble gas concentration in $\mu Ci/cm^3$ is determined by sampling and/or by obtaining a value from the monitor. The plant vent flowrate is determined by the number of operating auxiliary building exhaust fans. The release rate in Ci/s is determined by the following equation:

Release rate (Ci/s) = Concentration ($\mu Ci/cm^3$) x flowrate (ft^3/min) x conversion factor

The above method to determine noble gas release rate is described in emergency implementing procedures. During emergencies the release rate is calculated periodically as directed by the emergency director to determine whether the accident classification should be upgraded.

A3. Unit 1 and Unit 2 Plant Vent Stack Monitor - Particulate, Iodine, and Gas (PIG) Monitor Skid Assembly Unit 1 and Unit 2 R-29C

The PIG monitor skid assembly Unit 1 and Unit 2 R-29C is located in the mechanical equipment room at el 175 ft of the auxiliary building. The system consists of particulate, iodine, and noble gas detectors designed to continuously sample the air from the plant vent stack. Monitor Unit 1 and Unit 2 R-29C contains Channels R-29H, R-29I, and R-29J. The PIG measures particulate, iodine, and noble gas radioactive nuclide concentrations in fixed volume samples. It includes two beta detector subassemblies, one for particulate (fixed filter) and one for gas radiation countrate. These channels use a scintillator and PhotoMultiplier Tube (PMT) subassembly to produce an electrical signal proportional to the level of beta radioactivity sensed by the particulate and gas detectors. The third detector subassembly is a gamma detector for iodine radiation countrate. This channel uses a NaI crystal and PMT subassembly to produce an electrical signal proportional to the level of gamma radioactivity sensed by the iodine detector. A lead shield assembly surrounds the three detectors, minimizing the effects of background radiation on activity data. The detectors have the fixed type of background subtract capability.

The signal from these detectors is processed by a microprocessor. Radiation activity data, indication, and alarms are displayed locally at the skid and remotely in the main control room radiation monitoring system cabinets. Annunciation is provided on the main control board annunciation panel.

Grab sample ports are provided in the PIG monitor assembly's outlet sample return line to facilitate collection of required plant vent gaseous samples via a portable gas sample apparatus.

Calibration of all detector channels is by use of external calibration sources and is performed upon installation and at established intervals.

In addition, checksources are provided to verify channel operation. The following summarizes by channel number and type which checksources are provided:

- Channel R-29H - 0.1 μCi, Cl-36 Beta Checksource.
- Channel R-29I - 8 μCi Ba-133 Gamma Checksource.
- Channel R-29J - 0.1 μCi Cl-36 Beta Checksource.

A4. The R-29B/R-29D monitor performs a RG 1.97 function. The Unit 1 R-29B/R-29D and R-29C and Unit 2 R-29B/R-29D and R-29C monitors are Class 1E. The monitors have been environmentally qualified by the vendor for the environment in which they are located. The monitors are Q-Listed for the purpose of documenting repairs and modifications. As permitted by FSAR subsection 17.3.5, the replacement can be purchased to meet the quality requirements of the original specification. These quality requirements are the appropriate quality assurance provisions for the purpose of meeting the FSAR subsection 17.3.1 requirements for replacements.

B. Main Condenser Air Removal Monitor - Channel R-15B and C

The main condenser air removal exhaust systems for Units 1 and 2 are monitored using the existing monitor (described in paragraph 11.4.2.2) on the steam jet air ejector exhaust for the normal range of radioactivity. The accident range of radioactivity will be monitored for Units 1 and 2 by intermediate and high range detectors with overlapping ranges and located at the common vent duct for the turbine building. The accident monitor consists of two detectors and readouts. The intermediate range detector readout module has a range of indication of 0.1 to 100 mR/h. The high range detector readout module has a range of 10 mR/h to 1000 R/h. The relationship between mR/h and $\mu Ci/cm^3$ will be established for the noble gas isotopes present during an accident. The range of the accident monitors in $\mu Ci/cm^3$ is from 10^{-5} to 10^3, with the normal range monitor measuring concentrations down to 10^{-6} $\mu Ci/cm^3$. This is the required range for the case where the steam jet air ejector exhaust is combined with turbine building ventilation exhaust. The readout modules are located in the control room and provide continuous indication. The accident detectors are shielded from background radiation with 6 in. of lead. Calibration is by use of an external calibration source and is performed upon installation and at intervals specified in the Technical Requirements Manual.

C. Main Steam Line Monitors - Channel R-60 (A through C)

The steam generators main steam lines will be monitored by measuring the radiation levels in these steam lines. There are three detectors located adjacent to the main steam lines in the main steam valve room. One detector will be used to monitor the main steam line from each steam generator The monitors have a range of 0.01 mRad/h to 1000 mRad/h. Each detector will be connected to a readout module in the control room, providing continuous indication. Each detector is collimated and background shielded with 2.5 in. of lead.

Calibration for both units is by use of an external calibration source and is performed upon installation and at intervals specified in the Technical Requirements Manual.

D. Design for Noble Gas Effluent Monitors

The noble gas effluent monitors are powered from a vital instrument bus. Procedures have been developed for use, calibration of the system, and dissemination of release rate information. The original APC position was to monitor the main condenser air removal exhaust and the discharge from the steam generator safety relief valves and atmospheric relief valves with a portable gamma survey instrument. However, APC finalized the above position based on NRC questions during the latter part of 1980 and purchased the best available monitors upon finalization of this position. To ensure accurate reading of each of these monitors, a complex shielding design is required to discriminate actual readings from background, including containment shine.

11.4.2.2.21 Control Room Makeup Air Inlet Gas Monitors – Channel R-35A and B

A. Introduction

During normal plant operation, control room air is recirculated through air conditioning units to maintain control room design conditions of temperature and relative humidity. Fresh air makeup is provided by a supply duct from the computer room air conditioning unit. Redundant radiation monitors are provided on the makeup air supply duct. When a high radiation level is sensed by the monitors, a high radiation alarm is actuated in the control room, the air path is isolated to prevent entry of radioactive contaminants, and the control room ventilation system is manually aligned to the emergency recirculation mode. In addition, the radiation monitor signal also isolates the outside air intake to the technical support center (TSC) and realigns the TSC ventilation to the emergency filtration mode. After isolation of the control room and when conditions permit, fresh air can be brought in manually through redundant control room pressurization charcoal filter systems.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the control room makeup air gas monitors:

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The criteria are not applicable.

I. Conformance to Regulatory Guide 1.22

The control room makeup air isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

When the gaseous activity in the control room makeup air duct reaches the high level setpoint, automatic isolation of the outside air intake to the control room and the TSC ventilation system is shifted to the emergency filtration mode. Following the isolation of the normal control room intake, monitoring of activity in the control room and TSC is provided by the control room area radiation monitor (channel R-1), TSC area radiation monitor (R-01B), and health physics surveys.

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in subsection 9.4.1.

Q. Supporting Systems

Supporting systems for the redundant control room makeup air gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail-safe with respect to all of these supporting systems.

R. Nonsafety Systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity levels of the control room makeup air flow during normal plant operations.

A motor-driven positive displacement pump is used to draw a continuous sample from the control room makeup air line and direct the sample through a particulate removal prefilter. The sample is then routed to a 4π shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample is monitored for radioactivity by a thin beta crystal scintillation detector assembly placed within the sample chamber in contact with the sample, prior to being returned to the makeup air duct.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs energy in the form of a pulse (current) which is fed directly into a preamplifier at the base of the detector assembly and in turn provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits. The ratemeters have a five-decade range (10^1 to 10^6 cpm).

Power for channels R-35A and B is provided from vital motor control centers A and B, respectively.

A "loss of power" and "channel failure" are monitored for each detector providing annunciation in the control room.

A channel performance test is available to the operator. An electronic pulse signal is used to verify the performance of the readout instrumentation.

A radioactive checksource, controlled from the readout instrument in the control room, can be actuated to check system integrity. This checksource is used as a convenient operational and gross calibration check of the detection system.

The checksource is of the same or similar energy and range of the isotopes to be monitored.

A three-way, solenoid-operated valve at the sample chamber inlet is operable from the control room. It is provided to permit air purging of the sample chamber to facilitate background activity checks.

Visual/audible indication of channel failure and/or high radiation is provided in the control room.

11.4.2.3 Alarm Setpoint Basis

The alarm setpoints for the process radiation monitors are based on the following:

A. The methodology used to calculate setpoints for RE-13, RE-15, RE-18, RE-23B, RE-24, Unit 1 Channels R-29E and R-29J and Unit 2 Channels R-29E and R-29J is specified in the Offsite Dose Calculation Manual (ODCM). The RE-23A setpoint methodology, while not specified in the ODCM, will be the same as that for RE-23B.

B. The RE-15B setpoint will be based such that the monitor will alarm at one half decade before RE-15 goes offscale. The RE-15C setpoint will be based such that the monitor will alarm at one half decade before RE-15B goes offscale.

C. Detector response which will provide warning to the operator of leakage of activity into a normally low activity system. This includes channels RE-11, RE-12, RE-17A and B, RE-19, RE-20A and B, and RE-60A, B, and C.

D. The detector response which will provide to the operator warning of plant vent stack effluent in accordance with Regulatory Guide 1.97. This includes R-29B/R-29D. In addition, the R-29B low-range noble gas channel and the Unit 1 and Unit 2 Monitor RE-29C iodine channel setpoints are based on the NOUE emergency classification criteria, annual average meteorology, on ODCM-based dose conversion factors, and maximum plant vent stack flowrate.

Typical alarm setpoints for the process radiation monitors are listed in table 11.4-3.

11.4.2.4 Design Evaluation

The liquid and gaseous waste discharge monitors are provided to maintain surveillance over the release of radioactivity with the following features:

A. A checksource is provided to permit the operator to check the monitor before discharge by operating a switch on the radiation monitor system panel.

B. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

C. If the power supply to the channel fails, a high radiation alarm will annunciate. Control valves associated with the channel will also close.

An evaluation of instrumentation function relative to monitoring and for controlling release of radioactivity from various plant systems is discussed below.

A. Fuel Handling

For activity releases inside the containment and in the fuel handling area, the offline gas monitors (channel 24A and B or R-25A and B) would function. Each of these monitors initiates alarms in the control room and initiates ventilation isolation when the radiation level exceeds a preset level. Activity releases within the auxiliary building ventilation exhaust flow would cause the plant vent and area monitors to alarm on an increase in radiation level.

B. Liquid and Gas Wastes

For ruptures or leaks in the waste processing system, plant area monitors and the vent stack monitor will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of waste tanks, activity release is not controlled. The environmental consequences of the postulated accidents are based on no-instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The plant vent monitor will close the flow control valve in the waste decay tanks discharge line when the radiation level in the plant vent exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

C. Waste Gas Release Procedures

There is normally no need to vent the waste gas processing system, although occasional discharges will be required to perform maintenance. The waste gas release is an operator decision based on weather conditions and activity contained in the waste gas. When the operator has decided to release waste gas, he first samples the gas to determine its activity concentration. With this information and total pressure in the tank, the operator knows the quantity of activity to be released as well as the rate at which the gas can be released. To make the actual release, he must unlock and then open the manual isolation valve at the tank discharge and set the discharge flow control valve at the desired rate based on the plant vent activity monitor. Discharge flow is maintained at a constant rate by a pressure regulator upstream of the flow control valve. If the discharge flowrate results in an excessive radioactivity release rate, the flow control valve is tripped shut by the plant vent monitor.

D. Liquid Waste Release Procedure

The release of liquid waste is under administrative control. The normal procedure for discharging liquid waste is as follows:

1. A batch of waste is collected in one waste monitor tank.

2. The tank is isolated.

3. The tank contents are recirculated to mix the liquid.

4. A sample is taken for analysis.

5. If analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. Release is made when it is determined that the release is "as low as practicable" of necessity below permissible limits.

6. To release the liquid, an operator must unlock and open a stop valve in the discharge line, which is normally locked shut; open a second valve, which trips shut automatically on high radiation signal from the monitor (channel R-18); start a waste monitor tank pump and establish the normal flowrate using the flow indicator provided; and, finally, close the recirculation valve. Liquid is now being discharged.

E. Steam Generator Blowdown Release Procedure

See subsection 10.4.8 for a description of the steam generator blowdown release procedure.

F. Turbine Building Sump Release Procedure

There are two collection sumps located in each turbine building, each having a 30,000-gal capacity and isolated from each other. The radiation activity levels in these sumps are expected to be minimal. Each sump is provided with recirculation capability and will be grab sampled for composite prior to or during its discharge in conformance with Regulatory Guide 1.21. The discharge from this source is constant flow during a batch process. Radiation monitors at the discharge line would provide no meaningful information since the laboratory analysis of grab samples will be much more accurate; also, the instrument performance would be questionable as to the quality of water discharged from this system. Therefore, no in-process radiation monitors are justified, and none are provided.

G. Main Condenser Blowdown

The main condenser hotwell is blown down on occasion to assist in the maintenance of secondary water chemistry. The release is via the condensate pump discharge to the turbine building sump pump discharge line. This pathway is monitored as described in table 11.4-6.

11.4.3 SAMPLING

The following paragraphs present a detailed description of the radiological sampling procedures, frequencies, and objectives for all reactor plant process and effluent sampling. The process sampling system is described in subsection 9.3.2.

11.4.3.1 Process Sampling

The sample frequency, type of analyses, analytical sensitivity, and purpose of the sample are summarized in table 11.4-4 for each liquid process sample location and in table 11.4-5 for each gas process sample location. The analytical procedures used in sample analysis are presented in paragraph 11.4.3.3. This sampling monitors activity levels within various plant systems.

11.4.3.2 Effluent Sampling

11.4.3.2.1 Normal Operation Sampling

Effluent sampling of all potentially radioactive liquid and gaseous effluent paths will be conducted on a regular basis in order to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas. This effluent sampling program will be of such a comprehensive nature as to provide the information for the effluent measuring and reporting programs required by 10 CFR 50.36a, in annual reports to the NRC. The frequency of the periodic sampling and analysis described herein is a minimum. Table 11.4-6 summarizes the sample and analysis frequency schedules presented in the following paragraphs.

The following sample regime will apply to all potentially radioactive liquid effluents released to the plant discharge header from the liquid waste processing system.

A. Measurements will be made on a representative sample of each batch of effluent released and kept as a record together with the volume of the batch, the average dilution water flow used during discharge, and the time and date of release.

B. At least monthly a batch that is typical of average releases of radioactivity will be analyzed for dissolved fission and activation gases. This analysis has a minimum detectable concentration (MDC) as specified in table 11.4-6.

C. Proportional composite samples will be made up periodically to calculate total activity released. These will be samples in which the quantity of liquid added to the composite from each batch released will be proportional to the quantity of liquid in that batch. The composite will represent the average concentration prior to release and, by multiplying by the total volume released, will represent the quantity of radioactivity released during the compositing period. Such composite samples will be made up and analyzed in accordance with table 11.4-6

The steam generator blowdown system sample regime will be as specified in table 11.4-6.

Turbine building sump releases and condenser blowdown (during releases via this pathway) will be sampled and analyzed in accordance with table 11.4-6.

The following sample regime will apply to any intentional release from each containment purge exhaust:

A. The meteorological conditions of wind speed, wind direction, and atmospheric stability will be determined and averaged on an hourly basis for the purpose of determining the atmospheric dispersion during the period of release.

B. A representative gaseous sample of each release will be analyzed for individual noble gas nuclides in accordance with table 11.4-6. The gross noble gas activity released from the containment will be determined using grab samples.

C. A representative sample of each release will be analyzed for tritium in accordance with table 11.4-6. The samples will be collected by condensation or adsorption.

The following sample regime will apply to the potentially radioactive gaseous releases continuously discharged from the plant vent stack system and the condenser steam jet air ejector system:

A. Meteorological measurements of wind speed, wind direction, and atmospheric stability will be made and averaged over each 1-h period.

B. Gaseous activity releases will be quantitatively determined based on gaseous sample analyses and release flowrates for each of these effluent streams. The accumulated releases will be reported on a quarterly basis.

C. Within 1 month of initial criticality, at least monthly thereafter, and then following each refueling, process change, or other occurrence that could alter the mixture of radionuclide gas, samples will be analyzed for principal gamma emitting nuclides and tritium in accordance with table 11.4-6.

The following sample regime will also apply to the gaseous releases from the plant vent stack system:

A. A sample will be drawn through an iodine sampling device to determine the quantity of radioactive iodine isotopes released. The device will be analyzed at least weekly for I-131 and I-133 in accordance with table 11.4-6.

B. A continuous sample will be drawn through a particulate filter device and analyzed weekly for principal gamma emitting nuclides (Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, Ce-144, and I-131) in accordance with table 11.4-6.

C. A monthly analysis of gross alpha will be made on a composite of particulate filters for a duration of 1 month in accordance with table 11.4-6.

D. A quarterly analysis for Sr-89 and Sr-90 will be made on a composite of particulate filters for a duration of one calendar quarter in accordance with table 11.4-6.

11.4.3.2.2 Post-Accident Sampling and Analysis of Plant Effluents

Southern Nuclear Operating Company has the capability to provide continuous sampling of plant gaseous effluent for post-accident releases of radioactive iodine and particulates at the plant vent and the condenser air removal system. The sampling method involves passing a portion of the effluent gases through a filter assembly and transporting the filter to a counting room for analysis. The sampling system has the following capabilities:

A. Effective iodine absorption of > 90 percent for all forms of gaseous iodine.

B. Greater than 90-percent retention of particulates for 0.3-mm diameter particulates.

C. Design intent meets sampling requirements of American National Standards Institute N13.1-1969.

D. Continuous collection whenever exhaust flow occurs.

E. Analytical facilities and procedures considered in the design basis sample.

F. Shielding factors considered in the design.

Onsite laboratory capability exists to analyze or measure these samples. The sampling system design is such that plant personnel can remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the guidelines of General Design Criterion 19 of 5 rem whole body and 75 rem to the extremities during the duration of the accident, assuming the design basis shielding envelope of NUREG 0737.

The post-accident vent stack monitor draws its sample into its filter system downstream of HEPA filters that limit sample particle size to 0.3 μm and smaller. The smaller particulates behave like gas and eliminate the need for isokinetic sampling. In addition, there is a post-accident plant vent stack grab sampling system with particulate and iodine filters and a gas sampler that allow a technician to draw samples to be analyzed in the laboratory.

The steam jet air ejector sample point is located on the vertical section of the turbine building exhaust ventilation duct. Locating the sample point on the vertical section of the exhaust duct ensures that the absorber material is not degraded with entrapped water.

11.4.3.3 Analytical Procedures

Samples of process and effluent gases and liquids will be analyzed in the laboratory by the following techniques:

A. Gross alpha counting.

B. Gamma spectrometry.

C. Liquid scintillation counting.

D. Radiochemical separations.

Instrumentation that will be available in the laboratory for the measurement of radioactivity includes:

A. End-window Geiger-Mueller counter.

B. Windowless or low-background windowed 2μ internal proportional counter.

C. Liquid scintillation counter.

D. Gamma spectrometer:

    1. Intrinsic germanium detector.

    2. Multichannel analyzer system.

Gross alpha analysis of all liquid effluent samples may be performed with the internal proportional counter. Samples will be evaporated onto planchets for counting. Sample volume and counting time will be chosen to give the desired sensitivities. Corrections will be made for sample detector geometry, sample self-absorption, and other parameters as necessary to ensure accuracy.

Gross alpha analysis of air particulate will be performed by counting a portion of the filter paper in the internal proportional counter.

Gamma spectrometry will be used for isotopic analysis of gaseous, air particulate, and liquid samples. A high efficiency, high resolution intrinsic germanium detector will be available for resolving complex gamma spectra.

Gaseous tritium samples may be collected by condensation, adsorption, or freeze out. Liquid samples for tritium analysis with interfering impurities will be purified prior to analysis by ion exchange, distillation, and/or filtration. Samples will be counted on the liquid scintillation counter.

Liquid samples will be collected in polymer bottles to minimize adsorption of nuclides onto container walls.

Gaseous radioactive iodine sampling devices will be tested to determine sampling efficiency and/or sampling line losses. In lieu of in-plant testing, data from tests performed by other organizations will be used if available.

When abnormal activity levels are detected by the above procedures, plant conditions will be studied to determine the cause of the abnormal activity and corrective action taken to bring the abnormal conditions back to normal.

11.4.4 INSERVICE INSPECTION, CALIBRATION, AND MAINTENANCE

11.4.4.1 <u>Definitions</u>

Radiation monitor channel check - A channel check shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

Radiation monitor channel operational test - A channel operational test (COT) shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the operability of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

Radiation monitor channel calibration - A channel calibration shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The channel calibration shall encompass the entire channel, including the required sensor, alarm, interlock and trip functions. The channel calibration may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

11.4.4.2 Calibration Procedure

A primary calibration is performed on a one-time basis, utilizing typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and a secondary standard.

A remotely operated, long half-life radiation checksource is furnished as a secondary source in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause approximately 30 percent of full scale indication.

11.4.4.3 Test Frequencies

The radiation monitoring system channels will be channel checked daily and source checked monthly or quarterly. Channel operational testing will be done on a quarterly basis, and calibration will be performed every 18 months. These intervals for source checks, channel checks, channel operational tests, and channel calibrations are contained in plant procedures, Technical Specifications, ODCM, or the Technical Requirements Manual, as applicable.

REFERENCES:

1. NRC Safety Evaluation Report, J. M. Farley Nuclear Plant Unit 1 and Unit 2, NUREG-0117 Supplement No. 5 to NUREG-75/034, dated March 1981.

2. Letter from NRC, dated March 26, 1985, and enclosed SER related to the Post-Accident Sampling System.

3. Letter from NRC, dated January 7, 1987, and enclosed SER related to Regulatory Guide 1.97.

4. Letter from NRC, dated May 22, 2002, and enclosed SER related to FOL amendments 156 and 148 for Units 1 and 2, respectively.

TABLE 11.4-1

DETECTING MEDIUM CONDITIONS

Channel	Medium	Temperature Range (°F)
R-1	Air	40-95
R-1B	Air	40-104
R-2	Air	40-120
R-3	Air	40-104
R-4	Air	40-104
R-5	Air	40-104
R-6	Air	40-104
R-7	Air	40-120
R-8	Air	40-104
R-9	Air	40-104
R-10	Air	50-110
R-11	Air	50-120
R-12	Air	50-120
R-13	Air	50-500
R-15	Air	40-150
R-15B and C	Air	50-105
R-17A and B	Water	60-200
R-18	Water	60-500
R-19	Water	<100
R-20A and B	Water	30-110
R-23A and B	Air	50-120
R-24A and B	Air	40-120
R-25A and B	Air	40-110
R-27A and B	Air	60-367
R-29B	Air	40-120
1R-29C	Air	40-120
2R-29C	Air	40-120
R-35A and B	Air	8-107 Note 1
R-60A through C	Air	1-107
R-66A through F	Air	Note 2

Notes:

1) 8°F is the lowest allowable outside air temperature that can be heated to the detector low limit of 32°F before reaching the detector.

2) The monitors are located inside the low level radwaste building which has no air conditioning. Ventilation fans change the building air once each hour; therefore, the medium conditions are assumed to be the same as the outdoor temperature conditions of -1°F to 107°F.

TABLE 11.4-2 (SHEET 1 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-10	Penetration room filtration monitoring system	Air particulate	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137	85 kev to 2.28 mev-β	5000 cfm (Exh) 500 cfm (Recir)	2.0 cfm	Continuous	2.5	Normal operational penetration room activity	Post-LOCA penetration room activity	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-11	Containment monitoring system	Air particulate	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137	85 kev to 2.28 mev-β	N/A	2.0 cfm	Continuous	2.5	Normal operational containment activity	Post-LOCA containment activity levels	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-12	Containment monitoring system	Radioactive gases	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Ar-41 Kr-85 Xe-133 Xe-135	85 kev to 2.28 mev-β	N/A	2.0 cfm	Continuous	2.5	Normal operational containment activity	Post-LOCA containment activity levels	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-13	Waste gas processing system	Radioactive gases	Log rate meter and alarms on controls room monitoring panel	Geiger-Mueller	$1.0x10^{-1}$ to $1.0x10^{+4}$ *	Kr-85 Kr-87 Xe-133 Xe-135	100 kev to 3.0 mev-ϒ 200 kev to 3.0 mev-β	40 scfm @ 110 psig 140° F 2-in. SCH 40	10 cfm	Continuous	2.0		Maximum expected waste gas decay tank activity	1. Circuit failure 2. High radiation level 3. Test mode	None
R-15	Condenser air ejector monitoring system	Radioactive gases	Log rate meter and alarms on control room monitoring panel	Geiger-Mueller	$1.0x10^{-6}$ to $1.0x10^{-3}$ ($5x10^{-7}$ for Kr^{85})	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-ϒ (Kr^{85}) 200 kev to 3.0 mev-β (A^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	None

TABLE 11.4-2 (SHEET 2 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range $\mu Ci/cm^3$	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-15B	Condenser air ejector monitoring system	Radioactive gases	4-decade logarithmic scale meter and alarms on control room monitoring panel	Geiger-Mueller	$1.1x10^{-4}$ to $1.4x10^{-1}$	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-γ (Kr^{85}) 200 kev to 3.0 mev-β (Ar^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	None
R-15C	Condenser air ejector monitoring system	Radioactive gases	5-decade logarithmic scale meter and alarms on control room monitoring panel	Pressurized ion chamber	$1.1x10^{-2}$ to $1.4x10^{3}$	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-γ (Kr^{85}) 200 kev to 3.0 mev-β (Ar^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity activity	1. Circuit failure 2. High radiation level 3. Test mode	None
R-17 (A-B)	Component cooling water monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	3000	1-5 gpm	Continuous	2.0	Less than minimum detector sensitivity	Maximum expected CCW activity	1. Circuit failure 2. High radiation level 3. Test mode	Close CCW surge tank vent
R-18	Liquid waste processing monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	Max 100 gpm at 200 ft head 2-in SCH 40	1-5 gpm	Continuous	2.0	Normal radioactive waste activity level	Anticipated operational occurrences radioactive waste activity level	1. Circuit failure 2. High radiation level 3. Test mode	Close RCV-018 on Hi-Radiation
R-19	Steam generator radiation monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$ *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	1-5 gpm	1-5 gpm	Continuous	2.0	Normal steam generator activity	Post steam generator tube rupture accident activity level	1. Circuit failure 2. High radiation level 3. Test mode	Isolate steam generator sample lines

TABLE 11.4-2 (SHEET 3 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-20 (A-B)	Service water	Liquid	Log rate meter and alarms on control room radiation monitoring panel	NaI scintillation detector	1.0×10^{-5} to 1.0×10^{-2} *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	1600 to 4000 gpm	1-5 gpm	Continuous	2.0			1. Circuit failure 2. High radiation level 3. Test mode	None 55 55
-23 (A-B)	Steam generator blowdown processing monitoring system	Liquid	Log rate meter and alarms on control room radiation monitoring panel	NaI scintillation detector	1.0×10^{-6} to 1.0×10^{-3} *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	15 to 37.5 gpm 2" and 3" SCH 40	1-5 gpm	Continuous	1.0	Normal steam generator activity <1.0x10 μc/cc	Post steam generator tube rupture accident activity level	1. Circuit failure 2. High radiation level meeting low as practicable 10 CFR 50 for continuous discharge 3. Test mode	Isolate steam generator blowdown process system and discharge line on high alarm
R-24 (A-B)	Containment purge monitoring system	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	1.0×10^{-6} to 1.0×10^{-3} *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	50,000 cfm 25,000 cfm	10 cfm	Continuous	2.0	Normal operational containment activity	Post-LOCA containment activity levels	1. Circuit failure 2. High radiation level 3. Test mode	Containment purge vent isolation
R-25 (A-B)	Fuel handling monitoring system	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	1.0×10^{-6} to 1.0×10^{-3} *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	20,000 cfm	10 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels	Post fuel handling accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	Fuel handling area vent isolation
R-29B[a]	Plant vent stack effluent monitors, wide range gas	Gaseous	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (1); gamma cadmium telluride solid-state detector (2)	1.0×10^{-4} to 1.0×10^{5} *	Ar-41 Kr-85 Xe-133 Xe-135	85 kev to 2.28 mev-β 75 kev to 2.4 mev-γ	150,000 cfm (maximum) 5000 cfm (minimum)	Normal Skid Flow 2.5 cfm Accident Skid Flow 0.06 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels	Post-LOCA containment activity levels	1. Oper failure 2. High radiation level 3. Test mode	Closure of gas release valve in waste gas processing system

TABLE 11.4-2 (SHEET 4 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range $\mu Ci/cm^3$	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
1R-29C(b1)	Plant vent stack monitoring system	Particulate/ iodine/low range gas	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (2); NaI crystal/photomulti plier detector (1)	$1.0x10^{-9}$ to $1.0x10^{-6}$ (Particulate) $1.0x10^{-10}$ to $1.0x10^{-6}$ (iodine) $5.0x10^{-7}$ to $1.0x10^{-4}$ (Noble Gas)*	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137 Xe-133 Xe-135 Kr-85 Ar-41	85 kev to 2.28 mev-β 329 kev to 407 mev-γ	150,000 cfm (maximum) 5000 cfm (minimum)	2.5 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels		1. Oper failure 2. High radiation level 3. Test mode	None
2R-29C(b2)	Plant vent stack monitoring system	Particulate/ iodine/low range gas	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (2); NaI crystal/photomulti plier detector (1)	$1.0x10^{-9}$ to $1.0x10^{-6}$ (Particulate) $1.0x10^{-10}$ to $1.0x10^{-6}$ (iodine) $5.0x10^{-7}$ to $1.0x10^{-4}$ (Noble Gas)*	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137 Xe-133 Xe-135 Kr-85 Ar-41	85 kev to 2.28 mev-β 329 kev to 407 mev-γ	150,000 cfm (maximum) 5000 cfm (minimum)	2.5 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels		1. Oper failure 2. High radiation level 3. Test mode	None
R-35 (A-B)	Control room makeup air inlet	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	$1.0x10^{-6}$ to $1.0x10^{-3}$ *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	20,000 cfm	10 cfm	Continuous	2.0	Normal operational outside air activity	< 800 cpm	1. Circuit failure 2. High radiation level 3. Test mode	Control room/TSC ventilation isolation
R-60 (A-C)	Main steam line monitors	Gaseous	LCD display with scale meter and alarm indications and alarms on control room monitoring panel	NaI(Tl) scintillation detector	$1.0x10^{-2}$ to $1.0x10^{4}$ *	Ar-41 Kr-85 I-131 I-133 Xe-133 Xe-135	80 kev to 3.0 mev-γ			Continuous	2.0	Normal operational main steam line activity levels	Post steam generated tube rupture accident activity levels	1. Oper failure 2. High radiation level 3. Test mode	None

TABLE 11.4-2 (SHEET 5 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function

(a) This refers to monitor R-29B. Channels R-29E (Low), R-29F (Mid), R-29G (high), and R-29B - Composite Gas.

(b1) This refers to Monitor Unit 1 R-29C. Channels R-29H (Particulate), R-29I (Iodine), and R-29J (Low Gas).

(b2) This refers to Monitor Unit 2 R-29C. Channels R-29H (Particulate), R-29I (Iodine), and R-29J (Low Gas).

* range not for all isotopes

TABLE 11.4-3 (SHEET 1 OF 2)

PROCESS MONITOR ALARM SETPOINTS

Channel	Monitor	Alarm Level[a]
R-10	Penetration room filtration air particulate	1420 cpm
R-11	Containment air particulate	1420 cpm
R-12	Containment radioactive gas	320 cpm
R-13	Waste gas processing	18,000 cpm
R-15	Condenser air ejector	3000 cpm
R-15B	Condenser air ejector, intermediate range	4.5 mR/h
R-15C	Condenser air ejector, high range	50 mR/h
R-17A and B	Component cooling liquid	320 cpm
R-18	Waste processing liquid effluent	320 cpm
R-19	Steam generator liquid sample	320 cpm
R-20A and B	Service water liquid	320 cpm
R-23A and B	Steam generator blowdown processing system liquid	320 cpm
R-24A and B	Containment purge monitor	13,000 cpm
R-25A and B	Spent fuel pool exhaust flow gas monitor	55,000 cpm
R-29B[b]	Plant vent stack effluent monitors, wide-range gas	R-29E - 4.44×10^{-4} µCi/cc R-29B - Composite Gas 1.31×10^{-4} µCi/cc
1R-29C[c1]	Plant vent stack monitor particulate/iodine/low-range gas	R-29H - 6.7×10^{-8} µCi/cc R-29J - 1.31×10^{-4} µCi/cc
2R-29C[c2]	Plant vent stack monitor particulate/iodine/low-range gas	R-29H - 6.7×10^{-8} µCi/cc R-29J - 1.31×10^{-4} µCi/cc
R-60A through C	Main steam line monitors	7×10^{-2} µCi/cc

TABLE 11.4-3 (SHEET 2 OF 2)

PROCESS MONITOR ALARM SETPOINTS

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a. These are typical values. Actual values are contained in the plant Technical Specifications, the Technical Requirements Manual, and/or the plant radiological control and protection procedures.

b. This refers to Monitor R-29B, Channels R-29E and R-29B - Composite Gas.

c1. This refers to Unit 1 Monitor R-29C, Channels R-29H and R-29J.

c2. This refers to Unit 2 Monitor R-29C, Channels R-29H and R-29J.

TABLE 11.4-4 (SHEET 1 OF 5)

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES

Sampling Location	Sampling[(a)] Frequency	Analysis Performed	MDC[(b)] (μCi/cm^3)	Reason for Analysis
Reactor Coolant System				
Hot leg reactor coolant	Weekly	Gamma spectrometric	10^{-6}	Evaluate integrity of fuel cladding and monitor levels of fission and activated corrosion products.
	Weekly	Tritium	10^{-5}	
	Monthly	Gross $\propto$	10^{-6}	
	Weekly	Selected noble gases	10^{-5}	
	Monthly	Activated corrosion products	-	Determine purification requirements.
	Biweekly	Dose equivalent I-131	10^{-6}	
	Semiannually	Average energy $\bar{E}$	-	Determine limits for activity in the reactor coolant system.
Chemical and Volume Control System				
Upstream of mixed bed demineralizer	Monthly	Gamma spectrometric	-	Evaluate performance of letdown purification demineralizers.
Downstream of mixed bed and cation bed demineralizers[(c)]	Weekly	Gamma spectrometric	-	Evaluate performance of letdown purification demineralizers.
	Monthly	Gamma spectrometric	-	
Holdup tanks	Each batch	Gamma spectrometric	10^{-6}	Determine purification requirements of waste water. Evaluate performance of recycle evaporator feed demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	

TABLE 11.4-4 (SHEET 2 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] (μCi/cm^3)	Reason for Analysis
Downstream recycle evaporator feed demineralizers	Each batch	Gamma spectrometric	10^{-6}	Determine purification requirements of waste water. Evaluate performance of recycle evaporator feed demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	
Downstream recycle/holdup tanks	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator.
	Monthly	Gamma spectrometric	10^{-5}	
Recycle evaporator distillate	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of recycle evaporator.
Downstream recycle evaporator condensate demineralizer	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator condensate demineralizer
	Monthly	Gamma spectrometric	10^{-5}	
Recycle evaporator concentrates	Each batch	Gamma spectrometric	10^{-6}	Determine whether to recycle or solidify.
Safety Injection System				
Refueling water storage tank	Monthly	Gamma spectrometric	10^{-6}	General monitoring.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Accumulator tanks	Monthly	Gamma spectrometric	10^{-6}	General monitoring.
Component Cooling Water System				
Downstream of component cooling water pumps	Weekly	Gamma spectrometric	10^{-6}	Check on component cooling water system radiation monitors.

TABLE 11.4-4 (SHEET 3 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] ($\mu Ci/cm^3$)	Reason for Analysis
Service Water System				
Downstream of component cooling water heat exchangers	Weekly	Gamma spectrometric	10^{-6}	Monitor inleakage from component cooling water system.
Residual Heat Removal System	Once/72 h	Gross $\beta\gamma$	10^{-6}	Evaluate integrity of fuel cladding and monitor levels of fission and activated corrosion products.
	Once/72 h	Gamma spectrometric	10^{-6}	
Recyclable (Channel A) Waste Treatment System				
Reactor coolant drain tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to recycle or process.
Downstream waste holdup tank	Each batch	Gamma spectrometric	10^{-6}	Determine processing requirements
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of waste evaporator.
Waste evaporator distillate	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of waste evaporator.
	Monthly	Gamma spectrometric	10^{-5}	
Downstream waste evaporator distillate demineralizer	Each batch	Gross $\beta\gamma$	10^{-6}	Evaluate performance of waste evaporator condensate demineralizer
	Monthly	Gamma spectrometric	10^{-5}	
Waste condensate tank	Each batch	Gamma spectrometric	10^{-6}	Determine disposition of processed condensate
Waste evaporator concentrates	Each batch	Gamma spectrometric	10^{-6}	Determine whether to return to boron recycle system or to solid waste disposal.
	Monthly	Gamma spectrometric	10^{-5}	

TABLE 11.4-4 (SHEET 4 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] (μCi/cm^3)	Reason for Analysis
Nonreactor Grade (Channel B) Waste Treatment System				
Laundry and hot shower tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to process or discharge.
Floor drain tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to process or discharge.
Downstream waste monitor tank demineralizer	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
Waste monitor tanks	Each batch	Gamma spectrometric	5×10^{-7}	Determine whether to reprocess or discharge.
Primary Makeup Water System				
Primary water storage tank	Monthly	Gamma spectrometric	10^{-6}	Evaluation of systems for recycling waste water.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Spent Fuel Pool Cooling and Demineralizer System				
Spent fuel pool water	Biweekly	Gamma spectrometric	10^{-6}	Determine purification requirements and evaluate leakage from spent fuel.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Upstream fuel pool demineralizer	Monthly	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of demineralizer.
Downstream fuel pool demineralizer	Monthly	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of demineralizer.
Condensate and Feedwater System				
Steam generator feedwater	Periodically	Gamma spectrometric	10^{-7}	Determine radionuclide carryover in main

TABLE 11.4-4 (SHEET 5 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] ($\mu Ci/cm^3$)	Reason for Analysis
				steam.
	Periodically	Gamma spectrometric	5×10^{-7}	Determine radionuclide carryover in main steam.
Steam Generator Blowdown System				
Downstream of steam generator ion exchangers when in use	5 days/week	Gamma spectrometric	10^{-7}	Evaluate performance of demineralizer.
	Weekly	Gamma spectrometric	5×10^{-7}	Evaluate performance of demineralizer.

a. These are typical frequencies. Actual frequencies are contained in the plant Technical Specifications and the plant operating procedures.

b. For principle γ emitters as per STS.

c. When in use.

TABLE 11.4-5

RADIOLOGICAL ANALYSIS SUMMARY OF GAS PROCESS SAMPLES

Sampling Location	Sampling Frequency[a]	Analysis Performed	MDC ($\mu Ci/cm^3$)	Purpose of Sample
Chemical volume and control system holdup tanks vapor space	Weekly	Noble gas isotopic	10^{-4}	Evaluate tank leakage; determine whether to purge to GCH.
Containment atmosphere	Prior to personnel entry	Noble gas isotopic	10^{-7}	Personnel radiation protection; determine necessity for operation of cleanup recirculation units and/or containment purge systems.
		Halogens	10^{-9}	
		Air particulate	10^{-9}	
		Tritium	10^{-6}	

a. These are typical frequencies. Actual frequencies are contained in plant operating procedures.

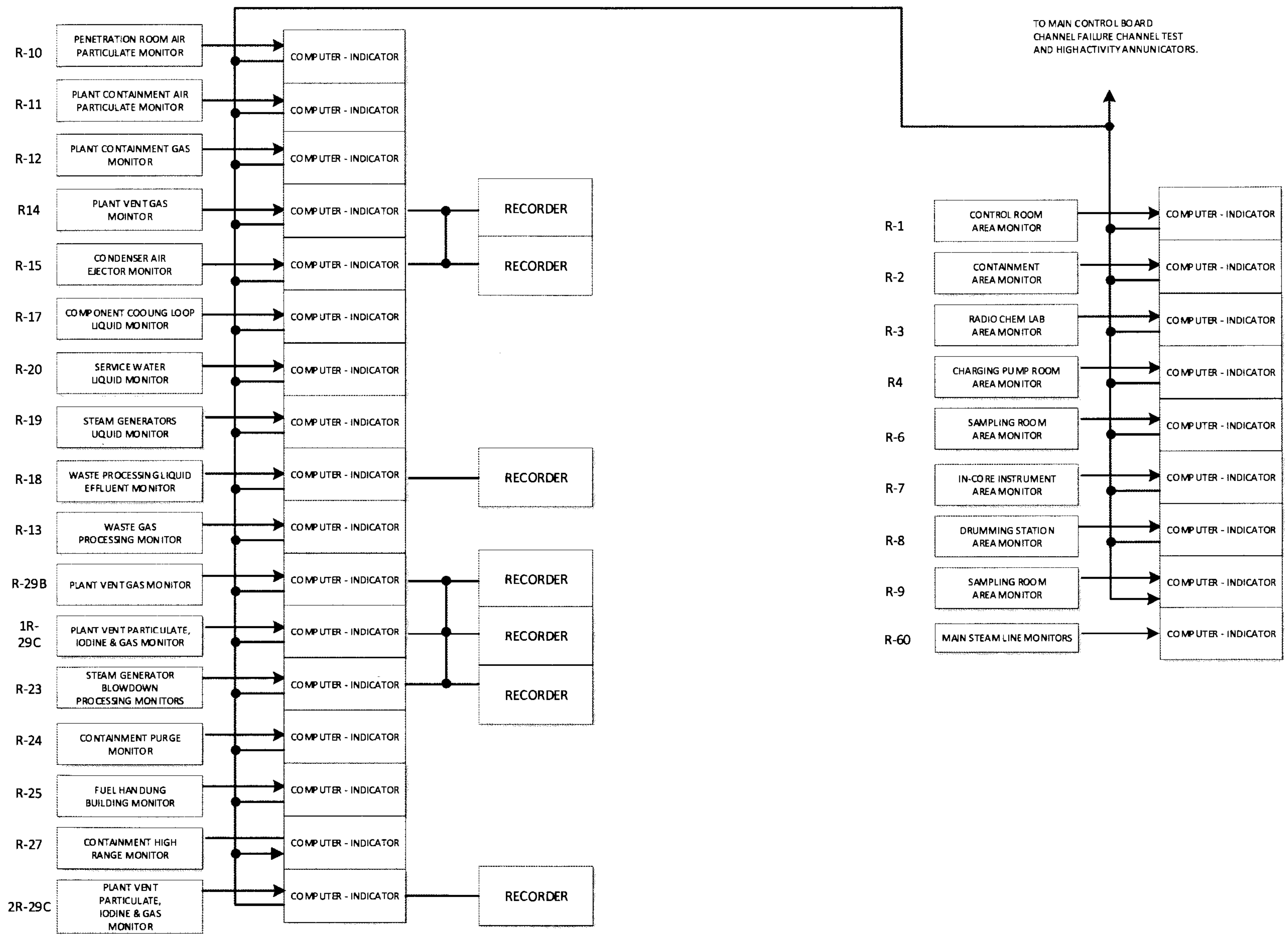
TABLE 11.4-6 (SHEET 1 OF 2)

EFFLUENT SAMPLE AND ANALYSIS SCHEDULE

Effluent Stream	Sample Frequency	Minimum Analysis Frequency	Type of Activity Analysis Performed	Minimum Detectable Concentration (MDC) (μCi/ml)
Radioactive waste processing system discharges (Batch Waste Release Tanks)	Each batch prior to discharge	Each batch prior to discharge	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	One batch/month prior to discharge	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Each batch prior to discharge	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Each batch prior to discharge	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Steam generator blowdown processing system discharge	Daily grab sample	Weekly composite	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	Monthly grab sample	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Daily grab sample	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Daily grab sample	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Condenser Blowdown	Daily grab sample During discharge	Weekly composite	Principal gamma emitters H-3	5×10^{-7} 1×10^{-5}
Turbine building sump	Grab sample prior to or during release	Weekly composite	Principal gamma emitters H-3	5×10^{-7} 1×10^{-5}

TABLE 11.4-6 (SHEET 2 OF 2)

Effluent Stream	Sample Frequency	Minimum Analysis Frequency	Type of Activity Analysis Performed	Minimum Detectable Concentration (MDC) (μCi/ml)
Waste gas storage tank releases	Grab sample each tank prior to release	Each tank prior to release	Principal gamma emitters	1×10^{-4}
Containment purge	Grab sample each purge prior to release	Prior to each purge	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Condenser steam jet air ejector discharge	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Plant vent stack	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Plant vent stack, Containment purge	Continuous (charcoal)	Weekly charcoal sample	I-131 I-133	$1 \times 10\text{-}^{12}$ $1 \times 10\text{-}^{10}$
	Continuous	Weekly particulate sample	Principal gamma emitters (I-131 and others)	1×10^{-11}
	Continuous	Monthly composite particulate sample	Gross alpha	1×10^{-11}
	Continuous	Quarterly composite particulate sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble gas monitor	Noble gases Gross Beta & Gamma	1×10^{-6}



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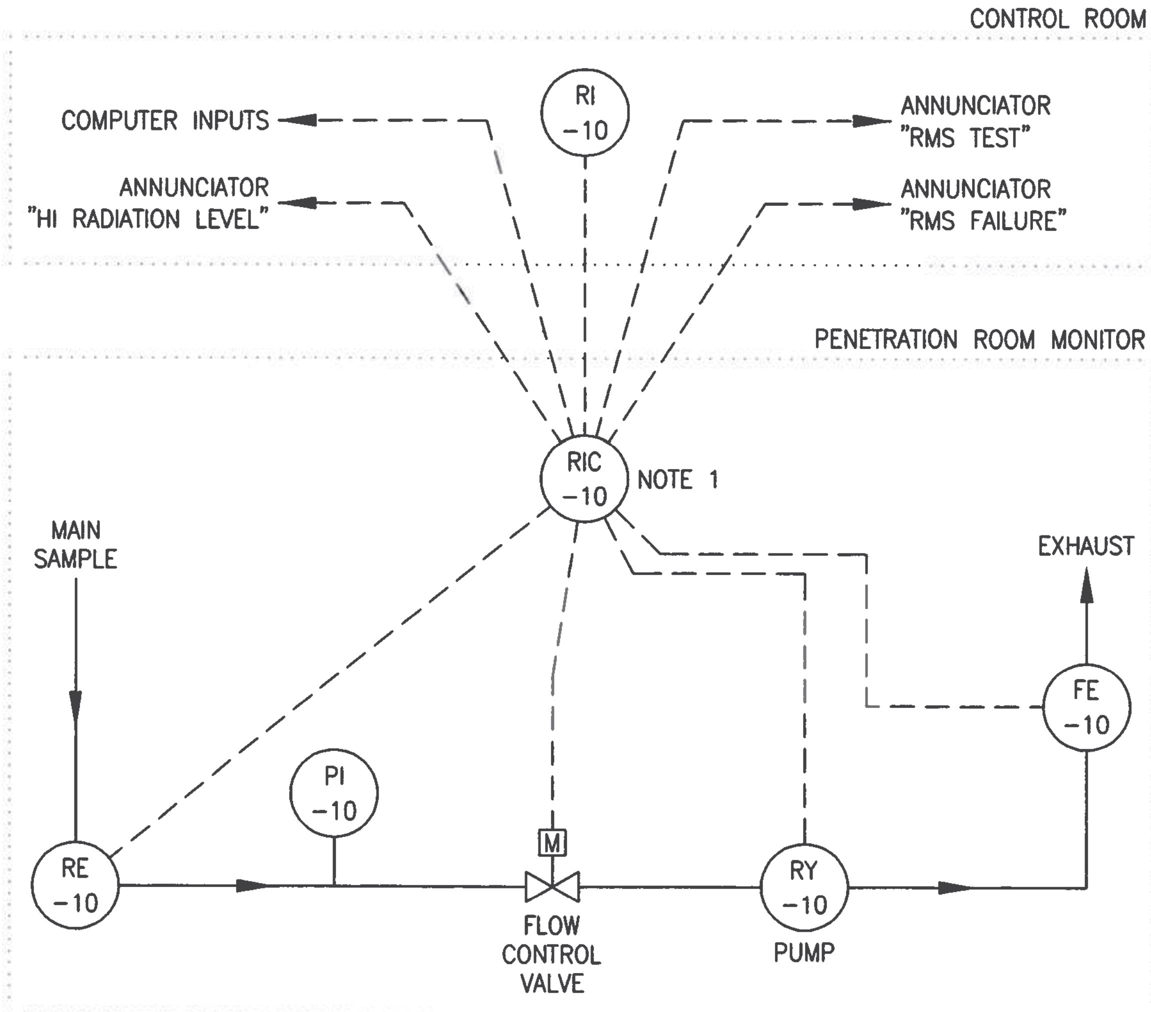
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM

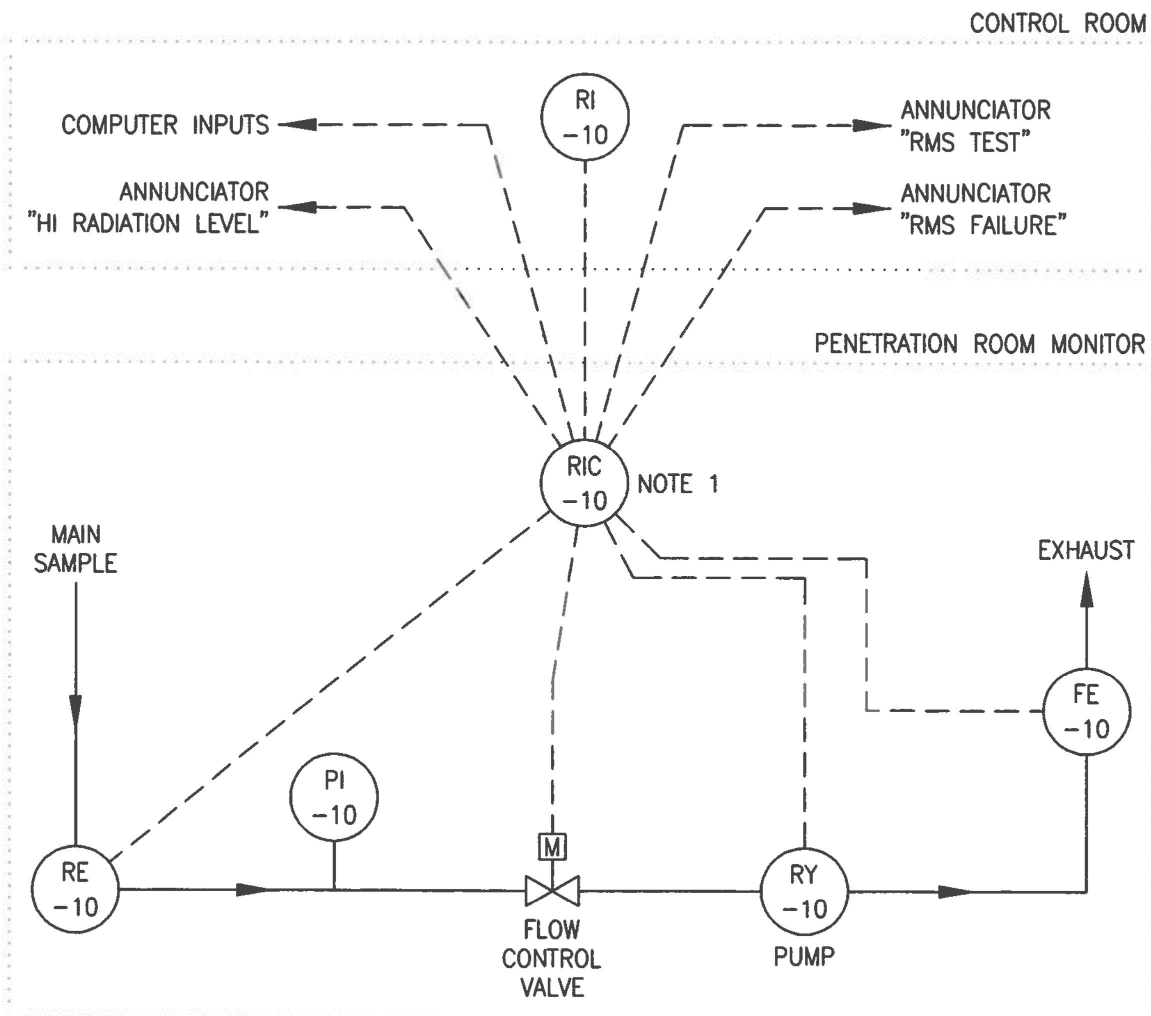
FIGURE 11.4-1



NOTES:
1. RIC-10 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMP, & FLOW CONTROL.

REV 23 5/11

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RADIATION MONITORING SYSTEM AREA RANGE R-10
		FIGURE 11.4-2



NOTES:
1. RIC-10 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMP, & FLOW CONTROL.

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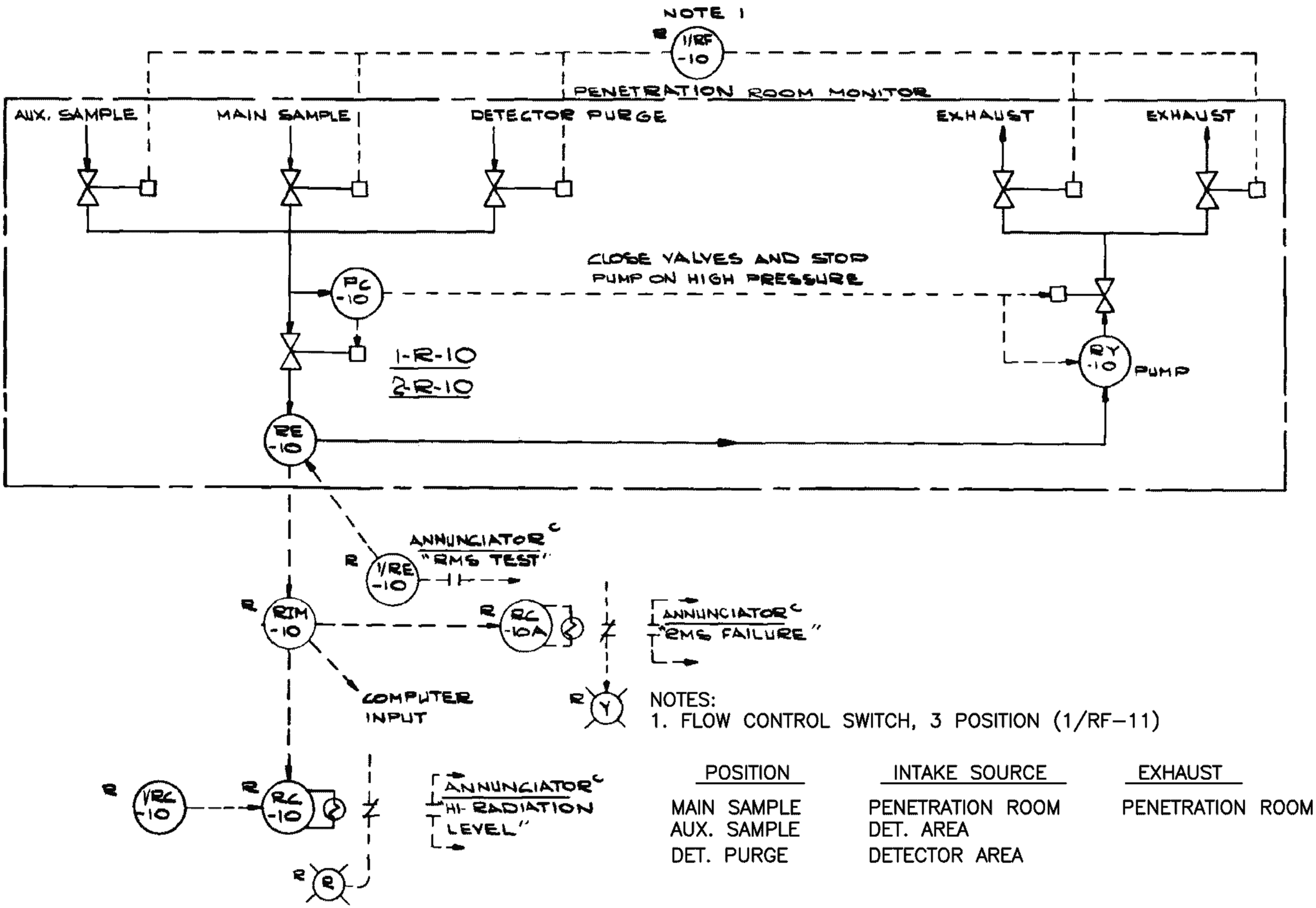
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1

RADIATION MONITORING SYSTEM
AREA RANGE R-10

FIGURE 11.4-2a



NOTES:
1. FLOW CONTROL SWITCH, 3 POSITION (1/RF-11)

POSITION	INTAKE SOURCE	EXHAUST
MAIN SAMPLE	PENETRATION ROOM	PENETRATION ROOM
AUX. SAMPLE	DET. AREA	
DET. PURGE	DETECTOR AREA	

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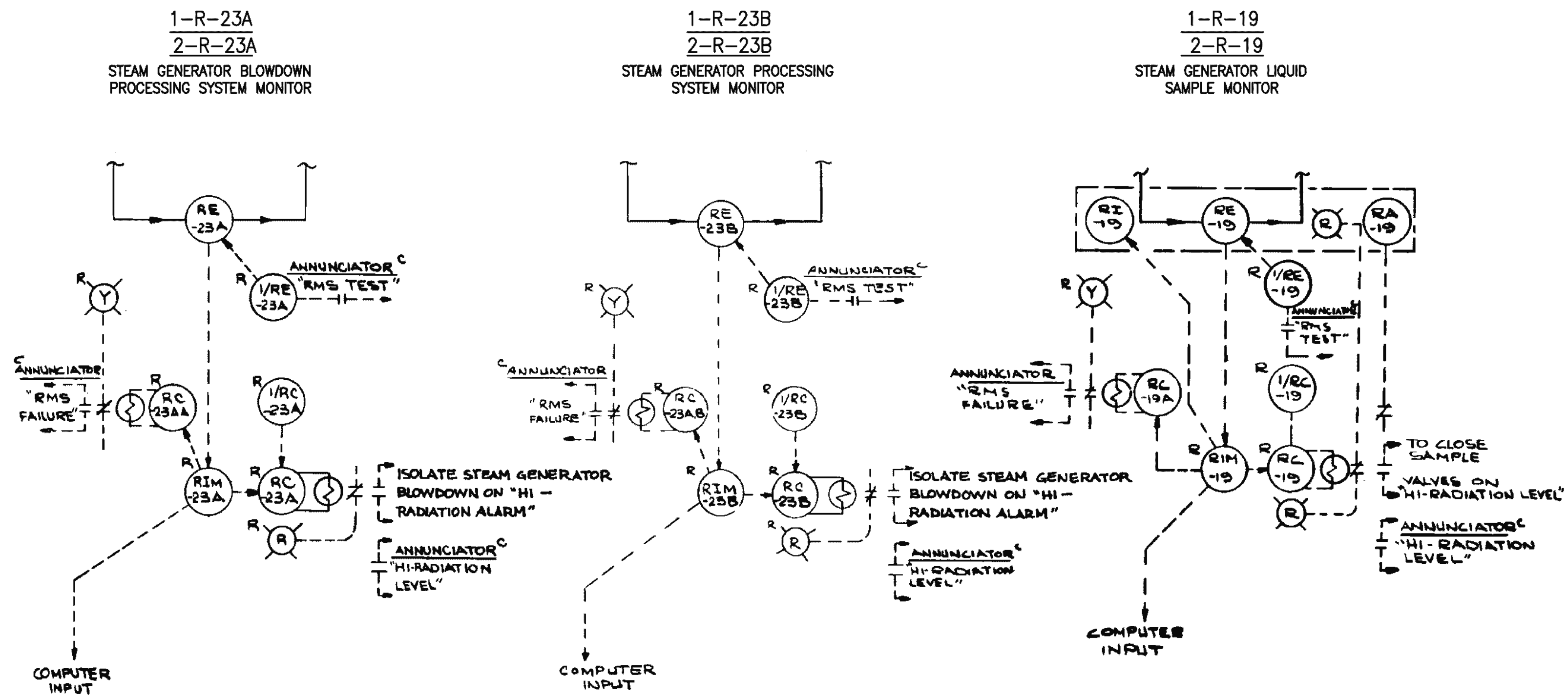
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-10

FIGURE 11.4-2b



REV 21 5/08

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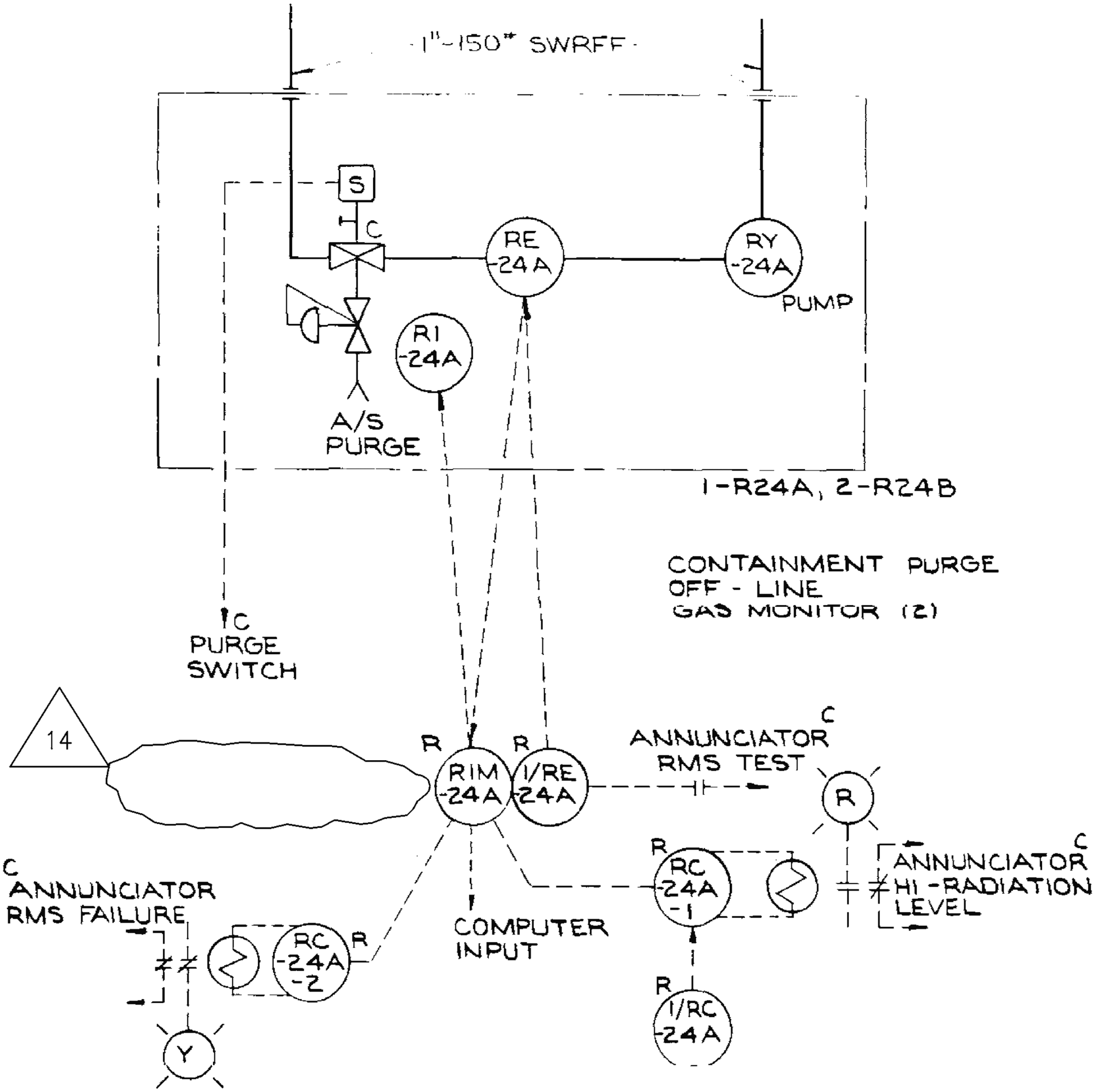
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-19, R-23A, AND R-23B

FIGURE 11.4-3



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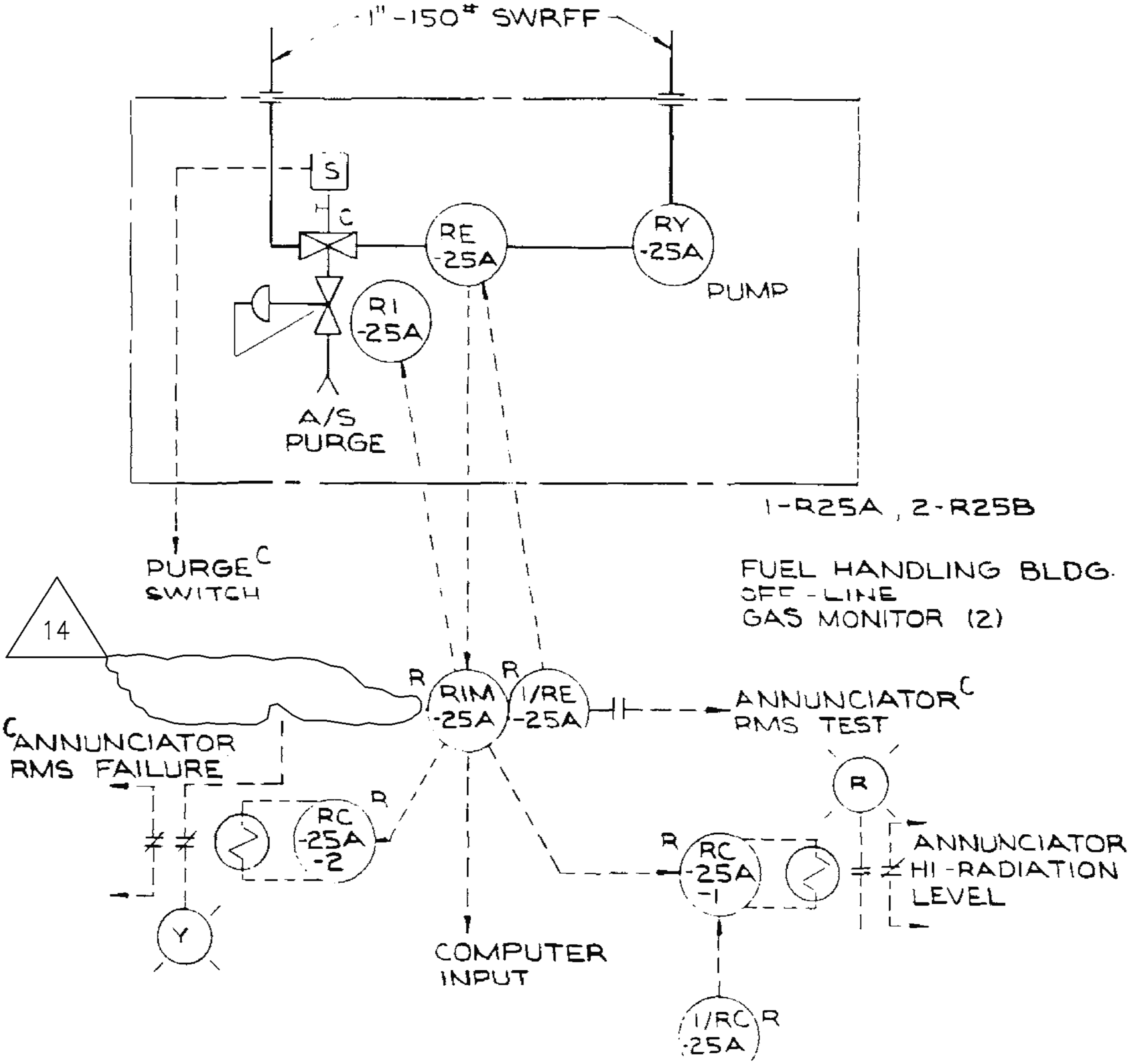
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
FUNCTIONAL BLOCK DIAGRAM (R-24)
CONTAINMENT PURGE MONITOR

FIGURE 11.4-4



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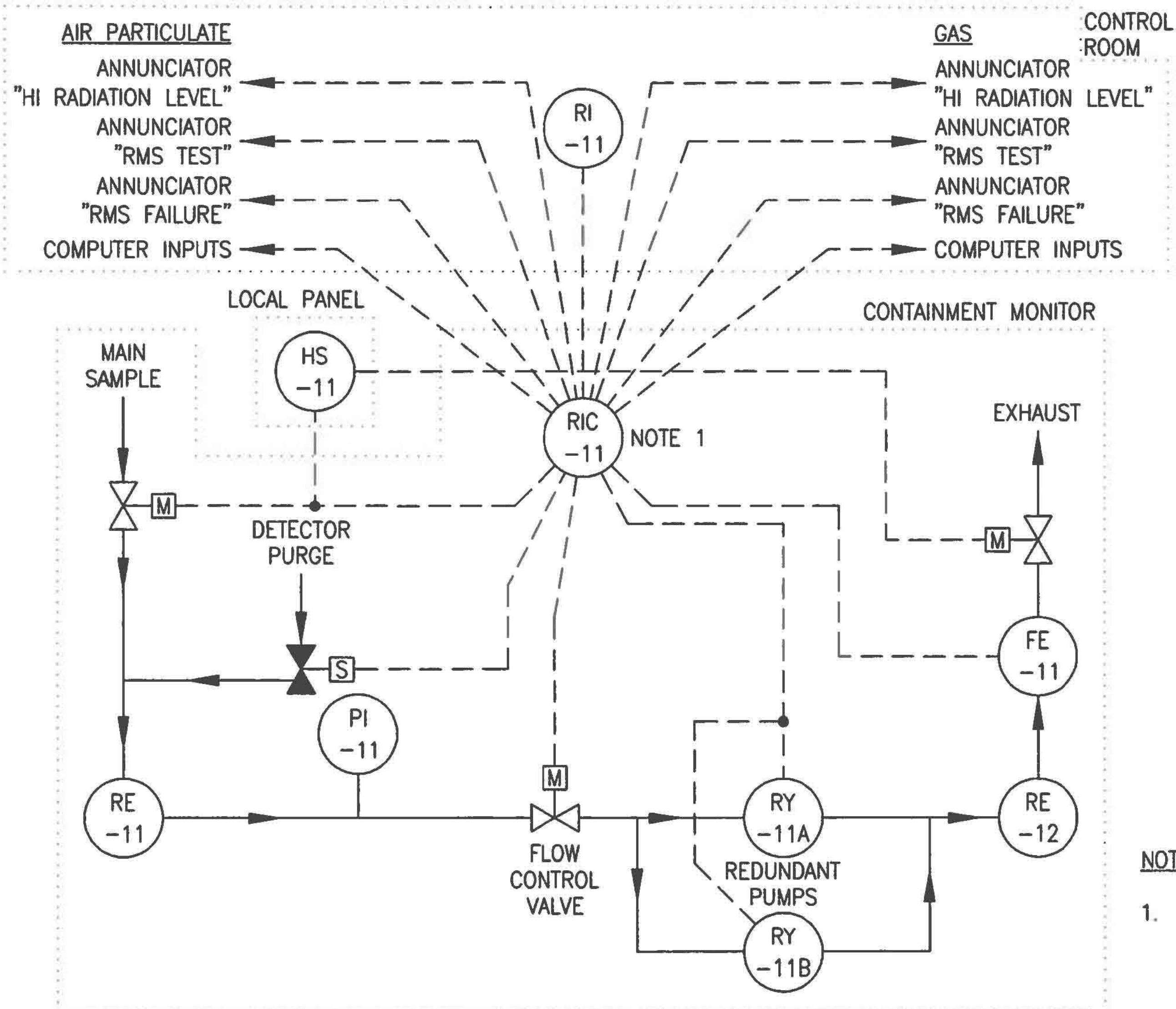
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
FUNCTIONAL BLOCK DIAGRAM (R-25)
FUEL HANDLING BUILDING MONITOR

FIGURE 11.4-5



NOTES:

1. RIC-11 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMPS, FLOW CONTROL, & PURGE VALVE OPERATION.

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-11 AND R-12

FIGURE 11.4-6

11.5 SOLID WASTE SYSTEM

11.5.1 DESIGN OBJECTIVES

The solid waste system is designed to transfer spent resins, evaporator concentrates, and chemical tank effluents. This system is installed in Unit 1 and has adequate capacity to serve both units.

To provide more efficient solidification and to ensure compliance with current burial ground license requirements (including volume restrictions), provision has been made for the use of a portable cement solidification system. The portable system is operated in the solidification and dewatering facility outside the Unit 1 and Unit 2 auxiliary building and is capable of solidifying resins, evaporator concentrates, and chemical drains from both units. The system also serves as a solidification system for the disposable demineralizer system, should solidification be required prior to shipment.

A separate system is available to compact dry active wastes such as paper, disposable clothing, rags, towels, floor coverings, shoe covers, plastics, cloth smears, and respirator filters.

It is estimated that about 15,000 ft^3 of solid waste are produced for burial each year which includes dry active waste (DAW).

The systems are used to package radioactive wastes within the limitations specified by 10 CFR 20, 10 CFR 61, 10 CFR 71, and 49 CFR 170 through 178. Bulk waste may be shipped to a licensed waste processor or to a disposal facility without encapsulation or solidification in accordance with these regulations and per applicable license and regulations for the receiver of the waste.

Shielding is designed to limit general area radiation levels in the drumming rooms, drum storage rooms and the low level radwaste building.

During normal work activities, tools, scrap, and other miscellaneous equipment and materials may become radioactively contaminated. The Solidification/Dewatering Facility (SDF) can be used as a decontamination area when needed. Appropriate administrative controls as determined by FNP personnel will be reviewed and implemented to ensure compliance with regulatory guidelines if used as a decontamination area.

Solidification via the portable system is accomplished with the liner inside a shipping cask or a shielded enclosure in the solidification and dewatering facility, which provides the necessary personnel shielding.

11.5.2 SYSTEM INPUTS

Input to the solid waste system comes from the spent resin storage tanks, waste evaporator, concentrated waste tank, and chemical drain tank. Solid, compressible wastes are products of the plant operation and maintenance. Input points are identified in figure 11.5-1.

Volumes and isotopic inventories are discussed in subsection 11.5.4.

11.5.3 EQUIPMENT DESCRIPTION

11.5.3.1 Processing System Design

The process flow diagram for the in-plant solid waste system is shown in figure 11.5-1. The solid waste system is designed to package all solid wastes in standard drums in Unit 1 and Unit 2 for removal to disposal facilities. In addition, the system has been designed to permit bulk shipment of wastes by transfer to a disposal liner at the SDF. The solidification and dewatering facility consists of a building with shielded pits and process lines located east of the Unit 1 and Unit 2 auxiliary building. This facility is capable of receiving waste from either unit. Bulk shipment will be performed by an approved carrier and sent to a licensed waste processor or waste disposal site. Procedures used for solidification will be reviewed and approved by Southern Nuclear Operating Company (SNC).

Spent resin and evaporator concentrates may be encapsulated in containers, while solid waste such as paper, clothing, rags, towels, etc., is compressed directly into the drums. The chemical drain tank effluent is sent to the waste holdup tank for further processing. In the case of metals, wood, etc., the material will be loaded into an appropriate sized container to facilitate shipment and burial.

A. Encapsulation Process - In-Plant System

The evaporator bottoms and spent resins are transported in pipes to the drumming area.

The evaporator bottoms and spent resins are dispensed from a common manifold using six separate valves. The chemical drain tank effluent is dispensed from a single valve on the tank drain line. These valves are fail-safe, air-operated diaphragm valves.

Waste evaporator bottoms are encapsulated in 55-gal drums that are prepared in a nonradiation area separate from the drumming room.

The drums are positioned upright and an injector assembly is suspended within the drum. A vibrator which is strapped to the vertical surface of the drum is energized, and four bags of vermiculite cement are gradually poured into the drum. This mixture completely surrounds the liquid injector assembly. The drum lid is installed, and the clamping ring is secured in position. The drum is now ready for use.

Spent resin slurries are encapsulated in 55-gal drums that are prepared in a nonradiation area separate from the drumming room.

The drums are positioned upright, and a mixture of water and cement is poured into the drum until the bottom surface is covered with a 1-in. thick layer. This operation is followed by placing a 16-gauge thick, carbon steel casting sleeve in the drum and filling the annulus between the casting sleeve and the inside diameter of the drum with the water cement mixture for a height of 29 in. After the cement liner has become compact, the drum vibrator is strapped to the outside surface of the drum and then energized. A 1-in. layer of dry vermiculite cement is then poured into the bottom of the casting sleeve. A resin cage assembly, fabricated of 12-gauge thick carbon steel and resembling a DOT-2R container, is suspended inside the casting sleeve. The void between the cage and sleeve and the area above the cage, extending to the top of the drum, is filled with the dry vermiculite cement. The drum lid is then installed, and the clamping ring is secured in position. The drum is now ready for use.

B. Encapsulation Process - Portable System

The in-container cement solidification system utilizes specially designed liners. These liners are equipped with internal mixer assemblies and fit into cylindrical shielded shipping casks. Figure 11.5-2 shows a cross-sectional view of a liner in place within a cask.

In-container cement solidification can be utilized in conjunction with the disposable demineralizers and filters. Liquid radwaste streams are processed, and then the depleted processing medium is solidified or dewatered, transported, and disposed of, all in the same vessel. Resin handling operations, such as sluicing from the demineralizer to the solidification container, are minimized, thereby resulting in savings in both cost and personnel radiation exposure. In addition, using this approach of disposable demineralizers and filters offers significant waste volume reduction benefits as compared to other techniques for concentrating liquid wastes, such as evaporation.

C. Baling Process

The baling process involves the use of drums. The baler is equipped with a dust shroud to prevent the escape of radioactive particulate matter during the compaction process. This shroud is connected to the building exhaust system. After the drum has been filled with compacted wastes, it is sealed and transferred to the storage area.

The in-plant radwaste encapsulation process described in paragraph 11.5.3.1.A has not been qualified in accordance with the stability criteria of 10 CFR 61.56(b). Subsequent to the startup of the disposable demineralizer system (described in paragraph 11.2.3.1.8) in 1981 and due to the advent of high integrity containers, the need for solidification has diminished substantially. Thus, the in-plant radwaste encapsulation system is not needed and will not be used at the Farley Nuclear Plant (unless subsequently qualified).

11.5.3.2 Component Design - In-Plant System

A. Drum Vibrator Package

The drum vibrator consists of an electric-operated ball race vibrator, mounting bracket, and mounting strap. The vibrator is mounted on the drum and is used to compact the vermiculite cement inside the drum.

B. Drum

The drum consists of a DOT-17H, 55-gal drum, drum lid with 2-in. bung, lid gasket, and closing ring. After proper assembly, the drum is placed at a filling position in the drumming room, enclosed in shielding if required, and connected to process piping.

C. Vermiculite Cement Blend

The vermiculite cement is a mixture of vermiculite and Portland cement. This blend, which is both a desiccant and a solidifying medium, can either be made on the plant site or obtained from commercial suppliers.

D. Liquid Injector Assembly

The liquid injector assembly is composed of standard, commercially available components and can be assembled on the plant site. The disposal item distributes the liquid waste evaporator bottoms with the vermiculite cement in the drum.

E. Drum Package Tool and Fixture

The drum package tool and fixtures consist of separate metal and wooden support and positioning devices. The wooden fixture holds either the resin cage assembly or the liquid injector in position inside the drum, while the vermiculite cement is being added to the drum. The metal positioning device maintains the position of the injector or resin cage in the prepackaged drum during transit from the prepackaging area to the drumming room filling position.

F. Resin Cage Assembly

The resin cage assembly, which resembles a DOT-2R container, can be fabricated at the plant site. This item aids in removing water from the spent resins and is encapsulated within the solidified cement vermiculite mixture in the drum.

G. Impact Wrench Package

The impact wrench is an air-operated impact wrench with a deep socket. This wrench is used primarily to obtain adequate and uniform application of torque to the closing ring bolt on drum closing.

H. Drum Shields

The drum shield consists of a two-piece, cylindrical lead shield, jacketed with steel. The shield provides radiological protection for operating personnel during the injection of waste liquid into the drums and during subsequent in-plant handling of the drums. Under the maximum radioactivity loading, the calculated surface contact reading is 10 mR/h.

I. Piping Header Assembly

The piping header assembly consists of flexible hose with drum filling header, a liquid level device (vacuum switch), and a gauge. This assembly directs liquid flow to the packages. It automatically controls the volume of liquid added to the packages and can be used whether or not the shield is in place.

J. Drum Header Installation and Removal Tool

The drum header installation and removal tool is of steel fabrication and is used to install and remove the drum filling header into or out of the drum packages.

K. Vacuum Pump Package

The vacuum pump package consists of a commercially available vacuum pump with motor, filter, manifold, vent valve, suction hose, gauge, and service cart. This pump evacuates the drum package prior to the filling operation.

L. Resin Cage Plug Installation Tool

The resin cage plug installation tool is a magnetic socket with a long extension; it also includes a shielding adapter. This tool permits an operator to install a plug in the top of the resin cage assembly which is enclosed by a shield. The shielding adapter provides radiological protection for the operator's hands.

M. Drum Plug Installation Tool

The drum plug installation tool consists of a magnetic socket with a long extension and also includes a shielding adapter. This tool permits an operator to install a plug in the 2-in. bung of the drum lid which is enclosed by a shield. The shielding adapter provides radiological protection for the operator's hands.

N. Drum Lifting Grab

The drum lifting grab is a lifting device with a 5-ton capacity. It is capable of both automatic and remote operation. It can also be used for other nonremote operations involving the handling of the drums.

O. Shield Spreader Bar

The load beam or spreader bar has a 6-ton capacity. The spreader bar is used, under normal conditions, to maneuver the shield assembly which may contain the drum. The design of the shield spreader bar is shown in figure 11.5-3.

P. Drumming Station Control Panel

The drumming station control panel is approximately 42 in. long by 42 in. wide by 16 in. deep and is used for all electrical controls. It has a mounted board for indicating instruments and a wall-mounted cabinet with a piano-hinged front door for access to the cabinet interior. The panel contains the switches that control the operation of the spent resin storage tank and the filling valves. It also contains indicators for open or closed positions of the valves. The arrangement of the drumming station control panel is shown in figure 11.5-4.

11.5.3.3 <u>Operation of Equipment</u>

A. Encapsulation - In-Plant System

The prepared drums are placed in the drumming room at the filling positions and connected to the dispensing manifolds and the vacuum pump. If required, the drums are enclosed in shields. Manifold arrangement and drum area layout are shown in figure 11.5-5 and figures 1.2-1 and 1.2-2, respectively.

Six 55-gal drums can be filled simultaneously with evaporator bottoms or spent resin slurries.

The vacuum pump is energized and the individual drums are evacuated. During the processing of waste evaporator concentrates or chemical tank effluent, the drums are evacuated to a 23-in. Hg vacuum. If spent resins are being processed, the drums are evacuated to a 21-in. Hg vacuum.

After each drum is evacuated, the vacuum pump is disconnected and the required capping performed. Following an interval of 5 min, the vacuum in each drum is recorded. The vacuum is then inspected at 2-min intervals for a minimum of three readings. If the loss in vacuum does not exceed 0.1-in. Hg V-ac/min and the package vacuum level is within limits, the drums are ready to be filled. The remaining encapsulation process is remotely controlled from the panel located outside the drumming area.

When the filling operation is complete, the vacuum switch opens and thus closes the dispensing valve to terminate the filling.

The pressure used to transfer the wastes to the manifold is released in the following manner: The waste evaporator bottom transferring operation is stopped by turning the transfer pump switch to the off position. The spent resin pressure is released by venting the spent resin storage tank.

The major advantages of the vacuum filling technique is that radioactive particulate or gaseous matter is not released to the work area environment.

After transfer of wastes and before the operator enters the drumming area, the filling hoses are flushed to the drums. The flush water is provided by the makeup water system and is manually operated by valves located in a shielded area of the drumming room. The system is so designed that all interconnecting lines between the storage drum, dispensing valve, and manifold are flushed back to the spent resin storage tank or to the remaining spaces in the individual drums. The waste in the chemical drain tank effluent valve and drain piping can also be flushed back into the chemical drain tank. A radiation detector, located in the drumming area with a readout instrument on the control panel, enables the operator to determine remotely the radiation levels in the drumming area. Throughout these operations, additional radiation surveys can be made by using portable survey equipment.

The header assemblies are removed from each drum. Using the shielded tools provided, a plug is inserted into the resin cage guide tube and tightened securely. This is followed by the insertion of a 2-in. bung closure and shielded plug. After the plugs are installed, the drum with shield is transferred from the drumming areas to the shield stripping area, using the overhead crane and spreader bar. To remove the shield from the drum, the operator releases the latching mechanism attaching the shield cylinder to the shield base. Following this operation, the operator moves to the observation area where succeeding operations may be performed remotely. Using the crane, the shields are stripped from the drums.

B. Encapsulation - Portable System

A typical configuration of the in-container cement solidification system is shown in figure 11.5-2. The design and operational features of the system, when utilized to solidify resins, are as follows:

1. The solidification liner is in place within the shielded shipping cask or shielded pit. The cask cover is in place, and the shield plug in the center of the cask cover is removed or the liner is in a shielded pit in the solidification facility.

2. The drive and fill-flange assembly is bolted to the liner opening, forming a sealed closure. The mixer drive unit is an integral part of the drive and fill-flange assembly, and it remains in place during the sluicing and dewatering operations. The mixer drive mates with the mixer shaft in the solidification liner.

3. Vent and overflow lines are used to prevent overpressurization of the liner and overflow into the cask, respectively. The vent line is directed to a filter/dust collector.

4. If necessary, resins are slurried into the liner through the waste inlet connection in the drive and fill-flange assembly.

5. Suction is maintained on the dewatering line, so that the slurry is dewatered during the sluicing operation unless material is already in a disposable liner.

6. High levels within the liner are monitored during the sluicing operation using a contact level probe.

7. Dewatering is continued after sluicing is completed in order to remove as much water as possible from the resin (as necessary).

8. Upon completion of dewatering, the resin inlet line is disconnected and the cement fill line inlet valve is opened.

9. Water is added in specified amounts, and the liner internal mixer is rotated using the drive unit. Cement is added, in amounts dictated by the process control plan, in order to produce the desired mixture.

10. Mixing is continued until the torque required to rotate the mixer increases, indicating setup of the mixture.

11. The drive and fill-flange assembly is removed, the solidification liner is capped, and if cask is used, the shipping cask shield plug is reinstalled. The cask is then prepared for shipment. If pit is used after solidification, the container is transferred into the shipping cask.

C. Baler Operation

The baler is used to compress low radiation level solid wastes into drums. The drum is placed in the baler, solid wastes are inserted in the open drum, and the shroud door is closed. The drum is automatically positioned to be coaxial with the baler ram. An operator initiates the compaction process by positioning an up/down switch in the down position, thus energizing the hydraulic pump motor. The hydraulic pressure forces the ram down into the drum, thereby compressing the wastes. The shroud door is opened, and additional wastes are added to the drum. The cycle is repeated until the drum is full and the lid is installed, the clamping ring tightened, and the drum stored pending shipment. An anti-spring hood device may be used to help keep compressed materials in the barrels during compaction.

The shroud is ducted to the plant ventilation system to remove dust or particles that may be emitted from the drum during compression of the wastes.

11.5.4 EXPECTED VOLUMES

[HISTORICAL]

[Table 11.5-1 gives the total solid waste generated per year through 1986. The associated Ci content of solid waste processed from Farley Nuclear Plant (FNP) to date is also given.

The principal nuclides shipped from the plant site include the following:

Chromium-51	*Molybdenum-99*
Manganese-54	*Iodine-131*
Manganese-56	*Cesium-134*
Iron-59	*Cesium-136*
Cobalt-58	*Cesium-137*
Cobalt-60	*Iron-55]*

The total solid waste generation at FNP is documented/reported in the Annual Radiological Effluent Report as required by the Technical Specifications.

11.5.5 PACKAGING

The solid waste system product is shipped to the appropriate burial ground facility using the necessary package, as required by Department of Transportation (DOT) regulations. These packages include but are not limited to: strong, tight containers; type A containers; type B containers; and large quantity containers.

11.5.6 STORAGE FACILITIES

Solid radwastes, dewatered resins, and waste oil are contained in their appropriate packages and stored at the following designated locations:

A. Low-level radwaste storage building (outside of auxiliary building). The low-level radwaste storage building (LLRB) has been constructed to supplement the plant's storage capabilities for equipment and some types of radwaste. This building will be utilized as the primary storage point for dry active waste as well as the loading facility for shipments of dry active waste. Storage containers normally will be transported to and placed in the storage building by means of a fork lift. The LLRB will utilize a first-in, first-out method of inventory control, when practical, since the building is designed to accommodate temporary material storage of up to a 4-year duration. Other waste forms may be stored in the LLRB as needed should primary storage locations not be available. Contaminated equipment and materials may be stored in the LLRB provided that suitable contamination control practices are implemented.

B. Drumming rooms (rooms 2420 and 2421 at el 155 ft in the Unit 2 auxiliary building).

C. New fuel storage rooms (Unit 1 auxiliary building room 459 and Unit 2 auxiliary building room 2459 at el 155 ft).

D. Drum storage area (Unit 1 auxiliary building room 603 and Unit 2 auxiliary building room 2603 at el 130 ft).

E. Nuclear laundry rooms (rooms 420 and 421 at el 155 in the Unit 1 auxiliary building).

F. Solidification dewatering facility oil and paint storage room.

G. Temporary storage during and after filling prior to transport to the LLRB is normally in the east end of Unit 1 or Unit 2 155-ft hallway.

H. Temporary storage of waste in process inside the solidification and dewatering facility.

I. Temporary storage of disposable demineralizer liners, oil drums, or other wastes may be in suitable containers outside the solidification and dewatering facility until shipment. This storage time will normally be kept to a minimum.

J. Low-level contaminated construction materials in suitable containers may be temporarily staged in the Complex III warehouse until shipment or movement to other storage locations.

11.5.7 SHIPMENT

Under normal operating conditions, solid radioactive wastes will be shipped from the site by carriers licensed to perform offsite disposal of such wastes. The normal mode of shipment will be by truck in containers which meet DOT standards. Facilities are available for shipment by rail, if necessary. Typically, the shipper will supply any special DOT-approved casks that he may have available for the shipment of radioactive material with an unusual configuration, with a high Ci content, and/or with a high radiation level. The shipper will be contracted to pick up waste as required.

The containers are transferred from the storage area to the shipper's vehicle by mechanical means. Large liners, including disposable demineralizers, will be handled by the spent fuel cask crane or by appropriate mobile cranes. Loading will be performed by SNC and/or contractor personnel and the vehicle will be assigned for sole use of SNC. Once the waste is loaded and secured on the truck, surveys are made to ensure compliance with 49 CFR 170 through 199, specifically, if shipped as "Radioactive - LSA" material:

A. The dose rate will be less than 2 mrem/h in the normally occupied areas of the cab, less than 10 mrem/h at 2 meters from the vehicle vertical surfaces, and less than 200 mrem/h at any point on the surface of the vehicle.

B. The vehicle shall smear less than 2200 dpm/100 cm2 beta gamma and less than 220 dpm/100 cm2 alpha.

C. Containers will be labeled or marked as required, and shipping papers that describe the hazardous materials that are being transported will be provided to the carrier.

Although it will not be a normal procedure to store full or partially loaded shipment vehicles on the site for extended periods of time, if the occasion should arise, the vehicles will normally be parked inside the owner controlled fenced area. Normally, drummed waste will not be stored in any areas other than those mentioned in subsection 11.5.6. Some exceptions to this are when drums are stored for super compactor support or if plant management gives permission to store in other areas due to unforeseen circumstances.

[HISTORICAL]
[TABLE 11.5-1 (SHEET 1 OF 3)]

ACTUAL TOTAL SOLID WASTE GENERATED PER UNIT AT FNP

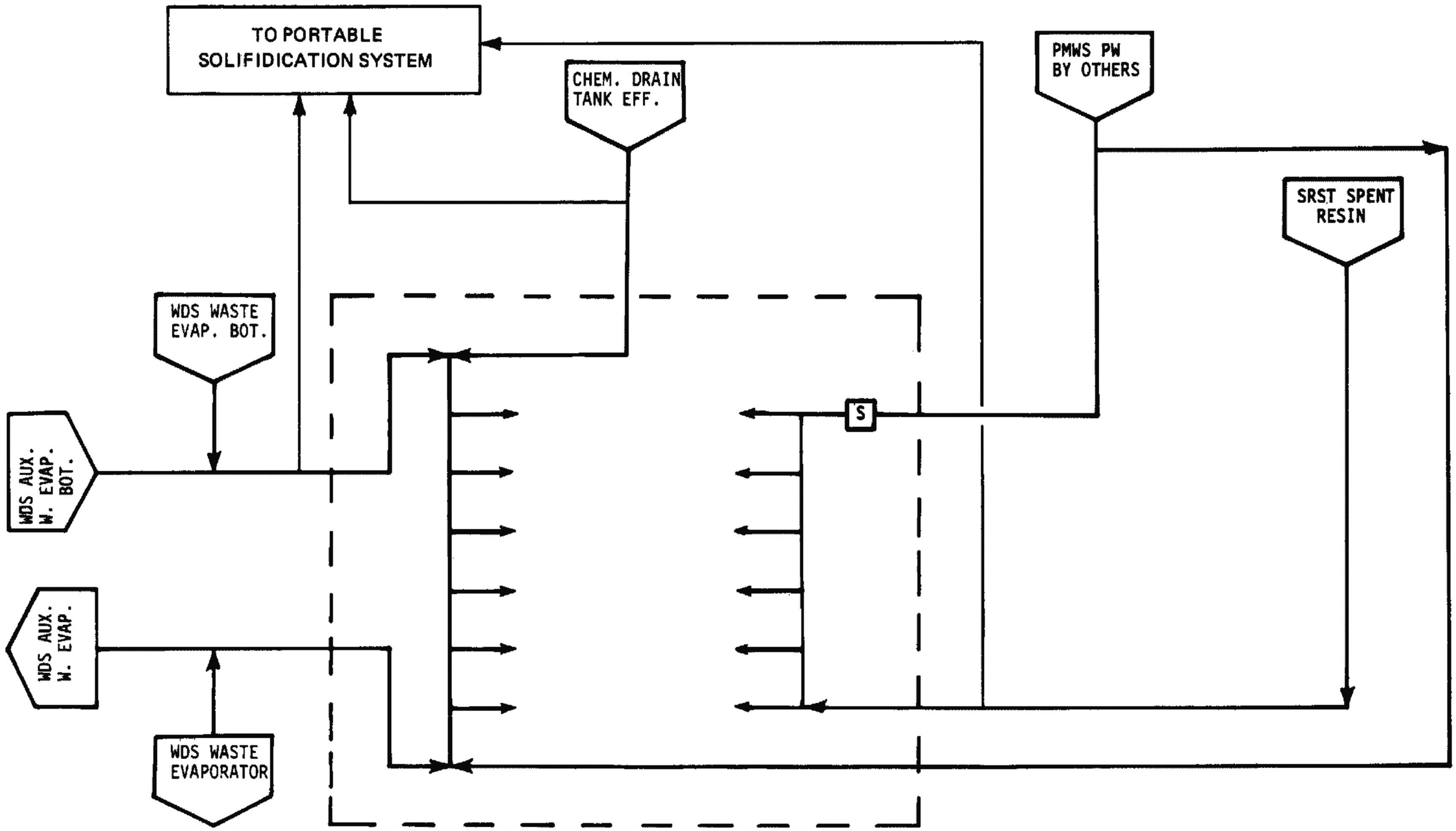
Period	*Total Solid Volume (m^3)*	*Total Ci*
1977, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*221*	*3.90*
Dry compressible waste	*0*	*0*
1978, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*68*	*0.117*
Dry compressible waste	*0*	*0*
1978, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*110*	*3.40*
Dry compressible waste	*91*	*2.20*
1979, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*340*	*150*
Dry compressible waste	*200*	*30*
1979, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*260*	*390*
Dry compressible waste	*310*	*20*
1980, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*88*	*180*
Dry compressible waste	*140*	*340*
1980, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*73*	*9.8*
Dry compressible waste	*140*	*1.8*

TABLE 11.5-1 (SHEET 2 OF 3)

Period	*Total Solid Volume (m^3)*	*Total Ci*
1981, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*133*	*589*
Dry active waste	*226*	*13.1*
1981, 3rd and 4th quarters		
Spent resins and filter sludges	*16.4*	*115*
Dry active waste	*189*	*2.47*
1982, 1st and 2nd quarters		
Spent resins and filter sludges	*19.3*	*25.1*
Dry active waste	*183*	*14.4*
1982, 3rd and 4th quarters		
Spent resins and filter sludges	*9.63*	*3.23*
Dry active waste	*163.2*	*60.2*
1983, 1st and 2nd quarters		
Spent resins and filter sludges	*33.13*	*788.04*
Dry active waste	*190.28*	*22.33*
1983, 3rd and 4th quarters		
Spent resins and filter sludges	*22.2*	*213.2*
Dry active waste	*214.8*	*20.1*
Irradiated components, control rods, etc.	*1.0*	*5.1*
1984, 1st and 2nd quarters		
Spent resins and filter sludges	*20.02*	*113.8*
Dry active waste	*288.4*	*12.09*

TABLE 11.5-1 (SHEET 3 OF 3)

Period	*Total Solid Volume (m^3)*	*Total Ci*
1984, 3rd and 4th quarters		
Spent resins and filter sludges	*13.93*	*153.65*
Dry active waste	*187.0*	*15.2*
1985, 1st and 2nd quarters		
Spent resins and filter sludges	*35.68*	*27.47*
Dry active waste	*277.0*	*14.85*
1985, 3rd and 4th quarters		
Spent resins and filter sludges	*35.39*	*773.5*
Dry active waste	*129.81*	*5.50*
Other (absorbed oil)	*17.20*	*0.0022*
1986, 1st and 2nd quarters		
Spent resins and filter sludges	*16.1*	*306.0*
Dry active waste	*60.1*	*0.201*
1986, 3rd and 4th quarters		
Spent resins and filter sludges	*26.03*	*920.3*
Dry active waste	*141.3*	*1.77*
Irradiated components (control rods, etc.)	*1.37*	*570.0]*



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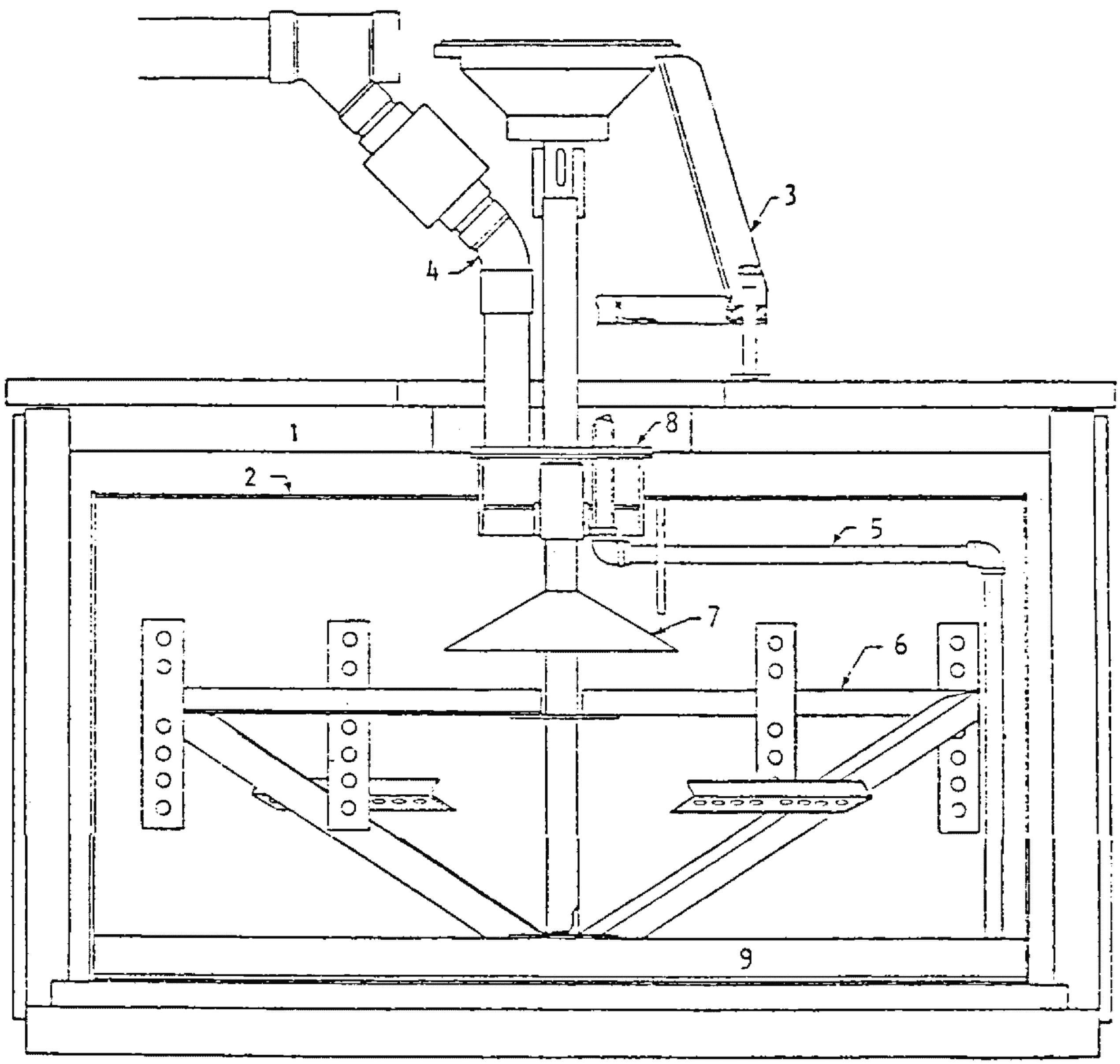
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAM
SOLID WASTE PROCESSING SYSTEM

FIGURE 11.5-1

1 CASK
2 LINER
3 DRIVE FRAME AND ASSEMBLY
4 CEMENT FEEDLINE
5 DEWATERING LINE
6 MIXER ASSEMBLY
7 DEFLECTOR PLATE
8 ACCESS CLOSURE PLATE
9 FILTER ELEMENTS



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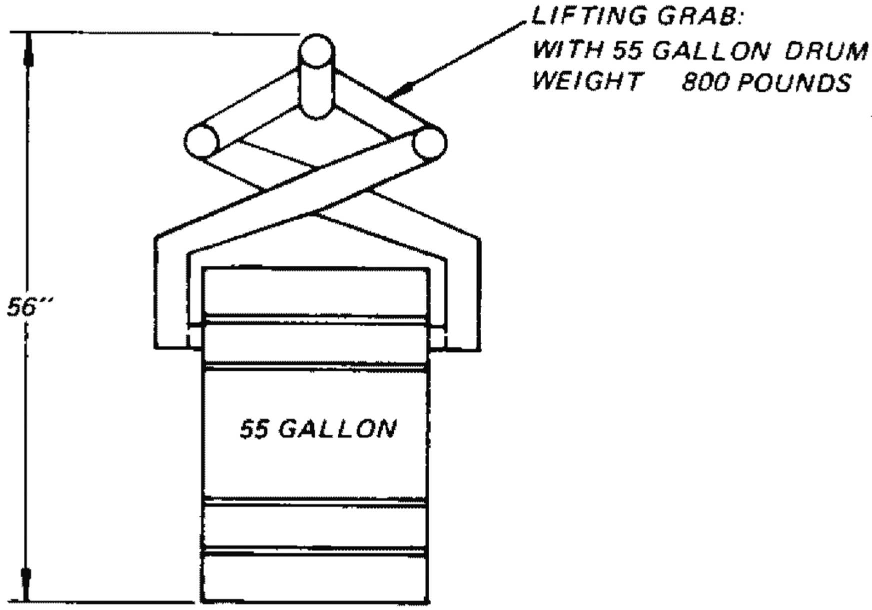
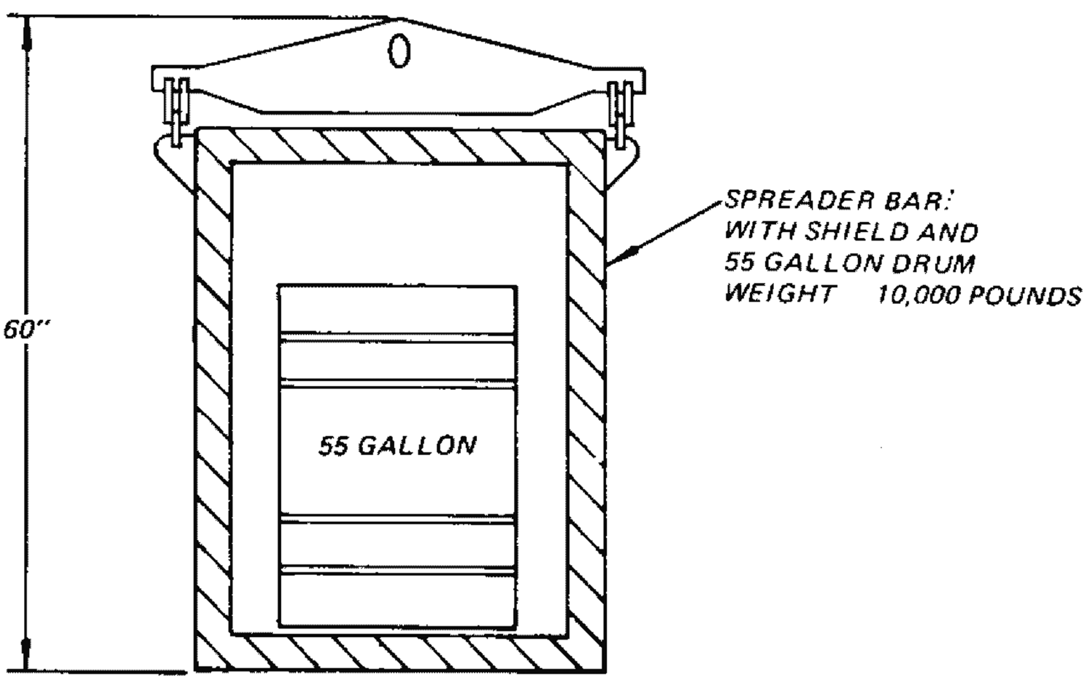
UNIT 1 AND UNIT 2

PORTABLE IN-CONTAINER CEMENT
SOLIDIFICATION SYSTEM

FIGURE 11.5-2





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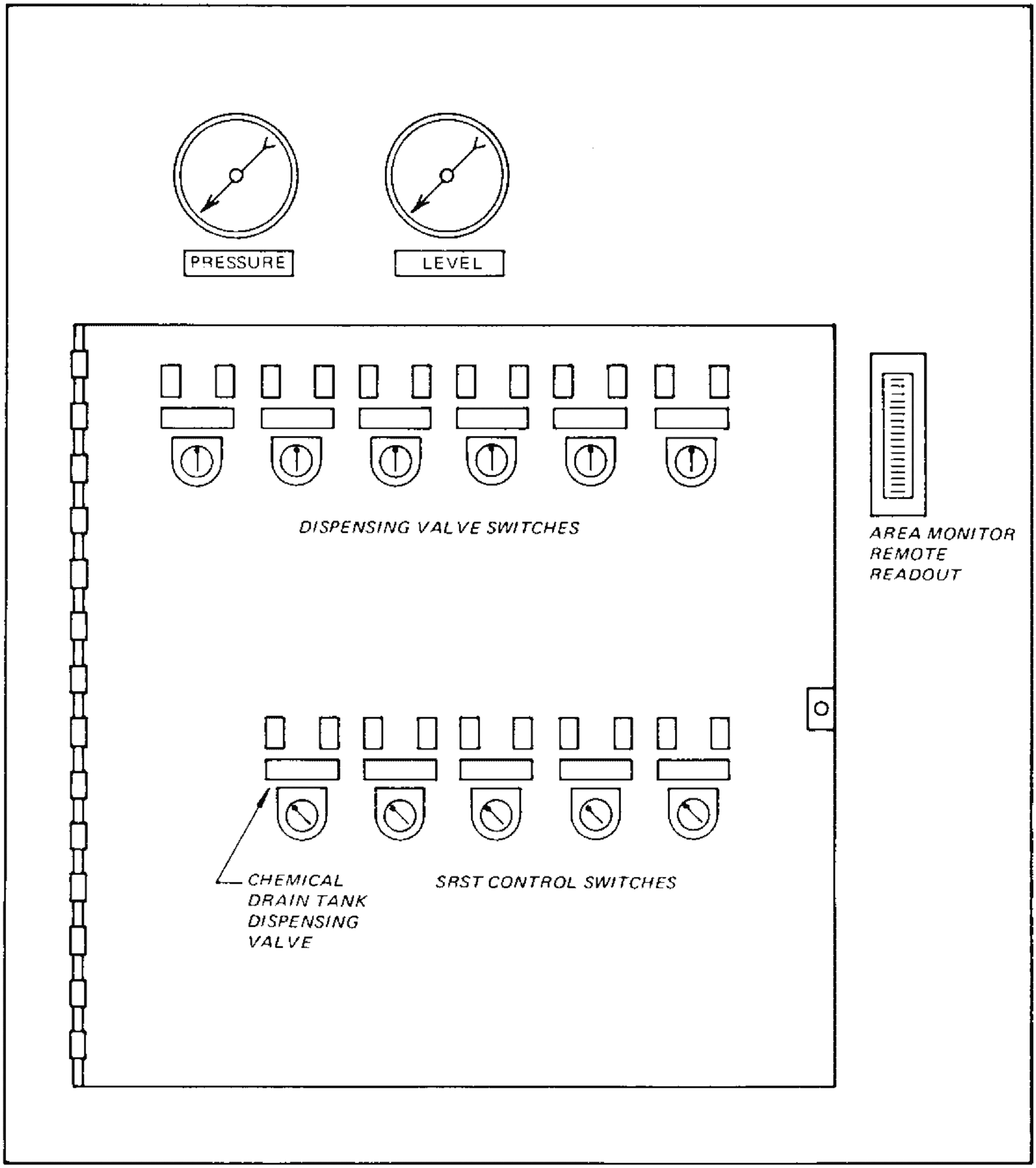
DRUM HANDLING ASSEMBLIES

FIGURE 11.5-3



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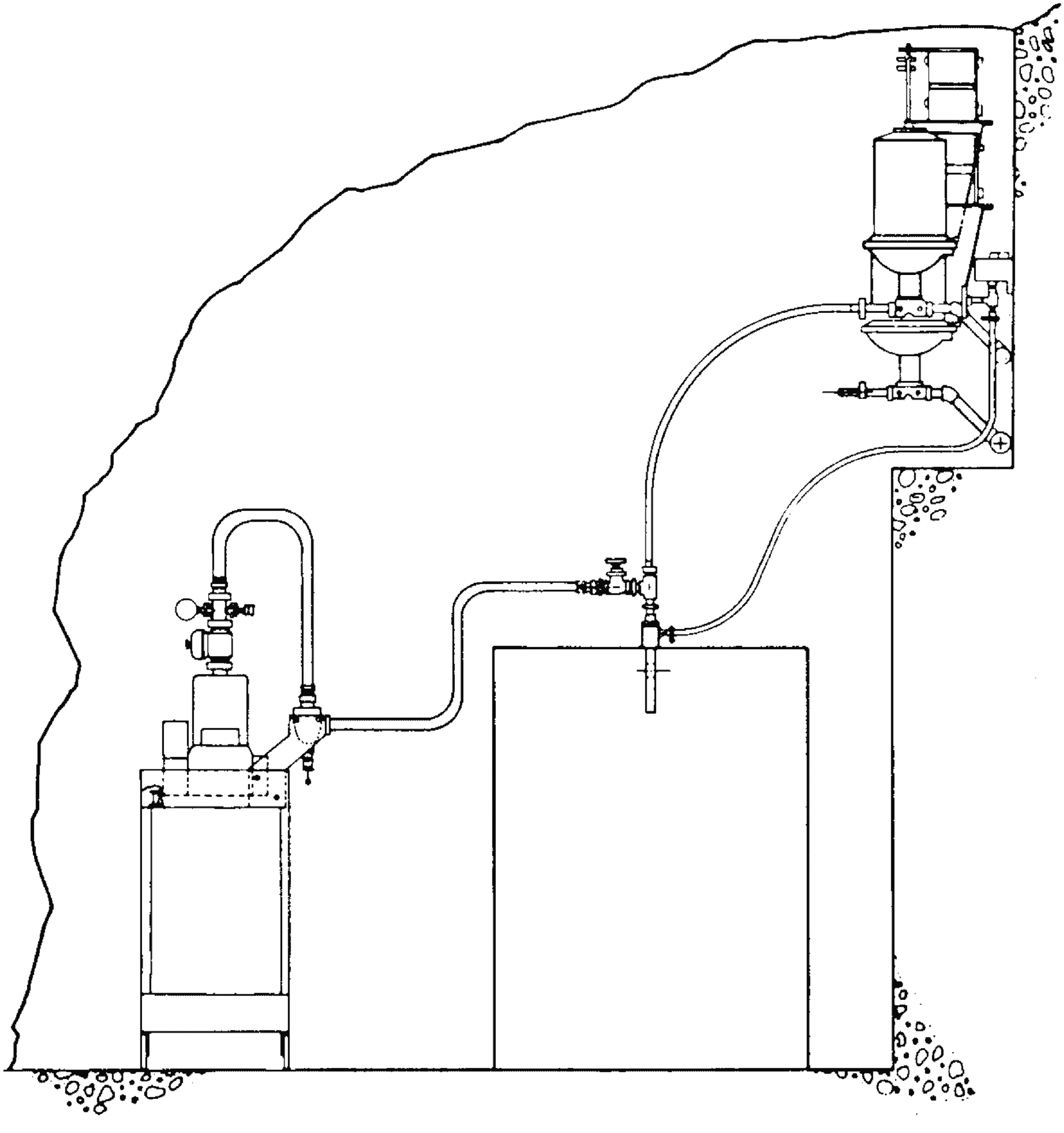
DRUMMING ROOM CONTROL PANEL

FIGURE 11.5-4



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UNIT 1 AND UNIT 2

SHIELDED DRUM PACKAGE INSTALLED
AND READY FOR EVACUATION

FIGURE 11.5-5

11.6 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Radiological Environmental Monitoring Program (REMP) is described in Chapter 4 of the Offsite Dose Calculation Manual (ODCM).

APPENDIX 11A TRITIUM CONTROL

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APPENDIX 11A

TRITIUM CONTROL

The release of tritium to the environment from operating Westinghouse pressurized water reactors (PWRs) has always been well below 10 CFR 20.1 - 20.601 limits. This section discusses the reduced tritium production in the plant as a result of employing zircaloy-clad fuel and silver-indium-cadmium control rods.

11A.1 SYSTEM SOURCES

There are two principal contributors to tritium production within the PWR system: the ternary fission source and the dissolved boron in the reactor coolant. Additional small contributions are made by Li-6, Li-7, and deuterium in the reactor water. Tritium production from different sources is shown in table 11A-1.

11A.1.1 THE FISSION SOURCE

This tritium is formed within the fuel material and may do one of the following:

A. Remain in the fuel rod uranium matrix.

B. Diffuse into the cladding and become hydrated and fixed there.

C. Diffuse through the clad for release into the primary coolant.

D. Release to the coolant through macroscopic cracks or failures in the fuel cladding.

Previous Westinghouse design has conservatively assumed that the ratio of fission tritium released into the coolant to the total fission tritium formed was approximately 0.30 for zircaloy-clad fuel. The operating experience at the R. E. Ginna Plant of the Rochester Gas and Electric Company, and at other operating reactors using zircaloy-clad fuel, has shown that the tritium release through the zircaloy fuel cladding is substantially less than earlier estimates predicted. Consequently, the release fraction may be revised downward from 30 percent to 10 percent based on this data.[1]

11A.1.2 CONTROL ROD SOURCE

The full and part length rods for this plant are of silver-indium-cadmium. There are no reactions in these absorber materials which would produce tritium, thus eliminating any contribution from this source.

11A.1.3 BORIC ACID SOURCE

A direct contribution to the reactor coolant tritium concentration is made by neutron reaction with the boron in solution. The concentration of boric acid varies with core life and load follow, so that this is a steady decreasing source during core life. The principal boron reactions are the B-10(n, 2α)H-3 and B-10(n, α)Li-7(n, nα)H-3 reactions. The Li-7 reaction is controlled by limiting the overall lithium concentration to approximately 2 ppm during operation. Li-6 is essentially excluded from the system by utilizing 99.9 percent Li-7.

11A.1.4 BURNABLE SHIM ROD SOURCE

These rods are in the core only during the first operating cycle and their tritium contribution is potential only during this period.

11A.1.5 LITHIUM AND DEUTERIUM

Lithium and deuterium reactions contribute only minor quantities to the tritium inventory, as shown in table 11A-1.

11A.2 DESIGN BASES

The design intent is to reduce the tritium sources in the reactor coolant system to a practical minimum in order to permit longer retention of the reactor coolant within the plant. Reduction of source terms is provided by utilizing silver-indium-cadmium control rods and the determination that the quantity of tritium released from the fuel rods with zircaloy cladding is less than originally expected.

11A.3 DESIGN EVALUATION

Table 11A-1 is a comparison of a typical design basis tritium production, which has been utilized in the past to establish system and operational requirements of the plant and present expected values.(1) It is noted that there are two principal contributors to the tritium production: ternary fission source and the dissolved boron in the reactor coolant. Of these sources it is noted that the 30 percent release of ternary fission through the cladding was the predominant contributor in past design considerations.

Because of the importance of this source on the operation of the plant, Westinghouse has been closely following operating plant data. Table 11A-2 represents tritium releases during one calendar year for different Westinghouse PWR plants. Further, a program is being conducted at the R. E. Ginna Plant to follow this in detail. The R. E. Ginna Plant has a zircaloy-clad core with silver-indium-cadmium control rods. The operating levels of boron concentration during the startup of the plant are approximately 1100 to 1200 ppm of boron. In addition, burnable poison rods in the core contain boron which will contribute some tritium to the coolant, but only during the first cycle. Data during the operation of the plant have indicated very clearly that the present design sources were indeed conservative. The tritium released is essentially from the boron dissolved in the coolant and a ternary fission source which is less than 10 percent. In addition to this data, other operating plants with zircaloy-clad cores have also reported very low tritium concentrations in the reactor coolant system after considerably longer operation.

For a leakage from the primary coolant system into the containment of 40 lb/day, with an assumed tritium concentration in the coolant of 2.5 $\mu Ci/cm^3$ (no containment ventilation purge), the tritium concentration in the atmosphere of the containment would be low enough to permit access without protective equipment by plant maintenance personnel for an average of 2 h/week.

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. Abnormal leakages in excess of the design estimate have occurred in operating plants. The leaking components have been identified and corrective measures have been taken. For example, bellows sealed valves, diaphragm sealed valves, and pump seal purge systems have been employed.

The total activity that would be released from the containment purge during refueling operations would range in the order of 20 to 40 Ci of tritium, depending on the core cycle, relative humidity, etc. It is not proposed that this amount of activity from evaporative losses be collected but that it be discharged from the plant. Similarly, any radioactive gases in the containment would be discharged. Evaporation of tritium from the refueling pool has been considered in evaluating the consequences to tritium on both operators and environmental releases. This indicates maximum tritium concentration in the containment consistent with 40 h/week occupancy and total tritium release of about 30 Ci/refueling.

The tritium source terms in the reactor coolant are at a low level (approximately 1110 Ci/cycle) such that it is possible to discharge tritium in amounts to preclude in-plant exposure problems without exceeding the "as low as practicable" design objective. Alternatively, without any intentional removal of tritiated water:

A. Tritium levels should not cause a problem during refueling through the 40-year original operating license term or the 60-year operating life resulting from the renewed licenses. Assuming no change in tritium production or in system leakage, the plant activities (and consequently, releases) would increase only 9% for the additional 20-year period of extended operation due to the short half-life. Considering system leakage, the actual increase is even less.

B. Special procedures (purging, etc.) prior to containment access may be required.

Credit is taken for dilution of the reactor coolant system water by refueling water and spent-fuel pool. Discharge of the tritiated water from the plant is therefore possible at extended intervals or, if discharged on a regular basis, would below the 10 CFR 20.1 - 20.601 limits because of the reduced production rates.

Based on the above, the following conclusions have been reached:

A. The tritium levels in plants operating with zircaloy-clad cores will be substantially lower than previous design predictions.

B. The tritium source in the plants will be reduced by utilizing silver-indium-cadmium control rods.

C. Containment access during power operation and refueling with continued storage of the tritium in the plant is possible with the application of special procedures (purging, etc.) prior to containment access.

D. The containment tritium purge is relatively small compared to the total available and will be discharged.

REFERENCE

1. Westinghouse Electric Corporation, "Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, Revision 1, July 1975.

TABLE 11A-1

TYPICAL DESIGN BASIS TRITIUM PRODUCTION

Tritium Source	Total Produced (Ci/year)	Release Expected to Reactor Coolant (Ci/year)
Ternary fission	8160	816
Burnable poison rods		
Initial cycle	599	60
Soluble boron		
Initial cycle	152	152
Equilibrium cycle	217	217
Lithium and deuterium reactions	82	82
Total initial cycle	8993	1110
Total equilibrium cycle	8459	1120

Basis

Power level, core thermal power (Mwt)	2766
Load factor	0.8
Release fraction from fuel (percent)	10
Release fraction from burnable poison rods (percent)	10
Burnable poison rod B-10 mass (g)	2374
Reactor coolant boron concentration, initial cycle (ppm)	700
Reactor coolant boron concentration, equilibrium cycle (ppm)	1000

TABLE 11A-2

TRITIUM RELEASES FOR 1971 FROM WESTINGHOUSE-DESIGNED OPERATING REACTORS

Plant	Type of Cladding	Total Released (Ci)	Average Discharge Concentration ($\mu Ci/cm^3$)	Fraction Column 2, Table II, Appendix B, 10 CFR 20.1-20.601 (3×10^{-3} $\mu Ci/cm^3$)
Yankee Rowe	Stainless steel	1633	5.9×10^{-6}	2.0×10^{-3}
Connecticut Yankee	Stainless steel	5830	7.7×10^{-6}	2.6×10^{-3}
San Onofre	Stainless steel	4570	6.7×10^{-6}	2.4×10^{-3}
Robert E. Ginna	Zircaloy	154	2.3×10^{-7}	7.7×10^{-5}
H. B. Robinson Unit 2	Zircaloy	118	1.7×10^{-7}	6.0×10^{-5}
Point Beach Unit 1	Zircaloy	266	4.7×10^{-7}	1.6×10^{-4}

APPENDIX 11B

MATHEMATICAL MODEL OF RADIOACTIVITY DOSES

[THIS SECTION INTENTIONALLY DELETED]

12.0 RADIATION PROTECTION

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12.0 RADIATION PROTECTION

12.1 SHIELDING

12.1.1 DESIGN OBJECTIVE

The primary objective of the shielding design and access control is to protect operating personnel and the general public from potential radiation sources in the reactor, the radwaste system, and other auxiliary systems including associated equipment and piping.

Shielding is designed to perform the following functions:

A. Limit the dose to plant personnel, construction workers, vendors, and visitors during normal operation, including anticipated operational occurrences, to within a few percent of the guidelines of 10 CFR 20.1 - 20.601.

B. Limit the dose to plant personnel, in the unlikely event of an accident, to within the requirements of 10 CFR 50.67, to permit termination of accident conditions without undue risk to the general public.

C. Limit dose to certain components in high radiation areas and very high radiation areas within specified radiation tolerances.

D. Protect certain components to prevent excessive neutron activation and facilitate access.

E. Limit dose to persons at the boundary of the restricted area to a small fraction of the guidelines of 10 CFR 20.1 - 20.601 due to direct radiation during normal operation.

The following guides are used in shield design to achieve the above objectives. All plant areas are divided into zones according to the dose rates given below. These zones are for planning purposes; actual dose rates will be determined by surveys. As documented in NUREG-75/034, dated May 2, 1975, appropriate design features recommended by Regulatory Guide 8.8 (NRC acceptance criteria) have been included to maintain radiation exposures ALARA.

Zone Designation	Dose Rate (mrem/h)
I-A	≤ 0.2
I	≤ 0.5
II	≤ 2.5
III	≤ 15.0
IV-A	≤ 25.0
IV	≤ 100.0
V	>100.0

A. Access control and shielding design are considered according to the above guidelines in determining optimum plant layout that will allow personnel to perform their normal functions, based on required stay times to perform these functions, with the minimum of exposure.

B. All pipes and ducts penetrating the primary and secondary shields are located in positions so that a direct radiation shine from high radiation sources such as the reactor vessel and components of the reactor coolant loops is avoided.

Penetrations from all pipes and ducts through the shield walls are located to avoid a direct line of sight with the radiation source to prevent streaming into lower radiation zones. Grouting materials have been used to fill voids between the penetration and the wall where necessary.

C. Shield discontinuities include concrete hatch covers, shielding doors, and access labyrinths. To reduce radiation streaming through gaps between the main shield and a removable section, offsets have been used and the gaps are not in line of sight of the radiation source where this is feasible. Access labyrinths into rooms containing radiation sources such as gas decay tanks, coolant sampling equipment, evaporators, and filters have been designed to eliminate a direct shine through the offset passage to the accessible areas.

D. Radioactive piping is routed to minimize exposure to plant personnel. This is accomplished by:

1. Minimizing radioactive pipe routing through the corridors and low radiation zones.

2. Using shielded pipe trenches when the above method is not feasible.

3. Separating the locations of radioactive and nonradioactive pipes.

E. Motor-operated or diaphragm valves are used whenever feasible. Provision is made for drainage of associated equipment to minimize radioactive exposure during valve maintenance. In case of manual valves, provision is made to protect the operator from the radioactive valve by use of shield walls and valve stem extensions (reach rods).

F. Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions so that the personnel do not receive radiation exposures greater than 5 rem whole body or its equivalent to any part of the body for the duration of the accident, in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

   The analysis of the control room doses, including ingress/egress, is covered in detail in chapter 15.

G. The principal shield material is concrete of a density of 145 lb/ft^3. Other miscellaneous material like steel, lead, high density concrete, or water are occasionally used as required.

H. The design shields major sources and activated components to allow access and provide adequate protection for inservice inspection.

I. Radioactive source data are based on full load plant operation with the equivalent of 1-percent fuel cladding defects.

12.1.2 DESIGN DESCRIPTION

12.1.2.1 General Description

Detailed drawings showing the layouts and cross-sections of buildings that contain process equipment for treatment of radioactive fluids are shown in figures 1.2-1 through 1.2-9. Drawing D-170084 is a detailed plot plan of the total plant layout within the site boundary, showing all outside storage areas and the location of the railroad siding.

Scaled isometric views and a layout drawing of the control room are illustrated in figures 1.2-1 and 12.1-1.

The radiation monitoring system functional block diagram is shown in drawing U-167647. The shield wall thickness, occupancy times, and maximum possible dose rates are shown in the radiation zones and controlled access diagrams (drawings D-176035, D-176036, D-176037, D-176038, D-176039, D-176040, D-176041, D-176042, D-176043, D-206035, D-206036, D-206037, D-206038, D-206039, D-206040, D-206041, D-206042, and D-206043).

Concrete radiation shields are designed to American National Standards Institute N101.6-1972, as modified by Regulatory Guide 1.69.

12.1.2.2 Justification for Shield Design - Physical and Mathematical Models

Shield design has been based on normal operations with 1-percent failed fuel or TID 14844 accident releases where applicable. Models for calculating source strengths of critical equipment and systems are discussed in subsection 12.1.3. The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, demineralizers, evaporators, and the containment is a finite cylindrical volume source shield and an infinite shielded cylinder in case of piping.

The mathematical models are based on formulations in "The Engineering Compendium of Radiation Shielding" and the "Reactor Shielding Design Manual." Nuclear data are derived from the "Table of Isotopes," "Reactor Physics Constants," ANL-5800, and XDC-59-8-179. Sources involving different isotopes are divided into different energy bins, corresponding to the gamma energies. The dose contribution from individual sources is calculated based on the above described model. The total dose to the receptor is taken as the sum of doses from each source During shutdown, in cases of corrosion products deposited on surfaces such as a pipe, the latter is treated as a cylindrical surface source.

The radiation shielding in the various plant buildings is described in the following paragraphs.

12.1.2.3 Containment

The containment shield is composed of a reinforced, prestressed, posttensioned, steel line concrete containment that completely surrounds the nuclear steam supply system (NSSS). This shield, together with the primary and secondary shields, reduces the radiation levels for accessibility outside the containment to 0.5 mrem/h. In case of an accident, the shielding will minimize the station doses to less than 5 rem whole body and the offsite doses to less than 0.5 mrem/h.

12.1.2.4 Primary Shield

The primary shield of 6-ft-thick reinforced concrete surrounds the reactor vessel. The cavity between the primary shield and the reactor vessel is air cooled to prevent overheating, dehydration, and degradation of the shielding properties of the concrete. The primary shield, in conjunction with the secondary shield, serves to attenuate the radiation from the reactor vessel and reactor coolant equipment. It permits limited access in the containment during normal power operation and allows limited access to reactor coolant equipment. The primary shield also reduces neutron activation of the components and structures over the life of the plant. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.5 Secondary Shield

The secondary shield consists of 2 to 3 1/2 ft of reinforced concrete and surrounds the reactor coolant equipment, steam generators, pressurizer, and associated piping. This shield supplements the primary shield by further attenuation of neutrons escaping the primary shield and permits limited access to the containment during full power operation by attenuating the nitrogen-16 gammas from the primary coolant system. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.6 Spent-Fuel Pool Shielding

Shielding is provided for protection during all phases of spent-fuel removal and storage. Operations requiring shielding of personnel are spent-fuel removal from reactor, spent-fuel transfer through refueling canal and transfer tube, spent-fuel storage, and spent-fuel cask loading operations.

Since all spent-fuel removal and transfer operations will be carried out under borated water, minimum water depths above the tops of the fuel assemblies have been established to provide radiation shielding protection. The dose rates at the water surface should normally be less than 2.5 mrem/h. The concrete walls of the fuel transfer canal and spent-fuel pool supplement the water shielding and limit the continuous radiation dose levels in working areas to normally less than 2.5 mrem/h. However, the radiation levels will be closely monitored during removal and transfer operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.1201-20.1208.

The refueling water and concrete walls also provide shielding from activated rod cluster control assemblies and reactor internals which will be removed at refueling times. Although dose rates will generally be less than 2.5 mrem/h in working areas, certain manipulation of fuel assemblies, rod cluster control assembly, or reactor internals may produce areas where dose rates exceed 2.5 mrem/h for short periods. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.1201 - 20.1208.

All spent-fuel pool penetrations are located higher than the minimum water depth above the fuel assemblies, so that a failure in any penetration will not drain the pool to less than the minimum water level.

12.1.2.7 Control Room

Control room shielding design is based on the requirements set forth in 10 CFR 50, Appendix A, General Design Criterion 19, which requires occupancy of and access to the control room under accident conditions. The dose to personnel will be limited to 5 rem total effective dose equivalent (TEDE) per 10 CFR 50.67 for the duration of the accident. The accident analysis in chapter 15 indicates that this dose to personnel will be less than 5 rem TEDE.

Protection of control room personnel from the fission product release in the containment is provided by the concrete walls between them. Emergency air conditioning and filtration systems are provided for accident conditions and are described in detail in subsection 9.4.1. Figure 12.1-1 contains the control room layout and isometrics of the control room and associated shielding.

12.1.2.8 Auxiliary Building

The auxiliary building shielding includes all concrete walls, covers, and removable blocks that protect personnel working near various system components of the waste processing system, chemical and volume control system, boron thermal regeneration system, and safety injection system. Typical radioactive sources are the waste evaporator, recycle evaporator, demineralizers, filters, waste gas decay tanks, waste holdup tanks, recycle holdup tanks, and the waste drumming area.

Equipment is shielded in compartments, and the shield walls in each compartment are evaluated on the basis of radiation levels within the compartment, the surrounding sources, and access and maintenance requirements.

All radioactive areas are accessible through service corridors that can be entered from the access control station. In the high radiation zones, manually operated valves necessary for system operation and normal maintenance of contaminated equipment have been provided with reach rods penetrating through the shield walls into the corridor or have remote manual operators. Gauges and instrumentation requiring visual checking periodically will be inspected from the corridors or on the local or central control boards.

12.1.2.9 Turbine Building

The turbine building is normally accessible during plant operation and shutdown. In the event of a maximum hypothetical accident, access to the turbine building is controlled for radiation protection. Access is normally controlled for security reasons. There is no direct radiation from the turbine building.

12.1.2.10 General Plant Yard Areas

The radiation field in the plant yard areas frequently occupied by plant personnel is limited to < 0.5 mrem/h. The exception being any radiation controlled area (RCA) being setup or established by Radiation Protection (RP).

12.1.2.11 Inspection of Steam Generator Tubing

Inspection of steam generator tubing when required is performed by use of eddy current techniques. Entrance is required to the primary side of the steam generator to install a probe positioner. This positioner is so designed that installation time is minimized. Depending on the number and location of tubes to be inspected, additional entrance to move the positioner may be required. The eddy current and probe inspection monitoring equipment is located and operated from a position remote from the steam generator.

The exposure to personnel is maintained as low as reasonably achievable by a combination of training, shielding, and location of the control, readout, and probe pushing equipment in a remote location.

12.1.2.12 Corridor Leading to the Personnel Access Hatch to the Containment at the 155-ft Level

The corridor has heating, ventilation, and air conditioning equipment rooms on either side which normally have no sources of radioactivity but might have a local hot spot on a contaminated filter. There are no sources in the containment near the access hatch. The steam generators and the pressurizer have concrete shielding around them. The reactor vessel is well below the 155-ft level. Above the corridor is the roof of the auxiliary building; there are no sources on the roof. Below the corridor are electrical penetration rooms, which have no sources. The corridor is in an area of the auxiliary building where radioactive contamination is possible; consequently, access to the area is controlled and personnel must wear an Optically Stimulated Luminescent dosimeter (OSLD) or other personnel dosimetry devices while in the area.

12.1.2.13 Operating Floor of the Containment During Cold Shutdown Condition

The only time that the reactor would normally be placed in a cold shutdown condition is for refueling. Degasification of the primary coolant system will be required in order to refuel the reactor. Therefore, the noble gases will be purged from the top of the pressurizer. The pressurizer and the steam generators have concrete shields around them on the operating floor. The reactor vessel is well below the operating floor and will be covered by water after the refueling canal is flooded. There may be miscellaneous hot spots around the sides of the steam generators and the pressurizer, but these pieces of equipment are shielded. The reactor head will be the hottest source on the operating floor during refueling; temporary shielding will be installed around it for personnel protection, if crud deposits cause hot spots that would otherwise contribute to excessive doses. There are no sources above the operating floor. Sources below the operating floor are shielded by the thick concrete floor.

12.1.2.14 Old Steam Generator Storage Facility

The six old steam generators removed from Unit 1 and Unit 2 primary containments are stored in the old steam generator storage facility (OSGSF), which is a reinforced concrete building that provides long-term storage of and shielding for the steam generators. Any reactor coolant system (RCS) elbows that may be replaced and concrete wall sections cut from the secondary shield walls during the steam generator replacement are also stored in the OSGSF. As shown on the site plot plan (drawing D-170084), this facility is located south of Unit 1, outside the protected area, but is within the owner controlled area and site boundary.

12.1.3 SOURCE TERMS

The shielding design source terms are based upon the three general plant conditions of normal full-power operation, shutdown, and design basis events.

Subsection 12.1.1 and paragraph 5.2.1.19 provide a complete description of design considerations and procedures used to ensure that field run process piping is designated and routed with appropriate regard for minimizing exposures to plant personnel.

12.1.3.1 Sources for Normal Full-Power Operation

The main sources of activity during normal full-power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. The activity level of N-16 at various locations in the RCS is shown in figure 12.1-2. The isotopic inventory of fission, corrosion, and activation products in the reactor coolant is given in table 11.1-2. All shielding is based on the maximum case of clad defects in fuel rods producing 1.0 percent of core thermal power. Expected sources would be based on defects in fuel rods producing 0.25 percent of core thermal power as discussed in section 11.2. Each plant system was shielded according to the amount of activity present and adjacent zoning and access criteria. These systems include:

- RCS.
- Chemical and volume control system (CVCS).
- Waste processing system.
- Boron recycle system.
- Spent-fuel pool cooling and purification system.
- Steam generator blowdown processing system.

The N-16 activity of the coolant is the controlling radiation source in the design of the RCS secondary shielding and is plotted in figure 12.1-2 as a function of transport time in a reactor coolant loop.

The radiation sources in the CVCS are given in table 12.1-1.

One of the purposes of the CVCS is to provide continuous purification of the reactor coolant water. The major equipment items include the regenerative and letdown heat exchangers, mixed-bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The boron thermal regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The seal water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger.

Table 12.1-1 gives a summation of the activity, by energy groups, of all isotopes listed in table 11.1-2. The delay time from the reactor coolant loop is sufficient for decay of the N-16 isotope.

The radiation sources in the ion exchangers, volume control tank, filters, and heat exchangers of the CVCS are also given in table 12.1-1.

The mixed-bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control and to supplement the mixed bed in removing Y, Cs, Mo, and the crud metals.

The BTR beds are used to regulate the boron concentration in the reactor coolant water. They are utilized during load follow operations and in removing boron from the coolant as the nuclear fuel is depleted. These demineralizers also collect radioactive anions, such as iodine, which may have passed through the mixed bed.

The regenerative and excess letdown heat exchangers are located in the containment building. They provide the initial cooling for the reactor coolant letdown and their sources include N-16 activity. The balance of the CVCS heat exchangers is located in the auxiliary building where N-16 activity is not a significant factor.

The letdown heat exchanger provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The activity at this point is identical to the letdown coolant source.

The thermal regeneration heat exchangers include the moderating, chiller, and letdown reheat units. The radiation sources in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

The seal water heat exchanger cools the water from the reactor coolant pump seals. In the source tabulation, credit has been taken for activity removed by the demineralizers and the volume control tank.

The radiation sources in the waste processing system are tabulated in table 12.1-2. The major equipment items in the waste gas portion are the waste gas compressors, hydrogen recombiners, and gas decay tanks. The radiation sources in this equipment are based on cold shutdown procedures during which the radioactive gases are stripped from the RCS. The radiation sources in the waste gas equipment are conservatively assumed to be identical.

The liquid waste processing system is considered as several subsystems, based on its intended use during normal operation. The equipment items normally associated with processing reactor grade water are the waste holdup tank, waste evaporator feed filter, and waste evaporator. The evaporator distillate is directed to the waste condensate tank and may be further processed through the waste evaporator condensate demineralizer and filter, if required. The waste evaporator concentrates are sent to the drumming station or solidification and dewatering building for packaging.

Low activity, nonreactor grade water is directed to the floor drain or laundry and hot shower subsystems. Normally this water is analyzed, then discharged. If activity levels prevent this, the water can be processed by a demineralizer/filter or the waste evaporator. The equipment included in the subsystem is the floor drain tank and filter, laundry and hot shower tank and filter, waste monitor tank demineralizer and filter, and two waste monitor tanks. The floor drain and waste monitor tanks provide surge capacity for the waste holdup tank during periods when abnormal volumes of liquid waste are encountered. Hence, for shielding purposes the radiation sources in these tanks are assumed to be the same, i.e., degassed reactor coolant. Similarly, the sources on the floor drain tank filter are the same (100 R/h contact) as the waste evaporator feed filter since they can operate in similar service. The sources on the waste monitor tank demineralizer and filter are based on circulating reactor coolant through these components.

Radioactive spent resins discharged from the various demineralizers are retained in the spent-resin storage tank. The mixed-bed demineralizer contains the most radioactive resin discharged to the storage tank; these sources determine the tank shielding required. The short-lived activity is allowed to decay (~30 days), and the resin is then directed to the solidification and dewatering facility for packaging. The associated equipment includes the spent-resin storage tank and the resin sluice pump and filter. The resin sluice filter is shielded for radiation levels of 100 R/h contact.

Radiation sources in the various pumps in this system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

Sources in the laundry, hot shower tank and filter, and waste condensate tank are negligible; these items do not require shielding.

The evaporator concentrates and the spent resin are packaged at the solidification and dewatering facility for shipment to an offsite burial facility. Prior to shipment, the packaged waste is stored as described in section 11.5. The shielding for the drum storage area is designed to accommodate the full storage capacity with each drum reading 1 R/h at 3 ft. Spent resin can be stored in a steel shipping shield, if necessary, to limit radiation levels.

The radiation sources in the boron recycle system are listed in table 12.1-3. The major equipment items included in this system are the recycle holdup tanks and the recycle evaporator with its associated equipment, i.e., feed demineralizers and filter, condensate demineralizer and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

The evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed-bed resins which remove nongaseous activity from the reactor coolant directed to the holdup tanks. A dilution factor of 10 across these beds is taken for all particulate activity.

The evaporator condensate demineralizer is charged with anion resin to remove any boron and iodine activity which may be carried over with the evaporator condensate.

The recycle holdup tanks are each equipped with a diaphragm. Gases which flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until /500 ft^3 of gas has accumulated; the gases are then removed to the waste gas system. The radiation sources in the holdup tanks are based on 50 percent of the gaseous activity flashing into the vapor phase.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers and serve to retain particulates and any resin fines which may escape from the demineralizers.

The maximum radiation sources on these filters are listed below. The sources for the feed filter correspond to a radiation level of 100 R/h contact. The condensate filter sources result in levels of less than 1 R/h contact. The maximum activity of the liquid concentrates in the recycle evaporator is 40 μCi/g. The resultant radiation sources on the concentrates filter correspond to an exposure rate of approximately 3 R/h.

The radiation sources in the spent-fuel pool cooling system are given in table 12.1-4. The system demineralizer and filter are used to maintain water clarity and remove activity released during refueling operations and the subsequent fuel cooling period. The filter sources correspond to an exposure rate of 100 R/h contact.

The radiation sources for the steam generator blowdown processing system are given in table 12.1-5. The sources are based on removal of all radioactive contaminants in one 75-ft^3 bed assuming 144 gal/day primary to secondary steam generator leakage, 1 percent-fuel clad defects, and a 90-day service lifetime for the bed.

The exposure rate at site boundary per Ci of stored waste (including shipping casks) is dependent not only on the energy of emissions of the radioactive materials stored but also on the amount of shielding used. The radiation exposure at the site boundary from direct radiation from stored radioactive materials is expected to be a small fraction of the natural background radiation.

12.1.3.2 Sources for Shutdown Conditions

In the reactor shutdown condition the only additional sources of radiation requiring shielding is the residual heat removal system.

The maximum specific source strengths in the residual heat removal loops are given in table 12.1-6. The residual heat removal loop is placed in operation approximately 4 h after reactor shutdown and reduces the reactor coolant temperature to approximately 120°F within about 20 h after shutdown. The sources are maximum values with credit taken for 4 h of activity decay and purification.

12.1.3.3 Sources for Design Basis Events

The fission product sources released to the containment building following a core meltdown accident are based on the assumptions stated in TID-14844.[1] These are as follows:

Core Meltdown Accident

NSSS power level (MWt)	2774
Equivalent fraction of core melting	1.0
Fission product fractional releases	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01

Minimum full-power operating time (days)	650
Cleanup rate following accident	0.0

The assumptions used in the gap release accident are listed below:

Gap Release Accident

NSSS power level (MWt)	2774
Fraction of gap activity released to containment	1.0
Fraction of gap activity absorbed by the sump water	
Noble gases	0.0
All others	1.0
Reactor coolant volume (ft^3)	9,107
Refueling water volume (ft^3)	40,100
Total volume (ft^3)	49,207

The fission product sources released to the containment following an equivalent 100-percent core meltdown (TID-14844 release) are listed in table 12.1-7. These sources are used to calculate the postaccident radiation levels outside the containment building.

The radiation sources circulating in the residual heat removal loop and associated equipment are tabulated in table 12.1-8.

These sources are based on an accident in which the fission products in the gap region between the fuel pellets and cladding are released to the containment. The nongaseous activity is assumed to be transferred to the sump water which flows in the residual heat removal loop.

The design basis for the postaccident recirculation system is that residual heat removal pump compartments have sufficient shielding to permit limited access in the pump compartments following a gap release accident. The integrated exposure does not exceed 3 rem for any 8-h period after the accident.

12.1.4 AREA MONITORING

12.1.4.1 Design Bases

The area radiation monitoring system is provided to supplement the personnel and area radiation monitoring provisions of the plant radiation protection program described in section

12.3. Included in this system are nine permanently located radiation detectors for Unit 1 and ten permanently located radiation detectors for Unit 2, which provide continuous local and remote indication and alarm of direct radiation dose rate levels. The primary objectives of the system are:

A. To immediately alert plant personnel entering or working in normally unlimited occupancy areas of increasing or abnormally high radiation levels, which, if unnoticed, might possibly result in inadvertent overexposures.

B. To inform the control room operators of the occurrence and approximate location of abnormal events 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.

D. To provide a continuous record of external dose rates at selected locations, thereby ensuring detection of transient increases in doses which are attributable to rapid changes in the radioactivity content of equipment and process streams.

E. To provide information on radiological conditions in the containment, in the event of a NUREG 0578 accident (Unit 1) or NUREG 0737 accident (Unit 2).

In addition, an exemption from 10 CFR 70.24, relative to the authorization to possess special nuclear material at Farley Nuclear Plant, has been granted by the Nuclear Regulatory Commission(4) that provides relief from the requirement to install criticality monitors. These monitors are not needed because inadvertent or accidental criticality will be precluded through compliance with the plant Technical Specifications, geometric spacing of fuel assemblies in the new fuel storage area and spent-fuel storage pool, administrative controls imposed on fuel handling procedures, and the use of nuclear instrumentation that monitors the behavior of nuclear fuel in the reactor vessel.

12.1.4.2 System Description

This system consists of multiple channels which monitor radiation levels in various areas of the plant, among which are the following:

Channel	Area Monitoring
R-1 (Unit 1 only)	Control room
R-1B (Unit 2 only)	Technical support center
R-2	Containment

R-3	Radiochemistry laboratory
R-4	Charging pump room
R-5	Spent-fuel building
R-6	Sampling room
R-7	Incore instrumentation area
R-8	Drumming station
R-9 (Unit 2 only)	Sampling panel room

These locations have been chosen as representative of plant locations where significant sources of radioactive material are stored and/or handled or where occupancy is highest.

Detecting medium for the channels is air with a corresponding temperature range of 40°F to 120°F. Each channel consists of a fixed position gross beta gamma Geiger-Mueller tube or ion chamber (R-2 & R-7 only) detector with range 1.0×10^{-4} to 1.0×10^{1} R/h (rad/h for R-2 & R-7). Drawing U-167647 contains a functional block diagram for the above area monitor channels.

The area radiation level is indicated locally at the detector, in the cable spreading room at the signal processing cabinet (R-2 & R-7 only), and at the radiation monitoring system cabinets. Radiation levels are recorded by a data acquisition system computer which can display data, on demand, to the operator. High radiation alarms are displayed at the radiation monitoring system cabinets and annunciated at the detector location and at the control board in the control room. The control board annunciator provides a single window which alarms for all area radiation monitor channels in addition to process radiation monitor channels R-10 through R-13, 2R-14, R-15, R-17 through R-20, 2R-21, 2R-22, R-23, Unit 1 R-29B (channels E & Composite Gas Channel)/R-29C (channels H & J), and Unit 2 R-29B (channels E & Composite Gas Channel)/R-29C (channels H & J). Channels R-24A and B and R-25A and B have individual annunciator windows on the control board. Verification of which area radiation monitor channel has alarmed is done at the radiation monitoring system cabinets in the control room.

To meet the requirements of NUREG 0578, Alabama Power Company (APC) has installed radiation detection systems R-27A and B to meet the requirements for a high-range containment radiation monitor. Each system consists of an ion chamber detector, signal processing cabinet, readout panel, and interconnecting cables. The detectors are located inside containment about 5 ft above the operating deck and approximately 90° apart. These locations ensure that detectors are not protected by massive shielding and they will provide a reasonable assessment of area radiation conditions inside the containment during and following an accident.

A. Each detector is designed to measure gamma radiation.

B. The range of each detector is 1 R/h to 10^{7} rad/h for photon radiation.

C. The energy response is ±14 percent to 81 keV to 3 MeV and ±12 percent from 100 keV to 3 MeV.

D. The calibration frequency will be once per refueling cycle as defined by the Technical Specifications. Capability exists for onsite calibration of the radiation detector to 10 rad/h.

12.1.4.3 Design Evaluation

Area monitors are located in areas of the plant which house equipment containing or processing radioactive fluid or where, because of personnel occupancy, it is deemed necessary to monitor continuously. These instruments continually detect and record operating radiation levels. If the radiation level should rise above the setpoint listed for each channel (see table 12.1-9), an alarm is initiated in the control room. Local annunciation is provided at the detector to indicate high radiation levels to personnel in the area. The radiation monitoring system operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning are thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed the limits of 10 CFR 20.1201 - 20.1208.

12.1.5 OPERATING PROCEDURES

12.1.5.1 General

The Radiation Protection manager is responsible for assisting the training manager in developing a radiation protection training program and for developing a radiation surveillance program to ensure that exposures of all personnel are kept within the limits of 10 CFR 20.1201 - 20.1208. See section 12.3 for a description of the Radiation Protection program as it relates to shielding operating procedures.

12.1.5.2 Procedures

The basic principles of time, distance, and shielding will be applied during operation and maintenance to ensure that personnel exposure will be within limits. Specifically, the following procedures and techniques will be employed:

A. During initial startup, neutron and gamma dose rate surveys will be performed to determine the adequacy of shielding.

B. During normal operations, dose rate surveys will be performed periodically throughout the plant and areas will be posted accordingly. This procedure will ensure that data are available for planning operation and maintenance activities.

C. Radiation areas will be conspicuously posted. High radiation areas will be conspicuously posted and barricaded. High radiation areas of 1 R/h measured at 30 cm but less than 500 rads/h measured at 1 m will be provided with locked or continuously guarded doors and the keys maintained under the administrative control of the shift supervisor or shift support supervisor and/or Radiation Protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. For individual high radiation areas with radiation levels, as measured at 30 cm from the radiation source or from any surface that the radiation penetrates, such that a major portion of the body could receive in 1 h a dose greater than 1000 mrem, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device. Entry into high radiation areas is controlled by the Technical Specifications. Very high radiation areas (levels greater than 500 rads/h at 1 m) will be conspicuously posted and access will be controlled in accordance with plant procedures. These control measures comply with the NRC acceptance criteria contained in 10 CFR 20.1601 and 20.1602, respectively.

D. A radiation work permit system will be employed to ensure proper administrative control over work in restricted areas. The permit is designed to ensure that the radiation conditions are known and that appropriate measures are taken to minimize the dose received by personnel.

E. Extension tools will be used when possible or practical to increase the distance from the radiation source to the worker.

F. Equipment will be moved to areas of lower radiation fields for maintenance when possible or practical.

G. Portable shielding in the form of lead bricks, lead sheets, lead shot, and/or high density concrete blocks will be considered for use when the requirements of items E and F are not possible or practical. Steel plates will be used in lieu of lead where high temperature may be a factor. (A shielding evaluation will be conducted prior to installing shielding on safety-related equipment.)

H. A personnel dosimetry program, as described in subsection 12.3.3, will be administered by the Radiation Protection group to ensure compliance with 10 CFR 20.1502.

Each permanent plant employee who is to be a radiation worker will attend radiation worker orientation prior to being allowed unescorted access.

Experience gained during the operation and maintenance of FNP and that of several nuclear plants with whom SNC has contact will be used to provide a basis for further evaluation and development of shielding procedures.

12.1.6 ESTIMATES OF EXPOSURE

12.1.6.1 Exposures in the Controlled area and in the Unrestricted Area

12.1.6.1.1 Normal Plant Operations

The total effective dose equivalent (TEDE) to individuals due to licensed operation will not exceed 100 mrem in a year in the controlled area, as defined in 10 CFR 20.1003.

Doses at the site boundary from radioactive liquid releases are given in subsection 11.2.9 and those due to gaseous effluents are dealt with in subsection 11.3.9.

At and beyond the site boundary, (unrestricted area, as defined in 10 CFR 20.1003), the interpreted man-rem values (for normal operations) as a function of distance are computed from the expected gaseous and liquid releases, and the atmospheric dilution factors are presented in subsection 11.3.8, liquid dilution and reconcentration factors in subsection 11.2.8, and the population density in subsection 2.1.3.

12.1.6.1.2 Operational Occurrences

Maximum radiation exposures resulting from operational occurrences are discussed in detail in chapter 15. Under the most severe conditions, the dose values at the site boundary, the low population zone distance, and at the visitors center will be well below 10 CFR 100 guidelines.

12.1.6.2 Exposures Within Restricted Areas

12.1.6.2.1 Normal Plant Operations

Administrative controls and controlled access will ensure that the plant personnel working in restricted areas, as defined in 10 CFR 20.1003, will not receive doses in excess of those established in 10 CFR 20.1201 - 20.1208.

The dose rates given in subsection 12.1.1 represent the upper limits, and the expected exposures will be significantly lower than these limits.

12.1.6.2.2 Operational Occurrences

Various operational occurrences are discussed in detail in chapter 15. Minor occurrences like spills or leakages will contribute no appreciable increase to normal exposures and will be of only local significance. However, if the radiation levels from an occurrence call for evacuation of the plant, this will be indicated on the radiation monitoring equipment and the emergency evacuation plan will remain in operation. Personnel essential to shut down the plant and maintain it in a safe condition under accident conditions will direct required operator actions from the control room. Maximum dose rates and protection of the control room are discussed in detail in paragraph 12.1.2.7.

12.1.6.2.3 In-Plant Radiation Monitoring

A program exists which will ensure the capability to accurately determine the airborne iodine concentration in certain plant areas where personnel may be present under accident conditions. This program shall include the following:

a. Training of personnel,
b. Procedures for monitoring, and
c. Provisions for maintenance of sampling and analyses of equipment.

12.1.6.3 <u>Comparison with Other Operating Plants</u>

Liquid effluent releases from operating pressurized water reactor plants are indicated in table 11.2-6 for years 1970 and 1977 and actual measured values for radioactive noble gas releases from FNP Unit 1 for years 1977 to 1983 are indicated in table 11.3-6.

The average anticipated releases from this plant during operation are included in tables 11.2-7 and 11.3-9, and doses from these releases are given in sections 11.3 and 11.2.

12.1.6.4 <u>Estimated Annual Exposures</u>

Table 12.1-10 gives typical doses received by personnel in operating plants based on data from 1977 to 1981. Those areas described in table 12.1-10 that have radiation levels greater than 100 mrem/h are classed as Radiation Zone 5 areas, as described in subsection 12.1.1.

The estimated annual exposure from the plant as designed is 350 man-rem per unit. This number is a typical value for relevant operating plants as shown in table 12.1-10 based on data from 1977 to 1981. The value of 350 man-rem per unit can be confirmed by analysis of other relevant plant data, as shown in table 12.1-11.

12.1.7 DESIGN REVIEW OF PLANT SHIELDING FOR POSTACCIDENT OPERATION

This subsection describes the design review of plant shielding of spaces for postaccident operations, as required by NUREG-0737, item II.B.2. Systems required to process primary reactor coolant outside the containment during postaccident conditions were selected for evaluation. Large radiation sources beyond the original plant design bases were postulated to be present in the selected systems. Areas and equipment which are vital for postaccident occupancy or operation were evaluated to determine whether access and performance of required operator activities might be unduly impaired due to the presence of the postulated radiation source in these systems.

12.1.7.1 Selection of Systems for Shielding Review

The criteria applied in selection of plant systems used in the shielding review resulted in several classifications of systems as discussed below.

A. Category A (Recirculation Systems)

The first group of systems are those required by plant design to mitigate a design basis loss-of-coolant accident (LOCA) and which might contain highly radioactive sources in excess of the current design basis. A first-priority safety concern is to ensure that operation of those systems containing a significant source will not adversely impact operator functions required outside the containment. Therefore, the following systems have been selected to ensure this first-priority safety concern is adequately addressed by the existing plant shielding design:

1. Those portions of the containment spray system used to recirculate water from the containment sump back into the containment.

2. Those portions of the residual heat removal system used to recirculate water from the containment sump back into the containment.

3. Those portions of the high-head safety injection system used to recirculate water from the containment sump via the residual heat removal system back into the containment.

B. Category B (Extensions of Containment Atmosphere)

In addition to systems listed above, there are other systems or portions of systems which would contain radioactivity by virtue of their connection to the containment following an accident. Proper operation of the emergency core cooling systems (ECCS) would prevent extensive core damage and mean that these systems would not be expected to contain the significant radioactive sources required by this special analysis. Nevertheless, such sources have been postulated in the following systems:

1. Those portions of the postaccident containment combustible gas control system external to the containment which would contain the atmosphere from the containment.

2. Those portions of the containment ventilation systems external to the containment up to the second isolation valve which could contain the atmosphere from the containment.

3. Those portions of the sampling system used to obtain a containment atmosphere sample.

C. Category C (Liquid Samples)

Item II.B.3 of NUREG 0737 required that certain postaccident liquid samples be obtained from the RCS for containment systems. Those portions of the sampling system which were identified for use to meet the intent of item II.B.3 were selected for this shielding review.

D. Category D (Letdown)

That portion of the letdown system from the RCS past the letdown heat exchanger up to the inlet valves to the letdown demineralizers has been selected for analysis.

12.1.7.2 <u>Quantification of Potential Radioactive Source Release Fractions</u>

The following release fractions were used as a basis for determining the concentrations for the shielding review:

A. Source A, containment atmosphere – 100% noble gases, 25% halogens.

B. Source B, reactor coolant – 100% noble gases, 50% halogens, 1- percent solids.

C. Source C, containment sump liquid – 50% halogens, 1% solids.

The above release fractions were applied to the total Ci available for the particular chemical species (i.e., noble gas, halogen, or solid) for an equilibrium fission product inventory for a light water reactor core.)

12.1.7.3 <u>Source Term Models</u>

The paragraph above outlines the assumptions used for release fractions for the shielding design review. These release fractions are, however, only the first step in modeling the source terms for the activity concentrations in the systems under review. The important modeling parameters of decay time and dilution volume also affect shielding analysis. The following sections outline the rationale for the selection of values for these key parameters.

12.1.7.3.1 Decay Time

For the first stage of the shielding design review process, no decay time credit was used with the above release. The primary reason for this was to develop a set of accident radiation zone maps normalized to no decay that could be used as a tool by the plant staff along with a set of decay curves to quantitatively assess the plant status quickly following any abnormal occurrence. However, the following decay times were used in assessing anticipated potential personnel radiation exposure due to those operator actions required post-LOCA.

For analyses of personnel exposures in vital areas outside the control room, radioactive decay equivalent to 10 min allowed for operator action was used as the minimum decay time.

A decay time of 24 min, which is consistent with the time for initiation of recirculation in accordance with chapter 6, was allowed for the review of those ECCS systems that are used to recirculate water from the containment sump back into the containment.

12.1.7.3.2 Dilution Volume

The volume used for dilution is important, affecting the calculations of dose rate in a linear fashion. The following dilution volumes were used with the release fractions and decay times listed above to arrive at the final source terms for the shielding reviews:

A. Source A, containment free volume - The volume occupied by the ECCS water was neglected.

B. Source B - RCS volume based on reactor coolant density at the operating temperature and pressure.

C. Source C - The volume of water present at the time of recirculation (RCS + refueling water storage tank + safety injection tanks).

12.1.7.3.3 Sources Used in Piping and Equipment for Each System Under Review

In defining the limits of the connected piping subject to contamination listed below, normally shut valves were assumed to remain shut.

A. Containment spray system - At the initiation of recirculation, source C was used.

B. High-head safety injection system - At the initiation of recirculation, source C was used.

C. Residual heat removal system - Source C was used for sump recirculation mode.

D. Sampling systems - The sources used in the shielding design review for sampling systems were as follows:

1. Containment air sample - Source A.

2. Reactor coolant sample - Source B.

E. Letdown system - The liquid source was source B.

12.1.7.4 Shielding Design Review Methodology

12.1.7.4.1 Analytical Shielding Techniques

The previous sections outlined the rationale and assumptions for the selection of the systems that would undergo a shielding design review as well as the formulation of the sources for those systems. The next step in the review process was to use these sources along with standard point kernel shielding analytical techniques to estimate dose rates from those selected systems. For compartments containing the systems under review, estimates were made for a general area dose rate rather than superimposing the maximum dose rate at contact with the surfaces of all individual components of that system in the compartment. For corridors outside compartments, reviews were done to check the dose rate transmitted into the corridor through the walls of adjacent compartments. Checks were also made for any piping or equipment that could directly contribute to corridor dose rate, i.e., piping that may be running directly into the corridor or equipment/piping in a compartment that could shine directly into corridors with no attenuation through compartment walls.

12.1.7.4.2 Accident Radiation Zone Maps

One of the two principal products of this review is the series of accident radiation zone maps. These zone maps represent the correlation of the dose rates as estimated above with the required operator actions and resultant necessary accessibility to vital areas.

The most conservative decay curve (source B) should be used if doubt exists about which source is causing the dose rate. The accident zone maps give total dose rates from all sources and do not distinguish between sources A, B, or C.

These zone maps are shown in drawings D-176075, D-176076, D-176077, D-176078, D-176079 (Unit 1), and drawings D-206075, D-206076, D-206077, D-206078, and D-206079 (Unit 2) for elevations 77 to 83 ft, 100 to 105 ft, 121 to 129 ft, 139 ft, and 155 ft.

The zone boundaries were formulated based on the following rationale:

Zone Designation	Rationale	Zone Dose Rate Limits ($\dot{D}$) (rem/h)
A-I	The first zone is consistent with personnel radiation exposure guidelines for vital areas requiring continuous occupancy.	$0 \leq \dot{D} \leq 0.015$
A-II	The second zone is consistent with the personnel radiation exposure guidelines for vital areas requiring occasional access or for corridors to these areas.	$0.015 \leq \dot{D} \leq 0.100$
A-III	The third zone is consistent with the personnel radiation exposure guidelines for vital areas requiring infrequent access or corridors to these areas.	$0.100 \leq \dot{D} \leq 5.0$

The subsequent zones were selected by grouping them by powers of 10 so that rapid assessment of additional shielding measures could be used via tenth-value layers of common shielding materials.

Zone Designation	Zone Dose Rate Limits ($\dot{D}$)(rem/h)
A-IV	$5 \leq \dot{D} \leq 50$
A-V	$50 \leq \dot{D} \leq 500$
A-VI	$500 \leq \dot{D} \leq 5000$
A-VII	$5000 \leq \dot{D} \leq 50{,}000$
A-VIII	$\dot{D} \geq 50{,}000$

These zone designations should not be confused with those used for the normal plant operation zone maps shown in drawings D-176035, D-176036, D-176037, D-176038, D-176039 (Unit 1), and drawings D-206035, D-206036, D-206037, D-206038, D-206039, (Unit 2).

12.1.7.4.3 Decay Curves

Figures 12.1-3 through 12.1-6, corresponding to sources A, B, and C outlined in paragraph 12.1.7.2, are a set of curves developed for the shielding design review to serve as generic tools to estimate transient decay credit. These generic curves were developed rather than developing a parametric set of curves for each source that would account explicitly for the effects of self-attenuation in the source material on actual dose rates. The primary assumption in the application of the curves was that the dose rate from a source was directly proportional to the total gamma ray energy release rate (in MeV/s) from the source in question, i.e., source A, B, or C. Therefore, the decay curves are more properly fission product energy release rate curves. These curves were developed in a similar manner as those from the work of Lurie, et al.[2] for the Sandia Laboratory research directed by Bonzon, et al.[3] All curves have been normalized to the initial energy release rate for the source in question.

12.1.7.5 Postaccident Access and Personnel Exposure

12.1.7.5.1 Access

Those operator actions required post-LOCA were reviewed to ensure that first-priority safety actions can be achieved in the postulated radiation fields. This review ensures that access is available and required operator actions can be achieved.

The following areas in the auxiliary building require postaccident access:

Area	Occupancy Period
Control room, technical support center	24 h/day
Radiation Protection area	24 h/day
Hallway 316	1 h/day
Hallway 409	1 h/day
Hallway 322 (outside sample room)	2 h/day
Cable spreading room	1/2 h
Filter rooms	2 h/day
Switchgear rooms (el 121 ft)	1/2 h
Hot shutdown panel	24 h/day
Component cooling water pump room	1/2 h
Corridor 161	1/2 h
Residual heat removal heat exchanger room	1/2 h
Stairway 1	Transit to elevations at west side of auxiliary building
Stairway 2	Transit to el 77 ft to el 83 ft
Stairway 8	Transit to elevations at north and east sides of auxiliary building

12.1.7.5.2 Personnel Radiation Exposure

The general basis for personnel radiation exposure guidelines was 10 CFR 50, Appendix A, General Design Criterion 19. The following additional radiation guidelines were used to evaluate occupancy and accessibility of plant vital areas. General area dose rates were used rather than maximum surface dose rates. Contributions from all sources were considered.

A. Vital areas requiring continuous occupancy – Vital areas such as control room and the onsite technical support center were verified to ensure the direct dose rate was less than 15 mR/h.

B. Vital areas requiring infrequent access or corridors to these vital areas - For these areas the dose rate was verified to be less than 5 R/h.

   For dose rates greater than 100 mR/h, a man-rem calculation including time and motion analysis was performed to ensure that the integrated exposure for an operator action would not exceed 5 rem as given in General Design Criterion 19.

REFERENCES

1. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

2. Lurie, N. A., Houston, D. H., and Naber, J. A., "Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions," SAND 78-0091, May 1978.

3. Bonzon, L. L., Gillen, K. T., and Salazar, E. A., "Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report: October-December 1978," NUREG/CR 0813 or SAND 79-0761, June 1979.

4. Letter from B. L. Siegel (Nuclear Regulatory Commission) to D. N. Morey (Southern Nuclear Operating Company), dated July 31, 1996, regarding exemption from the requirements of 10 CFR 70.24, “Criticality Accident Requirements,” for the Joseph M. Farley Nuclear Plant, Units 1 and 2.

TABLE 12.1-1 (SHEET 1 OF 4)

RADIATION SOURCES - CHEMICAL AND VOLUME CONTROL SYSTEM

Letdown Coolant Sources

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4

Mixed-Bed Demineralizer Sources

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.3×10^8
0.8	2.5×10^8
1.3	3.0×10^7
1.7	1.2×10^7
2.2	3.5×10^6
2.5	2.4×10^5
3.5	1.6×10^5

Cation Bed Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.4×10^5
0.8	2.2×10^8
1.3	5.9×10^6

TABLE 12.1-1 (SHEET 2 OF 4)

Boron Thermal Regeneration Demineralizers

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	2.3×10^5
0.8	1.4×10^5
1.3	5.2×10^4
1.7	2.3×10^4
2.2	9.1×10^3

Volume Control Tank

Vapor Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	2.6×10^6
0.4	5.7×10^5
0.8	1.8×10^5
1.7	1.4×10^5
2.2	3.3×10^5
2.5	6.8×10^5

Liquid Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	7.3×10^4
0.8	4.3×10^4
1.3	1.4×10^4
1.7	2.6×10^4
2.2	3.9×10^4
2.5	8.7×10^4

TABLE 12.1-1 (SHEET 3 OF 4)

Reactor Coolant Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	5.7×10^7
1.3	1.5×10^7

Seal Water Injection Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	4.8×10^7
1.3	1.2×10^7

Seal Water Return Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.1×10^7
1.3	3.0×10^6

Regenerative Heat Exchanger and
Excess Letdown Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4
6.1	2.2×10^6
7.1	1.8×10^5

TABLE 12.1-1 (SHEET 4 OF 4)

Letdown Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4

Boron Thermal Regeneration System (Moderating, Chiller, and Letdown Reheat Heat Exchangers)

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.1×10^5
0.8	6.7×10^4
1.3	1.4×10^4
1.7	3.9×10^4
2.2	1.2×10^5
2.5	1.6×10^5

Seal Water Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	7.3×10^4
0.8	4.3×10^4
1.3	1.4×10^4
1.7	2.6×10^4
2.2	3.9×10^4
2.5	8.7×10^4

TABLE 12.1-2 (SHEET 1 OF 4)

RADIATION SOURCES - WASTE PROCESSING SYSTEM

Waste Evaporator Condensate Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.4	2.1×10^4
0.8	4.0×10^4
1.3	2.6×10^3
1.7	1.5×10^3
2.2	4.5×10^2

Waste Monitor Tank Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.2×10^6
0.8	3.9×10^6
1.3	1.0×10^6
1.7	5.0×10^5
2.2	1.9×10^5

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

TABLE 12.1-2 (SHEET 2 OF 4)

Drumming Station

Spent Resin

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.0×10^8
1.3	1.0×10^7

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

Waste Holdup Tank, Floor Drain Tank, and Waste Monitor Tanks

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.4	3.7×10^4
0.8	2.1×10^5
1.3	1.0×10^5
1.7	4.8×10^4
2.2	1.7×10^4

Spent-Resin Storage Tank

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.9×10^8
0.8	2.5×10^8
1.3	3.0×10^7
1.7	1.2×10^7
2.2	3.5×10^6
2.5	2.4×10^5
3.5	1.6×10^5

TABLE 12.1-2 (SHEET 3 OF 4)

Hydrogen Recombiner, Waste Gas Compressor, and Gas Decay Tanks

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	2.0×10^6
0.4	3.5×10^5
0.8	9.4×10^4
1.7	7.5×10^4
2.2	1.4×10^5
2.5	3.3×10^5

Waste Evaporator Feed Filter and Floor Drain Tank Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	3.4×10^7
1.3	8.9×10^6

Spent-Resin Sluice Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.1×10^7
1.3	3.0×10^6

Waste Monitor Tank Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	6.7×10^6
0.8	2.1×10^7
1.3	5.7×10^6
1.7	2.8×10^6
2.2	1.1×10^6

TABLE 12.1-2 (SHEET 4 OF 4)

Waste Evaporator Condensate Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.2×10^5
0.8	2.2×10^5
1.3	2.5×10^4

Waste Evaporator Vent Condenser Vapor

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	1.1×10^7
0.4	3.6×10^6
0.8	1.5×10^6
1.7	1.5×10^6
2.2	3.8×10^6
2.5	5.3×10^6

TABLE 12.1-3 (SHEET 1 OF 3)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Evaporator Feed Demineralizers

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.7×10^6
0.8	3.9×10^7
1.3	3.7×10^6
1.7	1.9×10^6
2.2	6.5×10^5

Recycle Evaporator Condensate Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.5×10^4
0.8	2.2×10^4
1.3	5.4×10^3
1.7	2.5×10^3
2.2	9.7×10^2

Recycle Holdup Tanks

Vapor Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	8.8×10^5
0.4	2.0×10^5
0.8	6.7×10^4
1.7	5.2×10^4
2.2	1.1×10^5
2.5	2.5×10^5

TABLE 12.1-3 (SHEET 2 OF 3)

Liquid Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	1.6×10^5
0.4	5.5×10^4
0.8	2.9×10^4
1.3	1.0×10^3
1.7	1.6×10^4
2.2	5.9×10^4
2.5	8.1×10^4

Recycle Evaporator Feed Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.1×10^7
1.3	3.0×10^6

Recycle Evaporator Condensate Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.6×10^5
0.8	8.0×10^4
1.3	3.3×10^4

Recycle Evaporator Concentrates Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.2×10^6

TABLE 12.1-3 (SHEET 3 OF 3)

Recycle Evaporator

Vent Condenser Vapor

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	1.1×10^7
0.4	3.6×10^6
0.8	1.5×10^6
1.7	1.0×10^6
2.2	3.8×10^6
2.5	5.3×10^6

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

TABLE 12.1-4

RADIATION SOURCES - SPENT-FUEL POOL COOLING AND PURIFICATION SYSTEM

Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.4	2.1×10^6
0.8	7.2×10^5
1.3	2.2×10^3
1.7	4.4×10^3

Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.8	1.1×10^7
1.3	3.0×10^6

TABLE 12.1-5

RADIATION SOURCES - STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm³/s)
0.4	2.2×10^5
0.8	9.2×10^5
1.3	8.7×10^3
1.7	1.0×10^4

TABLE 12.1-6

RADIATION SOURCES - RESIDUAL HEAT REMOVAL SYSTEM

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	1.9×10^5
0.4	5.0×10^4
0.8	8.5×10^4
1.3	3.0×10^4
1.7	1.9×10^4
2.2	1.7×10^4
2.5	2.9×10^4

TABLE 12.1-7

INSTANTANEOUS DIRECT GAMMA SOURCE STRENGTH (MeV/s)

	Time after Release				
Gamma Energy (MeV)	0 hour	1 hour	2 hours	1 day	1 month
Gases					
0.4	1.9×10^{18}	1.8×10^{18}	1.7×10^{18}	5.3×10^{17}	1.0×10^{15}
0.8	3.4×10^{18}	1.8×10^{18}	1.3×10^{18}	9.5×10^{16}	9.0×10^{14}
1.3	1.1×10^{17}	8.8×10^{16}	6.9×10^{16}	3.0×10^{14}	0.0
1.7	1.1×10^{19}	5.3×10^{17}	4.2×10^{17}	1.8×10^{15}	0.0
2.2	6.3×10^{18}	1.9×10^{18}	1.2×10^{18}	4.9×10^{15}	0.0
2.5	4.7×10^{18}	3.2×10^{18}	2.3×10^{18}	6.7×10^{15}	0.0
3.5	4.7×10^{18}	0.0	0.0	0.0	0.0
Particulates					
0.4	5.0×10^{17}	4.4×10^{17}	4.3×10^{17}	4.1×10^{17}	2.2×10^{17}
0.8	8.2×10^{18}	6.4×10^{18}	4.8×10^{18}	9.5×10^{17}	1.1×10^{17}
1.3	2.6×10^{18}	1.7×10^{18}	1.3×10^{18}	8.2×10^{16}	1.8×10^{15}
1.7	1.1×10^{18}	5.5×10^{17}	4.1×10^{17}	1.0×10^{17}	2.2×10^{16}
2.2	3.9×10^{18}	3.4×10^{18}	3.0×10^{18}	2.9×10^{17}	1.3×10^{15}
2.5	4.3×10^{17}	2.3×10^{17}	1.7×10^{17}	1.8×10^{16}	2.0×10^{15}
3.5	4.6×10^{17}	9.9×10^{16}	2.9×10^{16}	3.5×10^{14}	9.2×10^{13}

TABLE 12.1-8

ACCIDENT SOURCE STRENGTH IN RESIDUAL HEAT REMOVAL LOOP (MeV/cm^3/s)

Gap Release Accident

Gamma Energy (MeV/ϒ	Time After Release: 0 hour	1 hour	2 hours	8 hours	1 day	1 week
0.4	1.7×10^7	1.5×10^7	1.4×10^7	1.2×10^7	1.1×10^7	1.0×10^7
0.8	1.4×10^8	1.2×10^8	1.1×10^8	8.3×10^7	7.4×10^7	5.2×10^7
1.3	9.1×10^6	6.6×10^6	4.9×10^6	7.7×10^5	5.2×10^4	1.2×10^2
1.7	5.2×10^6	3.7×10^6	2.7×10^6	4.4×10^5	1.9×10^4	3.1×10^2
2.2	4.9×10^6	3.9×10^6	3.2×10^6	6.0×10^5	1.9×10^5	----
2.5	1.8×10^6	1.3×10^6	8.8×10^5	1.4×10^5	7.6×10^3	----
3.5	4.8×10^5	3.0×10^5	2.1×10^5	2.7×10^4	1.1×10^2	----

TABLE 12.1-9

AREA MONITOR ALARM SETPOINTS

Channel	Area Monitor	Alarm Level (R/h)[a]
R-1A (Unit 1 only)	Control room	0.75×10^{-3}
R-1B (Unit 2 only)	Technical support center	0.75×10^{-3}
R-2	Containment	At power 90×10^{-3}; after shutdown 20×10^{-3}
R-3	Radiochemistry laboratory	2.0×10^{-3}
R-4	Charging pump room	Inside room 50×10^{-3}
R-5	Spent-fuel building	2.0×10^{-3}
R-6	Sampling room	15.0×10^{-3}
R-7	Incore instrumentation area	50.0×10^{-3}
R-8	Drumming station	15.0×10^{-3}
R-9 (Unit 2 only)	Sample panel room	15.0×10^{-3}
R-27A and B	Containment high radiation monitor	50.0

a. These setpoints are typical of those anticipated during initial plant operation and are subject to change during the life of the plant. Actual setpoints are incorporated in plant procedures.

TABLE 12.1-10 (SHEET 1 OF 3)

TYPICAL DATA FOR OPERATING PLANTS

	MWe	man-rem (1979)[a]
Reactor 1	800	132
Reactors 2 and 3	810@	805
Reactors 4 and 5	1044/1100	718
Reactor 6	197	495
Reactor 7	906	30
Reactors 8 and 9	0/859	1279
Reactor 10	911	636
Reactor 11	772	154
Reactor 12	802	472
Reactor 13	898	449
Reactors 14, 15, and 16	860@	1001
Reactor 17	873	126
Reactors 18 and 19	775@	3584
Reactor 20	788	1170
Average		553

Component

Inservice Inspection Doses

	man-rem
Reactor head	1.1
Reactor vessel	7.1
Pressurizer	1.66
Steam generator	2.275
Reactor coolant piping	3.235
Total	15.37

TABLE 12.1-10 (SHEET 2 OF 3)

Component Radiation Dose Rates (mrem/h)

	Reactor A	Reactor B	FNP[b]
Primary loop piping (outside)	200-350	50	200-300
Primary loop piping (inside)	1000-13,000		800-10,000
Steam generator plenum	15,000 (inside shutdown)	10,000 (outside, operating)	12,000-15,000 (inside, shutdown)
Reactor vessel head (outside)	400-500		250-300
Reactor vessel head (inside)	15,000		25,000
Reactor vessel nozzles (outside)	15-20		50-150
Reactor vessel nozzles (inside)	5000		-

Component Radiation Dose Rates (mrem/h)

	Reactor J	Reactor K	Reactor L	FNP[b]
Seal water filter	165,000	140	950	1000-2000
Letdown ion exchanger	>1,000,000	800	100,000	1,000,000 (before depletion)
Letdown filter	600,000	1200	50,000	25,000-50,000
Primary drain tank	100-500	35		
Liquid waste monitor tank		2	20	10
Compactor		3		3-5
Waste gas flash tank		19		20-50
Radwaste process filter		85		60-500
Liquid waste holdup tank		120		100-300
Liquid waste holdup pump (area)		8		10

TABLE 12.1-10 (SHEET 3 OF 3)

	Reactor J	Reactor K	Reactor L	FNP[b]
Charging pump (area)		5		5-10
Seal water heat exchanger		50		50-100
Radiochemistry lab drain tank (area)		140		5
Top of steam generator		1		1
Side of steam generator		5		20
Side of pressurizer		90		30
Top of pressurizer		30		100
Reactor coolant pump		33		30-50
Residual heat removal heat exchanger	500	120	30-50	20-250
Residual heat removal pump (area)		40		50-100
Containment (outside secondary shield)			50-200	10-200 (shutdown)
Boric acid drums	200	200		
Shutdown cooling (area)	100			50-100
Spent fuel pool heat exchanger (area)	30			10
Waste gas storage tank			100	10-50

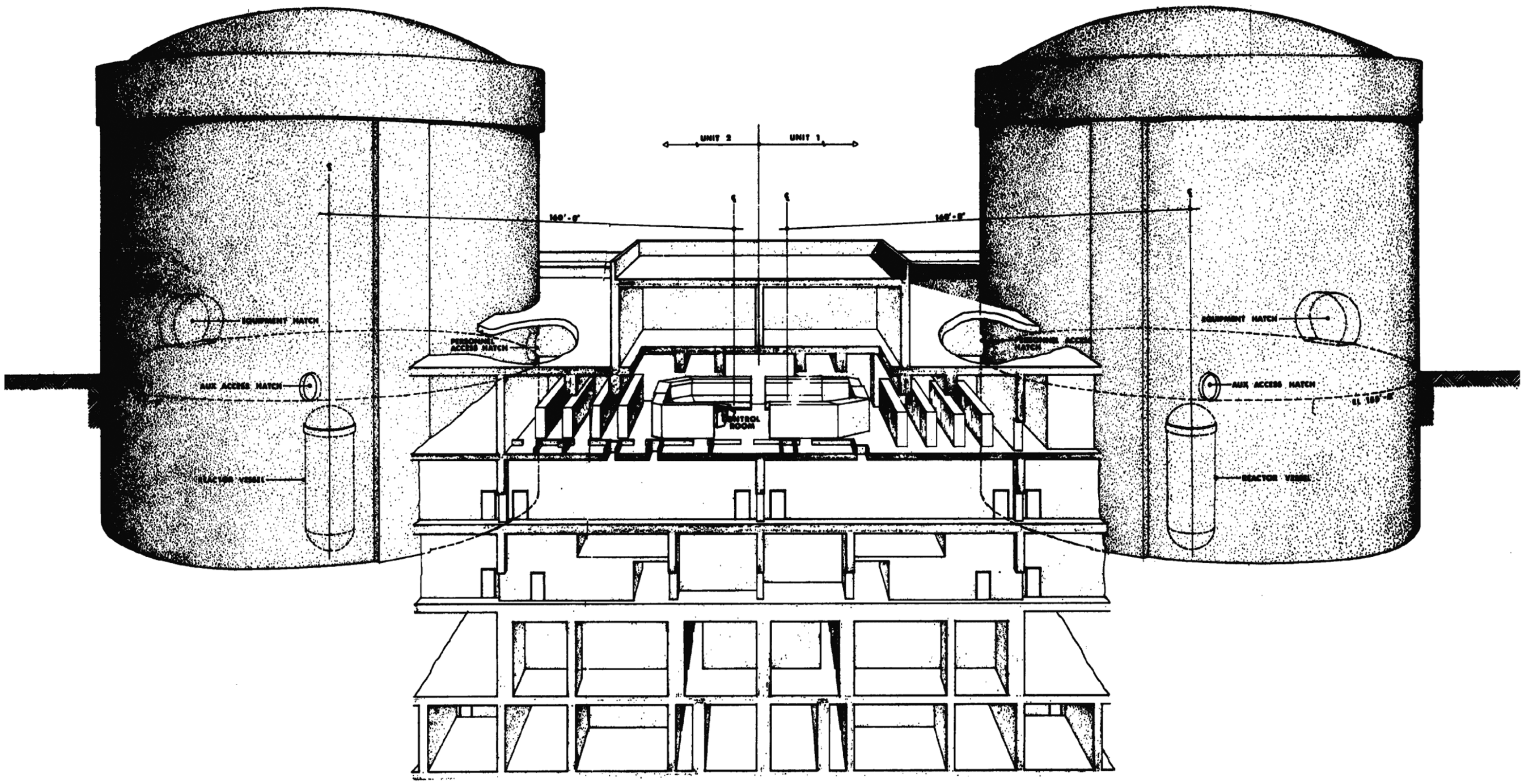
a. Based on NUREG 0713, Volume 1, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1979," Appendix A.

b. FNP Unit 1 (August 1977 through October 1981).

TABLE 12.1-11

EXPECTED ANNUAL MAN-REM PER UNIT BASED ON UNIT 1 OPERATING PLANT DATA

Group	rem/month	rem/year
Operators	5.09	61.1
Maintenance	8.23	98.7
Radiation Protection	4.72	56.6
Engineers	1.0	12.0
Administrators	0.6	7.2
Security	0.1	1.2
Contractors	9.0	108.0
Visitors	Negligible	Negligible
Total	28.74	344.88



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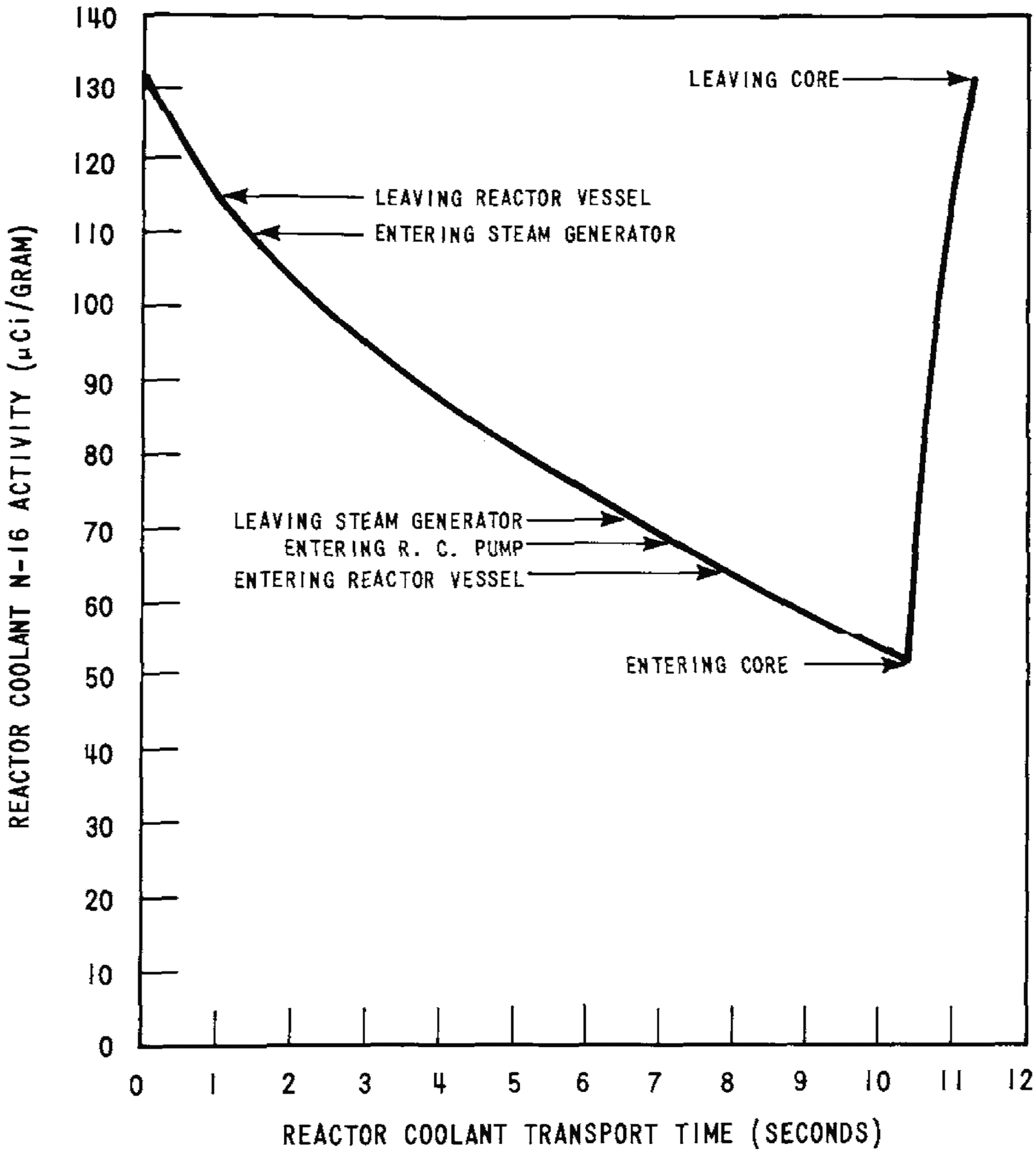
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PERSPECTIVE OF CONTROL ROOM

FIGURE 12.1-1



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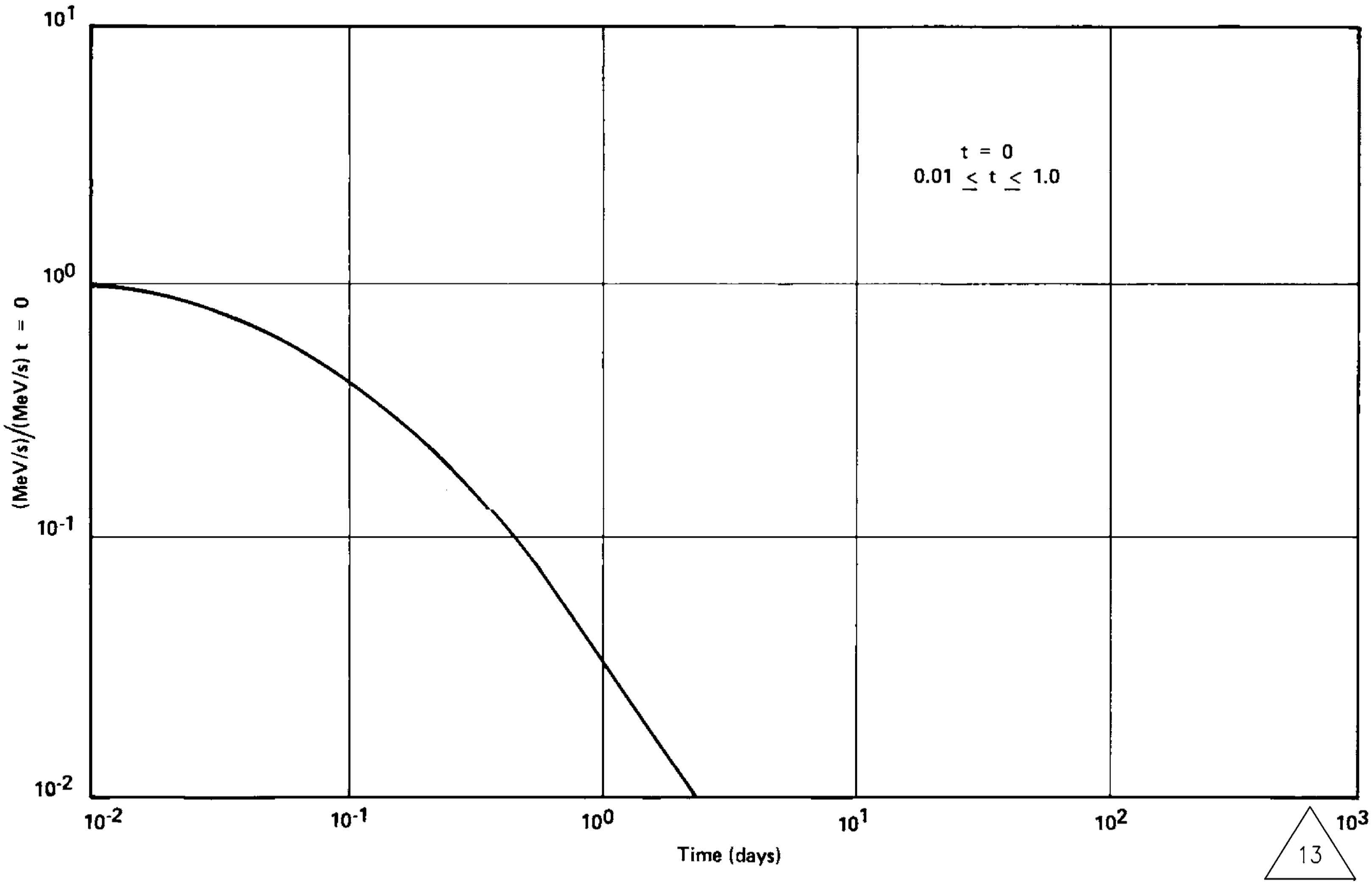
REACTOR COOLANT N-16 ACTIVITY

FIGURE 12.1-2



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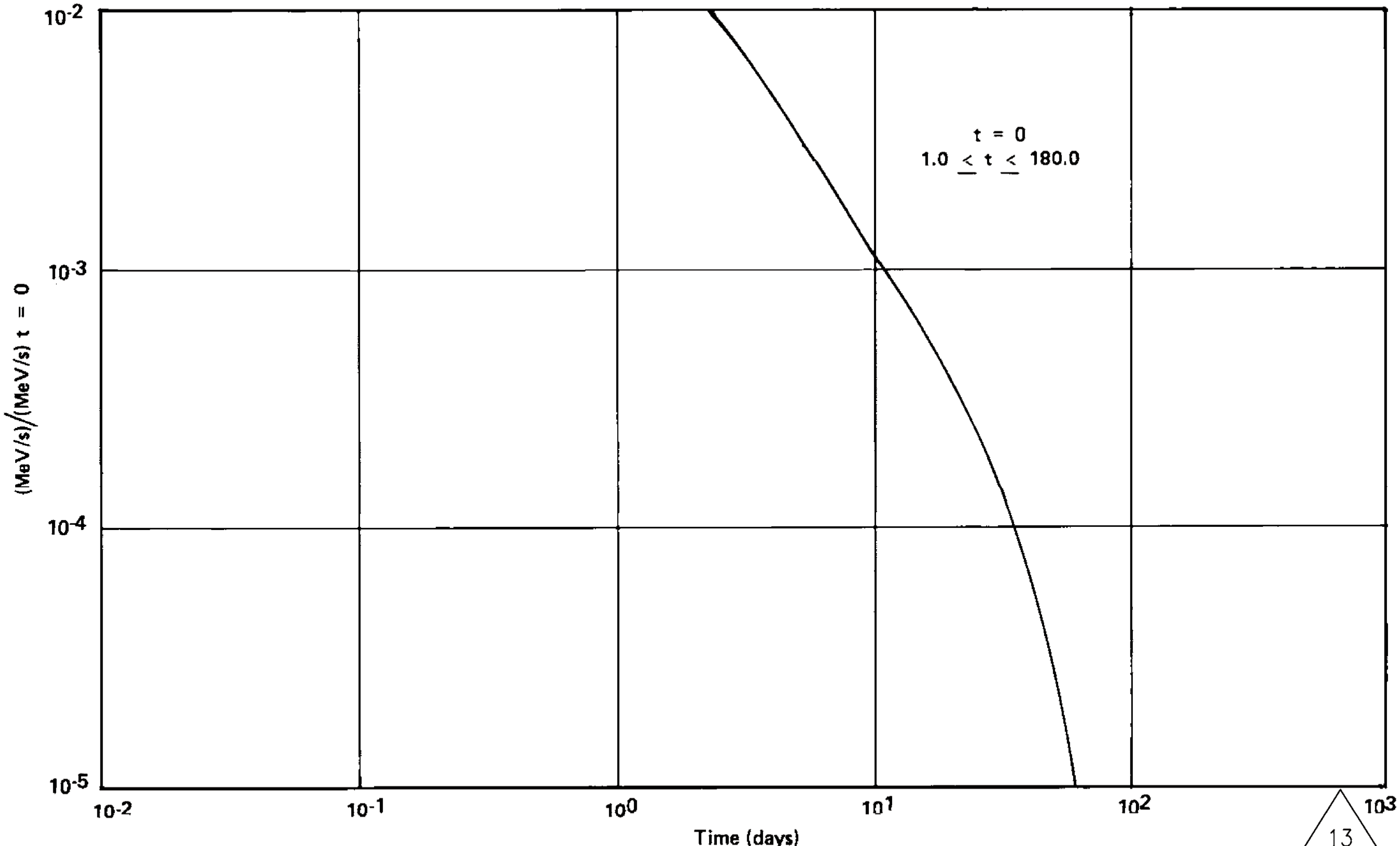
DECAY CURVE FOR SOURCE A, PART 1

FIGURE 12.1-3



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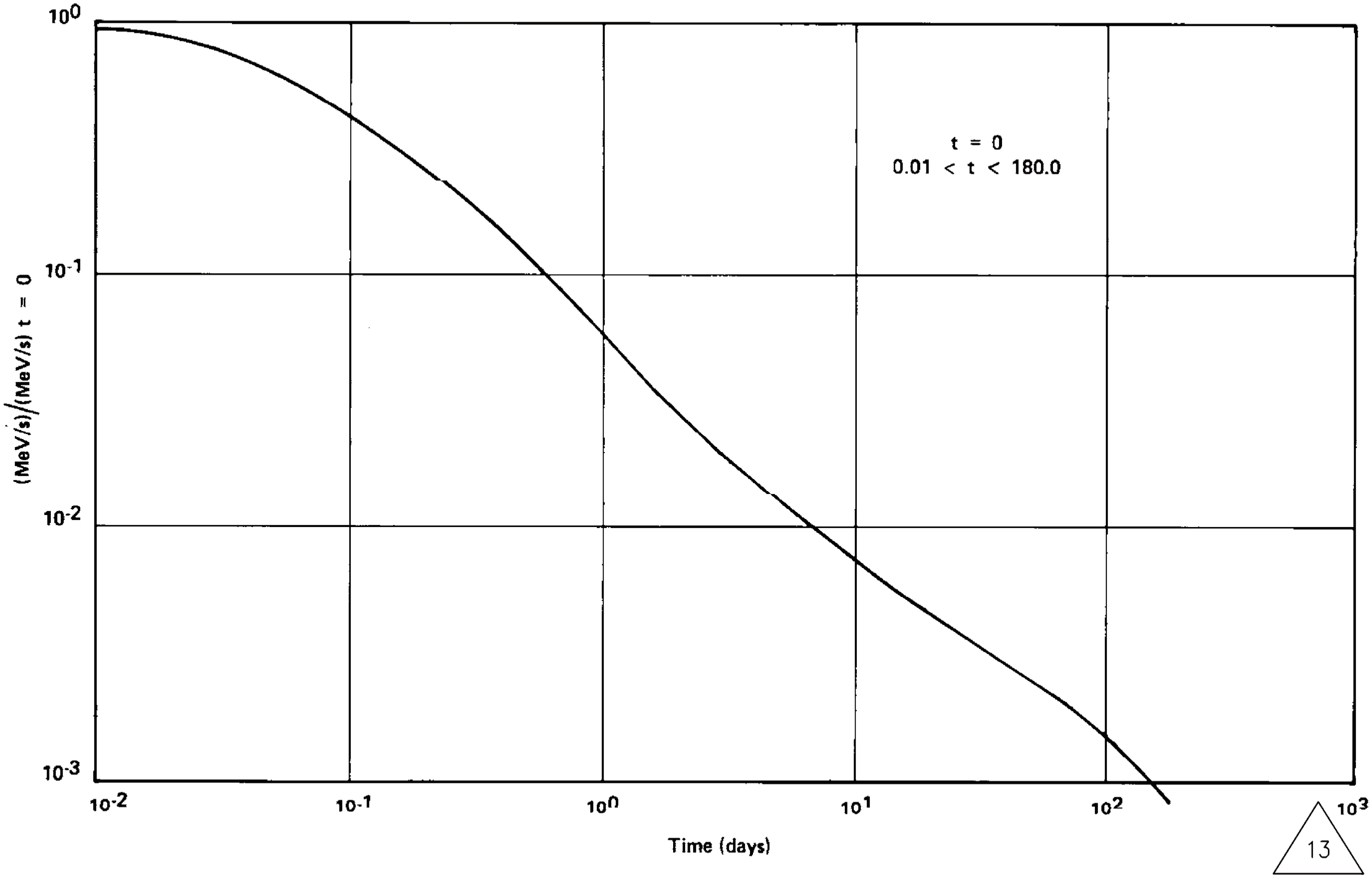
DECAY CURVE FOR SOURCE A, PART 2

FIGURE 12.1-4



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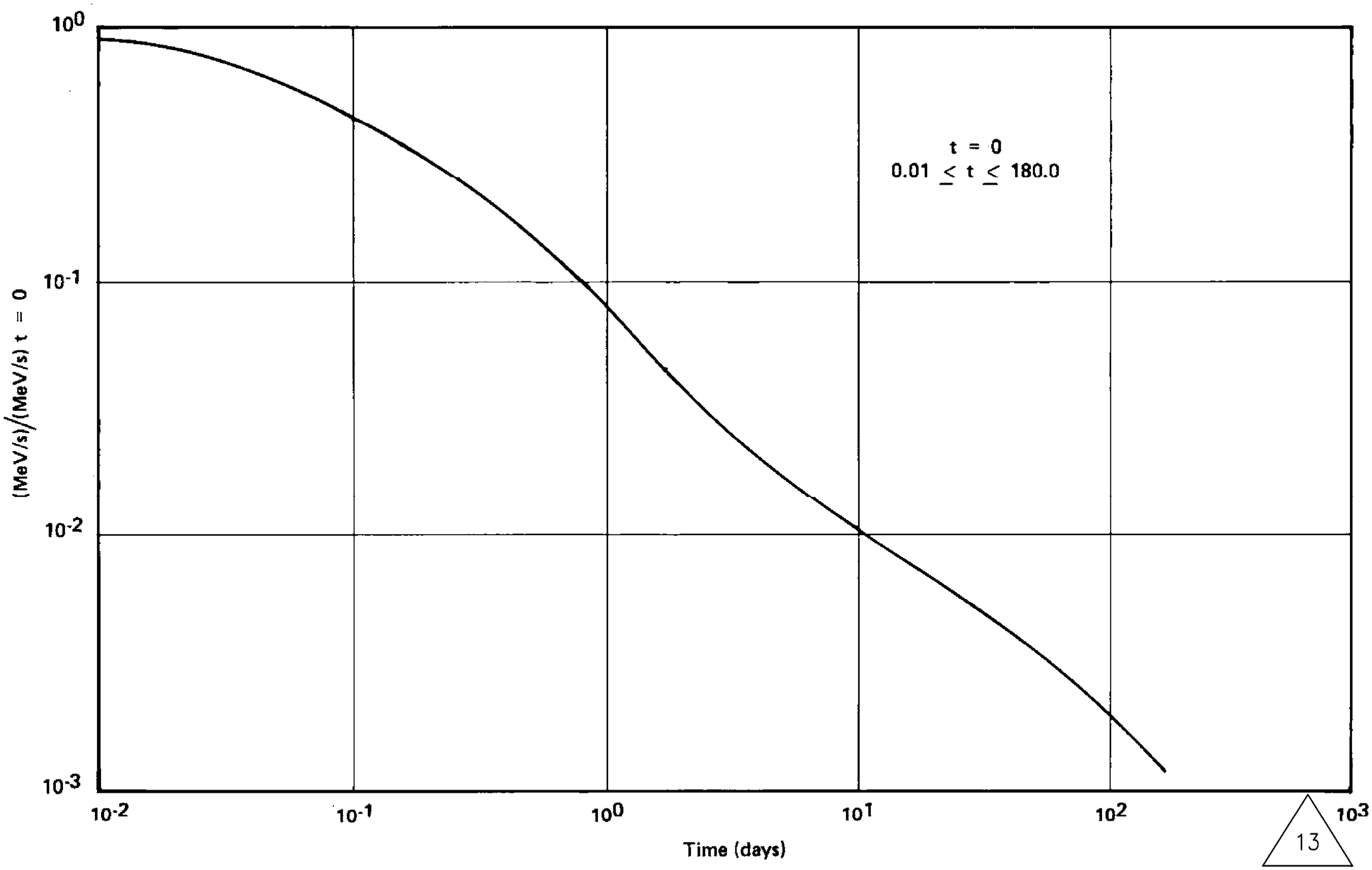
DECAY CURVE FOR SOURCE B

FIGURE 12.1-5



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UNIT 1 AND UNIT 2

DECAY CURVE FOR SOURCE C

FIGURE 12.1-6

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The plant ventilation systems, in addition to their primary function of preventing extreme thermal environmental conditions for operating personnel and equipment, will provide effective protection for operating personnel against possible airborne radioactive contamination in areas where this may occur.

The systems will operate to ensure that the maximum airborne radioactivity levels for normal operation, including anticipated operational occurrences, are within the limits of column 3, Table 1, Appendix B to 10 CFR 20 for areas within plant structures and on the plant site where construction workers and visitors are permitted. The average airborne radioactivity levels meet the requirements of column 3, Table 1, Appendix B to 10 CFR 20 and 10 CFR 50 and, in fact, will be considerably smaller since average coolant inventories and actual equipment leakages will be small.

The systems will operate to ensure compliance with normal operation offsite release limits as discussed in Section 11.3.

The control room ventilation system will also operate to provide a suitable environment for equipment and continuous personnel occupancy in the control room under postaccident conditions in accordance with 10 CFR 50, Appendix A, General Design Criterion 19 and 10 CFR 50.67.

The expected airborne radioactivity levels for normal operations and anticipated operational occurrences, in the containment and auxiliary buildings are presented in Table 12.2-1. The methods used are discussed in subsection 12.2.6. Assumptions used to calculate these airborne radioactivity levels are presented in Table 12.2-2. A discussion of the estimated doses to personnel at the site is also presented in subsection 12.2.6.

12.2.2 DESIGN DESCRIPTION

In order to accomplish the design objectives, certain general design guidelines are followed when possible and applicable:

A. Air movement patterns are provided from areas of lesser contamination potential to areas of progressively greater contamination potential prior to final exhaust.

B. Slightly negative pressures are maintained, where applicable, to prevent uncontrolled exfiltration of contamination. Slightly positive pressure is maintained in the control room to prevent infiltration of potential contaminants.

C. Valves and equipment are maintained as leaktight as possible in order to prevent leakage of radioactive water and subsequent airborne contamination.

D. Individual air supplies are provided for each building in order to keep potentially contaminated airflows separate from noncontaminated air.

E. High efficiency particulate air (HEPA) and charcoal filters are provided on the exhaust side of the radioactive and fuel handling ventilation systems to remove airborne activity and to reduce onsite and offsite radiation levels.

F. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered to prevent contamination of the control room.

These guides are incorporated in the heating and ventilation design described in Section 9.4. The following is a brief summary of those systems.

12.2.2.1 Control Room Ventilation

During normal plant operation, control room air is recirculated through air conditioning units to maintain control room design conditions of temperature and relative humidity. Fresh air makeup is provided by a supply duct from the computer room air conditioning unit. Redundant radiation monitors are provided on the makeup air supply duct. When a high radiation level is sensed by the monitors, a high radiation alarm is actuated in the control room, the air path is isolated to prevent entry of radioactive contaminants, and the control room ventilation system is aligned to the recirculation mode. After isolation of the control room and when conditions permit, fresh air can be brought in manually through redundant control room pressurization charcoal filter systems.

In the event of a loss-of-coolant accident, the control room is automatically sealed, the ventilation system automatically shifts to recirculation, and the pressurization systems automatically actuate to build up a positive internal pressure.

There are two 100-percent capacity control room filtration systems which are designed to recirculate air through charcoal filters following an accident. A complete description of control room ventilation is found in subsection 9.4.1. The control room area volume is 114,000 ft^3 which includes the space above the suspended ceiling. Control room ventilation system components are described in Table 9.4-1.

12.2.2.2 Containment

The containment cooling system consists of recirculating air cooling units to maintain the design containment temperature and relative humidity. A complete description of the containment ventilation system is found in subsection 6.2.3, and containment cooling system components are described in Table 6.2-28. Containment area volume is listed in Table 6.2-1.

12.2.2.3 Auxiliary Building

The auxiliary building for each unit is served by separate ventilation systems for the fuel handling area, the radioactive waste area, and the nonradioactive area. The shared control room is served by two separate and redundant air conditioning systems. A complete description of the auxiliary building ventilation is found in subsection 9.4.2. The area volume of the auxiliary building is 1.02×10^6 ft^3. Components in the system are discussed in detail in subsection 9.4.2.

12.2.2.4 Radwaste Area

Outside air will be filtered, tempered, and delivered to the clean areas such as the lower level corridors. A pressure gradient will be maintained to create airflow from the corridors into the equipment cells, where it will be exhausted after removing airborne contaminants. The exhaust air functions to maintain the area under a negative pressure with respect to the outside.

The volume of the radwaste area is 666,400 ft^3. A complete description of the radwaste area ventilation system is found in subsection 9.4.3, and principal components are described in Tables 9.4-8 and 9.4-9.

12.2.2.5 Turbine Building

The turbine building is provided a recirculating ventilation system which conditions the air for maximum safety and convenience for operating personnel. Passive smoke/heat vents in the turbine building roof allow smoke and heat to exit the turbine building.

The area volume of the turbine building is 4.25×10^6 ft^3. A complete description of the turbine building ventilation system is found in subsection 9.4.4. Heating, cooling, and filtration system component design parameters in the turbine building ventilation system are listed in Table 9.4-11.

12.2.2.6 Maintenance Considerations

12.2.2.6.1 Filter Housings

All nonsafety-related charcoal absorbers are of a vertical, fixed-bed design. Contaminated charcoal is conveyed pneumatically from the filter unit to disposal drums in a closed pipeline, with no necessity for personnel to enter the filter housing. Following removal of charcoal, personnel enter the filter housing and place each HEPA filter and prefilter in individual plastic bags, which are sealed before the filters are removed from the filter housing.

12.2.2.6.2 Temporary Ducting

Temporary ducting is used for pneumatic removal of contaminated charcoal from all filter units having fixed charcoal filters. The ducting is used in a closed and leaktight system, and no airborne radioactivity will be released.

12.2.3 SOURCE TERMS

During reactor operation, airborne activities can originate due to system leakage from the following sources:

A. Reactor coolant leakage to the containment building.

B. Reactor coolant leakage to the auxiliary building.

C. Secondary side system leakage.

D. Waste gas processing system leakage.

A complete identification of all radioactive sources and an estimate of resulting radioactive effluents are further described in Section 11.1.

12.2.3.1 Reactor Coolant Leakage to the Containment Building

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. The leakage is estimated to be 40 lb/day.

12.2.3.2 Reactor Coolant Leakage to the Auxiliary Building

This effluent represents nonrecyclable reactor coolant from system leaks in the auxiliary building. It is assumed that the total amount of leakage is 20 gal/day.

12.2.3.3 Secondary Side Leakage to the Turbine Building

The rate of steam leakage from the secondary system is estimated to be 5 gal/min when condensed.

In addition, liquid leakage from systems operating below 212°F is estimated to be 12.5 gal/min.

12.2.3.4 Waste Gas Processing System Leakage

The gaseous waste processing system is designed to contain the gaseous waste for the lifetime of the plant. However, although all precautions are taken to avoid any leakage from the system, an estimated leakage of 100 sf^3/year is assumed.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

An analysis of the auxiliary building was conducted in order to identify the potential points of releases of airborne radioactive material in the form of contaminated steam or liquid discharges from valves, pumps, tanks, sumps, and other release mechanisms. For plant design, an NRC acceptance criterion, discussed in subsection 12.1.2 of the FNP FSAR Safety Evaluation Report, required concentrations of airborne radioactive material to be controlled such that limits stated in 10 CFR 20 would not be exceeded. In-plant airborne radioactive materials concentration limits that were in effect at the time of plant design are specifically stated in 10 CFR 20.103, which references column 1, Table I of Appendix B to 10 CFR 20.1 - 20.601.

During plant operations, access to the rooms, enclosures, or operating areas containing release points, and having the potential of causing operating personnel to be exposed to airborne radioactive material to an average concentration in excess of the limits specified in Appendix B, Table 1, of 10 CFR 20.1001 - 20.2401, will be controlled by a program of:

A. Surveys or continuous online type of sampling.

B. Clear identification of spaces with appropriate caution signs.

C. Locked doors as appropriate.

D. Administrative controls through the use of radiation work permits and procedures.

Other areas of potential airborne contamination, such as the containment, penetration room, and spent fuel area, are monitored by the fixed airborne radiation monitoring instruments described in Section 11.4. The continuous radiation monitors in these areas will be augmented by the use of periodic portable air activity samplers.

The samplers will be used as a check on the fixed monitoring system during normal and maintenance operations and to determine airborne activity levels should an accident occur or after receipt of an alarm from the fixed monitoring system. Systems such as the plant vent air particulate monitor system and the plant vent gas monitor described in subsection 11.4.2 will be checked using grab samples.

The results of these checks will be logged and filed as part of the plant records.

12.2.4.1 Containment Airborne Radioactivity Monitoring

Two Kr-85 radiation monitors are provided in the containment purge exhaust ductwork. The monitors are capable of measuring and alarming 1 MPC-h of Kr-85 when operating in a background of 2 mR/h of 1 MeV gamma rays.

Radiation measurements from these monitors are included in the administrative controls utilized to evaluate the containment radioactivity levels prior to permitting personnel to enter the containment.

The location of the radiation monitors are shown in drawings D-175010, sheet 2, and D-205010, sheet 2.

12.2.4.2 Spent-Fuel Area Airborne Radioactivity Monitoring

The spent-fuel area is continuously exhausted to the plant vent during plant operation. The exhaust flow is continuously monitored for high radiation by two Kr-85 gas monitors capable of alarming when a level of 1 MPC-h is reached.

Dilution factors were not considered in the analysis since representative gaseous samples will exist in the exhaust ductwork over the period of 1 h, due to dispersion within the spent-fuel area.

The radiation monitors have a sensitivity of 5×10^{-7} $\mu Ci/cm^3$ in a background of 2 mR/h of 1 MeV gamma rays.

The location of the radiation monitors is shown in drawings D-175045 and D-205045.

12.2.4.3 Penetration Room Airborne Radioactivity Monitoring

Access to the penetration room will be on an as-required basis for maintenance or repair of equipment. Manual samples will be taken routinely with portable sampling equipment to permit personnel to enter the penetration room compartments as required.

12.2.5 OPERATING PROCEDURES

12.2.5.1 General

The radiation protection group is responsible for developing a radiation protection program which will ensure that inhalation exposure is kept as low as is reasonably achievable, consistent with 10 CFR 20.1101 and 20.1701 - 20.1704.

12.2.5.2 Procedures

Inhalation exposure will be minimized during operations and maintenance by using the following procedures and techniques to determine and cope with the hazards present:

A. Monitoring

Air samplers of various flowrates will be used to collect particulates on high efficiency filter media for subsequent counting. For tritium analysis, freeze-out methods may be used to obtain samples for counting.

Routine smear surveys will be performed to establish the levels of removable contamination throughout the plant so that personnel protection measures or decontamination may be effected.

Assay of noble gases will be performed by drawing an air sample into a sample container and analyzing it on a multichannel analyzer system.

B. Respiratory Protection

In areas where airborne radioactivity can cause exposures in excess of that allowed by 10 CFR 20.1201 - 20.1207 respiratory devices and/or portable HEPA filtration systems may be required. It is the responsibility of the radiation protection group to monitor such areas, to establish the requirement for respiratory equipment, and to control access to such areas through the radiation work permit program.

Each individual who enters a radiation controlled area will be trained or briefed in accordance with 10 CFR 19.12. A notice describing where radiation control procedures may be examined is posted in accordance with 10 CFR 19.11.

12.2.6 ESTIMATES OF INHALATION DOSES

Peak airborne radioisotopic concentrations in the different buildings of operating pressurized water reactor (PWR) plants have shown that these concentrations are insignificant for PWR plants. The inhalation doses to plant personnel at these plants have been found to be negligible.

The doses to plant personnel and construction workers from airborne radioactivity will depend upon the extent of their occupancy and the time when this occupancy occurs. These doses will be controlled by limiting personnel occupancy in the contaminated areas and by provision of respiratory protection equipment if required. The highest dose to plant personnel will therefore be limited to the maximum permissible dose for occupationally exposed individuals, as specified by 10 CFR 20.1201 - 20.1207.

The assumptions used to estimate concentrations and inhalation doses in the containment, turbine building, and certain regions within the auxiliary building are listed in Table 12.2-2. The airborne peak concentrations in each of the regions mentioned above are given in Table 12.2-1 In addition, the table gives the Derived Air Concentrations for airborne activity in these areas as defined in column 3, Table 1, of Appendix B to 10 CFR 20.

The annual inhalation doses to plant personnel due to the airborne radioisotopes in each of the above mentioned regions are presented in Table 12.2-3.

TABLE 12.2-1

PEAK AIRBORNE RADIOISOTOPIC CONCENTRATIONS IN THE DIFFERENT REGIONS OF THE PLANT

	Turbine Building		Containment @ Power		Containment @ Refueling		Waste Gas Processing Area		Waste Monitor Tank Rooms		Radwaste Area	
Isotope	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)[1]	Concentration (µCi/cc)	DAC for 4 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 2 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 2 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)
Kr-83m	8.03E-10	1.00E-02	0	1.00E-01	0	1.00E-02	0	2.00E-01	0	2.00E-01	0	1.00E-02
Kr-85	5.35E-10	1.00E-04	2.49E-08	1.00E-03	2.56E-09	1.00E-04	4.08E-06	2.00E-03	2.68E-07	2.00E-03	1.32E-09	1.00E-04
Kr-85m	6.85E-09	2.00E-05	1.20E-07	2.00E-04	5.01E-12	2.00E-05	6.24E-08	4.00E-04	3.86E-06	4.00E-04	1.89E-08	2.00E-05
Kr-87	1.77E-09	5.00E-06	2.73E-08	5.00E-05	3.79E-16	5.00E-06	5.76E-09	1.00E-04	2.47E-06	1.00E-04	1.22E-08	5.00E-06
Kr-88	1.01E-08	2.00E-06	1.48E-07	2.00E-05	8.95E-13	2.00E-06	6.24E-08	4.00E-05	7.29E-06	4.00E-05	3.60E-08	2.00E-06
Xe-131m	1.34E-09	4.00E-04	3.62E-08	4.00E-03	0	4.00E-04	0	8.00E-03	0	8.00E-03	0	4.00E-04
Xe-133	3.86E-07	1.00E-04	1.32E-05	1.00E-03	1.21E-07	1.00E-04	1.82E-05	2.00E-03	1.54E-04	2.00E-03	7.54E-07	1.00E-04
Xe-133m	8.43E-09	1.00E-04	2.25E-07	1.00E-03	8.50E-10	1.00E-04	2.11E-07	2.00E-03	2.90E-06	2.00E-03	1.44E-08	1.00E-04
Xe-135	2.49E-08	1.00E-05	4.97E-07	1.00E-04	1.49E-10	1.00E-05	5.28E-07	2.00E-04	1.07E-05	2.00E-04	5.31E-08	1.00E-05
Xe-135m	9.55E-11	9.00E-06	1.15E-09	9.00E-05	4.69E-36	9.00E-06	0	1.80E-04	3.86E-07	1.80E-04	1.89E-09	9.00E-06
Xe-138	2.93E-10	4.00E-06	2.94E-09	4.00E-05	5.86E-33	4.00E-06	0	8.00E-05	1.29E-06	8.00E-05	6.51E-09	4.00E-06
I-131	4.69E-11	2.00E-08	4.23E-09	2.00E-07	1.97E-14	2.00E-08	1.10E-10	4.00E-07	2.44E-10	4.00E-07	1.97E-11	2.00E-08
I-132	4.34E-12	3.00E-06	3.11E-10	3.00E-05	6.72E-19	3.00E-06	4.56E-13	6.00E-05	8.79E-11	6.00E-05	4.78E-12	3.00E-06
I-133	3.96E-11	1.00E-07	4.90E-09	1.00E-06	2.41E-15	1.00E-07	1.73E-11	2.00E-06	3.92E-10	2.00E-06	2.09E-11	1.00E-07
I-134	2.14E-13	2.00E-05	9.03E-11	2.00E-04	1.23E-23	2.00E-05	9.60E-14	4.00E-04	5.86E-11	4.00E-04	2.98E-12	2.00E-05
I-135	9.20E-12	7.00E-07	1.60E-09	7.00E-06	1.50E-16	7.00E-07	3.36E-12	1.40E-05	2.22E-10	1.40E-05	1.07E-11	7.00E-07

1. Derived Air Concentrations for 40 hours/week occupancy are from 10 CFR Part 20, Appendix B, Table 1, Column 3; other DACs are multiples of these values.

TABLE 12.2-2 (SHEET 1 OF 2)

ASSUMPTIONS USED TO ESTIMATE PEAK AIRBORNE CONCENTRATIONS AND INHALATION DOSES

Leak Rates (lb/day)	
Steam generator tube leak (primary coolant)	166.9
Leak into containment (primary coolant)	40
Leak into auxiliary building (primary coolant)	166.9
Steam leak into turbine building	6×10^4
Liquid leak into turbine building	1.5×10^5
Leak from waste gas processing system	100 sf^3/year
Partition Factors or Ratio of Liquid Activity to Airborne Activity (iodines)	
Steam generator	100
Air ejector	10,000
Liquid leakage to turbine building	100
Liquid leakage to auxiliary building	100
Containment building, primary coolant leakage	100
Leakage from waste gas processing system (partition in the volume control tank)	100
Ventilation (ft^3/min)	
Exhaust rate from turbine building	5000
Flowrate for recirculation in the turbine building	11,500
Containment purge rate (Main/Mini)	25,000/2500
Preaccess filter system in containment, flowrate for recirculation	20,000
Exhaust rate from waste gas processing region of radwaste area	3500
Exhaust rate from waste monitor tank rooms (region of radwaste area containing waste holdup, floor drain, and waste monitor tanks)	660
Exhaust rate from radwaste area (excluding the waste gas processing region and waste monitor tank rooms)	4.90×10^4
Filter Efficiency (percentage)	
Halogen recirculation filter efficiency in containment	90

TABLE 12.2-2 (SHEET 2 OF 2)

Occupancy in the Regions (h/week; weeks/year)

Turbine building	40; 50[(a)]
Waste gas processing area	2; 50[(a)]
Waste monitor tank rooms	2; 50[(a)]
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	40; 50[(a)]
Containment during refueling or shutdown purge	40; 4

Volumes of the Regions (ft^3)

Turbine building	4.25×10^6
Containment	2.05×10^6
Waste gas processing region	38,000
Waste monitor tank rooms	8400
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	6.2×10^5

Other Factors

Fuel defects (percent)	0.25
Plant load factor (percent)	100
Duration of the containment purge, hot shutdown (h)	24
Duration of the preaccess filter operation (h)	40[(b)]
Duration of the containment refueling shutdown purge (h)	8

a. Full occupancy a year means 50 weeks/year for plant load factor 1.0; 40 weeks/year for plant load factor 0.8.

b. For refueling or shutdown purge, recirculation through preaccess filters will be for a total of 16 + 8 = 24 h. Purging at power is done by a constant 2500-ft^3/min flow through the minipurge system.

TABLE 12.2-3

INHALATION DOSES DUE TO AIRBORNE RADIOISOTOPES

Region	Occupancy	Lung Dose (rem/year)	Thyroid Dose (rem/year)
Turbine building	40 h/week for 50 weeks/year[a]	6.13×10^{-4}	0.141
Containment during hot shutdown purges	4 h/purge for 3 purges/year	3×10^{-3}	0.144
Containment during refueling or shutdown purge	40 h/week for 40 weeks/year	3.56×10^{-9}	1.56×10^{-6}
Containment total		3×10^{-3}	0.145
Auxiliary building waste gas processing region	2 h/week for 50 weeks/year[a]	3.95×10^{-5}	1.48×10^{-2}
Auxiliary building waste monitor tank rooms	2 h/week for 50 weeks/year[a]	2.74×10^{-4}	4.14×10^{-2}
Auxiliary building radwaste area excluding waste gas processing region and waste monitor tank rooms	40 h/week for 50 weeks/year[a]	3.26×10^{-4}	6.14×10^{-2}

a. For plant factor 1.0, occupancy is 50 weeks/year; for plant factor 0.8, occupancy is 40 weeks/year equivalent.

12.3 RADIATION PROTECTION PROGRAM

12.3.1 PROGRAM OBJECTIVES

12.3.1.1 Objectives

It is the objective of the Farley Nuclear Plant Radiation Protection program to provide effective radiation protection for plant personnel and visitors during operations, maintenance, refueling, and emergencies, and further to keep exposures as low as reasonably achievable (ALARA). The Radiation Protection group is responsible for developing and administering such a program consistent with 10 CFR 20, Standards for Protection Against Radiation, paragraph 20.1101.

12.3.1.2 Organization

The Radiation Protection group consists of a radiation protection manager, radiation protection support superintendent, radiation protection operations superintendent, plant health physicist, radiation protection supervisors, and radiation protection technicians. Overall responsibility for plant operation lies with the plant manager, but the responsibility for radiation protection operations is delegated to the radiation protection manager. (See figure 13.1-6.)

The radiation protection manager reports to the plant manager and is responsible for keeping him informed at all times of radiation hazards and conditions related to potential exposure, contamination of plant equipment, or contamination of site and environs. As the administrator of the radiation protection program, the responsibilities of the radiation protection manager include:

A. Training and supervising the radiation protection technicians, supervisors, health physicist, and radiation protection superintendents.

B. Planning and scheduling radiation protection coverage and surveillance activities.

C. Establishing and maintaining data on plant radiation and contamination levels, personnel exposures, and work restrictions.

