NT

SECTION 11

RADIOACTIVE WASTE AND RADIATION PROTECTION AND MONITORING

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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, and demineralization 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
)perating 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 gpm 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 $1x10^{-20}$ 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.

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 (χ/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 χ/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

<u>Relatively Highly Concentrated Gaseous Wastes</u>

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 gpm 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.

Radioactive Gases Released from Containment

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.

Shop Tests

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 produced at the station include process resins, used waste and process filters, dewatered ion exchange and filtration media, and miscellaneous solid wastes.

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
Waste Filters	5	each filter assembly
Spent Fuel Pool Filter	5	per cycle
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

<u>Radioactive Liquid and Spent Resins</u>

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.

<u>Miscellaneous Solid Waste</u>

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 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.

NT

SECTION 14

SAFETY ANALYSIS

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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 Framatome ANP Richland, Inc., the fuel vendor for Cycles 6 through 10, performed the reanalysis of all events described in this chapter of the USAR that 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 (OPPD) 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 CE-1 correlation 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 CE-1 correlation minimum DNBR limit, which was 1.22 for Cycle 9 and 1.18 for current analyses. The uncertainties for other factors such as the Doppler and moderator temperature coefficients are treated deterministically.

For Cycle 20, the DNB performance for limiting transients was evaluated using Framatome ANP Richland, Inc.'s (the fuel vendor for Cycle 20) methodology with the XCOBRA-IIIC fuel assembly thermal-hydraulic code. The limiting assembly DNBR calculations were performed using the approved HTP Correlation, which has a safety limit of 1.141 (Reference 14.1-11). Since Cycle 20 contains fuel designs from two vendors, the 95/95 DNBR safety limit, includes a 2% mixed-core penalty (Reference 14.1-12). This methodology uses a deterministic application of uncertainties, thus the plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.1-10).

14.1.1 Identification of Occurrences and Accidents

The anticipated operational 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 20 analyses:

	Cycle 6	Cycle 20
Reactor Power Level, MWt	1530	1500
Core Inlet Temperature, °F	547	545
Minimum Pressurizer Pressure, psia	2053	2075
RCS Flow Rate, gpm	190,000	206,000
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. These parameter values were adjusted to account for measurement uncertainties in the minimum DNBR calculations. Steam generator pressure is dependent on the value of core inlet temperature and will vary depending on the analysis.

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. Pevious 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. The transient conditions were not re-evaluated for Cycle 20.

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 20 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 Seized Rotor event analysis for Cycle 20, however, was performed by Framatome ANP Richland, Inc. using the approved ANF-RELAP Non-LOCA Transient Analysis methodology (Reference 14.1-10). Thus, the ANF-RELAP plant transient thermal-hydraulic code was used versus CESEC-III.

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. For Cycle 20, the DNB performance for limiting transients was evaluated using Framatome ANP Richland, Inc.'s (the fuel vendor for Cycle 20) methodology with the XCOBRA-IIIC fuel assembly thermal-hydraulic code. The limiting assembly DNBR calculations were performed using the approved HTP Correlation, which has a safety limit of 1.141 (Reference 14.1-11). Since Cycle 20 contains fuel designs from two vendors, the 95/95 DNBR safety limit, inclused a 2% mixed-core penalty (Reference 14.1-12). This methodology uses a deterministic application of uncertainties, thus the plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.1-10).

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 20 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 +2°F for measurement error.
Primary Coolant 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 20 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 20 Input Value
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		
Integrated Radial Peaking Factor F_R^T	1.57	1.890
Reactor Coolant System Flow Rate, gpm	190,000	206,000 (1)
Core Flow Rate, fraction of RCS flow rate	0.9554	0.9546
Reactor Inlet Temperature, °F	547.0	545 (1)
Calculated average heat flux (2) Btu/hr-ft^2	176,210	177,975
Steam generators		
Calculated total steam flow (3) lb/hr	6.737×10^6	6.61×10^6 (4)(5)
Steam temperature, °F	525.2	N/A
Feedwater enthalpy, Btu/lb	423.11	421.4 (4)(5)

NOTES:

(1) Uncertainties accounted for in the 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.
(5) The plant transient simulations were not re-generated for Cycle 20.

The design parameter values representative of Framatome ANP RIchland, Inc. 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 20 (Ref. 14.1-9).

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 20 MTC is the Technical Specification value for BOC and the COLR limit value for EOC, which includes uncertainties. For Cycle 20, 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	0.377 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)	23,408*

* This value represents the maximum number of fuel rods that can be loaded into the core. This value is adjusted based on the number of stainless steel rods in the core. This value may also vary based on the fuel types used in a specific core design since some designs use fuel displacing shims.

Table 14.1-4 - "Fort Calhoun Reactivity Data"

Parameter	Cycle 6 Value BOC	Cycle 6 Value EOC	Cycle 20 Value BOC	Cycle 20 Value EOC
Moderator temperature coefficient ($\Delta\rho$/Fx10^{-4})	+ 0.5	-2.3	+0.5	-3.05
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	0.0063	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 E-4350-595-1,"Fort Calhoun Unit 1, Cycle 20 Non-LOCA Transient MDNBRs," December 2000.

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-00-011, Rev. 0, "Cycle 20 Safety Analysis Base Cases."

14.1-10 ANF-89-151 (P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15, Events, Advanced Nuclear Fuels Corporation, May 1992.

14.1-11 EMF-92-153 (P)(A) and Suppliment 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, " March 1994.

14.1-12 XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to mixed core configurations," Revision 1, September 1983.

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.

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 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 DNBR for the hot channel as a function of time and axial position.

For Cycle 20, the results of the transient simulation for the limiting HFP Case were used as input into the XCOBRA-IIIC fuel assembly thermal-hydraulic code (Reference 14.2-24). Like TORC, this code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hyraulic model of the limiting assembly (representing each sub-channel by a single "channel"). The limiting assembly DNBR calculations are performed using the approved HTP correlation (References 14.2-22, 23, and 26). The minimum DNBR safety limit includes a 2% mixed-core penalty (Reference 14.2-27).

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, with the exception of groups 3 and 4, which have a 40% overlap. The position of the groups under steady state conditions is a function of power level. (See Reference 14.2-6, Figure 2).

For the full power DNBR analysis, an MTC consistent with that utilized in Reference 14.2-3 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 LHR and zero power LHR and DNBR cases the most positive MTC is used to maximize the positive reactivity feedback from increasing coolant temperatures. To minimize negative reactivity feedback from increasing fuel temperature, the least negative Doppler coefficient of reactivity is used with the minimum multiplier. The initial RCS pressure is chosen to be 2100 psia, which corresponds to the nominal pressure and also the SCU pressure (References 14.2-4 and 5). These assumptions yield lower transient minimum DNBRs.

14.2.4.1 Hot Full Power Case

Table 14.2-1 contains a list of the initial conditions and assumptions including uncertainties 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 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 incident was partially reanalyzed for Cycle 20 to evaluate the DNB performance of the DNB-limiting HTP assembly in the Cycle 20 core. The reanalysis was limited to performing the minimum DNBR calculations using the HTP DNB correlation and a Cycle 20 DNB-limiting axial power distribution generated with Framatome ANP's set axial methodology (Reference 14.2-24). The CESEC plant simulations from the Cycle 19 analysis of the CEA Withdrawal incident (Reference 14.2-7) remain valid and were used as input into the minimum DNBR calculations (reference 14.2-25). The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.2-24).

The limiting at-power case for the Cycle 20 CEA Withdrawal incident results in a minimum DNBR value that is greater than the HTP correlation 95/95 safety limit plus a 2% mixed-core penalty (Reference 14.2-24).

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. The RPS 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 (Reference 14.2-7 and 25). 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 20 results in a minimum DNBR greater than the DNBR limit The analysis shows that although the peak linear heat rate limit of 22 kw/ft is exceeded for a very short time period, 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

When initiated from the LCOs at either HFP or HZP conditions, the CEA withdrawal incident for Cycle 20 will not produce a DNBR or a LHR (fuel centerline melt temperature) that 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.

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).

14.2-22 EMF-2062(P), Guidelines for PWR Safety Analysis, G104,025, "Uncontrolled Control Assembly Bank Withdrawal at Power (SRP-15.4.2), "June 1998.

14.2-23 ANF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992.

14.2-24 E-4350-595-1, "Ft Calhoun Unit 1, Cycle 20: Non-LOCA Transient MDNBRs, " December 2000.

14.2-25 EA-FC-00-028, Revision 0, "Cycle 20 Transients Summary."

14.2-26 EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994.

14.2-27 XN-NF-82-21 (P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Revision 1, September 1983.

