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11. RADIOACTIVE WASTE AND RADIATION PROTECTION AND MONITORING

11.1 RADIOACTIVE WASTE DISPOSAL SYSTEM

11.1.1 Design Bases

11.1.1.1 General

The radioactive waste disposal system (RWDS) is designed to protect plant personnel and the public from exposure to radioactive wastes in accordance with 10 CFR Part 20; 10 CFR 50, Appendix I; 40 CFR Part 190; 10 CFR 50 Appendix A General Design Criteria 60, 63, and 64; 10 CFR 50 Appendix B for reviews and audits; and the intent of NUREG-0472, Draft Revision 3 (see Section 11.3).

The RWDS has been reviewed against the requirements of NUREG-0472, Draft 7 of Revision 3, "Standard Radiological Effluent Technical Specifications (RETS) for Pressurized Water Reactors". As a result of the review, Technical Specifications were approved to govern effluent instrumentation calibration and operation, allowable dose rates, approved methodology to calculate dose rates, limiting conditions for operating the RWDS, requirements for environmental monitoring programs and requirements for maintaining records, ensuring adequate review and audits and reporting information as required. The details of RETS commitments for the liquid, gaseous and solid radioactive treatment systems are discussed in Sections 11.1.2, 11.1.3, and 11.1.4. RETS commitments for Radiation Monitoring are discussed in Section 11.2.3. Section 11.3 addresses overall requirements such as the Offsite Dose Calculation Manual (ODCM), reporting requirements, and summarizes the requirements of RETS as they are addressed by the Technical Specifications.

NRC Generic Letter 89-01 allowed licensees to remove the procedural details of the Radiological Effluent Technical Specifications from the Technical Specifications and place them in the ODCM. The administrative section of Technical Specifications was updated to include the programmatic controls necessary to ensure compliance with Federal Regulations. This change has placed the procedural requirements for equipment, sampling, analyses, monitoring, and dose limitations in the ODCM. Reference to specific sections of the ODCM will not be made in this document.

The RWDS includes equipment to collect, store, process and treat as required, monitor, and dispose of liquid, solid, and gaseous radioactive wastes.

The RWDS is designed to process and remove radioactive wastes from the plant adequately and safely when 1 percent of the core fuel elements have failed and corrosion and fission product concentrations in the reactor coolant are at design values. The design of the RWDS is based on the plant operating cycle shown in Table 11.1-1.

Table 11.1-1 - "Plant Operating Cycle"

Event	Number of Occurrences per Refueling Cycle
Refuel and start-up	1
Cold shutdown and restart immediately following initial full power operation	1
Hot shutdown and restart with one occurring within the last 40 days of core life	4
Cold shutdown and partial drain of reactor coolant loop for maintenance followed by restart occurring after the third hot shutdown and restart	1
Initiate operation of deborating demineralizer	1
Cold shutdown prior to refueling	1

11.1.1.2 Radioactive Waste Inventory

The waste volumes estimated to accumulate during one refueling cycle are shown in Table 11.1-2.

Table 11.1-2 - "Radioactive Waste Volumes"

	Volume (ft^3/cycle)	Basis
Liquids	150,000	Processed liquid at 70°F
Gases	50,800	At 70°F and 1 atm
Solids	5,000	Dry Activated Waste, filters, spent resins, depleted filtration/ion exchange media

11.1.1.3 Reactor Coolant Composition

The accumulated radioactive waste inventory has been calculated assuming operation with one percent failed fuel in the core. A CE analysis, Reference 11-1, was used to calculate the time dependent fission activity levels of individual nuclides in the fuel rods and coolant. The parameters used in the calculation are summarized in Table 11.1-3, the coolant chemistry is as summarized in Table 9.2-2, and the resulting coolant activity is given in Table 11.1-5. Credit has been taken for normal ion exchange purification in the chemical and volume control system (see Section 9.2); the ion exchangers are assumed to reduce the coolant activity level of most nuclides by a factor of 10, but no credit is taken for removal of corrosion products, noble gases, molybdenum, rubidium, tritium, or yttrium. The major area of conservatism in the calculation is the fission product release fractions, these are based upon The Reactor Safety Study (WASH-1400, 1975) and ANS/ANSI-5.4. The ANSI Standards suggest that under low temperature conditions, the cumulative fraction release is independent of temperature. The following equations are used to estimate the release fractions for long and short lived nuclides:

<u>Long Lived Nuclides (half life > 1 year, ANS/ANSI-5.4, 1982)</u>

$F = 7x10^{-8}$ (Bu)
F = the release fraction
Bu = the burnup in MWD/MTU

<u>Short Lived Nuclides (half-life < 1 year, ANS/ANSI-5.4, 1982)</u>

F = the release fraction

λ = the decay constant, sec^{-1}

P = the specific power in MWD/MTU

Table 11.1-3 - "Parameters Used for Calculation of Reactor Coolant Activity"

Percent failed fuel rods, %	1
Fuel Enrichment	4.5% ^{235}U
End Cycle Composition	44 Assemblies at 20 GWD/MTU 44 Assemblies at 40 GWD/MTU 45 Assemblies at 60 GWD/MTU
<u>Isotope</u>	<u>Release Fraction</u>
Xe-133	0.058
Xe-131m	0.131
I-131	0.088
All Other	0.018
Reactor coolant volume (includes the water volume in the pressurizer and CVCS), ft^3	6716
Refueling dilution factor	0.144
Purification flow rate (power operation), gpm	36
Refueling purification flow rate	0
Fraction of fission products remaining after each refueling	0.666

The release fractions were calculated for noble gas and iodines. For isotopes which have release fractions less than 1.8% (WASH 1400 value); the more conservative 1.8% was used. For isotopes with calculated fraction greater than 1.8%; the calculated values were used. A summary of the release fractions greater than 1.8%, are found in Table 11.1-6. These release fractions will be used in the analysis.

These release fractions are consistent with the guidance found in Regulatory Guide 1.77, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and NRC Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequence of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." These guides recommend using a release fraction for Noble Gases and Iodine of 10 percent.

Table 11.1-5 - "Average Fission and Corrosion Product Activity in The Reactor Coolant with 1% Failed Fuel"

Nuclide	Specific Activity at STP (μCi/cc)	Coolant Inventory (curies)
Xe-131m	1.60 E+0	4.76 E+2
Xe-133	1.24 E+2	3.69 E+4
Xe-135	9.83 E+0	2.93 E+3
Xe-135m	7.81 E+0	2.33 E+3
Xe-137	3.47 E+0	1.03 E+4
Xe-138	3.27 E+1	9.75 E+3
Kr-85	2.63 E-1	7.82 E+1
Kr-85m	5.07 E+0	1.51 E+3
Kr-87	9.71 E+0	2.89 E+3
Kr-88	1.36 E+1	4.05 E+3
Kr-89	1.66 E+1	4.94 E+3
I-129	1.21 E-8	3.61 E-5
I-131	1.37 E+0	4.09 E+3
I-132	4.05 E-1	1.20 E+3
I-133	5.72 E-1	1.70 E+3
I-134	6.26 E-1	1.86 E+3
I-135	5.36 E-1	1.60 E+3
Br-84	5.97 E-2	1.78 E+2
Ru-103	4.36 E-1	1.30 E+3
Ru-106	1.65 E-1	4.90 E+2
Te-129	8.87 E-2	2.64 E+2
Te-132	3.98 E-1	1.19 E+3
Te-134	4.73 E-1	1.41 E+3
Cs-134	6.50 E-2	1.93 E+2
Cs-137	4.19 E-2	1.25 E+2
Cs-138	5.23 E-1	1.56 E+3
Ba-140	4.92 E-1	1.47 E+3
La-140	5.28 E-1	1.57 E+3
Rb-88	2.00 E+0	5.94 E+2
Rb-89	2.56 E-1	7.61 E+2
Sr-89	2.65 E-1	7.89 E+2
Sr-90	3.04 E-2	9.05 E+0
Y-90	3.17 E-1	9.44 E+1
Sr-91	3.32 E-1	9.87 E+2
Y-91	3.43 E+0	1.02 E+3
Nb-95	4.74 E-1	1.41 E+3
Zr-95	4.71 E-1	1.40 E+3
Mo-99	5.13 E+0	1.53 E+3
H-3	1.00 E-1	1.38 E+1
Total =	2.44 E+2	1.05 E+5

Table 11.1-6 - "Release Fractions"

Isotope	F (Fraction of Activity Escaping)
^{133}Xe	0.058
^{131m}Xe	0.131
^{131}I	0.088
All Others	0.018

11.1.1.3a Calculations without Ion-Exchange

The 1% coolant activity without ion-exchangers in service was calculated. The 1% coolant inventory was divided by the volume of the Reactor Coolant (1.90×10^8 cm^3) to estimate the circulating concentration in the reactor coolant system. The concentrations were corrected to standard temperature and pressure by multiplying each concentration by 0.639. The correction factor for standard pressure and temperature (STP) was arrived at by taking the ratio of the specific volumes of saturated liquid at 2100 psia to that of 14.696 psia. To estimate the specific activities at operating conditions, the concentrations were multiplied by 1.56 and divided by 0.613 gm/cm^3 (the density of water at 2100 psia). The correction factor for the specific activity is the inverse of the correction factor at STP.

$A =$ $(0.01)Fk_2k_3A_{Core}$ where

$A =$ the coolant activity for 1% failed fuel, Ci

$A_{Core} =$ 100% core source term, Ci

$F =$ the fraction of activity release from fuel

$k_2 =$ the refueling fraction (0.666)

$k_3 =$ the refueling dilution factor (0.144, Particulates and Halogens only)

11.1.1.3b Calculations with Ion-Exchange

A decon factor of 10 was applied to each isotope listed with exception of Noble Gases, Molybdenum, Rubidium, and Yttrium; this factor is applied since the letdown ion-exchangers have a decontamination factor of approximately 10% for cations in the presence of Boron.

Table 11.1-7 - "Fort Calhoun Fission Product Coolant Activity (4.5% by Weight ^{235}U)"

Isotope	Core Activity (Ci) 100%	W/O Ion Exch. Coolant Activity (Ci) 1%	Isotopic %	Type	With Ion Exch. Coolant Activity (Ci) 1%	STP Coolant μCi/cc 1%
89-Kr	4.12E+07	4.94E+03	4.464%	Gas	4.94E+03	1.66E+01
131m-Xe	5.42E+05	4.76E+02	0.430%	Gas	4.76E+02	1.60E+00
133-Xe	9.61E+07	3.69E+04	33.325%	Gas	3.69E+04	1.24E+02
140-Xe	4.15E+07	4.98E+03	4.496%	Gas	4.98E+03	1.67E+01
135-Xe	2.44E+07	2.93E+03	2.644%	Gas	2.93E+03	9.83E+00
135m-Xe	1.94E+07	2.33E+03	2.102%	Gas	2.33E+03	7.81E+00
137-Xe	8.62E+07	1.03E+04	9.340%	Gas	1.03E+04	3.47E+01
138-Xe	8.13E+07	9.75E+03	8.809%	Gas	9.75E+03	3.27E+01
85-Kr	6.52E+05	7.82E+01	0.071%	Gas	7.82E+01	2.63E-01
85m-Kr	1.26E+07	1.51E+03	1.365%	Gas	1.51E+03	5.07E+00
87-Kr	2.41E+07	2.89E+03	2.611%	Gas	2.89E+03	9.71E+00
88-Kr	3.38E+07	4.05E+03	3.662%	Gas	4.05E+03	1.36E+01
129-I	2.09E+00	3.61E-05	0.000%	Halogen	3.61E-06	1.21E-08
131-I	4.83E+07	4.09E+03	3.696%	Halogen	4.09E+02	1.37E+00
132-I	6.98E+07	1.20E+03	1.089%	Halogen	1.20E+02	4.05E-01
133-I	9.87E+07	1.70E+03	1.540%	Halogen	1.70E+02	5.72E-01
134-I	1.08E+08	1.86E+03	1.685%	Halogen	1.86E+02	6.26E-01
135-I	9.24E+07	1.60E+03	1.442%	Halogen	1.60E+02	5.36E-01
84-Br	1.03E+07	1.78E+02	0.161%	Halogen	1.78E+01	5.97E-02
103-Ru	7.52E+07	1.30E+03	1.173%	Particulate	1.30E+02	4.36E-01
106-Ru	2.84E+07	4.90E+02	0.443%	Particulate	4.90E+01	1.65E-01
129-Te	1.53E+07	2.64E+02	0.239%	Particulate	2.64E+01	8.87E-02
132-Te	6.87E+07	1.19E+03	1.072%	Particulate	1.19E+02	3.98E-01
134-Cs	1.12E+07	1.93E+02	0.175%	Particulate	1.93E+01	6.50E-02
134-Te	8.16E+07	1.41E+03	1.273%	Particulate	1.41E+02	4.73E-01
137-Cs	7.22E+06	1.25E+02	0.113%	Particulate	1.25E+01	4.19E-02
138-Cs	9.02E+07	1.56E+03	1.407%	Particulate	1.56E+02	5.23E-01
140-Ba	8.49E+07	1.47E+03	1.325%	Particulate	1.47E+02	4.92E-01
140-La	9.10E+07	1.57E+03	1.420%	Particulate	1.57E+02	5.28E-01
88-Rb	3.44E+07	5.94E+02	0.537%	Particulate	5.94E+02	2.00E+00
89-Rb	4.41E+07	7.61E+02	0.688%	Particulate	7.61E+02	2.56E+00
89-Sr	4.57E+07	7.89E+02	0.713%	Particulate	7.89E+01	2.65E-01
90-Sr	5.24E+06	9.05E+01	0.668%	Particulate	9.05E+00	3.04E-02
90-Y	5.47E+06	9.44E+01	0.085%	Particulate	9.44E+01	3.17E-01
91-Sr	5.72E+07	9.87E+02	0.892%	Particulate	9.87E+01	3.32E-01
91-Y	5.92E+07	1.02E+03	0.924%	Particulate	1.02E+03	3.43E+00
95-Nb	8.17E+07	1.41E+03	1.275%	Particulate	1.41E+02	4.74E-01
95-Zr	8.12E+07	1.40E+03	1.267%	Particulate	1.40E+02	4.71E-01
99-Mo	8.85E+07	1.53E+03	1.381%	Particulate	1.53E+03	5.13E+00
H_3		1.38E+01		Gas	1.38E+01	0.100
Total	1.9E+09	1.1E+05	100%		8.7E+04	2.95E+02

11.1.1.4 Tritium Activity in Reactor Coolant

The analysis used for predicting the tritium activity in the reactor coolant consists of three parts listed as follows:

Coolant Activation

Tritium is produced in the coolant by the reactions shown in Table 11.1-8. The assumed concentration of the parent element for the reaction is also given.

Table 11.1-8 - "Tritium Reactions"

Reaction	Concentration of Target Material in Coolant
D $(n,\gamma)T$	150 ppm in hydrogen (naturally present in water)
B^{10} $(n, 2\alpha)T$	185*ppm in water (reactivity shim control)
B^{11} $(n, T)Be^{9}$	760*ppm in water (reactivity shim control)
Li^{7} $(n, nT)He^{4**}$	0.06 ppm in water (boron reaction product)*

* Concentration at beginning of life. The concentration is reduced by dilution (feed and bleed) throughout the core life in proportion to fuel burnup.

** Lithium 7 is the only isotope of lithium that is produced from the boron reactions.

Tritium from these sources account for 73.7% of the maximum concentration in the reactor coolant.

Fission

Tritium is also produced in the fuel as a fission product. Tritium production from fission is assumed to be one tritium atom per 1.25×10^4 fissions. The amount of tritium released to the coolant is based on operating the plant with 1% of the fuel failed.

Tritium from the fission source accounts for 25% of the maximum concentration in the reactor coolant.

Control Element Assemblies

Tritium is produced in the control element assemblies (CEA's). The tritium produced from the B4C in the control rods is based on the average number of control rods that are expected to be in the core during an operating cycle and a design value of 1% diffusion through the CEA cladding.

Tritium from this source accounts for 1.3% of the maximum concentration in the reactor coolant.

The production rates per core cycle are listed in Table 11.1-9.

Table 11.1-9 - "Production Rates in Reactor Coolant"

Source		Average Annual Activity, Ci
Coolant Activation		730
Fission		50
Control Element Assemblies		2
	Total	782

11.1.2 Liquid Wastes

11.1.2.1 Sources and Characteristics of Liquid Wastes

The liquid waste collection and storage system is divided into three sections; hydrogen bearing reactor coolant liquids, auxiliary systems process wastes, and hotel wastes. The original sources of liquid wastes and their routing to the collection points are shown in the flow diagrams, P&ID's 11405-M-6, 11405-M-7 and 11405-M-99.

Hydrogen Bearing Reactor Coolant Liquids

The principal sources for these liquids are:

a. Chemical and volume control system bleed for boron control;

b. Volume control tank relief and drains;

c. Pressurizer quench tank drains;

d. Reactor coolant loop drains;

e. Equipment drain header.

These liquids vary in composition, but approximate the reactor coolant in both chemical composition and activity.

Fuel transfer canal drains and safety injection system drains also enter the collection system, although they are not hydrogen bearing reactor coolant liquids. These liquids and the hydrogen bearing reactor coolant liquids are collected in three nitrogen blanketed tanks; the reactor coolant drain tank, the auxiliary building sump tank.

Auxiliary Systems Process Wastes

The principal sources for the liquids are:

a. Spent regenerate from deborating demineralizers;
b. Auxiliary building floor drain header;
c. Auxiliary building sump flows;
d. Laboratory and decontamination area drain header;
e. Spent resin sluice water;
f. Monitor tanks contaminated return flows;
g. Waste holdup tank relief valves;
h. Steam generator blowdown and secondary side drains (contaminated flows only);
i. Containment building sump flows.
j. Radioactive Waste Processing Building sump flows.
k. Chemical and Radiation Protection Building Laboratory drains.

Wastes from these sources are subject to contamination by reactor coolant. The drained liquids may be aerated prior to entering the waste disposal system and therefore, these wastes are collected in tanks that are not vented to the closed gas (nitrogen blanketed) circuit, due to possible oxygen contamination of the circuit. They are collected in the spent regenerant tanks which are vented to the auxiliary building ventilation system.

Aerated Domestic Wastes

The principal sources for these liquids are:

a. Laundry facility drains;

b. Shower drains;

c. Hand sink drains.

These wastes all originate in the auxiliary building and are transported in the laundry drain header which discharges to the hotel waste tanks. Aerated domestic wastes are normally low in activity.

11.1.2.2 Collection and Handling of Liquid Wastes

Hydrogen Bearing Reactor Coolant Liquids

The principal source for these liquid wastes is volume control tank bleed for boron control. Reactor coolant is "bled off" at the inlet valve of the volume control tank during the course of the plant operating cycle to reduce boron concentration as fuel is depleted. Other bleed-offs occur at this same point when heat-up of the reactor system produces an increase in coolant volume. The volume of waste entering the radioactive waste disposal system from this source is shown in Table 11.1-10.

Table 11.1-10 - "Reactor Coolant Waste Volumes"

Reactor Coolant Discharged to Waste Between Refuelings
Based on Postulated Events During an Equilibrium Core Cycle

	Event	Elapsed Time (equivalent full power days)	Waste Volume (liquid @ 70°F, ft³/event)
1.	Reactor refueled at 70°F through heatup to 570°F, initial full power and xenon equilibrium	2	4,230
2.	Cold shutdown No. 1 and restart following attainment of samarium equilibrium	23	3,278
3.	Hot shutdown No. 1 and restart	40	1,167
4.	Hot shutdown No. 2 and restart	120	1,717
5.	Hot shutdown No. 3 and restart	200	1,938
6.	Cold shutdown No. 2, partial drain for maintenance and restart	210	9,463
7.	Hot shutdown No. 4 and restart	280	5,671
8.	Initiate operation of deborating demineralizer No.3	307	--
9.	Cold shutdown	321	763
	Total from events		28,227
	Total from control of coolant boron concentration during 307 full power days		14,200
	Total per equilibrium cycle		42,427

Assumptions: (a) Base loaded plant; boron adjustment for load following is not required. (b) Reactivity effect of xenon during shutdown is not compensated by boron adjustment. Other reactor coolant type wastes are variable in flow and occur chiefly as periodic drains (such as the pressurizer quench tank drain), leak-offs, and occasional relief valve discharges. All liquid waste volumes are shown in Table 11.1-11.

Table 11.1-11 - "Liquid Waste Volumes"

		Volume liquid @ 70°F, (ft^3/321 full power days)	Remarks
1.	Reactor Coolant Wastes		
	Boron control	42,500	From Table 11.1-10
	Reactor coolant pump seal leak-offs	40	--
	CEDM leak-offs	730	Design value for RWDS purpose
	Charging pump seal leak-offs	3,000	--
	Stored energy safety injection tanks, check valve leak-offs	1,000	Flow to RWDS based on 0.1% leak-off
	Purification filters drain	20	2 replacements per cycle
	CVCS ion exchangers, drain and sluice water	400	3 parts sluice water per part resin
	Reactor coolant and CVCS sample wastes	10,000	Continuous analyzer operation plus normal sampling
	Valve leak-offs & safety relief) Valve discharge) Quench tank drain)	normally zero	--
2.	Spent Regenerant Chemicals		
	Deborating exchangers	700	Based on two regenerations per cycle
3.	Hotel Wastes	30,000	--
4.	Spent Fuel Pool Cooling System		
	Filter drain	50	Two replacements per cycle
	Ion exchanger drain and sluice water	60	--
5.	Radiochemical Lab Drains	-	Accounted for in sampling wastes
6.	Secondary Plant Steam Generator Blowdown	10,000	Normally zero Assumes discharge of water inventory of two steam generators per year.
	Total	98,500	

Reactor coolant liquids are collected as follows:

a. Reactor coolant drain tank (WD-1): This tank is the collection point for pressurizer quench tank drains, reactor coolant loop drains, CEDM leakage, safety injection system leakage, coolant pump seal leakage, and refueling pool drains. The tank is blanketed with nitrogen. Two pumps, automatically controlled by tank level, deliver these wastes to waste holdup tanks.

b. Auxiliary building sump tank (WD-25): This tank is the collection point for equipment drains in the auxiliary building (equipment drain header), and is provided with nitrogen blanketing. Two pumps can be automatically controlled by tank level to deliver these wastes to the waste holdup tanks.

c. Waste holdup tanks (WD-4A/B/C): These tanks receive the coolant wastes from the reactor coolant drain tank, spent regenerant tanks (WD-13A/B) and the auxiliary building sump tank. The function of these tanks is to provide temporary storage capacity. Three waste holdup tanks are provided, each capable of holding approximately one volume of reactor coolant in the reactor coolant system.

   When one of these tanks becomes filled, the waste flow is diverted to a second tank. The accumulated batch may be then thoroughly mixed by means of a recirculation pump. The recirculation pump is also capable of transferring the contents of one tank to another. Normally, the third waste holdup tank is on standby, ready to receive waste flow if the second tank becomes filled before the contents of the first tank have been discharged. These tanks are nitrogen gas blanketed.

Two waste holdup pumps take suction from the tanks and deliver the waste to the treatment inlet header or to the monitor tanks. The two pumps are manually controlled from the waste treatment control panel (see Section 7.6.3).

Auxiliary Systems Process Wastes

These wastes are collected in the spent regenerant tanks and include spent regenerant from the deborating demineralizers, floor drain header flows from the auxiliary building, sump flows from the auxiliary building, radioactive waste processing and containment buildings, and spent resin sluice water. The largest waste input occurs during the last few weeks of the refueling cycle when the deborating demineralizers in the chemical and volume control system are being regenerated. Gravity drains from the floor drain header and the drain header above floor elevation 971'-0" are collected directly in the spent regenerant tanks, whereas gravity drains from the sub-basement floor elevation and floor drains within the containment are collected in sumps and are delivered automatically by level-controlled pumps to the spent regenerant tanks.

Two spent regenerant tanks are provided and they are constructed of type 304 stainless steel due to the variety of liquids they might contain. Connection to the caustic dilution tank is provided for neutralization purposes, if required. The tanks are vented to the building ventilations exhaust system. Checked vent lines permit atmospheric inflow to the tanks on falling liquid level and exhaust to the ventilation system on rising liquid level.

A completed waste batch is normally delivered to the waste holdup tanks or the treatment inlet header. Delivery is made by two spent regenerant pumps, manually controlled, that also serve to mix the tank contents by recirculation. The tanks can also be transferred directly to the monitor tanks or the other spent regenerant tank or be recirculated and sampled if desired.

Hotel Wastes

These flows are chiefly from the laundry drain header, are usually low in activity, and are collected in the hotel waste tanks.

A filter has been placed in this line to prevent the passage of radioactive solids to the hotel waste tank from the laundry washers.

Two hotel waste tanks are provided, each designed to hold approximately one day's hotel waste flow. They are constructed of carbon steel, since corrosive liquids do not enter the laundry drain header. The tanks are simply vented to the atmosphere; there is no need for gas blanketing.

Mixing is accomplished by use of the hotel waste pumps as circulators, after which the waste batch is sampled and analyzed. The batch is then delivered to either the treatment inlet header or the monitor tanks or the overboard discharge header by the two manually controlled hotel waste pumps.

11.1.2.3 Liquid Waste Treatment

General

The RWDS is designed to provide filtration, evaporation, and demineralization in any combination, as needed to ready the waste for ultimate disposal. The process flow diagrams are shown in P&ID's 11405-M-8 and 11405-M-9.

Filtration

Suspended solids are removed by two waste filters. Solids are retained on the disposable filter element. Filter effluent is directed to the next treatment step or to the monitor tanks.

Filtration/Ion-Exchange

Filtration/ion-exchange (FIX) services are presently being used as the preferred method for liquid waste treatment and is located in the Radioactive Waste Processing Building.

The FIX system is designed to remove specific radioisotopes in the liquid waste stream.

The treated effluent from the FIX system is transferred to the monitor tanks.

Monitor Tanks

The two monitor tanks normally receive processed liquid wastes from the waste holdup tanks. The wastes are sampled and analyzed isotopically to confirm acceptability for controlled release to the overboard header. One tank can be undergoing recirculation for sampling while the other tank is being released to the overboard header.

11.1.2.4 Liquid Waste Disposal

During releases of radioactive liquid waste, the equipment and conditions shall be in accordance with the ODCM. The doses resulting from liquid releases shall not exceed, during any calendar year, 3 millirem to the total body (10 millirem to any organ) as required by 10 CFR Part 50 Appendix I.

The requirements for sample monitoring and testing prior to release and the requirements to ensure monitors are calibrated are included in the ODCM. Records of liquid releases must be maintained and are subject to the review, audits, and reporting requirements discussed in Section 11.3.

The overboard header is the only path through which the liquid rad wastes can be released from the containment, auxiliary, Radioactive Waste Processing and CARP buildings. It receives liquid from the monitor tanks, the hotel waste tanks, or blowdown from the steam generators. The overboard header originates at the monitor tanks or the hotel waste tanks and terminates in the condenser circulating water discharge tunnel, entering the tunnel in the section downstream of the warm water recirculation return (see P&ID 11405-M-257). Effluent from the monitor tanks or the hotel waste tanks is moved by two monitor tank pumps or hotel waste pumps and the flow rate is monitored on a recorder. The steam generator blowdown is controlled and monitored and recorded in accordance with the ODCM prior to the overboard header.

The overboard header is equipped with a radiation monitor that interrupts flow if waste activity reaches a predetermined setpoint (see Section 11.2.3).

11.1.2.5 System Components

The various components of the RWDS are divided into three groups for convenience of listing; tanks, pumps, and process equipment. These are shown in Tables 11.1-12, 13, and 14.

Table 11.1-12 - "Component Design Data, Waste Disposal System Tanks "

Tank	No. Installed/ Item No.	Tank Capacity gallons/ft^3	Pressure, psig Design/ Operating	Temperature °F Design/ Operating	Material*	Code
Reactor Coolant Drain Tank	1/WD-1	900/120	25/2	300/267	304 SS	ASME Section III, Class C, Feb. 1968
Waste Holdup Tanks	3/WD-4A, B&C	45,800/6,100	15/2	200/120	CS	ASME Section III, Class C, Feb. 1968
Spent Regenerant Tanks	2/WD-13A&B	5,530/739	5/Atmos	200/70	304 SS	ASME Section VIII, Feb. 1968
Hotel Waste Tanks	2/WD-15A&B	1,200/160	15/Atmos	200/140	CS	ASME Section VIII, Feb. 1968
Monitor Tanks	2/WD-22A&B	6,770/905	5/Atmos	200/140	304 SS	ASME Section VIII, Feb. 1968
Auxiliary Building Sump Tank	1/WD-25	700/95	25/2	200/120	304 SS	ASME Section VIII, Feb. 1968
Gas Decay Tank	4/WD-29A,B, C&D	3,571/477	150/100	200/140	CS	ASME Section III, Class C, Feb. 1968
Spent Resin Storage Tank	1/WD-33	3,250/434	25/2	250/120	304 SS	ASME Section VIII, Feb. 1968
Waste Metering Tank	1/WD-46	688/92	Atmos	-	316 SS	ASME Section VIII, Feb. 1968

* SS= Stainless Steel, CS= Carbon Steel

Pumps were in accordance with the Standards of the Hydraulic Institute and all motors conformed to NEMA standards. Materials were in accordance with the appropriate ASTM specifications. Other codes and standards are listed in the tables referenced above.

Table 11.1-13 - "Component Design Data, Waste Disposal System Pumps"

Pump	No. Installed/ Item No.	Type	Capacity	Fluid Side Material*
Reactor Coolant Drain Tank Pumps	2/WD-2A&B	Horizontal Centrifugal	2A, 250 gpm @ 75 ft. 2B, 50 gpm @ 75 ft.	316 SS 316 SS
Containment Sump Pumps	2/WD-3A&B	Vertical Centrifugal	50 gpm @ 40 ft.	AI
Waste Holdup Tank Pumps	2/WD-5A&B	Horizontal Centrifugal, Canned Rotor	50 gpm @ 177 ft.	316 SS
Waste Holdup Recirculation Pump	1/WD-6	Horizontal Centrifugal	500 gpm @ 85 ft.	AI
Spent Reg. Pumps	2/WD-14A&B	Horizontal Centrifugal	50 gpm @ 157 ft.	304 SS
Hotel Waste Pumps	2/WD-16A&B	Horizontal Centrifugal	50 gpm @ 130 ft.	AI

* AI = All Iron
SS = Stainless Steel
CS = Carbon Steel

Table 11.1-13 (Continued)

Pump	No. Installed/ Item No.	Type	Capacity	Fluid Side Material*
Monitor Tank Pumps	2/WD-23A&B	Horizontal Centrifugal	50 gpm @ 160 ft.	304 SS
Auxiliary Bldg. Sump Tank Pumps	2/WD-26A&B	Horizontal Centrifugal	35 gpm @ 110 ft.	304 SS
Auxiliary Bldg. Sump Pumps	6/WD-27A&B, 40A&B, 41A & B	Vertical Centrifugal	20 gpm @ 36 ft.	CI
Spent Resin Pump	1/WD-34	Horizontal Centrifugal	30 gpm @ 106 ft.	304 SS
Radioactive Waste Processing Bldg. Sump Pumps	4/WD-30A&B, WD/31A&B	Vertical Centrifugal	65 gpm @ 40 ft.	304SS

* SS = Stainless Steel
CS = Carbon Steel
CI = Cast Iron

Table 11.1-14 - "Component Design Data, Waste Disposal System Process Equipment"

Waste Filters, Item No's WD-17A&B	Description
Number	2
Type	Expendable element pressure type
Materials of Construction	304 stainless steel vessel
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	250
Vessel Code	ASME Section III, Class C, Feb. 1968
Flow Rate (filter), each, gpm	150
Average Efficiency, % (particles 50 microns)	43

Filtration and Ion-Exchangers	
Number	6
Type	Sluiceable vessel, disposable resin/media
Materials of Construction	304 L SS
Design Pressure, psig	150
Design Temperature, °F	130
Operating Pressure, psig	50
Operating Temperature, Max.°F	125
Vessel Code	ASME Section VIII
Vessel volume	1 - 69 ft^3, 5 - 30 ft^3
Flow Rate, Max gpm	50

Waste Gas Analyzer Item No. AI-110	
Type	Membranes
Determinations	Oxygen Content Hydrogen Content
Number of Stations Scanned	16

11.1.2.6 System Operation

The operation of the liquid waste section of the RWDS involves a combination of automatic and manual controls. The flow of liquids from two of the collection tanks (reactor coolant drain tank, and the auxiliary building sump tank) and the four drain sumps can be controlled automatically by liquid level. The control panels are described in Section 7.6.3.

At the waste holdup tanks, the hotel waste tanks, the spent regenerant tanks, and the monitor tanks, the operator must decide where to send the contents of a tank. The operator can send it through various tanks, filters, or the Filtration Ion Exchange System, depending on the processing required. Therefore, the flow leaving these tanks is manually controlled at the waste disposal control panel.

The waste filters are equipped with differential pressure indication and the filters are replaced when a predetermined pressure drop is reached.

The filtration/ion exchange system is designed to provide any flow logic through the system's pressure vessels. The flow logic is dependent upon the type of waste to be processed and is accomplished by manually valving the hose setup between vessels.

11.1.2.7 Design Evaluation

The anticipated performance of the liquid waste system has been calculated in accordance with the following assumptions.

The maximum annual quantity of liquid waste containing significant activity is approximately 98,500 cu. ft. As shown in Table 11.1-10, 42,500 cu. ft. of the total liquid waste is from the chemical and volume control system and has already passed through the purification ion exchangers. The activity of this liquid waste is assumed to be reduced by a factor of 10 for each nuclide except rubidium, molybdenum, noble gases, corrosion products and tritium for which a factor of unity has been assumed. An additional volume of 15,190 cu. ft., shown in Table 11.1-11, has an activity equal to that of reactor coolant. Hotel wastes are low in activity and with the addition of a filter on the discharge from the laundry washers, which collects radioactive solids, will remain low in activity at discharge to the hotel waste tanks. Waste volumes resulting from steam generator blowdown while normally zero, have been estimated on the basis that primary-to-secondary leakage requires that the zero load liquid inventory of both steam generators (6,000 cu. ft.) is discharged to the RWDS once per year and that the activity is consistent with having operated for 45 days with a 1 gph primary-secondary leak and one percent fuel failure.

The two waste filters are designed to remove insoluble corrosion products, some of which may be radioactive. However, no credit has been assumed for these filters in the system evaluation. The Filtration Ion Exchange System average total decontamination factor is 364. The normal liquid waste holdup time is 30 days. The fission and corrosion product activities in the liquid waste treatment system are shown in Table 11.1-15.

Table 11.1-15 - "Fission and Corrosion Product Activity in the Waste Treatment System at STP"

Nuclide	As Received µCi/cc	After 1 Day ** µCi/cc	After 30 Days ** µCi/cc
Xe-131m	1.60 E+0	1.51 E+0	2.76 E-1
Xe-133	1.24 E+2	1.09 E+2	2.35 E+0
Xe-135	9.83 E+0	1.59 E+0	1.69 E-23
Xe-135m	7.81 E+0	5.74 E-28	0.00 E+0
Xe-137	3.47 E+0	2.35 E-113	0.00 E+0
Xe-138	3.27 E+1	6.74 E-30	0.00 E+0
Kr-85	2.63 E-1	2.63 E-1	2.62 E-1
Kr-85m	5.07 E+0	1.32 E-1	1.49 E-47
Kr-87	9.71 E+0	2.03 E-5	3.92 E-170
Kr-88	1.36 E+1	4.22 E-2	7.52 E-75
Kr-89	1.66 E+1	5.80 E-137	0.00 E+0
I-129	4.08 E-9	4.08 E-9	4.08 E-9
I-131	4.62 E-1	4.24 E-1	3.48 E-2
I-132	1.36 E-1	1.33 E-4	6.70 E-92
I-133	1.93 E-1	8.69 E-2	8.03 E-12
I-134	2.11 E-1	1.19 E-9	7.98 E-249
I-135	1.81 E-1	1.52 E-2	1.01 E-33
Br-84	2.01 E-2	4.81 E-16	0.00 E+0
Ru-103	1.47 E-1	1.44 E-1	8.66 E-2
Ru-106	5.56 E-2	5.55 E-2	5.25 E-2
Te-129	2.99 E-2	1.75 E-8	3.16 E-189
Te-132	1.34 E-1	1.08 E-1	2.28 E-4
Te-134	1.59 E-1	6.64 E-12	0.00 E+0
Cs-134	2.19 E-2	2.19 E-2	2.13 E-2
Cs-137	1.41 E-2	1.41 E-2	1.41 E-2
Cs-138	1.76 E-1	6.42 E-15	0.00 E+0
Ba-140	1.66 E-1	1.57 E-1	3.26 E-2
La-140	1.78 E-1	1.18 E-1	7.49 E-7
Rb-88	6.74 E-1	3.56 E-25	0.00 E+0
Rb-89	8.63 E-2	6.34 E-30	0.00 E+0
Sr-89	8.93 E-2	8.81 E-2	5.92 E-2
Sr-90	1.02 E-2	1.02 E-2	1.02 E-2
Y-90	1.07 E-1	8.24 E-2	4.43 E-5
Sr-91	1.12 E-1	1.98 E-2	2.96 E-24
Y-91	1.16 E+0	1.14 E+0	8.10 E-1
Nb-95	1.60 E-1	1.57 E-1	8.83 E-2
Zr-95	1.59 E-1	1.57 E-1	1.15 E-1
Mo-99	1.73 E+0	1.34 E+0	8.99 E-4
H-3	1.00 E-1	1.00 E-1	9.95 E-2

****NOTE:** All noble gases are assumed to be released from solution immediately after entering the LRWS.

Anticipated annual quantities of liquid waste releases and the corresponding annual average concentrations in the discharge tunnel are given in Table 11.1-16 for those nuclides expected to have annual average concentrations greater than 1×10^{-20} µCi/cc. As illustrated by the table, it is expected that no single nuclide will exceed 1 percent of 10 CFR Part 20 limits on an annual average basis. Cumulative dose contributions from radioactive materials in liquid effluents released to unrestricted areas shall be determined on a quarterly basis in accordance with the ODCM. The total annual average concentration of liquid wastes discharged, excluding tritium, is not expected to exceed 1.13 E-9 µCi/cc. The expected annual average concentration of tritium in the discharge tunnel is approximately 1.29 E-6 µCi/cc.

For the purposes of calculating the anticipated concentrations, an annual average discharge tunnel flow of 305,000 gpm was used. This average flow was obtained by assuming the use of two circulating water pumps and one raw water pump during six cold months of the year and use of three circulating water pumps and one raw water pump during the six warmer months.

Effluents shall be limited to ten times 10 CFR Part 20, Appendix B, Table 2, Column 2 concentrations at discharge. While the exact circumstances that could lead to the closest approach to this limit are difficult to predict, hypothetical cases can be postulated. For example, if the entire waste batch was initially at the reactor coolant activity (Table 11.1-7), the waste evaporator decontamination factor was degraded to 10^3, and the performance of other components was as previously assumed, except that the discharge flow rate was 15 gpm and the circulating water flow rate was 120,000 gpm, the total concentration in the discharge tunnel would be 5.3 E-7 µCi/cc. This release concentration would be below the guidelines of 10 CFR Part 20.

Calculations have been made to determine the downstream concentration of radionuclides discharged in the circulating water discharge from the Fort Calhoun Station into the Missouri River. These calculations were based on a model developed and experimentally verified by Yotsukura, Fischer and Sayre in: "Measurements of Mixing Characteristics of the Missouri River between Sioux City, Iowa, and Plattsmouth, Nebraska, U. S. Geological Survey Water Supply paper 1899-G, U. S. Government Printing Office, Washington: 1970". The computer code described in this publication was obtained by OPPD and its applicability confirmed by comparison with experimental data contained in the paper for a center-of-stream source of dye and its dispersion in the river reach adjacent to the plant site.

The calculated maximum concentration of wastes is shown in Figure 11.1-1 as a function of distance. Conditions are shown for a maximum distance of 19.5 miles, which corresponds to the location of the municipal water intake for the city of Omaha.

The source is assumed to be a continuous release of material from the bank which is uniformly mixed with 5% of the total river discharge and the 5% stream tube has the same concentration from the point of injection to 200 feet downstream.

Table 11.1-16 - "Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel"

Nuclide	Total Quality Released, Ci	Average Annual Conc. (μCi/cc)	New 10CFR20 Limits Appendix B Table II, Col. 2(μCi/cc)
Xe-131m	4.51 E-2	7.44 E-11	NA
Xe-133	3.84 E-1	6.33 E-10	NA
Kr-85	4.27 E-2	7.04 E-11	NA
I-129	6.66 E-10	1.10 E-18	2 E-7
I-131	5.68 E-3	9.36 E-12	1 E-6
I-133	1.31 E-12	2.16 E-21	7 E-6
Ru-103	1.41 E-2	2.33 E-11	3 E-5
Ru-106	8.58 E-3	1.41 E-11	3 E-6
Te-132	3.72 E-5	6.13 E-14	9 E-6
Cs-134	3.48 E-3	5.74 E-12	9 E-7
Cs-137	2.30 E-3	3.79 E-12	1 E-6
Ba-140	5.33 E-3	8.78 E-12	8 E-6
La-140	1.22 E-7	2.02 E-16	9 E-6
Sr-89	9.67 E-3	1.59 E-11	8 E-6
Sr-90	1.67 E-3	2.75 E-12	5 E-7
Y-90	7.24 E-6	1.19 E-14	7 E-6
Y-91	1.32 E-1	2.18 E-10	8 E-6
Nb-95	1.44 E-2	2.38 E-11	3 E-5
Zr-95	1.87 E-2	3.09 E-11	2 E-5
Mo-99	1.47 E-4	2.42 E-13	2 E-5
H-3	7.82 E+2	1.29 E-6	1 E-3

Total Annual Average Concentration (excluding Tritium) = 1.13 E-9
Total of 10 CFR 20 Fractions = 1.3 E-3
Total Concentration at Discharge Tunnel (Bounding Case) = 5.27 E-7

The contribution of steam generator blowdown to the total liquid waste activity will be very small, since it is intended to secure blowdown if the second monitor setpoint is reached. This would happen about twelve hours after initiation of a 1 gph primary-to-secondary leak, if the coolant activity were consistent with 1 percent fuel failures. Assuming the plant was then operated for forty-five days with blowdown secured, and then the contents of the secondary sides of the steam generators were discharged to the waste plant so that the leak could be repaired, the quantities of activity discharged to the radioactive waste system would be as given in Table 11.1-17.

Table 11.1-17 - "Secondary Side Activity Released to Liquid Waste System"

Nuclide	STP Coolant µCi/cc 1%	SG Concentration after 45 days (µCi/cc)
I-129	1.21 E-8	4.95 E-8
I-131	1.37 E+0	1.41 E+0
I-132	4.05 E-1	5.31 E-3
I-133	5.72 E-1	6.52 E-2
I-134	6.26 E-1	3.00 E-3
I-135	5.36 E-1	1.97 E-2
Br-84	5.97 E-2	1.73 E-4
Ru-103	4.36 E-1	1.23 E+0
Ru-106	1.65 E-1	6.47 E-1
Te-129	8.87 E-2	5.62 E-4
Te-132	3.98 E-1	1.70 E-1
Te-134	4.73 E-1	1.80 E-3
Cs-134	6.50 E-2	2.60 E-1
Cs-137	4.19 E-2	1.71 E-1
Cs-138	5.23 E-1	1.54 E-3
Ba-140	4.92 E-1	7.53 E-1
La-140	5.28 E-1	1.16 E-1
Rb-88	2.00 E+0	3.25 E-3
Rb-89	2.56 E-1	3.59 E-4
Sr-89	2.65 E-1	8.09 E-1
Sr-90	3.04 E-2	1.24 E-1
Y-90	3.17 E-1	1.11 E-1
Sr-91	3.32 E-1	1.74 E-2
Y-91	3.43 E+0	1.09 E+1
Nb-95	4.74 E-1	1.28 E+0
Zr-95	4.71 E-1	1.52 E+0
Mo-99	5.13 E+0	1.85 E+0
H-3	1.00 E-1	4.07 E-1

11.1.2.8 Availability and Reliability

The liquid waste system is not dependent on a fixed or normal method of operation of the reactor coolant system or the chemical and volume control system but will function properly with wide variations in these two systems. For example, the system is designed to handle the large volume of boron control bleed needed at hot or cold startups as well as the comparatively small volume of bleed while operating at a constant power level.

The liquid waste process equipment is dependent on the electrical systems, the demineralized water system and on the nitrogen gas system for tank blanketing. Collection of waste is chiefly by gravity and is therefore, almost wholly independent of auxiliary systems.

The liquid waste system has a duplicate sampling and analyzing capability. Liquid waste is analyzed at the waste hold-up tanks and then again at the monitor tanks, thus ensuring that effluent to the overboard header has always had two independent analyses. In addition, the radiation monitor at the overboard header automatically stops this flow if it exceeds a pre-determined concentration of radioactivity.

The transport pumping sets in the liquid waste system have redundancies, with one of the two pumps being a spare for the other.

Redundant volume is provided in the waste holdup tanks, spent regenerant tanks, and the hotel waste tanks; two tanks are furnished for spent regenerant and two for hotel wastes whereas three tanks are furnished for waste holdup. In the case of two tanks, the second is normally a complete spare of the first in volume capacity. In the case of three tanks, the capacity of 1-1/2 tanks is spare volume. The usual mode of operation is for one tank to be collecting while another tank is being discharged to treatment.

11.1.2.9 Operation

The liquid waste processing system is operated to minimize the amount of radioactivity contained in liquid effluents from the plant. A program of equipment operation and maintenance will be in effect to provide maximum system availability. Only under unusual circumstances of severe need would a system component be bypassed if it could, within detectable limits, significantly reduce the activity of the waste liquid. Waste liquids are segregated as to radioactivity level and point of origin. Under normal operating conditions highly radioactive liquid wastes are held for sufficient duration to allow decay of short-lived radioactive nuclides prior to processing and release.

Hotel waste tanks are normally diverted for processing if the activity level is above the limits established for release. All liquids are sampled and analyzed prior to release.

Steam generator blowdown will be stopped if an alarm setpoint on either blowdown monitor (RM-054A or B) is exceeded.

System flexibility ensures that proper treatment brings waste quantities and activities well within the limits of 10 CFR Part 20 and 40 CFR 190. In addition to this flexibility, it is possible to reprocess any volume of liquid if this need should occur.

The radiation monitors may be inoperable and liquid releases may continue provided the requirements of the ODCM are complied with. All liquid radioactive wastes originating within the containment, CARP and Radioactive Waste Processing Building are pumped to the auxiliary building. All radioactive liquids in the auxiliary building are collected in the RWDS. The radiation monitors utilized for monitoring RWDS are described in Section 11.2.3.

11.1.2.10 Tests and Inspections

The purpose of the testing and inspection program was to ensure that the liquid waste system components meet design objectives and specifications.

All equipment in the system was subject to two types of test and inspections: manufacturer's shop tests and on-site tests.

<u>Shop Tests</u>

All equipment was tested and inspected in the manufacturer's shop in accordance with the then applicable codes and standards. In addition, some equipment was given performance type tests in the manufacturer's shop.

After preliminary operation to demonstrate the mechanical integrity and suitability of all components, a short term test program to demonstrate specific modes and methods of operation was undertaken. Chemical tests, such as boron concentration, and operating parameters, such as flow rates, were recorded during the test program. After the successful completion of the above tests, the equipment was partially disassembled and shipped to the plant site.

On-Site Tests

On-site tests of the performance type to ensure that the overall liquid waste system functions in a safe and efficient manner were conducted prior to actual plant startup. Provisions were made to test the full operational sequence of the system. Pumps were started, valves operated, and instruments put into service. Flow paths, flow capacity, and mechanical operability were thoroughly checked. Pressure, temperature, flow and level indicating instruments were calibrated and checked for performance. All safety equipment, including alarms were thoroughly tested. Special emphasis was placed on the proper functioning of the liquid waste instrumentation and controls on the waste control panel.

11.1.3 Gaseous Wastes

11.1.3.1 General

Radioactive waste gases are collected, compressed, stored, analyzed, and monitored in the radioactive waste disposal system. Waste gas found to be suitable for discharge in accordance with the requirements set forth in 10 CFR Part 20 are released under controlled conditions to the auxiliary building ventilation system for dilution prior to discharge at the plant stack (see Section 9.10). A radiation monitor in the plant stack (see Section 11.2.3) automatically interrupts the flow of waste gas in the gas discharge header if the activity reaches a predetermined concentration. The calculated annual air dose at any location which could be occupied by individuals in unrestricted areas shall not exceed 10 millirads for gamma radiation, 20 millirads for beta radiation and 15 millirems to any organ for iodine-131, tritium, and other particulates with half-lives greater than eight days as required by 10 CFR Part 50, Appendix I.

The methods of dose calculation are defined in the Offsite Dose Calculation Manual.

Additional amounts of radioactive gases may exist in relatively low concentrations in the containment and auxiliary building, where the gases can evolve from unconfined leakage of reactor coolant, and also in the condenser air ejector discharge, the vent from the blowdown flash tank, and turbine building exhaust under conditions when primary to secondary leakage exists coincident with fuel clad defects. The concentrations are too dilute and the volumes of carrier gases too large to permit collection and storage. However, the amounts of radioactivity released in low concentration waste gas will be known and releases will be terminated if the activity reaches predetermined limits.

There may be small amounts of radioactive gas in the Radioactive Waste Processing and CARP buildings. The amount of gas will be extremely low and releases will be measured and recorded.

The annual average dispersion factor (x/Q) for gaseous releases used to determine exposures in the unrestricted area is calculated using data obtained from the meteorological program. This program is described in detail in section 2.5. The annual average value of x/Q is specified in the ODCM. A revision of this value, either due to subsequent data or revised criteria, would affect the gaseous release concentration in direct ratio to the change. The ODCM ensures that all releases are within applicable criteria.

11.1.3.2 Sources of Waste Gas

Radioactive gases, normally present in trace amounts in reactor coolant liquids, collect in the vapor space above the various tanks and components as the liquid becomes depressurized. Hydrogen gas, used for corrosion control in the CVCS, enters the coolant in the volume control tank. Nitrogen gas is used to blanket the tanks and components, thereby greatly diluting the hydrogen and radioactive gases. As a tank fills, or a component operates, the gases occupying the vapor space are forced into the vent header (VH), where they are then known as waste gases. Table 11.1-18 lists the tanks and equipment that are waste gas sources.

Table 11.1-18 - "Waste Gas Sources"

Source	Operation
Pressurizer Quench Tank	N_2 gas blanket and intermittent purge
Reactor Coolant Drain Tank	N_2 gas blanket
Volume Control Tank	H_2 gas in vapor space during normal power cycle, N_2 prior to shutdown
Waste Holdup Tanks	N_2 blanket
Spent Resin Storage Tank	N_2 blanket, N_2 mix
Auxiliary Building Sump Tank	N_2 blanket
Gas Decay Tanks	N_2 purge
Automatic Gas Analyzer	Waste gas vent

Table 11.1-19 lists the constituents present in the waste gas system.

Table 11.1-19 -"Waste Gas Constituents"

	Concentration by Volume
Nitrogen	Background
*Hydrogen, %	Trace to 3 max
*Oxygen, %	Trace to 3 max
Radioactive Gases (xenon and krypton)	Trace
Water Vapor	Saturated
Other gases used for leak testing	See paragraph 11.1.3.10

* Hydrogen, depending on the amount of reactor coolant leakage or plant evolutions in progress such as degassing, can exceed 3% concentration in the waste gas. However, hydrogen and oxygen gas concentrations will not exceed 3% at the same time.

11.1.3.3 Processing of Waste Gases

Waste gases from all of the sources mentioned above are collected in the vent header as shown in the process flow diagram P&ID 11405-M-98. Two waste gas compressors take suction from the vent header, compress the gas, and then deliver it to one of the four gas decay tanks. Normally, when the vent header exceeds 2 psig, one of the two waste gas compressors is started to deliver the gas to a decay tank. The second compressor will be started if the waste gas flow exceeds the capacity of the operating compressor. The compressors will be run as required to reduce the vent header pressure to less than 2.0 psig. The waste gas can be compressed to 100 psig (nominal) in a gas decay tank, and then discharged on a batch basis.

The procedure for processing a waste gas batch is as follows:

a. Fill operation: A decay tank, initially at atmospheric pressure is pressurized to 100 psig (nominal) during the tank fill period. Upon, or prior to reaching 100 psig, the inlet pressure control valve and its manual inlet isolation valve are shut, and another waste gas decay tank is selected and placed in service.

b. Analysis: Analysis of the contents of a filled decay tank determines whether a batch of waste gas must be retained to permit radioactive decay or is suitable for controlled release to the atmosphere.

c. Controlled release: The contents of a decay tank can be held for the decay of short-lived radioactive gases. A batch found acceptable for discharge is released by manually opening the tank outlet valve and a block valve in the gas discharge header. Two parallel mounted split - range flow control valves in the discharge header, controlled by a microprocessor controller with temperature and pressure compensation automatically limit the discharge rate to the exhaust ventilation system to a preset rate to maintain the effluent gases at or below required activity limit during release.

A radiation recorder-controller (see Section 11.2.3) monitors the Auxiliary Building ventilation system exhaust for gaseous activity and automatically closes a control valve in the gas discharge header on high concentration of activity. A permanent record of waste gas released is obtained from the flow recorder-controller in the gas discharge header.

11.1.3.4 Gas Re-Use Option

Accumulated batches of low activity waste gas may be returned from a decay tank to a re-use header and thence to the waste holdup tank area. This option is not normally utilized due to possible O_2 contamination. If the option is used, the waste gas is split into two lines at this point; one line serves for tank blanketing of the waste holdup tanks whereas the other line supplies gas for sparging of the tanks. Gas sparging helps in mixing and also assists in partial degasification. Tank blanketing with re-use gas conserves nitrogen. The nitrogen supply for blanketing is normally used and would automatically flow into the tanks when the re-use gas flow subsides, if the option was used.

In addition to low activity, the waste gas batch must be 99 percent or greater nitrogen and essentially free of oxygen and oil vapor in order to be suitable for re-use.

11.1.3.5 Waste Gas Analyzer (AI-110)

The gas space in all tanks and equipment utilizing hydrogen gas can be monitored for hydrogen and oxygen gas content. A sixteen channel sampling system is provided (one channel is a nitrogen gas purge). The system is designed to sample one channel at a time. The waste gas analyzer panel is located in the auxiliary building adjacent to the waste disposal system control panel.

The waste gas analyzer system also provides grab samples from the 16 channels to an explosion proof hood adjacent to the waste gas analyzer panel. The grab samples are then analyzed in the hot lab.

11.1.3.6 System Components

The major components of the gaseous wastes section of the RWDS are as follows; the referenced tables summarize pertinent data:

a. Gas decay tanks (see Table 11.1-12);

b. Waste gas compressors (see Table 11.1-14);

c. Automatic gas analyzer (see Table 11.1-14).
Materials are in accordance with the appropriate ASTM specifications.

11.1.3.7 Design Evaluation

Relatively Highly Concentrated Gaseous Wastes

The volume of gaseous waste consists primarily of nitrogen, with concentrations of up to 3 percent hydrogen, and trace amounts of oxygen (see Table 11.1-19), xenon, krypton, ammonia and water vapor.

The vent header is the collection point for all waste gases, and is normally operated at a low pressure of 1/2 to 2 psig. The high nitrogen content of the waste gases prevents the formation of explosive mixtures of H_2 and O_2. This same high nitrogen content greatly dilutes any gaseous activity that may be contributed by xenon and krypton, in addition to acting as an inert carrier gas for all waste gases.

The compression of the gases leaving the vent header results in an increase in effective gas storage volume by about 7 times due to the pressure increase from approximately 16 psia to 115 psia.

Each of the four gas decay tanks has a volume of 400 cubic feet, giving a total volume of 1,600 cubic feet. When multiplied by the compression factor of 7.2 this results in a total storage volume of 11,520 cubic feet of waste gas as it is received in the radioactive waste disposal system. A waste gas volume of 4,800 standard cubic feet is produced during the cold shutdown assumed to occur at day 210 of the power cycle. This volume results from reactor coolant degassing. Nitrogen blanket displacement and H_2 removed from the reactor coolant system requires the holding capacity equal to approximately two and one-half gas decay tank volumes. Monitoring of the hydrogen and oxygen content of the gas decay tanks is required during waste gas transfer per TS 2.9. Daily channel checks of the hydrogen and oxygen monitoring systems are required only when the monitors are in service. (Reference 11-10)

At the above average waste gas generation rate and assuming that one gas decay tank must always be in the fill position ready to accept waste gas, three tanks provide an average holdup or decay period of 59 days. However, in estimating discharges of gaseous wastes from the plant, it has been conservatively assumed that the holdup time is 30 days.

The accumulated waste gas volumes during one cycle are shown in Table 11.1-20.

Table 11.1-20 - "Waste Gas Volumes"

Source		Volume ft^3 @ STP/cycle
Reactor Coolant Liquids		
Degasification of reactor coolant prior to cold shutdown (1)		3,024
Off-gas released from reactor coolant waste liquid (startups, shutdowns and boron control) (2)		1,270
Nitrogen Blanket Gas Displacement		46,535
	Total	50,829

(1) Based on three cold shutdowns per cycle and six volumes of purge gas (N_2) applied per volume of off-gas removed from coolant.

(2) Off-gas consisting of nitrogen, hydrogen, ammonia and fission gases released in the ratio of 30cc gas/Kg liquid waste.

Table 11.1-21 shows the activities of the gaseous waste in the treatment system with maximum coolant activity for the 1 percent failed fuel condition. The effect of decay on gaseous activities and total annual releases assuming a 30-day holdup are also shown. After 30 days holdup, there is negligible activity from the noble gas daughter products. There are small quantities of I-131 and particulates with long half lives present. The values shown in the table correspond to a DF of 1,000 in the volume control tank for halogens and particulates. The release normally goes through HEPA and charcoal filters.

Maximum Activity in a Gas Decay Tank

The maximum activity of a batch of waste gas initially introduced into a gas holdup tank can reach 16,900 curies. After the normal holdup time of 30 days the radioactive gas ultimately released would be mainly Kr-85 and Xe-133, with total activity of about 481 curies.

Table 11.1-21 - "Gaseous Activity in Waste Treatment System"

Nuclide	Specific Activity to Decay Tanks μCi/cc	Specific Activity After 30 Days μCi/cc	Annual Release From Decay Tanks Ci
Xe-131m	1.35 E-1	2.33 E-2	3.36 E+1
Xe-133	1.05 E+1	1.99 E-1	2.86 E+2
Kr-85	2.22 E-2	2.21 E-2	3.18 E+1
I-129	1.02 E-12	1.02 E-12	1.47 E-9
I-131	1.16 E-4	8.71 E-6	1.25 E-2
I-133	4.83 E-5	2.01 E-15	2.90 E-12
Ru-103	3.68 E-5	2.17 E-5	3.13 E-5
Ru-106	1.39 E-5	1.32 E-5	1.90 E-5
Te-132	3.36 E-5	5.71 E-8	8.21 E-8
Cs-134	5.49 E-6	5.34 E-6	7.69 E-6
Cs-137	3.54 E-6	3.53 E-6	5.09 E-6
Ba-140	4.16 E-5	8.18 E-6	1.18 E-5
La-140	4.46 E-5	1.88 E-10	2.70 E-10
Sr-89	2.24 E-5	1.48 E-5	2.14 E-5
Sr-90	2.57 E-6	2.56 E-6	3.69 E-6
Y-90	2.68 E-5	1.11 E-8	1.60 E-8
Y-91	2.90 E-4	2.03 E-4	2.92 E-4
Nb-95	4.00 E-5	2.21 E-5	3.18 E-5
Zr-95	3.98 E-5	2.88 E-5	4.14 E-5
Mo-99	4.33 E-4	2.25 E-7	3.24 E-7

Total Initial Concentration to Decay Tank = 1.90 E+1

Total Concentration after 30 Days = 2.45 E-1

Radioactive Gases Released from the Secondary System

If primary-to-secondary system leakage (in the steam generator for example) exists coincident with failed fuel, noble gases and halogens will be released from the air ejector discharge. To estimate the amount released, it has been assumed that the plant is operated for 45 days in succession once per year with a primary-to-secondary leak rate of 1 gph and with 1 percent failed fuel. All of the noble gases contained in the leakage flow and a small fraction of the halogens are assumed to be released. The halogen release fraction has been computed on the following bases. The partitioning coefficients between the gas and liquid phases in the steam generator and condenser are in accordance with references 11-1 and 11-2, respectively. The air ejector flow is 20 cfm. It is further assumed that the steam leakage to the turbine building is 100 pounds per hour.

It is assumed that the steam generator blowdown is secured as soon as the second setpoint of the blowdown monitor is reached. For the postulated conditions described above, this would occur within approximately 12 hours. The average blowdown flow rate over this period is assumed to be 1 gpm from each steam generator. The release of halogens to the atmosphere is assumed to be one-tenth of what is in the portion of the blowdown flow that flashes. The estimated release rates are listed in Table 11.1-22.

Table 11.1-22 - "Annual Gaseous Releases from Secondary System"

Nuclide	Specific Activity at STP (µCi/cc)	Total Ci Released from Secondary Side
Xe-131m	1.60 E+0	6.54 E+0
Xe-133	1.24 E+2	5.07 E+2
Xe-135	9.83 E+0	4.02 E+1
Xe-135m	7.81 E+0	3.19 E+1
Xe-137	3.47 E+0	1.42 E+1
Xe-138	3.27 E+1	1.34 E+2
Kr-85	2.63 E-1	1.08 E+0
Kr-85m	5.07 E+0	2.07 E+1
Kr-87	9.71 E+0	3.97 E+1
Kr-88	1.36 E+1	5.56 E+1
Kr-89	1.66 E+1	6.79 E+1
I-129	1.21 E-8	4.95 E-12
I-131	1.37 E+0	5.60 E-4
I-132	4.05 E-1	1.66 E-4
I-133	5.72 E-1	2.34 E-4
I-134	6.26 E-1	2.56 E-4
I-135	5.36 E-1	2.19 E-4
Br-84	5.97 E-2	2.44 E-5
Ru-103	4.36 E-1	1.78 E-4
Ru-106	1.65 E-1	6.75 E-5
Te-129	8.87 E-2	3.63 E-5
Te-132	3.98 E-1	1.63 E-4
Te-134	4.73 E-1	1.93 E-4
Cs-134	6.50 E-2	2.66 E-5
Cs-137	4.19 E-2	1.71 E-5
Cs-138	5.23 E-1	2.14 E-4
Ba-140	4.92 E-1	2.01 E-4
La-140	5.28 E-1	2.16 E-4
Rb-88	2.00 E+0	8.18 E-4
Rb-89	2.56 E-1	1.05 E-4
Sr-89	2.65 E-1	1.08 E-4
Sr-90	3.04 E-2	1.24 E-5
Y-90	3.17 E-1	1.30 E-4
Sr-91	3.32 E-1	1.36 E-4
Y-91	3.43 E+0	1.40 E-3
Nb-95	4.74 E-1	1.94 E-4
Zr-95	4.71 E-1	1.93 E-4
Mo-99	5.13 E+0	2.10 E-3

Radiological Gases Released from Auxiliary Building

It is expected that small amounts of radioactive gases, halogens and particulates may leak into the auxiliary building atmosphere. Potential sources include the following:

a. Venting of Spent Regenerant Tanks.

The vapor spaces of the Spent Regenerant Tanks (SRT) in the RWDS are vented to the Auxiliary Building Ventilating System.

The only liquids collected in the SRT are those which have been depressurized and aerated in the process of becoming a waste. WDS design in addition to reducing activity releases to the extent practicable, must also be inherently safe. The separation of unaerated and aerated liquids in the collecting circuits is an important plant safety consideration in that it avoids combining the hydrogen bearing and the oxygen (air) bearing wastes to avoid the formation of explosive mixtures in the vapor spaces above collected liquids.

b. Ventilating System Concentrations

In general, all reactor coolant quality wastes, with minor exceptions, are suitable for collection in the nitrogen blanketed collecting circuits. The exceptions consist of primary system sample wastes and CVCS system ion exchanger and filter drains. These latter sources are aerated and are therefore routed to the SRT along with laboratory and floor drains.

Liquids collected in the SRT along with their design activities are listed under "Auxiliary Systems Process Wastes", Section 11.1.2.1. Waste volumes for these sources as listed in Table 11.1-11 indicate that a total of 11,000 cu. ft. of liquid per cycle is discharged to the SRT. Design activities for liquids entering the SRT are expected to be variable over a range of 10^{-7} to 6.0 μCi/cc as shown. The maximum amount of gaseous activity that may be present in the Auxiliary Building Ventilating System from the SRT has been calculated and is summarized along with applicable design parameters in Table 11.1-23.

Radiological Gases released From CARP and Radioactive Waste Processing Buildings

It is expected that small amounts of radioactive gases, halogens and particulates may be released to the CARP and Radioactive Waste Processing Building HVAC systems. The HVAC systems in these two buildings are designed to capture such releases and maintain personnel exposure ALARA. The sources for airborne radioactivity in the CARP and Radioactive Waste Processing Buildings were previously located in the existing Auxiliary Building. Therefore they do not constitute a new source of airborne radioactive releases and the releases tabulated in Table 11.1 3 remain unchanged.

Potential sources include the following:

a. Radioactive Waste Processing Building
   1. DAW sorting.
   2. DAW compaction
   3. DAW Decontamination
   4. Radwaste Filtration and Ion Exchange System
   5. Radwaste Solidification System

b. CARP Building
   1. Laboratory

Table 11.1-23 - "Maximum Gaseous Release, Spent Regenerant Tanks"

Design

Liquid volume cycle [1] to SRT = 11,000 cu. ft.
Maximum average activity, liquid mixture = 3.0 μCi/cc
Fraction volatiles present in liquid [2] = 0.5
Fraction volatiles immediately released = 1.0
Auxiliary Building ventilation rate = 7.25×10^4 SCFM

Maximum Average Activity [3]

Concentration in Aux. Bldg. Vent. Sys. = 4.3E-7 μCi/cc
Maximum SRT release to Aux. Bldg. Vent. Sys. is approximately 1.3 Ci/day of noble gases, mainly Xe-133, and approximately 7.0 μCi/day of I-131.

[1] One cycle is equivalent to 321 full power days.

[2] Estimate is conservative since liquid has been previously aerated.

[3] Volatile composition as shown in Table 11.1-21.

c. Discussion of RWDS Vent Connections.

It is concluded that under design conditions for failed fuel the liquids contained in Spent Regenerant are not a significant source of gaseous activity release.

d. Relief Valve Discharges.

The RWDS nitrogen blanket circuit is designed, by making maximum use of connected tankage, to contain relief discharges with the system. Referring to P&ID 11405-M-98, the RWDS waste gas circuit flow diagram, the Vent Header is connected through unchecked piping to the vapor spaces of all three Waste Holdup Tanks during powered operation. Locked open valves WD-441, 442 and 443 and the tank vent lines as shown in P&ID 11405-M-8 provide an interconnecting manifold between the vapor spaces of the three tanks.

The tanks therefore provide a connected reserve vapor space equivalent to at least one half of a reactor coolant volume, 6,000 cu. ft. to absorb connected component pressure variations. This is based on the assumption that all three tanks are liquid filled to 80 percent of their operating capacity.

e. Total Releases from Auxiliary Building

Of the sources discussed above, the major one is projected to be released from venting of the concentrate tanks. Total gaseous releases from the auxiliary building over a year's time have been assumed to be 150% of the releases from the concentrate tank vents, based on the total quantity of liquid wastes to be processed (see Table 11.1-2). It is further assumed that the decontamination factors given in item c above apply and that the HEPA filters in the auxiliary building discharge have a 90% efficiency for removal of particulates. The resulting releases are as given in Table 11.1-25.

<u>Radioactive Gases Released from Containment</u>

While the amount of reactor coolant that will leak into the containment is uncertain, operating experience with other, generally similar reactors indicates that leakage of approximately 25 gallons per day could be expected. Assuming a reactor coolant activity consistent with 1 percent failed fuel, the release rates of noble gases and halogens to containment would be as shown in Table 11.1-25. It is assumed that all of the noble gases activity enters the containment atmosphere. A fraction of the halogen and particulate activity will remain in the liquid phase; and additional fraction will plateout on containment surfaces; and still more will be removed by recirculation through the charcoal filters of the containment cleanup system. Thus, it is assumed that only 10^{-5} of the halogen and particulate activity leaked into the containment remains airborne.

The containment will be purged prior to refueling and possibly at other times to limit personnel exposure during access to containment. For evaluation purposes, it is assumed that the containment is purged at 30 day intervals. The activity released during purges would be as shown in Table 11.1-25.

Total Radioactive Gaseous Releases

The total expected annual activity release to the atmosphere from the (1) waste gas system, (2) containment purges, (3) auxiliary building ventilation and (4) primary-to-secondary leakage and (5) Radioactive Waste Processing and CARP buildings are listed in Table 11.1-25. Also given are the average concentration at the boundary of the unrestricted area. An average-annual dispersion factor of 5.0×10^{-6} sec/m^3 has been used to determine the isotopic activities at the boundary (Amendment 113[11-11,11-12]). The maximum whole body dose at the boundary of the restricted area, consistent with the average concentrations at the boundary in Table 11.1-25, is approximately 1.04 millirad/year, based on continuous occupancy.