D. Writing and maintaining current radiation protection procedures which incorporate the provisions of Regulatory Guides 8.8, Revision 3, and 8.10, Revision 1 (this should include radiation protection reviews of appropriate design changes).

E. Ensuring that plant operations comply with 10 CFR 20.1001 - 20.2401.

F. Advising the emergency director during emergencies involving radiological hazards.

G. Advising other group supervisors with regard to dose equalization among their personnel. Personnel will be rotated insofar as practical for uniformity of occupational radiation exposure within each group.

H. Managing the shipment and disposal of all solid radwaste.

I. Supervising the ALARA program.

To carry out the responsibilities of the radiation protection manager, the Radiation Protection group is organized to:

A. Perform radiation monitoring for plant operations and maintenance activities as required and maintain records of all surveys performed.

B. Establish and maintain a radiological surveillance program to collect and document data concerning radiation and contamination levels throughout the plant and on the plant site.

C. Make plant personnel aware of radiological conditions by posting areas throughout the plant based on radiation and contamination levels.

D. Provide and maintain protective clothing and respiratory equipment for plant operation and maintenance and instruct plant personnel in their use.

E. Recommend procedures for dealing with radiation hazards in performing day-to-day operation and maintenance and verify effectiveness of such procedures.

F. Specify dosimetry requirements for radiation work.

G. Assist in the plant training program by providing specialized training in radiation protection when necessary to support the training group.

H. Make recommendations and assist in performing equipment, area, and personnel decontamination.

I. Assist with the receipt and shipment of radioactive materials to ensure compliance with Federal and State regulations.

J. Ensure that radiation protection equipment designated for service is operational and calibrated.

K. Establish and implement an active ALARA program.

12.3.1.3 Personnel Qualification and Training

The radiation protection manager will have qualifications equivalent to those in Regulatory Guide 1.8, Revision 1, September 1975 (Personnel Selection and Training).

Each permanent plant employee who is classified as a radiation worker is required to attend radiation protection training, the depth of which will depend on the work assignments, individual responsibilities, and the degree of radiation hazard anticipated. Personnel whose duties entail entering restricted areas are required to attend training and to demonstrate a minimum level of knowledge prior to being permitted unescorted access to restricted areas.

Requiring individuals to demonstrate a minimum level of knowledge in radiation protection ensures that each individual is qualified to perform his duties safely. Further, plant supervisory personnel are advised to screen their employees with respect to conscientiousness and responsibility in performing their duties with regard for approved radiation protection procedures.

12.3.1.4 Plans and Procedures

To ensure that the internal and external occupational exposure is kept ALARA during the activities of maintenance, inspection, refueling, and nonroutine as well as routine operations, radiation protection procedures are utilized. When practical, these procedures covering the areas listed below are developed in accordance with Regulatory Guide 8.8, Revision 3, Information Relevant to Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable, and Regulatory Guide 8.2, Revision 0, Guide for Administrative Practices in Radiation Monitoring.

A. Personnel monitoring.

B. Personnel, equipment, and area decontamination.

C. Access to controlled areas.

D. Use and cleaning of protective clothing.

E. Use of respiratory protection equipment.

F. Air and liquid sampling.

G. Radiation work permit applicability and use.

H. Portable and fixed radiation protection equipment calibration.

I. Area and process radiation monitoring calibration.

J. Receipt and shipment of radioactive materials.

K. Leak testing of radioactive sources in accordance with the Technical Requirements Manual.

12.3.2 FACILITIES AND EQUIPMENT

12.3.2.1 Facilities

The radiation protection facilities consist of a radiation protection office, briefing room, survey preparation room, and material and personnel frisking room located in the auxiliary building access control area at el 155 ft.

Clothing issue rooms, a calibration lab, a respirator issue room, a decontamination room, a drumming room, and a nuclear laundry are located in the auxiliary building radiation controlled area at el 155 ft. The radiation protection office is located near the boundary between the clean area and the radiation controlled area so that radiation protection services and decontamination may be conveniently provided to those who enter or leave this area. Personnel decontamination can be performed in the hot toilet rooms which are conveniently located adjacent to the radiation protection office. Radiation controlled area entry is through an administratively controlled one-way door. Prior to leaving the radiation controlled area, one passes through a portal monitor or uses friskers near the radiation protection office.

12.3.2.2 Shielding and Handling Methods

Shielding (e.g., lead, tungsten) in various forms, such as bricks, blankets, or sheets will be available for use as portable shielding. (A safety evaluation checklist must be completed before shielding can be applied to any safety-related equipment or systems.)

A radiation work permit will be employed as the principal means of ensuring that proper precautions are taken and that adequate planning is effected before work is performed in any area that presents a real or potential radiological hazard. Prior to a worker's entry into an area in which the radiological conditions are unknown, a survey is made and a radiation work permit completed which lists the radiation protection requirements for the particular work to be accomplished.

Other handling methods and procedures for keeping external and internal exposures ALARA are discussed in subsections 12.1.5 and 12.2.5.

12.3.2.3 Respiratory Equipment

Respiratory equipment will be available for use in areas in which airborne radioactive material exceeds those concentrations given in Table 1, Appendix B to 10 CFR 20.1001 - 20.2401. Typical respiratory devices which will be made available at the plant include the following:

A. Full-face masks with high efficiency particulate and charcoal filters.

B. Full-face masks with air line.

C. Hoods and suits with air line.

D. Full-face masks with self-contained breathing apparatus.

E. Full-face masks with battery-powered high efficiency particulate filters.

The respiratory protection program is designed to comply with 10 CFR 20.1701 - 20.1704. Respiratory equipment is selected and protection factors are assigned in accordance with 10 CFR 20.1001 - 20.2401, Appendix A. An exemption from 10 CFR 20 which allows the use of a protection factor for radioiodine has been granted to Farley Nuclear Plant by the NRC. Any changes to the October 23, 1984, NRC exemption will be incorporated into the program.

12.3.2.4 Protective Clothing

Protective clothing will be required in contaminated areas. Typical protective clothing that will be made available at the plant is listed below:

A. Coveralls.

B. Laboratory coats.

C. Plastic suits.

D. Canvas caps.

E. Hoods.

F. Shoe covers.

G. Booties.

H. Gloves.

12.3.2.5 Portable Instrumentation

The majority of the portable radiation protection instrumentation will be located in the auxiliary building near the radiation protection office or the radiation protection calibration laboratory. For purposes of emergency monitoring, instruments will be kept at various places as designated by emergency preparedness procedures. A listing and description of some of the portable radiation protection instruments are given in table 12.3-1.

The Radiation Protection group will be responsible for writing and implementing procedures for the use and calibration of this equipment. Detailed records on the maintenance and calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the National Institute of Standards and Technology (NIST) or other reputable vendors and/or using reference instruments having calibrations traceable to the NIST. In addition, reputable vendors will be used to calibrate and perform maintenance on some of the portable instruments. Vendors will implement their own calibration procedures but are subject to Southern Nuclear Operating Company (SNC) quality assurance requirements. Calibrations and preventive maintenance on portable radiation protection instrumentation will be performed semiannually or when required. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration.

12.3.2.6 Laboratory Equipment

Major fixed laboratory instrumentation will generally be located at the radiation controlled area exit and the radiation protection counting room, but use is not limited to these areas. A listing of typical equipment, including location and description, is given in table 12.3-1 and table 12.3-2.

The Radiation Protection group will be responsible for writing and implementing procedures for the use and calibration of equipment. Detailed records on the calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the NIST or other reputable vendors and/or using reference instruments having calibration traceable to the NIST. In addition, reputable vendors may be used to calibrate and perform maintenance on fixed laboratory instrumentation. Vendors will implement their own calibration procedures but are subject to SNC quality assurance requirements. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration. The equipment and instrumentation listed in tables 12.3-1 and 12.3-2 are typical of the devices which will be purchased.

12.3.3 PERSONNEL DOSIMETRY

Where applicable, the personnel dosimetry program will be developed in accordance with Regulatory Guide 8.4, Revision 0, Direct Reading and Indirect Reading Pocket Dosimeters, and Regulatory Guide 8.13, Revision 2, Instructions Concerning Prenatal Radiation Exposure.

12.3.3.1 External Dosimetry

Plant employees, visitors, support personnel, and construction workers will be required to wear one or more personnel dosimeters when they enter the radiation control area if they are likely to receive, in 1 calendar year, from sources external to the body, a dose in excess of 10 percent of the limits in 10 CFR 20.1201(a). A third party may be used or a complete in-house program may be implemented for processing dosimetry badges (e.g., OSLDs).

To minimize congestion during outages and other peak activity periods, issuance and storage of contractor dosimetry badges and subsequent monitoring may be established at an alternate location other than the primary access point.

Personnel dosimetry used at the plant will include a dosimetry badge and either a digital alarming dosimeter or pocket ion chamber. The dosimetry badge must be sensitive to beta-gamma radiation and the dosimeter must be sensitive to gamma radiation. The dose received on dosimeters and dosimetry badges will be tracked by plant personnel. Extremity dosimeters will be issued on a case-by-case basis, and neutron dosimetry will be accomplished by setting dose rates and time keeping, which must be performed by a qualified individual, or by issuing neutron dosimetry.

12.3.3.2 Internal Dosimetry

Whole body counting and bioassay will be used to supplement the dosimetry program. If internal dose assessment is deemed necessary, calculations will meet the intent of Regulatory Guide 8.34.

12.3.3.3 Records

Exposure data of all personnel will be collected and recorded on form NRC-5, Occupational Exposure Received for a Monitoring Period, or the equivalent. Occupational exposures incurred by individuals prior to working at the Farley Nuclear Plant, bioassay data, and whole body counting data will be summarized on form NRC-4, Cumulative Occupational Exposure History, or the equivalent. Records retained on form NRC-4 or its equivalent will be retained until the license is terminated. The records used in preparing form NRC-4 or its equivalent will be kept for 3 years, after which time they may be disposed of. Exposure data recorded on form NRC-5 or its equivalent will be retained until the license is terminated.

TABLE 12.3-1

PORTABLE AND SEMIORTABLE RADIATION PROTECTION INSTRUMENTS

Instrument	Radiation Detected	Range	Accuracy	Number	Location	Remarks
GM survey meter (portable)	Beta, gamma	0-70,000 cpm 0-50 mR/h	±10% full scale	15	Instrument locker	Equipped with end window, side window, or pancake probe
GM survey meter (semiportable)	Beta, gamma	0-500,000 cpm	±10% full scale	10	Various areas in plant	Equipped with pancake probe for smear checks and personnel frisking
GM survey meter	Gamma	0-1000 R/h	±25% full scale	2	Instrument locker	Extendible probe
Neutron survey meter	Thermal through fast neutrons	0-5000 mR/h	±10% full scale	2	Instrument locker	Detection of neutrons up to 10 MeV
Ion chamber survey meter	Beta, gamma	0-20,000 R/h	±10% full scale	2	Instrument locker	Ionization chamber
Proportional alpha counter	Alpha	0-500,000 cpm	±10% full scale	2	Instrument locker	Scintillation counter
Ion chamber survey meter	Beta, gamma	0-5 R/h	±10% full scale	9	Instrument locker	Ionization chamber
Ion chamber survey meter	Beta, gamma	0-50 R/h	±10% full scale	5	Instrument locker	Ionization chamber
High volume air samplers	-	0-30 cfm	±10%	6	Instrument locker	Air sampling
Low volume air samplers	-	0-100 cfm	±10%	10	Instrument locker	Air sampling

TABLE 12.3-2

FIXED INSTRUMENTATION

Instrument	Radiation Detected	Sensitivity(a)	Number	Location	Remarks
Automatic smear counter (gas proportional)	Alpha, beta, gamma		1	Radiation protection designated area	Used primarily for counting smears to determine contamination levels
Multiple channel analyzer	Gamma		1	Radiation protection designated area	Used for identification of isotopes
Small articles monitor/tool monitor	Gamma	Alarm ≥ 5000 dpm/100 cm^2	2	Radiation protection designated area	Plastic scintillator detectors, RCA exit survey
Contamination monitors - whole body	Beta, gamma	Alarm ≥ 5000 dpm/100 cm^2	2	RCA exit	Flow proportional detectors for RCA exit survey
Portal monitor	Gamma	Alarm 75 nCi of Cs-137[2]	1	RCA exit	Plastic scintillator detectors for RCA exit survey
Portal monitor	Gamma	Alarm 75 nCi of Cs-137[2]	2	PESB exit point	Plastic scintillator detectors for MPBPA exit
Contamination monitor - hand, cuff, and foot surface	Alpha, beta	Alarm ≥ 5000 dpm/100 cm^2	1	RCA exit toilets	Plastic scintillator detector, used to survey prior to entrance to the RCA exit restrooms

a. Instrument sensitivities will comply with measuring and reporting requirements of NRC IE Circular 81-07.

12.4 RADIOACTIVE MATERIALS SAFETY

12.4.1 MATERIALS SAFETY PROGRAM

Sealed and unsealed sources may be used at the Farley Nuclear Plant to calibrate reactor excore detectors, process and effluent radiation monitoring systems, area radiation monitoring systems, portable survey instruments, and fixed laboratory equipment. Storage and handling of these sources will be in accordance with 10 CFR 20, 30, 37, 40, and 70, with the radiation protection group being responsible for the control of such sources. A Nuclear Regulatory Commission license will be obtained for byproduct, source, or special nuclear material, as appropriate, prior to the procurement of radioactive sources.

High level sources such as those listed in table 12.4-1 will normally be housed in lockable, shielded containers and stored in an area that has been approved by the radiation protection group. Low level sources that are primarily used for calibration and quality control checks of fixed laboratory instruments and for portable survey instrument check sources are normally stored in locked cabinets located in radiation protection approved areas (e.g. on el 139 ft in the radiochemistry laboratory and counting room, or on el 155 ft in the radiation protection calibration laboratory at the training center).

Other information pertinent to the handling and use of radioactive sources is contained in subsections 12.3.1.4, 12.3.2.2, and 12.3.3.

12.4.2 FACILITIES AND EQUIPMENT

A discussion of the facilities utilized by the radiation protection groups is given in subsection 12.3.2.1. The radiochemistry laboratory, where unsealed radioactive sources would normally be stored, is equipped with two exhaust hoods that exhaust to the plant vent. The sampling room and gas analysis room are also equipped with one such hood each.

A discussion of portable radiation protection instrumentation is given in paragraph 12.3.2.5, with a listing and description of each instrument given in table 12.3-1. A discussion of fixed laboratory instrumentation is given in paragraph 12.3.2.6, with a listing and description of each instrument given in table 12.3-2.

12.4.3 PERSONNEL AND PROCEDURES

The radiation protection manager, the radiation protection support superintendent, and the plant health physicist are the key personnel responsible for handling and monitoring radioactive materials at the plant. The qualifications of these personnel are given in paragraph 13.1.3.1.

The qualification requirements for radiation protection supervisors, who direct/oversee the work activities of the radiation protection technicians, meet or exceed the minimum requirements set forth in American National Standards Institute (ANSI) N18.1-1971. The minimum qualification requirements for radiation protection supervisor are given in paragraph 13.1.3.1.

The qualification requirements for the health physicist/radiation protection technician, who reports to the radiation protection support superintendent and is responsible for activities related to radioactive material and radioactive waste shipments, meet or exceed the minimum requirements set forth in ANSI N18.1-1971. The minimum qualification requirements for the health physicist/radiation protection technician are given in paragraph 13.1.3.1.

The qualification requirements for radiation protection technicians, who handle and monitor radioactive materials under the direction of the radiation protection support superintendent or supervisors, meet or exceed the minimum requirements set forth in ANSI N18.1-1971. The minimum qualification requirements for radiation protection technicians are given in paragraph 13.1.3.1.

Procedures have been developed by the radiation protection group to cover the receipt, storage, and use of radioactive sources. These procedures are discussed in the group training sessions to ensure that all technicians who are required to handle radioactive sources are thoroughly familiar with the procedures.

12.4.4 REQUIRED MATERIALS

A list of sources that are likely to be purchased is given in table 12.4-1. This table includes a listing of isotope, quantity, form, and use for byproduct, source, and special nuclear materials that exceed the amounts of Table 1 of Regulatory Guide 1.70.3. At the time of procurement of radioactive sources that exceed the quantities in Table 1 of Regulatory Guide 1.70.3, an amendment will be made to table 12.4-1, if necessary.

TABLE 12.4-1

RADIOACTIVE SOURCES

Material	Isotope	Quality	Form	Use
Byproduct	Cs-137	300 μCi	Sealed source	Low range calibration of gamma radiation monitoring equipment
	Cs-137	<300 Ci	Sealed source	Intermediate and high range calibration of gamma radiation monitoring equipment
	Co-60	1 μCi	Sealed source	Low level calibration of portal monitors
	Co-60	36 μCi	Sealed source	Intermediate and high range calibration of gamma radiation monitoring equipment
Special nuclear	Pu-239	8.5 Ci	Pu-Be sealed source	Calibration of neutron radiation monitoring equipment
	Pu-239	5 μCi	Pu-Be sealed	Calibration of alpha radiation monitoring equipment
	Am-241	3.5 Ci	Am-Be	Source for in-line boron analysis instrumentation (one source per unit)

13.0 CONDUCT OF OPERATIONS

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13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF SOUTHERN NUCLEAR OPERATING COMPANY

13.1.1 CORPORATE ORGANIZATION

This section describes the structure and qualifications of Southern Nuclear Operating Company's (SNC's) corporate organization and the corporate organizations of the principal contractors: Bechtel Power Corporation and Westinghouse Electric Company. Technical qualifications of SNC and the principal contractors are also described in section 1.4. The corporate organization of SNC is shown in figure 13.1-7.

HISTORICAL
[*The construction phase corporate organizations of Southern Company Services (SCS), Bechtel, Westinghouse, and Fluor Constructors International, Inc.(FCII) (formerly Daniel Construction) are shown in figures 17.1-1, 17.1-3, 17.1-4, and 17.1-5, respectively.****]***

13.1.1.1 Corporate Functions, Responsibilities, and Authorities

SNC, as operator of Farley Nuclear Plant, is solely responsible for plant design, construction, quality assurance, testing, operation, and all other plant activities.

Alabama Power Company (APC) is the sole owner of FNP.

Southern Company Services, Inc. (SCS), an affiliated service company, served as the original architect-engineer. As a result of the consolidation of SCS and SNC nuclear expertise, and in addition to being the licensee, SNC also serves as its own architect/engineer and performs the functions previously performed by SCS. SNC is responsible for engineering and design of all portions of the plant except the switchyard and transmission system, which is designed by APC.

Bechtel Power Corporation is retained by SNC to assist in the engineering and design of the major portions of the plant, primarily the containments and auxiliary buildings.

Westinghouse Electric Corporation was contracted by APC to design and fabricate the nuclear steam supply systems and initial cores for FNP. Westinghouse was contracted to provide startup services during the preoperational test program at FNP.

The overall project quality assurance program and the programs of each of the principal contractors for the various plant activities are described in detail in the appropriate sections of chapter 17.

The responsibility for preoperational and startup testing was assigned to APC's nuclear generation organization. Advice and assistance was available from the vendors' engineers and startup specialists and from the engineering design groups and consultants. Details of the testing program are contained in chapter 14.

The responsibility for operation and maintenance is assigned to SNC.

13.1.1.2 Interrelationships with Contractors and Suppliers

The working interrelationships and organizational interfaces between the organizations described in paragraph 13.1.1.1 are described below.

SNC is responsible for coordinating and approving plant engineering and design. This includes design concepts, detail designs, specifications, and drawings whether developed by Bechtel, Westinghouse, or internally at SNC. SNC may call upon Bechtel or Westinghouse to provide comments on specifications and drawings affecting the Bechtel-Westinghouse interface.

Final acceptance by SNC of design concepts, documents, and equipment suppliers is based on recommendations from the responsible designers and consultation with the APC PGS, Purchasing, and Nuclear Oversight Department, as appropriate.

Construction and modifications to the plant are the responsibility of SNC. Onsite construction activities are monitored by SNC site management and Nuclear Oversight personnel, as appropriate.

The interrelationships and interfaces that existed between the various organizations during startup and preoperational testing are described in subsection 14.2.2.

The responsibility for ensuring that equipment suppliers and contractors conform to approved specifications is retained by the design organizations, although all equipment is procured by SNC. Conformance is verified through implementation of the quality assurance program described in the SNC Quality Assurance Topical Report (QATR).

13.1.1.3 Licensee's Technical Staff

The nuclear operations organization, under the supervision of the president/CEO, has direct responsibility for the operation and maintenance of Southern Company's nuclear plants. The nuclear operations organization consists of the plant operating staffs and corporate management, planning and performance, and quality assurance. Engineering support is provided primarily by the corporate and site engineering organizations as described herein.

As shown on figures 13.1-4, 13.1-5, and 13.1-6, the president and CEO, executive vice president, and vice president-Farley provide line management direction for the operation of the plant. The plant staff personnel are highly qualified to perform their responsibilities. The SNC corporate support staff consists of the vice president-fleet operations support and the vice president-engineering support and their staffs.

The structure of the nuclear operations organization is described in the SNC Quality Assurance Topical Report (QATR). Portions of the SNC Fleet Operations Support, Engineering, General Counsel and External Affairs, and Human Resources organizations are also described in the following paragraphs, or in the QATR.

13.1.1.3.1 Vice President and General Counsel

The vice president and general counsel reports to the president/CEO. This individual is responsible for the legal compliance and external affairs activities associated with operation of SNC plants. The vice president and general counsel is also the corporate secretary and directs the managing attorney/compliance manager and the public affairs manager.

13.1.2 OPERATING ORGANIZATION

The description of the operating organization in this section is for a two-unit operation. Any differences between one- and two-unit operations will be noted. The operating organization is described in the QATR, except for those positions described below.

13.1.2.1 Operations Group

The operations director, who reports to the plant manager, is responsible for the management and coordination of operations activities of the plant. Reporting to him/her are the operations superintendent-outage, the operations superintendent-support, and the operations superintendent-daily. He/she performs outage management functions as directed by the plant manager.

The operations director is responsible for the day-to-day operation of the plant in a safe and efficient manner in compliance with the operating license. The operations director is responsible for developing normal, emergency, and refueling operating procedures, department training, and retraining programs.

The plant operations superintendents are responsible for administering surveillance, coordinating investigation of personnel and equipment incidents, and for the on-shift operations staff including procedure development and coordination of shift clerks.

The shift manager reports to the shift operations manager. The shift managers are the senior management representatives of each shift and are responsible for safe and efficient operation of the plant. Reporting to the shift manager is a shift supervisor for each unit, shift support supervisors, plant operators, and systems operators.

The shift supervisor is in direct charge of his/her unit, including startup, power operations, and shutdown. He/she will initiate immediate action in the event of an abnormal situation to avoid violation of the operating license, to avert possible injury or undue radiation exposure of personnel, or to prevent damage to plant equipment.

The shift supervisor has the responsibility of supervising the actions of the station operators (plant operators, systems operators) to ensure safe and prudent operation of the facility. He/she will initiate immediate corrective action in any abnormal situation until assistance, if required, arrives.

The plant operators, who are supervised by the shift supervisor, control and direct the operation of their assigned unit according to detailed procedures. Normally, one plant operator will be assigned to each unit's main control center.

The shift support supervisor reports to the shift supervisor and assists in supervision of the system operators and equipment systems control. The shift technical advisor advises the shift supervisor during emergency conditions and has no command and control functions.

The systems operators, who work under the direction of the shift support supervisor and shift supervisor, inspect, service, and operate plant equipment.

13.1.2.2 Reactor Engineering Group

The reactor engineering group, under the supervision of the reactor engineering supervisor, is responsible for the evaluation of reports on reactivity, reactivity coefficients, boron concentration, control rod positions, reactivity worths, assisting in evaluating the fuel management program, assisting the operations manager in refueling operations, developing and writing various startup procedures, operating procedures and refueling procedures, originating any procedures involving tests on the reactor, periodically determining fuel composition and burnup, determining proper fuel loading sequences, and evaluating and determining control rod worths and operations sequences.

13.1.2.3 Supervisory Succession

The vice president-Farley is responsible for operation, maintenance, and technical support of FNP. In the absence of the vice president-Farley, the following members of the plant staff, in the order listed below, will assume this responsibility:

A. Plant manager.

B. Plant operations director.

C. Regulatory affairs manager.[a]

D. Other designated manager.

E. Shift manager.

F. Shift supervisor.

13.1.2.4 Shift Crew Composition

The FNP will be operated from one central control room. A shift supervisor will be directly responsible for the safe and efficient operation of each unit.

The normal shift complement for two-unit operation is as follows:

	Unit 1	Unit 2	Common
Shift manager (SRO)	-	-	1
Shift supervisor (SRO)	1	1	-
Shift support supervisor [d]	-	-	3
Shift technical advisor	-	-	1[b]
Plant operator (RO)	2	2	-
Systems operator[c]	2	2	3

The above manning levels are for normal two-unit operation. Deviations may be made as long as the minimum manning and license requirements of the Technical Specifications are met.

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a. The person filling this position may act as the plant manager provided this person meets the requirements of ANSI N18.1-1971 Section 4.2.1, FSAR paragraph 13.1.3.1.2, and has completed emergency director training.

b. The shift technical advisor may also be one of the shift support supervisors.

c. Although the systems operator will not be required to hold a reactor operator's (RO) license, he will be required to participate in regularly scheduled operator training programs and actively pursue a RO license. He will be trained and qualified in the operation of all auxiliary equipment and will work under the direction of a licensed operator. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

d. The shift support supervisor is not required to have a RO or senior reactor operator’s (SRO) license unless the individual is fulfilling the duties of another staff position that requires RO or SRO qualification.

13.1.3 QUALIFICATION REQUIREMENTS FOR NUCLEAR FACILITY PERSONNEL

13.1.3.1 Minimum Qualification Requirements for All Plant Personnel

Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in 10 CFR 55, except for the senior individual in charge of Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. Personnel who complete an accredited program which has been endorsed by the NRC shall meet the requirements of the accredited program in lieu of the above.

The following qualification requirements, used as a general guideline for personnel assignments, meet or exceed the minimum requirements set forth in the American National Standards Institute document N18.1-1971, Standard for Selection and Training of Personnel for Nuclear Power Plants.

13.1.3.1.1 Vice President-Farley / Plant Manager (Note 1)

A. A baccalaureate degree in an engineering or scientific field generally associated with power production.

B. A total of 10 years of power plant experience, of which a minimum of 3 years is nuclear power plant experience. Four of the remaining 7 years may be fulfilled by academic training generally associated with power production on a one-for-one basis.

C. Have acquired experience and training required for a senior operator's license.

Note 1: See ANSI N18.1-1971 for allowances if one or more of the requirements is not met for the vice president-Farley or plant manager.

13.1.3.1.2 Not used.

13.1.3.1.3 Regulatory Affairs Manager

A. A baccalaureate degree.

B. A total of 10 years of power plant experience of which a minimum of 3 years is nuclear power plant experience. Four of the remaining 7 years may be fulfilled by academic training generally associated with power production on a one-for-one basis.

13.1.3.1.4 Operations Director

A. High school education or equivalent.

B. A total of 8 years of power plant experience, of which a minimum of 3 years is nuclear power plant experience. Two of the remaining 5 years may be fulfilled by academic or technical training on a one-for-one basis.

C. The operations director or at least one operations superintendent shall hold a USNRC SRO license. If not currently licensed, the operations director shall have previously held a USNRC SRO license.

13.1.3.1.5 Shift Manager, Shift Supervisor, Operations Superintendent, and Shift Support Supervisor

A. High school education or equivalent.

B. A total of 4 years of power plant experience, of which a minimum of 1 year is nuclear power plant experience. A maximum of 2 years of the remaining 3 years of plant experience may be fulfilled by academic or related technical training on a one-for-one basis.

C. The shift manager and shift supervisor shall hold a USNRC SRO license. Also, the operations director or at least one operations superintendent shall hold a USNRC SRO license.

D. The shift support supervisor is not required to have a RO or SRO license unless the individual is fulfilling the duties of another staff position that requires RO or SRO qualification.

13.1.3.1.6 Plant Operator

A. High school education or equivalent.

B. A total of 2 years of power plant experience, of which a minimum of 1 year is nuclear power plant experience.

C. USNRC RO license.

13.1.3.1.7 Systems Operator

A. High school education or equivalent.

B. A total of 6 months of power plant experience or applicable experience.

13.1.3.1.8 Deleted

13.1.3.1.9 Maintenance Director and Maintenance Team Leaders and Maintenance Superintendents

A. High school education or equivalent.

B. A minimum of 7 years of responsible power plant experience or applicable industrial experience, of which a minimum of 1 year is nuclear power plant experience.

C. A maximum of 2 of the remaining 6 years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one basis.

D. Nondestructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

13.1.3.1.10 Maintenance Assistant Team Leaders

A. High school education or equivalent.

B. Four years of maintenance experience in a power plant or applicable industrial experience.

13.1.3.1.11 Mechanical Maintenance Personnel

A. Three years of power plant maintenance experience or applicable industrial experience.

13.1.3.1.12 Electrical Maintenance Personnel

A. Three years of electrical maintenance experience in a power plant or applicable industrial experience.

13.1.3.1.13 Instrumentation and Control Technician

A. High school education or equivalent.

B. Five years of experience in instrumentation and control, of which a minimum of 6 months is in nuclear instrumentation and control.

C. A minimum of 2 of the 5 years of experience should be related technical training.

D. A maximum of 4 of this 5 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.14 Control Technician

A. Two years of working experience in instrumentation and control.

13.1.3.1.15 Instrument Serviceman

A. Two years of working experience in instrumentation and control.

13.1.3.1.16 Deleted

13.1.3.1.17 Chemistry Manager/Chemistry Support Superintendent/Chemistry Operations Superintendent

A. High school education or equivalent.

B. A total of 5 years of experience in chemistry and environmental surveillance at a nuclear reactor facility, of which a minimum of 1 year of the chemistry experience is in radiochemistry.

C. A minimum of 2 of this 5 years of experience should be related technical training.

D. A maximum of 4 of this 5 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.18 Radiation Protection Manager/Radiation Protection Support Superintendent/Radiation Protection Operations Superintendent

A. High school education or equivalent. In addition, this person should have a baccalaureate degree or the equivalent in a science or engineering subject, including some formal training in radiation protection.

B. At least 5 years of professional experience in applied radiation protection. (A master's degree may be considered equivalent to 1 year of professional experience, and a doctor's degree may be considered equivalent to 2 years of professional experience where course work related to radiation protection is involved.)

C. At least 3 years of this professional experience should be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations, preferably in an actual nuclear power station.

13.1.3.1.19 Chemistry Supervisor/Radiation Protection Supervisor/Nuclear Specialist

A. High school education or equivalent.

B. Five years of experience in specialty (chemistry or radiation protection).

C. A minimum of 2 of this 5 years of experience should be related technical training in specialty (chemistry or radiation protection).

D. A maximum of 3 of this 5 years of experience may be fulfilled by a related technical or academic training.

13.1.3.1.20 Chemistry Technician and Radiation Protection Technician

A. Two years of experience in specialty (chemistry or radiation protection).

B. One year of training in the areas of chemistry, radiochemistry, or radiation protection principles or successful completion of the FNP chemistry and radiation protection technician course.

13.1.3.1.21 Engineering Director

A. A baccalaureate degree in engineering or the physical sciences.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year is in nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.22 Engineering Systems Manager

A. A baccalaureate degree in engineering or the physical sciences.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year is nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.23 Work Management Director

A. A high school education or equivalent.

B. A total of 8 years of responsible experience, at least 1 year of which should be nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.24 Engineering Supervisor-Programs, Engineering Supervisors-Mechanical and Electrical/I&C, Engineering Manager-CMO, Engineering Supervisor-Mechanical/Civil Design, Engineering Supervisor-Electrical Design, Engineering Supervisor-I&C/Digital Design, and Engineering Manager-EFIN

A. A high school education or equivalent. (A baccalaureate degree in engineering or the physical sciences is highly desirable and is required for the supervisor with responsibility for reactor engineering.)

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year involves nuclear power plant design or operation.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.25 Design Engineering Manager

A. A baccalaureate degree in engineering or the physical sciences.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year is in nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.26 Deleted

13.1.3.1.27 Deleted

13.1.3.1.28 Training Director

A. A high school education or equivalent.

B. A total of 4 years nuclear power plant experience of which a minimum of 1 year is nuclear power plant training experience. One of the remaining years may be fulfilled by academic or technical training on a one-for-one basis.

13.1.3.1.29 Facilities Supervisor

A. High school education or equivalent.

B. Sufficient maintenance or operational work experience to provide: a good knowledge of plant layout and functional operation, and an understanding of radiation protection regarding decontamination activities.

13.1.3.1.30 Deleted

13.1.3.2 <u>Qualifications of Plant Personnel</u>

The Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3, dated November 1978, Section 13, "Conduct of Operations," states: "This chapter of the SAR should provide information relating to the preparations and plans for operation of the plant." "Its purpose is to provide assurance that the applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the Licensee are adequate to protect public health and safety."

It should be noted the original purpose of identifying the specific qualifications of plant personnel in paragraph 13.1.3.2 was to assist the NRC staff in determining that Alabama Power Company had established an adequate staff to operate the facility. The NRC staff completed the review and has determined that the operating staff which was established is adequate as evidenced by the issuance of the Operating Licensees for Units 1 and 2. The purpose of paragraph 13.1.3.2 has, therefore, been accomplished and there is no further need to continually update the resumes previously contained therein. The resumes have therefore been deleted.

Paragraph 13.1.3.1 specifies the minimum qualification requirements for all plant personnel, and thereby fulfills the mandate of assuring "that the applicant will maintain...a staff of adequate size and technical competence...to protect public health and safety."

THIS FIGURE HAS BEEN DELETED.

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALABAMA POWER COMPANY
GENERAL ORGANIZATION

FIGURE 13.1-1

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POWER DELIVERY DEPARTMENT

FIGURE 13.1-2

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONSTRUCTION DEPARTMENT
ORGANIZATION

FIGURE 13.1-3

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY

Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SOUTHERN NUCLEAR OPERATING COMPANY
OFFSITE ORGANIZATION

FIGURE 13.1-4

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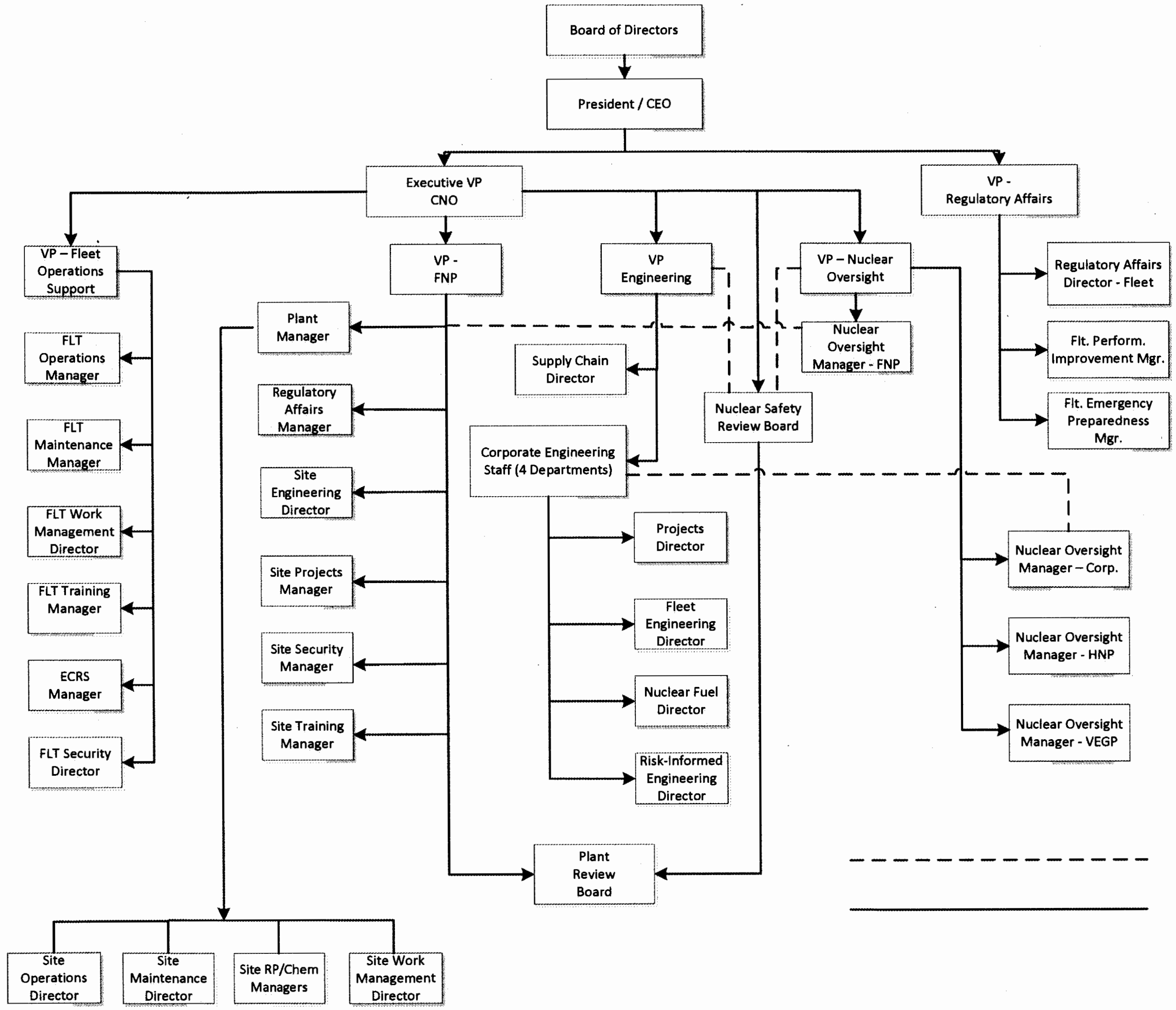
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

OFFSITE ORGANIZATION FOR FACILITY MANAGEMENT AND TECHNICAL SUPPORT

FIGURE 13.1-5



REV 26 11/15

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FACILITY ORGANIZATION

FIGURE 13.1-6

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SOUTHERN NUCLEAR OPERATING COMPANY
		FIGURE 13.1-7

13.2 TRAINING PROGRAM

Responsibility for administration of the overall training program for FNP rests with the vice president-Farley. Training programs administered at the plant site are supervised by the training manager. He/she is responsible to the vice president-Farley for ensuring that operations training, technical/maintenance, and plant access training maintain the educational level adequate for safe and efficient operation of the plant. The training program shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR 55, and shall include familiarization with the relevant operational experience. In lieu of ANSI N18.1-1971 requirements, the training program shall meet the requirements of an accredited program endorsed by the NRC.

13.2.1 PROGRAM CONTENT FOR OPERATIONS PERSONNEL

13.2.1.1 Operator License Training Program

The Operator License Training Program contains three phases and is presented on a level required for senior reactor operator.

Reactor operator (RO) license candidates and senior reactor operator (SRO) license candidates, with or without engineering degrees, attend this program. The Operator License Training Program satisfies the training requirement stipulated in ANSI N18.1-1971.

The three phases of operator license training are as follows:

A. Fundamental training

This phase of the program provides license candidates training in the areas of math, physics, chemistry, electrical theory, heat transfer/fluid flow, thermodynamics, reactor theory, and radiological protection.

B. Specialized training

This phase provides training in the areas of plant fluid, electrical, and instrumentation/control systems including fuel handling; procedures, technical specifications, and integrated plant operations including mitigating core damage and transient/accident analysis; and simulator training on FNP's Unit 1 plant-referenced simulator which includes reactor startups, shutdowns, and power operation during normal, transient, and accident conditions. Plant-specific procedures are used extensively during this portion of training.

C. On-the-Job training

During this period the student is assigned to various licensed operator crew positions and is required to perform day-to-day routine evolutions including a minimum of five reactivity manipulations while under the supervision of an RO or SRO licensed individual.

13.2.1.2 License Continuing Training Program

The License Continuing Training Program is designed to maintain a base level of knowledge through the use of a systematic approach to training (SAT). Farley's License Continuing Training Program received accreditation in December 1984 and accreditation is renewed at the frequency recommended by the National Academy for Nuclear Training.

The program meets or exceeds the requirements of 10 CFR 55. It is conducted on an SRO level.

The scope of the program is described in the applicable Training Center Procedures. Examinations are conducted to satisfy requirements of 10 CFR 55.

Each individual must demonstrate that he/she has adequately learned the material presented by passing the examinations required by 10 CFR 55. If an individual does not meet the passing criteria on any part of an examination required by 10 CFR 55, the individual will be removed from licensed duties and placed in a remedial program to correct his/her deficiencies.

The individual will be reexamined and must successfully meet the previously described passing criteria prior to resuming licensed duties.

If an exam or quiz is given that is not part of the exams required by 10 CFR 55 and if an individual scores less than the passing criteria, then he/she will be remediated and/or retested as deemed appropriate by training supervision.

The Simulator Continuing Training is based on a systematic approach to training (SAT) and is part of the accredited training program.

Integral with simulator continuing training, the trainee is given an annual simulator operational evaluation consisting of an overall operational ability demonstration. The trainee is also given an annual operational evaluation of job performance measures. Any deficiencies observed are discussed with the student, and corrective actions are given in a post training critique. The student will be removed from licensed duties if individual deficiencies result in failure of critical tasks during the simulator operational examination or job performance measures.

The student must be remediated and reexamined prior to resuming licensed duties.

13.2.1.3 System Operator Training Program (Nonlicensed Operator)

The System Operator Training Program contains three phases of training: classroom lecture and system qualification requirements.

The three phases of system operator (SO) training are as follows:

A. SO Fundamentals

This phase provides training in the areas of math, physics, thermodynamics, radiation protection, electrical theory, reactor theory, power plant equipment, communications, fire brigade, and OJT/OJE evaluator training.

B. Plant Systems

This phase provides training on plant fluid and electrical systems applicable to the duties of the SO. Included in this training are appropriate procedures and instructions on watch standing techniques and fire brigade training.

C. On-The-Job Training

This phase allows the student to complete the appropriate system qualification requirements. The student is also required to participate in routine shift activities such as equipment monitoring and observation, log taking, etc.

13.2.1.4 System Operator Continuing Training Program (Nonlicensed Operator Continuing Training)

The Systems Operator Continuing Training Program is designed to maintain a base level of knowledge through the use of a systematic approach to training (SAT). Farley's License Continuing Training Program received accreditation in December 1984 and accreditation is renewed at the frequency recommended by the National Academy for Nuclear Training.

Comprehensive written examinations shall be given to determine trainees' knowledge of subjects covered in the continuing training program. Quizzes may be used to determine the effectiveness of training. If an individual scores less than the passing criteria on a quiz, then he/she will be remediated and/or retested as deemed appropriate by training supervision. Operational exams will be given consisting of job performance measures. A score of 80 percent or greater must be achieved on the written exam and the JPM exam for successful completion of the continuing training program. Systematic observation and evaluation of the performance and competency of personnel shall be conducted by shift supervision.

13.2.2 PROGRAM CONTENT FOR TECHNICAL AND MAINTENANCE PERSONNEL

13.2.2.1 Radiation Protection Technician Training

Radiation Protection technicians will attend a formal training program. Technicians will normally complete this course during their first year of employment. However, they may be exempted from such training on the basis of previous training or job experience. Program courses will cover mathematics, nuclear physics, principles of radiation detection and protection, responsibilities and duties of radiation protection group personnel, radiation biology, plant systems, water chemistry, radioactive waste processing, gamma ray spectrometry, and on-the-job training.

13.2.2.2 Chemistry Technical Training

Chemistry technician IIs will attend a formal training program. Technician IIs will normally complete this training during the first year of employment. However, they may be exempted from such training on the basis of previous training or job experience. Program courses will cover responsibilities and duties of chemistry and environmental personnel, basic chemistry, corrosion, sampling considerations during normal plant operations and accident conditions, water purification and treatment, sewage treatment, chemistry technical specifications and limits, instrumental analysis and analytical procedures, plant chemistry control problems, group responsibilities required to support emergency activities, and on-the-job training.

13.2.2.3 Instrumentation and Control Training Program

Instrumentation and control group personnel will attend a formal training program. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover basic electricity and electronics; fundamentals of pressure; temperature; level and flow measurement and control; NSSS instrumentation such as 7300, SSPS, DRPI, rod control, incore and excore; primary and secondary plant systems; and on-the-job training.

13.2.2.4 Mechanical Maintenance Training Program

Mechanical maintenance group personnel will attend a formal training program. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover piping systems, diesel generators, rotating machinery, lubrication, machinery balancing, vibration and alignment, principles of rigging, hydraulics, primary and secondary plant systems, and on-the-job training.

13.2.2.5 Electrical Maintenance Training Program

Electrical maintenance group personnel will attend a formal training program. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover basic electricity fundamentals, single and three-phase motors, dc motors, ac and dc circuits, batteries, switchgear and protective devices, primary and secondary plant systems, and on-the-job training.

13.2.2.6 Vendor Supplied Training Courses

Personnel from the plant engineering, maintenance, and technical staff may attend training courses supplied by offsite vendors. Examples of this training are computer systems courses and station nuclear reactor engineer training courses.

13.2.3 PROGRAM CONTENT FOR PLANT ACCESS TRAINING

13.2.3.1 Radiological Health and Safety

All persons assigned to the plant who are granted unescorted access to the radiation controlled area will attend radiation protection and safety training. This training will be reviewed and updated as required and will be presented to plant personnel on a regularly scheduled basis. A typical outline of the program includes nuclear plant terminology, biological effects of radiation, plant specifics, 10 CFR 19 and 10 CFR 20, nonoccupational sources of radiation, warning signs and hazards, use of protective clothing, and frisking for contamination.

13.2.3.2 Emergency Plan Training

Training will be given on a regularly scheduled basis. The material presented will be relevant to the job requirements of each individual. Copies of the plant emergency plan will be available for review by all plant personnel.

13.2.3.3 Security Plan Training

Training covering the FNP security plan will be given on a regularly scheduled basis to appropriate plant personnel.

The material presented will be relevant to the job requirements of all individuals. Persons directly involved with the administration and implementation of the plan will receive adequate initial training, continuing training, and auditing to ensure the continued effectiveness of the plan.

13.2.3.4 Industrial Safety

Training covering industrial safety and health will be given on a regularly scheduled basis to appropriate plant personnel. A typical outline includes accident prevention, general plant safety, and SNC Safety and Health Policies and Procedures related to safety, including the Southern Nuclear Company Safety and Health Manual.

13.2.3.5 Plant Access Training (Continuing)

Appropriate plant personnel will be given training on a regularly scheduled basis covering the following subjects. The material presented in the training will be relevant to the individual's job requirements.

A. Emergency Plan implementing procedures.

B. Radiological health and safety.

C. Farley Nuclear Plant security plans.

Examinations should be given to determine the knowledge of subjects covered in the retraining.

13.2.4 RECORDS

A record of the person's qualifications and general records of correspondence and certifications will be maintained. Also, items such as examination results, continuing training examinations, training attendance, drill participation, and results of remediation administered in areas in which personnel have exhibited deficiencies will be maintained. Initial operator license training, license continuing training, and remediation files will be maintained in accordance with 10 CFR 55.

All of these records will be used to judge the effectiveness of the FNP initial training and continuing programs. The plant supervisory staff and the plant training staff will periodically review in detail each individual's progress in the plant training program. The plant supervisory staff and the plant training staff will also periodically review all accredited programs to determine how well these programs are supplying and maintaining qualified personnel to safely and efficiently operate Farley Nuclear Plant.

13.3 EMERGENCY PLANNING

The purpose of the Farley Nuclear Plant (FNP) emergency plan is to fulfill the requirements set forth in Appendix E, Emergency Plans for Production and Utilization Facilities, of 10 CFR 50, Licensing of Production and Utilization Facilities, with the fundamental objective of protecting the health and safety of the general public, persons temporarily visiting or assigned to the plant, and plant employees.

The NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, was used as a guide in developing the contents of the FNP emergency plan. This plan, entitled "Joseph M. Farley Nuclear Plant Units 1 and 2 Emergency Plan," is filed as a separate document. Detailed procedures concerning the implementation of the emergency plan are not included in the plan but are included in the emergency plan implementing procedures. In accordance with the latest requirements set forth in Supplement 1 to NUREG-0737, the emergency plan fully describes the functions of the Emergency Operations Facility (EOF and the Technical Support Center).

13.4 REVIEW AND AUDIT

A program of in-plant and independent reviews and audits has been developed to provide a system to determine that plant design, construction, startup, and operation are consistent with company policy and rules, approved procedures, and license provisions; to ensure that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and to detect trends which may not be apparent to a day to day observer. For convenience of administration of the program, the review and audit program is divided into the construction phase and the test and operation phase.

[HISTORICAL] [13.4.1REVIEW AND AUDIT - CONSTRUCTION

*Review and audit during design and construction of the Farley Nuclear Plant (FNP) is a part of the quality assurance program which is described in Section 17.1. This program does not utilize a formal review and audit committee, as such; however, through a comprehensive system of planned audits, compliance with all aspects of the quality assurance program is verified. Audits are performed on the design organizations, the construction site, and vendor facilities. The review and audit function during design and construction is fully described in Section 17.1.**]***

13.5 PLANT PROCEDURES

Actions concerning structures, systems, or components of the Farley Nuclear Plant (FNP) that are safety related are conducted according to written approved procedures. Safety-related structures, systems, and components are those that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public and such structures, systems, or components that are essential for the safe shutdown of the plant.

These procedures are written in sufficient detail so that a qualified individual may perform the required function without direct supervision. Procedures will contain the following significant aspects wherever they apply to the intent of each particular procedure:

A. Title - A concise descriptive statement concerning the activity covered in the procedure. Safety-related procedure titles are available in Document Control.

B. Purpose - A concise descriptive statement concerning the purpose and scope of the procedure.

C. References - Plant procedures, instructions, drawings, technical manuals, reports, the Final Safety Analysis Report, or other plant documents which contain information related to the procedure.

D. Precautions - Actions which if not taken or events which if not avoided when performing the procedure could result in hazardous personnel conditions or damage to plant equipment. Precautions will also appear in the main body of the procedure where applicable.

E. Prerequisites - Independent actions or procedures which shall be completed and plant conditions which shall exist prior to the procedure's use.

F. Limitations - Statements specifying limits on the parameters being controlled.

G. Main body - Step-by-step instructions in the degree of detail necessary to perform the required function or task.

H. Checkoff lists - Lists included in complex procedures requiring the person either performing or supervising the activity being performed to signify by his initials when important procedure steps have been completed.

I. Technical specifications - A reference to plant technical specifications where appropriate.

J. Symptoms - For emergency procedures, symptoms shall be included to aid in the identification of the emergency. They should include significant alarms, operating conditions, and, where possible, probable magnitudes of parameter changes.

K. Automatic actions - The automatic actions that will probably occur as a result of an emergency should be identified.

L. Immediate operator actions - For emergencies, steps should be specified for operation of controls or confirmation of automatic actions that are required to stop the degradation of conditions and mitigate their consequences.

M. Probable cause - For alarms the probable cause should be specified.

N. Acceptance criteria - The qualitative and/or quantitative criteria against which an evaluation of acceptability may be made. In certain procedures, the acceptance criteria will reference other sections of the procedure such as precautions, limitations, or checkoff list which may fulfill the acceptance criteria requirements.

Some procedural steps are required to be committed to memory, while others, which are routine actions, may be implied but not actually delineated.

Typical categories of plant procedures are as follows:

A. Administrative procedures.

B. Unit operating procedures.

C. System operating procedures.

D. Annunciator response procedures.

E. Critical safety function procedures.

F. Abnormal operating procedures.

G. Fuel handling procedures.

H. Surveillance test procedures.

I. Chemical and radiochemical control procedures.

J. Maintenance procedures.

K. General maintenance procedures.

L. Instrument maintenance procedures.

M. Radiation control and protection procedures.

N. Emergency Plan implementation procedures.

O. Security procedures.

P. Engineering technical procedures.

Q. Environmental sampling procedures.

R. Fire surveillance procedures.

S. Document control procedures.

T. Special nuclear material procedures.

13.5.1 ADMINISTRATIVE PROCEDURES

The vice president-Farley issues, in the form of standard operating orders, his instructions governing employee actions and established standards for plant operation. These rules and instructions provide a clear understanding of operating philosophy and management policies to ensure safe operation of the plant within the limits set by the facility licenses and the Technical Specifications. They provide that plant activities will be conducted in a manner that will protect the general public, plant personnel, and equipment.

In particular, written administrative procedures are provided to control issuance of documents, to ensure adherence to written procedures, records, and shift scheduling (including limiting working hours in accordance with the Technical Specifications), and to control the writing, approval, and implementation of written plant procedures. Administrative procedures are written, as referenced in the SNC Quality Assurance Technical Report (QATR), to implement certain aspects of the operations quality assurance program. Procedures covering plant operations, maintenance work, tests, equipment changes, radioactivity control, administration, and other activities which might adversely affect safety are put into effect only after being reviewed and approved as specified in subsection 17.2.20. It is the responsibility of the vice president-Farley to ensure that required reviews are completed before procedures are issued.

Permanent changes to procedures follow the same review and approval route as a new procedure. A permanent change is implemented only after formal approval.

Temporary procedures and temporary revisions to procedures are issued to direct operations during testing and maintenance, to provide guidance in unusual situations, and to ensure orderly and uniform operations for short time periods when existing procedures do not apply.

These administrative procedures delineate the general responsibilities and authorities of the plant staff. Instructions have been established to provide methods by which temporary changes to approved procedures can be made, including the designation of persons authorized to approve such temporary changes. In cases of emergency, personnel will be authorized to depart from approved procedures when necessary to prevent injury to personnel or damage to the facility in accordance with the provisions of 10 CFR 50.54 (x) and (y). In all such cases, changes will be documented and reviewed by the appropriate superintendent within 30 days. If appropriate, it will then be incorporated in the next revision of the affected procedure. More detail on the review, approval, changes, revisions, and implementing of procedures is found in the QATR.

Procedures are provided for control of equipment to maintain reactor and personnel safety and to avoid unauthorized operation of equipment. These procedures delineate control measures and actions such as locking, tagging, notification, removal of tags, and identification of equipment.

Standard operating procedures are documented and disseminated to the plant staff. They include such subjects as job turnover and relief, designation of confines of control room, definition of duties of operators and others, transmittal of operating data to management, filing and changing charts, maintaining operating log, and personal belongings carried onsite. Also, there is a mechanism for issuing special orders which have short term applicability and which require dissemination.

A number of programs are in place to ensure that plant procedures are adequate and applicable and to ensure that potential procedural impact is assessed and necessary revisions are made. The following programs, which include but are not limited to those listed below, adequately provide for procedure review and any necessary input to procedure revisions and changes:

- Plant personnel feedback.
- Incident investigation feedback.
- Design modification program.
- Operating experience evaluation program.
- Simulator training program.
- Technical specifications and FSAR revisions.
- Quality assurance program.

Additionally, as a part of the overall quality assurance program, the QA group performs various audits (described in the QATR) to assure that the procedural process is working and that procedures are being properly maintained.

13.5.2 OPERATING PROCEDURES

Operating procedures were written before initial fuel loading and include all the anticipated operating conditions that affect the safety of the plant and the public. These procedures provide a preplanned method of conducting operations to minimize reliance on memory. They cover the following areas: system operating procedures, unit operating procedures, fuel handling procedures, annunciator response procedures, emergency operating procedures, and abnormal operating procedures.

The following types of procedures have been written for the FNP.

13.5.2.1 System Operating Procedures

Instructions for energizing, starting up, shutting down, changing modes of operation, and other instructions appropriate for operations of systems related to the safety of the unit are delineated in system operating procedures. These procedures are concerned with systems and include valve lineup, control operation, and instrumentation within the system boundaries. Where needed to ensure a safe and proper sequence of operation, a procedure checkoff list is incorporated.

13.5.2.2 Unit Operating Procedures

Unit operating procedures have been written to provide instructions for the integrated operation of the unit. The system operating procedures covered in paragraph 13.5.2.1 were limited to individual systems, but these procedures integrate all auxiliary systems to the main nuclear steam supply system and turbine-generator to perform as a unit.

Unit operating procedures cover plant operation in the following areas.

13.5.2.2.1 Startup Procedures

Startup procedures have been written to provide instructions for starting the reactor from cold or hot conditions and establishing minimum load operation with the generator synchronized to the line. System procedures are referenced as required.

13.5.2.2.2 Shutdown Procedures

Procedures have been written to guide operations during and following controlled shutdown and include instructions for establishing or maintaining hot shutdown or cold shutdown conditions, as applicable. Such actions as monitoring and controlling reactivity, load reduction rates, cooldown rates, taking equipment or systems out of service and/or into service, electrical switching to ensure unit safety, etc., are covered in these procedures. System procedures are referenced as required.

13.5.2.2.3 Power Range Operation Procedures

Procedures have been written for steady-state power operation and load changing and include instructions on the use of control rods, chemical shim, long- and short-term control of reactivity, making deliberate load changes, and adjusting operating parameters to trim power operation. System procedures are referenced as required.

13.5.2.2.4 Reactor Startup Procedures

Procedures have been written to instruct the operator on prerequisites, equipment control, and control functions concerning the reactor in going critical. Actual control manipulations have been included in these written procedures. These procedures are included in the startup procedures described in paragraph 13.5.2.2.1.

13.5.2.3 Abnormal Operating Procedures

Procedures have been written for operation of the unit under abnormal conditions.

13.5.2.4 Annunciator Response Procedures

Procedures have been written to instruct the operator on the proper action to be taken in response to each safety-related annunciator in the control room. These procedures contain annunciator identification, inputs into this annunciator circuit, and logical responses to be taken to ensure corrective action.

13.5.2.5 Fuel Handling Procedures

Fuel handling operations will be performed in accordance with written procedures. These procedures, in conjunction with respective unit operating procedures, specify actions and philosophy for core alterations and partial or complete fueling operations. They include continuous monitoring of the neutron flux throughout core loading, periodic data taking, audible annunciation of abnormal flux increases, duties of personnel assigned to fueling, response actions to alarms during fueling, instructions for proper sequence of events, rules for periods when fueling is interrupted, verification and frequency of sampling to ensure the shutdown margin, communications between control room and the fuel loading station, criteria for stopping fueling and evacuation systems operation, and documentation of final fuel component serial numbers and locations. Prerequisites have been included in these procedures to ensure the status of plant systems is such that fueling can proceed. Specific procedures are written for each fueling to handle those actions and parameters that are unique to that particular refueling. System operating procedures are referenced as required.

13.5.3 MAINTENANCE PROCEDURES

A maintenance program was developed early in plant life to maintain safety-related equipment at the efficiency level required to perform its intended function. This program includes maintenance of safety-related mechanical, instrument, and electrical equipment. Maintenance or modifications which may affect the functioning of safety- related structures, systems, or components are performed to ensure operating quality at least equivalent to that specified in applicable codes, bases, standards, design requirements, materials specifications, and inspection requirements. To ensure a high degree of confidence, appropriate inspection in accordance with applicable standards is performed. Replacement components are used only when the proper quality assurance documents are available or when the required quality assurance can be obtained and documented by inspection and/or testing prior to being placed in service. These quality review measures are documented, and evidence of the documentation is retained.

Maintenance which can affect the performance of safety-related equipment is preplanned and performed in accordance with written procedures, approved documented instructions, and/or approved drawings appropriate to the circumstances. In maintenance situations where related vendor manuals or instructions or approved drawings do provide adequate instruction to ensure the required quality of work on safety-related equipment, these documents will be used. However, where related vendor manual or instructions or approved drawings do not provide adequate instruction to ensure the required quality of work on safety-related equipment, a suitable procedure to handle this maintenance work will be written. Skills normally possessed by qualified maintenance personnel are not covered in detailed procedures.

When failure occurs to safety-related equipment, the cause of the failure will be evaluated; however, since the probability of failure is usually unknown and the time and mode of failure are usually unpredictable, procedures will not be written for repair of most equipment prior to failure.

A preventive maintenance schedule has been developed which describes the time and type of maintenance to be performed. A preliminary schedule was developed early in plant life and will be refined and changed as experience with the equipment is gained. This schedule specifies equipment inspections, replacement of such items as filters and strainers, and inspection or replacement of parts that have a specific lifetime. As equipment baselines develop, a computer program will be considered to aid in preventive maintenance scheduling. Lubrication requirements will be scheduled separately.

Maintenance is scheduled and planned so as not to jeopardize the safety of the plant. Planning is done in order to consider the possible safety consequences of concurrent or sequential maintenance, testing, or operating activities. Maintenance is performed in such a manner that the license limits are not violated. Planning for maintenance includes evaluation of the use of special processes, equipment, and materials to be used in the performance of the job. This evaluation attempts to assess the potential hazards to personnel and equipment.

Modifications in equipment or systems which might degrade the plant quality will not be permitted.

Procedures to support the maintenance philosophy were written early in plant life. These procedures control plant maintenance activities during operation and at the time of equipment failure.

13.5.4 PERIODIC CALIBRATION AND TEST PROCEDURES

13.5.4.1 Instrument Calibration and Tests

In those instances where equipment technical manuals do not provide sufficient instruction, procedures will be written for periodic calibration and testing of all safety-related plant instrumentation. This instrumentation includes interlocks, alarm devices, sensors, readout instruments, transmitters, signal conditioners, laboratory equipment, key recorders, and protective logic circuits. A list of equipment to be calibrated, a calibration schedule, and calibration records will be kept and maintained by the instrument group. Manuals will be reviewed and procedures written with the intent of ensuring measurement accuracies adequate to keep safety parameters and controls within safety and operational limits. Calibration, testing, and check of instrumentation channels will be performed as specified in the plant Technical Specifications or the Technical Requirements Manual, as applicable.

13.5.4.2 Safety-Related Surveillance Tests

Safety-related surveillance tests and inspections are performed in accordance with the plant Technical Specifications or the Technical Requirements Manual, as applicable, to ensure that failures or substandard performance do not remain undetected and that the required reliability of safety systems is maintained. Testing of safety-related plant structures, systems, and components is performed in accordance with approved written test procedures which set forth the requirements and acceptance limits. Test procedures contain a description of the test objectives, the acceptance criteria that will be used to evaluate the test results, the prerequisites for performing the tests, including any special conditions to be used to simulate normal or abnormal operating conditions, and the test procedure. These procedures also specify any special test equipment or calibrations required to conduct the surveillance test. A surveillance schedule, reflecting the status of all planned in-plant surveillance testing, was established prior to fuel load. Additional control procedures have been established, as necessary, to ensure timely conduct of surveillance testing, appropriate documentation, reporting, and evaluation of test results.

Records are kept in sufficient detail to permit adequate confirmation of the test program. They identify, as a minimum, the data recorder, results of the test, the acceptability of the results, deviations and their cause or reason, and any corrective action resulting from the test. Significant deficiencies identified by the tests will be reported to management where the deficiencies will be evaluated and the condition corrected in a timely manner.

13.5.5 CHEMICAL AND RADIOCHEMICAL CONTROL PROCEDURES

Procedures have been written for chemical and radiochemical control activities. These procedures include the nature and frequency of sampling and analyses, instructions for maintaining coolant and condensate within prescribed quality limits, and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces, or become sources of radiation hazards due to activation. These procedures include laboratory instructions and calibration of laboratory equipment.

Sample intervals will be according to the plant Technical Specifications or the Technical Requirements Manual, as applicable.

13.5.6 PROCEDURES FOR COMBATING EMERGENCIES AND OTHER SIGNIFICANT EVENTS

Procedures have been written to guide operations during potential emergencies. They have been written so that a trained operator and crew will know in advance the expected course of events that will identify an emergency and the immediate action which should be taken.

Procedures that cover actions for manipulation of controls to prevent accidents or lessen the consequences are based on a general predictable sequence of observations and actions. Emphasis is placed on operator responses to observations and indications in the control room; that is, when immediate operator actions are required to prevent or mitigate the consequences of a serious condition, procedures delineate and require that these actions be implemented promptly. When initially available intelligence provided to operating personnel via instrument readings, annunciator alarms, physical conditions, and personal observations may not clearly indicate the difference between a simple operational problem and a serious emergency, the actions outlined in the procedures are based on a conservative course of action by the operating crew. The operator will be responsible for believing and responding conservatively to instrument indications unless they are proven to be false. Considerable judgment on the part of competent personnel will be used before departing from the procedure. In events where it is necessary to depart from approved procedures to prevent injury to personnel or damage to the facility, the deviation will be logged with the prevailing circumstances described. Sections of emergency operating procedures that require immediate response action from the operating crew are required to be committed to memory.

13.5.6.1 Events of Potential Emergency

Potential emergency conditions have been identified and procedures for coping with them have been prepared. Some of these procedures include immediate action to be taken, while others may guide operations in correcting an abnormal situation which could lead to an emergency.

13.5.6.2 Procedures for Implementing the Emergency Plan

Implementing procedures for emergency plan actions have been written to instruct all plant personnel in the integrated implementing actions of the emergency plan.

13.5.7 PROCEDURES FOR THE CONTROL OF RADIOACTIVITY

Procedures have been written to provide for personnel exposure control, for the control of radioactive materials on the plant site, and for the control, sampling, monitoring, storage, and disposal of solid, liquid, and gaseous radwastes.

13.5.7.1 Solid Radioactive Wastes Control Program

Written procedures shall be established, implemented and maintained covering the PROCESS CONTROL PROGRAM implementation.

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71;

State regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

The PCP shall be approved by the Commission prior to implementation.

Licensee initiated changes to the PCP:

A. Shall be documented and records of reviews performed shall be retained as required by subsection 13.6.2. This documentation shall contain:

   1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

B. Shall become effective after review and acceptance by the PRB and the approval of the vice president-Farley.

13.5.8 SECURITY PROCEDURES

Security procedures have been prepared to complement the security plan in describing the security requirements for the FNP and to guide security activities. These procedures were drafted and reviewed by members of the plant security group and are used to guide security activities. Because of the sensitive nature and content of most security procedures, copies will be placed only where required, with access restricted to those plant personnel having a need to know. Security procedures are serially numbered documents which are periodically inventoried and accounted for by the security (site) manager. Procedures are subject to reviews by the security (site) manager whenever a security threat or other security incident makes such a review desirable.

Revisions to procedures are subject to the same review and approval as original procedures.

Revised or obsolete security procedures are processed in accordance with approved plant procedures. Temporary security procedures and temporary revisions to procedures to cover unanticipated or emergency situations may be issued at the direction of the vice president-Farley. Temporary procedures shall reflect the purpose and limitations of their use. Unless declared obsolete and processed in accordance with approved plant procedures, temporary procedures shall be prepared as permanent procedures as soon as practicable.

13.5.9 SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY PROCEDURES

Special nuclear material procedures have been written to implement the special nuclear material accountability program. These procedures delineate personnel responsibilities and authorities, designate and describe item control areas, and provide instructions for special nuclear material control records and reports, receiving and shipping of special nuclear material, internal transfers, physical inventories, special nuclear material element and isotopic calculations, and special nuclear material review and audit.

13.5.10 ENGINEERING TECHNICAL PROCEDURES

Engineering technical procedures provide instructions for the performance of tests or essential calculations which, because of their nature or technical content, are normally performed by plant engineers. These procedures cover such subjects as reactor core physics tests, calculation of operating curves and data, component or systems performance evaluations, developmental testing, and other related functions.

13.5.11 ENVIRONMENTAL SAMPLING PROCEDURES

Environmental sampling procedures provide methods for measurement of radiation and of radioactive material in those exposure pathways which lead to the highest potential radiation exposures of individuals resulting from operation of the plant.

13.5.12 DOCUMENT CONTROL PROCEDURES

Document control procedures provide guidelines for handling various documents and records.

13.5.13 (Deleted)

13.5.14 FIRE SURVEILLANCE PROCEDURES

Fire surveillance procedures have been written to provide instructions for the testing of fire surveillance equipment to ensure its operability.

13.5.15 FIRE VENTILATION PROCEDURES

Fire ventilation procedures provide symptoms, automatic actions, initial actions, secondary actions, and restoration of systems for ventilation operations initiated and/or required due to a fire.

13.6 PLANT RECORDS

Records documenting the quality of the design, construction, testing, operation, maintenance, and modification of the Farley Nuclear Plant (FNP) are maintained at the plant site. The records maintained comply with the requirements of Criteria XVII of Appendix B of 10 CFR 50 and American National Standards Institute N18.7, Standard for Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.

These records are stored and maintained in the document control center and satellite document control center which provides facilities to preserve the records in a manner to preclude deterioration. A signout system ensures immediate location of any record that is temporarily removed from the files. Nuclear Oversight is responsible for auditing these quality records as described in the QATR.

13.6.1 PLANT HISTORY

Upon completion of the plant design, construction, and construction testing (Phase I of the FNP testing program), the Power Generation Services (PGS) Department (formerly known as the Construction Department) transferred or disclosed the location of all quality documentation to the plant staff and the Nuclear Oversight group (formerly known as the safety audit and engineering review group). Records not maintained at the plant site are kept at the material supplier's site due to codes or standards requirements or at SNC Corporate.

This documentation and the records generated during Phases II and III of the testing program, operation, maintenance, inspection, modification, and events of the FNP as described in subsections 13.6.2 and 13.6.3 serve as a recorded history of quality plant activities.

13.6.2 OPERATING RECORDS

Safety-related preoperational and startup test records generated during Phases II and III of the testing program are kept at the plant site along with appropriate operating records. These operating records include chemistry records; manuals and procedures; operating, maintenance, and testing records; special nuclear material records; records and reports required by regulatory agencies; and administrative records.

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

Operational records and logs that are kept at the plant are considered nonpermanent and shall be retained for at least 5 years, unless a longer period is required by applicable regulations. The following are examples of these type records:

A. Startup problems and resolutions.

B. Records and logs of normal plant operation, including power levels and period of operation at each level.

C. Records and logs of principal maintenance activities, including inspection, repair, substitution, or replacement of principal items of equipment related to nuclear safety.

D. Records of abnormal occurrences and unusual events.

E. Obsolete equipment instruction manuals and drawings.

F. Records of shipment of radioactive material.

G. All reportable occurrences submitted to the Nuclear Regulatory Commission.

H. Records of surveillance activities, inspections, and calibrations required by regulatory agencies or Technical Specifications.

I. Records of changes made to plant procedures.

J. Records of sealed source and fission detector leak tests and results.

K. Records of annual physical inventory of all sealed source material of record.

Operational records that are considered to be of a significant value in demonstrating capability of safe operation, in maintaining, replacing, or repairing an item, in determining the cause of an accident or malfunction of an item, or in providing baseline data for inservice inspection are maintained for the life of the plant. The following are examples of these type records:

A. Applicable plant procedures and drawings.

B. Records of inservice inspections.

C. Records of radiation exposure for all individuals entering radiation control areas.

D. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

E. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

F. Reactor water chemistry data.

G. Records of gaseous and liquid radioactive material released to the environs.

H. Records of transient or operational cycles for those unit components identified in table 5.2-2 and 5.2-2a.

I. Records of reactor tests and experiments.

J. Records of training and qualification for current members of the facility staff.

K. Records of safety-related quality assurance activities required by the QATR and not specifically described in the previous paragraph addressing records retained at least 5 years.

L. Records of reviews performed for changes made to procedures or equipment or reviews of tests or experiments pursuant to 10 CFR 50.59.

M. Records of meetings of the Plant Review Board (PRB) and the Nuclear Safety Review Board (NSRB).

N. Records of secondary water sampling and water quality.

O. Records of analyses required by the radiological environment monitoring program.

P. Records of the service lives of all hydraulic and mechanical snubbers required by the Technical Requirements Manual, including the date at which the service life begins and associated installation and maintenance records. (For Unit 1 this applies to all safety-related snubbers installed or replaced after July 1, 1981.)

Q. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

13.6.3 EVENT RECORDS

Records of occurrences such as radioactive releases and environmental surveys are maintained at the plant site. These records are considered to be of a significant value in demonstrating the safe operation capability of the plant and will therefore be kept for the life of the plant. The following are examples of these type records:

A. Records of plant radiation and contamination surveys.

B. Gaseous and liquid release data.

13.6.4 ENVIRONMENTAL RECORDS

Records of modifications to plant structures, systems, and components determined to potentially affect the continued protection of the environment are retained for the life of the plant. All other records, data, and logs relating to the environmental protection plan are retained for 5 years or, where applicable, in accordance with the requirements of other agencies.

13.7 INDUSTRIAL SECURITY

The industrial security requirements of 10 CFR 73.55 will be met for the Farley Nuclear Plant (FNP). This section discusses in general how FNP will meet these requirements. A description of detailed security measures for the physical protection of the plant was submitted to the Nuclear Regulatory Commission (NRC) in May 1977. As a result of the onsite review conducted by the NRC staff between August 2, 1977, and August 12, 1977, Alabama Power Company (APC) submitted a revised description of security measures on November 11, 1977. Following a meeting with the NRC on March 1, 1978, APC submitted general revisions to the NRC on April 14, 1978. On November 16, 1978, a final submittal was made to the NRC. On February 26, 1979, the NRC issued Amendment 9 to the Unit 1 operating license, incorporating the physical security plan into license NPF-2, effective February 23, 1979. This plan describes the security measures that are effective in meeting 10 CFR 73.55.

Security measures include provisions for:

A. Initial employee investigation and continuing evaluation/review.

B. A security force.

C. A lighted physical barrier.

D. A lock and key system.

E. A system of intrusion alarms.

F. Liaison with local law enforcement authorities and redundant communication links with these authorities.

G. Routine operating and administrative procedures that adequately monitor the condition of vital areas and equipment.

H. Control of the movement of personnel, materials, and vehicles into and within the plant controlled area to ensure an adequate security program.

I. Written procedures.

Several changes have been made to the Security Plan in accordance with 10 CFR 50.54(p).

In accordance with 10 CFR 50.54(p) and 10 CFR 73, Appendix C, APC submitted a Contingency Plan on October 8, 1979. Revisions to this plan were submitted on March 20, 1980. The NRC issued approval of the Contingency Plan on May 1, 1980.

Also in accordance with 10 CFR 73.55(b)(4), APC submitted a Security Personnel Training and Qualifications plan on August 16, 1979. On April 29, 1981, APC submitted Revision 1 to this plan. The NRC issued approval of this plan on August 18, 1981, and incorporated it into both unit operating licenses.

13.7.1 PERSONNEL AND PLANT DESIGN

13.7.1.1 Plant Design

The plant design and arrangement include features that enhance industrial security and reduce the vulnerability of the plant to deliberate acts that may adversely affect the plant and public safety. The plant design incorporates a system of locked fences that form protected areas for control of access to major plant structures and equipment. Vital outlying structures incorporate fences with structural design to assist in controlling access. Gates through fences utilize a lock and key system administered by the security (site) manager. Intrusion alarms and closed circuit television are also integrated, as needed, into the plant design for electrical/electronic surveillance. Portions of the fences, including those surrounding vital structures, are lighted between sunset and sunrise.

Doors allowing access to plant structures have a lock and key system with strict administrative control. Doors to vital structures have intrusion alarms. Vital structures are those enclosing vital equipment and facilities. These facilities and equipment are considered vital if their failure could lead to a radiological accident significantly affecting the health and safety of the public. Also, associated equipment or facilities designed to limit the consequences of such radiological accidents or required for safe shutdown of the reactor are considered vital equipment.

At least two independent communication links with the local law enforcement authorities are provided. One of these links utilizes electromagnetic waves. One link is available through the use of a two-way radio system. Intrusion alarms and communication links terminate in the central and secondary alarm stations.

The grading, ground cover, and landscaping within the protected area do not introduce barriers to surveillance by the security patrol. All-weather roads and pathways are provided within the protected areas for the use of the security patrol.

Intrusion alarms, security fences, lights, and communication links are maintained in operable and effective condition under the supervision of the security (site) manager. Lights and communication links are inspected and tested for operability and required functional performance. Intrusion alarms are tested periodically.

13.7.1.2 Employee Selection and Performance Review

In order to select reliable personnel to protect against industrial sabotage, employment standards and procedures have been established by SNC management. These include application forms, background investigation, physical examination, and interviews. The application, background investigation, physical examination, and initial interviews are coordinated by the SNC Personnel Department. Prior to employment for a position at the plant, the applicant's personnel file is reviewed and at least one detailed interview is conducted by SNC Personnel. All new employees have a 6-month probation period during which their performance is closely observed. Nuclear security officers (NSOs) are screened and qualified in accordance with the provisions of the FNP Training and Qualification Plan.

Personnel being considered for promotion and/or transfer to the plant from other plants or departments within the company are screened to ensure the effectiveness of the plant industrial security program.

An alert plant organization, cognizant of its responsibility for protection against industrial sabotage, is maintained. The performance of all employees is appraised annually and the results reported in order to: further aid in maintaining a high level of employee performance and the maximum utilization of employee abilities; provide recorded evidence of employee performance for use in making judgments concerning transfer, demotion, promotion, and terminations; ensure that employees are adequately and systematically informed of the effectiveness of their service; and further facilitate the maintenance of a high standard of supervision in SNC. A statement of the results of those appraised is signed by the employee's supervisor.

Observation of employee performance is a continuous supervisory function; such observation is made as a regular part of day-to-day supervision. Plant supervision is constantly on the alert for early detection of changes in behavioral patterns of employees under its supervision.

13.7.2 SECURITY PLAN

The following sections discuss in a general manner various aspects of the FNP Security Plan.

13.7.2.1 Means for Control of Plant Access

A security force of well-trained, uniformed NSOs polices plant property and provides access control and surveillance of the plant protected and vital areas. Other plant personnel, such as the operating group, help monitor access inside plant structures by observation of the plant area during equipment checks and by challenging any unauthorized individuals within the plant buildings.

Gates permitting entrance to the protected areas surrounded by a security fence are either locked or manned access control is in effect. The main security fence and fences surrounding outlying structures are regularly patrolled by roving NSOs. The roving NSOs have radio communication with the central alarm station.

Guidelines are included in the security plan for monitoring and controlling access to and from the plant and the movement of persons within the plant. The security plan also includes guidelines for searching vehicles and personnel entering and exiting the plant. In-plant access control is provided by operating personnel challenging the entry of unauthorized persons attempting to enter vital operating areas. Guidelines controlling the entry of unauthorized vehicles and the entry and exit of unauthorized materials through the fence and gate system are included in the security plan. Vehicle access to the protected area is limited to certain authorized vehicles and there are vehicle barriers installed to deny forced entry of a vehicle.

13.7.2.2 Control of Personnel by Categories

The general public is admitted inside the controlled area at the Central Security Control Building. A visitor identification system is used as the primary form of access control inside the security areas. Visitors are escorted while inside protected and vital areas by authorized personnel who are responsible for the visitors' actions, areas of movement, and safety.

The biometrics access control system will control access to the protected areas. Access is limited to properly authorized individuals. Authorized individuals are admitted by identification of the geometric hand reader and a proximity card reader. The readers positively identify each individual by their hand print and the RFID tag in their security badge.

However, override capability is provided to either allow or deny entry to the protected area. Permanent plant employees and other authorized employees are positively identified by the biometrics access control system.

13.7.2.3 Access Control During Emergencies

The Security Plan is compatible with the Emergency Plan and Contingency Plan procedures. NSOs are informed of any emergency situations within the plant and are informed as to what outside assistance and support personnel will arrive. If necessary, additional personnel are assigned to facilitate admitting emergency vehicles and personnel as fast as possible. While on the site, all outside emergency personnel are under the direction of the emergency director.

13.7.2.4 Equipment Monitoring

The plant operations personnel monitor the status of plant systems and equipment by means of annunciators, indicating lights, indicators, and recorders. Operating logs and computer printout data are frequently examined for changes in equipment performance. Most equipment is in continuous operation, and any changes are immediately detected by the operator. Standby and emergency equipment is periodically tested on a routine basis as required by the plant Technical Specifications and the Technical Requirements Manual. System operators inspect accessible equipment and areas for unusual operating characteristics and abnormal conditions on a frequency established by plant procedures. These inspections should detect foreign objects, evidence of a sabotage attempt, etc. System valve positions are periodically checked by the system operators using valve checklists. In addition, shift support supervisors, plant supervisors, and plant engineers make frequent unscheduled inspection tours through the plant. The combination of these efforts provide reasonable assurance that unauthorized physical changes in the status of components of equipment do not go undetected for long periods.

Key operating log sheets and selected recorder tracings are reviewed daily by plant operation and technical staff personnel. Abnormal changes observed are called to the attention of the plant manager and appropriate supervisors for investigation and corrective action, if required. This audit serves to ensure early detection of changes which have a significant bearing on plant performance.

13.7.2.5 <u>Potential Security Threats</u>

Procedures for dealing with potential dangers, such as bomb threats and civil disturbances, including provisions for timely notification of proper authorities, are included in the contingency plan. These procedures are written in sufficient detail so that a qualified individual can perform the required actions without direct supervision.

If there appears to be a real threat of disorder, off-duty NSOs will be recalled and local and State law enforcement authorities will be contacted for assistance.

13.7.2.6 <u>Administrative Procedures</u>

In the event of an incident of suspected sabotage or condition that threatens the security of the plant, security personnel immediately notify the emergency director and implement appropriate contingency plan events. The executive vice president will be notified of the situation as soon as possible by the vice president-Farley or his designated alternate. A report will be prepared by the security (site) manager that will include, as a minimum, the cause of the event, extent of damage, if any, and actions taken to prevent recurrence of a similar event. Copies of the report will be sent to the plant manager, regulatory affairs manager, and vice president-Farley. The NRC will be notified of the situation in accordance with the contingency plan, and a report containing all information indicated above will be submitted.

The nuclear oversight vice president will make or cause to be made audits of the plant security program as specified in the Security Plan. Based on these audits, the vice president-Farley will direct revising and updating the program, if needed. A report on such audits will be forwarded to the vice president-Farley, the regulatory affairs manager, and the plant manager.

The following security records are maintained in the plant records system:

A. NSO logs.

B. Test and maintenance records of security equipment.

C. Up-to-date copy of the Security Plan including all security procedures.

D. Reports of responses to intrusion alarms or threats to plant security.

E. Key and lock records.

F. Security program audits.

G. Up-to-date copy of the Contingency Plan and associated procedures.

H. Up-to-date copy of the security personnel training and qualification plan.

14.0 INITIAL TESTS AND OPERATION

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[HISTORICAL] [14.0 INITIAL TESTS AND OPERATION

An extensive testing program was conducted during the initial period of testing and operation of Farley Nuclear Plant (FNP) Units 1 and 2 to ensure that the health and safety of the public would not be endangered and that the structures, systems, and components met the safety-related requirements. The test program was directed by Alabama Power Company's (APC) startup staff. Technical assistance was provided during the test program by Westinghouse Electric Corporation, Bechtel Power Corporation, and vendor technical representatives.

An advisory committee was formed of onsite personnel from APC, Bechtel Power Corporation, and Westinghouse Electric Corporation. This advisory committee was called the joint test group. The function of the joint test group was to assist in resolving various discrepancies or deficiencies that occurred during the testing program; to review, comment on, and recommend approval of system test procedures; and to review and recommend approval of test data.

14.1 TEST PROGRAM

The test program for FNP Units 1 and 2 was divided into three phases: Phase I, construction tests; Phase II, acceptance and preoperational test; and Phase III, fuel loading and initial startup testing. This chapter is limited to the discussion of Phase II preoperational testing and all of Phase III.

Phase II preoperational testing started when the installation of individual systems or subsystems was completed and continued through the successful completion of a hot functional test of the reactor coolant system until the beginning of fuel loading. Primary objectives of preoperational testing were to verify that equipment and systems perform in accordance with design and safety requirements for initial fuel loading.

After the operating license was received, Phase III testing began with fuel loading, continued through initial criticality, zero power operation, and ascension to power operation, and was completed when the plant was fully licensed and placed in commercial operation. Primary objectives of Phase III testing were to verify nuclear parameters and that all portions of the plant can operate at rated capacity without endangering the health and safety of the public.

To ensure quality control, all testing was conducted using approved written procedures. Test procedures were prepared, reviewed, approved, and executed; the test results were evaluated and the completed test approved and documented in accordance with established administrative procedures.

When procedure changes or system modifications were required during the test program, they were effected in accordance with administrative procedures established for this purpose.

The testing program for FNP Units 1 and 2 was developed, administered, and conducted by APC's startup and operating staffs and by the Westinghouse startup services group under the direction of the plant manager.

Compliance with regulations, with regard to the development of the preoperational and startup test programs, was in accordance with Regulatory Guide 1.68, Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors, as outlined in subsections 14.1.2 and 14.1.3. The modified startup physics test program for Unit 2 is described in an APC letter to the Nuclear Regulatory

Commission (NRC) dated July 7, 1980. The preparation and content of procedures for preoperational tests, fuel loading, startup to criticality, and initial ascension to power was in accordance with Appendix C of Regulatory Guide 1.68. Participation in the testing program by the plant operations staff demonstrated that the operations staff was knowledgeable about the plant and plant operating procedures and was prepared to operate the facility in a safe manner.

Demonstration and evaluation of the procedures for operating the plant are discussed in subsection 14.1.4.

Startup observed housekeeping practices during the testing program that were in compliance with the requirements of Regulatory Guide 1.39, Housekeeping Requirements for Water Cooled Nuclear Power Plants. While there were few instances in which startup had direct control of area housekeeping, every effort was made to coordinate with construction and operation forces to ensure the maintenance of those standards which were applicable to startup activities. Control of facilities, utilization of tools and equipment, training of personnel, and maintenance of inspection requirements and records was in accordance with the applicable operations procedures which were prepared to comply with Regulatory Guide 1.39.

Startup activities involved with the installation, inspection and testing of instrumentation, electrical equipment, mechanical equipment, and systems were accomplished in accordance with Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment, and American National Standards Institute (ANSI) N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants.

Equipment was inspected by startup prior to acceptance from construction forces to verify that installation standards were met and that all nonconforming items and temporary conditions were properly identified. Any nonconforming items or temporary conditions which were identified/ necessitated by the testing program were handled in accordance with startup administrative procedures.

Instrument and electrical/mechanical devices were calibrated prior to the performance of Phase II tests. Phase II and III tests demonstrated that the installation and operation of these instruments and devices were in accordance with design requirements.

The program for the testing of systems was in compliance with the requirements of Regulatory Guide 1.30, ANSI N45.2.8, and other regulations and standards as referenced in this chapter.

The startup staff was responsible for ensuring the quality control of all safety-related startup activities as described in subsection 13.4.2 and section 17.2.

Preoperational cleaning and layup, and associated activities involving the cleanliness of safety-related fluid systems, were performed in accordance with Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants. These activities included preparation of procedures for cleaning and layup of systems and preparation of procedures for work activities performed by startup which could affect the maintenance of installation cleanliness. The administration of the cleaning program including review and approval of procedures and records were in accordance with the administrative procedures/practices discussed in this chapter.

A system of internal audits was established, as described in subsection 17.2.18, to verify that the elements of the operations quality assurance program which were applicable to startup were developed, documented, and implemented in accordance with specified requirements. Implementing procedures delineated the organizational responsibilities; personnel selection, qualification, and training requirements; planning, documentation, and implementation of guidelines; and the requirements for maintenance of records in accordance with ANSI N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants. These procedures included the development of a checklist specifying the various subjects to be audited and the frequency of audits.

14.1.1 ADMINISTRATIVE PROCEDURES (TESTING)

14.1.1.1 Procedure Development, Test Execution, Data Evaluation, and Documentation

The procedure followed for developing and approving test procedures, conducting tests, obtaining approval of test data, and documenting tests and test results for FNP Units 1 and 2 during the test program were as follows. (See figure 14.1-1.)

The responsibility of each planned test was assigned to a system engineer. A writer prepared a test procedure draft which was subject to internal review and a walkthrough of the test procedure. The writer resolved comments resulting from the review and walkthrough and incorporated the necessary changes into the test procedure.

The test procedure was then reviewed by an assigned engineer, members of the joint test group, and plant staff members (as required).

Once satisfied with the procedure, the joint test group recommended approval of the test procedure to the plant manager who, when he concurred, approved the test procedure by affixing his signature to the original. One copy of the approved procedure labeled "official test copy" was released for execution.

A test engineer or the assigned system engineer was responsible for having the test conducted in strict accordance with the official test copy of the approved procedure.

Upon completion of the test, the completed test procedure and test data were reviewed by the assigned system engineer, joint test group members, the plant manager, and other parties (as necessary).

Once the reviewers were satisfied with the test results, the plant manager approved the test results. The official test copy of the test procedure, data sheets, and documentation associated with the test and approval of the test results were filed with the permanent plant records.

For initial fuel loading and startup tests, the procedures were reviewed by the Plant Operations Review Committee (PORC) and approved by the plant manager. Startup test results were reviewed by the PORC and approved by the plant manager prior to ascending to the next power level as required by Regulatory Guide 1.68, November 1973.

14.1.1.2 Personnel Qualification

All startup personnel who were assigned the responsibility and authority to perform project functions involving inspection and testing activities affecting quality were certified according to their level of capability in accordance with Regulatory Guide 1.58, Revision 0, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel. Determination of the applicability of prior experience in the basis for certification was made by the plant manager or his designated representative.

Appropriate training and certification records for each person designated to perform project functions were maintained by the plant manager or his designated representative. Personnel performance evaluations were maintained in the general office and not as a part of the certification records.

For a discussion of plant manager qualifications, see paragraph 13.1.3.2.1.

14.1.2 TEST OBJECTIVES AND PROCEDURES

The following listing is a compilation of the preoperational tests to be conducted during the testing program for FNP Units 1 and 2. Those tests marked with an asterisk (*) are further designated as precritical tests.

1. *Reactor coolant system heatup*
2. *Reactor coolant system at temperature*
3. *Reactor coolant system cooldown*
4. *Reactor coolant system flow measurement**
5. *Reactor coolant system flow coastdown**
6. *Reactor coolant system thermal expansion*
7. *Reactor coolant system leak test**
8. *Reactor coolant system post-hot functional inspection, cleaning, and testing*
9. *Boric acid system*
10. *Boron thermal regeneration system*
11. *Chemical and volume control system*
12. *Automatic reactor control system*
13. *Incore movable detectors**
14. *Nuclear instrumentation system*
15. *Reactor protection time response measurement*
16. *Reactor protection operational check**
17. *Safeguards system operational check*
18. *Rod drive mechanism timing**
19. *Rod control system**
20. *Rod drop time measurement**
21. *Rod position indication system**
22. *Core loading instrumentation**
23. *Power conversion system thermal expansion*
24. *Power conversion system vibration measurements*
25. *Auxiliary feedwater system*
26. *Component cooling water system*
27. *Residual heat removal system*
28. *Fire protection system*

29. *Service water system*
30. *River water system*
31. *Control room ventilation system*
32. *Auxiliary building ventilation system (radioactive portion)*
33. *Plant response to loss of instrument air*
34. *Pressurizer relief tank*
35. *Pressurizer effectiveness test**
36. *Heat tracing system (boric acid)*
37. *120-V instrument power systems*
38. *600-V electrical load centers*
39. *600-V motor control centers*
40. *4160-V electrical system*
41. *Unit auxiliary and startup auxiliary transformers*
42. *Direct current systems*
43. *Communications system*
44. *Emergency diesel generators*
45. *Diesel fuel oil system*
46. *Containment integrated leak rate*
47. *Containment structural integrity*
48. *Containment cooling system*
49. *Containment spray and additive system*
50. *Containment isolation system*
51. *Postaccident containment combustible gas control system*
52. *Penetration room filtration system*
53. *Emergency core cooling system vibration measurement*
54. *Emergency core cooling system thermal expansion*
55. *Safety injection system*
56. *Reactor components and fuel handling tools and fixtures*
57. *Fuel transfer system*
58. *Spent fuel pool cooling system*
59. *Process and area radiation monitoring system*
60. *Personnel monitoring and survey instruments*
61. *Laboratory equipment*
62. *Water quality tests**
63. *Radioactive waste systems*
64. *Reactor coolant pressure boundary leakage detection system*
65. *Service water pond (shared)*

The following synopsis outlines the test objectives, prerequisites, test methods, and acceptance criteria for each preoperational test. Also included are the provisions to simulate normal and abnormal operating conditions, which are incorporated into the test methods where appropriate.

REACTOR COOLANT SYSTEM HEATUP

1.0 Objective

Perform functional checks on the reactor coolant system and associated systems components and instrumentation required to bring the plant from a cold shutdown condition to normal operating temperature and pressure.

2.0 Prerequisites

2.1 Reactor coolant system and all supporting systems valve lineups for normal operation completed and normal flow paths established.

2.2 Reactor coolant system cold hydrostatic test completed.

2.3 Specified preoperational and acceptance tests completed.

2.4 Specified instrumentation and control checkouts and calibrations completed.

2.5 Secondary system ready to receive steam and return feedwater to the steam generators.

2.6 Diesel generators fully operable and ready for emergency power requirements. Batteries and battery chargers are in service.

2.7 Specified systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Establish specified charging and letdown flowrate and seal water flow to the reactor coolant pumps.

3.2 Energize pressurizer heaters and conduct solid system pressure control demonstration.

3.3 Start reactor coolant pumps to commence plant heatup.

3.4 Commence recording reactor coolant pump vibration data.

3.5 Perform chemistry adjustment demonstrations.

3.6 Form pressurizer steam bubble.

3.7 Check operability of pressurizer power-operated relief valves, pressurizer spray valves, and steam generator atmospheric steam dump valves.

3.8 At approximately 100°F intervals, stabilize all system parameters and record required data, measurements, and observations for incore thermocouple and resistance temperature detector

(RTD) cross-calibration, reactor coolant pump vibration measurements, and reactor coolant system thermal expansion measurements.

3.9 Verify ability to maintain steam generator levels by operation of the atmospheric steam dump and the auxiliary feedwater system.

3.10 Continue heatup to specified conditions.

4.0 Acceptance Criteria

4.1 All systems, components, instrumentation, and controls function as described in the Final Safety Analysis Report (FSAR), vendors' instruction manuals, and applicant's inquiries.

4.2 Reactor coolant pump vibration readings are within the values specified in vendors' instruction manuals

REACTOR COOLANT SYSTEM AT TEMPERATURE

1.0 Objective

Perform functional checks on the reactor coolant system and associated systems components and instrumentation required during normal hot plant operation.

2.0 Prerequisites

2.1 Reactor coolant system heatup completed, with conditions of 515°F to 547°F and 2250 ± 25 psig being maintained.

2.2 Specified systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Check the response, stability, and general control characteristics of the pressure control system.

3.2 Transfer process systems controls to remote station. Demonstrate ability to maintain hot shutdown conditions.

3.3 Perform other tests which require the reactor coolant system to be at normal operating no-load temperature and pressure.

3.4 Check operational setpoints of the steam generator safety valves.

3.5 Conduct initial turbine roll test.

4.0 Acceptance Criteria

All systems, components, instrumentation, and controls function as described in the FSAR, vendors' instruction manuals, and applicant's inquiries.

REACTOR COOLANT SYSTEM COOLDOWN

1.0 Objective

Perform functional checks on the reactor coolant system and associated systems components and instrumentation required to bring the plant to the cooled down, depressurized condition.

*2.0 **Prerequisites***

2.1 Reactor coolant system at temperature test completed, with conditions of 515°F to 547°F and 2250 ± 25 psig being maintained.

2.2 Primary water storage tank contains sufficient quantity of Grade A water to accommodate the contraction of the primary coolant during cooldown.

2.3 Specified systems completed to the extent necessary to allow conduct of this test.

*3.0 **Test Methods***

3.1 Secure two reactor coolant pumps and commence plant cooldown by decreasing the set pressure of the steam dump valves.

3.2 Record data as required for incore thermocouple and RTD cross-calibration.

3.3 When reactor coolant temperature and pressure are below 350°F and 450 psig, place the residual heat removal system in operation

3.4 Collapse the steam bubble.

3.5 Continue pressurizer and reactor coolant system cooldown to 140°F and reduce pressure to 50 psig.

3.6 Establish conditions for reactor coolant system draining.

*4.0 **Acceptance Criteria***

All systems, components, instrumentation, and controls function as described in the FSAR, vendors' instruction manuals, and applicant's inquiries.

REACTOR COOLANT SYSTEM FLOW MEASUREMENT

1.0 Objective

Obtain the data to compute actual reactor coolant system flowrates as they relate to the design flowrates.

*2.0 **Prerequisites***

2.1 Core installed.

2.2 Reactor plant is in hot shutdown condition with all control rods fully inserted.

2.3 Reactor coolant pumps operable.

*3.0 **Test Methods**[a]*

3.1 Measure loop temperatures and loop elbow tap Δp indications at hot shutdown conditions with all reactor coolant pumps running.

3.2 Compute actual reactor coolant system flowrate.

*4.0 **Acceptance Criteria***

Reactor coolant system flowrates are verified to design values.

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a. Prior to going critical on Unit 2, with the reactor at 547°F and fully loaded, the measured loop elbow tap Δp was compared to the Unit 1 value to verify gross flowrate with respect to Unit 1. Absolute flow measurements were performed using a new elbow tap procedure at 50 percent power and above.

REACTOR COOLANT SYSTEM FLOW COASTDOWN

1.0 Objective

1.1 Measure the rate at which reactor coolant system flow changes subsequent to reactor coolant pump stops and starts.

1.2 Measure time delays associated with the loss of flow accident.

2.0 Prerequisites

2.1 Core installed.

2.2 Reactor plant is in hot shutdown condition with all control rods fully inserted.

3.0 Test Methods

3.1 Selectively trip reactor coolant pumps from various configurations of pump operation.

3.2 Measure required flow data and response times for each configuration of pump operation.

4.0 Acceptance Criteria

4.1 Time delays associated with the loss of flow accident are within the values specified in the approved test procedure.

4.2 Rate of change of reactor coolant flow is within the limits specified in the approved test procedure.

REACTOR COOLANT SYSTEM THERMAL EXPANSION

1.0 Objective

Verify that the reactor coolant system piping can expand without obstruction during initial heatup to normal operating conditions.

2.0 Prerequisites

2.1 To be performed in conjunction with the reactor coolant system heatup test.

2.2 Hanger lock pins removed and expansion clearances set to the proper cold values.

2.3 Reference points for measurements established.

3.0 Test Methods

3.1 Log cold settings on all hangers.

3.2 Heat up system to normal operating condition.

3.3 Log hot setting movements at specified points in the system.

3.4 Operate power conversion system under transient conditions.

3.5 Log movements.

4.0 Acceptance Criteria

4.1 All hangers remain within cold and hot setpoints.

4.2 Piping movements do not cause piping rubs or interference with other equipment.

4.3 Piping movements do not cause undue stresses on associated pumps or cause misalignments.

REACTOR COOLANT SYSTEM LEAK TEST

1.0 Objective

Verify that there is no leakage past the reactor vessel head and vessel seal following installation of the reactor vessel head after core loading.

2.0 Prerequisites

2.1 Core installed, reactor vessel head installed, and reactor vessel head studs torqued.

2.2 Reactor coolant system pressure integrity verified in accordance with American Society of Mechanical Engineers code prior to core loading.

3.0 Test Methods

3.1 Establish normal operating no-load temperature and pressure conditions for reactor coolant system.

3.2 Increase system pressure to 100 psi above operating pressure and check for leakage past the head and vessel seal.

4.0 Acceptance Criteria

No detectable leakage past reactor vessel head and vessel seal.

REACTOR COOLANT SYSTEM POST-HOT FUNCTIONAL INSPECTION, CLEANING, AND TESTING

1.0 Objective

1.1 Ensure that the reactor coolant system, including the reactor vessel internals, is properly inspected and cleaned after hot functional testing.

1.2 Ensure that baseline inservice inspections are completed and acceptable prior to core loading.

2.0 Prerequisites

2.1 Reactor coolant system cooled down in preparation for draining.

2.2 Preparations completed to the extent possible for removing vessel head and internals.

3.0 Test Methods

3.1 Drain the reactor coolant system.

3.2 Complete preparations for removal of reactor vessel head.

3.3 Remove reactor vessel head and internals.

3.4 Dye check thermal shield fixtures.

3.5 Visually inspect internal clad surfaces of the pressurizer, reactor vessel, and primary side of the steam generators as required for preservice inspection baseline data.

3.6 Flush internals packages with Grade A water.

3.7 Visually inspect internals.

3.8 Examine reactor vessel closure head, studs, nuts, and washers as required for preservice inspection baseline data.

3.9 Perform preservice inspection of reactor vessel shell and nozzle welds.

3.10 Complete final cleanness procedures and inspections of vessel, piping, and components.

4.0 Acceptance Criteria

4.1 Cleanness requirements meet specifications as described in the approved test procedure.

4.2 Inservice inspection data collected and documented in accordance with the approved test procedure.

BORIC ACID SYSTEM

1.0 Objective

Verify proper functioning of equipment and instrumentation utilized in batching, storage, transfer, and recirculation of boric acid solutions.

2.0 Prerequisites

2.1 Boric acid system installation and component checks completed.

2.2 Adequate supply of Grade A water available.

2.3 Steam supply available to batching tank jacket heater.

2.4 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Verify boric acid tank and batching tank level setpoints, controller functions, and steam delivery to batching tank heaters.

3.3 Verify capability of boric acid transfer pumps to deliver solution from the batching tank to the boric acid tanks and to recirculate each boric acid tank.

3.4 Verify capability of supplying emergency boration to the charging pump suction.

3.5 Verify boron injection tank and surge tank recirculation capability and temperature control functions.

4.0 Acceptance Criteria

4.1 System provides for batching, storage, transfer, and recirculation of boric acid solutions in accordance with the FSAR system description and the approved test procedure.

4.2 Interlocks, automatic functions, alarms, flows, and pressures are in accordance with the system description and the approved test procedure.

BORON THERMAL REGENERATION SYSTEM[a]

1.0 Objective

1.1 Operationally check out the boron thermal regeneration system and operate the system with borated letdown flow to determine the operational capabilities of the storage and release of boron at several reactor coolant system boron concentrations.

1.2 Verify calculated storage and release times for finite boron concentrations changes and determine the response times of the demineralizer to letdown flow temperature change.

2.0 Prerequisites

2.1 Boron thermal regeneration system installation and component checks completed.

2.2 The reactor coolant system at normal zero power operating temperature and pressure and borated to specified concentration.

2.3 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Method

3.1 Align the system for normal operation.

3.2 Operate the system in the dilution and the boration phases at specified reactor coolant system boron concentrations.

4.0 Acceptance Criteria

Dilution and boration times and temperature lag times within the design limits as specified in the approved test procedure.

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a. This section, related to testing at various boron concentrations, is applicable only to Unit 1.

CHEMICAL AND VOLUME CONTROL SYSTEM

1.0 Objective

Demonstrate that the chemical and volume control system performs as required during plant operation.

2.0 Prerequisites

2.1 Chemical and volume control system installation and component checks completed.

2.2 Reactor coolant system at the condition specified in the approved test procedure.

2.3 Adequate supply of Grade A water available in refueling water storage tank.

2.4 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align chemical and volume control system for normal operation and establish normal flow paths.

3.2 Verify capacities of letdown orifices and pressure drop of reactor coolant filter.

3.3 Check operation of the letdown line temperature and pressure controllers with the demineralizers bypassed.

3.4 Verify operation of the excess letdown and seal water subsystems.

3.5 Verify flowrates and pressure drops of demineralizers.

3.6 Verify charging pumps flowrates and the seal water flowrate for each reactor coolant pump.

3.7 Verify volume control tank level controller operation.

3.8 Check reactor makeup control system response to inventory changes of volume control tank. Verify flowrates in the dilute, alternate dilute, and borate modes.

3.9 Verify regulation of hydrogen supply to volume control tank.

4.0 Acceptance Criteria

4.1 System performance is in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

4.2 Interlocks, automatic functions, flows, alarms, temperatures, and pressures are in accordance with the system description and the approved test procedure.

AUTOMATIC REACTOR CONTROL SYSTEM

1.0 Objective

Verify proper functioning of the automatic reactor control system prior to power operation. (System performance in maintaining coolant average temperatures will be demonstrated during initial operations under steady state and transient conditions.)

2.0 Prerequisites

2.1 Automatic reactor control system installation and component checks completed.

2.2 All process instrumentation channels providing inputs to the automatic reactor control system calibrated and aligned.

2.3 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Energize equipment for the specified warmup period.

3.2 Perform static and dynamic test of automatic reactor control system as prescribed in vendors' instructional manuals and other procedures as appropriate.

4.0 Acceptance Criteria

System performance is in accordance with vendors' instruction manuals.

INCORE MOVABLE DETECTORS

1.0 Objective

Verify proper response of the individual channels of instrumentation and the ability to accurately position the detectors of the incore movable detector system.

2.0 Prerequisites

2.1 Incore movable detector system installation and component checks completed.

2.2 Manual local operation has been checked using a dummy cable.

2.3 Core installed.

2.4 Gas purge system and leak detection system installation and component checks completed.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Verify proper operation of all transfer devices, isolation valves, safety and limit switches, and readout and control equipment.

3.3 Compare position readouts with observed position of detectors.

4.0 Acceptance Criteria

4.1 System provides mapping capability as described in vendors' instruction manuals and FSAR system description.

4.2 Gas purge and leak detection system components, alarms, and interlocks function as described in vendors' instruction manuals and FSAR system description.

NUCLEAR INSTRUMENTATION SYSTEM

1.0 Objective

Verify that the nuclear instrumentation system performs the required indication and control functions through the source, intermediate, and power ranges of operation.

2.0 Prerequisites

2.1 Nuclear instrumentation system installed with calibration and initial alignments completed.

2.2 System energized for a minimum of 4 h prior to commencing this test.

3.0 Test Methods

Using the installed test facilities, verify proper performance of instrumentation, including output signals to the reactor control system, reactor protection system, and remote indications.

4.0 Acceptance Criteria

System performance is in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

REACTOR PROTECTION TIME RESPONSE MEASUREMENT

1.0 Objective

Verify the reactor protection system response times and functioning of each trip path, excluding the sensors.

2.0 Prerequisite

2.1 Reactor plant in cold shutdown condition prior to initial criticality.

2.2 All instrumentation and reactor protective systems installation checks and calibrations completed.

3.0 Test Methods

3.1 Utilizing test panels and temporary instrumentation as required, measure the time response and verify the functioning of each trip path in the reactor protective circuitry.

3.2 If measured times are greater than those in the specifications, analyze the test data to determine the suitability of the actual response times and corrective actions to be taken.

4.0 Acceptance Criteria

4.1 Response times of the individual trip paths are less than the maximum allowable times specified in the approved test procedure.

4.2 System functional responses to the various input signals are in accordance with vendors' instruction manuals and FSAR system description.

REACTOR PROTECTION OPERATIONAL CHECK

1.0 Objective

Verify the correct installation and proper operation of the reactor trip portion of the reactor protection system.

2.0 Prerequisites

2.1 Reactor plant in cold shutdown condition.

2.2 All instrumentation and reactor protection systems installation checks and calibrations completed.

3.0 Test Methods

3.1 Utilizing the appropriate train test panels, conduct individual tests of each train's tripping logic.

3.2 Conduct overall logic test for both trains simultaneously.

4.0 Acceptance Criteria

4.1 System performance is in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

4.2 Systems demonstrate the required redundancy in accordance with applicable design codes.

SAFEGUARDS SYSTEM OPERATIONAL CHECK

1.0 Objective

Verify the operation of the safeguards logic systems for all conditions of trip logic.

2.0 Prerequisites

2.1 All instrumentation and safeguards systems installation checks and calibration completed.

2.2 Reactor plant in cold shutdown condition prior to core loading.

3.0 Test Methods

3.1 Conduct individual train logic tests.

3.2 Conduct overall logic test for both trains simultaneously.

3.3 Verify redundant tripping of each safeguard channel through to the relay or controller that actuates the safeguards device.

4.0 Acceptance Criteria

4.1 System performance is in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

4.2 System demonstrates the required redundancy in accordance with applicable design codes.

ROD DRIVE MECHANISM TIMING

1.0 Objective

Verify proper timing of each rod control system slave cycler and conduct an operational check of each full length control rod drive mechanism.

2.0 Prerequisites

2.1 All full length control rod drive mechanism equipment installed with rod cluster control assemblies attached.

2.2 Reactor coolant system filled and vented.

2.3 Boron concentration equal to or greater than that required for refueling shutdown.

2.4 Baseline count rates established for each source range channel.

2.5 Test is to be performed at cold and hot shutdown conditions.

3.0 Test Methods

3.1 Verify the timing of each power cabinet's slave cycler.

3.2 Conduct individual mechanism operational checks by withdrawing and inserting each mechanism a specified number of steps while obtaining an oscillograph trace.

4.0 Acceptance Criteria

Mechanism timing and operational checks verified in accordance with the approved test procedure.

ROD CONTROL SYSTEM

1.0 Objective

Verify that the full length rod control system satisfactorily performs all required control and indication functions.

2.0 Prerequisites

2.1 Rod drop time measurement, rod position indication system test, and rod drive mechanism timing test completed.

2.2 Both source range protection channels in operation and an audible signal from one channel available in the control room.

2.3 Reactor plant in hot shutdown condition.

2.4 Boron concentration equal to or greater than that required for refueling shutdown.

3.0 Test Methods

3.1 Alternately withdraw and insert all banks the specified number of steps, verifying rod positions and status and alarm annunciator operation.

3.2 Check bank overlap settings by withdrawing and inserting control banks with bank selector switch in manual.

3.3 Conduct simultaneous rod drop test by initiating a manual scram with all banks withdrawn 50 steps or greater.

4.0 Acceptance Criteria

4.1 System performs all required control and indication functions in accordance with FSAR system description and the approved test procedure.

4.2 Ability to manually scram the reactor is satisfactorily demonstrated.

ROD DROP TIME MEASUREMENT

1.0 Objective

Determine the drop time for each full length control rod at cold no-flow, cold full-flow, and hot full-flow conditions.

2.0 Prerequisites

2.1 Core installed and reactor vessel head in place.

2.2 Boron concentration equal to or greater than that required for refueling shutdown.

2.3 Rod position indication system operable.

2.4 Both source range protection channels in operation with an audible signal from one channel available in the control room.

3.0 Test Methods

3.1 Withdraw selected bank to the fully withdrawn position.

3.2 Conduct individual rod drop tests, recording rod drop time, rod travel time, and other specified data.

3.3 Repeat for all banks of full length rods in required conditions of flow and temperature.

4.0 Acceptance Criteria

Drop time for all rods is less than the maximum value specified in the plant technical specifications.

ROD POSITION INDICATION SYSTEM

1.0 Objective

1.1 Demonstrate that the rod position indication system performs the required indication and alarm functions for each full length rod cluster control assembly.

1.2 Demonstrate performance of the full length rod cluster control assemblies over their full range of travel.

2.0 Prerequisites

2.1 Reactor coolant system at normal operating no-load temperature and pressure.

2.2 Boron concentration equal to or greater than that required for refueling shutdown.

2.3 Cold shutdown alignment and adjustments of rod position indication system completed.

3.0 Test Methods

3.1 Simulate the operation of rod position indication system for each rod cluster control assembly. Observe indications and alarms for proper operation.

3.2 Alternately insert and withdraw each control bank in selected increments. Collect data to calibrate the applicable step counters.

4.0 Acceptance Criteria

4.1 All indicators and alarms function in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

4.2 Rod cluster control assembly and bank position indicators properly calibrated.

4.3 Detector output voltages and rod bottom bistable setpoints are in accordance with the approved test procedure.

CORE LOADING INSTRUMENTATION

1.0 Objective

Verify proper operation of the source range instrumentation channels prior to fuel loading operations.

2.0 Prerequisites

2.1 Temporary source range instrumentation installation checks completed.

2.2 Permanent source range channels operable.

3.0 Test Methods

3.1 Perform calibration of each source range channel.

3.2 Verify response of each channel to a neutron source.

3.3 Verify audible signal from at least one permanent channel available in control room.

4.0 Acceptance Criteria

Instrumentation provides monitoring of source range neutron level for loading fuel as required by the plant technical specifications.

POWER CONVERSION SYSTEM THERMAL EXPANSION

1.0 Objective

Demonstrate that piping and hanger deflections are within acceptable limits during heatups, cooldowns, and power transients.

2.0 Prerequisites

2.1 Power conversion system operable and available for transient operations.

2.2 Hanger lock pins removed and expansion clearances set to the proper cold values.

2.3 Reference points for measurements established.

3.0 Test Methods

3.1 Log cold settings on all hangers.

3.2 Heat up system to normal operating conditions.

3.3 Log hot setting movements at specified points in the system.

3.4 Operate power conversion system under transient conditions.

3.5 Log movements.

4.0 Acceptance Criteria

4.1 All hangers remain within cold and hot setpoints.

4.2 Piping movements do not cause piping rubs or interference with other equipment.

4.3 Piping movements do not cause undue stresses on associated components or cause misalignments.

4.4 Piping and components return to approximate baseline position on cooldown.

POWER CONVERSION SYSTEM VIBRATION MEASUREMENTS

1.0 Objective

Demonstrate that power conversion system vibration levels are within acceptable limits.

2.0 Prerequisites

Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Line up system for normal operation.

3.2 Operate each selected item of rotating equipment at various plant conditions.

3.3 Measure vibration levels at specified locations for each plant condition.

4.0 Acceptance Criteria

Vibration levels within the limits stated in the vendors' instruction manuals and applicant's inquiries.

AUXILIARY FEEDWATER SYSTEM

1.0 Objective

Demonstrate that the auxiliary feedwater system is capable of providing adequate quantities of feedwater for the removal of decay heat.

2.0 Prerequisites

2.1 Auxiliary feedwater system installation and component checks completed.

2.2 To be performed in conjunction with hot functional testing of the reactor coolant system.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Verify manual and automatic initiation of system.

3.3 Verify pump performance curve.

3.4 Verify ability to control steam generator levels within specified band.

3.5 Verify operation of motor-operated supply valves from service water system (not done during hot functional test).

3.6 Simulate actuation signals to steam generator auxiliary feed inlet valves.

4.0 Acceptance Criteria

4.1 Feedwater flow capability of the system meets the design requirements.

4.2 All system interlocks, alarms, and logic function in accordance with vendors' instruction manuals, FSAR system description, and the approved test procedure.

4.3 All required valve operations take place within the time limits specified in the approved test procedure.

COMPONENT COOLING WATER SYSTEM

1.0 Objective

Demonstrate the capability of the component cooling system to supply adequate cooling water in all modes of operation.

2.0 Prerequisites

2.1 Component cooling system installation and component checks completed.

2.2 Adequate supply of demineralized water available.

2.3 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for operation and establish normal flow paths and rates.

3.2 Initiate safeguards actuation signal and evaluate postaccident operation.

3.3 Demonstrate operation for normal plant cooldown.

4.0 Acceptance Criteria

4.1 System flow requirements are met for all modes of operation.

4.2 System response to safeguards actuation signal is in accordance with FSAR system description and the approved test procedure.

RESIDUAL HEAT REMOVAL SYSTEM

1.0 Objective

Demonstrate the capability of the residual heat removal system to maintain the specified design cooldown rate of the reactor coolant system.

2.0 Prerequisites

2.1 To be performed in conjunction with reactor coolant system cooldown test.

2.2 Residual heat removal system installation and component checks completed.

3.0 Test Methods

3.1 When reactor coolant system temperature and pressure have been reduced to less than 350°F and 425 psig, place the residual heat removal system in service.

3.2 Adjust heat exchanger hand control valves and flow control valves to obtain specified cooldown rate.

4.0 Acceptance Criteria

4.1 System is capable of establishing and maintaining the specified cooldown rate in accordance with design requirements and FSAR system description.

4.2 System flow, pressure, interlock operation, and automatic functions are in accordance with design requirements, FSAR system description, and the approved test procedure.

FIRE PROTECTION SYSTEM

1.0 Objective

Demonstrate that the fire protection system is capable of providing adequate fire protection under all conditions, including loss of power to the electric-driven pump.

2.0 Prerequisites

2.1 Adequate supply of water available in fire water storage tanks.

2.2 Fire protection system installation and component checks completed

3.0 Test Methods

3.1 Align the system for operation and establish normal flow paths.

3.2 Check operation of water sprinkler, chemical, and cooling tower deluge systems. (Cooling tower deluge system has been removed with installation of new towers.)

3.3 Verify pump heads and flowrates under both normal and emergency power conditions.

3.4 Verify operation and response of detector systems.

4.0 Acceptance Criteria

4.1 Alarms, interlocks, and detection devices function as described in vendors' instruction manuals and applicant's inquiries.

4.2 System is capable of providing protection in accordance with applicable fire protection codes.

SERVICE WATER SYSTEM

1.0 Objective

Demonstrate the capability of the service water system to provide adequate cooling water in both the normal and engineered safeguards modes of operation.

2.0 Prerequisites

2.1 Service water system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Method

3.1 Align system for normal operation.

3.2 Verify pump flowrates.

3.3 Simulate a safety injection actuation signal.

3.4 Demonstrate performance from emergency power source.

3.5 Demonstrate normal and emergency recirculation to the pond.

4.0 Acceptance Criteria

4.1 System response to actuation signal is in accordance with FSAR system description and the approved test procedure.

4.2 System flow, pressure, and automatic functions are in accordance with design requirements, FSAR system description, and the approved test procedure.

RIVER WATER SYSTEM

1.0 Objective

Verify that the river water system provides adequate water flow to the storage pond/service water intake structure.

2.0 Prerequisites

2.1 River water system installation and component checks completed.

2.2 Service water intake structure wet pit completed.

2.3 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Shift to alternate river water supply line.

3.3 Check response of system to high and low pond level signals.

3.4 Check operation of river water system in each mode of operation from both normal and emergency power sources.

4.0 Acceptance Criteria

4.1 System flow in accordance with design requirements for all modes of operation.

4.2 System responds to controls in each mode of operation as required by design.

CONTROL ROOM VENTILATION SYSTEM

1.0 <u>Objective</u>

Demonstrate that the control room ventilation system is capable of providing a controlled environment during normal and abnormal conditions.

2.0 <u>Prerequisites</u>

2.1 Control room ventilation system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 <u>Test Methods</u>

3.1 Align system for normal operation.

3.2 Simulate smoke detection and recirculation signals and observe system response.

3.3 Measure airflows and temperatures maintained at specified locations in each mode of operation.

4.0 <u>Acceptance Criteria</u>

4.1 The dampers and fans respond to smoke detection and recirculation signals in accordance with FSAR system description and the approved test procedure.

4.2 Heating, cooling, and recirculation capabilities meet design requirements.

AUXILIARY BUILDING VENTILATION SYSTEM (RADIOACTIVE PORTION)

1.0 Objective

Demonstrate that the auxiliary building radioactive ventilation system functions in its various modes of operation and that it is capable of providing a controlled environment.

2.0 Prerequisites

2.1 Auxiliary building ventilation system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Test supply and exhaust fans for capacity and static pressure.

3.3 Test positioning on all pneumatically operated dampers.

3.4 Measure airflows and temperatures maintained at specified locations in each mode of operation.

4.0 Acceptance Criteria

4.1 System provides for control and disposal of airborne radioactivity in accordance with design requirements and FSAR system description.

4.2 Environment control maintained in all modes of operation in accordance with design requirements.

PLANT RESPONSE TO LOSS OF INSTRUMENT AIR

1.0 Objective

Demonstrate that pneumatically operated valves fail to their safe position on a loss of instrument air.

2.0 Prerequisites

2.1 Instrument air system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

Note: Specific valves and systems may be tested individually.

3.1 Align system for normal operation.

3.2 Reduce instrument air pressure to 0 psig.

3.3 Observe the response of pneumatically operated valves during loss of air pressure and record the position to which each valve fails.

4.0 Acceptance Criteria

Pneumatically operated valves fail to their safe position as specified in the approved test procedure.

PRESSURIZER RELIEF TANK

1.0 Objective

Verify that the pressurizer relief tank provides for adequate control of the discharge from the primary power reliefs and safety valves.

2.0 Prerequisites

2.1 Hydrostatic test of pressurizer relief tank completed.

2.2 Pressurizer relief tank installation checks completed.

2.3 Radioactive waste disposal system completed to the extent necessary to allow conduct of this test.

2.4 Adequate supply of Grade A water available.

3.0 Test Methods

3.1 Fill pressurizer relief tank with Grade A water.

3.2 As pressure increases, verify alarms, interlock operations, and spray flow control.

3.3 Demonstrate ability to maintain nitrogen blanket in pressurizer relief tank.

3.4 Verify transfer flow paths from pressurizer relief tank.

4.0 Acceptance Criteria

Pressurizer relief tank provides for control and disposal of primary plant coolant discharge in accordance with design requirements and FSAR system description.

PRESSURIZER EFFECTIVENESS TEST

1.0 Objective

1.1 Establish proper continuous spray flowrate.

1.2 Verify pressurizer normal control spray effectiveness.

1.3 Verify pressurizer heater effectiveness.

2.0 Prerequisites

2.1 Core installed.

2.2 Plant is in hot shutdown condition at approximately the normal operating no-load temperature and pressure.

3.0 Test Methods

3.1 Adjust continuous spray flowrate to the minimum which results in a 200°F or less ΔT between the pressurizer and spray lines and which keeps the spray line low temperature alarms clear.

3.2 Check normal control spray effectiveness by spraying down to approximately 2000 psig.

3.3 Check heater effectiveness by energizing all heaters with power-operated relief valves in close and spray and level controls in manual. Allow pressure to increase to approximately 2300 psig.

4.0 Acceptance Criteria

4.1 Continuous spray flow adjusted as specified in step 3.1.

4.2 Heater and normal control spray effectiveness are in accordance with design requirements and the approved test procedure.

HEAT TRACING SYSTEM (BORIC ACID)

1.0 Objective

Demonstrate the ability of the heat tracing system to maintain proper temperature control in the various piping systems involved in transporting/storing boric acid solutions.

2.0 Prerequisites

2.1 Heat tracing system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Energize heat tracing system.

3.2 Systematically place systems involved in transporting/storing boric acid in operation and establish transfer flow paths.

3.3 Monitor temperatures maintained by each heat tracing circuit at specified locations in each system.

4.0 Acceptance Criteria

Each heat tracing circuit maintains temperature as specified in the approved test procedure.

120-V INSTRUMENT POWER SYSTEMS

1.0 Objective

Demonstrate the capabilities of the 120-V vital instrument power system and the 120 V-ac regulated instrument power system to supply power to essential and nonessential instrumentation and control loads under normal and emergency conditions.

2.0 Prerequisites

2.1 The 120-V instrument power systems installation and component checks completed.

2.2 The 600 V-ac system available.

2.3 The 125 V-dc system operable.

3.0 Test Methods

3.1 Energize 120-V instrument power buses from their normal power sources.

3.2 Demonstrate ability to transfer each vital instrument bus manually to a regulated instrument bus and back to its static inverter.

3.3 Trip the normal power supplies to the static inverters. Verify automatic transfer to alternate dc source. Verify transfer back to normal supply when reenergized.

3.4 Demonstrate ability to transfer each regulated instrument panel manually to its alternate source.

4.0 Acceptance Criteria

4.1 All vital and regulated buses or panels can be manually transferred to alternate sources.

4.2 Voltage and frequency changes resulting from transient conditions do not exceed the design requirements.

4.3 All system interlocks and alarms function properly.

600-V ELECTRICAL LOAD CENTERS

1.0 Objective

1.1 Verify that the 600-V safeguard and nonsafeguard load centers can be energized from their normal and alternate sources.

1.2 Verify that electrical and mechanical interlocks function properly.

2.0 Prerequisites

2.1 Meters, relays, and protective devices calibrated and tested.

2.2 The 125 V-dc and 4.16-kV buses energized.

2.3 Phase rotation checked on 600-V buses.

3.0 Test Methods

3.1 Close 4.16-kV breakers to energize load center transformers and buses.

3.2 Measure voltage and verify phase relationship.

3.3 Shift buses to alternate power sources as applicable.

3.4 Initiate loss of power and safeguards actuation signals.

4.0 Acceptance Criteria

4.1 The 600-V safeguard and nonsafeguard buses are capable of being energized from their normal and alternate sources, and proper phase relationship is exhibited.

4.2 The 600-V load centers respond correctly to a loss of station power and safeguards actuation signals.

4.3 Interlocks function as described in vendors' instruction manuals and applicant's inquiries.

600-V MOTOR CONTROL CENTERS

1.0 Objective

1.1 Verify that the 600-V safeguard and nonsafeguard motor control centers can be energized from their normal and emergency sources.

1.2 Verify that electrical and mechanical interlocks function properly.

2.0 Prerequisites

2.1 Meters, relays, and protective devices calibrated and tested.

2.2 The 125 V-dc system available.

2.3 The 600-V buses energized.

2.4 Phase rotation checked on motor control centers.

3.0 Test Methods

3.1 Rack in and close motor control center supply breakers.

3.2 Manually transfer motor control centers to emergency power supplies as applicable.

3.3 Measure voltage and verify phase relationship.

3.4 Initiate loss of power and safeguards actuation signals.

4.0 Acceptance Criteria

4.1 Safeguard and nonsafeguard motor control centers are capable of being energized from their normal and emergency sources, and proper phase relationship is exhibited.

4.2 The 600-V motor control centers respond correctly to a loss of station power and safeguards actuation signals.

4.3 All system interlocks and alarms function properly.

4160-V ELECTRICAL SYSTEM

1.0 Objective

1.1 Verify that the 4160-V buses can be energized from their respective normal and alternate source.

1.2 Verify that all electrical and mechanical interlocks function properly.

2.0 Prerequisites

2.1 Meters, relays, and protective devices calibrated and tested.

2.2 The 125 V-dc system available.

2.3 Phase rotation checked on 4160-V buses.

3.0 Test Methods

3.1 Rack in and close 4160-V breakers to energize associated 4160-V buses.

3.2 Record voltage and verify phase relationship.

3.3 Shift buses to alternate power sources as applicable.

3.4 Initiate loss of power and safeguards actuation signals.

4.0 Acceptance Criteria

4.1 The 4160-V buses are capable of being energized from their normal and alternate source, and proper phase relationship is exhibited.

4.2 The 4160-V system responds correctly to a loss of power and safeguards actuation signals.

4.3 All system interlocks and alarms function properly.

UNIT AUXILIARY AND STARTUP AUXILIARY TRANSFORMERS

1.0 Objective

1.1 Demonstrate the capability of the unit auxiliary and startup auxiliary transformers to supply electrical power to the 4160-V buses.

1.2 Verify operation of protective devices and functional operation of controls and interlocks.

2.0 Prerequisites

2.1 All meters, relays, and protective devices calibrated and tested.

2.2 The 125 V-dc system available.

2.3 All erection work on transformers and switchgear completed.

2.4 Transformer oil and gas systems tested and in service.

2.5 Isolated phase bus tested and ready for service.

2.6 Breaker controls and transfer scheme verified.

2.7 PT and CT circuits checked for polarity and continuity.

3.0 Test Methods

3.1 Simulate signals to temperature controls and verify operation of transformer oil pumps and fans.

3.2 Simulate signals to verify annunciators for transformer protective devices.

3.3 Verify dead bus transfer capability to start up auxiliary transformers when unit auxiliary transformers are deenergized.

4.0 Acceptance Criteria

Transformers provide reliable source of electrical power to 4160-V buses in accordance with design requirements and FSAR system description.

DIRECT CURRENT SYSTEMS

1.0 Objective

Demonstrate the capability of the dc system to provide a source of reliable, uninterruptible dc power for all normal and emergency instrumentation, control, and power loads.

2.0 Prerequisites

2.1 The 600 V-ac power available.

2.2 Battery room ventilation system operable.

2.3 Batteries, battery chargers, and dc distribution system, including protective devices, installation, and component checks, completed.

3.0 Test Methods

3.1 Energize the battery chargers.

3.2 Adjust alarms and interlocks.

3.3 Discharge the batteries at a controlled rate and determine Ah capacity.

3.4 Adjust chargers to supply dc load and charge batteries simultaneously.

3.5 Deenergize battery chargers while the applicable busses are carrying their rated station power.

4.0 Acceptance Criteria

4.1 All system interlocks and alarms function properly.

4.2 Batteries are capable of supplying plant dc power upon deenergization of their chargers.

4.3 Battery chargers are capable of maintaining normal bus loads concurrently with charging the batteries.

COMMUNICATIONS SYSTEM

1.0 Objective

1.1 Demonstrate the adequacy of the plant public address system, intracommunication between all local stations, and interconnection to commercial telephone service.

1.2 Demonstrate that the evacuation signal can be heard from any location in the plant under all required conditions.

2.0 Prerequisites

2.1 All communications systems installation and component checks completed.

2.2 Sound levels established for locations where noise levels might interfere with communications.

3.0 Test Methods

3.1 Test the portable stations, hand set stations, and jack stations for proper operation in all modes.

3.2 Test interconnection to commercial phone service.

3.3 Test all alarms.

3.4 Shift applicable equipment to alternate power sources and verify operation.

4.0 Acceptance Criteria

4.1 Communication system provides for paging, normal plant communications, interconnection to commercial telephone service, and alarm signaling in accordance with design requirements and FSAR system description.

4.2 Evacuation alarm can be heard from any location in the plant.

EMERGENCY DIESEL GENERATORS

1.0 Objective

1.1 Demonstrate manual start and synchronization of the diesel generators.

1.2 Demonstrate automatic start and sequencing of diesel generators. Demonstrate load carrying capacity of diesel generators.

1.3 Demonstrate independence among redundant, onsite power sources and their load groups.

2.0 Prerequisites

2.1 Station batteries charged and dc control power available.

2.2 Relays calibrated and all normal bus protective devices checked and in service.

2.3 Diesel engine auxiliary systems installation, component checks, and acceptance tests completed as specified.

2.4 Diesel room ventilation and fire protection systems operable.

3.0 Test Methods

3.1 Demonstrate manual start and synchronization of each diesel generator.

3.2 Verify diesel generator response to engineered safeguards actuation signals, 4160-V buses undervoltage signals, and low pond level signals.

3.3 Verify timing of diesel generators starting sequence.

3.4 Verify capability to control diesel generators in all modes of operation.

3.5 Verify load group assignments of onsite emergency power systems as required in Regulatory Guide 1.41.

3.6 Conduct load carrying duration test.

4.0 Acceptance Criteria

4.1 Regulators function to regulate and maintain voltage in all modes of operation in accordance with design requirements.

4.2 Diesel generators function in maintaining the 4160-V emergency buses in accordance with design requirements, FSAR system description, and the approved test procedure.

4.3 Diesel generators do not overspeed when load is removed.

4.4 Each redundant onsite power source and its load group can function without any dependence upon any other redundant load group or portion thereof.

4.5 Direct current and onsite ac buses and related loads not under test will be monitored to verify absence of voltage at these buses and loads.

DIESEL FUEL OIL SYSTEM

1.0 Objective

Demonstrate that the diesel fuel oil system supplies adequate quantities of fuel oil to the diesel oil day tanks.

2.0 Prerequisites

2.1 Fire protection system operable.

2.2 Diesel fuel oil system installation and component checks completed.

3.0 Test Methods

3.1 Align system for operation and establish normal transfer flow paths.

3.2 Verify capability to transfer fuel oil at specified rate.

4.0 Acceptance Criteria

Fuel transfer capability of the system meets the design requirements.

CONTAINMENT INTEGRATED LEAK RATE

1.0 Objective

Demonstrate that the containment leak rate is within allowable limits.

2.0 Prerequisites

Containment structural integrity test completed.

3.0 Test Methods

Integrated leak rate testing of the containment will be conducted in accordance with the procedures described in the proprietary Bechtel Corporation Topical Report BN-TOP-1, Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants, Revision 1, November 1, 1972.

4.0 Acceptance Criteria

Integrated leak rate meets the requirements of the applicable Regulatory Guides and the approved test procedure.

CONTAINMENT STRUCTURAL INTEGRITY

1.0 Objective

Verify the structural integrity of the containment building.

2.0 Prerequisites

2.1 Containment penetrations installed and penetration leak tests completed.

2.2 Containment ventilation systems operable to extent required to control containment internal temperature.

3.0 Test Methods

3.1 In accordance with NRC Acceptance Criteria (NRC SER NUREG 75/034 dated May 2, 1975), prior to initial fuel loading, the containment will be subjected to a pressure equivalent to 115 percent of the containment design pressure. This test demonstrates that the containment is capable of resisting the postulated accident pressure. In addition, by measuring the structural response and comparing the results with analytical predictions, the test verifies that the structure does behave as anticipated.

3.2 Instrumentation, measuring systems, pressurization procedure, deformation, strain and temperature measurements, crack pattern mapping, and data acquisition schedules for the preoperational structural integrity test will be in accordance with the proprietary Bechtel Corporation Topical Report BC-TOP-5, Prestressed Concrete Nuclear Reactor Containment Structures, Revision 1, December 1972.

4.0 Acceptance Criteria

The containment structure meets structural integrity requirements as required by applicable Regulatory Guides and the approved test procedure.

CONTAINMENT COOLING SYSTEM

1.0 *Objective*

Demonstrate that the containment cooling system is capable of providing adequate ventilation and cooling in normal operation and in the engineered safeguards mode of operation.

2.0 *Prerequisites*

2.1 Containment penetration installed.

2.2 Containment cooling system installation and component checks completed.

2.3 Service water system operable.

2.4 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 *Test Methods*

3.1 Align system for normal operation.

3.2 Test capacity and static pressure of fans in various operating configurations.

3.3 Simulate safety injection signal and observe system response.

4.0 *Acceptance Criteria*

4.1 System response to safety injection actuation signal is in accordance with design criteria, FSAR system description, and the approved test procedure.

4.2 System interlocks, instrumentation, and alarms function properly.

CONTAINMENT SPRAY AND ADDITIVE SYSTEM

1.0 Objective

Demonstrate the capability of the containment spray and additive system to respond properly to a containment spray actuation signal.

2.0 Prerequisites

2.1 Containment spray and additive system installation and component checks completed.

2.2 Sufficient Grade A water available in the refueling water storage tank and spray additive tank.

3.0 Test Methods

3.1 Align system to recirculate to the refueling water storage tank.

3.2 Align eductor suction to the spray additive tank.

3.3 Initiate a containment spray actuation signal and observe sequencing of active components.

3.4 Remotely initiate recirculation spray flow with each spray pump.

3.5 Measure flowrates and pump heads in both injection and recirculation modes.

3.6 Force air or smoke through each spray nozzle to verify that nozzles are free of obstructions.

4.0 Acceptance Criteria

4.1 System responds to actuation signal and provides adequate cooling in accordance with design criteria, FSAR system description, and the approved test procedure.

4.2 System provides for chemical addition to spray flow in accordance with design requirements and FSAR system description.

CONTAINMENT ISOLATION SYSTEM

1.0 Objective

Demonstrate the capability of the containment isolation system to respond properly to a containment isolation actuation signal.

2.0 Prerequisites

2.1 Containment isolation system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow the conduct of this test.

3.0 Test Methods

3.1 Containment isolation system and the applicable isolation valves in associated systems aligned for normal operation.

3.2 Simulate a safety injection actuation signal.

3.3 Simulate containment isolation actuation signals.

3.4 Record isolation valve response times to the actuation signals.

4.0 Acceptance Criteria

System response to both safety injection and containment isolation actuation signals is in accordance with FSAR system description, design requirements, and the approved test procedure.

POSTACCIDENT CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM

1.0 Objective

Demonstrate the capability of the postaccident containment combustible gas control system to provide for circulation, sample collection, and removal of combustible gases following a loss-of-coolant accident.

2.0 Prerequisites

2.1 System installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Verify remote actuation of active components.

3.3 Check flowrates of postaccident containment mixing fans, reactor cavity hydrogen dilution fans, and postaccident containment air sample fan.

3.4 Demonstrate ability to obtain atmospheric samples from each sample point.

3.5 Verify proper operation of each hydrogen recombiner.

4.0 Acceptance Criteria

4.1 System provides for circulating and sampling containment atmosphere in accordance with FSAR system description.

4.2 Hydrogen recombiners function as described in vendors' instruction manuals and FSAR system description.

PENETRATION ROOM FILTRATION SYSTEM

1.0 <u>Objective</u>

Demonstrate the effectiveness of the penetration room filtration system in controlling the release of containment leakage to the atmosphere.

2.0 <u>Prerequisites</u>

2.1 Test assemblies installed to simulate filter pressure drops.

2.2 Penetration room filtration system installation and component checks completed.

2.3 Associated systems completed to the extent necessary to allow the conduct of this test.

3.0 <u>Test Methods</u>

3.1 With system operating, verify circulation flow paths within the penetration room.

3.2 Simulate containment isolation actuation signal and observe system response.

3.3 Check penetration room leak rate.

3.4 Inhibit operation of the recirculation fan exhaust valve in one system and observe performance of the system.

4.0 <u>Acceptance Criteria</u>

System responds to actuation signals and provides for controlled handling of containment leakage in accordance with FSAR system description and design requirements.

EMERGENCY CORE COOLING SYSTEM VIBRATION MEASUREMENT

1.0 Objective

Verify that emergency core cooling system rotating equipment vibration levels are within acceptable limits.

2.0 Prerequisites

2.1 Emergency core cooling system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation.

3.2 Operate rotating equipment in the various system operating conditions.

3.3 Measure vibration levels at specified points in the system in each mode of operation.

4.0 Acceptance Criteria

Vibration levels within the limits specified in the vendors' instruction manuals and applicable codes.

EMERGENCY CORE COOLING SYSTEM THERMAL EXPANSION

1.0 Objective

Verify that the emergency core cooling system piping can expand without obstruction upon system heatup to operating conditions.

2.0 Prerequisites

2.1 Emergency core cooling system installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow conduct of this test.

3.0 Test Methods

3.1 Record cold baseline data.

3.2 Heat up system to normal operating temperatures.

3.3 Record hot setting movements.

3.4 Record movements due to thermal expansion or contraction during the operation of injection pumps and recirculation pumps.

3.5 Verify that piping and components return to approximately cold baseline position upon cooldown.

4.0 Acceptance Criteria

4.1 Piping movements do not cause piping rubs, misalignments, or excessive hanger deflections.

4.2 Piping and components return to approximate baseline position upon cooldown.

SAFETY INJECTION SYSTEM

1.0 Objective

1.1 Verify operation of the boron injection tank heaters.

1.2 Verify that the boron injection tank remains full during normal operation.

1.3 Verify that safety injection accumulators discharge flow to the reactor coolant system.

1.4 Verify that the system properly responds to a safety injection actuation signal.

1.5 Verify that the flowrates delivered through each injection flow path, using all pump combinations, are within the design specifications (not including the recirculation mode from the containment sump).

1.6 Verify that the high pressure safety injection pumps are capable of taking suction from the residual heat removal pumps.

1.7 Verify that the safety injection pumps will not trip under conditions of maximum flow.

1.8 Verify the operability of the check valves in the safety injection system that are subject to an elevated temperature during normal operation, at as close as possible to accident conditions.

1.9 Verify proper motor-operated valve operation under maximum expected differential pressure conditions.

1.10 Verify that the accumulator isolation valves will open with zero pressure in the reactor coolant system and with normal pressure in the accumulator.

2.0 Prerequisites

2.1 Reactor vessel head removed prior to core loading.

2.2 Boron injection tank is filled and refueling water storage tank is filled to its normal level.

2.3 Installation and calibration checks completed on safety injection system instruments and components.

2.4 Temporary arrangements have been made to use an alternative source of water for the recirculation test.

3.0 Test Methods

3.1 Demonstrate boron injection tank heater operation in automatic and manual modes. Establish normal recirculation path from the boron injection surge tank to the boron injection tank and verify that injection tank level is maintained.

3.2 Establish specified conditions and initiate safety injection signals from each train. Verify proper actuation of active components in response to signals from each train.

3.3 Conduct miniflow tests of safety injection pumps.

3.4 Pressurize accumulators to minimum pressure required to move water and demonstrate injection through cold loop injection valves to reactor vessel from each accumulator.

3.5 Conduct system pressure/flow verifications.

4.0 Acceptance Criteria

4.1 Boron injection tank heaters maintain temperature in accordance with design requirements.

4.2 Boron injection tank remains full during normal operation.

4.3 System response to safety injection signals is in accordance with FSAR system description, design requirements, and the approved test procedure.

4.4 Accumulator injection flow path to reactor vessel is free of obstructions.

4.5 System pressure/flow characteristics meet the design specifications.

REACTOR COMPONENTS AND FUEL HANDLING TOOLS AND FIXTURES

1.0 Objective

Verify the adequacy of the special equipment required for refueling operations.

2.0 Prerequisites

Equipment to be checked out is onsite and inspected in accordance with the routine receiving inspection.

3.0 Test Methods

3.1 Inspect the mating surface fit and grip of each tool.

3.2 Check each tool for smooth performance and complete actuation.

3.3 Check adequacy of locating devices, guides, and chambers.

3.4 Verify operation of all interlocks and/or safety devices.

3.5 Load test all lifting devices.

4.0 Acceptance Criteria

Equipment provides for safe handling of fuel assemblies and reactor components as described in vendors' instruction manuals, applicant's inquiries, and FSAR system description.

FUEL TRANSFER SYSTEM

1.0 <u>Objective</u>

1.1 Provide functional demonstration of the fuel transfer system and fuel handling tools prior to initial core load.

1.2 Provide functional demonstration of the refueling canal water system.

2.0 <u>Prerequisites</u>

2.1 Reactor components and fuel handling tools and fixtures test completed.

2.2 Fuel transfer system and refueling canal water system installation and component checks completed.

2.3 Reactor vessel head and internals stored in the refueling positions.

2.4 Dummy fuel assembly stored in a new fuel storage rack.

3.0 <u>Test Methods</u>

3.1 Demonstrate flooding, draining, and adjusting level in the refueling water canal.

3.2 With canal drained, conduct the various fuel handling evolutions with the dummy fuel assembly.

4.0 <u>Acceptance Criteria</u>

4.1 System provides for storage, transfer, and handling of fuel assemblies in accordance with vendors' instruction manuals, FSAR system description, and design requirements.

4.2 Refueling canal water system provides for flooding, draining, and adjusting level in accordance with FSAR system description and design requirements.

SPENT FUEL POOL COOLING SYSTEM

1.0 Objective

Verify the cooling and purification capabilities of the spent fuel pool cooling system.

2.0 Prerequisites

2.1 Spent fuel pool cooling system installation and component checks completed.

2.2 Adequate supply of Grade A water available.

3.0 Test Methods

3.1 Demonstrate filling and draining the spent fuel pool.

3.2 Demonstrate circulation through demineralizer, heat exchanger, and skimmer loops.

3.3 Demonstrate that the spent fuel pool can be drained only by deliberate action.

4.0 Acceptance Criteria

System provides for filling, draining, and purification of the spent fuel pool in accordance with FSAR system description and design requirements.

PROCESS AND AREA RADIATION MONITORING SYSTEM

1.0 Objective

Demonstrate the capability of the process and area radiation monitoring system to monitor effectively the levels of radiation in the plant area and effluents and to initiate isolation and alarms as required.

2.0 Prerequisites

2.1 Process and area radiation monitoring systems installation and component checks completed.

2.2 Associated systems completed to the extent necessary to allow the conduct of this test.

3.0 Test Methods

3.1 Align system for normal operation. Position valves in associated systems as necessary to allow response to isolation signals.

3.2 Verify proper functioning of system detectors by utilizing test sources and other procedures as appropriate.

3.3 Verify proper system response to simulated alarm conditions by monitoring controller outputs, alarm indications, and the operation of isolation valves where possible.

4.0 Acceptance Criteria

System effectively monitors and responds to levels of radiation in the plant areas and effluents in accordance with vendors' instruction manuals, design requirements, and FSAR system description.

PERSONNEL MONITORING AND SURVEY INSTRUMENTS

1.0 Objective

Verify the proper functioning of all personnel monitoring and radiation survey instruments.

2.0 Prerequisites

All personnel monitoring and radiation survey instruments calibrated within the specified time frame for each instrument.

3.0 Test Methods

Verify proper functioning of all personnel monitoring and radiation survey instruments by exposure to test sources and other procedures as appropriate.

4.0 Acceptance Criteria

Instruments function as specified in vendors' instruction manuals and applicant's inquiries.

LABORATORY EQUIPMENT

1.0 Objective

Verify the proper functioning of laboratory equipment utilized in radiological control processes.

2.0 Prerequisites

All applicable laboratory equipment calibrated within the specified time period for each apparatus.

3.0 Test Methods

Verify proper functioning of each apparatus by exposure to test sources and other procedures as appropriate.

4.0 Acceptance Criteria

Equipment functions as specified in vendors' instruction manuals and applicant's inquiries.

WATER QUALITY TESTS

1.0 Objective

Verify acceptable water quality of reactor coolant system fill and makeup water prior to initial criticality.

2.0 Prerequisites

2.1 Reactor coolant system filled and vented in preparation for initial criticality.

2.2 Reactor makeup system water storage at operating level.

3.0 Test Methods

3.1 Sample reactor coolant system and analyze in accordance with approved plant procedures.

3.2 Sample reactor makeup system and analyze in accordance with approved plant procedures.

4.0 Acceptance Criteria

All analyses are within the limits specified in the plant chemistry specifications.

RADIOACTIVE WASTE SYSTEMS

1.0 Objective

Demonstrate the ability of the radioactive waste systems to provide controlled handling and disposal of solid, liquid, and gaseous radioactive wastes.

2.0 Prerequisites

2.1 Solid waste processing, liquid waste processing, and gaseous waste processing systems installation and component checks completed.

2.2 Demineralized water available to utilize as working fluid.

2.3 Associated systems completed to the extent necessary to allow the conduct of this test.

3.0 Test Methods

3.1 Align radioactive waste systems for normal operation and establish normal flow paths.

3.2 Measure flowrates, capacities, and alarm setpoints as specified.

3.3 Verify proper functioning of all components, controllers, valves, and indicators.

4.0 Acceptance Criteria

Systems provide controlled handling and disposal of radioactive wastes in accordance with vendors' instruction manuals, FSAR system description, and design requirements.

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM

1.0 Objective

Demonstrate system capability of detecting the presence of significant leakage from the reactor coolant loops to the containment atmosphere during normal operations.

2.0 Prerequisites

2.1 Reactor coolant pressure boundary leakage detection system installation and component checks completed.

2.2 Associated system completed to the extent necessary to allow the conduct of this test.

3.0 Test Methods

3.1 Verify proper functioning of containment air particulate monitor and radioactive gas monitor detectors by exposure to standard test sources.

3.2 Verify monitor's flowrates and associated controls, indications, and alarms.

3.3 Verify proper functioning of specific humidity monitoring devices in accordance with vendors' instruction manuals and the approved test procedure.

4.0 Acceptance Criteria

System provides for monitoring and indication of reactor coolant pressure boundary leakage in accordance with FSAR system description, design requirements, and the approved test procedure.

SERVICE WATER POND (SHARED)

1.0 Objective

Verify the seepage from the service water pond.

2.0 Prerequisites

2.1 Construction of the service water pond is complete.

2.2 Construction of the service water intake structure wet pit is complete.

2.3 The service water pond spillway is complete and operational.

2.4 The river water system is operational.

2.5 The river water flume is operational.

2.6 A volume versus elevation relationship has been established for the service water pond.

2.7 The rainfall 12 h prior to the test is less than 0.1 in./h.

3.0 Test Methods

3.1 The service water pond is filled to the normal level.

3.2 A measured water inventory is maintained on the service water pond throughout the test period.

4.0 Acceptance Criteria

Service water pond is deemed acceptable if the test verifies that the seepage rate does not exceed 15 ft^3/s.

14.1.3 FUEL LOADING AND INITIAL OPERATION

Fuel loading began when all prerequisite system tests and operations were satisfactorily completed and the NRC operating license received. Upon completion of fuel loading, the reactor upper internals and pressure vessel head were installed and additional mechanical and electrical tests performed prior to initial criticality. After final precritical tests were completed, initial operation of the reactor began.

The primary objectives of the fuel loading and initial operation phase were as follows:

- *A. To accomplish an orderly and safe initial fuel loading.*
- *B. To accomplish an orderly and safe approach to criticality.*
- *C. To accomplish an orderly and safe ascension to power.*

The procedures which will guide fuel loading, attainment of initial criticality, and ascension to power are described in subsections 14.1.3.1 and 14.1.3.2.

14.1.3.1 Fuel Loading

The reactor containment structure shall have been completed and the containment integrity established prior to commencing loading operations.

Fuel handling tools and equipment shall have been checked out and dry runs conducted in their use and operation.

The reactor vessel and associated components will be in a state of readiness to receive fuel. Water level will be maintained above the bottom of the nozzles and recirculation maintained to ensure a uniform boron concentration. Boron concentration can be increased via the recirculation system or by direct additions to the open vessel.

The overall responsibility and direction of the initial core loading will be exercised by the plant manager. The process of initial core loading will be directed from the charging floor of the containment structure. Procedures for the control of personnel access and the maintenance of containment security will be implemented prior to commencing loading operations.

The initial core configuration is specified as part of the core design studies, conducted well in advance of station startup.

In the event that during core loading operations mechanical damage is sustained to a fuel assembly of a type for which no spare is available onsite, core loading operations will be suspended until an alternate core loading scheme whose characteristics closely approximate those of the initially prescribed pattern has been determined.

The core will be assembled in the reactor vessel, submerged in Grade A water containing enough dissolved boric acid to maintain a calculated core effective multiplication factor $\leq$ 0.90. The refueling cavity will be dry during initial core loading. Core moderator chemistry conditions (particularly boron *concentration) will be prescribed in the core loading procedure documents and verified by chemical analysis of moderator samples taken prior to and at specified intervals during core loading operations.*

Core loading instrumentation consists of two permanently installed source range channels and two temporary incore source range channels plus a third temporary channel which can be used as a spare. The permanent channels are monitored in the main control room by licensed station operators; the temporary channels are installed in the containment structure and monitored by qualified engineering personnel and licensed station operators. At least one permanent channel is equipped with an audible count rate indicator. Both permanent channels have the capability of displaying the neutron flux level on a strip chart recorder. The temporary channels indicate on count rate meters with a minimum of one channel recorded on a strip chart recorder. Minimum count rates of 2 counts/s, attributable to core neutrons, are required on at least two of the four available source range channels at all times following installation of the initial nucleus of eight fuel assemblies.

At least two artificial neutron sources will be introduced into the core at specified points in the core loading program to ensure a minimum count rate of 2 counts/s for adequate monitoring of the core.

Fuel assemblies and inserted components will be placed in the reactor vessel one at a time in accordance with a previously established and approved sequence developed to provide reliable core monitoring while minimizing the possibility of core mechanical damage. The core loading procedure documents include detailed tabular check sheets which will prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks will be made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components. Fuel assembly status boards are maintained throughout the core loading operation both in the main control room and in the containment.

An initial nucleus of eight fuel assemblies, the first of which contains an activated neutron source, is the minimum source fuel nucleus which permits subsequent meaningful inverse count rate monitoring. This initial nucleus is determined by calculation and previous experience to be markedly subcritical (k_{eff} $\leq$ 0.90) under the required conditions of loading.

Each subsequent fuel addition will be accompanied by detailed neutron count rate monitoring to determine that the just loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count rate ratio is not decreasing for unexplained reasons. The results of each loading step will be evaluated before the next prescribed step is started.

Criteria for safe loading require that loading operations stop immediately if:

A. *An unanticipated increase in the neutron count rates by a factor of two occurs on all responding source range channels during any single loading step, after the initial nucleus of eight fuel assemblies are loaded (excluding anticipated change due to detector and/or source movement).*

B. *The neutron count rate on any individual source range channel increases by a factor of five during any single loading step, after the initial nucleus of eight fuel assemblies are loaded (excluding anticipated changes due to detector and/or source movements).*

An alarm in the containment and main control room is coupled to the source range channels with a setpoint at five times the current count rate. This alarm automatically alerts the loading operation personnel of high count rate and requires an immediate stop of all loading operations until the situation is evaluated. In the event the alarm is actuated during core loading and after it has been determined that no hazards to personnel exist, preselected personnel will be permitted to reenter the containment vessel to evaluate the cause and determine future action.

Core loading procedures specify the condition of fluid systems to prevent inadvertent dilution of the reactor coolant, the movement of fuel to preclude the possibility of mechanical damage, and the conditions under which loading can proceed.

14.1.3.2 Initial Operation

Upon completion of core loading, the reactor upper internals and the pressure vessel head were installed and additional mechanical and electrical tests performed prior to initial criticality. The final pressure test was conducted after filling and venting was completed to verify the integrity of the vessel head installation.

Mechanical and electrical tests were performed on the control rod drive mechanisms. These tests included a complete operational checkout of the mechanisms and calibration of the individual rod position indication.

Tests were performed on the reactor trip circuits to test manual trip operation. The actual control rod assembly drop times were measured for each control rod assembly. The reactor control and protection system was checked with simulated signals to produce a trip signal for the various conditions that require plant trip.

At all times when the control rod drive mechanisms were being tested, the boron concentration in the coolant moderator was maintained so that criticality could not be achieved with all control rod assemblies fully withdrawn.

A complete functional electrical and mechanical check was made of the incore movable detector system at operating temperature and pressure. After completion of precritical tests, nuclear operation of the reactor began. This final phase of startup and testing included initial criticality, low power testing, and power level escalation. The purpose of these tests was to establish the plant operational characteristics, to acquire data for the proper calibration of setpoints, and to ensure that operation is within license requirements. A brief description of the testing is presented in the following sections. Table 14.1-1 summarizes the major tests which were performed following initial core loading, and figure 14.1-2 shows the startup test sequence.

14.1.3.2.1 Initial Criticality

The approach to initial criticality was conducted according to approved written procedures which specify the plant conditions, safety and precautionary measures, and specific instructions. The procedures also delineate the chains of responsibility and authority in effect during this period of operation. Alignment of the fluid system was specified to provide controlled start and stop as well as adjustments of the rate of the approach to criticality.

Initial criticality was achieved by a combination of shutdown and control bank withdrawal and reactor coolant system boron concentration reduction.

Inverse count rate ratio monitoring, using data from the normal plant source range instrumentation, was used as an indication of the proximity and rate of approach to criticality. Inverse count rate ratio data were plotted as a function of rod bank position during rod motion and as a function of primary water addition during reactor coolant system boron concentration reduction.

Initially, the shutdown and control banks of control rods were withdrawn incrementally in the normal withdrawal sequence, leaving the last withdrawn control bank inserted far enough in the core to provide effective control when criticality was achieved.

The boron concentration in the reactor coolant system was then reduced by the addition of primary water. Criticality was achieved during boron dilution or by subsequent rod withdrawal following boron dilution. The rate of primary water addition, and hence the rate of approach to criticality, could have been reduced as the reactor approached criticality to ensure that effective control was maintained. Throughout this period, samples of the primary coolant were obtained and analyzed for boron concentration.

Written procedures specify the plant conditions, precautions, and specific instructions for the approach to criticality.

Successive stages of control rod assembly group withdrawal and of boron concentration reduction were monitored by observing changes in neutron count rate, as indicated by the permanent source range nuclear instrumentation, as functions of group position during rod motion, reactor coolant boron concentration, and primary water addition to the reactor coolant system during dilution. Throughout this period, samples of the primary coolant were obtained and analyzed for boron concentration.

Inverse count rate ratio monitoring was used as an indication of the proximity and rate of approach to criticality during control rod assembly group withdrawal and during reactor coolant boron dilution. The rate of approach was reduced as the reactor approached the time extrapolated for criticality to ensure that effective control was maintained at all times.

14.1.3.2.2 Low Power Testing[(a)]

A prescribed program of reactor physics measurements was undertaken to verify that the basic static and kinetic characteristics of the core were as expected and that the values of the kinetic coefficients assumed in the safeguards analysis were indeed conservative.

The measurements were made at low power and primarily at or near operating temperature and pressure. Measurements were made, including verification of calculated values of control rod assembly group reactivity worths, of isothermal temperature coefficient under various core conditions, differential boron concentration reactivity worth, and critical boron concentrations as functions of control rod assembly group configuration. In addition, measurements of the relative power distributions were made. Concurrent tests were conducted on the instrumentation, including the source and intermediate range nuclear channels.

Procedures were prepared to specify the sequence of tests and measurements to be conducted and the conditions under which each was to be performed to ensure both safety of operation and the relevancy and consistency of the results obtained. If any significant deviations from design predictions existed, unacceptable behavior had been revealed, or apparent anomalies developed, the testing could have been suspended and the situation reviewed by the PORC, with technical assistance as required to determine whether a question of safety was involved, prior to resumption of testing.

14.1.3.2.3 Power Level Escalation

When the plant operating characteristics were verified by low power testing, a program of power level escalation brought the unit to its full rated power level. Operational characteristics were closely examined at each stage and the conformance with the safeguards analysis verified before escalation to the next programmed level.

Measurements were taken to determine the relative power distribution in the core as a function of power level and control rod assembly group position.

Secondary system heat balances were performed to ensure that the indications of power level were consistent and to provide a basis for calibration of the power range nuclear channels. The ability of the reactor coolant system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified.

At prescribed power levels, the dynamic response characteristics of the reactor coolant and steam systems were evaluated. The responses of the systems were measured for design step load changes of 10 percent, rapid 50 percent load reductions, and plant trips.

Adequacy of radiation shielding was verified by gamma and neutron radiation surveys at selected points inside the containment and throughout the station site at various power levels. Periodic sampling was performed to verify the chemical and radiochemical analysis of the primary coolant.

The sequence of testing following core loading was used as a basis for planning and scheduling tests. The existing plant condition and status of plant systems and components were the primary factors in determining which tests and operations could be performed at a given time. The schedule was modified to meet the particular needs and conditions at the time, but in no event was a test or operation undertaken without satisfying the prerequisites for that test or operation.

14.1.3.2.4 Special Test Program

In response to the requirements of NUREG 0694, dated June 1980, entitled "TMI-Related Requirements for New Operating Licenses," Section I.G.1, Training During Low Power Testing, APC has reviewed the special low power testing requirements of NRC for FNP Unit 2. Following are the tests which have been considered in this review:

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a. The modified startup physics test program for Unit 2 is described in an APC letter to the NRC dated July 7, 1980.

Test No.	*Description*
1	*Natural circulation demonstration*
2	*Natural circulation with simulated loss of offsite power*
3	*Natural circulation with loss of pressurizer heaters*
4	*Effect of steam generator secondary side isolation on natural circulation*
5	*Natural circulation at reduced pressure*
6	*Cooldown capability of the chemical and volume control system*
7	*Simulated loss of all offsite and onsite ac power*
8	*Establishment of natural circulation from stagnant conditions*
9a	*Forced circulation cooldown*
9b	*Boron mixing and cooldown with natural circulation*

Alabama Power Company performed tests 1 through 7 and 9a prior to exceeding 5 percent of rated thermal power. Several of these tests could be combined in a manner similar to that performed at the North Anna facility. In lieu of performing test 9b, credit was taken for test results at other operating plants that were directly applicable to FNP. In lieu of performing test 8, training was provided for FNP operators via a simulator that has been updated as necessary using the Westinghouse and Tennessee Valley Authority test data from Sequoyah.

14.1.4 ADMINISTRATIVE PROCEDURES (SYSTEM OPERATION)

Whenever possible, test procedures incorporate the use of plant operating procedures to demonstrate the adequacy and feasibility of normal and emergency operating procedures. Test procedures incorporate only plant operating procedures that have been prepared and approved in accordance with subsection 13.4.2.

When a plant operating procedure is included in a test procedure, the plant operating procedure becomes a part of the test procedure and is conducted as part of the test procedure described in subsection 14.1.1.1.

Should modification of a plant operating procedure that is part of a test procedure be found to be required during the conduct of the test, the required modification will be accomplished as described in subsection 13.4.2. After the test is completed and accepted, the plant operating procedure will be changed, if required, for plant operation in accordance with subsection 13.4.2.]

[HISTORICAL]
[TABLE 14.1-1 (SHEET 1 OF 9)

STARTUP TESTS

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
Low Power Tests			
Radiation surveys	Reactor is critical at various power levels from 0 to 100 percent, as specified by detailed procedures. Personnel monitoring and survey instruments test completed.	1. Measure radiation dose levels at selected points throughout the plant to verify shielding effectiveness. 2. At specified reactor power levels, detailed radiation surveys are conducted at selected points throughout the plant.	Measured radiation levels are within the within the limits of 10 CFR 20 for the zone designation of each area surveyed.
Calibration of nuclear instruments with power and determination of overlap (at low power and during power ascension as applicable)	Reactor is critical at the power level specified in the detailed procedure. Necessary test equipment is installed for secondary heat balance measurements.	1. Obtain nuclear instrumentation system channel overlap data; calibrate the power range channels to reflect actual power levels; obtain temperature data for overtemperature and overpower ΔT trip setpoints. 2. At specified low power level and selected levels during escalation, the following are determined/performed: a. Power range detector currents vs power level. b. Secondary heat balance and adjustment of power range channel gain. c. Hot and cold leg RTD readings and ΔT amplifier output. d. Source, intermediate, and power range channel outputs to establish channel overlaps.	Power range channels display linear output over normal operating range. Power range channels accurately reflect heat balance data. ΔT setpoint adjustments entered in accordance with test procedure. Consistent overlap data obtained on power level changes through the source, intermediate, and power ranges

TABLE 14.1-1 (SHEET 2 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
Effluent radiation monitors	*Reactor has been at power for a time sufficient to produce representative effluents.* *Effluent monitors have been checked against known sources.*	*1. Verify the performance of the effluent monitors under actual plant discharge operations.* *2. Following standard discharge procedures, discharge commences and the response of effluent monitors are observed. Effluents are sampled and monitor performance verified by radio-chemical analysis. This is repeated at selected power levels.*	*Installed effluent monitors perform in accordance with design standards and properly indicate the radioactive content of the effluent.*
Physics measurements	*Reactor plant is in hot zero power condition.*	*1. Perform reactor physics measure-ment as outlined below to verify that characteristics of the core, coolant, and physics parameters are as expected and that co-efficients of reactivity are as assumed in the safety analysis.*	*Plant characteristics and coefficients of reactivity are consistent with the safety analysis.*
a. Moderator temperature reactivity coefficient		*2a. At normal no-load temperature and no nuclear heating, reactor coolant system cooldown and heatup are accomplished using the steam dump and reactor coolant pumps operation as required. An approximate 5°F change in temperature is initiated, and during these changes T_{avg} and reactivity are recorded on an X-Y recorder. The temperature co-efficient is determined from these data.*	*a. Isothermal temperature coefficient is negative under all conditions of critical operation.*

TABLE 14.1-1 (SHEET 3 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
b. Control rod reactivity worths		2b. Under zero power conditions at near operating temperature and pressure, the nuclear design predictions for rod cluster control assembly (RCCA) groups differential worths are validated. These validations are made from boron concentration sampling data, RCCA group positions, and recorder traces of reactivity. From these data, the integral RCCA group worths are determined. The minimum boron concentration for maintaining the reactor shutdown with the most reactive RCCA stuck in the full-out position is determined. The determination is made from analysis of boron concentration and RCCA worths.	b. Control rod reactivity worths meet FSAR design requirements for shutdown considerations; minimum shutdown boron concentration is within the limits of the safety analysis.
c. Boron reactivity worth measurement		2c. Differential boron worth measurements are made by monotonically increasing or decreasing reactor coolant boron concentration. Compensation for the reactivity effect of the boron concentration change is made by withdrawing or inserting respective control rods to maintain a moderator average temperature and power level constant and by observing the resultant accumulated change in core reactivity corresponding to these successive rod movements.	c. Measured boron worths are consistent with the trend of design values.
d. Determination of boron concentration of initial criticality and reactivity allocation		2d. All-rods-out boron concentration is determined as part of the approach to initial criticality, in that criticality is achieved by boron dilution with all but the controlling group of rods fully withdrawn. The amount of reactivity of the controlling group is then subsequently determined by withdrawal of the group, noting the amount of reactivity inserted, and converting this value to an equivalent amount of boron.	d. Critical boron concentration measurements are within the limits specified in the approved test procedure.

TABLE 14.1-1 (SHEET 4 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
e. Flux distribution measurements with normal rod patterns		2e. Incore movable detector system is used to map flux distribution for normal rod patterns	e. Analysis of flux distribution measurements yields hot channel factors less than or equal to design safety limits
f. Pseudo rod ejection test to verify safety analysis		2f. Incore measurements are made under pseudo ejected rod conditions simulating the zero power accident to determine the hot channel factors and verify that they are within assumptions made in the accident analysis.	f. Analysis of flux distribution measurements yields hot channel factors less than or equal to design safety limits.
Chemical tests to demonstrate ability to control water quality	At hot zero power conditions and during power escalation.	1. At hot zero power conditions and during power escalation, perform sampling and analysis to verify that plant chemistry is within specifications.	Reactor plant chemistry is controlled within the limits of the plant chemistry specifications.
		2. Demonstrations of adjustment of plant chemistry are performed as required.	
Power Ascension Tests			
Power reactivity coefficient evaluation and power defects measurements (approximately 30, 50, 75, and 100 percent)	Reactor is critical at various power levels from0 to 100 percent as specified	1. Determine the differential power coefficient of reactivity and the integral power defect.	Differential power coefficient is equal to or more conservative than the power coefficient assumed in the safety analysis.
	Necessary test equipment is installed for secondary heat balance measurements.	2. During each power escalation, recorder traces are made of reactor power vs reactivity changes; at selected power levels, plant systems are stabilized and secondary heat balances obtained to determine core power accurately; power coefficient and power defect are calculated from data obtained over the range from hot zero power to full power.	Measured power defect is compatible with shutdown margin calculations.

TABLE 14.1-1 (SHEET 5 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
Manual and automatic plant load changes (approximately 35, 50, 75, and 100 percent)	Reactor is critical at various power levels from 0 to 100 percent as specified.	1. Verify plant response to load change conditions. 2. Plant response to the following load changes is demonstrated: a. Step load change of $\pm$ 10 percent from Approximately 35, 75, and 100 percent power. b. Load reductions of 50 percent. c. Plant trips from power levels as specified in the approved test procedure. During the performance of these tests, recordings are analyzed for control systems behavior and requirements for realignment. At approximately 15 to 30 percent power, the automatic control systems are checked by simulating controlling parameters with a test signal, observing controller response and programmed step changes in the control parameter, switching to automatic, and observing the ability of the parameter to achieve the net setting without appreciable over-shoot or oscillation. During the transient tests, these systems are operationally checked under actual design load changing conditions.	Acceptance criteria, such as the plant not tripping(where applicable), relief and safety valves not lifting, and steam dump operating correctly, are identified in the individual procedures. Basic acceptance criteria are the proper response of individual systems and integrated plant response to each load change operation as described in the various sections of the FSAR.
		1. Verify the core performance margins are within design predictions.	Core performance margins are within design predictions
Evaluations of core performance (30, 50,75, and 100 percent)	Reactor is critical at various power levels from 0 to 100 percent as specified.	2. At steady state power points, incore data are obtained and analysis performed to verify that the core performance margins are within design predictions for expected normal and abnormal rod configurations.	Nuclear and temperature instrumentation is responsive to reactor conditions, both changing and steady state.

TABLE 14.1-1 (SHEET 6 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
		The data/measurements to be taken include: a. Power range detector currents vs power level. b. Secondary heat balance and adjustment of power range channel gain. c. RTD values and ΔT amplifier outputs. d. Excore detector signal voltages vs currents e. Overlap data for power and intermediate ranges. f. Data for calibration of steam and feedwater flow instruments.	
Turbine trip	Reactor plant at power level as specified in the approved test procedure.	1. Demonstrate capability of the automatic control systems and secondary plant to sustain a plant trip from power and achieve stable shutdown conditions; determine overall response time of reactor coolant system hot leg RTDs. 2. At steady state power level as specified in the approved test procedure and with all control systems in automatic mode, the turbine is manually tripped. Plant parameters are recorded on high speed recorders.	Pressurizer and steam generator safety valves do not rise. Safety injection is not initiated. RTD response time no greater than design specifications. Neutron flux must drop to15 percent within specified time. All full length control rods must drop. Controlled temperature reduction to no load.

TABLE 14.1-1 (SHEET 7 OF 9)

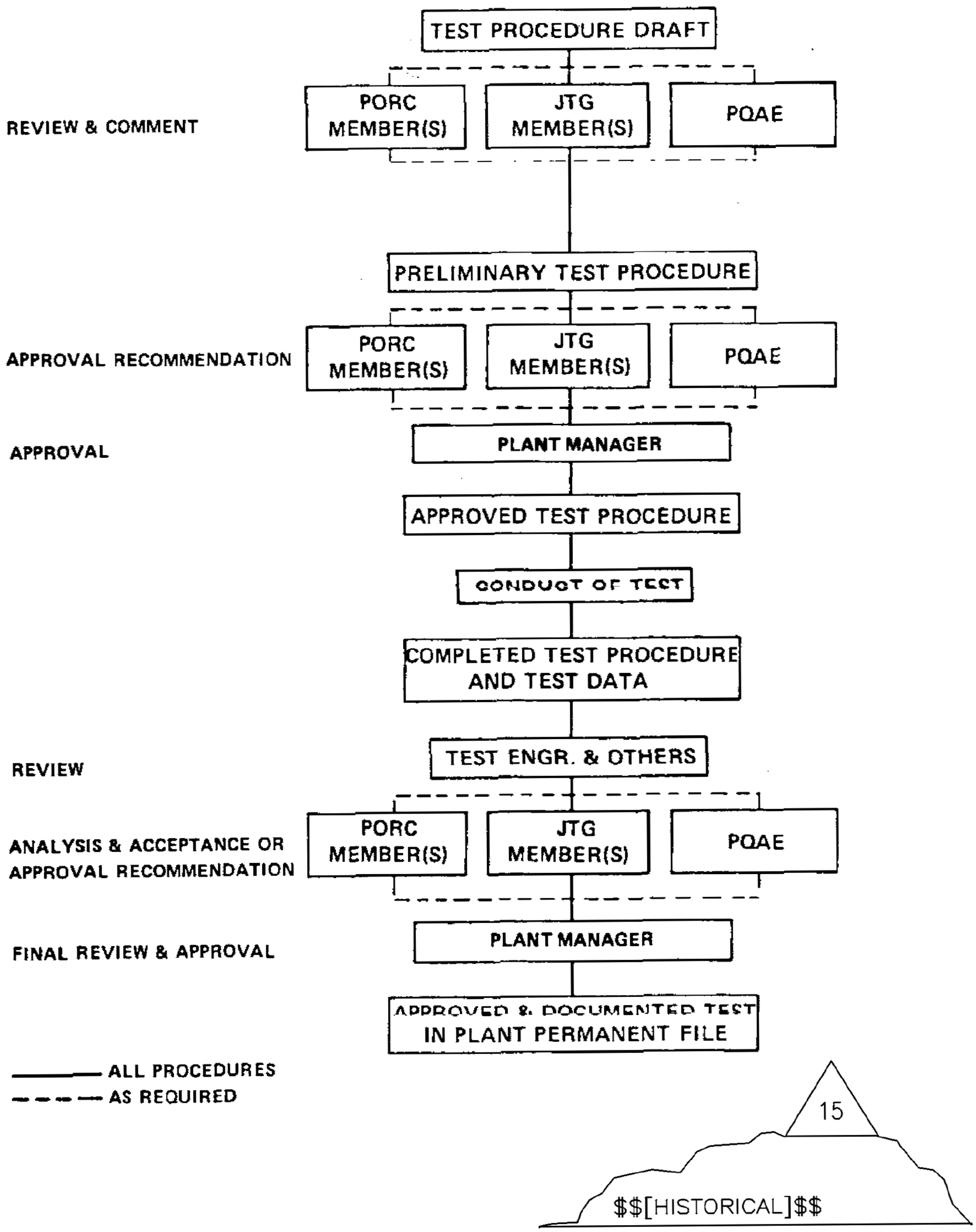
Test/Measurement	*Prerequisite*	*Test Objective (1)* *Test Summary (2)*	*Acceptance Criteria*
Incore/excore detector calibration	*Reactor plant critical at approximately 75 percent.*	*1. Establish relationships between incore and excore generated axial offsets and determine F (ΔI) setpoints.* *2. Data for power distribution measurements are obtained using incore movable detector system and thermocouples* *Additional data for generation of F (ΔI) setpoints are obtained.*	*Calibrated excore axial offset agrees with incore axial offset to within the values specified in the approved test procedure.*
Static RCCA drop and RCCA below-bank position measurements	*Reactor plant critical at approximately 50 percent.*	*1. Obtain worth of the most reactive below-bank RCCA; demonstrate excore and incore instrumentation response to a unit RCCA moving below bank; determine hot channel factors as a function of RCCA position.* *2. Unit RCCA worths are determined by RCCA movement in response to boron dilution.* *During RCCA insertion, the following data are recorded: excore detector currents, thermo-couple maps, and movable detector traces. This allows the computation of hot channel factors, core tilt, and excore sensitivity as a function of RCCA position.*	*Hot channel factors are within design safety limits when the unit RCCA is completely misaligned.* *The excore and/or incore instrumentation will detect a misaligned RCCA when the misalignment causes a significant power maldistribution.* *Misalignment within the limits of resolution of the rod position indicators will not cause a significant power maldistribution.*
Pseudo rod ejection and RCCA above-bank position measurement	*Reactor critical with plant at approximately 30 percent power.*	*1. Verify ejected rod worth and hot channel factors assumed in the accident analysis; demonstrate instrumentation response to an RCCA above-bank position and to an ejected rod.* *2. Unit RCCA worths are determined by RCCA movement in response to a continuous boration.*	*Flux tilt settings are made such that hot channel factors and power distributions assumed in the safety analysis will not be exceeded by a single RCCA out of bank.*

TABLE 14.1-1 (SHEET 8 OF 9)

Test/Measurement	Prerequisite	Test Objective (1) Test Summary (2)	Acceptance Criteria
		During RCCA withdrawal, the following data are recorded: excore detector currents, thermocouple maps, and movable detector traces. This allows the computation of hot channel factors, flux tilt, and excore sensitivity as a function of RCCA position.	
Loss of offsite power	Reactor plant at power level specified in the approved test procedure.	1. Verify that, upon a loss of offsite power, the plant can be maintained in a safe hot shutdown condition.	Turbine and reactor trips function as described in the FSAR.
		2. While operating at power, the 4160-V busses F, G, H, J, K, and L are isolated from the unit and auxiliary startup transformers. This test may be conducted one train at a time.	Emergency power systems function as described in the FSAR. Plant is maintained in a safe hot shutdown condition in accordance with plant emergency procedures.
Shutdown from outside control room	Reactor plant at power level specified in the approved test procedure.	1. Demonstrate the capability of shutting down the reactor plant and maintaining a hot shutdown condition from outside the control room.	Capability to shutdown the reactor plant from outside the control room is demonstrated from an initial condition of power operation.
		2. While operating at a power level greater than 10 percent MWe, the reactor is tripped and hot shutdown is maintained from the hot shutdown panel in accordance with approved procedures.	
Generator trip	Reactor plant at power level as specified in the approved test procedure.	1. Demonstrate capability of the primary and secondary plant to sustain a loss of external load and to bring the plant to stable conditions following the transient.	Safety injection does not occur.

TABLE 14.1-1 (SHEET 9 OF 9)

Test/Measurement	*Prerequisite*	*Test Objective (1)* *Test Summary (2)*	*Acceptance Criteria*
		2. At steady state power level as specified in the approved test procedure and with all control systems in automatic mode, the generator breaker is tripped. *Plant parameters are recorded on high speed recorders.]*	



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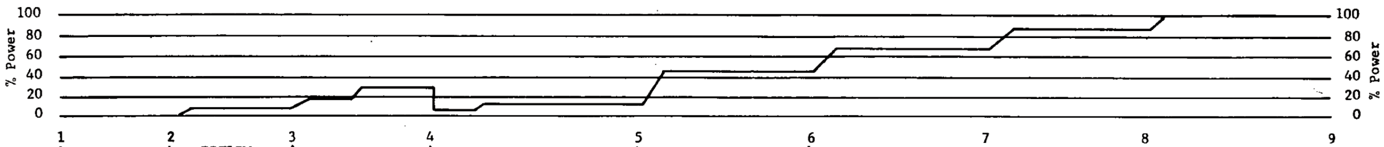
[TEST PROCEDURE DEVELOPMENT, TEST EXECUTION, DATA EVALUATION, AND DOCUMENTATION

FIGURE 14.1-1]



INITIAL CRITICALITY	PRELIM. ZERO POWER	ESCALATION TO 30%	ZERO POWER TESTS	ESCALATION TO 50%	ESCALATION TO 75%	ESCALATION TO 90%	ESCALATION TO 100%
Initial Criticality Chemical Test	Determination of Boron Concentration at Initial Criticality	Radiation Surveys NIS Power Range Calibration & Overlap Data Automatic Control Systems (Checkout STM. Generator, Reactor Control Turbine) Effluent & Effluent Monitoring System	Moderator Temperature Reactivity Coefficient Control Reactivity Rod Worth Boron Reactivity Worth Measurement Flux Distribution Measurements with Normal Rod Patterns Pseudo Rod Ejection Test	Chemical Analysis NIS Power Range Calibration Flux Distribution Measurements with Normal Rod Pattern Power Reactivity Coefficient Eval. & Power Defects Pseudo Rod Ejection Test Evaluation of Core Performance Radiation Surveys & Shielding Effectiveness Effluent & Effluent Monitoring System Plant Response to ± 10% Loadswing	Radiation Surveys NIS Power Range Calibration Flux Distribution Measurements with Normal Rod Pattern Power Reactivity Coefficient Eval. & Power Defects Plant Response to ± 10% Loadswing Evaluation of Core Performance Effluent & Effluent Monitoring System Part Length Rod Insertion/Withdrawal to Control Xenon Transient Chemical Analysis	NIS Power Range Calibration Power Reactivity Coefficient Eval. & Power Defects	Radiation Survey NIS Power Range Calibration Flux Distribution Measurements With Normal Rod Pattern Rod Reactivity Coefficient Eval. & Power Defects ± 10% Load Swing 50% Load Reduction Evaluation of Core Performance Effluent & Effluent Monitoring System Chemical Analysis

Loss of offsite power, shutdown from outside the control room, plant response to 50 percent load reduction, and turbine trip will be conducted at power levels as specified in the approved test procedures.

$$[HISTORICAL]$$

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NUCLEAR PLANT
UNIT 1 AND UNIT 2

[STARTUP TEST SEQUENCE

FIGURE 14.1-2]

[HISTORICAL] [14.2 AUGMENTATION OF APPLICANT'S STAFF FOR INITIAL TESTS AND OPERATION

During the period of initial testing and operation of Farley Nuclear Plant Units 1 and 2, Alabama Power Company's plan to use a separate startup staff, under the direction of the plant manager, was unique in that the plant operating staff served as an augmenting organization rather than being the primary organization. In addition to the plant operating staff, the startup staff was augmented by technical specialists furnished by Westinghouse Electric Corporation, Bechtel Power Corporation, and other contractors and vendors as required. Also, technical assistance was available from Southern Company Services, Inc., Alabama Power Company's Production and Engineering Departments, and competent technical personnel from other company facilities as needed.]

15.0 ACCIDENT ANALYSES

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15.0 - ACCIDENT ANALYSES

The accident analyses for the Farley plant have been done using the American Nuclear Society (ANS) classification of plant conditions which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

A. Condition I - Normal operation and operational transients.

B. Condition II - Faults of moderate frequency.

C. Condition III - Infrequent faults.

D. Condition IV - Limiting faults.

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public and that those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single-failure criterion in fulfilling this principle. Specific considerations are listed in the analysis of each accident.

In evaluating radiological consequences associated with initiation of a spectrum of accident conditions, numerous assumptions must be postulated. In many instances, these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are presently not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

This chapter addresses itself to the accident conditions listed on pages 15T-1, 15T-2, and 15T-3 of the NRC Guide, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 1), as they apply to the Farley plant.

The events listed in table 15-1 of the NRC Guide and the FSAR sections which address these events are cross-referenced as follows:

Item 1 - See subsection 15.2.1.

Item 2 - See subsection 15.2.2.

Item 3 - See subsections 15.2.3 and 15.3.6.

Item 4 - See subsection 15.2.4.

Item 5 - See subsections 15.2.5, 15.3.4, and 15.4.4.

Item 6 - See subsection 15.2.6.

Item 7 - See subsection 15.2.7.

Item 8 - See subsection 15.2.8.

Item 9 - See subsection 15.2.9.

Item 10 - There are no pressure regulators or regulating instruments in the Farley design whose failure could cause heat removal greater than heat generation.

Item 11 - The reactor coolant flow controller is not a feature of the Farley design. Treatment of the performance of the reactivity controller in a number of accident conditions is offered in this chapter.

- Item 12 - The analysis of specific effects of internal and external events, such as major and minor fires, floods, storms, or earthquake, are discussed in appropriate sections of chapters 2, 3, and 8.

  An extensive fire protection system is provided for onsite fires as described in subsection 9.5.1. Floods and flood protection are discussed extensively in sections 2.4 and 3.4.

  Storms, probable frequency of occurrence, wind and tornado loadings, and missile protection (against tornado-generated missiles) are discussed in sections 2.3, 3.3, and 3.5.

  Earthquake analysis is discussed in subsection 3.2.1 and in section 3.7.

  The reactor coolant system (RCS) components whose failure could cause a Condition III or Condition IV loss-of-coolant accident (LOCA) are Safety Class I components designed to withstand consequences of the safe shutdown earthquake (SSE) occurrence. In the analysis of the Condition IV maximum credible accident, a rupture of the largest pipe in the RCS is assumed to occur in conjunction with an earthquake occurrence which may result in the loss of offsite power.

Item 13 - See subsections 15.2.12, 15.3.1, and 15.4.1.

Item 14 - See subsections 15.2.13, 15.3.2, and 15.4.2.

Item 15 - See subsection 15.3.3.

Item 16 - See subsection 15.3.5.

Item 17 - Applicable to boiling water reactors (BWRs) only.

Item 18 - See subsection 15.4.3.

Item 19 - Applicable to BWRs only.

Item 20 - See subsection 15.4.6.

Item 21 - Applicable to BWRs only.

Item 22 - No instrument lines from the RCS boundary in the Westinghouse pressurized-water reactor (PWR) design penetrate the containment.[a]

Item 23 - See subsection 15.4.5.

Item 24 - Small spills or leaks of radioactive material outside the containment are events which release relatively small amounts of radioactive material into the environment. Accidents for which dose analyses are presented in this report are those which release significant amounts of radioactive material and which therefore provide an acceptable basis for demonstrating the adequacy of the Farley design to prevent undue risk to the health and safety of the public.

Item 25 - See subsections 15.4.1 and 15.4.6. Dose analyses assuming steam generator leakage are performed for all accidents which cause fuel damage.

Item 26 - See subsection 15.4.1. Section 7.4 contains an analysis showing that the plant can be brought to hot shutdown and maintained in that condition from outside the control room.

Item 27 - The residual heat removal (RHR) is protected from inadvertent overpressurization by ASME code relief valves. Two main control board annunciator windows are installed to alert the operators when the RHR suction/isolation valve(s) is not fully closed and the RCS pressure exceeds the alarm setpoint. Power is removed from RHR suction/isolation valves when in Modes 1, 2, and 3.

Leak testing is performed on the isolation valves as described in the technical specifications.

The operability of the RHR isolation valves and associated interlocks is assured by strict administrative controls.

Item 28 - Loss of condenser vacuum is considered in the analyses of subsection 15.2.7, Loss of External Electrical Load and/or Turbine Trip.

Item 29 - Same as item 28 above.

a. For the definition of the RCS boundary, refer to ANS 18.2 Section 5, Nuclear Safety Criteria for the Design of Stationary BWR Plants.

Item 30 - The service water system is designed to preclude complete loss of service water as discussed in subsection 9.2.1.

Item 31 - Loss of one (redundant) dc system is considered in subsection 8.3.2.

Item 32 - See subsection 15.2.14.

Item 33 - The effects of turbine trip on the RCS are presented in subsection 15.2.7. Equipment described in subsection 8.2.1.2 will handle the consequences of a turbine trip with failure of the generator breaker to open.

Item 34 - Loss of instrument air is considered in chapter 9.

Item 35 - Malfunction of the turbine gland sealing system is only of significance in BWRs.

Accident analyses presented in this chapter were originally applicable to the first fuel cycle. These analyses have been updated to remain bounding for current cycles and are typical of expected values for cycles through the equilibrium cycle. They include the maximum expected core average burnup for an equilibrium cycle based on Westinghouse design methods and reload fuel.

The operator action times assumed in this chapter include conservative actions to provide an adequate safety margin for the purpose of nuclear safety system design and nuclear safety analysis of the design basis events. However, they are not intended to serve as a basis for actual operator action times in procedures or training. The assumed time periods are considered in the basis of plant design to permit credit for operator actions. The Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG's) provide a basis for operator actions in response to design basis accidents.

15.1 CONDITION I - NORMAL OPERATION

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with a margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation.

The NRC acceptance criteria for ensuring that Condition I occurrences will not result in fuel rod failures is that the minimum departure from nucleate boiling ratio (DNBR $\geq$ 1.23-1.25 as described in section 4.4.1.1) satisfies the 95/95 criterion. That is, there is at least 95-percent probability at a 95-percent confidence level (95/95 probability/confidence) that DNB will not occur on the limiting fuel rod. In addition, there is also a 95/95 probability/confidence that the peak kW/ft fuel rods will not exceed the melting temperature of uranium dioxide, taken as

4900°F (unirradiated) and 4800°F at end of life. A bounding value of 4700°F is used in the safety analyses as described in section 4.4. The NRC acceptance criterion used to ensure that the reactor coolant pressure boundary integrity is maintained is that the reactor coolant pressure remains below the ASME Section III Code pressure limit (2750 psia).

A typical list of Condition I events is listed below:

A. Steady-state and shutdown operations

   1. Power operation (≈ 15 to 100 percent of full power)

   2. Startup or standby (critical, 0 to 15 percent of full power)

   3. Hot shutdown (subcritical, RHR system isolated)

   4. Cold shutdown (subcritical, RHR system in operation)

   5. Refueling

B. Operation with permissible deviations

   Various deviations which may occur during continued operation as permitted by the plant technical specifications must be considered in conjunction with other operational modes. These include:

   1. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service)

   2. Leakage from fuel with cladding defects

   3. Activity in the reactor coolant

      a. Fission products
      b. Corrosion products
      c. Tritium

   4. Operation with steam generator leaks up to the maximum allowed by technical specifications

C. Operational transients

   1. Plant heatup and cooldown (up to 100°F/h for the reactor coolant system; 200°F/h for the pressurizer)

   2. Step load changes (up to + 10 percent)

   3. Ramp load changes (up to 5 percent/min)

4. Load rejection up to and including design load rejection transient (~ 50-percent steam dump capability)

15.1.1 OPTIMIZATION OF CONTROL SYSTEMS

15.1.1.1 Setpoint Study

A setpoint study has been performed in order to simulate performance of the reactor control and protection systems. Emphasis was placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints was determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance were made and verified. A consistent set of control system parameters for power levels between 15 and 100 percent was derived satisfying plant operational requirements throughout the cycle life. The study comprised an analysis of the following control systems: rod cluster control (RCC), assembly control, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.1.1.2 End of Life Coastdown

Coastdowns at the end of an operating cycle may be performed by a power reduction on the normal temperature program (power coastdown), or by a combination of RCS temperature reduction (temperature coastdown) followed by a power coastdown. In the latter case, RCS temperature and power may initially be reduced by maintaining a constant turbine control valve position and allowing the temperature feedback of the reactor core to control the rate of temperature and power reduction. If the valves are not at the valves wide open position when depletion of reactivity is reached at the end of an operating cycle, the valves can be gradually opened as temperature and power begin to decrease.

In order to perform a power coastdown on the normal temperature program, no specific adjustments to the control or protection system settings are required.

For the combination of a temperature coastdown followed by a power coastdown, the steam dump load rejection controller must be reset with trip open bistable and gain settings corresponding to a full-load reference temperature. This ensures that the steam dumps provide adqequate heat removal for load rejections as described in sections 7.7 and 10.4, and that the requirements of no challenges to the pressurizer PORVs are met for a turbine trip without a reactor trip below the P-9 setpoint, as described in paragraph 7.2.1.1.1.F. This method of steam dump control may be used for any combination of temperature coastdown followed by a power coastdown within the analyzed range of temperature programs. No changes to the plant trip controller settings are required. To improve the reactor control system response to transients, and to provide the operators with a target temperature for manual control and trip recovery, the programmed reference temperature should be reset periodically during the temperature coastdown. Once a final temperature program is reached, no further changes are required during the subsequent power coastdown. The overtemperature (OTDT) and overpower (OPDT) setpoint reference temperatures may remain at their corresponding pre-coastdown settings for the duration of the coastdown.

15.1.2 INITIAL POWER CONDITIONS ASSUMED IN ACCIDENT ANALYSES

Table 15.1-1 lists the principal power rating values assumed in analyses performed in this chapter. The rating values listed in table 15.1-1 are based on the nuclear steam supply system (NSSS) thermal power output which includes the thermal power generated by the reactor coolant pumps (RCPs) and other sources.

The thermal power attributed to the RCPs and other sources is the total RCS heat addition less the heat loss from the RCS.

For most accidents which are DNB limited, nominal values of the initial conditions are assumed. The uncertainty allowances on power, temperature, pressure, and RCS flow are included on a statistical basis and are included in the limit DNBR value, as described in reference 18. This procedure is known as the Revised Thermal Design Procedure (RTDP). For accidents analyses which are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum steady-state errors to rated values (this procedure is commonly known as Standard Thermal Design Procedure or STDP).

The following steady-state errors are considered in the analyses:

A. Core power - ± 2-percent allowance for calorimetric error (note that this error is conservatively applied in the positive direction in non-LOCA accident analyses).

B. Average RCS temperature - ± 6°F allowance dead band and system measurement error, including -1°F bias due to cold leg streaming.

C. Pressurizer pressure - ± 50-psi allowance for steady-state fluctuations and measurement errors.

Accidents employing RTDP assume a minimum measured flow (MMF); accidents employing STDP assume a thermal design flow (TDF). In addition to being the flow used in the DNB analysis for RTDP methodology, the MMF is specified in the technical specifications as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup. The RTDP DNB transients are analyzed using a MMF with a 2.1% uncertainty based on a precision heat balance measurement. The MMF with a 2.4% uncertainty is based on RCS cold leg elbow tap ΔP measurements. The TDF equals the MMF minus the plant flow measurement uncertainty.

Table 15.1-2A summarizes the initial conditions and computer codes used in the accident analyses. The values of other pertinent plant parameters used in the accident analyses are given in table 15.1-2B.

The outer surface of the fuel rod at the hotspot operates at a temperature of approximately 660°F for steady-state operation at rated power throughout core life due to the onset of nucleate boiling. Initially [beginning of life (BOL)], this temperature is that of the cladding metal outer surface. During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the cladding surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise.

Since the thermal hydraulic design basis limits departure from nucleate boiling (DNB), adequate heat transfer is provided between the fuel cladding and the reactor coolant so that the core thermal output is not limited by considerations of the cladding temperature. Figure 4.4-4 shows the axial variation of average cladding temperature for a typical rod (17-x-17 fuel assembly) both at beginning and end of life. End of life (EOL) is after three typical cycles of operation (approximately 60,000 MWd/MTU) for the most highly exposed assemblies in the core. These temperatures are calculated using the Westinghouse fuel rod model[(1)] which has been reviewed and approved by the NRC.

15.1.2.1 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods and by operation instructions. The power distribution may be characterized by the radial peaking factor $F_{\Delta H}$ and the total peaking factor F_Q. The peaking factor limits are given in the Core Operating Limits Report.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in figure 15.1-1A. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with (or greater than) the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is the 1.55 chopped cosine as discussed in paragraph 4.4.3.2.2.

For transients which may be overpower limited, the total peaking factor F_Q is of importance. The value of FQ may increase with decreasing power level such that full-power, hotspot heat flux is not exceeded, i.e., $F_Q \times$ power = design hotspot heat flux. All transients that may be overpower limited are assumed to begin with a value of F_Q consistent with (or greater than) the initial power level as defined in the Technical Specifications.

The value of peak kW/ft can be directly related to fuel temperature as illustrated in figures 4.4-1 and 4.4-2. For transients which are slow with respect to the fuel rod thermal time constant, the fuel temperatures are illustrated in figures 4.4-1 and 4.4-2. For transients which are fast with respect to the fuel rod thermal time constant, e.g., rod ejection, a detailed heat transfer calculation is made.

15.1.3 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in table 15.1-3.

A reactor trip signal acts to open two trip breakers connected in series which feed power to the control rod drive mechanisms (CRDMs). The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies (RCCAs) which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from when the monitored parameter reaches the trip setpoint until the rods are free and begin to fall.

Table 15.1-3 refers to the overtemperature and overpower ΔT trip shown in figure 15.1-1A.

These trip setpoints bound mixed LOPAR/VANTAGE 5 cores and a full core of VANTAGE 5 fuel within the requirements of the technical specifications. The associated OTΔT f(ΔI) penalty is shown in figure 15.1-1B.

For all the reactor trips, the difference between the trip setpoints assumed in the analysis and the nominal trip setpoints account for instrumentation channel error and setpoint error. The plant technical specifications specify the nominal trip setpoints. Response time limits for the reactor trip systems are maintained in table 7.2-5. The calibration of protection system channels and the periodic determination of instrument response times are in accordance with the plant technical specifications.

15.1.4 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS

The VANTAGE 5 fuel design features, the modified safety analysis assumptions, and the application of new methodologies (i.e., RTDP, WRB-1, and WRB-2) as discussed in section 4.4 (with respect to the changes associated with the instrument uncertainties for the NSSS control parameters of power, pressure, temperature, and flow) are covered in reference 2 and reference 19.

Westinghouse Technical Bulletin, ESBU-TB-92-14-R1, “Decalibration Effects of Calorimetric Power Level Measurements on the NIS High Power Reactor Trip at Power Levels Less Than 70% RTP,” identified a potential non-conservative bias which could be introduced if NIS channel indicated power is adjusted in the decreasing power direction based on a part power calorimetric. To assure a reactor trip below the safety analysis limit, the Power Range Neutron Flux - High bistables are set $\leq$ 85% RTP: 1) whenever the NIS channel indicated power is adjusted in the decreasing power direction due to a part power calorimetric below 50% RTP; and 2) for a post refueling startup. Before the Power Range Neutron Flux - High bistables are reset $\leq$ 109% RTP, the NIS channel calibration must be confirmed based on a calorimetric performed $\geq$ 50% RTP (reference 19).

15.1.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTIC

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time from the start of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses, it is conservatively assumed that the insertion time to dashpot entry is 2.7 seconds. The RCCA position versus time assumed in accident analyses is shown in figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from a xenon oscillation or can be considered as representing a transient axial distribution which would exist after the rod cluster control assembly bank has already traveled some distance after trip. This curve has been conservatively selected to bound future reloads, which can include axial blankets of natural uranium.

There is inherent conservatism in the use of this curve in that it is based on a skewed distribution which would exist relatively infrequently. For cases other than those associated with xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in figure 15.1-4. The curve shown in this figure was obtained from figures 15.1-2 and 15.1-3. A total negative reactivity insertion following trip of 4.8-percent $\Delta k/k$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in table 4.3-3. Both the trip reactivity and reactivity insertion rate are verified to be conservative with respect to the core design as part of the reload design process (reference 3).

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (figure 15.1-4) is used in transient analyses. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of reactor trip (figure 15.1-2) is used as code input.

15.1.6 REACTIVITY COEFFICIENTS

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in chapter 4.

The use of a slightly positive moderator temperature coefficient was initially incorporated into the core design by Amendments 37 and 27 to the Operating Licenses for Units 1 and 2, respectively. Subsequently, the moderator temperature coefficient technical specification unit was increased by Amendments 92 and 85 for Units 1 and 2, respectively. These amendments were the result of the reanalysis of those transients which are sensitive to a positive moderator temperature coefficient. In general, these are transients which cause an increase in the reactor coolant temperature such as an uncontrolled RCCA withdrawal, partial loss of forced reactor coolant flow, loss of external electrical load and/or turbine trip, accidental depressurization of the reactor coolant system, complete loss of forced reactor coolant flow, single reactor coolant pump locked rotor, and RCCA ejection. In all cases, the results indicated that the safety criteria and the NRC acceptance criteria are met. That is, peak fuel and clad temperatures remained acceptable, the DNB design basis is met, and/or reactor coolant system pressure remained below 110 percent of design pressure.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values (see figure 15.1-5). Some analyses, such as loss of reactor coolant from cracks or ruptures in the RCS, do not depend on reactivity feedback effects.

The values used are given in table 15.1-2A. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. To facilitate comparison, individual sections in which justification can be found for the use of large or small reactivity coefficient values are referenced below:

	Condition II Events	Subsection
A.	Uncontrolled RCCA bank withdrawal from a subcritical condition	15.2.1
B.	Uncontrolled RCCA bank withdrawal at power	15.2.2
C.	RCCA misalignment	15.2.3
D.	Uncontrolled boron dilution	15.2.4
E.	Partial loss of forced reactor coolant flow	15.2.5
F	Startup of an inactive reactor coolant loop	15.2-6
G.	Loss of external electrical load and/or turbine trip	15.2.7
H.	Loss of all ac power to the station auxiliaries	15.2.9
I.	Excessive heat removal due to feedwater system malfunctions	15.2.10
J.	Excessive load increase incident	15.2.11
K.	Accidental depressurization of RCS	15.2.12
L.	Accidental depressurization of main steam system	15.2.13

M.	Inadvertent operation of emergency core cooling system (ECCS) during power operation	15.2.14

Condition III Events

A.	Complete loss of forced reactor coolant flow	15.3.4
B.	Single RCCA withdrawal at full power	15.3.6

Condition IV Events

A.	Rupture of a steam line	15.4.2.1
B.	Rupture of a feed line	15.4.2.2
C.	Single reactor coolant pump locked rotor	15.4.4.3
D.	Rupture of a control rod drive mechanism housing (RCCA ejection)	15.4.6.3

15.1.7 FISSION PRODUCT INVENTORIES

15.1.7.1 Activities in the Core

Fuel burnup and fission product values were modeled via the ORIGEN2 code[15,16]. ORIGEN2 is a versatile point-depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating nuclide compositions. This code takes into account the transmutation of all isotopes in the material. For the relatively high fluxes in the core region, burn-in and burn-out of isotopes can have an important effect, particularly when high burnup cases are being considered.

15.1.7.2 Core Inventory Release Fractions

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are in accordance with Table 2 of Regulatory Guide (RG) 1.183 Regulatory Position 3.2. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3 of RG 1.183 Regulatory Position 3.2. These gap fractions are provided in table 15.1.4. The release fractions are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. These fractions are applied to the equilibrium core inventory described in FSAR paragraph 15.1.7.1.

15.1.8 RESIDUAL DECAY HEAT

15.1.8.1 Decay Heat Model for Non-LOCA Analyses

For the non-LOCA analyses, conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS decay heat standard (reference 8) plus uncertainty was used for calculation of residual decay heat levels. Figure 15.1-6 presents the curve as a function of time after shutdown.

15.1.8.2 Distribution of Decay Heat Following Loss-of-Coolant Accident

During a LOCA, the power generation in the core decreases rapidly due to void formation or RCCA insertion, or both. A large fraction of the remaining heat generation comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. During steady-state operation, as high as 97.4 percent of the hot rod power is generated directly in the pellets and cladding. When the fission power is reduced due to void formation and/or RCCA insertion, more of the hot rod power is redistributed. In the small break LOCA analysis, this is accounted for by reducing the power generated directly in the hot rod power from 97.4 to 95%. In the large break Best Estimate LOCA analysis, a detailed model is used to calculate the energy redistribution as a function of time, as described in Section 8.0 of reference 17.

15.1.9 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident such as the BE WCOBRA/TRAC code as in the analysis of the RCS pipe rupture (section 15.4) and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in table 15.1-2.

15.1.9.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod (LOPAR or VANTAGE 5 –see figure 15.1-7) and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, density). The code uses a fuel model which exhibits the following features simultaneously:

A. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

B. Material properties which are functions of temperature and a sophisticated fuel to clad gap heat transfer calculation.

C. The necessary calculations to handle post-DNB transients (film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials).

The gap heat transfer coefficient is calculated according to an elastic pellet model (figure 15.1-7). The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings.

Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded clad, there is no fuel clad contact and the gap conductance is calculated on the basis of the thermal conductivity of the gas contained in the gap. If the pellet outside radius so calculated is larger than the clad inside radius (negative gap), the pellet and the clad are pictured as exerting a pressure upon each other sufficiently important to reduce the gap to zero by elastic deformation of both. This contact pressure determines the gap heat transfer coefficient.

The effects of IFBA are implicitly included in the fuel model by appropriately modifying the initial fuel temperatures.

FACTRAN is further discussed in reference 9.

15.1.9.2 LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates up to a 4-loop system by modeling the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and pressurizer. The pressurizer heaters' spray, relief, and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary sides of the steam generators utilize a homogeneous, saturated mixture for the thermal transients, and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on neutron flux, overpower and overtemperature reactor coolant delta T, high and low pressure, low flow, and high-pressurizer level. Control systems, including rod control, steam dump, feedwater control, and pressurizer pressure control are also simulated. The safety injection system (SIS), including the accumulators, are also modeled.

LOFTRAN is a versatile program suited to accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of the DNBR based on the input from the core limits illustrated in figure 15.1-1A. The core limits represent the combination of the safety analysis DNBR exit boiling and exit quality limits as calculated for a typical or thimble cell.

LOFTRAN is further discussed in reference 10.

15.1.9.3 CROSS-SECTION GENERATION COMPUTER CODE

The lattice codes which have been used for the generation of group constants needed in the spatial two-group diffusion codes are described in chapter 4.

15.1.9.4 SPATIAL TWO-GROUP DIFFUSION CALCULATION CODE

Spatial few-group diffusion calculations are described in chapter 4.

15.1.9.5 TWINKLE

The TWINKLE program is a multidimensional spatial neutron kinetics code patterned after steady-state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion, fuel-clad, coolant heat transfer model for calculating, by point, Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits provide channel power, axial offset, enthalpy, volumetric surge, point power, fuel temperatures, and so on.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in reference 13.

15.1.9.6 THINC

The THINC code is described in paragraph 4.4.3.4.

15.1.9.7 ANC

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

ANC is further discussed in reference 14.

15.1.9.8 RETRAN

In addition to LOFTRAN, the RETRAN program is used for studies of the transient response of a PWR system to specified perturbations in process parameters. RETRAN is a one-dimensional, best estimate, thermal hydraulic analysis computer code developed under the sponsorship of the Electric Power Research Institute to provide for the analysis of light water reactor systems. The EPRI RETRAN code was approved by the USNRC in references 21 and 22.

The RETRAN code is a variable nodalization code; therefore, the user builds the desired plant model by defining the control volumes and flow paths (i.e., junctions) with heat slabs (i.e., conductors) to account for heat transfer in both the primary and secondary elements.

The RETRAN code allows either point neutron kinetics or one-dimensional space time kinetics to be used for the neutronics. Various component models are available, including a two-region nonequilbrium pressurizer, centrifugal pumps, valves, and non-conduction heat exchanges. In addition, special purpose models include a subcooled void fit, bubble rise, trips, and a flexible control system which allows the user to implement a wide range of auxiliary calculations/systems.

The Westinghouse RETRAN model consists of a point kinetics core model, a multi-node vessel model, which provides flexibility to address a wide range of upper and lower plenum mixing characteristics and asymmetric flow transitions, explicit models of each reactor coolant loop, multi-node steam generator models, and detailed models of the protection and control systems. Details of the NRC-approved Westinghouse RETRAN model are documented in reference 23.

15.1.10 FISSION PRODUCT BARRIERS ASSUMED IN ACCIDENT ANALYSES

One objective of Title 10 of the Code of Federal Regulations is to establish requirements directed toward protecting the health and safety of the public from an uncontrolled release of radioactivity. The design of the Farley Nuclear Plant applies defense-in-depth by providing adequate physical barriers to maintain uncontrolled releases of radioactivity within the guidelines of 10 CFR 100 and NUREG-0800. Physical barriers to the uncontrolled release of fission products credited in each of the analyses of the events as described in the updated FSAR are listed in table 15.1-6.

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TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Item	Rating (MWt)
Core thermal power (MWt)	2775
Thermal power generated by the RCPs [(b)] (nominal)	10[(a)]
Engineered safety features (ESF) design rating	2775
Thermal power generated by the RCPs[(b)] (ESF, nominal)	10[(a)]
Thermal power generated by the RCPs[(b)] (ESF, maximum)	15[(a)]

a. Nominal pump heat is considered to be 10 MWt for the NSSS power of 2785 MWt. The non-LOCA analyses assume a conservative maximum of 15 MWt for those transients in which larger values of pump heat are conservative. For transients in which pump heat would provide a transient benefit, no (zero) pump heat is assumed.

b. Analytical representation of Total Net Heat Input into the RCS from all sources.

TABLE 15.1-2A (SHEET 1 OF 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: BOL Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: EOL Moderator Density (Δk/g/cm^3)	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
Condition II					
Uncontrolled RCCA bank withdrawal from a sub-critical condition	TWINKLE, FACTRAN, THINC	+ 7.0	--	Coefficient is consistent with a defect of -960 pcm	0 (subcritical)[e]
Uncontrolled RCCA bank withdrawal at power DNB transient	LOFTRAN	+ 7.0	0.50	Lower and upper (see figure 15.1-5)	279, 1674, and 2790[c,f]
Uncontrolled RCCA bank withdrawal at power pressure transient	LOFTRAN	+7.0	--	Lower (see figure 15.1-5)	Varies from 223 to 2841 [a,e]
RCCA misalignment	THINC, ANC, LOFTRAN	-	--	-	2775[b,f]
Uncontrolled boron dilution	NA	NA	NA	NA	N/A
Partial loss of forced reactor coolant flow	LOFTRAN, FACTRAN, THINC	+ 7.0	--	Upper (see figure 15.1-5)	2790[c,f]
Startup of an inactive RCP	NA	NA	NA	NA	NA
Loss of external electrical load and/or turbine trip	RETRAN	+ 7.0, ≤ 70% RTP Ramping to 0 at 100% RTP	--	Lower (see figure 15.1-5)	2785[e,f]
Loss of normal feedwater	RETRAN	+ 7.0, ≤ 70% RTP Ramping to 0 at 100% RTP	--	Upper (see figure 15.1-5)	2785[c,e]

TABLE 15.1-2A (SHEET 2 OF 4)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed BOL Moderator Temperature (pcm/°F)	 EOL Moderator Density (Δk/g/cm^3)	 Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
Loss of all ac power to the station auxiliaries	RETRAN	+ 7.0, $\leq$ 70% RTP Ramping to 0 at 100% RTP	--	Upper (see figure 15.1-5)	2785[c,e]
Excessive heat removal due to feedwater system malfunctions	LOFTRAN	-	0.50	Lower (see figure 15.1-5)	0 and 2785[a,f]
Excessive load increase	LOFTRAN	0.0[g]	0.50	Upper and lower (see figure 15.1-5)	2785[a,f]
Accidental depressurization of the RCS	LOFTRAN	+ 7.0	--	Lower (see figure 15.1-5)	2785[c,f]
Accidental depressurization of the main steam system	LOFTRAN	Function of moderator density (see subsection 15.2.13 and figure 15.2-40 Sh. 1)	--	See figure 15.2-40 Sh. 2	0 (subcritical)[e]
Inadvertent operation of the ECCS during power operation	LOFTRAN	+ 7.0	0.50	Lower (see figure 15.1-5)	2785[c,e,f]
Condition III					
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	NOTRUMP, LOCTA-IV	--	--	--	2775[b]
Inadvertant loading of a fuel assembly into an improper position	LEOPARD, TURTLE	--	--	--	2775

TABLE 15.1-2A (SHEET 3 OF 4)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: BOL Moderator Temperature (pcm/°F)	EOL Moderator Density ($\Delta k/g/cm^3$)	Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
Complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, THINC	+ 7 ≤ 70% RTP Ramping to 0 at 100% RTP	--	Upper (see figure 15.1-5)	2790[c,f]
Waste gas decay tank rupture	--	--	--	--	--
Single RCCA withdrawal at full power	ANC	--	--	--	2775[b]
Condition IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the RCS (LOCA)	WCOBRA/TRAC HOTSPOT, COCO	Function of moderator density (see subsection 15.4.1)	--	Function of fuel temperature (see subsection 15.4.1)	2775[b]
Major secondary system pipe rupture up to and including double- ended rupture of a steam pipe	RETRAN, THINC	Function of moderator density (see paragraph 15.4.2.1 and figure 15.2-40 Sheet 1)	--	See figure 15.2-40 Sh. 2	0(subcritical)[e]
Major rupture of a main feedwater pipe	RETRAN	--	0.50	Upper (see figure 15.1-3)	2790[c,e]
Steam generator tube rupture	--	--	--	--	2775[b]

TABLE 15.1-2A (SHEET 4 OF 4)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: BOL Moderator Temperature (pcm/°F)	EOL Moderator Density ($\Delta k/g/cm^3$)	Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
RCP shaft seizure (locked rotor) peak clad temperature transient	LOFTRAN, FACTRAN, THINC	+ 7 ≤ 70% RTP Ramping to 0 at 100% RTP	--	Upper (see figure 15.1-5)	2846[(d,e)]
RCP shaft seizure (locked rotor) pressure transient	LOFTRAN	+7 ≤ 70% RTP Ramping to 0 at 100% RTP	--	Upper (see figure 15.1-5)	2841[(h,e)]
Fuel handling accident	--	--	--	--	2831
Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN, THINC	Refer to Section 15.4.6.2.2.3	-	Coefficient is consistent with a defect of: - 954 pcm (BOL HZP) - 955 pcm (BOL HFP) - 909 pcm (EOL HZP) - 909 pcm (EOL HFP)	0 and 2775[(b,e)]

a. Nominal pump heat of 10 MWt is assumed.

b. No pump heat (core thermal power) assumed.

c. Maximum pump heat of 15 MWt is assumed.

d. Uprated NSSS power with maximum pump heat increased by 2 percent.

e. STDP with a TDF of 86,000 gal/min/loop assumed.

f. RTDP with a MMF of 87,800 gal/min/loop assumed.

g. More limiting than + 7.0 pcm/°F.

h. Uprated NSSS power with nominal pump heat increased by 2 percent.

TABLE 15.1-2B

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS USED IN THE ACCIDENT ANALYSES

Parameter	STDP Value	RTDP Value
NSSS thermal output (includes 10 MWt generated by RCPs, MWt)	2785[c]	2785
Steam generator tube plugging (%)	20	20
Vessel average temperature (°F)		
High T_{avg}	577.2[c]	577.2
Low T_{avg}	567.2[c]	567.2
Core inlet temperature (°F)		
At High T_{avg}	541.1[c]	541.8
At Low T_{avg}	530.6[c]	531.3
Pressurizer pressure (psia)	2250[c]	2250
Reactor coolant flow, loop (gpm)	86,000[a]	87,800[b][d]
Steam flow at 2775 MWt, total (lbm/hr)		
At High T_{avg}	12,220,000	12,220,000
At Low T_{avg}	12,200,000	12,200,000
Steam pressure at steam generator outlet (psia)		
At High T_{avg}	724	724
At Low T_{avg}	656	656
Maximum steam moisture content (%)	0.10	0.10
Feedwater temperature at steam generator inlet (°F)	443.4	443.4
Average core heat flux (Btu/h-ft^2)		
LOPAR fuel	189,818.7	189,818.7
VANTAGE 5 fuel	197,200.5	197,200.5

a. TDF assumed in the non-LOCA analyses.

b. MMF assumed in the non-LOCA analyses based on 86,000 gpm/loop TDF and a conservative 2.1% flow uncertainty associated with RCS flow measurement verification using a precision heat balance.

c. Does not include uncertainties. See the appropriate accident sections.

d. The technical specifications MMF criterion of 88,100 gpm/loop is based on 86,000 gpm/loop TDF and a conservative 2.4% flow uncertainty associated with RCS flow measurement verification using cold leg elbow taps.

TABLE 15.1-3

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed in Analyses	Time Delay (s)
Power range high neutron flux, high setting	118% of RTP	0.5
Power range high neutron flux, low setting	35% of RTP	0.5
Power range high positive neutron flux rate	9% of RTP 2 seconds lag time constant	0.65
OTΔT	Variable (See figure 15.1-1A & 1B)	[a]
OPΔT	Variable (See figure 15.1-1A)	[a]
High pressurizer pressure	2425 psig	1.0
Low pressurizer pressure	1831 psig	2.0
Low reactor coolant flow (from loop flow detectors)	85% loop flow	1.0
Low-low steam generator water level	0% of narrow range level span (Feedline rupture event)[d]	2.0
	16% of narrow range level span (Loss of offsite power event)[d]	2.0
	19% of narrow range level span (Loss of normal feedwater event)[d]	2.0
High-high steam generator water level trip of the closure of feedwater closure of feedwater system valves and turbine trip	100% of narrow range[d] level span	7.0[b] 2.5[c]
Reactor trip (following turbine trip)	NA	1.0

a. The response time test criteria provided in chapter 7 are based on the FSAR chapter 15 analyses which model the channel response time. The specific safety analyses channel time delay is a function of the transient and code model. The model includes:

i) a first order lag for the 5-s RTD time constant
ii) a first order lag for the 6-s filters on measured ΔT and T_{avg}
iii) dynamic $T_{avg}/\Delta T$ signal compensation as defined by the Technical Specifications
iv) a 2-s pure time delay. This 2-s pure delay accounts for the channel electronics delay and the trip logic circuit delay, plus the time for the reactor breakers to open and the time for the CRDM stationary grippers to disengage (gripper release time).

b. From time setpoint is reached to feedwater isolation.

c. From time setpoint is reached to turbine trip.

d. Narrow range level span is from 375 to 587 in. above the top of the tube sheet.