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

System Parameter	Units	Value	Reference
Initial Core Power Level	MWt	1531.85(1)	14.2-7
Reactor Coolant System			
Initial Core Inlet Coolant Temperature	°F	545(1)	14.2-7
Initial RCS Flow Rate	gpm	202,500(1)(2)	14.2-2
Pressurizer Pressure	psia	2100(1)	14.2-7
Reactor Protective System			
VHPT Setpoint	% of rated thermal power	112.0	14.2-7
Reactivity Control System			
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	14.2-7
CEA Group Withdrawal Rate	in/min	46	14.2-7
CEA Holding Coil Delay	sec	0.5	14.2-2
System/Parameter			
Max. Reactivity Insertion Rate	$\%\Delta\rho$/Second	1.0	14.2-2

1. The plant transient simulations, which are based in Reference 14.2-7, were adjusted to account for power, temperature, pressure, and flow measurement uncertainty in the DNBR calculations for Cycle 20.
2. The initial RCS flow rate was adjusted to reflect the T.S. flow of 206,000 gpm minus the measurement uncertainty (i.e., 198,584 gpm) in the DNBR calculations for Cycle 20.

Table 14.2-2 - " 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 (HZP)"

System Parameter	Units	Value	Reference
Initial Core Power Level	MWt	1*	14.2-7
Reactor Coolant System			
Initial Core Inlet Coolant Temperature	°F	532*	14.2-7
Initial RCS Flow Rate	gpm	202,500	14.2-2
Pressurizer Pressure	psia	2100*	14.2-7
Reactor Protective System			
VHPT Setpoint	% of rated thermal power	30	14.2-7
Reactivity Control System			
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	14.2-7
CEA Group Withdrawal Rate	in/min	46	14.2-7
CEA Holding Coil Delay	sec	0.5	14.2-2
System/Parameter			
CEA Time to 100% Insertion (including Holding Coil Delay)**	sec	3.1	14.2-18

* 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
CEA Worth at Trip (HZP PDIL with ARO insertion is most limiting)	%Δρ	4.7791	14.2-19
Max Reactivity Insertion	%Δρ/second	1.0	14.2-2

Table 14.2-4 - " 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 DNBR	≥minimum DNBR Limit

14.3 BORON DILUTION INCIDENT

14.3.1 General

The boron dilution incident was reanalyzed for Cycle 20 in Reference 14.3-4. The reactivity and power distribution anomalies for Cycle 20 are bounded by the CEA withdrawal incident. The chemical and volume control system regulates both the chemistry and the quantity of coolant in the reactor coolant system. Changing the boron concentration in the reactor coolant system is a part of normal plant operation, compensating for long term reactivity effects such as fuel burnup, xenon buildup and decay, and plant cooldown. For refueling operations, borated water is supplied from the safety injection and refueling water tank.

Boron dilution is a manual operation, conducted under strict procedural controls which specify permissible limits on the rate and magnitude of any required change in boron concentration. Boron concentration in the reactor coolant system can be decreased either by controlled addition of unborated makeup water with a corresponding removal of reactor coolant (feed and bleed) or by using the deborating ion exchangers. The deborating ion exchangers are used for boron removal when the boron concentration is low, and the feed and bleed method becomes inefficient.

During normal operation, concentrated boric acid solution is mixed in the manual mode with primary makeup water (demineralized water) to achieve the concentration required for proper plant operation and added to the volume control tank as needed to maintain the proper level. To effect boron dilution, the makeup controller mode selector switch must be set to "Dilute" and the demineralized water batch quantity selector set to the desired quantity. When the specific amount has been injected, the demineralized water control valve is shut automatically.

Dilution of the reactor coolant can be terminated by isolation of the primary makeup water system or by stopping both the deaerated water booster pump and the charging pumps or by closing the charging isolation valves. A charging pump must be running in addition to a deaerated water booster pump for boron dilution to take place.

The chemical and volume control system is equipped with the following indications and alarm functions which will inform the reactor operator when a change in boron concentration in the reactor coolant system may be occurring:

a. Volume control tank level and high level alarm;
b. Letdown diverter valve position;
c. Makeup controller flow indication and alarms which alert the operator to flow deviation from the set value; or
d. Letdown flow temperature indication at outlet of regenerative heat exchanger.

Because of the procedures involved and the numerous alarms and indications available to the operator, the probability of a sustained or erroneous dilution is very low.

14.3.2 Applicable Industry and Regulatory Requirements

The boron dilution incident is an anticipated operational occurrence for which the DNBR and LHR SAFDLs can not be exceeded. This requirement is met by building in a large enough margin into the DNBR and LHR limiting conditions of operation (LCOs) to ensure that the SAFDLs are not exceeded during power operation mode (Mode 1), and by ensuring that sufficient time exists for the operator to take corrective actions to terminate the event before all of the minimum required shutdown margin is lost in modes 2 through 5. Currently, the acceptance criteria for the minimum dilution time intervals are 15 minutes for hot standby, hot shutdown, and cold shutdown modes (Modes 2-4), and 30 minutes for the refueling mode (Mode 5). These criteria are summarized in Table 14.3-1.

14.3.3 Analysis and Results

Although the possibility is remote, a boron dilution incident could occur either with the reactor shutdown or operating. The purpose of the boron dilution analysis is to ensure that there is a large enough "time interval" for the operator to manually terminate a boron dilution event for any mode of operation before SAFDL limits are violated. The analysis is performed in accordance with the methods established in References 14.3-1 and 14.3-3.

The "time interval" referred to in this analysis is defined as the minimum time required to lose all of the minimum required shutdown margin allowed for that mode of operation per the Technical Specifications. For modes 3 through 5, this "time interval" can also be referred to as the time to reach criticality since the reactor is assumed to be sub-critical by the minimum shutdown margin per the Technical Specifications. The minimum required shutdown margins required are 4%$\Delta\rho$ for modes 1 through 3, and 3%$\Delta\rho$ for mode 4. The refueling mode (mode 5) requires a minimum shutdown margin of 5%$\Delta\rho$.

Previous analysis, including that for Cycle 18 (Reference 14.3-2), calculated the time interval in which the minimum Technical Specification shutdown margin (SDM) is lost. These calculations were based on critical boron concentrations (CBC) and inverse boron worths (IBW) for specific operating modes. The Boron Dilution analysis for Cycle 20 consists of calculating a bounding CBC for modes 2 through 4b based on the minimum accepted time interval during which the prescribed SDM is lost due to an inadvertent boron dilution. This bounding CBC is compared with the cycle-specific calculated CBD to ensure sufficient margin. For mode 5, a direct calculation for the time interval is performed based on the refueling boron concentration (RBC).

Currently, the acceptance time interval is 15 minutes for hot standby, hot shutdown and cold shutdown modes (2 through 4), and 30 minutes during the refueling operations in mode 5. The time interval, t, for each mode of operation is calculated using the following equation:

$$t = MT \ln\left[\frac{CBC + (SDM * IBW)}{CBC}\right]$$

where,

- M = multiplier accounting for the effects of density differences between the RCS and the makeup water.
- T = boron dilution time constant which is the ratio of RCS minimum volume to the maximum makeup charging flow rate. This time constant has units of time.
- CBC = maximum critical boron concentration in the range of operation for the mode in evaluation (ppm).
- SDM = minimum required shutdown margin per the technical specification (%$\Delta\rho$).
- IBW = minimum inverse boron worth in the range of operation for the mode in evaluation in (ppm/%$\Delta\rho$).

This equation is solved for the bounding CBC value in terms of the acceptance criteria for the time interval, SDM, and IBW:

$$CBC_b = SDM \ * \ IBW * \left[\ e^{t/mt} - 1 \ \right]^{-1}$$

14.3.3.1 Dilution at Power (Mode 1)

Inadvertent charging of unborated primary makeup water into the reactor coolant system while the reactor is at power would result in a reactivity addition producing power and temperature increases which result in a reduction in the margin to both the DNBR and kW/ft SAFDL's.

Since the Thermal Margin/Low Pressure (TM/LP) trip system monitors the transient behavior of core power level and core inlet temperature, the TM/LP trip assures that the DNBR SAFDL is not exceeded for power increases less than the Variable High Power Trip (VHPT) setpoints.

For power excursions in excess of the VHPT, a reactor trip is actuated. The approach to the kW/ft SAFDL is terminated by either the Axial Power Distribution trip, VHPT or the TM/LP trip. For a boron dilution initiated from hot zero power critical, the power transient resulting from the slow reacting insertion rate is terminated by the VHPT prior to approaching the SAFDL's.

The boron dilution event is similar to and bounded by the CEA withdrawal event with the exceptions that the dilution transient has a slower reactivity insertion rate and lacks the local power peaking associated with a withdrawn CEA.

Alarms and/or indications that the event is taking place are the same as in Section 14.3.1. Because of the available alarms and indications, there is ample time and information available to allow the operator to take corrective action. Protracted, unidentified erroneous dilution is improbable.