Table 11.1-25 - "Annual Releases of Radioactive Gases and Particulates"

Nuclide	Gas Decay Tank (Ci)	Containment Purge (Ci)	Auxiliary Building (Ci)	Secondary Side (Ci)	Total Curies Released Annually	Concentration at Boundary (μCi/cc)	Fraction of 10CFR20
Xe-131m	3.36 E+1	1.04 E+1	9.19 E+0	6.54 E+0	5.97 E+1	9.47 E-12	4.74 E-6
Xe-133	2.86 E+2	6.43 E+2	7.12 E+2	5.07 E+2	2.15 E+3	3.41 E-10	6.81 E-4
Xe-135	2.05 E-21	6.12 E+0	5.65 E+1	4.02 E+1	1.03 E+2	1.63 E-11	2.33 E-4
Xe-138	0.00 E+0	5.26 E-1	1.88 E+2	1.34 E+2	3.23 E+2	5.11 E-11	2.56 E-3
Kr-85	3.18 E+1	2.09 E+0	1.51 E+0	1.08 E+0	3.65 E+1	5.78 E-12	8.26 E-6
Kr-85m	0.00 E+0	1.58 E+0	2.91 E+1	2.07 E+1	5.14 E+1	8.15 E-12	8.15 E-5
Kr-87	0.00 E+0	8.43 E-1	5.58 E+1	3.97 E+1	9.63 E+1	1.53 E-11	7.64 E-4
Kr-88	0.00 E+0	2.67 E+0	7.81 E+1	5.56 E+1	1.36 E+2	2.16 E-11	2.40 E-3
I-129	1.47 E-9	9.62 E-14	6.95 E-11	2.47 E-11	1.56 E-9	2.48 E-22	6.20 E-12
I-131	1.25 E-2	8.18 E-6	7.87 E-3	2.80 E-3	2.32 E-2	3.67 E-15	1.84 E-5
Ru-103	3.13 E-5	3.26 E-8	2.50 E-3	8.91 E-4	3.42 E-3	5.43 E-16	6.03 E-7
Ru-106	1.90 E-5	1.30 E-8	9.48 E-4	3.37 E-4	1.30 E-3	2.07 E-16	1.03 E-5
Te-129	0.00 E+0	7.02 E-11	5.10 E-4	1.81 E-4	6.91 E-4	1.10 E-16	1.22 E-9
Cs-134	7.69 E-6	5.15 E-9	3.73 E-4	1.33 E-4	5.14 E-4	8.14 E-17	4.07 E-7
Cs-137	5.09 E-6	3.33 E-9	2.41 E-4	8.57 E-5	3.32 E-4	5.26 E-17	2.63 E-7
Ba-140	1.18 E-5	3.25 E-8	2.83 E-3	1.01 E-3	3.85 E-3	6.11 E-16	3.05 E-7
Sr-89	2.14 E-5	2.01 E-8	1.52 E-3	5.42 E-4	2.08 E-3	3.30 E-16	1.65 E-6
Sr-90	3.69 E-6	2.42 E-9	1.42 E-3	6.21 E-5	1.49 E-3	2.36 E-16	3.93 E-5
Sr-91	0.00 E+0	2.18 E-9	1.91 E-3	6.79 E-4	2.59 E-3	4.10 E-16	8.21 E-8
Y-91	2.92 E-4	2.62 E-7	1.97 E-2	7.01 E-3	2.70 E-2	4.28 E-15	2.14 E-5
Nb-95	3.18 E-5	3.52 E-8	2.72 E-3	9.69 E-4	3.72 E-3	5.90 E-16	2.95 E-7
Zr-95	4.14 E-5	3.61 E-8	2.71 E-3	9.63 E-4	3.71 E-3	5.89 E-16	1.47 E-6
Totals	3.51 E+2	6.67 E+2	1.13 E+3	8.05 E+2	2.95 E+3	4.68 E-10	6.83 E-3

11.1.3.8 Availability and Reliability

The gaseous waste system is designed to collect, analyze, compress, store and release waste gases. While not presently being used, some portions of the gas in the vent header could be reused for tank blanketing in place of the normally used nitrogen. The system can handle gaseous wastes resulting from widely varying reactor coolant system and chemical and volume control system operational modes.

The gaseous waste system is dependent on the nitrogen gas system, as it is based on a nitrogen gas blanketing network. The gaseous waste system is also dependent on the electrical systems (See Section 8.), the component cooling water system (See Section 9.7.), and the demineralized water system. Component cooling water is used at the gas compressor heat exchangers and demineralized water is used as water seal at the gas compressors.

The automatic gas analyzing system has built in redundancy; any of the sixteen sampling streams can be directed to a gas sampling bottle and then analyzed in the hot laboratory.

The waste holdup tanks are equipped with a redundant gas blanketing supply, either nitrogen or re-use gas. These tanks require the largest volume of blanketing gas. All waste gas must pass through a gas decay tank prior to release to the atmosphere. One of the two gas compressors can handle the largest anticipated waste gas flow; the other compressor is a spare.

Radioactive gaseous effluents can be released from the plant without being so indicated on an installed radiation monitor if the requirements of the ODCM are complied with. The monitoring system is described in Section 11.2.3.

A redundant method of radioactivity detection is provided at the gas decay tanks before final release at the ventilation discharge duct. A sample from the tank is first isolated in a gas sample bottle at the automatic gas analyzer station and then checked for radioactivity level in the laboratory. If found suitable, the batch of gas is gradually released to the discharge duct via the gas release header.

The radiation monitor at the discharge duct provides a second check on radioactivity, and if the activity exceeds a predetermined limit, stops the flow completely.

In order to empty a gas decay tank to the ventilation discharge duct, a block valve at the tank outlet must be manually opened. In addition, a block valve in the gas release header must be manually opened. This double valving ensures the safest possible operation at this very critical point.

The vent header, where all of the waste gases are combined, can be sampled and analyzed for H_2 and O_2. This serves as a rough check on the contribution being made by a single component.

Interlocks and other design features have been incorporated in the RWDS to preclude in so far as practical any gaseous release except under fully controlled conditions. Typical among these features are:

a. Maximum use of available RWDS tankage by an unchecked, interconnected vapor space arrangement as previously described in part "e" of this section (Relief Valve Discharges).

b. Vents and drains arrangements as described in part "a" of this section (Venting of Spent Regenerant Tanks) provides three separate liquid drain circuits and a closed vent circuit arranged to retain activity within the auxiliary building.

c. Interlocks on RWDS components provide equipment shutdown in the event of malfunction.

11.1.3.9 Operation

The operation of the gaseous waste system is such that values for radioactive effluent release are maintained as low as reasonably achievable (ALARA). The normal operation for waste gas systems is collection, compression, retention to allow decay of short-lived radionuclides, and analysis prior to the controlled release of individual batches of waste gas. The release rates for radioactive materials, other than noble gases, in gaseous effluents is controlled such that concentrations of radionuclides do not exceed ten times 10 CFR 20 Appendix B, Table 2, Column 1 limits. For noble gases, the concentration shall be limited to five times 10 CFR 20, Appendix B, Table 2, Column 1 limits. Concentrations shall be calculated based upon the annual average Chi/Q[11-11, 11-12]. Cumulative dose contributions must be determined in accordance with the Offsite Dose Calculation manual (ODCM) on a quarterly basis. Prior to discharge of radioactive materials in gaseous effluents, the equipment used in processing gaseous effluents is operated in accordance with the requirements of the ODCM. The setpoints for the effluent radiation monitors are calculated in accordance with the ODCM. The requirements for equipment operability are defined in the ODCM. The requirements for sampling and activity analyses for radioactive gaseous waste and the requirements for verification of equipment operability are given in the ODCM. Ninety-two percent (92%) of the waste gas is due to nitrogen with low activity ($\leq 10^{-6}$ µCi/cc).

11.1.3.10 Tests and Inspections

All equipment in the gaseous waste system was subject to both manufacturer's shop tests and on-site tests.

<u>Shop Tests</u>

Some equipment was tested and inspected in the manufacturer's shop in accordance with then applicable codes and standards. In addition, some equipment was given performance type tests in the manufacturer's shop.

On-Site Tests

These tests were primarily of the performance type and were designed to ensure that the overall gaseous waste system functions in a safe and efficient manner and were conducted prior to actual plant startup.

Provision was made to test the full operational sequence of the system. Compressors were started, valves operated, instruments put into service. Flow paths, flow capacity, and mechanical operability were thoroughly checked. Pressure, temperature, flow and level indicating instruments were calibrated and checked for performance. All safety equipment, including alarms, were thoroughly tested. The automatic gas analyzer was calibrated with hydrogen and oxygen gases. Special emphasis were placed on the proper functioning of the waste gas compressor controls on the waste control panel.

Tracer gases, such as P-10 (10% Methane - 90% Argon), Helium and Sulpur Hexafluoride can be used as a leak detection medium in conjunction with a suitable detector, to locate leaks in the waste gas system outside containment. P-10 is non-flammable, non-toxic and does not become radioactive unless subjected to a neutron radiation field, which is not found outside containment.

11.1.4 Solid Wastes

11.1.4.1 General

The general types of radioactive solid wastes are produced at the station; solidified concentrate and process resins, used waste and process filters, dewatered ion exchange and filtration media, and miscellaneous solid wastes.

Solidified concentrate is placed in containers, stored in the storage area and finally shipped from the plant to an approved disposal area. Spent resin from the filtration/ion exchange system is sluiced to a high integrity container which is deterred and eventually shipped for disposal. Used filters are placed in a shielded container, stored in the cask decontamination area and eventually shipped from the plant. Miscellaneous solid wastes, such as equipment parts and laboratory glassware, are stored prior to off-site shipment.

The flow diagram, P&ID 11405-M-8, shows the process portion of the solid waste disposal system.

11.1.4.2 Sources of Solid Waste

a. Radioactive liquid waste is processed either through a filtration/ion exchange system with the processed water being directed to the monitor tanks.

b. Process wastes containing spent resins are obtained from the filtration/ion exchange system, purification ion exchangers, the cation ion exchanger, the deborating ion exchanger, and the spent fuel storage pool demineralizer.

   The resins from other sources and their sluice water are collected in the spent resin storage tank. The contents of this tank are mixed and solids are kept in suspension by nitrogen gas sparging. The contents of the tank are forced by pressurized demineralized water into a shielded resin cask after which the contents are dewatered and shipped from the plant. At this point it is considered to be a solid waste.

c. Used filter baskets originate from the purification filters, the waste filters and the spent fuel pool cooling system filter. Solids removed from the liquid are retained on the filter elements which form the basket.

d. Miscellaneous solid waste consist of contaminated articles such as equipment parts, laboratory glassware, clothing, gloves, cleaning tools, rags, towels, and plastic covers originating in the controlled access areas of the plant.

Table 11.1-26 shows the anticipated waste volumes on an annual basis.

Table 11.1-26 - "Solid Waste Volumes"

Sources	Volume (ft^3/cycle)	Basis
Spent Resins		
Filtration/Ion-Exchanger	60	3 vessels/cycle
Purification Exchangers	30	1 vessel/cycle
Cation Exchanger	20	1/2 vessel/cycle
Deborating Exchangers	10	1/5 vessel/cycle
Spent Fuel Pool Demineralizer	10	1/5 vessel/cycle
Filter Elements		
Purification Filters	15	One replacement of each filter assembly per cycle
Waste Filters	5	
Spent Fuel Pool Filter	5	
Miscellaneous Solids	2,000	Assumed value for low activity solids.
Total	2,155	

11.1.4.3 System Components

The major components of the solid wastes system of the RWDS are as follows; the referenced tables summarize pertinent data:

a. Spent resin storage tank (see Table 11.1-12);
b. Spent resin pump (see Table 11.1-13);
c. Mobile Radwaste Processing System/Filtration/Ion Exchanger (FIX)

11.1.4.4 System Operation

Radioactive Liquid and Spent Resins

The following operation is followed for the processing of liquid and resin.

a. If the filtration/ion exchange system is in operation, the radioactive liquid is transferred from the waste holdup tanks using the waste holdup transfer pumps. The water that has been processed is directed to the monitor tanks to be analyzed and discharged to the Missouri River through the overboard discharge piping. Depleted filtration ion exchange media is sluiced to a high integrity container and then dewatered using vendor supplied system prior to being shipped offsite for disposal.

b. The resin is flushed from the resin storage tank by demineralized water to a shielded resin cask with liner located in the Radioactive Waste Processing Building through shielded piping. The resin is then dewatered/solidified. The liner with resin is placed in the cask which is shipped offsite.

Miscellaneous Solid Waste

Compactable wastes are placed in a drum which is then placed on the waste baler located in the Radioactive Waste Processing Building. The waste is then hydraulically compressed. This is repeated until the drum is filled with compressed material. The drum is then removed from the baler, sealed and stored to await off-site removal. Non-compactable waste are placed in large steel boxes for disposal. The activity of this material is normally low and special shielding is not necessary.

11.1.4.5 Design Evaluation

The spent resin storage tank has a volume of 400 cu. ft. and is designed to hold at least two to three years production of spent resins. Excess transport water used to convey resins to the tank is removed by pumping from a screened lateral connection in the tank. Transport water returns to the spent regenerant tanks. Nitrogen is admitted through the bottom lateral at a sufficient rate to mix the resin slurry.

Spent resin can have high activity; therefore the resin casks are equipped with internal shields designed to reduce the external dose rate to a level permitting in-plant handling.

11.1.4.6 Availability and Reliability

The solid waste system is normally operated on a batch basis, and is available to perform abnormal or emergency functions. The system can handle wastes resulting from widely varying reactor coolant system and chemical and volume control system operational modes.

The solid waste system is dependent on the operation of the filtration/ion exchange system. These systems are also dependent on the electrical systems, the demineralized water system, the plant compressed air system, and the nitrogen gas system.

The Process Control Program (PCP) is used to verify satisfactory solidification of waste prior to shipment offsite. The PCP calls for examination of at least one representative test specimen from at least every twelfth batch of wet radioactive waste and provides for followup actions if the specimen fails to verify proper solidification. [11-4]

The Radioactive Waste Processing Building is sized to accumulate a number of containers (e.g., liners, drums, high integrity containers) to permit scheduling of off-site shipments.

11.1.4.7 Tests and Inspections

All equipment in the solid waste system was subject to both shop and on-site tests.

Shop Tests

All equipment was tested and inspected in the manufacturer's shop in accordance with the then applicable codes.

In addition, some equipment was given performance type tests in the manufacturer's shop.

On-Site Tests

These tests were primarily of the performance type and were designed to ensure that the overall solid waste system functions in a safe and efficient manner. These tests were conducted prior to actual plant startup.

Provision was made to test the full operational sequence of the system. Pumps were started, valves operated, instruments put into service.

Inspection of Containers in Storage

Provisions are included for inspection of containers while in storage by using TV cameras or boroscope for high radiation level conditions, and by direct observation when radiation levels are low.

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12. CONDUCT OF OPERATIONS

12.1 ORGANIZATION AND RESPONSIBILITY

Paragraph 50.34(b)(6)(i) of 10 CFR Part 50 requires that applications for a license to operate a nuclear power plant include information concerning organizational structure, personnel qualifications and related matters. Technical Specification 5.3.1 endorses ANSI N18.1-1971 and Regulatory Guide 1.8 Revision 1 for personnel qualifications and provides additional guidance concerning the qualifications for the position of Shift Technical Advisor.

Section 12 delineates these requirements and provides organizational information required by Technical Specification 5.2.2 in accordance with guidance contained in Generic Letter 88-06 and as approved by the NRC in Amendment 115 to the Technical Specifications. Information concerning the personnel qualifications of individuals involved with the original plant startup may be obtained from the Final Safety Analysis Report as submitted to the commission and on file in the Public Document Room.

The plant organization for the Fort Calhoun Station is shown in Figure 12.1-1. The relationship of the station organization to the balance of Omaha Public Power District's Nuclear Operations Division is shown in Figure 12.1-2.

The plant organization, under the Manager-Fort Calhoun Station, is responsible for the safe and efficient operation and maintenance of the facility. In the absence of the Manager-Fort Calhoun Station, his duties are assumed by a previously appointed designee.

OPPD was directly responsible for the initial plant startup including preoperational testing, core loading, initial criticality, low power physics testing, and the approach to full power. Technical assistance was provided by Combustion Engineering; Gibbs, Hill, Durham and Richardson; and equipment suppliers.

During normal operations, an operating shift consists of two Senior Reactor Operators, two Reactor Operators, at least two unlicensed operators and a Shift Technical Advisor. At least one Senior Reactor Operator is stationed in the control room and may be relieved by another licensed operator for short periods of time. The other Senior Reactor Operator is normally stationed in the control room, but may leave for a period of time to perform other operating functions. At least one Reactor Operator, or the second Senior Reactor Operator if both Senior Reactor Operators are in the control room, must be present at the controls at all times. The unlicensed operators perform routine checks and operations on auxiliary systems at locations outside the control room. The plant is normally staffed with six crews which is sufficient to cover vacations and illness. The plant is manned with licensed Senior Reactor Operators and Reactor Operators in accordance with NRC Regulations. At least one Chemistry and one Radiation Protection technician is assigned to every shift.

The plant maintenance organization consist of a Manager-Maintenance, Maintenance Supervisors, mechanical, electrical, and instrumentation and control craftspeople. The maintenance force is supplemented by OPPD personnel from other locations for other than normal maintenance such as turbine-generator inspection or major equipment repair.

12.1.1 Qualifications of Various Onsite Personnel

The following plant staff positions are filled by onsite personnel:

Manager-Fort Calhoun Station
Assistant Manager-Fort Calhoun Station
Manager-Operations
Supervisor-Operations
Manager-Maintenance
Manager-Radiation Protection
Supervisor-I&C Maintenance
Manager-Chemistry
Supervisor-Systems Chemistry
Supervisors
Shift Technical Advisors
Principal Reactor Engineer

12.1.1.1 Manager-Fort Calhoun Station, Assistant Manager-Fort Calhoun Station

At time of appointment to the active position, the Manager-Fort Calhoun Station, and the Assistant Manager(s)-Fort Calhoun Station shall have a minimum of ten years of responsible power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one time basis. To be acceptable this training shall be in an engineering or scientific field generally associated with power production. The plant manager shall have acquired the experience and training normally required for examination for a Senior Reactor Operator's License whether or not the examination is taken.

Where one or more persons, including the Assistant Manager-Fort Calhoun Station, who are designated as principal alternates for the plant manager and who meets the nuclear power plant experience and NRC examination requirements established for the plant manager, the requirements for the Manager - Fort Calhoun Station may be reduced such that only one of the ten years of experience need be nuclear power plant experience and the manager need not be eligible for NRC examination.

At least one of the persons filling this position should have a recognized baccalaureate or higher degree in an engineering or scientific field generally associated with power production.

12.1.1.2 Manager-Operations

At the time of appointment to the active position, the Manager-Operations shall meet the requirements for the Assistant Manager-Fort Calhoun Station.

12.1.1.3 Supervisor-Operations

At the time of appointment to the active position, the Supervisor-Operations, shall have a minimum of eight years of responsible power plant experience of which a minimum of three shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one basis. At the time of appointment to the active position the Supervisor-Operations shall hold a Senior Reactor Operator's License.

12.1.1.4 Manager-Maintenance

At the time of appointment to the active position the Manager-Maintenance shall have a minimum of seven years of responsible power plant experience or applicable industrial experience, a minimum of one year of which shall be nuclear power plant experience. A maximum of two years of the remaining six 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. The Manager-Maintenance further should have nondestructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

12.1.1.5 Manager-Radiation Protection

The Manager-Radiation Protection (MRP) at the time of appointment to the active position, shall meet the requirements set forth in Regulatory Guide 1.8 dated September 1975 entitled "Personnel Selection and Training". The MRP shall have at least five years of professional experience in applied radiation protection. A master's degree may be considered equivalent to one year of professional experience, and a doctor's degree may be considered equivalent to two years of professional experience where course work related to radiation protection is involved. At least three years of this professional experience shall 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 as specified in the Technical Specifications. The MRP is considered to meet the educational and experience qualifications set forth above with at least five years of experience in applied radiation protection and extensive formal training in radiation protection.

12.1.1.6 Supervisor-I&C Maintenance

At time of appointment to the active position, the Supervisor-I&C Maintenance shall have a minimum of five years experience in instrumentation and control, of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

12.1.1.7 Manager-Chemistry

At time of initial appointment to the active position, the Manager-Chemistry, should have a minimum of eight years in responsible positions, of which one year shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training.

The individual shall have a minimum of five years experience in chemistry of which a minimum of one year shall be in radiochemistry. A minimum of two of this five years experience should be related technical training.

12.1.1.8 Supervisors

Supervisors not requiring an NRC License shall, at the time of initial appointment to the active position, have a high school diploma or equivalent and a minimum of four years experience in the craft or discipline supervised.

12.1.1.9 Shift Technical Advisors

In accordance with Technical Specification 5.3.1, Shift Technical Advisors shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

12.1.1.10 Supervisor-Systems Chemistry

At the time of appointment to the active position, the responsible person shall have a minimum of five years experience in chemistry of which a minimum of one year shall be in radiochemistry. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

12.1.1.11 Principal Reactor Engineer

At the time of appointment to the active position, the Principal Reactor Engineer shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences and two years experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing programs.

12.1.2 Support Personnel

The Fort Calhoun Station plant staff is technically supported during operation by personnel in the OPPD organization and outside consultants who, collectively, are technically competent in all necessary areas. The technical areas which must be covered by the backup technical personnel include: reactor operations, reactor engineering, chemistry and radiochemistry, metallurgy and radiation damage, instrumentation and control, radiation safety, mechanical and electrical engineering, and Quality Control.

The OPPD personnel who support the plant staff are shown in Figures 12.1-2, 12.1-3, and 12.1-4. Those personnel who perform the major role in support of the plant staff are identified by the following titles:

1. Vice President
2. Manager-Nuclear Training
3. Manager-Nuclear Licensing
4. Manager-Security and Emergency Planning
5. Supervisor-Central Maintenance
6. Division Manager-Nuclear Operations
7. Manager-Nuclear Procurement Services
8. Manager-Design Engineering Nuclear
9. Supervisor-Reactor Physics and Reactor Engineering
10. Manager-Nuclear Construction Management
11. Manager-Nuclear Process Computer Services
12. Division Manager-Nuclear Engineering
13. Manager-System Engineering
14. Supervisor-Special Services Engineering
15. Division Manager-Nuclear Assessments
16. Manager-Quality Assurance & Quality Control
17. Manager-Nuclear Safety Review Group
18. Manager-Nuclear Administrative Services
19. Division Manager-Environmental and Governmental Affairs
20. Manager-Corrective Action Group
21. Division Manager-Nuclear Support Services

12.1.2.1 Manager-System Engineering, and Manager-Design Engineering Nuclear

Shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences. The individuals should have a minimum of eight years in responsible positions. A maximum of four of the eight years should be fulfilled by satisfactory completion of academic training. The individuals shall have a minimum of three years of professional level experience in nuclear services, nuclear plant operation, or nuclear engineering, and the necessary overall nuclear background to determine when to call consultants and contractors for dealing with complex problems beyond the scope of owner-organization expertise.

12.1.3 Technical Support from Outside Organizations

OPPD has several outside organizations that provide technical support to the plant staff as necessary. This is only a limited listing; other organizations may be utilized as appropriate.

Raytheon has been retained as in-service inspection specialists for the Fort Calhoun Station. This organization has a staff which is also fully competent in the fields of welding technology, metallurgy, and radiation damage.

ABB/Combustion Engineering, Inc. (ABB/CE), provides technical support to the OPPD organization in various areas including instrumentation and control, safety analysis and other areas related to the NSSS design and operations.

The Gibbs and Hill (G&H) consulting engineering organization provided the design for all major mechanical and electrical systems in the Fort Calhoun plant.

Exxon Nuclear (EN) was the fuel supplier for the Fort Calhoun Plant for cycles 6 through 10. ABB/CE supplied fuel for cycles 11 through 13. Westinghouse supplied fuel for cycles 14 through 19. Siemens (formerly Exxon Nuclear and then Advanced Nuclear Fuels Corporation), ABB/CE and Westinghouse also supply (or supplied) technical support for the fuel. Future reloads may be obtained from these suppliers or any other qualified reload fuel vendor.

Stone and Webster (S&W) provides the Fort Calhoun Plant with technical support in the areas of Instrumentation and Control and Mechanical. S&W also provides general support as required by the plant staff.

Wyle Labs (WL) provides the Fort Calhoun Plant with technical support in the areas of instrumentation testing and qualification.

12.1.4 Summary

The OPPD and consulting personnel presented in sections 12.1.2 and 12.1.3 provide technical competence in all areas of nuclear power plant technology to ensure safe and efficient operation of the facility. The major areas of expertise for each individual and group are presented in Table 12.1-1.

12.1.5 Plant Staff Working Hours

Plant Staff working hour limitations are implemented in response to Generic Letters 82-02 and 82-12, and apply to plant staff who perform safety-related functions (e.g., Operations staff, Shift Technical Advisors, Shift Radiation Protection Technicians, Shift Chemist and key maintenance personnel, including apprentice and above).

12.1.5.1 Working Hours Guidelines and Limitations

The following guidelines and limitations shall be adhered to for controlling working hours for applicable plant staff:

- An individual shall not be permitted to work more than 16 hours straight (excluding shift turnover time).
- An individual shall not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period (all excluding shift turnover time).
- A break of at least 8 hours shall be allowed between work periods (including shift turnover time). A work period is defined as eight (8) hours or more.

- Except during extended shutdown periods, the use of overtime shall be considered on an individual basis and not for the entire staff on shift.

- If special circumstances arise or emergency conditions exist, deviation from these guidelines is permitted. Such deviations shall be approved in advance by the Manager-Fort Calhoun Station or designated alternate, or higher levels of management.

- Working hours shall be reviewed at least monthly by the Department Head, Manager-Fort Calhoun Station or their designated alternates, or higher levels of management, to ensure working hours for applicable individuals are in compliance with these guidelines.

Table 12.1-1 - "Support Organization Areas of Competence"

Area	OPPD	Outside Organization
Reactor Operations/ Engineering	Supervisor-Reactor Performance Analysis Supervisor-Reactor Physics and Reactor Engineering Principal Reactor Engineer	ABB/CE W
Metallurgy and Radiation Damage	Supervisor-Mechanical Systems Principal Engineer-Metallurgical	Raytheon ABB/CE
Instrumentation and Control	Supervisor - Electrical/I&C Engineering Principal Engineer-Electrical/I&C	ABB/CE WL
Mechanical and Electrical Engineering	Division Manager - Nuclear Engineering Manager-Design Engineering Nuclear Manager - System Engineering Supervisor - Electrical/I&C Engineering Supervisor-Mechanical Systems Supervisor-Engineering Mechanics Engineers (Electrical & Mechanical) - Nuclear Engineering	ABB/CE S&L S&W
Quality Assurance/ Quality Control	Division Manager - Nuclear Assessments Manager - Quality Assurance and Quality Control	Raytheon
Emergency Planning	Manager-Security and Emergency Planning	

Abbreviations:

S&W	Stone and Webster
S&L	Sergant & Lundy Engineers
W	Westinghouse
WL	Wyle Labs
ABB/CE	Asea Brown Boveri/Combustion Engineering

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14. SAFETY ANALYSIS

14.1 GENERAL

Earlier sections of this report described and evaluated the reliability of major systems and components of the plant from a safety standpoint. For the Safety Analysis it is assumed that certain incidents may occur notwithstanding the precautions taken to prevent their occurrence. The potential consequences of such occurrences are then examined to determine their effect on the plant, to determine whether the plant design is adequate to minimize the consequences of such occurrences, and to provide assurance that the health and safety of the public is protected from the consequences of even the most severe of the hypothetical accidents analyzed.

On August 15, 1980, Fort Calhoun Station was issued a license amendment to allow operation at a steady state full rated power level of 1500 MWt. Prior to that amendment the licensed full rate power level was 1420 MWt, even though the station was designed to operate at 1500 MWt, and certain safety analyses in the FSAR were based on this higher power level. While most of the anticipated operational occurrences and postulated accidents considered in the FSAR were reanalyzed to justify operation at 1500 MWt, the original safety analysis remains valid for certain events initiated from lower power or zero power initial conditions as well as for events originally analyzed at a power level of 1500 MWt. In addition, some events analyzed for a full power rating of 1420 MWt are more severe than their counterparts at 1500 MWt, due to more restrictive core and system parameters existing for that cycle. The most restrictive cycle's analysis will be referred to as the limiting cycle while the most recent analysis will be labeled the reference cycle.

Exxon Nuclear Company (ENC) now known as Siemens Nuclear Power Corporation, the fuel vendor for Cycles 6 through 10, performed the reanalysis of all events described in this chapter of the USAR which were affected by the increase in rated power to 1500 MWt (in Cycle 6). The analyses, which bounded Cycle 6 operation, were performed using the ENC plant transients simulation model which is further described in Section 14.1.5. The ENC DNBR analyses utilized the W-3 correlation which has a minimum DNBR limit of 1.30.

Beginning with Cycle 8, Omaha Public Power District has performed the reanalysis of all events affected by Technical Specification changes, core physics or thermal-hydraulics parameter changes, and plant modifications with the exception of the Loss of Coolant and CEA ejection accidents. The methodology, described in Reference 14.1-1, and simulation code (CESEC-III) used are consistent with that being used by the NSSS vendor Combustion Engineering, Inc. (CE). The DNBR analysis utilizes the CE-1 correlation which when used in a deterministic simulation had a limit of 1.19 for Cycles 8 and 9. A value of 1.15 was used for Cycle 10. The change occurred as a result of the NRC final approval of the CE-1 correlation with a limit of 1.15 as contained in Reference 14.1-3. The 1.19 value represented an NRC-approved interim value. The CESEC-III code is further discussed in Section 14.1.5.

Since Cycle 9 the use of a statistical combination of uncertainties program has been incorporated into the DNBR analysis (Reference 14.1-2) method. Simulation of the DNBR-related events assume initial values without uncertainties for core average heat flux, core flow rate, core inlet temperature, RCS pressure, and integrated radial peaking factor. The uncertainties associated with these parameters are combined statistically and included in the minimum DNBR limit of 1.22 for Cycle 9 and a limit of 1.18 for current analyses. The uncertainties for other factors such as the Doppler and moderator temperature coefficient are treated deterministically.

14.1.1 Identification of Occurrences and Accidents

The anticipated operation occurrences and postulated accidents analyzed in this section fall into three principle categories. One category includes events which do not involve any break of the reactor coolant boundary. While these events do result in power, temperature or pressure increases in the reactor core, they do not involve any release of radioactive material from the reactor fuel to the reactor coolant. Events in this category are discussed in Sections 14.2 through 14.12 and Section 14.22. With the exception of the Main Steam Line Break Incident (14.12) and seized rotor event (14.6.2), which are considered as postulated accidents, all incidents in this grouping are classified as anticipated operational occurrences in which protection from exceeding the Specified Acceptable Fuel Design Limits is provided by either the Reactor Protective System or is dependent on the maintenance of an initial over power margin.

A second category includes those postulated incidents which do involve a failure of the reactor coolant system boundary. These are considered in Sections 14.13 through 14.17, and include the control element assembly (CEA) ejection, steam generator tube rupture, and the loss-of-coolant accidents. Such accidents most likely will not occur during the life of the plant. Nevertheless, in order to evaluate the protection afforded to the public by the safety features of plant design and operation, the consequences of such incidents are analyzed in terms of the resulting potential releases of radioactive material and the potential radiological exposure to persons outside the plant site boundaries. To assure that adequate protection is provided for the public, conservative assumptions are incorporated into the analyses. In all cases, the potential exposures which are calculated are shown to be less than the limits specified in 10 CFR 100.

In addition to the two categories described above, a number of postulated accidents are also considered which do not involve the reactor core or coolant system, but which could involve a release of radioactive or toxic material to the environment. They are discussed in Sections 14.18, 14.19, 14.20, 14.23 and 14.24. Analysis of these incidents shows that safeguards incorporated in the plant design would limit any release of radioactive material to inconsequential amounts.

The maximum hypothetical accident (MHA) (Section 14.21 which now references the Loss of Coolant Accident, Section 14.15) involves a release of substantial quantities of fission products from the reactor core to the containment. This accident is not considered credible because of the numerous protective devices and systems incorporated in the plant, but it is analyzed to show that even this incredible accident does not result in an unacceptable risk to the health and safety of the public. This requires a containment leak rate of 0.1% per day, and this limit is incorporated in the Technical Specifications for the Fort Calhoun Station.

14.1.2 System Parameters

The parameters used as input to the analyses are in general consistent with those listed in Section 3. For the purposes of the safety analyses, the following values of the major parameters were assumed as shown for the Cycle 6 deterministic analyses and the Cycle 19 analyses incorporating the statistical combination of uncertainties and appropriate Technical Specification changes:

	Cycle 6	Cycle 19
Reactor Power Level, MWt	1530	1500
Core Inlet Temperature, °F	547	545
Minimum Pressurizer Pressure, psia	2053	2075
RCS Flow Rate, gpm	190,000	202,500
Steam Generator Pressure, psia	850	825

Deviation from the first four parameter values in any of the analyses is specifically discussed including the reasons and effects. Steam generator pressure is dependent on the value of core inlet temperature and will vary depending on the analysis. This parameter is also not a statistical combination of uncertainties based input.

14.1.3 Trip Settings

The reactor is protected by the Reactor Protective System and the Engineered Safeguards Systems. In case of abnormal transients, the Reactor Protective System is set to trip the reactor and prevent core damage. The elapsed time between the time when the setpoint condition exists at the sensor and the time when the control element assembly clutches are de-energized is defined as the trip delay time. The values of the trip setpoints and trip delay times used for the purpose of the safety analyses are shown in Table 14.1-1.

The high rate of change of power (HRCP) trip is developed from a signal generated by the Wide Range Nuclear Instrumentation. It is provided to protect against power excursion events (e.g., boron dilution, uncontrolled CEA withdrawal, or CEA ejection) initiated from subcritical conditions. With the HRCP trip operational, events initiated from subcritical conditions are assured of having much less severe consequences than events initiated from critical conditions. Therefore, specific analyses of events initiated from subcritical conditions are not performed.

A reactor trip signal acts to open the trip contactors feeding power to the CEA drive mechanism clutches (see Section 7.2.5). The loss of power to the clutches causes the mechanisms to release the CEA's which then fall by gravity into the core.

The safety analyses presented in this chapter of the USAR, when performed deterministically, are based on the worst credible combinations of parameters including the given uncertainties. Analyses employing the statistical combination of uncertainties assume nominal initial values without uncertainties for the core average heat flux, core flow rate, core inlet temperature, RCS pressure, and integrated radial peaking factor in conjunction with a deterministic combination of all other parameters.

Table 14.1-1 - " Reactor Protective System Trips and Safety Injection for Safety Analyses Setpoints"

Trip	Setpoint	Uncertainty	Used in Analysis Delay Time (Sec)	Safety Analyses Setpoint
High Rate-of-Change of Power	2.6 dec/min	N/A	N/A	N/A (4)
High Power Level	107%	5.0%	0.4	112%
Low Reactor Coolant Flow	95%	±2%	0.65	93%(5)
High Pressurizer Pressure	2350 psia	±22 psi	0.9	2422 psia (3)
Thermal Margin/Low Pressure(1)	1750 psia	±22 psi	0.9	1728 psia
Low Steam Generator Pressure	500 psia	±22 psi	0.9	478 psia
Low Steam Generator Water Level	31.2% of narrow range span	±10 in. (5.7% of narrow range span)	0.9	25.5% of (4) span
Asymmetric Steam Generator Differential Pressure	135 psid	±40 psid	0.9	175 psid
Containment Pressure High	5 psig	±0.4 psi	0.9	5.4 psig
High Pressure Safety Injection	1600 psia	±22 psi	12(2)	1578 psia

(1) Values represent the low limit of the thermal margin/low pressure trip. The setpoint of this trip is discussed in Section 7.2.
(2) Pump start - loop valve opening time.
(3) The pre-8/92 setpoint was 2400 psia which was subsequently reduced to 2350 psia. The analysis setpoint is conservatively retained at 2400 psia plus the 22 psia uncertainty.
(4) Currently not credited in USAR Section 14 transient-accident analyses.
(5) A conservative bounding value of 90% was used for Cycle 18 and re-evaluated for Cycle 19 at 93%.

14.1.4 Radiation Monitoring During Accident Conditions

Gaseous radioactivity is continuously sampled and monitored from the containment building (RM-051) and the ventilation discharge duct (RM-062). A swing monitor (RM-052) can also monitor gaseous radioactivity and continuously sample particulates and iodine from either the containment building or the ventilation discharge duct. Particulate activity from the containment is sampled and monitored continuously by RM-050. Particulate and iodine are also sampled continuously by RM-062. The ventilation discharge duct post-accident wide range noble gas detector, RM-063 and the post-accident particulate and iodine sampling system will be used in the event RM-062 monitor goes off-scale due to very high radiation releases under severe accident conditions. The main steam line monitor, RM-064, will be used to monitor the gaseous effluent releases from the main steam safety relief valves, atmospheric dump valve, and auxiliary feed pump turbine exhaust in the event of a steam generator tube rupture. A twenty-three channel area monitoring system is provided to measure radiation levels in the containment and auxiliary building. Additionally, the condenser off gases, steam generator blowdown, waste disposal system liquid effluent, and component cooling water are continuously monitored. The radiation monitoring equipment, (described in detail in Section 11.2.3) in conjunction with installed process instruments and data from the on-site meteorological tower will be used to monitor, locate, quantify, control and plan releases of radioactivity from the plant during normal operation and following an accident of less severity than a major loss of coolant accident. In the extremely unlikely event of a LOCA, the operator would quickly be alerted by the containment high pressure, pressurizer low pressure and low reactor coolant flow alarms and the containment isolation signal.

Once containment isolation is initiated, the containment sample lines are automatically closed and the gaseous and particulate monitoring equipment is effectively isolated. Immediate and continuous quantitative indication of the magnitude of radioactivity in the containment would be obtained, however, from the six (RM-070 thru RM-075) area radiation monitoring channels, which allow surveillance of the containment if necessary. Containment wide range area monitors RM-091A and RM-091B will be used in the event of very high radiation releases under severe accident environments.

14.1.5 Calculation Methods and Input Parameters for Transient Reanalysis

The Cycles 8 through 19 analyses performed by OPPD utilize the CESEC-III code (Reference 14.1-4 14.1-5) to simulate non-LOCA plant responses which include all the anticipated operational occurrences and all accidents except the CEA Ejection and Loss of Coolant.

The CESEC code, which numerically integrates one dimensional mass and energy conservation equations, assumes a node/flow-path network to model the NSSS. The primary system components considered in the code include the reactor vessel, the reactor core, the primary coolant loops, the pressurizer, the steam generators and the reactor coolant pumps. The secondary system components includes the secondary side of the steam generators, the main steam system, the feedwater system and the various steam control valves. In addition, the program models most of the control and plant protection systems.

The code self-initializes for any given, but constant, set of reactor power level, reactor coolant flow rate and steam generator conditions. During the transient calculations, the time rate of change in the system pressure and enthalpy are obtained from the solution of the conservation equations. These derivatives are then numerically integrated in time under the assumption of thermal equilibrium to give the system pressure and nodal enthalpies. The fluid states recognized by the code are subcooled and saturated; superheating is allowed in the pressurizer. Fluid in the reactor coolant system is assumed to be homogeneous.

The CESEC-III code contains a wall heat transfer model to permit simulation of voiding in any node in which steam formation occurs. Voiding may occur in events such as a steam line break or steam generator tube rupture. Nodalization of the closure head, allows for the formation of a void in the upper head region when the pressurizer empties.

The DNBR analyses performed for Cycles 8 through 19 use the TORC code (Reference 14.1-7) or the CETOP code (Reference 14.1-8) which incorporate the CE-1 correlation.

To ensure conservative predictions of systems responses with resulting minimum values for the DNB ratios, as well as maximum values for the system pressure, conservative assumptions are applied to the input data. These assumptions can be grouped into the following three general categories:

1. Generic assumptions, applicable to all transients, based on steady state operational and instrumentation errors (measurement uncertainties).

2. Assumptions which conservatively encompass reload fuel neutronic parameters.

3. Transient specific assumptions yielding the most adverse system response.

The generic assumptions (Category 1) used in a deterministic analysis are applied to all full power transients to account for steady state instrumentation errors. The Cycle 6 and 19 initial core conditions were obtained by adding the maximum steady state uncertainties to the rated values as follows:

Reactor Power	=	1500 MW_{th} + 2 percent (30 MW_{th}) for calorimetric error.
Reactor Inlet temperature	=	543°F + 2°F for dead band and measurement error.
Primary Coolant fluctuation System pressure	=	2075 - 22 psia for steady state and measurement errors.

The combination of the above parameters minimizes the initial minimum DNB flux ratio. These values are consistent with those in the Plant Technical Specifications for 1500 MW_{th} operation. Table 14.1-2 shows a list of typical operating parameters used in the Cycle 6 and Cycle 19 analyses. The trip setpoints incorporated into the model for the Fort Calhoun Station are the same as previously listed in Table 14.1-1.

Table 14.1-2 - "Typical Operating Parameter Values Used in the Analysis of the Fort Calhoun Station"

	Cycle 6 Input Value	Cycle 19 Input Value (Reference 14.1-6)
Core Total core heat output, MW	1530.0	1500.0[1]
Heat generated in fuel, %	97.5	97.5
Pressurizer pressure (minimum) psia	2053.0	2075[1]
Hot channel factors		
Unrodded Radial Peaking Factor, F_{xy}^{T}	1.62	1.816
Integrated Radial Peaking Factor F_R^T	1.57	1.732
Reactor Coolant System Flow Rate, gpm	190,000	202,500[1]
Core Flow Rate, fraction of RCS flow rate	0.9554	0.9554
Reactor Inlet Temperature, °F	547.0	545[1]
Calculated average heat flux[2] Btu/hr-ft^2	176,210	178,378
Steam generators		
Calculated total steam flow[3] lb/hr	6.737×10^6	6.61×10^6[4]
Steam temperature, °F	525.2	*
Feedwater enthalpy, Btu/lb	423.11	421.4[4]

*Not an input value.

NOTES:

(1) Uncertainties combined statistically in DNBR analysis.
(2) Calculated from total thermal power and total cladding surface. (@100% power).
(3) Calculated from thermal power, feedwater, and steam conditions.
(4) Corresponds to 100% power.

The design parameter values representative of Westinghouse reload fuel are summarized in Table 14.1-3. Table 14.1-4 lists the neutronic parameter values which conservatively bounded representative reload fuel for both the beginning (BOC) and end of cycle (EOC) conditions for Cycles 6 and 19 (Ref. 14.1-6).

The assumptions in Category 2 refer to the reactivity feedback effects from moderator temperature changes and Doppler broadening. For a given transient, the BOC or EOC conditions (as given in Table 14.1-4) are used depending on which would result in the more limiting plant responses. For Cycle 6 analyses, the nominal moderator temperature coefficient (MTC) and Doppler coefficient were adjusted by 20 percent to ensure conservative results. The Cycle 19 MTC is the Technical Specification value for BOC and the COLR limit value for EOC, which includes uncertainties. For Cycle 19, the Doppler uncertainty was conservatively applied and bounded using appropriate multipliers. These multipliers are used for every applicable transient.

The assumptions in Category 3 apply to plant control and protection systems. As an example, pressurizer spray and pressurizer relief valve action are ignored in the seized rotor accident. Since these assumptions are considered separately for each transient, they are detailed in the appropriate section where each transient is described.

Table 14.1-3 - "Fort Calhoun Fuel Design Parameter Values for Representative Reload Fuel"

Fuel Pellet diameter	0.3765 inch
Inner cladding diameter	0.384 inch
Outer cladding diameter	0.440 inch
Active length	128.0 inch
Number of active fuel rods in core (design)	23,368*

* This value is used as a baseline for the Reference 14.1-10 assumption of allowing up to ten replacement/reconstituted stainless steel rods. The CETOP/TORC models utilize 23,358 fuel rods.

Table 14.1-4 - "Fort Calhoun Reactivity Data"

Parameter	Cycle 6 Value		Cycle 19 Value	
	BOC	EOC	BOC	EOC
Moderator temperature coefficient ($\Delta\rho$/Fx10^{-4})	+ 0.5	-2.3	+0.5	-3.5
Doppler coefficient ($\Delta\rho$/Fx10^{-4})	-0.08	-0.213	(1)	(1)
Pressure coefficient ($\Delta\rho$/psi 10^{-4})	-0.01	+0.04	-	-
Moderator density coefficient %$\Delta\rho$/(g/cm^3)	-6.0	+40.0	-	-
Inverse boron worth coefficient (ppm/%$\Delta\rho$)	-125	-111	(2)	(3)
Delayed neutron fraction	0.0072	0.0045	0.0062	0.0052
Total rod worth at HFP PDIL (%$\Delta\rho$)	-5.85	-5.80	(4)	(4)
Shutdown margin at HZP (%$\Delta\rho$)	-2.7	-2.7	-4.0	-4.0

(1) Generic bounding equation as a function of fuel temperature is used. Additional uncertainty is assessed in each of the analyses as appropriate.
(2) Value reported in Boron Dilution Incident.
(3) Value reported in Main Steam Line Break Accident.
(4) Value depends on the transient being evaluated.

14.1.6 Specific References

14.1-1 "Omaha Public Power District Reload Analysis Methodology - Transient and Accident Methods and Verification," OPPD-NA-8303-P, Revision 4, January 1993.

14.1-2 "Statistical Combination of Uncertainties," Parts 1-3, CEN-257(0)-P, November, 1983.

14.1-3 "CE Critical Heat Flux, Part 2: Non-Uniform Axial Power Distribution," CENPD-207-P-A, December, 1984.

14.1-4 "CESEC, Digital Simulation of a Combustion Engineering NSSS," December 1981, transmitted as enclosure 1-P to LD-82-001, January 6, 1982.

14.1-5 "Response to questions on CESEC," CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.1-6 EA-FC-98-044, Rev 0, "Cycle 19 CESEC Basedeck."

14.1-7 "Users Manual for TORC," CENPSD-628-P, Rev. 04-P, March 1994.

14.1-8 "CETOP: Thermal Margin Model Development," CENPSD-150-P, Rev. 01-P, April 1991.

14.1-9 EA-FC-98-036, Rev. 0, "Cycle 19 Design Depletions."

14.1-10 EA-FC-98-045, Rev. 0, "Cycle 19 Thermal Hydraulics Analysis."

14.1.7 General References

14.1.7.1 Kahn, J. D., "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurizer Water Reactors (PTSPWR)," XN-74-5, Revision 1, May 1975.

14.1.7.2 Koester, G. E., et al, "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt," XN-NF-77-18.

14.1.7.3 Galbraith, K. P. and Patten, T. W., "Verification and Justification of Exxon Nuclear Company DNB Correlation for PWRs, XN-75-48," October 1975.

14.2 CONTROL ELEMENT ASSEMBLY WITHDRAWAL INCIDENT

14.2.1 General

The sequential CEA group withdrawal event is assumed to occur as a result of a failure in the control element drive mechanism control system or by operator error. The CEA Block System, which was installed after Cycle 1, has eliminated the possibility of an out-of-sequence bank withdrawal or a single CEA withdrawal due to a single failure.

This event was reanalyzed for Cycle 19 (Reference 14.2-7). The Cycle 19 DNB-ROPM results were determined to be less limiting than the Cycle 18 analysis for HFP. The Cycle 19 MDNBR and PLHGR were found to be more limiting than the Cycle 18 results at HZP.

An uncontrolled or unplanned withdrawal of the CEAs results in a positive reactivity addition, which causes the core power, core average heat flux, and reactor coolant system temperature and pressure to rise in turn decreasing the DNB and the linear heat rate (LHR) margins. The pressure increase, if large enough, activates the pressurizer sprays which mitigate the pressure rise. In the presence of a positive moderator temperature coefficient (MTC) of reactivity, the temperature increase results in an additional positive reactivity addition further increasing the severity of the power transient and reducing the DNB and LHR margins.

The withdrawal of the CEAs also causes the axial power distribution to shift to the top of the core. The associated increase in the axial peak is partially compensated by a corresponding decrease in the integrated radial peaking factor. The magnitude of the 3-D peak change depends primarily on the initial CEA configuration and the axial power distribution. Furthermore, the neutron flux measured by the excore detectors becomes decalibrated due to CEA motion (i.e., rod shadowing effects). This decalibration of excore detectors, however, is partially compensated by reduced neutron attenuation arising from moderator density changes (i.e., temperature shadowing effects).

As the core power and heat flux increase, a reactor trip on high power, variable power, or thermal margin/low pressure may occur to terminate the event depending on the initial operating conditions and the rate of reactivity addition. Other potential reactor trips include axial power distribution and high pressurizer pressure. If a trip occurs, the CEAs drop into the core and insert negative reactivity which quickly terminates further thermal margin degradation. If no trip occurs and corrective action is not taken by the operator, the CEAs fully withdraw and the NSSS achieves a new steady state equilibrium with higher power, temperature, peak LHR and a lower hot channel DNBR value.

14.2.1.1 Hot Full Power CEA Withdrawal

Withdrawal of CEAs from full power operating conditions results in a small reactivity addition since the lead bank (normally a low worth bank) can only be inserted 25%. The small positive reactivity addition causes the core average heat flux, and reactor coolant system pressure and temperature to rise. This rise in power is mitigated by the high power trip.

14.2.1.2 Hot Zero Power CEA Withdrawal

A CEA withdrawal event initiated from lower power levels will exhibit trends similar to the full power CEA withdrawal except that the rate of reactivity addition (and margin degradation) will be greater due to the greater insertion of CEAs allowed by the Technical Specification Power Dependent Insertion Limit LCO (Reference 14.2-6, Figure 2). The rate and magnitude of the power, temperature, heat flux and pressure increase are therefore, greater due to the greater reactivity addition. At hot zero power (including subcritical conditions) the withdrawal can result in a significant power spike. The heat flux follows the fission power but is limited by the fuel temperature feedback. The event is terminated by the variable high power trip. The TM/LP trip will not occur because the Pvar calculated pressure will be less than the actual reactor coolant system pressure.

14.2.2 Applicable Industry and Regulatory Requirements

The CEA Withdrawal event is classified as an anticipated operational occurrence (AOO) which does not require an RPS trip at HFP to provide protection against exceeding the DNB and LHR SAFDLs (Reference 14.2-1). These requirements are met by adding sufficient margin to the DNB and LHR LCOs to ensure that the SAFDL limits will not be exceeded during a CEA withdrawal event. However, for some initial conditions and reactivity insertion rates, the Variable High Power Trip in conjunction with the initial steady state LCOs, is required to prevent the DNBR limits from being exceeded. The Variable High Power Trip (VHPT) and the Axial Power Distribution (APD) trip, in conjunction with the steady state LCOs, prevent the LHR limits from being exceeded.

14.2.3 Method of Analysis

The methodology employed in analyzing this event, is described in References 14.2-1, -2 and -3. Depending on the initial conditions and the reactivity insertion rate associated with the CEA Withdrawal (CEAW), the Variable High Power Trip in conjunction with the initial steady state LCOs, prevents DNBR limits from being exceeded. An approach to the centerline to melt (CTM) limit is terminated by either the Variable High Power Trip or the Axial Power Distribution Trip. The analysis takes credit for only the Variable High Power Trip (utilizing input from the excore detectors) in both the determination of the required initial overpower margin for DNBR using TORC/CE-1 and the peak linear heat generation rate for the CTM SAFDL (Reference 14.2-15).

In order to maximize the overpower margin requirements for the LCO, the CEAW is analyzed parametrically to obtain a maximum steady state power level just below the high power trip setpoint. Any higher level of reactivity insertion will cause a high power trip and terminate the event, a lower reactivity insertion will be bounded by the overpower margin requirements.

The CEAW incident is analyzed using the CESEC-III computer code (References 14.2-8, -9, -10 and -11) which models neutron kinetics with fuel and moderator temperature feedback, the reactor control system, the reactor coolant system, the steam generators, and the main steam and feedwater systems. The results of the transient simulation, the transient average core heat flux, average channel mass flow rate, reactor core inlet temperature, and reactor coolant system pressure serve as input to TORC (References 14.2-12 and -16) which uses open channel pressure balancing calculations. This code uses the CE-1 correlation (References 14.2-13 and -14) to calculate the DNB ratio for the hot channel as a function of time and axial position.

Reactivity addition by withdrawal of CEA regulating groups is dependent on the initial position of the groups prior to the withdrawal and on the integral worth of these groups. The regulating groups are withdrawn in a specified sequence having 20 percent group overlap, with the exception of groups 3 and 4 which have a 40 percent overlap, and the position of the groups under steady state conditions is a function of power level (see COLR, Figure 2).

For the full power DNBR analysis, an MTC consistent with that utilized in Reference 3 and a gap thermal conductivity consistent with the assumption of References (14.2-1 and -2) are used in conjunction with a variable reactivity insertion rate. The range of reactivity insertion rates examined is given in Table 14.2-1.

For both the full power CTM and zero power CTM and DNBR cases the most positive MTC was used to maximize the positive reactivity feedback from increasing coolant temperatures. To minimize negative reactivity feedback from increasing fuel temperature, a 1.0 multiplier was applied to the Doppler coefficient of reactivity. The initial RCS pressure for Cycle 19 was chosen to be 2100 psia which corresponds to the SCU pressure (Reference 14.2-4). These assumptions yield lower transient minimum DNBRs.

14.2.4 Inputs and Assumptions

Reactivity addition by withdrawal of CEA regulating groups is dependent on the initial position of the groups prior to the withdrawal and on the integral worth of these groups. The regulating groups are withdrawn in a specified sequence having 20 percent group overlap. The position of the groups under steady state conditions is a function of power level.

For the full power DNBR analysis, an MTC identical to that utilized in Reference 14.2-3 and a gap thermal conductivity consistent with the assumption of Reference 14.2-1 are used in conjunction with a variable reactivity insertion rate.

For both the full power LHR and zero power LHR and DNBR cases the most positive MTC was used to maximize the positive reactivity feedback from increasing coolant temperatures. To minimize negative reactivity feedback from increasing fuel temperature, a 1.0 multiplier was applied to the Doppler coefficient of reactivity. The initial RCS pressure was chosen to be 2100 psia which corresponds to the minimum allowed pressure minus uncertainties. These assumptions yield lower transient minimum DNBRs. The maximum positive reactivity insertion rate, due to the CEA withdrawal, was determined to be bounded by $1.0 \times 10^{-4} \Delta\rho$/sec for both the full power and zero power conditions.

14.2.4.1 Hot Full Power Case

Table 14.2-1 contains a list of the initial conditions and assumptions including uncertainties for Cycle 19 used in the analysis of the full power CEA withdrawal. For the full power case it is conservative not to take credit for the decalibration of the excores due to CEA motion, i.e., the rod shadowing factor including uncertainties is less than 1.0. A trip on High Power at 112.0% of rated thermal power was assumed in the analysis (Ref. 14.2-20).

14.2.4.2 Hot Zero Power Case

The list of the initial conditions and assumptions including uncertainties for Cycle 19 used in the zero power CEA withdrawal case can be found in Table 14.2-3. In this case, decalibration of the excores due to the CEA motion was not accounted for. A reactor trip, initiated by the Variable High Power Trip at 30% (20% plus 10% uncertainty) of rated thermal power, was assumed in the analysis.

14.2.5 Results

The CEA Withdrawal event was analyzed for Cycle 19 to determine the initial margins that must be maintained by the Limiting Conditions for Operation (LCO's) such that the DNBR, LHR and the CTM design limits will not be exceeded in conjunction with the High Power and Variable High Power Trips.

Protection against exceeding the LHR limit for the CEA withdrawal at full power is provided by the initial steady state thermal margin which is maintained by adhering to the Technical Specifications LCOs on LHR margin and by the response of the RPS which provides an automatic reactor trip on high power level. The analysis shows that the peak LHR is well below the acceptable value of 22 kW/ft. The sequence of events for the full power case with the maximum reactivity insertion rate is presented in Table 14.2-2. Figures 14.2-1 through 14.2-4 show the representative transient behavior of core power, core average heat flux, reactor coolant system temperatures, and the RCS pressure for the full power case.

The zero power case initiated at limiting conditions of operation for Cycle 19 results in a minimum CE-1 DNBR of much greater than 1.18. The analysis shows that although the peak linear heat rate limit of 22 kw/ft is exceeded, the results are acceptable because the peak centerline temperature remains below the 4700^{0}F centerline melt limit of Reference 14.2-21. For rapid power spikes of short duration a time at power is more significant than the peak linear heat generation rate achieved (Page 3-1 of Reference 14.2-3). The axial power distribution trip is not credited in this event, because the change in axial power shape during the event is not large enough to actuate this trip. Table 14.2-4 contains the sequence of events for the zero power case. The representative transient behavior of the core power, core average heat flux, reactor coolant system temperatures, and the RCS pressure are presented in Figures 14.2-5 through 14.2-8.

14.2.6 Affected Plant Systems

For this event the affected plant systems are the reactor coolant system, the reactor protective system (VHPT and APDT), and the reactivity control system. The specific system parameters affected are provided in Tables 14.2-1 and 14.2-3.

14.2.7 Limiting Parameters for Reload Analysis

Reevaluation of the CEA withdrawal event is required when either of the following conditions exist:

- Core physics and/or thermal-hydraulic parameters change in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent Technical Specification limiting condition of operation (LCO).

Any changes to parameters and/or Technical Specifications must result in a DNBR and peak LHR which do not exceed the SAFDLs.

14.2.8 Conclusions

The CEA Withdrawal Incident when initiated at either hot full power conditions or hot zero power conditions for Cycle 19 from the LCOs will not lead to a DNBR or a LHR (centerline fuel temperature), which will violate the design limits. Neither of the design limits are violated for this event and no pins are predicted to fail.

14.2.9 Specific References

14.2-1. "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification," OPPD-NA-8303-P, Rev. 04, January 1993.

14.2-2. "CE Transient Analysis Methods for Fort Calhoun Station Unit No. 1," CENPSD. 152-1 Rev. 1-P, July 1981.

14.2-3. "CEA Withdrawal Methodology," CEN-121(B) - P, November 1979.

14.2-4. "Statistical Combination of Uncertainties," CEN-257(0) - P, November 1983.

14.2-5. Supplement 1-P (of Reference 14.2-4), Aug. 1985.

14.2-6. Fort Calhoun Technical Data Book, Section VI, Core Operating Limits Report

14.2-7. EA-FC-98-047, Rev. 0. "Cycle 19 CEA Withdrawal Analysis."

14.2-8. "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, CE Proprietary Report, April 1974.

14.2-9. "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply Steam," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.2-10. "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply Steam," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.2-11. Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.2-12. "TORC; A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P-A, April 1986.

14.2-13. "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution," CENPD-162-P-A, September 1976.

14.2-14. "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution," CENPD-270-P, June 1978.

14.2-15. "WCAP-12978 Westinghouse Reload Fuel Mechanical Design Evaluation for the Fort Calhoun Station, Unit 1," June 1991. (contained in EA-FC-90-004)

14.2-16. "TORC Code, Verification and Simplification Modeling Methods," CENPD-206-P-A, June 1981.

14.2-17. Fort Calhoun Station, Unit 1 Operating License DPR-40 and Technical Specifications, including all Amendments through Amendment 190, April 15, 1999.

14.2-18 EA-FC-98-037, Rev. 0, "Cycle 19 Safety Analysis Base Cases".

14.2-19 EA-FC-98-041, Rev. 0, "Cycle 19 SLB Cooldown and Scram Curves".

14.2-20 EOS-FC-97-0449, dated November 13, 1997; Telecon Thomas Heng (OPPD) to Kim Jones (ABB/CE), "Treatment of Uncertainties for the High Power Trip Setpoint", contained in EA-FC-97-027, Rev. 1, "Cycle 18 CEA Withdrawal Analysis".

14.2-21 Letter 99 CF-G-0021, "Omaha Public Power District Fort Calhoun Fuel Centerline Temperature Limit", dated 07/13/99, from M.F. Muenks (Westinghouse) to T.A. Heng (OPPD).

Table 14.2-1 - "Key Parameters Assumed in the CEA Withdrawal Analysis Cycle 19 (HFP)"

System Parameter	Units	Value	Reference	Affected Tech Spec Value
Initial Core Power Level	MWt	1531.85*	14.2-7	Ref. 14.2-17, Rated Power Definitions, pg. 1 including Reactor Coolant Pump Heat. Conservative initial power level per Ref. 14.2-20.
Reactor Coolant System				
Initial Core Inlet Coolant Temperature	°F	545*	14.2-7	Maximum allowed plus uncertainty, Ref. 14.2-17, Section 2.10.4(5)(a)(i) within the limit for Core Inlet temperature provided in the COLR
Initial RCS Flow Rate	gpm	202,500	14.2-2	Reference 14.2-17, Section 2.10.4(5)(a)(iii) requires nominal flow
Pressurizer Pressure	psia	2100*	14.2-7	Minimum allowed minus uncertainty. Ref. 14.2-17, Section 2.10.4(5)(a)(ii) requires ≥2100 psi nominal
Reactor Protective System				
VHPT Setpoint	% of rated thermal power	112.0	14.2-7	Reference 14.2-17, Basis for 1.3(1)
Reactivity Control System				
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	14.2-7	Maximum allowed (Ref. 14.2-17, Section 2.10.2(3) requires less positive than +0.5 below 80% power, less positive than +0.2 at or above 80% of rated power)
CEA Group Withdrawal Rate	in/min	46	14.2-7	N/A
CEA Holding Coil Delay	sec	0.5	14.2-2	N/A
System/Parameter				
Max. Reactivity Insertion Rate	%Δρ/Second	1.0	14.2-2	N/A

* For DNBR calculations, the uncertainties on these parameters have been statistically combined.

Table 14.2-2 - "Cycle 19 Sequence of Events for the HFP CEA Withdrawal Event Maximum LHR"

Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	----
2.32	High Power Trip Conditions Sensed	112.0% of 1500 MWt
3.72	Reactor Trip Breakers Open	----
4.22	CEAs Begin to Drop Into Core	----
3.72	Maximum PLHR	19.31% kw/ft
3.84	Maximum Core Power	121.14% of 1500 MWt
4.24	Maximum Heat Flux	111.10% of 1500 MWt

Table 14.2-3 - "Key Parameters Assumed in the CEA Withdrawal Analysis Cycle 19 (HZP)"

System Parameter	Units	Value	Reference	Affected Tech Spec Value
Initial Core Power Level	MWt	1*	14.2-7	Ref. 14.2-17, Rated Power Definitions, pg. 1 including Reactor Coolant Pump Heat
Reactor Coolant System				
Initial Core Inlet Coolant Temperature	°F	532*	14.2-7	Limit for core inlet temperature provided by COLR at HZP
Initial RCS Flow Rate	gpm	202,500	14.2-2	Reference 14.2-17, Section 2.10.4(5)(a)(iii) requires nominal flow
Pressurizer Pressure	psia	2100*	14.2-7	Minimum allowed minus uncertainty. Ref. 14.2-17, Section 2.10.4(5)(a)(ii) requires ≥2100 psi nominal
Reactor Protective System				
VHPT Setpoint	% of rated thermal power	30	14.2-7	Reference 14.2-17, Basis for 1.3(1)
Reactivity Control System				
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	14.2-7	Maximum allowed (Ref. 14.2-17, Section 2.10.2(3) requires less positive than +0.5 below 80% power, less positive than +0.2 at or above 80% of rated power)
CEA Group Withdrawal Rate	in/min	46	14.2-7	N/A
CEA Holding Coil Delay	sec	0.5	14.2-2	N/A
System/Parameter				
CEA Time to 100% Insertion (including Holding Coil Delay)**	sec	3.1	14.2-18	Maximum (Ref. 14.2-7, Section 2.10.2(8) requires ≥2.5 sec to 90% insertion)

* For DNBR calculations, the uncertainties on these parameters have been statistically combined.
**CE generic scram curve specifies 100% insertion in 3.1 sec which includes 0.5 sec signal delay plus holding coil delay.

Table 14.2-3 - (Continued)

System Parameter	Units	Value	Reference	Affected Tech Spec Value
CEA Worth at Trip (HZP PDIL with ARO insertion is most limiting)	%Δρ	4.7791	14.2-19	N/A
Max Reactivity Insertion	%Δρ/second	1.0	14.2-2	N/A

Table 14.2-4 - "Cycle 19 Sequence of Events for the HZP CEA Withdrawal Event Maximum LHR"

Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	----
19.13	Variable High Power Trip Signal Generated	30% of 1500 MWt
20.53	Reactor Trip Breakers Open	----
21.03	CEAs Begin to Drop Into Core	----
21.38	Maximum Core Power	85.71% of 1500 MWt
22.08	Maximum Heat Flux	50.43% of 1500 MWt
22.08	Minimum CE-1 DNBR	>>1.18

14.4 CONTROL ELEMENT ASSEMBLY DROP INCIDENT

14.4.1 General

The CEA drop incident is defined as the inadvertent release of a CEA causing it to drop into the reactor core. The CEA drive is of the rack and pinion type, with the drive shaft running parallel to and driving the rack through a pinion gear and a set of bevel gears. The drive mechanism is equipped with a mechanical brake which maintains the position of the CEA. A CEA drop may occur due to either an inadvertent interruption of power to the CEA holding coil (i.e. magnetic clutch) or an electrical or mechanical failure of the mechanical brake in the CEA drive mechanism when the CEA's are being moved. This incident was reanalyzed for Cycle 19.

The drop of a single CEA into the core reduces the fission power in the vicinity of the dropped CEA and adds negative reactivity on a core-wide basis. The negative reactivity insertion causes a prompt drop in core power and heat flux with the magnitude ranging from approximately 4 to 30%, depending on the worth of the dropped CEA. The turbine runback circuitry at the Fort Calhoun Station has been removed along with the automatic mode of operation. Therefore, the turbine continues to demand the same power as it did prior to the drop. This results in a power mismatch between the primary and secondary systems resulting in a cooldown of the reactor coolant system. In the presence of a negative moderator temperature coefficient (MTC) of reactivity, the decreasing average coolant and fuel temperatures add positive reactivity to the core. The radial and axial power distributions begin to shift as a result of the reactivity feedback effects and the neutron flux asymmetry caused by the dropped CEA. A new tilted asymptotic radial power distribution with higher radial peaking is reached within a few minutes. Xenon redistribution will cause further tilting and increase the radial peak by approximately 5% within one hour if the event is not terminated. The positive reactivity addition due to feedback from the moderator and Doppler is eventually sufficient to compensate for the dropped rod's negative reactivity. The final result is that the core power may return to the pre-drop level and the coolant temperature will be slightly reduced. With this configuration or in the process of achieving it, local power densities and heat fluxes may exist which are in excess of the design limits.

Detection of a dropped CEA is accomplished from any one of three sources. Alarms indicating four and eight inch deviations from the group position are provided from the position indications for every CEA. This means of detection is independent of the location and reactivity worth of the dropped CEA and is also independent of spatial distribution of core power. The rod block circuitry, which contains a visual display of rod positions, provides another method of determining that a rod drop has occurred. The CRT screen will flash for this condition, and the circuitry will limit CEA movement to the manual individual mode (where only one rod can be moved at a time). A third method for sensing a dropped CEA utilizes the out-of-core power range nuclear instruments. A first order time lag network is used to distinguish between the relatively rapid power reduction caused by a dropped CEA as compared with normal changes in load demand. Dropping of even the most remote CEA (a CEA near the core center) is expected to cause a reduction of approximately 10 percent in the signal from the out-of-core detectors. Should a CEA drop from a partially inserted position, causing a smaller change in neutron flux, the corresponding change in power distortion would be smaller.

14.4.2 Applicable Industry and Regulatory Requirements

The CEA drop event is classified as an anticipated operational occurrence (AOO) which does not require a reactor protective system trip to maintain a DNBR (using the CE-1 correlation) greater or equal to 1.18 and a peak linear heat rate (PLHR) less than the linear heat rate (LHR) limiting condition of operation (LCO) and limiting safety system setting (LSSS). The DNBR criterion is met by maintaining the following parameters within their LCO limits:

(1) Cold leg temperature $\leq$ 545°F

(2) Pressurizer pressure $\geq$ 2075 psia

(3) Reactor coolant flow $\geq$ 202,500 gpm

(4) Axial shape index within limits of Technical Specification 2.10, (Limiting Condition of Operation for DNB Monitoring)

(5) CEA configurations within power dependent insertion limits of Technical Specification 2.10.

(6) Integrated radial peaking factor F^T_R, within limits of Technical Specification 2.10.

During the reload analysis, sufficient initial steady state margin must be built into these LCO's to allow the reactor to ride out the event.

14.4.3 Method of Analysis

The CEA drop incident analysis was performed using the computer code CESEC 14.4-5,6,7 and 8 which models neutron kinetics with fuel and moderator temperature feedback, the reactor control system, the reactor coolant system (RCS), the steam generators, and the main steam and feedwater systems.

The methodology used in deriving the DNBR and LHR Required Overpower Margins* (ROPMs) is consistent with that used in Reference 14.4-1 which utilizes the statistical combination of uncertainties described in Reference 14.4-2. Table 14.4-1 contains a list of the assumptions including uncertainties for the analysis. The most negative Doppler coefficient was used to enhance the positive reactivity feedback from the reactor coolant temperature decrease. Likewise, the most negative moderator temperature coefficient of reactivity was chosen.

The initial pressurizer pressure was chosen to be 2075 psia which corresponds to the minimum allowed pressurizer pressure. This results in a lower final RCS pressure and thus in a lower minimum DNBR. The minimum dropped rod worth allowed by the PDIL was chosen so that the prompt drop in power and inlet temperature drops would be minimized. Consequently, the initial condition LCOs are more restrictive, because the inlet temperature remains higher resulting in a lower DNBR value.

* The ratio between the margin that is available at the initiation of the transient and that which exists for the most adverse conditions at any time during the transient expressed as a percentage change.

Table 14.4-1 - "CEA Drop Assumptions Including Uncertainties"

Parameter	Units	Cycle 18
Initial Core Power Level	MWt	100% of 1500*
Core Inlet Temperature	°F	545*
Pressurizer Pressure	psia	2075*
RCS Mass Flow Rate	gpm	202,500
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.5
CEA Insertion at Full Power	% Insertion of Bank 4	25.0
Dropped CEA Worth	% Δρ-unrodded	-0.246
	PDIL	-0.240
Radial Peaking Distortion Factor		
Integrated Radial Peaking	Unrodded	1.223
	PDIL	1.223

*The uncertainties on these parameters were combined statistically and included in the DNBR analysis.