TABLE 15.1-4

CORE INVENTORY FRACTION RELEASED INTO CONTAINMENT

LOCA

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali Metals	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Non-LOCA

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

TABLE 15.1-5

DELETED

TABLE 15.1-6 (SHEET 1 OF 7)

FISSION PRODUCT BARRIERS

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Accidental Releases of Liquid Effluents in Ground and Surface Water (2.4.13.3)	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
RCS Pressure Control During Low-Temperature Operation (5.2.2.4)	Fuel Cladding	None credited	N/A
	RCS Pressure Boundary	RCS within ASME stress limits (Appendix G, heatup/cooldown)	FSAR 3.1.27, 3A-1.65 5.2.2, 5.2.4, 5.4.2, PTLR
	Containment	None credited	N/A
Uncontrolled RCCA Withdrawal from Subcritical Condition (15.2.1)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature – ≤ 4700°F	FSAR 15.2.1.3, 4.4.1.1 FSAR 15.2.1.2.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.1, Table 5.2-19
	Containment	None credited	N/A
Uncontrolled RCCA Bank Withdrawal At Power (15.2.2)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Linear heat rate ≤ 22.4 kW/ft Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.2.3, 4.4.1.1 FSAR 15.2.2.1C, 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.2, Table 5.2-19
	Containment	None credited	N/A
RCCA Misalignment (15.2.3)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Linear heat rate ≤ 22.4 kW/ft Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.3.3, 4.4.1.1 FSAR 15.2.3.2.2C, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 2 OF 7)

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Uncontrolled Boron Dilution (15.2.4)	Fuel Cladding, RCS Pressure Boundary	Bounded by 15.2.2	FSAR 15.2.4.2.5
	Containment	None credited	N/A
Partial Loss of Forced Reactor Coolant Flow (15.2.5)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.5.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A
Startup of Inactive RCS Loop (15.2.6)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.6.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A
Loss of External Electrical Load and/or Turbine Trip (15.2.7)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.7.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.7.3, Table 5.2-19
	Containment	None credited	N/A
Loss of Normal Feedwater (15.2.8)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.8.3, 4.4.1.1 FSAR 15.2.8.3, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.8.3, Table 5.2-19
		Pressurizer H_2O vol. < 1400 ft^3	FSAR 15.2.8.3, Table 5.5-9
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 3 OF 7)

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Loss of All ac Power to the Station Auxiliaries (15.2.9)	Fuel Cladding	Bounded by 15.3.4 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.9.3 FSAR 15.2.9.3, Table 4.4-1
		I_2 spike ≤ 60 μCi/gm	FSAR 15.2.9.4, Table 15.2-3
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.9.3, Table 5.2-19
		RCS-to-SG leak ≤ 1 gpm	FSAR 15.2.9.4, Table 15.2-3
	Containment	None credited	N/A
Excessive Heat Removal due to Feedwater System Malfunctions (15.2.10)	Fuel Cladding	DNBR bounded by 15.2.1 and 15.2.11	FSAR 15.2.10.3
		DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.10.3, 4.4.1.1 FSAR 15.2.10.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A
Excess Load Increase (15.2.11)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.11.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A
Accidental Depressurization of the RCS (15.2.12)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.12.2, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2, Table 5.2-19
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 4 OF 7)

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Accidental Depressurization of the Main Steam System (15.2.13)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.13.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.13.3, Table 5.2-19
	Containment	None credited	N/A
Inadvertent ECCS During Power Operation (15.2.14)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Fuel centerline temperature - ≤ 4700°F	FSAR 15.2.14.3, 4.4.1.1 FSAR 15.2, Table 4.4-1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia Przr H_2O volume < 1400 ft^3	FSAR 15.2, Table 5.2-19 FSAR 15.2.14.3, Table 5.5-9.
	Containment	None credited	N/A
Loss of Reactor Coolant from Small Ruptured Pipes (15.3.1)	Fuel Cladding	PCT < 2200°F	FSAR 15.3.1.3
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.3, Table 5.2-19
	Containment	Leak rate ≤ 0.15% / day	FSAR 15.3, Table 6.2-1
Minor Secondary System Pipe Breaks (15.3.2)	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.4.2.1	FSAR 15.3.2.3
Misloaded Fuel Assembly (15.3.3)	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
Complete Loss of RCS Flow (15.3.4)	Fuel Cladding	DNBR limit ≥ 1.23-1.25	FSAR 15.3.4.4, 4.4.1.1
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.3, Table 5.2-19
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 5 OF 7)

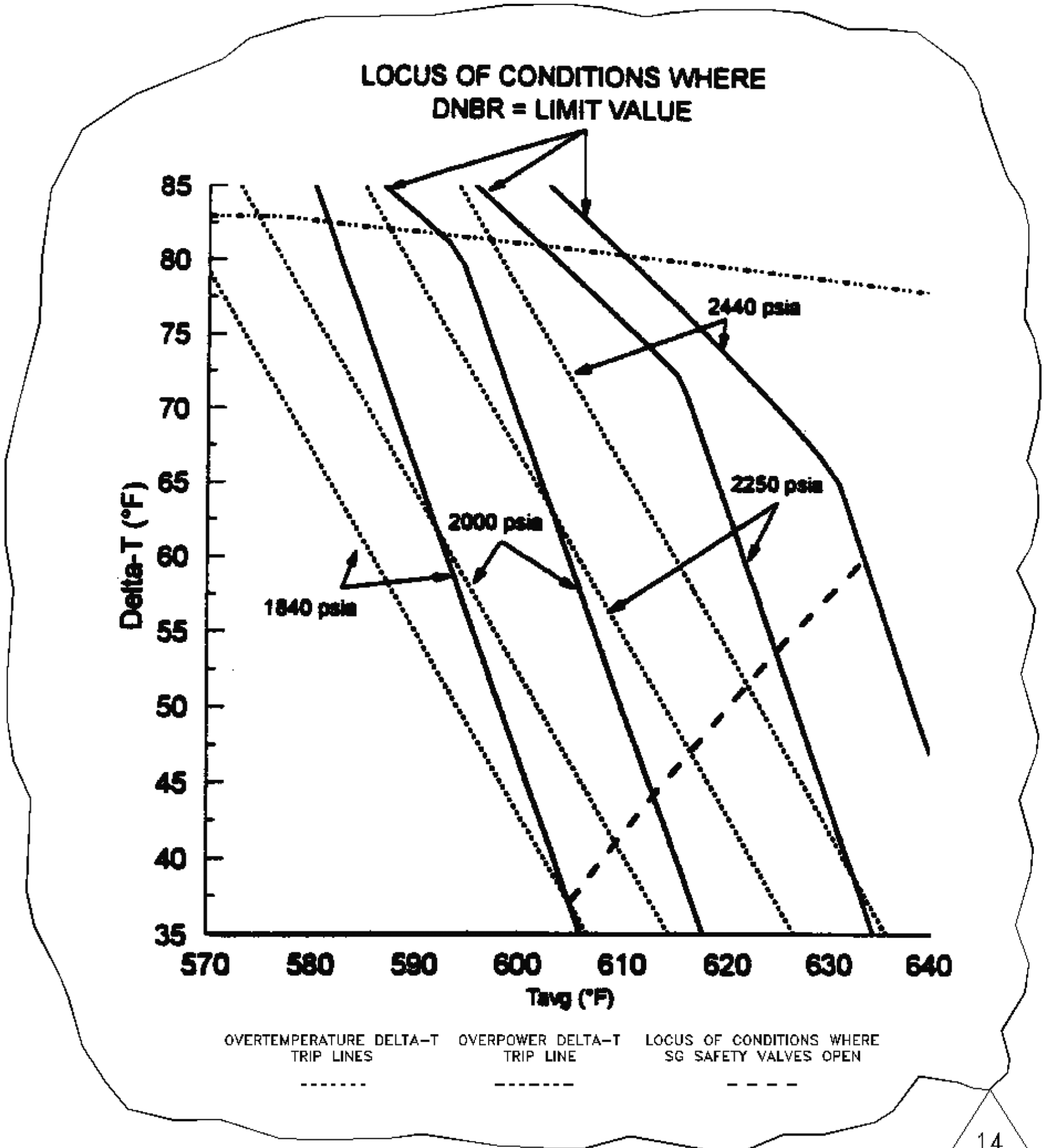
Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Gas Decay Tank Rupture (15.3.5)	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
Single RCCA Withdrawal at Power (15.3.6)	Fuel Cladding	≤ 5 % fuel rods exceed DNBR	FSAR 15.3.6.2
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.3, Table 5.2-19
	Containment	None credited	N/A
LOCA (15.4.1)	Fuel Cladding	PCT ≤ 2200°F Clad oxidation ≤ 17% locally	FSAR 15.4.1.5.4.1 FSAR 15.4.1.5.4.2
	RCS Pressure Boundary	ECCS leakage outside containment is ≤ 40,000 cm^3/h RCS meets limits in Tables 5.2-3 through 5.2-7	FSAR 15.4.1.10, Table 15.4.14 FSAR 15.4.1.1D and E5.2.1.10
	Containment	Stress ≤ limits in tables (including pressure ≤ 54 psig and temperature ≤ 280°F)	FSAR 3.8.1.5 FSAR Table 6.2-1
		Containment leakage ≤ 0.15% / day H_2 concentration < 4 $^v/_o$ Minipurge close ≤ 6 seconds	FSAR 15.4.1.7.3B, Table 15.4-14 FSAR 15.4.1.6.5 FSAR Table 15.4-15
Steam Line Break (15.4.2.1)	Fuel Cladding	DNBR limit ≥ 1.23-1.25 Linear heat rate ≤ 22.4 kW/ft	FSAR 15.4.2.1.1.2, 4.4.1.1 FSAR 15.4.2.1.2.2, Table 4.4-1
		I_2 spike ≤ 60 μCi/gm and 500 x normal appearance rate	FSAR 15.4.2.1.4E, Table 15.4-23
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.4.2.1.1.1, Table 5.2-19
		RCS-to-SG leak ≤ 1 gpm total (0.65 gpm to intact SG/0.35 gpm to faulted SG)	FSAR Table 15.4-23
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 6 OF 7)

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
Feedwater Line Break (15.4.2.2)	Fuel Cladding	Core stays water covered	FSAR 15.4.2.2.4
	RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.4.2.2.4, Table 5.2-19
See additional criteria for FSAR sections 15.2.8 and 15.4.2.1	Containment	None credited	N/A
SGTR (15.4.3)	Fuel Cladding	I_2 spike ≤ 30 μCi/gm and 335 x normal appearance rate	FSAR Table 15.4-24
		Core stays water covered	FSAR 15.4.3.2.2
	RCS Pressure Boundary	RCS stress ≤ limits in Tables 5.2-3 through 5.2-7	FSAR 15.4.3.3, 5.2.1.10
		RCS-to-SG leak ≤ 1 gpm total (0.65 gpm to intact SG/0.35 gpm to faulted SG)	FSAR 15.4.3.4, Table 15.4-24
	Containment	None credited	N/A
Locked Rotor (15.4.4)	Fuel Cladding	≤ 20% fuel rods exceed DNBR PCT ≤ 2700°F (ZIRLO) PCT ≤ 2375°F (Optimized ZIRLO)	FSAR 15.4.4.3C FSAR 15.4.4.3B
	RCS Pressure Boundary	RCS stress ≤ limits in Tables 5.2-3 through 5.2-7	FSAR 15.4.4.3A, 5.2.1.10
		RCS-to-SG leak ≤ 1 gpm	FSAR 15.4.4.4, Table 15.4-25A
	Containment	None credited	N/A
FHA (15.4.5)	Fuel Cladding	Pool water depth ≥ elevation 153'-3"	FSAR 9.1.3 and support documentation for Amendments 137/129
	RCS Pressure Boundary	None credited	N/A
	Containment	None credited	N/A

TABLE 15.1-6 (SHEET 7 OF 7)

Accident Description (FSAR Section)	Fission Product Barrier	Design Basis Limit	Reference(s)
RCCA Ejection (15.4.6)	Fuel Cladding	Avg. fuel enthalpy < 200 cal/g	FSAR 15.4.6.1.2A, 3A-1.183
		≤ 10% fuel rods exceed DNBR ≤ 10% fuel centerline temperature ≥ 4700°F	FSAR 15.4.6.4.3.A FSAR 15.4.6.4.3.C.2
	RCS Pressure Boundary	RCS stress ≤ limits in Tables 5.2-3 through 5.2-7	FSAR 15.4.6.1.2B
		RCS-to-SG leakage ≤ 1 gpm for all generators	FSAR 15.4.6.4.2A, Table 15.4-31
	Containment	Containment leakage ≤ 0.15% / day	FSAR 15.4.6.4.3G Table 15.4-31



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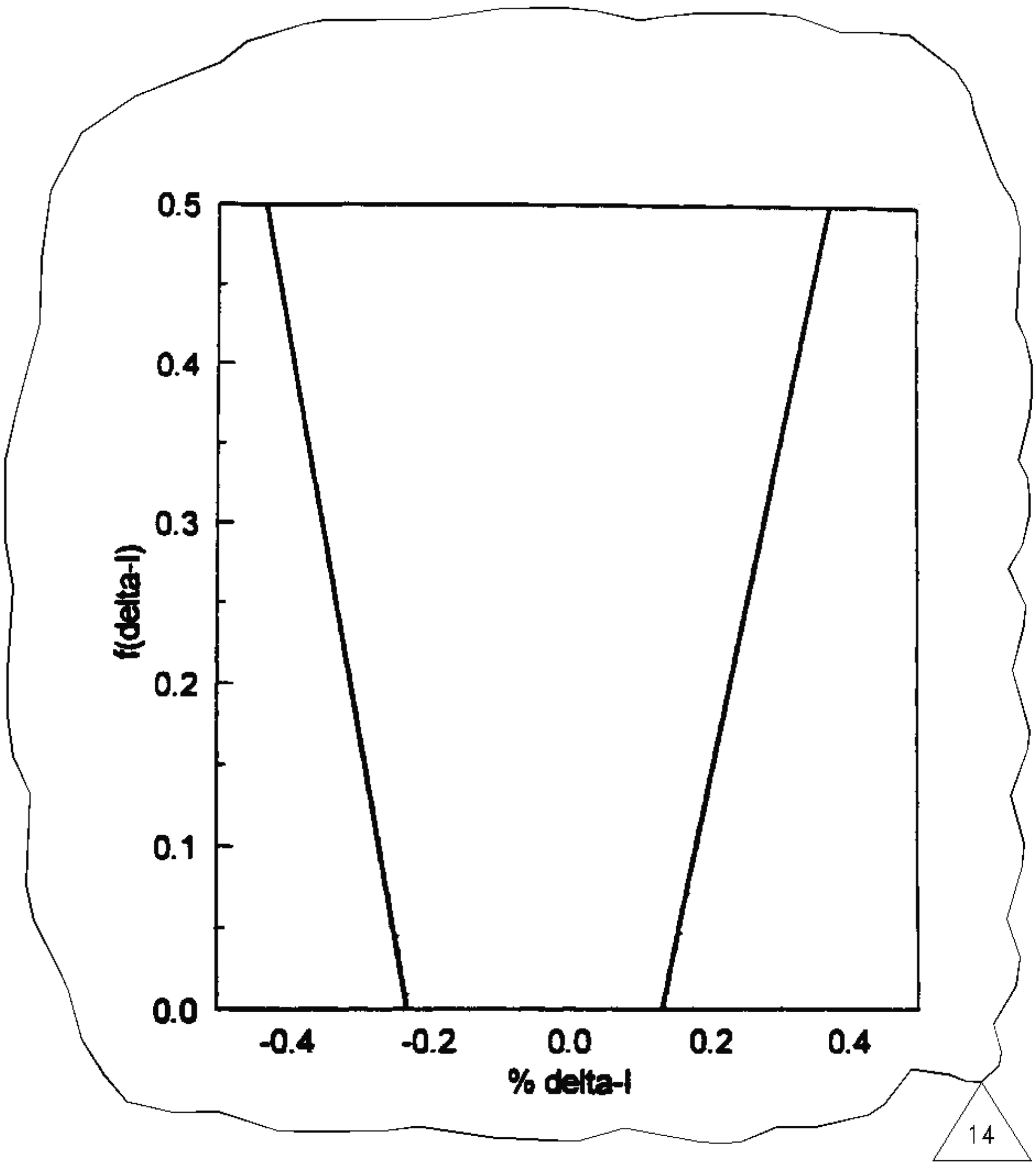
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

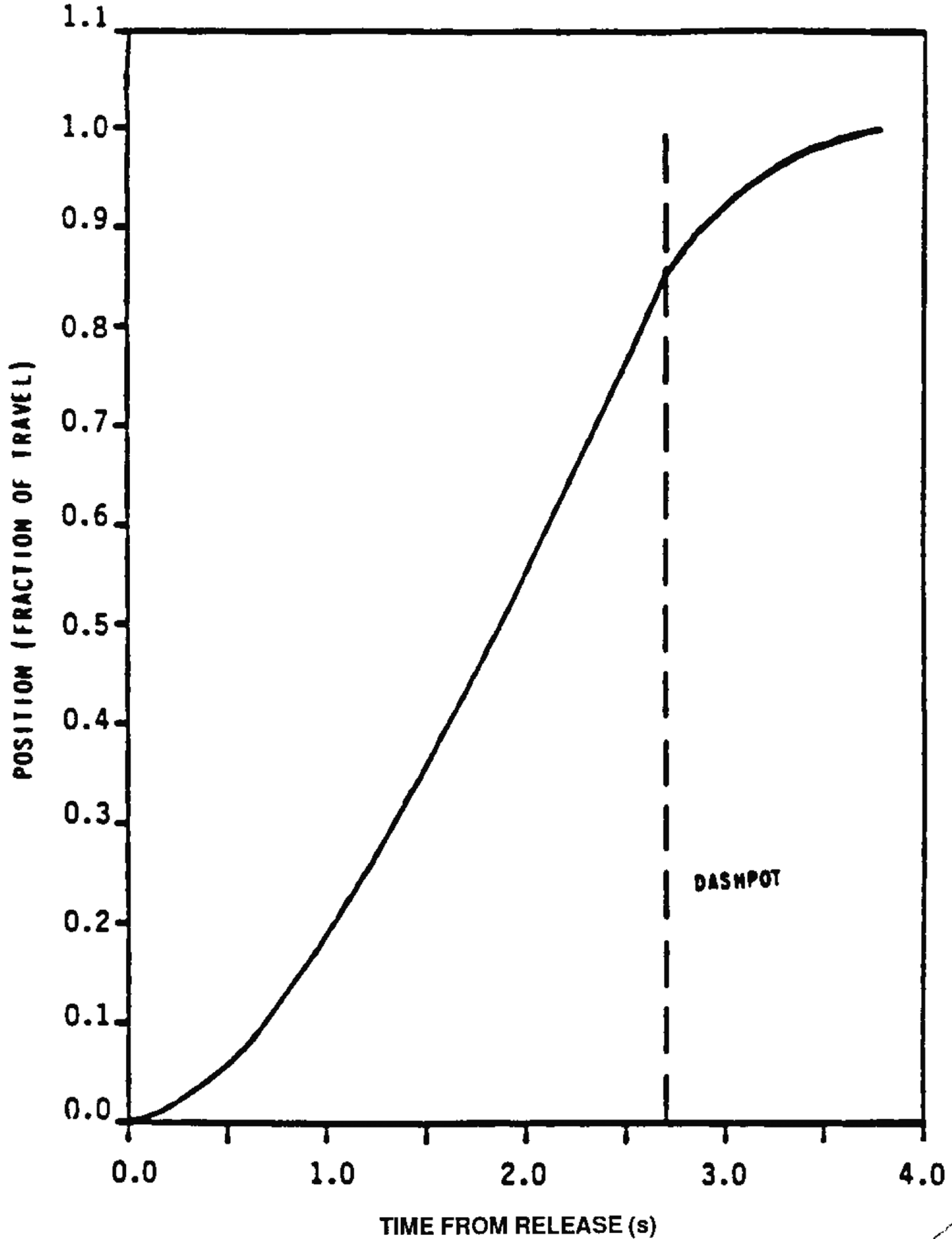
OVERPOWER AND OVERTEMPERATURE
DELTA-T PROTECTION

FIGURE 15.1-1A



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	OVERTEMPERATURE DELTA-T SETPOINT F-DELTA-I PENALTY
		FIGURE 15.1-1B



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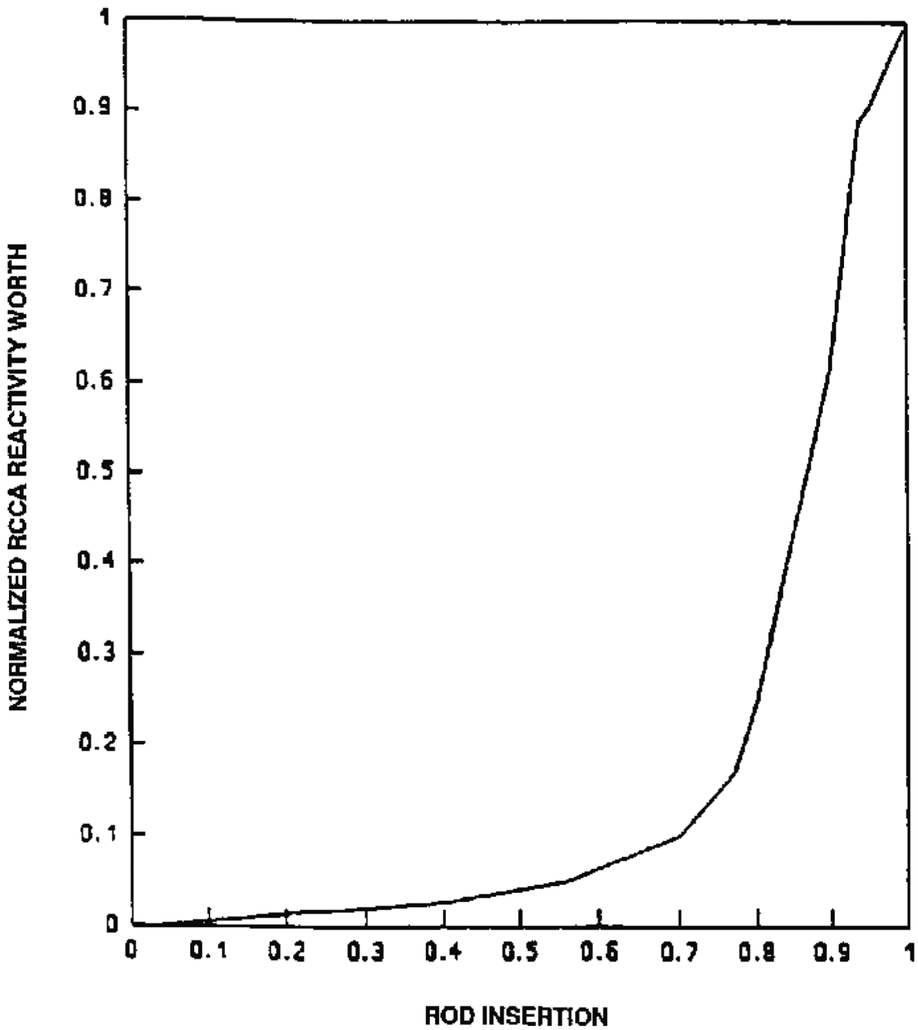
UNIT 1 AND UNIT 2

ROD POSITION VERSUS TIME
ON REACTOR TRIP

FIGURE 15.1-2



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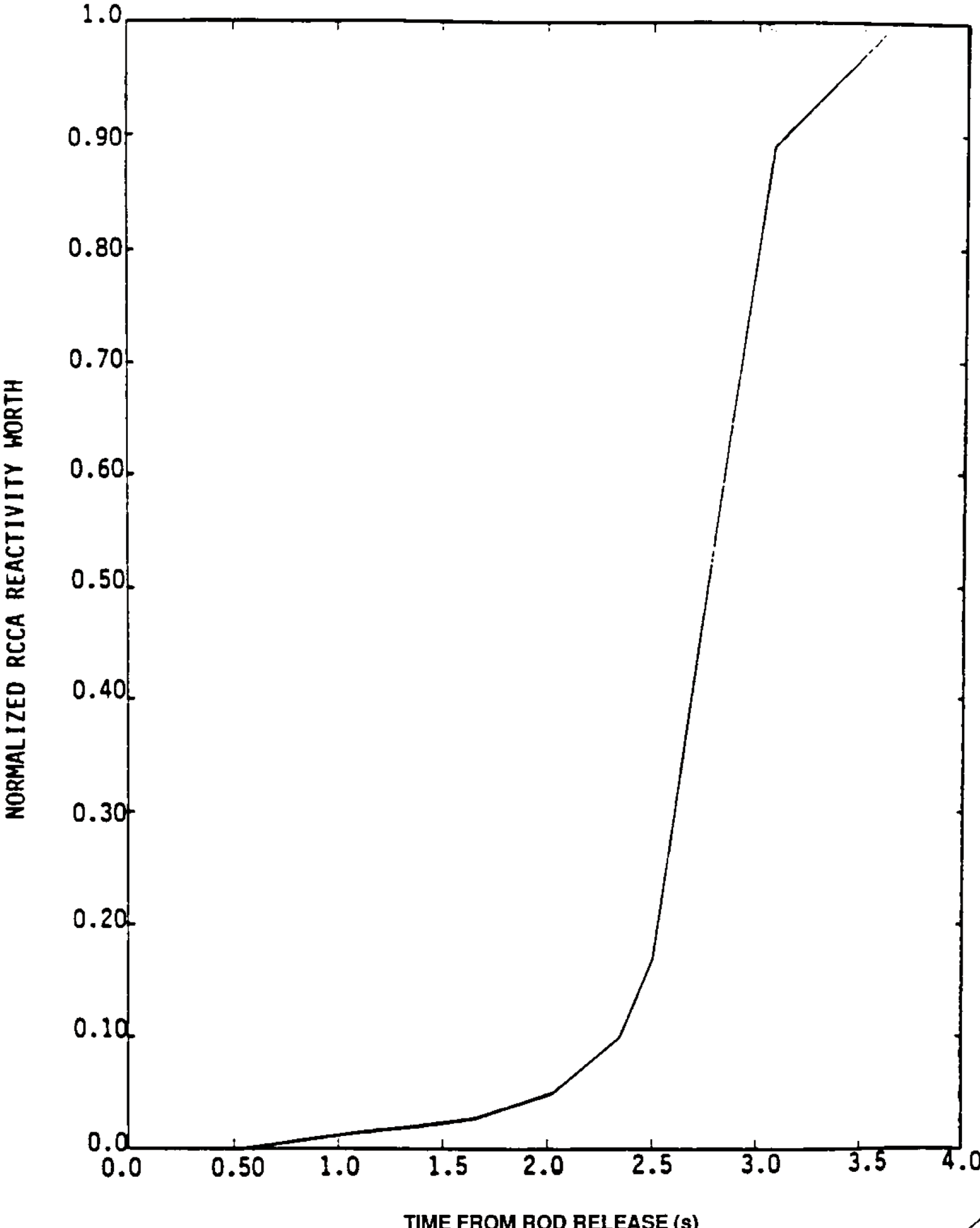
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NORMALIZED ROD WORTH
VERSUS FRACTION INSERTION

FIGURE 15.1-3



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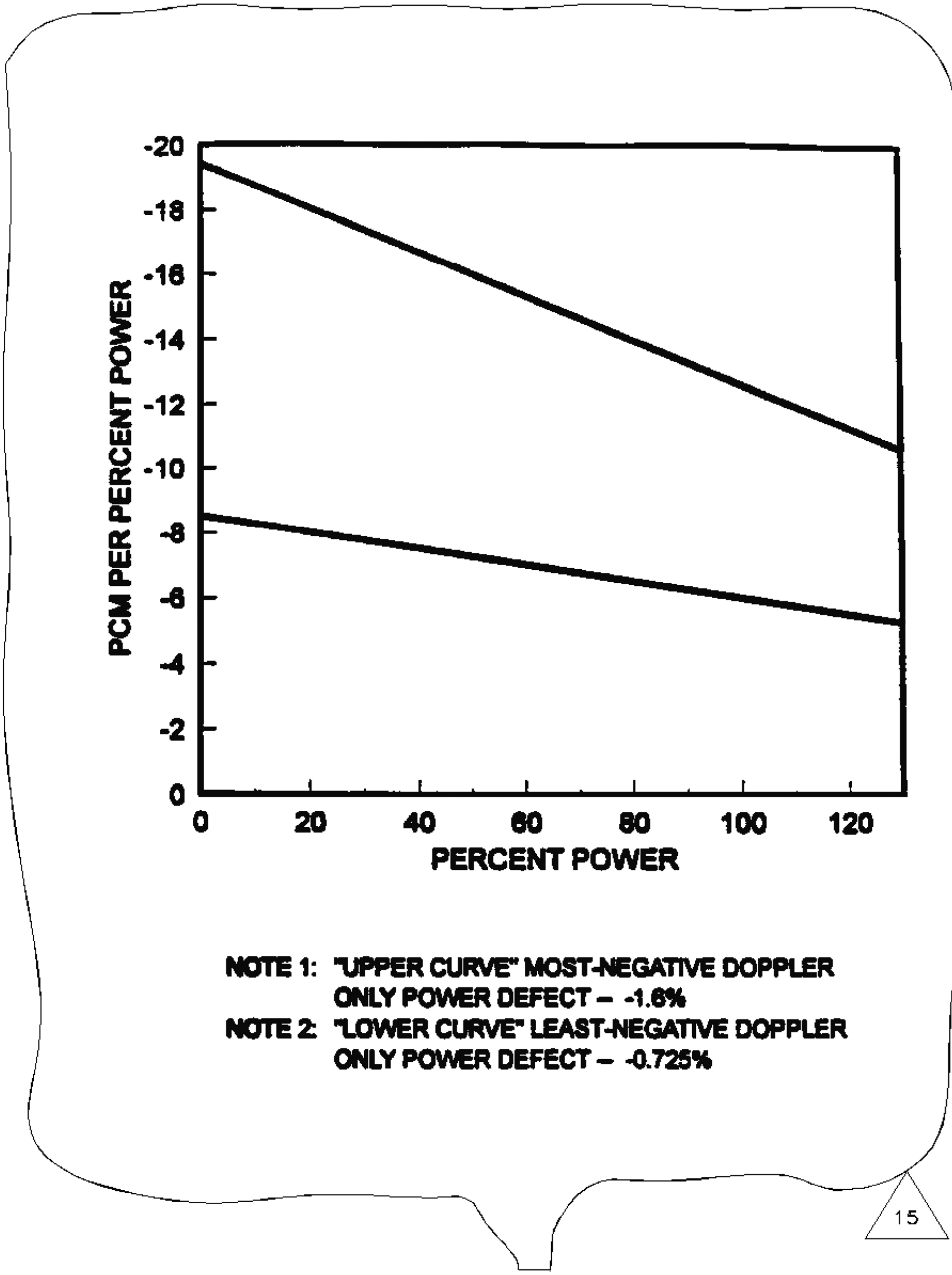
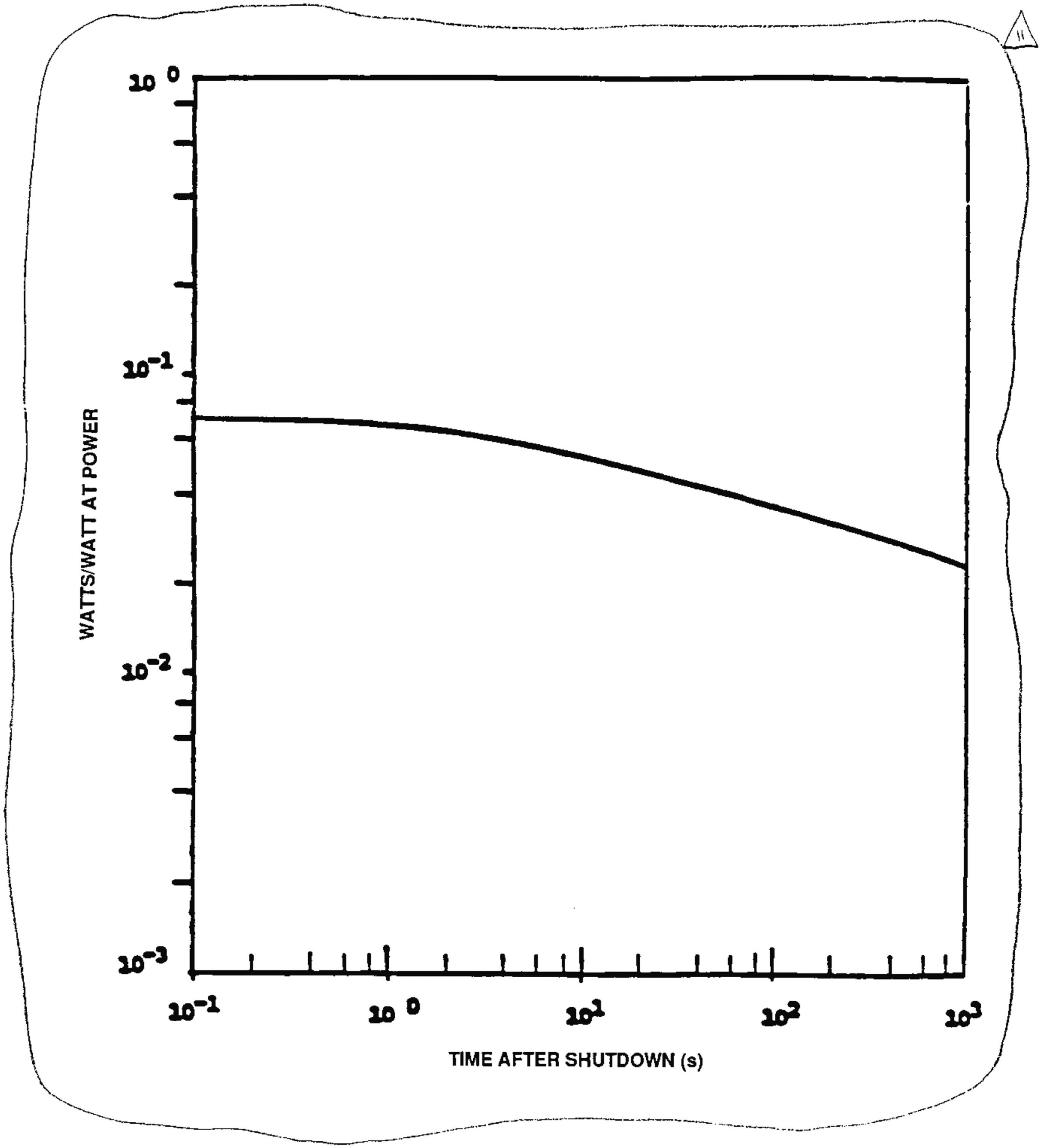
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS TIME AFTER TRIP
		FIGURE 15.1-4



REV 21 5/08

DOPPLER POWER COEFFICIENT
BOUNDED BY ACCIDENT ANALYSES

FIGURE 15.1-5



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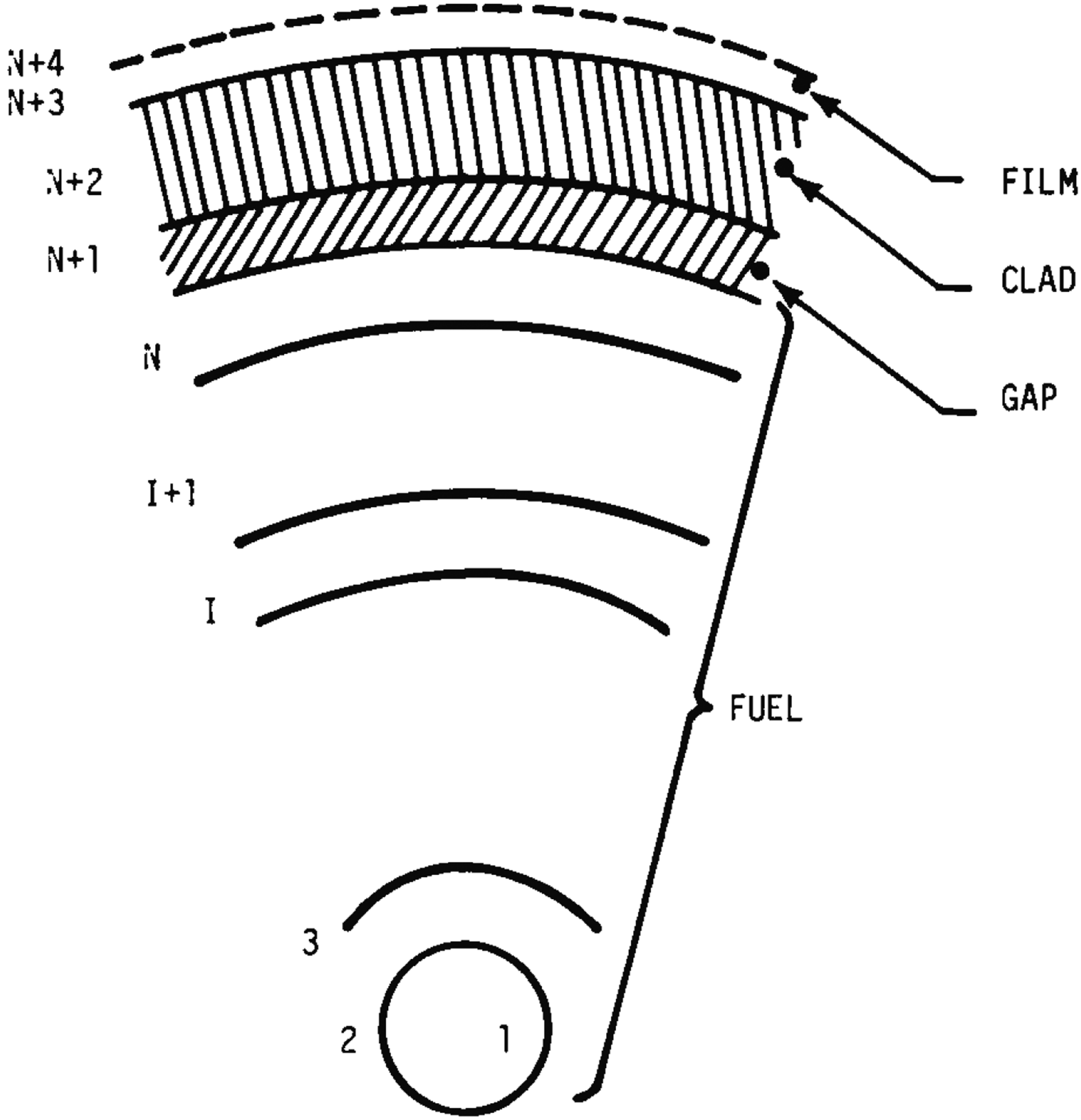
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

1979 ANS DECAY HEAT

FIGURE 15.1-6



REV 21 5/08

FUEL ROD CROSS-SECTION

FIGURE 15.1-7

15.2 CONDITION II - INCIDENTS OF MODERATE FREQUENCY

These faults at worst result in reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault; i.e., a Condition III or IV event. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system (RCS) overpressurization. The NRC acceptance criterion for ensuring that Condition II events will not result in fuel rod failures is that the minimum departure from nucleate boiling ratio (DNBR $\geq$ 1.23-1.25 as described in paragraph 4.4.1.1) satisfies the 95/95 criterion. That is, there is at least a 95-percent probability at a 95-percent confidence level (95/95 probability/confidence) that departure from nucleate boiling (DNB) will not occur on the limiting fuel rod. In addition, there is also a 95/95 probability/confidence that the peak kW/ft fuel rods will not exceed the melting temperature of uranium dioxide taken as 4900°F (unirradiated) and 4800°F at end of life (EOL). A bounding value of 4700°F is used in the safety analyses as described in section 4.4. The NRC acceptance criterion used to ensure that the reactor coolant pressure boundary integrity is maintained is that the reactor coolant pressure remains below the ASME Section III Code pressure limit (2750 psia). For the purposes of this report, the following faults have been grouped into this category:

A. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical condition.

B. Uncontrolled RCCA bank withdrawal at power.

C. RCCA misalignment.

D. Uncontrolled boron dilution.

E. Partial loss of forced reactor coolant flow.

F. Startup of an inactive reactor coolant loop.

G. Loss of external electrical load and/or turbine trip.

H. Loss of normal feedwater.

I. Loss of all ac power to the station auxiliaries.

J. Excessive heat removal due to feedwater system malfunctions.

K. Excessive load increase.

L. Accidental depressurization of the RCS.

M. Accident depressurization of the main steam system.

N. Inadvertent operation of the emergency core cooling system (ECCS) during power operation.

The Farley design incorporates a solid-state reactor protection system (RPS). Reference 1 describes the techniques used to evaluate the reliability of relay protection logic and demonstrates that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} for random component failures). The solid-state RPS design has been evaluated by the same methods as those used to evaluate the relay protection system design, and the same order of magnitude of reliability has been demonstrated.

Hence, because of the high reliability of the protection system, no special provision is proposed to be taken in the design to cope with the consequences of Condition II events without trip.

The time sequence of events during each Condition II fault is shown in table 15.2-1.

15.2.1 UNCONTROLLED RCCA BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION

15.2.1.1 Identification of Causes and Accident Description

A RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical, at hot zero power, or at power. The at-power case is discussed in subsection 15.2.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see subsection 15.2.4, Uncontrolled Boron Dilution).

The RCCA drive mechanisms are wired into preselected bank configurations. These circuits prevent the assemblies from being withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time. The RCCA drive mechanisms are of the magnetic latch type; coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the RPS:

A. Source Range High Neutron Flux Reactor Trip

This trip function is automatically actuated when either of two independent source range channels indicates a neutron flux level above a preselected, manually-adjustable setpoint. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

B. Intermediate Range High Neutron Flux Reactor Trip

This trip function is automatically actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually-adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels are reading above approximately 10 percent of full power. It is automatically reinstated when three of the four channels indicate a power level below this value.

C. Power Range High Neutron Flux Reactor Trip (Low Setting)

This trip function is automatically actuated when two of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when three of the four channels indicate a power level below this value.

D. Power Range High Neutron Flux Reactor Trip (High Setting)

This trip function is automatically actuated when two of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

E. High Positive Nuclear Flux Rate Reactor Trip

The high nuclear flux rate reactor trip is actuated when the positive rate of change of neutron flux on two-out-of-four nuclear power range channels indicates a rate above the preset setpoint. This trip function is always active.

In addition, control rod stops on a high intermediate range flux level (one of two) and a high power range flux level (one of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.2.1.2 Analysis of Effects and Consequences

15.2.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: First a spatial neutron kinetics computer code, TWINKLE (reference 2), is used to calculate the core average nuclear power transient, including the various core feedback effects, i.e., Doppler and moderator reactivity. FACTRAN (reference 3) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the average heat flux calculated by FACTRAN is used in THINC for transient DNBR calculations.

In order to give conservative results for a startup accident, the following assumptions are made.

A. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler power reactivity coefficient, a conservatively low (absolute magnitude) value for the Doppler power defect is used (960 pcm). (Note: Although this value of Doppler power defect is larger than that given in figure 15.1-5, it is still a conservatively low value for the Farley units.)

B. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much larger than the neutron flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. The analysis assumes a moderator temperature coefficient which is +7 pcm/°F at the zero power nominal temperature.

C. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, a larger initial fuel and water stored energy, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the nuclear flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux. The initial effective multiplication factor is assumed to be 1.0 since this results in the maximum nuclear power peak.

D. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent; no credit is taken for the source and intermediate range protection. Figure 15.2-1 shows that the rise in nuclear flux is so rapid that the effect of error in the trip setpoint on the actual time at which the rods release is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. (See subsection 15.1.5 for RCCA insertion characteristics.)

E. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at a conservative speed (45 in./min, which corresponds to 72 steps/min).

F. The DNB analysis assumes the most limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their high worth position.

G. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

H. The analysis assumes two reactor coolant pumps (RCPs) to be in operation (Mode 3 Technical Specification allowed operation). This is conservative with respect to the DNB transient.

I. The accident analysis employs the Standard Thermal Design Procedure (STDP) methodology. The use of STDP stipulates that the RCS flowrates will be based on a fraction of the thermal design flow for two RCPs operating and that the RCS pressure is 50 psi below nominal. Since the event is analyzed from hot zero power, the steady-state non-RTDP uncertainties on core power and RCS average temperature are not considered in defining the initial conditions.

15.2.1.2.2 Results

Figures 15.2-1 through 15.2-3 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35-percent nominal power. This insertion rate is greater than that for the two highest worth control banks, both assumed to be in their highest incremental worth region.

Figure 15.2-1 shows the neutron flux transient. The neutron flux overshoots the full-power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown in figure 15.2-2. The beneficial effect on the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full-power nominal value. Figure 15.2-3 shows the response of the hot spot average fuel and clad inner temperatures. The hot spot average fuel temperature increases to a value lower than the nominal full-power value. The analysis demonstrates that the DNB design basis is met.

15.2.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature continue to meet the DNB design basis. Thus, no fuel or clad damage is predicted as a result of this transient.

15.2.2 UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

15.2.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in fuel cladding damage and/or RCS overpressurization. Therefore, in order to avert damage to the fuel and/or fuel cladding, the RPS is designed to terminate any such transient before the DNBR falls below the limit value or the allowable fuel linear heat generation rate is exceeded. The RPS and pressurizer safety valves are designed to preclude the RCS pressure boundary safety limit.

The automatic features of the RPS which prevent core damage and preclude RCS overpressurization during the postulated accident include the following:

A. Power range neutron flux instrumentation actuates a reactor trip if two of four channels exceed an overpower setpoint. (This trip is credited for the reactor core protection and RCS overpressure analyses.)

B. Reactor trip is actuated if any two of three ΔT channels exceed an overtemperature ΔT (OTΔT) setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

C. Reactor trip is actuated if any two of three ΔT channels exceed an overpower ΔT (OPΔT) setpoint. This setpoint is automatically varied with coolant average temperature to ensure that the allowable fuel linear heat generation rate (≤ 22.4 kW/ft) is not exceeded.

D. A high-pressurizer pressure reactor trip is actuated from any two of three pressure channels set at a fixed point. This set pressure is less than the set pressure for the PSVs.

E. A high-pressurizer water level reactor trip is actuated from any two of three level channels set at a fixed point.

F. Power range neutron flux instrumentation actuates a reactor trip if two of four channels exceed a specified positive flux rate setpoint. (This trip is credited in the RCS overpressure analyses. It is not credited in the reactor core protection analyses.)

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

A. High neutron flux (one of four).

B. OPΔT (two of three).

C. OTΔT (two of three).

The manner in which the combination of OPΔT and OTΔT trips provide protection over the full range of RCS conditions is described in chapter 7. This description is illustrated by figures 15.1-1A and 15.1-1B which present the OPΔT and OTΔT safety analysis limits for the combination of allowable reactor coolant loop average temperatures and ΔTs for the design power distribution and flow. The OTΔT safety analysis limits are also varied as a function of primary coolant pressure. The boundaries of operation defined by the OPΔT trip and the OTΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure, and temperature) is bound by the combination of reactor trips: high neutron flux (fixed setpoint), high pressure (fixed setpoint), low pressure (fixed setpoint), OPΔT and OTΔT (variable setpoints).

The pressurizer safety valves are required to provide overpressure protection. The pressurizer safety valves, which have water filled loop seals, open to allow steam relief and thus RCS pressure relief when the pressurizer pressure exceeds the respective lift setpoint for each pressurizer safety valve. The main steam safety valves open to allow secondary pressure relief, thus increasing the heat removal capability of the secondary side when the steam generator pressure exceeds the respective lift setpoint for each main steam safety valve.

15.2.2.2 Analysis of Effects and Consequences

15.2.2.2.1 Method of Analysis

The reactor core protection and RCS overpressure transients are analyzed by the LOFTRAN[4] code. This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level. The core limits, as illustrated in figures 15.1-1A and 15.1-1B, are used as input to LOFTRAN to conservatively estimate the minimum DNBR during the transient.

The following assumptions are made for the DNB analysis in order to obtain conservative values of DNBR.

A. This accident is analyzed with the RTDP as described in WCAP-11397-P-A (reference 5). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in reference 5.

B. For reactivity coefficients, two cases are analyzed, reflecting minimum or maximum feedback conditions, that are described in table 15.1-2A.

C. The reactor trip on high neutron flux is assumed to be actuated at a conservative value which is defined in table 15.1-3. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

D. The RCCA trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

E. The maximum reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at a conservative speed (45 in./min, which corresponds to 72 steps/min).

F. The impact of a full power RCS T_{avg} window was considered for the uncontrolled RCCA bank withdrawal at power analysis. Explicit analysis performed for this event models the maximum T_{avg} value, which is limiting with respect to the calculated minimum DNBR.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in the overtemperature trip setpoint proportional to a decrease in margin to DNB (see figure 15.1-1B).

For the RCS overpressure transient, a conservative analysis was performed to demonstrate that the RCS overpressure safety limit will not be exceeded for an uncontrolled rod withdrawal during power operation (reference 15). The following assumptions are made in order to obtain conservative values of RCS pressure:

A. This accident is analyzed using the STDP. Therefore, initial power, pressure, and RCS temperatures are assumed to be within their respective allowable operating ranges with uncertainties applied in the conservative directions.

B. A range of initial reactor power levels (from 8% RTP to 102% RTP) are analyzed.

C. A range of reactivity insertion rates (from 15 pcm/s to 110 pcm/s) are analyzed. The largest reactivity insertion rate considered (110 pcm/s) is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at a conservative speed (45 in./min, which corresponds to 72 steps/min).

D. Minimum reactivity feedback conditions are modeled as described in table 15.1-2A.

E. Reactor trips on high neutron flux, high pressurizer pressure, and power range high positive neutron flux rate are assumed to be actuated at conservative values which are defined in table 15.1-3.

F. Pressurizer heater are conservatively modeled with setpoints appropriately modified based on the initial pressurizer pressure modeled. Pressurizer sprays are not modeled as the operation of the pressurizer spray valves is nonconservative for the overpressure transient.

G. Pressurizer safety valves are modeled with an opening delay to account for water-filled loop seals and the opening setpoint as the design pressure plus a set pressure shift. Pressurizer power operated relief valves are not modeled as their operation is nonconservative for the overpressure transient.

H. The main steam safety valves are modeled with bounding opening pressures in order to conservatively prolong the mismatch between core heat generation and secondary heat removal capability.

15.2.2.2.2 Results

Figures 15.2-4 and 15.2-5 show the transient response for a rapid RCCA bank withdrawal incident starting from full power with maximum feedback. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because of the rapid reactor trip with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result, and margin to DNB is maintained.

DNB Case:

The transient response for a slow RCCA bank withdrawal from full power with maximum feedback is shown on figures 15.2-6 and 15.2-7. Reactor trip on OTΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.2-8 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for both minimum and maximum reactivity feedback. It can be seen that the high neutron flux and OTΔT reactor trip functions provide DNB protection over the range of reactivity insertion rates considered. The minimum DNBR is never less than the limit value. Figures 15.2-9 and 15.2-10 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60-percent and 10-percent power, respectively. The results are similar to the 100-percent power case; however, as the initial power level decreases, the range over which the OTΔT trip is effective is increased. In neither case does the DNBR fall below the limit value. The calculated sequence of events for the DNB transient is shown on table 15.2-1.

Overpressure Case:

The results of the overpressure analysis demonstrated that protection against overpressure was maintained for all analyzed conditions. The calculated sequence of events for the overpressure analysis is shown in table 15.2-1. The transient response for the most limiting set of conditions is shown on figures 15.2-7A and 15.2-7B. Protection for the most limiting case is provided by the high pressurizer pressure reactor trip function, which occurs shortly after the start of the accident.

Figure 15.2-10A shows the peak RCS pressure as a function of reactivity insertion rate and initial power levels. Figure 15.2-10B illustrates the effect of reactivity insertion rate and initial power level on the RPS trip function. It can be seen that the high pressurizer pressure, high neutron flux, and power range high positive neutron flux rate trip functions provide RCS overpressure protection over the range of conditions considered. The peak RCS pressure is never greater than the overpressure safety limit value.

With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.2.2.3 <u>Conclusions</u>

The high neutron flux and OTΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; i.e., the analysis demonstrates that the DNB design basis is met. Thus, there will be no cladding damage and no release of fission products to the RCS.

For overpressure cases, the high pressurizer pressure, high neutron flux, and power range high positive neutron flux rate trip functions along with the pressurizer safety valves and steam generator safety valves provide adequate protection over the entire range of possible reactivity insertion rates and initial conditions analyzed; i.e., the analysis demonstrates that the integrity of the RCS pressure boundary is maintained as the maximum transient pressure does not exceed the RCS pressure boundary safety limit.

15.2.3 RCCA MISALIGNMENT

15.2.3.1 Identification of Causes and Accident Description

RCCA misalignment accidents include:

A. One or more dropped full-length assemblies.

B. A dropped full-length assembly bank.

C. A statically misaligned full-length assembly.

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod bottom signal which actuates a local alarm and a control room annunciator. Group demand position is also indicated. RCCAs move in preselected banks, and the banks always move in the same preselected sequence. Each bank of RCCAs consists of two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of the control rod drive mechanism withdraws the RCCA held by the mechanism. Mechanical failures are in the direction of insertion or immobility.

A dropped assembly or assembly banks may be detected by:

A. A sudden drop in the core power level as seen by the nuclear instrumentation system.

B. An asymmetric power distribution as seen on out-of-core neutron detectors.

C. Rod bottom signal.

D. A rod deviation alarm.

E. A rod position indication.

Misaligned assemblies may be detected by:

1. An asymmetric power distribution as seen on out-of-core neutron detectors.

2. A rod deviation alarm.

3. Rod position indicators.

The deviation alarm alerts the operator to rod deviation with respect to group demand position in excess of 5 percent of span. If the rod deviation alarm is not operable, the operator is required to log the RCCA positions in a prescribed time sequence to confirm alignment.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to ensure the alignment of the nonindicated assemblies.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

A. One or More Dropped RCCAs From the Same Group

The LOFTRAN computer code (reference 4) calculates the transient system response for the evaluation of the dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Transient reactor statepoints (temperature, pressure, and power) are calculated by LOFTRAN, and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with the methodology described in reference 7.

B. Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in reference 7, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

C. Statically Misaligned RCCA

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The analysis examines the case of the worst rod withdrawn from bank D inserted at the insertion limit with the reactor initially at full power. The analysis assumes this incident to occur at BOL since this results in the minimum value of the moderator temperature coefficient (least negative). This assumption maximizes the power rise and minimizes the tendency of the large moderator temperature coefficient (most negative) to flatten the power distribution.

The THINC code is used to confirm that the DNB design basis is met for the peaking factors associated with the statically misaligned RCCA.

15.2.3.2.2 Results

A. One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will reestablish power.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Thus, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller, after which the control system will insert the control bank to restore program conditions. Figure 15.2-11 shows a typical transient response to a dropped RCCA (or RCCAs) in the automatic rod control mode. In all cases, the DNB design basis is met.

Following plant stabilization, the operator may manually retrieve the RCCA by following approved operating procedures.

B. Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period since power is decreasing rapidly. The transient will proceed as described in part A; however, the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

C. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions.

The insertion limits in the Core Operating Limits Report may vary from time to time depending on several limiting criteria. The full-power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full-power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangement.

The DNB design basis is met for the RCCA misalignment with bank D inserted to its full-power insertion limit and one RCCA fully withdrawn. The analysis of this case assumes that the initial reactor power, pressure, and RCS temperature are at the nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

The DNB design basis is met for the RCCA misalignment with one RCCA fully inserted. The analysis of this case assumes that initial reactor power, pressure, and RCS temperatures are at the nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation rate is well below that which would cause fuel melting ($\leq$ 22.4 kW/ft).

After identifying an RCCA group misalignment condition, the operator must take action as required by the plant Technical Specifications and operating instructions.

15.2.3.3 <u>Conclusions</u>

The DNB design basis is met for cases of dropped RCCAs or dropped banks. For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNB design basis is met.

15.2.4 UNCONTROLLED BORON DILUTION

15.2.4.1 <u>Identification of Causes and Accident Description</u>

Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the chemical and volume control system (CVCS). Boron dilution is a manual operation under strict administrative controls with operating procedures limiting the total amount of dilution allowed. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve provides makeup to the RCS which can dilute the reactor coolant.

Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump. The rate of addition of unborated makeup water to the RCS when it is not at pressure is limited by the capacity of the primary water makeup pumps. The maximum addition rate in this case is 300 gal/min with both pumps running. The 300 gal/min reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flowrates of boric acid and primary grade water on the control board.

With the exception of dilution by chemical addition, two separate operations are required in order to dilute the RCS. The operator must switch from the automatic makeup mode to the dilute mode, and the start button must be depressed. Omitting either step would prevent dilution. Dilution by chemical addition does not affect the maximum dilution rates assumed for this accident.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or makeup water flowrates deviate from preset values as a result of system malfunction. The signals initiating these alarms will also cause the closure of control valves terminating the addition to the RCS.

15.2.4.2 Analysis of Effects and Consequences

15.2.4.2.1 Method of Analysis

Plant operation during refueling, cold and hot shutdown, startup, and power operation are considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident. Table 15.2-2 presents results of the boron dilution analysis for power, startup, and refueling operations. Also included in this table are pertinent analysis assumptions. Perfect mixing is assumed in the analysis. This assumption results in a conservative rate of RCS boron dilution.

15.2.4.2.2 Dilution During Refueling

During refueling, the following assumptions are made:

A. One residual heat removal (RHR) pump is operating to ensure continuous mixing in the reactor vessel.

B. The seal injection water supply to the reactor coolant pumps is isolated.

C. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution.

D. The boron concentration in the refueling water is approximately 2200 ppm[a], corresponding to a shutdown margin of at least 5-percent Δk with all RCCAs in; periodic sampling ensures that this concentration is maintained.

E. The source range detectors outside the reactor vessel are active and provide an audible count rate.

A minimum water volume of 3290 ft^3 in the RCS is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the RHR loop. A maximum dilution flow of 300 gal/min, limited by the capacity of the two primary water makeup pumps, and uniform mixing are assumed.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment. In addition, a high source range flux level is alarmed in the control room. The count rate increase is proportional to the subcritical multiplication factor.

For dilution during refueling, the boron concentration must be reduced from greater than 2200 ppm[a] to approximately 1750 ppm before the reactor will go critical. This would take at least 18 min. Within this time, the operator must recognize the high count rate signal and isolate the primary water makeup source by closing any one of several valves and stopping the reactor makeup water pumps.

The safety analyses for the refueling (Mode 6) uncontrolled boron dilution event use very conservative initial (2200 ppm) and critical (1750 ppm) boron concentrations, and assume all rods (control and shutdown) are inserted, to determine a limiting time for operator action to terminate an uncontrolled boron dilution event. The “higher” (i.e., greater than the Mode 6 limit) initial boron concentration assumed in the safety analyses results in a faster dilution of the boron in the RCS, compared to “lower” (such as those associated with the Mode 6 limit contained in the COLRs) boron concentrations, for a given dilution flowrate and active RCS volume. Thus, the “lower” Technical Specification values contained in the COLR for the Mode 6 boron concentration associated with maintaining a k_{eff} of ≤ 0.95 as discussed above, are bounded by the conservative assumptions made for the Mode 6 uncontrolled boron dilution analyses presented in FSAR paragraph 15.2.4.2.2.

The Mode 6 boron dilution accident described above assumes a maximum unborated water flow of 300 gal/min from the primary water makeup pumps. Isolation of the unborated water source from the RCS will preclude the analyzed accident described above. The isolation of the unborated water source to prevent a boron dilution is consistent with Required Action C.1 of

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a. The minimum refueling water storage tank (RWST) boron concentration is 2300 ppm. The 2200-ppm value bounds the case of an initial boron concentration of 2300 ppm (see table 15.2-2).

Technical Specification 3.9.2, “Nuclear Instrumentation,” which is applicable for the condition where the audible count rate is not available. Required Action C.1 specifies that the unborated water sources be isolated. The Bases for this Required Action explain that this action will ensure an inadvertent dilution of the RCS is prevented.

15.2.4.2.3 Dilution During Shutdown

A plant-specific evaluation of the boron dilution event during plant shutdown (hot and cold) was performed. This evaluation is based upon the operating procedure outlined in reference 8. The operating procedure is based upon a generic boron dilution analysis assuming active RCS and RHR volumes which are conservative with respect to the Farley units. Additionally, the operating procedure accommodates mid-loop cold shutdown operation. The operating procedure is applicable for maximum dilution flowrates up to 300 gal/min and minimum RHR flowrates of 1000 gal/min. Current plant procedures require one reactor makeup water pump to be secured when no reactor coolant pumps are running, limiting the maximum dilution flowrate to 150 gal/min. In the event of a boron dilution accident during plant shutdown, use of the operating procedure provides the plant operator with sufficient information to maintain an appropriate boron concentration to conservatively assure at least 15 min will be available for operator action to terminate the dilution prior to the reactor reaching a critical condition.

15.2.4.2.4 Dilution During Startup

In this mode, the plant is being taken from one long-term mode of operation, hot standby, to another, power. Typically, the plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions assumed for the analysis are as follows:

A. The dilution flow is the maximum capacity of the two primary water makeup pumps, 300 gal/min.

B. A minimum RCS water volume of 7735 ft^3, corresponding to the active RCS volume minus the pressurizer and surge line.

C. An initial boron concentration of 2100 ppm (see table 15.2-2), corresponding to a critical hot zero power condition, rods to insertion limits, and no xenon.

D. A critical boron concentration of 1800 ppm following reactor trip. This represents the maximum boron concentration at which the core can obtain critical conditions with all control rods inserted (less the most-reactive RCCA stuck out of the core), at hot zero power conditions. The 300-ppm change from the initial condition noted above is a conservative minimum value.

The startup mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching critically, thus ensuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range (nominally at 10^5 cps). Too fast of a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip, and the reactor would immediately shut down.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power until the power range high neutron flux low setpoint is reached and a reactor trip occurs. From the time of reactor trip, a time period > 15 min is available for operator action prior to return to criticality. (See table 15.2-2.)

15.2.4.2.5 Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions assumed for this analysis are the following:

A. With the units at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging flow control valve. Although only one charging pump is normally in operation, the analysis is performed assuming the dilution flow is the maximum capacity of two charging pumps at power operation conditions. Although the dilution flowrate is less, a conservatively large dilution flowrate of 300 gal/min. is assumed in this analysis. This flowrate is the maximum deliverable dilution flowrate and can be assumed to include seal injection water.

B. A minimum RCS water volume of 7735 ft^3, corresponding to the active RCS volume minus the pressurizer.

C. An initial boron concentration of 2100 ppm, corresponding to a critical hot full-power condition, with the rods at their insertion limits.

D. A critical boron concentration of 1800 ppm following reactor trip. This represents the maximum boron concentration at which the core can obtain critical conditions at hot zero power conditions with all control rods inserted (less the most reactive RCCA stuck out of the core). No credit is taken for xenon. The 300-ppm change from the initial condition noted above is a conservative minimum value.

With the reactor in automatic rod control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in available shutdown margin. The rod insertion limit alarms (low and low-low settings) alert the operator at least 15 min. prior to criticality. (See table 15.2-2.) This is sufficient time to determine the cause of dilution, isolate the reactor makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control, a rod stop alarm is initiated 3 percent below the OTΔT reactor trip setpoint, which would alert the operator. If no operator action is taken, however, the power and temperature rise will cause the reactor to reach the OTΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be 3.5 pcm/s, which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis (FSAR subsection 15.2.2). Following reactor trip, there are greater than 15 min. prior to criticality. (See table 15.2-2.) This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

15.2.4.3 Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the conditions. The maximum reactivity addition due to the dilution is slow enough to allow the operator sufficient time to determine the cause of the addition and take corrective action before shutdown margin is lost.

15.2.5 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.2.5.1 Identification of Causes and Accident Description

A partial loss-of-coolant flow accident can result from a mechanical or electrical failure in a RCP, or from a fault in the power supply to the pump. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the RCPs is supplied through separate buses from a transformer connected to the generator. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines so that the pumps will continue to provide forced coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 s after reactor trip before any transfer is made. Since each pump is on a separate bus, a single bus fault will not result in the loss of more than one pump. The simultaneous loss of power to all the RCPs (subsection 15.3.4) is a highly unlikely event.

The necessary protection against a partial loss-of-coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two of three low flow signals in any reactor coolant loop. Above the P-8 setpoint (see table 7.2-2), low flow in any loop will actuate a reactor trip. Between approximately 10% power (P-7 setpoint) and the P-8 setpoint, low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below P-7.

A reactor trip signal from OPDT and OTDT provide backup protection to the low flow signal.

15.2.5.2 Analysis of Effects and Consequences

15.2.5.2.1 Method of Analysis

A loss of one pump with three loops operating has been analyzed. The lost pump is assumed to be coasting down.

This transient is analyzed by three digital computer codes. First, the LOFTRAN[(4)] code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the flows, the primary system pressure and temperature transients, and the nuclear power transient following reactor trip. Next, the FACTRAN[(3)] code is used to calculate the heat flux transient based on the nuclear power and the flow from LOFTRAN. Finally, the THINC code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

15.2.5.2.2 Initial Conditions

The accident is analyzed using the RTDP. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397 (reference 5).

15.2.5.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. The total integrated Doppler reactivity from 0- to 100-percent power is assumed to be -0.016 Δk.

The most positive moderator temperature coefficient (+7 pcm/ΔF) is assumed since this results in the maximum core power and hotspot heat flux during the initial part of the transient when the minimum DNBR is reached.

15.2.5.2.4 Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the as-built pump characteristics; it is based on high estimates of system pressure losses.

15.2.5.2.5 Results

Figures 15.2-12 through 15.2-17 show the transient response for the loss of one reactor coolant pump with three loops initially in operation. The figures include trends of the core flow, loop flow, nuclear power, and core heat flux coastdowns. The reactor is tripped on a low loop flow signal. Figure 15.2-17 shows that the DNB design basis is met.

For the case analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperatures do not increase far above the respective initial values.

The calculated sequence of events is shown in table 15.2-1. The affected RCP will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation (two RCPs). With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.2.5.3 Conclusions

The analysis shows that the DNB design basis is met during the transient. Thus, there will be no cladding damage and no release of fission products to the RCS.

15.2.6 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

15.2.6.1 Identification of Causes and Accident Description

If the plant were to operate with one pump out of service, there would be reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power with an inactive loop, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop. With the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Starting an idle reactor coolant pump without first bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity excursion and subsequent power increase due to the moderator density reactivity feedback effect.

Based on the expected frequency of occurrence, the Startup of an Inactive Loop event is classified as a Condition II event (an incident of moderate frequency) as defined by the American Nuclear Society Nuclear Safety Criteria for the Design of Stationary PWR Plants.

15.2.6.2 Sequence of Events and Systems Operation

Following the startup of the inactive reactor coolant pump, the flow in the inactive loop will accelerate to full flow in the forward direction over a period of several seconds. Since the Technical Specifications require all reactor coolant pumps to be operating while in Modes 1 and 2, the maximum initial core power level for the Startup of an Inactive Loop transient is approximately 0 MWt. Under these conditions, there can be no significant reactivity insertion because the RCS is initially at a nearly uniform temperature. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. Thus, there will be no increase in core power and no automatic or manual protective action is required.

15.2.6.3 Conclusions

The Startup of an Inactive Coolant Loop event results in an increase in reactor vessel flow while the reactor remains in a subcritical condition. No analysis is required to show that the minimum DNBR is satisfied for this event.

15.2.7 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP

15.2.7.1 Identification of Causes and Accident Description

Major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. A loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating condition. For either case, offsite power is available for the continued operation of plant components such as the RCPs. The case of loss of all ac power is analyzed in subsection 15.2.9.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant is designed to accept a step loss of load from 100-percent to 50-percent load without actuating a reactor trip with all NSSS control systems in automatic (reactor control system, pressurizer pressure and level, steam generator water level control, and steam dumps). The automatic steam dump system with 40-percent dump capacity to the condenser, together with the reactor control system, is able to accommodate the load rejection. Reactor power is reduced to a new equilibrium value consistent with the capability of the rod control system. The pressurizer power-operated relief valves (PORVs) may be actuated but the PSVs and the steam generator safety valves do not lift for the 50-percent load rejection with steam dump.

For a turbine or generator trip, such as would result from a loss of condenser vacuum, the reactor would be tripped directly (unless it is below P-9, approximately 50-percent power) from a signal derived from the turbine autostop oil pressure and/or turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere, and main feedwater flow would be lost. For this situation, steam generator level would be maintained by the auxiliary feedwater (AFW) system to ensure adequate residual and decay heat removal.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high-pressurizer pressure signal, the high-pressurizer water level signal, the OTΔT signal, or the low-low steam generator water level signal.

The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The PSVs and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming operation of the steam dump system, pressurizer spray, pressurizer PORVs, automatic RCCA control, or the direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safeguard design rating (105 percent of nominal full-power steam flow) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss-of-heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The PSVs are then able to maintain the RCS pressure within 110 percent of the RCS design pressure without a direct reactor trip on turbine trip action.

The Farley RPS and primary and secondary system designs preclude overpressurization. A more complete discussion of overpressure protection can be found in reference 6.

15.2.7.2 Analysis of Effects and Consequences

15.2.7.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This assumption is made to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins; it delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst-case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for AFW (except for long-term recovery) to mitigate the consequences of the transient.

The total loss of load transient is analyzed with the RETRAN (references 12, 13, and 14) computer code. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves.

The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below.

A. For DNB considerations, the accident is analyzed using the RTDP. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation (see table 15.1-2B). Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397.[5]

B. The total loss of load transient is analyzed with minimum reactivity feedback (BOL) conditions for 2 cases, with and without pressurizer pressure control. The cases analyzed model a least-negative moderator temperature coefficient and least-negative Doppler coefficient as indicated in table 15.1-2A. Historically, cases modeling EOL reactivity feedback conditions were also performed; however, analyses have demonstrated that this event is limiting at BOL conditions. Therefore, a bounding analysis at BOL conditions is performed.

C. From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual rod control. If the reactor were in automatic rod control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. The loss of load event is analyzed both with and without pressurizer pressure control. The pressurizer PORVs and sprays are assumed operable for the case with pressure control. The case with pressure control minimizes the increase in primary pressure which is conservative for the DNBR transient. The case without pressure control maximizes the pressure increase which is conservative for the RCS overpressurization criterion.

E. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur.

F. Only the OTΔT, high pressurizer pressure, and low-low steam generator water level reactor trips are assumed operable for the purpose of this analysis. No credit is taken for a reactor trip on high pressurizer level or the direct reactor trip on turbine trip.

G. No credit is taken for the operation of the steam dump system or steam generator PORVs. This assumption maximizes secondary pressure. The main steam safety valve model includes allowances for safety valve setpoint uncertainty and accumulation.

H. The analysis value for the pressurizer safety valve set pressure includes a $\pm$2 percent uncertainty. For the case analyzed primarily for DNBR (pressurizer pressure control case), the uncertainty is applied in the negative direction, thus reducing the safety analysis set pressure. For the case analyzed primarily for peak RCS pressure, the uncertainty is applied in the positive direction. The peak pressure case also considers a 1.6-s water purge time due to the pressurizer safety valve loop seals. Steam relief occurs following the 1.6-s purge time.

15.2.7.2.2 Results

Two cases were analyzed for a total loss of load from 100 percent of NSSS power.

A. Minimum feedback with pressure control.

B. Minimum feedback without pressure control.

The calculated sequence of events for each case is presented in table 15.2-1.

Case A

Figures 15.2-19 through 15.2-21 show the transient response for the total loss of steam load event under BOL conditions, including a +7 pcm/°F moderator temperature coefficient, with pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux increases until the reactor is tripped, and although the DNBR value decreases below the initial value, it remains well above the design basis limit throughout the entire transient. The pressurizer relief valves and sprays maintain primary pressure below 110 percent of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

Case B

Figures 15.2-22 through 15.2-25 show the transient response for the total loss of steam load event under BOL conditions, including a zero moderator temperature coefficient without pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux remains essentially constant at full power until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The PSVs are actuated and maintain primary pressure below 110 percent of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

15.2.7.3 Conclusions

The results of this analysis show that the plant design is such that a total loss of external electrical load without a direct reactor trip presents no hazards to the integrity of the RCS or the main steam system. The analysis demonstrates that the DNB design basis is met. The peak primary and secondary system pressures remain below 110 percent of design at all times.

15.2.8 LOSS OF NORMAL FEEDWATER

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, pipe breaks, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss-of-heat sink. If an alternate supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Loss of significant water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the core does not approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

A. Reactor trip on low-low water level in any steam generator.

B. Two motor-driven auxiliary feedwater pumps which are started automatically on one of the following:

   1. Low-low water level in any steam generator.
   2. Loss of both main feedwater pumps.
   3. Any SI signal.
   4. Loss of offsite power (automatic transfer to diesel generators).

   The motor-driven auxiliary feedwater pumps can also be started manually from the control room.

C. One turbine-driven auxiliary feedwater pump which is started automatically on one of the following:

   1. Low-low water level in any two of three steam generators.
   2. Undervoltage on any two of three reactor coolant pump buses.

   The turbine-driven auxiliary feedwater pump can also be started manually from the control room.

The motor-driven AFW pumps are connected to vital buses which are powered by diesel generators if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary system. The controls are designed to start both types of pumps within 60 s, even if a loss of all ac power occurs simultaneously with loss of normal feedwater. The AFW pumps are normally aligned to take suction from the condensate storage tank for delivery to the steam generators. A backup source of water for the pumps is provided by the safety-related portion of the service water system (see section 6.5). The RPS and AFW system designs ensure that reactor trip and AFW flow will occur following any loss of normal feedwater.

The analysis shows that, following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat, thus preventing overpressurization of the RCS, overpressurization of the secondary side, or uncovery of the reactor core. Consequently, the plant is able to return to a safe condition.

15.2.8.2 <u>Analysis of Effects and Consequences</u>

15.2.8.2.1 Method of Analysis

A detailed analysis using the RETRAN-02 (references 12, 13, and 14) computer code is performed in order to determine the plant transient following a loss of normal feedwater. The code describes the core neutron kinetics; RCS, including natural circulation, pressurizer, pressurizer PORV heaters and sprays, steam generators, and main steam safety valves; and the AFW system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The following assumptions are made in the analysis.

A. The plant is initially operating at 102 percent of the NSSS power (2785 MWt) with all three RCPs in operation providing a constant reactor coolant volumetric flow equal to the thermal design. A conservatively high RCP heat addition of 15 MWt (5 MWt/pump) is assumed. It is assumed that the operator manually trips two of three RCPs 10 min after reactor trip (rod motion). At this time the RCP heat addition is reduced from 15 MWt to 5 MWt.

B. An uncertainty of $\pm$ 6°F on the initial reactor vessel average coolant temperature is conservatively assumed to account for the temperature uncertainty on nominal temperature and also includes a -1.0°F bias due to cold leg streaming. The initial pressurizer pressure uncertainty is 50 psi and is conservatively added to the nominal pressure value.

C. Reactor trip occurs on steam generator low-low water level at 19% of narrow range span.

D. It is assumed that two motor-driven AFW pumps are available to supply a minimum of 350 gal/min to three steam generators, 60 s following a low-low steam generator water level signal. The worst single failure, for this analysis, is the loss of the turbine-driven AFW pump.

E. The AFW system is actuated by a low-low steam generator water level signal at 19% of narrow range span. AFW flow begins following a 60-s delay. The AFW line purge volume is conservatively assumed to be the maximum value for either unit of 140 ft^3, and the initial AFW enthalpy is assumed to be 80.83 Btu/lbm.

F. The pressurizer sprays and PORVs are assumed operable. This maximizes the pressurizer water volume. If the spray valves and/or the pressurizer pressure system did not operate, the PSVs would prevent the RCS pressure from exceeding the RCS design pressure limit during this transient.

G. The pressurizer proportional and backup heaters are assumed operable. The proportional heaters output is modulated by the master pressure PI controller to maintain the reference pressure of 2,235 psig. Maximum output is provided when the pressurizer pressure decreases below the reference pressure (equivalent error of -15 psi). The capacity of the proportional heaters is 0.375 MWt. The backup heaters are also actuated on decreasing pressure (equivalent error of -25 psi) or on a pressurizer water level in-surge greater than 5% span above the programmed reference level. The capacity of the backup heaters is 1.025 MWt. The total capacity of the pressurizer heaters is 1.4 MWt. This represents an addition to the RCS energy which must be removed by the AFW system.

H. Secondary system steam relief is achieved through the self-actuated main steam safety valves. Note that steam relief will, in fact, be through the steam generator atmospheric relief valves or condenser dump valves for most cases of loss of normal feedwater. However, since the condenser dump valves and controls are not safety grade and their availability would lessen the consequences of the event, they have been assumed unavailable.

I. The main steam safety valves are modeled assuming a 3% tolerance and a conservative accumulation model (3% accumulation for Banks 1, 2, and 3; 2% accumulation for Bank 4, and 10-psi accumulation for Bank 5, respectively, beginning with the safety valve with the lowest setpoint).

J. Core residual heat generation is based on the 1979 version of ANS 5.1 (reference 9). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

K. This analysis bounds steam generator tube plugging levels of 0% to 20%.

The assumptions detailed above are designed to minimize the heat removal capability of the secondary system and to maximize the potential for water relief from the RCS by maximizing the expansion of the primary system.

Note that the analysis assumption addressing the securing of 2 of 3 RCPs is met by incorporating a continuous action step in the EOPs to secure 2 of 3 RCPs if dump steam is not effective at controlling RCS temperature following a reactor trip. The time constraint of 10 min is an analysis input assumption. The procedure action is sufficient to ensure that the analysis remains bounding even if the RCPs are not secured within 10 min.

15.2.8.2.2 Results

Figure 15.2-26 shows plant parameters following a loss of normal feedwater with the assumptions listed in the previous subsection. Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to reduction of the steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the motor-driven AFW pumps automatically start; consequently, reducing the rate at which the steam generator water level is decreasing.

The capacity of the motor-driven AFW pumps is such that sufficient heat transfer is available to dissipate core residual heat without water relief through the RCS pressurizer relief or safety valves. From figure 15.2-26, sheet 1, it can be seen that at no time does the pressurizer water volume exceed the capacity of the pressurizer (1400 ft^3). Therefore, at no time is there water relief from the pressurizer. If the AFW delivered is > 350 gal/min, or the initial reactor power is < 102% of the NSSS rating, or the steam generator water level in one or more steam generators is above the conservatively low 19% narrow-range span level assumed for the low-low steam generator setpoint, the results for this transient will be bounded by the analysis presented. The calculated sequence of events for this accident is listed in table 15.2-1.

15.2.8.3 <u>Conclusions</u>

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the AFW capacity is such that the reactor coolant water is not relieved from the pressurizer relief or safety valves. In addition to the NRC acceptance criteria for Condition II events described in section 15.2, the analysis of the loss of normal feedwater event meets the NRC acceptance criteria specific to a loss of normal feedwater event. That is, the analysis demonstrates that there is no overpressurization of the primary or secondary side. In addition, the Westinghouse criterion that the pressurizer does not go water solid is also satisfied.

15.2.9 LOSS OF ALL AC POWER TO THE STATION AUXILIARIES

15.2.9.1 <u>Identification of Causes and Accident Description</u>

A complete loss of nonemergency ac power will result in a loss of power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip or by a loss of the onsite ac distribution system. The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below.

A. The emergency diesel generators will start on a loss of voltage on the plant emergency buses and begin to supply plant vital loads.

B. Plant vital instruments are supplied by emergency power sources.

C. As the steam system pressure rises following the trip, the steam system PORVs are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the PORVs are not available, the self-actuated main steam safety valves will lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

D. As the no-load temperature is approached, the steam system PORVs (or the self-actuated safety valves, if the PORVs are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

The following provide the necessary protection against a loss of all ac power.

A. Reactor trip on low-low water level in any steam generator.

B. Two motor-driven AFW pumps that are started on:

    1. Low-low water level in any steam generator.
    2. Trip of both main feedwater pumps.
    3. Any SI signal.
    4. Loss of offsite power (automatic transfer to diesel generators).
    5. Manual actuation.

C. One turbine-driven auxiliary feedwater pump that is started on:

    1. Low-low water level in any two steam generators.
    2. Undervoltage on any two RCP buses.
    3. Manual actuation.

The AFW system is initiated as discussed in the loss of normal feedwater analysis (subsection 15.2.8). The turbine-driven pump utilizes steam from the secondary system and exhausts it to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The AFW pumps are normally aligned to take suction from the condensate storage tank for delivery to the steam generators. A backup source of water for the pumps is provided by the safety-related portion of the service water system (see section 6.5). The RPS and AFW system designs ensure that reactor trip and AFW flow will occur following any loss of normal feedwater, including from a loss of ac power to the station auxiliaries.

Following the loss of power to the RCPs, coolant flow is necessary for core cooling and the removal of residual and decay heat.

Heat removal is maintained by natural circulation in the RCS loops. Following the RCP coastdown, the natural circulation capability of the RCS will remove decay heat from the core, aided by the AFW flow in the secondary system. Demonstrating that acceptable results can be obtained for this event proves that the resultant natural circulation flow in the RCS is adequate to remove decay heat from the core.

The first few seconds after the loss of ac power to the RCPs will closely resemble a simulation of the complete loss of flow event (subsection 15.3.4, where it is demonstrated that the DNB design basis is satisfied). Therefore, the DNB aspects for the station blackout event are not explicitly evaluated in this analysis. The analysis shows that, following a loss of all ac power to the station auxiliaries, RCS natural circulation and the AFW system are capable of removing the stored and residual heat, consequently preventing overpressurization of the RCS, overpressurization of the secondary side, or uncovery of the reactor core. The plant is, therefore, able to return to a safe condition.

15.2.9.2 Analysis of Effects and Consequences

15.2.9.2.1 Method of Analysis

A detailed analysis using the RETRAN-02 (references 12, 13, and 14) computer code is performed in order to determine the plant transient following a loss of all ac power. The code describes the core neutron kinetics; RCS, including natural circulation, pressurizer, pressurizer PORV heaters and sprays, steam generators, and main steam safety valves; and the AFW system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The major assumptions used in this analysis are identical to those used in the loss of normal feedwater analysis (subsection 15.2.8) with the following exceptions. Note that with the exception of the assumed safety analysis limit, the remaining AFW system modeling assumptions (items E and F, below) are consistent with those used in the loss of normal feedwater.

A. Loss of ac power is assumed to occur at the time of reactor trip on low-low SG water level. No credit is taken for the immediate insertion of the control rods as a result of the loss of ac power to the station auxiliaries.

B. Power is assumed to be lost to the RCPs. To maximize the amount of stored energy in the RCS, the power to the RCPs is not assumed to be lost until after the start of rod motion.

C. A heat transfer coefficient in the steam generators associated with RCS natural circulation is assumed following the RCP coastdown.

D. The RCS flow coastdown is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, the as-built pump characteristics, and conservative estimates of system pressure losses.

E. It is assumed that two motor-driven AFW pumps are available to supply a minimum of 350 gal/min to three steam generators, 60 s following a low-low steam generator water level signal. The worst single failure, for this analysis, is the loss of the turbine-driven AFW pump.

F. The AFW system is actuated by a low-low steam generator water level signal at 16.0% of narrow range span. AFW flow begins following a 60-s delay. The AFW line purge volume is conservatively assumed to be the maximum value for either unit of 140 ft^3, and the initial AFW enthalpy is assumed to be 80.83 Btu/lbm.

Plant characteristics and initial conditions are further discussed in section 15.1. Consistent with the loss of normal feedwater analysis, the most-limiting single failure occurs in the AFW system.

15.2.9.2.2 Results

Figure 15.2-27 shows plant parameters following a loss of offsite power with the assumptions listed above.

The first few seconds after the loss of ac power to the RCPs will closely resemble a simulation of the complete loss of flow incident, i.e., core damage due to rapidly increasing core temperatures is prevented by the reactor trip on the low-low steam generator water level signal. After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The RETRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

The capacity of the motor-driven AFW pumps is such that sufficient heat transfer is available to dissipate core residual heat without water relief through the RCS pressurizer relief or safety valves. From figure 15.2-27, sheet 1, it can be seen that at no time does the pressurizer water volume exceed the capacity of the pressurizer (1400 ft^3). Therefore, at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in table 15.2-1. As shown in figure 15.2-27, the plant approaches a stabilized condition following reactor trip, pump coastdown, and AFW initiation.

15.2.9.3 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. The DNBR transient is bounded by the complete loss of flow event (subsection 15.3.4) and remains above the safety analysis limit value. AFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves. The RCS and main steam system pressures remain within their respective pressure limits.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long-term heat removal capability exists following RCP pump coastdown to prevent fuel or clad damage.

In addition to the NRC acceptance criteria for Condition II events described in section 15.2, the analysis of the loss of all ac power to station auxiliaries meets the NRC acceptance criteria specific to the loss of all ac power to station auxiliaries event. That is, the analysis demonstrates that (1) there is no overpressurization of the primary or secondary side and (2) the natural circulation capacity of the RCS provides sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown. In addition, the Westinghouse criteria that the pressurizer does not go water solid are also satisfied.

15.2.9.4 Environmental Consequences of a Postulated Loss of ac Power to the Plant Auxiliaries

The postulated incidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage. This analysis incorporates assumptions of defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the primary and secondary systems. Parameters used in the analysis are listed in table 15.2-3.

The conservative assumptions used to determine the equilibrium concentrations of isotopes in the secondary system are as follows:

A. The primary to secondary leakage in steam generators occurs when the reactor starts up and the leakage remains constant during plant operation.

B. The primary to secondary leakage is evenly distributed in steam generators.

C. Primary coolant noble gas activity is associated with 1-percent defective fuel given in table 11.1-2 and iodine activity at 1.0 μCi/gm DEI_{131}. The secondary side concentration of iodine is assumed to be at 0.1 μCi/gm DEI_{131}.

D. The iodine partition factor is as follows:

$$\frac{\text{amount of iodine/ unit mass steam}}{\text{amount of iodine / unitl mass liquid}} = 0.1 \text{ in the steam generators}$$

E. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off-gas system.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated loss of ac power to the plant auxiliaries:

A. Offsite power is lost; main steam condensers are not available for steam dump.

B. Eight hours after the accident, the RHR system starts operation to cool down the plant.

C. After 8 hours following the accident, no steam and activity are released to the environment.

D. There is no air ejector release and no steam generator blowdown during the accident.

E. The primary to secondary leakage is evenly distributed in steam generators.

F. An iodine spike of 60 μCi/gm DEI_{131} is assumed to exist previous to the accident.

G. No noble gas is dissolved in the steam generator water.

H. The iodine partition factor is as follows:

$$\frac{\text{amount of iodine/ unit mass steam}}{\text{amount of iodine / unitl mass liquid}} = 0.1 \text{ in the steam generators}^{(a)}$$

I. During the postulated accident, iodine carryover from the primary side is uniformly mixed with the water in the steam generators and is diluted by the incoming feedwater.

J. The steam release for cooling down the plant is equally contributed by all steam generators.

K. The 0 to 2- and 2 to 8-h atmospheric diffusion factors given in appendix 15B and the 0 to 8-h breathing rate of 3.47×10^{-4} m^3/s are applicable.

The steam releases to the atmosphere for the loss of ac power are given in table 15.2-3.

The gamma, beta, and thyroid doses for the loss of ac power to the plant auxiliaries for the conservative analysis at the site boundary and the low-population zone are a small fraction of the limits as defined in 10 CFR 100 (25-rem whole body and 300-rem thyroid) as shown in table 15.2-3.

The potential for uncovery of the steam generator tubes during the event has also been evaluated for impact on doses. The tube uncovery was assumed to exist for the first 1/2 h of the accident and the tube leakage locations were assumed to all be near the top of the tube bundle and, thus, subject to the uncovery. With the primary to secondary leakage entering the vapor space, no credit was taken for mixing with the secondary coolant, nor was credit taken for a partition factor within the steam generator (i.e., the primary coolant was assumed to be released directly to the environment). The uncovery does not impact the release of noble gases to the environment; thus, the gamma and beta doses are not affected. The uncovery does result in an increase in the accident releases of iodine, but the thyroid dose remains well within the limits as defined in 10 CFR 100.

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

15.2.10.1 Identification of Causes and Accident Description

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (high neutron flux, OPΔT, and OTΔT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit value.

An example of excessive heat removal from the RCS is excessive feedwater flow due to full opening of a feedwater control valve. The valve opening may be due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater will cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

A second example of excessive heat removal from the primary system is the transient associated with the accidental opening of the high pressure heater bypass valve which diverts flow around the number six feedwater heaters. In the event of an accidental opening of the high pressure heater bypass valve, there is a sudden reduction in inlet feedwater temperature to the steam generators. This increased subcooling will create a greater load demand on the RCS.

The steam generator high-high level trip is provided to prevent the continuous addition of excessive feedwater to a steam generator.

15.2.10.2 Analysis of Effects and Consequences

15.2.10.2.1 Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN (reference 4) computer code. This code simulates a multiloop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate acceptable consequences in the event of a feedwater system malfunction. Feedwater temperature reduction due to number six heater bypass in conjunction with second stage reheater drains dumped to the condenser is considered. Additionally, excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully is considered.

Two excessive feedwater flow cases are analyzed as follows.

A. Accidental opening of one feedwater control valve with the reactor just critical at zero-load conditions assuming a conservatively large moderator density coefficient characteristic of EOL conditions.

B. Accidental opening of one feedwater control valve with the reactor in automatic rod control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions.

A. This accident is analyzed with the RTDP as described in WCAP-11397-P-A (reference 5); therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR described in reference 5.

B. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 184 percent of nominal feedwater flow to one steam generator.

C. For the feedwater control valve accident at zero-load condition, a feedwater valve malfunction occurs that results in an increase in flow to one steam generator from zero to the nominal full-load value for one steam generator.

D. For the zero-load condition, feedwater temperature is at a conservatively low value of 32°F.

E. The initial water level in all the steam generators is at the nominal level.

F. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

G. No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

H. The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal that closes all feedwater main control and feedwater control-bypass valves, indirectly closes all feedwater isolation valves, and trips the main feedwater pumps and turbine generator.

Normal reactor control systems and engineered safety systems (e.g., SI) are not required to function. The RPS may actuate to trip the reactor due to an overpower condition. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event. The steam generator overfill protection system meets the requirements of Generic Letter 89-19.

15.2.10.2.2 Results

Opening of the high pressure heater bypass valve and dump of the reheater drain causes a reduction in feedwater temperature which increases the thermal load on the primary system. The reduction in feedwater temperature is < 65°F, resulting in an increase in heat load on the primary system of < 10 percent of full power. The increased thermal load due to the opening of the high pressure heater bypass valve thus would result in a transient very similar (but of reduced magnitude) to that presented in subsection 15.2.11 for an excessive load increase incident, which evaluates the consequences of a 10-percent step-load increase. Therefore, the results of analyses are not presented.

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the abovementioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in subsection 15.2.1, Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition, and therefore, the results of the analyses are not presented. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent.

The full-power case (EOL maximum reactivity feedback with automatic rod control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in the affected steam generator reaches the high-high level setpoint. Assuming the reactor to be in manual rod control results in a slightly less severe transient. The rod control system is, however, not required to function for this event.

For all cases of excessive feedwater flow, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control valves, closure of manual feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. In addition, the feedwater isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Following turbine trip, the reactor will automatically be tripped either directly due to the turbine trip or due to one of the reactor trip signals discussed in subsection 15.2.7 (Loss of External Electrical Load). If the reactor were in automatic control, the control rods would be inserted at the maximum rate following the turbine trip, and the resulting transient would not be limiting in terms of peak RCS pressure.

Transient results (see figure 15.2-28) show the core heat flux, pressurizer pressure, core average temperature, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator water level rises until the feedwater addition is terminated as a result of the high-high steam generator water level trip. The analysis demonstrates that the DNB design basis is met.

Since the power level rises during this event, the fuel temperature will also rise until the reactor trip occurs. The core heat flux lags behind the neutron flux due to the fuel rod thermal time constant and, as a result, the peak core heat flux value does not exceed 118 percent of nominal. Thus, the peak fuel temperature will remain well below the fuel melting point.

The calculated sequence of events is shown in table 15.2-1. The transient results show that DNB does not occur at any time during the feedwater flow increase transient; thus, the ability of the primary coolant to remove heat from the fuel rods is not reduced. Therefore, the fuel cladding temperature does not rise significantly above its initial value during the transient.

15.2.10.3 <u>Conclusions</u>

The decrease in feedwater temperature transient due to an opening of a heater bypass valve is less severe than the excessive load increase event (see subsection 15.2.11). Based on the results presented in subsection 15.2.11, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

For the excessive feedwater addition at power transient, the results show that the DNB ratios encountered are above the safety analysis limit value; hence, no fuel damage is predicted. Additionally, it has been shown that the reactivity insertion rate which occurs at no-load conditions following an excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawn from a subcritical condition event.

15.2.11 EXCESSIVE LOAD INCREASE INCIDENT

15.2.11.1 <u>Identification of Causes and Accident Description</u>

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The

RCS is designed to accommodate a 10-percent step-load increase or a 5-percent per minute ramp-load increase in the range of 15 to 100 percent of full power, taking credit for all control systems in automatic. Any loading rate in excess of these values may cause a reactor trip actuated by the RPS.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control. For excessive loading by the operator or by system demand, the turbine load limiter function in the DEH control system keeps maximum turbine load demand at 100-percent rated load.

During power operation, steam dump to the condenser is controlled by comparing the RCS temperature (median T_{avg}) to a reference temperature based on turbine power, where a high temperature difference in conjunction with a loss of load or a plant trip indicates a need for steam dump. A single controller or control signal malfunction does not cause steam dump valves to open. Interlocks are provided to block the opening of the valves unless a large turbine load decrease or a plant trip has occurred. In addition, the reference temperature and loss of load signals are developed by independent sensors.

Protection against an excessive load increase accident is provided by the following RPS signals:

A. OPΔT.

B. OTΔT.

C. Power range high neutron flux.

D. Low pressurizer pressure.

15.2.11.2 <u>Analysis of Effects and Consequences</u>

15.2.11.2.1 Method of Analysis

Four cases are analyzed to demonstrate the plant behavior following a 10-percent step-load increase from rated load. These cases are as follows:

A. Manually-controlled reactor with BOL (minimum moderator) reactivity feedback.

B. Manually-controlled reactor with EOL (maximum moderator) reactivity feedback.

C. Reactor in automatic control with BOL (minimum moderator) reactivity feedback.

D. Reactor in automatic control with EOL (maximum moderator) reactivity feedback.

This accident is analyzed using the LOFTRAN[4] code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generators, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

At BOL minimum moderator feedback, the core has the least-negative moderator temperature coefficient of reactivity and the least-negative Doppler only power coefficient curve, and therefore, the least-inherent transient response capability. Since a positive moderator temperature coefficient would provide a transient benefit, a zero moderator temperature coefficient was assumed in the minimum feedback cases. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its most-negative value and the most-negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

Normal reactor control systems and engineered safety systems are not required to function. A conservative limit on the turbine valve opening is assumed. The analysis does not take credit for the operation of the pressurizer heaters. The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required.

The RPS is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure in any system or component required for mitigation will adversely affect the consequences of this accident.

This accident is analyzed with the RTDP as described in WCAP-11397-P-A (reference 5). Initial reactor power, RCS pressure, and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in reference 5.

15.2.11.2.2 Results

Figures 15.2-29 through 15.2-32 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL case, there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases (after a slight decrease) above its initial value. For the EOL manually-controlled case, there is a much larger increase in reactor power due to the moderator feedback. A minimum DNBR is reached does not violate the DNB design basis.

Figures 15.2-33 through 15.2-36 illustrate the transient assuming the reactor is in the automatic control mode. Both the BOL and the EOL cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. The minimum DNBR for the BOL case and for the EOL case does not violate the DNB design basis.

The calculated time sequence of events for the excessive load increase event is shown on table 15.2-1. Note that a reactor trip signal was not generated for any of the four cases.

15.2.11.3 Conclusions

It has been demonstrated that for an excessive load increase, the minimum DNBR during the transient will not go below the safety analysis limit value and thus will neither affect fuel cladding integrity nor result in the release of fission products to the RCS.

15.2.12 ACCIDENTAL DEPRESSURIZATION OF THE RCS

15.2.12.1 Identification of Causes and Accident Description

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a pressurizer safety valve is sized to relieve approximately twice the steam flowrate of a relief valve and will allow a much more rapid depressurization upon opening, the most-severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot leg saturation conditions without RPS intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably; however, the pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via the moderator density feedback; however, if the plant is in the automatic mode, the rod control system functions to maintain the power essentially constant throughout the initial stages of the transient. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip because of the decreased reactor coolant density.

The reactor will be tripped by the following RPS signals:

A. Pressurizer low pressure.

B. Overtemperature ΔT.

15.2.12.2 Analysis of Effects and Consequences

15.2.12.2.1 Method of Analysis

The accidental depressurization of the RCS is analyzed by the detailed digital computer code LOFTRAN.[4] The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

In calculating the DNBR, the following conservative assumptions are made:

A. The accident is analyzed using the RTDP. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation (see table 15.1-2B). Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397 (reference 5).

B. A most positive moderator temperature coefficient of reactivity (table 15.1-2A) is assumed in order to provide a conservatively high amount of positive reactivity feedback due to changes in the moderator temperature.

C. A small (absolute value) Doppler coefficient of reactivity is assumed, such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator feedback.

D. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. In fact, it should be noted that the power peaking factors are kept constant at their design values, while the void formation and resulting core feedback effects would result in considerable flattening of the power distribution. Although this would significantly increase the calculated DNBR, no credit is taken for this effect.

15.2.12.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in figures 15.2-37 through 15.2-39. Figure 15.2-37 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly until reactor trip occurs on OTΔT. The pressure decay and core average temperature transients following the accident are given in figure 15.2-38. The DNBR decreases initially, but increases rapidly following the reactor trip as shown in figure 15.2-39. The analysis demonstrates that the DNB design basis is met.

The calculated sequence of events is shown in table 15.2-1.

15.2.12.3 <u>Conclusions</u>

The results of the analysis show that the pressurizer low pressure and OTΔT RPS signals provide adequate protection against the RCS depressurization event. Thus, there will be no cladding damage or release of fission products to the RCS.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.13.1 <u>Identification of Causes and Accident Description</u>

The most severe core conditions resulting from an accidental depressurization of the main steam system, which is classified as an ANS Condition II event, result from an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam pipe, which is classified as an ANS Condition IV event, are given in section 15.4.

The steam released as a consequence of this accidental depressurization results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity and subsequent reduction of core shutdown margin.

For an accidental depressurization of the main steam system, the radiation releases must remain within the requirements of 10 CFR Part 20.1 - 20.601. This is the ANSI N18.2 criterion for Condition II events, "Faults of Moderate Frequency." Although the plant may return to criticality, the above limit can be met by showing there is not consequential damage, i.e., that the DNB design basis is met. Therefore, the analysis is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA and a single failure in the engineered safety features (ESF), the limit DNBR value will be met after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, power-operated relief or safety valve.

The following systems provide the necessary protection against an accidental depressurization of the main steam system:

A. Safety injection system (SIS) actuation from any of the following:

    1. Two of three low-pressurizer pressure signals.

    2. High steam line differential pressure.

B. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal.

C. Redundant isolation of the main feedwater lines; sustained high feedwater flow would cause additional cooldown. Therefore, an SI signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and indirectly close the feedwater isolation valves (2/2 steam generator feedwater pump tripped).

15.2.13.2 <u>Analysis of Effects and Consequences</u>

15.2.13.2.1 Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

A. A full plant digital computer simulation using LOFTRAN[4] to determine RCS temperature and pressure during cooldown.

B. An analysis to ascertain that the DNB design basis is met.

The following conditions are assumed to exist at the time of a secondary system depressurization incident:

A. EOL shutdown margin at no load, equilibrium xenon conditions, and with the most reactive RCCA assembly stuck in its fully withdrawn position. Operation of RCCA banks during core burnup is restricted in such a way that addition of positive reactivity due to a secondary system break accident will not lead to a more adverse condition than the case analyzed.

B. A negative moderator coefficient corresponding to the EOL rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature at 1000 lb/in.^{2} corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown on figure 15.2-40, sheet 1. The effect of power generation in the core on overall reactivity is shown in figure 15.2-40, sheet 2.

C. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the SIS. The injection curve assumed is shown in figure 15.2-41. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. No credit has been taken for the low-concentration boric acid which must be swept from the SI lines downstream of the RWST prior to the delivery of concentrated boric acid (2300 ppm from the RWST to the reactor coolant loops.

D. The case studied consists of a steam flow of 224.3 lb/s at 1004 psia from one steam generator with offsite power available. This is the calculated maximum capacity of any single steam dump or safety valve. Initial hot standby conditions with minimum required shutdown margin at no load T_{avg} are assumed since this represents the most conservative initial condition. The conclusions of this case are valid for the limiting steam flow from any single steam dump or safety valve as specified in tables 10.3-1 and 10.3-2 (i.e., 890,000 lb/h at 1085 psig) since this case is less limiting than the rupture of a main steam pipe case presented in section 15.4 which also satisfies the acceptance criteria for accidental depressurization of the main steam system.

E. In computing the steam flow, the Moody Curve for $\frac{fL}{D} = 0$ is used.

F. Perfect moisture separation in the steam generator is assumed.

G. A boric acid solution of 0 ppm in the high head injection lines and the equivalent volume of the boron injection tank (BIT), which has been deleted, is assumed.

15.2.13.2.2 Results

Since it is postulated that all of the conditions described above occur simultaneously, the results presented are a conservative indication of the events which would occur ssuming a secondary system steam release.

Figure 15.2-42 shows the transients arising as the result of a steam release having an initial steam flow of 224.3 lb/s at 1004 psia with steam release from one steam generator. The assumed steam release is that for the maximum capacity of any single steam dump or safety valve. In this case, SI is initiated automatically by low-pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 2300 ppm enters the RCS from the RWST, providing sufficient negative reactivity to prevent core damage. The calculated transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 min, the neglected stored energy will have a significant effect in slowing the cooldown.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot standby condition through control of auxiliary feedwater flow and SI flow, as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow, and to control steam generator level and RCS coolant temperature using the AFW system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 min following SI actuation.

The calculated time sequence of events for this accident is listed in table 15.2-1.

15.2.13.3 Conclusions

The analysis has shown that the criterion stated earlier in this section is satisfied. For an accidental depressurization of the main steam system, (where the boron concentration in the high head injection line and the equivalent volume of the BIT, which has been deleted is 0 ppm), the minimum DNBR remains well above the limiting value and no system design limits are exceeded. This case is less limiting than the rupture of a main steam pipe case presented in section 15.4.

15.2.14 INADVERTENT OPERATION OF ECCS DURING POWER OPERATION

15.2.14.1 Identification of Causes and Accident Description

Inadvertent operation of the ECCS at power could be caused by operator error, test sequence error, or a false electrical actuation signal. A spurious signal initiated after the logic circuitry in one solid-state protection system train for any of the following Engineered Safety Feature (ESF) functions could cause this incident by actuating the ESF equipment associated with the affected train:

A. High containment pressure.

B. Low pressurizer pressure.

C. High steam line differential pressure.

D. Low steam line pressure.

Following the actuation signal, the suction of the coolant charging pumps diverts from the volume control tank (VCT) to the RWST. Simultaneously, the valves isolating the high head injection lines from the charging pumps automatically open and the normal charging line isolation valves close. The charging pumps force the borated water from the RWST through the pump discharge header, the injection line, and into the cold leg of each loop. The passive accumulator tank SI and low-head system are available; however, they do not provide flow when the RCS is at normal pressure.

An SI signal normally results in a direct reactor trip and a turbine trip; however, any single fault that actuates the ECCS will not necessarily produce a reactor trip. If an SI signal generates a reactor trip, the operator should determine if the signal is spurious. If the SI signal is determined to be spurious, the operator should terminate SI and maintain the plant in the hot-standby condition as determined by appropriate recovery procedures. If repair of the ESF actuation system instrumentation is necessary, future plant operation will be in accordance with the technical specifications.

If the RPS does not produce an immediate trip as a result of the spurious SI signal, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved out of the core. The transient is eventually terminated by the RPS low pressurizer pressure trip or by manual trip.

The time to trip is affected by initial operating conditions. These initial conditions include the core burnup history which affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

15.2.14.2 Analysis of Effects and Consequences

15.2.14.2.1 Method of Analysis

Inadvertent operation of the ECCS is analyzed using the LOFTRAN[4] computer code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generators, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and,

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

It is easy to conclude that criterion (c) is met if it can be demonstrated that the pressurizer does not become water solid in the minimum allowable operator action time. However, if ECCS flow is not terminated before the pressurizer becomes water solid, it is more difficult to demonstrate that this Condition II event does not lead to a more serious plant condition.

ANS 51.1/N18.2-1973 (reference 11), lists Example (15) of a Condition II event as a “minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only.” In reference 11, normal makeup systems are defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power. Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak). Therefore, the above example of a Condition II event is met provided “orderly reactor shutdown” is also met.

To ensure “orderly reactor shutdown” can occur, the RCS pressure boundary must ultimately be isolatable once the source of the ECCS flow is terminated. To ensure the RCS pressure boundary can be isolated, the pressurizer safety relief valves (PSRVs) must function as designed and the power-operated relief and/or block valves must be available to the operator (after the minimum allowable operator action time) to provide isolation functions.

The capability of the PSRVs to function properly following the discharge of significantly subcooled water through the PSRVs has not been demonstrated and, therefore, is not certain. Hence, for continued conservatism in the safety analysis methodology, it is assumed the PSRVs must not pass water in order to ensure their integrity and continued availability. With one or more PORVs available, the PSRV setpoint will not be reached.

Any water discharge from the RCS would be through the PORV(s). Isolation of the RCS following operator action to terminate ECCS flow would then be obtainable via the PORV block valves(s).

Therefore, to address criterion (c), the analysis uses the criterion that a water-solid pressurizer condition be precluded when the pressurizer is at or above the set pressure of the PSRVs. For the potential condition of the plant operating with all the PORVs blocked, either action to terminate the ECCS flow to avert a water-solid condition or to confirm that at least one PORV is unblocked and available for water relief prior to reaching a water-solid condition must be taken. This addresses any concerns regarding subcooled water relief through the plant PSRVs. Should water relief through the pressurizer PORVs occur, the PORV block valves would be available to isolate the RCS.

The Inadvertent ECCS Action at Power event is analyzed to determine both the minimum DNBR value and maximum pressurizer water volume (or minimum time to a pressurizer water-solid condition). The most limiting case with respect to DNB is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

For maximizing the potential for pressurizer filling, the most limiting case is a maximum reactivity feedback condition with an immediate reactor trip, and subsequent turbine trip, on the initiating SI signal. The transient results are presented for each case.

The analysis assumptions are as follows:

A. Initial Operating Conditions

The DNB case is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A (reference 5). Initial reactor power, RCS pressure, and temperature are assumed to be at the nominal full-power values (see table 15.1-2B). Uncertainties in initial conditions are included in the limit DNBR as described in reference 5.

For the pressurizer filling case, initial conditions (see table 15.1-2B) with maximum uncertainties on power (+2%), vessel average temperature (-6°F), and pressurizer pressure (-50 psia) are assumed.

B. Moderator and Doppler Coefficients of Reactivity

The minimum feedback case (DNB) assumes a zero (0 pcm/°F) moderator temperature coefficient and a low absolute value Doppler power coefficient at BOL. The maximum feedback case (pressurizer filling) assumes a most-negative moderator temperature coefficient and a most-negative Doppler power coefficient representative of EOL Conditions.

C. Reactor Control

For the DNB case (without direct reactor trip on SI) the reactor is assumed to be in manual rod control. In the case of the pressurizer filling scenario, the reactor is assumed to trip at the time of the SI signal. Thus, the reactor control mode is of no consequence.

D. Pressurizer Pressure Control

Pressurizer spray is assumed available for each case in order to minimize the RCS pressure. In the pressurizer filling case, minimizing the RCS pressure conservatively maximizes the incoming SI flow. For the DNB case, the pressurizer heaters are assumed to be inoperable. This assumption yields a more rapid pressure decrease for the DNB case, which is conservative. For the pressurizer filling case, the pressurizer heaters are assumed operable since this maximizes the heat addition to the pressurizer water, thus maximizing the fluid expansion, resulting in an earlier time to pressurizer filling.

For the DNB case, the PORVs are assumed to be operable to conservatively minimize the RCS pressure. PORVs are not assumed as an automatic pressure control function for the pressurizer filling case. Automatic pressure control operation with one or more PORVs would preclude the pressurizer pressure from reaching the PSRV set-pressure and, hence, preclude water discharge through the PSRVs. If one or more PORVs are available and water relief through a PORV occurs, operator action to manually block the PORV (after the operator terminates the ECCS flow) ensures the integrity of the RCS pressure boundary is maintained. Also, permissive P-11 automatically interlocks the PORVs closed on decreasing pressure.

E. Boron Injection

At the initiation of the event, two charging pumps inject borated water into the cold leg of each loop. The analysis assumes zero injection line purge volume for calculational simplicity; thus, the boration transient begins immediately in the analysis.

F. Turbine Load

For the DNB case (without direct reactor trip/turbine trip on SI), the turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops. In the case of pressurizer filling, the reactor and turbine both trip at the time of SI actuation with the turbine load dropping to zero simultaneously.

G. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal for the DNB case. The pressurizer filling case assumes an immediate reactor trip on the initiating SI signal.

H. Decay Heat

The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip), whereas in the pressurizer filling case, the availability of decay heat and its expansion effects on the RCS liquid volume have been taken into account.

Core residual heat generation is based on the 1979 version of ANS 5.1 (reference 9). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

I. Operator Action Time

The PSRVs must not be exposed to subcooled liquid discharge as a result of reaching a water solid pressurizer condition. Consequently, PORV availability must be assured by manually opening a block valve to allow a PORV to actuate on demand. Per ANSI/ANS-58.8-1984 (reference 10), the operator action times for event indication are based on specific time tests. Inadvertent ECCS Actuation at Power is a Condition II event per ANSI N18.2-1973 which relates to a Plant Condition II event per ANSI/ANS-58.8-1984. For a Plant Condition II event time test 1 requires 5 min and time test 2 requires 1 + n * 1 min where "n" signifies the number of discrete manipulations required. PORVs would be expected to be available unless they were blocked due to a leaking PORV condition. Therefore, any operator action associated with assuring PORV availability consists of manually opening a block valve to allow it to actuate on demand. The appropriate time to assume initial operator action is 7 min. This consists of 5 min to evaluate the incident and decide upon corrective measures plus 1-min fixed time delay to receive simple readout information, i.e., status of PORV block valves, and 1 min to begin the appropriate action.

J. Pressurizer Safety Valves

The safety valves are conservatively assumed to open at a pressure of 2425 psia which corresponds to a tolerance of -3% relative to the set pressure of 2500 psia. The valves are assumed to close at a pressure of 2300 psia, which corresponds to a blowdown of 5% below the opening pressure of 2425 psia.

15.2.14.2.2 Results

The transient responses for the DNB and pressurizer filling cases are shown in figures 15.2-43 through 15.2-45. Table 15.2-1 shows the calculated sequence of events.

DNB Case:

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

Pressurizer Filling Case:

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases throughout the transient. At 7 min, the analysis assumes that the operatorf takes action to open a PORV (i.e., opens PORV block valve). A 40.0-s delay is assumed from initial operator action until the time one PORV is fully open. At this point in the transient, the operational PORV begins relieving water and steam from the pressurizer. This occurs prior to the pressurizer reaching a water-solid condition. Pressurizer pressure never rises above the PSV setpoint during the transient. Thus the analysis demonstrates that water relief through the PSVs is precluded.

15.2.14.3 <u>Conclusions</u>

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, the pressurizer will not reach a water-solid condition prior to the operator opening a PORV block valve and PORV and the RCS pressure dropping to the pressure where the PSRVs reseat, thereby precluding water relief through the PSVs.

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1. Gangloff, W. C., "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized-Water Reactors," WCAP-7468, May 1971.

2. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-A (Proprietary) and WCAP-8028-A (Nonproprietary), January 1975.

3. Hargrove, H. G., "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

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8. 90AP*-G-0041, Letter from D. E. McKinnon to L. K. Mathews, "Joseph M. Farley Nuclear Plant, Units 1 and 2 Operating Procedure for Mode 4/5 Boron Dilution," with attachments, July 6, 1990.

9. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors," August 29, 1979.

10. ANSI/ANS-58.8-1984, “Time Response Design Criteria for Nuclear Safety Related Operator Actions.”

11. ANS-51.1/N18.2-1973, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

12. Letter from C. O. Thomas (USNRC) to T. W. Schnatz (UGRA), “Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, RETRAN-A Program for One Dimensional Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems,” and C. E. Peterson, et al., “RETRAN-02-A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” EPRI NP-1850-CCM, Rev. 6, December 1995.

13. Letter from A. C. Thandani to W. J. Boatwright (TU Electric), “Acceptance for Use of RETRAN MOD005.0,” November 1, 1991.

14. D. S. Huegel, et al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” WCAP-14882-P (Proprietary), June 1997.

15. Westinghouse Project letter ALA-09-121 dated 11/10/2009.

TABLE 15.2-1 (SHEET 1 OF 6)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (s)
Uncontrolled RCCA withdrawal from a subcritical condition	Initiation of uncontrolled rod withdrawal (78.75 pcm/s) reactivity insertion rate from 10^{-9} fraction of nominal power	0.0
	Power range high neutron flux low setpoint reached	9.6
	Peak nuclear power occurs	9.7
	Rods begin to fall into core	10.1
	Peak heat flux occurs	11.4
	Peak average clad temperature occurs	12.0
	Peak average fuel temperature occurs	12.5
Uncontrolled RCCA bank withdrawal at power (full power with maximum feedback), DNB cases		
Case A	Initiation of uncontrolled RCCA withdrawal at maximum insertion rate (110 pcm/s)	0.0
	Power range high neutron flux high setpoint reached	3.8
	Rods begin to fall into core	4.3
	Minimum DNBR occurs	4.6
Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (20 pcm/s)	0.0
	OTΔT reactor trip setpoint initiated	58.3
	Minimum DNBR occurs	60.1
	Rods begin to fall into core	60.3

TABLE 15.2-1 (SHEET 2 OF 6)

Accident	Event	Time (s)
Uncontrolled RCCA bank withdrawal at power (limiting overpressure case)		
	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (27 pcm/s)	0.0
	High pressurizer pressure reactor trip setpoint reached	10.17
	Rods begin to fall into the core	11.17
	Maximum RCS pressure occurs	13.50
Uncontrolled boron dilution		
Dilution during refueling	Dilution begins	0
	Shutdown margin lost (if dilution continues)	> 1100
Dilution during startup	Power range - low setpoint reactor trip due to dilution	0
	Shutdown margin lost (if dilution continues)	> 900
Dilution during full-power operation		
Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0
	Shutdown margin lost (if dilution continues)	> 900
Manual reactor control	Reactor trip on OTΔT due to dilution	0
	Shutdown margin is lost (if dilution continues)	> 900
Partial loss of forced reactor coolant flow		
All pumps initially in operation, one pump coasting down	One pump begins coasting down	0.0
	Low flow reactor trip	1.6

TABLE 15.2-1 (SHEET 3 OF 6)

Accident	Event	Time (s)
	Rods begin to drop	3.1
	Minimum DNBR occurs	3.9
Loss of external electrical load		
With pressurizer pressure control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	13.3
	Peak RCS pressure occurs	15.0
	Rods begin to drop	14.3
	Initiation of steam release from steam generator safety valves	15.5
	Minimum DNBR occurs	15.0
Loss of external electrical load		
Without pressurizer pressure control (BOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	6.6
	Rods begin to drop	7.6
	Peak RCS pressure occurs	9.9
	Initiation of steam release from steam generator safety valves	11.6
	Minimum DNBR occurs	N/A

TABLE 15.2-1 (SHEET 4 OF 6)

Accident	Event	Time (s)
Loss of normal feedwater	Main feedwater flow stops	20.0
	Low-low steam generator water level reactor trip 19% NRS	75.0
	Rods begin to drop	77.0
	Two motor-driven pumps begin to deliver AFW (350 gpm)	135.0
	Operator action to trip two RCPs	677.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~2400.0
	Peak water level in pressurizer occurs (post reactor trip)	2444.0
Loss of all ac power to the station auxiliaries	Main feedwater flow stops	20.0
	Low-low steam generator water level reactor trip 16% NRS	81.6
	Rods begins to drop	83.6
	ac power is lost and RCPs begin to coast down	85.6
	Two motor-driven pumps powered by diesel generators, begin to deliver AFW (350 gpm)	141.6
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~1800.0
	Peak water level in pressurizer occurs (post reactor trip)	1969.0

TABLE 15.2-1 (SHEET 5 OF 6)

Accident	Event	Time (s)
Excessive feedwater flow at full power	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level signal generated	50.0
	Minimum DNBR occurs	51.5
	Turbine trip occurs due to high-high steam generator water level	52.5
	Reactor trip due to turbine trip (rod motion begins)	53.5
	Feedwater control valves fully closed	57.0
Excessive load increase	10% step-load increase	0.0
Manual reactor control (BOL)	Equilibrium conditions reached (approximate time)	140.0
Manual reactor control (EOL)	10% step-load increase	0.0
	Equilibrium conditions reached (approximate time)	75.0
Automatic reactor control (BOL)	10% step-load increase	0.0
	Equilibrium conditions reached (approximate time)	250.0
Automatic reactor control (EOL)	10% step-load increase	0.0
	Equilibrium conditions reached (approximate time)	75.0

TABLE 15.2-1 (SHEET 6 OF 6)

Accident	Event	Time (s)
Accidental depressurization of the RCS	Inadvertent opening of one RCS safety valve	0.0
	Overtemperature ΔT reactor trip setpoint reached	25.1
	Rods begin to drop	27.1
	Minimum DNBR occurs	27.8
Accidental depressurization of the main steam system	Inadvertent opening of one main steam safety or relief valve	0.0
	Borated water from the RWST reaches the core	252.7
	Pressurizer empties	263.7
	Criticality reached	445.7
Inadvertent operation of ECCS during power operation		
DNBR Case:	SI pumps begin injecting borated water	0.0
	Low pressurizer pressure reactor trip setpoint reached	61.2
	Rods begin to drop	63.2
	Minimum DNBR occurs	(a)
Pressurizer Filling Case:	SI pumps begin injecting borated water, rods begin drop	0.0
	Operator action to confirm one PORV available	420.0
	One PORV is fully open	460.0
	Pressurizer becomes water solid	461.5

a. DNBR does not decrease below its initial value.

TABLE 15.2-2

**SUMMARY OF BORON DILUTION
ANALYSIS RESULTS AND ANALYSIS ASSUMPTIONS**

Mode of Operation	Flowrate Dilution (gal/min)	Active Volume (ft^3)	Operator Action Time (min)
Power operation			
Auto rod control	300	7735	21.2
Manual rod control	300	7735	20.4
Startup	300	7735	22.1
Refueling	300	3290.0	18.4

Other Important Analysis Assumptions

Mode of Operation	Assumed Initial Boron Conc. (ppm)	Assumed Critical Boron Conc. (ppm)	Average Core Coolant Temperature(°F)
Power operation			
Auto rod control	2100	1800	583.2
Manual rod control	2100	1800	583.2
Startup	2100	1800	554.5

TABLE 15-2.3

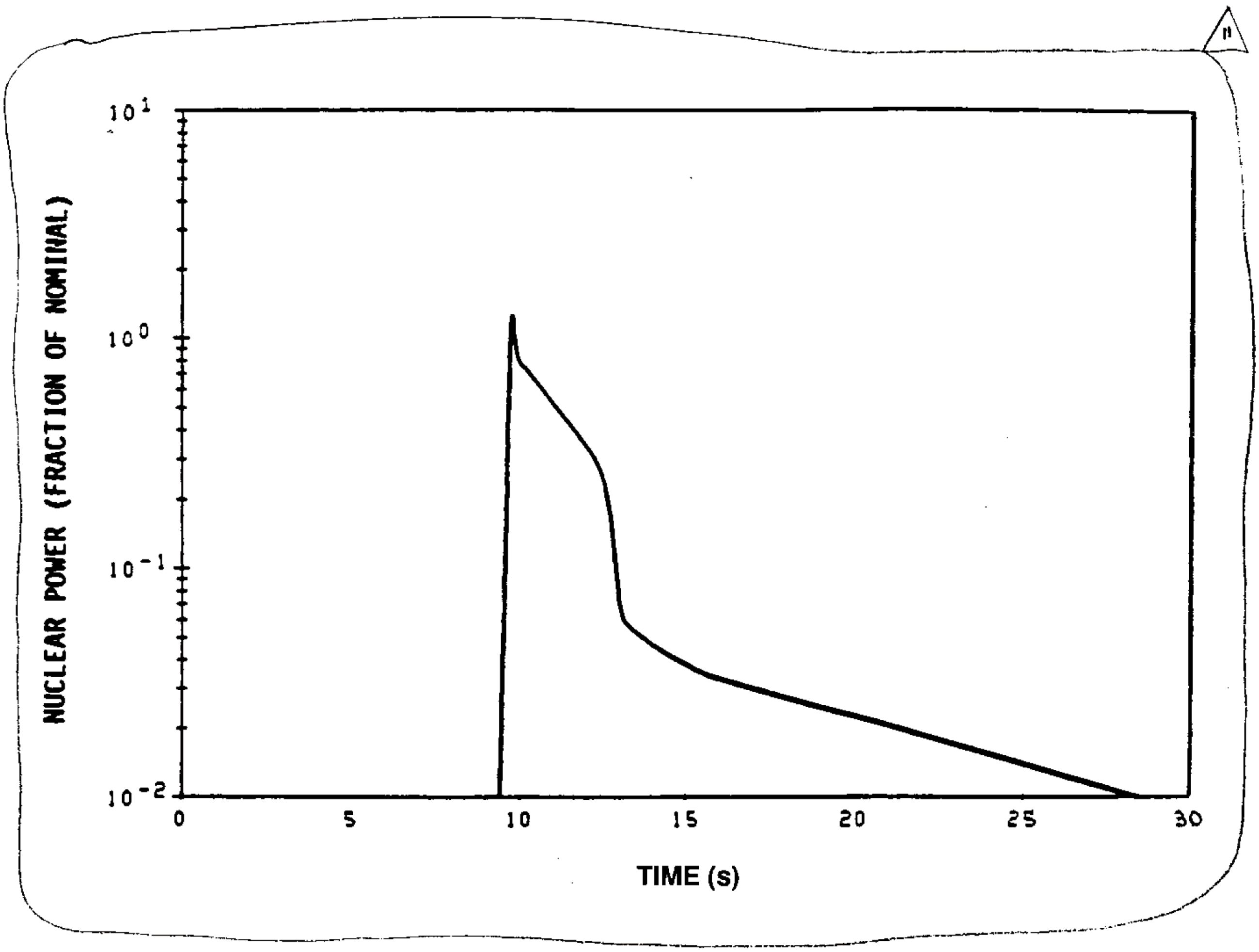
PARAMETERS USED IN LOSS OF ac POWER ANALYSES

Core thermal power	2831 MWt
Steam generator tube leak rate prior to and during accident	1 gpm
Offsite power	Lost
Fuel defects	1%[(a)]
Iodine partition factor in steam generators prior to and during accident	0.1
Secondary side iodine activity	0.1 μCi/gm dose equivalent I_{131}
Duration of plant cooldown by secondary system after accident	8 h
Steam release from three steam generators	538,000 lb (0-2 h) 875,000 lb (2-8 h)
Feedwater flow to three steam generators	728,000 lb (0-2 h) 887,000 lb (2-8 h)
Meteorology	Accident (see appendix 15)

OFFSITE DOSES FROM LOSS OF ac POWER

	Thyroid Dose (Rem)	Whole Body Dose (Rem)	B-skin Dose (Rem)
Site boundary (0-2 hour)	1.2	2×10^{-3}	2×10^{-3}
Low Population Zone (0-8 hour)	0.89	1×10^{-3}	1×10^{-3}

a. A pre-existing iodine spike of 60 μCi/gm dose equivalent I_{131} is assumed.



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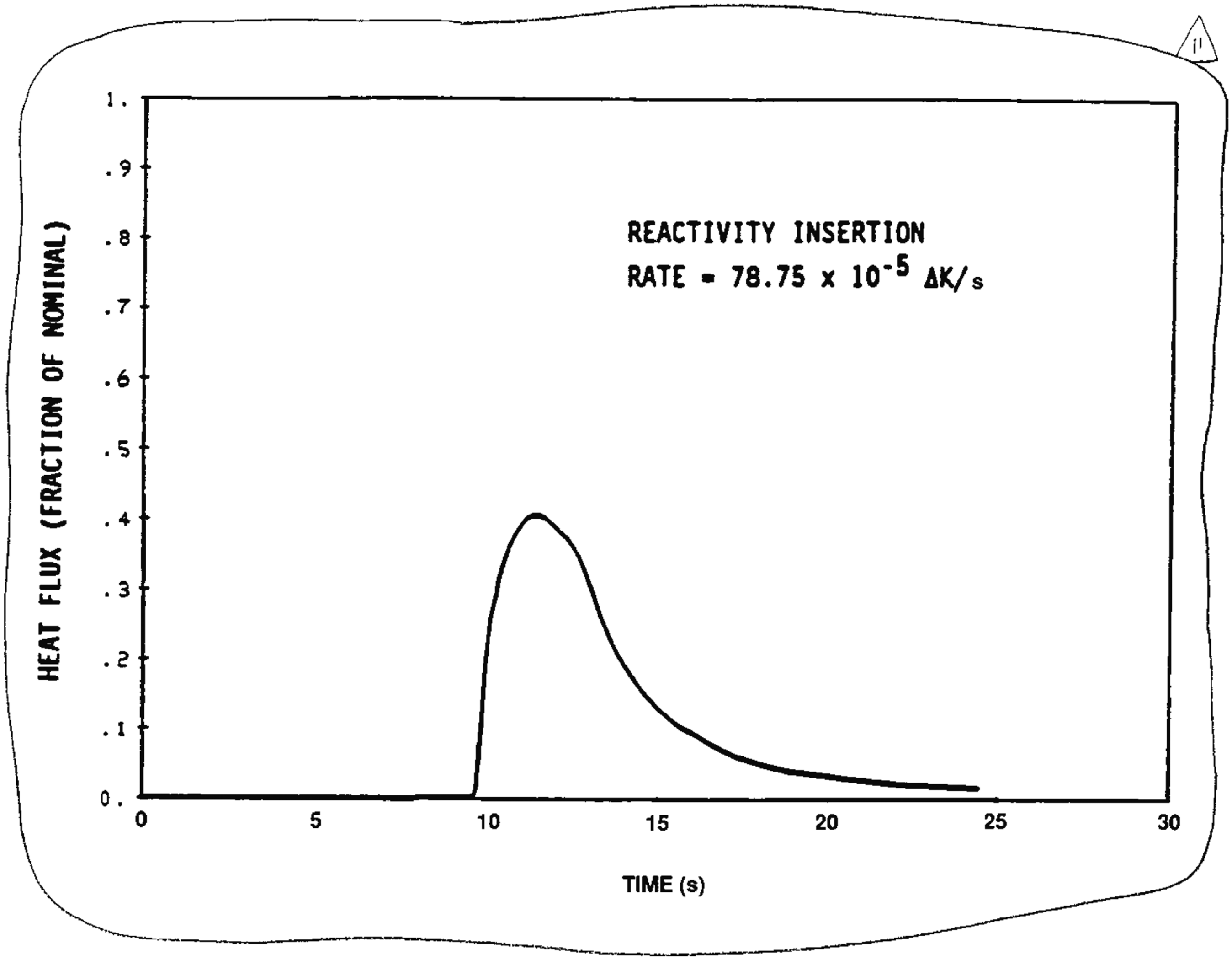
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STARTUP FROM SUBCRITICAL FOR 3-LOOP, 17 X 17 PLANT NUCLEAR POWER
		FIGURE 15.2-1



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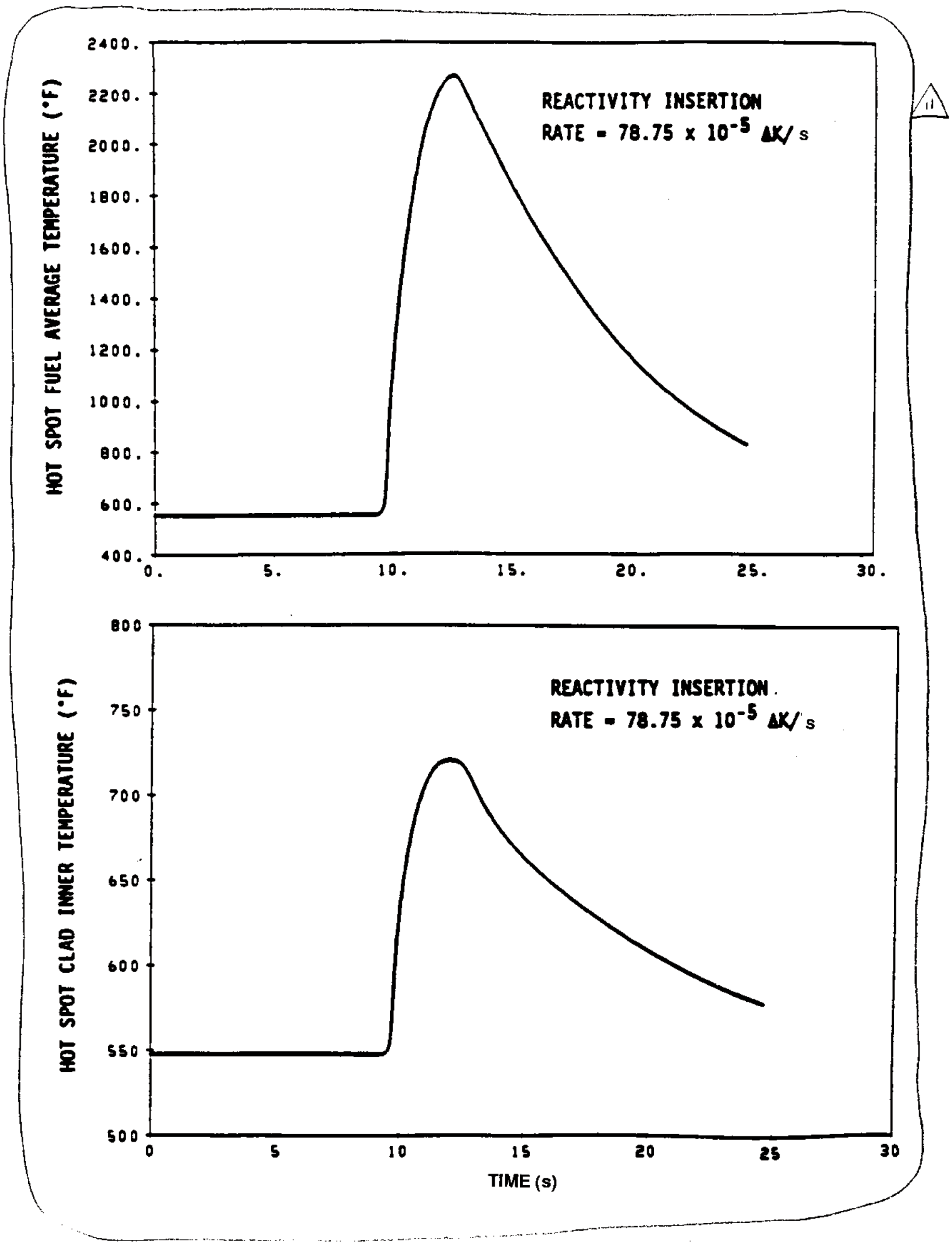
UNIT 1 AND UNIT 2

STARTUP FROM SUBCRITICAL FOR
3-LOOP, 17 X 17 PLANT FACTRAN
HEAT FLUX

FIGURE 15.2-2



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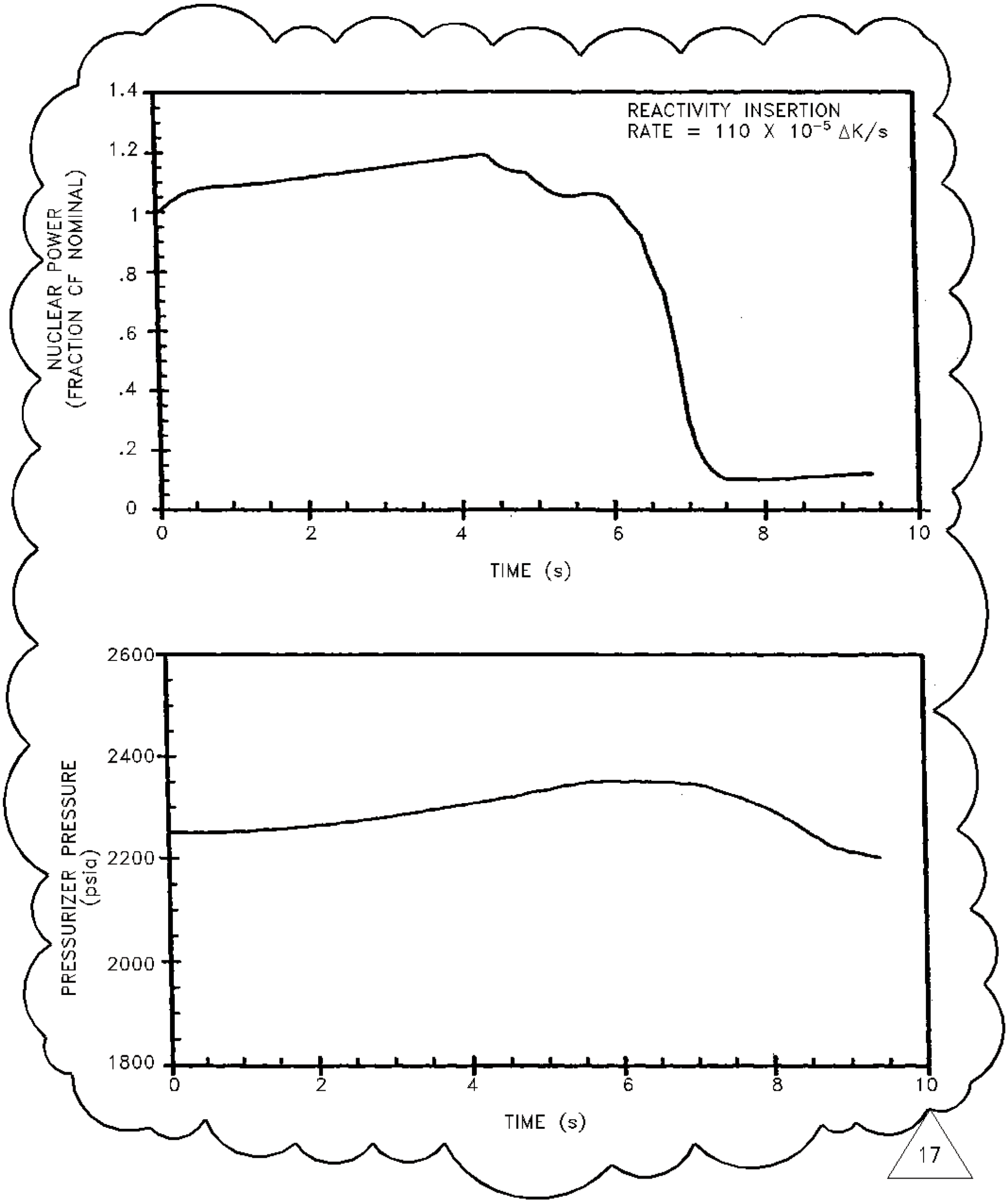
UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION –
TEMPERATURE VERSUS TIME

FIGURE 15.2-3



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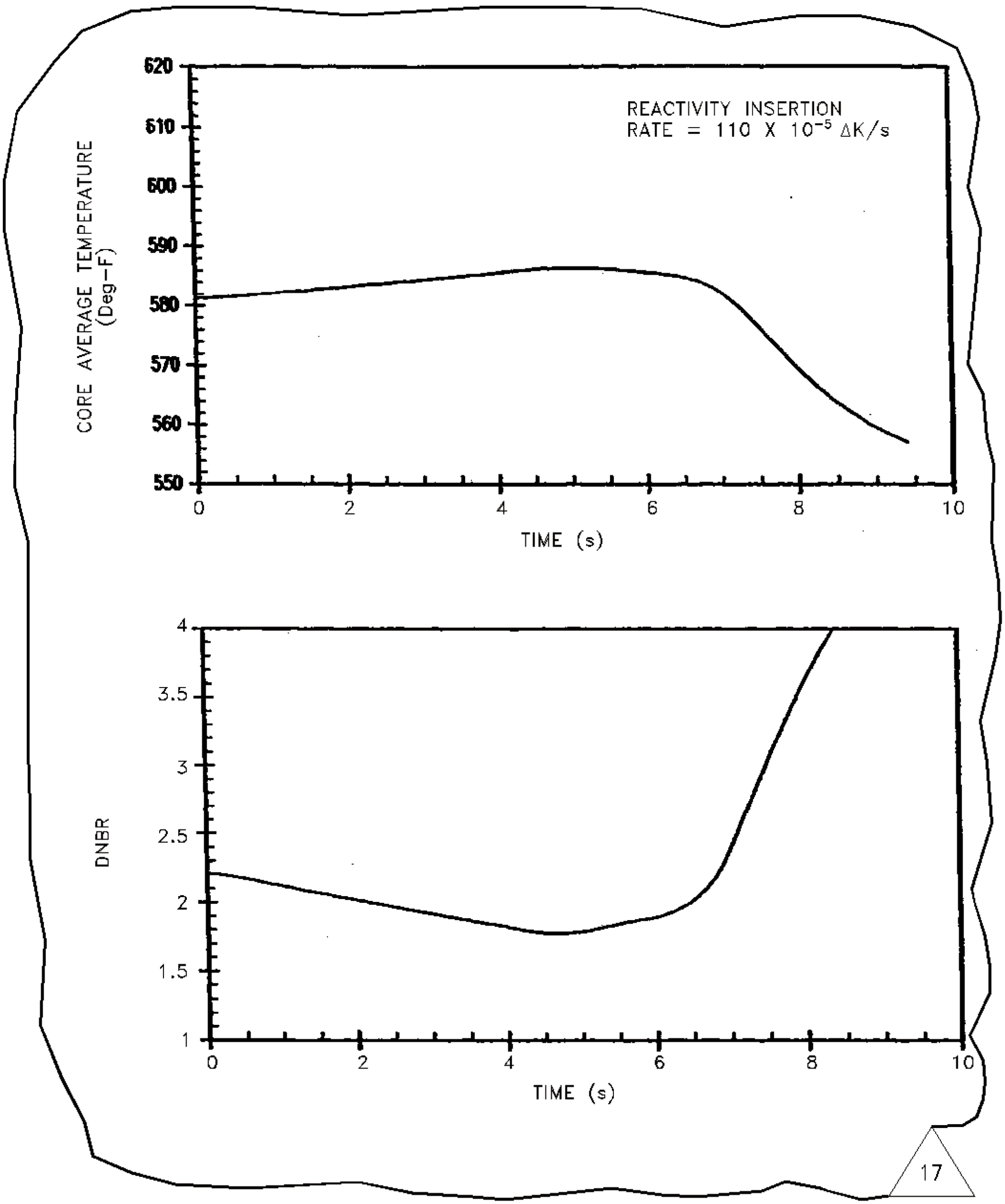
TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY HIGH NEUTRON FLUX TRIP

FIGURE 15.2-4



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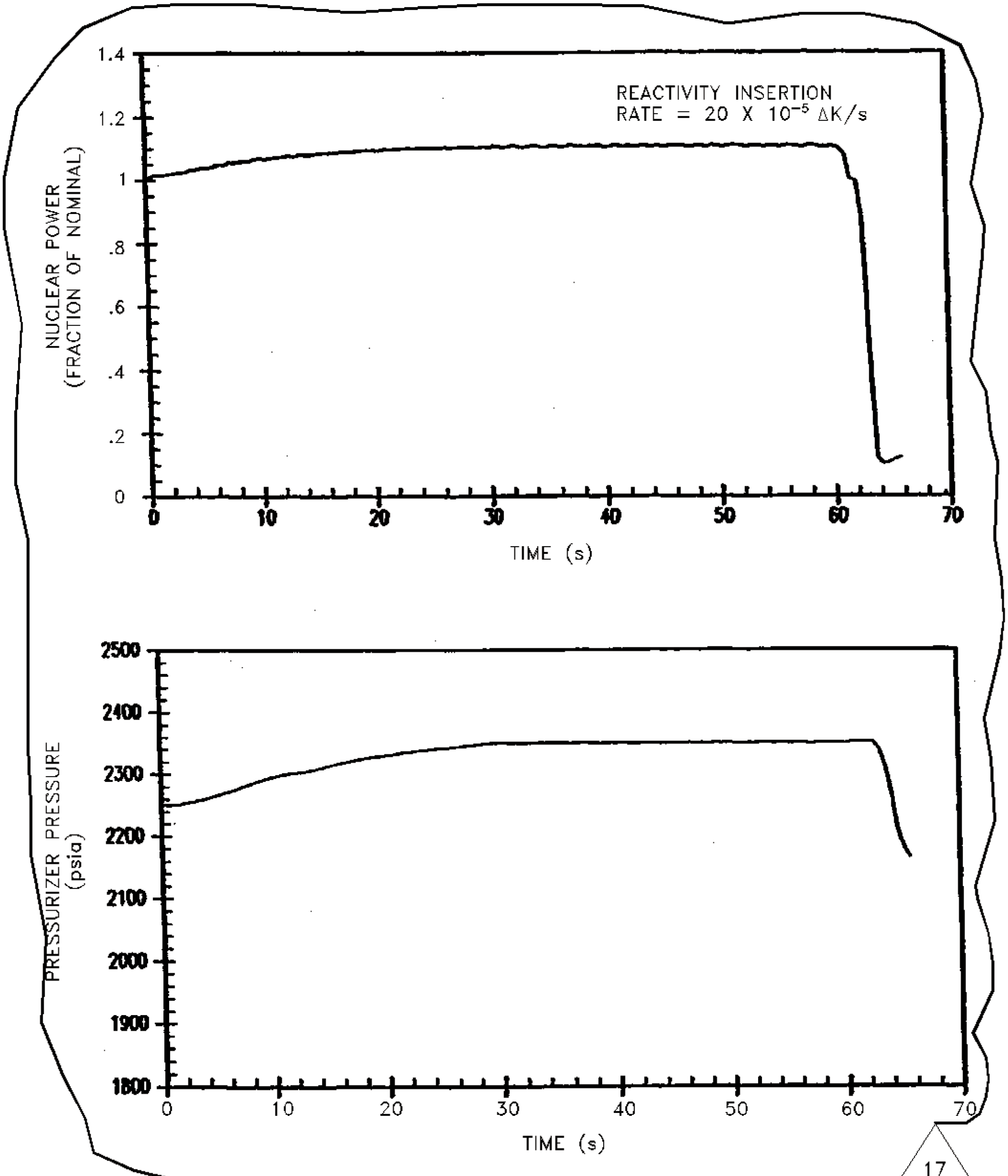
TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY HIGH NEUTRON FLUX TRIP

FIGURE 15.2-5



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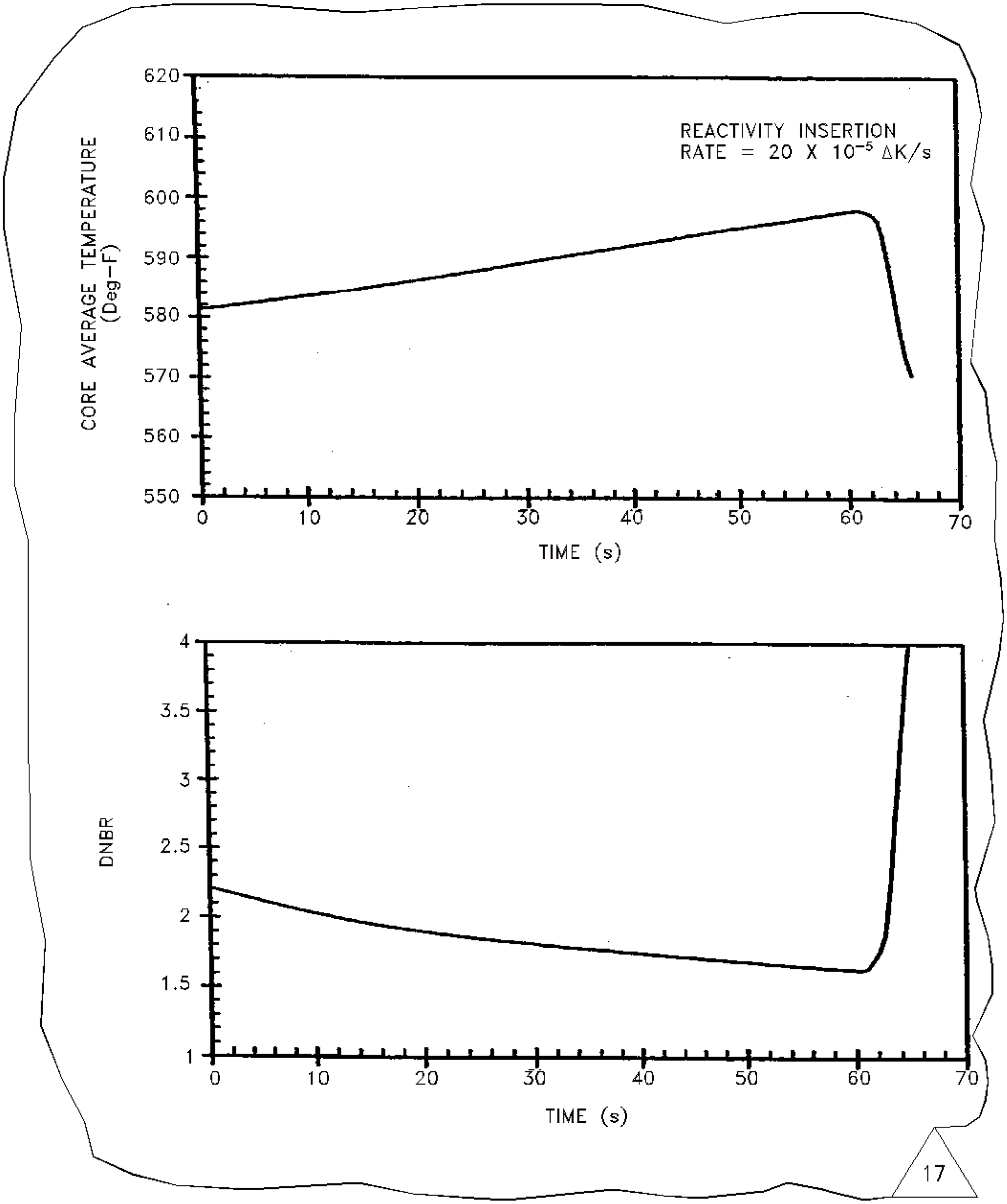
TRANSIENT RESPONSE FOR UNCONTROLLED
ROD WITHDRAWAL FROM FULL POWER
TERMINATED BY OVERTEMPERATURE DELTA-T TRIP

FIGURE 15.2-6



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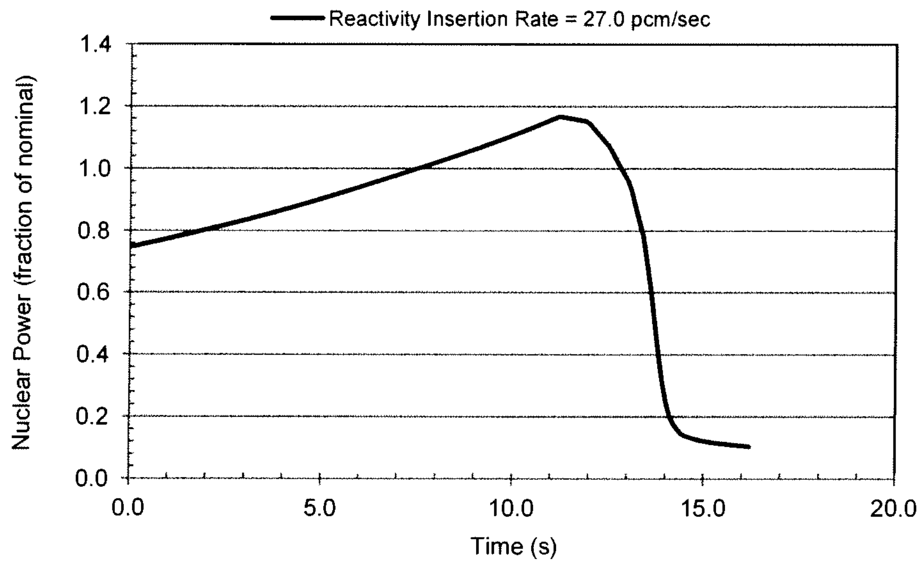
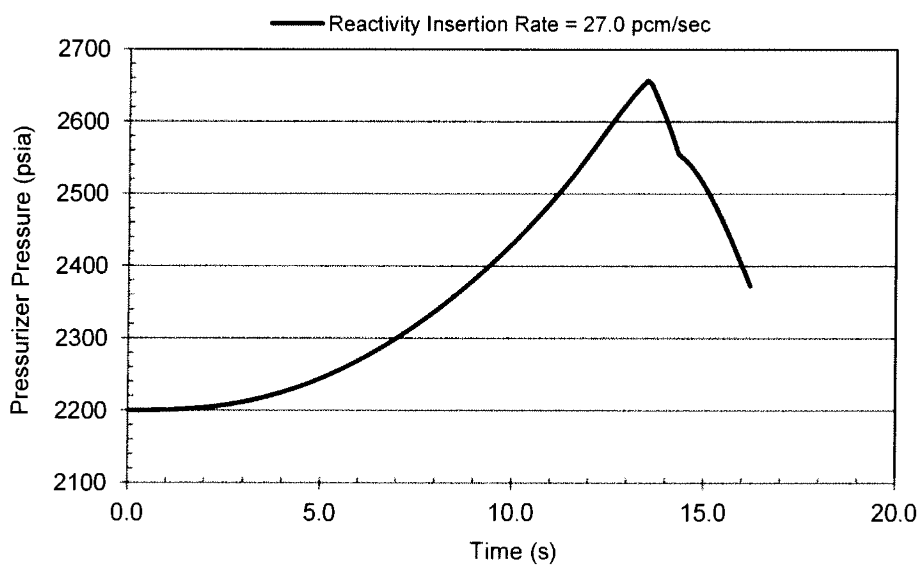
SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY OVERTEMPERATURE DELTA-T TRIP
		FIGURE 15.2-7





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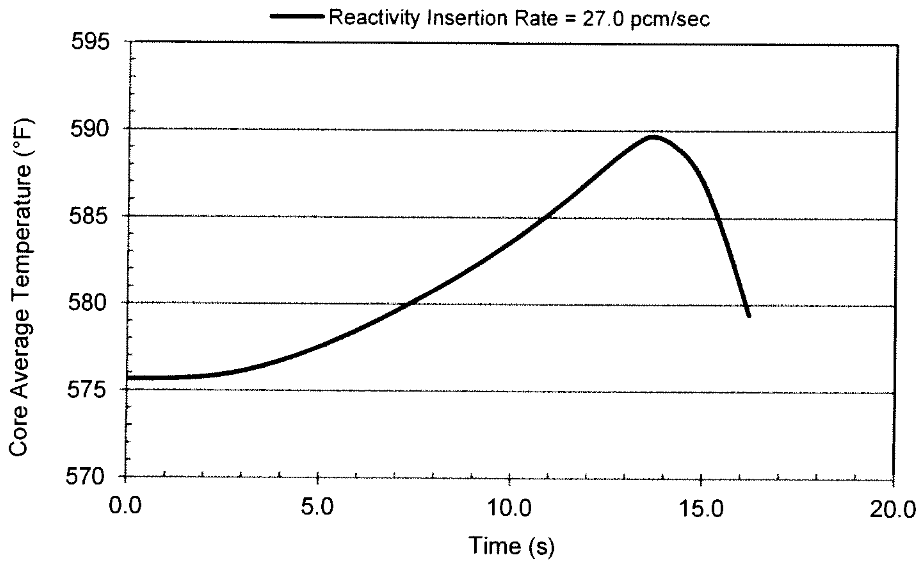
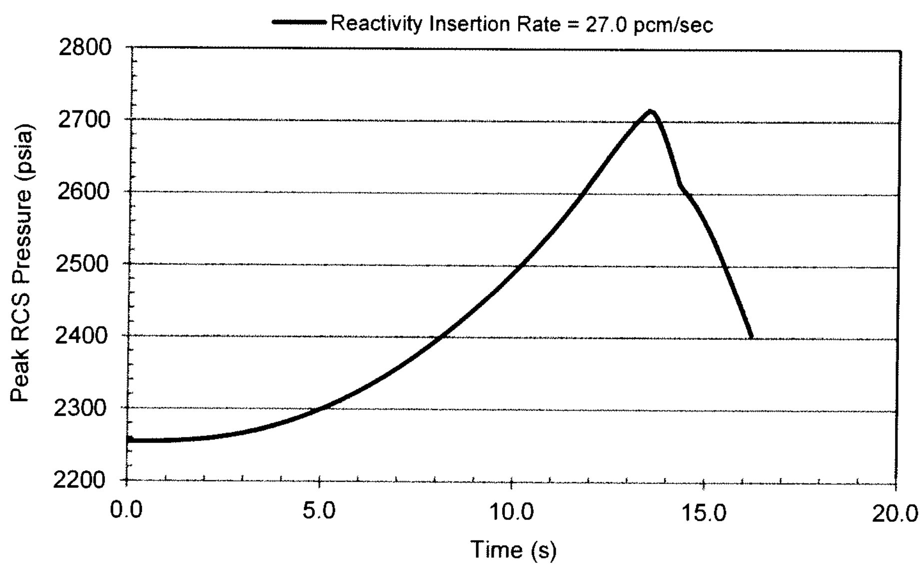
TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM AT POWER CONDITIONS (LIMITING OVERPRESSURE CASE)

FIGURE 15.2-7A





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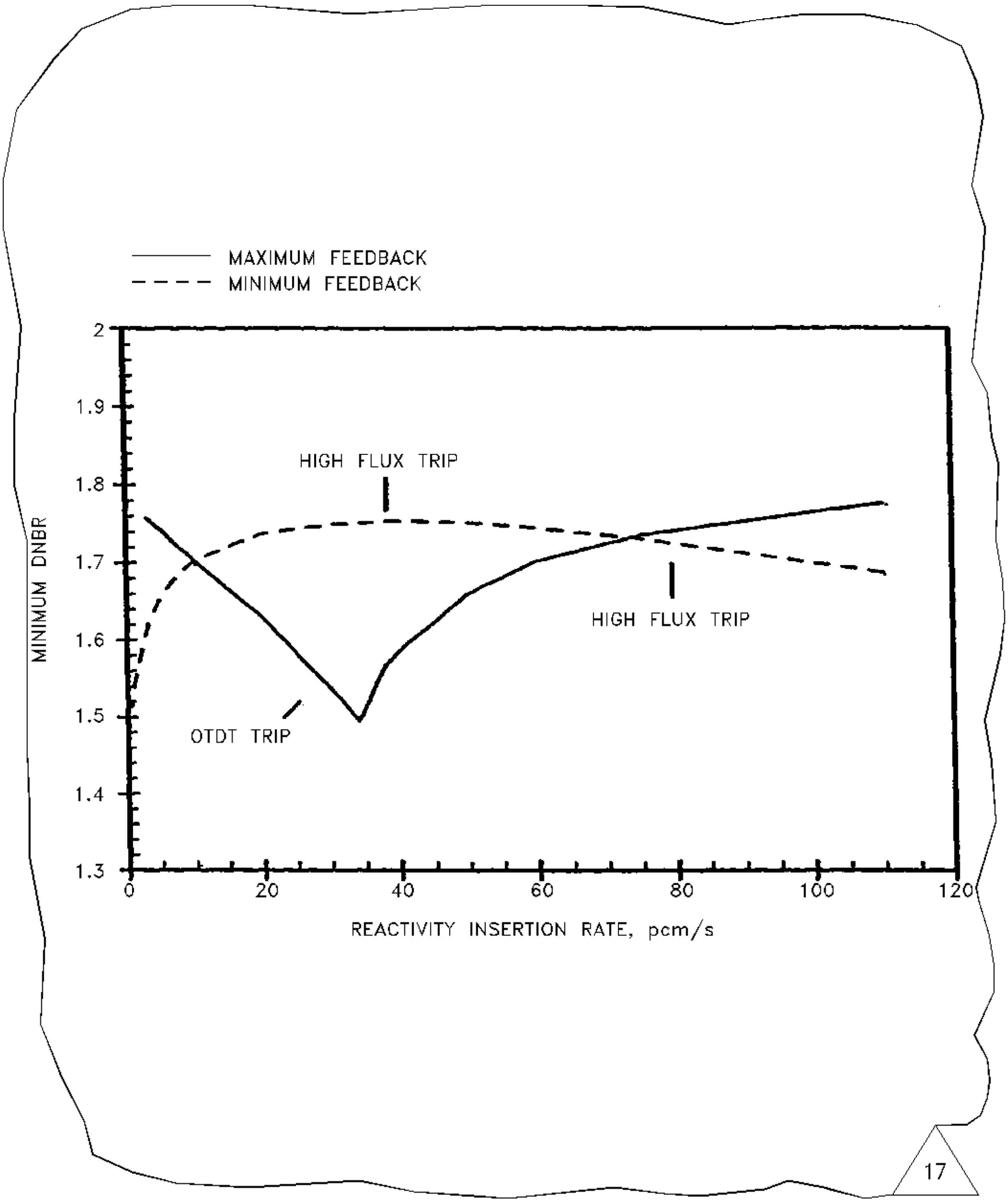
SOUTHERN COMPANY *Energy to Serve Your World®* JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM AT POWER CONDITIONS (LIMITING OVERPRESSURE CASE)
	FIGURE 15.2-7B



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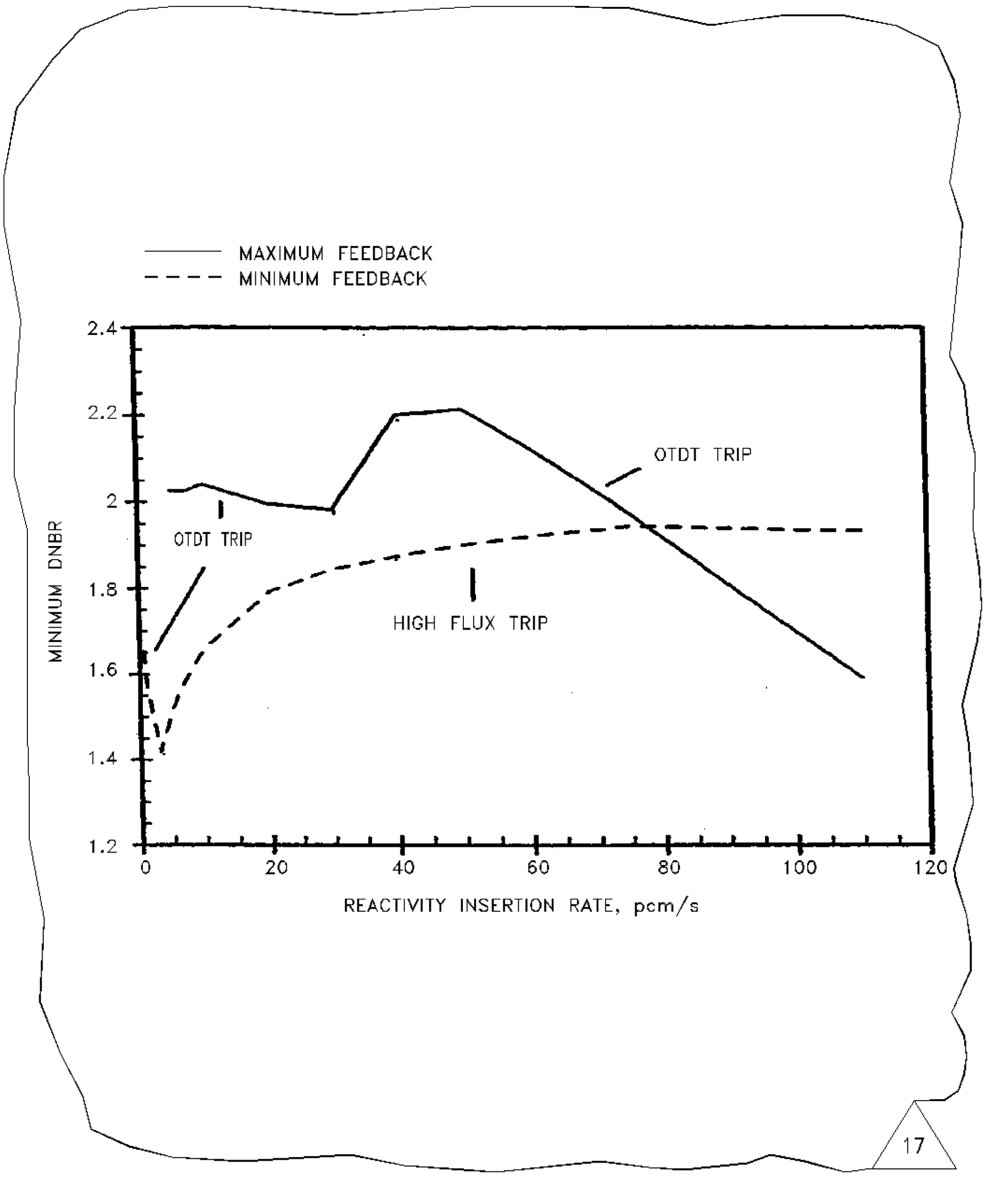
EFFECT OF REACTIVITY INSERTION
RATE ON MINIMUM DNBR FOR A ROD
WITHDRAWAL ACCIDENT FROM 100% POWER

FIGURE 15.2-8



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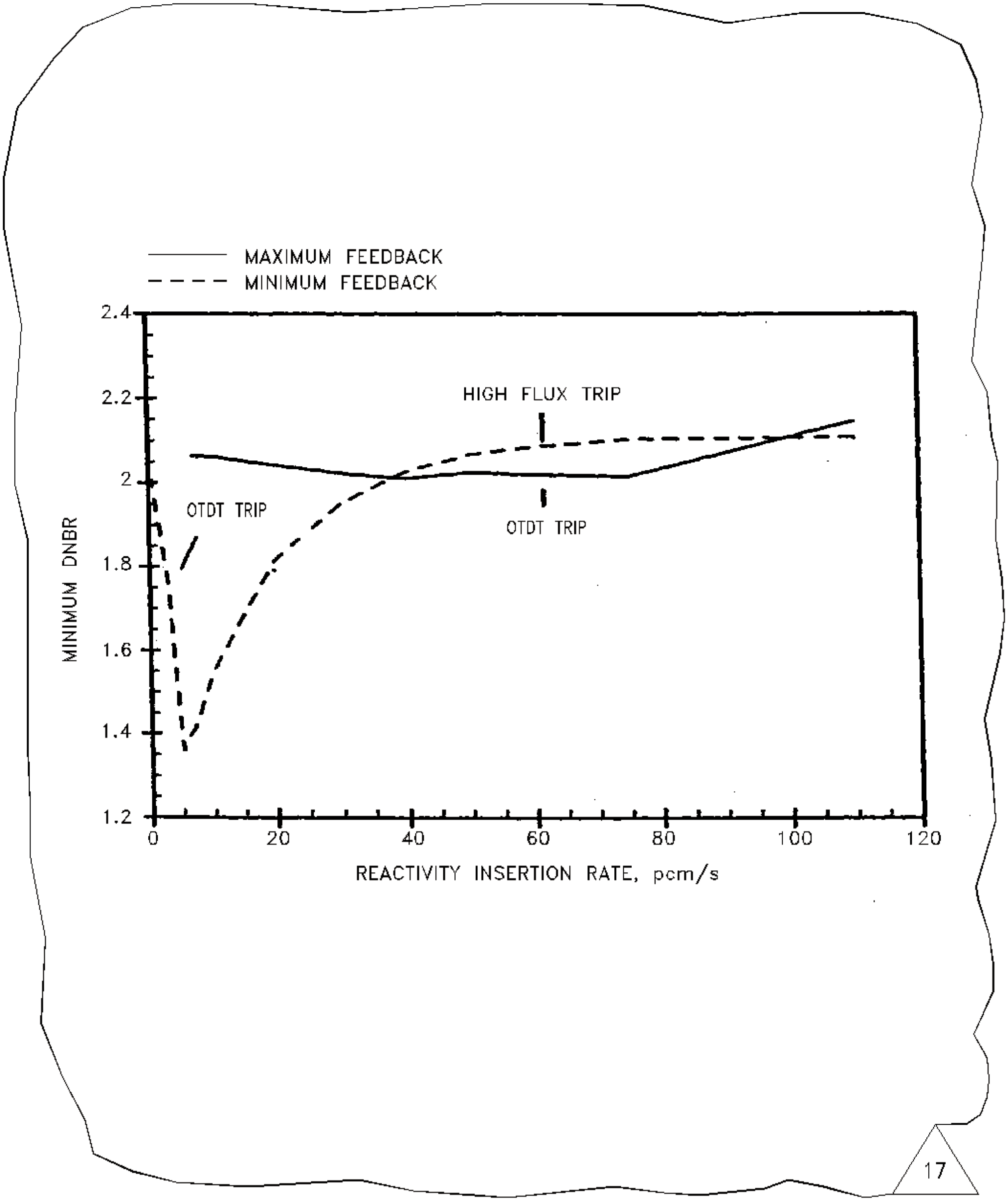
EFFECT OF REACTIVITY INSERTION
RATE ON MINIMUM DNBR FOR A ROD
WITHDRAWAL ACCIDENT FROM 60% POWER

FIGURE 15.2-9



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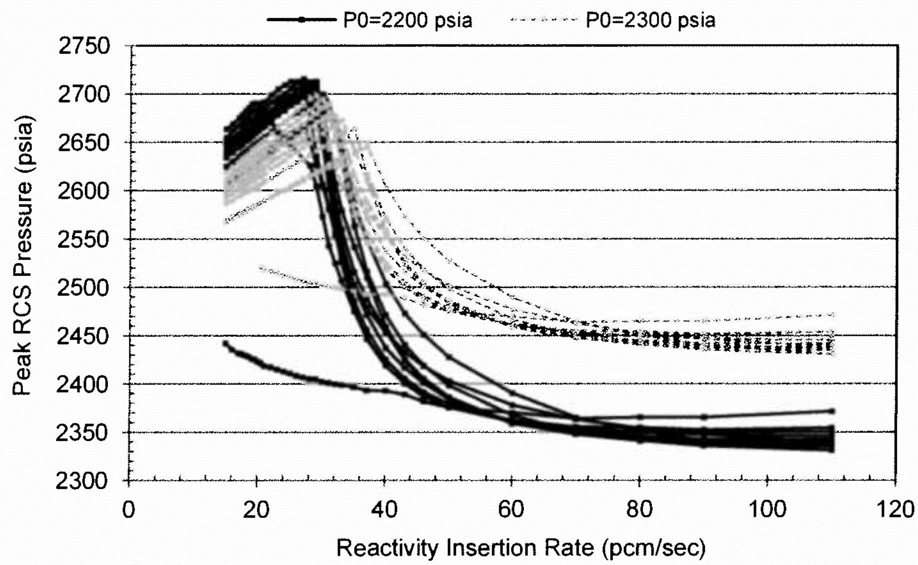
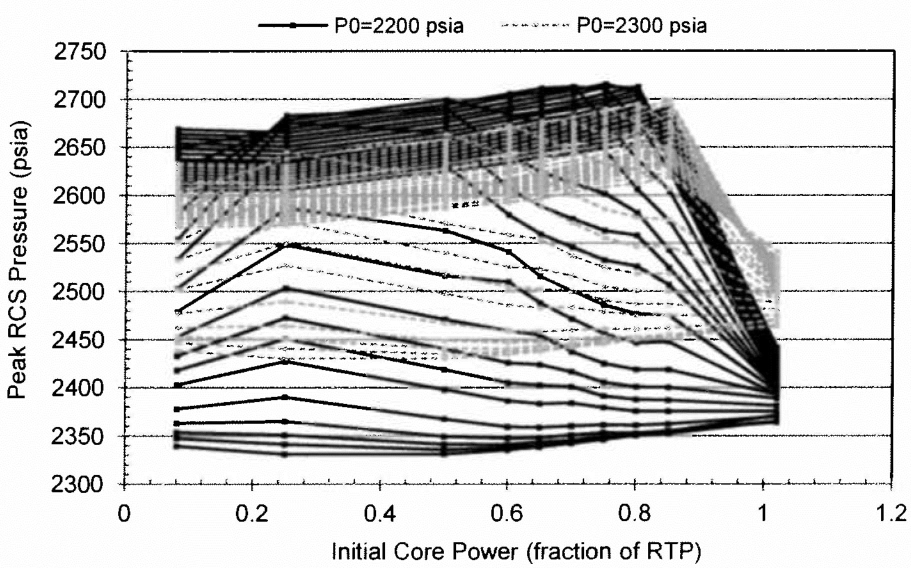
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNBR FOR A ROD WITHDRAWAL ACCIDENT FROM 10% POWER

FIGURE 15.2-10





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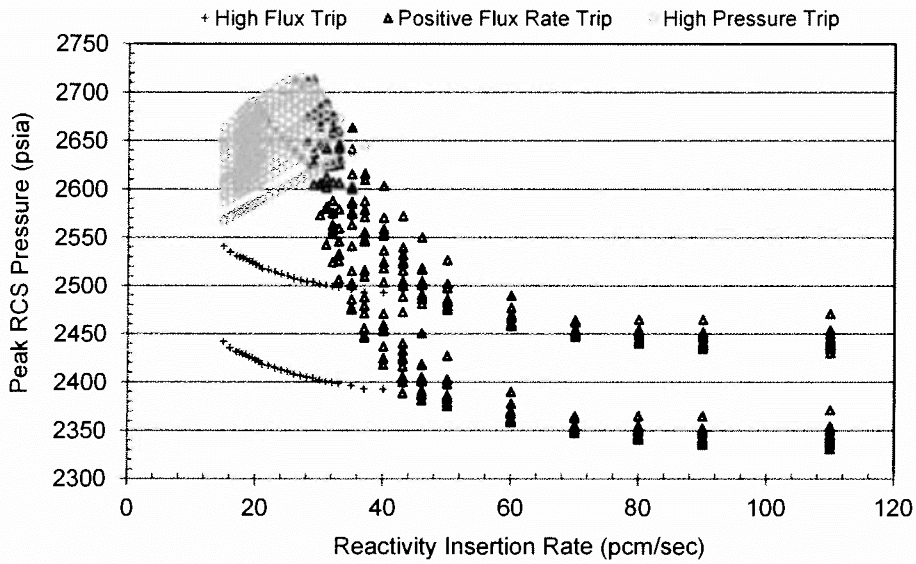
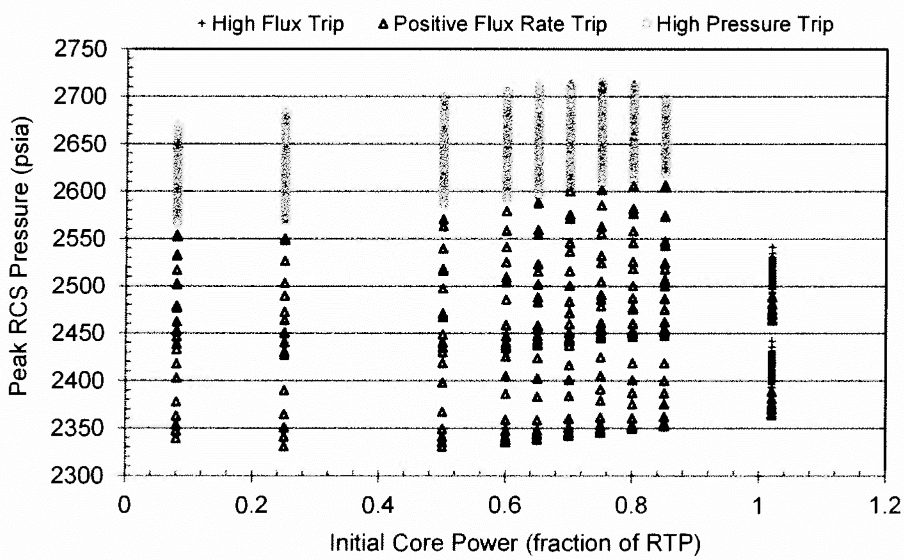
UNIT 1 AND UNIT 2

EFFECT OF REACTIVITY INSERTION RATE AND INITIAL POWER LEVEL ON PEAK RCS PRESSURE FOR A ROD WITHDRAWAL ACCIDENT

FIGURE 15.2-10A





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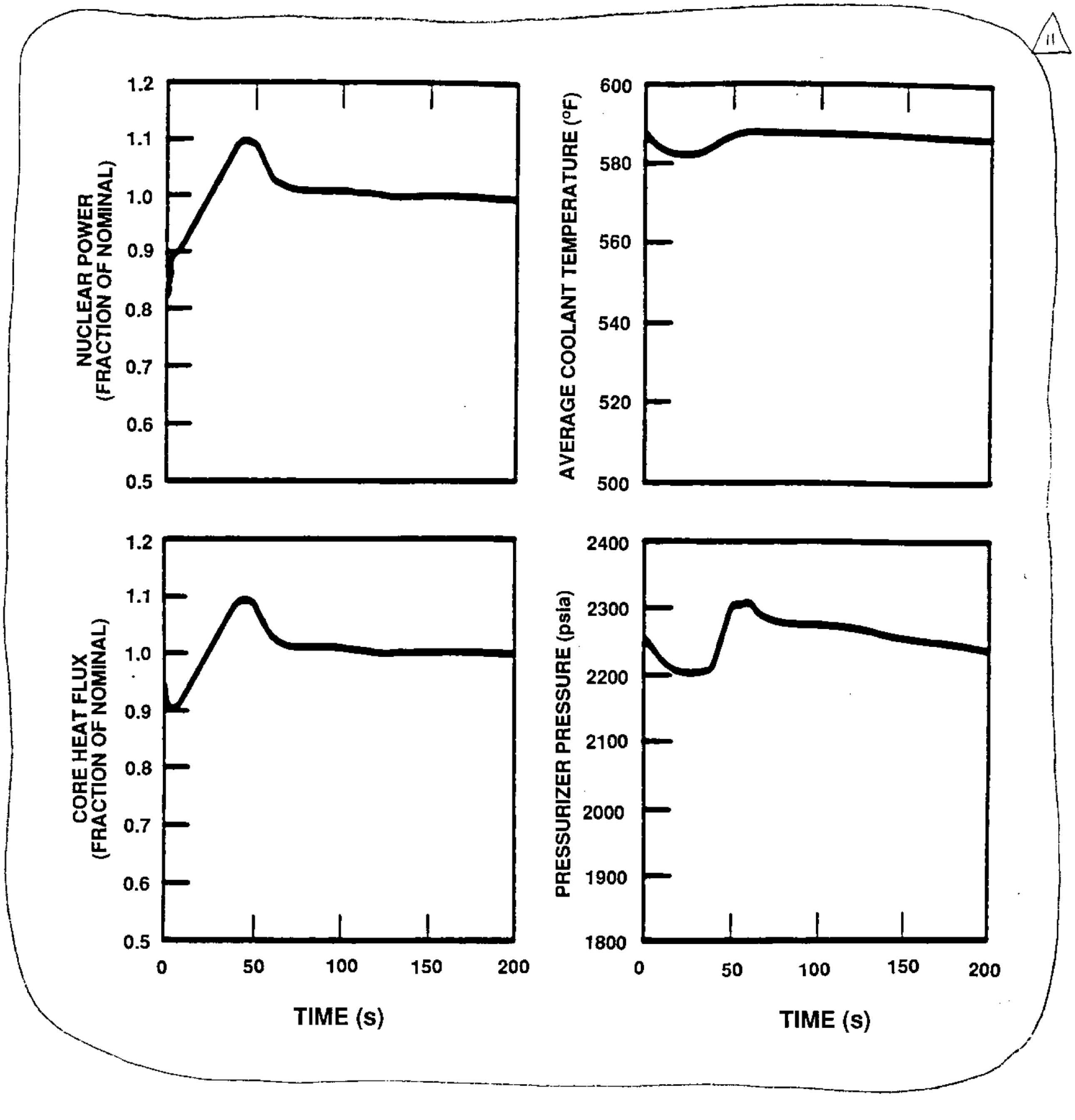
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	EFFECT OF REACTIVITY INSERTION RATE AND INITIAL POWER LEVEL ON PEAK RCS TRIP FUNCTION FOR A ROD WITHDRAWAL ACCIDENT (OVERPRESSURE CASE)
		FIGURE 15.2-10B



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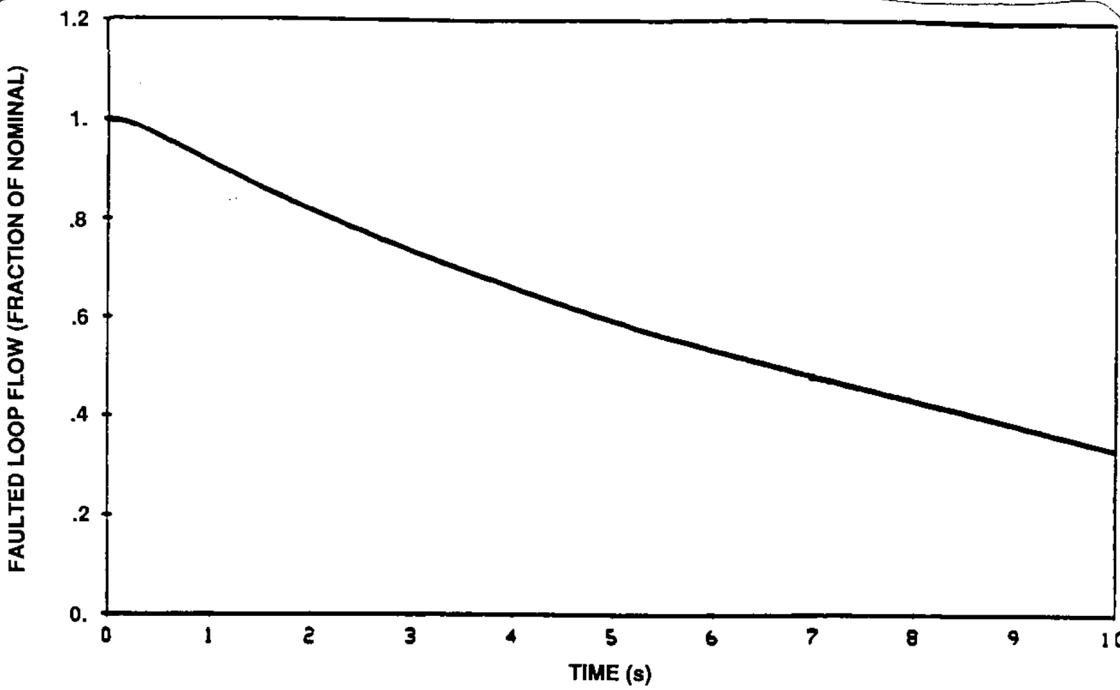
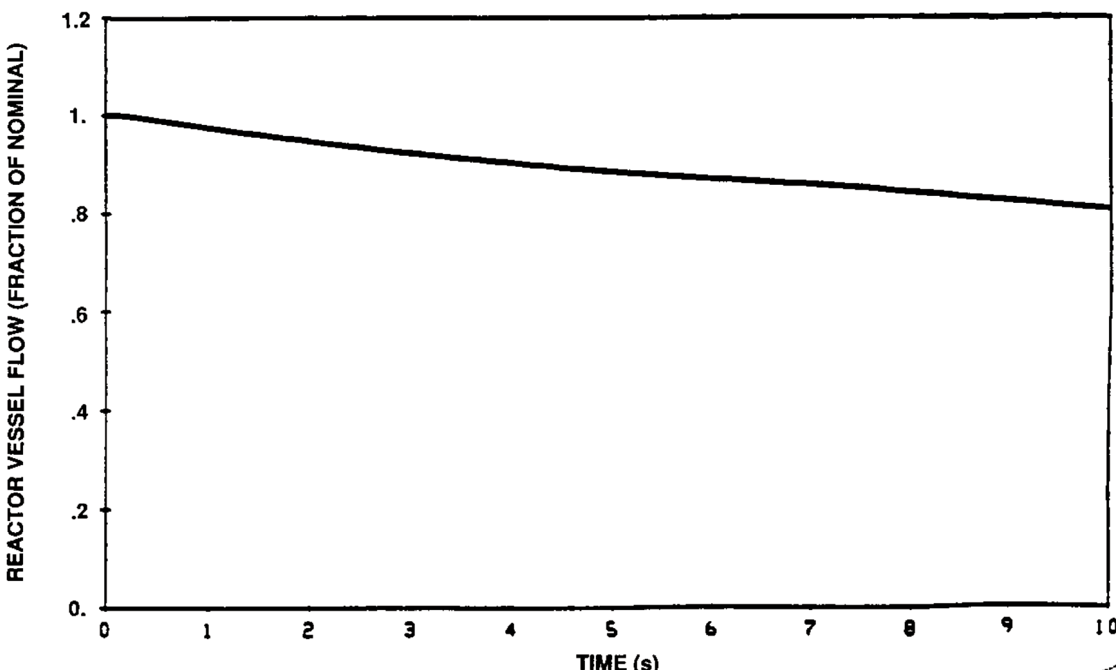
TRANSIENT RESPONSE TO A DROPPED ROD CLUSTER CONTROL ASSEMBLY

FIGURE 15.2-11





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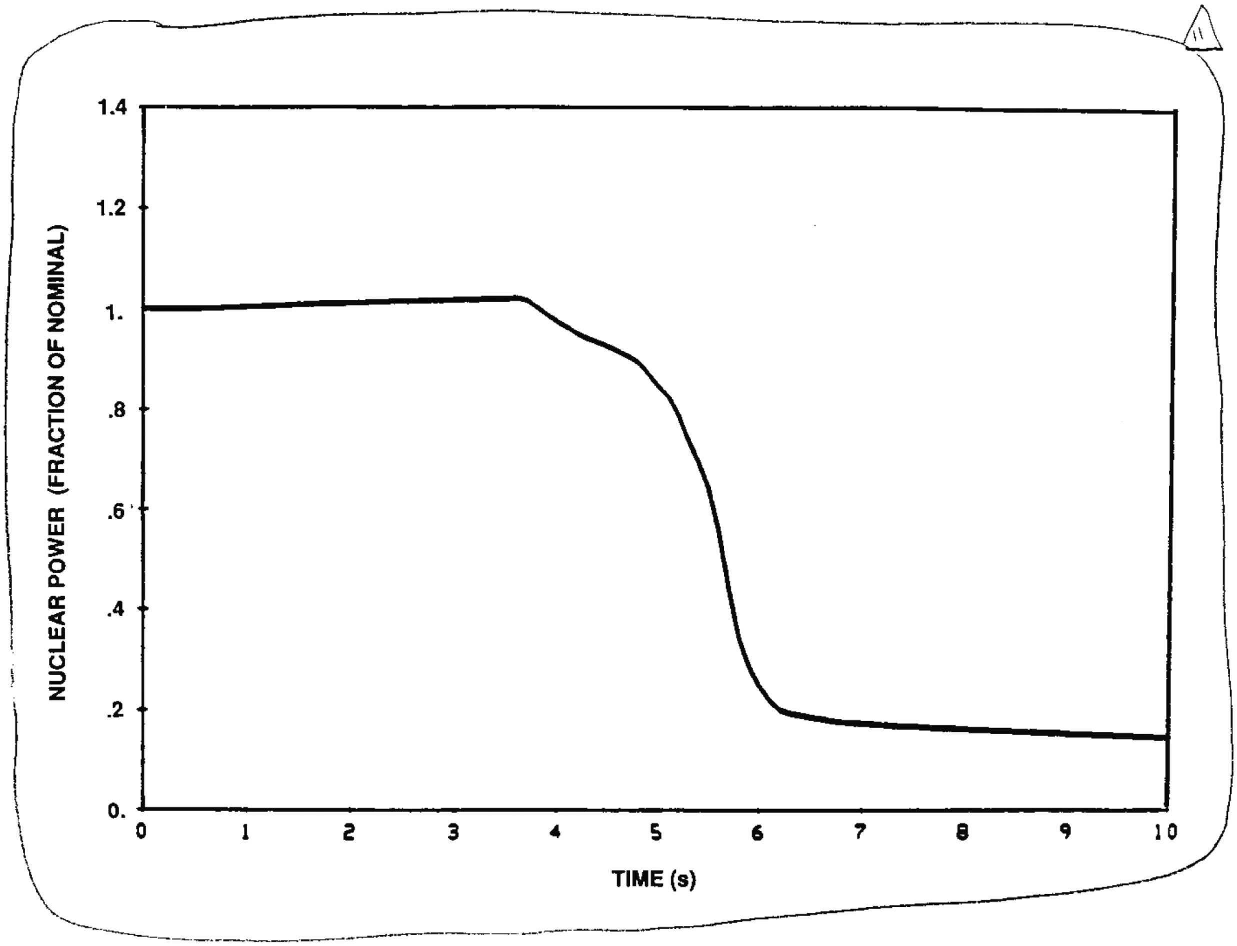
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPERATING, ONE LOOP COASTING DOWN – FLOW COASTDOWNS VERSUS TIME
		FIGURE 15.2-12



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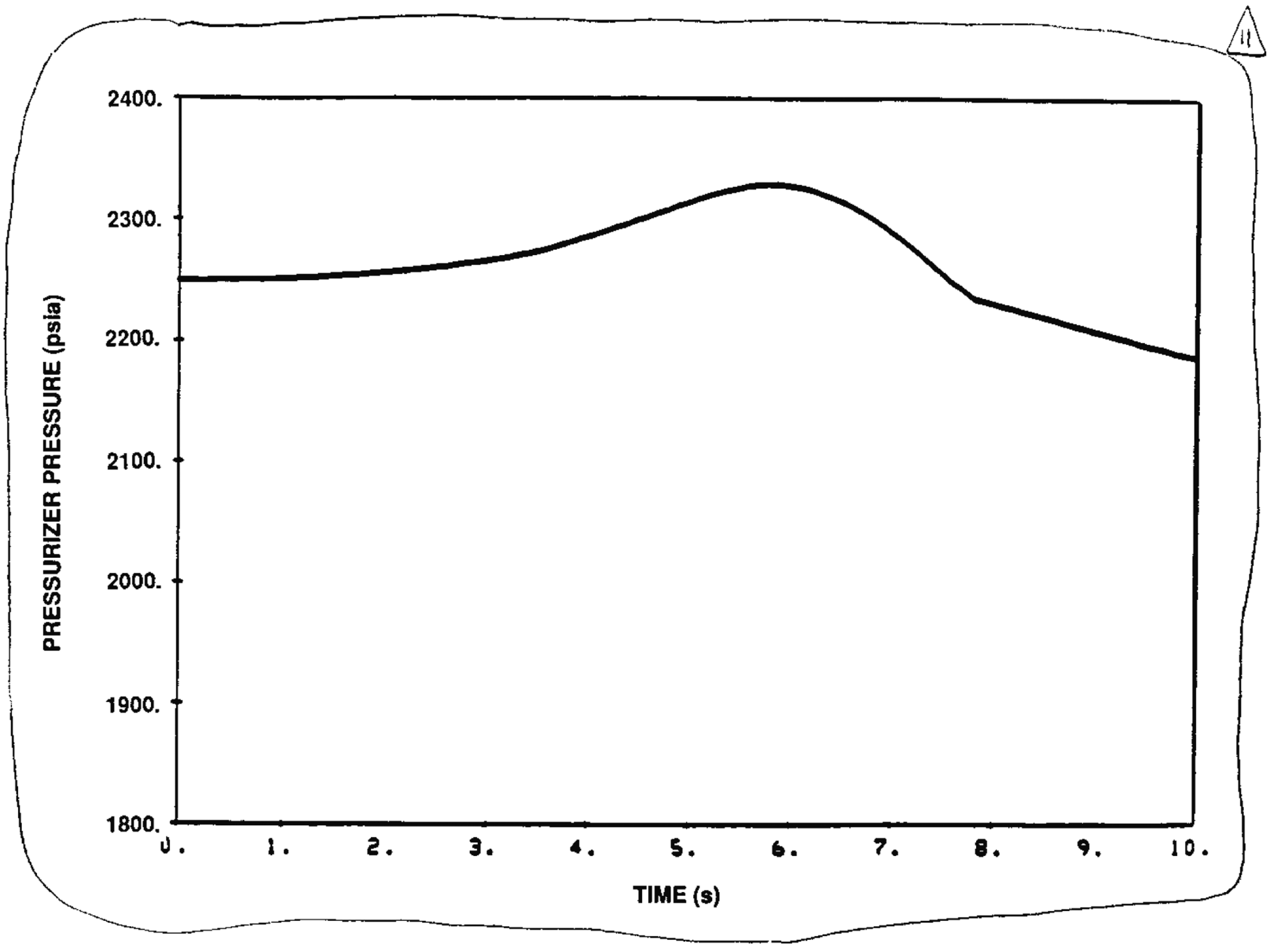
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPERATING, ONE LOOP COASTING DOWN – NUCLEAR POWER VERSUS TIME FIGURE 15.2-13



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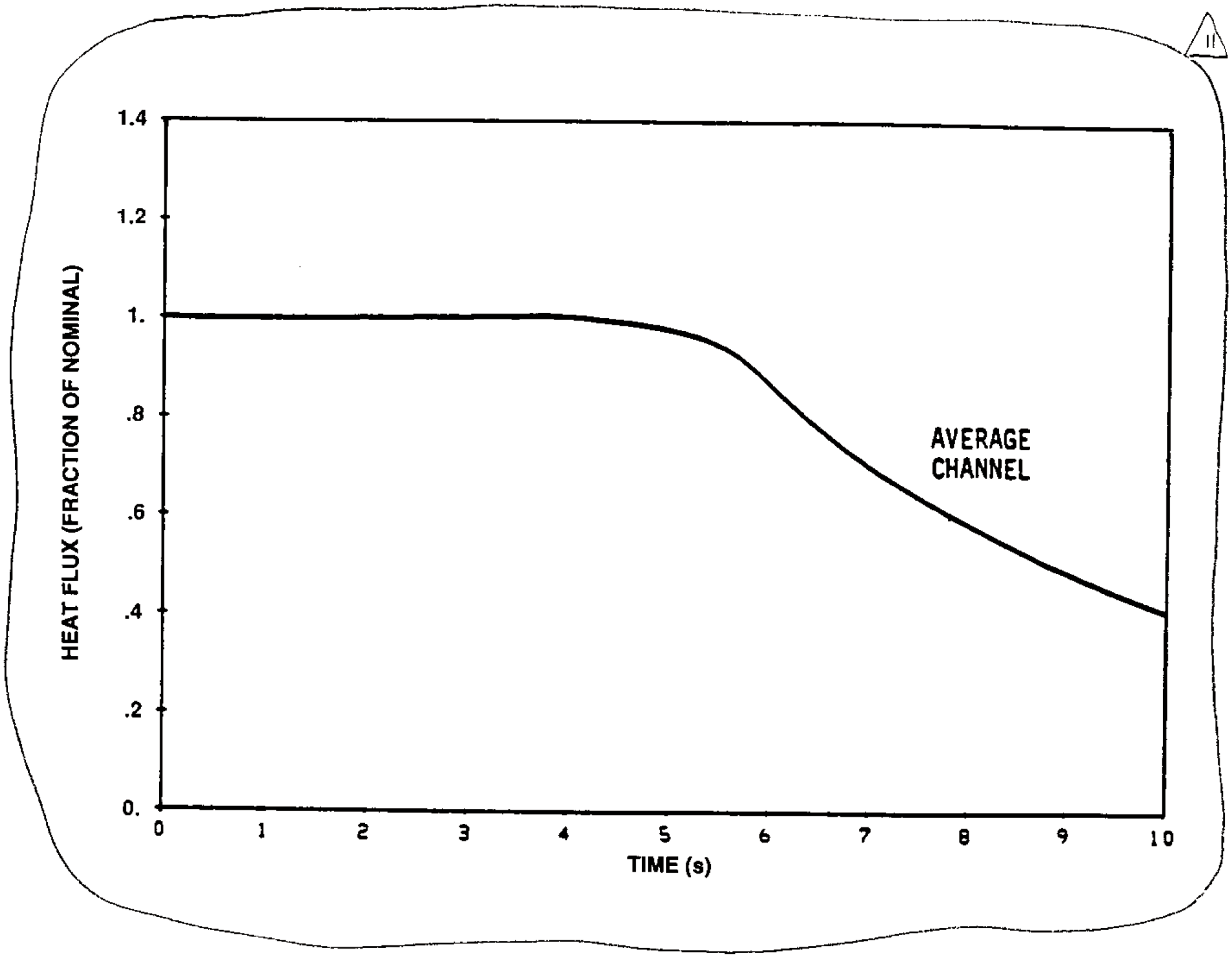
UNIT 1 AND UNIT 2

ALL LOOPS INITIALLY OPERATING,
ONE LOOP COASTING DOWN –
PRESSURIZER PRESSURE VERSUS TIME

FIGURE 15.2-14



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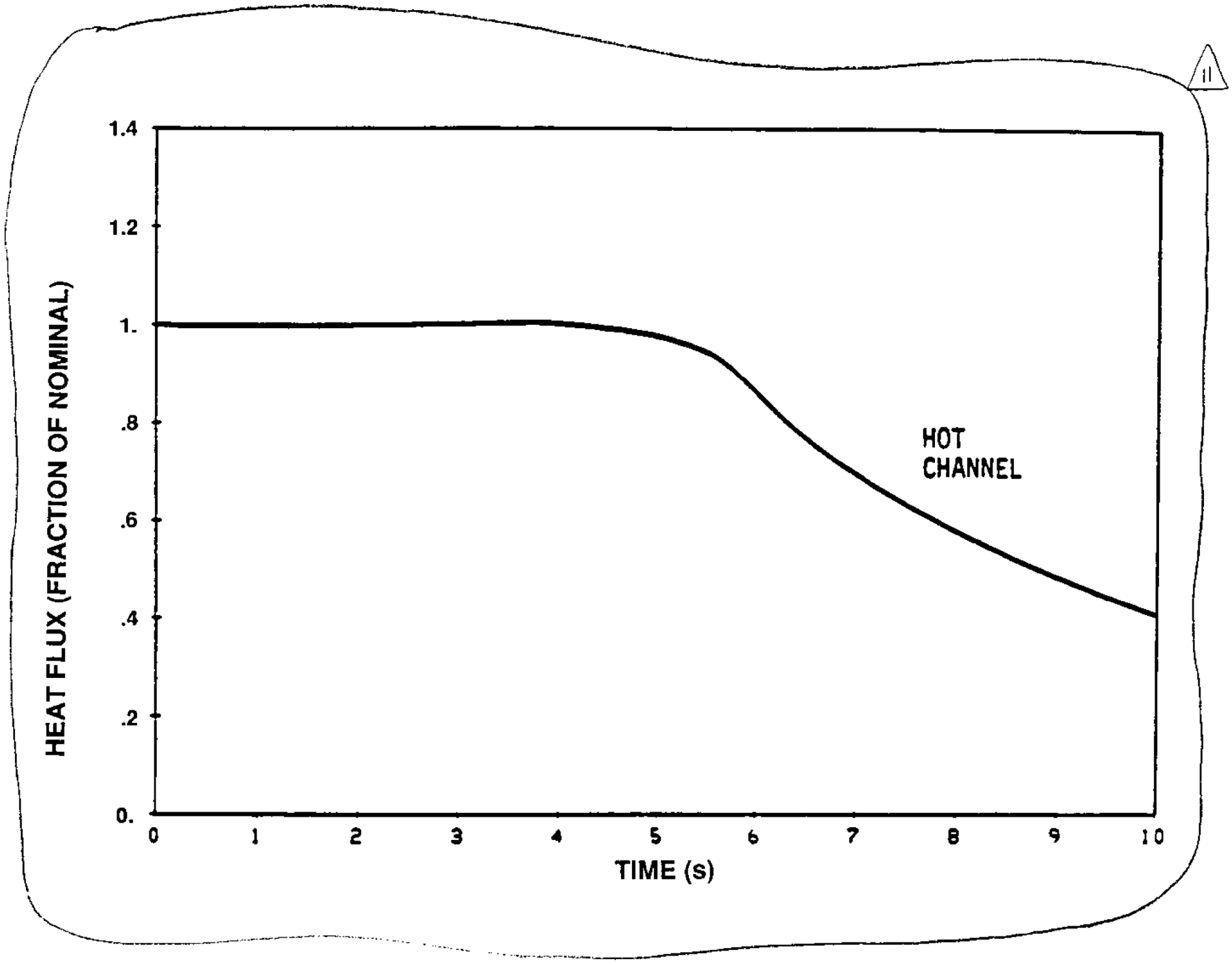
ALL LOOPS INITIALLY OPERATING,
ONE LOOP COASTING DOWN – AVERAGE
CHANNEL HEAT FLUX VERSUS TIME

FIGURE 15.2-15



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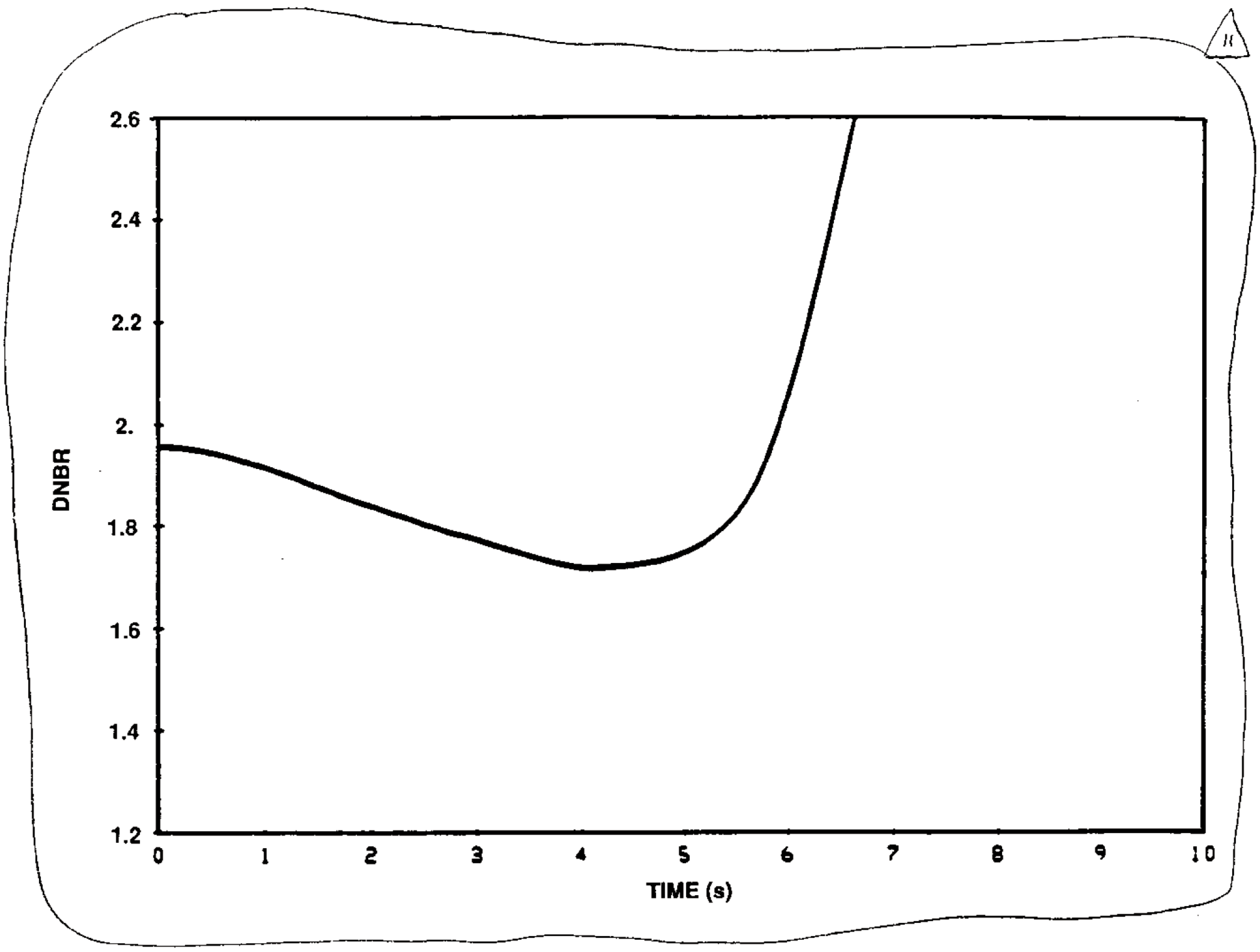
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS INITIALLY OPERATING, ONE LOOP COASTING DOWN – HOT CHANNEL HEAT FLUX VERSUS TIME

FIGURE 15.2-16



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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPERATING, ONE LOOP COASTING DOWN – DNBR VERSUS TIME
		FIGURE 15.2-17

THIS FIGURE HAS BEEN DELETED.

15

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SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STARTUP OF AN INACTIVE REACTOR COOLANT LOOP – 17 X 17 CORE FIGURE 15.2-18 (SHEET 1 OF 3)

THIS FIGURE HAS BEEN DELETED.

15

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SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STARTUP OF AN INACTIVE REACTOR COOLANT LOOP – 17 X 17 CORE
		FIGURE 15.2-18 (SHEET 2 OF 3)

THIS FIGURE HAS BEEN DELETED.