14.3.3.2 Dilution at Hot Standby (Mode 2)

The RCS average temperature range for this mode is 515°F to 532°F with the reactor being critical (Technical Specifications definitions Ref. 14.3.9-1). The minimum required shutdown margin is 4%$\Delta\rho$ (Technical Specification). The input parameters assumed in the analysis are listed in Table 14.3-1. The minimum acceptable time interval for dilution to critical is 15.0 minutes. Alarms and indications that a dilution is taking place are the same as for the event at power except that an audible count rate indication is available. The results of the analysis are summarized in Table 14.3-2. As shown, the boron dilution in mode 2 is acceptable.

14.3.3.3 Dilution at Hot Shutdown (Mode 3)

Two cases were considered in this mode of operation. One case is for RCS temperature range of 515° to 532°F with the reactor being subcritical per the definition of hot shutdown mode in the technical specifications. The second case was for RCS temperatures above the cold shutdown temperature (above 210°F) all the way to 515°F. For both cases, a shutdown margin of 4%$\Delta\rho$ was utilized per the technical specifications. The input parameters assumed in the analysis are listed in Table 14.3-1. The minimum acceptable dilution to critical time interval is 15.0 minutes.

The calculations for this mode demonstrate acceptable results. The results are summarized in Table 14.3-2

14.3.3.4 Dilution at Cold Shutdown (Mode 4)

The cold shutdown, boron dilution event, was analyzed for two cases - one with the RCS at normal volume and the other with a partially drained volume. The second configuration may occur when the RCS is drained to the centerline of the reactor vessel outlet nozzles. To be conservative, the drained volume, excluding the cold leg volume, was utilized as the minimum RCS volume. For both cases, the minimum required shutdown margin is 3% $\Delta\rho$. The minimum acceptable time interval is 15.0 minutes. The input parameters assumed in the analysis are listed in Table 14.3-1.

The results of the analysis are summarized in Table 14.3-2. The results for both cases show that the boron dilution in mode 4 is acceptable. Rather than assume an ARO configuration, which is overly conservative, the partially drained system analysis uses a critical boron concentration that is derived with the Shutdown Groups A and B fully withdrawn and all Regulating Groups fully inserted. Additionally, the most reactive Regulating Rod is assumed to be in the fully stuck-out position. These assumptions are consistent with the Technical Specifications for cold shutdown conditions.

14.3.3.5 Dilution During Refueling (Mode 5)

The boron dilution event analysis for refueling conditions contained the following assumptions:

a. Reactor refueling has just been completed and the head is in place, but the coolant volume is just sufficient to fill the reactor vessel to the centerline of the hot legs with no cold leg inventory.

b. Demineralized water is added by the charging system at a maximum 3 pump charging rate of 132 gpm.

c. The minimum permissible boron concentration allowed by Technical Specifications for refueling exists, which is based on all CEAs withdrawn from the core. This is conservative since the CEAs are now moved with the fuel and not shuffled over the core.

These assumptions represent a shutdown condition wherein the core reactivity is the greatest, the water volume and total boron content is at a minimum, and the rate of dilution is the largest possible. Hence, this condition represents the minimum time to achieve inadvertent criticality in the event of uncontrolled boron dilution.

The minimum required shutdown margin for mode 5 is 5%$\Delta\rho$. The minimum acceptable time interval is 30.0 minutes. The input parameters assumed in the analysis are listed in Table 14.3-1. The results of the analysis are summarized in Table 14.3-2.

The dilution time from refueling boron concentration to critical allows approximately 33 minutes for the operator to acknowledge the audible count rate signal and makeup controller alarm prior to criticality. Corrective action can then be taken to isolate the primary makeup water source by closing valves and/or stopping the primary makeup water pumps or the charging pumps. With the control rods in the all-in position, significantly more time is required to achieve a critical condition.

Should dilution occur, the operator would have additional indirect indication of the condition from the volume control tank level alarms and from operation of the letdown diverter valves. Should the makeup controller fail to shut the primary makeup water stop valve, the operator also has control room indication and manual control of the makeup water flow.

14.3.4 Affected Plant Technical Specifications

The boron dilution incident analysis uses inputs from the following Technical Specifications:

- LSSS 1.3 Limiting Safety System Settings, Reactor Protective System
- LCO 2.8 Refueling Operations
- LCO 2.10.2 Reactivity Control Systems and Core Physics Parameters Limits

14.3.5 Affected Plant Systems

For this event, the affected plant systems are the reactor coolant system, reactivity control systems, chemical and volume control system, and the reactor protective system.

14.3.6 Limiting Parameters for Reload Analysis

Reevaluation of the boron dilution incident is required when one of the following conditions exists:

- Core physics parameters change (Inverse boron worth or critical boron concentration) is a nonconservative manner.

- A plant design modification is expected to cause a change to a pertinent system such as the chemical and volume control system (CVCS). For example, the boron dilution incident would need to be reevaluated if the maximum charging flow increased.

- A nonconservative change is made to a pertinent Technical Specification LCO such as the required shutdown margin or refueling boron concentration.

Any changes to parameters and/or technical specifications must result in dilution times to critical that are greater than the acceptance criteria stated in Table 14.3-1.

14.3.7 Conclusions

Because of the equipment and controls and the administrative procedures provided for the boron dilution operation, the probability of erroneous dilution is considered very small. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. For the hot standby, hot shutdown, cold shutdown, and refueling modes, the maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the dilution and take corrective action before all of the initially required shutdown margin is completely lost. The boron dilution event at power is less severe than and bounded by the CEA withdrawal event.

Table 14.3-1 - "Cycle 20 Assumed Input Parameters for Boron Dilution Incident Analysis"

Mode	Minimum Acceptance Criteria for Time to Lose SDM (min.)	Inverse Boron Worth (ppm/%$\Delta\rho$)	Charging Flow Rate (gpm)	RCS Volume (ft^3)	Minimum Required SDM (% $\Delta\rho$)
Hot Standby	15.0	-55	132	5700.0	4.0
Hot Shutdown RCS Temperature (515°F 532°F)	15.0	-55	132	5700.0	4.0
Hot Shutdown RCS Temperature (210°F 515°F)	15.0	-55	132	5700.0	4.0
Cold Shutdown Normal RCS Volume	15.0	-55	132	5700.0	3.0
*Cold Shutdown Minimum RCS Volume	15.0	-75	132	2026.8	3.0
Refueling	30.0	N/A	132	2026.8	5.0

* Shutdown Groups A and B out, all Regulating Groups inserted except most reactive rod stuck out.

Table 14.3-2 - " Results of Boron Dilution Incident "

Mode	Bounding CBC (ppm)	Calculated CBC (ppm)
(2) Hot Standby	3500.41	1722
(3) Hot Shutdown RCS Temperature (515°F to 532°F)	3500.41	1722
Hot Shutdown RCS Temperature (210°F to 515°F)	3574.28	1741
(4) Cold Shutdown - Normal RCS Volume	3323.90	1638
Cold Shutdown - Minimum RCS Volume	1587.98	1399
(5) Refueling*	N/A	1614

* The critical time interval was calculated directly from the RBC (2155.0 ppm) and calculated CBC (1614 ppm) and the value is 32.72 minutes.

14.3.8 Specific References

14.3-1 OPPD-NA-8303, Rev. 4 (January 1993) Omaha Public Power District Reload Core Analysis Methodology, "Transient and Accident Methods and Verification."

14.3-2 EA-FC-97-026, Rev. 0 Cycle 18 Boron Dilution Analysis

14.3-3 CENPSD-164-P, Rev. 1-P, "Part II, CE Transient Analysis Methods for Fort Calhoun Unit No. 1," September 1981.

14.3-4 EA-FC-00-024, Cycle 20 Boron Dilution Analysis, Revision 0.

14.3.9 General References

14.3.9-1 Fort Calhoun Station Unit 1, Operating License DPR-40 and Technical Specifications, Section 2.3(1)a, including amendments through Amendment 180, February 1997.

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.

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 greater than or equal to the minimum DNBR limit 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 $\geq$ 206,000 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 CESEC plant transient thermal-hydraulic code is used to simulate the overall response of the reactor coolant and steam systems during the transient (References 14.4-5, 6, 7, and 8). The CESEC code models neutron kinetics with fuel and moderator temperature feedback, the reactor control system, the reactor coolant system (RCS), and the main steam and feedwater systems.

Based on the overall core conditions calculated by CESEC at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hydraulic model of the limiting assembly (representing each sub-channel by a single "channel"). The limiting assembly DNBR calculations are performed using the approved HTP correlation (References 14.4-2, 14.4-3, and 14.4-11). The DNBR safety limit includes a 2% mixed-core penalty (Reference 14.4-12).

Table 14.4-1 contains a list of the assumptions for the analysis. All parameters listed except the Radial Peaking Augementation Factor and the Integrated Radial Peaking Factor are based on the Cycle 19 plant transient simulations, which are valid based on Reference 14.4-13. The most negative Doppler coefficient is used to enhance the positive reactivity feedback from the reactor coolant temperature decrease. Likewise, the most negative moderator temperature coefficient of reactivity is chosen to maximize the positive reactivity insertion associated with the RCS cooldown following the rod drop. This maximizes the steady-state heat flux and minimizes the DNBR value. The initial pressurizer pressure corresponds to the minimum value allowed by procedure minus uncertainty. This results in a lower final RCS pressure and thus in a lower minimum DNBR. The minimum dropped rod worth allowed by the PDIL is 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.