14.4.4 Results

The CEA Drop incident was reanalyzed for Cycle 19 to accommodate the extreme low leakage fuel management used.

Table 14.4-2 presents the sequence of events for the CEA Drop incident initiated from the full power initial conditions contained in the Table 14.4-1. Figures 14.4-1 through 14.4-4 show the results of a representative CEA Drop simulation as presented in plots of core power, core heat flux, reactor coolant system temperatures, and RCS pressure versus time.

The incident is initiated by the insertion of the dropped CEA worth listed in Table 14.4-1 over a time period of 1.0 second. The distortion factors which were used in the calculation of the DNBR Required Overpower Margin were derived on the basis of a three-dimensional power distribution analysis rather than on the synthesis of a two dimensional model using separate rodded and unrodded distortion factors in conjunction with the rodded and unrodded F_{xy} values and the most adverse axial distribution.

For Cycle 19, the full power CEA Drop initiated at an ASI of -0.16 and with a Bank 4 insertion of 25% results in a limiting minimum DNBR value of $\geq$1.18 using CE-1 correlation in conjunction with the (References 14.4-2 and 3) limit of 1.18. The maximum DNBR ROPM is 116.7% of initial power (PDIL CASE, ASI = -0.16) (Reference 14.4-4). The maximum LHR ROPM is 122.3% at ARO HFP.

For a CEA Drop, a maximum allowable initial linear heat generation rate greater than the LOCA limit of 15.5 kW/ft could exist as an initial condition without exceeding the Specified Acceptable Fuel Design Limit of 22 kW/ft during this transient. This amount of margin is maintained operationally by setting the Linear Heat Rate LCO based on the allowable linear heat rate for LOCA.

Since the limit conditions for operation maintain the required DNB thermal margin, and the allowable linear heat generation rate LCOs are based on more stringent LOCA limits, the Specified Acceptable Fuel Design Limits (SAFDLs) will not be exceeded during a CEA drop incident.

Table 14.4-2 - "Sequence of Events for Full Length CEA Drop Incident From ARO"

TIME (Sec)	Event	Value
0.0	CEA begins to drop into core	
1.0	CEA reaches fully inserted position	100% insertion
1.1	Core power level reaches a minimum and begins to return to power due to reactivity feedbacks	70.0% of 1500 MWt
103.3	Core inlet temperature reaches a minimum value	542.06°F
200.0	RCS pressure reaches a minimum value	2025.1 psia
200.0	Minimum DNBR is reached	≥1.18 (CE-1 correlation)
200.0	Core power returns to its maximum value	93.2% of 1500 MWt

14.4.5 Affected Plant Technical Specifications

The following Technical Specifications are used as input to the CEA Drop analysis (Ref. 14.4-9).

- LCO 2.10.2 Reactivity Control Systems and Core Physics Parameter Limits
- LCO 2.10.4 Power Distribution Limits

14.4.6 Affected Plant Systems

For this event the affected plant systems are the reactor coolant system and the reactivity control system. The main steam system and turbine generator are indirectly affected due to the power reduction without a corresponding change in steam flow rate.

14.4.7 Limiting Parameters for Reload Analysis

Reevaluation of the CEA drop incident is required when either of the following conditions exist:

Core physics and/or thermal-hydraulic parameters change in a nonconservative direction.

A plant design modification is expected to cause a change to a pertinent Technical Specification limiting condition of operation (LCO).

Any changes to parameters and/or technical specifications must result in a DNBR and peak LHR which do not exceed the SAFDLs.

14.4.8 Conclusions

The results of the CEA drop rod incident analysis for Cycle 19 show that the DNBR LCO limits of the core and RCS ensure that the reactor will ride out the event without tripping while maintaining a DNBR greater or equal to 1.18 using the CE-1 correlation.

14.4.9 Specific References

14.4-1 OPPD-NA-8303-P, Rev. 04, "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification," January 1993.

14.4-2 CEN-257(0)-P, "Statistical Combination of Uncertainties Methodology, Parts 1-3," November 1983.

14.4-3 Supplement 1-P (of Reference 14.4-2), August, 1985.

14.4-4 EA-FC-98-048, Rev. 0, "Cycle 19 CEA Drop."

14.4-5 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.4-6 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.4-7 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.4-8 Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.4-9 Fort Calhoun Station Unit No. 1 Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 172, December 1995.

14.4.10 General References

14.4.10-1 "Users Manual for TORC," CE-NPSD-628-P, CE Proprietary Report, March 1994.

14.6 LOSS OF COOLANT FLOW INCIDENT

The Loss of Coolant Flow event was reanalyzed for Cycle 18 (Ref. 14.6-5). The seized rotor event was reviewed for Cycle 19, (Ref. 14.6-16).

A loss of normal coolant flow may result from either a loss of electrical power to one or more of the four reactor coolant pumps or from a mechanical failure, such as shaft seizure, of a single pump. Simultaneous mechanical failure of two or more pumps, however, is not considered credible. The loss of electrical power to one or more reactor coolant pumps will hereafter be referred to as the Loss of Coolant Flow while the mechanical failure will be called the Seized Rotor event. These two events will be analyzed separately below.

14.6.1 Loss of Coolant Flow Event

14.6.1.1 General

The three failure modes resulting in a loss of coolant flow due to an electrical failure include, in order of severity:

a. Simultaneous loss of power to all four reactor coolant pumps;

b. Loss of one auxiliary transformer (two pumps in opposite loops);

c. Loss of power to one pump.

The loss of power to all four reactor coolant pumps at full power represents the most limiting case of the above failure modes in terms of DNBR margin degradation, and hence the other two cases are not analyzed. Event (a) may occur due to either the complete loss of ac power to the plant or the failure of the fast transfer breakers to close after a loss of offsite power. In the event that this incident occurs, the high rotational energy in the pumps (N = 1192 rpm, I = 71,000 lb-ft^2 per pump) will cause the core flow rate to drop at such a rate that the minimum DNBR is always in excess of 1.18 using the CE-1 correlation (i.e., at which point there is a 95% probability at a 95% confidence level that DNB does not occur) in conjunction with the statistical combination of uncertainties described in References 14.6-1 and 14.6-3.

Reactor trip for the loss of coolant flow incident is initiated by a low coolant flow rate as determined by a reduction in the sum of the steam generator hot to cold leg pressure drops. This signal is compared with a setpoint which is a function of the number of reactor coolant pumps in operation (which current Technical Specifications require to be four). For all loss of flow events, a trip would be initiated when the flow rate drops to 93 percent of full flow (95 percent minus 2 percent uncertainty). For the Cycle 18 analysis, a bounding value of 90% of full flow was used.

14.6.1.2 Applicable Industry and Regulatory Requirements

The loss of coolant flow event is considered an anticipated operational occurrence. For this incident, the specified acceptable fuel design limits (SAFDL) must not be exceeded. This is achieved through automatic action of the reactor protective system (RPS) which was designed in accordance with the applicable design criteria as stated in Appendix G.

14.6.1.3 Method of Analysis

The Loss of Coolant Flow event is analyzed using the methodology described in Reference 14.6-2. This entails simulating the event using the CESEC-III computer code (Ref. 14.6-6,7,8 and 9) which contains explicit modeling of the four reactor coolant pumps, i.e., the conservation equations for mass flow rate and momentum are solved using the pump torque values as given by the manufacturer's four- quadrant curves wherein the torque is related to the pump angular velocity and discharge rate. The event is analyzed parametrically in initial axial shape and rod configuration.

The transient heat fluxes, core mass flow rate, inlet temperature, and RCS pressure are then used as input to the TORC code (Ref. 14.6-10) or CETOP code (Ref. 14.6-4) which uses the CE-1 correlation (Refs. 14.6-11,12 and 13) for performing DNBR calculations for the hot channel as a function of time. The minimum DNBR is then compared to the References 14.6-1 and 14.6-3 limit of 1.18. The minimum initial margin that must be maintained by the LCOs is such that, in conjunction with the RPS low flow trip, the DNBR limit will not be exceeded.

14.6.1.4 Inputs and Assumptions

Reactor trip for the loss of coolant flow incident is initiated by a low coolant flow rate as determined by a reduction in the sum of the steam generator hot to cold leg pressure drops. This signal is compared with a setpoint which is a function of the number of reactor coolant pumps in operation (which current Technical Specifications require to be four). For all loss of flow events, a trip would be initiated when the flow rate drops to 93 percent of full flow (95 percent minus 2 percent uncertainty). As previously noted, a conservative value of 90% versus 93% was used for the Cycle 18 analysis to bound future cycles.

Coolant flow coast-down is calculated by CESEC utilizing the manufacturer's four-quadrant homologous pump curves, wherein the torque is related to the pump annular velocity and discharge rate. The resultant coast-down curve is presented in Figure 14.6-1. The event is analyzed parametrically in initial axial shape and rod configuration assuming:

- A low flow trip response time of 0.65 seconds,
- The most reactive CEA is stuck in the fully withdrawn position,
- The moderator temperature coefficient of reactivity chosen is the most positive allowed by Technical Specifications

Table 14.6-1 lists the key transient parameters used in the Cycle 18 analysis.

14.6.1.5 Results

The Loss of Coolant Flow event was reanalyzed for Cycle 18 to determine the minimum initial margin that must be maintained by the Limiting Conditions for Operation (LCO's) such that in conjunction with the RPS (low flow) trip, the DNBR limit will not be exceeded.

Table 14.6-2 presents the NSSS and RPS responses during a four pump loss of flow initiated at an axial shape index of -0.182 and PDIL condition which bound the DNBR related axial shape index LCO. The low flow trip setpoint (of 90% as analyzed for Cycle 18) is reached at 3.67 seconds and the reactor trip breakers open at 5.07 seconds. The minimum CE-1 DNBR is reached at 5.45 seconds. Figures 14.6-2 to 14.6-5 present the core power, heat flux, core coolant temperatures, and RCS pressure as a function of time. The maximum ROPM (Required Over Power Margin) for this event is 113.030%.

Table 14.6-1 - "Fort Calhoun Cycle 18 Key Parameters Assumed in the Loss Of Coolant Flow Analysis"

Parameter	Units	Cycle 18 Value
Initial Core Power Level	MWt	1500*
Initial Core Inlet Coolant Temperature	°F	545*
Initial RCS Flow Rate	gpm	202,500
Pressurizer Pressure	psia	2075*
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5
LFT Analysis Setpoint	% of initial flow	90.0**
LFT Response Time	sec	0.65 (0.90 conservatively used)
4-Pump RCS Flow Coastdown		Figure 14.6-1
CEA Holding Coil Delay	sec	0.5
CEA Time to 100% Insertion (Including Holding Coil Delay)	sec	3.1
CEA Worth at Trip (all rods out)	$10^{-2}\Delta\rho$	-6.32
Total ARO Integrated Radial Peaking Factor (F_R^T)		1.768
Total PDIL Integrated Radial Peaking Factor (F_R^T) with Group 4&3 Insertion		1.8909

* The uncertainties on these parameters have been statistically combined and included in the DNBR Analysis.

** This value must be 93% or less to comply with Technical Specification 1.3(2).

Table 14.6-2 - "Fort Calhoun Cycle 18 Sequence of Events for Loss of Flow"

Time (Sec)	Event	Setpoint or Value
1.0	Loss of Power to all Four Reactor Coolant Pumps	-----
3.67	Low Flow Trip Conditions Reached	90% of 4-Pump Flow
4.57	Trip Breakers Open	-----
5.07	Shutdown, CEA's Begin to Drop into Core	-----
5.45	Minimum CE-1 DNBR	>>1.18
5.92	Maximum RCS Pressure, psia	2109.88

14.6.1.6 Affected Plant Technical Specifications

The loss of flow event analysis uses input from the following Technical Specifications (Ref. 14.6-14).

- LSSS 1.3 — Limiting Safety System Settings, Reactor Protective System
- LCO 2.1.1 — Operable Components
- LCO 2.10.2 — Reactivity Control Systems and Core Physics Parameter Limits
- LCO 2.10.4 — Power Distribution Limits

14.6.1.7 Affected Plant Systems

For this event, the affected plant systems are the reactor coolant system (reactor coolant pumps), the reactivity control systems and the reactor protective system (low flow trip). The specific system parameters are provided in Table 14.6-1.

14.6.1.8 Limiting Parameters for Reload Analysis

Reevaluation of the loss of coolant flow event is required when either of the following conditions exists:

- Core physics and/or thermal-hydraulic parameters change in a nonconservative direction.
- A plant design modification is expected to cause a change in a pertinent technical specification limiting condition of operation (LCO).

Any changes to parameters and/or Technical Specifications must result in a minimum DNBR of 1.18 or greater. This minimum is required in order to maintain adequate heat transfer from the core and limit the fuel cladding temperature rise during the loss of flow event.

14.6.1.9 Conclusions

It may be concluded that the Four-Pump Loss of Flow event, when initiated from within the Technical Specifications LCO's in conjunction with the low flow trip will not exceed the design DNBR limit.

14.6.2 Seized Rotor Event

14.6.2.1 General

The Seized Rotor event is assumed to be the result of a mechanical failure of a single reactor coolant pump.

In this event, the most limiting circumstance would be an instantaneous shearing of the rotor, leaving a low inertia impeller attached to a bent shaft. The shaft and impeller are assumed to stop instantaneously causing a very rapid decrease in core flow. The reduction in flow would initiate a reactor trip on low flow within the first few seconds of the transient.

14.6.2.2 Applicable Industry and Regulatory Requirements

The seized rotor event, as described is classified as a postulated accident for which the dose rates due to radiological releases must be within the 10 CFR 100 guidelines (Ref. 14.6-15). Assurance of meeting this requirement is met if less than one percent of the fuel pins in the core fail during the event (Ref. 14.6-2).

14.6.2.3 Method of Analysis

The Seized Rotor event is analyzed in accordance with the methodology described in Reference 14.6-2. This method calculates the number of pin failures assuming that the core flow instantaneously decreases to the 3-pump flow rate and utilizes the TORC analysis (Ref. 14.6-10) with a 3-pump inlet flow distribution. The initial RCS pressure and core inlet temperature are used as input to TORC and the core average heat flux is conservatively assumed to remain at its initial value.

The maximum F_R^T value allowed by Technical Specifications (Ref. 14.6-14) is combined with a conservatively flat power distribution using the TORC code to generate absolute DNBR's. These results, when combined with the probability of failure from the CE-1 correlation, determine the total number of fuel pins that have failed (Ref. 14.6-16).

14.6.2.4 Results

The Seized Rotor event was analyzed for Cycle 19 to demonstrate that a small fraction of fuel pins are predicted to fail during this event.

The methods used to analyze this event are consistent with the OPPD Fort Calhoun Unit No. 1 transient methodology described in Reference 14.6-2. A DNBR limit of 1.18 was used to be consistent with the Statistical Combination of Uncertainties program (Reference 14.6-1 and 14.6-3).

The total number of pins predicted to fail was less than 1% (0.4%) of all the fuel pins in the core. Based on this result the resultant site boundary dose would be well within the limits of 10 CFR 100.

14.6.2.5 Affected Plant Technical Specifications

This event is analyzed utilizing inputs from the following Technical Specifications (Ref. 14.6-14).

- LSSS 1.3 — Limiting Safety System Settings, Reactor Protective System
- LCO 2.1.1 — Operable Components
- LCO 2.10.2 — Reactivity Control Systems and Core Physics Parameter Limits
- LCO 2.10.4 — Power Distribution Limits

14.6.2.6 Affected Plant Systems

The affected plant systems are the reactor coolant system (reactor coolant pumps), the reactivity control systems, and the reactor protection system (low flow trip). The specific system parameters affected are provided in Table 14.6-3.

14.6.2.7 Limiting Parameters for Reload Analysis

Reevaluation of the Seized Rotor event is required when either of the following conditions exists:

- Fuel supplier changes: core physics and/or thermal-hydraulic parameters change in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent Technical Specification LCO.

Parameters and Technical Specification changes have to be such that the resultant site boundary dose is within the limits set forth by 10 CFR 100. This criterion is met if the number of failed fuel pins is less than one percent of the total number of fuel pins in the core.

14.6.2.8 Conclusions

It may be concluded that the Seized Rotor event, when initiated from within the Technical Specification LCO's, in conjunction with the low flow trip, will not result in site boundary doses in excess of the limits imposed by 10 CFR 100.

14.6.3 Specific References

14.6-1 "Statistical Combination of Uncertainties, Parts 1-3," CEN-257(0)-P, November 1983.

14.6-2 "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification," OPPD-NA-8303-P, Rev. 4, January 1993.

14.6-3 Supplement 1-P of Reference 14.6-1, Aug. 1985.

14.6-4 "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2", CEN-191(B)-P, December 1981.

14.6-5 EA-FC-97-029, Rev. 0, "Cycle 18 Loss of Flow Analysis."

14.6-6 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.6-7 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.6-8 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.6-9 Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.6-10 "Users Manual for TORC," CE NPSD-628-P, Rev. 09-P, August 1996.

14.6-11 OPPD Letter No. LIC-81-0134, W. C. Jones (OPPD) to R. Clark (NRC), 9/17/81 (CT 1011, Frame 805).

14.6-12 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution," CENPD-162-P-A, September 1976.

14.6-13 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution," CENPD-162-P-A, June 1978.

14.6-14 Fort Calhoun Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 190, April 1999.

14.6-15 10 CFR 100, Reactor Site Criteria, as amended effective January 5, 1987.

14.6-16 EA-FC-98-053, Rev.0 "Cycle 19 Seized Rotor Analysis."

Table 14.6-3 - "Key Parameters Assumed in the Reactor Coolant Pump Seized Rotor Event-Cycle 19"

System/Parameter	Units	Value	Source	Affected Tech. Spec Value
Initial Core Power Level	MWt	1536	Ref. 14.6-16	(Ref. 14.6-14, Rated Power Definitions, pg 1)
Reactor Coolant System				
Initial Core Inlet Coolant Temperature	°F	547	Ref. 14.6-16	Maximum Allowed (Ref. 14.6-14, Section 2.10.4 (5) (a) (i) requires ≤545°F)
3 Pump RCS Flow Rate	gpm	151,875 (0.75 of 202,500)	Ref. 14-6-16	Minimum Allowed (Ref. 14.6-14, Section 2.10.4 (5) (a) (iii) requires 4 pump flow ≥197,000 gpm)
Pressurizer Pressure	psia	2053	Ref. 14.6-16	Minimum Allowed (Ref. 14.6-14, Section 2.10.4 (5) (a) (ii) requires ≥2075 psia)
Axial Shape Index	asiu	-0.150	Ref. 14.6-16	Most Negative Allowed at 100% Power (Ref. 14.6-14, COLR Figure 8)

14.9 LOSS OF LOAD

14.9.1 Loss of Load to Both Steam Generators

14.9.1.1 General

The loss of load to both steam generators event is analyzed to ensure that the peak RCS pressure remains below 110% of the design pressure (2750 psia) in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Another criteria that needs to be satisfied is that a sufficient thermal margin be maintained in the hot fuel assembly to assure that DNB does not occur throughout the transient (Ref. 14.9-13).

The loss of load incident is defined as a rapid and large reduction of secondary system power demand which may be caused by a turbine trip which could result from a loss of external electrical load or abnormal variations in the electrical network frequencies. Other mechanisms that would result in the loss of the secondary steam flow include simultaneous closure of the turbine stop valves or main steam isolation valves. Partial to total reduction in heat removal capability from the reactor coolant system has the potential for core damage if appropriate protection were not provided.

Upon the loss of power demand, i.e., termination of the secondary system steam flow, the rate of heat removal from the primary system is considerably reduced. As a result, the reactor coolant temperature and pressure increase. The reactor coolant pressure continues to increase until a reactor trip on high pressurizer pressure is initiated terminating the event and the pressurizer safety valves open which mitigates the pressure increase (Ref. 14.9-1).

14.9.1.2 Applicable Industry and Regulatory Requirements

The Loss of Load to Both Steam Generators event is classified as an Anticipated Operational Occurrence (AOO) for which peak RCS pressure must remain below 110% of the design pressure per Section III of ASME Boiler and Pressure Vessel Code. Also, a sufficient margin must be maintained in the LCO to prevent DNB from happening during the transient.

The reactor protective system provides reactor protection through a reactor trip initiated by either the high pressurizer pressure trip or the thermal margin/low pressure trip. Although a turbine trip would initiate a reactor trip, the turbine trip is an equipment trip and is not safety grade. Thus, the turbine trip is not credited in analyzing this event.

The plant is designed to accept a 10 percent step reduction in load without actuating a reactor trip. In the event of a complete loss of load, the steam dump and bypass system and the PORVs are available to remove energy from the reactor coolant system. In the transient safety analyses, no credit is taken for the steam dump and bypass system and the PORVs. However, the pressurizer and steam generator safety valves provide assurance that both the reactor coolant system and steam generator pressures would not exceed design limits.

14.9.1.3 Methods of Analysis

The analysis of loss of load to both Steam generators was performed using the CESEC digital computer simulation code (Ref. 14.9-3 to 6). The simulation includes neutron kinetics with fuel and moderator temperature feedback, the effect of the shutdown group of CEAs and the reactor coolant and main steam systems including steam dump and bypass valves. The initial pressurizer pressure is chosen such that it would result in a maximum RCS peak pressure.

14.9.1.4 Inputs and Assumptions

The reactor trip credited in the safety analysis is the pressurizer high pressure trip resulting from the RCS pressure spike upon loss of load. Another reactor trip which would provide protection is the TM/LP trip, however, this automatic trip is not credited. Table 14.9-1 shows the major input parameters.

14.9.1.5 Results

Two cases have been analyzed to ensure that the acceptance criteria of Section 14.9.1.2 are satisfied.

MDNBR Case:

The case which results in the minimum DNBR was analyzed in Cycle 6 (Ref. 14.9-7, 14.9-8 and 14.9-9) and was initiated from 2053 psia. The pressurizer spray and relief valves are assumed to be operable, but the steam dump and bypass to the condenser are assumed inoperable. Figures 14.9.1-1 through 14.9.1-4 show the plant responses for this case. The increase in primary pressure at an average rate of approximately 50 psi/sec is not as rapid in this case as in the peak RCS pressure case. This is the result of the pressurizer spray and relief valves operation which also delays reactor trip until about 8 seconds after initiation of the transient. Although the DNBR is lower for this case than for the peak PCS pressure case, it never decreases below the initial value. The safety valves are actuated at about 9 seconds and limit the primary pressure to 2500 psia.

Peak RCS Pressure Case:

In order to bound the effects of primary and secondary valves piping pressure drop, the loss of load event was reanalyzed for Cycle 17 assuming a maximum primary safety valve drift up to 1% and a secondary safety valve setpoint drift of 3%.

The analysis (Ref. 14.9-1) was performed to ensure that the RCS peak pressure upset limit of 2750 psia would not be exceeded. The Loss of Load event with a primary safety setpoint drift of 1%, initiated from the conditions given in Table 14.9.1-1 results in a high pressurizer pressure trip signal at 10.37 seconds. At 11.59 seconds, the RCS pressure reaches its maximum value of 2649 psia. At 16.10 seconds, the steam generator secondary side reaches its maximum value of 1081 psia.

An additional case, run to show the effects of a 6% primary safety setpoint drift, reached a maximum RCS pressure of 2736 psia at 12.30 seconds.

Table 14.9-2 presents the sequence of events for this transient. Figures 14.9-5 through 14.9-8 show the transient behavior of power, RCS pressure, and RCS coolant temperatures and steam generator pressure.

DNBR and peak linear heat rate (LHR) calculations were not performed for the Cycle 17 analysis, because the previous analysis (Cycle 6) confirmed that negligible changes/changes in the conservative direction in DNBR and peak LHR occur.

14.9.1.6 Affected Plant Technical Specifications

Changes in inputs from the following Technical Specifications may influence the validity of the current loss of load to both steam generators analysis:

- LSSS 1.3 Limiting Safety System settings, Reactor Protective System
- LCO 2.1.1 Operable Components
- LCO 2.1.6 Pressurizer and Steam System Safety Valves
- LCO 2.10.2 Reactivity Control Systems and Core Physics Parameter Limit

For specific parameters involved, refer to Table 14.9-1.

14.9.1.7 Affected Plant Systems

For this event, the affected plant systems are the reactor coolant system, reactivity control system, reactor protective system (high pressurizer pressure trip) and secondary steam system. Some of the specific system parameters affected are provided in Table 14.9-1.

14.9.1.8 Limiting Parameters for Reload Analysis

Reevaluation of the loss of load to both steam generators event analysis is required when any of the following conditions exist:

- Thermal hydraulic parameters change (e.g. RCS temperature, RCS pressure, etc) in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent technical specification LCO.
- A change of system configuration or operation that may change any of Table 14.9-1 parameters in a nonconservative direction.

Table 14.9-1 - "Key Parameters Assumed in the Loss of Load to Both Steam Generators Analysis Cycle 17"

System/Parameter	Units	Value	Source	Non-Conservative Affected Tech. Spec. Value
Initial Core Power Level	MWt	1535.6	Ref. 14.9-1	Full power plus RCP heat dissipation and uncertainty (Ref. 14.9-2, Rated Power definitions, pg. 1).
Reactor Coolant System				
Initial Core Inlet Coolant Temperature	°F	547	Ref. 14.9-1	Maximum Allowed plus uncertainty (Ref. 14.9-2, Section 2.10.4(5)(a)(i) requires ≤543°F nominal)
Initial RCS Flow Rate	gpm	192,000	Ref. 14.9-1	Minimum Allowed (Ref. 14.9-2, Section 2.10.4 (5)(a)(iii) requires ≥197,000 gpm)
Pressurizer Pressure	psia	2053	Ref. 14.9-1	Minimum Allowed minus uncertainty (Ref. 14.9-2, Section 2.10.4 (5)(a)(ii) requires ≥2075 psia nominal)
Reactor Protective System				
High Pressurizer Pressure	psia	2422	Ref. 14.9-1	Maximum allowed plus uncertainty (Ref. 14.9-2, Section 1.3(3) requires high pressurizer pressure trip at 2400 psia)
Reactivity Control System				
Moderator Temperature Coefficient	10	+0.5	Ref. 14.9-1	Maximum Allowed (Ref. 14.9-2, Section 2.10.2(3) requires less positive +0.5 at or above 80% power, less positive than +0.2 at below 80% of rated power)
CEA Holding Coil Delay	sec	0.5	Ref. 14.9-1	N/A
CEA Time to 100% Insertion (Including Holding Coil Delay)	sec	3.1	Ref. 14.9-1	Maximum (Ref. 14.9-2, Section 2.10.2(8) requires ≤2.5 sec to 90% insertion)
CEA Worth at Trip (all rods out)	10	-6.12	Ref. 14.9-1	N/A
Miscellaneous				
SG Tubes plugged	% of total	20	Ref. 14.9-1	N/A
Initial Secondary Pressure	psia	800	Ref. 14.9-1	N/A
Charging Flow	gpm	116	Ref. 14.9-1	Three pump operation (nominal three pump flow) . Ref. 14.9-2, Section 2.2.

Table 14.9-2 - "Fort Calhoun Cycle 14 Sequence of Events for the Loss of Load Event to Maximize Calculated RCS Peak Pressure"

Time (sec)	Event	Setpoint or Value
0.1	Loss of Secondary Load	-----
9.1	Steam Generator Safety Valves Open	1044.4 psia
10.37	High Pressurizer Pressure Analysis Trip Signal is Generated	2422 psia
10.63	Pressurizer Safety Valves Open	2564 psia*
11.59	Maximum RCS Pressure	2649 psia
16.10	Maximum Steam Generator Pressure	1081 psia

* Includes 1% drift and run pipe losses on nominal setpoint.

14.9.2 Loss of Load to one Steam Generator

14.9.2.1 General

The transients resulting from the malfunction of one steam generator are analyzed to determine the thermal margin requirements which must be built into the LCO's to prevent the DNBR and fuel centerline melt (kW/ft) SAFDLs from being exceeded.

The four events which affect a single generator are:

1. Loss of load to one steam generator;

2. Excess load to one steam generator;

3. Loss of feedwater flow to one steam generator; and

4. Excess feedwater flow to one steam generator.

Of the four events listed above, it has been determined that the Loss of Load to One Steam Generator (LL/1SG) transient is the limiting asymmetric event. Hence, only the results of this analysis are reported (Ref. 14.9-13).

The event is initiated by the inadvertent closure of a single main steam isolation valve. Upon the loss of load to the single steam generator, its pressure and temperature increase to the opening pressure of the secondary safety valves. The intact steam generator "picks up" the lost load, which causes its temperature and pressure to decrease, thus causing the core average inlet temperature to decrease and enhance the asymmetry in the reactor inlet temperature. In the presence of a negative moderator temperature coefficient this causes an increase in core power and radial peaking. Thus, the most negative value of this coefficient is used in the analysis. With this assumed sequence of events, the LL/1SG event results in the greatest asymmetry in core inlet temperature distribution and the most limiting DNBR for the transients resulting from the malfunction of one steam generator.

14.9.2.2 Applicable Industry and Regulatory Requirements

The loss of load to one steam generator event is an Anticipated Operational Occurrence (AOO) for which thermal margin must be built into the LCO to prevent the DNBR and fuel centerline melt (kW/ft) SAFDL's from being exceeded.

Maintaining the DNBR and fuel centerline melt within the SAFDLs is achieved by the timely intervention of the Reactor Protective System (RPS) in conjunction with building sufficient margin into the LCOs.

14.9.2.3 Method of Analysis

The Loss of Load to One Steam Generator was performed using the CESEC III digital computer simulation code.

The transient heat fluxes, core mass flow rate, inlet temperature, RCS pressure, and F_R^T are then used as input to the CETOP code which uses the CE-1 correlation for performing DNBR calculations for the limiting channel as a function of time.

14.9.2.4 Inputs and Assumptions

The reactor trip credited in the Safety Analysis is the Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) which is based on a differential pressure trip setpoint between the two steam generators. Table 4.1.12-1 shows the major input parameters.

The event was simulated at full power assuming that:

1. A single main steam isolation valve closes instantaneously, isolating the steam flow from the associated steam generator. Therefore, the temperature and pressure of the isolated steam generator increases until the secondary safety valves open.

2. The unaffected steam generator picks up the load lost by the isolated steam generator resulting in a core inlet temperature asymmetry due to the overcooling of this loop and undercooling from the isolated steam generator.

3. The most negative moderator temperature coefficient of reactivity permitted by the Technical Specifications in conjunction with the most negative Doppler coefficient (with a 1.15 multiplier) is used to maximize power peaking in the colder half of the core.

4. The most reactive CEA is assumed to be stuck in the fully withdrawn position. The scram worth and response utilized correspond to initiation from the Technical Specification PDIL and a top peaked axial shape.

14.9.2.5 Results

The Loss of Load to One Steam Generator was analyzed for Cycle 9 (Reference 14.9-11) to determine the minimum initial margin that must be maintained by the LCOs such that in conjunction with the RPS ASGTPTF, the DNBR limit will not be violated. The LL/1SG was conservatively assumed to be initiated at the initial conditions given in Table 4.1.12-1 with an axial shape index of -0.182 which bounds the DNBR-related axial shape index LCO. A reactor trip is generated by the Asymmetric Steam Generator Trip at 3.0 seconds based on high differential pressure between the steam generators.

Table 14.9.2-2 presents the sequence of events for the Loss of Load to One Steam Generator event. The transient behavior of key NSSS parameters is presented in Figures 14.9-9 to 14.9-14. The minimum transient DNBR calculated for the LL/1SG event is 1.53 as compared with the acceptable CE-1 correlation DNBR limit of 1.22. Note that in Cycle 9 the NRC approved CE-1 correlation limit was 1.18, with the application of the statistical combustion of uncertainties (Ref. 14.9-14) the limit became 1.22. In Reference 14.9-12 the NRC approved the CE-1 correlation with a 1.15 limit and the corresponding Fort Calhoun limit became 1.18.

A maximum allowable initial linear heat generation rate which can exist as an initial condition without exceeding the acceptable fuel to centerline melt of 22 kW/ft exceeds the LOCA Linear Heat Rate LCO and is thus less limiting for this event.

It may be concluded that the LL/1SG event, when initiated from the extremes of the LCOs in conjunction with the ASGTPTF will not lead to DNBR or centerline fuel temperatures which exceed the DNBR and centerline to melt design limits. It is worthy to note that the ROPM for this event was calculated to be approximately 105% which is significantly less than Loss of Flow, CEA Drop-CEA Withdrawal. Therefore, the event does not need to be reanalyzed on any "frequent basis".

14.9.2.6 Affected Plant Technical Specifications

Changes in inputs from the following Technical Specifications may influence the validity of the current loss of load to one steam generator analysis.

•	LSSS 1.3	Reactor Protective System
•	LCO 2.1.1	Operable Components
•	LCO 2.1.6	Pressurizer and Steam System Safety Valves
•	LCO 2.10.2	Reactivity Control Systems and Core Physics Parameter Limits.

For specific parameters involved, refer to Table 14.9.2-1

14.9.2.7 Affected Plant Systems

For this event, the affected plant systems are the reactor coolant system, reactivity control system, reactor protective system, (ASGTPTF) and secondary steam system. Some of the specific system parameters affected are provided in Table 14.9.2-1.

14.9.2.8 Limiting Parameters for Reload Analysis

Reevaluation of the loss of load to one steam generator event analysis is required when any of the following conditions exist:

- Thermal hydraulic parameters change (e.g. RCS temperature, RCS pressure, etc) in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent technical specification LCO.
- A change of system configuration or operation that may change any of Table 14.9.2-1 parameters in a nonconservative direction.

14.9.3 Specific References

14.9-1 EA-FC-97-004, "Evaluation of the Effect of Increased Line Pressure Drop on MSSVs and PSVs Setpoints", Rev. 0.

14.9-2 Ft. Calhoun Station, Unit No. 1, Operating License DRR-40 and Technical Specifications.

14.9-3 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.9-4 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.9-5 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.9-6 Response to questions on CESEC, CEN-234(c)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.9-7 "CETOP: Thermal Margin Model Development," CE-NPSD-150-P, CE Proprietary Report, May 1981.

14.9-8 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution," CENPD-162-P-A, September 1976.

14.9-9 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution," CENPD-207-P, June 1978.

14.9-10 "NX-NF-79-79 Plant Transient Analysis for the Fort Calhoun Reactor at 1500 MWt". Exxon Nuclear Company, EX-NF-79-79, October 1979.

14.9-11 Loss of Load to One Steam Generator Event, OSAR 83-37, Cycle 9 Reload Analysis .

14.9-12 Fort Calhoun Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 122, June 1989.

14.9-13 CE Transient Analysis Methods for Fort Calhoun Unit 1, Part 1: Transient Input for Generating DNB and LHR Technical Specification Limits, CE NPSD-152-P, Revision 1-P, Combustion Engineering Proprietary, July, 1981.

14.9-14 "Statistical Combination of Uncertainties," Parts 1-3, Supplement 1-P to CEN-257(0)-P, August 1985.

Table 14.9-4 - "Key Parameters Assumed in the Loss of Load to One Steam Generator Analysis Cycle 9"

System/Parameter	Units	Value	Reference	Affected Tech. Spec. Value
Initial Core Power	MW_{th}	1535.6	Ref. 14.9-11	Full power plus RCP heat dissipation and uncertainty (Ref. 14-9-12 Rated Power Definitions, pg. 1)
Reactor Coolant System				
Initial Core Inlet Temperature	°F	547	Ref. 14.9-11	Maximum Allowed plus uncertainty (Ref. 14-9-12, Section 2.10.4(5)(a)(i) requires ≤545°F nominal).
Initial RCS Flow Rate	gpm	208,280	Ref. 14.9-11	N/A
Pressurizer Pressure	psia	2053	Ref. 14.9-11	Minimum Allowed minus uncertainty (Ref. 14-9-12, Section 2.10.4 (5)(a)(ii) requires ≥2075 psia nominal)
Reactor Protective System				
Asymmetric Steam Generator Transient Protection Trip Function (SGTPTF)	psia	175.0	Ref. 14.9-11	Actual Setpoint plus uncertainty. (Ref. 14-9-12, Section 1.3(9) requires setpoint ≤135 psid)
Reactivity Control System				
Doppler Coefficient Multiplier**	---	1.15	Ref. 14.9-11	N/A
Moderator Temperature Coefficient	$10^{4}\Delta p$/°F	-2.7	Ref. 14.9-11	Minimum Allowed (Ref. 14-9-12, Section 2.10.2(3) limits MTC to -2.7X10
CEA Holding Coil Delay	sec	0.5	Ref. 14.9-11	N/A
CEA Time to 100% Insertion	sec	3.1	Ref. 14.9-11	Maximum. (Ref. 14-9-12, Section 2.10.2(8) requires ≤2.5 sec to 90% insertion)
CEA Worth at Trip (all rods out)	$10^{4}\Delta p$/°F	-6.52	Ref. 14.9-11	N/A

Table 14.9-5 - "Fort Calhoun Cycle 9 Sequence of Events for Loss of Load to One Steam Generator"

Time (sec)	Event	Setpoint or Value
0.0	Spurious closure of a single main steam isolation valve	---
0.0	Steam flow from unaffected steam generator increases to maintain turbine power	---
3.0	ASGTPTF* Trip Signal Generated	175 psid
3.8	Safety valves open on isolated Steam Generator	1015 psia
3.9	Trip Breakers open	---
4.4	CEA's begin to drop into core	---
4.7	Minimum DNBR occurs	1.53
6.1	Maximum Steam Generator Pressure	1051 psia

*ASGTPTF - Asymmetric Steam Generator Transient Protection Trip Function

14.11 EXCESS LOAD INCREASE

14.11.1 General

An excess load transient is defined as any rapid increase in steam generator steam flow other than a steam line rupture (discussed in Section 14.12). Such rapid increases in steam flow result in a power mismatch between the reactor core power and steam generator load demand. In addition, there is a decrease in reactor coolant temperature and pressure. Under these conditions the negative moderator temperature coefficient of reactivity causes an increase in core power.

The nuclear steam supply system is designed to accept ramp increases in load up to 10% per minute or step increases in load of up to 10% of full power. The variable high power trip provides a high power trip at 107% of full power when the plant is at power and at 20% of full power when the plant is under hot standby conditions. Protection against damage to the reactor core as a consequence of an excessive load increase is also provided by other trip signals, including high rate-of-change of power, thermal margin/low pressure, low steam generator water level, and low steam generator pressure.

In this section, the consequences of a rapid opening of the turbine admission valves, or the steam dump and bypass to condenser valves are discussed. The turbine valves are not sized to accommodate steam flow for powers much in excess of 1500 MWt. The steam dump valves and the steam bypass valve to the condenser are sized to accommodate 33% and 5%, respectively, of the steam flow at 1500 MWt. The hot full power increase incidents considered are:

Case (a) Rapid opening of the turbine control valves at hot full power: The maximum increase in steam flow due to the turbine control valves opening is limited by the turbine load limit control. The load limit control function is used to maintain load, so unless valve failure occurs the control valves will remain where positioned. If the turbine control valves are rapidly opened, a new steady state condition is attained without initiating a reactor trip. Reactor coolant temperature and pressure decrease somewhat with a corresponding small increase in reactor power level.

Case (b) Opening of all dump and bypass valves at hot full power due to steam dump control interlock failure: The circuit between the steam dump controller and the dump valves is open while the turbine-generator is on-line. Accidental closing of the steam dump control interlock under full load conditions would, according to the temperature program of the controller, cause full opening of the dump and bypass valves. Since the reactor coolant temperature decreases during the event, these valves would be closed again after the average reactor coolant temperature decreased to 535°F. The turbine admission valves would close on reactor trip. When the steam dump and steam bypass valves are suddenly opened there is a larger and more rapid increase in reactor coolant temperature and pressure than for Case (a). The resulting increase in reactor power level is also correspondingly larger, resulting in a reactor trip on high power.

14.11.2 Applicable Industry and Regulatory Requirements

The Excess Load Increase is classified as an anticipated operational occurrence (A00) for which the transient minimum DNBR and the peak linear heat rate must not exceed the DNBR and LHR SAFDLs.

These requirements are met by building sufficient margin into the DNB and LHR LCOs to ensure that protection is provided by the RPS and initial margin in the LCOs. DNB and LHR ROPMs are calculated for the Excess Load event, and are then compared to the ROPMs calculated for the AOOs such as CEA Withdrawal and CEA Drop to determine the limiting ROPMs which are to be incorporated into the LCO calculations.

14.11.3 Methods of Analysis

The methodology for the Excess Load event is described in References 14.11-1, 14.11-11 and 14.11-13. The Excess Load event is analyzed using the CESEC computer code (Refs. 14.11-2, 14.11-3, 14.11-4 and 14.11-5). The CESEC code models neutron kinetics with fuel and moderator temperature feedback, the reactor control system, the reactor coolant system, the steam generators, and the main steam and feedwater systems. The results of the transient simulation are input to the TORC computer code (Ref. 14.11-6) which uses the CE-1 critical heat flux correlation (Refs. 14.11-7 and 14.11-8) to calculate the minimum DNBR for the hot channel. The DNB ROPM is calculated to account for the degradation of margin between the start of the event and the time of MDNBR.

14.11.4 Excess Load Increase From Hot Full Power

Being the limiting excess load event, Case (b) of Section 14.11.1 was reanalyzed for Cycle 19 using the methodology described in References 14.11-1, 14.11-11 and 14.11-13 to verify that the minimum DNBR during the event is greater than the SCU SAFDL 1.18 using the CE-1 correlation. The results of this analysis (Reference 14.11-10) showed that the minimum DNBR reached during the event is 1.284 at 60.3 seconds.

The results described in this section demonstrates that the excessive load increase events are not limiting transients, and the resulting CE-1 DNB ratio is maintained well above the SCU SAFDL of 1.18. The key parameters assumed are per Table 14.11-1. The sequence of events are per Table 14.11-2. The results are shown in Figures 14.11-1 through 14.11-7.

14.11.5 DNB and LHR Required Overpower Margin

The Excess Load event was reclassified in Cycle 14 from an event that is protected by the Thermal Margin/Low Pressure (TM/LP) trip to one that is protected by sufficient initial margin maintained by the Limiting Conditions for Operation (LCOs) and by the Reactor Protective System.

In previous cycles, including Cycle 13, the Excess Load event was protected by the TM/LP trip which required that a γ-bias factor be calculated and incorporated into the γ term of the TM/LP equation. The Excess Load event is now protected by the RPS and sufficient initial margin which is maintained by the LCOs (Ref. 14.11-9). This change does not result in a net gain in margin. It only transfers the margin requirements from the LSSS to the LCO.

The methodology for calculating the DNB and LHR ROPMs is described in Reference 14.11-1. For Cycle 19 the DNB ROPM was determined to be 115.48% and the LHR ROPM was determined to be 117.24%. These values will be utilized in the setpoint analysis for determining the DNB and LHR LCOs for Cycle 19.

14.11.6 Affected Plant Technical Specifications

The following Technical Specifications are used as input to the Excess Load analysis.

- SL 1.1 Safety Limits - Reactor Core
- LSSS 1.3 Limiting Safety System Settings, Reactor Protective System
- LCO 2.10.2 Reactivity Control Systems and Core Physics Parameter Limits
- LCO 2.10.4 Power Distribution Limits

14.11.7 Affected Plant Systems

For this event the affected plant systems are the reactor coolant system, the reactor protective system (TM/LP, VHPT), the reactivity control system, the main steam system, and the condenser. The specific system parameter affected are provided in Table 14.11-1.

14.11.8 Limiting Parameters for Reload Analysis

Reevaluation of the Excess Load event is required when either of the following conditions exist:

- Core physics and/or thermal-hydraulic parameters change in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent Technical Specification Limiting Condition of Operation (LCO).

Any changes to parameters and/or Technical Specifications must result in a DNBR and peak linear heat rate which do not exceed the SAFDLs.

14.11.9 Specific References

14.11-1 OPPD-NA-8303-P, Rev. 04, "Omaha Public Power District Reload Core Analysis Methods Transient and Accident Methods and Verification," Section 5.6, January 1993.

14.11-2 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.11-3 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.11-4 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.11-5 Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.11-6 "Users Manual for TORC" CE-NPSD-628-P, Rev. 01-P through 04-P, CE Proprietary Report, March 1994.

14.11-7 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution," CENPD-162-P-A, September 1976.

14.11-8 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Uniform Axial Power Distribution," CENPD-207-P, June 1978.

14.11-9 Ft. Calhoun Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 172, December 1995.

14.11-10 "Cycle 19 Excess Load Analysis," EA-FC-98-052, Rev. 0.

14.11-11 "CE Transient Analysis Methods for Fort Calhoun Unit 1, Part 1," CENPDS-152-P, July 1981.

14.11-12 "CESEC Code Verification and Cycle 18 Update," EA-FC-97-017, Rev. 0.

14.11-13 O-90-003, T. G. Ober (CE) to W. O. Weber (OPPD), "Methodology for the Excess Load Event," January 12, 1990.

14.11.10 General References

14.11.10-1 "Main Steam and Turbine Steam Extraction Design Basis Document," SDBD-MS-125, Rev. 0, Attachment 0, March 1989.

Table 14.11-1 - "Key Parameters Assumed in the Excess Load Analysis Cycle 19"

System/Parameter	Units	Value	Reference	Affected Tech. Spec. Value
Initial Core Power Level	MWt	1531.85*	14.11-10	(Ref. 14.11-9, Rated Power Definitions, pg 1) including Reactor Coolant Pump Heat
Reactor Coolant System				
Initial Core Inlet Coolant Temperature	°F	545*	14.11-10	Maximum allowed plus uncertainty (Ref. 14.11-9, Section 2.10.4(5)(a)(i)) within the limit for Core Inlet Temperature provided in the COLR
Initial RCS Flow Rate	gpm	202,500	14.11-10	**
Pressurizer Pressure	psi	2075*	14.11-10	Minimum allowed minus uncertainty (Ref. 14.11-9) Section 2.10.4(5)(a)(ii) requires ≥2075 psia (nominal)
Reactivity Control System				
Moderator Temperature Coefficient	10^{-4} Δρ/°F	-1.17	14.11-10	
CEA Worth at Trip (PDIL) Total Integrated Unrodded	10^{-2} Δρ	5.501	14.11-10	N/A
Radial Peaking Factor ($F_R T$)		1.732	14.11-10	
Reactor Protective System				
Higher Power Trip	% of rated power	112	14.11-10	(Ref. 14.11-9, Section 1.3(1))

* For DNBR calculations, the uncertainties on these parameters have been statistically combined.

** Nominal flow (Reference 14.11.9, Section 2.10.4(5)(a)(iii)) plus uncertainties.

Table 14.11-2 - "Sequence of Events for Cycle 19 (Reference 14.11-10)"

Time	Event	Setpoint or Value
0	Steam Dump and Bypass Valves Open	----
58.8	High Power Trip Conditions Reached	112% of Rated Power
60.2	High Power Trip Signal Generated	112% of Rated Power
60.3	Minimum DNBR Value Reached	>1.18

14.12 MAIN STEAM LINE BREAK ACCIDENT

14.12.1 General

The main steam line break accident was reanalyzed for Cycle 19 (Ref. 14.12-16). It was determined that peak LHR and minimum DNBR did not violate their respective SAFDLS.

In the event of a large pipe break in the main steam system, rapid depletion of the steam generator inventory causes an increased rate of heat extraction from the primary coolant. The resultant cooldown of the primary coolant, in the presence of a negative moderator temperature coefficient of reactivity, will cause an increase in nuclear power and trip the reactor. If the most reactive control element assembly (CEA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will return to power and criticality.

A severe decrease in main steam pressure will also initiate a reactor trip on low steam generator pressure and cause the main steam line isolation valves to trip closed. If the steam line rupture occurs between the isolation valve and the steam generator outlet nozzle, blowdown of the affected steam generator would continue until the steam generator inventory is depleted. (However, closure of the check valve in the ruptured steam line, as well as closure of the isolation valves in the unaffected steam lines, will terminate blowdown from the intact steam generator.) The fastest blowdown, and therefore, the most rapid reactivity addition, occurs when the break is at a steam generator nozzle. This break location is assumed for the cases analyzed. Inadvertent opening of valves in the main steam system is discussed in Section 14.11 (Excess Load Increase event).

The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

14.12.2 Applicable Industrial and Regulatory Requirements

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

Site boundary doses do not exceed the guidelines of 10 CFR 100 (Ref. 14.12-1). Acceptable doses are demonstrated by showing that the peak LHR and minimum DNBR do not violate their respective SAFDLS.

Although DNB and possible clad 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 control element assembly stuck in its fully withdrawn position.

The objective of the radiological analysis is to ensure that the site boundary doses following the accident are within the 10 CFR 100 limits (Ref. 14.12-1). These limits are divided into two parts, as follows:

(1) A person located at the Exclusion area Boundary for two hours immediately following the onset of a postulated fission product release would receive a total radiation dose of no greater than 25 Rem to the whole body or 300 Rem to the thyroid from iodine exposure.

(2) A person located at the Low Population Zone during the entire period of the passage of the postulated fission product release would receive a total radiation dose of no greater than 25 Rem to the whole body or 300 Rem to the thyroid from iodine exposure.

14.12.3 Method of Analysis

The analyses of the main steam line break (MSLB) accident are performed using the digital computer code CESEC (Refs. 14.12-2,3,4 and 5) which models neutron kinetics with fuel and moderator temperature feedback, the reactor protection system, the reactor coolant system, the steam generators, and the main steam and feedwater systems.

14.12.4 Inputs and Assumptions

The main steam line break accident is reviewed for each reload cycle (Ref. 14.12-9).

Technical Specification 2.1.1 prohibits operation with less than four reactor coolant pumps in use (with the exception of physics testing done at less than 10^{-1} percent power). Both full power and no-load (hot standby) initial condition cases are considered for two-loop operation (i.e., four reactor coolant pumps). The objectives of the analysis are to demonstrate that the minimum DNBR and Peak LHR for the reload core no-load two-loop and full-load two-loop main steam line break cases do not violate their respective SAFDLS.

Since the steam generators are designed to withstand reactor coolant system operating pressure on the tube side with atmospheric pressure on the shell side (Ref. 14.12-7), the continued integrity of the reactor coolant system barrier is assured.

The MSLB accident is assumed to start from steady state conditions with the initial power being 1530 MWt (102%) for the full power case and 1 MWt for the no load case. The reactor coolant system cooldown causes the greatest positive reactivity insertion into the core when the moderator temperature coefficient (MTC) of reactivity is the most negative. For this reason the COLR negative MTC limit corresponding to the end-of-cycle is assumed for the analysis. Since the reactivity change associated with moderator feedback varies significantly over the temperature range covered in the analysis, a curve of reactivity insertion versus temperature rather than a single value of MTC is assumed. The RCS cooldown curves utilized for Cycle 19 is shown in figure 14.12-1. The cooldown curve for Cycle 17 is shown for reference purposes only.

These curves are derived on the basis that upon reactor trip the most reactive CEA is stuck in the fully withdrawn position thus yielding the most adverse combination of scram worth and reactivity insertion. Although no single value of MTC is assumed in the analysis, the moderator cooldown reactivity function is calculated assuming an initial MTC equal to the most negative Technical Specification limit, i.e., $-2.3 \times 10^{-4} \Delta\rho/°F$ for Cycle 1, and $-2.5 \times 10^{-4} \Delta\rho/°F$ for Cycle 8, and the most negative COLR limit of $-3.5 \times 10^{-4} \Delta\rho/°F$ for Cycle 19.

The moderator density reactivity insertion curve for the hot zero power steam line break case is calculated by successively lowering the inlet temperature of the SIMULATE-3 Computer Code (Ref. 14.12-8) model from 532°F and allowing only moderator temperature feedback in the model. The moderator density reactivity insertion curve for the full power case is calculated by decreasing the power level and core average coolant temperature from full power to the hot zero power inlet temperature and then successively lowering the inlet temperature as in the hot zero power case. Only moderator temperature feedback is utilized in the SIMULATE- 3 model. Since the moderator reactivity insertion curve corresponds to an MTC which is bounded by the EOC MTC COLR limit, no additional uncertainty is added to this curve.

Reactivity feedback effects from the variation of fuel temperature (i.e., Doppler) are included in the analysis. The most negative Doppler effect function, when used in conjunction with the decreasing fuel temperature, causes the greatest positive reactivity insertion during the MSLB event. For Cycle 19, in addition to assuming the most negative Doppler feedback function, a 1.4003 multiplication factor was used which resulted in a larger return-to-power. The Doppler multiplier is a cycle specific value calculated from reactor physics methods.

The Doppler reactivity insertion for the hot zero power case is determined in the same manner as the HFP case. The fuel temperature feedback in the simulate-3 model allows the production of a curve of Doppler reactivity as a function of fuel temperature. All zero power calculations are performed assuming there is no decay heat and no credit is taken for local voiding in the region of the stuck CEA.

The minimum Beta fraction at EOC conditions with uncertainties was the most limiting. This beta fraction maximizes the return to power and was used for this event.

The most probable trip signals resulting from a MSLB (Ref. 14.12-9) include low steam generator pressure, high power, low steam generator level, thermal margin/low pressure, and high rate-of-change of power (for the no-load case). The steam generator low pressure trip, which occurs at 478 psia (including a 22 psia uncertainty below the nominal trip setting of 500 psia), is the trip assumed in the analysis. No credit is taken for the high power trip which occurs at approximately the same time for the full power case. For the cases analyzed, it is assumed that the most reactive CEA was stuck in the withdrawn position. The CEA configuration at no load operation is such that the most reactive CEA of those in the withdrawn position is worth less than the most reactive CEA of those withdrawn at full power. If all CEAs insert (no stuck CEA), there is no return to criticality and no power transient following trip.

The power distribution following CEA insertion is distorted by the stuck CEA. The coincident high radial peaking and low reactor coolant pressure can lead to local boiling at moderate power levels. The power flattening effect of the voids and of the locally high fuel temperature is included in the analysis, but no credit is taken for the corresponding reactivity feedback. In addition, cold edge temperatures are used to calculate moderator reactivity insertion during the cooldown, thus maximizing the return-to-critical and return-to-power potentials. The computed power peaks after trip are thus conservative.

The Emergency Operating Procedures were revised during operation of Cycle 11 (eleven) to implement the Trip 2/ Leave 2 RCP trip Strategy (Ref. 14.12-15).

The MSLB case with the RCPs tripped is similar to the MSLB case with a loss of offsite power (LOOP) since the RCPs coastdown in both events. As discussed in Reference 14.12-10, the loss of offsite power delays safety injection due to the time delay for the emergency diesel generators to restore power to the safety injection pumps and causes a coastdown of the RCPs. The coastdown affects the degree of overcooling and increases the time for safety injection borated water to reach the core midplane.

Because manual tripping of the RCPs results in a later coastdown of the RCPs and because safety injection is not delayed since offsite power is available (i.e., the diesel generator startup and pump loading delays are not present), the injected boron will arrive at the core midplane sooner for a MSLB with the RCPs tripped than for a MSLB with a loss of offsite power. Therefore, the reactivity effects of a MSLB with the RCPs tripped are less severe than for the MSLB with a loss of offsite power.

Reference 14.12-6 states that the MSLB case with a loss of offsite power results in the injection boron being dominant over the RCS cooldown and concludes that the reactivity effects of a MSLB accident would be reduced in severity with a concurrent loss of offsite power when compared to the same event with offsite power available and the RCPs operating. Because the reactivity effects of a MSLB with the RCPs tripped after Safety Injection Actuation Signal (SIAS) are less severe than a MSLB with a concurrent loss of offsite power, it is concluded that the reactivity effects for the MSLB case with the RCPs tripped after SIAS are less severe than for a MSLB with offsite power available and RCPs operating (Ref. 14.12-6). Therefore, to maximize the severity of the reactivity effects, the Cycle 19 MSLB analysis was performed with the four reactor coolant pumps operating at the limiting condition of operation volumetric flow rate.

The reactor coolant volumetric flow rate is assumed to be constant during the incident. A flow rate of 197,000 gpm was used in Cycle 19 in order to obtain the most adverse results. A lower flow rate increases the initial fuel and average primary coolant temperature and consequently results in a higher steam generator pressure and a greater steam generator mass inventory.

These effects cause a longer blowdown, a greater blowdown rate, and a greater decrease in average primary coolant temperature. After MSIV closure the lower flow rate decreases the rate of reverse heat transfer from the intact steam generator, thereby increasing the heat extracted from the primary system by the ruptured steam generator. The overall effect is that the potential for a return-to-power is maximized.

Maximum values for the heat transfer coefficient across the steam generator are used for the no-load initial condition case, while nominal values are used for the full-load initial condition. These heat transfer coefficients result in the most severe conditions during the incident because of the shape of the reactivity versus moderator temperature function and the difference in average moderator temperature for the maximum and minimum values of the steam generator heat transfer coefficients.

The fast cooldown following a MSLB results in a rapid shrinking of the reactor coolant. After the pressurizer empties, the reactor coolant pressure is assumed to be equal to the saturation pressure corresponding to the highest temperature in the system.

No credit is taken for safety injection via HPSI pumps or charging flow.

Since the rate of temperature reduction in the reactor coolant system increases with rupture size and with steam pressure at the point of rupture, it is assumed that a circumferential rupture of a 26-inch (inside diameter) steam line occurs at the steam generator main steam line nozzle, with unrestricted blowdown. Critical flow is assumed at the point of rupture, and all of the mass leaving the break is assumed to be in the steam phase. This assumption results in the maximum heat removal from the reactor coolant per pound of secondary water, since the latent heat of vaporization is included in the net heat removal. A single failure of the reverse flow check valve in the ruptured steam generator is assumed; so that the intact steam generator will have steam flow through the unaffected steam line and back through and out the ruptured line. The analysis credits a choke which is installed in each steam line immediately above the steam generator and assumes the steam flow from the intact steam generator is through a 50% area reduction in a 24 inch steam line. This flow will be terminated upon MSIV closure.

The feedwater flow at the start of the MSLB corresponds to the initial steady state operation. For the full load initial condition, it is automatically reduced from 100 percent to 0.00 percent within 40 seconds following a steam generator isolation signal. For the no load initial condition, feedwater flow is assumed to match energy input by the reactor coolant pumps and the 1 MWt core power.

Table 14.12-1 contains the conditions from which the Cycle 19 no-load, two-loop MSLB event was initiated and the assumptions used. It also lists the Technical Specifications affected by the inputs and assumptions.

Table 14.12-3 contains similar inputs and assumptions for Cycle 19 full load, two-loop operation.

14.12.5 Results

The MSLB event case initiated from HFP was simulated for 200 seconds using CESEC C89300mod5 with parameters that maximize the potential for Return to Power (R-T-P) or/and Return to Criticality (R-T-C). The limiting MSLB accident occurs with all RCPs running. The results of this case is per Table 14.12-4. This case shows a peak R-T-P of 25.61%, a peak reactivity of -0.069%$\Delta\rho$. The peak LHR and minimum DNBR did not violate their respective SAFDLS.

The MSLB event case initiated from HZP was simulated for 300 seconds using CESEC C89300mod5 with parameters that maximize the potential for R-T-P or/and R-T-C.

The HZP case was run with the TS 2.10.2(1) LCO requirement for Shutdown Margin of 4.0% $\Delta k/k$ substituted for scram worth. It is conservatively assumed that at the HZP condition the minimum CEA worth available for negative reactivity addition at time of trip will be equivalent to the minimum allowable Shutdown Margin of TS 2.10.2(1). The TS reactivity control limits require that whenever the reactor is in hot standby or power operation conditions with T_{cold} >210°F, a Shutdown Margin of $\geq$4%$\Delta\rho$ must be available.

The limiting HZP case shows a peak R-T-P of <1.0% and a peak reactivity of +0.172%$\Delta\rho$. The peak LHR and minimum DNBR did not violate their respective SAFDLs.

14.12.6 Radiological Consequences of a MSLB

The radiological consequences of main steam line break (MSLB) are determined based on the conservative assumption that there is a complete severance of a main steam line outside the containment with the plant in a hot zero power condition where the transient is initiated shortly after full power operation. The hot zero power condition assures the maximum water inventory in the steam generators and the shutdown from full power assures the maximum decay heat which must be removed by manual control of the Air Assisted Main Steam Safety Valve (MSSV) MS-291 or MS-292 associated with the intact steam generator. The MSIVs are installed in the main steam lines from each steam generator, downstream from the safety relief valves and Air Assisted MSSVs outside the containment. The MSLB is assumed to be upstream of the MSIV. Following a reactor trip, the affected steam generator blows down completely and the steam is vented directly to the atmosphere. Mass release from the intact steam generator is terminated when the shutdown cooling system is initiated at a reactor coolant system temperature of 300°F.

14.12.6.1 Methods of Analysis

The offsite doses for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) are calculated in accordance with the methods outlined in Reference 14.12-1. For the MSLB the gas gap activity from the fuel leaks into the secondary system from the primary system and is concentrated in the steam generator. Table 14.12-5 lists the Fuel Fission Product Inventory for the gas gap. The Whole Body Dose Source Calculation is shown below (Ref. 14.12-1):

Based on 1 Rod:

$$DEQ_{Xe_{133}} = K_{\gamma}\sum_{i=83}^{138} A_i * RDCF_{\gamma_i} * E_{\gamma} + K\sum_{i=83}^{138} A_i * RDCF_{\beta_i} * E_{\beta}$$

where,

$DEQ_{Xe\text{-}133}$	= Dose Equivalent Xe-133 (Rem-M^3/S)
K	= Conversion Factor (Rem-M3-Disintegration/Mev-Ci)
K_{γ}	= .25, for whole body gamma radiation
K_{β}	= .23, for skin beta radiation
A_i	= Activity For Noble Gas Isotope (Ci)

$RDCF_{\gamma i}$ or β_i = Relative Dose Conversion Factor For Noble Gas Isotope
E_γ or β = Average Gamma or Beta Decay Energy of Xe-133 (Mev/Disintegration)

The activity for the Noble Gas Isotopic Parameters is obtained from Table 14.12-6, while the conversion factors are contained in Table 14.12-7.

The Thyroid Dose Source Calculation is shown below (Ref. 14.12-1):

Based on 1 Rod:

$$DEQ_{I-131} = 2.5\ DCF_{I-131} \sum_{i=131}^{135} A_i * RDCF_i$$

where,

DEQ_{I-131}	=	Dose Equivalent I-131 (Rem)
A_i	=	Activity For Iodine Isotope (Curie Per Rod)
$RDCF_i$	=	Relative Dose Conversion Factor For Iodine Isotope
DCF_{I-131}	=	Dose Conversion Factor of Iodine-131 (Rem/Ci)

The thyroid dose calculation relates the iodine activity released to the affected body organ - the thyroid. The isotopic parameters for iodine as well as the relative dose conversion factors are shown in Tables 14.12-6 and 14.12-7, respectively.

The release path to the environment is from the Main Steam line in Room 81 to the atmosphere. The input parameters are described in Section 14.12-6. The steam release is determined by the MSLB analyses in Section 14.12-5. After the release path has been determined the total quantity released at the end of the 2 hour limit is evaluated as outlined below (Ref. 14.12-1):

1. First Find Total Heat Generation Rate During Cooldown

$$\dot{Q}_T = \dot{Q}_C + \dot{Q}_D + \dot{Q}_P$$

where

$\dot{Q}_C$ = (Heat Stored in Metals on Primary Side + Heat Stored in Primary and Secondary Water + Heat Stored in Pressurizer+ Water + Heat Stored in Pressurizer Steam) x (cooldown rate)

$\dot{Q}_D$ = $\bar{P}$ ($\bar{P}$ = Average Power Produced, BTU/sec)

$$\dot{Q}_P = \frac{\text{\# Pumps During DBE}}{\text{\# Pumps Initially}} \text{ (Pump Heat)}$$

2. Calculate Steam Release Rate

$$\dot{W}_S = \frac{\dot{Q}_T}{_{Hg}MIN - H_{AFW}}$$

where

$\dot{Q}_T$ = Taken from Step 1

$_{Hg}MIN$ = Minimum Enthalpy of Secondary Steam

H_{AFW} = Enthalpy of AFW Flow

From the above values the secondary dose calculations for the whole body and thyroid can be completed utilizing the equations that follow:

Whole Body Dose =

$$\frac{DEQ_{Xe\text{-}133} * N * \chi/Q *}{V_{RCS}} \sum_{I}^{135} \frac{L * W_{STM} * t}{M_{SG}}$$

where,

$DEQ_{Xe\text{-}133}$ = Dose Equivalent Xe-133 (REM-M^3/S)

N = Number Of Failed Rods

V_{RCS} = Minimum RCS Volume (GAL)

χ/Q = Atmospheric Dispersion Factor (S/M^3)

L = Maximum Primary-To-Secondary Leak Rate During Time Interval t (Gal/Min)

W_{STM} = Steam Release In Time Interval t (LBM)

M_{SG} = Minimum SG Mass In Time Interval t (LBM)

t = Time Interval (Min)

Thyroid Dose =

$$\frac{DEQ_{Xe\text{-}131} * N * \chi/Q * Q *}{V_{RCS}} \sum_{I} \frac{L * W_{STM} * p * t}{M_{SG}}$$

where,

DEQ_{I-131}	= Dose Equivalent I-131 (REM-M^3/S)
N	= Number Of Failed Rods
V_{RCS}	= Minimum RCS Volume (Gal)
χ/Q	= Atmospheric Dispersion Factor (S/M^3)
B	= Breathing Rate (M^3/S)
L	= Maximum Primary-To-Secondary Leak Rate In Steam Interval t (Gal/Min)
W_{STM}	= Steam Release In Time Interval t (LBM)
M_{SG}	= Minimum SG Mass In Time Interval t (LBM)
p	= Partition Factor(s) In Time Interval t
t	= Time Interval (Min)

Additional input values are obtained from Section 14.12.6-2.

14.12.6.2 Inputs and Assumptions

The following assumptions are postulated in the calculation of radiological consequences:

(1) The reactor coolant equilibrium activity is based on long term operation at 100 percent of the ultimate core power level of 1500 MWt and 1% failed fuel. The RCS equilibrium activity is 60 μCi/gm DEQ I-131.

(2) The activity in the secondary coolant is assumed to be equal to 0.1 μCi/gm DEQ I-131.

(3) The primary-to-secondary leakage of 1 gpm was assumed to continue through the affected steam generator at a constant rate until shutdown cooling is initiated.

(4) Offsite power is lost; the main condenser is not available for steam relief via the turbine bypass system.

(5) The activity released from the steam generators is immediately vented to the atmosphere. No credit is taken for radioactive decay for isotopes in transit to the dose points.

(6) The mass of primary-to-secondary leakage for the 30-minute duration is 491 lbs.

(7) The secondary mass release to atmosphere from the affected steam generator is 233,498 lbs.

(8) A post-accident steam generator decontamination factor between steam and water phase is 1.0.

(9) The total activity released from the steam generator for various nuclides is provided in Table 14.12-10.

(10) The dispersion factors for the EAB and the LPZ outer boundary are 4.4 E-04 sec/m3, respectively.

(11) The adult breathing rate for the EAB and LPZ is 3.47 E-04 m^3/sec.

14.12.6.3 Results

Based on the above assumptions, the resulting doses are as follows:

	Thyroid (Rems)	Whole Body (Rems)
EAB	5.41	0.000566
LPZ	0.193 E-01	0.0000202

The results of radiological consequences due to a postulated MSLB are presented above. The values for thyroid dose and whole body dose show that the calculated doses using the conservative assumptions are well within the limits of 10 CFR Part 100.

14.12.7 Affected Plant Technical Specifications

The main steam line break accident analysis uses inputs from the following technical specifications:

- LSSS 1.3 Limiting Safety System Settings, Reactor Protective System
- LCO 2.1.1 Operable Components
- LCO 2.2 Chemical and Volume Control Systems
- LCO 2.5 Steam and Feedwater Systems
- LCO 2.10.2 Reactivity Control Systems and Core Physics Parameter Limits
- LCO 2.10.4 Power Distribution Limits
- LCO 2.14 Engineered Safety Features System Initiation Instrumentation Settings.

For the specific parameters involved, refer to Table 14.12-1 and Table 14.12-3.

14.12.8 Affected Plant Systems

For this accident, the affected plant systems are the reactor coolant system, the control element drive system, safety injection system, reactor protective system, chemical volume control system and the steam power conversion system. The specific system parameters affected are given in the text and in Tables 14.12-1 and 14.12-3.

14.12.9 Limiting Parameters for Reload Analysis

Reevaluation of the main steam line break event is required when any of the following conditions become nonconservative.

- Core physics, and/or thermal-hydraulic parameter changes (moderator cooldown curve and scram worth).
- A plant design modification is expected to cause a change to a pertinent technical specification limiting condition of operation (LCO).
- A plant design modification which causes a change to the system parameters described in Section 14.12.7.

Any changes to parameters and/or technical specifications must result in a return to power less than that calculated for Cycle 19.

14.12.10 Specific References

14.12-1 Code of Federal Regulations, Energy, 10 CFR 100, Reactor Site Criteria, Jan. 1979.

14.12-2 CESEC - "Digital Simulation of a Combustion Engineering Nuclear Steam Supply System" CENPD-107, CE Proprietary Report, April 1974.

14.12-3 CESEC - "Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, Supplement 6, CE Nonproprietary Report, August, 1979.

14.12-4 CESEC, December, 1981, Transmitted as enclosure 1-P to LD-82-001, January 6, 1982.

14.12-5 Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford 3, Docket 50-382, December, 1982.

14.12-6 OSAR 82-09, Cycle 8 Transient Analysis".

14.12-7 "Steam Generator Performance", CE Calculation T-601 rev. 0 dated May 20, 1968.

14.12-8 "Omaha Public Power District Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302-P, Rev. 04, May 1994.

14.12-9 "Omaha Public Power District Transient and Accident Methods and Verification", OPPD-NA-8303 Rev. 4, January 1993.

14.12-10 "Safety Analysis and Setpoints for Licensing of the Safety Grade Auto Feedwater", OSAR 81-03.