15

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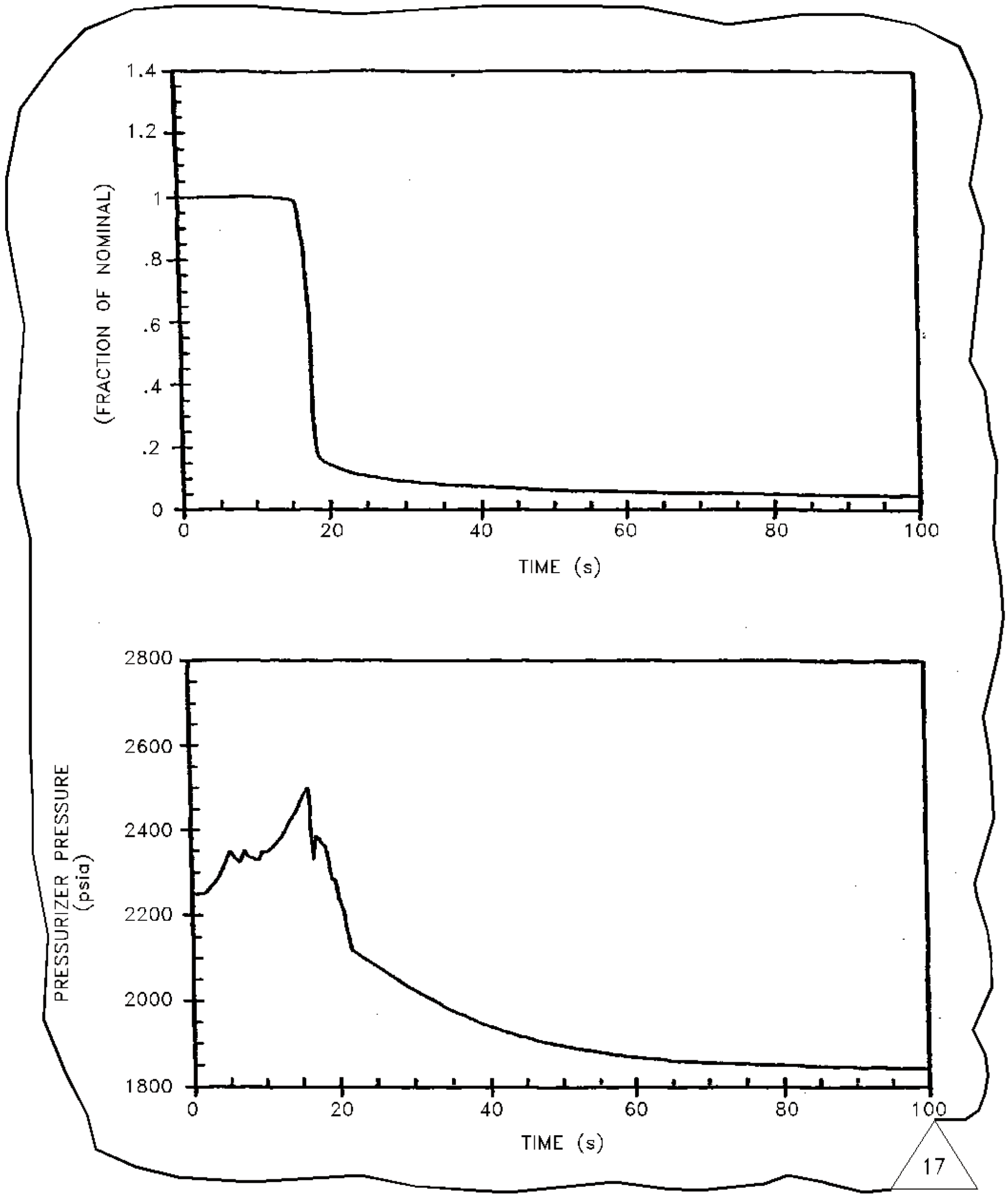
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STARTUP OF AN INACTIVE REACTOR COOLANT LOOP – 17 X 17 CORE
		FIGURE 15.2-18 (SHEET 3 OF 3)



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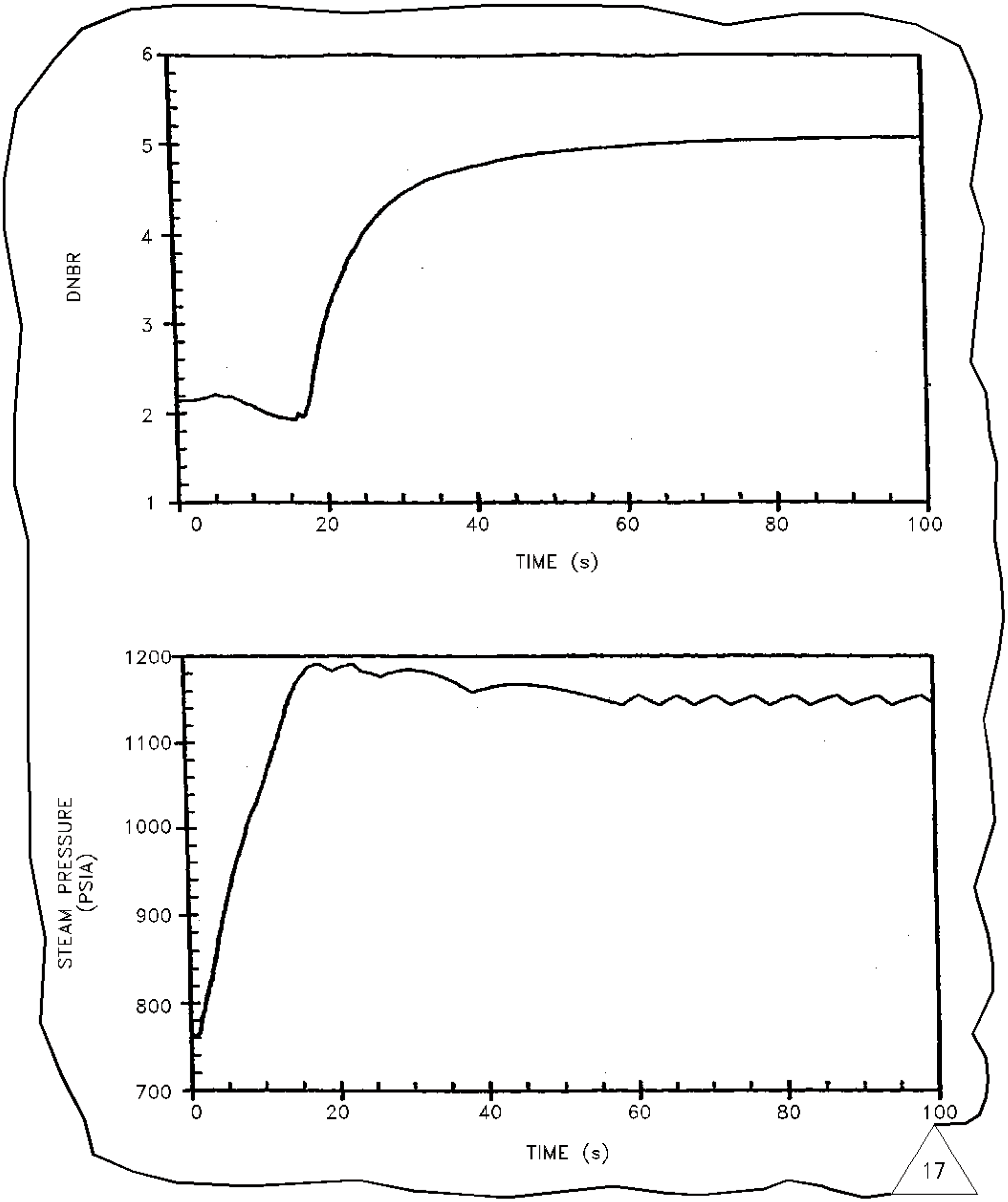
SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	LOSS-OF-LOAD ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVE – BOL
		FIGURE 15.2-19 (SHEET 1 OF 2)



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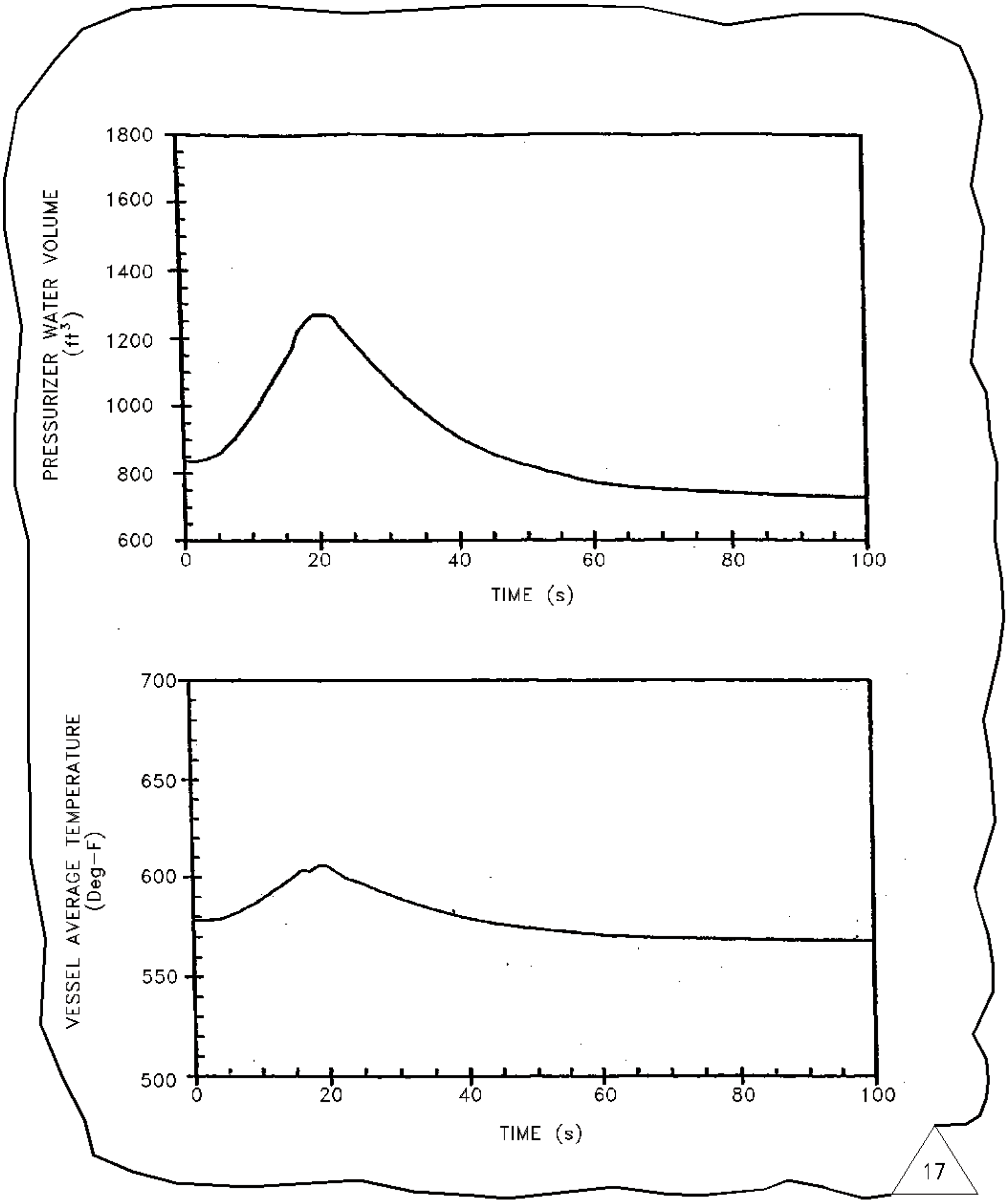
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITH
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-19 (SHEET 2 OF 2)



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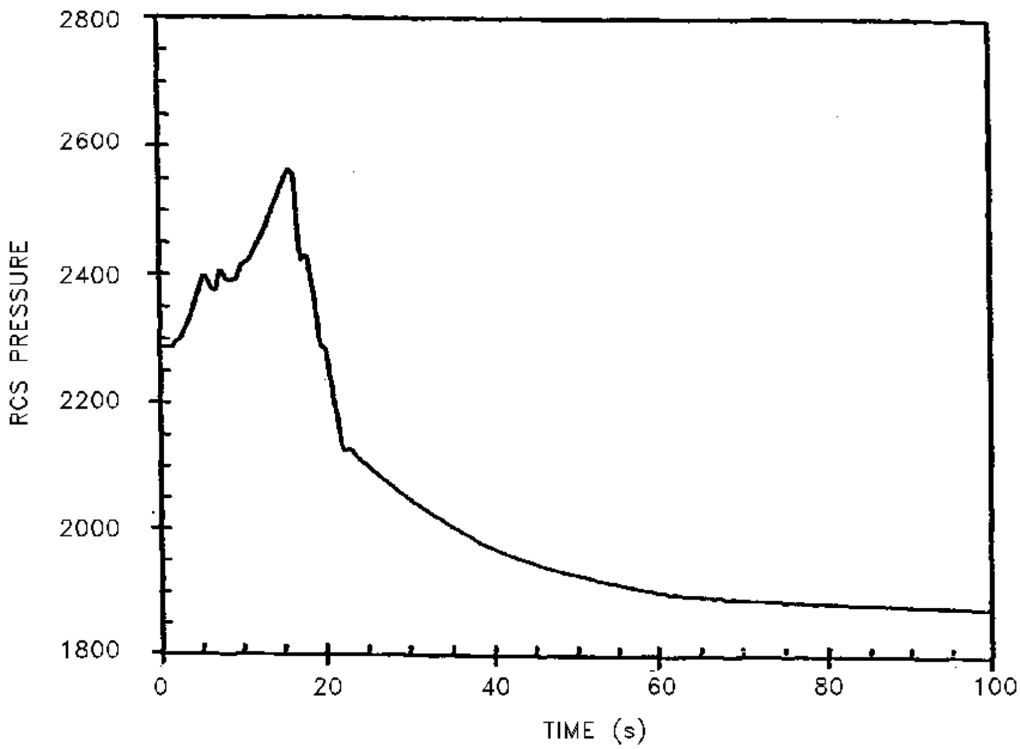
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS-OF-LOAD ACCIDENT WITH PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVE – BOL
		FIGURE 15.2-20



17

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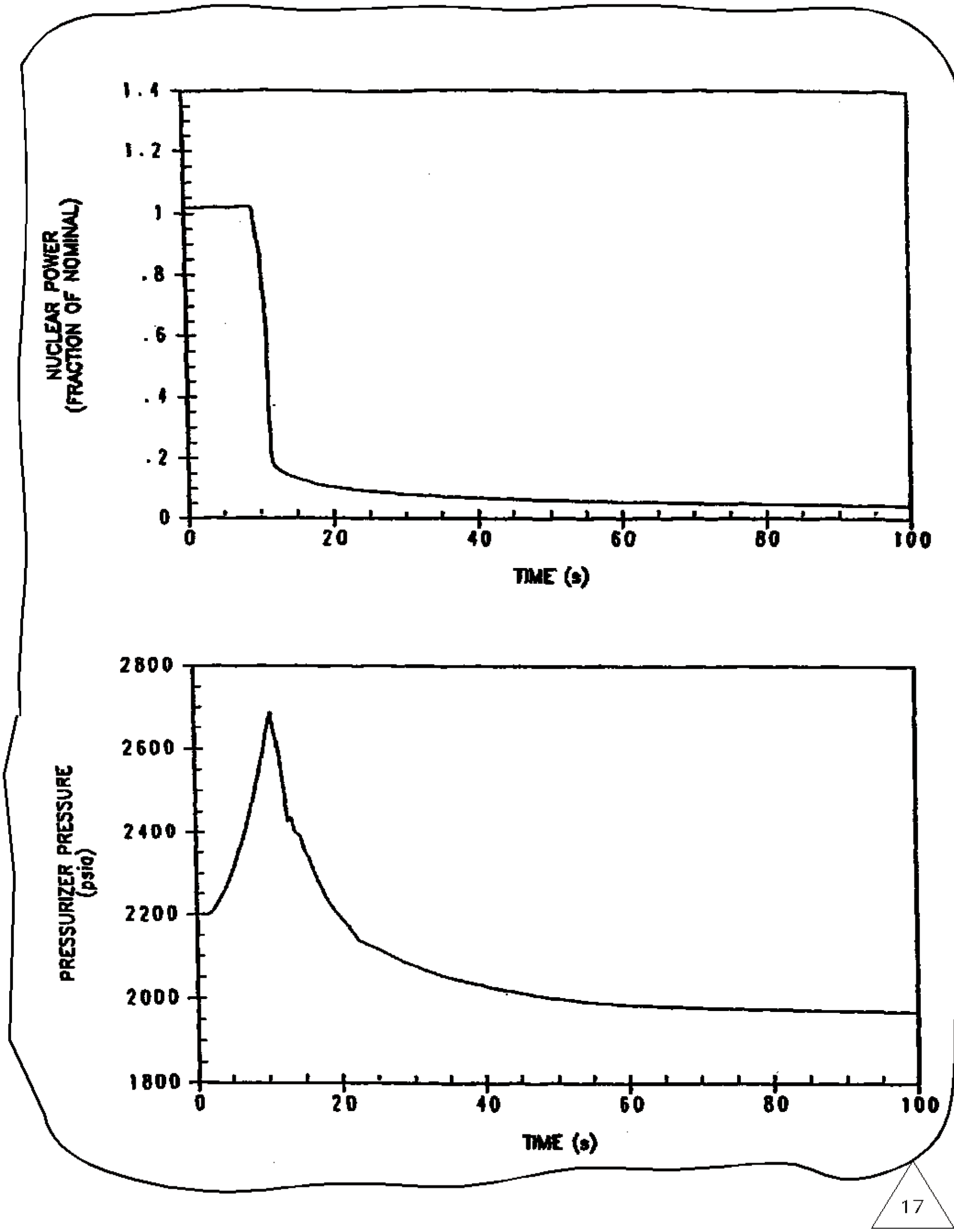
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITH
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-21



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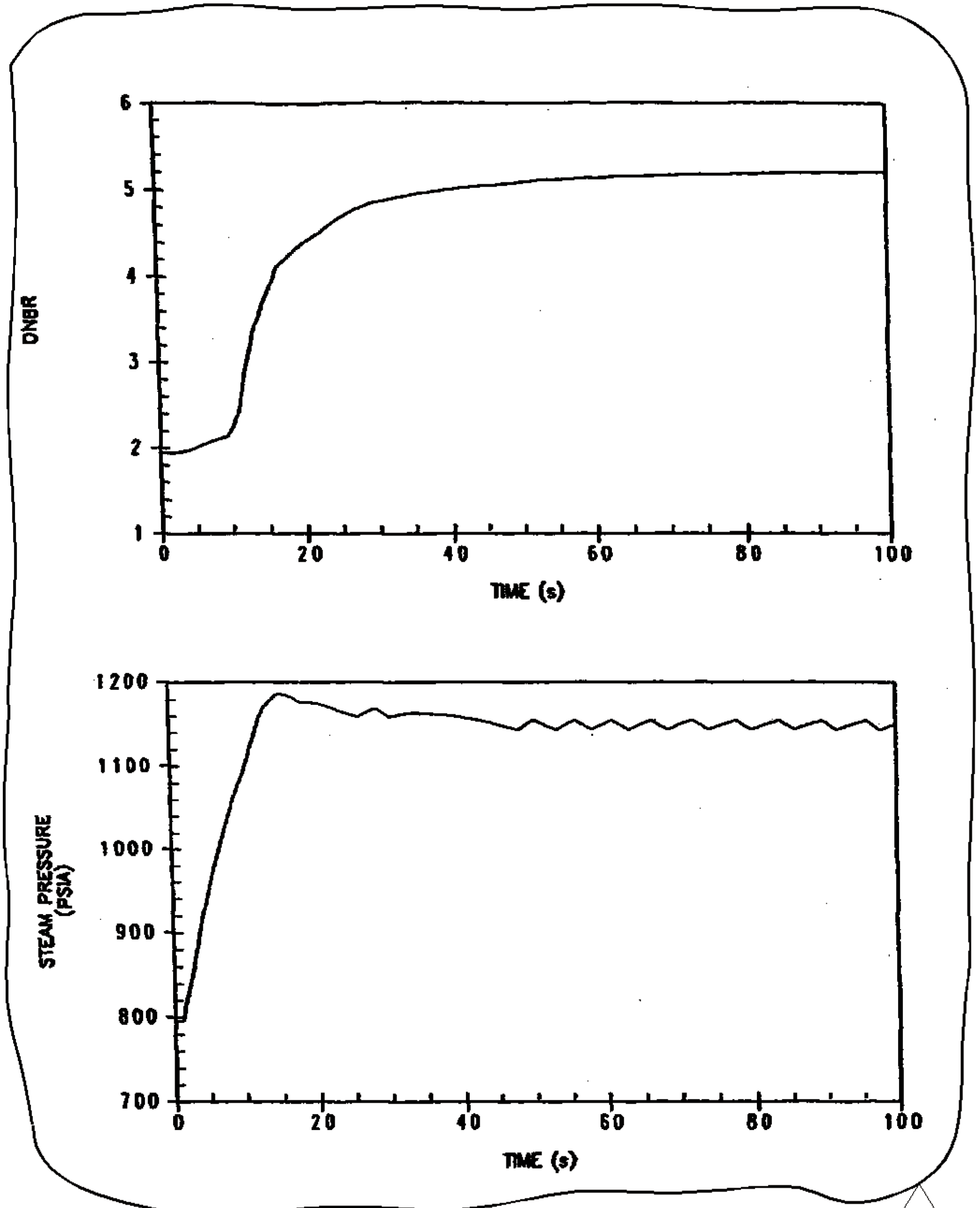
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-22



17

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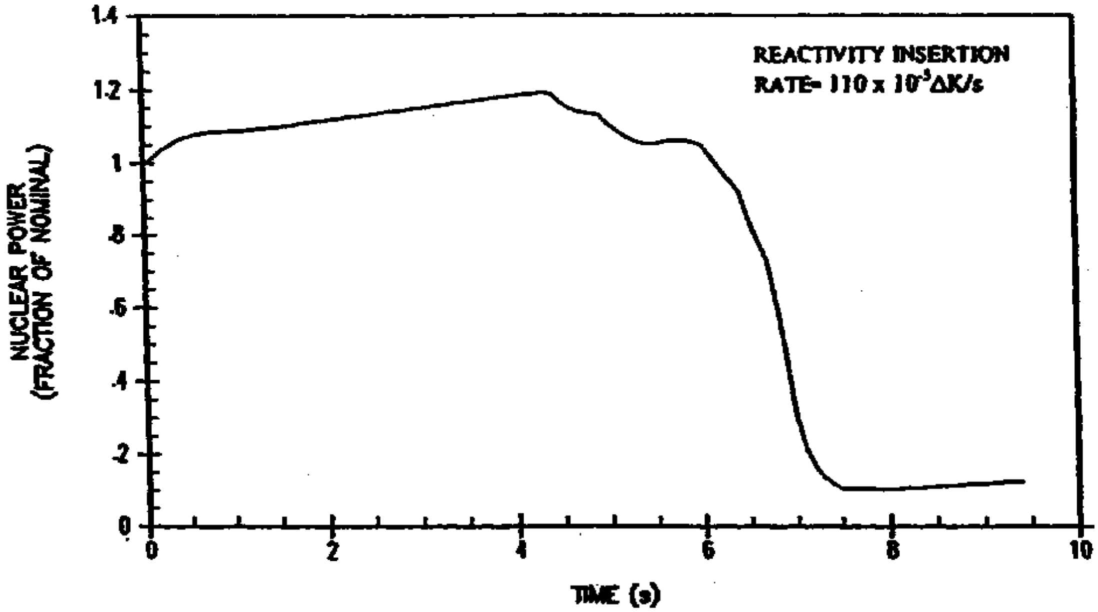
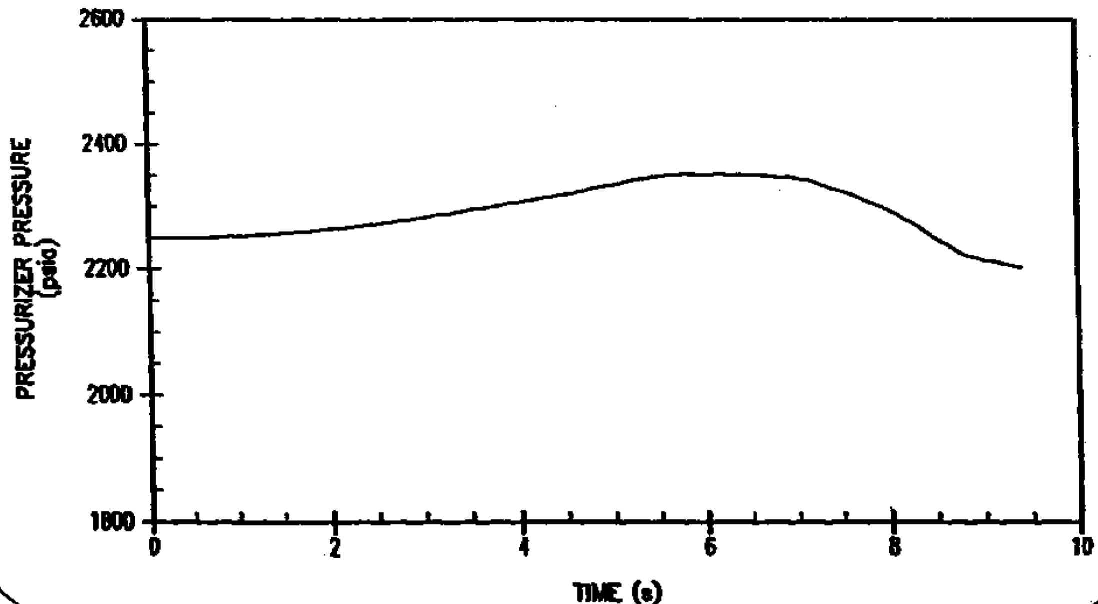
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-23





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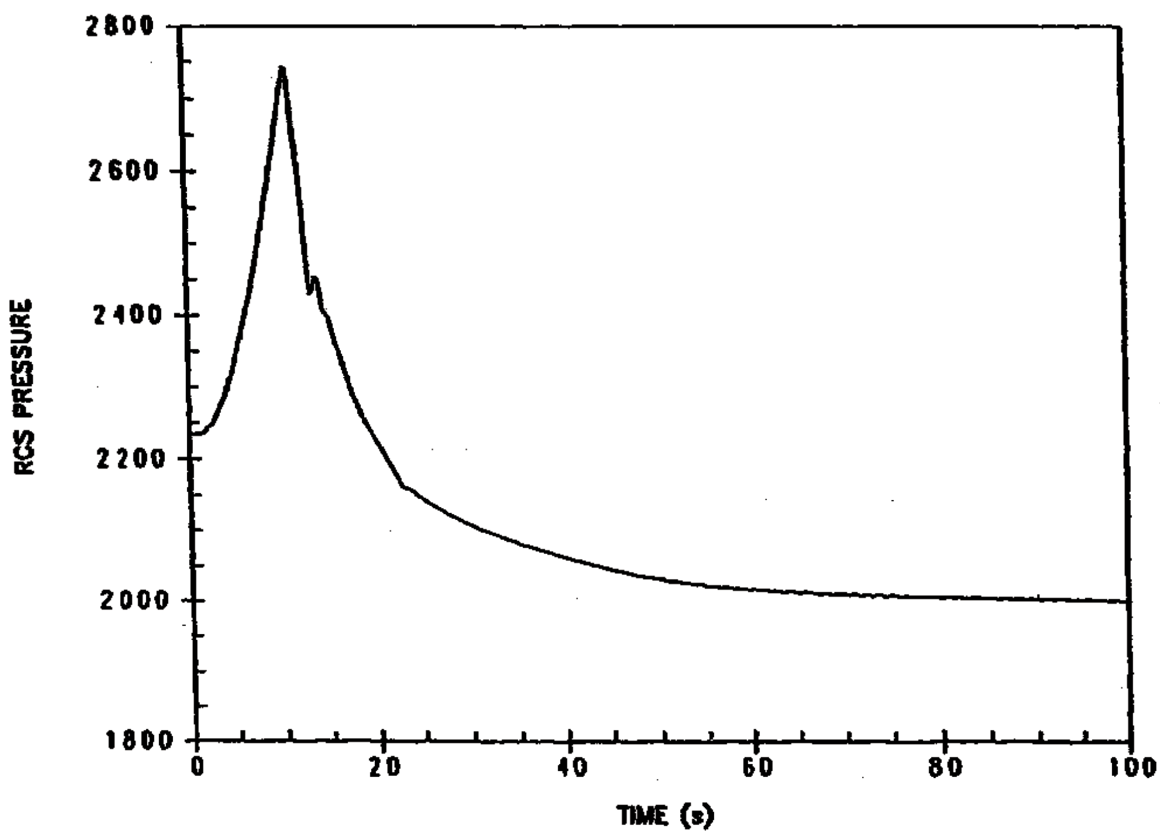
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-24



17

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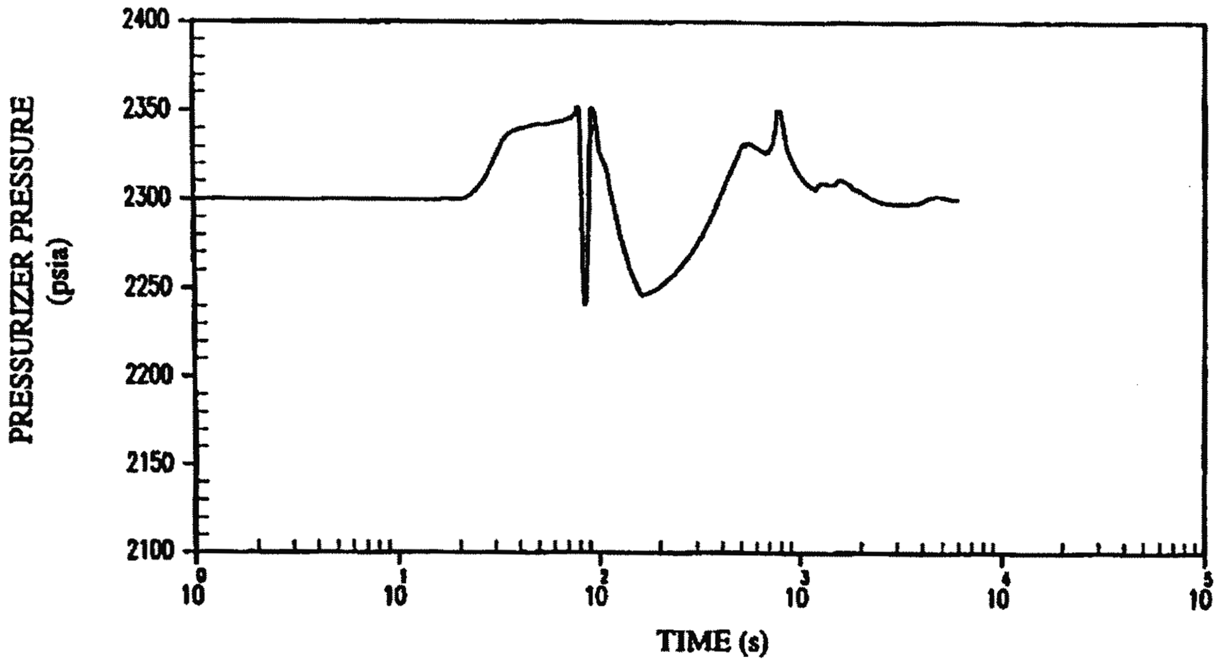
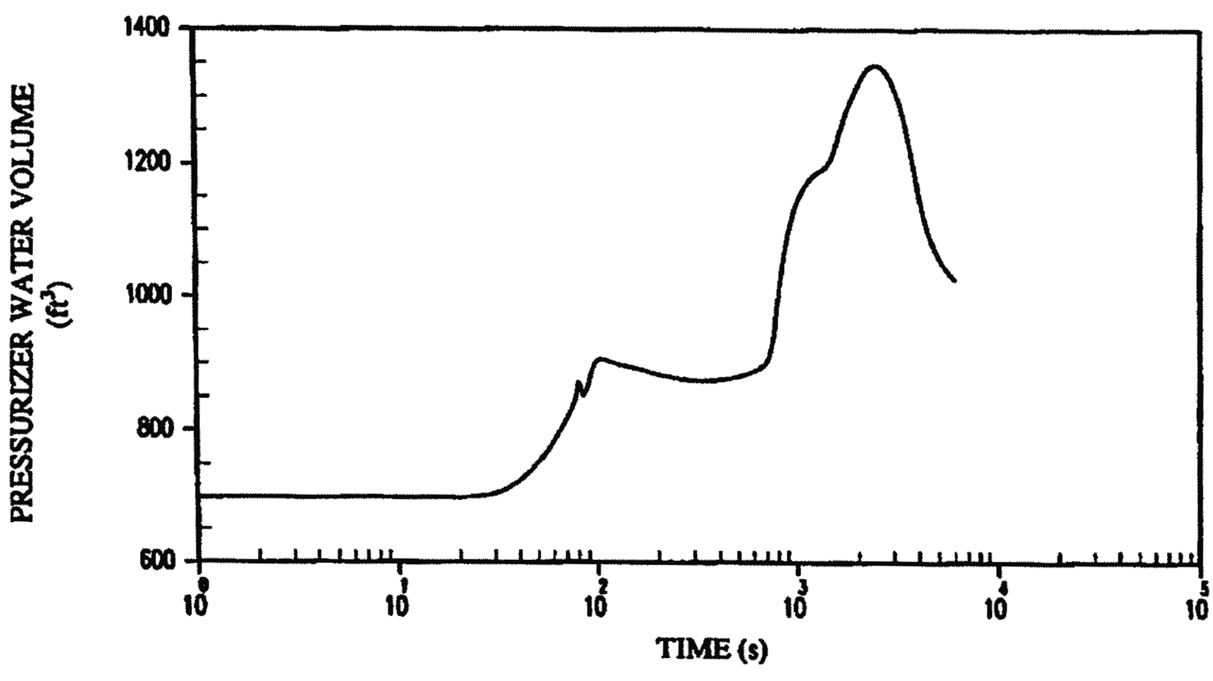
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVE – BOL

FIGURE 15.2-25





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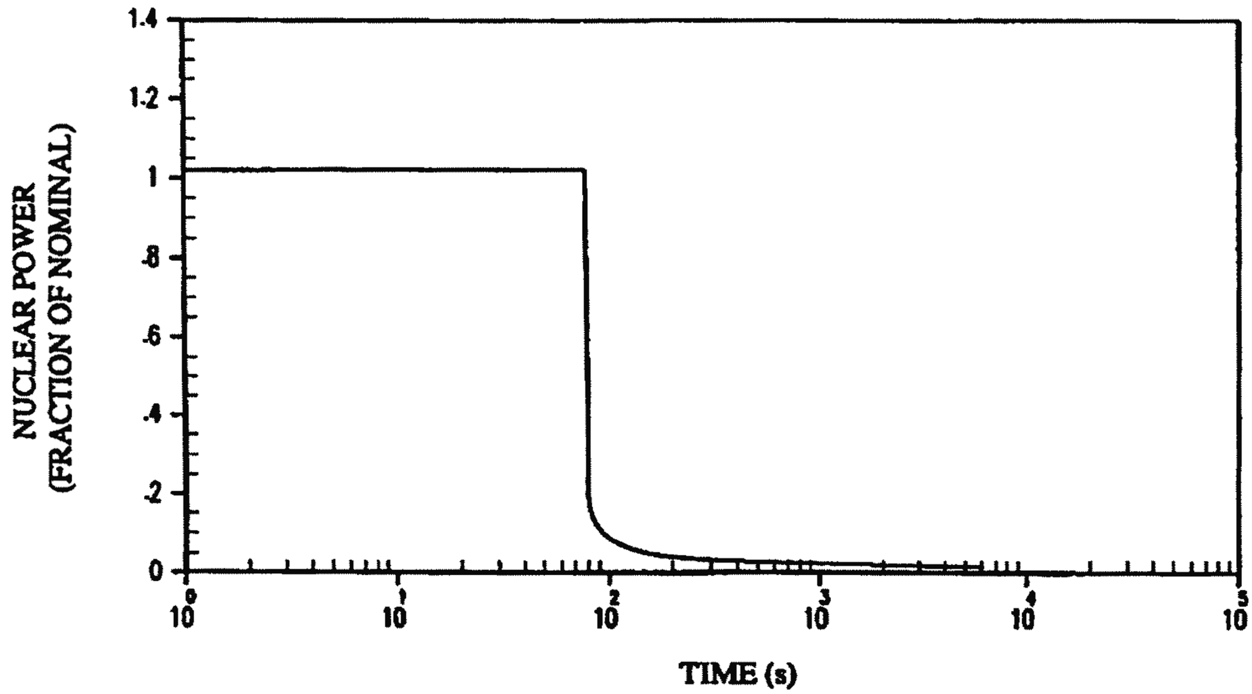
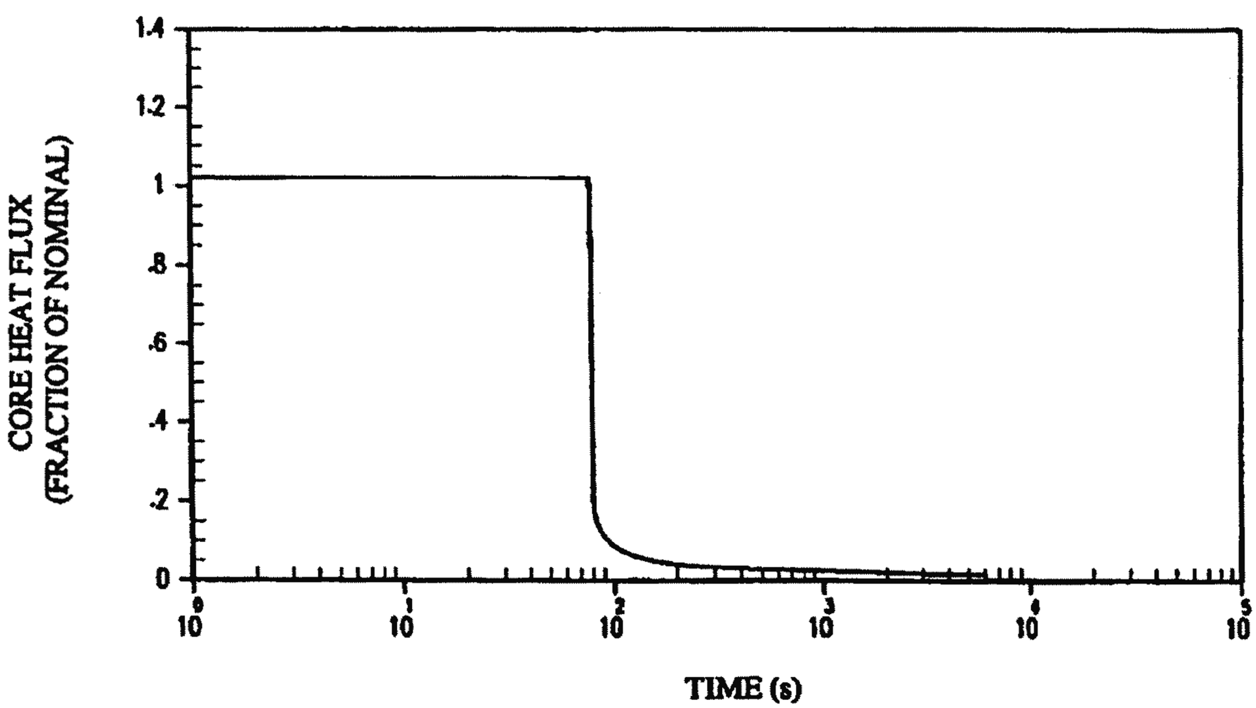
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS OF NORMAL FEEDWATER PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME
		FIGURE 15.2-26 (SHEET 1 OF 4)





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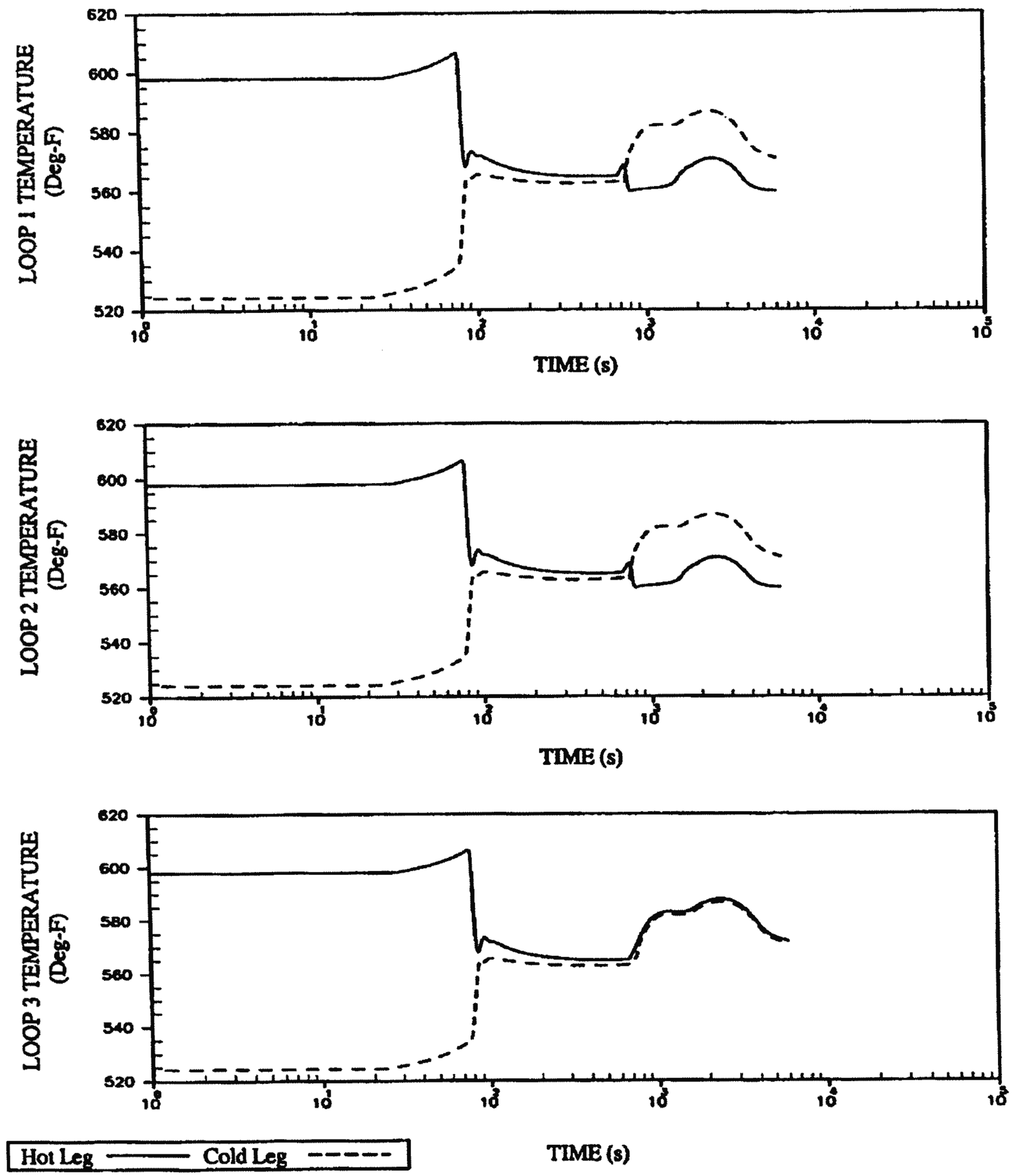
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS OF NORMAL FEEDWATER
NUCLEAR POWER AND CORE HEAT FLUX
VERSUS TIME

FIGURE 15.2-26 (SHEET 2 OF 4)



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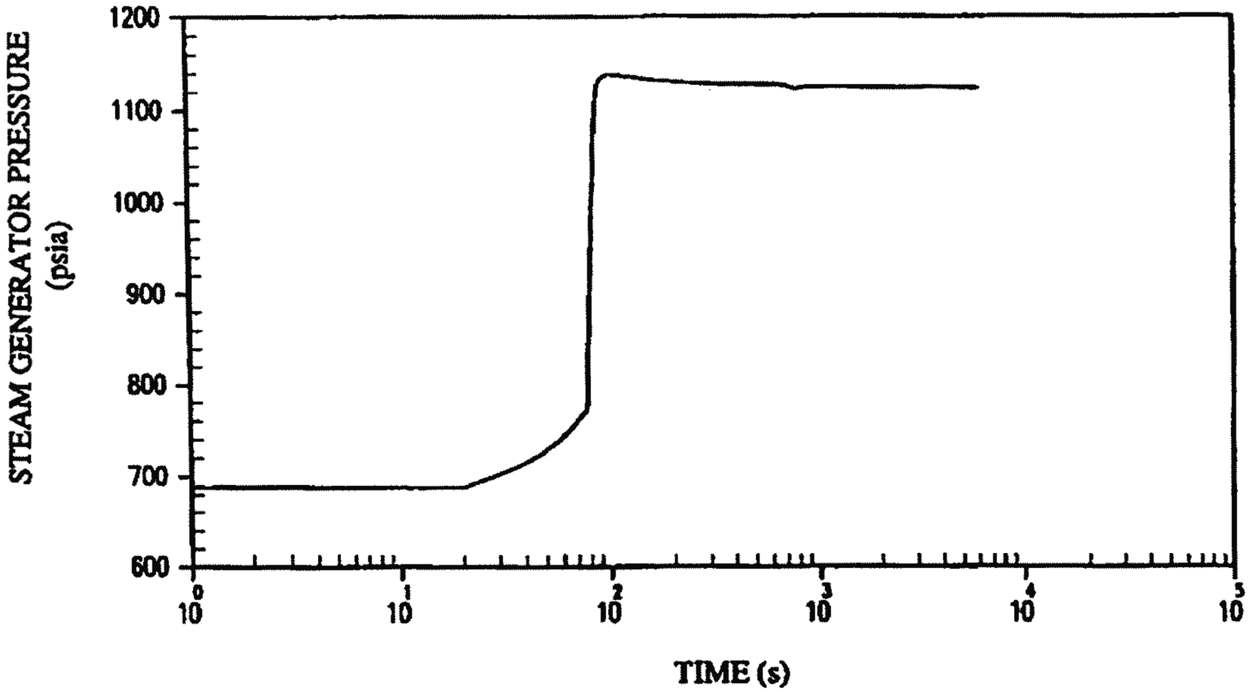
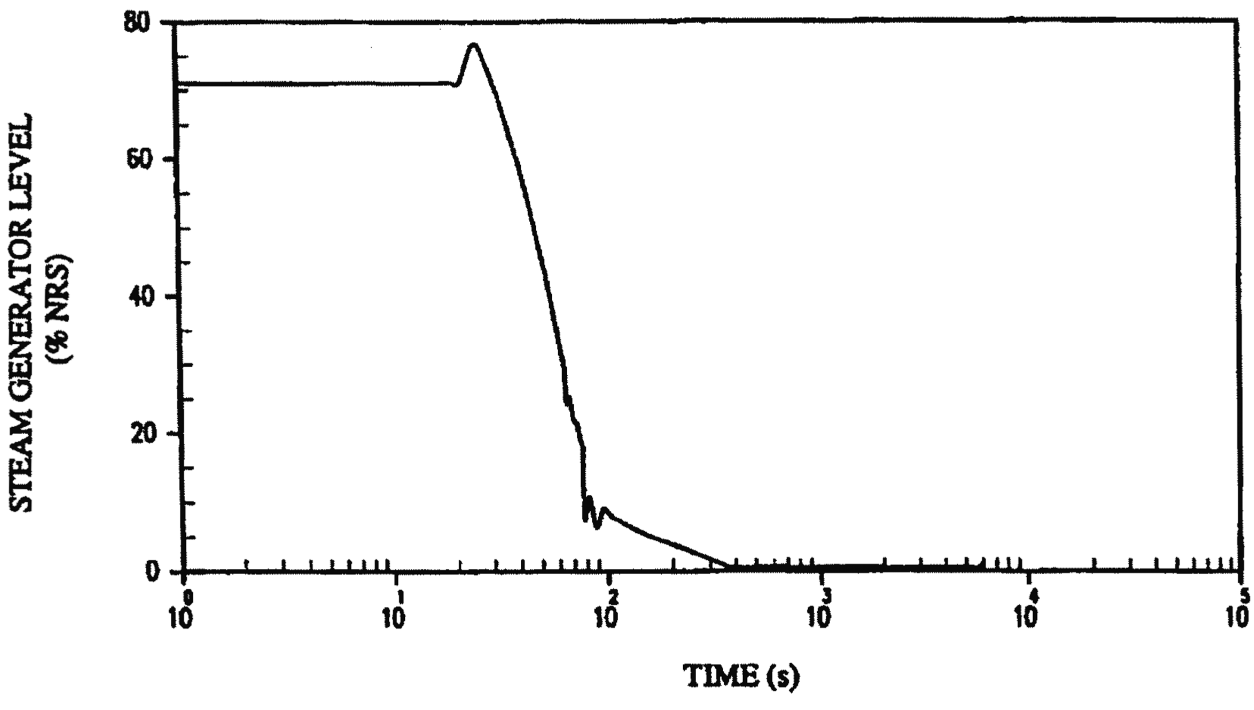
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS OF NORMAL FEEDWATER
LOOP 1, 2, AND 3 HOT AND COLD LEG
TEMPERATURES VERSUS TIME

FIGURE 15.2-26 (SHEET 3 OF 4)





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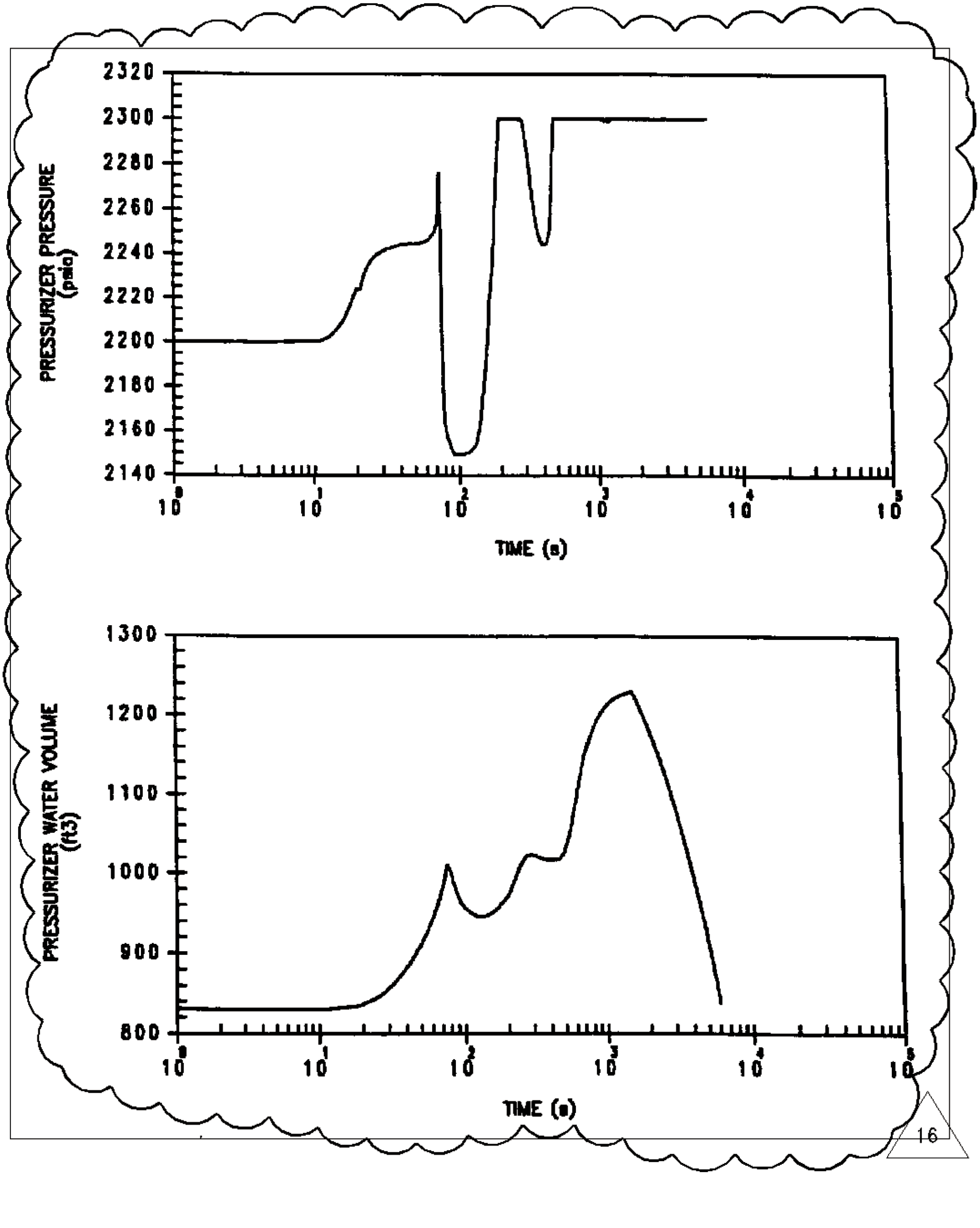
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS OF NORMAL FEEDWATER
STEAM GENERATOR PRESSURE AND LEVEL
VERSUS TIME

FIGURE 15.2-26 (SHEET 4 OF 4)



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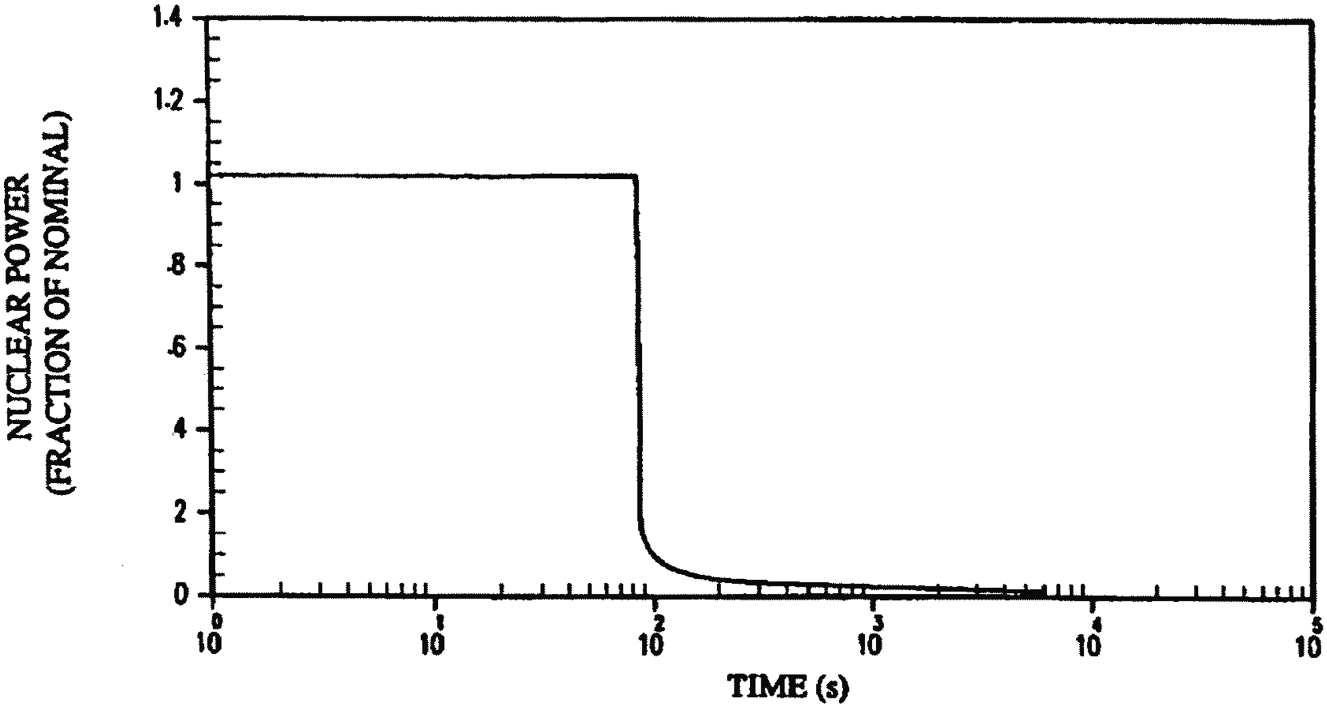
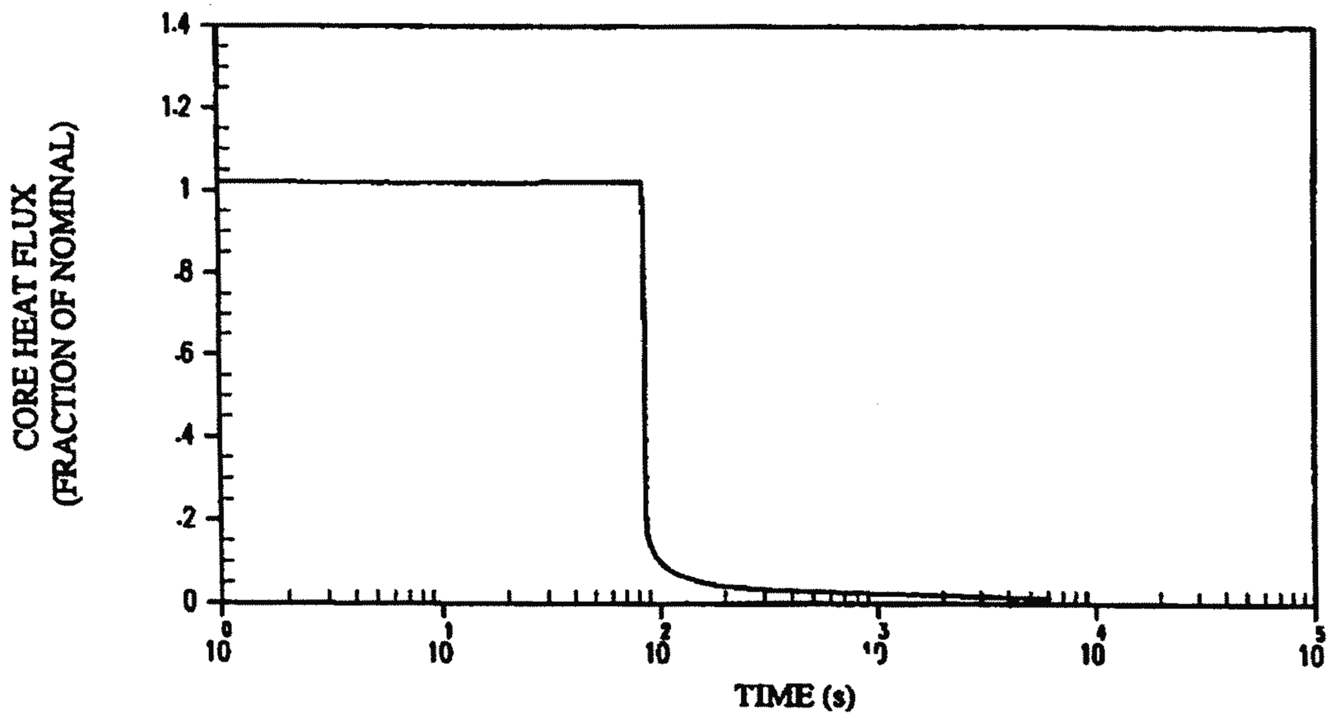
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME
		FIGURE 15.2-27 (SHEET 1 OF 4)





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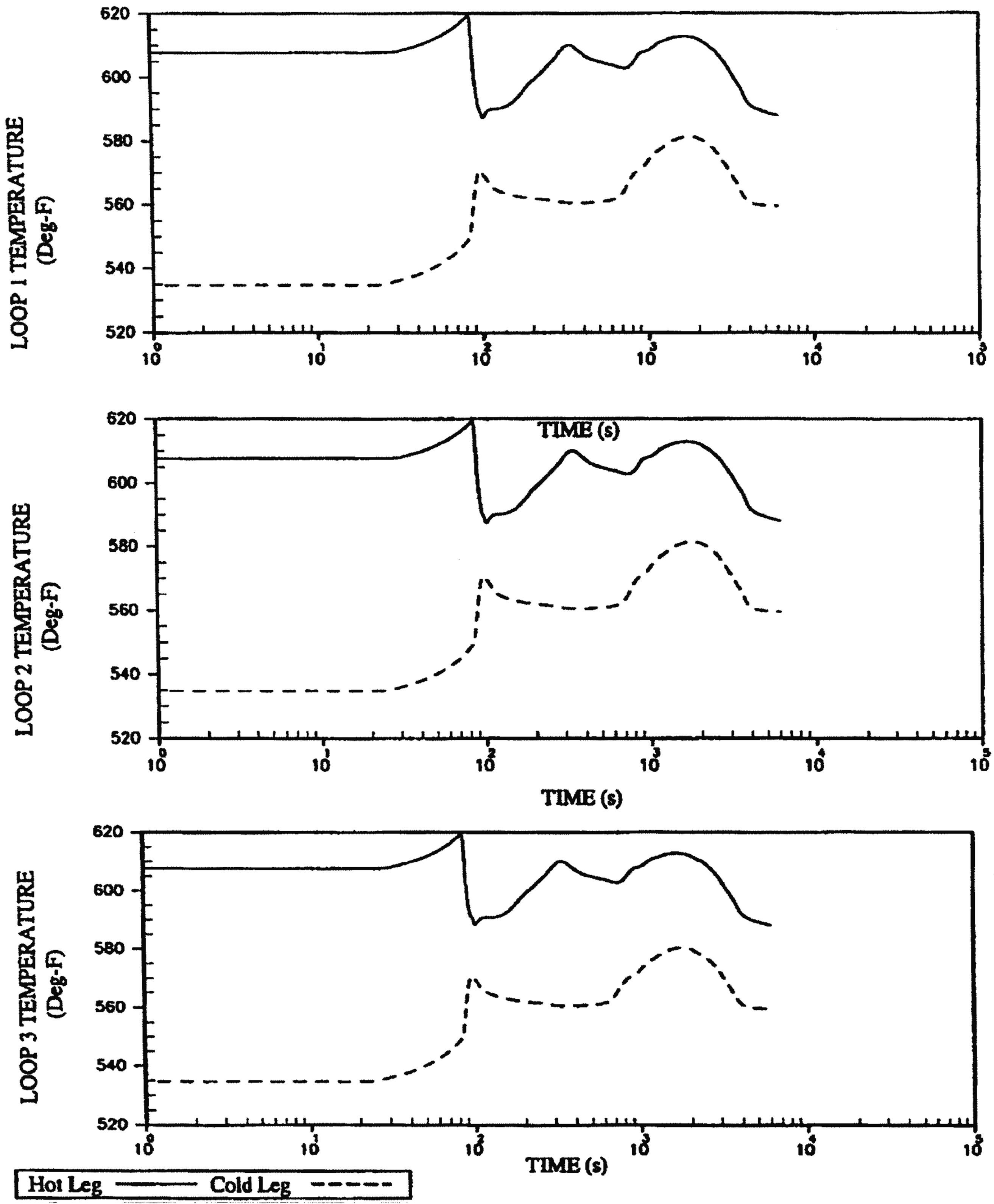
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME FIGURE 15.2-27 (SHEET 2 OF 4)



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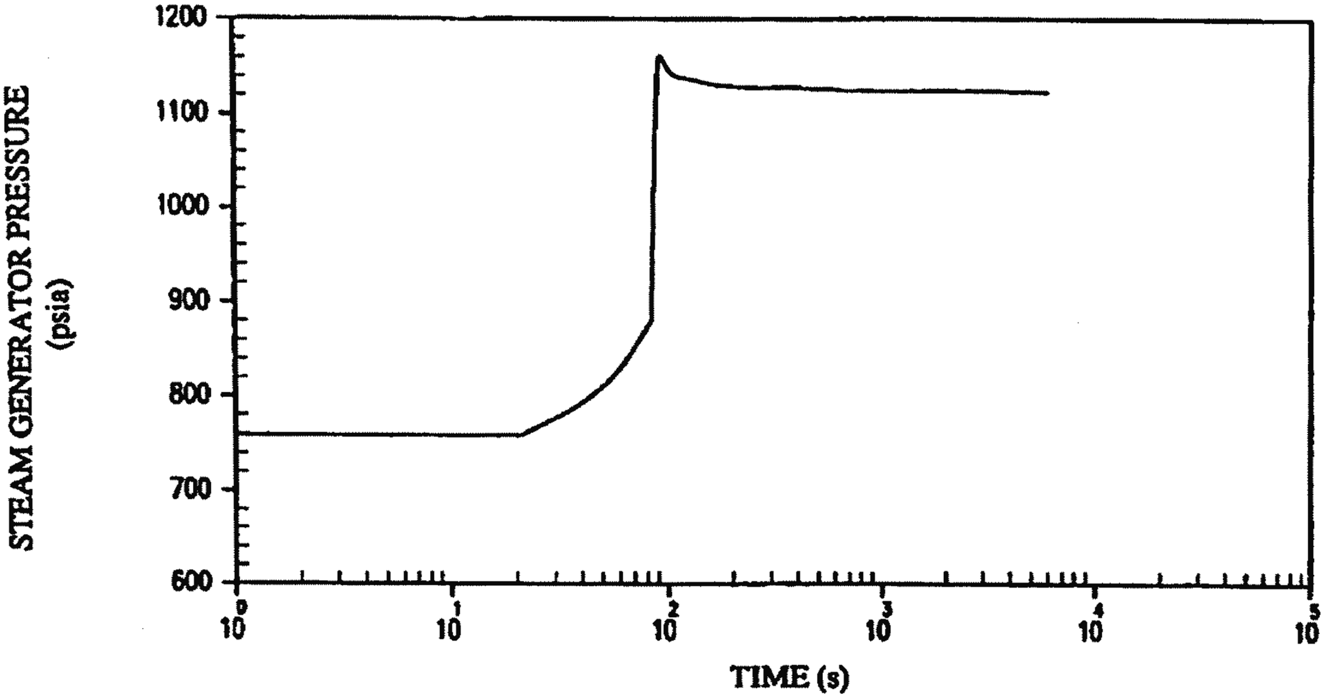
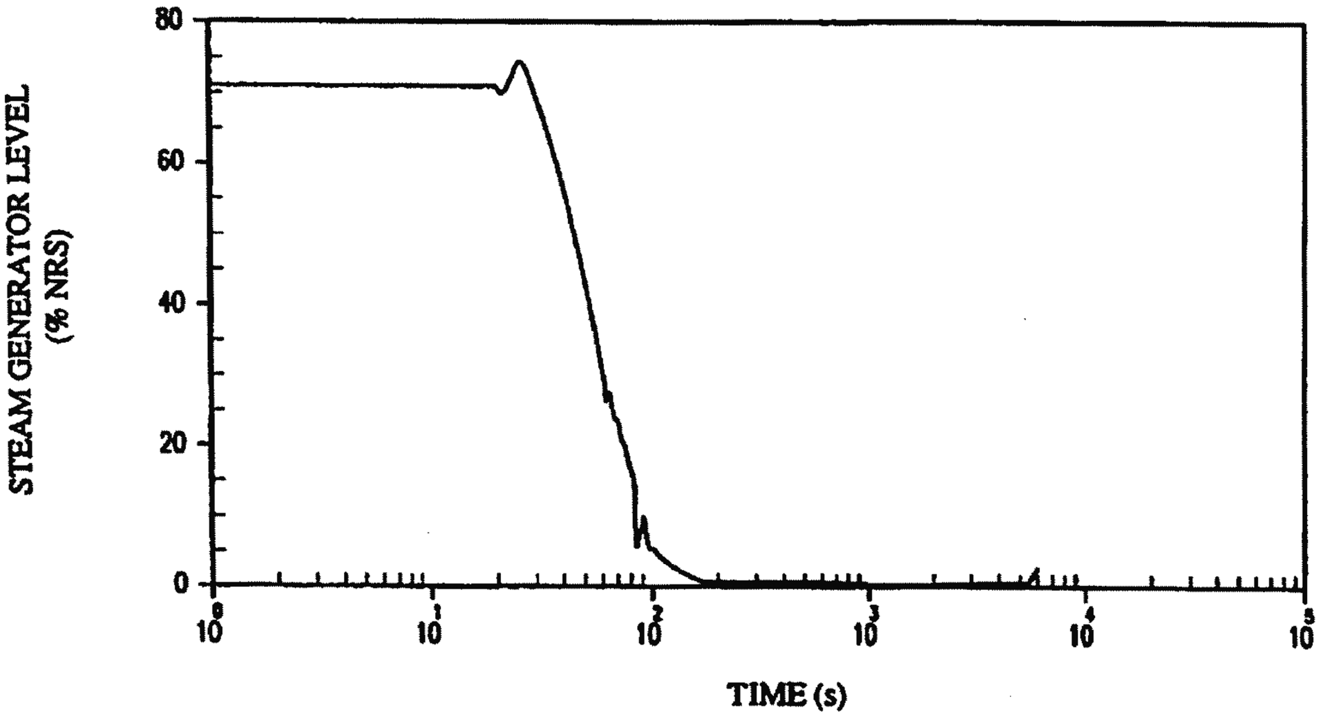
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – LOOP 1, 2, & 3 HOT AND COLD LEG TEMPERATURES VERSUS TIME

FIGURE 15.2-27 (SHEET 3 OF 4)





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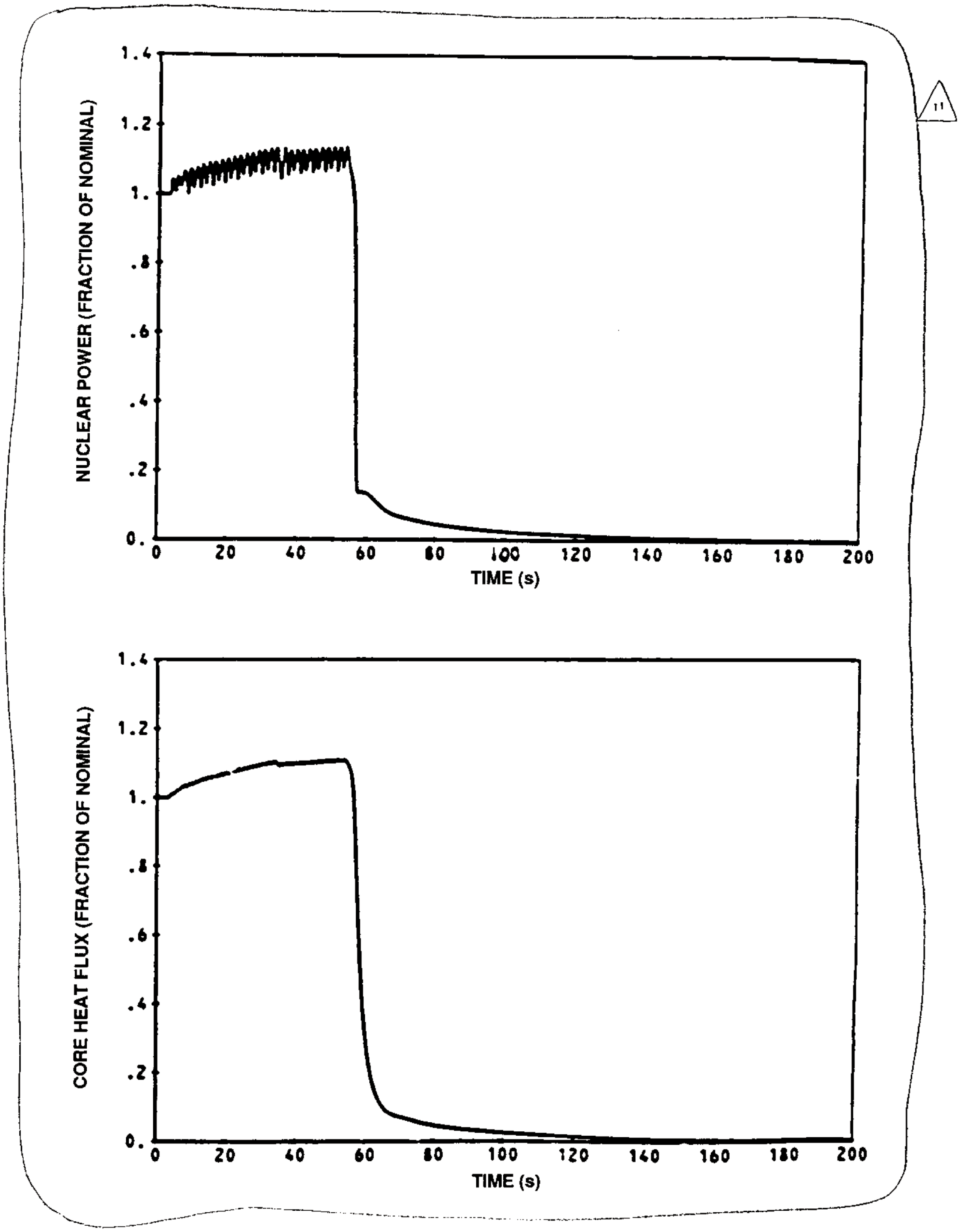
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS OF ALL AC POWER TO THE STATION AUXILIARIES – STEAM GENERATOR PRESSURE AND LEVEL VERSUS TIME

FIGURE 15.2-27 (SHEET 4 OF 4)



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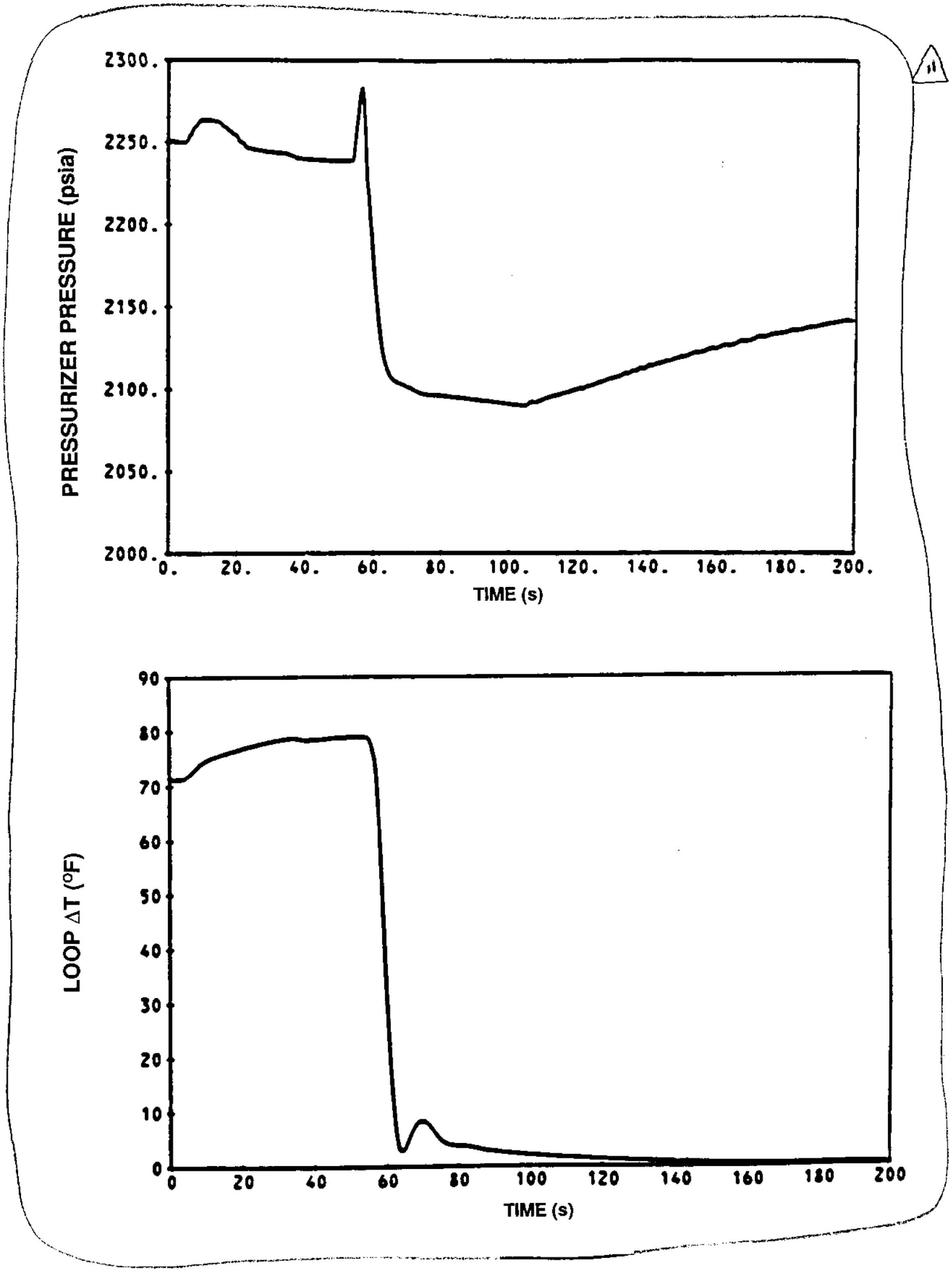
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FEEDWATER CONTROL VALVE MALFUNCTION-
NUCLEAER POWER AND CORE HEAT FLUX
VERSUS TIME

FIGURE 15.2-28 (SHEET 1 OF 3)



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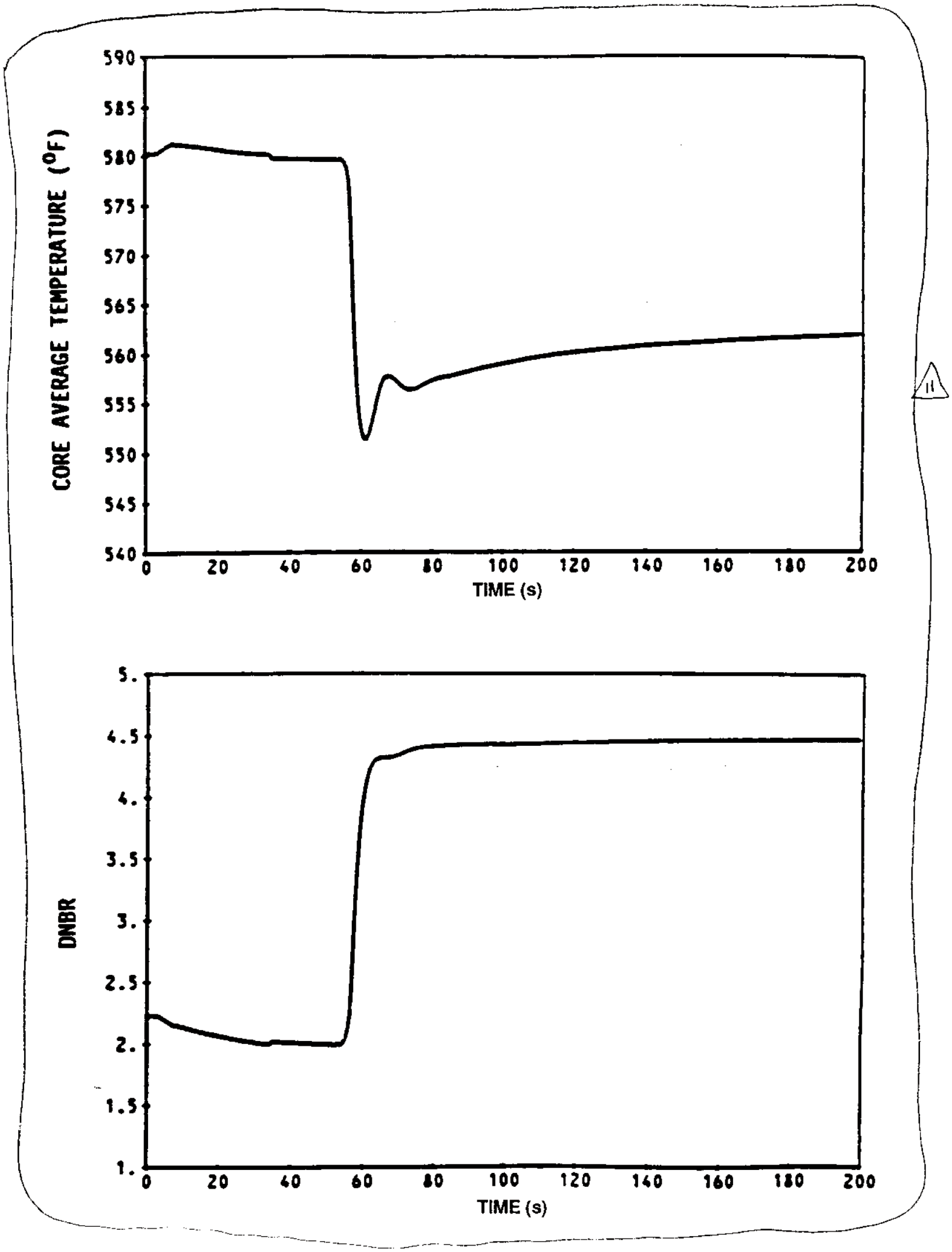
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FEEDWATER CONTROL VALVE
MALFUNCTION-PRESSURIZER PRESSURE
AND LOOP ΔT VERSUS TIME

FIGURE 15.2-28 (SHEET 2 OF 3)



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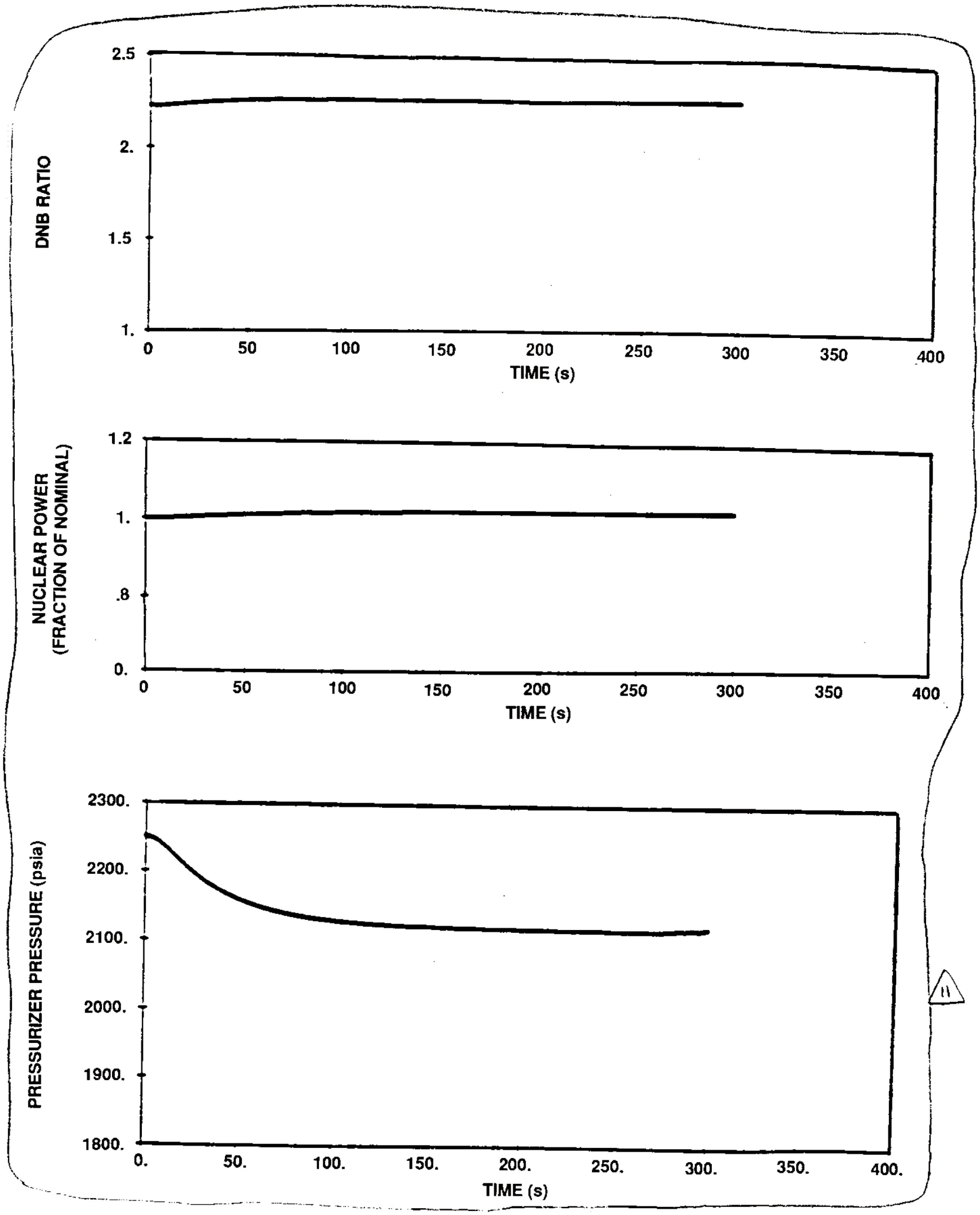
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FEEDWATER CONTROL VALVE MALFUNCTION-
CORE AVERAGE TEMPERATURE AND
DNBR VERSUS TIME

FIGURE 15.2-28 (SHEET 3 OF 3)



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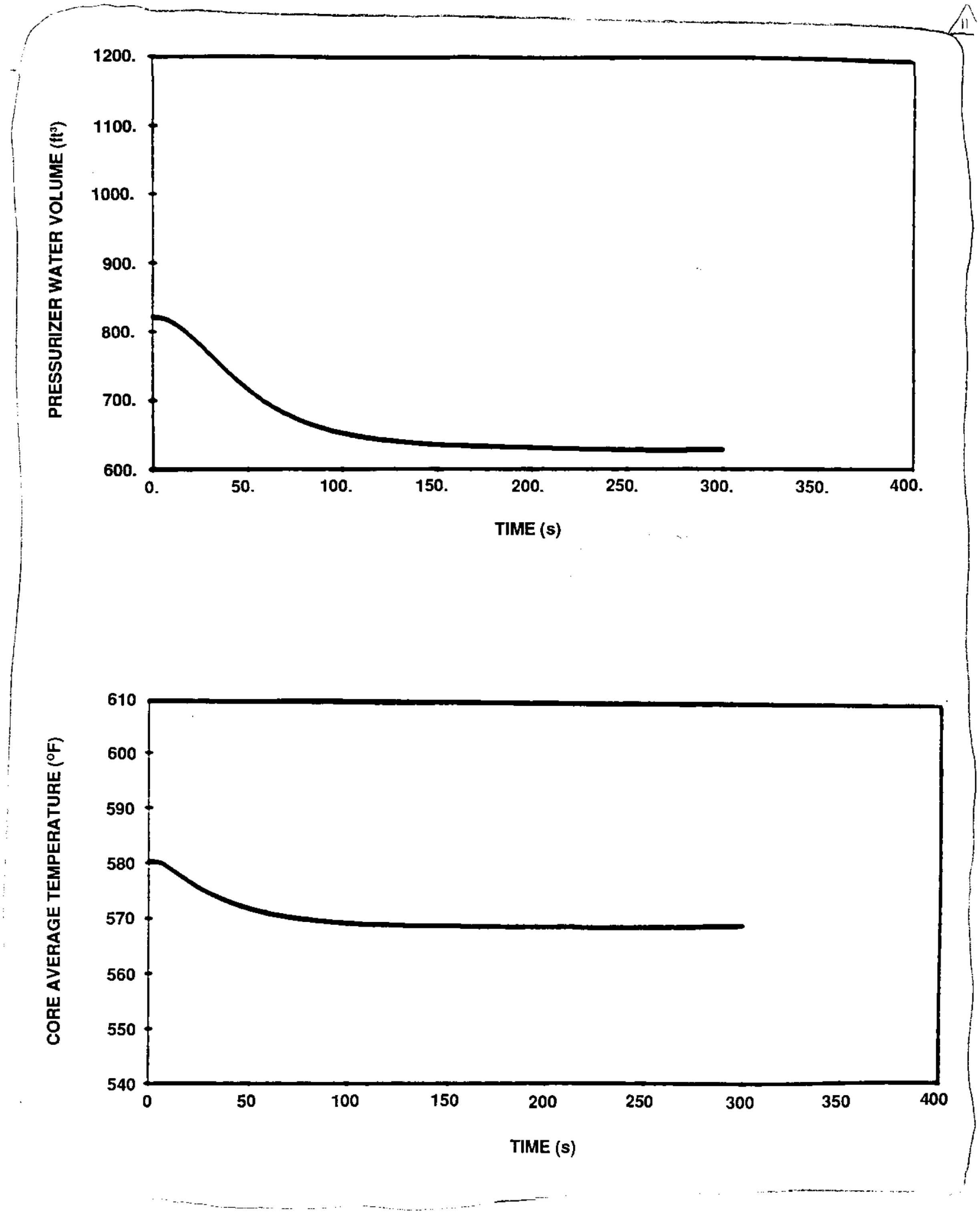
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION – BOL – DNBR, NUCLEAR POWER, AND PRESSURIZER PRESURE AS A FUNCTION OF TIME

FIGURE 15.2-29



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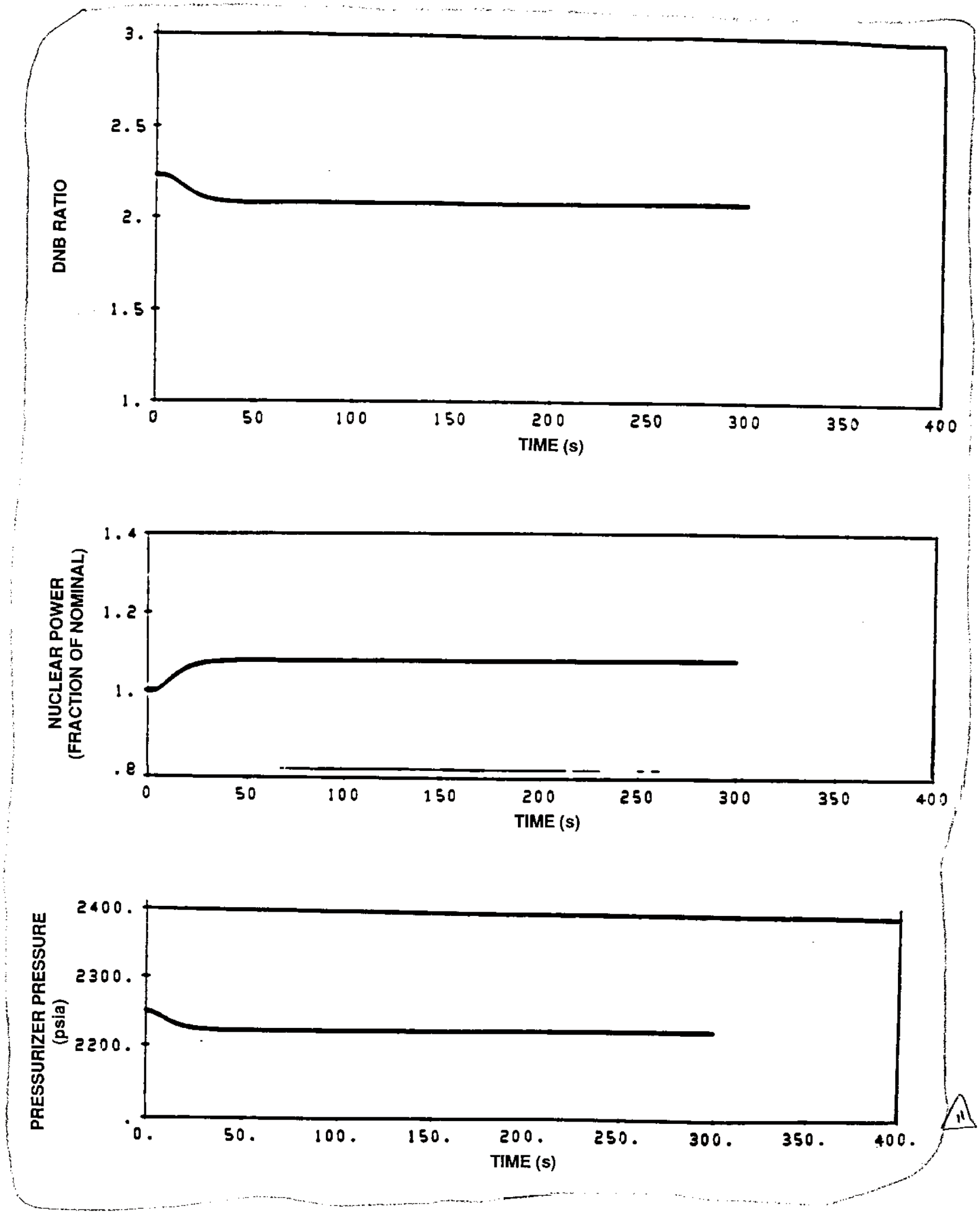
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION – BOL – T_{AVG} AND PRESSURIZER VOLUME AS A FUNCTION OF TIME

FIGURE 15.2-30



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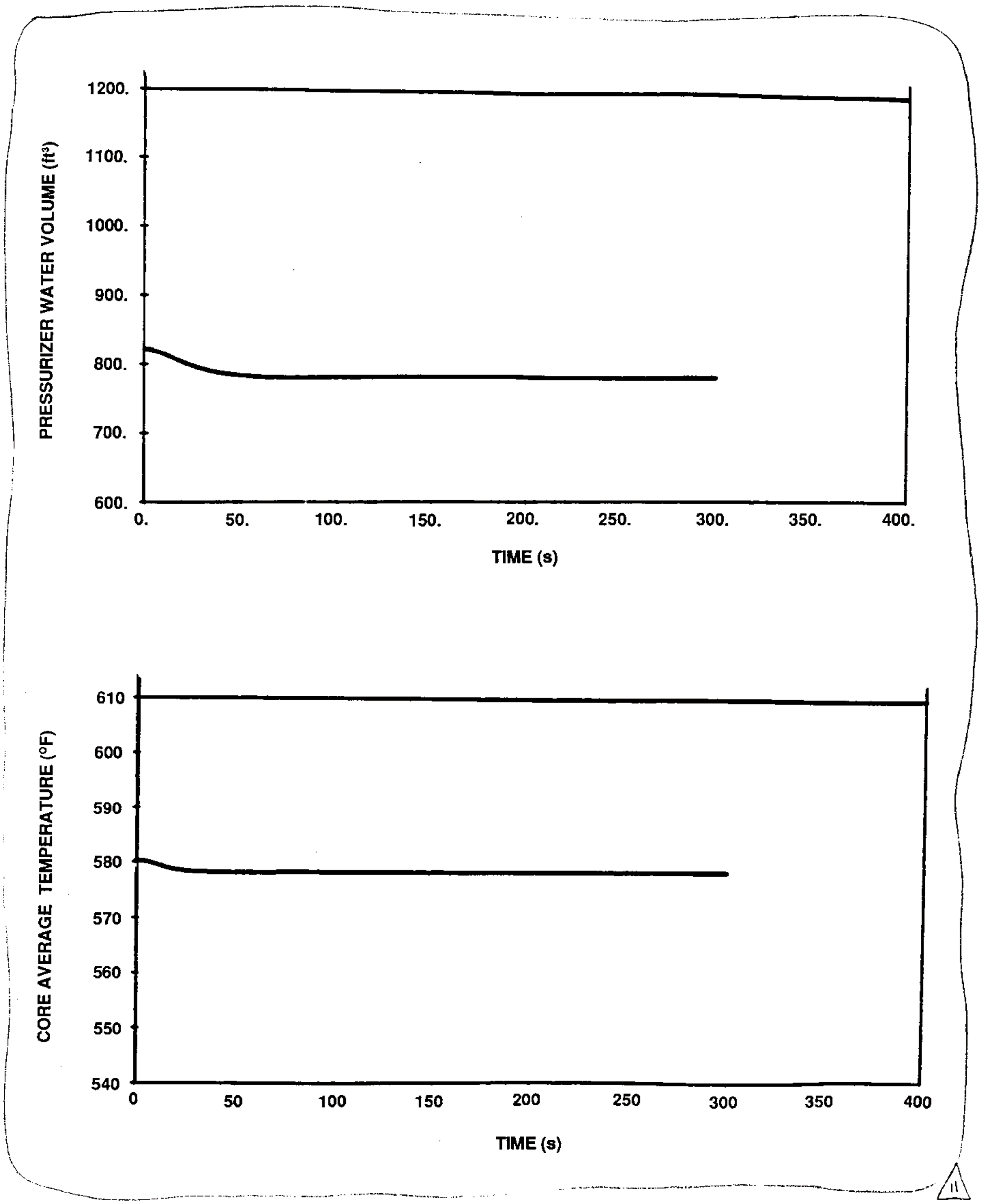
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION – EOL – DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME

FIGURE 15.2-31



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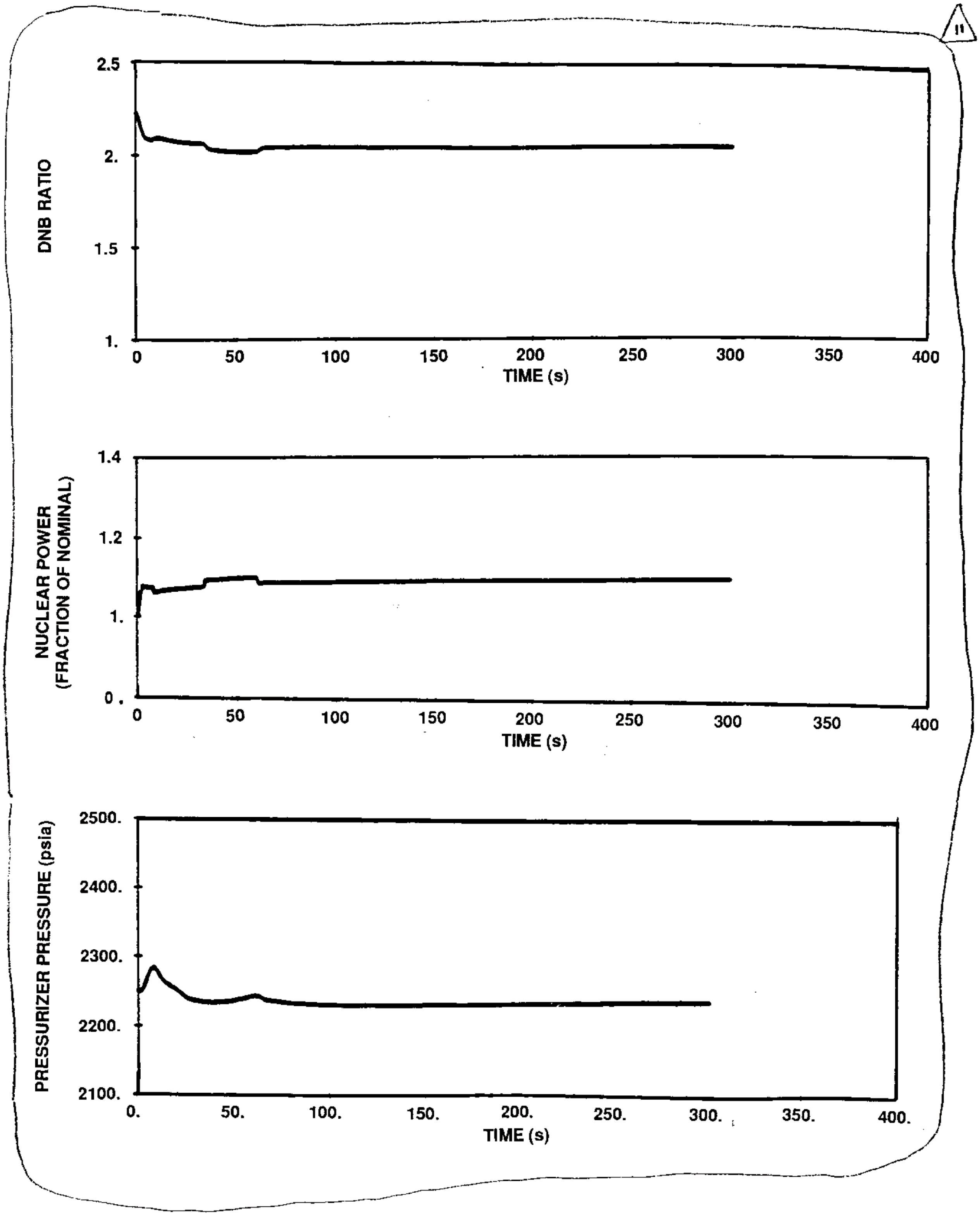
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION – EOL – T_{AVG} AND PRESSURIZER VOLUME AS A FUNCTION OF TIME FIGURE 15.2-32



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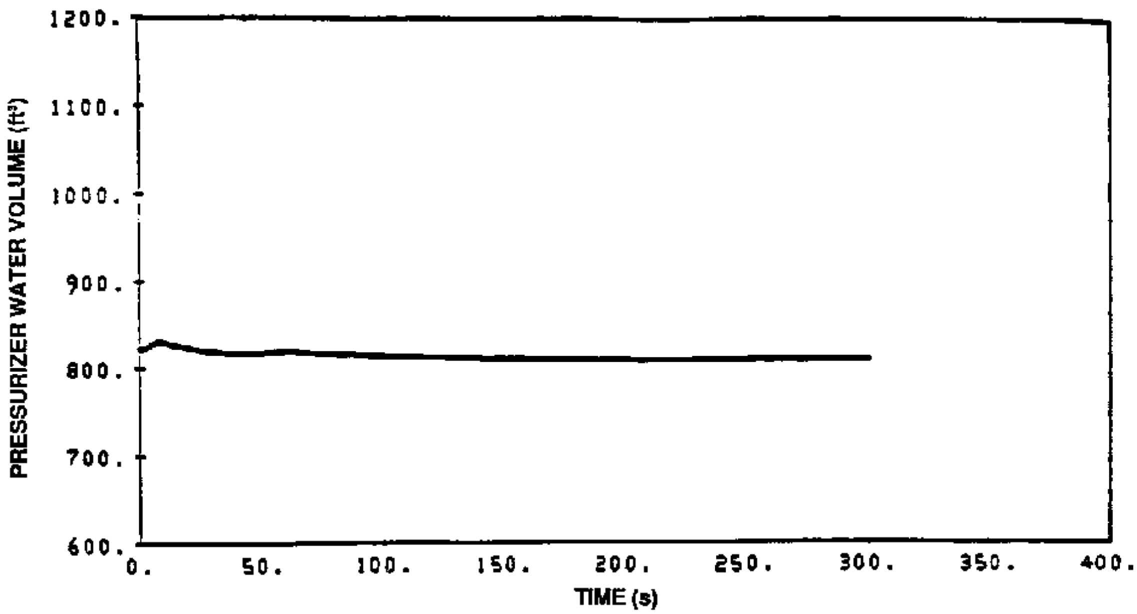
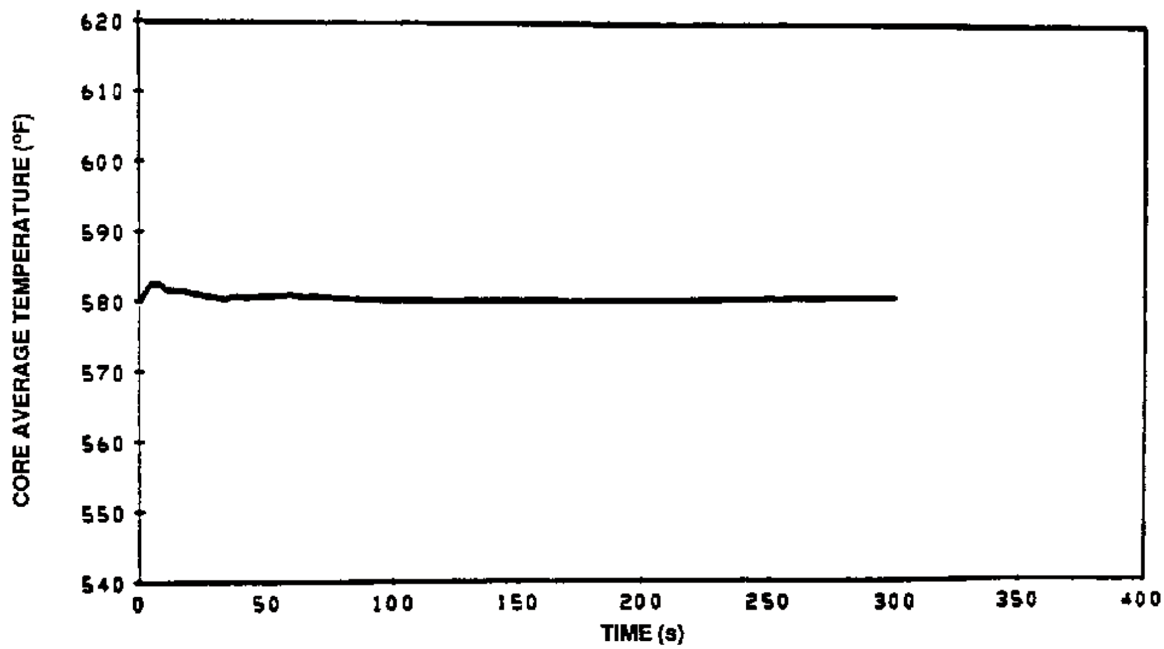
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL – BOL – DNBR, NUCLEAR POWER, AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME

FIGURE 15.2-33





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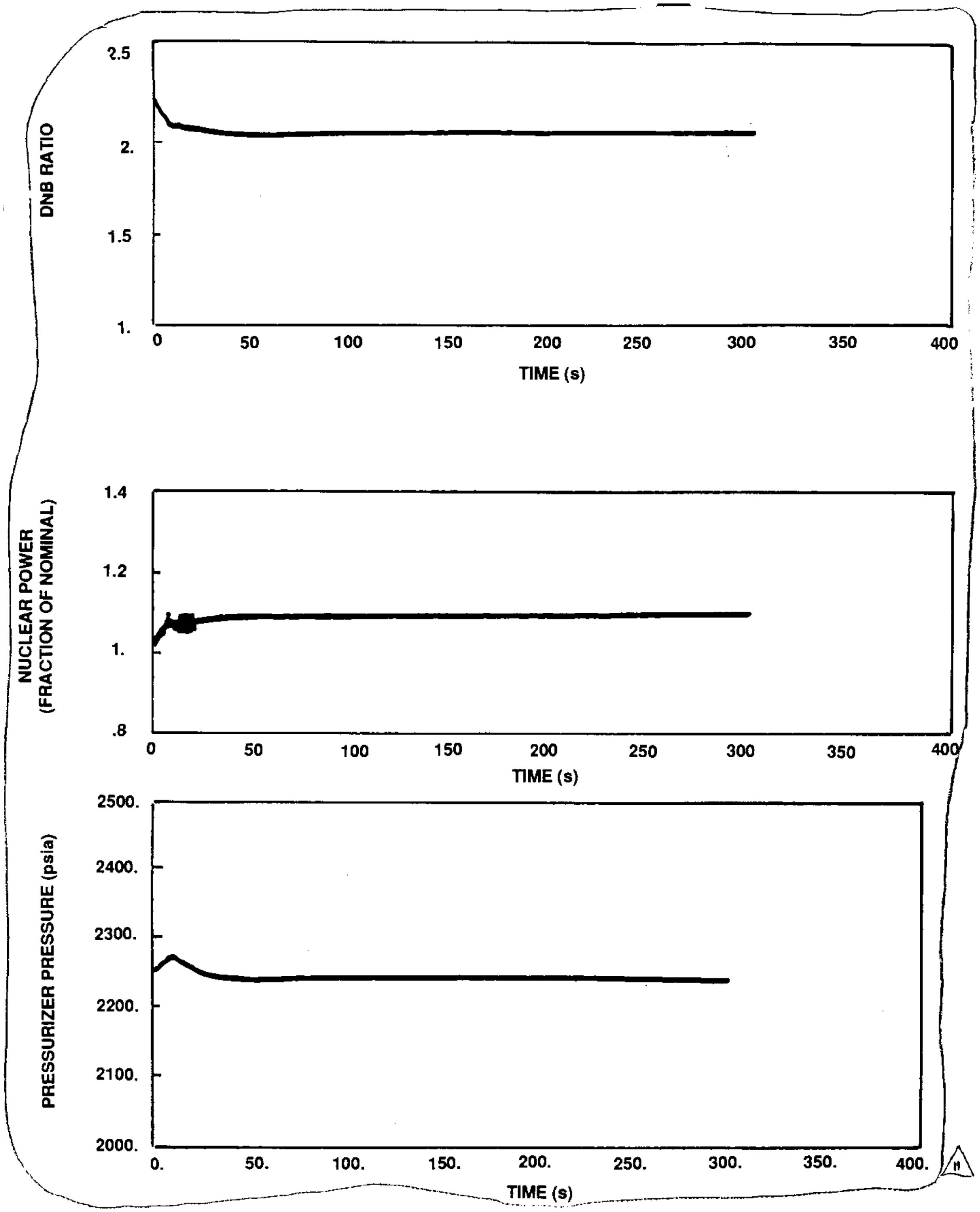
EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL – BOL – T_{AVG} AND PRESSURIZER VOLUME AS A FUNCTION OF TIME

FIGURE 15.2-34



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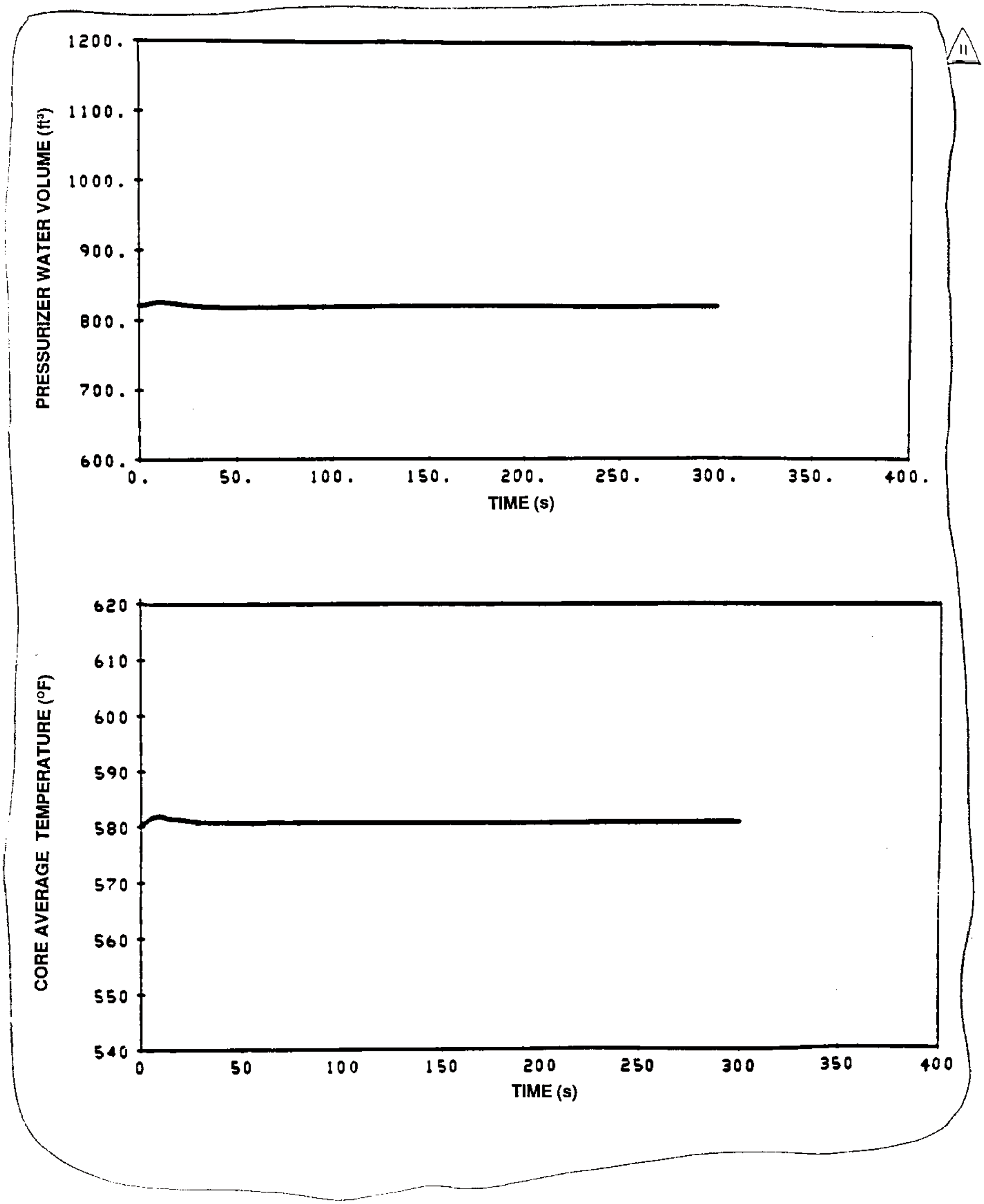
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL – EOL – DNBR, NUCLEAR POWER, AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME
		FIGURE 15.2-35



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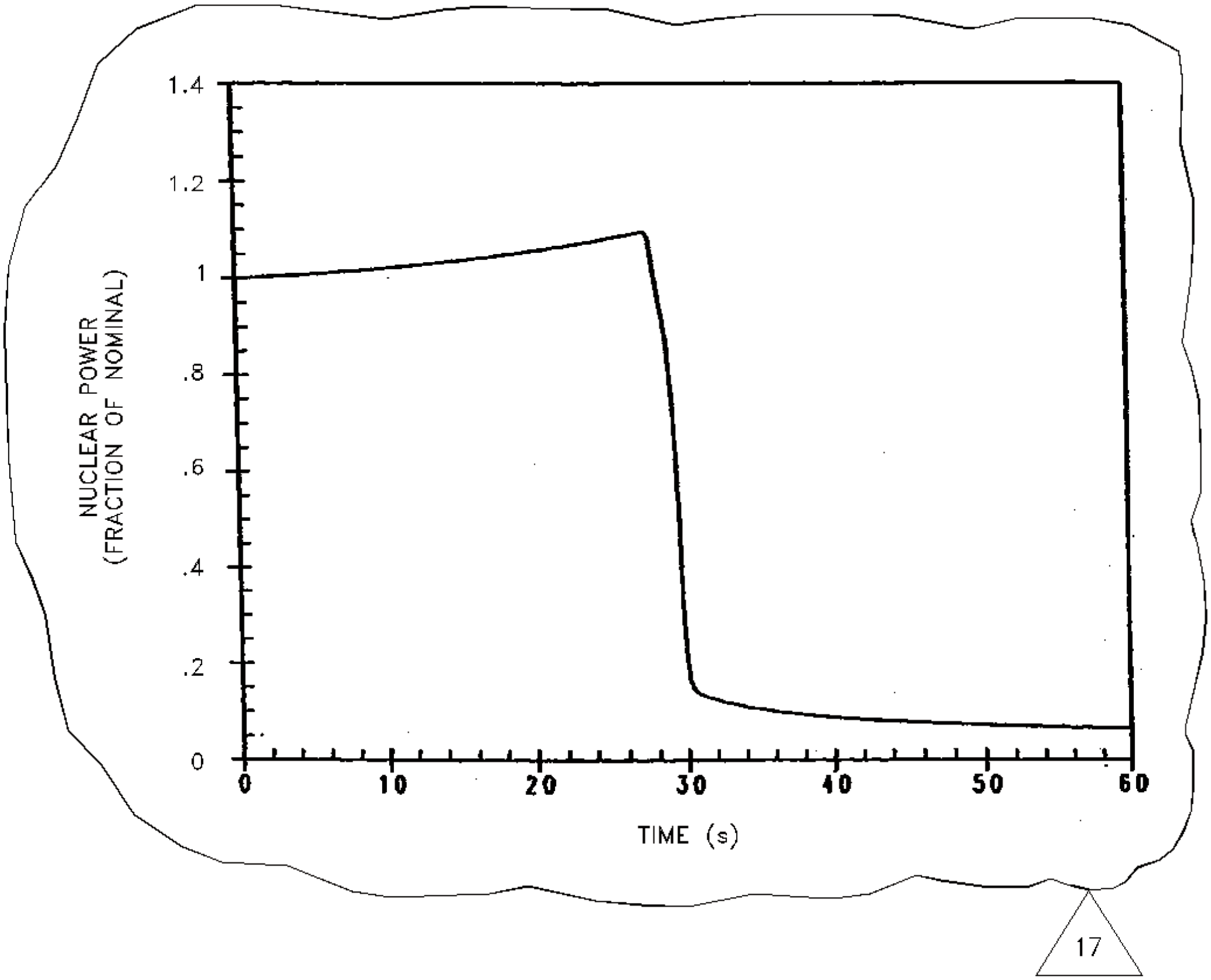
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL – EOL – T_{AVG} AND PRESSURIZER VOLUME AS A FUNCTION OF TIME

FIGURE 15.2-36



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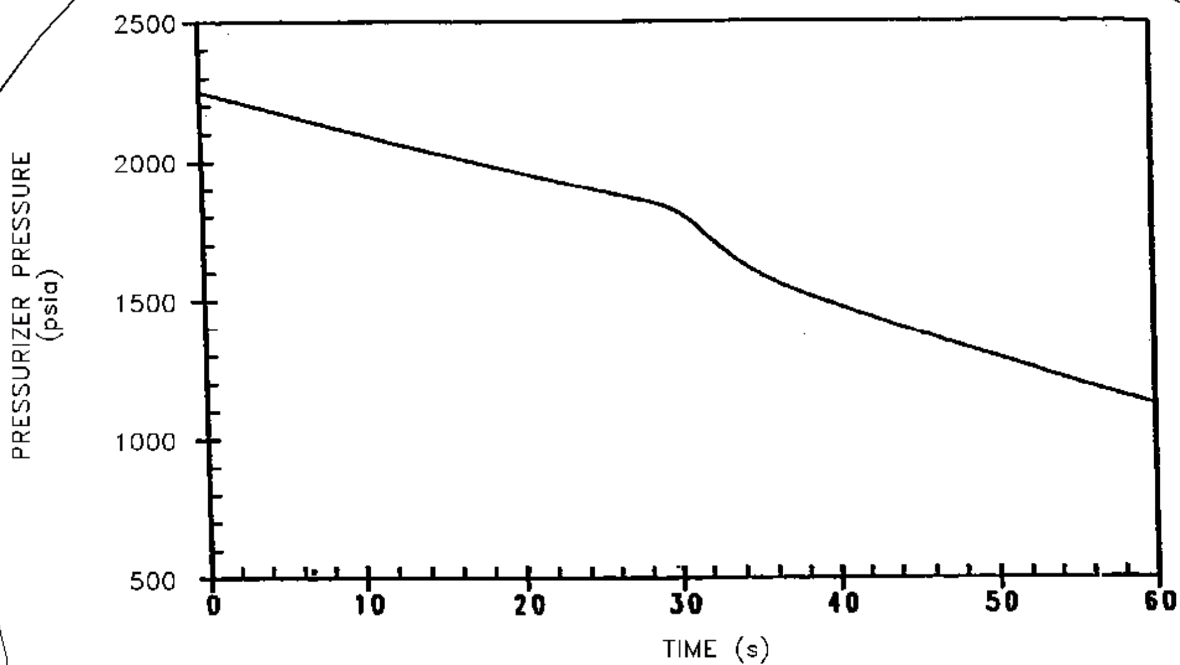
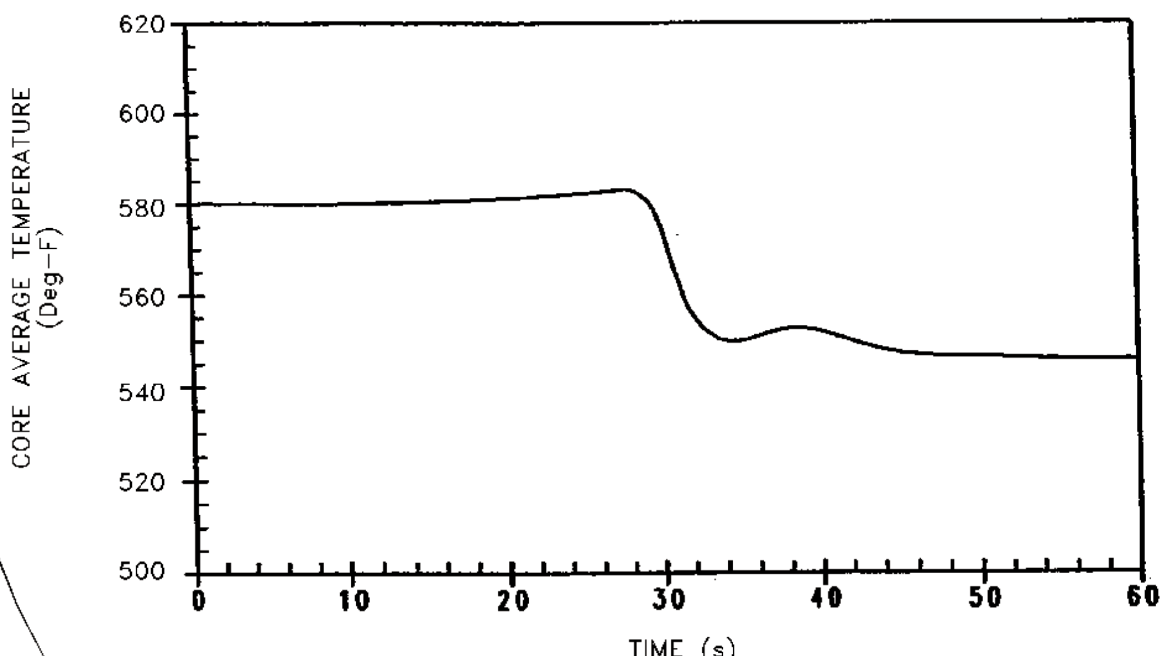
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ACCIDENTAL RCS DEPRESSURIZATION NUCLEAR POWER VERSUS TIME
		FIGURE 15.2-37





17

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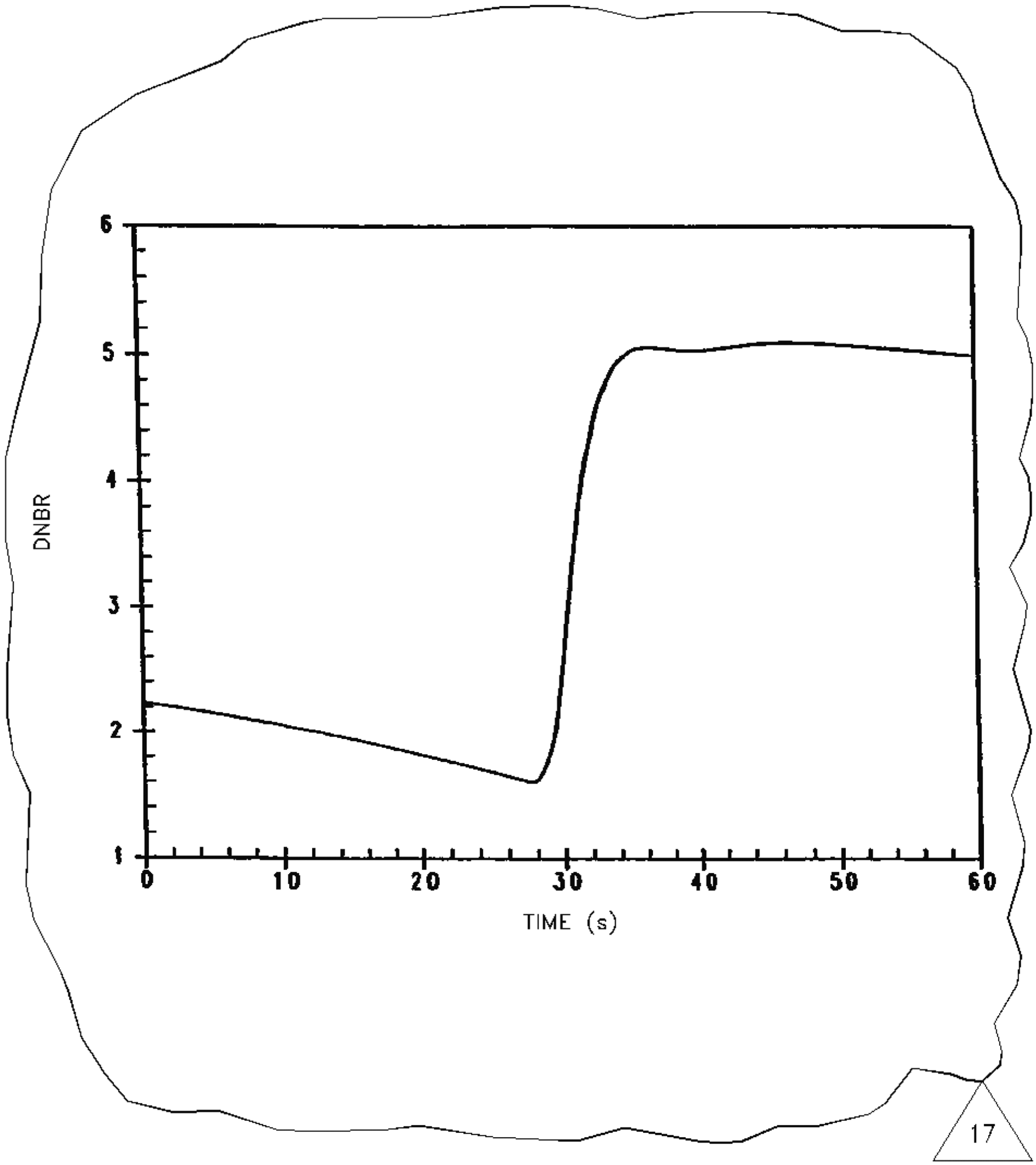
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ACCIDENTAL RCS DEPRESSURIZATION
T_{AVG} AND PRESSURIZER PRESSURE
VERSUS TIME

FIGURE 15.2-38



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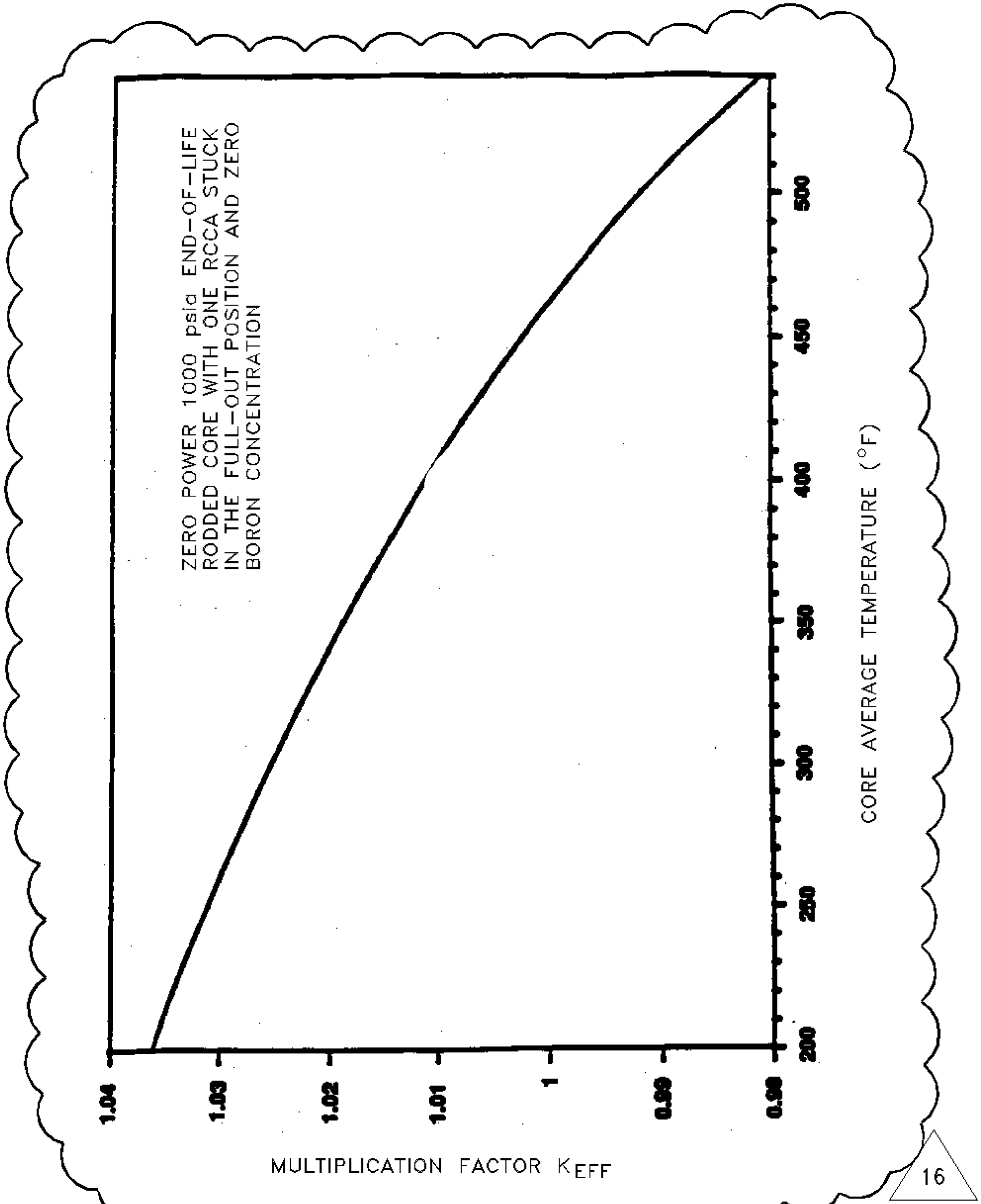
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ACCIDENTAL RCS DEPRESSURIZATION
DNBR VERSUS TIME

FIGURE 15.2-39



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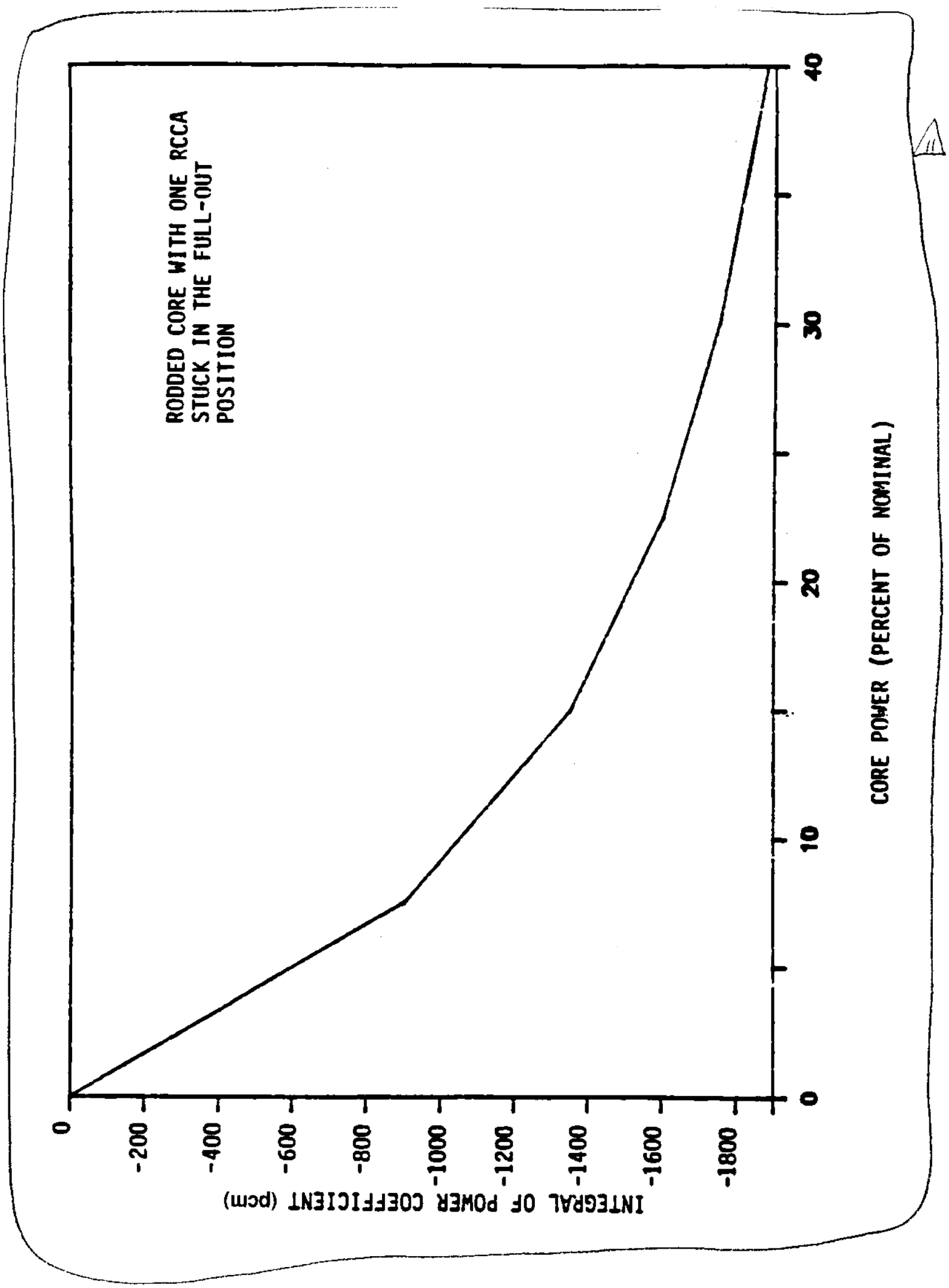
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	VARIATION OF K_{EFF} WITH CORE TEMPERATURE FIGURE 15.2-40 (SHEET 1 OF 2)



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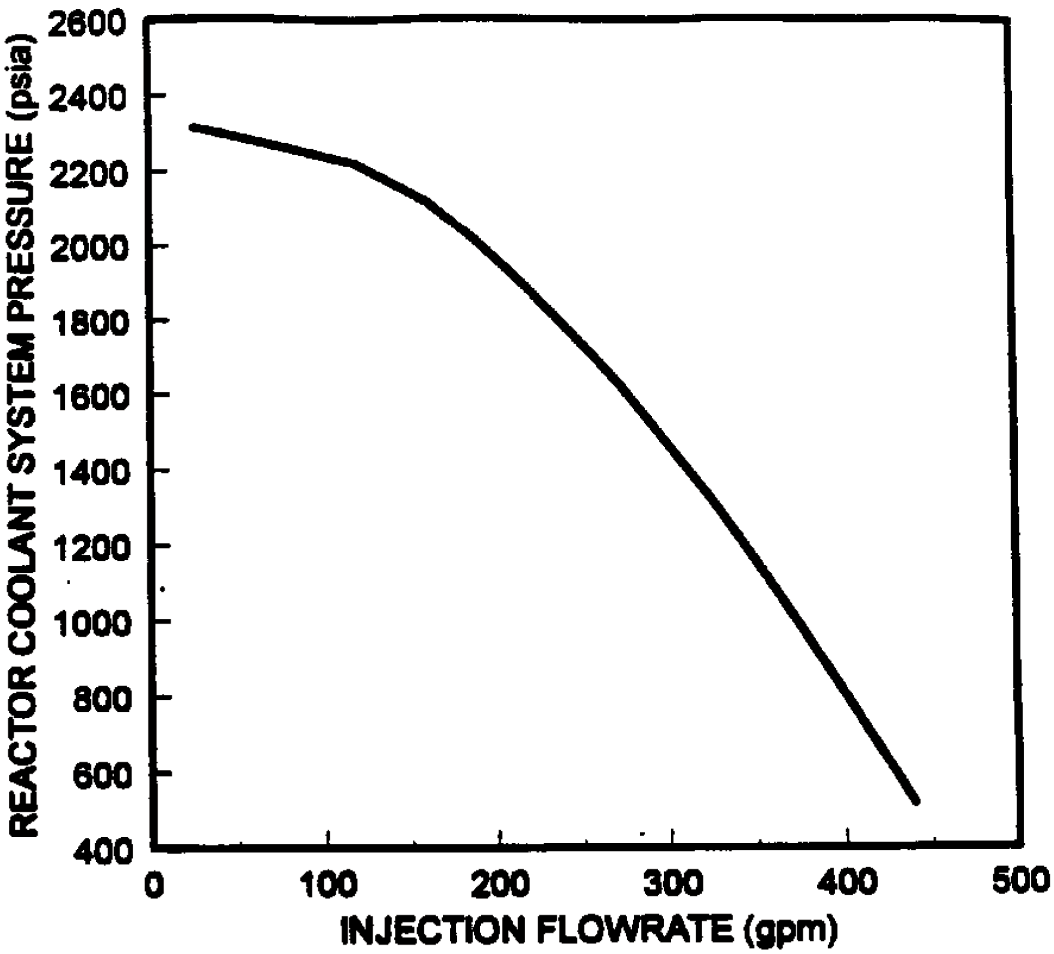
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DOPPLER POWER FEEDBACK

FIGURE 15.2-40 (SHEET 2 OF 2)



15

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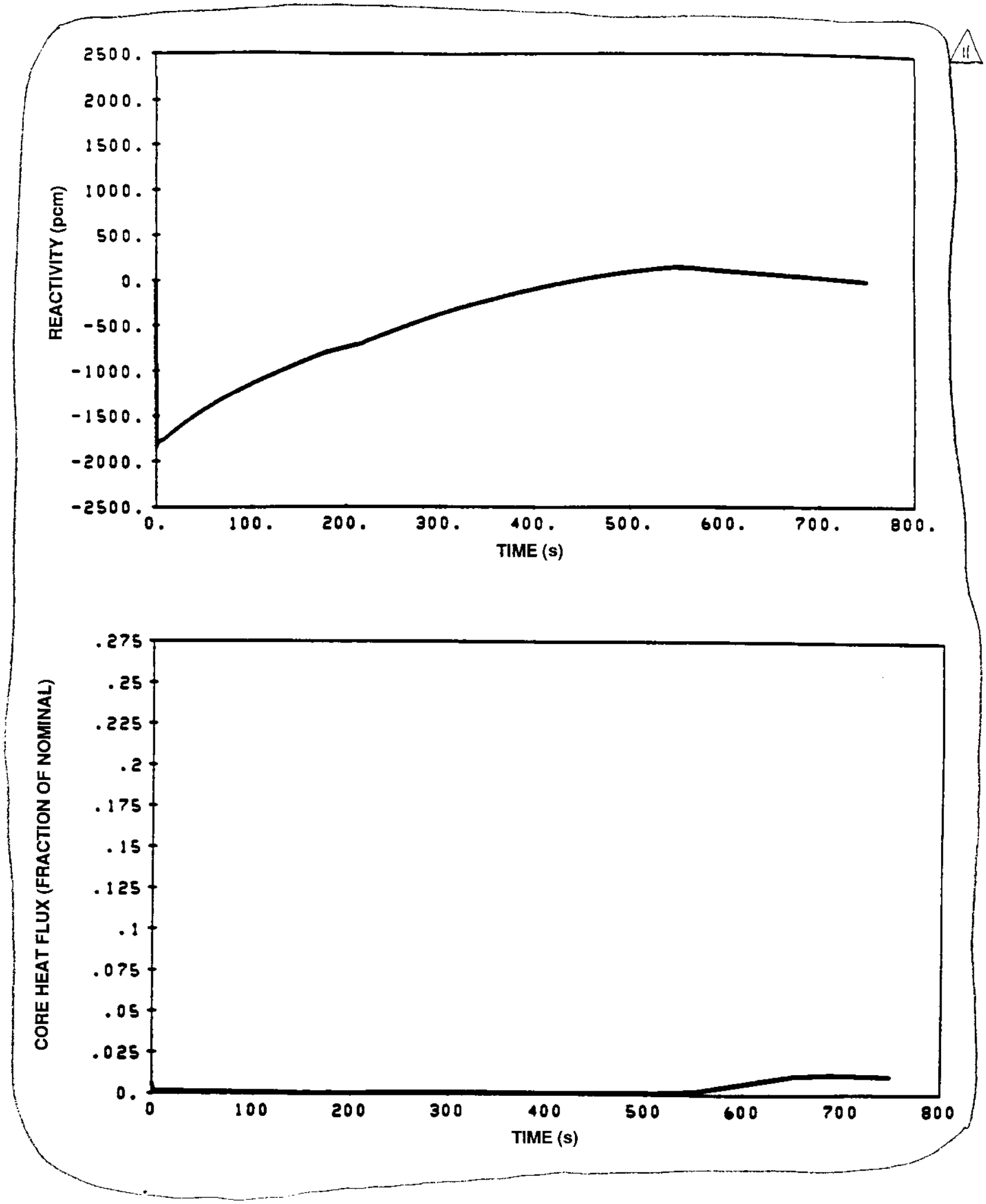
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

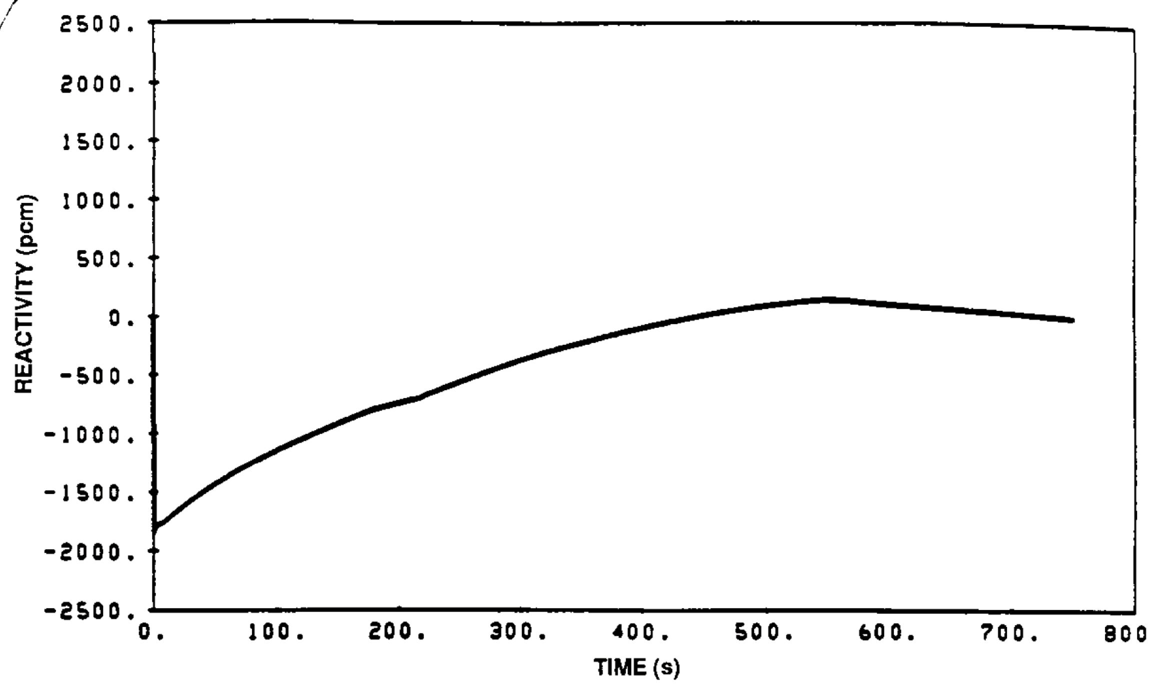
SAFETY INJECTION FLOWRATE
VS. RCS PRESSURE

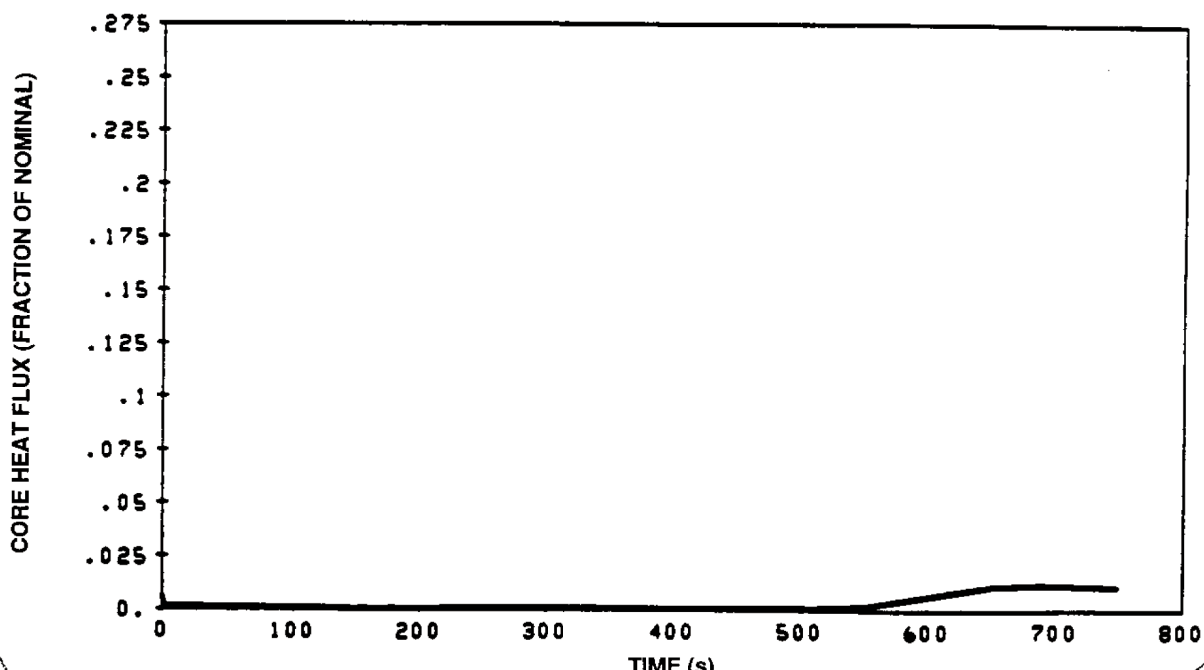
FIGURE 15.2-41



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM LINE BREAK TRANSIENT CREDIBLE BREAK, FAILED OPEN VALVE FIGURE 15.2-42 (SHEET 1 OF 3)





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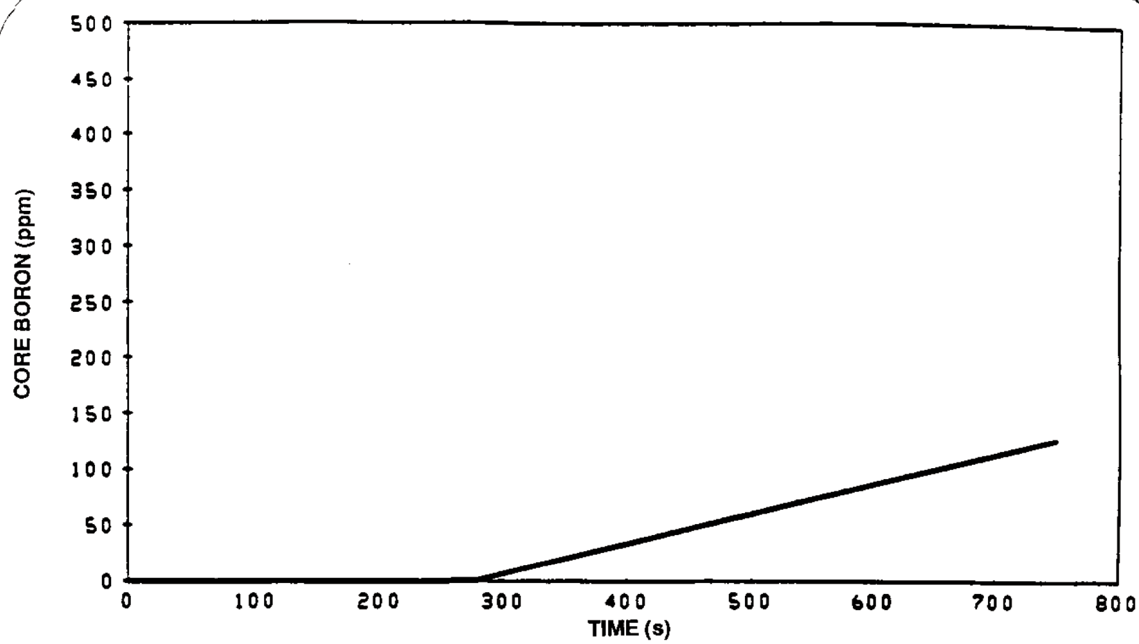
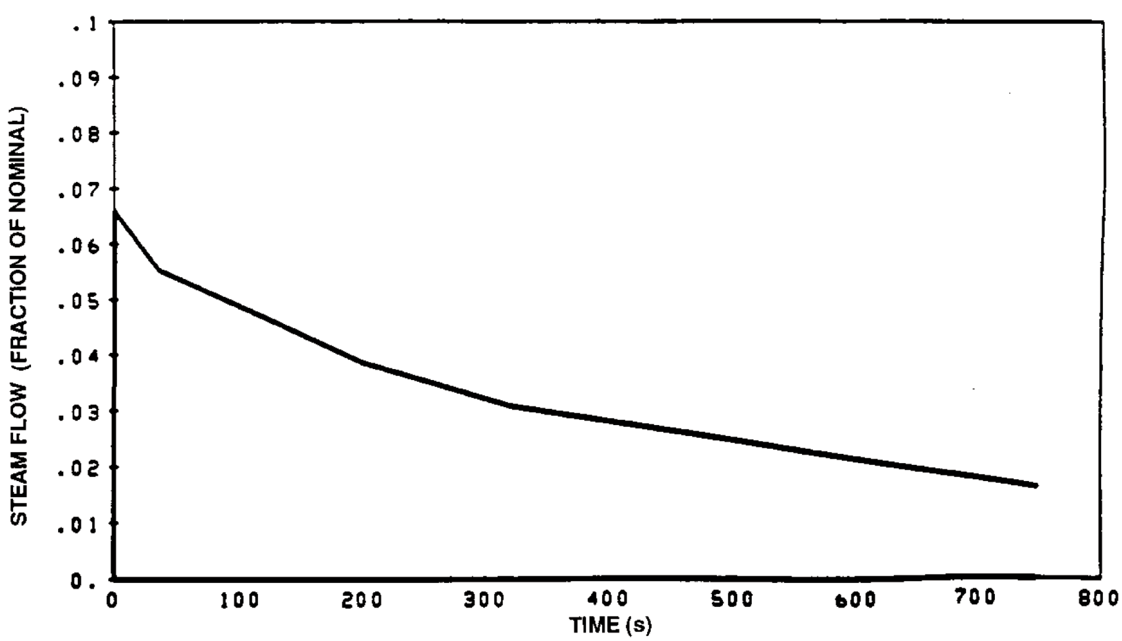
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT CREDIBLE BREAK, FAILED OPEN VALVE

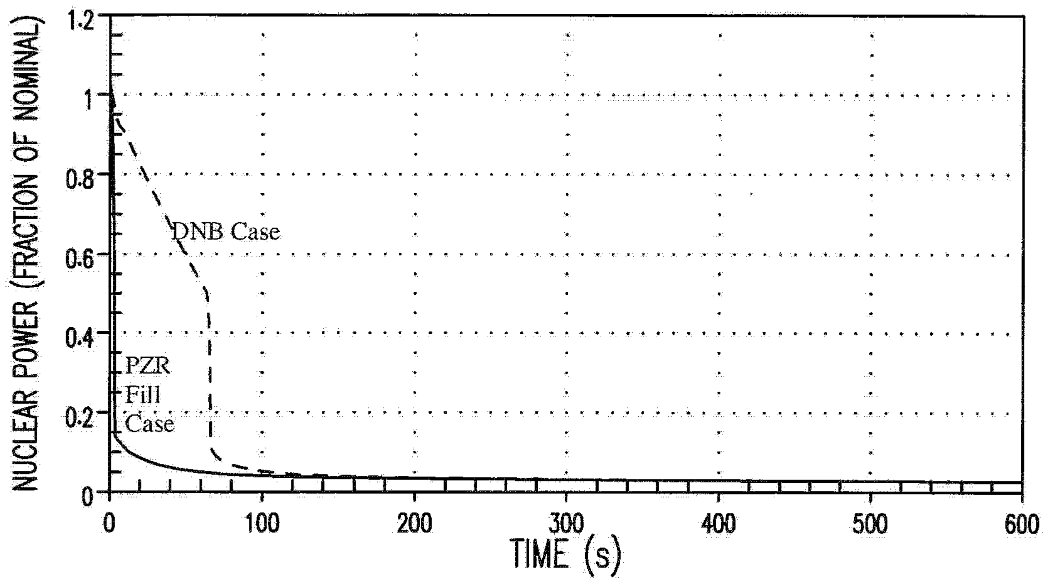
FIGURE 15.2-42 (SHEET 2 OF 3)

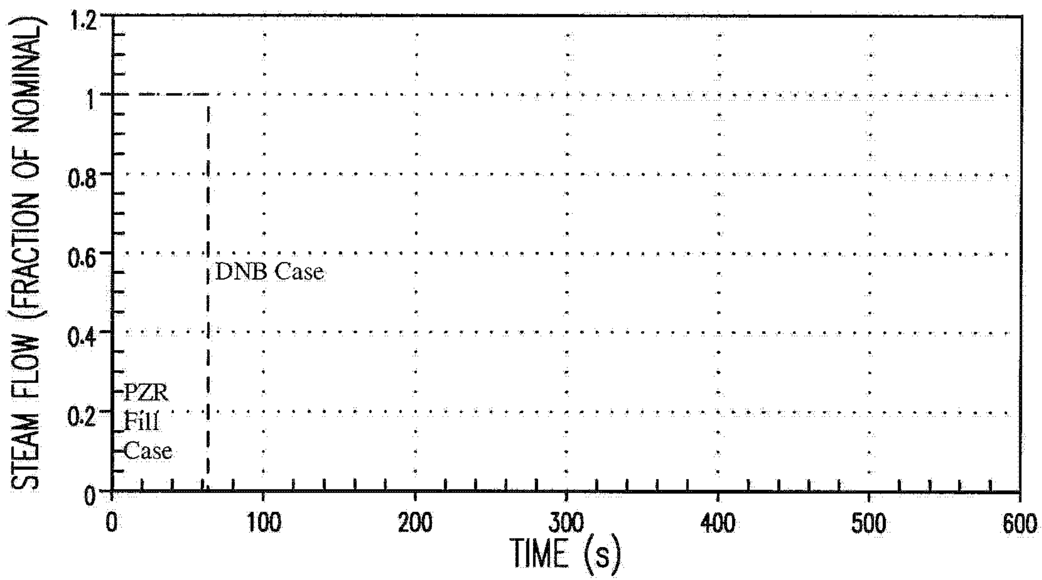




REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM LINE BREAK TRANSIENT CREDIBLE BREAK, FAILED OPEN VALVE FIGURE 15.2-42 (SHEET 3 OF 3)



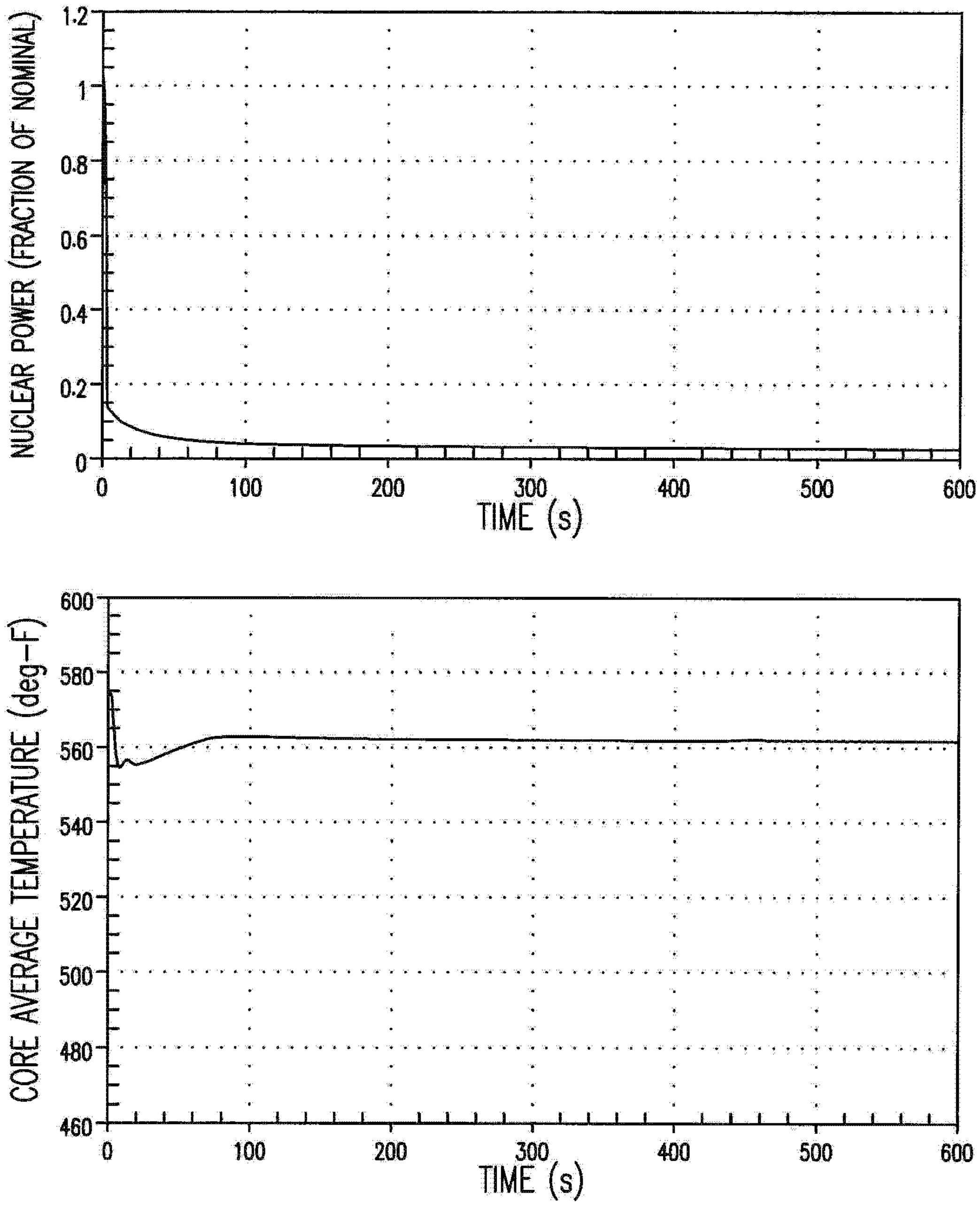


REV 22 8/09

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SPURIOUS ACTUATION OF THE SI SYSTEM
NUCLEAR POWER AND STEAM FLOW
VERSUS TIME

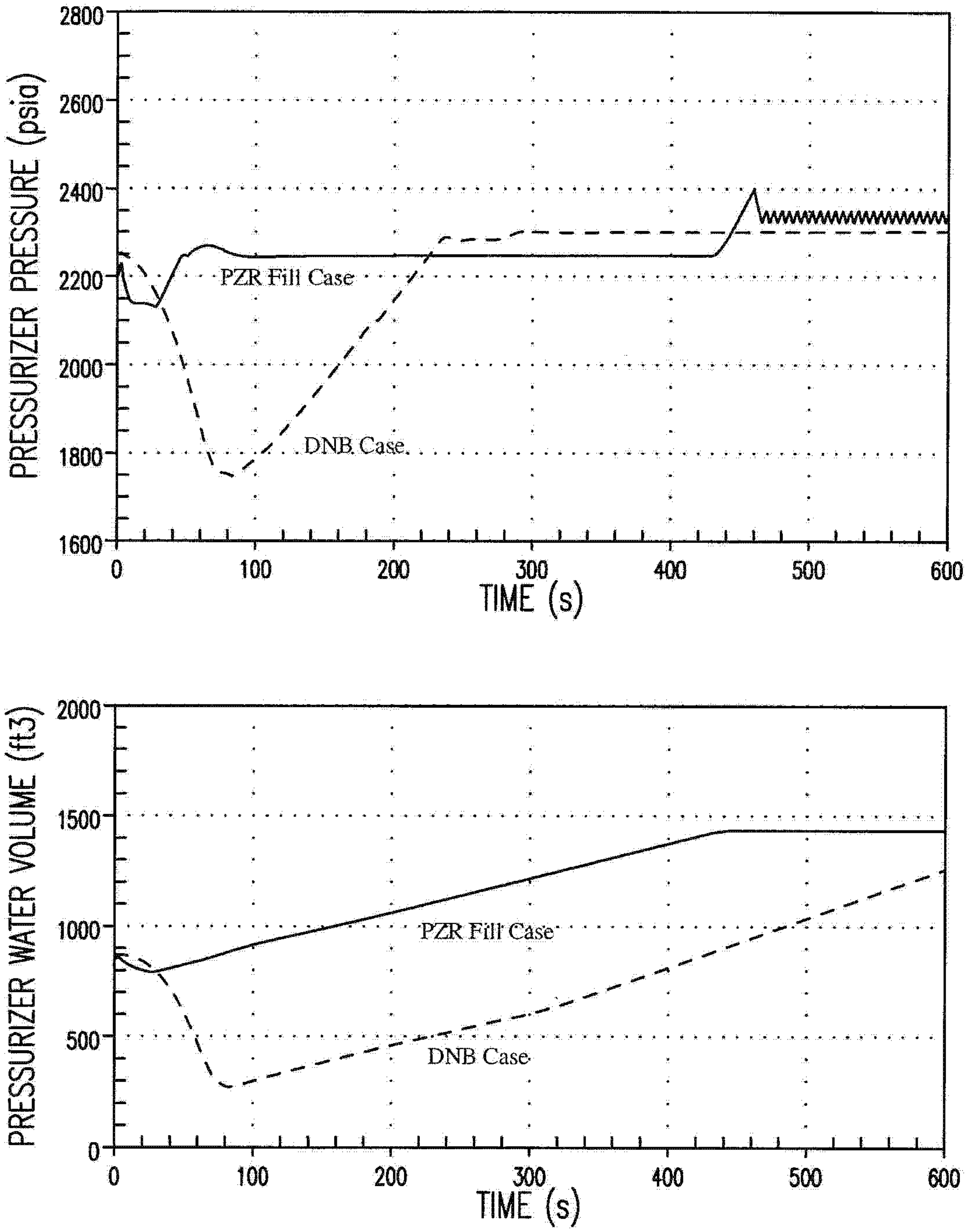
FIGURE 15.2-43



REV 22 8/09

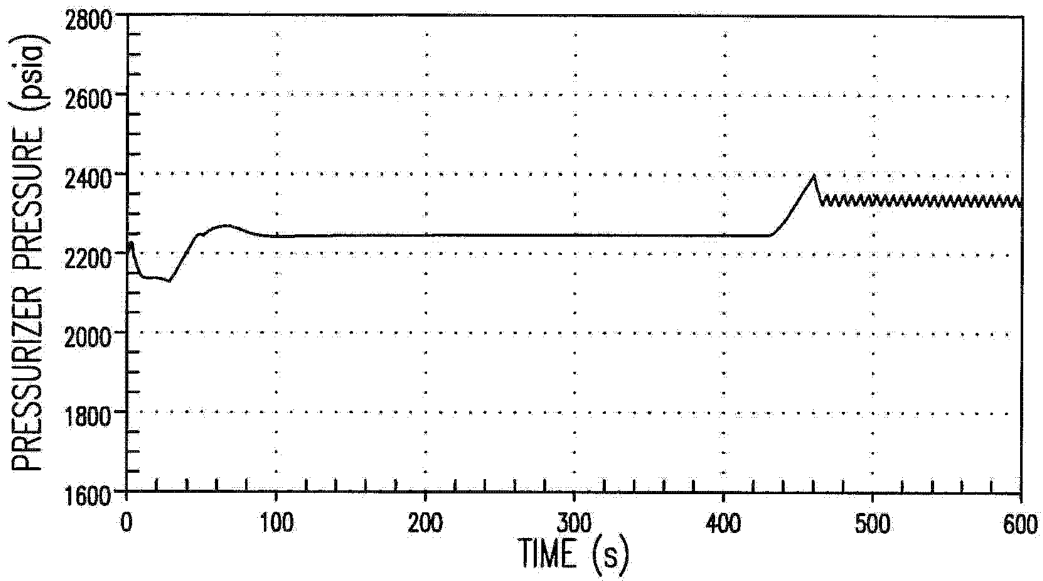
SPURIOUS ACTUATION OF THE SI SYSTEM
NUCLEAR POWER AND CORE AVERAGE TEMPERATURE
VERSUS TIME (PRESSURIZER FILL CASE)

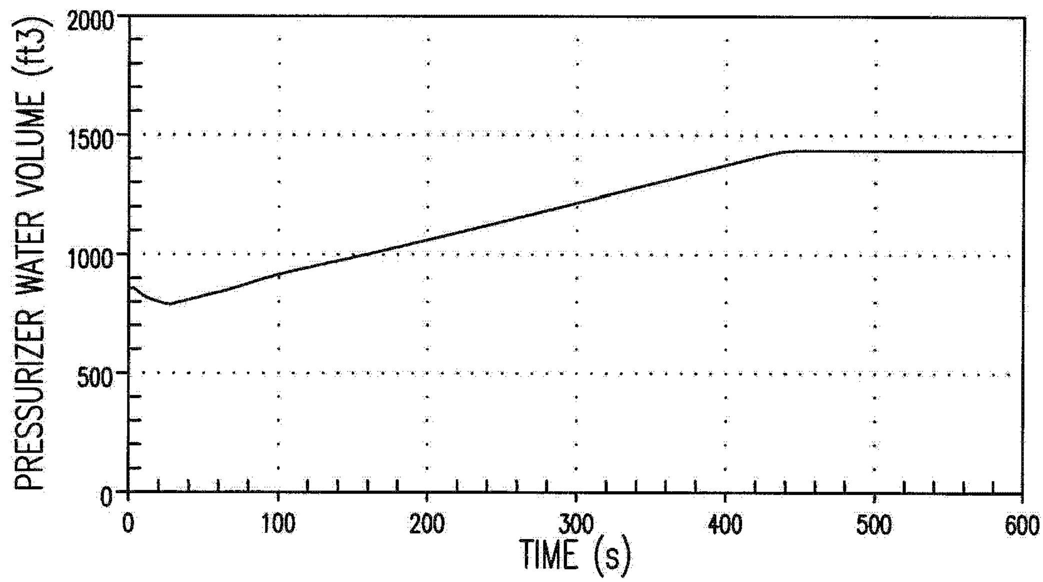
FIGURE 15.2-43A



REV 22 8/09

SOUTHERN COMPANY	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SPURIOUS ACTUATION OF THE SI SYSTEM PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME
		FIGURE 15.2-44





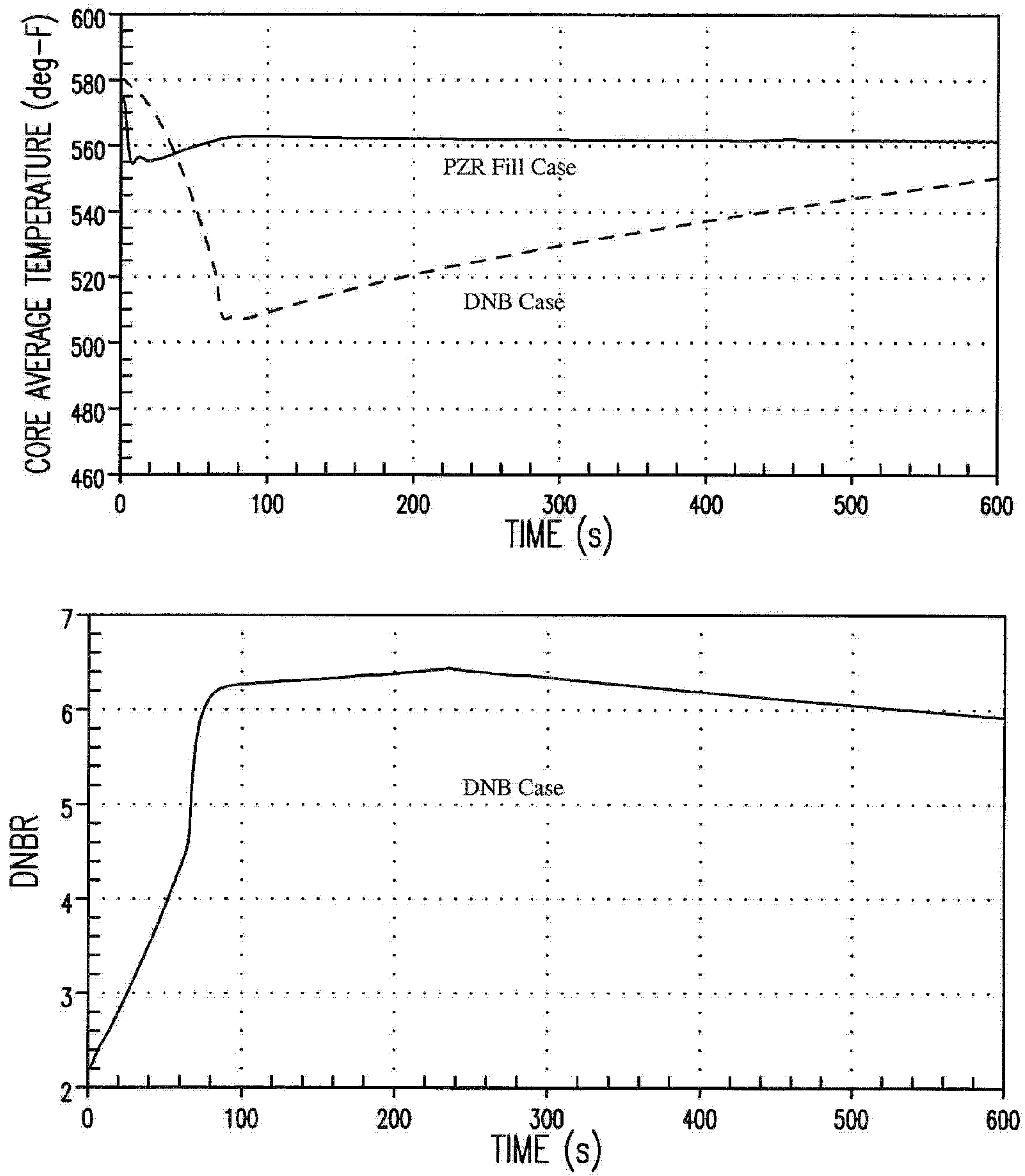
REV 22 8/09

SOUTHERN COMPANY

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SPURIOUS ACTUATION OF THE SI SYSTEM
PRESSURIZER PRESSURE AND PRESSURIZER WATER
VOLUME VERSUS TIME (PRESSURIZER FILL CASE)

FIGURE 15.2-44A



REV 22 8/09

SOUTHERN COMPANY	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SPURIOUS ACTUATION OF THE SI SYSTEM CORE T_{avg} AND DNBR VERSUS TIME
		FIGURE 15.2-45

THIS FIGURE HAS BEEN DELETED.

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SPURIOUS ACTUATION OF SAFETY INJECTION SYSTEM AT ZERO POWER
		FIGURE 15.2-46

15.3 CONDITION III – INFREQUENT INCIDENTS

By definition, Condition III occurrences are faults which may occur very infrequently during the life of the plant. The NRC acceptance criteria for Condition III events are that (1) only a small fraction of the fuel rods will fail, although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time and (2) the release of radioactivity will not be sufficient to interrupt or restrict public use of these areas beyond the exclusion radius (i.e., 10 CFR 100 limits are not exceeded). A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system (RCS) or containment barriers. The latter acceptance criteria require, in part, maintaining the RCS $\leq$ 2750 psia and containment leak rate $\leq$ 0.15% per day, respectively. For the purposes of this report, the following faults have been grouped into this category:

A. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuate emergency core cooling system (ECCS).

B. Minor secondary system pipe break.

C. Inadvertent loading of a fuel assembly into an improper position.

D. Complete loss of forced reactor coolant flow.

E. Waste gas decay tank rupture.

F. Single rod cluster control assembly (RCCA) withdrawal at full power.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATE EMERGENCY CORE COOLING SYSTEM

This section presents results of the small break loss-of-coolant accident (LOCA) which are in conformance with the NRC acceptance criteria found in 10 CFR 50.46 (reference 1) and Appendix K of 10 CFR 50.

15.3.1.1 Identification of Causes and Accident Description

A LOCA is defined as a rupture of the RCS piping or of any line connected to the system. Ruptures of small cross-sections will cause expulsion of the coolant at a rate which can be accommodated by the high-head SI pumps and which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing in it.

The maximum break size for which the normal makeup system can maintain the pressurizer level can be obtained by comparing the calculated flow from the RCS through the postulated break against the high-head SI makeup flow at normal RCS pressure; i.e., 2250 psia.

A small break, as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area < 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure.

For small break LOCAs, the most limiting single active failure is the one that results in the minimum ECCS flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit can be taken for only one high-head SI pump and one residual heat removal (RHR) (low-head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one for each loop). For the 2-in., 2.25-in., 2.5-in., 2.75-in., 3-in., 3.25-in., and 4-in. small break LOCA analysis cases, the broken loop injection line is assumed to spill to RCS backpressure. For the 6-in. small break LOCA analysis case, which has the break larger than the SI line (inner diameter = 5.189 in.), the broken loop injection line is assumed to spill to containment backpressure.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer, resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. Loss-of-offsite-power, including a loss of ac power to the station auxiliaries (LOOP), is assumed to occur coincident with reactor trip on the affected unit. A safety injection (SI) signal is generated when the pressurizer low pressure SI setpoint is reached. After the SI setpoint is reached, an additional 27-s delay ensues. This delay conservatively models the 2-s instrumentation delay, the full 15-s diesel generator start time, plus the up to 10 s necessary to align the appropriate valves and increase the pumps to full speed (diesel generator start on the SI signal versus LOOP is a conservative modeling assumption). These countermeasures will limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water; however, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of RCCAs subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full-out position.

B. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is assumed to be in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps (RCPs) through the core as the pumps coast down following LOOP. Upward flow through the core is maintained; however, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continue to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In this case, continued heat addition to the secondary results in increased secondary system pressure which leads to steam relief via the atmospheric relief valve and/or safety valves. Makeup to the secondary is automatically provided by the auxiliary feedwater (AFW) pumps. The SI signal isolates normal feedwater flow by closing the main feedwater control and bypass valves. LOOP, assumed concurrent with reactor trip, initiates AFW flow by starting the AFW pumps. The secondary flow aids in the reduction of RCS pressure. However, this analysis conservatively models AFW delivery 60 s following SI. Also due to the LOOP assumption, the RCPs are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops; however, the vessel mixture level starts to increase to cover the fuel with ECCS pumped injection before the accumulator injection for most breaks.

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

For small breaks (< 1.0 ft^2) the NOTRUMP digital computer code (references 2, 3, and 13) is employed to calculate the transient depressurization of the RCS as well as to describe the mass and energy of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Also, SI into the broken loop is modeled using the COSI condensation model (reference 13). The NOTRUMP small break LOCA ECCS evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address NRC concerns expressed in NUREG-0611 (reference 4), "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

The RCS model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enable a proper calculation of the behavior of the loop seal during a LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in references 2, 3, and 13.

Peak clad temperature calculations are performed with the LOCTA-IV code (reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow, and mixture heights as boundary conditions (see figure 15.3-1). Figure 15.3-2 depicts the hot rod axial power shape used to perform the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small break LOCAs because it minimizes coolant level swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes the core continues to operate at full power until the control rods are completely inserted; however, for conservatism, it is assumed that the most reactive RCCA does not insert.

After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer pressure signal (1840 psia). Soon after the reactor trip signal is generated, the SI actuation signal is generated due to a low pressurizer pressure (1700 psia). SI systems consist of gas pressurized accumulator tanks and pumped injection systems. The small break LOCA analysis assumed nominal accumulator water volume with a cover gas pressure of 600 psia (the minimum pressure allowed by the Technical Specifications). Minimum ECCS availability is assumed for the analysis at the maximum reactor water storage tank (RWST) temperature. Assumed pumped SI characteristics as a function of RCS pressure used as boundary conditions in the analysis are shown in figure 15.3-3A and table 15.3-5A (2-in., 2.25-in., 2.5-in., 2.75-in., 3-in., 3.25-in., and 4-in. cases) and figure 15.3-3B and table 15.3-5B (6-in. case). The SI flowrates presented are based on pump performance curves degraded 10 percent from the design head and an assumed charging system branch line cold leg imbalance of 20 (-5, +15) gal/min. The effect of flow from the RHR pumps is not considered in the 2-in., 2.25-in., 2.5-in., 2.75-in., 3-in., 3.25-in., and 4-in. cases since their shutoff head is lower than the RCS pressure during the time portion of the transients considered here. SI is delayed 27 s after the occurrence of the low pressure condition. This accounts for signal initiation (2 s), diesel generator startup, and emergency power bus loading consistent with the assumed LOOP with reactor trip (15 s), as well as the delay involved in aligning the valves and bringing the pumps up to speed (10 s). The small break LOCA analysis also assumed that the rod drop time is 2.7 s.

On the secondary side, a main feedwater isolation signal is conservatively assumed to be generated in conjunction with a reactor trip with a 2-s signal delay and a 5-s valve closure time. (At Farley, feedwater isolation is initiated by the SI signal.) The AFW pumps are assumed to start and deliver full flow (one turbine-driven and one motor-driven pump) 60 s after SI. (At Farley, the motor-driven AFW pump is started by the ESF bus loss of voltage (i.e., LOSP signal), and the turbine-driven AFW pump is started by the RCP bus loss of voltage.) The AFW enthalpy is assumed to be that of the main feedwater until after an additional bounding feedwater purge volume (140 ft^3/loop) has been displaced.

15.3.1.2.2 Results

15.3.1.2.2.1 __Limiting Break Case__. This section presents results of the limiting small break LOCA analysis (as determined by the highest calculated peak clad temperature) from a range of break sizes and RCS average temperatures at full power.

NUREG-0737 (reference 6), Section II.K.3.31, required a plant-specific small break LOCA analysis using an evaluation model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-35 (reference 7), generic analyses using NOTRUMP (references 2 and 3) were performed and are presented in WCAP-11145 (reference 8). Those results demonstrate that in a comparison of cold leg, hot leg, and pump suction leg break locations, the cold leg break location is limiting. An eight-break spectrum analysis performed at high RCS average temperature demonstrates that the limiting break is a 2.75-in. diameter cold leg break. This conclusion is also applicable at low RCS average temperature. The results of the 2-in., 2.25-in., 2.5-in., 2.75-in., 3-in., 3.25-in., and 4-in. breaks are based on the Unit 2 analysis, while the results of the 6-in. break are based on the Unit 1 analysis. However, the results and conclusions apply to both units since the two units are hydraulically similar. A list of input assumptions used in the analyses is provided in table 15.3-2. The results of the analyses are summarized in table 15.3-2A, while the key transient event times are listed in table 15.3-2B. The peak clad temperature in a small break LOCA is largely a function of the depth of core uncovery which in turn is dependent on the overall mass inventory and, ultimately, the primary side pressure.

Figures 15.3-4A through 15.3-11A show the following parameters, respectively, for the limiting 2.75-in. break transient for high RCS average temperature.

- RCS pressure.
- Core mixture level.
- Clad temperature transient at peak clad temperature elevation.
- Core exit steam flow.
- Clad surface heat transfer coefficient at the peak clad temperature elevation.
- Fluid temperature at the peak clad temperature elevation.
- Cold leg break mass flowrate.
- ECCS pumped injection flowrate.

During the initial period of the small break transient the effect of the break flowrate is not strong enough to overcome the flowrate maintained by the RCPs as the pumps coast down following LOOP. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following reactor trip, the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transient for the limiting break (2.75-in. break) calculation shown in figure 15.3-6A, it is seen that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is steam cooled. This time is accompanied by the highest vapor superheating above the mixture level. The peak clad temperature attained during the transient at RSG conditions with high RCS average temperature was 1903.6°F. This result is applicable to both Units 1 and 2 since both units are hydraulically similar. At the time the transient was terminated, the safety mass flowrate that was delivered to the RCS exceeded the mass flowrate that was delivered to the RCS exceeded the mass flowrate out the break in each case, with the exception of the 4-in. and 6-in. break cases. For these breaks the clad temperature transient has ended and the RCS mass inventory is increasing. Although the core mixture level has not yet covered the entire core (see figure 15.3-5A), there is no longer a concern of exceeding the 10 CFR 50.46 criteria since the RCS pressure is gradually decaying and there is a net mass inventory gain. The decreasing RCS pressure results in greater SI flow as well as reduced break flow. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel clad temperatures will continue to decline.

Additionally, only one core channel is modeled in the NOTRUMP computer code since the core flowrate during a small break LOCA is relatively slow. This provides enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break LOCA, it is not necessary to perform a small break LOCA evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the VANTAGE 5 fuel design as bounding for all transition cycles. Further, the results documented herein pertain to both Zirc-4 and ZIRLO clad fuel (see reference 12). However, a minimum burnup of 6000 MWD/MTU was assumed for Zirc-4 clad fuel to ensure it remains nonlimiting.

Reference 14 concluded that the LOCA ZIRLO models are acceptable to Optimized ZIRLO cladding in small break analyses, and that no additional calculations are necessary for evaluating the use of Optimized ZIRLO cladding provided plant specific ZIRLO calculations were previously performed.

15.3.1.2.2.2 <u>Additional Break Cases</u>. Studies documented in reference 3 determined that the limiting small break size occurred for breaks < 10 in. in diameter. To ensure that the worst possible small break size has been identified, calculations were performed for a spectrum of breaks (2.0, 2.25, 2.5, 3.0, 3.25, 4, and 6 in.) in addition to the limiting 2.75-in. break. (The 6-in. break case conservatively models a break in the SI line (inner diameter = 5.189 in.)) The results of these calculations are shown in the Results table (15.3-2A) and the Sequence of Events table (15.3-2B).

For all cases analyzed, plots of the following transient parameters are presented:

- RCS pressure.
- Core mixture level.
- Clad temperature transient at peak clad temperature elevation.

The plots at high RCS average temperature are shown in figures 15.3-4B through 15.3-6B for the 2-in. break, figures 15.3-4C through 15.3-6C for the 2.25-in. break, figures 15.3-4D through 15.3-6D for the 2.5-in. break, figures 15.3-4E through 15.3-6E for the 3-in. break, figures 15.3-4F through 15.3-6F for the 3.25-in. break, figures 15.3-4G through 15.3-6G for the 4-in. break, and figures 15.3-4H through 15.3-6H for the 6-in. break. As seen in table 15.3-2A, the peak clad temperatures in all cases were calculated to be less than that for the 2.75-in. break at high

RCS average temperature. The plots for the Unit 2 3-in. break at high RCS average temperature are shown in figures 15.3-4G through 15.3-6G. As seen in table 15.3-2A, the peak clad temperature in this case was calculated to be less than that for the Unit 2 3-inch break at low RCS average temperature.

15.3.1.3 Conclusions

Analyses presented in this subsection show that one high-head SI pump and one residual heat removal pump, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below the NRC acceptance criteria of 2200 ºF, as specified by 10 CFR 50.46. Adequate protection is, therefore, afforded by the ECCS in the event of a small break LOCA.

15.3.1.3.1 Breaks During Startup and Shutdown

During startup and shutdown, studies have shown that for breaks < 2 in., manual initiation of SI may be required. The studies also show that ample time exists for the operator to take such action (see figures 15.3-26, 15.3-27, 15.3-28, and 15.3-29).

15.3.1.3.2 NUREG-0737

Item II.K.3.30 of NUREG-0737 outlines the commission requirements for the industry to demonstrate that its small break LOCA methods continue to comply with the NRC acceptance criteria of Appendix K to 10 CFR 50. The technical issues to be addressed were listed in NUREG-0611, including comparison with semiscale experimental test results.

In response to Item II.K.3.30, the Westinghouse Owners Group (WOG) elected to reference the NOTRUMP code as the new licensing basis for the small break LOCA model. The NOTRUMP code and methodology are described in WCAP-10079[2] and WCAP-10054[3]. The NRC staff reviewed and approved NOTRUMP as the new licensing tool for calculating small break LOCA response for Westinghouse plant designs. The NRC staff further concluded that the WOG actions had met the requirements of Item II.K.3.30 of NUREG-0737 and that the responses to NUREG-0611 concerns, as calculated in the NRC's TMI Action Item II.K.3.30 SER, were found acceptable.

Item II.K.3.31 of NUREG-0737 required that each license holder or applicant submit a new small break analysis using the model approved under Item II.K.3.30. NRC Generic Letter 83-35 dated November 2, 1983, provided clarification of the requirements of Item II.K.3.31 by allowing license holders and applicants to comply on a generic basis by demonstrating that the WFLASH analyses are conservative when compared to analyses performed using NOTRUMP. As a result, the WOG submitted WCAP-11145[8] which contained generic comparisons to WFLASH analyses for various plant types, including comparisons for 3-loop plants of the design similar to Plant Farley. Initially, Alabama Power Company chose to reference WCAP-11145 to resolve Item II.K.3.31 and received NRC approval by letter dated January 16, 1987. However, the small break LOCA was reanalyzed using the NOTRUMP code, in conjunction with the transition to

VANTAGE-5 fuel, and was found acceptable by the NRC as documented in their SER dated March 11, 1992. The small break LOCA was again reanalyzed using the NOTRUMP code in conjunction with the uprate in core power to 2775 MWt, and was again found to be acceptable by the NRC as documented in their SER dated April 29, 1998. Finally, the small break LOCA was reanalyzed using the NOTRUMP code in conjunction with Model 54F replacement steam generators and was found to be acceptable by the NRC as documented in NRC SER dated December 29, 1999.

15.3.2 MINOR SECONDARY SYSTEM PIPE BREAKS

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-in. diameter break or smaller.

15.3.2.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in subsection 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analysis for minor secondary system pipe breaks is not required.

The analyses of the more probable accidental opening of a secondary system steam dump, relief, or safety valve are presented in subsection 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.3.2.3 Conclusions

The analysis presented in paragraph 15.4.2.1 demonstrates that the consequences of a minor secondary system pipe break are acceptable since a departure from nucleate boiling ratio (DNBR) of less than the limit value does not occur even for a more critical major secondary system pipe break.

15.3.3 INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION

15.3.3.1 Identification of Causes and Accident Description

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, the loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5-percent uncertainty margin included in the design value of the power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number will be checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Power distribution in the x-y plane of the core and resulting thermal hydraulic conditions are analyzed with the steady-state computer programs briefly discussed in chapter 4. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The assembly power distributions in the x-y plane for a correctly loaded core are also given in chapter 4 based on enrichments given in that chapter.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (see figures 15.3-15 through 15.3-19).

15.3.3.2.2 Results

The following core loading error cases have been analyzed:

A. Case A

In this case, a region 1 assembly is interchanged with a region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (see figure 15.3-15).

B. Case B

In this case, a region 1 assembly is interchanged with a neighboring region 2 fuel assembly. Two analyses have been performed for this case (see figures 15.3-16 and 15.3-17).

1. In case B-1, the interchange is assumed to take place with the burnable poison rods transferred with the region 2 assembly mistakenly loaded into region 1.

2. In case B-2, the interchange is assumed to take place closer to core center and with burnable poison rods located in the correct region 2 position but in a region 1 assembly mistakenly loaded into the region 2 position.

C. Case C

This is an enrichment error case in which a region 2 fuel assembly is loaded in the core central position (see figure 15.3-18).

D. Case D

In this case, a region 2 fuel assembly, instead of a region 1 assembly, is loaded near the core periphery (see figure 15.3-19).

15.3.3.3 <u>Conclusions</u>

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the incore moveable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.3.4 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

15.3.4.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in a departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor were not tripped promptly.

The following signals provide the protection against a complete loss of flow accident.

A. Low reactor coolant loop flow

B. Undervoltage or underfrequency on RCP power supply buses (Unit 2 only)

The loss of measured loop flow is the primary trip credited in accident analysis. The RCP bus undervoltage and underfrequency trips are backups to the low flow trip (even though they perform in an anticipatory fashion).

The reactor trip on low primary coolant loop flow is provided as the primary trip to protect against loss of flow conditions. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10% power (permissive P-7) and the power level corresponding to permissive P-8, low flow in any two loops will actuate a reactor trip.

The reactor trip on reactor coolant pump undervoltage is provided as an anticipatory trip to protect against conditions which can cause a loss of voltage to all RCPs, i.e., loss of offsite power. This function is blocked below approximately 10% power (permissive P-7). See FSAR table 7.2.2 for a definition of permissive setpoints.

The RCP underfrequency function is provided as an anticipatory trip to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid.

The RCP underfrequency reactor trip function is blocked below P-7. In addition, the underfrequency function will open all RCP breakers whenever an underfrequency condition occurs (no P-7 or P-8 interlock) to ensure adequate RCP pump coastdown.

15.3.4.2 Method of Analysis

This transient is analyzed by three digital computer codes. First, the LOFTRAN (reference 9) code is used to calculate the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN (reference 10) code is then used to calculate the heat flux transient based on the nuclear power and flow from

LOFTRAN. Finally, the THINC code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical and thimble cells.

Two cases have been analyzed:

1. Complete loss of all three RCPs with three loops in operation;

2. Frequency decay event resulting in a complete loss of forced reactor coolant flow.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in section 15.1, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by low reactor coolant flow. (Note: With respect to the reactivity coefficient assumptions, the analysis conservatively bounds a +7 pcm/°F MTC ≤ 70% power, ramping to 0 at full power, by assuming a moderator temperature coefficient of +2 pcm/°F at full power.)

The accident is analyzed using the Revised Thermal Design Procedure (RTDP). Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397 (reference 11).

15.3.4.3 <u>Results</u>

The calculated sequence of events for the limiting case (Frequency Decay) is shown in table 15.3-1. Figures 15.3-20 through 15.3-25 also show the transient response for this limiting loss of flow event. The reactor is assumed to be tripped on a low flow signal. Figure 15.3-25 represents the general DNBR trend for the limiting cell type.

15.3.4.4 <u>Conclusions</u>

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNB design basis is met during the transient; thus, there is no clad damage or release of fission products to the RCS.

15.3.5 WASTE GAS DECAY TANK RUPTURE

15.3.5.1 <u>Accident Description</u>

The gaseous waste processing system (GWPS), discussed in chapter 11, is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, hydrogen recombiners, waste gas decay tanks for service at power, and other waste gas decay tanks for service at shutdown and startup.

The accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping.

15.3.5.1.1 Method of Analysis

Nonvolatile fission product concentrations are greatly reduced as the coolant is passed through the purification demineralizers. An iodine removal factor of 10 is expected in the mixed-bed demineralizers, and an iodine partition factor of the order of 10,000 is expected between the liquid and vapor phases. Based on the above analysis and operating experience at Yankee-Rowe and Saxton, activity stored in a gas decay tank consists of that from the noble gases released from the processed coolant and only negligible quantities of less volatile isotopes.

The maximum noble gas activities in the waste gas decay tanks and the assumptions on which the calculations are based are given in chapter 11.

15.3.5.2 Environmental Consequences

The GWPS is designed to be operated in such a manner that the maximum activity in any one gas decay tank is such that the offsite doses resulting from a rupture of the tank will be well within (25% of) 10 CFR 100 guidelines for accidents with the plant operating at the design fuel defect level of 1%. The system is equipped with a radiation monitor which is installed so that it always indicates activity level in the tank onstream at the time. The monitor will alarm at the waste panel when the activity level in the onstream tank reaches a predetermined level. The initiation of the alarm on the waste panel also initiates a general alarm in the control room. When the alarm is received in the control room, the operator will go to the gas decay tank valving area and manually isolate the high-activity tank and open another tank to receive gaseous radioactive waste. The gaseous activity in the high-activity tank will then decay while the other tanks in the system are being filled with gaseous radioactivity. The order of filling the gas decay tanks will be such that the tank being filled (or to be filled) has the lowest quantity of radioactivity stored in it at the time filling of the tank begins.

Therefore, the maximum activity that could be released as a result of a gas decay rupture is the activity stored in one gas decay tank immediately after it has been isolated from the GWPS.

Parameters used for the analysis are listed in table 15.3-3. The conservative evaluation of the radiation doses resulting from the postulated rupture of a gas decay tank is based on the following assumptions:

A. The maximum content of the decay tank assumed to fail is used for the purpose of computing the noble gas inventory released. The noble gas inventory of the tank is given in table 15.3-4. The inventory is based on the maximum inventory allowed by the Technical Requirements Manual.

B. The tank rupture is assumed to occur immediately after isolation of the tank from the GWPS, releasing the entire contents of the tank at ground level to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that the valving of the decay tanks in the GWPS has been designed such that a release from one gas decay tank due to any means will not result in any additional release of radioactivity stored in any of the other gas decay tanks.

C. The 0- to 2-h atmospheric diffusion factor given in appendix 15B is applicable.

These offsite doses are substantially below (< 0.5-rem whole body in accordance with NUREG 0133) the limits for accidents as defined in 10 CFR 100, as shown in table 15.3-3.

15.3.6 SINGLE RCCA WITHDRAWAL AT FULL POWER

15.3.6.1 Accident Description

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The event analyzed must result from multiple wiring failures, multiple significant operator errors, or subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low so that the limiting consequences may include slight fuel damage.

Each bank of RCCAs in the system is divided into two groups of four mechanisms each. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are required to drive the RCCAs. The RCCAs in a group are driven in parallel. Any single failure which would cause withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are in the direction of insertion or immobility. (Note: The operator can deliberately withdraw a single RCCA in a control or shutdown bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped.)

In the unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, the plant annunciator will display both the rod deviation and rod control urgent failure, and the rod position indicators will indicate the relative positions of the RCCAs in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indication. The overtemperature ΔT (OTΔT) reactor trip provides automatic protection for this event, although due to the increase in local power density, it is not possible to always provide assurance that the core safety limits will not be exceeded.

15.3.6.1.1 Method of Analysis

Power distributions are analyzed using appropriate nuclear physics computer codes. The DNB evaluation conservatively assumes that any fuel rod with an $F_{\Delta H}$ above the Core Operating Limits Report limit is in DNB and fails. The analysis examines the case of the worst rod withdrawn from bank D, inserted at the insertion limit with the reactor initially at full power.

15.3.6.1.2 Results

Two cases have been considered as follows:

A. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in subsection 15.2.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the safety analysis limit value. Evaluation of this case at the power and coolant conditions at which the OTΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5%.

B. If the reactor is in the automatic rod control mode, the multiple failures that result in the withdrawal of a single RCCA cause immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as case A, described above. For such cases, reactor trip will ultimately ensue, although not quickly enough in all cases to prevent a minimum DNBR in the core of less than the safety analysis limit value. Following reactor trip, normal shutdown procedures are followed.

15.3.6.2 <u>Conclusions</u>

For the condition of one RCCA fully withdrawn with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5% of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB would occur. For case B discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

REFERENCES

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2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

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13. Thompson, C. M., et al., “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model,” WCAP-10054-P-A, Addendum 2, Revision 1, July 1997.

14. Shah, H. H., et al., “Addendum 1 to WCAP-126-P-A and CENPD-404-P-A Optimized ZIRLO,” February 2003.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time (s)
Complete loss of forced reactor coolant flow (Underfrequency)	Frequency decay begins	0.0
	Low reactor coolant flow trip setpoint reached	1.9
	Rods begin to drop	2.9
	Minimum DNBR occurs	5.0

TABLE 15.3-2

PLANT INPUT PARAMETERS USED IN SMALL BREAK LOCA ANALYSIS FOR 17 X 17 VANTAGE 5 FUEL

Core power	102% of 2775 MWt
Total core peaking factor (F_Q)	2.50
Enthalpy rise peaking factor ($F_{\Delta H}$)	1.70
Steam generator tube plugging level	20% (peak uniform)
Accumulator conditions:	
Cover gas pressure	600 psia
Water volume[(a)]	980 ft^3
Total volume	1450 ft^3
RCS initial conditions:	
Core T_{avg}	583.2 °F (high) / 561.2 °F (low)
Pressure	2300 psia
Vessel flowrate	258,000 gal/min
Reactor trip signal	1840 psia
Signal delay time	2.0 s
Safety injection signal	1700 psia
Safety injection delay time	27 s
Rod drop time	2.7 s
MFW isolation time	
Delay time	2.0 s
Valve closure time	5.0 s

a. The initial accumulator water volume does not include the undeliverable piping volume of 45 ft^3.

TABLE 15.3-2A

SMALL BREAK LOSS-OF-COOLANT ACCIDENT CALCULATION AT RSG CONDITIONS

Parameter	Results[a]							
	Case A 2-in. High T_{avg}	Case B 2.25-in. High T_{avg}	Case C 2.5-in. High T_{avg}	Case D 2.75-in. High T_{avg}	Case E 3-in. High T_{avg}	Case F 3.25-in. High T_{avg}	Case G 4-in. High T_{avg}	Case H[c] 6-in. High T_{avg}
Burnup	BOL	BOL	BOL	10,000 [b] MWD/MTU	10,000 [b] MWD/MTU	BOL	BOL	BOL
Peak clad temperature (°F)	985.7	1455.1	1570.6	1903.6[d]	1834.6	1702.1	1456.7	1035.0
elevation (ft)	11.25	11.75	11.75	12.00	11.75	11.75	11.25	10.50
Zr/H_2O cumulative reaction								
Local maximum (%)	0.04	0.88	1.06	8.68	7.90	1.78	0.30	0.01
elevation (ft)	11.25	11.50	11.75	12.00	11.75	11.75	11.25	10.50
Total core (%)	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	<1.0
Rod Burst time (s)	None	None	None	1640	1324.4	None	None	None
elevation (ft)				12	11.75			

a. All results are for ZIRLO cladding; use of Optimized ZIRLO was qualitatively evaluated as acceptable.

b. The limiting time-in-life for the 2.75-in. and 3.0-in. break cases is determined to be at 10,000 MWD/MTU for both PCT and oxidation.

c. The 6-in. break results are based on Unit 1 analysis; all other break sizes were based on Unit 2 analysis. Note that all results are applicable to both units, since both units are hydraulically similar.

d. The analyzed peak cladding temperature (PCT) attained during this case was 1903.6 ºF. For 10 CFR 50.46 reporting, the limiting PCT value (1903.6 ºF) is increased by 10 ºF to account for an annular blanket penalty and is rounded up to the nearest integer. For 10 CFR 50.46 reporting, this value (1914 ºF) is referred to as the Analysis-of-Record PCT.

TABLE 15.3-2B

SMALL BREAK LOCA CALCULATION AT RSG CONDITIONS

Event	Time(s)							
	Case A 2-in. High T_{avg}	Case B 2.25-in. High T_{avg}	Case C 2.5-in. High T_{avg}	Case D 2.75-in.[i] High T_{avg}	Case E 3-in.[i] High T_{avg}	Case F 3.25-in. High T_{avg}	Case G 4-in. High T_{avg}	Case H 6-in. High T_{avg}
Break occurs	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor trip signal[f]	39.9	30.5	24.0	19.7	16.6	14.0	9.6	7.4
Safety injection signal[f]	51.2	40.6	33.7	29.2	25.7	22.9	18.1	11.4
Start of safety injection delivery[g]	78.2	67.6	60.7	56.2	52.7	49.9	45.1	38.4
Start of AFW delivery[h]	111.2	100.6	93.7	89.2	85.7	82.9	78.1	71.4
Loop seal venting (initial)[d]	860	668	541	452	379	324	222	60
Loop seal core uncovery	953	749	626	(c)	(c)	(c)	212	130
Loop seal core recovery	960	764	633	(c)	(c)	(c)	244	175
Boiloff core uncovery	1204	877	881	567	503	642	450	243
Accumulator injection begins	(a)	2524	1880	1375	1119	973	616	270
Peak clad temperature occurs	3184.4	2177.3	1906.3	1642.5	1326.6	1180.7	755.7	334
Top of core recovered	(b)	(b)	(b)	(b)	(b)	(b)	(b)	365
SI flowrate exceeds break flowrate	2836	2258	2800	2809	2940	2810	(e)	(e)

a. System pressure never drops below the accumulator cut-in pressure (600 psia).
b. For the cases where core recovery is greater than the transient time, basis for transient termination can be concluded based on some or all of the following: (1) The RCS pressure is decreasing or is stable, which will increase safety injection (SI) flow, (2) Total RCS system mass is increasing or stable, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.
c. No core uncovery occurs during the loop seal clearing period for this transient.
d. Loop seal venting is considered to occur when the broken loop seal vapor flow rate is above 1 lbm/s.
e. Although SI flow has not yet matched break flow, the core is covered, the clad temperature transient has ended, the total RCS mass is increasing, and the core is covered (for the 6-in. break).
f. Time where system pressure reaches the actuation setpoint at the pressure sensor.
g. SI is assumed to begin 27 s after SI signal.
h. AFW delivery is assumed to begin 60 s after SI signal.
i. The limiting time in like for the 2.75-in. and 3.0-in. break cases for the PCT was determined to be 10,000 MWD/MTU. All other PCT times are for beginning-of-life (BOL).

TABLE 15.3-3

PARAMETERS USED IN WASTE GAS DECAY TANK RUPTURE ANALYSES

Plant load factor	1.00
Activity released from GWPS	Contents of one tank
Tank Contents	See Table 15.3-4
Number of tanks (normal operation)	6.00
Iodine partition factor in volume control tank	0.01
Time of accident	Immediately after isolation of tank from GWPS
Meteorology	Accident (see appendix 15B)

OFFSITE DOSES FROM WASTE GAS DECAY TANK RUPTURE

	Whole Body Dose (Rem)	B-Skin Dose (Rem)
Site Boundary	0.30	0.57
Low Population Zone	0.11	0.21

TABLE 15.3-4

WASTE GAS DECAY TANK INVENTORY
(Technical Requirements Manual Limit for Conservative Analysis)

Isotope	Activity (Ci)
Xe-133	6.77×10^{4}
Xe-133m	1.02×10^{3}
Xe-135	6.77×10^{2}
Xe-135m	2.88×10^{0}
Xe-138	2.63×10^{-1}
Kr-85	(a)
Kr-85m	8.03×10^{1}
Kr-87	9.15×10^{0}
Kr-88	8.53×10^{1}

(a) The dose conversion factor for Kr-85 is much less than the other isotopes, and it accumulates much slower than the other isotopes, thus it is conservatively ignored.

TABLE 15.3-5A

SAFETY INJECTION FLOWRATE[a]

RCS Pressure (psia)	Intact Loop SI Flowrate (spill to RCS) (lb/s)	Broken Loop SI Flowrate (spill to RCS) (lb/s)
14.7	45.8	25.3
114.7	44.8	24.7
214.7	43.8	24.2
314.7	42.6	23.5
414.7	41.5	22.9
514.7	40.3	22.2
614.7	39.1	21.6
714.7	37.8	20.9
814.7	36.5	20.2
914.7	35.2	19.5
1014.7	33.9	18.7
1114.7	32.6	18.0
1214.7	31.2	17.2
1314.7	29.8	16.5
1414.7	28.3	15.6
1514.7	26.7	14.8
1614.7	25.1	13.9
1714.7	23.3	12.9
1814.7	21.4	11.8
1914.7	19.5	10.8
2014.7	17.3	9.6
2114.7	14.8	8.2
2214.7	11.0	6.1
2314.7	2.4	1.3
2414.7	0.0	0.0

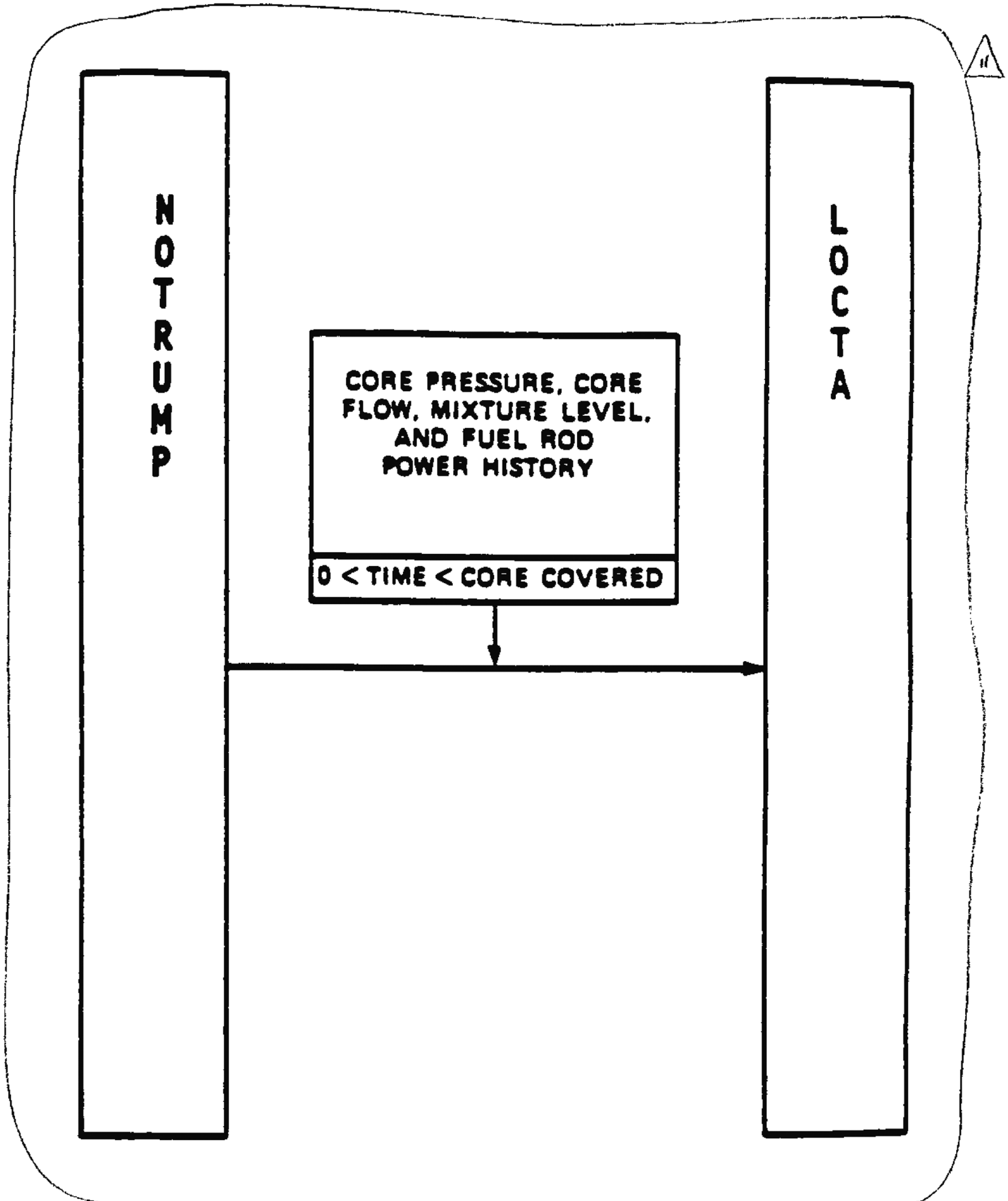
a. This table assumes flow from one high-head safety injection pump.

TABLE 15.3-5B

SAFETY INJECTION FLOWRATE[a]

RCS Pressure (psia)	Intact Loop SI Flowrate (spill to containment) (lb/s)	Intact Loop RHR Flowrate (spill to containment) (lb/s)	Total Intact Loop ECCS Flow (lb/s)
14.70	45.00	352.90	397.90
34.70	43.70	309.90	353.60
54.70	43.70	264.60	308.30
74.70	43.70	212.80	256.50
94.70	43.70	155.30	199.00
114.70	43.70	88.30	132.00
134.70	42.10	5.00	47.10
154.70	42.10	0.00	42.10
214.70	42.10	0.00	42.10
314.70	40.40	0.00	40.40
414.70	38.70	0.00	38.70
514.70	36.90	0.00	36.90
614.70	35.10	0.00	35.10
714.70	33.30	0.00	33.30
814.70	31.30	0.00	31.30
914.70	29.30	0.00	29.30
1014.70	27.00	0.00	27.00
1114.70	24.30	0.00	24.30
1214.70	21.60	0.00	21.60
1314.70	18.70	0.00	18.70
1414.70	15.60	0.00	15.60
1514.70	12.40	0.00	12.40
1614.70	9.00	0.00	9.00
1714.70	5.00	0.00	5.00
1814.70	0.40	0.00	0.40
1914.70	0.00	0.00	0.00

a. This table assumes flow from one high-head safety injection pump and one RHR pump.



REV 21 5/08

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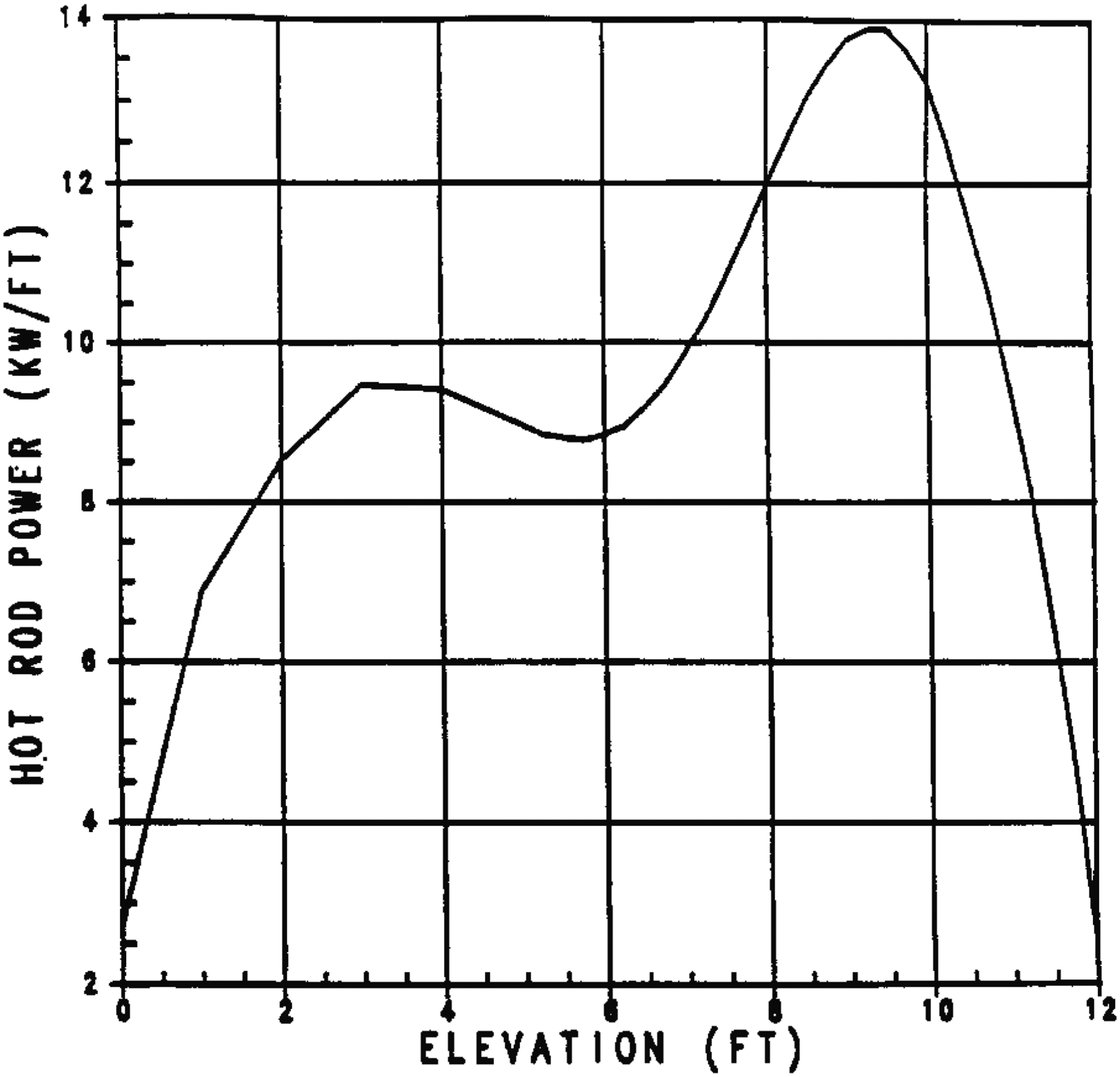
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CODE INTERFACE DESCRIPTION
FOR SMALL BREAK MODEL

FIGURE 15.3-1



15

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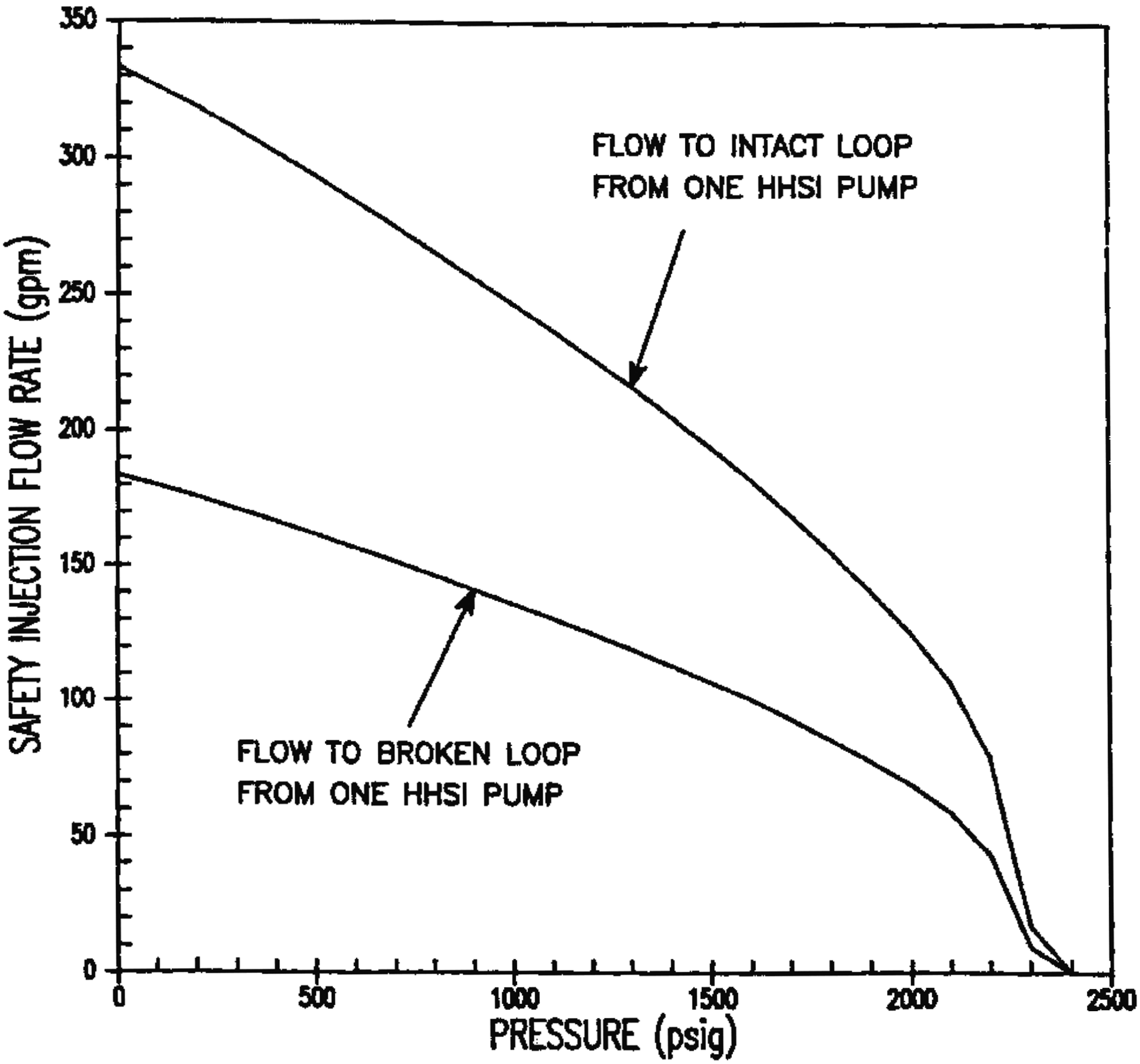
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SMALL BREAK HOT ROD POWER SHAPE

FIGURE 15.3-2



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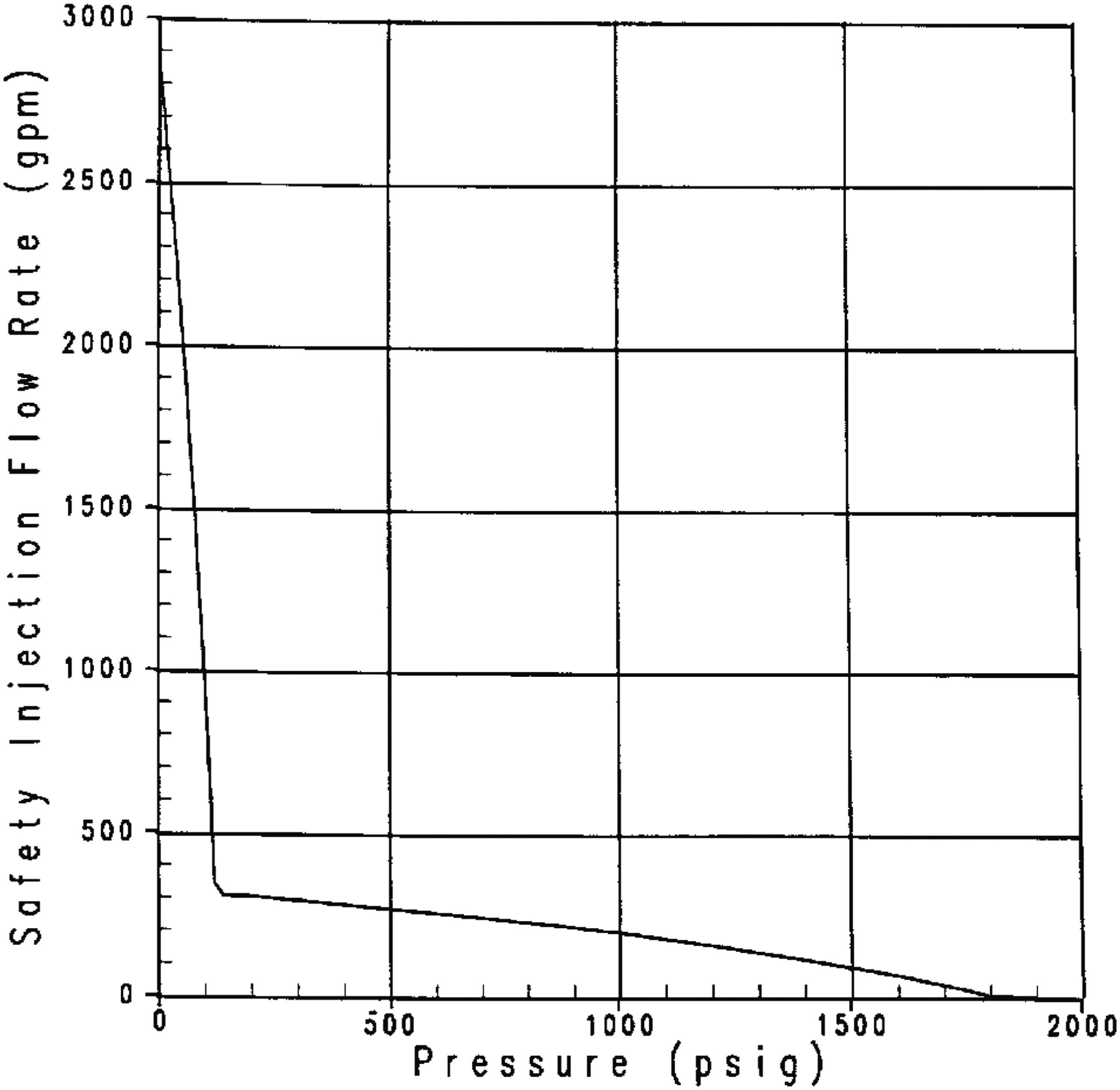
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SMALL BREAK SAFETY INJECTION FLOWRATE – 1 HIGH-HEAD SAFETY INJECTION PUMP
		FIGURE 15.3-3A



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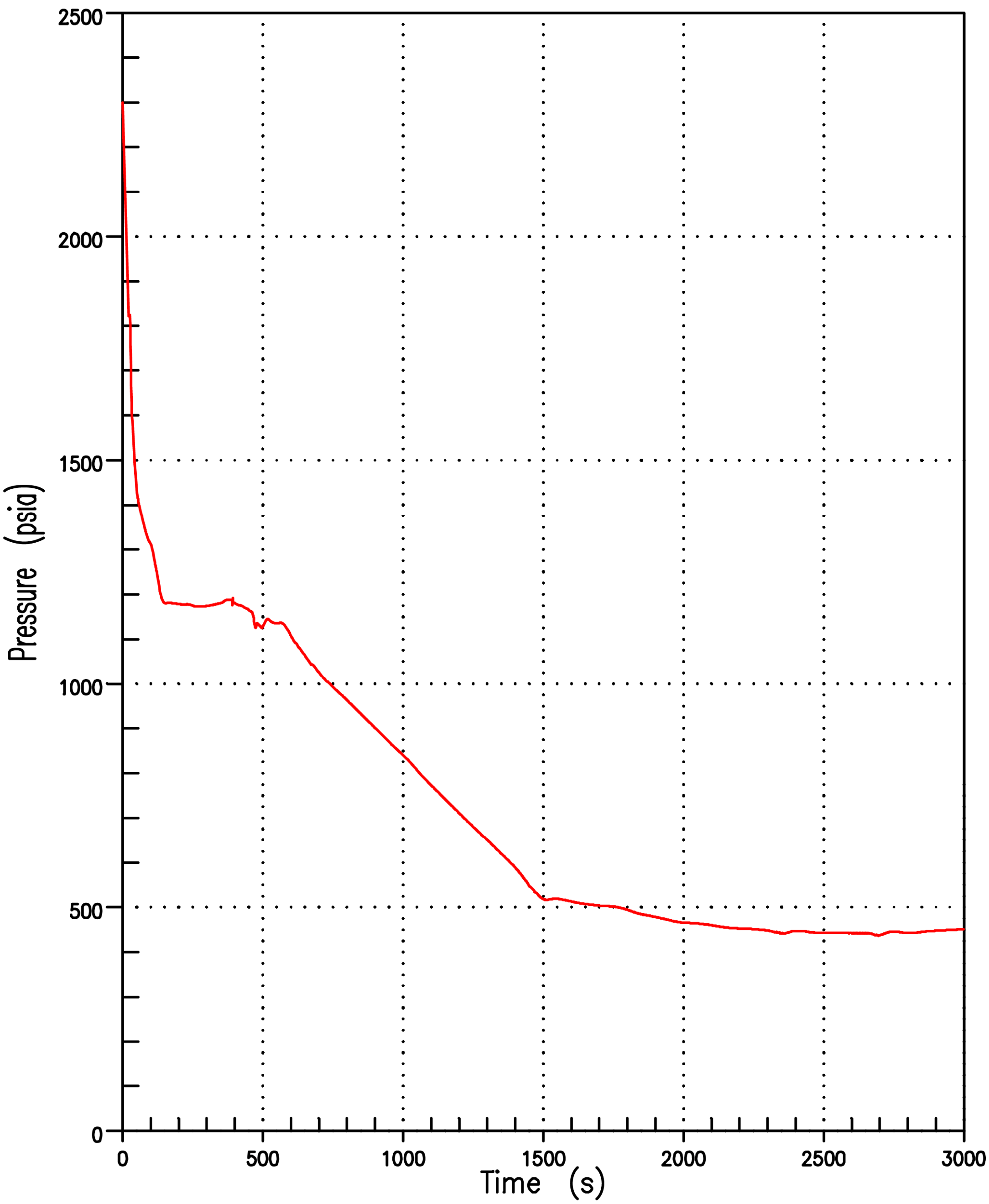
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

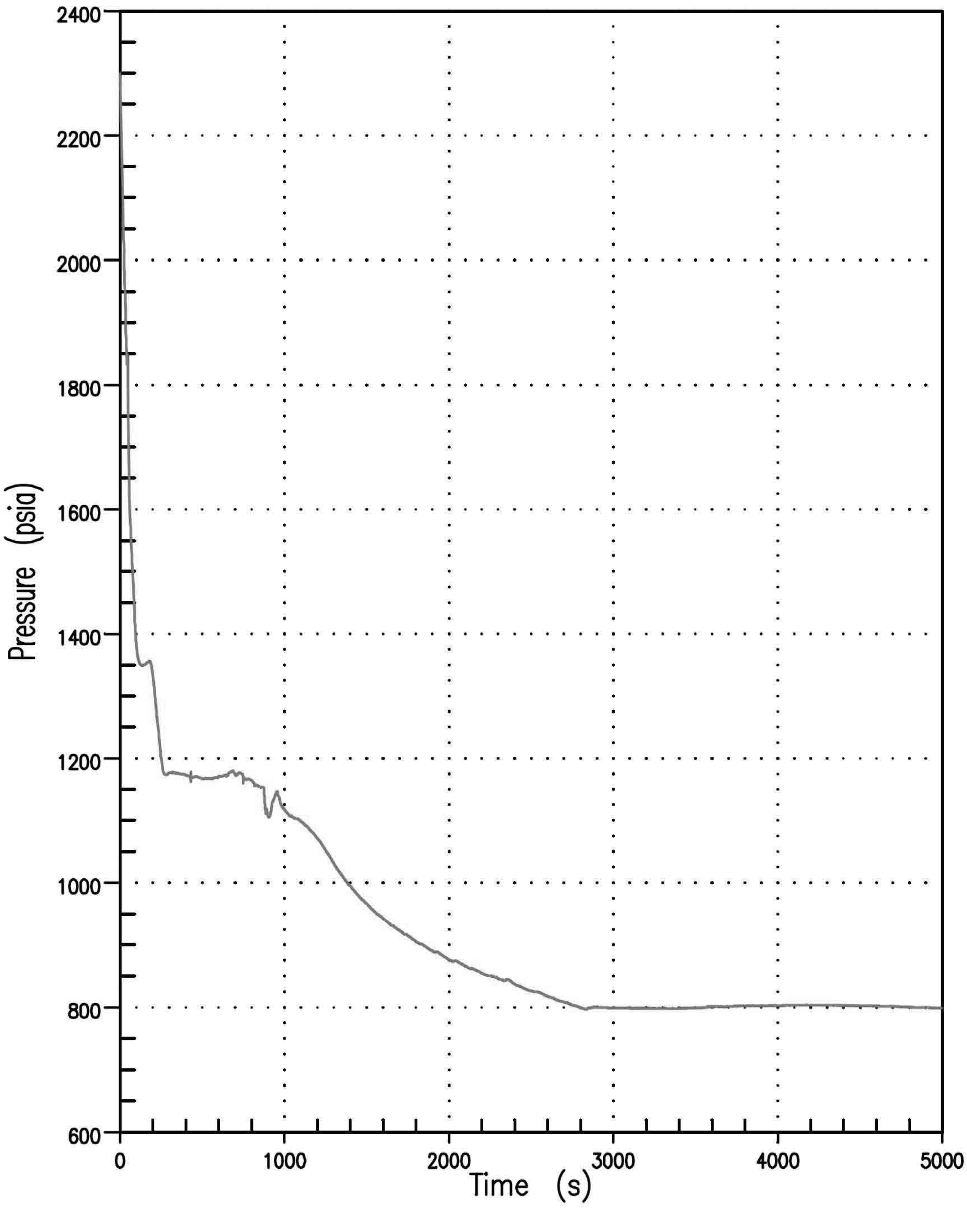
SMALL BREAK SAFETY INJECTION
FLOWRATE – 1 HIGH-HEAD SAFETY
INJECTION PUMP & 1 RHR PUMP

FIGURE 15.3-3B



REV 23 5/11

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RCS PRESSURE TRANSIENT (2.75-IN. BREAK) FIGURE 15.3-4A



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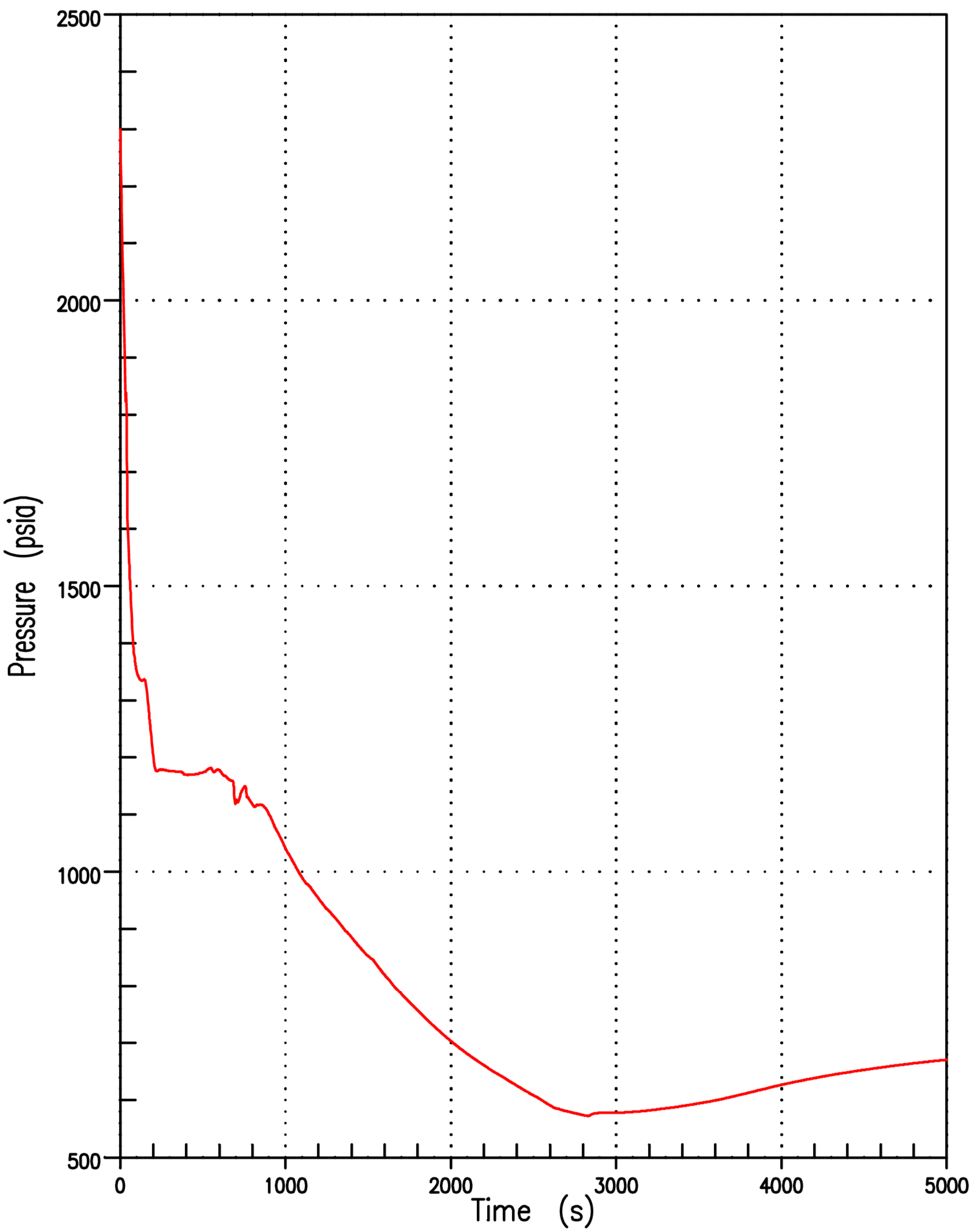
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RCS PRESSURE
(2.0-IN. BREAK)

FIGURE 15.3-4B



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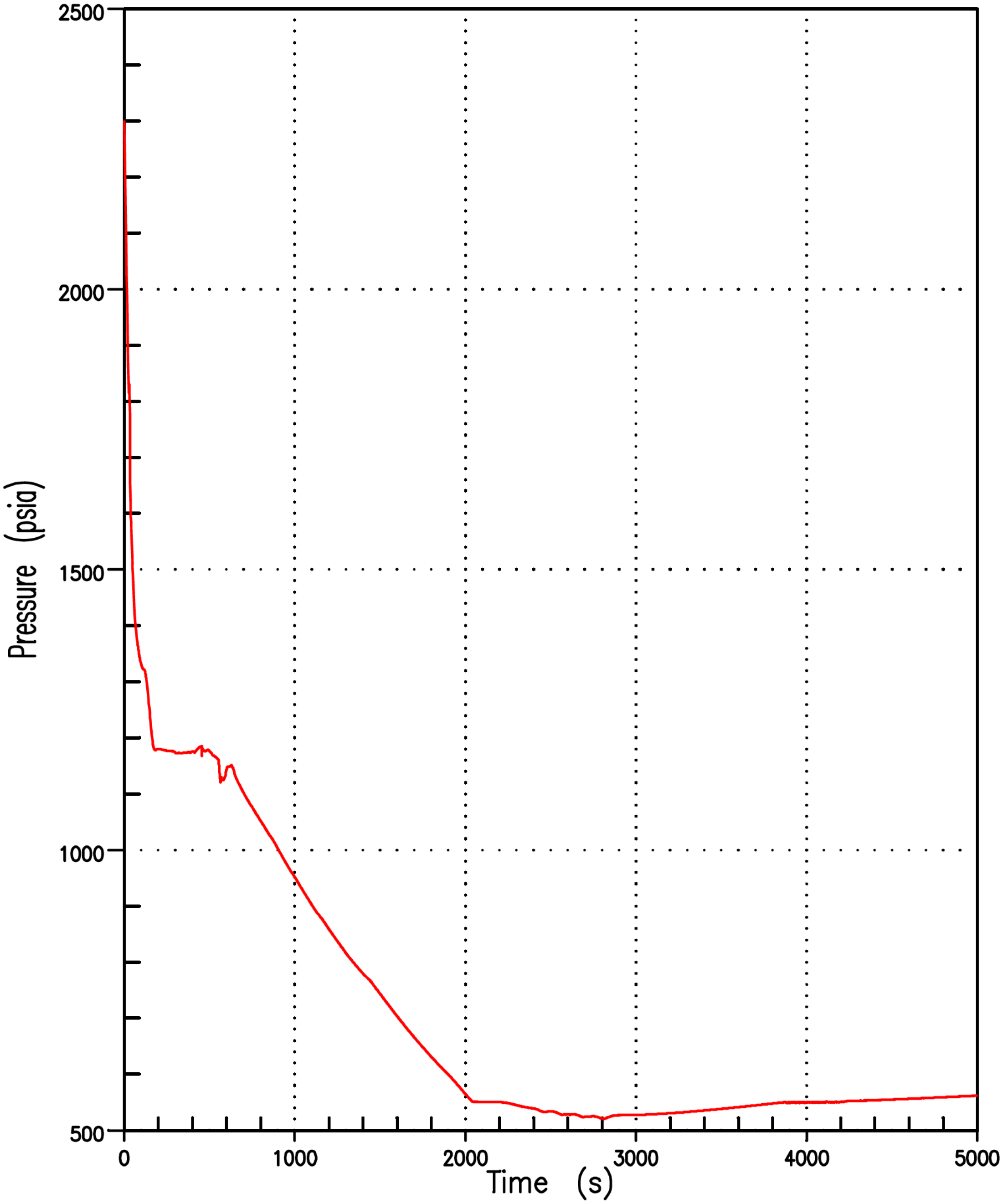
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RCS PRESSURE TRANSIENT
(2.25-IN. BREAK)

FIGURE 15.3-4C

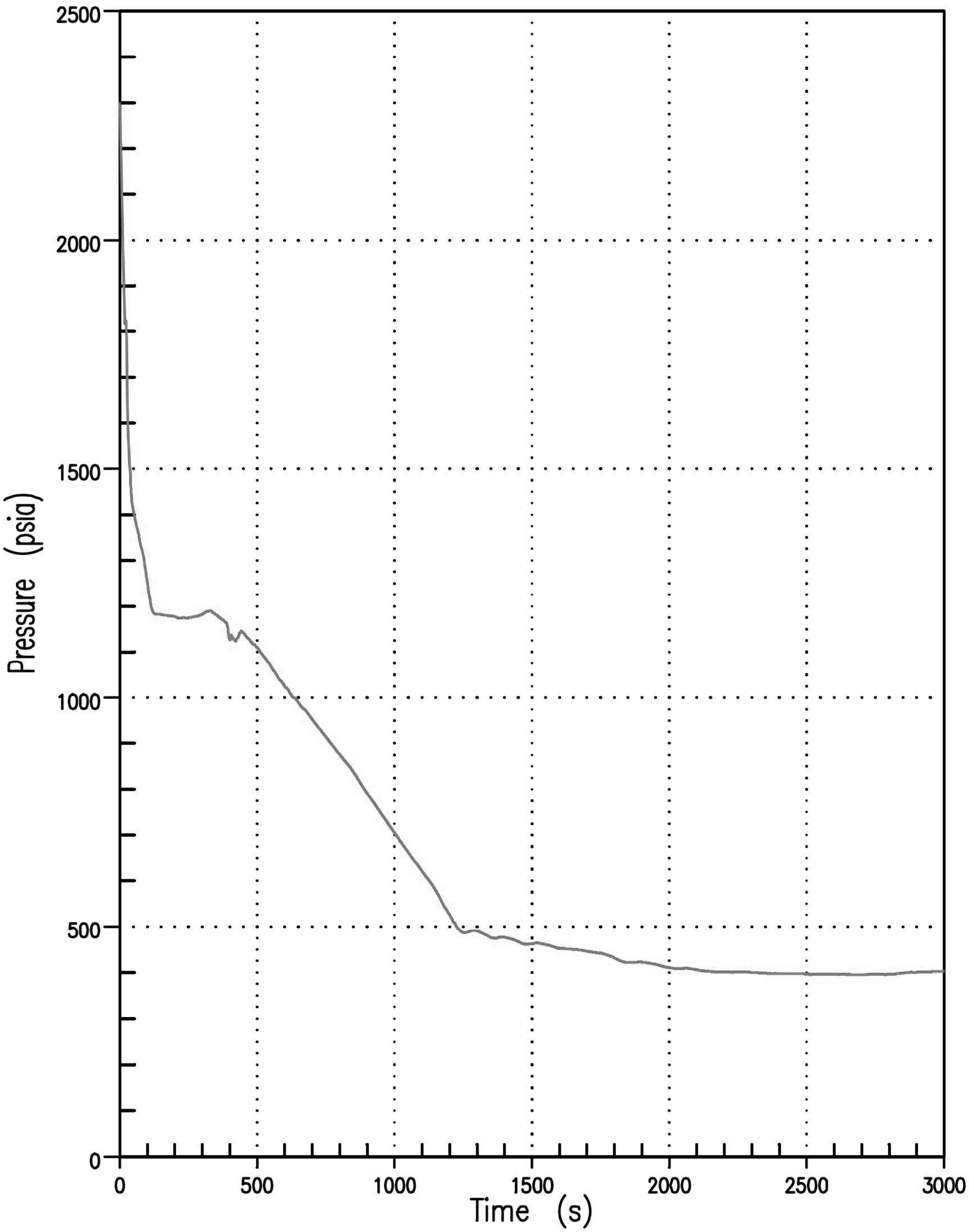


REV 23 5/11

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RCS PRESSURE TRANSIENT (2.5-IN. BREAK)

FIGURE 15.3-4D



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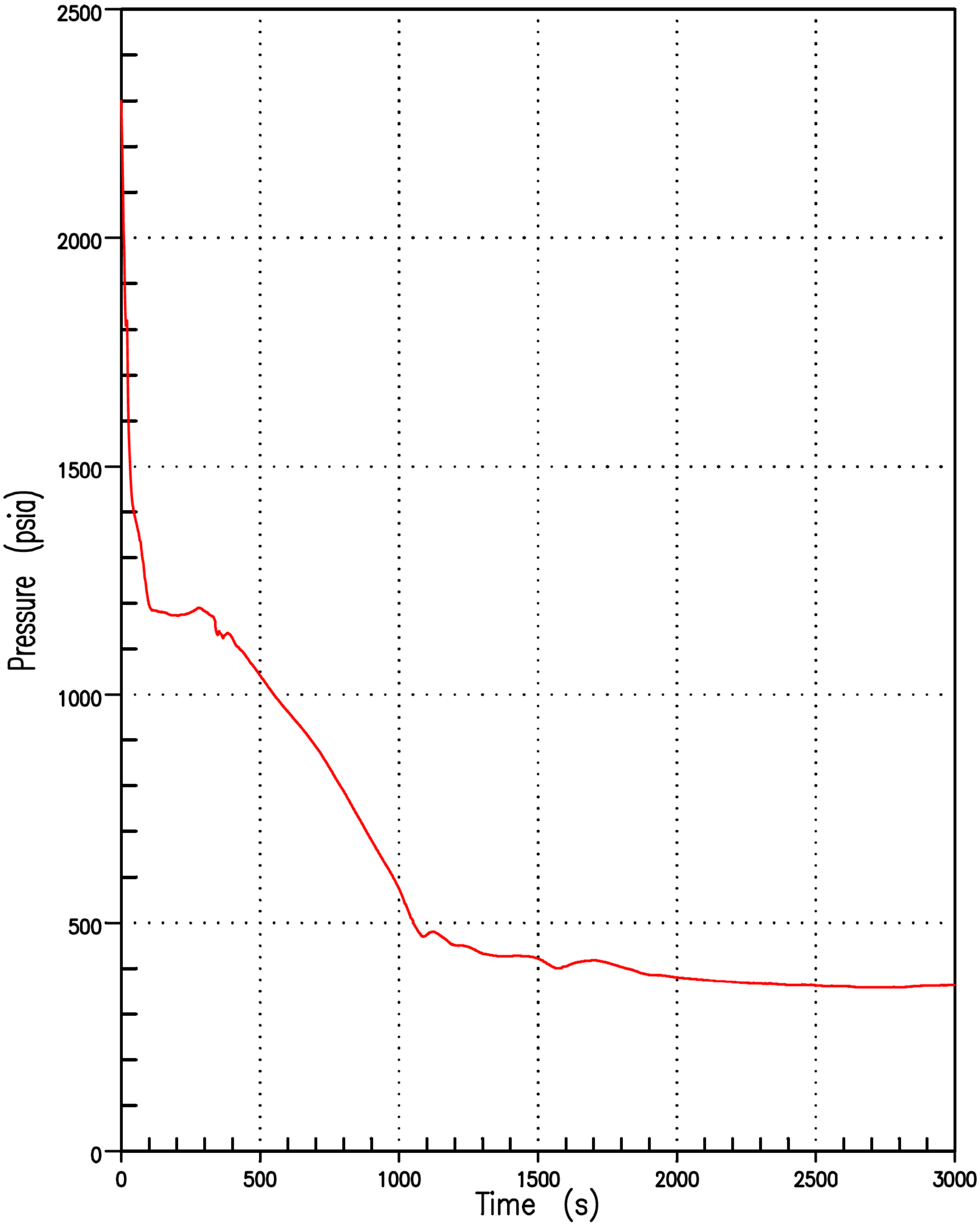
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RCS PRESSURE TRANSIENT (3.0-IN. BREAK)

FIGURE 15.3-4E



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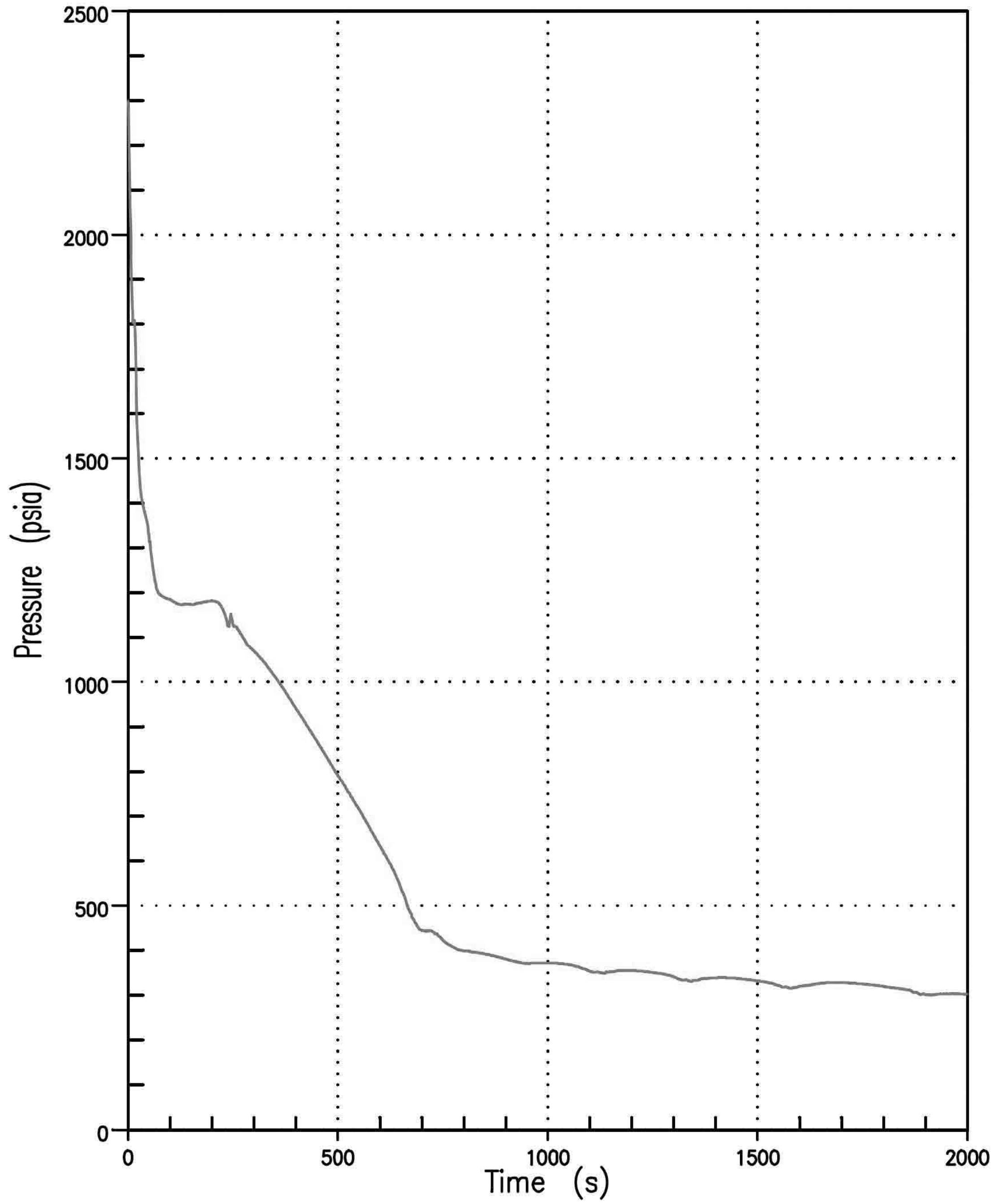
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

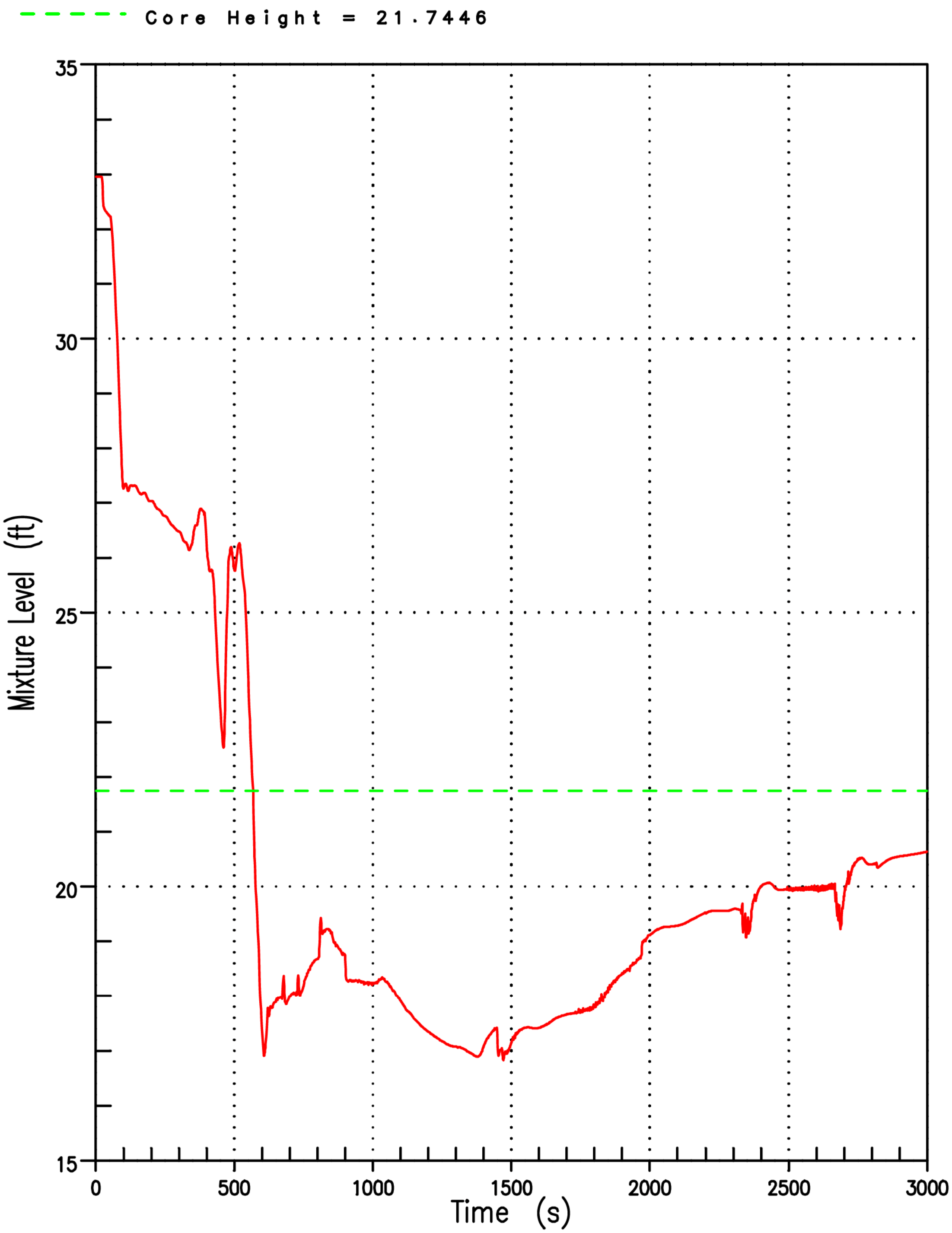
RCS PRESSURE TRANSIENT
(3.25-IN. BREAK)

FIGURE 15.3-4F



REV 23 5/11

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RCS PRESSURE TRANSIENT (4.0-IN. BREAK)
		FIGURE 15.3-4G



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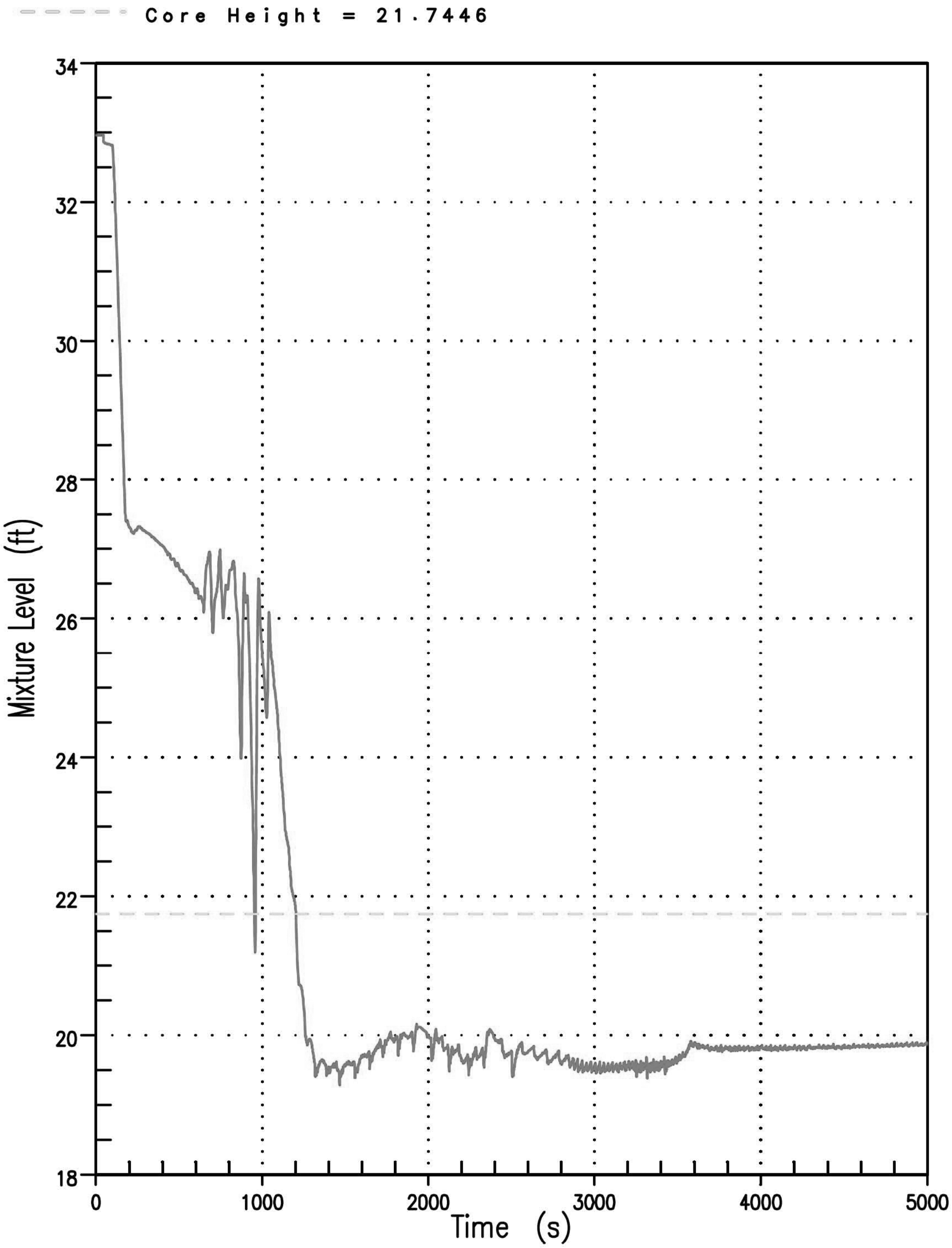
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

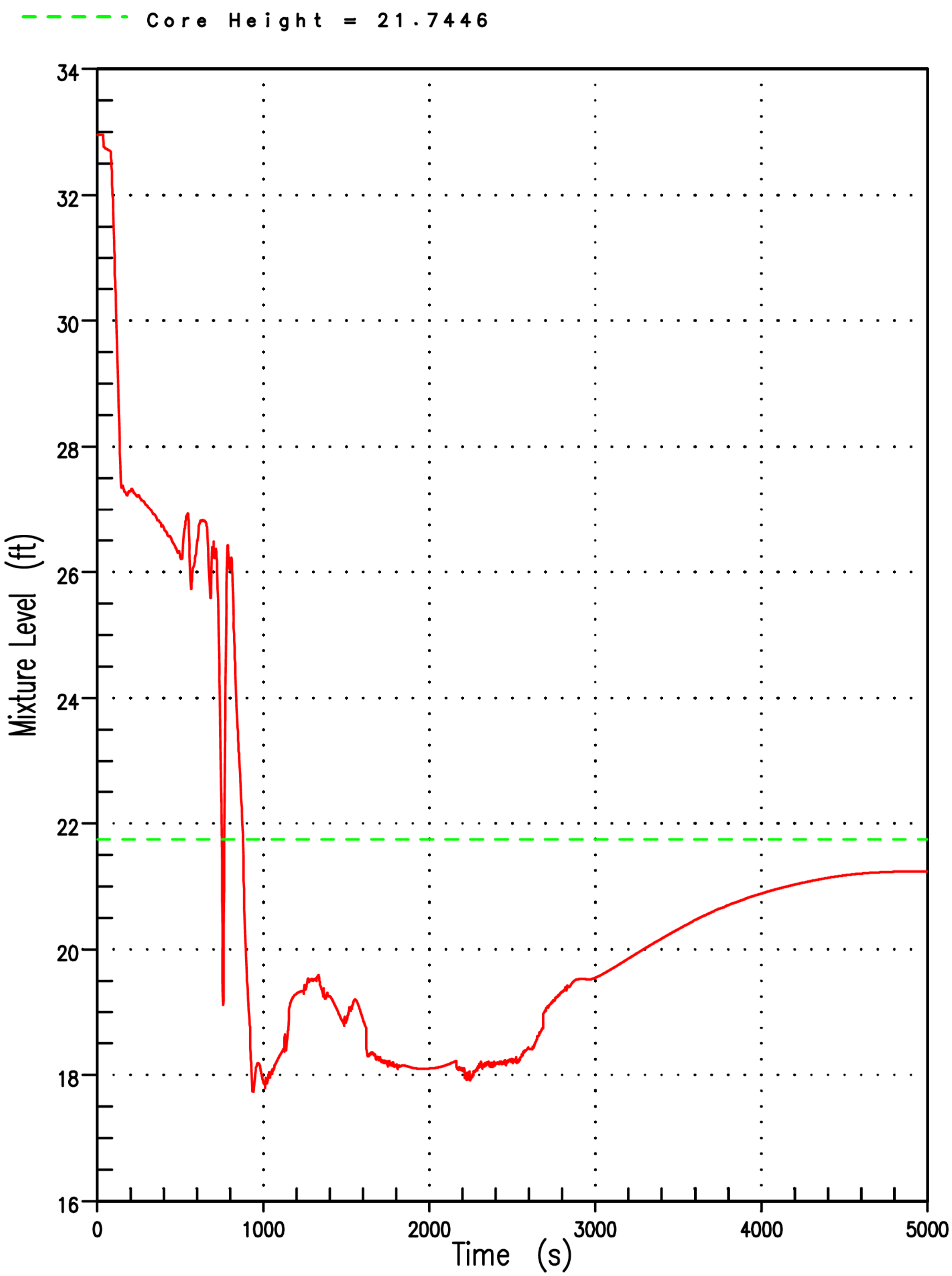
CORE MIXTURE LEVEL
(2.75-IN. BREAK)