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

Parameter	Units	Value
Initial Core Power Level	MWt	1500 (1)
Core Inlet Temperature	°F	545 (1)
Pressurizer Pressure	psia	2075 (1)
RCS Flow Rate	gpm	202,500 (1)(2)
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.5
CEA Insertion at Full Power	% Insertion of Bank 4	25.0
Dropped CEA Worth	% $\Delta\rho$	-0.246 (unrodded) -0.240 (PDIL)
Radial Peaking Augmentation Factor	N/A	1.195
Integrated Radial Peaking	N/A	1.890

1. The plant transient simulations, which are based on Reference 14.4-4, were adjusted to account for power, temperature, pressure, and flow measurement uncertainty in the DNBR calculations for Cycle 20.

2. The initial RCS flow rate was adjusted to reflect the T.S. flow of 206,000 gpm minus the measurement uncertainty (i.e., 198,584 gpm) in the DNBR calculations for Cycle 20.

14.4.4 Results

The CEA Drop incident was partially reanalyzed for Cycle 20 to evaluate the DNB performance of the DNB-limiting HTP assembly in the Cycle 20 core. The reanalysis was limited to performing the minimum DNBR calculations using the HTP DNB correlation and a Cycle 20 DNB-limiting axial power distribution generated with Framatome ANP's setpoint axial methodology (Reference 14.4-10). The CESEC plant simulations from the Cycle 19 analysis of the CEA Drop incident (Reference 14.4-4) remain valid and were used as input into the minimum DNBR calculations (Reference 14.4-13).

The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.4-10).

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. The incident is initiated by the insertion of the dropped CEA worth over a time period of 1.0 seconds. 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.

For Cycle 20, the full power CEA drop initiated at an ASI of -0.144, and with a Group 4 insertion of 25%, results in a minimum DNBR value greater than the HTP correlation 95/95 DNBR safety limit plus a 2% mixed-core penalty (Reference 14.4-10).

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 limiting 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	≥Minimum DNBR Limit
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 that do not exceed the SAFDLs.

14.4.8 Conclusions

The results of the CEA drop rod incident analysis for Cycle 20 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 than or equal to the minimum DNBR limit.

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 EMF-2062(P), Guidelines for PWR Safety Analysis, G104,026, "Dropped Control Rod/Blank (SRP 15.4.3.1), "June 1998.

14.4-3 ANF-89-151 (P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992.

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.

14.4-10 E-4350-595-1, "Ft. Calhoun Unit 1, Cycle 20: Non-LOCA Transient MDNBRs," December 2000.

14.4-11 EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994.

14.4-12 XN-NF-82-21 (P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, "Revision 1, September 1983.

14.4-13 EA-FC-00-028, Revision 0, "Cycle 20 Transients Summary."

14.5 MALPOSITIONING OF THE NON-TRIPPABLE CEA's

14.5.1 Malpositioning of the Non-Trippable CEA's

For Cycle 1 through Cycle 10, there were four part length control rods. These part length rods were replaced and four non-trippable full length rods were installed prior to Cycle 11 (Reference 14.5-1).

Originally, the four part length CEA's were intended for use in controlling the axial power distribution, however, Technical Specification 2.10.2(5) required them to be withdrawn to at least 114 inches when critical. In practice they were withdrawn to the all-rods-out position.

The use of the non-trippable CEA's for axial power distribution has been prohibited, because movement from a high reactivity region, in which their residence time has been long enough to allow xenon decay, could cause the effects of both adding positive reactivity and distorting the axial and radial power distributions.

The inadvertent insertion of the non-trippable group from the withdrawn condition is still possible.

14.5.2 Non-Trippable CEA Drop

The non-trippable CEA's are not connected to any reactor trip circuit and will not drop into the core on a reactor trip or loss of power, but a mechanical failure in a CEDM could cause a non-trippable CEA to drop into the lower region of the core. The drop of this rod (from the fully withdrawn position) produces a transient similar to that of the CEA Drop Incident (Section 14.4). In conclusion, this event is bounded by the CEA Drop and no analysis is required.

14.5.3 Section 14.5 References

14.5-1 Amendment No. 109 to Facility Operating License DPR-40 for Fort Calhoun Station Unit No. 1, issued May 4, 1987.

14.6 LOSS OF COOLANT FLOW INCIDENT

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 the minimum DNBR limit. At this point there is a 95% probability at a 95% confidence level that DNB does not occur.

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).

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. The CESEC plant transient thermal-hydraulic code is used to simulate the overall response of the reactor coolant and steam systems during the transient (References 14.6-6, 7, 8, and 9). The CESEC code 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 CESEC code explicity models 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).

Based on the overall core conditions calculated by CECES at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hydraulic model of the limiting assembly (representing each sub-channel by a single "channel").

The limiting assembly DNBR calculations are performed using the approved HTP correlation (References 14.6-1, 14.6-14, and 14.6-3). The DNBR safety limit included a 2% mixed-core penalty (Reference 14.6-15).

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). 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. The CESEC plant transient data for Cycle 18 was used as input for the Cycle 20 DNBR calculations, which is described in the next section.

14.6.1.5 Results

The Loss of Coolant Flow event was partially reanalyzed for Cycle 20 to evaluate the DNB performance of the DNB-limiting HTP assembly in the Cycle 20 core. The reanalysis was limited to performing the minimum DNBR calculations using the HTP DNB correlation and a Cycle 20 DNB-limiting axial power distribution generated with Framatome ANP's setpoint axial methodology (Reference 14.6-4). The CESEC plant simulation from the Cycle 18 analysis of the Loss of Coolant Flow event (Reference 14.6-5) remain valid and were used as input into the minimum DNBR calculations (Ref. 14.6-16). The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.6-4).

The sequence of events for the Loss of Coolant Flow event is presented in Table 14.6-2. Figures 14.6-2 through 14.6-5 present the core power, heat flux, core coolant temperatures, and RCS pressure as a function of time.

The Loss of Coolant Flow event for Cycle 20 results in a minimum DNBR value that is greater than the HPT correlation 95/95 DNBR safety limit plus 2% mixed-core penalty (Reference 14.6-4).

Table 14.6-1 - "Key Parameters Assumed in the Loss Of Coolant Flow Analysis"

Parameter	Units	Value
Initial Core Power Level	MWt	1500 (1)
Initial Core Inlet Coolant Temperature	°F	545 (1)
Initial RCS Flow Rate	gpm	202,500 (1)(2)
Pressurizer Pressure	psia	2075 (1)
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5
LFT Analysis Setpoint	% of initial flow	90.0 (3)
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 Integrated Radial Peaking Factor (F_R^T)		1.890

1. The plant transient simulations, which are based on Reference 14.6-5, were adjusted to account for power, temperature, pressure, and flow measurement uncertainty in the DNBR calculations for Cycle 20.

2. The initial RCS flow rare was adjusted to reflect the T.S. flow of 206,000 gpm minus the measurement uncertainty (i,e., 198,584 gpm) in the DNBR calculations for Cycle 20.

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

Table 14.6-2 - "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	-----
5.70	Minimum DNBR Reached	≥Minimum DNBR Limit
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. 14.6-10).

- 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 greater than or equal to the minimum DNBR limit. 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 Analysis for Ft. Calhoun Cycle 20 was performed in accordance with the approved Framatome ANP Richland, Inc. ANF-RELAP Non-LOCA Transient Analysis methodology (Ref. 14.6-3) and has been documented in Ref. 14.6-13.

The ANF-RELAP plant transient thermal-hydraulic code is used to simulate the overall response of the reactor coolant and steam systems during the transient. The ANF-RELAP model includes a thermal model of the fuel, a hydraulic model of the Reactor Coolant System (RCS), a point-kinetics model of the reactor, a hydraulic model of the steam system, and control logic which represents various Reactor Protection System (RCS) trips. The RCS hydraulic model simulates the hot legs, pressurizer, steam generators (primary sides), cold legs, reactor coolant pumps (RCPs), reactor vessel, and core. The steam system hydraulic model simulates the steam generators (secondary sides), main steam lines, and turbine.

Based on the overall core conditions calculated by ANF-RELAP at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single "channel"). The limiting assembly DNBR calculations are performed using an approved Framatome ANP Richland, Inc. DNB correlation (Reference 14.6-14).

14.6.2.4 Results

The Seized Rotor event is initiated at beginning-of-cycle (BOC) full-power conditions by an instantaneous seizure of an RCP, coincident with a loss of offsite power and trips to the turbine and unaffected RCPs (see Tables 14.6-3 and 14.6-4 and Figure 14.6-6).