14.12-11 "CETOP: Thermal Margin Model Development", CE-NPSD-150-P, CE Proprietary Report, May, 1981.

14.12-12 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution", CENPD-162-P-A, CE Proprietary Report, September, 1976,.

14.12-13 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution'< CENPD-270-P, CE Proprietary Report, June, 1978.

14.12-14 Amendment 75 to Facility Operating License DPR-40, Omaha Public Power District Fort Calhoun Station, Unit 1, Docket No. 50-285.

14.12-15 "Uncontrolled Heat Extraction, EOP-05" Omaha Public Power District Fort Calhoun Station, Unit 1, Rev. 05, September 1994.

14.12-16 EA-FC-98-050, Rev. 0, "Cycle 19 Main Steam Line Break Analysis".

14.12-17 EA-FC-94-026, Rev. 0, "Cycle 16 Steam Line Break Analysis".

Table 14.12-1 - "Key Parameters Assumed in the Main Steam Line Break Analysis for HZP Operation"

Parameter	Units	Cycle 19
Initial Core Power*	MWt	1.0
Initial Core Inlet Temperature	°F	532
Initial Pressurizer Pressure	psia	2172
Initial Steam Generator Pressure	psia	890
Initial Steam Generator Water Mass Inventory	lbm	123,685
RCS Flow Rate	gpm	197,000
Minimum CEA Worth Available at Trip (Shutdown Margin)	%Δρ	-4.0
Doppler Multiplier		1.4003
Moderate Cooldown Curve	%Δρ vs temp.	See Figure 14.12-1
Effective MTC	$x10^{-4}\Delta\rho/°F$	-3.5
Inverse Boron Worth	ppm/%Δρ	-110.4
βFraction (including uncertainty)		0.005223
Min. MSL Stop Valve Closure Time	sec	4
SG Low Pressure Trip (MS + MF iso) (includes a 22 psi margin)	psia	478 psia

* Reactor coolant pump heat assumed to be zero.

Table 14.12-2 - "Sequence of Events for the Main Steam Line Break Event for HZP Operation"

TIME (sec)	EVENT	SETPOINT or VALUE
0.0	Main Steam Line Break Occurs	-
3.8	Low Steam Generator Pressure Trip	478 psia (Setpoint 500)
	Main Steam and Feedwater Isolation Signal	478 psia
4.8	Main Steam and Feedwater Isolation Valves Begin to Close	-
5.2	CEAs Begin to Drop into Core	-
8.8	Main Steam Isolation Valves Completely Closed	-
18.1	Pressurizer Empties	-
44.8	Main Feedwater Isolation Valves Completely Closed	-
93.2	Return-to-Critical	$>0.0\%\Delta\rho$
121.8	Peak Reactivity	$+0.172\%\Delta\rho$
138.6	Dryout of Ruptured Steam Generator	-
155.0	Subcritical	$<0.0\%\Delta\rho$

Table 14-12-3 - "Key Parameters Assumed in the Main Steam Line Break Analysis for HFP Operation"

Parameter	Units	Cycle 19
Initial Core Power*	MWt	1530 (=102% of 1500)
Initial Core Inlet Temperature	°F	547
Initial Pressurizer Pressure	psia	2172
Initial Steam Generator Pressure	psia	890
Initial Steam Generator Water Mass Inventory	lbm	76,329
RCS Flow Rate	gpm	197,000
Minimum CEA Worth Available at Trip (Shutdown Margin)	%ρ	-6.0914
Doppler Multiplier		1.4003
Moderate Cooldown Curve	%Δρ vs temp.	See Figure 14.12-1
Inverse Boron Worth	ppm/%Δρ	112.6
Effective MTC	$x10^{-4}$Δρ/°F	-3.5
βFraction (including uncertainty)		0.005223
Min. MSL Stop Valve Closure Time	sec	4
SG Low Pressure Trip (MS + MF iso) (includes a 22 psi margin)	psia	478

* Reactor coolant pump heat of 5.6 MWt not included in this value.

Table 14.12-4 - "Sequence of Events for the Main Steam Line Break Event for HFP Operation"

TIME (sec)	EVENT	SETPOINT or VALUE
0.0	Main Steam Line Break Occurs	-
3.9	Low Steam Generator Pressure Trip	478 psia (Setpoint 500)
	Main Steam and Feedwater Isolation Signal	478 psia
4.9	Main Steam and Feedwater Isolation Valves Begin to Close	-
5.3	CEAs Begin to Drop into Core	-
8.9	Main Steam Isolation Valves Completely Closed	-
19.3	Pressurizer Empties	-
44.9	Main Feedwater Isolation Valves Completely Closed	-
72.5	Peak Post-Trip Reactivity	-0.069%$\Delta\rho$
73.0	Peak Return to Power	18.50%
78.7	Dryout of Ruptured Steam Generator	-

Table 14.12-5 - "Fuel Fission Product Inventory"

Isotope	Core[1] Inventory (Ci)	Maximum Rod Gas[2] Gap Inventory (Ci)
Kr-83m	7.64(+6)	3.52(+1)
Kr-85	1.12(+6)	5.17(+0)
Kr-85m	1.59(+7)	7.33(+1)
Kr-87	2.84(+7)	1.31(+2)
Kr-88	4.08(+7)	1.88(+2)
Kr-89	4.86(+7)	2.24(+2)
Kr-90	4.74(+7)	2.18(+2)
Kr-91	3.45(+7)	1.59(+2)
Kr-92	1.80(+7)	8.29(+1)
Kr-93	6.87(+6)	3.17(+1)
Kr-94	2.34(+6)	1.08(+1)
I-128	1.70(+6)	7.83(+0)
I-131	7.51(+7)	3.45(+2)
I-132	1.09(+8)	5.02(+2)
I-133	1.47(+8)	6.78(+2)
I-133m	5.55(+6)	2.56(+1)
I-134	1.57(+8)	7.24(+2)
I-134m	1.85(+7)	8.52(+1)
I-135	1.36(+8)	6.26(+2)
I-136	6.04(+7)	2.79(+2)
I-136m	3.53(+7)	1.62(+2)
I-137	5.82(+7)	2.69(+2)
I-138	2.90(+7)	1.34(+2)
I-139	1.39(+7)	6.41(+1)
I-140	4.22(+6)	1.94(+2)
XE-133	1.48(+8)	6.82(+2)
XE-133m	4.83(+6)	2.23(+1)
XE-134m	1.52(+6)	7.01(+0)
XE-135	2.84(+7)	1.31(+2)
XE-135m	3.13(+7)	1.44(+2)
XE-137	1.29(+8)	5.93(+2)
XE-138	1.13(+8)	5.21(+2)
XE-139	8.49(+7)	3.92(+2)
XE-140	5.47(+7)	2.52(+2)
XE-141	1.88(+7)	8.66(+1)
XE-142	7.54(+6)	3.48(+1)
XE-143	1.43(+6)	6.59(+0)

(1) Assumes all rods have burnup of 51,000 MWD/MTU, maximum for three 18 month cycles, 4.05 w/o enrichment 2700 Mwt. (This inventory bounds the inventory associated with 4.5 w/o at 1500 mWt for Fort Calhoun Station.)
(2) Assumes 10% of isotopes released to gap as per Regulatory Guide 1.77

Table 14.12-6 - "Noble Gas Isotopic Parameters"

Isotope	Half-Life	$\bar{E}\gamma$ [MeV/Disintegration]	$\bar{E}$ [MeV/Disintegration]
Kr-83m	1.86h	.002	.037
Kr-85m	4.48h	.159	.253
Kr-85	10.73y	.002	.251
Kr-87	76.31m	.793	1.324
Kr-88	2.80h	1.95	.375
Kr-89	3.16m	----	----
Kr-90	32.3s	----	----
Kr-91	9.0s	----	----
Kr-92	1.84s	----	----
Kr-93	1.27s	----	----
Kr-94	.21s	----	----
Xe-133m	2.23d	.0146	.190
Xe-133	5.29d	.0454	.135
Xe-134m	.29s	----	----
Xe-135m	15.3m	.432	.095
Xe-135	9.17h	.247	.316
Xe-137	3.84m	----	----
Xe-138	14.17m	1.183	.606
xe-139	39.7s	----	----
Xe-140	13.6s	----	----
Xe-141	1.72s	----	----
Xe-142	1.22s	----	----
Xe-143	.30s	----	----

Table 14.12-7 - "Noble Gases Dose Conversion Factors"

Radionuclide	Beta Skin DCF ($Rem\text{-}m^3/Ci\text{-}s$)	Whole Body Gamma DCF	β-DCF	γ-DCF
Kr-83m	0	$5.02x10^{-6}$	0	.000538
Kr-85m	$4.64x10^{-2}$	$3.72x10^{-2}$	4.79	3.99
Kr-85	$4.25x10^{-2}$	$5.25x10^{-4}$	4.38	.0563
Kr-87	$3.08x10^{-1}$	$1.87x10^{-1}$	31.8	20.0
Kr-88	$7.50x10^{-2}$	$4.64x10^{-1}$	7.74	49.7
Xe-133m	$3.14x10^{-2}$	$8.00x10^{-3}$	3.24	.857
Xe-133	$9.69x10^{-3}$	$9.33x10^{-3}$	1.00	1.00
Xe-135m	$2.25x10^{-2}$	$9.92x10^{-2}$	2.32	10.6
Xe-135	$5.89x10^{-2}$	$5.72x10^{-2}$	6.07	6.13
Xe-138	$1.31x10^{-1}$	$2.81x10^{-1}$	13.5	30.1

Table 14.12-8 - "Iodine Isotopic Parameters"

Isotope	Half-Life	Dose Conversion Factor [Rem-Thyroid/Ci]
I-128	25.0m	----
I-131	8.06d	$1.48x10^6$
I-132	2.28h	$5.35x10^4$
I-133	20.8h	$4.00x10^5$
I-134	52.6m	$2.50x10^4$
I-135	6.59h	$1.25x10^5$
I-136	85.0s	----
I-137	24.6s	----
I-138	6.5s	----
I-139	2.4s	----

Table 14.12-9 - "Dose Equivalent I-131 Thyroid Relative Dose Conversion Factors (RDCF's) "

Isotope	Relative DCF
I-131	1.0
I-132	.0362
I-133	.27
I-134	.017
I-135	.084

Table 14.12-10 - "Activity Released from the Steam Generator"

Nuclide	Activity (Curies)
DEC I-131	2.39 E+01
Kr-83m	1.86 E-02
Kr-85m	1.08 E-01
Kr-85	1.93 E+00
Kr-87	4.80 E-02
Kr-88	2.12 E-01
Xe-131m	1.61 E-01
Xe-133m	2.44 E-01
Xe-133	2.20 E+01
Xe-135m	4.83 E-03
Xe-135	3.63 E-01
Xe-138	1.54 E-02

14.13 CONTROL ELEMENT ASSEMBLY EJECTION ACCIDENT

14.13.1 General

The CEA ejection accident is defined as the mechanical failure in the form of a complete circumferential rupture of a CEDM housing or nozzle on the reactor vessel head resulting in the ejection of a control rod. The consequence of this mechanical failure is a rapid reactivity insertion which when combined with an adverse power distribution may result in localized fuel damage. The CEA ejection accident was reanalyzed for Cycle 19 (Ref. 14.13-10).

In design and fabrication, the CEDM is considered to be an extension of the reactor coolant system boundary; hence the probability of such a failure is equivalent to any other rupture of the reactor coolant system and is considered highly unlikely. Further, even if the CEA nozzle should separate from the reactor vessel head, its potential vertical upward travel is limited by the missile shield blocks placed over the reactor head and drive mechanisms. The missile shield block placement will allow an upward movement of only 18 inches; therefore, an additional failure in the drive train must be postulated for a continued CEA ejection. In addition, if the ejection continues, it will do so at a substantially lower rate.

In the following analysis, it is assumed that a CEA is ejected instantaneously from the core, although no mechanism for such an event has been identified. The analytical results presented in this section deal with the nuclear portion of the transient, which is terminated within 3 seconds.

The analysis was performed for hot zero power and hot full power initial conditions assuming the most adverse initial CEA configurations which are determined from the Technical Specification on power dependent insertion limits (PDIL). Additionally, the analysis was performed both at Beginning of Cycle (BOC) and End of Cycle (EOC). Dual CEA's are not considered, because the PDIL prohibits their insertion when critical. At zero power Groups 1 and 2 must be totally withdrawn and Group 3 at least 20% withdrawn. At full power all groups except Group 4 must be fully withdrawn, and the Group 4 insertion is limited to 75% withdrawn.

If the reactor is subcritical, Technical Specifications require all shutdown CEA's to be withdrawn before any regulating CEA's are withdrawn and all regulating CEA's to be inserted before any shutdown CEA's can be inserted. These specifications require that during shutdown dissolved boron concentration must be maintained such that all shutdown CEA's and Groups 1 and 2 regulating CEA's must be fully withdrawn and Group 3 regulating CEA's must be at least 20% withdrawn in order to achieve criticality. Ejection of any one dual CEA when the reactor is subcritical under the above conditions cannot result in criticality, because the worth of any one dual CEA is less than the combined worth of all shutdown and regulating CEA's.

Following the rapid ejection of a CEA, either from full power or zero power (critical) initial conditions, the core power rises rapidly for a brief period until the increasing reactivity loss due to the widening absorption resonances (Doppler effect) in U-238 terminates and reverses the increasing power transient. Increasing power will initiate a variable high power trip at 20% for the zero power case and a high power trip for the full power case, causing the CEA banks to insert which reduces the neutron power to negligible levels.

The loss of coolant resulting from the circumferential rupture of a CEDM housing or nozzle, and its consequences are within the scope of the small break loss of coolant accident which is discussed in Section 14.15.

14.13.2 Method of Analysis

The computer codes used in the analysis are TWINKLE and FACTRAN (Ref. 14.13-1 and 14.13-2). The calculation of the CEA ejection event is performed in two stages. First, an average core channel calculation is done using TWINKLE; and then, a hot spot analysis is done using FACTRAN.

The average core calculation is performed using spatial neutron kinetics to determine the average power generation with time, including the various core reactivity feedback effects, i.e., Doppler and moderator reactivity. The nuclear power increase during this transient will lead to elevated fuel pellet and fuel cladding temperatures. The TWINKLE code is utilized, in conjunction with Fort Calhoun Unit 1 plant specific physics data (Ref. 14.13-3 and 14.13-10), to perform a one-dimensional (axial) average core neutron kinetic analysis allowing for a more realistic representation of the spatial effects of axial moderator feedback and CEA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods of calculating the CEA worth and hot channel factor as discussed below.

The resulting average core nuclear power transient is input into FACTRAN along with the appropriate parameters such as fuel geometry, initial power, nominal average heat flux and core flow rate, initial and final hot spot total peaking factors, pellet power distribution, and gap heat transfer coefficients vs. time. Enthalpy and temperature transients in the hot spot are determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. During the transient, the steady state heat flux hot channel factor is linearly increased to the transient value in 0.05 second, the assumed time for full ejection of the CEA. Therefore, the assumption is made that the hot spots before and after ejection are at the same axial location. Prior to ejection, the power in this region will be depressed. Therefore, this is conservative since the peak power after ejection will occur in or adjacent to the assembly with the ejected CEA.

In the hot spot analysis, the transient temperature distribution in a cross-section of a metal-clad uranium dioxide fuel rod, and the heat flux at the surface of the rod, is calculated, 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. At hot full power conditions, the radial power distribution in the fuel pellet is represented by an inverted parabola which has been flattened so as to place the emphasis on the pellet centerline. This assumption causes the center of the pellet to heat up and maximizes the fraction of the fuel melt for these cases. At hot zero power conditions, the radial power distribution is an exaggerated parabolic shape placing the energy at the pellet perimeter. This causes an increase in the heat transfer to the cladding and maximizes the clad temperature increase effect.

The FACTRAN computer code uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation after DNB. Prior to DNB, the code automatically selects between the forced convection (Dittus-Boelter) and local boiling (Jens-Lottes) correlations based on the clad temperatures calculated by each. The Bishop-Sandberg-Tong correlation is conservatively used, assuming zero bulk fluid quality. The DNBR is not calculated; instead, for the full power cases, the code is forced into DNB 0.05 seconds after the start of the transient while in the zero power cases, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can 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.

The power distribution calculated without feedback is conservatively assumed to persist throughout the transient. FACTRAN calculates the percent of fuel melting and the hot spot clad and fuel temperatures. The fuel melting is assumed to be spread over a 5°F zone instead of taking place at a constant temperature. Changes in fuel rod geometry due to melting are not represented in the core except for fuel volume increase.

14.13.3 Results

The magnitude of fuel failure can be determined by examination of the following criteria:

1. The average fuel pellet enthalpy at the hot spot is below 200 cal/gm (360 Btu/lb) for irradiated fuel; the criterion for unirradiated fuel is 225 cal/gm. However, since the 200 cal/gm is more limiting, it is reflected as the enthalpy criterion here in the Fort Calhoun Unit 1 USAR.

2. Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is below the limits of the fuel pellet criteria discussed above. The calculated values for the fuel melt fraction are less than the limit value in accordance with 10 CFR 50 Appendix A.

3. The average clad temperature at the hot spot is below 3000°F. This criterion is not part of the licensing basis for Fort Calhoun Unit 1; however, Westinghouse internally applies this limit as a conservative value for the melting temperature of Zirconium. Reference 14.13-4 explains to the Nuclear Regulatory Commission that Westinghouse no longer considers peak clad average temperature to be a criterion for acceptance as part of a plant's licensing basis. However, since the validity of the FACTRAN clad temperature results above 3000°F may be questionable, this limit will be maintained as an internal Westinghouse acceptance criterion for CEA ejection. In addition to the 3000°F peak clad average temperature limit, Westinghouse applies a maximum Zirconium-water reaction limit of 16% at the hot spot.

Table 14.13-1 lists the significant input variables at full and zero power and at BOC and EOC (Ref. 14.13-3, 14.13-10). All the ejected CEA worths and radial peaking factors include appropriate allowances for calculation uncertainties. In all cases analyzed, a conservative value of 0.05 seconds was assumed for the total ejection time. For the full power and zero power case, a Variable Overpower trip is conservatively assumed to initiate at 112% and 30% (20% + 10% uncertainty) of full power, respectively. The initial conditions assumed the core was operating at 102% of full power for the full load case while 1.5 MWt was assumed for the zero power case.

Table 14.13-1 - "CEA Ejection Accident Assumptions"

Core Information	Condition Assumption (1)
Isothermal Temperature coefficients, $10^{-4}\Delta\rho/°F$	
BOC HZP	+ 0.50 (most pos.)
BOC HFP	+ 0.20 (most pos.)
EOC HZP	-1.30 (least neg.)
EOC HFP	-1.60 (least neg.)
Doppler-only power defect, $\%\Delta\rho$	(least neg.)
Beginning of Cycle	-0.800
End of Cycle	-0.800
Delayed neutron fraction, β	(min.)
Beginning of Cycle	0.0060
End of Cycle	0.0050
Trip Reactivity, $\%\Delta\rho$	(min.)
Hot Zero Power	-4.2
Hot Full Power	-1.5
Core Average, kw/ft	(max.) 6.3
Initial Fuel Average Temperature (inc. unc.), °F	(max.)
BOC HZP	NA
BOC HFP	2450
EOC HZP	NA
EOC HFP	2397
Ejected CEA Worth, $\%\Delta\rho$	(max.)
BOC HZP	0.690
BOC HFP	0.380
EOC HZP	0.690
EOC HFP	0.380
Peaking factor (Fq) before/after CEA ejection (max.)	
BOC HZP	NA/10.9
BOC HFP	2.66/ 5.70
EOC HZP	NA/10.9
EOC HFP	2.66/ 5.70

(1) Assumed parameter value as a minimum, maximum or NA.

The results of the full and zero power CEA ejection events for the reference cycle are compared to those of the most limiting cycles in Table 14.13-2. The reference cycle was assessed against the Regulatory Guide 1.77 criterion (Ref. 14.13-5) which limits the average hot pellet enthalpy to less than 280 cal/gram. The previous acceptance criteria of 200 cal/gram is more conservative with respect to the Regulatory Guide limit. The centerline melt criteria were not assessed in the reference cycle analysis, because the Regulatory Guide does not require it.

Table 14.13-2 - "CEA Ejection Accident Results"

Average fuel pellet enthalpy limit	200 (cal/gm)

The maximum average fuel pellet enthalpy for each of the cases reported is presented below.

Beginning of Cycle, Hot Zero Power	109.6 (cal/gm)
Beginning of Cycle, Hot Full Power	178.6 (cal/gm)
End of Cycle, Hot Zero Power	88.6 (cal/gm)
End of Cycle, Hot Full Power	159.1 (cal/gm)

Fuel melting limit	<10%

The maximum amount of fuel melting for each of the cases reported is presented below. Westinghouse applies an internal criterion that no more than 10% of the innermost portion of the hot spot may experience melting.

Beginning of Cycle, Hot Zero Power	0.0%
Beginning of Cycle, Hot Full Power	6.6%
End of Cycle, Hot Zero Power	0.0%
End of Cycle, Hot Full Power	1.2%

Peak Zirconium-Water reaction limit	16 wt. %

The weight percent of Zirconium reacting with water for each of the cases reported is presented below. Westinghouse internally applies this limit to ensure clad integrity.

Beginning of Cycle, Hot Zero Power	0.21 wt. %
Beginning of Cycle, Hot Full Power	1.18 wt. %
End of Cycle, Hot Zero Power	0.10 wt. %
End of Cycle, Hot Full Power	0.63 wt. %

14.13.4 Radiological Consequences of a CEA Ejection Accident

A conservative analysis of the potential radiological consequences of a CEA Ejection event has been performed in accordance with the guidelines presented in Regulatory Guide 1.77 (Ref. 14.13-5).

Two radioactivity release paths to the environment are assumed to contribute to the radiological consequences of a CEA ejection accident. The first is through containment leakage of fission products contained in the primary system. The second is through leakage from the primary system to the steam generators (primary-to-secondary leakage) and release to the environment via the secondary side relief valves.

The salient assumptions used to calculate the activity releases and offsite doses follow.

1. Prior to the accident, the primary coolant iodine and noble gas concentrations are assumed to equal the 1% fuel defect level, based on plant operation at 1500 MWt (Ref. 14.13-6).

2. Prior to the accident, the secondary coolant iodine concentration is assumed to equal the Technical Specification limit for full power operation -0.1 µCl/gram of dose equivalent I-131.

3. Ten percent of the core is assumed to fail as a result of DNB (Reference 14.13-7). This is assumed to result in the instantaneous release of 10% of the core gap activity to the primary coolant. The fraction of core activity contained in the gap (gap fraction) is assumed to be 10% for all nuclides. Thus, a total of 1% of the core activity is released. For the containment leakage release, 100% of this activity is assumed to be instantaneously released to the containment atmosphere. For the secondary system release, 100% of this activity is assumed to be contained in the reactor coolant. The core activity is presented in Reference 14.13-8.

4. One fourth of one percent (0.25%) of the core is assumed to melt. For the containment leakage release, 100% of the noble gases and 25% of the iodines are assumed to be instantaneously released to the containment atmosphere. For releases through the secondary system, 100% of the noble gases and 50% of the iodines are assumed to be released to the primary coolant (Ref. 14.13-5).

   The melted fuel fraction was determined as follows:

   a. A conservative upper limit of 50% of the rods experiencing clad damage are assumed to experience centerline melting (5% of the core) (Ref. 14.13-5).

   b. For rods experiencing centerline melting, 10% percent of the rod volume is assumed to actually melt (equivalent to 0.5% of the core) (Ref. 14.13-7).

   c. A conservative maximum of 50% of the axial length of the rod is assumed to experience melting due to the power distribution (0.5 of 0.5% of the core is equal to 0.25% of the core) (Ref. 14.13-5).

5. The total primary-to-secondary leak rate is assumed to be at the Technical Specification limit of 1 gpm (Ref. 14.13-5).

6. Activity released to the environment via the primary to secondary leakage pathway is assumed to be released directly to the environment without mixing with the secondary coolant. An iodine decontamination factor of 10 is applied to this activity release.

7. Offsite power is lost at the initiation of the event.

8. Steam release to the environment: 0 to 50 seconds - 9354 lbm

   This steam release is used with an iodine partition coefficient of 0.1 to determine the release of the initial secondary coolant iodine activity (Item 2).

9. Containment leakage rate (volume percent/day): 0 to 24 hours-0.1
   1 to 30 days-0.05

10. Atmospheric dispersion factors (sec./cu. meter) (Ref. 14.13-9).

site boundary (0 to 2 hour)	2.55 E-4
low population zone (0 to 30 days)	4.53 E-6

11. Breathing rates (cu. meter/sec.): 0-8 hr, 3.47 E-4
    (Ref. 14.13-5) 8 - 24 hr, 1.75 E-4
    > 24 hr, 2.32 E-4

12. Thyroid dose conversion factors (rem/curie): ICRP Publication 2

Results

The activity released to the environment from the secondary system is presented in Table 14.13-3 (Ref. 14-13-6).

Table 14.13-3 - "Activity Released from the Secondary System"

Nuclide	Activity (Curies) (0 - 50 sec.)
Kr-85m	1.5 E 0
Kr-85	7.5 E-2
Kr-87	2.8 E 0
Kr-88	3.9 E 0
Xe-131m	6.6 E-2
Xe-133	1.1 E 1
Xe-135m	2.2 E 0
Xe-135	2.8 E 0
Xe-138	9.2 E 0
I-131	8.9 E-1
I-132	6.8 E-1
I-133	9.6 E-1
I-134	1.1 E 0
I-135	9.0 E-1

The activity released to the environment from the containment is presented in Table 14.13-4 (Ref. 14.13-6).

Table 14.13-4 - "Activity Released from the Containment"

Nuclide	Activity (Curies) 0-2 hours	0-30 days
Kr-85m	2.3 E 1	8.3 E 1
Kr-85	1.4 E 0	2.5 E 2
Kr-87	3.1 E 1	4.6 E 1
Kr-88	5.6 E 1	1.4 E 2
Xe-131m	1.2 E 0	1.1 E 2
Xe-133	2.0 E 2	1.0 E 4
Xe-135m	7.6 E 0	7.6 E 0
Xe-135	4.8 E 1	3.1 E 2
Xe-138	3.4 E 1	3.5 E 1
I-131	8.5 E 1	6.0 E 3
I-132	9.3 E 1	2.1 E 2
I-133	1.7 E 2	2.1 E 3
I-134	9.6 E 1	1.2 E 2
I-135	1.5 E 2	7.6 E 2

The resulting doses at the exclusion area boundary (EAB) and at the outer boundary of the low population zone (LPZ) are presented below.

Table 14.13-5 - "Resulting Doses"

	Dose (rem)	
Containment release	Thyroid	Whole body gamma
0-2 hour dose at EAB	19.5	1.4 E-2
0-30 day dose at LPZ	10.0	9.0 E-4
Secondary System release		
0-2 hour dose at EAB	1.7 E-1	1.5 E-3
0-30 day dose at LPZ	3.0 E-3	2.7 E-5
Total offsite dose		
0-2 hour dose at EAB	19.7	1.6 E-2
0-30 day dose at LPZ	10.0	9.3 E-4

14.13.5 Conclusions

The analyses of the CEA ejection event demonstrate only a small fraction of fuel melting and no clad damage will occur following a CEA ejection from full or zero power at beginning or end of cycle.

The dose acceptance criteria is based on the recommendations of Standard Review Plan (NUREG-0800) Section 15.4.8, Appendix A, i.e., 75 rem thyroid and 6 rem whole-body. The calculated doses for the CEA ejection event are within the acceptance criteria. Specific results of radiological consequences are presented in Table 14-13-5.

14.13.6 Section 14.13 References

14.13-1 Risher, D.H., Jr. and Barry, R.F., "TWINKLE-A Multi-Dimensional Kinetics Computer Code, "WCAP-7979-P-A, January 1975 and WCAP-8028-A, January 1975.

14.13-2 Hunin, C., "FACTRAN Code Description, "WCAP-7337, Rev. 1-P, December 1989.

14.13-3 "Westinghouse Mixed Vendor Core Data List (MVCDL) for the Fort Calhoun Unit 1." [EA-FC-90-004, Rev. 0].

14.13-4 Letter NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," letter from W. J. Johnson, W Nuclear Safety, to Robert C. Jones, NRC Reactor Systems Branch, dated October 23, 1989.

14.13-5 Regulatory Guide 1.77, "Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," U. S. Nuclear Regulatory Commission, May, 1974.

14.13-6 EA-FC-91-001, Rev. 0 "1% Failed Fuel."

14.13-7 Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Spacial Kinetics Methods," WCAP-7588, Revision 1-A, January, 1975.

14.13-8 EA-FC-90-004, Revision 0. "MVCDL"

14.13-9 OPPD Calculation PED-FC-91-1357, "Atmospheric Dispersion: USAR Calculations."

14.13-10 EA-FC-98-054, Rev. 0, "Cycle 19 CEA Ejection Confirmation."

14.15 LOSS-OF-COOLANT ACCIDENT

14.15.1 General

A loss-of-coolant accident (LOCA) is defined as a breach of the reactor coolant system boundary which results in interruption of the normal mechanism for removing heat from the reactor core. Emergency core cooling is provided to prevent clad and fuel melting which could occur as a result of decay heat and possible chemical reactions. The Emergency Core Cooling System (ECCS) provides adequate protection for the core in the unlikely event of a LOCA.

The safety injection system, which provides the emergency core cooling, consists of three high-pressure pumps, two low-pressure pumps, and four safety injection tanks. Although the three charging pumps would normally operate, no analysis credit is taken for their operation. Emergency operation of the safety injection pumps is initiated either by a low-low pressurizer pressure signal or by a high containment building pressure signal. Water is also delivered to the reactor coolant system from the safety injection tanks when the cold leg pressure drops below the driving head which consists of nitrogen gas (minimum gas pressure = 240 psig) within the safety injection tanks plus an elevation head. Thus, the tanks operate as a passive system requiring no manual or automatic action for their operation.

The injection water for the high- and low-pressure injection systems is supplied from the borated safety injection and refueling water (SIRW) tank. This analysis assumes a minimum usable SIRW Tank inventory of 250,000 gallons (Ref. 14.15-45). When the SIRW tank is nearly empty, water is recirculated from the containment sump, as described in Section 6.2.

The ECCS is designed such that its calculated cooling performance following a postulated LOCA conforms to the criteria specified in 10 CFR Part 50.46. The models used for the evaluation of ECCS performance during the various postulated LOCA's include the required and acceptable features specified in Appendix K to 10 CFR Part 50.46, and ECCS performance has been calculated for a number of postulated LOCA's of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated LOCA's is covered.

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break, assuming minimum availability of the safety injection system corresponding to the following assumptions. The entire contents of all four safety injection tanks are available for emergency core cooling, but the contents of one of the tanks are assumed to be lost through the break in the reactor coolant system. In addition, of the three high-pressure safety injection pumps (HPSI's) and the two low-pressure safety injection pumps (LPSI's), it is assumed that one high-pressure and one low-pressure pump operate for the large break analysis and one of each type is assumed to operate in the small break analysis. No credit for charging pump operation is taken in either the large or small break LOCA analyses. The maximum SI flow condition assumes that all required SI equipment is operational and available for use. For the large break LOCA, it is assumed that 25% of the combined HPSI-LPSI discharge rate and the flow from one safety injection tank is lost through the break in the reactor coolant system. At 30 minutes into a large break LOCA there is sufficient HPSI flow to remove decay heat and keep the core covered with 35% spillage. For the small break LOCA, 25% of the HPSI flow, 50% of the LPSI flow, and the flow from one safety injection tank is assumed to be lost through the break.

A complete large break LOCA analysis for operation of the Fort Calhoun Station at 1500 MWT was performed for Cycle 14 by Westinghouse. A complete small break LOCA analysis was performed for Cycle 14. The limiting break was based on a fuel pellet burnup of 60,000 MWD/MTU and integrated radial peaking factor 1.80. Steam Generators are assumed to have 6% plugging in each generator.

For a postulated LOCA, a reactor trip is initiated when the pressurizer pressure-low setpoint is reached. A Safety Injection Actuation Signal (SIAS) is actuated by either a containment pressure high or a pressurizer pressure low-low signal. The consequences of the accident are limited in two ways:

1. Reactor trip and safety injection (of borated water) supplement void formation in causing a rapid reduction of the nuclear power to a residual level corresponding to the delayed fission product decay. For a postulated Large Break LOCA, credit for CEA insertion to keep the reactor subcritical post-LOCA is not taken since large break forces may degrade the trip function of the CEAs.

The requirement for post-LOCA subcriticality is met by maintaining a sufficiently high boron concentration in the Safety Injection and Refueling Water (SIRW) Tank.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive fuel temperatures.

Before the reactor trip occurs, the reactor is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat and heat from the vessel and internals is transferred to the RCS fluid, and then to the containment and the secondary system dependant upon the break size.

The reactor coolant pumps are tripped at the initiation of the accident due to an assumed loss of offsite power or remain running until operator actions trip the pumps in accordance with the trip 2/leave 2 operating criteria (Ref. 14.15-37). The effects of the RCP coastdown are included in the blowdown analyses. Without a loss of offsite power, a Steam Generator Isolation Signal (SGIS), which occurs as a result of containment pressure-high, terminates normal feedwater flow by closing the main feedwater isolation valves. With the assumed loss of offsite power, main feedwater is lost with the coastdown of the electric motor driven main feedwater pumps. If offsite power is available, the steam is dumped to the condenser, although not credited in the analysis; if offsite power is not available, the steam is assumed to be dumped to the atmosphere via the main steam safety valves. Makeup to the steam generators is initiated by the auxiliary feedwater pumps if steam generator level falls below the auxiliary feedwater system actuation setpoint. If not terminated previously by High Pressure Safety Injection (HPSI) pump flow, the core uncovery transient is turned around when the RCS pressure falls below approximately 225 psia (analysis value, Reference 14.15-51) and the Safety Injection Tanks inject borated water.

14.15.2 Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand the thermal effects caused by a loss-of-coolant accident including the double-ended severance of the largest Reactor Coolant System pipe. The reactor core and internals together with the Emergency Core Cooling System are designed so that the essential heat transfer geometry of the core is preserved following the accident. The Emergency Core Cooling System, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46[1], "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors".

These requirements are:

1. Peak clad temperatures (PCT) do not exceed 2200°F.

2. The amount of cladding that chemically reacts with the coolant does not exceed 1% of the zircalloy cladding surrounding the fuel, excluding the cladding surrounding the plenum volume.

3. Oxidation of the cladding does not exceed 17% of the original cladding thickness, which precludes embrittlement problems.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After initial operation of the ECCS, core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived radioactivity remaining in the core.

14.15.3 Thermal Analysis Description

The analysis specified by 10 CFR 50.46 is presented in Sections 14.15.4 and 14.15.5 for large and small breaks, respectively. The results of the large and small break loss-of-coolant accident analyses which are summarized in Tables 14.15-3 and 14.15-7 show compliance with the above Acceptance Criteria. The highest PCT calculated was for a double-ended cold leg guillotine (DECLG) break with a Moody discharge coefficient (C_D) of 0.4.

The large break analysis is based on the initial reactor conditions shown in Table 14.15-1. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are described in the topical report, "Westinghouse ECCS Evaluation Model-Summary".[2,3] The description of the various aspects of the LOCA analysis methodology is provided in References 14.15-3,4,5 and 6.

These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria.

The method of analysis to determine peak clad temperature is divided into two types of analysis: 1) large break LOCA, and 2) small break LOCA. The methods of analysis for the large and small break LOCAs are described below and results are given.

14.15.4 Large Break LOCA Analysis

14.15.4.1 Description of Analysis and Assumptions

Should a major break occur, depressurization of the Reactor Coolant System results in a decrease in pressurizer pressure. A reactor trip signal occurs when the pressurizer pressure-low trip setpoint is reached. The Safety Injection System is actuated when either the pressurizer pressure low-low setpoint or the containment pressure-high setpoint is reached. At the beginning of the blowdown phase, the entire Reactor Coolant System contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with the requirements of Appendix K of 10 CFR 50 (Ref. 14.15-1). Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

When the Reactor Coolant System pressure falls below approximately 225 psia (the analysis setpoint which includes and bounds the uncertainties of Reference 14.15-52), the Safety Injection Tanks begin to inject borated water. A conservative assumption is made that some injected water bypasses the core and goes out through the break until the termination of bypass. This conservatism is consistent with the requirements of Appendix K of 10CFR50 (Ref. 14.15-1). The termination of bypass is defined as the commencement of a continuous flow of liquid down the downcomer into the lower plenum.

Sensitivity studies (Ref. 14.15-7) for large break LOCA were performed which covered a break spectrum utilizing a double-ended cold leg guillotine (DECLG) break with various values of discharge coefficient (C_D), a double-ended hot leg guillotine (DEHLG) break, a double-ended pump suction guillotine (DEPSG) break, and a range of split-type break sizes ranging from a 1.0 ft^2 area to a full double-ended area of the cold leg. This study determined that, for Westinghouse plant designs, the DECLG type break was both the most limiting type and location. Furthermore, Combustion Engineering's Analysis of the Fort Calhoun plant has shown that the DECLG type break was both the most limiting type and location for that unit. Therefore, the spectrum of break was limited to only DECLG types for Fort Calhoun Station. The initial large break spectra was analyzed with a core which contains burnable shims. It was performed at 102 percent of 1500 MW_t (the core licensed power), with a vessel flow rate of 196,000 gpm, a 6% steam generator plugging level in each generator, and a 30.9 second delay in delivery of pumped SI flow with loss of offsite power.

A range of axial core power distribution was studied, as required for LOCA analysis (Appendix K of 10 CFR 50 (Ref. 14.15-1)), and the distribution resulting in the most severe calculated consequences selected for analysis. The power distributions considered hot spot elevations from mid-core to 8.75 ft, and an Axial Shape Index (ASI) range of 0.0 to -0.16 asiu. The axial core power distribution in Figure 14.15-2 was found to be limiting. This power distribution, determined by Westinghouse methods, was found to be similar to that used in the most recent LOCA analysis performed by Combustion Engineering.

To determine the limiting plant conditions, Loss of Offsite Power, and No Loss of Offsite Power for both the minimum and maximum SI flow conditions with modeling of pumped SI/SI Tank flow interactions were analyzed. No Loss of Offsite Power with Minimum Safeguards was found to produce the most severe consequences.

The large break LOCA analysis presented here was performed with the 1981 + BART for CE NSSS Evaluation Model (EM) (Ref. 14.15-6). Important features of this model are described in other Westinghouse Evaluation Models (References 14.15-8, 9, 10 and 11). The 1981 + BART for CE NSSS Evaluation Model incorporates the core heat transfer portions of the BART computer code to obtain a mechanistic core heat transfer model. The 1981 + BART was approved by the NRC for use in PWRS with Westinghouse fuel (Ref. 14.15-8).

Table 14.15-1 - "Fort Calhoun Large Break LOCA Analysis Input Parameters and Assumptions"

NSSS Power - 102% of 1500 MWt	1530 MWt
Peak Linear Heat Rate - at 102% of 1500 MWt	15.5 KW/ft
Radial Peaking Factor (F_R^T)	1.80
Maximum Allowable Peaking Factor (F_q)	2.545
Axial Power Distribution	See Figure 14.15-2
Reactor Coolant System Pressure	2100 psia
Reactor Coolant System Flow Rate	196,000 gpm
Reactor Inlet Temperature	545°F
Reactor Trip Signal (Including uncertainties)	1728 psia, Pressurizer Pressure LOW
SI Signal (Including uncertainties)	1578 psia, Pressurizer Pressure LOW-LOW
Safety Injection Tank Water Volume	825 ft^3/Tank
Safety Injection Tank Minimum Pressure	225 psia
Steam Generator Tube Plugging Level	6% (Uniform)

14.15.4.2 Method of Large Break Analysis

The large break LOCA transient can be conveniently divided into three periods: blowdown, refill, and reflood. Also, three physical parts of the transient are analyzed for each period: the thermal-hydraulic transient in the reactor coolant system, the containment pressure and temperature, and the fuel and cladding temperatures of the hottest rod. These considerations lead to the use of a system of computer codes designed to model the LOCA transient. A detailed description of the various aspects of the LOCA analysis is given in WCAP-8339 (Ref. 14.15-2).

The SATAN-VI (Ref. 14.15-12) code evaluates the thermal-hydraulic transient during blowdown. The WREFLOOD (References 14.15-6 and 13) code, using output from the SATAN-VI code, computes the time to bottom of core recovery (BCR), RCS conditions at BCR and mass and energy release from the break during the reflood phase of the LOCA. The COCO (Ref. 14.15-14) code is used to model the containment pressure transient. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. The BCR conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the LOCBART code.

In the 1981 + BART for CE NSSS model, the cladding heat-up transient is calculated by LOCBART, which is a combination of the LOCTA (References 14.15-5 and 15) code with the BART code (Reference 14.15-8). In LOCBART, the empirical FLECHT correlation has been replaced by the BART methodology. BART methodology employs mechanistic models to generate heat transfer coefficients appropriate to the actual flow, and heat transfer regimes experienced by the fuel rods. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. A more detailed description of the LOCBART code can be found in References 14.15-8 and 14.15-16.

The calculation of peak cladding temperature is performed by modeling the hottest fuel assembly (from the reactor core) in LOCBART. The Improved Fuel Performance model (Reference 14.15-17) is used to generate the initial input parameters for fuel rod conditions for LOCBART. The hottest fuel assembly is subdivided into three regions:

1. The hottest rod
2. Adjacent rod to the hottest rod
3. The average fuel channel in the hot assembly.

Figure 14.15-1 shows the interaction of the 1981 + BART for CE NSSS model and the relationship of the computer codes to the LOCA sequence of events. In this analysis an additional node was included in the SATAN lower plenum model to accurately describe the Fort Calhoun configuration and WREFLOOD was expanded to accurately model both cold legs in the broken loop.

14.15.4.3 Results of Large Break Analysis

A break spectrum sensitivity analysis with a range of Moody discharge coefficients (C_D = 0.4, 0.6 and 0.8) was run assuming a loss of offsite power, minimum SI flow, F_R^T = 1.75 with non-Integral Fuel Burnable Absorber (IFBA) fuel rods. The time sequence of events for this break spectrum is given in Table 14.15-2 and the associated peak cladding temperatures (PCT) and hot spot metal reactions are summarized in Table 14.15-3. Based upon the results of Table 14.15-3 an additional spectra of analyses in which F_R^T = 1.80 (in order to bound Cycle 14 and future cycle operation) and a matrix of minimum and maximum SI flows, with and without the loss of offsite power, were analyzed. The limiting (CD = 0.4) break size remains valid for the increased F_R^T value since the increase in PCT would be consistent for all the break sizes previously analyzed.

Table 14.15-2 - "Large Break Sequence of Events Fort Calhoun Large Break LOCA Analysis"

RESULTS	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.4	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.6	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.8
Start	0.0	0.0	0.0
Rx Trip Signal	0.60	0.59	0.59
S.I. Actuation Signal	0.97	0.77	0.67
S.I. Tank Injection	22.80	16.80	14.00
Pump Injection Begins	31.87	31.67	31.57
End of Bypass	28.92	20.59	17.48
End of Blowdown	28.92	20.59	17.48
Bottom of Core Recovery	39.34	31.73	28.52
S.I. Tanks Empty	94.92	90.14	88.01

NOTE: All times are in seconds.

Table 14.15-3 - "Break Spectrum Sensitivity Analysis Results
Fort Calhoun Large Break LOCA Analysis"

RESULTS	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.4	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.6	MIN SI FLOW $F^T_R = 1.75$ DECLG C_D=0.8
Peak Clad Temperature (°F)	1981.	1869.	1815.
Peak Clad Temp. Elevation (Ft.)	9.25	9.25	9.25
Peak Clad Temperature Time (Sec.)	113.9	98.3	86.8
Max Local Zr/H_2O Reaction (%)	2.98	2.88	2.38
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Hot Assy. Burst Time (Sec.)	47.4	69.5	61.1
Hot Assy. Burst Elevation (Ft.)	8.75	9.00	8.75
Blockage on Hot Rod (%)	41.0	35.2	38.8

Table 14.15-4 - "Large Break LOCA Results - Fort Calhoun Large Break LOCA Analysis"

RESULTS	MIN SI FLOW, F^T_R = 1.80 LOSS OF OFF-SITE POWER, DECLG C_D=0.4	MIN SI FLOW, F^T_R = 1.80 NO LOSS OF OFF-SITE POWER, DECLG C_D=0.4	MAX SI FLOW, F^T_R = 1.80 LOSS OF OFF-SITE POWER, DECLG C_D=0.4
Peak Clad Temperature (°F)	2006.	2066.	2032.
Peak Clad Temp. Elevation (Ft.)	9.50	9.25	9.50
Peak Clad Temperature Time (Sec.)	118.5	94.4	117.1
Max Local Zr/H_2O Reaction (%)	3.38	5.77	3.66
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Hot Assy. Burst Time (Sec.)	51.2	51.1	51.2
Hot Assy. Burst Elevation (Ft.)	8.75	8.75	8.75
Blockage on Hot Assembly(%)	39.0	37.6	38.8

The core axial power shape sensitivity was also examined, in the break spectrum sensitivity analysis, with a set of shapes covering a range of ASIs from 0.0 to -0.16 asiu. The hot spot elevations correspondingly ranged from 5.33 ft to 8.75 ft. The limiting shape was found to occur at the ASI limit of -.16 asiu, as shown in Figure 14.15-2.

The second set of analyses also included the case of no loss of offsite power to establish the limiting plant conditions. Both Minimum and Maximum SI conditions were evaluated with the reactor coolant pumps running and as noted above, the Minimum SI case found to be limiting with a PCT of 2066°F, as indicated in Table 14.15-4. The limiting discharge coefficient C_D = 0.4 and the limiting power shape were included in both cases. For the Maximum safeguards (Maximum SI) calculations, no loss of Safety Injection was assumed and nominal SI pump flows were used. Maximum SI calculations were then performed for the case of loss of offsite power. The maximum PCT for the cases of Maximum SI analyzed was 2032°F, as indicated in Table 14.15-4. Thus, the plant is Minimum SI, no loss of offsite power limited for the large break LOCA.

A study was performed on the IFBA fuel rods to establish the limiting fuel type. As typically observed with W fuel designs, the non-IFBA fuel is limiting, with the IFBA fuel providing a PCT benefit of not less than 15°F over non-IFBA fuel for the corresponding conditions. An evaluation was also performed by OPPD comparing Westinghouse and CE fuel rod temperatures over the appropriate range of power and burnup. It was determined that Westinghouse fuel was limiting due to higher fuel rod temperatures, i.e. higher stored energy.

Figures 14.15-3 through 14.15-18 present the limiting (C_D = 0.4) LOCA case (PCT = 2066°) of Minimum SI flow, no loss of offsite power and F_R^T = 1.80. Figures 14.15-19 through 14.15-32 provide the principal parameters for the initial break spectra (C_D = 0.4), using Maximum SI loss of offsite power, and 1.75 as input assumptions[1]. The following parameters are provided for the limiting break case:

1. Core Power Transient
2. Core Pressure Transient
3. Break Flow Rate
4. Containment Pressure

5. Safety Injection Tank Flow (Blowdown)
6. Pumped ECCS Flow (Reflood)
7. Reflood Core and Downcomer Levels
8. Core Reflood Rate
9. Core Mass Velocity
10. Core Flow (Top and Bottom)
11. Core Fluid Quality
12. Core Heat Transfer
13. Core Fluid Temperature
14. Peak Cladding Temperature

In addition, Break Energy Release Blowdown (Figure 14.15-6), and Containment Wall Heat Transfer Coefficient (Figure 14.15-7), are provided for the limiting case.

14.15.5 Small Break LOCA Analysis

14.15.5.1 Description of Analysis and Assumptions

The Fort Calhoun small break LOCA analysis was performed using the Westinghouse ECCS Small Break Evaluation model (Ref. 14.15) which utilizes the NOTRUMP (References 14.15-19, 20, 21, 22 and 6) and LOCTA-IV (Reference 14.15-6, 15 and 22) computer codes. The small break was analyzed with a core which contained 424 burnable shims. Figure 14.15-33 shows the interaction of the computer codes used to evaluate the small break cases.

The core power level utilized in the small break LOCA analysis was 102% of 1500 MWT, the licensed power. The peak linear heat rate and peaking factor used in the analyses are also given in Table 14.15-5. Additional assumptions for the analysis of Fort Calhoun Station were: operation at a total primary system flow rate of 196,000 gpm, a 6% steam generator tube plugging in each of the two steam generators, and a 30.9 second delay in delivery of pumped ECCS flow assuming Loss of Offsite Power coincident with Reactor Trip. Eight of the ten Main Steam Safety valves were assumed to be operable. They were assumed to be set at 3% above the Technical Specification setpoint value and require an additional 3% accumulation before being assumed fully open.

Figure 14.15-34 presents the axial power shape utilized to perform the small break analysis. This axial power shape was chosen because it provides distribution of power versus core height which will maximize PCT using -.16 asiu for the Technical Specification Axial Shape Index to provide the most top peaked power distribution.

Table 14.15-5 - "Input Parameters and Assumptions Fort Calhoun Small Break LOCA Analysis"

NSSS Power - 102% of 1500 MWt	=1530 MWt
Peak Linear Heat Rate - at 102% of 1500 MWt	=15.5 KW/ft
Radial Peaking Factor (F_R^T)	=1.80
Maximum Allowable Peaking Factor (F_q)	=2.545
Axial Power Distribution	See Figure 14.15-34
Reactor Coolant System Pressure	=2100 psia
Reactor Coolant System Flow Rate	=196,000 gpm
Reactor Inlet Temperature	=545°F
Reactor Trip Signal (including uncertainties)	=1728 psia, Pressurizer Pressure LOW
SI Signal (including uncertainties)	=1578 psia, Pressurizer Pressurizer LOW-LOW
Safety Injection Tank Water Volume	=825 ft^3/Tank
Safety Injection Tank Minimum Pressure	=225 psia*
Steam Generator Tube plugging Level	=6% (Uniform)
MSSV Setpoint Uncertainties	=$\pm$ 3% Nominal Setpoint Pressure
	=$\pm$ 3% Valve Accumulation Pressure

A spectrum of cold leg break sizes (.022, .049 and .087 ft^2), was analyzed in order to determine the most limiting break size. These breaks were analyzed following the method presented in Section 14.15-2. Additionally, the no loss of offsite power with the trip 2/leave 2 RCP strategy case was studied and found to be non-limiting.

*Actual analysis used 255 psia, however 225 psia is justified by Reference 14.15-51.

14.15.5.2 Method of Small Break Analysis

The NOTRUMP (References 14.15-6, 19, 20, 21 and 22) and LOCTA-IV (References 14.15-6, 15 and 22) computer codes are used to perform the analysis of Loss-Of-Coolant Accidents for small breaks in the Reactor Coolant System (RCS). The NOTRUMP computer code, approved for use by the Nuclear Regulatory Commission (NRC), is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the flow through the reactor core and through the break. This code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these features are the utilization of nonequilibrium thermal calculation in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stack fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants".

In NOTRUMP, the RCS is subdivided into fluid filled control volumes (fluid nodes) and metal nodes interconnected by flowpaths and heat transfer links. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied to these nodes. Both the broken loop and the intact loop are modeled explicitly with two cold legs and two reactor coolant pumps in each providing the option for analysis of the "trip 2/leave 2" RCP strategy. A detailed description of the NOTRUMP code is provided in References 14.15-6, 20, 21 and 22.

In the NOTRUMP model (Ref. 14.15-18), the reactor core is represented as a vertical stack of heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables the explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

Clad thermal analysis is performed with the LOCTA-IV (Ref. 14.15-15) computer code which uses as input the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history form the NOTRUMP hydraulic calculations as input. For all computations, the NOTRUMP and LOCTA-IV calculations were terminated when the core mixture level returned to the top of the core.

A schematic representation of the computer code interfaces is given in Figure 14.15-33.

14.15.5.3 Results of Small Break Analysis

The small break LOCA analyses were performed with the assumptions contained in Table 14.15-5. The time sequence of events and results of key parameters for the .022, .049 and .087 ft^2 breaks analyzed are shown in Tables 14.15-6 and 14.15-7, respectively. The .049 ft^2 break is shown to be the limiting break. This is the largest break terminated by the HPSI pumps; with each Safety Injection Tank injecting less than 2 ft^3 of SI flow. The peak clad temperature for the .049 ft^3 was 1444°F, occurring for both the IFBA and non-IFBA fuel rods at the beginning of cycle, 0 MWD/MTU. The maximum local zirconium oxidation was 0.40% which is less than the 1% core average criteria as required by 10 CFR 50.46. These results indicate that a coolable geometry would be maintained for small break LOCAs. Figures 14.15-35 through 14.15-40 show RCS pressure, core mixture height, peak clad temperature, steam flow rate, rod film coefficient, and hot spot fluid temperature versus time for the limiting break size (0.049 ft^2).

Table 14.15-6 - "Small Break Sequence of Events Fort Calhoun Small Break LOCA Analysis"

	Cold Leg Break .022 SQ FT	Cold Leg Break .049 SQ FT	Cold Leg Break .087 SQ FT
Start	0.0 Sec	0.0 Sec	0.0 Sec
Reactor Trip Signal	23.0 Sec	10.6 Sec	7.2 Sec
SI Actuation Signal	36.6 Sec	17.2 Sec	10.5 Sec
Pumped SI begins	67.5 Sec	48.1 Sec	41.4 Sec
Top of Core Uncovered	2178.5 Sec	1095.1 Sec	710.7 Sec
S.I. Tank Injection Begins	NA	NA	932.9 Sec
PCT Occurs	3075.8 Sec	1898.0 Sec	1022.1 Sec
Top of Core Recovered	4713.8 Sec	3100.3 Sec	1368.9 Sec

Table 14.15-7 - "Small Break LOCA Results Fort Calhoun Small Break LOCA Analysis"

RESULTS	BOC IFBA Cold Leg Break .022 SQ FT	BOC IFBA Cold Leg Break .049 SQ FT	BOC IFBA Cold Leg Break .087 SQ FT
Peak Clad Temperature (°F)	1076.	1444.	1166.
Peak Clad Temperature Elevation (Ft.)	10.25	10.25	10.00
Peak Clad Temperature Time (Sec.)	3075.8	1898.0	1022.1
Max Local Zr/H_2O Reaction (%)	0.05	0.40	0.03
Max Local Zr/H_2O Rx Elev. (Ft.)	10.25	10.25	10.00
Total Zr/H_2O Reaction (%)	<1.00	<1.00	<1.00
Hot Rod Burst Time (Sec.)	NO BURST	NO BURST	NO BURST
Hot Rod Burst Elevation (Ft.)	NA	NA	NA

14.15.6 Long Term Cooling Considerations (ECCS)

General

An evaluation of the post-LOCA Long Term Cooling ECCS performance by Westinghouse of Fort Calhoun station has demonstrated conformance with criterion (5) of 10 CFR Part 50.46(b). Procedures have been defined for utilizing the ECCS to remove decay heat and thereby maintain core temperatures at acceptable low values for an indefinite period of time.

Long term cooling is initiated when the core is reflooded after a LOCA and is continued until the plant is secured. The objective of long term cooling is to maintain the core temperature at an acceptably low value while removing decay heat for the extended period of time required by the long-lived radioactive isotopes remaining in the core. In satisfying this objective, the post-LOCA long term cooling requirements (as contained in the Emergency Operating Procedures) make provisions for maintaining core cooling and boric acid flushing by simultaneous hot and cold leg injection, or for initiating cooldown of the Reactor Coolant System (RCS) if the break is sufficiently small with natural circulation present, such that success of such operation is assured.

Within 8.5 hours of the start of the LOCA and if shutdown cooling has not been established, then simultaneous hot and cold leg injection is established in accordance with the Emergency Operating Procedures (EOPs). The EOPs are based on the NRC approved CE Emergency Procedure Guidelines and no distinction/classification between large and small breaks is made. The HPSI pumps discharge lines are realigned so that the total injection flow is split between the hot and cold legs. The hot side injection is achieved by injection in the RCS through the pressurizer auxiliary spray system. Boron precipitation/accumulation will not occur as long as all of the following conditions are satisfied:

1. Maximum Safety Injection Tank and Safety Injection Refueling Water Tank boron concentration does not exceed 2400 ppm.

2. Simultaneous Hot and Cold Leg Injection is initiated within 8.5 hours of the start of the LOCA if shutdown cooling cannot be established.

3. Both Hot and Cold Leg injection rates must be at or above 140 gpm (Ref. 14.15-23).

In the event that the auxiliary spray line becomes inoperable, the hot side injection is achieved by realigning the LPSI pump discharge to the shutdown cooling suction line in conjunction with the opening of a PORV. With the PORV open, the depressurization is sufficient to create a flushing flow either by allowing the RCS to refill or by establishing adequate hot side LPSI flow. Sufficient injection flow is provided to both cool the core and flush the reactor vessel for an indefinite period of time. This injection mode provides cooling for the RCS and prevents boric acid precipitation/accumulation in the vessel following a LOCA.

For the small break LOCA with the steam generator(s) available to remove decay heat, the cooldown of the RCS and long term decay heat removal is provided by natural or forced circulation cooling. For the small break LOCA, without heat removal through a steam generator, cooldown of the RCS and long term decay heat removal is provided by once-through-cooling, i.e., opening of the pressurizer power operated relief valves (PORVs) to release steam from the RCS. This action depressurizes and maintains the RCS pressure below the HPSI shutoff head, allowing the HPSI pumps to flush the core and accelerate refilling of the RCS.

14.15.6.1 Large Break LOCA Results

For a large break LOCA, boric acid precipitation/ accumulation is minimized by the core flushing flow which is provided by the simultaneous hot side and cold side injection from the HPSI pumps. The simultaneous hot side and cold side injection mode is initiated within 8.5 hours post-LOCA if shutdown cooling cannot be established in accordance with the EOPs.

Large break long term cooling without establishment of shutdown cooling is achieved for those break sizes for which simultaneous hot and cold side injection can both flush and cool the core. A break size of .005 ft^2 is the smallest for which the large depressurization maintains the RCS pressure sufficiently low to allow a HPSI pump to flush and cool the RCS. This process is also successful for small breaks as large as 0.015 ft^2.

The above description considered only the condition where off-site power is unavailable. With offsite power available, it is possible to more quickly cool down the RCS using the turbine bypass system and thereby initiate operation of the shutdown cooling system. However, opening of the PORVs with HPSI flow is sufficient to maintain decay heat removal for an indefinite period of time such that it is not necessary to initiate operation of the shutdown cooling system to assure continued heat removal.

14.15.6.2 Small Break LOCA Results

For a small break LOCA operator response is proceduralized in the LOCA and Functional Recovery EOPs. If less than 8.5 hours have elapsed since the start of the LOCA and Shutdown Cooling can be established, cooldown of the RCS is accomplished with the steam generators, if available. If the steam generators are not available, long term cooling is initiated by once-through-cooling by opening the PORVs. Opening of the PORVs results in cooling and reducing the RCS pressure sufficiently such that the HPSI pump refills and cools the RCS. The refilling and cooling of the RCS results in maintaining the boric acid concentration in the vessel well below the precipitation limit by dispersing the boron through the RCS by natural circulation.

The results of the analysis demonstrate that, for break sizes of 0.015 ft^2 or smaller, the RCS will refill and subsequently achieve a subcooled condition. The boric acid concentration in the vessel is also maintained well below the Westinghouse LOCA methodology (References 14.15-47 and 14.15-48) precipitation limit of 23.53 wt% as approved by the NRC, prior to refill.

14.15.7 Other LOCA Event Analyses

14.15.7.1 Loss of Coolant Accident During Shutdown

At any time after the Safety Injection Tanks (SITs) are isolated, the Reactor Coolant System pressure is 400 psia or less (Ref. 14.15-40), and the total stress in any component will be less than the total stress at the design pressure. Therefore, the possibility of a LOCA during shutdown cooling becomes even more remote than while at power.

The pressurizer pressure low signal (PPLS) has been bypassed below 1600 psia, and the safety injection tanks will be valved out when the system temperature and pressure reach 400°F and 400 psia respectively during shutdown. Using a maximum cooling rate of 75°F/hr, the above conditions are reached in less than 2 1/2 hours. Less than 1 1/2 hours later, when the system temperature and pressure reach 300°F and 250 psia, the system is placed in the shutdown cooling mode. In this mode, the coolant temperature is reduced from 300°F to 140°F in about 24 hours.

This shutdown procedure will usually occur only once per year or at most a few times per year. For each shutdown, a period of about 25 hours exists during which automatic initiation of the ECCS is not available, before refueling temperature is reached.

Bypassing of the PPLS does not bypass the containment pressure high signal (CPHS). The CPHS will still initiate safety injection and the automatic sequences involved. Furthermore, after the SITs are isolated, the operator will still be alerted to a LOCA by a combination of the following alarms and/or indications:

1. Low pressurizer level alarm
2. Low reactor coolant system pressure
3. High containment pressure alarm
4. Containment activity alarm
5. Containment temperature
6. Containment sump level alarm
7. Shutdown cooling temperature
8. Component cooling water temperature to and from the containment air cooling and ventilating system
9. Low volume control tank level alarm

For large breaks within the reactor coolant system, all of the above alarm indications will normally be present. As break size decreases, so will the number of indications and alarms. For the break size equal to or less than the capacity of one charging pump, the least number of indications and alarms that can be postulated to occur is three; namely, increases in: (1) containment activity, (2) temperature, and (3) temperature of the component cooling water to and from the containment atmosphere.

The containment air cooling and ventilating system can be used to reduce the high containment building pressure and to remove decay heat from the building if it becomes necessary to stop shutdown cooling. The capacity of the containment air cooling and ventilating system will be in excess of that required since the energy release will be far less than for a LOCA at full power; thus, no spray system backup should be required. Diesels will also be started to provide a standby source of power if outside power is lost during the accident.

An evaluation was performed to define the minimum Emergency Core Cooling (ECC) equipment operability requirements necessary to mitigate a LOCA during shutdown conditions (Ref. 14.15-49 and 14.15-50).

Minimum safety injection required to be operable during shutdown conditions are as follows:

<u>Mode 3 with RCS Pressure ≥1700 psia</u>

Safety Injection required is two LPSI and two HPSI trains aligned for automatic actuation. The HPSI and LPSI flow paths consist of piping, valves, and pumps that enable water to be injected into the Reactor Coolant System.

Four operable safety injection tanks (SITs) with isolation valves open.

Mode 3 with RCS Pressure <1700 psia and RCS Temperature ≥300°F

Safety injection required is one HPSI train aligned for automatic actuation. The exception is during startup (shutdown for at least 24 hours), one HPSI train is available for manual actuation below an RCS temperature and pressure of 450° F and 1700 psia. The HPSI pumps shall be disabled in accordance with Technical Specification 2.3 LTOP requirements. The HPSI flow path consists of piping, valves, and a pump that enables water to be injected into the Reactor Coolant System.

Four operable safety injection tanks (SITs) with isolation valves open. The SIT isolation valves may be closed when RCS pressure is less than 400 psia. The exception is during startup (shutdown for at least 24 hours), the SITs are required to be unisolated before reaching an RCS temperature and pressure of 515°F and 1700 psia.

14.15.7.2 Steam Generator Tube Failures in Conjunction with a Double-Ended Cold Leg Guillotine Break

An analysis was undertaken to determine how many steam generator tubes must fail, in conjunction with rupture of a reactor coolant system cold leg pipe, to cause steam binding sufficient to prevent emergency cooling water form rising above the midplane of the core.

The analysis showed there is no limit to the number of steam generator tubes that can rupture concurrent with a 24-inch double-ended cold leg break which will prevent water from rising to the core midplane. This is because the leaking steam generator will eventually discharge its contents to the primary system which will then blow down to the containment reducing the reactor coolant pressure to an acceptable value for refill. The nature of the refill inside the core barrel will depend upon the time at which the core pressure drops below that of the containment plus the static head of water in the downcomer annulus. Should the above condition exist shortly (about 1 minute) after the reactor coolant pressure drops to about 200 psig, then the safety injection tanks will contribute to the refill. However, should the primary system blowdown take longer, the refill of the barrel will be based on flow from the high and low pressure safety injection pumps.

14.15.7.3 Break Size Consistent With Charging Pump Capacity

Consideration has been given to the maximum reactor coolant system break size for which the charging pumps will make up the flow loss to the containment; so that a normal shutdown may occur.

As described in Section 9.2, there are three 40 gpm charging pumps. The number of charging pumps in operation and the corresponding maximum break area for which the charging pumps will make up the flow loss are given in Table 14.15-8. The discharge rate was determined from the orifice flow equation with a value of unity employed for the discharge coefficient.

Table 14.15-8 - "Maximum Break Area Consistent with Charging Pump Capacity"

No. of Pumps	Area (ft^2)	Equivalent Circular Diameter (in.)
1 (40 gpm)	1.37×10^{-4}	0.160
2 (80 gpm)	2.75×10^{-4}	0.224
3 (120 gpm)	4.12×10^{-4}	0.276

14.15.7.4 Core and Internals Integrity Analysis

The consequences of a LOCA on the reactor internal structure have been analyzed for reactor coolant system pipe breaks up to a double-ended rupture of a 32-inch pipe. Following a pipe rupture, two types of loading occur sequentially. The first is an impulse load of 15 to 30 milliseconds duration caused by rapid system depressurization from initial subcooled conditions to saturated conditions. This initial blowdown phase is followed by a two-phase fluid blowdown which persists for time periods varying up to several seconds, depending on the size of the postulated rupture.

In the early portion of the blowdown, acoustic waves propagate through the Reactor Coolant System. The WHAM code (Ref. 14.15-24) is used to calculate the pressure variations in the system following pipe rupture. WHAM calculates the impulse pressure loadings which the system is subjected to during passage of the pressure waves through the system.

For the saturated portion of the blowdown, the loadings on the reactor internals are associated with the fluid drag forces imposed by the high velocity two-phase fluid in its flow to the break location. The short term impulse forces are generally greater than the long term drag forces, except for the loads on some of the CEA shrouds in the case of a pipe rupture near the pressure vessel outlet nozzle.

The results of the blowdown force analyses for the reactor core support system are presented in Table 14.15-9, for a 32-inch hot leg break and for a 24-inch cold leg break, both for full power and zero power initial conditions.

Table 14.15-9 - "Maximum Pressure Difference Across Core and Core Support Structure"

	Pressure Difference, psi	
	32 in. Outlet Pipe Break	24 in. Outlet Pipe Break
Zero Power		
Lower Structure	54	40
Reactor Core	163	154
Upper Guide Structure	91	35
Full Power		
Lower Structure	29	30
Reactor Core	95	151
Upper Guide Structure	52	45

The maximum calculated stresses and deflections during blowdown for critical reactor components were found to be below allowable stresses (Ref. 14.15-36). Reference 14.15-36 lists the corresponding allowable values of stress, pressure, or deformation based on the design criteria specified in Section 3.2, along with the estimated values of stress, pressure, or deformation at which failure would occur. It is emphasized that the shutdown mechanism for large breaks is voiding of the core. It is not necessary for the CEAs to insert (see Section 1.5.5).

During an inlet pipe break, the core support barrel is subjected to time dependent axial loads and axially varying radial pressure differentials. Axial stresses in the core support barrel and shear and bending stresses in the lower support structure were evaluated, using conservative stress analysis methods. The peak of the axial pressure pulse, calculated from WHAM, was applied as a steady loading. The SEAL-SHELL computer program was used to assure that stresses and deformation are within design limits when the barrel is subjected to the peak of the time-dependent axially varying, radial pressure distribution.

During an outlet break, the core support barrel is subjected to a time dependent upward force and external radial pressure. Loads on the upper guide structure were evaluated by calculating the acceleration of the core under the time varying axial pressures (during the subcooled portion of the LOCA) and equating the kinetic energy of the core to the strain energy of the upper guide structure and core after impact. The strain energy is then related to stresses in the system. Bending stresses in the upper guide structure were evaluated during two-phase flow by using peak values for the axially varying pressure forces during this regime. The upper guide structure, modeled as a system of continuous and discrete elements, was subjected to a pressure time history, and axial stresses were evaluated.

The analyses which have been performed indicate that design limits are not exceeded even when dynamic effects are taken into account.

14.15.7.5 Reactor Vessel Thermal Shock

The effect of operation of the emergency core cooling system on the reactor vessel following a loss-of-coolant has been discussed in CE report A-68-9-1, "Thermal Shock Analysis on Reactor Vessel Due to Emergency Core Cooling System Operation", by W.H. Tupenny et al, March 15, 1968. This was submitted as Amendment 9 to the Maine Yankee License Application (AEC Docket No. 50-309). Additional information for this condition appears in CE report A-68-10-2, "Experimental Determination of Limiting Heat Transfer Coefficients During the Quenching of Thick Steel Plates in Water", by J.H. Simon, M.W. Davis and W.H. Tupenny, December 13, 1968. This report was placed in the public record in January, 1969. This work is discussed in Section 1.5.4.

14.15.7.6 Hydrogen Accumulation in Containment

Hydrogen accumulation in the containment is discussed in Section 14.17. The hydrogen produced by radiolysis is released to the containment building where it mixes with the steam-air atmosphere. In addition to the radiolytic hydrogen, the hydrogen produced by reaction of steam with the zircaloy cladding is considered.

To control hydrogen concentration, a purge of the containment building would be instituted when 3 percent (volume) hydrogen is reached. This value is below the flammability limit.

14.15.7.7 Reactor Operator Action

An evaluation has been performed to determine what actions the reactor operator would have to perform or may have to perform in the unlikely event of a design basis LOCA. Consideration was also given as to the allowable time periods within which the action would or might have to be performed. The results of the evaluation are contained in EOP-03.

In this evaluation it was assumed that all engineered safeguards and support systems functioned to fulfill the design objectives of each system. Certain additional actions may have to be performed in the unlikely event of a malfunction of portion(s) of certain system(s); however, this evaluation was performed to specifically identify reactor operator actions that will or may be required in the event of a loss-of-coolant accident.