FIGURE 15.3-5A



REV 23 5/11

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CORE MIXTURE LEVEL (2.0-IN. BREAK) FIGURE 15.3-5B



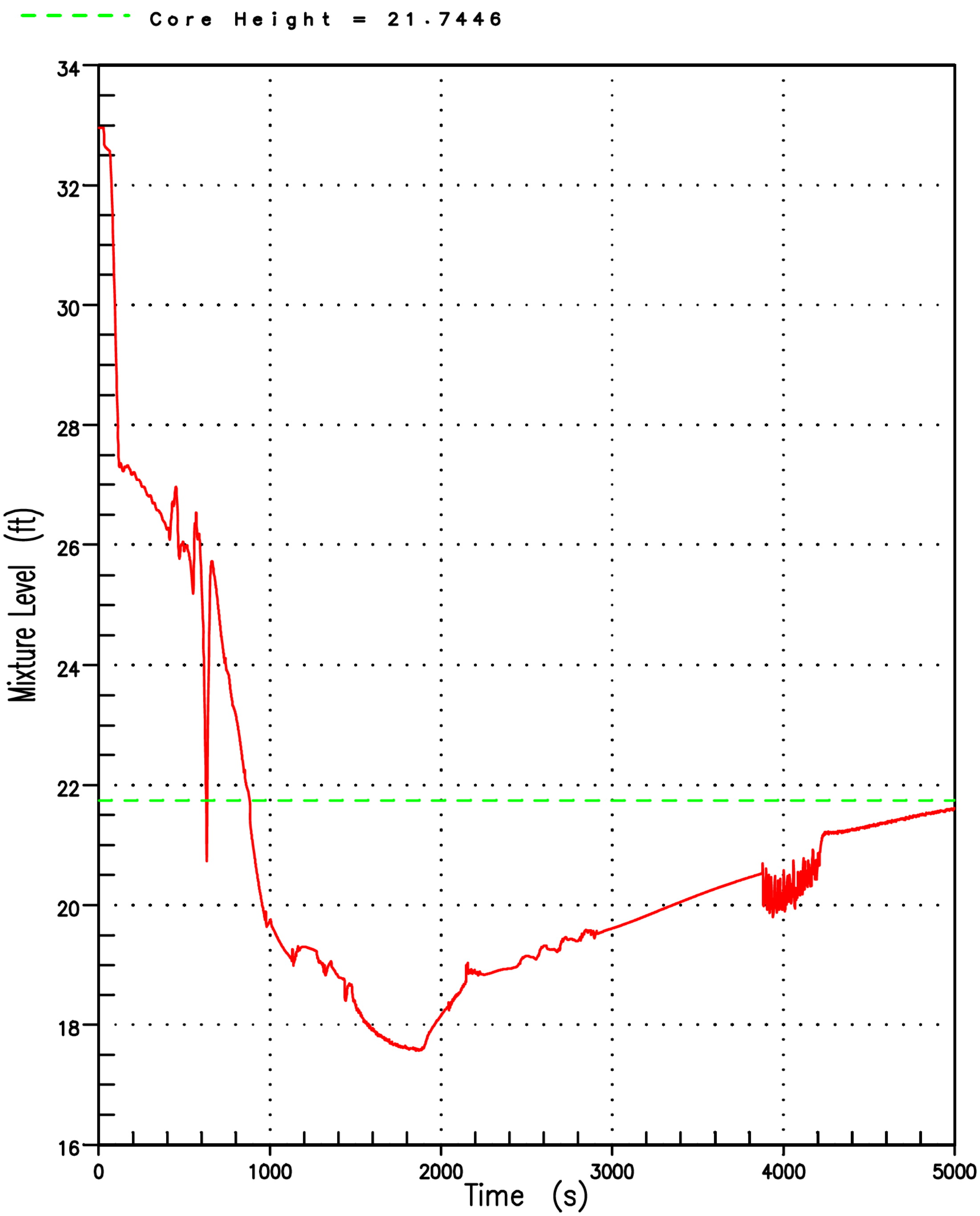
REV 23 5/11

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(2.25-IN. BREAK)

FIGURE 15.3-5C



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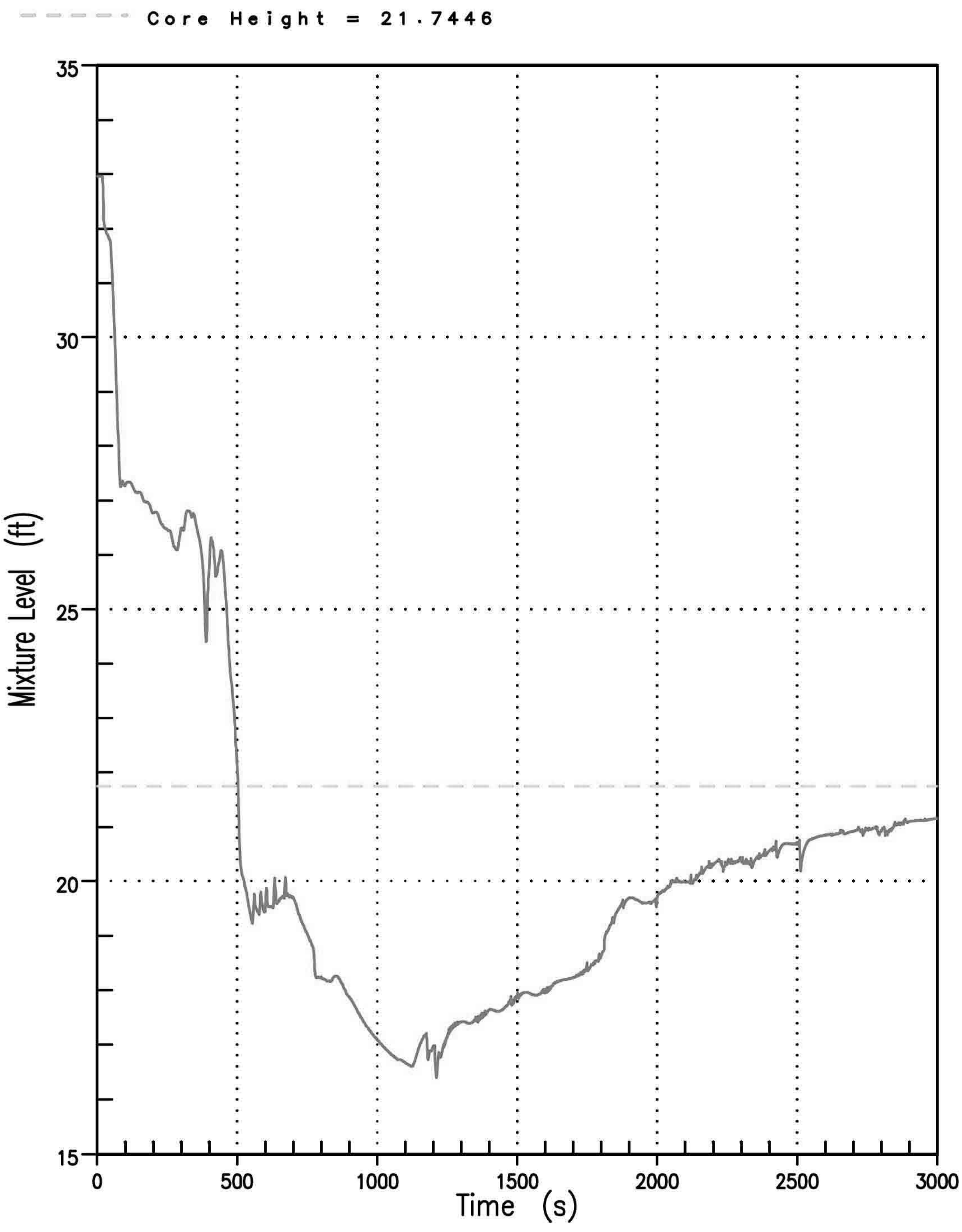
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(2.5-IN. BREAK)

FIGURE 15.3-5D



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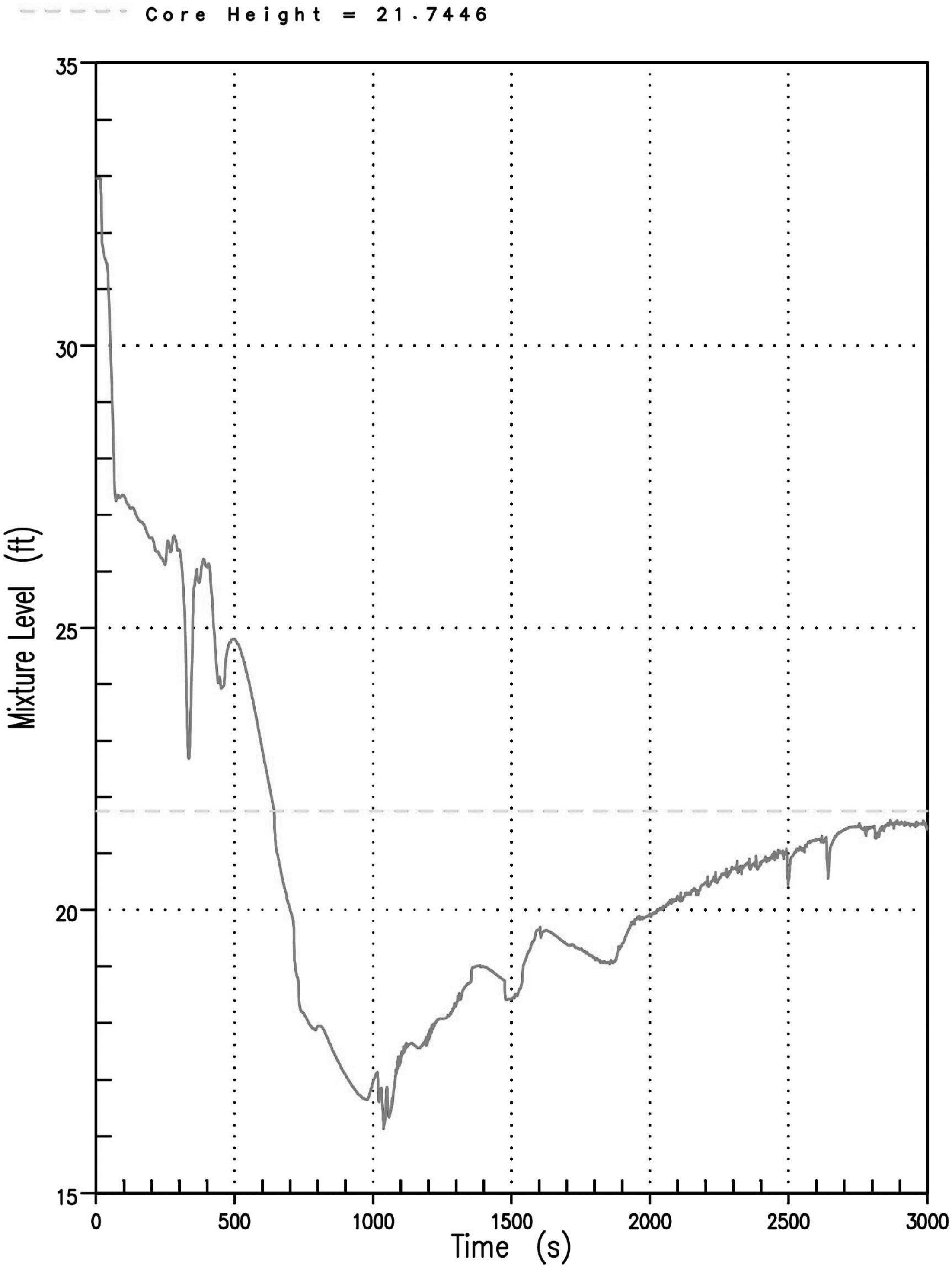
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(3.0-IN. BREAK)

FIGURE 15.3-5E



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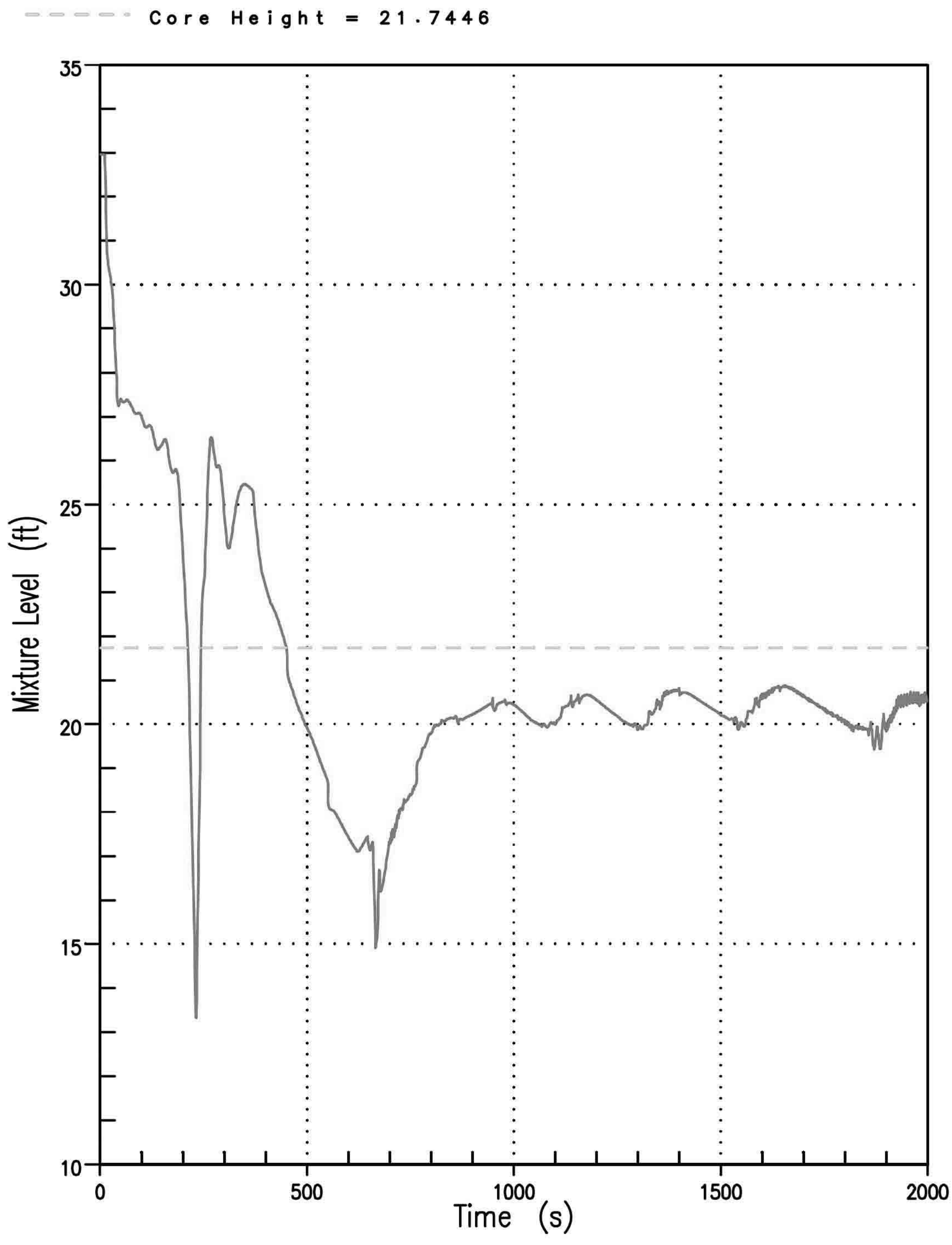
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(3.25-IN. BREAK)

FIGURE 15.3-5F



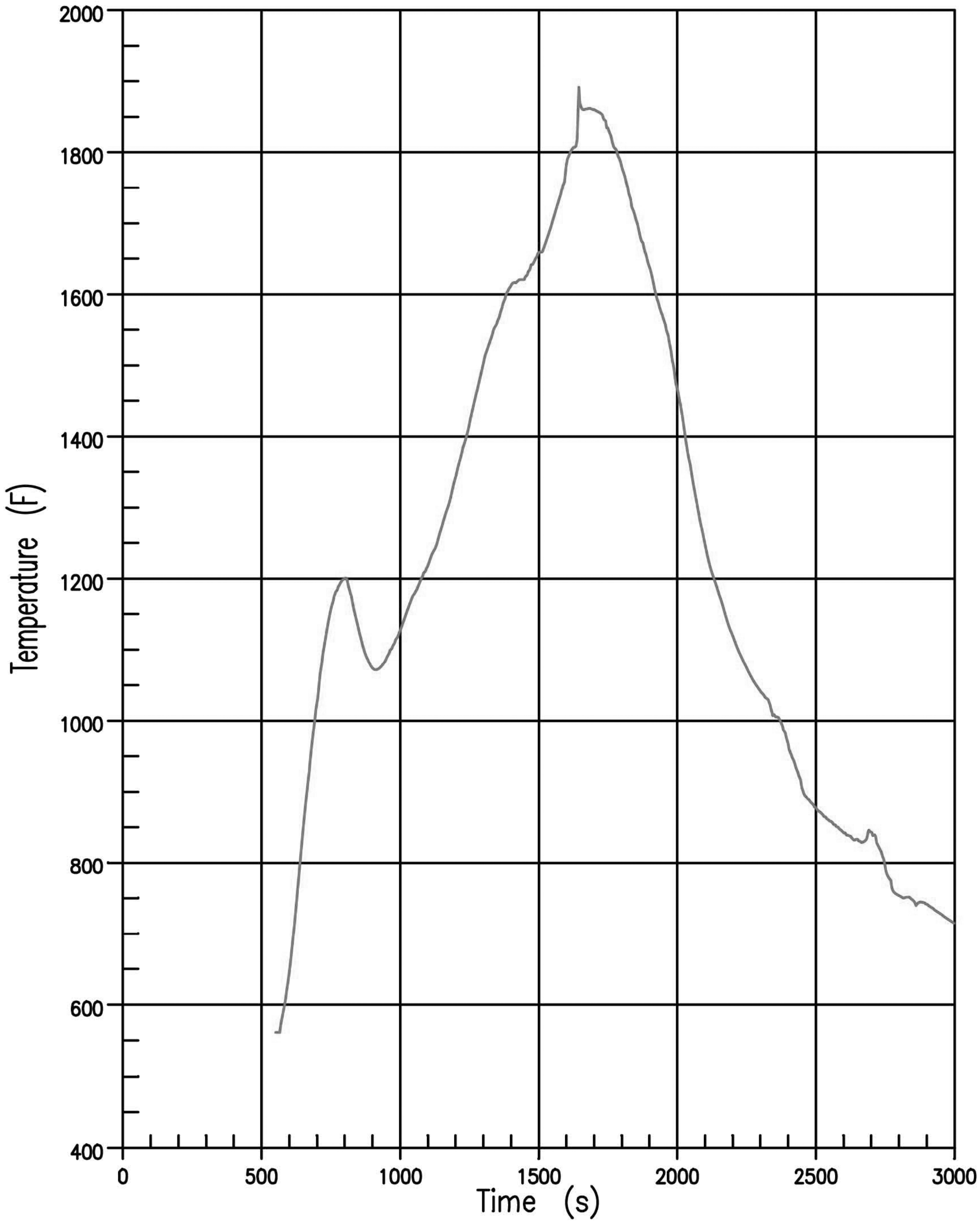
REV 23 5/11

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE MIXTURE LEVEL
(4.0-IN. BREAK)

FIGURE 15.3-5G



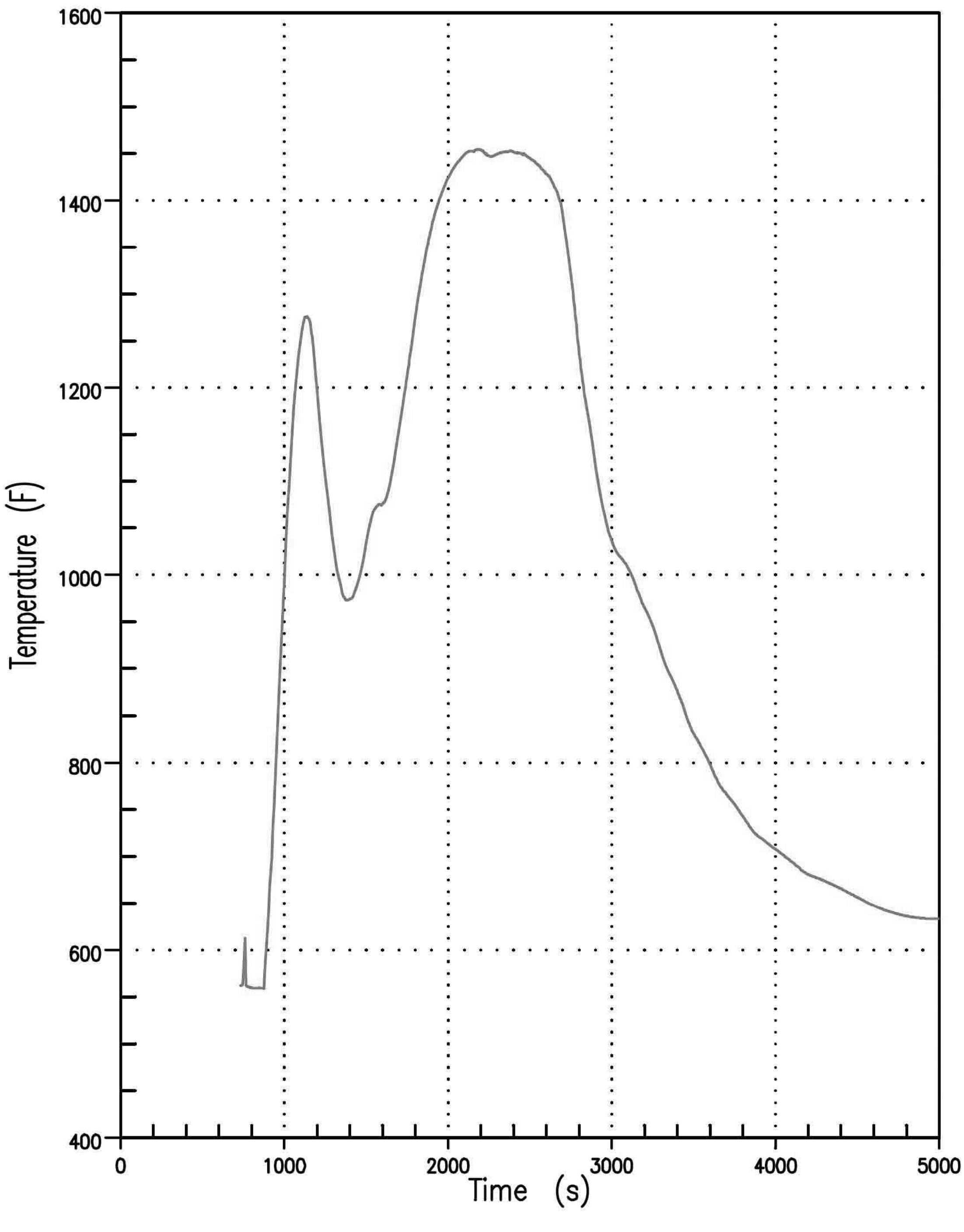
REV 23 5/11

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (2.75-IN. BREAK)

FIGURE 15.3-6A



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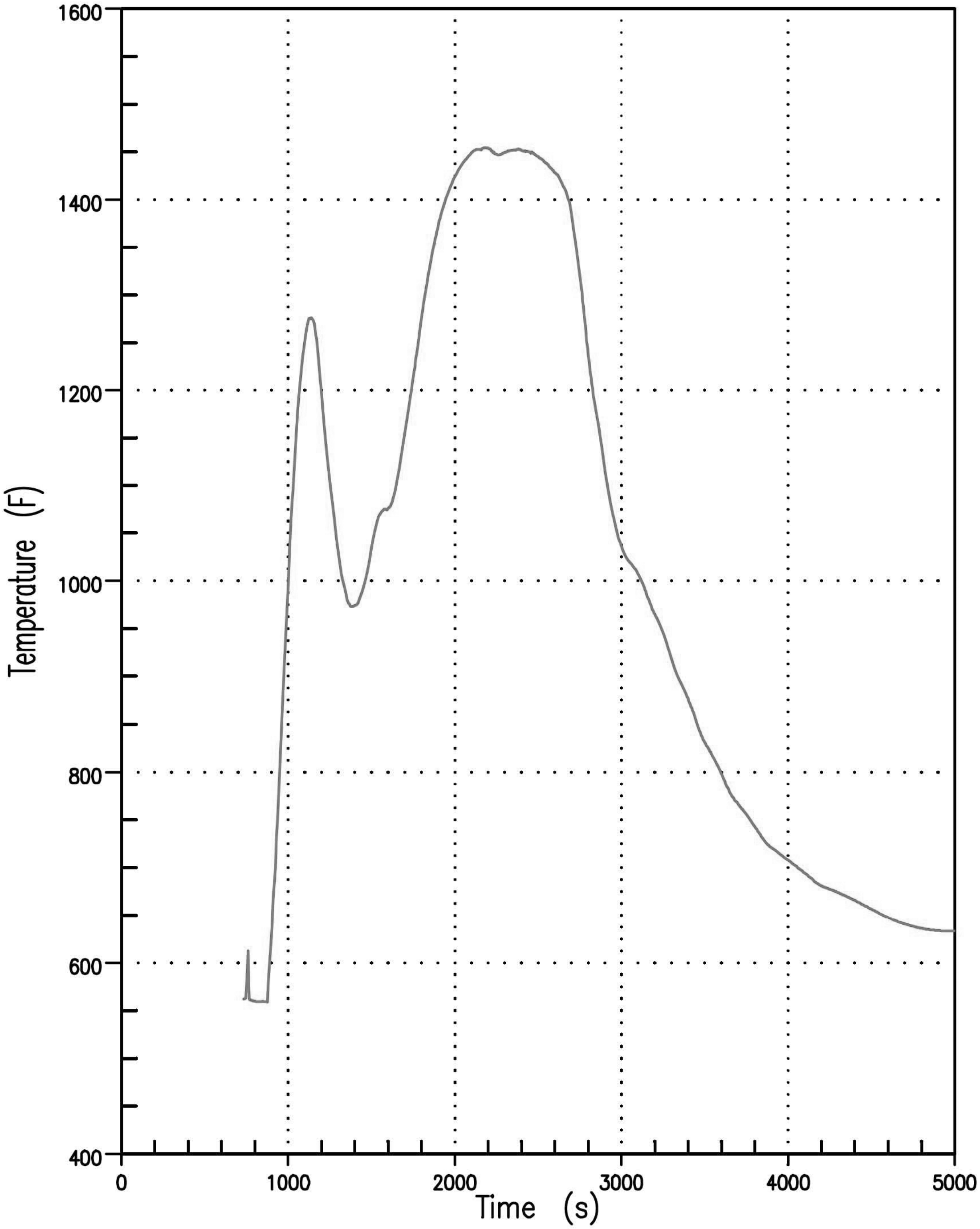
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (2.25-IN. BREAK)

FIGURE 15.3-6C



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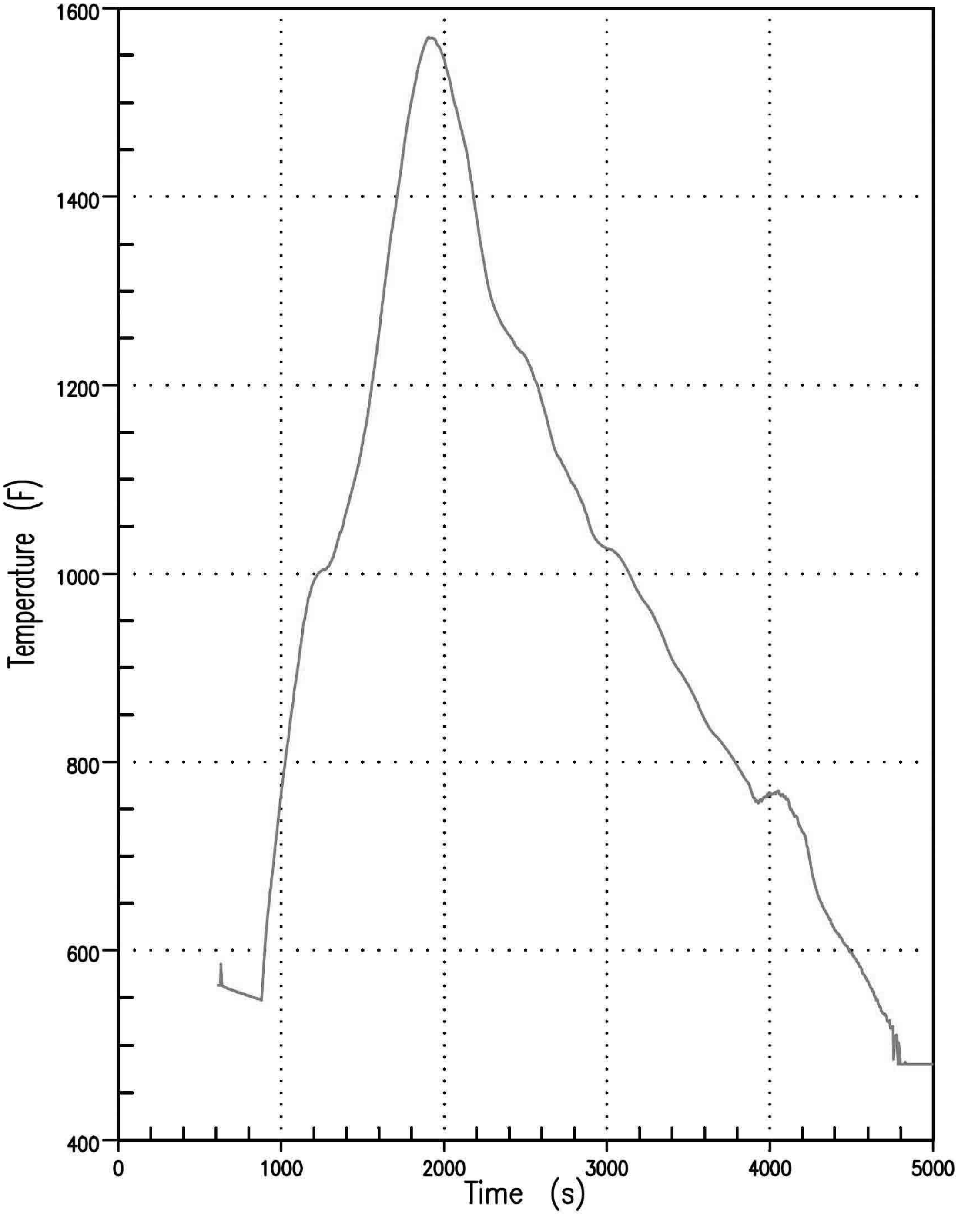
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (2.25-IN. BREAK)
		FIGURE 15.3-6C



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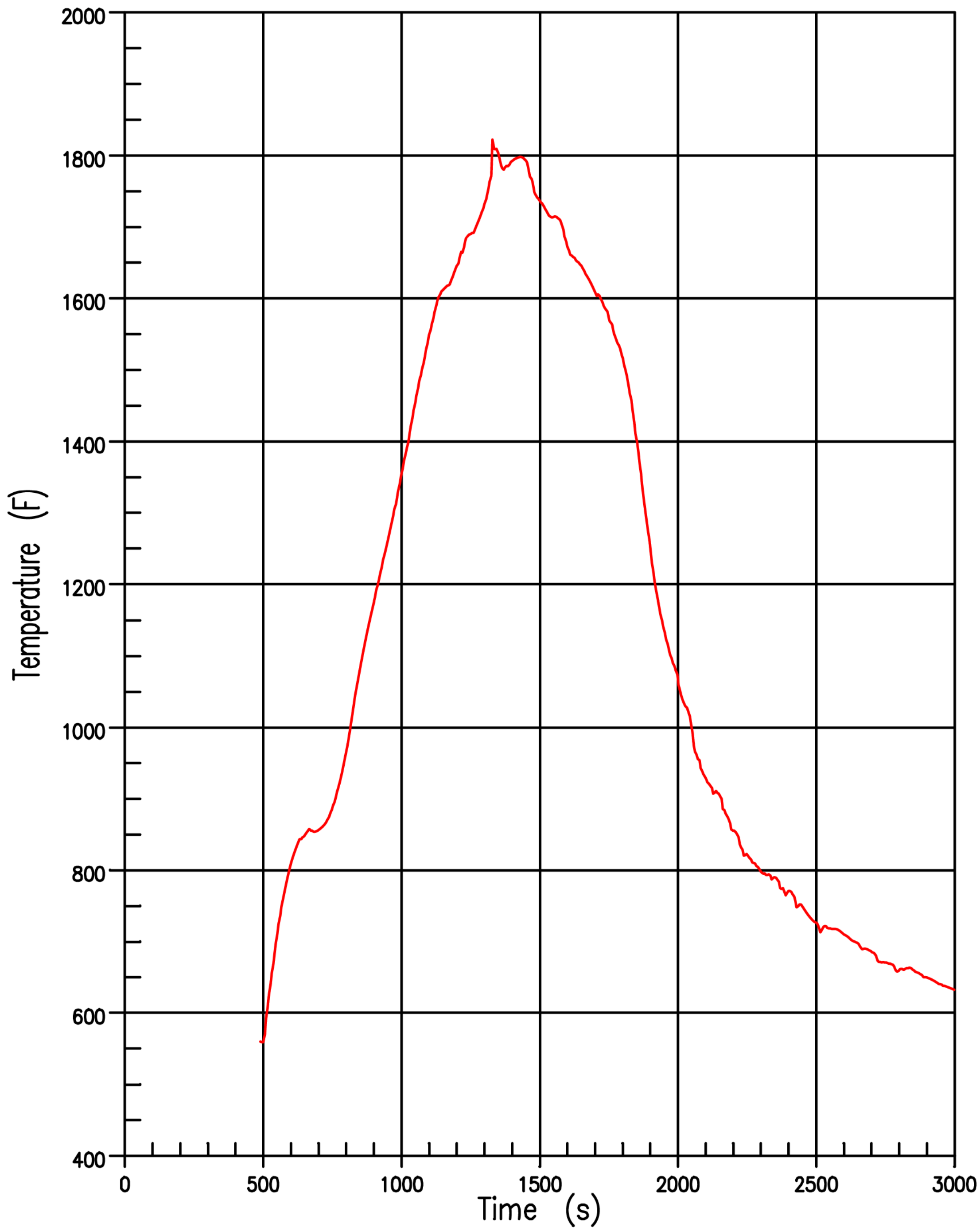
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (2.5-IN. BREAK)

FIGURE 15.3-6D



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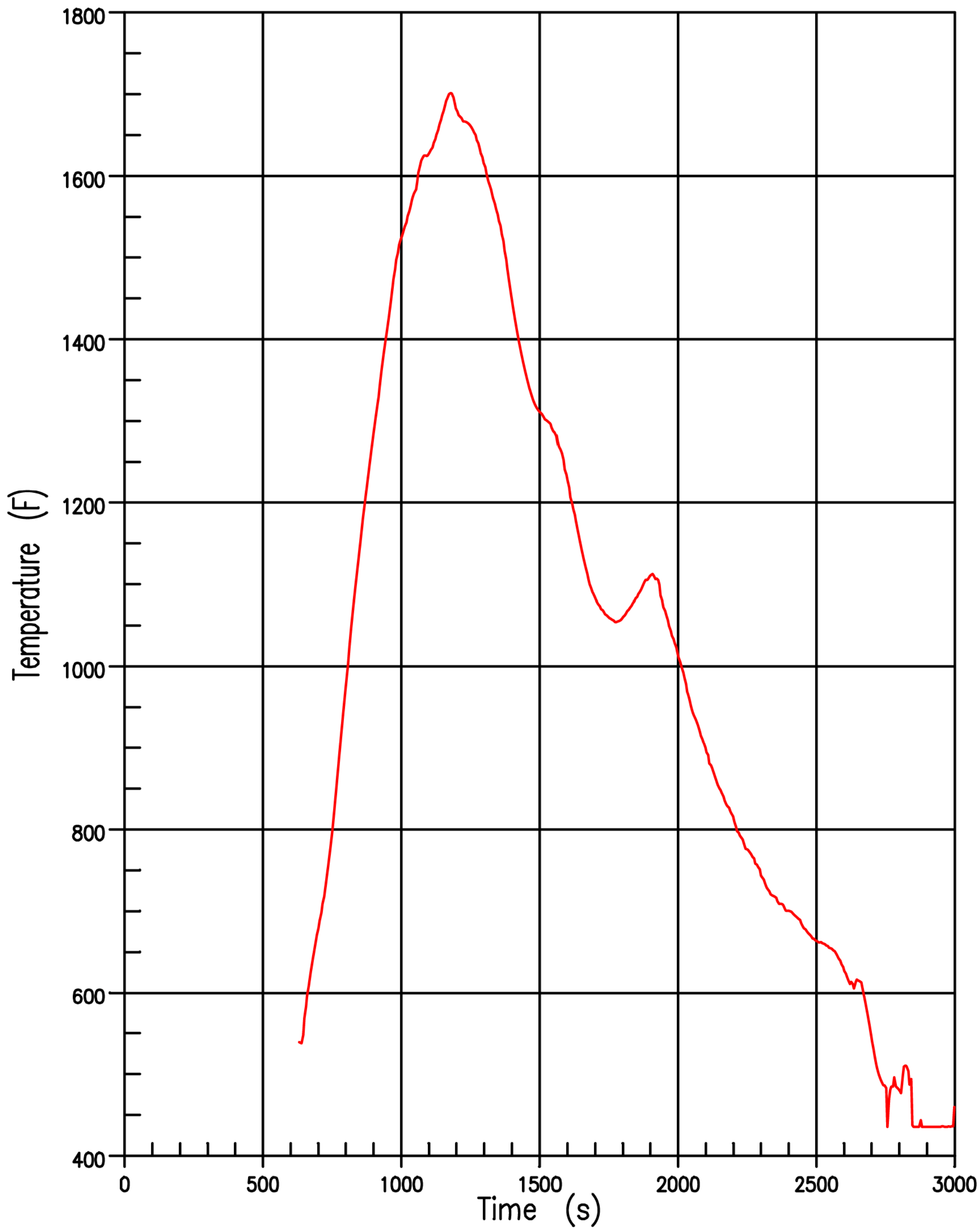
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (3.0-IN. BREAK)

FIGURE 15.3-6E



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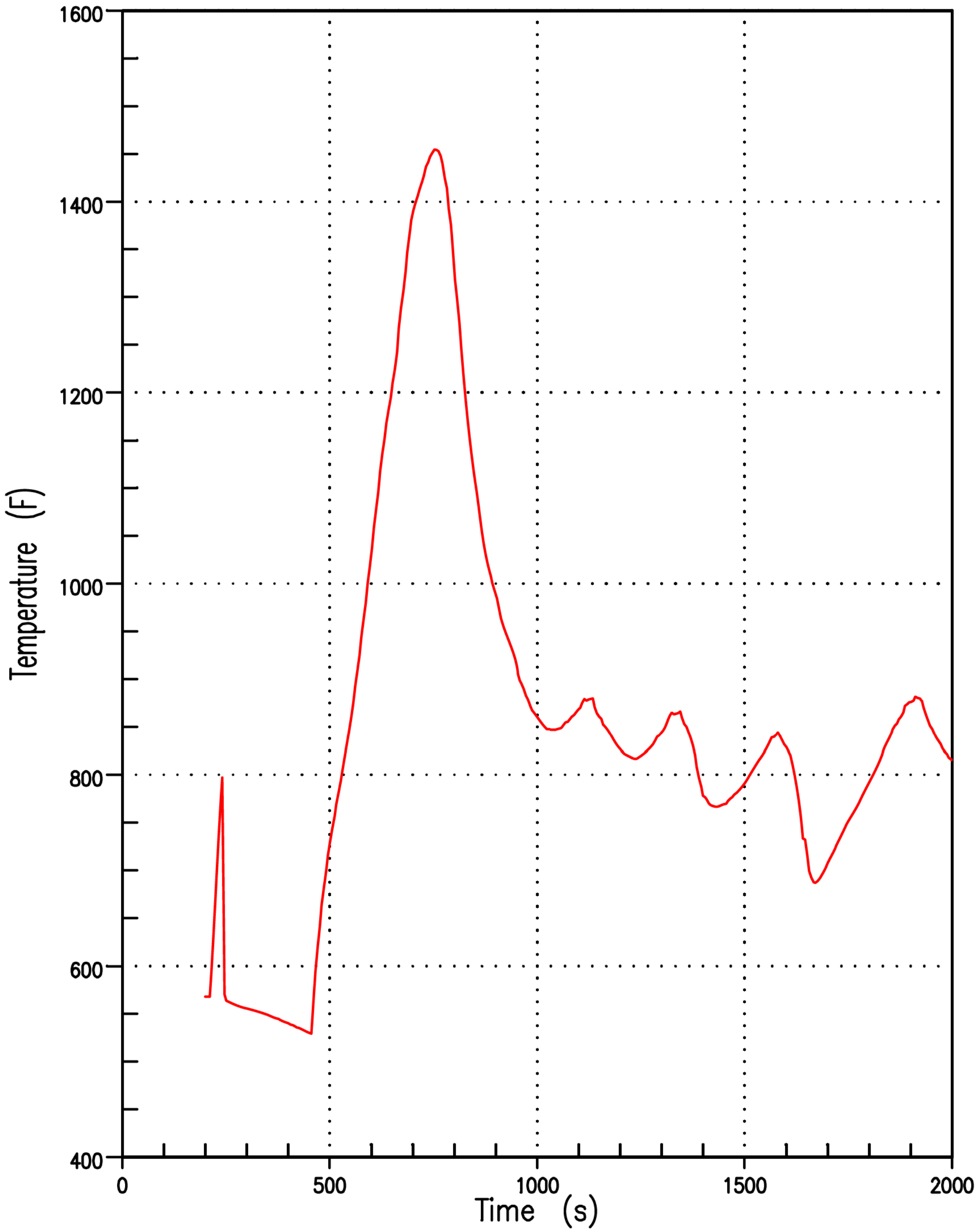
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (3.25-IN. BREAK)

FIGURE 15.3-6F



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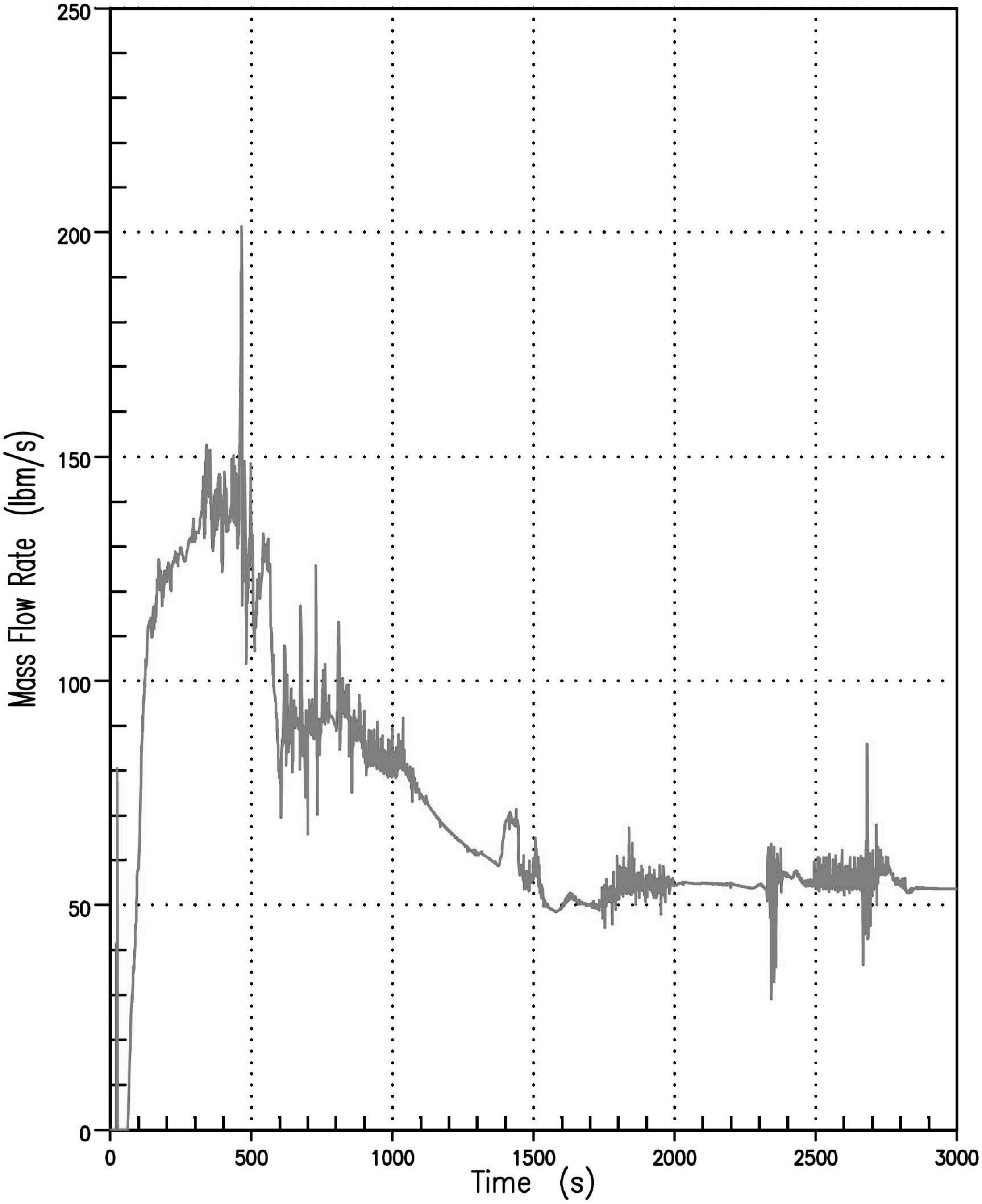
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION (4.0-IN. BREAK)

FIGURE 15.3-6G



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE EXIT STEAM FLOW (2.75-IN. BREAK)

FIGURE 15.3-7A

THIS FIGURE HAS BEEN DELETED.

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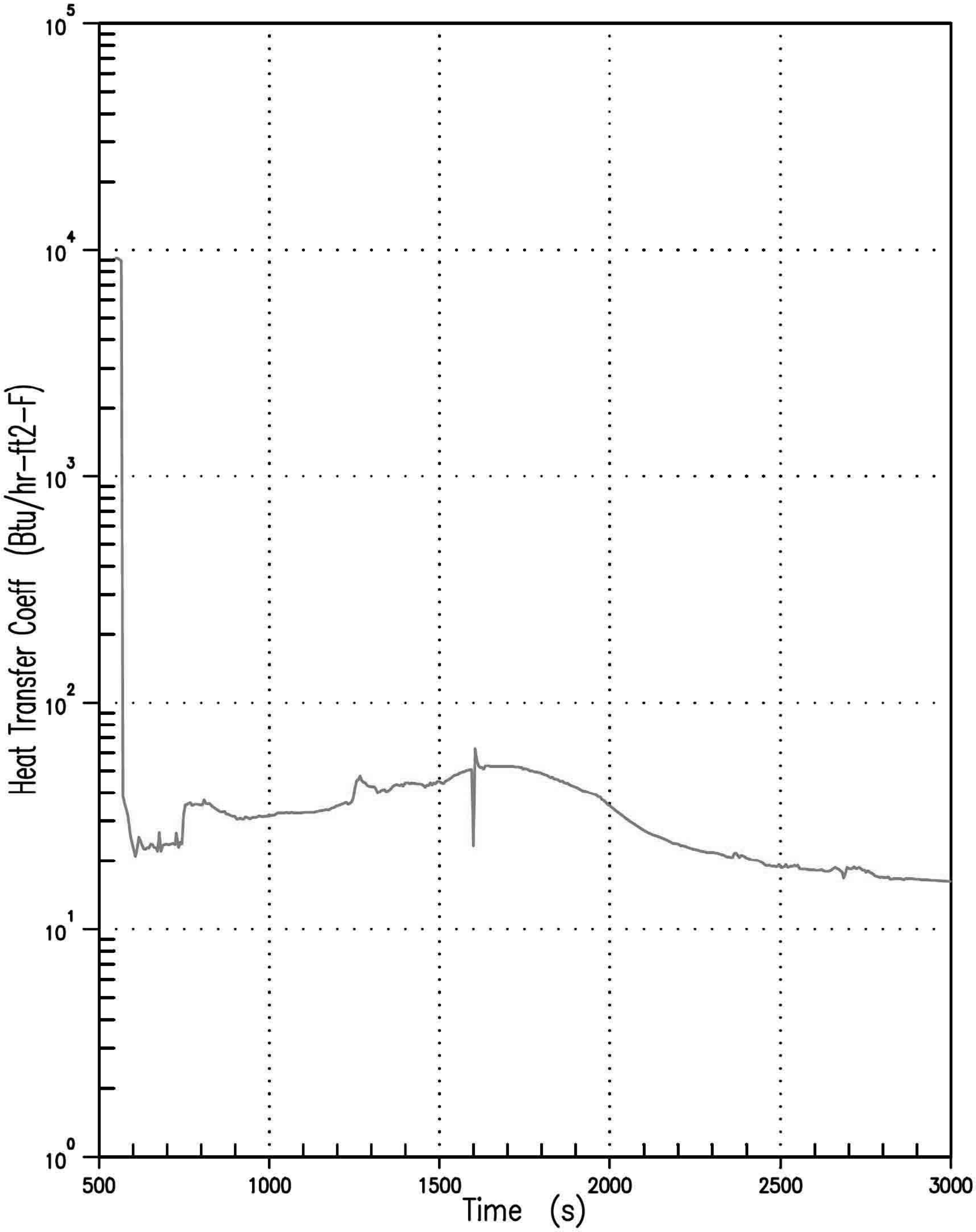
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE EXIT STEAM FLOW – UNIT 2
(LOW T_{AVG}) (3-IN. BREAK)

FIGURE 15.3-7B



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD SURFACE HEAT TRANSFER COEFFICIENT AT PCT ELEVATION (2.75-IN. BREAK)

FIGURE 15.3-8A

THIS FIGURE HAS BEEN DELETED.

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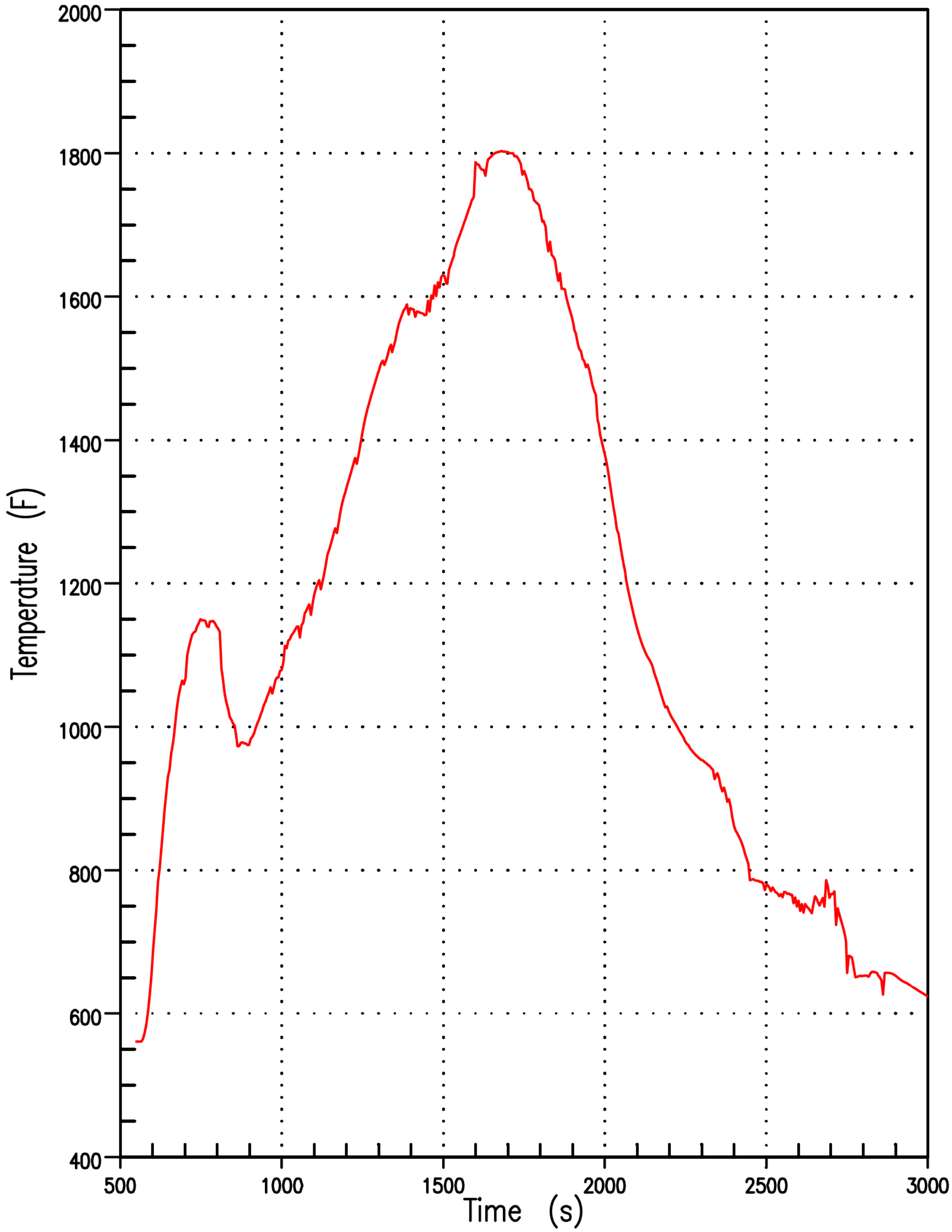
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	HEAT TRANSFER COEFFICIENT – HOT SPOT ROD SURFACE – UNIT 2 (LOW T_{AVG}) (3-IN. BREAK) FIGURE 15.3-8B



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE AT PCT ELEVATION
(2.75-IN. BREAK)

FIGURE 15.3-9A

THIS FIGURE HAS BEEN DELETED.

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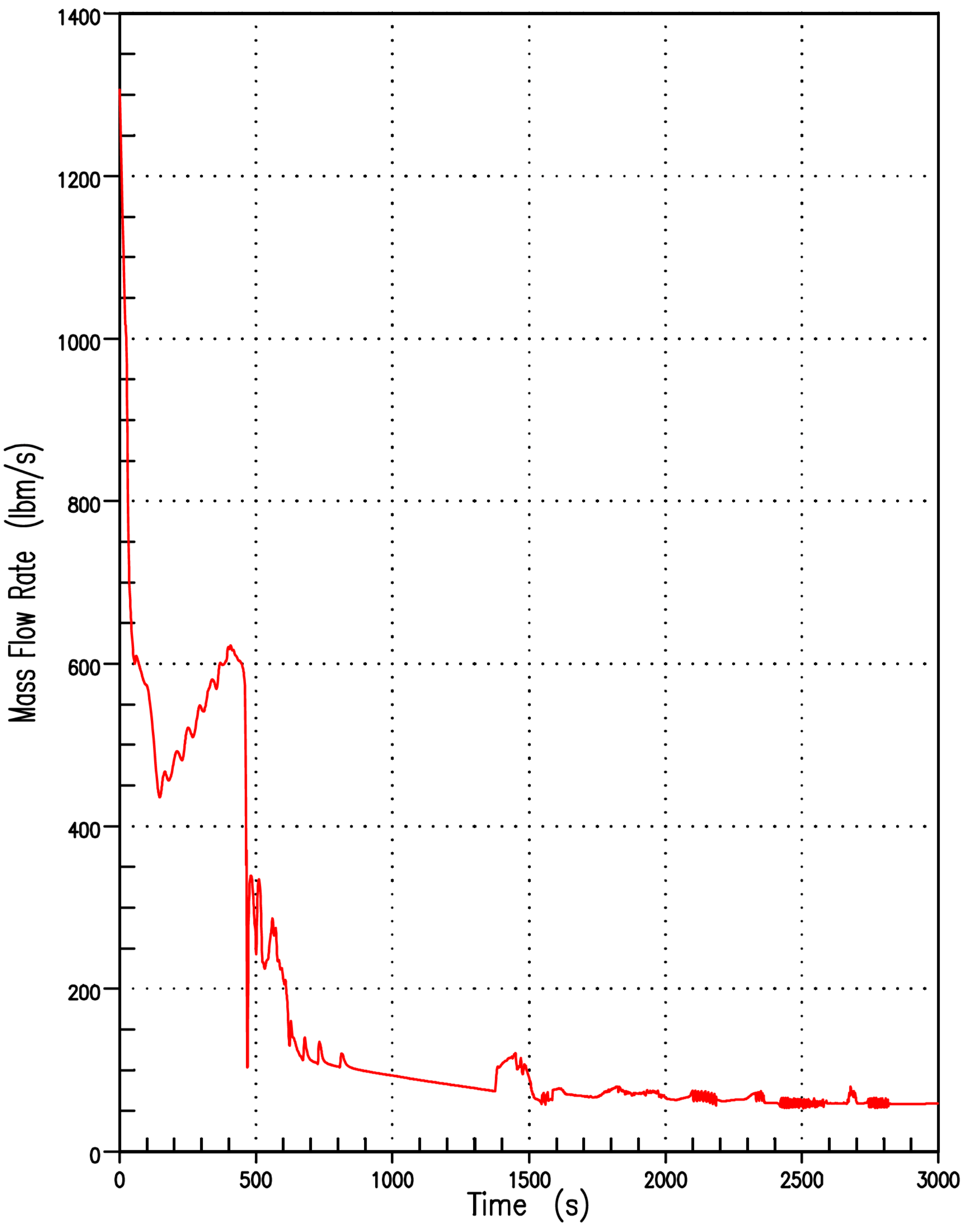
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FLUID TEMPERATURE – HOT SPOT – UNIT 2
(LOW T_{AVG}) (3-IN. BREAK)

FIGURE 15.3-9B



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

COLD LEG BREAK MASS FLOW (2.75-IN. BREAK)

FIGURE 15.3-10A

THIS FIGURE HAS BEEN DELETED.

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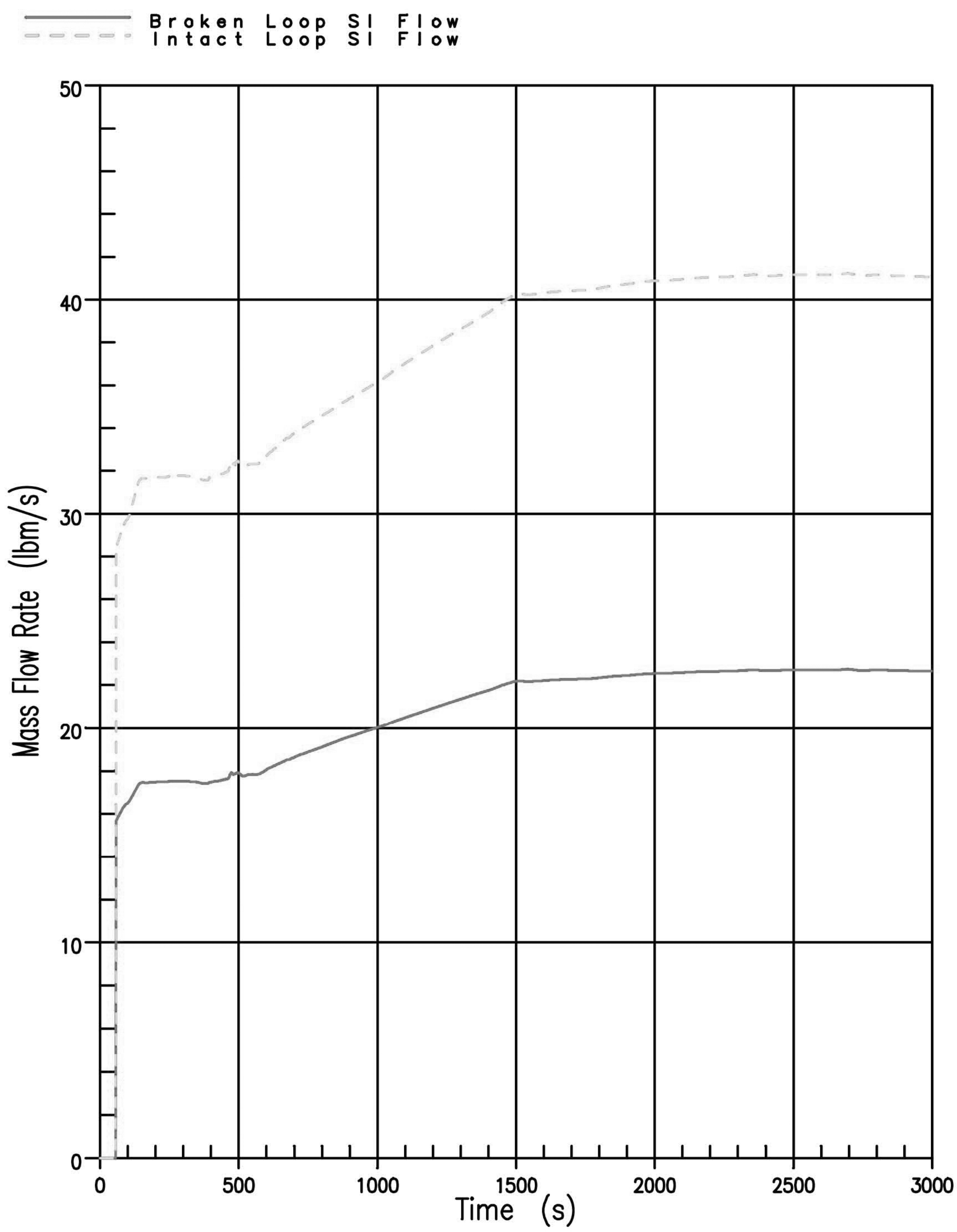
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

COLD LEG BREAK MASS FLOW – UNIT 2
(LOW T_{AVG}) (3-IN. BREAK)

FIGURE 15.3-10B



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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ECCS PUMPED SAFETY INJECTION (2.75-IN. BREAK)
		FIGURE 15.3-11A

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ECCS PUMPED SAFETY INJECTION – UNIT 2 (LOW T_{AVG}) (3-IN. BREAK)
		FIGURE 15.3-11B

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CORE MIXTURE HEIGHT (4-IN.)
		FIGURE 15.3-12

THIS FIGURE HAS BEEN DELETED.

11

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT
(3-IN.)

FIGURE 15.3-13

THIS FIGURE HAS BEEN DELETED.

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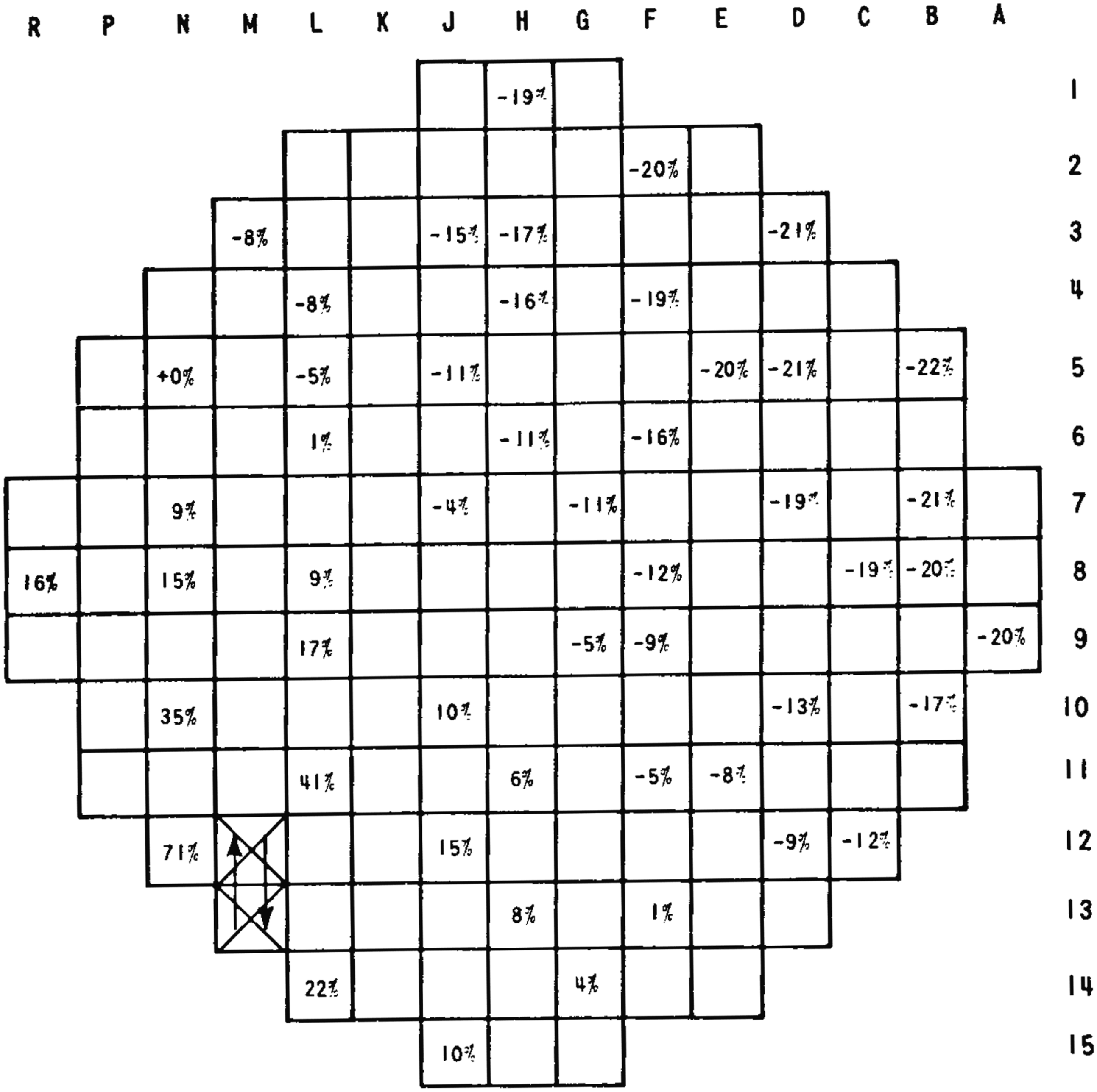
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLAD TEMPERATURE TRANSIENT
(4-IN.)

FIGURE 15.3-14



CASE A

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTERCHANGE BETWEEN REGION 1
AND REGION 3 ASSEMBLY

FIGURE 15.3-15

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								1%							
2										2%					
3				0%			1%	1%				2%			
4					0%			1%		2%					
5			-0%		0%		1%				3%	3%		3%	
6					0%			1%		3%					
7			-0%				0%		2%			4%		2%	
8	-1%		-1%	-0%	-0%					4%			2%	1%	
9					-1%				-1%	-3%					+1%
10			-1%				-1%					-3%		-1%	
11					-1%			-2%		-3%	-3%				
12			-1%				-2%					-2%	-2%		
13								-2%		-2%					
14					-2%				-2%						
15							-2%								

CASE B-1

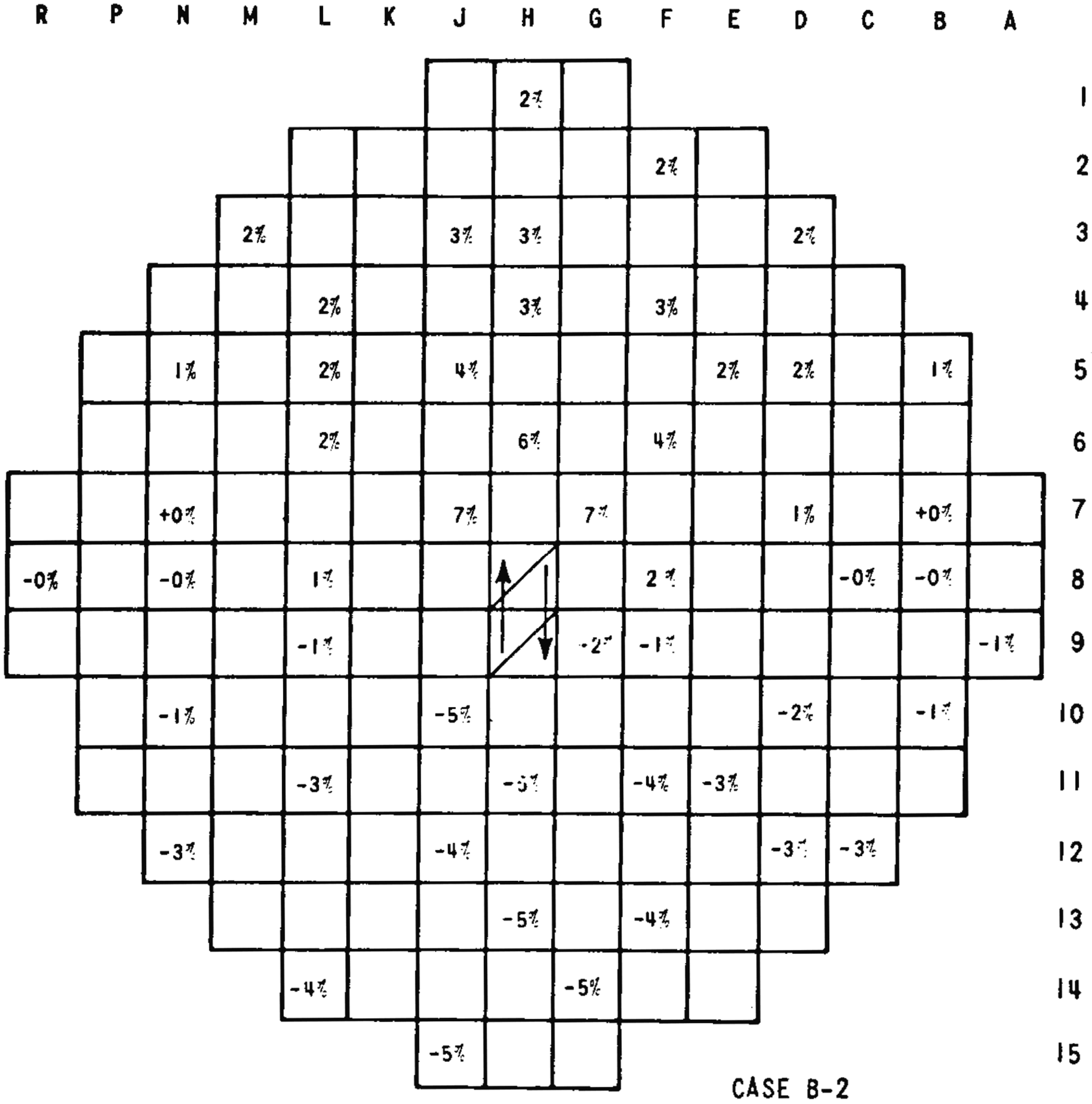
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTERCHANGE BETWEEN REGION 1 AND 2 ASSEMBLY, BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY

FIGURE 15.3-16



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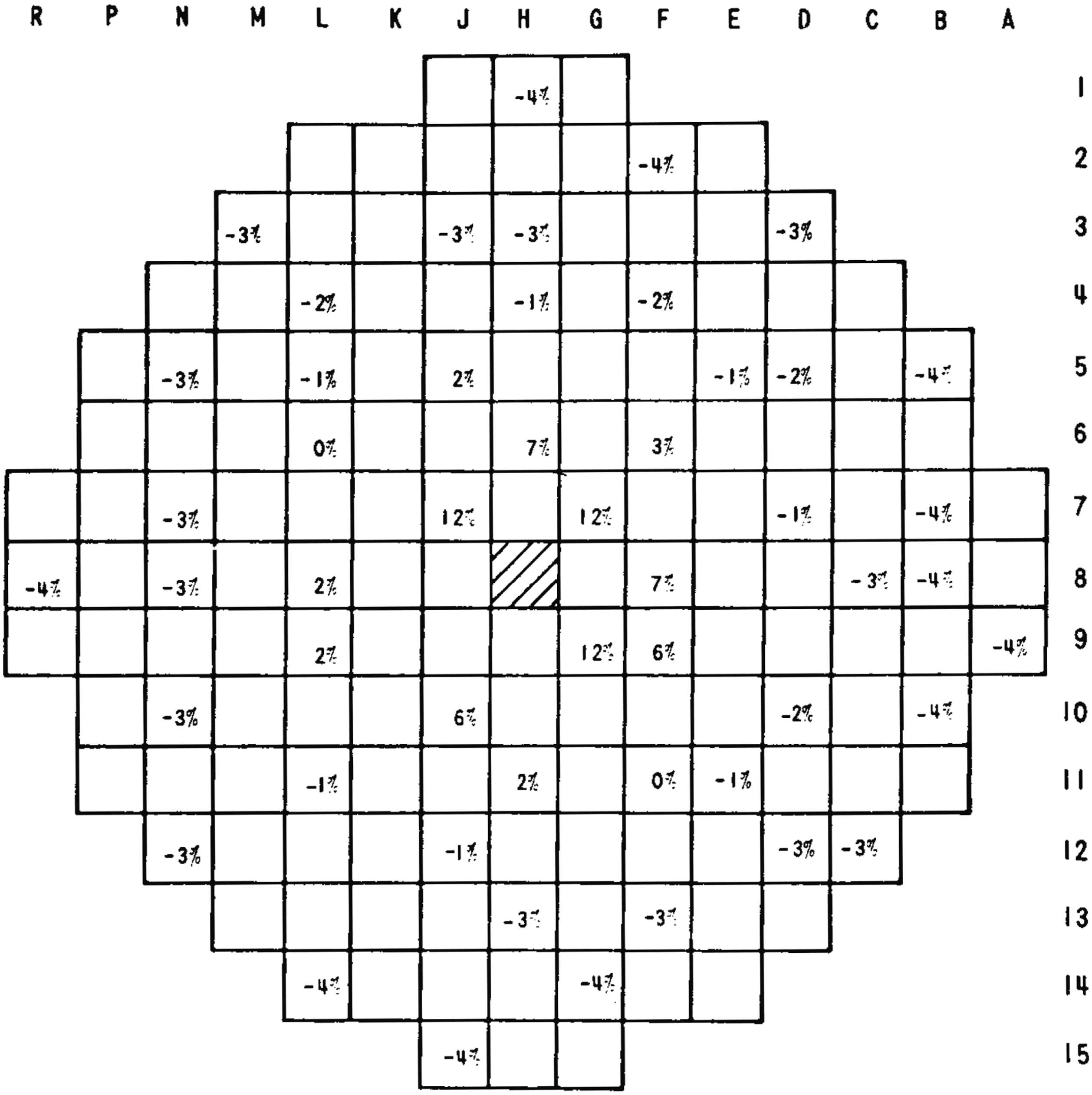
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING TRANSFERRED TO THE REGION 1 ASSEMBLY

FIGURE 15.3-17



CASE C

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ENRICHMENT ERROR – A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION

FIGURE 15.3-18

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1								-18%							
2										-19%					
3				-9%			-15%	-17%				-20%			
4					-9%			-15%		-18%					
5			-2%		-6%		-11%				-18%	-19%		-20%	
6					-3%			-11%		-15%					
7			5%				-4%		-11%			-17%		-19%	
8	11%		11%		7%					-11%			-17%	-18%	
9					14%				-4%	-8%					-17%
10			28%				10%					-11%		-15%	
11					35%			7%		-3%	-6%				
12			60%				17%					-7%	-9%		
13								11%		1%					
14					35%				7%						
15							15%								

CASE D

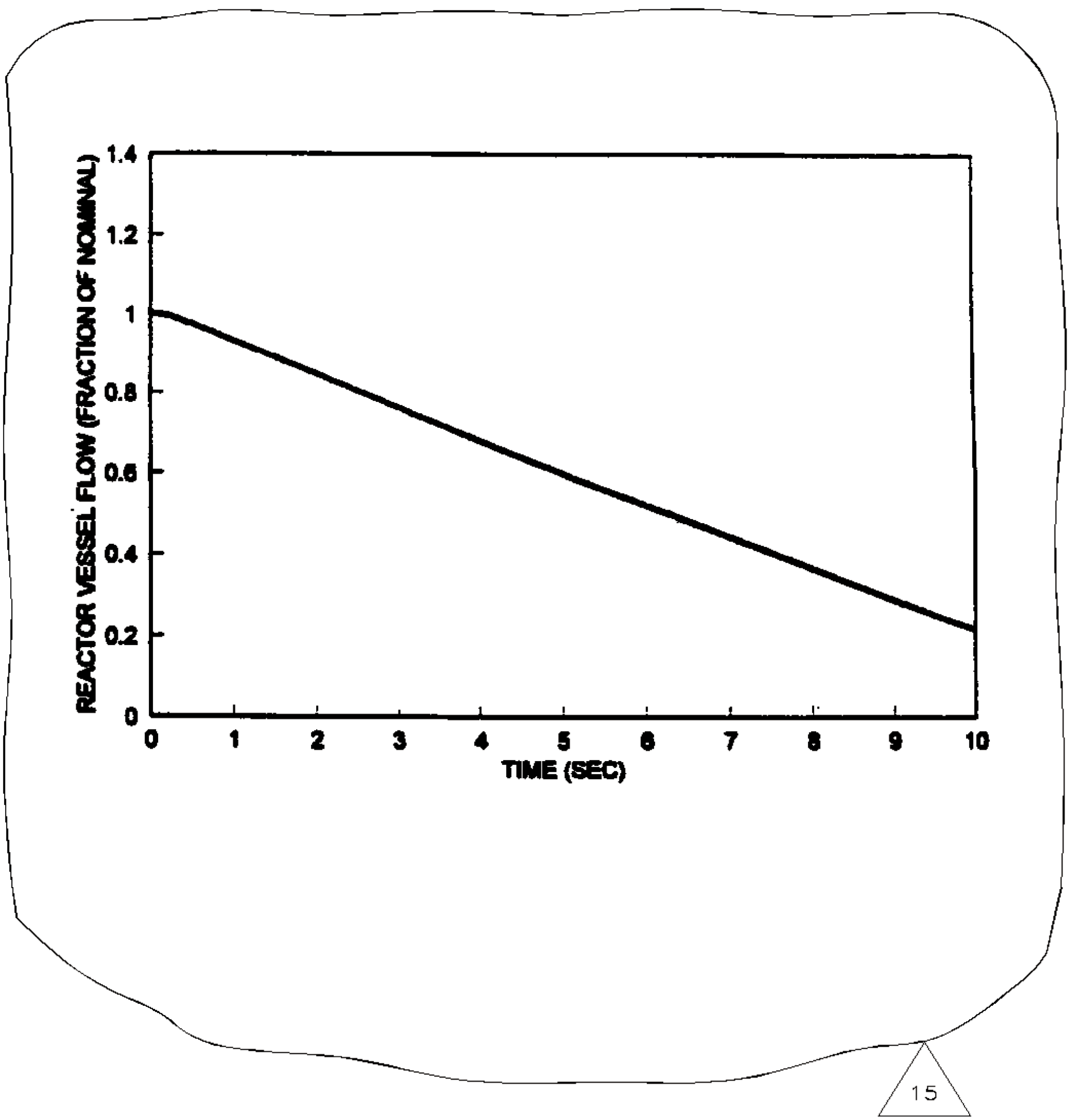
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOADING A REGION 2 ASSEMBLY
INTO A REGION 1 POSITION NEAR CORE
PERIPHERY

FIGURE 15.3-19



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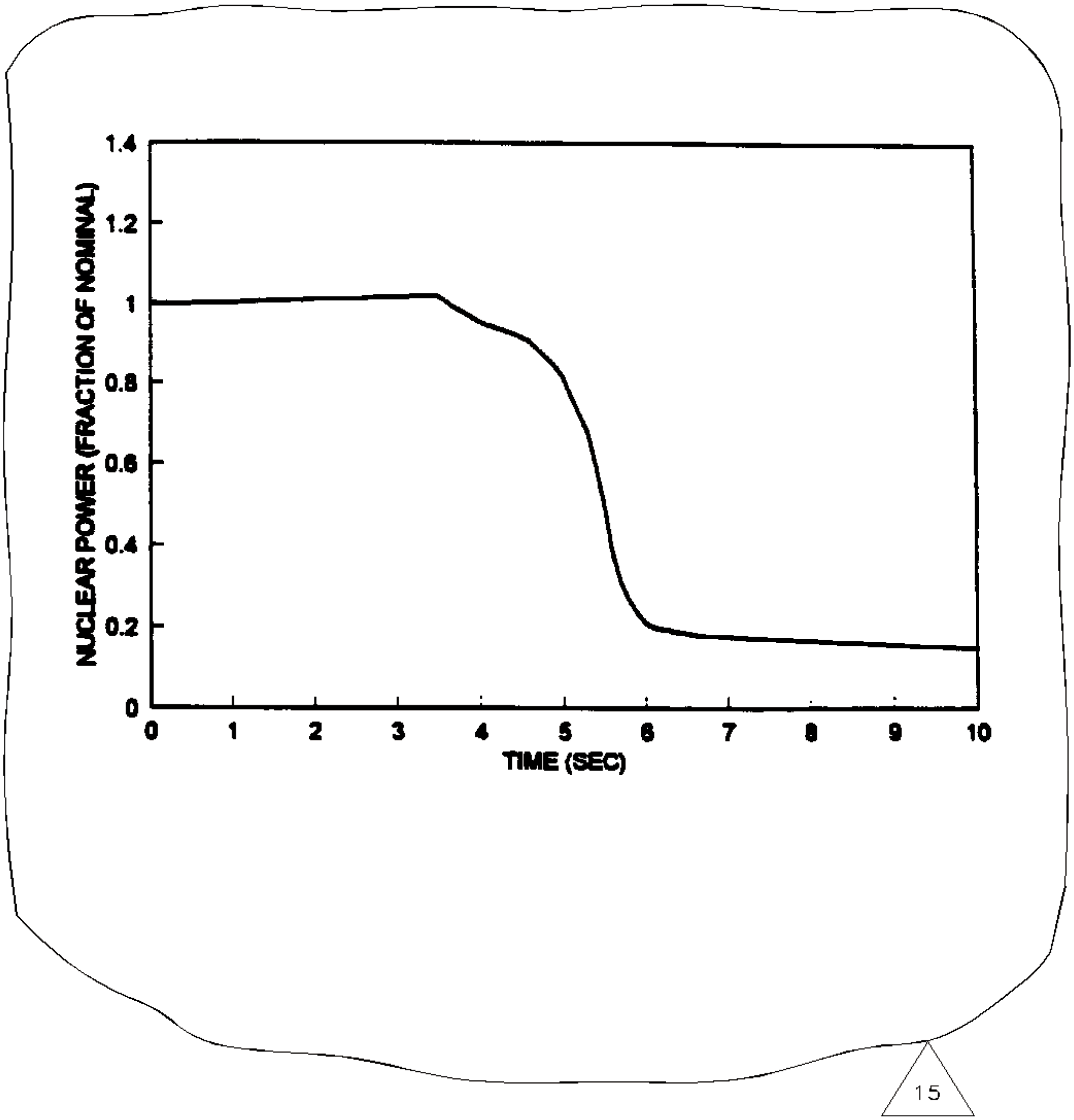
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OEPRATING, FREQUENCY DECAY
TO ALL LOOPS –
CORE FLOW VERSUS TIME

FIGURE 15.3-20



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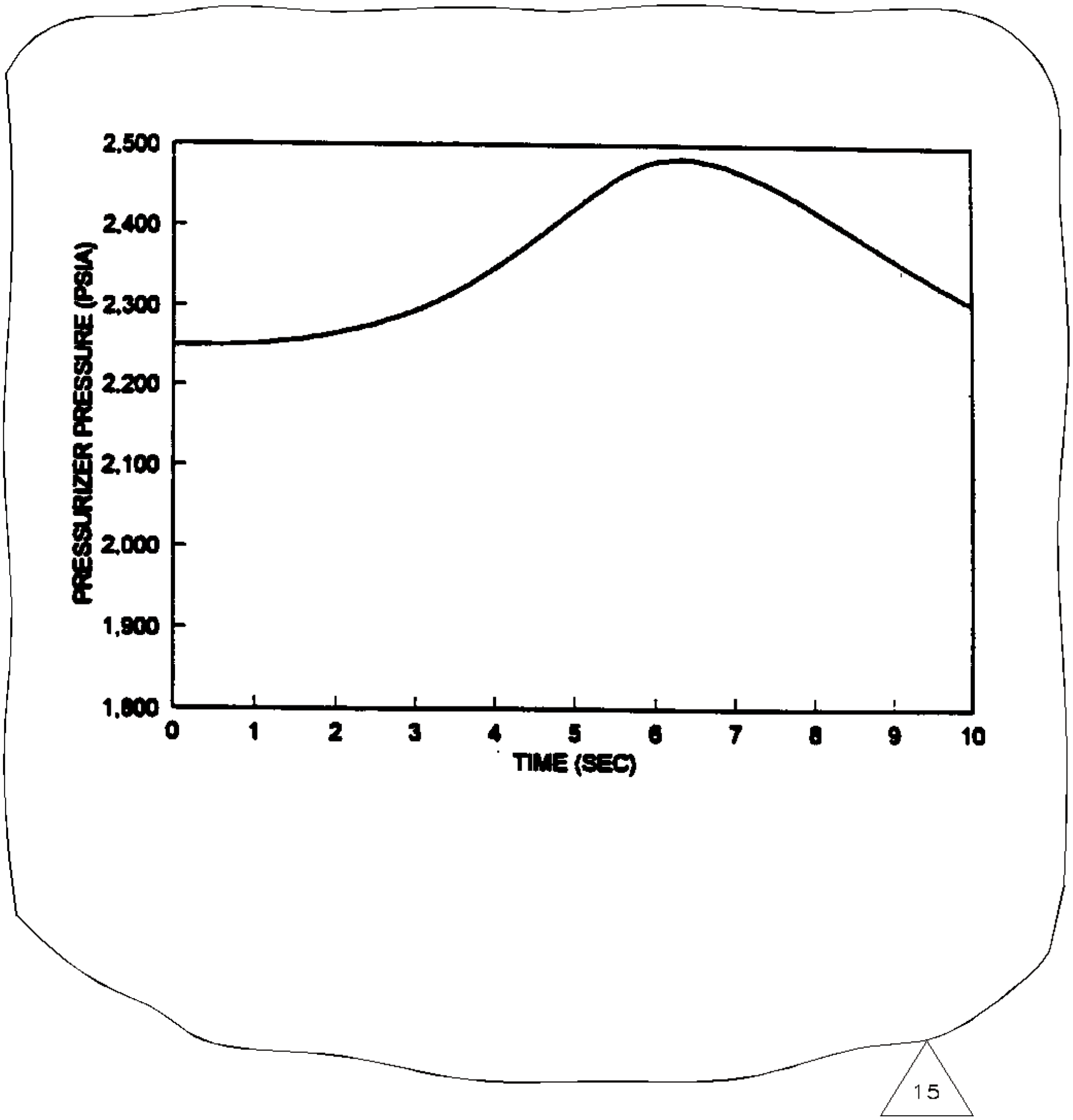
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING, FREQUENCY DECAY
TO ALL LOOPS –
NUCLEAR POWER VERSUS TIME

FIGURE 15.3-21



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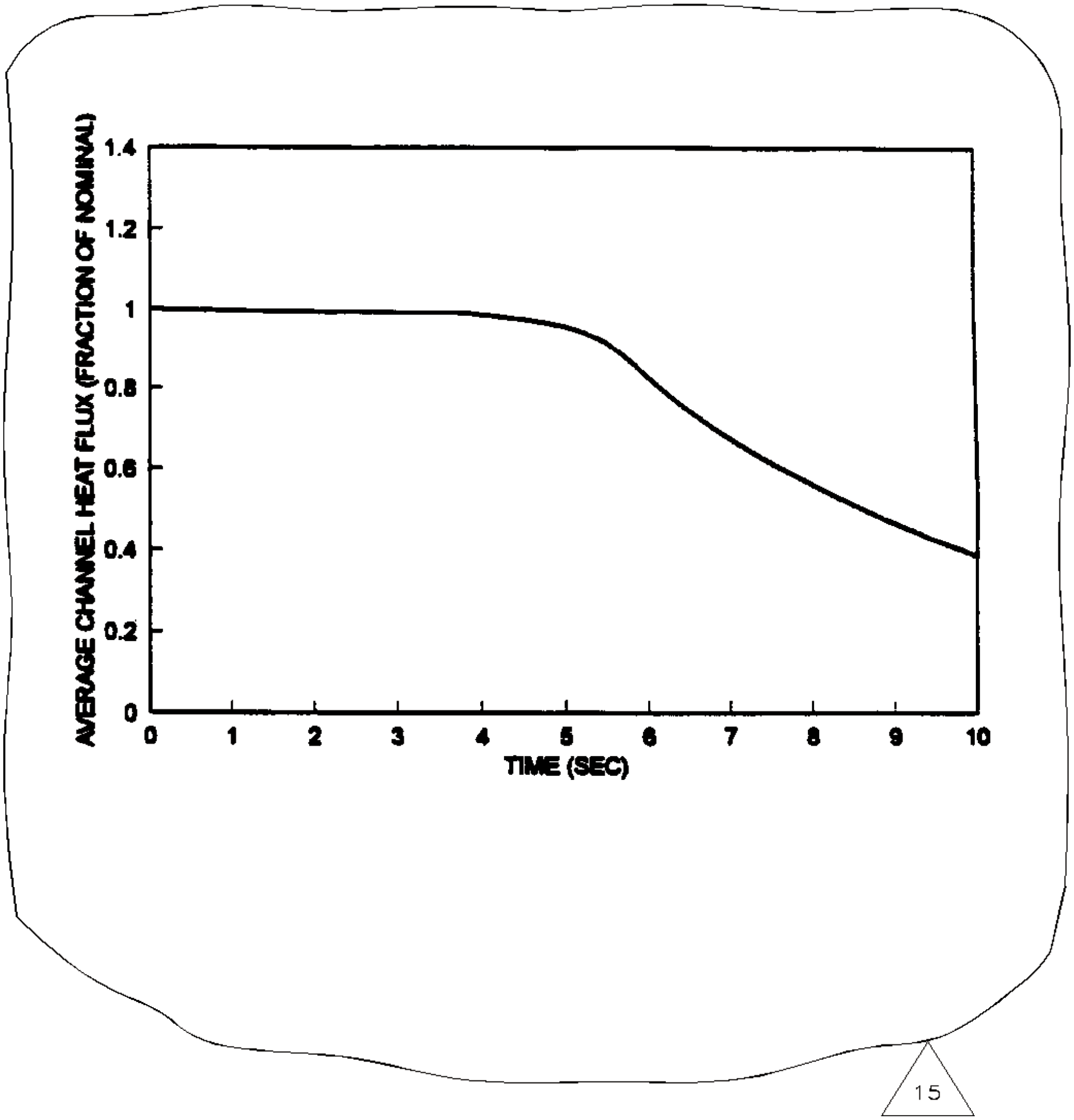
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING, FREQUENCY DECAY
TO ALL LOOPS –
PRESSURIZER PRESSURE VERSUS TIME

FIGURE 15.3-22



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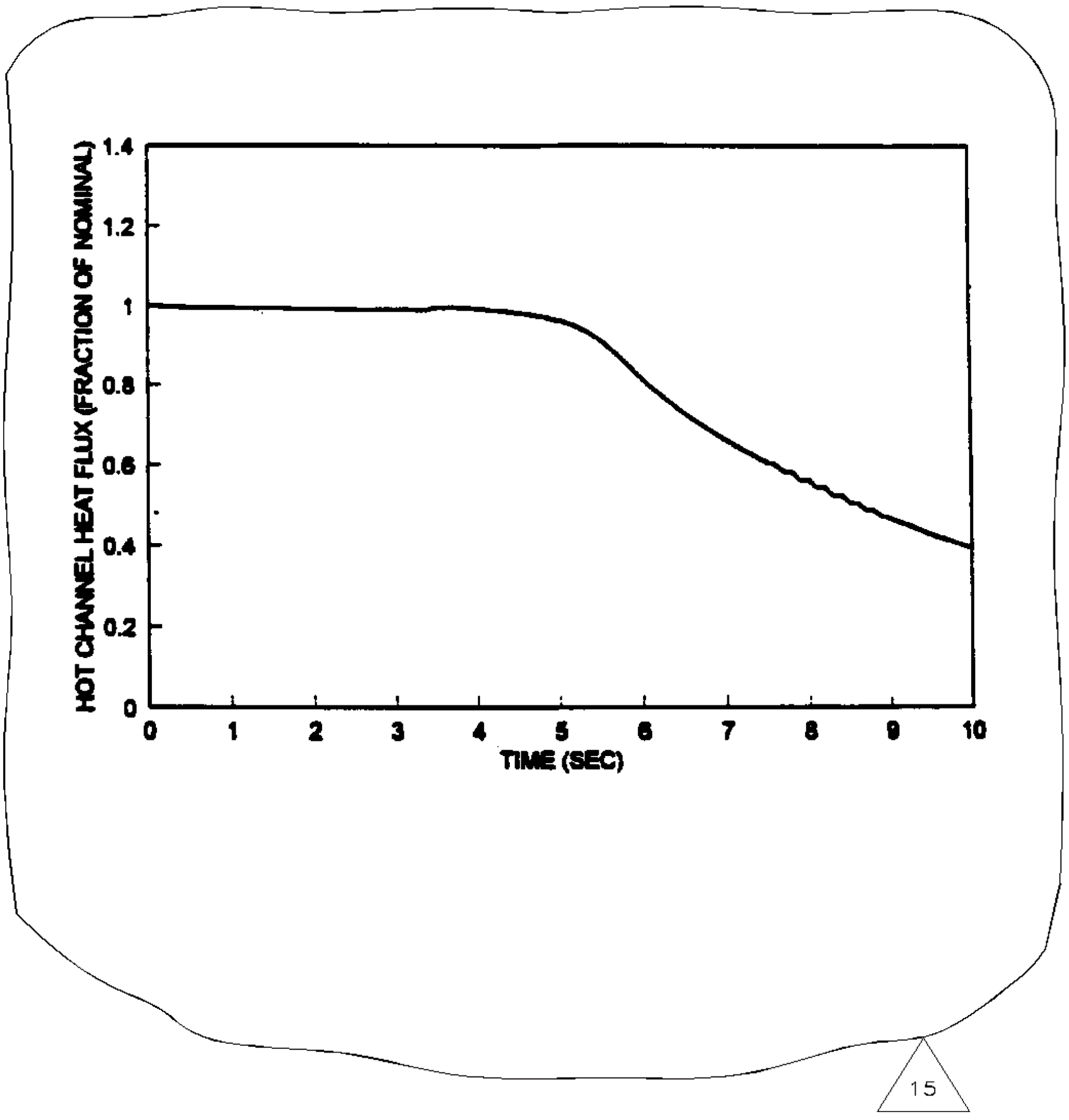
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING, FREQUENCY DECAY
TO ALL LOOPS – AVERAGE CHANNEL
HEAT FLUX VERSUS TIME

FIGURE 15.3-23



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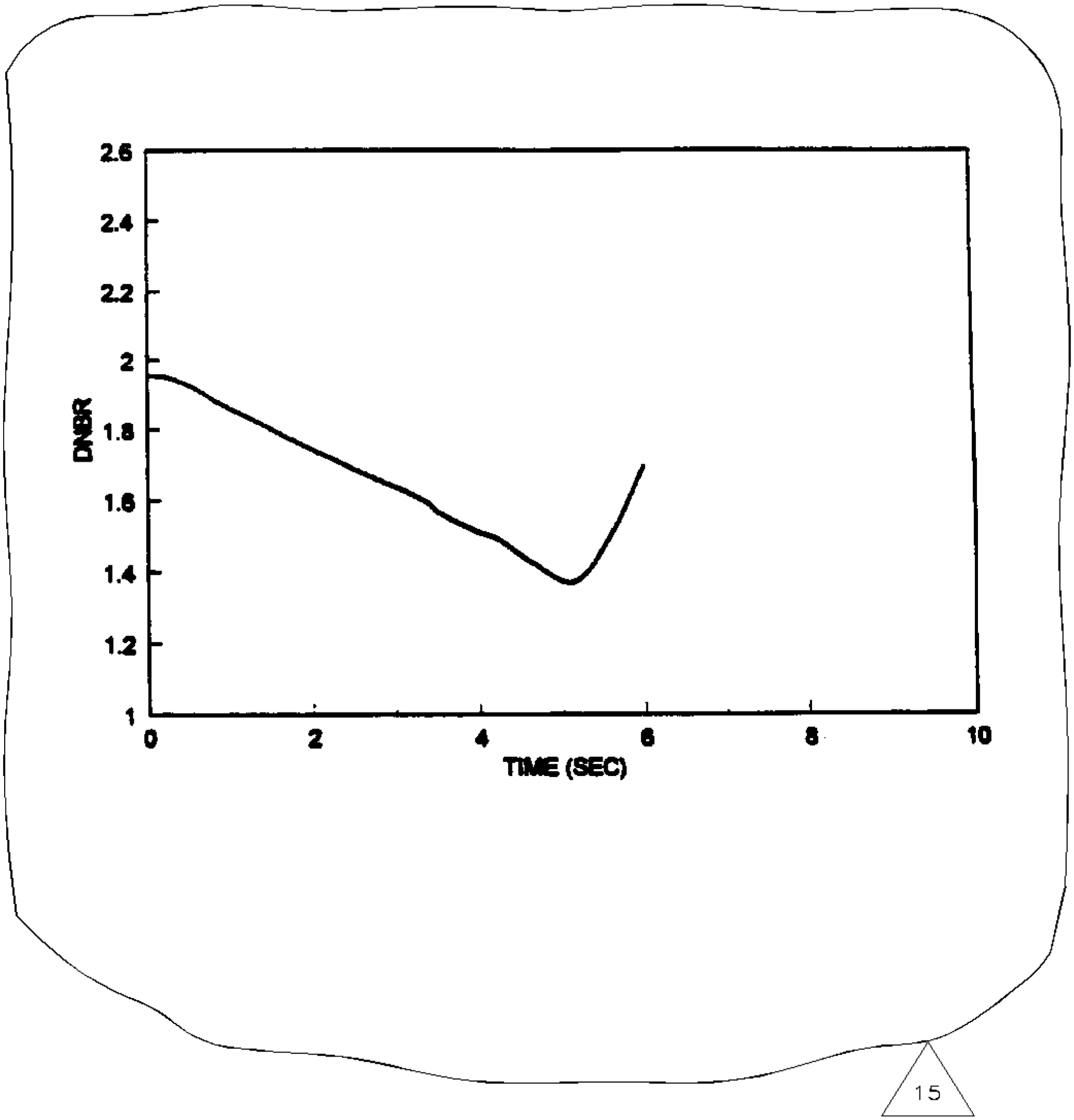
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING, FREQUENCY DECAY TO ALL LOOPS – HOT CHANNEL HEAT FLUX VERSUS TIME

FIGURE 15.3-24



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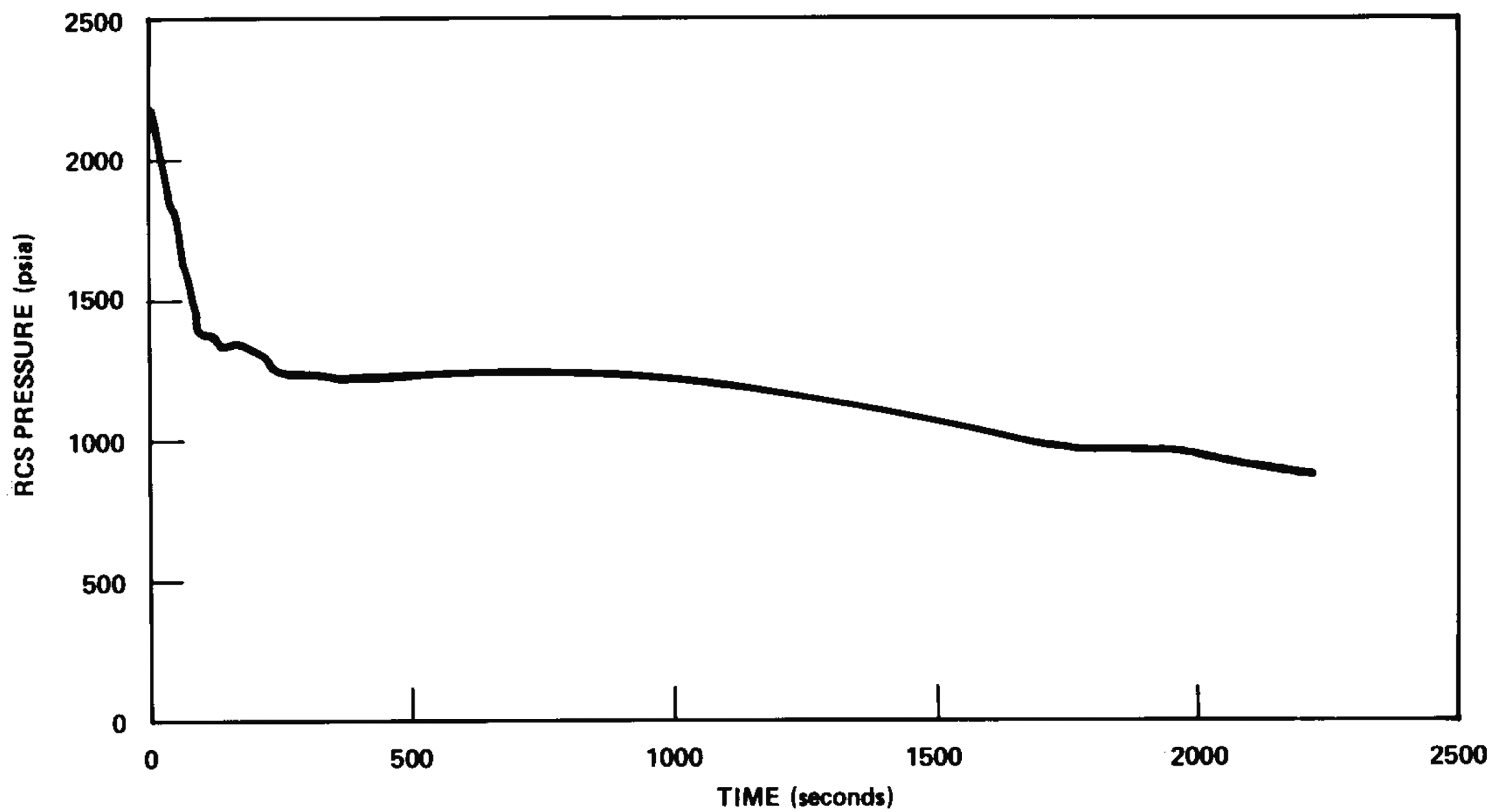
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING, FREQUENCY DECAY
TO ALL LOOPS –
DNBR VERSUS TIME

FIGURE 15.3-25



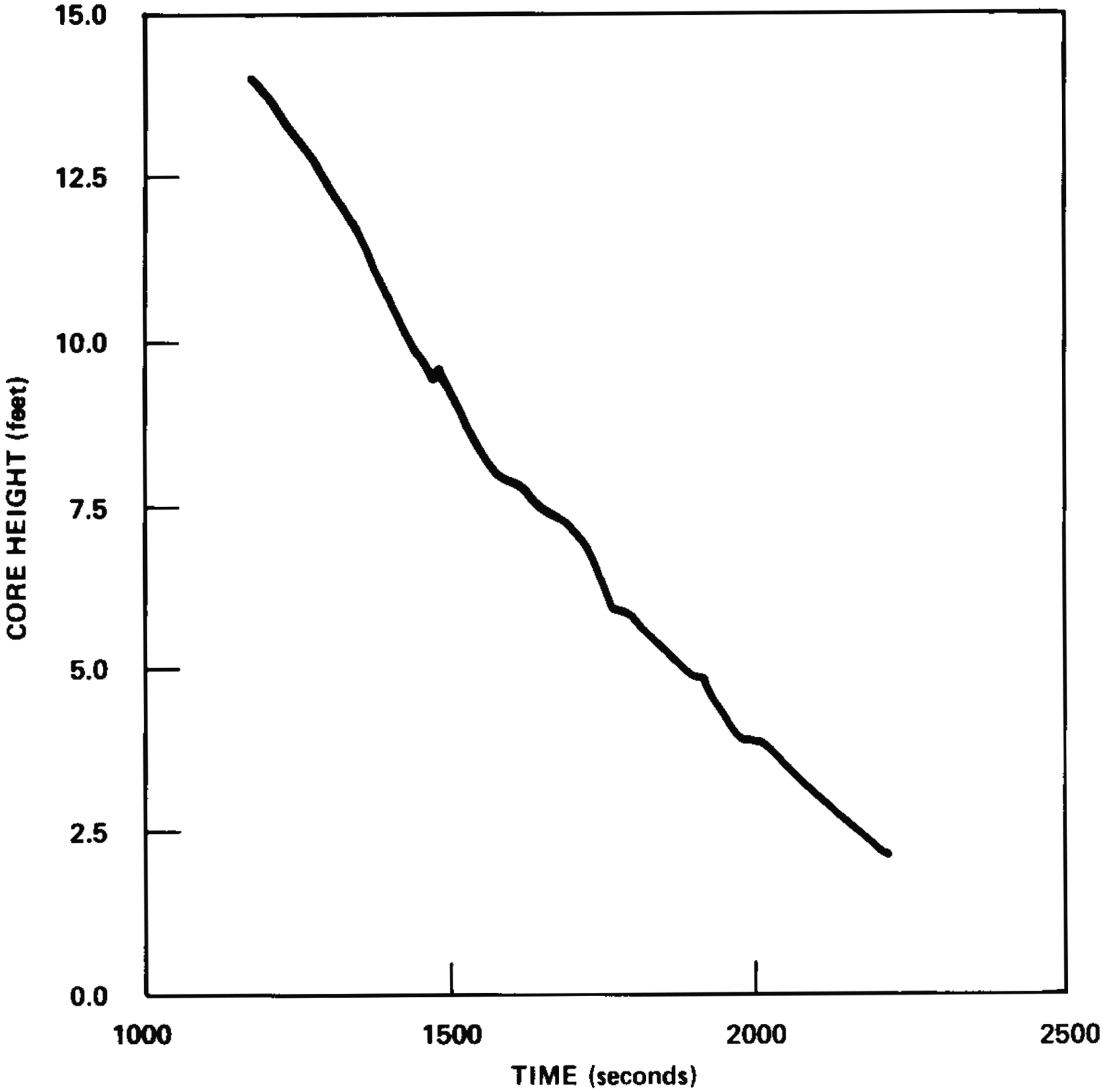
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

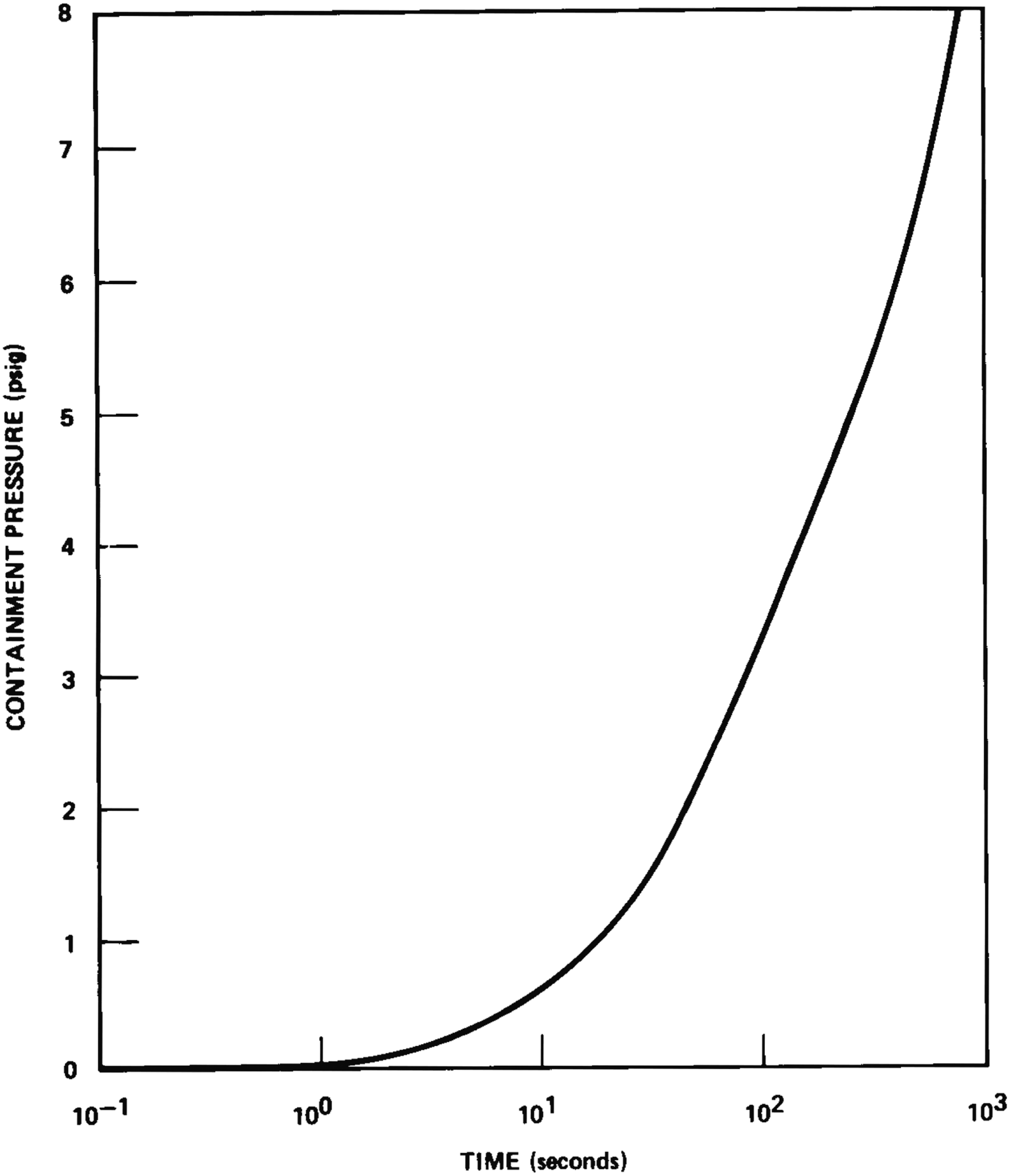
2-IN. BREAK,
NO SI, RCS PRESSURE

FIGURE 15.3-26



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	2-IN. BREAK, NO SI, CORE MIXTURE HEIGHT
		FIGURE 15.3-27



REV 21 5/08

SOUTHERN COMPANY
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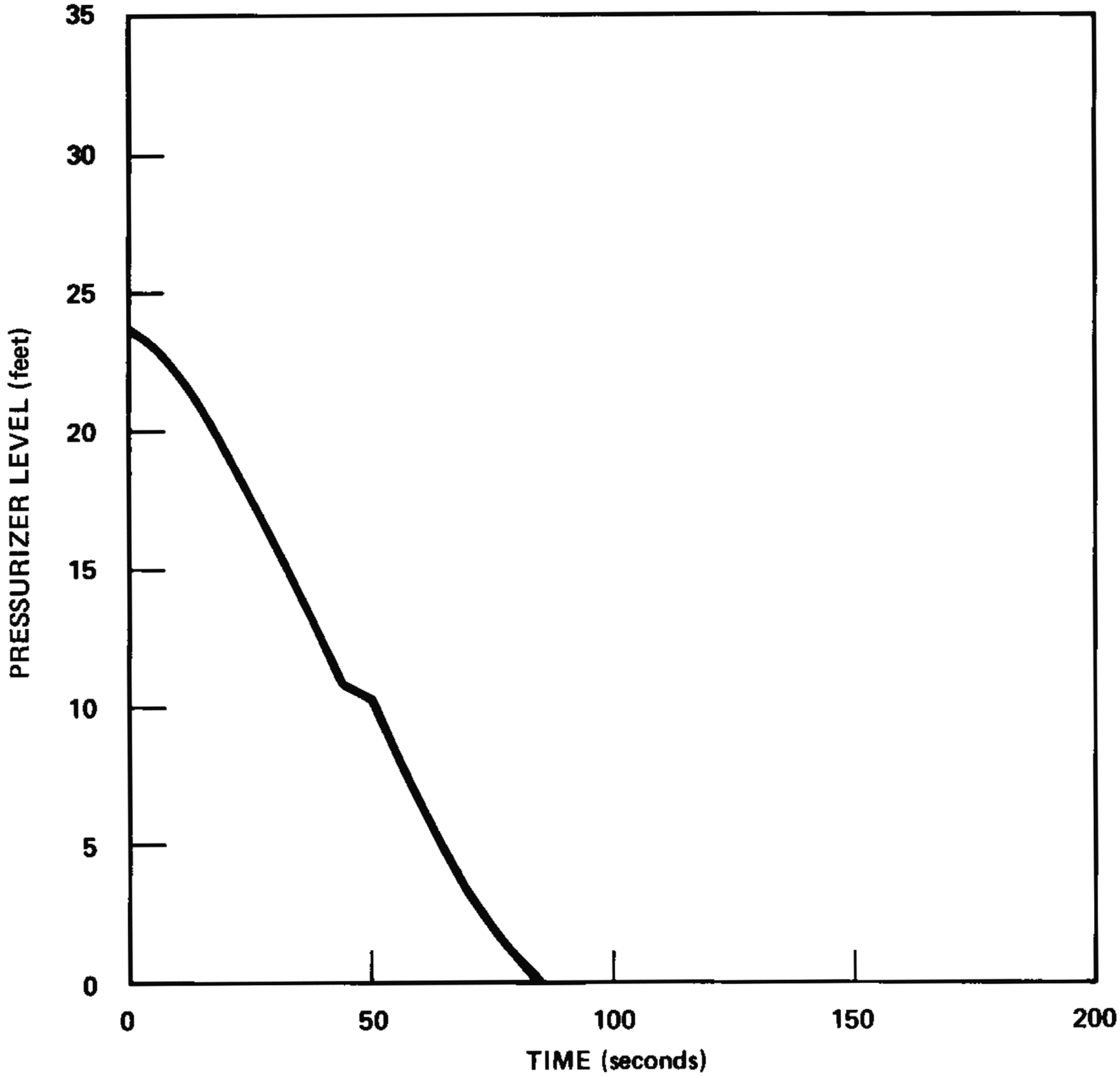
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SMALL BREAK

FIGURE 15.3-28



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

2-IN. BREAK, NO SI,
PRESSURIZER LEVEL (FT)

FIGURE 15.3-29

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic occurrences which must be designed against and thus represent limiting design cases. The Nuclear Regulatory Commission (NRC) acceptance criterion for Condition IV faults is that doses outside of the plant exclusion boundary will be less than the limits established in 10 CFR 50.67 in order to ensure that there will be no undue risk to the health and safety of the public. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system (ECCS) and the containment as described in sections 6.3 and 6.2 respectively.

For the purposes of this report, the following faults have been classified in this category:

A. Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (RCS) [loss-of-coolant accident (LOCA)].

B. Major secondary system pipe rupture up to and including double-ended rupture (rupture of a steam pipe).

C. Steam generator tube rupture.

D. Single reactor coolant pump locked rotor.

E. Fuel handling accident (FHA).

F. Rupture of a control rod mechanism housing (rod cluster control assembly (RCCA) ejection).

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENTS)

15.4.1.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area ≥ 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

For large break LOCAs, the most limiting single failure is the one which produces the lowest containment pressure. The lowest containment pressure would be obtained only if all containment spray pumps and fan coolers operated subsequent to the postulated LOCA. Therefore, for the purposes of large break LOCA analyses, the most limiting single failure would only be the loss of one residual heat removal (RHR) pump with full operation of the spray pumps and fan coolers (with the lowest containment pressure). However, the large break LOCA analyses conservatively assume both maximum containment safeguards (lowest containment pressure) and minimum ECCS safeguards (the loss of one complete train of ECCS components which includes one RHR pump and one high-head safety injection (SI) pump), which results in the minimum delivered ECCS flow available to the RCS. Minimum ECCS flow has been shown to be a conservative assumption for best-estimate large break LOCA (reference 7). The NRC acceptance criteria for the LOCA are described in 10 CFR 50.46 (reference 1) as follows:

A. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

B. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for an extended period of time required by the long-lived radioactivity.

These criteria were established to provide significant margin in ECCS performance following a LOCA. WASH-1400 (reference 2) presents a study in regards to the probability of occurrences of RCS pipe ruptures.

15.4.1.2 <u>Sequence of Events and Systems Operations</u>

Should a major break occur, depressurization of the RCS would result in a pressure decrease in the pressurizer. The reactor trip signal would subsequently occur when the pressurizer low-pressure trip setpoint is reached. An SI signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. An average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the present Westinghouse design, the most limiting large break single failure is the loss of one high-head pump and one low-head pump. This assumption is consistent with the current procedure for large break analyses.

For the large break analysis, one ECCS train, including one high-head SI pump and one RHR (low-head) pump, starts and delivers flow through the injection lines (one for each loop) with one branch injection line spilling to the containment backpressure. However, both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment fan coolers and spray pumps. Modeling full containment heat removal systems operation is required by 10 CFR 50 Appendix K and Branch Technical Position CSB 6-1 and is conservative for the large break LOCA.

To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, both the high-head SI pump and the RHR pump performance curves were degraded by 10% and a 20 gal/min (-5, +15) flow imbalance was assumed for the high-head SI pumps.

In the large break ECCS analysis presented here, single failure is conservatively accounted for via the loss of an ECCS train, the spilling of the minimum resistance injection line, and by assuming all containment spray pumps and fan coolers are available. Therefore, the analysis assumed one high-head pump, one RHR pump, two containment spray pumps, and four fan coolers are operating.

15.4.1.3 Description of Large Break LOCA Transient

The RCS is assumed to be operating normally at full power. Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed by the secondary system. A large cold leg break is assumed to open nearly instantaneously in one of the main coolant pipes. Calculations where the location and size of the break have been varied indicate that a break in the cold leg between the pump and the vessel leads to the most severe transient. For the break location, a rapid depressurization occurs, along with a core flow reversal as subcooled liquid flows out of the vessel into the broken cold leg. Boiling begins in the core, and the reactor core begins to shut down. Within approximately 2 s, the core is highly voided, and core fission is terminated. The cladding temperature rises rapidly as heat transfer from the fuel rods is reduced.

Within approximately 5 s, the pressure in the pressurizer has fallen to the point where reactor trip and SI signals are initiated. It is highly likely that these signals will have been initiated sooner as a result of a high containment pressure signal. Along with the SI signal, the containment isolation signal is also initiated.

In the first 5 s, the coolant in all regions of the vessel begins to flash. In addition, the break flow becomes saturated and is substantially reduced. This reduces the depressurization rate, and may also lead to a period of positive core flow or reduced downflow as the reactor coolant pumps in the intact loops continue to supply water to the vessel, and as flashing continues in the vessel lower plenum and downcomer. Cladding temperatures may be reduced, and some portions of the core may rewet during this period.

The positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg. Core cooling occurs as a result of the reverse flow.

At approximately 10 s after the break, the pressure falls to the point where accumulators begin injecting cold water into the cold legs. Because the break flow is still high, much of the injected ECCS water, which flows into the downcomer of the vessel, is bypassed out to the break.

Approximately 25 s after the break, most of the original RCS inventory has been ejected or boiled off. The system pressure and break flow are reduced and the ECCS water, which has been filling the downcomer, begins to fill the lower plenum of the vessel. Additional ECCS water pumped from the RWST begins to flow into the vessel. During this time, core heat transfer is relatively poor and cladding temperatures increase.

The blowndown phase of the transient ends when the RCS pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowndown, termination of bypass occurs and refill of the reactor vessel lower plenum begins. Refill is completed when ECCS water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core (BOC) recovery time).

Approximately 35 s after the break, the lower plenum has re-filled, and ECCS water enters the core. The flow into the core is oscillatory, as cold water rewets hot fuel cladding, generating steam. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generator, and the reactor coolant pump before it can be vented out the break. The resistance of this flow path of the steam flow is balanced by the driving force of water filling the downcomer. Shortly after reflood begins, the accumulators exhaust their inventory of water, and begin to inject the nitrogen gas which was used to pressurize the accumulators. This results in a short period of improved heat transfer as the nitrogen forces water from the downcomer into the core. When the accumulators have exhausted their supply of nitrogen, the reflood rate may be reduced and peak cladding temperatures may again rise. This heatup may continue until the core has reflooded to several feet. Approximately 3 min after the break, all locations in the core begin to cool. The core is completely quenched within 10 min, and long-term cooling and decay heat removal begin. Long-term cooling for the next several minutes is characterized by continued boiling in the vessel as decay power and residual heat in the reactor structures are removed.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures would be reduced to long-term steady-state levels associated with the dissipation of residual heat generation. After the water level of the RWST reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation mode of operation in which spilled borated water is drawn from the containment sump by the RHR pumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment emergency sumps and continue to operate to further reduce containment pressure and temperature.

At 7.5 h after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs and cold legs in order to control the boric acid concentration in the reactor vessel. Long-term cooling includes long-term criticality control. To achieve long-term criticality control, a mixed-mean sump boron concentration is determined and verified against core design margins to assure core subcriticality, without credit for RCCA insertion. A mixed-mean sump boron concentration is calculated based on minimum volumes for boron sources and maximum volumes for dilution sources. The calculated mixed-mean sump boron concentration is verified against available core design margins on a cycle-specific basis. The current Technical Specifications range is 2300 to 2500 ppm boron for the RWST and 2200 to 2500 ppm for the accumulators.

The sequence of events described above is summarized in figure 15.4-1.

15.4.1.4 Analysis of Effects and Consequences

15.4.1.4.1 Method of Analysis

When the final acceptance criteria (FAC) governing the LOCA for light water reactors was issued in Appendix K of 10 CFR 50.46 (reference 1), both the NRC and the industry recognized that the rule was highly conservative. That is, using the then accepted analysis methods, the performance of the ECCS would be conservatively underestimated, resulting in predicted peak clad temperatures (PCTs) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1. Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could not be quantified with additional data so that a less prescriptive future approach might be allowed.

2. Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analysis and confirmed that some relaxation of the rule can be made without a loss in safety to the public. It was also found that some plants were being restricted in operating flexibility by overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in SECY-83-472 (reference 55). The SECY-83-472 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, SECY-83-472 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1988, as a result of the improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA (reference 3). Under the amended rules, best estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the analysis, an assessment of the uncertainty of the best estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance limits. Further guidance for the use of best estimate codes was provided in Regulatory Guide 1.157 (reference 4).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (reference 5). This method outlined an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was recently approved by the NRC (reference 6). The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (reference 7).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical TReatment of Uncertainly Method (reference 56). This method is still based on the CQD methodology and follows the steps in the CSAU methodology. However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations (reference 57).

The three 10 CFR 50.46 criteria (PCT, maximum local oxidation, and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95 percentile of three parameters with a 95% confidence level.

15.4.1.4.2 Best-Estimate Large Break LOCA Evaluation Model

The thermal-hydraulic computer code which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC Version MOD7A Revision 1 (WCAP-12945-P-A, reference 7). Since its approval, the code has been upgraded to Revision 6. WCOBRA/TRAC MOD7A Revision 6 is an evolution of Revision 1. The differences between these frozen versions include logic to facilitate the automation aspects of ASTRUM, user conveniences, and error corrections. WCOBRA/TRAC MOD7A Revision 6 is documented in reference 56.

WCOBRA/TRAC combines two-fluid, three-field, multidimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features.

- Ability to model transient three-dimensional flows in different geometries inside the vessel;

- Ability to model thermal and mechanical nonequilibrium between phases;

- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes;

- Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ. Dividing the liquid phase into two fields is a convenient and physically accurate way of handling flows where the liquid can appear in both film and droplet form. The droplet field permits more accurate modeling of thermal-hydraulic phenomena such as entrainment, de-entrainment, fallback, liquid pooling, and flooding.

WCOBRA/TRAC also features a two-phase, one-dimensional hydrodynamics formulation. In this model, the effect of phase slip is modeled indirectly via a constitutive relationship which provides the phase relative velocity as a function of fluid conditions. Separate mass and energy conservation equations exist for the two-phase mixture and for the vapor.

The reactor vessel is modeled with the three-dimensional, three-field model, while the loop, major loop components, and SI points are modeled with the one-dimensional model.

All geometries modeled using the three-dimensional model are represented as a matrix of cells. The number of mesh cells used depends on the degree of detail required to resolve the flow field, the phenomena being modeled, and practical restrictions such as computing costs and core storage limitations.

The equations for the flow field in the three-dimensional model are solved using a staggered difference scheme on the Eulerian mesh. The velocities are obtained at a mesh cell faces, and the state variables (e.g., pressure, density, enthalpy, and phasic volume fractions) are obtained at the cell center. This cell is the control volume for the scalar continuity and energy equations. The momentum equations are solved on a staggered mesh with the momentum cell centered on the scalar cell face.

The basic building block for the mesh is the channel, a vertical stack of single-mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

One-dimensional components are connected to the vessel. The basic scheme used also employs the staggered-mesh cell. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state, initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood follows continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component of the loop model using a time-dependent pressure table. Containment pressure is calculated using the COCO code (references 9 and 11) and mass and energy releases from the WCOBRA/TRAC calculation.

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in references 6, 7, 56, and 58. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis (reference 58).

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, local maximum oxidation (LMO), and core-wide oxidation (CWO) at 95% probability, is described in the following sections.

1. Plant Model Development

   In this step, a WCOBRA/TRAC model of the plant is developed. A high level of noding detail is used in order to provide an accurate simulation of the transient. However, specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among other plant models, except for specific areas dictated by hardware differences such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions

   In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a “key LOCA parameters” list which was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the “initial transient.” Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

3. Assessment of Uncertainty

   The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of reference 56. The determination of the PCT uncertainty, LMO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, LMO, and CWO).

   The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which is randomly sampled for each WCOBRA/TRAC calculation, includes initial conditions, power distributions, and model uncertainties. The time in the cycle, break type (slip or double-ended guillotine), and break size for the split break are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for LMO and CWO. The highest rank of PCT, LMO, and CWO will bound 95% of their respective populations with 95% confidence level.

4. Plant Operating Range

The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range or may be narrower for some parameters to gain additional margin.

15.4.1.4.3 Analytical Input Assumption Differences Between Units 1 and 2

The WCOBRA/TRAC models for Farley Units 1 and 2 were originally developed for the power uprate (reference 10). Two models were utilized in the original Analysis of Record (AOR), mainly because Unit 1 had an upflow barrel/baffle (B/B), whereas Unit 2 had a downflow B/B. A parametric study was performed at that time to determine the limiting unit. Unit 2 was determined to be the limiting unit at that time. Therefore, the Unit 2 model was utilized for the subsequent steps of the original application of the best-estimate large break LOCA evaluation model.

Subsequent to the original analysis, replacement steam generators (RSGs) have been implemented, and Unit 2 has been converted to an upflow B/F configuration. These changes were incorporated into the ASTRUM analysis. Moreover, investigations revealed that the remaining differences in the vessels were small enough to justify the use of a single WCOBRA/TRAC geometric model for both Units 1 and 2. Consequently, there are no WCOBRA/TRAC model differences between the two units at this time.

15.4.1.4.4 Farley Units 1/2 Model Results

A series of WCOBRA/TRAC calculations was performed, using the Farley Unit 1/2 plant input model, to determine the effect of variations in several key LOCA parameters on the PCT. From these studies, an assessment was made of the parameters which had a significant effect, as described in the following sections. These parameters, once established, become the bounding conditions of the reference transient. The peak clad temperature (PCT) curve of the reference transient is presented in figure 15.4-2.

15.4.1.4.4.1 Units 1/2 Reference Transient Description. The Units 1/2 initial transient is a double-ended cold leg guillotine break which used the conditions listed in table 15.4-1. Since many of these parameters are at their bounded values, the calculated results of the reference transient of table 15.4-2 are a conservative representation of the response to a large break LOCA.

The LOCA transient can be divided into time periods in which specific phenomena are occurring. A convenient way to divide the transient is in terms of the various heatup and cooldown transients that the hot assembly undergoes. For each of these phases, specific phenomena and heat transfer regimes are important, as discussed below. Results of the initial transient are shown on figures 15.4-3 to 15.4-14. In these figures, the transient starts at 0 s.

Critical Heat Flux (CHF) Phase

Immediately following the cold leg rupture, the break flowrates are subcooled and high. The regions of the RCS with the hottest initial temperatures (core, upper plenum, upper head, and hot legs) begin to flash to steam within the first 0.5 s following the break. Flow in the core reverses, and the fuel rods begin to go through departure from nucleate boiling (DNB). Voiding in the core also causes the fission power to drop rapidly. The discharge flowrates decrease sharply as the break flows become two-phase (figures 15.4-3 and 15.4-4). This phase is terminated when the water in the lower plenum and downcomer (DC) begin to flash.

Upward Core Flow Phase

For three-loop plants, double-ended cold leg guillotine (DECLG) breaks exert a strong downflow pull on core flow, such that for the larger breaks, there is no evidence of an upward core flow phase. Flashing in the lower plenum and pumped flow supplied by the intact loops reduces the magnitude of the core downflow. The degradation of the pump head due to voiding and the large outflow at the vessel-side broken cold leg increases the magnitude of the downward core flow. This phase ends as the lower plenum mass is depleted, the loops become two-phase, and the pump head degrades.

Downward Core Flow Phase

Downward flow into the core increases as the pump head continues to be degraded and upward flow in the DC is firmly established (figure 15.4.7).

Due to the downflow during this phase, the cladding temperature was turned around at about 6 s after the initiation of the transient. As the system pressure continues to fall (figure 15.4-8), the break flow and, consequently, the core flow are reduced. The vessel pressure reaches the containment pressure at the end of this phase, which occurs about 22 s after the initiation of the transient. The core begins to heat up as the system reaches containment pressure, and the vessel begins to fill with ECCS water.

Refill Phase

The refill period is characterized by a rapid increase in the lower plenum liquid level and the vessel fluid mass (figures 15.4-9 and 15.4-10). In this period, the cladding temperature at all elevations increases rapidly due to the lack of liquid and steam flow in the core region and resulting poor cooling (figure 15.4-2). This phase ends when the lower plenum fills with water (figure 15.4-9) and the ECCS water enters the core (bottom of core recovery, BOC). This initiates the reflood phase, where entrainment begins, with a resulting improvement in heat transfer.

Reflood Phase

At the beginning of this phase, the accumulators empty around 30 s after the transient begins (figure 15.4-11) and nitrogen enters the system, which causes a surge of water into the core (figure 15.4-13) and a temporary cooldown (figure 15.4-2). The early part of this period is characterized by a significant vapor generation as the lower elevations of the core quench. This temporarily increases the core pressure, reversing the core inlet flow. As the steam generated in the core is vented through the loops and the DC level rises further, the DC pressure increases above the core pressure and positive core flow is reestablished. The resulting core/DC level oscillations can be seen in the core and DC liquid level plots (figures 15.4-13 and 15.4-14). At approximately 130 s after the transient begins, ECCS water accumulated in the lower plenum starts to boil, causing a reduction in the core and DC liquid levels and the vessel mass (figure 15.4-10), as the two-phase level swell pushes water out the break (figure 15.4-3).

15.4.1.4.4.2 Units 1/2 Confirmatory Studies. A few sensitivity studies were performed to establish the limiting conditions for the uncertainty evaluation. In the sensitivity studies performed, key LOCA parameters are varied over a range and the impact on the peak clad temperature is assessed.

The results for the sensitivity studies are summarized in table 15.4-2. A full report on the results is included in Section 4 of reference 58. In summary, the limiting conditions for the plant at the time the design basis accident is postulated to occur are reflected in the final reference transient. They are as follows:

- Loss of offsite power.
- High RCS average temperature.
- High steam generator tube plugging (SGTP) of 10%.
- Low average power fraction in the assemblies on the core periphery (fraction of power in outer assemblies (PLOW) = 0.2).

15.4.1.5 Uncertainty Evaluation and Results

15.4.1.5.1 Uncertainty Evaluation

The ASTRUM methodology (reference 56) differs from the previously approved Westinghouse best-estimate methodology (reference 7), primarily in the statistical technique used to make a singular probabilistic statement with regard to the conformance of the system under analysis to the regulatory requirement of 10 CFR 50.46.

The ASTRUM methodology applies a nonparametric statistical technique to generate output (e.g., PCT, LMO, and CWO from a combination of WCOBRA/TRAC and HOTSPOT (reference

56) calculations. These calculations are performed by applying a direct, random Monte Carlo sampling to generate the input for the WCOBRA/TRAC and HOTSPOT computer codes.

This approach allows the formulation of a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46. Based on the nonparametric statistical approach, the number of Monte Carlo runs is only a function of the tolerance interval and associated confidence level required to meet the desired level of safety.

The singular statement of uncertainty chosen in the ASTRUM methodology is based on a 95% tolerance interval with a 95% confidence level for each of the 10 CFR 50.45 criteria, (b)(1), (2), and (3), i.e., PCT, LMO, and CWO, respectively. This requires 124 large break LOCA calculations (reference 57).

The uncertainty attributes have been divided into the following categories; initial conditions uncertainty, power distribution uncertainty, global model uncertainty, and local model uncertainty. Each category is discussed in greater detail in Section 5 of reference 59. The results for Farley Units 1 and 2 are given in table 15.4.3.

15.4.1.5.3 Additional Evaluations

COCO Evaluation

The ASTRUM methodology (reference 56, Section 11-3-1) designates that the containment pressure utilized in the analysis will be conservatively low and based on the mass and energy releases from the reference transient. The mass and energy releases from the updated reference transient were utilized in the execution of a COCO minimum pressure study that demonstrated that the containment backpressure inputs used in the updated reference transient were conservative. The updated reference transient extends to 280 s. The same containment backpressure inputs to 280 s were used in the confirmatory suite and the 124-case ASTRUM runset. However, the ASTRUM runset extends to 500 transient seconds. During the period from 280 to 500 s, the as-executed containment pressure remained the same as the last value to that point. It was subsequently determined that the as-executed runset was nonconservative in the time period from 340 to 500 s. The majority of the transients, however, had already quenched by 340 s and would not be expected to be impacted. The WCOBRA/TRAC PCT for all 124 ASTRUM transients occurs prior to 340 s and is not anticipated to be impacted. The oxidation is anticipated to be marginally impacted, since during the timeframe of the discrepancy, the PCT is significantly reduced. For completeness, a case was reexecuted to 500 s and incorporated an updated backpressure curve (the updated values were extrapolated). The HOTSPOT PCT, HOTSPOT LMO, and WCOBRA/TRAC hot assembly rod 2 total oxidation were completely unchanged. Hence, the overall analysis results are deemed to remain valid.

RHR Miniflow

WCOBRA/TRAC does not currently have enough flexibility to precisely model the timing and delivery characteristics of the plant RHR miniflow open and closed configuration. Westinghouse's approach is to sanction a bounding RHR delivery approach that would only credit RHR delivery flow once the miniflow valve is fully closed.

Quarterly RHR Test Configuration Evaluation

RHR pump test procedures are performed regularly for the Farley units. This test configuration includes degraded RHR injection, but credits two charging SI pumps, whereas the SI performance in the ASTRUM analysis credits only one charging/SI pump. Both SI performance configurations utilize broken loop spill to 10 psig, and the test configuration yields approximately 6% SI reduction in the low pressure range of interest to LBLOCA analyses. The ASTRUM analysis program consideration of the test configuration is not included in the ASTRUM analysis proper, but is evaluated subsequently by analyzing a larger SI reduction during the early and middle Reflood portion of the transient. The PCT results of this study led to a net PCT penalty of 25°F. See table 15.4-3 for the application of this penalty.

Reactor Coolant Pump Inputs Error

Several errors were discovered in the pump two-phase degraded homologous curve. In addition, minor errors were also found in the pump inputs resulting in a slight loop-to-loop asymmetry. The corrected pump inputs have been evaluated. The result from the plant-specific WCOBRA/TRAC runs yielded an estimated PCT increase of 18°F. See table 15.4-3 for the application of this penalty.

Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown

Fuel pellet thermal conductivity degradation (TCD) and peaking factor burndown were not explicitly considered in the best estimate large break loss-of-coolant accident (BE LBLOCA) AOR. A quantitative evaluation was performed to assess the PCT effect of fuel pellet TCD and peaking factor burndown on the BE LBLOCA analysis and concluded that the estimated PCT impact is 150°F. See table 15.4-3 for the application of this penalty.

Revised Heat Transfer Multiplier Distributions

Several changes and error corrections were made to WCOBRA/TRAC and the impacts of these changes on the heat transfer multiplier uncertainty distributions were investigated. During this investigation, errors were discovered in the development of the original multiplier distributions, including errors in the grid locations specified in the WCOBRA/TRAC models for the G2 Refill and G2 Reflood tests and errors in processing test data used to develop the reflood heat transfer multiplier distribution. Therefore, the blowdown heatup, blowdown cooling, refill, and reflood heat transfer multiplier distributions were redeveloped. For the reflood heat transfer multiplier development, the evaluation time windows for each set of test experimental data and each test simulation were separately defined based on the time at which the test or simulation exhibited dispersed flow film boiling heat transfer conditions characteristic of the reflood time period. The revised heat transfer multiplier distributions have been evaluated for impact on existing analyses. A plant transient calculation representative of Farley Units 1 and 2 transient behavior was performed with the latest version of WCOBRA/TRAC. Using this transient, a matrix of HOTSPOT calculations was performed to estimate the effect of the heat transfer multiplier distribution changes. Using these results and considering the heat transfer multiplier uncertainty attributes from limiting cases for Farley Units 1 and 2 resulted in an estimated PCT effect of -40°F. See table 15.4-3 for the application of this penalty.

Changes to Grid Blockage Ratio and Porosity

A change in the methodology used to calculate grid blockage ratio and porosity for Westinghouse fuel resulted in a change to the grid inputs used in the LBLOCA analysis. Grid inputs affect heat transfer in the core during a LBLOCA. The estimated penalty associated with the changes is 24°F. See table 15.4-3 for the application of this penalty.

Error in Burst Strain Application

An error in the application of the burst strain was discovered in HOTSPOT. Correction of the erroneous calculation results in thinner cladding at the burst node and more fuel relocating into the burst node, leading to an increase in PCT at the burst node. The estimated penalty associated with this error is 21°F. See table 15.4-3 for the application of this penalty.

15.4.1.5.4 Deleted

15.4.1.5.5 10 CFR 50.46 Requirements

It must be demonstrated that there is a high probability that the limits set forth by 10 CFR 50.46 (reference 4) will not be exceeded. The demonstration that these limits are met for Farley Units 1 and 2 are as follows:

1. The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95% confidence level. Since the resulting PCT (including changes/errors discovered subsequent to the AOR model development) for the limiting case is 2013°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., “Peak Clad Temperature less than 2200°F,” is demonstrated. The results are shown in table 15.4-3.

2. The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95% confidence level. Since the resulting LMO for the limiting case is 2.9%, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., “Local Maximum Oxidation of the cladding less that 70 percent,” is demonstrated. The results are shown in table 15.4.3.

3. The limiting core-wide oxidation corresponds to a bounding estimate of the 95th percentile CWO at the 95% confidence level. The limiting hot assembly rod (HAR) total maximum oxidation is 0.22%. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because

there is significant margin to the regulatory limit, the CWO calculation is, therefore, not needed because the outcome will always be < 0.22%. Since the resulting CWO is 0.22%, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., “Core-Wide Oxidation less than 1 percent,” is demonstrated. The results are shown in table 15.4-3.

4. 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2) and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for best-estimate LOCA applications. The approved methodology (reference 7) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 28 assemblies in the low-power channel. This situation has not been calculated to occur in Farley Units 1 and 2. Therefore, acceptance criterion (b)(4) is satisfied.

5. 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. The approved Westinghouse position on this criterion is that this requirement is satisfied if a coolable core geometry is maintained, and the core remains subcritical following the LOCA (reference 59). This position is unaffected by the use of best-estimate LOCA methodology.

15.4.1.5.6 Plant Operating Range

The expected PCT and its uncertainty developed previously are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 15.4-4 summarizes the operating ranges for Farley Units 1/2 as defined for the proposed operating conditions, which are supported by the best-estimate LBLOCA analysis. Table 15.4-6 summarizes the LBLOCA containment data used for calculating containment pressure. It should be noted that other non-LBLOCA analyses may not support these ranges. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T_{avg}).

Note that the LBLOCA analysis was performed with ZIRLO cladding. However, reference 61 concluded that the LOCA ZIRLO models are acceptable for application to Optimized ZIRLO cladding in large break analyses. No additional calculations are necessary for evaluating the use of Optimized ZIRLO cladding provided plant specific ZIRLO calculations were previously performed.

15.4.1.6 Hydrogen Production and Accumulation

Hydrogen accumulation in the containment atmosphere following the design basis accident (DBA) can be the result of production from several sources. The potential sources of hydrogen are the zirconium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, includes both core solution radiolysis and sump solution radiolysis.

15.4.1.6.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the ECCS. The criteria for evaluation of the ECCS require that the zircaloy-water reaction be limited to 1% by weight of the total quantity of zirconium in the core. ECCS calculations have shown the zircaloy-water reaction to be < 0.6%, much less than required by the criteria.

The use of aluminum and zinc inside the containment is limited, and is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum and zinc are more reactive with the containment spray alkaline borate solution than with other plant materials such as carbon and stainless steel, copper, and copper-nickel alloys. By limiting the use of aluminum and zinc, the aggregate source of hydrogen over the long term is essentially restricted to that arising from radiolytic decomposition of core and sump water. The upper limit rate of such decomposition can be predicted with ample certainty to permit the design of effective countermeasures.

During the recirculation phase, trisodium phosphate dissolved in the sump maintains a pH of 7.0 to 10.5; thus, hydrogen production due to aluminum and zinc corrosion is minimized.

It should be noted that the zirconium-water reaction and aluminum and zinc corrosion with containment spray are chemical reactions and, thus, essentially independent of the radiation field inside the containment following a LOCA. Radiolytic decomposition of water is dependent on the radiation field intensity.

The radiation field inside the containment is calculated for the maximum credible accident in which the fission product activities given in TID-14844[19] are used.

The hydrogen generation calculation is performed; one using the Westinghouse model discussed below, the other using the NRC model discussed in Regulatory Guide 1.7.[26] As described in subsection 6.2.5, the resultant hydrogen concentrations from the NRC release model were used in the development of the design criteria for the containment combustible gas control systems.

15.4.1.6.2 Assumptions

The following discussion outlines the assumptions used in the calculations:

A. Zirconium-Water Reaction

The zirconium-water reaction is described by the chemical equation:

$$Zr + 2\,H_2O \rightarrow ZrO_2 + 2\,H_2 + \text{Heat}$$

The hydrogen generation due to this reaction will be completed during the first day following the LOCA. The Westinghouse model assumes a 2% zirconium-water reaction and the NRC model assumes a 5% zirconium-water reaction. The hydrogen generated is assumed to be released immediately into the containment atmosphere.

B. Corrosion of Plant Materials

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in the containment in the emergency core cooling solution at DBA conditions. Metals tested include zircaloy, inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel, and copper. Tests conducted at Oak Ridge National Laboratory (ORNL)[21][22] have also verified the compatibility of the various materials (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates, the results of the corrosion tests have shown that only aluminum and zinc corrode at a rate that may significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2Al + 3\,H_2O \rightarrow Al_2O_3 + 3\,H_2$$

Therefore, three moles of hydrogen are produced for every two moles of aluminum that are oxidized. (Approximately 20 sf^3 of hydrogen for each pound of aluminum corroded.)

Corrosion of zinc may be described by the overall reaction:

$$Zn + 2\,H_2O \rightarrow Zn(OH)_2 + H_2$$

One mole of hydrogen gas is produced for each mole of zinc that is oxidized. Approximately 6 sf^3 of hydrogen gas is produced for each pound of zinc corroded. The time temperature cycle (table 15.4-8) considered in the calculation of aluminum and zinc corrosion is based on a conservative step representation of the postulated post-accident containment transient. The corrosion rates at the various steps were determined from the aluminum corrosion rate design curve shown in figure 15.4-18. The corrosion rate for zinc at various temperatures is shown in table 15.4-7. These corrosion rates and the zinc inventory described in section 6A.2 were used in the hydrogen generation calculation. Aluminum and zinc corrosion data points include the effects of temperature, alloy, and spray solution conditions. Based on these corrosion rates and the aluminum inventory given in drawing A-508597 (Farley Unit 1) and A-508928 (Farley Unit 2), the contribution of aluminum corrosion to hydrogen accumulation in the containment following the DBA has been calculated. For conservative estimation, no credit was taken for protective shielding effects of insulation or enclosures from the spray, and complete and continuous immersion was assumed.

Drawing A-508597 (Farley Unit 1) and A-508928 (Farley Unit 2) depict the basis for maximum aluminum inventory in containment used in the calculation of post-LOCA hydrogen generation. Table 15.4-13, hydrogen release and generation analyses, is based on aluminum inventory described in drawings A-508597 (Farley Unit 1) and A-508928 (Farley Unit 2) and the zinc inventory as described in section 6A.2. The current aluminum inventory in the containments of Unit 1 and Unit 2 is documented and tracked by these drawings.

The above calculation based on Regulatory Guide 1.7 was performed by allowing an increased corrosion rate during the final step of the post-accident containment temperature transient (table 6A-1) corresponding to 200 mil/yr (15.7 mg/dm^2/h) for aluminum and 5 mils/yr (0.395 mg/dm^2/h) for zinc. The corrosion rates earlier in the accident sequence are the higher rates determined from figure 15.4-18.

C. Radiolysis of Core and Sump Water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$H_2O \leftrightarrow H_2 + \frac{1}{2}O_2$$

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following the DBA.

An extensive program has been conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the DBA. In the course of this investigation, it became apparent that two separate radiolytic environments exist in the containment at DBA. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In the other case, the decay of dissolved fission products that have escaped from the core results in the radiolysis of the sump solution. The results of these investigations are discussed in reference 22.

15.4.1.6.3 Core Solution Radiolysis

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission product in the fuel. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from knowledge of the fission products contained in the core and a detailed analysis of the dissipation of the decay energy between core materials and the solution. The results of Westinghouse studies show essentially all of the beta energy will be absorbed within the fuel and cladding; this represents approximately 50% of the total beta-gamma decay energy. This study further shows that, of the gamma energy, a maximum of 7.4% will be absorbed by the solution incore. Thus, an overall absorption factor of 3.7% of the total core decay energy ($\beta + \gamma$) is used to compute solution radiation dose rates and the time-integrated dose. Table 15.4-9 presents the total decay energy ($\beta + \gamma$) of a reactor core, which assumes a full-power operating time of 830 days prior to the accident. For the maximum credible accident case, the contained decay energy in the core accounts for the assumed TID-14844 release of 50% halogens and 1% other fission products. To be conservative, the noble gases have been assumed by the TID-14844 model to escape to the containment vapor space where little or no water radiolysis would result from decay of these nuclides.

The radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse and Oak Ridge National Laboratory (ORNL). The results of static capsule tests conducted by Westinghouse indicate that hydrogen yields much lower than the maximum of 0.44 molecules per 100 eV would be the case incore. With little gas space to which the hydrogen formed in solution can escape, the rapid back reactions of molecular radiolytic products in solution to reform water are sufficient to result in very low net hydrogen yields.

However, it is recognized that there are differences between the static capsule tests and the dynamic condition incore where the core cooling fluid is continuously flowing. Such flow is reasoned to disturb the steady-state conditions which are observed in static capsule tests, and while the occurrence of back reactions would still be significant, the overall net yield of hydrogen would be somewhat higher in the flowing system.

The study of radiolysis in dynamic systems was initiated by Westinghouse, which formed the basis for experimental work performed at ORNL. Both studies clearly illustrate the reduced yields in hydrogen from core radiolysis; i.e., reduced from the maximum yield of 0.44 molecules per 100 Ev. These results were recently published.(24)(25)

For the purposes of this analysis, the calculations of hydrogen yield from core radiolysis are performed with the very conservative value of 0.44 molecules per 100 Ev. That this value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works. The Westinghouse results from the dynamic studies show 0.44 to be a maximum at very high solution flowrates through the gamma radiation field. The referenced ORNL(23) work also confirms this value as a maximum at high flowrates. Allen(24) presents a very comprehensive review of work performed to confirm the primary hydrogen yield to be a maximum of 0.44 to 0.45 molecules per 100 Ev.

On the foregoing basis, the production rate and total hydrogen produced from core radiolysis as a function of time has been conservatively estimated for the maximum credible accident case.

Calculations based on Regulatory Guide 1.7 assume a hydrogen yield value of 0.5 molecules per 100 Ev, 10 percent of the gamma energy produced from fission products in the fuel rods is absorbed by the solution in the region of the core, and the noble gases escape to the containment vapor space.

15.4.1.6.4 Sump Solution Radiolysis

Another potential source of hydrogen assumed for the post-accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products. In this consideration, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis.

The energy deposited in solution is computed using the following basis:

A. For the maximum credible accident, a TID-14844 release model[19] is assumed where 50% of the total core halogens and 1% of all other fission products, excluding noble gases, are released from the core to the sump solution.

B. The quantity of fission product release is equal to that from a reactor operating at full power for 830 days prior to the accident.

C. The total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the solution.

Within the assessment of energy release by fission products in water, account is made of the decay of halogens and a separate accounting for the slower decay of the 1% other fission products. To arrive at the energy deposition rate and time-integrated energy deposited, the contribution from each individual fission product class was computed. The overall contribution from each of the two classes of fission products is shown in table 15.4-10.

The yield of hydrogen from sump solution radiolysis is most nearly represented by the static capsule tests performed by Westinghouse and ORNL with the alkaline sodium borate solution. The differences between these tests and the actual conditions for the sump solution, however, are important and render the capsule tests conservative in their predictions of radiolytic hydrogen yields.

In this assessment, the sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solution, as compared to the case with shallow-depth capsule tests. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The buildup of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield in the same manner as a reduction in gas-to-liquid volume ratio will reduce the yield. This is illustrated by the data presented in figure 15.4-19 for capsule tests with various gas-to-liquid volume ratios. The data show a significant reduction in the apparent or net hydrogen yield from the published primary maximum yield of 0.44 molecules per 100 Ev. Even at the very highest ratios, where capsule solution depths are very low, the yield is < 0.30 with the highest scatter data point at 0.39 molecules per 100 Ev.

With these considerations taken into account, a reduced hydrogen yield is a reasonable assumption to make for the case of sump radiolysis. While it can be expected that the yield will be on the order of 0.1 or less, a conservative value of 0.30 molecules per 100 Ev has been used in the maximum credible accident case.
Calculations based on Regulatory Guide 1.7 do not take credit for a reduced hydrogen yield in the case of sump radiolysis and a hydrogen yield value of 0.5 molecules per 100 Ev has been used.

15.4.1.6.5 Results

Table 15.4-13 shows the results of the calculations for hydrogen production and accumulation from the following sources for the Reg. Guide 1.7 model:

A. Zirconium-water reaction.

B. Aluminum and zinc corrosion.

C. Radiolytic decomposition of core and sump solution.

Table 15.4-13 shows the total hydrogen production rate as a function of time following a LOCA for core and sump radiolytic decomposition and the total quantity of hydrogen accumulated in the containment due to all sources as a function of time for the maximum credible accident case up to 100 days.

The hydrogen generation resulting from the zirconium-water reaction, as described previously, is considered an instantaneous input to the containment and represented as the quantity at zero time. Hydrogen from the other sources is reflected in the overall time function.

These results show that the post-LOCA hydrogen concentration inside containment will not reach 4% by volume with one hydrogen recombiner placed in service 1 day after the start of LOCA (see figure 6.2-94).

15.4.1.7 <u>Environmental Consequences of Postulated Loss-of-Coolant Accident</u>

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a LOCA do not result in doses which exceed the NRC acceptance criteria specified in 10 CFR 50.67 The analysis is based on the use of the alternative source term (AST) methodology in Regulatory Guide 1.183[27].

The parameters used in the analysis are listed in table 15.4-14. There are four release pathways to the environment modeled in the LOCA analysis: a release during containment purge, direct leakage from containment, ESF leakage outside containment, and leakage from the RWST. The RADTRAD (Version 3.10) code is used in this analysis to calculate the immersion and inhalation dose contributions to both the onsite and offsite radiological dose consequences. Tables 15B-2 and 15B-3 provide the offsite and control room atmospheric dispersion factors used in the radiological dose consequence analysis.

15.4.1.7.1 Containment Purge Pathway

The containment mini-purge system normally operates during modes 1, 2, 3, and 4 to provide an acceptable working environment inside containment. Following a LOCA, the mini-purge system isolates prior to the onset of the gap release (as described in Table 4 of RG 1.183), so only those nuclides in the RCS source term are available for release.

For the containment purge release, the entire RCS inventory is assumed to be instantaneously and thoroughly mixed throughout the containment and no sprays or iodine deposition are credited. Since the containment is well mixed with no iodine removal, it is modeled as a single compartment with a volume of 2,030,000 ft^3.

The mini-purge exhaust fan discharges to the plant vent through the containment purge filtration unit. For conservatism, no credit is taken for this filtration unit in this analysis. The containment mini-purge system is assumed to be in operation with a fan flow rate of 2850 ft^3/min, and terminates within 30 s of the start of the event. Atmospheric dispersion factors for the plant vent are applied.

15.4.1.7.2 Containment Leakage Pathway

The RADTRAD model used to evaluate the dose contribution due to leakage from the containment includes four compartments and seven pathways between those compartments.

The volume of the sprayed region of containment is the fraction of the total volume covered by containment sprays (82.2% of 2,030,000 ft^3). The source term is distributed uniformly throughout the containment; therefore, the source fraction applicable to this compartment is equal to the 82.2% sprayed fraction. Removal of elemental iodine and aerosols by both containment sprays and natural deposition is modeled in this compartment.

The unsprayed region of containment is 17.8% of the total 2,030,000 ft^3 containment volume. The source term fraction is set equal to the sprayed volume fraction. Elemental iodine and aerosol removal by wall deposition is credited in this compartment.

The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment corresponds to two turnovers of the unsprayed regions per hour. Containment spray flow begins at 90 s and is terminated after 8 h. Circulation flow is not credited between the sprayed and unsprayed region when the containment sprays are secured. The leakage from each of the sprayed and unsprayed regions of containment to the environment is 0.15% per day, which is reduced to 0.075% per day after 24 h.

15.4.1.7.3 ESF Leakage Pathway

ESF leakage is the leakage of sump fluid through valve packing, pump seals, and similar components into the auxiliary building. The ESF leakage RADTRAD model is represented by four compartments and five pathways.

With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to be instantaneously and homogeneously mixed in the primary sump water. The containment sump is modeled as a single compartment with a volume of 49,200 ft^3. The release is due to ESF leakage egress to the penetration room, and is treated by the penetration room filtration system (PRFS) prior release to the plant vent. No credit is provided for holdup or dilution in the auxiliary building.

Sump fluid release to the environment begins with the start of ECCS recirculation, which occurs at 20 min. The leakage rate is analyzed as 40,000 cc/h in the LOCA dose calculation. The flashing fraction applied to the ESF leakage is 10%. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

The PRFS is aligned to filter the release 30 min after the start of the LOCA. The PRFS filter efficiencies are 89.5% for particulates and all forms of iodine.

15.4.1.7.4 RWST Release Pathway

The ESF leakage pathways include those through valves that isolate containment sump water from interfacing systems. Seat leakage past valves which isolate recirculation flow to the RWST is included. The adjusted leakage rate from the sump, through the RWST, to the environment is modeled as a direct connection between the sump and the environment. All of the radioactive materials in the recirculating liquid with the exception of the iodines are assumed to be retained in the liquid phase, as the leakage path to the RWST is below the RWST waterline.

15.4.1.7.5 Discussion of Results

The atmospheric dispersion factors that were used in calculating the offsite radiological dose consequences are listed in table 15B-2. The total effective dose equivalent (TEDE) from the LOCA at the site boundary and low-population zone are given in table 15.4-15. The dose limits for this accident are defined in 10 CFR 50.67, as shown in table 15.4-15. The doses for this conservatively analyzed accident are within the 10 CFR 50.67 requirements. The doses from a LOCA with all safeguards operating as designed would be several orders of magnitude lower than the doses presented in table 15.4-15.

15.4.1.7.6 Radiological Consequences of a Small Break LOCA

For this evaluation, a small break LOCA is defined as the release of 100 percent of the activity in the fuel-clad gas gap. The input parameters are as shown in table 15.4-14. The whole body, skin, and thyroid doses from a small break LOCA at the site boundary and low-population zone meet NRC acceptance criteria; i.e., they are within the 10 CFR 100 guidelines.

15.4.1.8 Radiological Consequences to Control Room

A conservative analysis is performed to determine the radiological consequences to control room personnel following the postulated LOCA. The parameters used are summarized in table 15.4-16.

15.4.1.8.1 Control Room Ventilation System

The design of the control room air-conditioning and filtration system is described in subsection 9.4.1. Actuation logic for the emergency pressurization system is described in paragraph 9.4.1.5.

Following the postulated LOCA, the control room will be pressurized by a nominal flow of 300, +75-30 ft^3/min of filtered air into the control room. However, 10 ft^3/min of unfiltered outside air inleakage is conservatively assumed to account for opening and closing of the control room doors and 325 ft^3/min unfiltered outside air inleakage is assumed for the control room envelope. Filtered recirculation of control room air occurs at a nominal rate of 3000, $\pm$ 300 ft^3/min (2700 ft^3/min, the conservative rate, is used in the analysis).

15.4.1.8.2 Atmospheric Dilution Factors

The atmospheric dilution factors (χ/Q) were computed at the control room intakes for each hour of meteorological data from January 2000 through December 2004 using ARCON96 as described in Regulatory Guide 1.194. Values were determined for Unit 1 and Unit 2 containment vent stack and RWST release points and the most conservative values were used in calculating radiological consequences to the control room. Table 15.B-2 gives the bounding values for each averaging time. The higher values resulted from a Unit 2 release point.

15.4.1.8.3 Discussion of Results

The TEDE to the occupants of the control room for the duration of the LOCA are given in table 15.4-17.

The dose limit applicable to personnel in the control room is 5 rem TEDE, as specified in 10 CFR 50.67.

15.4.1.9 Environmental Consequences of Containment Purging to Control Hydrogen After a Loss-of-Coolant Accident

Post-LOCA containment purging provides a backup method to the electric recombiners for controlling the potential hydrogen accumulation in the containment.

Two analyses of environmental consequences of purging are performed: a realistic analysis and an analysis based on Regulatory Guide 1.7.[26] The parameters used for each of the analyses are listed in table 15.4-18.

The purging system requires a differential pressure between the containment and the outside atmosphere in order to permit purging. The Regulatory Guide 1.7 analysis is based on a pressure of 2 psig in the containment. If required, the containment is pressurized to 2 psig with diluent air when the hydrogen reaches 3.5 volume percent after the LOCA in the conservative analysis. The hydrogen concentration is reduced by this pressurization. Purging is thus delayed until the next day’s hydrogen concentration in the containment has been estimated to exceed 3.5 volume percent.

The 3.5% hydrogen level was selected as the point of starting the purge because of the following factors:

A. This level allows a sufficient margin of safety below the lower flammability limit of 4%.

B. It provides a sufficient margin so that purging could be delayed a few days if so desired.

C. The optimum starting time for the purge, from the standpoint of minimizing the doses, is the latest time.

This level allows sufficient margin of safety below the lower flammability limit of 4.1%.

The optimum starting time for the purge, from the standpoint of minimizing the doses, is the latest time. For power uprate the purge begins at approximately 18 days and continues for this duration of the accident.

The purge rate was selected to match the rate of hydrogen generation at the time of initiation of the purge. The hydrogen concentration in the containment will be maintained below 4% as purging continues.

The dose analysis is based on the activity released from the containment after the time of the postulated LOCA until 30 days. The thyroid and whole body doses as a function of distance from the plant due to activity release from containment leakage following the postulated LOCA are computed using the activity release model described in paragraph 15.4.1.7. Additionally, the dose analysis is based on 100% of the noble gases and 50% of the iodines released to the containment. This is due to the containment spray system and plateout reducing the amount of iodine available for release to the environment.

For the Regulatory Guide 1.7 analysis, containment purge system filter efficiencies of 89.5%, 30%, and 98.5% are used for the removal of elemental, methyl, and particulate iodines, respectively, which have been reduced by 0.5% for bypass leakage.

The dose models discussed and the atmospheric diffusion factors given in appendix 15B are used in determining doses.

The beta, gamma, and thyroid doses due to containment purging at the low-population zone are given in table 15.4-19 and meet the NRC acceptance criteria. That is, the calculated doses for the post-LOCA containment purging are well within (25% of) the guidelines of 10 CFR 100.

15.4.1.10 Conclusions

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will meet the NRC acceptance criteria as presented in 10 CFR 50.46. That is as follows:

A. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

B. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for an extended period of time required by the long-lived radioactivity.

15.4.2 MAJOR SECONDARY SYSTEM PIPE RUPTURE

15.4.2.1 Rupture of Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem, mainly because of the high power peaking factors that would exist assuming the most-reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid delivered by the ECCS.

For a double-ended rupture of a main steam line, the radiation releases must remain within the requirements of 10 CFR 50.67. These are the ANSI N18.2 criteria for Condition IV events, "Limiting Faults." The criteria are conservatively met by demonstrating that the DNB design basis is met, a criterion typically used for Condition II events. Therefore, the analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

1. Assuming a stuck RCCA (with or without offsite power), and assuming a single failure in the ESF, there is no consequential damage to the primary system and the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 100. Note: Conformance to Part 100 is superseded by the radiological limits of 10 CFR 50.67 for the Farley main steam line break accident.

2. Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

   The rupture of a major steam line, which is classified as an ANS Condition IV event, is the most limiting cooldown transient. It is analyzed at zero power with no decay heat since decay heat would retard the cooldown, thereby reducing the return to power. A detailed discussion of this transient with the most limiting break size, (a double-ended rupture), is presented here.

The following functions provide the necessary protection against a steam pipe rupture:

A. Safety injection system actuation from any of the following:

   1. Two out of three low-pressurizer pressure signals.

   2. High steam line differential pressure.

   3. Low main steam line pressure in two out of three steam lines.

   4. Two out of three high containment pressure signals.

B. The overpower reactor trips and the reactor trip occurring in conjunction with receipt of the SI signal.

C. Redundant isolation of the main feedwater lines to prevent sustained high feedwater flow which would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and indirectly close the feedwater isolation valves that back up the control valves. In addition, trip of the steam generator feedwater pumps results in automatic closure of the respective pump discharge isolation valve.

D. Trip of the fast-acting main steam line isolation valves (MSIVs, assumed to close in < 10 s) or main steam line isolation bypass valves (MSIBVs, assumed to close in < 10 s) after receipt of an ECCS or main steam line isolation signal on:

   1. High steam flow in two out of three main steam lines (one of two per line) in coincidence with two out of three low-low RCS average temperature signals.

   2. Low steam line pressure signal in any two out of three steam lines.

   3. Two out of three high-high (hi-2) containment pressure signals.

For breaks downstream of the isolation valves, closure of all valves will completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close. Circuit design assures that the MSIBVs are closed whenever the MSIVs are closed.

Following a steam line break, only one steam generator can blow down completely. Each main steam line is provided with two isolation valves located outside of the containment immediately downstream of the steam line safety valves. The isolation valves are signal-actuated valves which close to prevent flow in the normal (forward) flow direction. The valves on all three steam lines will be driven closed and isolate the other steam generators. Thus, only one steam generator can blow down, minimizing the potential steam release and resultant RCS cooldown. In addition, the remaining two steam generators will still be available for dissipation of any decay heat after the initial transient is over. In the case of LOSP, this heat is removed to the atmosphere via the atmospheric dump valves which have been sized to handle this situation.

Steam flow is measured by monitoring pressure difference between pressure taps in the steam drum and downstream of the integral flow restrictor nozzles. The effective throat diameter of flow restrictors is 14 in., of considerably smaller diameter than the main steam pipe. These restrictors are located in the steam generators outlet nozzle and serve to limit the maximum steam flow for any break at any location.

15.4.2.1.2 Analysis of Effects and Consequences

15.4.2.1.2.1 **<u>Method of Analysis</u>.** The analysis of the steam pipe rupture has been performed to determine:

A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The RETRAN-02[52, 53, 54] code has been used.

B. The thermal and hydraulic behavior of the core following the steam line break. A detailed thermal and hydraulic digital computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in A above.

The following conditions were assumed to exist at the time of the main steam line break accident:

A. End of life (EOL) shutdown margin at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

B. The negative moderator coefficient corresponding to the EOL rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus average coolant temperature at 1000 lb/in.^{2} corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown in figure 15.2-40, along with the effect of power generated in the core on overall reactivity.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high-power region near the stuck rod. To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the hot-water enthalpy near the stuck RCCA, power redistribution, and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high-flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the true reactivity verifying conservatism, i.e., underprediction of negative reactivity feedback from power generation.

C. Minimum capability for injection of high concentration boric acid solution (2300 ppm from the RWST) corresponding to the most restrictive single failure in the ECCS.

The 2300-ppm boron solution corresponds to the minimum boron concentration in the RWST. A boric acid solution of 0 ppm is assumed in the high-head injection lines and the equivalent volume of the boron injection tank (BIT), which has been deleted. No credit has been taken for the low concentration of boric acid that must be swept from the ECCS lines downstream of the RWST isolation valves prior to the delivery of the concentrated boric acid (2300 ppm from the RWST) to the reactor coolant loops.

The SI curve assumed is shown in figure 15.2-41. The flow corresponds to that delivered by one charging pump delivering full flow to the cold leg header. The variation of the mass flowrate due to water density changes is included in the calculations, as is the variation in flowrate in the ECCS due to changes in the RCS pressure. The ECCS flow calculation includes the line losses as well as the SI pump head curve. The modeling of the ECCS in the Westinghouse PWR RETRAN model is described in reference 54.

The boric acid solution from the ECCS is assumed to be uniformly delivered to the three reactor coolant loops. The boron in the loops is then delivered to the inlet plenum where the coolant (and boron) from each loop is mixed and delivered to the core. The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the core. The concentration after mixing depends on the relative flowrates of the RCS and the ECCS. The stuck RCCA is conservatively assumed to be located in the core sector near the faulted steam generator.

For the cases where offsite power is assumed, the sequence of events in the ECCS is as follows. After the generation of the SI signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high-head injection pump starts. In 27 s, the valves are assumed to be in their final position and the pump is assumed to be at full speed and to be drawing suction from the RWST. The 27 s can be assumed to include 2 s for electronic delay, 10 s for the RWST valve to open, and 15 s for the VCT valve to close. The SI system piping is assumed to contain no boron (0 ppm). This delays the 2300-ppm boron concentration RWST water from reaching the RCS. This delay in the 2300-ppm solution reaching the RCS is inherently included in the RETRAN model.

In cases where offsite power is not available, an additional 15-s delay is assumed to start the diesels and to reenergize the ESF electrical buses. That is, after a total of 42 s following the time an SI setpoint is reached at the sensor, the ECCS is assumed to be capable of delivering flow to the RCS.

D. To maximize primary-to-secondary heat transfer, 0% steam generator tube plugging is assumed.

E. Since the steam generators are provided with integral flow restrictors with a 1.069-ft^2 throat area, any rupture with a break area greater than 1.069 ft^2, regardless of location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.069 ft^2 break. The following cases have been considered in determining the core power and RCS transients:

1. Complete severance of a pipe, with the plant initially at no-load conditions, and full reactor coolant flow with offsite power available.

2. Complete severance of a pipe with the plant initially at no-load conditions with offsite power unavailable. Loss-of-offsite power (LOSP) results in coolant pump coastdown.

F. Power-peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void, in conjunction with the large negative moderator coefficient, partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, operating history, temperature, pressure, and flow, and thus are different for each case studied.

Both cases assume initial hot-standby conditions at event initiation since this represents the most conservative initial condition.

Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection when the power level or ΔT reaches a trip setpoint. Following a trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero. In addition, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the RCS cooldown are less for steam line break occurring at power.

G. In computing the steam flow during a steam line break, the Moody Curve (reference 25) for fl/D = 0 is used. The moody multiplier is 1 with a discharge at dry saturated steam conditions.

H. Perfect moisture separation in the steam generator is assumed unless the mixture level reaches the top of the steam generator. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core.

I. The maximum feedwater flow is assumed. Increasing the feedwater flowrate aggravates cooldown accidents like steam line rupture. All main and auxiliary feedwater (AFW) pumps are assumed to be operating at full capacity when the rupture occurs. The analysis of the RCS and main steam system transients, presented in figures 15.4-28 through 15.4-31, assumes an AFW flow of 1000 gal/min delivered to the faulted steam generator with total flow to all steam generators not to exceed 2200 gal/min.

J. The effect of heat transferred from thick metal in the pressurizer and reactor vessel upper head is not included in the cases analyzed. Studies previously performed show that the heat transferred from these sources is a net benefit in DNBR and RCS energy when the effect of the extra heat on reactivity and peak power is considered.

15.4.2.1.2.2 **Results.** The time sequence of events for postulated steam line rupture accidents with and without offsite power is presented in table 15.4-5. The results presented are a conservative indication of the events that would occur assuming a steam line rupture since it is postulated that all of the conditions described in the prior section occur simultaneously.

Figures 15.4-28 and 15.4-29 show the RCS transients and core heat flux following a main steam pipe rupture. Offsite power is assumed to be available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator.

As can be seen, the core attains criticality with RCCAs inserted (with the design shutdown margin assuming one stuck RCCA) just after the boric acid solution at 2300 ppm enters the RCS from the ECCS which is drawing from the RWST. The delay time consists of the time to receive and actuate the SI signal, to start the high-head SI (HHSI) pumps, and to completely align valve trains in the ECCS lines. The HHSI pumps are then ready to deliver flow. At this stage, a further delay is incurred before 2300-ppm boron solution can be injected to the RCS due to the low concentration solution being swept from the SI lines. Should a partial LOSP occur such that power is lost to the ESF functions, an additional SI delay of 15 s would occur while the diesel generators start up and reenergize the ESF buses. Allowing for these delays, a peak core power well below the nominal full-power value is attained.

Should the core be critical at near zero power when the rupture occurs, the initiation of the SI signal by high steam line differential pressure, low steam line pressure, or high containment pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic closure of the isolation valves in the steam lines by low steam line pressure, a high steam flow signal in coincidence with low-low RCS temperature, or high-high containment pressure. The main steam isolation valves (MSIVs) and the MSIBVs are assumed to be fully closed in < 10 s after receipt of a closure signal. Complete steam line isolation occurs when both the MSIVs and MSIBVs are fully closed.

Figures 15.4-30 and 15.4-31 show the responses of the salient parameters for the case discussed above with a total LOSP at the time of the rupture. This assumption results in a coastdown of the reactor coolant pumps (RCPs). In this case, the core power increases at a slower rate and reaches a lower peak value than in the case in which offsite power is available to the RCPs. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS.

It should be noted that following a steam line break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In case of a LOSP, this heat is removed to the atmosphere via the steam line safety valves.

Following blowdown of the faulted steam generator, the plant can be brought to a stabilized hot-standby condition through control of the AFW flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the AFW system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 min following SI.

In conjunction with analyses supporting a relaxed setpoint for the overpower ΔT reactor trip function, an analysis of the steam line break event initiated from full power conditions was performed. This event was also analyzed to support the Model 54F steam generators.

Reactor protection in the limiting case is explicitly provided via a reactor trip on the overpower ΔT function. The appropriate reactor trip delays are modeled as indicated in table 15.1-3. The analysis results demonstrate that the minimum DNBR does not go below the limit value and that core power generation does not reach that which would result in fuel damage.

15.4.2.1.3 Conclusions

The analysis has shown that the criteria stated earlier in the accidental depressurization of the secondary system section are satisfied. Although preventing clad damage is not necessary for Condition IV events, the results show that the DNB design basis is met. The dose evaluation, as shown in paragraph 15.4.2.1.4, continues to demonstrate that the Condition IV accident criteria are satisfied.

Additionally, the NRC acceptance criteria contained in IE Bulletin 80-84 are met relative to the core transient (reactivity increase) for a main steam line rupture with continued feedwater addition. All potential sources of water were identified. Although a reactor return-to-power is predicted, there is no violation of specified acceptable fuel design limits. The conclusions of this analysis remain valid.

15.4.2.1.4 Environmental Consequences of a Postulated Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generators. A conservative analysis of the potential offsite doses resulting from a steam line break outside containment is presented. This analysis incorporates assumptions of defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. Parameters used in the analysis are listed in table 15.4-23.

The conservative assumptions used to determine the equilibrium concentrations of isotopes in the secondary system are as follows:

A. The primary to secondary leakage in steam generators occurs when the reactor starts up; leakage remains constant during plant operation.

B. The primary to secondary leakage is evenly distributed in steam generators.

C. Primary coolant noble gas activity is associated with 1% defective fuel given in table 11.1-2 and a limiting concurrent iodine spike activity is 0.5 μCi/gm DEI_{131}. The secondary side concentration of iodine is assumed to be 0.1 μCi/gm DEI_{131}.

D. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off-gas system.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for a steam line break:

A. Prior to the accident, an equilibrium activity of fission products exists in the primary and secondary systems due to a primary to secondary leakage in steam generators.

B. Offsite power is lost, and main steam condensers are not available for steam dump.

C. Eight hours after the accident, the RHR system starts operation to cool down the plant.

D. The primary to secondary leakage is 0.35 gpm in the faulted steam generator and 0.65 gpm in the intact steam generators.

E. A preaccident iodine spike or an accident initiated iodine spike is assumed.

F. Twenty-four hours following the accident, no steam and activity are released to the environment.

G. There is no air ejector release and no steam generator blowdown during the accident.

H. No noble gas is dissolved in the steam generator water.

I. In the intact steam generators, the iodine partition factor is 100. The alkali partition factor is 1000.

J. During the postulated accident, iodine carryover from the primary side in the two intact steam generators is diluted in the incoming feedwater.

K. In the faulted steam generator, all the water boils off and releases through the break immediately after the accident. The partition factor for the iodine released is assumed to be 1.0. After this initial release, further iodine is released due to the primary to secondary leakage in the affected steam generator. A partition factor of 1.0 is also assumed for this iodine release.

L. The primary pressure remains constant at 2235 psig for 0 to 2 h and decreases linearly to atmospheric from 2235 psig during the period of 2 to 24 h.

M. The 0- to 2-h, 2- to 8-h, and 8- to 24-h atmospheric diffusion factors given in appendix 15B, and the 0- to 8-h and 8- to 24-h breathing rates of 3.5×10^{-4} m^3/s and 1.8×10^{-4} m^3/s respectively are applicable.

The steam releases to the atmosphere for the steam line break are given in table 15.4-23.

The TEDE doses for the steam line break accident for the conservative analysis at the site boundary and the low-population zone are given in table 15.4-23. The doses from this accident are within the NRC acceptance criteria described in Regulatory Guide 1.183.

The potential for uncovery of the tubes in the intact steam generators during the event has previously been evaluated for impact on doses, and has not been updated for the implementation of 10 CFR 50.67. The tube uncovery was assumed to exist for the first 1/2 h of the accident, and the tube leakage locations were assumed to all be near the top of the tube bundle and, thus, subject to the uncovery. With the primary to secondary leakage entering the vapor space, no credit was taken for mixing with the secondary coolant, nor was credit taken for a partition factor within the steam generator (i.e., the primary coolant was assumed to be released directly to the environment). The uncovery does not impact the release of noble gases to the environment; thus, the gamma and beta doses are not affected. The uncovery does result in an increase in the accident releases of iodine. The effect on the conservative thyroid dose at the site boundary (assuming a primary to secondary leak rate of 1 gal/min is that this dose remains well within the limits as defined in 10 CFR 100.

15.4.2.1.5 Radiological Consequences to Control Room

A conservative analysis is performed to determine the radiological consequences to control room personnel following the postulated main steam line break. Parameters used in the analysis and radiological dose result are listed in table 15.4.23a.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators.

If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedline check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in subsection 15.2.8).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break), or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in paragraph 15.4.2.1, Rupture of Main Steam Line. Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the RCS because of the following reasons:

A. Feedwater to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

B. Liquid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

C. The break may be large enough to prevent the addition of any main feedwater.

An AFW system is provided to ensure that adequate feedwater will be available to provide heat removal such that:

A. No substantial overpressurization of the RCS shall occur.

B. Liquid in the RCS shall be sufficient to cover the reactor core at all times.

The following provide the necessary protection against a main feedwater rupture:

A. A reactor trip on any of the following conditions:

   1. High-pressurizer pressure.

   2. Overtemperature ΔT (OTΔT).

   3. Low-low steam generator water level in any steam generator.

SI signals from either of the following:

a. Two of three low-pressurizer pressure signals.

b. Two of three high-differential pressure signals between any steam line and remaining steam lines.

c. Low main steam line pressure in any two lines.

d. Two of three high containment pressure.

(Refer to chapter 7 for a description of the actuation system.)

B. An AFW system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to chapter 6 for a description of the AFW system.)

15.4.2.2.2 Analysis of Effects and Consequences

15.4.2.2.2.1 <u>Method of Analysis</u>. A detailed analysis using the RETRAN-02[52, 53, 54] code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics and the RCS, including natural circulation, pressurizer, steam generators, and feedwater system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The RETRAN code is used to calculate the course of the system transient through the time that the auxiliary feedwater system heat removal capacity exceeds decay heat generation.

<u>Case A</u>

The major assumptions for the major feedwater rupture analysis are as follows:

A. The plant is initially operating at 102% of the NSSS design rating (2785 MWt), including a conservatively large RCP heat of 15 MWt.

B. Initial reactor coolant average temperature is 6°F above the nominal value, and the initial pressurizer pressure is 50 psi above its nominal value.

C. No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

D. No credit is taken for the high-pressurizer pressure reactor trip. (Note: This assumption is made for calculational convenience.) Pressurizer power-operated relief valves and spray could act to delay the high-pressure trip. Assumptions C and D permit evaluation of one hypothetical limiting case rather than two possible cases: one with a high-pressure trip and no pressure control, and one with pressure control but no high-pressure trip.

E. Main feed to all steam generators is assumed to stop at the time the break occurs.

F. Saturated liquid discharge (no steam) is assumed from the affected steam generator through the feedline rupture. This assumption minimizes energy removal from the NSSS during blowdown.

G. No credit is taken for the low-low water level trip on the affected steam generator until the steam generator water level reaches 0% of the narrow range span.

H. The worst possible break area is assumed; i.e., one that empties the affected steam generator and causes a reactor trip on low-low steam generator water level as assumed above. This assumption minimizes the steam generator fluid inventory at the time of trip, and thereby maximizes the resultant heatup of the reactor coolant.

I. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

J. No credit is taken for charging or letdown.

K. The cases are analyzed without offsite power. The loss of offsite electrical power occurs after reactor trip (at the time of rod motion). Reactor coolant flow then decreases to that provided by natural circulation.

L. Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

M. ANS-5.1-1979 standard residual heat generation (reference 32) is assumed based upon long-term operation at the initial power level preceding the trip.

N. The AFW is initiated by the operator 10 min after the trip with a feed rate of 350 gal/min. The cold AFW is mixed with the hotter water occupying the AFW purge lines until a homogeneous temperature distribution in the purge lines is achieved for each time step. This produces a conservatively slow transition to the cold AFW.

O. Limiting reactivity coefficients reflecting maximum feedback are assumed. (Note: Separate cases were analyzed with maximum and minimum reactivity feedback. The maximum feedback case yielded more limiting results for Case A.)

<u>Case B</u>

An analysis has also been performed to demonstrate that the operator has at least 30 min to increase AFW flow to the intact steam generators without hot leg boiling prior to transient turnaround. The major assumptions used in this analysis are as follows:

A. The plant is initially operating at 102% of the NSSS design rating (2785 MWt), including RCP heat generation of 15 MWt.

B. Initial reactor coolant average temperature is 6°F above the nominal value, and the initial pressurizer pressure is 50 psi above its nominal value.

C. No credit is taken for the pressurizer power-operated relief valves (PORVs) or pressurizer spray.

D. No credit is taken for the high-pressurizer pressure reactor trip. (Note: This assumption is made for calculational convenience.)

E. Main feed to all steam generators is assumed to stop at the time the break occurs.

F. Saturated liquid discharge (no steam) is assumed from the affected steam generator through the feedline rupture. This assumption minimizes energy removal from the NSSS during blowdown.

G. Credit is taken for the low-low water level trip on the affected steam generator.

H. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

I. No credit is taken for charging or letdown.

J. Loss of offsite electrical power is assumed after the reactor trip, and reactor coolant flow decreases to natural circulation.

K. Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

L. ANS-5.1-1979 standard residual heat generation (reference 32) is assumed based upon long-term operation at the initial power level preceding the trip.

M. The AFW is initiated at 1 min after receipt of a low-low steam generator water level signal with a feed rate of 150 gal/min. This assumes a single failure of the turbine-driven AFW pump. The cold AFW is mixed with hotter water occupying the AFW purge lines until a homogeneous temperature distribution in the purge lines is achieved for each time step. This produces a conservatively slow transition to the cold AFW.

N. Limiting reactivity coefficients reflecting minimum feedback are assumed. (Note: Separate cases were analyzed with maximum and minimum reactivity feedback. The minimum feedback case yielded more limiting results for Case B.).

15.4.2.2.3 Results

Case A

Figure 15.4-32 (sheets 1 and 2) shows the calculated plant parameters following a feedline rupture for Case A. The assumed auxiliary feedwater flowrate is capable of removing decay heat 1820 s after trip. After this time, core decay heat decreases below the auxiliary feedwater heat removal capacity and reactor coolant temperatures and pressures decrease. The calculated sequence of events for this case is presented in table 15.4-5.

Case B

Figure 15.4-32 (sheets 3 and 4) show the calculated plant parameters following a feedline rupture for Case B. The assumed AFW is capable of removing decay heat approximately 2115 s after trip. After this time, core decay heat decreases below the AFW heat removal capacity and reactor coolant temperatures and pressures decrease. The calculated sequence of events for Case B is presented in table 15.4-5.

The system response following the feedwater line rupture is similar for both Case A and Case B. Pressurizer pressure increases until the reactor trip occurs on low-low steam generator narrow range level. Pressure then decreases, due to the loss of heat input, until the SI system is actuated on low steam line pressure in the ruptured loop. Coolant expansion occurs due to reduced heat transfer capability in the steam generators. The pressurizer safety valves open to maintain primary pressure at an acceptable value. The calculated relief rates are well within the relief capacity of the pressurizer safety valves. Addition of the SI flow aids in cooling down the primary and helps to ensure that sufficient fluid exists to keep the core covered with water.

The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the auxiliary feedwater system and makeup is provided by the SI system. Bulk boiling does not occur in the RCS at any time in the transient.

15.4.2.2.4 Conclusion

Results of the analysis show that for the postulated feedline rupture, the assumed AFW system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS and main steam system, and to prevent uncovering the reactor core. The analysis results also verify that the natural circulation capacity of the RCS provides sufficient heat removal capacity following reactor coolant pump coastdown. In addition to the NRC Acceptance Criteria for Condition IV events described in paragraph 15.4, the analysis of the feedwater system pipe break meets the NRC acceptance criteria specific to feedwater system pipe breaks. That is, the primary and secondary side pressures do not exceed allowable limits and the core remains adequately covered.

15.4.3 STEAM GENERATOR TUBE RUPTURE

15.4.3.1 Accident Description

The bounding accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident LOSP or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or PORVs.

In view of the fact that the steam generator tube material is Inconel 600 and is a highly-ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the affected steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the affected unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion for a complete severance of a single steam generator tube, that the isolation procedure can be completed within 30 min of accident initiation.

Assuming normal operation of the various plant control systems, the following events are initiated following the complete severance of a single steam generator tube:

A. Pressurizer low-pressure and low-level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side, there is a steam flow/feedwater flow mismatch alarm as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

B. Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low-pressurizer pressure or overtemperature ΔT. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level; the SI signal, initiated by low-pressurizer pressure, follows soon after the reactor trip. The SI signal automatically terminates normal feedwater supply and initiates AFW feedwater addition. Although the original design bases do not consider a single failure, the analyses prepared for power uprate and thereafter consider the failure of the steam-driven auxiliary feedwater pump. This failure is not the most limiting single failure; however, the consideration of this single failure is more conservative than was required by the original design and is acceptable (reference 51).

C. The secondary side radiation monitors will alarm, indicating a sharp increase in radioactivity in the secondary system, and will isolate steam generator sample and blowdown lines as shown in table 11.4-2.

D. The reactor trip automatically trips the turbine and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a coincident LOSP, the steam dump valves would automatically close to protect the condenser. In this case, the steam generator pressure would rapidly increase, resulting in steam discharge to the atmosphere through the steam generator safety and/or PORVs.

E. Following reactor trip, the continued action of AFW supply and borated SI flow (supplied from the RWST) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of LOSP, steam relief to atmosphere, is attenuated during the 30 min in which the recovery procedure leading to isolation is being carried out.

F. SI flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Method of Analysis

In estimating the mass transfer from the RCS through one completely severed tube, the following conservative assumptions are made:

A. Reactor trip and SI injection occur automatically as a result of low-pressurizer pressure SI setpoint actuation.

B. Following the initiation of the SI, all centrifugal charging/SI pumps are actuated and continue to deliver flow for 30 min.

C. After reactor trip, the break flow reaches equilibrium at the point where incoming SI flow is balanced by outgoing break flow as shown in figure 15.4-23. The resultant break flow persists from plant trip until 30 min beyond initiation of the accident.

D. The steam generators are controlled at the safety valve setting minus 3% main steam safety valve (MSSV) tolerance rather than at the PORV setting. The lowest safety valve setpoint is modeled with 13% blowdown.

E. The operator identifies the accident type and terminates break flow to the affected steam generator within 30 min of accident initiation.

Mass and energy balance calculations are performed to determine primary to secondary mass release and to determine the amount of steam vented from each of the steam generators.

15.4.3.2.2 Recovery Procedure

Immediately apparent symptoms of a tube rupture accident, such as falling pressurizer pressure and level and increased charging pump flow, are also symptoms of small steam line breaks and LOCAs. It is therefore important for the operator to determine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The steam generator tube rupture may be identified by a secondary side radiation monitor (see table 11.4-2) indication or alarm; and the operator will proceed with the following recovery procedures only if at least one of these alarms is received. In the event of a relatively large rupture, it will be clear soon after trip that the level in one steam generator is rising more rapidly than in the others. This too is a unique indication of a tube rupture accident.

The operator normally carries out the following major actions subsequent to a reactor trip:

1. Identify the affected steam generator.

   The affected steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication on the corresponding radiation monitor or sample. In some cases, the affected steam generator may be obvious prior to reactor trip due to steam flow/feed flow mismatch or steam generator level deviation alarms. For larger tube failures, rapidly increasing water level should be evident soon after trip. However, sampling/monitoring for high activity may be necessary to locate smaller tube failures. This response provides confirmation of an SGTR event and also identifies the affected steam generator.

2. Isolate the affected steam generator from the intact steam generators and isolate feedwater to the affected steam generator.

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the affected steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the affected steam generator.

3. Cool down the RCS using the intact steam generators.

   After isolation of the affected steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the affected steam generator pressure by dumping steam from the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the affected steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost or the normal steam dump system is unavailable, the RCS is cooled using the power-operated relief valves PORVs on the intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, RCS pressure must be reduced to stop primary to secondary leakage. The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. However, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and affected steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the affected steam generator.

6. Prepare for cooldown to cold shutdown.

   Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time, a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the affected steam generator.

After the RHR system is placed in operation, the condensate accumulated in the secondary system can be examined and processed as required.

Figure 15.4-24 gives estimated primary and secondary system pressure histories which could be expected to occur during a steam generator tube rupture transient and subsequent recovery. Injected flow equals or exceeds leakage flow at the time of SI actuation. It is conservatively assumed that steam venting through the affected steam generator safety valves occurs until 30 min after the accident. Normally, the operator would isolate the affected steam generator as soon as possible after identifying the accident.

Paragraph 15.4.3.1 describes the accident sequence as analyzed. The flow from a severed tube is assumed to reach an equilibrium at the point where SI flow is balanced by break flow. This break flow is conservatively assumed to persist until 30 min following accident initiation, at which time the operator will have terminated the break flow to the affected steam generator.

Paragraph 15.4.3.2 outlines those operations which the operator could perform to terminate flow to the affected steam generator and prepare for cooldown to cold shutdown. The core will remain completely covered by liquid throughout the accident. Thus, clad temperatures will remain very near the saturation temperature of the coolant, even if DNB were postulated to occur.

There is ample time available to carry out the above recovery procedures such that isolation of the affected steam generator is established before water level rises into the main steam pipes. The available time scale is improved by the termination of AFW flow in the affected steam generator. Normal operator vigilance therefore assures that excessive water level will not be attained.

15.4.3.2.3 Results

Figure 15.4-23 illustrates the break flowrate following reactor trip/SI that would result through the severed steam generator tube. The previous assumptions lead to a conservative upper-limit estimate of 158,000 lb for the total amount of reactor coolant transferred to the secondary side of the affected steam generator as a result of a tube rupture accident. The amount of steam released from the affected steam generator is shown in table 15.4-24

15.4.3.3 Conclusions

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneous LOSP. Offsite dose consequences may be calculated based on a conservative estimate of 158,000 lb of reactor coolant transferred to the secondary side of the affected steam generator following the accident. The amount of steam released from the affected steam generator is shown in table 15.4-24.

15.4.3.4 Environmental Consequences of a Postulated Steam Generator Tube Rupture

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS of the secondary system in the steam generators. A conservative analysis of the postulated steam generator tube rupture assumes the LOSP and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident is presented. This analysis incorporates assumptions of defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system.

Parameters used in the conservative analyses are listed in table 15.4-24.

The conservative assumptions used to determine the equilibrium concentrations of isotopes in the secondary system are as follows:

A. The primary to secondary leakage in steam generators occurs when the reactor starts up; leakage remains constant during plant operation.

B. The primary to secondary leakage is evenly distributed in steam generators.

C. Primary coolant noble gas activity is associated with 1% defective fuel given in table 11.1-2 and a limiting pre-accident iodine activity of 30.0 μCi/gm DEI_{131}. The secondary side concentration of iodine is assumed to be 0.1 μCi/gm DEI_{131}.

D. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off-gas system.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated steam generator tube rupture:

A. Prior to the accident, an equilibrium activity of fission products exists in the primary and the secondary systems due to a primary to secondary leakage in steam generators.

B. Offsite power is lost, and main steam condensers are not available for steam dump.

C. Eight hours after the accident, the RHR system starts operation to cool down the plant.

D. The primary to secondary leakage is 0.35 gpm in the faulted steam generator and 0.65 gpm in the intact steam generators.

E. A preaccident iodine spike or an accident initiated concurrent iodine spike is assumed.

F. Eight hours following the accident, no steam and activity are released to the environment.

G. There is no air ejector release and no steam generator blowdown during the accident.

H. No noble gas is dissolved in the steam generator water.

I. In the intact steam generators, the iodine partition factor is 100. The alkali partition factor is 1000.

J. During the postulated accident, iodine carryover from the primary side in the two intact steam generators is diluted in the incoming feedwater.

K. Steam release to atmosphere and the associated activity release from the intact steam generators is terminated at 8 h after the accident, when the RHR system takes over in cooling down the plant.

L. Thirty minutes after the accident the pressure between the ruptured steam generator and the primary system is equalized. The ruptured steam generator is isolated. No steam and fission product activities are released from the ruptured steam generator thereafter.

M. The 0- to 2- and 2- to 8-h atmospheric diffusion factors given in appendix 15B and the 0- to 8-h breathing rate of 3.5×10^{-4} m^3/s are applicable.

The steam releases to the atmosphere for the steam generator tube rupture are given in table 15.4-24.

Results

For the conservative analysis, doses at the site boundary and at the low population zone are shown in Table 15.4-24. The doses are within the NRC acceptance criteria described in Regulatory Guide 1.183.

The potential for uncovery of the steam generator tubes during the event has also been evaluated for impact on doses. The tube uncovery was assumed to exist for the first 30 min of the accident, and the tube rupture location was assumed to all be near the top of the tube bundle and, thus, subject to the uncovery. With the primary to secondary leakage entering the vapor space, no credit was taken for mixing with the secondary coolant, nor was credit taken for a partition factor within the steam generator (i.e., the primary coolant was assumed to be released directly to the environment). The uncovery does not impact the release of noble gases to the environment; thus, the gamma and beta doses are not affected. The uncovery does result in an increase in the accident releases of iodine. The effect on the conservative thyroid dose at the site boundary (assuming a primary to secondary leak rate of 500 gal/day per intact steam generator) is that this dose meets the NRC acceptance criteria. That is, the doses from the accident are a small fraction (10 percent) of the limits as defined in 10 CFR 100.

15.4.3.5 Radiological Consequences to Control Room

A conservative analysis is performed to determine the radiological consequences to control room personnel following the postulated steam generator tube rupture accident. Parameters used in the analysis and radiological dose result are listed in table 15.4.24a.

15.4.4 SINGLE REACTOR COOLANT PUMP LOCKED ROTOR

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a RCP rotor (such as is discussed in subsection 5.5.1) or the sudden break of the shaft of a RCP. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip due to turbine trip on reactor trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the PORVs, and opens the pressurizer safety valves, in that sequence. The two PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the spray, is not included in the analysis.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of reduction of coolant flow is slightly greater for the locked rotor event. However, with a failed shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of such reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis is performed, representing the most limiting condition for the locked rotor and pump shaft break accidents.

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN[30] code is used to calculate the resulting loop and core coolant flow following the pump seizure, the time of reactor trip (based on the loop flow transient), nuclear power following reactor trip, and to determine the peak RCS pressure. The thermal behavior of the fuel located at the core hotspot is investigated using the FACTRAN[40] code which uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

Two cases are analyzed: a peak RCS pressure evaluation and a peak clad temperature evaluation. In each case, one locked rotor/shaft break with three loops in operation is modeled.

The accident is evaluated without offsite power available. Power is assumed to be lost to the unaffected pumps 2 s after rod motion following a reactor trip on low RCS flow.

15.4.4.2.2 Initial Operating Conditions

At the beginning of the postulated locked rotor accident, the plant is assumed to be operating under the most-adverse steady-state operating conditions. These include the maximum steady-state power level, pressure, and coolant average temperature. The reactivity coefficients assumed in the analysis (table 15.1-2A) include a conservative moderator temperature coefficient of 0 pcm/°F (+7 ≤ 70% RTP ramping to 0 at 100% RTP) and a conservatively large (absolute value) of the Doppler-only power coefficient. The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.016 Δk. For this analysis, the curve of trip reactivity versus time (figure 15.1-4) was used with a 4.8% Δk trip reactivity which includes the most-reactive RCCA stuck out of the core.

15.4.4.2.3 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin 1 s after the flow in the affected loop reaches 85% of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip.

Although these systems are expected to function and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are actuated at 2550 psia. This includes 2% uncertainty over the nominal setpoint of 2500 psia. Additionally, the flow through the pressurizer safety valves was modeled with 5-psi accumulation, i.e., the flow ramps from zero to full-rated flow (steam relief of 287.5 lbm/s) over the range of 2550 to 2555 psia.

15.4.4.2.4 Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core; therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hotspot condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

In the evaluation, the rod power at the hotspot is assumed to be 2.5 at the initial core power level.

15.4.4.2.5 Film Boiling Coefficient

The Bishop-Sandberg-Tong film boiling correlation is used in the FACTRAN code. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures).

The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flowrate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

15.4.4.2.6 Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. For the initial portion of the transient, a high gap coefficient produces higher clad temperatures since the heat stored and generated in the fuel redistributes itself in the cooler cladding. This effect is reversed when the clad temperature exceeds the pellet temperature in cases where a zirconium-steam reaction is present. Based on investigations of the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperatures to 10,000 Btu/h-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel is released to the clad at the initiation of the transient.

15.4.4.2.7 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). In order to take this phenomenon into account, the following Baker-Just correlation, which defines the rate of the zirconium-steam reaction, was introduced into the model[(34)]:

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp - \left[\frac{(45{,}500)}{1.986T}\right]$$

where:

w	=	amount Zr reacted (mg/cm^2)
t	=	time (s)
T	=	temperature (°K)

The reaction heat is 1510 cal/g.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

15.4.4.2.8 Results

Peak RCS Pressure Case

The calculated sequence of events for the peak RCS pressure case is shown in table 15.4-5. The transient results without offsite power available are shown in figures 15.4-33A through 15.4-35A. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits.

Peak Clad Temperature Case

The calculated sequence of events for the peak clad temperature case is shown in table 15.4-5. The transient results without offsite power available are shown on figures 15.4-33 through 15.4-38. The peak clad surface temperature is less than 2700 ºF (for ZIRLO), and the more restrictive limit of 2375°F (associated with Optimized ZIRLO cladding). It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

The results of these calculations (peak pressure, peak clad temperature, and zirconium-steam reaction) are also summarized in table 15.4-25.

15.4.4.3 Conclusions

The analysis has shown the following:

A. Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

B. Since the peak clad surface temperature calculated for the hotspot during the worst transient remains less than 2700°F (for ZIRLO) and 2375°F (Optimized ZIRLO) and the amount of zirconium-water reaction is small, the core will remain in place and intact with no loss of core cooling capability.

C. The number of fuel rods in DNB calculated for dose analysis is less than 20% of the core.

15.4.4.4 Environmental Consequences of a Single Reactor Coolant Pump Locked Rotor

The radiological effects of a single reactor coolant pump locked rotor have been analyzed using assumptions as discussed below. The models used to calculate offsite doses are discussed in appendix 15B.

A. The accident occurs when the reactor has been operating at 102% of full power (2831 MWt).

B. 100% of the gas gap inventory from 20% of the core is released to, and mixes uniformly with, the RCS. The RCS primary to secondary leakage is 1 gpm total for all steam generators.

C. Secondary side concentration of iodine is assumed to be 0.1 μCi/gm DEI_{131}. No noble gas is assumed to be dissolved in the secondary side water.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that fraction of the activity leaking from the primary coolant through the steam generator tubes following the accident. The leakage of primary coolant to the secondary side of the steam generator is assumed to continue at its initial rate, which is assumed to be the same rate as the leakage prior to the accident, for the 8-h duration of the accident. The steam releases and other assumptions are shown in table 15.4-25A. Site boundary, low population zone, and control room doses are also shown in table 15.4-25.

15.4.4.5 Radiological Consequences to Control Room

A conservative analysis is performed to determine the radiological consequences to control room personnel following the locked rotor accident. Parameters used in the analysis and radiological dose result are listed in table 15.4.25a.

15.4.5 FUEL HANDLING ACCIDENT

15.4.5.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent-fuel assembly onto the spent-fuel pool floor or the refueling canal floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.4.5.2 Analysis of Effects and Consequences

The radiological effects of dropping a spent-fuel assembly have been analyzed for two separate cases, depending on whether the accident occurs inside the auxiliary building or inside the containment. Both cases are analyzed conservatively using assumptions outlined in Regulatory Guide 1.183, as discussed below. The models used to calculate offsite doses are discussed in appendix 15B.

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident (FHA):

A. The accident occurs at 100 h following the reactor shutdown; i.e., the earliest time after shutdown at which spent-fuel operations would begin. Radioactive decay of the fission product inventory is taken into account during this interval.

B. In an FHA, only the outer row of rods in an assembly is expected to be damaged. However, in this analysis, it is assumed that all the rods in an assembly are damaged.

C. The damaged assembly is, conservatively, the one operating at the highest power level in the core region to be discharged. See table 15.4-26 for activities.

D. The entire activity in the clad gap of the damaged assembly at the time of the accident is assumed to be released. For the Regulatory Guide 1.183 auxiliary building and containment analyses, this consists of 8% for I-131, 10% for Kr-85, and 5% for other halogens and noble gases. The iodine released from the fuel is 99.85% elemental and 0.15% organic.

E. The spent-fuel pool or refueling canal water retains a large fraction of the iodine released from the damaged fuel assembly. For the Regulatory Guide 1.183 auxiliary building and containment analyses, an overall decontamination factor (DF) of 200 is assumed for iodine (for elemental iodine DF = 500 and for organic iodine no scrubbing removal is assumed, so DF = 1). Noble gases are also assumed not to be retained by the water, that is, DF = 1.

F. For an FHA in the spent-fuel pool, all radioactivity released to the auxiliary building is assumed to be released to the environment over 2 h through the penetration room filtration system.

   For an FHA in the containment, no credit is taken for the containment purge exhaust radiation monitor; the containment purge is assumed to be exhausted through the open equipment hatch and personnel airlock. Although the containment purge filters remain in place, no credit is taken for filtration of contaminated air exhausted.

G. Iodine escaping from the spent-fuel pool will be detected in the pool sweep ductwork and an alarm signaled to the control room operator. The normal ventilation system will be isolated automatically and the activity will be exhausted through the penetration room filtration (PRF) system. Both of the 100% capacity PRF systems will receive an automatic start signal. A single PRF system is capable of meeting all requirements of the FHA analyses. In the event of low flow in the normal ventilation system, the PRF system will automatically start, thus assuring a negative pressure inside the fuel handling area. Charcoal filter efficiencies of 89.5% for elemental and organic iodine and aerosols are assumed, which includes a 0.5% reduction for bypass leakage.

   Movement of new fuel over spent fuel with the spent-fuel area roof new fuel access hatch open creates the potential for an FHA with a release pathway which bypasses the radiation monitors in the exhaust duct and, consequently, a bypass of the PRF. This configuration is required to move new fuel assemblies from the new fuel storage area into the spent-fuel pool (SFP) prior to a core reload. In addition, the PRF system may not provide a negative pressure differential between the SFP area and adjacent areas with the hatch removed. For this case, the accident occurs 676 h after reactor shutdown and 100% of the released radioactivity is assumed to bypass the PRF system filters.

For the FHA inside containment, the released radioactivity is discharged unfiltered until all activity is purged. The FHA assumes essentially 100 percent of the fission products released from the reactor cavity are releases to the environment in 2 h without any credit for filtration.

As noted in the Technical Specification Bases, the potential for containment pressurization as a result of an accident in Mode 6 (refueling) is not likely; therefore, requirements to isolate the containment from outside atmosphere can be less stringent. The Technical Specification requirements are referred to as “refueling integrity” rather than “containment OPERABILITY.” Refueling integrity means that all potential escape paths are closed or are capable of being closed. During periods of unit shutdown, when refueling integrity is not required, the personnel airlock door interlock mechanism may be disabled, allowing both doors of an airlock to remain open for extended periods when frequent containment entry is required. During core alterations or movement of irradiated fuel assemblies within containment, refueling integrity is required and one airlock door must always remain capable of being closed. The 2-h closure is not included in the Technical Specification Bases, since the analysis did not extend beyond 2 h. However, as a defense in depth, the commitment was made to have one door in each personnel airlock closed following evacuation.

FNP procedures demonstrate the capability to close the Unit 1 or Unit 2 equipment hatches when the hatches are open during core alterations or movement of irradiated fuel, in the event of an FHA inside containment within the 2-h time limitation. A designated, trained maintenance closure response team (MCRT) is available to facilitate prompt closure of at least one door each of the personnel and auxiliary access locks, after evacuation of containment, in the event of an FHA (reference 60).

H. Short-term atmospheric dilution factors are taken from table 15B-2 for the site boundary and low-population zone.

I. The control room ventilation will be switched to the emergency ventilation system as described in subsection 9.4.1.

15.4.5.3 Environmental Consequences of a Fuel Handling Accident

The assumptions used to analyze the consequences of an FHA in the SFP (auxiliary building) or refueling canal (containment) are discussed in paragraph 15.4.5.2. The assumptions are summarized in table 15.4-27.

The activity released to the environment following the postulated FHA is dependent on the location of the accident.

Table 15.4-26 lists activity releases to the environment for an FHA occurring in the SFP or in the refueling canal. The corresponding TEDE doses at the control room, site boundary, and low-population zone are presented in tables 15.4-29 and 15.4-30. The NRC acceptance criteria require offsite doses to be within 6.3 rem TEDE per Regulatory Guide 1.183.

15.4.6 RUPTURE OF A CONTROL ROD DRIVE MECHANISM (CRDM) HOUSING (RCCA EJECTION)

15.4.6.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a CRDM pressure housing resulting in the ejection of a drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.6.1.1 Design Precautions and Protection

Certain features in Westinghouse pressurized-water reactors (PWRs) are intended to preclude the possibility of a rod ejection accident or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at power.

15.4.6.1.1.1 Mechanical Design. The mechanical design is discussed in section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA to be rapidly ejected from the core are listed below:

A. All full-length CRDM housings are completely assembled with the reactor vessel head and shop tested as an assembly at 3107 psig.

B. Deleted.

C. Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

D. The latch mechanism housing and rod travel housing are each a single length of forged type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and rod travel housing are threaded joints reinforced by canopy-type rod welds. Administrative procedures require periodic inspections of these welds.

15.4.6.1.1.2 Nuclear Design. Even if a rupture of an RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger-than-normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one assembly deviates from its bank. There are low and low-low level insertion alarm circuits for each bank. In addition, bank positions and the low-low limit are provided by the rod insertion limit recorder. Operating instructions require boration at low level alarm and emergency boration at the low-low level alarm.

15.4.6.1.1.3 Reactor Protection. The RPS response to a rod ejection accident has been described in reference 35. The protection for this accident is provided by the power range high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in section 7.2.

15.4.6.1.1.4 Effects on Adjacent Housing. Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings leading to increased severity of the initial accident.

15.4.6.1.2 Limiting Criteria

Due to the extremely low probability of an RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation.[36] Extensive tests of UO_2 zirconium clad fuel rods representative of those in PWR-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/g. However, other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT[37] results which indicated a failure threshold of 280 cal/g. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/g for unirradiated rods and 200 cal/g for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/g.

The real physical limits of this accident are that the rod ejection event and any consequential damage to either the core or the RCS must not prevent long-term core cooling and any offsite dose consequences must be within the guidelines of 10 CFR 100. More specific and restrictive criteria are applied to ensure that fuel dispersal in the coolant, gross lattice distortion, or severe shock waves will not occur. In view of the above experimental results, the conclusions of WCAP-7588, Rev. 1-A (reference 38) and reference 39, the limiting criteria are as follows:

A. Average fuel-pellet enthalpy at the hotspot must be maintained below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

B. Peak reactor coolant pressure must be less than that which would cause stresses to exceed the faulted condition stress limits.

C. Fuel melting will be limited to less than 10% of the fuel volume at the hotspot even if the average fuel-pellet enthalpy is below the limits of criterion A.

15.4.6.2 <u>Analysis of Effects and Consequences</u>

15.4.6.2.1 Method of Analysis

The calculation of the rod ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core power calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hotspot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in reference 40.

15.4.6.2.1.1 <u>Average Core Analysis</u>. The spatial kinetics computer code TWINKLE[42] is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equations in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel clad coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in subsection 15.1.9.

15.4.6.2.1.2 <u>Hotspot Analysis</u>. The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hotspot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and a transition boiling correlation (Bishop-Sandburg-Tong)(27) to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient may be calculated by the code; however, it is adjusted in order to force the full-power, steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in subsection 15.1.9.

15.4.6.2.1.3 <u>System Overpressure Analysis</u>. Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hotspot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient taking into account fluid transport in the system and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.6.2.2 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-12 presents the parameters used in this analysis.

15.4.6.2.2.1 Ejected Rod Worths and Hot Channel Factors. The values for ejected rod worths and hot channel factors are calculated using either three-dimensional static methods or a synthesis of one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation. The total transient hot channel factor, F_q, is then obtained by combining the axial and radial factors, even though the axial peaks are not coincident under the conditions of calculation.

The ejected rod worth and hot channel factors include appropriate margins to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

15.4.6.2.2.2 Reactivity Feedback Weighting Factors. The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single-channel analysis. Physics calculations are carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes are compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, the axial weighting is not used. In addition, no weighting is applied to the moderator feedback. A very conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors are shown to be conservative compared to three-dimensional analysis.[38]

15.4.6.2.2.3 Moderator and Doppler Coefficient. The critical boron concentrations at the beginning of life (BOL) and end of life (EOL) are adjusted in the nuclear core in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results. The resulting moderator temperature coefficient is at least +7 pcm/°F at the appropriate zero- or full-power nominal average temperature for the BOL cases.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. This weighting factor will increase under accident conditions (as discussed above).

15.4.6.2.2.4 Delayed Neutron Fraction, β. Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at BOL and 0.50% at EOL. The accident is sensitive to β if the ejected rod worth is equal to or greater than β_{eff} as in zero-power transients. In order to allow for future fuel cycles, conservative estimates of β at beginning of cycle and at end of cycle were used in the analysis. (See table 15.4-12.)

15.4.6.2.2.5 <u>Trip Reactivity Insertion</u>. The trip reactivity insertion is assumed to be 4.8% from hot full power and 1.77% from hot zero power, including the effect of one stuck rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 s after the high neutron flux trip setpoint is reached. It is assumed that insertion to dashpot does not occur until 2.7 s after the rods begin to fall. This time to full insertion, together with the 0.5-s trip delay, overestimates the time for significant insertion of shutdown reactivity into the core. The choice of such a conservative insertion rate means that there is over 1 s after reaching the trip point before significant shutdown reactivity is inserted into the core. This is a significant conservatism for hot full-power accidents.

The minimum design shutdown margin available for this plant at hot zero power may only occur at EOL in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that two stuck RCCAs (one of which is the worst ejected rod) reduce the shutdown margin by about an additional 1% $\Delta\rho$. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to hot zero power.

15.4.6.2.3 Results

The calculated sequence of events is shown in table 15.4-5. The values of the parameters used in the analysis, as well as the results of the analysis, are presented in table 15.4-12 and are discussed in the following paragraphs.

15.4.6.2.3.1 <u>Beginning of Cycle, Full Power</u>. Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated. Key analysis assumptions and results are presented in table 15.4-12.

15.4.6.2.3.2 <u>Beginning of Cycle, Zero Power</u>. For this condition, control bank D was assumed to be fully inserted and bank C was at its insertion limit. The worst ejected rod is located in control bank D. Key analysis assumptions and results are provided in table 15.4-12.

15.4.6.2.3.3 <u>End of Cycle, Full Power</u>. Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated. Key analysis assumptions and results are presented in table 15.4-12.

15.4.6.2.3.4 <u>End of Cycle, Zero Power</u>. The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and bank C to be at its insertion limit. The worst ejected rod is in control bank. Key analysis assumptions and results are presented in table 15.4-12.

A summary of the cases presented in paragraphs 15.4.6.2.3.1 through 15.4.6.2.3.4 above is given in table 15.4-12. The nuclear power and hotspot fuel temperature transients for two cases (BOL full power and EOL zero power) are presented in figures 15.4-40 through 15.4-43.

15.4.6.2.3.5 Fission Product Release. It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three-dimensional THINC analysis. Although limited fuel melting at the hotspot was predicted for the full-power cases, in practice, melting is not expected since the analysis conservatively assumed that the hotspots before and after ejection were coincident.

15.4.6.2.3.6 Pressure Surge. A detailed calculation of the pressure surge for an ejection rod worth of $1 at BOL hot full power indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits. Since the severity of the present analysis does not exceed this "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further adverse effects to the RCS.

15.4.6.2.3.7 Lattice Deformations. A large temperature gradient will exist in the region of the hotspot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotspot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hotspot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly and it is considered inconceivable that cross-flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling sufficient to distort the lattice is hypothetically postulated, the large void fraction in the hotspot region would produce a reduction in the total core moderator to fuel ratio. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analyses.

15.4.6.3 Conclusions

Despite the conservative assumptions, the analyses indicate that the described fuel limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated that the fuel rods entering DNB are less than 10% of the fuel rods in the core; therefore, the assumption of 10% of the fuel rods in the core entering DNB for the fission product release calculation is conservative. In addition, the maximum average fuel pellet enthalpy was < 200 cal/g for all control rod ejection events, thus meeting the NRC acceptance criteria of < 280 cal/g.

15.4.6.4 Environmental Consequences of a Postulated Rod Ejection Accident

A conservative analysis of a postulated rod ejection accident is performed based on Regulatory Guide 1.183.[40] The parameters used for the analysis are listed in table 15.4-31.

The conservative analysis of the doses resulting from a rod ejection accident is based on the analysis given previously in this section which demonstrates a conservative fission product release of the gap activity from 10% of the fuel rods in the core.

For the conservative analysis, it is assumed that the plant is operating at equilibrium levels of radioactivity in the primary and secondary systems prior to the postulated rod ejection accident as a result of coincident fuel defects and steam generator tube leakage. Following a postulated rod ejection accident, two activity release paths contribute to the total radiological consequences of the accident. The first release path is via containment leakage resulting from release of activity from the primary coolant to the containment. The second path is the contribution of contaminated steam in the secondary system dumped through the relief valves since offsite power is assumed to be lost.

15.4.6.4.1 Model

Prior to the accident, it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a sufficient period of time to establish equilibrium levels of activity in the primary and secondary coolant.

Following a postulated rod ejection accident, the activity released from the fuel-pellet clad gap due to failure of a portion of the fuel rods is assumed to be instantaneously released to the primary coolant. It is assumed that this release to the primary coolant is uniformly mixed throughout the coolant instantaneously. Thus, the total activity released from the fuel rod gaps is assumed to be immediately available for release from the RCS.

Of the activity released to the containment from the coolant through the rupture in the reactor vessel head, 100% is assumed to be mixed instantaneously throughout the containment and is available for leakage from the containment at the design leak rate. The removal processes considered in the containment are plateout, radioactive decay, and leakage from the containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that fraction of the activity leaking from the primary coolant through the steam generator tubes following the accident. The leakage of primary coolant to the secondary side of the steam generator is assumed to continue at its initial rate, which is assumed to be the same rate as the leakage prior to the accident, until the pressures in the primary and secondary system are equalized. No mass transfer from the primary system to the secondary system through the steam generator tube leakage is assumed thereafter. Thus, in the case of coincident LOSP, activity is assumed released to the atmosphere from a steam dump through the relief valves for 98 s.

15.4.6.4.2 Assumptions for Conservative Analysis of Equilibrium Concentrations of Isotopes in the Secondary System

The following conservative assumptions were used in the analysis of the release of secondary system radioactivity to the environment in the event of a postulated rod ejection accident. A summary of parameters used in the analysis is given in table 15.4-31.

A. The primary to secondary leakage in the steam generators occurs when the reactor starts up; leakage remains constant at 1 gpm total for all steam generators.

B. The primary to secondary leakage is evenly distributed in the steam generators.

C. Primary coolant noble gas activity is associated with 0.25% melted fuel as given in table 11.1-2 and iodine activity is 0.5 μCi/gm DEI_{131}. The secondary side concentration of iodine is assumed to be 0.1 μCi/gm DEI_{131}.

D. No noble gas is dissolved or contained in the steam generator water; i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser offgas system.

15.4.6.4.3 Assumptions for Regulatory Guide 1.183 Analysis

Design criteria applied to ensure that fuel dispersal into the coolant will not occur include:

> "Fuel melting limited to less than the innermost 10 percent of the fuel pellet at the hotspot...."[38]

Even though centerline melting in a small fraction of the core is not expected, a conservative upper limit of fission product release from the core as a result of a rod ejection accident can be estimated. This limit would include the release of 100% of the noble gases and 50% of the iodines from that portion of the fuel which could experience centerline melting under the above criteria.

The upper limit of fission product release from the core for this very conservative case was determined using the following assumptions:

A. One hundred percent of the noble gases and iodines in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10% of the rods in the core[38]) is assumed released to the reactor coolant.

B. Fifty percent of the iodines and 100% of the noble gases in the fuel that melts is assumed released to the reactor coolant. This is a very conservative assumption since only centerline melting could occur for a maximum time period of 6 s.

C. The fraction of fuel melting was conservatively assumed to be 0.25% of the core as determined by the following method:

   1. A conservative upper limit of 50% of the rods experiencing clad damage may experience centerline melting (a total of 5% of the core).

   2. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10% of the rod volume will actually melt (equivalent to 0.5% of the core that could experience melting).

   3. A conservative maximum of 50% of the axial length of the rod will experience melting due to the power distribution (0.5 of the 0.5% of the core is equal to 0.25% of the core).

D. Instantaneous mixing in the containment of all activity released from the coolant.

E. It is assumed that aerosols released to the containment atmosphere immediately plate out on containment surfaces. No credit is taken for elemental iodine plateout.

F. No credit is assumed for removal of iodine in the containment due to containment sprays.

G. The containment leaks for the first 24 h at its design leak rate (as specified in the Technical Specifications) of 0.15%/day. Thereafter, the containment leak rate is 0.075%/day.

H. No credit is taken for the penetration room filters.

I. Primary and secondary system pressures are equalized after 2500 s, thus terminating primary to secondary leakage in the steam generators.

J. For the case of LOSP, 426,000 lb of steam are discharged from the secondary system through the relief valves the first 98 s following the accident. Steam dump is terminated after 98 s.

K. All releases to the atmosphere are assumed to be at ground level.

L. Dose models and isotopic data used in the analysis are presented in appendix 15B of this report.

15.4.6.4.4 Results

The TEDE doses at the site boundary, low-population zone, and control room for the rod ejection accident for the conservative Regulatory Guide 1.183 analyses are given in table 15.4-31. For the 2-h and 30-day periods after a postulated rod ejection accident, the doses at the site boundary and low-population zone, respectively, meet the NRC Regulatory Guide 1.183 acceptance criteria for rod ejection accidents. Control room doses are within the limits of 10 CFR 50.67.

The potential for uncovery of the steam generator tubes during the event has also been evaluated for impact on doses. The tube uncovery was assumed to exist until the primary to secondary leakage is terminated when the primary and secondary pressures are equalized. The tube leakage locations were assumed to all be near the top of the tube bundle and, thus, subject to the uncovery. With the primary to secondary leakage entering the vapor space, no credit was taken for mixing with the secondary coolant, nor was credit taken for a partition factor within the steam generator (i.e., the primary coolant was assumed to be released directly to the environment). The uncovery does not impact the release of noble gases to the environment; thus, the gamma and beta doses are not affected. The uncovery does result in an increase in the accident releases of iodine. The effect on the conservative thyroid dose at the site boundary is that this dose remains within the limits as defined in 10 CFR 100.

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TABLE 15.4-1 (SHEET 1 OF 4)

KEY LOCA PARAMETERS AND INITIAL TRANSIENT ASSUMPTIONS (UNITS 1 AND 2)

Parameter	Initial Transient	Uncertainty Treatment
1.0 Plant Physical Description		
a. Dimensions	Nominal	Sampled[a]
b. Flow resistance	Nominal	Sampled[a]
c. Pressurizer location	Opposite broken loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	17 x 17 V5 w/ZIRLO clad[d]	Bounded
f. SG tube plugging level	High (10%)	Bounded
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal - Based on core power of 2831 MWt (102% of uprated power)	0.0[b]
b. Hot rod average linear heat rate (PLHR)	Derived from TS limit of 2.5 and maximum baseload FQ	Sampled[a]
c. Hot rod average linear heat rate (HRFLUX)	Derived from TS $F_{\Delta H}$ = 1.7	Sampled[a]
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	Sampled[a]
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	Sampled[a]

TABLE 15.4-1 (SHEET 2 OF 4)

Parameter			Initial Transient	Uncertainty Treatment
	f.	Axial power distribution (PBOT, PMID)	Figure 15.4-16	Sampled[(a)]
	g.	Low power region relative power (PLOW)	0.2	Bounded
	h.	Cycle burnup	500 MWD//T	Sampled[(a)]
	i.	Prior operating history	Equilibrium decay heat	Bounded
	j.	Moderator Temperature Coefficient (MTC)	TS MAXIMUM (0)	Bounded
2.2	Fluid Conditions			
	a.	T_{avg}	High T_{avg} = 577.2°F	Bounded; Sampled[(a)]
	b.	Pressurizer Pressure	Nominal (2250 psia)	Sampled[(a)]
	c.	Loop flow	86,000 gal/min	Bounded
	d.	T_{UH}	T_{HOT}	0
	e.	Pressurizer level	Nominal (54.9%) at High T_{avg}	0
	f.	Accumulator temperature	Nominal (105°F)	Sampled[(a)]
	g.	Accumulator pressure	Nominal (640 psia)	Sampled[(a)]
	h.	Accumulator liquid volume	Nominal (980 ft^3)	Sampled[(a)]
	i.	Accumulator line resistance	Nominal	Sampled[(a)]
	j.	Accumulator boron	Minimum	Bounded

TABLE 15.4-1 (SHEET 3 OF 4)

Parameter		Initial Transient	Uncertainty Treatment
3.0	Accident Boundary Conditions		
a.	Break location	Cold leg	Bounded
b.	Break type	Guillotine (DECLG)	Sampled[a]
c.	Break size	Nominal (cold leg area)	Sampled[a]
d.	Offsite power	On (RCS pumps running)	Bounded
e.	Safety injection flow	Minimum	Bounded
f.	Safety injection temperature	Nominal (85°F)	Sampled[a]
g.	Safety injection delay	Max delay (12 s)[c]	Bounded
h.	Containment pressure	Bounded - Based on minimum containment pressure of 14.7 psia. The WCOBRA/TRAC pressure curve is based on the COCO containment pressure curve (figure 15.4-17) obtained using conditions supplied in table 15.4-6.	Bounded
i.	Single failure	ECCS: Loss of 1 SI train Containment pressure: all trains operational	Bounded
j.	Control rod drop time	No control rods	Bounded

TABLE 15.4-1 (SHEET 4 OF 4)

Parameter	Initial Transient	Uncertainty Treatment
4.0 Model Parameters		
a. Critical Flow	Nominal (C_D = 1.0)	Sampled[a]
b. Resistance uncertainties in broken loop	Nominal (as coded)	Sampled[a]
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	Sampled[a]
d. Core heat transfer	Nominal (as coded)	Sampled[a]
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Noncondensible gases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	Sampled[a]

a. Sampling distribution defined in table 5.2.1 of reference 58.
b. This is used in the methodology to allow for "mini-uprate" and will bound any potential mini-uprate; i.e., any calorimetric uprate in which the increase in core power and reduced uncertainty sum to 2%.
c. SI injection actuation delay is calculated to bound RHR miniflow effects.
d. All results are for ZIRLO cladding; use of Optimized ZIRLO cladding was qualitatively evaluated as acceptable.

TABLE 15.4-2

RESULTS FROM CONFIRMATORY STUDIES FOR FARLEY UNITS 1 AND 2

Study Description	Parameter Varied	Value	PCT (°F)
Initial Transient	N/A	N/A	1680
Confirmatory Runs	Offsite Power	Yes	1667
	Power in Outer Assemblies (PLOW)	0.8	1608
	Steam Generator Tube Plugging (SGTP)	0%	1631
	Average Temperature (T_{avg})	567.2°F	1678
Final Reference Transient	Offsite Power	No	1728
	PLOW	0.2	
	SGTP	10%	
	(T_{avg})	577.2°F	

TABLE 15.4-3

BEST ESTIMATE LARGE BREAK LOCA RESULTS

	Result	Criterion
95/95 PCT (°F)	1836[a]	< 2200
95/95 LMO (%)	2.9	< 17.0
95/95 CWO (%)	0.22	< 1.00
Coolable Geometry	Core remains coolable	
Long Term Cooling	Core remains cool in long term	
Additional Assessment Quarterly RHR Pump Test[a]	+ 25°F	N/A
Reactor Coolant Pumps Input Error[b]	+18°F	N/A
Evaluation of Fuel Pellet Thermal Conductivity Degradation and Peaking Factor Burndown[b]	+150°F	N/A
Revised Heat Transfer Multiplier Distributions[b]	- 40°F	N/A
Changes to Grid Blockage Ratio and Porosity[b]	+24°F	N/A
Error in Burst Strain Application [b]	+21°F	N/A
Licensing Basis PCT$^{95\%}$ + Margin Allocations (°F)	2034[c]	< 2200

PCT – Peak Clad Temperature
LMO – Local Maximum Oxidation
CWO – Core-Wide Oxidation

a. An evaluation was performed to assess the ECCS performance during the quarterly residual heat removal (RHR) surveillance testing. The assessment concluded that a 25°F PCT penalty applies to the 95/95 PCT during the testing period. The penalty will be tracked as a temporary PCT penalty and will apply only during the period of the test. See paragraph 15.4.1.5.3 for further explanation on the RHR test.
b. See paragraph 15.4.1.5.3 for further information.
c. Includes temporary PCT penalty, per footnote a.