Flow through the affected RCS cold leg abruptly decreases, reaching the RPS low flow setpoint at 0.04 seconds from transient initiation and reversing direction at 0.43 seconds (see Figure 14.6-7). Flow through the core decreases less abruptly, responding to the combined effects of the unaffected RCPs coasting down and the affected RCP not rotating at all.

During the RPS signal processing and scram control element assembly (CEA) holding coil release delays (see Table 14.6-3), the reactor is still at power, and the decreasing core flow causes core temperatures to increase (see Figure 14.6-8).

The increasing core temperatures, in conjunction with the positive moderator temperature coefficient (MTC) used in the analysis, cause the reactivity and power to increase, until scram CEA insertion begins at 1.19 seconds (see Figures 14.6-9 and 14.6-10). When scram CEA insertion begins, the power decreases, but ongoing transfer of the heat stored in the rods to the coolant causes rod heat fluxes to remain elevated for a short period of time thereafter.

During this short period of elevated rod heat fluxes, the decreasing core flow and increasing core temperatures further degrade the margin to DNB. The MDNBR occurs at 1.90 seconds when the rod heat fluxes begin to decrease, as the combined effects of decreasing rod heat fluxes and increasing core pressures (see Figure 14.6-11) begin to outweigh the combined effects of decreasing core flow and increasing core temperatures.

The MDNBR is greater that the HTP correlation 95/95 DNBR safety limit plus 2% mixed-core penalty. This indicates that no fuel failure due to DNB would occur and, therefore, that the acceptance criterion which is challenged by this event is met.

14.6.2.5 Affected Plant Technical Specifications

This event is analyzed utilizing inputs from the following Technical Specifications (Ref. 14.6-10).

• 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).

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.

14.6.2.8 Conclusions

It is concluded that no fuel failures due to DNB result from the Seized Rotor Event and therefore this event 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 EMF-2062(P), Guidelines for PWR Safety Analysis, G104,022, "Loss of Forced Reactor Coolant Flow (SP 15.3.1 and 15.3.2)," June 1998.

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 ANF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992.

14.6-4 E-4350-595-1, "Fort Calhoun Unit 1, Cycle 20: Non-LOCA Transient MDNBRs," December 2000.

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 Fort Calhoun Operating License DPR-40 and Technical Specifications.

14.6-11 10 CFR 100, Reactor Site Criteria, as amended effective January 5, 1987.

14.6-12 EA-FC-98-053, Rev.0 "Cycle 19 Seized Rotor Analysis."

14.6-13 EMF-2499, Rev. 0, "Fort Calhoun RCP Rotor Seizure Analysis," Siemens Power Corporation, December 2000.

14.6-14 EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, " March 1994.

14.6-15 XN-NF-82-21 (P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Revision 1, September 1983.

14.6-16 EA-FC-00-028, Revision 0, "Cycle 20 Transients Summary."

Table 14.6-3 - "Key Parameters Assumed for Seized Rotor Event Analysis"

Parameter	Biasing	Value
Initial reactor power	Rated + 2.0% uncertainty	1530 MW
Initial RCS total flow rate	Tech. Spec. minimum indicated value - 3.6% uncertainty	198,584 gpm
Core bypass flow rate	Maximum analysis value	4.54 % of RCS total
Initial core inlet temperature	Programmed value for rated power Maximum analyzed value	543°F(1) 547°F
RCS pressure for DNBR calculations	Minimum allowed value - 22 psi uncertainty	2053 psia
MTC	Tech. Spec. maximum for indicated power of 80% and above	$+ 0.2 \times 10^{-4}\ \Delta\rho/°F$
Event Initiator	Prescribed	Instantaneous seizure of single RCP
Trip assumption for unseized RCPs	Conservative	Coincident with loss of offsite power at time of rotor seizure
Low-flow reactor trip setpoint	Setpoint - 2.0% uncertainty	93.0% of initial flow
Low-flow reactor trip signal delay	Maximum analysis value	0.65 sec
Scram CEA holding coil delay	Maximum analysis value	0.50 sec after signal received

1. The initial ANF-RELAP core inlet temperature was set to the HFP nominal inlet temperature. The core inlet temperature assumed for the XCOBRA-IIIC boundary conditions for the MDNBR calculations is the transient ANF-RELAP-calculated core inlet temperature plus 4°F to account for the operating band and measurement uncertainty.

Table 14.6-4 - "Sequence of Events for Seized Rotor Event"

Time	Event	Value
0.00 sec	Plant is operating at BOC full-power condition	----
0.00 sec	RCP seizes	----
0.00 sec	Offsite power is lost, and turbine is tripped	----
0.00 sec	Other RCPs are tripped (loss of power)	----
0.04 sec	Cold leg flow reaches RPS low flow setpoint	93.0% of initial
0.43 sec	Cold leg flow reverses direction	
0.69 sec	Low flow signal initiates reactor trip	---
1.19 sec	Scram CEA insertion begins, and reactor power peaks	1553 MW
1.90 sec	MDNBR occurs	≥Minimum DNBR limit
2.40 sec	Pressurizer spray turns on	----

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 in accordance with Contract 750 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 $\leq$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 $\geq$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 $\geq$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 $\leq$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 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

14.10 MALFUNCTIONS OF THE FEEDWATER SYSTEM

14.10.1 Loss of Feedwater Flow

14.10.1.1 General

This incident has been reevaluated for Cycle 17 (Reference 14.10-1) to determine the impact of the pressure drops caused by piping losses to the primary and main steam safety valves. The loss of feedwater flow incident is defined as a reduction in feedwater flow to the steam generators when operating at power without a corresponding reduction in steam flow from the steam generators. The limiting case is a total loss of main feedwater which most likely would result from:

a. inadvertent closure of the main feedwater control or regulating valves or feedwater isolation valves due to a feedwater controller malfunction or manual positioning by the operator, or

b. loss of all feedwater or condensate pumps.

Upon the loss of feedwater flow to the steam generators and a continued steam demand by the turbine, water inventories in the steam generators begin decreasing. Primary system temperatures begin to increase due to the reduced heat removal capability of the secondary system.

The Reactor Protective System provides reactor protection through a reactor trip actuated by low water level in either steam generator with additional protection for the reactor provided by the high pressurizer pressure trip and by the thermal margin/low pressure trip and the variable high power trip. Moreover, the steam generators are designed to withstand the thermal loading imposed by a total loss of water and the subsequent refill transient (Reference 14.10-2).

The feedwater system is designed to avoid a complete loss of feedwater flow. Three, one-half capacity motor driven feedwater pumps, having common suction and discharge are provided (Ref. 14.10.4-2). Two safety grade auxiliary feedwater pumps, one motor driven and one steam turbine driven, are also available (Ref. 14.10.4-3). Both discharge into a common header from which a line leads to each steam generator. Either of the two auxiliary feedwater pumps can supply sufficient feedwater to remove decay heat from the reactor coolant system, even at peak steam generator pressures.

Complete loss of feedwater flow from the main feedwater system could occur in the event of a rupture of a feedwater line. Check valves in the feedwater lines to each steam generator prevent a steam generator blowdown should such an unlikely event occur. Rupture of a feedwater line downstream of one of these check valves would result in blowdown of one steam generator. The rupture of a main steam line, discussed as in Section 14.12 represents an upper limit for such an accident.

The reactor is assumed to be initially operating at full power conditions with all parameters within the LCOs (e.g., PDIL, nominal coolant temperatures and pressures, etc.) and nominal steam generator water levels and pressures. The plant then experiences a complete loss of main feedwater flow.

Since subcooled feedwater flow is terminated, the heat extracted from the primary system goes directly into vaporizing the saturated fluid/steam mixture in the steam generators and causes a reduction in the primary-to-secondary heat transfer. The primary system inlet temperatures starts rising within a few seconds, since the temperature change occurs in the primary fluid as it leaves the steam generators. The primary inlet temperature rise causes a corresponding increase in the average and exit coolant temperatures. In the presence of a negative MTC and fuel temperature coefficient, the increasing coolant and fuel temperatures add negative reactivity to the core causing the core power to decrease.

The control rods are in the manual mode of operation and remain stationary. Localized moderator reactivity feedback effects, however, cause the axial power distribution to shift slightly towards the bottom of the core as the moderator temperature increases. Radially there will be a redistribution and the radial peaks will either increase or decrease a few percent, the sense of the change depending on what the initial distribution looked like.

In any event, the radial and 3-D peaks increase at most by a few percent and, in conjunction with the decreasing core power (and heat flux), yield a negligible change in DNBR and peak LHR. Thus, the SAFDLs are not approached.

During the first few seconds of the event, the secondary temperature and pressure rise. The steam generator water level drops since the turbine is continuing to demand full power in addition to shrinkage caused by the secondary pressure increase. The steam generator water level continues to decrease until a reactor trip on low water level occurs and subsequently initiates a turbine trip. During this time, the primary pressure increase is mitigated by the action of the pressurizer sprays. The primary pressure increase is not sufficient to lift either the pressurizer power operated relief or primary safety valves.