This evaluation began by assuming a double-ended rupture of a 32-inch reactor coolant system pipe. This is followed by the response of the systems in the order and manner in which the various components of the systems are expected to function.

The results of this evaluation indicate that, in the event of a LOCA with the engineered safeguards and support systems functioning to fulfill design objectives, no control actions are required by the reactor operator for several hours following the accident.

14.15.8 Radiological Consequences of a LOCA

The ECCS, following a design basis LOCA (double-ended break), limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the environs. However, in order to demonstrate that the Fort Calhoun Station does not represent any undue radiological hazard to the public, the off-site doses and doses to Control Room personnel from a postulated LOCA have been calculated to assure that the radiological hazards are below the limits of 10 CFR Part 100 (Ref. 14.15-39) or Standard Review Plan Section 6.4 (Ref. 14.15-25) as appropriate.

14.15.8.1 Off-Site Radiological Consequences

The off-site doses from a postulated LOCA have been calculated based on the following assumptions and conditions (Ref. 14.15-42):

1. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at a power level of 1500 MWt.

2. 100 percent of the core noble gas inventory and 25 percent of the core iodine inventory are immediately available for release from the containment.

3. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% is in the form of particulate iodine, and 4% is in the form of organic iodine.

4. No credit is taken for spray removal of halogens even though the redundant spray systems should be effective in removing the elemental and particulate halogens.

5. The containment leak rate is 0.2 percent of the free volume for the first 24 hours, and 0.1 percent of the free volume for the remaining duration of the accident.

6. The Containment Air Recirculating, Cooling and Iodine Removal System filter removal constant for halogens is 5.14 per hour based on a filtration system flow rate of 100,000 scfm (only 50 percent of the installed capacity assumed available) with removal efficiency of 85% (Ref. 14.15-38) and a containment net free volume of 1.05×10^6.

7. Containment purge system is started when the hydrogen concentration in the containment reaches three (3) volume percent (NOTE: THE RADIOLOGICAL CONSEQUENCES OF HYDROGEN PURGE SYSTEM ARE FULLY DISCUSSED IN SECTION 14.17 OF THE USAR).

8. The dispersion factor for the EAB is $2.55x10^{-4}$ sec/m^3, and the dispersion factor for LPZ outer boundary is conservatively assumed (based on a dispersion factor for 0 - 8 hours) as $4.53x10^{6}$ sec/m^3 (Ref. 14.15-41).

9. The end of cycle iodine and noble gas activity in the reactor core assumes a core average enrichment of 4.5 weight percent U-235 and a maximum rod burnup of approximately 60,000 MWD/MTU (Reference 14.15-46).

Reactor Core Activity for Extended Burnup

The nuclide inventories (curies) in the reactor core used in determining the EAB and LPZ doses are shown below (from Reference 14.15-46).

Nuclide	Inventory (curies)
I-131	4.83E7
I-132	6.98E7
I-133	9.87E7
I-134	1.08E8
I-135	9.24E7
Kr-85m	1.26E7
Kr-85	6.52E5
Kr-87	2.41E7
Kr-88	3.38E7
Kr-89	4.12E7
Xe-131m	5.42E5
Xe-133	9.61E7
Xe-135m	1.94E7
Xe-135	2.44E7
Xe-137	8.26E7
Xe-138	8.13E7

Based on these assumptions, the resulting off-site doses are below the 10 CFR Part 100 limits:

	Thyroid (REM)	Whole Body (REM)
EAB	58.9	3.3
LPZ	5.5	0.43
10 CFR 100	300.0	25.0

14.15.8.2 Post-LOCA Doses to Control Room Personnel

Control Room Radiation Shielding

Radiation shielding is provided for the control room envelope by concrete walls with a density of 145 pounds/ft^3 as follows (Ref. 14.15-26):

Control Room Wall	Thickness
North	2' 0"
South	1' 3"
East	1' 6"
West	1' 6"
Roof	1' 6"

In addition, as a result of a previous control room shielding review (Ref. 14.15-27), a 1 foot thick concrete wall is provided to shield personnel from containment spray pipes in the auxiliary building. Major penetrations in the bulk shielding include two door entrances in the east wall, two duct penetrations in the far western corner of the south wall, and two penetrations in the roof: one for the toilet exhaust and one for the elevator machine room exhaust.

Radiological Habitability Analyses

Radiological analyses have been performed to assure that the radiation doses to control room personnel do not exceed the limits of GDC 19 (Ref. 14.15-43) and the guidelines of SRP 6.4 for postulated design basis accidents. The methodology, data, assumptions, and calculated results for each design basis accident is presented below.

Design parameters for the control room ventilation and emergency filtration system used in the radiological habitability analyses are as follows:

1. The emergency filtration system has a capacity of 2,000 cfm to filter an outside makeup air flow rate of 1,000 cfm $\pm$100 cfm plus recirculated control room air. The radiological analyses assume an outside makeup flow rate of 1,100 cfm with 900 cfm on recirculated air.

2. The emergency filtration system provides a filtration efficiency of 99 percent for both elemental and organic iodine.

3. No unfiltered inleakage is assumed since the control room is maintained at a positive pressure with respect to adjacent areas and air lock type vestibules are installed for each entrance to the control room.

4. Iodine filtration of the recirculated air is assumed to be unavailable for 4 hours post accident to account for repair of the non-redundant recirculation duct isolation damper.

5. The normal operation mode unfiltered outside air intake flow rate of 1,000 cfm is isolated within 10 seconds upon actuation of the emergency filtration system.

6. The length of the intake ventilation ductwork upstream of the normal intake isolation damper is 112 feet with a transit time of 24.5 seconds.

<u>Loss of Coolant Accident</u>

The radiological consequences due to a design basis LOCA have been analyzed and include an evaluation of the radiological impact due to airborne radioactivity inside and outside the control room, as well as direct shine from contained radiation sources. Dose contributions from the following sources have been evaluated:

1. Airborne Radioactivity within the Control Room (Ref. 14.15-28)
2. Overhead Cloud Shine (Ref. 14.15-31)
3. Control Room Emergency Charcoal Absorbers (Ref. 14.15-32)
4. Piping containing Post-LOCA Radioactivity (Ref. 14.15-33)
5. Containment Shine (Ref. 14.15-34)
6. ESF Internal/External Radiological Leakage (Ref. 14.15-35)

Each dose contributor is discussed in detail below.

Airborne Radioactivity within the Control Room

Control room doses due to airborne radioactivity releases from containment and engineered safety feature (ESF) leakage have been evaluated. The calculated doses are as follows (Ref. 14.15-28):

Dose (rem)	
Whole Body Gamma:	0.59
Thyroid:	7.6
Beta Skin:	8.9

The doses have been calculated based on the following assumptions and conditions:

1. The reactor core equilibrium noble gases and iodine inventories are based on long-term operation as a power level of 1500 MWT.

2. 100 percent of the core noble gas inventory and 25 percent of the core iodine inventory are immediately available for release from the containment.

3. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% is in the form of particulate iodine, and 4% is in the form of organic iodine.

4. No credit is taken for spray removal of halogens even though the redundant spray systems should be effective in removing the elemental and particulate halogens.

5. The control room emergency iodine filtration system is automatically initiated upon a safety injection actuation signal which occurs 0.55 seconds post-LOCA.

6. Credit for decay of fission products in containment during the duration of the accident is taken.

7. The containment leak rate is 0.1 volume percent per day for the first 24 hours and 0.05 volume percent for the remainder of the accident (1-30 days).

8. The Containment Air Recirculating, Cooling and Iodine Removal System filter removal efficiency for halogens is conservatively assumed as 85% (Ref. 14.15-38) for both organic and inorganic species with a filtration system flow rate of 100,000 scfm (only 50% of the available capacity is assumed available). However, the credit is not taken until 0.02 hours following the accident, and reduction of the organic iodides is taken only until they are reduced to 50% of their initial concentration.

9. The containment hydrogen purge operation is assumed to be initiated 46 days following the accident. Therefore, this source is not considered for control room habitability purposes.

10. The control room net free volume is 100,000 cubic feet.

11. The leakage from ESF equipment (Ref. 14.15-35), i.e. high and low pressure SI pumps, etc, is assumed to be twice the maximum proposed Technical Specification limit of 1243 cc/hr beginning at the time of the recirculation actuation signal (27.4 minutes) and continuing for the duration of the accident (30 days). The volume of the containment sump water is 209,000 gallons and contains 50% of core iodine inventory.

12. A decontamination factor of 10 is assumed for ESF leakage between the water and steam phases.

13. Containment free air volume is 1.05 X 10^6 ft^3.

14. Core radioactivity inventory (Ref. 14.15-42):

Nuclide	Activity (Ci)
Kr-85	3.34×10^5
Kr-85m	1.05×10^7
Kr-87	1.93×10^7
Kr-88	2.73×10^7
Xe-131M	2.94×10^5
Xe-133	8.46×10^7
Xe-135	1.51×10^7
Xe-135m	1.71×10^7
Xe-138	6.75×10^7
I-131	4.16×10^7
I-132	6.04×10^7
I-133	8.44×10^7
I-134	9.12×10^7
I-135	7.84×10^7

15. Meteorological dispersion factors for the containment release and for the main plant vent (ESF leakage) are given in Reference 14.15-29. Chi/Q values are calculated using the Murphy/Kampe methodology (Ref. 14.15-30).

Overhead Cloud Shine

The direct radiation shine dose to control room personnel from an overhead cloud of airborne radioactivity due to containment and ESF leakage was calculated (Ref. 14.15-31). The 30-day integrated radiation dose for bulk control room shielding (no penetrations) was calculated to be 1.2 rem gamma whole body (Ref. 14.15-27) and is based on the following data and assumptions:

1. Airborne radioactivity releases are based on the same data and assumptions outlined previously for calculating airborne radioactivity within the Control Room.

2. Concrete shield wall thickness is assumed to be 1'6" for all walls and the roof. This is a conservative average based upon the actual dimensions.

3. Dose is based on an unshielded semi-hemispherical cloud model per Reg. Guide 1.4 (Ref. 14.15-44) with attenuation credit for 1'6" of concrete at a density of 145 pounds/ft^3.

4. The airborne cloud radioactivity concentration is based on meteorological dispersion factors for the centerline of the control room envelope which is about 21 meters from the containment in the north sector. The χ/Qs for the containment are from Reference 14.15-29 and are as follows:

Time Period	χ/Q (sec/m^3) Main Plant Vent Containment
0-8 hrs	$1.10X10^{-3}$$6.96X10^{-3}$
8-24 hrs	$6.49X10^{-4}$$4.22X10^{-3}$
1-4 days	$2.53X10^{-4}$$2.30X10^{-3}$
4-30 days	$7.26X10^{-5}$$9.21X10^{-4}$

The direct radiation doses in localized areas of the control room due to the major penetrations have also been evaluated. These penetrations include the following:

- Two door entrances in the east wall.
- Two ventilation duct penetrations (12" and 16" diameter) in the far western corner of the south wall.
- Toilet exhaust penetration (12" X 12") in the roof above the mezzanine office area.
- Elevator machine room exhaust penetration (22" X 22") in the roof above the elevator machine room.

The radiation shine from the ventilation duct penetrations in the south wall has not been quantified since this impacts the mechanical equipment room, which is not habitable for continuous occupancy as discussed in Reference 14.15-29. Similarly, radiation shine through the elevator machine exhaust penetration has not been quantified since this area does not require post-accident occupancy.

Radiation shine from the control room doors was calculated as a function of distance into the control room (Ref. 14.15-31). Based on these results, at a distance of 8 feet a 3 rem integrated dose is calculated which when added to other doses is less than the 5 rem GDC 19 limit.

Locations within 8 feet of the doors will not be occupied to any significant extent, and, therefore, will not present a radiation dose concern.

Control Room Emergency Charcoal Absorbers

The maximum calculated dose in the control room due to direct shine from the radioactivity buildup on the charcoal absorber is 0.043 rem gamma whole body (Ref. 14.15-32). The airborne releases are calculated using the data and assumptions outlined previously. Shielding credit is taken for the 1'3" thick south concrete wall.

Piping Containing Post-LOCA Radioactivity

The only significant dose contribution from piping sources is from containment spray piping at elevations 1032" and 1036'8" of the auxiliary building. The 30-day integrated dose at various control room locations has been calculated as follows (Ref. 14.15-33):

Location	Dose (rem)
Mechanical Equipment room	9.8
Toilet Room/Lunch Room	1.54
Auxiliary Control Panel Area	0.50
Main Control Board Area	0.008
Mezzanine Office	1.4

The calculated doses within the mechanical equipment room exceed GDC 19; however, this area will be administratively controlled during post-accident conditions. The maximum dose for all other areas of the control room envelope is 1.54 rem. The most representative dose for the control room envelope is 0.008 rem and corresponds to the dose calculated for the main control board area, this value is used as input to the calculation of the total gamma whole body dose from significant direct shine contributors.

Containment Shine

The direct shine dose from the containment, which includes the dose from the airborne radioactivity in the containment atmosphere and the radioactivity buildup on the containment recirculation and iodine removal filters, was calculated to be 0.128 rem (Ref. 14.15-34).

Summary of LOCA Doses

A summary of the radiological consequences due to a design basis LOCA is as follows:

	30-Day Post-LOCA Integrated Dose (rem)		
Dose Contribution	Gamma	Thyroid	Beta Skin
1. Airborne Radioactivity within Control Room	0.59	7.6	8.9
2. Overhead Cloud Shine	1.2	N/A	N/A
3. Control Room Emergency Charcoal Absorbers	0.043	N/A	N/A
4. Piping containing Post-LOCA Radioactivity (Main Control Board Area)	0.008	N/A	N/A
5. Containment Shine	0.128	N/A	N/A
6. ESF Internal/External Radionuclide Leakage (Ref. 14.15-35)	2.53	19.4	18.1
Total	4.5*	27.0	27.0
GDC 19/SRP 6.4 Dose Limits	5	30	30

* This total dose applies to the main control board area of the control room. The maximum dose within the control room envelope occurs in the toilet/lunch room area and is calculated to be 3.5 rem.

Based upon the above results it can be seen that the calculated doses are within the limits of GDC 19 and the guidelines of SRP 6.4. The 5 rem gamma dose limit is met throughout the control room envelope except for the mechanical equipment room, which is not continuously occupied at any time. The area directly inside the two control room doors is also calculated to exceed 5 rem; however, these areas will not be occupied to a large extent during post-LOCA conditions.

14.15.9 Conclusions

The LOCA analysis demonstrates that the ECCS provides adequate core cooling, by keeping the core in a coolable geometry, over the entire spectrum of breaks, including a double-ended hot leg guillotine.

The results of radiological consequences show that the thyroid and whole body doses, using the conservative assumptions, are well within the limits of 10 CFR Part 100 at the EAB and LPZ, and well below the limits of Standard Review Plan 6.4 for Control Room Personnel.

14.15.10 Specific References

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14.15-32 Stone & Webster Calculation No. 16472.26-UR(B)-015-0, 30 Day Integrated Dose (gamma) to Operators in the Control Room due to Direct Shine from the Control Room Intake/Recirculation Filters; 6/28/89.

14.15-33 Stone & Webster Calculation No. 16472.19-UR-001-0, 30 Day Post-LOCA Total Integrated Doses in the Control Room Due to Piping in the Auxiliary Building; 11/18/88.

14.15-34 Stone & Webster Calculation No. 16472.26-UR(B)-007-01, 30 Day Integrated Dose (gamma) to Operators in the Control Room due to Direct Shine from the Containment (airborne activity and carbon filters); 6/28/89.

14.15-35 Radiological Consequences of Internal and External ESF Equipment Leakage, FC05934; 5/26/92.

14.15-36 Maximum Stresses, Pressures and Deflections for Critical Internals Components; CENPSD-110-P.

14.15-37 CEN-268 Rev. 1, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients" May 1987.

14.15-38 Westinghouse Calculation CN-CRA-93-142, Rev. 0 "Post LOCA Iodine Removal"

14.15-39 Code of Federal Regulations, Title 10 Part 100.11 "Determination of Exclusion Areas Low Population Zone and Population Center Distance" Current Revision.

14.15-40 WCAP-12476 "Evaluation of LOCA during Modes 3&4 Operation for W NSSS," November 1991

14.15-41 Calculation FC-05526, Rev. 0 "Atmospheric Dispersion USAR Calculations.

14.15-42 Letter Westinghouse (R.G. Creighton) to OPPD (R. L. Phelps) 91CF*-G-0054 dated August 22, 1991.

14.15-43 Code of Federal Regulations, Title 10, Part 50 Appendix A General Design Criteria #19 "Protection Systems Reliability."

14.15-44 U.S. Atomic Energy Commission Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" Rev. 2, June 1974.

14.15-45 Mixed Vendor Core Data List for Fort Calhoun Unit 1, EA-FC-90-004, R0.

14.15-46 Engineering Analysis EA-FC-90-111, Rev. 0, Shielding Calculation Revision - Phase I - Comparison Study.

14.15-47 Letter CLC-NS-309 from C.L. Caso (Westinghouse) to T.M. Novak (NRC), dated April 1, 1975.

14.15-48 Westinghouse Memorandum SE-SE-1200, from H.V. Julian to M. Harding and I. Ratsep, dated May 26, 1977.

14.15-49 Letter from J.M. Cleary (ABB-CE) to J.L. McManis (OPPD), "Recommended ECCS Equipment Operability Requirements when the Reactor is not Critical", ST-98-272, May 21, 1998 (Calculation FC06738).

14.15-50 Letter from J.M. Cleary (ABB-CE) to A.W. Richards (OPPD), "Relaxation of ECCS Equipment Requirements While Shutdown," ST-99-221, March 19, 1999 (Calculation FC06739)

14.15-51 Letter from R.G. Creighton (Westinghouse) to T.A. Heng (OPPD), "Evaluation of 225 psia SIT Pressure," 99CF-G-001, January 5, 1999.

14.15-52 Calculation FC-06286, Rev. 0, "Safety Injection Tank TLU Calculation."

14.15.11 General References

14.15.11.1 Bordelon F.M., *et al*; "The Westinghouse Evaluation Model and Supplementary Information", WCAP-8471-P-A (Proprietary), WCAP-8472 (Non-Proprietary); January 1975.

14.15.11.2 Letter from R.G. Creighton (Westinghouse) to R.L. Phelps (OPPD), "USAR Update", 91 CF*-G-0054; August 22,1991.

14.22 REACTOR COOLANT SYSTEM DEPRESSURIZATION INCIDENT

14.22.1 General

The RCS Depressurization event is characterized by a rapid decrease in the primary system pressure caused by either the inadvertent opening of both power operated relief valves (PORV's) or a single primary safety valve while operating at rated thermal power.

Following the initiation of the event, steam is discharged from the pressurizer steam space to the quench tank where it is condensed and stored. To compensate for the decreasing pressure, the water in the pressurizer flashes to steam and the proportional heaters increase the heat added to the water in the pressurizer in an attempt to maintain pressure. During this time, the pressurizer level also begins to decrease causing the letdown control valves to close and additional charging pumps to start in an attempt to maintain level. As the pressure continues to drop, the backup heaters energize to further assist in maintaining the primary pressure. A reactor trip is initiated by the TM/LP trip to prevent exceeding the DNBR SAFDL (Refs. 14.22-11 and 12).

In order to ensure that enough margin is built into the TM/LP trip, to prevent the DNBR SAFDL from being exceeded, a conservative pressure bias term for the TM/LP trip must be calculated. The pressure bias term accounts for the DNBR margin degradation, caused by the depressurization, between the time reactor trip conditions exist and the time of minimum DNBR. This time is primarily due to the signal processing delays in the TM/LP trip logic and the CEA clutch coil delay time.

14.22.2 Method of Analysis

The RCS Depressurization incident was analyzed using the CESEC computer code which models neutron kinetics with fuel and moderator temperature feedback, the reactor control system, the reactor coolant system, the steam generators, and the main steam and feedwater systems (Reference 14.22-2,3,4 and 5). The results of the transient simulation, the transient average core heat flux, average channel mass flow rate, reactor core inlet temperature, and reactor coolant system pressure serve as input to CETOP (Reference 14.22-6) which performs open channel pressure balancing calculations. This code uses the CE-1 critical heat flux correlation (References 14.22-7 and 14.22-8) to calculate the DNB ratio for the hot channel as a function of time and axial position (see Section 3.6).

The most negative moderator temperature coefficient (MTC) of reactivity was used to increase the coolant temperature feedback effects which result in higher heat fluxes and thus greater residual heat thereby minimizing DNBR. In order to maximize the negative reactivity feedback from the increasing fuel temperature a 1.45 multiplier was applied for Cycle 19 to the Doppler coefficient of reactivity. The initial pressurizer pressure was chosen to be 2172 psia which corresponds to the maximum allowed pressure plus uncertainties. The charging pumps, the pressurizer proportional heaters and the pressurizer backup heaters were assumed to be inoperable with the letdown valves open at the maximum flow. The higher initial pressure, maximum letdown flow and the inoperability of the pressurizer heaters and charging pumps result in a faster rate of depressurization. These assumptions yield a lower transient minimum DNBR and a maximum pressure bias term (Reference 14.22-1).

Table 14.22-1 contains the list of initial conditions and assumptions including uncertainties for Cycle 19 used in the analysis of the RCS Depressurization event.

Table 14.22-1 - "Cycle 19 Key Parameters for the RCS Depressurization Event"

Parameter	Units	Value
Initial Core Power Level	MWth	1541.6
Core Inlet Coolant Temperature	°F	547
Pressurizer Pressure	psia	2172
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.5
RCS Flow Rate	gpm	230,000
Total Trip Delay Time (Processing plus CEA holding coil delay)	sec	1.4

14.22.3 Affected Plant Technical Specifications

The RCS depressurization event analysis uses inputs from the following Technical Specifications (Reference 14.22-10):

LCO 2.10.2	Reactivity Control Systems and Core Physics Parameters Limits
LCO 2.10.4	Power Distribution Limits

The results of the RCS depressurization event analysis are used as inputs to Technical Specification 1.3, Limiting Safety System Setting, Reactor Protective System.

14.22.4 Affected Plant Systems

For this event the affected plant systems are the reactor coolant system, the reactor protective system (TM/LP), and the reactivity control system.

14.22.5 Limiting Parameters for Reload Analysis

Reevaluation of the RCS depressurization incident is required when either of the following conditions exists.

- Core physics and/or thermal-hydraulic parameters change in a nonconservative direction.
- A plant design modification is expected to cause a change to a pertinent Technical Specification limiting condition of operation (LCO).

Any changes to parameters and/or technical specifications must result in a DNBR which is greater than 1.18 (Reference 14.22-11 and 14.22-12). This minimum is required in order to maintain adequate heat transfer from the core and limit the fuel cladding temperature rise during the RCS depressurization event.

14.22.6 Results

The RCS Depressurization event was reanalyzed for Cycle 19 to determine the pressure bias term input to the TM/LP trip. The trip setpoints incorporating this bias factor will provide adequate protection to prevent the DNBR SAFDL from being exceeded during the transient.

The analysis of this event shows that a conservative pressure bias term is 30.0 psia (Reference 14.22-9). The sequence of events for the RCS Depressurization event is presented in Table 14.22-2. Figures 14.22-1 through 14.22-4 show the transient behavior of the core power, core average heat flux, reactor coolant system temperatures, and reactor coolant system pressure.

Table 14.22-2 - "Cycle 19 Sequence of Events for the RCS Depressurization Event"

Time (sec)	Event	Setpoint or Value
0.0	Inadvertent Opening of both Pressurizer Relief Valves	-----
7.77	Manual Trip	2079.37 psia
9.36	Time of Minimum DNBR	2054.41 psia

14.22.7 Conclusions

The analysis of this event shows that a pressure bias term of 24.96 psia has been calculated, but a value of 30.0 psia will be used to be conservative for Cycle 19 operation (Reference 14.22-9).

14.22.8 Specific References

14.22-1 OPPD-NA-8303-P, Rev. 04, Omaha Public Power District Reload Core Analysis Methodology "Transient and Accident Methods and Verification," Section 5.7.

14.22-2 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.22-3 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.22-4 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.22-5 Response to questions on CESEC, CEN-234(c)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.22-6 "CETOP: Thermal Margin Model Development," CE-NPSD-150-P, CE Proprietary Report, May 1981.

14.22-7 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution," CENPD-152-P-A, September 1976.

14.22-8 "CE Critical Heat Flux, Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution," CENPD-207-P, June 1978.

14.22-9 "Cycle 19 RCS Depressurization Analysis," EA-FC-98-051, Rev. 0.

14.22-10 Fort Calhoun Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 190, April 15, 1999.

14.22-11 "Statistical Combination of Uncertainties," Parts 1-3, CEN-257(0)-P, November 1983.

14.22-12 "Statistical Combination of Uncertainties," Parts 1-3, Supplement 1-P to CEN-257(0)-P, August 1985.

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APPENDIX A

QUALITY ASSURANCE PROGRAM

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1. INTRODUCTION

This Appendix describes the Omaha Public Power District's (OPPD) Quality Assurance Program for the operation of Fort Calhoun Station. The program is based on the criteria of Appendix B to 10CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; General Design Criterion 3, Appendix A to 10CFR Part 50, "Fire Protection"; Subpart H of 10CFR Part 71, "Packaging and Transportation of Radioactive Material;" the applicable guidance provided in American National Standard, ANSI N18.7 "Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants" and, ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants," and its associated daughter standards; and, Regulatory Guide 1.120, Revision 1, "Fire Protection Guidelines for Nuclear Power Plants" and Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." Attachment 1 details OPPD's specific commitments with respect to the ANSI N45.2 series, other industry QA standards, and associated NRC Regulatory Guides, including clarifications or alternatives used as a basis for OPPD's Quality Assurance Program and QA Plan.

The program is applied to: Critical Quality Elements (CQE) defined as those structures, systems, components, or items whose satisfactory performance is required to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public; those fire protection systems and equipment used or installed in areas housing safety-related equipment, and other areas where an unsuppressed fire could potentially damage safety-related structures, systems or components; those activities affecting the components of radioactive material packaging for transport which are significant to safety; Limited Critical Quality Elements (Limited CQE) defined as those structures, systems, components or items whose satisfactory performance is required to prevent or mitigate the failure of those structures, systems, components or items identified as CQE.

2. ORGANIZATION

OPPD's organization for carrying out an effective operations phase Quality Assurance Program is shown in Figure A-1.

2.1 Vice President

The Vice President is the upper level management position which has primary responsibility for plant operations; formulation, implementation, and assessment of the effectiveness of the fire protection program; implementation and maintenance of the ALARA Radiation program; and packaging of radioactive material for transport. This individual is also responsible for approval and overall implementation of the Quality Assurance Plan.

2.2 Division Manager-Nuclear Assessments

The Division Manager-Nuclear Assessments is the upper level management position which has primary responsibility for formulation, implementation, and assessment of the effectiveness of the Quality Assurance Program. This individual also has primary responsibility for providing support in the areas of computing services, health physics, nuclear safety review, and the employee concerns program.

2.3 Division Manager-Nuclear Engineering

The Division Manager-Nuclear Engineering is the upper level management position which has primary responsibility for the design of modifications and additions to Fort Calhoun Station and the designation of structures, systems, components, or items classified as CQE, Limited CQE or covered by the fire protection program. This individual also has primary responsibility for providing system engineering and technical support for Fort Calhoun Station.

2.4 Division Manager-Nuclear Operations

The Division Manager-Nuclear Operations is the upper level management position which has primary responsiblity for the safe operation of Fort Calhoun Station and for licensing activities, training, providing security, and emergency planning activities at Fort Calhoun Station.

2.5 Division Manager-Nuclear Support Services

The Division Manager-Nuclear Support Services is the upper level management position which has primary responsibility for administrative services, the Condition Reporting system, procurement services, receipt inspection, and quality assurance material and vendor audits in support of Fort Calhoun Station.

2.6 Division Manager-Material Management

The Division Manager-Material Management has primary responsibility for formulation and implementation of OPPD's procurement policy. Implementation of the QA Plan requirements for procurement is shared by those Division Managers performing nuclear procurement activities.

2.7 Division Manager-Information Technology

The Division Manager-Information Technology has primary responsibility for development, implementation and maintenance of corporate computing systems in accordance with the Information Technology General Policy.

2.8 Manager-Technical Services (Production Operations Division)

The Manager-Technical Services has primary responsibility for Fort Calhoun maintenance and modification activities performed by Central Maintenance and OPPD's maintenance contractors.

2.9 Division Manager-Electric Operations

The Division Manager-Electric Operations has primary responsibility for certain Fort Calhoun electrical maintenance activities.

2.10 Division Manager-Fuels

The Division Manager-Fuels has primary responsibility for the procurement of nuclear fuel.

2.11 Division Manager-Human Resources

The Division Manager-Human Resources has primary responsibility for the administration and management of the Fitness For Duty Program, including chemical testing, employee assistance, and employee and supervisory training programs.

2.12 Manager-Quality Assurance & Quality Control

The Manager-Quality Assurance & Quality Control, under the Division Manager-Nuclear Assessments, has primary responsibility for the development and implementation of the Quality Assurance Program for Fort Calhoun Station designs, modifications, operations, and procurements. This responsibility extends into all projects and operations activities affecting safety including engineering, design, procurement, and construction. This individual has the duty and authority to identify quality-related problems; to recommend solutions; and to verify the implementation of corrective actions taken. This individual has the authority to "stop work" associated with activities affecting safety if the work in progress does not conform to applicable requirements, specifications, or established procedures. By reporting to the Division Manager-Nuclear Assessments the Manager-Quality Assurance & Quality Control is isolated from budgeting and scheduling influences of plant operations.

The principal duties and responsibilities of the Manager-Quality Assurance & Quality Control include the following:

- Manages the OPPD Quality Assurance Program.

- Manages and coordinates an independent audit and QA surveillance program and review of OPPD quality activities to assure compliance with, and effectiveness of, the Quality Assurance Program.

- Training and indoctrination of appropriate OPPD personnel in the Quality Assurance program.

- Prepares Quality Assurance procedures, instructions, and reports, including the description, preparation, and maintenance of OPPD's QA Plan and other QA program documents.

- Acts as an internal consultant on Quality Assurance related matters.

- Maintains a file of quality-related codes and standards for the operation, maintenance and modification of nuclear power plants.

- Reviews design data and changes, operating procedures, and test reports for quality requirements.

- Manages quality control activities at Fort Calhoun Station.

The qualification requirements of the Manager-Quality Assurance and Quality Control are:

(1) A college degree or equivalent experience. Registration as a professional engineer in the State of Nebraska is desirable. Advanced degree in management or work in quality related discipline is desirable.

(2) Eight (8) or more years in a supervisory or management position related to the operation, maintenance, testing, and construction of nuclear power plants, including quality assurance.

(3) Working knowledge and understanding of regulations, standards, and guides relating to nuclear power plant quality assurance.

(4) Thorough management, administrative and supervisory skills.

(5) A very high degree of emotional stability, maturity, and personal integrity. Must have a high degree of initiative, and the ability to make sound decisions and defend them to top management.

(6) The ability to work closely and in harmony with others. The ability to coordinate several complex operations simultaneously.

The Manager-Quality Assurance and Quality Control has assigned staff to assist in the execution of the manager's duties and responsibilities.

2.13 Manager-Nuclear Safety Review Group

The Manager-Nuclear Safety Review Group, under the Division Manager-Nuclear Assessments, has primary responsibility for independently reviewing and advising management of all aspects of nuclear safety.

2.14 Manager-Fort Calhoun Station

The Manager-Fort Calhoun Station, under the Division Manager-Nuclear Operations, has direct responsibility for the safe operation of Fort Calhoun Station including operation, maintenance, planning and scheduling, radiation protection and chemistry activities.

2.15 Manager-Nuclear Licensing

The Manager-Nuclear Licensing, under the Division Manager-Nuclear Operations, has primary responsibility for licensing activities, maintaining the FCS Operating License and Technical Specifications, NRC and INPO interfaces, 10 CFR Part 21 program coordination, and commitment tracking.

2.16 Manager-Security and Emergency Planning

The Manager-Security and Emergency Planning, under the Division Manager-Nuclear Operations, has primary responsibility for developing and maintaining security policies and procedures for Fort Calhoun Station in compliance with 10 CFR Part 73, the Site Security Plan, and the QA Plan. This position also has primary responsibility for emergency planning activities, including development, administration and maintenance of the Fort Calhoun Station Radiological Emergency Response Plan (RERP) and Implementing Procedures (EPIPs).

2.17 Manager-Training

The Manager-Training, under the Division Manager-Nuclear Operations, has primary responsibility for nuclear technical training programs and the Fort Calhoun Simulator.

2.18 Manager-Nuclear Procurement Services

The Manager-Nuclear Procurement Services, under the Division Manager-Nuclear Support Services, has primary responsibility for development and review of nuclear procurement specifications, procurement, material control activities, receipt inspection and qualification of OPPD vendors and suppliers.

2.19 Manager-Nuclear Administrative Services

The Manager-Nuclear Administrative Services, under the Division Manager-Nuclear Support Services, has primary responsibility for document control of design documents, drawings, and procedures associated with Fort Calhoun Station.

2.20 Manager-Design Engineering - Nuclear

The Manager-Design Engineering - Nuclear, under the Division Manager-Nuclear Engineering, has primary responsibility for detailed designs for modification of and additions to Fort Calhoun Station; maintenance of as-built design data; preparation of design drawings; classification of CQE, Limited CQE, fire protection, and radioactive waste disposal structures, systems, and components; Updated Safety Analysis Report (USAR) engineering updates; fuel reload analysis and technical evaluations.

2.21 Manager-Nuclear Process Computing Services

The Manager-Nuclear Process Computing Services, under the Division Manager-Nuclear Assessments, has primary responsibility for development, implementation and maintenance of process computing systems and nuclear business unit applications in support of Fort Calhoun Station.

2.22 Manager-System Engineering

The Manager-System Engineering, under the Division Manager-Nuclear Engineering, has primary responsibility for operability of fire protection systems, system engineering, and assigned technical support services for Fort Calhoun Station.

2.23 Manager-Nuclear Construction Management

The Manager-Nuclear Construction Management, under the Division Manager-Nuclear Engineering, has the primary responsibility for planning and execution of assigned outage and on-line modifications, other construction activities and assigned technical support services at Fort Calhoun Station.

2.24 Manager-Nuclear Projects

The Manager-Nuclear Projects, under the Division Manager-Nuclear Engineering, has the primary responsibility for providing design drafting and configuration control support for design documents, drawings and engineering procedures associated with Fort Calhoun Station.

2.25 Manager-Corrective Action Group

The Manager-Corrective Action Group, under the Division Manager-Nuclear Support Services, has primary responsibility for administering the Condition Reporting system.

2.26 Corporate Health Physicist

The Corporate Health Physicist, under the Division Manager-Nuclear Assessments, has primary responsibility for direction and oversight of radiation protection functions at the Fort Calhoun Station.

2.27 Plant Review Committee (PRC)

A committee composed of key management personnel designated as the Plant Review Committee (PRC) acts in an advisory capacity to the Manager-Fort Calhoun Station and serves in accordance with the Technical Specifications and the Plant Standing Orders.

2.28 Safety Audit and Review Committee (SARC)

A committee composed of highly qualified and experienced OPPD management personnel and consultants, designated as the Safety Audit and Review Committee (SARC), functions to provide independent review and audit of activities in accordance with the Technical Specifications and the SARC Charter. The SARC reports to and advises the Vice President on reviews and audits of the designated activities.

3. QA PROGRAM

3.1 Corporate Policy

The Omaha Public Power District (OPPD), as the owner and operator of Fort Calhoun Station, has established a Company policy to maintain and operate the facility with due regard for public and plant safety as prescribed by various regulatory requirements. Since there is a close correlation between safety and plant quality, the control of quality is a responsibility of every individual associated with station design, procurement, modification, maintenance, and operation.

The OPPD Quality Assurance Program delineates the established policy and quality requirements, and is implemented and applied to those activities as specified therein. It establishes the OPPD Quality Assurance Plan, sets forth the quality policies, defines the requirements, and specifies responsibilities within OPPD for implementing the program. Compliance with the QA Plan, as well as with the implementing procedures developed from it, is mandatory. Management gives full support to maintaining an effective quality program. Compliance with applicable requirements of the QA Plan is made a condition of contract for supporting companies.

The Vice President has overall responsibility and authority for the implementation of the Quality Assurance Program for Fort Calhoun Station. Revisions to the QA Plan are approved by the Vice President.

The CQE/Limited CQE structures, systems, and components controlled by the QA Program are identified in the CQE list for Fort Calhoun Station. Limited CQE items include consumables such as gaskets, packing and lubricants among many other off-the-shelf items. With respect to fire protection, the QA program is applicable to the following fire protection equipment in support of nuclear safety related equipment areas: fire water supply (pumps, main piping, and valves); fire suppression systems and hose racks; fire detection and alarm systems; fire area barriers and penetrations; emergency lighting that supports fire event credited manual actions; and communications and breathing equipment intended for Fire Brigade use. The pertinent sections of the QA Plan are applied to the fire protection system to an extent consistent with safety. Therefore, Sections A.7, A.9, A.10, A.13 and A.14 of this program description are not applicable to the fire protection program.

The Quality Assurance Program applies to the procurement, maintenance, repair, and use of packaging for the transport of radioactive material. This shall include receptacles, wrappers, and their contents excluding fissile material and other radioactive material, but including absorbent material, spacing structures, thermal insulation, radiation shielding, devices for cooling and for absorbing mechanical shock, external fittings, neutron moderators, nonfissile neutron absorbers, and other supplementary equipment which has safety significance. All other activities (such as design, fabrication, assembly, and modification) are not covered by OPPD's QA Program and shall be satisfied by obtaining certifications from package suppliers that these activities were conducted in accordance with an NRC-approved QA Program. All transportation activities shall meet the requirements of 10 CFR 71 and Department of Transportation Regulations.

Any disputes which can not be resolved to the satisfaction of the Manager-Quality Assurance & Quality Control shall be brought before the Vice President via the Division Manager-Nuclear Assessments. The Vice President would then be responsible for resolving the dispute after considering all aspects of the issue.

Changes to OPPD's QA Program description are included in the update of the USAR.

3.2 QA Plan

The OPPD QA Plan requires that OPPD organizations and companies under contract to supply technical services or products for the plant comply with the following requirements:

a. The authority and duties of individuals and groups performing quality assurance functions are clearly established and delineated in writing. They have sufficient authority and organizational freedom to:

   (1) identify quality problems

   (2) recommend solutions for conditions adverse to quality

   (3) verify implementation

b. An individual or group assigned responsibility for auditing that an activity has been correctly performed, is not directly responsible for performing the specific activity.

Copies of the QA Plan are issued in a controlled manner. A distribution list is maintained showing recipients of controlled QA Plan copies. Personnel signify the receipt of their copy of the QA Plan by signing and returning a receipt acknowledgment. Recommended changes to this plan are solicited and such recommendations are given due consideration by the Manager-Quality Assurance & Quality Control and OPPD management. Necessary revisions are prepared, reviewed for adequacy, approved and issued in a controlled manner. These revisions are also controlled by means of a receipt acknowledgment. Revisions are dated and identified with formal revision numbers as they are issued.

3.3 QA Program Procedures

The QA Plan requires that the various QA Program procedures be derived from approved QA policies by means of a review of these procedures, both prior to issuance and during audits of the activity prescribed by the procedure. Procedure reviews are accomplished in accordance with established procedures.

3.4 Training and Indoctrination

Personnel responsible for performing activities affecting quality are instructed as to the purpose, scope, and implementation of the QA Plan and QA Program manuals, instructions, and procedures by participation in training programs and on-the-job training.

The QA Plan requires that personnel performing activities affecting quality possess documented evidence that they are trained and qualified in the principles and techniques of the activity being performed. Procedures provide for training and qualification in the principles and techniques of the activity being performed, including:

(1) Nondestructive evaluation (NDE) personnel administration

(2) Auditor training and qualification

(3) Indoctrination and training of quality assurance personnel

Established procedures specify the training and qualification requirements. The QA auditing and surveillance programs provide assurance that the personnel are trained in the activity.

The scope, the objective, and the method of implementing the various indoctrination and training programs are prescribed in writing and records are maintained to verify the progress and success of the programs. This documentation is audited and the programs receive periodic reviews within the applicable divisions to verify their adequacy.

The indoctrination and training programs assure that the proficiency of personnel performing activities affecting quality is maintained by specifying retraining, re-examining, and/or recertifying in accordance with the specified requirements. The indoctrination and training programs provide for documenting the training sessions, describing the content, the date held, the attendees, and the results of any examinations conducted.

4. DESIGN CONTROL

The OPPD Quality Assurance Plan provides for several levels of design control for modification. OPPD design activities meet applicable QA Plan requirements for activities affecting quality. QA audits assure that OPPD's design control measures provide a clear definition of design interfaces, review and approval of designs, including changes or revisions, and that those performing design review activities are independent of those originating the design. The verification of engineering and design adequacy of the contractors' design documents is performed in accordance with an OPPD approved Quality Assurance program and procedures.

Requirements for OPPD design development and review are contained within the QA Plan. Administrative instructions for initiating, controlling and documenting modification of station equipment and facilities is provided. The Manager-Fort Calhoun Station is responsible for reviewing and approving designs prior to their implementation at Fort Calhoun Station. Utilization of the Plant Review Committee is governed by the station Operating Manual and plant Standing Orders for their review function.

If an unreviewed safety question is involved, the design is further reviewed by the Safety Audit and Review Committee as specified in the SARC Charter prior to submittal to the NRC for approval.

Procedures require an independent review of design documents for CQE/Limited CQE designs. Procedures assure that design characteristics can be controlled, inspected, and tested. Independent design review and verification activities are required by the QA Plan to be performed in accordance with approved procedures by appropriately qualified engineers for engineering calculations, specifications, and design drawings for items within the QA Program boundary.

The QA Plan and established procedures require that selected documents be reviewed to determine that they contain, as appropriate:

a. Applicable design bases, technical requirements, regulatory requirements, component and material identification, drawings specifications, codes and industry standards, tests and inspection requirements, and special process instructions for such activities as fabrication, cleaning, erection, packaging, handling, shipping, storage, and inspection;

b. Requirements that identify the documentation to be prepared, maintained, submitted, and made available to the purchaser for review and comment, such as drawings, specifications, procedures, inspection and test records, personnel and procedure qualifications, and chemical and physical test results on materials;

c. Requirements for the retention, control, and maintenance of documents and records for activities affecting quality.

The QA Plan and established procedures require that design adequacy be verified by systematic evaluation of the elements of the design with respect to requirements for design, safety, function, and quality. Verification may be accomplished by performing design reviews, by the use of alternate or simplified calculational methods, or by conducting a suitable test program. The verifying process shall be performed by individuals or groups other than those who performed the original design.

Detailed design or design changes involving Critical Quality Elements (CQE) that are performed by the Design Engineering-Nuclear Department are performed in accordance with approved procedures. Procedures require technical calculations and safety analyses be provided by the design engineer and describe how safety analyses and technical calculations are processed. Procedures provide design controls for compatibility of materials and accessibility for inservice inspection, maintenance, and repair.

Materials, parts, equipment and processes essential to CQE/Limited CQE and the fire protection program are required to be selected and reviewed for suitability of application. The methods of assurance of suitability are required to include independent design verification by individuals or groups competent in the applicable field of design and related nuclear power plant requirements.

The methods of selection and review are required to provide for (as applicable): reactor physics, stress, thermal, hydraulic and accident analyses; compatibility of materials; as low as practicable radiation levels; accessibility for in service inspection, maintenance and repair; test requirements and delineation of acceptance criteria for inspections and tests.

The QA Plan requires that measures be established to assure that applicable fire protection program guidelines and requirements are included in design and procurement documents prepared and that deviations there from are controlled. Field changes and design deviations that affect the intent of the modification shall be subject to the same level of controls, reviews, and approvals that were applicable to the original document. Quality standards are specified in the design documents such as appropriate fire protection codes and standards. Deviations or changes from these standards are individually approved. New designs and plant modifications, including fire protection systems, are reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements. These reviews include items such as:

a. Reviews to verify adequacy of wiring isolation

b. Reviews to verify appropriate requirements for room isolation

c. Reviews to verify appropriate material is used

Materials, parts, and equipment for CQE/Limited CQE structures, systems, and components are procured in accordance with established procedures regardless of commercial or previous approval status. Procedures require an engineering and quality review of procurement documents for CQE and limited CQE items.

Procedures require that CQE/Limited CQE design changes affecting the design basis be processed through the design engineer and that an independent reviewer reviews the changes. The process is documented and retained as records.

The OPPD QA Plan requires that OPPD's manufacturers' and contractors' design activities meet applicable regulatory requirements for quality-related activities. The OPPD QA Plan requires verification that applicable regulatory requirements have been incorporated in activities affecting quality design review, audit, and surveillance of manufacturers and contractors. This assures that design input (applicable regulatory requirements and design bases as specified in the license application for safety related structures, systems, and components) for Fort Calhoun Station Unit No. 1 are correctly translated into design output documents (specifications, drawings, procedures, and instructions). QA audits assure that OPPD's manufacturers' or contractors' design control measures provide a clear definition of design interfaces, review and approval of initial design, including changes or revisions, and that those performing design review activities are independent of those originating the design.

The design activities of contractors for safety-related structures, systems, or components are required to comply with OPPD approved design development and control requirements.

5. PROCUREMENT DOCUMENT CONTROL

Appropriate requirements have been established by the OPPD Quality Assurance Plan to assure that procurement documentation is controlled and accurately reflects applicable regulatory requirements, design bases, and other appropriate requirements, such as industry codes and standards. Procurement documents and specifications require that bidders or suppliers submit for review by OPPD written quality assurance programs consistent with the importance and complexity of the materials, equipment, or service procured. Such vendors quality assurance programs are required to be consistent with pertinent provisions of Appendix B to 10CFR, Part 50, or Subpart H of 10CFR Part 71, as appropriate. OPPD satisfies these requirements as follows:

(1) Review of procurement documentation for CQE and limited CQE listed materials, equipment, and services is performed in accordance with established procedures which require OPPD personnel to review CQE and limited CQE procurement documents and document their review.

(2) Procurement documents for fire protection and radioactive material packaging materials, equipment, and services are reviewed, approved and documented by qualified personnel to verify the adequacy of fire protection and quality requirements. This review assures that fire protection requirements and quality requirements are correctly stated, inspectable and controllable; that there are adequate acceptance and rejection criteria; and that the procurement document has been properly prepared, reviewed, and approved.

(3) Procurement documents are reviewed to assure that the item is materially compatible with the environment in which it will be used and that applicable documentation is specified.

(4) Planned, periodic, and documented audits are performed by responsible OPPD personnel to provide assurance that the procurement activities of OPPD are being carried out in accordance with approved procedures.

The QA Plan and established procedures require that quality data be included in or appended to the procurement documents or engineering data attachments, as appropriate. The quality data prescribes as necessary:

(1) Quality requirements including use of procedures or instructions

(2) Requirements for a supplier quality program and documentation

(3) Requirements for documentary evidence of quality to be furnished by the supplier (e.g., test results, certification that specific requirements have been met, or traceability to the source)

(4) Access requirements for surveillance, inspection, and audits at the supplier's work site

The OPPD QA Plan requires that revisions or amendments which affect the safety or quality aspects of purchase orders, procurement documents or contracts be prepared, reviewed, and approved in the same manner as the original documents.

The procurement process for spares and replacement parts for Fort Calhoun Station, as required by the OPPD QA Plan and further delineated in established procedures, is more controlled than the original procurement process. The procurement process for FC1 occurred from 1967 to 1970; the 10CFR, Part 50, Appendix B, QA requirements were not invoked until 1971.

6. INSTRUCTIONS, PROCEDURES AND DRAWINGS

Appropriate requirements have been established in the OPPD Quality Assurance Plan to assure that activities affecting quality are prescribed by documented instructions, procedures, or drawings; that they are accomplished in accordance with such documents, and are approved only when acceptance criteria are met. The responsibility for the development of the instructions, procedures, or drawings is delegated to the organization responsible for the activity; however, the developed instructions, procedures, and drawings are subject to OPPD QA audit. The Quality Assurance Plan contains the specific requirements pertaining to the instructions, procedures, and drawings associated with activities affecting quality.

The QA Plan requires that approved changes be promptly included where applicable into instructions, procedures, and drawings associated with the change. The OPPD QA Plan requires that changes be reviewed for their effect on present instructions, procedures, and/or drawings.

The OPPD QA Plan requires that procedures include a description of the sequence of activities or operation for fabrication, processing, assembly, inspection and test. Instructions indicate the operations or processes to be performed, type of characteristics to be measured or observed, the methods of examination, the applicable acceptance criteria and documentation requirements. The QA Plan also requires establishment of those inspections, tests, and holdpoints at which time conformance of parts, components, and subsystems to requirements will be verified.

OPPD personnel review such documentation to assure that it adequately reflects applicable quality requirements. In reviewing activities, OPPD personnel assure that instructions, procedures, and drawings contain appropriate quantitative (such as dimensions, tolerances, and samples) acceptance criteria for determining that important activities have been satisfactorily accomplished.

Inspections, tests, administrative controls, fire drills, and training that govern the fire protection program are prescribed by the QA Plan and established instructions, procedures, or drawings and are accomplished in accordance with these documents. Instructions and procedures for design, installation, inspection, test, maintenance, modification and administrative controls are reviewed in accordance with the established procedures to assure the proper inclusion of fire protection requirements.

7. DOCUMENT CONTROL

The OPPD QA Plan requires that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel. These requirements also provide that contractors include, in their internal programs, measures to assure that changes to documents will be reviewed and approved by the same organization that performed the original review and approval. The OPPD QA Plan requires that changes to documents that have been reviewed and approved by OPPD organizations be reviewed and approved by the same OPPD organizations that performed the original review and approval. These requirements also provide that the documents are distributed to and used at the location where the prescribed activity is performed. The scope of these requirements apply to OPPD as well as to contractors and subcontractors.

The QA Plan requires a document control system that utilizes numbering of documents requiring control, predetermined distribution lists, and review and approval procedures. Controlled documents associated with Fort Calhoun Station have been controlled by document change transmittal letters instructing the recipient to remove and destroy obsolete or superseded pages. The QA Plan requires:

(1) maintenance of distribution lists

(2) use of receipt acknowledgments which indicate that superseded pages/documents are destroyed or marked as superseded

The Quality Assurance Plan requires that design engineering and procurement documentation, except for fire protection equipment, which consists of specifications, drawings, USAR material, instruction, procedures, reports, and changes thereto, and manufacturing and construction documents and records required for traceability, evidence of quality, and substantiation of the as-built configuration, be controlled.

Instructions, procedures, specifications, drawings, and procurement documents are controlled in accordance with the QA Plan and established division/department procedures.

A "Table of Contents" or "Index" system is used by OPPD departments to identify the current revision number of instructions, procedures, and procurement documents. The controlled copies are distributed to predetermined, responsible personnel, and a distribution list is maintained. Superseded documents are returned to the originator or destroyed as directed in the transmittal letter.

The QA Plan and established procedures identify those individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto.

8. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Measures utilized by OPPD to control purchased material, equipment, and services for an operating plant consist of individual and committee reviews, audits, and inspections. These measures are described in the OPPD Quality Assurance Plan and the established procurement procedures.

Potential manufacturers or contractors who are to be considered by OPPD or its prime contractors for the supply of items will normally be evaluated in advance of their use as an OPPD vendor. OPPD's evaluation of potential vendors is performed in accordance with established procedures. The evaluation involves the review of available historical data on manufacturers' or contractors' performance and capability; review of their quality assurance programs; or results of previous shop surveys and audits. Quality assurance program documents are required to be submitted with bids for CQE listed items. The manufacturer or contractor selected to supply the material, equipment, or services is approved by the Manager-Nuclear Procurement Services. If required, a pre-award survey at the supplier's facility is conducted before award of contract.

Documented, objective evidence such as certifications, chemical and physical analyses, inspection reports, test results, personnel and process qualification results, code stampings and nondestructive test reports are required to be evaluated by OPPD and suppliers or contractors. This verification will assure conformance to design requirements, drawings, specifications, codes, standards, regulatory requirements and other applicable criteria. These documents become a part of the quality verification records to be retained as a QA record in accordance with Section A.18.

Source inspection, when deemed necessary, is required by the applicable procurement document. The purchasing organization shall require that holdpoints be determined as necessary for this activity. Manufacturers are required to give sufficient notice of approaching holdpoints to allow scheduling of personnel.

Both in-process and final source inspections cover review of the quality verification documentation. An inspection document is used to establish the inspection sequence and for recording inspection results. This document also becomes part of the quality verification records. Provision is made for reporting deviations and nonconformances, if any; for recommending disposition and corrective action; for reinspection, if required; and for release for shipment, if appropriate. OPPD or its contractor may elect to participate in selected source inspections.

The OPPD QA Plan requires that manufacturers or contractors provide the quality verification documentation at the plant prior to the scheduled time of installation or use of the subject material and equipment. Audits will assure that the contractor is implementing a records management system. Delivered components will not be used until objective evidence of the quality verification package has been received unless there is a documented waiver.

Receiving inspection of purchased products is accomplished in accordance with established procedures. These procedures require that shipments delivered to the station be checked for shipping damage, agreement of actual count with the purchase order and packing slip, and agreement of the individual item identification with the purchase order and packing slip.

Procedures require that receiving inspection records be prepared for each purchase order requiring delivery of CQE, Limited CQE, fire protection and radioactive material packaging items to the station.

Receiving inspection records include a copy of the purchase order and material inspection records. Special instructions may be included for complex inspection requirements and tests to be performed at the plant or the supplier's work site as determined from the purchase order. Drawings and/or specification documents are included as appropriate. The inspector(s) perform(s) the receiving inspection in accordance with the above instructions and/or specifications.

The QA Plan requires that inspection records or certificates of conformance attesting to the quality of materials and equipment be submitted to OPPD for permanent retention. Such records are available for review during audits and are forwarded prior to or concurrent with material or equipment shipments to which they are related. In addition, prior to acceptance of material, the Nuclear Procurement Services Department is notified to verify that necessary documentation has been received.

Products intended for use as CQE/Limited CQE are inspected upon receipt in accordance with established procedures, which require a "nonconforming material" tag to be affixed to rejected material, and the material segregated in the receiving area to prevent inadvertent use. Accepted material is identified, and there are records traceable to the material indicating acceptance.

9. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Appropriate requirements have been established by the OPPD Quality Assurance Plan to assure accurate identification traceability and control of materials, parts, and components so that the use of incorrect or defective material, parts, or components is prevented.

Material received at the storeroom for use as CQE/Limited CQE or as packaging for radioactive material for transport are identified to prevent the use of incorrect or defective material. The identification of the item is maintained by an appropriate code, letter, or number so that the identity of the material is maintained. Items shipped to the plant are normally identified by nameplate or other identification marking on the item. In those instances when it is not practical to provide identification markings on the individual items, identification information is provided in shipping paperwork that is transmitted with each shipment.

The traceability of materials is assured through the use of established procedures. The receiving inspection records contain the documentation needed for the traceability of the item. Those documents which are not included are referenced as to their location. The method of identification to be applied to purchased materials is specified as part of the purchase document. Codes and standards referenced in the purchase document have incorporated the appropriate marking method, such that the fit, function, or quality of the item is not affected. The correct identification of materials is verified and documented prior to release.

Contractors are required to utilize procedures which establish and document a system or method of identifying the material (e.g., physical marking, tagging, labeling, color code). This system clearly indicates whether materials are acceptable or unacceptable for further use, as required by the quality program. Material traceability is provided as specifically required by applicable codes; otherwise, material identification, either on the item or on records traceable to the item, are used, as appropriate. Where identification marking of an item is employed, the marking will be clear, understandable, and legible, and applied in such a manner as not to affect the function of the item. The identification and control measures provide for relating the item of production (batch, lot, components, part) at any stage, from materials receipt through fabrication, shipment, and installation to an applicable drawing, specification, or other technical document.

OPPD requires its suppliers to establish and implement a program for inspecting, marking, identifying and documenting material prior to use or storage. This program must be documented. Holdpoints are required where inspections must be made and certified complete before start of the next operation. Inspection of materials include the following:

(1) Verification that identification and markings are in accordance with applicable codes, standards, specifications, drawings, and purchase orders.

(2) Visual examination of materials and components for physical damage or contamination.

(3) Examination of quality verification records to assure that the material received was manufactured, tested and inspected prior to shipment in accordance with applicable requirements.

(4) Actual inspection, as required, of workmanship, configuration and other characteristics.

These inspections are documented and controlled. OPPD performs surveillance of vendor facilities as necessary to assure implementation of the program.

OPPD requires that contractors establish specific measures to assure compliance with approved procedures for identification and control of materials, parts, and components, including coatings and partially fabricated assemblies. OPPD verifies conformance by one or more of the following methods:

(1) Review and approval of contractors' quality assurance programs and procedures.

(2) Surveillance of selected manufacturing, fabrication, construction and installation activities by quality assurance personnel.

(3) Auditing:

(a) of contractors for satisfactory performance of committed quality actions; and

(b) of OPPD activities for adherence to quality requirements.

10. CONTROL OF SPECIAL PROCESSES

The QA Plan requires that written procedures and controls be prepared to assure that special processes, including welding, heat treating, protective coatings, and nondestructive testing are accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. These procedures describe the operations to be performed, the sequence of operations, the characteristics involved (e.g., flow temperature, fitup, finish, hardness, and dimensions), the limits of these characteristics, process controls, measuring and testing equipment to be utilized, and documentation requirements.

Examination, tests, and inspections are conducted to verify conformance to the specified requirements.

Written procedures are required to cover training, examination, qualification, certification, and verification of personnel as well as the maintenance of required personnel records.

Compliance with these procedures is required for plant maintenance personnel, contractors, and vendors. Procedures for control of special processes are subject to review and approval by OPPD on an individual basis.

OPPD assures conformance with these requirements by:

(1) Review of procedures by the plant and QA & QC Department personnel for inclusion of special processes requiring control; definition of requirements for training, qualification and certification; conformance to applicable codes, standards, drawings, specifications, or other criteria.

(2) Audits to verify the adequacy of selected plant and vendor shop activities and the effectiveness of the special process procedures being implemented.

11. INSPECTION

OPPD will establish with its personnel and contractors a division of responsibility which will determine the services, structures, systems, components, and materials for which each is responsible. The organization having the responsibility for maintenance or repair of such items is also responsible to assure that adequate inspection is accomplished. OPPD's QA & QC Department, however, retains the responsibility and authority for review, approval, and surveillance or audit of the inspection procedures utilized by plant personnel or contractors.

OPPD QA & QC Department personnel are responsible for the inspection of work performed by OPPD maintenance personnel on nuclear safety related structures, systems, or components, and on radioactive material packaging. Quality Control personnel or other personnel who are independent of the individuals performing the activity being inspected and who are qualified in the design and installation requirements for fire protection will inspect activities affecting fire protection to verify conformance with documented installation drawings and test procedures for accomplishing the activities.

The review and approval of a contractor's inspection program and procedures is accomplished as an integral part of OPPD's review of the organization's Quality Assurance/Quality Control programs. The QA & QC Department uses the following criteria in evaluating inspection methods proposed by plant OPPD personnel or organizations under contract to OPPD:

(1) Inspection procedures for functional groups such as procurement, project engineering, construction, and shop inspectors, must be described including measures to identify inspection and test status.

(2) Duties and responsibilities of personnel performing quality activities must be clearly established.

(3) Qualifications of personnel performing quality activities must be commensurate with their duties and responsibilities.

(4) Documentation methods for inspection activities of each group must be established (e.g., inspection forms, reports).

(5) Documentation control systems for identification and distributing inspection documents must be defined.

(6) Review and approval procedures for inspection documentation must be provided.

(7) Surveillance methods must be established to assure proper implementation of inspection procedures.

(8) Planning of inspection sequence activities by plant maintenance personnel or the contractors includes the type of characteristics to be measured, the methods of examination, and the criteria. OPPD will approve inspection holdpoints in the sequence.

The Manager-Fort Calhoun Station assures that the periodic inspections made by OPPD Personnel include:

(1) Periodic inspections of fire protection systems, breathing equipment, emergency lighting, and communication equipment to assure the acceptable conditions of these items.

(2) Periodic inspections of materials subject to degradation such as fire stops, seals, and fire retardant coatings to assure that such items have not been damaged or deteriorated.

Inspection planning is utilized to assure conformance to procedures, drawings, specifications, codes, standards, and other documented instructions. Inspections are not to be performed by those individuals who performed the activity being inspected. Sufficient inspections are conducted to verify conformance particularly in areas rendered inaccessible by further processing. Process monitoring may be utilized in lieu of inspection in those cases where inspection is impossible, disadvantageous, or destructive. When required for adequate control, a combination of inspection and process monitoring is employed. Holdpoints verify (by review of inspection reports, visits to supplier shops, and plant surveillance) that inspections are being performed and documented by personnel in conformance with approved procedures.

The provisions which assure inspection is performed with the necessary drawings and specifications are covered in established procedures.

Modifications are inspected in accordance with established procedures. A plan for inspection and monitoring is developed and incorporated in planning documents of work segments, including designation of mandatory holdpoints. The inspection and monitoring plan is designed to verify conformance of work and products with the planning documents, applicable design documents, and specific quality standards and requirements. The plan provides for inspection and monitoring during critical stages in the progression of work and for inspection at the conclusion of each work segment.

Repairs and replacements are inspected in process or during receiving inspection in accordance with established procedures. Holdpoints for inspection or witnessing are specified in accordance with the plant Standing Orders.

OPPD inspectors are qualified and maintain their qualification by participation in the training and indoctrination delineated in Section A.3.4. OPPD QA & QC personnel performing nondestructive examination are trained and qualified in accordance with established procedures. Consultant and contractor inspectors performing inspection duties for OPPD are required to provide documentary evidence that they are qualified and that the certifications are current.

12. TEST CONTROL

The OPPD Quality Assurance Plan requires that OPPD personnel, contractors, and suppliers designate appropriate tests to be performed at specific stages of manufacturing, fabrication, construction, and operation. Conduct of tests are governed by written procedures which incorporate requirements and acceptance limits to assure that the structures, systems, and components tested will perform satisfactorily in service. Tests are conducted in accordance with these procedures and are properly documented.

OPPD assures that necessary tests are conducted by contractors performing maintenance or repair service for an operating plant. Such testing is performed in accordance with quality assurance and engineering test limits contained in applicable design documents. Test requirements and acceptance criteria are provided by the organization responsible for the specification of the item under test, unless otherwise designated. The entire test program covers required testing including, as appropriate, performance testing of production equipment, calibration testing of instruments, hydrostatic testing of pressure boundary components and surveillance testing.

Measures are established which assure that modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptance alternatives. Documentation of tests conducted is included in the completed design package, with the completed maintenance order by special procedure, or included in the receiving inspection packet.

Following modification, repair, or replacement, sufficient testing is performed to demonstrate that fire protection equipment in support of nuclear safety related equipment areas will perform satisfactorily in service and that design criteria are met. Written test procedures for installation tests are prepared by the responsible engineering group and incorporate the requirements and acceptance limits contained in applicable design documents.

Test procedures are evaluated for the following criteria and includes them where applicable:

(1) Requirements that prerequisites for the test have been met. Test prerequisites may include, but are not limited to, the following:

(a) calibrated instrumentation

(b) adequate and appropriate equipment

(c) trained, qualified and, as appropriate, licensed or certified personnel

(d) preparation, condition, and completeness of item to be tested

(e) suitable and, if required, controlled environmental conditions

(f) mandatory inspection holdpoints, where applicable, for witness by OPPD, contractor, or authorized inspector

(g) provisions for data collection and storage

(h) acceptance and rejection criteria

(i) methods of documenting or recording test data results

(2) Designation of specific test methods to adequately assess appropriate parameters.

(3) Designation of measuring and test equipment to be used.

(4) Specific environmental considerations.

(5) Measures to prevent damage to the item or system under test.

(6) Safety considerations.

(7) Documentation requirements.

Test results are evaluated to verify as applicable:

(1) Proper functioning of the system, structure, or component.

(2) Conformance to design specifications.

(3) Compliance with stated test requirements.

(4) That test results are within allowable limits.

(5) That recording and documentation is complete and accurate.

Audits by OPPD QA, vendor surveillance, and witness of specific tests serve to assure the functional adequacy of, and verify compliance with, the testing program.

13. CONTROL OF MEASURING AND TEST EQUIPMENT

The OPPD Quality Assurance Plan requires that organizations performing activities affecting quality involving measuring and test equipment have written procedures to govern these actions. The QA Plan requires that the standards used for calibration and accuracy verification of measuring and test equipment be traceable to the National Institute of Standards and Technology or other appropriate sources. In addition, only properly calibrated measuring and test equipment is used. A calibration program is established to which the tools, instruments, gauges, and other devices shall conform. Records of calibrations are maintained and the calibration equipment appropriately marked to indicate the date and acceptance of the calibration. Calibration activities being performed by OPPD personnel are in accordance with Standing Orders or other procedures. If a standards error exceeds the guaranteed accuracy, then the standard is replaced. Calibration standards are procedurally controlled to guarantee accuracy ratios consistent with industry standards.

When inspection and testing equipment is found to be out of calibration due to use or damage, or when out of limits at recalibration, items inspected, tested, or measured with that equipment since the latest valid calibration are considered as being potentially unacceptable. Resolution of these cases is determined on a case basis.

14. HANDLING, STORAGE AND SHIPPING

OPPD's QA Plan requires that instructions or guidance for plant handling, preservation, storage, and control of products are prepared and approved prior to arrival of the products at the plant. These procedures specify, as required, that special environmental facilities, such as inert gas, humidity control, or temperature controlled storage area are established prior to the receipt of the products. Contractors performing maintenance or repair services are required to provide procedures for the handling of products to prevent damage or deterioration. The procedures are reviewed and approved by OPPD.

To assure existence of the requirements for procedures in the procurement documents, OPPD verifies the inclusion during its review prior to authorization for document issuance. OPPD personnel procure, receive, store and handle CQE/Limited CQE and radioactive material packaging products, material, and components in accordance with established procedures.

15. INSPECTION, TEST AND OPERATING STATUS

OPPD's QA Plan requires that procedures be established to identify the inspection, test, and operating status of radioactive material packaging and safety-related structures, systems, and components. Identification of the inspection, test, and operating status of structures, systems, and components is provided in the surveillance test program. Schedules and methods for periodic testing of fire protection systems and components have been developed and documented by the Manager-Fort Calhoun Station. Fire protection equipment in support of nuclear safety related equipment areas is tested periodically to assure that the equipment will properly function and continue to meet the design criteria. Test results are documented, evaluated, and reviewed for acceptability.

The application and removal of inspection and welding stamps and status indicators are procedurally controlled, and nonconforming, inoperative, or malfunctioning structures, systems, or components are identified in accordance with established procedures. System completeness and acceptance at the end of a maintenance or repair phase are determined by:

(1) reviewing for adequacy, completeness, and conformance to quality assurance requirements for each system or component being accepted;

(2) performing surveillance and monitoring of the test activities associated with the approved test program;

(3) reviewing the test records to verify that test results comply with established requirements.

The suppliers' and contractors' inspection and test status of items are required to be maintained through the use of status indicators such as physical location, tags, markings, shop travelers, stamps, or inspection records. These measures provide for assuring that only items that have received the required inspections and tests are used in manufacturing and are released for shipment. The procedures for control of status indicators, including the authority for application and removal of tags, markings, labels or stamps are documented in approved manufacturing or quality assurance procedures.

16. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

The OPPD Quality Assurance Plan requires that measures be established to control the identification, documentation, segregation, and disposition of nonconforming material, parts, or components. The implementing instructions which fulfill these requirements are prescribed in established procedures. The QA Plan identifies those individuals or groups delegated the responsibility and authority for the disposition and approval of nonconforming items. Nonconforming items are controlled and identified in accordance with written procedures to prevent inadvertent use or installation. Control measures include tagging or marking and segregation when feasible. Control measures are maintained until the item has been removed from the plant site or corrective work has been completed and accepted.

Established procedures cover:

(1) Initiation of the documentation for material rejected at receiving inspection or in-plant activities including the tagging of nonconforming items. The documentation identifies the nonconforming item and describes the nonconformance.

(2) Assignment of disposition and/or corrective action responsibilities, including the inspection requirements and signature approval of the disposition.

(3) Control of correction work planning and acceptance.

Nonconformance reports are analyzed to detect adverse quality trends. Trending of nonconformances is conducted in accordance with procedures.

The OPPD Quality Assurance Plan requires that measures be taken and documented by contractors and suppliers to control the identification, documentation, segregation, and disposition of nonconforming material, parts, or components. These measures prevent inadvertent use or installation of defective components and are subject to review and approval by OPPD. Written procedures will be required for investigation of the nonconforming item, decisions on its disposition, and preparation of adequate reports. Procedures also control further processing, fabrication, delivery, or installation of items for which disposition is pending. Reports documenting actions taken on nonconforming items are made available to OPPD for evaluation. Departures from design specifications and drawing requirements that are dispositioned "use as-is" and "repair" are reported to affected organizations and OPPD management.

The effectiveness of nonconformance control procedures is assured by:

(1) Contractor quality assurance and manufacturing, fabrication, or construction personnel being involved in processing nonconforming reports.

(2) OPPD participation in dispositions and approvals.

(3) Document review at final inspection or shipping release and at receiving inspection by OPPD.

(4) Audits and/or Surveillances by OPPD and contract personnel.

17. CORRECTIVE ACTION

The QA Plan requires that measures be established to assure that conditions adverse to quality are promptly identified, reported, and corrected. Responsibility for performing corrective action is assigned to OPPD personnel and contractors and suppliers so that each will be alert to those conditions adverse to quality within his own area of responsibility. In the case of significant conditions adverse to quality, measures are taken to assure that the cause of the condition is determined and corrective action is implemented to preclude repetition. Corrective action procedures require thorough investigation and documentation of significant conditions adverse to quality. The cause and corrective action are reported in writing to the appropriate levels of management. The corrective action to be applied is subject to review and approval by the PRC. Corrective action followup and closeout procedures provide that corrective action commitments are implemented in a systematic and timely manner and are effective.