TABLE 15.4-4 (SHEET 1 OF 2)

PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE-BREAK LOCA ANALYSIS (UNITS 1 and 2)

			Parameter	Operating Range
1.0	Plant Physical Description			
	a.	Dimensions		No in board assembly grid deformation during LOCA + SSE
	b.	Flow resistance		N/A
	c.	Pressurizer location		N/A
	d.	Hot assembly location		Anywhere in core
	e.	Hot assembly type		Fresh 17 x 17 V5, w/ZIRLO or Optimized ZIRLO cladding
	f.	SG tube plugging level		$\leq$ 10%
	g.	Fuel assembly type		VANTAGE 5, ZIRLO, or Optimized ZIRLO cladding, 1.5 x IFBA
2.0	Plant Initial Operating Conditions			
	2.1	Reactor Power		
		a.	Core avg linear heat rate (AFLUX)	Based on core power $\leq$ 102% of 2775 MWt
		b.	Peak linear heat rate (PLHR)	$F_Q \leq 2.5$
		c.	Hot rod avg linear heat rate (HRFLUX)	$F_{\Delta H} \leq 1.7$
		d.	Hot assembly average heat rate (HAFLUX)	$\bar{P}_{HA} \leq 1.7/1.04$ [(a)]
		e.	Hot assembly peak heat rate (HAPHR)	$F_{Q,HA} \leq 2.5/1.04$
		f.	Axial power distribution (PBOT, PMID)	Figure 15.4-16
		g.	Lower power region relative power (PLOW)	$0.2 \leq PLOW \leq 0.8$
		h.	Hot rod burnup	$\leq$ 75,000 MWD/MTU, lead rod[(b)]
		i.	Prior operating history	All normal operating histories
		j	MTC	$\leq$ 0 at HFP
	2.2	Fluid Conditions		
		a.	T_{avg}	$567.2 \pm 6°F \leq T_{avg} \leq 577.2 \pm 6°F$
		b.	Pressurizer pressure	P_{RCS} = 2250 psia $\pm$ 50 psi
		c.	Loop flow	$\geq$ 86,000 gpm/loop
		d.	T_{UH}	Current upper internals

TABLE 15.4-4 (SHEET 2 OF 2)

Parameter			Operating Range
	e.	Pressurizer level	Normal level, automatic control
	f.	Accumulator temperature	90°F ≤ T_{ACC} ≤ 120°F
	g.	Accumulator pressure	600 psia ≤ P_{ACC} ≤ 680 psia
	h.	Accumulator volume (tank only)	965 ft^3 ≤ V_{ACC} ≤ 995 ft^3
	i.	Accumulator fL/D	Current line configuration
	j.	Minimum accumulator boron	≥ 2100 ppm
3.0		Accident Boundary Conditions	
	a.	Break location	N/A
	b.	Break type	N/A
	c.	Break size	N/A
	d.	Offsite power	On or Off
	e.	Safety injection flow	≥ values used in reference case
	f.	Safety injection temperature	70°F[(3)] ≤ SI Temp ≤ 100°F
	g.	Safety injection delay	≤ 12 s (with offsite power) ≤ 27 s (without offsite power)
	h.	Containment pressure	Bounded - Based on minimum containment pressure of 14.7 psia. The WCOBRA/TRAC pressure curve is based on the COCO containment pressure curve (figure 15-4-17) obtained using conditions supplied in table 15.4-6.
	i.	Single failure	Loss of one ECCS train
	j.	Control rod drop time	N/A

1. Note that this $\overline{P}_{HA}$ limit is a maximum value. For purposes of core design calculations or in core measurements, the maximum value must be reduced by an additional 4%, yielding a value of $\overline{P}_{HA} \leq 1.7 / 1.08 = 1.574$.
2. Based on generic BE LBLOCA studies.
3. 70°F is a statistical lower limit for the SI temperature based on actual plant data. Temperatures as low as the TS lower limit of 35°F are acceptable.

TABLE 15.4-5 (SHEET 1 OF 4)

Accident	Event	Time (s)
Major steam pipe rupture	Steam line ruptures	0.0
With offsite power		
	Borated water from the RWST reaches the core	66.3
	Criticality attained	51.6
	Accumulators actuate	148.2
Without offsite power	Steamline ruptures	0.0
	Borated water from the RWST reaches the core	86.7
	Criticality attained	108.3
Reactor coolant pump shaft seizure (locked rotor/shaft break)		
Peak RCS Pressure Case	Rotor on one pump locks or the shaft breaks	0.0
	Low flow reactor trip setpoint reached	0.03
	Rods begin to drop	1.03
	Remaining pumps lose power and begin coasting down	3.03
	Maximum RCS pressure occurs	3.4
Peak Clad Temperature Case	Rotor on one pump locks or the shaft breaks	0.0
	Low flow reactor trip setpoint reached	0.03
	Rods begin to drop	1.03
	Remaining pumps lose power and begin coasting down	3.03

TABLE 15.4-5 (SHEET 2 OF 4)

Accident	Event	Time (s)
	Maximum clad temperature occurs	3.8
Rupture of main feedwater pipe		
CASE A	Feedline rupture occurs	20.0
	Affected steam generator liquid discharge, low-low level reactor trip setpoint reached	27.8
	Reactor trip occurs (rods fall)	29.8
	Steam line isolation occurs	64.6
	Auxiliary feedwater flow initiated at 350 gal/min	627.8
	Peak Pressurizer volume	863.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1850.0
CASE B	Feedline rupture occurs	20.0
	Affected steam generator liquid discharge, low-low level reactor trip setpoint reached	27.8
	Reactor trip occurs (rods fall)	29.8
	Steam line isolation occurs	65.0
	Auxiliary feedwater flow initiated at 150 gal/min	87.8
	Peak pressurizer volume	1736.0
	Auxiliary feedwater flow increased to 350 gal/min	1827.8
	Core decay heat decreases to auxiliary feedwater heat removal capacity	2120.0

TABLE 15.4-5 (SHEET 3 OF 4)

Accident	Event	Time (s)
RCCA ejection accident		
BOL, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.30
	Peak nuclear power occurs	0.36
	Rods begin to fall into core	0.80
	Peak clad average temperature occurs	2.54
	Peak heat flux occurs	2.55
	Peak fuel average temperature occurs	2.80
BOL, full power	Initiation of rod ejection	0.0
	Power range high neutron flux high setpoint reached	0.05
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.44
	Peak clad average temperature occurs	2.56
	Peak heat flux occurs	2.57
EOL, zero power	Initiation of rod ejection	0.0
	Power range high neutron flux low setpoint reached	0.17
	Peak nuclear power occurs	0.20
	Rods begin to fall into core	0.67
	Peak clad average temperature occurs	1.35

TABLE 15.4-5 (SHEET 4 OF 4)

Accident	Event	Time (s)
	Peak heat flux occurs	1.35
	Peak fuel average temperature occurs	1.66
EOL, full power	Initiation of rod ejection	0.0
	Power range high neutron flux high setpoint reached	0.05
	Peak nuclear power occurs	0.13
	Rods begin to fall into core	0.55
	Peak fuel average temperature occurs	2.18
	Peak clad average temperature occurs	2.30
	Peak heat flux occurs	2.31

TABLE 15.4-6 (SHEET 1 OF 2)

LARGE BREAK LOCA CONTAINMENT DATA USED FOR COCO CALCULATION OF CONTAINMENT PRESSURE

Net Free Volume	2,150,000 ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	90.0°F
RWST temperature	35.0°F
Service water temperature	40.0°F
Temperature outside containment	20.0°F
Initial spray temperature	35.0°F
Spray System	
Runout flow for a spray pump	3400 gal/min
Number of spray pumps operating	2
Post-accident spray system initiation delay	18 s
Maximum spray system flow	6800 gal/min(1)
Containment Fan Coolers	
Post-accident initiation fan coolers	0.0 s(a)
Number of fan coolers operating	4

TABLE 15.4-6 (SHEET 2 OF 2)

Structural Heat Sinks

WALL	T_{air} (°F)	Area (ft^3)	Height (ft)	T_{init} (°F)	Nominal[b] Thickness (in)
1. Containment wall and dome	20 (Unit 2)	75,000	10	90	0.25 Carbon steel / 45 Concrete
2. Containment penetrations, plates, and liner stiffeners	20 (Unit 2)	5,170	10	90	0.51 Carbon steel /45 Concrete
3. Unlined concrete	90	82,733	10	90	9 Concrete
4. Galvanized carbon steel (excluding cable trays)	90	91,985	10	90	0.003 Zinc /.08 Steel
5. Thin painted carbon steel (< 0.5 in.)	90	137,102	10	90	0.18 Steel
6. Painted steel (< 1.0 in.)	90	34,982	10	90	0.59 Steel
7. Painted steel (< 2.0 in.)	90	12,674	10	90	1.35 Steel
8. Thick painted steel (≥ 2.0 in.)	90	5,030	10	90	3.59 Steel
9. Floor	50	13,275	10	90	108 Concrete
10. Refueling pool liner (Stainless steel)	90	7,900	10	90	0.25 Stainless steel / 18 Concrete
11. Unpainted stainless steel	90	14,567	10	90	0.12 Stainless steel
12. Galvanized steel	90	31,916	10	90	0.003 Zinc / .05 steel
13. Uninsulated stainless steel pipe	90	1,535	10	90	0.22 stainless steel

a. Bounds delay with and without Loop.
b. The nominal thicknesses are increased by 15% in the ASTRUM containment pressure analysis.

TABLE 15.4-7

CORROSION RATE USED IN THE POST-ACCIDENT CONTAINMENT HYDROGEN GENERATION ANALYSIS

Temperature (°F)	Zinc Corrosion Rate (lb-mole$_{ZN}$/ft^2-hr)	Zinc Corrosion Rate (mg/dm^2-hr)
266	1.00×10^{-4}	319
240	1.99×10^{-5}	63.6
234	1.59×10^{-5}	50.7
205	2.78×10^{-6}	8.9
190	1.39×10^{-6}	4.43
175	5.13×10^{-7}	1.64
160	2.21×10^{-7}	0.71
147	12.4×10^{-8}	0.395

TABLE 15.4-8

POSTACCIDENT CONTAINMENT TEMPERATURE TRANSIENT USED IN THE CALCULATION OF ALUMINUM AND ZINC CORROSION

Time Interval (s)	Temperature (°F)
0 - 2000	266
2000 - 4000	240
4000 - 6000	234
6000 - 40000	205
40000 - 86400	190
86400 - 172800	175
172800 - 259200	160
>259200	

TABLE 15.4-8A

This table has been deleted.

TABLE 15.4-9

CORE FISSION PRODUCT ENERGY AFTER 830 FULL-POWER DAYS

	Core Fission Product Energy[a, b]	
Time After Reactor Trip (day)	Energy Release Rate (W/MWt x 10^{-3})	Integrated Energy Release (W-day/MWt x 10^{-4})
1	3.887	0.574
5	2.595	1.777
10	2.211	2.967
20	1.760	4.934
30	1.475	6.541
40	1.291	7.919
50	1.163	9.143
60	1.068	10.259
70	0.992	11.289
80	0.926	12.249
90	0.867	13.139
100	0.814	13.979

a. Assumes release of 50-percent core halogens and 1-percent other fission products, including 100-percent noble gases. Values are for total (β and γ) energy.

b. For power uprate, a 5% increase in these values was assumed. Table values are pre-uprate and do not include the 5% increase.

TABLE 15.4-10

FISSION PRODUCT DECAY DEPOSITION IN SUMP SOLUTION

	50-Percent Halogens		1-Percent Other Fission Products		Total[(a)]	
Time After Reactor Trip (day)	Energy Release Rate (W/MWt)	Integrated Energy Release (W-day/MW x 10^{-2})	Energy Release Rate (W/MWt x 10^{-1})	Integrated Energy Release (W-day/MWt x 10^{-2})	Energy Release Rate (W/MWt x 10^{-1})	Integrated Energy Release (W-day/MWt x 10^{-3})
1	145.00	4.27	3.78	0.536	18.28	0.481
3	49.40	5.88	2.90	1.18	7.85	0.707
5	31.00	6.65	2.59	1.78	5.69	0.338
10	18.20	7.82	2.22	2.92	4.03	1.07
20	7.63	9.03	1.77	4.89	2.53	1.39
30	3.22	9.54	1.49	6.51	1.31	1.61
40	1.36	9.76	1.30	7.90	1.44	1.77
60	0.241	9.89	1.08	10.30	1.10	2.02
80	0.043	9.91	0.935	12.30	0.940	2.22
100	0.008	9.92	0.822	14.00	0.823	2.39

a. For power uprate, a 5% increase in these values was assumed. Table values are pre-uprate and do not include the 5% increase.

TABLE 15.4-11

This table has been deleted.

TABLE 15.4-12

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP Beginning	HFP Beginning	HZP End	HFP End
Power level (%)[a]	0	102	0	102
Ejected rod worth (% Δρ)	0.75	0.20	0.945	0.21
Delayed neutron fraction (%)	0.54	0.54	0.44	0.44
Doppler feedback reactivity weighting	2.07	1.30	3.19	1.30
Trip reactivity (%Δρ)	1.52	4.8	1.715	4.8
F_Q before rod ejection	-	2.50	-	2.50
F_Q after rod ejection	13.0	6.0	22.0	7.0
Number of operational pumps	2	3	2	3
Maximum fuel pellet average temperature at the hotspot (°F)	3316	4023	3833	3878
Maximum fuel center temperature at the hotspot (°F)	3935	> 4900	4319	> 4800
Maximum fuel stored energy at the hotspot (cal/g)	139.4	175.5	165.7	168.0
Fuel melt (%)	0	< 10	0	< 10

a. Power level is percent of 2775 MWt.

TABLE 15.4-13

FARLEY HYDROGEN GENERATION NRC BASIS (5-percent Zr-Water Reaction)[b]

Total Hydrogen generated by Zr-Water reaction = 1.541E + 04 sf[3]

TIME (DAYS)	SUMP RADIOLYSIS (CU.FT.)	CORE RADIOLYSIS (CU.FT.)	AL & ZN CORROSION (CU.FT.)	HYDROGEN RECOMBINER (CU.FT.)	GRAND TOTAL[a] (CU.FT.)	CONTAINMENT CONCENTRATION (% VOL)
0						0
1	4956	2793	23748	0	47937	2.71
10	11025	15330	34326	<32631>	44489	2.52
20	14385	25410	39905	<62344>	33796	1.93
30	16485	33705	45484	<85426>	26688	1.53
40	18165	40845	51064	<104201>	22313	1.28
50	19530	47145	56643	<120262>	19495	1.12
60	20790	52815	62222	<134575>	17692	1.02
70	21840	58170	67801	<147752>	16500	0.95
80	22785	63105	73381	<160126>	15585	0.90
90	23730	67620	78960	<171879>	14871	0.86
100	24675	72030	84539	<183211>	14474	0.83

a. Includes hydrogen generated by the zirconium-water reaction and initial RCS/pressurizer hydrogen.

b. The NRC hydrogen generation model results were used in the development of the design criteria for the containment combustible gas control systems.

TABLE 15.4-14 (SHEET 1 OF 2)

PARAMETERS USED IN THE LOCA ANALYSIS

Parameter	Value
Core power level (2% uncertainty and a rated power of 2775 MWt)	2831 MWt
Fuel release fractions	Per RG 1.183
Fuel release timing	Per RG 1.183
Reactor coolant system mass	440,900 lb mass
RCS concentration	Based on 1% failed fuel and 0.5 µci per gram (DEI) per TS 3.4.16
Containment Mini-Purge Parameters	
Minimum containment free volume chemical	2.03E =06 ft^3
Form of iodine released	4.85% Elemental 95% Particulate 0.15% Organic
Containment purge filtration	None
Containment purge flow rate	2,850 ft^3/min
Containment purge isolation	30 s or less
Removal by wall deposition or containment sprays	None
Containment Leakage Parameters	
Containment volume	2.03E+06 ft^3 Sprayed: 1,668,660 ft^3 Unsprayed: 361,340 ft^3
Chemical form of iodine released	4.85% Elemental 95% Particulate 0.15% Organic
Containment spray removal coefficient	Element iodine: 13.7 per hour Aerosol: 5.45 per hour during injection mode and 5.13 per hour during recirculation mode Organic: None
Natural deposition of aerosols	0.1 per hour after sprays are terminated
Containment spray	Initiation time: 90 s Termination time: 8 h

TABLE 15.4-14 (SHEET 2 OF 2)

PARAMETERS USED IN THE LOCA ANALYSIS

Containment spray flow rate	2,480 gal/min in injection mode 2,290 gal/min in recirculation mode
Long term sump water pH	≥ 7.0
Maximum allowable DF for fission product removal	Elemental iodine: 200
Containment leak rate	0 to 24 h: 0.15% weight per day 1 to 30 days: 0.75% weight per day
Engineered Safety Features System Leakage Parameters	
Sump volume	49,200 ft^3
Minimum time after LOCA when recirculation is initiated	20 min
Leakage duration	30 days
Maximum ECCS fluid temperature after initiation of recirculation	265°F
ECCS leak rate	40,000 cubic centimeters /h
ECCS leakage iodine flashing fraction	10%
Chemical form of iodine released	97% Elemental 3% Organic
Refueling Water Storage Tank (RWST) Back Leakage Parameters	
Minimum time after LOCA when recirculation is initiated	20 min
RWST volume at transfer to recirculation mode	29.002 gal
RWST capacity	505,562 gal
RWST leakage inflow rate	2 gal/min
RWST leakage iodine flashing fractions	0% to 13.9%
Atmospheric Dispersion Factors	
Offsite	Table 15B-2
Control room	Table 15B-3

TABLE 15.4-14a

PARAMETERS USED IN THE SMALL BREAK LOCA ANALYSIS

Core thermal power	2831 MWt (2775 x 1.02)
Core thermal power (control room dose analysis)	2831 MWt (2775 x 1.02)
Containment free volume	2.03 x 10^6 ft^3
Volume fractions	
Sprayed	0.822
Unsprayed	0.178
Mixing rate between sprayed and unsprayed containment volumes	12,000 ft^3/min
Core fission product inventories	See table 15.1-4.
Activity released to containment	Gas gap activity
Plateout of elemental iodine activity released to containment	2.7 h^{-1} (DF < 100) 0.27 h^{-1}(100 $\leq$ DF < 1000) 0.0 h^{-1} (DF $\geq$ 1000)
Form of iodine activity in containment available for release	
Elemental	95.5%
Organic	2.0%
Particulate	2.5%
Spray removal constants	0.0 h^{-1}
Time to reach decontamination factor	
Elemental	24 min
Methyl	N/A
Particulate	8 h
Containment leak rate	
0-24 h	0.15%/day
1-30 days	0.075%/day
Atmospheric dilution estimates	See table 15.B-2a.

TABLE 15.4-15

OFFSITE DOSES FROM LOCA

	Site Boundary (rem)	Low Population Zone (rem)
Total	13.2	6.0
Standard Review Plan 15.0.1 and RG 1.183 Limit (rem)	25	25

TABLE 15.4-16

PARAMETERS USED IN ANALYSIS OF POST-LOCA CONTROL ROOM DOSES

CREFS Initiation Time		Pressurization: 60 s
Filtered pressurization rate (ft^3/min)		375
Filtered recirculation rate (ft^3/min)		2700
Unfiltered inleakage rate (ft^3/min)		10 (ingress/egress) 325 (control room envelope)
Filter efficiencies (%)		
Pressurization air	(all forms of iodine)	98.5[(a)]
Recirculation air	(elemental/organic iodine)	94.5[(a)]
	(particulate iodine)	98.5[(a)]
Volume (ft^3)		114,000
Operator breathing rate (m^3/s)		3.5×10^{-4}
Percent of time operator is in control room following LOCA		
0-1 day		100
1-4 days		60
4-30 days		40

a. Filter efficiencies have been reduced by 0.5% for all forms of iodine to account for bypass leakage.

TABLE 15.4-17

CONTROL ROOM DOSES FOLLOWING A LOCA

	TEDE Dose Results (rem)
Total	4.9*
10 CFR 50.67 Limit	5

* Includes 0.2 rem TEDE for control room ingress/egress.

TABLE 15.4-18

PARAMETERS USED IN THE ANALYSES OF HYDROGEN PURGING FOLLOWING A LOCA

Parameter	Realistic Analysis	Regulatory Guide 1.7 Analysis
Model used to determine hydrogen generation	(a)	Regulatory Guide 1.7
Hydrogen concentration limit in containment (vol/%)	-	4.0
Time after LOCA at which hydrogen concentration limit is reached and containment pressurization is initiated (days)	-	5
Containment pressurization (psig)	-	2.0
Time after LOCA at which hydrogen concentration limit is reached after containment pressurization (days)	-	7
Containment purge rate (sf^3/min)	-	54
Containment purge filter efficiencies[(b)]		
Elemental iodine (%)	-	89.5
Methyl iodine (%)	-	30.0
Particulate iodine (%)	-	98.5
Activity released to containment available for release		
Noble (%)	-	100 of core inventory
Iodines (%)	-	50 of core inventory as reduced by decay, plateout, and spray
Meteorology	-	Accident (see appendix 15B)

a. No activity release due to purging since redundant electric recombiners would operate and purging would not be required.

b. Filter efficiencies have been reduced by 0.5% to account for bypass leakage.

TABLE 15.4-19

OFFSITE DOSES FROM CONTAINMENT PURGING TO CONTROL HYDROGEN FOLLOWING A LOCA

	Thyroid Dose (rem)
	Low-Population Zone (0-30 day) (3219 m)
Realistic analysis	0.0
Regulatory Guide 1.7 analysis	86.0

	Whole Body Dose (rem)
	Low-Population Zone (0-30 day) (3219 m)
Realistic analysis	0.0
Regulatory Guide 1.7 analysis	0.4

TABLE 15.4-20

Deleted

TABLE 15.4-21

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

This table has been deleted.

TABLE 15.4-22

SUMMARY OF IMPORTANT PARAMETERS FOR THE STEAM BREAK ANALYSIS REPORTED IN SUBSECTION 15.4.2

This table has been deleted.

TABLE 15.4-23 (SHEET 1 OF 2)

PARAMETERS USED IN STEAM LINE BREAK ANALYSES

	Conservative Analysis
Core thermal power (MWt)	2831
Steam generator tube leak rate prior to accident and initial 24 h following accident (gal/min)	1 gal/min[b] (0.65 gal/min to intact SG/0.35 gal/min to faulted SG)
Offsite power	Lost
Fuel defects (%)	1[a]
Iodine partition factor for initial steam release from faulted steam generator	1.0
Iodine partition factor in intact steam generators prior to and during accident	100
Alkali metals partition factor in intact steam generators	1000
Steam release from faulted steam generator (lbm)	483,000 (0-24 h)
Duration of cooldown by secondary system after accident (time to terminate RCS leak and steam releases (h)	24
Steam release from two intact steam generators (lbm))	348,000 (0-2) 774,000 (2-8) 1,040,000 (8-24)
Feedwater flow to two intact steam generators (lbm)	481,000 (0-2) 783,000 (2-8) 1,040,000 (8-24)
Meteorology	Accident (see Appendix 15B)

TABLE 15.4-23 (SHEET 2 OF 2)

Offsite Dose (rem)[c]	Pre-Accident Iodine Spike	Accident-Initiated Iodine Spike
Site Boundary	0.9	1.0
LPZ	0.4	0.5
Regulatory Guide 1.183 Offsite Limit	25	2.5

a. A pre-existing iodine spike of 30 μCi/gm or an accident-initiated iodine spike 500 times the normal appearance rate based on an initial RCS DEI-131 concentration of 0.5 μCi/gm and 145 gal/min letdown is assumed.

b. Mass released is in addition to steam releases shown.

TABLE 15.4-23a

PARAMETERS USED IN ANALYSIS OF MAIN STEAM LINE BREAK CONTROL ROOM DOSES

CREFS initiation time	Safety injection signal generated; 27 s Pressurization < 60 s
Filtered pressurization rate (ft^3/ min)	375
Filtered recirculation rate (ft^3/min)	2700
Unfiltered inleakage rate (ft^3/min)	310 (includes 10 ft^3/min for ingress/egress)
Filter efficiencies (%)	
Pressurization air (all forms of iodine)	98.5
Recirculation air (elemental/organic iodine)	94.5
(particulate iodine)	98.5
Volume (ft^3)	114,000
Operator breathing rate (m^3/s)	3.5×10^{-4}
Meteorology	Accident (see appendix 15B)
Control room dose	10 CFR 50.67 Limit (rem)
Pre-iodine spike	0.2
Concurrent iodine spike	0.5
10 CFR 50.67 Limit	5

TABLE 15.4-24 (SHEET 1 OF 2)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

Parameter	Value	
Core thermal power (MWt)	2831	
Steam generator tube leak rate prior to and during accident	1 gal/min (0.65 gal/min to intact SG/0.35 gal/min to faulted SG)	
Offsite power	Lost	
Fuel defects (%)	1[(a)]	
Alkali metal partition factor flow out of the steam generators during accident	1000	
Iodine partition factors for secondary sidewater in steam generators during accident	100	
Iodine and alkali partition factor for primary side water (RCS flashing) during accident	Non-Flash	Flashing
0-324 s (before trip)	1.27	4.76
324 s - 1800s (after trip)	1.18	6.67
Time to reactor trip (s)	324[(b)]	
Time to isolate defective steam generator (min)	30	
Duration of plant cooldown by secondary system after accident (h)	8	
Steam release from ruptured steam generator	367,000 lb (0-324 s) 79, 000 lb (324 s -30 min)	
Steam release from two intact steam generators	734,000 lb (0-324 s) 422,000 lb (324 s -2 h) 934,000 lb (2-8hr)	
Feedwater flow to two intact steam generators	327,000 lb (0-2 h) 981,000 lb (2-8 h)	

TABLE 15.4-24 (SHEET 2 OF 2)

Reactor coolant released to the ruptured steam generator		26,600 lbm (0-324 s) 136,400 lbm (>324 s)
Meteorology		Accident (see appendix 15B)
Offsite Dose (rem)[c]	Pre-Accident Iodine Spike	Accident-Initiated Iodine Spike
Site Boundary	2.4	0.8
LPZ	0.9	0.3
Regulatory Guide 1.183 Offsite Limit	25	2.5

a. Pre-accident iodine spike 60 times the Technical Specification limit or accident-initiated iodine spike 335 times the normal appearance rate based on an initial RCS DEI-131 concentration of 0.5 μCi/gm and 145 gal/min letdown.

b. Steam release prior to reactor trip is to the condenser; however, no credit is taken for partitioning or cleanup in the condenser.

TABLE 15.4-24a

PARAMETERS USED IN ANALYSIS OF STEAM GENERATOR TUBE RUPTURE ACCIDENT CONTROL ROOM DOSES

CREFS initiation time	Safety injection signal generated; 27 s Pressurization < 60 s
Filtered pressurization rate (ft^3/min)	375
Filtered recirculation rate (ft^3/min)	2700
Unfiltered inleakage rate (ft^3/min)	310 (includes 10 ft^3/min for ingress/egress)
Filter efficiencies (%)	
Pressurization air (all forms of iodine)	98.5
Recirculation air (elemental/organic iodine)	94.5
(particulate iodine)	98.5
Volume (ft^3)	114,000
Operator breathing rate (m^3/s)	3.5×10^{-4}
Meteorology	Accident (see appendix 15B)
Control room dose	10 CFR 50.67 Limit (rem)
Pre-iodine spike	0.2
Concurrent iodine spike	0.5
10 CFR 50.67 Limit	5

TABLE 15.4-25

SUMMARY OF RESULTS FOR THE LOCKED ROTOR TRANSIENT

Criteria	3 Loops Initially Operating, One Locked Rotor
Maximum RCS pressure	2727 psia
Maximum clad temperature at core hotspot	2165°F
Zr-H_2O reaction at core hotspot	1.0 wt%
Fraction of rods in DNB	< 20 percent
Site boundary dose (0 - 2 h)	1.0 rem
Low population zone dose (0 - 8 hour)[(a)]	0.7 rem
Regulatory Guide 1.183 Offsite Limit	2.5 rem
Control room dose	0.4 rem
10 CFR 50.67 Limit	5 rem

TABLE 15.4-25A

PARAMETERS USED IN RCP LOCKED ROTOR ANALYSES

Core thermal power (MWt)	2831
Offsite Power	Lost
Steam generator tube leak rate prior to and during accident (gal/min)	1
Activity released to RCS	20% of gap inventory
Radial peaking factor	1.7
Secondary side iodine activity	0.1 μCi/gm DEI_{131}
Iodine partition factor in steam generators	100
Duration of plant cooldown by secondary system after accident (h)	8
Steam release from three steam generators (lb)	564,000 (0-2 h) 917,000 (2-8 h)
Feedwater flow to three steam generators (lb)	763,000 (0-2 h) 929,000 (2-8 h)
Manual CREFS initiation	20 min
Filtered pressurization rate (ft^3/min)	375
Filtered recirculation rate (ft^3/min)	2700
Unfiltered inleakage rate (ft^3/min)	325 (includes 10 ft^3/min for ingress/egress)
Filter efficiencies (%)	
Pressurization air (all forms of iodine)	98.5
Recirculation air (elemental/organic iodine)	94.5
(particulate iodine)	98.5
Volume (ft^3)	114,000
Operator breathing rate (m^3/s)	3.5×10^{-4}
Meteorology	Accident (see appendix 15B)

TABLE 15.4-26

RELEASE ACTIVITIES FOR FUEL HANDLING ACCIDENT

Group	Isotope	100-h Released Activity (Curies)
Noble Gases	Kr-85	8.97E+02
	Xe-131m	4.52E+02
	Xe-133	5.68E+04
	Xe-133m	1.12E+03
	Xe-135	1.12E+02
Halogens	I-131	2.41E+02
	I-132	1.28E+02
	I-133	1.59E+01
	I-135	1.14E-02

TABLE 15.4-27 (SHEET 1 OF 3)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSIS

	Accident in Spent-Fuel Pool (Auxiliary Building)	Accident in Refueling Canal (Containment)[c]
Core thermal power	2831 MWt	2831 MWt
Time between plant shutdown and accident	100 h	100 h
Minimum water depth between tops of damaged fuel rods and water surface	23 ft	23 ft
Damage to fuel assembly	All rods ruptured	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged	Highest powered fuel assembly in core region discharged
Activity release from assembly	Gap activity in ruptured rods	Gap activity in ruptured rods
Radial peaking factor	1.7	1.7
Decontamination factor in water		
Elemental iodine (99.85%)	500	500
Organic iodine (0.15%)	1	1
Noble gases	1	1
Amount of mixing in building	No mixing credited	1.0×10^6 ft^3

TABLE 15.4-27 (SHEET 2 OF 3)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSIS

	Accident in Spent-Fuel Pool (Auxiliary Building)	Accident in Refueling Canal (Containment) [c]
Exhaust flowrate	5,000 ft^3/min	55,000 ft^3/min (containment hatch) 25,000 ft^3/min (personnel airlock)
Isolation time	N/A	N/A
Iodine filtration system	Penetration room filtration system	Containment purge system (not credited)
Filter efficiency (all species)	89.5[a]	N/A
Atmospheric dilution factors	Accident (see table 15B-2)	Accident (see table 15B-2)

Control Room Parameters

Normal HVAC unfiltered intake (ft^3/min)	2,340
Unpressurized unfiltered infiltration (ft^3/min)	600
Filtered pressurization makeup rate (ft^3/min)	375
Pressurized unfiltered inleakage (ft^3/min)	325
Filtered recirculation rate (ft^3/min)	2,700

TABLE 15.4-27 (SHEET 3 OF 3)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSIS

Control Room Parameters

Unfiltered ingress / egress rate (ft^3/min)	10
Filter efficiencies (all forms of iodine) %[(a)]	
Pressurization air	98.5
Recirculation air	94.5
Isolation time (s)	60
Pressurization Sys Manual Start Time (min)	21
Volume	114,000
Operator breathing rate (m^3/s)	3.5×10^{-4}
Percent of time operator is in control room following fuel handling accident (%)	
0 – 8 h	100

a. Filter efficiency has been reduced by 0.5% to account for bypass leakage.

b. Deleted

c. During the postulated FHA, the containment equipment hatch and personnel air locks are open.

TABLE 15.4-28

DELETED

TABLE 15.4-29

DOSES FROM FUEL HANDLING ACCIDENT IN SPENT FUEL POOL

100 HR DECAY, NEW FUEL ACCESS ROOF HATCH CLOSED

Site Boundary Dose (rem)	Low-Population Zone Dose (rem)
0.5	0.2

Control Room Dose (rem)
0.1

676 HR DECAY, NEW FUEL ACCESS ROOF HATCH OPEN

Site Boundary Dose (rem)		Low-Population Zone Dose (rem)	
Thyroid	Whole Body	Thyroid	Whole Body
25.7	< 0.1	9.5	< 0.1

Control Room Thyroid Dose (rem)
12.3

TABLE 15.4-30

DOSES FROM FUEL HANDLING ACCIDENT INSIDE CONTAINMENT WITH EQUIPMENT HATCH AND PERSONNEL AIRLOCKS OPEN

Location	Dose type	Dose (rem)	Accident Limit (rem)
Site Boundary	TEDE	2.4	6.3
Low – Population Zone (LPZ)	TEDE	0.9	6.3
Control Room [a]	TEDE	2.3	5

a. The control room doses comply with the acceptance criteria of 10 CFR 50.67.

TABLE 15.4-31 (SHEET 1 OF 4)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSES

	Conservative Analysis
Core thermal power (MWt)	2831
Containment free volume (ft^3)	2.03×10^6
Containment leak rates	
0-24 h	0.15 weight %/day
> 24 h	0.0075 weight %/day
Steam generator tube leak rate prior to and during steam (gal/min total)	1
Failed fuel	10% of fuel rods incore
Activity released to reactor coolant from failed fuel and available for release	
Iodine isotopes and noble gases (%)	10.0 of gap inventory
Other halogens (%)	5 of gap inventory
Alkali metals (%)	12
Melted fuel (%)	0.25 of core inventory
Activity released to reactor coolant from melted fuel and available for release	
Noble gases (%)	0.25 of core inventory
Iodines (%)	0.06 of core inventory

TABLE 15.4-31 (SHEET 2 OF 4)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSES

	Conservative Analysis
Radial peaking factor	1.7
Iodine partition factor in steam generators prior to and during accident	100
Alkali metal partition factor in steam generators prior to and during accident	1000
Plateout of aerosol activity released to containment (h)	2.74E-2

TABLE 15.4-31 (SHEET 3 OF 4)

	Conservative Analysis
Form of iodine activity in containment available for release	
Elemental iodine (%)	4.85
Organic iodine (%)	0.154
Aerosols (%)	95
Containment leak rate (%/day)	0.15 (0-24 h) 0.075 (1-30 days)
Time between accident and equalization of primary and secondary system pressures (s)	2500
CREFS initiation time	Safety injection signal generated: 27 s Pressurization: < 60 s
Filtered pressurization rate (ft^3/min)	375
Filtered recirculation rate (ft^3/min)	2700
Unfiltered inleakage rate (ft^3/min)	325 (includes 10 ft^3/min for ingress/egress)
Filter efficiencies (%)	
Pressurization air (all forms of iodine)	98.5
Recirculation air (elemental/organic iodine)	94.5
(particulate iodine)	98.5
Volume (ft^3)	114,000

TABLE 15.4-31 (SHEET 4 OF 4)

	Conservative Analysis
Operator breathing rate (m^3/s)	3.5×10^{-4}
Meteorology	Accident (see appendix 15B)
Off-Site Dose (Rem)	
Site Boundary	
Containment release	2.5
Secondary release	0.5
LPZ	
Containment release	1.9
Secondary release	0.2
Control Room Dose (rem)	
Containment release	2.7
Secondary release	< 0.1

Blowdown	0 sec.	CTMT Cooling System Running (No LOSP)
		Break Occurs
		Reactor Trip (Low Pressurizer Pressure or High CTMT Pressure)
		SI Signal (Low Pressurizer Pressure or High CTMT Pressure)
		Accumulator Injection Begins
		Pumped ECCS Injection Begins (No LOSP)
	20-25 sec.	End of Bypass
		End of Blowdown
RefIll		CTMT Spray Begins (No LOSP)
	25-30 sec.	Bottom of Core Recovery
Reflood		Accumulators Empty
	10 min.	Core Quenched
Long Term CoolIng		Switch to Cold Leg Recirculation on RWST Low Level Alarm
		Switch to Hot Leg/Cold Leg Recirculation
	24 hr.	

16

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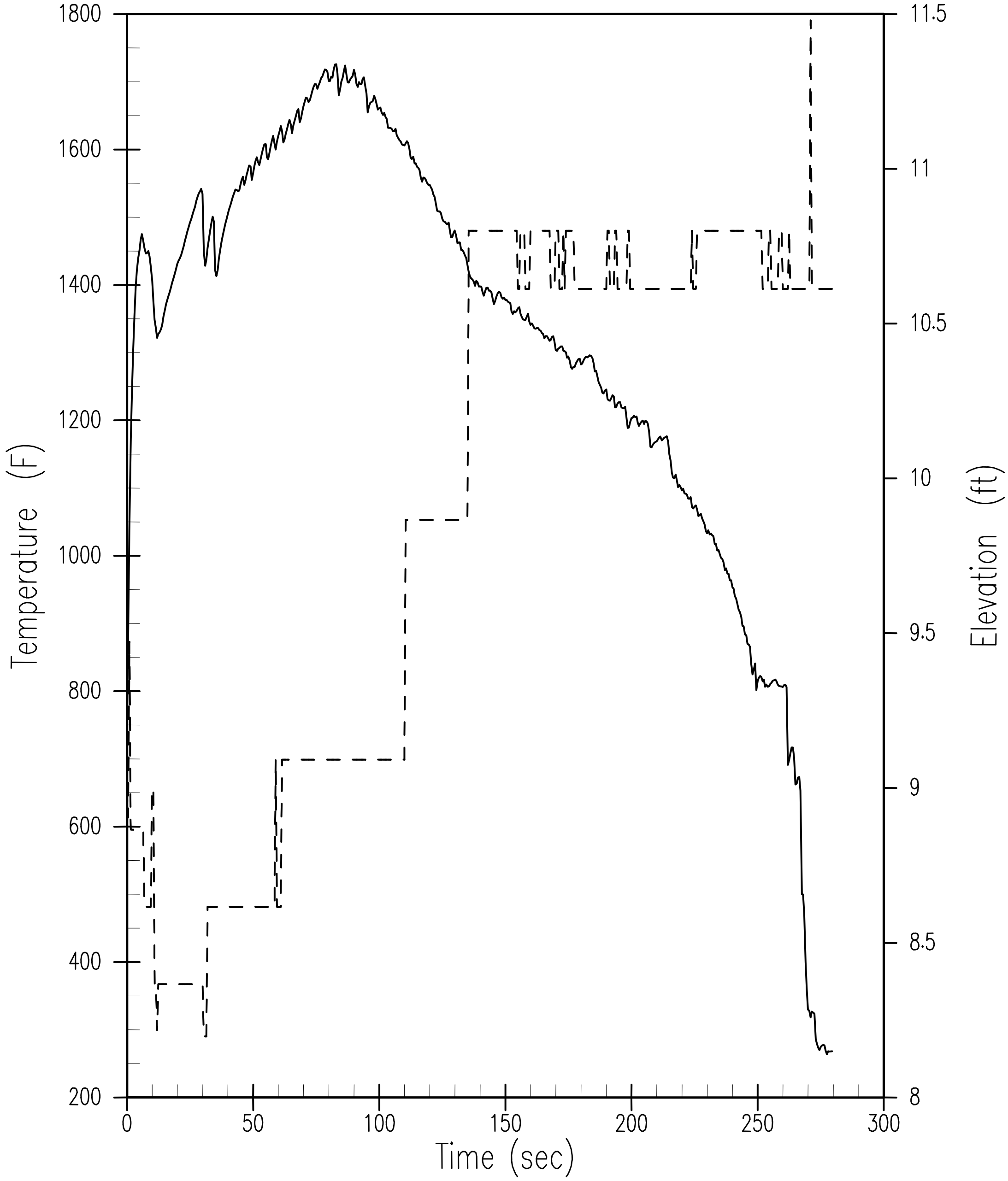

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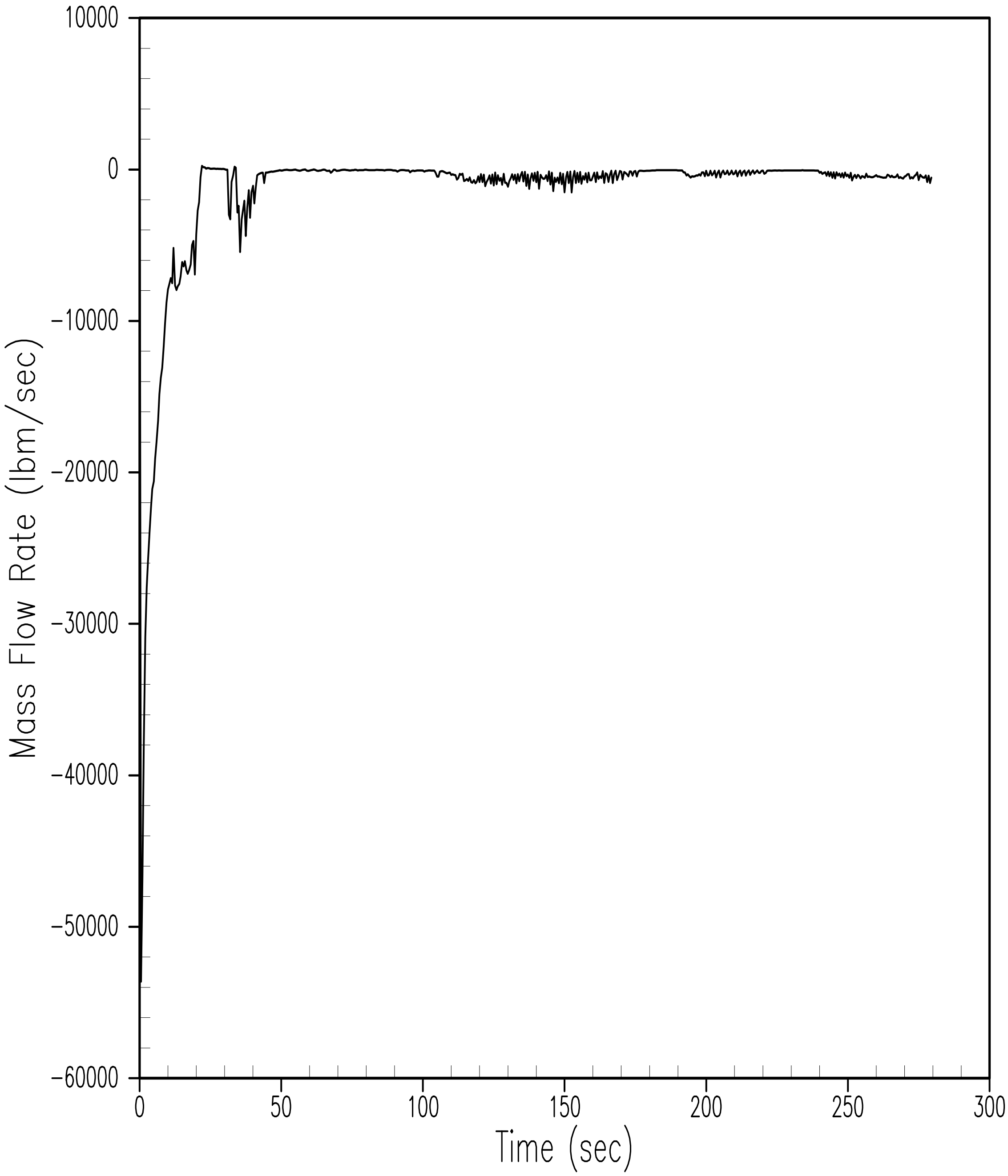
PEAK CLADDING TEMPERATURE AND ELEVATION (REFERENCE DECLG TRANSIENT)

FIGURE 15.4-2



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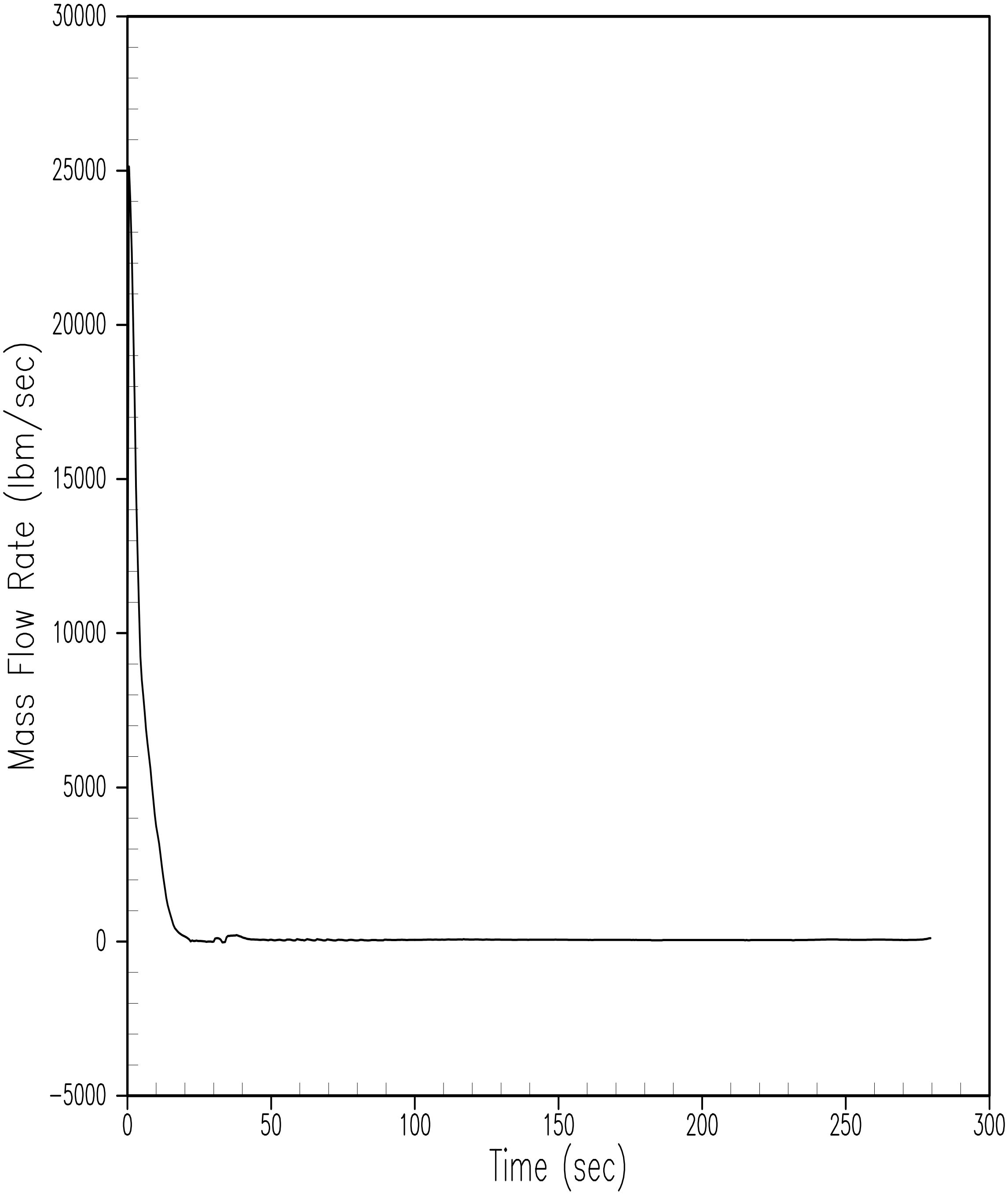
BREAK FLOW (VESSEL SIDE)

FIGURE 15.4-3



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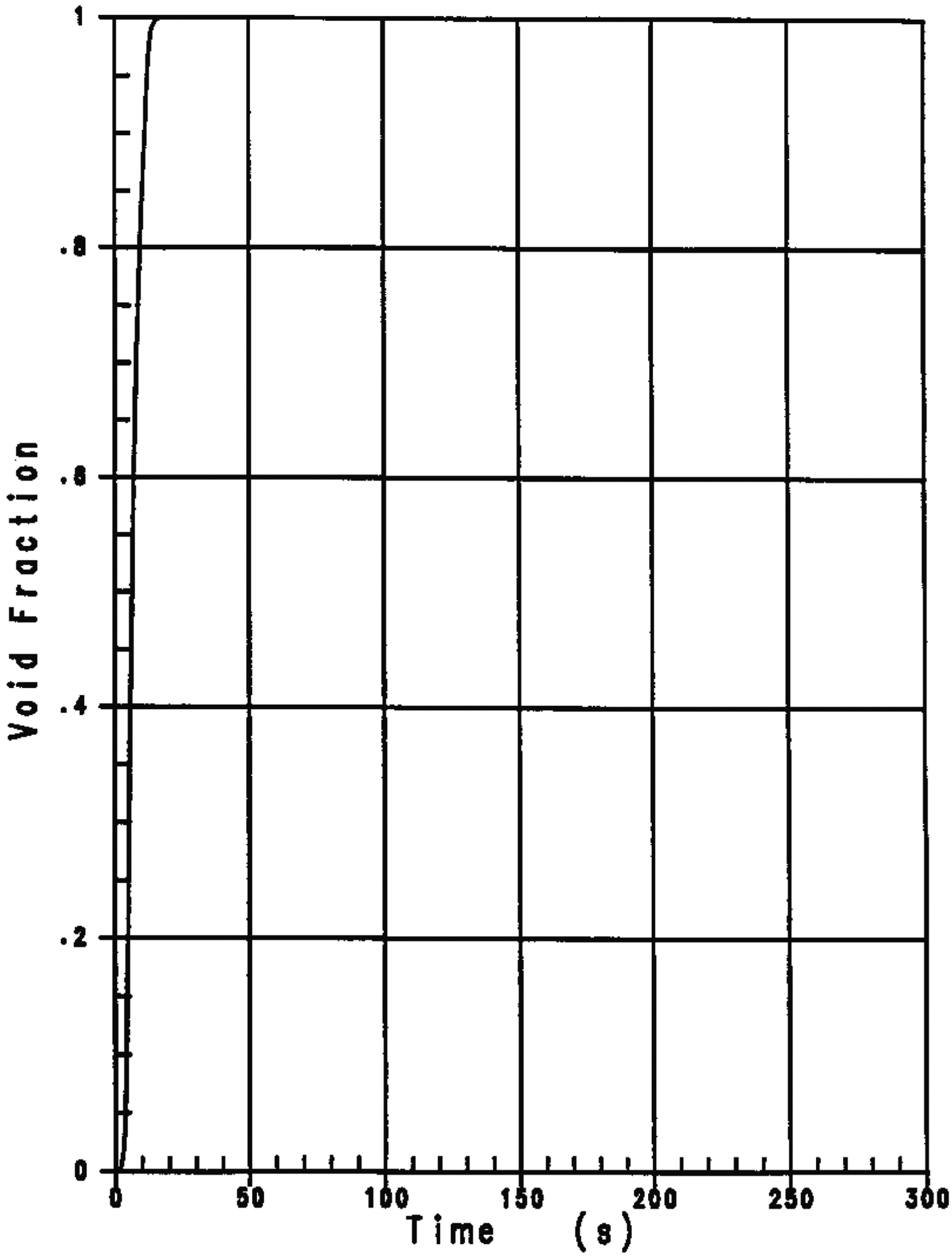
BREAK FLOW (LOOP SIDE)

FIGURE 15.4-4



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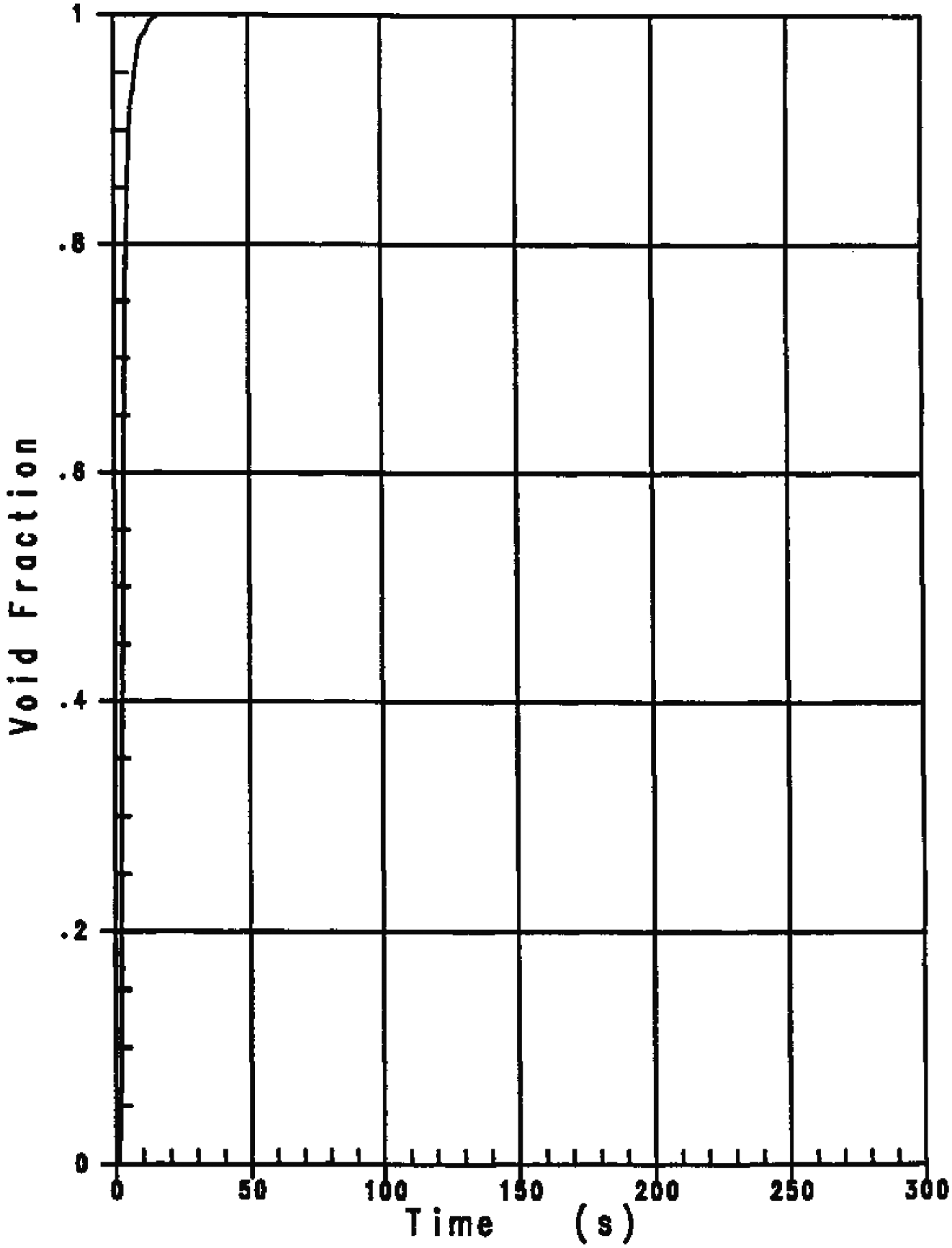
INTACT LOOP PUMP INLET
VOID FRACTION

FIGURE 15.4-5



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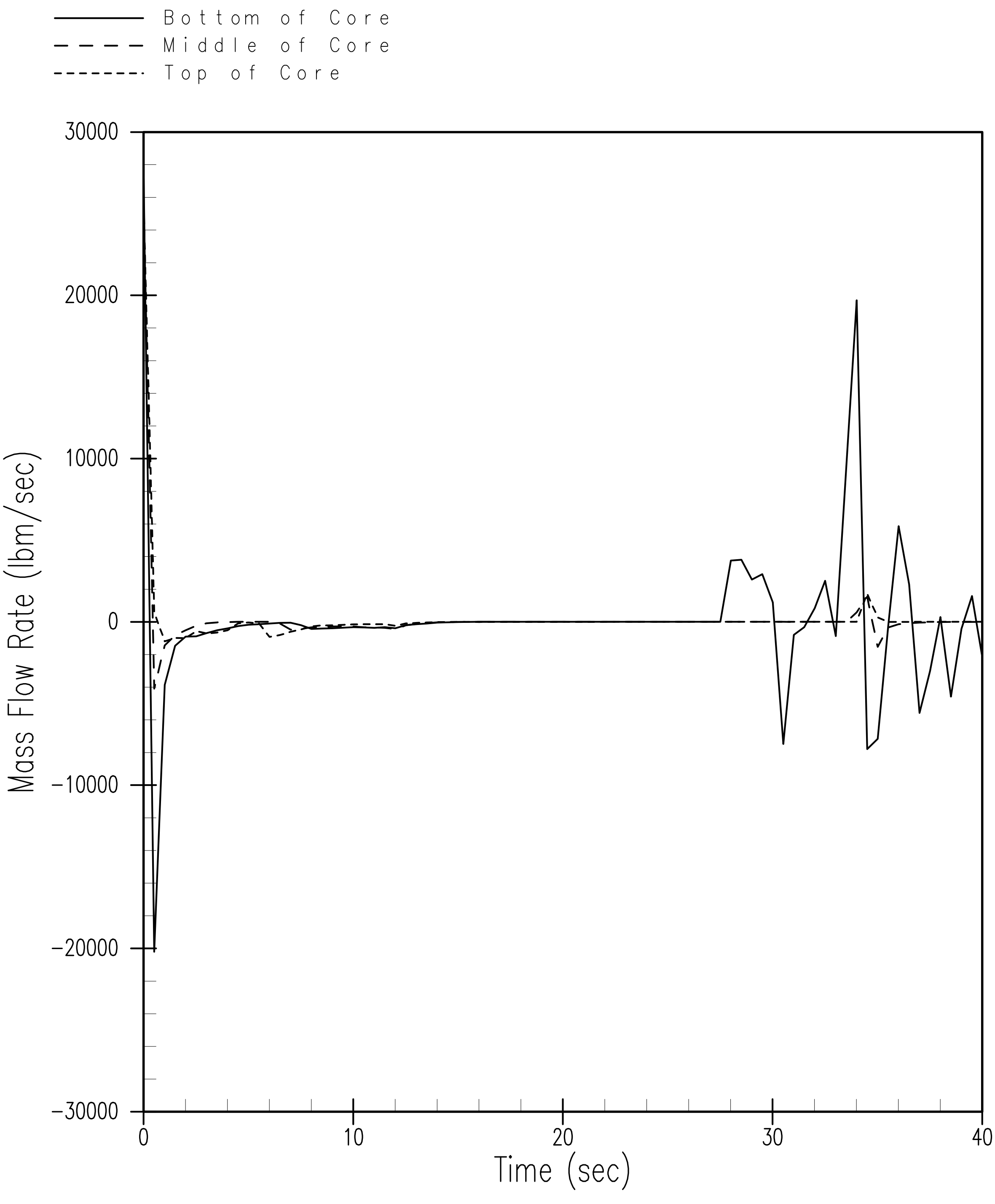
BROKEN LOOP PUMP INLET
VOID FRACTION

FIGURE 15.4-6



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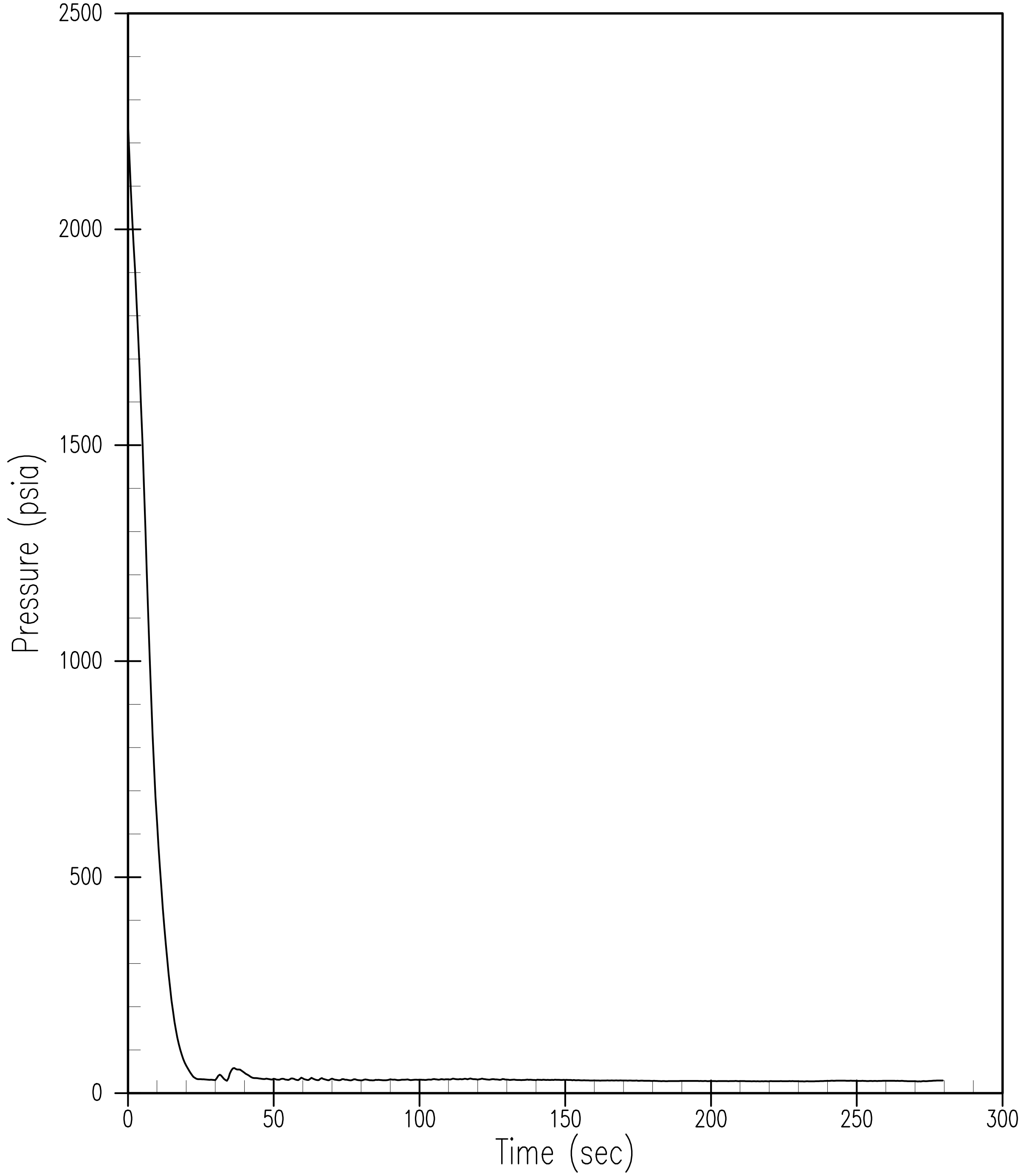
FLOWRATE AT TOP, MIDDLE, AND BOTTOM OF CORE DURING BLOWDOWN

FIGURE 15.4-7



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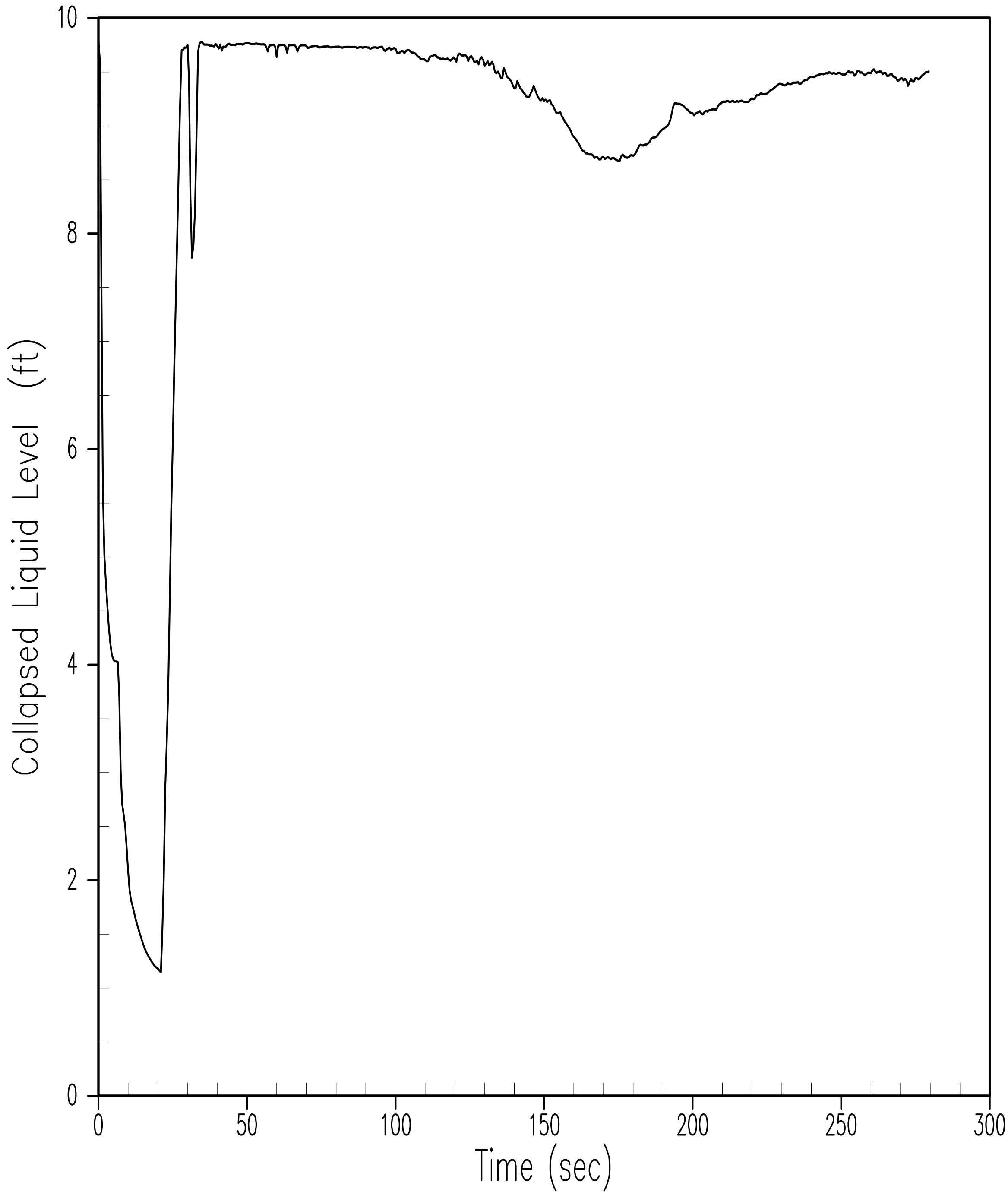
PRESSURIZER PRESSURE

FIGURE 15.4-8



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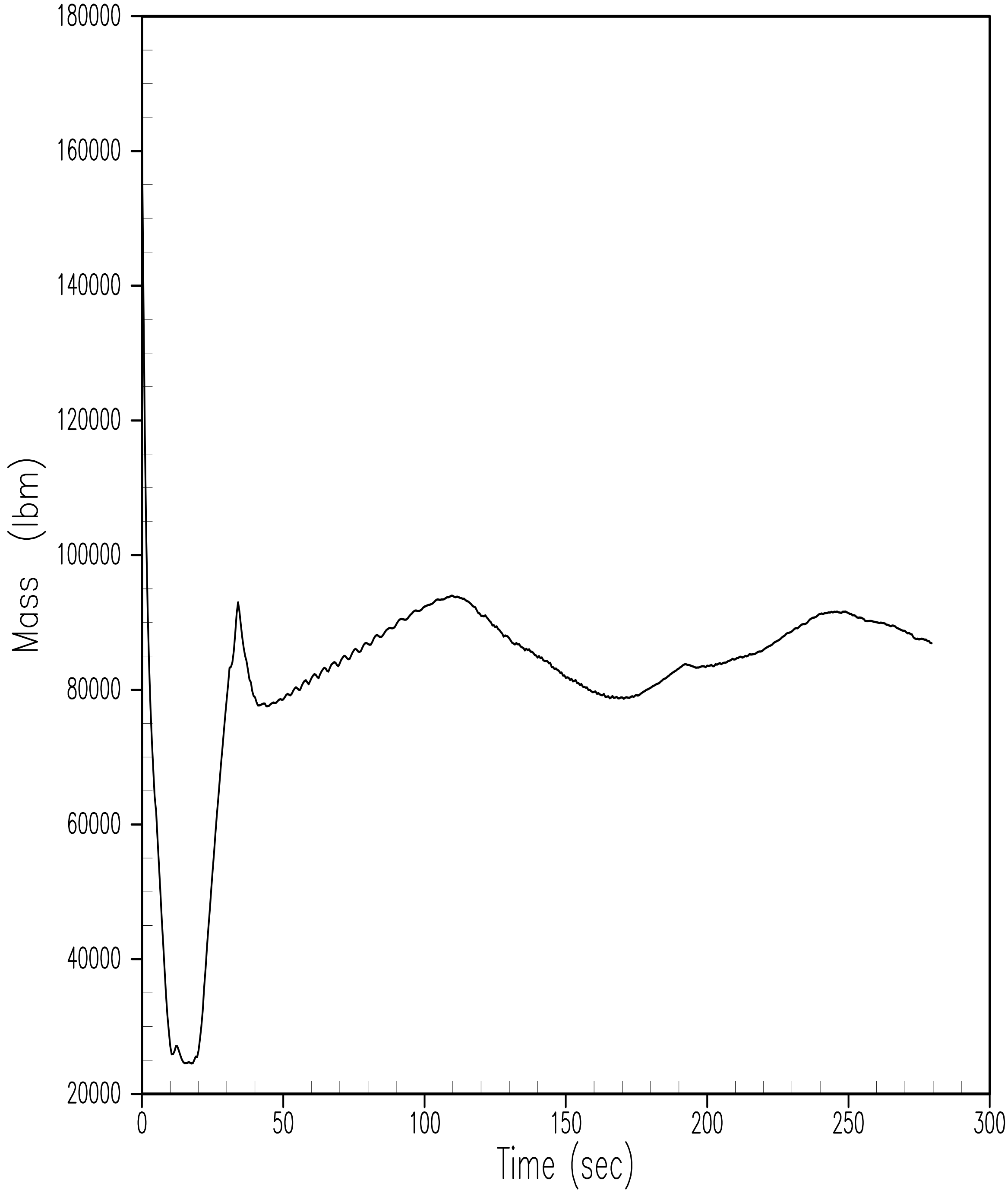
UNIT 1 AND UNIT 2

LOWER PLENUM COLLAPSED LIQUID LEVEL

FIGURE 15.4-9



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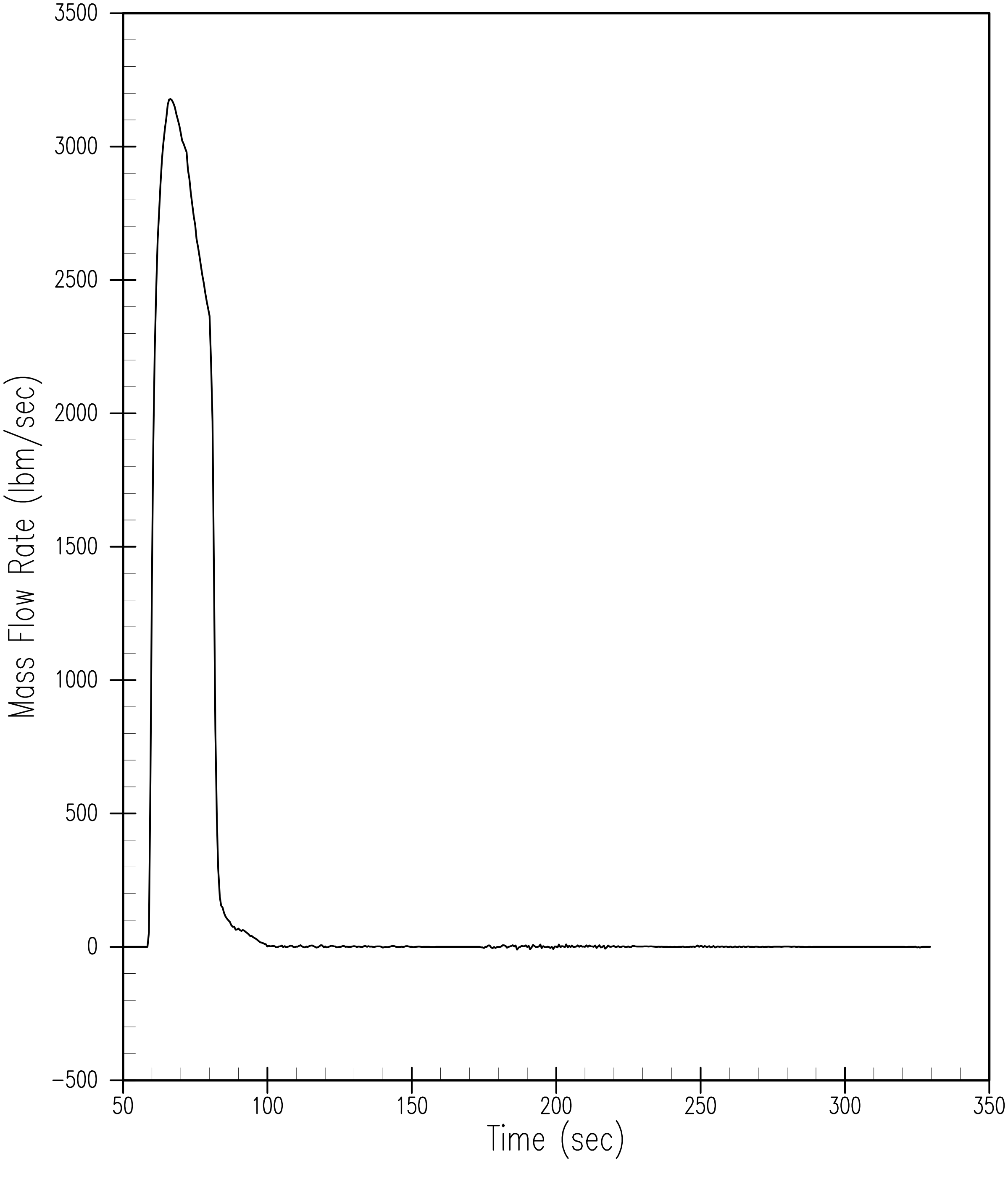
VESSEL FLUID MASS

FIGURE 15.4-10



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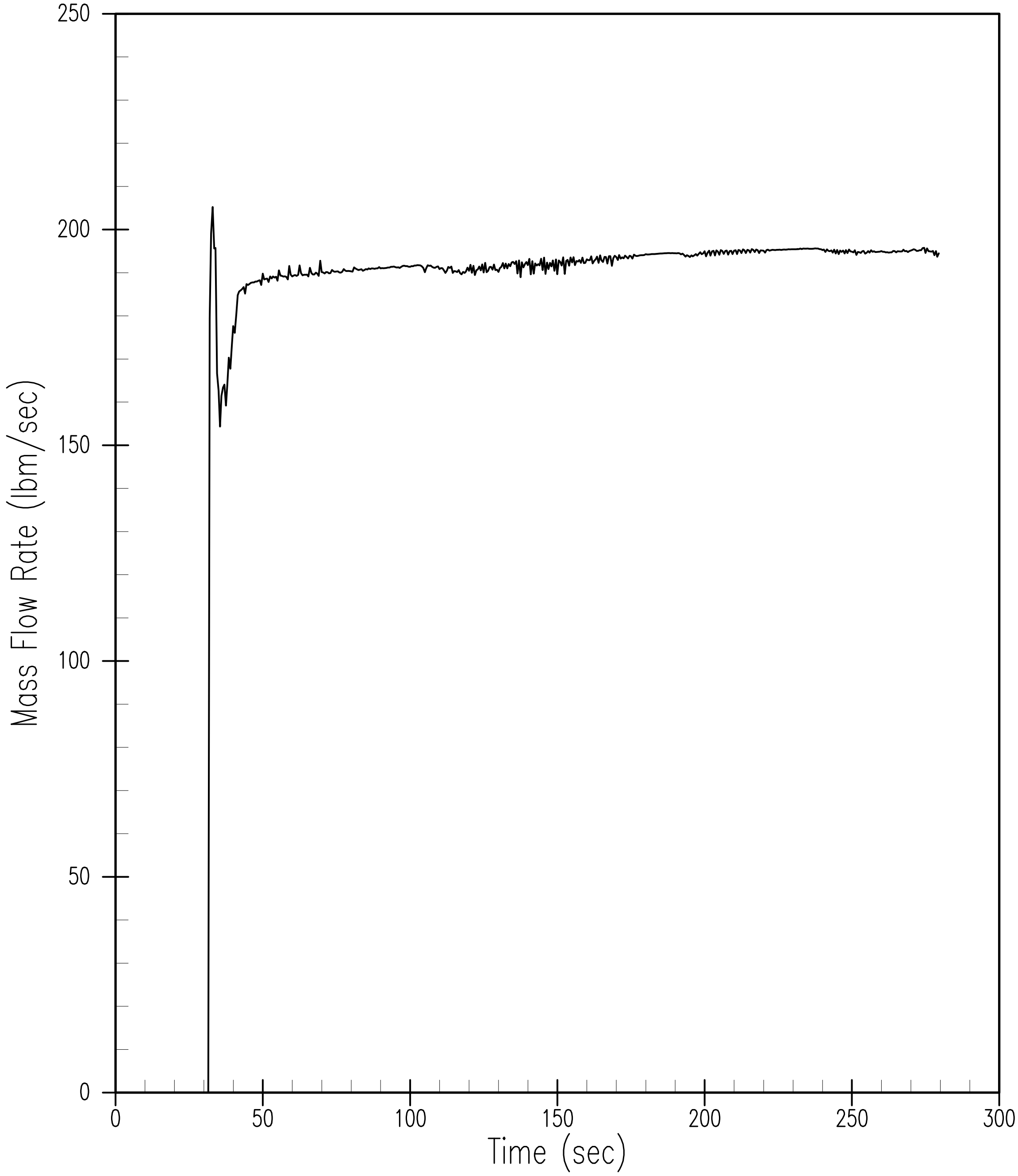
ACCUMULATOR LIQUID FLOW

FIGURE 15.4-11



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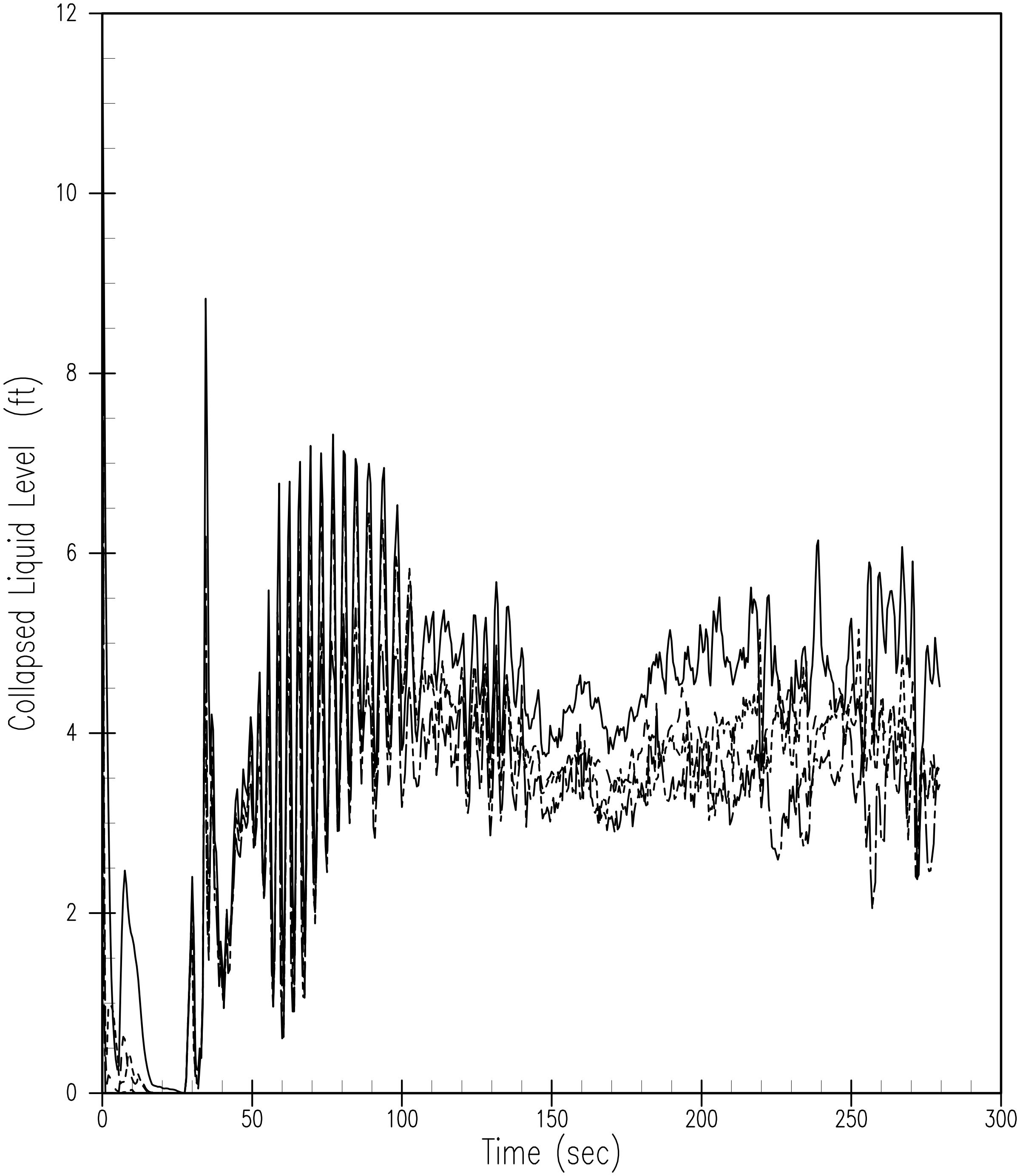
SAFETY INJECTION FLOW

FIGURE 15.4-12



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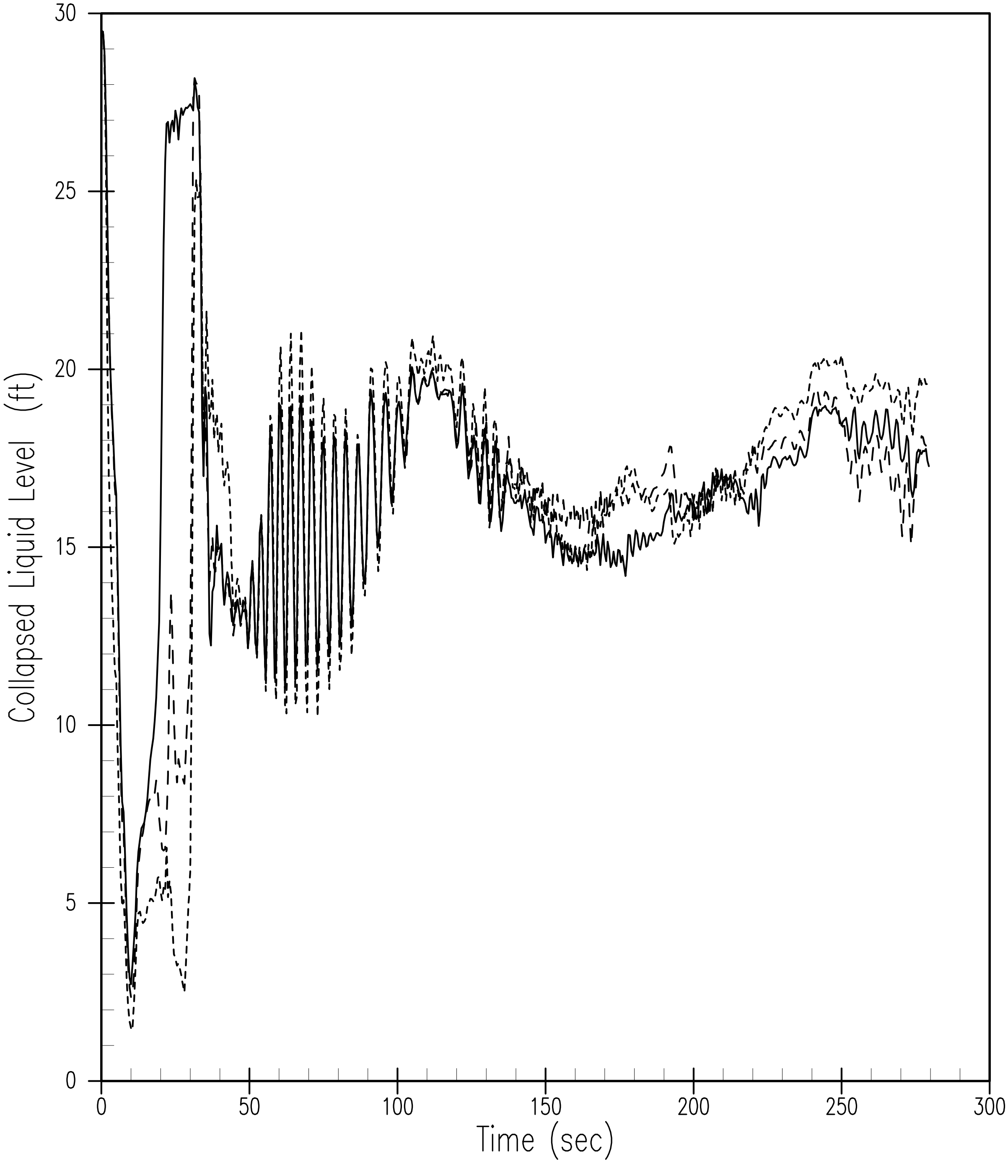
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE CHANNELS COLLAPSED LIQUID LEVEL

FIGURE 15.4-13



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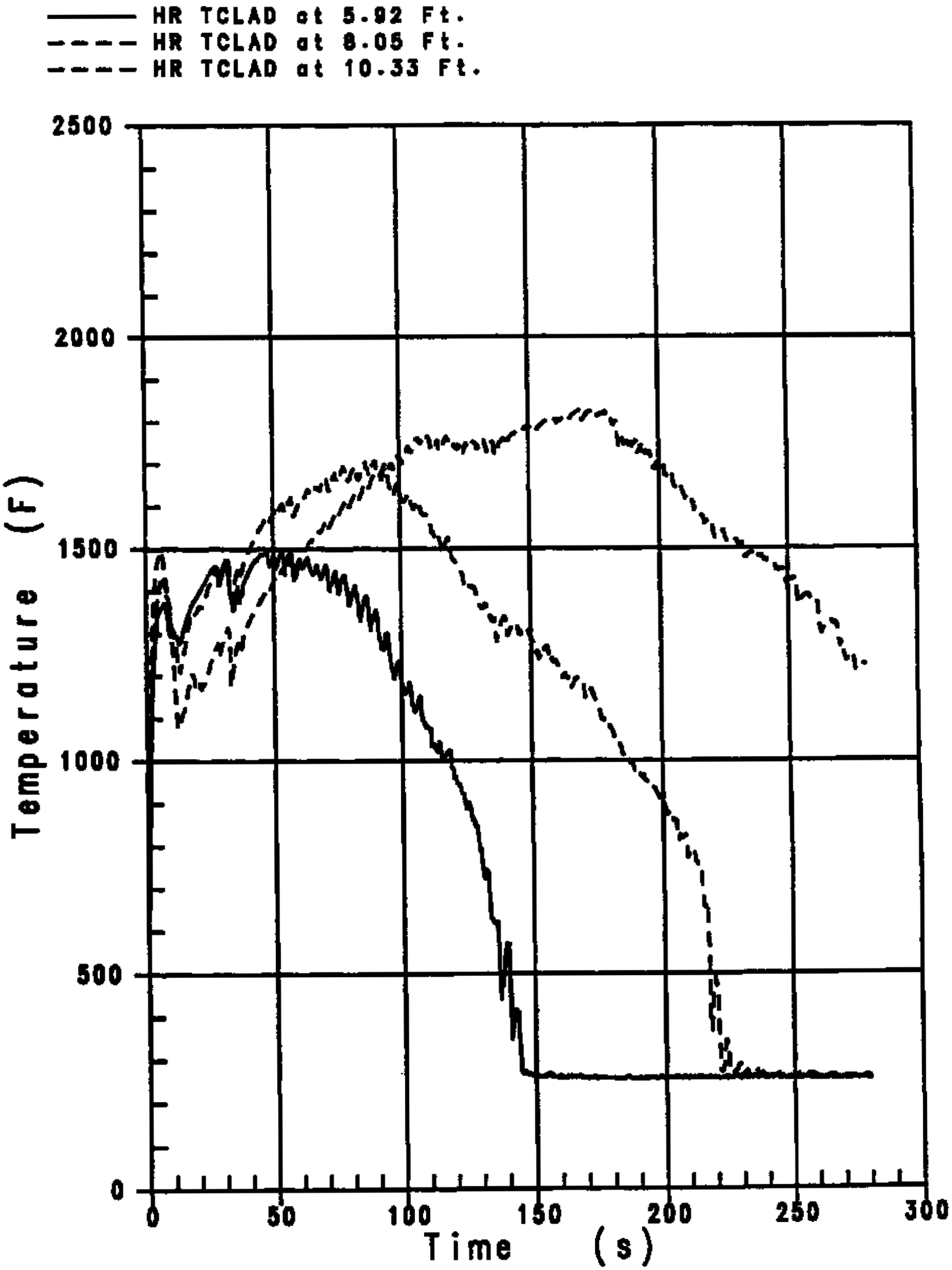
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DOWNCOMER CHANNELS COLLAPSED
LIQUID LEVEL

FIGURE 15.4-14



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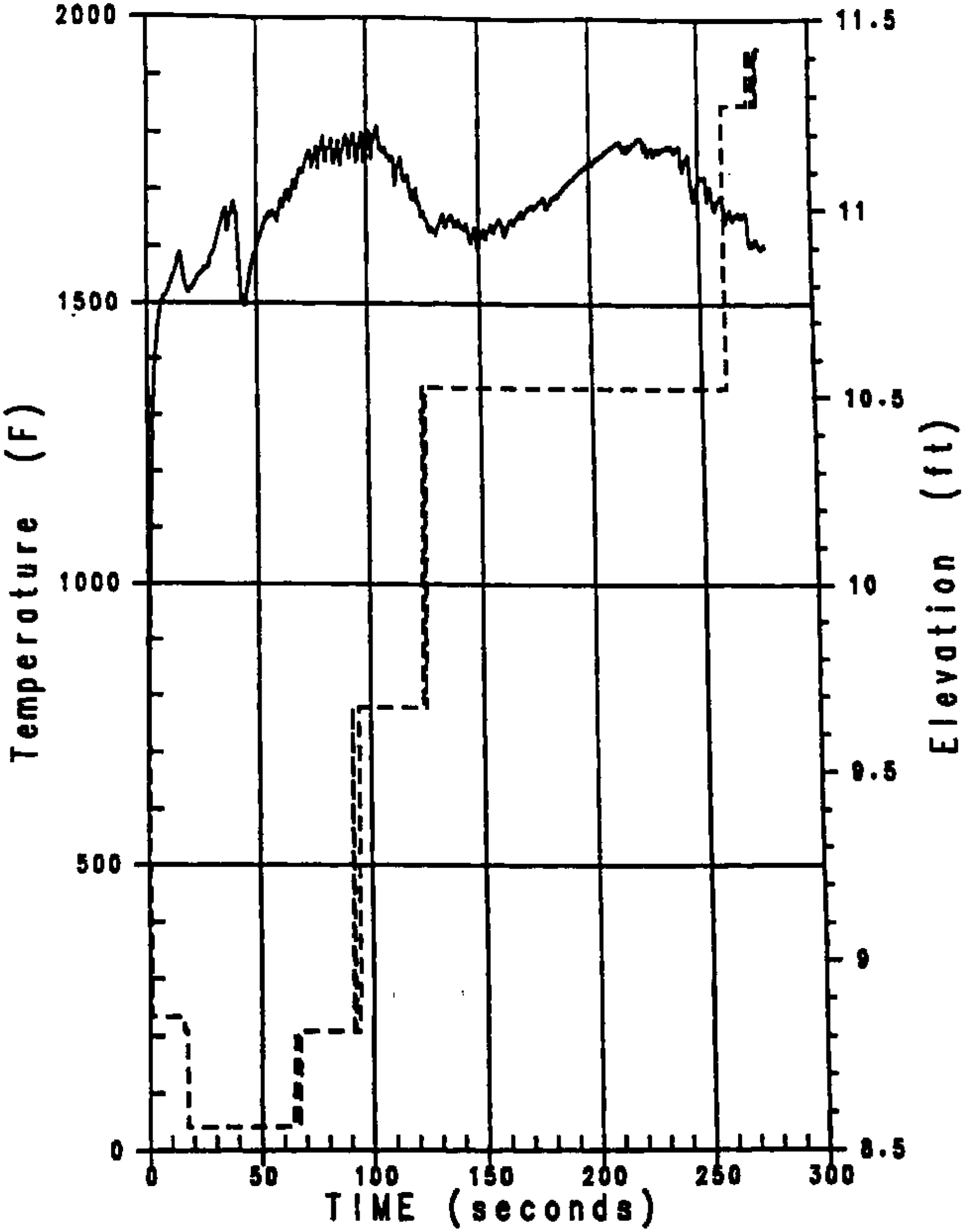
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CLADDING TEMPERATURE AT VARIOUS ELEVATIONS (HOT ROD) FIGURE 15.4-15



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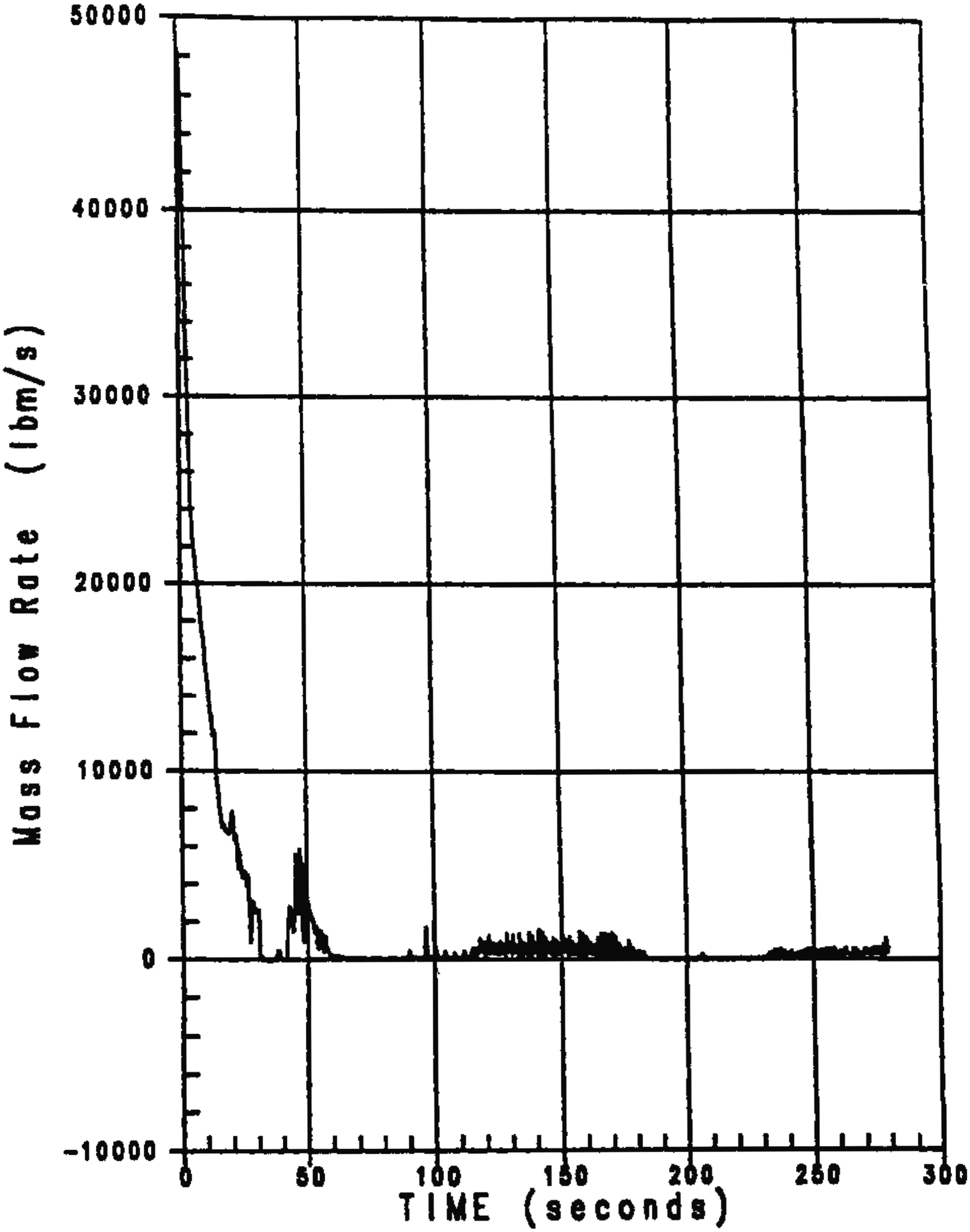
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PEAK CLADDING TEMPERATURE
(REFERENCE SPLIT TRANSIENT)

FIGURE 15.4-15A



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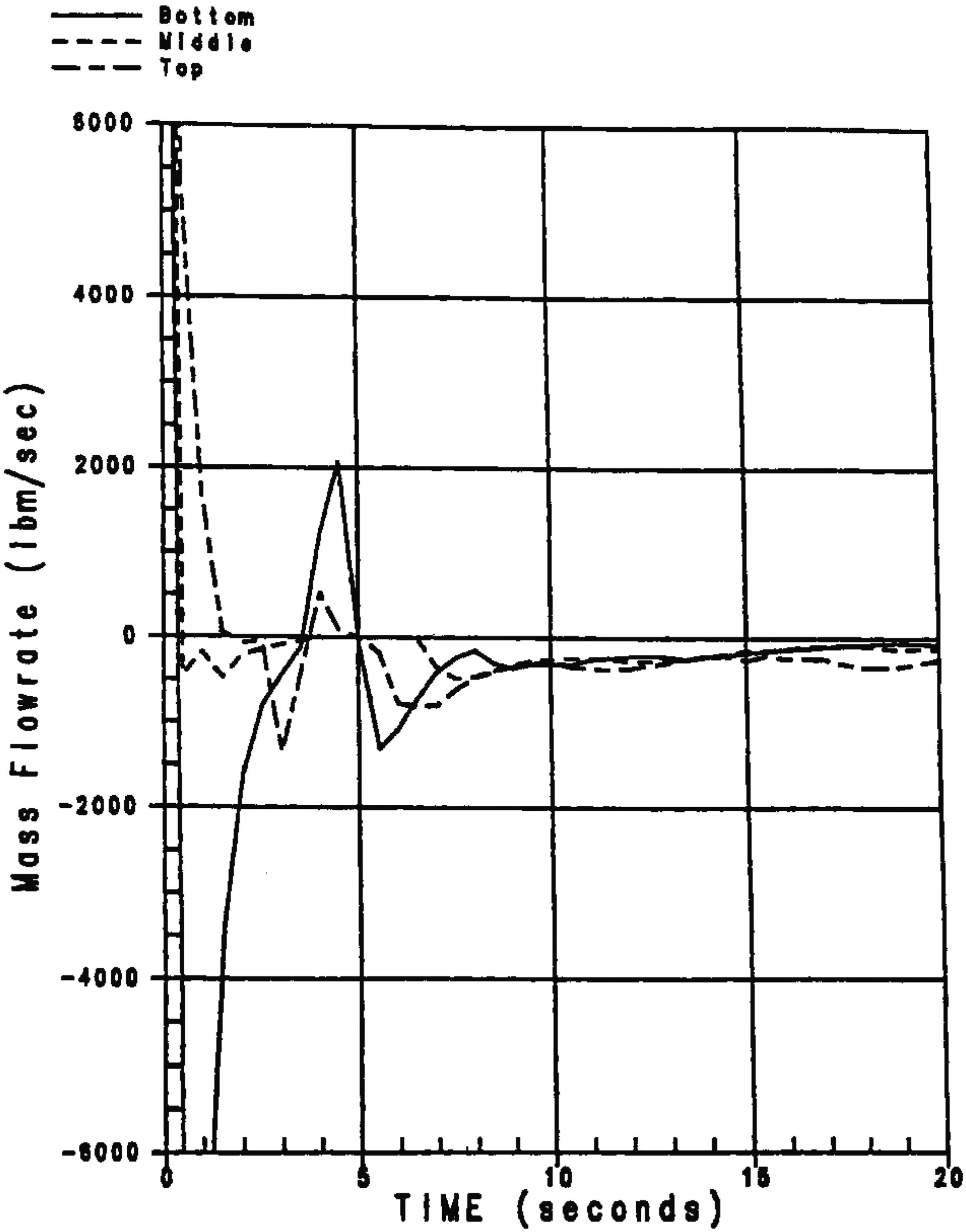
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

BREAK FLOW
(SPLIT BREAK)

FIGURE 15.4-15B



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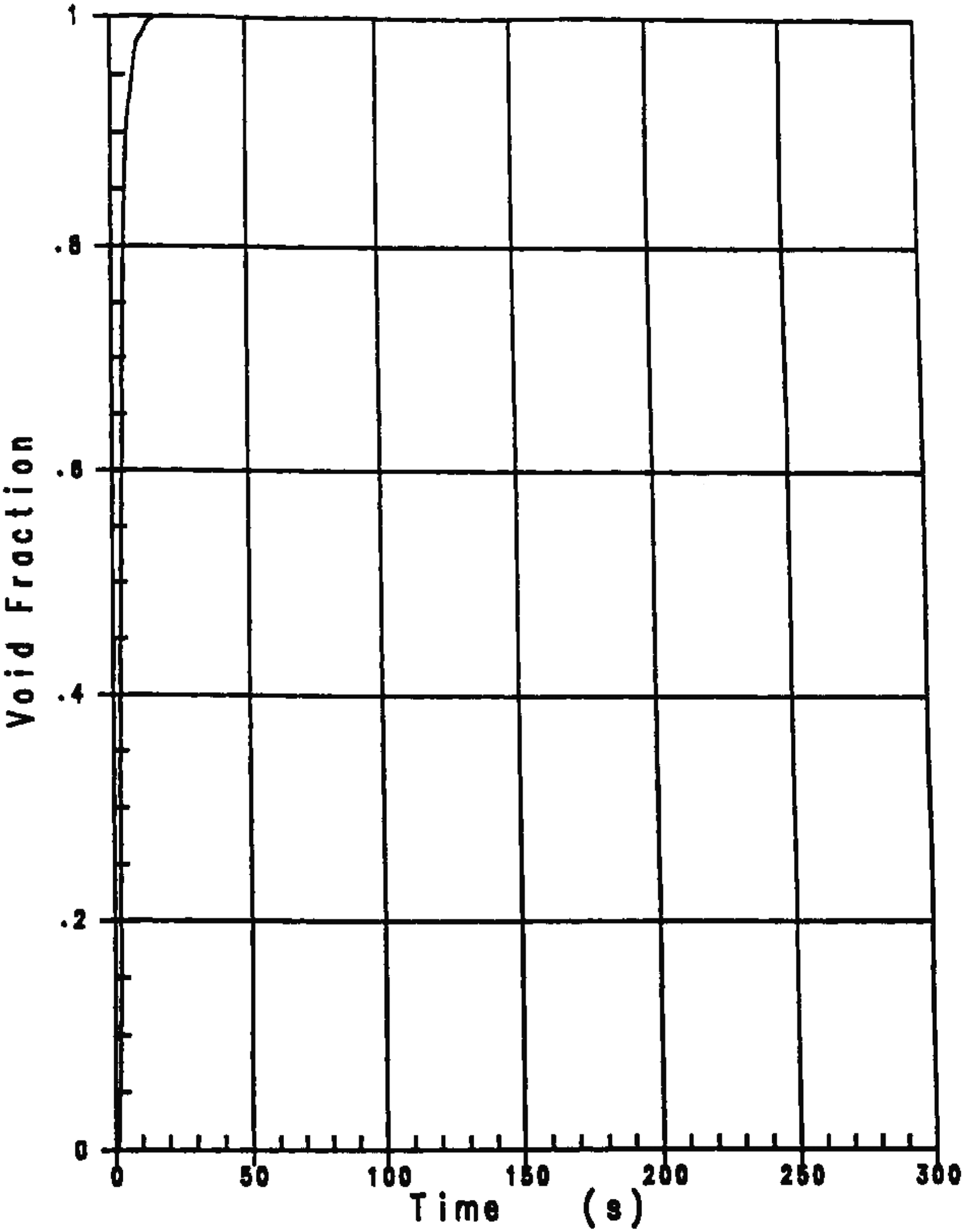
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FLOWRATE AT TOP, MIDDLE, AND BOTTOM
OF CORE DURING BLOWDOWN

FIGURE 15.4-15C



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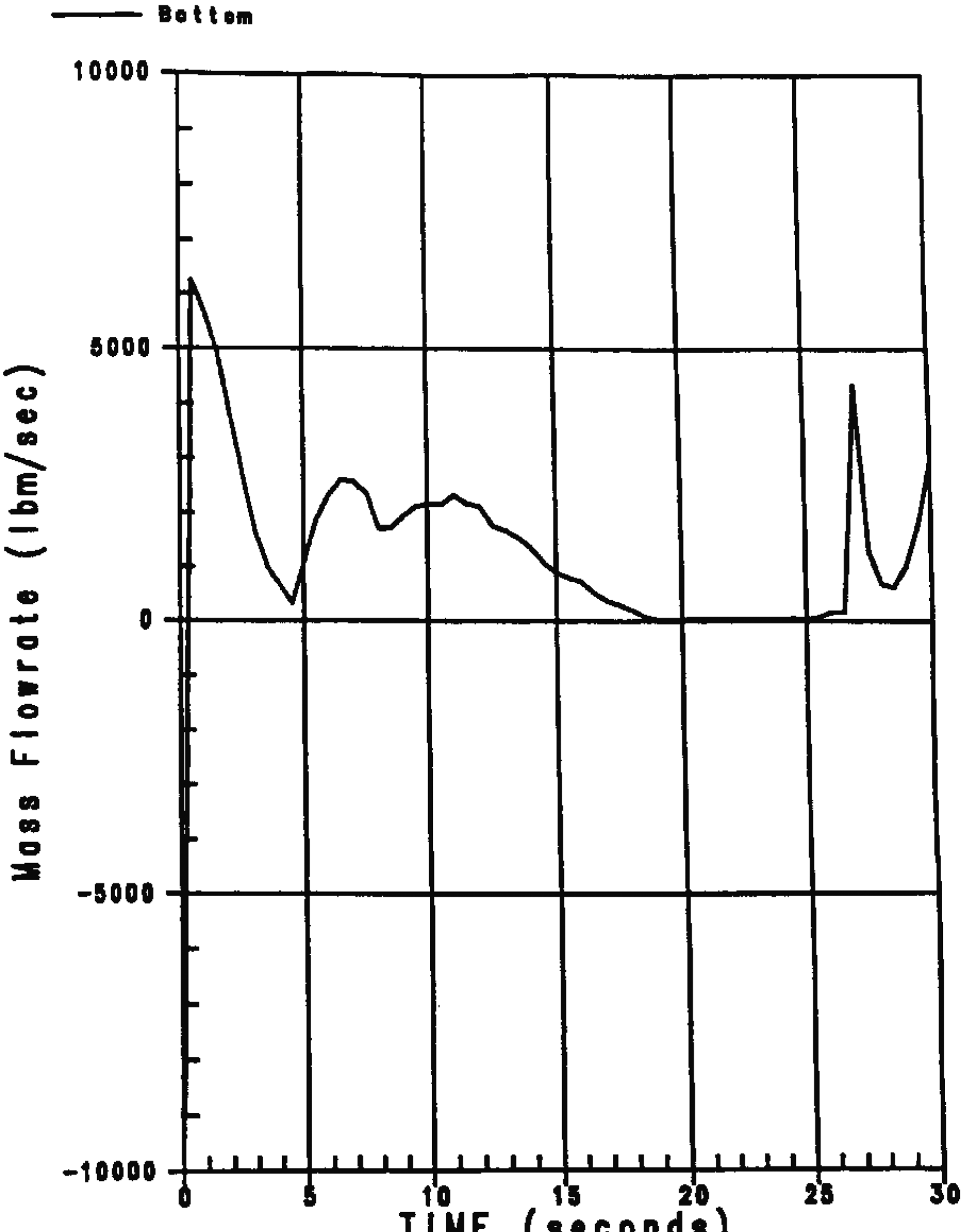
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

INTACT LOOP PUMP INLET
VOID FRACTION

FIGURE 15.4-15D



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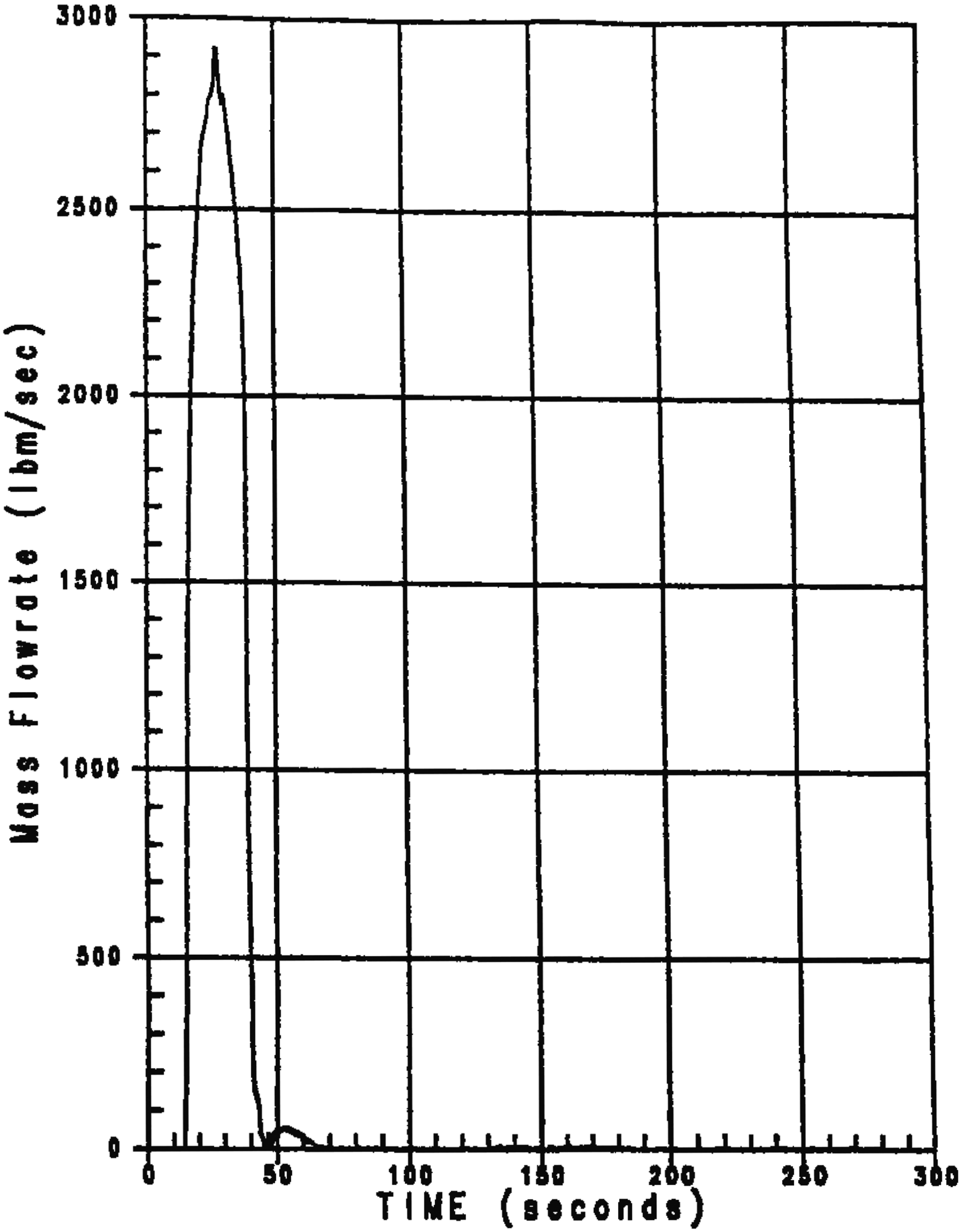
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	FLOWRATE AT BOTTOM OF BROKEN DC CHANNEL FIGURE 15.4-15E



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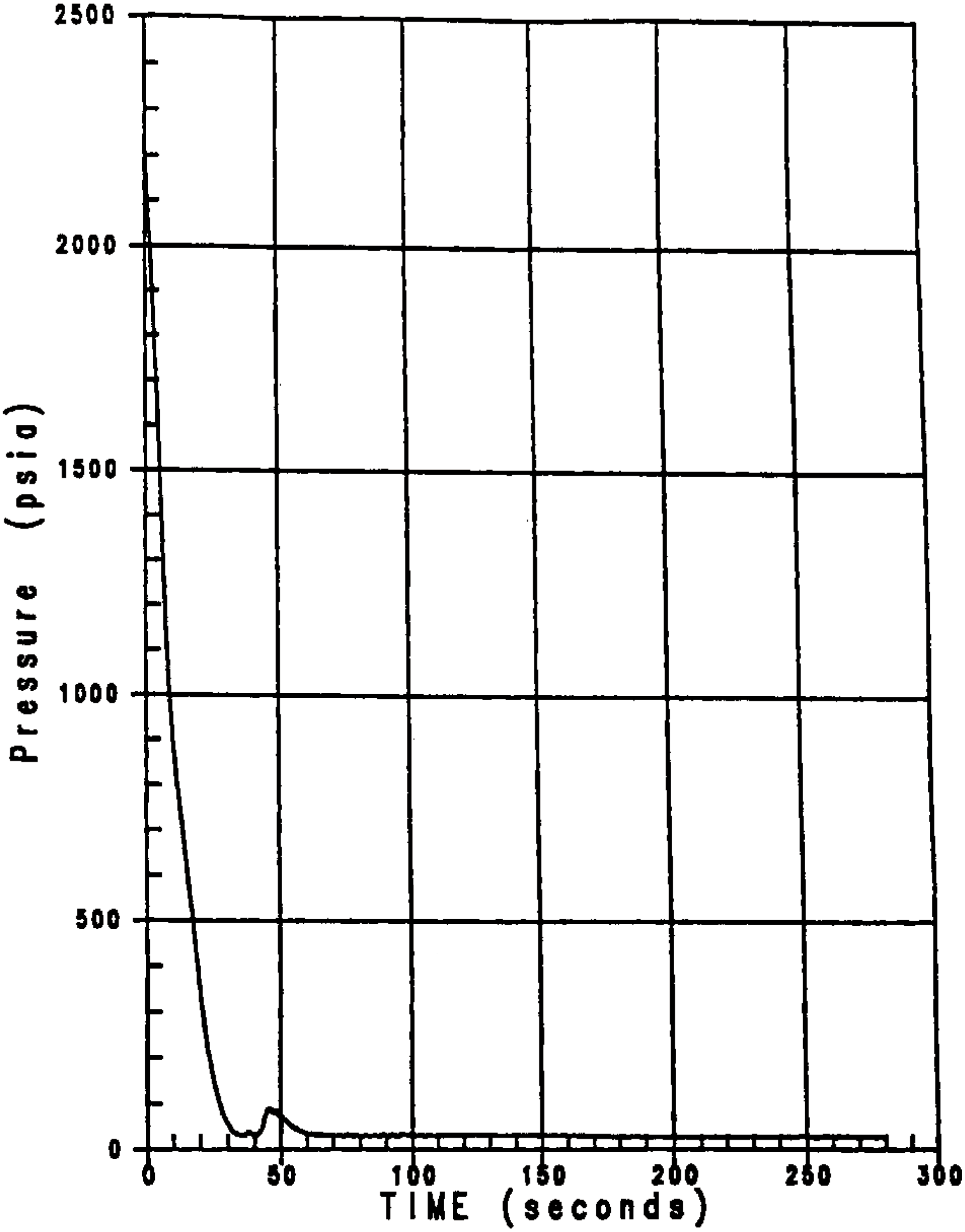
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ACCUMULATOR LIQUID FLOW

FIGURE 15.4-15F



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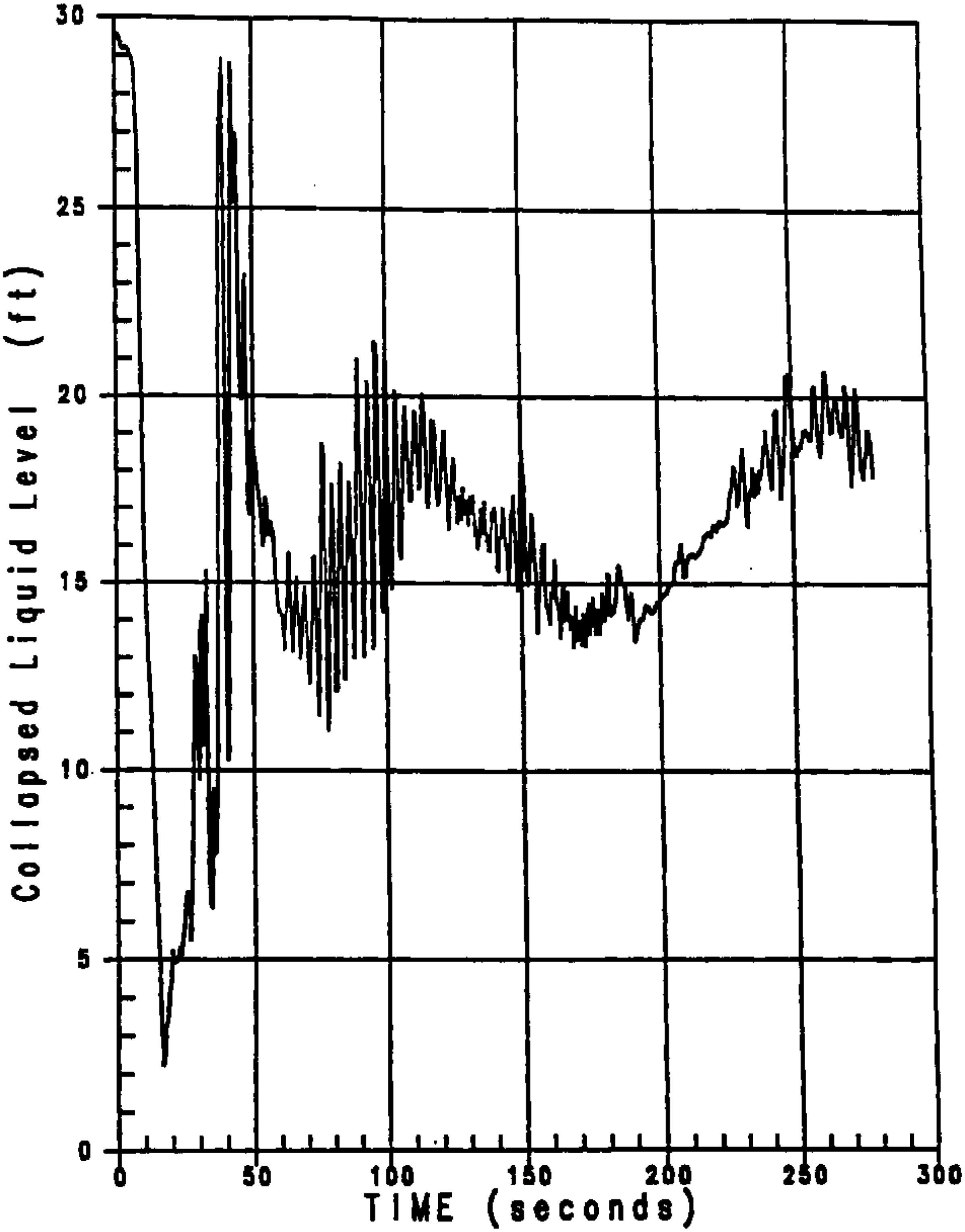
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURIZER PRESSURE

FIGURE 15.4-15G



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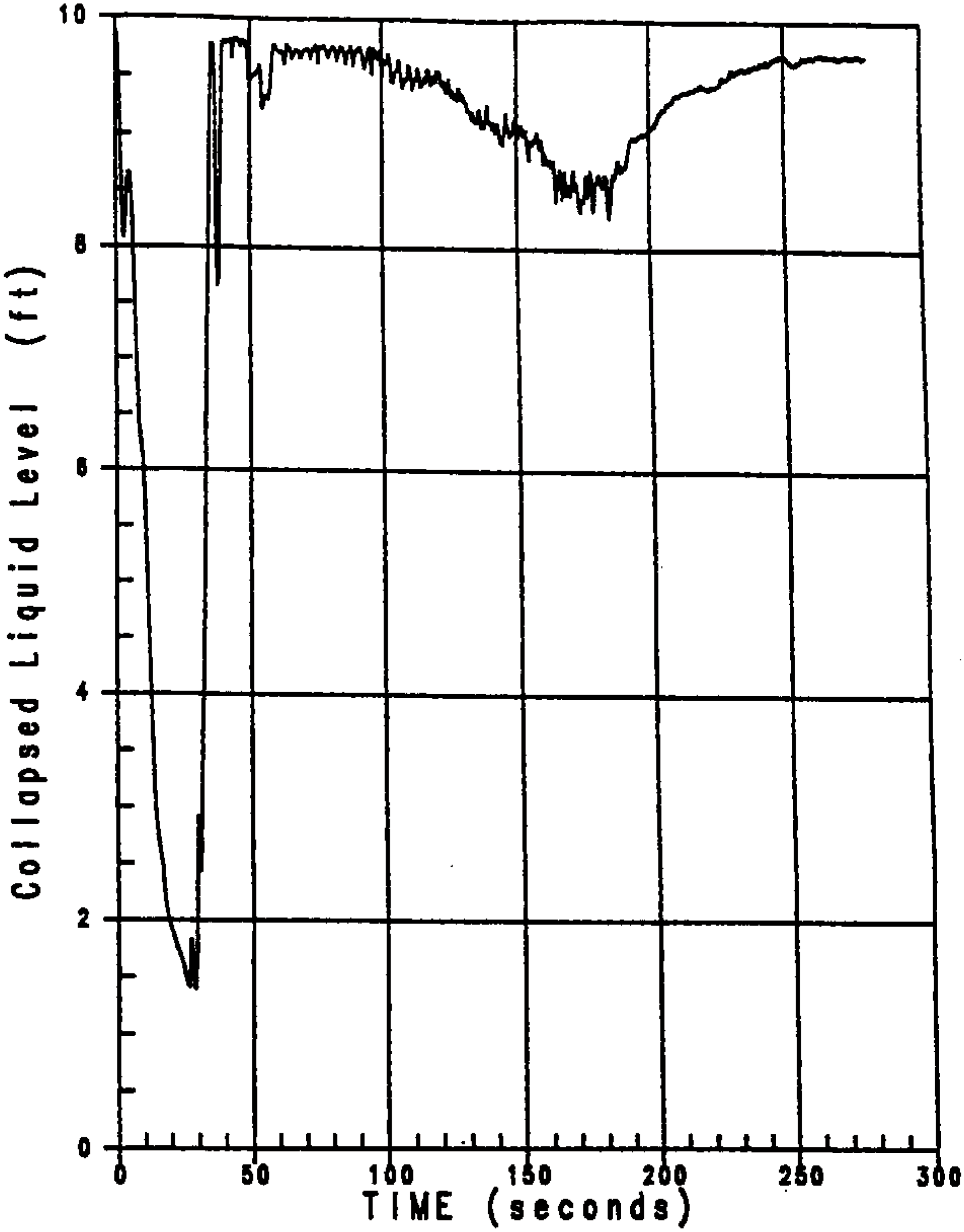
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DOWNCOMER CHANNELS
COLLAPSED LIQUID LEVEL

FIGURE 15.4-15H



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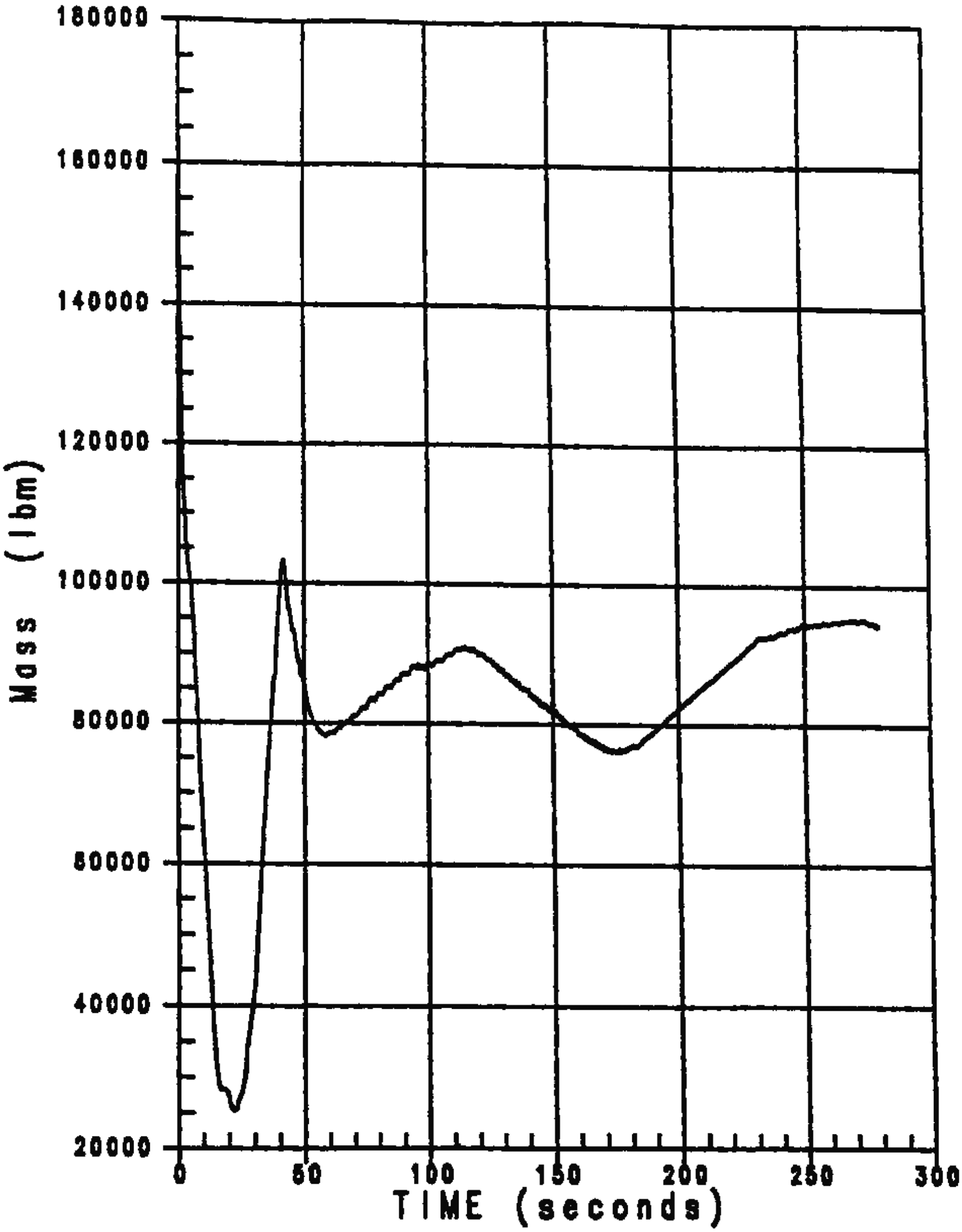
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOWER PLENUM COLLAPSED LIQUID LEVEL

FIGURE 15.4-15I



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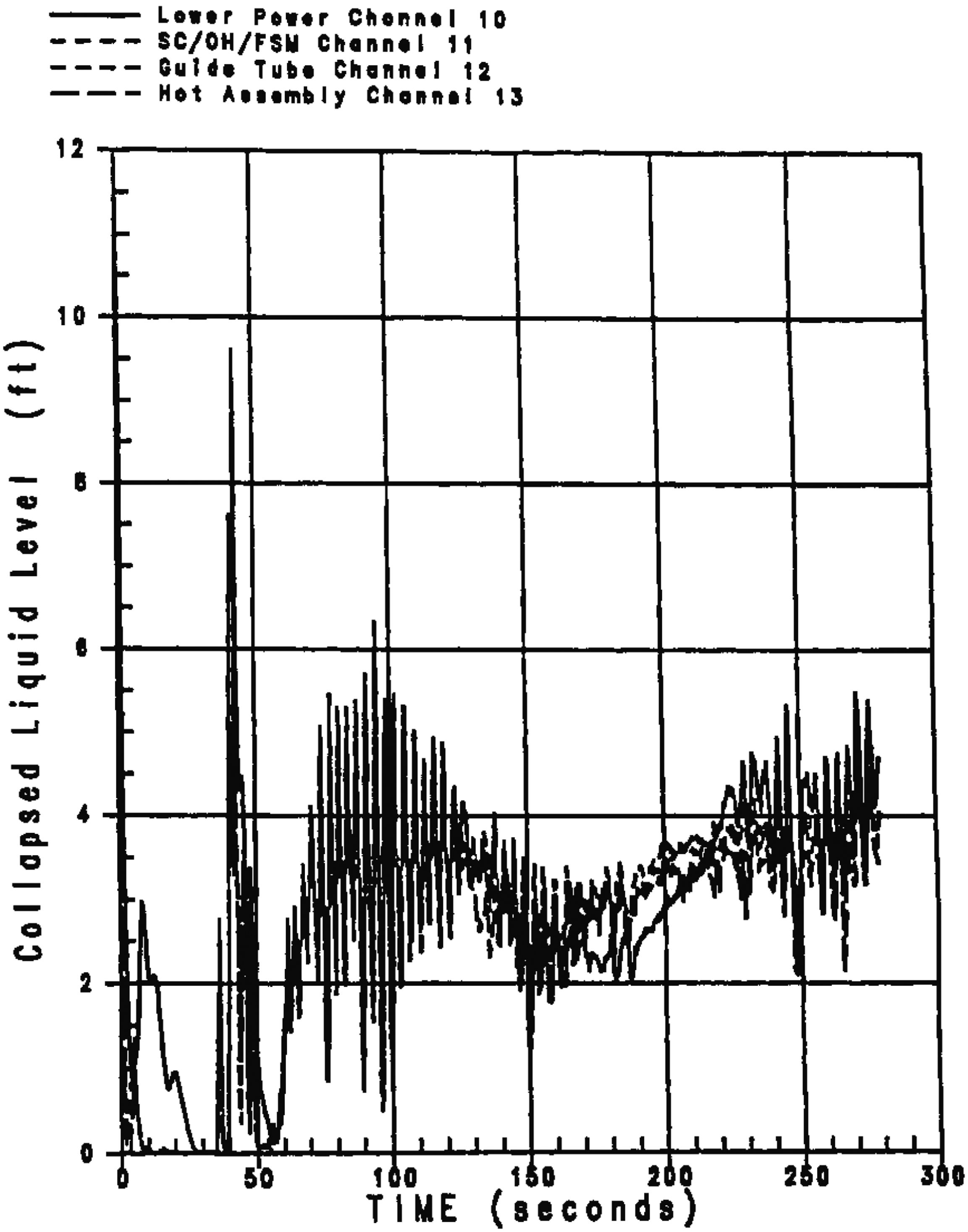
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VESSEL FLUID MASS

FIGURE 15.4-15J



Lower Power Channel 10
SC/OH/FSM Channel 11
Guide Tube Channel 12
Hot Assembly Channel 13
Collapsed Liquid Level (ft)
TIME (seconds)
0
2
4
6
8
10
12
0
50
100
150
200
250
300

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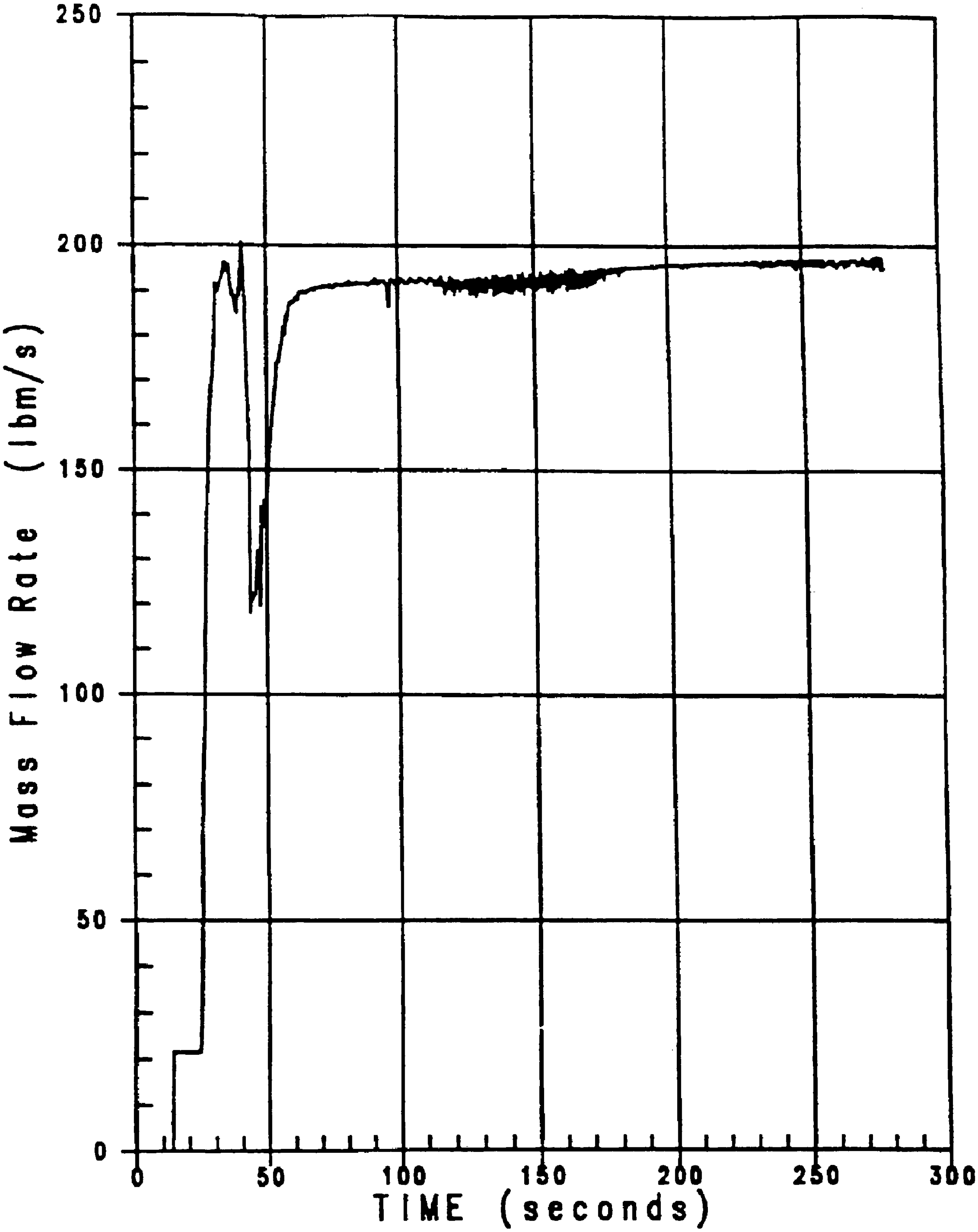

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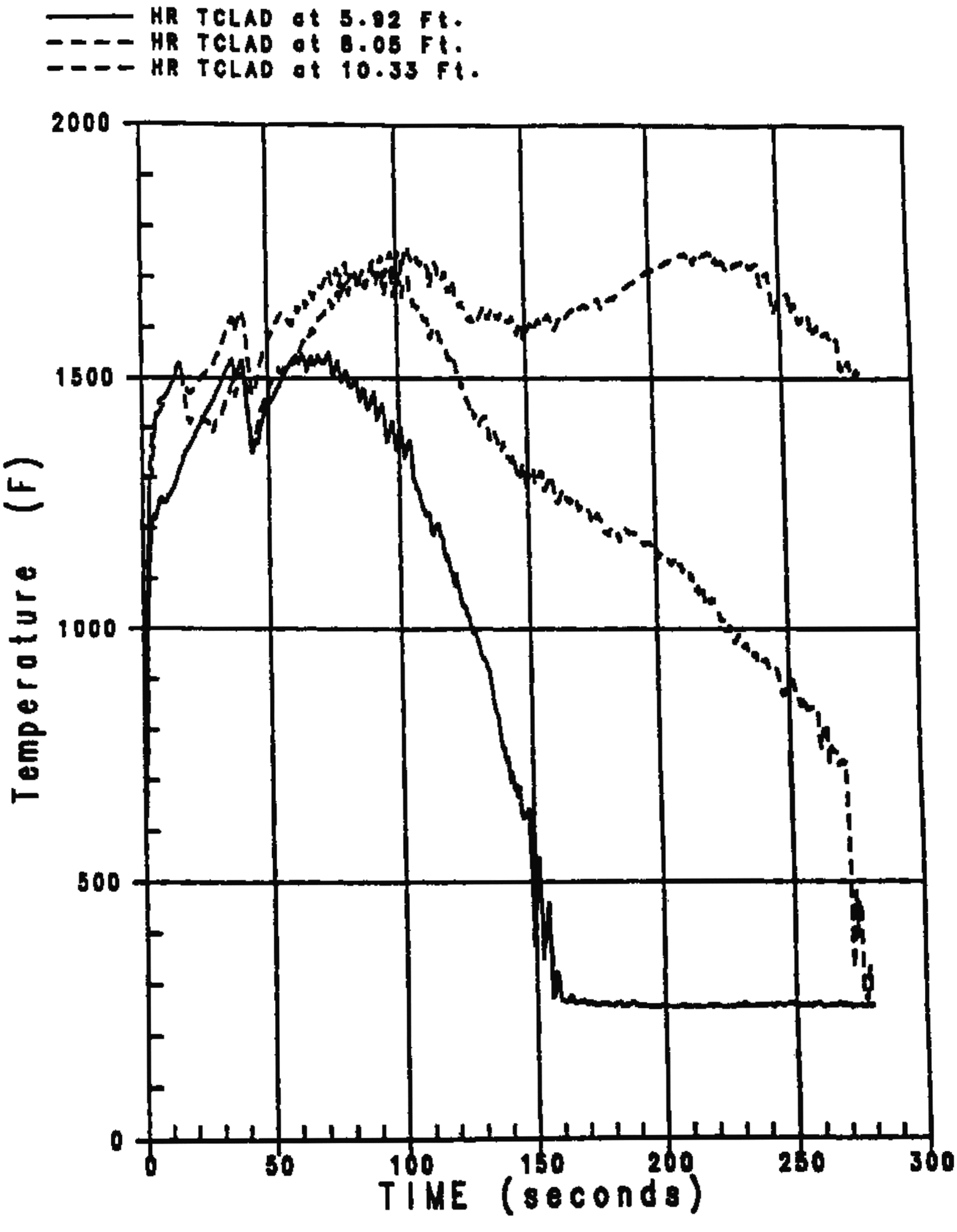
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SAFETY INJECTION FLOW

FIGURE 15.4-15L



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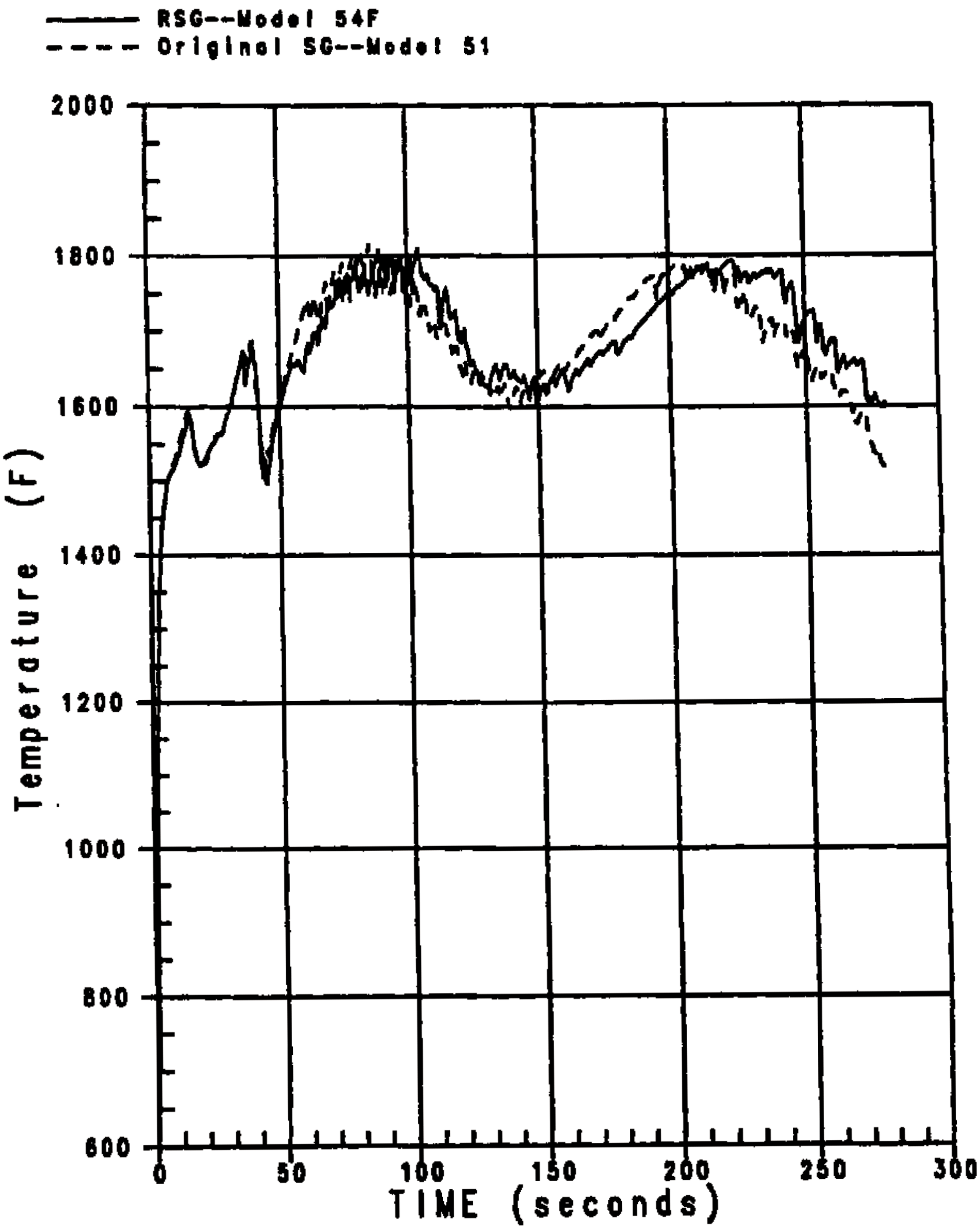
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CLADDING TEMPERATURE
AT VARIOUS ELEVATIONS (HOT ROD)

FIGURE 15.4-15M



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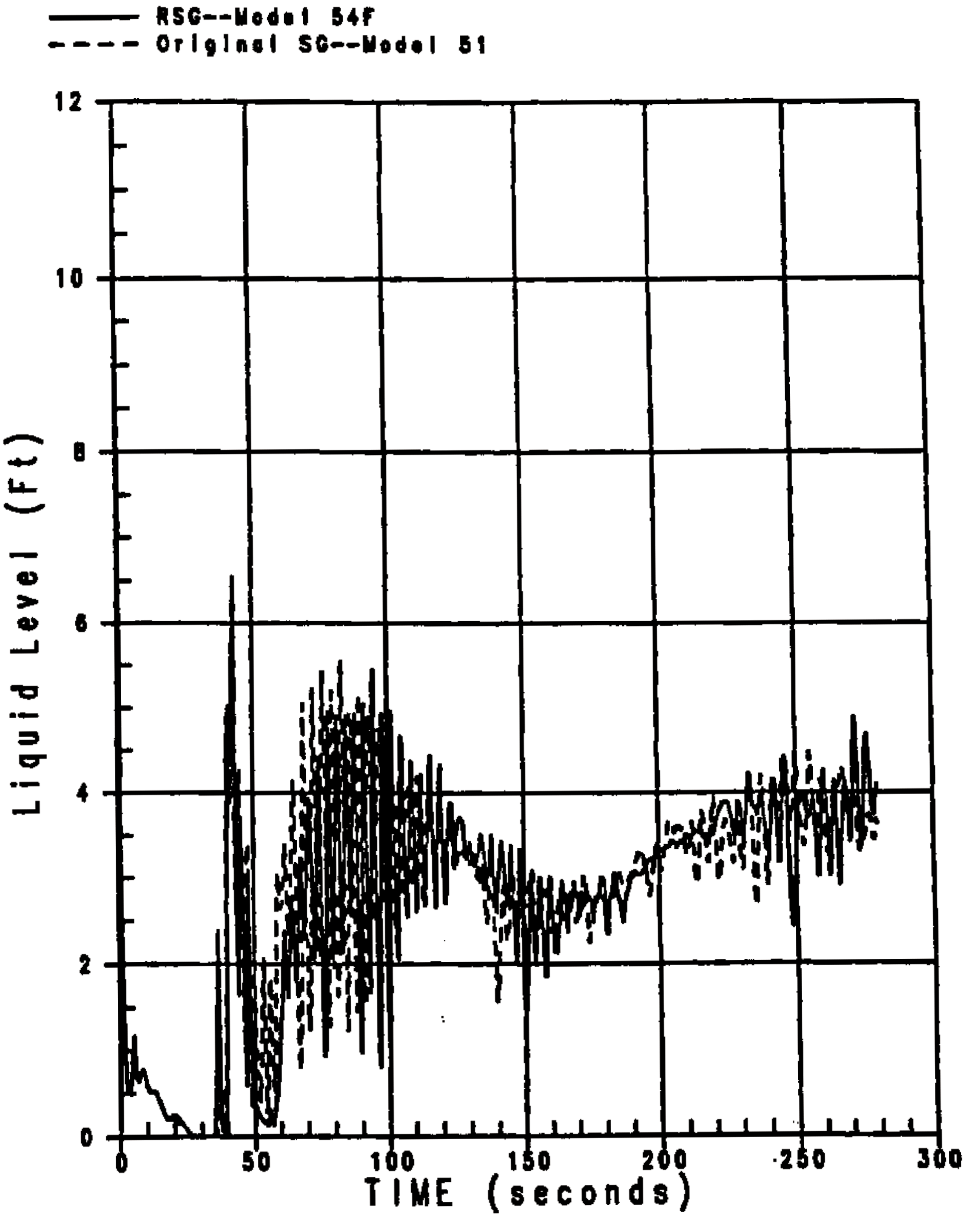
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FAREY REFERENCE SPLIT BREAK –
CD = 1.0 PEAK CLADDING TEMPERATURE
HOT ROD

FIGURE 15.4-15N



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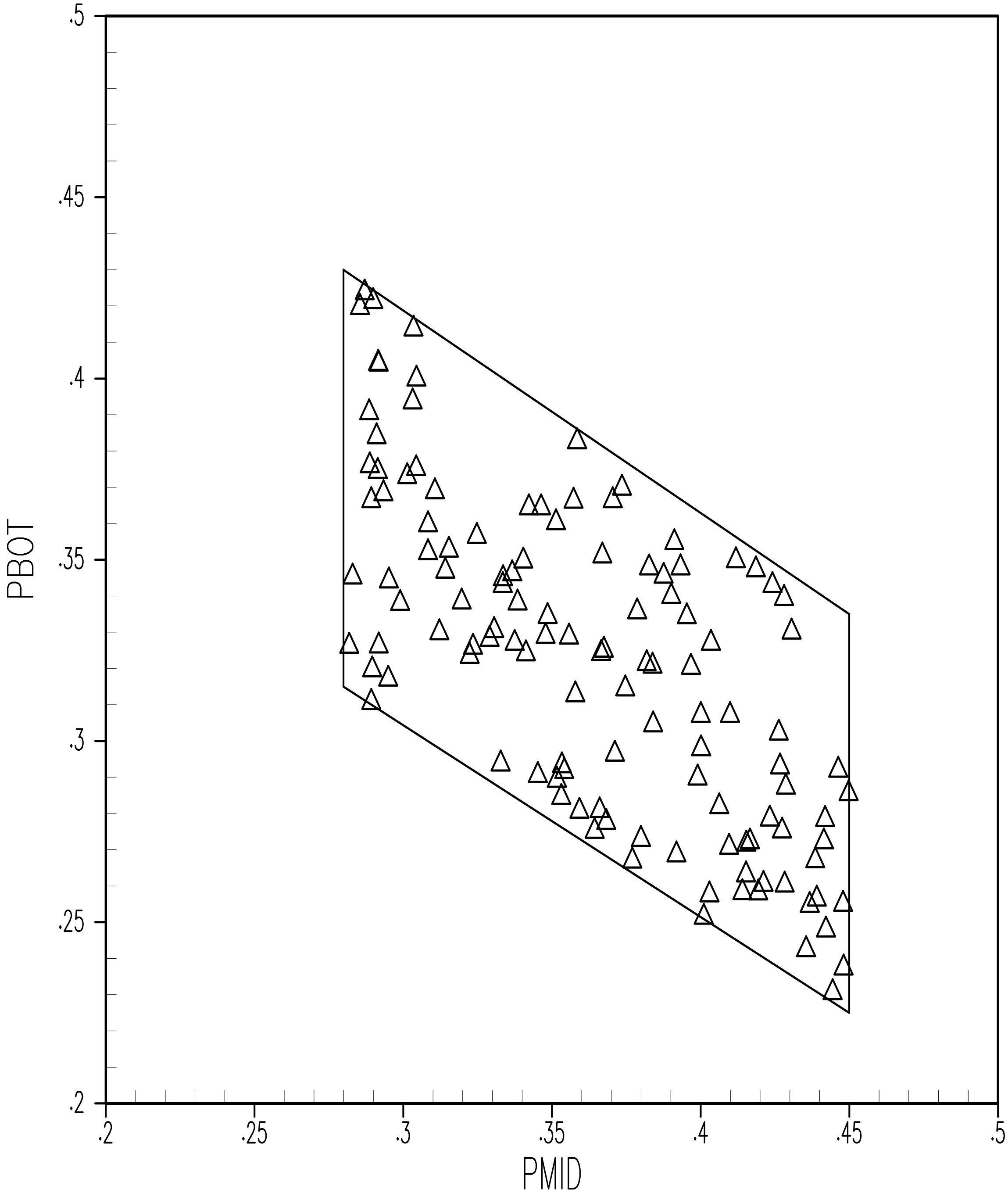
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FARLEY REFERENCE SPLIT BREAK –
CD = 1.0 AVERAGE CORE LIQUID
LEVE – CH 10, 11, 12, 13

FIGURE 15.4-15P



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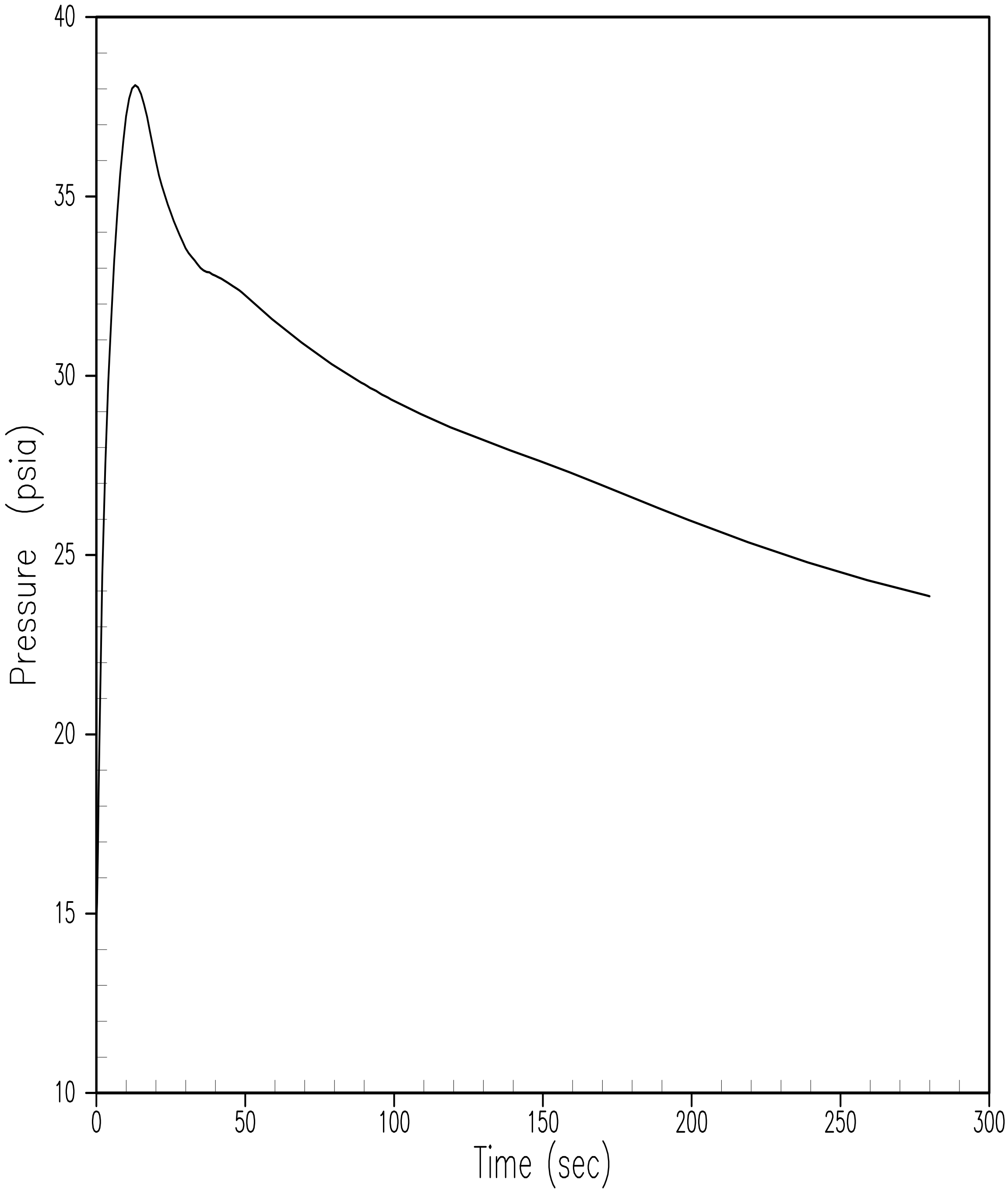
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PBOT/PMID LIMITS USED IN
BEST-ESTIMATE LOCA ANALYSIS

FIGURE 15.4-16



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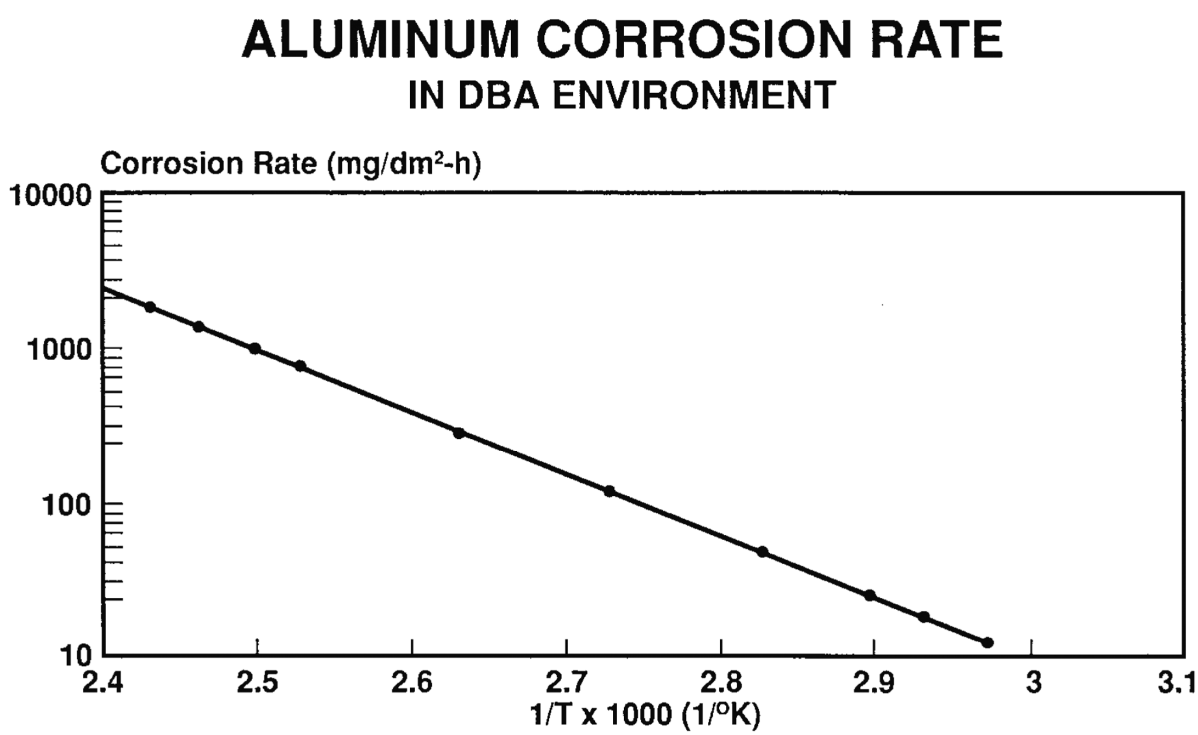
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

COCO CONTAINMENT PRESSURE USED
IN BEST-ESTIMATE LOCA ANALYSIS

FIGURE 15.4-17



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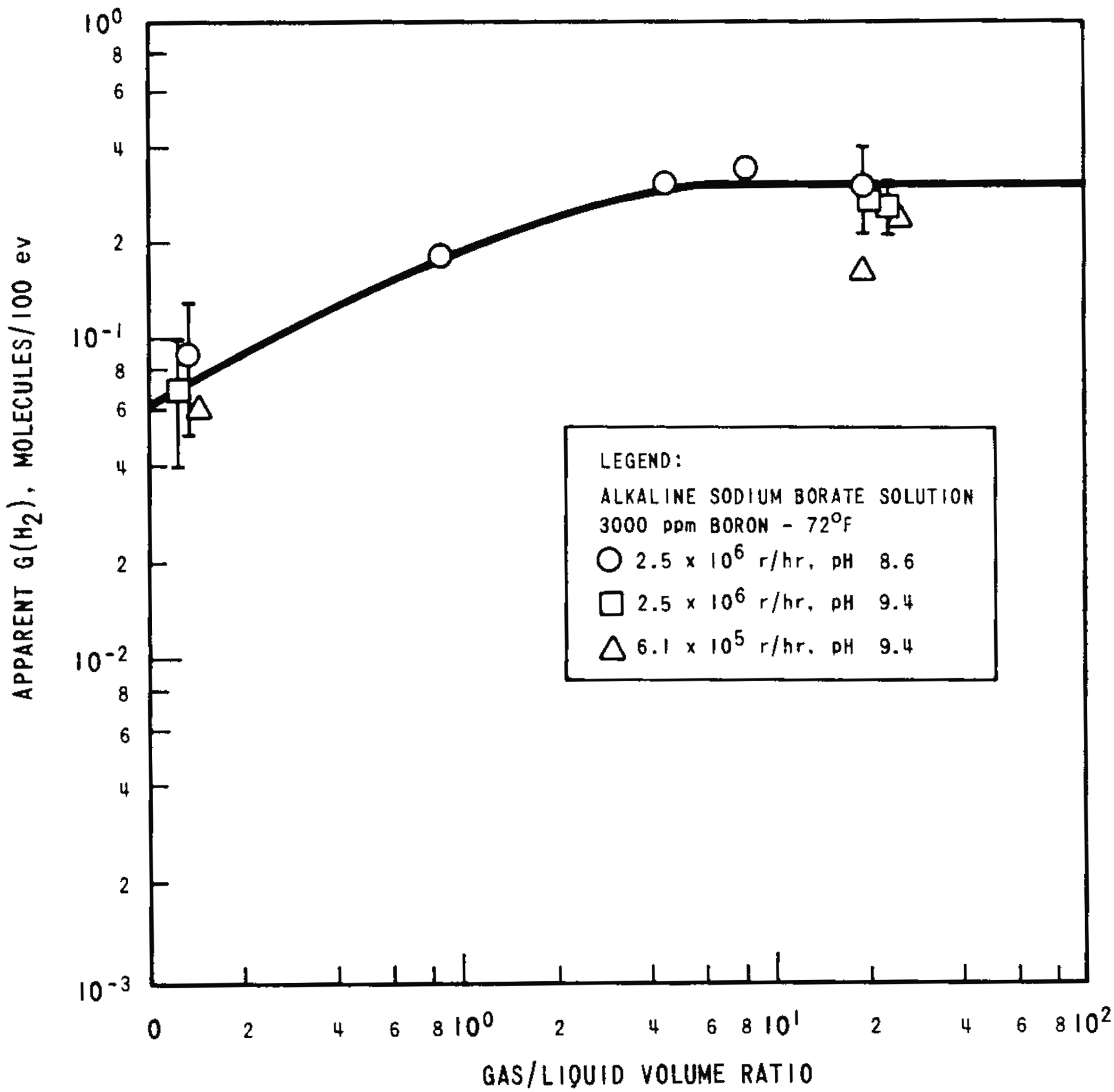
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALUMINUM CORROSION IN DBA ENVIRONMENT
		FIGURE 15.4-18



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RESULTS OF WESTINGHOUSE
CAPSULE IRRADIATION TESTS

FIGURE 15.4-19

THIS FIGURE HAS BEEN DELETED.

15

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK GAMMA DOSE

FIGURE 15.4-20

THIS FIGURE HAS BEEN DELETED.

15

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NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK BETA DOSE

FIGURE 15.4-21

THIS FIGURE HAS BEEN DELETED.

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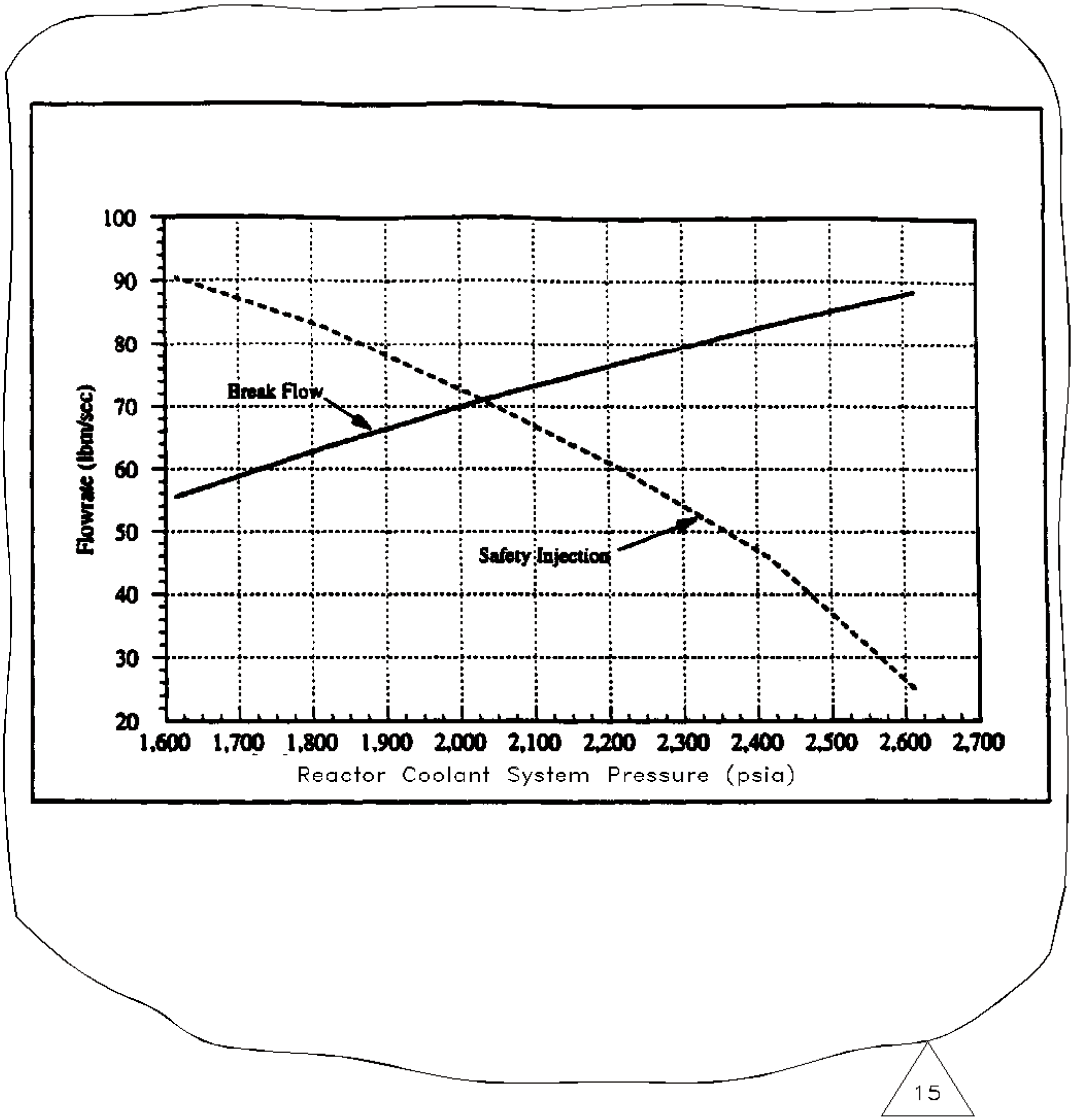
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK THYROID DOSE

FIGURE 15.4-22



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

BREAK AND INJECTED MASS FLOWS

FIGURE 15.4-23

THIS FIGURE HAS BEEN DELETED

15

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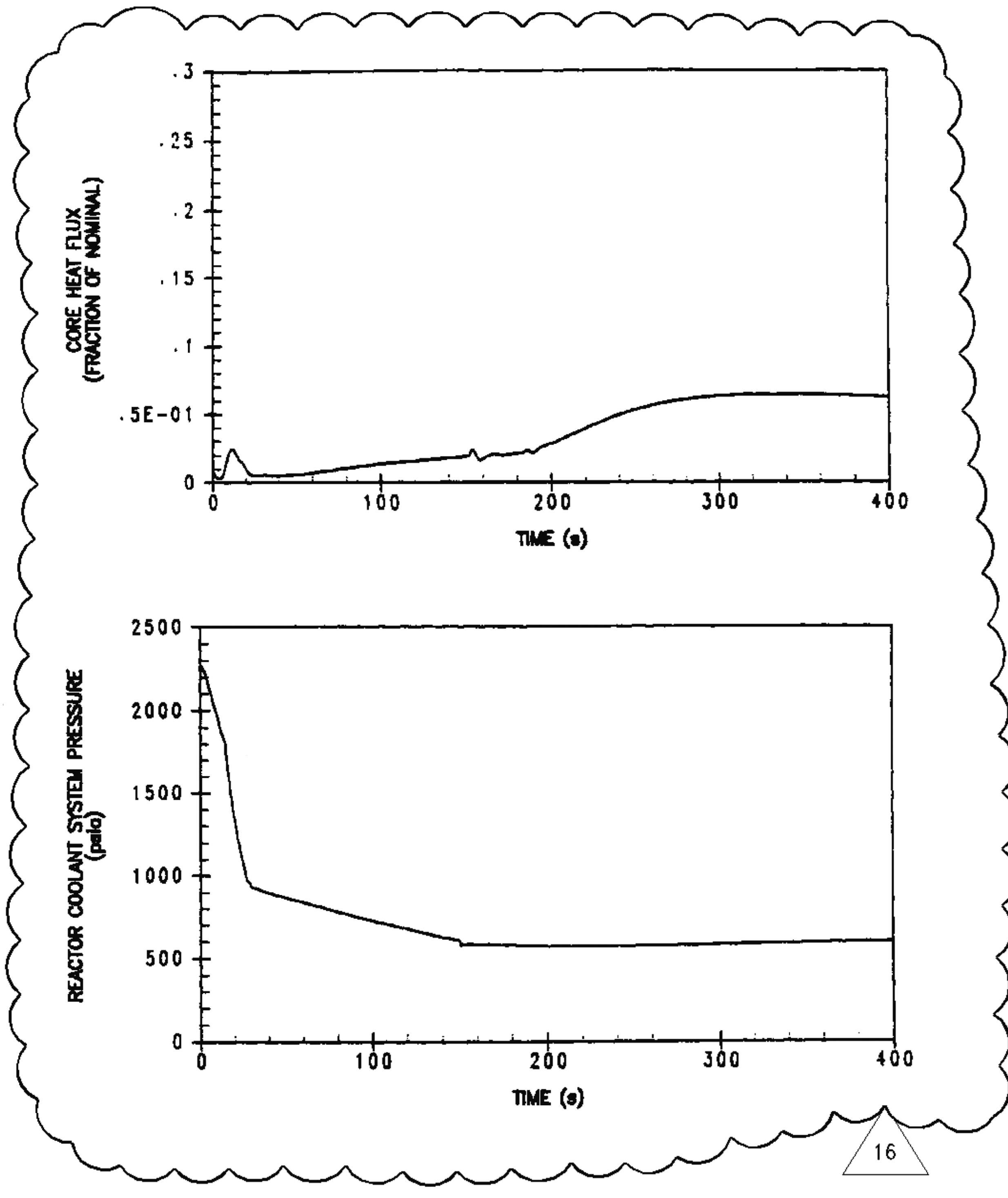
STEAM GENERATOR TUBE RUPTURE –
TYPICAL PRIMARY AND SECONDARY SYSTEM
PRESSURE AS A FUNCTION OF TIME

FIGURE 15.4-24



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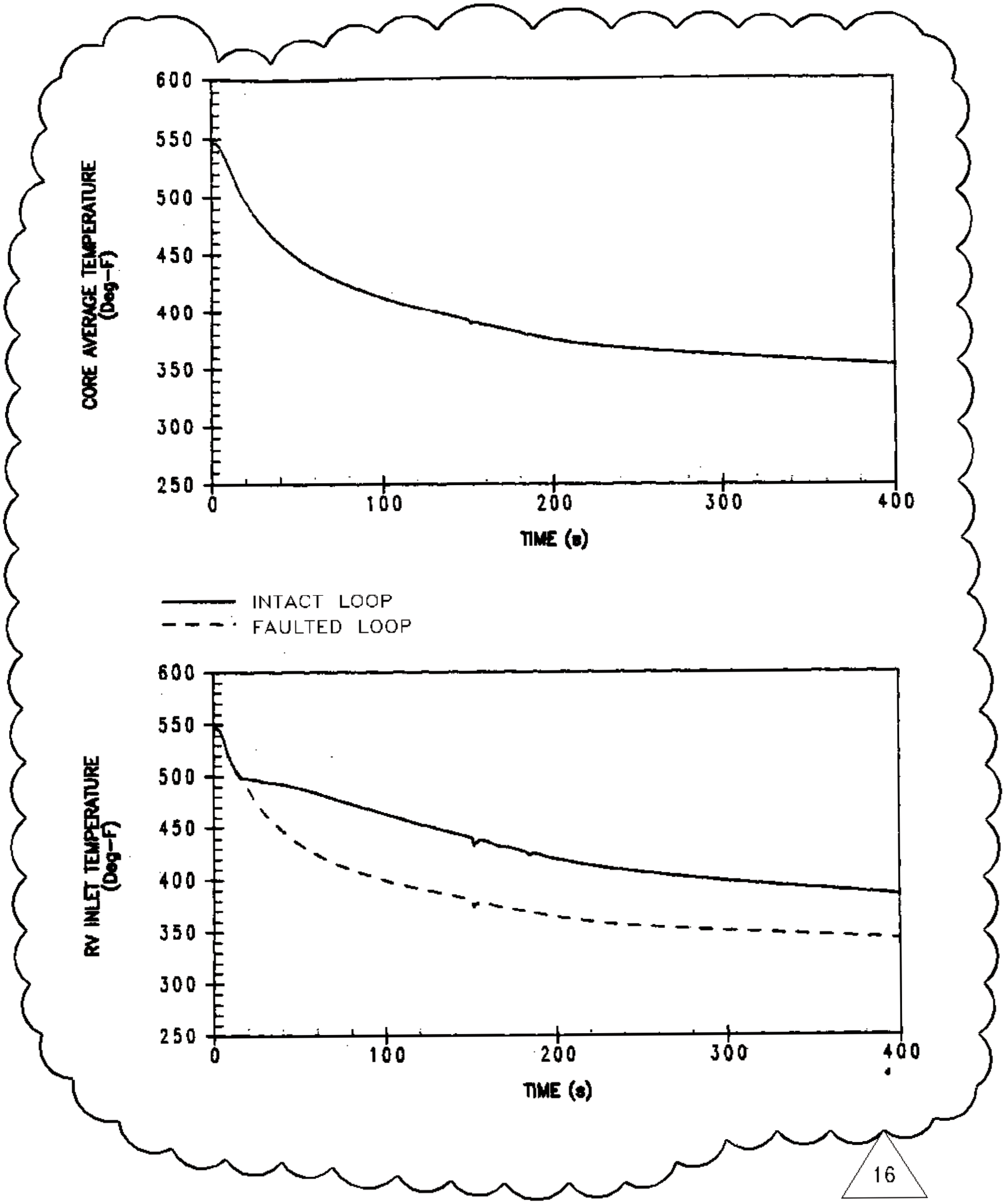
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-28 (SHEET 1 OF 3)



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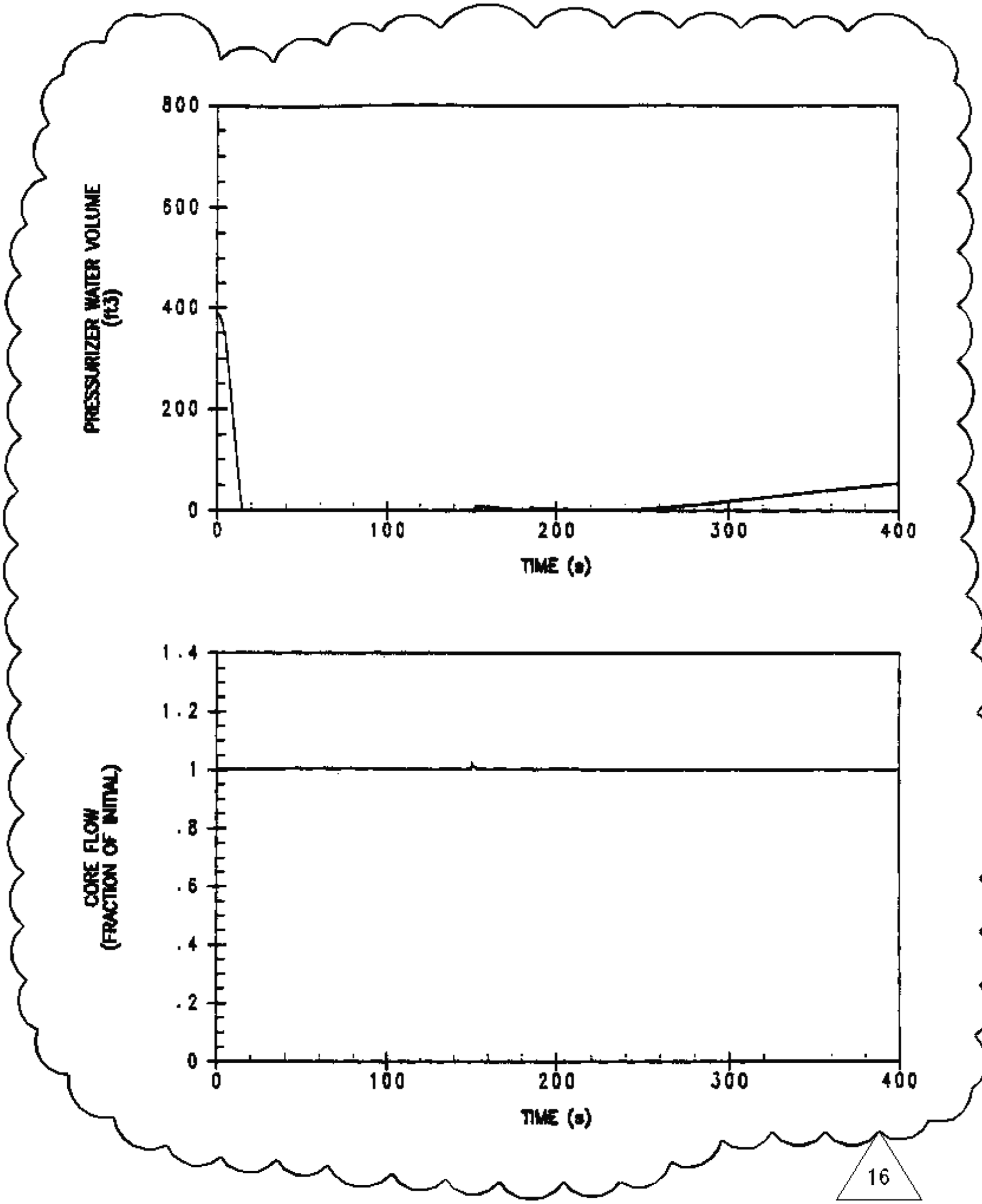
STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-28 (SHEET 2 OF 3)



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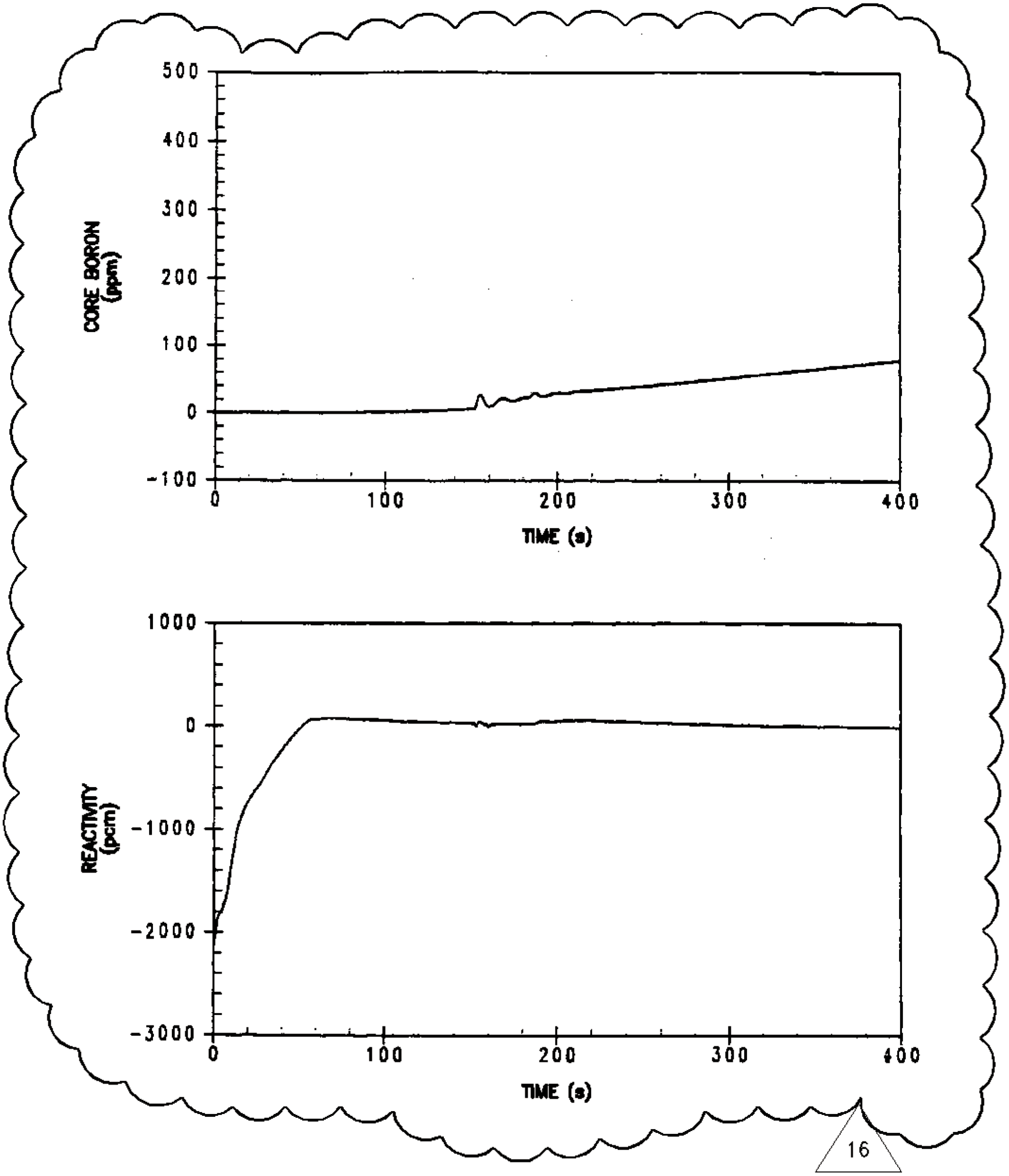
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-28 (SHEET 3 OF 3)



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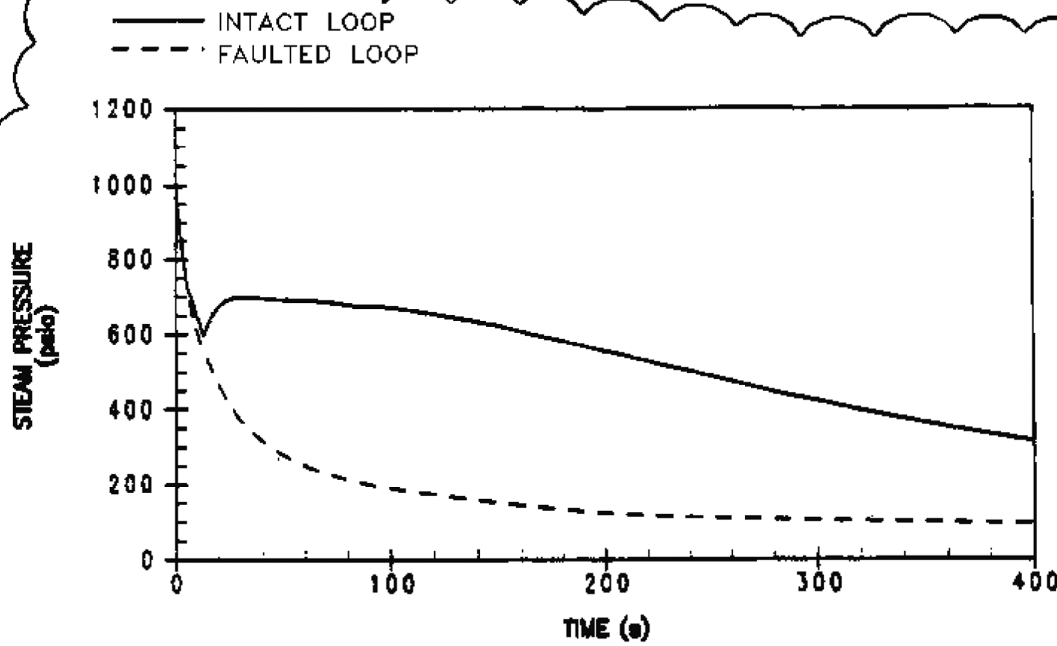
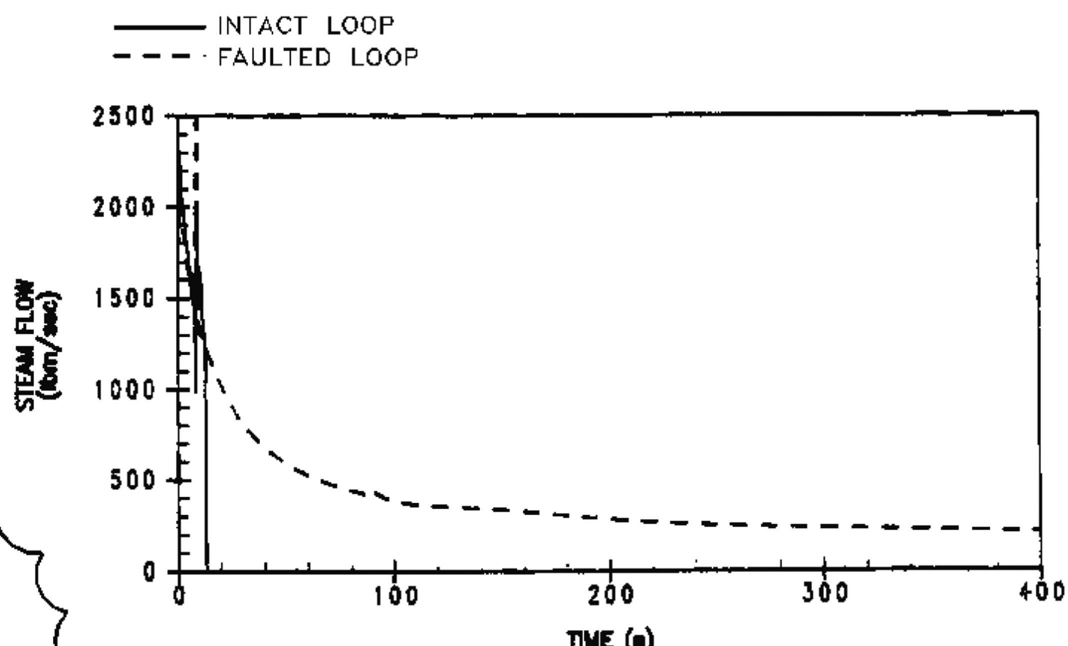
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-29 (SHEET 1 OF 3)





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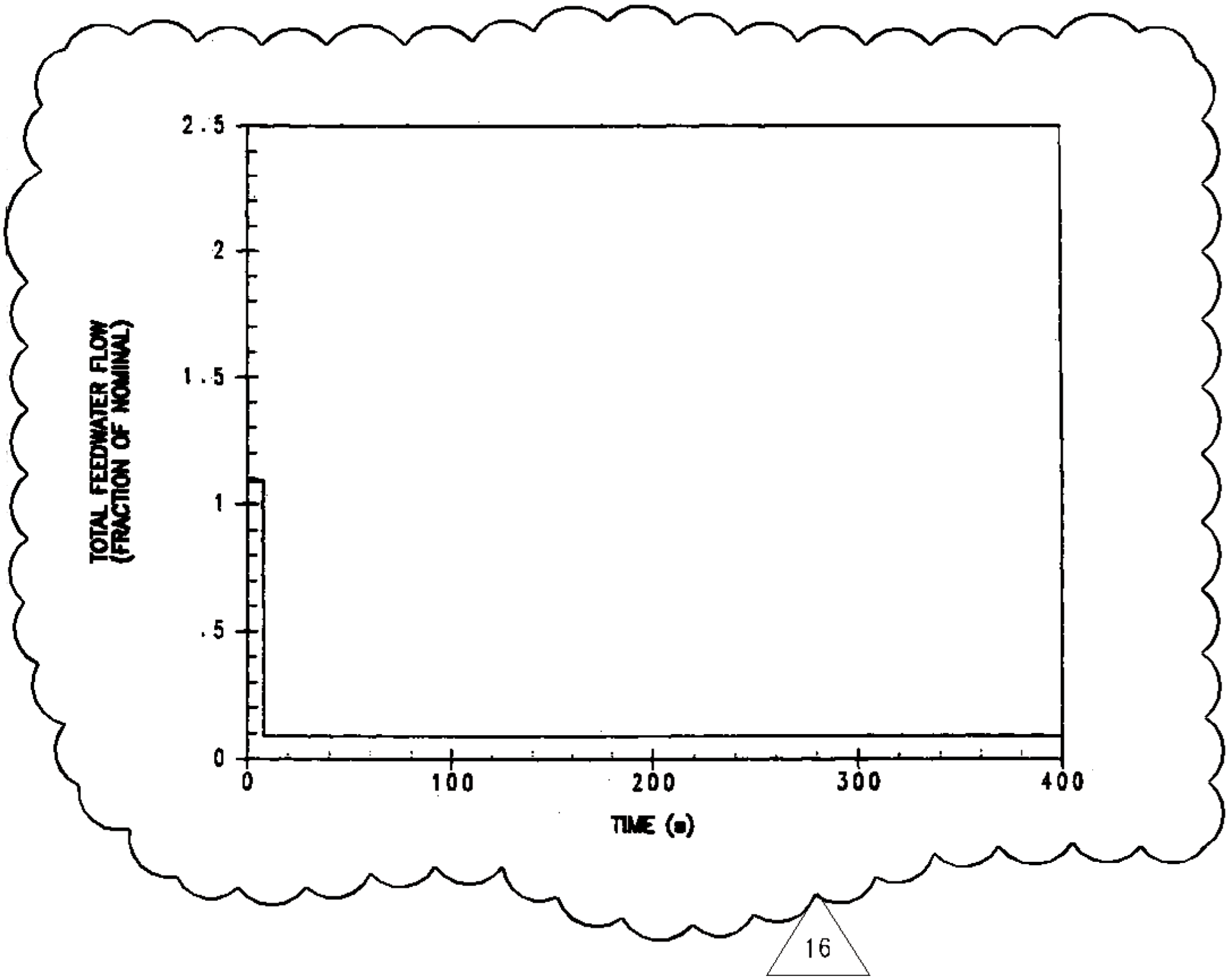
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-29 (SHEET 2 OF 3)



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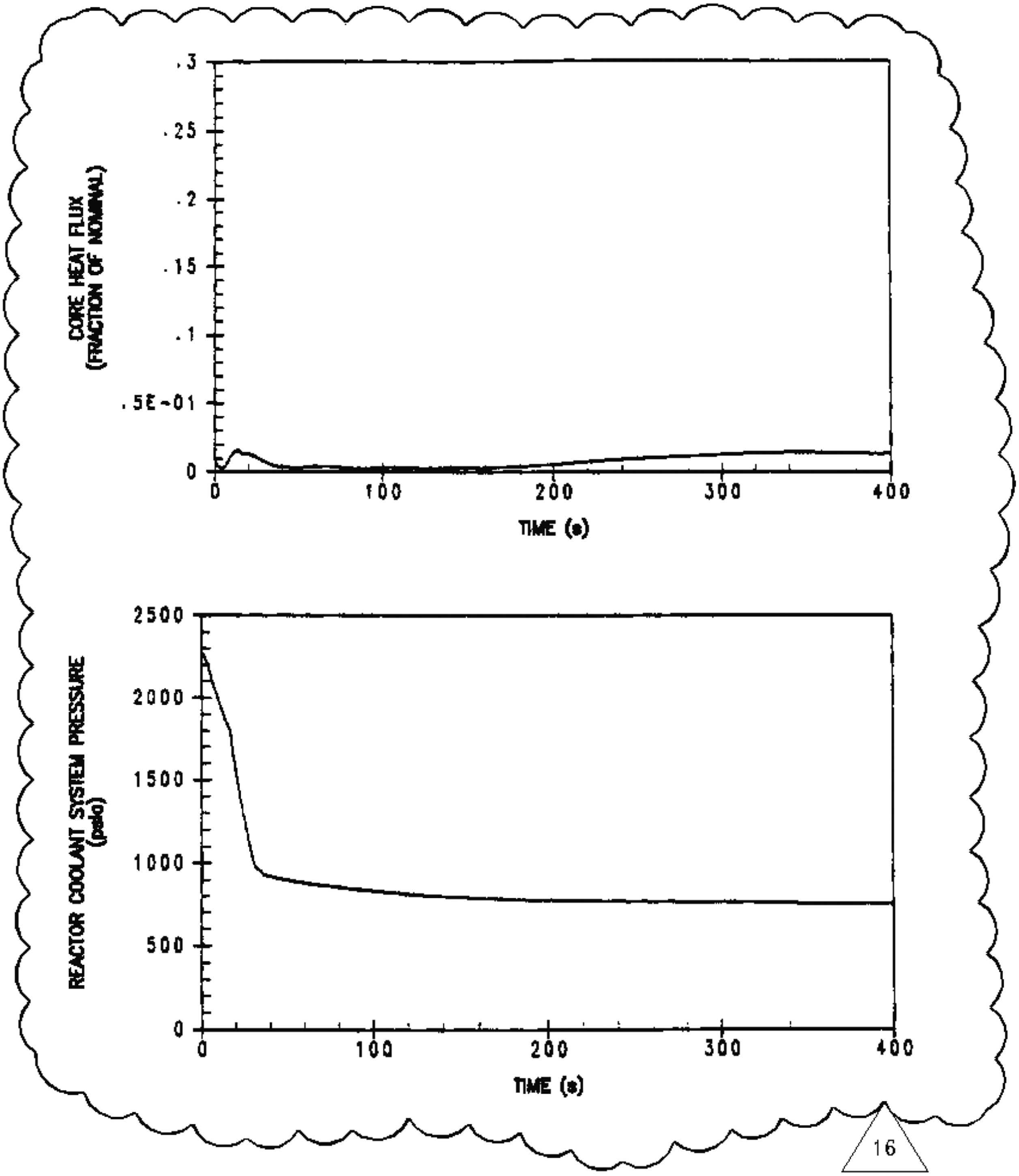
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITH OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-29 (SHEET 3 OF 3)



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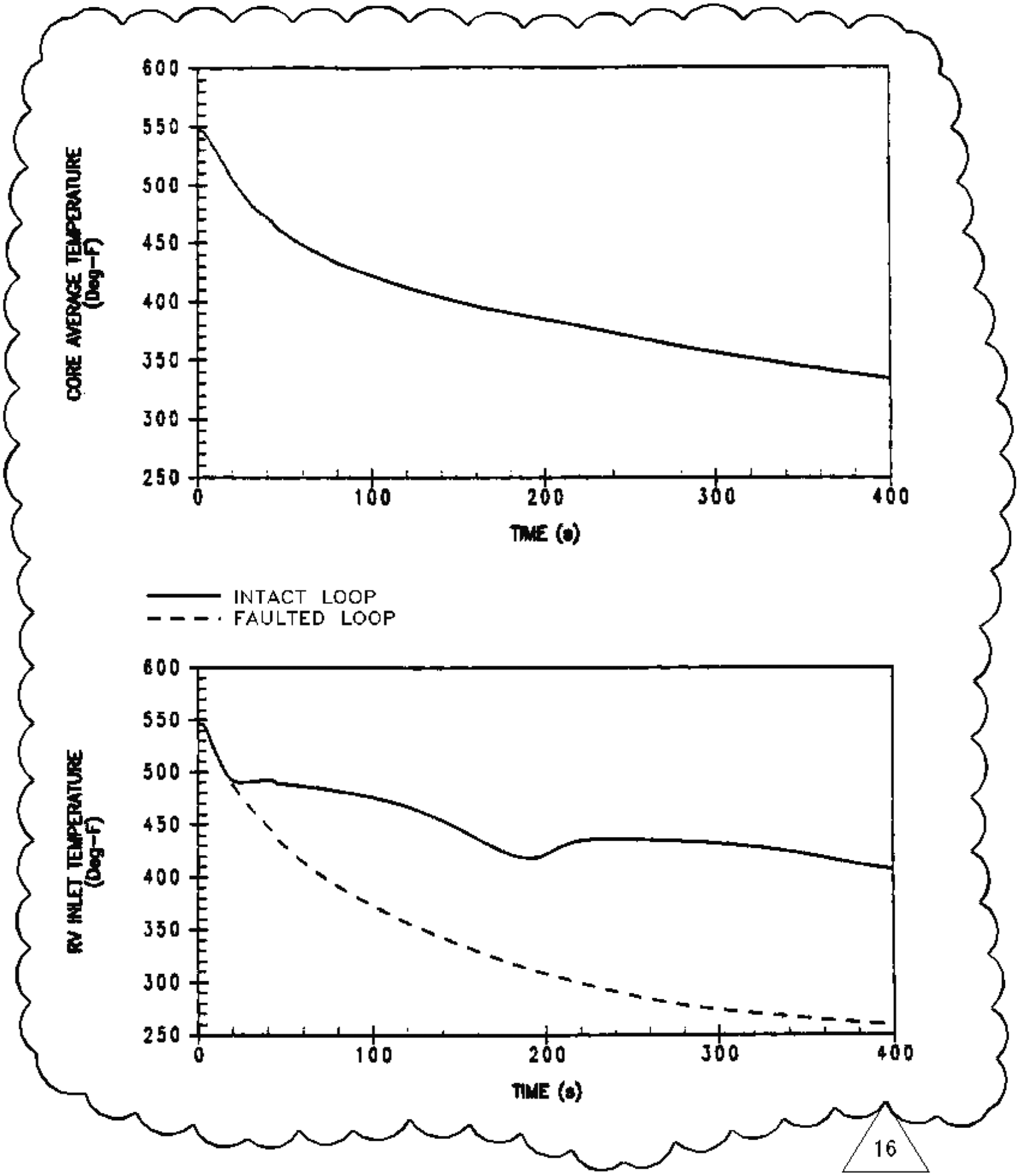
SOUTHERN COMPANY *Energy to Serve Your World®* JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM LINE BREAK TRANSIENT WITHOUT OFFSITE POWER 1.069 ft^2 DOUBLE-ENDED RUPTURE
	FIGURE 15.4-30 (SHEET 1 OF 3)



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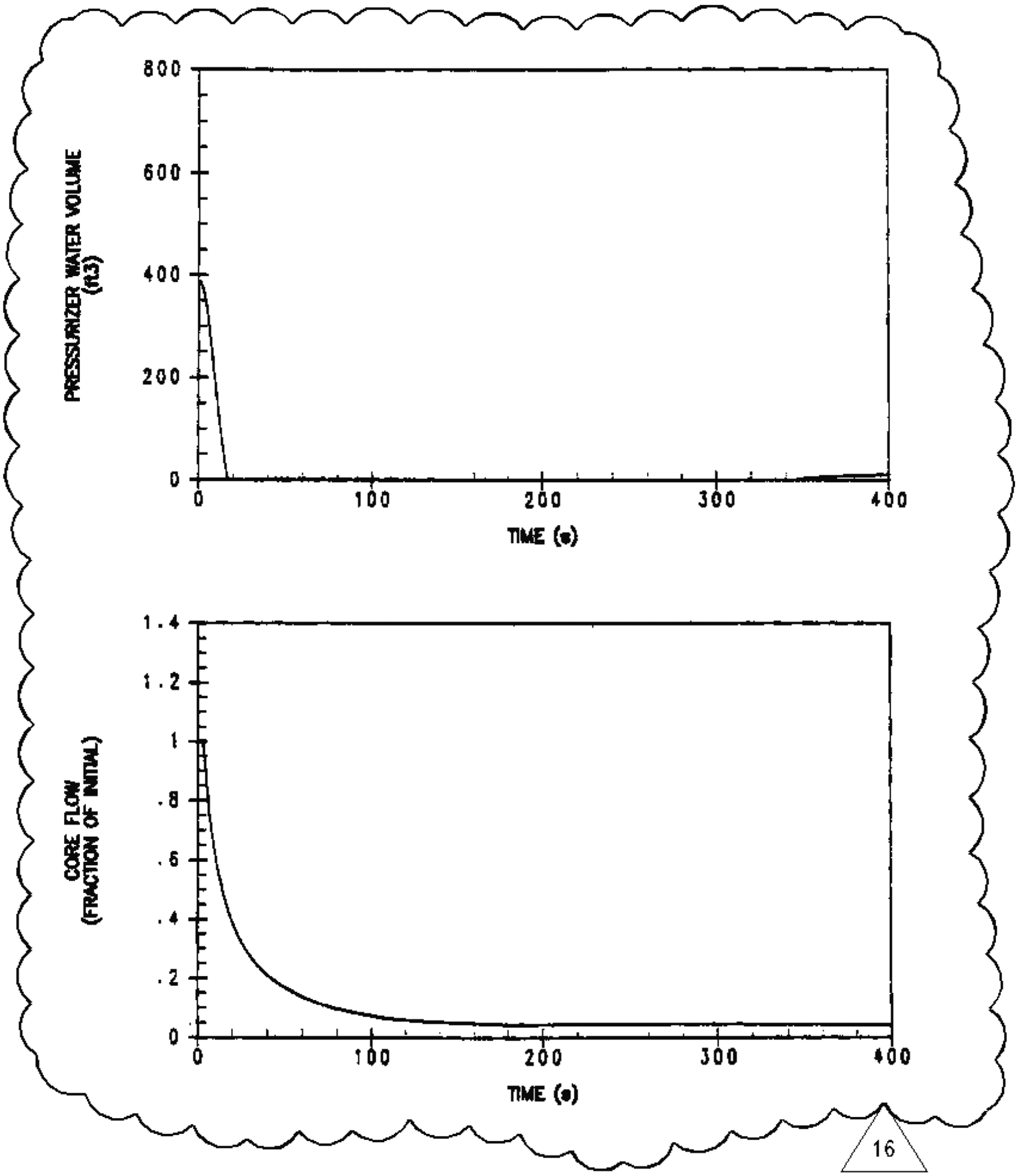
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT WITHOUT OFFSITE POWER 1.069 ft^2 DOUBLE-ENDED RUPTURE

FIGURE 15.4-30 (SHEET 2 OF 3)



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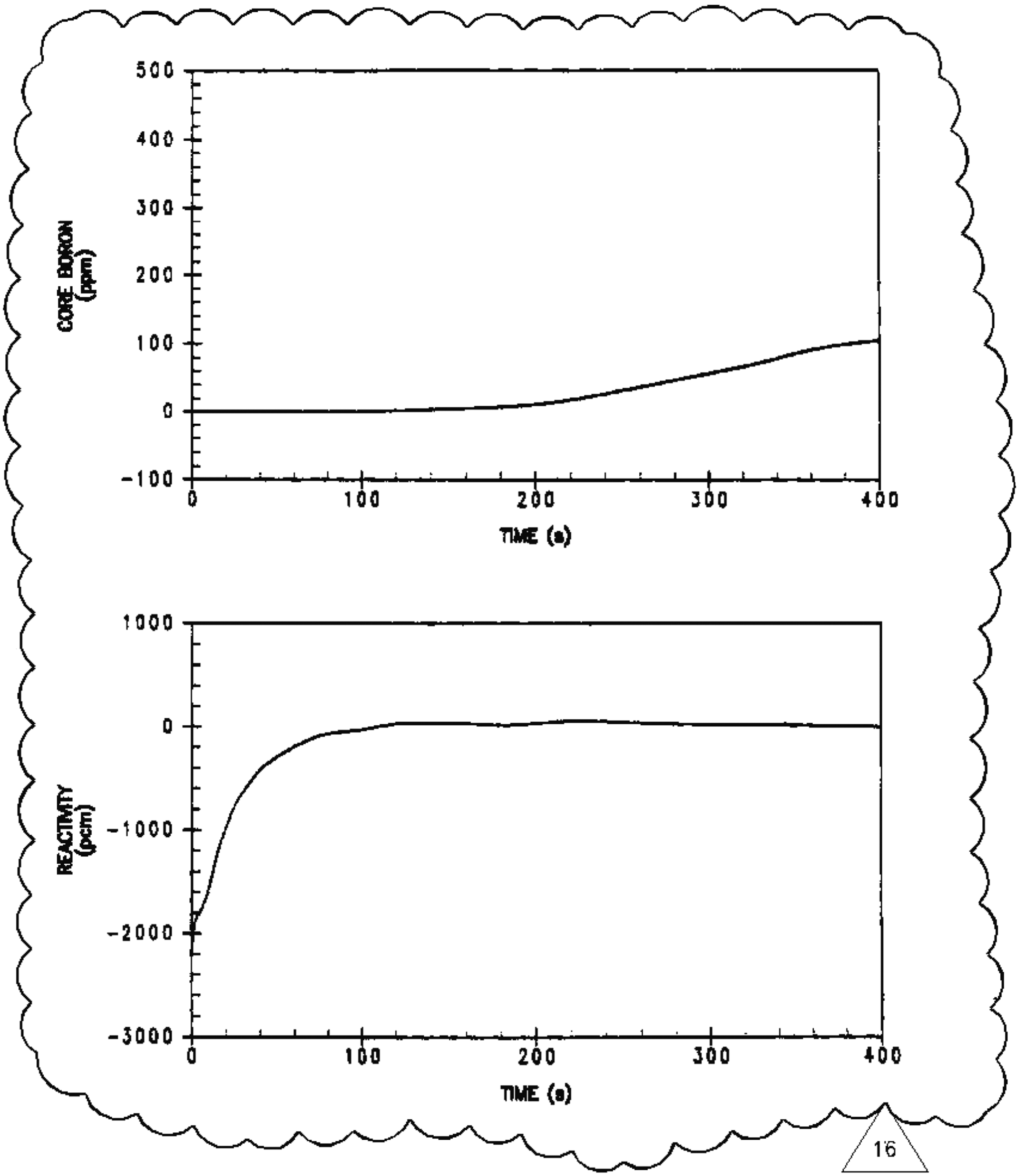
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT WITHOUT OFFSITE POWER 1.069 ft^2 DOUBLE-ENDED RUPTURE

FIGURE 15.4-30 (SHEET 3 OF 3)



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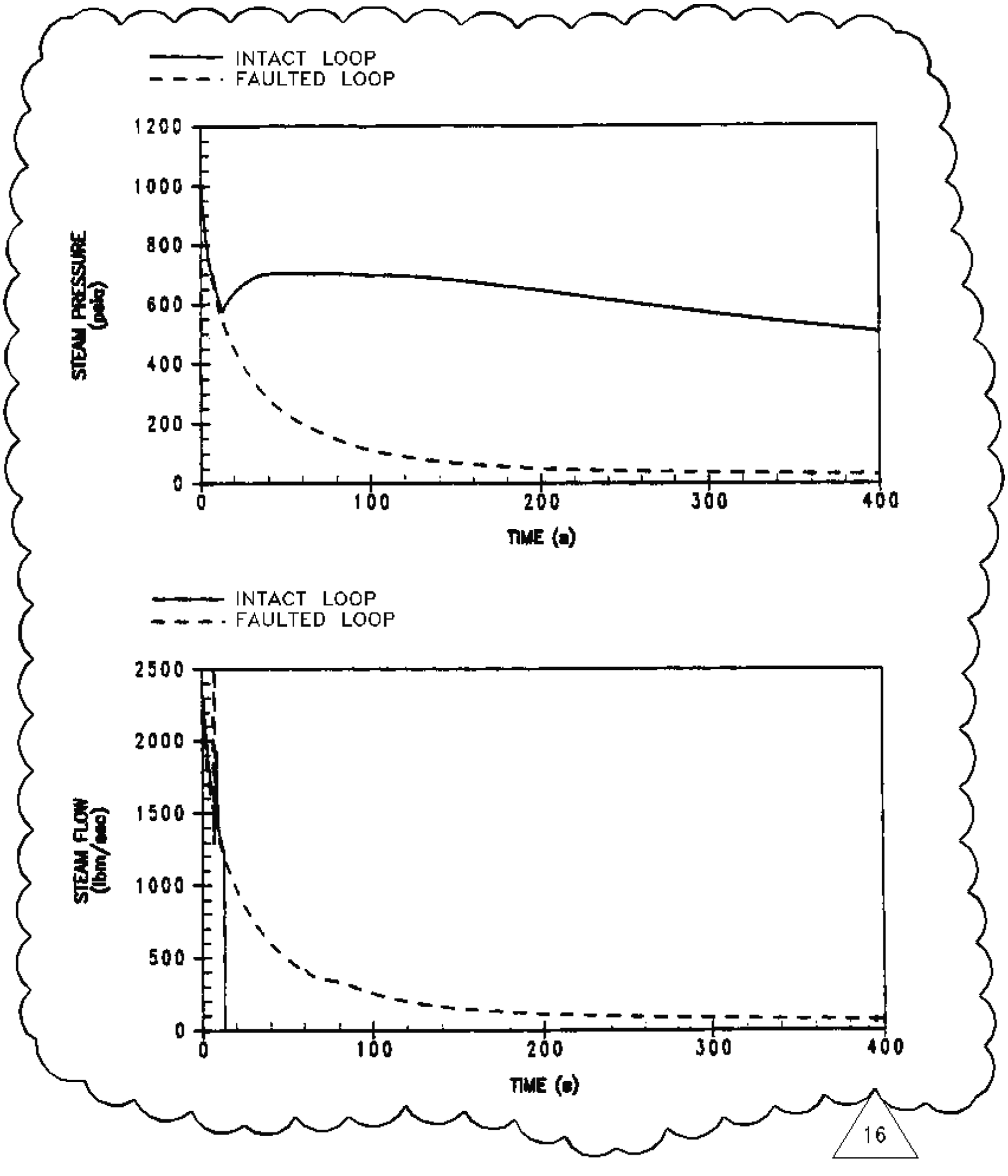
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM LINE BREAK TRANSIENT
WITHOUT OFFSITE POWER 1.069 ft^2
DOUBLE-ENDED RUPTURE

FIGURE 15.4-31 (SHEET 1 OF 3)



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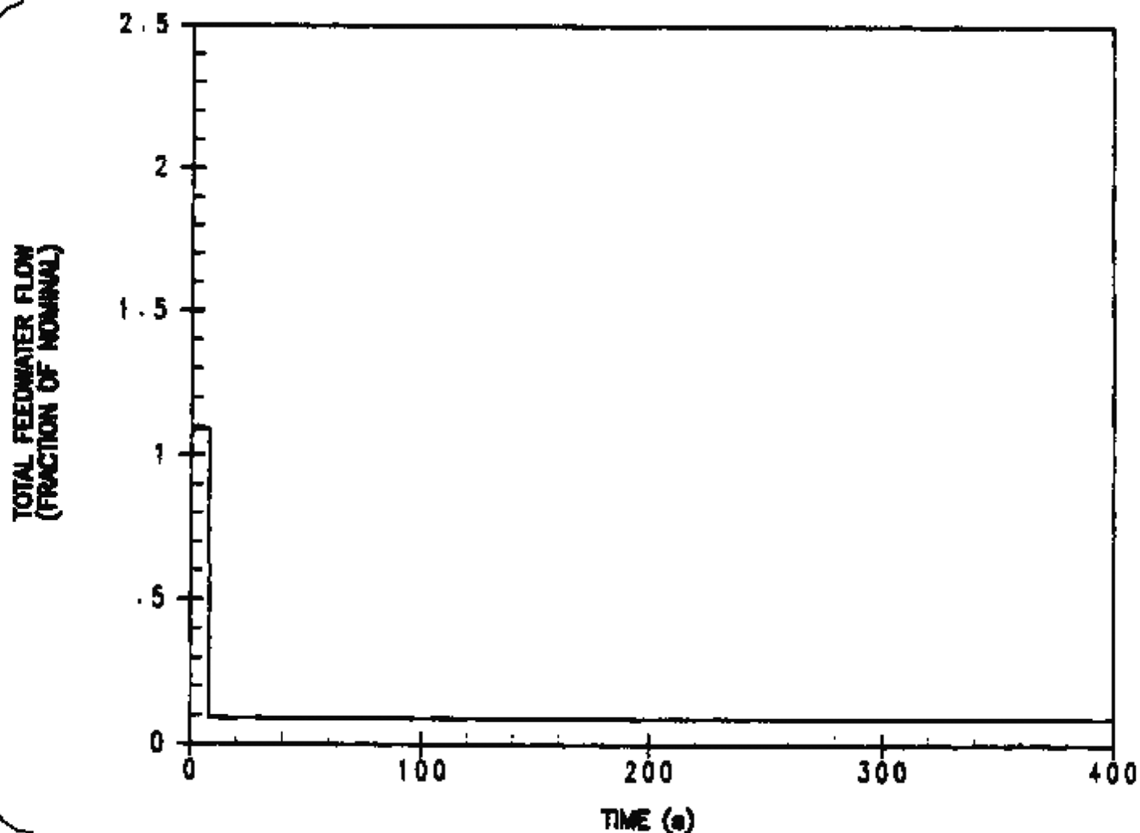
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM LINE BREAK TRANSIENT WITHOUT OFFSITE POWER 1.069 ft^2 DOUBLE-ENDED RUPTURE
		FIGURE 15.4-31 (SHEET 2 OF 3)



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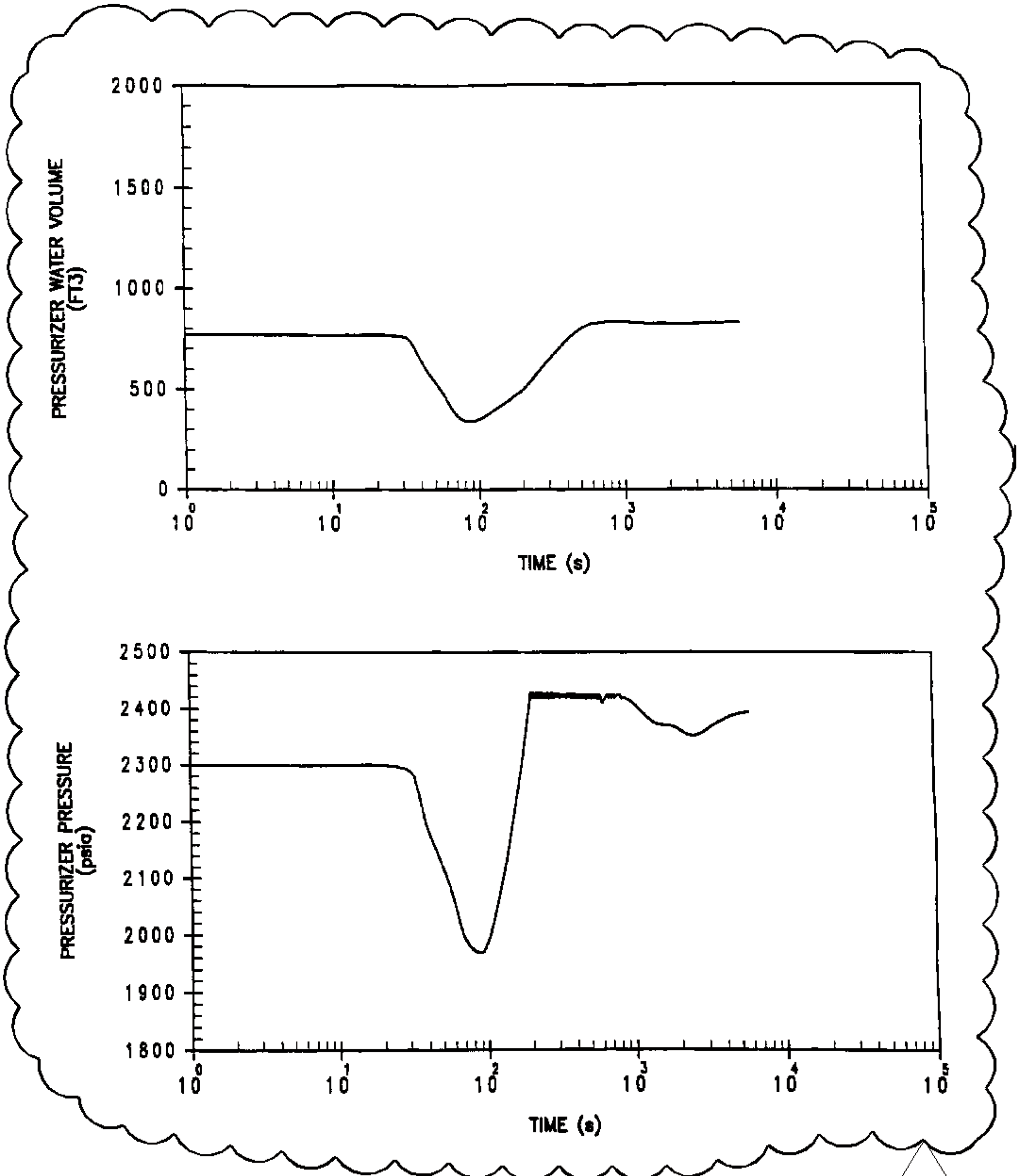
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	STEAM LINE BREAK TRANSIENT WITHOUT OFFSITE POWER 1.069 ft^2 DOUBLE-ENDED RUPTURE
		FIGURE 15.4-31 (SHEET 3 OF 3)



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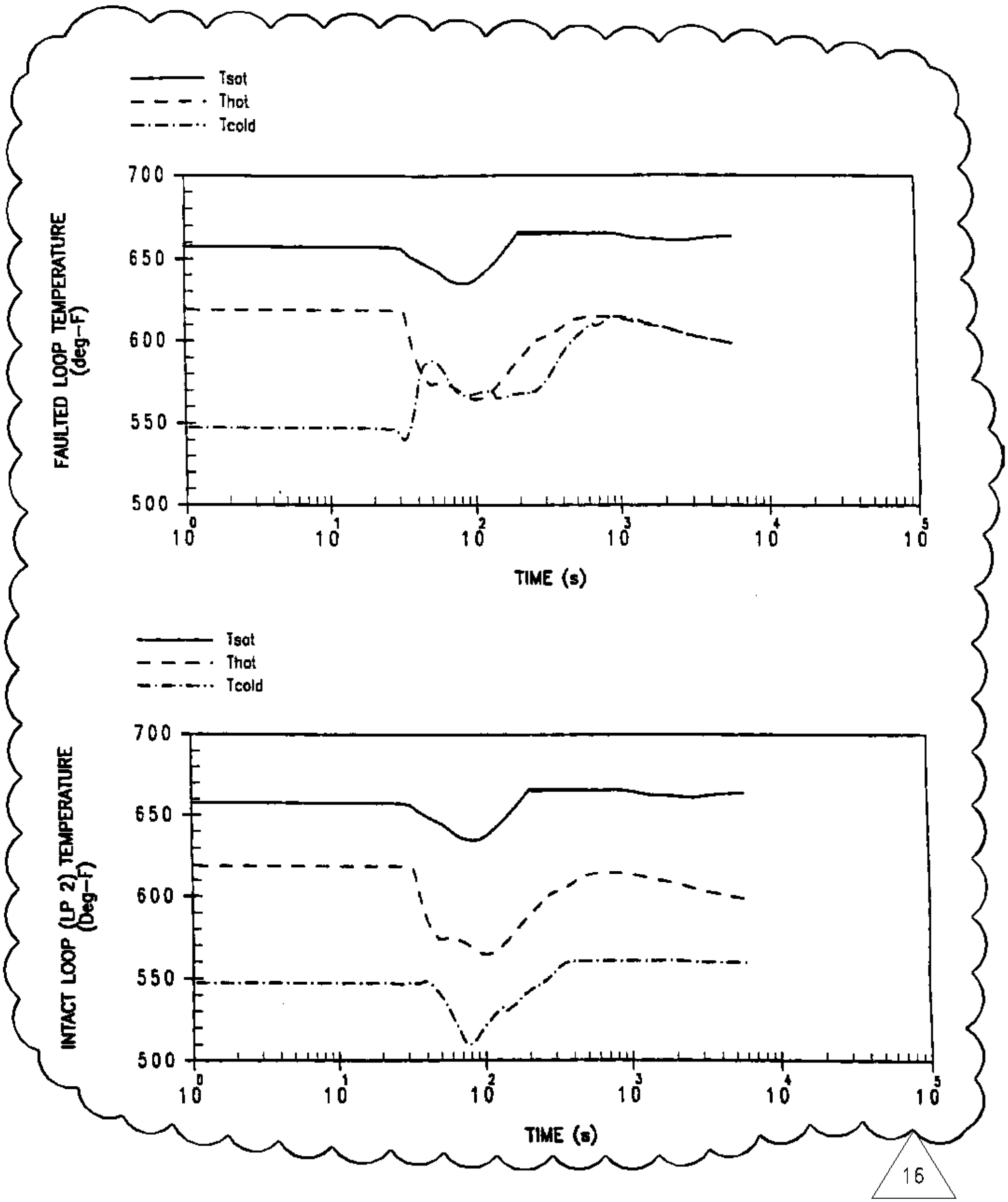
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MAIN RUPTURE OF A MAIN FEEDWATER PIPE
WITHOUT OFFSITE POWER – PRESSURIZER PRESSURE
AND WATER VOLUME VERSUS TIME
(CASE A)

FIGURE 15.4-32 (SHEET 1 OF 4)



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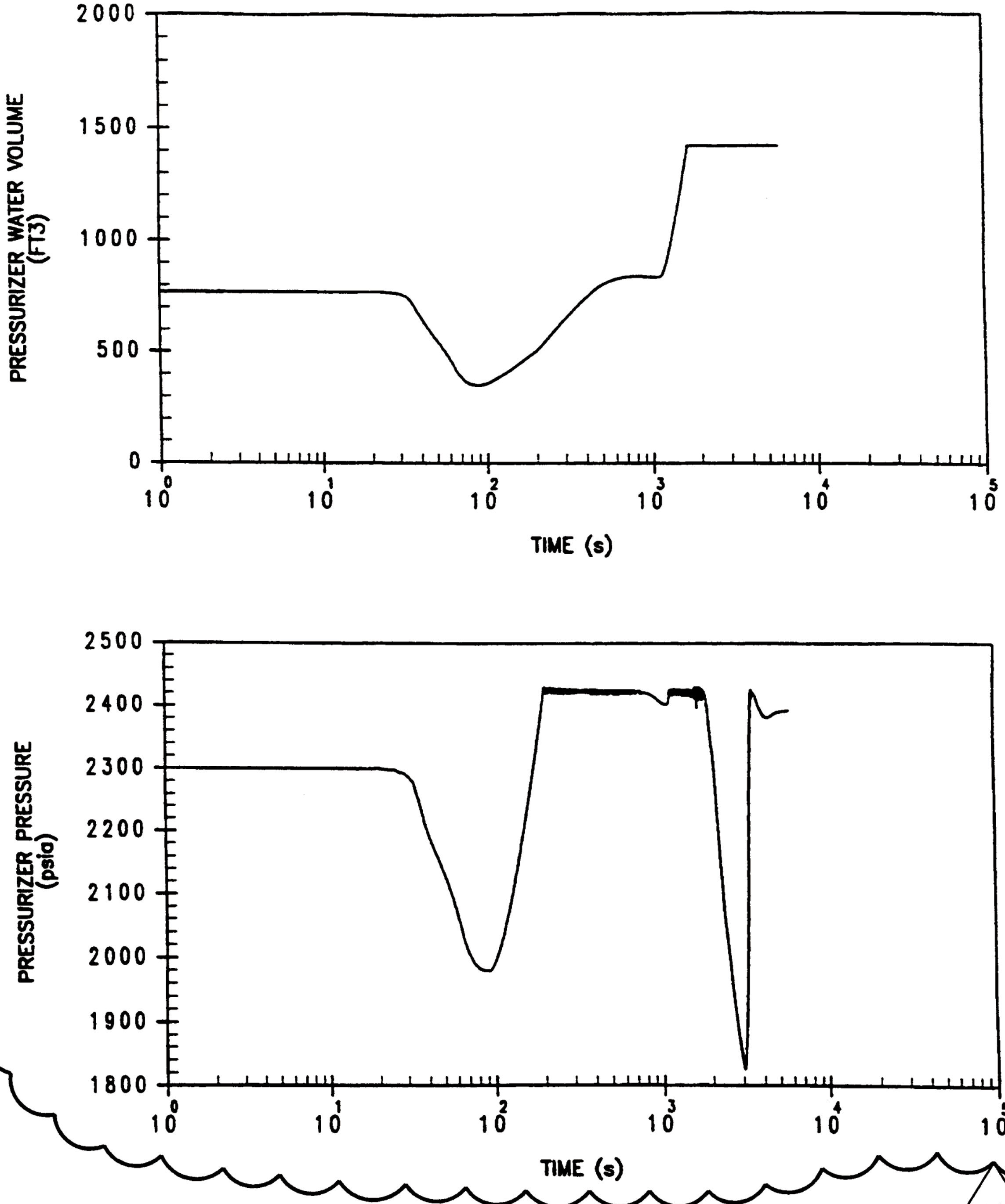
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MAIN RUPTURE OF A MAIN FEEDWATER PIPE WITHOUT OFFSITE POWER – INTACT AND FAULTED LOOP TEMPERATURE VERSUS TIME (CASE A)

FIGURE 15.4-32 (SHEET 2 OF 4)