After the reactor trip occurs, the reactor core power rapidly decreases to the decay power levels. The amount of residual heat contained in the fuel and structural materials determines the rate at which the liquid inventory in the steam generators is depleted. The higher the fuel temperatures (i.e., low H_{gap}) and the higher the fission product inventory (i.e., higher power and burnup) the greater the rate of steam generator liquid mass loss. The turbine trip leads to a quick opening of the steam dump and bypass valves, normally in the automatic mode of operation. The RCS and steam generator pressures and temperatures are regulated by this system to remove decay heat which is extracted by forced coolant flow through the core.

The inventory remaining in the steam generators after trip will not be completely depleted until about 30 minutes (Ref. 14.10-3) assuming no operator action and no additional feedwater. During this time interval, automatic actuation of the safety grade auxiliary feedwater system on low S.G. level (32% wide range level) would occur to assure that a secondary heat sink is maintained. This will allow the cooldown of the plant to proceed in an orderly fashion using the power operated safety valves (MS-291 and MS-292), after which, shutdown cooling can be initiated.

14.10.1.2 Applicable Industry and Regulatory Requirements

The loss of feedwater incident, is an anticipated operational occurrence for which the following criteria must be met:

(a) the transient minimum DNBR ≥1.18 (CE-1 correlation (Ref. 14.10-4 to 14.10-6)).
(b) the peak linear heat rate (PLHR) ≤22kw/ft (Ref. 14.10-7)
(c) the maximum peak RCS pressure <2750 psia (110% of design pressure of 2500 psia)
(d) the maximum peak secondary system pressure <1100 psia (110% of design pressure at 1000 psia)
(e) dose rates within 10 CFR 100 guidelines (i.e., 25 Rem whole body and 300 Rem thyroid dose)
(f) at least 10 minutes is available for the operator to initiate the auxiliary feedwater system before dryout of the steam generators occurs.

NOTE: Criterion (e) is not a major concern because the primary release paths for this event are (a) the secondary safety valves to the atmosphere and b) the steam dump and bypass system to the condenser venting to the atmosphere via the condenser air ejectors which have a large decontamination factor. Explicit dose calculations are not performed on the basis that dose rates associated with other transients, such as the Loss of AC event, are more limiting.

NOTE: Criterion (f) is not a significant concern since auxiliary feedwater flow is automatically initiated to both SGs upon generation of a low S.G. level signal at 32% wide range. Automatic initiation ensures that subcooled feedwater will be supplied in sufficient time to provide a secondary heat sink and prevent S.G. dryout. This criteria has traditionally been instituted in the past in analyzing this event to demonstrate that an adequate secondary heat sink exists for at least 10 minutes to remove primary residual heat in the absence of either automatic or manual by the operator, initiation of auxiliary feedwater.

14.10.1.3 Methods of Analysis

A complete loss of feedwater flow was assumed in this analysis, since that condition requires that the most rapid response from the reactor control and protective system.

There are three combinations of parameters which produce the limiting case for (a) approaching the SAFDLs and the RCS upset pressure limit of 2750 psia, (b) approaching the secondary system upset pressure limit of 1100 psia and (c) steam generator dryout. The severity of the limiting case depends on what is assumed for single failures and the operating modes assumed for control equipment. The two cases analyzed to date are:

(1) For the limiting RCS peak pressure and DNBR transient, the steam dump and bypass system and pressurizer sprays are assumed to be inoperable. No credit for the steam generator low level trip is assumed. Credit for only the high pressurizer pressure reactor trip is conservatively assumed. The safety analysis high pressurizer pressure trip setpoint assumed corresponds to the Technical Specifications value plus the 22 psia pressure measurement uncertainty allowance (e.g., 2422 psia).

(2) For the limiting steam generator pressure transient, the pressurizer pressure and level control systems are assumed to be in auto to maximize the time to trip which maximizes the primary to secondary heat transfer and hence pressure rise on the secondary system. The steam dump and bypass system is assumed to be inoperable. Credit for a reactor trip on low SG water level is assumed. The safety analysis low level trip setpoint assumed corresponds to the Technical Specifications value minus a 10" uncertainty for instrument error (i.e., 68 inches below the nominal SG water level).

(3) For the limiting steam generator liquid inventory depletion transient, the steam dump and bypass system is assumed to be operable. Credit for a reactor trip on low S.G. water level is assumed. The safety analysis low level trip setpoint assumed corresponds to the Technical Specifications value minus a 10" uncertainty for instrument error (i.e., 68 inches below the nominal SG water level).

For all three of these cases the physical sequence of events is basically the same as the expected case except that a positive MTC is assumed which causes a power increase (with RCS heatup) rather than a decrease. In addition, with the steam dump and bypass system inoperable, the steam generator pressures will increase and cause the lifting of the main steam safety valves. The increase in the primary pressure may also be high enough for this case to open the pressurizer safety valves for a short time; assuming no credit for the Pressurize Power Operated Relief Valves (PORVs).

The following methods and procedures are used to determine the (a) minimum transient DNBR and peak LHR, (b) peak RCS pressure, (c) peak S/G pressure, and (d) time to steam generator dryout.

For the case of determining the peak RCS pressure, the method used is to setup and execute a CESEC (Ref. 14.10-8 through 14.10-11) case simulating a Loss of Feedwater Flow event for approximately 75 seconds where the key parameters including uncertainties are indicated in Table 14.10.1-1. The peak pressure developed during the event is then directly extracted from the CESEC run.

Since the key parameters assumed in Table 14.10.1-1 give the maximum power rise, the results of this case can be used to verify that the LHR SAFDL is not violated.

For determining the minimum transient DNBR for the event, it should be noted that the parameters assumed in Table 14.10-1 are also the worst case conditions for DNBR <u>except</u> that the pressurizer pressure control system is assumed to be inoperable. A conservative absolute minimum DNBR can be calculated without taking credit for the RCS pressure rise. The explicit procedures for establishing the time of minimum DNBR and the absolute minimum DNBR using TORC (or CETOP) are outlined in References 14.10-4 through 14.10-6.

The specific trip signal generated during the event is dependent on initial conditions at the onset of event and the single failures assumed.

A minimum cycle specific value of the Doppler multiplier was used to minimize the amount of negative reactivity feedback which would mitigate the transient increases in both the core heat flux and the RCS pressure. Likewise, the most positive reactivity insertions into the core and thus the increases in core heat flux, RCS pressure, and primary system temperatures. The initial pressurizer pressure was chosen to be 2053 psia which corresponds to the minimum allowed pressurizer pressure including uncertainties; this results in a maximum peak RCS pressure. In addition, no credit was taken for the low steam generator level trip. The high pressurizer pressure trip was assumed to terminate the event.

14.10.1.4 Inputs and Assumptions

The inputs utilized in the loss of feedwater incident are indicated in Table 14.10.1-1. The most limiting conditions which lead to conservative results are shown below:

Peak RCS Pressure and Closest Approach to the SAFDLs

Core Power Level	Full power +2% measurement uncertainty
Doppler Curve	BOC curve -18.1% uncertainty
MTC	Most positive value allowed by Tech Specs (e.g., $+.5X10^{-4}\Delta\rho/°F$)
Kinetics Parameters	Minimum absolute β (i.e., EOC) with λ_i and $SB\ell^*$ consistent with β
Core Average H_{gap}	Minimum predicted value in fuel cycle
T_{in}	Maximum allowed by Technical Specifications LCOs including +2°F measurement uncertainty
RCS Pressure	Minimum allowed by Technical Specifications LCOs (-22 psia measurement uncertainty)
Vessel Flow Rate	Minimum Technical Specifications guaranteed determination vessel flow rate
Response Times	Largest delay times allowed by Technical Specifications (e.g., .9 seconds for RPS trip signal processing, .5 seconds for CEA holding coil decay time, etc.)
CEA Drop Time	Largest allowed by Technical Specifications for time to 90% insertion
Scram Reactivity	Minimum available scram worth corresponding to deepest rod insertion allowed by Technical Specifications PDIL LCO
Scram Reactivity Curve	Hot Full Power Curves at ASI = +0.20 for ARO

MSSV Drift Allowance	3% of Setpoint Pressure
Initial Secondary Pressure	Low secondary pressure which delays opening of secondary safety valves and maximizes primary system temperature and pressure rise (e.g., 800 psia)
Reactor Regulating System	MANUAL mode at ARO
Steam Dump and Bypass Control System	MANUAL mode (i.e., steam dump and bypass valves inoperative)
Pressurizer Pressure Control System	MANUAL mode (i.e., pressurizer sprays and relief valves inoperative)
Pressurizer Level Control System	MANUAL mode (i.e., maximum charging and zero letdown flow)

14.10.1.5 Results

Two cases have been analyzed to ensure that the acceptance criteria of Section 14.10.1.2 are satisfied. Analysis of the loss of feedwater flow event for Cycle 6 operation at 1500 MWT showed that due to the pressure increase on the primary side and a negative pressure feedback coefficient, a negative reactivity is introduced which results in a slowly decreasing power level prior to reactor trip. As a result, the MDNBR does not decrease below the initial steady state value. MDNBR was not evaluated in the Cycle 17 analysis.