The effectiveness of the suppliers' or contractors' corrective action program is assessed during audits by the supplier, the contractor, and by OPPD. Stop work authority is exercised as required.

A quarterly report of internal deficiencies that have occurred is prepared and distributed to the management of OPPD organizations participating in the QA program.

18. QUALITY ASSURANCE RECORDS

OPPD's Quality Assurance Plan requires that OPPD and its contractors have a quality records system which provides documentary evidence of the performance of activities affecting quality. The requirements include that:

(1) Records are maintained that show evidence of performance of activities affecting quality. Typical records to be maintained include quality assurance programs and plans, design data and studies, design review reports, specification procurement documents, procedures, inspection and test reports, material certifications, personnel certifications, test reports, audit reports, reports of nonconformances and corrective actions, as-built drawings, operating logs, calibration history, maintenance data, and failure and incident reports.

(2) Inspection and test records, as a minimum, identify the date of the inspection or test, the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any nonconformances noted.

(3) Records are protected against deterioration and damage.

(4) Criteria are established for determining the classification of the record as well as the length of the retention period.

(5) A method of identification and indexing of records for ease of retrievability is established.

(6) Responsibility for record keeping during design, fabrication, construction, preoperational testing, and commercial operation is documented.

(7) Method of transfer of records between organizations and ultimate transfer to OPPD shall be established.

Requirements and responsibilities for the handling, storage, and retention of records which furnish documentary evidence of quality are prescribed by established procedures. The records are accumulated and handled in a controlled manner in accordance with these procedures.

19. AUDITS

The OPPD QA Plan requires that planned and periodic audits be performed to verify compliance of activities affecting quality and to determine the effectiveness of the QA program. OPPD QA and Nuclear Procurement Services personnel perform such performance based audits on OPPD internal activities, contractors, suppliers, and others as necessary to provide an objective evaluation of the effectiveness of their programs; to determine that their programs are in compliance with established requirements, methods, and procedures; and to verify implementation of recommended corrective action.

The Safety Audit and Review Committee (SARC) provides independent review and audit of activities as delineated in the Fort Calhoun Station Technical Specifications.

The internal audit cycle for activities affecting quality and the fire protection program is promulgated in establishing procedures and is based on the safety importance of the activities being performed. An audit schedule is distributed on a calendar year basis and is updated as necessary to ensure coverage of status changes. If, in the opinion of the Manager-Quality Assurance & Quality Control, a given area requires added emphasis, the frequency of audits is increased until the situation is clarified.

The OPPD audits, both internal and external, are conducted in accordance with the established procedures. Consultants may be utilized by OPPD on audits as required. The QA Plan specifies that the auditing system used by OPPD, its contractors, and suppliers:

(1) utilizes an audit planning document which defines the organizations and activities to be audited and the frequency of audits;

(2) requires auditors to be familiar with the type of activities to be audited and have no direct responsibilities in the type of activities to be audited;

(3) provides auditing checklists or other objective guidelines to identify those activities which affect quality;

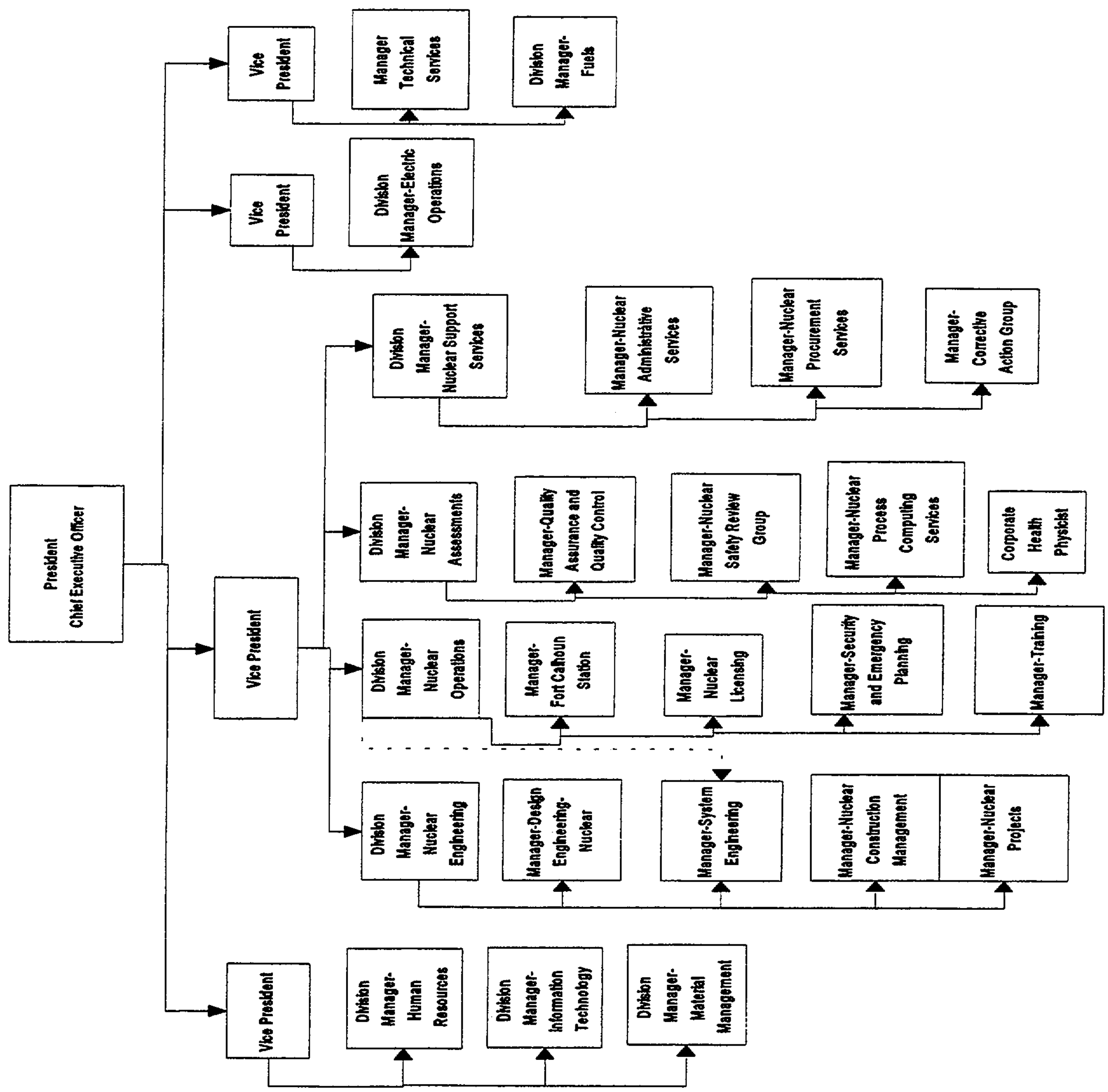
(4) requires examination of the essential characteristics of the quality activity examined;

(5) requires an audit report to be prepared and that it notes the extent of examination and deficiencies found.

Established procedures provide the means which assure that audits are performed in a thorough and professional manner. OPPD audits determine the existence of a control system and the deficiencies of that system, and the actual practice of the system. Audit checklists are used to ensure that audits include the objective evaluation of work areas, activities, processes and items and the review of documents and records.

Established procedures require that a formal report be prepared upon completion of each audit. The audit report identifies any deficiencies or nonconformances found during the audit, and recommended solutions.

Figure A-1- "Nuclear Quality Assurance Program Organization for Fort Calhoun Station"



Attachment 1 - "Industry Standards and Associated Regulatory Guides Used as a Basis for the OPPD QA Program"

Introduction

This attachment identifies the specific industry standards and NRC Regulatory Guides which form the base of OPPD's QA Program as described in the USAR Appendix, and as delineated in the QA Plan. A position statement is provided for each standard/Regulatory Guide which describes the nature and extent of OPPD's commitments, including any alternatives used or exceptions taken. OPPD interprets the verbs (shall, should and may) used in industry standards to mean the following:

- Shall: Indicates a requirement.
- Should: Indicates a recommendation.
- May: Indicates permission or an option.

Where the below listed standards make reference to other documents to be included as a part of the referencing standard, it is assumed that the reference is to the specific standard(s) identified and described in this attachment, or as defined in other applicable District commitment documents. In the development of the QA Plan, requirements of construction based standards were incorporated to the extent that they are applicable to operations phase activities, retaining the basic quality assurance controls, but not necessarily encompassing the specific implementation associated with that control or measure delineated for the construction phase.

Commitments

A. Standard: ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"

   Regulatory Guide: RG 1.28, Revision 2, "Quality Assurance Program Requirements (Design and Construction)"

   Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of Fort Calhoun Station.

B. Standard: ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.33, Revision 2, "Quality Assurance Program Requirements (Operations)"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. Preoperational testing of Fort Calhoun Station is completed, therefore, these requirements are implemented only to the extent required by the Station Technical Specifications and as applicable to preoperational testing associated with station modification activities and post-maintenance testing.

2. OPPD's audit program is a four-tiered program consisting of (1) regularly scheduled internal and external audits conducted on a 3 year cycle by the Quality Assurance and Quality Control and Nuclear Procurement Services Departments, (2) regularly scheduled QA surveillances conducted by the Quality Assurance & Quality Control Department, (3) scheduled audits performed by, or under the cognizance of, the Safety Audit and Review Committee in accordance with the Station Technical Specifications, and (4) a management review to determine the adequacy and effectiveness of OPPD's Quality Assurance Program performed under the auspices of the Safety Audit and Review Committee.

3. Audits performed under the cognizance of the Safety Audit and Review Committee are conducted at the following frequencies:

   a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per two years.

   b. The training and qualifications of the facility staff at least once per two years.

   c. Actions taken to correct deficiencies occurring in facility equipment, structures, systems, components or method of operation that affect nuclear safety at least once per two years.

   d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per two years.

   e. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidifications of radioactive waste at least once per two years.

f. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

g. An inspection and audit of the fire protection and loss prevention program by an outside qualified fire consultant shall be performed at intervals no greater than three years.

4. OPPD uses a dynamic procedure review process instead of a static two year review cycle to prevent the use of outdated or inappropriate documents. This dynamic process ensures applicable procedures and instructions are reviewed for possible revision upon the identification of new or revised source material.

5. Written procedures and administrative policies affecting Fort Calhoun Station are also controlled by requirements contained in the Administrative Controls section of the FCS Technical Specifications.

C. Standard: ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.37, Revision 0, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of the Fort Calhoun Station as delineated in OPPD's QA Plan.

D. Standard: ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)"

Regulatory Guide: RG 1.38, Revision 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of Fort Calhoun Station.

E. Standard: ANSI N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.39, Revision 2, "Housekeeping Requirements for Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operation phase of the Fort Calhoun Station. The Fort Calhoun Station is divided into zones for security, storage, fire protection, and radiation protection and since zones designated by ANSI 45.2.3 are primarily designed for control of units during construction, additional zones have not been established for housekeeping purposes. Limitations are applied on eating, drinking, and smoking in specified areas at the Fort Calhoun Station. Housekeeping practices and controls have been established for the control of activities that can affect the quality of CQE related systems and components.

F. Standard: ANSI N45.2.4-1972, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

Regulatory Guide: RG 1.30, Revision 0, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

G. Standard: ANSI N45.2.5-1974, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.94, Revision 1, "Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

H. Standard: ANSI N45.2.6-1978, "Qualifications of Inspection Examination, and Testing Personnel for Nuclear Power Plants"

Regulatory Guide: RG 1.58, Revision 1, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. OPPD and contractor inspectors performing Quality Control inspections are certified in accordance with this standard; however, these certification requirements are not applied to personnel performing operational surveillance testing and inspection in accordance with the Technical Specifications, to investigative inspections or to the conduct of preliminary inspections for purpose of planning corrective or improvement actions, or to the surveillance of plant operations to verify compliance with procedures. Certification of inspectors for nondestructive examinations is accomplished in accordance with SNT-TC-1A guidelines.

I. Standard: ANSI N45.2.8-1975, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.116, Revision 0, "Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

J. Standard: ANSI N45.2.9-1974, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants"

Regulatory Guide: RG 1.88, Revision 2, "Collection, Storage, and Maintenance of Nuclear Power Quality Assurance Records"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. The Fort Calhoun Station File Room meets the criteria of NUREG-0800, Standard Review Plan, Part 17.1, Acceptance Criteria 17.4, Alternative (3); ANSI N45.2.9-1979; NFPA 232; and will withstand a maximum wind velocity of 110 miles per hour.

2. Fire rated file cabinets used for interim record storage meet a one hour or greater fire rating.

K. Standard: ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"

Regulatory Guide: RG 1.74, Revision 0, "Quality Assurance Terms and Definitions"

Position: OPPD's QA Program and QA Plan comply with the general terms and definitions of this standard.

L. Standard: ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

Regulatory Guide: RG 1.64, Revision 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety performed in the operations phase of Fort Calhoun Station. Additionally, in unique circumstances which occur in a detailed specialty field (such as reactor physics, seismic, stress analysis, etc.) where the only technically qualified individual within the licensee's organization available to perform analysis verification is the immediate supervisor, such review will be allowed when:

1. Justification allowing the review is documented and approved in advance by the Division Manager - Engineering and Operations Support and

2. All other Regulatory Guide 1.64, Revision 2, independence criteria are met.

M. Standard: ANSI N45.2.12-1977, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants"

Regulatory Guide: RG 1.144, Revision 1, "Auditing of Quality Assurance Programs for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety performed in the operations phase of Fort Calhoun Station"

N. Standard: ANSI N45.2.13-1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

Regulatory Guide: RG 1.123, Revision 1, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety performed in the operations phase of Fort Calhoun Station.

O. Standard: ANSI N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Regulatory Guide: RG 1.146, Revision 0, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide.

P. Standard: ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel"

Regulatory Guide: RG 1.8, Revision 1, "Personnel Selection and Training"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. Qualification requirements for the Supervisor-Radiation Protection and for the Shift Technical Advisor are in accordance with the Fort Calhoun Station Technical Specifications.

2. Qualification requirements for the Manager-Operations, as described in this standard, shall be met by the FCS Supervisor-Operations.

Q. Standard: ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

Regulatory Guide: RG 1.54, Revision 0, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

NT

APPENDIX F

CLASSIFICATION OF STRUCTURES AND EQUIPMENT AND SEISMIC CRITERIA

Table of Contents

List of Tables

List of Figures

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

1. CLASSIFICATION OF STRUCTURES AND EQUIPMENT

1.1 Definition of Classes

Structures and components including instruments and controls designated as Class I as specifically defined in this section (not to be confused with ASME Class I) are those whose failure might cause or increase the severity of an accident which could result in an uncontrolled release of radioactivity. Components and structures vital to safe shutdown and isolation of the reactor are also included in the Class I classification. All other structures and components are classified as Class II.

1.2 Classification of Buildings and Structures

Class I buildings and structures are listed below. Buildings and structures not listed are Class II; these contain conventional facilities.

a. Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures);

b. Auxiliary building (including the control room, spent fuel storage pool, safety injection and refueling water storage tank and emergency diesel-generator rooms);

c. Intake structure.

1.3 Classification of Systems and Equipment

Systems and equipment designated as Class I are listed below. Where necessary the description is amplified to indicate the Class I items. All supports, hangers, etc., associated with Class I equipment are also to Class I standards. Systems, equipment and other items not listed are Class II.

a. Engineered Safety Features systems;

1. Chemical and volume control system (safety related components and piping only)

2. Safety injection system

3. Containment spray system

4. Containment air recirculation, cooling and iodine removal system.

5. Auxiliary feedwater system (excluding portions of the system outside the auxiliary building; excluding portions of the alternate auxiliary flow path in auxiliary building Room 81 downstream of valve FW-746 and HCV-1384; and excluding the emergency feedwater storage tank fill line)

6. Containment isolation

b. Essential Auxiliary Support systems;

1. Component cooling water system

2. Raw water system

3. Control room HVAC system

4. Auxiliary building HVAC system

5. Emergency power - Diesel generators (including starting air, fuel oil transfer and storage), station batteries

6. Normal station electrical power - switch gear, control boards, control centers, bus ducts, and cables required for Class I systems and equipment.

c. Engineered Safeguards Controls and Instrumentation;

1. Reactor control and protective system (excluding in-core instrumentation)

2. Instruments and control devices required for Class I systems and equipment

d. Other Systems and Equipment

1. Reactor (including vessel, internals, fuel assemblies, CEA's and CEDM's)

2. Reactor coolant system

3. Main steam and feedwater piping (in containment and auxiliary building up to containment isolation valves)

4. Shutdown cooling system

5. Spent fuel pool cooling system

6. Portions of the radioactive waste disposal system required for containment and waste gas isolation

7. Fuel handling equipment

8. Primary water storage tank

9. Fire protection system (pump house only)

10. Nuclear detector well cooling system

11. Containment crane and refueling crane

12. Radiation monitoring system in containment and auxiliary building

13. Containment purge system

2. SEISMIC CRITERIA, ANALYSIS AND INSTRUMENTATION

2.1 Class I Seismic Criteria

2.1.1 Design and Maximum Hypothetical Earthquakes

The following criteria was applied to components, structures and equipment for the design earthquake and maximum hypothetical earthquake.

Design Earthquake

All Class I components, systems and structures are designed so that the seismic stresses resulting from the response to a ground acceleration of 0.08g acting in the horizontal direction and two-thirds of 0.08g acting in the vertical direction simultaneously, in combination with the primary steady state stresses, are maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards; e.g., the ASME Boiler and Pressure Vessel Code, USAS B31.1 (1967), USAS B31.1 (1955) for Reactor Coolant loop piping, and B31.7 (1968) Codes for Pressure Piping, ACI 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

Maximum Hypothetical Earthquake

All Class I components, systems and structures are designed so that seismic stresses resulting from the response to a ground acceleration of 0.17g acting in the horizontal direction and two-thirds of 0.17g acting in the vertical direction simultaneously, in combination with the primary steady state stresses, are limited so that the function of the component, system or structure is not impaired in such a manner that a safe and orderly shutdown of the plant is prevented.

2.1.2 Stress and Deformation Criteria

The design loading and stress criteria for the reactor coolant system are presented in Section 4.2.5.3. The stress and deformation criteria for other Class I piping systems, vessels and supports for the various design load combinations are presented in Table F-1.

Stress and deformation criteria for seismic Class I HVAC systems are available in the Alternate Seismic Criteria and Methodologies (ASCM) (briefly described in Section F.2.2.3) which may be used, provided the analysis is performed in accordance with the caveats, requirements, and methods identified therein. (Reference 6)

For Class I systems the necessary restraints or energy absorbing devices employed to limit deformations during the maximum hypothetical earthquake are such that stresses will not cause rupture. The natural frequency of each system was determined analytically to ensure proper positioning of these restraints or energy absorbing devices.

Class I equipment essential to a safe shutdown is capable of functioning during and following a maximum hypothetical earthquake. Analyses were made to provide assurance that elements would not come into contact because of displacements occurring during the seismic disturbance. Where necessary, clearances were increased accordingly.

GIP-3 (Reference 13) may be used as an a alternative method for showing that systems and equipment will not be adversely affected by potential seismic interactions with nearby equipment and structures. See Section F.2.2.2 for a description of the caveats and requirements for this alternative seismic qualification method.

Table F-1 - "Loading Combinations and Primary Stress Limits"

Loading Combinations	Primary Stress Limits: Vessels	Primary Stress Limits: Piping	Primary Stress Limits: Supports
1. Design Loading + Design Earthquake	$P_M \leq S_M$ $P_B + P_L \leq 1.5S_M$	$P_M \leq 1.2S_h$ $P_B + P_M \leq 1.2S_h$	Working Stress Anchor Bolts F.S.$\geq$4.0 (d)
2. Normal Operating Loadings + Maximum Hypothetical Earthquake + (Fluid Transient Loadings (d))	$P_M \leq S_D$ $P_B \leq 1.5 \left[1-\left(\frac{P_M}{S_D}\right)^2 \right] S_D$ (b)	$P_M \leq S_D$ $P_B \leq \frac{4}{\pi} S_D \cos \frac{\pi}{2} \cdot \frac{P_M}{S_D}$ (c)	Within Yield Anchor Bolts F.S.$\geq$ 2.0 (d)
3. Normal Operating Loadings + Pipe Rupture + Maximum Hypothetical Earthquake	$P_M \leq S_L$ $P_B \leq 1.5 \left[1-\left(\frac{P_M}{S_L}\right)^2 \right] S_L$ (b)	$P_M \leq S_L$ $P_B \leq \frac{4}{\pi} S_L \cos \frac{\pi}{2} \cdot \frac{P_M}{S_L}$ (a), (c)	Deflection of supports limited to maintain supported equipment within limits shown

Table F-1 (Continued)

NOTES:

(a) These stress criteria are not applied to a piping run within which a pipe break is considered to have occurred.

(b) Loading combinations 2 and 3, stress limits for vessels, are also used in evaluating the effects of local loads imposed on vessels and/or piping, with the symbol P_M changed to P_L.

(c) The tabulated limits for piping are based on a minimum "shape factor". These limits are modified to incorporate the shape factor of the particular piping being analyzed.

(d) These load cases and limits apply only to the Pressurizer relief valve piping and supports.

Table F-1 (Continued)

Legend

P_M = Calculated Primary Membrane Stress

P_B = Calculated Primary Bending Stress

P_L = Calculated Primary Local Membrane Stress

S_M = Tabulated Allowable Stress Limit at Temperature from ASME Boiler and Pressure Vessel Code, Section III.

S_Y = Tabulated Yield at Temperature, ASME Boiler and Pressure Vessel Code, Section III

S_h = Tabulated Value at Temperature from USAS B31.7

S_D = Design Stress

= S_Y (for ferritic steels)

= $1.2S_M$ (for austenitic steels)

$S_L = S_Y + \frac{1}{3}(S_u - S_Y)$

S_u = Tensile Strength of Material at Temperature

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lbs/sq.in.

Material	S_Y (1)	S_u	S_D	S_L
A-106B	25.4	60.0 (2)	25.4	36.9
SA-533B	41.4	80.0 (2)	41.4	54.3
304 SS	17.0	54.0 (3)	18.35	29.3
316 SS	18.5	58.2 (3)	22.2	31.7

(1) From ASME Boiler and Pressure Vessel Code, Section III, 1968 Revision, at 650°F.

(2) Minimum value at room temperature which is approximately the same at 650°F for ferritic materials.

(3) Estimated.

Piping runs are designed with sufficient flexibility to accept differential movement between structures without exceeding the allowable stress criteria presented in Table F-1. However, the containment and auxiliary building are on a common mat and, therefore, movement between these structures is not significant in contributing to piping stress levels. Estimates of these displacements are available in Reference 6. In addition, seismic anchor motion (SAM) displacements between the Auxiliary Building and the Turbine Building, which are significant were calculated. These displacements can be used for analysis of piping, such as Main Steam and Main Feedwater, which pass between the two structures. (Reference 6)

2.1.3 Damping Factors

Damping factors used in the design of Class I components and structures are shown in Table F-2. Alternatively, the damping factors of Reference 6 may be used, provided the analysis is performed in accordance with the caveats, requirements, and methods described for the ASCM. Another alternative which may be used for mechanical and electrical equipment is to use the damping factors in GIP-3 (Reference 13), provided the evaluation is performed in accordance with the caveats, requirements and methods described in Reference 11. See Section F.2.2.2 for a discussion of the use and limitations of GIP-3.

Table F-2 - "Damping Factors"

Component or Structure	Percent Damping: Design Earthquake	Percent Damping: Maximum Hypothetical Earthquake
Containment Structure	2.0	2.0
Concrete Support Structures for Reactor Vessel and Steam Generators	2.0	2.0
Steel Assemblies		
Bolted or Riveted	2.0	2.0
Welded	1.0	1.0
Vital Piping Systems	0.5	0.5
Rigid Vault Type Concrete Structures	2.0	5.0
Framed Concrete Structures	5.0	7.0

2.1.4 Response Curves

Response curves are shown in Figures F-1 and F-2 for the design and maximum hypothetical earthquakes respectively and were used for the design of Class I components and structures.

The response spectrum concept provides a conservative approach which has been found generally to be satisfactory for other sites with similar sub-surface conditions. The spectra conform to the average spectra developed by Housner (and presented in Reference 1) for frequencies higher than about 0.33 cycles per second. The spectra for frequencies lower than about 0.33 cycles per second were prepared utilizing data presented by Newmark (Reference 2).

The spectra have been 'normalized' to a horizontal ground acceleration of eight percent of gravity for the design earthquake and seventeen percent of gravity for the maximum hypothetical earthquake.

2.1.5 Refueling Equipment

All refueling equipment (including refueling pool crane) is designed for a seismic loading of .09g vertical and .19g horizontal applied simultaneously. The stress under the combined deadweight live and seismic loads will not exceed the allowable stress of the material. Furthermore, the equipment will withstand a simultaneous vertical acceleration of .13g and a horizontal acceleration of .27g in conjunction with normal loads without exceeding material minimum yield stresses. Guide rollers restrict lateral movement of the refueling machine and the spent fuel handling machine on their rails and have been designed for seismic loads in excess of the above values. In addition, because of its high center of gravity, the spent fuel handling machine is provided with keepers to prevent overturning under seismic shock conditions.

2.2 Methods of Analysis for Class I Structures and Components

2.2.1 General

The following methods of analysis were applied to Class I structures, systems and equipment:

a. The natural frequency of vibration of the structure or component was determined.

b. The response acceleration of the component to the seismic motion was taken from the response spectrum curve at the appropriate natural frequency and damping factor.

c. Stresses resulting from the combined influence of normal loads and the additional load from the design earthquake were calculated and checked against the limits imposed by the design standard.

d. Stresses and deflections resulting from the combined influence of normal loads and the additional loads from the maximum hypothetical earthquake were calculated and checked to verify that deflections would not prevent functional performance and that stresses would not produce rupture.

2.2.2 Structures or Equipment Supported In or On Other Structures

Structures or equipment supported in or on other structures are classified into three groups based on their natural frequency and the frequency of the supporting structure as follows:

a. Rigid Group: $f_n/f > 2.0$
b. Resonance Group: $0.7 \leq f_n/f \leq 2.0$
c. Flexible Group: $f_n/f < 0.7$

Where: f_n is the natural frequency of the structure or equipment, and f is the frequency of the supporting structure in the corresponding location.

Rigid Group

The analysis of equipment in the rigid group was based on applying a static load at its center of gravity. The load assumed was equal to the mass of the equipment multiplied by the maximum acceleration at the floor level on which the equipment was mounted.

Resonance Group

If a structure or item of equipment was found to fall in the resonance group, the supporting structure or equipment was modified to alter the frequency and prevent resonance wherever possible. If the resonance condition could not be avoided the possibility of large amplitudes is avoided by the use of stops or damping devices. Dynamic design considering resonance vibration was performed when this was not possible. If the restriction of vibration was such as to make the element rigid, an examination assuming rigid behavior was also carried out.

Flexible Group

Flexible elements were designed using induced accelerations corresponding with their frequencies. A careful check was made for elements which could come into contact because of excessive displacements.

Piping and Equipment

The B31.7 piping design to be analyzed for seismic effects has been previously defined by:

a. Pipe routing based on flow diagrams, equipment arrangement, accessibility, radioactivity considerations, drainage, venting, and other considerations.

b. Preliminary thermal analysis for hot lines confirmed the routing. Preliminary design was in accordance with B31.1.

c. Pipe thickness based on B31.1. This selection usually allows a margin over required thickness according to B31.7.

d. Piping details based on the appropriate code. For example, B31.7 Class I does not permit branches to intersect at angles less than 60 degrees.

e. Hanger locations were determined in accordance with the appropriate codes.

The first step in seismic analysis of piping was to position seismic restraints closely enough to ensure that the natural frequency of piping in the auxiliary building and containment building was 6 hertz horizontally and 18 hertz in the vertical direction. A proprietary nomogram was developed to calculate the required restraints spacing. The weights of valves were also taken into consideration in spacing of the restraints. This ensures that the piping is in the rigid range when compared to the building in which it is situated. The response of the building to the seismic ground motion was determined as discussed in Section F.2.2.3, the resulting accelerations of which are shown on Figure F-4. Values based on the ASCM are available and can be used in analyses which adopt the ASCM as design basis (Reference 6). To simplify the piping analysis, the maximum accelerations obtained, those of the uppermost level of the structure, were used in all subsequent calculations. These accelerations are considered to act at the anchors and seismic restraints of the piping. In order to obtain the response of the piping between restraints, the building accelerations were multiplied by a magnification factor giving the design acceleration of the piping. The magnification factor was derived from the response spectra of Figures F-1 and F-2 normalized for the maximum building acceleration and using 0.5% damping and a natural frequency of 6 hertz horizontally and 18 hertz vertically.

The design thus determined was put into a proprietary computer program developed by the A.D. Little Company and modified to suit Gibbs & Hill requirements. This code was entitled ADL pipe. It calculated stresses in accordance with equations (9), (10) and (11) of B31.7. The following comments apply to ADL pipe as used by Gibbs & Hill:

a. The final combination of stresses was done manually.

b. Stresses are combined at the stress level, rather than moment level as permitted by B31.7.

c. Seismic displacement of support points was considered in the following cases.

    1. Part of the pipe was supported by the containment shell and part by the containment internal structure.

    2. Part of this pipe was supported by the containment shell and part by the auxiliary building structure.

In all other cases, piping support points were considered fixed with respect to one another.

d. Movement of pipe supports were considered when necessary. For example, the movement of the main steam line supports due to containment post tensioning was taken into account.

e. The program did not include cyclic loading or thermal gradient thru the wall as required in equations 10 and 11.

B31.1 piping in the auxiliary building received the same attention with regard to selection of hangers and restraints as the B31.7 piping. Seismic stresses were combined with longitudinal stress due to pressure, weight and other sustained loads and limited to S_h (allowable stress in hot condition).

The reactor coolant loop piping is B31.1. The adequacy of the design of the components of the Reactor Coolant System to sustain the loadings which result from seismic conditions was confirmed by the methods of dynamic analysis. The analysis employed a multi-mass, lumped parameter, model of the Reactor Coolant System loops and normal mode theory as described in TID-7024.

Starting in 1979, in response to NRC IE Bulletins numbered 79-02, 79-04, 79-07, 79-14, and 81-01 and NRC Generic Letter No. 81-14, a complete seismic verification of all Critical Quality Element (CQE) piping systems, as defined in the Fort Calhoun Station's CQE list (except those portions of the system that were inaccessible), was initiated. The initial part of the verification process consisted of identifying where adequate documentation existed to demonstrate the seismic qualification of each CQE system. Where adequate documentation did not exist, those systems 2-1/2" in diameter and larger were reanalyzed using the stress criteria identified in Section F.2.1 with a newer, more sophisticated dynamic loading model than previously described. The net result of the reverification program has been that in some cases, piping systems were determined to be in the resonance group. Piping in the resonant group was reviewed to determine if excessive displacement or stress was present. If stresses were within allowables and no interference due to the deflection is created the piping is adequate. The piping that did not meet this criteria was modified by use of stops and dampers to adequately prevent large amplitude deflections or an attempt was made to increase the systems natural frequency to 6 hz horizontal and 18 hz vertical by adding additional restraints.

Special seismic restraints (either rigid or snubbers) were provided on control valve mechanisms to prevent overstress when the control mechanism forms a mass center outside the pipe center line and generates over 1500 psi bending stress on the piping system due to earthquake G loading.

The waste disposal tanks are fitted with horizontal restraints at the upper and lower ends, to avoid overloading the equipment support under seismic loading.

Special seismic restraints were installed on the electrical cable trays. The cable trays were supported vertically and horizontally so as to meet the stress criteria under all conditions including the postulated earthquakes. Spacing of vertical supports is sufficiently close to maintain the lowest vertical dominant natural frequency of the cable trays above 18 hz., which is double the dominant natural frequency of the building. Where multiple tray arrangements make determination of the natural frequencies unfeasible the system has been analyzed for a resonant condition, using a time-history approach. The cable trays are braced horizontally at a spacing to ensure a minimum natural frequency of 6 hz., double the dominant horizontal natural frequency of the building. The stresses due to the resulting seismic response are maintained within the allowable limits.

In order to avoid resonance between any of the elements of the reactor coolant system and the reactor building, additional seismic restraints are installed at the steam generators and at the reactor coolant pumps. Four columns are installed at each steam generator between the underside of the support frame and the foundation mat. These columns are pinned top and bottom to permit thermal movement of the generators. Similar columns are provided under the three support pads of each reactor coolant pump. In addition, horizontal hydraulic snubbers are installed at approximately the vertical center of gravity of each reactor coolant pump. This system of restraints stiffens the entire reactor coolant system and moves the lowest dominant natural frequency of the steam generators and reactor coolant pumps away from the peak of the applicable floor response spectrum. Resulting stresses in all components of the reactor coolant system are within allowable limits.

In order to assure proper functioning of the containment air handling units under all design conditions including maximum hypothetical earthquake, seismic response and resulting stresses were determined. In-place testing of units was performed to evaluate natural frequencies and determine where stiffening was required. Dynamic analyses were then performed to the extent necessary.

The components of the Reactor Protective System and the Engineered Safeguards Actuation System were seismically qualified prior to operation. Qualification was done by test and/or analysis. In either case, qualification was performed in accordance with IEEE 344 "Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations".

Tests on representative samples, as well as analytical methods, were chosen for the qualification of racks, panels, cabinets, or other passive structures. Test methods were chosen for equipment which is required to function during and after the occurrence of the DBE. During seismic testing, the operation of the equipment is monitored by suitable means.

Analysis of racks, panels, and cabinets performed so far indicate that the natural frequency of these structures are substantially higher than those at which the peaks in the response spectra occur. Therefore, these supports are assumed to be rigid and no amplification was considered when rack-mounted equipment was tested.

Tests have been carried out by the equipment manufacturers, by the NSSS supplier, and by independent laboratories. In general, tests have been performed with input accelerations of 1.0 G or higher. This is at least a factor of two higher than the maximum input acceleration that any component may experience under DBE conditions.

In 1980, the NRC initiated an Unreviewed Safety Issue (USI) A-46 to review the seismic adequacy of equipment in certain operating nuclear power plants against seismic criteria not in use when these plants were licensed. Fort Calhoun Station was identified as one of the A-46 plants which must be reviewed. OPPD joined the Seismic Qualification Utility Group (SQUG) which published the Generic Implementation Procedure, Revision 2 (GIP-2) (Reference 8) for evaluating these plants. The NRC accepted the SQUG procedure for resolving USI A-46 in Supplementary Safety Evaluation Report No. 2 (SSER No. 2) (Reference 9). OPPD used GIP-2 in its entirety, including the clarifications, interpretations, and exceptions identified in SSER No. 2, as clarified by the August 21, 1992, SQUG letter (Reference 10), to evaluate the seismic adequacy of selected safe shutdown equipment in the Fort Calhoun Station (Reference 11). The NRC issued a Safety Evaluation Report to OPPD on July 30, 1998, which accepted the results of the USI A-46 program for Fort Calhoun Station, including the approach used to resolve the outliers (Reference 12).

SQUG issued Revision 3 of the GIP (GIP-3) on May 16, 1997, to include additional restrictions and certain editorial and typographical changes to GIP-2 (Reference 13). The NRC accepted these changes in Supplemental Safety Evaluation Report No. 3 (SSER No. 3) (Reference 14).

The elements of the OPPD submittal for resolution of USI A-46 are maintained using the GIP to demonstrate that the Fort Calhoun Station can be brought to a safe shutdown condition following a safe shutdown earthquake (Reference 11).

The GIP-3, including the clarifications, interpretations and exceptions identified in SSER No. 2, as clarified in Reference 10 and in SSER No. 3 (Reference 14), may be used as an alternative method for seismic qualification of mechanical and electrical equipment, electrical relays, and cable and conduit raceway systems, and portions thereof. The use of GIP-3 is optional (i.e., the original design basis may continue to be used). This method can apply to the re-analysis or modification of existing items and to new or replacement items (except as noted below) and will be documented, therein, by reference to GIP-3 as the design basis for those calculations. This alternative seismic qualification method will not supercede specific commitments to use Regulatory Guide 1.100 (IEEE 344-1975) for certain equipment within the scope of Regulatory Guide 1.97 for seismic qualification of post-accident monitoring instrumentation in harsh environments.

The GIP-3 was originally developed for resolution of USI A-46. As such, portions of this document contain administrative, licensing, and documentation information which is only applicable to the USI A-46 program. Therefore, only the sections of GIP-3 listed below will be used to perform seismic qualification evaluation of equipment and systems. These sections will be used in their entirety, i.e., all the applicable criteria and methods defined in GIP-3 for an item of equipment or system will be used.

a. Part I, Section 2.3.4, Future Modifications and New and Replacement Equipment.

b. Part II, Section 2, Seismic Evaluation Personnel.

c. Part II, Section 4, Screening Verification and Walkdown.
   (**NOTE**: For new installations and for newly designed anchorages in modified or replaced items, the anchorage criteria in GIP-3, Part I, Section 2.3.4 will be used.)

d. Part II, Section 6.4, Comparison of Relay Seismic Capacity to Seismic Demand, and Section 6.5, Relay Walkdown.
   (**NOTE:** It is not necessary to identify "essential relays" as defined in other parts of GIP-3, Part II, Section 6.)

e. Part II, Section 8, Cable and Conduit Raceway Review.
   (**NOTE:** The additional evaluations and alternative methods for resolving raceway outliers in subsections 8.4.1 through 8.4.8 may be used. However, the generic methods for resolving outliers in GIP-3, Part II, Section 5 will not be used).

f. Part II, Section 10, References.

g. Appendix B, Summary of Equipment Class Descriptions and Caveats.

h. Appendix C, Generic Equipment Characteristics for Anchorage Evaluations.

i. Appendix D, Seismic Interaction.

j. Appendix G, Screening Evaluation Work Sheets (SEWS).

2.2.3 Containment and Auxiliary Building

The analytical model for seismic design of the containment and auxiliary building consists of several lumped masses distributed as depicted in Figure F-3. Mass M_1 represents the dome with the ring girder; masses M_2 and M_3 represent the cylindrical shell; mass M_4 represents the auxiliary building; mass M_5 represents the mat and the internal masses of the containment structure.

The analytical model of the five masses was assumed to have 13 degrees of freedom, namely; two horizontal translations in the principal directions of each mass, two rotations of the entire assumed rigid system, and uncoupled vertical motion of the entire mass of the system. The five translational movements of the masses and the rotational movement of the entire mass in the same plane are coupled. The uncoupling of the vertical motion and the motions in the two vertical planes was based on the fact that the center of gravity of the entire system reasonably coincides with the center of gravity of the elastic foundation.

The lateral stiffness coefficient of the foundation, k_H, depends on the soil modulus and elastic properties of the piles and was determined from the field test of piles. Vertical and rotary stiffness coefficients, k_v and k_o respectively, were obtained on the basis of elastic properties of the piles and bedrock. The stiffness matrix was formed on the basis of the lumped mass system with elastic properties in discrete parts which included bending and shear characteristics.

The structure's viscous-friction was assumed to be 5 percent of the structure's critical damping for the maximum hypothetical earthquake and 2 percent for the design earthquake. For concrete structural components that would not crack under the maximum hypothetical earthquake, a damping factor of 2 percent was employed.

The choice of damping factors and their use in the seismic design of the containment structure was based on the following considerations. The containment structure and the auxiliary building are supported on a continuous mat, common to both structures. This mat is supported on 803 steel pipe piles driven to rock. During an earthquake the response of this system would include horizontal and vertical translational movements of the whole structure, rocking motions of the containment and of the auxiliary building involving deformation of the piles, and an oscillatory motion of individual masses comprising the superstructure. The predominant damping effect during this motion would be in the soil-pile system and in the soil-mat system; for this damping effect a value of damping coefficient of 0.02 was used for the design earthquake and 0.05 for the maximum hypothetical earthquake.

The following procedure was utilized for the seismic design of the containment and auxiliary building structure:

a. Determination of undamped modes and frequencies;

b. Determination of the response of all elements of the system using a damping coefficient of 0.02 for the design earthquake and maximum hypothetical earthquake;

c. Determination of the response of all the elements of the system using a damping coefficient of 0.05 for the maximum hypothetical earthquake.

d. For the design earthquake, the absolute accelerations obtained with a damping coefficient of 0.02 were applied to the containment and auxiliary building structures. For the maximum hypothetical earthquake, the absolute accelerations were obtained using both .02 and .05 damping factors, independently. The same damping factors were used in each mode. Then, the absolute accelerations resulting from a damping factor of .02 were applied to the containment shell only, and the absolute accelerations resulting from a damping factor of .05 were applied to the rest of the masses. This approach yields more conservative results in this case than would a method of analysis which uses different damping in different modes. For example, an approximate rule for determining the appropriate damping for a mode is to weight the damping associated with the individual springs according to the stored energy in each spring:

$$D_n = \frac{D_s E_{sn} + D_h E_{hn} + D_r E_{rn}}{E_{sn} + E_{hn} + E_{rn}}$$

Where D_n is the weighted average damping for the n-th mode; D_s, D_h and D_r are the damping factors for motion of the superstructure, for swaying due to soil-structure interaction and for rocking due to soil structure interaction, respectively, and E_{sn}, E_{hn}, and E_{rn} are the energies stored in the n-th mode in the superstructure and in the swaying and rocking springs, respectively. Since it is known that a large damping factor is associated with swaying and due to the fact that the principal contribution, in this particular case, is in the lowest frequency mode which is predominantly associated with swaying, one would expect that the accelerations obtained as described above, using relatively low damping factors, are conservative.

The absolute modal accelerations were combined by the root-mean-square method, i.e., the square root of the sum of the squares of the modal accelerations was obtained. This method of combining the absolute modal accelerations yielded conservative results with regard to moments and shears. For instance, for the maximum hypothetical earthquake, the maximum base shear resulting from combining the absolute modal accelerations is in the order of 55,587 kips in comparison to 55,225 kips obtained by combining the modal base shears by the root-mean-square method. By the same token, the order of magnitude of the maximum base moment resulting from combining the absolute modal accelerations is 2,837,693 foot kips in comparison to 2,759,675 foot kips which results from combining the modal base moments by the root-mean-square method.

e. The amplification due to the vertical ground excitation of the containment and auxiliary building with a common mat was carried out in the dynamic analysis. The entire structure was lumped into one mass for the vibration in the vertical direction. Furthermore, the vertical motion was assumed to be uncoupled from the rest of the motions of this system. The vertical rigidity constant of the foundation was obtained on the basis of elastic properties of the piles and bedrock.

Diagrams of the containment shell absolute accelerations as a function of height above the mat are shown in Figure F-4 for the design earthquake and maximum hypothetical earthquake.

Sufficiently wide joints are provided between the containment shell and adjacent structures, including the structure within the containment vessel, to avoid any possible collision as the structures deflect independently under the seismic action.

An additional dynamic analysis of the Containment and Auxiliary Building structures, as well as the Intake Structure and Turbine Building, was performed utilizing a more refined structural model and state-of-the-art methods of Soil Structure Interaction (SSI). These analyses utilize a time history methodology for deriving structural response to seismic input. The time histories used as input to these analyses were mathematically derived by generating artificial time histories having response spectra which conservatively envelope the ground spectra described in Section F.2.1.4. The results of these analyses were used to derive alternate floor response spectra and seismic anchor motions for use with damping values, criteria and methods (i.e., Alternate Seismic Criteria and Methodologies) more current than the original design basis. The details of the analyses performed and the resultant criteria and floor response spectra can be found in Reference 6. A review of these analyses and criteria was performed by the NRC, and Reference 5 documents NRC acceptance of them as the basis for an alternate (i.e., in lieu of the original design basis) seismic criteria and methodology for analyses of equipment and structures at Ft. Calhoun.

The use of the ASCM is optional (i.e., the original design basis may continue to be used) but it will become the design basis for those systems, equipment or portions thereof for which the criteria is used in analysis and design. This can apply to reanalysis of existing items or first time analysis of new items and will be documented, therein, by reference to EA-FC-94-003 as design basis for those calculations. The caveats and requirements for use of the ASCM are set forth in Reference 6.

2.2.4 Containment and Refueling Cranes

The containment and refueling cranes were analyzed considering that the supporting runway girders are subject to the accelerations obtained from the seismic analysis of the containment and auxiliary building structure.

A structural analysis of all crane structural members was performed for the combination of seismic loads (horizontal and vertical) and dead load.

The design bases for the containment crane are predicted upon the following considerations. During the postulated loss-of-coolant accident (DBA), the containment atmosphere would undergo a large temperature rise. The temperature of the steel structure of the crane would very rapidly assume the temperature of the containment atmosphere whereas the mass of the concrete containment initially would be unaffected.

In order to avoid large forces on the crane structure and on the areas of the containment supporting the crane runway, the crane wheels must be permitted to move transversely across the rail. For this reason, the wheels are double flanged with sufficient clearance between the flanges to accommodate the maximum differential movement. For this reason also, rail clamps are not provided. However, stops are furnished to assure that the crane cannot be displaced from the rail during an earthquake.

The stops are stiffened steel assemblies bolted to the crane girders and located so as to contact the edge of the runway griders should the wheel flange fail to stop the motion of the crane in the direction normal to the rail. The striking surface of the stops on the bridge girders is inclined from the vertical so as to minimize the impact but still limits the amount of movement so that the wheel cannot move completely off the rail.

Multi-mass modal analyses of the containment and refueling cranes have been performed utilizing the vertical floor responses spectrum applicable to the structure supporting the crane. The floor spectra have been developed by a modal analysis time-history method utilizing as input the ground motions of 1940 El Centro and 1952 Taft normalized to the ground acceleration of the maximum hypothetical earthquake. These analyses show that during the maximum hypothetical earthquake neither the main crane trucks nor the trolley of either crane will be subject to a net uplift.

Similar steel stops, but with vertical striking surfaces, are provided on the trolleys to prevent the trolley wheels from being displaced from their rails.

The stops described above for the main crane trucks and trolley were furnished on both the containment crane and the refueling crane.

The acceleration of the crane runway during the maximum hypothetical earthquake, the mass of the crane, the friction between the wheels and the rail, and the strain energy and potential energy developed during impact have been considered in the design of the stops. Movement of the polar crane in a direction parallel to the rail during an earthquake is limited by friction between the locked crane wheels and the rail. End bumpers were also provided on the trolleys and on the runway of the refueling crane.

2.3 Plant Seismic Instrumentation

Seismic instrumentation is provided to determine the response of the containment and auxiliary building structure in the event of an earthquake so that such response can be compared with that used as the basis of design.

Two strong motion triaxial accelerographs are provided, one at the top of the foundation mat in the basement of the containment building and one directly above it on the operating deck at elevation 1045'-0". Since the foundation mat is common to both the containment and the auxiliary buildings and the dynamic analysis utilizes a model which includes the auxiliary building, the containment and their common foundation; the measured response at these two points will permit an evaluation of the actual response versus the seismic design values.

The upper accelerograph is located directly above the lower accelerograph and the horizontal axes of both instruments coincide. Each accelerograph is connected to a magnetic tape recording system located in the control room. The magnetic tape playback system can be connected to a strip chart recorder, thus providing a permanent record of the acceleration time history in both the vertical and the two orthogonal, horizontal directions. Operation of both accelerographs is initiated simultaneously by a single seismic trigger which responds to vertical or horizontal acceleration at a preset low level. Switch closure is maintained for a minimum duration after the vibrational level drops below the preset value. The seismic trigger also automatically resets itself after each actuation so that any aftershocks are also monitored.

The instruments are operated by batteries which are automatically recharged and are enclosed in sealed metal containers rigidly bolted to the supporting structure. The accelerographs are accessible for periodic maintenance which consists of calibration runs of the instrumentation and replacement of batteries as required.

Should a seismic disturbance occur in the neighborhood of the plant, the accelerations recorded within the plant will be the basis for a decision as to continued plant operation. If the recorded maximum acceleration does not exceed 1.2 times the acceleration predicted during the design basis earthquake (see Figure F-28) for the structure at the point where the seismic instrumentation is located, plant operation will continue unless other circumstances or indications of malfunction dictate otherwise. However, while plant operation is maintained and after the danger of aftershocks is considered to have become negligible, the complete on-line testing routine will be performed to ensure that all safeguards systems and controls are undamaged and in proper working order. A visual inspection of the plant will also be carried out to verify that no structural damage has occurred. If the recorded maximum acceleration within the plant exceeds 1.2 times the acceleration predicted during the design basis earthquake at the location of the seismic instrumentation, the plant will be brought to a shutdown condition. When danger of aftershocks is considered to have become negligible, the plant will be thoroughly inspected visually for any structural or functional damage and the shutdown testing routine will be performed. Startup will not be initiated until all systems have been verified.

Maintenance of the seismic instrumentation consists of periodic checks to ensure continuous operability. Exact maintenance procedures and intervals are in accordance with the manufacturer's recommendations.

2.4 Class II Seismic Criteria

Seismic design for all Class II structures was governed by the requirements of the National Building Code, 1967 edition. The National Building Code is the governing code for Washington County, Nebraska, in which the plant is located. This code is identical with the Uniform Building Code for earthquake design requirements. The numerical coefficient representing seismic intensity is equal to 0.25 and is the value of "Z" as established by this code for locations in zone 1, the zone designated for the area in which the plant site is located on the map of seismic probability included as a part of the code.

Class II equipment and components conform to the applicable design codes and standards. No special provisions were made against seismic effects.

The Turbine Building, which is seismic Class II, was dynamically analyzed, in Reference 6, to determine values of Seismic Anchor Motion (SAM) and Floor Response Spectra (FRS) for use in analyzing the Main Steam and Main Feedwater piping which pass between these structures. The Turbine Building was not structurally evaluated for the associated moments and forces from this analysis.

2.5 Seismic Design of Equipment and Piping

For the Fort Calhoun plant, the criterion is that if the lowest dominant natural frequency of the equipment or piping is 6 Hz or above horizontally or 18 Hz or above vertically, the ground motion response spectra, normalized to the acceleration at the elevation of the equipment support, may be used as an input for the analysis to determine the equipment response.

The basis for this criterion is that the lowest dominant natural frequency of both the reactor building and the auxiliary building is approximately 3 Hz horizontally and 9 Hz vertically and that higher modes of vibration of the building will not contribute significantly to the response of equipment attached to the building. The latter statement is made specifically for this particular plant and follows from the fact that the combined mat which supports both buildings is founded on long end-bearing steel piles. This type of foundation with its high vertical rigidity and relatively low horizontal rigidity, responds to horizontal seismic excitation primarily in horizontal translation and only very slightly in rocking. Thus the principal contribution to the total response of the internal structure of the reactor building and the auxiliary building is in the lowest frequency mode due primarily to horizontal translation arising from the soil-pile interaction.

The response of the five lumped masses of the reactor and auxiliary buildings model for six modes of vibration in a vertical plane have been obtained from the seismic analysis of the structure. These values are tabulated in Table F-3 and F-4 together with the total response of each mass calculated as the square root of the sum of the squares of the modal responses. On Figures F-5 thru F-11, the mode shapes of the system are shown for each of the modes. It can be readily seen that the principal contribution to the response of the internal structure is in the lowest frequency mode and that the rocking associated with this mode is insignificant. In the second frequency mode of about 6 Hz, there is some distortion arising from flexural and shearing effects of the containment dome and the upper and lower portions of the containment shell but the principal contribution to the total response of these masses is in the lowest frequency mode.

It should be noted that the seismic analysis of the structure is based on a somewhat conservative assumption that the sum of the mass moments of inertia of the individual masses, which assumes no vertical distortion between the masses, can be replaced by the moment of inertia of the assumed rigid entire mass. This assumption results in higher values associated with the rotary inertia effect. Moreover, the modal analysis of the structure is based on flexible rocking springs calculated on the conservative assumption that the axial rigidity of the piles alone, without the effect of the surrounding soil, contributes to the stiffness of the rocking springs. These two assumptions increase the calculated rocking effect.

Table F-3 - "Seismic Responses of Reactor and Auxiliary Buildings Obtained by Modal Analysis Using Response Spectrum Concept"

Operating Basis Earthquake

Direction of Ground Excitation		N-S							VERT.	E-W						
Modal Natural Frequencies -H_z	Mass	3.02	6.23	8.85	20.30	24.40	29	COMB RMS	9.12	3.03	6.37	8.78	19	23.3	30.7	COMB RMS
Location		Absolute Accelerations - g's and g-rad/ft														
Dome	1	.258	-.076	.005	-.001	.002	0.	.269	.090	.250	-.067	.005	-.002	.002	0.	.260
Upper Containment Cylindrical Shell	2	.227	-.044	.003	.002	-.003	.001	.231	.090	.224	-.039	.004	.002	-.003	.001	.227
Lower Containment Cylindrical Shell	3	.182	.001	.001	.002	-.001	-.001	.182	.090	.183	-.001	.001	.003	-.001	.001	.183
Auxiliary Building	4	.175	.013	-.001	-.001	-.001	0.	.177	.090	.177	0.12	-.002	-.001	-.001	0.	.178
Internal Structure Within Containment and Mat	5	.161	.016	0.	.001	.001	0.	.163	.090	.163	.013	0.	.001	.001	0.	.164
Entire Structure	Rotary Interia	.0263 x10^{-2}	-.0214 x10^{-2}	-.0037 x10^{-2}	.0032 x10^{-2}	.0016 x10^{-2}	0.	.0344 x10^{-2}	0.	.0208 x10^{-2}	-.0147 x10^{-2}	-.0045 x10^{-2}	.0003 x10^{-2}	.0002 x10^{-2}	0.	.0258 X10^{-2}

NOTE: RMS = Square root of the sum of the squares of modal responses.

Table F-4 - "Seismic Responses of Reactor and Auxiliary Buildings Obtained by Modal Analysis Using Response Spectrum Concept"

Design Basis Earthquake

Direction of Ground Excitation		N-S							VERT.	E-W						
Modal Natural Frequencies -H_z	Mass	3.02	6.23	8.85	20.30	24.40	29	COMB RMS	9.12	3.03	6.37	8.78	19	23.3	30.7	COMB RMS
Location		Absolute Accelerations - g's and g-rad/ft														
Dome	1	.538	-.143	.009	-.003	.004	-.0007	.555	.130	.525	-.130	.011	-.004	.004	-.0006	.540
Upper Containment Cylindrical Shell	2	.474	-.083	.006	.003	-.006	.0018	.482	.130	.465	-.077	.008	.004	-.007	.0016	.472
Lower Containment Cylindrical Shell	3	.380	.001	.002	.005	-.002	-.0028	.381	.130	.380	-.002	.002	.007	-.003	-.0023	.381
Auxiliary Building	4	.254	.017	-.002	-.002	-.002	0.	.255	.130	.257	.016	-.002	-.003	-.001	0.	.258
Internal Structure Within Containment and Mat	5	.234	.021	0.	.001	.002	0.	.236	.130	.236	.017	0.	.002	.002	.0003	.238
Entire Structure	Rotary Interia	$.0380 \times 10^{-2}$	$-.0273 \times 10^{-2}$	$-.0052 \times 10^{-2}$	$.0006 \times 10^{-2}$	$.0005 \times 10^{-2}$	0.	$.0416 \times 10^{-2}$	0.	$.0300 \times 10^{-2}$	$-.019 \times 10^{-2}$	$-.0064 \times 10^{-2}$	$.0008 \times 10^{-2}$	$.0003 \times 10^{-2}$	$0. \times 10^{-2}$	$.0362 \times 10^{-2}$

Design Basis Earthquake

NOTE: RMS = Square root of the sum of the squares of modal responses.

In order to substantiate the statement that the lowest frequency mode of the structure governs the response of equipment, a time-history analysis has been performed for a piece of equipment of a single-degree-of-freedom representation with a natural frequency in resonance with the lowest frequency mode of the structure. The analysis has been performed using six modes of vibration of the structure in a vertical plane and also using only the lowest dominant frequency mode of the structure. The time-history is the 1940 El Centro N-S ground motion record. Damping is assumed to be one percent of critical for the equipment and two percent for the structure corresponding to the Operating Basis Earthquake. The equipment supports are assumed to be located at an elevation of 995'-0", (i.e., at the center of gravity of the mass of the analytical model representing the internal structure of the containment and the mat). The maximum acceleration obtained using six structure modes 2.33 of gravity as compared to 2.19 of gravity using the lowest frequency mode of the structure. These results also indicate that the rocking motion due to soil-pile-structure interaction is negligible.

Floor response spectra have been developed for masses #2, #4 and #5, by a modal analysis time-history method using as input the normalized El Centro ground motion. For this purpose the recorded El Centro N-S horizontal time-history has been scaled to .08g and .17g which correspond to the Fort Calhoun OBE and DBE ground accelerations respectively. The analytical model used for the construction of the horizontal floor response spectra (as shown in Figure F-5 Part A) consists of five masses, namely mass #1 representing the concrete dome, masses #2 and #3 simulating the upper and lower portion of the containment shell respectively, mass #4 representing the auxiliary building and mass #5 representing the containment internal structure and mat. Each mass was assumed to have two translational degrees of freedom in the two principal directions. The two additional degrees of freedom are due to rotary inertias of the entire structure about the two horizontal principal axes. All six structure modes of vibration were used in the development of the horizontal response spectra. Equipment damping is taken as 0.5% of critical. The procedure used to generate the floor response spectra is described below:

Having established the normal modes, the equation of structure motion in matrix form can be written:

$$\ddot{q}_n + 2w_n\beta_s \dot{q}_n + w_n{}^2 q_n = -R\,\ddot{S}_{(t)}\,[\gamma_n]$$

where $\ddot{q}_n$, $\dot{q}_n$ and q_n are the column matrices of model relative accelerations, velocities and displacements of the structure in normal coordinates, w_n is the circular natural frequency of the structure of mode n, β_s is the fraction of critical damping of the structure in mode n, $[\gamma_n]$ is the column matrix of the modal participation factors,
$S_{(t)}$ is the time varying acceleration of a recorded ground motion and R is the scaling factor by which the maximum acceleration of an actually recorded accelerogram is scaled to the maximum acceleration predicted for the site of the structure involved.

In order to obtain response of any one-degree system mounted within the structure, the following equation of motion can be used:

$$\ddot{q}_m + 2w_m\beta_e \dot{q}m + w^2 q_m = [\sum_{n=1}^{N} \rho_{in}\ \ddot{q}_n + R\,\ddot{S}_{(t)} \qquad (2)$$

where $\ddot{q}_m$, $\dot{q}_m$ and q_m are the equipment relative acceleration, velocity and displacement, respectively, w_m is the circular frequency of the equipment, β_e is the critical damping ration of the equipment, and ρ_{in} is the modal shape of the structure of mass i in mode n.

Equations (1) and (2) are solved by numerical integration utilizing the computer program presented in reference (1) for IBM 1130 computer. The output consists of the following accelerations:

$R\,\ddot{S}_{(t)}$	=	scaled predicted ground motion
$W_i = \sum_{n=1}^{N} \rho_{in}\ddot{q}_m + R\,\ddot{S}_{(t)}$	=	absolute acceleration for any mass (i) of the structure
$\ddot{q}_m$	=	relative acceleration of the equipment
$\ddot{W}_i + \ddot{q}_m$	=	absolute acceleration of the equipment

Suitable time intervals Δt for a step-by-step numerical integration were selected.

It should be noted that a system of equations of motion, consisting of several masses and spring constants of a structure subjected to a time-history support motion, may also be solved by numerical integration. In this case, the numerical integration technique must be performed simultaneously for all of the coupled equations. This procedure is cumbersome, requiring a large amount of computations, and is susceptible to computational difficulties. For example, it is difficult to know how small the time intervals should be to avoid mathematical instability. Furthermore, there is no really satisfactory way to determine all of the damping coefficients in these equations. Because of these difficulties, the modal method of analysis was used as described above.

The resulting spectra for masses #2, #4, and #5 for DBE and OBE are shown on Figure F-12 thru F-17. The normalized ground response spectra for frequencies exceeding 3 Hz are also shown on these figures.

For the modal analysis of the reactor coolant system, horizontal response spectra have been developed by the procedure described above, but using the computed responses to actual time-history inputs of two ground motion records: 1940 El Centro N-S component and 1952 Taft N-21E component. Also, the San Francisco Golden Gate recorded accelerogram was used in some instances in order to check the conservatism of the response. The responses obtained from Taft and El Centro accelerograms govern the peaks in all cases. The envelope of maximum peaks was used for the construction of the spectra for equipment frequencies in the horizontal direction between approximately 6 Hz and 1 Hz and in the vertical direction between approximately 15 Hz and 5 Hz. As for the spectra discussed above, all six structure modes of vibration were used. Uncertainties associated primarily with the evaluation of the foundation rigidities were taken into account by parametric studies which resulted in shifting of the resonance peaks of the floor spectra. Normalized ground motion response spectra were used for equipment frequencies of about 6 Hz and above for the construction of horizontal spectra and for equipment frequencies of about 18 Hz and above for the vertical spectra. The resulting horizontal floor response spectra of the containment internal structure and mat (mass #5), shown in Figures F-18 and F-19, have been developed at an elevation of 995'-0", the center of gravity of mass #5. Figures F-20 and F-21 show these spectra normalized to an elevation of 1006'-4 1/2" which is the elevation of the reactor vessel inlet and outlet nozzles. Normalizing of these spectra is in accordance with the variation in structure acceleration shown on Figure F-28 and F-29. The internal structure within the containment consists of box type heavy concrete shear wall assemblies, especially the reactor support walls whose heavy thicknesses are provided for radiation shielding purposes. It has been found that the lowest natural frequency of this internal structure, based on conservative assumptions, is approximately 18 Hz. This indicates that the internal structure is relatively rigid and, therefore, validates the assumption made in the analytical model for the dynamic analysis of the reactor and auxiliary buildings that the internal structure within the containment can be considered as an integral part of the mat.

The vertical response spectra shown in Figures F-22 thru F-27 were constructed for all elevations of the reactor and auxiliary buildings since the analytical model of these buildings was assumed as a single-degree-of-freedom system, that is the entire mass of these buildings was considered to be lumped into one.

Absolute accelerations of masses #2, #4 and #5 as obtained by the modal analysis time-history using the scaled El Centro ground motion are shown in Tables F-5 and F-6. Comparison of Tables F-3 thru F-6 verifies that the specified ground response spectrum is compatible with the scaled El Centro ground motion, as has been stated in Reference 2.

It can be seen from Figures F-12, F-13, F-15 and F-16 that for equipment or piping supported on mass #4 or #5 and which is in resonance with the second frequency mode of the structure (6.23 Hz) the acceleration given by the normalized ground response spectrum for this frequency exceeds the acceleration obtained by a time-history analysis using the El Centro ground motion. These results in conjunction with the demonstrated fact that for the Fort Calhoun plant the higher structure modes of vibration have an insignificant contribution to the response of the structure justify the proposed approach to the seismic design of piping and equipment supported within the reactor internal concrete or auxiliary building.

This approach was that if equipment or piping within the containment internal structure or auxiliary building has a lowest dominant natural frequency of 6 Hz or more horizontally or 18 Hz or more vertically, the acceleration of the equipment or piping was taken directly from the normalized ground response spectrum. In the case of piping, restraints were spaced to limit the natural frequency of the piping to this criterion.

Figures F-14 and F-17 indicate that for mass #2, which represents the upper portion of the containment shell, the response obtained by the El Centro time-history analysis for equipment with a frequency of 6.23 Hz slightly exceeds the acceleration given by the normalized ground response spectra. Therefore, for those piping runs which penetrate the containment shell or are otherwise connected to it the spacing of restraints were such as to assure a lowest dominant natural frequency of 12 Hz horizontally and 18 Hz vertically for the pipe run up to the first point of full fixity. The normalized ground response spectra was used and relative movements of the shell and adjacent structures was considered in the stress analysis of the piping.

The following criteria have been applied:

a. The greater of the responses shown in Figure F-12 through F-17 was used for the appropriate frequency of the system analyzed.

b. Items which cannot be removed from resonance were designated to 1.3 times the peak of the appropriate floor response spectra (to account for participation of the higher modes in the system or equipment); allowable stresses calculated on an elastic basis were limited in these analyses to those corresponding to the emergency condition limit of the B31.7 (1968) piping code.

c. Items which cannot meet the criteria (b) above were analyzed by the response spectrum modal method using the appropriate floor response spectra.

Where piping is connected to equipment whose lowest dominant natural frequency is less than 6 Hz horizontally and 18 Hz vertically, restraints were spaced to limit the lowest dominant natural frequency of the piping to 25 Hz up to the first point of full fixity. If this was not feasible, a modal analysis was performed on an analytical model of a lumped mass system which included the piping and the equipment to which it is attached. The appropriate floor response spectrum was used. Stops or damping devices were provided to limit stress to acceptable values.

An alternative procedure to that described in the preceding paragraph was to space the piping restraints to limit the lowest natural frequency of the piping to 6 Hz and assume a response equal to the peak of the appropriate floor response spectrum. The piping was then designed in accordance with code requirements for the Emergency Condition. As noted in Section F.2.2.3, alternative structural dynamic analyses, criteria and methods, to those described above, were developed and the results may be used as an alternative design basis (i.e., in lieu of the original design basis described above) for analysis and design of piping.

OPPD requested relief from modifying pipe supports SIH-3 and RCH-13. The relief request was based on guidance provided in NRC Inspection and Enforcement Bulletin (IEB) No. 79-14, Supplement 2, "Seismic Analysis for As-Built Safety Related Piping Systems," dated September 7, 1979. The limiting component for SIH-3 is an expansion anchor with a factor of safety of 3.1 which is less than 4 as required. The limiting component for RCH-13 is an angle iron, where the maximum bending stress exceeds the licensing basis upset allowable by less than 2%, while it meets the licensing basis faulted allowable stress by an adequate margin. The NRC staff finds due to ALARA, the repair of these supports a hardship. The staff has concluded that relief be granted from modifying pipe supports SIH-3 and RCH-13 (Reference 7).

As noted in Section F.2.2.2, an alternative method of evaluating the seismic adequacy of equipment was developed and the results may be used as an alternative design basis method (i.e., in lieu of the original design basis described above) for analysis and design of mechanical and electrical equipment.

Table F-5 - "Seismic Response of Reactor and Auxiliary Buildings Obtained by Modal Analysis Using Time-History Concept"

Operating Basis Earthquake

Input Earthquake		El Centro - May 18, 1940 Scaled N-S Component				Taft - July 21, 1952 Scaled N21°E Component			
Direction of Ground Excitation		N - S		VERT.		N - S		VERT.	
Time - Sec.		2.6674	2.6464	2.6704	2.4814	9.8314	9.8344	9.9965	9.1385
Location	Mass	Absolute Accelerations - g's							
Upper Containment Cylindrical Shell	2	.2137	.2292*	.2056	.1369*	-.3165	-.3231	.3894*	.1161*
Auxiliary Building	4	.1942	.1791	.1947*	.1369*	-.2956	-.2962*	.2453	.1161*
Internal Structure Within Containment and Mat	5	.1837*	.1663	.1834	.1369*	-.2819*	-.2816	.2193	.1161*

* Maximum Response

Table F-6 - "Seismic Response of Reactor and Auxiliary Buildings Obtained by Modal Analysis Using Time-History Concept"

Operating Basis Earthquake

Input Earthquake		El Centro - May 18, 1940 Scaled N-S Component				Taft - July 21, 1952 Scaled N21°E Component			
Direction of Ground Excitation		N - S		VERT.		N - S		VERT.	
Time - Sec.		2.6614	2.6644	2.6464	2.4824	9.6504	9.8435	9.8465	9.1364
Location	Mass	Absolute Accelerations - g's							
Upper Containment Cylindrical Shell	2	.469	.462	.487*	.2338*	.8994*	-.720	-.7275	.1803*
Auxiliary Building	4	.3289	.3289*	.3167	.2338*	.4313	-.4418	-.4429*	.1803*
Internal Structure Within Containment and Mat	5	.3127*	.3120	.2981	.2338*	.4010	-.418*	-.4165	.1803*

* Maximum Response

3. APPENDIX F REFERENCES

1. Nuclear Reactors and Earthquakes, TID-7024, Division of Licensing and Regulation, AEC, Washington, D.C., August, 1963.

2. Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards, Newmark, N. M., Department of Civil Engineering, University of Illinois, (paper presented in Tokyo, 1968).

3. IBM Application Program, H20-0282-1 1130 Continuous System Modeling Program, (1130-CX-13X), Program Reference Manual.

4. Report to AEC Regulatory Staff, Adequacy of the Structural Criteria for Fort Calhoun Station - Unit No. 1, Omaha Public Power District (Docket No. 50-285), by N. M. Newmark, W. J. Hall and A. J. Hendron, January 12, 1968.

5. USNRC Safety Evaluation Report of Alternate Seismic Criteria and Methodologies - Fort Calhoun Station, April 16, 1993, TAC No. M71408 (NRC-93-0150)

6. EA-FC-94-003, Alternate Seismic Criteria and Methodologies, Rev. 0

7. USNRC Safety Evaluation Report, Fort Calhoun Station Unit No. 1 - Request for Relief from Modifying Pipe Supports SIS-63/65, SIH-3 and RCH-13 (TAC-No. M95547), OPPD Tracking No. NRC-96-188.

8. Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment, Revision 2, Corrected 02/14/98, Seismic Qualification Utility Group (SQUG), February 1992.

9. NRC letter to SQUG Members dated May 22, 1992, Supplemental No. 1 to Generic Letter 87-02 transmitting Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision 2, Corrected February 14, 1992 (GIP-2).