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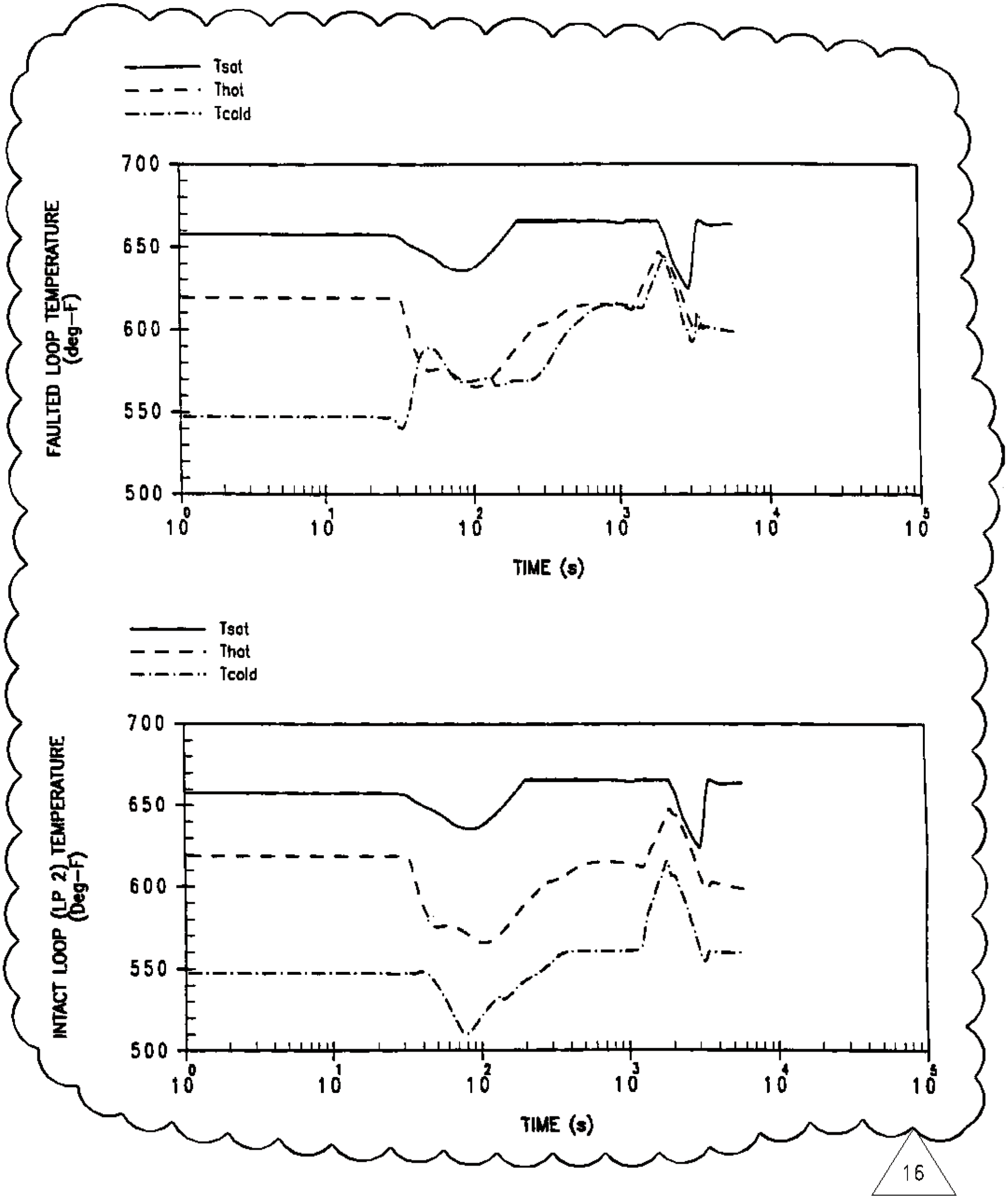
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MAIN RUPTURE OF A MAIN FEEDWATER PIPE
WITHOUT OFFSITE POWER – PRESSURIZER PRESSURE
AND WATER VOLUME VERSUS TIME
(CASE B)

FIGURE 15.4-32 (SHEET 3 OF 4)



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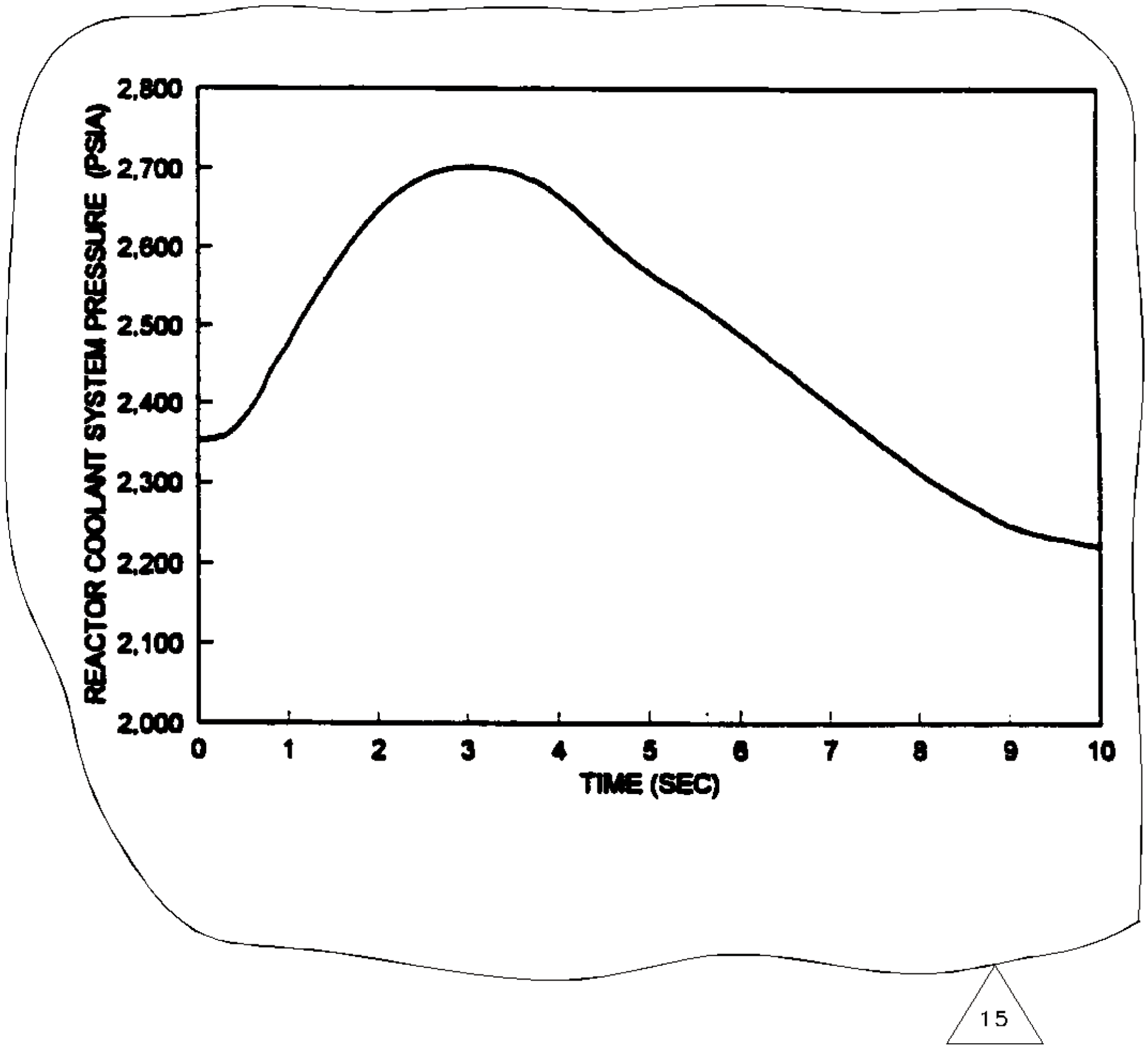
SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	MAIN RUPTURE OF A MAIN FEEDWATER PIPE WITHOUT OFFSITE POWER – INTACT AND FAULTED LOOP TEMPERATURE VERSUS TIME (CASE B)
		FIGURE 15.4-32 (SHEET 4 OF 4)



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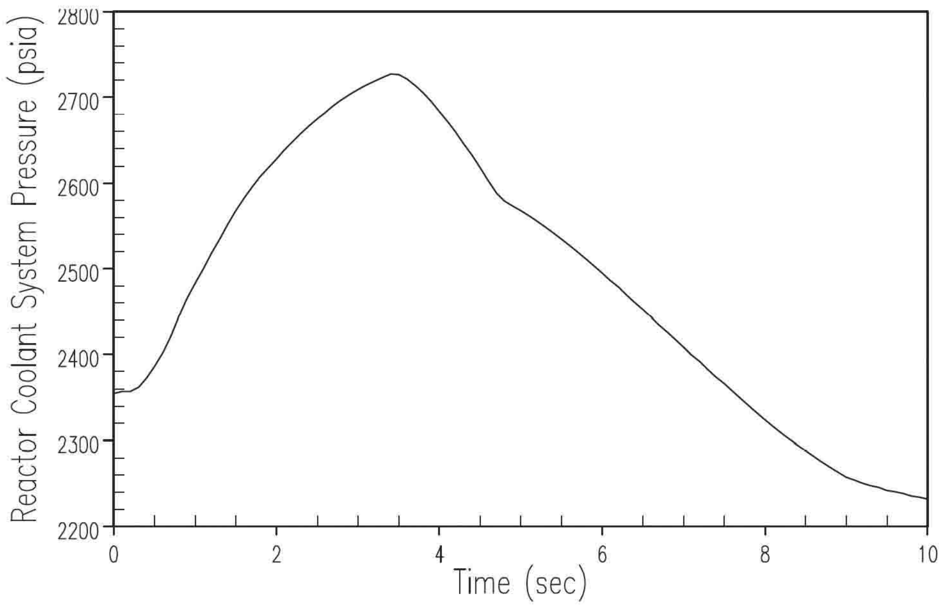
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS INITIALLY
OPERAITNG, ONE LOCKED ROTOR –
REACTOR COOLANT SYSTEM PRESSURE

FIGURE 15.4-33



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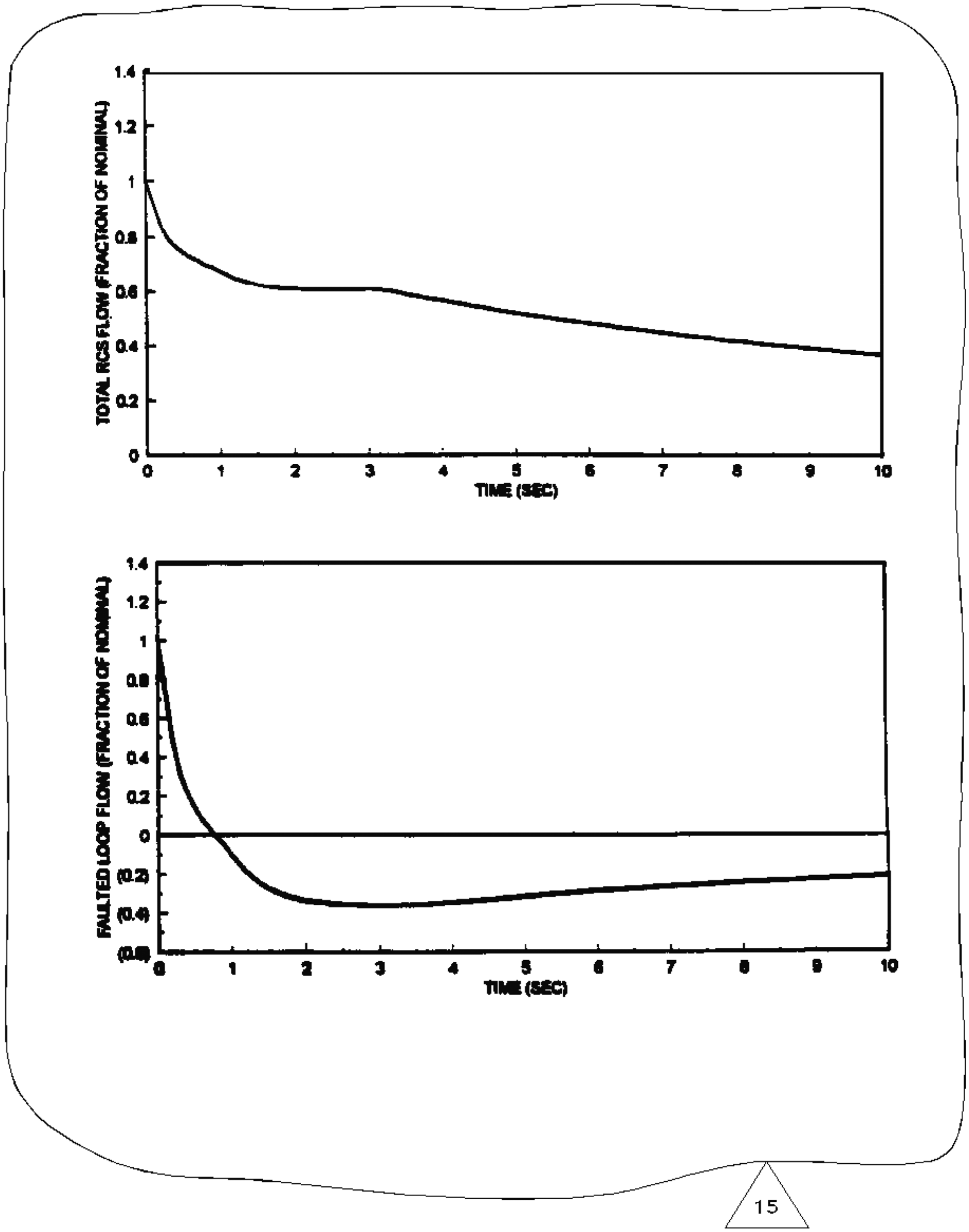
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

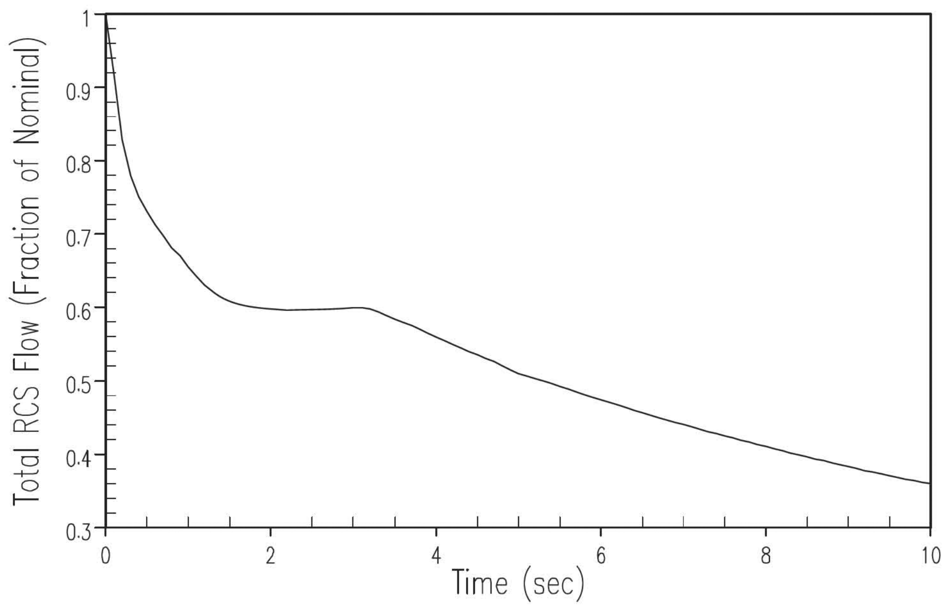
ALL LOOPS INITIALLY OPERATING, ONE LOCKED ROTOR –
RCS PRESSURE

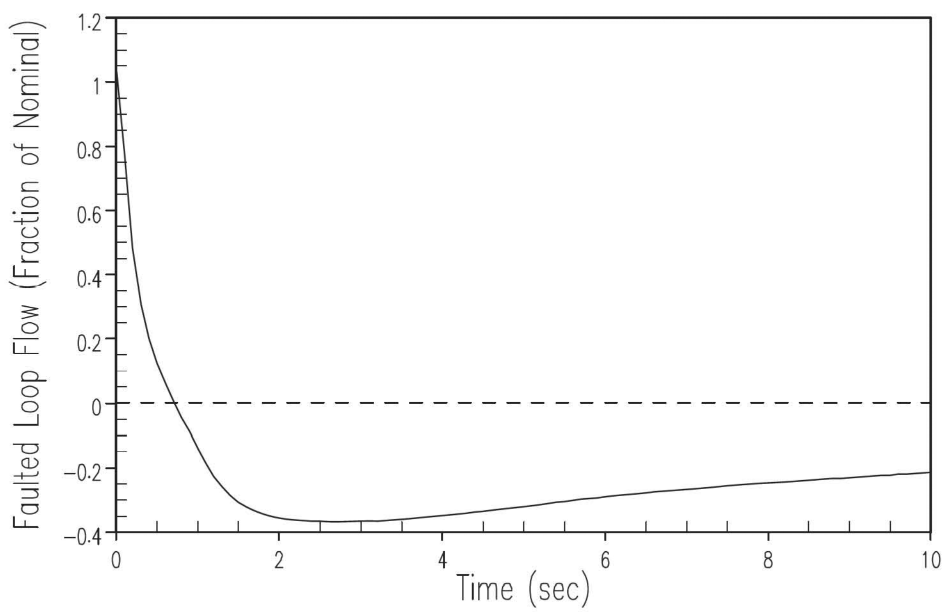
FIGURE 15.4-33A



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPERATING, ONE LOCKED ROTOR – TOTAL RCS FLOW AND FAULTED LOOP FLOW FIGURE 15.4-34





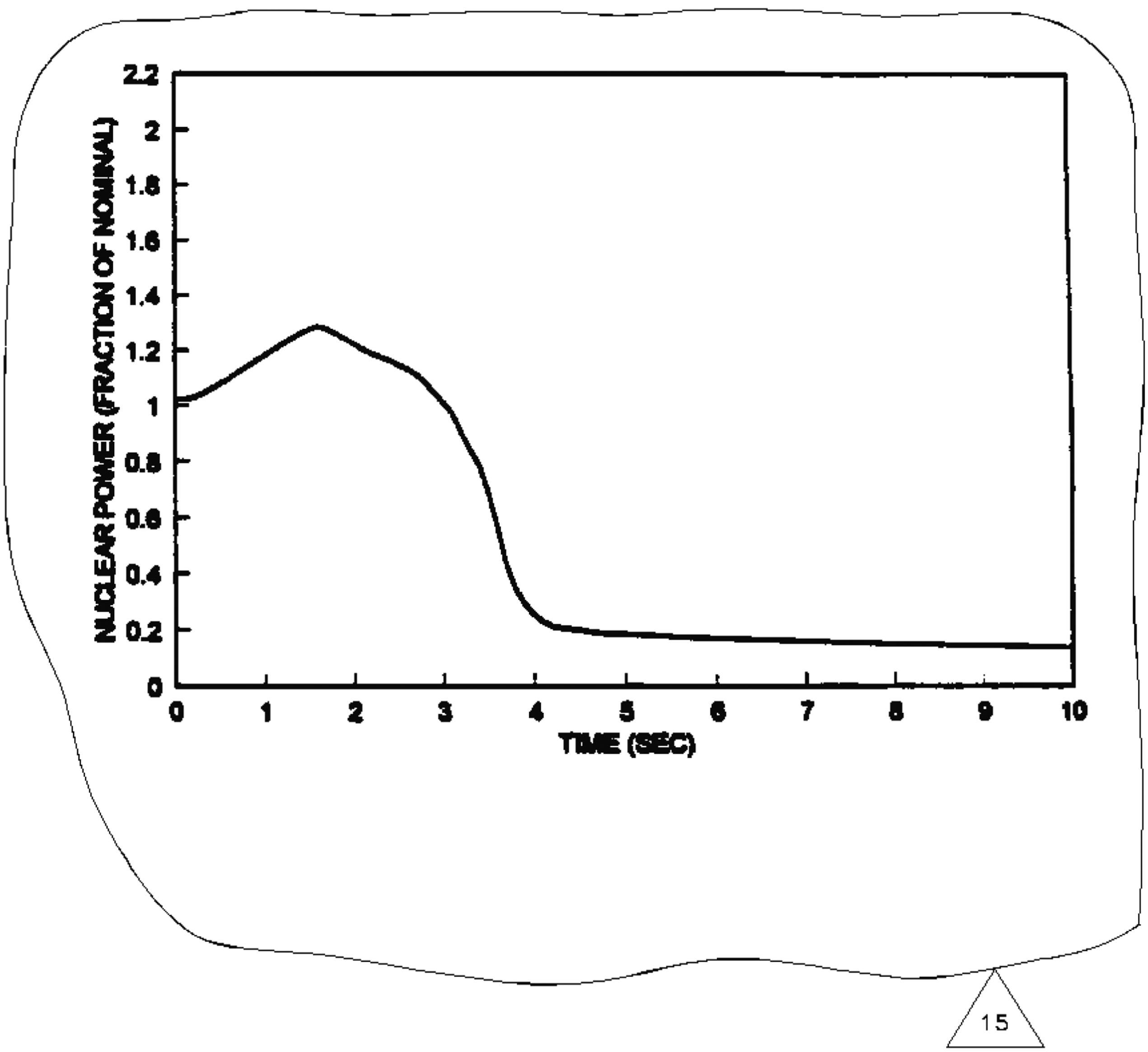
REV 23 5/11

SOUTHERN COMPANY

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS INITIALLY OPERATING, ONE LOCKED ROTOR –
RCS FLOW AND FAULTED LOOP FLOW

FIGURE 15.4-34A



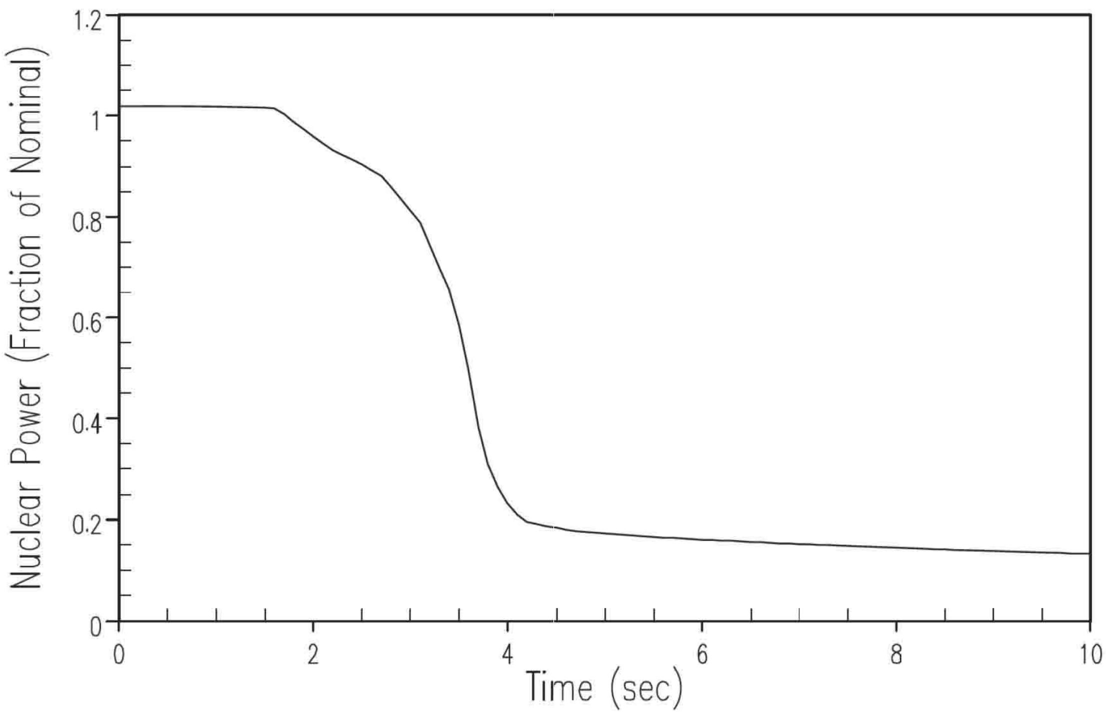
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

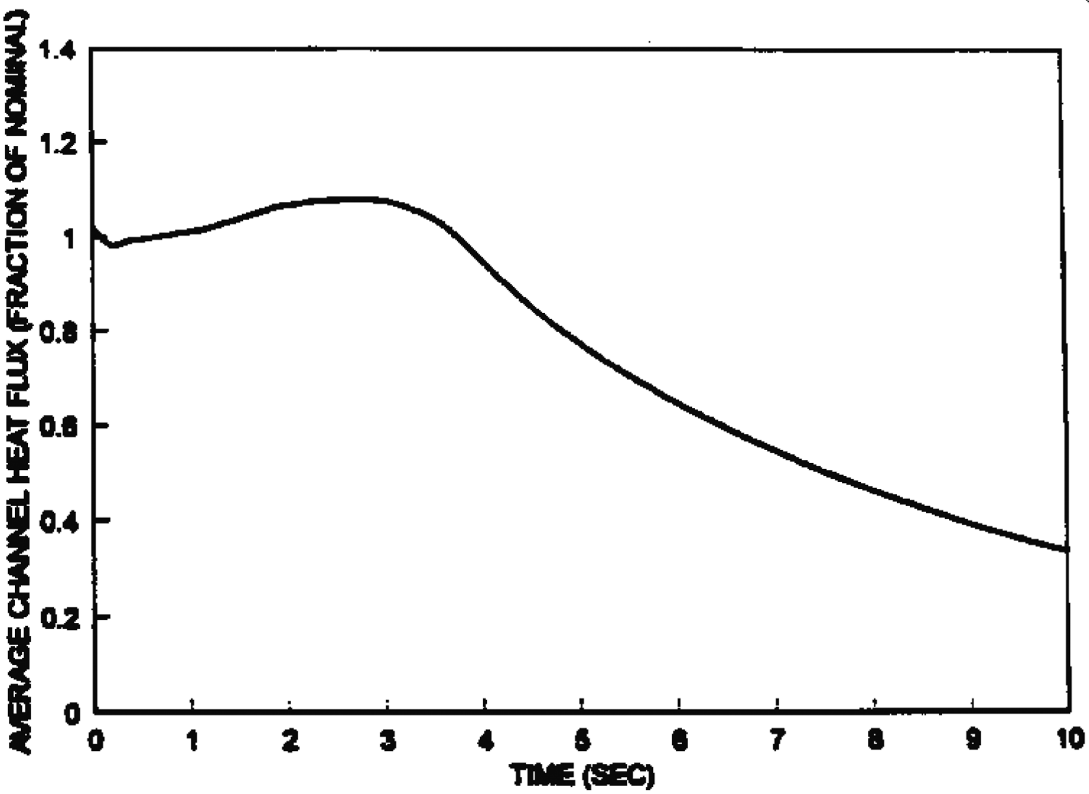
ALL LOOPS INITIALLY
OPERATING, ONE LOCKED ROTOR –
NUCLEAR POWER

FIGURE 15.4-35



REV 23 5/11

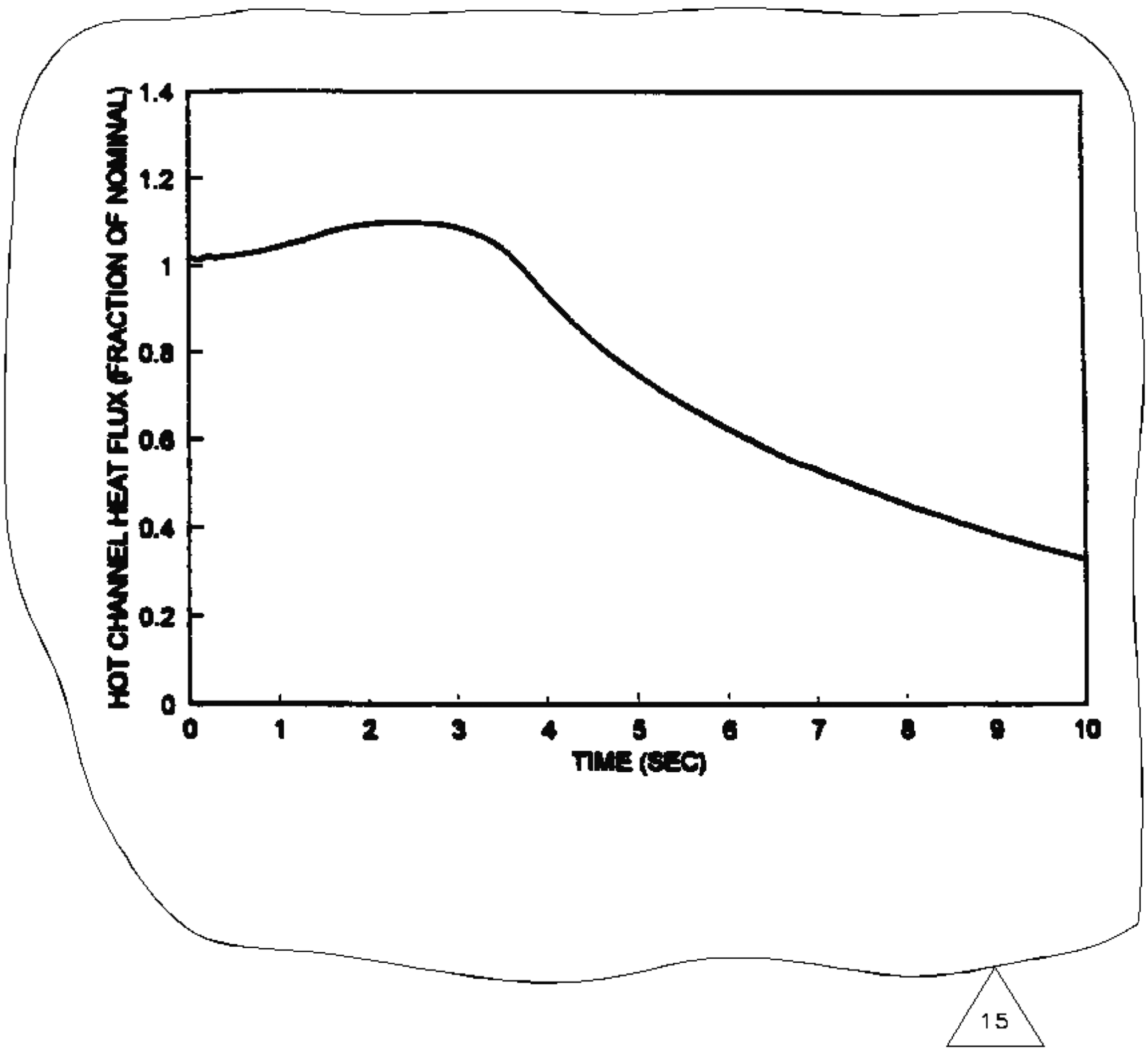
SOUTHERN COMPANY	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPERATING, ONE LOCKED ROTOR – NUCLEAR POWER
		FIGURE 15.4-35A



15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ALL LOOPS INITIALLY OPRATING, ONE LOCKED ROTOR – AVERAGE CHANNEL HEAT FLUX
		FIGURE 15.4-36



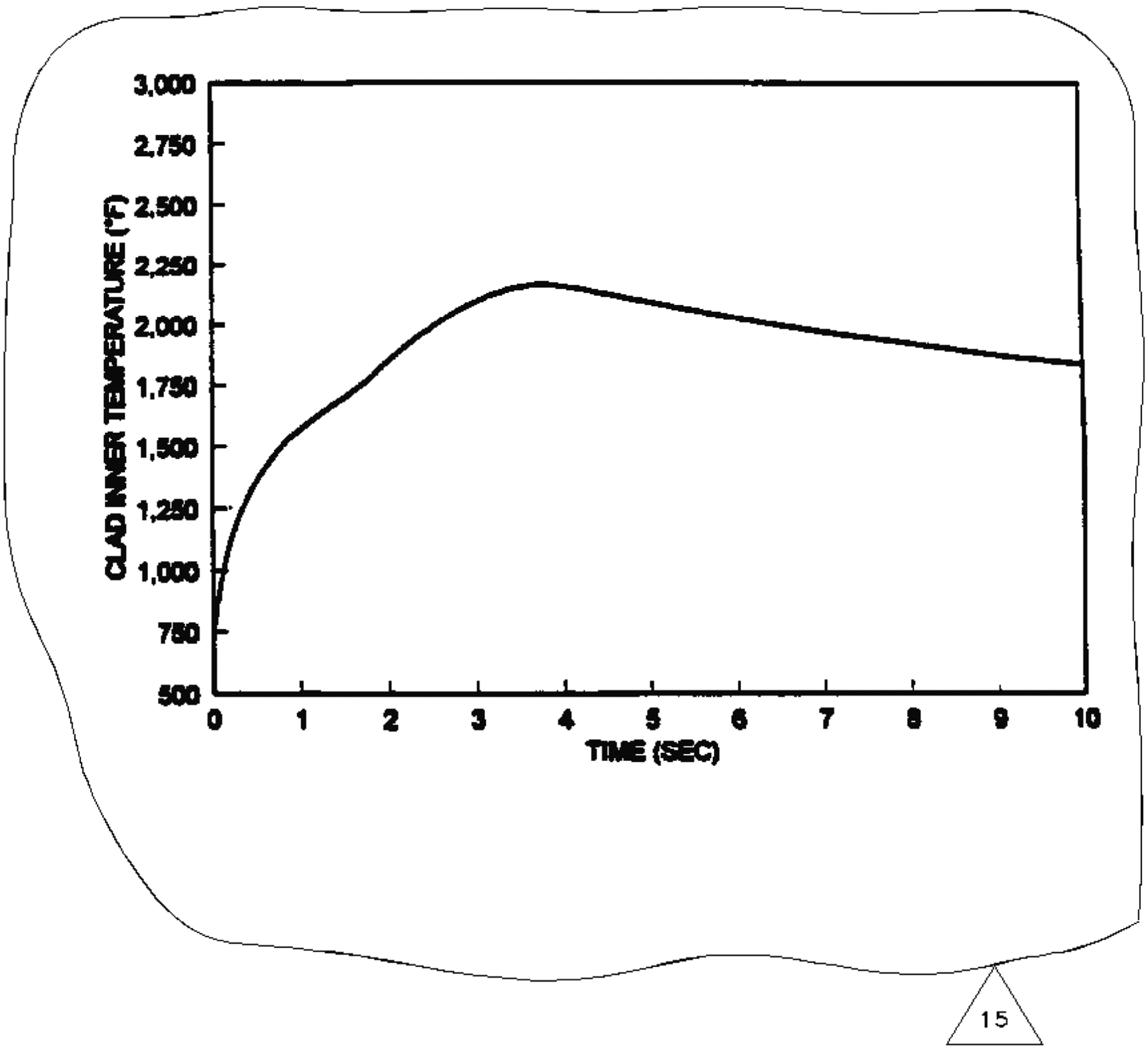
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS INITIALLY
OPERATING, ONE LOCKED ROTOR –
HOT CHANNEL HEAT FLUX

FIGURE 15.4-37



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	ALL LOOPS INITIALLY OPERATING, ONE LOCKED ROTOR – CLAD INNER TEMPERATURE
		FIGURE 15.4-38

THIS FIGURE HAS BEEN DELETED.

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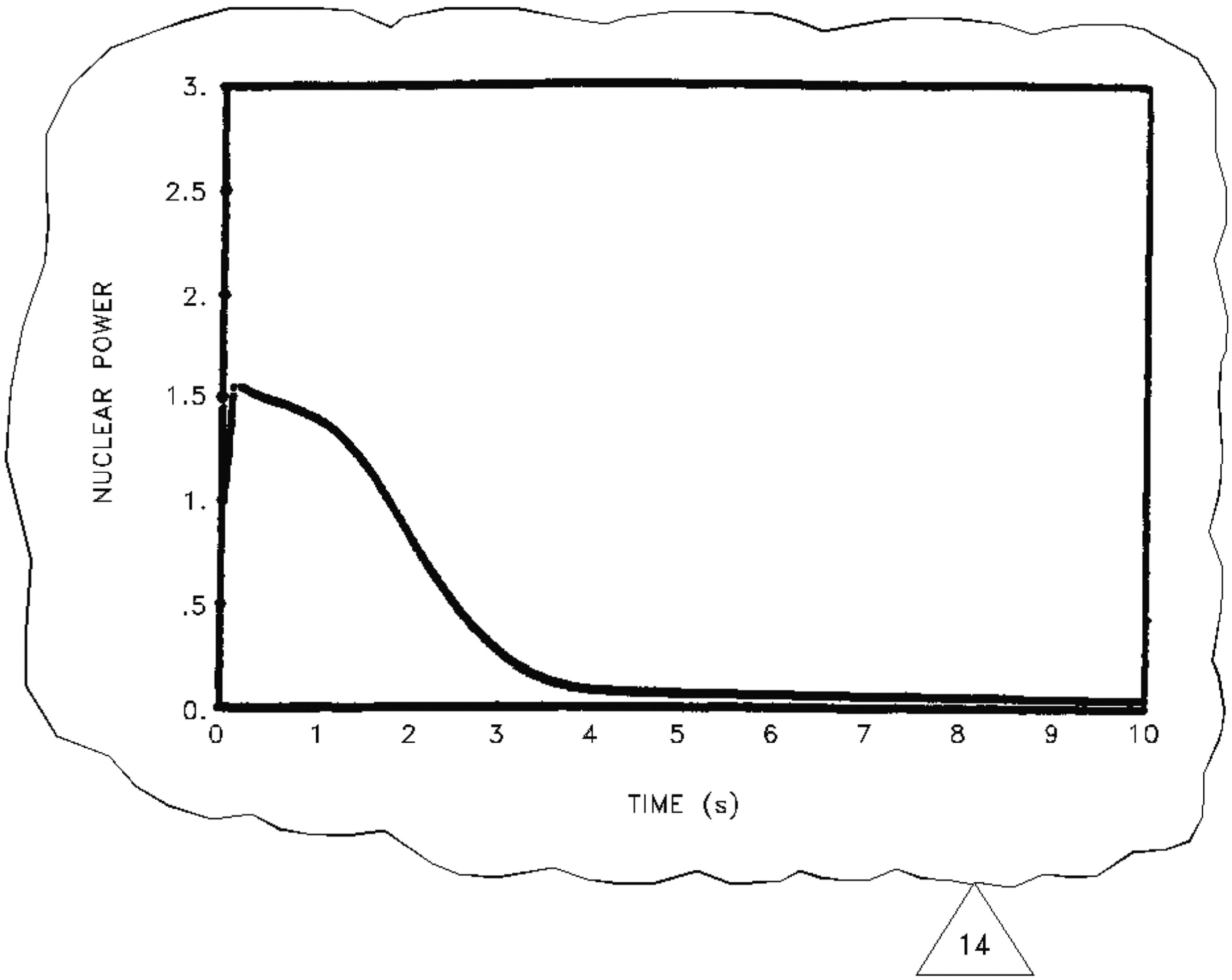
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ALL LOOPS OPERATING,
ONE L OCKED ROTOR –
CLAD TEMPERATURE VERSUS TIME

FIGURE 15.4-39



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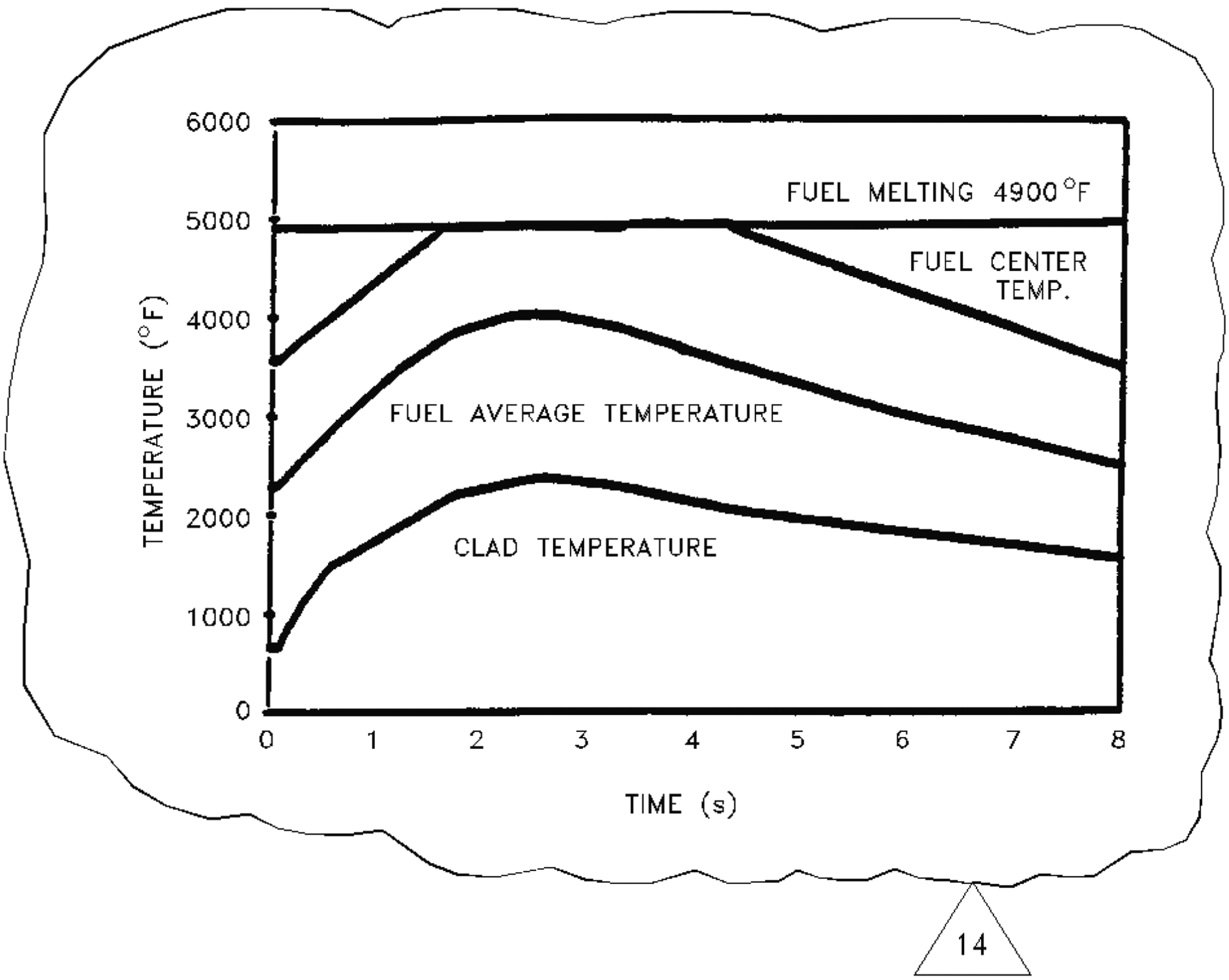
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

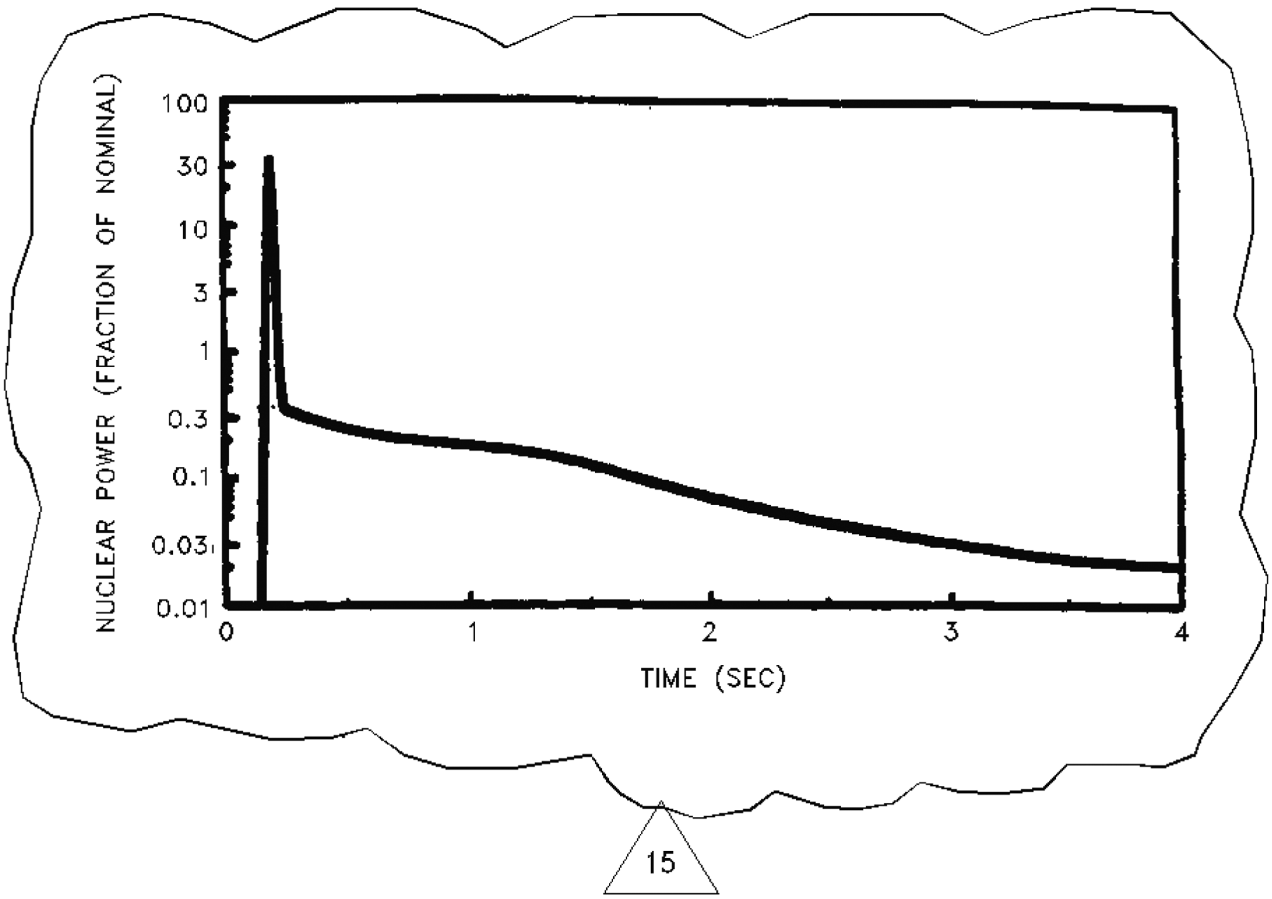
NUCLEAR POWER TRANSIENT BOL HOT FULL-POWER ROD EJECTION ACCIDENT

FIGURE 15.4-40



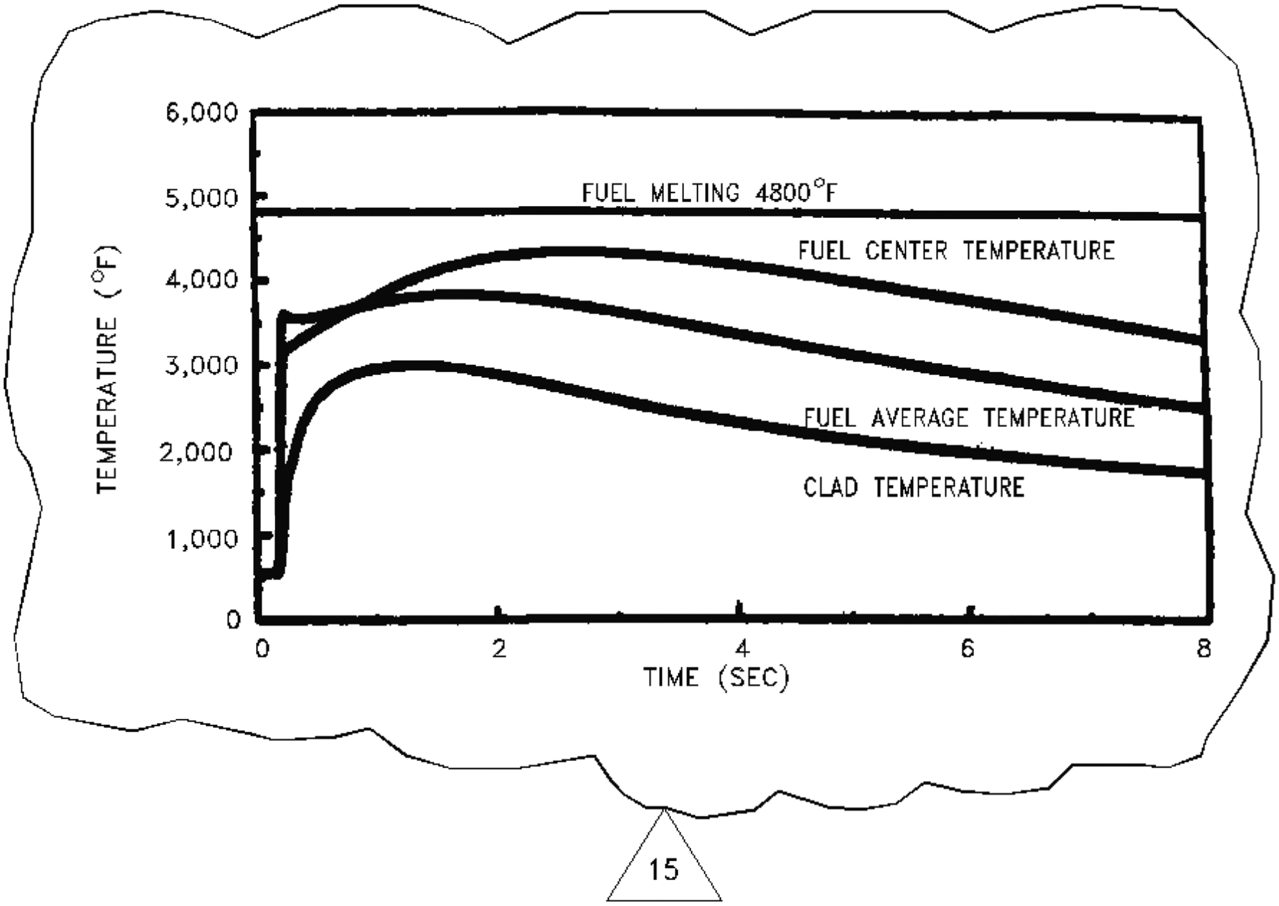
REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	HOTSPOT FUEL AND CLAD TEMPERATURE VERSUS TIME – BOL HOT FULL-POWER ROD EJECTION ACCIDENT
		FIGURE 15.4-41



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	NUCLEAR POWER TRANSIENT EOL HOT ZERO-POWER ROD EJECTION ACCIDENT
		FIGURE 15.4-42



REV 21 5/08

HOTSPOT FUEL AND CLAD
TEMPERATURE VERSUS TIME – EOL HOT
ZERO-POWER ROD EJECTION ACCIDENT

FIGURE 15.4-43

THIS FIGURE HAS BEEN DELETED.

15

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SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ROD EJECTION ACCIDENT (NO MELTING) GAMMA DOSE
		FIGURE 15.4-44

THIS FIGURE HAS BEEN DELETED.

15

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ROD EJECTION ACCIDENT
(NO MELTING) BETA DOSE

FIGURE 15.4-45

THIS FIGURE HAS BEEN DELETED.

15

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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ROD EJECTION ACCIDENT
(NO MELTING) THYROID DOSE

FIGURE 15.4-46

THIS FIGURE HAS BEEN DELETED.

15

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Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ROD EJECTION ACCIDENT
(0.25-PERCENT MELTING) GAMMA DOSE

FIGURE 15.4-47

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

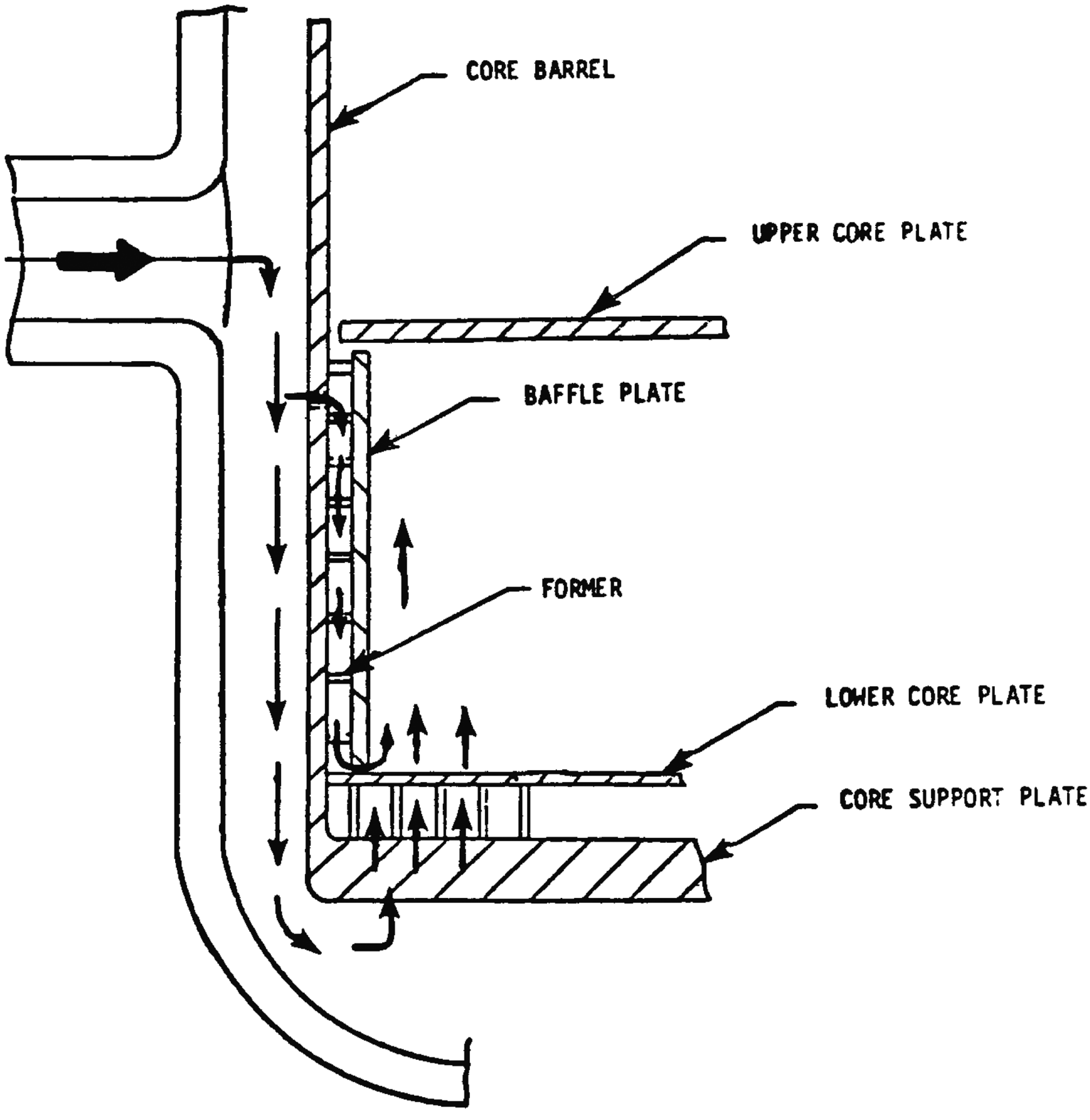
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ROD EJECTION ACCIDENT (0.25-PERCENT MELTING) BETA DOSE FIGURE 15.4-48

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ROD EJECTION ACCIDENT (0.25-PERCENT MELTING) THYROID DOSE FIGURE 15.4-49



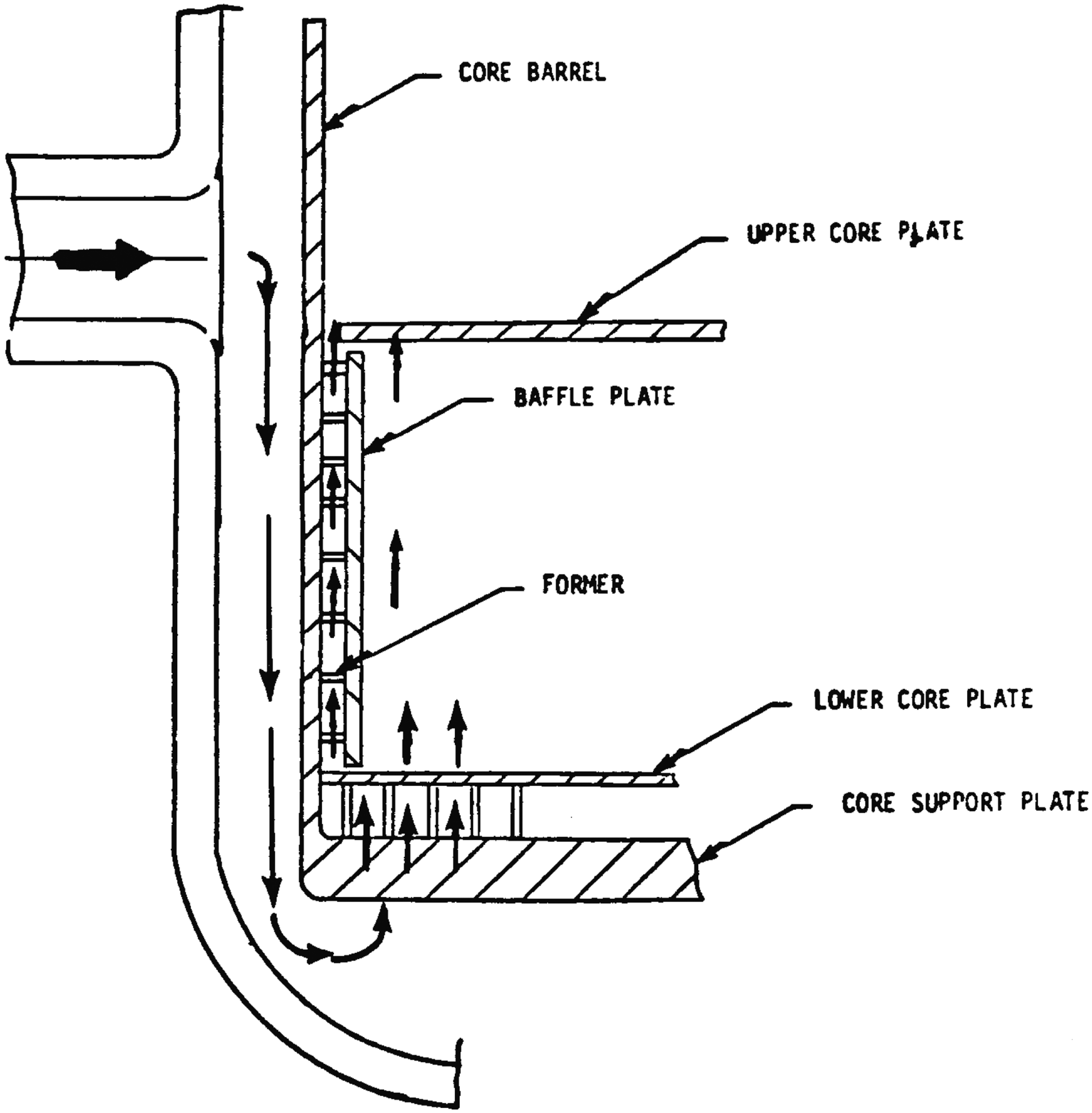
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPICTION OF THE DOWNFLOW BARREL
BAFFLE DESIGN CONFIGURATION

FIGURE 15.4-50



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPICTION OF THE UPFLOW BARREL BAFFLE DESIGN CONFIGURATION FIGURE 15.4-51

15.5 ANTICIPATED TRANSIENTS WITHOUT SCRAM

The worst common mode failure which is postulated to occur is the failure to scram the reactor after an anticipated transient has occurred. A series of generic studies[1,2] on anticipated transients without scram (ATWS) showed acceptable consequences would result provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. The final NRC ATWS rule[3] requires that Westinghouse designed plants install an ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow independent of the reactor protection system.

The Farley AMSAC design is described in section 7.8

REFERENCES

1. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

2. Anderson, T.M., "ATWS Submittal," Westinghouse Letter NS-TMA-2182 to S. H. Hanauer of the NRC, December 1979.

3. ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

APPENDIX 15A

HEAT TRANSFER COEFFICIENTS USED IN THE LOCTA-R2 CORE THERMAL ANALYSIS

This appendix has been deleted from the FSAR.

APPENDIX 15B

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

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APPENDIX 15B

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15B.1 INTRODUCTION

This section identifies the models used to calculate offsite radiological doses that would result from releases of radioactivity due to various postulated accidents. The postulated accidents are:

A. Fuel handling accident (FHA).

B. Waste gas decay tank rupture.

C. Steam generator tube rupture (SGTR).

D. Main steam line break (MSLB) accident.

E. Control rod ejection (CRE) accident.

F. Loss-of-coolant accident (LOCA).

G. Locked rotor accident (LRA).

15B.2 ASSUMPTIONS

The following assumptions are basic to both the model for the gamma and beta doses due to immersion in a cloud of radioactive material and the model for the TEDE and thyroid dose due to inhalation of radioactive material:

A. Direct radiation from the source point is negligible compared to gamma and beta radiation due to submersion in the radioactive material leakage cloud.

B. All radioactive material releases are treated as ground-level releases regardless of the point of discharge.

C. The dose receptor is a standard man as defined the International Commission on Radiological Protection (ICRP)[1].

D. Radioactive decay from the point of release to the dose receptor is neglected.

E. Isotopic data such as decay rates and decay energy emissions are taken from standard industry documents.[2, 7]

15B.3 GAMMA DOSE AND BETA DOSE

The gamma and beta doses delivered to a dose receptor are obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane; i.e., an "infinite semispherical cloud." The concentration of radioactive material within this cloud is taken to be uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release.

The beta dose is a result of external beta radiation and the gamma dose is a result of external gamma radiation. Equations describing an infinite semispherical cloud were used to calculate the doses for a given time period as follows:[3]

$$\text{Beta Dose} = 0.23 \cdot \frac{\chi}{Q} \cdot \sum_i \left[A_{R_i} \cdot \bar{E}_{\beta_i}\right]$$

$$\text{Gamma Dose} = 0.25 \cdot \frac{\chi}{Q} \cdot \sum_i \left[A_{R_i} \cdot \bar{E}_{\gamma_i}\right]$$

where:

A_{R_i} = the activity of isotope i release during a given time period (Ci)

$\frac{\chi}{Q}$ = the atmospheric dilution factor for a given time period (s/m^3)

$\bar{E}_{\beta_i}$ = the average beta radiation energy emitted by isotope i per disintegration (MeV)

$\bar{E}_{\gamma_i}$ = the average gamma radiation energy emitted by isotope i per disintegration (MeV)

As an alternative, doses may be calculated as

$$\text{Beta or Gamma Dose} = \frac{\chi}{Q} \cdot \sum_i \left[A_{R_i} \cdot DCF_i\right]$$

where DCF_i is the Dose Conversion Factor for isotope i taken from standard industry documents[4-7]. The dose may be modified by an occupancy factor for non-continuous occupancy, a geometry factor for other than an infinite hemispherical source, etc., as appropriate to the problem being analyzed.

15B.4 THYROID INHALATION DOSE

The thyroid dose for accidents not utilizing the 10 CFR 50.67 Alternative Source Term for a given time period t is obtained from the following expression:[(4)]

$$D = \frac{\chi}{Q} \cdot B \cdot \sum_i \left[Q_i \cdot DCF_i \right]$$

where:

D	=	thyroid inhalation dose (rem)
B	=	breathing rate for time interval t (m^3/s)
Q	=	total activity of iodine isotope i released in time period t (Ci)

The isotopic data and standard-man data are given in table 15B-1. The atmospheric dilution factors used in the analysis of the environmental consequences of accidents are given in chapter 2 of this report and are reiterated in table 15B-2a of this appendix.

The gamma energies, $E\gamma$, in table 15B-1 include the X-rays and annihilation gamma rays if they are prominent in the electromagnetic spectrum. Also, the beta energies, $E\beta$, include conversion electrons if they are prominent in the electromagnetic spectrum. The beta energies are averaged quantities in the sense that the continuous beta spectra energies are computed as one-third the maximum beta energies.

15B.5 TOTAL EFFECTIVE DOSE EQUIVALENT DOSE

The total effective dose equivalent (TEDE) dose for accidents that utilize the 10 CFR 50.67 Alternative Source Term for a given time period t is derived from the methodology described in Regulatory Guide 1.183.

The atmospheric dispersion factors (X/Q) for the exclusion area boundary and the low population zone were established during the initial licensing of the facility, as described in paragraph 2.3.4.2. The X/Q values used for each averaging period are shown in table 15B-2.

The atmospheric dispersion factors were computed at the control room intake for each hour of meteorological date, for the years 2000 through 2004, using the ARCON96 computer code as described in Regulatory Guide 1.194. ARCON96 evaluates ground level, vent, and elevated releases. A vent release is one that takes place through a rooftop vent with an uncapped vertical opening. Building wake effects are also considered in the model for estimating X/Q values from ground-level releases. Momentum rise and thermal plume rise are not considered in calculating the effective release height in the model. Additionally, under calm wind conditions, the receptor location is assumed to be directly downwind of the release point. Considering the release height, the receptor height, and the horizontal distance from the release point to the receptor, the model will calculate a “slant range distance” as the straight-line distance between the release point and the receptor. The values of X/Q for each averaging period were calculated and the 5-percent probable values determined. The X/Q values for each averaging time for each accident are shown in table 15B-3.

REFERENCES

1. "Report on ICRP Committee III on Permissible Dose for Internal Radiation (1959)," Health Physics, Vol 3, pp 30 and 146-153, 1960.

2. Leaderer, C. M., et al., Table of Isotopes, 6th ed., 1968, or 7th ed., 1978.

3. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences for a Loss of Coolant Accident for Pressurized-Water Reactors," Regulatory Guide 1.4, June 1974.

4. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

5. Nuclear Regulatory Commission, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, Revision 1, October 1977.

6. “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30.

7. “Users’ Guide for the TACT5 Computer Code,” NUREG/CR-5106, June 1988.

TABLE 15B-1

PHYSICAL DATA FOR ISOTOPES

Isotope	Decay Constant (HR-1)	Gamma Energy[a] (MeV/Disintegration)	Beta Energy[a] (MeV/Disintegration)	Dose Conversion Factor[b] (rem/Ci)
I-131	3.5856×10^{-3}	0.371	0.197	1.48×10^{6}
I-132	2.97×10^{-1}	2.400	0.448	5.35×10^{4}
I-133	3.31×10^{-2}	0.477	0.423	4.00×10^{5}
I-134	7.92×10^{-1}	1.939	0.455	2.50×10^{4}
I-135	1.03×10^{-1}	1.779	0.308	1.24×10^{5}
Xe-133	5.47×10^{-3}	0.030	0.146	-
Xe-133m	1.26×10^{-2}	0.033	0.155	-
Xe-135	7.60×10^{-2}	0.246	0.322	-
Xe-135m	2.72×10^{0}	0.422	0.097	-
Xe-138	2.45×10^{0}	2.870	0.800	-
Kr-85	7.95×10^{-6}	0.0021	0.223	-
Kr-85m	1.49×10^{-1}	0.151	0.233	-
Kr-87	5.33×10^{-1}	1.375	1.050	-
Kr-88	2.50×10^{-1}	1.743	0.341	-

BREATHING RATES

Time Period (h)	Control Room (m^3/s)	Offsite (m^3/s)
0 - 8	3.47×10^{-4}	3.47×10^{-4}
8 - 24	3.47×10^{-4}	1.75×10^{-4}
24 - 720	3.47×10^{-4}	2.32×10^{-4}

a. See reference 2.

b. See reference 4. Subsequent to the issuance of the Operating License, the dose conversion factors from Regulatory Guide 1.109 (reference 5) have been used and may continue to be used when calculating doses to the thyroid due to inhalation. Subsequent to FSAR Revision 13, dose conversion factors from ICRP 30 (reference 6) may be used when calculating doses to the thyroid due to inhalation.

TABLE 15B-2

OFFSITE ACCIDENT ATMOSPHERIC DISPERSION FACTORS (s/m^3)

Time Period	Site Boundary (1262 m)	Low-Population Zone (3219 m)
0 – 2 h	7.6×10^{-4}	2.8×10^{-4}
2 – 8 h		1.1×10^{-4}
8 – 24 h		1.0×10^{-5}
24 – 96 h		5.4×10^{-6}
96 – 720 h		2.9×10^{-6}

TABLE 15B-2a

SMALL BREAK LOCA ATMOSPHERIC DILUTION FACTORS (s/m^3)

Time Period	Site Boundary (1262 m)	Low-Population Zone (3219 m)	Control Room
0 – 30 s	-------------	-------------	8.79×10^{-4} [(b)] / 5.06×10^{-3} [(c)(e)]
30 – 2 h	-------------	-------------	8.79×10^{-4} [(b)] / 1.66×10^{-3} [(d)(e)]
0 – 2 h[(a)]	7.6×10^{-4}	2.8×10^{-4}	1.66×10^{-3}
2 – 8 h	2.9×10^{-4}	1.1×10^{-4}	1.38×10^{-3}
8 – 24 h	3.3×10^{-5}	1.0×10^{-5}	7.20×10^{-4}
24 – 96 h	1.9×10^{-5}	5.4×10^{-6}	5.60×10^{-4}
96 – 720 h	1.1×10^{-5}	2.9×10^{-6}	4.21×10^{-4}

a. These values are actually the 0-1-h X/Q values and are used for the 0 to 2-h period following an accident in accordance with NRC practice.
b. Equipment hatch – control room (emergency intake)
c. Containment – TSC used for control room (normal intake)
d. Containment – control room (emergency intake)
e. These control room atmospheric dispersion factors reflect the values resulting from analysis performed to support a Technical Specification change allowing the equipment hatch and personnel airlocks to remain open during refueling operations with appropriate administrative controls. Reference NRC SERs documented in NRC letters LC14842 (dated 29 September 2008) and LC 14149 (dated 30 September 2004).

TABLE 15B-3 (SHEET 1 OF 2)

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (s/m^3)

Containment		Accident
Time Period	Control Room	
0 – 2 h	1.66×10^{-3}	LOCA, MSLBA, SGTR, CRE, LRA
2 – 8 h	1.36×10^{-3}	LOCA
2 – 8 h	1.38×10^{-3}	MSLBA, SGTR, CRE, LRA
8 – 24 h	6.81×10^{-4}	LOCA
8 – 24 h	7.20×10^{-4}	MSLBA, CRE
24 – 96 h	5.60×10^{-4}	LOCA, CRE
96 – 720 h	4.21×10^{-4}	LOCA, CRE

Containment Hatch		Accident
Time Period	Control Room	
0 – 2 h	8.70×10^{-4}	FHA
2 – 8 h	6.77×10^{-4}	FHA
8 – 24	3.32×10^{-4}	FHA

Plant Vent		
Time Period	Control Room	
0 – 2 h	1.62×10^{-3}	FHA
0 – 0.0167 h	2.79×10^{-3}	LOCA
0.0167 – 2 h	1.65×10^{-3}	LOCA
2 – 8 h	1.37×10^{-3}	FHA
2 – 8 h	1.38×10^{-3}	LOCA

TABLE 15B-3 (SHEET 2 OF 2)

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (s/m^3)

8 – 24 h	7.10×10^{-4}	FHA
8 – 24 h	7.20×10^{-4}	LOCA
24 – 96 h	5.47×10^{-4}	LOCA
96 – 720 h	3.63×10^{-4}	LOCA

Time Period	RWST Control Room	
0 – 2 h	4.97×10^{-4}	LOCA
2 – 8 h	3.82×10^{-4}	LOCA
8 – 24 h	1.70×10^{-4}	LOCA
24 – 96 h	1.28×10^{-4}	LOCA
96 – 720 h	1.00×10^{-4}	LOCA

APPENDIX 15C

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16.0 TECHNICAL SPECIFICATIONS

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16.1 INSTRUMENTATION

16.1.1 SEISMIC INSTRUMENTATION

The seismic instrumentation requirements are addressed in TR 13.3.6 in the Technical Requirements Manual.

16.1.2 METEOROLOGICAL INSTRUMENTATION

The meteorological instrumentation requirements are addressed in TR 13.3.7 in the Technical Requirements Manual.

16.1.3 CONTAINMENT HYDROGEN MONITORS

The Containment Hydrogen Monitor requirements are addressed in TR 13.3.8 in the Technical Requirements Manual.

16.2 TECHNICAL REQUIREMENTS MANUAL

The conversion of the FNP Technical Specifications, based on NUREG-0452, to the FNP Technical Specifications, based on NUREG-1431, Revision 1 resulted in the creation of the Technical Requirements Manual (TRM) which includes certain technical requirements which do not meet the 10 CFR 50.36 criteria for inclusion in the Technical Specifications. Technical requirements that are licensing commitments, but which may be controlled by the licensee in accordance with the process for changes, tests, and experiments as provided in 10 CFR 50.59, can be maintained in the TRM.

The TRM contains selected requirements that apply to the operation of FNP with the intent being to provide a single, prominent, and easily accessible document for operating staff to reference and which will support the operating staff's compliance with these requirements with a minimum of effort. These requirements are conditions for operation, associated action requirements, and surveillance requirements with the format for presentation of the requirements being the same as used in the FNP NUREG-1431 based Technical Specifications.

The administrative controls for the TRM are the same as used for the control of the FSAR. These administrative controls ensure proposed TRM changes do not require NRC approval pursuant to 10 CFR 50.59, or if a change does require NRC approval, the controls ensure NRC approval prior to implementation of the change.

17.0 QUALITY ASSURANCE

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17.0 - QUALITY ASSURANCE

A. General

Chapter 17 describes the QA programs developed and implemented during the design, construction, and operation of FNP. Appendixes describing the design and construction QA programs of four major vendor organizations follow this chapter.

B. Introduction

The responsibility for the design, construction, testing and operation of the Joseph M. Farley Nuclear Plant (FNP) rests with the applicant, Alabama Power Company (APC)[a] To provide assurance that the design and construction of the FNP conforms with applicable regulatory requirements and with the design bases specified in the license application, a quality assurance (QA) program was developed and implemented under the supervision of APC's executive vice president. This program was applicable to all safety-related structures, systems, and components. The responsibility for developing and implementing certain phases of the overall program was delegated by APC to Southern Company Services, Inc. (SCS) subject to the review and approval of the applicant. The portion of the program delegated to SCS included the review or audit of design concepts, detail designs, specifications, drawings (including compliance with the requirements of the FSAR), and certain vendor shop quality control surveillance. APC was responsible for the development and implementation of the quality control program at the construction site through its general contractor, Daniel Construction Company of Alabama (Daniel).

C. Definitions

Quality Assurance (QA) - All those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

Quality Control (QC) - Those quality assurance actions related to the physical characteristics of a material, component, or system which provide a means to control the quality of the material, component, or system to predetermined requirements.

Quality Assurance Manual (QAM) - A manual prepared for the FNP setting forth the procedures and methods to be employed to ensure compliance with applicable codes, standards, criteria, and other requirements in the design and construction of the FNP for all safety-related systems, structures, and components.

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a. Southern Nuclear Operating Company became the plant licensed operator on December 23, 1991.

Owner - The persons, company, or corporation responsible for the nuclear power plant construction permit or operating license.

Contractor - Any organization under contract for furnishing items or services to an organization operating with the FNP QA program. It includes the terms vendor, supplier, subcontractor, and subtier levels of these where appropriate.

Safety - Related Structures, Systems, and Components – Those structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. They will be listed in the Quality Assurance Manual (QAM) for the FNP in a Q-List.

Procurement Documents - Binding documents that identify and define the requirements to which items or services must comply in order to be accepted by the owner.

Item - Any level of unit assembly, including structure, system, subsystem, subassembly, component, part, or material.

Objective Evidence - Any statement of fact, information, or record, either quantitative or qualitative, pertaining to the quality of an item or service based on observations, measurements, or tests which can be verified.

Documentation - Any written or pictorial information describing, defining, specifying, reporting, or certifying activities, requirements, procedures, or results.

Nonconformance - A deficiency in characteristic, documentation, or procedure that renders the quality of an item unacceptable or indeterminate. Examples of nonconformance include physical defects, test failures, incorrect or inadequate documentation, and deviation from prescribed processing, inspection, or test procedures.

Repair - The process of restoring a nonconforming characteristic to a condition such that the capability of an item to function reliably and safely is unimpaired even though that item may still deviate from an original requirement.

Rework - The process by which a nonconforming item is made to conform to a prior specified requirement by completion, remachining, reassembling, or other means.

Quality Assurance List (Q-List) - The list identifying FNP safety-related items is included in section 17.3.

[HISTORICAL]

[17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

Section 17.1 contains historical information implemented during design and construction of FNP. The current QA program is described in section 17.2.

17.1.1 ORGANIZATION

The major organizations participating in the design and construction of the Farley Nuclear Plant (FNP) are:

A. Alabama Power Company (APC) - Owner.

B. Southern Company Services, Inc. (SCS) - Architect/engineer (A/E) (agent for Alabama Power Company).

C. Bechtel Power Corporation - Architect/engineer (subcontractor responsible to SCS for major portion of plant design).

D. Westinghouse Electric Corporation - Designer and supplier of the nuclear steam supply system (NSSS).

E. Daniel Construction Company of Alabama (Daniel) - General contractor for the construction of the FNP.

The organization chart for each of these companies appears at the end of this section (figure 17.1-1).

The following describes the quality assurance (QA) responsibilities and authorities of each major organization.

17.1.1.1 Alabama Power Company

Alabama Power Company is the owner and is responsible for the overall development and implementation of the total QA program for the FNP during design and construction.

The manager - quality assurance (design and construction) (MQA) and his staff, reporting ultimately to the executive vice president via the vice president - nuclear generation and the senior vice president, manages the FNP QA program and ensures through a system of audits that all facets of the program are properly documented, implemented, and enforced. The MQA and his supporting staff are located in APC's General Office Building in Birmingham, Alabama; their primary duties and responsibilities are as noted in paragraph 17.1.2.2. The quality *assurance field representatives (QAFRs) provide a completely independent review and evaluation of the adequacy and effectiveness of the construction site quality control (QC) program. The QAFRs report regularly to the MQA on the status and adequacy of the construction site QA program. They have been provided with sufficient organizational freedom to monitor field construction and erection activities and to identify quality problems. Those problems requiring management decisions are referred to the MQA for appropriate action.*

The Nuclear Engineering and Technical Support Section of the Nuclear Generation Department is responsible for the audit and acceptance of all specifications, general design drawings, and procedures, with particular emphasis on the coordination of audits by other interested APC departments. This responsibility involves assessment of the adequacy of applicable codes and standards made a part of any specifications related to the project and ensuring that adequate quality assurance programs are clearly made a contractual responsibility of contractors, vendors, and suppliers. Assessing the qualifications of all outside consultant specialists used on the project is also a responsibility of this section.

The Construction Department has the responsibility for all site construction activities, including monitoring functions of Daniel, to see that terms of the contract (cost, accounting, scheduling, and QC) are met and that any corrective actions that may be required are taken. The Construction Department also provides limited procurement and inspection services at the site. QC activities of Daniel, which encompass its work as well as that of its subcontractors, will be monitored by Construction Department personnel.

Corrective actions shall be one of the following:

A. *Daniel supervisors are cautioned regarding any observable trends leading toward laxity in QC.*

B. *An actual ordering of a shutdown of some phase of the work is made because of an observed deviation.*

In the latter case, this action will be immediately reported to the Daniel project manager or his representative and to the representative of the APC MQA for handling by the prescribed procedure.

Notwithstanding any inspection performed above, the final responsibility for the adequacy of all field quality control and assurance procedures is the responsibility of the MQA and his field representatives who will monitor and audit all QC activities and will assist in establishing an effective program.

The Nuclear Generation Section is responsible for component/preoperational testing and startup, maintenance, and operation at the FNP.

The Purchasing Department is responsible for maintaining APC's bidders' list and reviewing the proposed bidders submitted by SCS, Bechtel, and Daniel. APC Purchasing will cooperate with Daniel in the preparation of a bidders' list and will approve purchasing recommendations by Daniel in excess of a specified monetary value. APC purchase orders will be placed based on quotations which may be obtained by SCS, Bechtel, and APC. Inquiries or purchase orders will be issued on requisitions prepared by SCS, Bechtel, and the Construction Department. Requisitions from SCS and Bechtel may be assigned to Daniel, in which case Daniel will issue the inquiry, obtain the bids, issue the purchase order, and perform vendor surveillance. In some instances, APC purchase orders are assigned to Daniel. In these cases, Daniel will perform vendor surveillance.

17.1.1.2 Southern Company Services, Inc.

Southern Company Services, Inc. is the architect/engineer for the FNP. SCS has developed and implemented that portion of the QA program relating to the review or audit, approval, and documentation of basic design concepts, detail designs, drawings, and specifications. SCS assures that all drawings and specifications for structures, systems, and components clearly set forth the requirements, codes, and special procedures which must be met to render all items suitable for their intended service and to provide for quality manufacture, fabrication, and construction installation.

SCS is also responsible for the analysis of all proposals for the furnishing and installation of equipment and structures to ensure that contractors and manufacturers have an adequate program to meet all QA requirements and codes which are a part of the specifications.

SCS is responsible for the administration of the vendor surveillance program.

For a description of the SCS QA program, refer to appendix 17A. For the SCS organization chart, see figure 17.1-1.

17.1.1.3 Bechtel Power Corporation

Bechtel Power Corporation has been retained by Southern Company Services, Inc. to act as its consultant on the nuclear portion of the plant. In this capacity, Bechtel is responsible for the review or audit, approval, and documentation of basic design concepts, detail designs, drawings, and specifications for certain structures, systems, and components. Bechtel assures that all drawings and specifications for structures, systems, and components for which they are responsible clearly set forth the requirements, codes, and special procedures which must be met to render all items suitable for their intended service.

Bechtel is also responsible for the analysis of all proposals for the furnishing and installation of equipment and structures for which they are responsible and for ensuring that the involved *contractors and manufacturers have an adequate program to meet all QA requirements and codes which are a part of the specifications.*

For a description of the Bechtel QA program, refer to appendix 17B. For the Bechtel organization chart, see figure 17.1-1.

17.1.1.4 Westinghouse Electric Corporation

APC has contracted with Westinghouse Electric Corporation to design and fabricate the nuclear steam supply system and the initial reactor core (comprised of the Westinghouse standard 3-loop plant) for the FNP.

The Westinghouse QA program is applicable to the design, procurement, and inspection of all systems and components in the Westinghouse scope of supply whether manufactured by Westinghouse or purchased through other suppliers.

Over the course of performing the design and initial procurement activities for the Joseph M. Farley Plant, the Westinghouse quality assurance program was upgraded to reflect changes in regulatory requirements and industry standards. These changes first culminated in WCAP-8370, Revision 7A. This revision of the Westinghouse QA program was applicable to activities within the Westinghouse scope performed for the FNP which were initiated from January 1, 1975 to October 1, 1977. Subsequently, the Westinghouse QA program, which is described in WCAP-8370, Revision 8A, was applicable to activities within the Westinghouse scope which were initiated after October 1, 1977 and through October 1979. The Westinghouse QA program, described in WCAP-8370, Revision 9A, is applicable to activities within the Westinghouse scope which were initiated after October 31, 1979 and through February 1, 1981. The most recent Westinghouse QA plan, described in WCAP-8370, Revision 12A, issued in 1992 was recently replaced with QMS Rev. 1 (Reference 6 of Chapter 4.2) which is applicable to activities within the Westinghouse scope initiated after January, 1996.

The original quality assurance program implemented by Westinghouse for the Joseph M. Farley Plant is described in appendix 17C. For the Westinghouse organization chart, see figure 17.1-1.

17.1.1.5 Daniel Construction Company of Alabama

Daniel Construction Company of Alabama has been retained by Alabama Power Company as the general contractor for FNP construction activities. Daniel will execute a quality control program in full accord with APC's QA program. The quality control program includes the procedures, instructions, and control actions necessary to assure that the field fabrication and construction, material and equipment, and workmanship are controlled to meet applicable requirements of the drawings and specifications. All personnel performing quality control functions have been delegated sufficient operational authority to exercise their knowledge and *responsibility through quality control surveillance and inspections to assure that the specified requirements are achieved.*

The accumulation, filing, and storage of quality-related documentation shall be the responsibility of Daniel.

Daniel is responsible for administering QA supplier surveillance for Daniel-originated procurements and APC procurements which have been assigned to Daniel.

For a description of the Daniel QA program, refer to appendix 17D. For the Daniel organization chart, see figure 17.1-1.

17.1.2 QUALITY ASSURANCE PROGRAM

The FNP QA program is applicable to those structures, systems, and components classified as safety related. These items are identified in section 17.3 along with the associated QA responsibilities of the major participating organizations. The QA program shall be in force throughout the design and construction of the FNP.

The APC QAM requires procedures and instructions which govern the activities of APC in the design and construction of the FNP. In addition, each contractor of safety-related structures, systems, or components is required to develop and implement his own QA program subject to acceptance by APC. Audits are conducted by APC to ensure that the QA provisions are met.

The APC design and construction program is composed of the Quality Assurance Committee (QAC), the manager – corporate quality assurance, the manager - quality assurance (design and construction), and the Quality Assurance Manual (QAM). These elements are discussed in the following paragraphs.

17.1.2.1 Quality Assurance Committee

The Quality Assurance Committee advises and assists the executive vice president of APC on all phases of the QA program. This executive vice president serves as chairman of the Committee. Other members of the Committee are the senior vice president and the vice presidents of the Nuclear Generation Department and the Construction Department of APC, the senior vice president of SCS and the vice president - nuclear of SCS.

The Committee meets semiannually, or more often if called by the chairman (either at his discretion or at the request of any member), to review the adequacy and practicality of the QA program, the functioning of the program in regard to implementation and effectiveness, and any proposed modifications to the program.

The QAC has the duty of proposing revisions or modifications to the chairman in the event its review indicates the need for such. The MQA, his staff, and any other personnel of APC and SCS are available to assist the Committee in its review and to record the minutes of all meetings.

The MQA reports to the Committee at each semiannual meeting, or at such other times as requested, on the overall effectiveness of the program, other matters which he considers significant, and any phase of the program on which any member of the Committee requests a report.

17.1.2.2 Manager-Quality Assurance

APC has appointed an experienced graduate engineer with a broad general background of construction management to function as manager-quality assurance for the FNP project. He will report to the vice president - nuclear generation but will have direct access to the executive vice president. Specific duties and responsibilities of the MQA, which may be delegated to personnel in the QA Section, include:

A. The maintenance of close communication with Southern Company Services, Bechtel, and Daniel to ensure that the portion of the QA program assigned to them is being properly developed and implemented.

B. The maintenance of close communication with APC Construction, Nuclear Generation, and Purchasing Departments and with the Nuclear Engineering and Technical Support Section of the Nuclear Generation Department with respect to the APC portion of the QA program.

C. Reporting periodically via the vice president - nuclear generation to the executive vice president of APC regarding the overall progress and status of the QA program and any deviations. Any deficiency or discrepancy considered a significant deviation must be reported immediately.

D. Auditing specifications with respect to quality assurance requirements.

E. Prior to the award of a contract, examining the supplier's proposal and the recommendations, including quality assurance programs, to verify that each vendor recommended as a supplier of a safety-related structure, system, or component has an adequate QA program at his manufacturing or fabricating plant to meet the requirements of the specifications, drawings, and contract documents.

F. Coordinating through SCS and Daniel the activities of outside organizations and special consultants engaged to monitor and document the QA programs being utilized at the manufacturing or fabricating plants of vendors furnishing safety-related structures, systems, orcomponents.

G. Other duties as may be assigned by APC's executive vice president via the vice president – nuclear generation to ensure proper development and adequate implementation of the QA program.

The MQA and his supporting staff are located in the APC General Office Building. Onsite representatives reporting directly to the MQA keep him fully informed regarding day-by-day progress of construction and compliance with the provisions of the QA program.

The MQA visits the construction site frequently for consultation with his representatives and for persona observations to ensure compliance with the provisions of the QA program. He periodically participates with representatives of outside organizations and special consultants on visits to manufacturing plants of vendors to ensure proper monitoring and documentation of their QA programs.

The MQA or his representatives have authority to stop any work in progress at the construction site and to require the removal of any item not conforming to the approved specifications and drawings or which is not in accordance with the provisions of the QA program.

17.1.2.3 The Quality Assurance Manual

The Quality Assurance Manual (QAM) defines the policies and procedures employed to implement the QA program and to ensure compliance with applicable codes, standards, design criteria, and other requirements identified in the design, procurement, and construction documents of the FNP for all safety-related structures, systems, and components. The QAM contains a detailed listing, referred to as the Q-List, identifying these structures, systems, and components.

The QAM references the QC Procedure Manual which contains procedures for work in the construction of the FNP. The QAM and the QC Procedure Manual are amended to include changes and additional procedures as they are developed. The changes and additional procedures are prepared, approved, and released prior to the initiation of any work governed by changes or new procedures.

17.1.3 DESIGN CONTROL

17.1.3.1 Procedure Manual

A procedure manual has been developed for the FNP which contains detailed instructions regarding design control measures. This manual includes engineering correspondence procedures, design and engineering approval procedures, engineering division of responsibility, and the checks, reviews, and audits required to integrate Westinghouse, Bechtel, Southern Company Services, and Alabama Power Company into a common effort on plant design and, at the same time, provide a system of checks and balances to assure both quality design and adequate participation by SCS and APC.

17.1.3.2 Designs Originating with SCS

The design bases and performance criteria for structures, systems, and components under SCS's responsibility are contained in the FSAR. They were reviewed and approved by APC and served as the starting point in design by SCS's engineers.

All drawings, specifications, and calculations are subject to internal review. Each discipline (Mechanical, Electrical, Structural) has a project engineer who is responsible for assuring:

A. *The incorporation of requirements and design bases as outlined in the FSAR into specifications and drawings.*

B. *The incorporation of QA requirements into design documents commensurate with the function of the item.*

C. *General conformance to good engineering practices which ensures compatibility of items incorporated into the plant.*

D. *Proper coordination with project engineers of other disciplines.*

Staff specialists and outside consultants are available as required by project engineers. The vice president – design engineering has overall responsibility for all designs submitted by SCS.

17.1.3.3 Designs Originating with Bechtel

The design bases and performance criteria for structures, systems, and components under Bechtel's responsibility are contained in the FSAR. They were audited and approved by SCS and APC and served as the starting point in design by Bechtel's engineers. All drawings, specifications, and calculations are subject to internal review. In these design reviews, *independent checks of the drawings, specifications, and/or calculations are made to ensure accuracy and adherence to FSAR requirements. During various stages of design, these engineers consult with the discipline chief engineers and their staff specialists. By means of a design control checklist, chief engineers designate those documents they want to review and approve. Review and approval is indicated by the signature of the appropriate chief engineer on the document. Outside specialists are consulted if the occasion demands. The project engineer, who reports to the engineering manager, has overall responsibility for all designs submitted to SCS and APC.*

17.1.3.4 Design Interfaces

The APC QAM requires procedures that govern the interface relationship among the design organizations with regard to review, approval, release, distribution, and revision of engineering data. Verification of design adequacy is accomplished by the following procedures:

A. Westinghouse specifications and drawings affecting the Bechtel-Westinghouse interface are submitted by Westinghouse to Southern Company Services, Inc., and Bechtel for comment and are subject to final acceptance by the APC Nuclear Generation Department.

B. The design concepts and specifications developed by Bechtel covering the nuclear aspects of the FNP are audited and documented by SCS and are subject to final acceptance by the APC Nuclear Generation Department.

Drawings and documents covering the portion of the FNP developed by Bechtel are submitted to Westinghouse for comment when they have a bearing on the Westinghouse NSSS or result from criteria supplied by Westinghouse; they are also submitted to the APC Nuclear Generation Department.

C. The design concepts, detail designs, specifications, and drawings covering the portion of the FNP developed by SCS are, where appropriate, audited for nuclear aspects only and documented by Bechtel subject to final acceptance by the APC Nuclear Generation Department.

17.1.3.5 <u>Design Changes</u>

Design changes, including field changes, are governed by design control measures commensurate with those applied to the original design and are reflected in accurate "as-built" drawings and specifications. All design changes are reviewed and approved by the organizations that performed the original design, review, and approval. The APC QAM requires that procedures be prepared to control design changes.

17.1.4 PROCUREMENT DOCUMENT CONTROL

Procurement documents are prepared by APC, SCS, Bechtel, and Daniel. In most cases, procurement packages contain detail specifications which are prepared in accordance with preceding and subsequent subsections. Purchase orders not described by a specification are reviewed by APC's QAM to ensure that adequate QA/QC requirements are included prior to contract award. The QAM sets forth the review and approval procedures that are followed by the preparing agency. Changes in procurement documents are subject to the same degree of control that is utilized in the preparation of the original document.

Procurement documents include provisions for the following:

A. *Supplier Quality Assurance Program*

   Each bidder shall submit with his proposal a description of the quality assurance control program which will be followed to ensure meeting the requirements of the procurement documents.

B. *Basic Technical Requirements*

   Procurement includes provisions regarding drawings, codes, and standards, with applicable revision data, inspection requirements, and special instructions and requirements such as for designing, fabrication, cleaning, erecting, packaging, handling, shipping, and storage at the construction site.

C. *Source Inspection and Audit*

   Procurement documents shall provide for access to the plant facilities and records.

D. *Documentation Requirements*

   Procurement documents shall require records (such as drawings, procedures, procurement documents, inspection and test records, personnel and procedure qualifications, and material, chemical, and physical test results) to be prepared, *maintained, submitted, or made available for review. Instructions on record retention and disposition shall be provided.*

APC reviews procurement documents to ensure that appropriate requirements are included to provide a quality product. Such requirements, as appropriate, include reference to applicable codes and standards, welding requirements, testing and inspection requirements, and any special requirements dictated by the uniqueness of the item.

17.1.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

The QAM requires procedures governing the interfacing activities of the major design organizations and the activities of the APC General Office and field forces.

17.1.5.1 Major Design Organizations

The quality assurance manuals of major design organizations are reviewed by APC to ensure that instructions and procedures exist for the preparation of drawings and specifications. Of particular concern is the assurance that provisions are made for the incorporation of design requirements imposed by codes, standards, and the FSAR; adequate checking of design documents; and control of design changes. Audits by APC's MQA or his designee are performed periodically to verify compliance.

17.1.5.2 Contractors

Contractors' quality assurance manuals are reviewed to ensure that all activities are described by procedures and instructions adequate to provide satisfactory accomplishment of activities. APC's inspectors audit offsite contractors to ensure compliance with approved procedures while the onsite contractors are monitored by Daniel and by APC's MQA and QAFRs.

17.1.5.3 Construction Site

The QAM requires that procedures be prepared to set forth the QA/QC requirements and practices that are followed at the construction site. These requirements are applicable to all APC personnel, contractors, and subcontractors performing work at the plant site. They include, but are not limited to, the following:

A. Receipt, control, distribution, updating, filing, and utilization of approved drawings and specifications and the retrieval of void drawings and specifications.

B. Receipt and inspection of materials and equipment upon arrival.

C. Identification, control, and proper utilization of material and equipment.

D. Storage and handling of material and equipment.

E. Appropriate fabrication or erection processes.

F. Destructive and/or nondestructive testing as may be required.

G. Calibration and control of test and measurement equipment.

H. Documenting, recording, and retention of results of inspection and tests.

I. Reporting and documenting deviations in or from the drawings, specifications, or procedures.

J. Resolution of construction deviations from drawings, specifications, or procedures.

K. Reporting and documenting the results of incidents.

All new and revised procedures are reviewed and approved by APC's MQA prior to release.

17.1.6 DOCUMENT CONTROL

The QAM requires that procedures be prepared to provide control of approved drawings, specifications, and instructions which apply to the various phases of work. This ensures that work is performed in accordance with the latest approved documents. The procedures include the following:

A. The documents applicable to various phases of the field construction work.

B. The distribution of each of the documents to responsible individuals in the field construction organization and to the contractor and their subcontractor performing work at the plant site.

C. The method by which revisions to these documents are issued and distributed and void documents are retrieved.

D. A means by which a document may be marked to indicate that a portion of the document may or will be changed by a revision at some later date.

E. A means by which persons involved in the work can verify that the copy of any document which they hold is an up-to-date, complete, and approved copy

Design changes, including field changes, are subjected to the same control procedures as the original documents and are reviewed and approved by the same organization that performed the original review and approval. Periodic audits are performed to ensure that the latest engineering data are being used at the construction site.

17.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

To qualify as a prospective bidder for the FNP, a manufacturer, supplier, or contractor must not only have a good record of quality performance on all types of work, either nuclear or nonnuclear, but must provide special assurance to APC of its capability, competence, and willingness to produce under contract the high performance level which is consistent with APC's complete QA program on the FNP.

A list of qualified bidders for each type of equipment and material is maintained by the APC Purchasing Department. This list is kept current by the addition of newly-qualified bidders and prompt deletion of any bidder whose performance is unsatisfactory. Names of prospective qualified bidders are obtained from various sources such as:

A. Bidder lists on past APC work.

B. Recommendations by SCS, Bechtel, Daniel, consultants, and other electric utilities.

C. Vendor's lists of major component manufacturers.

The QAM requires procedures for qualifying and approving bidders by APC when procurement is initiated by APC, SCS, Bechtel, or Daniel.

Manufacturing and fabricating facilities of prospective bidders may be inspected as part of the qualification procedure. Meetings with prospective bidders may be held to appraise technical expertise and QA/QC competence. All bidders are required to submit a QA/QC program with their bid. APC's MQA verifies, prior to award of a contract, that the successful bidder has a QA program adequate to meet the requirements of the procurement documents at his manufacturing or fabricating plant.

SCS administers vendor shop quality control surveillance for components procured by APC. Daniel administers vendor shop quality control surveillance for components which are supplied as part of their scope of work. Prior to shipment of components from manufacturers' facilities, the release is approved by the source inspector who verifies compliance with procurement documents.

Procured materials are inspected at the construction site for damage, identification, and conformance to the procurement documents. Receiving, storing, and handling of materials and equipment at the construction site are performed in accordance with approved procedures referenced in the QAM which preclude acceptance of material that does not conform to the procurement documents and ensures that correctly identified, acceptable materials are properly controlled to preclude damage or deterioration prior to use in construction. The receiving QC representative initiates required QC documentation and is also responsible for initiating corrective action and control for damaged or nonconforming materials or equipment according to approved procedures.

Documentation for purchased material and equipment is evaluated to ensure compliance with the procurement documents. When nonconformances are noted, immediate action is taken in accordance with approved procedures.

Periodic supplier evaluations by APC representatives ensure that the QA programs are effective and in compliance with approved procedures and the QAM.

17.1.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Identification of materials, parts, and components is required of all suppliers by specifications, drawings, and purchase orders. Requirements for identification are ensured through the process of purchase document review and audit.

Unique equipment numbers are assigned to items of equipment or mechanical devices. These unique numbers are used to identify the equipment in the field and on drawings, schematics, and similar documents. These numbers appear beside or below the equipment description and are used in a variety of listings and tabulations to clearly identify each piece of equipment. Items are permanently identified to permit identification to supporting documentation. Items are traceable from such identification to a specific purchase order, to manufacturers' records, and to quality assurance records and documentation. Identification of material or equipment to the corresponding mill test reports, certifications, and other required documentation is maintained from receipt of the material or equipment throughout the operating life of the plant

Control of material, parts, and components is governed by APC acceptance of contractor procedures and QA programs. Specific control requirements include:

A. Each organization receiving items is required to determine that they are properly identified and that supporting documentation has been obtained.

B. Nonconforming or rejected materials, parts, or components are identified and segregated to ensure against their misuse.

17.1.9 CONTROL OF SPECIAL PROCESSES

Contractors' QA programs and special process procedures are reviewed and accepted to ensure that special processes employed are adequately controlled and documented and that they conform to established codes and standards. Each contractor's QA program is reviewed and accepted by APC's MQA and SCS and/or Bechtel. Special process procedures are reviewed and approved by the appropriate design group within SCS and/or Bechtel. Any document not reviewed by APC is subject to audit by APC. Typical processes include welding, heat treatment, cleaning, preservation, nondestructive examination, and plating.

For APC purchases, APC inspection is utilized to verify the control of processes at the contractor's facilities; for Daniel purchases, this assurance is verified by Daniel personnel. At the construction site, special processes are monitored by Daniel, the Construction Department, and APC's QAFR to ensure that approved procedures are followed. The following aspects of special process control are checked for compliance at the contractor's facilities and at the construction site:

A. Training, testing, and qualification of operator and inspection personnel involved with special process operations.

B. Certification of equipment utilized in the performance of special process operations.

C. Documentation of results.

17.1.10 INSPECTION

17.1.10.1 Contractors

For equipment purchased by APC, the vendor quality surveillance program for contractors is administered by SCS, implemented by companies providing source surveillance, and periodically audited by APC's MQA. The vendor shop quality control surveillance program for equipment purchased by Daniel is administered and implemented by Daniel and periodically audited by APC's MQA.

APC's quality surveillance representatives (QSRs) are required to verify through a program of scheduled surveillance activities that the contractor is abiding by approved procedures, purchaser specifications, and codes during the fabrication process. Specific areas requiring *surveillance include fabrication practices, dimensional accuracy, cleaning, NDE procedures and documentation, and packaging and shipping procedures and documentation. Mandatory hold points are established; the QSR is required to witness and evaluate such tests or procedures as required. Performance tests required by specification may also be witnessed by the QSR.*

Each surveillance visit is documented in a formal report which is distributed for review and evaluation. Deficiencies noted during surveillance are reaudited to ensure prompt and satisfactory closeout.

Deviations from approved specifications, repairs, and corrective procedures are documented and submitted to the appropriate design organization for evaluation and approval prior to the equipment being released for shipment by the QSR.

17.1.10.2 Construction Site

Work performed at the construction site is inspected to ensure compliance with applicable contracts, purchase orders, specifications, and drawings. This effort is conducted by the Construction Department Quality Control Group and Daniel's Quality Control Group. The division of responsibility is clearly defined by QC procedures prepared by the respective groups; the division of responsibility is approved by the MQA. Each group is composed of inspectors and technicians who are thoroughly familiar with the specifications, drawings, codes, welding procedures, and NDE procedures applicable to their discipline. Inspection activities may be performed as required by independent testing laboratories.

All inspections are documented and reviewed to ensure that all requirements are satisfactorily fulfilled.

APC's MQA and QAFR audit all inspection and testing activities to ensure that approved procedures are being utilized.

17.1.11 TEST CONTROL

The testing program for the FNP includes all tests necessary to demonstrate that structures, systems, and components will perform satisfactorily in service. This program is organized into the categories expounded in the following subsections.

17.1.11.1 Contractor Tests

Procurement documents require that performance tests be performed by contractors on specific materials and equipment purchased from them. Test requirements and acceptance criteria are provided in the specification by the organization responsible for the design of the item to be *tested. Testing is performed in accordance with approved written test procedures and incorporates all requirements contained in the applicable design documents. Procurement documents require that test results be documented and submitted to the applicable design organization for evaluation and acceptance.*

17.1.11.2 <u>Construction Proof Tests</u>

APC's QAM and site contractors' QAMs require that specific testing be performed onsite during construction of the FNP. Such tests include but are not limited to soil tests, rebar splice tests, concrete tests, vacuum box tests, and other special tests as may be required. Such tests are performed in accordance with previously approved procedures requiring results to be documented. Tests results are evaluated and accepted when they are in compliance with engineering requirements.

17.1.11.3 <u>Construction Testing</u>

A division of responsibility between APC's Construction and Nuclear Generation Departments has been established delineating functions and responsibilities concerning the FNP testing program. At the completion of construction, systems and components will be turned over to the Nuclear Generation Department for system and component testing.

17.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

17.1.12.1 <u>Contractor Facilities</u>

Prior to the award of any contract for equipment or services, the QA programs of each contractor are reviewed by APC's MQA to ensure that procedures are defined for the control of measuring and test equipment. The procedures are evaluated for compliance with the following:

A. *Identification of equipment by serial number or the equivalent.*

B. *Frequency of calibration schedule.*

C. *Preparation and maintenance of calibration records to indicate identity of equipment, date of calibration, and due date for recalibration.*

D. *Assurance that equipment is removed from service when calibration date is exceeded or when equipment is damaged or suspected to be inaccurate.*

E. Proper handling and storage facilities for equipment.

For APC-purchased equipment, control of measuring and test equipment is audited by APC inspectors for conformance to procedures. For Daniel-purchased equipment, Daniel inspection personnel perform this audit function.

17.1.12.2 <u>Construction Site</u>

The QAM requires that procedures be established for the control of measuring and test equipment to ensure that inspection and testing of material and equipment at the construction site is performed with devices that are properly calibrated.

Onsite contractors are audited by Daniel and the APC QAFR for compliance with procedures for the control of measuring and test equipment.

17.1.13 HANDLING, STORAGE, AND SHIPPING

17.1.13.1 <u>Contractor Facilities</u>

Procurement documents are reviewed by APC's MQA to ensure that special handling, storage, shipping, cleaning, and preservation requirements are included.

The QA programs of contractors providing items or services are evaluated to ensure that adequate procedures exist for the special handling, storage, shipping, cleaning, and preservation of materials and equipment.

For APC-procured equipment, compliance with approved procedures is ensured through the shop surveillance program administered by SCS. Shop inspection for Daniel-procured equipment is implemented by Daniel and procedure compliance is verified by them.

17.1.13.2 <u>Construction Site</u>

The QAM requires that procedures be established for the control, identification, protection, and handling of material and equipment from the time they are received onsite until turnover to APC's Nuclear Generation Department. These procedures require:

A. Adherence to suppliers instructions for storage and handling equipment.

B. Special storage areas and facilities for various types of materials and equipment.

C. Special storage methods for various types of materials and equipment.

D. Inspections to be performed during the storage period.

E. Identification and marking of equipment to enable tracing of its source of documentation.

F. Control steps to ensure that material and equipment are used only as indicated by approved design documents.

G. Special handling tools and equipment to ensure safe and adequate handling.

H. Records of receipt and storage inspections.

I. Identifying nonconforming items.

Compliance with approved procedures is ensured by the construction site audit program conducted by Daniel and the APC QAFR.

17.1.14 INSPECTION, TEST, AND OPERATING STATUS

17.1.14.1 Contractor Facilities

The QA programs of contractors are reviewed by APC's MQA to ensure that adequate control exists to identify the status of required inspections and tests. Generally, a document, such as a shop traveler, is required to accompany a component or assembly throughout the manufacturing, inspection, and testing process. This document lists the required activities and provides for signature of the individual responsible for accepting them. These documents are retained by the contractor for use and retention by APC as required.

Tagging procedures or the equivalent are evaluated to ensure that contractors have some means for identifying the inspection status of a component or assembly.

17.1.14.2 Construction Site

Items arriving onsite are accompanied by documented evidence which ensures that all requirements of the procurement documents have been satisfied.

The QAM requires that procedures be established for maintaining the inspection and test status of items. The procedure provides for the methods that are used for identifying material and equipment received at the construction site and for controlling their status throughout the *construction phase in accordance with approved design documents. This is accomplished by the use of tags which are affixed to and remain on the items from receipt through installation inspection.*

17.1.15 NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

17.1.15.1 Contractor Facilities

Contractors' QA programs are reviewed by APC's MQA to ensure that procedures are defined for the control of nonconforming materials, parts, or components. The procedures are evaluated to include the following:

- *A. Identification of the nonconforming item.*
- *B. Documentation of the nonconformance.*
- *C. Segregation of the nonconforming item.*
- *D. Disposition of the nonconformances and notification of the affected organization.*

For APC-procured equipment, compliance with approved procedures is ensured through the shop surveillance program administered by SCS. Shop inspection for Daniel-procured equipment is implemented by Daniel and procedure compliance is verified by them.

17.1.15.2 Construction Site

The QAM requires procedures for the control of nonconforming materials, parts, or components. Compliance with approved procedures is ensured by the construction site audit program conducted by Daniel and APC's QAFR. The procedures include the following:

- *A. The responsibility and authority for the identification, reporting, and resolution of nonconformances.*
- *B. The classification of nonconformances into accept, repair, rework, and reject categories.*
- *C. The method by which nonconformances are identified, documented, segregated, and by which affected organizations are notified and resolution is reached.*

D. The means by which the deviant item is processed to fulfill the requirements of the directed resolution.

17.1.16 CORRECTIVE ACTION

17.1.16.1 Contractor Facilities

APC's MQA reviews contractors' QA programs to ensure that adequate procedures are in effect which govern the identification and disposition of conditions adverse to quality. The procedures are evaluated to ensure the documentation of nonconformances or deficiencies, and to ensure that measures for corrective action and trend analysis to prevent recurrent problems are provided.

17.1.16.2 Construction Site

The QAM requires procedures be established for identifying, reporting, resolving, recording, and analyzing construction site conditions adverse to quality. The procedures include the following:

A. A method to appropriately mark or identify nonconforming items so that before related work continues, a course of corrective action is established.

B. The means for reporting nonconformances and the actions taken to resolve them.

C. Identification of the groups and/or persons having authority to approve the resolution of nonconformances.

D. Identification of the groups and/or persons who shall, for information purposes, be made aware of the nonconformances.

E. The means by which nonconformances are resolved.

F. A system for keeping adequate records of nonconformances and for periodically reporting on the status to management.

G. A method for analyzing nonconformances to determine appropriate corrective actions based on trend, rate, and occurrence.

Compliance with approved procedures is verified by the construction site audit program conducted by Daniel and APC's QAFR.

17.1.17 QUALITY ASSURANCE RECORDS

Procurement documents delineate the QA records that are to accompany or precede equipment to the construction site and specify those records which are to be maintained by the manufacturer in his facility. Approved procedures, referenced in the QAM, require construction site-generated records which reflect the as-built condition of items in the plant.

As QA data is received from manufacturers, it is checked against procurement document requirements to ensure that no nonconformances or deficiencies exist. Site-generated records are reviewed for compliance with construction requirements.

QA records for the FNP are collected, evaluated, cataloged, and maintained by Daniel. Daniel employs a filing system which provides for easy identification and access to all records and which encompasses all of the systems and components of the FNP. All documentation for a specific system is filed together with a subfiling for each component within the system. Records include, as applicable, all QA records received from the manufacturer; records of shop inspections; records of field inspections, tests, and audits; records of personnel qualifications and procedures; and all other supplementary records which may be generated.

QA records received by APC are part of the permanent records of the FNP and will be retained at the plant site in accordance with applicable requirements. Those records retained by a manufacturer are available to APC if needed.

17.1.18 AUDITS

17.1.18.1 Design Audits

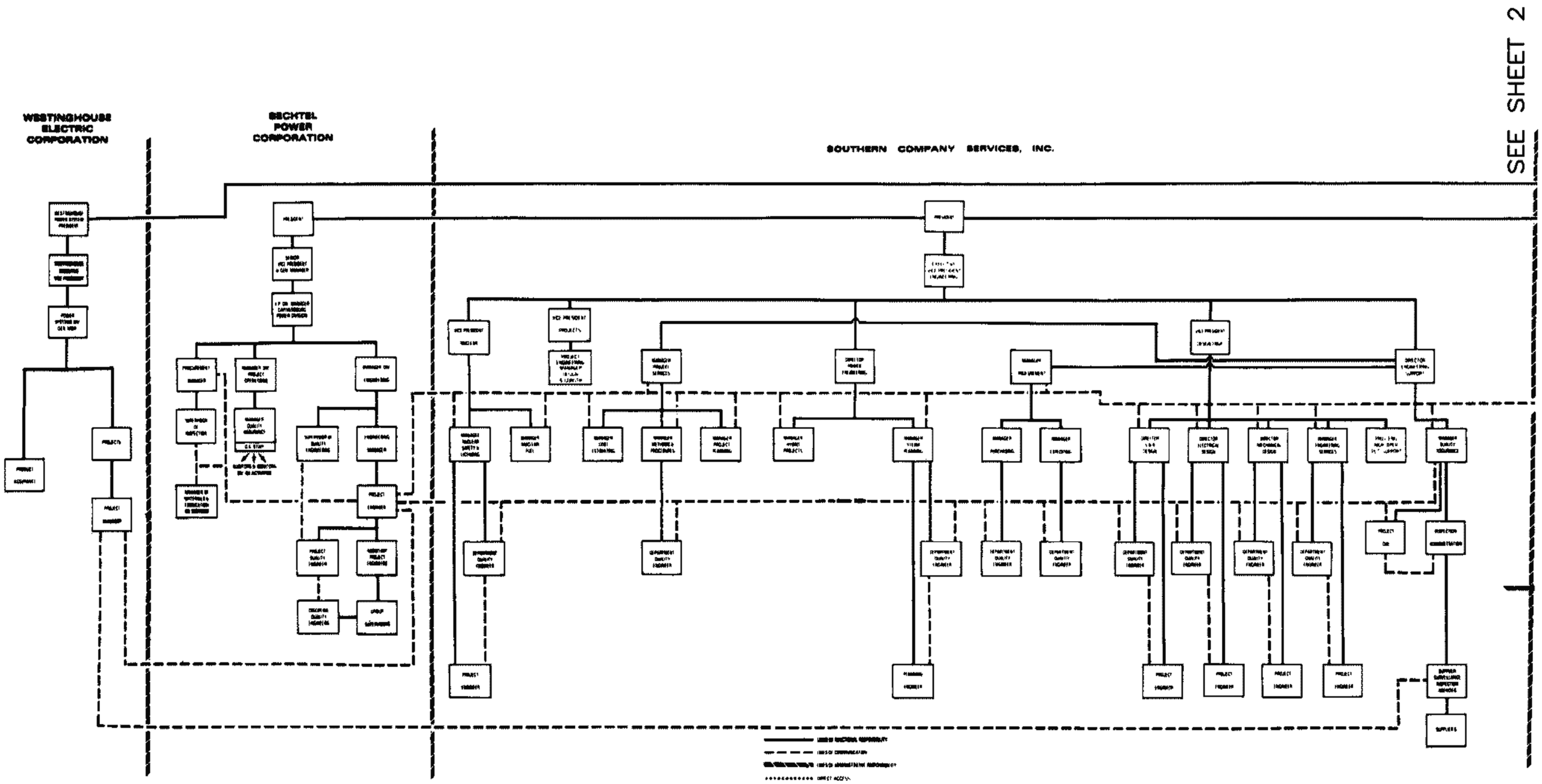
Design audits are intended to evaluate the design organization for compliance with procedures, codes, specifications, and other pertinent areas. On a semiannual basis, the design activities of SCS, Bechtel, and Westinghouse are audited based upon a prepared checklist. The audit of SCS and Bechtel is performed by SCS's manager of quality assurance, and the audit of Westinghouse is performed by Bechtel's project quality engineer. APC's MQA or a representative from his staff participates in the audit of each design organization. Audit reports are prepared and forwarded to appropriate management levels for review.

17.1.18.2 <u>Construction Site Audits</u>

To verify compliance with and to determine the effectiveness of the construction site QA program, audits are conducted by APC's QAFR. Audits are based upon a prepared checklist and are performed every two weeks on specific areas of work at the construction site. On a periodic basis, APC's MQA or a representative from his general office staff assists in performing the audit. Results are documented and audit findings reviewed with appropriate levels of management from the organizations audited. Open items are closely monitored until resolved. Every two weeks, Daniel's project quality assurance manager conducts audits on Daniel's construction site activities. The audits are based upon a prepared checklist and are performed by appropriately trained personnel not having direct responsibilities in the areas being audited. Results are documented and distributed to responsible Daniel management personnel and to APC's MQA for review. Open items are closely monitored until resolved.

17.1.18.3 <u>Vendor Audits</u>

*On a selective basis, the MQA or members from his staff accompany SCS or Daniel personnel on audits of vendor facilities. These audits consist of checks on specific areas of a vendor's operation; results are documented and distributed to appropriate management personnel.***]**



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SOUTHERN COMPANY
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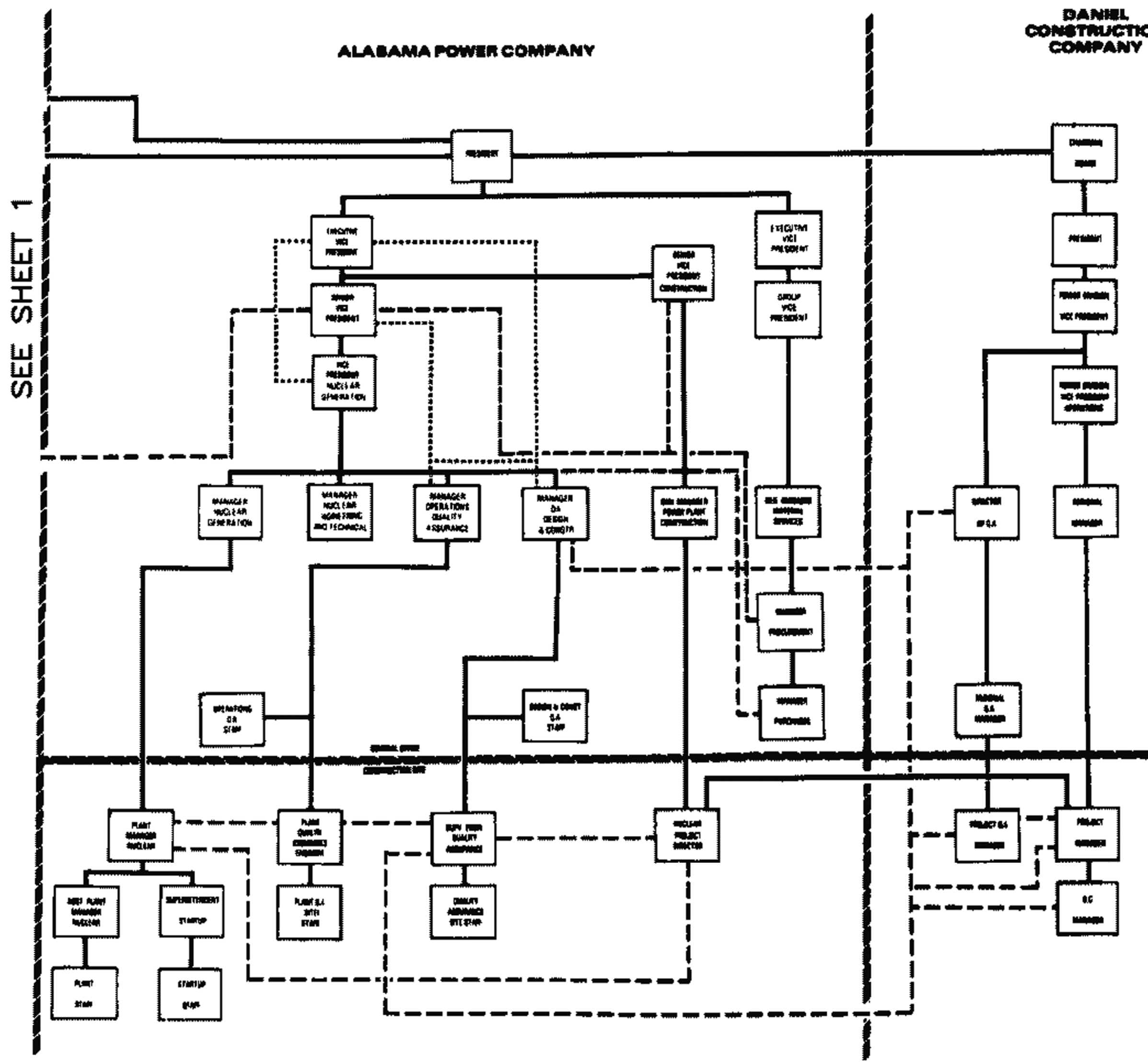
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PROJECT ORGANIZATION CHART FOR QUALITY ASSURANCE

FIGURE 17.1-1 (SHEET 1 OF 2)]



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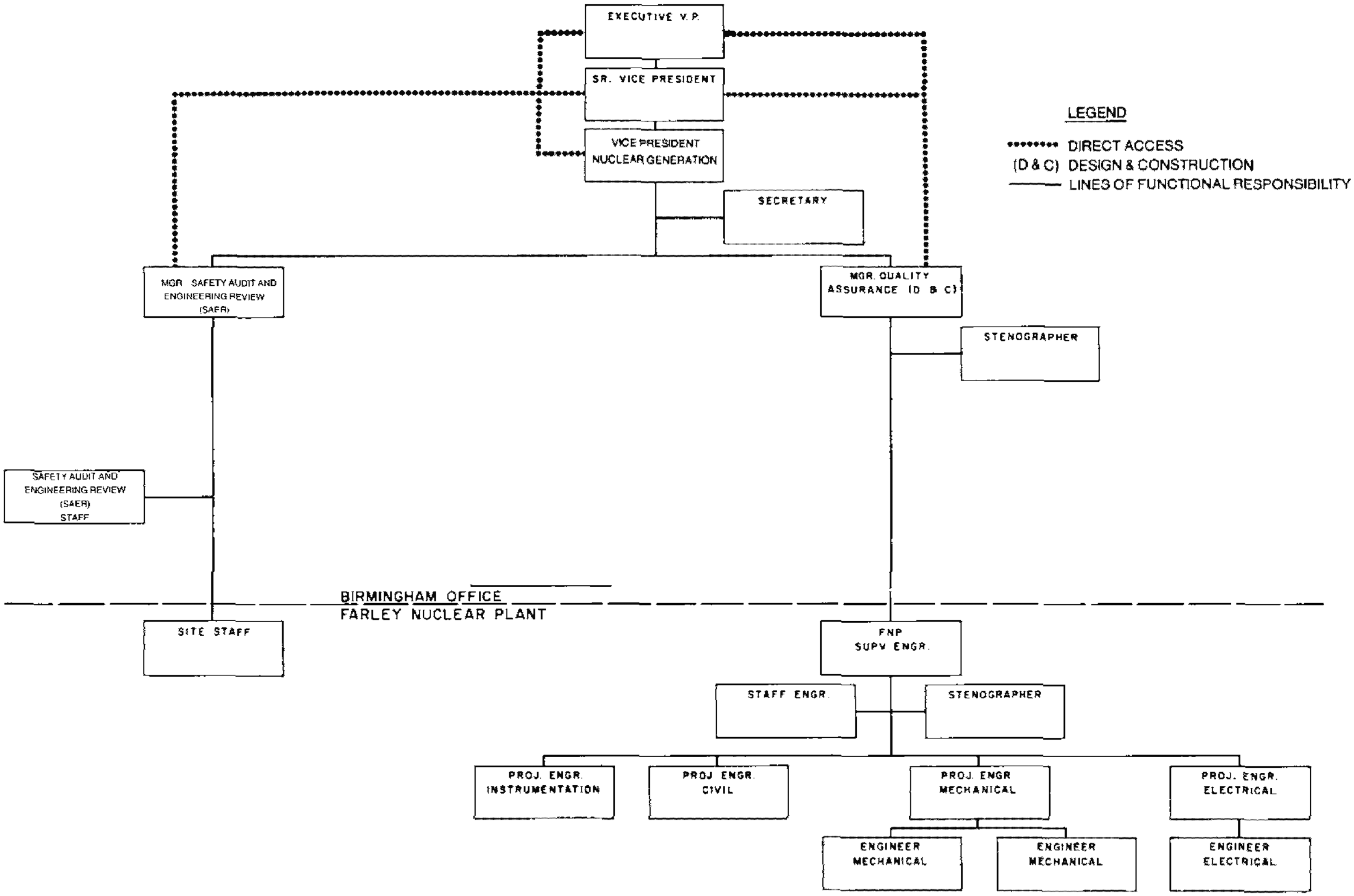
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PROJECT ORGANIZATION CHART FOR QUALITY ASSURANCE

FIGURE 17.1-1 (SHEET 2 OF 2)]



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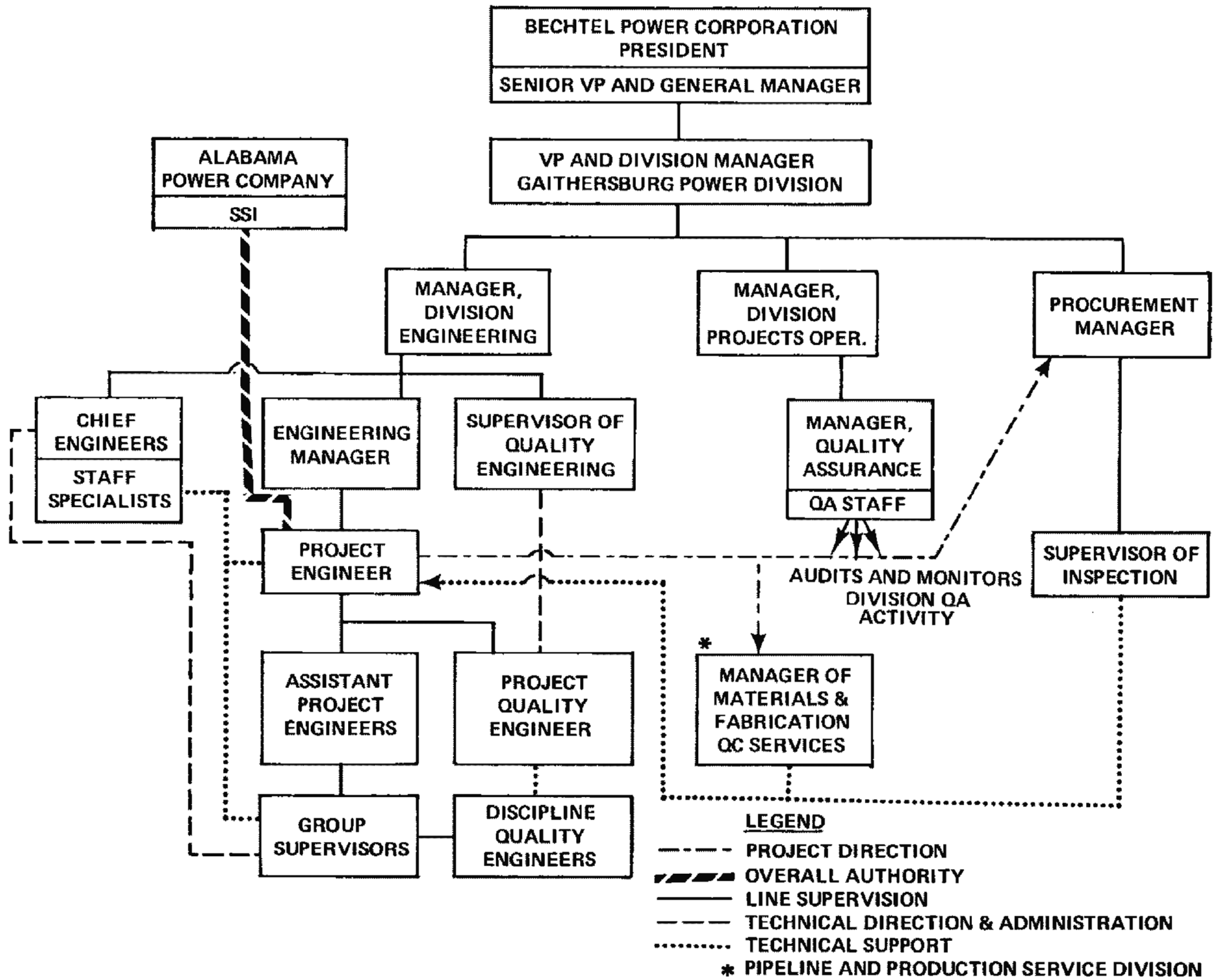
NUCLEAR PLANT
UNIT 1 AND UNIT 2

*[ALABAMA POWER COMPANY
QUALITY ASSURANCE DESIGN AND
CONSTRUCTION ORGANIZATION*

FIGURE 17.1-2]



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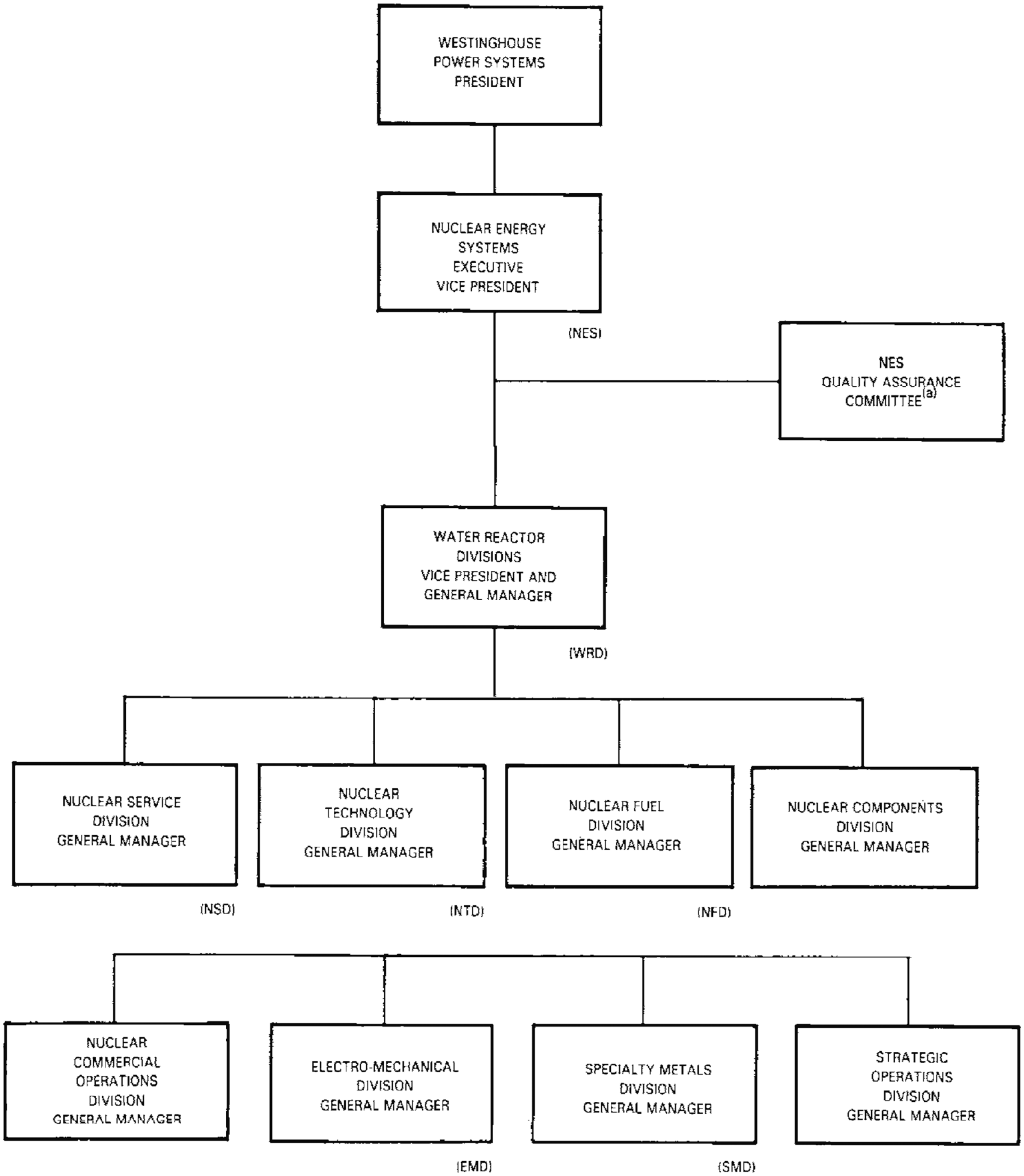
UNIT 1 AND UNIT 2

[BECHTEL POWER CORPORATION
GAITHERSBURG POWER DIVISION
QUALITY ASSURANCE ORGANIZATION
JOSEPH M. FARLEY NUCLEAR PLANT

FIGURE 17.1-3]



a) THE NES QUALITY ASSURANCE COMMITTEE IS COMPOSED OF THE QUALITY ASSURANCE AND RELIABILITY MANAGERS FROM EACH OF THE NES DIVISIONS. THE COMMITTEE'S CHAIRMAN IS THE NTD PRODUCT ASSURANCE MANAGER.

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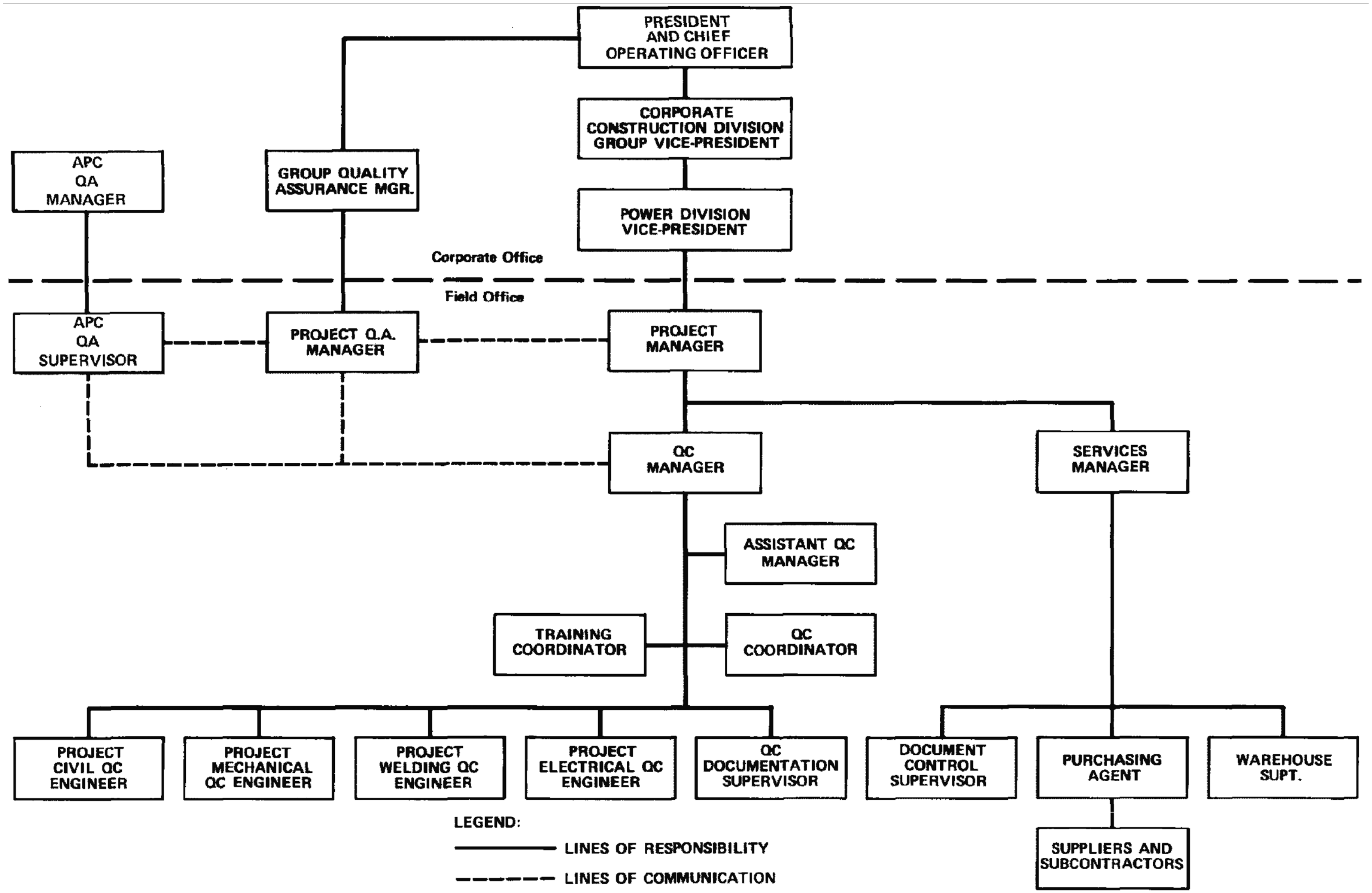
SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	*[NUCLEAR ENERGY SYSTEMS ORGANIZATION* *FIGURE 17.1-4]*



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

*[DANIEL CONSTRUCTION COMPANY
OF ALABAMA
QUALITY ASSURANCE ORGANIZATION*

FIGURE 17.1-5]

17.2 OPERATIONS QUALITY ASSURANCE PROGRAM (OQAP)

The operations-phase quality assurance (QA) program for Farley Nuclear Plant (FNP) is designed to assure the plant's safe and reliable operation and to satisfy the QA requirements of Appendix B to 10 CFR Part 50. The QA program applicable to operations-phase activities for FNP is described in the Southern Nuclear Operating Company (SNC) Quality Assurance Topical Report (QATR). QA program requirements formerly contained in FNP FSAR section 17.2 are superseded by those contained in the SNC QATR.

TABLE 17.2-1

ONSITE QA STAFF

This table has been deleted.

TABLE 17.2-2

TYPICAL AUDIT FREQUENCIES

This table has been deleted.

TABLE 17.2A-1

CLASSIFICATION OF INDEPENDENT SPENT-FUEL STORAGE INSTALLATION'S STRUCTURES AND COMPONENTS

I. CATEGORY A

- Items specified as Category A in the dry cask storage vendor's Topical Safety Analysis Report (TSAR), unless other category is assigned by this document.

II. CATEGORY B

- Items specified as Category B in the dry cask storage vendor's TSAR, unless other category is assigned by this document.

III. CATEGORY C

- Items specified as Category C in the dry cask storage vendor's TSAR, unless other category is assigned by this document.
- Concrete storage pad
- ISFSI Soil Test and Analysis
- Roadways for transport of cask and associated equipment

IV. NOT IMPORTANT TO SAFETY

- Items specified as not important to safety by the dry cask storage vendor's TSAR, unless other category is assigned by this document.
- Security System
- Dose rate boundary fence
- Facility lighting
- Electric power system and backup
- Railways for transport of cask and associated equipment

(DELETED)

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SOUTHERN NUCLEAR OPERATING COMPANY
FARLEY PROJECT

FIGURE 17.2-1

SEE FIGURE 13.1-6

13

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FACILITY ORGANIZATION

FIGURE 17.2-2

17.3 JOSEPH M. FARLEY NUCLEAR PLANT QUALITY ASSURANCE Q-LIST

17.3.1 INTRODUCTION

The Q-List consists of those structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Any item contained in the current Q-List that was properly designated and installed as a non-Q item will not be modified, upgraded, or replaced as a result of this subsequent change in designation. However, when these items are replaced, modified, or repaired, the appropriate quality assurance provisions will be applied to the replacement, modification, or repair parts.

The quality assurance provisions to be applied to the structures, systems, and components on the list will be consistent with the safety function which that structure, system, or component is to perform.

17.3.2 GROUP 1 - STRUCTURES

A. Containment Building

1. Substructure

a. Concrete

b. Rebar

c. Liner plate

2. Superstructure

a. Concrete

b. Rebar

c. Structural steel

d. Liner plate

e. Tendon system

f. Tendon grease

g. Hatches - equipment, auxiliary and personnel

h. Penetrations - electrical and mechanical (includes nonsafety systems that penetrate containment boundary up to and including the containment isolation valves)

3. Painting and special coatings

4. Biological shielding

5. Missile barriers (table 3.5-1)

B. Auxiliary Building

1. Substructure

a. Concrete

b. Rebar

2. Superstructure

a. Concrete

b. Rebar

c. Superstructure steel

3. Biological shielding

4. Missile barriers (table 3.5-6)

5. All masonry walls in proximity to or with safety-related equipment attached to them

C. Service Water System

1. Storage pond and dam

2. Service water intake structure at pond

3. Foundation soil for structures and embedded piping

4. All masonry walls in proximity to or with safety-related equipment attached to them

D. Diesel Generator Building

1. Foundation soil and/or piles

2. Concrete

3. Rebar

4. Structural steel

5. All masonry walls in proximity to or with safety-related equipment attached to them

E. Vent Stack

Note: The vent stack is functionally non-nuclear safety (NNS); however, the appropriate criteria of 10 CFR 50, Appendix B, are applied to maintain the structure as Seismic Category 1 to prevent a failure that could impact safety-related structures, systems, and components.

F. Cable Tunnel Structure

1. Concrete

2. Rebar

17.3.3 GROUP 2 - MECHANICAL SYSTEMS

A. Reactor Coolant System (RCS)

1. Reactor vessel and associated equipment, including:

a. Vessel shell

b. Vessel head with control rod drive mechanism (CRDM) adapters

c. Upper internals assembly

d. Lower internals assembly

e. Control rod guide tubes

f. Control rod drive mechanism adapter plugs

g. Integral support pads and brackets

h. Support shoes and shims

i. Closure studs, nuts, and washers

j. Flange leak-off stub

k. Reactor neutron panels

l. Control rod assemblies

m. Fuel assemblies

n. Core support structure

o. Reactor vessel internals other than A.1.m, A.1.n, and A.1.o above

2. Reactor vessel supports

3. Control rod clusters

4. Control rod drive mechanism housing

5. Nuclear instrumentation - out of core

6. Steam generators, including:

   a. Shell and tubes

   b. Integral support pads

   c. Steam generator external supports

   d. Steam generator steam flow restrictors

7. Pressurizer, including:

   a. Relief and safety valves

   b. Integral support pads

   c. Pressurizer external supports

d. Pressurizer spray nozzle assembly (RCPB components only: spray head is NNS.)

8. Reactor coolant pumps, including:

a. Motor supports

b. Motor flywheel

c. Two seal assemblies nearest high-pressure coolant (seal assemblies 1 and 2)

d. Reactor coolant pump seal bypass orifice

9. Reactor coolant system, including:

a. Branch lines up to and including the second isolation valve or first valve outside the containment

Exempt from this list are the pressurizer safety valve water seal drain lines downstream of the first manual isolation valve and the pressurizer vent line beyond the first manual isolation valve

b. Reactor coolant piping

B. Residual Heat Removal (RHR) System

1. RHR piping system

2. Residual heat removal pumps and motors (low-head safety injection pumps; same as E.4)

3. Residual heat removal heat exchangers

C. Containment Cooling System

1. Containment air cooler piping system

2. Containment air cooler fans and drives

3. Containment air cooler coils and housings

4. Fan discharge transition and fusible link plate (Containment ductwork dampers and supports are not Q; fusible links disconnect the ductwork from the cooler discharge after a LOCA. Fan motors and fusible links are Q.)

D. Containment Spray System

1. Containment spray piping system

2. Containment spray nozzles

3. Spray additive eductors (pressure boundary only)

4. Refueling water storage tank

5. Containment spray pumps and motors

E. Emergency Core Cooling System (ECCS)

1. ECCS piping systems

2. Accumulator tanks

3. High-head safety injection pumps (charging pumps) and motors (same as H.5)

4. Low-head safety injection pumps (RHR pumps) and motors (same as B.2)

5. Containment sump

6. Containment sump screening apparatus

F. Spent-Fuel Pool

1. Liner plate

2. Storage racks

3. Fuel transfer tube and blind flange

4. Concrete structure

G. New Fuel Storage Racks

H. Chemical and Volume Control System (CVCS) (Excluding Boron Recycle Loop)

1. Piping systems

2. Volume control tank

3. Boric acid tanks

4. Reactor makeup water storage tank

5. Charging/high-head safety injection pumps and motors (same as E.3)

6. Boric acid transfer pumps and motors

7. Nonregenerative letdown heat exchanger

8. Excess letdown heat exchanger

9. Regenerative heat exchanger

10. Seal water heat exchanger

11. Reactor coolant filter pressure housing only

12. Boric acid filter pressure housing only

13. Seal water injection and return filter housing only

14. Boric acid blender

15. Letdown orifices

I. Waste Disposal System

1. Mechanical alternators, instrumentation, and controls which give operating status of sump pumps in RHR pump rooms, containment spray pump rooms, and other rooms where sump pumps are part of the leak detection system

J. Main Steam System

1. Main steam piping system from steam generator up to and including the first isolation valve outside containment

2. Main steam safety and relief valves, isolation valves, and associated piping system for main steam headers

K. Condensate and Feedwater System

1. Feedwater piping system from steam generator up to and including the first isolation valve outside the containment

2. Condensate storage tank

L. Auxiliary Feedwater System

1. Auxiliary feedwater piping system, including supply lines from service water system

2. Auxiliary feedwater pumps and drives

3. Steam piping to auxiliary feedwater pump steam turbine

M. Component Cooling System

1. Component cooling water (CCW) piping system to safeguard equipment and associated valves

2. Component cooling surge tank

3. Component cooling pumps and motors

4. Component cooling heat exchangers

N. Emergency Diesel Generator System

1. Diesel fuel oil transfer piping system

2. Diesel generator system packages

3. Diesel fuel oil storage tanks

4. Diesel fuel oil day tanks

5. Diesel fuel oil transfer pumps and motors

6. Diesel generator building ventilation fans

O. Containment Cranes

1. Reactor cavity manipulator crane

2. Containment polar crane

P. Auxiliary Building Cranes

1. Spent-fuel pool bridge crane and hoist

2. Cask crane

Q. Penetration Room Filtration System

1. Penetration room filtration fans and drives

2. Penetration room filters and housing

3. Penetration room ductwork and isolation valves

R. Control Room Ventilation System

1. Air conditioning refrigeration system

2. Supply air handling units and drives

3. Filtration fans and drives

4. Filtration filters and housing

5. Isolation valves

S. Pump Room Ventilation Systems

1. High-head safety injection pump room cooler fans and drives, cooling coils, and housings

2. Low-head injection pump room cooler fans and drives, cooling coils, and housings

3. Component cooling pump room cooler fans and drives, cooling coils, and housings

4. Containment spray pump room cooler fans and drives, cooling coils, and housings

5. Auxiliary feedwater pump room cooler fan and drives, cooling coils, and housings

T. Service Water System

1. Service water piping system

2. Service water strainers

3. Service water pumps and motors

4. Service water intake structure heating and ventilation system

U. Spent-Fuel Cooling System

1. Spent-fuel pool cooling system piping

2. Spent-fuel pool heat exchangers

3. Spent-fuel pool pumps

V. Post-LOCA hydrogen Control System

1. Post-LOCA hydrogen recombiners

2. Containment Post-LOCA hydrogen mixing system

3. Post-LOCA containment hydrogen monitoring equipment (The purge supply system inside containment, the containment penetrations, and associated penetration isolation valves are Q. Beyond the isolation valves outside containment, the system receives supply air from the instrument air system, which is non-Q.)

W. Nonsafety Systems

Nonsafety systems that penetrate containment are Q up to and including the containment isolation valves

X. Sampling System

Sampling system lines connected to safety system components are Q up to and including the containment isolation valves

17.3.3.1 <u>Notes on Group 2 - Mechanical Systems</u>

Where a piping system is specified, such system includes the necessary valves, supports, and restraints.

For each system, those portions of the instrumentation and controls that are safety related are included with that system. Q instruments are identified as such in the applicable instrument indexes.

Motor operators for active valves are Q. Active valves are defined as valves which must change position to mitigate the consequences of a design basis accident. Nonactive valves are not required to change positions to mitigate accidents.

17.3.4 GROUP 3 - ELECTRICAL SYSTEMS

A. 4160-V Switchgear (Engineered Safeguard Buses)

B. 4160-V to 600-V transformers (Associated with Engineered Safeguard Systems)

C. 600-V Load Centers (Engineered Safeguard Buses)

D. 600-V and 208-V Motor Control Centers (Associated with Engineered Safeguard Systems)

E. DC Electrical Distribution System (Auxiliary Building and Service Water Building)

1. 125-V dc station batteries

2. Inverters, 125-V dc to 120-V ac (vital instrument buses and control rod drive indicator)

3. 125-V dc distribution panels

4. 125-V dc switchgear

5. 125-V dc battery chargers

6. Battery racks

E. Vital ac Instrumentation and regulated ac distribution panels

F. Control Panels and Vertical Control Boards

1. Protective relay boards and racks, safeguard systems

2. Protective relay boards and racks, reactor protection systems

3. Instrument boards and racks, safeguard systems

4. Instrument boards and racks, reactor protection systems

G. Class 1E Supports for Conduits and Trays

H. Class 1E Power Cables

I. Class 1E Instrumentation and Control Cables

J. DC Emergency Lighting - Battery Pack

K. Onsite AC Power Systems

1. Diesel generators, including auxiliaries

2. Transformers (4160-V-600-V transformers which are part of the Class 1E 600-V switchgear)

3. Protective relays (those mounted in Class 1E switchgear)

4. Containment electrical penetrations

L. Motor Operators

Motor operators for active valves are Q

M. Turbine-Driven Auxiliary Feedwater Pump Uninterruptible Power Supply

17.3.5 GROUP 4 - OTHER SYSTEMS

The following systems were not originally purchased as Q systems. Repairs and modifications to these systems will be documented as if these systems were purchased on a level of quality similar to other systems on this list. Replacement parts will be purchased to meet the original specifications.

A. Reactor Coolant Leakage Detection System

1. Containment Air Particulate Monitor (R-11)

2. Containment Radioactive Gas Monitor (R-12)

3. Condensate Measuring System

4. Dewpoint Measuring System

B. Containment Mini-Purge System Fans

C. Vent Stack Monitor (R-29B/R-29D, and Unit 1 and Unit 2 R-29C)

D. Main Condenser Air Removal Monitor (R-15B, R-15C)

E. Main Steam Line Monitors R-60 (A through C)

17.3.6 GROUP 5 - EXPENDABLE AND CONSUMABLE ITEMS

Administrative procedures ensure that applicable regulatory requirements, design bases, and other quality assurance requirements are included or referenced in procurement documents for expendable and consumable items necessary for the functional performance of critical structures, systems, and components.

APPENDIX 17A

SOUTHERN COMPANY SERVICES, INC. QUALITY ASSURANCE PROGRAM

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LIST OF FIGURES

[HISTORICAL]
[APPENDIX 17A

SOUTHERN COMPANY SERVICES, INC. QUALITY ASSURANCE PROGRAM

Appendix 17A contains historical information implemented during design and construction of FNP. The current quality assurance policies are delineated in the SCS Quality Services Policy and Procedures Manual.

17A.1.1 ORGANIZATION

The executive vice president - engineering has overall responsibility for quality of all services performed by Southern Company Services, Inc. (SCS) as the architect/engineer (A/E) on the Farley Plant. The SCS quality assurance (QA) program is implemented by all concerned departments. Each department manager, through his project engineer and other department members, is directly responsible for the quality of the products of that department. Procedures for the preparation and quality verification of work accomplished by the department are contained in corporate, departmental, and project procedures. First-line quality verification of design calculations, drawings, and other documents is accomplished by engineers other than those performing the work. Additional reviews are made by supervisory personnel and/or other engineers. Audits to verify quality are performed by the SCS Quality Assurance Department.

The Quality Assurance Department verifies that engineering procedures and other departmental and project procedures affecting quality are followed through audits and other monitoring techniques. This department is also responsible for assuring that other agencies, such as outside consultants and the nuclear steam supply system (NSSS) supplier, who are contracted by SCS to provide portions of the nuclear plant design, maintain adequate quality assurance programs through coordination of quality activities and audits. Another responsibility includes administration of the vendor shop inspection program through which vendors are qualified and monitored for quality performance by inspectors assigned to the vendors' shops. This is accomplished principally through contracts with vendor shop inspection agencies. The SCS QA Department participates in a number of these audits and inspections on a selective basis.

The manager - quality assurance department reports to the director - engineering support services, who in turn reports to the executive vice president - engineering. This provides a separate line of authority independent of the departments and groups involved in the design work.

The organization chart for the SCS quality assurance program is shown in figure 17A-1.

17A.1.2 QUALITY ASSURANCE PROGRAM

The SCS quality assurance program assures that all of the tasks performed by Southern Company Services in their role as architect/engineer on plants designed for the operating companies of The Southern Company are in accordance with the quality standards of SCS and meet the intent of 10 CFR 50, Appendix B.

The primary responsibility of quality rests with the department responsible for the design and procurement of a given item or system. This quality is verified by procedures that provide for independent checks and reviews of all design documents by engineering personnel other than those originating the work. Additionally, design audits are conducted by the SCS QA Department with participation by operating company personnel to assure that quality program procedures are utilized throughout the design and procurement phases.

The Southern Company Services Quality Assurance Department is responsible for:

- *A. Coordinating and administering the quality aspects of design, procurement, and other related functions within SCS, and providing interface for quality activities with operating companies in The Southern Company.*
- *B. Coordinating the auditing of quality programs of contractors and vendors providing services and materials for SCS.*
- *C. Administering the vendor shop surveillance program to assure that materials and equipment manufactured for SCS meet the desired quality.*

The SCS Engineering Policy and Procedures Manual (EPPM) contains procedures governing the operation of the SCS quality assurance program. The procedures are designed to provide guidelines for achieving the established goals of the program.

17A.1.2.1 <u>Design-Related Procedures</u>

Quality assurance program procedures establish quality guidelines to follow in providing design, verification, and documentation control.

QA procedures assist in the implementation of controls over design-related activities such as the following:

- *A. Working relationships among organizations involved in the program, such as owner, architect/engineer, and NSSS supplier.*

B. Administrative and technical instruction within design organizations, such as procedure manuals and guidelines for performing technical work.

C. Information exchange across external and internal interfaces.

D. Document control, including review, approval, release, distribution, and revision of documents.

E. Record keeping of the evolution of relevant work changes and final issues of them, which shall be complete, applicable, accessible, and understandable.

F. Keeping management apprised of the quality posture of the program.

17A.1.2.2 Procurement-Related Procedures and Other Procedures

In addition to procedures related directly to the design function, quality assurance program procedures assist in implementing controls of SCS procurement-related activities such as:

A. Initial quality planning which provides guidelines for inclusion of quality requirements in procurement documents issued by SCS.

B. Procedures used for the review and approval of vendor's quality assurance programs.

C. Procedures used for assignment of vendor shop inspection.

D. Procedures used for performing audits of vendors.

E. Routines used for assuring that all deficiencies are corrected and documented.

F. Procedures used to assure that the product arriving at the site is exactly as specified and approved.

G. Routines used for assuring that all site deficiencies related to vendor-supplied material or equipment are conducted as documented.

17A.1.3 DESIGN CONTROL

The SCS EPPM provides for control of engineering design and assures that technical and quality requirements are met.

General design criteria are developed from the current sections of applicable codes, standards, regulations, NRC Regulatory Guides, and safety analysis reports (SARs), including all appendixes, addenda, and references as contained in the design section of the preliminary safety analysis report (PSAR).

Functional design criteria are developed from basic plant capacity requirements determined by extensive studies involving load demand, location, timing, and overall economics. These inputs include:

A. *Number of generating units.*

B. *Type of units.*

C. *Capacity.*

D. *Location.*

E. *Other physical and functional requirements.*

Procedures provide for the independent review and checking of design documents, calculations, basic functional and physical criteria, drawings, flow sheets, applicability of materials, parts, and processes to the desired performance, and other engineering information. Engineering documents are prepared by personnel assigned to the project from the appropriate engineering department. The documents are given a comprehensive check and are reviewed by other personnel having technical qualifications comparable to those of the originator.

Identification and control of design interfaces are accomplished by use of document and correspondence handling procedures located in the Farley Project Procedure Manual. These procedures provide a routing and review system to assure that each document is reviewed and approved by the appropriate groups at the proper time. Periodic audits of the system by the Quality Assurance Department provide assurance that the system is operating properly.

Engineering group supervisors are responsible for review and approval of engineering documents. Dependent upon their nature, engineering documents may require approval by specialists, engineering managers, or other departments within SCS.

Control of changes and/or deviations from approved design practices are described in the SCS EPPM. Design changes are reviewed and approved in a manner similar to that used in the control of original design drawings and other documents. Final drawings will reflect all changes and provide an as-built set for retention.

Regular Farley Project meetings are held and problems, changes, and progress are discussed among all concerned groups within and outside of SCS. Design work and specifications originating within SCS are reviewed by Bechtel Engineering while work originating within

Bechtel is audited by SCS. Design work and specifications originating with a vendor are reviewed and/or audited by SCS and Bechtel.

17A.1.4 PROCUREMENT DOCUMENT CONTROL

SCS procedures include detailed information for implementing procurement document control measures, including controls over:

- *A. Vendor qualification.*
- *B. Specifications, drawings, and instructions to bidders - preparation and review.*
- *C. Inquiry preparation and distribution.*
- *D. Bid analysis and recommendation of successful bidder.*
- *E. Requisition preparation and issuance.*
- *F. Inspection, test, and audit.*
- *G. Vendor document review and storage.*

The engineering departments have basic responsibility in the above activities. Other SCS departments involved in the procurement function are the Purchasing Department, and the Quality Assurance Department on nuclear-related items. Also involved are the Steam Projects Planning Department, operating companies in the Southern electric system, consulting firms under contract for specific projects, and certain government regulatory agencies.

The technical and design aspects of specifications, drawings, inquiries, purchase orders, and other procurement documents are developed by the appropriate engineering department. All quality assurance programs, inspection programs, and vendor documentation requirements are also included in procurement documents prepared by the responsible engineering department.

Vendor drawings, specifications, design information, quality program information, and other input supplied by the vendor are reviewed to assure that applicable requirements of 10 CFR 50, Appendix B; specification requirements; and other requirements are being met and are included in the procurement documents as appropriate. The engineering department responsible for the preparation of a particular procurement document consults with all appropriate departments, consultants, contractors, operating companies, and other groups to assure a complete package is prepared. Control of these documents is maintained through procedures which provide for review and approval by competent personnel other than the originator. Preaward, preproduction, and other meetings are scheduled as required with the vendor to assure that specifications are understood and met. Vendor performance is reviewed through monitoring *shop inspection reports, audit reports, and participation in audits, meetings, etc., as appropriate in order to assure compliance with specifications.*

The SCS Quality Assurance Department performs periodic design audits of SCS, Bechtel, and other organizations with responsibilities for preparing design and procurement documents.

Internal review of documents among SCS departments is performed as appropriate. Each procurement document package is reviewed prior to release by each engineering department concerned, Bechtel, and Southern Nuclear Operating Company (SNC).

Procurement responsibility for certain items related to plant construction has been delegated to SNC. In these cases, the responsibility for procurement document control rests with them.

17A.1.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality are defined by instructions, procedures, drawings, and other documents found in:

- *A. Safety analysis reports.*
- *B. Quality Assurance Department Policy and Procedures Manual.*
- *C. SCS Engineering Policy and Procedures Manual.*
- *D. SCS department manuals.*
- *E. SCS Engineering Standards Manual.*
- *F. Project procedures manual.*

The preparation, review, and approval of the above manuals are the responsibility of the department having primary input. Distribution is made on the basis of need.

The SCS quality assurance program provides for the accomplishment of activities affecting quality in accordance with the above referenced instructions and procedures. Periodic audits of SCS, Bechtel, Westinghouse, and vendors by the Quality Assurance Department assure that the instructions and procedures are being followed.

17A.1.6 DOCUMENT CONTROL

Procedures to control the transmittal, review, comment, identification, changes, approval, storage, and current status of engineering documents are followed by SCS personnel. Technical *documents relating to design, including calculations; SCS, Bechtel, and vendor drawings; specifications; studies; and others are included in the document control procedures. The procedures provide for the review and approval of documents by qualified personnel other than those originating the work.*

Control measures are taken to assure that drawings and design documents transmitted to and received at the construction site are properly identified and are current.

Periodic audits by the SCS Quality Assurance Department assist in maintaining the integrity of the system.

17A.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

QA and engineering procedures provide means for assuring that a vendor is properly qualified and that he maintains control of his operation throughout the procurement and manufacturing process. The following phases are included:

A. *The identification of potentially-acceptable vendors.*

B. *The preliminary evaluation prior to vendor qualification.*

C. *The qualification procedures.*

D. *Inquiry document issuance and control.*

E. *Bid analysis and recommendation of successful bidder selection.*

F. *Requisition preparation resulting in issuance of purchase order.*

G. *Vendor inspections and audits.*

H. *Vendor reports and QA documents.*

I. *Final product inspection.*

The engineering department that prepared the requisition and bid analysis has primary responsibility in the above activities. Other SCS departments involved in the procurement function are the Purchasing Department, the Steam Projects Planning Department on major steam components, and the Quality Assurance Department.

In order to qualify as a bidder, a vendor must answer questions concerning his qualifications to produce a product(s) that will meet all specifications and other requirements imposed by the customer. Additionally, the prospective vendor must demonstrate that he has a financially sound *organization that has proper procedures and controls for manufacturing, quality assurance, testing, inspection, documentation, and scheduled delivery of all products.*

Control over vendor evaluation and approval is maintained by a system that provides for review and check of vendor evaluation reports, audits, etc., by cognizant groups other than those preparing the reports.

The SCS Quality Assurance Department administers a vendor inspection program through which vendor job performance is monitored. This program includes:

A. *Determining and defining inspection needs by the responsible engineering design department.*

B. *Inspection request, assignment, and scheduling.*

C. *Monitoring and auditing inspection reports.*

D. *Action by the responsible engineering department to solve problems, deviations, etc.*

E. *Conducting periodic vendor audits.*

F. *Reviewing and taking appropriate action on inspector's final shop inspection and release-for-shipment documentation.*

G. *Reviewing and acting as necessary on problems located during inspection and final acceptance by authorized jobsite inspectors.*

H. *Reviewing vendor final documentation at the vendor's shop.*

Control over vendor performance and quality of the products is assured by audits of the vendor inspection program, vendor shops, and vendor documentation.

Control over design services is maintained by design audits performed by the SCS Quality Assurance Department on SCS design departments and design contractors. Participation in the audits by Bechtel, Southern Nuclear Operating Company, and others is included as appropriate.

In cases where procurement responsibility lies with the operating company or an outside contractor, the control of purchased material, equipment, and services also rests with the operating company or with the contractor.

17A.1.8 DENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

SCS specifications contain procedures for the making, control, and traceability of applicable vendor products for which SCS has procurement responsibility.

These procedures include parts, components, subassemblies, equipment, partially-fabricated items, and final products. The type of identification depends upon the nature of the product and the manufacturing process, and may include strip marking, imprinted tape, color coding, tags, heat, batch, and log number.

Current status of the item(s) as it flows through the manufacturing process must be maintained by appropriate changes in marking. Deficient items must be distinctly marked and removed from regular product flow. The deficiency must be corrected and documented before the special marking is removed. Traceability, as required by codes, standards, or specifications, is included that permits the vendor to identify components, raw materials, subassemblies, and other items that went into the manufacture of each finished product unit that is shipped.

Control over these procedures is obtained through routine shop inspections, documentation reviews, and special program audits by the SCS Quality Assurance Department and other appropriate groups.

17A.1.9 CONTROL OF SPECIAL PROCESSES

SCS specifications include procedures for the control of special processes where required by codes, standards, and SCS requirements. These processes include welding, heat treating, and nondestructive testing.

The first-line control and implementation of special processes are the responsibility of vendors and subcontractors who must provide an adequate quality control program applicable to materials and personnel. Procedures, methods, and instructions must adequately describe the work to be performed for the qualification of equipment and personnel. The vendor is required to submit his procedures to SCS for review and verification of their acceptability. Specialists with SCS (or consultants) perform this review as required.

The vendor shop surveillance program includes checking on the vendor's performance in carrying out special processes during the manufacturing phase. Written surveillance reports document the vendor's activities and call out any deviations from acceptable operation. Surveillance reports are reviewed by the responsible SCS engineering department and the Quality Assurance Department. Periodic audits are conducted to evaluate the vendor and the surveillance programs. Appropriate action on deviations uncovered by surveillance or audits is taken by the responsible engineering department.

Vendor documentation packages are sent to the Farley site for retention after acceptance and release for shipment by the vendor shop inspector. These packages contain results of nondestructive testing and other special process testing as required by SCS specifications.

17A.1.10 INSPECTION

Inspection by vendors providing materials or equipment procured by SCS is assured through the inclusion of inspection requirements in the procurement specification which are appropriate and applicable to the item. The implementation of inspection requirements by the vendor is assured by shop survey, audit, and/or vendor shop surveillance assignment.

SCS specifications require a vendor shop inspection program on safety-related and certain other items in accordance with SCS and regulatory requirements. Vendor inspection activities are audited by Bechtel, SCS, and SNC personnel as appropriate. Inspection assignments are coordinated and administered by the SCS QA Department. Written inspection reports are prepared by the assigned inspector and contain inspection results, job progress, deviations, problems, and other pertinent information. These reports are reviewed and any problems are resolved. The responsible SCS engineering department uses the report as one means of assuring that the vendor is complying with specifications.

Periodic vendor audits and surveillance visits are conducted by the SCS Quality Assurance Department to monitor the inspection program and to assure compliance with specifications. In all cases, the group performing the inspection and the group auditing the inspection program are independent of the group performing the activity.

In certain cases where procurement responsibility is with the operating company or an outside contractor, vendor inspection program control also rests with the operating company or the contractor.

17A.1.11 TEST CONTROL

SCS specifications contain reference to required testing as described in applicable codes; they also contain written details of other test procedures required for use by the vendor. The vendor has the responsibility of submitting a detailed testing program as a part of the overall inspection program for review and approval by SCS engineering departments.

Compliance with the approved testing program is assured by routine surveillance of the vendor's activities during production and by periodic audits of the inspection program by the SCS Quality Assurance Department. Documentation is required of test data and witness points according to specifications. Inspection reports and test documentation are reviewed by the responsible engineering department to assure compliance with specifications. Any deviations are corrected before final approval and release of the item.

17A.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

SCS, vendors, and subcontractors are required to maintain adequate control, calibration, and storage of all measuring and test equipment so that material testing can be performed in accordance with specifications. Control of measuring and test equipment is assured by checking during surveillance and audits carried out by SCS and contracted inspection agencies. Documentation of findings are included in the surveillance reports and audit reports.

17A.1.13 HANDLING, STORAGE, AND SHIPPING

SCS specifications include requirements and procedures for the handling, storage, and shipping of vendor items. Compliance with the specifications is assured by implementation of the vendor shop surveillance program. Review and audit of the program by the SCS Quality Assurance Department provides assurance that materials are being shipped and handled according to specifications.

In certain cases where procurement responsibility is with the operating company or an outside contractor, control over handling, storage, and shipping also lies with the operating company or the contractor.

17A.1.14 INSPECTION, TEST, AND OPERATING STATUS

SCS specifications require that the vendor provide a system that identifies the status of items during manufacturing and item acceptance, with provisions for signoff inspections, tests, and traceability. A system of inspection using tags or other suitable marking to identify inprocess or completed status is required. Material and equipment shipped to the construction site must be accompanied by a certificate of conformance and supported by records of the vendor's inspections and the required test and operational documents. Records of vendor's actions and dispositions of nonconforming materials must be available for review.

SCS procedures provide for keeping records on the status of purchased items. The vendor shop inspection program provides regular documented review of vendor activities in this area and periodic audits by the SCS Quality Assurance Department provide review and control of the vendor's inspection, test, and operating status. Vendors providing products for which the *operating company or an outside contractor has procurement responsibility are monitored, inspected, and audited by the operating company or the contractor as required.*

17A.1.15 NONCONFORMING MATERIAL, PARTS, AND COMPONENTS

SCS specifications require that vendors and subcontractors provide and use a system that will detect nonconformances, identify and segregate nonconforming material, conduct material reviews, assume the proper disposition of material, and provide adequate records. Deficient items corrected and returned to regular production must be accompanied by documentation that identifies the deficiency, corrective measures taken, and results of subsequent testing to establish adequate quality. Where required, final documentation will include deficiencies uncovered during the manufacturing process and the corrective action taken.

SCS procedures provide for the control of nonconforming materials. The vendor shop surveillance program provides for regular reviews and audits of vendor activities and for reports of any nonconforming items. The responsible engineering department reviews the surveillance reports, takes the necessary action to correct any problems, and notifies concerned groups. Periodic audits by the SCS Quality Assurance Department provide a check on the surveillance program and assure quality requirements are met.

In cases where procurement responsibility is with the operating company or an outside contractor, control responsibility over nonconforming material, parts, and components also lies with the operating company or the contractor.

17A.1.16 CORRECTIVE ACTION

SCS procedures assure that prompt, corrective action is taken when a discrepancy or deviation is discovered during the manufacture and procurement of materials. The responsible engineering department reviews shop surveillance reports and audit reports and takes whatever action is necessary to see that the vendor complies with SCS requirements. Access to corporate authority is provided as necessary to obtain prompt responses.

Reviews are made of drawings, calculations, specifications, inquiries, procedures, vendor drawings, and other documents relating to design and procurement. At any point during the project life, the responsible engineering department takes whatever action is required to correct a deficiency or deviation from the specification or procedure requirements.

Deviations and problems detected upon arrival at the site are also reviewed and resolved by the responsible engineering department. The SCS Quality Assurance Department reviews deviation reports and is developing a vendor quality file for future reference and vendor performance evaluation.

Periodic design and vendor audits are conducted by the SCS Quality Assurance Department with participation by appropriate SCS, Bechtel, and Southern Nuclear Operating Company personnel.

These audits provide a check on corrective measures taken by the vendor and assure that adequate testing and documentation is prepared by the vendor and verified by the authorized inspector prior to final release of the item for use in the plant.

The outside contractor or operating company has responsibility for assuring that proper corrective action is taken concerning items for which the contractor or operating company has procurement responsibility.

17A.1.17 QUALITY ASSURANCE RECORDS

SCS specifications require vendors to retain production quality records in a safe and readily-accessible system. This documentation includes traceability records that permit identifying components used to manufacture the finished product. Prior to disposal of any records, the vendor will notify Southern Nuclear Operating Company of his intention. Southern Nuclear Operating Company will retain records within its organization as appropriate. Additionally, the specifications require that the vendor provide certain quality documentation to the Alabama Power Company site at the time of final shipment. The material is reviewed, evaluated, and stored at the site in accordance with the requirements of ANSI N45.2.9.

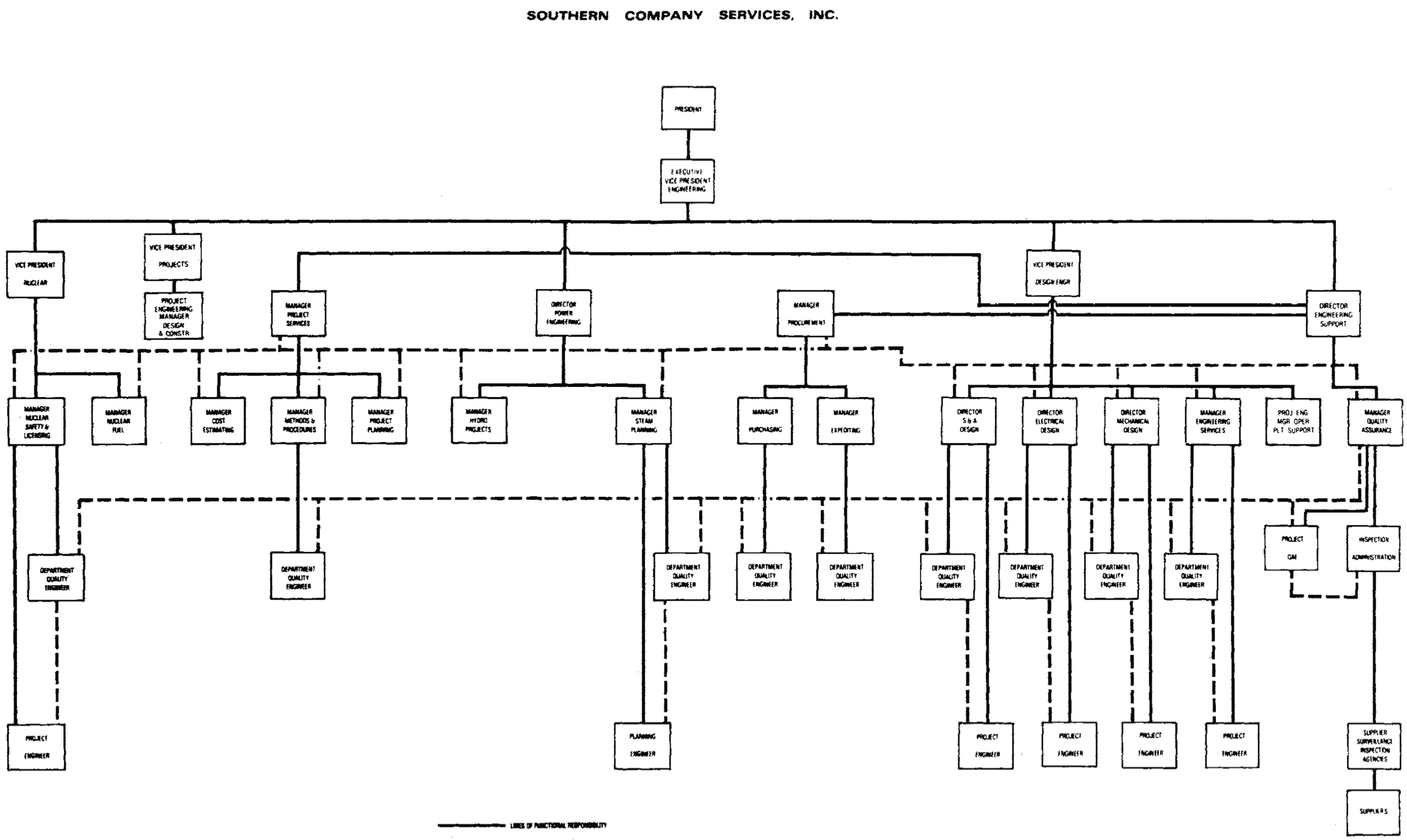
Periodic audits conducted by the SCS Quality Assurance Department assure that the interim records retention system of SCS is functioning satisfactorily. Permanent record storage is the responsibility of the operating company.

17A.1.18 AUDITS

The SCS Quality Assurance Department is responsible for conducting design audits of SCS, Bechtel, and Westinghouse. These audits are conducted on the average of once per year and can be called at anytime. An evaluation to determine the need for an audit is conducted according to procedures. Participation in the audits may include personnel from SCS engineering, the SCS QA Department, and the Southern Nuclear Operating Company SAER Group.

The audit results are written in formal reports and distributed to appropriate personnel at Southern Nuclear Operating Company, Bechtel, Westinghouse, and SCS for review, comment, and action as required. Procedures include provisions for prompt and efficient action to be taken by the concerned engineering department to resolve any problems and deficiencies uncovered by the audit. Followup audits and inspections are made as required to verify that all quality problems have been resolved in a satisfactory manner.

Audits of vendors for which SCS or Bechtel has procurement responsibility are so conducted to assure performance according to all specifications. The SCS Quality Assurance Department schedules and participates in the audits along with appropriate SCS, Bechtel, and Southern Nuclear Operating Company personnel. These audits are documented in formal reports and are reviewed by responsible engineering and management personnel. The responsible engineering department assures that appropriate action is taken to correct any discrepancy or deviation.]



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SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

*[SOUTHERN COMPANY SERVICES, INC.
QUALITY ORGANIZATION FOR
JOSEPH M. FARLEY NUCLEAR PLANT*

FIGURE 17A-1]

APPENDIX 17B

BECHTEL POWER CORPORATION QUALITY ASSURANCE PROGRAM

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BECHTEL POWER CORPORATION QUALITY ASSURANCE PROGRAM

Appendix 17B contains historical information implemented during design and construction of FNP. The current quality assurance policies are delineated in the Bechtel Nuclear Quality Assurance Manual.

Control of quality is the responsibility of the organization which performs the work operation. Quality verification is performed by individuals other than those directly responsible for the work operation; however, they may be members of the same organization.

Assurance of quality is a management function which includes coordination of the quality assurance (QA) program plus monitoring and auditing of the organizations performing the work.

17B.1.1 ORGANIZATION

The vice president and division manager - Bechtel Power Corporation, Gaithersburg Power Division is responsible for the total program and will promulgate the division policy and requirements for quality assurance. Formulation of quality assurance policy and technical direction of the quality assurance program is assigned to the quality assurance manager; the QA manager reports to the vice president and division manager.

The authority and duties of personnel and organizations involved in the quality assurance program are described in subsections 17B.1.1.1 through 17B.1.1.10. Figure 17B-1 is an organization chart showing Bechtel quality assurance program relationship.

17B.1.1.1 Division Quality Assurance Manager/QA Staff

Administrative supervision for quality assurance personnel, technical coordination, and project audits is the responsibility of the quality assurance manager. He is assisted in these functions by a quality assurance staff. The staff monitors and audits engineering activities to assure conformance with the overall quality assurance program.

17B.1.1.2 Division Manager of Engineering

The division manager of engineering establishes division engineering policy and provides overall direction of Engineering Department activities. He monitors project activity and *progress through an engineering manager and through periodic engineering management reviews.*

17B.1.1.3 Supervisor of Quality Engineering

The supervisor of quality engineering is responsible for defining Engineering Department quality program procedures for the division and for providing technical direction for the project quality engineer. He reports to the division manager of engineering.

17B.1.1.4 Chief Engineers

The Bechtel organization provides a chief engineer for each discipline (civil, mechanical, electrical, control systems, nuclear, plant design, and architecture) to assign and provide technical support and coordination of group supervisors, engineers, and designers on the project.

The chief engineers provide independent, documented review of items on the design control checklists. In so doing, they coordinate and assure necessary technical review by specialists and consultants. Chief engineers may delegate review to qualified specialists on their staffs.

17B.1.1.5 Project Engineer

The project engineer is responsible for all matters relating to the performance of the project and is the primary point of contact for the owner. He establishes specific project requirements and conducts regular reviews of the project to ensure that it is proceeding as planned. When problems arise in the operation of the project, he secures necessary corrective action from the cognizant Bechtel groups. He directs the operation of the project engineering team, which has primary responsibility for the quality and technical adequacy of engineering. The team, under the supervision of group supervisors, prepares drawings, specifications, bid evaluations, procedures, and instructions in accordance with quality requirements. They prepare and implement the Q-List and design control checklist. The project quality engineer and individual discipline quality engineers provide verification that the quality control requirements are met and defined in engineering documents. The team also reviews QA documentation submitted by the vendor and shop inspection reports prepared by procurement supplier quality.

17B.1.1.6 Project Quality Engineer

The project quality engineer assists the project engineer in the planning and development of the project quality engineering program and performs routine surveillance of Engineering Design

Group activities to assure compliance with quality assurance requirements and quality control procedures. The project quality engineer is assigned by and receives technical and administrative supervision from the supervisor of quality engineering.

17B.1.1.7 Discipline Quality Engineer

The discipline quality engineer has the responsibility for the review of engineering design documents within his discipline for compliance with project procedures, instructions, and quality program requirements. He maintains administrative control of all requests for engineering changes. He receives technical direction from the project quality engineer and is assigned by the chief engineer through his respective group supervisor.

17B.1.1.8 Materials and Quality Services

The quality assurance aspects of special processes are coordinated by the Materials and Quality Services Group of the Scientific Development Division. Their function includes preparation of standards, procedures, and forms for materials, fabrication, coatings, and nondestructive examination, plus providing technical consultation and guidance for engineering and procurement personnel. In particular, they provide technical direction and service in response to requirements of ASME Section III, Boiler and Pressure Vessel Code for Nuclear Power Plant Components.

17B.1.1.9 Procurement Supplier Quality Manager

The procurement supplier quality manager assigns procurement supplier quality representatives and supervises their activities to assure that purchased material, equipment, and required documentation conforms to the quality requirements of the specifications, drawings, and codes.

17B.1.1.10 Procurement Supplier Quality Representatives

Procurement supplier quality representatives (PSQRs) are responsible for shop qualification audits, inprocess surveillance or inspection of work in vendor shops, checking of vendor documentation and inspection, and release of equipment for shipment. Activities are performed in accordance with the Procurement Supplier Quality Manual as supplemented by the drawings, specifications, and additional instructions provided by Project Engineering.

17B.1.2 QUALITY ASSURANCE PROGRAM

Appendix M of the Bechtel Joseph M. Farley Project Procedures Manual, reviewed and approved by the division QA manager, defines the quality assurance program for the Bechtel scope of work on the Farley Project. The requirements for implementing and maintaining the program are contained in that document and are supplemented by the Procurement Supplier Quality Manual. The program provides for indoctrination and training, as appropriate, of personnel affecting quality. The status and adequacy of the program is regularly reviewed by management.

The purpose of the Bechtel project quality assurance program is to assure that the design, materials, and equipment conform to high standards of quality consistent with the requirements of the owner, regulatory agency criteria, and Bechtel standards. Documentation is provided to confirm that the requirements are met.

The policies and procedures followed by Bechtel in implementing and maintaining an effective, documented quality assurance program during the design and procurement stages of the owner's plant are described in this subsection.[a]

The Bechtel project quality assurance program reflects the requirements of 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, and applies to those safety-related structures, systems, and components for which Bechtel has the design and procurement responsibility.

The scope of the Bechtel project quality assurance program is defined by the Q-List prepared for the project. The Q-List is the master control document for identifying safety-related structures, systems, and components of the nuclear power plant. The Q-List is a working document and is therefore expanded during the design effort to maintain it current.

The following principles are applied in accomplishing the Bechtel project quality assurance program:

A. *The requirements of 10 CFR 50, Appendix B are imposed in all phases of the program to the extent they are applicable.*

B. *The project engineering team has responsibility for quality in the design phase.*

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a. *The term procurement as used to describe Bechtel functions in chapter 17 refers only to the preparation of specifications, soliciting and analyzing bids, and recommending suppliers for those items falling within the scope of Bechtel responsibility. Actual placement of the order is the responsibility of the owner. Procurement supplier quality services are provided when contracted for by the owner.*

C. Specifications assign vendors the responsibility for quality of materials, equipment, and services furnished by them and require them to provide a quality assurance program and organization consistent with the scope of their contract.

D. One or more levels of inspection or checks are provided within the organization having quality responsibilities.

The Bechtel project quality assurance program provides for at least one level of monitoring or auditing by individuals not under the direct control of the group having primary responsibility for quality; e.g., Quality Assurance monitors Engineering, Bechtel PSQRs survey vendors, etc. Quality assurance audits of Engineering and Procurement are performed under the direction of the division quality assurance manager.

17B.1.3 DESIGN CONTROL

Several levels of design review and approval are applied to the design of Bechtel work. These standard procedures include:

A. Check and review by design and engineering personnel within the project engineering team having technical qualifications comparable to those of the engineer or designer who originated the work.

B. Review and approval by the originating engineer's group supervisor.

C. Review and approval by the appropriate chief engineer of design drawings, specifications, and documents identified on design control checklists.

D. Review and approval by the project engineer.

E. Review and approval by the owner of selected design drawings, specifications, and procedures.

Design control checklists are prepared which identify drawings, specifications, and other data for review by chief engineers or technical specialists. When periodic design reviews are deemed necessary, chief engineers and the project engineer agree on appropriate schedules and procedures. When an item identified in the design control checklist has been completed, the cognizant chief engineer reviews and signs it, signifying that the necessary reviews have been performed. Specifications, design and interface information, and systems criteria developed by the supplier of the nuclear steam supply system (NSSS) are submitted to Bechtel for review. Interfaces with vendors are coordinated by Bechtel Project Engineering.

The project engineering team employs several documents to establish requirements for the project. These documents include or incorporate applicable NRC regulatory requirements and design bases, owner-furnished data defining plant requirements, basic engineering data, NSSS supplier-furnished criteria and data, project criteria, standard specifications, and data sheets. Testing of prototype units under the most adverse conditions is required when considered necessary to prove the adequacy of a design.

Design changes are subject to design control measures commensurate with those applied to the original design. Design changes are reviewed and approved by the person or organization that performed the original review and approval. If review and approval of design changes by the original person or organization are not practical, another equally qualified responsible person or organization is formally designated to perform such activities. Persons or organizations so designated are judged to have competence in the specific design area of interest and are given access to pertinent background information upon which to base their review and approval.

17B.1.4 PROCUREMENT DOCUMENT CONTROL

Technical aspects of procurement documents; i.e., specifications, drawings, etc., are prepared by the project engineering team. Owner-supplied vendor quality assurance program requirements are incorporated in the procurement documents. Provisions are made for periodic and shipment inspections in vendor shops. When contracted for by the owner, Bechtel PSQRs visit vendor shops to perform inspection functions employing specifications and quality assurance requirements established by the project engineering team.

Technical changes in procurement documents are subject to the same degree of design control as was exercised in the preparation of the original document.

A vendor print control register is maintained by Project Engineering and is regularly revised to show current status. Review of vendor documents is performed as required by Project Engineering, Quality Engineering, Materials and Quality Services, and the chief engineer's staff specialists. PSQRs are kept advised of the current status of approved vendor documents and drawings.

17B.1.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

The documented instructions and procedures used to implement the Bechtel quality assurance program and provide assurance that the activities affecting quality during the engineering and procurement phases of the project are contained in the following manuals and documents:

A. *Joseph M. Farley Project Procedures Manual contains the detailed procedures for the project quality assurance program and general project operation. Appendix M* *of the manual defines the specific requirements of the project quality assurance program.*

B. *Procurement Supplier Quality Manual contains PSQR instructions, guidelines, and procedures.*

Approval and distribution of these instructions and procedures are controlled by the responsible department manager. Appendix M of the Bechtel Farley Project Procedures Manual is approved by the division QA manager. Other groups affected by these instructions and procedures review the applicable documents prior to their approval.

The Bechtel quality assurance program provides that activities affecting quality will be accomplished in accordance with documented instructions and procedures, and that appropriate means of verifying quality are satisfactorily accomplished and included.

17B.1.6 DOCUMENT CONTROL

The review and approval of Bechtel design documents are covered in Design Control, subsection 17.1.3. Approved drawings, specifications, and procedures are promptly distributed to organizations and individuals performing the work and to those responsible for inspection. Control of distribution and maintenance of current status and files is the responsibility of the recipient organization. Changes made to approved documents by the project engineering team or proposed by the field are reviewed and approved by the project engineering team in accordance with procedures for review of the initial issue. Proposed changes to the Q-list are reviewed by cognizant chief engineers and/or technical specialists.

17B.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

The Bechtel project quality assurance program provides for preparation of procurement specifications which require an appropriate vendor quality assurance program and organization, procurement inspection when necessary, vendor preparation and maintenance of appropriate test and inspection records, certificates and other quality assurance documentation, and vendor submittal of quality records considered necessary to verify quality of completed work.

Recommendation of bidders to the owner and evaluation of bids by Bechtel is made by the Procurement Department and Project Engineering based on the potential vendor's previous performance and capability, information concerning the vendor's quality assurance program, results of shop surveys, and audits by Bechtel. The final decision on a bidders' list, the selection of a vendor, and the placement of a purchase order are the responsibilities of the owner.

As requested by the owner, Bechtel PSQRs review and verify vendor quality assurance records, prepare progressive surveillance inspection reports, witness tests, and identify discrepancies. Inspectors perform audits and document the results on audit checklists. Periodic inspection is performed in the vendor's shop prior to and including release for shipment. Release for shipment is not an indication of acceptance because final acceptance is always at the jobsite.

Files of currently qualified vendors are maintained by the Procurement Department.

17B.1.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

As it applies to vendors, appropriate requirements for identification and control of materials, parts, and components are established in specifications and through review of the vendor's quality assurance program and procedures.

17B.1.9 CONTROL OF SPECIAL PROCESSES

Use of qualified process procedures and application thereof, as required by established codes and standards, are prescribed in procurement specifications prepared by Bechtel. For other special processes identified by equipment suppliers or Bechtel Project Engineering, procedures are prepared by the equipment supplier or Bechtel Project Engineering and are approved by Project Engineering or chief engineer staff specialists.

The Bechtel Materials and Quality Services Group furnishes specialized evaluation of procedures covering metallurgy, corrosion control, metal fabrication techniques, welding, coating, and nondestructive testing.

17B.1.10 INSPECTION

When contracted for by the owner, Bechtel performs periodic and preshipment inspections of vendor work as described in subsection 17.1.7.3. This is performed by PSQRs; however, in special cases, engineering personnel may participate. Inspection practices include witnessing of tests or inspection at mandatory hold points where, in the opinion of Bechtel or the owner, work should not proceed without prior examination by the PSQR.

17B.1.11 TEST CONTROL

The Bechtel quality assurance program requires that vendors have a quality assurance program with requirements that the qualification, functional, proof, acceptance, and operational testing *be performed under controlled conditions in accordance with Bechtel-approved test procedures. These procedures are required to meet the requirements and acceptance limits contained in applicable regulatory specifications, codes, and standards. Bechtel PSQRs review vendor test procedures, including changes thereto, for verification of Project Engineering approval prior to and during the manufacturing process. Bechtel PSQRs and/or engineers are required to personally witness vendor shop tests when specified by the purchase order, specifications, or regulatory code. Checksheets are provided for use by the PSQR in his inspection surveillance and documentation functions.*

17B.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Vendor quality assurance programs are required to have procedures for control of measuring and test equipment. These procedures are reviewed during evaluation of their quality assurance program.

17B.1.13 HANDLING, STORAGE, AND SHIPPING

Special handling, storage, shipping, and preservation requirements are identified in procurement specifications for vendor work.

17B.1.14 INSPECTION, TEST, AND OPERATING STATUS

Vendors are required to provide a system that identifies the status of items during manufacture and test. Provisions for signing off inspections and tests are required. The system of inspection to identify inprocess or completed status is subject to review and approval by Project Engineering. Specifications require that material and equipment shipped to the jobsite be accompanied by documentary verification that the inspections, tests, and operations have been accomplished.

17B.1.15 NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Procurement specifications require vendors to maintain nonconformance procedures as part of their quality assurance program. Their procedures must provide for Project Engineering and/or owner review and concurrence of major nonconformance dispositions. Vendor programs for handling nonconforming material are reviewed and approved by Project Engineering.

17B.1.16 CORRECTIVE ACTION

The Bechtel quality assurance program incorporates corrective action procedures for identification, reporting, and correction to prevent recurrence of situations which are deemed adverse to quality.

The documents generated in these procedures are routed to appropriate levels of Bechtel management and affected organizations for their information and action. Documentation relating to nonconformances and corrective action is filed in project files. Procurement specifications prepared by Bechtel require vendors to have quality assurance programs that provide for similar corrective action programs appropriate to the work they perform.

Procedures for reporting deficiencies are required by 10 CFR 50.55(e) and are described in the Bechtel Farley Project Procedures Manual.

17B.1.17 QUALITY ASSURANCE RECORDS

Quality documentation prepared by Bechtel or obtained from vendors which is collected during the design and procurement phases of the project is identified, reviewed, and filed in project files.

These records are available for audit by the owner and regulatory agencies. The project will maintain these records in compliance with Bechtel practices regarding retention, location, duration, and responsibility until they are turned over to the owner at the completion of the project.

17B.1.18 AUDITS

The Bechtel quality assurance program includes three specific audit activities to verify compliance with the program and to determine the effectiveness of the program.

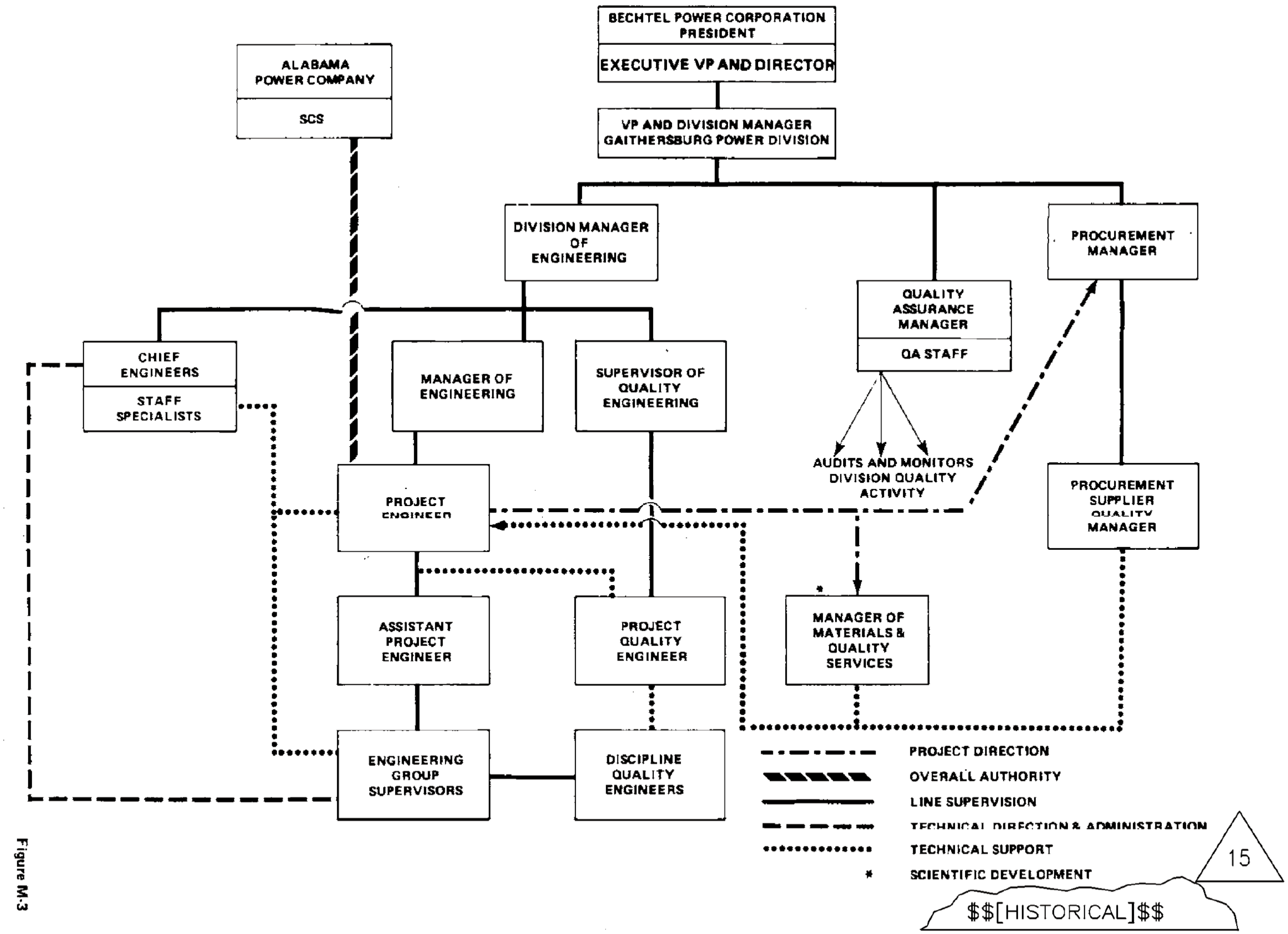
A. *Audits of Project Engineering and Procurement activities and records by or under the direction of the division quality assurance manager.*

B. *Audits of vendor's quality assurance program and records by Bechtel's PSQR.*

C. *Informal monitoring of Project Engineering design activities by the project quality engineer.*

These audits are conducted on a sampling basis during the design and procurement phases of the project.

The results of these audits are documented and distributed to affected management personnel within Bechtel and/or the appropriate vendor organization. Problems found during audits are noted in reports and corrective action is required. Followup audits are performed to assure effectiveness of the corrective action.]



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[BECHTEL POWER CORPORATION
GAITHERSBURG POWER DIVISION
QUALITY ASSURANCE ORGANIZATION
JOSEPH M. FARLEY NUCLEAR PLANT

FIGURE 17B-1]

APPENDIX 17C

WESTINGHOUSE CORPORATION ASSURANCE PROGRAM

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WESTINGHOUSE CORPORATION QUALITY ASSURANCE PROGRAM

Appendix 17C contains historical information implemented during design and construction of FNP. The current quality assurance policies are delineated in the Westinghouse Quality Management Systems Manual.

17C.1 INTRODUCTION

This appendix is the Westinghouse Nuclear Energy Systems (NES) Division's quality plan. Its purpose is to describe the quality assurance (QA) program used by Westinghouse NES to assure that the design, materials, and workmanship on nuclear steam supply system (NSSS) equipment meet applicable safety requirements.

This Westinghouse NES Division's quality plan is a requirement for those NSSS components, systems, and structures having a vital role in the prevention or mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This plan complies with NRC quality assurance criteria, 10 CFR 50, Appendix B and with ANSI N45.2 to the extent that these criteria apply to the design and fabrication of safety-related NSSS equipment.

Several safety guides have been issued on acceptable methods of implementing portions of the NES quality assurance program.

Safety Guide 28, Quality Assurance Program Requirements (Design and Construction), recognizes ANSI N45.2-1971, Quality Assurance Program Requirements for Nuclear Power Plants, as an acceptable basis for complying with 10 CFR 50, Appendix B requirements.

The Westinghouse quality assurance plan for safety-related NSSS equipment described within complies with the requirements of ANSI N45.2 as those requirements apply to the design and fabrication of safety-related equipment, and therefore to the QA plan. The Westinghouse QA plan satisfies Safety Guide 28.

<u>Safety Guide 30, QA Requirements for Installation, Inspection, and Testing o Instrumentation and Electric Equipment</u>, recognizes ANSI 45.2.4-1972, Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations, as an adequate basis for complying with 10 CFR 50, Appendix B requirements. The guide also recognizes that ANSI 45.2.4 was approved by the IEEE Standards Committee as IEEE-336-1971.

The design criteria for Westinghouse instrumentation and controls, described in chapter 7 of the reference safety analysis report (RESAR), requires that safety-related systems comply with IEEE-336-1971 and therefore satisfies the safety guide.

<u>Safety Guide 33, Quality Assurance Program Requirements (Operation)</u>, describes an acceptable method of complying with the Commission's regulations with regard to overall quality assurance program requirements for the operation of nuclear power plant structures, systems, and components.

The responsibility for operation rests with the applicant; however, Westinghouse, in an interface relationship, may perform activities affecting quality. The quality assurance plan for safety-related NSSS equipment described within establishes Westinghouse/applicant interface controls; the plan therefore satisfies Safety Guide 33 for any contracted services during plant operation. Specifically, the quality assurance plan described herein does not address any of these services.

The quality plan is structured to provide a statement of quality assurance philosophy followed by an overview of the NES quality assurance program. Subsections 17C.1.1 through 17C.1.18 address each of the NRC quality assurance criteria, 10 CFR 50, Appendix B. For each criterion, the measures employed are described in sufficient detail to allow the reader to understand the quality assurance program.

<u>NES Quality Assurance Philosophy</u>

Westinghouse Nuclear Energy Systems (NES) is an organizational group of operating divisions whose purpose is to provide nuclear power plant services and equipment. Figure 17C-1 depicts the organization. The Pressurized-Water Reactor Systems Division (PWRSD) is the lead division with respect to design and procurement. The other water reactor divisions, including the Nuclear Fuel Division (NFD), Pensacola Division (PD), and the Nuclear Service Division (NSD), together with the nuclear equipment divisions (NEDs), comprised of the Electro-Mechanical Division (EMD), Tampa Division (TD), and Specialty Metals Division (SMD), provide nuclear power plant equipment and services.

The NES philosophy of quality assurance is to provide reliable, high-quality products. This philosophy has existed since Westinghouse began furnishing nuclear power plant services and equipment. This philosophy is set forth in a policy statement by the general manager - PWRSD:

> *"The PWR Systems Division policy is to furnish nuclear power equipment and services that will provide an electric utility with a safe, reliable, efficient plant throughout its design life. Our quality assurance program must be designed to achieve this objective, starting from the conceptual design, supported by key research and development programs, and carrying through the specification of detailed engineering, manufacturing, inspection, and test requirements, to the installation and operation of the plant. Our program must be a coordinated, routine, in process effort of all the departments whose functions contribute to the quality and reliability of our equipment and services. It must be supported by adequate documentation to assure objective evidence that the program is effective.*
>
> *The Quality Assurance Department has the independence and authority to assure that the program is effective. On matters of quality and reliability, the department manager has direct access to the project managers, the engineering manager, and the division general manager.*
>
> *The Quality Assurance and Reliability Manual describes the PWRSD quality assurance program. All employees whose work contributes to carrying out the division's quality assurance policy should be familiar with it and follow its procedures."*

<u>NES Quality Assurance Program Summary</u>

The NES quality assurance program is designed and implemented to provide safe and reliable nuclear power plant equipment. The quality organization in each of the NES divisions provides the mechanism through which the quality assurance program is administered and monitored. The operation of the program is documented in written procedures and instructions.

The activities of NES are complex in that there are many disciplines involved. Table 17C-1 and figure 17C-2 describe the flow of information and the effort required to design and fabricate NSSS equipment. As shown, the process begins with the identification of technical requirements and ends with a description of the monitoring accomplished to assure process adequacy. The flow schedule simplifies many complex activities for the purpose of showing the overall design and fabrication process. An example is the flow schedule's treatment of equipment specifications, item 7 of table 17C-1. Equipment specifications are the basic method by which NES specifies technical requirements for NSSS equipment. Before an equipment specification can be prepared, functional engineering information is required from many sources. Preliminary specifications are reviewed by many groups. After issue, the specifications are further reviewed by the applicant and suppliers. For visibility, the generation of equipment specifications is shown as one entry on the flow schedule. Figure 17C-2 depicts a functional *rather than formal organizational structure. Like the flow schedule, the functional chart shows the communication network in brief form. On the chart, both horizontal and vertical lines show communication flow paths.*

The quality assurance program provides for the control of design information. Contractual requirements from the applicant and the contents of safety analysis reports (SARs) are inputs to the design process and are reviewed at several points as the design progresses. Analyses of seismic calculations are, for example, accomplished in accordance with specified standards. Drawings and equipment specifications are independently reviewed prior to issue by knowledgeable groups within Westinghouse. Essential drawings and equipment specifications are reviewed by the applicant and his architect/engineer (A/E). Suppliers' detail designs and procedures are reviewed by cognizant NES personnel to assure compliance with equipment specifications and drawing requirements. Design changes which occur during design, fabrication, or installation are controlled in a manner similar to the initial design.

In addition to the above, independent design verification activities, formal in-depth design reviews, and environmental performance testing are performed on a selected basis to confirm that equipment will perform satisfactorily. All design control activities are documented. Interfaces between participating design organizations are defined. These interfaces include the supplier NES interface, the interfaces among NES divisions, and the interfaces among groups within the divisions. The philosophy is that an experienced design engineer is the focal point through which all other participating groups work. This concept provides for single-point responsibility and accountability. At the same time, each participating group has access to higher management for arbitration of unresolved issues.

The NES quality assurance program provides for the control of purchased material, systems, and services. Prospective suppliers are evaluated for quality system capability. Purchase orders are reviewed for technical and quality-related requirements. As applicable, source surveillance and receipt inspection are performed. Supplier documentation essential to demonstrating product quality is reviewed and retained by NES. Audits and feedback of discrepancy data are used by NES quality engineers to measure supplier performance.

All the NES divisions have systems which control the review, approval, distribution, and revision of instructions, procedures, specifications, and drawings. Because of the varying needs of the divisions, these systems differ in detail. The objective of each of the document control systems is to provide a means for controlling the use of documents so that NSSS equipment is designed and fabricated in accordance with the stated requirements.

NES manufacturing divisions operate under controlled systems. These systems require the performance of important operations in accordance with instructions and procedures. These instructions and procedures are substantiated prior to use by actual demonstration. The means for accomplishing the operation and the criteria for accepting the operation are included in instructions. Examples of operations covered include welding, heat treating, nondestructive testing, performance testing, welder qualifications, receipt inspection, final inspection, gage *control, material handling, and material identification. The manufacturing control systems provide for control of the overall manufacturing process. The control system indicates inspection and test status.*

The NES quality assurance program controls nonconforming material by procedures which provide for documented results. All nonconforming material is segregated or, if segregation is physically impractical, clearly identified so that its inadvertent use is prevented. Data from nonconformances are collected and summarized for use in design changes to prevent recurrence of nonconformance.

The NES quality assurance program maintains sufficient records to clearly establish the quality of the product. A microfilmed copy of the fabrication and inspection records is provided to the applicant for permanent retention prior to plant acceptance. To document equipment acceptability prior to site installation, a copy of the purchase order, the applicable design specification, and the quality release are provided to the applicant.

A comprehensive audit program is part of the NES quality assurance program. This audit program provides NES management with information pertaining to the effectiveness of the quality program. Planned and scheduled audits are conducted with results reported to appropriate management levels and corrective action taken as necessary.

The applicant's need to assure himself of the adequacy of the NES quality assurance program is recognized by NES management. The NES QA program provides an extensive amount of design information for applicant review and use. On a typical nuclear power plant, 1000 equipment specifications and drawings, as well as various design installation, testing, and quality assurance manuals, are transmitted to the applicant for review. Audits by the applicant are performed on NES activities covering design, manufacturing, and documentation. The applicant also participates with NES quality assurance personnel in selectively performing supplier surveillance. Quality-related procedures and instructions, as well as the results of tests and inspections, are available to the applicant for his review to verify the operational adequacy of the quality assurance program.

17C.1.1 ORGANIZATION

NES is comprised of a number of operating divisions under an executive vice president, as shown in figure 17C-1. The authority and responsibility of each activity shown on this chart and subsequent charts is set forth in an approved, written statement of group responsibility. In addition, written position descriptions are prepared for each management and professional position. These descriptions specify the educational and experiential qualifications of the position.

The quality assurance aspects of NES activities are overseen and coordinated by the NES Quality Assurance Committee. This Committee, appointed by the NES executive vice president, *is made up of the quality assurance and reliability managers of the NES divisions. The Committee monitors activities throughout NES to provide assurance to NES management that requirements relating to quality assurance are effectively met. The Committee also considers matters of policy to improve and unify the divisions' quality assurance systems.*

Overall contract responsibility for supplying the NSSS is assigned to a project manager within the PWRSD. He provides the focal point for communications among the NES divisions, the applicant, and the architect/engineer.

The following is a summary of design and manufacturing responsibilities of the NES divisions involved in furnishing NSSS equipment and service.

17C.1.1.1 PWR Systems Division

The PWR Systems Division, as shown on figure 17C-3, is the lead NES division with regard to the project management, design, and procurement of NSSS equipment.

The Project Department of PWR Systems Division, through a designated project manager, has the primary responsibility within NES for supplying the NSSS equipment and services to the applicant.

The Purchases and Traffic Department provides the PWRSD procurement interface with suppliers and with other NES divisions. This department is also responsible for administering the transportation of equipment from supplier's facilities to the construction site. Material and equipment is protected appropriately against the hazards of mechanical damage and weather during shipping. This includes such provisions as painting with suitable rust inhibitors, taking into account ease of removal of all protective substances during the chemical cleaning and preoperational periods, polyethylene or suitable wrapping, heavy boxes or crates, bolted wooden flange protectors, suitable barriers or blocks, resilient supports, inert gas purge, and other protective provisions.

Additionally, Westinghouse furnishes accelerometer instrumentation with shipments of certain critical equipment. All equipment furnished by Westinghouse is shipped, insofar as possible, completely assembled. Technical assistance for unloading, handling, and storage is furnished to the applicant through the Field and Technical Operations Departments of the Nuclear Service Division.

The Engineering Department of PWR Systems Division has responsibility for the overall design of the NSSS. This responsibility includes:

A. Fluid and electrical systems design by Systems Engineering.

B. Mechanical equipment design and materials support by Plant Apparatus.

C. Control and electrical equipment design by Control and Electrical Systems.

The Nuclear Safety Department is responsible for providing the nuclear steam supply system the safety system performance requirements, safety system criteria, safety analysis methods, and safety evaluations to provide the required analytical and statistical verification of postulated accidents. Further, the department is charged with providing the licensing activity to support the applicant in obtaining the construction permit and operating license for the nuclear steam supply system.

The quality program management performing QA-related activities (checking, auditing, inspecting, or verifying that an activity has been correctly performed) is structured as shown in figure 17C-4. Quality management exercises both technical direction and administrative control.

Quality management does not have prime responsibility for schedule or cost, but does have the authority to stop work pending resolution of quality matters. Quality management also has the freedom to:

A. Identify quality problems.

B. Initiate, recommend, or provide solutions through designated channels.

C. Verify implementation of solutions.

D. Control further processing, delivery, or installation of a nonconforming item, deficiency, or unsatisfactory condition until proper dispositioning has occurred.

Within PWRSD, responsibility for quality assurance activities is assigned to the Product Assurance Department. This includes having lead responsibility for developing the capabilities and demonstrating compliance with the 18 criteria of Appendix B. The manager of product assurance reports directly to the division general manager and is in parallel with the other major departments within the division as shown in figure 17C-3. Thus, matters pertaining to product and system quality can be related directly from the product assurance manager to the division general manager, independent of other functional activities.

The Product Assurance Department is divided into two groups, Product Assurance Systems and Quality Assurance. The efforts of each group are directed by a separate manager.

The Product Assurance Systems Group has five major functional responsibilities. These are:

A. The investigation and analyses of the PWRSD's procedures for compliance with the criteria of 10 CFR 50, Appendix B, as well as other industry and corporate quality standards.

B. Preparation and maintenance of division level policies and procedures.

C. Administration of centralized files and quality records.

D. Internal auditing for compliance with established procedures.

E. Design reviews. The Product Quality Assurance Group also compiles, audits, stores, and retrieves the various QA records associated with the NSSS equipment.

The Quality Assurance Department consists of five sections: Quality Engineering, Quality Assurance Surveillance Zone 1, Quality Assurance Surveillance Zone 2, Quality Assurance Electrical, and Reliability Engineering.

Quality Engineering provides the necessary QA input into engineering and procurement activities (e.g., drawings, specifications, purchase orders, etc.), develops QA plans for surveillance activities, participates with Engineering and Purchasing in the evaluation of proposed suppliers, and coordinates customer audits at PWRSD.

The two Quality Assurance Surveillance Sections monitor the activities of PWRSD suppliers and verify conformance to procurement quality requirements. This is done using both resident and itinerant QA representatives. To provide the most effective coverage of suppliers, each group is assigned responsibility for performing surveillance in a specified geographic area encompassing both domestic and international supplies. Zone 1 encompasses Pennsylvania and the states east and south of the Mississippi and Ohio Rivers. Zone 2 includes the New England states, New York, and the mid-west and western states. The surveillance representatives perform the in-process monitoring and release suppliers' equipment by issue of the PWRSD quality release document.

The Quality Assurance Electrical Section performs the combined function of the Quality Engineering and Surveillance Sections as it applies to electrical/electronic equipment. Since the nature and volume of electrical equipment is considerably different from the other NSSS equipment, the QA functions have been merged into this one section. However, the methods and procedures used by this section are the same as used for the other NSSS equipment.

The Reliability Engineering Section performs the product assessment functions, including formal design reviews and reliability analyses. In addition, this section conducts the internal audits of PWRSD systems and procedures related to product quality.

17C.1.1.2 <u>Nuclear Fuel Division</u>

The Nuclear Fuel Division is responsible for the detailed design of first cores based upon PWRSD equipment specifications and drawings, for manufacture of fuel assemblies and core components, and for all aspects of the design and manufacture of fuel assemblies for repeat cores. The organization of this division and a description of the Nuclear Fuel Division quality assurance program is contained in reference 1.

17C.1.1.3 <u>Electro-Mechanical Division</u>

The Electro-Mechanical Division designs, manufactures, and tests control rod drive mechanisms (CRDMs), reactor coolant pumps, loop stop gate, and check valves. Figure 17C-5 shows the organization for the Electro-Mechanical Division.

This division is a qualified manufacturer per requirements of ASME Section III, Boiler and Pressure Vessel Code for Nuclear Power Plant Components and Hold Certificates of Authorization for the Use of the "N" and "NPT" Symbol Stamps.

The Design Engineering Section develops the detailed design drawings and specifications, including those for procurement from contractual equipment specifications provided by PWR Systems Division. Formal design reviews precede the finalization and issuance of all new product designs.

The Manufacturing Systems Section is responsible for the control of manufacturing information designed by the Manufacturing Engineering Section.

The QA organization has total responsibility for assuring compliance to all contractual requirements. In addition, each department is responsible for applicable controls as outlined in the Quality Program Manual.

The overall responsibility for the implementation of the quality assurance program is vested in the quality assurance manager who also has the authority to enforce full compliance with all quality requirements relative to safety, reliability, operation, and maintenance.

Quality Assurance Engineering is responsible for planning controls to assure product quality. This includes: review of all contractual/governing specifications, engineering design drawings and specifications, purchasing information, and detailed manufacturing instructions.

This organization is responsible for the design and implementation of the inspection point programs, including non-destructive testing incorporated into manufacturing work instructions; qualification of nondestructive testing personnel; audits; compilation of documentation maintained as objective evidence of inspections performed; and attendant functions of analyses and preventive actions to eliminate problem areas.

Field Assurance Engineering is responsible for surveys and qualification of suppliers, design and control of quality programs for purchased material, in-process inspections, and tests, audits, final inspection, and release of supplied product.

Design/Test Engineering is responsible for test specifications, hydrostatic and performance tests, evaluation of test results, and issuance of the test release.

Quality Assurance Engineering is responsible for the final release and certification of all products prior to shipment.

17C.1.1.4 Tampa Division

The Tampa Division designs and manufactures steam generators and pressurizers. Figure 17C-6 shows the organization for the Tampa Division.

Tampa Engineering performs the detail design from equipment specifications provided by PWR Systems Division.

The manufacturing groups are responsible for the fabrication and testing.

The reliability manager directs the activities of the Reliability Engineering, Quality Assurance, and Metallurgy Departments.

The metallurgical organization is responsible for the quality-related functions of material specification, material source approval, welding process qualification, welder qualification, and resolution of shop metallurgical problems.

Overall responsibility for the implementation of the quality assurance program is vested in the quality assurance manager who has the authority and responsibility to stop any operation to assure compliance with the ASME Code, customer requirements, and Westinghouse requirements.

Responsibility for planning controls to assure product quality resides with Quality Assurance Engineering. Quality assurance planning includes preparation of the inspection point programs to assure compliance with drawings, specifications, and ASME Code requirements; the inspection point programs provide process and product verification and are also utilized as a permanent record of the inspection operations. The details of the inspection point programs *follow the manufacturing sequence of the operational lineup; the format delineates the applicable forms, charts, reports, and documentation required.*

The responsibility for conducting internal audits resides with the manager - reliability engineering. The audit program provides management with a continuing overview of the compliance with the quality program. In addition, Reliability Engineering coordinates the Tampa efforts to resolve field discrepancies and affect corrective action. Reliability engineers participate in the design review function and provide additional engineering support in the area of failure modes and effects analyses.

17C.1.1.5 Pensacola Division

The Pensacola Division is responsible for the design and manufacture of reactor internals and other associated internals equipment. The organization is shown in figure 17C-7.

The Pensacola Quality Assurance Division provides measures to control the design, manufacture, purchase, inspection, test, packaging, shipment, and site installation of reactor internals.

The organization of Quality Assurance, Manufacturing Groups, and Manufacturing Planning permits the Quality Assurance Department direct access to responsible management. This permits the Quality Assurance Department to independently identify quality problems, and to initiate, recommend, and provide appropriate solutions. The Quality Assurance Department is vested with the authority and responsibility to stop production until acceptable solutions have been provided.

17C.1.1.6 Specialty Metals Division

The Specialty Metals Division (SMD) manufactures tubing used by the Tampa Division and the Nuclear Fuel Division (NFD). Figure 17C-8 shows the SMD organization. Manufacture is in accordance with the specifications provided by Tampa and the NFD.

Quality Assurance has direct control of the division gage calibration system, instrument service and calibration, quality engineering, and product verification and certification. Quality Assurance has the responsibility for the training and certification of inspectors in particular fields.

17C.1.1.7 Nuclear Service Division

The responsibility of NES at the plant site involves construction consultation to the applicant. This responsibility is within the Field Operations and Technical Operations Departments of the Nuclear Service Division. Figure 17C-9 shows the NSD organization.

Work on nuclear steam supply equipment, as performed by the construction contractor and subcontractors, is monitored by Westinghouse representatives assigned to the construction site. The necessary procedures and actions are coordinated with the construction contractor. Special processes, such as welding, cleaning, and nondestructive testing, are observed by qualified Westinghouse personnel to assure the work is performed in accordance with written procedures.

During component installation, Westinghouse NSD monitors work on nuclear steam supply and engineered safeguards equipment. Qualified personnel provide technical advice on various disciplines of construction such as welding, mechanical and electrical systems, instrumentation and control equipment, and preoperations and startup testing. Guidance documents are provided to the applicant detailing Westinghouse-recommended programs for site activities such as receiving, inspection, and storage; installed equipment inspection; cleaning and flushing; equipment checkout; initial operation and adjustment; integrity tests; system functional tests; and plant operational tests and measurements.

The construction site manager is responsible for overseeing that the Westinghouse nuclear steam supply equipment is in good condition when received and that it is stored, handled, and installed properly according to applicable specifications, procedures, and manufacturers' instructions.

A written procedure describes the system for identifying, reporting, and obtaining disposition of nonconforming material or equipment discovered at the site. NSD personnel fill out a field deficiency report to provide the cognizant engineering group with the information necessary for making proper and timely disposition of each problem. After the cognizant personnel make a disposition, it is noted on the field deficiency report and returned to the field for action. Files of these reports are maintained to record all field deficiencies and to provide for long-term corrective action.

The Service Operations Department provides optional services to the applicant, such as nuclear training services, renewal parts and components services, post-operational services, etc.

The Quality Assurance Department within the Technical Operations Department of NSD is responsible for conducting independent audits of Westinghouse personnel activities at the construction site. Additional quality assurance functions for NSD are provided as necessary by the quality organization within the PWRSD.

17C.1.1.8 Functional Responsibilities

The functional responsibilities of designing and fabricating NSSS equipment are shown in table 17C-2. The responsibilities are broken down into three categories: design criteria, detail design, and manufacture. For each category, the organization responsible for performing the particular function is identified. The table identifies the scope of the quality assurance program for both safety and nonsafety equipment. The identification of safety-related equipment is covered in other chapters of this report.

17C.1.2 QUALITY ASSURANCE PROGRAM

The NES policy is to provide nuclear power equipment and services that will provide an electric utility with a safe and reliable plant throughout its design life. To meet this policy, each NES division is committed to comply with quality assurance criteria of 10 CFR 50, Appendix B.

This plan is a description of the NES quality assurance program. The program is supported by written policies and procedures governing quality-related functions and activities from initiation of design through fabrication and shipment. Identification of the principal quality assurance documents is contained throughout this plan.

Table 17C-3 gives a typical and representative listing of the written procedures within NES for implementing the NES quality assurance program. Listed are the various manuals, the subjects covered, and a short description of their purpose. The manuals referenced each contain procedures dealing with other topics unrelated to the criteria of Appendix B. Only those procedures felt to be responsive to the requirements of 10 CFR 50, Appendix B are detailed in table 17C-3.

This plan demonstrates that the NES quality assurance program complies with the criteria of 10 CFR 50, Appendix B. In order to facilitate the presentation, measures established for the quality assurance program are described for each criterion.

The NES quality assurance program requires that contractors and suppliers of NSSS equipment have quality systems consistent with the requirements of Appendix B quality assurance criteria. A summary description of the NES quality assurance program is found on pages 17C-3 through 17C-6.

17C.1.3 DESIGN CONTROL

Each of the NES divisions involved in NSSS design provides measures to assure effective design control. Below is a description of the design control procedures which provide methods for controlling activities such as specifying quality standards, selection and review, design changes, design interfaces, and implementation of procedures.

17C.1.3.1 PWR Systems Division

The project manager is responsible for identifying to Engineering, Purchasing, Licensing, and Quality Assurance Groups the technical requirements of a nuclear power plant. This identification process is formal and documented. The distribution of this technical information is the start of the design activity on a nuclear power plant. Changes to distributed information are also issued by the project manager.

Nuclear Safety prepares safety analysis reports. Prior to the submittal of NSSS portions of safety analysis reports to the applicant, licensing engineers obtain engineering, projects, and quality assurance review and concurrence of technical content. The review process is formal and documented.

Based upon the identified technical parameters, Systems Engineering Groups design the nuclear power plant to meet functional, safety, and regulatory requirements. Mechanical and electrical design engineers participate in the functional design process by identifying equipment limitations and resolving functional requirements with equipment capabilities. The output of the Systems Engineering Groups are written functional parameter documents.

Control and electrical system engineers, plant apparatus mechanical design engineers, and nuclear service engineers are responsible for designing or specifying NSSS equipment. Equipment specifications are prepared by the electrical and mechanical design engineers. The term "equipment specification" as used in this plan includes drawings when they are used instead of equipment specifications. Detailed quality control requirements are specified in the equipment specification or its references. Examples of these specifications are nondestructive tests, acceptance standards, functional tests, and recording the measured values of key characteristics. In the few cases where equipment specifications or design drawings are not used, the specific quality control requirements, tests, and acceptance standards are identified in the purchase order. The design of equipment also provides for access to components for *inservice inspection and maintenance as required to assure continued integrity throughout the life of the plant.*

Preliminary equipment specifications are reviewed within Westinghouse by systems engineers, materials and process engineers, licensing engineers, Quality Assurance, projects, and others as required. These independent reviews verify that equipment specifications meet system requirements; conform to established engineering standards; are adequate from a metallurgical and welding point of view; meet code requirements; satisfy safety requirements, including those specified in safety analysis reports; and contain necessary quality control requirements. Written engineering instructions prescribe preparation, review, and approval of equipment specifications.

Documented procedures control design changes. These procedures require appropriate groups to review and approve the changes according to written engineering instructions.

Westinghouse interprets as-built drawings and specifications to meet those documents which specify the functional parameters of an item for procurement, manufacturing, installation, and operational purposes. Whenever changes are necessary to these parameters, as identified by engineering, manufacturing organizations, or the applicant, Westinghouse Engineering reviews these proposed changes to the original design. Upon approval, Engineering initiates the required action to change the drawings and specifications to accurately reflect the design change. When approved for release, copies of the revised documents are provided to the applicant as well as other organizations needing the documents for subsequent work. As discussed in subsection 17C.1.6, this distribution system is controlled.

Aspects of the equipment design that have an effect on that part of the plant design performed by the applicant or architect/engineer are forwarded to them for their review. Applicant or architect/engineer drawings which have an effect on the NES scope of supply are likewise sent to NES engineers for their review.

The implementation of the design control system is audited by Product Assurance.

In addition to the verification of technical requirements discussed above, formal design reviews are conducted by Reliability Engineering on critical systems, subsystems, and components to improve their reliability and to reduce fabrication, installation, and maintenance costs. The design reviews are comprehensive, systematic studies by personnel representing a variety of disciplines not directly associated with the development of the product. Specialists from other Westinghouse divisions and outside consultants are used in the reviews as necessary. Information developed by the reviews is recorded for evaluation and action by the cognizant design engineer. The design review procedure requires the resolution of open items within specified periods. Reliability engineers verify completed action.

The design review program is projected over a substantial period of time because of the comprehensive nature of each review. Both the scheduling of the review and the selection of *specific equipment for review are based upon many considerations, including whether the equipment is of a new design, its importance to public health and safety, its importance to plant availability and performance, and previous experience with the equipment. In this priority scheme, some equipment of proven design may not receive a formal design review.*

Verification calculations and performance testing are accomplished as necessary. A discussion of the means by which seismic requirements are satisfied describes the decision and control process involved. Seismic criteria are provided by the applicant. These criteria are forwarded by the project manager, as previously described, to the Mechanics and Materials Technology Group within Plant Apparatus. A seismic coordinator distributes the seismic criteria to equipment design engineers for inclusion in equipment specifications. These specifications, which are reviewed by Mechanics Technology personnel, require supplier submittal of either calculations or test data demonstrating that the equipment is seismically qualified. The design engineer reviews and checks the supplier submittals. Seismic calculations are forwarded to the Mechanics Technology Group for final review and certification. The final review process includes an independent recalculation when the seismic adequacy is doubted. The various events within this process (e.g., equipment specification review) are performed in accordance with procedures which require documented results.

17C.1.3.2 Electro-Mechanical Division

Upon receipt of an equipment specification from the PWR Systems Division, the lead design engineer is responsible for correct translation of reactor coolant pumps, control rod drive mechanisms, and loop stop valves into specifications, drawings, procedures, and instructions.

Engineering instructions specify that all designs be reviewed by a design review committee chaired by an individual who is not a direct supervisor of the lead design engineer or directly involved with the original design group. Technical reviews by Engineering personnel verify design adequacy for compliance with performance requirements. Subsequently, divisional design reviews, with participants from Engineering, Manufacturing, and Quality Control as a minimum, coordinate all departments to assure compatibility with code and quality requirements. In addition to the formal review process described above, drawings, equipment specifications, and manufacturing routings are reviewed by cognizant groups within the division, including Quality Engineering. Design changes and document revisions are released after a review for adequacy and approval for release by the same groups involved in the initial review. This includes deviations controlled under a nonconforming materials review system.