The fastest rate of steam generator liquid inventory depletion is characterized by the input parameters indicated in Section 14.10.1.4 with the following exceptions:

Doppler Curve	EOC curve +15% uncertainty
Kinetics Parameters	Maximum absolute β (i.e., BOC) with λ_i and ℓ^* consistent with β
Core Average H_{gap}	Minimum predicted value in fuel cycle

Initial Secondary Pressure	High secondary pressure which results in faster opening of the secondary safety valves and greater rate of S.G. liquid inventory depletion
Steam Dump Bypass Control System	AUTO mode i.e., steam dump and bypass valves functional)
Pressurizer Pressure Control System	AUTO mode (i.e., pressurizer sprays and valves functional)

In order to bound the effects of the pressure drop caused by piping losses in the main steam system, the loss of feedwater flow event was reanalyzed for Cycle 17 (Reference 14.10-1) assuming a pressure drop of 35 psid in the main steam piping between the S/G and MSSVs and up to 6% plugging per steam generator. The analysis was performed to ensure that the peak RCS pressure during the event would remain less than the 2750 psia limit and the peak secondary pressure would remain less than the 1100 psia limit. The loss of feedwater flow event, initiated from the conditions given in Table 14.10.1-1, results in a high pressurizer pressure trip signal at 38.7 seconds.

At 43.0 seconds, the primary pressure reached its maximum value of 2566 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 46.11 seconds. The secondary pressure reaches its maximum value of 1088 psia at 44.7 seconds after initiation of the event.

Table 14.10.1-2 presents the sequence of events for this transient. Figures 14.10-1 through 14.10-4 show the transient behavior of power, RCS pressure, 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 the change in DNBR and peak linear heat rate which occurred was negligible.

Another loss of feedwater case was run in Reference 14.10-1 to maximize the peak secondary system pressure, since some of the analysis input is different than the case for peak primary pressure. This case, which has the pressurizer pressure and level control systems in auto, maximizes the time required to trip and requires the Low Steam Generator Level Trip (which occurs prior to the High Pressurizer Pressure Trip used in the peak RCS pressure case) to mitigate its effects. The peak secondary system pressure from this case occurs at 42.3 seconds at 1095 psia. This case is run with the lowest set-pressure 6" MSSV inoperable on each generator, which is the limiting case to verify acceptability of Technical Specification 2.1.6 (3).

14.10.1.6 Affected Plant Technical Specifications

The loss of feedwater incident analysis uses inputs from the following Technical Specifications:

- LSSS 1.3 — Limiting Safety System Settings, Reactor Protective System
- LCO 2.10.2 — Reactivity Control Systems and Core Physics Parameters Limits
- LCO 2.5 — Steam and Feedwater Systems
- Section 3.9 — Surveillance Requirements for the Auxiliary Feedwater System

14.10.1.7 Affected Plant Systems

For this event, the affected plant systems are the reactor coolant system, reactivity control system, main feedwater system, auxiliary feedwater system and the reactor protective system.

14.10.1.8 Limiting Parameters for Reload Analysis

Reevaluation of the loss of feedwater incident is required when one of the following conditions exist:

- Core physics parameters change (Doppler, CEA insertion time) in a nonconservative direction.
- A plant modification is expected to cause a change to a pertinent system such as main steam safety valves (MSSV). For example the loss of feedwater would have to be reevaluated if the setpoint tolerance would be changed on the primary or secondary safety valves.
- A nonconservative change is made to a pertinent technical specification LCO such as the maximum system pressure.

Any changes to parameters and/or technical specifications must result in peak reactor coolant system pressures less than 2750 psia, peak S/G pressure less than 1100 psia, a peak linear heat rate less than 22 kw/ft and a transient minimum DNBR value greater than 1.18.

Table 14.10.1-1 - "Fort Calhoun Cycle 17 Key Parameters Assumed in the Loss of Feedwater Flow Analysis"

Parameters	Units	Value
Initial Core Power Level	MWt	1535.6 (102% + RCP Heat)
Initial Core Inlet Temperature	°F	547
Initial Pressurizer Pressure	psia	2,053
Initial Steam Generator Pressure	psia	800
Initial RCS Flow Rate	gpm	192,000
Moderator Temperature Coefficient	10^{-4} $\Delta\rho$/°F	+0.5
Fuel Temperature Coefficient	10^{-4} $\Delta\rho$/°F	Least negative predicted during core life.
CEA Time to 100% Insertion (Including Holding Coil Relay)	seconds	3.1
Scram Reactivity Worth	%$\Delta\rho$	-6.12
Kinetics Parameters,	β	.005214
Maximum number of steam generator tubes assumed to be plugged per steam generator		300
MSSV Drift Allowance	%	+3
PSV Drift Allowance	%	+1

Table 14.10.1-2 - "Fort Calhoun Cycle 17 Sequence of Events for the Loss of Feedwater Flow Event to Maximize Calculated RCS Peak Pressure"

Time (sec)	Event	Setpoint or Value
0.0	Loss of Feedwater	------------
38.7	High Pressurizer Pressure Analysis Trip Setpoint is Reached	2422 psia
42.1	Steam Generator Safety Valves Open	1045 psia *
43	Maximum RCS Pressure	2566 psia
44.7	Maximum Steam Generator Pressure	1088 psia

Table 14.10.1-3 - "Fort Calhoun Cycle 17 Sequence of Events for the Loss of Feedwater Flow Event to Maximize Calculated Secondary System Pressure"

Time (sec)	Event	Setpoint or Value
0.01	Loss of Feedwater	------------
33.5	Low Steam Generator Water Level Setpoint is Reached	25.5%
37.2	Maximum RCS Pressure	2289 psia
38.0	Steam Generator Safety Valves Open	1045 psia *
42.3	Maximum Steam Generator Pressure	1095 psia

* Includes run pipe differential pressure and 3% setpoint drift allowance.

14.10.2 Loss of Feedwater Heating

14.10.2.1 General

Typically about one-third of the thermal energy flowing to the turbine is directed through the extraction lines to the feedwater heaters. If the feedwater heating is turned off slowly by closing all extraction valves, the steam generator could deliver full thermal power at a reduced steam flow and increase the enthalpy difference between steam and feedwater. The logarithmic mean temperature difference for the steam generator would be increased, and the overall thermal cycle efficiency would be decreased. Exxon Nuclear Company analyzed this event for Cycle 6 (Ref. 14.10-12) operation as it was not known if this would be a bounding event for stretch power (1420 to 1500 MW). The results of the analysis were found to be bounded by the loss of feedwater flow event, thus, this event has not been reanalyzed in any later operating cycles by OPPD personnel. The method of analysis described below would be utilized by OPPD if a reanalysis of this event was required.

14.10.2.2 Applicable Industry and Regulatory Requirements

The loss of feedwater heating incident is classified as an anticipated operational occurrence which requires a reactor trip at hot full power to provide protection against exceeding the DNB and LHR SAFDL's (Ref. 14.10-13). These requirements are met by adding sufficient margin to the DNB and LHR LCOs to ensure that the SFDL will not be exceeded during a loss of feedwater heating event.

For some limiting conditions and reactivity insertion rates, the variable High Power Trip (VHPT) in conjunction with the steady state LCOs, is required to prevent the DNBR limits from being exceeded. The VHPT and the Axial Power Distribution Trip, in conjunction with the steady state LCOs, prevent the LHR limits from being exceeded.

14.10.2.3 Method of Analysis

The loss of feedwater analysis methodology would be similar to the excess load events analysis methods, Section 14.11, except there is not an uncontrolled heat extraction during the event. The cooldown event is characterized by a gradual loss of feedwater heating which delivers feedwater to the steam generators with a reduced enthalpy value. The loss of feedwater heating event is analyzed using the CESEC computer code (Refs. 14.10-8 through 14.10-11). 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: The transient average core heat flux, average channel mass flow rate, reactor core inlet temperature, and reactor coolants system pressure serve as inputs to the CETOP computer code (Ref. 14.10-6). This code utilizes the CE-1 critical heat flux correlation (Ref. 14.10-4 and 14.10-5) to calculate the minimum DNBR for the hot channel DNB and LHR ROPMS are then calculated for the event using the methodology described in Reference 14.10-13.

14.10.2.4 Inputs and Assumptions

In order to determine the most limiting reactor response, the transient initial conditions and assumptions are selected to maximize the greatest possible cooldown. All bypass valves on the feedwater heaters are postulated to fail open simultaneously; this will divert all extraction flow to the condenser without significantly impacting the amount of extraction steam flow from the turbine. This is simulated by ramping the feedwater enthalpy down to condenser conditions in a 30 second period, without any changes in main steam or feedwater mass flow rates. The inputs are listed on Table 14.10.2-1 for each key parameter.