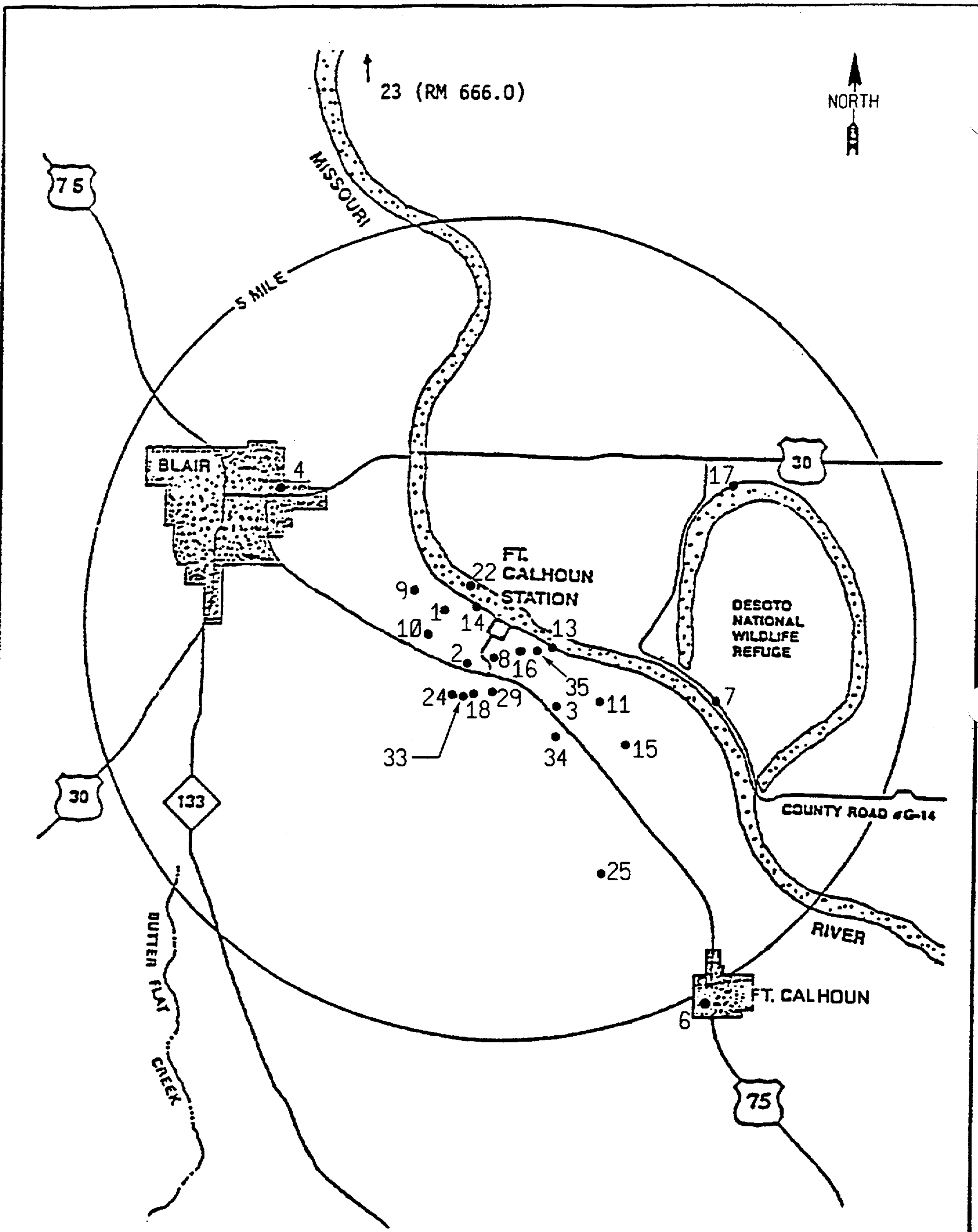
10. SQUG Letter to NRC dated August 21, 1992, SQUG Response to Generic Letter 87-02, Supplement 1 and Supplementary Safety Evaluation Report No. 2 on the SQUG GIP.

11. EA-FC-93-085, NRC USQ A-46 and Seismic IPEEE Resolution.

12. NRC Letter to OPPD dated July 30, 1998, Fort Calhoun Station, Unit No. 1 - Closeout of Unresolved Safety Issue A-46 (TAC No. M69447), OPPD Tracking No. NRC-98-129.

13. Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment, Revision 3, Updated 05/16/97 (GIP-3), Seismic Qualification Utility Group (SQUG), May 1997.

14. NRC Letter for SQUG dated December 4, 1997, Supplemental Safety Evaluation Report No. 3 (SSER No. 3) on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment, Updated May 16, 1997 (GIP-3).



OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	Environmental Radiological Sampling Points 0-5 Miles	**USAR** Figure 2.10-2 REV.SH. 25464 FILE 36056 APVD 5/9/00 REV 4

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

					F1 40.977	H1 38.643
		C2 35.894	D2 35.894	E2 20.094	G2 0.000	J2 29.109
	B3 38.732	C3 13.645	D3 0.000	E3 0.000	G3 21.375	J3 0.000
	B4 6.510	C4 0.000	D4 15.954	E4 31.762	G4 0.000	J4 34.006
A6 40.823	B5 21.600	C5 0.000	D5 32.202	E5 21.994	G5 21.831	J5 20.027
A8 39.134	B7 0.000	C7 21.506	D7 0.000	E7 21.815	G7 17.018	J7 0.000
	B9 29.126	C9 0.000	D9 34.026	E9 20.118	G9 0.000	J9 31.278

OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. 23643	APVD mJr 11/19/99	REV 5
FILE 36416		

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

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

F1	H1
43.958	42.197

		C2 40.538	D2 44.047	E2 31.703	G2 17.098	J2 42.050
	B3 42.001	C3 27.215	D3 21.073	E3 22.721	G3 39.501	J3 22.919
	B4 11.316	C4 20.724	D4 36.591	E4 49.444	G4 24.030	J4 51.445
A6 45.533	B5 33.162	C5 22.609	D5 49.780	E5 41.082	G5 40.104	J5 38.674
A8 45.207	B7 18.538	C7 39.730	D7 23.862	E7 39.952	G7 38.163	J7 23.234
	B9 43.477	C9 23.460	D9 51.516	E9 38.697	G9 23.142	J9 49.579

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23644	APVD	REV
FILE 36417	MJG 11/19/97	5

Assy Avg Burnup Distribution End of Cycle (15,500 MWD/MTU)	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-2

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

F1 U4	H1 U4

		C2 M/	D2 A U7	E2 W2	G2 2 X2	J2 U3
	B3 U6	C3 4 W1	D3 X3	E3 A X5	G3 W4	J3 N X6
	B4 A U1	C4 X3	D4 B W1	E4 U2	G4 X6	J4 U2
A6 U6	B5 W3	C5 A X4	D5 U5	E5 B W5	G5 W3	J5 3 W2
A8 U5	B7 2 X2	C7 W3	D7 1 X6	E7 W3	G7 W1	J7 X1
	B9 U3	C9 N X6	D9 U2	E9 3 W2	G9 X1	J9 4 M/

FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION		
REV.SH. 23645	APVD MJG 11/15/95	REV 5
FILE 36419		

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

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

				F1 0.185	H1 0.223	
		C2 0.238	D2 0.488	E2 0.727	G2 1.156	J2 0.841
	B3 0.190	C3 0.867	D3 1.337	E3 1.403	G3 1.155	J3 1.342
	B4 0.264	C4 1.326	D4 1.448	E4 1.196	G4 1.449	J4 1.105
A6 0.301	B5 0.746	C5 1.446	D5 1.207	E5 1.333	G5 1.233	J5 1.230
A8 0.395	B7 1.318 1.713	C7 1.208	D7 1.462	E7 1.228	G7 1.415	J7 1.347
	B9 0.986	C9 1.431	D9 1.132	E9 1.234	G9 1.341	J9 0.986

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23646	APVD	REV
FILE 36420	mrr 11/19/55	5

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

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

					F1 0.180	H1 0.216
		C2 0.245	D2 0.488	E2 0.719	G2 1.117	J2 0.823
	B3 0.188	C3 0.852	D3 1.336	E3 1.426	G3 1.158	J3 1.391
	B4 0.261	C4 1.313 1.696	D4 1.405	E4 1.182	G4 1.506	J4 1.127
A6 0.285	B5 0.722	C5 1.442	D5 1.182	E5 1.309	G5 1.238	J5 1.253
A8 0.373	B7 1.240	C7 1.189	D7 1.505	E7 1.230	G7 1.452	J7 1.474
	B9 0.939	C9 1.457	D9 1.143	E9 1.251	G9 1.467	J9 1.099

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23647	APVD MJL 11/19/98	REV 5
FILE 36421		

Core Radial Power Distribution at 1500 MWD/MTU Burnup ARO	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-6

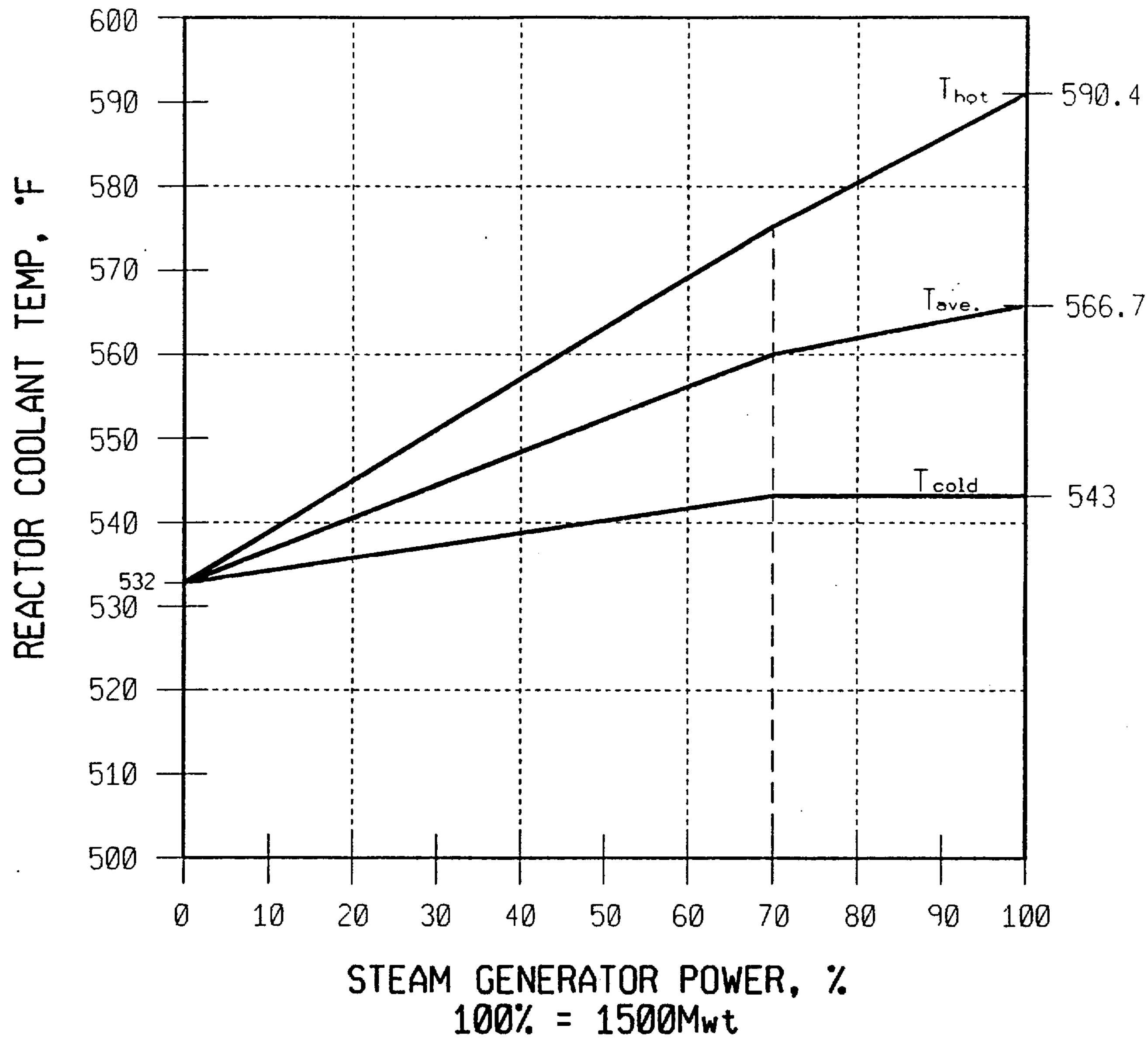
AA	FUEL ASSEMBLY LOCATION NUMBER
B.BBB	RELATIVE ASSEMBLY POWER DENSITY
C.CCC	MAXIMUM Fxy

				F1 0.222		H1 0.261	
		C2 0.341	D2 0.595	E2 0.794	G2 1.118	J2 0.874	
	B3 0.255	C3 0.935	D3 1.363	E3 1.445	G3 1.149	J3 1.467	
	B4 0.396	C4 1.354	D4 1.288	E4 1.125	G4 1.497	J4 1.119	
	B5 0.808	C5 1.448	D5 1.121	E5 1.171	G5 1.122	J5 1.143	
A6 0.366	B7 1.221	C7 1.164	D7 1.495	E7 1.118	G7 1.272	J7 1.403	
A8 0.464	B9 0.973	C9 1.498 1.619	D9 1.124	E9 1.141	G9 1.400	J9 1.064	

OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

REV.SH. 23648	APVD	REV
FILE 36422	MJR 11/15/95	5

Core Radial Power Distribution EOC (15,500 MWD/MTU) ARO	Omaha Public Power District Fort Calhoun Station Unit No. 1	Figure 3.4-7



OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO. 1

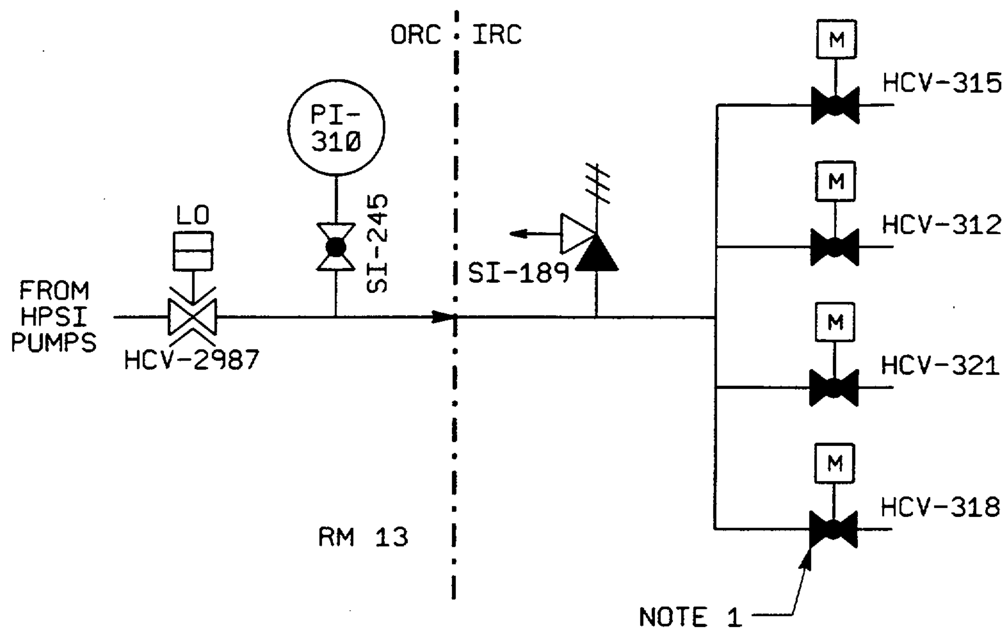
TEMPERATURE CONTROL PROGRAM

USAR

FIGURE 4.3-9

REV.SH. 36684	APVD DSM 2-11-99	REV 2
FILE 36459		

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-315	AS IS	OPEN	SIAS
HCV-312	AS IS	OPEN	SIAS
HCV-321	AS IS	OPEN	SIAS
HCV-318	AS IS	OPEN	SIAS
SI-245	—	—	—
HCV-2987	NOTE 2	—	—
SI-189	—	—	—



NOTES:

1. These valves are located as close to the first major system component as is possible.
2. Valve fails AS-IS on loss of instrument air, and fails OPEN on loss of DC control power to air solenoid.

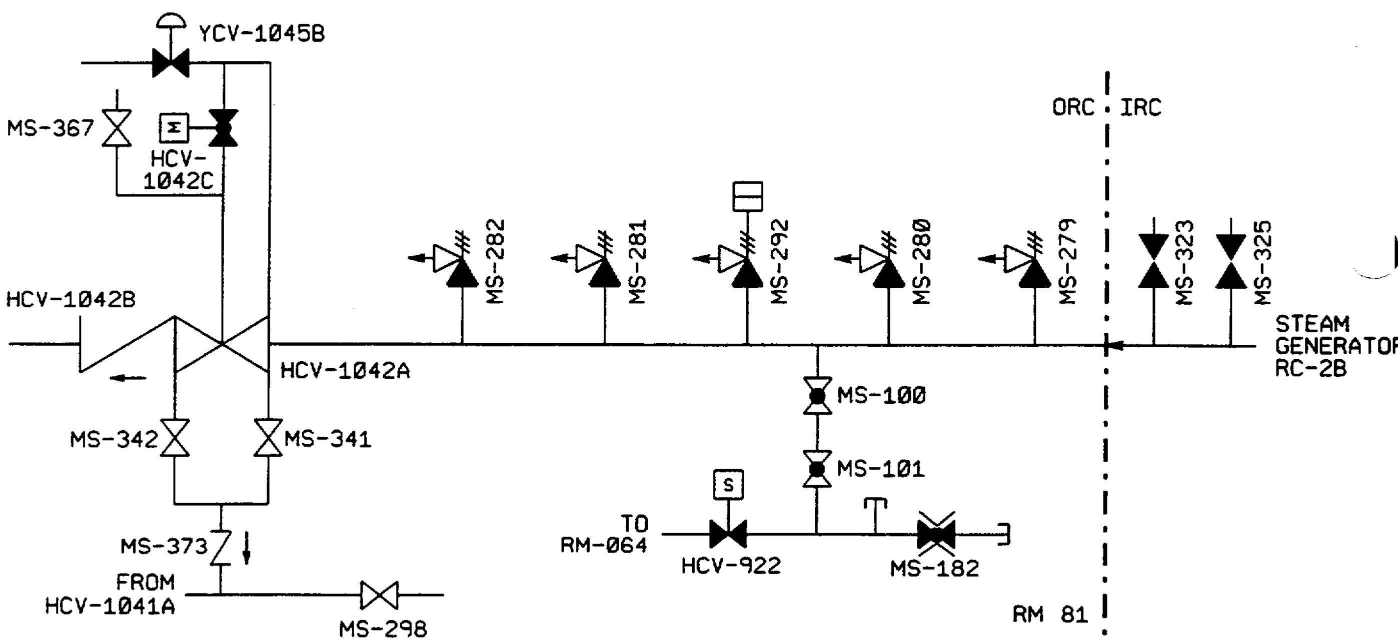
Reference Drawings:

1. 11405-MECH-1
2. E-23866-210-130

OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION UNIT NO. 1	PENETRATION M-5 SAFETY INJECTION (Reactor Coolant Exposed System)	USAR Figure 5.9-13, Sh. 5 REV.SH. 19990 APVD REV 6 FILE 40126

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY
	FAIL	ACCIDENT	
HCV-1042A	CLOSED	CLOSED	SGIS
HCV-1042C	AS-IS	CLOSED	SGIS
YCV-1045B	OPEN	OPEN	AFAS
HCV-922	CLOSED	NOTE 2	—
MS-367	—	NOTE 3	—
MS-341	—	—	—
MS-342	—	—	—
MS-100	—	—	—
MS-101	—	—	—
MS-182	—	—	—
MS-279	—	—	—
MS-280	—	—	—
MS-281	—	—	—
MS-282	—	—	—
MS-292	—	—	—
MS-323	—	—	—
MS-325	—	—	—
MS-298	—	NOTE 3	—

NOTE 1



Reference Drawings:

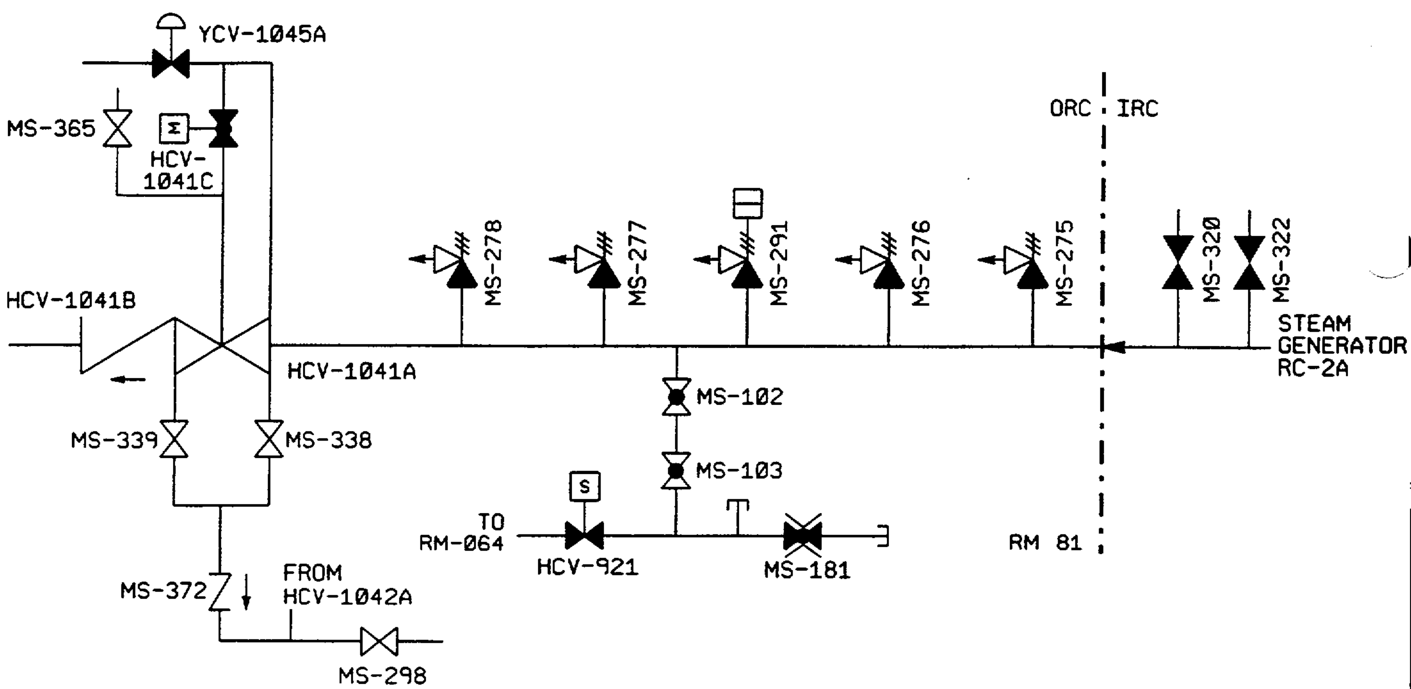
1. 11405-MECH-1
2. 11405-M-252

NOTE:

1. YCV-1045B can be closed in the event of SG tube rupture.
2. Refer to OI-PAP-8
3. Refer to EOP-4 for steam generator tube rupture accident

OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	PENETRATION M-94 MAIN STEAM FROM RC-2B (Closed System)	**USAR** Figure 5.9-13 Sheet 62 REV.SH. 20047 / FILE 40183 / APVD KCH / REV 8

VALVE NUMBER	VALVE POSITION		REPOSITIONED BY	
	FAIL	ACCIDENT		
HCV-1041A	CLOSED	CLOSED	SGIS	
HCV-1041C	AS-IS	CLOSED	SGIS	
YCV-1045A	OPEN	OPEN	AFAS	NOTE 1
HCV-921	CLOSED	NOTE 2	—	
MS-365	—	NOTE 3	—	
MS-338	—	—	—	
MS-339	—	—	—	
MS-102	—	—	—	
MS-103	—	—	—	
MS-181	—	—	—	
MS-275	—	—	—	
MS-276	—	—	—	
MS-277	—	—	—	
MS-278	—	—	—	
MS-291	—	—	—	
MS-320	—	—	—	
MS-322	—	—	—	
MS-298	—	NOTE 3	—	



Reference Drawings:

1. 11405-MECH-1
2. 11405-M-252

NOTE:

1. YCV-1045A can be closed in the event of SG tube rupture.
2. Refer to OI-PAP-8
3. Refer to EOP-4 for steam generator tube rupture accident

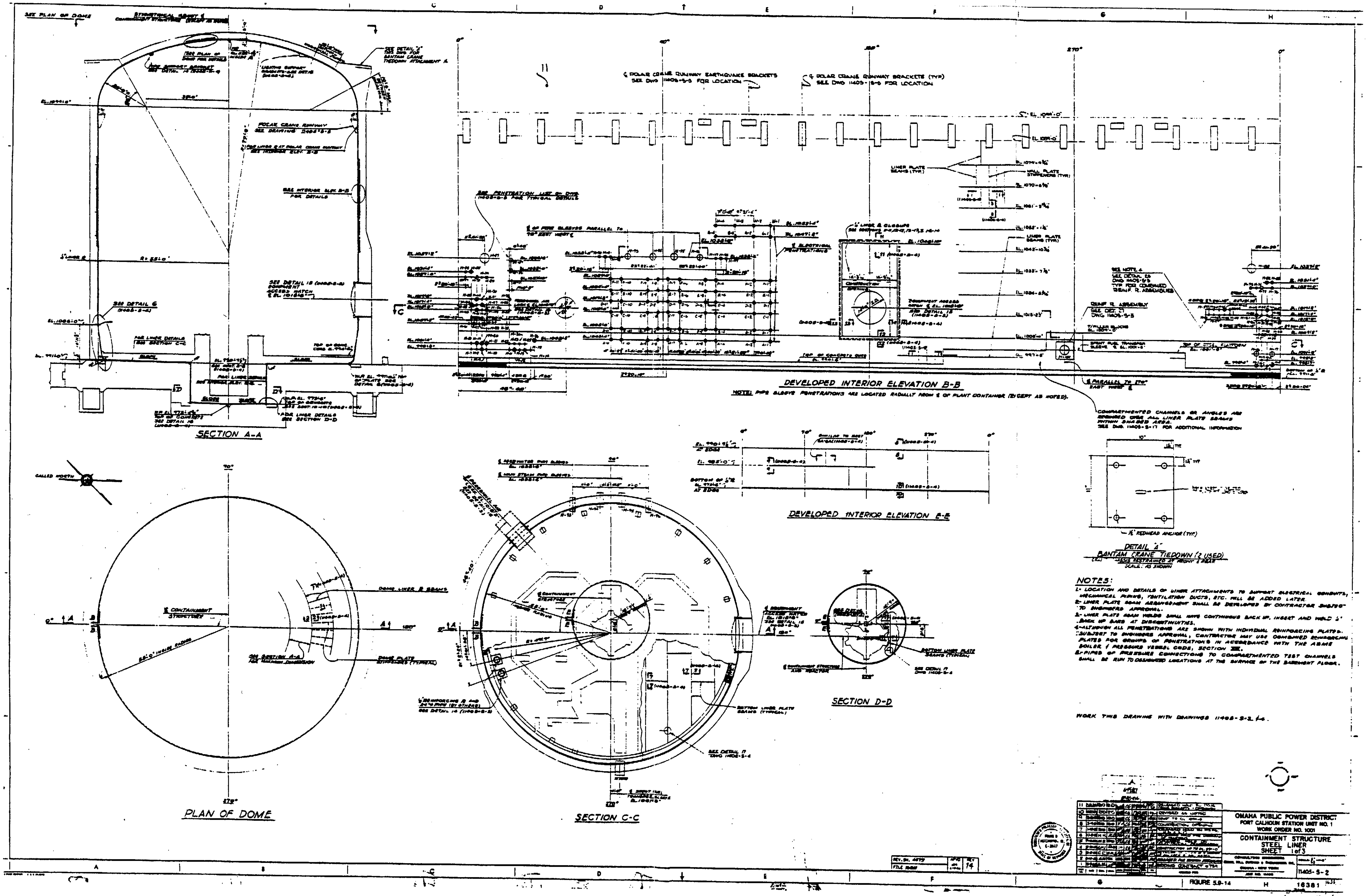
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO. 1

PENETRATION M-95
MAIN STEAM FROM RC-2A
(Closed System)

USAR

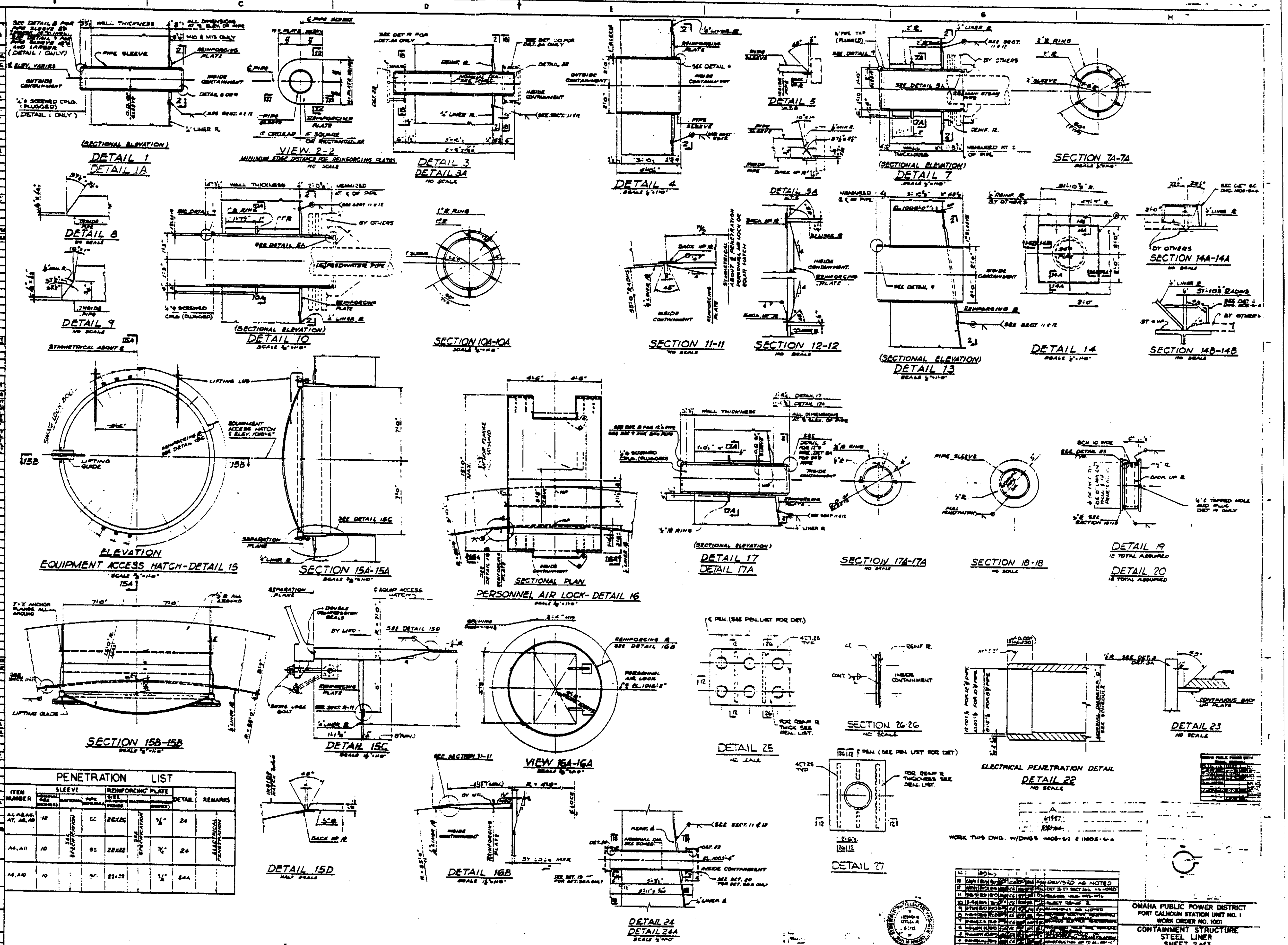
Figure 5.9-13 Sheet 63

REV.SH. 20048	APVD	REV
FILE 40184	KH	8


SECTION A-A
PLAN OF DOME
SECTION C-C
SECTION D-D
DEVELOPED INTERIOR ELEVATION B-B
NOTE: PIPE SLEEVE PENETRATIONS ARE LOCATED RADIALLY FROM ℄ OF PLANT CONTAINER (EXCEPT AS NOTED).
DEVELOPED INTERIOR ELEVATION E-E
℄ POLAR CRANE RUNWAY EARTHQUAKE BRACKETS SEE DWG 11405-S-5 FOR LOCATION
℄ POLAR CRANE RUNWAY BRACKETS (TYP) SEE DWG 11405-S-5 FOR LOCATION
DETAIL "A"
BANTAM CRANE TIEDOWN (2 USED)
SCALE: AS SHOWN
NOTES:
1- LOCATION AND DETAILS OF LINER ATTACHMENTS TO SUPPORT ELECTRICAL CONDUITS, MECHANICAL PIPING, VENTILATION DUCTS, ETC. WILL BE ADDED LATER.
2- LINER PLATE SEAM ARRANGEMENT SHALL BE DEVELOPED BY CONTRACTOR SUBJECT TO ENGINEERS APPROVAL.
3- LINER PLATE SEAM WELDS SHALL HAVE CONTINUOUS BACK UP. INSERT AND WELD ¼" BACK UP BARS AT DISCONTINUITIES.
4- ALTHOUGH ALL PENETRATIONS ARE SHOWN WITH INDIVIDUAL REINFORCING PLATES, SUBJECT TO ENGINEERS APPROVAL, CONTRACTOR MAY USE COMBINED REINFORCING PLATES FOR GROUPS OF PENETRATIONS IN ACCORDANCE WITH THE ASME BOILER & PRESSURE VESSEL CODE, SECTION III.
5- PIPES OF PRESSURE CONNECTIONS TO COMPARTMENTED TEST CHANNELS SHALL BE RUN TO DESIGNATED LOCATIONS AT THE SURFACE OF THE BASEMENT FLOOR.
WORK THIS DRAWING WITH DRAWINGS 11405-S-3, &-4.
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO. 1
WORK ORDER NO. 1001
CONTAINMENT STRUCTURE
STEEL LINER
SHEET 1 of 3
11405-S-2
FIGURE 5.9-14
16381

PENETRATION LIST

ITEM NUMBER	SLEEVE: NOMINAL SIZE (INCHES)	SLEEVE: MATERIAL	SLEEVE: PIPE SCHEDULE	REINFORCING PLATE: SIZE (INCHES)	REINFORCING PLATE: MATERIAL	REINFORCING PLATE: THICKNESS (INCHES)	DETAIL	REMARKS
M1	12		80	26x26		½"	7	SPARE
M2	8			18x18				CHARGING
M3	8			18x18				CHARGING
M4	[illegible]			[illegible]				SPARE
M5	12			26x26			17	HPSI
M6	12			26x26			17	HPSI
M7	[illegible]			18x18				[illegible]
M8	[illegible]			[illegible]			1A	W.D.
M9	12			[illegible]				[illegible]
M10	8			[illegible]				[illegible]
M11	[illegible]			[illegible]				A.C.
M12	12			[illegible]				[illegible]
M13	8			18x18				[illegible]
M14	12			[illegible]			1A	VENT. H. DR.
M15	8			[illegible]			1A	A.C.
M16	[illegible]			[illegible]			17A	[illegible]
M17	24			[illegible]			17A	LPSI
M18	12			26x26				A.C.
M19	12			[illegible]			1A	A.C.
M20	12			26x26				[illegible]
M21	12			26x26				SPARE
M22	8			18x18				S.I.
M23	8							SPARE
M24	8						1A	GAS SAMPLE
M25	8			18x18			1A	GAS SAMPLE
M26	16			32x32			1A	SPARE
M27	8			18x18				
M28	12			26x26				
M29	8			18x18				
M32	12	SEE SPECIFICATION		26x26	SEE SPECIFICATION			
M33	8			18x18				
M34	12			26x26				
M35	8			18x18				
M36	8			18x18				
M37	8			18x18				
M38	12			26x26			1A	[illegible]
M39	8			18x18			1A	AC
[illegible]								SPARE
M42	8			18x18			1A	N2
M43	12			26x26			1A	N2
M44	8			18x18			1A	[illegible]
M45	8							R.C. SAMPLE
M46	8						1A	[illegible]
M47	8						1A	[illegible]
M48	8						1A	[illegible]
M49	8			18x18				[illegible]
M50	12			26x26			1A	[illegible]
M51	8			18x18			1A	[illegible]
M52	8			18x18			1A	[illegible]
M53	12			18x18				A.C.
M54	12			26x26				SPARE
M55	8			18x18				
M56	12			26x26				
M59	8							
M60	8							
M61	8			18x18				
M62	12			26x26				
M63	8			18x18				[illegible]
M64	8			18x18				SPARE
M65	8			18x18				
M66	12			26x26				
M67	8			18x18				
M68	8			18x18				
M70	8			18x18				
M71	8			18x18				
M72	12			26x26				
M73	12			26x26			1A	INSTR. AIR
M74	8			18x18			1A	PLANT AIR
M75	14			28x28				AUX. COOLING
M76	12			26x26				
M77	12			26x26			1A	
M78	14			28x28			1A	
M79	8			18x18			1A	[illegible] WATER
M80	12			26x26			1A	[illegible] WATER
M81	8			18x18				SPARE
M82	14			28x28				AUX. COOLING
M83	12			26x26				
M84	12			26x26			1A	
M85	14			28x28			1A	
M86	12		80	26x26			1A	S.I.
M87	[illegible]		[illegible]	[illegible]		1	4	PURGE AIR
M88	[illegible]		[illegible]	[illegible]		1	4	PURGE AIR
M89	12		[illegible]	[illegible]		½"	1A	S.I.
M90	8			18x18			1A	SPARE
M91	12			26x26		½"	1	EMERG. F.W.
M92	8		80	18x18			7	SPARE
[illegible]	30		[illegible]	[illegible]		1	10	[illegible]
[illegible]	44			88x88		1	7	[illegible]
[illegible]	44			88x88		1	7	[illegible]
[illegible]	30			[illegible]		1	10	[illegible]
[illegible]	8		[illegible]	18x18		½"	1	[illegible]
[illegible]	12		80	26x26		½"	1	SPARE
[illegible]	8		80	18x18		½"	1	SPARE
[illegible]	[illegible]		[illegible]	[illegible]		1	13	[illegible]
C4,C7,C8,D1, D2,[illegible]	12		80	26x26		¾"	3	ELECTRICAL PENETRATION
[illegible]	10		80	22x22		¾"	3	
[illegible]	8		80	18x18		¾"	3	
[illegible]	12		80	26x26		¾"	3A	
[illegible]	10		80	22x22		¾"	3A	
F2	8		80	18x18		¾"	3A	



PENETRATION LIST

ITEM NUMBER	SLEEVE: NOMINAL SIZE (INCHES)	SLEEVE: MATERIAL	SLEEVE: PIPE SCHEDULE	REINFORCING PLATE: SIZE (INCHES)	REINFORCING PLATE: MATERIAL	REINFORCING PLATE: THICKNESS (INCHES)	DETAIL	REMARKS
A1, A2, A6, A7, A8, A9	12	SEE SPECIFICATION	80	26x26	SEE SPECIFICATION	¾"	24	ELECTRICAL PENETRATION
A4, A11	10		80	22x22		¾"	24	
A5, A10	10		80	22x22		¾"	24A	

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

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

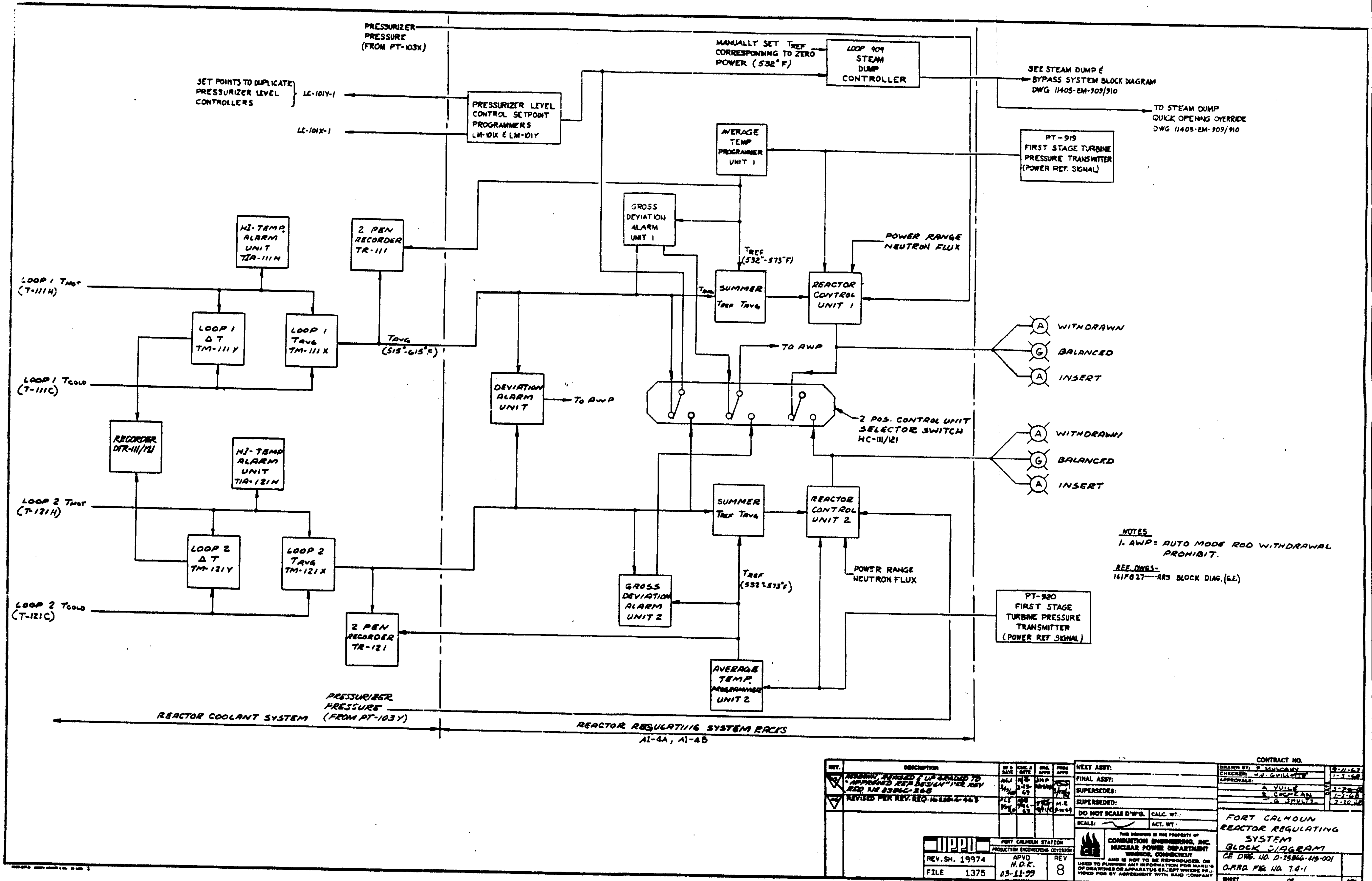
CONTAINMENT STRUCTURE
STEEL LINER
SHEET 2 of 3

SCALE AS NOTED

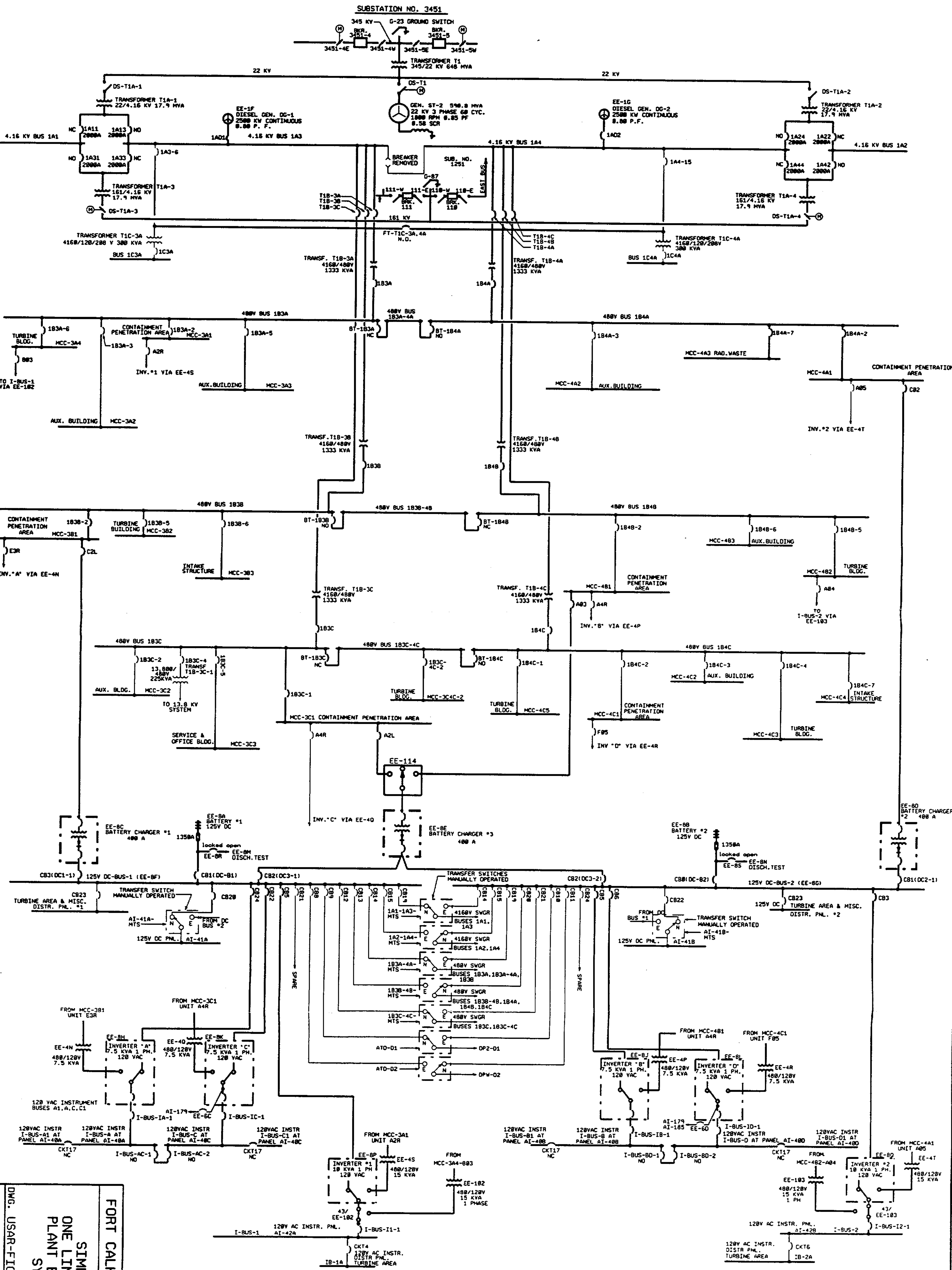
11405-S-3

FIGURE 5.9-15

16382



PRESSURIZER PRESSURE (FROM PT-103X)
SET POINTS TO DUPLICATE PRESSURIZER LEVEL CONTROLLERS
LC-101Y-1
LC-101X-1
PRESSURIZER LEVEL CONTROL SETPOINT PROGRAMMERS LM-101X & LM-101Y
MANUALLY SET TREF CORRESPONDING TO ZERO POWER (532°F)
LOOP 909 STEAM DUMP CONTROLLER
SEE STEAM DUMP & BYPASS SYSTEM BLOCK DIAGRAM DWG 11405-EM-909/910
TO STEAM DUMP QUICK OPENING OVERRIDE DWG 11405-EM-909/910
AVERAGE TEMP PROGRAMMER UNIT 1
PT-919 FIRST STAGE TURBINE PRESSURE TRANSMITTER (POWER REF. SIGNAL)
GROSS DEVIATION ALARM UNIT 1
POWER RANGE NEUTRON FLUX
TREF (532°-575°F)
SUMMER TREF TAVG
REACTOR CONTROL UNIT 1
HI-TEMP ALARM UNIT TIA-111H
2 PEN RECORDER TR-111
LOOP 1 THOT (T-111H)
LOOP 1 ΔT TM-111Y
LOOP 1 TAVG TM-111X
TAVG (515°-615°F)
LOOP 1 TCOLD (T-111C)
DEVIATION ALARM UNIT
TO AWP
WITHDRAWN
BALANCED
INSERT
2 POS. CONTROL UNIT SELECTOR SWITCH HC-111/121
RECORDER DTR-111/121
HI-TEMP ALARM UNIT TIA-121H
LOOP 2 THOT (T-121H)
SUMMER TREF TAVG
REACTOR CONTROL UNIT 2
LOOP 2 ΔT TM-121Y
LOOP 2 TAVG TM-121X
GROSS DEVIATION ALARM UNIT 2
TREF (532°-575°F)
POWER RANGE NEUTRON FLUX
LOOP 2 TCOLD (T-121C)
2 PEN RECORDER TR-121
PT-920 FIRST STAGE TURBINE PRESSURE TRANSMITTER (POWER REF SIGNAL)
AVERAGE TEMP. PROGRAMMER UNIT 2
PRESSURIZER PRESSURE (FROM PT-103Y)
REACTOR COOLANT SYSTEM
REACTOR REGULATING SYSTEM RACKS
A1-4A, A1-4B
NOTES
1. AWP = AUTO MODE ROD WITHDRAWAL PROHIBIT.
REF. DWGS-
161F827----RRS BLOCK DIAG. (G.E.)
REV. DESCRIPTION
REDRAWN, REVISED & UPGRADED TO "APPROVED REF DESIGN" PER REV REQ No 23866-268
REVISED PER REV. REQ. No 23866-463
NEXT ASSY:
FINAL ASSY:
SUPERSEDES:
SUPERSEDED:
DO NOT SCALE D'W'G.
CALC. WT.:
SCALE:
ACT. WT.:
THIS DRAWING IS THE PROPERTY OF COMBUSTION ENGINEERING, INC. NUCLEAR POWER DEPARTMENT WINDSOR, CONNECTICUT AND IS NOT TO BE REPRODUCED, OR USED TO FURNISH ANY INFORMATION FOR MAKING OF DRAWINGS OR APPARATUS EXCEPT WHERE PROVIDED FOR BY AGREEMENT WITH SAID COMPANY
CONTRACT NO.
DRAWN BY: P. MULCAHY 9-11-67
CHECKER: J. GUILLOTTE 1-3-68
APPROVALS:
A. YUILE
R. COCHRAN
G. SHULTZ
FORT CALHOUN REACTOR REGULATING SYSTEM BLOCK DIAGRAM
CE DWG. NO. D-23866-419-001
O.P.P.D. FIG. NO. 7.4-1
SHEET OF
REV.
OPPD FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION
REV. SH. 19974
FILE 1375
APVD H.D.K. 03-11-99
REV 8



NOTES:
① THIS DRAWING SHOWS THE OVERALL BASIC STRUCTURE OF THE PLANT ELECTRICAL DISTRIBUTION SYSTEM.

FORT CALHOUN STATION

SIMPLIFIED ONE LINE DIAGRAM PLANT ELECTRICAL SYSTEM

DWG. USAR-FIG 8.1-1

REV.SH. 27563

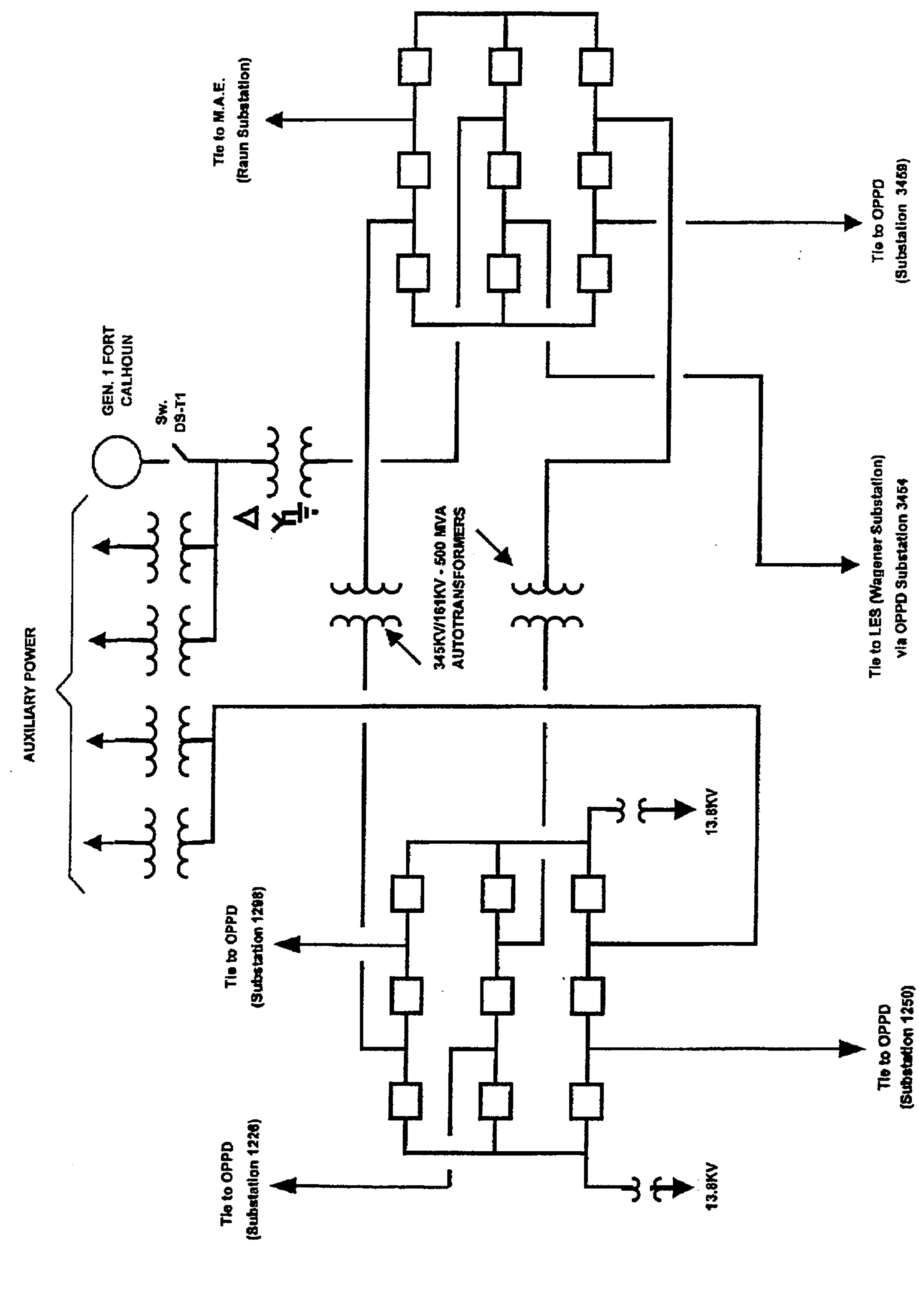
FILE 54187

APVD

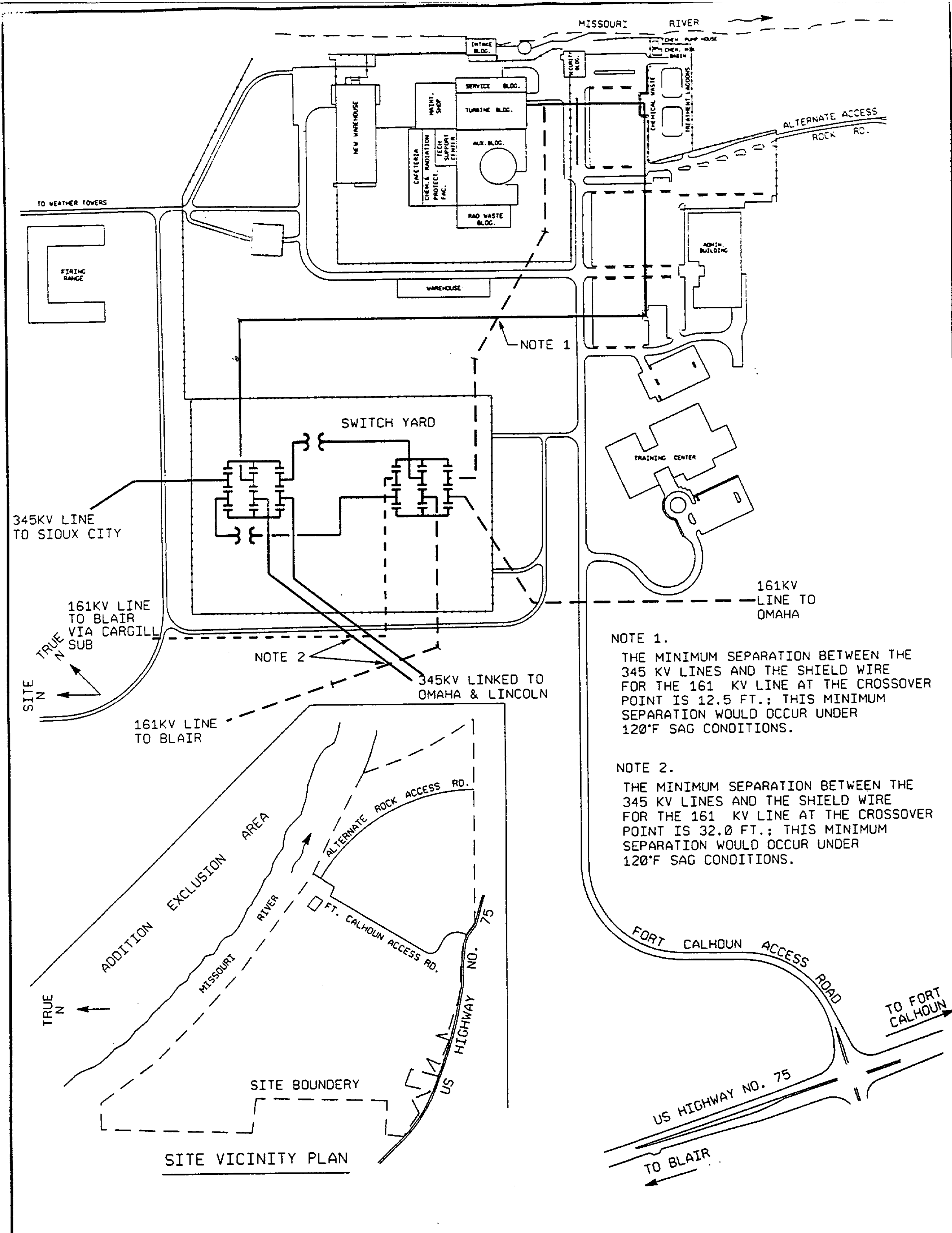
REV 10

LEGEND

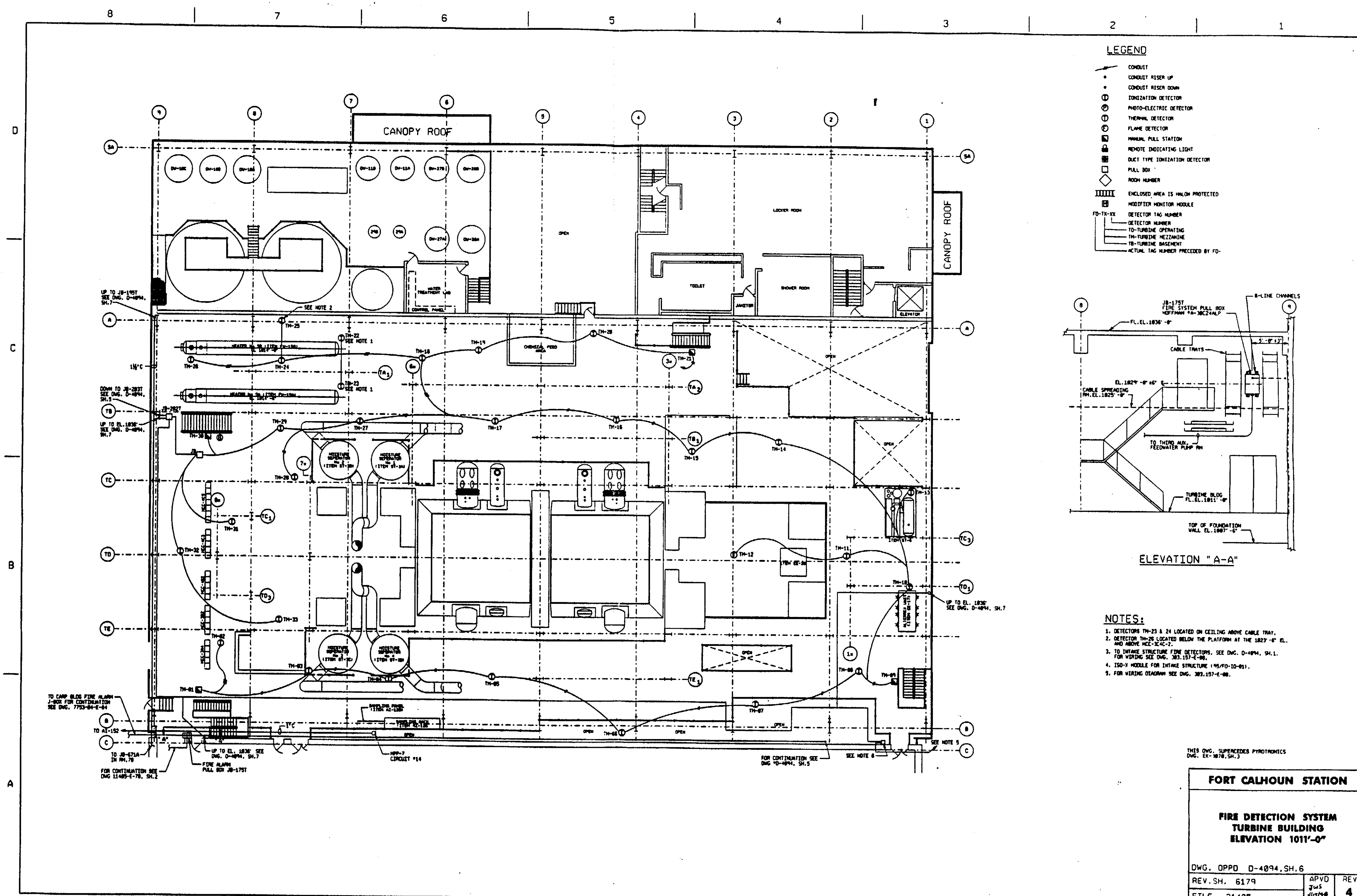
SF 6 CIRCUIT BREAKER



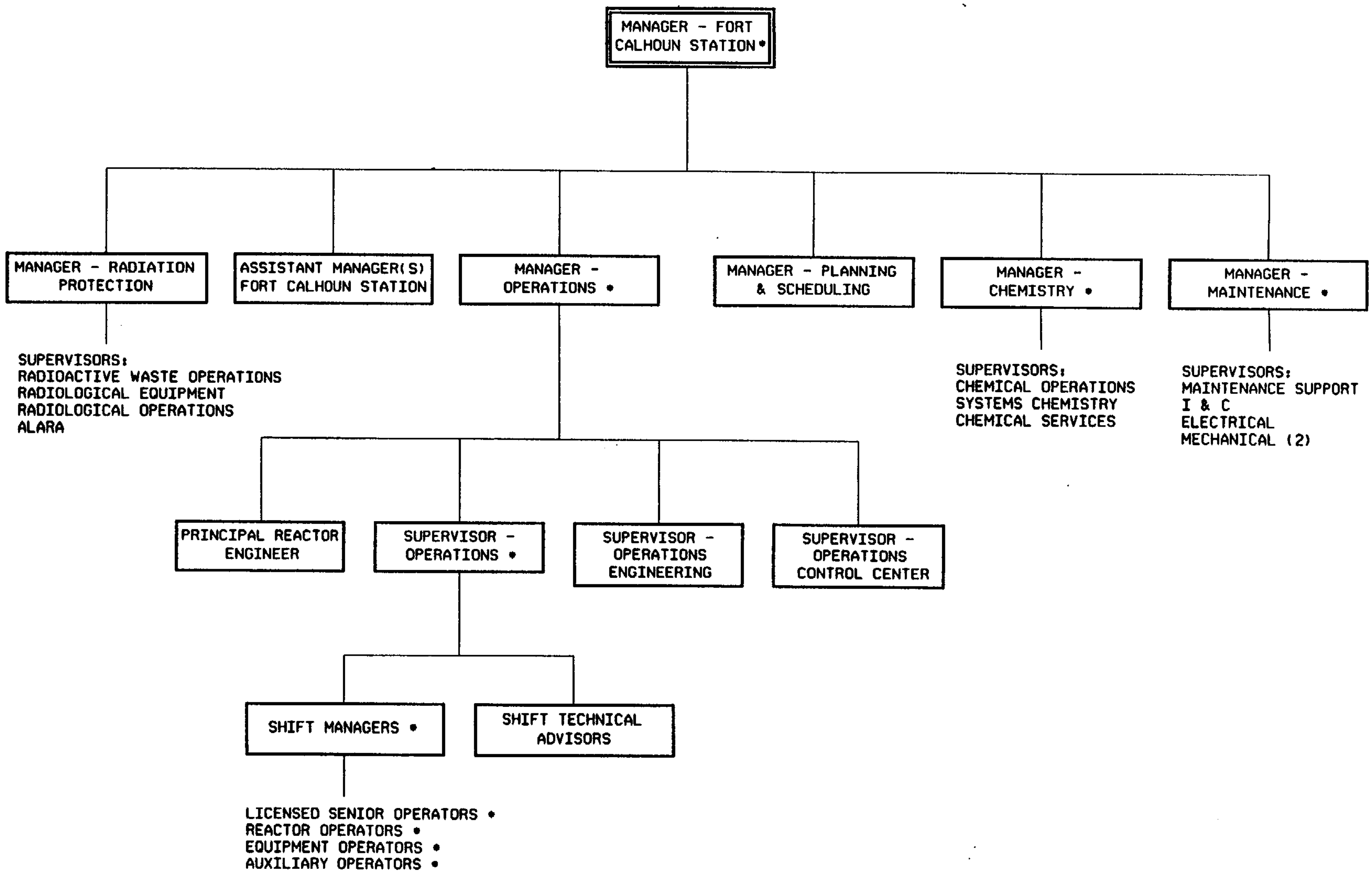
OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	ELECTRICAL NETWORK INTERCONNECTIONS	**USAR** **FIGURE 8.2-1** REV.SH. 34707 · FILE 36564 · APVD 1/13/00 · REV 5



OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	TRANSMISSION LINE ROUTING	**USAR** **FIGURE 8.2-2** REV.SH. 23689 FILE 36565 7/13/00 REV 7


LEGEND
CONDUIT
CONDUIT RISER UP
CONDUIT RISER DOWN
IONIZATION DETECTOR
PHOTO-ELECTRIC DETECTOR
THERMAL DETECTOR
FLAME DETECTOR
MANUAL PULL STATION
REMOTE INDICATING LIGHT
DUCT TYPE IONIZATION DETECTOR
PULL BOX
ROOM NUMBER
ENCLOSED AREA IS HALON PROTECTED
MODIFIER MONITOR MODULE
FO-TX-XX DETECTOR TAG NUMBER
DETECTOR NUMBER
TO-TURBINE OPERATING
TM-TURBINE MEZZANINE
TB-TURBINE BASEMENT
ACTUAL TAG NUMBER PRECEDED BY FO-
ELEVATION "A-A"
NOTES:
1. DETECTORS TM-23 & 24 LOCATED ON CEILING ABOVE CABLE TRAY.
2. DETECTOR TM-26 LOCATED BELOW THE PLATFORM AT THE 1023'-6" EL. AND ABOVE MCC-3C4C-2.
3. TO INTAKE STRUCTURE FIRE DETECTORS, SEE DWG. D-4094, SH.1. FOR WIRING SEE DWG. 303.157-E-00.
4. ISO-X MODULE FOR INTAKE STRUCTURE (95/FO-IO-01).
5. FOR WIRING DIAGRAM SEE DWG. 303.157-E-00.
CANOPY ROOF
CANOPY ROOF
LOCKER ROOM
SHOWER ROOM
TOILET
JANITOR
ELEVATOR
OPEN
CHEMICAL FEED AREA
WATER TREATMENT LAB
THIS DWG. SUPERCEDES PYROTRONICS DWG. EX-3070,SH.3
FORT CALHOUN STATION
FIRE DETECTION SYSTEM
TURBINE BUILDING
ELEVATION 1011'-0"
DWG. OPPD D-4094,SH.6
REV.SH. 6179
FILE 21435
APVD
REV
4