17C.1.3.3 Tampa Division

The Tampa Division is responsible for the design and manufacture of steam generators and pressurizers. The design effort is based upon an equipment specification from the PWR Systems Division. The Design Group is responsible for heat transfer, material evaluation, *hydraulic analysis, and operation. The Structural Analysis Group is responsible for vibration and shock analysis, experimental stress analysis, general stress analysis, and materials behavior.*

Prior to releasing drawings for manufacture or purchase, Quality Assurance Engineering reviews drawing for conformance to the ASME Code. Included in the review is assurance that dimensions and tolerances are shown and requirements for special tools are established; that material specified is in accordance with ASME Code requirements; that welding specifications are compatible with material; that the correct nondestructive tests are specified; and that any special characteristics are clearly identified on the drawing. The quality assurance engineer signifies review by signing the drawing when the above points have been satisfied. All drawing revisions are reviewed and approved in a manner similar to the original drawings.

Planned and documented design reviews are conducted to assure that the product being designed and manufactured meets all contractual and code requirements. These reviews provide assurance that nuclear effects, mechanical, thermal, hydraulic, safety, and similar type studies are complete; and that research and development programs and test provide adequate substantiation of the design if necessary. Compatibility of materials and design interfaces is assured and maximum use of qualified, standard, or approved parts, materials, components, and processes are used where possible. Adequate accessibility for inservice inspection, maintenance, or repair is designed into the product as well as specifying acceptance criteria for inspections tests. Each design review is documented for permanent filing and includes coverage of significant problems, decisions, and the action taken or proposed.

17C.1.3.4 Pensacola Division

The Pensacola Division is responsible for the design of structural components for internals of the Westinghouse pressurized-water reactor (PWR). Prior to release of drawings and specifications at the Pensacola Division, Quality Engineering and Reliability reviews each drawing and specification to assure compliance with the contract and ASME Code requirements. This review shall include the following considerations as a minimum:

A. *The material defined is code-approved.*

B. *Proper and adequate nondestructive tests are specified to assure compliance with the code and applicable specifications.*

C. *Any special processes specified are adequately defined and compatible with the material.*

D. *Dimensioning is clearly defined to permit manufacture and subsequent piece-part inspection.*

To assure that each drawing and specification receives the above review, the cognizant quality engineer indicates approval for drawing release by signing each drawing. Subsequent revisions to the drawing are also reviewed by the quality engineer to assure continued compliance with the above stipulated considerations.

The Pensacola Division maintains a computer-controlled printout for identification and control of all drawings applicable to each product line. As revisions become applicable, the printout is updated and distributed to appropriate work areas.

17C.1.3.5 Specialty Metals Division

Detail design is not performed by the Specialty Metals Division.

17C.1.3.6 Interface Control

Written instructions define the interfaces among participating design organizations. Within the PWR Systems Division, shop-order logic flow diagrams document the relationships among the many design, procurement, control, and administrative activities required to conduct the business of the line organizations. Additionally, the flow diagrams serve to document and show by road map how PWRSD complies with Appendix B. The level of detail depicted on the flow diagram is intended to optimally portray management controls, provide an easy means of education, and facilitate the auditability of these controls.

PWRSD, as the lead division with NES, establishes the design criteria and parameters for systems, structures, and equipment. This information is transmitted in the form of equipment specifications or drawings to the manufacturer. In some cases, the manufacturer is responsible for providing a detail design or process procedure based upon the PWR criteria and parameters. These are submitted by the vendor to PWRSD where they are reviewed and approved prior to manufacture. Review and approval requirements are clearly stated in purchase orders, or in the case of other NES divisions, in written interface instructions.

One example of the latter is the PWRSD/NFD interface instruction. This instruction clearly defines the division of design responsibilities in terms of which groups originate review and distribute the design documents, deviation reports, and design change reports involved in the PWRSD/NFD interface.

Tables 17C-6 and 17C-7, which are two sheets extracted from a typical shop-order logic flow diagram, depict the above process for review and approval of vendor submittals, noting the specific interfaces, applicable documents for detailed instructions, and appropriate criteria of Appendix B. It is the responsibility of the cognizant manager for each shop order to maintain the currency of his particular logic flow diagram. Product Assurance Systems is responsible *for the distribution and control of these and for any support required in the updating of the diagrams.*

In addition to the interface between PWRSD and manufacturers, there is an interface with the applicant and his design agents.

All PWRSD equipment specifications, flow diagrams, and procurement drawings that are outline or assembly drawings and are used in lieu of equipment specifications are transmitted to the applicant or his design agents for review. Each project manager has a written procedure defining the process for transmittal and resolution of comments.

17C.1.4 PROCUREMENT DOCUMENT CONTROL

In general, the procurement of components, systems, structures, and material within NES falls into three distinct areas:

A. Components procured by the PWRSD from other NES divisions.

B. Components, systems, and structures procured by PWRSD from suppliers and non-NES divisions.

C. Materials procured by Pensacola, Tampa, EMD, and SMD.

Relationships of the various NES divisions is discussed in detail in subsection 17C.1.1.

17C.1.4.1 PWR Systems Division

As described in subsection 17C.1.3, equipment specifications and drawings receive a detailed review prior to issue. Purchase orders reference equipment specifications and drawings as the technical basis of procurement. A quality assurance procedure requires quality engineers to review purchase orders. The review process assures that the purchase order defines the equipment being procured and clearly specifies technical and quality requirements. When discrepancies are noted, a written request for corrections is initiated.

Quality requirements that specifically apply to a component are contained in the equipment specification. Quality system requirements of a general nature are contained in two standard documents.

The first document is entitled, Administrative Specification for the Procurement of Nuclear Steam Supply System Components. This document is applied in all component purchase orders. The administrative specification requires that the supplier not only manufacture equipment that conforms to purchase order requirements, but to assure himself and

Westinghouse by means of appropriate inspections and tests that the equipment conforms to these requirements. The quality control section (QCS) of this specification contains specific requirements in areas such as:

A. Organization.

B. Purchasing control.

C. Receiving inspection.

D. Material control.

E. Control of drawings and procedures.

F. Calibration of measuring and test equipment.

G. Personnel qualifications.

H. Deviations from specifications.

I. Special process and test procedures.

J. Handling and storage procedures.

K. Inspection and manufacturing control.

L. Quality records.

M. Quality release.

N. Quality systems audits.

The second document that specifies quality requirements is QCS-1, Manufacturer's Quality Control Systems Requirements. This document is applied to orders for more critical safety equipment. This document requires the supplier to maintain an adequate quality control system. This specification meets NA4000 of Section III of the ASME Boiler and Pressure Vessel Code in the area of quality control system requirements. QCS-1 requires, among other things, the following:

A. Establishment and maintenance of a system for the control of quality that assures that all supplies and services meet all specification, drawing, and contract requirements.

B. Application of the system to subcontracted items.

C. Written procedures that implement the system.

D. Qualification of personnel.

E. Qualification and control of processes, including welding, heat treating, nondestructive testing, quality audits, and inspection techniques.

F. Operation under a controlled manufacturing system such as process sheets, travelers, etc.

G. Written inspection plans for in-process and final inspection.

H. Submittal of inspection checklists for approval.

I. Recording of results of inspection operations.

J. Written work and inspection instructions for handling, storage, shipping, preservation, and packaging.

As required, inspection hold points are specified in the equipment specification or elsewhere in the purchase order. These are points of witness or inspection by Westinghouse beyond which work may not proceed without approval by the PWRSD.

NSSS equipment ordered by the PWR Systems Division from other NES divisions is specified by equipment specifications or drawings. Quality assurance program requirements are satisfied by requiring NES divisions to perform their work in accordance with 10 CFR 50, Appendix B.

17C.1.4.2 Electro-Mechanical Division

Upon receipt of an order from the PWRSD, written procedures require that the overall quality requirements of the contract are reviewed by cognizant engineering personnel and action is initiated to assure that contractual quality requirements will be referenced in documents for procurement of material, equipment, and services and will be met during procurement, manufacturing, and shipment. Procurement documents delineate the quality assurance program requirements consistent with application of the material, component, or service being provided.

Quality engineers review purchase orders to assure that the supplier is furnished all applicable requirements affecting quality.

17C.1.4.3 <u>Tampa Division</u>

The Quality Assurance Department has developed an inspection code which is used in the determination of the supplier inspection requirement level for all purchased materials. The inspection code, Codes 1 through 4, is used on all purchase orders, with Code 4 being applied to items such as light bulbs, stationery, etc. All purchase orders for materials or parts having Code 1, Code 2, or Code 3 requirements are reviewed by Quality Engineering to determine that proper and essential quality requirements are specified.

17C.1.4.4 <u>Pensacola Division</u>

Prior to placement of a material purchase order, the purchase requisition is approved by the Quality Assurance Department. The purchase requisition review assures that applicable drawings and specifications are listed together with correct revision references, and that required destructive and nondestructive tests are specified.

An addendum form to the purchase order requisition titled, Purchase Order Supplementary Technical Requirement (POSTR), is used to delineate the specifics of the referenced requirements.

17C.1.4.5 <u>Specialty Metals Division</u>

The Manufacturing Department is responsible for initiating purchase requisitions in accordance with Tampa and NFD requirements. Purchase requisitions are approved by the Quality Assurance Department prior to issue. Applicable Quality System requirements are specified, as well as references to technical specifications.

17C.1.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Within NES, written procedures and instructions are in use to implement the quality assurance program and to provide assurance that all activities affecting quality in the context of 10 CFR 50, Appendix B are documented (table 17C-3) and are in formats appropriate to their applications, such as:

- *A. Management responsibility statements.*
- *B. Position descriptions of management and professional personnel.*
- *C. Engineering instructions.*

D. Quality assurance and reliability procedures.

E. Projects procedures.

F. Purchasing procedures.

G. Construction site procedures.

Each of the above contains detailed procedures and instructions relating to the functioning of the quality program. Approval and distribution of the procedures is controlled by the manager responsible. For example, engineering procedures within the PWRSD are approved by the engineering manager and distribution is controlled by his staff. Other groups affected by one department's procedures review the procedures prior to their approval.

Table 17C-3 relates the various NES manuals and written procedures in relation to the applicable NRC criteria.

Technical and contractual information necessary to assure effective implementation of these policies and procedures is developed, documented, and controlled through a standard Westinghouse system which consists in part of the establishment of:

A. System design parameters.

B. Equipment specifications.

C. Corporate process specifications.

D. Corporate material test specifications.

E. Corporate Purchasing Department specifications, including specifications for materials.

F. Component specifications.

G. Drawings, drawing lists, and bills of material.

H. Purchase orders.

I. Operating procedures.

J. Job and work orders.

K. Quality assurance procedures.

The quality assurance program provides that all activities affecting quality will be accomplished in accordance with documented instructions, procedures, and drawings and that appropriate quantitative and qualitative means of verifying quality are satisfactorily accomplished and included as appropriate.

17C.1.6 DOCUMENT CONTROL

Each of the NES divisions provides measures to assure effective document control. Below is a description of the document control procedures which provide methods for establishing control of instructions, drawings, and procedures related to quality and safety. In addition, these procedures provide a means to assure that obsolete documents are not used, that controls are exercised for document changes, and that review and approval of changes is performed by organizations originating the document.

17C.1.6.1 PWR Systems Division

Within the PWRSD, there are a variety of documents used in the design and procurement of the PWR plant equipment. In the paragraphs below, the controls in use to assure content adequacy and the correct distribution are discussed.

The various sources of PWRSD procedures relating to quality assurance are summarized in table 17C-3. Each of the manuals has written instructions describing the review, approval, distribution, and revision of procedures. Typically, the preparation of new or revised procedures are controlled by the department responsible for the manual. Prior to issue, proposed procedures are routed to affected groups. Written comments are received and resolved. Approval is the responsibility of the department manager, who assures the completeness and resolution of the review. Manuals are serialized and assigned to specific individuals. Distribution of new and revised procedures is made to each person assigned a manual. Manual holders are responsible for updating their manuals. Implementation of most procedures is at the date of issue and is clearly defined by the distribution letter. In exceptional instances, when implementation varies from the issue date, specific instructions are provided in the body of the procedure.

Since design information is provided by equipment specifications and drawings, both types of documents are controlled by specific instructions. For both new and revised equipment specifications, these transmittal forms designate which groups review each document and approval requirements are clearly established. The manager of the originating group is the person responsible for assuring that before approving the document, all steps required by instructions have been completed satisfactorily. This includes the proper review and resolution of written comments as well as the technical adequacy of the document. Both drawings and equipment specifications are distributed to central control groups from which formal and demand distributions are made.

Approved drawings are microfilmed and distributed to satellite files. Obsolete drawings are exchanged for revised drawings. Obsolete issues are returned to the central file and destroyed. Past revisions are available only from the central file. All full- and half-size copies of drawings are informational. Since more than one revision to a drawing is applicable to different plants, a computerized drawing control system defines applicability. When a new or revised drawing is sent to central files, it is accompanied by a applicability form which is the input to the computer system. Drawing lists are issued to all satellite files and project offices monthly; partial change lists are issued more frequently. The written drawing control system requires division personnel to determine applicability of a drawing by referral to the drawing lists.

Process specifications; i.e., specifications that detail fabrication, inspection, and testing requirements, are handled in a manner similar to equipment specifications except that their development, approval, and distribution is coordinated by the Mechanics and Materials Technology Group.

Procurement documents are controlled by the Purchasing Department. The purchasing manual contains written instructions which detail how purchase orders, purchase order change notices, and procurement advisory releases are originated, reviewed, approved, and distributed. The instructions specify that sequential unique numbers be applied to all procurement documents. A computerized system identifies the latest serial number used on each document within the purchase order. The PWRSD has no specific responsibility at the construction site. Documents at the site are transmitted through the applicant or are sent by PWRSD supplies. Written instructions define the requirements for transmittal of documents to the applicant. Prior to distribution, the applicability of each document is assured by the approval of the cognizant engineer and project manager. Quality releases are forwarded to the site by equipment manufacturers. The quality release system is described in subsection 17.1.7. Computerized reports are issued twice a month to identify the quality releases applicable to site-delivered PWR plant equipment.

17C.1.6.2 <u>Electro-Mechanical Division</u>

Manuals, as summarized in table 17C-3, are developed and maintained in the manner described for PWRSD manuals in subsection 17C.1.6.1. The quality manual is the responsibility of the quality assurance manager. Where action and responsibilities of the departments are affected, the managers of these departments shall also approve related sections. The manual is reviewed on a scheduled basis and revised as necessary. As revisions are approved, the quality assurance manager shall supply copies of changed sections to manual holders.

Maintenance or correlated documentation to material, component, and assembly processing, inspection, and test is determined during initial planning. Documentation instructions are incorporated directly into the manufacturing routing or are referenced in separate inspection and test instructions. All documents, including changes and revisions, are reviewed for adequacy and approved for release by authorized personnel as described in subsections 17C.1.3.2 and 17C.1.4.2.

Electronic data processing is utilized in preparing and maintaining drawing and specification lists showing the applicable revision. These lists are strategically located in the factory and office area. The drawing and specification distribution control system assures that current information is available to the user and that manufacturing information is upgraded.

17C.1.6.3 <u>Tampa Division</u>

Control is maintained over the issuance of all design, welding, nondestructive testing, and manufacturing documents affecting quality. Each drawing is reviewed and signed by Design Engineering, Metallurgy, Quality Assurance, and Manufacturing prior to release to the Drawing Control Center. Copies of the drawing are prepared from aperture cards for transmittal to the Production Planning Department, who in turn prepares a feeder package that includes the necessary drawing and distributes the complete package to the applicable manufacturing group. The Drawing Control Center issues a master engineering drawing list. This drawing list contains all drawing numbers in the active category, including the latest revision. Distribution of the list is made twice a month to appropriate department managers, including the quality assurance manager. Each department is responsible for maintaining only up-to-date drawings.

Procedures that include welding, nondestructive tests, and manufacturing are maintained and controlled through the Metallurgical Group.

Serialized Tampa Specifications Manuals are issued to appropriate individuals in each department. New and revised specifications are distributed to manual holders who are responsible for inserting the new specification and destroying the outdated specification. Each specification is identified by a specific number. This number is shown on the drawing and *dictates the procedure to be used for each specific operation; i.e. welding, nondestructive test, cleaning, etc.*

Tampa quality control instructions (TQCIs) are issued to all QA personnel by quality assurance engineers or supervisors. The purpose of these instructions is to detail specific functions and responsibilities of QA personnel; i.e., N-1 form and data plate processing procedure, productive work station budget charges, inspection stamp issue, etc. These instructions are numbered and updated as necessary.

Tampa specifications are reviewed, approved, and signed similar to the drawings, with one notable exception; the division safety engineer also reviews and approves each Tampa process specification and manufacturing procedure. Additional discussions of document controls are contained in subsections 17C.1.3.3, 17C.1.4.3, 17C.1.9.3, and 17C.1.10.3.

17C.1.6.4 Pensacola Division

Within the Pensacola Division, operational and administrative procedures are developed and implemented as described within the PWRSD's subsection 17C.1.6.1.

The Quality Assurance Manual (QAM) is the responsibility of the Quality Assurance Department. This manual is approved by division management. Revisions are issued to holders of "controlled copies" as listed on records maintained by the Quality Assurance Department.

All drawings and product process specifications are controlled by Pensacola Division Design Engineering. Contract applicability is controlled through an engineering design release. The design release generates input to the business systems master drawing and specification list (computer listing). This computer master drawing/specification list is the contractual core internals configuration control document.

All material procurements, manufacturing, quality assurance plans, and inspection plans are completed and coordinated with the master drawing specification list.

All changes are keyed to the master drawing/specification list through design releases which trigger and control all implementations of manufacturing planning, quality, and inspection planning.

Additional discussions of document controls are contained in subsection 17C.1.3.4.

17C.1.6.5 <u>Specialty Metals Division</u>

The content and issuance of manufacturing, engineering, and quality information to the manufacturing, inspection, and test areas of the shop is controlled. An authorized change notice system controls process and/or inspection changes. Discussions of document controls are contained in subsections 17C.1.3.5, 17C.1.4.5, 17C.1.9.5, and 17C.1.10.5. Manuals are controlled as discussed for the PWRSD in subsection 17C.1.6.1.

17C.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Each of the NES divisions maintains its own system for control of purchased items. In general, the items purchased by the manufacturing divisions are in the raw materials category; therefore, the controls used are of a different nature than those used by the PWRSD. However, the principles of evaluation, selection, auditing, and documentation of quality are applied by all the NES divisions.

NES furnishes for each component at the construction site a copy of the purchase order (including changes), the design specification, and a quality release. These documents certify component quality and satisfy regulatory requirements pertaining to site documentation.

The quality release is a NES certification document which provides for:

A. *The specific identification of the procured material by purchase order number.*

B. *Certification that the equipment meets all requirements of the purchase order, drawings, and specifications. Identification of those procurement requirements which have not been met. Requirements which have been deferred; i.e., to be accomplished at the site, are clearly stated. Contingent conditions; i.e., conditions that are to be corrected by the supplier, are identified and correction is documented by a certification by the supplier. The supplier's certification describes the action taken, is signed by a responsible member of the supplier's organization, and is attached to the quality release. In addition, the quality release identifies the deviation notices which have dispositioned nonconformances to purchase order requirements.*

C. *The authorizing signature on a quality release is that of a NES quality assurance representative, or in specifically authorized situations, a member of the supplier's quality organization.*

Each NES division has a documented procedure which describes the requirements for completing, authorizing, issuing, distributing, and revising quality releases. Periodically, the PWRSD audits site releases from all NES divisions. These audits are performed in accordance with a written checklist. Audit reports are distributed to cognizant management for correction.

17C.1.7.1 <u>PWR Systems Division</u>

Prior to considering a new supplier for placement of a purchase order, a supplier evaluation is conducted. This is done in accordance with a written checklist. The results are documented in a report issued to management personnel of Purchasing, Engineering, Quality Assurance, and Projects. The evaluation is conducted by a team consisting of personnel from Purchasing, Engineering, and Quality Assurance. Other personnel, such as material and process personnel and manufacturing engineers, participate as required.

Considerations of the evaluation include all elements of the NRC's quality assurance criteria to the extent these criteria are applicable to the equipment being procured. Deficiencies in the supplier's organization or systems are resolved with the supplier's management prior to placing a purchase order. If an existing supplier does not maintain the required quality level on PWRSD orders, a similar team will review the supplier's problems and make recommendations to his management to correct the situation immediately. When problems arise, Westinghouse specialists aid the supplier in specific areas, such as welding, manufacturing, and nondestructive testing, to resolve the problem. In this manner, Westinghouse assures the continued high level of supplier performance necessary to obtain the quality level required by the contract.

PWRSD surveillance of suppliers during fabrication, inspection, testing, and shipment of components is planned in advance and performed in accordance with written quality plans. These plans are prepared by QA engineers and are based on the technical requirements of the purchase order. The plans are reviewed and approved by Quality Assurance Department management.

The purpose of a quality plan is to provide planned guidance to the QA field representative by identifying those characteristics which are most important to quality and reliability; providing specific instructions for the witnessing, documentation, and acceptance of the equipment; and providing a summary of quality releases issued for the specific purchase order. The plan identifies those supplier documents requiring approval and the points during manufacturing and test that Quality Assurance intends to witness. Special emphasis is placed on the aspects of manufacture and inspection that most directly affect performance of the equipment. Lead units of a new design get particular attention in the supplier's shop by both Quality Assurance and Engineering Department representatives.

When planning the surveillance activities, Quality Assurance develops a visit schedule. Visits are more frequent during the initial stages of manufacture, particularly to a new supplier, with frequency diminishing as the supplier demonstrates his capability. The purpose of PWRSD surveillance of suppliers is to provide Westinghouse management first-hand objective assurance of compliance with specified requirements. The principle followed is that the supplier is responsible for inspecting and testing his product. The PWRSD field representative assures that the supplier has done this, rather than attempting to perform the supplier's inspection for him or duplicate the work he has done.

The frequency and scope of surveillance varies with the degree of importance of equipment, supplier performance, complexity of the component, and other factors. This determination is made by Quality Assurance in conjunction with Engineering. Quality Assurance residents are established as necessary.

Surveillance is accomplished in accordance with the quality plans described above. During the surveillance visits, the field representative sees that written instructions and procedures are kept current, that corrective action is implemented, and that other necessary controls are effective. The QA representative informs, in writing, the supplier directly of problems he discovers and obtains commitments to correct them. He brings these problems to the attention of the supplier's management as required to obtain resolution. PWRSD management is made aware of the surveillance activities, including supplier discrepancies and audit results, by means of the trip report issued by the QA representative for each visit to a vendor.

When the QA representative is satisfied that the equipment can be released for shipment, he prepares a quality release form, and distributes copies of the form to the supplier and cognizant personnel within the PWRSD. The equipment can then be released through normal engineering purchasing channels for shipment. The supplier forwards the quality release with the equipment to the plant site.

The PWRSD has no direct responsibility for receipt inspection of equipment at the site. The applicant or his designated representative establishes the site-receiving activities. The PWRSD provides recommendations to the applicant for handling and storage of equipment and the documentation as described in subsection 17C.1.7 to assure quality.

In some instances, the supplier is authorized by the PWRSD to prepare a supplier quality release. This authorization is given only to those suppliers who have, over a period of time, demonstrated an effective quality system. PWRSD QA personnel periodically audit the supplier's system to assure continued performance.

17C.1.7.2 Electro-Mechanical Division

Prior to the award of purchase orders, Quality Assurance performs a survey to evaluate and approve all procurement sources for purchased material or services covered by the quality assurance program.

Based upon the type of component to be manufactured, the vendor may be required to submit process outlines and procedures (covering nondestructive testing, manufacturing, and/or inspection) to EMD for information or approval prior to manufacture. Such submittals, when specified, are reviewed by cognizant engineering personnel to assure adequate material control and conformance to drawing, specification, and purchase order requirements. In addition, EMD maintains a comprehensive supplier surveillance program. Suppliers are visited on a scheduled basis and audit reports formulated, evaluated, and maintained on file for future *reference. On complex purchase items, a QA field representative may visit the supplier's facility to witness nondestructive or destructive testing or to perform verification of dimensional inspection. As required by applicable purchase orders, Quality Assurance releases material for shipment. The QA field representative, after acceptance, documents the results of source inspections and releases by means of the quality control field release report. This report, along with other specified supplier documentation, accompanies shipments of material to EMD receiving inspection.*

Upon the receipt of supplier-furnished material, certified reports and related documents are reviewed and components inspected by the Receiving Inspection Section for compliance with the purchase order and related ordering data in accordance with instructions prepared and issued by Quality Assurance. After acceptance, the received material is forwarded to controlled stores or released directly to manufacturing. Nonconforming material is identified and held in quarantine until proper disposition is made.

17C.1.7.3 Tampa Division

Established controls assure that all purchased materials conform to purchase order requirements, including material and drawing specifications. It is the Tampa Division's policy to formally release to the Manufacturing Department all materials and parts that are to be used in NSSS equipment.

A Tampa Division team, consisting of two or more selected personnel, audits a new supplier's operation to determine acceptability as a supplier. For pressure-boundary or safety-related components, this supplier survey is conducted prior to procurement. The auditors are selected from the Quality Assurance, Metallurgy, Purchasing, Manufacturing, Planning, or Production Departments. The Quality Assurance Department conducts surveillance inspection and audits as necessary to assure acceptable quality products.

Source inspection is performed on all pressure-boundary material, plates, forgings, castings, and tubes. Prior to shipment, the Tampa quality representative will inspect, complete a source inspection form, and identify the material with the assigned test number and purchase order number.

A copy of the purchase order's applicable specifications, drawings, and prior source inspection data is furnished to the Receiving Inspection Section. Inspection of incoming material, not subject to source quality assurance but requiring in-house inspection, is accomplished by the Quality Assurance Department.

17C.1.7.4 Pensacola Division

Various techniques are used to monitor supplier performance. Prior to considering a supplier acceptable, a supplier evaluation is performed and results reported. After purchase order award, a quality history is maintained based on inspection results, and reevaluations are conducted as necessary. When necessary, a request is issued to the supplier by the Pensacola Division for corrective action to maintain product quality and to request a statement of the specific corrective action initiated by the supplier. Source inspection and surveillance ratings are used to determine supplier quality qualification.

Upon receipt of supplier-furnished material, the following quality assurance actions are initiated:

A. *Verification that all certified test reports, letters of compliance, dimensional data, welding records, heat treat charts, etc., required by the purchase order have been supplied by the vendor and are complete and correct.*

B. *Inspection of the product or material to determine acceptability according to instructions issued by the cognizant quality engineer.*

C. *If the product is accepted and released by Quality Assurance, the material, component, or assembly is forwarded to a controlled storage area to await future use, or is released directly to Manufacturing Operations.*

D. *If the product is rejected, the defective product is held in a controlled area until proper evaluation and disposition is made.*

17C.1.7.5 Specialty Metals Division

Subcontractors are limited to those which have demonstrated their capabilities. They are formally evaluated and selected on the basis of the capability of their quality system. Suppliers are classified in three categories for the purposes of quality acceptance because of the marked differences in the types of suppliers used by the Specialty Metals Division (SMD). Class I, Raw Material Suppliers, and Class II, Conversion Suppliers of SMD Material, are both surveyed by questionnaires to determine their acceptability. Then, all incoming lots from new vendors are checked chemically and by sampling techniques to verify supplier test reports and certifications. Class III, Conversion Suppliers Subcontracted Work, are surveyed to an audit format to assure that the supplier has procedures and processes that will meet the requirements of the intended purchase order. The Quality Assurance Department maintains records of the quality performance of each supplier. These records are maintained to rate suppliers as to their performance and to aid in developing and improving the suppliers' quality program.

When required, Quality Assurance Department personnel make surveillance visits of quality organizations of suppliers to assure continuous quality of purchased material and to assure that objective evidence of quality is maintained. Information obtained through these visits and through suppliers' audits provides data for determining the continuing acceptability of a supplier.

Suppliers are classified into categories for purposes of surveillance because of the marked difference in the type of vendors used by SMD. The first category applies to raw material suppliers whose products are evaluated by receipt inspection sampling techniques. The second category applies to new suppliers. All incoming lots from new suppliers are checked chemically for required elements and compared to suppliers' certifications. Any discrepancies are resolved by investigation; comparison of analysis techniques are used. When these have been resolved as evidenced by five lots received with no discrepancies, material is accepted on the sampling of subsequent lots. Suppliers performing more critical fabrication comprise the third category. These suppliers are subject to periodic surveillance by Quality Assurance Department personnel in addition to confirmation at receipt inspection.

17C.1.8 IDENTIFICATION AND CONTROL OF MATERIAL, PARTS, AND COMPONENTS

Within each of the NES divisions, procedures exist establishing measures which assure that identification and traceability of items are maintained during the production of components for delivery to the nuclear power plant site.

17C.1.8.1 PWR Systems Division

QCS-1 and the administrative specification contain requirements that a supplier have measures to maintain identification and control of material, parts, and components. The procedures used to establish these measures and the application of the procedures are reviewed for adequacy during supplier selection and monitored for compliance during the surveillance activities.

17C.1.8.2 Electro-Mechanical Division

To assure that unacceptable items are not used, identification in the form of a pre-assigned sequential serial number is placed on material at the supplier's plant or at EMD, depending on part-end use and processing. This identification remains with the material throughout subsequent manufacturing operations as a control number enabling tracing to the supplier's heat, slab or lot, and test data.

Serialization requirements are determined during the initial planning stages and fall into the following general categories:

A. Specification, contract, material, and equipment requirements.

B. Critical components.

C. Nondestructive test control.

D. Manufacturing control.

The minimum level of identification is shown on the detail subassembly or assembly drawing, including marking location and method, and takes into consideration that the location and method do not effect function or quality. This identification is also reflected in the manufacturing routing. Sequential serial numbers are issued by the Production Department and assurance against repetition is maintained through serialization log books. Serial numbers are pre-assigned to certain purchased items and included as part of the purchase order requirements.

Heat identity is maintained through all operations when required by contract, material, equipment, or code specifications. This identity is maintained by transfer of the heat number from operation to operation and/or by appropriate documentation through the use of sequential serial numbering and serial number log books.

17C.1.8.3 Tampa Division

A test number system is used as positive and permanent identification of materials and items purchased. The number identifies the supplier's heat number, slab or lot number, and the physical and chemical property records. In instances of multi-piece orders, subnumbers are utilized in the event a common melt was used for all items. One major exception is the tube bundle material. In this application, the heat number is used and recorded for each specified tube location in the bundle from definite orientation reference points.

The test numbers consist of a letter and a five-digit number assigned and affixed or stamped on each piece of material, component, assembly, or set of materials and parts which are for use in the manufactured product. The test number is also recorded as a permanent record in the inspection point program.

Nonconforming material and components are properly identified until corrective action has been taken. Materials that require additional tests are tagged and held until tests have been accomplished and results evaluated.

17C.1.8.4 Pensacola Division

Control identification is maintained of all materials and products to insure traceability to heat number. Major components and/or assemblies are normally serialized. All components rejected in-house have a permanent serial number assigned and marked on them. Material supplied to the Pensacola Division for product use is identified in a manner traceable to the original heat identity and/or purchase order as applicable. This identity is maintained when material is placed in the storage area prior to assignment for specific product fabrication.

Prior to the issuance of any material or component to fabrication, verification is made that the item issued satisfies the related drawing requirement. Routing information specifies material identity and the type of identification required on the components to be fabricated.

17C.1.8.5 Specialty Metals Division

Permanent identification and marking methods are used for control of materials throughout the manufacturing area. An identifying number is applied or attached to the material as it enters the plant by the receiving inspector. This number is modified as the material is processed; however, the basic number is used for identification in both processing and storing of the material. When a shipment is received, the receiving checker identifies the material and notifies the receiving inspector. After inspection, the receiving report is stamped according to the determined disposition (accept, reject, repair). Traceability to heat number is maintained.

17C.1.9 CONTROL OF SPECIAL PROCESSES

All NES divisions have established measures and procedures which maintain control over special processes. These include the qualification of processes and personnel for welding and inspection in accordance with ASME requirements, nondestructive inspection in accordance with SNT-TC-1A (1980) standards, and other processes as may be necessary for adequate control.

Recognizing the importance that valve bodies and other cast components may have to nuclear safety, the PWRSD as the lead division employs the following program to demonstrate that these items meet design requirements.

For valves, the PWRSD has included in its procurement requirements by an addendum to QCS-1 the following actions to be performed by the valve manufacturer.

A recorded dimensional survey shall be made by the seller of both the body and bonnet as follows:

A. *The first piece of each style as patterned in the "as-cast" or "wrought" condition to ensure that final valve assembly tolerances can be achieved.*

B. *The first piece, every multiple of 10, and the last piece of the finished machined body and bonnet shall be inspected for wall thickness at the location of the minimum wall by design. Three wall thickness readings shall be recorded on the dimensional survey. In addition, the weld preparation configuration and the maximum envelope dimension in the x, y, and z plane shall be inspected and recorded as a checkmark on the dimensional survey provided they are within the design tolerances.*

These requirements are applicable to valves larger than 2-inch nominal pipe size.

The intent of this requirement is not to provide an inspection survey on each valve, but to impose process controls during manufacturing to ensure that the process starts out in control and is sampled to verify that continued control is maintained. The dimensional surveys give documented evidence that the controls are operative, and additional requirements are implemented where required to meet expended applicant commitment.

For other cast components, such as pump casings, piping, fittings, etc., similar controls as noted above are contained in purchase orders, equipment specifications, or drawings. The degree of these controls depends on the component type, configuration, and application. The various controls include, as appropriate, checks of thickness as a part of receiving inspection, checks after machining, hydrostatic "proof" tests, and checks during final inspection. Surveillance representatives perform system and process audits of vendors to ensure continued control of the various cast components. These quality control and assurance techniques are designed to demonstrate that the components meet design requirements.

Other special processes, such as welding, nondestructive testing, electrochemical machining explosive forming, cleaning, and painting, are prescribed by means of documented procedures. For example, paint applications are detailed in documents known as process specifications. These specifications, similar to the equipment specifications discussed in subsection 17C.1.3, are process-oriented and contain requirements such as scope of paint application, selection of paint, surface preparation and condition, method of application, curving, repair of coating, methods of removal, etc. Quality provisions provide for monitoring the process, criteria for visual examination, and checks of paint characteristics such as adhesion, flexibility, and thickness.

17C.1.9.1 <u>PWR Systems Division</u>

QCS-1 and the administrative specification contain requirements that a supplier have measures for control of special processes. The procedures used to implement these measures and the application of the procedures are reviewed for adequacy and monitored for compliance during *the surveillance activities. In addition, equipment specifications or purchase orders identify certain processes or personnel qualifications which require PWRSD review and/or approval. Special process procedures and personnel qualifications are maintained under the document control and records retention systems.*

PWRSD personnel are qualified in accordance with a nondestructive testing certification program which conforms to SNT-TC-1A (1980).

Supplier procedures for special processes must be approved by PWRSD. The QA surveillance representative monitors the supplier's activities to ensure that all special processes are performed by properly qualified personnel using approved procedures.

17C.1.9.2 Electro-Mechanical Division

Recognizing the need to control special processes, the EMD has established departmental responsibilities for developing, reviewing, implementing, and controlling special processes, including the requirements for associated personnel qualifications. Special processes, including welding, heat treatment, and nondestructive testing procedures, are reviewed by Quality Assurance to assure compliance to applicable codes, standards, specifications, and criteria.

Welder procedure qualification is conducted in accordance with the ASME Boiler and Pressure Vessel Code Section IX. In addition, the welding process includes a program of weld electrode control. Manufacturing penetrant operators (Level I) and quality control nondestructive test inspectors (Level II) are qualified to SNT-TC-1A (1980)standards. Heat treatment processes are controlled through EMD's calibration program. Special process procedures and credentials of qualified personnel are maintained under document control and records keeping systems.

17C.1.9.3 Tampa Division

Definite departmental responsibilities are established to identify the need for documentation and review of special process procedures, as well as any associated personnel qualifications. The Metallurgical Department writes all welding and associated procedures, including preheat and heat treatment. The Quality Assurance Department writes all nondestructive test specifications, such as radiography, ultrasonics, magnetic particle, and liquid penetrant. These process specifications are in accordance with ASME Sections III and IX. Each process specifically defines the personnel qualification required by the applicable code. Welding qualification procedures and the required documentation are in accordance with the applicable code requirement. All process documents are subject to the controls and reviews noted in subsection 17C.1.6.

Lists of qualified welders and nondestructive personnel are issued to departmental supervisors and the quality assurance technicians; welding records and nondestructive reports indicate the individual that performed the welding and the technician that performed the nondestructive test. Credentials of qualified personnel are maintained under the records program described in subsection 17.1.17.3.

17C.1.9.4 Pensacola Division

The Quality Assurance Department, in conjunction with the Manufacturing Engineering Department, identifies, defines, and establishes special processes and process controls. Advanced quality planning includes identification of the need for qualification programs for special processes, equipment, and personnel. Prior to issuing manufacturing information for the processing of the product, all such documents are to be forwarded to the Quality Assurance Department for review. This information includes all drawings, specifications, routings, and other documents directly involved in processing the products.

Review of these documents by the Quality Assurance Department consists of verification that special tooling, fixturing, or gauging used to determine product quality is indicated at the correct operation/sequence, and verification that special testing together with acceptance criteria is in compliance with ordering information.

Upon completion of this review and determination that the required information is complete, correct, and adequate, the Quality Assurance Department completes the document by adding inspection operations to the routing at the applicable phase in processing, inspection methods and procedures, inspection forms to be completed, and sampling plans to be applied if applicable. Special emphasis is focused on control of heat treating and welding processes. Qualification of personnel, conformance of process to applicable requirements, and records of process data are constantly evaluated. Special process procedures and qualification records of personnel are maintained under document control and records retention programs.

17C.1.9.5 Specialty Metals Division

Responsibilities have been established to assure that special processes, such as heat treatment, pickling, cleaning, etc., are specified and defined in manufacturing specifications. Procedure responsibilities, controls, and qualifications, as necessary, are outlined in the documents. Inspectors are used by the Quality Control Department for maintaining a uniform quality level, controlling manufacturing processes, and for overall product quality assurance. Several techniques of nondestructive testing are used by the Quality Control Department and are required by process and quality specifications. These are ultrasonic inspection, fluorescent magnetic particle, and liquid penetrant. Special process specifications and credentials of personnel qualifications are maintained under document control and records retention programs.

17C.1.10 INSPECTION

Each NES division, in order to ensure that attributes affecting quality are controlled, has established measures by which inspections are performed. As noted earlier, adequate independence exists between inspection groups and manufacturing functions to allow effective, overall controlled conditions.

Physical examinations, measurements, and tests are conducted as appropriate to demonstrate product quality. Various job positions within the quality organizations are detailed by written position descriptions to assure that qualified personnel, with specialized training as necessary, are utilized in the inspection and quality assurance function.

17C.1.10.1 PWR Systems Division

Since the PWRSD does not manufacture anything directly, emphasis is placed on supplier surveillance. The principle followed is that the supplier is responsible for inspecting his product and PWR QA personnel verify his controls to assure the adequacy of inspection. As such, inspection as PWR is more appropriately described as supplier surveillance. Details of the PWR surveillance program are contained in the description of Control of Purchased Material, Equipment, and Services, subsection 17C.1.7.

17C.1.10.2 Electro-Mechanical Division

The quality program provides for assurance that all fabrication, welding, machining, and other operations are performed under controlled conditions. Features include verification of documented work instructions (routings), preparation of procedures for monitoring product quality (inspection point program), and the physical examination and testing at significant points during the manufacturing cycle. Criteria for approval or rejection is established by Quality Assurance in accordance with engineering drawings and specifications.

In-process, final inspection, and test operations are incorporated into manufacturing routings, which are approved and signed by Quality Assurance Engineering in accordance with internal requirements. The need for special inspection tools, fixtures, and gages; inclusion of all inspections; and adequacy and completeness of the routing information are considered by Quality Assurance Engineering during their review.

Before an in-process or final inspection operation is performed, reference is made to appropriate document control lists to assure the use of proper revisions of drawings, specifications, and procedures. Measuring and testing equipment is checked before use to assure proper inspection and calibration status by making reference to the calibration sticker. After a manufacturing sequence is inspected and prior to proceeding to the next operation, the acceptance of the operation is recorded by Inspection on the inspection control card. Each *inspector verifies that all prior operations are listed and signed off on the card. Upon performing all inspection operations as required by the manufacturing routing or other approved internal instruction, Inspection is responsible for documenting the inspection utilizing inspection forms, checklists, suitable log book, etc. Prior to functional testing or to shipment of completed parts or components, suitable releases are obtained, as required, from the Quality Assurance Department records center.*

Nonconforming conditions noted during inspections are documented and processed in accordance with documented procedures. Sampling inspection by attributes and/or variables is used. Normally, appropriate government sampling references are used; however, sampling plans may be developed from recognized texts and techniques to suit EMD needs. The sampling and quality levels are based on the function of the component and/or characteristics. Records of sampling inspection are maintained on appropriate documentation and filed in the Quality Assurance Department records center or in the receiving inspection area per internal instructions.

17C.1.10.3 Tampa Division

To provide process and product verification, a detailed inspection point program is generated for each manufactured unit. The inspection point program parallels the manufacturing operation sequence prepared by the Industrial Engineering Section and approved by the Quality Assurance Department. The quality assurance engineer reviews the operational lineup for compliance with ASME Code Section III, drawings, and applicable Westinghouse specifications. The pertinent inspection points are inserted by the quality assurance engineer. These points specify the type of inspection, applicable nondestructive tests, data to be recorded, and charts or forms to be completed for documentation of inspection operations. As discussed in subsection 17C.1.3.3, the need for special inspection tools, fixtures, and gages is determined during Quality Assurance Engineering reviews prior to the release of drawings for manufacture or purchase.

The applicable inspection point program is distributed to the quality assurance technician in the applicable manufacturing area for the specific component. The QA technician initials or stamps the inspection points upon completion of inspection, signifying acceptance. All operations within a manufacturing section are detailed and accepted on the inspection checklist. Quality assurance technicians verify the completeness of the checklist to assure that all operations are complete within the section. Any deviation from the specified routing between manufacturing groups requires documentation and Quality Assurance concurrence. The program and its applicable data forms, charts, and logs become a permanent Quality Control Department record to provide objective evidence of the inspection operations.

Inspection point programs are audited by Quality Assurance personnel, customer representatives, and the authorized code inspector. Specific mandatory notification points *which require witnessing or inspection by the customer representative are established and are so designated in appropriate documents.*

17C.1.10.4 Pensacola Division

The Quality Assurance Department maintains inspection planning to obtain assurance of the following:

A. *Inspection instructions are clear, concise, and adequately definitive.*

B. *Inspection operations are referenced and applied at the most effective points in the process to monitor product quality.*

C. *Relatively complex inspection procedures are reviewed with inspection supervisors for concurrence with the information reflected.*

D. *Inspection methods requiring training of personnel are issued prior to production and a training program is initiated on a timely basis so that personnel involved will be qualified for production inspection processing.*

E. *Special tooling, fixturing, and gaging equipment required for product quality evaluation are designed, built, evaluated, and released for inspection use.*

F. *All documentation formats which will report inspection results are prepared and issued.*

Required inspections and tests are displayed by the shop routing and inspection instructions referenced. Within the Pensacola Division, the movement of hardware from cost center to cost center is controlled by means of a document known as the "routing." This document is a brief description of the fabrication sequence, including inspection operations distributed throughout the manufacturing cycles at those points which will verify the quality of the product. It is related to a particular engineering release to manufacturing. Each operation must be signed off prior to proceeding to the next operation. Documentation is evaluated by Quality Assurance to assure that all operations, inspections, and tests have been performed and are acceptable. Objective evidence of the inspection performed is documented on nondestructive testing reports and detailed dimensional inspection reports. After manufacture and inspection, the components/assembly must be released by Quality Assurance before utilization on the next assembly. Major components are evaluated by Quality Assurance to assure that all inspection tests have been performed and are acceptable.

17C.1.10.5 Specialty Metals Division

The primary function of the Quality Control Group is to provide assurance that processing capability requirements are being met. This includes the observation and evaluation of operator's adherence to manufacturing instructions, written procedures, and other control documents. In process Inspection inspects the product at designated stages of manufacture, including first-piece and patrol where appropriate, to assure compliance with the intermediate and final product specifications.

Procedures for inspection points are preplanned and are prepared by the quality assurance engineer in the form of quality specification cards or they are incorporated as part of manufacturing instructions. The need for special inspection tools, fixtures, and gages is considered by both Quality Assurance and Manufacturing Engineering during the development and review of these instructions. These specifications indicate that the inspection be performed during the process so that it can be determined that the product has met specifications. These are reviewed prior to issue by Manufacturing Engineering. If the inspections indicated are acceptable, the material is released by inspection and may continue on to the next process or operation. Results of inspections are posted on the designated quality form by the inspector. He also applies his numbered stamp opposite the inspection step on the manufacturing or process specification follow card.

17C.1.11 TEST CONTROL

Means are established at each of the NES divisions to control testing. These measures provide for the development of procedures, a means of assessing adequacy of the tested items, and designation of the responsibility for performing the various phases of the testing activities.

17C.1.11.1 PWR Systems Division

QCS-1 details that tests required by a contract be described by clear and current written procedures which assure that tests are performed as specified. The criteria for acceptance or rejection shall be included. The procedures for meeting the above are a part of the supplier quality plan submitted to the PWRSD for approval. The administrative specification contains similar requirements. These two documents also require that the supplier maintain records showing the results of the tests. These records are reviewed for acceptability by the PWRSD. Tests are conducted by groups within the supplier organization considered acceptable during supplier selection; they are monitored during PWRSD surveillance.

The Quality Assurance Department also participates in the PWRSD development test program for critical new equipment designs. Test plans and specifications are drawn to clearly define the number of units to be tested, the conditions under which tests should be conducted, and the types of data to be collected and analyzed. Development tests are designed through the use of *statistical theory and engineering judgment to obtain the optimum relevant information to assure that performance, life, and cost requirements are met. Quality Assurance reviews the test plans, monitors the setup and conduct of the test, and reviews the test reports. Assistance is also provided from independent laboratories and testing agencies in following test programs.*

17C.1.11.2 Electro-Mechanical Division

In order to assure that desired product quality is maintained by clear and complete instructions of a type appropriate to the circumstances, Test Engineering prepares and maintains test work instructions and monitors their application. These instructions are contained in routings, drawings, and test specifications with the support of auxiliary test procedures.

Initial application of new test instructions is jointly performed by Test Engineering and the tester to ensure their feasibility and adequacy. Test results are validated by Test Engineering and the tester and evaluated by cognizant engineering personnel.

17C.1.11.3 Tampa Division

The quality assurance program provides for assurance that tests are performed under controlled conditions. Features include verification of documented work instructions, preparation of procedures for monitoring product quality, and testing at significant points during the manufacture cycle. Criteria for acceptance or rejection is established by Quality Assurance in accordance with engineering drawings and specifications.

17C.1.11.4 Pensacola Division

Quality Assurance prepares detailed procedures to implement required nondestructive techniques. The methods and techniques used, as a minimum, meet the requirements of the ASME Boiler and Pressure Vessel Code. All nondestructive tests are performed per drawing requirements, and substitute methods are utilized to substantiate questionable data. Nondestructive test (NDT) personnel are qualified to applicable standards, and a current qualification status is maintained for all NDT technicians.

17C.1.11.5 Specialty Metals Division

The SMD performs chemical, physical, and metallographic tests to assure that its products conform to required specifications.

Testing is performed in accordance with written procedures. The details of testing are recorded in log books in the individual labs and results are formally reported to the Quality Assurance office where they become part of the order and heat data system.

17C.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

The NES divisions maintain a separate means of controlling measuring and test equipment. Each division has developed and maintains a separate basis for its own program, considering such attributes as inherent stability of their equipment, purpose or use, desired accuracy, and degree of usage. All measuring and test equipment used for the acceptance and verification of product quality are maintained under control systems. Such specifications as Mil-C-45662 and handbook Mil-MDBK-52 serve as a basis and provide guidance in the determination of an effective program for the control of test and measuring equipment. Typical of this equipment are micrometers, plug gages, height gages, dial verniers, voltmeters, temperature recorders, pressure gages, hardness testers, etc. Documented procedures detail the requirements for the calibration of measuring and test equipment and the use of appropriately traceable measurement standards.

17C.1.12.1 PWR Systems Division

The requirement for a supplier to maintain a system for calibration of all examination, measuring, and test equipment is contained in the administrative specification and in QCS-1. All calibration must be traceable to national standards. PWRSD verifies the acceptability of the system during the supplier selection and monitors for compliance during the surveillance activities.

17C.1.12.2 Electro-Mechanical Division

The EMD, under the direction of the Quality Assurance Department, maintains an extensive tool and gage control program utilizing electronic data processing. All tools and gages used in the manufacture and inspection of completed products are inspected and calibrated in accordance with established procedures. Control of the use of measuring and test equipment is maintained by the tool crib approach where equipment is logged out to individuals or assigned to specific areas. The program requires that any equipment which becomes damaged or out of calibration be forwarded for repair or recalibration as required. Under this program, precision tools and gages are inspected and calibrated at specified intervals based on their stability, purpose, and degree of usage. All tool and gage inspection and calibration is performed in a controlled environment. Calibration stickers are affixed to all equipment, excluding personal tools which have been found acceptable under the program. Personal tools are identified by name with the calibration status maintained by the gage inspector. Reference *standards used are certified and traceable to the National Institute of Standards and Technology.*

17C.1.12.3 Tampa Division

Formalized procedures defining calibration frequency and maintenance of gages and test equipment used for inspections are in effect and implemented by the Quality Assurance Department. Quality control tools and gages are identified by quality control serial numbers which are color coded to indicate calibration status, and are controlled by a tool crib card index system. Established calibration schedules for each type of tool or gage used for inspection purposes are implemented. Frequency of calibration is based on engineering judgment and verified by Quality Assurance review of calibration records. Damaged or inaccurate measuring and test equipment is removed from the cycle until repaired, recalibrated, or replaced. Master measuring standards are maintained and calibrated on a frequency cycle by a qualified laboratory with standards traceable to the National Institute of Standards and Technology.

Electrical test equipment such as magnetic particle equipment is on a scheduled calibration cycle. The Works Engineering Department is responsible for maintenance and calibration. This effort is audited by the Quality Assurance Department.

Pressure test gages used for hydrostatic and gas leak tests are checked and calibrated on a frequency schedule; deadweight test equipment is used to verify calibration. The procedures are designed to assure accuracies within established standards and include disposition and/or corrective measures when discrepancies are noted.

17C.1.12.4 Pensacola Division

All decisions on the acceptance of any product or quality characteristic are made by utilizing inspection and test equipment under calibration by the Quality Assurance Department; this calibration is traceable to the National Institute of Standards and Technology. Each gage is identified with a unique identifying serial number. For each individual gage, there is a gage inspection record card used to record the results of periodic inspections.

Calibration frequencies are initially established by an engineered estimate of the total useful life of the gage and the frequency of recalibration at one-fifth of this estimated time.

Calibration frequencies are adjusted based on an evaluation comparison of the gage usage versus the wear recorded on the gage inspection record card. Any gage which passes through three calibration cycles without being used is placed in an inactive status until needed at future time.

All gaging and testing equipment is tagged with a sticky label which identifies the date calibrated, the date of next calibration, and an identifying stamp of the gage inspector who performed the calibration. Calibration control is maintained by advancing the gage calibration record card in a pigeonhole filing system where each pigeonhole represents one workweek within the 52 workweek year. Gage record cards which appear in the current workweek slot are calibrated within the current workweek.

The area supervisor, whether manufacturing or inspection, has the responsibility to promptly report any gage in his area known to be functioning improperly. Defective equipment is red tagged and scheduled for repair or replacement.

17C.1.12.5 Specialty Metals Division

Measuring instruments, gages, fixtures, standards, masters, and any item needed to facilitate precise measurement are under the jurisdiction of the Quality Assurance Department. Quality Assurance is responsible for the control of gaging equipment used by Quality Control and Manufacturing personnel. Quality Assurance Engineering checks and maintains all gages and fixtures assigned to Quality Control for use in accepting products. The gage calibration lab is under the supervision of Quality Assurance and has written procedures for the periodic recall, inspection, and calibration of gages. When calibration is complete, proper notation is made on each gage or instrument in addition to recording calibration results on the tool inspection master card. To assure continued accuracy, a safety check is made when a gage is dropped, mishandled, or the calibration status is questionable. Calibration frequencies established by Quality Assurance and based on experience are verified periodically by a review of the tool inspection master card.

17C.1.13 HANDLING, STORAGE, AND SHIPPING

Measures to establish control over handling, storage, and shipping are in documented procedures in use at each of the NES divisions.

17C.1.13.1 PWR Systems Division

Since the PWRSD does not manufacture equipment, emphasis is placed on controlling the supplier's handling, storage, and shipping activities. QCS-1 and the administrative specification specify that a supplier's quality program require the use of handling procedures and handling equipment inspection procedures to prevent damage to a product. The vendor must have adequate written work and inspection instructions for storage, preservation, packaging, and shipping to protect the products from damage, loss, deterioration, or substitution. As required by the equipment specification, these procedures may be subject to approval by the PWRSD.

A supplier's procedures and systems for handling, storage, and shipping are evaluated during source selection and monitored for compliance during PWRSD quality assurance surveillance.

17C.1.13.2 Electro-Mechanical Division

The EMD has established procedures defining a system of inspection and usage control for all lifting fixtures and devices used in the factory in accordance with existing specifications and applicable industrial safety standards. In addition, a review committee has the responsibility to review lifting and handling fixtures and equipment and to arrange for marking their identification and limits. The review committee also has the responsibility to determine necessary corrective action when noncompliance to a required standard is found.

In process movement and storage of material, components, subassemblies, and assemblies is defined by manufacturing procedures that provide the instructions necessary to maintain identity and protect finished attributes and surfaces from damage.

Packaging and shipping requirements required by contract are reviewed and appropriate manufacturing routing prepared. Specific packaging instructions are referenced within manufacturing routing; they include pertinent inspections to assure that preservation, packaging, and packing is accomplished to protect the products and/or supplies from damage, loss, deterioration, degradation, or substitution.

17C.1.13.3 Tampa Division

Established procedures and training is provided for materials-handling personnel. A procedure book on safe practices in rigging and crane operation, including sketches and handling methods for all major production lifts, is supplied to the riggers and crane operators. The guidelines set forth in the procedure book are established by the Tampa Division Lifting Committee and includes requirements for inspection of chain slings, wire rope slings, shackles, eye bolts, plate clamps, and hoisting ropes. In addition to visual examination, all hooks are periodically nondestructively tested.

Protective covers are used on nozzles after final machining to protect weld preparations, preserve cleanliness, and minimize damage.

Design Engineering generates shipping drawings that detail arrangements for barge or car shipments, cradles to be used, and size and number of tiedown straps and rods. In conjunction with the drawing, a formal engineering procedure is issued that specifies strap and tiedown locations, welding to be performed, special reinforcement, etc., plus liquid envelope protection application instructions.

17C.1.13.4 Pensacola Division

Handling, storing, and shipping methods are defined by written procedures. Quality Assurance reviews all procedures, makes recommendations for improvements, and audits the procedures for compliance. Special frames, jigs, and containers are used for in process handling and storage to protect the dimensional and finished attributes of component parts.

17C.1.13.5 Specialty Metals Division

Facilities are available and procedures are written for crating, packaging, preserving, and identifying products for overseas and domestic shipments in accordance with various commercial and federal specifications.

The material is handled and stored according to procedures which describe the manner of storage, protection of finishes, and control of limited life supplies. Quality Assurance conducts periodic audits of storage areas to assure conformance to applicable storage procedures and requirements.

17C.1.14 INSPECTION, TEST, AND OPERATING STATUS

Each of the NES divisions has established procedures to indicate the inspection, test, and operating status of materials, parts, and components. The purpose of these procedures is to preclude inadvertent bypassing of inspection and tests.

17C.1.14.1 PWR Systems Division

QCS-1 and the administrative specification contain requirements that a supplier have measures to indicate an inspection, test, and operating status of an item. The procedures used to establish these measures and the application of the procedures are reviewed for adequacy during supplier selection and monitored for compliance during the surveillance activities.

17C.1.14.2 Electro-Mechanical Division

The positive identification of inspection status for each product is accomplished by the inspection control card (ICC) which travels with the product from one manufacturing section to the next. The quality control inspection stamps on the ICC indicate acceptance of the product at specific checkpoints. The inspection stamp system maintains control of each stamp symbol and the individual to whom it is issued.

17C.1.14.3 Tampa Division

Procedures are used to maintain identity of all material and manufactured components, beginning with receipt inspection and through final shipment of the product. Material is identified by use of a Tampa test number as described in subsection 17C.1.8. Test numbers are not changed when material is transferred to another order. These numbers are recorded in the applicable inspection point program. The required test reports can be identified by the master test number log.

The inspection point program readily identifies the completed manufacturing operations as well as the completed inspection and/or nondestructive tests. Also, the Production Planning Department keeps an up-to-date status report on each component for all orders.

Nonconforming material is identified by use of tags as described in the error appraisal notice procedure (subsection 17C.1.15).

The inspection documents identify the technician that performed the inspection or test; the individuals qualifications are in accordance with SNT-TC-1A (1980) as required by Section III of the ASME Code.

17C.1.14.4 Pensacola Division

All manufactured material is identified by a route sheet throughout the manufacturing operation. The route sheet is stamped at all inspection, NDT, and manufacturing steps. If an item is discrepant, the number of the rejection document is entered on the route sheet; therefore, the status of each item is always available. Control and use of stamp issuance is maintained by the Quality Assurance Department.

17C.1.14.5 Specialty Metals Division

A formal record system is followed to identify the stage of manufacture of a product at the SMD. Manufacturing instructions indicate inspection and test points in the sequence of operations and require formal release to proceed. An identifying number is applied to the material as it enters the plant by the receiving inspector. This number is modified as the material is processed; however, the basic number is used for identification. Process follow cards readily identify the status of the manufacturing operations as well as inspection and test status. Because of the nature of the operations, these cards are frequently damaged or obliterated; therefore, the cards do not represent the official status of materials. Verification of the status indicated can be made from records maintained by Production Control, from manufacturing reports, and from Quality Control releases.

Nonconforming materials are identified and held for disposition. Actions become part of the records and data system.

17C.1.15 NONCONFORMING MATERIAL, PARTS, OR COMPONENTS

Each NES division has documented procedures to control nonconforming material, parts, and components which prevent their inadvertent use and provide for their identification, segregation, and disposition. Normally, each NES division makes disposition of these nonconforming material reports which vary from specifications and standards established within the division. Nonconformances of PWRSD equipment specification requirements are controlled by the PWRSD. All approved nonconformance reports are identified on the final quality release. In addition, nonconformances which affect site installation, test, maintenance, or operation are submitted to the applicant. Additionally, Westinghouse will notify the applicant of each significant deficiency found in the process of design, manufacture, fabrication, installation, construction, and testing and inspection, which:

- *A. If left uncorrected, could adversely affect the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.*
- *B. Represents either,*
    1. *A significant breakdown in any portion of the quality assurance program. Deficiencies found during the normal operation of the quality assurance program, such as inspection, test, audits, design reviews, etc., are not considered as indication of a breakdown.*
    2. *A significant deficiency in final designs approved and released for construction.*
    3. *A significant deficiency in the construction of or significant damage to a structure, system, or component requiring corrective action involving extensive effort.*
    4. *A significant deviation from performance specifications requiring corrective action involving extensive effort.*

Notification by Westinghouse will be as defined in 10 CFR 50, 55(e).

17C.1.15.1 PWR Systems Division

17C.1.15.1.1 Deficiencies at Suppliers' Plants

QCS-1 and the administrative specification described above contain specific contractual requirements for controlling nonconforming material or workmanship.

Suppliers are required to provide a system for the identification, documentation, and evaluation of discrepancies, and for alerting the supplier cognizant management to the need for corrective action. A Westinghouse deviation form is initiated at the supplier's and completed per instructions on the back of the form. Upon receipt at the PWRSD, the deviation is processed in accordance with documented instructions to assure proper review and disposition by Design Engineering and Quality Assurance, with concurrence of Materials and/or System Engineering as appropriate. Possible dispositions are accept, repair, scrap, and hold and resubmit.

When repair is indicated, acceptance of the repaired item upon completion of the repair is noted by a QA signature on the deviation form. A permanent file of the deviation records is maintained by the PWRSD.

17C.1.15.1.2 Deficiencies at the Construction Site

A written procedure provides for documented reporting of deficiencies on NSSS equipment found during plant construction by Westinghouse personnel. These reports are submitted by Westinghouse site engineering personnel to the cognizant engineering department. Like reports from suppliers' plants, these reports are reviewed for necessary action, formally approved by the cognizant engineer, and permanently filed. Summary reports are developed to alert appropriate levels of management of the deficiencies found and the actions taken.

17C.1.15.2 Electro-Mechanical Division

Nonconforming material is identified and held in quarantine until disposition is received. All deviations are reviewed and disposition is made by engineering personnel. The nonconforming material control and evaluation program provides for review and evaluation of cause and corrective action to prevent recurrence.

Data concerning nonconforming material is categorized to type of deviation and component. Appropriate reports are formulated under the program for evaluation and management action toward reduction in costly defectives and toward quality improvement.

The EMD utilizes a multicopy material review report (MRR) with nonrepetitive sequential numbers. These numbers are applied to nonconforming components, subassemblies, or assemblies depending on marking restrictions. In addition, the EMD utilizes electronic data processing to summarize material review reports by order, part, serial, or lot number. This summary provides convenient quality history trace back.

The MRR system provides:

- *A. Identification of nonconforming materials and their status.*
- *B. Segregation of nonconforming materials from production material by physical means where possible or by positive identification where not possible by physical means.*
- *C. Formal disposition of the nonconforming materials from Engineering and Quality Assurance Departments.*
- *D. Verification of rework or repair to correct the nonconforming material by inspection personnel in the form of a stamp or authorized signature on appropriate documentation.*
- *E. Correction of the causes of nonconforming materials to prevent recurrence during all phases of procurement and fabrication.*

17C.1.15.3 Tampa Division

Materials, parts, or components which do not meet design drawings or specifications, process specifications, or quality standards are considered defective material. All deviations are documented by the error appraisal notice (EAN) system. An EAN is issued upon discovery of nonconforming or defective work produced by any department, and a caution tag is attached to the material signifying a discrepant condition. Upon issuance of an EAN by the quality assurance technician, the quality assurance engineer verifies the technician's findings. The quality assurance or manufacturing engineer recommends corrective repair when applicable. The EAN is then transmitted to Metallurgical Engineering for all discrepancies that involve welding and related functions, such as heat treatment, preheat, etc., and/or Design Engineering for all other conditions. Design or Metallurgical Engineering agrees with the recommended repair or issues further instructions. The answered EAN is returned to Quality Assurance who then initiates an attachment that details the manufacturing operations and inspections required to conform to the specified engineering instructions. The EAN with attachment is distributed to Manufacturing, Production, Quality Assurance, and other departments involved. Upon completion of the repair, the EAN attachment is signed off by the area supervisor and the quality assurance technician and is added to the inspection records for that particular component or part; the caution tag is removed from the piece by the QA technician.

The system provides identification of discrepant material, a formal notification to individuals involved, and the formal signature signifying repair and acceptance of the material. In the event that material or parts are not repairable and are to be scrapped, the piece is removed from the manufacturing area to prevent inadvertent use.

17C.1.15.4 Pensacola Division

The Pensacola Division maintains systematic control of the identification, segregation, and disposition of all nonconforming materials, components, subassemblies, and equipment. A specific form is assigned for each type of rejection that could occur in the processing of all material and the subsequently fabricated product. These forms provide for the specific delineation of existing conditions resulting in a "reject" disposition.

The issuance, processing, and dispositioning of this documentation is under the control of the Quality Assurance Department. When the determination is made that a discrepancy exists, the material or product involved is immediately tagged and segregated, if feasible, until disposition is made on the governing document.

Disposition of nonconforming material is defined by formal procedures. All dispositions must be approved by the pertinent quality engineer before any rework, repair, scrap, or vendor return action is taken. All dispositions are formally documented; files are kept for future reference and management evaluation.

The Inspection Department is responsible for performing the routed inspection operations and tests, signing off the rework/repair routing, and repairing required nondestructive reports and detailed dimensional reports.

17C.1.15.5 Specialty Metals Division

A formal procedure is used in identifying and controlling all defective material detected at receiving, in any stage of processing, or at final inspection. It is the responsibility of the Quality Control Department to identify any production material that varies from contract, drawings, specifications, procedures, process, or quality standards. Quality Control then informs the proper personnel, determines or obtains disposition, approves and signs off the corrections or modifications, maintains records of the occurrence, and assures that fundamental corrective action is taken.

Deviations from product or process contractual requirements are reported to the NFD or Tampa in accordance with their instructions for their disposition.

17C.1.16 CORRECTIVE ACTION

Each NES division has a corrective action program which has a means for determining the need for corrective action, documenting the need and the action taken, and reporting the need and action taken to appropriate levels of management.

17C.1.16.1 PWR Systems Division

QCS-1 requires that the supplier's quality system provides for the identification and evaluation of significant or recurring discrepancies and for alerting supplier's cognizant management of the need for corrective action. The supplier must review corrective action for effectiveness and the need for further action. The supplier's corrective action program is reviewed for adequacy during supplier selection and monitored for compliance during the surveillance activities.

Through a computerized coding system, Quality Assurance receives deficiency data on Westinghouse-supplied equipment from suppliers and construction sites to determine patterns of occurrence by supplier, by component, or by process. With this as a guide, Quality Assurance and cognizant engineers determine corrective actions needed to prevent recurrence. This action is in addition to assuring that the supplier or site personnel take corrective action on the individual deficiencies reported. Through periodic reports, management is informed of the need for action and the action taken. Several of the reports are trip reports, field discrepancies report summaries, and audit reports.

17C.1.16.2 Electro-Mechanical Division

The EMD's quality program provides, through a computer system, for the early detection of nonconforming material, summarization of recurring or significant quality problems, analysis of trends, and diagnosis of causes. Appropriate levels of management are notified of significant failures, malfunctions, and nonconformances. The corrective action program covers vendor quality performance, in-plant operations, and field installation problems.

17C.1.16.3 Tampa Division

The quality assurance engineer is responsible for reviewing all EANs and other data relating to the quality of products and operations under his cognizance. As a result of this review, the engineer is responsible for initiating positive corrective action when a quality problem of significant magnitude is indicated on the basis of safety, cost, or possibility of shipping undetected discrepancies. Recurring discrepancies indicate a need for correction of design, process, or method.

Systems have been established to identify and document trends in specific operations, such as tube welding, inspection and test, and in all major pressure welds. Reports are issued to cognizant management personnel for action when deemed necessary.

17C.1.16.4 Pensacola Division

The cause of deficiencies and action taken by the responsible group to prevent the recurrence of discrepancies are documented. Recurring deficiencies are analyzed by cognizant quality and manufacturing engineers and appropriate action is taken to prevent reoccurrence. A quality costs program permits computer tabulation of specific quality costs so that problem areas may be readily identified, investigated, and corrected.

17C.1.16.5 Specialty Metals Division

Formal procedures require the documented reporting of all material and manufacturing deficiencies. The documentation includes a complete description of the deficiencies, the specification or requirement involved, and the disposition. These conditions require formal review by appropriate levels of management. Recurring deficiencies are analyzed by Quality Assurance and cognizant engineers for corrective action taken to prevent recurrence. The recommended action requires the review and approval of responsible Manufacturing, Engineering, and Quality Assurance Department managers.

In addition to corrective action covering deficiencies, standard statistical evaluations are performed on current manufacturing data to determine manufacturing trends to prevent the manufacture of defective material.

17C.1.17 QUALITY ASSURANCE RECORDS

The NES quality assurance program requires the retention of those fabrication, inspection, and surveillance records essential to demonstrating product quality. Records are reviewed by Westinghouse QA personnel, microfilmed, and submitted to the applicant prior to plant acceptance. Records relating to the design and fabrication of NSSS equipment are available for review.

17C.1.17.1 PWR Systems Division

The administrative specification previously described requires suppliers to maintain records for each test (nondestructive, electrical, performance) specified in the purchase order. The administrative specification and equipment specification also require maintenance of other records, as required, such as material test reports, welder qualifications, inspection records, *etc. Records such as trip reports, deviation notices, and other quality-related documents form a part of the records maintained by the PWRSD.*

All suppliers are required to maintain these records for specified periods, after which they notify the PWRSD for disposition. Copies of records covering significant inspections on critical portions of the component are transmitted to the PWRSD. These inspection records, along with quality-related documents generated by PWRSD QA personnel, comprise the permanent quality file for each component; these records will be maintained for the life of the plant.

Table 17C-4 is a typical listing of the documents and records kept as a part of the NES quality assurance program. The eighth item, fourth listing shows the supplier and NES having a retention responsibility; the supplier retains all data relative to the component, and NES obtains copies of significant data to maintain as a part of the NES history file. Table 17C-5 lists some typical components and details the data retained by NES for each.

Records generated at the construction site are filed and maintained there.

17C.1.17.2 Electro-Mechanical Division

The EMD maintains sufficient product-related records to furnish documentary evidence of activities affecting quality. The documentation and data requirements are determined during the initial quality planning stage for each contract, and appropriate instructions and documentation checklists are prepared for internal records audits to assure that required records are generated. Records include results of technical and divisional reviews, inspections, audits, material analyses, data on work performance, operation logs, and test results. Closely-related data such as qualifications of personnel, procedures, and equipment are also retained.

All records are not necessarily maintained by Quality Assurance. Retention is determined by the respective department which conducts the review, test, qualification, etc.

17C.1.17.3 Tampa Division

Documentary evidence and records of inspection and other related manufacturing information are maintained. The records include materials test reports, nondestructive test reports and radiographic film, heat treatment logs and charts, inspection point programs, and all related documents including the EANs. The records will identify the inspector, the results, and the action taken to correct deficiencies.

17C.1.17.4 Pensacola Division

A records system is maintained by Quality Assurance to furnish documentary evidence of all results affecting quality. This system includes but is not limited to logging and filing of material certifications, inspection results, discrepancy documentation, test results, audit results, and other closely-related data such as qualification of personnel, procedures, and equipment.

17C.1.17.5 Specialty Metals Division

Objective evidence and records of the various inspection operations during the manufacturing cycle are shipped with the material and maintained by Quality Control. Records of in process inspection are maintained in Quality Assurance Department files for six months after contract shipment. Records related to source or receiving inspection are accumulated in the receiving inspection files. A permanent record of the results of calibration and checking of gages is maintained in the Quality Assurance files in line with the established gage control procedure.

17C.1.18 AUDITS

To verify the effectiveness of the quality assurance program, NES has a comprehensive system of audits. Planned and scheduled audits are conducted by:

- *A. The corporate quality staff of the NES divisions.*
- *B. The NES QA Committee of the NES divisions.*
- *C. NES divisions of other NES divisions for intra-NES purchases.*
- *D. NES divisions of their suppliers.*
- *E. NSD of Westinghouse site activities.*
- *F. Each NES division of its own internal programs.*

Quality Assurance audits are conducted in accordance with defined audit procedures. As required by these procedures, checklists are utilized in many cases. As a minimum, a documented audit report detailing discrepant areas with needs for corrective measures and records of resolution are maintained. The NES quality assurance program requires the originator of an audit report to follow an open item until action is taken to satisfy an audit action item. Areas subject to audit include all procedures and operations within each of the divisions which affect or have an active part in the total quality program as defined by 10 CFR 50, Appendix B. The schedule and sequence of operations to be audited is planned in advance.

For example, PWRSD audits of NES divisions furnishing equipment to the PWRSD are established on a calendar year basis by mutual agreement between the PWRSD and the particular division. The calendar year schedule identifies the operations to be audited.

A description of the various audits within NES is found in subsections 17C.1.18.1 through 17C.1.18.8.

17C.1.18.1 Westinghouse Corporate Audits

The Westinghouse Corporate Headquarters Quality Control staff has a formal audit program which applies to all divisions in Westinghouse, including divisions furnishing equipment or services to the nuclear industry.

The purpose of the audits is to provide an independent verification that the quality assurance programs of the Westinghouse divisions are effectively assuring that the product quality complies with the requirements of their customers and that the programs include the most effective approaches to prevent the manufacture of defective products.

Audits are performed of each division's quality assurance effort by a two-man team, consisting of a member of the Corporate Headquarters Quality Control staff and the quality assurance manager of another division in the same product group as the division audited. The audit normally takes five days. The Corporate Headquarters Quality Control audit of each Westinghouse division is held on the average of once every three years.

The quality assurance systems and procedures that have been established by the division are reviewed to determine if these systems and procedures are sufficient to provide an effective program. Observations are then made to assure that the established systems and procedures are being correctly followed.

An oral presentation of the findings and conclusions of the audit is made to the division general manager, quality assurance manager, and other personnel affected by the audit findings. The items recommended for improvement in the quality assurance program are presented as well as recommendations of approaches for accomplishing these improvements.

Following the audit, a written report containing the findings and recommendations reviewed in the oral report is prepared and sent to the responsible division personnel. In addition, a copy of the report is sent to the executive vice president to whom the division reports and to the corporate vice president -manufacturing.

17C.1.18.2 NES Quality Assurance Committee Audits

The Westinghouse Nuclear Energy Systems Quality Assurance Committee has established an audit program which applies to all NES Westinghouse divisions engaged in nuclear supply system design or manufacture of PWR equipment. The purpose of the audits is to provide in-depth evaluation of the quality assurance policies and processes of the various Westinghouse NES divisions in order to verify that they result in products and services which meet safety and reliability requirements. Particular emphasis during the audit of the quality assurance programs is placed on compliance with the requirements of 10 CFR 50, Appendix B.

In addition to carrying out audits, the Committee serves as a forum to communicate quality and reliability activities, and to establish improved and consistent division policies of quality assurance in light of nuclear industry requirements.

Annual quality assurance system audits are conducted of each Westinghouse NES division by an audit team composed of representatives from the Committee. Typical team membership is three men. Each audit normally takes three days. The Corporate Headquarters Quality Control audit of the Westinghouse NES division described above substitutes for the annual Westinghouse NES audit the year it is held.

At the conclusion of each audit, an oral presentation is made by the audit team to the division general manager and quality assurance manager of the division which has been audited. Following the audit, a written report containing the findings of the audit and recommendations for improvement in the quality assurance program and its implementation is sent to the responsible division personnel, to the committee members, and to the Westinghouse NES executive vice president. This procedure assures high-level management attention to actions needed to carry out recommendations of the audit.

17C.1.18.3 PWR Systems Division

17C.1.18.3.1 Suppliers' Plants

The Westinghouse PWRSD's audit function of suppliers is described in subsection 17C.1.7, Control of Purchased Material, Equipment, and Services. The NES manufacturing divisions are also considered as suppliers to PWRSD and scheduled audits of the divisions are conducted by PWRSD Quality Assurance.

17C.1.18.3.2 Internal

The Quality Assurance Department performs audits within the PWRSD. These audits cover procedures and implementation of the procedures. The audits are performed periodically by a *team headed by QA personnel and selected from appropriate engineering groups of the PWRSD and from outside divisions as necessary. Audit findings are documented and sent to management for review and corrective action, where necessary.*

Additional audits of the PWRSD are conducted by the NES Quality Assurance Committee and the Westinghouse Corporate Headquarters Quality Control staff.

17C.1.18.4 Electro-Mechanical Division

The EMD maintains comprehensive internal audit programs to assure that established systems are being followed and that systems adjustments are made. Audit programs also provide management with a continuing overview of quality trends, methods, and functions. Informal audits are performed by various functions within their areas of responsibility; they report findings and corrective actions in writing as required. Formal audits are conducted by a team consisting of members from Engineering, Manufacturing, Quality Assurance, and other functions as required. Deviations noted during the audit are corrected and appropriate action taken to assure against recurrence. Audit reports are formulated for review and action by management.

Surveillance audits involve reinspection of previously-accepted work, verification of required documentation, and reviews of failure analysis and corrective action methods.

Additional audits of EMD are conducted by the NES Quality Assurance Committee, the Westinghouse Corporate Headquarters Quality Control staff, and by PWR QA personnel.

17C.1.18.5 Tampa Division

A continuing program of surveillance audits is conducted to assure conformance to standards, procedures, and methods for all activities affecting product quality. The program includes reinspection of previously accepted work, verification of required documentation, and reviews of error appraisal notice data and corrective action methods. The audit program also provides management with a continuing overview of quality trends, methods, and functions.

The Reliability Engineering Department also has a formal internal audit program. The internal audit program is designed as a cooperative effort by all activities performing quality-related functions for the purpose of assuming compliance with requirements and identifying and resolving problem areas. Audit teams consisting of two members from outside Quality Assurance and a Quality Assurance Department advisor conduct approximately six scheduled audits per year. The results of each audit are discussed with the manager of the area audited and documented in a audit report issued by the Reliability Engineering Department to designated management. As necessary, the reliability engineering manager assigns personnel to initiate and follow-up on required corrective action. When re-audits are deemed necessary, *they are performed by the Quality Assurance Department to provide assurance that corrective action has been effective in resolving problem areas.*

Additional audits are conducted by the NES Quality Assurance Committee, the Westinghouse Corporate Headquarters Quality Control staff, and by PWR QA personnel.

17C.1.18.6 Pensacola Division

Periodic internal audits are conducted by Quality Assurance Engineering to assure compliance with applicable procedures. These audits cover general manufacturing practices and adherence to quality procedures. A request for corrective action is issued to document adverse audit results. This document is sent to management for appropriate action. Follow-up reviews are made to ascertain that the stipulated corrective action has been instituted.

Additional audits of the Pensacola Division are conducted by the NES Quality Assurance Committee, the Westinghouse Headquarters Quality Control staff, and by PWR QA personnel.

17C.1.18.7 Specialty Metals Division

Internal quality audits are held to determine the adequacy of established procedures for controlling quality and to evaluate the degree of compliance with the procedures.

The Quality Assurance audit team, performing as a management audit function, is responsible for the following:

A. *Investigating potential and actual problem areas which directly or indirectly affect the quality and performance of the SMD's products.*

B. *Reporting favorable and unfavorable conditions to supervision directly responsible for corrective action.*

C. *Reviewing and re-auditing corrective action measures taken and the conditions that caused them to assure that such conditions have been eliminated.*

Subjects of primary concern to the team are documentation, procedure follow, process conformance, product quality, and housekeeping functions. Correction of discrepancies noted by the audit team rests with the supervisor of the area affected, but solution and permanent elimination of the basic problem remain the responsibility of the manager to whom that supervisor reports.

Additional audits of the SMD are made by the NES QA Committee, the Westinghouse Headquarters Quality Control staff, and by Tampa and NFD Quality Assurance personnel. The PWRSD does not audit the SMD.

17C.1.18.8 Nuclear Services Division

Quality Assurance conducts independent audits of Westinghouse personnel activities at the construction site to assure that proper procedures and instructions are available and in use, and that adequate controls exist and are effective. Reports of audits are sent to top management of the PWR Systems Division. Additional audits of NSD are made by the NES Quality Assurance Committee and the Westinghouse Corporate Headquarters Quality Control staff.

REFERENCES

1. *Dollard, W. J., "Nuclear Fuel Reliability and Quality Assurance Program Plan," WCAP-7800, Revision 4-A, April 1975.***]***

[HISTORICAL] [TABLE 17C-1 (SHEET 1 OF 3)

NSSS FUNCTIONAL RELATIONSHIP FLOW SCHEDULE[(a)]

Item	*Function*	*Originating Group*	*Participating Group*	*Subsequent Action*
1	*Dissemination of contractual requirements*	*Project manager*		*Nuclear safety, quality assurance and reliability, and functional engineering groups, PWRSD equipment design groups, field operation group, and purchasing*
2	*Identification of regulatory requirements*	*Nuclear safety*	*Projects and functional design groups*	*PWRSD equipment design groups, functional design groups, and quality assurance and reliability*
3	*NES quality assurance program*	*NES quality assurance groups (through the NES Quality Assurance Committee)*	*NES equipment design and manufacturing groups, projects, purchasing, and safety and licensing*	*Applicant and NES quality assurance groups*
4	*Quality assurance and reliability procedures*	*NES quality assurance and reliability groups*	*NES equipment design and manufacturing groups, projects, NES purchasing groups, and safety and licensing*	*NES quality assurance and reliability groups and applicant*
5	*Design control procedures*	*NES equipment design groups, functional design, and NES reliability groups*	*NES quality assurance, manufacturing, and purchasing groups, projects, and safety and licensing*	*NES equipment design and reliability groups and applicant*
6	*Specification of system and equipment functional requirements*	*Functional design groups*	*Projects, safety and licensing, and applicant*	*PWRSD equipment design groups, architect/engineer, and applicant*

TABLE 17C-1 (SHEET 2 OF 3)

Item	*Function*	*Originating Group*	*Participating Group*	*Subsequent Action*
7	*Equipment specifications or drawings*	*NES equipment design*	*Functional design groups, NES equipment design groups, quality assurance and reliability, projects, safety and licensing, applicant, and architect/engineer*	*Applicant, architect/engineer, constructor, purchasing, NES quality assurance and manufacturing groups, field operations group, and suppliers*
8	*NES manufacture*	*NES manufacturing groups*	*NES equipment design, and quality control and quality assurance groups*	*Constructor, applicant, and field operations group*
9	*Supplier selection and approval*	*NES quality assurance, equipment design, and purchasing groups*	*Projects and NES manufacturing groups*	*Supplier and purchasing*
10	*Supplier detail design, fabrication, and inspection documents*	*Supplier design, manufacturing, and quality control groups*	*NES equipment design groups, and NES quality assurance and reliability groups*	*Supplier and NES quality assurance groups*
11	*Product surveillance and process audits*	*NES quality assurance groups*	*Applicant and purchasing*	*Projects, NES equipment design groups, and supplier or*
12	*Quality release*	*Supplier and NES quality assurance groups*	*NES equipment design and purchasing groups and projects*	*Applicant, constructor, and field operations group*
13	*System layout drawings*	*Architect/engineer*	*Functional design groups and PWRSD equipment design groups*	*Applicant and constructor*
14	*Receipt inspection and erection of NSSS equipment at nuclear power plant site*	*Constructor*	*Applicant and field operations group*	*NES equipment design and quality assurance groups and projects*
15	*Plant testing and acceptance*	*Applicant*	*Architect/engineer, safety and licensing, field operations group, functional engineering groups, and NES equipment design groups*	*Applicant*

TABLE 17C-1 (SHEET 3 OF 3)

Item	*Function*	*Originating Group*	*Participating Group*	*Subsequent Action*
16	*Audits of*			
	a. NSSS quality assurance program	*Applicant, headquarter QC staff, and NES Quality Assurance Committee*	*NES quality assurance and reliability groups*	*All NES groups*
	b. NSSS design and fabrication	*Applicant and NES quality assurance groups*	*Architect/engineer and all NES groups*	*All NES groups*
	c. NSSS site work	*NSD quality assurance*	*Projects*	*Field operations group*

*a. The groups identified on this table relate to the functional chart depicted in figure 17C-2.**]***

[HISTORICAL] [TABLE 17C-2 (SHEET 1 OF 9)

NSSS FUNCTIONAL RESPONSIBILITIES

	Design Criteria		Detail Design		Manufacture		
Component	*Responsible*	*QA*	*Responsible*	*QA*	*Responsible*	*QC*	*QA*
Reactor Coolant System (RCS)							
Reactor vessel	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor vessel support shoes and shims	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor vessel insulation	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Full-length control rod drive mechanism (CRDM) housing	*PWRSD*	*A and PWRSD*	*EMD*	*PWRSD*	*EMD*	*EMD*	*PWRSD*
Part-length CRDM housing	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor coolant pump casing	*PWRSD*	*A and PWRSD*	*EMD*	*PWRSD*	*S*	*S*	*EMD*
Reactor coolant pump internals	*PWRSD*	*A and PWRSD*	*EMD*	*PWRSD*	*END*	*E*	*PWRSD*
Reactor coolant pump motor	*PWRSD*	*A and PWRSD*	*EMD*	*PWRSD*	*S*	*S*	*EMD*
Reactor coolant loop isolation valves	*PWRSD*	*A and PWRSD*	*EMD*	*PWRSD*	*EMD*	*EMD*	*PWRSD*
Steam generator (tube side)	*PWRSD*	*A and PWRSD*	*TD*	*PWRSD*	*TD*	*TD*	*PWRSD*
Steam generator (shell side)	*PWRSD*	*A and PWRSD*	*TD*	*PWRSD*	*TD*	*TD*	*PWRSD*
Pressurizer	*PWRSD*	*A and PWRSD*	*TD*	*PWRSD*	*TD*	*TD*	*PWRSD*
Reactor coolant piping	*PWRSD*	*A and PWRSD*	*PWRSD*	*A and PWRSD*	*S*	*S*	*PWRSD*
Reactor vessel internals	*PWRSD*	*A and PWRSD*	*PC*	*PWRSD*	*PD*	*PD*	*PWRSD*
Primary and secondary sources	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
CRDM seismic support structure	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*

TABLE 17C-2 (SHEET 2 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	*Responsible*	*QA*	*Responsible*	*QA*	*Responsible*	*QC*	*QA*
CRDM dummy baffle cans	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
CRDM cooling shroud assembly	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Bypass manifold	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Safety valves	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Relief valves	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Valves to reactor coolant system boundary	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Piping to reactor coolant system boundary	*PWRSD*	*A and PWRSD*	*A/E*	*A and PWRSD*	*C*	*C*	*A*
Reactor coolant pump seal bypass orifice	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Pressurizer relief tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Seal table assembly	*PWRSD*	*A and PWRSD*	*PD*	*PWRSD*	*PD*	*PD*	*PWRSD*
Instrumentation tubing and fittings	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Control rod clusters	*PWRSD*	*A and PWRSD*	*NFD*	*PWRSD*	*NFD*	*NFD*	*PWRSD*
Rod cluster control (RCC) thimble plug	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSDc*
Control rod drive mechanism head adapter plugs	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Chemical and Volume Control System (CVCS)							
Regenerative heat exchanger	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Letdown heat exchanger	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Mixed bed demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Cation bed demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*

TABLE 17C-2 (SHEET 3 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	Responsible	QA	Responsible	QA	Responsible	QC	QA
Reactor coolant filter	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Volume control tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Centrifugal charging pump	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Seal water injection filter	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Letdown orifice	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Excess letdown heat exchanger	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Seal water return filter	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Seal water heat exchanger	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Boric acid tanks	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Boric acid filter	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Boric acid transfer pump	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Boric acid blender	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Resin fill tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Boric acid batching tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Chemical mixing tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
System piping	PWRSD	A and PWRSD	A/E	A and PWRSD	C	C	A
System valves	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Boron Thermal Regeneration Subsystem							
Moderating heat exchanger	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD

TABLE 17C-2 (SHEET 4 OF 9)

Component	Design Criteria Responsible	Design Criteria QA	Detail Design Responsible	Detail Design QA	Manufacture Responsible	Manufacture QC	Manufacture QA
Letdown chiller heat exchanger	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Thermal regeneration demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Letdown reheat heat exchanger	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Boron Recycle System							
Recycle holdup tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator feed pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator feed demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator feed filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator condensate demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator condensate filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Recycle evaporator concentrate filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor coolant drain tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor coolant drain pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor coolant drain tank heat exchanger	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Liquid Waste Processing System							
Waste holdup tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator feed pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator feed filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*

TABLE 17C-2 (SHEET 5 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	*Responsible*	*QA*	*Responsible*	*QA*	*Responsible*	*QC*	*QA*
Waste evaporator	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator condensate demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator condensate filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator condensate tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste evaporator condensate tank pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Chemical drain tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-resin storage tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-resin sluice pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-resin sluice filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Laundry and hot shower tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Laundry and hot shower tank pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Laundry and hot shower filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Floor drain tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste monitor tank	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste monitor tank pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Waste monitor tank demineralizer	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Waste monitor tank filter	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Floor drain tank pump	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*

TABLE 17C-2 (SHEET 6 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	Responsible	QA	Responsible	QA	Responsible	QC	QA
Floor drain tank filter	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Gaseous Waste Processing System							
Gas compressor	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Gas decay tanks	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Hydrogen recombiner	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Safety Injection System (SIS)							
Refueling water storage tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Accumulator	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Safety injection pump	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Boron injection tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Boron injection tank recirculation pump	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Boron injection tank surge tank	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
System piping	PWRSD	A and PWRSD	A/E	PWRSD	C	C	A
System valves	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Residual Heat Removal (RHR) System							
Residual heat removal pump	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Residual heat exchanger	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
System piping	PWRSD	A and PWRSD	A/E	A and PWRSD	C	C	A
System valves	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD

TABLE 17C-2 (SHEET 7 OF 9)

Component	Design Criteria Responsible	Design Criteria QA	Detail Design Responsible	Detail Design QA	Manufacture Responsible	Manufacture QC	Manufacture QA
Refueling Equipment							
RCC changing fixture	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Control rod drive shaft handling fixture	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
New fuel storage racks	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-fuel storage racks	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Control rod drive shaft storage racks	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Guide tube cover handling tool	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
New fuel elevator	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-fuel pit bridge and hoist	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Vessel head lifting device	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Upper internals storage stand	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Lower internals storage stand	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor vessel internals handling device	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Reactor cavity manipulator crane	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
RCC thimble plug handling tool	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Spent-fuel assembly handling tool	*PWRSD*	*A and PWRSD*	*PWRSD*	*PWRSD*	*S*	*S*	*PWRSD*
Fuel Transfer System							
Fuel transfer tube and flange	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*
Fuel transfer components	*PWRSD*	*A and PWRSD*	*S*	*PWRSD*	*S*	*S*	*PWRSD*

TABLE 17C-2 (SHEET 8 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	Responsible	QA	Responsible	QA	Responsible	QC	QA
Nuclear Instrumentation Power Range							
Detectors	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Rack-mounted equipment	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Balance of system	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Rod Control Systems/Rod Position Indication System	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Radiation Monitors	PWRSD	A and PWRSD	PWRSD	PWRSD	S	S	PWRSD
Solid-State Shutdown System							
Input relay cabinet	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Logic cabinet	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Output relay cabinet	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Balance of equipment	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Reactor Trip Switchgear							
Switchgear and cabinets	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Bus duct	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Process Control Systems - Reactor Coolant Flow							
Rack-mounted equipment	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Field-mounted equipment	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD
Flow elements	PWRSD	A and PWRSD	S	PWRSD	S	S	PWRSD

TABLE 17C-2 (SHEET 9 OF 9)

	Design Criteria		Detail Design		Manufacture		
Component	Responsible	QA	Responsible	QA	Responsible	QC	QA

Legend

A	=	Applicant or designated representative
A/E	=	Architect/engineer
C	=	Constructor
EMD	=	Electro-Mechanical Division
NFD	=	Nuclear Fuel Division
NSSS	=	Nuclear steam supply system
PD	=	Pensacola Division
PWRSD	=	Pressurized-Water Reactor Systems Division
S	=	Supplier to Nuclear Energy Systems
TD	=	Tampa Division
QA	=	Quality assurance
QC	=	Quality control

[HISTORICAL] [TABLE 17C-3 (SHEET 1 OF 15)

WRITTEN PROCEDURES WITHIN NES FOR IMPLEMENTING QUALITY ASSURANCE

Procedure	*I*	*II*	*III*	*IV*	*V*	*VI*	*VII*	*VIII*	*IX*	*X*	*XI*	*XII*	*XIII*	*XIV*	*XV*	*XVI*	*XVII*	*XVIII*
PWRSD Policy and Procedures Manual	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
PWRSD Quality Assurance and Reliability Manual	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
PWRSD Engineering Policies and Procedures Manual	X	X	X	X	X	X	X		X									
PWRSD Purchasing Manual			X	X	X		X						X	X	X			
PWRSD Project Manual			X	X	X	X	X											X
Safety and Licensing Manual		X	X		X	X												
EMD Quality Assurance Manual	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
EMD Purchasing Department Manual						X												
EMD Engineering Department Instructions			X															
Pensacola Quality Program Manual	X	X	X	X	X	X	X	X	X	X	X	X	(a)	X	X	X	X	X
Tampa Standard Division Procedure			X	X	X	X	X	X	X	X					X			
Tampa Product Control and Design Procedure			X															
Tampa Quality Assurance Manual	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
SMD Quality Assurance Manual	X	X		X	X	X	X	X	X	X	X	X	(a)	X	X	X	X	X

a. Handled by specific instructions on shop travelers or in individual component specifications.

TABLE 17C-3 (SHEET 2 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
PWRSD Policy and Procedures Manual (PPM)	*To set forth and define division policies and procedures*	
Organization charts, charters, and personnel rosters	*To set forth policy and procedural instructions for establishing and maintaining organization charts, organization charters, departmental personnel rosters, and documentation of the authorities and duties of personnel and organizations for all PWRSD functions and activities that affect the quality of safety-related structures, systems, components, and services*	*I*
Quality assurance program	*To set forth policy and procedural instruction for establishing and maintaining a quality assurance program that ensures and demonstrates PWRSD compliance with applicable regulatory, industry, and Westinghouse quality assurance requirements for PWR nuclear power plants*	*II and V*
Classification of safety-related PWR plant components and services	*To set forth policy and procedure requirements covering the identification and classification of all safety-related structures, systems, components, and and services*	*II*
Design verification and design reviews	*To set forth PWRSD policy, procedural instruction, and design review guidelines to be followed in verifying designs at all significant design stages of structures, systems, and components involved in the nuclear steam supply system (NSSS) and appropriate system auxiliaries to the NSSS*	*III*
Quality requirements and standards	*To set forth policy and procedural instructions for specifying quality requirements and quality standards in design documents, such as specifications, drawings, etc., in such a manner that manufacturers and the plant constructor can demonstrate through inspection or testing that the structures, systems, and components meet the specified requirements*	*III*
Design change control	*To set forth policy and procedural instructions to control changes to PWRSD design documents in order to know and control the configuration of the facility, structure, system, or component throughout design, construction, and operation of the PWRSD plant*	*III*
Interface control	*To set forth PWRSD policy and procedural instructions necessary to assure that adequate information on quality, safety, and reliability requirements are included or referenced in procurement documents for items and services*	*IV*

TABLE 17C-3 (SHEET 3 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Procurement document control	*To set forth PWRSD policy and procedural instructions necessary to assure that adequate information on quality, safety, and reliability requirements are included or referenced in procurement documents for items and services*	*IV*
Document control	*To set forth PWRSD policy and procedural instructions to assure effective control (review, comment resolution, approval, issue, change, and disposition) of documents which prescribe all activities affecting quality, safety, reliability, and performance of safety-related PWRSD structures, systems, and components*	*VI*
Supplier quality assurance program surveys	*To set forth policy and procedural instructions for performing surveys of prospective and current suppliers' quality assurance programs to determine their acceptability and to ensure the required quality of purchased materials, items, and services essential to the overall effectiveness of the PWRSD quality assurance program*	*VII*
Handling, storage, shipping, and receiving	*To set forth policy and procedural instructions for the control of handling, storage, shipping, and receiving, including cleaning, packaging, and preservation of material structures, systems, and components for PWR plants to prevent damage, deterioration, and loss*	*XIII*
Control of nonconformances, reporting deficiencies, and corrective actions	*To set forth policy and procedural instructions for controlling nonconformances in material and equipment, reporting significant deficiencies, and exercising necessary corrective actions*	*XV and XVI*
Documentation and records	*To set forth policy and procedural requirements regarding control, maintenance, and disposition of records associated with the documentation of PWRSD activities relative to the assurance of quality, safety, and reliability of Westinghouse nuclear power plants*	*XVI*
Quality assurance audits	*To set forth policy and procedural instructions for establishing and executing a comprehensive system of planned and documented audits to verify compliance with all aspects of the PWRSD quality assurance program and to assess program performance*	*XVIII*
Qualification of personnel	*To set forth policy and procedural instructions for establishing and maintaining the qualification of personnel associated with the quality, safety, and reliability of structures, systems, components, and services provided directly by PWRSD or by suppliers for PWR plants*	*IX*

TABLE 17C-3 (SHEET 4 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Safety analysis report (SAR) control	*To set forth policy and procedural instructions for ensuring that the requirements specified in design documents for a particular PWR plant technically support all the functional design and safety system performance requirements specified in the SAR for the particular plant*	*II, III, and VI*
PWRSD Quality Assurance and Reliability Manual	*To set forth specific quality instructions for implementing the policies prescribed in the PWRSD Policy and Procedures Manuals*	
PWR quality assurance plan	*To describe the procedures and actions used by Westinghouse to assure that the design, materials, and workmanship employed in the fabrication and construction of systems, components, and installations within the Westinghouse scope of responsibility in a nuclear power plant are control led and meet all applicable requirements of safety, reliability, operation, and maintenance*	*I and II*
Equipment specification and drawing review	*To assure, through independent review of these documents, the adequacy of specifications and drawings prepared for equipment. The PWRSD method of review is explained in this procedure. The term "E-spec" refers to equipment specifications or drawings when' they are used in lieu of equipment specifications*	*III*
Purchase order review	*To describe Quality Assurance review of purchase orders for clarity and for adequacy of quality requirements*	*IV*
Quality control plans	*To define the procedure followed in developing quality control plans*	*V, VII, and XIV*
Drawing control	*To briefly describe how satellite files are controlled and what steps are to be taken by Quality Assurance and Reliability personnel to assure the validity of drawings used or referenced in the performance of their work*	*VI*
Control of nonconforming material	*To define an instruction for controlling, reviewing, and disposing of nonconforming materials through the use of the deviation notice. This procedure describes how nonconforming materials are documented, reported, and disposed of for discrepancies in equipment at suppliers' plants reactor internals assembly*	*XV*
Customer audits or PWRSD documentation	*This procedure deals with customer audit of the quality control documentation maintained by the PWRSD Quality Assurance Department*	*XVIII*

TABLE 17C-3 (SHEET 5 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Surveillance techniques	*To describe the method used in surveying suppliers' facilities to provide for control of quality of equipment delivered to PWRSD. Supplier surveillance is divided into two categories: audits that monitor the suppliers' operating systems and process procedures, and product verification in specific areas to exercise control over critical fabrication and test points. These two surveillance categories are taken into account in the quality control plan*	*VIII, VII, X, XI, XII, XIII, and XIV*
Quality control levels	*PWRSD applies three levels of quality control to procured materials. This procedure defines these levels and describes how they are assigned*	*II*
Preaward surveys and postaward audits	*Formal evaluations of new, prospective, or existing suppliers are necessary to determine and record their ability to meet Westinghouse Nuclear Energy Systems requirements for the manufacture of NSSS equipment. This procedure describes the means used by Quality Assurance for these evaluations*	*VII*
Customer participation in surveillance	*In fulfillment of contractual and regulatory requirements, customers' quality assurance representatives periodically visit Westinghouse and its suppliers' fabricating facilities to observe the Westinghouse Quality Assurance surveillance effort and to assure that supplier fabrication is control led; this procedure describes how this activity is coordinated*	*II, VII, and XVII*
Quality releases	*To describe the use of quality releases*	*VII*
Corporate audits	*To describe the audit program which is under the direction of the corporate direction of reliability control which is organizationally independent from the operating divisions of the corporation*	*XVIII*
Product documentation files	*To define the responsibilities and method of processing and maintaining the product documentation files*	*XVII*
Long-range file retention	*To clearly define the retention responsibilities for equipment design and fabrication records*	*XVII*
Training and certification of nondestructive testing (NDT) personnel	*To provide a uniform system for training, qualifying, and certifying personnel who need to utilize NDT methods and techniques and to assure they have optimum knowledge of the current state of the art commensurate with their specific job function*	*IX*
Design review	*To provide guidelines for conducting a design*	*III*

TABLE 17C-3 (SHEET 6 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Reliability analysis	*To describe the approach used in applying reliability analysis to component, subsystem, or system design*	*II*
Data feedback and analysis system	*To define the methods and responsibilities for acquiring, classifying, filing, retrieving, and analyzing empirical data on Westinghouse NSSS*	*XVI*
Equipment deficiency analysis	*To identify equipment defects through a systematic review of the deficiency data file*	*XVI*
PWRSD Engineering Policies and Procedures Manual (EPPM)	*To set forth specific engineering instructions for implementing the policies prescribed in the PWRSD PPM*	
Objectives and policies	*To state overall objectives and policies of Engineering within the scope defined by division policy*	*I and II*
Technical policies and procedures	*To set forth policy and procedural instruction governing the areas of position papers; preparation of E-specs; control of design changes; and control and distribution of licensing reports, SARs, reports supporting licensing applications, and drawings*	*III, V, and VI*
Quality assurance	*To define a procedure for stipulating the quality requirements of equipment designed or specified by PWRSD. In addition, key sections of the Quality Assurance Manual are referenced*	*All*
Administration	*To set forth policy and procedural instructions for such areas as supplier evaluation, purchase order submittals, and documentation and storage*	*IV, VII, and IX*
PWR Purchasing Manual	*To set forth specific purchasing instructions for implementing the policies prescribed in the PWRSD PPM*	
Selection of suppliers	*To set forth the policy on selection of suppliers*	*VII, XIII, and XV*
Order preparation and administration	*To set forth the instructions for the preparation and administration of an order (including requisitions, purchase orders, and change notices), procurement advisory releases, and acknowledgment*	*IV and V*
Coordination with other activities	*To set forth instructions for interface relationships with other groups*	*III, and XV*

TABLE 17C-3 (SHEET 7 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
General procurement considerations	*To set forth specific instructions on the evaluation of suppliers, visits, terms and conditions, proposal evaluation, and order review*	*VII, XIII, and XV*
PWRSD Project Manual	*To set forth specific project instructions for implementing the policies prescribed in the PWRSD PPM*	
Communications	*To set forth procedures for drawing transmittal, projects communications, communication with other NES divisions, review of engineering specifications, and the specification control system*	*III, V, and VI*
Engineering	*To set forth procedures for the interfacing relationships dealing with customer quality assurance, surveillance, and visitations*	*XVIII*
Procurement	*To set forth procedures for project activities associated with procurement*	*IV and VII*
Safety and Licensing Manual	*To set forth specific instructions for the Safety and Licensing Group on implementing the policies prescribed in the PWRSD PPM*	
Process assurance plan	*To address each of the 18 criteria applicable to Safety and Licensing and give an overview of means of compliance*	*II*
SARs	*To set forth instructions for the preparation, review, issuance, and revision of SARs. In addition, the means of handling questions is detailed*	*III, V, and VI*
Open licensing issues (OLIs)	*To set forth instructions for identifying, assessing, and communicating the the status of open licensing issues to achieve problem resolution*	
Regulations, codes, and industrial standards	*To identify responsibility for preparation of fundamental safety criteria to assist in the design process by assuring conformance with NRC Safety and Licensing requirements*	*III*
Review of designs	*To set forth instructions for review of design by Safety and Licensing Groups*	*III*
WCAP preparation	*To establish guidelines within Safety and Licensing for the handling of topical reports*	*VI*
EMD Quality Assurance Manual	*To set forth specific quality instruction for implementing the quality program*	

TABLE 17C-3 (SHEET 8 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Manual and administration	*To define the Quality Program Manual and its administration; e.g., scope and format of the manual itself; the provisions for issuance, distribution, and maintenance; approval; and applicability*	*II*
Quality organization and organization charts	*To describe the organizational position of the Quality Assurance Department within the Electro-Mechanical Division and the manner in which the responsibilities and authority of the Quality Assurance Department are implemented by its internal organization*	*I*
Quality planning	*To identify total contract quality requirements as early as possible after order receipt and to assure that actions are undertaken in a timely and organized manner*	*III, IV, and V*
Integrated manufacturing and quality planning instructions	*To describe the program to assure that the desired product quality is maintained by clear and complete instructions of a type appropriate to the circumstances*	*V*
Quality control records	*To outline the system implemented to generate, compile, and store those records necessary to be maintained as objective evidence of compliance*	*XVII*
Equipment control and calibration	*To establish necessary controls for the periodic inspection and calibration of all measuring and testing equipment used for the acceptance of quality*	
Control of purchases	*To define the procedures for controlling the quality of vendor-furnished deliverable materials*	*IV and VII*
Inprocess and final inspection and testing	*To establish necessary controls covering inprocess inspection of materials, parts, and components, including final inspection and testing of complete products*	*X and XI*
Corrective action for nonconforming material	*To establish a procedure for insuring timely corrective action for nonconforming material*	*XVI*

TABLE 17C-3 (SHEET 9 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Control of documents and changes	*To define the procedure for release and control of drawings and specifications and document status reporting; to assure that proper document revisions are used by manufacturing and inspection and provided to the authorized inspector; to provide and effectively maintain shop order documents; to establish a uniform method of processing engineering changes and supplementary information in order to evaluate and control technical aspects; and to provide availability of previous and current revisions of documents compatible to Westinghouse EMD product lines*	*VI*
Handling, storage, and delivery	*To provide the work and inspection instructions for handling, storage preservation, packaging, and shipping to protect the quality of products and prevent damage, loss, deterioration, degradation, or substitution of products*	*XIII*
Control of nonconforming material	*To establish a procedure for the identification, segregation, and disposition of material that does not conform to the requirements; to provide a means of analyzing the causes of nonconformance so that corrective action can be taken to prevent recurrence; and to provide immediate solutions to problems generated by manufacturing processes during the course of manufacture*	*XV*
Inspection status indicator	*To define the methods used to show the inspection status of products*	*XIV*
Personnel certification for nondestructive testing	*To establish the practice for certifying personnel to perform nondestructive testing duties at Westinghouse Electro-Mechanical Division*	*IX*
Design control	*To outline the control system implemented for review involving designing, design engineering development, and design interface activities*	*III*
Manufacturing control (identification of materials, parts, and components)	*To describe the methods for identification, control, and traceability of materials, parts, and components through all stages of processing*	*VIII*
Audits	*To establish the procedure for audits to insure compliance with all aspects of the quality assurance program and to determine effectiveness of the program*	*XVIII*
EMD Purchasing Department Manual	*To detail responsibilities, policies, and procedures of the Purchasing Department for the control of purchased materials, equipment, and services*	*VII*

TABLE 17C-3 (SHEET 10 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
EMD Engineering Department Instructions	*To establish procedures for design control, including such elements as reviews, verifications of adequacy, tests, approvals, releases, and control of changes*	*III*
Pensacola Quality Program Manual	*To set forth specific instructions for implementing the quality program*	
General	*To provide a comprehensive overview of the quality system utilized at Westinghouse Pensacola Division, defining organization, responsibilities, and authority*	*I, II, and V*
Contract control	*To describe the system utilized for dissemination of current contractual information to all affected internal departments*	*III and VI*
Design drawing and design specification control system	*To define the system for release and control of design drawing and specifications to provide the proper document revisions to the authorized inspector*	*III, V, and VI*
Control of purchase	*To define the activities associated with control of purchase, including requisition review, supplier evaluation, approval requests, quality release, material rejection, and corrective action*	*IV, VII, XV, and XVI*
Manufacturing and quality assurance planning	*To define the routine actions of manufacturing and quality engineering in planning for the acquisition, manufacture, and inspection of material, components, or assemblies*	*II, V, VIII, XI, X, and XIV*
Manufacture control	*To define the system for control of the manufacturing process*	*VIII, IX, X, and XIV*
Inspection	*To define the activities required to inspect and evaluate results, and to document and release manufactured items of further processing or assembly*	*X and XVI*
Welding	*To define the procedure to be used to assure that all welds made on products in process of manufacture are made per qualified procedures by qualified welders using properly controlled and traceable filler metal*	*IX*
Nondestructive examination, qualification, and control	*To define the methods for qualifying, certifying, and controlling nondestructive examination personnel, equipment, and procedures*	*IX and XI*

TABLE 17C-3 (SHEET 11 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Gauge and instrument control	*To provide "off-the-shelf" availability of reliable calibrated inspection instruments and gages, and to maintain records of gage location and calibration status*	*XII*
Special process control and heat treating	*To describe control of special processes*	*IX*
Nonconforming material control	*To describe the nonconforming review and documentation system*	*XV*
Corrective action	*To define the corrective action program*	*XVI*
Internal quality audits	*To describe the system for audit of conformance to internal procedures and processes*	*XVIII*
Quality documentation	*To define the product documentation necessary and the method of data retrieval*	*XVII*
Customer witness points	*To describe the system utilized to assure customer notification prior to occurrence of contracted witness points*	*II and X*
Tampa Standard Division Procedure	*To set forth standard division procedures*	
Drawing control	*To control the release and distribution of drawings and corresponding change notices prior to and during the manufacturing cycle*	*III, V, and VI*
Material, finish, and process specification handling	*To request new and/or revised specifications for material, finish, process, and manufacturing procedures. Procedure includes the control for issuance and distribution*	*III, V, and VI*
Preparing manufacturing lineups	*To prepare and review new and revised manufacturing lineups*	*V, VI, and X*
Procedure for ordering feeders	*To control and release manufacturing operational lineups from which inspection point programs are generated*	*V and VI*
Preparation of emergency manufacturing information	*To prepare and control manufacturing information released to Manufacturing for one-time job activities*	*V*
Welder qualification procedure	*To control recording and reviewing welder qualifications for code compliance*	*V and IX*

TABLE 17C-3 (SHEET 12 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Error appraisal notification (EAN) procedure	*To control nonconforming material*	*III, V, VI, VII, and XV*
Receiving and inspection procedure	*To control receipt and inspection of material*	*IV, V, VII, VIII, and X*
Contingency EANs	*To control open EANs at the time of shipment on quality release forms that must be satisfied and cleared at the field sites*	*V and IX*
Purchase order procedure	*To prepare and issue instructions for material and services*	*IV and V*
G-letter procedure	*To expedite drawing changes for a specific customer order only and to incorporate drawings into units when time does not permit waiting for drawing change and release by the standard procedure*	*III, V, VI, and XV*
Equipment specifications	*To maintain design control of requirements specified in the equipment specification and to insure that all design requirements are incorporated into the product*	*III*
Design codes and addenda	*To maintain design control of all requirements in ASME Code Section III and to assure that all design requirements are incorporated into the product design drawings and documents within six months after issue date*	*III*
Design reviews and checklists	*To conduct design reviews*	*III*
Stress report compliance with hardware	*To insure that the final stress report is reviewed by the product design section in which design originated and wherein all changes and variations are recorded*	*III*
Tampa Quality Assurance Manual	*This manual has been prepared for the Nuclear Energy Systems, Large Components Division, Tampa Division by the Quality Assurance Section; it provides information on procedures, systems, and activities performed by Quality Assurance personnel*	
Quality assurance control program	*To provide a positive system for controlling the quality of nuclear vessel products supplied by the Tampa Division of Westinghouse Electric Corporation*	*II*
Applicability	*To establish applicability of the Quality Assurance Manual*	*II*

TABLE 17C-3 (SHEET 13 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Design control	*To define the Quality Assurance function in design and design change control*	*III*
Quality assurance authority and administration of manual	*To set forth the authority and responsibility for administration of the manual*	*I and VI*
Organization and responsibilities	*To define the responsibilities and functions of the Quality Assurance Department*	*I*
Quality assurance planning	*To establish the preplanning requirements for specification review and inspection point plan development by Quality Assurance*	*VI, X, and XIV*
Control of purchased material	*To establish Quality Assurance requirements associated with ordering information, purchase order (PO) inspection codes, Quality Assurance review of POs, supplier evaluations and audits, source inspection, and receiving inspection*	*IV, VI, VII, and VIII*
Inspections, examinations, nondestructive examinations, and process verification	*To set forth instruction for in-process inspection, patrol inspection, final inspection, shipping inspection, nondestructive examination, post-weld heat-treat verification, and the development of quality assurance instructions*	*IX, X, XI, and XIII*
Control of nonconforming operations and/or material	*To establish the guidelines for writing and handling EANs that were initiated by Westinghouse Tampa personnel performing in-house inspections and by Westinghouse field representatives performing either source inspections at supplier's plant or assembly and test inspections at customer's plant sites*	*XV*
Corrective action program	*To set forth the procedure and responsibility for initiating positive corrective action*	*XVI*
Control of specifications and drawings	*To establish procedures to assure that specifications and drawings used are current and that outdated, nonapplicable specifications are removed and destroyed*	*V*
Welding and weld rod control	*To define welding controls which are established to provide assurance and objective evidence that all welding operations performed during the manufacturing cycle adhere to and conform with engineering drawings, process specifications, and customer and and code requirements*	*IX and XIII*

TABLE 17C-3 (SHEET 14 OF 15)

Procedure	Purpose	10 CFR 50, Appendix B Criteria
Personnel qualification	To establish the Tampa program for qualification of nondestructive testing and examination personnel	X, XIV, VIII, and XVIII
Technician's stamp control	To specify the method of assurance that all rubber stamps used by Quality Assurance personnel are recorded and identified to specific individuals	X, XIV, and XVII
Material handling	To define the material handling program and inspection of lifting equipment	XII
Tool and gauge control	To define the calibration frequency and maintenance for gauges and equipment used by the Quality Assurance Department and to establish responsibility for compliance to the procedure	XII
Objective evidence of quality	To define the records program	VIII and XVII
Quality auditing	To set forth the instructions for the quality auditing program	XVIII
SMD Quality Assurance Manual		
Organization	To define the functions of the various components of the quality control organization	I
Administration of product quality control	To establish the quality planning and order processing procedure	II
Inspection and test philosophy	To establish responsibility for all phases of inspection and test	X and XI
Supplier and material quality assurance	To establish procedures for purchase requisition control and vendor evaluations	IV, V, and VII
Material identification	To describe the system used by Quality Control in identifying and controlling material throughout the entire manufacturing cycle	VIII
Process control and inprocess inspection	To set forth instructions for process control and inprocess inspection	X
Special processes	To establish special process controls and nondestructive test requirements and qualifications	IX

TABLE 17C-3 (SHEET 15 OF 15)

Procedure	*Purpose*	*10 CFR 50, Appendix B Criteria*
Statistical planning and application	*To define statistical methods and techniques utilized for sampling inspection*	*X*
Records and data system	*To set forth the records program for maintaining objective evidence of quality*	*XVII*
Drawing, specification, change review, and control system	*To define the measure for document control including changes*	*V and VI*
Control of nonconforming material	*To describe the procedure used in identifying and controlling all defective material detected at receiving, or in any stage of processing, and to assure that corrective action is taken*	*XV and XVI*
Quality audit system	*To establish responsibilities and instructions for audits*	*XVIII*
Use and control of Inspection stamps	*To set forth instructions for the use and control of inspection stamps*	*XIV*
Control laboratories	*To establish responsibilities and instructions for performing tests on raw materials and intermediate and final product*	*XI*
Gauge control	*To establish measures for control of measuring and test equipment*	*XII]*

[HISTORICAL] [TABLE 17C-4

RECORDS RETENTION

All documents and records to be kept as a part of the Westinghouse program for all safety-related and additional selected major components are listed in this table.

Document Description	*Primary Retention Responsibility*
Equipment specifications	*Engineering*
Process specifications	*Engineering*
Instrumentation and control standards	*Engineering*
Drawings	*Central files, Drafting, and Reproduction*
Design review documentation	*Quality Assurance*
Purchase order, change notices, and procurement advisory releases	*Purchasing*
Equipment prototype test data, design calculations, and stress reports	*Engineering*
Supplier-fabricated equipment	
Supplier fabricating, testing, and nondestructive testing procedures	*Engineering when NES approval is required; supplier when NES approval is not required*[a]
Radiographs	*Supplier*[a]
Testing equipment calibration records and personnel qualifications	*Supplier*[a]
Mill test reports, final acceptance inspection records, special process records, and performance test records	*Supplier*[a] *and Quality Assurance*
Quality releases, code forms, inspection plans, and nameplate rub-offs	*Quality Assurance*
QA trip reports, QC plans, deviation notices, and supplier survey reports	*Quality Assurance*
Technical manual, instruction book, and spare-part forms	*Engineering*
Field deficiency reports	*Construction and Services*

a. *Quality Assurance has the responsibility for reviewing documentation maintained by the supplier and assuring that the supplier has adequate facilities for long-term retention. Purchasing has the responsibility for enforcing the contractual retention and retrieval requirements.]*

[HISTORICAL] [TABLE 17C-5

TYPICAL DATA (RETAINED BY WESTINGHOUSE) FOR REPRESENTATIVE COMPONENTS

Data Type	*Steam Generator*	*R.C. Piping*	*R.C.Fittings*	*Letdown Hx*	*Seal Water Injection Filter*	*Accumulator*	*Safety Injection Pump*	*Boron Injection Tank*	*R.C. Loop Stop Valve*	*Pressurizer Relief Valve*	*Waste Evaporator Feed Pump*	*HD Adapter Plugs*
Quality release	X	X	X	X	X	X	X	X	X	X	X	X
Inspection checklist	X	X	X	X	X	X	X	X	X			
Pressure envelope material certifications	X	X	X	X	X	X	X	X	X	X	X	X
RT records	X	X	X	X	X	X	X	X	X	X		
PT/MT records	X	X	X	X	X	X	X	X	X	X	X	X
UT records	X	X				X		X	X	X		X
Dimensional records	X	X	X	X	X	X	X	X	X			X
Performance test data							X		X	X	X	
Code form	X	X		X	X	X		X				
Heat-treat records	X	X	X			X	X	X	X			
Trip report summary sheet		X	X	X	X	X	X	X	X			
Nameplate rub-off	X			X	X	X		X				
Supplier certificate of compliance				X	X	X			X			
Pressure test certificate				X	X	X		X	X	X	X]	

[HISTORICAL] [TABLE 17C-6

EXAMPLE FROM TYPICAL SHOP ORDER LOGIC FLOW DIAGRAM

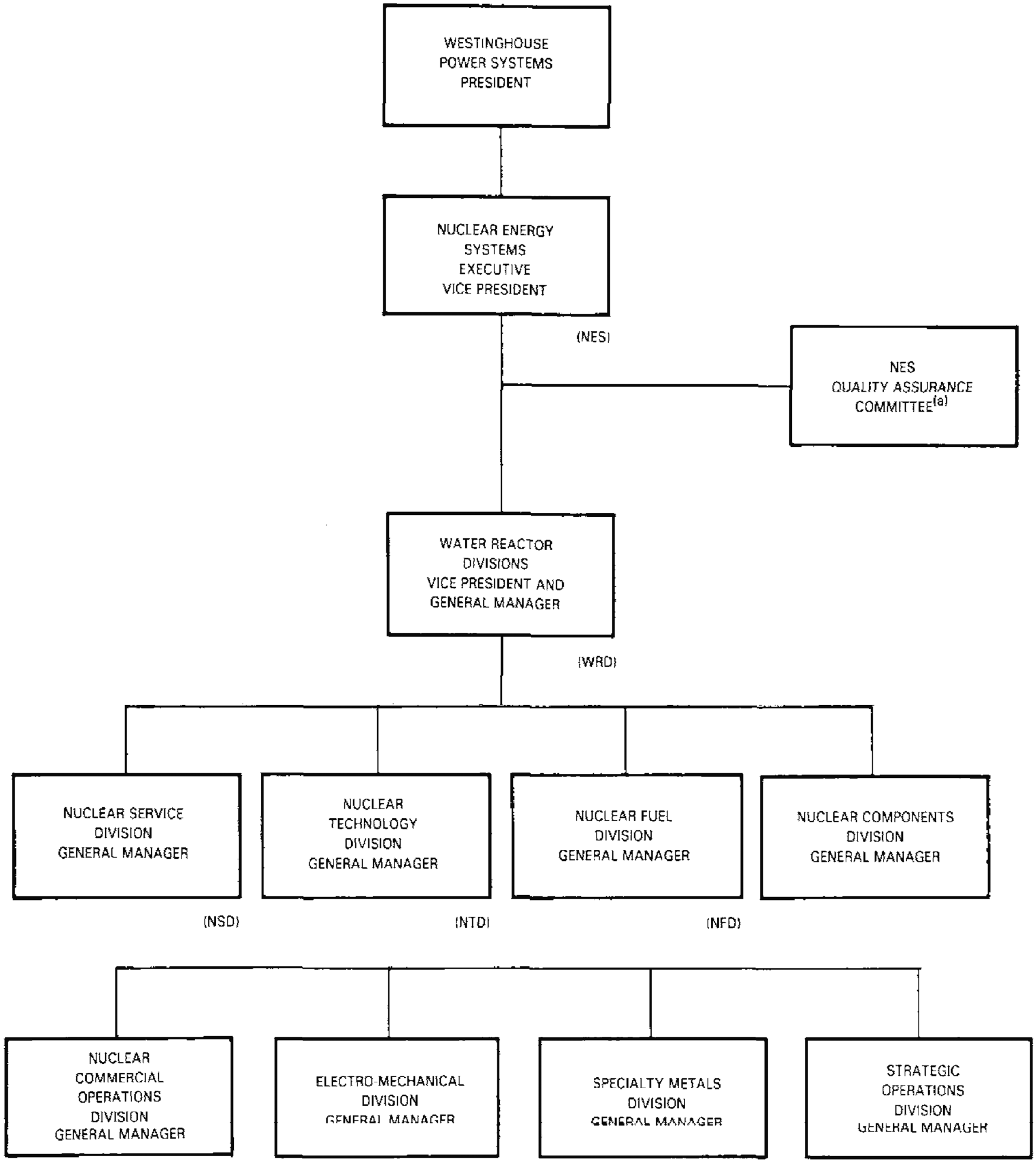
Shop Order Standard | *Page 7 of 14*

ACTIVITY DESCRIPTION	CUSTOMER	PROJECTS	OTHER DEPTS & FUNCTIONS	SHOP ORDER DEPARTMENT	PURCHASING	SUPPLIER	QUALITY ASSURANCE		FILE	10CFR50-B CRITERIA	APPLICABLE DOCUMENT
49. Supplier submits drawings, procedures, calculations and other submittals as required						49. Issue Submittals				III -DSGN Control IV -Procurement Doc. Control V -Instructions, Proc. & DWGS VI -Doc. Control VII -Control of Purch. Atl., Equipment & Services	
50. Purchasing processes supplier submittals					50. Process Submittals					IV -Procurement Doc. Control	Purchasing Procedure C.2.8 -Submittals & Distribution of Vendor DWGS & Documents
51. Purchasing forwards drawings to Drawing Control & other submittals to Engineering.					51. Issue Submittals to ENG					IV -Procurement Doc. Control VII -Control of Pur. Mtl., Equipment & Services	Purchasing Procedure C.2.8 -Submittals & Distribution of Vendor Drawings & Documents
52. Engineering Shop Order Department reviews submittals				52. Review Submittals						III -DSGN Control IV -Procurement Doc. Control	EI 23: Processing APRVL Req. Forms – Vendor Doc. Submittals EI 29: DSGN Control & Documentation
53. Engineering Shop Order Department issues submittals for review by applicable departments.				53. Issue Submittals for Review						III -DSGN Control IV -Procurement Doc. Control	EI 23: Processing APRVL Req. Forms – Vendor Doc. Submittals EI 29: DSGN Control & Documentation
54. Applicable departments review submittals.			54. Review Submittals				54. Review Submittals			III -DSGN Control IV -Procurement Doc. Control	EI 23: Processing APRVL Req. Forms – Vendor Doc. Submittals EI 29: DSGN Control & Documentation
55. Applicable departments issue comments on submittals to Engineering Shop Order Department.			55. Issue Comments				55. Issue Comments			III -DSGN Control IV -Procurement Doc. Control	EI 23: Processing APRVL Req. Forms – Vendor Doc. Submittals EI 29: DSGN Control & Documentation
56. Engineering Shop Order Department resolves comments.				56. Resolve Comments						III -DSGN Control IV -Procurement Doc. Control	EI 23: Processing APRVL Req. Forms – Vendor Doc. Submittals
57. Engineering Shop Order Department documents resolution, distributes resolution and files.			57. ENG Use	57. Issue Resolution of Comments			57. QA Use		57. PA	III -DSGN Control IV -Procurement Doc. Control	EI 23: APRVL Req. Forms – Vendor Doc. Submittals
58 Engineering Shop Order Department reviews and signs off submittals.				58. Review & Signoff						III -DSGN Control IV -Procurement Doc. Control	EI 23: APRVL Req. Forms – Vendor Doc. Submittals]

[HISTORICAL] [TABLE 17C-7

[EXAMPLE FROM TYPICAL SHOP ORDER LOGIC FLOW DIAGRAM

							Shop Order Standard				*Page 8 of 14*
ACTIVITY DESCRIPTION	*CUSTOMER*	*PROJECTS*	*OTHER DEPTS & FUNCTIONS*	*SHOP ORDER DEPARTMENT*	*PURCHASING*	*SUPPLIER*	*QUALITY ASSURANCE*		*FILE*	*10CFR50-B CRITERIA*	*APPLICABLE DOCUMENT*
59. Engineering Shop Order Department issues PO/CN or PAR to purchasing as required.				*59. Issue PAR*					*59. PA*	*III -DSGN Control* *IV -Procurement Doc. Control*	*Purchasing Procedure* *2.2: Writing PO & CN's* *2.3: Completing general order & GO/CN* *2.4: Multiplant purchase orders* *2.6: Procurement advisory release* *EI 17:* *ENGG processing of purchase reqs & change notices*
60. Purchasing issues PO/CN to supplier. Supplier acknowledges, revises, and resubmits as required. Recycle to step 49 as necessary.					*60. Prepare PAR* *60. Issue PAR*	*60. Supp. Use*				*III -DSGN Control* *IV -Procurement Doc. Control*	*Purchasing Procedure* *2.2: Writing PO & CN's* *2.3: Completing general order & GO/CN* *2.4: Multiplant purchase orders* *2.6: Procurement advisory release* *2.7: Acknowledgements*
61. Engineering Shop Order Department prepares release to fabricate via PAR.				*61. Prepare PAR to Fabricate*						*IV -Procurement Doc. Control* *VI -Document CTRL* *VII -Control of Purchased Materials, Equipment & Services*	*Purchasing Procedure* *2.6: Procurement advisory release*
62. Engineering Shop Order Department issues PAR to purchasing.				*62. Issue PAR to Purchasing*					*62. PA*	*IV -Procurement Doc. Control* *VI -Document CTRL* *VII -Control of Purchased Materials, Equipment & Services*	*Purchasing Procedure* *2.6: Procurement advisory release*
63. Purchasing processes PAR.					*63. Process PAR*					*IV -Procurement Doc. Control* *VI -Document CTRL* *VII -Control of Purchased Materials & Services*	*Purchasing Procedure* *2.6: Procurement advisory release*
64. Purchasing issues PAR to supplier.					*64. Issue PAR to Supplier*	*64. Supp. Use*				*IV -Procurement Doc. Control* *VI -Document CTRL* *VII -Control of Purchased Materia, & Services*	*Purchasing Procedure* *2.6: Procurement advisory release*
65. Engineering Shop Order Department issues approved vendor data, as applicable, to customer via projects.		*65. CUST Use*	*65. Issue Data to Customer*	*65. Issue Data to Projects*						*III -DSGN Control* *VI -Document CNTRL*	*EI 29:* *DSGN Control & Documentation]*



a) THE NES QUALITY ASSURANCE COMMITTEE IS COMPOSED OF THE QUALITY ASSURANCE AND RELIABILITY MANAGERS FROM EACH OF THE NES DIVISIONS. THE COMMITTEE'S CHAIRMAN IS THE NTD PRODUCT ASSURANCE MANAGER.

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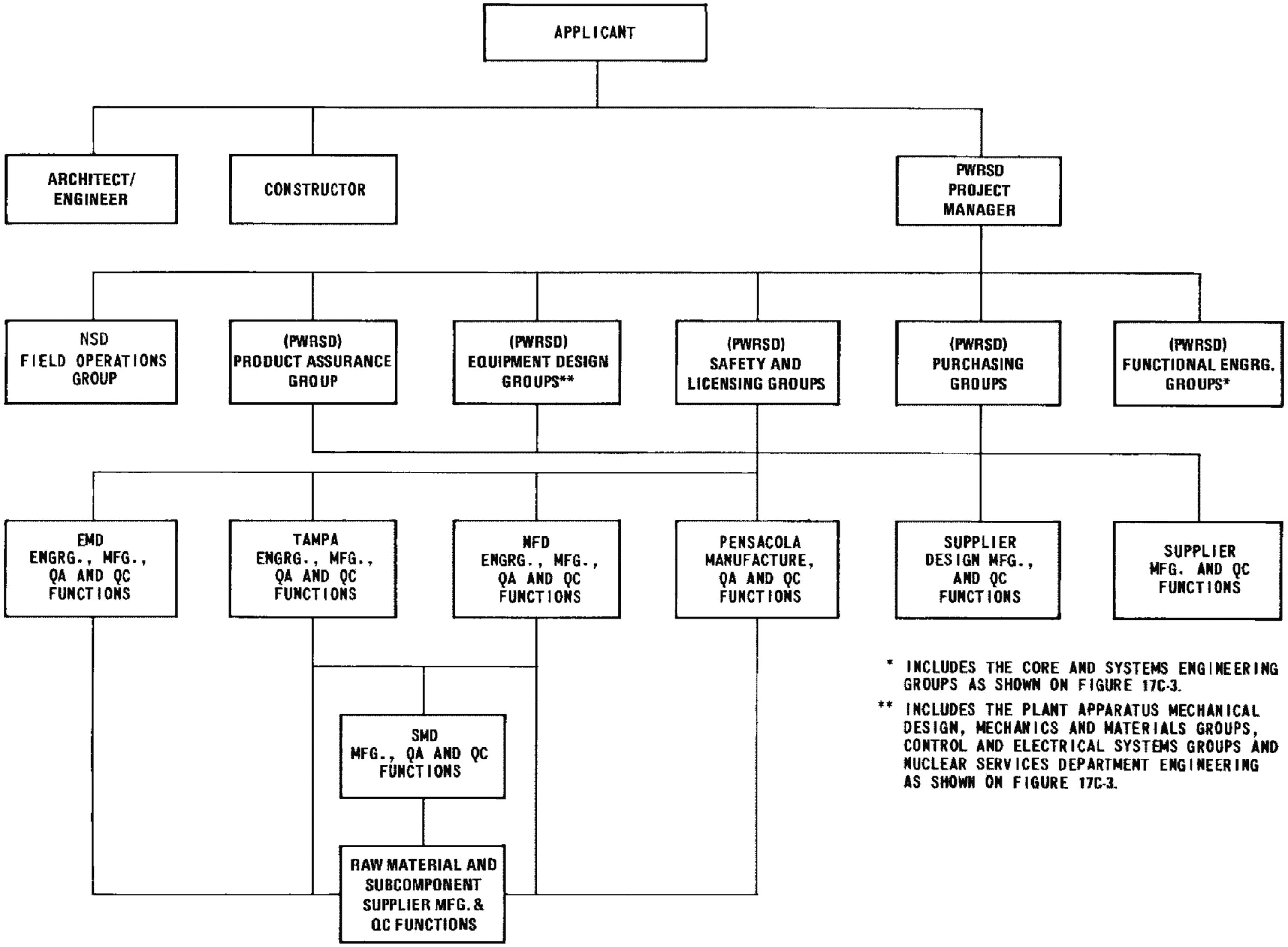

* INCLUDES THE CORE AND SYSTEMS ENGINEERING GROUPS AS SHOWN ON FIGURE 17C-3.

** INCLUDES THE PLANT APPARATUS MECHANICAL DESIGN, MECHANICS AND MATERIALS GROUPS, CONTROL AND ELECTRICAL SYSTEMS GROUPS AND NUCLEAR SERVICES DEPARTMENT ENGINEERING AS SHOWN ON FIGURE 17C-3.

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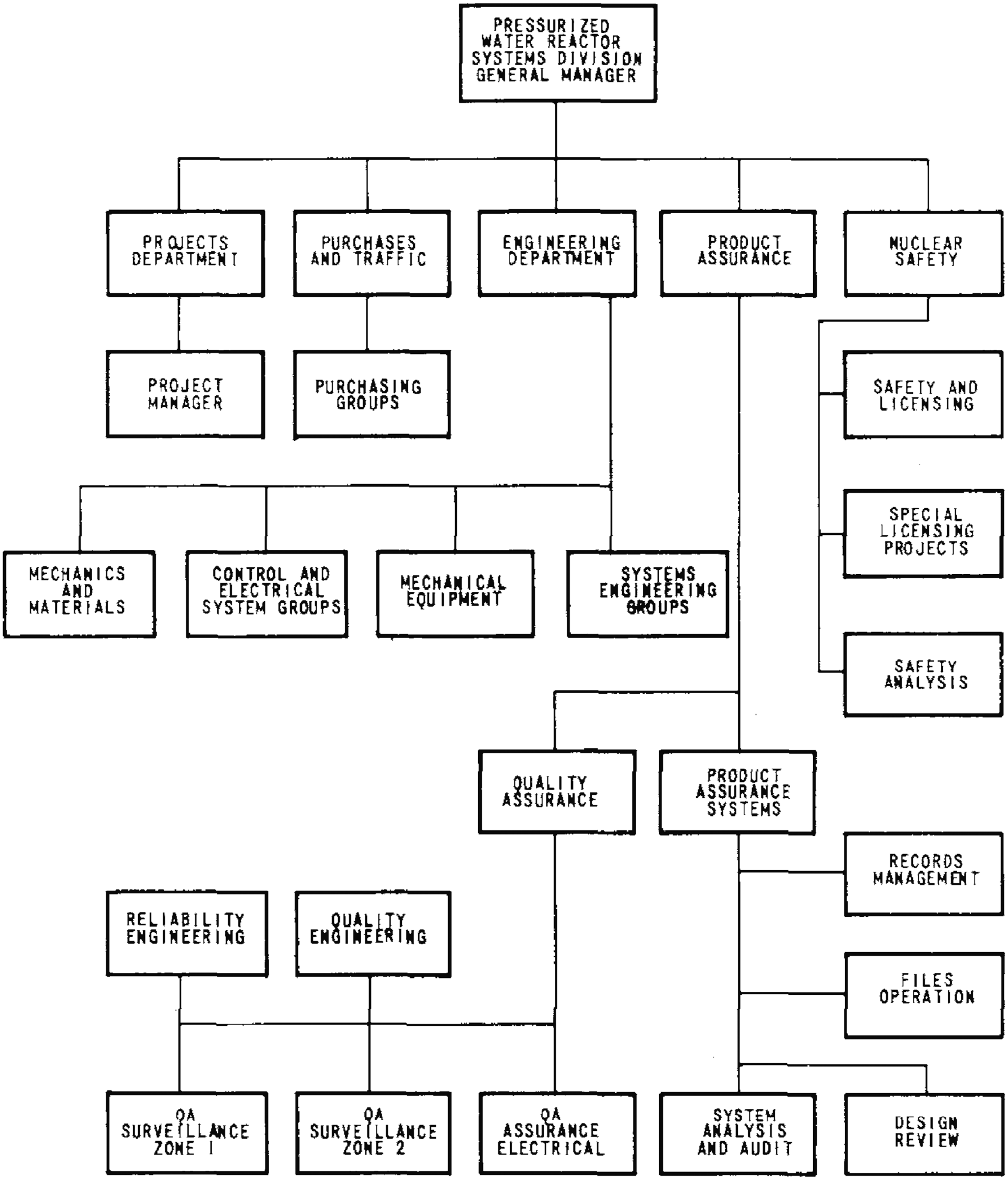
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[NSSS FUNCTIONAL RELATIONSHIP CHART

FIGURE 17C-2]



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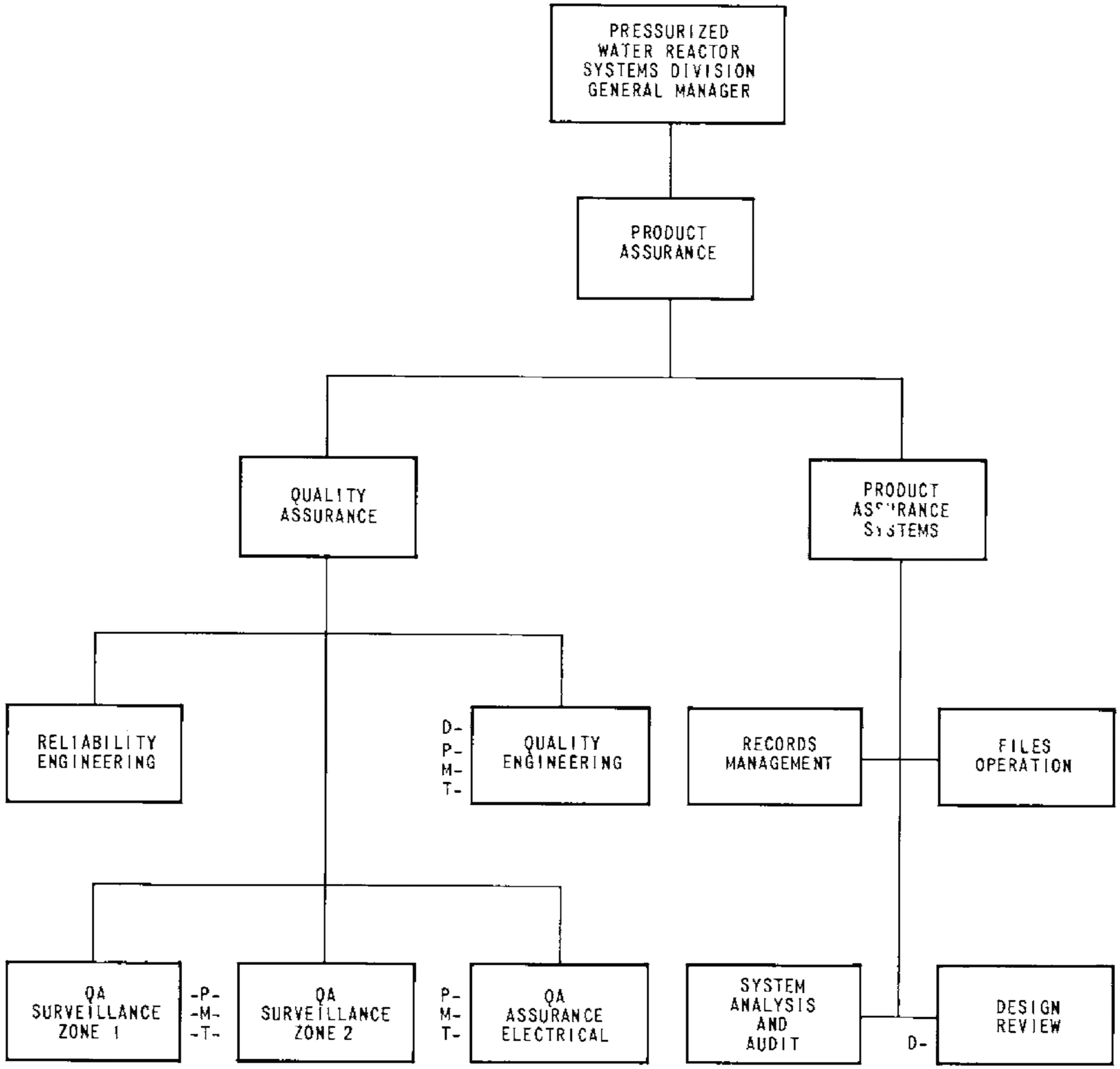
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PWRSD QUALITY ASSURANCE PROGRAM ORGANIZATION

FIGURE 17C-3]



LEGEND:

D - DESIGN
P - PROCUREMENT
M - MANUFACTURING
T - TESTING

REV 21 5/08

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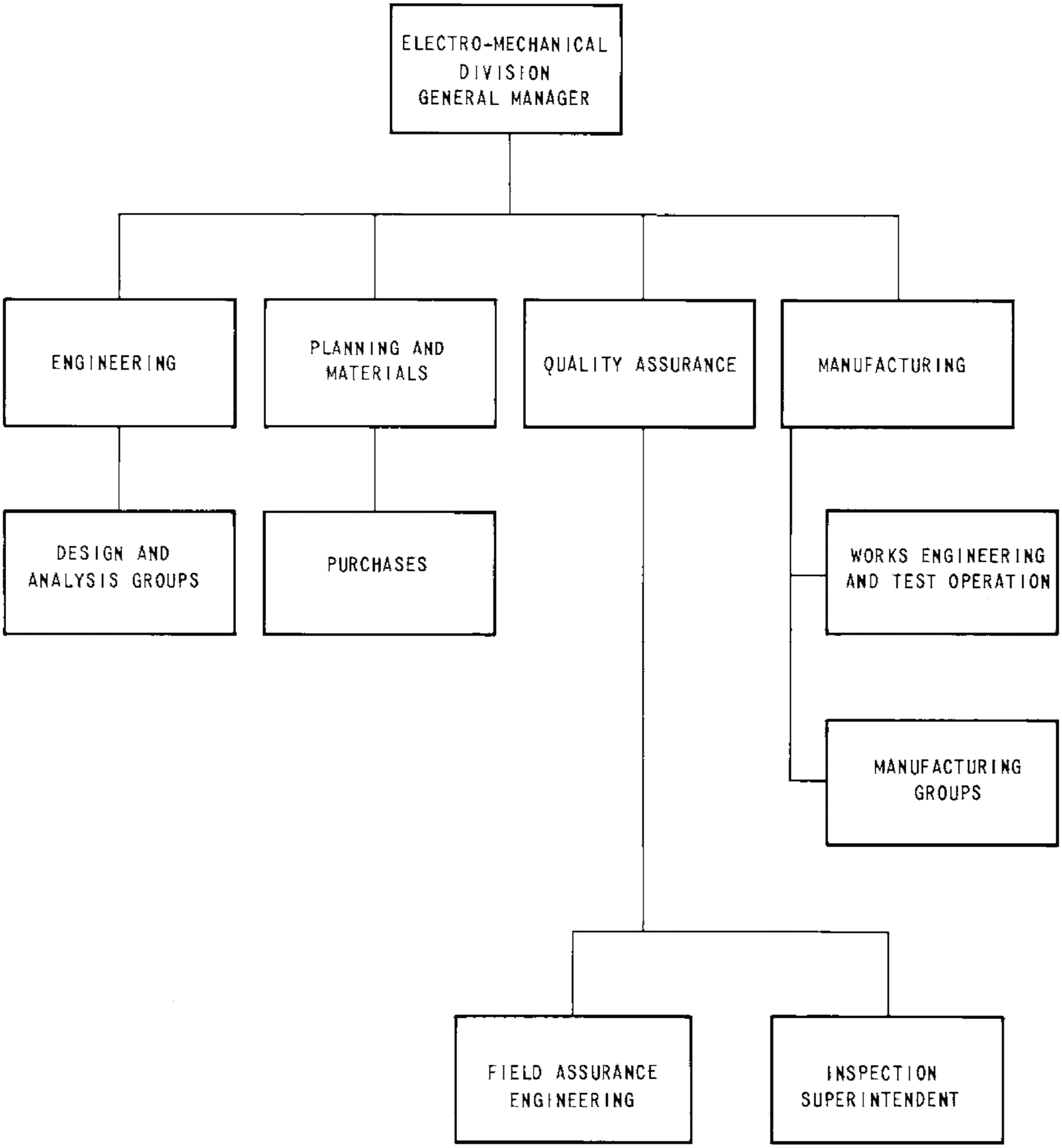
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PWRSD PRODUCT ASSURANCE QA-RELATED FUNCTIONS

FIGURE 17C-4]



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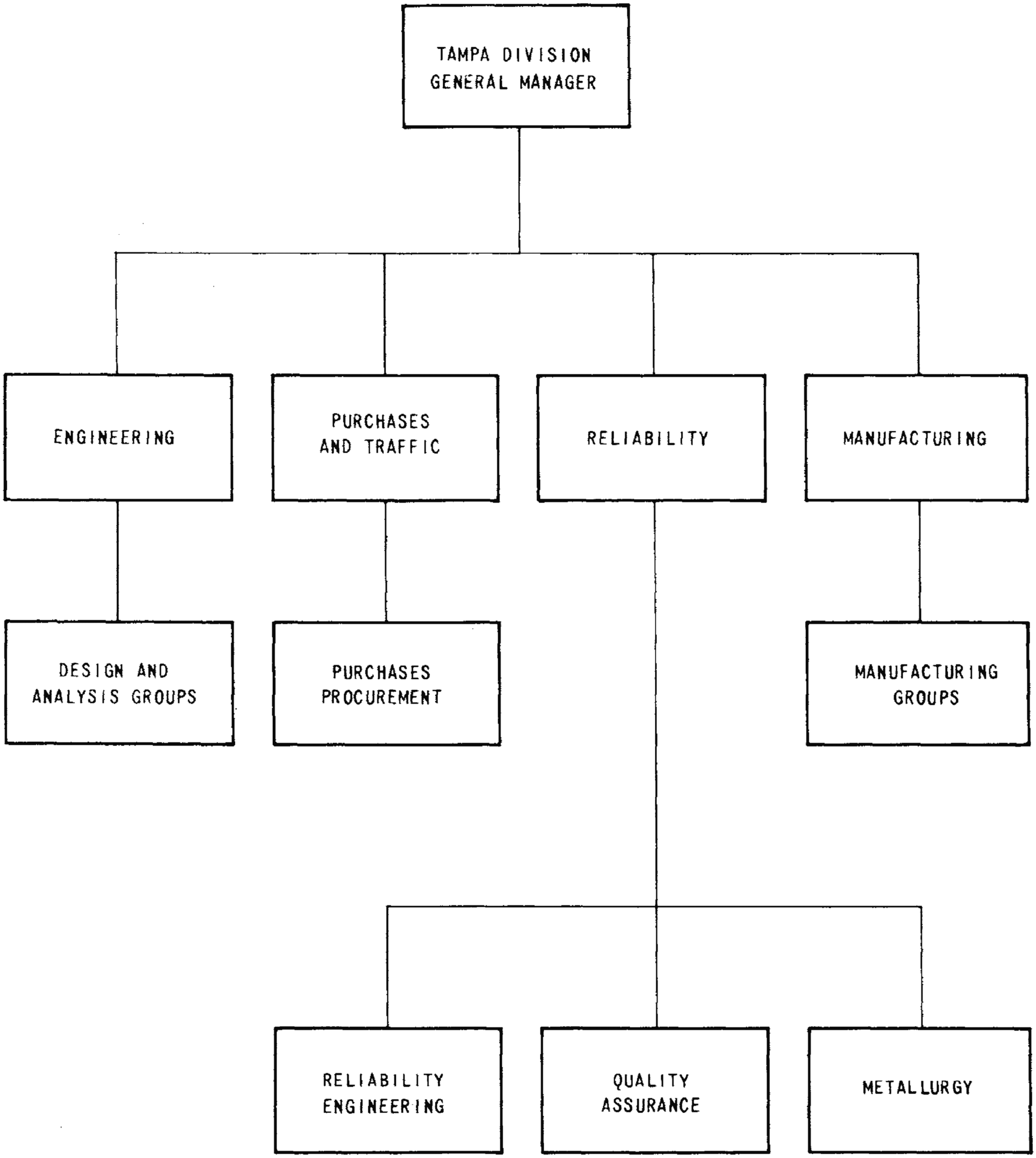
UNIT 1 AND UNIT 2

*[ELECTRO-MECHANICAL DIVISION
QUALITY ASSURANCE PROGRAM
ORGANIZATION*

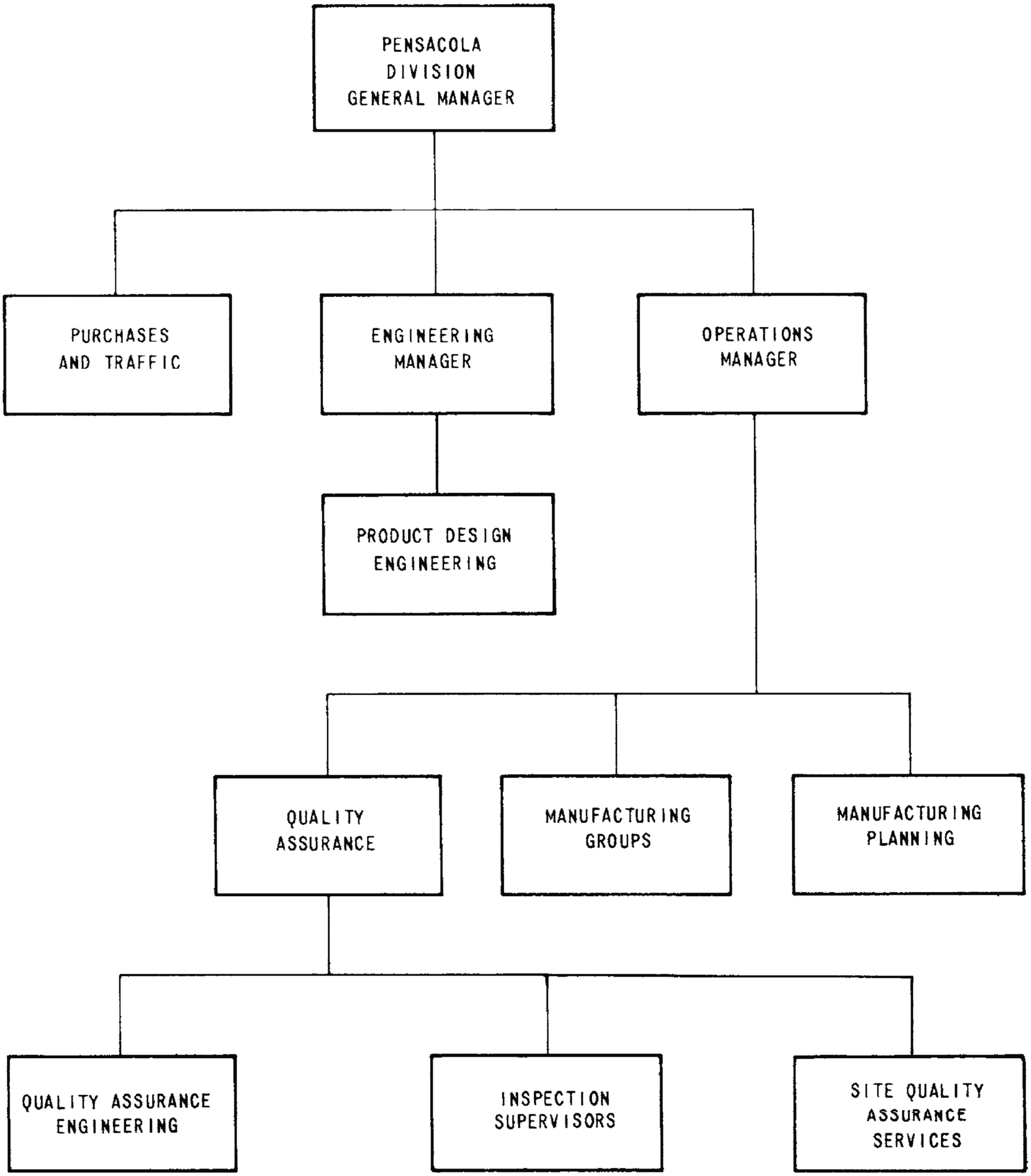
FIGURE 17C-5]



REV 21 5/08

*[TAMPA DIVISION QUALITY
ASSURANCE PROGRAM ORGANIZATION*

FIGURE 17C-6]



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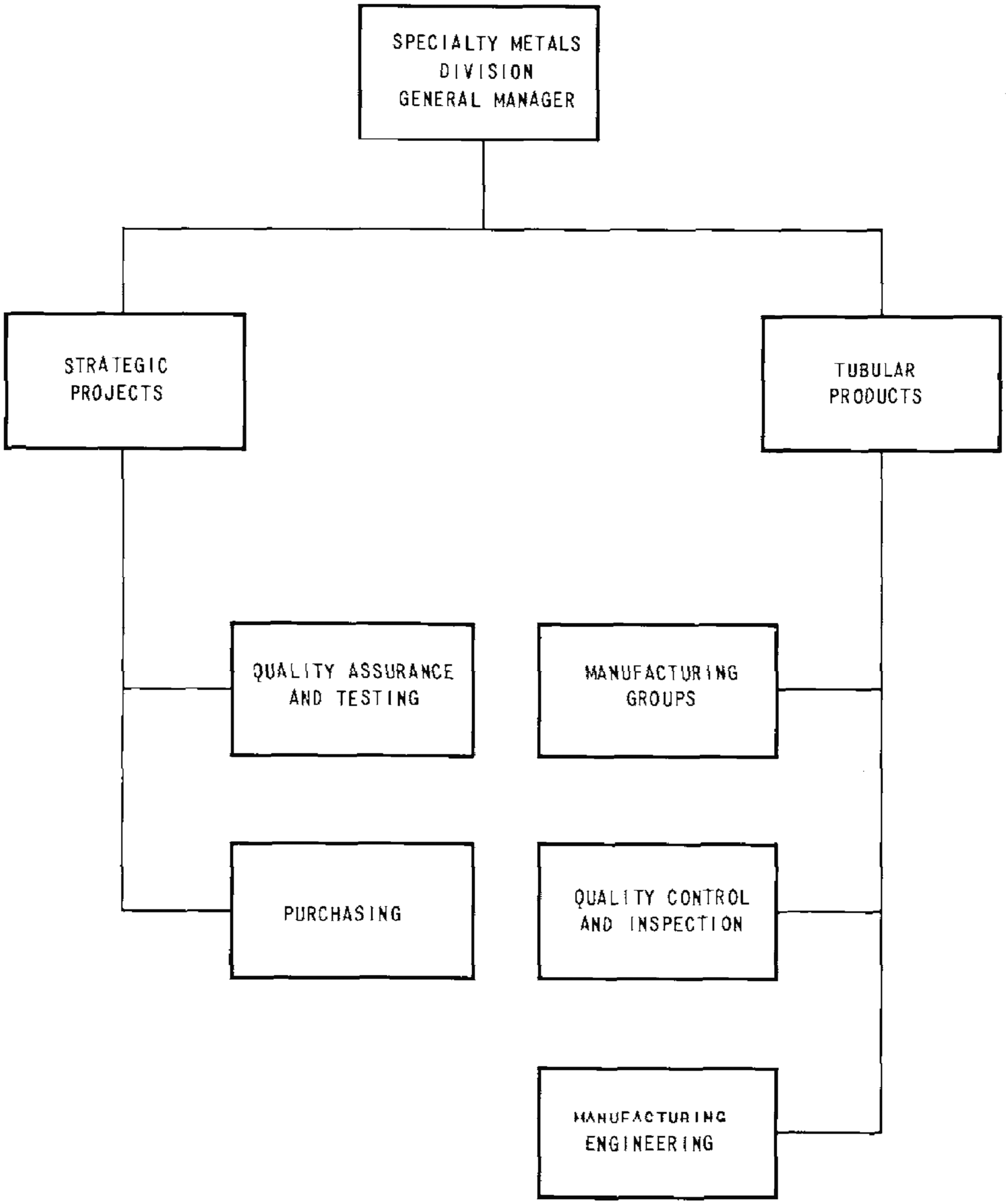
UNIT 1 AND UNIT 2

*[PENSACOLA DIVISION
QUALITY ASSURANCE
PROGRAM ORGANIZATION*

FIGURE 17C-7]



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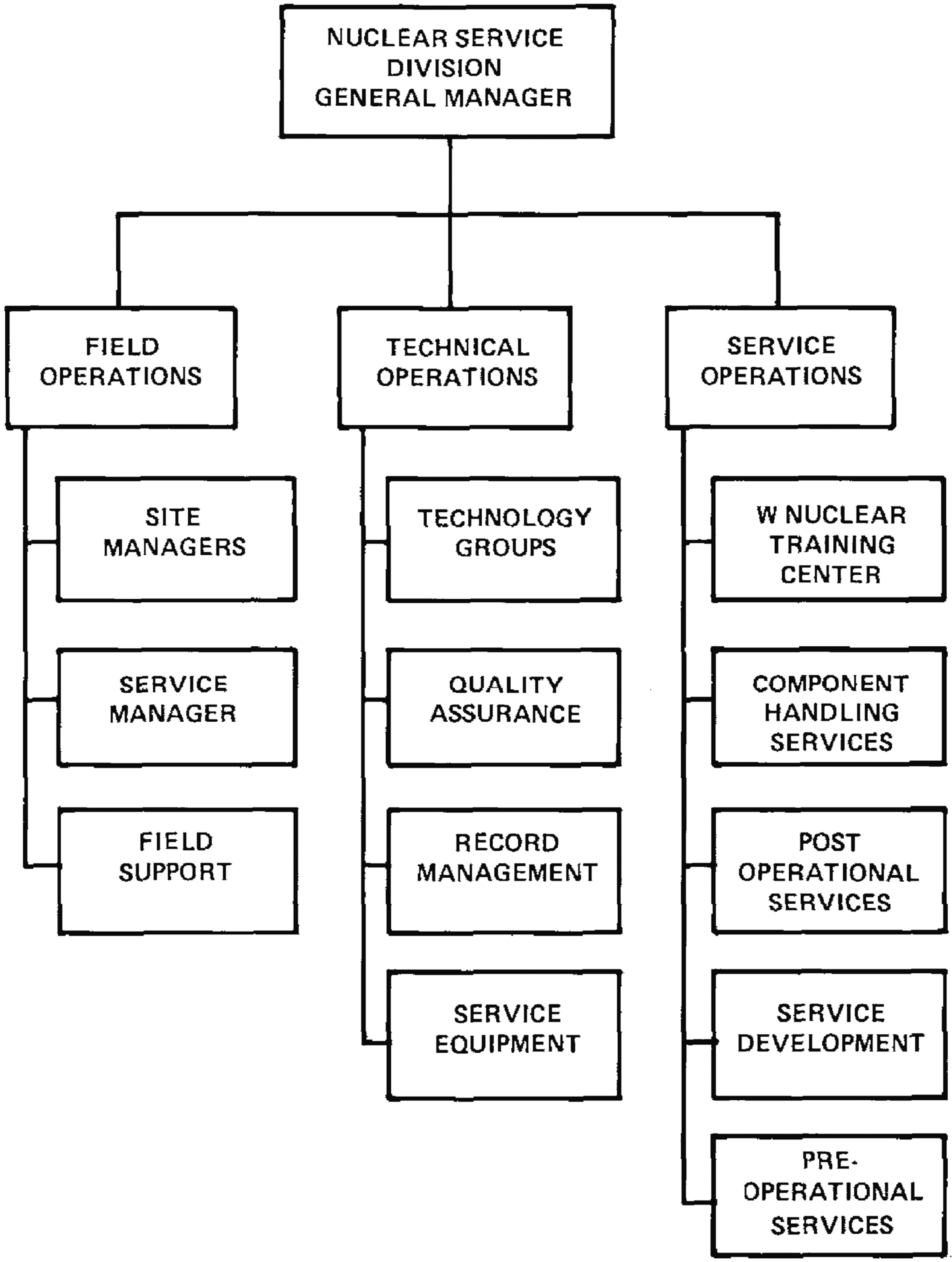
*[SPECIAL METALS DIVISION
QUALITY ASSURANCE
PROGRAM ORGANIZATION*

FIGURE 17C-8]



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[NUCLEAR SERVICE DIVISION QUALITY ASSURANCE PROGRAM ORGANIZATION

FIGURE 17C-9]

APPENDIX D

DANIEL CONSTRUCTION COMPANY OF ALABAMA QUALITY ASSURANCE PROGRAM

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[HISTORICAL]
[APPENDIX 17D

DANIEL CONSTRUCTION COMPANY OF ALABAMA
QUALITY ASSURANCE PROGRAM

Appendix 17D contains historical information about the Daniel Construction Company quality assurance program implemented during design and construction of FNP.

As the plant constructor, Daniel Construction Company of Alabama (Daniel) has developed and is implementing a quality assurance (QA) program to control the quality of all assigned construction, field purchasing, field engineering, and other related services.

The Daniel quality assurance program includes the procedures, instructions, and control actions used by Daniel to assure that field fabrication and construction, purchased materials, and workmanship are controlled to meet all applicable specifications, codes, and regulatory requirements. The program is directed toward, but not limited to, providing the necessary quality control for safety-related structures, systems, and components and for providing quality assurance surveillance and audits to assure that the specified requirements are achieved. The procedural controls and instructions are developed and administered by project discipline management. A separate surveillance and audit activity is provided by the resident project quality assurance manager assigned by the Daniel corporate director - quality assurance.

17D.1.1 ORGANIZATION

The project manager represents Daniel on all matters relating to the Farley Nuclear Plant (FNP) and is fully responsible for the proper implementation and satisfactory operation of the construction QA program.

The project manager has delegated to the quality control (QC) manager responsibility for the development, implementation, and administration of the Daniel quality control program. Figure 17D-1 shows the functional relationships among the individuals and units performing the construction quality control activities. Figure 17D-2 shows the functional relationship of the Daniel corporate director - quality assurance with the project and corporate organization. The corporate director - quality assurance assigns a resident project quality assurance manager (PQAM) to provide a completely independent audit and surveillance of the adequacy and effectiveness of the Daniel quality assurance program.

The independent quality assurance audit within the Daniel organization is in addition to the continuous field surveillance audit being performed by the Alabama Power Company (APC).

The duties and responsibilities of each individual performing a quality control function have been set forth in writing in the Daniel FNP Project Procedure Manual (figures 17D-1 and 17D-2). Referring to figure 17D-1, the development and implementation of the construction QC procedures and instructions are carried out through QC responsibilities assigned to the QC disciplines under the direction of the quality control manager, and through the purchasing agent and the warehouse superintendent under the direction of the services manager. As indicated in figures 17D-1 and 17D-2, the project civil QC engineer, the project mechanical QC engineer, the project welding QC engineer, and the project electrical QC engineer have principal responsibility for quality control in their respective construction disciplines; the purchasing agent and warehouse superintendent have a principal responsibility for quality control related to the procurement, inspection, receiving, storage, preservation, and issuance of material and equipment.

The quality control manager has under his supervision a project civil QC engineer whose principal responsibility and duty is quality control. He is responsible for directing the quality control inspectors and technicians, and all testing and inspection, whether on or off the site or by independent testing laboratories or consultants, for those functions under the jurisdiction of the Civil Discipline.

The quality control manager has under his supervision a project mechanical QC engineer whose principal responsibility and duty is quality control. He is responsible for directing the quality control inspectors and technicians, and all testing and inspections, whether on or off the site or by independent testing laboratories or consultants, for those functions under the jurisdiction of the Mechanical Discipline.

The project electrical QC engineer has the responsibility of quality control of the electrical work. He has under his supervision a group of quality control inspectors and technicians and is responsible for all testing and inspection, whether on or off the site or by independent testing laboratories or consultants, for those functions under the jurisdiction of the Electrical Discipline.

The project welding QC engineer has principal responsibility for the quality of all welding and related control activities of metal fusion processes on the project. He is responsible for the direction of welding inspectors and nondestructive examination (NDE) activities performed or subcontracted by Daniel.

The QC manager also has under his direction the Quality Control Documentation Section. This section is responsible for the filing of records and documents related to quality control. It is also responsible for the completeness of QC documentation.

The inspectors and technicians assigned to the above disciplines will be required to be thoroughly familiar with the approved specifications, drawings, procedures, codes, and other instructions pertaining to their area of responsibility; they shall be suitably qualified for their *assigned responsibilities by training, experience, and test when required. These inspectors and technicians shall provide or initiate the required construction test and inspection documentation.*

A separate Document Control Section reporting to the services manager is responsible for the administration and control of records and documents related to quality control, and for the control and distribution of all approved drawings, specifications, and construction procedures.

The purchasing agent and the warehouse superintendent are functionally responsible for quality control related to the procurement and receipt of material and equipment. However, they rely upon the quality control engineers and the cognizant engineers within the Project Civil, Mechanical, Welding, and Electrical Sections for checking and verifying material and equipment compliance with procurement requirements.

Engineering and QC personnel also perform required vendor or subcontractor preaward reviews and postaward surveillance and audit.

The function of the project quality assurance manager is to provide a completely independent review and evaluation of the adequacy and effectiveness of the construction QC program. He has been provided with sufficient organization freedom to be able to identify quality problems; to initiate, recommend, or provide solutions to quality problems; and to verify corrective action. He has a direct line of responsibility and communication to the Daniel corporate director – quality assurance. The PQAM prepares periodic reports on the status, adequacy, and conformance of the construction QC program with job requirements. The PQAM reports are distributed to Daniel corporate and project management and to APC QA representatives. The PQAM also works closely with APC's manager – quality assurance (MQA) field representatives. This close association, along with good communication between responsible Quality Control and Quality Assurance personnel at all levels, assures a minimum of interface problems and a good understanding of quality control and quality assurance objectives.

17D.1.2 QUALITY ASSURANCE PROGRAM

The objectives of the Daniel quality assurance program are to: develop written procedures and instructions to assure compliance by field work forces and vendors with specified quality requirements and acceptance criteria; select manufacturing facilities or subcontractors that can assure achievement of the required quality levels; and monitor the manufacturing (vendor shops), subcontractors, and field construction activities as appropriate to assure that the established quality level has been achieved.

Quality control procedures required to implement the construction QC program for safety-related structures, systems, and components are considered to meet the intent of the applicable criteria of Appendix B to 10 CFR 50, Quality Assurance Criteria for Nuclear Power Plants.

17D.1.3 DESIGN CONTROL

Design changes originating in the field as field change requests are processed to the owner/designer in accordance with Daniel QC procedure 5.3.2.1A, Field Change Requests. Approved changes, as documented on the field change request or design change notice, are incorporated by the owner/designer in revisions to the original approved design documents. Latest revisions of all design documents are processed at the project by Daniel in accordance with QC procedure 5.3.2.1, Document Control.

17D.1.4 PROCUREMENT DOCUMENT CONTROL

Inquiries and purchase orders for construction materials or plant equipment purchased by Daniel are based on approved procurement specifications provided by APC.

The supplier is held responsible for meeting the requirements of the bid inquiry or purchase order, and is required to possess or develop a QC/QA organization and program with a QC/QA manual of formal policies and practices by which the supplier can assure the control, verification, and record of product quality. Written QC/QA manuals or programs are obtained from bidders when required for review and approval by Daniel, APC, and others (where applicable) prior to placement of purchase orders.

Prospective vendors for Daniel-procured items or services not on the approved qualified bidders' list are surveyed in accordance with Daniel QC procedure 5.2.1, Supplier/Sub-contractor Qualification Survey, prior to being recommended to APC for approval as a qualified bidder.

The supplier has the responsibility for assuring the quality of materials or services in accordance with QC requirements of the purchase order and for assuring that any subvendors also meet all applicable QC requirements of the Daniel purchase order.

A supplier's or subcontractor's adherence to his QC program and to the procurement requirements is audited by Daniel's QC/QA representatives as required; the supplier or subcontractor is also subject to audit by APC QA representatives.

17D.1.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Daniel implements its project QC program by the use of written procedures and instructions which set forth the practices to be followed at the construction project. These procedures are applicable to all field personnel performing work at the plant site. All quality control procedures are developed by the Quality Control Section in accordance with QC procedure 5.1.3, Preparation, Control, and Implementation of Procedures; they are approved by APC's MQA prior to implementation.

Daniel corporate office personnel assist the project organization as requested in developing selected detailed procedures such as welding, heat treating, and NDE procedures.

The PQAM assists Project Engineering QC personnel in developing detailed QC procedures, reviews QC procedures for adequacy, and audits implementation of QC procedures for effectiveness.

17D.1.6 DOCUMENT CONTROL

The Document Control Section is responsible for the administration, control, and filing of records and documents related to quality control, and for the control and distribution of all approved drawings, specifications, and construction procedures, in accordance with QC procedure 5.3.2.1, Document Control.

The Daniel project filing system has been set up in accordance with the total plant numbering system (TPNS) procedure supplied by APC which covers the systems and equipment for the FNP. Quality control and assurance records and other applicable documents are filed by system or area in accordance with the TPNS requirements to assure a complete history record of each system and its components.

A quality control history file will contain a record of inspections, tests, audits, and other documentation required by applicable specifications, codes, standards, and approved procedures. QC personnel generate the project inspection documentation, and review supplier and subcontractor documentation. This documentation is filed by the Quality Control Documentation Section and controlled by the Document Control Section. The Document Control Section also controls approved field use of documents, such as specifications, procedures, and drawings, in accordance with QC procedures for document control, field change requests, and reproduction of drawings to assure that only latest approved revisions are used to control project construction activities.

17D.1.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

The description in paragraph 17.1.5.4, Procurement Document Control, applies to equipment and services subcontractors; i.e., material, equipment, and services must conform to the procurement documents. The supplier's/subcontractor's adherence to his QC program and the procurement requirements is subject to audit by Daniel QC/QA representatives and representatives of APC. Daniel QC personnel develop procedures, checklists, and instructions to assure conformance with the procurement documents. Receiving inspection includes verification of receipt and adequacy of required documentary evidence of conformance with the specified adequacy by the Daniel PQAM and APC's PQAM or his representative(s).

Procured materials are inspected at the FNP site for damage, identification, and conformance to the procurement documents. The receiving, storage, and handling of materials and equipment at the plant site is performed in accordance with approved QC procedures 5.2.3, Receiving QC Inspection, and 5.2.4, Storage and Handling. These procedures contain measures to preclude receiving and acceptance of material which does not conform to the purchase documents and to ensure that correctly identified, acceptable materials are properly controlled in storage to preclude damage or deterioration prior to use in the construction phase. The Daniel QC engineer or inspector performing the receiving inspection initiates the required QC documentation, and documents and initiates corrective action for damaged or nonconforming materials or equipment in accordance with QC procedure 5.3.1.2, Nonconformance and Corrective Action. Procurement documents for safety-related materials or components included in the text or by attachments, specifications, drawings, and QC/QA provisions which reflect the requirements of the applicable criteria in Appendix B of 10 CFR 50. Daniel controls onsite receiving QC inspection, storage, preservation, handling, material identification, status control, and nonconformances with approved QC procedures. These procedures control the status of the material or equipment from its arrival on the site until its final acceptance for operation. Documentation generated by QC personnel in the implementation of these procedures is maintained in Document Control in accordance with paragraph 17.1.5.6.

17D.1.8 IDENTIFICATION AND CONTROL OF MATERIAL, PARTS, AND COMPONENTS

The following QC procedures are implemented by Daniel for onsite receiving QC inspection: 5.2.3, Receiving QC Inspection; 5.2.4, Storage and Handling; 5.2.5, Material Identification and Status Control; and 5.3.1.2, Nonconformance and Corrective Action. These procedures cover the status of the material or equipment from its arrival on the site until its final acceptance for operation. The QC inspection procedures provide for identification of material and equipment with approved for installation, quality control hold, or reject tags, and segregated storage area for nonconforming items. The storage and handling procedure defines environmental storage conditions and requires surveillance and audit of storage and handling by QC inspectors. The material identification and status control procedure describes the tags and marking methods to *be used and the checks to be made to ensure the status identification of materials and equipment whether approved for use in QC hold or rejected. The nonconformance procedure defines the steps to be followed in controlling deficient material and equipment, and for obtaining disposition instructions such as rework, repair, rejection, or hold for receipt of required documentation. The nonconformance procedure requires that materials, parts, or components which are reworked or repaired be reinspected and accepted by the QC inspector prior to use.*

17D.1.9 CONTROL OF SPECIAL PROCESSES

Engineering provides technical direction for assigned construction, fabrication, and equipment erection. QC personnel are responsible for developing and implementing approved procedures and instructions to assure that all field construction is in conformance with approved specifications, drawings, codes, and other specified requirements. They develop status tags, checklists, quality documents, etc., necessary to comply and to demonstrate compliance with specified requirements. Welding, heat treating, nondestructive examination (NDE), and other special processes are controlled by written procedures in accordance with approved specifications. Quality Control personnel perform and document inspections and audits to verify conformance with requirements.

Field construction and erection activities are closely monitored and audited by the PQAM and APC QA. Work on the nuclear steam supply system equipment is also monitored by Westinghouse representatives.

17D.1.10 INSPECTION

Tests and inspections are performed by Quality Control personnel in accordance with written procedures developed by the Engineering and Quality Control Sections and approved by APC prior to implementation. The program of testing and inspection assures conformance with specification requirements, procedures, and drawings. Nonconformities or damages are reported, documented, and controlled by written approved procedures until corrective action approved by APC resolves the nonconformance. Certain types of testing and inspection at the plant site are performed directly by Daniel personnel, while personnel from outside testing laboratories or subcontractors may be called upon to perform other types of inspection and testing. For example, Daniel operates the project concrete laboratory and test facility. X-Ray Engineering is subcontracted to perform a part of the NDE work. Radiography and other nondestructive examination of the welding on the containment liner are the responsibility of the erector (CB&I) and surveillance of subcontracted erection activities is performed by the cognizant Daniel Quality Control Discipline. Daniel inspectors carefully follow the work of all subcontractors at the site to ensure that both workmanship and materials are in accordance with the approved specifications, drawings, and the provisions of the contract or purchase order documents.

Mandatory inspection hold points, where inspection is required by Quality Control personnel or a third party inspector before the work sequence may proceed, shall be indicated in the QC procedure and process sheets in accordance with the Daniel procedures for control of installation of piping, equipment, and instrumentation.

17D.1.11 TEST CONTROL

Construction proof testing is performed in accordance with written test procedures approved by APC; test results shall be documented by and acceptable to APC. Daniel will assist APC as required during the preoperational and startup phase. Procedures developed by Daniel for testing receive internal Daniel review and approval and APC approval prior to release for implementation.

17D.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Devices (tools, gauges, instruments, etc.) used in the testing and inspection activities are calibrated and controlled in accordance with QC procedure 5.3.3, Calibration and Control of Testing and Measuring Equipment, to assure accuracy, calibration, periodic recalibration, and disposition control of affected work when out of calibration is discovered.

17D.1.13 HANDLING, STORAGE, AND SHIPPING

The receiving, storage, and handling of materials and equipment at the plant site is performed in accordance with QC procedures for receiving QC inspection, storage, and handling. These procedures contain measures to preclude receipt and acceptance of materials which do not conform to the purchase documents and to assure that correctly identified, acceptable materials and equipment are properly controlled to preclude damage or deterioration prior to use. Handling of materials and equipment is documented in accordance with the above procedure when applicable.

17D.1.14 INSPECTION, TEST, AND OPERATING STATUS

QC personnel are responsible for developing and implementing approved procedures and instructions to assure that project construction is in conformance with approved specifications, drawings, codes, and other specified requirements. They develop the tags, checklists, process controls, quality documents, etc., necessary to comply and to demonstrate compliance with specified requirements.

17D.1.15 NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Daniel Quality Control personnel implement procedures, checklists, and instructions to assure conformance with specification requirements, procedures, and drawings. Nonconformities or damages are reported, documented, and controlled by written procedures until corrective action approved by APC resolves the nonconformance. QC procedure 5.3.1.4, Work Stoppage, is implemented by Daniel. Basically, work stoppage authority is vested in the quality control manager. The project discipline QC engineers are responsible for initiating action to stop work if the quality of work does not meet requirements or if there is a breakdown in a work process. The QC manager will also implement a work stoppage upon a recommendation from the APC MQA field representative or the Daniel PQAM. Material involved in a work stoppage is controlled in accordance with written procedures which contain instructions for identification, documentation, corrective action, notification of affected management, and followup to verify implementation of the approved disposition.

17D.1.16 CORRECTIVE ACTION

All nonconformances are handled in accordance with QC procedure 5.3.1.2, Nonconformance and Corrective Action. The nonconformance procedure defines the steps to be followed in controlling deficient material and equipment disposition, whether repair, rejection, or hold for receipt of required documentation. The procedure requires that materials, parts, or components which are reworked or repaired be reinspected and accepted by the QC inspector prior to use. Measures to prevent reoccurrence are specified in a disposition of nonconformances.

17D.1.17 QUALITY ASSURANCE RECORDS

Required record documentation verifying conformance to project requirements for Daniel's assigned activities in the construction of the Farley Nuclear Plant shall be maintained by Daniel Document Control in accordance with the TPNS and QC procedures for document control, records, and filing. Record documentation of Daniel QC/QA audit, inspection, and test activities are included in the Document Control files. The Document Control files are continuously monitored by Quality Control personnel and the project quality assurance manager to assure that records of activities affecting quality are adequate and retrievable.

17D.1.18 AUDITS

The project quality assurance manager is responsible for providing a completely independent review, audit, and evaluation of the adequacy and effectiveness of the project QC/QA program. He is resident at the project and reports to the Daniel corporate director - quality assurance. He is assisted, as required, by resident Daniel QA engineers assigned to the project by the QA *manager. Daniel QA engineers resident at the project perform continuous auditing of project activities to assure compliance with the approved program and procedures and to determine the effectiveness of the control system.*

Daniel QA engineer audits consist of preplanned periodic surveillance and random spot audits to measure the overall effectiveness of the QC/QA program and systematic in-depth audits and reviews of individual program elements and procedures to evaluate the adequacy of the control system. QA audits are performed in accordance with specific instructions or checklists prepared by the resident QA engineers.

Daniel QA engineers report identified noncompliance with requirements to the responsible project management or supervision for immediate correction, or to QC personnel for documentation and control in accordance with the nonconformance and corrective action procedure.

Daniel QA auditing provides for reporting and followup to verify proper disposition and corrective action on deficiencies identified during previous audits.

The project QA manager issues biweekly reports of QA audit results with distribution to responsible project management, to the Daniel corporate director - quality assurance, and to APC as required. Responsible project management shall identify in writing the action taken on QA-identified control system and performance deficiencies.

The project QA manager uses QA deficiency reports or special reports to initiate involvement of the Daniel corporate director - quality assurance when necessary to effect corrective action or work stoppage.

Daniel Quality Assurance has unrestricted access to all project activities, and unrestricted, informal communication paths to all individuals and organizational components.

In addition to Daniel Project QA audit activities, the Daniel Corporate QA Department performs periodic audits of the adequacy and effectiveness of the project control system under the Daniel corporate project monitoring program described in the Project Procedure Manual. The corporate project monitoring program has provisions for reporting results and recommendations for corrective action to Daniel corporate management, Daniel project management, and APC with followup audit of identified deficiencies.]

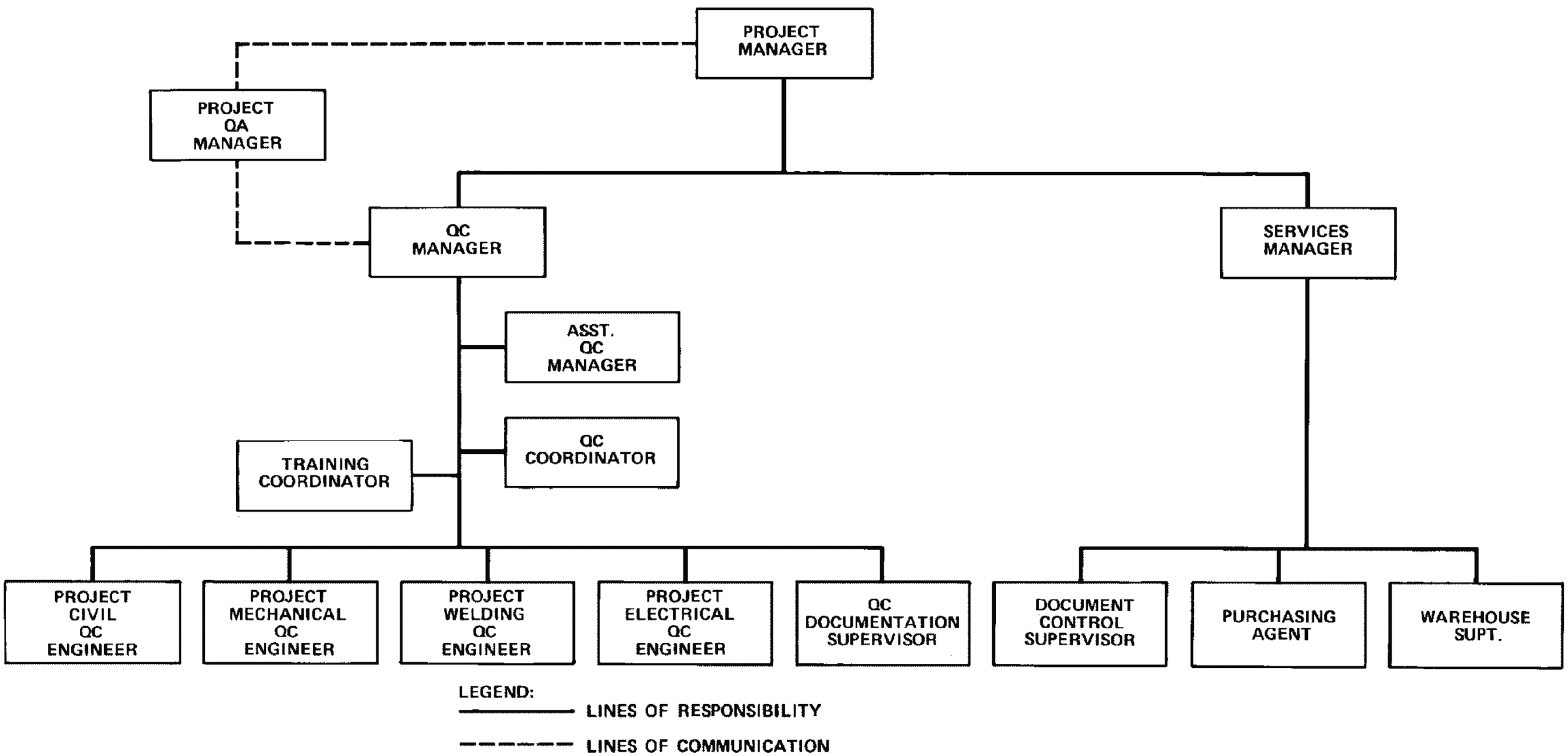

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FIGURE 17D-1]

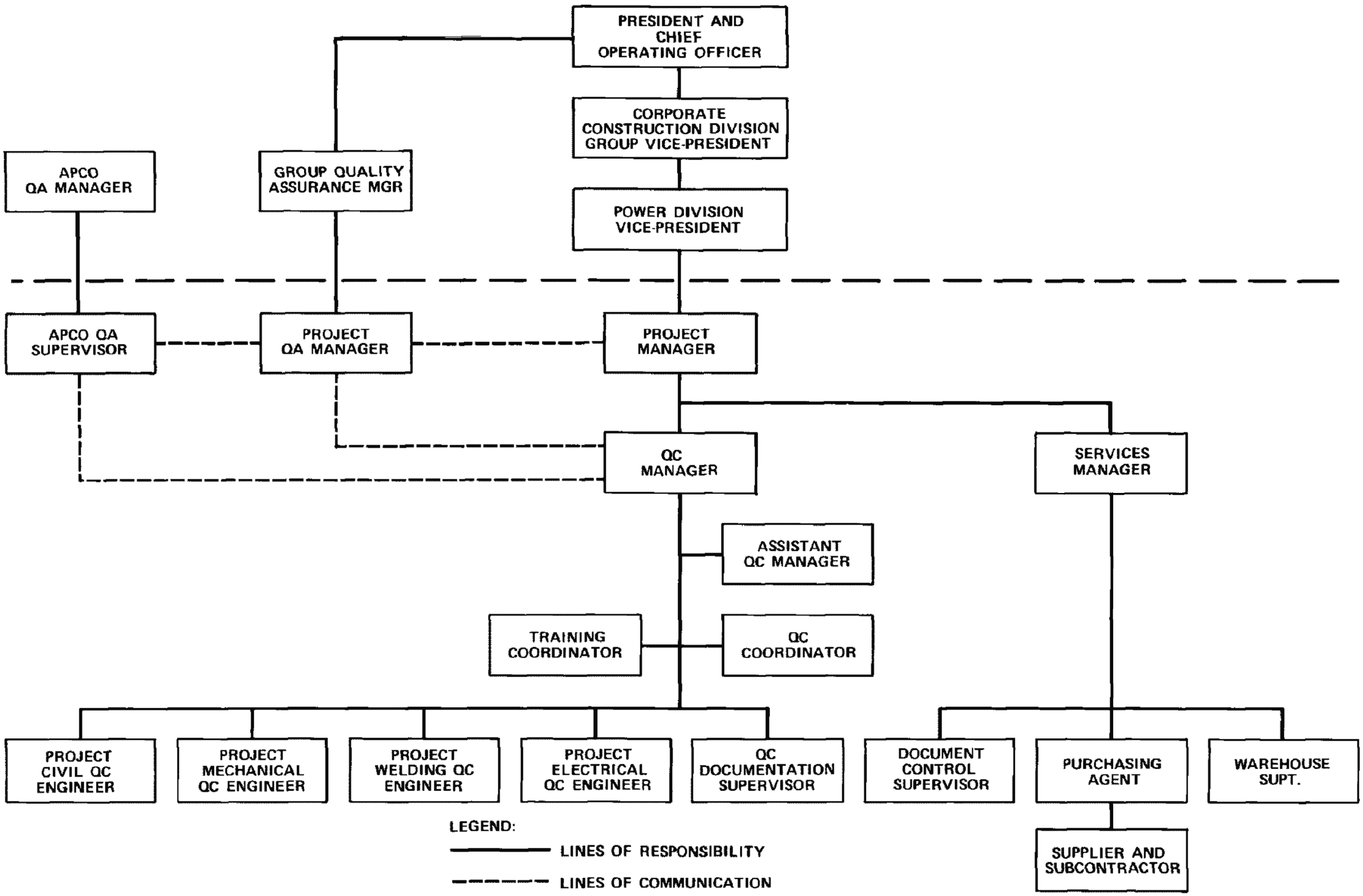

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FIGURE 17D-2]

18.0 LICENSE RENEWAL - AGING MANAGEMENT PROGRAMS AND ACTIVITIES

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18.0 – LICENSE RENEWAL – AGING MANAGEMENT PROGRAMS AND ACTIVITIES

18.1 INTRODUCTION

18.1.1 BACKGROUND

Renewed operating licenses for Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2 were issued on May 12, 2005, extending the original licensed operating term by 20 years. FNP Units 1 and 2 will enter the period of extended operation on June 26, 2017 and April 1, 2021 for Units 1 and 2, respectively.

18.1.1.1 License Renewal Rule and Process

10 CFR Part 54, the license renewal rule, establishes the procedures, criteria, and standards governing nuclear plant license renewal.

Plant systems, structures, and components (SSCs) within the scope of license renewal are defined in 10 CFR 54.4(a) as:

- Safety-related SSCs (i.e., perform a safety-related function as defined in 10 CFR 54.4(a) (1)).

- Nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of safety-related functions.

- All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

The license renewal rule focuses on managing the effects of aging on the passive intended functions of long-lived structures and components, and on evaluation of time-limited aging analyses (TLAA), as defined in 10 CFR 54.21. (See paragraph 18.1.1.3 for a discussion of the definition of a TLAA.)

The license renewal rule generically excludes structures and components associated only with active functions from an aging management review. Functional degradation resulting from the effects of aging on active functions is more readily determinable and detectable, and existing programs and regulatory requirements are expected to directly detect the effects of aging. The license renewal rule credits the continued applicability of existing programs and regulatory requirements, and the maintenance rule requirements (10 CFR 50.65), to monitor the performance and condition of systems, structures, and components that perform active functions.

The license renewal process includes the identification of systems, structures, and components within the scope of the license renewal rule, determining the in-scope structures and components subject to aging management review (i.e., are passive and long-lived), and assuring the effects of aging on the intended functions are adequately managed through the identification and/or development of various aging management programs and activities. The process also includes the identification and evaluation of TLAAs, including any exemptions containing TLAAs.

The license renewal rule and the renewed operating licenses require that a summary description of the aging management programs and activities and the TLAA evaluations become part of the FSAR. To meet this requirement, sections 18.2 through 18.5 are incorporated into the FSAR. After issuance of the renewed license, 10 CFR 54.37(b) requires that, for newly identified systems, structures, and components that would have been subject to aging management review or evaluation of TLAAs in accordance with 10 CFR 54.21, the FSAR be updated to describe how the effects of aging will be managed such that the intended functions(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation.

18.1.1.2 Aging Management Programs

The NRC, in the Standard Review Plan for License Renewal (NUREG-1800), Appendix A.1, “Aging Management Review – Generic (Branch Technical Position RLSB-1),” describes the elements of an acceptable aging management program to the NRC Staff. Additionally, NUREG-1801, “Generic Aging Lessons Learned Report,” describes aging management programs that have been found acceptable to the NRC Staff to manage the aging effects of SSCs for license renewal.

In support of the NRC’s license renewal application review process, the FNP aging management programs are evaluated for consistency with the corresponding programs described in NUREG-1801, when applicable. A program is considered reasonably and materially consistent with NUREG-1801 when it meets the key elements of the attributes described for that program. The FNP programs are identified as being consistent with, or consistent with exceptions to, the corresponding program(s) described in NUREG-1801 or as plant specific. Program consistency with NUREG-1801 means the program is consistent with a program described in Revision 0 of NUREG-1801, unless otherwise specified.

In many cases, programs and activities existing at the time of the license renewal application were found adequate for managing aging for the period of extended operation. In some cases, the existing programs or activities required some degree of enhancement. Also, some new programs and activities were identified. It is important to note that only a portion of certain programs or activities may be relied upon for managing the effects of aging under the license renewal rule.

More than one program or activity may be credited to manage aging in a single system, structure, or component. Conversely, in other cases, one program or activity may manage the effects of aging in multiple systems.

18.1.1.3 Time-Limited Aging Analyses

The license renewal rule requires that TLAA be evaluated to capture certain plant-specific aging analyses explicitly based on the original 40-year operating life of the plant. In addition, the Rule requires that any exemptions based on TLAAs be identified and analyzed to justify extension of those exemptions through the renewal term.

TLAA evaluations are defined by the license renewal rule in 10 CFR 54.3 as those calculations and analyses that meet all of the following six criteria:

- Involve SSCs within the scope of license renewal.
- Consider the effects of aging.
- Involve time-limited assumptions defined by the operating term, e.g., 40 years.
- Were determined to be relevant in making a safety determination.
- Involve conclusions or provide the bases for conclusions related to the capability of the SSC to perform its intended functions, as delineated in the Rule.
- Are contained or incorporated by reference in the current licensing basis.

Once a TLAA has been identified, the Rule in 10 CFR 54.21 (c) requires it to be dispositioned by one of the following three specific criteria:

- The analyses remain valid for the period of extended operation.
- The analyses have been acceptably projected to the end of the period of extended operation.
- The effects of aging on the intended functions(s) will be adequately managed (e.g., programs or activities are in place) for the period of extended operation.

After the renewed license has been issued, 10 CFR 54.37 (b) requires that any newly identified calculations or analyses that would have been a TLAA be evaluated and a summary description placed in the FSAR.

18.1.2 AGING MANAGEMENT PROGRAMS

The following programs are credited to manage the effects of aging during the period of extended operation for license renewal and are described in section 18.2 as listed below:

- Inservice Inspection Program (Including Subsections IWB, IWC, IWD, IWE, IWL, and IWF) (18.2.2).
- Water Chemistry Control Program (18.2.3).
- Service Water Pond Dam Inspection Program (18.2.4).
- Reactor Vessel Surveillance Program (18.2.5).
- Boric Acid Corrosion Control Program (18.2.6).
- Overhead and Refueling Crane Inspection Program (18.2.7).
- Steam Generator Program (18.2.8).
- Flow Accelerated Corrosion Program (18.2.9).
- Fuel Oil Chemistry Control Program (18.2.10).
- Structural Monitoring Program (18.2.11).
- Service Water Program (18.2.12).
- Fire Protection Program (18.2.13).
- Reactor Vessel Internals Program (18.2.14).
- Flux Detector Thimble Inspection Program (18.2.15).
- External Surfaces Monitoring Program (18.2.16).
- Buried Piping and Tank Inspection Program (18.2.17).
- One-Time Inspection Program (18.2.18).

- Nickel Alloy Management Program (18.2.19).
- Non-EQ Cables Program (18.2.20).
- Periodic Surveillance and Preventive Maintenance Activities (18.2.21).
- Selective Leaching Program (18.2.22).

18.1.3 AGING MANAGEMENT PROGRAMS – TIME LIMITED AGING ANALYSES (TLAA)

The aging management programs credited for managing the associated TLAAs during the period of extended operation are described in section 18.3 as listed below:

- Environmental Qualification Program (18.3.1).
- Fatigue Monitoring Program (18.3.2).

18.1.4 TLAA EVALUATIONS

The evaluation of TLAAs for the period of extended operation is provided in section 18.4. The TLAAs evaluated for the period of extended operation are listed below:

- Reactor Vessel Neutron Embrittlement Analyses (18.4.1).
- Metal Fatigue Analysis (18.4.2).
- Containment Tendon Pre-Stress Analysis (18.4.3).
- Environmental Qualification calculations (18.4.4).
- Ultimate Heat Sink Silting calculations (18.4.5).
- Leak-Before-Break Analysis (18.4.6).
- RHR Relief Valve Capacity Verification calculations (18.4.7).

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

18.2 AGING MANAGEMENT PROGRAM DESCRIPTIONS

18.2.1 QUALITY ASSURANCE REQUIREMENTS

The FNP Operations Quality Assurance Program will apply the quality assurance criteria of 10 CFR 50, Appendix B to the program elements of corrective actions, confirmation process, and administrative controls for the license renewal aging management program activities (described in sections 18.2 and 18.3) and their implementing documents during the period of extended operation. These criteria will be applied to the license renewal aging management activities for all safety-related and nonsafety-related structures and components that perform an intended function for license renewal.

18.2.1.1 Corrective Action

The FNP Corrective Actions Program is initiated following the identification of conditions adverse to quality, and documented as required by appropriate procedures. Various processes are used to identify problems requiring corrective action. The primary vehicle for initiating corrective action is the condition reporting process described in the Corrective Action Program.

18.2.1.2 Confirmation Process

Condition reports are reviewed to determine regulatory reportability and significance. Those items determined to be significant conditions adverse to quality are also reviewed by site management. Corrective actions taken for significant items are reviewed for assurance that appropriate action has been taken.

18.2.1.3 Administrative Controls

Activities affecting quality are prescribed by documented instructions, procedures, or drawings of a type appropriate to the condition and are accomplished in accordance with these instructions, procedures, or drawings. They contain appropriate acceptance criteria and documentation requirements for determining whether important activities have been satisfactorily accomplished. Procedures establish review and approval requirements.

18.2.2 INSERVICE INSPECTION PROGRAM

The Inservice Inspection Program will be implemented during the period of extended operation in accordance with 10 CFR 50.55a, which imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 (Subsections IWB/IWC/IWD) pressure-retaining components and their integral attachments, containment and integral attachments (Subsections IWE/IWL), and the applicable component supports (Subsection IWF). In addition, Farley Class 1 and 2 piping weld examinations will be performed per an NRC staff-approved risk-informed ISI program (Examination Categories B-F, B-J, C-F-1, and C-F-2).

The continued implementation of applicable 10 CFR 50.55a requirements, with approved alternatives and relief requests, will provide reasonable assurance that the aging effects will be managed such that the systems and components within the scope of the program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

This program is consistent with 10 attributes of the collection of acceptable programs described in NUREG-1801 Sections XI.M1, XI.M3, XI.M12, XI.S1, XI.S2, XI.S3, and XI.S4 with the clarification that exceptions to ASME Code requirements granted by approved alternatives or relief requests are not considered to be exceptions to the NUREG-1801 aging management program criteria.

18.2.3 WATER CHEMISTRY CONTROL PROGRAM

The Water Chemistry Control Program will manage aging during the period of extended operation through maintenance of low levels of detrimental impurities and the use of chemical additives.

The Primary Water Chemistry Control Program will be based upon the guidance provided in the EPRI PWR Primary Water Chemistry Guidelines (Volumes 1 & 2).

The Secondary Water Chemistry Control Program will be based upon the guidance provided in the EPRI PWR Secondary Water Chemistry Guidelines.

The Closed Cooling Water Chemistry Control Strategic Plan will be based upon the guidance contained in the EPRI Closed Cooling Water Chemistry Guideline.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M2. It is also consistent with the 10 attributes of the aging management program described in Section XI.M21A.

18.2.4 SERVICE WATER POND DAM INSPECTION PROGRAM

The service water pond dam and spillway will be inspected during the period of extended operation on a periodic basis in accordance with Nuclear Regulatory Commission (NRC) Regulatory Guide 1.127, Rev. 1, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” The service water pond dam inspection performed in accordance with Regulatory Guide 1.127 is an acceptable basis for inservice inspection and surveillance of the dam, its slopes, and associated spillway. The service water pond dam inspection(s) include the earthen dam, the service water pond embankments, and the spillway slopes.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.S7.

18.2.5 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program will be used to predict changes in reactor vessel beltline material fracture toughness during the period of extended operation. The program will be used to evaluate neutron embrittlement through surveillance capsule testing and evaluation, fluence calculations and benchmarking, and monitoring of effective full power years (EFPYs). For fluence calculations, FNP uses Regulatory Guide 1.190, which provides for a "best estimate" fluence calculation.

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion.

For each unit, FNP will install alternative dosimetry to monitor neutron fluence on the reactor vessel after removal of the last surveillance capsule in that unit.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section XI.M31, with the exception of the surveillance capsule removal schedule.

18.2.6 BORIC ACID CORROSION CONTROL PROGRAM

The Boric Acid Corrosion Control Program implements the plant-specific commitments made in response to NRC Generic Letter 88-05 and subsequent NRC communications on boric acid corrosion and leakage detection which include NRC Bulletins 2001-01, 2002-01, 2002-02, 2003-02, and NRC Order EA-03-009 (as revised). The program is applicable to areas where there are carbon steel and low-alloy steel structures or components, or electrical components, on which borated reactor water might leak.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M10.

18.2.7 OVERHEAD AND REFUELING CRANE INSPECTION PROGRAM

The Overhead and Refueling Crane Inspection Program will be used during the period of extended operation to manage the effects of general corrosion of the crane bridge and trolley structural girders and beams and the crane rails and support girders for the reactor cavity manipulator, spent-fuel bridge, spent-fuel cask, and the containment polar cranes. The contacting surfaces of the steel rails of these components will be periodically inspected in accordance with plant procedures.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M23.

18.2.8 STEAM GENERATOR PROGRAM

The Steam Generator Program used to perform replacement steam generator tube surveillance in accordance with the Technical Specifications will be continued during the period of extended operation. The program includes monitoring of steam generator secondary side internal components the failure of which could prevent the steam generator from fulfilling its intended safety-related function. The program will be based upon NEI 97-06, "Steam Generator Program Guidelines" or its successors.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M19.

18.2.9 FLOW ACCELERATED CORROSION PROGRAM

Flow Accelerated Corrosion Program activities include, but are not limited to, analysis to determine susceptible locations, license renewal in-scope susceptible locations, baseline inspections of wall thickness, follow-up inspections, and predictive modeling techniques. These activities will provide reasonable assurance that systems will perform their intended safety function(s) during the period of extended operation.

The Flow Accelerated Corrosion Program will be enhanced prior to entering the period of extended operation by adding the auxiliary feedwater pump turbine exhaust piping to the scope of the program.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M17.

18.2.10 FUEL OIL CHEMISTRY CONTROL PROGRAM

The Fuel Oil Chemistry Program is governed by Technical Specifications (emergency diesel generators fuel oil systems) and the approved fire protection program (diesel-driven fire pumps fuel oil systems). It will continue to include surveillance and maintenance procedures to mitigate corrosion as well as measures to verify the effectiveness of this aging management program and confirm the absence of an aging effect. Fuel oil quality will be maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines contained in selected American Society for Testing Materials (ASTM) standards.

The specific ASTM standards that FNP uses as guidelines for sampling and sample analysis are governed by the plant Technical Specifications (and the approved fire protection program) and differ from those cited in NUREG-1801, Section XI.M30. Parameters important to corrosion are monitored by the FNP program, and no significant differences exist in the ability of the program to manage aging effects.

SNC will evaluate the scope of the program and the need to improve procedural guidance for maintaining and monitoring the diesel-driven fire pump fuel oil system such that there is reasonable assurance that the system will perform its intended function during the period of extended operation. If changes are necessary, FNP will make them prior to the period of extended operation.

The FNP Fuel Oil Chemistry Program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M30, except as noted above.

18.2.11 STRUCTURAL MONITORING PROGRAM

The FNP Structural Monitoring Program (SMP) is based upon the requirements and guidance set forth in 10 CFR 50.65 and Regulatory Guide 1.160. SNC will continue to use the SMP to monitor the condition of structures and structural components within the scope of the Maintenance Rule, thereby providing reasonable assurance that there is no loss of structure or structural component intended function during the period of extended operation. The SMP also addresses the masonry wall considerations identified in NRC IE Bulletin 80-11 and NRC Information Notice 87-67.

The FNP SMP will be enhanced to include provisions to monitor structures and components during the period of extended operation which are in-scope for license renewal but are not currently monitored under the program.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.S5 and S6.

18.2.12 SERVICE WATER PROGRAM

The Service Water (SW) Program activities implement the recommendations of NRC Generic Letter 89-13. Mitigation, as well as performance and condition monitoring techniques, are used to manage fouling and loss of material in the SW system and components it serves. Collectively, these activities provide reasonable assurance that the SW system will perform its intended safety function(s) during the period of extended operation.

The scope of the SW Program will be enhanced prior to the period of extended operation to include inspection of piping from the main service water header to the air compressor and the service water pump columns.

This program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M20.

18.2.13 FIRE PROTECTION PROGRAM

The Fire Protection Program will provide inspections, performance testing, monitoring, and aging management activities during the period of extended operation for water- and gas-based fire protection systems, fire dampers, fire doors, fire penetration seals, cable wrap, and fire pump diesels (including the external surfaces of exposed fuel oil piping) requiring aging management of license renewal.

SNC will implement the following enhancements to the FNP Fire Protection Program prior to entering the period of extended operation through the use of administrative controls and procedures.

- The fire protection sprinkler system piping will be subjected to wall thickness evaluations (e.g., nonintrusive volumetric testing and/or visual internal inspections during plant maintenance) prior to the period of extended operation and at specific intervals thereafter. The plant-specific inspection interval will be established from the initial inspection results and revised as appropriate for subsequent inspection results.

- A sample of sprinkler heads will be tested by using the guidance of National Fire Protection Association (NFPA) 25 (2002), Section 5.3.1.1.1, at or before 50 years service and every 10 years thereafter.

- Diesel-driven fire pump surveillance procedures will be upgraded to provide more detailed instructions related to inspection of the fuel oil supply piping.

- The current practice of replacing CO_2 hoses at 5-year intervals will be formalized in fire protection procedures.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.M26 and M.27, as amended by Interim Staff Guidance ISG-04.

18.2.14 REACTOR VESSEL INTERNALS PROGRAM

The FNP Reactor Vessel Internals Program was implemented prior to entering the period of extended operation and provides an integrated inspection program that addresses the reactor internals. It is governed by administrative controls and procedures to supplement the inspection requirements of ASME Section XI, IWB Category B-N-3 to ensure that aging effects do not result in a loss of intended function of internal components during the period of extended operation.

The program manages the effects of crack initiation and growth due to irradiation-assisted stress corrosion cracking; loss of fracture toughness due to irradiation embrittlement, thermal embrittlement, or void swelling; or changes in material properties (dimension) due to void swelling.

SNC will continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting reactor vessel internals in accordance with the SNC commitment to the Nuclear Energy Institute (NEI) 03-08 Materials Initiative.(6) These initiatives resulted in industry guidance promulgated under reference 6 for operating PWR designs. SNC incorporated the results of these initiatives (to the extent that they are applicable to the FNP reactor internals) into the scope, inspection requirements (inspection locations, methods, qualifications, and frequencies), acceptance criteria, and corrective actions of the Reactor Vessel Internals Program.

SNC submitted an aging management program/plan documenting the plant-specific consistency with industry guidance and interim staff guidance LR-ISG-2011-04 for the FNP Reactor Vessel Internals for NRC review and approval at least 22 months prior to entering the period of extended operation for the FNP units.(12) Via reference 8, industry guidance was incorporated into NUREG-1801 aging management program X1.M16A “PWR Vessel Internals”. Consistent with a living program, future revision to industry guidance will be incorporated as applicable into the Reactor Vessel Internals Program when issued under NEI 03-08.(6)

The FNP Reactor Vessel Internals Program is consistent with the 10 attributes of the aging management program described in LR-ISG-2011-04 which revised NUREG-1801, Section X1.M16A.

18.2.15 FLUX DETECTOR THIMBLE INSPECTION PROGRAM

The new Flux Detector Thimble Inspection Program will be implemented prior to entering the period of extended operation to formalize examinations already being performed. The program will be administratively controlled by plant procedures. It will be used to identify loss of material due to fretting/wear in the detector thimble tubes during the period of extended operation.

18.2.16 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program will be a new plant-specific condition monitoring program that will be implemented prior to entering the period of extended operation. It will include periodic visual inspections of external surfaces of carbon steel, low alloy steel, and other susceptible materials in components requiring aging management for license renewal.

Plant procedures and administrative controls will be developed to provide for surface condition monitoring of selected equipment and components for signs of corrosion or wear. Periodic inspections of accessible portions of piping and tubing will be performed to detect signs of loss of material, flange leakage, missing or damaged insulation, damaged coatings, and fretting of tubing.

Accessible in-scope polymers or elastomers will also be inspected for loss of material, cracking, and change in material properties. Susceptible materials or components will include accessible fasteners, ventilation systems seals and collars, other polymers and elastomers, copper, aluminum, and coated steel structural components which are not within the scope of the Structural Monitoring Program.

18.2.17 BURIED PIPING AND TANK INSPECTION PROGRAM

The new Buried Piping and Tank Inspection Program will be used to manage the loss of material from external surfaces of in-scope pressure-retaining buried carbon steel piping and tanks and buried stainless steel and copper alloy piping during the period of extended operation. Administrative controls and procedures will be put in place to ensure that buried piping and tanks will be inspected when they are excavated for maintenance or when those components are exposed for any reason. This new program will be implemented prior to the period of extended operation.

SNC will perform an inspection of buried piping and tanks within 10 years after entering the period of extended operation, unless an opportunistic inspection has occurred within this 10-year period. Before the tenth year, SNC will perform an engineering evaluation to determine if sufficient inspections have been conducted to draw a conclusion regarding the ability of the underground coatings to protect the underground piping and tanks from degradation. If not, SNC will conduct a focused inspection to allow that conclusion to be reached.

This program will be consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M34, with the exception that it also includes provisions for inspection of buried stainless steel and copper alloy piping.

18.2.18 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program will be implemented prior to the period of extended operation. The One-Time Inspection Program will include measures to verify the effectiveness of various other aging management programs and confirm the absence of aging effects requiring management. These measures will include use of examinations of reactor coolant system small-bore (< 4-in. NPS) ASME Class 1 piping components for cracking as an indicator of the potential for stress corrosion cracking in other stainless steel components exposed to a borated water environment. Also included is a thickness measurement of the bottom of the condensate storage tank. Insofar as practical with respect to scheduled outages, the inspections for selective leaching will be performed within a window of 5 years immediately preceding the period of extended operations and all other one-time inspections will be performed within a window of up to 10 years immediately preceding the period of extended operations.

The program will be administratively controlled by plant procedures. Administrative controls and procedures will be developed to identify the specific components which must be included, as well as the systems from which the remaining sample set will be collected.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.M32 and M33 except that some inspections for Unit 2, not including inspections for selective leaching, will be performed more than 5 but less than 10 years before the period of extended operation.

18.2.19 NICKEL ALLOY MANAGEMENT PROGRAM

The plant-specific Nickel Alloy Management Program was implemented prior to the period of extended operation to address the potential for primary water stress corrosion cracking (PWSCC) in nickel alloy components exposed to the reactor coolant environment. This program assessed nickel base alloy component susceptibility to PWSCC and provided for any required augmented inspection requirements to ensure that the susceptible components will be maintained within ASME acceptance criteria during the period of extended operation. Administrative controls and procedures were developed to implement the program in accordance with industry initiatives. Subsequent to this initial assessment, ASME code cases were developed and mandated by rulemaking under §10 CFR 50.55a which has superseded direct industry guidance and is applicable to the original and extended periods of operation.(9)(10)

The scope includes nickel base alloy reactor coolant pressure boundary components with known or potential susceptibility to PWSCC, excluding steam generator tubes, which are specifically addressed by the Steam Generator Program, and reactor internals which are addressed by the Reactor Internals Inspection Program. The scope and frequency of examinations and acceptance criteria for detected degradation not already covered by ASME Code Section XI have been incorporated into ASME Code Section XI code cases as conditioned in §10 CFR 50.55a.

FNP will continue to participate in industry initiatives (such as the PWR Owners Group and the EPRI Materials Reliability Program) in accordance with the SNC commitment to the NEI 03-08 Materials Initiative.(6) Susceptibility rankings and program inspection requirements are consistent with the latest version of the EPRI Materials Reliability Program safety assessment regarding Alloy 82/182 pipe butt welds or successor code cases as conditioned in §10 CFR 50.55a.

SNC submitted correspondence indicating that all plant components under the scope of the Nickel Alloy Management Program has been incorporated into the ISI Program 24 months prior to entering the period of extended operation for the FNP units.(11)

18.2.20 NON-EQ CABLES PROGRAM

The Non-EQ Cables Program will be a new inspection and testing program that will be implemented prior to the period of extended operation. It will be used to maintain the function of electrical cables and connections which are not subject to the environmental qualification requirements of 10 CFR 50.49, but are exposed to adverse localized environments caused by heat, radiation, or moisture.

The program will be administratively controlled by procedures. The scope will include: 1) accessible electrical cables and connections (connectors, splices, terminal blocks, fuse holders, and electrical penetration assembly pigtails) installed in adverse localized environments caused by heat or radiation, coupled with the presence of oxygen; 2) electrical cables used in circuits with sensitive, high voltage, low-level signals such as radiation monitoring and nuclear instrumentation and; 3) inaccessible medium voltage cables that are exposed to significant moisture and voltage at the same time.

This program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Sections XI.E1 and E3, and for Section XI.32 as amended by Interim Staff Guidance ISG-15 and the alternate program drafted by the License Renewal Working Group.

18.2.21 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE ACTIVITIES

The periodic surveillance and preventive maintenance activities are plant-specific periodic inspections and tests that are relied upon for license renewal to manage the aging effects applicable to the components included in the program that are not managed by other aging management programs. The periodic surveillance and preventive maintenance activities are implemented through repetitive tasks and surveillances.

The periodic surveillance and preventive maintenance activities credited for license renewal will be implemented prior to the period of extended operation.

The specific items included in this program are as follows:

- Periodic visual inspection of a sample set of tank diaphragms for the boric acid tanks, reactor makeup water storage tanks, and condensate storage tanks.

18.2.22 SELECTIVE LEACHING PROGRAM

The Selective Leaching Program will be implemented prior to the period of extended operation. The Selective Leaching Program will include measures to verify the integrity of components made of cast iron, bronze, brass, and other alloys exposed to raw water, treated water, or a groundwater environment that may lead to selective leaching of one of the metal components. Insofar as practical with respect to scheduled outages, the inspections for selective leaching will be performed within a window of 5 years immediately preceding the period of extended operations and continue on a opportunistic basis into the period of extended operations. The aging management program includes a one-time visual inspection and hardness test to determine whether the loss of material due to selective leaching is occurring.

The program will be administratively controlled by plant procedures. Administrative controls and procedures exist to identify the specific components which must be included, as well as the systems from which the remaining sample set will be collected.

The program is consistent with the 10 attributes of the aging management programs described in NUREG-1801, Section XI.M33.

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

6. Nuclear Energy Institute (NEI), NEI 03-08 Revision 2, "Guideline for the Management of Materials Issues," January 2010.

7. Regulatory Issue Summary (RIS) 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Aging Management," July 21, 2011.

8. NRC Final License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," noticed in Federal Register June 3, 2013.

9. Federal Register Volume 73, No. 176, September 10, 2008, "NRC Rulemaking amending §10 CFR 50.55a to incorporate by reference ASME Code Cases N-729-1 and N-722.

10. Federal Register Volume 76, No. 119, June 21, 2011, "NRC Rulemaking amending §10 CFR 50.55a to incorporate by reference ASME Code Cases N-770-1 and N-722-1.

11. Southern Nuclear letter NL-15-0336 dated June 22, 2015.

12. Southern Nuclear letter NL-15-1507 dated August 12, 2015.

18.3 AGING MANAGEMENT PROGRAMS – TIME LIMITED AGING ANALYSES (TLAA)

18.3.1 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification (EQ) Program manages component thermal, radiation, and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49 (f) qualification methods. As required by 10 CFR 50.49, EQ components whose qualified lives expire before the end of the applicable license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section X.E1.

18.3.2 FATIGUE MONITORING PROGRAM

The design basis metal fatigue analyses for the FNP reactor coolant pressure boundary are TLAAs. The Fatigue Monitoring Program will be used to monitor plant transients that are significant contributors to the fatigue cumulative usage factor. Demonstration that plant cycles have not exceeded design assumptions during the period of extended operation will ensure that the design limit on fatigue usage will not be exceeded. If projected plant cycles exceed a design assumption, SNC will take corrective action which may include a more refined analysis or replacement or an inspection program approved by the NRC. As an alternative to monitoring the number of certain transients, stress-based monitoring of the plant transient may be used to compute the actual fatigue usage of each transient, at a bounding location.

SNC will fully implement the program prior to entering the period of extended operation. When fully implemented, the program will include monitoring for thermal stratification at susceptible locations in addition to the current transient counting required by Technical Specifications. SNC has evaluated the effects of environmentally assisted fatigue on piping and components comparable to the locations evaluated in Section 5.4 of NUREG/CR-6260. The results of that evaluation are given in paragraph 18.4.2.1.

This program is consistent with the attributes of the aging management program described in NUREG-1801, Section X.M1.

REFERENCES

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

18.4 EVALUATION OF TIME LIMITED AGING ANALYSES (TLAA)

18.4.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSES

The reactor vessels are subjected to neutron irradiation from the core. This irradiation results in embrittlement of the reactor vessel materials. The following FNP analyses address the effects of neutron embrittlement of the reactor vessels for both units.

- Upper-Shelf energy (USE).
- Pressurized thermal shock (PTS).
- Pressure-Temperature (P-T) limits.
- Adjusted reference temperature (ART).
- Neutron fluence.

18.4.1.1 USE Calculation

Appendix G of 10 CFR Part 50 requires that the reactor vessel beltline materials must maintain a Charpy USE of no less than 50 ft-lb throughout the life of the reactor vessel.

SNC has projected the FNP analyses to the end of the period of extended operation for the limiting component of the beltline region materials. The limiting Unit 1 location has a projected end-of-life (EOL) USE of 52.8 ft-lb. For Unit 2, the limiting USE location has a projected EOL USE of 58 ft-lb. These TLAAs have been shown to be acceptable for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.2 PTS Calculation

The requirements of 10 CFR 50.61 provide for protection against PTS events in pressurized water reactors. The screening criterion in 10 CFR 50.61 is 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. According to this regulation, if the calculated RT_{PTS} for the limiting reactor beltline materials is less than the specified screening criterion, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients.

SNC has updated the RT_{PTS} calculation for FNP Units 1 and 2 to include the period of extended operation, and has determined that the screening criteria are met for both units. The limiting material for FNP Unit 1 has a 54 EFPY RT_{PTS} value of 191°F. The limiting material for FNP Unit 2 has a 54 EFPY RT_{PTS} value of 239°F. These TLAAs have been shown to be acceptable for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.3 P-T Limits Calculation

Appendix G of 10 CFR Part 50 requires heatup and cooldown of the reactor pressure vessel be accomplished within established limits for P-T. Plant-specific calculations establish these limits. The calculations utilize materials and fluence data obtained through plant-specific reactor surveillance capsule programs.

The P-T limit curves that apply for the current operating conditions at FNP are included in the Pressure and Temperature Limits Report (PTLR) for each unit. When the operating conditions of each unit merit the use of a different curve, the PTLR for that unit is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit. SNC has updated the FNP P-T calculations, including the ART values, to account for 54 EFPY in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.1.4 ART Calculation

SNC updated the calculations to determine the ART for the critical components of the reactor vessel for 54 EFPY in accordance with 10 CFR 54.21 (c) (1) (ii). The ART values that apply for the current operating conditions at FNP are included in the PTLR for each unit. When the PTLR is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit, updated ART values are included.

18.4.1.5 Neutron Fluence Calculation

SNC updated the reactor vessel neutron embrittlement calculations including the neutron fluence calculations for the critical components of the reactor vessel for 54 EFPY in accordance with 10 CFR 54.21 (c) (1) (ii). The neutron fluence values that apply for the current operating conditions at FNP are summarized in the PTLR for each unit. When the PTLR is updated to include P-T limit curves that bound the current level of neutron embrittlement for the unit, changes in neutron fluence values are included.

18.4.2 METAL FATIGUE ANALYSIS

The thermal fatigue analyses of the FNP mechanical components have been identified as TLAAs.

18.4.2.1 ASME Section III, Class 1 Component Fatigue Analysis

Section III of the ASME Code requires a discrete analysis of the thermal and dynamic stress cycles on components that make up the reactor coolant pressure boundary. The required analysis completed for FNP incorporated a set of design transients. SNC reviewed the transient cycle assumptions and determined that the assumed transient cycles are conservative for 40 years and bounding for the period of extended operation, except in the specific cases described below.

The design basis for the FNP pressurizer surge line includes a stress analysis to ensure that cumulative fatigue usage will remain below the ASME Code allowable. SNC has evaluated the cumulative fatigue on the pressurizer surge line and will manage it during the period of extended operation using the Fatigue Monitoring Program in accordance with 10 CFR 54.21 (c) (1) (iii).

The design basis for the FNP RHR suction lines includes an analysis of the impact of thermal stratification on certain portions of these lines. The analysis meets the definition of a TLAA, pursuant to 10 CFR 54.3. In accordance with 10 CFR 54.21 (c) (1) (iii), SNC will monitor the actual transients on these lines using the Fatigue Monitoring Program described in subsection 18.3.2 to show that the assumptions used in the analysis will not be exceeded during 60 years of operation.

Thermal stratification of the pressurizer surge line and the resultant fatigue effects are similarly treated for FNP. As part of the FNP response to NRC IE Bulletin 88-11, SNC prepared an evaluation of the impact of thermal stratification on the surge line. The fatigue usage value calculated using the Fatigue Monitoring Program includes the impact of thermal stratification upon the cumulative fatigue of the surge line (demonstration in accordance with 10 CFR 54.21 (c) (1) (iii)).

SNC has evaluated the effect of environmentally assisted fatigue (EAF) for locations equivalent to those presented in Section 5.4 of NUREG/CR-6260. The application of the appropriate environmental factors from NUREG/CR-6583 resulted in an acceptable environmentally assisted fatigue adjusted value $<$ 1.0.

- Reactor vessel shell and lower head.
- Reactor vessel inlet and outlet nozzles.
- Surge line hot leg nozzle.
- Safety injection nozzle.
- Charging nozzles and alternate charging nozzles.
- Residual heat removal 6-in. RHR/SI nozzles to the RCS cold leg.

18.4.2.2 ASME Section III, Non-Class 1 Component Fatigue Analysis

For cracking due to thermal fatigue for in-scope FNP components outside the reactor coolant pressure boundary (non-Class 1), thermal stresses on piping will bound thermal stresses on other components in a system. The design of ASME Code Section III Class 2 and 3 piping systems at FNP incorporates the Code stress reduction factor for determining the acceptability of the piping design with respect to thermal stresses. Those in-scope components that are designed in accordance with ASME B31.1 Code requirements also incorporate the stress reduction factor based upon an assumed number of thermal cycles. In general, 7000 thermal cycles are assumed, leading to a stress reduction factor of 1.0 in the stress analyses. SNC evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicate that the 7000 thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the existing pipe stress calculations are valid for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (i).

SNC has determined that 22000 thermal cycles are assumed in the design for FNP small bore piping systems that receive a blowdown from the main steam or reactor coolant systems. This assumption has also been evaluated and determined to be bounding for 60 years of operation. For the air start system of the emergency diesel generators, 60 years of operation will produce more than 7000 thermal cycles. This piping has also been evaluated and found to be acceptable as designed. Therefore, the TLAA for these components is adequate for the period of extended operation, in accordance with 10 CFR 54.21 (c) (1) (i).

18.4.2.3 Reactor Coolant Pump Flywheel Fatigue

Westinghouse has generically analyzed the potential for cracking due to fatigue in reactor coolant pump (RCP) flywheels in WCAP-14535A and WCAP-15666. These two Westinghouse analyses are applicable to FNP. The evaluations of the growth of an assumed crack in the flywheel uses the assumption that the RCPs will experience 6000 start/stop cycles over 60 years of operation. The evaluations show that the crack growth is negligible for the flywheel model that bounds those in the RCPs at FNP. The number of start/stop cycles for the FNP RCPs is estimated to be significantly $<$ 6000 through the period of extended operation. Therefore, these analyses are valid for FNP through the period of extended operation (demonstration in accordance with 10 CFR 54.21 (c) (1) (ii)).

18.4.3 CONTAINMENT TENDON PRESTRESS ANALYSIS

To meet the requirements on 10 CFR 50.55a (b) (2) (ix) (B), SNC used an analysis to predict the amount of residual prestress in the containment tendons for FNP. This analysis meets the definition of a TLAA. SNC performed a new analysis to estimate the amount of residual prestress on the tendons after 60 years of operation (demonstration in accordance with 10 CFR 54.21 (c) (1) (ii)).

The new calculation includes the latest measurements of containment tendon prestress taken since the plant began commercial operation. The calculation indicates that acceptable containment tendon prestress will continue to exist throughout the period of extended operation.

The minimum required prestressing forces for the vertical, hoop, and dome tendons (kip/wire) are 6.81, 6.01, and 6.35, respectively.

18.4.4 ENVIRONMENTAL QUALIFICATION CALCULATIONS

The FNP EQ program described in subsection 18.3.1 meets the requirements of 10 CFR 50.49. Aging evaluations which meet the definition of a TLAA can be found in test reports, test report evaluations (10 CFR 50.49 checklists), and calculations. Qualified service lives for the EQ components have already been determined. EQ components are tracked to determine when a component is nearing the end of its service life.

For those components that are nearing the end of their qualified service life, the EQ program has provisions for the components to be reevaluated for longer service, refurbished, requalified, or replaced. The EQ program described in subsection 18.3.1 will be continued through the extended term of operation as an aging management program in accordance with 10 CFR 54.21 (c) (1) (iii).

18.4.5 ULTIMATE HEAT SINK SILTING CALCULATIONS

The FNP ultimate heat sink (UHS) is a pond in which excessive silting could reduce the total volume of water available to maintain long-term shutdown cooling following a design basis accident. SNC conducts a regular surveillance to confirm water volume in the pond. The acceptance criteria for this surveillance involves a volume versus pond level curve that is calculated with 40-year assumptions as to the amount of silting that could occur without adversely impacting the volume of the pond and, therefore, meets the definition of a TLAA.

SNC has updated the calculations to include the pertinent depth-sounding data and to address the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii).

18.4.6 LEAK-BEFORE-BREAK ANALYSIS

A leak-before-break (LBB) analysis has been performed for the FNP primary coolant loop and the pressurizer surge line. LBB analyses evaluate postulated flaw growth in the piping for the reactor coolant loops and the surge line. These analyses meet the definition of a TLAA.

For the primary coolant loop, SNC has updated the LBB analysis to account for the period of extended operation in accordance with 10 CFR 54.21 (c) (1) (ii). For the LBB analysis of the pressurizer surge line, SNC has determined that the current analysis is bounding for 60 years, in accordance with 10 CFR 54.21 (c) (1) (i).

18.4.7 RHR RELIEF VALVE CAPACITY VERIFICATION CALCULATION

SNC takes credit for the relief capacity of the RHR relief valves in the cold overpressure mitigation analysis for FNP. SNC performed a calculation that verifies relief valve capacity given the safe operating P-T limit curves. The calculation adjusts the P-T limit curves to account for the flow-induced pressure drop from the beltline of the reactor vessel to the RHR relief valves. The calculation meets the definition of a TLAA. Pursuant to 10 CFR 54.21 (c) (1) (ii), SNC will update this calculation to include the calculated 54 EFPY P-T limit curves prior to entering the period of extended operation.

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1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, Initial Report, July 2001.

3. Joseph M. Farley Technical Specifications, Units 1 and 2.

4. NRC Interim Staff Guidance (ISG)-04, Aging Management of Fire Protection Systems for License Renewal, December 3, 2002.

5. NRC Interim Staff Guidance (ISG)-15, Interim Staff Guidance on the Identification and Treatment of Electrical Fuse Holders for License Renewal, March 10, 2003.

6. WCAP-14535A, Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination, November 1, 1996.

7. WCAP-15666, Extension of Reactor Coolant Pump Motor Flywheel Examination, October 2003.