14.10.2.5 Results

The resulting transient is rather slow with the reactor power and heat flux increasing as a result of the decreasing core average coolant temperature and the feedback of the negative moderator temperature coefficient. The reactor power level increases at an average rate of about 0.6% of rated power per second, which results in a variable high power reactor trip at approximately 25 seconds. The minimum DNBR decreases from an initial value of 1.69 to a minimum value of 1.43 at about 27 seconds. The above results demonstrate that an adequate DNBR margin is maintained throughout this transient.

14.10.2.6 Affected Plant Technical Specifications

The following Technical Specifications are used as input to the loss of feedwater heating analysis:

- SL 1.1 — Safety Limits - Reactor Core
- SL 1.2 — Safety Limit, Reactor Coolant System Pressure
- LSSS 1.3 — Limiting Safety System Setting, Reactor Protective System
- LCO 2.1 — Reactor Coolant System
- LCO 2.5 — Steam and Feedwater Systems
- LCO 2.10.2 — Reactivity Control Systems and Core Physics Parameter Limit
- LCO 2.10.4 — Power Distribution Limits

14.10.2.7 Affected Plant Systems

For the loss of feedwater heating event the affected plant systems are the reactor coolant system, reactor protective system, the reactivity control system, the main steam and main feedwater systems, condensate system and waste drain system.

14.10.2.8 Limiting Parameters for Reload Analysis

Reevaluation of the loss of feedwater heating event 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 14.10.2-1 parameters in a nonconservative direction.

Table 14.10.2-1 - "Fort Calhoun Cycle 6 Key Parameters Assumed in the Loss of Feedwater Heating Analysis"

Parameters	Units	Value
Initial Core Power Level, MWt	MWt	1530 (102%)
Initial Core Inlet Temperature,	°F	547
Initial Pressurizer Pressure,	psia	2,053
Initial Steam Generator Pressure,	psia	815
Initial RCS Flow Rate, gpm	196,000	
Moderator Temperature Coefficient,	10^{-4} $\Delta\rho$/°F	+0.5
Fuel Temperature Coefficient,	10^{-4} $\Delta\rho$/°F	Least negative predicted during core life
Fuel Temperature Coefficient Multiplier		0.85
CEA Time to 100% Insertion, (Including Holding Coil Relay)	seconds	3.1
Scram Reactivity Worth	%$\Delta\rho$	-6.65
Kinetics Parameters	β	.004696
Maximum number of steam generator tubes assumed to be plugged per steam generator		300

14.10.3 Specific References

14.10-1 EA-FC-97-004, Rev. 1, dated March 13, 1998, "Evaluation of the Effect of Increased Line Pressure Drop on MSSVs and PSVs Setpoints".

14.10-2 "Steam Generator Performance" Rev. 0, CE Calculation T-601, dated May 20, 1968, "Steam Generator Circulation" Rev. 0, CE Calculation T-602, dated May 20, 1968, "Omaha Steam Generator Orificed Performance Characteristic" CE Calculation ST-603, dated October 22, 1973 and CE report CENC-1676 "Thermal-Hydraulic Analysis of the Omaha Public Power District Fort Calhoun Steam Generator" dated December 1984, Rev. 0.

14.10-3 EA-FC-89-11 Analysis of Applicability of Auxiliary Feedwater Actuation Setpoint with Reduced AFW Flow.

14.10-4 "CE Critical Heat Flux, Critical Heat Flux Correction for CE Fuel Assemblies with Standard Spacer Grids, Part 1: Uniform Axial Power Distribution" CENPD-162-P-A, September 1976.

14.10-5 "CE Critical Heat Flux, Critical Heat Flux Correction for CE Fuel Assemblies with Standard Spacer Grids, Part 2: Nonuniform Axial Power Distribution" CENPD-270-P, June, 1978.

14.10-6 CETOP: Thermal Margin Model Development, CE-NPSD-150-P, May 1981.

14.10-7 "Omaha Batch N Reload Fuel Design Report" CEN 374(O)-P, Revision 0, May 1988.

14.10-8 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, CE Proprietary Report, April 1974.

14.10-9 "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.10-10 "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.

14.10-11 Response to Questions on CESEC, CEN-234(C)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.10-12 XN-NF-79-79 "Fort Calhoun Cycle 6 Reload Plant Transient Analysis Report," October 1979.

14.10-13 "Omaha Public Power District Reload Core Analysis Methodology-Transient and Accident Methods and Verification," OPPD-NA-8303, Rev. 2, Section 5, April 1988.

14.10.4 General References

14.10.4-1 CE Transient Analysis Methods for Fort Calhoun Unit 1 Part Two: Transient Analysis for Generation of Technical Specification Limits, CENPSD-164-p Revision 1-P, September 1981.

14.10.4-2 SDBD-FW-116 Feedwater, Design Basis Document

14.10.4-3 SDBD-FW-AFW-117 Auxiliary Feedwater, Design Basis Document

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 (AOO) for which the transient minimum DNBR and the peak LHR 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.

14.11.3 Methods of Analysis

The methodology for analyzing the Excess Load Event is described in References 14.11-1, 14.11-11, and 14.11-13.

The CESEC plant transient thermal-hydraulic code is used to simulate the overall response of the reactor coolant and steam systems during the transient (References 14.11-2, 3, 4, and 5). The CESEC code 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.

Based on the overall core conditions calculated by CESEC at selected times during the transient, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly at those times. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hydraulic model of the limiting assembly (representing each sub-channel by a single "channel"). The limiting assembly DNBR calculations are performed using the NRC approved HTP correlation (References 14.11-6, 14.11-14 and 14.11-7).

The DNBR Safety limit includes a 2% mixed-core penalty (References 14.11-15).

14.11.4 Excess Load Increase From Hot Full Power

The limiting excess load event, Case (b) of Section 14.11.1, was partially reanalyzed for Cycle 20 to evaluate the DNB performance of the DNB-limiting HTP assembly in the Cycle 20 core. The reanalysis was limited to performing the minimum DNBR calculations using the HTP DNB correlation and a Cycle 20 DNB-limiting axial power distribution generated with Framatome ANP's setpoint axial methodology (Reference 14.11-8). The CESEC plant simulations from the Cycle 19 analysis of the Excess Load event (Reference 14.11-10) remain valid and were used as input into the minimum DNBR calculations. The plant simulations were adjusted to account for power, temperature, pressure, and flow measurement uncertainties in the minimum DNBR calculations (Reference 14.11-8).

The key parameters assumed in the analysis are summarized in Table 14.11-1. Table 14.11-2 presents the sequence of events for the Excess Load event. Figures 14.11-1 through 14.11-7 show the results of a representative Excess Load simulation.

The Excess Load event for Cycle 20 results in a minimum DNBR value that is greater than the HTP correlation 95/95 DNBR safety limit plus 2% mixed-core penalty (Reference 14.11-8).

14.11.5 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

14.11.6 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 parameters affected are provided in Table 14.11-1.

14.11.7 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.

14.11.8 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 EMF-2062(P), Guidelines for PWR Safety Analysis, G104,014 Revision 1, "Increase in Steam Flow (SRP 15.1.3)," April 1999.

14.11-7 ANF-89-151(P)(A), ANF-RELAP Methodology for Pressurizer Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992.

14.11-8 E-4350-595-1, "Ft. Calhoun Unit 1, Cycle 20: Non-LOCA Transient MDNBRs," December 2000.

14.11-9 Ft. Calhoun Operating License DPR-40 and Technical Specifications.

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.

14.11-14 EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994.

14.11-15 XN-NF-82-21 (P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Revision 1, September 1983.

14.11.9 General References

14.11.9-1 "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"

System/Parameter	Units	Value	Reference
Initial Core Power Level	MWt	1531.85[1]	14.11-10
Reactor Coolant System			
Initial Core Inlet Coolant Temperature	°F	545[1]	14.11-10
Initial RCS Flow Rate	gpm	202,500[1][2]	14.11-10
Pressurizer Pressure	psi	2075[1]	14.11-10
Reactivity Control System			
Moderator Temperature Coefficient	10^{-4} $\Delta\rho$/°F	-1.17	14.11-10
CEA Worth at Trip (PDIL) Total Integrated Unrodded	10^{-2} $\Delta\rho$	5.501	14.11-10
Radial Peaking Factor (F_RT)	N/A	1.890	14.11-6
Reactor Protective System			
Higher Power Trip	% of rated power	112	14.11-10

1. The plant transient simulations, which are based on Reference 14.11-10, were adjusted to account for power, temperature, pressure, and measurement uncertainty in the DNBR calculations for Cycle 20.

2. The initial RCS flow rate was adjusted to reflect the T.S. flow of 206,000 gpm minus the measurement uncertainty (i.e., 198,584 gpm) in the DNBR calculations for Cycle 20.

Table 14.11-2 - "Sequence of Events for the Excess Load Event"

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
60.2	Minimum DNBR Value Reached	>Minimum DNBR Limit