1 FIGURE 8.59



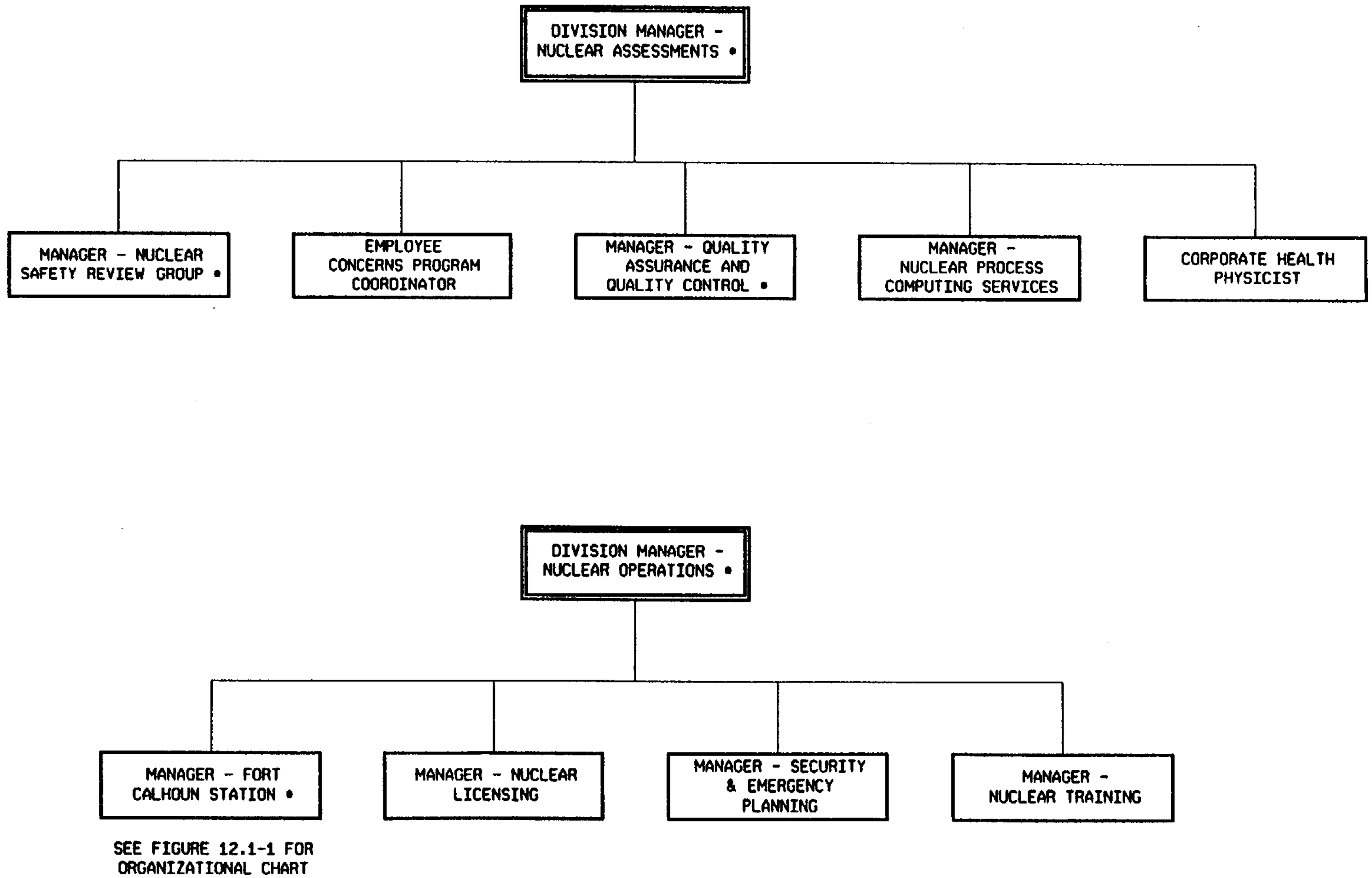
• POSITIONS AND/OR GROUPS POSSESSING FIRE PROTECTION RESPONSIBILITIES

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO. 1

STATION ORGANIZATION CHART

USAR
Fig. 12.1-1

REV.SH. 23624
FILE 36586
APVD MLE 11-1-99
REV 13



• POSITIONS AND/OR GROUPS POSSESSING FIRE PROTECTION RESPONSIBILITIES

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO.1

NUCLEAR ASSESSMENTS DIVISION &
NUCLEAR OPERATIONS DIVISION
ORGANIZATION CHART

USAR

Fig. 12.1-2

REV.SH. 23625

FILE 36587

APVD 11-1-99 MLE

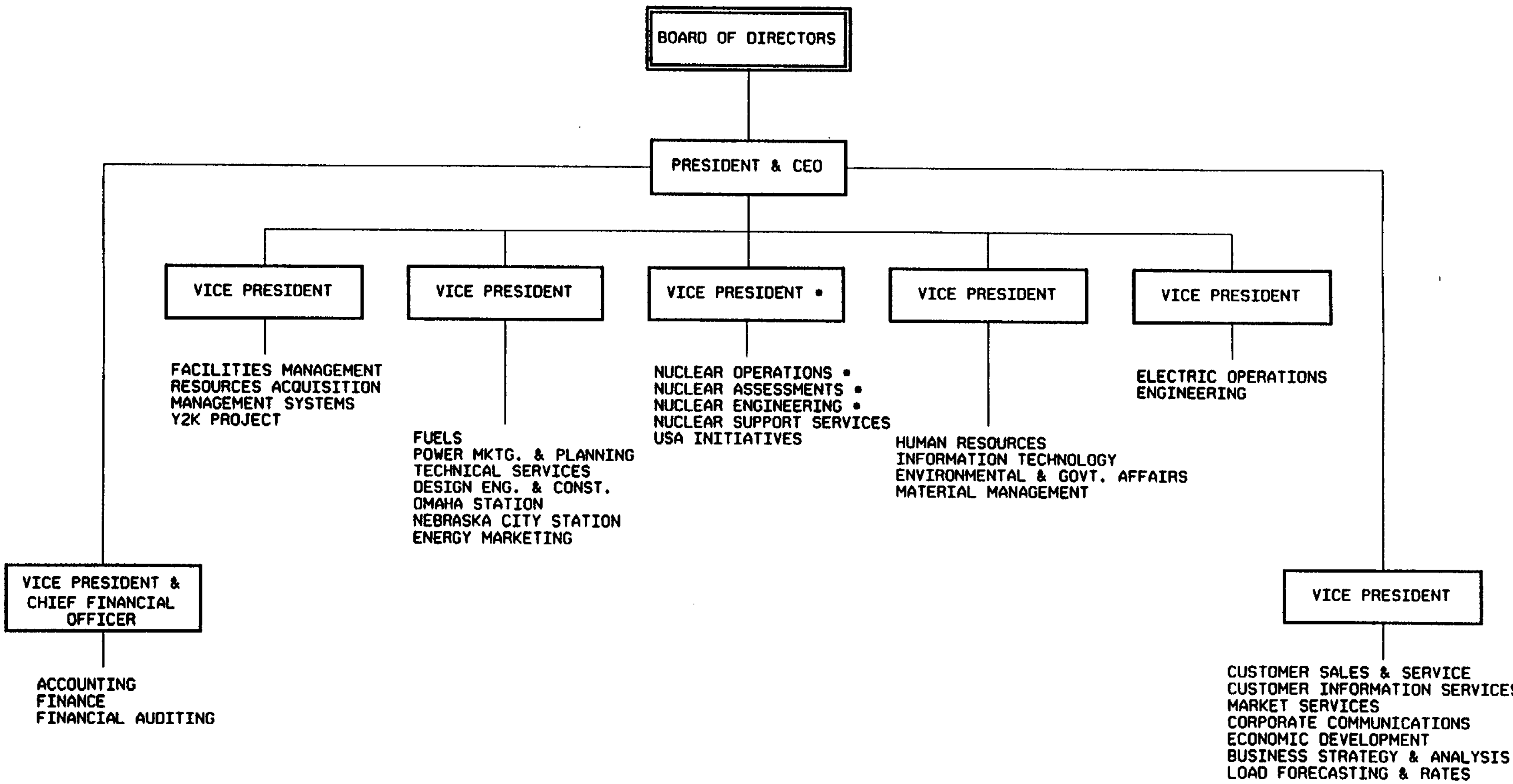
REV 12

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT NO.1

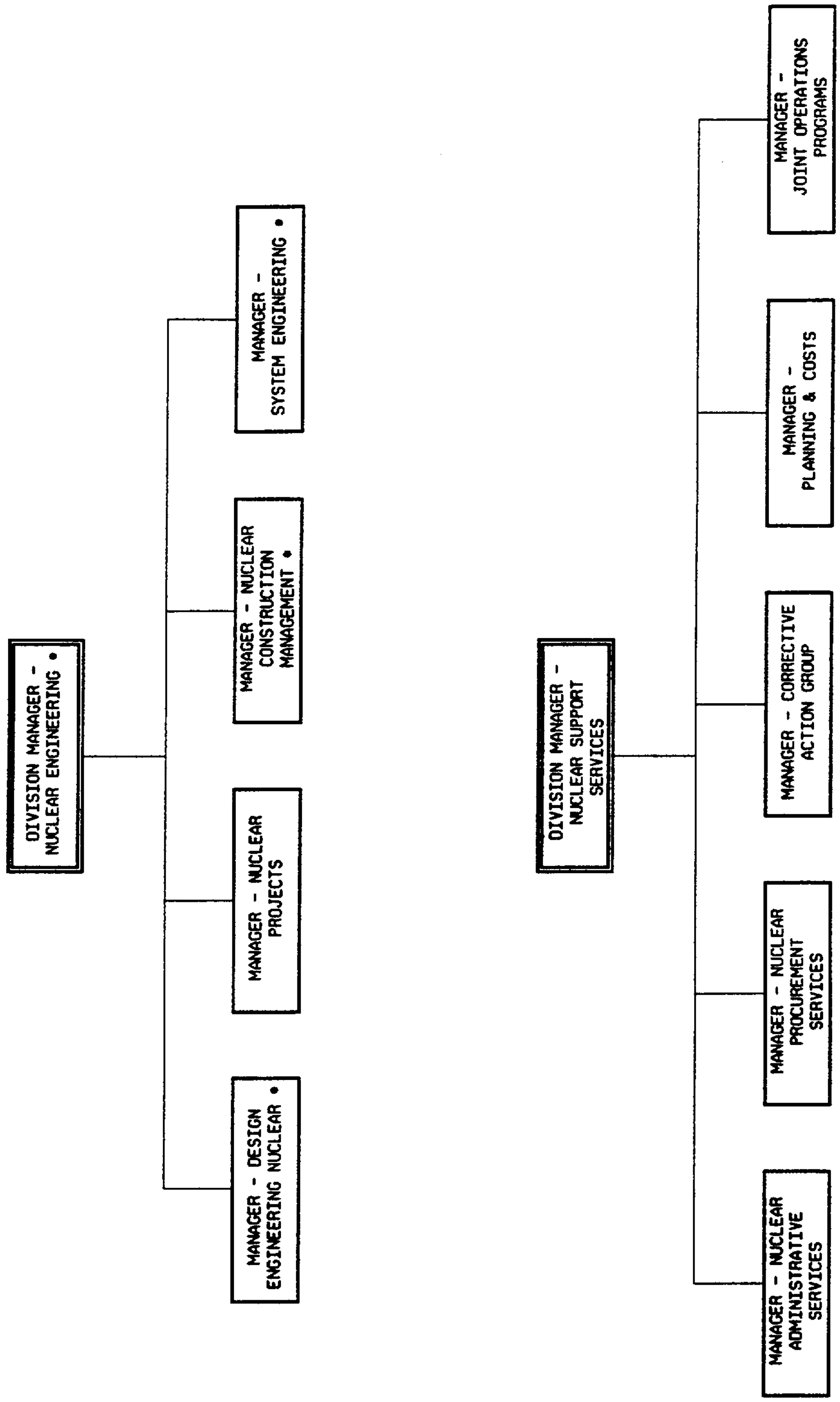
MANAGEMENT STRUCTURE

USAR
Fig. 12.1-3

REV.SH. 23626
FILE 36588
APVD MKE 11-1-99
REV 10

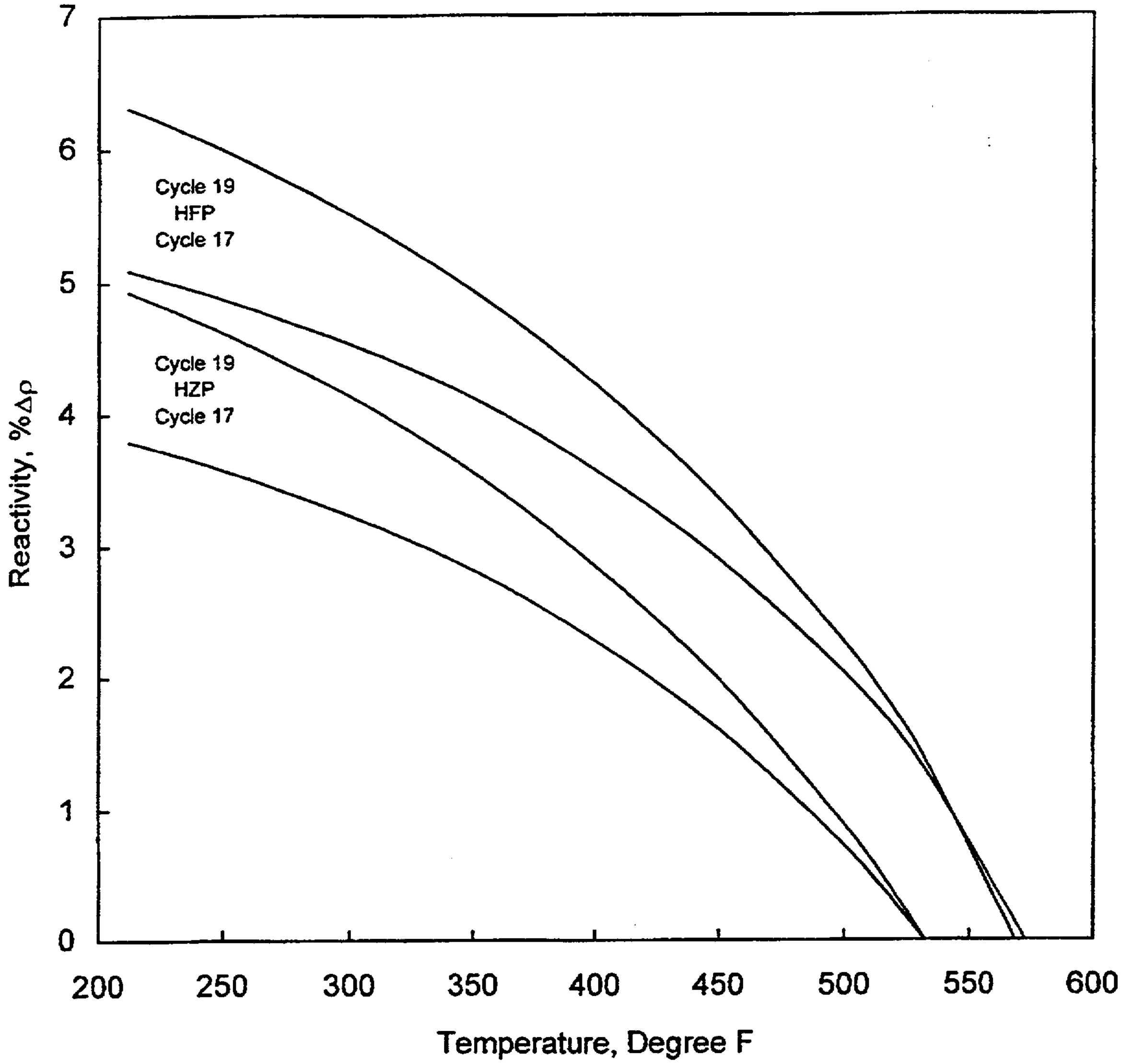


• POSITIONS AND/OR GROUPS POSSESSING FIRE PROTECTION RESPONSIBILITIES



• POSITIONS AND/OR GROUPS POSSESSING FIRE PROTECTION RESPONSIBILITIES

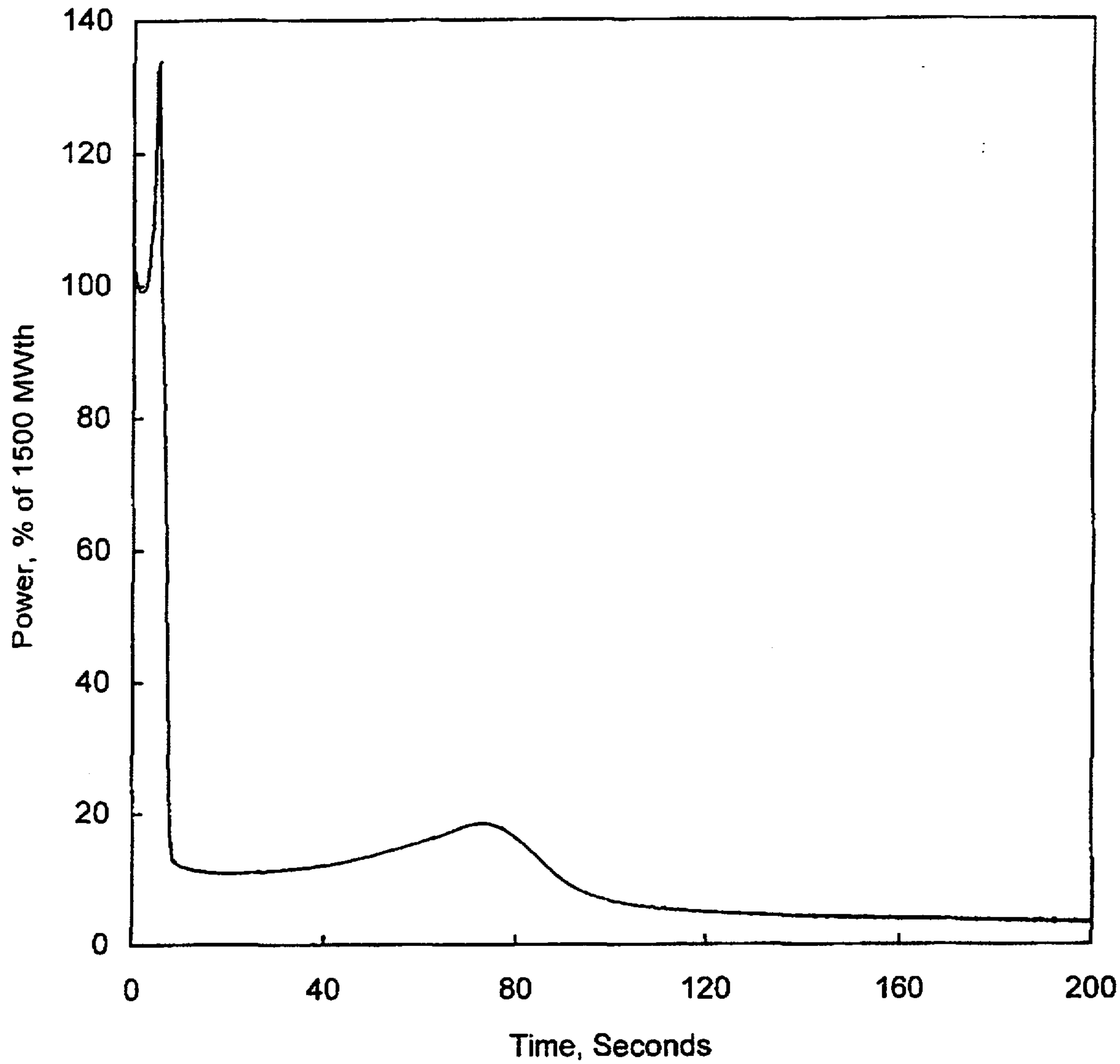
OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	NUCLEAR ENGINEERING DIVISION & NUCLEAR SUPPORT SERVICES DIVISION ORGANIZATION CHART	**USAR** Fig. 12.1-4 REV.SH. 25533 / FILE 36589 / APVD MLE 11-1-99 / REV 9



OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

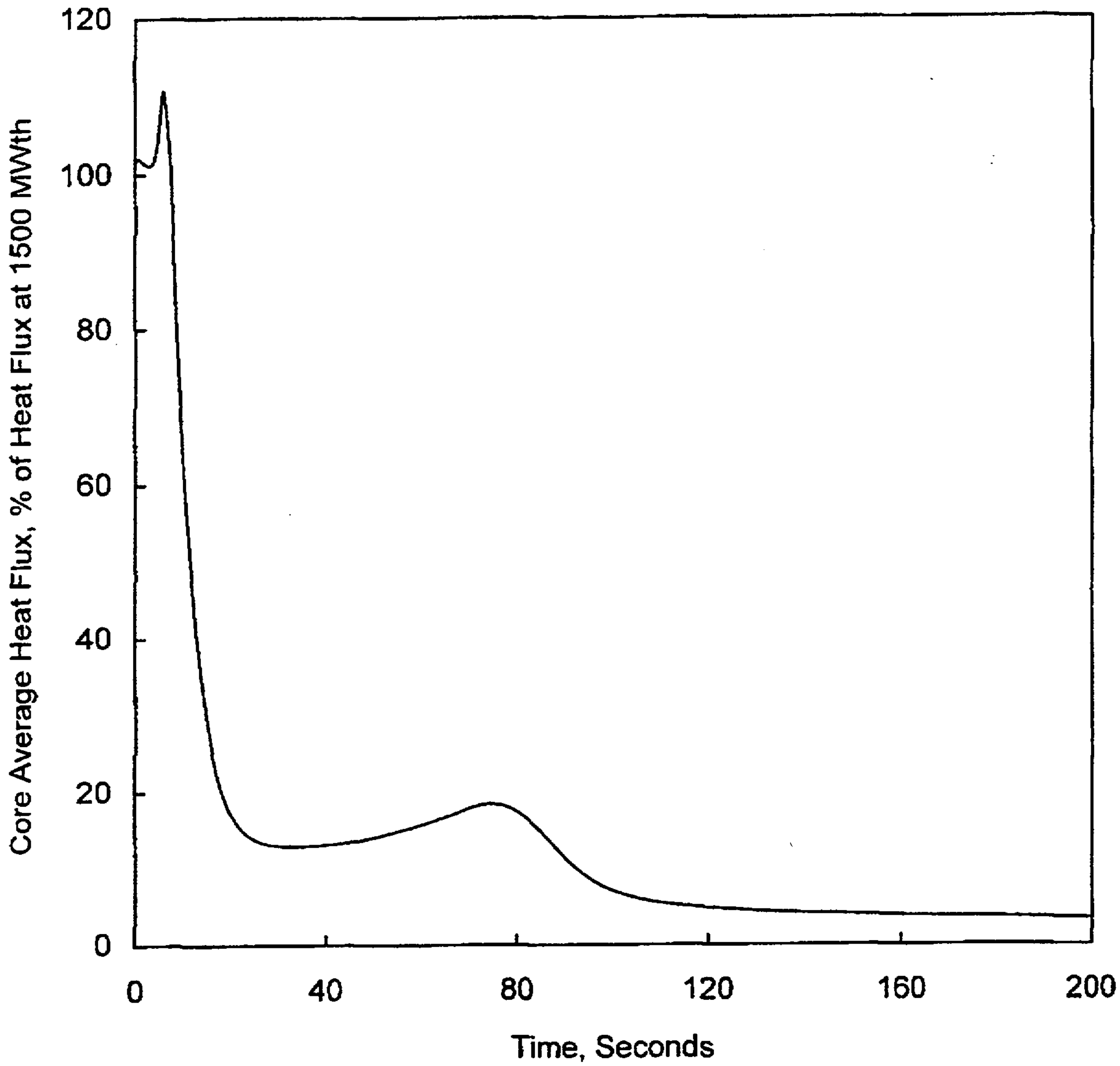
REV.SH. -----	APVD	REV
FILE 36630	11/19/99	1

Cycle 19 Main Steam Line Break Moderator Reactivity vs Temp	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-1



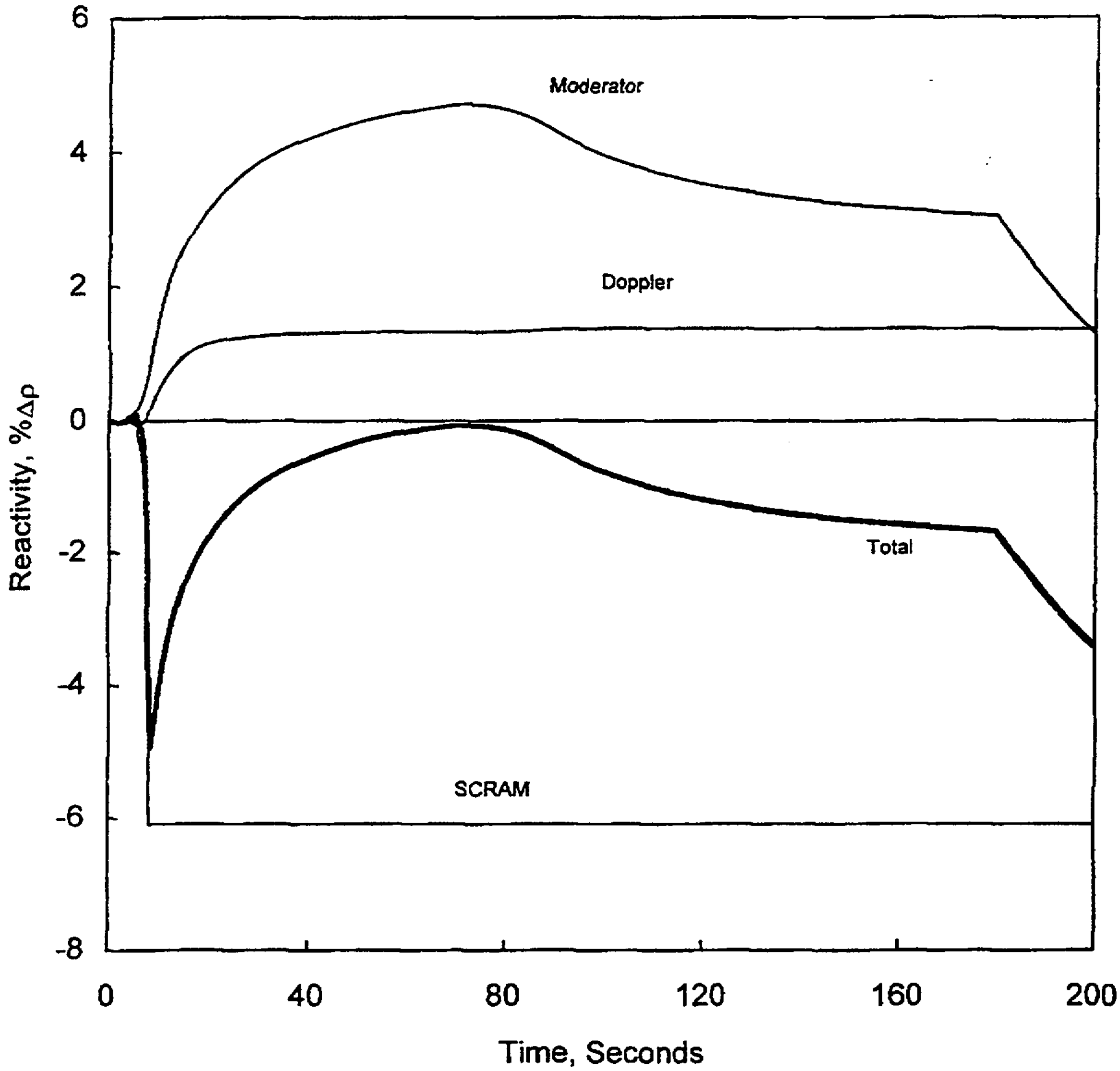
OPPD	FORT CALHOUN STATION	
	PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36631	11/19/99	1

Cycle 19 HFP Steam Line Break Core Power vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-2



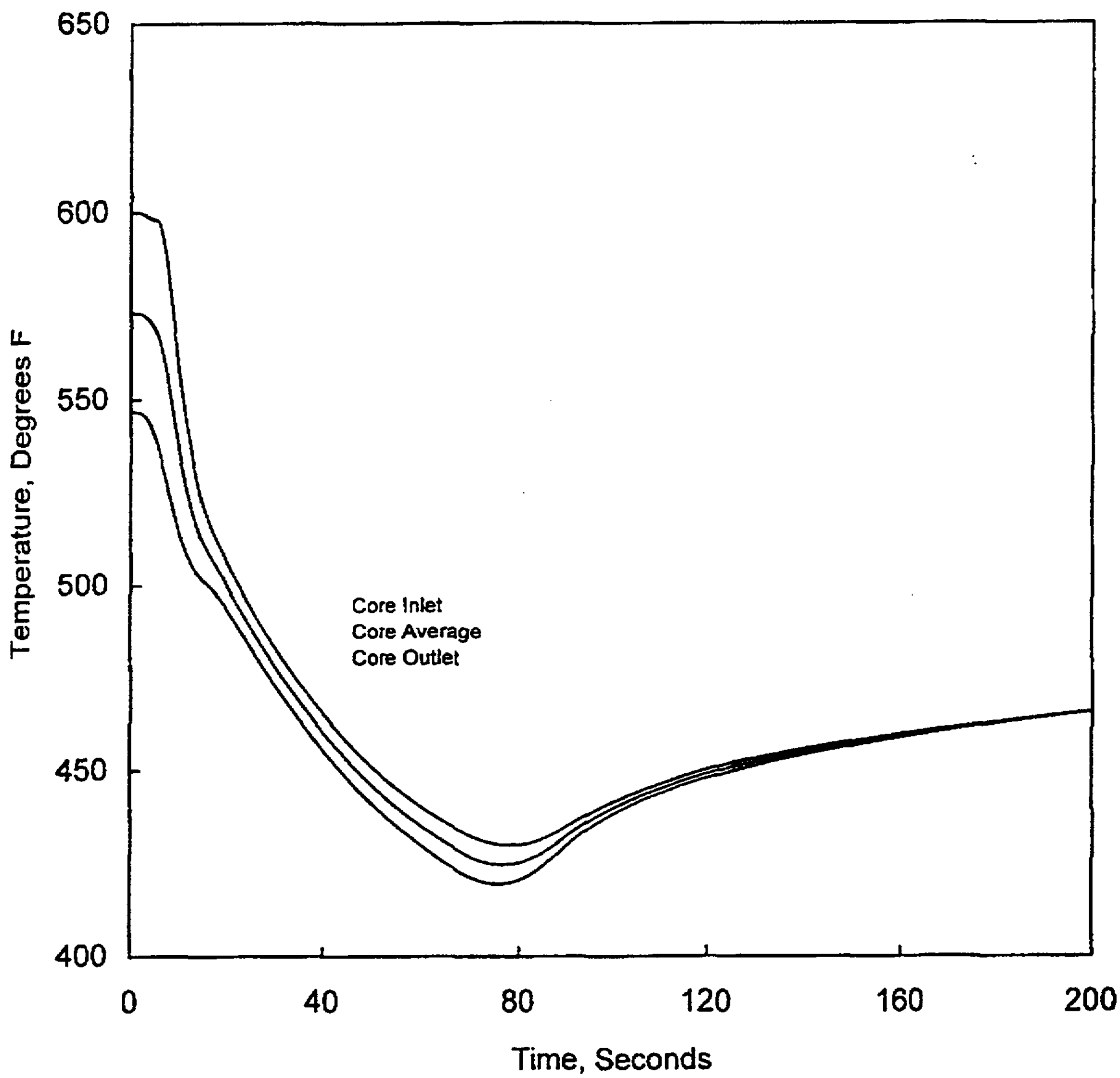
FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION		
REV.SH. -----	APVD mJz 11/19/99	REV 1
FILE 36632		

Cycle 19 HFP Steam Line Break Core Average Heat Flux vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-3



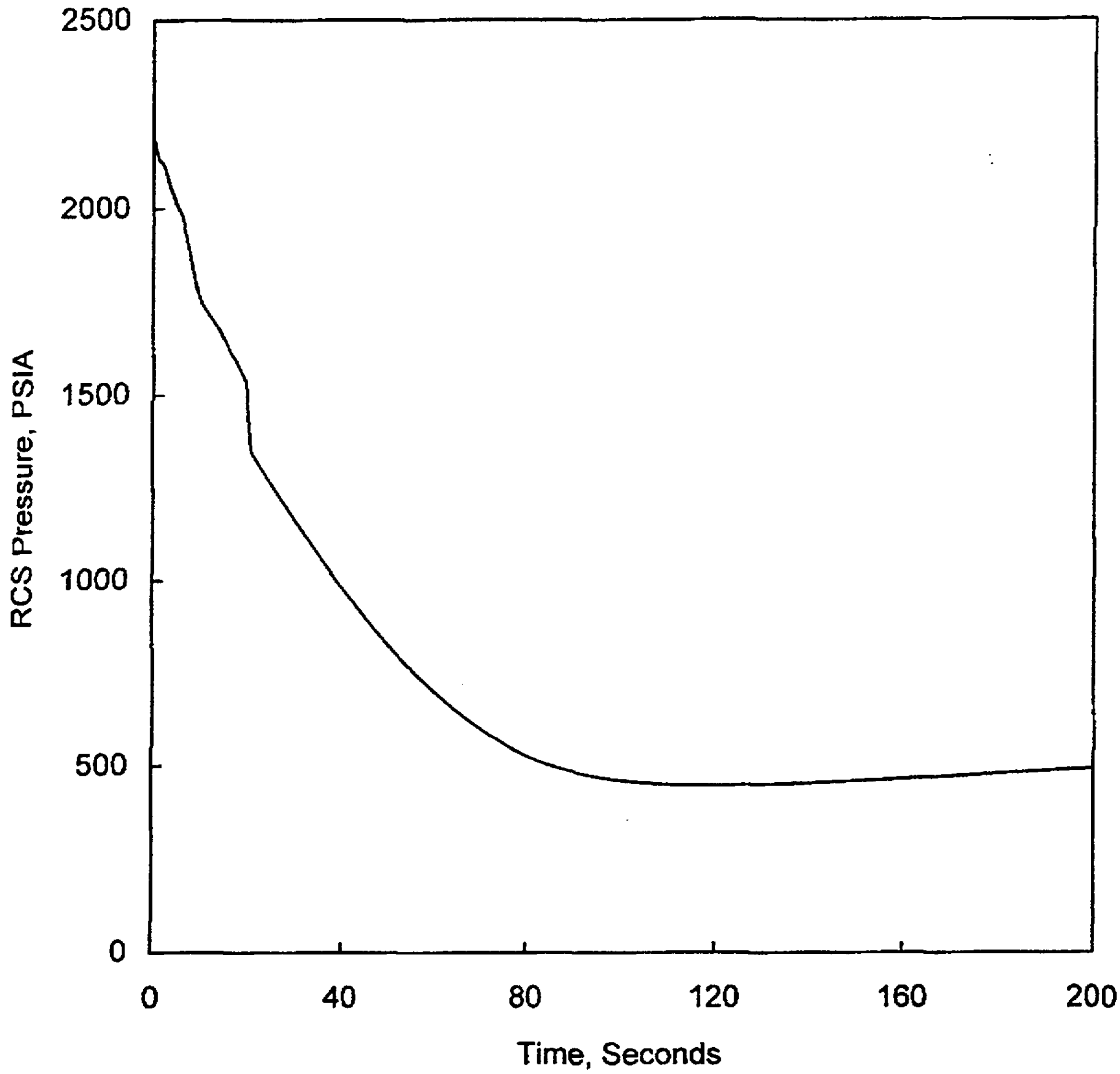
OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD mJL 11/19/95	REV 1
FILE 36633		

Cycle 19 HFP Steam Line Break Reactivity Changes vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-4



OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36634	MTL 11/13/99	1

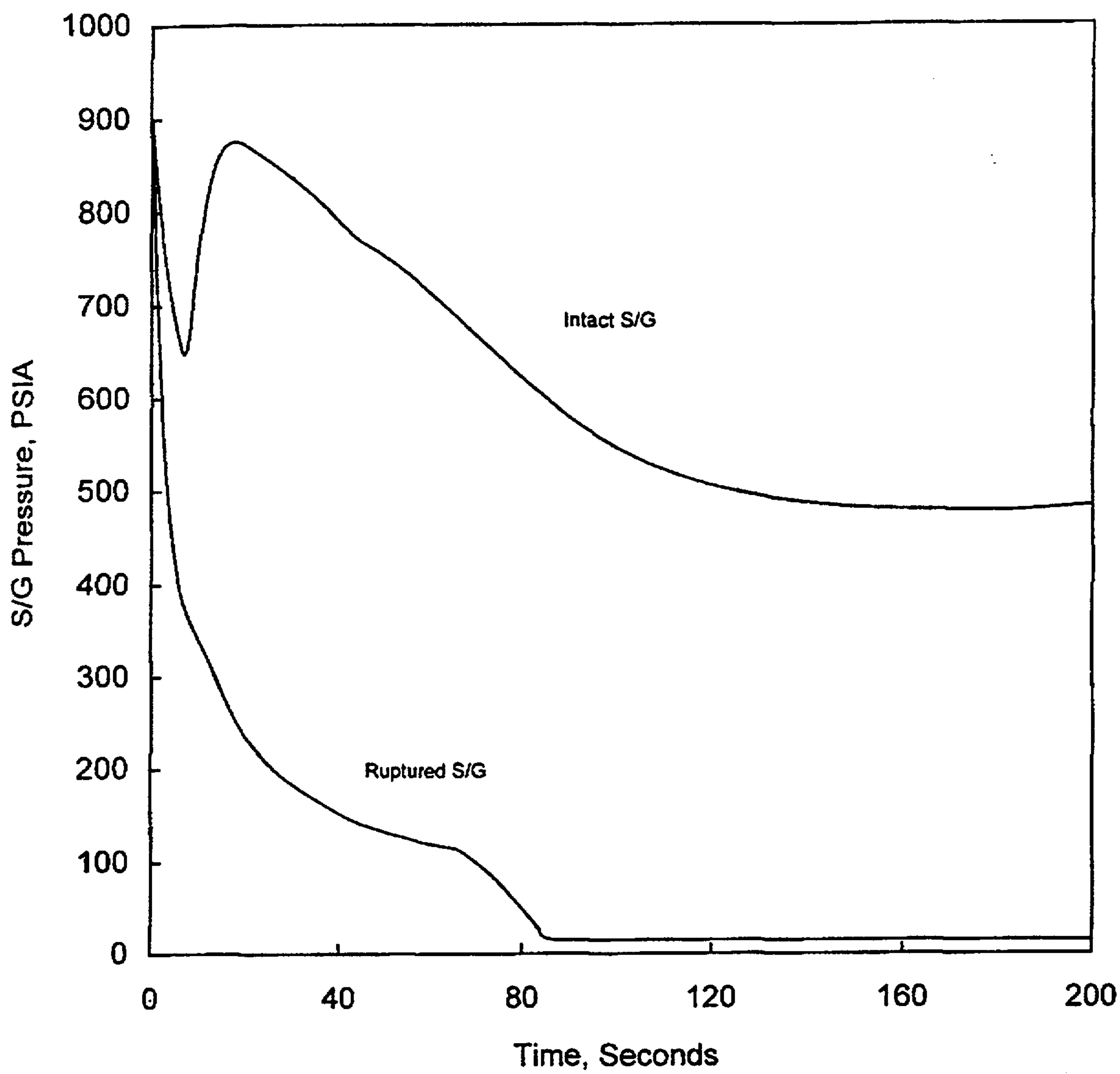
Cycle 19 HFP Steam Line Break Reactor Coolant Temps vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-5



OPPD FORT CALHOUN STATION
PRODUCTION ENGINEERING DIVISION

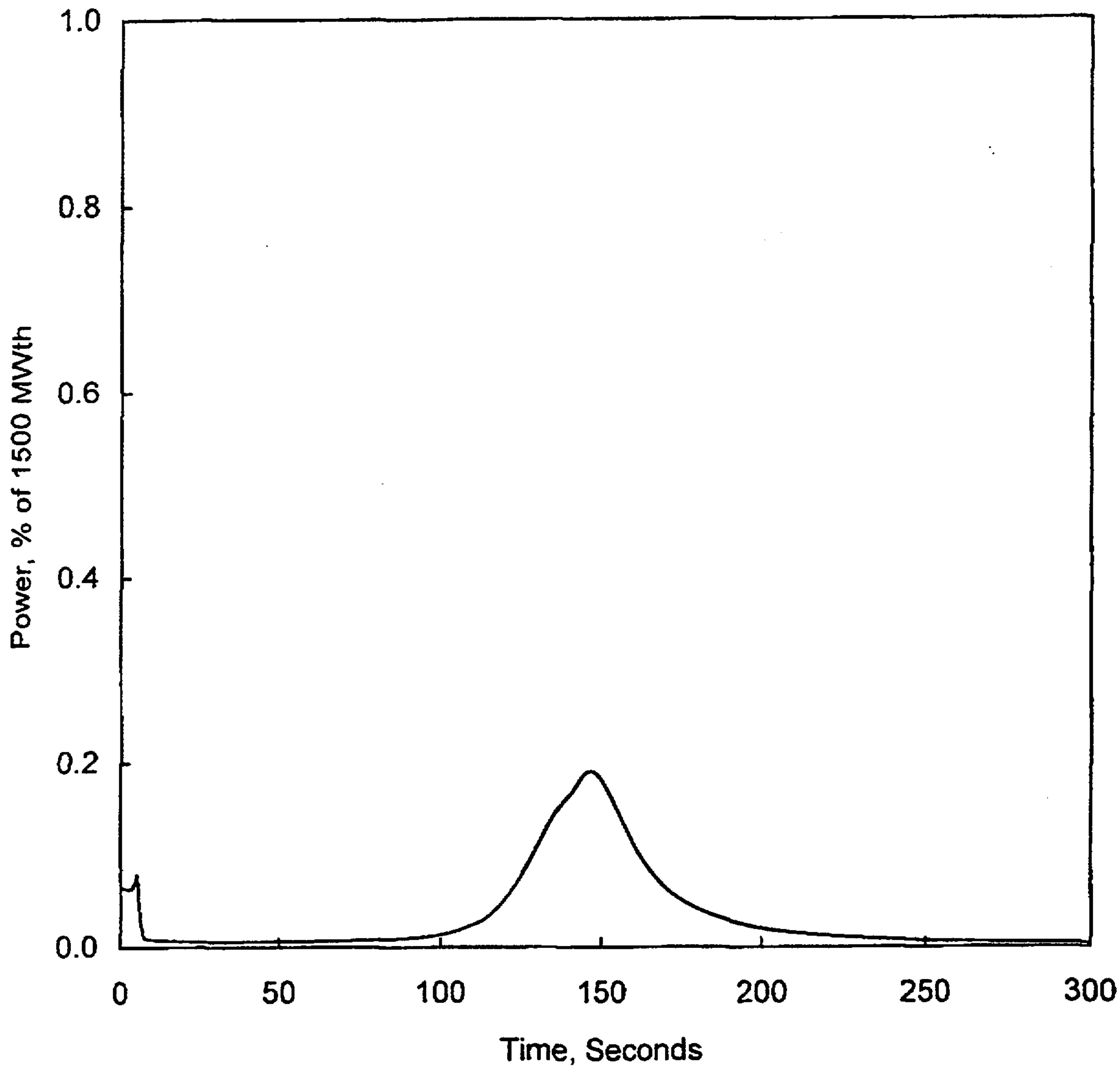
REV.SH. -----	APVD	REV
FILE 36635	11/19/99	1

Cycle 19 HFP Steam Line Break Reactor Coolant Pressure vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-6



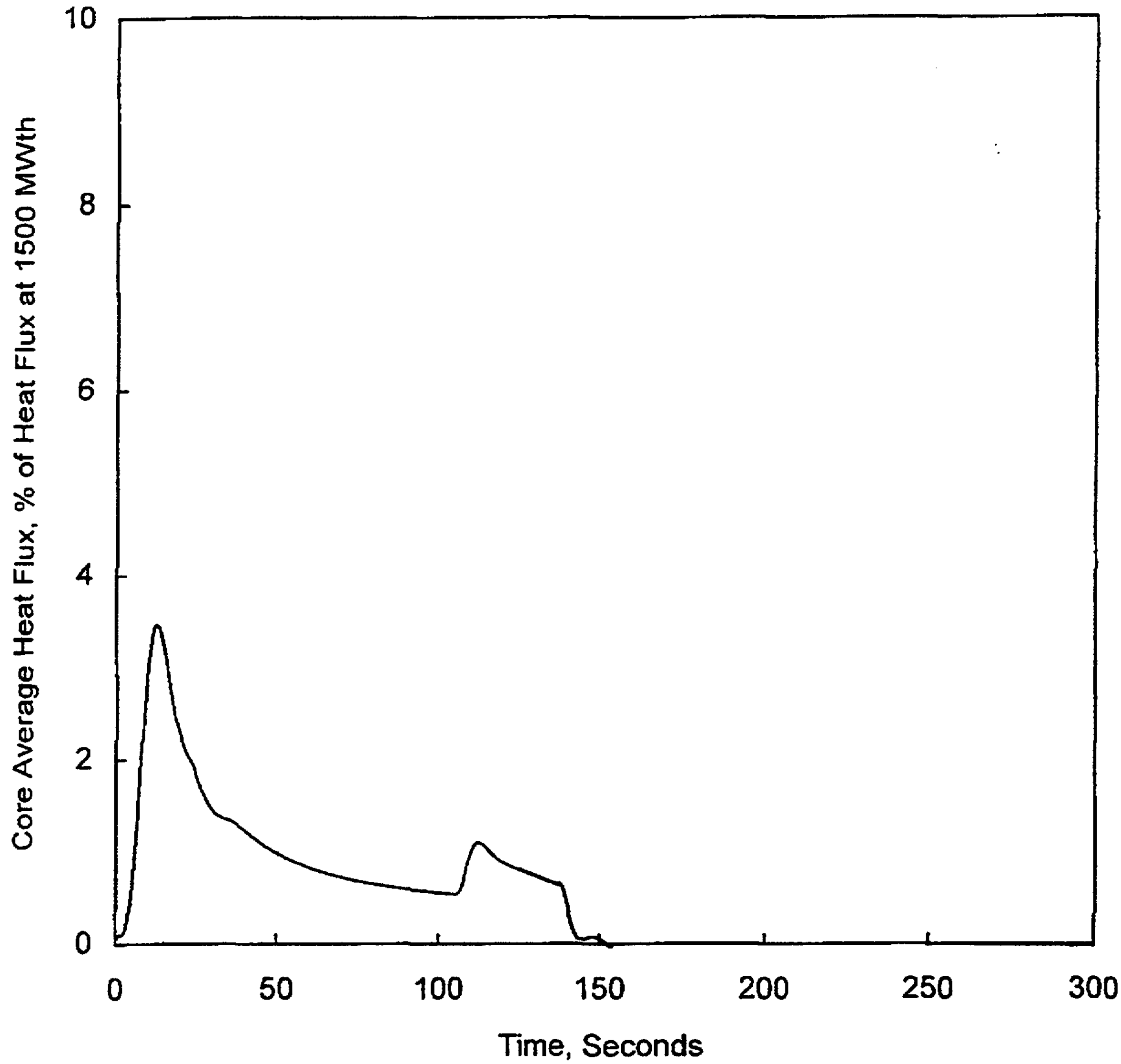
OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD mJL 11/19/95	REV 1
FILE 36636		

Cycle 19 HFP Steam Line Break Steam Generator Pressure vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-7



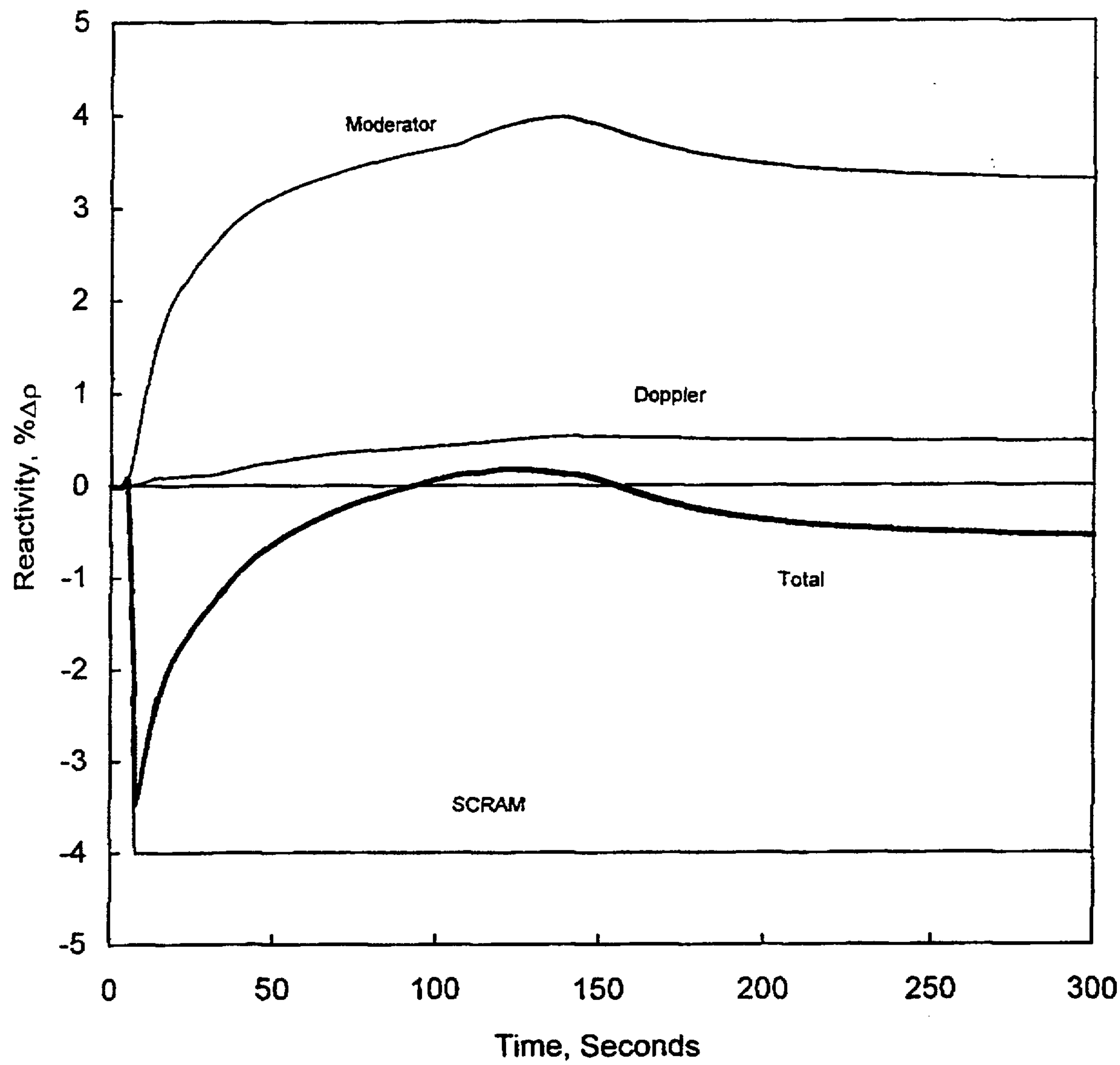
OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36637	MJR 11/13/95	1

Cycle 19 HZP Steam Line Break Core Power vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-8



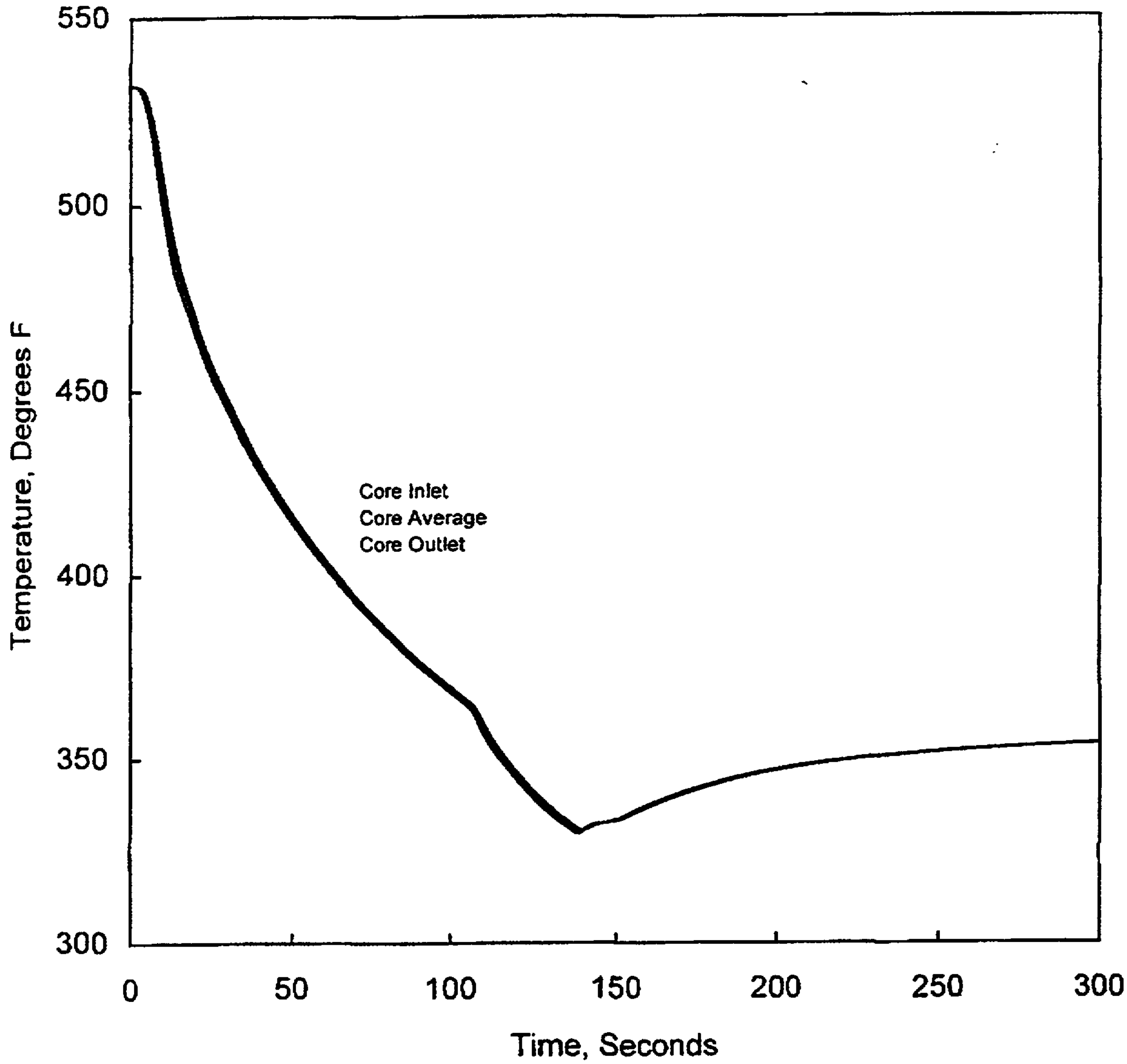
OPPD	FORT CALHOUN STATION	
	PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36638	mJZ 11/19/99	1

Cycle 19 HZP Steam Line Break Core Average Heat Flux vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-9



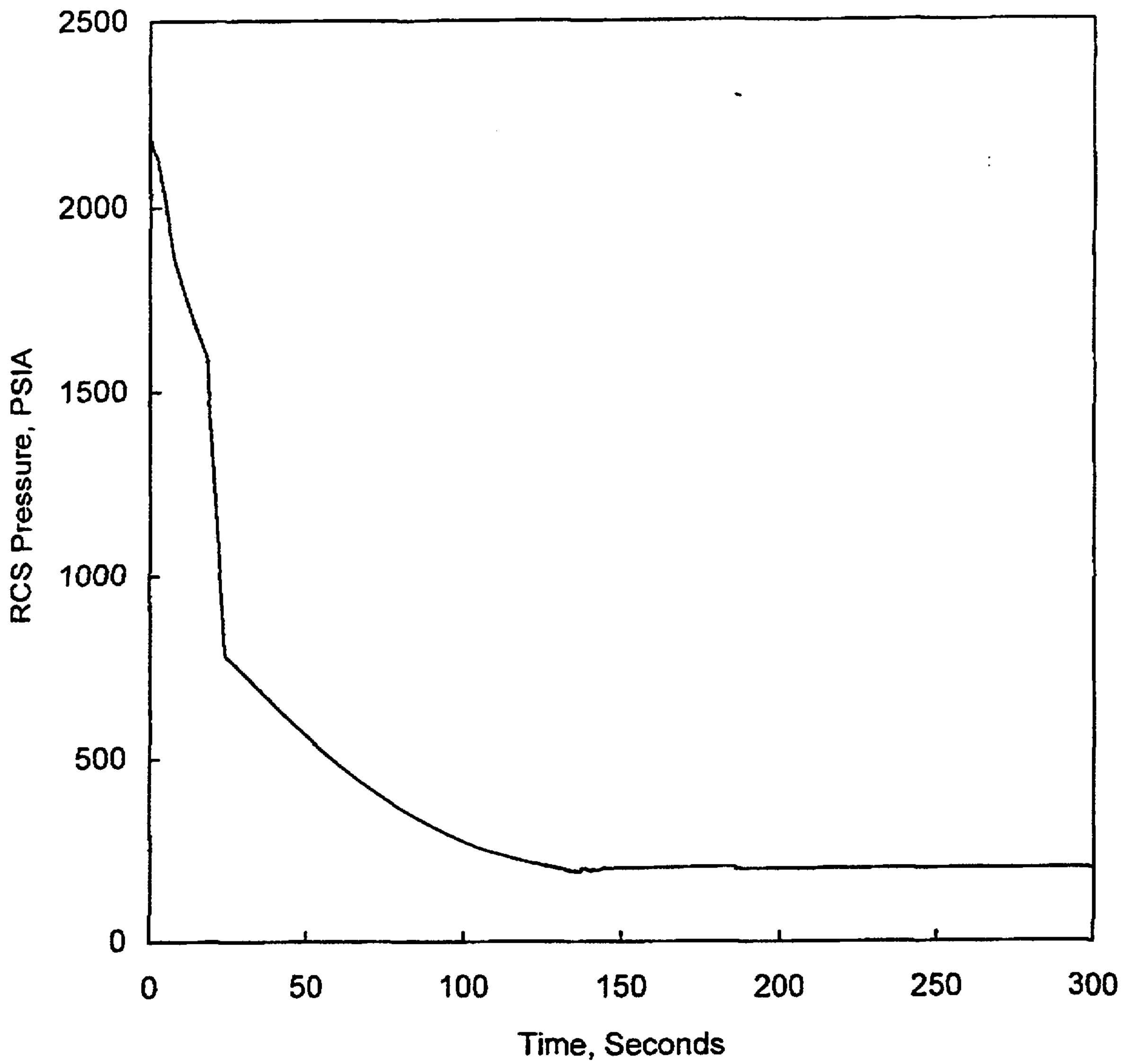
OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. ------	APVD	REV
FILE 36639	11/15/95	1

Cycle 19 HZP Steam Line Break Reactivity Changes vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-10



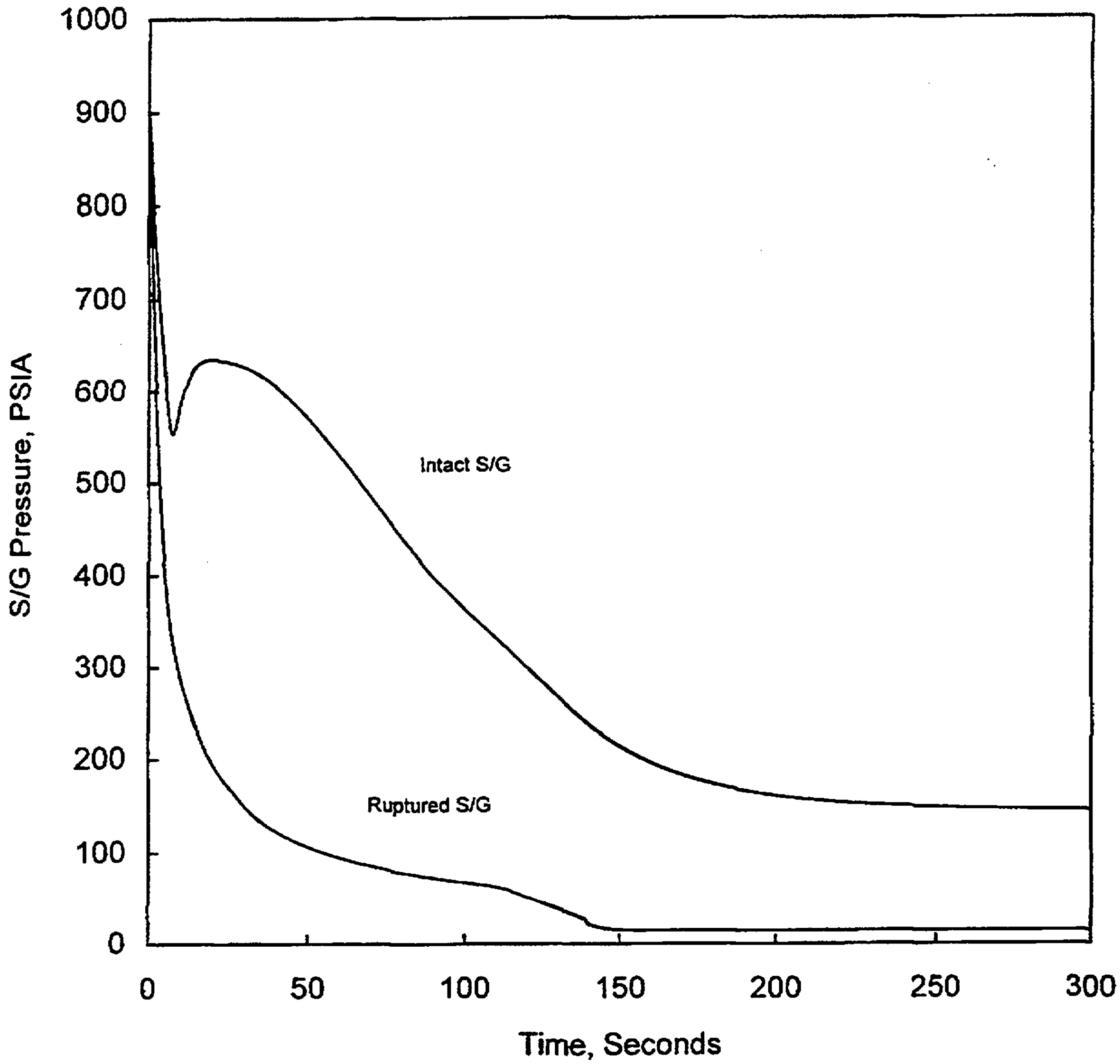
OPPD	FORT CALHOUN STATION	
	PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36640	MJL 11/19/96	1

Cycle 19 HZP Steam Line Break Reactor Coolant Temps vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-11



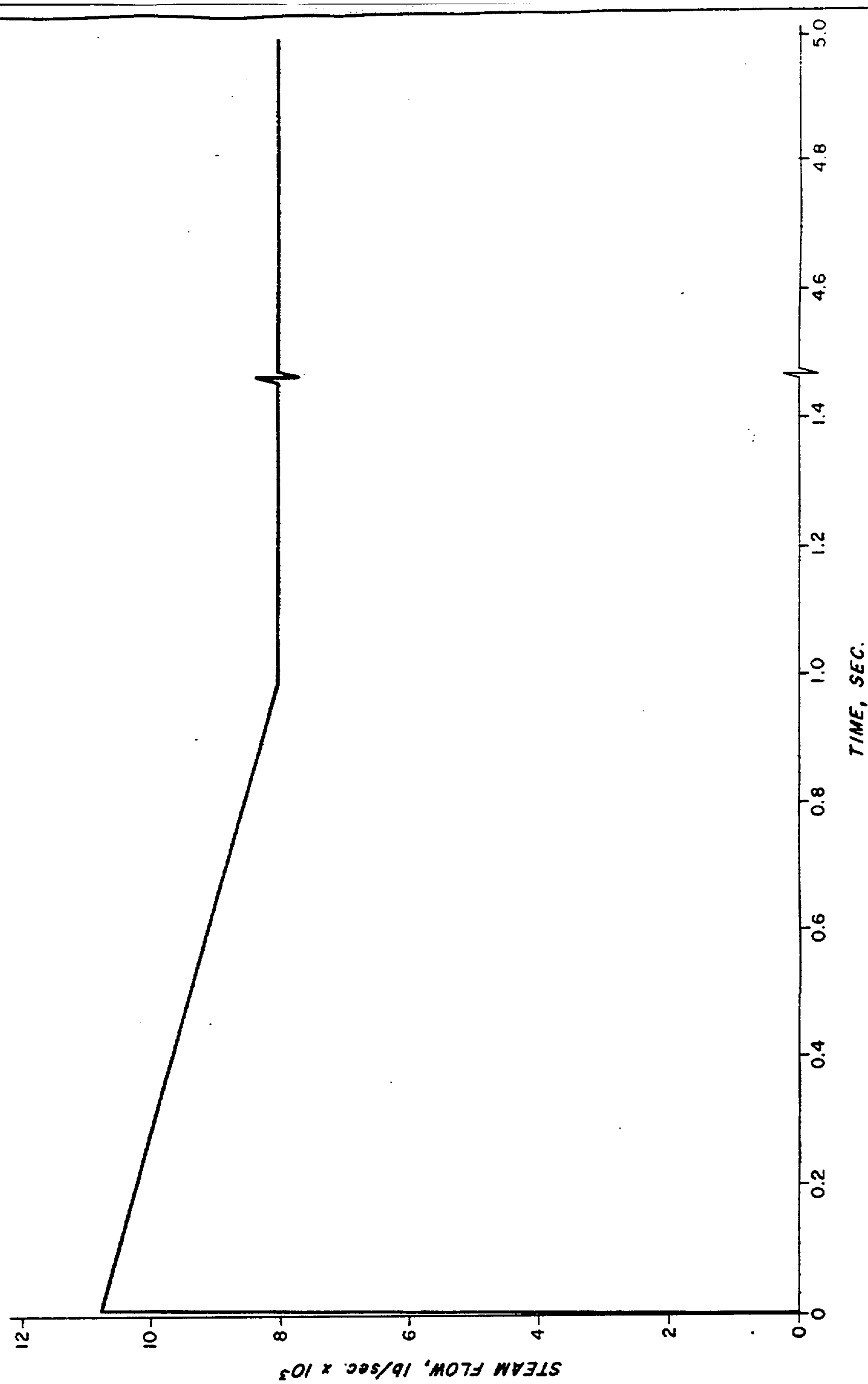
OPPD	FORT CALHOUN STATION PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36641	MNR 11/15/99	1

Cycle 19 HZP Steam Line Break Reactor Coolant Pressure vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-12

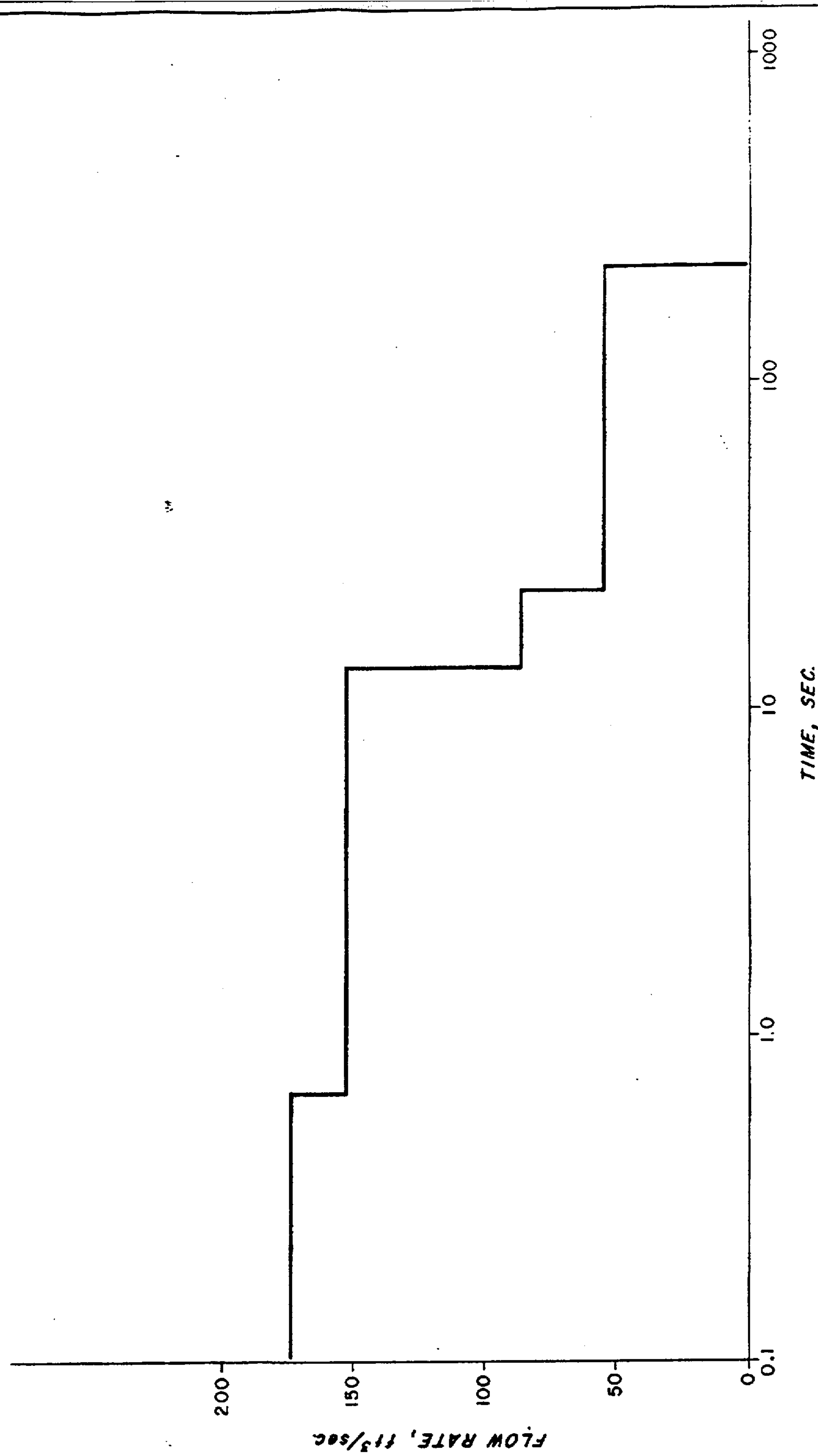


OPPD	FORT CALHOUN STATION	
	PRODUCTION ENGINEERING DIVISION	
REV.SH. -----	APVD	REV
FILE 36642	11/19/99	1

Cycle 19 HZP Steam Line Break Steam Generator Pressure vs Time	Omaha Public Power District Ft. Calhoun Station Unit No. 1	Figure 14.12-13



OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	STEAM FLOW INTO ROOM *81	**USAR** Fig. M-8 REV.SH. ------ / FILE 36796 / APVD DW 6/7/99 / REV 1



OMAHA PUBLIC POWER DISTRICT **FORT CALHOUN STATION** **UNIT NO. 1**	WATER FLOW RATE FROM FEEDWATER LINE BREAK	**USAR** Fig. M-10

REV.SH. ------	APVD LPW 6/1/99	REV 1
FILE 36798		